CONF-800403--28

ON THE DETERMINATION OF POST-DNB AND POST-BT FUEL DESIGN LIMITS*

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ABSTRACT

Categories of light water reactor transients and the departure from nucleate boiling (DNB) and boiling transition (BT) fuel design limits in light water reactors are reviewed. These fuel design limits for reactor licensing may be overly conservative because experiments have shown that fuel rods do not fail and may not experience damage as a result of momentary operation in film boiling or dryout conditions. Damage to the fuel rod is strongly dependent on the peak cladding temperature and the length of time at that temperature during the transient. Testing of two potential licensing fuel design limits is suggested: (a) fuel rod functional capabilities are retained and fuel system dimensions remain within operational tolerances; and (b) cladding deformation is permitted, but no significant oxidation is allowed. Damage mechanisms which may affect post-DNB or post-BT operation of fuel rods are permanent rod bowing and pellet-cladding interaction. The data necessary to support a fuel design limit and a means of obtaining these data are outlined.

INTRODUCTION

Light water reactors (LWRs) are designed and operated such that the fuel rods do not depart from nucleate boiling (DNB) or experience boiling transition (BT) during normal operation or anticipated operational occurrences. This fuel design limit is conservative because experiments, reviewed in References 1 and 2, have shown that fuel rods do not necessarily fail as a result of operation in film boiling or dryout conditions. This criterion protects the fuel rods from damage due to overheating during anticipated operational occurrences, but it is not a mechanistic limit in terms of fuel rod damage or failure. This is recognized in the Standard Review Plan[3] which states that other mechanistic methods may be acceptable. By replacing the current DNB and BT limit with mechanistic fuel design limits, the true thermal margin could be more precisely defined and the limit placed on a more realistic, "best estimate" basis. Further, it is expected that benefits could be obtained by increasing reactor power, thermal efficiency, or operating flexibility. The purpose of this paper is to review those anticipated transients which limit plant operation, briefly examine fuel rod damage mechanisms, and suggest potential post-DNB and post-BT fuel design limits and how they may be determined.

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*Work supported by the U. S. Department of Energy Assistant Secretary for Nuclear Energy, under DOE Contract No. DE-AC07-76ID01570.

NUCLEAR POWER PLANT TRANSIENTS

Nuclear power plant conditions are divided into four categories by the American National Standards Institute[4] in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows: (a) Condition I - Normal Operation; (b) Condition II - Incidents of Moderate Frequency; (c) Condition III - Infrequent Incidents; and (d) Condition IV - Limiting Faults. Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, and maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Condition II occurrences include "incidents, any one of which may occur during a calendar year for a particular plant"[4] and are classified as "anticipated" transients. By definition, these events do not result in fuel rod damage or failures or reactor coolant system overpressurization. "Not damaged" means that the rods do not fail, that functional capabilities are retained, and that the fuel system dimensions remain within operational tolerances. Condition III occurrences include "incidents, any one of which may occur during the lifetime of a particular plant."[4] Only a small fraction of the fuel rods may fail although sufficient fuel damage might occur to preclude resumption of power operation for a considerable period. "Failure" means that the fuel rod hermeticity is lost and that the first fission product barrier (the cladding) has been breached. Condition IV faults are not expected to occur, "but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material."[4] The present DNB and BT fuel design limit serves as a damage threshold for Condition II events, and a failure threshold during Condition III and IV events.

Condition II events can be summarized into two types: (a) power ramps; and (b) coastdowns in primary coolant flow. Of the overpower transients in pressurized water reactors (PWRs), the DNB limit is most closely approached during an uncontrolled rod withdrawal at power and an increase in feedwater flow. Depending on plant design, peak cladding surface heat fluxes reach 109%[5] of the nominal operating heat flux over a period of about 30 s. The BT limit on boiling water reactors (BWRs) is most closely approached during the overpower transients resulting from an electric load rejection or a turbine trip. Peak surface heat fluxes of up to 112% are achieved over a period of 2 s.[6] One-pump and total coastdowns of primary coolant flow are also among the limiting transients for both PWRs and BWRs. If a post-DNB or post-BT fuel design limit were implemented, the most severe events may cause the cladding to experience momentary (5-20 s) excursions to temperatures, based on measurements[2] during dryout on BWR rods and on PWR licensing calculations[5] for Condition III events, that range from approximately 800 to 1300 K.

