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POST-CHF NUCLEAR FUEL OPERATION WITH TIME-AT-TEMPERATURE CRITERIA

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Abstract. The dryout and departure from nucleate boiling criteria are used to set safety limits on nuclear heat transfer and temperature values in nuclear reactors. This review argues for the use of a Time-at-Temperature criterion to improve efficiency and utilization of nuclear fuels and reduce costs. To illustrate the importance of such criteria, the review presents the different phenomena of cladding and fuel rods that could impact the continued operation of nuclear reactors, as well as the experiments and operating events that occurred in light-water reactors.

Keywords: Time at Temperature, Critical Heat Flux, Departure from Nucleate Boiling, dryout, nuclear fuel.

1. Introduction

This document reviews critical safety considerations in nuclear reactors, focusing on the use of time-at-temperature (TaT) strategies to maintain both cladding and pellet integrity, avoiding degradation phenomena after Departure from Nuclear Boiling (DNB) or dryout events. Rather than using the conservative thermal limits and margins currently set by the regulators, which treat Dryout and DNB with significant margins as operational limits, it proposes a more flexible TaT criterion, which could improve fuel utilization, reduce the volume of waste, and offer significant economic benefits. The review details various degradation phenomena that might occur during DNB or Dryout events, such as cladding creepdown or embrittlement. It also reviews experimental tests, calculations, and operating events, providing real-world data to support theoretical discussions. The IFA test series performed within the Halden reactor project is explained in more detail. In general, the review advocates for an evolution in safety margins based on empirical understanding rather than conservative assumptions, with the aim of optimizing reactor performance without compromising safety.

2. Critical Heat Flux

When a reactor faces sudden loss or reduction in cooling capabilities or a local increase in heat transfer above a certain value, the heat flux reaches a value known as the Critical Heat Flux (CHF). This phenomenon might result in an increase in the local temperature and accelerated degradation of the fuel rod. As a result, thermal design limits expressed in terms of dryout in Boiling Water Reactor (BWR) and DNB in Pressurized Water Reactors (PWR) have been established [1]. The current operational envelope does not allow DNB to occur during normal operation or Anticipated Operational Occurences (AOOs). Under normal conditions, there is always a thin layer of coolant around the cladding in both types of reactors. However, due to an increase in heat flux, the coolant around the cladding surface vaporizes (nucleate boiling), and then the bubbles start to coalesce (transition boiling), finally resulting in the formation of a thin layer of vapor around the cladding (film boiling) [2]. The vapor film acts as an insulator, significantly reducing the heat transfer coefficient and local heat-removal capabilities. Therefore, as heat transfer declines, the temperature increases, which can lead to fuel or cladding failure through different phenomena [2].

CHF is reached between Nucleate Boiling and Transition Boiling, at the point where the heat transfer coefficient locally reaches its highest value as shown in Figure 1. After the Leidenfrost point, the system cannot be rewet as the cladding temperature is too high. This is the point where the vapor layer stabilizes. Heat transfer is now mainly induced by radiation rather than convection as before, further decreasing the effective heat transfer coefficient [2].

3. CPR and DNBR

What are the actual margins of the CHF in practice? In both BWR and PWR, a design limit has been set under which the integrity of the fuel rod is threatened [1]. A parameter was defined for each type of reactor system: Critical Power Ratio (CRP) in BWRs and DNB Ratio (DNBR) in PWRs. These ratios must remain above a minimum allowable value of 1.0 to maintain the integrity of the fuel cladding and prevent damage [1], but substantial margins are included in the analytical limits due to the high uncertainties

