Public Workshop on the RIA Interim Acceptance Criteria

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Workshop Agenda

1. Two-Stage Approach to Revising RIA Criteria and Guidance
2. Control Rod Worth and Pulse Width
3. Draft Core Coolability Criteria
4. Draft Fuel Rod Failure Criteria
5. Draft Guidance

Regulatory Strategy

Interim Criteria and Guidance

- Finalize – December 2006
- ACRS Committee – Jan/Feb 2007
- Issue SRP Update – March 2007

Revision to Regulatory Guides

- Finalize criteria and guidance over 18 months based on forthcoming RIA tests.
- Revise RG 1.77, 1.195, and 1.183 – End of 2008

Control Rod Worth and Pulse Width

Industry Discussion:

- Control rod worth and its relation to fuel assembly burnup, core power level, and core operating limits for both Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR).
- The amount of time that a plant operates with the maximum calculated control rod worth and the conditions necessary to achieve this maximum worth.
- The axial length and location of high fuel pellet enthalpy.
- Calculated pulse widths for both BWRs and PWRs.

Problems with RG 1.77 Criteria

In 1980, MacDonald et al. reviewed earlier test data from SPERT and TREAT (which form the basis of the current regulatory limits) and then compared these earlier tests to the then recent PBF test results. MacDonald concluded:

- LWR fuel rods subjected to the regulatory limit, radial average fuel enthalpy of 280 cal/g, will be severely damaged and post-accident cooling may be impaired.
- The NRC expressed the RIA criteria in terms of fuel enthalpy, whereas the SPERT and TREAT data were reported in terms of total energy deposition. Based on this difference, a more appropriate value for the RIA criteria would have been 230 cal/g.
- PCMI clad failure may result at a radial average fuel enthalpy of 140 cal/g on irradiated LWR fuel rods as compared to the 170 cal/g failure criteria.
- Fuel grain-boundary separation and powdering also contribute to a loss of rod geometry during quenching.
- The mode of fuel rod failure is strongly dependent on previous irradiation history.

Long-Term Core Cooling and Reactor Pressure Vessel Integrity

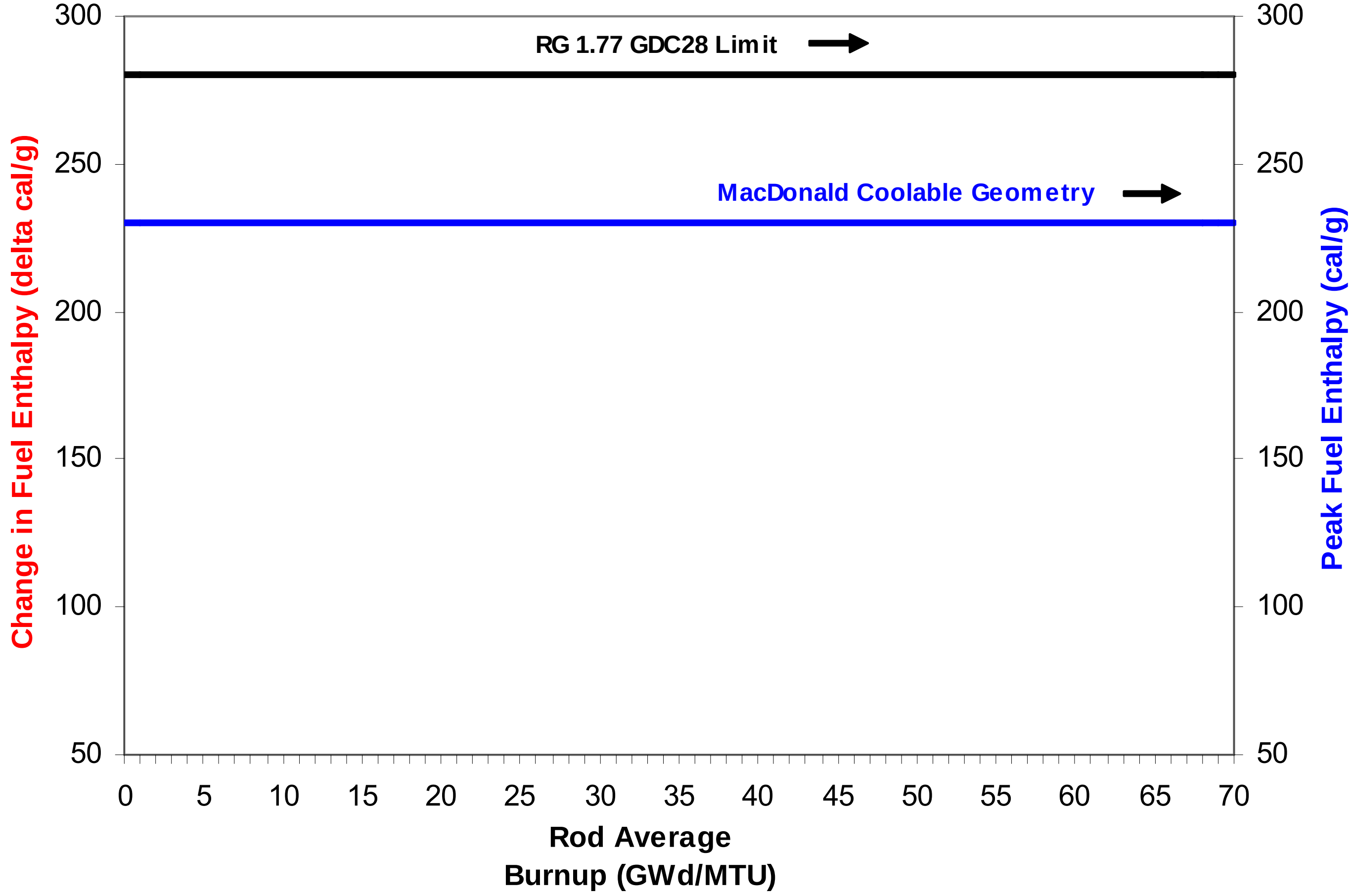
Phenomena

- Pressure pulse generated by the violent expulsion of molten or near molten fuel fragments and ensuing interaction with reactor coolant.
- Flow blockage due to fission product-induced swelling of molten or near molten fuel coupled with cladding plastic deformation.
- Flow blockage due to fuel pellet and cladding fragmentation and dispersal.
- Fuel powdering and dispersal within the reactor coolant system.

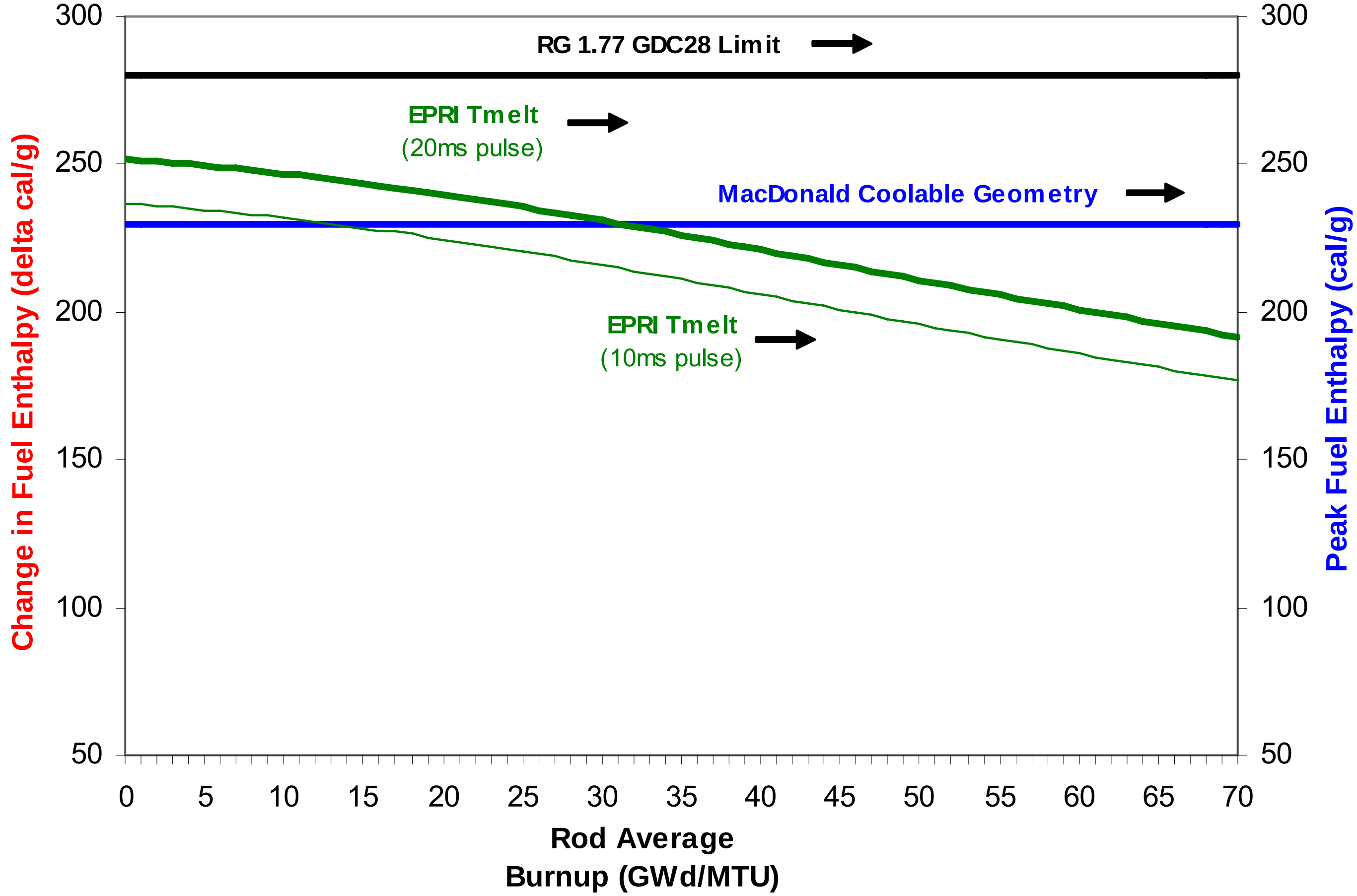
Draft Core Coolability Criteria

1. Peak radial average fuel enthalpy remains below 230 cal/g.
2. Peak fuel temperature remains below incipient melt based upon design-specific calculation using NRC-approved models.
3. No loss of coolable geometry due to fuel pellet and cladding fragmentation and dispersal.
4. Mechanical energy generated as a result of non-molten fuel fragmentation and dispersal must be addressed with respect to reactor pressure integrity, reactor internals, and reactor fuel.

