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How Industry Guide (Rev5) addresses NRC comments

RFP-NRC Meeting

December 6, 2000

Terry Rieck (Exelon)

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NRR Comments on Industry Guide

- General Comments
  1. *How will new limits or analysis methods that are not currently used but which may be needed for burnup extension be identified? For example: a limit on corrosion and spallation*
     - Review General Design Criteria to identify consequences the fuel limits are designed to guard against
     - Extensive technical experience from experimental and PIE programs such as Halden, CABRI, RFP, etc.
     - Develop integration matrix between fuel limits in SRP 4.2 and phenomena

Attachment 3

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Phenomena/Limit Cross-Reference Table

Table 1. Cross Reference of Phenomena and Fuel Design Limits

Comprehensive list of Fuel Behavior Impacted by Burnup	**Fuel Design Limits that address the burnup-related fuel behavior** (the numbers refer to Table 0-1 of the Industry Guide Rev.5 or Table 2 below)
Fuel Thermal Conductivity	1.1, 1.2, 1.3, 1.10 2.5, 2.6, 2.7 3.2
Cladding Oxidation	1.1, 1.2, 1.3, 1.5, 1.6, 1.7, 1.9, 1.10 2.7, 2.8, 2.9 3.1, 3.4, 3.5
Swelling	1.1, 1.2, 1.3, 1.10 2.2
Irradiation Growth	1.8, 1.9, 1.11
Etc...	Etc...

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NRR Comments on Industry Guide

- General Comments
  2. *In areas where data are needed, data should be developed up to the requested target burnup with prototypical operating conditions including power distributions and power histories.*
     - references to data from fuel rods that are required to demonstrate compliance will indicate that the operating conditions should be prototypical of the application

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NRR Comments on Industry Guide

- General Comments (cont'd)
  3. *There are several types of data that will be needed including data to (1) establish the effect of burnup on a particular parameter or phenomena, (2) data to justify an existing criterion or justify revised criteria, and (3) data to show compliance with the criteria. It would be useful to include tables differentiating between data needed to justify or establish criteria and data needed to verify compliance with criteria. These tables should include the type of data needed, where and how it will be obtained, the time frame for acquisition and analysis, the amount of data, the test conditions and other pertinent details.*
     - Industry Guide is focused on
       - data to establish impact of burnup (1)
       - data to justify existing or revised criterion (2)
     - Vendor submittals should focus on
       - data to show compliance with the criteria

Table 2. Data to Support Burnup Extension to 75 GWd/tU

Design Basis Limits/Criteria	Data to Justify Limit at 75 GWd/tU	Data to Demonstrate Compliance
1. Fuel System Damage		
1.1 Design Stress	Yield Strength/Ultimate Tensile Strength	Yield Strength/Ultimate Tensile Strength
1.2 Design Strain	Uniform Elongation/Total Elongation	Uniform Elongation/Total Elongation
1.3 Strain Fatigue	Use Existing Fatigue Data	Use Existing Fatigue Data
1.4 Fretting Wear	Use Existing Fretting Wear Data	Use Existing Fretting Wear Data
1.5 Oxidation	TBD	Maximum Oxide Thickness
1.6 Hydriding	TBD	Hydrogen Content
1.7 Crud	TBD	Crud Thickness
1.8 Rod Bow	Not Required	Rod-to-Rod Gap Closure
1.9 Irradiation Growth	Not Required	Fuel Rod/Assembly Length
1.10 Internal Gas Pressure	Not Required	Internal Gas Pressure
1.11 Hydraulic Lift Loads	TBD	Holddown Spring Force
2. Fuel Rod Failure		
2.1 Internal Hydriding	Not Required	Not Required
2.2 Cladding Collapse	Not Required	Not Required
2.3 Fretting	See Above	See above
2.4 Overheating of Cladding	Not Required	Vendor Flow Loop Tests (Maybe not necessary)
2.5 Overheating of Fuel Pellets	Halden/?	Fuel Centerline Temperature Calcs
2.6 Excess Fuel Enthalpy	CABRI	Fuel Enthalpy Calcs
2.7 Pellet/Cladding Interaction	Uniform Elongation/Total Elongation	Uniform Elongation/Total Elongation
2.8 Clad Rupture	EdF-Edgar and ANL	TBD
2.9 Mechanical Fracturing	Yield Stress	Yield Stress
3. Fuel Coolability		
3.1 Cladding Embrittlement	EdF/JAERI/ANL	Quench Tests
3.2 Violent Expulsion of Fuel	CABRI	3-D Neutronic Calc's
3.3 Generalized Clad Melting	Not Required	Not Required
3.4 Fuel Rod Ballooning	EdF-Edgar and ANL	TBD
3.5 Structural Deformation	Yield Stress	Yield Stress

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NRC Comments on Industry Guide

- Specific Comments
  - Addition of bounding power history in several locations
  - Rearrangement of Regulatory Requirement for Excessive Fuel Enthalpy
  - Modifications to discussion summarizing impact of hydrides on cladding mechanical properties

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Status of Industry Guide Document

- Document Revisions
  - Rev. 0 - June 1999
  - Rev. 1 - July 1999
  - Rev. 2 - September 1999
  - Rev. 3 - January 2000
  - Rev. 4 - March 2000
    - Interim report released for NRC review/comment
  - Rev. 5 - October 2000
    - Update of interim report with NRC comments addressed

Review of:
- 1.1 Design Stress
- 1.10 Rod Internal Pressure
- 2.6 Excessive Fuel Enthalpy
- 3.2 Violent Expulsion of Fuel

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Future Plans

- Future Document Releases
  - Rev. 6 - February 2001
    - Planned release of interim report to NRC
    - Four limits (1.1 - 1.10 - 2.6 - 3.2) completed
  - Draft Final Document - September 2001

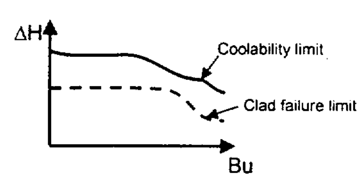
Proposal for RIA fuel failure and core coolability limits