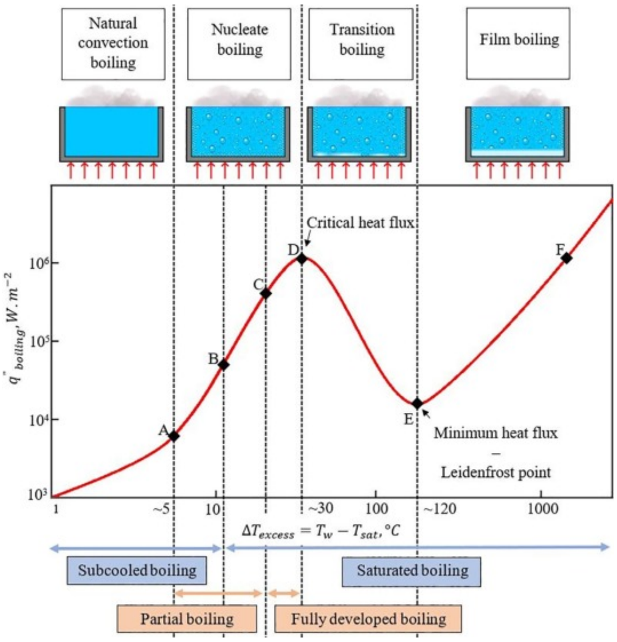


Figure 1. Water pool boiling curve [3].

related to the knowledge of local power and actual CHF values at given operating conditions.

The analytical dryout limit is represented in terms of the critical power ratio because it depends on channel thermal hydraulic history, whereas DNB is represented in terms of the heat flux ratio (DNBR) as it is a local condition [1].

$$\mathrm{CPR} = \frac{\text{predicted correlation power}}{\text{actual operating power}} \tag{1}$$

$$\mathrm{DNBR} = \frac{\text{predicted correlation heat flux}}{\text{actual operating heat flux}} \tag{2}$$

4. Time at Temperature

Currently, the US NRC uses the CPR and DNBR parameters to set thermal limits that should not be exceeded to ensure the preservation of the cladding and the fuel rod [4]. The main issue with choosing the DNB point as the margin that cannot be exceeded is that the fuel rods have not been utilized to their full potential. It leads to a potential waste of raw material or underutilization of fuel. This underuse of fuel has significant economic and environmental costs. The more fuel bundles that need to be used, the more irradiated assemblies that will need to be stored and disposed of afterward.

A TaT criterion would enable the current margins to be adjusted and operating the LWRs at a higher local power than the conservative limits used today. When the allowable times and temperatures are established, they will allow operation in mild DNBs or dryouts without affecting the reactor operation and further utilization of those fuel rods. Having this criterion would also accelerate return to operation after an event, reducing the financial impact of restarting a plant after CHF [5].

In BWRs, it is estimated that a TaT could be used in 75 % of the BWR fleet, saving an average of $ 1 M per cycle. In PWRs, TaT could increase the power output by 5 %, resulting in a saving of $ 0.5 M per plant per year. Given that there are 94 BWRs and 304 PWRs worldwide, the potential economic gains of using TaT in our light water reactors are clear [5].

The TaT criteria are more likely to be applied to AOOs, which have a short-duration power increase with a peak cladding temperature that is usually under 800 °C ; as well as an extremely short-duration accident caused by reactivity of the power, as the temperature quickly returns to normal during these events, but there are also significant benefits for PWRs [4].

5. Cladding and fuel phenomena

CHF can cause deformation phenomena that can damage the cladding or fuel rod, which is the main barrier between the fission products and the coolant. These phenomena have not yet been studied in detail; therefore, before establishing the new TaT limit, it is required to better understand them, observe how they interact, and determine how to operate post-CHF while ensuring the conservation of the fuel rods [5].

5.1. Power-Cooling Mismatch

CHF mostly occurs due to a reduction or loss of cooling (with loss of power or at constant power), or an excessive overpowering. These events are known as power-cooling mismatches and can occur simultaneously [6].

5.2. Recovery from irradiation hardening

Irradiation can harden the Zircaloy cladding and cause dislocations in its crystallographic structure. Heating the cladding to a temperature that exceeds its normal operating range can cause it to locally recover its initial material and mainly mechanical properties. This recovery could result in an accelerated creep and localized deformations resulting in cladding collapse during operation, an accelerated creepout during the drying of the fuel rod before the storage of spent dry fuel, or a change in cladding geometry in general [5].

