INL/EXT-13-29957

Advanced Fuels Campaign Light Water Reactor Accident Tolerant Fuel Performance Metrics

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February 2014

The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

Idaho National Laboratory

INL/EXT-13-29957
FCRD-FUEL-2013-000264

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February 2014

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Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

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Abstract

The safe, reliable and economic operation of the nation's nuclear power reactor fleet has always been a top priority for the United States' nuclear industry. As a result, continual improvement of technology, including advanced materials and nuclear fuels, remains central to industry's success. Decades of research combined with continual operation have produced steady advancements in technology and yielded an extensive base of data, experience, and knowledge on light water reactor (LWR) fuel performance under both normal and accident conditions. In 2011, following the Great East Japan Earthquake, resulting tsunami, and subsequent damage to the Fukushima Daiichi nuclear power plant complex, enhancing the accident tolerance of LWRs became a topic of serious discussion. As a result of direction from the U.S. Congress, the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) initiated an Accident Tolerant Fuel (ATF) Development program.

The complex multiphysics behavior of LWR nuclear fuel makes defining specific material or design improvements difficult; as such, establishing qualitative attributes is critical to guide the design and development of fuels and cladding with enhanced accident tolerance. This report summarizes a common set of technical evaluation metrics to aid in the optimization and down selection of candidate designs. As used herein, "metrics" describe a set of technical bases by which multiple concepts can be fairly evaluated against a common baseline and against one another. Furthermore, this report describes a proposed technical evaluation methodology that can be applied to assess the ability of each concept to meet performance and safety goals relative to the current UO_2 – zirconium alloy system and relative to one another. The resultant ranked evaluation can then inform concept down-selection, such that the most promising accident tolerant fuel design option(s) can continue to be developed for lead test rod or lead test assembly insertion into a commercial reactor within the desired timeframe (by 2022).

Executive Summary

The safe, reliable and economic operation of the nation's nuclear power reactor fleet has always been a top priority for the United States' nuclear industry. As a result, continual improvement of technology, including advanced materials and nuclear fuels, remains central to industry's success. Decades of research combined with continual operation have produced steady advancements in technology and yielded an extensive base of data, experience, and knowledge on light water reactor (LWR) fuel performance under both normal and accident conditions. In 2011, following the Great East Japan Earthquake, resulting tsunami, and subsequent damage to the Fukushima Daiichi nuclear power plant complex, enhancing the accident tolerance of LWRs became a topic of serious discussion. As a result of direction from the U.S. Congress, the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) initiated an Accident Tolerant Fuel (ATF) Development program. Prior to the events at Fukushima, the emphasis in the fuel development activities was on improving nuclear fuel performance in terms of increased burnup for waste minimization, increased power density for power upgrades, and increased fuel reliability.

The current nuclear power industry is based on mature technology and has an excellent safety and operational record. Except for a few extremely rare events, the current UO_2 – zirconium alloy fuel system meets all performance and safety requirements while keeping nuclear energy an economically competitive clean-energy alternative for the United States. Any new fuel concept must be compliant with and evaluated against current design, operational, economic, and safety requirements. The overall fuel cycle must also be considered, especially for concepts that represent a significant departure from the current technology.

After the March 2011 events at Fukushima, enhancing the accident tolerance of LWRs became a topic of discussion within the U.S. and internationally. In the Consolidated Appropriations Act, 2012, Conference Report 112-75, the U.S. Congress directed DOE-NE to give "priority to developing enhanced fuels and cladding for light water reactors to improve safety in the event of accidents in the reactor or spent fuel pools." Fuels with enhanced accident tolerance are those that, in comparison with the standard UO_2 – zirconium alloy system currently used by the nuclear industry, can tolerate loss of active cooling in the reactor core for a considerably longer time period during design-basis and beyond design-basis events (depending on the LWR system and accident scenario) while maintaining or improving the fuel performance during normal operations and operational transients.

The Fuel Cycle Research & Development (FCRD) Advanced Fuels Campaign (AFC) research, development and demonstration (RD&D) effort currently focuses on applications in currently operating reactors or reactors with design certifications. New fuel concepts will be evaluated with respect to the accident scenarios and specific plant designs for LWRs and fuel fabrication facilities. The candidate advanced fuel concepts must also be evaluated within the context of other potential improvements being developed to enhance overall safety (e.g., access to emergency cooling water, additional battery power, etc.) to fully characterize the impact of the candidate fuels on reactor operations. Overall safety assessments should be performed to the extent possible in the Phase I feasibility studies; these evaluations will be enhanced as more data becomes available in subsequent phases of development.

The accident tolerant fuels (ATF) development effort adopts a three-phase approach to commercialization (Figure ES-1). Phase 1 includes feasibility assessment and down-selection during which fuel concepts will be developed, tested, and evaluated. Feasibility assessments of the new concepts will be performed to reduce the number of concepts for further development. These assessments include: laboratory scale experiments, e.g., fabrication, preliminary irradiation, material properties measurements; fuel performance code updates; and analytical assessment of economic, operational, safety, fuel cycle, and environmental impacts. In Phase 2, the fabrication process will expand to industrial scale for lead test

rods (LTRs) and lead test assemblies (LTAs). Finally, Phase 3 establishes commercial fabrication capabilities. Each development phase roughly corresponds to the Technology Readiness Levels (TRL) defined for nuclear fuel development, where TRL 1-3 corresponds to the "proof-of-concept" stage, TRL 4-6 to "proof-of-principle," and TRL 7-9 to "proof-of-performance" (INL 2013; Carmack 2014). The draft evaluation methodology presented in this document focuses on Phase 1 assessment and down-selection, but it also establishes the framework necessary to move a new fuel concept through further development and analysis in Phase 2.

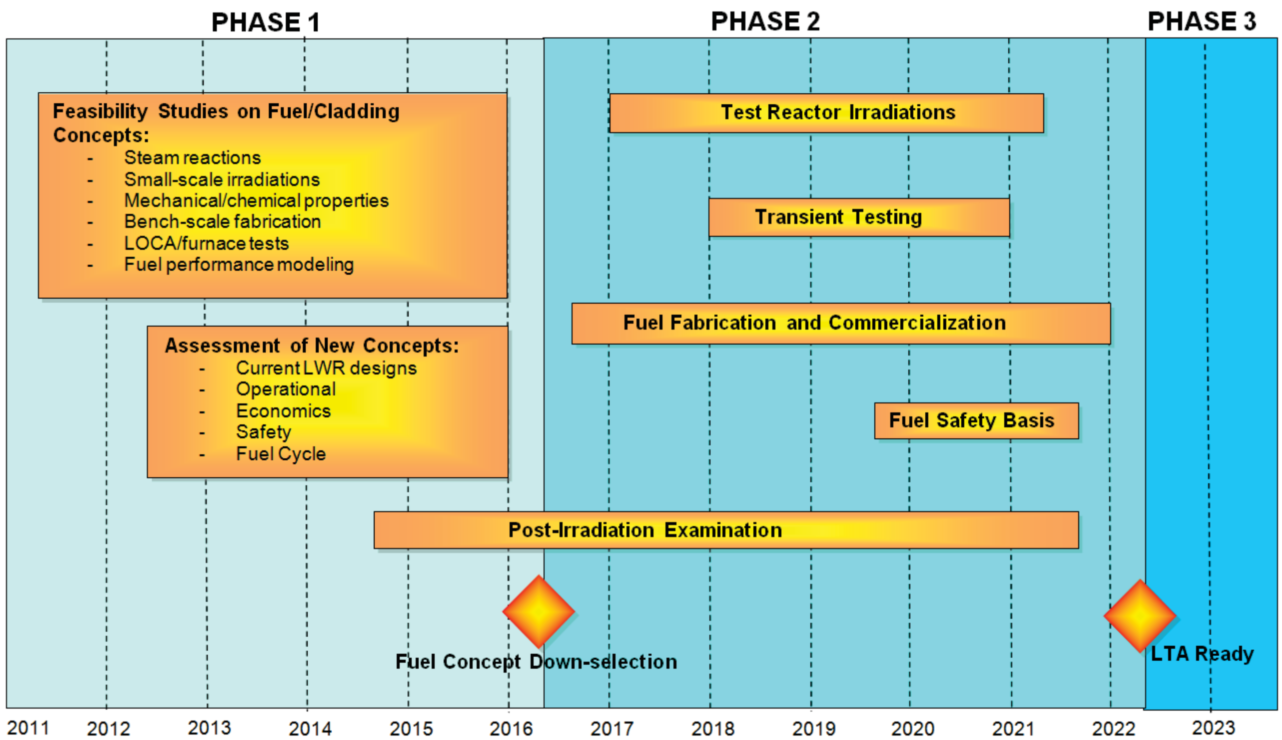

Figure ES-1. Research, development, and demonstration strategy for enhanced accident tolerant fuel development; an estimated timeline for each phase is included.

Attributes with potential impact for fuel designs with enhanced accident tolerance include reduced steam reaction kinetics, lower hydrogen generation rate (or generation of other combustible gases), and reduction of the initial stored energy in the core, while maintaining acceptable cladding and fuel thermo-mechanical properties, fuel-clad interactions, and fission-product behavior. Targeting improvements in these attributes provides guidance in establishing the critical parameters that must be considered in the development of fuels and cladding with enhanced accident tolerance. A common set of qualitative technical metrics will aid in the optimization and down-selection of candidate designs on a more quantitative basis.

"Metrics" describe a set of technical bases by which multiple concepts can be fairly evaluated against a common baseline and against one another. In some cases this may equate to a specific quantitative target value for selected properties or behaviors. "Metrics" can also describe a clear technical methodology for evaluation that can be used to rank two or more concepts. Because of the complex multiphysics behavior of nuclear fuel and the large set of performance requirements that must be met, the latter definition is adopted for the current evaluation of candidate accident tolerant fuel options. A series of national and international meetings were held in FY2013 to begin establishing a consensus on how to approach ATF design, optimization and evaluation for down-selection (Braase 2013; Braase and Bragg-Sitton 2013; OECD/NEA 2013). Each of these meetings provided expert direction on an appropriate set of enhanced accident tolerant fuel attributes, metrics, and associated screening evaluations for different classes of fuel and cladding material.

Beginning with the guidance on qualitative ATF metrics, a small team from across the DOE laboratories has begun to define a technical evaluation approach for accident tolerant fuels. Preliminary analyses have been conducted on a handful of ATF concepts, and sensitivity analyses have been performed for key performance parameters, including fuel thermal conductivity, cladding thermal conductivity, and cladding oxidation rate. Sensitivity analyses provide early insight to the key parameters to be used for cladding and fuel optimization, while more detailed analyses allow evaluation of currently proposed concepts based on the available property and performance data.

An assessment of the potential beneficial impact or unintended negative consequences of candidate ATF concepts must address the obvious "fuel-specific" characteristics of the concept but, perhaps more importantly, the assessment must address how implementation of the concept will affect reactor performance and safety characteristics. This assessment would include neutronics and thermal-hydraulics analyses to ensure that the reactor would operate as intended with the candidate fuel system. Coupled thermal hydraulic-neutronic analysis of candidate ATFs is essential for understanding the synergistic impact of the thermal properties and reactivity feedback.

Industry, academia and the DOE national laboratories are currently investigating multiple ATF concepts. The proposed technical evaluation methodology will be applied to gauge the ability of each of these concepts to meet performance and safety goals relative to the current UO_2 – zirconium alloy system and relative to one another. This ranked evaluation will enable the continued development of the most promising ATF design options given budget and time constraints, with a goal of inserting one (or possibly two) concepts as an LTR or LTA in a commercial LWR by 2022.

The steps described in Figure ES-2 (numbered 1 – 8) address the full scope of activities that need to be considered in evaluating the feasibility of candidate ATFs. Carrying out all the indicated steps can require significant investment of time and resources. However, the fidelity or level of detail involved depends on the stage of evolution/development of a concept. During the "screening" stage (Step 1), the level of detail associated with analyses will be limited, based on the current state of knowledge for the selected concept. The level of detail may range from literature reviews and expert judgment through limited experiments and computational analyses. The goal is to have sufficient confidence in the results of the assessment (with a reasonable investment of time and resources) that identified changes relative to the reference UO_2 – zirconium alloy fuel system are known well enough to proceed with continued development of the concept, or conclude that the concept should be modified or abandoned. For further description of the corresponding fuel technology readiness level related to Figure ES-2, see *Technology Readiness Levels for Advanced Nuclear Fuels and Materials Development* (Carmack 2014).

The design / development team for a candidate fuel system would be expected to rely on preliminary analyses and scoping studies performed as a part of the initial technology selection to complete the candidate fuel screening table (Table ES-1). The attributes defined in the table correspond to a complete fuel system (fuel plus cladding) over the full fuel life cycle (fabrication – operation – used fuel management). Table ES-1 is designed to identify "Benefits" and "Vulnerabilities" for each concept. The candidate fuel system should be ranked within each category on a scale of 0 to 5, where "0" indicates no notable change from the current UO_2 – zirconium alloy system and "5" indicates either a significant benefit relative to the current system or a significant vulnerability. If a specific property or behavior is not yet known (or not known conclusively), this would correspond to higher vulnerability. For attributes that are currently unknown or assumed, recommended actions should then be noted by the review panel. If a decision is made to continue development to Steps 2a, 2b despite these noted vulnerabilities, these recommended actions should be resolved before convening the second expert review panel that would be tasked with ranking the remaining candidate technologies (shown between Phase 1 [Steps 1, 2a, 2b] and Phase 2 [Steps 3-7]) based on both perceived benefits and remaining vulnerabilities. Concepts that are further in their development may have already completed elements listed in items 2a and 2b prior to the initial expert panel review, while others may have larger uncertainties associated with the expected fuel system behavior. The former case would have fewer recommended actions, while the latter may have more recommended actions necessary to reduce uncertainty in the performance estimates.

The independent Initial Expert Panel Review will be tasked with reviewing input (currently available data and analysis results) provided by each of the design teams and using this information to make a qualitative assessment of the relative benefits or vulnerabilities associated with the candidate design for each "Performance Attribute" relative to the specified "Performance Regimes:"

1. Fabrication / Manufacturability (to include Licensibility)
2. Normal Operation and Anticipated Operational Occurrences (AOOs)
3. Postulated Accidents (Design Basis)
4. Severe Accidents (Beyond Design Basis)
5. Used Fuel Storage / Transport / Disposition (to include potential for future reprocessing)

Upon completion of Table ES-1, the scores in the "benefits" and "vulnerabilities" columns should be tallied. The scores in each of these columns provide an indication of both development stage and expected performance benefits. Technologies that do not depart significantly from the current UO_2 – Zr alloy fuel system might be expected to have a modest benefit score and low vulnerability. This candidate technology would be considered low risk, modest payoff, but could potentially be developed in the near term for commercial demonstration. On the contrary, a relatively immature concept that is a significant departure from the current UO_2 – Zr system may have a high vulnerability score due to the existing technology or data gaps, while simultaneously scoring well in the benefits column based on the limited data that are available. For example, candidate ceramic cladding may score very well due to its ability to operate to very high temperature and minimal interaction with steam, but challenges in fabricating a hermetic fully ceramic cladding in the lengths necessary for LWR application may result in high vulnerability scores and a large number of recommended actions. A concept scored with a high benefit and high vulnerability would be considered high risk but potentially high payoff, requiring a longer period of time for development relative to technologies that are nearer to the current UO_2 – Zr system.

The same screening table (Table ES-1) can be applied in both the initial and secondary expert panel reviews. The level of uncertainty in each of the performance attributes would be expected to decrease at each evaluation stage, allowing quantitative estimates for some of the behaviors of interest as more property data become available. To continue with the above example of a high risk, potentially high payoff technology, the vulnerability score and number of recommended actions would be expected to decrease significantly between these reviews to allow a more informed decision to proceed (or not proceed). Note that there is also provision for "off-ramps" prior to a secondary expert review should a concept design or specific material demonstrate that it cannot meet the minimum performance requirements during the fundamental scoping tests (Step 2a) or core level analysis (Step 2b).

The convened expert review panel would be comprised of technology experts selected based on their knowledge of the technologies under review. This review panel should include experts specializing in materials (metals and ceramics), neutronics, thermal-hydraulics, and severe accidents. The totaled "benefits" and "vulnerabilities" scores for each technology will result in a ranked, prioritized list of candidate technologies. The review panel may choose to develop two ranked lists, one for near-term technologies, fitting within the defined 10-year development window, and a second for longer term technologies that appear to have a significant benefit at this early development stage but are unlikely to meet the defined development timeframe. The number of technologies selected to proceed for additional testing and development will be dependent on budget availability. It is anticipated that a few technologies will continue development beyond the first expert panel review, but only one or two would proceed beyond the second panel review. Upon completion of Phase 2, a lead test rod or assembly would be fabricated for industry testing in a currently operating light water reactor.

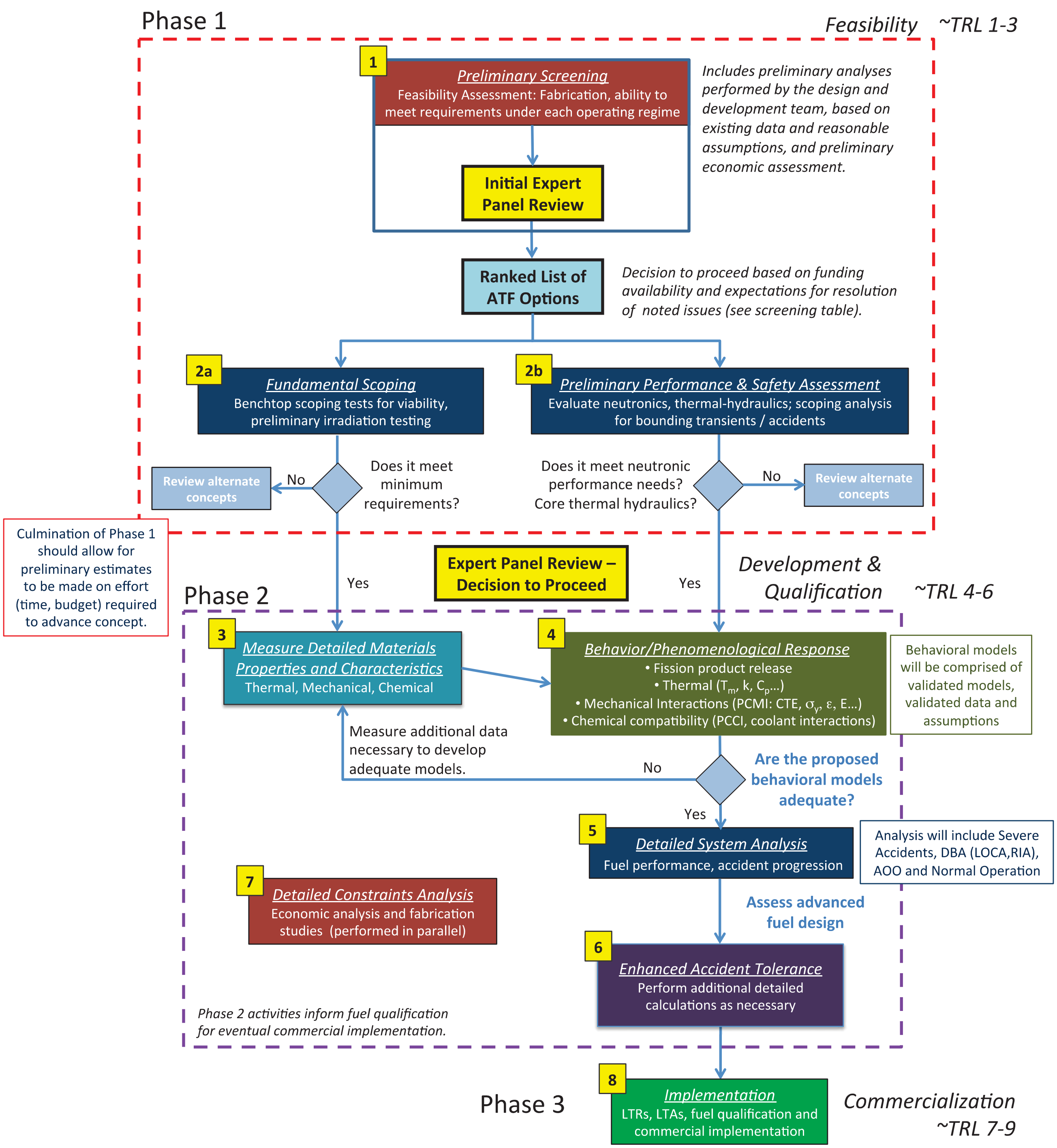

Figure ES-2. Proposed Accident Tolerant Fuel evaluation methodology. Preliminary concept down-selection would occur within Step 1 (see step numbers 1-8 noted), with secondary down-selection occurring at the end of Phase 1 prior to detailed tests and behavior model development.

Table ES-1. Candidate Fuel Screening Attributes Assessment Table

Performance Regime	Performance Attributes (For large-scale deployment)	Expert Opinion Assessment		Recommended Actions
		Benefit	Vulnerability	
Fabrication/Manufacturability *Considerations:* Millions ft of clad/year ~300 million pellets/year Economics - cost of raw materials and fabrication process Current fabrication plant enrichment limits	Manageable fissile material content			
	Compatible with large-scale production needs (material availability, fabrication techniques, waste, etc.)			
	Compatible with quality and uniformity standards			
	Licensibility			
Normal Operation and AOOs *Considerations:* Overall neutronics Linear Heat Generation Rate (LHGR) to centerline melt Power ramp, ~100 W/m/min Reduced flow (departure from nucleate boiling, DNB) Flow-induced vibrations Surface roughness effects Safe shutdown - earthquake External pressure (~2750 psi, 10% above PWR design pressure) Axial growth (less than upper nozzle gap)	Utilization or Burnup (12, 18, or 24 month/cycle)			
	Thermal hydraulic interaction			
	Reactivity control systems interaction			
	Mechanical strength, ductility (beginning of life and after irradiation)			
	Thermal behavior (conductivity, specific heat, melting)			
	Chemical compatibility (fuel-cladding) / stability			
	Chemical compatibility with and impact on coolant chemistry			
	Fission product behavior			
Postulated Accidents (Design Basis) *Considerations:* Prompt reactivity insertion Post-DNB behavior (T > 800^{o}C for Zr-UO_2 system) Loss of coolant conditions Thermal shock Steam reactions (~1000^{o}C +)	Thermal hydraulic interaction			
	Mechanical strength and ductility			
	Thermal behavior (conductivity, specific heat, melting)			
	Chemical compatibility/ stability (e.g. oxidation behavior)			
	Fission product behavior			
	Combustible gas production			
Severe Accidents (Beyond Design Basis) *Considerations:* Thermal shock Chemical reactions Combustible gas release Long-term stability in degraded state	Mechanical strength, ductility			
	Thermal behavior (conductivity, specific heat, melting)			
	Chemical compatibility/ stability (including high temperature steam interaction)			
	Fission product behavior			
	Combustible gas production			
Used Fuel Storage/ Transport/ Disposition *Considerations:* Handling, placement, and drying loads; future reprocessing potential	Mechanical strength, ductility			
	Thermal behavior			
	Chemical stability			
	Fission product behavior			

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Acronyms

AFC	Advanced Fuels Campaign
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
AOO	anticipated operational occurrences
ASTM	American Society for Testing and Materials
ATF	accident tolerant fuel
ATR	Advanced Test Reactor
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox
BAF	bottom of active fuel
BDBA	beyond design basis accident
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CDF	core damage frequency
CFR	Code of Federal Regulations
CRD	control rod drive
CSAU	code scaling, applicability, and uncertainty
CTE	coefficient of thermal expansion
DBA	design basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DOE-NE	U.S. Department of Energy, Office of Nuclear Energy
ECCS	emergency core cooling system
ECR	extent of cladding reacted
EFPD	effective full power days
FCCI	fuel-cladding chemical interaction
FCMI	fuel-cladding mechanical interaction
FCRD	Fuel Cycle Research and Development
FCT	fuel centerline temperature
FOA	Funding Opportunity Announcement
FPR	fission product retention
GE	General Electric
GEN-III+	Generation III+ reactors
hcp	hexagonal close-packed
HPCI	high-pressure coolant injection
IASCC	irradiation assisted stress corrosion cracking
INL	Idaho National Laboratory
IRP	Integrated Research Projects
LANL	Los Alamos National Laboratory

LBLOCA	large break loss-of-coolant accident
LEU	low enriched uranium
LHR	linear heat rate
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LTA	lead test assembly
LTR	lead test rod
LTSBO	long-term station blackout
LWR	Light Water Reactor
MCNP	Monte Carlo N-Particle
MCPR	minimum critical power ratio
MOX	molybdenum oxide
MSTSBO	mitigated short-term station blackout
NEUP	Nuclear Energy University Programs
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PCCI	pellet-clad chemical interaction
PCMI	pellet-clad mechanical interaction
PCT	peak cladding temperature
PIE	post-irradiation examination
PNNL	Pacific Northwest National Laboratory
PQD	post quench ductility
PWR	pressurized water reactor
PZR	pressurizer
R&D	research and development
RD&D	research, development and demonstration
RCCA	rod cluster control assembly
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RIA	reactivity initiated accident
RPV	reactor pressure vessel
S/G	steam generator
SATS	Severe Accident Test Station
SBLOCA	small break loss-of-coolant accident
SBO	station blackout
SCC	stress corrosion cracking
SNL	Sandia National Laboratories
SRV	safety relief valve
SSC	systems, structures, and components
STSBO	short-term station blackout
TAF	top of active fuel
TD	theoretical density
TMI	Three-Mile Island

TREAT	Transient Reactor Test Facility
TRL	technology readiness level
Zr-4	Zircaloy-4

Advanced Fuels Campaign Enhanced Light Water Reactor Accident Tolerant Fuel Performance Metrics

1. Introduction

The safe, reliable and economic operation of the nation's nuclear power reactor fleet has always been a top priority for the United States' nuclear industry. As a result, continual improvement of technology, including advanced materials and nuclear fuels, remains central to industry's success. Decades of research combined with continual operation have produced steady advancements in technology and yielded an extensive base of data, experience, and knowledge on light water reactor (LWR) fuel performance under both normal and accident conditions. In 2011, following the Great East Japan Earthquake, resulting tsunami, and subsequent damage to the Fukushima Daiichi nuclear power plant complex, enhancing the accident tolerance of LWRs became a topic of serious discussion. As a result of direction from the U.S. Congress, the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) initiated an Accident Tolerant Fuel (ATF) Development program. Prior to the events at Fukushima, the emphasis in the fuel development activities was on improving nuclear fuel performance in terms of increased burnup for waste minimization, increased power density for power upgrades, and increased fuel reliability.

After the March 2011 events at Fukushima enhancing the accident tolerance of LWRs became a topic of serious discussion. In the Consolidated Appropriations Act, 2012, Conference Report 112-75, the U.S. Congress directed DOE-NE to give "priority to developing enhanced fuels and cladding for light water reactors to improve safety in the event of accidents in the reactor or spent fuel pools." Fuels with enhanced accident tolerance are those that, in comparison with the standard UO_2-zirconium alloy system currently used by the nuclear industry, can tolerate loss of active cooling in the reactor core for a considerably longer time period during design-basis and beyond design-basis events (depending on the LWR system and accident scenario) while maintaining or improving the fuel performance during normal operations and operational transients.

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"Metrics" describe a set of technical bases by which multiple concepts can be fairly evaluated against a common baseline and against one another. In some cases this may equate to a specific quantitative target value for selected properties or behaviors. "Metrics" can also describe a clear technical methodology for evaluation that can be used to rank two or more concepts. Because of the complex multiphysics behavior of nuclear fuel and the large set of performance requirements that must be met, the latter definition is adopted for the current evaluation of candidate accident tolerant fuel options. This report describes a proposed technical evaluation methodology that can be applied to evaluate the ability of each ATF concept to meet performance and safety goals relative to the current UO_2 – zirconium alloy system and relative to one another. The resultant ranked evaluation can then inform concept down-selection, such that the most promising ATF design option(s) can continue to be developed toward qualification given budget and time constraints.

1.1 Challenges in Developing Accident Tolerant Fuel

Development and qualification of nuclear fuel is a well-established process. However, due to the scientific and engineering challenges associated with nuclear technology, along with the conservative approach to adopting new technology, fuel qualification is a long, complicated process. The ATF development effort adopts a three-phase approach to commercialization, as illustrated in Figure 1. Each development phase roughly corresponds to the Technology Readiness Levels (TRL) defined for nuclear fuel development, where TRL 1-3 corresponds to the "proof-of-concept" stage, TRL 4-6 to "proof-of-principle," and TRL 7-9 to "proof-of-performance" (INL 2013; Carmack 2014). The draft evaluation methodology presented in this document focuses on Phase 1 assessment and down-selection, but it also establishes the framework necessary to move a new fuel concept through further development and analysis in Phase 2. Each step in the fuel development effort will necessarily consider the requirements set out by the Nuclear Regulatory Commission (NRC) to support the fuel licensing effort in planning, executing and documenting all phases of the fuel development, including fabrication, experiments, model development and validation, etc.

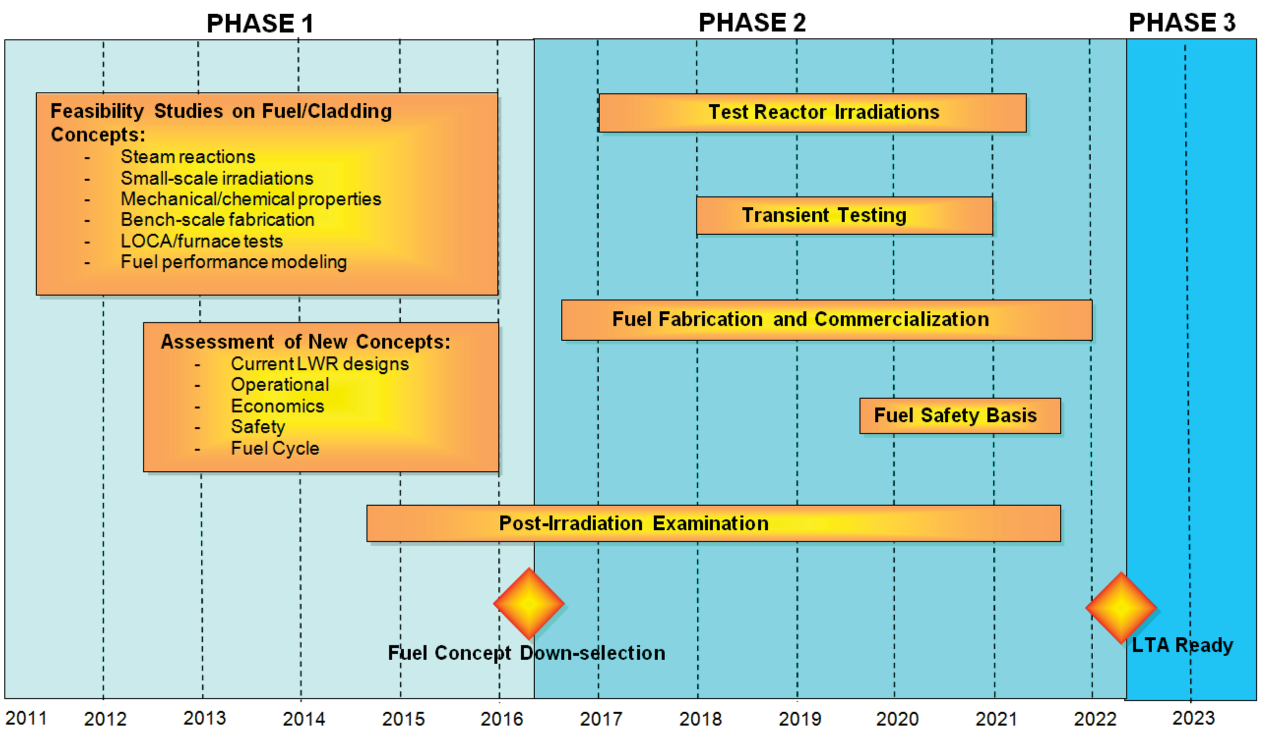

Figure 1. Research, development, and demonstration strategy for enhanced accident tolerant fuel development; an estimated timeline for each phase is included.

PHASE 1: Feasibility Assessment and Down-Selection

Feasibility assessment focuses on obtaining data from initial small-scale and phenomenological testing in order to conduct an informed down-selection of concepts. This work includes activities such as: laboratory scale experiments, e.g., fabrication, preliminary irradiation, material properties measurements; fuel performance code updates for specific concepts, applying measured property and behavior data; and analytical assessment of economic, operational, safety, fuel cycle, and environmental impacts. Fuel performance codes will be used during this phase to the degree the relevant fuel and cladding property measurements and/or models are available for the various concepts. Analytical assessments will be performed during this phase to evaluate promising concepts against the defined attributes for ATF.

