

## A Generalized Failure Map for Fuel Elements Subject to a Power Pulse

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### ABSTRACT

This paper reviews the various power pulse tests that have been conducted in research reactors over the past three decades and summarizes the dominant fuel failure modes that have been observed in these tests. Using physically based analytical models a map of failure limits for Zircaloy-clad uranium dioxide fuel is developed. This failure map is used to assess the experimental data from the power pulse tests. It is shown that the fuel failure map provides a good basis for establishing criteria that demarcate the observed fuel failure modes for the various fuel designs that have been used in these tests.

### INTRODUCTION

The behaviour of nuclear fuel subjected to a short duration power pulse is of relevance to LWR and CANDU reactor safety. Reactivity Initiated Accidents (RIA) in LWR's would subject fuel to a short duration power pulse of large amplitude, whereas in CANDU a large break Loss of Coolant Accident (LOCA) would subject fuel to a longer duration, lower amplitude power excursion. The energy generated in the fuel during the power pulse is a key parameter governing the fuel response.

This paper reviews the various power pulse tests that have been conducted in research reactors over the past three decades and summarizes the fuel failure modes that have been observed in these tests. A simple analytical model is developed to characterize fuel behaviour under power pulse conditions and the model is applied to assess the experimental data from the power pulse tests. It is shown that the simple model provides a good basis for establishing criteria that demarcate the observed fuel failure modes for the various fuel designs that have been used in these tests. These criteria are developed into a generalized failure map applicable to zircaloy clad uranium dioxide fuel. In addition, it is shown that the analytical model provides a basis for application of the available experimental data to quantify the behaviour of CANDU-type fuel.

### SUMMARY OF RIA EXPERIMENTAL DATA

Since the early 1990's a significant amount of data on RIA tests performed on LWR, VVER and CANDU-type fuels in various research reactors has become available in the open literature. The tests are briefly summarized below [1].

#### LWR-Type Fuel

Some of the earliest RIA tests were performed in the Special Power Excursion Reactor Test (SPERT) program conducted in the Capsule Driver Core (CDC) facility [2]. These tests were performed at atmospheric pressure, ambient temperature conditions, stagnant water coolant conditions and zero initial power representing zero power cold startup conditions in a Boiling Water Reactor (BWR). Fuel burnup ranged from 0 GWd/tU (fresh) to approximately 33 GWd/tU (792 MWh/kgU). These tests indicated rapid fuel dispersal (fragmentation) occurring at radial fuel enthalpy in excess of 275 cal/g (1150 kJ/kg).

A series of RIA tests were performed in the Power Burst Facility (PBF) at Idaho Falls [2]. The tests employed three different designs of Pressurized Water Reactor (PWR) fuel at zero irradiation and burnup of approximately 5 GWd/tU (120 MWh/kgU). Conditions in the tests simulated zero power, hot startup in a BWR (coolant temperature of 538°K, coolant pressure of 6.45 MPa and flow of 85 cm<sup>3</sup>/s). Fuel failures involving fuel fragmentation due to gross fuel melting were observed in a number of instances at radial average fuel enthalpies in the range 277 cal/g to 285 cal/g. Failures due to Pellet Clad Mechanical Interaction (PCMI) was also observed for some of the irradiated fuel rods. Typical half-height width of the pulse in the SPERT and PBF tests is approximately 4 ms which is narrower than typical LWR RIA pulse widths of approximately 20ms and is more than two orders of magnitude narrower than CANDU LOCA power pulses.

A very large number of RIA tests have been conducted in the Nuclear Safety Research Reactor (NSRR) test program of the Japan Atomic Energy Research Institute (JAERI). The NSRR is a modified TRIGA-ACPR (Annular Core Pulse Reactor) with a flexible pulse shaping control system and high pulse power capability. In excess of 1200 tests have been performed to date [4-16] covering a wide range of fuel designs, fuel burnup, coolant conditions and power transient characteristics. However, as is the case with the majority of pulse reactors high energy deposition levels are achieved only

with short pulse widths. As in the SPERT and PBF tests typical half-height pulse width in these tests is of the order of 3 to 4ms. Separate effects tests in NSRR have investigated a wide range of factors [4-16].

