Time-at-Temperature Operation of Light Water Reactors: Survey of Literature

Existing and Planned Research of Heat Transfer and Post-DNB Fuel Operation

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Abstract

Current United States Nuclear Regulatory Commission (NRC) regulations use departure from nucleate boiling (DNB) and dryout criteria to determine cladding heat transfer capability for loss-of-coolant accident (LOCA) and other selected non-LOCA accident analyses. However, the standard DNB criterion for the LOCA emergency core cooling system evaluation methodology does not account for the duration of the sustained temperature and its effect on material properties. Many anticipated operational occurrences (AOOs) involve a reactor trip that results in a short duration power increase, limiting the amount of heat that must be removed by the coolant. A set of fuel integrity criteria that allow for fuel to operate in post-DNB conditions for short durations would provide additional margin that might be used to greatly improve plant operational flexibility, allow faster startup times from outages, or justify increased reactor thermal power.

This report provides an overview of publicly available past, current, and planned research, including that by government agencies, related to or supports the potential development of time-at-temperature (TAT) criteria for fuel integrity assessment after operation in post-DNB conditions, specifically focusing on the conditions present during AOOs. The criteria would assess fuel integrity based on the time spent at elevated temperatures beyond the point of DNB, instead of precluding the occurrence of DNB.

The report also assesses the potential advantages of accident tolerant fuel (ATF) designs within a regulatory environment that adopts a set of TAT DNB fuel integrity criteria.

Keywords

Accident tolerant fuel (ATF)
Fuel integrity
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Post-dryout
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PRIMARY AUDIENCE: Safety analysis engineers, nuclear power plant operators, fuel licensing engineers

SECONDARY AUDIENCE: Risk assessment engineers, fuel vendors, thermal-hydraulic research groups

KEY RESEARCH QUESTION

Current United States Nuclear Regulatory Commission (NRC) regulations use departure from nucleate boiling (DNB) and dryout to determine cladding heat transfer capability for loss-of-coolant accident (LOCA) and other selected non-LOCA accident analyses, as required per 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." What research currently exists that would support adopting a time-at-temperature (TAT) post-DNB fuel integrity assessment that would allow limited operation in post-DNB conditions?

RESEARCH OVERVIEW

This report provides an overview of the results of a literature review of past, current, and planned research including that by government agencies related to or supports the potential development of a TAT criteria for fuel integrity assessment after operation in post-DNB conditions, specifically for AOO transients. Research on the failure modes, modeling considerations, and regulatory information relevant to post-DNB criteria is summarized. The applicability of accident tolerant fuel properties currently in development is also examined. Any gaps found in the current body of research are identified. Publicly available information from national laboratories, industry groups and organizations, scientific journals, regulatory documents, and industry experience, from both inside the United States and internationally are included in the literature search.

KEY FINDINGS

- Industry experience (Oskarshamn 2, Forsmark 2) and experimental findings (Halden Reactor Project IFA-613) provide evidence that fuel can experience dryout and safely continue operation.
- Empirical correlations have been developed and validated for film boiling heat transfer and fuel quench front propagation during rewetting after a power reduction.
- The Atomic Energy Society of Japan (AESJ) approved a fuel integrity assessment standard in 2003 based on dryout duration and cladding temperature.
- ATF might provide operational advantages in an environment that uses TAT fuel assessment criteria, especially since ATF materials are designed for more extreme conditions than those experienced by fuel in short duration dryout conditions.
- The most likely area of application of a TAT assessment is in AOOs that involve a quick power reduction or reactor trip, where this approach might only benefit plant operations that are currently limited by a transient fitting this description.
- Results presented in this document includes a summary of Department of Energy (DOE) planned research related to the topic.

WHY THIS MATTERS

A TAT fuel integrity criterion that allows for fuel to operate in post-DNB conditions for short durations might provide for improved plant operational flexibility, faster startup times from outages, faster power level adjustments, improved fuel cycle economics, or better optimized boiling water reactor control rod sequencing. These improvements have the potential to increase the efficiency and improve the economics of operating nuclear power plants.

HOW TO APPLY RESULTS

The results presented in this report are intended to serve as general information related to the status of potential development of TAT criteria, applicable to both BWR and PWRs worldwide. The results are not intended to imply that currently used safety criteria can be replaced or relaxed. The findings indicate to technological gaps that need to be addressed in furthering the interest.

LEARNING AND ENGAGEMENT OPPORTUNITIES

Findings summarized here affords the reader a concise collection of publicly available information and a discussion of the applicability of their contents towards the development of future TAT fuel integrity criteria. The gap analysis presented categorically identifies areas where future research efforts may be best profited.

Idaho National Lab, in support of the DOE Office of Nuclear Energy ATF program, has resumed operation of the Transient Reactor Test (TREAT) facility to support industry initiatives, including those by fuel vendors in testing their new and emerging ATF technologies. TREAT will serve as an independent test facility and the related test program offers learning and engagement opportunities to ATF developers and researchers.

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Nomenclature

AESJ	Atomic Energy Society of Japan
AOO	Anticipated Operational Occurrence
ATF	Accident Tolerant Fuels
ATWS	Anticipated Transient Without Scram
BT	Boiling Transition
BWR	Boiling Water Reactor
CAP	Cumulative Annealing Parameter
CHF	Critical Heat Flux
CPR	Critical Power Ratio
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
ECR	Equivalent Cladding Reacted
LOCA	Loss-of-Cooling Accident
LTA	Lead Testing Assembly
NRC	United States Nuclear Regulatory Commission
ODS	Oxide Dispersion Strengthening
PCI	Pellet-Cladding Interaction
PCM	Power Cooling Mismatch
PCMI	Pellet-Cladding Mechanical Interaction
Post-BT	Post Boiling Transition (BWR)

Post-DNB	Post Departure from Nucleate Boiling (PWR)
PWR	Pressurized Water Reactor
RIA	Reactivity Initiated Accident
RXA	Recrystallized, Annealed
SCC	Stress Corrosion Cracking
SPP	Secondary Phase Particle
SRA	Stress-Relieved, Annealed
TAT	Time-at-Temperature
TEPCO	Tokyo Electric Power Company
TREAT	Transient Reactor Test (Facility)

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Section 1: Background and Motivations

General Notes

The list of references, including a bibliography, is given at the end of this report represents all relevant sources discovered as part of this literature review. Many are discussed throughout the report, but not all. The additional bibliography section is provided so that the reader may have a consolidated listing of all discovered sources for further investigation. This review focuses on publicly available information, either from free postings by organizations, published journal articles available by subscriptions, libraries, or other public databases. Proprietary fuel vendor test and licensing data are generally outside of the scope of this report.

Post-departure from nucleate boiling (post-DNB) and *post-boiling transition (post-BT)* are used as synonymous terms in the report. *Post-DNB* is typically used in relation to pressurized water reactor (PWR) operations, while *post-BT* is typically used in relation to boiling water reactors (BWRs). They refer to the thermal-hydraulic regimes that exceed the point of critical heat flux (CHF) in the boiling curve when film boiling starts to develop in PWRs or critical power in BWRs is exceeded and the rod surfaces are exposed due to the evaporation of parts of the liquid film.