Within the DOE ATF development efforts, collaboration between DOE, the nuclear industry, utilities, and others, including the international community, is an important first step. This phase, led by DOE, includes formation of teams made up of the nuclear industry/utilities, national laboratories, universities, and international partners. These teams will work in close coordination to develop and evaluate fuel concepts using the evaluation methodology described in this report to allow down-selection of promising concepts for further development.

PHASE 2: Development and Qualification

During this phase, the fabrication process will expand to industrial scale, and fabrication of lead test assemblies (LTAs) or lead test rods (LTRs) will occur. Requirements for LTA/LTR testing will be established during the development phase. If the assembly design differs substantially from that of currently used UO_2 – zirconium alloy assemblies, the qualification will likely require testing of a full assembly. If the assembly design is similar to that of the current design, a few LTRs incorporated into a fuel assembly containing UO_2 – zirconium alloy rods may be sufficient for qualification.

Test reactor irradiation using long rodlets (about 36-inch-long fuel column) will cover fabrication variations, temperature, and linear heat-rate limits. Characterization, post-irradiation examination (PIE), and the development of a fuel performance code will be part of the qualification process. Sufficient testing must be completed to establish the statistical database. By 2018 a transient testing capability in a water loop will also need to be established. Transient experiments on unirradiated and irradiated rodlets will begin in ≈FY 2018 to establish fuel-failure modes and failure margins.

At the end of this phase, LTRs or LTAs will be fabricated and the safety basis for irradiation in a commercial reactor will be completed. The irradiation and subsequent PIE of the LTRs / LTAs will complete the demonstration phase for LWR fuels with enhanced accident tolerance.

PHASE 3: Commercialization

The commercialization phase entails the establishment of commercial fabrication capabilities and the conversion of LWR cores to the new fuel. This phase will primarily be a commercial activity performed by industry.

1.2 Current Development Phase

This LWR ATF guidelines document proposes measurements, tests and calculations necessary during the ***feasibility assessment phase*** (Phase I) to guide the down-selection among the various ATF concepts currently proposed. Beginning with expert guidance on ATF metrics (Braase 2013), a small team from across the DOE laboratories has begun to define a technical evaluation approach for accident tolerant fuels. Rather than establishing specific quantitative targets for individual performance parameters (e.g. thermal conductivity, melting temperature, oxidation resistance, combustible gas production, etc.), these guidelines present a common set of technical evaluation measures that can be applied by a panel of expert reviewers. In this manner, ATF can be designed to target quantitative improvement in several interrelated areas (e.g. reduction in the oxidation rate, or increased fuel and cladding thermal conductivity relative to the current fuel system) without being constrained to specific quantitative targets individually for each performance parameter. It is possible that improvements in one area may negatively impact other areas of performance due to the complex interaction of the various fuel performance parameters; a broad, multi-variable evaluation approach should be capable of identifying these instances.

An assessment of the potential beneficial impact or unintended negative consequences of candidate ATF concepts must address the obvious "fuel-specific" characteristics of the concept but, perhaps more importantly, the assessment must address how implementation of the concept will affect reactor performance and safety characteristics. This assessment would include neutronics and thermal-hydraulics analyses to ensure that the reactor would operate as intended with the candidate fuel system. Coupled thermal hydraulic-neutronic analysis of candidate ATFs is essential to understanding the synergistic impact of the thermal properties and reactivity feedback. Variations in the fuel performance characteristics relative to the current UO_2 – zirconium alloy system will likely result in variations in the manner in which the core is operated under normal and off-normal conditions, such that a core level analysis must be completed to assess the overall performance and safety improvements that might be offered by a candidate ATF.

Industry, academia, and the DOE national laboratories are currently investigating multiple ATF concepts; various international laboratories and organizations are also developing concepts. The proposed technical evaluation methodology can be applied to gauge the ability of each of these concepts to meet performance and safety goals relative to the current UO_2 – zirconium alloy system and relative to one another. A ranked evaluation will enable the continued development of the most promising ATF design options given budget and time constraints, with a goal of inserting one (or possibly two) concepts as an LTR or LTA in a U.S. commercial LWR by 2022.

1.3 Proposed Evaluation Approach

The proposed evaluation approach independently addresses each "performance regime" of interest, including:

1. Fabrication / Manufacturability (to include Licensibility)
2. Normal Operation and Anticipated Operational Occurrences (AOOs)
3. Postulated Accidents (Design Basis)
4. Severe Accidents (Beyond Design Basis)
5. Used Fuel Storage / Transport / Disposition (to include potential for future reprocessing)

If preliminary evaluations suggest that a new concept has the potential to provide technical improvements in safety and performance during the operational, postulated accident, and severe accident regimes, the economic impact of the concept must be evaluated to determine commercial viability. Economic assessment must include potential material and fabrication costs; costs associated with uranium enrichment requirements above 5% (must consider costs associated with any required modifications to fuel fabrication facilities and associated licenses to accommodate higher enrichment); modifications to reactor facility infrastructure; post-irradiation activities (storage, transportation and disposition; possibility for future reprocessing); etc. While all new fuel concepts will incur significant qualification and licensing costs, they will demonstrate different economics upon commercialization based on per-unit costs in addition to the impact on reactor operations and fuel cycle. If all other things are equal between the fuels being considered (e.g., cycle length [18 months minimum is suggested], thermal power, fuel bundle design), evaluation should include such comparison as mass of ^{235}U per megawatt-day thermal. At the end of the feasibility assessment phase a concept will be elevated to TRL 3 or 4, and sufficient information should be available to estimate the time and development cost associated with research, development and demonstration of a proposed concept to elevate it to TRL 6 by the end of Phase 2.

2. Accident Tolerant Fuel Definition

Enhanced, accident tolerant fuels are defined as fuels that can tolerate a severe loss of active cooling in the reactor core for a considerably longer time period than the current UO_2 – zirconium alloy fuel system, while maintaining or improving the fuel performance during normal operations and operational transients. This section defines key ATF attributes and the associated constraints to the development and deployment of ATF. The current performance levels for the UO_2 – zirconium alloy system, which provide a baseline for ATF evaluation, are discussed in detail in Appendix A. The associated Nuclear Regulatory Commission (NRC) guidelines for evaluating fuel system performance are also presented, as these guidelines provide the framework for the quantitative evaluation of fuel performance. In some cases, the existing regulations are specific to the limitations of the UO_2 – Zr alloy system. Hence, limits may need to be reevaluated in light of the characteristics of the proposed fuel. The NRC is currently exploring performance-based, technology-independent regulations that would be more easily applied to new concepts.

2.1 Attributes for Enhanced Accident Tolerant Fuel

To mitigate or reduce the consequences of fuel failure due to steam exposure at elevated temperatures that may occur during the progression of a severe accident, the attributes identified in the following subsections are considered. These attributes provide qualitative guidance for developing fuels with enhanced accident tolerance. It may be unnecessary or impossible to improve on all attributes. It is also likely that some attributes or combination of attributes will provide meaningful gains in accident tolerance, while others may provide only marginal benefits. Additionally, the currently described potential consequences of some accident conditions (e.g. hydrogen generation) may be specific to the current UO_2 – zirconium alloy fuel system. Other fuel and cladding systems could present additional effects not expressed here (e.g. generation of CO) that also must be considered in evaluation of the proposed system. A brief description of the desired ATF attributes is provided in this section and summarized in Figure 2.

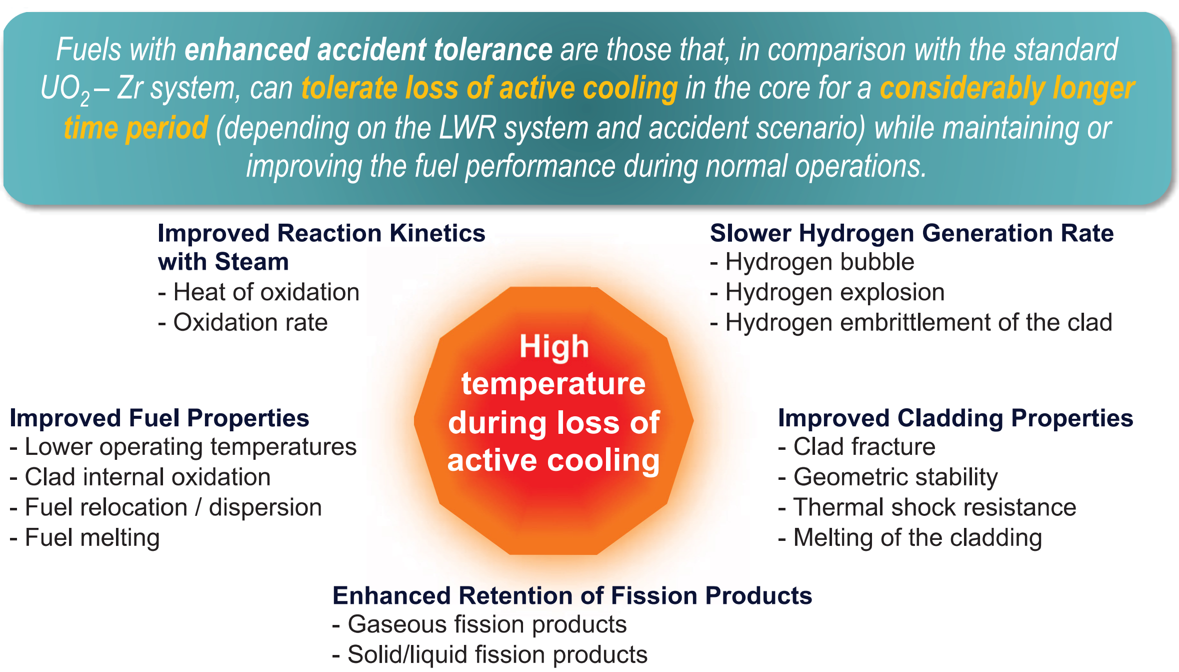

Figure 2. Major issues that need to be addressed in establishing accident tolerant fuel attributes.

Hydrogen Generation Rate (generation of combustible gases)

Hydrogen production during a severe accident can lead to energetic explosions as seen in the Fukushima events. Under a high-temperature steam environment that can result in rapid oxidation of cladding, it is not possible to totally avoid hydrogen generation or generation of other combustible gases. Important parameters include both the heat of oxidation and the oxidation rate. The result of reducing both of these parameters is a decreased rate of gas generation, allowing additional time for mitigation strategies to be implemented. If a severe accident is allowed to proceed without mitigation (such that all of the cladding material is allowed to interact with the steam) it is likely that any proposed cladding option would produce essentially equivalent volumes of hydrogen (Farmer et al. 2014).

Rapid oxidation of standard zirconium alloy cladding results in prolific hydrogen generation. The exothermic nature of the oxidation reaction increases the cladding temperature, which further accelerates hydrogen generation. A related issue is the diffusion of free hydrogen into the cladding metal, resulting in enhanced embrittlement and potential cladding failure.

A desirable accident tolerant cladding material would resist oxidation or experience a reduced oxidation rate relative to zirconium alloys, thereby resulting in a slower free hydrogen generation rate. Materials with lower heat of oxidation may be important in limiting the temperatures during an accident, and materials that are less susceptible to hydrogen diffusion may address the rapid embrittlement issue. More generally, a desirable accident tolerant fuel system would exhibit a reduced generation rate of any combustible gas, such as hydrogen, carbon monoxide, etc.

Fission Product Retention

Fission product retention (FPR) is a function of both fuel and cladding properties. Cladding provides the initial barrier to fission products in nuclear fuel. Upon cladding failure, retention of the fission products (gaseous and solid) within the vessel is required to minimize releases to the environment. Due to the potential severity of fission product release to the environment, retention within the fuel is of the utmost importance. While total retention may not be possible in the event of a severe accident, even partial retention (especially for highly mobile fission products) would provide a substantial safety margin.

An ATF design goal is to prevent melting or dispersion of the fuel by utilization of high temperature, high strength cladding materials that would retain cladding integrity beyond the current limitations of zirconium alloy cladding. Alternately, a desirable concept may have less stored energy due to high thermal conductivity, such that the material temperature would remain much lower during both normal and accident conditions. Fission product retention techniques or chemically linking the fission products in a fuel matrix may also be options, as long as the concepts can tolerate high temperatures. Building additional barriers around the fuel to contain fission products (as a backup to containment provided by the cladding) have also been envisioned.

Cladding Reaction with Steam

Multiple issues need to be considered if cladding is exposed to steam at high temperature during the course of a severe accident. As previously discussed, the high temperature steam interaction with zirconium alloy cladding causes an exothermic oxidation reaction and resulting hydrogen generation. In addition, this reaction deteriorates the structural integrity of the cladding and could result in subsequent fission product release into the reactor vessel.

Advanced cladding materials should demonstrate enhanced tolerance to radiation under normal operating conditions and to oxidation under extreme high-temperature exposure while specifically considering mechanical strength and structural integrity at the end of life (at maximum burnup) and when exposed to high-temperature steam for an extended duration. Evaluation of ATF concepts should address all possible reaction products, noting that oxidation of some of the proposed concepts could result in combustible gases other than hydrogen.

Fuel Cladding Interactions

In the event of cladding failure, fuel behavior is important. Key concerns are fuel melting and relocation, as well as fuel dispersion into the coolant. Fuel-cladding chemical interactions (FCCIs), fuel-cladding mechanical interactions (FCMIs) and fuel heating are important properties that must be understood during normal operation and accident conditions.

Advanced fuel designs should consider fuel systems that would experience reduced FCCI and FCMI, and have lower operating temperatures and stored energy relative to the UO_2 – zirconium alloy system (e.g. via increased thermal conductivity). Higher melting point and structural integrity at high temperatures (i.e. less dispersive) are also desired.

2.2 Constraints on Development of Enhanced Accident Tolerant Fuel

The current nuclear power industry is based on mature technology and has a stellar safety and operational record. The current UO_2 – zirconium alloy fuel system meets all performance and safety requirements while keeping nuclear energy an economically competitive, clean-energy alternative. With the exception of a few extremely rare events, the current LWR fuel has performed exceptionally well. Any new fuel concept proposed for enhanced accident tolerance under rare events must be compliant with and evaluated against current design, operational, economic, and safety requirements. The complete fuel cycle also must be considered, especially for concepts that represent a significant departure from the current technology. Candidate advanced fuel concepts should also be evaluated within the context of other potential improvements being developed to enhance overall safety (e.g., access to emergency cooling water, additional battery power, etc.) to fully characterize the impact of the candidate fuels on reactor operations. A brief summary of the constraints associated with commercial nuclear fuel development and deployment is provided below and summarized in Figure 3. Details on the performance characteristics of the current UO_2 – zirconium alloy fuel system are included in Appendix A.

2.2.1 Current LWR Designs

In order to meet the desired development timeline, advanced fuel and/or cladding concepts developed under this initiative must be suitable for use in existing LWRs or reactor concepts with design certifications (GEN-III+). Longer term concepts may be considered in conjunction with the near-term focus as resources permit. Regardless of whether the actual deployment target is a current or future reactor, the fuel is likely to be qualified and initially demonstrated in existing commercial reactors. Proposed fuel concepts should not require plant modifications to the host reactor for demonstration irradiation and should be as near prototypic as possible. This constraint includes compatibility with existing fuel handling equipment, fuel rod or assembly geometry, and co-resident fuel in the reactor. If these advanced fuels are eventually deployed in existing commercial reactors it is likely that they will be phased in over multiple reloads and, hence, must be capable of operating in concert with the existing reactor fuel.

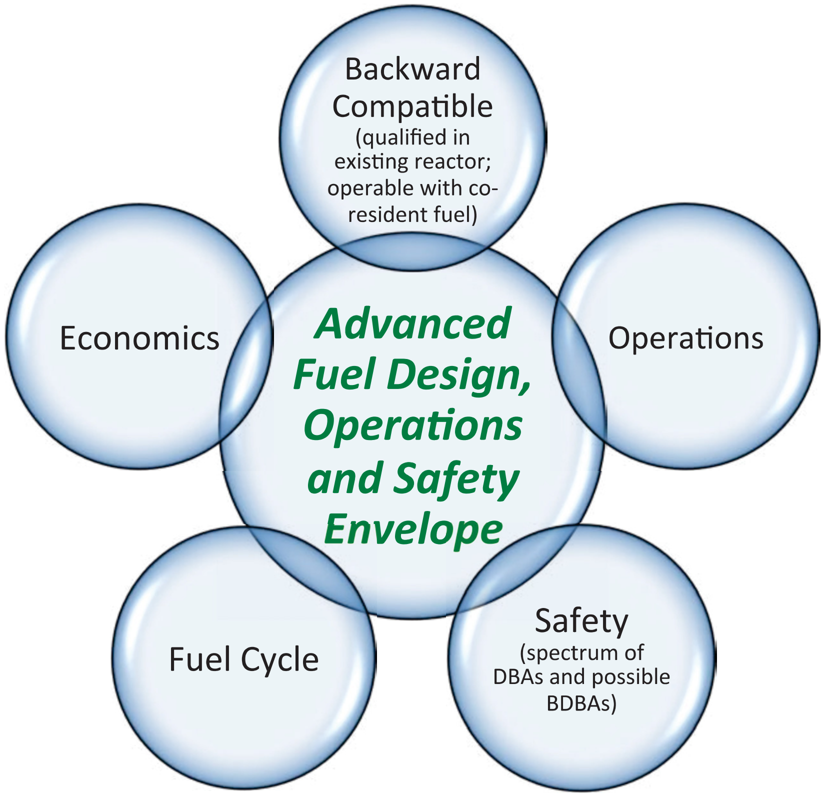

Figure 3. Considerations that constrain new fuel designs.

2.2.2 Operational Considerations

Before introducing a new fuel into an existing, or planned, reactor system, the potential impact on plant operations must be considered. The new fuel system must maintain or extend plant operating cycles, reactor power output, and reactor control. Reducing the availability or power output would be disruptive to utilities that would not readily accept this change unless the benefits outweigh the lost productivity. To maintain current operation, some of the fuel system concepts would require higher fuel enrichment. While the impact of higher enrichments is fairly well understood from a technical perspective, regulatory issues would have to be addressed both for the fuel fabrication facility and the operating plant, including handling, transportation and storage issues. With regard to lead test rod or assembly testing, the operating utilities will strongly prefer structure and materials that are similar to the co-resident fuel to ensure that there is no impact on the core thermal hydraulics, operating performance or safety. Utility participants in the metrics development meetings have stressed that the primary mission of commercial power reactors is to produce electricity for their customers, not to test new fuels. Plant operators should be included in ATF feasibility evaluations to ensure that utility concerns are addressed in a satisfactory manner.

2.2.3 Economic Impacts

After decades of development and optimization, the UO_2 – zirconium alloy fuel system is a streamlined technology that accounts for a relatively small percentage of the overall nuclear electricity production cost. Any proposed fuel system is unlikely, at least initially, to be able to compete economically with the current system. Fuels that require enrichment higher than that of current fuel (about 4 to 4.5%) are especially likely to cost more because enrichment is a major cost contributor. Increased enrichment could additionally require modifications to fuel fabrication facilities and fuel handling / transport operations (including license modification). Therefore, it is important to carefully and fully assess the economic impact of the new technology and to determine how much additional fuel cost the utilities will accept. As a potential solution, higher burnup (extended cycle with reduced waste and reduced refueling cost) and higher power densities (power upgrades) could mitigate some of the impact. While focusing on enhanced safety it is imperative to maintain enhanced performance goals as an economic consideration.

2.2.4 Safety Envelope

Performance of a new fuel system will be compared to the performance of the UO_2 – zirconium alloy fuel system to assess its accident tolerance. Operational transients and design-basis accidents must be considered in evaluating the new fuel system. Specific emphasis will be placed on long-term station blackout (LTSBO), loss-of-coolant accidents (LOCAs), and reactivity insertion accidents (RIAs) in the proposed analyses. Fuel performance during anticipated transients without scram must also be evaluated and must be shown to be similar to or better than the current system.

For design-basis LOCAs, the U.S. NRC is currently in the process of evaluating the safety envelope for high burnup fuels (>45 Giga-Watt-days/metric ton). Because some of the issues are similar to those that need to be addressed by proposed new fuel systems, this is an opportunity to closely work with NRC on assessment methodologies.

2.2.5 Fuel Cycle Impacts

The impact of new fuels and cladding on the front-end of the nuclear fuel cycle must be carefully assessed within the framework of current and future regulations and policies. Some of the fuel systems that will be considered require higher fuel enrichment. For instance, if an advanced stainless steel cladding replaces zirconium-based alloys, the required enrichment could increase by 1 to 2% (referred to as the U-235 penalty). On the other hand, very robust fuel forms with multiple layers of containment and fission-product barriers (e.g. microencapsulated fuels) could require enrichment up to the low-enrichment limit of < 20%. In addition to the economic penalty, higher enrichments would result in lower uranium utilization and would have a major impact on the current enrichment and fuel fabrication plants. Higher enrichment requirements would be considered a vulnerability to the potential licensing and commercial adoption of the proposed fuel system design in the evaluation methodology presented in section 4.

A new fuel system could also have an impact on the back-end of the fuel cycle. The storage (wet and dry) and repository performance of the fuel (assuming a once-through fuel cycle) must not be degraded; otherwise, engineering solutions must be augmented during storage and disposal. Over the long term, U.S. policy changes to transition to a closed fuel cycle with reprocessing and recycling would require evaluation of the impact of the new fuel form on reprocessing, requiring licensing modifications in addition to changes in operational procedures and handling equipment. Assessment of the impact of a new fuel on potential future fuel cycles should be coordinated with the FCRD Separations and Waste Forms Campaign and the Fuel Cycle Options Campaign.

2.3 NRC Guidance on Definition of Reactor Operating Conditions

Figure 4 provides an overview of the NRC guidance for categorizing operating conditions in evaluating the performance of the reactor and fuel system to meet the requirements for performance and safety. The more recent NUREG-0800 Rev. 3 (US NRC 2007) event categorization approach defines conditions of normal operation as those that are expected to occur once or more during the lifetime of the plant, sometimes called Anticipated Operational Occurrences (AOOs). Postulated Accidents are unanticipated occurrences that can have consequences of significant radioactivity release that must be within off-site dose limits. In addition to categorizing the reactor conditions by event frequency, reactor conditions are also classified by type of event; that is, by the effect on the plant. Beyond normal operation, the types of reactor events are generally grouped into seven categories:

1) Increase in heat removal by the secondary system (e.g. increase in feedwater flow)
2) Decrease in heat removal by the secondary system (e.g. loss of external load)
3) Decrease in reactor coolant system flow rate (e.g. seized pump shaft accident)
4) Reactivity and power distribution anomalies (e.g. control rod ejection accident; RIAs)
5) Increase in reactor coolant inventory (e.g. inadvertent emergency core cooling system [ECCS] actuation)
6) Decrease in reactor coolant inventory (e.g. LOCA)
7) Radioactive release from a subsystem or component (e.g. steam generator tube rupture)

ANSI N18.2/ANS 51.1/52.1 Event Categorization	NUREG-0800 Rev. 3 Event Categorization	Example Events
Category I – Normal Operation and Operational Transients	Normal Operation and Anticipated Operational Occurrences (AOOs) (Events expected to occur once or more per reactor lifetime)	• Increase in feedwater flow • Loss of feedwater heater • Loss of external load • RCCA withdrawal at power • Complete loss of forced reactor flow • Reactor Coolant Pump Shaft Break/Seizure • Control Rod Ejection • Control Rod Drop • Small Break LOCA • Large Break LOCA
Category II – Moderate Frequency (once per calendar year)		
Category III – Infrequent Incidents (once per reactor lifetime)		
Category IV – Limiting Faults (not expected to occur – consequences of significant radioactive release)	Postulated Accidents (not expected to occur – consequences of significant radioactive release)	

Figure 4. Operating conditions categories for safety evaluation of nuclear power plants.

The operating conditions listed in Figure 4 are used in the design of the reactor systems, structures, and components (SSCs) critical to safety. The events in the figure are generally called the design basis conditions and the accidents are referred to as design basis accidents (DBAs). Acceptance criteria to ensure successful operation of the SSCs are defined and safety analyses are used to demonstrate compliance with these criteria. Situations in which key SSCs are inoperable are called severe accidents, falling outside the design basis, or sometimes referred to as beyond design basis accident (BDBA) conditions. Severe accidents generally lead to significant core damage, high temperature conditions leading to chemical reactions, generation of hydrogen and the release of radioactive fission products into the containment. While not normally included in the design of the reactor or fuel systems, severe accident conditions are an important element in the review of enhanced accident tolerant fuel concepts.

The potential impact of different material properties and fuel behaviors on the important characteristics or attributes of fuel performance are discussed below using several operational scenarios, each of which should be considered in evaluating the potential performance of a proposed fuel design. Specific scenarios include: normal operation; two AOOs (loss of coolant flow and reactor overpower); two postulated accidents (LOCA, RIA); and two severe accidents (with and without quench). These scenarios were selected to provide the spectrum of initial and boundary conditions defined by the reactor/fuel power level; coolant flow rate, temperature, and pressure; and duration of the event that fuel would be expected to endure. The thermal, mechanical, and chemical stability attributes of the fuel must be considered throughout the entire spectrum of conditions, not just at select state points (i.e. maximum temperature) because of the complex relationships between the properties and these attributes. A short description of each operating/accident environment is given below.

Normal Operation

Normal operation spans the conditions from beginning of life to fuel discharge. The power conditions during normal operation include reactor startup at the beginning of each cycle, mid-cycle power maneuvers, possible load-following, and restart from unplanned outages. Coolant chemistry conditions needed to maintain reactivity control, corrosion control of other components in the primary system, and impurities derived from leaching of piping materials can have an impact on the cladding surface, and erosion of novel cladding material can impact the coolant chemistry. Fuel burnup levels required to meet current operating strategies are in the 50 to 60 GWd/tU range. Resident times for the fuel range from 4 to 6 years.

Overpower Anticipated Operational Occurrence

The overpower AOO is a moderate frequency event that occurs as a result of a number of initiating conditions, such as loss of feedwater heater or uncontrolled control rod withdrawal at power. The power increase during this event leads to an increase in fuel temperatures, fuel thermal expansion strain, and cladding heat flux to the coolant. The event is terminated by a reactor scram on a high power signal. The fuel performance behaviors of interest during this event are the power to melt ratio for the fuel, the cladding fracture strain and the potential for stress corrosion cracking by fission products. Furthermore, the increase in cladding heat flux reduces the margin to departure from nucleate boiling (DNB).

Decrease in Coolant Flow Anticipated Operational Occurrence

The decrease in coolant flow AOO is a moderate frequency event when initiated from a reactor pump trip and as a postulated accident in the case of a seized pump rotor or pump shaft breakage. The decrease in coolant flow reduces the cladding-to-coolant heat transfer conditions, with a corresponding decline in the margin to DNB. The fuel performance behavior of interest during this event is the potential for post-DNB heat transfer conditions. Such conditions cause a significant increase in cladding temperature, increased reaction with the coolant water, and cladding collapse. Rewetting of the cladding by either reduction in power (scram) or increase in flow can result in thermal shock loads due to quench.

Reactivity Initiated Accident Design Basis Accident

A control rod ejection event in a PWR or a control rod drop event in a BWR are postulated reactivity initiated accidents (RIAs) that can lead to prompt reactivity insertion and rapid increase in reactor power to many times 100% rated power. This design basis accident is used to determine the impact of excess reactivity within the core on the reactivity worth of a single control rod. The rapid insertion of reactivity during this event causes power pulses that have widths ranging from 10 to 50 milliseconds. These conditions produce near-adiabatic heat up of the fuel material for most of the energy deposition phase. Heat conduction from the fuel to the coolant begins during later portions of the power pulse or after the power pulse is completed. As a result, the performance of the fuel during a RIA event can generally be viewed as having two phases, a near-adiabatic heat-up phase during the first 10-50 ms (Phase 1) and a heat transfer phase (Phase 2). The duration and significance of the heat transfer phase depends on the severity of the accident. A very sharp RIA pulse, such as a prompt critical pulse from a zero power condition, will initiate significant heat transfer to the coolant on a relatively short time scale (< 1 second). However, the heat transfer phase can also be slower (~ seconds), such as during a sub-prompt critical RIA from a full power condition.

Phase 1 is characterized by the density, heat capacity, conductivity, and thermal expansion of the fuel material and the cladding ductility at operating temperature. The stored energy capacity of the fuel influences the fuel enthalpy and temperature rise during the rapid energy deposition that occurs in Phase 1. During rapid energy deposition, the temperature profile in the pellet follows the radial power profile, which is flat for lower burnup fuel and peaked at the pellet surface for higher burnup fuel. At large energy depositions, the rapid heat-up of the fuel pellet can cause melting and fragmentation of the pellet, and, potentially, melting of the cladding. The increase in fuel enthalpy causes pellet thermal expansion and, depending on the pellet-cladding gap size, mechanical strain on the cladding. The ability of the cladding to accommodate the pellet thermal expansion strain is defined by the material ductility. The fuel thermal expansion stabilizes once heat conduction to the cladding and coolant begins.

Heat-up of the cladding is the start of Phase 2. This phase is controlled by the high temperature mechanical and chemical response of the cladding. For large energy depositions, DNB is likely with a significant drop in the cladding-to-coolant heat transfer and a corresponding increase in cladding surface temperature. After a short period of post-DNB heat transfer conditions, rewetting of the cladding will occur, producing thermal shock stresses from quench in the cladding. Depending on the cladding embrittlement level, fracture of the cladding is possible.

RIA response depends on the worth of the ejected rod, the neutron flux in the core, the reactor kinetics parameters, the fuel temperature feedback (Doppler) coefficient, and the heat capacity of the fuel, among other parameters. In contrast to some other accident conditions, which are primarily thermal hydraulic, RIA response is holistic and dependent on the synergistic feedback between the reactor neutronics and thermal hydraulics.

Loss of Coolant Accident Design Basis Accident

The LOCA event is a postulated accident that spans a range of primary system pipe breaks leading to a reduction in the coolant inventory. There are two main LOCA events: the large break LOCA (LBLOCA) and the small break LOCA (SBLOCA). The LBLOCA considers a rupture of a major coolant pipe that results in a rapid depressurization of the primary coolant system and uncovering of the reactor core. The LBLOCA causes an increase in the cladding temperature that is controlled by the stored energy of the reactor at the initial blowdown and by the decay heat during the reflood phase. Scram of the reactor during the depressurization terminates the energy generation from fission at the initiation of the event. At some point, the emergency core cooling system (ECCS) actuates to provide coolant make-up and reflood of the core.

The response of the fuel in the core during a LBLOCA is used to size and time the actuation of the ECCS to ensure compliance with the fuel damage limits specified in Title 10 of the U.S. Code of Federal Regulations Part 50.46b (U.S. 2009). The fuel and cladding experience several important thermal, mechanical, and chemical processes during a LBLOCA. These include rapid heating to temperatures in excess of 800°C during the blowdown phase. For zirconium alloy cladding, the fuel rod pressure exceeds the coolant pressure and rapid thermal creep/plasticity leads to ballooning and burst of the cladding between 700 and 800°C. Prior to core reflood by the ECCS, the decay heat combined with the steam heat transfer conditions causes the fuel rod to continue to heat up, with cladding temperatures reaching between 1000°C and 1200°C for worst-case assumptions. During this high temperature period the steam reacts with the zirconium-alloy cladding, oxidizing the outer surface and generating hydrogen. Rapid thermal quench occurs between 5 and 10 minutes after the oxidation process starts. The quench process produces thermal shock loads that can cause cladding fracture depending on the level of embrittlement from oxygen uptake and pre-transient hydrogen content.

In the case of a SBLOCA, the size of the pipe break limits the depressurization event. The loss of coolant inventory occurs at a slower rate, but the core can still become uncovered, leading to increased fuel rod temperatures. The anticipated temperature excursion in a SBLOCA generally remains below that of a LBLOCA, but the time at elevated temperature is considerably longer. A SBLOCA can last between 30 and 60 minutes before the ECCS system terminates the event. In optimally designing a new fuel system it may also be necessary to consider modification to the ECCS design and operation to take full advantage of the fuel system properties.

Severe Accidents (with and without quench)

Severe accident conditions include temperature, time, and coolant conditions well above those considered in design basis accidents. The maximum allowable cladding temperature during a LOCA event is 1200°C (based on limits for the current fuel system); the fuel material temperature remains well below melting due to appropriate actuation of the ECCS. Severe accidents, where heat removal capabilities are insufficient to remove the decay heat, can result in cladding temperatures that exceed 1200°C, the formation of eutectics between the fuel, cladding, control rod and structural material that can reduce melting temperatures, and extended duration of poor heat removal conditions for long periods of time (> 8 hrs). Severe accident response could include core reflood with coolant, leading to quench (thermal shock loads); alternately, failure to reflood could lead to long times at elevated temperature, fuel melting and relocation, and eventual failure of the reactor pressure vessel.