Starting in 1992 a test program involving high burnup UO<sub>2</sub> and MOX (Mixed Oxide) PWR fuel has been conducted by the French regulator, IPSN, in collaboration with the utility, EDF [20],[21],[22]. The tests were conducted in the CABRI reactor, a low pressure liquid sodium cooled loop research reactor that is operated by the French CEA at the Cadarache Nuclear Research Centre. Pulse height widths of between 9 ms and 75 ms have been applied in the tests performed to date.

In the first test, Rep Na-1, with high burnup PWR fuel from the Graveline Unit 5 reactor, rod (element) failure occurred at a very low energy deposition level of 30 cal/g and resulted in dispersal of a significant amount of fine fuel particles. As a consequence of this test, an increased interest in RIA fuel behaviour at high fuel burnup has led to a better understanding of the factors influencing high burnup fuel performance under RIA conditions. The primary influencing factors – clad ductility deterioration due to water-side corrosion and development of a “rim” region [21],[22] at the pellet edge – are not relevant to CANDU fuel.

### VVER-Type Fuel

Tests on VVER annular pellet fuel have been performed in the Impulse Graphite Reactor (IGR) is located in Semipalatinsk, Kazakstan. The test program conducted by the Russian Research Centre - Kurchatov Institute of Moscow, Russia involved RIA type tests with fresh and high burnup VVER fuel under atmospheric pressure conditions with both water and air cooling [17],[18],[19]. These tests had very wide power pulses of between 0.63 to 0.84 second half-height width, which is more typical of a CANDU LOCA power pulse. However, the VVER fuel pellets are of annular design with a hollow central region. The fuel clad was Zr-1%Nb alloy, unlike the U.S. and Japanese tests which were performed on fuels with Zr-4 cladding. The Russian Zr-1%Nb alloy has greater corrosion resistance than Zr-4 and, consequently, no significant deterioration in fuel performance was observed at high burnup in the region of 50 GWd/tU (1200 MWh/kgU).

### CANDU-Type Fuel

Experimental data exists for CANDU-type fuel under power pulse conditions that are representative of large LOCA and for RIA conditions similar to those for LWR-type fuel. This data is summarized below.

The PBF-CANDU test was performed in the PBF reactor under conditions simulating a large LOCA [26]. Four test elements were used, three were pre-irradiated to 5 GWd/tU (120 MWh/kgU) in the NRX reactor at Chalk River and the fourth element was fresh. Two of the irradiated test elements were pre-pressurized to 0.44 MPa and 1.1 MPa, respectively, and the fresh element was pre-pressurized to 0.52 MPa. Initial loop coolant water conditions were a temperature of 300°C, a pressure of 10.4 MPa and a flow per test element of 1 l/s. The test was conducted with a loop blowdown transient similar to a CANDU LOCA transient with a controlled increase in element power from approximately 53 kW/m initially to a peak of 100 kW/m in 1 second, followed by a reduction to approximately 10 kW/m in 0.5 seconds. Maximum sheath temperatures in the region of 1250°K to 1305°K were attained and temperatures were maintained at these elevated temperatures until about 105 seconds into the test, at which time quench cooldown was initiated. Two elements failed in the test. One failure was due to atypical oxidation under an imperfectly laser welded spacer pad and the second failure was due to non-uniform diametral strain that caused a small localized cracking of the sheath oxide layer.

RIA type power pulse tests have been performed in a TRIGA-ACPR (Annular Core Pulse Reactor) at the Nuclear Research Institute, Pitesti, Romania. A total of 24 tests have been performed on CANDU type fuel samples starting from zero power cold conditions in a stagnant, water-filled capsule at atmospheric pressure. The results from 12 tests are reported in the open literature [23] and an assessment of some of the tests has been published [24]. As is typical of the TRIGA-ACPR reactors, a high amplitude pulse power is achieved with short pulse width of approximately 4 to 6 ms half-height width. The energy deposition rates are two orders of magnitude more rapid than in a typical CANDU power pulse. However, this data is of relevance in comparing the behaviour of CANDU-type fuel to that of LWR- type fuel under very rapid RIA power pulses.

### FUEL FAILURE MODES

The following fuel failure modes have been identified in RIA type power pulse tests (see Table 1):

*Oxygen embrittlement* of fuel sheaths due to Zircaloy-steam oxidation reaction followed by clad cracking on subsequent thermal quench of the test rod. This failure mode occurs well after the power pulse and is associated with energy deposition levels sufficient to induce departure from nucleate boiling (DNB) on the test rod clad.