Thermal events induced by insulating crud buildup are not discussed in the report, as they typically result in extended dryout-like conditions, and therefore are outside of the short duration anticipated operational occurrence (AOO) conditions of interest of this review.

Project Motivations

Current U.S. Nuclear Regulatory Commission (NRC) regulations use DNB and dryout criteria to determine cladding heat transfer capability for loss-of-coolant accident (LOCA) and other selected non-LOCA accident analyses, as required per 10 CFR50 Appendix K General Design Criteria. Fuel reaching DNB or dryout conditions during a design basis accident (DBA) may not be suitable for continued use and must be replaced.

The driving force behind the NRC design criteria is the sudden deterioration in heat transfer that occurs after the DNB point is reached. The effective rod surface heat flux corresponding to this point is referred to as the *critical heat flux (CHF)*. At this point, the conditions are such that the liquid coolant in contact with the fuel rods begins transitioning from nucleate boiling to film boiling. The developing vapor film acts as an insulator on the hot fuel rod, and the heat transfer rate drops drastically until the rod is hot enough that radiation becomes the dominant mode of heat transfer. During that transition, heat transfer is insufficient, leading to rapid heat buildup and temperature increases in the fuel and cladding materials. This may ultimately lead to damage and possible failure of the cladding.

The NRC therefore uses the DNB point as a conservative measure to ensure that fuel is not damaged by these conditions. However, the standard DNB criterion does not account for the rate of the temperature increase, the duration of the sustained temperature and its effect on material properties, or the cooling potential of rewetting or quenching during transients such as LOCA. Many AOOs involve a trip, which limits the amount of heat that must be removed by the coolant. These AOOs, particularly those that result in very short durations in post-DNB conditions, would be the primary beneficiary of time-at-temperature (TAT) fuel acceptance criteria.

A fuel integrity criterion that allows for fuel to operate in post-DNB conditions for short durations would provide additional margin that could be applied in various areas, including improved plant operational flexibility, faster startup times from outages, faster power level adjustments, improved fuel cycle economics, or better optimized BWR control rod sequencing. An improved criterion would define an acceptable time-temperature space, which would specify an allowable amount of time that the fuel could spend at a given temperature and still be considered operable.

Industry Events

The international nuclear industry has some isolated experience with reactor operation in dryout conditions. It provides an indication that perhaps the strict dryout criteria currently employed are overly conservative.

The Swedish BWR Oskarshamn Unit 2 experienced a dryout of fuel rods during operation in 1988. It was not detected at the time and continued to operate. The damage was discovered during refueling and is thought to be caused partially by local effects from rod bowing. The fuel in the reactor was of Westinghouse's SVEA design. This event serves as an unintentional experiment on the continued operation of fuel after dryout and has been analyzed by various entities [1, 2, 3].

The Swedish BWR Forsmark Unit 2 experienced an event that adversely affected the main recirculation pumps. Specifically, an electrical grid disturbance caused by a lightning strike propagated into the unit and through a series of malfunctions, resulting in a trip of all reactor recirculation pumps. Analysis shows that, during the resulting loss of flow, some of the fuel assemblies experienced a short dryout period. Those fuel assemblies were removed and subjected to a retesting program to determine their suitability for further operation, which essentially involved assessing the fuel against its original licensing design basis. All affected assemblies were eventually reloaded and were able to continue operation until their planned end of life. Analysis of this dryout event provides evidence that short fuel operation duration in dryout conditions is potentially an acceptable occurrence [4, 5, 6]. The peak cladding temperature reached during the transient was calculated to be between 450°C and 550°C, which was only sustained for about one second before reverting to lower temperatures [4, Figure 1].

Additional Background

Kono et al. concluded through a set of experiments that near-instantaneous fuel failure upon experiencing DNB is not as likely, as previously assumed [7]. They found that the cladding temperature increase is moderate, especially during transients that experience a relatively quick reactor trip; it is unlikely that the cladding will reach a temperature that compromises fuel integrity.

Ritterbusch and Baily concluded from experiments that a maximum cladding temperature of 597°C for 15 seconds can be sustained without fuel damage [8]. They discussed the potential benefits of increased DNB margin in a nine-plant survey. The survey concluded that reactors limited by DNB can benefit by increased reactor power, which translates directly into economic gains at the time of survey in 1980. Those that are not limited by DNB can benefit through less restrictive scram setpoints, along with other benefits previously listed in this report.

Recent work by Shirvan [9, Section 3.3] suggests that the actual temperature response of nuclear fuel to a DNB may not be as rapid as past testing indicates. A simulation of a four-loop Westinghouse PWR was performed using SIMULATE-3K that examined DNB occurrence resulting from either a loss of flow condition or initiated from steady-state operation. Even without a reactor scram, the peak cladding temperature does not increase to a level that the materials would not be able to survive for brief periods. While more detailed simulation analysis is required before drawing concrete conclusions, this type of first pass analyses provides encouraging potential for future investigations. It is postulated in the paper that past CHF testing with electrically heated rods does not fully mimic the power response the fuel experiences; the thermal capacity of the fuel materials or negative reactivity feedback effects are unaccounted for. Neglecting these effects is a conservative assumption and is usually adequate for safety analysis, but some margin may be available if they are fully considered.

Operating Conditions of Interest

The most likely area of application of a TAT fuel integrity assessment is for AOOs. These events are determined to be frequent enough to occur one or more times during the life of the plant and are delineated in Chapter 15 of the US NRC Standard Review Plan [10]. A few examples of these events are turbine trips, recirculation pump trips, and interruption of coolant flow. The events may lead to short duration power increases or other conditions that affect the coolant thermal hydraulics enough to cause a section of the fuel to approach the DNB point, before a reduction of power or recovered flow quickly rewets the section in dryout. Using the comparable Atomic Energy Society of Japan (AESJ) standard as a guide, as described in Section 4 of this report, the total time spent past the DNB point would be on the order of 100 seconds at most, with peak cladding temperatures generally less than a maximum of 800°C, where oxidation-induced cladding embrittlement becomes a concern. The majority of AOO events that are limited by DNB are mitigated by a reactor trip within 10 seconds or less. After the reactor trip and the resulting reduction of power, DNB will no longer be a concern. Extremely short duration at-power reactivity initiated accident (RIA) conditions where temperatures spike above those ranges, but quickly return below, may also be viable candidates. These events are typically completed in less than 10 seconds [11, Figure 3-2,]. DBAs, such as a LOCA, are longer events and involve conditions at much higher temperatures; they result in effects that are unlikely to allow for fuel reuse. For this reason, a TAT criterion is not likely to be feasible for LOCA and other accidents with similar conditions. This report focuses on the application of TAT to AOO-like conditions.