Station Blackout

Additional accident scenarios that will be discussed in the preliminary concept evaluations include a short-term station blackout (STSBO) and a long-term station blackout (LTSBO). In a STSBO, the reactor is shut down via scram, all power is lost and all coolant injection capability is lost at time zero. The LTSBO assumes the reactor is shut down via scram, AC power is lost at time zero and DC power (batteries) is lost some time into the event (e.g. at 8 hours). During the LTSBO event, the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) systems would be available for operation until the loss of battery power. For these scenarios, it can also be assumed that the ability to inject water via the control rod drive (CRD) pumps (for BWRs), low pressure ECCS pumps (no AC power) and diesel driven pumps is also assumed to be lost. A mitigated STSBO (MSTSBO) event can also be evaluated. In this case it is assumed that operators are able to inject water through the CRD pumps to provide a steady flow of high-pressure water injected into the bottom of the reactor pressure vessel (RPV).

2.4 Selected Baseline Accident Scenarios

It is clear from the discussion of operating scenarios in section 2.3, ranging from normal operation to design basis accident conditions, that a large number of fuel and cladding properties become important in each of these scenarios. While some scenarios are dominated by thermal hydraulic mechanisms, others are more dependent on the feedback between the reactor neutronics and thermal hydraulics. Hence, the complete spectrum of mechanical, thermal and neutronic properties of a proposed fuel and cladding system must be considered over a spectrum of potential operating conditions to assess the potential performance and accident tolerance of the proposed concept.

Reference transient and accident cases were discussed in detail at an International Workshop on Accident Tolerant Fuels for LWRs, as noted in the published workshop proceedings (OECD/NEA 2013). There was general consensus reached that station blackout (SBO) is a good reference transient to evaluate the potential benefits of new, more robust, fuel designs. Figures of merit in the evaluation of results would include “grace time” (the time into the event before the onset of core melt, during which additional recovery actions can be made to halt the accident progression), amount of combustible gases produced and the potential for release of radioactive material. Although this baseline accident case was identified for preliminary analysis, it should be noted that other accident scenarios might be relevant for a complete evaluation of the “accident tolerance” of a particular fuel design.

Two reference severe accident scenarios were selected for reactor performance simulations:

- Boiling water reactor (BWR) scenario: Fukushima 3 (3-F)
- Pressurized water reactor (PWR) scenario: Three Mile Island Unit 2 (TMI-2)

Analyzing performance of a proposed fuel design relative to a given accident scenario is challenging. Initial evaluation might consider performance of a selected concept (or the baseline UO_2 – zirconium alloy) using the known time-history of the actual event. This approach assumes that, for each fuel system, active components, such as pumps, are actuated at the exact same times; coolant levels and coolant chemistry are equivalent; etc. However, this approach fails to consider the likely differences in the chemistry response of each cladding type during the event. The response of an alternate fuel and cladding concept could actually change the environment itself, affecting the progression of the event. Production of hydrogen gas, for instance, could modify the natural circulation characteristics in the core, resulting in very different cooling conditions between the historical accident case and the considered new fuel case. A more complete assessment of accident performance should take these differences into consideration as development for a particular fuel and cladding design progresses.

Sensitivity analyses can provide insight into the relative (beneficial) effects of modifications to various fuel and cladding physics parameters, including thermal conductivity, oxidation kinetics, combustible gas production, exothermic reactions, fission product retention, mechanical behavior, etc. In evaluating fuel performance and failure probability during a postulated event, it should be recognized that the performance is not absolute for a given fuel design, but is instead a complex function of operating history (power history, fuel burnup, etc.) and evolution of a given transient (Youngblood and Smith 2013).

3. Key Material Properties and Behavior Characteristics

Candidate fuel systems must first hold to the principle of "do no harm," meaning that the fuels must, under all operating conditions, perform at least as well as or better than the current UO_2 – zirconium alloy fuel system. As discussed in section 2.2, a candidate fuel should preserve or improve upon:

- Burnup limits / cycle length (while maintaining criticality and fuel performance)
- Operational parameters (power distribution, peaking factors, safety margins, etc.)
- Reactivity coefficients and control parameters (shutdown margin, rod worths)
- Handling, transportation and storage (consideration of fuel isotopics, handling dose, mechanical integrity)
- Compatibility with existing infrastructure (e.g. fabrication facilities, loading, in-core operations, post-irradiation handling and storage, etc.; necessary to maintain acceptable economics)

To be considered for the "accident tolerant" moniker the fuel system must additionally provide improved response to AOOs, DBAs (RIA, LOCA, SBO) and BDBAs.

It is clear that a very large number of properties and performance parameters must be considered in fuel development. Fuel performance is the result of a complex system with interaction between the components of the fuel, cladding, reactor configuration and protection systems. Fuel system performance cannot be evaluated in isolation. Note that some "go / no-go" criteria must necessarily be established for any fuel system before defining more specific performance targets. Briefly stated, these include:

<u>Go / No-Go Criteria</u>

– The fuel design *must* meet current LWR constraints without changing assembly thermal-hydraulics.

– The fuel *must* have a quantifiable benefit under accident scenarios (e.g. longer time to onset of fuel melt) relative to the current fuel system to be deemed "accident tolerant".

– Reactivity feedback coefficients must be similar in magnitude and parametric behavior to the reference UO_2 – Zr alloy system to ensure backward compatibility in existing reactors. Reactivity coefficients for candidate fuels *must not reduce the safety of the reactor system or the safety margin* and should fall within the existing safety envelope for UO_2 – Zr fuels.

    - In some cases the calculated moderator temperature coefficients can be slightly positive (for current fuel systems) at the beginning of life with high soluble boron concentration. It is possible that the moderator temperature coefficient may also be slightly positive for some candidate ATFs under similar conditions.
    - Reactivity coefficients that are more negative than the reference UO_2 – Zr system may be problematic if they interfere with shutdown margin or system stability.

– The fuel *must* maintain current cycle length and power output at allowable enrichment levels (increased cycle length, number of fuel batches in the core management scheme, and power density may be desirable but not required).

While it may be desirable to improve performance in all areas identified in section 2.1, this likely is not an achievable goal. Hence, leading design objectives beyond the stated go/no-go criteria provide guidance to fuel design after first applying the practical constraints for use of the candidate design in a currently operating LWR. Stated more concisely than in section 2.1, design objectives include:

Leading Design Objectives

1) Maintain or improve upon the thermal, mechanical and chemical properties observed for the current state-of-the-art fuel systems.
2) Provide "accident tolerant" improvements that increase coping time ("grace period") under severe accident scenarios.
    a. Increase the time before the onset of core melt, during which additional recovery actions can be made to halt the accident progression.
    b. Reduce the impact of a severe accident by reducing core damage frequency (CDF); maintain coolable geometry; reduce combustible gas production as well as the amount of radioactive materials potentially released.
3) Offer the capability for power uprate and increased burnup to allow an economic case to be made for adoption of the new fuel system.

Key fuel system attributes that influence the thermal, mechanical, and chemical performance of the fuel system include chemical and dimensional stability, fission product retention, and impact on reactivity control systems during normal operation or accident conditions. The fundamental regulatory criteria for a potential accident tolerant fuel are the key reactor safety parameters versus the reference UO_2 – zirconium alloy fuel system. These criteria include comparison of lattice and full-core reactivity coefficients, study of thermal margin, full-core hot spot and hot channel factors during normal operation (e.g. F_q and $F_{\Delta h}$), and categorization of potential challenges in fuel management.

Applicability of existing regulatory limits to the alternate fuel system designs must also be considered. It is likely that, for some concepts, reasonable thermal limits will differ significantly from the current UO_2 – zirconium alloy system due to the temperatures at which eutectics might form between the fuel and cladding, fuel and cladding melt temperatures, etc. Additionally, recent NRC-sponsored work has suggested that the current regulatory acceptance criteria may not be conservative for higher burnup UO_2 – Zr alloy fuel under postulated LOCA conditions. The work found that embrittlement mechanisms may exist that were not considered when the original criteria were established. Hence, the current 17% limit on oxidation for zirconium-based cladding may not be adequate to preserve the level of ductility originally considered necessary for protection. As a result, a new NRC rule is current being considered. The proposed rule "would establish new requirements for zirconium-based alloys to prevent breakaway oxidation and account for oxygen diffusion from the oxide fuel pellet during the operating life of the fuel… necessary to prevent embrittlement of fuel cladding and to restore the rule to the level of reasonable assurance of adequate protection of public health and safety." [Excerpt from NRC proposed rulemaking (Borchardt 2012)]. In short, the expected performance of a particular fuel and cladding must be examined under all postulated normal operation and accident conditions throughout its intended lifetime (e.g. at high burnup) to ensure that the established limits meet the desired level of protection. Limits originally set for zirconium alloys (based on operation to lower burnup than is current industry practice) may not be applicable to new fuel designs or to the operation of the current fuel system to higher burnup. Effort should be expended to search the regulatory issue space for the limiting case for each fuel concept considered (Youngblood and Smith 2013).

3.1 Defining Key Fuel and Cladding Properties and Behavior

Key fuel system properties fall roughly into the categories of Thermal, Mechanical, Chemical and Neutronic. Some of the key properties in each of these categories, and a brief summary of their desired trends relative to the current system or their relative impact are listed below.

Fuel Thermal Properties:

1. *Melting Point,* T_{melt}
   Higher is preferred; important under RIA and LOCA conditions
2. *Thermal Conductivity,* k
   Higher is preferred; may impact DNB under loss of flow conditions; controls rate of stored energy release during a LOCA; as k increases fuel temperature profile flattens, providing further margin to melt under severe accidents and increasing the ability to retard Zr oxidation (acts as an effective heat sink)
3. *Density * Specific Heat,* ρC_p
   Controls initial stored energy at onset of LOCA; may impact DNB under loss of flow conditions (needs study)
4. *Diffusivity*
   Higher is preferred; controls rate of stored energy release during a LOCA; may impact DNB under loss of flow conditions (needs study)
5. *Coefficient of Thermal Expansion, CTE*
   Lower is preferred; important under overpower conditions

Fuel Mechanical Properties:

1. *Yield Strength*
   Lower is preferred; important under AOO conditions due to relationship to swelling
2. *Toughness*
   Higher is preferred; important under AOO conditions due to relationship to swelling; high toughness desirable during LOCA and severe accidents for mechanical stability and integrity to be preserved
3. *Creep Rate*
   Prefer rapid creep during normal operation; important under AOO conditions due to relationship to swelling; more ductile fuel is less likely to breach cladding during RIA
4. *Modulus of Elasticity*
   Structural rigidity required during normal operation; important under AOO conditions due to relationship to swelling

Fuel Chemical Properties:

1. *H_2O Reactivity*
   Lower is preferred; if cladding is breached due to LOCA or severe accident, fuel interaction with H_2O is very important
2. *Clad Compatibility*
   No adverse reactions between fuel and cladding under normal conditions; fuel and cladding in contact can react during AOOs, LOCA or severe accidents
3. *Phase Stability*
   Detrimental phase change and formation should be avoided under all conditions; phase stability during RIA only a concern at high burnup
4. *Fission Product Chemistry*
   New concepts should retain fission products as well as UO_2

Fuel Neutronic Properties:

1. *Fissile Density*
   Higher is preferred
2. *Cross Sections*
   Desire high fission cross section, low parasitic absorption
3. *Reactivity Feedback Coefficients*
   Similar in magnitude and parametric behavior to the reference UO_2 – Zr alloy system (see go / no-go criteria above for details)

"Other" Fuel Properties:

This general category allows one to capture aspects of fuel design and development that are not as easily classified or tied to a specific quantitative value but are equally important as those above.

1. Manufacturability
2. Transportability
3. Toxicity
4. Control Rod Compatibility
5. Reprocessing Potential
6. Proliferation Potential

Cladding Properties of Interest:

1. *Cladding thermal conductivity*
   Higher is preferred
2. *Cladding mechanical properties*
   a. Low pellet-clad mechanical interaction (PCMI)
   b. High strength
   c. High ductility
   d. No fission gas release under normal operation
   e. Minimal fretting wear
3. *Chemical compatibility*
   Cladding must be compatible with coolant and fuel (pellet-clad chemical interaction [PCCI])
4. *Cladding neutron cross-sections*
   Low parasitic absorption, activation

Each of these properties impacts the overall fuel performance across the various operating regimes that will be considered in evaluating a particular concept, as will be discussed in section 4.

All of these categories are important in understanding the anticipated performance under normal conditions and the potential enhanced accident tolerance of a specific design option. Quantification of target values for the various attributes/properties is extremely challenging due to the interrelationship between the properties. Rather, it is important to consider the integrated effects of the material properties of any particular advanced fuel design. These integrated effects become evident via analyses that consider the impact on the overall system; further discussion on a recommended analysis progression is provided in section 4.

In some cases standardized test procedures endorsed by the American Society for Testing and Materials (ASTM) can be used to characterize the materials of interest. However, for some materials, there are no existing standards to evaluate their applicability for nuclear service. It may be necessary to support the development of such standards in the course of ATF development and experimental evaluation or rely on existing regulatory limits. For instance, ASTM standards are not available for nuclear fuel. There are, however, standard characterization tests and NRC regulatory limits placed on hydrogen and moisture content in the fuel material. Many of the existing requirements for the UO_2 – zirconium alloy system are discussed in Appendix A. Note that any proposed rulemaking currently under review by the NRC should also be considered (Borchardt 2012).

The following sections provide discussion of an analysis framework (section 4), proposed characterization tests (section 5) and selected properties for which preliminary sensitivity analyses have been conducted (see section 6). This selection of initial sensitivity studies does not imply that other properties are insignificant in the evaluation of fuel and cladding concepts, but provides insight into the impact of some of the key properties of interest. Discussions on the impact of two key properties, fuel thermal conductivity and cladding oxidation behavior, are included below.

3.2 Fuel Thermal Conductivity

Under transient conditions, design criteria are put in place to limit core damage and maintain a controllable and coolable geometry in the core under DBA scenarios. As previously discussed for a loss-of-coolant accident, the existing design criteria limits the peak cladding temperature to below 1204°C as well as the maximum thickness of the cladding converted to oxide as a result of reaction with steam (currently to less than 17% of the cladding wall thickness for the zirconium alloy – UO_2 system; see U.S. NRC [2012] for proposed rulemaking that would change this criteria).

The thermal conductivity of UO_2 determines the temperature gradient and controls heat transfer across the UO_2 pellet in LWR fuel. Its magnitude is of interest since it impacts, while in turn being affected by, thermal-hydraulic and neutronic phenomena that govern reactor operation. By affecting the temperature profile, thermal conductivity influences the mechanical and thermodynamic state of the fuel pellet, in addition to affecting reactivity via Doppler broadening. Design requirements limit the maximum fuel centerline temperature to avoid excessive fission gas release under normal operating conditions and incipient melting under transient scenarios. An increase in the fuel thermal conductivity (and/or a reduction in the fuel pellet diameter) will reduce the peak fuel temperature at constant assembly power. The stored energy in the fuel is inversely proportional to the fuel thermal conductivity. Evaluation of performance sensitivity to the fuel thermal conductivity allows insight to the effect of stored energy in the fuel (viz. directly related to the temperature gradient across fuel pellets) on the thermal hydraulic response of the core. Note that while increased thermal conductivity is generally desirable, it will reduce the change in temperature that is observed in accident scenarios. This results in reduced negative Doppler feedback, which for some transients/accidents is key to limiting the power increase. This example illustrates the potential impact of considering fuel and cladding properties in isolation. The impact of improvements in one property must be weighed relative to the effect(s) on other properties.

Section 6 presents TRACE analysis results that illustrate the effect of enhanced fuel thermal conductivity on fuel temperature evolution in both a PWR and BWR during a LBLOCA. Full-core simulations were performed with fuel thermal conductivity at nominal and arbitrarily increased values. In this manner, the effect of thermal conductivity on the fuel temperature rise during an applied transient can be analyzed and compared across the two reactor platforms (see Terrani et al. [2013] for detailed analysis).

3.3 Cladding Oxidation Behavior

Candidate "accident tolerant" cladding materials should illustrate greater stability than zirconium alloy cladding (Zircaloy-2, Zircaloy-4, ZIRLO, M5, as applicable) in steam environments, and provide the potential for greater structural stability during severe accidents. Researchers have studied the oxidation behavior of various materials in steam and oxygen, and some materials have been found to have corrosion behavior that differs substantially. Key parameters of importance in the oxidation behavior of a material include the exothermic (or endothermic) nature of the reaction, the oxidation rate, and the products of oxidation, including both gaseous products (potentially combustible) and oxide products that often form a scale on the surface of the material. The diffusion rate of oxygen or water vapor through this oxide layer is also important, as it may temporarily protect the underlying material.

Section 6 presents MELCOR analysis results for candidate accident tolerant cladding materials applied in both PWR and BWR accident scenarios. Complete details on the oxidation behavior of some of the candidate cladding materials are still being defined experimentally. Hence, an additional parametric study was performed to predict the effect of changing the cladding oxidation rate by a factor of 10. Such parametric studies should also be completed for variation in the heat of oxidation to better understand the properties that drive the cladding behavior under severe accident conditions.

3.4 Interaction of Key Properties

It should be noted that while sample parametric studies for fuel thermal conductivity and for cladding oxidation rate have been performed to date (example results presented in section 6), these are not the only properties that should be considered in designing and selecting accident tolerant fuel and cladding materials. Additional fuel and cladding properties are likely to play key roles depending on the specific materials being assessed. The complex nature of the interaction between the various fuel and cladding properties and behaviors must be characterized via systems analysis under normal and off-normal conditions. Adopting a fuel technology with completely different vulnerabilities than the current fuel technology (i.e. ceramic vs. metallic cladding) would presumably be predicated, in part, on the adoption of different limiting conditions of operation. These differences would be expected to impact the sensitivity of the system to changing a selected property or behavior. Ideally, a new fuel system would allow for optimization of all plant systems (e.g. emergency core cooling system, coolant chemistry control, etc.), and a fair comparison of technologies would require such optimization of ancillary plant systems in addition to changing the fuel. Optimization of the plant adopting the new fuel system would be expected to some extent (at least with regard to operations), but is considered to be beyond the scope of these initial screening analyses intended for down-selection of ATF concepts.

A proposed analysis framework for ATF is presented in section 4 and sample calculations are included in section 6.

4. Evaluation of Accident Tolerant Fuel Designs

Design of an advanced fuel system demonstrating enhanced performance and safety relative to the current fuel system first requires understanding of the current state-of-the-art fuel system performance under the various system operating regimes. The performance of the current UO_2 – zirconium alloy fuel system and the current life limiting challenges and abilities are summarized in Appendix A. The current section presents a proposed evaluation and screening methodology to allow ranking and down selection of concepts. Scoping analyses are followed by more detailed analyses for those concepts that pass through the initial down selection. Results at each stage of analysis will have decreasing levels of uncertainty (increasing accuracy) as additional data are gathered to support the development of behavioral models for a particular concept. Results will allow for comparative ranking of candidate fuel system designs, relative to the existing fuel system and to one another, at differing levels of detail, enabling down-selection for further development of one or more concepts toward qualification.

Beginning with the qualitative guidance on ATF discussed in section 2, a small team from across the DOE laboratories has defined a technical evaluation approach for accident tolerant fuels. An assessment of the potential beneficial impact or unintended negative consequences of candidate ATF concepts must address the obvious "fuel-specific" characteristics of the concept. However, the assessment must also address how implementation of the concept will affect reactor performance and safety characteristics. This includes coupled neutronics and thermal-hydraulics analyses to ensure that the reactor would operate within the established performance and safety envelope.

4.1 ATF Evaluation and Screening

A preliminary approach to advanced fuel evaluation for enhanced accident tolerance is presented in Figure 5. The steps described in Figure 5 address the full scope of activities that must be considered in evaluating the feasibility of candidate ATFs. Carrying out all the indicated steps can require significant investment of time and resources, but each of these steps needs to be executed at the appropriate point in concept development. The fidelity or level of detail involved depends on the maturity of a concept; down selection is necessary to focus limited resources on concepts that have a greater chance of success. During the "screening" stage that is performed for all proposed concepts (*Step 1*, as numbered in Figure 5), the level of detail associated with analyses will be limited, based on the current state of knowledge for the selected concept. The level of detail may range from literature reviews and expert judgment through limited experiments and computational analyses. The goal is to have sufficient confidence in the results of the assessment (with a reasonable investment of time and resources) that identified changes relative to the reference UO_2 – zirconium alloy fuel system are known well enough to proceed with continued development of the concept, or conclude that the concept should be modified or abandoned. Note that concepts may be proposed and developed by national laboratories, industry, or academic institutions. Regardless of the concept origin, the leading ATF design would be transferred to industry for ultimate commercialization in Phase 3. Regulators should be engaged early in the development process to ensure that the characterization and testing data developed through Phase 2 will be appropriate for future licensing.

For further description of the corresponding fuel technology readiness levels (TRL) related to Figure 5, see *Technology Readiness Levels for Advanced Nuclear Fuels and Materials Development* (Carmack 2014). Roughly speaking, the "Feasibility" phase, noted as Phase 1 in the figure, corresponds to TRL 1-3; the "Development and Qualification" phase, Phase 2, corresponds to TRL 4-6; and the "Commercialization" phase, Phase 3, corresponds to TRL 7-9.

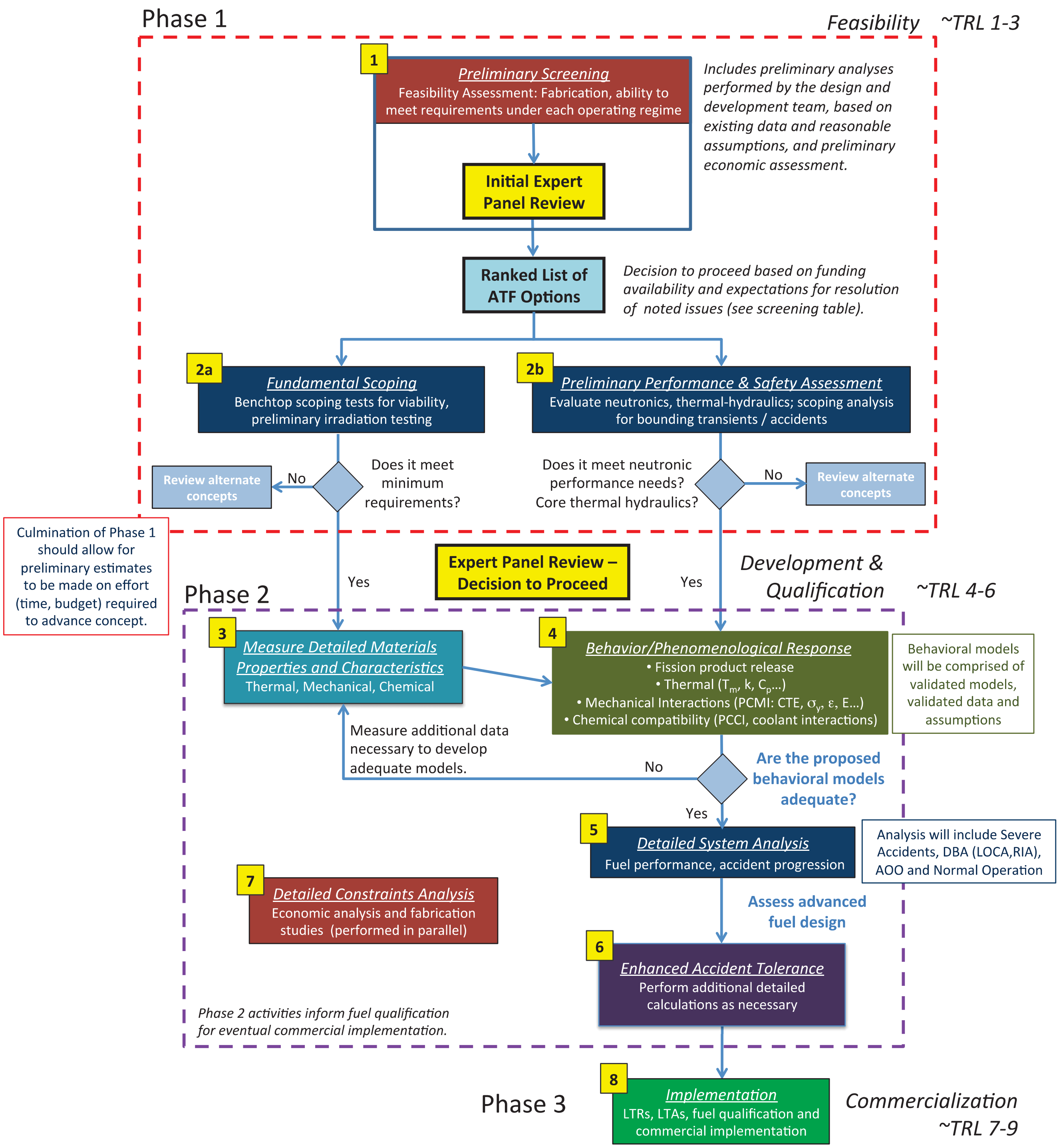

Figure 5. Proposed Accident Tolerant Fuel evaluation methodology. Preliminary concept down-selection would occur within Step 1 (see step numbers 1-8 noted), with secondary down-selection occurring at the end of Phase 1 prior to detailed tests and behavior model development.

4.1.1 Phase 1: Feasibility Assessment

Step 1, Preliminary Screening, describes the preliminary screening analyses that candidate designs will be subjected to in order to assess their performance under normal conditions and potential enhancements to safety under DBA and BDBA conditions in a generic PWR or BWR application. These preliminary analyses would apply existing materials data, fabrication experience, and cost data. Gaps in this data are likely, requiring assumptions to be made for some of the pertinent details. Scoping analyses are expected to have relatively large uncertainty bands and should not be applied to specific design activities. The purpose in these analyses is to eliminate proposed designs that are not likely to improve upon the current UO_2 – zirconium alloy fuel system and, hence, do not warrant additional investment. Proposed designs demonstrating promise as accident tolerant fuel candidates via an initial Expert Panel Review would then be recommended for additional scoping tests and property measurement (*Step 2a*) to assess the concept viability, followed by more detailed characterization and testing to fill the data gaps as necessary to develop applicable behavioral models. Considerations for the Expert Panel are summarized in Table 1, which is discussed further in section 4.1.3.

Following the initial Expert Panel Review, *Step 2, (a) Fundamental Scoping and (b) Preliminary Performance & Safety Assessment* includes both preliminary experimental characterization and evaluation (*Step 2a*) and preliminary core level analyses to determine the potential impact of a proposed design on core neutronics and thermal hydraulics under normal operating conditions (*Step 2b*). *Step 2b* additionally includes scoping analyses for a limited set of bounding transients and accident scenarios to confirm that a proposed concept will provide benefits in coping time, accident consequences, etc. relative to the current fuel system. At the culmination of Phase 1, concepts will be elevated to TRL 3 or 4 and sufficient data and experience should be available to inform a preliminary estimate of the effort required in terms of time and budget to advance a concept through Phase 2 development to TRL 6. Assuming that a selected concept makes it through the second Expert Panel Review ("Decision to Proceed") following *Step 2*, focused development and qualification-related activities would commence.

4.1.2 Phase 2: Development and Qualification Activities

For those concepts advancing past the second Expert Panel Review, additional data would be collected via both out-of-pile measurements and irradiation testing (*Step 3*). This data would be used in advanced behavioral models (*Step 4*) necessary to perform more detailed fuel performance and core-level analyses for normal and off-normal conditions at higher fidelity (reduced uncertainty) than the scoping analyses in *Step 2b* allowed. The enhanced behavioral models would then be applied in a more detailed system analysis (*Step 5*) intended to evaluate the specific concept under normal and accident conditions with a reduced uncertainty relative to what was possible earlier in the development (*Steps 2a/b*); results of this analysis would feed into the *Step 6* decision point regarding the level of accident tolerance of a particular concept, and performance of any remaining calculations to establish confidence in the design prior to insertion as a lead test rod in *Step 8* and Phase 3 Commercialization activities.

Step 7, Constraints Analysis, would be performed during Phase 2 in parallel with characterization, testing and model development. This step encompasses evaluation of the economics associated with the proposed fuel system (covering the complete fuel life cycle); challenges to fuel fabrication at the enrichment level determined by neutronics analysis; and potential fabrication challenges that could arise when ramping up production to quantities necessary for full-scale deployment in the operating LWR fleet. Preliminary engagement of regulators should also occur in parallel with Phase 2 to aid future licensing activities.

4.1.3 ATF Candidate Screening

Table 1 is a proposed screening table for the Expert Panel Review of candidate fuel systems. The attributes defined in the table correspond to a complete fuel system (fuel plus cladding) over the entire fuel life cycle (fabrication – operation – used fuel management). The listed attributes are intended to apply to both PWR and BWR plants, although some of the noted "*Considerations*" include constraints that are specific to one system or the other. This table is designed to be applicable to a candidate fuel system at all stages of development. The design and development team for a candidate fuel system would be expected to rely on preliminary analyses and scoping studies performed as a part of the initial technology selection to inform the initial Expert Panel Review, which may include significant uncertainty in some areas. As more data becomes available, the detailed performance attributes included in the table will be refined, reducing the uncertainty in the estimated performance assessment.

The Expert Panel Review Committee will provide independent assessment of the technology feasibility for near term research and development of candidate ATF design concepts and prioritization of those concepts. Table 1 will aid the review committee in ranking the potential performance of multiple concepts across the fuel life cycle and range of potential operating conditions. The reviewed concepts may be at varying stages of development; it is unlikely that any of the concepts will have been fully characterized upon commencement of the review. The goal for the initial committee review is to have sufficient confidence in the assessment results to either proceed with continued development of the concept, or to conclude that the concept should be modified or abandoned.

Table 1 is designed to identify "Benefits" and "Vulnerabilities" for each concept. The candidate fuel system should be ranked within each category on a scale of 0 to ±5. A score of "0" would indicate no notable change from the current UO_2 – zirconium alloy system; i.e. evaluation of UO_2 – Zr would result in an overall score of 0 for both benefits and vulnerabilities. A score of "+5" in the benefits column would indicate a significant benefit/improvement relative to the current system; a "-5" in the vulnerabilities column would indicate a significant data gap or potential issue for the indicated performance attribute. A "vulnerability" could be a known operational vulnerability that results from a particular fuel system behavior or requirement, such as low melting temperature, or an attribute could be scored with high vulnerability based on unknown performance for a selected parameter. If a specific property or behavior has not yet been measured or tested (or if it is not known conclusively), this would correspond to higher vulnerability and should be scored as such. For attributes that are currently unknown or assumed, recommended actions should then be noted by the Review Panel.

Scoring of benefits and vulnerabilities is inherently subjective in nature. The purpose in establishing a range of possible scores from 0 to ±5 is to allow for more discretization by the review committee that will be necessary in ranking the various concepts. Scoring of benefits and vulnerabilities *separately* ensures that technologies can easily be parsed into categories of moderate benefit – low risk; moderate benefit – high risk; significant benefit – low risk; and significant benefit – high risk. Selection of the panel members based on significant expertise in one or more areas relevant to fuel and reactor performance will be critical to development of an accurate review of all proposed concepts. It is not expected that any one reviewer would have expertise in all necessary areas.

If a decision is made to continue development to *Steps 2a/b* despite noted vulnerabilities, any recommended actions should be resolved before convening the second Expert Review Panel that would be tasked with ranking the remaining candidate technologies (shown between Phase 1 [*Steps, 1, 2a/b*] and Phase 2 [*Steps 3-7*]) based on both perceived benefits and remaining vulnerabilities. Concepts that are further in their development may have already completed elements listed in *Steps 2a/b* prior to the initial Expert Panel Review, while others may have larger uncertainties associated with the expected fuel system behavior. The former case would have fewer recommended actions, while the latter may have more recommended actions necessary to reduce uncertainty in the performance estimates.