*Pellet Clad Mechanical Interaction* (PCMI), which typically occurs during, or shortly after the power pulse, with the fuel sheath still relatively cool. The failure is associated with high strain rate loading of the sheath due to rapid thermal expansion or internal pressurization of the rod. Failures, such as that observed in the CABRI Rep-Na-1 test, can occur at relatively low energy deposition levels if the clad has lost ductility due to waterside corrosion effects, such as hydriding, development of thick oxide layers, and spalling of the oxide.

*High temperature burst* (creep strain failure) resulting in a local bulge and split in the sheath. Fuel sheath temperature in excess of the alpha-beta Zircaloy transition temperature and internal fission gas pressure higher than the external coolant pressure are pre-requisites for this failure mode to occur (i.e. to drive the rapid creep strain mechanism).

*Fuel fragmentation*, which can occur at high fuel enthalpy and involves extensive fuel melting and internal pressurization of the test rod prior to clad failure. Significant amounts of molten fuel can be relocated out of the failed rod and the evidence of prior melting is apparent from the spherical shape of the solidified fuel “particles” collected after the test.

*Fuel dispersal*, which may correspond to fuel fragmentation for fuel with burnup less than approximately 42 GWd/tU, but which is a significantly different process for high burnup LWR fuel. At high burnup (greater than 42 GWd/tU) a “rim” region develops at the outer periphery of the fuel (Ref. 22, 25), of approximately 100  $\mu\text{m}$  depth, in which the fuel microstructure is significantly altered. The rim region is characterized by defect clusters of extremely tangled dislocations, loss of as-fabricated grain boundary, fine recrystallized grains of 50-200 nm size and coarsened pores surrounded by recrystallized grains. Following sheath failure, the fuel in the rim region can be ejected and dispersed as fine fuel particles that do not present the recognizable spherical shape of prior melted fuel particles. This failure mode is not applicable to CANDU fuel since burnup is too low to give rise to a rim region.

**TABLE 1  
SUMMARY OF RIA EXPERIMENTS**

TEST REACTOR	FUEL TYPE	ENRICHMENT	NUMBER OF TEST RODS	PELLET OD [mm]	CLAD MATERIAL	CLAD DIAMETER [mm]	CLAD THICKNESS [mm]	INTERNAL PRESSURE [MPa]	FAILURE MODE	LOWER BOUND ENTHALPY [cal/g]	REF.
SPERT-CDC	BWR	7.0%	22	10.6	Zr-4	12.52	0.86	0.1 0.1 0.1	PCMI CMELT FR	154 206 275	2
SPERT-CDC	BWR	10.5%	5	10.6	Zr-4	12.52	0.86	0.1 0.1 0.1	PCMI CMELT FR	226 274 292	2
PBF	PWR	5.8%	5	9.29	Zr-4	10.72	0.62	0.1 0.1	PCMI FR	185 350	2
NSRR	PWR	10.0%	>1200	9.29	Zr-4	10.72	Multiple 0.57,0.62, 0.64,0.86, 0.9	0.1 0.1 1.2 2 3 5	OE PCMI HTB HTB FR	199 164 195 162 135 220	4,5,6,7 8,9,10,11 12,13,14,15,16
IGR	VVER	4.4%	23 Watercooled 23 Aircooled	7.48	Zr-1%Nb	9.1	0.63	1.7 - 2 1.7 - 2 1.7 - 2	HTB HTB FR	141 102 284	17,18,19
CABRI	PWR	High Burnup	10	8.2	Zr-4	9.5	0.64	0.4	BRIT PCMI	30 82	20,21,22
PITESTI	CANDU Type	10.0%	12	12.15	Zr-4	13.08	0.4	0.1 - 2.2	OE HTB	170 136	23,24
PBF LOCA	CANDU	10.0%	4	12.15	Zr-4	13.08	0.4	0.4 - 1.09	OE		26

## ANALYSIS OF LIMITS FOR FUEL FAILURE MODES

The average fission energy generation rate (power) in the fuel pellets of a fuel element is proportional to the neutron flux,  $\phi(t)$ , which for pulse type excursions can be approximated by a point kinetic equation of the form:

$$\frac{d\phi}{dt} = \frac{(\rho_{insertion} - \rho_{feedback} - \beta_{dn})}{\ell^*} \phi + \phi_0 \quad (1)$$