Section 2: Physical Phenomena

Nuclear fuel that has experienced post-DNB or post-dryout operating conditions is subject to various microstructural and mechanical effects that may lead to fuel failure or degraded performance. The lack of adequate heat transfer capability of the coolant in those conditions leads to rapid temperature increases within the fuel pellets and the cladding materials. A TAT-based assessment will not be applicable to all transients or postulated events but may be expected to provide additional margin for select limiting transients, shifting which transients are considered limiting.

Operation in the post-DNB regime can lead to damage or fuel failure from a variety of physical phenomena. The benefits of relaxation of DNB limits would primarily be focused on scenarios with fast operational transients that do not spend much time in post-DNB conditions, which typically describe many AOOs. Most RIA events are unlikely to gain relief from DNB limit relaxation, except those initiated from at-power conditions. The conditions of LOCA events are expected to be too extreme and would fall outside the expected allowed TAT region.

Power-Cooling Mismatch

A power-cooling mismatch is a condition in which the cooling capacity lags the heat transfer from the fuel rods, leading to temperature spikes in the fuel. If left uncorrected, this can lead to failures or degradation of the cladding, discussed later in this section. An example of this would be during a pump trip that decreases the core flow. Until the reactor power is reduced, or the coolant flow restored, there is more heat generated in the fuel than there is capacity to remove it. Events of this type that the plant design can quickly remedy are good candidates to benefit from a TAT limit.

Cladding Annealing and Material Properties

Sufficiently elevated temperatures experienced during transients can lead to structural changes in zirconium alloy fuel cladding that greatly alter its material properties, including the strength, corrosion resistance, and ductility [12]. The dominant mechanism causing the physical changes is the annealing process, in which time spent in an elevated temperature environment alters the grain size and shape of the metal and its impurities compared to its as-manufactured state. The manufacturing process of the fuel cladding is precisely controlled to deliver the necessary physical properties and thermo-mechanical performance for operation at design conditions.

Mizokami et al. suggest that material recovery due to annealing that occurs during short dryout durations is predictable and may be able to be credited in a TAT criterion [13].

A common source of cladding embrittlement is radiation hardening during the life of the fuel. As Bauer and Lowry showed in their work, embrittlement from irradiation anneals rapidly when temperatures rise above 538°C, and the damage anneals completely when it reaches 704°C under transient heating conditions [14]. It was also noted that annealing of the irradiation damage occurs quicker than annealing of the original cold work material structure applied during the manufacturing process.

As previously discussed in the introductory section, Forsmark Unit 2 experienced a short transient condition in which some fuel experienced dryout conditions. This presented the opportunity for analyzing the effect on the fuel of an operating reactor. In two studies, authors calculated that the maximum cladding temperature reached was between 450°C and 550°C [4, 6]. They then proceeded to run experiments on sections of the fuel (sections that did not experience dryout) in which the sections were heated to those temperatures for either 0.3 or 3 seconds. They observed the annealing effects and changes in microhardness that were a result of those conditions [4, Figure 7].

The annealing process during cladding manufacturing is typically performed by holding the subject material at a constant temperature. The annealing that would have to be accounted for as part of a TAT assessment would likely take place during transient heating conditions. This adds the additional variable of heating rate, along with time and temperature. Most experiments found in literature either focus on fast transients (~1000°C/s) or slow (~1°C/s); although most studies are focused on accident conditions, the effect of heating rate is important to consider. As part of experiments on these two classes of transients, Jailin et al. noted significant effects on changes in material and grain structure based on the speed of the transient, as evidenced by a structural analysis and strain tests [15]. Experiments by Yvon et al., conducted with different heating rates between 480°C and 600°C, also show differences in tensile strength with varying heating rates [16, Figure 4]. The experiments also included spalled samples and various levels of cladding oxidation layers and how they responded to different heating rates.

Secondary phase particles (SPPs) present in the cladding material are also important in determining the material properties of the alloy and its response to various stresses and corrosion. These particles may respond differently to elevated temperatures than the main bulk of zirconium, which could be detrimental to overall material performance. Sparse non-proprietary data exist on the evolution of SPPs during a heating transient, but it is likely that fuel vendors have much more detailed data on the evolution of SPPs in response to different heating environments. Work by Toffolon-Masclet et al. gives an indication of the dissolution/precipitation threshold temperature of SPPs in two different groups of zirconium alloys [17]. Zirconium-Tin alloys showed a dissolution threshold of about 800°C and exhibited a mostly reversible, low hysteresis allotropic phase transformation. Zirconium-Niobium alloys showed a dissolution threshold of about 600°C–700°C, along with a higher hysteresis. In both cases, the dissolution threshold was very similar to the temperature of the beginning of the allotropic phase transformation. This is significant, since cladding that has undergone the $\alpha \rightarrow \beta$ phase transformation is unlikely to be a viable candidate for fuel reuse after a short-term dryout condition. If that transformation is reached, continued fuel use may be restricted. Information on the behavior of SPPs is important in understanding whether the strength or corrosion resistance of the fuel cladding has been compromised by the transient heating.

Robson used the cumulative annealing parameter (CAP) to quantify the effects of multiple annealing steps, usually applied during the manufacturing process of the zirconium alloy cladding [18]. Experiments performed by Robson, while mostly focused on longer timescales that are typical for annealing steps during cladding production, also show some insight to SPP radius evolution between 400°C and 800°C [18, Figure 3]. SPPs are shown to begin growth even in the very early stages of the annealing process.

Understanding the change in material properties and the effects on cladding strength, resistance to corrosion, and response to various stresses due to time spent at elevated temperature is critical to assessing cladding integrity using a potential TAT criterion. As demonstrated by the literature cited, even short periods of time spent at temperatures above design conditions can measurably affect the structure and material properties of the cladding. Material property considerations will likely play a major role in determining an acceptable post-DNB, time-temperature region.

Cladding Deformation

Cladding can fail due to deformations, including ballooning rupture, cracking, buckling, and collapse. At high temperatures that may be experienced in prolonged post-DNB operations, the cladding material becomes highly ductile. Internal rod pressure from fission gas release can cause higher tensile hoop (circumference) stress, and, when high enough and combined with elevated temperatures, may result in localized ballooning due to localized creep to the point of rupture. Buckling can occur in a ductile segment subject to axial forces. Experimental evidence suggests that cladding collapse may occur approximately at 647°C under expected conditions for a fuel rod [12]. Instances of pellet-pellet gaps within the fuel stacks are better controlled in modern manufacturing processes, which mitigate cladding collapse. Sufficiently high differential pressure across the cladding causes ballooning (higher interior pressure) or collapse (higher exterior pressure). In a similar fashion to cladding annealing, deformation characteristics of the materials at high temperatures will be an important factor for determining time limits for operation at elevated temperatures.