RFP proposal to NRC
December 6, 2000

Nicolas Waeckel
Robert Montgomery
Rosa Yang

Outlines

- Background
  - Regulatory basis
  - RIA database
    - » integral tests and test conditions
    - » separate effects tests
  - Current understanding
    - » fuel clad failure
    - » fuel dispersal and coolability
    - » post-DNB type of failure needs not be considered
- Approach
- Proposal
    - » fuel clad failure criterion
    - » coolability criterion
- Summary



Attachment 4

Regulatory Background

- Core Coolability Limit for Reactivity Initiated Accident
  - Satisfy the requirements of General Design Criterion 28
    - » no damage to the reactor coolant pressure boundary greater than limited local yielding
    - » maintain the capability to cool the core
  - The **maximum radially average fuel enthalpy** should be less than 280 cal/gmUO_2 (Regulatory Guide 1.77) to maintain rod geometry and to avoid damaging pressure pulses
    - » Limit based on RIA tests performed in CDC-SPERT and TREAT on unirradiated rods and reported in terms of **radially average total energy deposition**
  - Later assessment by MacDonald, et.al. show
    - » that the limit should be 230 cal/gmUO_2 (instead of 280 cal/gm UO2) if expressed as maximum radially average fuel enthalpy
    - » Large pressure pulses and high energy conversion ratios (>1%) occur above 300 cal/gmUO_2

Regulatory Background

- Fuel failure threshold for Hot Zero Power (HZP) RIA
  - Established to define fuel failures for radiation dose limit calculations as required by 10CFR Part 100
  - Regulatory Guide 1.77
    - » PWR - Departure from Nucleate Boiling (DNB)
  - Standard Review Plan Section 4.2
    - » BWR - radially average peak fuel enthalpy greater than or equal to 170 cal/gmUO_2
    - » Criterion established as a surrogate for Critical Power Limit

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Fuel Failure Criterion Below 40 GWd/tU

- Analytical evaluations and test data show PCMI failure threshold exceeds 170 cal/gm for ductile cladding material
  - Irradiated cladding total elongation greater than 4 or 5% below 40 GWd/tU (Prometra data [1])
    - » maximum strain is less than 2-3% for 170 cal/gm [2],[3]
  - High integrity cladding can withstand 200 cal/gm
    - » REP Na-2 survived with more than 3% of strain [2],[3]

⇨ Use of 170 cal/gm as a limit for fuel clad failure below 40 GWd/TU is conservative

(1) M. Balourdet AND al. ANS Fuel Performance Topical Meeting, Portland April 1997
(2) F. Schmitz, J. Papin, Nuclear Safety, Vol 37, No. 4, 1996
(3) EPRI-Anatech Evaluation of irradiated fuel during RIA simulation tests TR-106387 Aug, 1996

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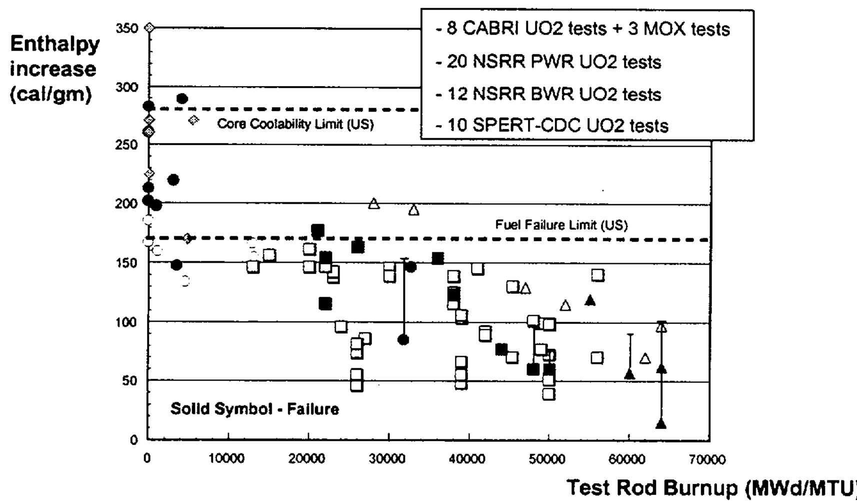
RIA-Simulation Test Database



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Test Conditions vs. LWR

	SPERT-CDC	NSRR	CABRI	LWR
Coolant Conditions				
Type	Stagnant Water	Stagnant Water	Flowing Sodium	Flowing Water
Temp (°C)	25	25	280	280 - BWR 290 - PWR
Pressure (atm)	1	1	3	70 - BWR 150 - PWR
Pulse Characteristics				
Full-Width Half Max. (msec)	13 to 31	4.5 to 6.6	10 natural 30-80 pseudo	25 to 90
Deposited Energies (cal/gm)	160 to 350	20 to 200	100 to 200	TBD

⇨ Need analytical tools and separate effect tests to assess tests results and compare to LWR conditions

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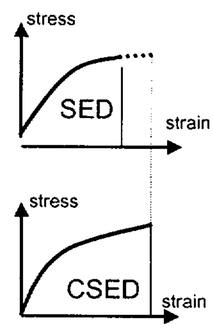
Separate effect tests

- Clad mechanical property tests
  - cladding ductility and CSED assessment
- Thermal hydraulic tests (Patricia)
  - critical heat flux and heat transfer coefficient during RIA transients

Clad mechanical property data

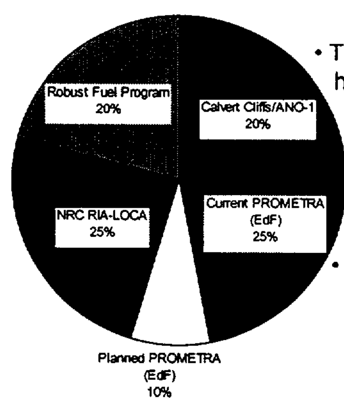
- SED is a measure of loading intensity on the cladding
  - SED is a calculated response parameter, based on integrating stress and strain
- CSED is a measure of cladding failure potential or cladding residual ductility
  - CSED is determined from mechanical property tests
  - depends mainly on H level, temperature and materials
- Cladding failure occurs when SED reaches the CSED for a given clad material