The Doppler feedback can be approximated by a function of the integrated power (energy deposited) in the fuel, since this quantity determines the energy stored in the fuel and hence the fuel temperature. Hence:

$$\rho_{feedback} = \xi \int \phi(t) dt = \xi E(t) \quad (2)$$

The radial power distribution in a fuel pellet of radius R can be represented by:

$$P(r,t) = K f(r) \phi(t) \quad (3)$$

Where,

$K$  = neutron flux to power conversion constant

$f(r) = \Psi_0 \{I_0(\kappa r) + \beta e^{\lambda(r-R)}\}$ , and  $\Psi_0$ ,  $\kappa$ ,  $\beta$  and  $\lambda$  are functions of fissile enrichment and burnup.

The corresponding radial distribution of UO<sub>2</sub> enthalpy in the fuel pellet due to a radial averaged specific energy deposition  $E_a$  (kJ/kg) is given by:

$$h(r) = f(r)E_a + h_i(r) \quad (4)$$

$h_i(r)$  is the initial radial distribution of enthalpy corresponding to an initial temperature distribution  $T_i(r)$ . Note that the majority of RIA tests have been performed under cold zero-power initial conditions for which  $T_i(r)$  is constant and approximately equal to 20°C. The energy deposition during the short width power pulses typical of RIA tests is effectively adiabatic and the radial temperature distribution in the pellet can be evaluated from:

$$T(r) = g\{h(r)\} \approx c_0 + c_1 h(r) + c_2 h(r)^2 \quad (5)$$

Where  $g\{h(r)\}$  is the inverse functional relationship between temperature and enthalpy of UO<sub>2</sub> derived from the functional relationship between enthalpy and temperature for UO<sub>2</sub> [27],[28], namely  $h_{UO_2} = H(T_{UO_2})$ . Limits for the various fuel failure modes observed in the RIA tests in different tests reactors can be established from fundamental consideration of the governing physical phenomena. These limits are derived as functions of the radial averaged specific energy deposition,  $ED_a$ , as follows.

### Oxygen Embrittlement (OE)

The fuel clad (sheath) temperature  $T_c$  can be expressed as a function of the fuel pellet surface temperature  $T_s = T(R)$ , the coolant temperature  $T_l$ , the gap conductance  $h_g$  the clad thickness  $\tau$  and thermal conductivity  $k$  and the coolant convective heat transfer coefficient  $h_c$  according to:

$$T_c = \frac{h_{eff} T_s + h_c T_l}{h_{eff} + h_c} \quad \text{with} \quad h_{eff} = \left( \frac{1}{h_g} + \frac{\tau}{k} \right)^{-1} \quad (6)$$

and  $T_s = g\{h(R)\} = g\{f(R)ED_a + h_i(R)\}$

If the energy deposition and resultant pellet surface temperature are sufficiently large then the clad outer surface will experience departure from nucleate boiling (DNB) and subsequent film boiling. If the clad temperature increases sufficiently, then extensive oxidation can occur during the period that the deposited energy is transferred from the fuel to the surrounding coolant. Zircaloy oxidation kinetics exhibits parabolic rate behaviour in which the thickness of the Zircaloy oxide layer,  $\delta_o$ , as a function of time,  $t$ , under isothermal conditions can be expressed as:  $\delta_o^2 = D t_o$  with

$$D = C \exp(B/(T + 273)) = C \exp(B/T')$$

The Zr-4 clad oxidation kinetics parameters  $C$  and  $B$  are evaluated from the Prater-Cathcart correlation [29].

The thickness of Zircaloy consumed,  $\delta_c$ , to produce a Zircaloy oxide layer of thickness  $\delta_o$  is given by  $\delta_o = 1.56 \delta_c$ . The thickness of the oxide layer on the clad for a PWR fuel element as a function of the power pulse energy deposition (in units of cal/g) is shown in Figure 1 for different fissile enrichments in the range of the RIA tests. The calculated values of oxide thickness for the 10% enrichment are in good agreement with those measured in NSRR tests in Japan [13]. As the extent of clad oxidation increases the higher will be the thermal stresses experienced on quench cool-down and the more likely will be clad failure.