5.3. Cladding creepdown and collapse

The increase in temperature after CHF leads to the reduction of mechanical strength, which is temperature dependent. This drastically reduces the creep strength of the cladding, that is, its ability to resist deformation or rupture over a long period. As a result, the cladding tends to collapse onto the pellets and the chamfer. Consequently, the cladding is bonded to the fuel, and the dimensional movement of every pellet is transferred to the cladding, leading to high stress that can cause considerable damage to both the cladding and the pellet. After the interaction

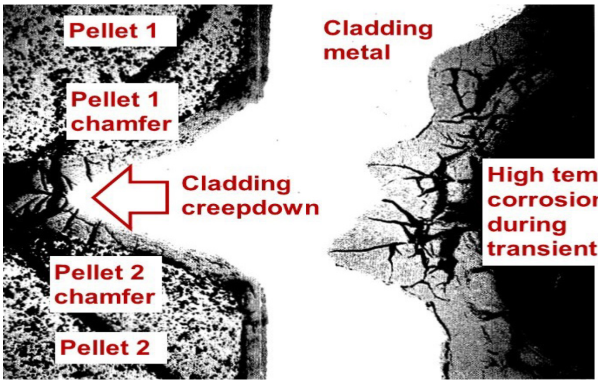


Figure 2. Collapse of the cladding into the pellet chamfers [5].

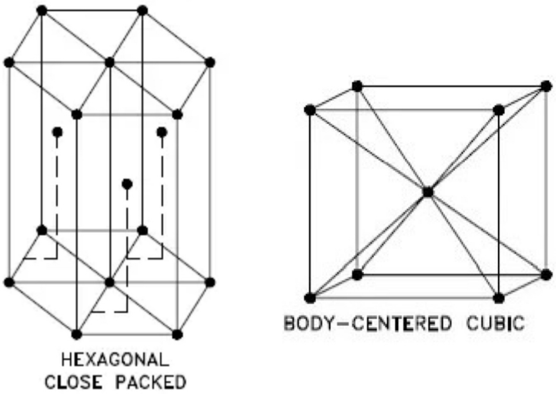


Figure 3. Crystallographic structure of α-Zr and β-Zr [8].

between the pellets and the cladding, the local wall thinning [5]. The interaction of the cladding and fuel after creepdown is shown in Figure 2.

5.4. ALPHA-BETA TRANSITION

The α-β transition is a change in the crystallographic structure of the cladding (Figure 3). The material can transition from the α phase to the α+β mixed phase and to the β phase as the temperature increases. Although a decrease in temperature can reverse the transition, there will still be a percentage of the β-phase material in the cladding. Once the α to α+β transition has occurred, the cladding material may not be suitable for continued operation, as the transition can affect the microstructure and mechanical properties [7].

5.5. Cladding corrosion and embrittlement

When Zircaloy cladding oxidizes, mostly due to irradiation-assisted corrosion (Figure 4) or steam oxidation, hydrogen is absorbed into the cladding, decreasing its ductility. Once a certain amount of hydrogen has been absorbed, hydrogen precipitates in the form of hydrides and the cladding starts to become brittle and is therefore more likely to fail. Hydrides form in cooler regions and also regions with higher stress. These hydrides can be redissolved at very high temperatures, leading to a cycle between hydrogen precipitation and hydride dissolution. Both processes contribute to a loss in cladding ductility and therefore to cladding embrittlement [5].

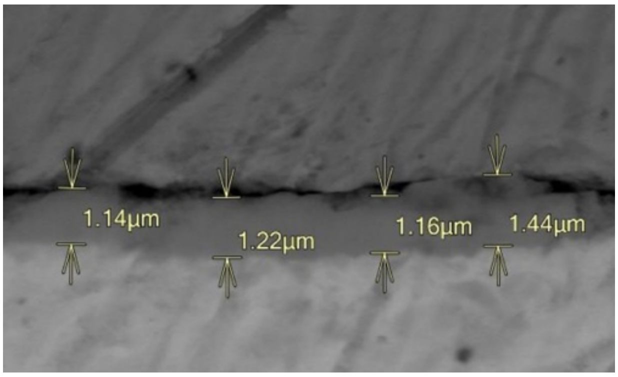


Figure 4. Corrosion of the cladding [9].

Figure 5. Rod bowing with rod-to-rod contact [5].

5.6. Rod bowing

During normal operation, it is possible to experience a rod bow with rod-to-rod contact. Although it does not cause cladding failure, it can locally disrupt the flow path and heat transfer, resulting in a very localized CHF with minimal impact. It can also result in a local increase in temperature and corrosion in the contact region, leading to wall thinning [5]. An example of rod bowing with a contact is shown in Figure 5.