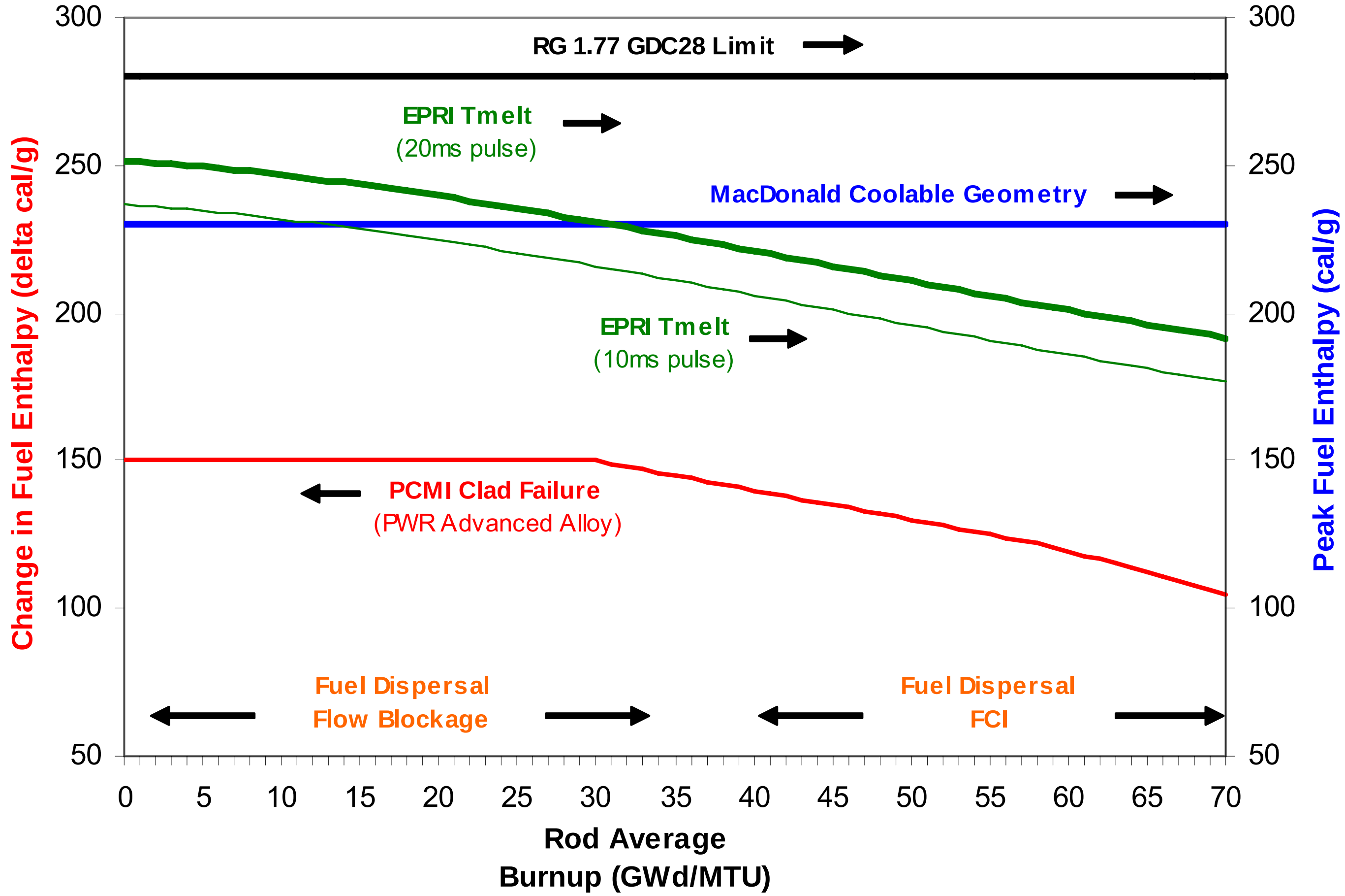
MacDonald's Coolability Criteria



Incipient Fuel Melting



Coolability versus Clad Failure



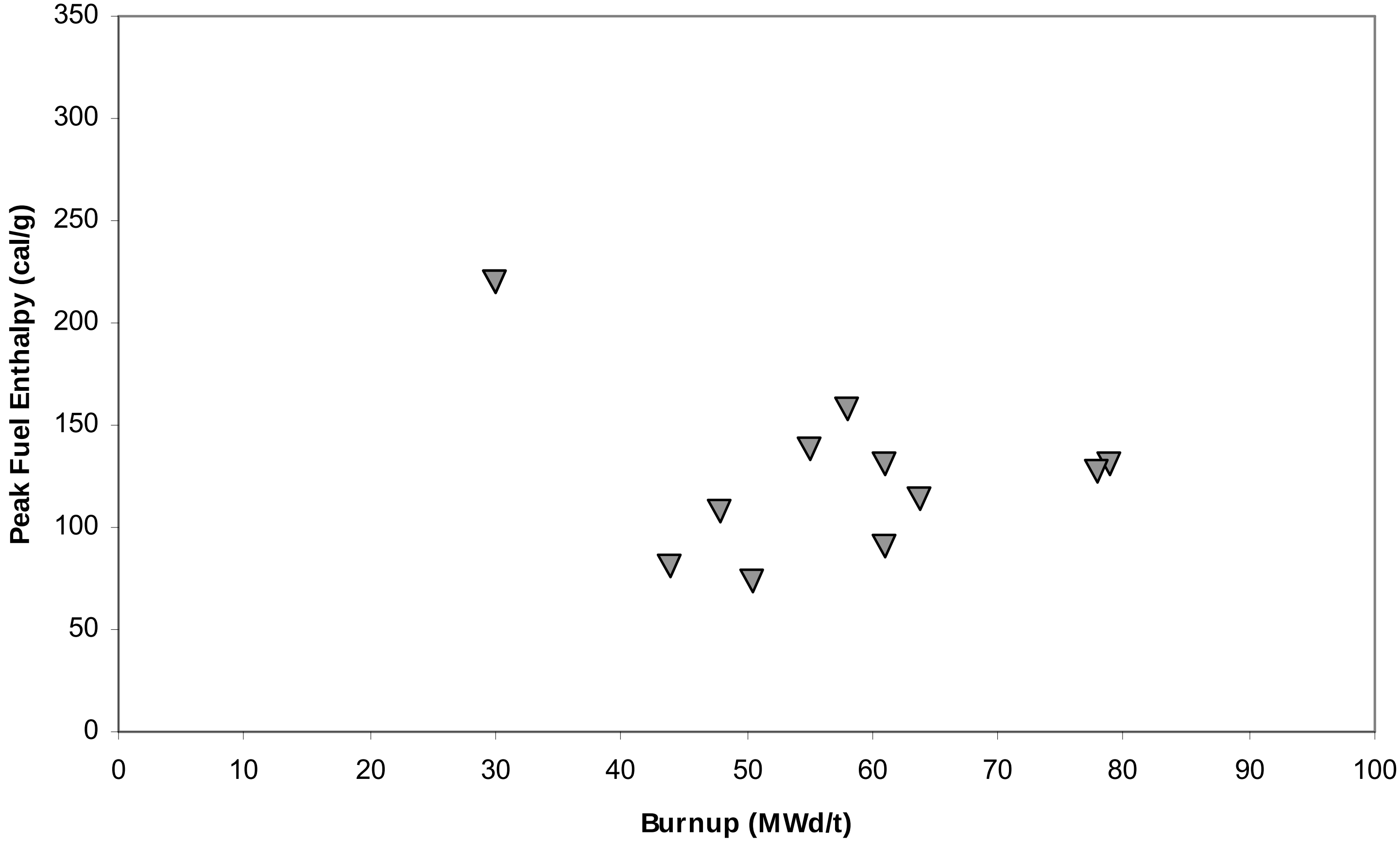
Technical Challenges to Address Fuel Dispersal

1. The amount of fragmented fuel and cladding particles must be limited to ensure that flow blockage does not impede long-term cooling.