FUEL ROD DAMAGE MECHANISMS

During Condition II transients, fuel rod behavior can be characterized in terms of cladding temperature, the parameter which most strongly affects fuel rod integrity. For those transients in which fuel rod damage is significant, the damage mechanisms may include permanent rod bowing, pellet-cladding interaction, cladding annealing, cladding deformation, and cladding embrittlement. The question with each of these mechanisms is, "Given that DNB or BT occurs, what effect will the peak cladding temperature have on rod performance during the transient and on its subsequent operation?"

Fuel Rod Bowing

Due to the potential for circumferential temperature gradients in the cladding, rod bowing can be assumed to occur during film boiling or dryout at any cladding temperature. Rod bowing is expected to be transitory when cladding temperatures are below 920 K; some permanent rod bowing is expected when cladding temperatures exceed 920 K.

Bowing during the transient would affect the post-DNB or post-BT cladding surface heat transfer coefficient. Groenveld[7] has determined that reductions in the pitch-to-diameter ratio decrease the surface heat transfer coefficient and, consequently, increase the cladding temperature at the rod-to-rod gap in dryout conditions. The effect of rod bowing on post-DNB heat transfer at the lower qualities typical of PWR conditions is also expected, but little or no data exist.

If permanent rod bowing occurs as a result of film boiling or dryout, the critical heat flux (CHF) may be reduced for subsequent operation. Out-of-pile bundle tests at PWR conditions indicate that CHF is unaffected by rod bowing unless the clearance between adjacent heated rods is reduced by more than 50%.[8] A heated rod bowed to simultaneously contact two adjacent heated rods reduces CHF by an amount dependent on pressure and heat flux.[8,9] In the bubbly flow regime (low quality), the penalty is more severe than in the annular flow regime.[8,9,10] Correlations for CHF penalty due to rod bowing are available in terms of pressure and rod average heat flux.[8,9] When an unheated rod is bowed to contact two adjacent heated rods, no reduction in CHF is indicated.[8] A heated rod bowed to contact a guide tube is also not expected to reduce CHF.

Pellet-Cladding Interaction

A very small percentage of commercial reactor fuel rods have failed during Condition I operation due to brittle fracture of the cladding. These Pellet-Cladding Interaction (PCI) failures are due to a combination of localized pellet-induced strain resulting from power increases, a reduction in the strain-to-failure capability due to irradiation, and chemically aggressive fission products generated and released as vapor to the fuel-cladding gap from those portions of the fuel pellets that operate at high temperatures. The probability for occurrence of a PCI failure generally increases with increasing power, the rate and amount of the power increase, time at high power, and burnup.[11] Fuel temperature is a controlling parameter since it effects the stress on the cladding due to differential fuel-cladding thermal expansion and the release of aggressive fission products to the fuel-cladding gap. It has been concluded[12] that the chemical species most aggressive to zircaloy are iodine, cadmium, and cesium and that crack formation is the critical step in this failure mechanism. Irradiation effects tend to predominate over differences in initial microstructure, surface finish, temperature, or alloy composition of zircaloy.

In contributing to the PCI failure mechanism, the magnitude of the stress, a mechanical process, and the availability of aggressive fission products, a chemical process, appear to be reciprocal.[11] During a brief transient which increases the fuel temperature, the stress in the cladding may or may not be sufficient to produce cladding failure. However, fission products released from the fuel to the fuel-cladding gap as a result of the transient may subsequently enhance PCI during normal operation.

Cladding Annealing

During transients which generate cladding temperatures above 750 K, partial annealing of the initial cold-working of the zircaloy and the accumulated irradiation damage is expected. The yield strength of irradiated cladding following isothermal annealing is illustrated in Figure 1.[13] Recrystallized alpha-zircaloy is expected to form at temperatures above 800 K and contribute to the annealing process. Based on the data of Figure 1, it is estimated that the yield strength of the annealed cladding drops below the value of two-thirds the yield strength of the unirradiated cladding, which corresponds to the stress intensity limit for zircaloy cladding, if a temperature of about 900 K is maintained for 20 to 30 seconds.