(1) R. Yang and Al, ANS Fuel Performance Topical Meeting, Park City, March 2000

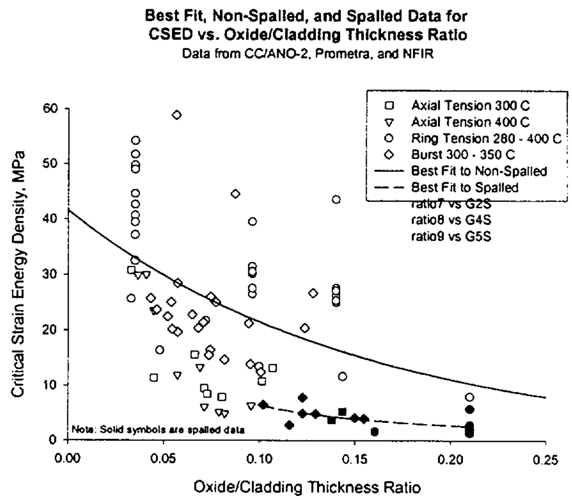
(2) Joe Rashid and Al, Windermere Meeting, June 2000

Separate effect tests- Clad mechanical properties



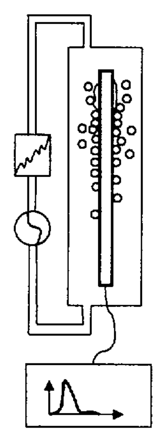
- Temperature and hydrogen effects
- Hydrogen and strain rate effects

CSED vs. Oxide Thickness for Zr-4



Separate effect tests- Transient heat exchange coefficient

- Out of pile thermal hydraulic experiments at CEA (Patricia test program) (1)
  - simulate an RIA-like pulse on a cladding tube in PWR conditions
  - study the influence of the kinetics on the heat exchange coefficients
- Outcomes:
  - no kinetic effect on Critical Heat Flux (CHF) value
  - no kinetic effect on post-DNB heat transfer coefficient
    - » current code correlations are valid
    - » clad temperatures are properly calculated



(1) Patricia test program. Synthesis report. IPSN Setex/ LTDF98/05 01-1999 T. Oulman

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Current understanding of RIA mechanisms

- Clad failure mechanism is Pellet-Clad Mechanical Interaction (PCMI) resulting from fuel thermal expansion and fuel matrix fission gas swelling
  - cladding **ductility** is the key determining factor
- Fuel rod failure depends mainly on cladding ductility NOT on burnup
  - corrosion/hydriding and fuel duty define clad residual ductility
  - spalled rods have significantly less ductility than non-spalled rods
    - » CABRI database shows NO fuel failure up to 64 GWd/TU for non-spalled rods
  - higher failure threshold expected for advanced alloys

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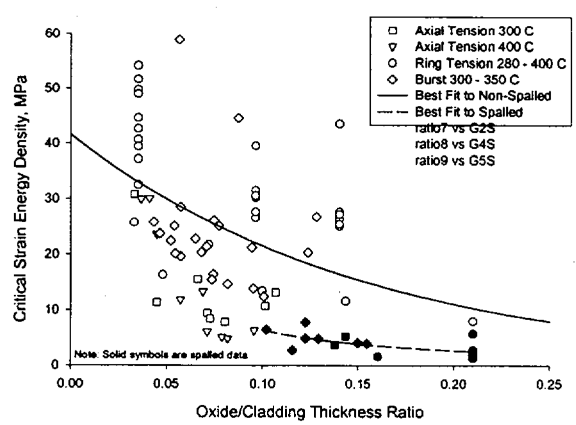
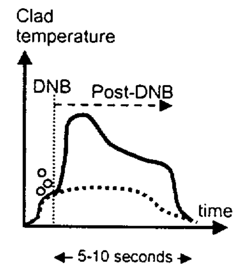
CSED vs. Oxide Thickness for Zr-4

Best Fit, Non-Spalled, and Spalled Data for CSED vs. Oxide/Cladding Thickness Ratio
Data from CC/ANO-2, Prometra, and NFIR



Is DNB a suitable failure limit ?

- DNB does not result in fuel failure
  - » DNB is NOT a failure mechanism
  - » Represents a transition from high to low heat transfer rates
  - » Cladding surface temperature excursion (potential for burnout)
  - » Used as a conservative limit for cladding failure



- Failure by post-DNB operation is by two modes :
  - (1) Oxidation-induced embrittlement
  - (2) Ballooning and burst

Failure by DNB during an RIA event

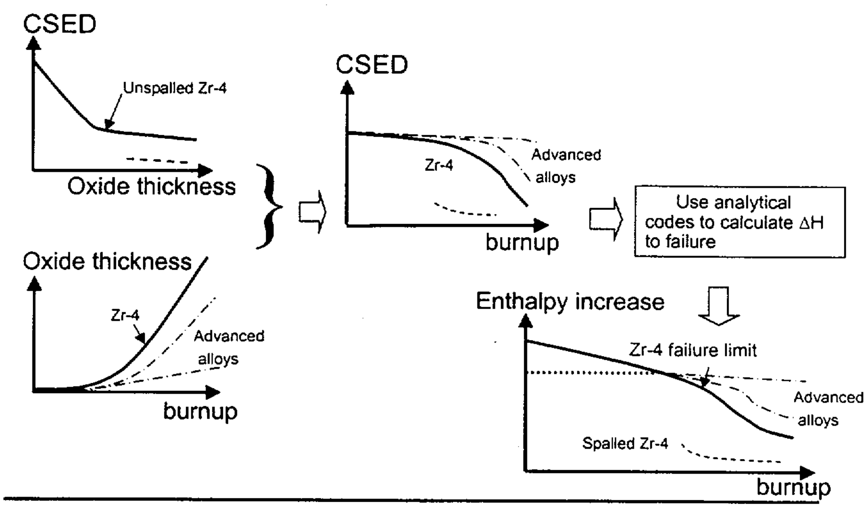
- Industry Position
  - Post-DNB failures need NOT be considered for irradiated fuel
    - » Below 170 cal/gm cladding temperatures are too low to produce failure by oxidation-induced embrittlement [1]
    - » Insufficient internal gas pressure to produce large ballooning and rupture deformation [2], [3]

(1) NUREG-0562

(2) Ishikawa and al. International colloquium on irradiation tests for reactor safety programs June 25-28, 1979

(3) N. Waeckel and al. ANS Fuel Performance Topical meeting Park City, March 2000

Fuel failure criterion: proposed approach
How to link clad ductility to burnup ?