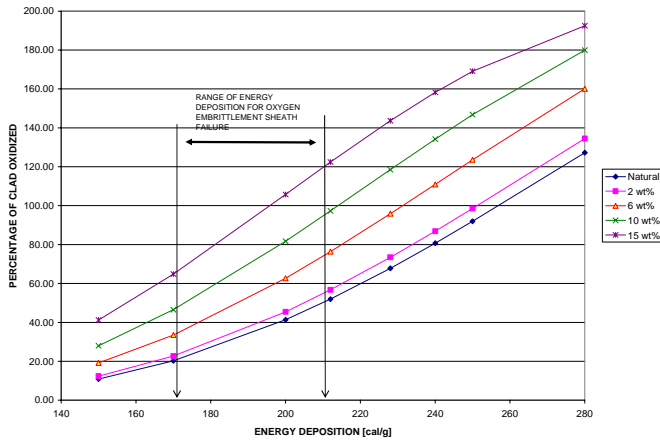


Figure 1. Percentage of Clad Oxidation vs Energy Deposition

The thermal hoop stress in the clad due to a temperature difference,  $\Delta T$ , between inner and outer surfaces induced by quenching the hot clad is given by the well known relationship:

$$\sigma_\theta = \frac{E\alpha\Delta T}{2(1-\mu)\ln(1+m)} \left( 1 - \frac{m}{3} \right) \quad (8)$$

$$(1+m) = \frac{R_i + \tau_m}{R_i}$$

### Pellet-Clad Mechanical Interaction (PCMI)

The radial expansion  $\Delta r$  of a UO<sub>2</sub> fuel pellet subject to a change in temperature  $\Delta T$  can be approximated as:

$$\Delta r \approx 0.5 D_{pellet} \alpha \Delta T \quad (9)$$

Where  $\alpha$  is the coefficient of thermal expansion of  $UO_2$ . The corresponding radial strain is  $\epsilon \approx \alpha \Delta T$ . Using the relationship between fuel temperature and fuel enthalpy defined previously, the strain rate can be described by:

$$\dot{\epsilon} = \alpha \frac{dT}{dt} = \alpha \frac{\partial g}{\partial h} \frac{dh}{dt} \tag{10}$$

Now the rate of change of enthalpy can be characterized by the ratio of the specific energy deposited in the power pulse,  $ED$  in units of kJ/kg, to the half height pulse width,  $\Delta\tau$ , yielding the following expression for strain rate,

$$\dot{\epsilon} = \alpha \frac{dT}{dt} = \alpha \frac{\partial g}{\partial h} \frac{ED}{\Delta\tau} \tag{11}$$

The calculated strain rate as a function of energy deposition and half height pulse width is shown in Figure 2. Based upon the data in Table 1, PCMI failures were observed with irradiated fuel in tests with pulse widths of approximately 4 ms and fuel enthalpy values in excess of 600 kJ/kg (154 cal/g). this corresponds to a threshold strain rate of 0.5 per sec. As can be seen from Figure 2 PCMI failures are strongly dependent upon pulse width and become less likely as pulse width increases. A very low probability of PCMI failure is expected for CANDU-typical power pulses which have pulse widths of the order of 1000 ms.

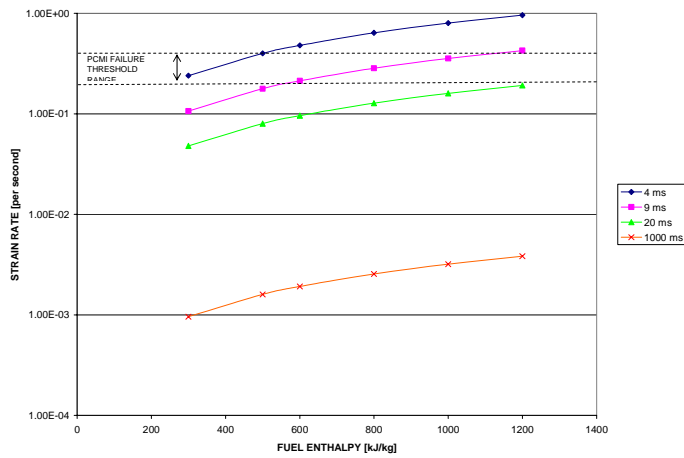


Figure 2. Clad Strain Rate vs Fuel Enthalpy

**Clad Melting (CMELT)**

The energy deposition values for clad melting failure are similar to those associated with oxygen embrittlement failure and the two mechanisms frequently occur together [13]. The energy deposition limits for Zircaloy melting (clad temperature of 1860°C) are shown as a function of pellet fissile enrichment in Figure 3.