Cladding Deformation

Cladding in the temperature range of 920 to 1100 K becomes ductile. Depending upon the differential pressure across the cladding and the period of time the cladding is at these elevated temperatures, the cladding may buckle, collapse uniformly onto the fuel column, waist (flow into cracks or gaps in the fuel column such as pellet-to-pellet interfaces), or balloon to rupture.

Relevant information on the dependence of cladding collapse upon temperature, pressure, and time has been obtained from out-of-pile tests and Test LOC-11[14] conducted in the Power Burst Facility (PBF). During Test LOC-11, a test to determine fuel rod behavior during a hypothetical loss of coolant accident (LOCA), two fuel rods had negative (compressive) differential pressures of 3.1 and 8.8 MPa across the cladding while the cladding was at high temperatures. On the basis of cladding microstructures at the ends of the zone of cladding collapse, incipient cladding deformation was found to occur at a temperature of about 920 K. The data suggest that measurable plastic strain is coincident with partial recrystallization of the stress-relieved alpha-zircaloy. Figure 2, based on isothermal out-of-pile tests lasting fifteen seconds,[15] shows the cladding temperatures required for cladding buckling, collapse, and waisting at different differential pressures across the cladding. Similar tests at cladding temperatures of 1144 and 1172 K at negative differential pressures of 6.9 and 4.8 MPa, respectively, indicated that cladding collapse and waisting may occur within two seconds.[15]

From this experimental evidence, it is concluded that incipient cladding collapse occurs at about 920 K at differential pressures that are expected during an operating cycle of a LWR fuel rod. The time required for collapse is of the same magnitude as the expected duration of film boiling or dryout during limiting Condition II transients if operation is not limited by the present DNB or BT criterion. The effect of collapsed cladding on the subsequent operation of a LWR rod is not well known. Recent data from the PBF suggest that fuel rods subjected to a collapsing event (in this case a LOCA) operate with lower temperatures during subsequent steady-state operation and successive LOCA events. Thus, one could speculate that fuel with collapsed cladding may actually be less susceptible to stress-corrosion cracking (SCC) induced PCI. Annealed cladding may be less susceptible to SCC, but collapse and waisting of the cladding will enhance stress concentrations in the zircaloy.

Should the internal pressure of the rod exceed the coolant pressure, cladding ballooning may occur. In Test LOC-11, incipient cladding ballooning occurred at about the same temperature as incipient cladding collapse, 920 K.[14] The cladding may balloon until rupture occurs. The dependence of the cladding burst temperature on the differential pressure across the cladding is shown in Figure 3.[16]

In the temperature range between about 1100 and 1255 K, the zircaloy is in the alpha-plus-beta phase. Cladding at these temperatures is less ductile. Collapse or ballooning of the cladding may occur, but the strain-to-failure will be relatively small.

Cladding Embrittlement

When cladding temperatures exceed 1200 K, embrittlement of the cladding due to oxidation by zircaloy-H_2O and zircaloy-UO_2 reactions can become significant. As oxygen diffuses into both surfaces of the cladding, the zircaloy undergoes a metallurgical phase transformation from the beta phase to the oxygen-stabilized alpha phase. The extent of cladding oxidation, and thus embrittlement, depends upon the cladding temperature and time of operation at temperatures greater than 1200 K. Highly oxidized cladding fractures upon quenching and during handling due to the brittle nature of the ZrO_2 and oxygen-stabilized alpha-zircaloy. Criteria to define the onset of cladding embrittlement are, therefore, based on behavior at room temperature.

In-pile data from regions of cladding fracture and maximum oxidation are compared with the Pawel criteria[17] in Figure 4 for various cladding temperatures and durations of film boiling achieved during testing.[18] The comparison indicates that the embrittlement of the zircaloy-clad rods tested under film boiling conditions is consistent with room temperature embrittlement criteria based on oxygen content (distribution) in the beta-phase zircaloy, with the exception of fuel rods which operated in film boiling with breached cladding. The rods with breached cladding picked up relatively large amounts of hydrogen (340-1000 ppm), probably due to a stagnant steam atmosphere[19] inside the rods, thus producing embrittlement in excess of that caused by oxygen alone.