Fuel failure criterion: Analytical approach

1) Cladding ductility (CSED, measured by mechanical property data) is a function of oxide thickness (H concentration), NOT burnup
2) Oxide thickness can be correlated with burnup through power history and alloy-specific correlation
3) From 1) and 2), clad ductility (CSED) can be related to burnup
4) SED-CSED criterion and analytical codes (FALCON, SCANAIR or FRAPTRAN) are used to calculate the enthalpy increase (ΔH) that results in fuel failure as a function of burnup
   - consistent with current criteria (ΔH versus burnup)
   - clad ductility taken into account
   - different curves for different alloys

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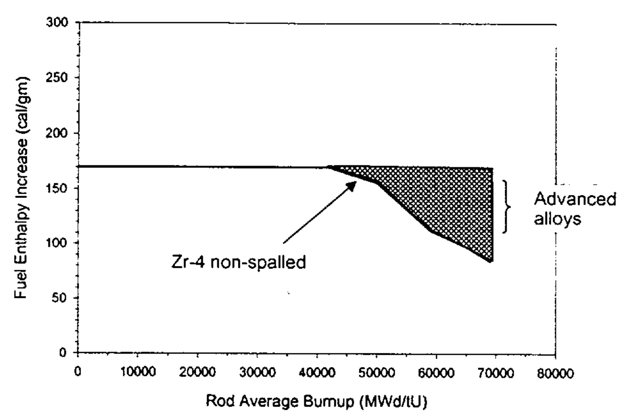
Fuel failure limit



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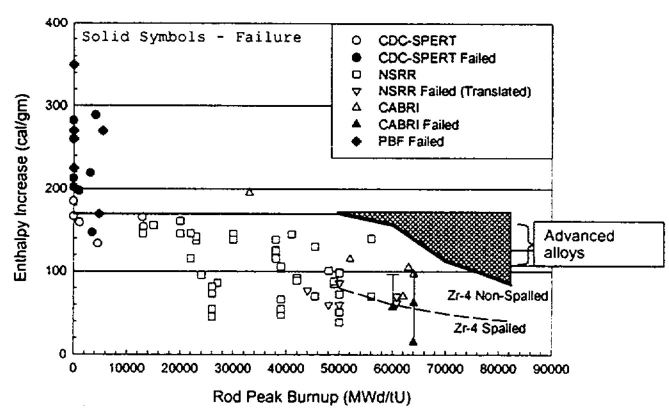
Proposed failure limit bound all RIA Test Data

(NSRR Failures have been translated to 300 °C)



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Current understanding of fuel dispersal and related core coolability issue

- Fuel dispersal
  - risk may increase above 40 GWd/T due to rim formation in fuel pellets
  - rim particles dispersal may occur during power pulse following cladding failure
    - » flow blockage and loss of coolable geometry ?
    - » pressure pulse generation and threat on pressure vessel integrity ?
- Data show that potential for fuel dispersal is a function of :
  - energy deposition following cladding failure
  - pulse width
- NO fuel dispersal observed experimentally in RIA simulation tests with pulse widths > 20 ms
  - representative LWR pulse widths ~25-90 ms

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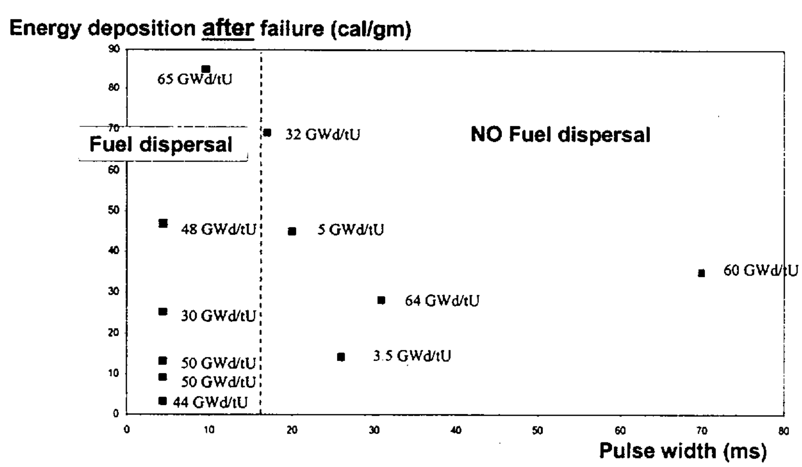
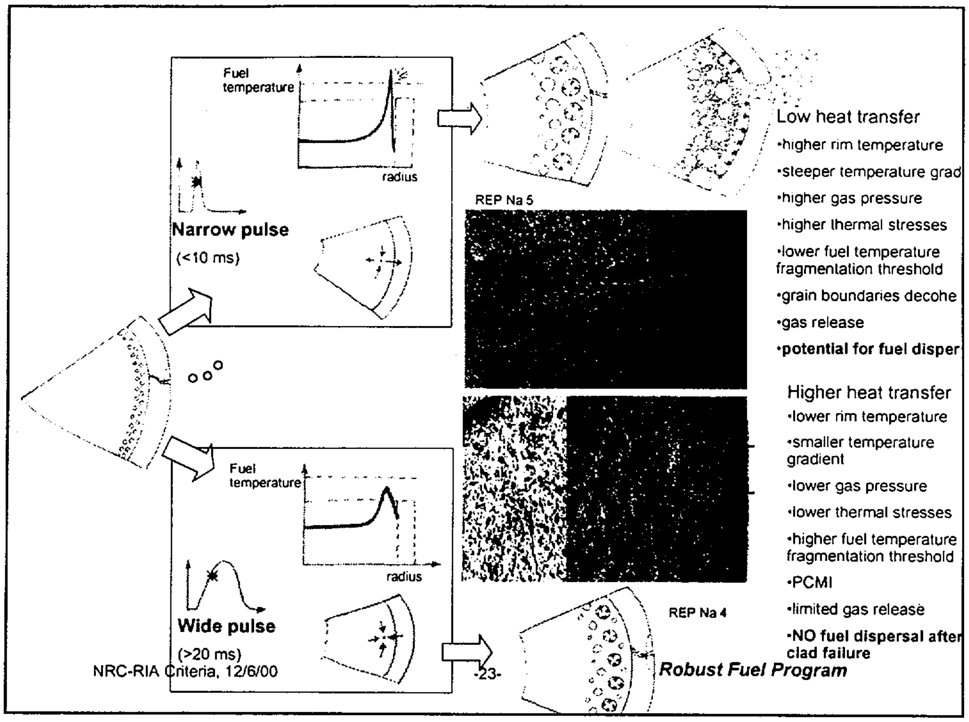
Pulse Width Effect on Fuel Dispersal