Table 1. Candidate Fuel System Attributes Assessment Table

Performance Regime	Performance Attributes (For large-scale deployment)	Expert Opinion Assessment		Recommended Actions
		Benefit	Vulnerability	
Fabrication/Manufacturability *Considerations:* Millions ft of clad/year ~300 million pellets/year Economics - cost of raw materials and fabrication process Current fabrication plant enrichment limits	Manageable fissile material content			
	Compatible with large-scale production needs (material availability, fabrication techniques, waste, etc.)			
	Compatible with quality and uniformity standards			
	Licensibility			
Normal Operation and AOOs *Considerations:* Overall neutronics Linear Heat Generation Rate (LHGR) to centerline melt Power ramp, ~100 W/m/min Reduced flow (departure from nucleate boiling, DNB) Flow-induced vibrations Surface roughness effects Safe shutdown - earthquake External pressure (~2750 psi, 10% above PWR design pressure) Axial growth (less than upper nozzle gap)	Utilization or Burnup (12, 18, or 24 month/cycle)			
	Thermal hydraulic interaction			
	Reactivity control systems interaction			
	Mechanical strength, ductility (beginning of life and after irradiation)			
	Thermal behavior (conductivity, specific heat, melting)			
	Chemical compatibility (fuel-cladding) / stability			
	Chemical compatibility with and impact on coolant chemistry			
	Fission product behavior			
Postulated Accidents (Design Basis) *Considerations:* Prompt reactivity insertion Post-DNB behavior (T > 800°C for Zr-UO_2 system) Loss of coolant conditions Thermal shock Steam reactions (~1000°C +)	Thermal hydraulic interaction			
	Mechanical strength and ductility			
	Thermal behavior (conductivity, specific heat, melting)			
	Chemical compatibility/ stability (e.g. oxidation behavior)			
	Fission product behavior			
	Combustible gas production			
Severe Accidents (Beyond Design Basis) *Considerations:* Thermal shock Chemical reactions Combustible gas release Long-term stability in degraded state	Mechanical strength, ductility			
	Thermal behavior (conductivity, specific heat, melting)			
	Chemical compatibility/ stability (including high temperature steam interaction)			
	Fission product behavior			
	Combustible gas production			
Used Fuel Storage/ Transport/ Disposition *Considerations:* Handling, placement, and drying loads; future reprocessing potential	Mechanical strength, ductility			
	Thermal behavior			
	Chemical stability			
	Fission product behavior			

The assessment table attempts to list key considerations under each performance regime, and the associated performance attributes for those regimes. *This list is not exhaustive, but is intended to identify the major contributors/considerations to the identified regime.* Some of these items may need to be modified for particular plant designs. In the absence of a specific plant design, which would include details of emergency response systems applicable to accident performance, these assessments must be made on a general basis. As discussed in section 3, some “go / no-go” criteria must be met prior to completing the more detailed assessment. A concept that fails to meet one of these criteria would be omitted from further consideration. For example, a fuel enrichment requirement of 20% or higher (above the low enriched uranium [LEU] limit), as determined by neutronic calculations, would be “no-go.” This equates to a vulnerability that cannot be overcome, hence removing the concept from consideration. A required enrichment of 5-20% would be assigned an increasing vulnerability as the enrichment requirement increases (i.e., 19.9% → vulnerability of -5).

4.2 Phase 1 Basic Core Level Analysis

An assessment of the potential impact(s) of candidate ATF concepts must address how implementation of the concept will affect reactor performance and safety characteristics during each operational regime defined in Table 1. Assessments performed in *Step 2* would include the following elements:

- **Neutronics Analyses:** These analyses can often be performed initially at the fuel-assembly level (especially for PWRs) using the linear reactivity model (or an appropriate enhancement) to estimate resulting cycle length and discharge burnup as a function of the number of batches in the fuel management scheme, power peaking, etc., and to provide estimates of reactivity and control coefficients relative to the reference UO_2 configuration. Monte Carlo codes such as MCNP and Serpent can also be applied at both the assembly and core levels; TRITON/CASMO/HELIOS could be used for lattice physics calculations. The Monte Carlo codes provide results that are essentially benchmark quality and are constrained only by the available nuclear data and the geometric detail and statistics employed in the modeling. In principle they can be coupled to thermal-hydraulic analyses to include feedback effects; however, a significant penalty in computational efficiency must generally be addressed in these cases.

  Ultimately, core-level analyses are required to assess the potential benefits, as well as any negative aspects, associated with the implementation of a specific concept. In addition to serving the initial screening function described above, the assembly lattice level analyses provide the nuclear data (e.g., neutron cross sections, etc.) for subsequent full-core analyses with deterministic (typically nodal diffusion) three-dimensional core simulator codes. These full-core analyses, performed using tools such as PARCS (Downar et al. 2012, 2013), include thermal-hydraulic feedback and can be used to evaluate the time-dependent response of the reactor under transient/accident conditions. Both the thermal properties and the reactivity coefficients will change relative to the reference UO_2-Zr alloy system.

  Coupled thermal hydraulic-neutronic analysis of candidate ATFs is essential for understanding the synergistic impact of the thermal properties and reactivity feedback.

- **Thermal-Hydraulic Analyses:** These analyses can also be performed at the assembly or core levels for steady-state estimates of the Departure from Nucleate Boiling Ratio (DNBR) or Minimum Critical Power Ratio (MCPR). Codes that can be used for these analyses include VIPRE and COBRA.

- **Fuel Performance:** See section 4.3 for detailed description.

Successful completion of initial candidate ATF concept screening will optimally address the desired objective of evaluating the potential that a proposed concept can at least have equivalent performance to the current UO_2 – Zr alloy fuel. The decision point following the core-level screening analysis (*Step 2b*) will provide a preliminary assessment of the potential performance and safety benefits of the proposed ATF to address the over-arching question: *What do I gain if I implement a particular concept?*

Assessment of the potential safety benefits should address the full spectrum of reactor accidents/transients up to and including Beyond Design Basis Accidents, as summarized in section 2.3. This phase of the feasibility assessment would include the full spectrum (or an appropriate subset) of the accidents in Chapter 15 of a standard Safety Analysis Report. Selected accidents were identified in section 2.4 only for preliminary assessment and down selection. Assessment could be performed by a systems analysis code such as TRACE or RELAP, where the detailed behavior of the reactor core does not need to be explicitly modeled, or using a coupled PARCS/TRACE simulation where the three-dimensional core response is evaluated with PARCS, and TRACE provides the balance-of-plant boundary conditions for the detailed simulation of transients/accidents where a detailed model of the core behavior is necessary.

A key aspect/objective/constraint of the feasibility assessment (*Step 1*) with respect to these reactor performance and safety characteristics is the necessary level of sophistication for these analyses (recognizing that development of an actual core design for a particular ATF concept requires a significant investment of resources). The results of the screening analysis should be of sufficient reliability to provide confidence that the conclusions are "actionable." This requirement essentially establishes a minimum uncertainty band on the scoping analyses to ensure that further development efforts for a selected design are focused, justified and would be expected to yield benefits with respect to the established criteria.

Key core-level operational considerations for normal operation include cycle length, reactivity coefficients, core power distribution, and thermal margins. Performance must be maintained relative to the current fuel system with regard to the following factors:

- Burnup and cycle length (ensure criticality, maintain or improve fuel performance)
- Operations (power distribution, peaking factors, margins, thermal hydraulics, etc.)
- Reactivity coefficients and control (shutdown margin, rod worth)
- Handling, transportation, storage (fuel must be able to endure these activities)

Each of these factors is integral to the pragmatic and economic application of a new accident tolerant fuel. The key question about any potentially accident tolerant fuel is: *Does the fuel enhance accident performance while satisfying operational constraints, and while maintaining compatibility with existing infrastructure?*

Thus, a key component in the core-level analysis of an ATF concept is the corresponding analysis of the resultant reactor system. For each accident tolerant fuel, the impact on the following performance characteristics must be understood at a given state-point and as a function of burnup:

- Steady-state core power, neutron flux distribution, fuel management challenges
- Reactivity coefficients, control parameters
- Delayed neutron parameters (β_{eff}, λ_i), neutron lifetime
- Hot channel factors, hot spot factors, thermal margins
- DNBR and MCPR
- Criticality safety

These assessments are covered by the "considerations" and "performance attributes" summarized in Table 1. Several available tools may be relevant to these analyses. Lattice-level tools include components of the SCALE package, such as TRITON/NEWT (ORNL 2011). Reactor core analysis can be performed using well-established regulatory-grade tools, such as the state-of-the-art PARCS core simulator (Downar et al. 2012, 2013).

The fundamental regulatory criteria for a potential accident tolerant fuel are the key reactor safety parameters versus the reference UO_2 – Zr alloy fuel system. These parameters include comparison of lattice and full-core reactivity coefficients, study of thermal margin, full-core hot spot and hot channel factors during normal operation (e.g. F_q and $F_{\Delta h}$), and categorization of potential challenges in fuel management. Applicability of the existing thermal limits must also be considered. An example core-level analysis using this approach for several arbitrary ATF concepts is provided in section 6.1. Section 6.2 presents results of a sensitivity study performed to assess the impact of modifying one of the identified key fuel parameters (fuel thermal conductivity) as an example of a scoping study that might be performed within *Step 2a* in Figure 5.

4.3 Phase 2 Detailed Analysis

Following the second Expert Panel Review, more detailed testing and characterization is proposed to inform development of behavioral models necessary to conduct detailed systems analysis (*Step 5*) to better assess the potential performance of enhanced accident tolerant fuel concepts. Detailed property measurements and characterization data will reduce the uncertainty that would be present in the preliminary scoping analysis. The detailed concept review attempts to eliminate bias to UO_2 – Zr alloy designs.

- *Step 3: Materials Properties and Characteristics*
  Following the screening analysis for basic material viability (*Steps 1-2*), more detailed measurement of materials properties and characteristics is necessary to perform a fuel performance calculation for those concepts that will be investigated beyond the initial down selection. Most of the necessary properties were described in section 3.

- *Step 4: Behavior/Phenomenological Response*
  Several behavioral models that are required to perform a fuel performance simulation are developed. These models will include the material properties previously described, which will come from validated models, experimental data or assumptions.

  An assessment of the behavioral models that exist for the proposed advanced fuel design must be performed (shown as a decision point below *Step 4* in Figure 5). The team proposing any particular design should provide a collection of behavioral models with a full description of the source of the data and/or assumption relied upon to generate the models. If the behavioral models are deemed adequate to perform a fuel performance calculation, then the evaluation should proceed with a detailed system and fuel performance analysis (*Step 5*). However, if the models or the data used to generate the behavioral models are deemed inadequate, then effort should be redirected to generate the necessary data to develop adequate models rather than continue an attempt to assess the performance and enhanced safety aspects of a design that fails this assessment. Some designs that are significantly different than the current UO_2-Zr alloy system may require development of significant modifications to the existing analysis tools.

- *Step 5: Detailed System Analysis*
  Detailed system analysis must be performed using the established behavioral models and associated data for both PWR and BWR cases (if the fuel system is to be considered for both reactor types). For an integrated analysis, relevant codes include: severe accident behavior (e.g. MELCOR, MAAP), design basis accidents (e.g. RELAP/RETRAN), and core depletion (e.g. TRITON/PARCS). Accidents that require 3-D neutronic feedback would be studied using

appropriate tools such as TRACE/PARCS or RELAP/PARCS. For single effects analysis, such codes are: for fuel performance analysis (e.g. FRAPCON/FRAPTRAN, FALCON) and focused codes on specialized/unique properties. See section 4.4 for details on severe accident analysis.

- *Step 6: Enhanced Accident Tolerance*
  Upon completion of the detailed analyses, sufficient information will be available to perform an assessment of a particular design's enhanced accident attributes with a reasonable level of confidence (acceptable uncertainty). In performing this assessment, it is important to identify the objectives, e.g. as good as or better than UO_2 – Zr alloy in safety and performance, and identify requirements to meet performance/safety objectives, e.g. fission product retention, cladding stress, fuel-cladding reaction layer, fraction of melt, etc. That is, calculations are performed using applicable codes/tools with representative models/data to compare baseline (UO_2 – Zr alloy) and new designs under different accident scenarios.

4.4 Severe Accident Analysis

Scoping simulations of severe accident conditions can be performed using a severe accident code, such as MELCOR, to investigate the influence of advanced materials on the BDB accident progression and to identify any existing code limitations. The MELCOR code is the primary code used by the NRC to model and analyze the progression of severe accidents (Gauntt et al. 2005; SNL 2000). It has been developed and maintained for the NRC by the Sandia National Laboratories (SNL). MELCOR is a systems level severe accident code which includes the major phenomena of the system thermal hydraulics, fuel heat up, cladding oxidation, radionuclide release and transport, fuel melting and relocation, etc.

MELCOR is designed for current LWR core material configurations. As such, the code contains material property definitions for "UO_2", "Zircaloy", "ZrO_2", "steel", "steel oxide" and "Inconel" for the fuel, cladding, spacer grids, support plates and channel boxes. The cladding and structural materials in the core are coupled in the code such that a change to the cladding material via user input also changes the composition and properties of structural materials assigned to the same material properties (e.g. "Zircaloy"). The user can change the defined properties of one of these materials, such as Zircaloy, in a relatively straightforward manner through user input. Internally, however, the code assigns material composition according to core component. As a result, a material property change to one of the core components, e.g. the cladding, will also change the material composition and properties of other core structures, e.g. channel boxes. This assumption can only be changed by modifying and recompiling the code to include additional materials for any given core component. Modifications that have been made to the MELCOR code to evaluate the performance of candidate accident tolerant materials are presented in further detail in section 6.3 and are summarized in Merrill and Bragg-Sitton (2013). Additional modifications to the MELCOR code structure are currently under way to allow more detailed analysis of candidate ATF designs under severe accident conditions.

The Three Mile Island Unit 2 (TMI-2) loss-of-coolant accident was selected as the baseline accident for examining the potential safety merits of PWRs constructed using the proposed accident tolerant fuel and cladding options (OECD/NEA 2013). For BWRs, short term and long term station blackout (STSBO, LTSBO) accidents have been simulated for Peach Bottom, a GE BWR/4 with a Mark-I containment. Fukushima Unit 3 (3-F) has been identified as the baseline BWR severe accident case (OECD/NEA 2013). Preliminary accident analysis results for selected zirconium alloy replacement materials are provided in section 6.3 and 6.4. As discussed previously, applying the same timeline for the accident sequence in the ATF analysis as was observed in the actual event may be somewhat misleading, as it does not take into account the affect that the ATF design may have on the coolant environment itself that could alter the overall system response.

5. Characterization and Scoping Tests

Prioritized characterization and scoping tests are proposed to provide early indication of the "accident tolerance" of a specific concept and to begin definition of some of the material properties data that will be needed in the details behavioral models. Many of the scoping tests focus on either the fuel or the cladding, while some require testing of a complete fueled rodlet.

5.1 Cladding Evaluation

Three primary "classes" of cladding materials are considered: fully metallic, fully ceramic, and hybrid cladding. The latter "hybrid" classification could include coatings or wraps on an inner cladding tube (likely metallic). The proposed test prioritization assumes that, based on known properties, the candidate material meets basic requirements for use in a fuel rod (e.g. it is not known to be a neutron poison material that would significantly impact core neutronics unless its negative impact can be reduced by having thinner walls). As additional data are generated, researchers must reassess the impact of the test and measurement results on the ability to design a functional fuel pin.

An initial set of prioritized cladding screening tests was proposed at the *Enhanced Accident Tolerant LWR Fuels National Metrics Workshop* in October 2012 (Braase 2013). This set of proposed tests was later refined at a meeting specifically focused on coated cladding options, which would be categorized as a "hybrid" cladding design (Braase and Bragg-Sitton 2013). The purpose of the proposed set of preliminary tests is to demonstrate basic concept viability, to provide justification that the concept could meet the definition of "accident tolerant," and to provide additional material property data to inform the associated fuel performance and core-level analyses. The series of tests proposed in the following sections assume that basic fabrication constraints have been met. The required fabrication processes (e.g., fabrication temperatures, annealing steps, etc.) must be designed such that the substrate cladding material will retain sufficient corrosion resistance and mechanical properties.

A test matrix for candidate fuel and cladding shall include all licensing criteria and shall identify existing test standards or characterization tests for which standards must be developed for future qualification. Categories for cladding tests and proposed test objectives, constraints, or associated details are provided in Appendix B. A preliminary list of proposed screening tests for each class of cladding design options (metallic, ceramic and coated or hybrid cladding) is provided below. These tests are not intended to fully characterize the material, only to demonstrate viability. Cladding candidates that pass the initial screening tests and screening analyses would then be considered for more detailed testing in the correct geometry and environments (e.g. fueled rodlets). Data from these later tests would then be applied in behavioral models and detailed fuel and core analyses. The proposed prioritization of screening tests is as follows:

Metallic Cladding

1. Steam oxidation, post-steam ductility / strength tests
2. Environmental testing (corrosion, irradiation assisted SCC, SCC, erosion)
3. Basic chemical compatibility at normal operating conditions
4. Irradiation environmental testing (determine effects on mechanical properties)
5. Mechanical testing (assuming baseline unirradiated data are already available, additional mechanical tests would not be required until after a material has passed the other screening tests identified)

Ceramic Cladding

1. Mechanical integrity tests (simple tests for hermeticity, break)
2. Steam oxidation, post-steam ductility / strength tests
3. Basic chemical compatibility at normal operating conditions
4. Environmental testing (LWR water exposure over a range of chemistry conditions)
5. Mechanical properties measurements
6. Irradiation environmental testing (determine effects on mechanical properties)

Hybrid Cladding / Coatings on Cladding Tube

1. Assess coating adhesion – initial, thermal cycling, with flow (spall), wear (fretting)
2. Based on substrate and "coating" materials and configuration, follow metallic or ceramic cladding priorities as defined above; additional details for coated cladding options include:
    a. Environmental testing with a small exposed region (autoclave)
    b. Chemical compatibility to include coating – substrate compatibility (establish temperature limit)

    Note for hybrid / coated cladding options: During cladding fabrication, the substrate must retain sufficient corrosion resistance and mechanical properties (e.g. must be able to withstand annealing temperatures).

Coated cladding options are expected to present unique questions in their preliminary evaluation for accident tolerant application. Although coated cladding designs represent a smaller departure from the current Zr-alloy cladding (assuming Zr-alloy substrate), there are concerns about coating adhesion to the substrate surface during operating and accident conditions. To address these concerns, specific details on the proposed scoping tests for coated cladding are provided below.

1. *Coating Adhesion* – Address the effects of thermal cycling and exposure to coolant flow (spall) and wear (fretting) and cladding creep-down on coating adhesion. These tests would likely include some mechanical testing to demonstrate adhesion under handling scenarios (e.g., bend tests).
2. *Environmental Testing* – Address coating performance at temperature before and after a standardized scratch test (application of a set force, which may or may not expose the underlying substrate) and with a small exposed region (e.g. full depth scratch down to the substrate). Evaluations should include corrosion, erosion, stress corrosion cracking, etc., when exposed to LWR water over a range of chemistry conditions for both PWR and BWR designs.
3. *Chemical Compatibility* – Address coating – substrate compatibility (establish temperature limit). For designs that deviate from a standard zirconium alloy substrate and UO_2 fuel, compatibility between the fuel and substrate cladding material must also be demonstrated.
4. *Severe Accident Conditions* – Steam oxidation testing and measurement of post-steam ductility and strength. The effect of pressure, flow rate and temperature should be evaluated.
5. *Mechanical Tests* – Assuming baseline unirradiated data are already available, additional mechanical tests would not be required until after a cladding design has passed the other screening tests identified; for non-metallic substrate materials (e.g. ceramic substrate) mechanical tests may be performed earlier in the test series to determine ductility, fracture behavior, and hermeticity.

6. *Irradiation Environmental Testing* – Irradiation of small sample coupons may precede irradiation of fueled cladding tubes; determine the effect of irradiation on properties (mechanical strength, coating adhesion, etc.). Testing could entail ion irradiation in advance of neutron irradiation. Note that demonstration of corrosion and erosion performance under irradiation can only be conducted with the cladding tube in contact with coolant flow at the appropriate PWR or BWR conditions. While initial irradiation could be conducted in a static capsule, coatings must also be demonstrated in contact with flowing coolant under irradiation.

Discussions among the ATF development community resulted in preliminary guidance for pre-screening evaluation tests under normal operating conditions (Table 2), tests for off-normal conditions (Table 3), and irradiation tests and associated post-irradiation examination (Table 4). While not comprehensive, these tables identify potentially applicable ASTM or industry standards for testing the property or behavior of interest.

Table 2. Guidance for Initial Pre-Screening Evaluation Tests – Normal Operating Conditions

Performance	Test	Condition	Analysis
Coating adhesion	Autoclave without scratch	72 hours ASTM G2M (Base metal) 400°C pure water/dry steam 1500 psi	Visual Weight change Recommendation: Consider performing tests based on this ASTM standard. The goal is to justify cladding performance to move forward.
	Address scratch tests per ASTM standards	ASTM C1624 for ceramic coatings	
Adhesion, Strength, and Ductility	Mechanical Adhesion Test Options for consideration • Bend Test-performance at various or increasing angles. How and when does it fail? • Burst test • Tensile test • Creep test • Pressurized tube • Thermal cycling • Coupon or coated tubing tests	Must survive a 1% strain ASTM C633 Adhesion, cohesion, strength of thermal sprays and coatings ASTM D4541-09E1 Strength of coatings using a portable tester	Determine the failure mode and strain at which failure occurs. When does catastrophic failure occur?
Compatibility	Chemical compatibility (substrate – coating – coolant) Thermal interdiffusion between coating and substrate	400°C – 450°C water/dry steam with relevant chemistry	Measure hydrogen pickup; NRC H_2 limits (ppm) are material specific (alloy dependent) Hydride issues Licensing issues Zirconium issues

Table 3. Tests for Off-Normal Conditions (to prove enhanced ATF behavior)

Test	Condition	Analysis
Simulated LOCA Test	<17% oxidation at 1200°C for <300 seconds (two-sided oxidation) or <1000 s (outer-surface oxidation) of 0.57-mm wall cladding; longer test times for thicker cladding Use existing standard for current alloys for guidance [10CFR50.46(b)]	Measure weight change or analyze cross-section Measure porosity Post Quench Ductility Test (PQD) Goal: Demonstrate improved performance relative to existing fuel/cladding
Incremental Temperature Increase Test	Raise steam temperature incrementally until failure; document process	Determine mode and time of failure

Table 4. Irradiation Tests

Test	Condition	Analysis
Ion Irradiation	Introduce damage via ion irradiation for initial (lower cost) assessment of behavior under irradiation	Microstructural analysis Coating adhesion Assessment of compatibility
Reactor – Steady state	Initial neutron irradiation of cladding coupons or tubes, with or without fuel, in a drop-in capsule	Microstructural analysis Coating adhesion Assessment of compatibility
Reactor – Loop test	Neutron irradiation of cladding coupons or tubes with flowing coolant (e.g. loop test)	Microstructural analysis Coating adhesion Assessment of compatibility
Reactor – Transient testing	To be determined	

5.2 Fuel Evaluation

Some of the necessary early fuel scoping tests were addressed in section 5.1, as some of the identified tests require testing of a fueled rodlet. Scoping evaluations that can be conducted on cold, clean fuel material prior to initial irradiation testing include:

1. Fuel – Clad Compatibility: Evaluation of chemical compatibility with the proposed cladding material at temperature for an extended period of time. The fuel-cladding interaction should be mapped over a range of temperatures, from normal operating conditions to accident scenarios.
2. Fuel – Coolant Compatibility: In the event of a cladding breach, the potential interaction between the fuel and coolant (liquid water and steam) must be understood over a range of possible temperatures.
3. Determination of Key Properties: Basic pre-irradiation fuel properties should be measured, including thermal conductivity, density, etc. (see section 3.1), over a range of temperatures. These properties should again be measured following irradiation.
4. Irradiation Testing and Post-Irradiation Analysis: Irradiation of fueled cladding rodlets (see Table 4).

6. Example Analyses for Hypothetical ATF Concepts

This section presents example analysis results for hypothetical accident tolerant fuel and / or cladding options (Figure 5, *Step 2b*). These analyses illustrate the impact of selected properties on fuel performance (sensitivity studies) and demonstrate how ATF concepts might be evaluated and subsequently ranked for their potential improvements over the current state-of-the-art fuel system during normal operation, design basis accident and severe accident conditions.

Where appropriate, the hypothetical design concepts used to generate the preliminary analysis results are denoted generically (ATF-1, ATF-2, etc.) to focus on the *process* of evaluating candidate materials, not to begin the down-selection itself. The example analyses do not present the economics associated with the evaluated options, nor do they presume to offer a ranking of the options. Fabricability, licensibility and the associated economics are necessary components of concept ranking and down-selection. Phase 2 analysis (*Step 5*, Detailed System Analysis in Figure 5) will require additional materials testing to fill existing data gaps for which material properties are either unavailable or must be assumed based on properties of related materials.

6.1 Core-Level Analysis

As discussed in section 4.1.1, a key element to the analysis of an accident tolerant fuel concept is the corresponding analysis of the resultant reactor system. Impact of a fuel design on the core power and flux distribution, reactivity coefficients, control parameters, neutronic behavior and thermal hydraulic behavior must be evaluated and understood at a given state-point and as a function of burnup.

This section presents an example core-level analysis for several hypothetical composite fuel types, denoted as ATF-1, -2, -3, -4. These hypothetical concepts only modify the fuel material properties, maintaining the standard cladding properties. A comparison of cycle length calculations is shown in Table 5 for these arbitrary composite fuel options. All results are shown relative to the reference UO_2 fuel system and a hypothetical pure UN fuel, which is not a viable candidate for a water-cooled reactor (the hypothetical ATF concepts are all nitride composites). The studied cases include a reference oxide case and two reference nitride cases. These cases are selected to illustrate the types of results that may be derived from the proposed analysis method rather than to provide comparative results for specific fuels. These composites contain two phases: (1) a primary oxide or nitride phase, and (2) a secondary phase. For the hypothetical fuel concepts represented in Table 5 the secondary phase is varied to determine the operational impact on fuel cycle length. These calculations were completed using the TRITON/NEWT lattice physics tools, included as modules in the SCALE package. These cycle length calculations, reported in effective full power days (EFPD), show the potential impact of the hypothetical fuels on reactor operation, where increased discharge burnup and cycle length may be desirable to industry. However, this result must be considered comprehensively with other analysis results to assess the potential overall impact of the proposed fuel option. Options shown in Table 5 that provide a longer cycle length alternatively would allow a reduction in the enrichment for comparable performance to the reference UO_2 system with resultant economic benefits.

Such analytical studies can be used to estimate the impact of fuel design and core fuel management parameters on cycle length. Table 6 includes example results for arbitrary micro-encapsulated fuel concepts (versus the composite fuels applied in Table 5). One fabrication parameter of interest for micro-encapsulated fuels is the particle packing fraction in the fuel pellets. A non-linear reactivity model was applied in these analyses to provide insight into the impact of two different packing fractions and the estimated leakage reactivity on the cycle length. Neutron leakage is a function of the neutron flux-energy spectrum, neutron mean free path, geometry, and fuel management scheme, among other factors. The example analysis results shown in Table 6 were calculated with the Serpent Monte Carlo reactor physics tool (Fridman and Leppänen 2011). If fuel fabrication studies indicate that the higher packing fraction is not feasible, then the preliminary analysis results would suggest that fuel cycle lengths would need to be reduced versus a reference UO_2 system (noted as 456 EFPD in Table 5).

Table 5. Calculated cycle length for hypothetical composite fuel options.

Fuel	UO_2 (0.95 TD)	UN (0.95 TD)	UN (0.8 TD)	ATF-1	ATF-2	ATF-3	ATF-4
Batch burnup (GWd/t)	17.6	17.7	18.0	17.9	17.9	17.7	17.8
Discharge burnup (GWd/t)	52.7	53.1	54.1	53.8	53.7	53.2	53.5
Cycle length (EFPD)	456	647	552	579	601	543	584

TD = theoretical density

Table 6. Calculated cycle length for hypothetical micro-encapsulated fuel options.

Leakage reactivity	Cycle burnup (EFPD) - 0.44 packing fraction	Cycle burnup (EFPD) - 0.5 packing fraction
$\rho_{LR} = 0.03$	411.3	474.8
$\rho_{LR} = 0.05$	400.8	461.9

Core reactivity parameters versus the relevant design limits are also vital parameters to consider. As an example, Table 7 shows the reactivity parameters for the hypothetical nitride composites ATF-2 and ATF-3 that were previously introduced in Table 5. The results are from a full-core equilibrium calculation applying PARCS with thermal feedback and a three-batch fuel management scheme in a classical out-in (scatter arrangement) fuel loading similar to a reference PWR. The reference reactor design is a state-of-the-art PWR offered by a large domestic reactor vendor. This analysis is a necessary boundary condition for transient safety studies that incorporate relevant neutronic feedback in events where feedback is important. As noted for Tables 5 and 6, all of the cases shown in Table 7 assume Zr-based cladding (modification made only to the fuel material). The noted “Reference Reactor Design Limits” are based on the safety analysis report for the reference PWR. These types of analyses can highlight when a proposed concept might produce unintentional consequences in the reactor operation. The results provided in Table 7 indicate that the hypothetical ATF-2 composite fuel would have a lower boron coefficient than allowable in the reference PWR. This result implies that soluble boron (with natural boron) would be less effective for reactivity control, and alternative methods of reactivity control or enriched boron would be required if this fuel were to be integrated into an existing PWR.

Core power and burnup distributions are important parameters that act as a vital boundary condition for almost all safety studies. They are also significant in evaluating key safety parameters in normal operation and transient conditions, such as DNBR and MCPR. Example core power distributions for a PWR equilibrium core with thermal hydraulic feedback and a realistic fuel management scheme as calculated by PARCS are shown in Figure 6. The color scales represent the relative power peaking in the assemblies. Red colors represent assemblies with the highest (above average) relative power and green colors represent assemblies with the lowest (below average) relative power, with yellow colors representing average (1.0) power peaking. The burnup values and radial power are axially averaged in Figure 6 for illustrative purposes. The actual core calculations, however, are fully three-dimensional with thermal hydraulic feedback. Only one-eighth of the core is shown because the core geometry and fuel management scheme considered in the calculations is octant symmetric.

Table 7. Example calculated equilibrium core reactivity coefficients for hypothetical composite fuel options (results produced using the PARCS tool with modified thermal properties).

Reactivity Coefficient	Reference Reactor Design Limit	UO_2	UN	ATF-2	ATF-3
Fuel Temperature Coefficient, pcm/K	-6.3 to -1.8	-2.7	-2.8	-2.9	-2.9
Moderator Temperature Coefficient, pcm/K	-72.0 to 0.0	-32.8	-32.1	-33.2	-33.4
Boron Coefficient, pcm/ppm	-13.5 to -5.0	-6.0	-5.0	-4.5	-5.0

Local power peaking factors are necessary to define the local power peaking relative to the design limit. For the reference PWR design in the example results shown here, the two key parameters are the three-dimensional local power peaking factor and the coolant enthalpy rise peaking factor. The local power peaking factor is shown in Figure 7 and the radial peaking factor is shown in Figure 8. In this case the enthalpy rise peaking factor is practically analogous to the radial peaking factor because the flow rate is assumed uniform in each thermal hydraulic channel. These results are for an equilibrium core calculated in an identical and realistic fuel management scheme with thermal hydraulic feedback. The PARCS code was modified to integrate appropriate thermal properties for the hypothetical nitride composites considered. For the example evaluation, the analysis cases show that, for the modeled core configuration, the hypothetical ATF options would potentially have greater safety margin than the reference oxide fuel due to the lower peaking factor. Additionally, the hypothetical ATF concepts would offer longer cycle lengths when compared to the UO_2 reference, a potential benefit. The reduced power peaking is not entirely an intrinsic property of the analyzed fuel concepts, but is a holistic impact of the modeled fuel properties and the parameters applied in the analysis (burnable poison loading, fuel management scheme, reactor geometry, etc.).

Core-level studies help determine appropriate thermal margins for proposed fuel concepts, which may be different than the conventional UO_2 system. Operational considerations, such as soluble boron concentration, are also important parameters that must be thoroughly studied, because they act as a boundary condition for studies of chemical reactions with the coolant. Critical boron concentration is shown in Figure 9 for the hypothetical composite fuel concepts ATF-2 and ATF-3. The soluble boron concentration was solved iteratively within each burnup step using the PARCS core simulator. As the fuel depletes, the concentration of boron in the moderator decreases. This behavior results in the so-called “boron letdown curve” of the calculated equilibrium cores. The example analyses show the calculated critical boron concentration to be higher for the hypothetical ATF concepts. Higher critical boron concentration is undesirable from the perspective of coolant chemistry and would impact potential behavior during a boron dilution accident; use of enriched boron may mitigate some of the negative consequences.