5.7. Pellet-Cladding mechanical interaction

During normal operation, the pellet swells mainly due to the accumulation of fission products, resulting in its expansion and closure of the original gap between the pellet and the cladding, resulting in contact. As the temperature increases, the pellet expands

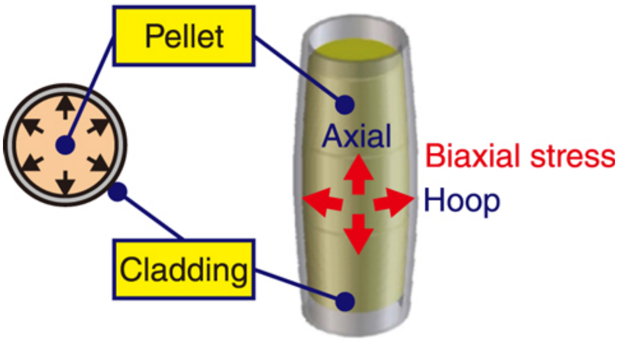


Figure 6. Internal forces onto the cladding imposed by expanding pellet [10].

further, exerting internal forces on the cladding surface, leading to a potential rupture and to cladding failure. These interactions can be exacerbated in a corrosive environment, as the cladding undergoes much higher mechanical stress [5]. The inner loading of the cladding by an expanding pellet is shown in Figure 6.

5.8. Conclusion of Section 5

All of the deformation criterion cited above can affect the fuel and cladding integrity if the heat flux reaches the CHF. Therefore, as we argue for the use of a TaT criterion to improve LWR productivity, it is important to take special care to detect every deformation criterion, as the temperature used in the reactors could exceed the usual operating temperature, even for a short amount of time.

6. Review

In the History of Nuclear power plants, scientists overview different scenarios where a reactor operated post-CHF, wether it was wanted in the experiments or due to an accident, and faced a deformation criterion without impacting the continuation of operations. It proves that it is possible to operate at a higher temperature while ensuring the reactor integrity, giving credit to the use of a TaT criterion in LWRs.

6.1. Calculations

6.1.1. PWR locked rotor

In a PWR, a reactor coolant pump seizure could potentially lead to DNB. The nuclear power remains high for around 3 seconds, exceeding the initial fuel power. However, the coolant flow rate rapidly decreases, leading to DNB within a fraction of a second. The inner cladding continues to overheat for 10 seconds, reaching a maximum temperature of more than 1 000 °C, as shown in Figure 7 [5, 11].

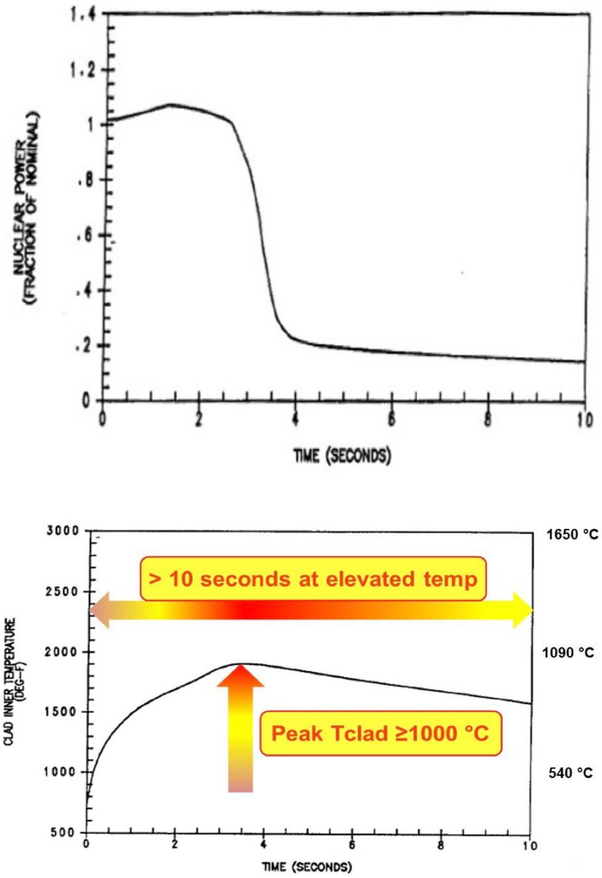


Figure 7. Calculated reactor power and peak cladding temperature in the PWR locked rotor [5, 11].