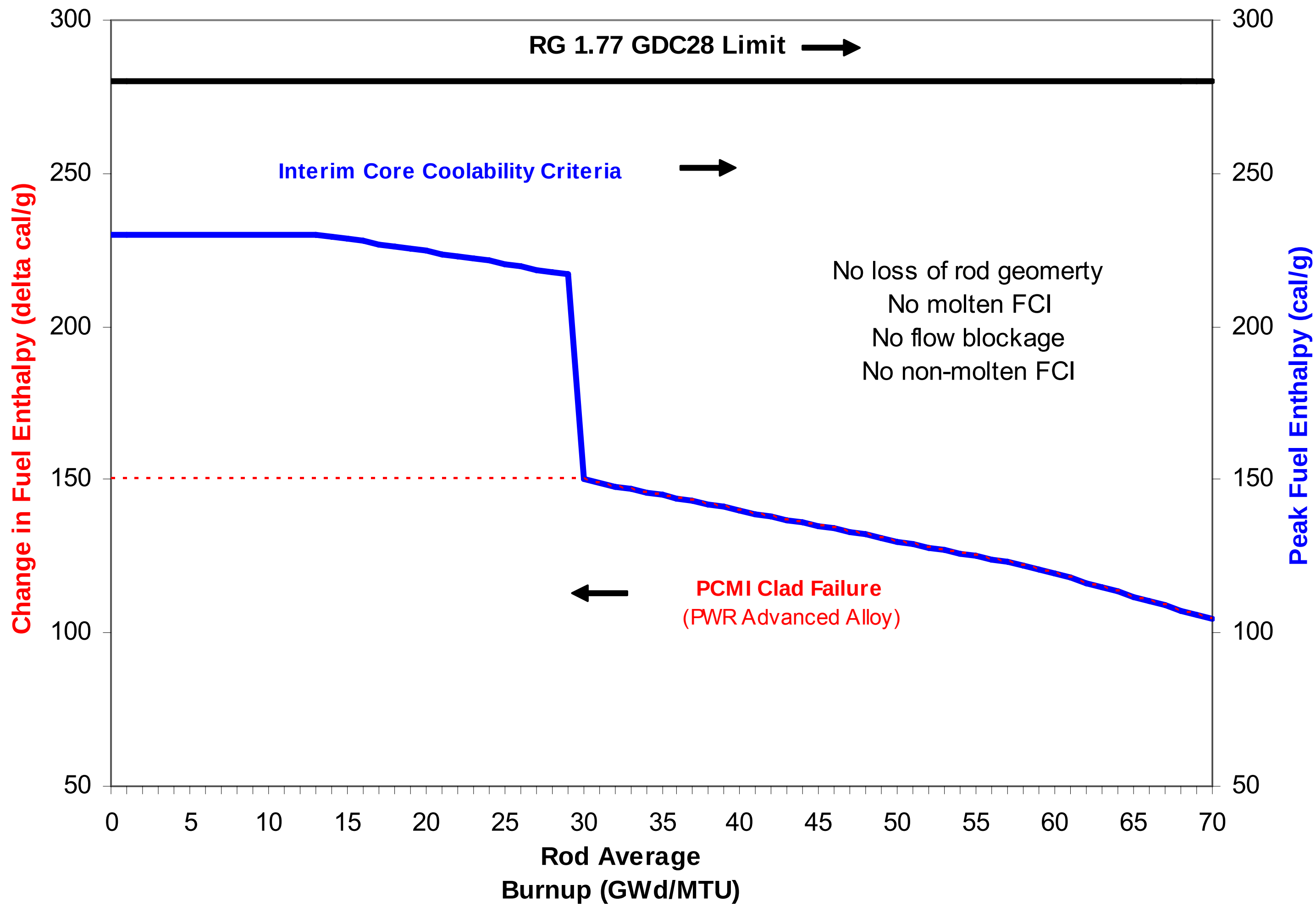
2. The FCI mechanical energy from the dispersal of non-molten fuel fragments, although significantly less than that associated with molten FCI, must be quantified and addressed with respect to reactor pressure integrity, reactor internals, and reactor fuel.

3. The transportation of fragmented fuel particles throughout the reactor coolant system needs to be assessed with respect to radiological consequences (public and workers), plant EQ, coolability, and criticality.

