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Summary of Fuel Safety Research Meeting 2005

March 2-3, 2005, Tokyo

Fuel Safety Research Group
Nuclear Safety Research Center

March 2006

Japan Atomic Energy Agency

日本原子力研究開発機構

Summary of Fuel Safety Research Meeting 2005
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Fuel Safety Research Group†

Nuclear Safety Research Center
Japan Atomic Energy Agency
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Fuel Safety Research Meeting 2005, which was organized by the Japan Atomic Energy Agency (Establishment of the new organization in Oct. 1, 2005 integrated of JAERI and JNC) was held on March 2-3, 2005 at Toshi Center Hotel, Tokyo. The purposes of the meeting are to present and discuss the results of experiments and analyses on reactor fuel safety and to exchange views and experiences among the participants. The technical topics of the meeting covered the status of fuel safety research activities, fuel behavior under Reactivity Initiated Accident (RIA) and Loss of coolant accident (LOCA) conditions, high fuel behavior, and radionuclide release under severe accident conditions. This summary contains all the abstracts and sheets of viewgraph presented in the meeting.

Keywords: Fuel Safety, RIA, LOCA, Severe Accident, High Burnup, Fuel Behavior

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燃料安全研究会合 2005 要約集
2005 年 3 月 2 日、3 日　東京

日本原子力研究開発機構安全研究センター
燃料安全評価研究グループ†

(2006 年 1 月 23 日受理)

3 月 2 日（水）及び 3 日（木）の両日、東京の都市センターホテルにおいて「燃料安全研究国際会議 2005（Fuel Safety Research Meeting 2005）」を開催した。本会議は、原子炉の安全性研究に関する最新の研究成果の発表と、専門家との情報交換及び討論を目的としている。本会議における技術的な話題は、燃料安全研究の現状、反応度事故時及び冷却材喪失事故時の燃料挙動、高燃焼度燃料のふるまい、及びシビアアクシデント時の放射性物質放出をカバーしている。本要約集は、本会議の発表において使用された要旨及び OHP をまとめたものである。

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Contents

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Outline of the FSRM 2005

Fuel Safety Research Meeting (FSRM2005) was held at Toshi Center Hotel, Tokyo on March 2-3, 2005. The technical scope of the meeting includes aspects of nuclear reactor fuel safety from performance in normal operation to behaviors during reactivity-initiated accident (RIA) and loss-of-coolant accident (LOCA) as well as radionuclides release from fuels under severe accident conditions. The meeting was held with seventy participants including ten from overseas. The eighteen presentations in the following six session were made and were comprised of ten from Japan Atomic Energy Agency (JAEA), and eight from foreign institutes.

Session1; General report

Outline of fuel safety research in U.S.A., on-going licensing issues and R&D study on high burnup fuel in France, fuel safety activities in Korea and recent progress in fuel safety research at JAEA were presented.

Session2; Fuel behavior under RIA condition

The results of recent NSRR experiments on high burnup LWR fuel under RIA conditions, analytical research on fission gas release during RIA conditions and fresh fuel rod experiment focused on hydride rim were presented by JAEA. Progresses of RANNS code simulating the fuel behavior during RIA were reported.

Session 3; High burnup fuel behavior

The NSRR experiment of simulated power oscillation in BWRs and progress of mechanical property evaluation on cladding by JAEA were reported. The 'cross-over program' which is to simulate a formation process of high burnup rim structure was introduced. Performance of fuel cladding material characterization for ALPS program was introduced by PSI.

Session 4; Fuel behavior under LOCA condition

Progress of LOCA experiment, second in-pile test and preliminary PIE was presented by Halden reactor project. Results of ring compression ductility screening test with advanced-alloy cladding and LOCA integral tests focused on post-quench ductility were introduced by Argonne National Laboratory. Recent results of LOCA experiment on high burnup fuel and, develop or modify test methodology to evaluate cladding performance and safety in LOCA were presented by JAEA.

Session 5; Radionuclides release under severe accident condition

Main results of VEGA (Verification Experiments of Radionuclides Gas/Aerosol Release) project on the radionuclide release were presented by JAEA. The general status and interpretation of results obtained by the VEROCORS program were introduced by IRSN.

Session7; JAEA's future program

The research program 'Advanced LWR Fuel Performance and Safety (ALPS)' was introduced. In the program, first NSRR pulse-irradiation test, VA-1, with 79 GWd/t Vandellos PWR fuel were performed in Feb. 2004. High burnup UO_2 and MOX fuels are subjected to variety of experiments, such as NSRR test and LOCA test.

The primary purposes of the meeting are to present and discuss results from fuel safety research program of JAEA, including RIA-simulating experiments in the NSRR (Nuclear Safety Research Reactor), LOCA experiments, VEGA program, and fuel behavior code development. Discussions among the meeting participants were active and informative. The meeting provided a great help in promoting JAEA research activities and collaborations with domestic and foreign organizations.

1.Session 1 General report
Session 1-1

Fuel Safety Research in EPRI

Rosa Yang, 3412 Hillview Ave, Palo Alto, CA ryang@epri.com

The Fuel Reliability Program, supported by international utilities, was established in 1998 to proactively address fuel operational and regulatory issues. Four Working Groups have been established : PWR corrosion & crud control, Fuel regulatory issues, Fuel reliability and performance margins and BWR corrosion & crud control. The WG 2, fuel regulatory issues, is the industry interface with US NRC on all fuel-related regulatory issues; interacting with NRR, thru NEI, on licensing issues; collaborate with NRC research (RES) on research issues.

The current focus of safety research in EPRI include :
-Burnup extension licensing to 75 GWD/T rod average burnup
-Criteria for reactivity initiated accidents (RIA)
-Loss of coolant accidents (LOCA) criteria
-Analytic tool (FALCON) for understanding transient fuel behavior and criteria development

The presentation will describe the status, accomplishment and issues to address in each area. In addition, examples will be given on how EPRI uses the FALCON code to analyze and design safety experiments and develop safety criteria.

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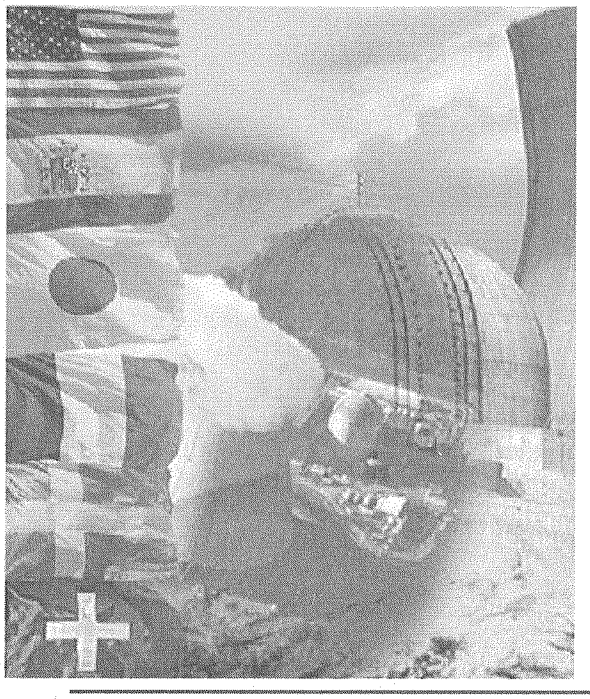


Fuel Safety Research in EPRI

Rosa Yang

Fuel Safety Research Meeting
Tokyo, Japan
March 2-3, 2005

Fuel Reliability Program

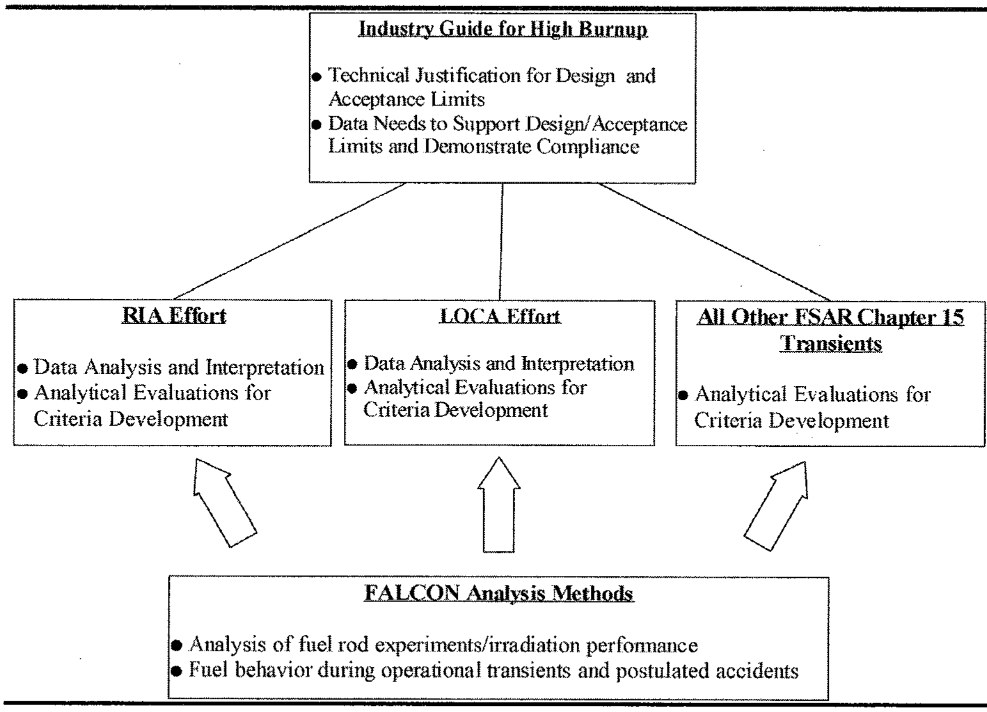
EPRI

Outline of EPRI's Safety Research

- Fuel Reliability Program – Working Group 2 (WG2) is the industry's interface with the USNRC on issues related to fuel
  - Through Nuclear Energy Institute for interactions with NRC-NRR on licensing issues
  - Directly with NRC-RES on technical matters
  - Utilities and fuel vendors actively participate in WG2
- Currently focus:
  - Burnup extension licensing (Industry Guide)
  - Criteria for Reactivity Initiated Accidents (RIA)
  - Loss of Coolant Accident (LOCA) criteria
  - Analytical tools for criteria development (FALCON code)

EPRI

EPRI Approach to High Burnup Licensing



EPRI

Role of Analytical Methods - FALCON

- Understand the phenomena important to fuel rod behavior
  - Evaluate experimental results
  - Integrate results from separate effects tests to represent overall fuel behavior, e.g. material property tests and PCMI loading
- Assist in design of experiments to highlight key behavior and minimize test artifacts
  - Identify important phenomena to observe/measure
  - Define specimen & equipment designs
- Translate behavior observed in integral tests and develop acceptance criteria
  - Account for differences in test conditions and postulated LWR accident conditions

EPRI

FALCON: LWR fuel rod behavior code

- Robust numerical structure and material constitutive models
  - State-of-the-art numerical methods in thermo-mechanics
    - Calculates the thermal and mechanical behavior of the fuel, cladding, gap, and internal void regions
  - Uses latest material property data on UO_2 and Zircaloy behavior that include the effects of high burnup
    - High burnup fuel performance
  - Applications include steady state fuel performance, off-normal/transient behavior, and fuel failure root cause evaluations
    - Large database of fuel rods for code validation

EPRI

Material Modeling in FALCON

- Fuel and cladding properties
  - MATPRO augmented by literature & NFIR material properties data
- Robust constitutive formulation
  - Elastic-Plastic-Creep for fuel and cladding
  - Large-displacement/finite-strain theory
  - 3D Cracking for fuel, compression / friction for gaps
- Behavioral models
  - Steady state and transient FGR
  - Cladding failure by PCI, mechanical fracture and rupture
  - High temperature phase transformation, oxidation, and oxygen diffusion in LOCA conditions

The Industry Guide (IG)

- Objective:
  - Provide a consistent, industry-wide approach for licensing burnup extensions for LWR fuel
  - Simplify the licensing process by obtaining NRC agreement on review of criteria
  - Simplify the burden of demonstrating adequate performance at targeted burnup limits (70 -75 GWD/T)
  - Provide licensees a reference for use as a basis (starting point)
    - Prepared in response to NRC request for an industry-wide consistent approach
    - Similar to a "Reg Guide" issued by NRC
    - Consistent with NUREG-0800 Standard Review Plan Section 4.2 and fuel vendor topical reports.

The Industry Guide (Continued)

Approach used in the IG

- Review current licensing bases – assess applicability to extended burnups
- Identify if modifications or additions to existing bases are needed
- Justify existing or modified criteria
- Identify what data will be needed to demonstrate compliance at extended burnups

Status

- Development work completed – Report published in 2004 (Ref: *Licensing Criteria for Fuel Burnup Extension Beyond 62 GWd/tU – Industry Guide*, EPRI Report 1001808)
- RIA and LOCA –related criteria reference the existing RIA topical report (EPRI 1002865) and a separate LOCA-specific topical report (to be prepared in 2006)
- Being submitted to the NRC for review and approval as an "industry topical" - for use as a reference in individual vendor applications

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Work related to RIA

- Conducted analysis & evaluation of all relevant RIA-simulation tests (EPRI Report 1002863)
- Developed failure threshold for LWR fuel (consistent with measured material property data)
- Proposed revisions to current RIA criteria at high burnup in a Topical Report (EPRI Report 1002865)
  - Topical report under review by NRR
- Developed 3-D Rod Ejection Analysis methodology for PWRs (EPRI report 1003385)
- Ongoing work - continued participation in CABRI water loop & analysis of results
  - Advanced claddings
  - Pulse width effects
  - BWR cladding

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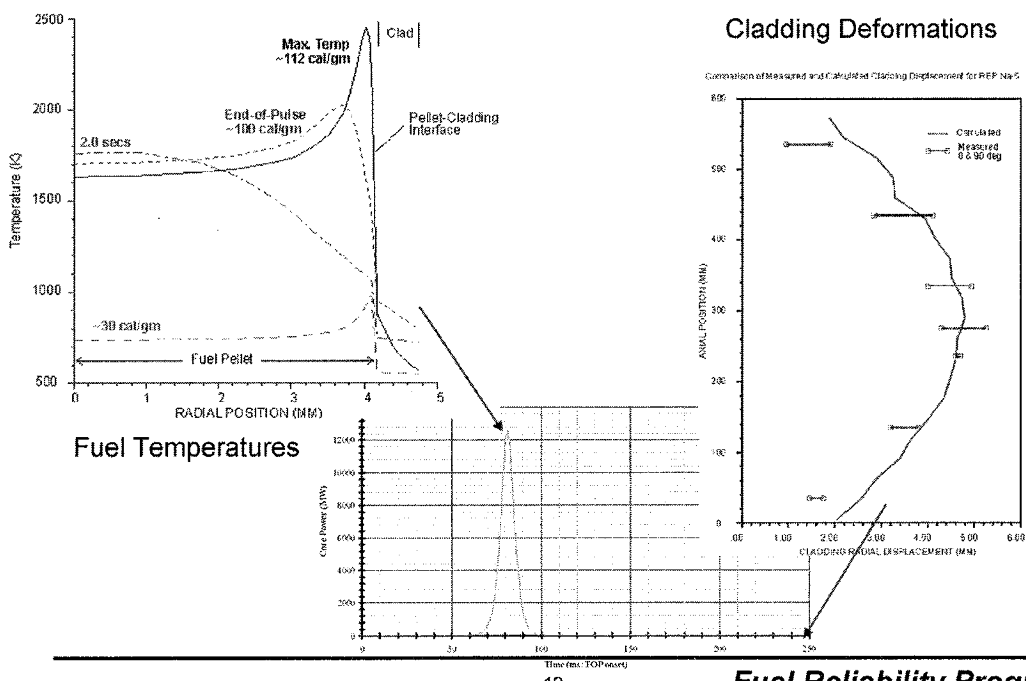
FALCON Analysis of RIA Fuel Behavior



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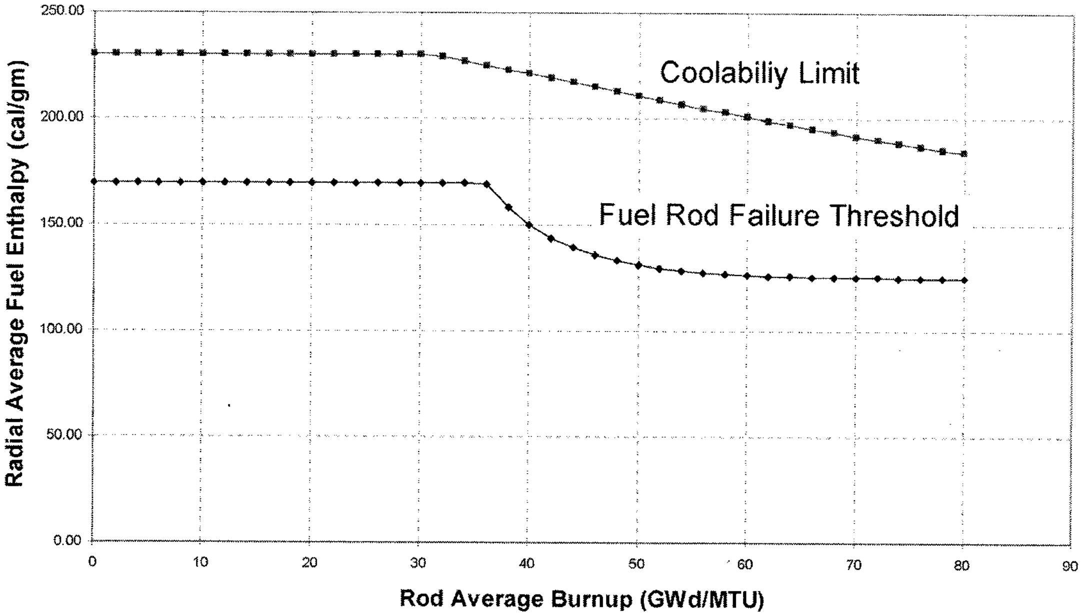
Proposed RIA Criteria



Japan-March 2005 11 *Fuel Reliability Program*

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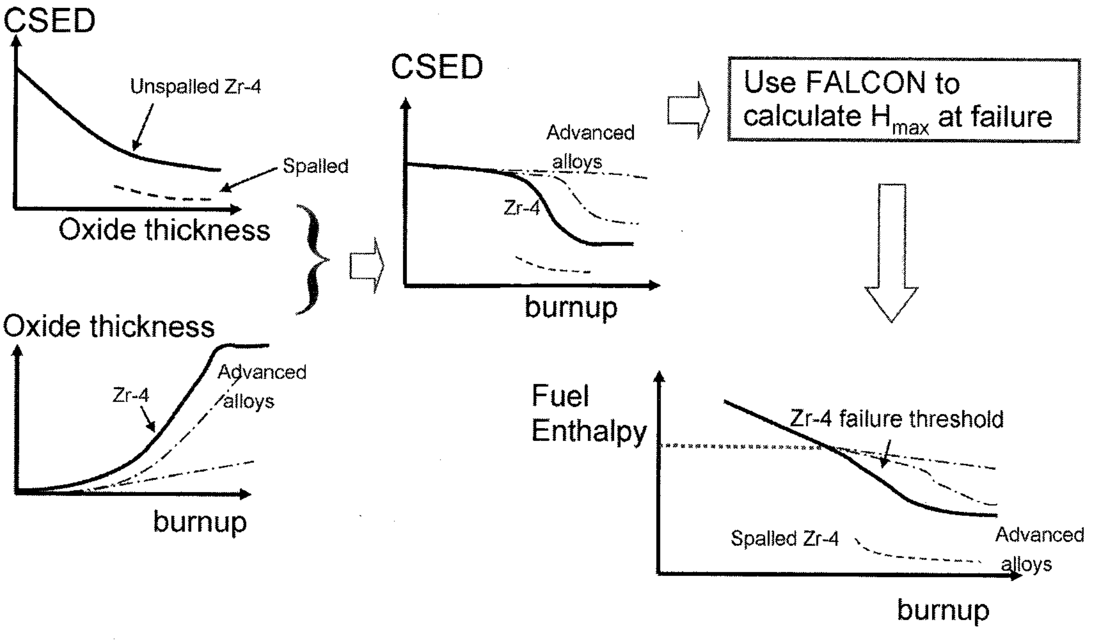
Analytical Approach to Develop Cladding Failure Threshold