6.2. Tests

6.2.1. Dryout tests in Halden test reactor

Three series of tests were conducted at the Halden reactor within IFA-613 experiments (Instrument Fuel Assembly) to provide information on the consequences of short-term dryout incidents for BWR fuel. The coolant flow was restricted at a more or less constant power until the cladding reached the desired temperature, then the system was quenched. The tests consisted of three loadings: the first loading consisted of 1 fresh fuel rod and 2 pre-irradiated non-barrier segments; the second loading consisted of 1 non-barrier segment and 2 barrier fuel segments; the third loading consisted of 1 fresh rod and 1 barrier fuel segment. A barrier segment is a part of a nuclear pellet containing an inner layer of zirconium metal, in addition to the cladding. It helps prevent stress corrosion cracking. Such segments are recommended in BWRs. A non-barrier segment does not have the additional Zirconium metal layer [12]. Each rod was contained within a flow channel (channel A, B, or C), whose purpose was to allow different flow transients to be implemented for each of the three rods [4, 5].

6.2.2. BWR pressure transient

A pressure increase was tested in the Peach Bottom 2 BWR, leading to a void collapse and therefore to an increase in local power. It was started by tripping a turbine, but CHF or dryout was not reached, due to slow heat release from the pellet. They briefly pulsed the power up to 450 % of full power and then

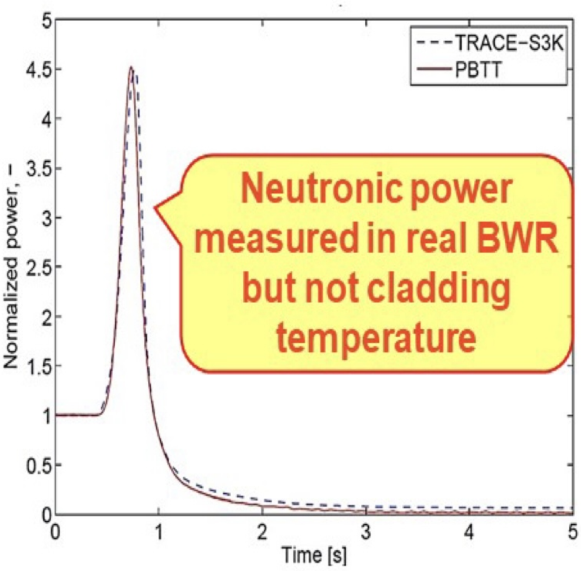


Figure 8. Measured power and reactor pressure in the Peach Bottom 2 reactor [5].

immediately decreased it, which is a typical reactivity-initiated accident. As the temperature of the cladding in such a transient cannot be measured, an out-of-pile simulation was used to determine the cladding response. Note that the cladding temperature transient is longer in time than the power transient because of the heat transfer. However, this result may differ from a transient real fuel cladding, as in the experiment the cladding is electrically heated [5]. The measured power and reactor pressure are shown in Figure 8.

6.3. Operating events

6.3.1. BWR recirculation pumps trip

In 2008, the Forsmark 2 BWR experienced rapid shutdown of all of its internal recirculation pumps, due to a disturbance of the electrical grid caused by a lightning strike. The core flow decreased from around 105 000 $\mathrm{kg\,s^{-1}}$ to approximately 2 800 $\mathrm{kg\,s^{-1}}$ within about one second, resulting in an increase in the void fraction. The negative void feedback resulted in a 20 % decrease in power, causing the pellet temperature to fall below its normal range of temperature in steady-state operation. The few seconds needed for heat to transfer from the pellet to the coolant caused 84 fuel assemblies to fall below SLMCPR, and 18 of these assemblies briefly experienced a dryout during which the cladding and pellet temperature exceeded their initial temperatures, with the inner surface of the cladding reaching 460 °C at most during one second according to the BISON/SLAVE Code calculations [13]. However, there was no creep of the cladding or recovery from the irradiation hardening caused by dryout. All fuel assemblies that experienced dryout were reloaded into the core and operated until their assumed end of life [4, 5]. The temperature history reconstruction of the event done using the BISON code is shown in Figure 9.