Jailin et al. examined the creep deformation of Zircaloy-4 claddings under post-DNB conditions [15, 19]. The test rig used is capable of up to 1200°C/s temperature transients. They subjected materials to fast transient temperature increases and analyzed the resulting deformations, ballooning, or creep. The effects on the material structure, strain, and creep rate were determined and categorized across three temperature domains, within an 840°C to 1020°C overall range. The speed of the temperature transient was found to have a significant effect on the observed creep and resulting mechanical structure [15, 19]. The temperatures are beyond the level expected for fuel to experience (800°C–1100°C), yet still be considered a candidate for use. However, the experiments show the deformation that is caused by the different heating rates and therefore can inform the selection of an upper bound of temperature, as well as identify the likelihood of deformation based on fast versus slow transients.

Mizokami et al. cite experiments in their summary paper showing that the degree of cladding deformation was predictable given a known cladding temperature and the time spent at that temperature [13]. This work forms some of the basis for a time-at-temperature AESJ standard for fuel assessment following a dryout period. That standard is discussed further in Section 4.

Pellet-Cladding Mechanical Interaction and Stress Corrosion Cracking

Pellet-cladding mechanical interaction (PCMI) refers to a situation in which increasing fuel temperature causes swelling of the fuel pellets due to thermal expansion. Eventually, the pellet swells enough to close the gap between the pellet and the cladding, contacting the inside of the cladding. Increasing fuel temperature continues to cause the pellet to expand, resulting in radial forces exerted against the cladding. The resulting increase in hoop stress can push the cladding to stretch, crack, or rupture. Claddings that have become embrittled due to prior irradiation and/or oxidation are less resistant to the hoop stress and can result in quicker failure relative to fresh materials [12, 20].

Stress corrosion cracking (SCC) is defined as "crack initiation and sub-critical crack growth of susceptible alloys under the influence of tensile stress and a 'corrosive' environment." As a subset to PCMI, the combined failure mode of pellet-cladding interaction, stress corrosion cracking (PCI-SCC) is a complex process that is a significant source of material degradation for fuel cladding materials. The mechanical stresses brought on by fuel dryout/rewetting processes could accelerate or exacerbate degradation due to PCI-SCC and would need to be considered. PCI-SCC typically has at least three prerequisites: sustained tensile stress, chemically active fission products, and a susceptible cladding metallurgical state [21]. PCI-SCC has been observed to exacerbate PCMI, eventually leading to failure in operating reactors, as examined by Smith et al. at Maine Yankee [22].

Fuel failure due to PCMI or PCI-SCC typically occurs during normal operating conditions that are below the DNB point, so this is not a specifically transient-driven failure mode. The resulting stresses from the PCMI and PCI-SCC may be a potential issue as the material properties of the cladding change due to time spent at an elevated temperature.

Fuel Rod Bow

High temperature operation, especially above 647°C, during post-DNB conditions can lead to temporary or permanent rod bowing [12]. This is near the top of the temperature range of interest (400°C–800°C), but the potential for rod bow may inform an upper limit on allowed post-DNB temperature. Rod bowing can affect reaction kinetics or alter coolant flow paths that may affect the heat transfer characteristics of the impacted and surrounding fuel assemblies. Bow effects must be considered when computing heat transfer coefficients, as shown by Groeneveld [23]. Aleshin et al. and Gabrielsson et al. analyzed and validated methods of predicting rod bow within a fuel assembly [24, 25]. Bowed rods reduce the amount of coolant that can flow within certain subchannels, degrading heat transfer ability and potentially inducing DNB conditions earlier than normally expected.

Cladding Embrittlement

Irradiation over the assemblies' lifetime is a primary factor of cladding embrittlement. An additional source of embrittlement is the steam oxidation reaction that occurs at high temperatures. Zirconium alloy claddings subjected to contact with high temperature steam experience rapid oxidation and absorption of hydrogen into the cladding metal [26]. The cladding material becomes increasingly less ductile with increasing levels of oxidation and hydrogen uptake, reaching a transition point where it becomes brittle and much more susceptible to cracking or rupture failures [27]. An embrittled cladding is much more likely to fail from stresses resulting from the combination of PCMI and high differential pressure across the cladding due to fission gas release from the fuel pellets. The effects of hydrogen and hydriding will be further discussed later in this section.

Steam oxidation becomes a concern only at temperatures above the operating range (400°C–800°C) that this report is focused on; the oxidation reaction typically occurs at temperatures higher than 800°C. The Cathcart-Pawel correlation, used to calculate the amount of cladding oxidation due to steam based on exposure duration, is valid at peak cladding temperatures starting at 1000°C [26]. At these temperatures, the transient has moved beyond the conditions that the plant would experience during a short AOO. Conditions and metallurgical changes are extreme enough that the fuel will likely be considered inoperable.

Halden Reactor Project Experiments

A test series examining dryout performance of nuclear fuel was performed as part of the OECD Halden Reactor Project in Norway, co-sponsored by TEPCO, using instrumented test assembly 613 (IFA-613). The objective of the dryout test series was to provide information on the consequences of short-term dryout incidents for the fuel in a BWR. The experimental method employed is exposing fuel rods with different levels of burnup to single or multiple dryout events, and to follow this by either unloading or continuing operation in the reactor. After exposure to the given operating conditions, the materials were examined and tested with emphasis on fuel cladding properties [28, section 4.2].

After subjection to varying dryout durations and occurrences, the rods were examined, using both destructive and nondestructive techniques. Some test fuel rods were operated for a month after the dryout period and subsequent rewetting. While the events altered the material structure of the cladding, none of the fuel rods or claddings failed during the return to operation period. The tests provide valuable data on the consequences of fuel operation in the post-DNB and dryout regimes and have been analyzed by multiple international groups [28–31].

Hydrogen and Hydride Redistribution

During the corrosion reaction of zirconium-based alloy fuel rod claddings under normal reactor operating conditions, hydrogen is released and a fraction of it is adsorbed at the oxide-metal interface of the cladding. This hydrogen is then absorbed into the bulk of the cladding metal. Upon exceeding the temperature dependent solid solubility limit, the hydrogen combines with zirconium metal and forms hydrides (namely, $ZrHx$ where $1<x<2$). These hydrides are small platelets that agglomerate into long stringers in the metal matrix with a length-to-thickness ratio significantly greater than unity [32, 33].

The formation and morphology of hydrides are also dependent on the microstructure of the cladding. As-fabricated texture will determine the orientation of the hydrides relative to the radial-transverse plane of the cladding. In stress-relieved, annealed (SRA) textured clad, the hydrides are predominantly circumferentially oriented; in other words, the long axis of hydride stringers is along the hoop direction of the clad. There may be some radially oriented hydrides in SRA cladding, but their fraction is much lower than those with a circumferential orientation. For recrystallized, annealed (RXA) textured clad, the hydrides are both circumferentially and radially oriented. In addition to this effect of as-fabricated texture, hydride morphology and orientation also depend on the applied tensile stress, especially in the hoop direction of the cladding, so that radial hydrides are more likely to form [34, 35].