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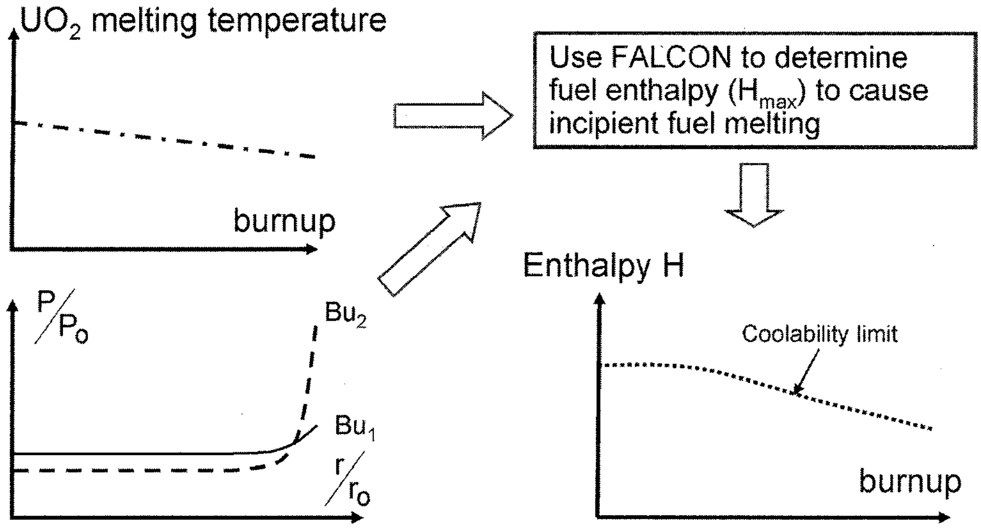
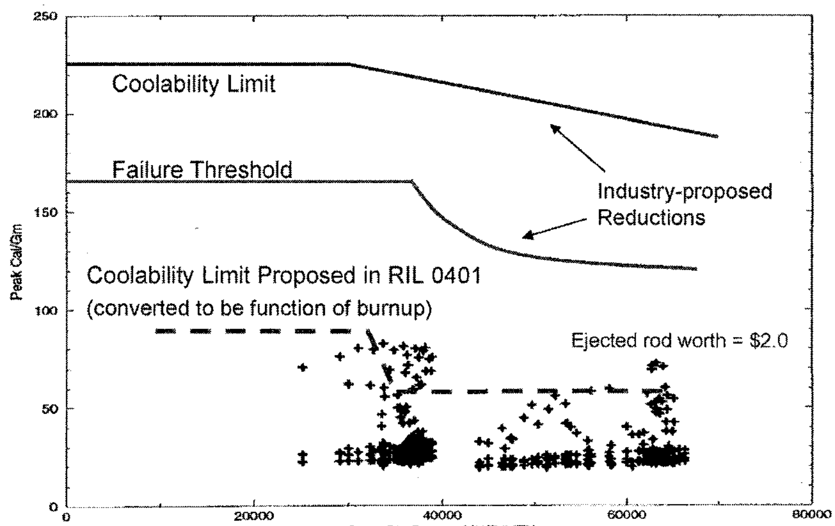
Approach to develop RIA coolability limit based on energy to incipient fuel melting



Application to PWR HZP REA

Comparison of Licensing 3-D Neutron Kinetics Results (W 4-Loop Plant)[3] with Proposed RIA Criteria



3. Risher D., et. al., "Generic Assessment of the High Burnup Reactivity Insertion Accident Issue in Westinghouse PWRs," Proceedings of the Fifth International Topical Meeting on Nuclear Thermal-Hydraulics, Operations,and Safety, Beijing, China, April 1997

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LOCA-Related Work

- FRP is actively participating in the LOCA program at ANL
  - Supply irradiated (high-burnup) BWR & PWR fuel rods for LOCA-related testing
  - Supply rods with advanced claddings (M5 & Zirlo) at high burnup
  - Analytic support for design of experiments
  - Independent analytic evaluation of results
- Ongoing analysis of results from international programs
  - Halden
  - JAERI
  - ANL
- Planning to develop a Topical Report on LOCA criteria in 2006

Japan-March 2005 15 **Fuel Reliability Program**

EPRI

Analytical Evaluations for LOCA

- Integral LOCA experiments
  - Calculate ballooning and burst behavior to evaluate burnup and alloy effects
  - Calculate cladding high temperature oxidation profiles to understand cladding temperature distribution (fuel relocation effects)
- Post-Quench ductility tests
  - Evaluate impact of hydrogen on embrittlement
    - Assess/correlate ring compression test results
  - Fundamental understanding of embrittlement factors
    - oxygen profile in alpha and beta layers
    - hydrogen distribution/morphology

Japan-March 2005 16 **Fuel Reliability Program**

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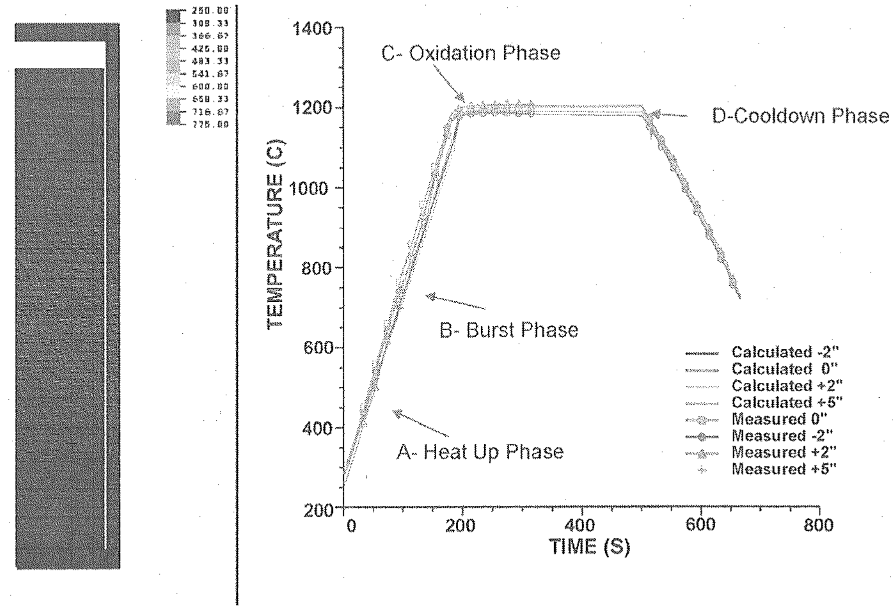
Deformation Response During a LOCA Integral Tests at ANL



Japan-March 2005 17 *Fuel Reliability Program*

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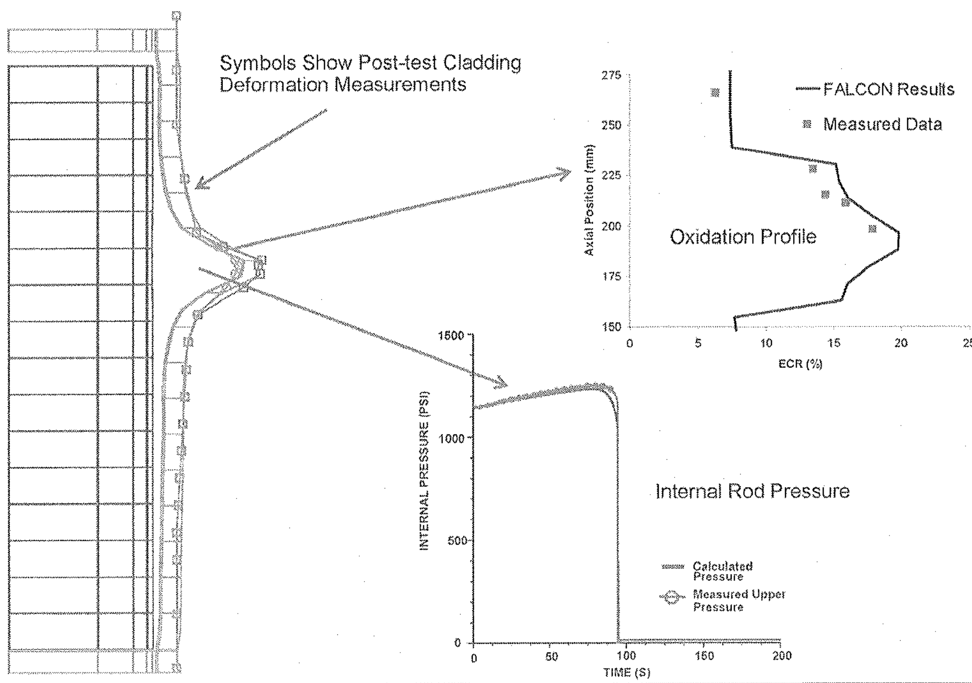
Comparison to ANL Results



Japan-March 2005 18 *Fuel Reliability Program*

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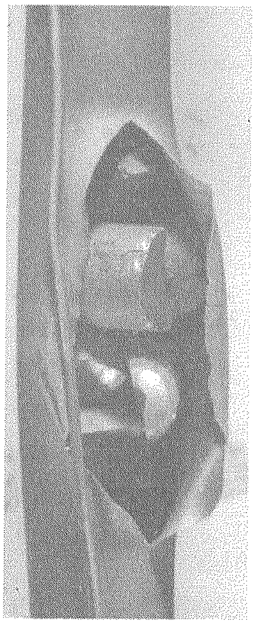
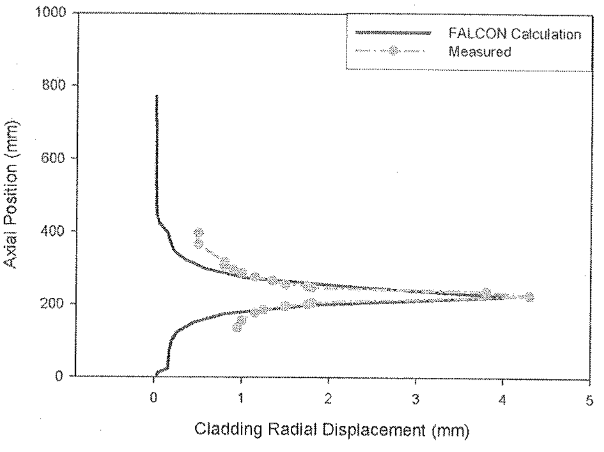
Halden IFA-650.2: FALCON Strain Calculation





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Hydrogen Effect on Post-Quench Ductility

- Hydrogen influences two key factors that control the deformation behavior of zirconium alloys
  - Increases the oxygen solubility in the beta phase
  - Results in formation of d-hydrides along grain boundaries during quench
- Both factors cause an increase in the material hardness
  - Solid solution hardening by higher oxygen content
  - Hardening by hydride precipitates and dislocations
- Possible synergistic effects between hydrogen and oxygen that may influence material ductility
  - Enhancement of oxygen diffusion in beta layer by hydrogen
  - Interplay of dislocations caused by hydride precipitation and solid solution hardening

Japan-March 2005 20 *Fuel Reliability Program*

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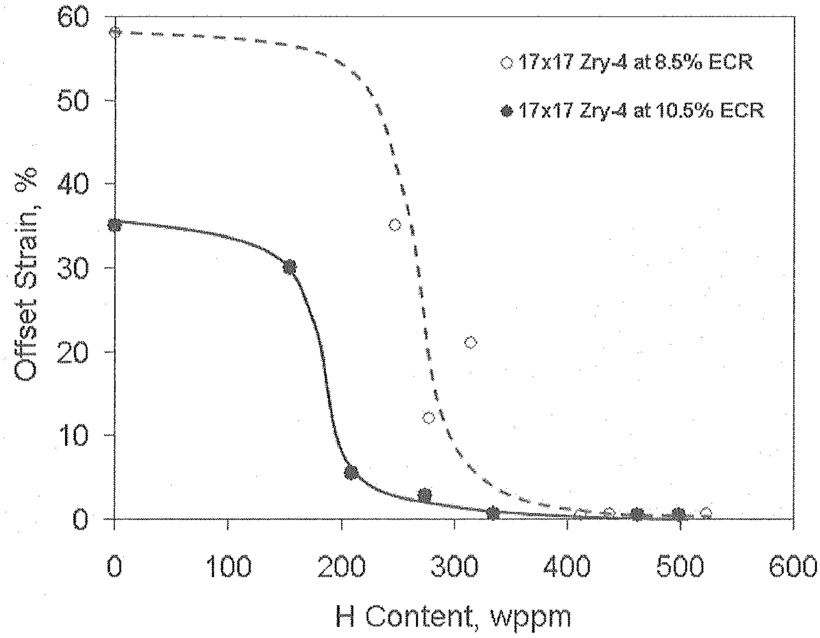
Ring Compression Tests on Pre-hydrided Zr-4

RCT performed at 135° C



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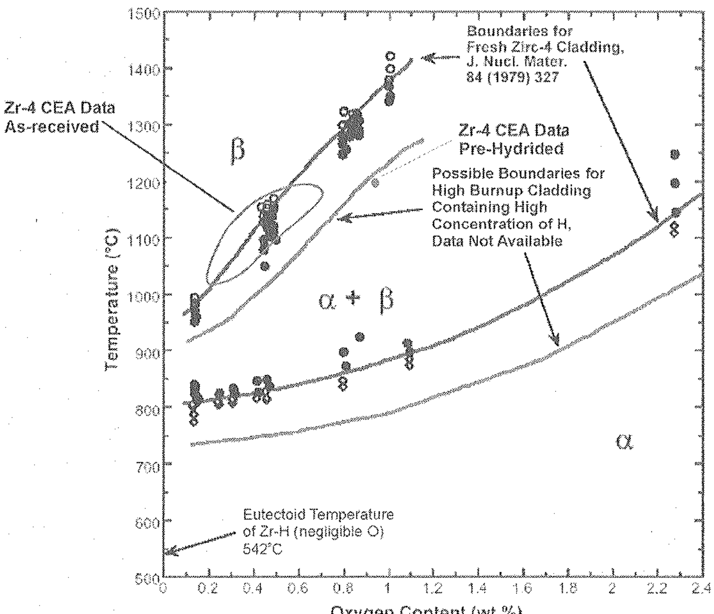
Effect of Hydrogen on Zr-O Phase Transformations

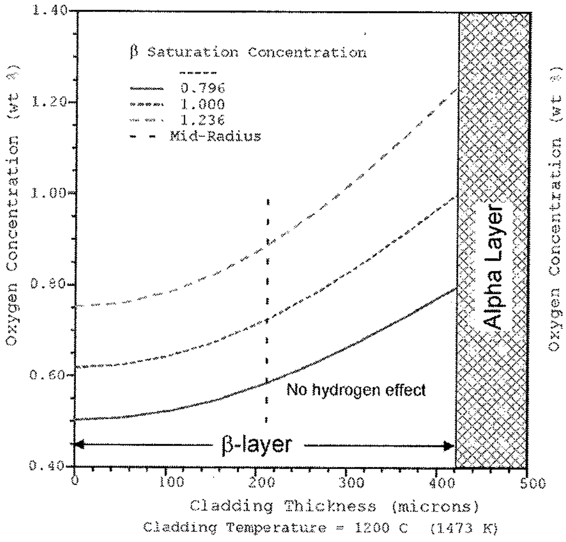


Ref: Hee Chung, ANL

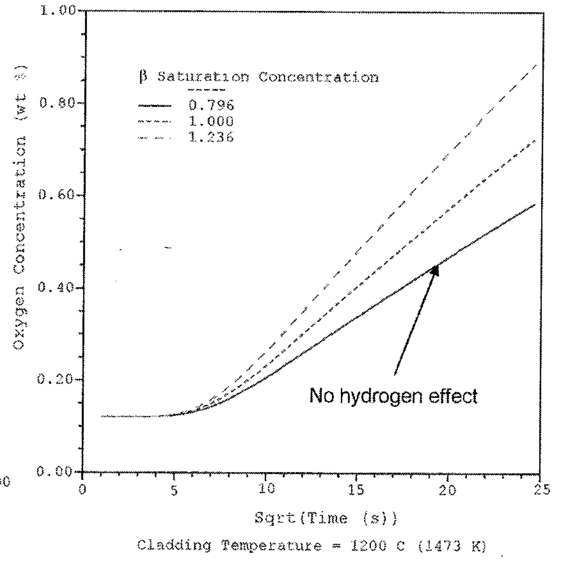
Oxygen Profile in β Layer Calculated by FALCON Using COBILD Model

Effect of Hydrogen on the β Layer Saturation Concentration

Oxygen Profile in β layer at ECR ~ 12% ECR



Mid-Radius Oxygen Profile vs. Time



LOCA Issue Is At A Critical Stage

- NRC has proposed an aggressive schedule for rule making to start September, 2005
  - Some tests may not be completed
  - No tests on irradiated advanced cladding
- Not all tests have been completed / most recent results need interpretation
- A reduction of LOCA limits will have major impact
  - New rule making: models, methodologies, additional analysis
  - Reduced margins, operation flexibility and fuel cycle economics
  - Impact on new cladding materials (performance-based criteria?)

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Summary

- The Fuel Reliability Program - WG2 is the industry focal point to address fuel-related regulatory issues
- The industry Guide is complete – Expect NRR review during 2005-06
- The RIA issue is being evaluated by NRR – FRP work will consist of:
  - Support for the NRR review
- Major near-term issues in the area of LOCA criteria
  - Continuing participation in ANL and other international experimental programs, like ALP
  - Continue analytical work to guide experiments and interpret data
  - Interact with NRC on LOCA rulemaking in 2005-2006
  - Develop technical report to support revisions of LOCA acceptance criteria

Session 1-2

Fuel Safety Research at EDF

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Tokyo, FSRM 2005

Fuel Management Schemes licensing process in France

In France, the licensing process typically involves a specific Fuel Management Scheme for a type of Nuclear Power Plants (NPP). A Fuel Management Scheme usually includes a specific fuel product and a set of design analysis methodologies. There are currently 3 types of operating NPPs (900Mwe, 1300 Mwe and 1450 Mwe NPPs). The new EPR (European Pressurized Reactor) will be built in Flamanville, north west coast of France, to be operational in 2012.
Three main Fuel Management Schemes are currently undergoing licensing:

- **MOX Parity** for the 900 Mwe NPPs featuring 3.7% UO2, equivalent 3.25% MOX fuel, M5 cladding and discharge burnup of 52GWd/TM (Fuel Assembly (FA) average). The key challenges to address are the MOX behavior in transient conditions including the FGR, the helium release, the End-of-Life rod internal pressure and the source term.
  The license is expected to be approved by mid-2006.
- **Burnup extension to 62 GWd/TU** (FA average) for the 1300Mwe NPPs featuring 4.5% UO2 fuel, M5 or optimized ZIRLO claddings, flexible 15 to 21 months cycles. The key issues to address are the transient behavior (RIA, LOCA and source term) of the fuel and the fuel assembly structure at high burnup.
  The license is expected to be approved by mid-2008.
- **EPR Fuel Management Scheme to 60-70 GWd/TM** (FA average) featuring 5% enrichment doped UO2 fuel (no PCI-related operational constraints), MOX as an option, M5 cladding and flexible 12 to 22 months cycles. The key challenges are the qualification process of the doped fuel at high burnup (steady state and transient, RIA, LOCA and source term), the related models and methodologies.
  The license is expected to be approved by 2011.