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How to define the coolability limit?

- Below 40 GWd/tU coolability is controlled by high temperature behavior (melt response) of fuel and cladding
  - » Molten cladding can lead to loss of rod geometry
  - » Molten fuel increases fuel coolant interaction kinetics
- Above 40 GWd/tU, fuel rim material dispersal may occur if sufficient energy is injected following cladding failure. Coolability may be impaired by two phenomena:
  - » loss of **coolable geometry** due to a large amount of dispersed material and/or massive clad fragmentation
  - » mechanical energy release (**pressure pulse**) in the coolant that may affect the pressure vessel integrity.

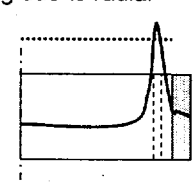
How to address coolability for high burnup fuel?

- NO fuel dispersal is expected for prototypical pulse widths
- At high energy or for narrow pulse, small amount of pellet material may be dispersed through failure opening but has low impact on:
  - Coolable geometry
    - » experimental data (NSRR) show less then 10% of pellet material loss - mostly from rim region (1)
    - » rod geometry is maintained in all cases (1)
  - Pressure pulse
    - » Tests exhibited low mechanical energy conversion (1)
      - temperature of dispersed material lower than melting
      - limited amount of material

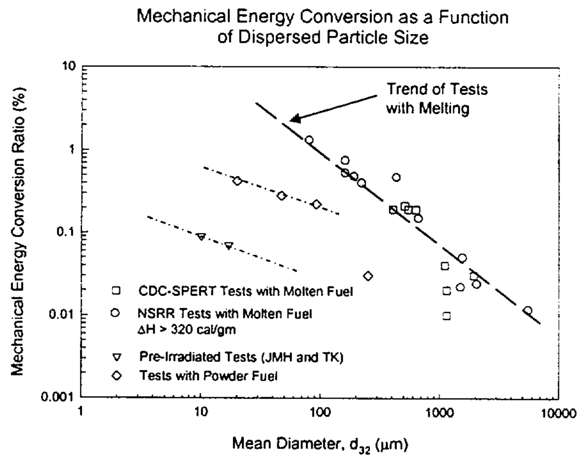
(1) T. Sugiyama and al. "Mechanical energy generation during high burnup fuel failure under RIA conditions". Journal of Nuclear Sciences and Technology, Vol 37, No. 10 October 2000

A coolability limit based on melting is proposed

- Data show molten fuel produce higher mechanical energy conversion ratios
  - Incipient melting in JMH-5 Test at 210 cal/gm and 30 GWd/tU show no adverse impact on coolable geometry or pressure vessel integrity
- To use incipient fuel melting as a precursor for coolability limit is very conservative
  - Maintains clad temperatures below melting to ensure rod geometry
  - Small region of high burnup fuel near incipient melting due to radial temperature peaking
    - » Majority of fuel well below peak temperature
  - Limits mechanical energy conversion ratio

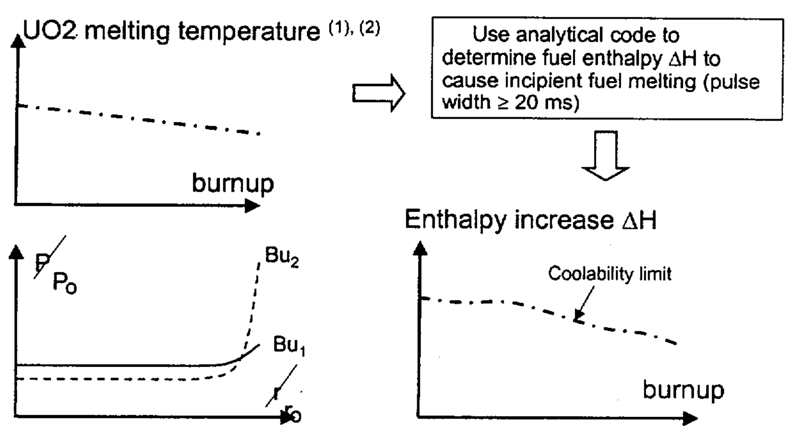


RIA Tests FCI Data



(1) T. Sugiyama and al. Journal of Nuclear Science and Technology, Vol 37, No 10, Oct 2000