The distribution of hydrides in the cladding metal also depends on the texture, as well as the thermal gradient across the cladding wall integrity of the outer-surface corrosion (oxide) layer, and presence of a barrier layer or liner on the inner surface of the cladding. As the number density of hydride stringers increases (along with bulk hydrogen content due to increased corrosion and the amount of hydrogen adsorbed by the corrosion reaction), there is a higher probability that the hydrides agglomerate toward the outer, cooler surface of the cladding. If the number density of hydrides is sufficiently low, then the hydrides tend to be mostly uniformly distributed across the cladding wall, with some texture effects limiting the number of hydrides present toward the inner surface. In the presence of an inner-surface barrier or liner, hydrides tend to agglomerate in the liner due to a lower solubility limit as the liner has different alloying elements compared to the rest of the cladding. Some of these material behaviors are competing, so the distribution of hydrides is very complex and dependent on the thermo-mechanical state of the cladding [36, 37].

During elevated temperatures associated with AOOs, a portion of the hydrides redissolve according to the solubility, and more hydrogen is put back into solution within the zirconium alloy matrix. This soluble hydrogen is then available to diffuse along temperature and concentration gradients. Some AOOs are short in duration and may be cyclical; hydride dissolution and reprecipitation are very complex and dynamic processes. Additionally, AOOs are likely to increase the mechanical stress on the cladding so that reprecipitation of hydrides may be more favorable in the radial direction. If the corrosion oxide layer spalls from the cladding during AOOs, hydrogen and hydrides may further agglomerate at these colder regions [35, 38].

The resulting microstructural changes of the cladding and changes to the hydride morphology, orientation, and distribution all need to be considered in evaluating the subsequent mechanical behavior of fuel rod cladding in post-AOO conditions. As noted previously, increasing amounts of hydrogen and hydrides in the fuel rod cladding contributed to a loss of ductility (that is, increased embrittlement).

Section 3: Modeling Considerations

Modeling Overview

An updated fuel integrity criterion may require modeling methodologies capable of simulating the mechanical forces and heat transfer processes in a post-DNB transient. There are multiple valid strategies, but full simulation of post-DNB performance is a multi-component process, with different software specializing in various areas. In broad terms, a high-level modeling strategy would be accomplished using three categories of software: systems modeling, subchannel thermal hydraulics, and fuel performance and heat transfer. The various software would provide applicable boundary conditions to the others, in addition to the system boundary conditions. This type of approach is typical in existing reactor safety analysis but would need to be extended to accurately model the new operating regimes.

Systems analysis codes such as RELAP, RETRAN, and TRACE have been used extensively throughout the nuclear industry for simulation and prediction of both large- and small-scale systems, especially in the simulation of plant transients. Primary and secondary cooling systems, loop components, control systems, emergency systems, and so forth, are modeled to predict plant system performance and response during postulated accident conditions.

Subchannel analysis codes such as the COBRA and VIPRE code families specialize in modeling the core flow and heat transfer from the reactor fuel rods in a fuel bundle to the core coolant. These codes, equipped with CHF and critical power correlations, are used to predict, among other things, the occurrence of DNB or dryout along portions of the heated rods. The scope of these codes is limited to the reactor core.

Fuel performance codes include Idaho National Lab's BISON/MARMOT, the Electric Power Research Institute's (EPRI) FALCON, the NRC's FRAPTRAN and FRAPCON, and vendor-owned codes. These codes are validated to predict a wide range of fuel operation phenomena, including mechanical stresses in the fuel and cladding, deformation, fuel and cladding temperatures, and so forth.

The abovementioned codes are all validated to predict fuel element behavior within their specific scopes. However, the current NRC licensing procedures require that operation under DNB is prevented within the enveloped postulated accidents, and thus the codes are typically not used or validated for operation in post-DNB conditions. The following sections describe additional considerations that are important to the modeling of transients that include post-DNB operation.

Modeling a plant transient in which the fuel is allowed to experience DNB and then recover requires reliable transient models or correlations to model the heat transfer that occurs in the boiling transition and film boiling regimes, the subsequent rewetting/quenching behaviors and associated material effects, the effects of time and temperature on material properties, and other mechanical and chemical fuel models for use at higher temperatures as opposed to those available for lower temperature operation.

Shirvan [9] provides a good overview and discussion of the considerations and effects of accident tolerant fuels (ATFs) on thermal hydraulic modeling and research, most of which is also applicable to non-ATF materials.

Fuel Mechanical Properties

Fuel performance codes include verified models for material properties, pellet physics, cladding physics, and so forth, applicable at times during or leading to fuel failure. Additional areas of concern for post-DNB operation and the associated high temperatures include changes in strength or ductility (annealing), effects on material corrosion behavior due to microstructural changes, SPP distribution and composition evolution, hydrogen pickup, and other effects related to these operating regimes. Some of these aspects will be covered by existing knowledge of material properties, because LOCA analysis involves temperatures and conditions much higher than the operating regime in question. On a related note, cladding embrittlement due to steam oxidation is not considered to be a contributing factor for short duration AOO events, as the temperatures at which steam oxidation occurs are generally higher than the regime of interest for this report. The widely accepted Cathcart-Pawel correlation, used to calculate oxidation due to steam, is only valid for local temperatures starting from 1000°C and higher [26].

The canonical source for material properties for light water reactor analysis is the MATPRO library [39]. Chapter 4 of this source contains the material property correlations for zircaloy. The library contains the basic properties needed to model zirconium in the temperature region of interest, including the effect of annealing of the cold work applied to the cladding alloy. An additional source of zirconium material properties under transient heating conditions is the properties database presented by Papin et al. [40]. The experiments were conducted under postulated RIA conditions (fast transients), but the resulting properties database covers a large range of temperatures, heating rates, strain rates, and loading conditions.

It is well documented that zirconium cladding alloys experience hardening and embrittlement from irradiation. This may be counteracted at high temperatures due to annealing [14]. This recovery factor may help prevent cladding cracking and subsequent failures in certain circumstances [13]. Models should therefore take the history of the fuel and cladding into account.

A primary area of concern that does not appear to have been addressed by the currently available body of research and material properties databases is the situation in which the cladding materials experience elevated temperatures for some time and then return to normal conditions, perhaps multiple times during their operating lifetime. This loading pattern would be expected to occur if a TAT-based fuel assessment allowed continued use of fuel that experienced transient post-DNB or post-dryout. As shown by the research presented in Section 2 of this report, the cladding microstructure, hardness, and SPP distribution may be altered under such conditions. Although explored for a few conditions, as mentioned previously [17], the amount of hysteresis accumulated by the potential repeated transients is not addressed in the current, publicly available literature. This is especially true in the operating regime expected for short duration AOO-driven post-DNB and post-dryout, and will be critical information for designing a fuel assessment methodology, as well as upper limits on the severity and frequency of the allowed transient conditions.

Kremer et al. created a model and validated a simulation code for predicting high burnup structures and gas release in nuclear fuel pellets [41]. This work shows the burnup and temperature dependency of fission gas release, which contributes potential stresses on the cladding. Excessive fission gas release can cause compromised cladding material to balloon and rupture. This work suggests that burnup levels will need to be accounted for when determining whether a transient would lead to potential failure by ballooning if the material properties have deteriorated during the short post-DNB or post-dryout operation.