Safety related experimental programs

RIA, LOCA and Source-Term issues are addressed using integral tests (e.g. CABRI) and separate effect tests. The experimental programs are aimed to provide the codes and models with relevant data to validate and qualify them.

The current **RIA** empirical safety domain has been extended to 62 GWd/TU (FA average) and will be replaced with an analytical fuel failure limit based on the mechanical properties of the cladding (Critical Strain Energy Density : CSED) similar to the one developed within the Fuel Reliability Program. EDF has recently performed a theoretical work to improve the relevance of the mechanical property data by correcting the raw data to account for the anisotropy, the biaxiality, the strain rate of the material and the tests. With this new approach to treat the data, the RIA failure limit can be defined using a convincing lower bound of the mechanical property tests data base.
In France, the **LOCA** limits are currently based on the behavior upon quench of the cladding. To address the high burnup related issues raised in the international arena, various separate effect tests have been performed. The tests include the study of the pre-transient corrosion/hydriding impact on high temperature oxidation, the ballooning and burst behavior (including measurement of the phase transformation kinetics as a function of H), the post-quench mechanical tests (including ring compression tests, 3 points bend tests, impact tests on pre-hydrided unirradiated samples), the study of the impact of steam pressure on the oxidation kinetics (to address the small break LOCA conditions), specific annealing tests on high burnup fuel pellets to quantify the source term, etc...
To complement these tests, theoretical work has been conducted to eventually move from a post-quench ductility based criterion to a strength based criterion. To avoid any debates concerning the ability to calculate the impact loads during post LOCA events, the load the cladding should withstand after a LOCA to ensure long term coolability might be related to other accepted design safety limits (e.g. the buckling of the FA grids or the maximum deflection allowed by the FA to FA gap closure within the core vessel).

Fuel Safety Research at EDF

FSRM
Tokyo, Japan
March 2-3, 2005

Nicolas WAECKEL

On-going licensing issues in France

Three main files currently under a licensing process

- **MOX Parity** (900 MWe, 3.7% UO2, eq. 3.25% MOX, 52GWd/TM FA av.)
  - » Issues : transient behavior, FGR, EOL Rod Internal Pressure
  - » Approval expected mid-2006
- **Burn-up extension to 62 GWd/TM** (1300 Mwe, 4.5 % UO2, 15-21 months cycles)
  - » Issues : FA behavior, RIA and LOCA criteria
  - » Approval expected mid-2008
- **EPR Fuel Management** (5% UO2, 12-22 Months cycles, No PCMI operational constraint, 60-70 GWd/TM)
  - » Issues : doped fuel qualification in transient
  - » Approval expected in 2011

RIA related Research

- The current empirical Safety Domain has been extended to 62 GWd/TU (FA average)
  - ΔH <60 cal/g, LH/2 >30 ms, Zr02<100μm
- Develop a new Analytical Failure Limit : CSED or ΔH =f(Burnup)
  - *Improved analysis of the mechanical property data base used to define the failure limit*
- Participate in CIP (Cabri International Program)

- Develop separate effect to address post-DnB cladding behavior
- Develop new separate effect tests to scale the behavior of the fuel pellet during rapid transients
  - Avoid running integral tests for each type of fuel
  - Identify the key parameters needed to optimise new fuel types (doped UO2 and MOX fuel)

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LOCA related Research

- Separate effect tests on pre-hydrided samples to qualify new claddings
  - Burst, constitutive laws, phase transformation diagram
  - Oxidation kinetics,
  - Failure upon quench, post-quench mechanical tests (ring compression tests, impact tests, 3 points bend tests)
- Separate effect test to study high temperature oxidation under high steam pressure
  - To address small break LOCA issues
- Separate effect tests to quantify the source term for high burnup fuels (UO2 fuel, MOX, doped fuel)
- LOCA criteria at high burnup
  - International effort
  - Cross check tests with ANL LOCA test program
  - *Calculations and alternative strength based approach*

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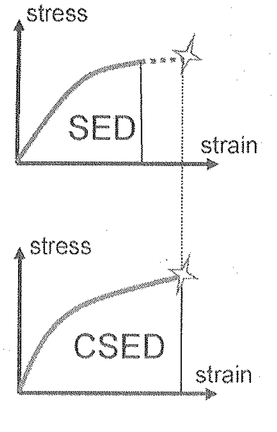
A new approach to correct the Mechanical Property Data used to define the RIA failure criterion

*Theoritical work by Sylvain Leclerq and Gérard Rousselier (**EDF-R&D**) to be presented at the ANS LWR Reactor in Oct 2005)*

EDF Electricité de France

Clad Failure Model for PCMI Conditions

- Strain Energy Density (SED) is a measure of loading intensity on the cladding
  - SED is a calculated response parameter, based on integrating stress and strain
- Critical SED is a measure of cladding failure potential or cladding residual ductility
  - **CSED is determined from Mechanical Property Tests**



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The Mechanical Property Tests data base includes various type of tests

Program – TYPE OF TEST	Fuel Type	Max. Bu (GWd/tU)	Max. Fast Fluence (n/cm^2)	Range of Oxide Thickness (μm)	Temperature Range (K)	Strain Rate (/sec)
ESEERCO Hot Cell Program on Zion Rods						
Burst	15x15	49	9.4x10^{21}	15 - 25	588	2x10^{-5}
ABBCE-DOE Hot Cell Program on Fort Calhoun Rods						
Burst	14x14	53	8x10^{21}	30 - 50	588	6.7x10^{-5}
EPRI-B&W Hot Cell Program on Oconee-1 Rods						
Axial Tension Ring Tension Burst	15x15	25	5x10^{21}	< 20	616	8x10^{-5}
EPRI-ABBCE Hot Cell Program on Calvert Cliffs-1 Rods						
Axial Tension	14x14	68	12x10^{21}	24 - 110‡	313 - 673	4x10^{-5}
Ring Tension				24 - 115‡	573	4x10^{-5}
Burst				36 - 110‡	588	6.7x10^{-5}
ABBCE-DOE Hot Cell Program on ANO-2 Rods						
Axial Tension	16x16	58	12x10^{21}	24 - 46	313 - 673	4x10^{-5}
Burst				24 - 46	588	7x10^{-5}
EdF-IPSN PROMETRA Program						
Ring Tension	17x17	63	10x10^{21}	20 - 120‡	298 - 673	.01 - 5
Nuclear Fuel Industry Research Program-III						
Burst	15x15	51	9x10^{21}	40 - 110‡	573 - 623	5x10^{-5}

‡ - Several samples were obtained from cladding with spalled oxide layers.

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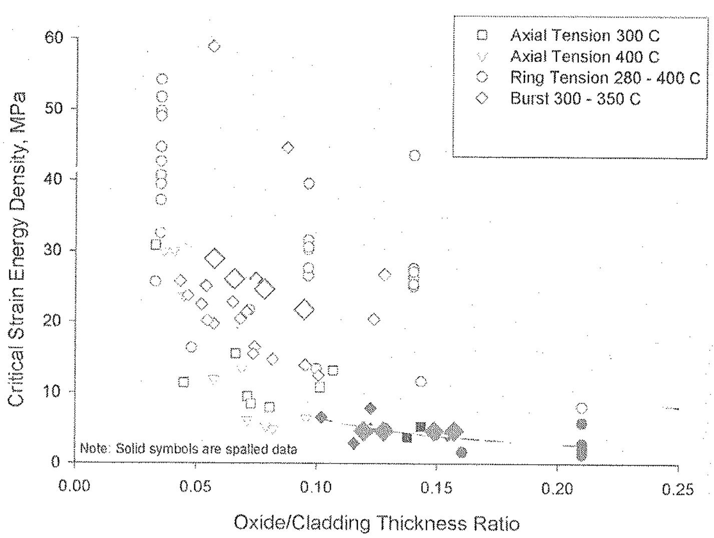
Problem : the Mechanical Property Test data has large scatter

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



- Scatter is more related to test conditions and specimen design artifacts rather than to material variability

Data base includes
-Axial Tension Test,
-Ring Tension Tests,
-Burst Tests

Correction factors have to be applied to the raw data to account for the various test conditions

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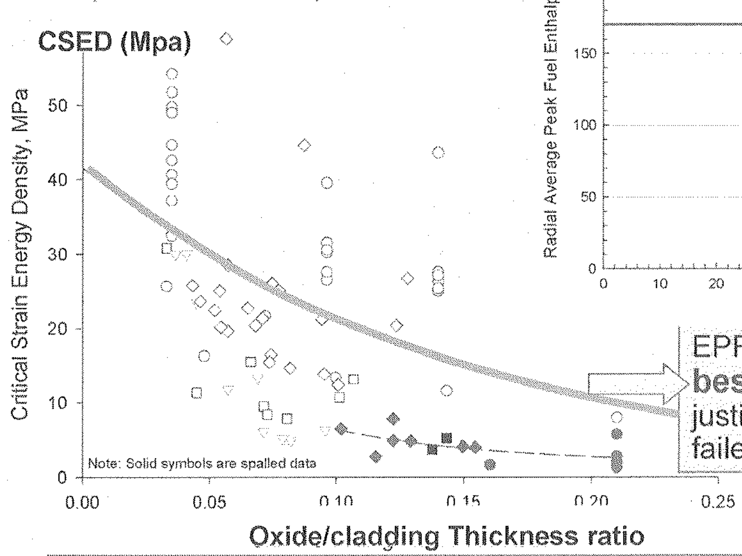
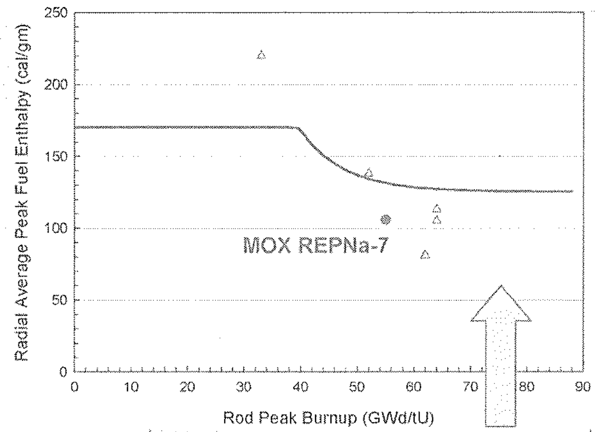
A Best Fit curve has been first used to define the failure limit

EPRI defined a first set of correction factors (RIA Topical Report June 2002)





EPRI showed that using a **best-fit CSED curve** is justified when compared with failed-unfailed RIA database

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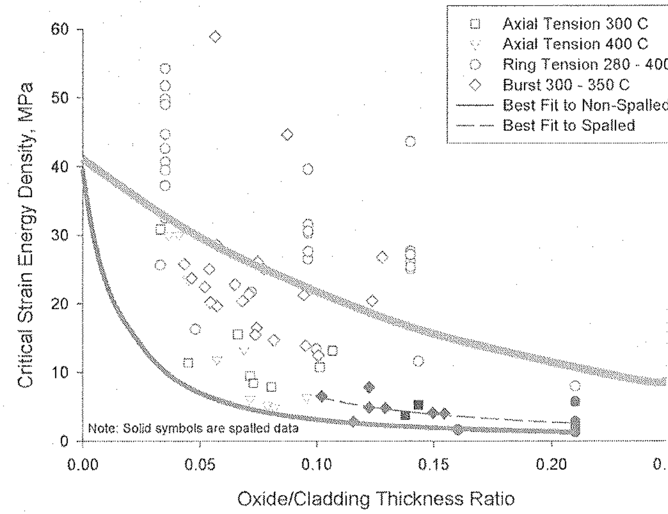
A lower bound CSED curve would have been more convincing than a best fit CSED curve

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



This point has been raised again and again by the Reviewers of the RIA Topical Report

Question : Can the analysis of the mechanical property data base be improved ?

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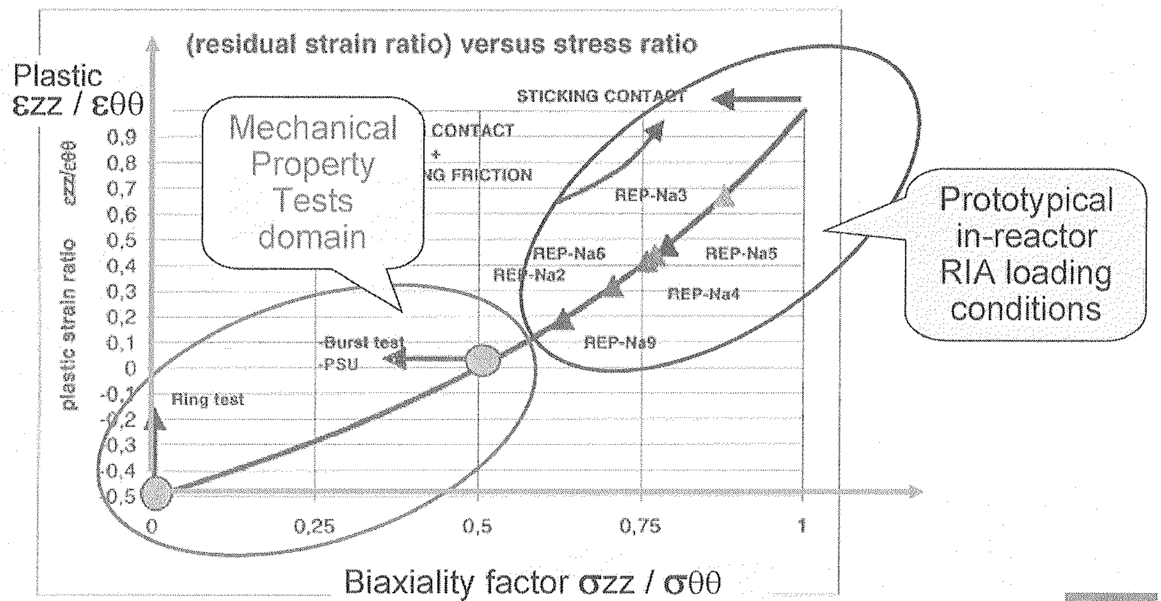
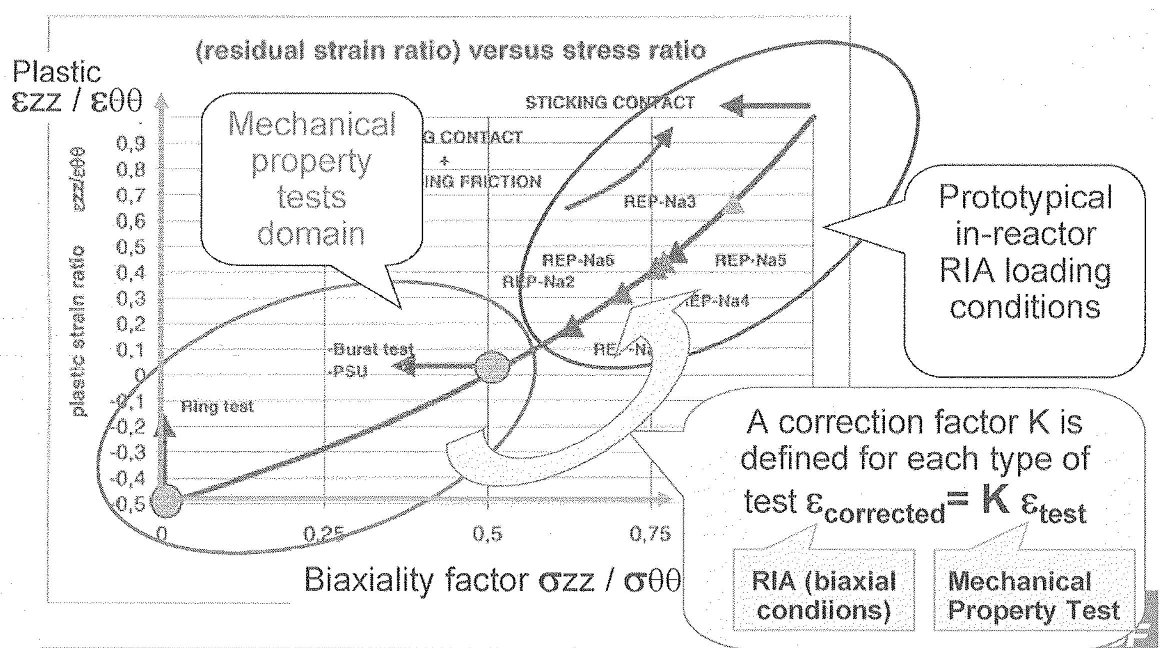
The Mechanical Property Tests are not representative of the in-reactor conditions



Homogenize the data base and make it more representative



EDF approach

- Select the most relevant Mechanical Property Tests of the data base
  - Axial tension test are rejected (not representative of a RIA loading)
- Homogenize the data base
  - Define a correction factor for each type of test
  - Account for the strain rate (it varies from 2.10^{-5} to $5\ s^{-1}$, depending on the test)
  - Account for the anisotropy and the specific property of the cladding
    - » Use a Hill Equivalent Strain Tensor
- Calculate the CSED of each test of the data base
- Define the lower bound as a fonction of oxide thickness

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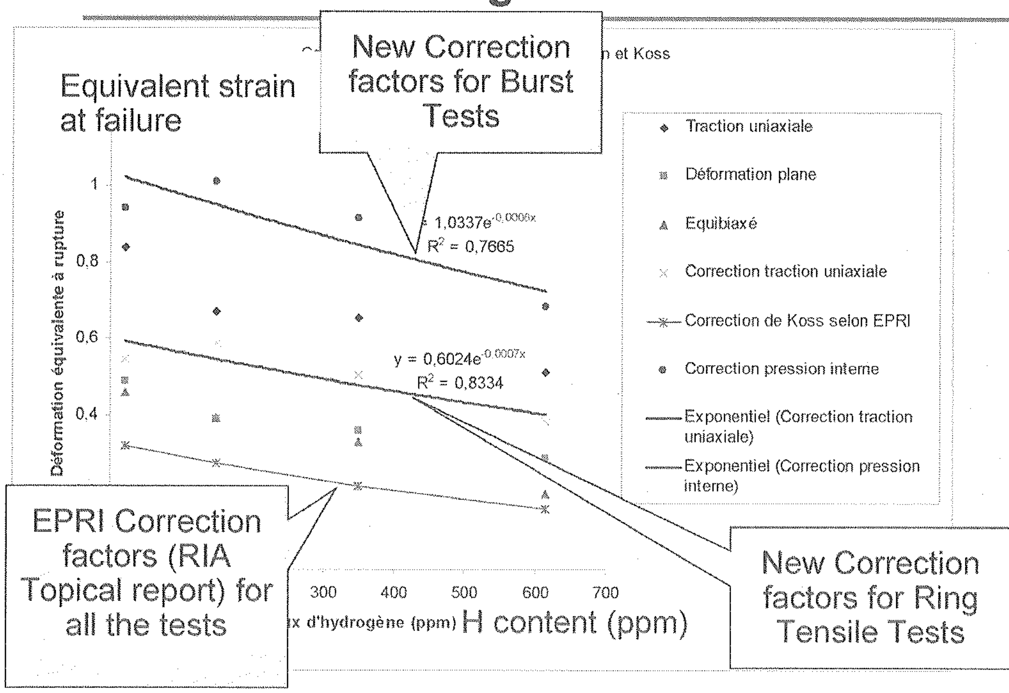
The Correction factors are determined using Fan & Koss data