0.49	0.41			
0.0	0.0			
0.79	0.91	0.73	0.47	
32.9	11.9	0.0	0.0	
0.89	1.12	1.18	0.84	0.61
40.4	0.0	11.6	35.2	0.0
1.06	0.99	1.16	1.36	1.14
34.1	45.2	34.5	12.9	14.2
1.30	1.39	1.36	1.28	
24.7	18.3	24.5	32.9	
1.04	1.33	1.16		
45.3	0.0	40.4		
1.15	1.00			
24.7	43.4			
0.96				
43.4				

Legend:

1.40
1.20
1.00
0.80
0.60
0.40

1/8-Core UO_2

Peak relative power:

1.39	Relative power
18.3	Burn-up (GWd/t)

0.57	0.49			
0.0	0.0			
0.83	0.98	0.82	0.52	
33.9	12.8	0.0	0.0	
0.91	1.20	1.20	0.85	0.64
40.3	0.0	12.3	34.8	0.0
1.01	0.97	1.10	1.29	1.10
34.5	44.1	35.0	13.4	15.1
1.23	1.31	1.26	1.18	
24.7	19.1	24.1	33.9	
1.03	1.34	1.12		
44.6	0.0	40.3		
1.09	0.99			
24.7	42.8			
0.91				
42.8				

Legend:

1.40
1.20
1.00
0.80
0.60
0.40

1/8-Core ATF-2

Peak relative power:

1.34	Relative power
0.0	Burn-up (GWd/t)

Figure 6. Eighth core radial power and burn-up distributions for equilibrium cycle at end-of-cycle for UO_2 (0.95 TD) and the hypothetical ATF-2 composite fuel concept.

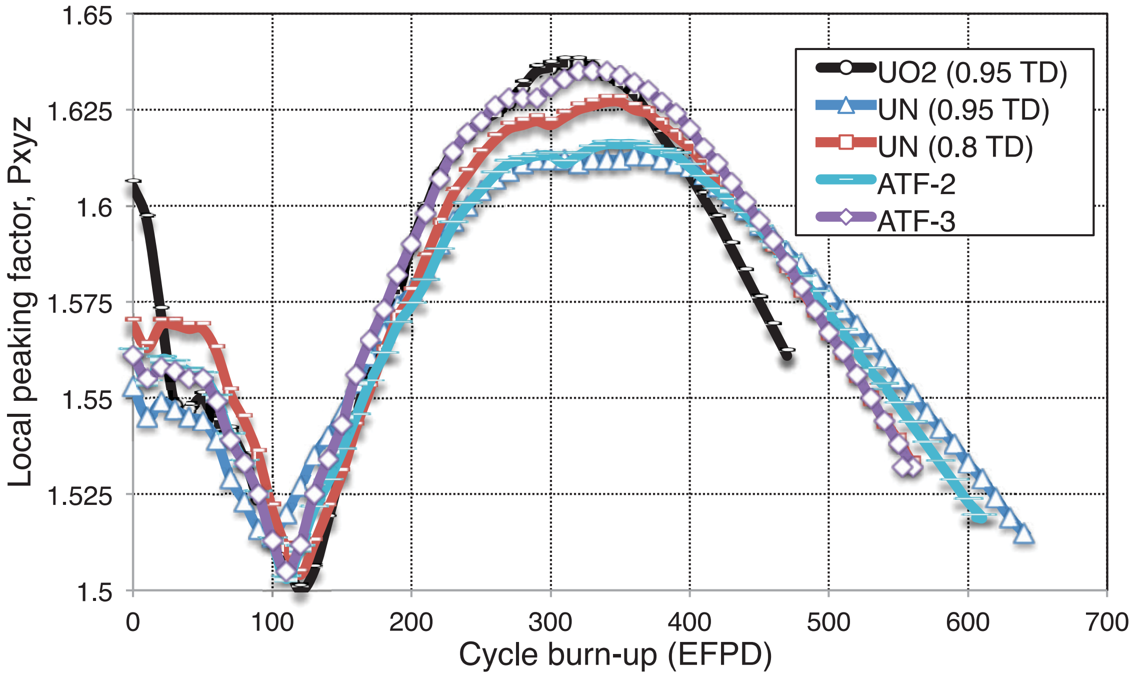

Figure 7. Local power peaking factors in an equilibrium cycle with identical fuel management for several hypothetical composite accident tolerant fuel concepts.

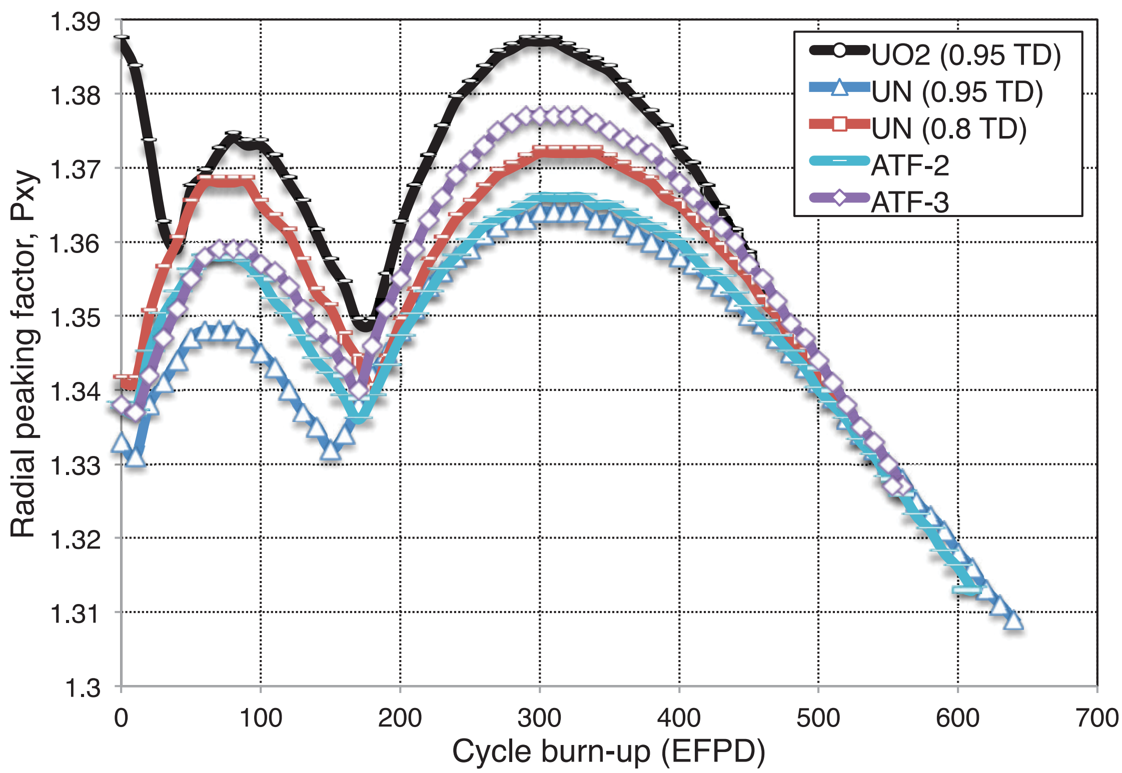

Figure 8. Radial peaking factor in an equilibrium cycle with identical fuel management for several hypothetical composite accident tolerant fuel concepts.

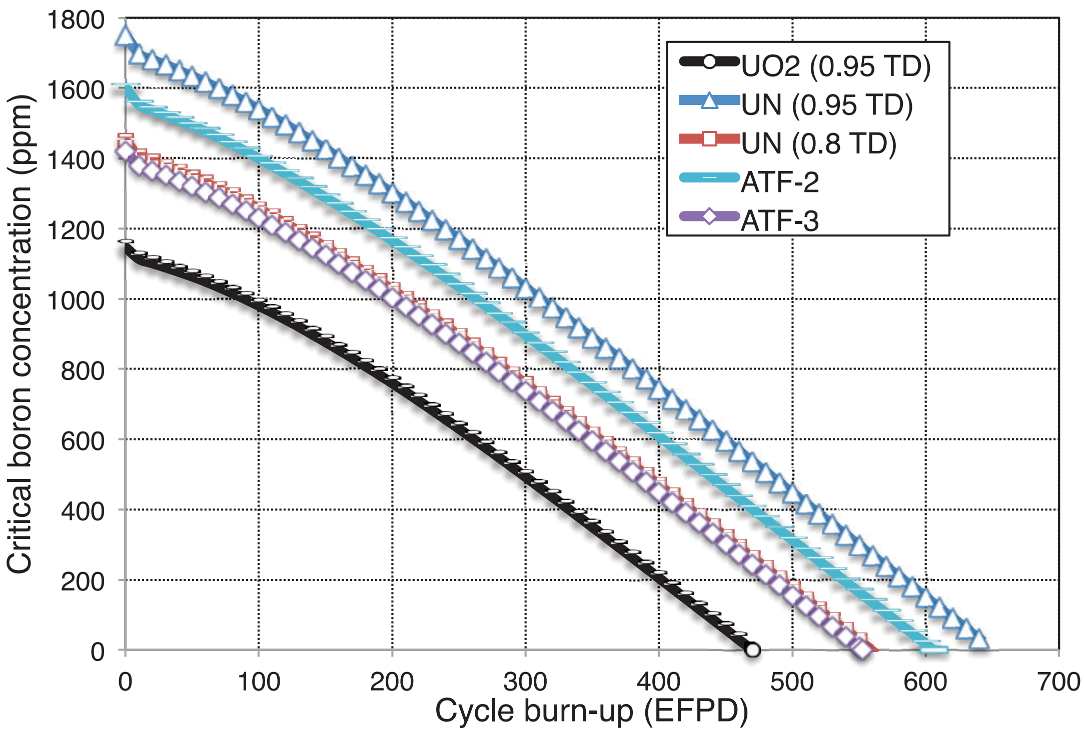

Figure 9. Critical boron concentration in an equilibrium cycle with identical fuel management for several hypothetical composite accident tolerant fuel concepts.

6.2 Impact of Enhanced Fuel Thermal Conductivity

Recent studies have addressed the impact of fuel thermal conductivity on enhanced accident tolerance during DBA and BDBA loss of coolant scenarios (Terrani et al. 2013). The code scaling, applicability, and uncertainty (CSAU) evaluation methodology initiated in the 1970s was used to identify and rank the most important factors contributing to core response during design basis large break (LB) LOCAs (Boyack et al. 1990). Many of the studies carried out based on CSAU methodology have been specific to PWRs, and the stored energy in the fuel has been considered to be an important parameter in determining the fate of the reactor under LBLOCAs (Wilson et al. 1990). The stored energy in the fuel is inversely proportional to the fuel thermal conductivity. The purpose of the current study was to examine the effect of stored energy in the fuel (which is directly related to the temperature gradient across the fuel pellets) on the thermal hydraulic response of the core for both BWRs and PWRs during a design basis LBLOCA scenario. A LBLOCA is considered the case under which core response is most sensitive to the initial stored energy in the fuel.

The magnitude and the dissipation rate of stored energy from the reference fuel and the hypothetical fuel having enhanced thermal conductivity were initially examined using a simple numerical model. This model solves the one-dimensional transient heat conduction equation in a PWR fuel rod subjected to a scram without any loss of cooling. Results indicate that the temperature profile in the fuel rapidly flattens and the stored heat is almost completely dispensed into the coolant after about 15 s. The small temperature gradient remaining across the fuel pellet is subsequently due to the small amount of decay heat that is being generated in the fuel volume.

The effect of assumed enhanced fuel thermal conductivity on fuel temperature evolution during a LBLOCA is examined using the TRACE code (US NRC 2012) to perform full-core simulations with fuel thermal conductivity at nominal and arbitrarily increased values. In this manner, the sensitivity of the fuel temperature rise to fuel thermal conductivity during the specified transient is analyzed and compared within the PWR and BWR reactor platforms.

6.2.1 PWR Performance under LOCA Conditions (TRACE Analysis)

A typical 4-loop Westinghouse design PWR loaded with Westinghouse 17×17 fuel assemblies was employed for this study. The PWR TRACE input file models all of the major flow paths in the system needed to perform large and small break LOCA and steam generator tube rupture simulations. All four recirculation loops are modeled explicitly (Figure 10). Geometry and physical property details are included in Terrani et al. (2013).

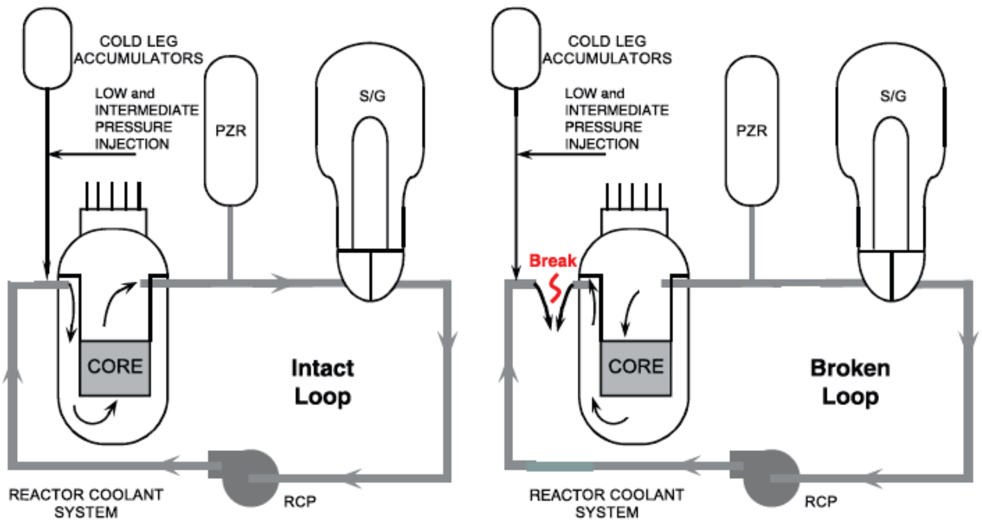

Figure 10. Schematic drawings of conceptual intact and broken loops in a PWR core subjected to LBLOCA (modified image after U.S. NRC).

A LBLOCA was simulated using the TRACE steady-state input model of the PWR. The LBLOCA was a double-ended guillotine break in cold leg 1. For this event, it was assumed that all ECCS systems were available in the simulation. The large break (80% break area) is assumed to be located between the ECCS injection line and the RPV. Double-ended cold leg break, reactor coolant pump (RCP) trip, and reactor/turbine trip are all assumed to initiate at time zero. The safety injection actuation signal is delivered 6 seconds later, and accumulator injection (intact loops) begins after 13 seconds. ECCS pumped flow is initiated at 36 seconds.

The results presented in this section (6.2) are for the hottest rod in the core with the axial linear heat rate (LHR) distributions shown in Figure 11 for the BWR and PWR platforms separately. Taking into account all the pertinent conduction, convection, and radiation heat transfer phenomena, the code reports global maxima for fuel centerline and cladding temperatures.

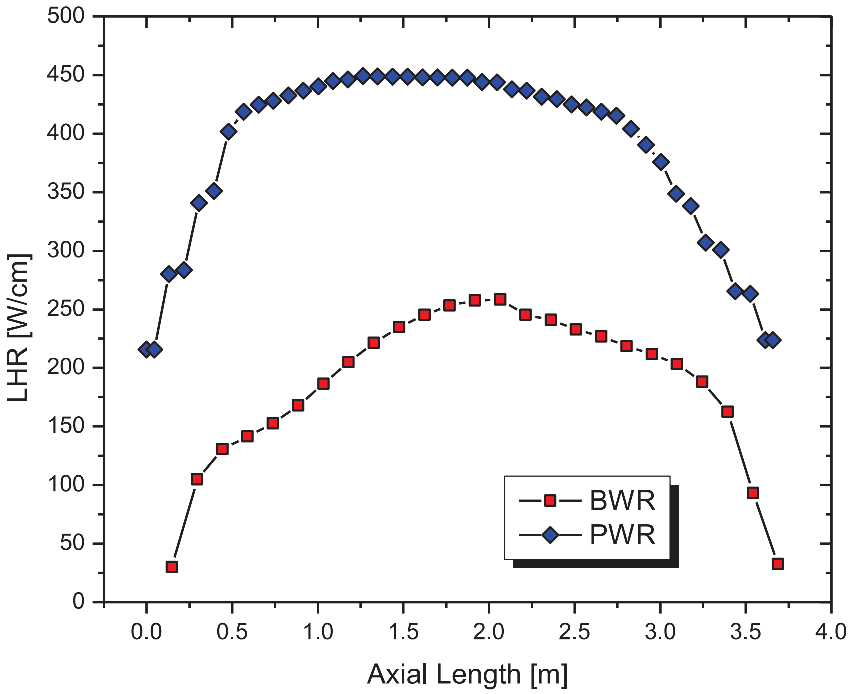

Figure 11. Linear power profile for the hottest rod under normal operating conditions in the modeled BWR and PWR cores prior to LBLOCA simulation.

Figure 12 shows the fuel centerline temperature (FCT) and the peak cladding temperature (PCT) for the hottest rod in the 4-loop PWR during the LBLOCA simulation with nominal and (computationally) enhanced fuel thermal conductivity. A notable difference in the magnitude of FCT and PCT during the course of the accident is apparent for various cases. Note that the assumed increases in fuel thermal conductivity are arbitrary and the associated analyses are intended to clarify the sensitivity of fuel and cladding temperatures to thermal conductivity during accident conditions. These thermal conductivities may or may not be achievable in practice. Analysis results indicate that an increase in fuel thermal conductivity by 200% and 500% would reduce PCT by 56°C and 92°C, respectively; also, increased fuel thermal conductivity would result in faster quench times [for nominal, 200% and 500%, the calculated quench times are 213, 194, and 177 seconds, respectively (from Figure 12)]. The magnitude of this change normalized against the thermal resistance of the fuel (directly proportional to the stored energy) is consistent with earlier analyses focused on parameter uncertainty (Wulff et al. 1990).

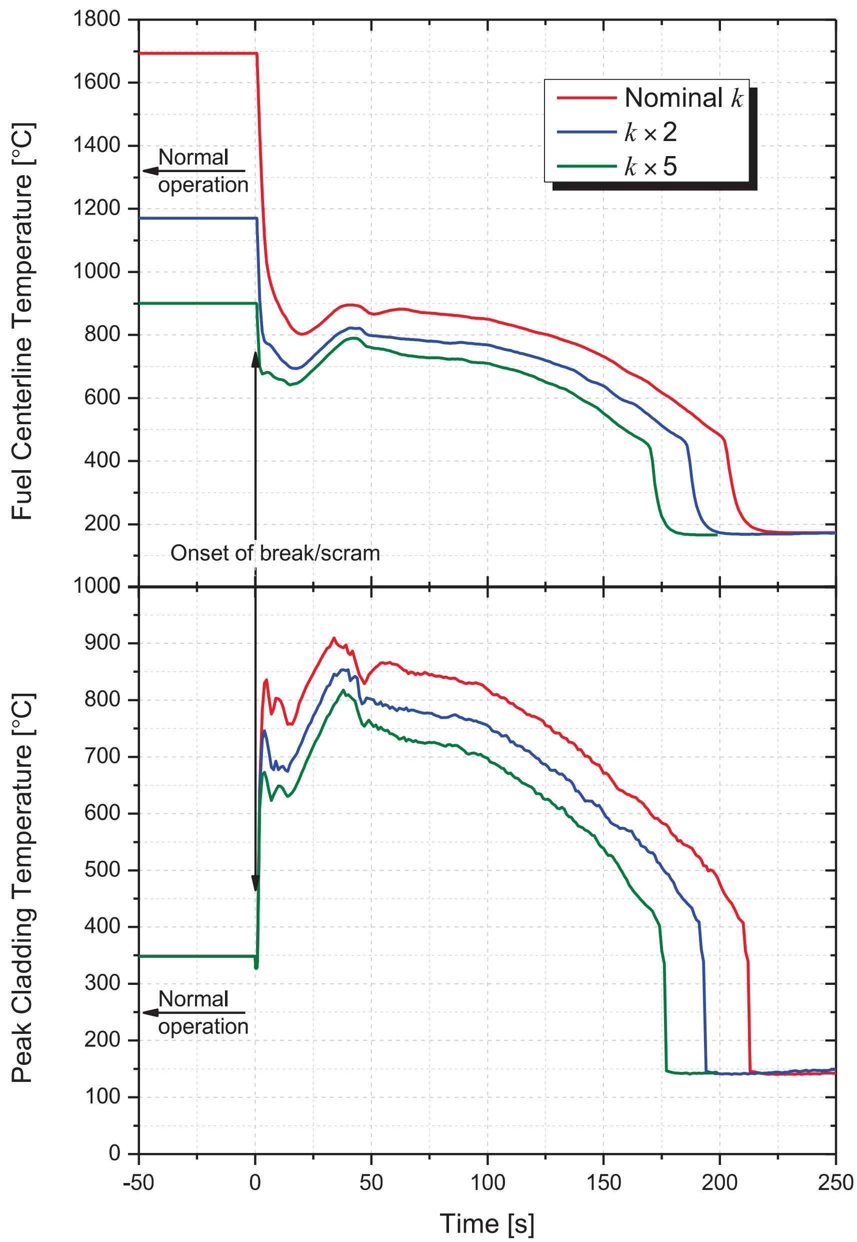

Figure 12. FCT and PCT evolution of the hottest rod during a simulated LBLOCA in typical 4-loop PWR loaded with fuel at nominal and computationally enhanced thermal conductivity.

6.2.2 BWR Performance under LOCA Conditions (TRACE Analysis)

A typical General Electric BWR4 plant with a Mark-I containment loaded with GE14 bundles (Nuclear Engineering International 2004) was employed to evaluate the potential impact of increased fuel thermal conductivity on BWR system response to a LOCA. The BWR TRACE model includes all of the major flow paths and system components to perform large and small break LOCA simulations. Both recirculation loops are modeled explicitly. More detailed description of the analysis framework is described in Ott, Robb and Wang (2013).

A double-ended guillotine recirculation suction line break (0.36 m^2) is assumed to mark the onset of transient at time zero, as shown in Figure 13. The two recirculation pumps are tripped and the reactor is scrammed at the same time. Only two LPCIs are assumed to be available. High-pressure core sprays and low-pressure core sprays, as well as the automatic depressurization system, are all assumed to be unavailable. LPCI injection is actuated as designed starting at 91 seconds after the break.

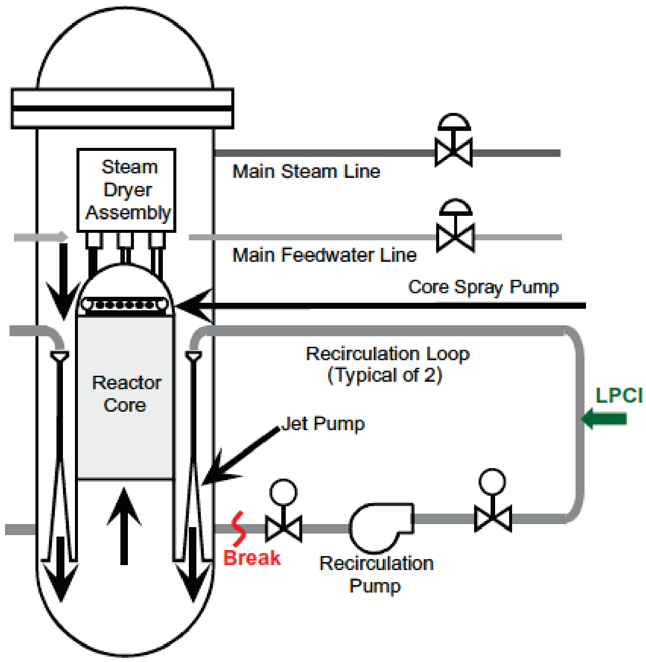

Figure 13. Schematic of a conceptual BWR core subjected to LBLOCA (modified image after U.S. NRC).

Figure 14 shows the fuel centerline and peak cladding temperature (FCT and PCT) for the hottest rod in the GE BWR-4 (Mark I containment) LBLOCA simulation with nominal and computationally enhanced fuel thermal conductivity. As noted for the PWR cases, the assumed increases in fuel thermal conductivity are arbitrary and may or may not be achievable in practice; the associated analyses are intended to clarify the BWR system sensitivity to thermal conductivity during accident conditions. The radial fuel temperature profile flattens rapidly for all three cases upon initiation of the scram. The FCT and PCT are essentially identical for all cases 15-s after the scram, with only a negligible difference between fuels with nominal and computationally enhanced thermal conductivity. At any time after the onset of the break/scram, the PCT for fuel with nominal thermal conductivity is nearly identical with that of the fuel with enhanced (up to 500%) thermal conductivity. The LPCI that is actuated at 91 seconds after the scram eventually results in core quench at about 170 seconds with only minor differences due to fuel thermal conductivity. Note that while the assumed increase in fuel thermal conductivity would significantly alter the temperature profile across the pellet during normal operation, results indicate that it would have no discernible effect on the fuel and cladding temperature under this LBLOCA scenario.

The lack of any notable difference in PCT evolution among these three BWR cases is inherent to the thermal-hydraulics of the BWR. Although pumps trip at the onset of the break, the pumping action coasts down slowly as the jet pump continues to inject water upwards into the core. The water moves upward within the fuel bundles, undergoes phase change as under normal operating conditions, and removes the heat generated inside the fuel rods. As shown in Figure 14, this cooling action continues for roughly 30 seconds during which the stored energy in the fuel is removed (FCT for the various cases is nearly identical) and the PCT for all cases drops. Once the stored energy from the fuel is removed, the rod responses are identical among all the examined cases since the decay heat solely depends on the power history.

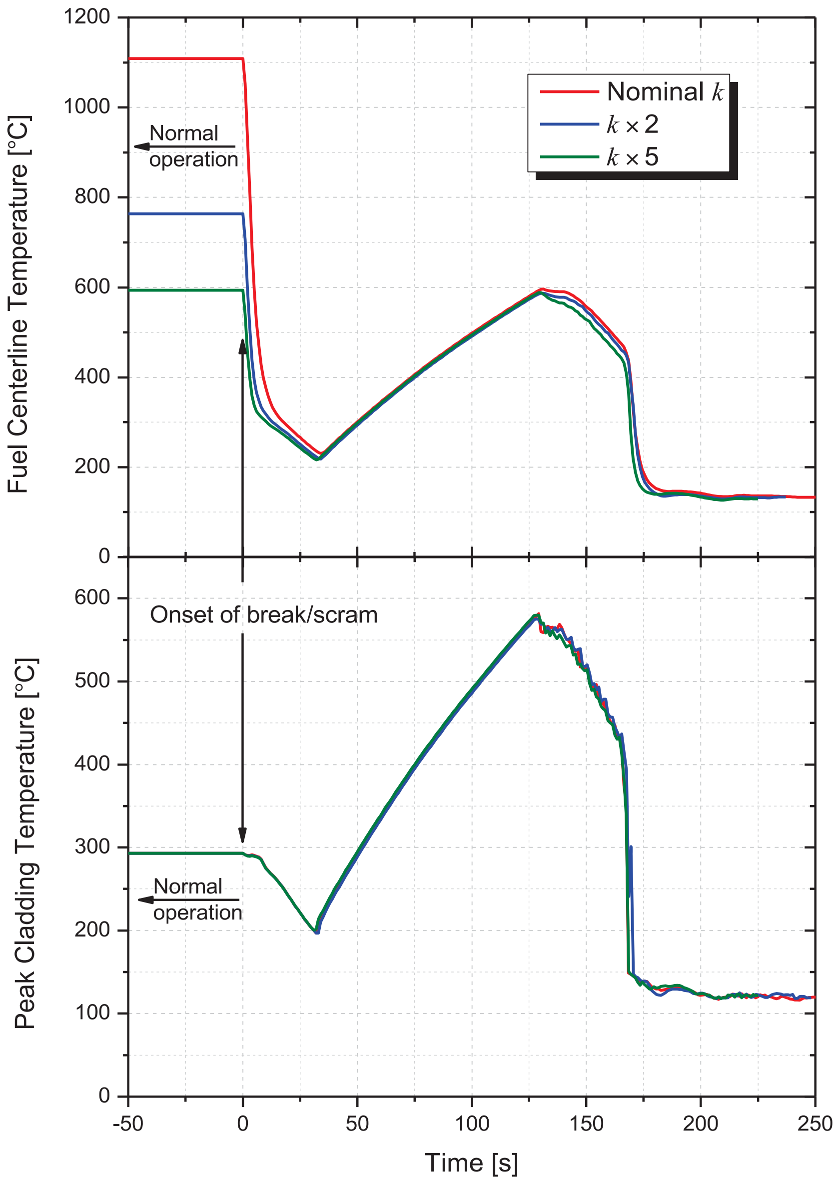

Figure 14. FCT and PCT evolution of the hottest rod during a simulated LBLOCA in typical BWR-4 loaded with fuel at nominal and computationally enhanced thermal conductivity.

6.2.3 Peak Cladding Temperature Sensitivity to Fuel Thermal Conductivity

While calculations indicate almost no effect from fuel thermal conductivity on PCT evolution for BWR rods, the sensitivity studies indicated a notable difference for PWR fuel pins. This disparity results from the thermal-hydraulics aspects of the two different cores and their ability to remove the stored energy in the fuel during the transient. In the BWR core, a break in the suction line does not alter the flow direction. Although both of the recirculation pumps are tripped, they continue to inject water down the jet pumps during coast down and facilitate upward coolant flow into the core. The water, as is the case under normal operating conditions, flashes into steam and removes heat as it climbs upward within the fuel bundles. The cooling process continues for ~30 seconds, which is sufficient compared to characteristic heat transport time across the radial direction of the fuel (~ 15 seconds, as noted previously). Accordingly, the magnitude of the remaining stored energy in the fuel is negligible and no discernible effect on PCT evolution is observed during later stages of the transient.

For the PWR, the break in the cold leg results in rapid coolant discharge from the core. The coolant entering from the cold leg in the three remaining intact loops is unable to continue flowing upward into the core during the depressurization process. Accordingly, no significant heat removal from the fuel takes place and the initial stored energy in the pellet simply redistributes across the pellet and cladding (with, as a result, increased PCT), as opposed to being removed. The larger magnitude of stored energy in the fuel at the nominal thermal conductivity results in a larger PCT shortly after the onset of the break. This higher temperature persists across the core until it is quenched. The expected drop in PCT upon arbitrarily doubling the fuel thermal conductivity (56°C per the current calculation) would provide some cladding burst margin across the core, resulting in a lower burst fraction (Powers and Meyer 1980).

6.3 Baseline PWR Accident Case: MELCOR Analyses

The Three Mile Island Unit 2 (TMI-2) loss-of-coolant accident (LOCA) was selected as the baseline PWR accident for examining the safety merits of the proposed ATF concepts. The TMI-2 accident was caused by a small-break LOCA in a two-loop Babcock & Wilcox (B&W) PWR (Henry 2001). Unaware of the small LOCA caused by a stuck-open pilot-operated relief valve, throughout the first 300 minutes of this event the reactor operators manually overrode and operated the Emergency Core Cooling System (ECCS) to inject water into the reactor pressure vessel (RPV) to cool the reactor core, while simultaneously draining the primary cooling system by way of the coolant letdown system in an attempt to prevent the primary system from becoming incompressible (that is, becoming completely filled by water).

Two reactor coolant system (RCS) pumps were operated during the first 100 minutes of this event, providing adequate core cooling, but increasing pump vibration caused by two-phase water entering these pumps forced the operators to stop RCS pump operation. At ~140 minutes, the operators became aware of the LOCA and terminated this loss of coolant by closing the appropriate electromatic relief block valve. However, as a result of the accident, the core became uncovered and heated up, causing spacer grid loss by melting, cladding ballooning, control rod meltdown, fuel rod oxidation, hydrogen production, cladding melting, melt candling, fuel melting, molten fuel/cladding pool formation, and a portion of this molten pool pouring into the RPV lower head.

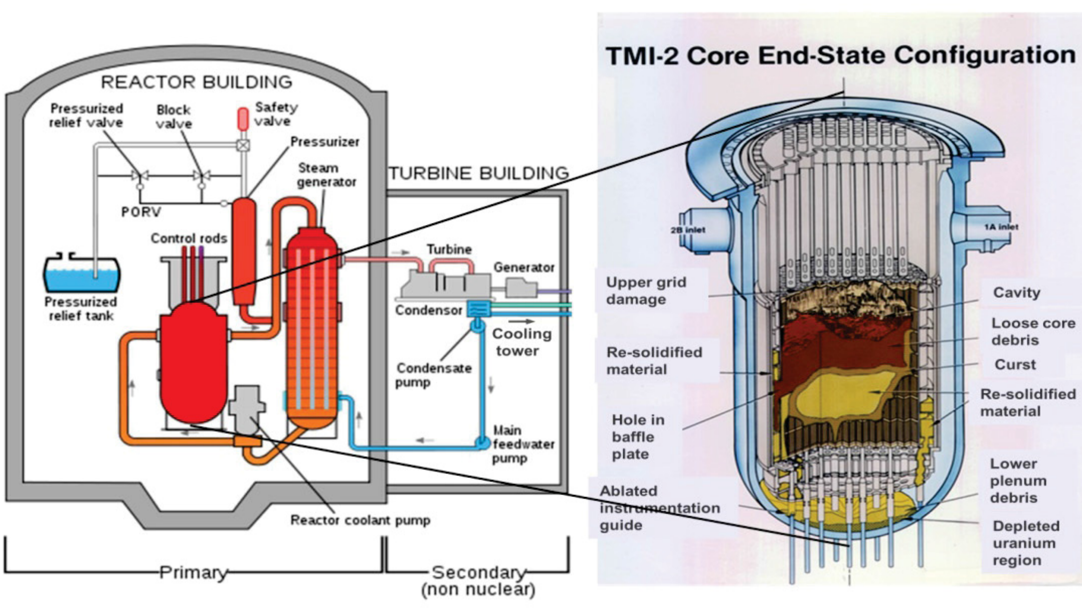

Figure 15. Schematic of TMI-2 reactor LOCA accident.