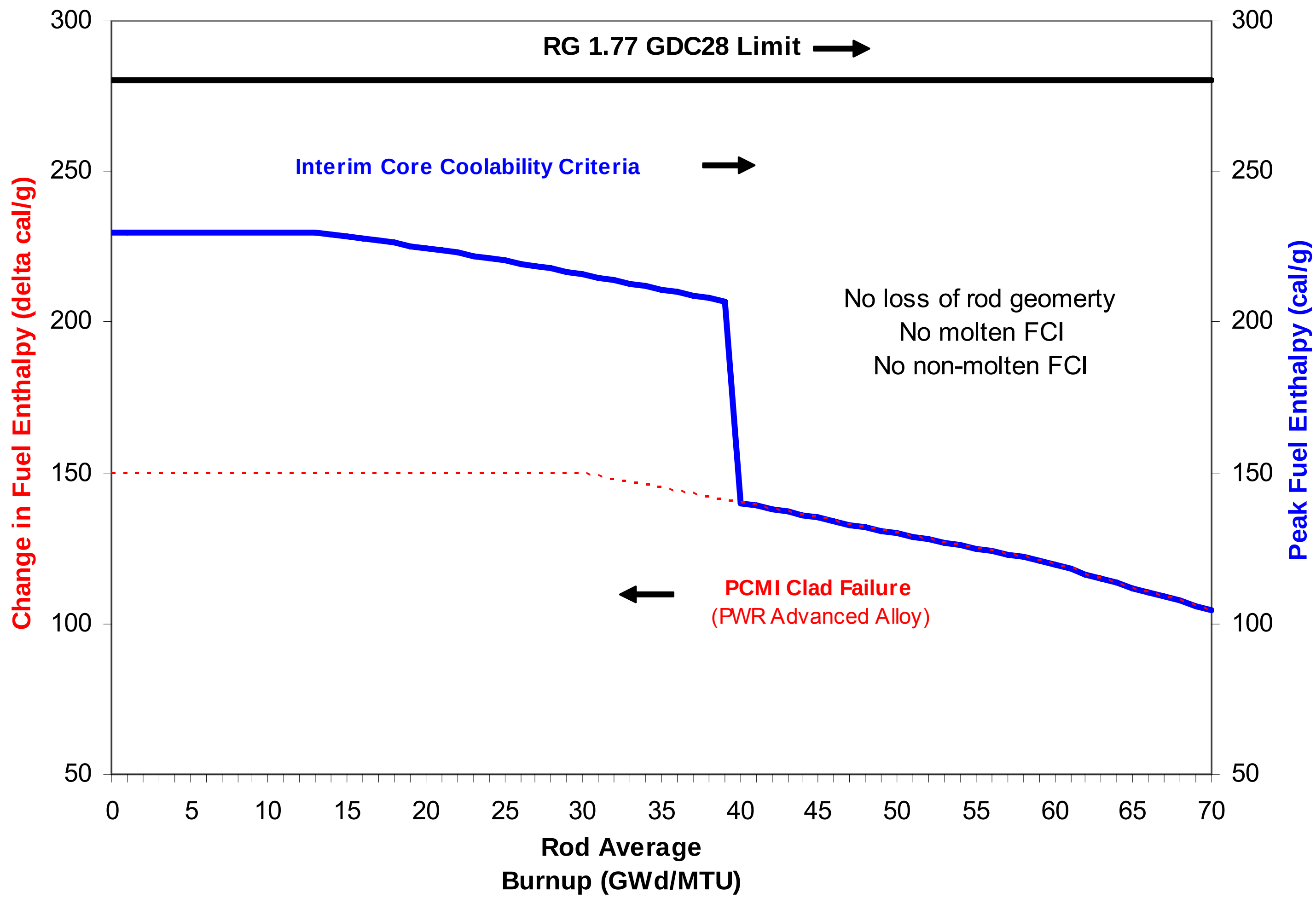
RIA Fuel Dispersal Database



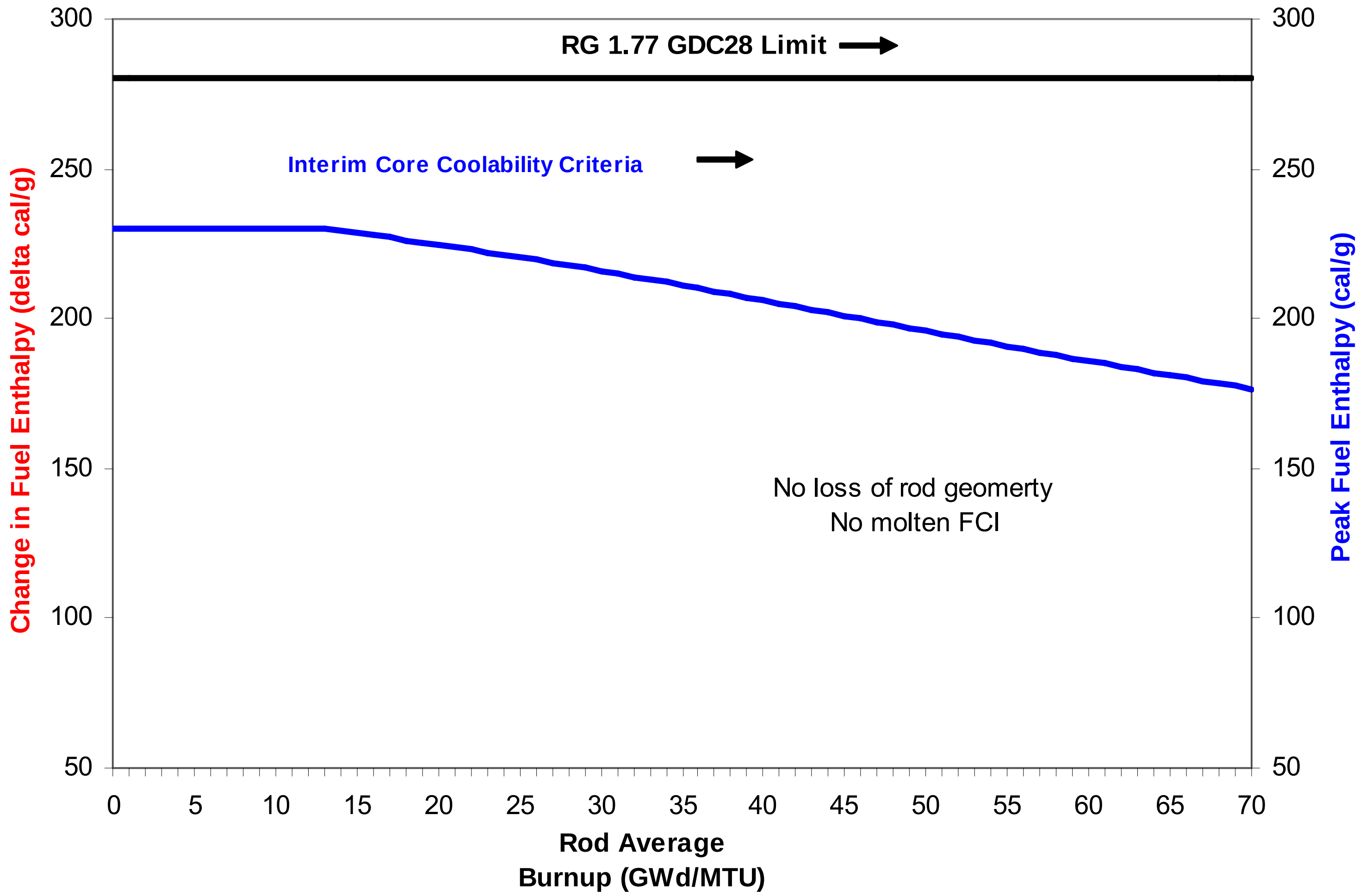
Draft Coolability Criteria – Option #1



Draft Coolability Criteria – Option #2



Draft Coolability Criteria – Option #3



Industry Discussion

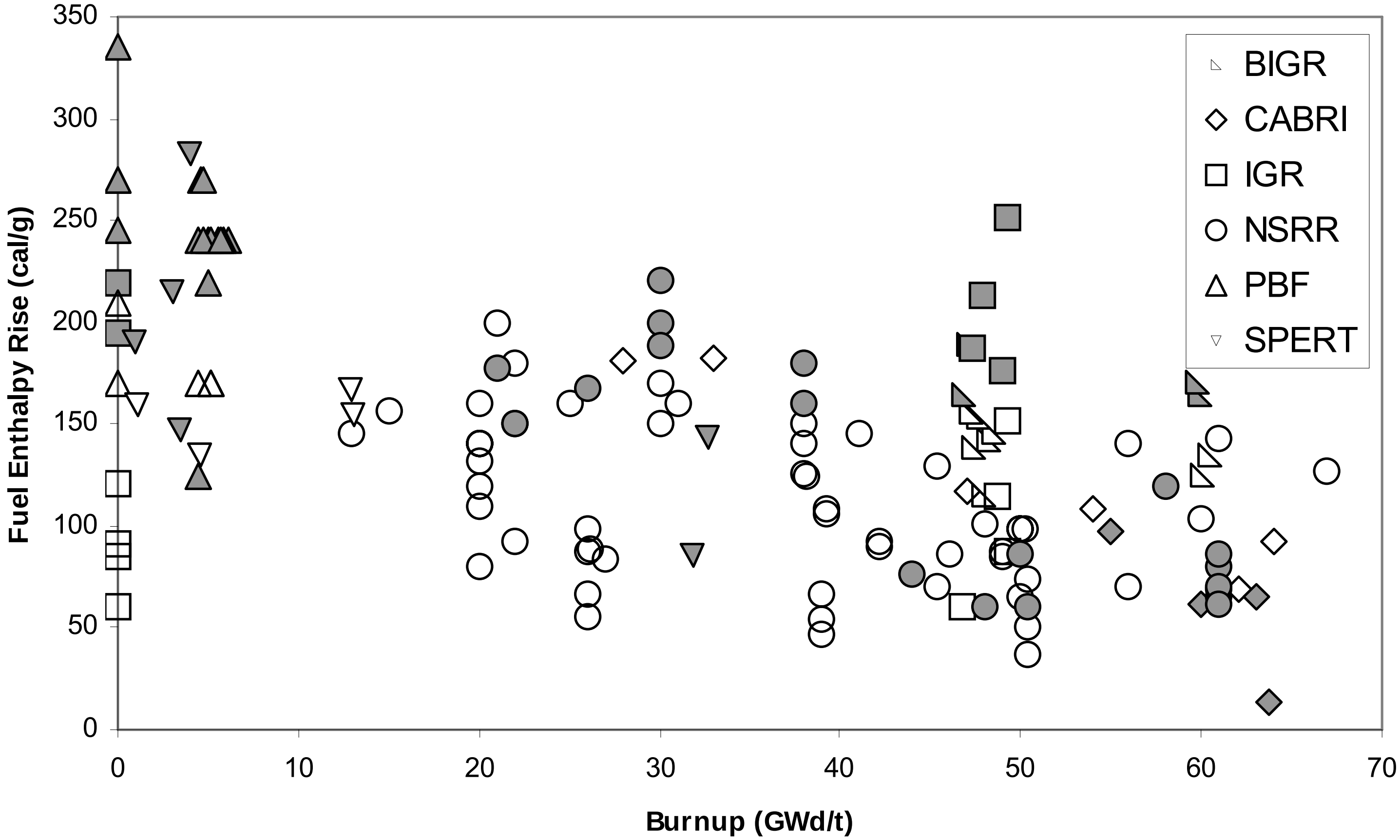
- Fuel pellet microstructure and fission gas accumulation as a function of fuel rod burnup as its relation to grain boundary separation (fuel powdering) and fuel dispersal during a RIA.
- The effect of pulse width on fuel dispersal during a RIA.