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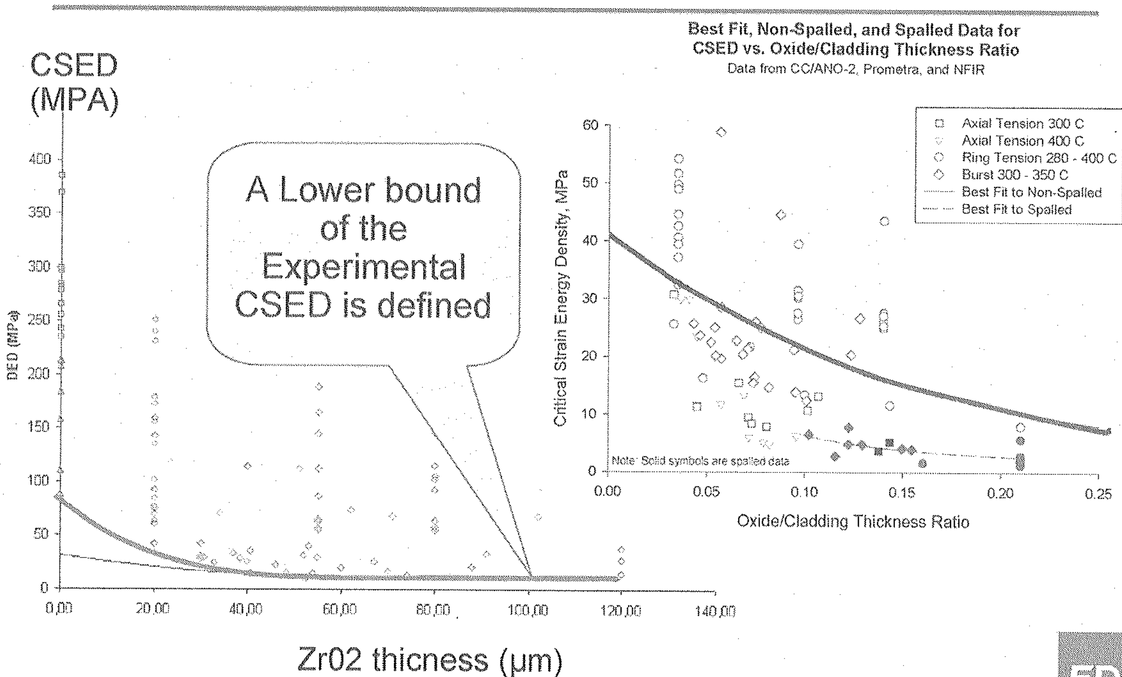
The CSED are higher after applying the new correction factors



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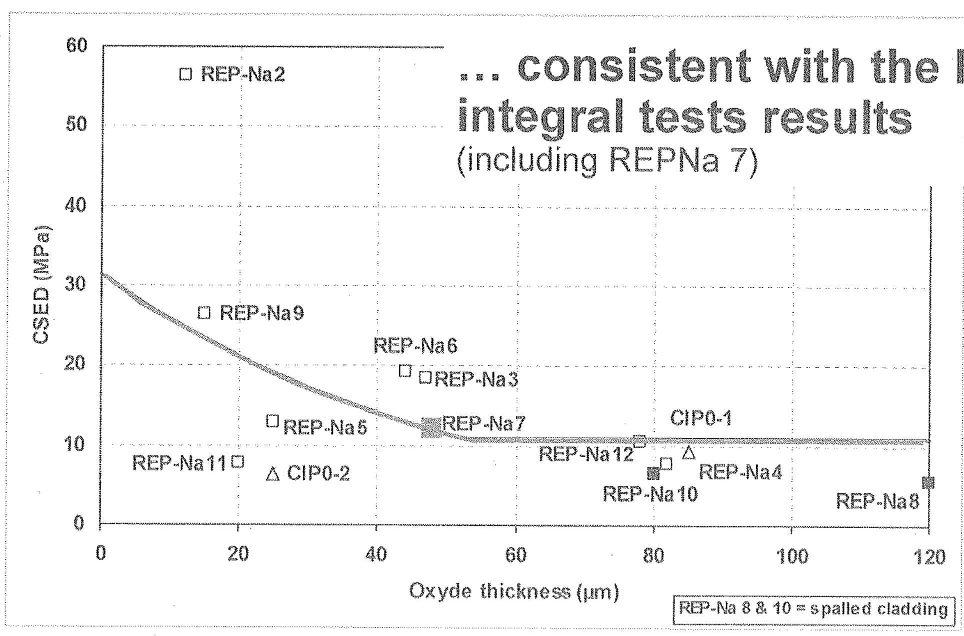
The RIA failure limit based on the corrected CSED (lower bound) is...



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LOCA limits at high burnup

Are the post-quench ductility tests appropriate ?

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Performance based approach (US-NRC)

- **Maintain coolable geometry : basis for the LOCA criterion**

→ keep the fuel inside the cladding

→ don't let the cladding fragment or break in several pieces

→ need some ductility (→ how much is enough ?)

→ limit Temperature and Oxidation (PCT and ECR)

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Post-quench ductility or strength ?

- NRC chose a zero-ductility limit to ensure long term coolability
  - Did not want to calculate any post-LOCA event loading (too complex)
  - Potential loadings are mainly impacts (transmitted by the grids to the rods) and rod bowing as a result of FA distorsion
- A way other than post-quench zero ductility might be needed to demonstrate long term coolability.
- Proposal : bound the potential loadings by another design limits
  - Assess the maximum loadings expected when **other design limits** are reached, e.g. :
    - » Grid buckling
    - » Max rod bow when all the FA-to-FA gaps are closed

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Loads applied to the fuel rods in their grid cells

- **Static** holding forces from the grid springs
  - Maximum at BOL
- **Static** loading related to the overall deflection of the FA
  - Contact forces between the fuel rod and the grid spring and dimples, within the grid cells
- **Dynamic forces** directly induced by the impact of the grid against a baffle or another FA
  - Impact forces transmitted to the rod through the grid springs and dimples
- **Additional Dynamic forces** induced by the elastic deformation of the grid (due to the impact of the grid against a baffle or another FA)

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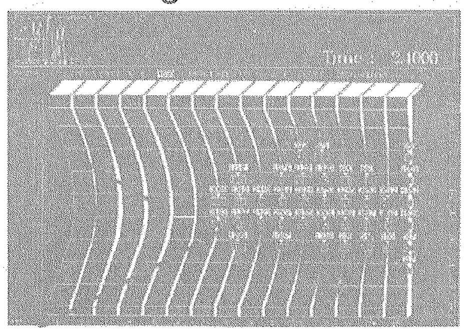
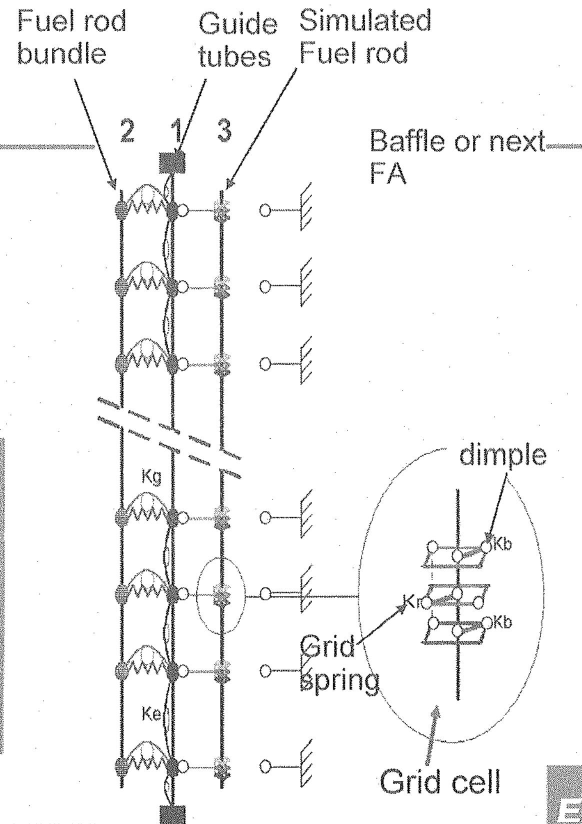
To be conservative the maximum impact force applied to the grid is the **buckling strength** of the grid





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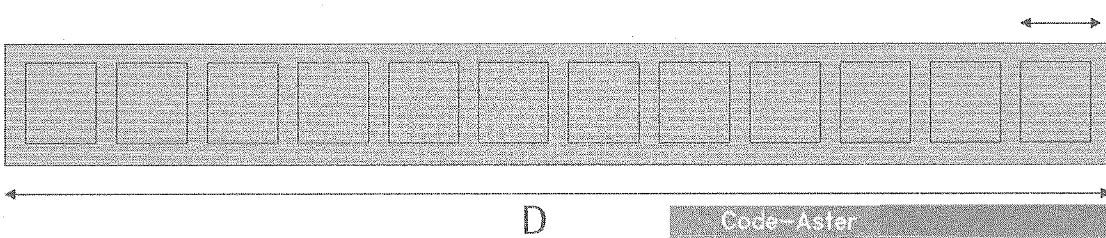
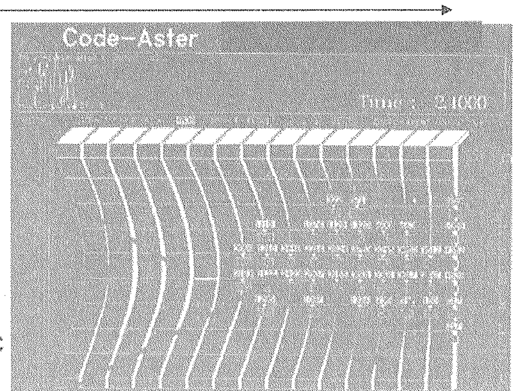
Assessment of the FA deflection

Row of N Fuel Assemblies





Maximum possible FA deflection :

Deflection = D – Nxd + Gap_{el}

D : Core dimension (cold conditions)

d : Grid dimension

Gap_{el} : additional gap induced by the elastic deformation of the grids when impacted

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Synthesis : bounding loadings

		BOL	EOL	Gap 0 μm	Gap 50 μm	Gap 100 μm
Total Forces (including initial holdown forces)	Spring	62.7 N incl. 40 N static	34.1 N incl. 5 N static	32.3 N incl. 2 N static	24.8 N incl. 0 N static	19.5 N incl. 0 N static
	Dimple	**115.7 N incl. 80 N static**	83.7 N incl. 36.5 N static	**97.5 N incl. 36 N static**	92.4 N incl. 34.3 N static	88.2 N incl. 33 N static
FA Deflection		**26.1 mm 14 ft. 22.1 mm 12 ft. mode 1**	15.9 mm 14 ft 13.8 mm 12 ft. mode 1	**15.9 mm 14 ft 13.8 mm 12 ft. mode 1**	15.9 mm 14 ft 13.8 mm 12 ft. mode 1	15.9 mm 14 ft 13.8 mm 12 ft. mode 1

- Maximum force applied to the rod (BOL conditions) :

  120 N = 80 N static + 30 N in 45 ms + 10 N in 3 ms

- Overall max FA deflection :

26.1 mm (corresponding to < 4 Mpa (axial stress) or **55 N (axial force)**

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Conclusion : what could be a strength based post-LOCA tests like ?

- Based on the grid buckling design limit
  - Post LOCA diametral compression test (RCT)
    - » The sample should survive a ~120 N loading
  - Post LOCA impact tests
    - » The cladding should survive a ~0.08 J impact
- Based on the maximal possible FA deflection
  - Post-LOCA axial tension test
    - » The sample should survive a ~55N load

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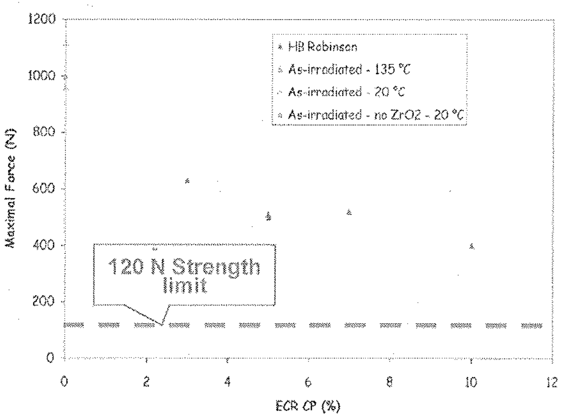
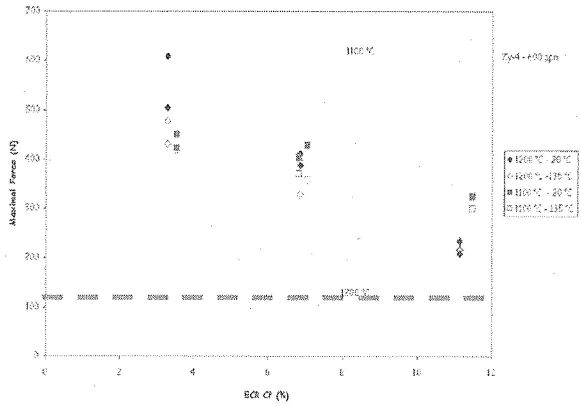
Irradiated post-oxidation and unirradiated prehydrided post-quench ring compression tests





ANL Results :

As-irradiated defueled Zr4 cladding

CEA-EDF-FRA DEZIROX Results

As received pre-hydrided Zr4 (600ppm)

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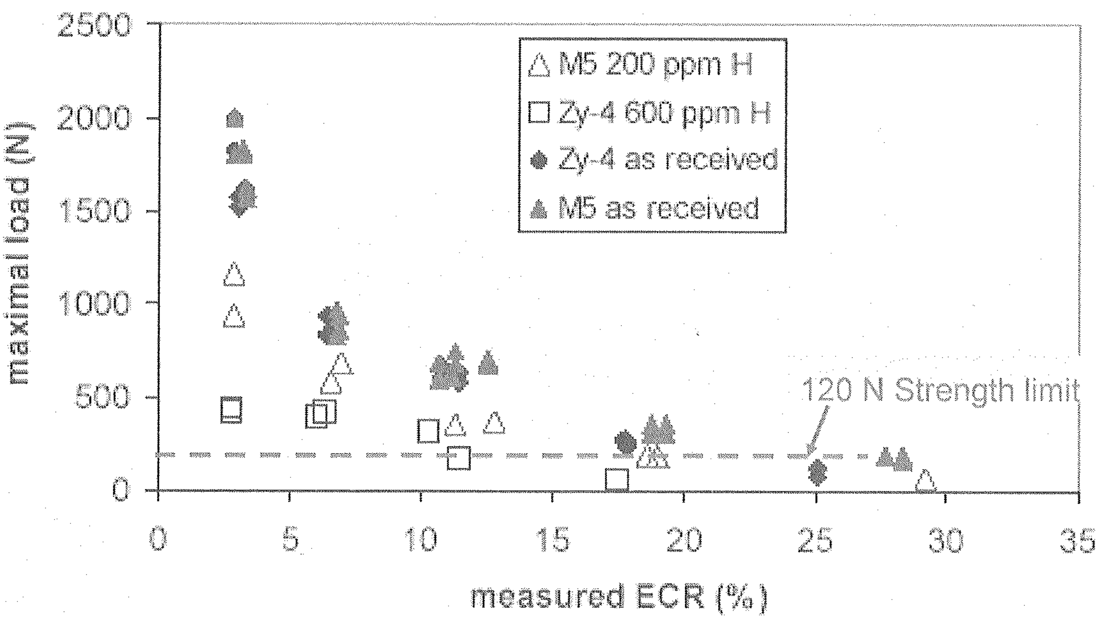

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Session 1-3

IRSN R&D STUDIES ON HIGH BURNUP FUEL BEHAVIOUR UNDER ACCIDENTAL CONDITIONS

J. Papin
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Direction de la Prévention des Accidents majeurs

The optimisation of the fuel management strategy in the French PWRs as foreseen by the utilities with burn-up increase for UO_2 fuel and large introduction of MOX fuel, created the need for new investigation of fuel behaviour under design basis accidents such as loss of coolant and reactivity initiated accidents. Within this framework, IRSN has launched R&D studies in these two areas in order to get bases for assessment of new safety criteria, evaluation of safety margins and development of computational tools.

Concerning LOCA matters, an extensive state of the art and the new results from foreign experimental on-going programs, allowed to identify the main pending questions which are the impact of the fuel relocation in the ballooned areas on the clad temperature evolution and oxidation rate, the secondary hydriding, the clad mechanical behaviour during the whole sequence and the coolability of an irradiated bundle (maximum blockage ratio). In parallel to the interpretation of the available experiments, the development of an advanced analytical tool is envisaged (3D multi-rod).

Concerning RIA, after the REP-Na programme (1992-2000) in the sodium loop of the CABRI reactor, the CABRI International Programme (CIP) launched under OECD auspices with a broad international cooperation and partnership of EDF, is now underway.
The CIP aims at providing, under typical pressurised water reactor conditions, the necessary knowledge for assessment of new RIA related criteria for advanced high burn-up UO_2 and MOX fuels.
The first two reference tests CIP0, have been performed in 2002 in the existing sodium loop with UO_2 fuel at 75 GWd/t and advanced claddings (Zirlo ENUSA rod, M5 EDF rod) and did not led to rod failure (maximum fuel enthalpy of 90 cal/g). The general objectives of the other test series are the following:

- CIPQ : qualification test of the water loop (checking of the absence of artefacts, study of post-boiling rod behaviour)
- CIP1 : test in the water loop with rod failure and comparison to REP Na basis
- CIP2 : study of very high burn-up rods behaviour (85 GWd/t)
- CIP3 : understanding of the physical phenomena : influence of power pulse width (10ms/30 ms) depending on clad corrosion level, effect of initial power level, post-failure phenomena, irradiation history, advanced fuel micro-structure
- CIP4 : study of high burn-up MOX fuel
- CIP5 : open tests (VVER, BWR fuels ?)

The precise definition of the tests is based on quantitative studies using the SCANAIR code.
In addition, the mechanical characterisation of the advanced claddings (Prometra program for Zirlo and M5) is underway, including high temperature hoop tensile and Penn-State tests.
A program aiming at the study of the fission gas dynamic behaviour for UO2 and MOX fuel (impact on clad loading, FGR kinetics) has been proposed for performance in NSRR in cooperation with JAERI.
The complementarity of the CIP and ALPS programs regarding pulse width and coolant conditions should provide valuable knowledge on fuel bahaviour under RIA conditions.