Film Boiling Correlations

Post-DNB heat transfer occurs in the transition and film boiling regimes. Appropriate heat transfer coefficients must be used for these regimes when modeling and predicting post-DNB operating conditions.

The Groeneveld correlation [23] is the original work on which many subsequent film boiling correlations are based. Subsequent studies endeavor to improve upon it by accounting for various additional considerations.

Akiyama et al. performed a series of experiments studying post-BT heat transfer in a BWR 8×8 rod bundle [42]. A full bundle was used to include grid effects that are not captured by single rod or reduced-array experiments. The authors concluded that the post-BT temperature rise in their experiments is not excessive and indicated that short durations in that regime are tolerable. Rods were electrically heated and rod temperature was limited to 500°C. The transients were slow and controlled to study response along the heating curve. Additionally, five separate film boiling correlations were examined against the experimental data and compared for accuracy. The Dougall-Rohsenow, Groeneveld, and Condie-Bengston correlations tended to under-predict the heat transfer coefficient in the transition boiling region, but they accurately predicted the film boiling region. The Koizumi and Sugawara correlations were found to predict the heat transfer over the whole data range well.

Tsukuda et al. performed a series of heat transfer experiments for post-DNB heat transfer in PWRs [43]. Either water or freon was used as a working fluid, cooling electrically heated rods. Film boiling was studied using steady-state tests and rewet/quenching behavior by transient tests. The authors propose a new evaluation model for post-DNB rod behavior, which predicts the experimental results well. The new model combines a (new) modified Dougall-Rohsenow correlation for film boiling, the Blair rewet velocity correlation, and the MIRC-1 DNB correlation. The film boiling correlation is used for areas experiencing DNB. This model was found to perform well in predicting the maximum rod surface temperature and to conservatively predict film boiling duration (43, Figures 16 and 17), which are the significant parameters in judging fuel rod integrity.

Sibamoto et al. performed experiments measuring post-BT heat transfer and proposed a model for heat transfer coefficients [44]. The proposed model applies to the transition region between portions of the heated rod experiencing film boiling and the portions that are still under the DNB point. This is accomplished by combining vapor film and direct coolant/cladding contact section heat transfer coefficients, accounting for droplet deposition downstream of the quench front. 6A simple heat transfer coefficient exponential decay model dependent on quench front location is also presented.

Rewetting Correlations

When conditions return to the nucleate boiling regime, the rods go through a quenching process as the liquid coolant returns to direct contact with the hot cladding surface. This rewetting process is characterized by the propagation of the quench front of the liquid, which separates the wet and the dry regions. A model predicting the rewetting velocity is important when predicting the duration in post-dryout fuel operation, and for post-DNB, the rate at which the boundary of the vapor region recedes.

Yasuteru et al. performed a series of experiments that examined rewetting phenomena on a post-dryout fuel rod operating in post-BT operating conditions [45]. Their measured quench front velocity was approximately 10 times faster than a typical reflood rewetting case. It was found that droplets entrained in the vapor layer contributed a significant amount to the heat transfer via a so-called “precursory cooling” process. Droplets mixed within the vapor deposited on the hot surface ahead of the quench front, increasing the rate of heat transfer and allowing the quench front to progress faster than would typically be predicted. A new quench front velocity model was developed that takes the precursory cooling into account, which correlated well with the measured experimental data.

Satou et al. also examined rewetting heat transfer for Japan Atomic Energy Agency [46], including the precursory cooling effects mentioned above. Their tests were performed to simulate conditions for an anticipated transient without scram (ATWS) on reduced rod arrays. They showed the effects on heat transfer coefficient of swirl spacers on post-BT heat transfer and rewetting, as well as other effects. The authors specifically cite the potential for a short duration dryout acceptance criterion as a driver for the research. They note future planned work to examine heat transfer under oscillatory thermal hydraulic conditions, which is an important gap in the current research.

Existing Post-DNB Modeling

A recent simulation study that coupled SIMULATE-3K and BISON was performed by He et al. that calculated preliminary predictions of cladding failure times while operating under post-DNB conditions [47]. Failure times ranged from less than a day to more than a week, depending on the set of conditions used. A failure condition was assessed based on a range of criteria: (1) % ECR, (2) plastic strain limit, (3) hydride formation, (4) creep strain, and (5) hydrogen concentration. The authors also identified areas of refinement for the model. The paper shows a promising start and a review of the required models and considerations needed for successfully modeling post-DNB fuel behavior. The authors note that the work does not have a directly applicable validation data set and consider their findings as preliminary and based on numerous assumptions.

Existing CHF Thermal Hydraulic Database

EPRI compiled a report in the 1980s that summarized and presented the copious amount of research performed at the Columbia University Heat Transfer Research Facility related to CHF [48]. The report describes the experimental setups and conditions, and contains the resulting CHF data found during the tests. The data presented in the reports were reviewed, and it was determined that they did not contain any information on post-CHF operation and heat transfer that would be useful for modeling post-DNB conditions. The testing was typically performed until a rod experienced CHF, and then the experiment was terminated. The available temperature data are not presented in a format that allows the extraction of information related to post-DNB operation. The correlations and subsequent work the experiments support form the backbone of thermal hydraulic information that should be used in the beginning of the boiling curve, up until CHF is reached. Models must then transition to using the other heat transfer models presented above to model the rest of the transient.

Section 4: Regulatory Status

United States NRC Relevant Regulations

A review of publicly available NRC documents did not identify any regulatory work in progress examining an allowed duration of time spent in post-DNB or post-dryout operation. To date, the NRC has not adopted or proposed rules, or conducted investigations, to allow continued use of fuel operated in post-DNB or post-dryout conditions.

The only related activity is a newly proposed set of rules that govern fuel cladding performance in response to cladding embrittlement resulting from steam oxidation. A proposed rule (RIN 3150-AH42), titled *Performance-Based Emergency Core Cooling Systems Cladding Acceptance Criteria*, has completed the public comment phase, along with a Draft Regulatory Guide, DG-1263 [27, 49, 50]. However, these rules focus on accounting for oxidation from steam that occurs at temperatures around 1000°C and above, which is well beyond the operating conditions that occur during short-term AOOs.

Westinghouse Fuel Licensing Documents—Dryout Performance

Westinghouse submitted licensing documentation on critical power experiments and a new critical power ratio (CPR) correlation for Westinghouse SVEA-96 Optima3 BWR fuel assemblies [1]. The document version referenced here is heavily redacted to remove proprietary data; however, a request for additional information (RAI) response [2] during the NRC review includes discussion of fuel dryout performance in its Appendix 1. Discussions of dryout experiments and Oskarshamn 2 industry experience are cited to demonstrate that fuel operation in dryout can be tolerated for a short period of time without adverse effects. For example, the document cites that the Halden IFA-613 experiment concluding that fuel rods were able to sustain dryout for 80 seconds then continue to operate for 30 days without observed failures. A figure from the same experiment is also presented that gives a creep strain limit that is a function of time spent in dryout at a given peak cladding temperature [2, Figure A1-1].