Fuel Rod Failure Criteria

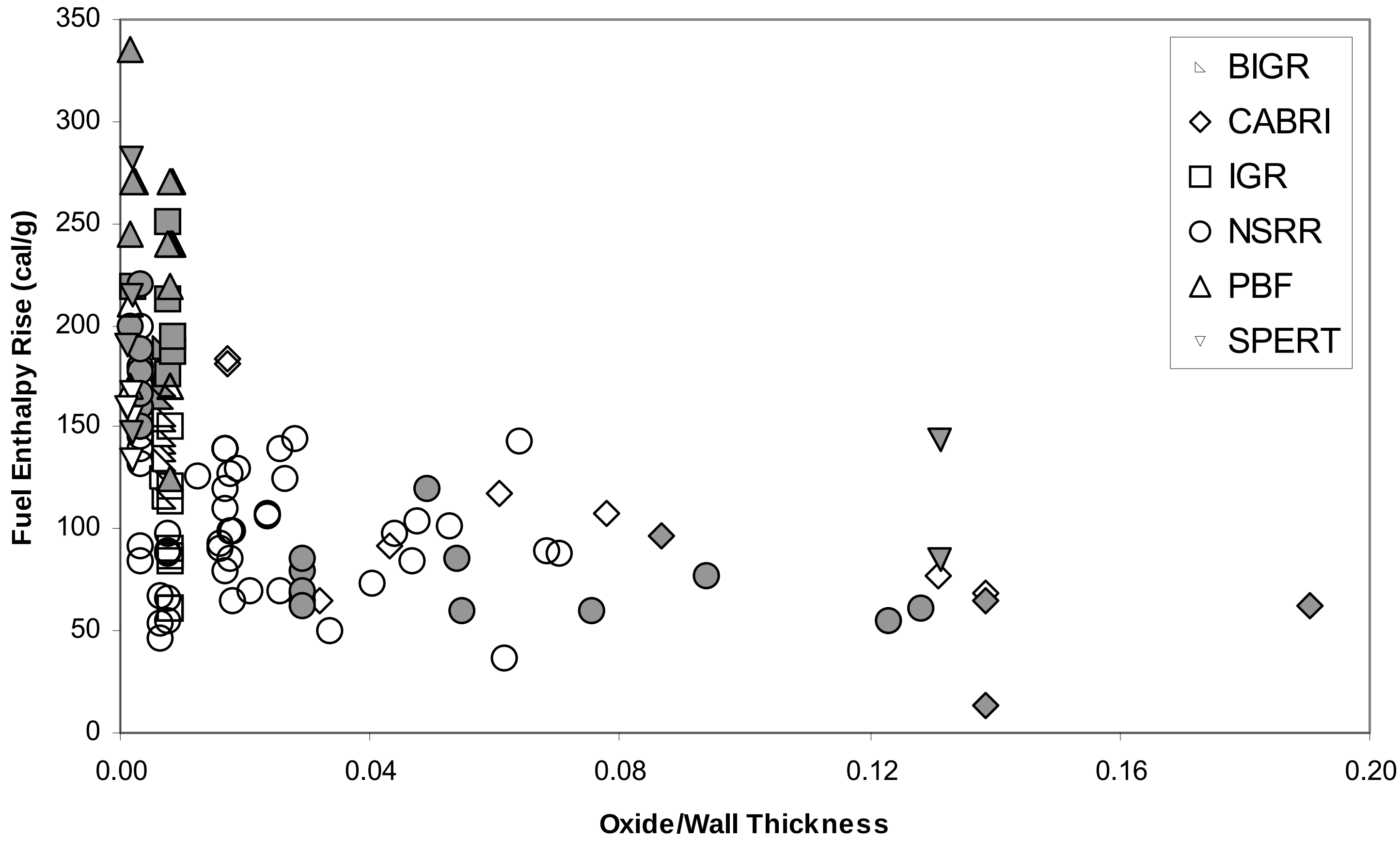
Draft Fuel Rod Failure Criteria

1. Empirically-based PCMI failure criteria.
2. Post-DNB failure criteria.

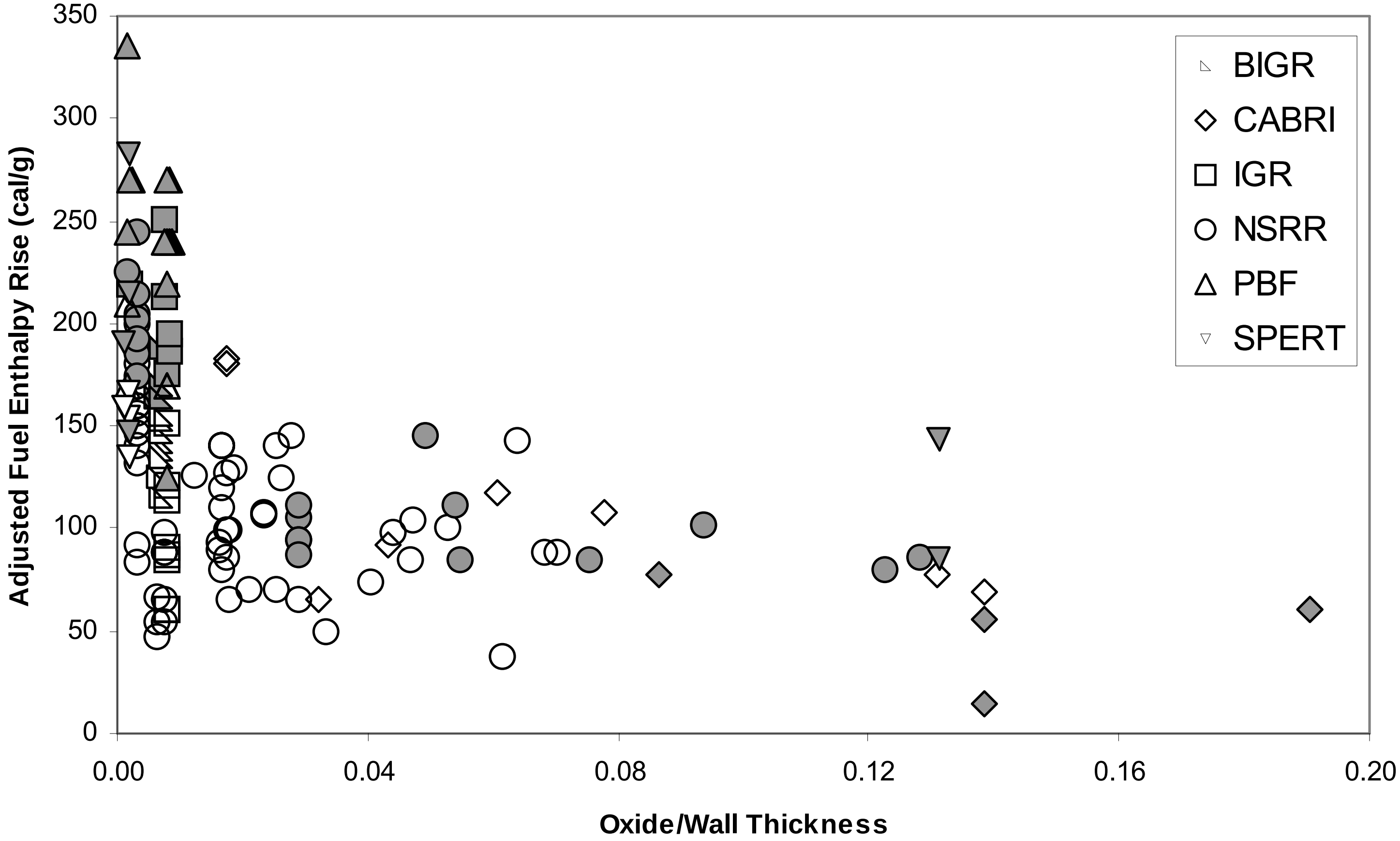
RIA Empirical Database



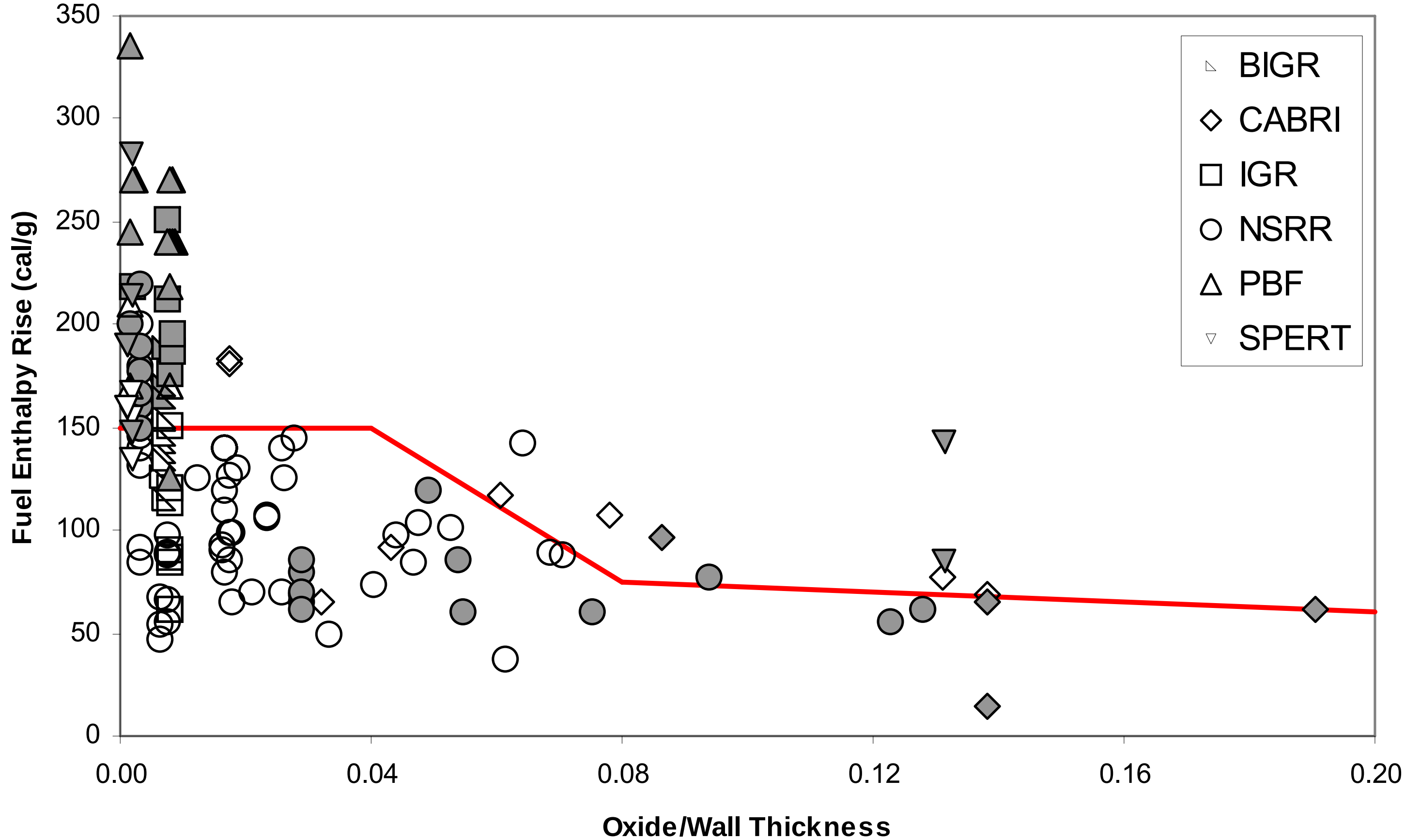
RIA Empirical Database



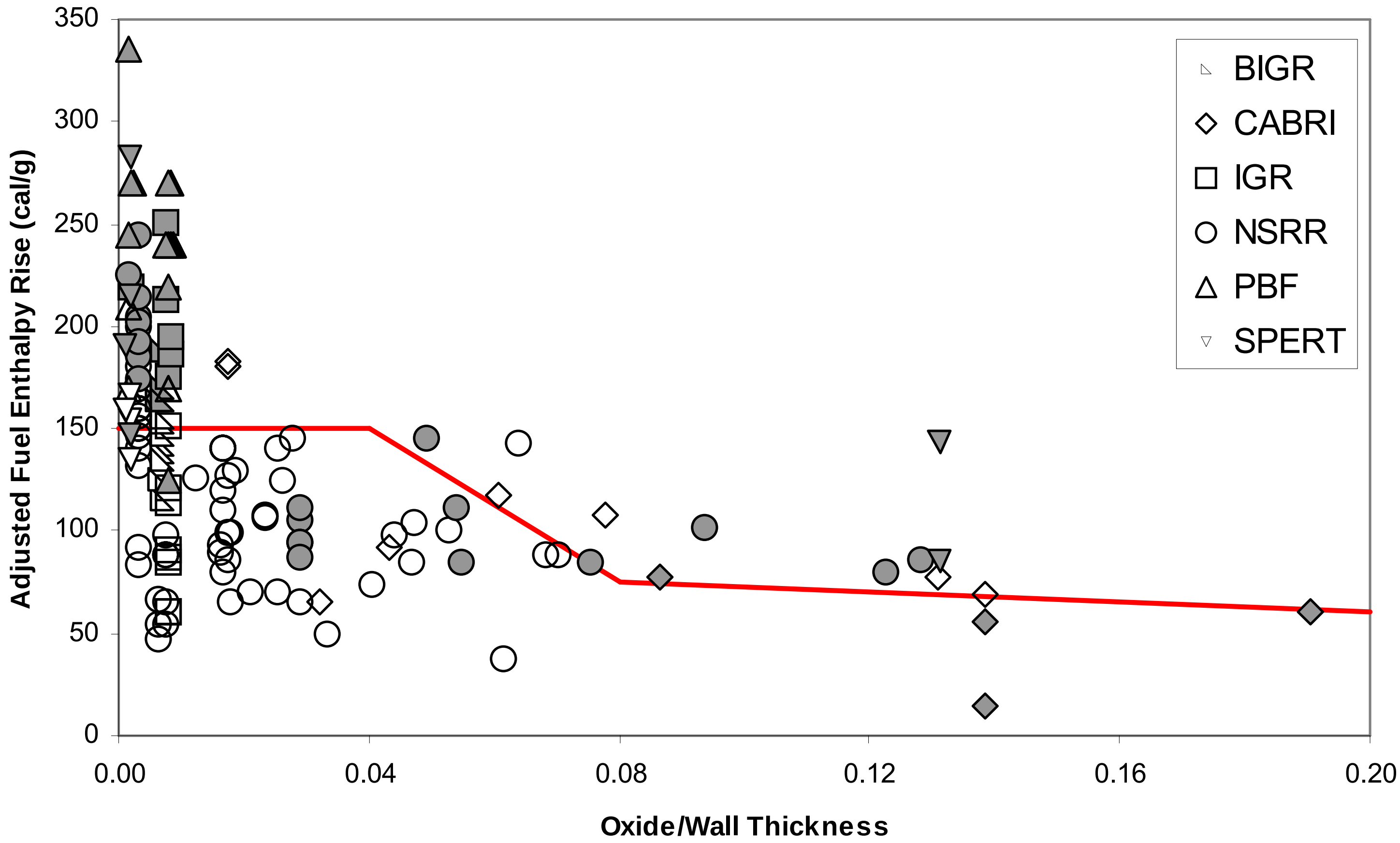
RIL 0401 Adjusted Failure Points



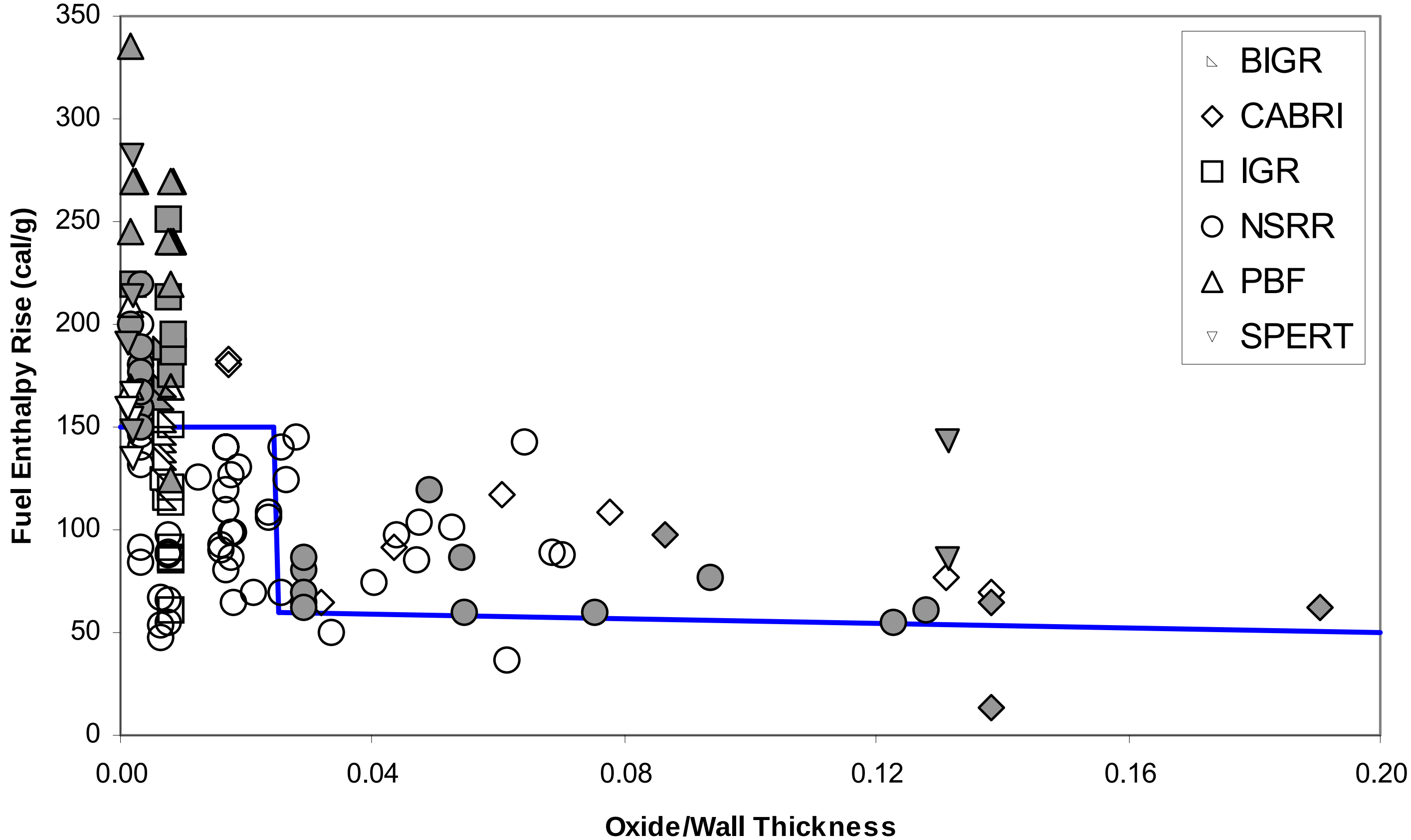
Draft Fuel Rod Failure Criteria PWR & BWR Hot Conditions



Draft Fuel Rod Failure Criteria