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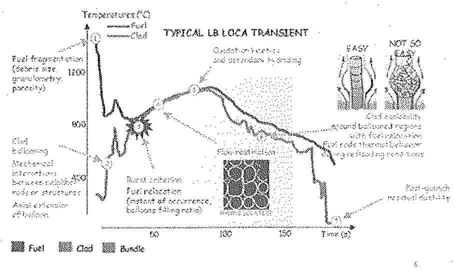

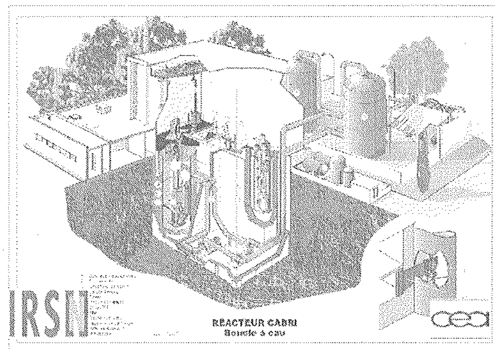

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1

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THE LOCA AND RIA STUDIES

Context and Objectives :

- ✓ **Fuel burn-up increase and use of new products (advanced claddings, MOX, doped fuel,...)**
- ✓ **Need of providing knowledge on UO2 and MOX irradiated fuel for :**
  - **checking/evolution of safety criteria**
  - **Evaluation of safety margins**
  - **Physical understanding, modeling, validation of calculational tools**

The LOCA studies
The CABRI program and CIP tests definition
The Scanair modeling : status and future work

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IRSN LOCA STUDIES AND PROGRAMME

- ✓ Experimental programmes performed to adress clad oxidation and quench resistance :TAGCIS, TAGCIR, HYDRAZIR (hydrided cladding)
- ✓ In-pile tests PHEBUS-LOCA (1980-1984) with fresh fuel in bundle geometry (25 rods, LB)
- ✓ State of the art performed (2000-2003) relative to:
  - clad ballooning-failure-fuel relocation-blockage,
  - coolability of blocked regions,
  - oxidation-quench-post-quench loads resistance

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3

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IRSN LOCA STUDIES AND PROGRAMME

LOCA Pending Questions :

- Impact of fuel relocation on PCT, ECR, secondary hydriding (ANL, Halden tests, possible in CABRI facility)
- Cladding mechanical behaviour under LOCA conditions including PQD
- Coolability of irradiated bundles : maximum blockage ratio with impact of fuel relocation

IRSN R&D programme under elaboration :

- Development of advanced analytical tool (3D multi-rod)
- Interpretation/analysis of the available tests underway (Halden, ANL, JAERI)
- Definition of associated experimental basis for providing lacking knowledge

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THE CABRI RESEARCH PROGRAMME FOR RIA

1st step : CABRI REP Na programme **(sodium loop, 12 tests)**

PCMI phase, rod failure threshold and mechanism, fuel ejection conditions

(1992-2000 with EDF cooperation + US-NRC support)

2nd step : CABRI International programme-CIP **(pressurised water loop)**

(started in 2000)

In parallel : SCANAIR code development, separate effect tests programmes for clad mechanical characterisation, clad to coolant heat transfer , fission gas dynamics.

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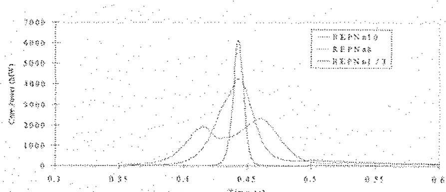
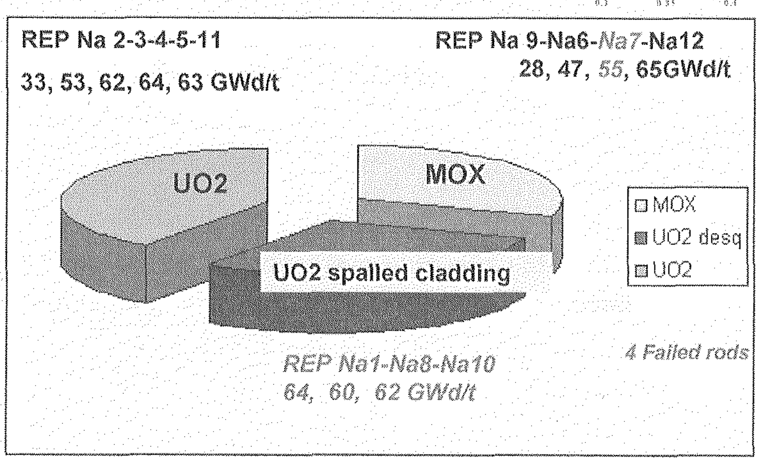
THE CABRI REP Na TESTS PERFORMED IN SODIUM

12 tests

Burnup 28 to 65GWd/t, Cladding: Zr4, M5 (Na11)

Hot zero power RIA simulation ($T_{inlet\ coolant}$=280°C)





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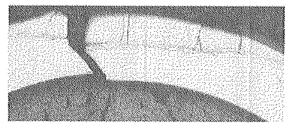
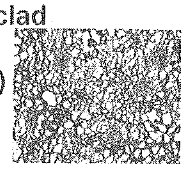
MAIN OUTCOMES OF THE CABRI REP NA PROGRAMME

- ✓ Need of evolution of present safety criteria **(rod failure with fuel ejection at low enthalpy level : 30-113 cal/g)**
- ✓ Enhanced risk of rod failure due to high corrosion level with hydride concentration **(in reactor oxide spalling, rim)**



REP Na8 clad at PPN

- ✓ Contribution of fission gases from grain boundary **(fuel fragmentation, clad loading and fission gas release increasing with burnup), especially in MOX (heterogeneous microstructure due to Pu rich zones)**



REP Na6 pellet periphery

- ✓ Influence of energy injection rate **: failure and fuel dispersion risk increased with short pulses**
- ✓ Transient oxide spalling **influencing clad to coolant heat transfer**

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THE CABRI CIP PROGRAMME

Objectives :

- **Advanced fuel (UO2+MOX) behaviour under RIA conditions in fully typical conditions (fuel, pulse, coolant conditions : 155b, 280° C)**
- **Data acquisition for :**
  - **Bases for assessment of new RIA criteria,**
  - **Understanding and modeling of physical phenomena, SCANAIR validation**
  - **Evaluation of safety margins**

The Programme : *14 partners, 11 countries*

- **12 tests : 2 in the sodium loop(CIP0, 2002) + 10 in the pressurised water loop**
- **Mechanical characterisation of advanced cladding (Prometra tests)**
- **Complementarity with the NSRR/ALPS programme : use of twin rods in different conditions (pulse width, coolant conditions)**

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PENDING QUESTIONS AND PHYSICAL PHENOMENA TO BE STUDIED

- Clad failure conditions in pressurized water environment
- Rod behaviour after clad heat-up (boiling crisis onset, clad creep, FGR, fuel swelling)
- Post-failure phenomena : ejection of fragmented solid fuel, fuel/coolant interaction
- Impact on phenomena of : energy injection rate (pulse width), clad corrosion, fuel micro-structure, initial power level, irradiation history.
- Fission gas dynamic behaviour for UO2 and MOX; impact on clad loading, FGR kinetics : *program proposal in NSRR in cooperation with JAERI*

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THE CIP TESTS

CIP0 series : Advanced UO2 fuel rods (CIP0-1 and CIP0-2 in Na loop)

CIPQ test : Post-boiling phase (WL qualification test)

CIP1 series : Link with REP Na data /rod failure +post-failure phenomena

CIP2 series : Impact of very high burnup (UO2 ~ 85 GWd/t)

CIP3 series : - pulse width (10/30ms) influence,
- irradiation history influence
- advanced fuel microstructure
- RIA at intermediate power level

CIP4 series : MOX fuel at high burnup : influence of microstructure

CIP5 series : On going discussion on BWR, VVER tests

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10

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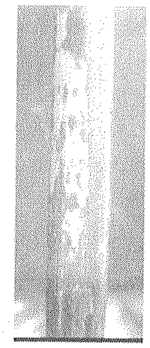
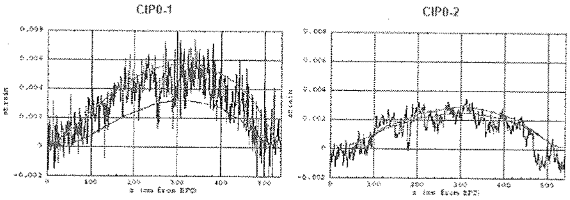
The CIP0 Tests

- CIP 0 : 2 reference tests in the sodium loop, UO_2 fuel ~75 GWj/t, Zirlo (CIP0-1) and M5 (CIP0-2) claddings, 30ms pulse width
- No rod failure for Hmax~ 90 cal/g (CIP0-1), 80cal/g (CIP0-2)



Desquamation transitoire CIP0-1

Scanair calculations :



Legend: —— SCANAIR with min ZrO_2 thickness after test ······ SCANAIR with max ZrO_2 thickness after test

Max. experimental results: 0.5 +/- 0.1%
Max. SCANAIR calculation: 0.45 +/- 0.1%

Max. experimental results: 0.27 +/- 0.04%
Max. SCANAIR calculation: 0.26 +/- 0.01%

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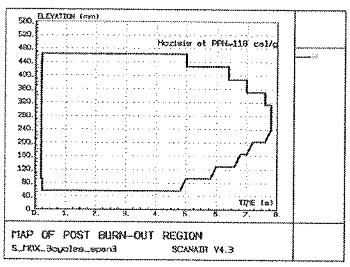
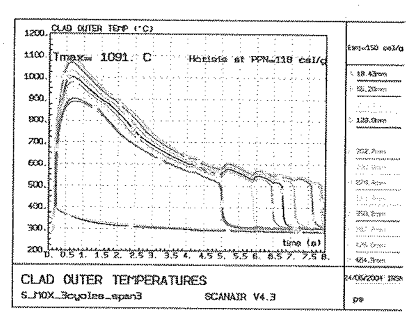
CIPQ test definition

CIPQ : Water loop qualification test
- verification of absence of artefacts linked to test device
- rod behaviour after boiling crisis +comparison with REP-Na for PCMI phase

Based on Scanair studies :

Twin test of REP-Na 6 (MOX 3 cycles-47 GWd/t, 156 cal/g Einj, 35ms)
- minimised risk of failure before boiling
- H Boiling crisis in the range 80-120 cal/g
- Dry-out duration : several seconds , allowing post-BO clad deformation





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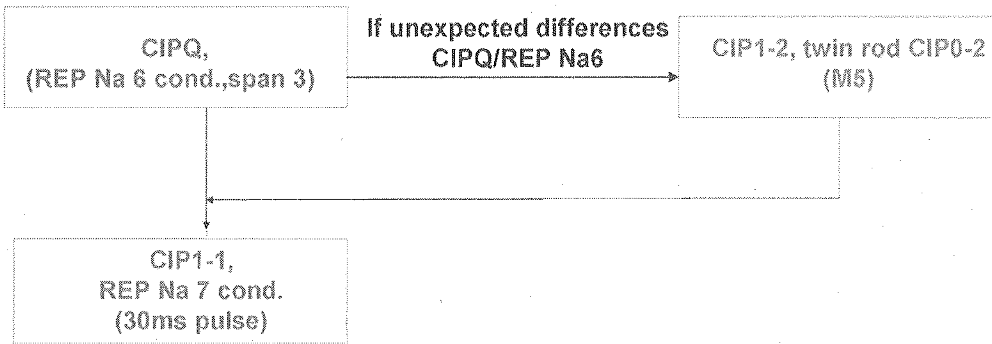
CIP1 test definition

CIP 1 : failure conditions : comparison with REP-Na



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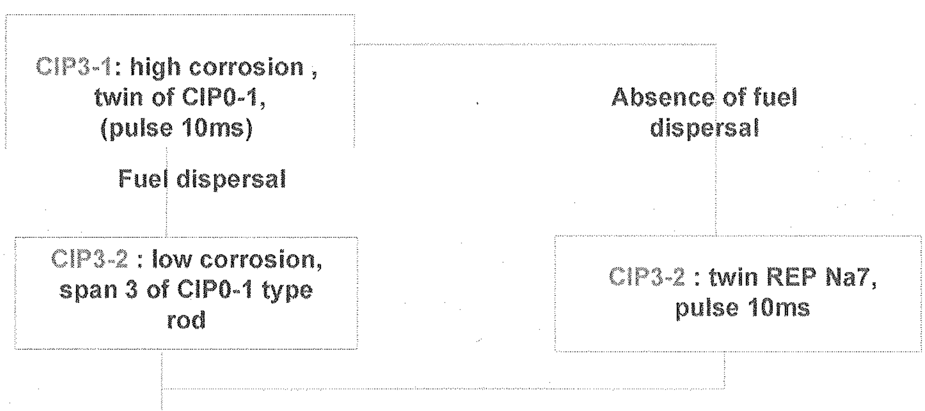
CIP3-1 et CIP3-2 tests

Objective: influence of a fast pulse (10ms/30ms) on failure + post-failure phenomena, depending on clad corrosion



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PRESENT CIP TEST MATRIX

SERIES	TEST N°	OBJECTIVES/DESCRIPTION	TEST ROD	COMMENTS
CIP0	CIP0-1	Reference tests in Na loop, pulse 30ms, Edep≈ 100cal/g	UO2 75 GWJ/t, Zirlo clad (Vandellos,Spain)	Tests realized in 2002 Link with NSRR/ALPS
	CIP0-2		UO2 77GWJ/t, M5 clad (EDF-Alix 6 cycles)	
CIPQ	CIPQ	Qualification test of the WL Activation of post-DNB phenomena	Span 3 of REP Na6 (MOX-3cycles, Zr4, EDF)	1er test in WL comparison with REP Na6 (phase initiale)
CIP1	CIP1-1	Link with REP Na database / failure, post-failure (pulse 30ms)	Twin REP Na7 (MOX-4cycles, Zr4)	Comparison with REP Na7
	(CIP1-2)	Link with CIP0 if unexpected differences CIPQ/REP Na6	Twin CIP0-2, UO2 77GWJ/t M5 (EDF)	*Option depending on CIPQ results*
CIP2	CIP2-1	Impact of ultra high burn-up on failure risk	UO2-85GWJ/t, M5 (EDF Alix 7 cycles)°	Référence Link with NSRR/ALPS
	(CIP2-2)		UO2-85 GWJ/t, Zirlo (Vandellos)	Fuel with large grain size, *test under discussion candidate for CIP3,* Link with NSRR/ALPS

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15

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PRESENT CIP TEST MATRIX

SERIES	TEST N°	OBJECTIVES/DESCRIPTION	TEST RODS	COMMENTS
CIP3	CIP3-1	Impact fast pulse (10ms) + high corrosion ; rod failure, post-failure phenomena, fuel dispersal	UO2 75 GWJ/t, Zirlo (Vandellos, span 5) twin CIP0-1	Comparison with CIP0-1 (pulse 10/30ms)
	CIP3-2	Fuel dispersal (if not obtained in CIP3-1), 10ms pulse	MOX 4 cycles, Zr4 (EDF), twin REP Na7	Choice of objectives depending on CIP3-1
		Impact fast pulse (10ms) + low clad corrosion (fuel influence)	UO2 75 GWJ/t, Zirlo (Vandellos, span 3) twin CIP0-1	
CIP4	CIP4-1	Impact of MOX micro-structure	MOX-E, M5, ~65 GWJ/t (EDF)	Heterogeneous structure Link with NSRR/ALPS
	CIP4-2		MOX-SBR-BNFL, Zr4, ~60 GWJ/t (Beznau, Switzerland)	Homogeneous tructure Link with NSRR/ALPS
Under discussion : simulation of RIA at intermediate initial power , study of new fuel microstructures, impact of irradiation history, open tests (BWR, VVER)				

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16

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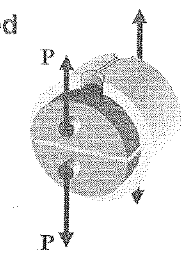
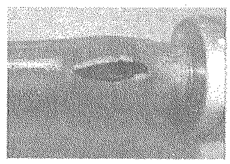
The Clad Mechanical Characterisation Programme PROMETRA

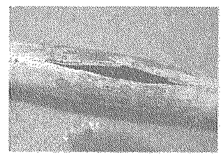
Objectives : provide the stress-strain laws (hoop tensile tests and their interpretation) and failure data (Penn-State + burst tests)

- ✓ Extensive data base for irradiated Zr4 *(cf SMIRT-15 1999, ASTM-2004)*
- ✓ Advanced cladding M5, Zirlo
  - 12 hoop tensile tests : 280, 480, 600, 700, 800° C at 1/s+ 480° C at 0,01/s (high T tests with induction heating)
  - 2 burst tests :T=280° C, 0.0025/s
  - Penn-state tests with irradiated to be performed end 2005 (280, 480, 600, 800° C at 1/s)





M5



Zirlo

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The SCANAIR Modeling (1/2)

Validation on REP Na tests

Thermal

Fig. 31. Na max. temperatures: comparison between SCANAIR calculations and measurements (UO$_2$ + MOX tests).

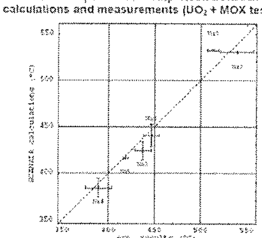
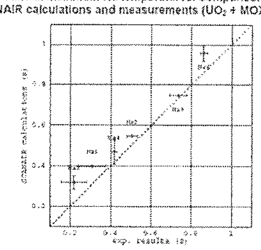
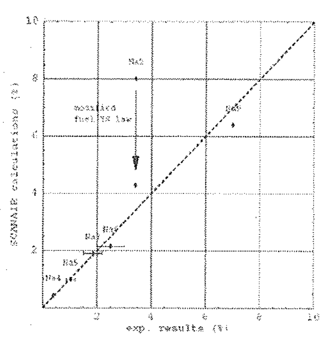


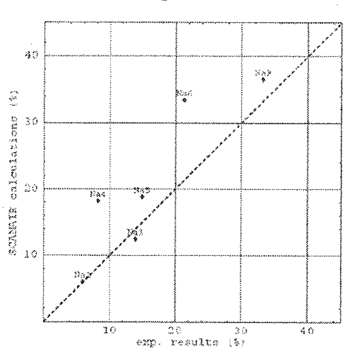
Fig. 32. Time at maximum Na temperatures: comparison between SCANAIR calculations and measurements (UO$_2$ + MOX tests).