Unfortunately, for the purposes of this literature review, a later revision of the appendix containing the information above was fully redacted in the final version of the non-proprietary report, so it could not be determined whether there was any change to the information when finalized. It can therefore be relied on only as preliminary without increased access to the documents. However, at a minimum, it demonstrates that Westinghouse and the NRC have previously considered this general type of dryout under operating conditions.

AESJ Standard for Fuel Integrity Assessment

The Atomic Energy Society of Japan Standard Committee published a standard in 2003 for the assessment of fuel integrity in case of an AOO in which the fuel experiences post-BT operation [51]. While the standard itself was not able to be procured, an article published in the *Journal of Nuclear Science and Technology* provides an adequate summary [13]. Two standards are provided: one for judging the integrity of a fuel bundle that experienced a BT event, and another for judging whether a fuel bundle may be reused in BWRs. The standards consider, among other things, rewetting propagation studies, experimental evidence of operation in dryout, cladding deformation, and annealing effects to define a methodology and acceptability threshold for continued reactor operation.

Figures 17 and 18 of Hara et al.'s article in the *Journal of Nuclear Science and Technology* [13] depict the AESJ acceptability criteria. After analysis using the specified methodology, acceptability is determined by how long the fuel experienced dryout during post-BT operation and at what temperature. If the combination lies inside the region on both charts, the fuel can continue operation. Allowable temperature limits range from maximums of 600°C to 800°C, up to a duration of 100 seconds.

The standard aims to provide additional operational margin and flexibility for reactor operations during some AOOs in which the rise in reactor power is limited by the reactor trip, which leads to very short times of high power that could result in the fuel experiencing post-BT conditions.

It is undetermined whether the standard was ever adopted or approved by the Japanese Nuclear Regulation Authority. During examination of the Forsmark 2 dryout event, Ramenblad et al. used the AESJ standard as one of their reuse criteria, signaling that there is some level of acceptance of it within the international community [6].

Section 5: Accident Tolerant Fuels

Overview

Recent efforts from nuclear fuel vendors have resulted in the development of fuel and claddings that use a variety of new materials that are more resistant to the types of damage and processes that occur under accident conditions. The fuels are designed to withstand much higher temperatures or to be more resistant to degradation than those currently in use. Industry experience from the Fukushima Daiichi accident has spurred the industry to develop these new fuel designs to better cope with those kinds of accident conditions.

ATFs would naturally complement and support a TAT fuel integrity criterion due to their higher tolerance to accident scenarios. The higher resistance to degradation under accident conditions would allow for improved margins and flexibility for plant operations for many postulated AOOs. Fuel that can operate in post-DNB and post-dryout conditions for short durations without requiring their replacement would lead to operational cost savings and increased operational flexibility.

Fuel vendors, partnered with multiple U.S. nuclear plant operators, are actively testing ATF designs in operating reactors around the country, alongside conventional fuel. These lead testing assembly (LTA) programs are expected to provide data to the fuel vendors by subjecting the new designs to normal operating conditions and allowing the examination of the effects of irradiation and burnup on the new materials [52].

Shirvan provides an overview and assessment of the thermal-hydraulic considerations that will be needed to capitalize on the advantages of ATFs [9]. The author credits the potential for replacement of the DNB criteria with performance-based criteria as an area that could provide for large benefits to the operational economics and safety of plants.

Many of the ATF materials identified here are designed to handle much more severe conditions than the short duration AOOs that are the focus of this report. The properties that allow the new materials to operate in extreme design conditions are likely to provide significant benefits at lower temperatures and smaller accident durations, since there will be improved safety margins, in general. For example, a lower susceptibility to oxidation during LOCA may also make the cladding material more resilient to slow oxidation or diffusion of hydrogen or other degradation mechanisms that affect fuel at lower temperatures during the core life of the fuel.

Materials and Material Properties

There are multiple types of materials in current or planned development for use in ATF designs [52–55].

Fuel vendor organizations, Framatome and Westinghouse, are developing UO_2 fuel pellets that are doped with either chromia or alumina. The doping materials result in larger grain sizes within the pellets, which decrease the effect of mechanical forces exerted on the cladding during a potential pellet-cladding interaction. The grain size also helps control the amount of fission gases released under accident conditions. These advantages lessen the mechanical stress exerted on the cladding [53]. This reduced mechanical stress would help prevent fuel failure due to PCI and clad ballooning.

For example, Westinghouse is developing a new type of fuel pellet that is made from uranium silicide (U_3Si_2), instead of the UO_2 used in conventional fuel products. The melting point of U_3Si_2 is lower than UO_2, but the thermal conductivity is about 5 times greater. The increased thermal conductivity would allow for more efficient heat transfer out of the of the fuel pellets, which is advantageous for controlling the maximum fuel temperature during transients when adequate core cooling capabilities are present. The effect of the operating conditions of a reactor on the mechanical properties of U_3Si_2 is still largely unknown, or at least not public information [53].

Fuel vendor organizations—General Electric, Global Nuclear Fuels, Westinghouse, Framatome—are developing various coatings that are applied to the outside of conventional zircaloy claddings [53, 56]. The coatings consist of materials such as chromium and aluminum alloys and are designed primarily to protect the substrate from steam oxidation. The coating reacts and oxidizes before the zircaloy and forms a protective oxide layer (referred to in literature as *scale*) that decreases the rate of oxidation in the cladding by orders of magnitude, thereby providing protection against corrosion and embrittlement. The oxide layer also can offer some additional protection from mechanical wear and fretting. Retarded oxidation rates on zircaloy cladding would be advantageous for a TAT-based evaluation criterion, as cladding embrittlement oxidation is heavily time dependent. Slower oxidation rates provide for a longer time to failure. In general, higher chromium content leads to a reduction in the cladding oxidation rate.

Westinghouse and Framatome are independently developing new types of claddings made from silicon carbide composite (SiC) materials. Pure SiC is resistant to steam oxidation and has high mechanical strength and melting points, with the potential for melting temperatures somewhat similar to those of the fuel itself. SiC is hard, but brittle, which is advantageous under some loading and wear conditions, but could be more susceptible to failure in others. Another potential downside is the decrease in conductivity compared to zircaloy, which can lead to higher internal fuel temperatures. Similar to the coated claddings, SiC-based cladding materials are likely to have increased resistance to steam oxidation, which would also be advantageous in allowing longer operations at elevated temperatures under accident scenarios [53].

Westinghouse has recently produced LTAs using U_3Si_2 fuel and chromia-doped fuel pellets, and Framatome and Global Nuclear Fuels have produced LTAs using chromium-coated zirconium claddings. These are currently placed in selected operating commercial reactors for testing and subsequent analysis.