6.3.1 Simulation Setup

Candidate "accident tolerant" cladding materials should illustrate greater stability than zirconium alloys in steam environments and provide the potential for greater structural stability during severe accidents. Researchers have studied the oxidation behavior of various materials in steam and oxygen, and some materials have been found to have corrosion behavior that differs substantially. Key parameters of importance in the oxidation behavior of a material include the exothermic (or endothermic) nature of the reaction, the oxidation rate, and the products of oxidation, including both gaseous products (potentially combustible) and oxide products that often form a scale on the surface of the material. The diffusion rate of oxygen or water vapor through this oxide layer is also important, as it may temporarily protect the underlying material.

The specifics of the oxidation behavior and material properties (thermal conductivity, specific heat, density, emissivity, and heat of formation) of a candidate cladding and / or structural material can be input into the MELCOR input model and subsequently benchmarked against the literature to verify that the models properly represent the documented behavior of the material in question.

To analyze this event, a version of the MELCOR severe accident analysis code (Gauntt et al. 2005) is being developed at the INL to accommodate the proposed cladding options. An initial version has been completed that substitutes the properties of a candidate cladding material for Zircaloy in MELCOR's reactor core oxidation and material property routines (denoted "Candidate Cladding 1" in the ensuing results). The MELCOR input model used for this accident analysis is that developed by Sandia National Laboratories (New Mexico) and Innovative Technology Solutions Corporation, Albuquerque, NM (Gauntt et al. 2002). Figure 16 contains the predicted RPV pressure during this accident. Also noted in this figure are the times when the RCS pumps were tripped in both cooling loops of the reactor (A and B) because of excessive pump vibration, the ERV was closed, a single pump in the B loop was restarted, and the high-pressure injection system restarted. This modified version of MELCOR was benchmarked against available experimental data to ensure that air and steam oxidation theory were correctly implemented in the code. Additional code modifications are being implemented to further improve the specificity in defining components fabricated from non-standard materials.

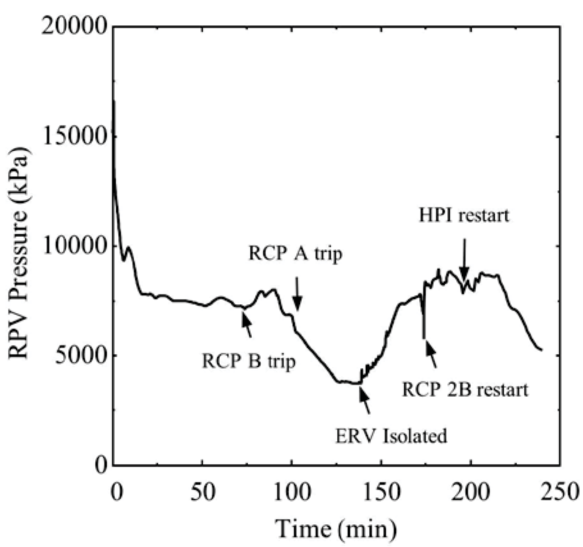

Figure 16. MELCOR predicted RPV pressure during the TMI-2 accident sequence.

Figure 17 presents a comparison between the predicted peak-cladding temperature for the TMI-2 accident sequence for Zircaloy and Candidate Cladding 1 (peak occurs at the top-center region of the core). For the first 100 minutes of this accident, cladding cooling was maintained by the RCPs in spite of the loss of reactor coolant inventory created by both the LOCA and operator action of draining water from the RCS through the letdown system. According to the MELCOR prediction, the RPV water level drops below the top of active fuel height 100-min into the accident, and the uncovered fuel rod cladding starts to heat up by decay heating in what is now a slow moving steam environment produced by boil-off of remaining core water inventory. By 125 minutes, the Zircaloy clad case reaches temperatures where oxidation heating produced by the steam/Zircaloy reaction starts to accelerate the cladding temperature rise. Cladding failure is predicted to occur at ≈145 minutes, producing a loss of cladding and core geometry in this core region. In the MELCOR code, the failure criterion is simply a temperature set point based on experimental data and modeling experience, which coincides with the melting temperature of a UO_2-ZrO_2 eutectic (≈2500 K). At this time, MELCOR transfers the rod's fuel and cladding into a debris field in the coolant channel, where channel blockage and debris heating accelerates core damage in lower regions of the core.

In contrast, the core using Candidate Cladding 1 reaches a predicted peak cladding temperature of 1830 K, which is below the melting temperature of the protective oxide scale, and well below the failure temperature for the material modeled. Temperatures begin to decrease after ERV isolation even under the low makeup water injection. Large decreases are seen when a pump in loop B restarts (at ≈175 minutes) and the Emergency Core Cooling System high-pressure injection is restored (at ≈195 minutes).

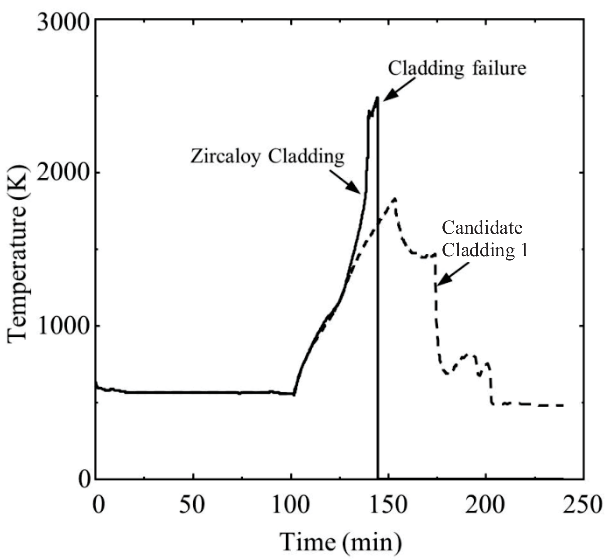

Figure 17. Comparison of MELCOR-predicted maximum cladding temperature during the TMI-2 accident sequence for standard Zircaloy cladding and a possible accident tolerant cladding, denoted as "Candidate Cladding 1."

A primary reason for the starkly different results between the Zircaloy cladding and the Candidate Cladding 1 during the TMI-2 accident sequence is that the oxidation heating of the candidate material is approximately two orders of magnitude less than that for Zircaloy, as is evident in the predictions contained in Figure 18. The Zircaloy heating at 150 minutes exceeds the core decay heat by a factor of 5. In addition, because of the lower temperatures, the Candidate Cladding 1 material would produce more than an order of magnitude less combustible gas during this accident sequence than Zircaloy (note Figure 20).

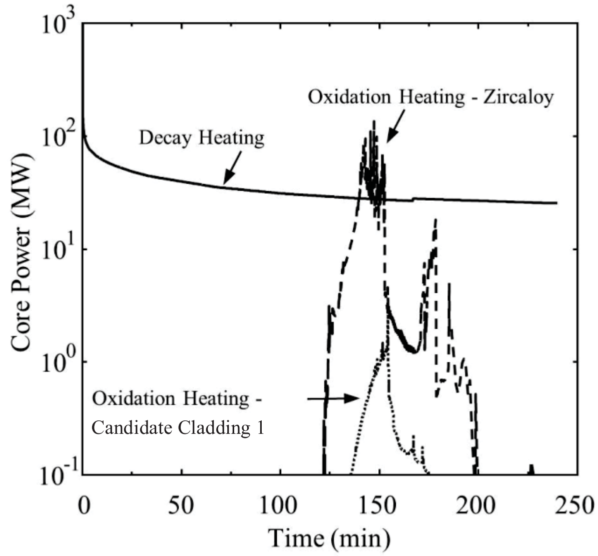

Figure 18. Comparison of MELCOR-predicted clad oxidation heating produced during the TMI-2 accident sequence for standard Zircaloy cladding and a possible accident tolerant cladding, denoted as "Candidate Cladding 1."

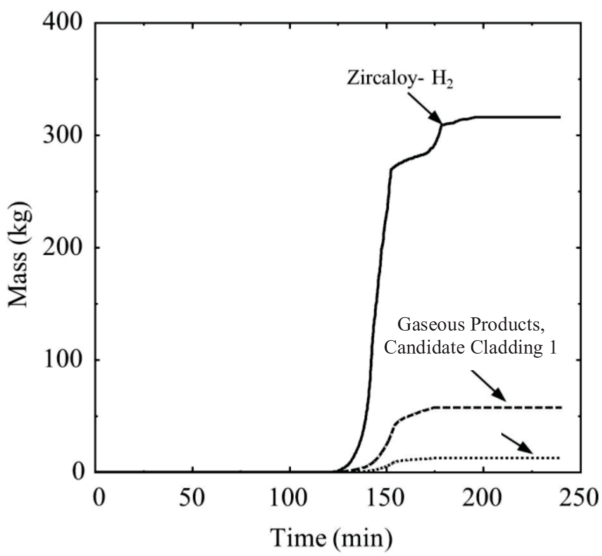

Figure 19. Comparison of MELCOR-predicted combustible gases produced during the TMI-2 accident sequence for standard Zircaloy cladding and a possible accident tolerant cladding, denoted as "Candidate Cladding 1."

6.3.2 MELCOR Parametric Studies on Oxidation Rate and Cladding Conductivity

The present uncertainty associated with the benchmarking of the Candidate Cladding 1 oxidation model prompted a parametric study that was undertaken to determine the impact of uncertainties associated with the oxidation model that was applied in this accident analysis. For this case, the assumed Candidate Cladding 1 oxidation energy was arbitrarily increased by a factor of 2 and 5 over the assumed base oxidation energy. Figure 20 presents a comparison of the MELCOR-predicted peak-cladding temperature for each of these cases. In this example, the 2x oxidation heating assumption is still a factor of ≈5 below the calculated Candidate Cladding 1 decay heat at 150 minutes (still significantly below Zr-alloy oxidation heating). As a result the predicted peak-cladding temperature differs between the base case and the 2x oxidation rate case by only ≈45 K. When the assumed oxidation rate is arbitrarily increased by a factor of 5 over the expected rate (based on limited data available for the modeled material) the predicted peak-cladding temperature reaches 2770 K, approaching the failure temperature of the evaluated material. These results confirm that the oxidation energy and rate are significant factors in predicting the performance of candidate ATF concepts under accident cases. Hence, these values should be determined experimentally for each of the specific materials under consideration to improve the accuracy of analysis predictions.

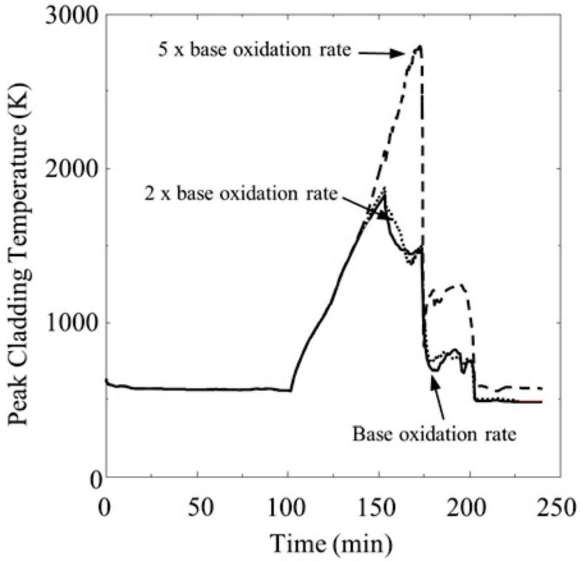

Figure 20. Comparison of MELCOR-predicted peak cladding temperature during the TMI-2 accident sequence for Candidate Cladding 1 using the benchmarked oxidation model with the assumed oxidation base rate (per the available data for this material) and with the oxidation rate arbitrarily increased by a factor of 2 and 5.

For some candidate cladding materials, the material thermal conductivity decreases with increasing irradiation fluence. To illustrate the impact this would have on MELCOR-predicted results for the TMI-2 accident, a second parameter case was performed in which the material thermal conductivity assumed for Candidate Cladding 1 was decreased by a factor of ten to simulate the impact of irradation. The predicted maximum fuel cladding temperatures for these cases are shown in Figure 21. The impact of reduced cladding thermal conductivity on the modeled accident appears to be negligible. This result is not surprising given that the modeled cladding is less than 1 mm thick. Even at a reduced conductivity, the primary heat transfer resistance is on the steam side since the high-temperature steam flowrate is only ≈ 2 m/s.

Based on the results presented in this section, the modeled Candidate Cladding 1 demonstrates a potential safety advantage over Zircaloy for an accident where cladding oxidation becomes a dominant heat source. The results of this model are presented as an example of the type of analyses that could be employed to evaluate the potential advantage of candidate materials. Although preliminary I nature, the presented results suggest a safety advantage for Candidate Cladding 1 during a TMI-2-type accident assuming that the material could survive the thermal-mechanical loads under these harsh accident conditions. The preliminary nature of this prediction is due to the uncertainty regarding material oxidation rates, and the structural survivability of the candidate material at very high temperatures and during rapid thermal quenching. These uncertainties can be resolved by additional research to further characterize the behavior and evolution of key properties at temperature and under irradiation.

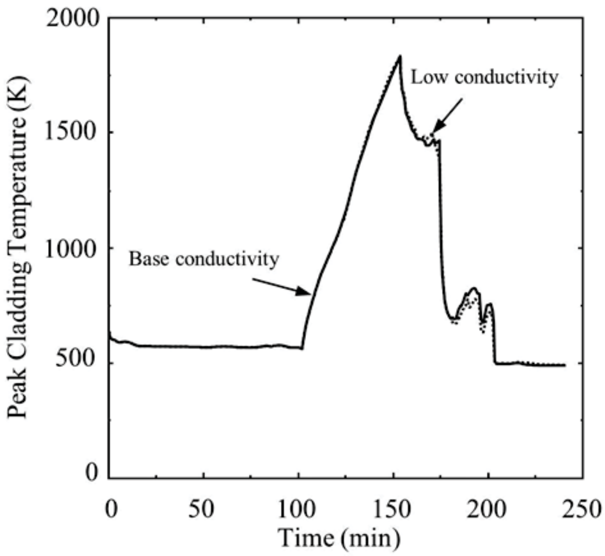

Figure 21. Comparison of MELCOR predicted peak cladding temperature during the TMI-2 accident sequence for the assumed Candidate Cladding 1 thermal conductivity (per the available data for this material) and with the cladding thermal conductivity reduced by a factor of 10.

Figure 22 shows comparative results for the MELCOR-predicted peak cladding temperature for two candidate accident tolerant cladding materials, "Candidate Cladding 1" and an additional "Candidate Cladding 2" material. These modeled materials differ in oxidation behavior, heat of oxidation and failure temperature, all of which were adjusted either via user input or by adjustment and recompilation of the MELCOR code. Estimated properties were defined and informed assumptions were made using existing data for both candidates studied. The heat of reaction estimated for "Candidate Cladding 2" was ≈5 times less on a per unit mass basis than "Candidate Cladding 1." The peak cladding temperatures for both candidate materials are similar for this accident because this temperature is primarily a result of the fuel decay heat. The melting or decomposition temperature for each candidate material becomes the key discriminator between these options.

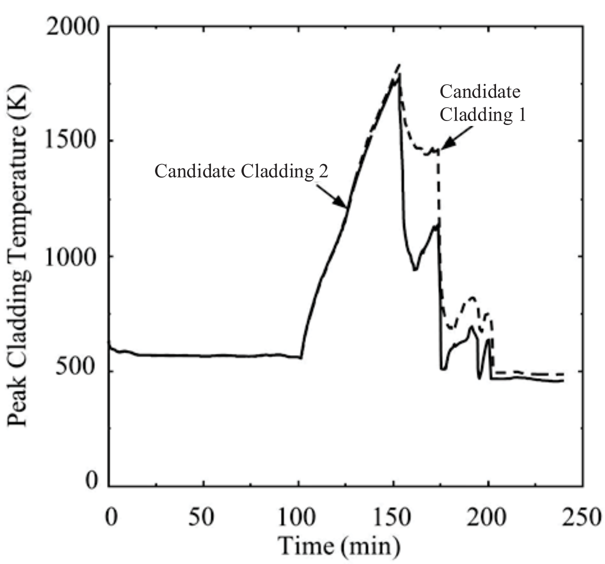

Figure 22. Comparison of MELCOR-predicted peak cladding temperature during a TMI-2 accident for two candidate cladding materials differing in differ in oxidation behavior, heat of oxidation and failure temperature.

6.4 Baseline BWR Accident Case: MELCOR Analyses

MELCOR analyses were also conducted for a baseline BWR case. The analyses used the Peach Bottom (GE BWR4/Mark I) MELCOR 1.8.5 model described by Francis (2006). The model contains the key equipment important in modeling severe accidents (reactor core isolation cooling [RCIC] and high pressure coolant injection [HPCI] systems, safety relief valves [SRVs], water storage tanks, etc.). Although the model contains a detailed representation of the reactor building, the current study focused on the impact of accident tolerant fuels on the in-vessel accident progression.

6.4.1 Simulation Setup

As discussed in section 4.4, the fuel and cladding properties must be modified in MELCOR to reflect the candidate ATF materials, with the material melt temperature, oxidation reactions, oxide properties, and relocation characteristics explicitly specified. This preliminary BWR analysis was conducted for a candidate accident tolerant fuel and cladding material. However, some characteristics and material properties for these advanced materials had to be assumed in the model prior to their determination experimentally. As discussed previously, due to the manner in which the base code is constructed, modification of cladding properties to those of an alternate material in the standard MELCOR code also results in modification of the channel boxes and other structural materials initially designated as "Zircaloy" in the model.

As was assumed in the PWR case, the hypothetical fuel and cladding were assumed to have the same geometry, dimensions, and, thus, flow characteristics as the baseline UO_2-Zr alloy core. To accomplish this the material masses were scaled by their densities to retain the same material volumes. No modifications were made to the core flow paths and flow resistances; radionuclide composition, distribution, and release physics; or other model parameters not discussed, such as heat transfer correlations. Modification of these parameters could be incorporated in future studies.

The fuel rod collapse (failure) temperature was set to 2800 K. The holdup of molten material by an oxide shell was set to occur at a minimum oxide thickness of 1 mm and for oxide temperatures below 1700 K. The "Zircaloy" material oxidation kinetics in MELCOR is governed by parabolic kinetics with consideration of gaseous diffusion limits and accounts for oxidation by both H_2O and O_2. Oxidation of the fuel material is not modeled in MELCOR.

For the example analyses discussed in this section, the oxidation rate equations for the hypothetical accident tolerant cladding material were modified to reflect 1% of the values of Zircaloy. In addition, a simulation where oxidation was effectively turned off (0%) and a simulation using the default UO_2 and Zircaloy properties were conducted to bound the range of possible cladding oxidation behavior, similar to the parametric study conducted for the PWR case.

A major code limitation was encountered in that the heat of reaction during oxidation is hard coded in MELCOR (Clad-H_2O: 5.797×10^6 J/kg clad at 298.15 K, Clad-O_2: 1.2065×10^7 J/kg clad at 298.15 K). While this was modified in the PWR TMI-2 analysis that employed MELCOR 1.8.6 (via modification of the base code and recompilation) it was not adjusted in the MELCOR 1.8.5 BWR model. The Zircaloy heat of reaction is approximately 10 times that of stainless steels. As was shown in the PWR case for Zr-alloy cladding and will be shown for the BWR case, the heat released due to cladding oxidation can be a major contributor to the overall heat load.

The BWR accident scenarios investigated include a short-term station blackout (STSBO) and a long-term station blackout (LTSBO). The STSBO assumes that the reactor shuts down (scrams), all power is lost and all injection is lost at time zero. The LTSBO assumes the reactor scrams and AC power is lost at time zero and DC power (batteries) is lost at 8 hours. The RCIC and HPCI are available for operation during the LTSBO until battery power fails. For these scenarios, the ability to inject water via the control rod drive (CRD) pumps and low pressure ECCS pumps (no AC power) and diesel driven pumps is also assumed to be lost. An additional scenario, a mitigated STSBO (MSTSBO), is analyzed in which operators take action to inject water through the CRD pumps at a rate of 110 GPM (25 m^3/h). This action provides a steady flow of high-pressure water, injected into the bottom of the RPV.

6.4.2 BWR Short Term Station Blackout

For the standard UO_2-Zr alloy system, rapid cladding oxidation occurs after the core is mostly uncovered. A large amount of hydrogen is generated (see Figure 25), releasing a substantial amount of heat 1 to 2.5 hours after reactor scram (Figure 27). The relocating core melt dries out the lower head approximately 4 hours after scram (Figure 23).

The progression of the UO_2 case with the hypothetical accident tolerant cladding material (denoted "UO_2-AT clad") and the UO_2-Zr alloy case with oxidation (computationally) prohibited closely align, as shown in Figure 23. Note that TAF and BAF refer to top and bottom of the active fuel height, respectively. After 10 hours, the UO_2-AT clad case generates approximately half of the hydrogen as the reference UO_2-Zr case (Figure 25). The relocating core melt dries out the lower head approximately 8.5 hours after scram (Note: There are additional uncertainties involved with these proposed candidate core materials since core degradation mechanisms [material interactions, eutectic formation, failure temperatures, etc.] have not been experimentally determined).

The case using the accident tolerant fuel <u>and</u> clad is designated "ATF-A" in the ensuing discussion. Results for ATF-A closely align with the UO_2-AT case; however, the core stays mostly intact and relocation to the lower plenum is reduced. This behavior is due to the high melting temperature specified for the considered accident tolerant fuel material. The relocation characteristics and modeling of the candidate accident tolerant fuel material (ATF-A) that were applied in this preliminary analysis should be further investigated.

The steam dryer temperature and main steam line temperature, before the main steam isolation valve, are given in Figures 26 and 28. While the water level drops, the fuel superheats the gas flow passing through the core and into the steam separators, steam dryers, main steam line and finally out the SRVs. In the reference UO_2-Zr system, the rapid release of energy from cladding oxidation further accelerates the heat-up of the upper structures. However, the UO_2 fuel and Zr alloy cladding begins to relocate into the bottom head sending additional steam upwards. The additional steam flow limits the temperature rise in the upper structures. Soon afterwards, the bottom head fails and the melt relocates ex-vessel. However, for the other modeled cases for which there is low/no cladding oxidation and the core remains intact for a longer period, the upper structures continue to heat up. A limitation in MELCOR and the current Peach Bottom model is the inability to melt and relocate upper structures and core shroud. Thus, even though temperatures are predicted well above the structural failure limits and are in the range of the material melting point (1675-1725 K), the structures are modeled as remaining intact.

Table 8 summarizes the differences in the predicted accident progression timing between the reference UO_2-Zr case and the UO_2-AT clad case. The modeled advanced cladding case (UO_2-AT) indicates slower accident progression, providing a few extra hours for accident response measures.

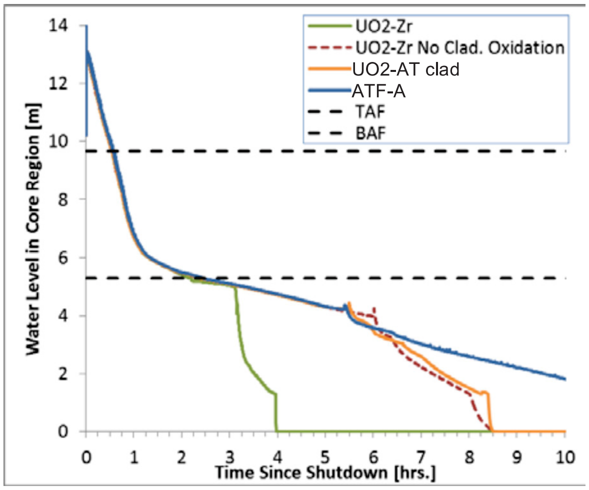

Figure 23. Core water level, STSBO.

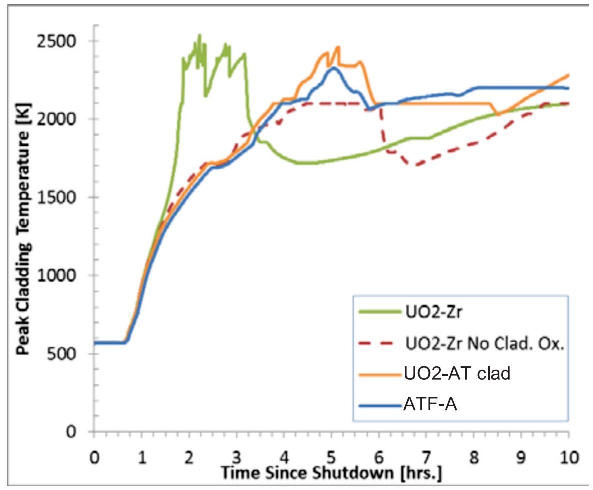

Figure 24. Peak intact cladding temperature, STSBO.

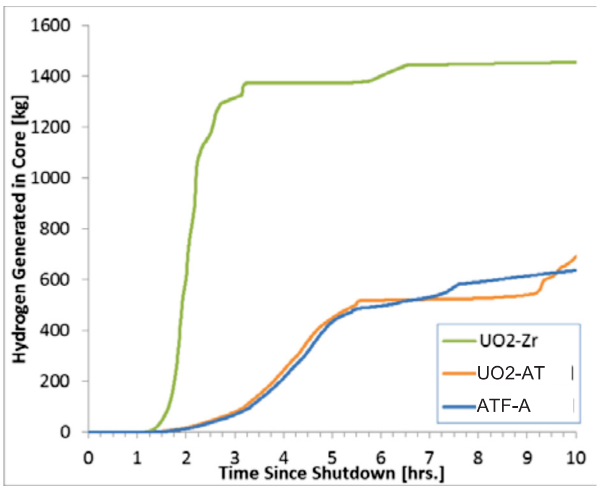

Figure 25. In-core hydrogen generation, STSBO.

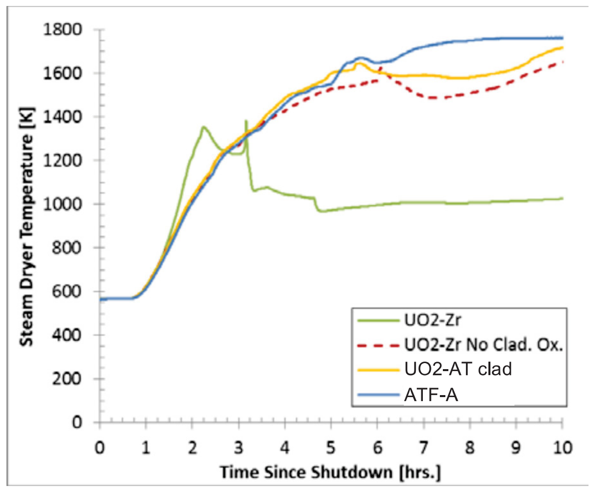

Figure 26. Steam dryer temperature, STSBO.

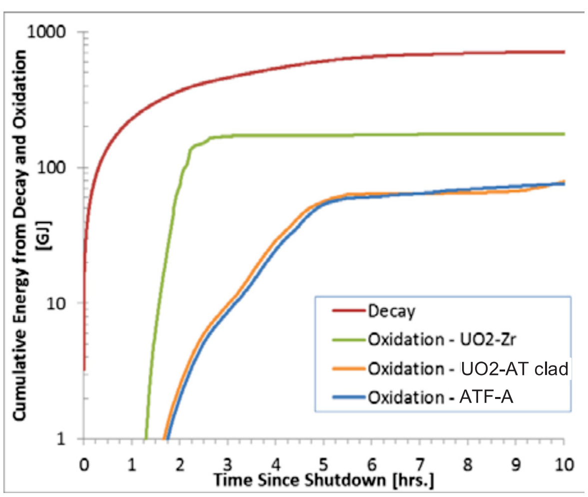

Figure 27. Energy source terms, STSBO.

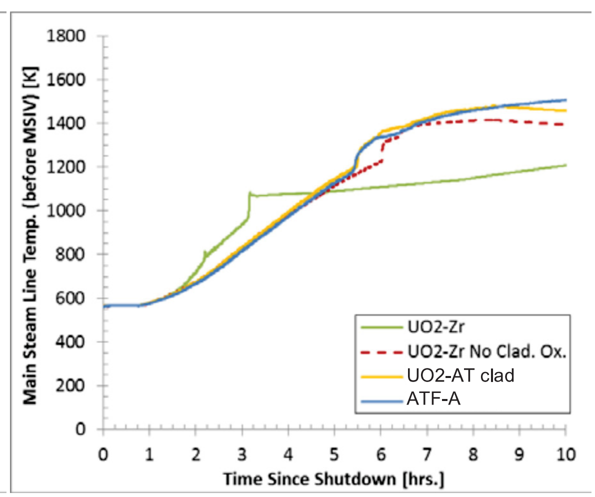

Figure 28. Main steam line temperature, STSBO.

Table 8. STSBO Event Timing Comparison.

Event	UO_2-Zr Case Time [hrs.]	UO_2-AT Clad Case Time [hrs.]	Difference ΔTime [hrs.]
First cladding gap release	1.23	1.26	0.03
Time to 50 kg H_2 generated	1.52	2.59	1.07
First channel box failure	1.87	4.22	2.35
First cladding relocation	2.07	4.93	2.86
Lower head dries out	3.98	8.52	4.54
Lower head failure	4.64	12.14	7.50

6.4.3 LTSBO

The LTSBO scenario results are generally consistent with the STBSO scenario results, except the time scales are extended (Figure 29-32). The first 12 hours of the accident (scram, RCIC operation, boildown) are similar between the cases.

Around 12 hours, cladding oxidation in the reference UO_2-Zr case accelerates, generating hydrogen and causing the excursion in cladding temperature, shown in Figure 30. The first cladding failure and relocation occurs at 14.1 hrs. In contrast, the cladding temperature for the cases with advanced materials increases at a slower rate due to the reduced cladding oxidation (or lack of cladding oxidation for the UO_2-Zr case in which Zr oxidation was prohibited [UO2-Zr No Clad Ox.]). By reducing the cladding oxidation, the time to the first cladding relocation is delayed by 6.1 hours, occurring at a predicted 20.3 hours after shutdown for the UO_2-AT clad case.

The UO_2-AT clad and ATF-A cases became unstable during core relocation and the simulations were terminated in MELCOR. However, the UO_2-Zr case with prohibited cladding oxidation was extended through the 24-hour simulation duration. The UO_2-AT clad case is expected to closely follow the UO_2-Zr case without cladding oxidation (UO2-Zr No Clad Ox.).

The steam dryer temperature response, shown in Figure 32, is similar to that predicted in the STSBO scenario. The reference UO_2-Zr case results in a rapid heat-up of the steam dryer during cladding oxidation; however, the temperature is reduced as the melt relocates into the lower head and then ex-vessel. In contrast, the UO_2-AT and ATF-A cases employing advanced materials have a more gradual and long-term heat-up of the steam dryers reaching very high temperatures. The integrity of the upper internals at such high temperatures over such long durations is questionable.

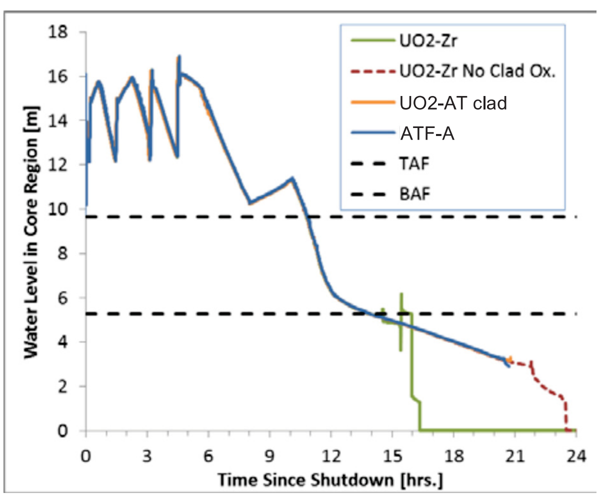

Figure 29. Core water level, LTSBO.