Mechanical



Fission gas release



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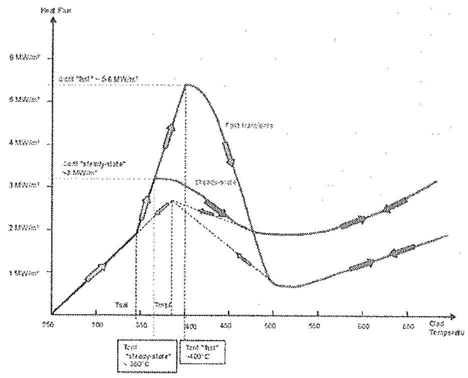
The SCANAIR Modeling (2/2)

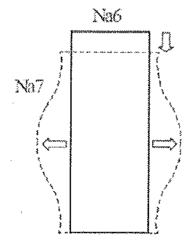
Recent developments :

- PCMI failure probability based on fracture mechanics
- Clad to coolant heat transfer based on PATRICIA experiments (*ref FSRM 2004)*
    - impact of Zr02 layer : not investigated
    - validation on NSRR tests to be performed



Future work

- Post-boiling crisis behaviour : clad creep
- Fuel creep : MOX fuel (REP Na6/Na7)
- Post-failure phenomena: fuel ejection, channel pressure build-up



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19

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CONCLUSION

- Need of an advanced tool taking into account detailed thermal and thermo-mechanical rod behaviour for LOCA prediction
- Important knowledge for understanding and modeling of irradiated fuel behaviour under RIA, provided from the former CABRI REP Na and separate effect test programs
- Proposal for fission gas dynamic behaviour study in NSRR with JAERI cooperation
- CIP program now underway : representative PWR conditions, quantitative studies for test definition

Complementarity of ALPS and CIP programs is of high interest and value for RIA study

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20

Session 1-4

FUEL SAFETY RESEARCH AT JAERI

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To provide a data base for the regulatory guide of light water reactors, behaviors of reactor fuels during off-normal and postulated accident conditions such as reactivity-initiated accident (RIA) and loss-of-coolant accident (LOCA) are being studied at the Japan Atomic Energy Research Institute (JAERI). The research program at JAERI is comprised of ;

- A series of RIA-simulating experiments with high burnup fuel rods by using pulse-irradiation capability of the NSRR (Nuclear Safety Research Reactor),
- Separate-effect tests in the NSRR to study specific phenomena observed in RIA fuel behavior, such as tests with fuel particles in zip-lock in order to study mechanical forces generation in PCMI failure and tests with artificially-hydrided cladding to simulate a PCMI failure.
- LOCA-related experiments including rod quench test, tube burst test and cladding oxidation test at high temperature,
- Development and verification of fuel behavior codes including high burnup fuel behavior code FEMAXI and newly-developed RIA fuel behavior code RANNS,
- Development of new methods in cladding mechanical testing, such as cladding fracture toughness measurement and EDC (expansion due to compression) test, and
- VEGA (Verification Experiments of radionuclides Gas/Aerosol release) program to understand mechanisms of radionuclides release from fuels under simulated severe accident conditions.

The presentation gives an outline of the above research activities. Regarding the RIA study, in particular, recent proposals of the regulatory criteria for high burnup fuels are also discussed.

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1

This presentation will cover;

✓ Outline of JAERI research activities on fuel safety, including;

- NSRR/RIA-simulating experiments with high burnup LWR fuels, ➔ T. Sugiyama, H. Sasajima
- Separate-effect tests in the NSRR for hydrided-assisted PCMI failure and post-DNB/post-failure events, ➔ K. Tomiyasu
- LOCA-related experiments such as rod quench test, ➔ F. Nagase
- Development and verification of fuel behavior codes, ➔ M. Suzuki
- New methods in cladding mechanical tests, ➔ N. Ikatsu
- High burnup fuel properties and rim structures ➔ M. Amaya
- VEGA program for radionuclides release from fuels under simulated severe accident conditions, ➔ T. Kudo

✓ RIA database and criteria

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Fuel safety research at JAERI

②

The objectives are;

- ✓ To evaluate adequacy of present safety criteria and safety margins,
- ✓ To provide a database for regulation on higher burnup fuels and MOX fuels in LWRs, and
- ✓ To promote a better understanding of phenomena appearing in high burnup region and MOX, such as rim-effect and Pu agglomerates in MOX, and to evaluate those effects on fuel behavior in accident conditions.

The research activities are being performed in;

- ✓ Department of Reactor Safety Research
  - Fuel Safety Research Laboratory
  - NSRR Operating Division
- ✓ Department of Hot Laboratory

collaborating with Department of Safety Research Technical Support and Department of JMTR

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Separate-Effect Tests in the NSRR

③

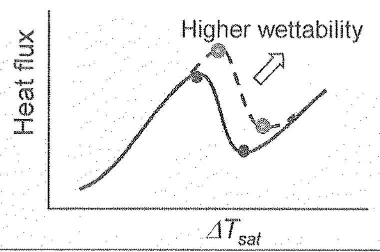
Pulse-irradiation with;

Fresh rod with artificially-hydrided cladding

to evaluate an effect of hydride rim on a PCMI failure

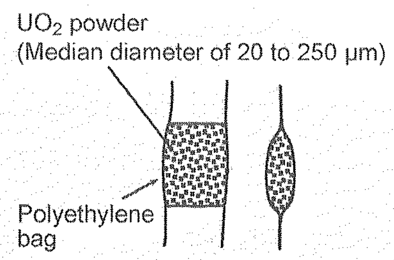
Fresh rod with pre-oxidized cladding

to investigate an effect of surface wettability on transient heat transfer, e.g. occurrence of DNB



Un-claded fuel powders

to assess thermal interactions between fragmented fuel pellets and coolant water after PCMI failure in high burnup fuels



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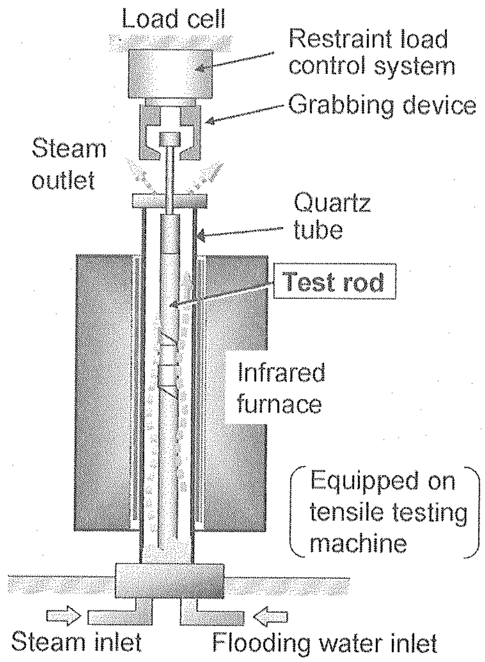
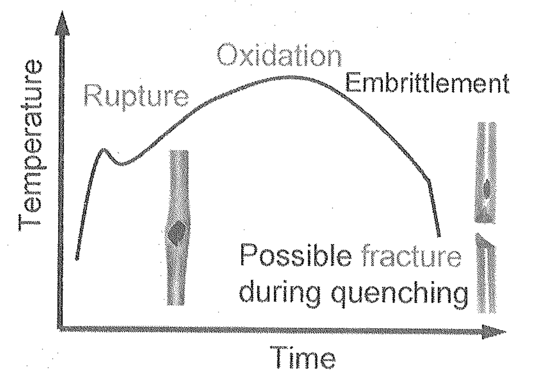
LOCA Tests

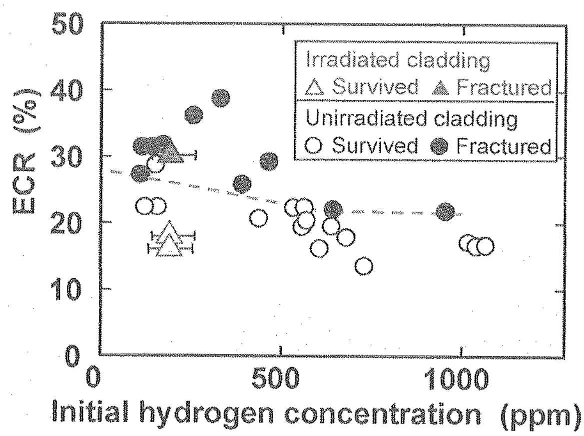
④

Integral thermal shock test



✓ With fully controlled axial constraint





LOCA Tests

⑤

Tube burst tests

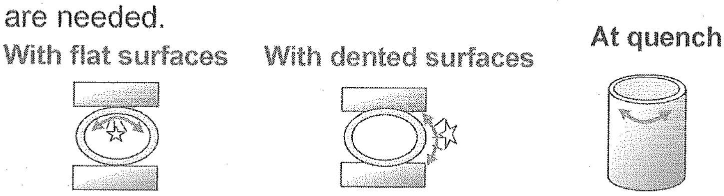
Oxidation test at high temperature

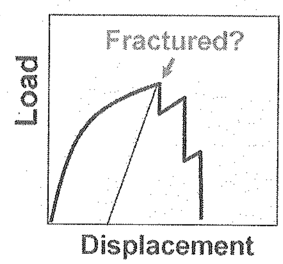
Ring Compression tests

Ring compression test is simple and can produce numerous data, but

- ▶ Well-defined 'Zero-ductility', and
- ▶ Standardization of test methodology

are needed.





JAERI LOCA-database

Data at higher burnups		(pellet burnup: MWd/kg)
	FY2004	until FY2007
UO_2 :	PWR 48	PWR ~80 BWR ~65

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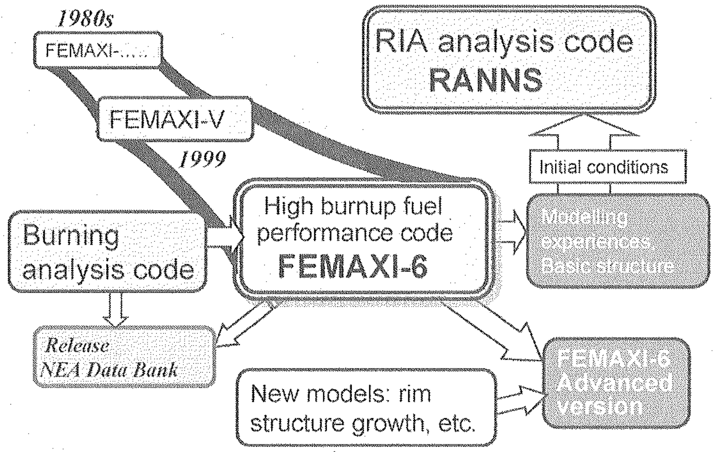
Development of Fuel Behavior Codes

⑥

Fuel behavior code FEMAXI has been developed at JAERI from 1980s, and FEMAXI-6 for high burnup fuel behavior was released. Upgrade of the code is being done in terms of rim effect, etc.

Based on the structure of the FEMAXI-6 code, a code RANNS is being developed for quantitative evaluation on fuel behaviors during RIAs.



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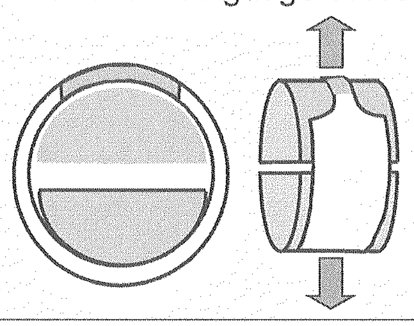
Cladding Mechanical Tests

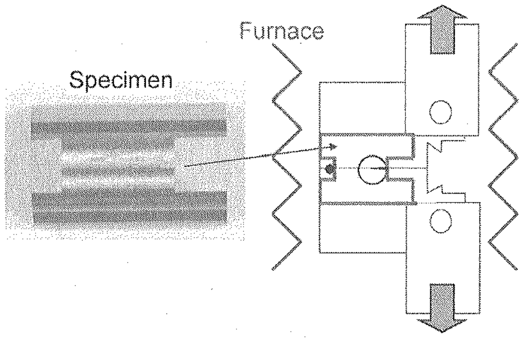
⑦

In order to assess hydrides effect on cladding integrity, artificially hydrided cladding samples are being subjected to several mechanical tests. Hydrogenation was performed in mixture gas of hydrogen and argon at about 600 K.

✓ Ring-tensile test with one-side gauge section



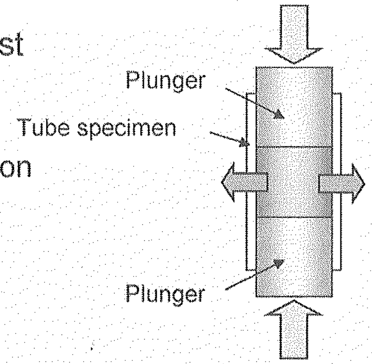
✓ Fracture toughness measurement



✓ Transient tube burst tests

✓ EDC test

Expansion Due to Compression

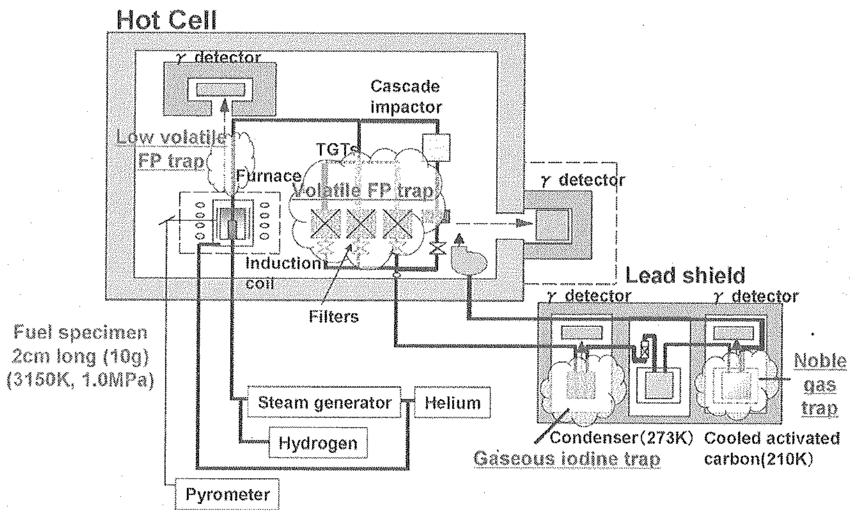


VEGA Program

8

VEGA (Verification Experiments of Radionuclides Gas/Aerosol Release) program is being performed to investigate release of FPs and actinides from fuels under severe accident conditions.

Temperature up to 3150 K and pressure up to 1 MPa can be achieved with the apparatus under helium and steam environment. Both of irradiated UO_2 and MOX fuel specimens are subjected to the test.



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International and Domestic Collaboration

9

International

- ✓ USNRC (ANL, PNNL, PSU, etc.)
- ✓ IRSN
- ✓ OECD Halden Reactor Programme
- ✓ OECD-IRSN CABRI Water Loop Project (under discussion)

Domestic

- ✓ JNC, JNES, BWR&PWR utilities, CRIEPI, MHI, NFI, GNF

Advanced LWR Fuel Performance and Safety Research Program

performed as a program of METI/NISA
collaborating with BNFL, Framatome, Studsvik, OECD Halden, PSI, FEPC, MHI, NFI, BWR&PWR utilities, etc.

'Crossover' program

A study for irradiation-induced material processes

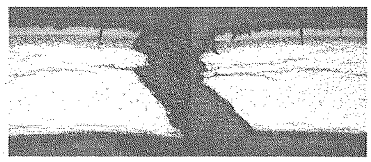
JAERI (managing organization), M. Kinoshita, CRIEPI (program leader)
NIMS (National Institute for Materials Science)
Hokkaido University, the University of Tokyo, Kyushu University

10

NSRR/RIA Experiments with irradiated LWR fuels

Test fuels	Fuel burnup (MWd/kg) 10 20 30 40 50 60	Number of tests
PWR (14x14, 17x17)		26
BWR (7x7, 8x8)		16
ATR/MOX		6
JMTR pre-irradiated		22

✓ To provide a database for regulatory criteria regarding failure threshold, coolability limit, etc.



Key observations

✓ Hydride-assisted PCMI failure
✓ Fuel dispersal and mechanical energy generation
✓ Large rod expansion and fission gas release

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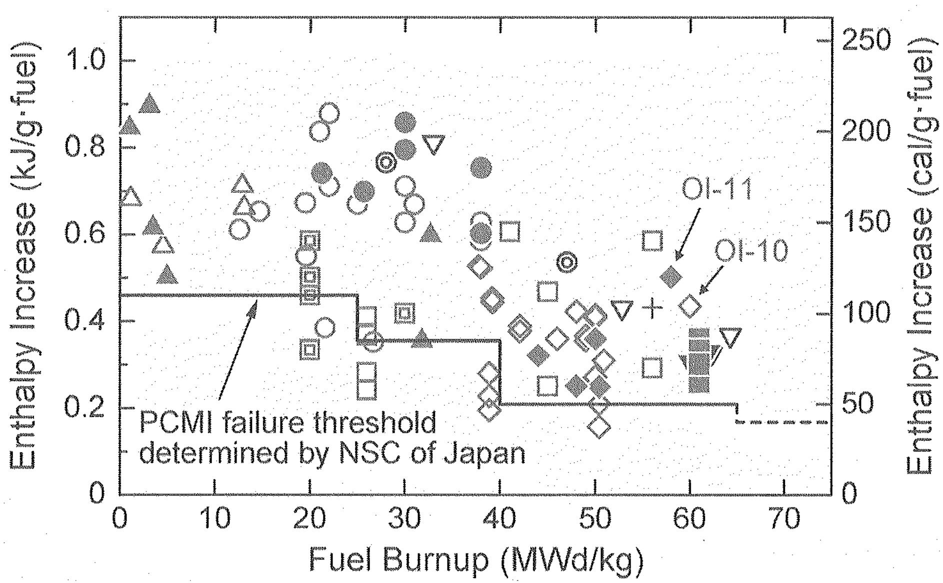
11

Existing data for failure threshold

Test ID		No failure	Failure	Test ID	No failure	Failure
NSRR	PWR	◇	◆	SPERT, PBF	△	▲
	BWR	□	■	CABRI UO_2	▽	▼
	ATR/MOX	⊡	——	CABRI MOX	◎	+
	JMTR	○	●	——	——	——



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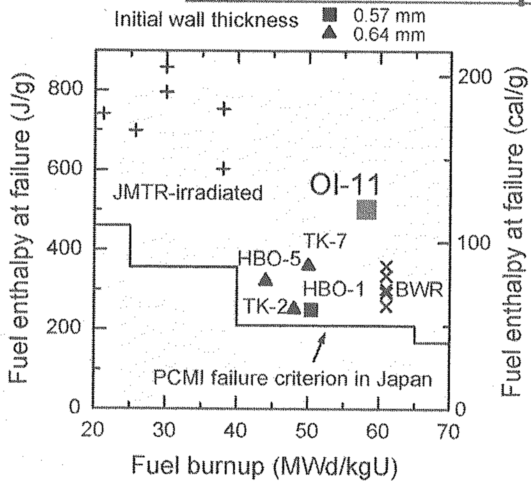
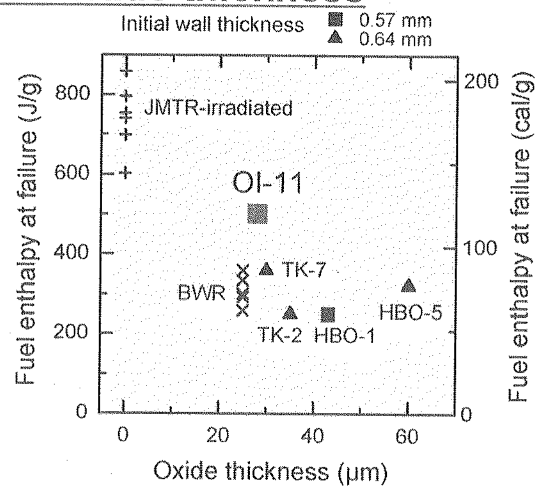
PCMI failure thresholds in terms of burnup and oxide thickness

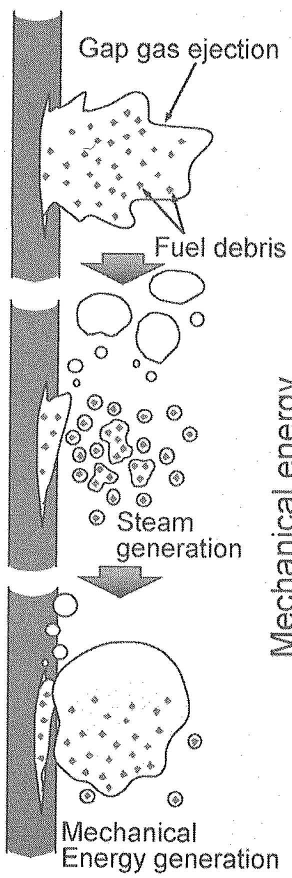
(12)





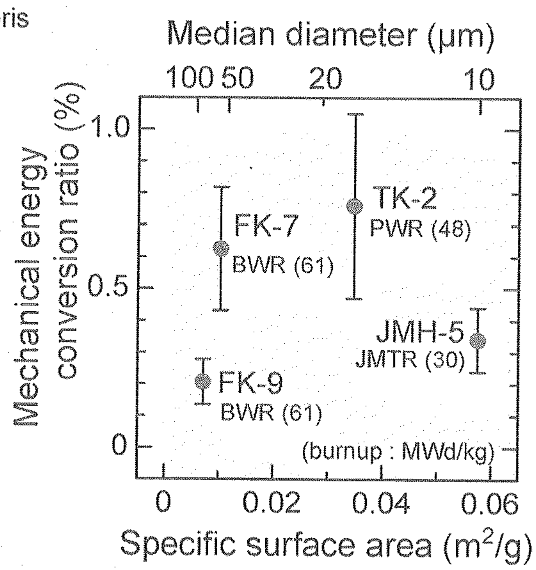
- ✓ Better performance of new design, such as improved corrosion-resistance in new alloys, can not be reflected to the current criterion.
- ✓ A threshold as a function of oxide thickness includes the corrosion-resistance only, and requires validation testing for the oxide thickness.
- ✓ Choosing better parameters for ordinate and abscissa axes is being widely and actively discussed. The bounding condition can be described better with numerous parameters. Regulatory criteria, however, must look simple.
- ✓ A final goal is a regulation with a reliable, mechanistic and well-verified code.