Kim et al. describes the potential to enhance the zirconium alloy surface using oxide dispersion strengthening (ODS), a widely used metallurgical technique for metal strengthening [55]. The treatment is known to significantly strengthen the material, especially in high temperature situations. ODS provides an increased burst temperature, along with higher resistance to ballooning and creep deformation [55]. However, it also results in an increased oxidation rate relative to normal zirconium.

Economics—EPRI Publication

EPRI produced a report in 2019 that examined the economic viability of ATFs [57]. The report evaluates the advantages of ATF in beyond design basis accidents, design basis accidents, and anticipated operational occurrences. The report concludes that ATFs may provide an additional 1–3 hours of coping time during severe accidents, reduce the likelihood of core damage in a probabilistic risk assessment analysis, and provide other advantages. It also indicates that a TAT criterion, combined with ATF, could present opportunities to realize economic gains due to the increased operational flexibility to nuclear plants.

Current Research

Terrani et al. performed a series of experiments at Oak Ridge National Laboratory that evaluated and compared the oxidation kinetics of proposed ATF fuel claddings with traditional zirconium-based alloys [26]. They focused on cladding materials or coatings that form protective chromia, alumina, or silica layers that are expected to slow the rate of oxidation of the cladding. In Terrani et al.'s study, Figure 6 shows that the oxidation rate of some of the new materials can achieve up to three orders of magnitude slower compared to zirconium alloys alone [26]. This is also supported by other work by the same research group [58].

Lee et al. performed experiments that examined the durability of the ATF cladding coatings and oxide layers in response to scratching and wear [59]. Portions of the fuel rods that are in contact with the spacer grids or other assembly support structures are subject to wear over the core life of the assembly. The coatings showed an increased resistance to wear and fretting in comparison to zirconium alloys and showed potential to remain intact and resistance to high temperature steam oxidation. The ability of the coatings to withstand this type of wear is critical to credit the lower oxidation rates in safety analysis.

Faucett et al. used MELCOR to examine potential benefits of ATF in severe accidents and showed improved core degradation timelines [60].

Westinghouse has tested chromium cladding coatings up to 1500°C and SiC claddings up to 1700°C [52]. Tests have been performed, are in progress, or are planned to examine corrosion rates, steam oxidation rates, and burnup effects.

Unocic et al. performed experiments to quantify the effect of varying aluminum and chromium content in certain cladding alloys (FeCrAl and commercial AMPT alloys) on the rate of steam oxidation [61].

Future Research—INL Publication

Further testing of current ATF materials will be required as the involved technologies are refined and improved. Idaho National Laboratory, in support of the Department of Energy Office of Nuclear Energy ATF program, has returned the Transient Reactor Test (TREAT) facility to operation. The lab will partner with industry groups and fuel vendors to use the test reactor in support of testing their developing technologies. It will also serve as an independent test facility to evaluate the emerging ATF products. The TREAT reactor was constructed in the 1950s and suspended operation in 1994 before resuming operation in 2017. The reactor is designed to conduct controlled fast transient nuclear reaction tests using actual fuel designs. A high degree of control is available to shape the transient and define the operating conditions.

INL plans to support U.S. fuel vendors in developing ATF products, starting in 2019, in a five-year effort with multiple series of tests that include chromium-coated claddings in various TAT transients [53].

Section 6: Summary and Gap Analysis

Summary

This report provides an overview of the results of a literature review of past, current, and planned research that is related to or supports the potential development of TAT criteria for fuel integrity assessment after fuel operation in post-DNB and post-dryout conditions. Various scientific journals, reports, industry group publications, and regulatory documents that address the topics of post-DNB/post-dryout fuel operation and failure, modeling techniques, and ATF developments were reviewed. All materials referenced herein are available publicly or via a journal or membership subscription.

While not complete and requiring additional work in various areas, a basis of supporting literature was discovered that shows the potential to develop TAT acceptance criteria. A large body of research performed in Japan in the early 2000s is particularly relevant to the subject; the Atomic Energy Society of Japan (AESJ) published a significant body of work supporting a similar type of standard. Most of the directly relevant information was found in international publications, including the Halden Reactor Project, Forsmark 2, and the aforementioned Japanese research. Multiple publications were found to agree with the potential to develop TAT fuel integrity criteria to provide additional margin that could translate to economic or operational benefits for nuclear power plants. The most likely area of acceptance of this type of methodology is in short duration AOOs, involving a quick reactor trip, in which the core could feasibly be allowed to enter a short period of dryout without suffering fuel damage. Fuel that experiences extended duration or high temperature dryout events is unlikely to be a candidate for reuse due to changes in cladding material properties and other concerns.

An overview of the types of fuel failure relevant to post-DNB and post-dryout operations, the modeling considerations in those regimes, and regulatory views of the processes were presented. The reader is encouraged to seek further details from the included reference list. Cladding material property changes appear to be dependent on the time-temperature operating space, and thus will represent the primary limiting factor. ATFs, which are designed for operating conditions more extreme than those experienced during short-term AOOs, show promise in the ability to take advantage of post-DNB/post-dryout operations through TAT criteria.

Gap Analysis

Based on the findings of this literature review, it is concluded that the following technical aspects would benefit from additional experimentation and investigation that will support the justification of fuel integrity assessment criteria based on TAT for short duration (~10 seconds before a reactor trip) AOOs:

- Experiments focusing on repeated dryout-rewet cycling and its effects on material properties. Any type of cumulative effect of this cycling would be important input to establishing the new criteria. The mechanical stresses on the fuel assemblies in boiling flow are not necessarily representative of normal operation. This is also suggested by Shirvan [9]. There may be unknown hysteresis effects on the cladding material properties that may potentially compromise its strength, corrosion resistance, or susceptibility to PCI after returning to normal operation. Studies on the thermal hydraulics of this cycling for BWRs are planned by Satou et al. [46].
- Studies examining repeated excursions into post-DNB and post-dryout operation over longer time scales. An upper limit of the allowed number of DNB/dryout episodes for a specific rod over its lifetime would need to be defined based on cumulative damage or the metallurgical changes experienced as the result of each transient.
- The mechanical stresses brought on by fuel dryout/rewetting processes in BWRs and post-DNB/return to nucleate boiling in PWRs could accelerate or exacerbate existing cladding degradation due to PCI-SCC. This type of failure mode should be considered for future research.
- Targeted experiments measuring the response of zirconium claddings and the subsequent material changes for short duration DNB/dryout events. This would supplement data from the Halden Reactor experiments.
- Experiments and empirical correlations for understanding annealing recovery effects on embrittlement and corrosion resistance based on changes to SPPs in zirconium-based cladding materials.
- For fast transients, additional in-pile transient testing in facilities like the INL TREAT test reactor would provide valuable insight in the actual temperature response of the fuel and cladding materials during AOO scenarios that are the most likely applications of the criteria. In-pile testing of short duration DNB/dryout would strengthen the known validation data sets for models that may potentially be used to predict the effects of operation under post-DNB and post-dryout conditions.

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