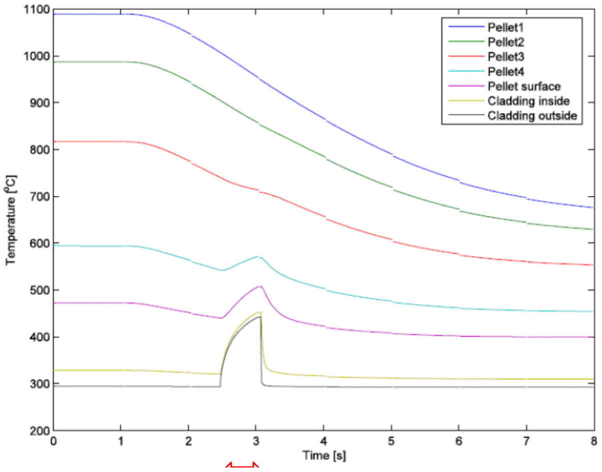


Figure 9. Temperature calculated at different radial positions based on the BISON/SLAVE Code calculations in the Forsmark 2 BWR [5].

6.3.2. Oskarshamn 2 dryout event 1987–1988

The only known dryout that has ever occurred during normal operation happened in the Oskarshamn 2 BWR. Some fuel rods experienced much higher power than expected due to systemic errors that had not been correctly accounted for. This resulted in an unnoticed dryout during normal steady-state operation, which was discovered during refueling. After CHF, the reactor continued to operate for at least six months until its scheduled termination, despite four failed rods (all corner rods in different assemblies). Some other intact rods experienced localized overheating, resulting in significant creepdown of the cladding. This event served as an unintentional experiment of continued operations after dryout [4, 5].

7. Halden reactor IFA 613

Based on the explanation about the Halden reactor made in Section 6.2.1, it was seen that the reactor experienced dryouts with a maximum temperature of 950–1200 °C for the first two tests and 750–850 °C for the final test. The fuel rods continued further operations for about a month after dryout transients. With respect to potential deformations, there was extensive creep and collapse of the cladding on the pellets and the chamfers of the cladding, as well as cladding oxidation when the cladding had reached significant temperatures. However, there were no cladding failures [4, 5]. The measured temperature history of selected IFA-613 tests is shown in Figure 10.

There were only two to three thermocouples on each of the rods of the Halden IFA 613 experiment. It is not enough to have a complete understanding of the behavior of the rods during the tests. The Halden Legacy Database, recently released by the Nuclear Energy Agency, contains all reports, drawings, and data regarding the experiments, such as power history and fuel rod and cladding properties. The FAST code [15] is used to calculate the fuel performance

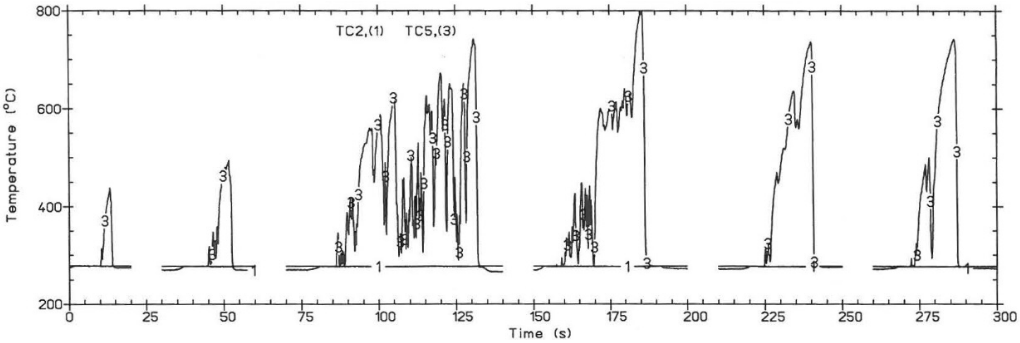
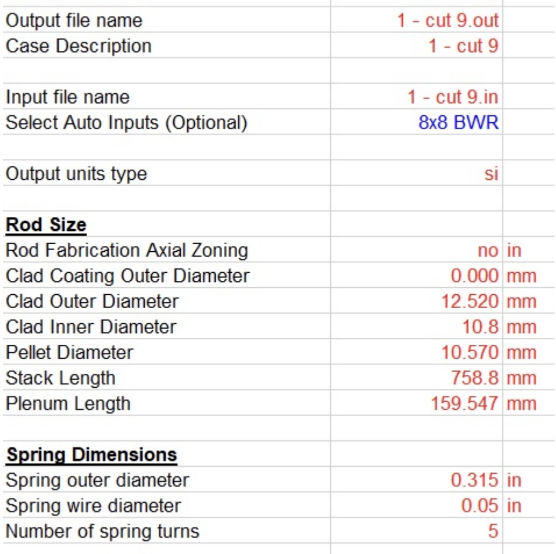