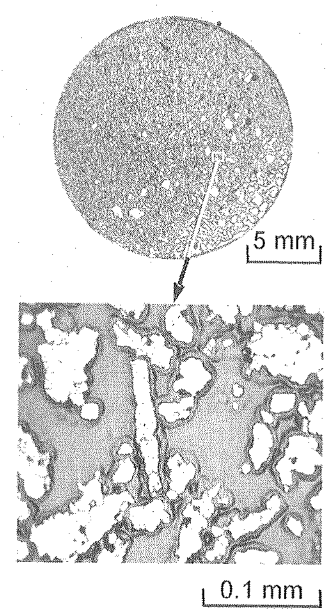
(13)



✓ Fuel dispersal and mechanical energy generation

- Database for coolability limit





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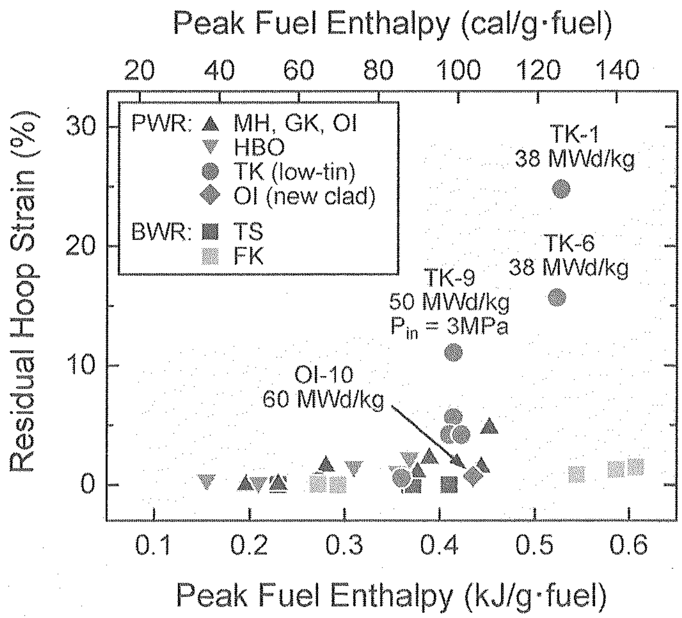
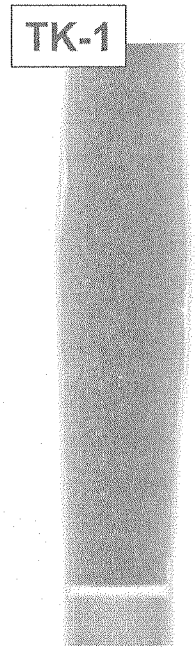
✓ Post-DNB large rod expansion

(14)

Residual Hoop Strain as a function of Peak Fuel Enthalpy





✓ **24.8% at TK-1 and 15.7% at TK-6 with ~0.52 kJ/g (~125 cal/g)**

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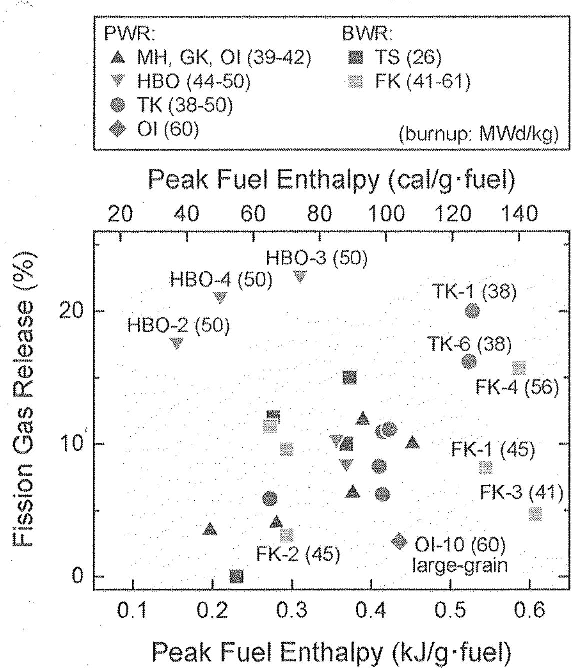
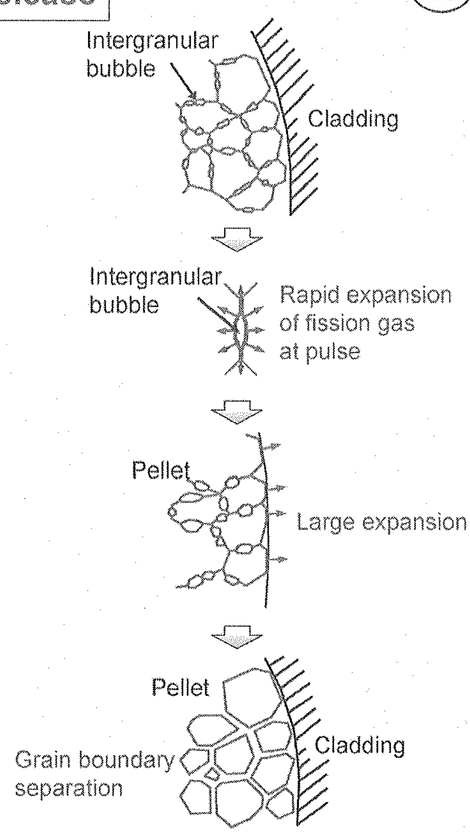
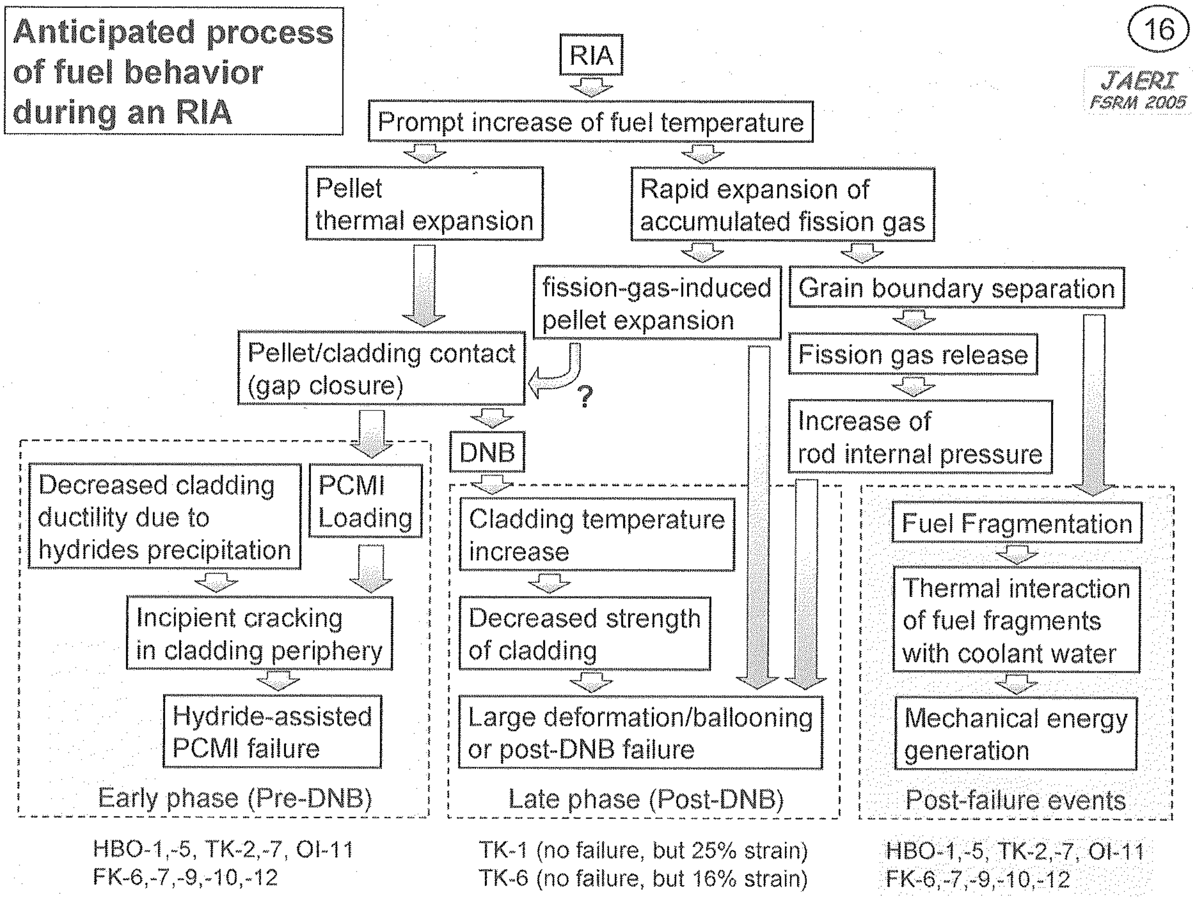
✓ Large fission gas release

(15)

Fission Gas Release as a function of Peak Fuel Enthalpy






Anticipated process of fuel behavior during an RIA
16
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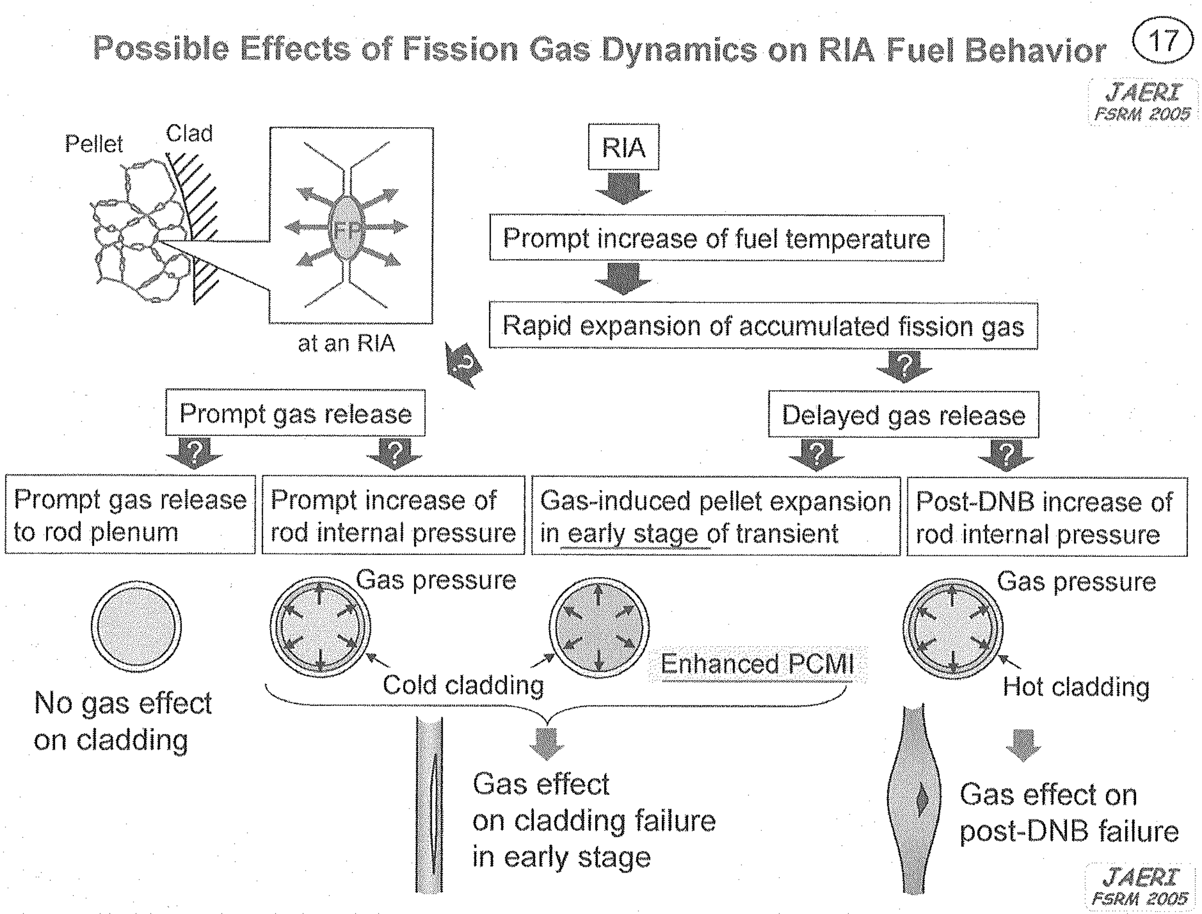
RIA
Prompt increase of fuel temperature
Pellet thermal expansion
Rapid expansion of accumulated fission gas
fission-gas-induced pellet expansion
Grain boundary separation
Pellet/cladding contact (gap closure)
?
Fission gas release
Increase of rod internal pressure
DNB
Decreased cladding ductility due to hydrides precipitation
PCMI Loading
Incipient cracking in cladding periphery
Hydride-assisted PCMI failure
Early phase (Pre-DNB)
Cladding temperature increase
Decreased strength of cladding
Large deformation/ballooning or post-DNB failure
Late phase (Post-DNB)
Fuel Fragmentation
Thermal interaction of fuel fragments with coolant water
Mechanical energy generation
Post-failure events
HBO-1,-5, TK-2,-7, OI-11
FK-6,-7,-9,-10,-12
TK-1 (no failure, but 25% strain)
TK-6 (no failure, but 16% strain)
HBO-1,-5, TK-2,-7, OI-11
FK-6,-7,-9,-10,-12


Possible Effects of Fission Gas Dynamics on RIA Fuel Behavior
17
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Pellet
Clad
FP
at an RIA
RIA
Prompt increase of fuel temperature
Rapid expansion of accumulated fission gas
?
?
Prompt gas release
Delayed gas release
?
?
?
?
Prompt gas release to rod plenum
Prompt increase of rod internal pressure
Gas-induced pellet expansion in early stage of transient
Post-DNB increase of rod internal pressure
Gas pressure
Gas pressure
No gas effect on cladding
Cold cladding
Enhanced PCMI
Hot cladding
Gas effect on cladding failure in early stage
Gas effect on post-DNB failure
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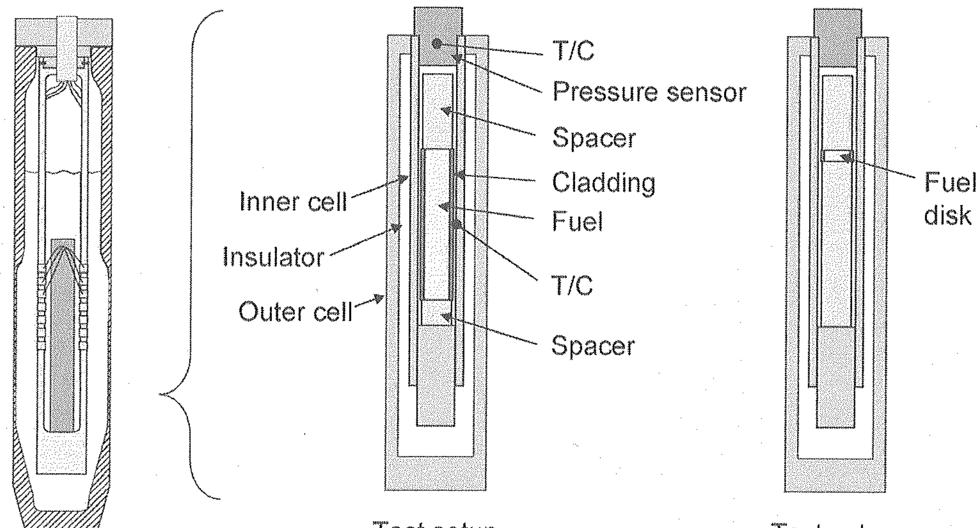
FGD (Fission Gas Dynamics) test

18

- ✓ To evaluate an effect of fission gas dynamics on high burnup fuel behaviors, such as PCMI, ballooning and fuel fragmentation.
- ✓ To investigate an influence of high burnup structure on the behaviors.

Transient measurement of fission gas release, i.e. measurement of fission gas release rate, with an accurate detection of pressure increase.



Test capsule

Test setup
(segmented fuel)

Test setup
(Fuel disk)

To be defined and prepared jointly with French IRSN and performed in the NSRR

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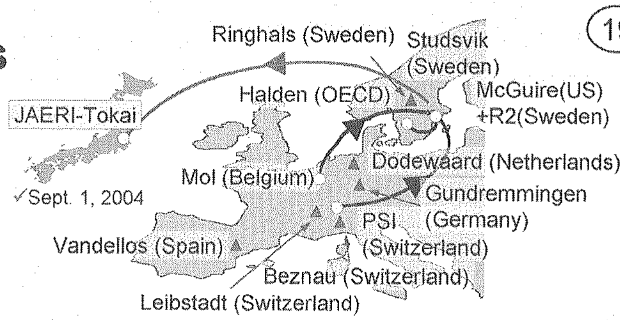
NSRR/RIA experiments

19

Experiments on
high burnup UO_2 and MOX fuels
from European/American reactors.



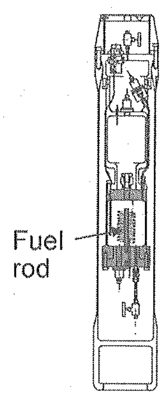
Data at higher burnups

(pellet burnup: MWd/kg)

		FY2003		
UO_2 :	PWR	60	~80	until FY2005
	BWR	61	~73	
MOX :	ATR	30	PWR ~60	until FY2006
			BWR ~45	

Test at higher temperature

Coolant water temperature
from 20 (room temp.) to 90 deg C → 286 deg C



HTHP (high-T, high-P)
test capsule

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RIA Criteria in Japan

20

RIA Guideline (July 19, 1984)

- Fuel enthalpy limit (Absolute limit)
- Safety design limits (Rod failure threshold)
- Mechanical energy generation at failure of water-logged fuel

⇐ **NSRR tests with fresh fuels**

- Provisional failure threshold for burnup fuels ⇐ **SPERT and PBF tests**

RIA Criteria for Burnup Fuels (April 13, 1998)

- PCMI failure threshold
- Mechanical energy generation at failure of burnup fuels
- Evaluation on fission gas release, coolability of dispersed fuels, etc.