PWR & BWR Hot Conditions



Draft Fuel Rod Failure Criteria
BWR Cold Startup Conditions



Draft Fuel Rod Failure Criteria

Empirically-based PCMI failure criteria

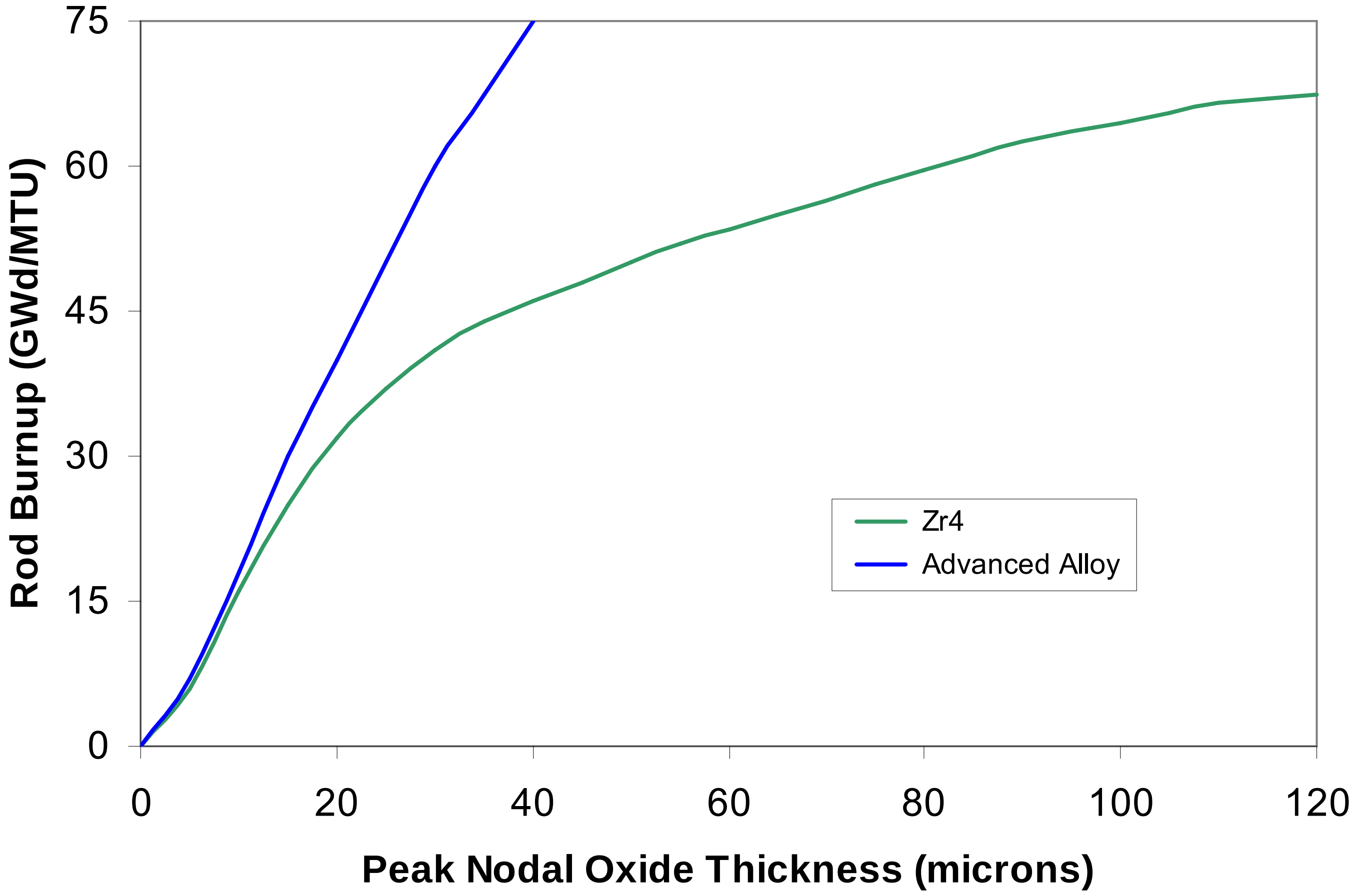
- Convert corrosion based empirical PCMI failure criteria to burnup based on NRC-approved, best-estimate peak nodal oxidation model.
- NRC-approved methods used to predict number of fuel pin failures due to PCMI.

Post-DNB failure criteria

- NRC-approved DNB and CPR correlations used to predict number of fuel pins in DNB.