Figure 10. Dryout tests – 29/30 October 1996 – Channel B second loading [14].



Output file name	1 - cut 9.out	
Case Description	1 - cut 9	
Input file name	1 - cut 9.in	
Select Auto Inputs (Optional)	8x8 BWR	
Output units type	si	
Rod Size		
Rod Fabrication Axial Zoning	no	in
Clad Coating Outer Diameter	0.000	mm
Clad Outer Diameter	12.520	mm
Clad Inner Diameter	10.8	mm
Pellet Diameter	10.570	mm
Stack Length	758.8	mm
Plenum Length	159.547	mm
Spring Dimensions		
Spring outer diameter	0.315	in
Spring wire diameter	0.05	in
Number of spring turns	5	

Figure 11. Extract of one of the Excel window of the FAST Code, where the data are implemented.

during the experiments. The input files were created by implementing the data extracted from the report file (Figure 11). The plot files processed using the FRAPlot tool will allow the creation of interpretable graphics of the behavior of the cladding and rods from the beginning to the end of the experiment.

8. Future work regarding TaT criterion

Based on observations of the calculations, experiments, and operating events that happened in LWRs, scientists' interest for a TaT criterion is increasing. About TaT criterion, CTU will perform in cooperation with Karlsruhe Institute of Technology (KIT) post-CHF tests to support the establishment of the TaT criterion using KIT's COSMOS loops. The test matrix was designed to address gaps in current material-related research such as those mentioned above. The main phenomena on which the test would focus on are the α-β phase transition, cladding embrittlement and corrosion, and cladding creepdown and collapse. There is a Test Matrix for each of these phenomena, detailing the tests that need to be conducted to fill the gaps in actual experiments. Several tests have indeed been conducted, but the relevant existing experimental database is still being reviewed. These data have to be compared with the data required for post-CHF licensing activities and establish which important gaps still need to be addressed before creating a test matrix on the matter.

Therefore, for now, the focus should be on the α-β phase transition and the cladding creepdown and collapse tests. It will provide a basis for the integrity of the cladding and the spent fuel handling criteria, as well as information on the potential benefits of using coated cladding.

9. Conclusion

This review summarizes various events and experiments related to post-CHF nuclear fuel operation and allows a better understanding of the need for the TaT criterion and its impact on the operation and performance of nuclear reactors. But how could such a criterion be implemented? It must be applied to every unstudied phenomenon that could occur in BWRs and PWRs to ensure cladding and pellet integrity in all situations.

The conduct of several experiments at KIT will allow the support of implementation of such criteria, focusing either on the conditioning parameters (irradiation, temperature, and time conditions) or on the testing parameters (the parameter of the phenomenon under investigation) until a new margin could be defined. These tests must be conducted on both BWRs and PWRs geometries and conditions separately, as testing priorities differ in these reactors due to variations in transient response.

The use of a TaT criterion would benefit different technical aspects, such as experiments focusing on the effect of repeated dryout/rewet cycling and its effect on material properties, or studies examining repeated excursions into post-DNB and post-dryout operation over longer timescales, and much more.

List of abbreviations

AOOs	Anticipated Operational Occurrences
BWR	Boiling Water Reactor
CHF	Critical Heat Flux

CPR Critical Power Ratio
DNB Departure from Nucleate Boiling
DNBR Departure from Nucleate Boiling Ratio
FAST Fuel Analysis under Steady-state and Transients
IFA Instrument Fuel Assembly
KIT Karlsruhe Institute of Technology
LWR Light Water Reactor
NRC Nuclear Regulatory Commission
PWR Pressurized Water Reactor
SLMCPR Safety Limit Minimum Critical Power Ratio
TaT Time at Temperature

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