POTENTIAL FUEL DESIGN LIMITS AND REQUIRED DATA

Two potential fuel design limits are suggested by the above cladding behavior: (a) fuel rod functional capabilities are retained and fuel system dimensions remain within operational tolerances (that is, the cladding temperature shall not exceed, for example, 870 K for 20 s); and (b) deformation such as collapse is allowed, but no significant oxidation is permitted (that is, the cladding temperature shall not exceed, for example, 1400 K for 100 s). The first criterion is a fuel rod damage threshold based on cladding annealing that applies to Condition II events. The primary concern is that if DNB or BT occurs on some fuel rods during a Condition II event, those rods should be able to safely operate to the end of their current cycle. They can then be inspected during the normal refueling period. The design limits may take the form of a statistical distribution, perhaps assuring with a 95% confidence that none of the fuel rods that were subjected to DNB or BT operation will breach prior to reaching the end of the fuel cycle in which the transient occurred. The second criterion is a failure threshold based on cladding embrittlement that applies to Condition III events. Adoption of this or a similar failure threshold would reduce the number of fuel rods currently calculated to "fail" during a Condition III event.

Data necessary to determine and support post-DNB and post-BT fuel design limits are required in the areas of post-DNB heat transfer, propagation of

DNB, and fuel rod performance following film boiling and dryout. Cladding surface heat transfer correlations in the film boiling regime are available, but, due to the small data base, their uncertainties are large. A reasonable data base exists for post-BT heat transfer[7]. Tests must show that film boiling or dryout does not propagate so that any film boiling or dryout that may occur during an event is restricted to a small number of channels in the core. It is also necessary to subject rods which have experienced DNB or BT to additional normal operation, power ramps, and limiting Condition II events to determine whether or not the rods are any more susceptible to failure. The primary factors which may affect fuel rod behavior during post-DNB or post-BT operation are PCI and rod bowing. Under the first design limit, PCI during and following the event may be enhanced, however, the data are insufficient to make this judgement. Tests on irradiated fuel rods of moderate burnup are essential for evaluating any increased propensity for PCI damage and whether or not it may be enhanced by fission product release or by cladding collapse. It is necessary to show that the rods may be safely operated to the end of their current fuel cycle. It must also be verified that the degree of permanent rod bowing that may occur does not affect the coolability of the fuel rod. More data are required to understand the behavior of unirradiated and irradiated zircaloy cladding annealed for brief periods (5-20 seconds) at temperatures in the range of 750 to 950 K. Tests on high burnup fuel assemblies are not required since it is unlikely that they will experience film boiling or dryout due to their low operating power.

CONCLUSIONS

To precisely define the true thermal margin in LWRs and place the fuel design limits on a "best estimate" basis, it is recommended that tests be performed to assess two potential mechanistic fuel design limits that are based on cladding annealing and embrittlement. They are: (a) that fuel rod functional capabilities are retained and that fuel system dimensions remain within operational tolerances; and (b) that cladding deformation be permitted, but that no significant oxidation be allowed. A large bundle in-pile test program to evaluate the post-DNB and post-BT fuel rod performance is necessary to properly assess the PCI damage mechanism and develop a thermal-hydraulic data base that will lead to the development of appropriate statistical damage criteria.

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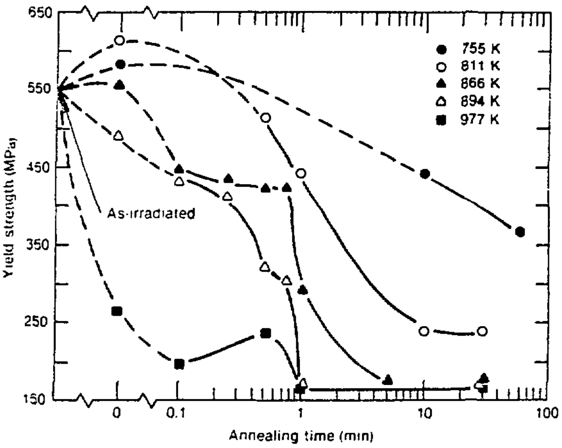

Fig. 1 Yield strength of irradiated zircaloy-4 cladding as a function of annealing temperature and time. (Specimens with a fluence of 4.2 X $10^{[illegible]}$ n/cm^2 1 MeV were heated at a rate of 28 K/s to the annealing temperature, held at that temperature for the annealing time shown, and quenched. The yield strength of the annealed material was obtained from tensile tests conducted at 644 K and at a strain rate of 0.025/min.) (From Ref. 13)