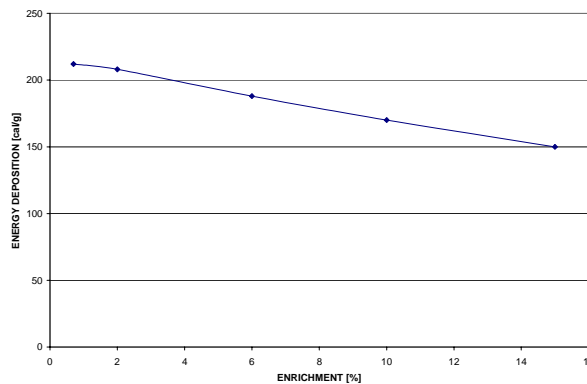
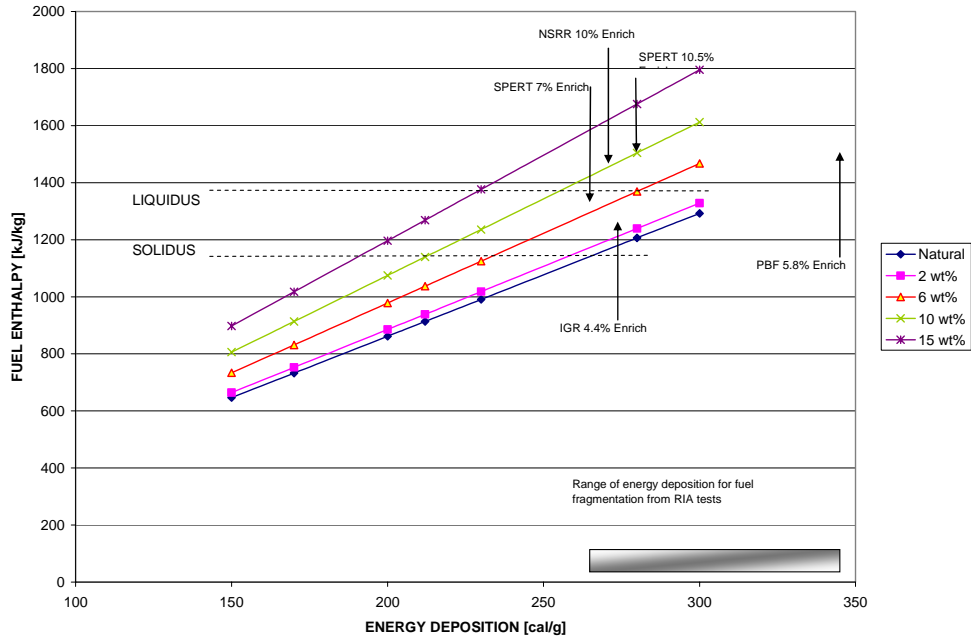


Figure 3 Energy Deposition for Clad Melting

**Fuel Fragmentation (FR)**

Evidence from the RIA tests on fresh fuel and irradiated fuel with burnup less than 40 Gwd/tU is that fragmentation of the fuel is associated with ejection of molten material from the fuel. As shown above, the maximum fuel temperature for power pulses from cold zero-power initial conditions occurs at the surface of the fuel pellet which, in turn, is where melting is likely to first occur. The enthalpy at the surface of the pellet as a function of the radial average energy deposition is shown in Figure 4 for various fissile enrichments. Also shown is the enthalpy for liquid UO<sub>2</sub> formation, which is the sum of the enthalpy for onset of melting in the solidus phase plus the heat of fusion (259 kJ/kg). The lower bounds on radial average fuel enthalpy for fuel fragmentation observed in the various RIA tests are shown in Figure 4. This clearly indicates that fragmentation of fuel with ductile cladding is associated with melting at the periphery of the fuel pellet and is strongly influenced by the radial fission power distribution, as governed by the fissile enrichment in the tests.



**Figure 4 Fuel Enthalpy at Pellet Surface as a Function of Energy Deposition**

**FUEL FAILURE MAP FOR POWER PULSE EVENTS**

A fuel failure map for the various fuel failure modes is developed from the analytical criteria described above. For each failure mode the limiting conditions are evaluated by converting where necessary the energy deposition into an equivalent radially averaged fuel enthalpy using standard properties for UO<sub>2</sub> and Zircaloy obtained from the literature. Adjustment to the fuel enthalpy is made to account for the energy removal during the power pulse. For the majority of the RIA test data conducted with very short half pulse widths, this adjustment is small. The resultant failure map is plotted in Figure 5.