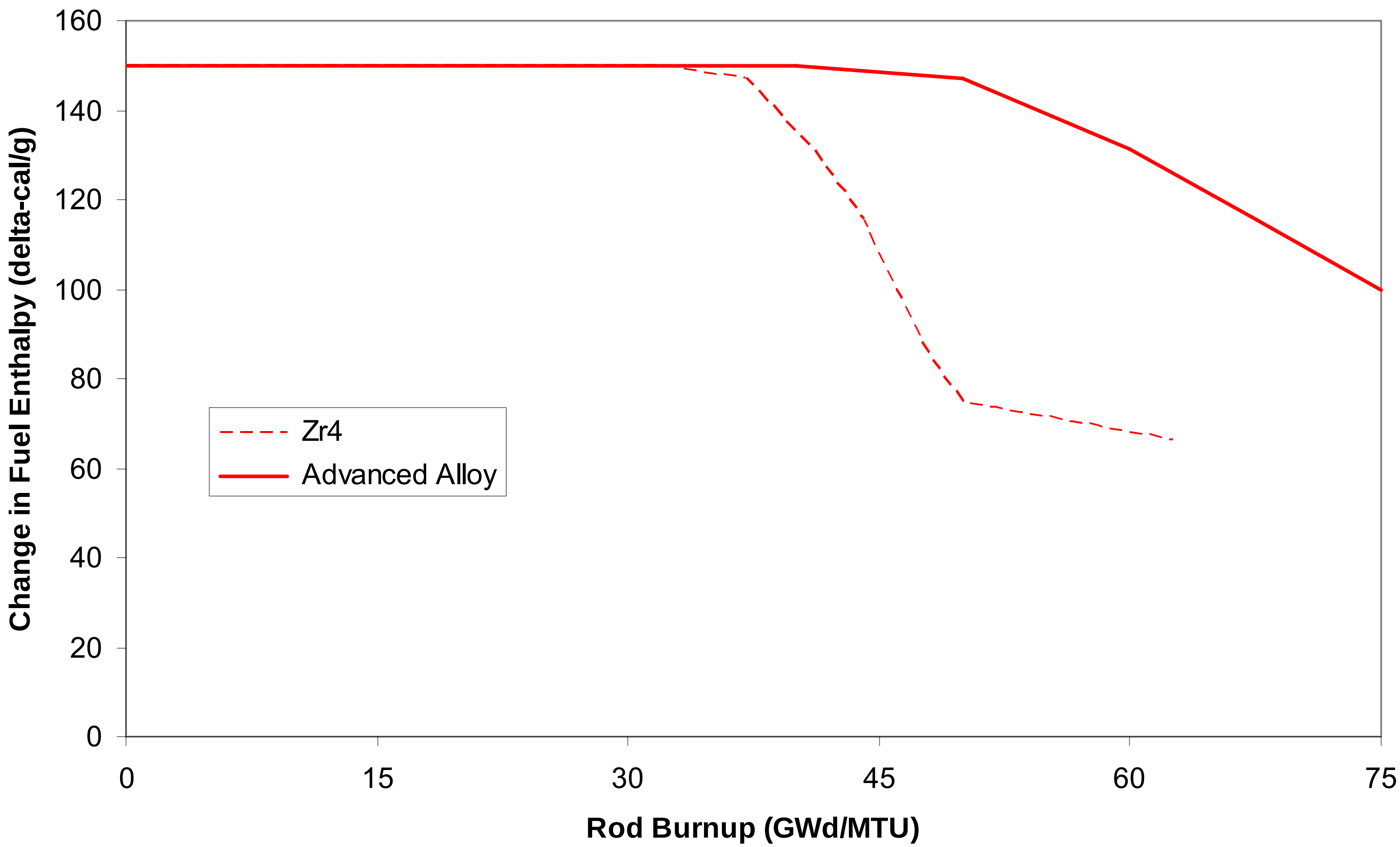
Dose calculation based on total number of failed fuel pins.

PWR Fuel Rod Corrosion Model



Sample PWR PCMI Failure Criteria

[Oxide/Wall Converted to Rod Burnup based on Modern PWR Design]



Industry Discussion

- BWR cold zero power scenarios and the effect of temperature on cladding failure threshold.
- The range of BWR coolant temperature and flow values at conditions during which Control Rod Drop Accident may occur.
- Fuel failure based upon post-Departure from Nucleate Boiling (dryout) conditions and the use of current methods and critical heat flux correlations to predict cladding failure.
- Hydrogen pickup fraction in BWR cladding versus PWR cladding.
- The accuracy of predicted oxide thickness and use of best-estimate peak nodal oxide to convert corrosion-dependent criteria to a more useable burnup dependent criteria.

Radiological Guidance

Draft Regulatory Guidance

Appendix B to RG 1.77 (1974) provides guidance and assumptions for evaluating the radiological consequences of a control rod ejection accident. These assumptions are supplemented by guidance given in RG 1.183 (2000) and RG 1.195 (2003).

RG 1.77 Appendix B, Assumption 1.c:

"*The amount of activity accumulated in the fuel-clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life, assuming continuous maximum full power operation.*"

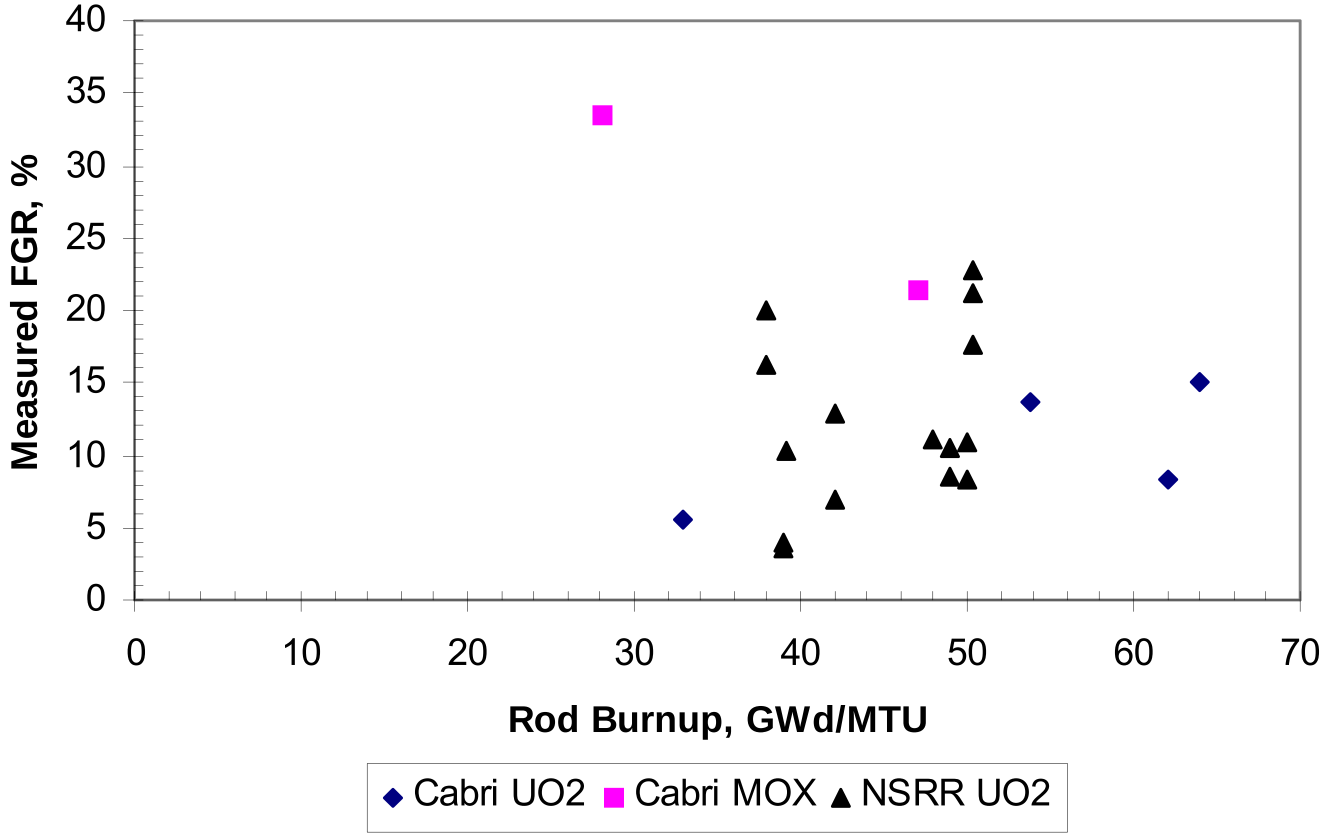
Fission Product Inventory

Total FP Inventory =	Steady-State Gap Fraction	+	Transient-Induced FGR from Pellet