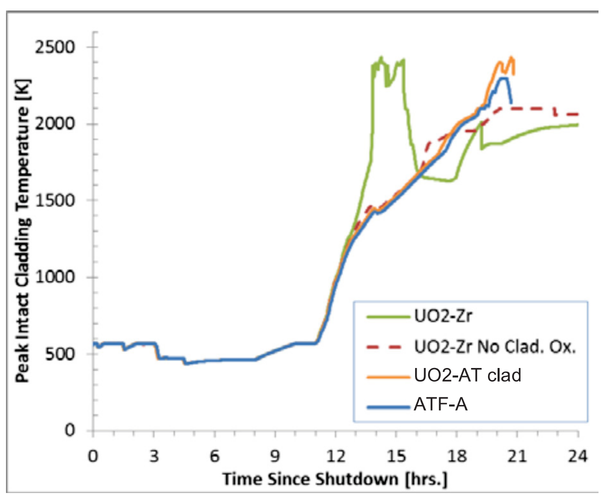

Figure 30. Peak intact cladding temperature, LTSBO.

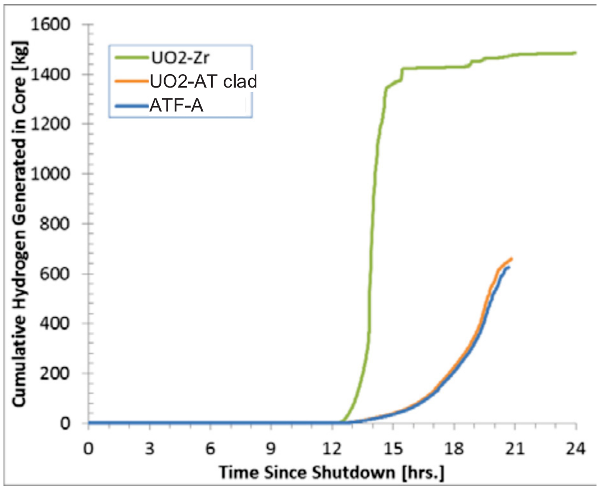

Figure 31. In-core hydrogen generation, LTSBO.

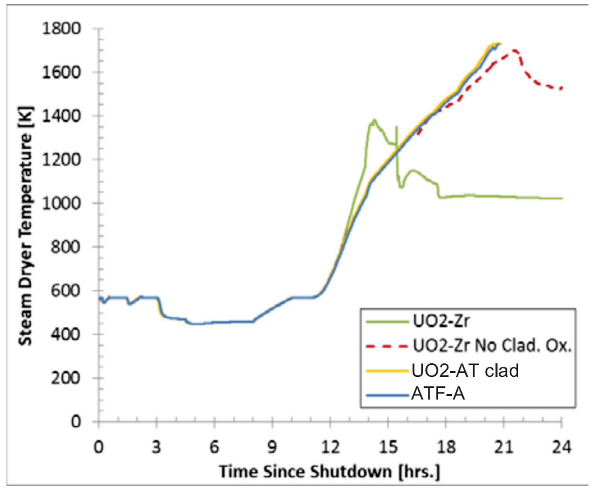

Figure 32. Steam dryer temperature, LTSBO.

Table 9 summarizes the differences in the predicted accident progression timing between the nominal UO_2-Zr case and the UO_2-AT clad case. The advanced cladding slows the accident progression, providing several extra hours for accident response measures.

Table 9. LTSBO Event Timing Comparison.

Event	Reference UO_2-Zr Case Time [hrs.]	UO_2-AT Clad Case Time [hrs.]	Difference ΔTime [hrs.]
First cladding gap release	11.83	12.54	0.71
Time to 50 kg H_2 generated	12.89	15.42	2.53
First channel box failure	13.83	19.38	5.55
First cladding relocation	14.12	20.25	6.13
Lower head dries out	16.38	*23.52	*7.14
Lower head failure	17.55	*24.48	*6.93

6.4.4 Mitigated STSBO (MSTBO) with Water Injection via Control Rod Drive

In the modeled MSTBO scenario the STSBO is successfully mitigated by operators recovering water injection capability through the control rod drive (CRD) pumps. In these simulations the CRD pumps inject water from the condensate storage tanks at the rate of 110 gpm (0.42 m^3 per min), which is the nominal PB CRD pump capacity after scram. The core slowly refills and remains partially exposed for many hours, as indicated by the core water level in Figure 33. Only the reference UO_2-Zr and UO_2-AT clad cases were analyzed.

The reference UO_2-Zr case reaches a predicted peak cladding temperature of 1720 K (Figure 34) while the UO_2-AT clad case reaches a predicted peak cladding temperature of 1455 K. Coincidentally, the UO_2-AT clad case remains below the regulatory peak cladding temperature limit of 1204°C (2200°F or 1477.7 K). The reference UO_2-Zr case produces a predicted 583 kg of hydrogen (Figure 35) over the time interval of 1.75 to 8 hours after scram, while the UO_2-AT clad case is predicted to produce only 31 kg. In the reference UO_2-Zr case, the steam dryer reaches temperatures (Figure 36) similar to those predicted in the STSBO and LTSBO scenarios. However, for the UO_2-AT clad case, the predicted steam dryer temperature is much lower than that predicted to occur in the STSBO and LTSBO scenarios.

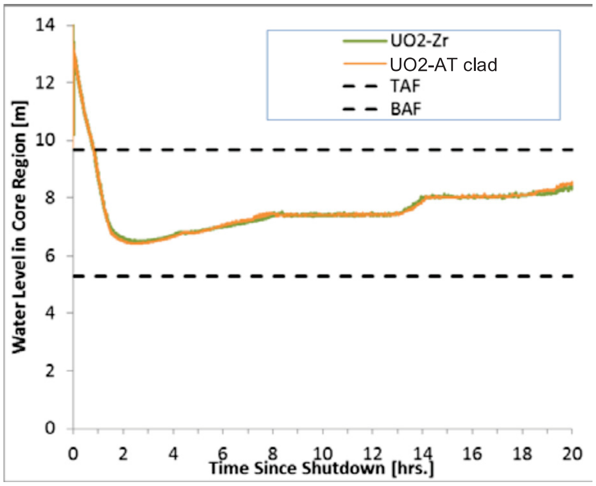

Figure 33. Core water level, MSTSBO.

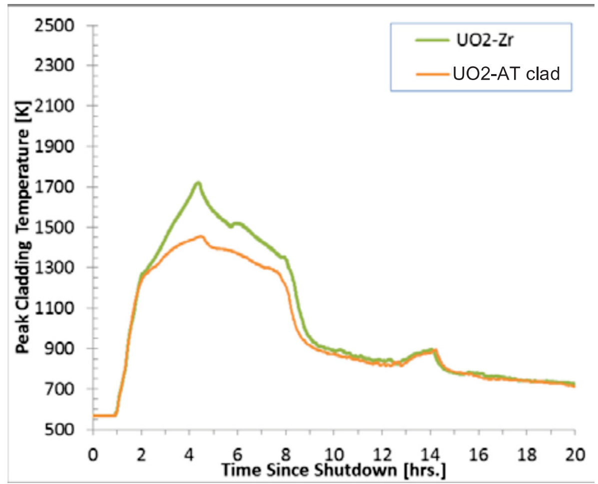

Figure 34. Peak intact cladding temperature, MSTSBO.

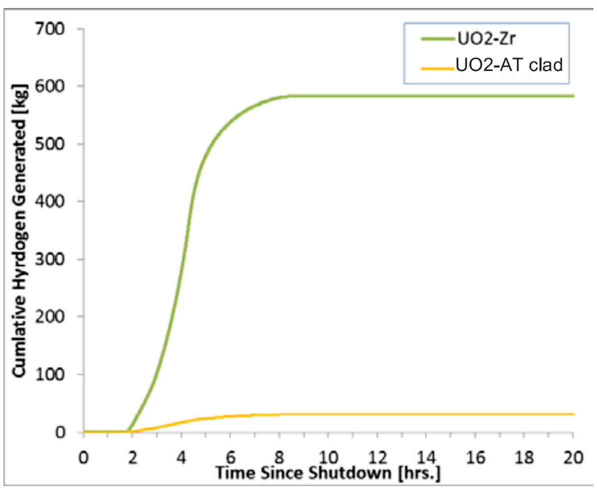

Figure 35. In-core hydrogen generation, MSLTSBO.

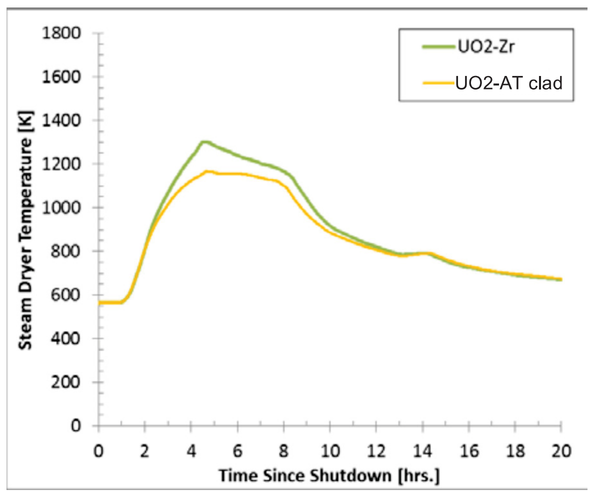

Figure 36. Steam dryer temperature, MSTSBO.

6.4.5 MELCOR Simulation Limitations for the BWR Case

Section 6.4 presents example analyses that could be performed to predict possible accident tolerant fuel and cladding performance for BWR accident cases. A few modeling limitations were encountered in the BWR MELCOR model. First, for a BWR simulation, the channel box and fuel cladding are both hard coded to use the same material definition. Thus, the material specification and properties cannot be altered independently for these components without significant modification to the code. Second, the heat of reaction (that of Zircaloy) during oxidation is hard coded in MELCOR. Another code limitation is that fuel oxidation is not modeled by MELCOR. While this is of limited interest for UO_2, future fuel materials may be more susceptible to oxidation. These limitations restrict a user's ability to evaluate advanced fuel and cladding materials that are outside the scope of the traditional UO_2, Zirconium alloy, and steel system. Additional modifications to the structure of the MELCOR code are possible and could be available for future concept evaluations, but additional code development activity would be required.

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

This study focused on a potential analysis approach that could be applied to assess the in-vessel impact of candidate advanced fuels and cladding on BWR accident progression with respect to accident progression timing, hydrogen production and component temperatures. Future analyses may also encompass the impact of advanced fuels and claddings on the ex-vessel phase of an accident (Farmer et al. 2014). Note that the potential additional material that could derive from melting and relocation of the upper structures may also impact the ex-vessel progression. Finally, the fission product release characteristics for candidate advanced fuels will need to be developed and integrated into system analysis tools to analyze their risk reduction potential.

6.5 Conclusions from Preliminary Beyond Design Basis Analysis Results

This section presented several analysis results for hypothetical accident tolerant fuel and cladding materials. These types of analyses will be useful in evaluating various candidate materials relative to the reference UO_2-Zr case, from a full-core and system perspective, to predict their potential performance and safety benefits.

Preliminary, example analysis results indicate that higher melting point / lower hydrogen producing core components *will not* preclude a severe accident. There are no “silver bullets” in fuel and cladding design. If core cooling cannot be restored, the severe accident will continue. The time for boil-off and uncovering of the core is primarily dependent upon the fuel decay heat rather than the latent heat of the fuel at the start of the accident. Assuming that the decay heat from candidate fuel materials is comparable to the traditional UO_2-Zr system, changing the cladding and fuel will have a minor impact on the early stages of a BDBA. Potential ATF benefits begin to be realized when the system reaches a point at which the cladding begins to oxidize. The use of ATF concepts can potential provide increased margin (time) for accident response and mitigation measures. This additional time available to restore core cooling and to mitigate the accident is on the order of an hour to a few hours for the hypothetical example cases presented. The rate of hydrogen generation, which can exacerbate an accident, is also potentially slowed by adoption of candidate ATF materials. Both the PWR and BWR analyses highlighted some of the limitations with the selected MELCOR code with respect to modeling ATFs; these limitations should to be addressed for subsequent analyses that employ MELCOR for the evaluation of ATF performance.

A significant number of experiments (materials-interaction and degraded-core) conducted over the previous thirty years have yielded the current state of knowledge on the progression of BDB accidents in LWRs with UO_2-Zircaloy cores. In general, the initiation of core degradation in current cores starts with failure (via eutectic interactions) of the control elements, which then propagates into the fuel assemblies and accelerates the fuel/cladding failure. It is imperative with the study or introduction of ATFs into nuclear cores that the influence of all the core components be considered on the core degradation process (e.g. the stainless steel control blade with B_4C absorber in a BWR and other RPV stainless steel components).

As noted, in the event that core cooling cannot be restored, the severe accident will continue. Under these late phase situations that can include vessel failure, combustible gas production from currently proposed advanced cladding types is expected to be similar to that produced by Zr-alloy clad fuel (Farmer et al. 2014).

7. Discussion and Path Forward

This report describes a proposed technical evaluation methodology that can be applied to assess the anticipated performance and safety of proposed accident tolerant fuel concepts relative to the current UO_2 – zirconium alloy system. Rather than focus on individual properties, the approach considers the confluence of properties that results in a particular behavior during all phases of possible operation and also considers challenges associated with fabrication of each concept. Evaluation tables completed for each concept will provide a clearer overview of each new concept relative to one another, highlighting expected benefits and vulnerabilities – which can be translated to the risk / benefit ratio for each concept and can be linked to the near term versus far term nature of the concept development. The intended goal of this exercise is to inform concept down-selection, such that the most promising accident tolerant fuel design option(s) can continue to be developed toward qualification.

Preliminary analyses have been conducted on a handful of ATF concepts, and sensitivity analyses have been performed for key performance parameters, including fuel thermal conductivity, cladding oxidation rate and cladding heat of oxidation. Sensitivity analyses provide early insight to the key parameters to be used for cladding and fuel optimization, while more detailed analyses allow evaluation of currently proposed concepts based on the available property and performance data.

There are no “silver bullets”; if core cooling cannot be restored during the course of a severe accident, the severe accident will continue. Multiple ATF concepts are currently being investigated by industry, academia and the national laboratories based on their expected potential to increase the time before which damage in the core becomes irreversible in an accident sequence. These materials are selected based on demonstrated lower oxidation kinetics rates, lower heat of oxidation, and reduced hydrogen generation (cladding), or due to the potential for significantly higher thermal conductivity (fuel), which would reduce fuel centerline temperature.

The proposed technical evaluation methodology will be applied to evaluate the ability of each of these concepts to meet performance and safety goals relative to the current UO_2 – zirconium alloy system and relative to one another. This ranked evaluation will be used to enable the continued development of the most promising ATF design options given budget and time constraints with a goal of inserting one (or possibly two) concepts as an LTR or LTA in a commercial LWR by 2022.

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Appendix A
Baseline UO_2-Zirconium Alloy Fuel System

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Design of an advanced fuel system demonstrating enhanced performance and safety relative to the current fuel system first requires understanding of the current state-of-the-art fuel system performance under the various system operating regimes. A light water reactor (LWR) core is comprised of hundreds of fuel assemblies consisting of a two-dimensional array of rods roughly 4-m in length. In the vast majority of commercial reactors, the fuel rods contain urania fuel pellets inside a zirconium alloy tube. The performance of the fuel system and current life limiting challenges and abilities are summarized in this section. Section A.2 outlines the performance requirements and phenomena relevant to normal operation, while section A.3 provides information on the zirconium-urania system under transient conditions such as RIAs and LOCAs.

A.1 UO_2-Zirconium Baseline Properties

This section discusses the key baseline unirradiated physical properties that characterize the performance of uranium dioxide and zirconium alloys used in the current fleet of reactors. The collection of baseline property data will aid in the comparison to potential new fuel and cladding materials. The physical properties of the fuel system are critical to understanding how the fuel will perform under reactor conditions. Many properties, such as melting temperature of the fuel, directly impact the operating limits of the reactor.

Table A-1 summarizes key thermal and mechanical properties for the uranium fuel and the currently deployed zirconium alloys in the unirrradiated state. Due to the propriety nature of M5™ and ZIRLO™ alloys the public data are limited; however, the primary improvements over Zircaloy-4 are the decreased coolant-side corrosion and reduced hydrogen pickup. Most of these properties are significantly impacted by irradiation. The effects of irradiation on some of these properties are discussed in section A.2. Many of the properties, particularly mechanical properties of zirconium, are dependent on fabrication process and heat treatment, which are not discussed in this document. An overview of physical properties of the fuel and Zircaloy alloys used in MATPRO and published by NRC can be found in NUREG/CR-6150 (Hagman 1993).

Table A-1. Baseline unirradiated properties of uranium oxide and zirconium alloys.

	UO_2 Surface Temperature (600K)	**UO_2 Centerline (1000K)**	**Zr-2 (600K)**	**Zr-4 (600K)**	**ZIRLO™**	**M5™**
Thermal	(Carbajo et al. 2001)		(Hagman 1993; Adamson 2011)		*Limited public data available.*	
Thermal Conductivity (W/m*K)	6.70	4.01	12.65	12.65		
Heat Capacity (J/kg*K)	281	311	290	290		
CTE (1/K)	9.91E^{-06}	1.05^{e-05}	1.49E^{-3} (radial) 3.78E^{-3} (axial)	1.49E^{-3} (radial) 3.78E^{-3} (axial)		
Density (kg/m^3)	10864	10733	65510	65510		
Melting Temp (K)	3120	3120	1858	1858		
Mechanical	(Hagman 1993)		(ASTM 2013; Northwood, London and Bähen1975; Bai et al. 1994; Forurgeaud et al. 2009)		(Pan, Garde and Atwood 2013; Charit and Murty 2008)	(Forurgeaud et al. 2009)
Yield Strength (MPa)	N/A	N/A	120-160	544	436 (@ 658K)	
Young's Modulus (GPa)	218	208	79	79	51 (@ 600 K)	51 @ 600K
Ultimate Tensile Strength (MPa)	N/A	N/A	200-235	745	508 (@ 658K)	

A.2 Normal Operation

Normal operations and anticipated operational occurrences as defined by the NRC were discussed in section 2.3. The normal operations along with AOO conditions are the primary basis of experimental operating experience for nuclear reactors. The majority of operational data and history support this regime, and the phenomena limiting these operations directly affect the economic viability of plants.

A.2.1 Nuclear Regulatory Licensing Requirements

The primary goals of the NRC requirements for normal operation and AOO regimes is to ensure safe operation of nuclear plants with no fuel failures from known causes during normal operation and to minimize fuel damage to limit severity of fuel failures during accident conditions (U.S. NRC 2007). Fuel failure is defined as loss of cladding hermeticity (gas tightness). NUREG-800 provides details on the structural and thermal hydraulic analyses that are required to ensure safe operation of plants during normal operation and to minimize the damage to fuel rods. The mechanical analysis must include damage from all possible irradiation and burn-up effects. The analysis should state a maximum fretting wear, which is then used in the stress and safety analysis. The cumulative number of strain fatigue cycles on the structural members should be significantly less than the design limit, which includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. NUREG-800 also sets out specific requirements for fuel damage and performance levels. No cladding liftoff can occur during normal operation, and the ability to insert control rods must be maintained. During normal operation and AOOs the fuel must be designed such that it will not fail by a specific mechanism. A limited number of failed rods is allowed due to the unavoidable nature of operation. In order to minimize the risk of fuel failure the cladding must not experience more than 1% uniform strain.

A.2.2 Corrosion

Zirconium alloys in general are highly resistant to corrosion; however, they are not immune to oxidation in the aggressive conditions that exist inside a commercial nuclear reactor (Franklin and Lang 1991). The corrosion issues for zirconium alloys in BWRs and PWRs are unique due to the differences in operating conditions and alloys used. BWRs utilize Zircaloy-2 (Zr-2), while PWRs previously used Zircaloy-4 (Zr-4) and are have now transitioned to Zr-Nb cladding (M5™ and Zirlo™). Other major differences between the reactor types that affect corrosion are: coolant boiling in BWRs; relatively high concentration of hydrogen in PWR coolant; relatively high concentration of oxygen in BWR coolant; and high operating temperature in PWRs (Adamson 2011).

Corrosion in zirconium alloys occurs via three modes: uniform, nodular, and shadow. Both BWRs and PWRs experience uniform corrosion, while shadow and nodular corrosion are observed only in BWRs [12]. Uniform oxidation follows a typical power law up to a thickness of 1.5-2 mm (typically achieved at ~30 GWd/tHM), at which point it transitions to linear growth rate (Lemaignan and Motta 1994; Dickson et al. 1979). M5™ and ZIRLO™ have improved corrosion behavior relative to Zr-4, remaining below ~30 GWd/tHM, and the transition to faster linear corrosion is delayed in M5™ and Zirlo™ resulting in improved corrosion performance (Bossis et al. 2006).

Corrosion is currently a design limiting issue in LWRs, for which the mechanisms are extremely complex and poorly understood. Despite a large database of irradiation performance on zirconium-based alloys, the mechanism and a mechanistic understanding of the corrosion behavior of the alloys have evaded scientists. Design correlations and limits are almost exclusively empirically based on irradiation data (Lemaignan and Motta 1994).

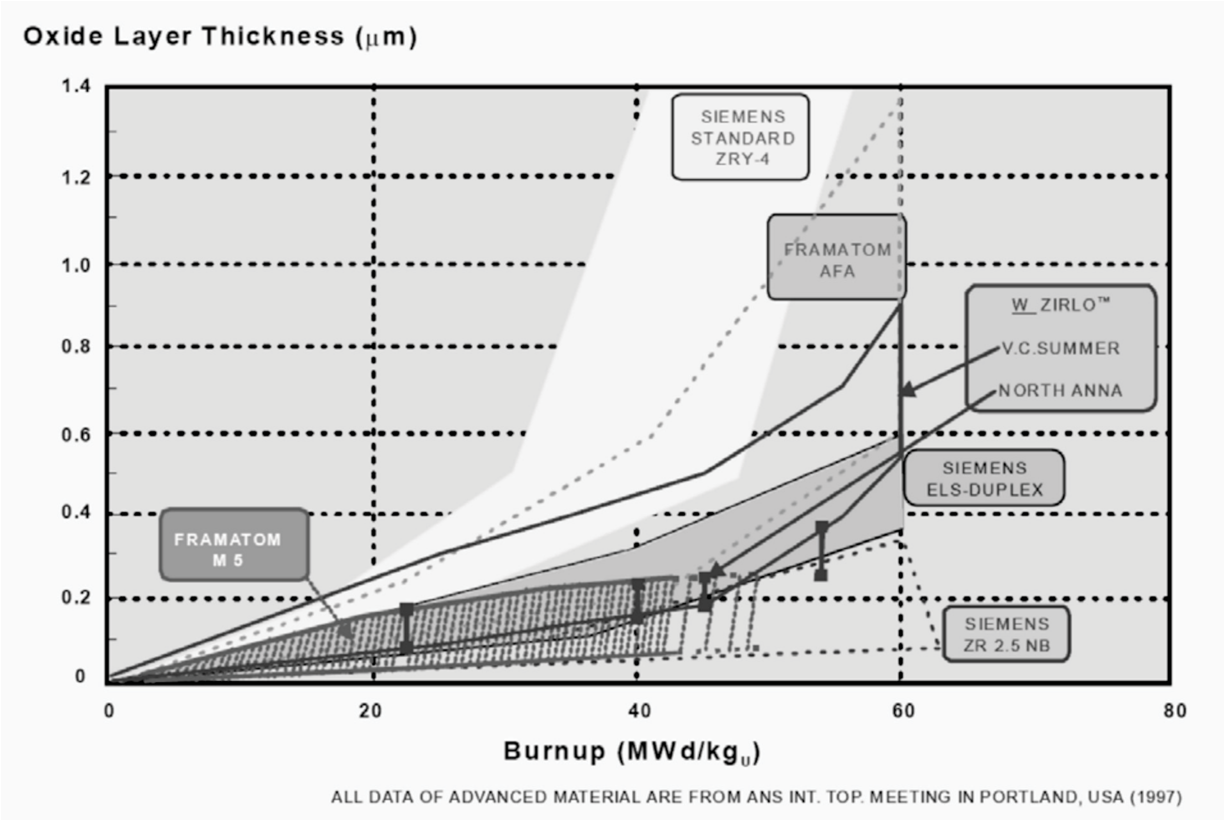

Figure A-1. Peak oxide layer thickness for a variety of zirconium alloys, showing the improved performance of Zr-Nb alloys (Weidinger 2001).

A.2.3 Hydrogen Pick-up

Hydrogen pick-up is the absorption into Zircaloy of hydrogen generated during the corrosion process. The oxidation of zirconium by water generates free hydrogen ions that can permeate into the zirconium metal. The solubility of hydrogen is extremely low at operating temperatures (80-100 ppm); as a result, hydrogen precipitates out as hydrides (Adamson 2011). These hydrides are deleterious to the corrosion properties, dimensional stability, and mechanical properties of the zirconium alloys. The hydrides precipitate and then migrate to areas of high stress, which can result in delayed hydride cracking. The presence of hydrides also results in increased uniform corrosion rate, although the mechanism for this increase is not understood. Additionally, due to the low density of the hydrides, the hydrogen pickup causes swelling in the zirconium alloys. Another concern with the presence of hydrides is their effect on long term stability of the cladding during long term dry storage (Adamson 2011; Lemaignan and Motta 1994; Rudling 2007).

Zr-2 used in BWRs at burnups above ~50 GWd/tHM experiences accelerated hydrogen pick-up (Figure A-2), which currently limits increasing burnup in BWRs (Lemaignan and Motta 1994; Bossis et al. 2006; Rudling 2007). Large scatter in the hydrogen content, even within the same fuel rods, is observed at high burnups. Zirconium alloys used in PWRs have a higher hydrogen content on average due to the higher operating temperatures in PWRs versus BWRs. The hydrogen content for M5 has been reported to go up to 60-80 ppm at high burn-ups (Geelhood 2009). The exact values for M5™ and Zirlo™ are not publicly available; however, reports to NRC indicate they are lower than Zr-4.

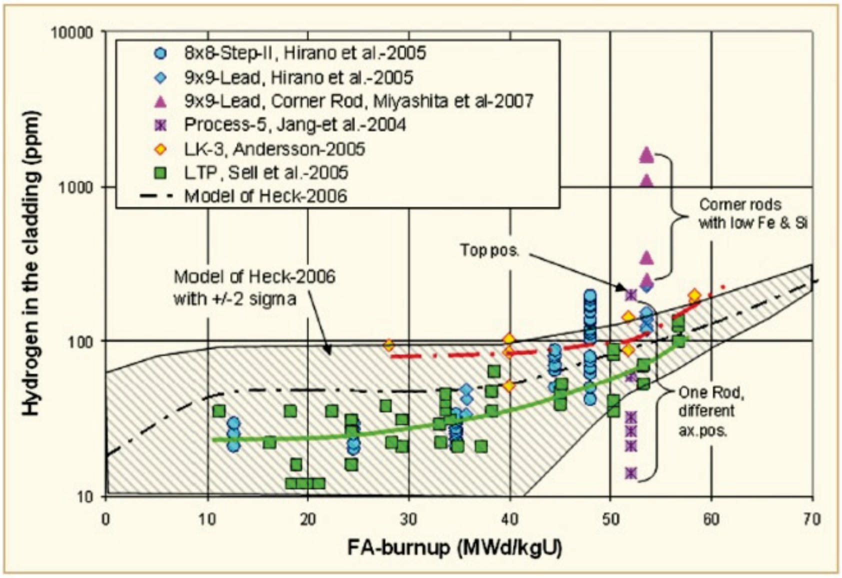

Figure A-2. Hydrogen content in advanced (intermediate sized second phase precipitates) Zr-2 claddings (Adamson 2011[a]).

A.2.4 Dimensional Stability of Cladding

Dimensional stability is critical for components within reactors, which are designed to tight tolerances. Deformation can lead to fuel channel bowing and fuel assembly bowing which reduce thermal margin in plant design. Dimensional instability in zirconium alloys is due to hydride volume changes, irradiation growth due to the hexagonal close-packed (hcp) structure, and irradiation creep (thermal creep is insignificant at operating temperatures) (Lemaignan and Motta 1994).

Hydride formation causes growth within the zirconium alloys due to its lower density (~16% less dense than Zr). At 1000 ppm hydrogen ~0.35% growth is seen, which correlates to a 0.5-in growth in the fuel column length (Adamson 2011).

Irradiation growth in zirconium is anisotropic due to the hcp structure and is strongly dependent on texture and fluence. The strong correlation with fluence and the non-uniform flux profile in reactors results in non-uniform growth within long core components such as fuel rods (Lemaignan and Motta 1994; Franklin 1982). Both Zr-2 and Zr-4 alloys experience breakaway irradiation growth above ~10-15 dpa, while Zirlo™ and M5™ show no breakaway irradiation growth rates out to 20-25 dpa (the limit of data collected) (Adamson 2011).

Irradiation creep is critical to the interaction of the cladding with the fuel pellets. Initially a gap exists between the fuel pellet and cladding. The cladding then creeps down to close this gap. Understanding the high burnup creep properties of zirconium alloys is critical to knowing when the gap reopens and how large the gap will become (Rudling 2007, 2011). Creep is also the limiting property for many accident scenarios, such as LOCA and RIA, due to the high temperatures encountered during these accidents. Zirlo™ and M5™ have improved irradiation and thermal creep properties, allowing a larger safety

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[a] ANT Reports are conglomerations of international work; full references for included data can be found in the reports.

margin during accident scenarios than earlier zirconium alloys (Lemaignan and Motta 1994; Rudling 2007, 2011; Matsuo 1989).

Due to the complex interaction between irradiation growth and irradiation creep in zirconium alloys the mechanisms are poorly understood, although several mechanisms have been suggested. Empirical correlations are currently used for design basis analysis, which limits the extension of their use to design space outside of currently operating reactors with UO_2-Zr alloy fuel (i.e. accident tolerant fuels comprised of different materials) (Adamson 2011; Lemaignan and Motta 1994; Matsuo 1989).

A.2.5 Fuel Swelling

Fuel swelling is important in calculating cladding strain, fuel cladding liftoff, initial gap size, and PCMI. Fuel swelling is the result of both solid and gaseous fission product generation during operation. Swelling due to solid fission products shows a wide range of values due to the fission rate and temperature profile across the fuel pellet; however, it is assumed to be independent of temperature, stoichiometry and stress (Lambert and Strain 1994). The swelling rates range from ~0.32% per 10 GWd/MTU (historical value) to ~0.61% per 10 GWd/tHM (MATPRO/FRAPCON value) (Lambert and Strain 1994; Geelhood et al. 2011). For calculation purposes the most conservative value is used; therefore, the higher rate is used for cladding strain evaluations while the lower rate is used for critical internal gas pressure for cladding liftoff (Patterson 2012). Swelling due to gaseous fission products varies strongly with temperature, fission rate, and stress within the fuel. The gaseous swelling rate also changes as function of burnup, increasing from 0.36% per 10 GWd/MTU at lower burnup to 0.7% per 10 GWd/MTU at high burnup (80 GWd/MTU) (Spino et al. 2005). Due to the critical nature of fuel swelling in fuel performance, the swelling behavior over the life of the fuel and under a large variety of conditions is required to adequately inform fuel design. A fuel having minimal swelling is desirable to maintain dimensional stability. A more accurate swelling model for the current fuel system could result in reduced conservatism.

A.2.6 Fission Gas Release

Fission gas release from the oxide fuel during normal operation is critical to performance in both normal and accident scenarios. Approximately 25% of all fission products are gaseous species (Xe, Kr), and the inventory of the fission gas in the fuel rod increases with increasing burnup. The majority of fission gas atoms are initially stored in the fuel lattice. These fission gases then diffuse to small bubbles, which then coalesce on grain boundaries (Geelhood 2005). The primary route for fission gas release during normal operating conditions is diffusion along the grain boundaries to the exterior of the pellets (Lambert and Strain 1994; Lambert et al. 2007). The fission gas release fraction is highly temperature and burnup dependent, as shown in Figure A-3. Low fission gas release is desirable to prevent over pressurization of the cladding, which can result in cladding liftoff at high burnups. Liftoff is disallowed by NRC during normal operations (U.S. NRC 2007; Lambert and Strain 1994; Patterson 2012). The release of fission gas to the plenum during normal operation is also deleterious in the case of fuel failure, since the radioactive fission gas can be released into the coolant (Lambert and Strain 1994).