**CONCLUSIONS**

The following conclusions are drawn regarding the failure map for fuel failure modes under power pulse conditions such as RIA:

- 1) Threshold values of fuel enthalpy for clad melting and fuel fragmentation failures are not strongly dependent upon the power pulse width (half height width) and increase slightly with increase in pulse width due to the increased heat removed by the coolant. Clad melt failure occurs at similar, but slightly higher enthalpy values than oxygen embrittlement failure.
- 2) Threshold values of fuel enthalpy for oxygen embrittlement failure tend to increase with increasing pulse width due to the larger sensitivity of fuel sheath temperature to energy removed by the coolant. Fuel sheath failure mode changes from oxygen embrittlement failure, during a thermal quench, to high temperature burst as the pressure difference between the inside of the fuel element and the outside coolant increases. This was the predominant

failure mode seen in the NSRR tests with pressurized rods, the IGR tests with high burnup VVER rods and the Romanian tests with CANDU type fuel.

- 3) PCMI failure has a strong sensitivity to pulse width and the threshold fuel enthalpy for failure increases rapidly with increasing half height pulse width. Additionally, for this mode of failure very small gaps between fuel pellet and sheath is necessary. It is for this reason that PCMI failures have been observed in RIA tests with short pulse widths (4 ms) and irradiated fuel, while unirradiated fuel tends not to exhibit this failure mode. PCMI is an unlikely failure mode for CANDU fuel because of the long power pulse.
- 4) Fuel fragmentation for fresh and irradiated rods with fuel sheaths that maintained ductility (i.e. sheaths without the excessive corrosion observed on LWR high burnup fuel cladding) is associated with gross melting of the fuel.
- 5) Avoidance of fuel melting precludes fuel fragmentation.

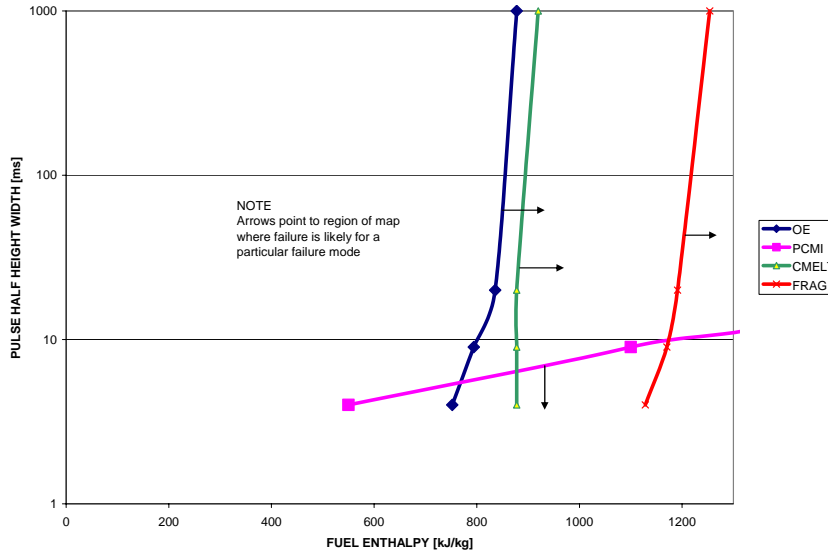


Figure 5. Failure Map for UO<sub>2</sub> Fuel Subject to a Power Pulse

#### MISCELLANEOUS NOMENCLATURE

$\rho_{insertion}$  = positive reactivity insertion that initiates the power pulse,

$\rho_{feedback}$  = negative fuel temperature (Doppler) reactivity feedback,

$\bar{\lambda}$  = prompt neutron generation time

$\beta_{dn}$  = total fraction of delayed neutron precursors

$\phi_0$  = initial neutron source

$t_o$  = time over which clad is oxidized.

$T$  = temperature (C)

$T$  = temperature (K)

$D$  = rate constant, a function of  $T$ .

$ED$  = deposited fission energy

$E$  = Young's modulus of clad

$\mu$  = Poisson ratio for clad

$\tau_m$  = thickness of non oxidized (metal) region of clad

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