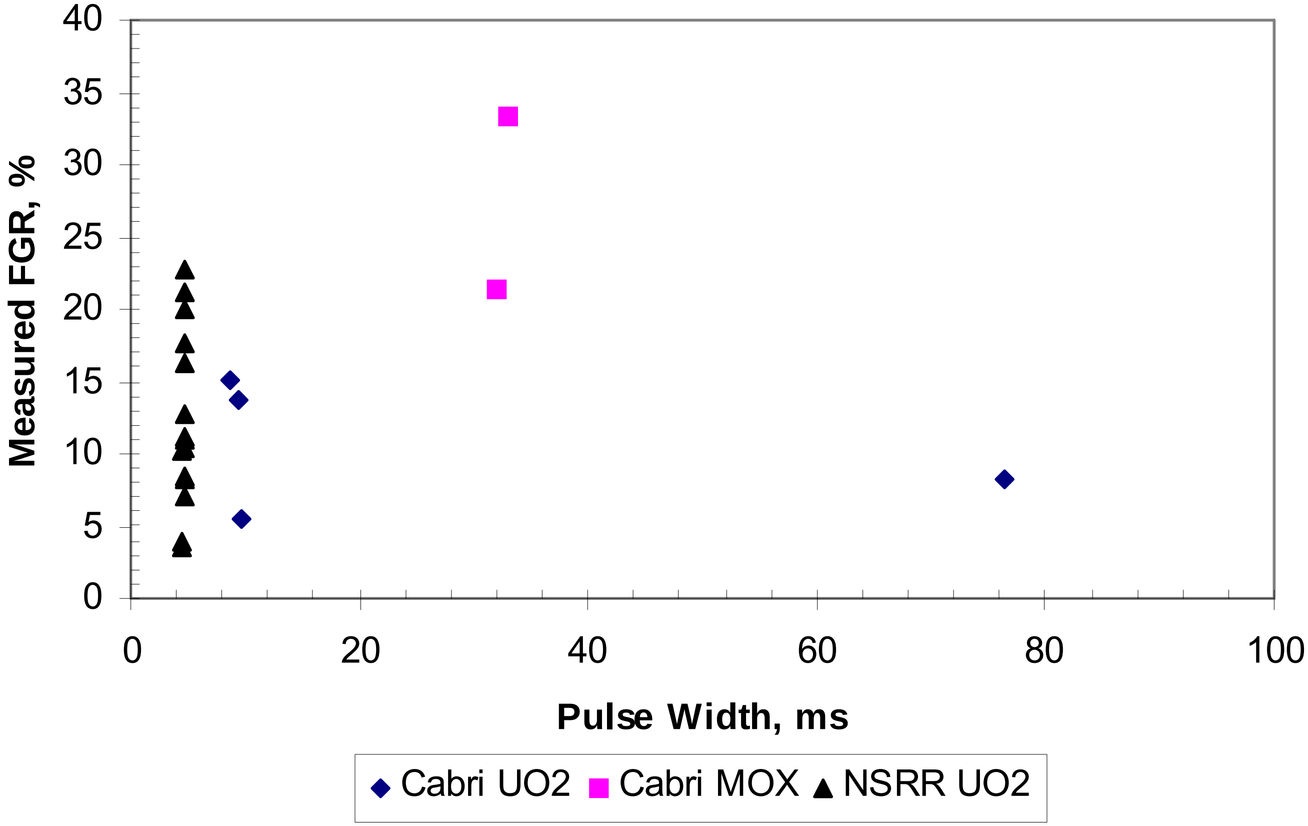
Where:

- Steady-state fission-product gap inventory governed by diffusion and related to power history.
- Transient-induced FGR mechanisms include pellet fracturing and grain boundary decohesion.
- Amount of transient release dependent on local burnup (fission gas accumulation along grain boundaries and within the porous rim region) and local power increase.

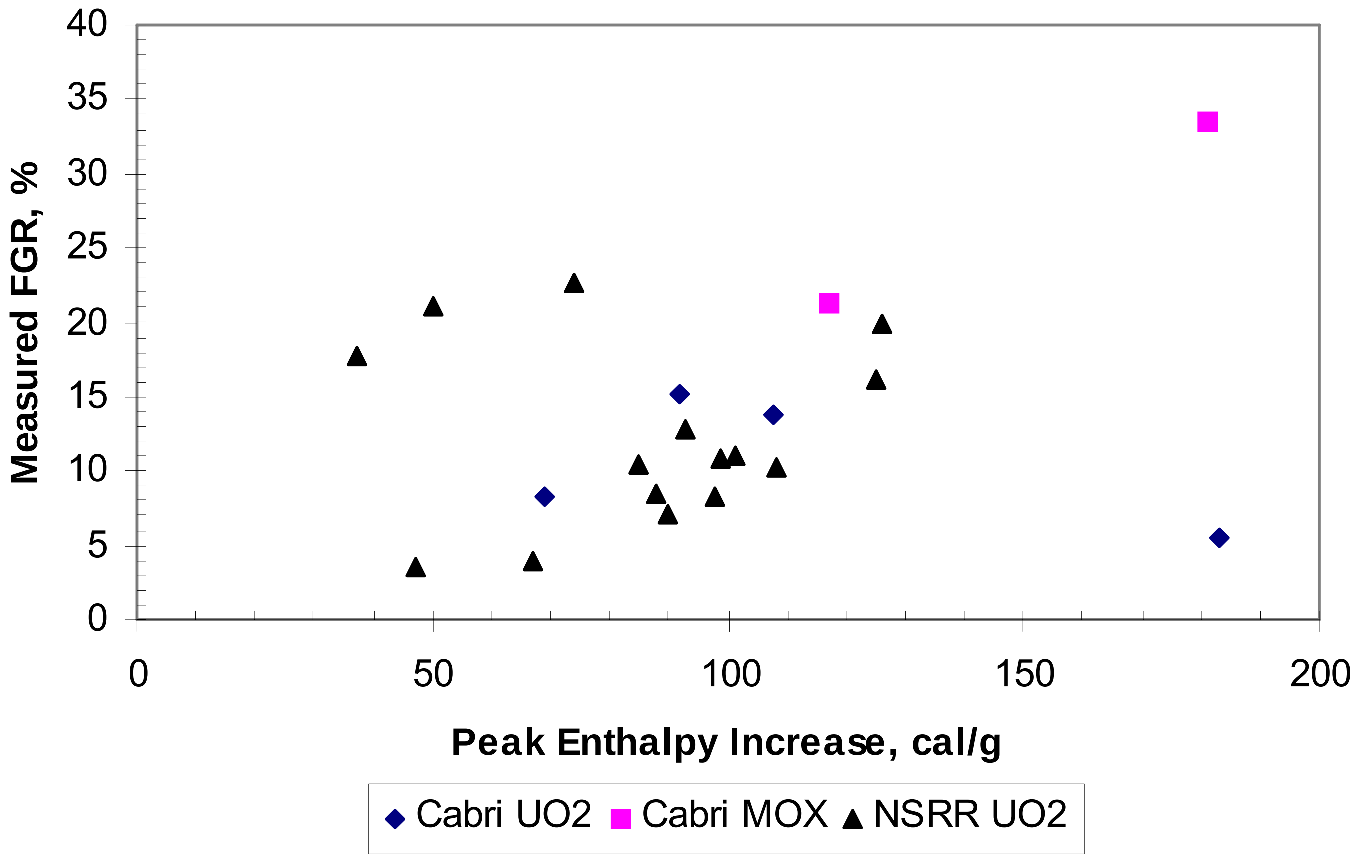
RIA FGR Measurements



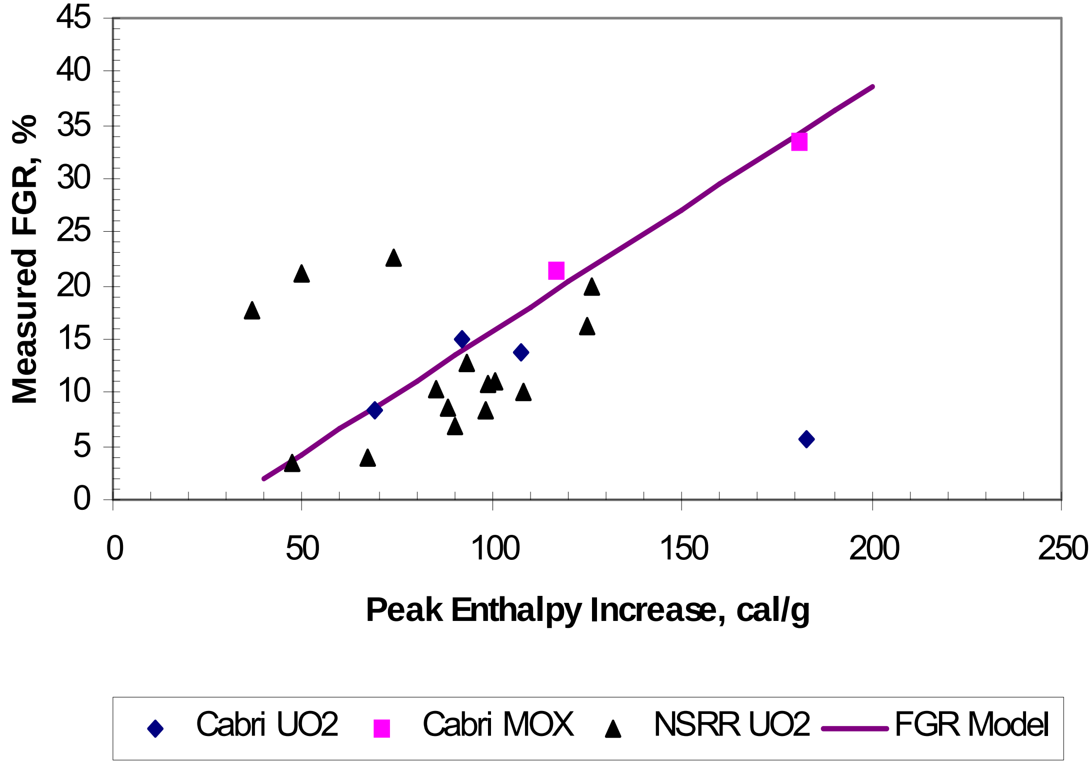
RIA FGR Measurements



RIA FGR Measurements



Transient FGR Correlation



Transient FGR (%) = 0.229*ΔH – 4.09

Draft Regulatory Guidance

RIA scenarios to be evaluated:

- PWR hot zero power, intermediate powers, and full power conditions.
- BWR hot zero power, intermediate powers, and full power conditions.
- BWR cold startup conditions.

Industry Discussion

- Transient-induced fission gas release (from the pellet) and radiological source term.
- Available analytical methods and implementation of burnup and corrosion dependent criteria.

Conclusions

1. Coolability criteria more restrictive than current 280 cal/g (radial average).
2. Fuel failure criteria more restrictive than current 170 cal/g (radial average).
3. Fission-product inventory more severe.
4. More detailed analytical techniques required over entire plant LCO.