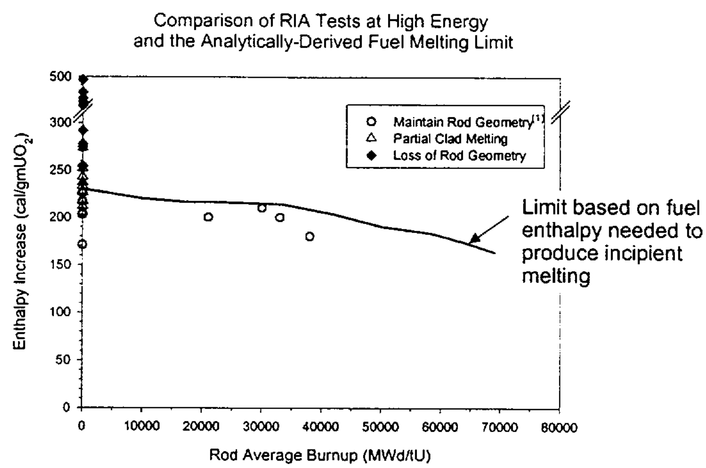
RIA coolability limit based on energy to incipient fuel melting vs burnup- Proposed approach



(1) Y. Philipponeau CEA technical Report LPCA n0 27

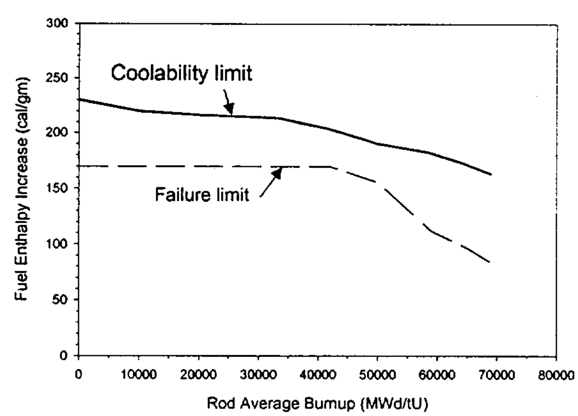
(2) J. Komatsu and al Journal of Nuclear Materials n0 154, vol 38 (1988)

Comparison to High Energy Tests

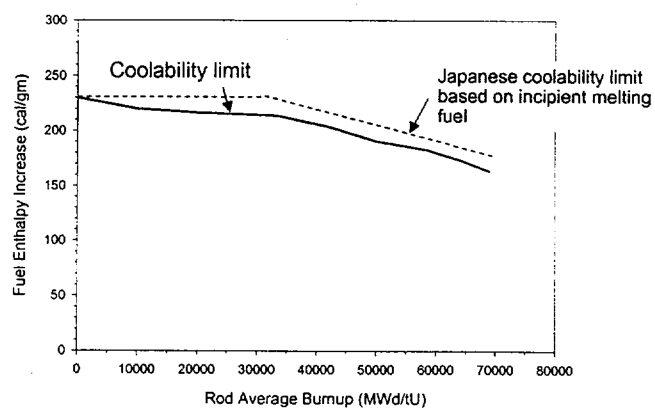


(1) T. Sugiyama and al. Journal of Nuclear Science and Technology, Vol 37, No 10, Oct 2000

Industry has proposed 2 separate limits



Comparison to Japanese coolability limit



Summary (1)

- Proposed clad failure and coolability limits as a function of burnup
- Limits are given in terms of enthalpy increase
  - directly usable for core reload designs
  - consistent with current practice
  - incorporated key controlling parameters (corrosion/hydriding)

Summary (2)

- Fuel Failure Criterion
  - Based on integral tests results, mechanical properties tests data and analytical approach
  - PCMI based failure mechanisms for HZP Rod Ejection Accident (REA)
    - » DNB occurs after PCMI completed and needs NOT be considered
    - » limit based on DNB remains valid for Hot Full Power REA
  - Cladding ductility is the controlling factor for HZP REA
    - » confirmed by the database
  - Limit represents upper bound of data for no-fail tests on non-spalled Zr-4 test rods
    - » Limit based on Zr-4 is a lower bound for advanced alloys
      - a material with a lower H pick-up ratio will contain less H for a given corrosion level and will exhibit a higher residual ductility and a higher failure limit
      - confirmed by recent M5 test

Summary (3)

- Coolability limit
  - The proposed limit is based on the enthalpy increase (ΔH) necessary to cause incipient fuel melting as a function of burnup
    - » the limit is supported by data for both loss of coolable geometry and mechanical energy release issues
    - » the limit is conservative
      - some RIA simulation tests at the limit level did NOT exhibit any incipient fuel melting suggesting melting temperature is higher
      - proposed limit is lower than JAERI licensing limit

Status of the Industry Collaborative Effort to Develop 3D Rod Ejection Analysis Methodology

EPRI RFP WG#2
REA 3D Methodology Focus Group

G. B. Swindlehurst
Duke Power Company

EPRI RFP / NRC Meeting
December 6, 2000

Objectives

- Develop a generic PWR REA 3D analysis guideline for the cal/gm acceptance limit
- Applicability to all U.S. PWR designs
- Meet future delta cal/gm acceptance limits
- Determine an appropriate level of conservatism based on the risk significance of REA .
- Independent of computer codes
- Do not impact core design strategies and economics
- Support industry goals to achieve higher burnup
- Reasonable scope of analysis and resources
- Less licensing effort by organizations performing REA analyses by establishing a generic method as a standard
- Facilitate NRC review and efficiency in the licensing process

Attachment 5

RFP REA 3D Focus Group Membership

- Dominion Generation
- Duke Power
- EdF
- Framatome Cogema Fuels
- Nuclear Management Company
- Westinghouse
- Westinghouse (CE-ABB)

- Duke Engineering & Services (contractor)

Approach

- Review existing licensed REA analysis methodologies
- Discuss and consider elements of Reg. Guide 1.77 "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs"
- Discuss and consider elements of SRP Section 15.4.8
- Discuss progress and outcomes of industry PIRT on REA
- Discuss use of probability-based arguments
- Discuss key physics parameters
- Discuss uncertainties in key parameters
- Discuss use of statistical methods
- Consider insights from REA literature

Approach (cont.)