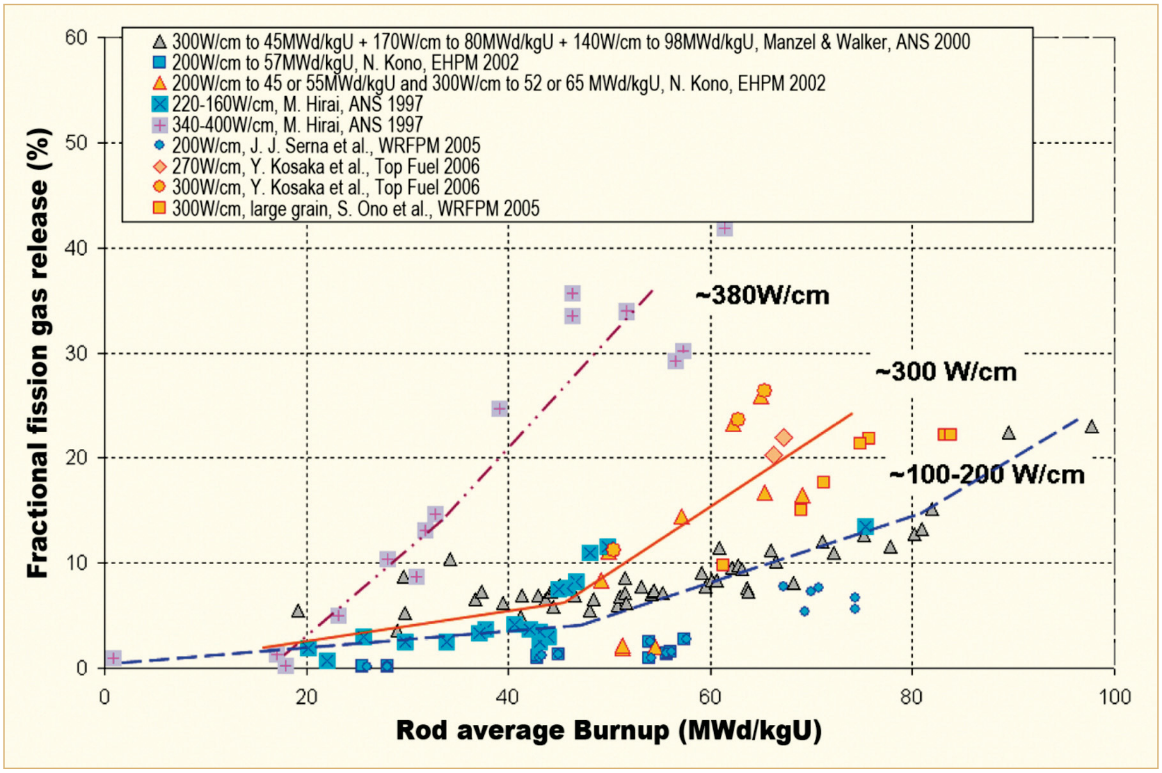

Figure A-3. Fission gas release as a function of temperature (linear heat rate) and rod average burnup (Patterson 2012[b]).

A.2.7 Fuel Thermal Conductivity

Fuel thermal conductivity is critical to the removal of heat from a system and is a potentially limiting factor in reactor design. The heat in the reactor is generated throughout the fuel. Hence, the fuel thermal conductivity limits the rate the rate at which heat is tranferred out to the coolant. The relatively low thermal conductivity of oxide fuels is one of its major drawbacks. The thermal conductivity of oxide fuels decreases rapidly at the begining of life due to radiation damage and fission products acting to produce scattering. The effect of irradiation temperature and burnup on the thermal conductiivty of LWR fuel was most extensively reviewed by Ronchi et al. (2004) and is summarized in Figure A-4.

[b] ANT Reports are conglomerations of international work; full references for included data can be found in the reports.

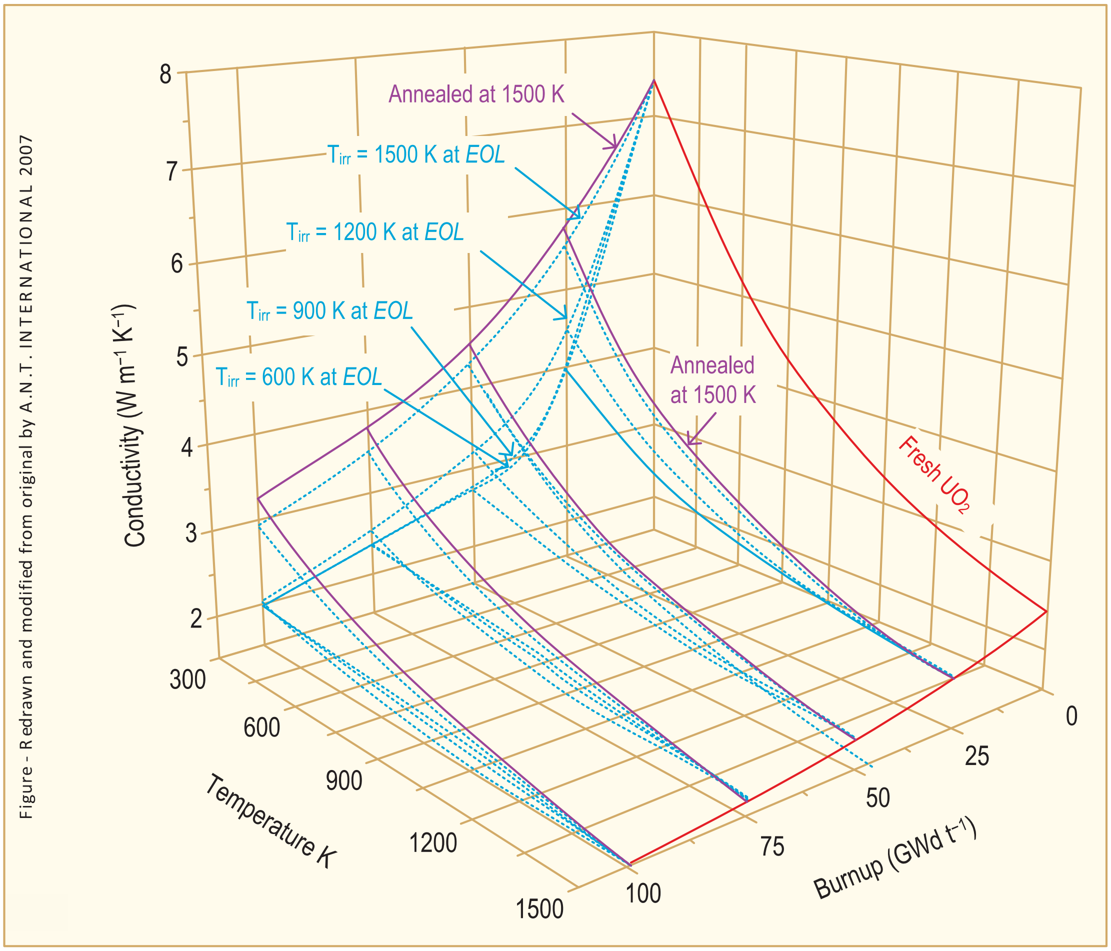

Figure A-4. Experimental data showing the decrease in thermal conductivity as a function of burn-up and irradiation temperature for oxide fuel (Ronchi et al. 2004).

A.3 Performance under Accident Conditions

The primary goal of the accident tolerant fuel program is to develop and deploy advanced fuel systems that are more resistant to accident situations than the current LWR fuel. Understanding how the current UO_2-zirconium alloy system performs under extreme accident scenarios that have occurred, and testing to ensure survival in postulated scenarios, is critical to developing fuels having improved performance. Section 2.3 describe the NRC-defined accident scenarios in detail.

A.3.1 NRC Requirements

The primary NRC requirements for performance under accident conditions are aimed at reducing dose to the public from the damaged plant (U.S. NRC 2007). In order to meet the dose limitations set out in 10 CFR 100, certain criteria and restrictions have been put in place by the NRC. The ultimate goal of the requirements is to ensure that coolable geometry is maintained during and following an accident. The regulations currently in place were developed for the UO_2-zirconium alloy fuel system and would need to be modified for new fuel types. Coolable geometry is defined in NUREG-800 as the retention of the rod-bundle geometry with adequate coolant channels to remove residual heat (U.S. NRC 2007). The ability to cool the reactor after an accident can be negatively impacted by cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. To limit cladding embrittlement, the peak cladding temperature during either a RIA or LOCA cannot exceed 1204^oC (2200^oF) and the peak oxidation must remain below 17% of the thickness of the cladding prior to oxidation. The coolability of the fuel during an accident is further restricted by the regulations in that the average fuel enthalpy must remain below 230 cal/g (960 J/g) and the peak fuel temperature must remain below the fuel melting temperature. Additionally, no loss of coolable geometry can be caused by fuel pellet and cladding fragmentation and dispersal or fuel rod ballooning. NRC 10 CFR 50.46b also restricts the total amount of hydrogen that can be generated during a LOCA from reaction with the zirconium cladding to less than 1% of the amount that would be generated if all of the cladding material were to react.

For purposes of calculating the potential dose to the public the NRC requires that, for PWRs, clad failure should be assumed when the calculated heat flux equals or exceeds the DNBR for zero power, low power and full power RIA events. For BWRs, cladding failure is assumed for rods that experience a maximum radially averaged fuel enthalpy greater than 170 cal/g (711 J/g) for RIA events initiated from zero or low power. For rated power conditions, fuel rods that experience cladding dryout should be assumed to fail (U.S. NRC 2007).

In summary, the NRC Requirements establish the following criteria under design basis accident scenarios:

- Reactor must maintain coolable geometry
- Peak cladding temperature less than 1204^oC
- Enthalpy of the fuel must be <230 cal/g (960 J/g)
- Failure is assumed for PWR cladding with heat flux ≥ DNBR
- Failure is assumed for BWR cladding when enthalpy > 170 cal/g (711 J/g)

A.3.2 Reactivity Insertion Accident

The schematic of fuel performance during a RIA, Figure A-5, describes the various scenarios that can occur within the fuel depending on the fuel history and intensity of the accident.

A.3.2.1 Cladding Failure

Fuel failure during a RIA event is largely controlled by failure of the cladding. At low burnup, oxidized, embrittled cladding can fail due to post-DNB fracture. At higher burnup, PCMI failures can occur, and post-DNB ballooning and creep burst can occur for fuel rods that have an internal overpressure. These failure mechanisms and the change in pellet gap and clad ductility are shown schematically in Figure A-6 (Rudling 2012). The change in the failure mechanisms at increasing burnup results from changes in the mechanical properties of the cladding. The combination of cladding embrittlement due to corrosion and hydrogen pick-up and fuel swelling makes PCMI a dominate failure mechanism for high burnup fuel rods during an RIA (Rudling 2007).

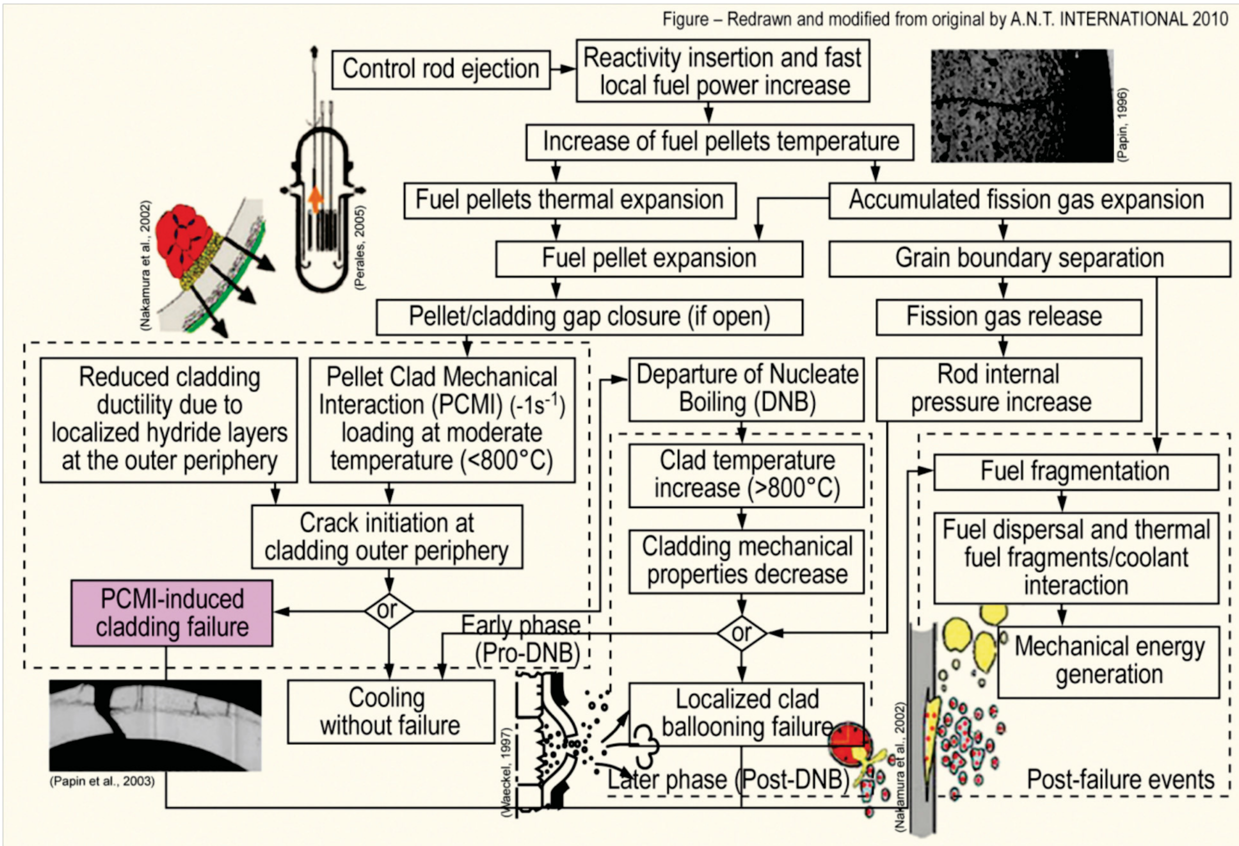

Figure A-5. Schematic showing the possible routes to failure and fuel dispersal during a postualted RIA event (Le Saux 2007).

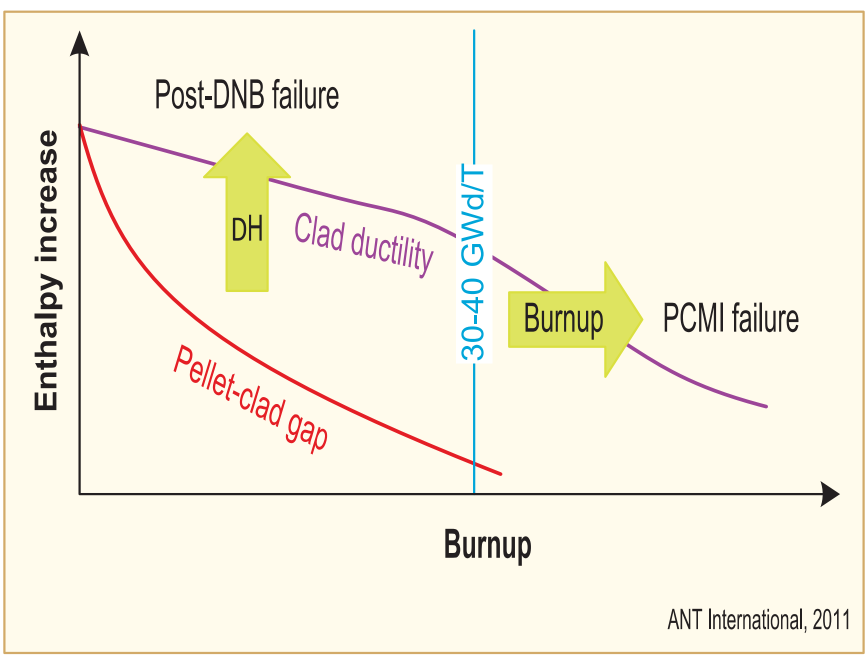

Figure A-6. The effect of enthalpy increase and burn-up on the failure mechanisms active during an RIA event (Sunderland et al. 2004).

A.3.2.2 Fuel Dispersal

Fuel dispersal is undesirable due to the potential for rapid generation of steam interacting with the fuel particles, which can cause damage to nearby fuel assemblies and even reactor pressure vessels if large enough. Fuel rods become more susceptible to fuel dispersal at lower enthalpy spikes with increasing burnup, as seen in Figure A-7. The reason for this increase in potential fuel dispersal is two-fold. First, the higher burnup cladding is more brittle, resulting in larger cracks. Second, the high burnup structure that forms in the fuel turns into small fragments under RIA conditions and is more easily dispersed through cladding cracks (Rudling 2007; Jernkvist and Massih 2010).

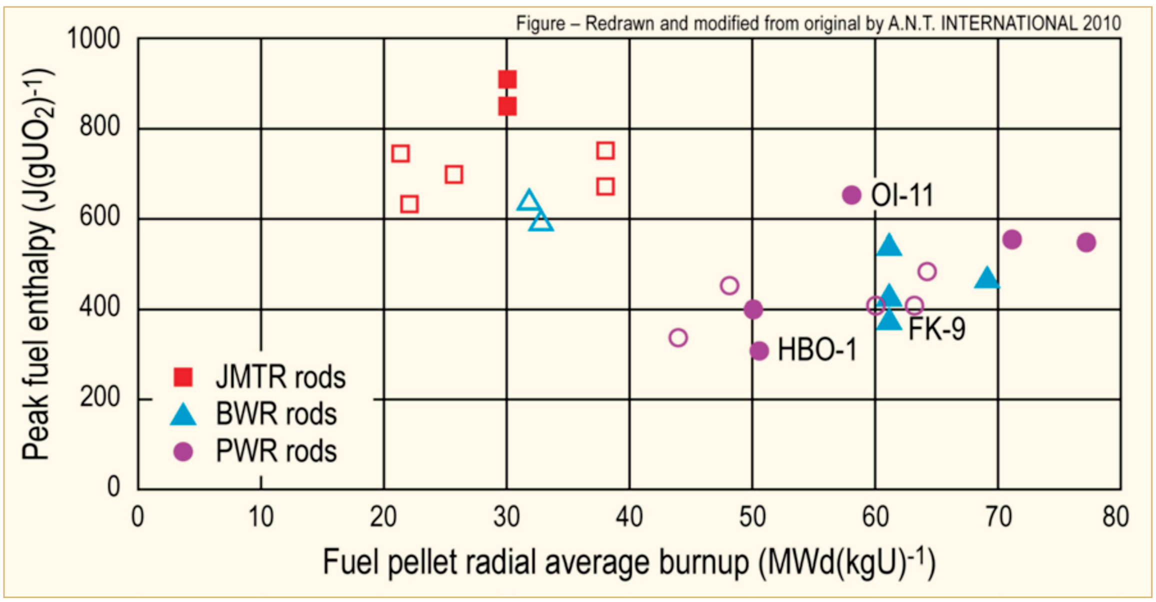

Figure A-7. Closed symbols are fuels which experiences significant (>10% inventory) fuel dispersal, while the open symbols are fuel pins which showed <10% inventory fuel dispersal (Jernkvist and Massih 2010).

A.3.3 LOCA

Postulated LOCA scenarios considered when designing a reactor for NRC licensing were previously discussed in section 2.3. This section provides more detail on the uranium-zirconium system performance and potential issues under such an event. Figure A-8 shows a temperature-time plot of the cladding during a hypothetical LOCA event, along with potential failure points during the LOCA. Note that ECR is extent of cladding reacted; other acronyms have been previously defined.

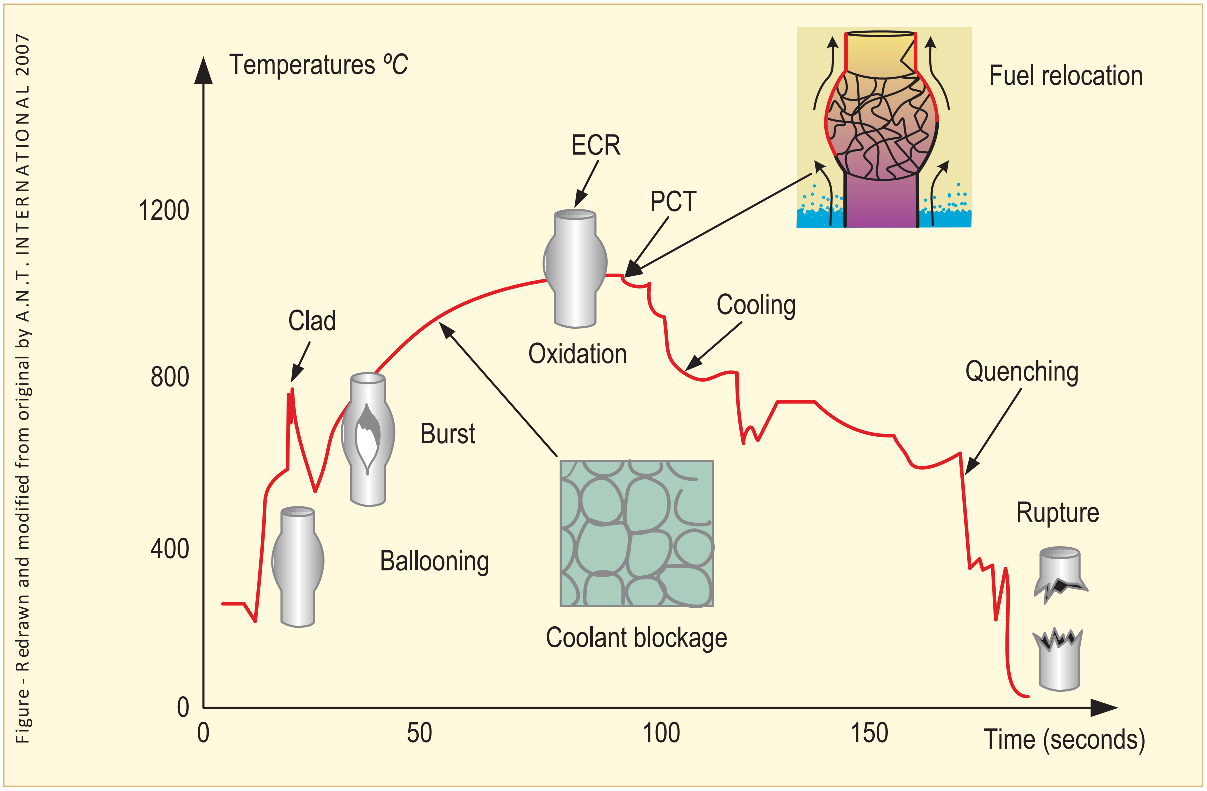

Figure A-8. Schematic of the fuel cladding temperature as a function of time during a hypothetical LOCA event (Grandjean et al. 2004).

A.3.3.1 Ballooning/Creep Rupture

During the initial blowdown phase of a LOCA the rapid decrease in system pressure along with increase in fuel temperature and fission gas pressure can result in ballooning of the cladding material. The ballooning requires the cladding to have retained sufficient ductility during normal operations, and is also related to the amount of fission gas that has been released. The extent of ballooning and potential for bursting are highly dependent on the cladding stress, temperature, and creep strength. The creep strength is highly dependent on oxidation and hydrogen pick-up. The concern associated with cladding ballooning without rupture is the potential for coolant channel blockage. If the stress is too large, bursting may occur. A cladding burst would allow steam oxidation of the interior of the cladding surface, which can significantly reduce the ductility of the remaining cladding and affects the extent of cladding reacted (ECR) later in the LOCA. Bursting also results in the release of noble gases, iodine, and cesium; however, due to the small size of the rupture, very little fuel dispersal is observed (Pettersson 2009).

A.3.3.2 Cladding Embrittlement, Breakaway Oxidation, and Quenching

The temperature rise during a LOCA event results in an increased cladding reaction rate. This reaction causes thicker layers of ZrO_2 to form on the cladding that eventually begin to spall. This behavior results in breakaway oxidation and high hydrogen pick-up fraction in the cladding. The amount of hydrogen pick-up decreases the cladding ductility (Adamson et al. 2007). The engagement of the ECCS results in rapid decrease in cladding temperature, which can result in the fracture of the embrittled cladding due to thermal shock. M5^{TM} and Zirlo^{TM} alloys have been demonstrated to have similar LOCA behavior to Zircaloy alloys (Sabol 2005).

A.3.4 Testing for Accident Conditions

No nuclear accidents have occurred that fall within the current DBA. Commercial reactor operators cannot perform accident testing without damage out-of-pile, and test reactors for in-pile testing are required to determine predicted performance under accident conditions. Unfortunately, the currently available reactor test facilities are unable to directly simulate many key parameters of commercial LWRs. The facilities available for RIA testing are outlined in Figure A-9 and compared to conditions in a typical PWR. The primary difficulty is the lower temperature and narrower pulses in the test reactor conditions that would result in reduced clad ductility and increased tendency for fuel fragmentation and fuel dispersal than in commercial PWR. These differences make transferring data from these test reactors to commercial reactors very difficult.

		SPERT US	PBF US	IGR KZ	BIGR RU	NSRR JP	CABRI FR	PWR/ BWR
Test conditions								
Coolant medium		Stagnant water	Flowing water	Stagnant water	Stagnant water	Stagnant water	Flowing sodium	Flowing water
Coolant temperature	[K]	293	538	293	293	293*	553	553 (BWR) 563 (PWR)
Coolant pressure	[MPa]	0.1	6.45	0.1	0.1	0.1*	0.5	7(BWR) 15.5 (PWR)
Power pulse width	[ms]	13-31	11-16	600-950	2-3	4-7	9-75	25-75

SPERT (Special Power Excursion Reactor Test Program) – US, shutdown 1970
PBF (Power Burst Facility) - US, shutdown 1985
IGR (Impulse Graphite Reactor) – Kazakhstan, operational
BIGR - Russia, operational
NSRR (Nuclear safety research reactor) – Japan, operational
CABRI - France, operational

Figure A-9. RIA testing capabilities compared to commerical RIA conditions (Jernkvist and Massih 2010).

The vast majority of LOCA testing has consisted of isothermal heating of fuel rods which are then quenched to simulate a LOCA; however, Maillat, Grandjean and Clement (2003) highlighted the deficiency in previous testing using this approach. A key shortcoming of this test method is the inability to match the temperature profile that results from neutron heating in a reactor environment. The fuel profile before and after quench during a LOCA is shown in Figure A-10. The middle of the fuel experiences a ~500^{o}C drop, while the edge of the fuel temperature increases by ~700^{o}C. These results can only be replicated via neutron heating. OECD is currently planning on LOCA tests that are neutron heated in Halden to determine the impact of having realistic temperature profiles during a LOCA quenching test on fuel performance (Rudling 2012).

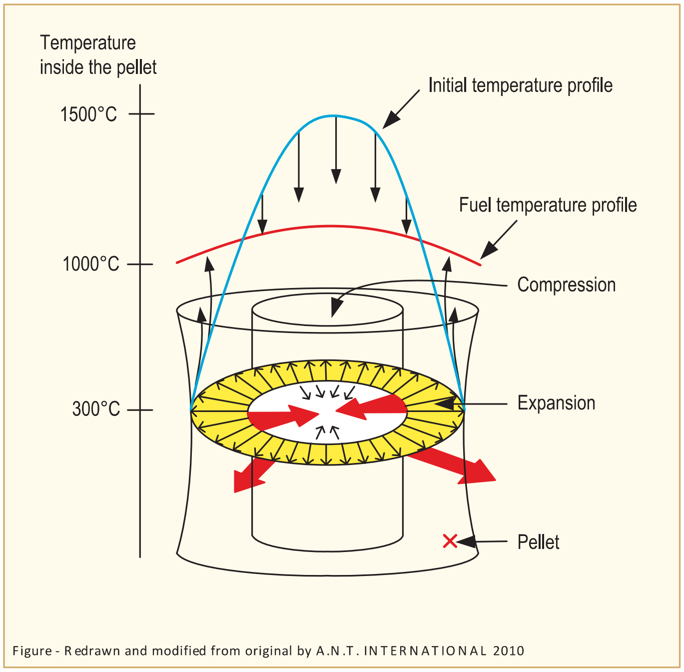

Figure A-10. Fuel temperature profile and resulting stresses during LOCA event (Maillat, Grandjean and Clement 2003).

Appendix A References

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Appendix B
Detailed Cladding Tests Proposed

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Establishment of a test matrix for candidate fuel and cladding shall include all licensing criteria and shall identify existing test standards or characterization tests for which standards must be developed for future qualification. Categories for cladding tests and proposed test objectives, constraints, or associated details are provided below.

- **Steam testing**
  - Leading cladding material candidates can be identified via a large array of tests of cladding coupons. Initial coupon screening could be followed by steam testing of short tubes before conducting a reduced test matrix with full-length rodlets under accident conditions.
  - ATF should demonstrate "longer" survival at elevated temperature before loss of strength and ductility relative to the current system; baseline tests should be included for the current cladding alloys.
    Note: No specific, desired coping time has yet been defined for accident tolerant fuels. However, it is assumed that a minimum increase in the anticipated survival time on the order of hours would be required for the industry to consider pursuing advanced cladding options. This assumption was generally agreed upon at the international metrics meeting [OECD/NEA 2013].
  - The definition of the test matrix must include:
    - Temperature ramp rate
    - Hold time at 800°C, 1000 °C, 1204 °C, plus higher temperature testing to determine point of failure
  - Matrix of environmental conditions:
    - 100% steam at atmospheric pressure and at elevated pressure
    - Mixed steam / hydrogen environment
    - Rapid quench – simulate ECCS injection
    - Steam shock test – expose heated cladding with dry steam (e.g. cladding at 1400 °C shocked with 1000 °C steam)
  - Quench at lower temperature (e.g. 800 °C)
    - Essentially freezes the H_2 in the cladding, preventing H_2 from precipitating out at higher T
    - This test could set a worst case for loss of cladding ductility
  - Measure mechanical properties before / after steam exposure
    - Tensile test, burst test, ring compression test, post-LOCA screening test for ductility
  - Measure onset of oxidation, oxidation rate, and heat of oxidation for specified conditions
  - Assess possible chemical interactions with other core materials, including the expected temperatures at which those interactions might be expected to occur
- **Pellet-Clad Chemical Interaction (PCCI)**
  - Diffusion couple tests at defined temperatures in a helium environment, at pressure
  - Test at normal operating temperature with contact between fuel and cladding
  - Test under stress at possible accident temperatures – 1130°C, 1204 °C, etc.
  - Note: Specific test temperatures should be established based on knowledge of eutectic formation in the proposed fuel / clad system

- **Erosion flow testing – normal conditions**
  - Flow velocity, thermal cycling, water chemistry control (pH, dissolved O_2, contaminants)
  - Assess adhesion of coatings, possible spalling of material
  - Evaluate effects of friction, wear and fretting
  - Determine effect of cladding selection (surface features) on heat transfer coefficient, DNB, etc.
- **Evaluate stress corrosion cracking (SCC) of cladding in coolant environment (autoclave test)**
  - Evaluate cold (pre-irradiation) and neutron irradiated material
- **Assess material stability**
  - Coefficient of thermal expansion (CTE) mismatch (particularly important for hybrid cladding options)
  - Test with external pressure on cladding tube
  - Coupon test under neutron irradiation
  - Measure irradiation creep
  - Test with 0.5 – 1% cladding creep down
- **Drop test (1-ft and 30-ft) to meet transportation requirements (10 CFR 71)**
  - Conduct normal tensile test; determine break point for strain
  - Evaluation criteria: fuel and cladding material should not deform at established acceleration limits (shock testing for fresh fuel)
- **Standard corrosion testing (normal operating conditions)**
- **Mechanical properties measurement**
  - Tensile testing (cold, at temperature, post-irradiation) – hoop and axial testing
  - Burst (pressurize to failure) – at room temperature and elevated temperature (~350^{o}C); higher temperature testing for LOCA analysis
  - Fracture toughness (testing of cold, elevated temperature, and irradiated sample coupons; success criteria must be established)
  - Fatigue testing – apply standard test for Zr-based clad (vendors currently conduct these tests)
    - Used to determine safe stress level
    - Cyclic loading under basic reactor conditions
  - Determine ductile to brittle transition – pre- and post-irradiation
  - Assess potential benefits of ion irradiation experiments – analysis should be conducted to clarify potential value and limits of ion irradiation for initial comparative studies of candidate materials
- **Measurement of thermophysical-mechanical properties – pre- and post-irradiation**
  - Elastic properties
  - Density
  - Thermal Conductivity
  - Heat capacity
  - Diffusivity
  - Emissivity
  - Microstructure and microstructure evolution

  - Evaluate possible processing-induced differences in the material, effect of radiation damage, multi-layer cladding issues
- **Measure tritium permeability**
- **Evaluate fabrication related issues** (e.g., ability to fabricate long tubes, seal end caps)
- **Robustness tests** (e.g., assess possible handling concerns)
  - Identify industry standards for fuel rod insertion into grid with regard to material hardness, scratching, etc.
  - Acceptable scratch depth for current Zr cladding is a maximum of 9% of wall thickness; evaluate how this requirement translates to the candidate cladding material
  - Measure material microhardness
- **Assess joining issues** (overall cladding hermeticity, weldability, joint mechanical strength, etc.)