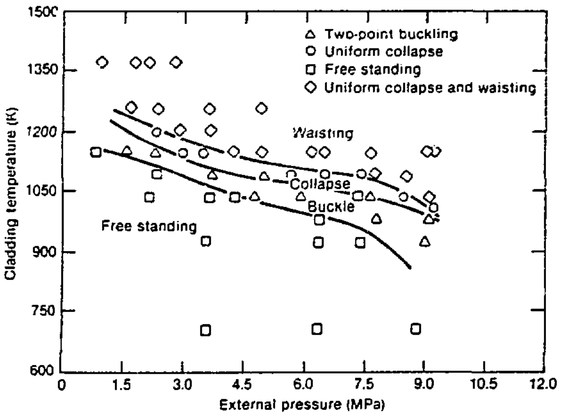

Fig. 2 Modes of cladding deformation at different pressures and temperatures, maintained for 15 s with the cladding internal pressure at 0.1 MPa. (From Ref. 15)

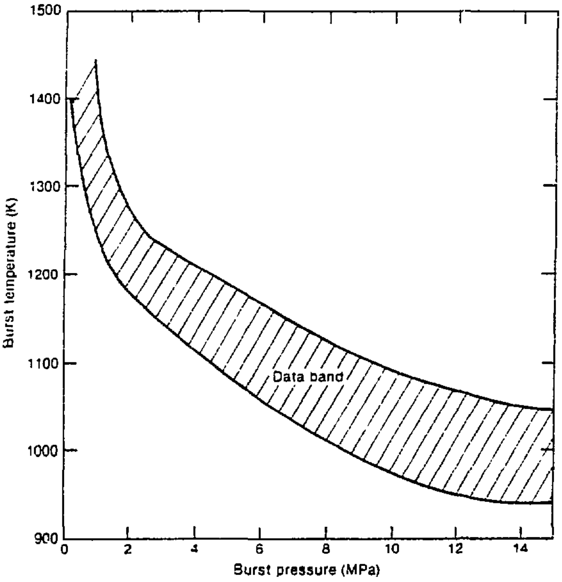

Fig. 3 Summary of data describing dependence of cladding burst temperature upon burst pressure. (The data was obtained from tests on unirradiated cladding and cladding from rods irradiated to 35 MWd/kg heated at a rate between 5 and 115 K/s.) (From Ref. 16)

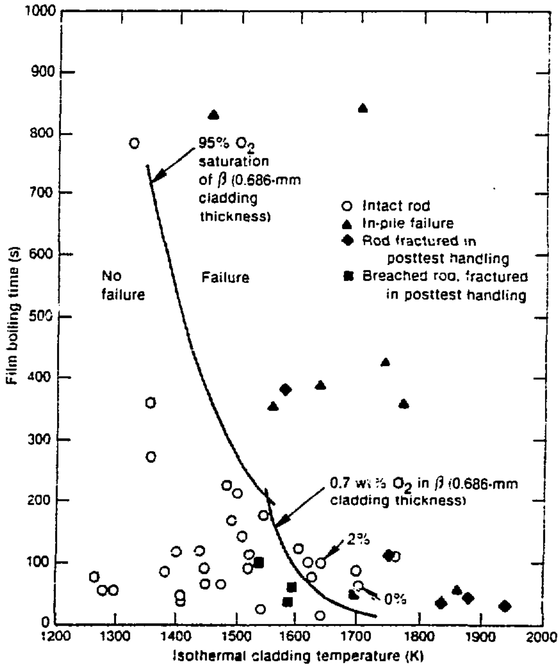

Fig. 4. Fracture map for fuel rods tested under power-cooling-mismatch conditions. (The percent ductility remaining in two cladding samples is also shown. The isothermal cladding temperature is the constant temperature during film boiling which would have produced an oxide layer on the surface of the cladding equal to the measured oxide layer.) (From Ref. 18)

CLEARANCE STATEMENT

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I hereby certify that all clearances regarding patents, classification, proprietary information and other requirements of my organization and/or government with respect to publication of the attached paper entitled: On the Determination of Post-DNB and Post-BT Fuel Design Limits

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