- Perform sensitivity analyses to obtain insights and trends
- Discuss off-normal core conditions
- Discuss related Technical Specifications issues
- Discuss concepts for documenting the methodology
- Discuss NRC review of the methodology
- Discuss need for and benefits of demonstration analyses
- Discuss absence of new acceptance limits (normally these are known prior to developing a methodology)
- Periodic discussions with EPRI RFP WG#2 to obtain input

These were accomplished in several meetings and telecons

Methodology Overview

- Limit the scope to REAs that can result in significant delta-cal/gm results. These are mainly zero/low power critical initial conditions.
- Methodology elements related to the other REA acceptance limits (DNBR, pressure, doses) not included since not related to the new concern regarding high burnup effects
- A simplified probability-based method for determining the range of core initial conditions to be analyzed
- A simplified method for addressing the effect of post-trip xenon on the initial condition
- Deterministic and statistical approaches

Probability-Based Method for Determining Initial Core Conditions

Only consider REA sequences that can result in a significant delta cal/gm result, and that have a frequency of >1E-7/yr. Other sequences are dropped from consideration.

- P-total = P-rea X P-zp X P-hw X P-ic (Must be >1E-7/yr)
- P-rea: The frequency of a rod ejection accident per year. This is based on zero events in PWR world history.
- P-zp: The frequency of being at critical zero/low power.
- P-hw: The probability of an ejected rod having a high enough worth to result in a significant delta cal/gm result
- P-ic: The frequency of an off-normal core initial condition that would result in higher delta cal/gm results

Probability-Based Method for Determining Initial Core Conditions (cont.)

Example Calculation

P-total = P-rea X P-zp X P-hw X P-ic (>1E-7/yr)

P-rea = 2.5E-4/yr This is a mean value based on zero occurrences in 3362 calender years of PWR operation, assuming pressurized and capable of a rod ejection accident 60% of the time.

P-zp = 2.8E-3/yr This is based on 24 hours per year during which the core conditions are critical and zero/low power. This can be calculated separately for initial startup following refueling and for all other critical and zero/low power conditions.

P-hw = 0.17 This is based on 9 of 53 control rods capable of resulting in a significant delta cal/gm result.

P-ic = 1E-7 / 1.2E-7 = 0.83 An off-normal core initial condition must have a probability of >83% at critical and zero/low power conditions to be considered in the REA analysis.

Post-trip Xenon Condition

- The core xenon distribution can have a significant effect on the ejected rod worth and the transient core power distribution. This can significantly increase the delta cal/gm result.
- Xenon conditions will be very different depending on the core conditions prior to the critical and zero/low power conditions of interest.
- For the HZP-BOC REA case following a refueling outage, no xenon will be present
- For other critical and zero/low power conditions, which mainly consist of restarts following a reactor trip, credit the minimum number of hours following a reactor trip during which the core is maintained subcritical. The xenon conditions during this subcritical time interval will be excluded from consideration.

Key REA Physics Parameters

The key REA physics parameters have been determined based on REA 3D kinetics experience and sensitivity studies

- Ejected rod worth in dollars (ERW$ = ERW / beta-effective)
- Fuel temperature (Doppler) feedback
- Moderator density feedback
- The core power distribution

Uncertainties

The uncertainties in the key physics parameters and the model will be included using one of two methods

First

- Establish the uncertainties for the key physics parameters (ERW$, fuel temperature feedback, moderator density feedback) based on the code/model
- Establish the uncertainty in the core power distribution code/model

Then

- Include the uncertainties in a conservative deterministic analysis approach OR
- Include the uncertainties in a statistical analysis approach

Statistical Method

The statistical approach will combine the uncertainties using the SRSS methodology

- A reference case will be run using nominal values for the key parameters. The result will be the reference delta cal/gm.
- A sensitivity case off the reference case will be run for each of the three key physics parameters, with the uncertainty included in the key parameter. For each of these three cases a delta-delta cal/gm result will be obtained by subtracting the reference delta cal/gm.
- The contribution of code/model uncertainty will be quantified by multiplying the reference delta cal/gm by the code/model uncertainty. This will produce a fourth delta-delta cal/gm value
- The statistical result will be the SRSS of the above four delta-delta cal/gm results added to the reference delta cal/gm value.

Typical REA Analysis Process

- An organization will employ a 3D transient neutronics code and a transient fuel rod heat transfer code (this capability may be within the 3D code)
- The reactor operating conditions in the critical and zero/low power range of interest will be defined
- The frequency and probability values needed to determine the range of initial core conditions will be determined
- The set of cases to be analyzed will be defined using the probability model
- The uncertainty values for the key physics parameters and the code/model will be determined
- The analysis will be performed for a bounding core design **OR** for a typical design, with either the deterministic or the statistical approach

Typical REA Analysis Process (cont.)

- The analysis results will be compared to the acceptance limits. This is expected to involve acceptance limits that are a function of burnup or some other parameter.
- If the acceptance limit is an indicator of cladding failure, then a pin census will be performed to quantify the percentage of failures.
- For subsequent core designs the validity of the analysis must be confirmed, or the analysis must be repeated. Note that if the core design does not include any potential ejected rods with sufficient worth to result in significant delta cal/gm results, then no analysis is required (i.e. P-hw = zero)
- Any changes in core operation or changes in values used in the probability model or the uncertainty parameter values must be evaluated.
- If unacceptable analysis results are obtained, then the core must be re-designed, or the operation of the core changed to achieve acceptable results.

Current Status and Future Plan

- Demonstration analyses are in progress
- The EPRI report is being written
  - Proposed revisions to R. G. 1.77
  - Proposed revisions to SRP 15.4.8
- The current intent is for the EPRI report to be submitted to the NRC for review
- NRC approval of the EPRI report would establish a standard methodology for optional use by the industry for REA analyses
- Organizations could then reference the approved EPRI report and identify all deviations from the standard method
- Vendor and licensee resources to implement will be reduced
- NRC resources to review will be reduced

Licensing Criteria for Fuel Burnup Extension Beyond 62 GWd/tU

"Industry Guide Development"

Status Report

Robert Montgomery

NRC-EPRI-NEI Meeting
December 6, 2000
Nuclear Regulatory Commission
Rockville, MD

Outline

- Status of Review and Documentation
  - Limits/Criteria that have been reviewed
- Example
  - Overheating of Fuel Pellets
- Future Plans

Attachment 6

Four Stage Review Process

(I) Establish Baseline for Current Fuel System Designs
(Requirements, Limits, Parameters, Methods, Measures and Data)

(II) Perform Burnup Effects Screening Evaluation
(Is there a burnup effect?)

(III) Perform Comprehensive Burnup Effects Evaluation
(How to address burnup effects?)

(IV) Establish New Baseline for Licensing Fuel System Designs for Extended Burnup
(Requirements, Limits, Parameters, Methods, Measures and Data)

Status of Review and Assessment

Design Basis Limits/Criteria	*Status*
1. Fuel System Damage	
1.1 Design Stress	Completed
1.2 Design Strain	Initiated
1.3 Strain Fatigue	Draft Completed
1.4 Fretting Wear	Draft Completed
1.5 Oxidation	Initiated
1.6 Hydriding	Initiated
1.7 Crud	Draft Completed
1.8 Rod Bow	Draft Completed
1.9 Irradiation Growth	Draft Completed
1.10 Internal Gas Pressure	Completed
1.11 Hydraulic Lift Loads	Draft Completed
2. Fuel Rod Failure	
2.1 Internal Hydriding	Draft Completed
2.2 Cladding Collapse	Draft Completed
2.3 Fretting	Draft Completed
2.4 Overheating of Cladding	
2.5 Overheating of Fuel Pellets	Draft Completed
2.6 Excess Fuel Enthalpy	Draft Completed
2.7 Pellet/Cladding Interaction	Initiated
2.8 Clad Rupture	
2.9 Mechanical Fracturing	
3. Fuel Coolability	
3.1 Cladding Embrittlement	
3.2 Violent Expulsion of Fuel	Draft Completed
3.3 Generalized Clad Melting	Draft Completed
3.4 Fuel Rod Ballooning	
3.5 Structural Deformation	

Limits/Criteria Reviewed Since Last Meeting

- Focused on areas that should have burnup independent criteria
  - Fuel System Damage
    - 1.3 Fatigue Strain
    - 1.4 Fretting Wear
    - 1.7 Crud
    - 1.8 Rod Bow
    - 1.9 Irradiation Growth
    - 1.11 Hydraulic Lift Loads
  - Fuel Rod Failure
    - 2.1 Internal Hydriding
    - 2.2 Cladding Collapse
    - 2.5 Overheating of Fuel Pellets
  - Fuel Coolability
    - 3.3 Generalized Clad Melting

2.5 Overheating of Fuel Pellets

- Stage I - Establish Baseline for Current Fuel Designs
  1. Application: Fuel rod failure during normal operation, AOO's and postulated accidents
  2. Standard Review Plan 4.2: It has also been traditional practice to assume that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and anticipated operational occurrences, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

EPRI

2.5 Overheating of Fuel Pellets

- Stage I - Establish Baseline for Current Fuel Designs (cont'd)
  3. Regulatory Requirement:
  – No fuel failures during normal operation and AOO's
    - Specified Acceptable Fuel Design Limit (SAFDL) - GDC 10, 12, 17, 20, and 25.
  – Number of fuel failures for postulated accidents
    - 10CFR100 radiation dose limits
  4. Design Limit:
    - maximum temperature will not exceed UO_2 melting temperature for normal operation and AOO's
    - number of rods exceeding UO_2 melting temperature are assumed failed for postulated accidents

EPRI

2.5 Overheating of Fuel Pellets

- Stage I - Establish Baseline for Current Fuel Designs (cont'd)
  5. Design Basis Approach:
    - Use fuel performance codes to calculate maximum fuel temperature during normal operation, AOO's and postulated accidents
    - Include effects of burnup and burnable absorbers
      – UO_2 thermal conductivity
      – Gap Conductance
      – Outer surface oxide and crud heat resistances
    - UO_2 melting temperature
      – Function of burnup and burnable absorbers
      – Various methods used (empirical models, penalty factors, etc)

EPRI

2.5 Overheating of Fuel Pellets

- Stage II - Burnup Effects Screening Evaluation
  - *Does burnup have an effect on the key parameter(s) or measures identified for the fuel limit?* - **Yes**
    - Key Parameters - maximum fuel centerline temperature
    - Influenced through two ways
    - Degradation of heat conduction in rod
      - UO_2 thermal conductivity, gap conductance, oxide and crud buildup
      - Important for postulated accidents
    - Power level restrictions to meet rod internal pressure and strain SAFDL's
      - high fission gas release
      - PCMI by pellet thermal expansion
      - appropriate for normal operation and AOO's

EPRI

2.5 Overheating of Fuel Pellets

- Stage II - Burnup Effects Screening Evaluation (cont'd)
  - *Does burnup have an effect on the current fuel limit?* - **No**
    - Design limit of no fuel failures by melting or assumed failure at melting is not burnup dependent

  - *Can the effect of burnup be addressed through expansion of current methods, processes, programs or data?* - **Yes**
    - Include effects of burnup on rod heat conduction
    - Evaluate design methods against applicable temperature measurement data at high burnup
    - Demonstrate that maximum fuel temperature limits do not exceed the melting temperature at extended burnup, e.g.
      - UO_2 melting temperature with burnup i.e., MATPRO
      - Maximum temperature set below actual melting temperature

2.5 Overheating of Fuel Pellets

- Stage IV - Assessment
  - For normal operation and AOO's
    - maximum achievable fuel temperature is limited by power restrictions to satisfy other fuel design limits
      - internal gas pressure and strain
      - well below melting temperature
  - For Analysis of postulated accidents
    - Methods should include the effects of burnup on the heat conduction in the fuel rod
    - Demonstrate that the maximum temperature limit used does not exceed the UO_2 melting temperature at target burnup

Future Activities

- Start Review of Limits Related to Cladding Mechanical Properties
  - 1.2 Design Strain
  - 1.5 Oxidation
  - 1.6 Hydriding
  - 2.7 Pellet-Cladding Interaction