⇐ **NSRR tests with burnup fuels**

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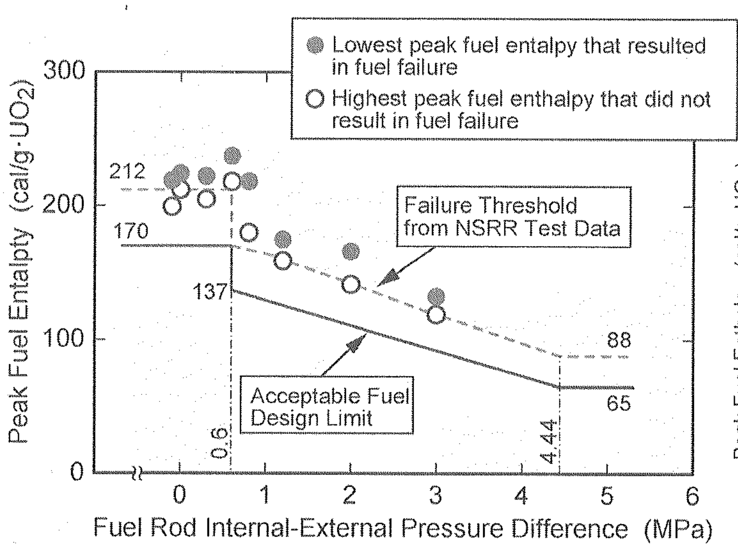
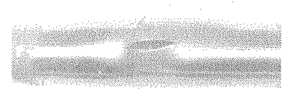
Safety Design Limits

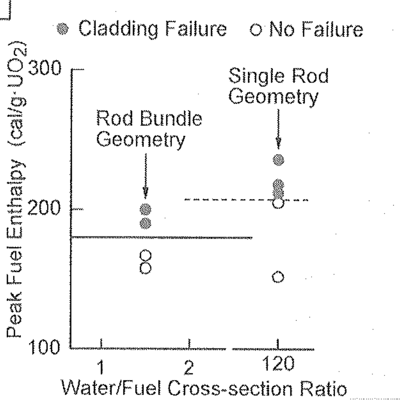
21

Threshold of fuel failure in terms of peak fuel enthalpy and fuel rod internal pressure to prevent fuel failure during abnormal transients and to estimate number of failed rods during an RIA

Threshold from the NSRR single pin experiments
➡ 15% reduction due to decreased coolability in bundle
10 cal/g reduction due to uncertainty of the data







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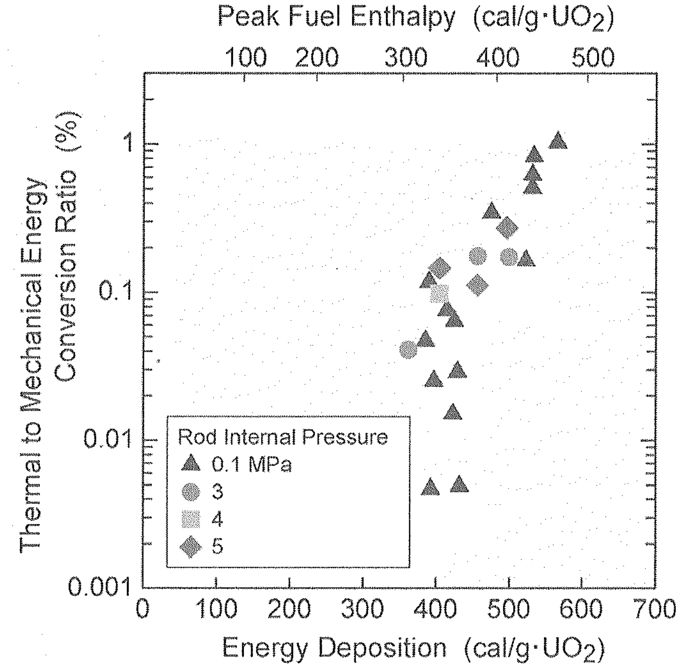
Fuel Enthalpy Limit

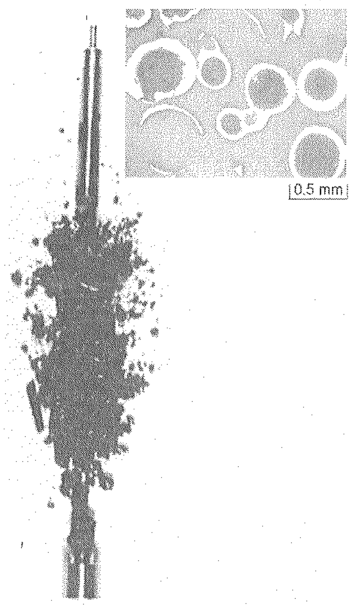
(22)

Threshold of fuel fragmentation and mechanical energy generation in the NSRR single pin experiments ➡ **285 cal/g**

➡ 15% reduction due to decreased coolability in bundle

10 cal/g reduction due to uncertainty of the data ➡ **230 cal/g**





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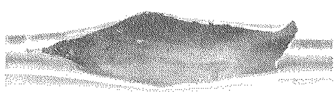
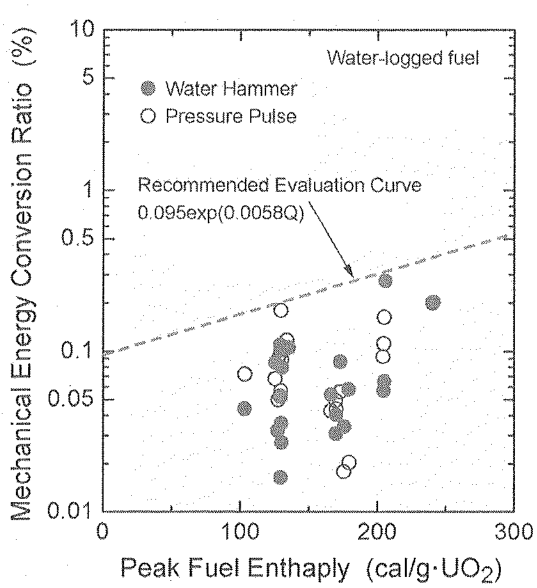
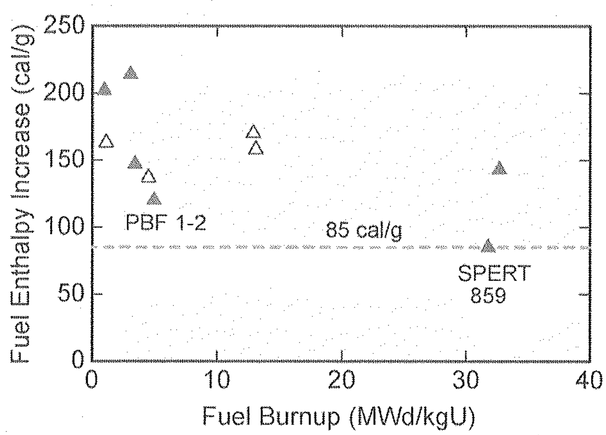
Mechanical Energy Generation at Failure of Water-Logged Fuel

(23)





Provisional Failure Threshold for Irradiated Fuels



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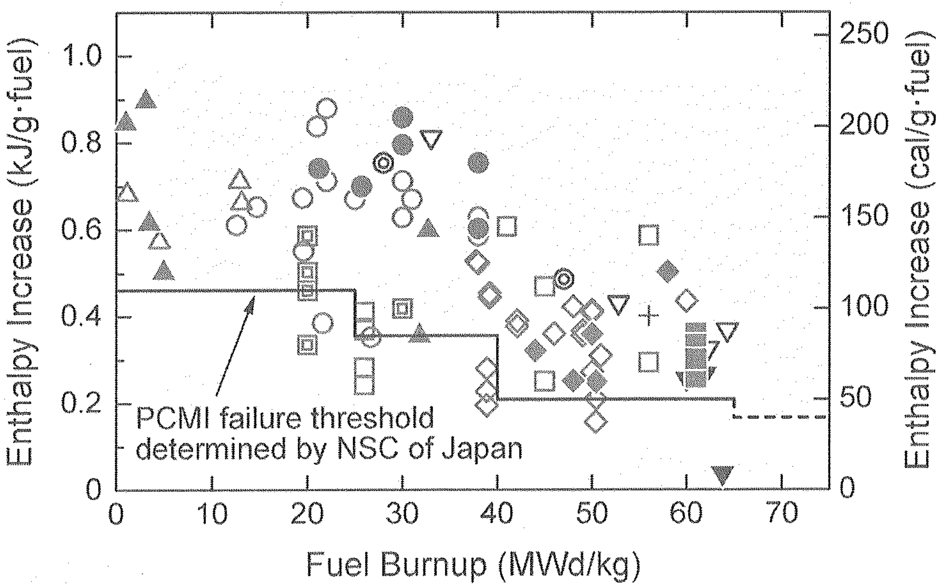
24

Threshold of PCMI Failure



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25

Absolute limit to avoid fuel fragmentation and mechanical energy generation due to fuel melting (**Coolability Limit**)

Max. enthalpy must be less than 230 cal/g minus enthalpy corresponding to decrease of melting point due to burnup, additives, MOX effect, etc. One can ignore the decrease up to 30 MWd/kgU and can assume the decrease of UO_2 melting point as 32 degree per 10 MWd/kgU.

Mechanical energy due to PCMI failure

Failure of whole grid span containing a pellet with enthalpy higher than the PCMI failure threshold should be assumed. Regarding thermal to mechanical energy conversion ratio, a formula identical to that in rupture of water-logged fuel. Safety evaluation must assume that the PCMI failure of burnup fuels and rupture of water-logged fuel occur simultaneously.

Coolability of dispersed fuel fragments accumulated in the bottom of PV

20% of fuels in grid span containing a pellet with enthalpy higher than the PCMI failure threshold is assumed to be dispersed and deposited. Safety evaluation must assume that the PCMI failure of burnup fuels and rupture of water-logged fuel occur simultaneously. Use Lipinski's 0-dimensional model in dryout evaluation.

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26

New RIA Criteria

- ✓ New RIA criteria have been and are being proposed by industry and safety organizations in many countries.

- ✓ Can be categorized into two regulatory approaches.

  #1 A coolability limit (an absolute limit of fuel enthalpy) and a fuel failure threshold. ➔ Japanese criteria, EPRI's proposal

  #2 Only one limit ➔ French limit, a proposal from USNRC/RES

  USNRC/RES proposed an RIA limit in the research information letter (RIL#0401) to USNRC/NRR. "Without cladding failure, there can be no fuel dispersal, no damaging pressure pulses, and no loss of coolable geometry."

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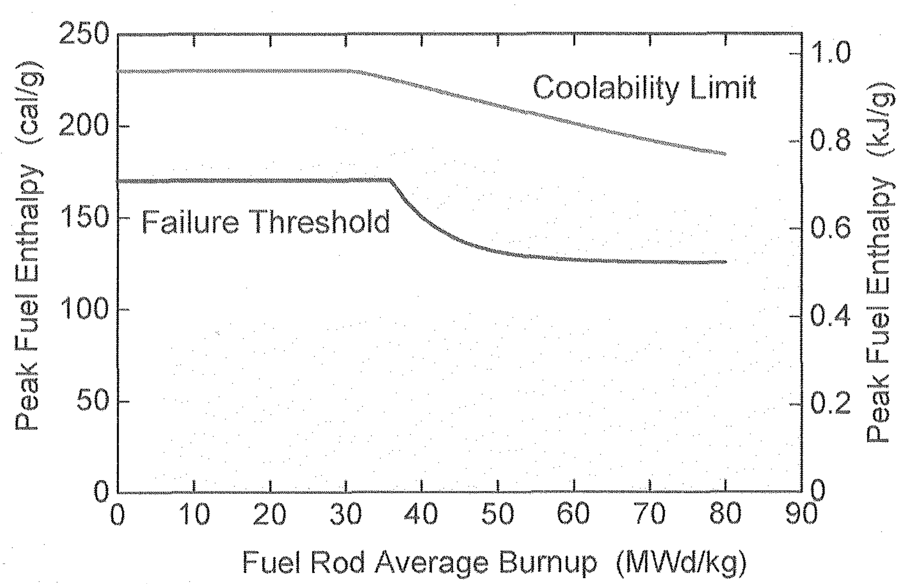
27

Example of Proposed Fuel Coolability Limit and Failure Threshold

(Courtesy of R. Yang, EPRI)



A coolability limit and a failure threshold.

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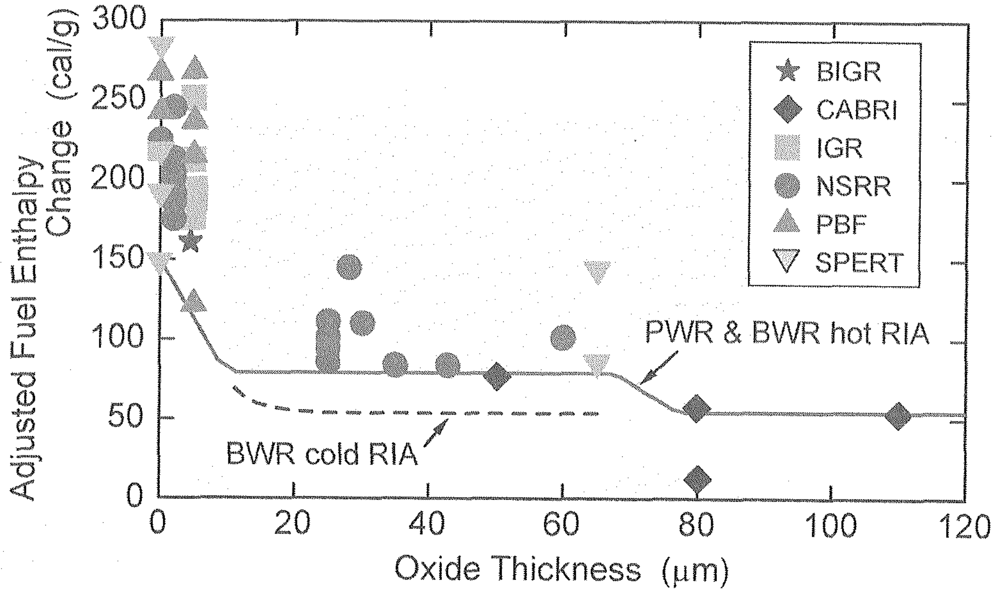
28

Example of Proposed Failure Threshold

Research Information Letter No.0401 from USNRC/RES (March 31, 2004)



Only one limit, i.e. the failure threshold.

✓ Avoiding fuel failure ensures coolability.

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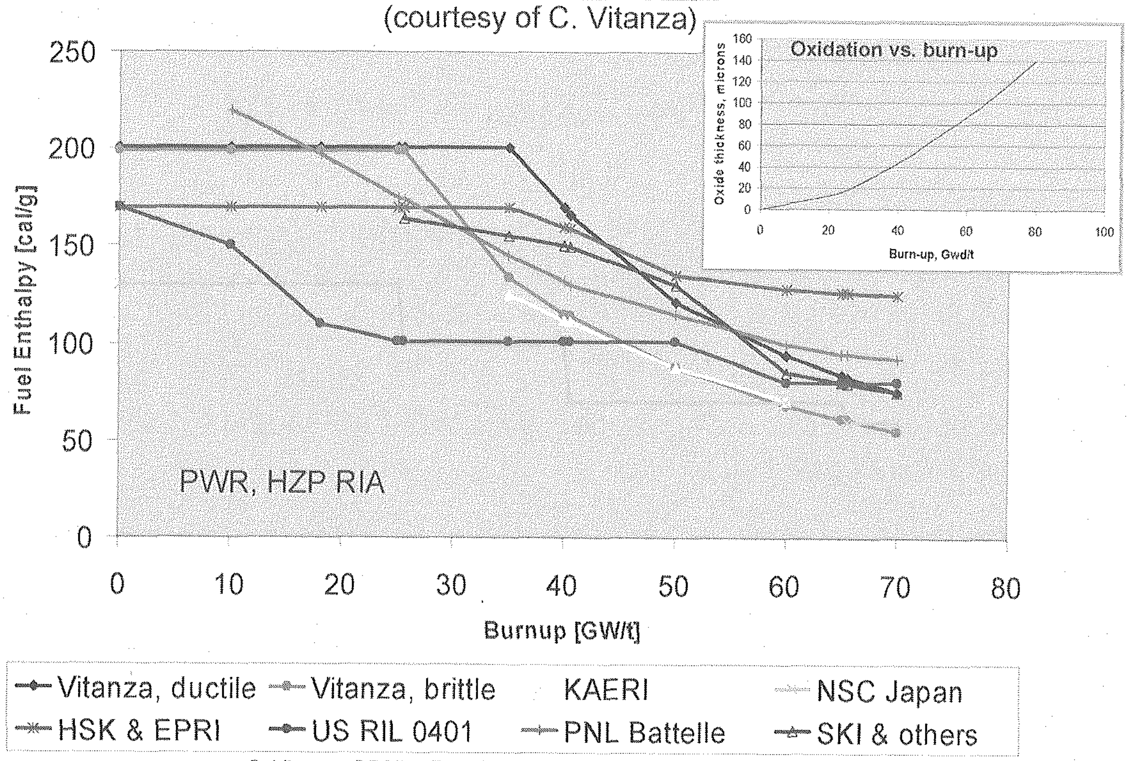
Comparison of Existing and Proposed Failure Thresholds

29

(courtesy of C. Vitanza)



C. Vitanza, OECD-NEA, "Overview of Burn-up Dependent RIA and LOCA Criteria", Joint CNRA-CSNI Topical Discussion on High Burn-up Fuel Safety Issues, Paris 1 December 2004

Summary

30

- 1/2 -

RIA

- ✓ According to the previous plot, the Japanese criterion established 6 years ago and a USRNC/RES's proposed last year look conservative as a failure threshold. The Japanese criterion is just a failure threshold and applied with a coolability limit. On the other hand, the RES's threshold is used as a coolability limit. Overall, an approach proposed by RES sounds the most conservative.

- ✓ There is an appreciable database in currently licensed burnup level, but the database must be extended to higher burnup and new cladding materials. Better understandings on, in particular, post-failure events, a role of fission gas during the early stage of the transient, and an influence of rim formation, are needed. The FGD test will be one of the efforts to answer the pending questions.

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Summary

31

- 2/2 -

Fuel Safety Research at JAERI

- ✓ In the field of fuel safety, an extensive research is being performed at JAERI by using some unique facilities such as NSRR, large hot cells, etc. Research on high-burnup and MOX fuel behaviors under accident conditions has a particular importance, since it provides technical basis of future regulation.

- ✓ Mechanistic fuel behavior codes, such as FEMAXI and RANNS, are being developed and verified.

- ✓ The JAERI research program is going to be extended to higher burnup UO_2 and MOX fuels, newly developed pellets and claddings, and advanced reactor fuels.

- ✓ International and domestic collaboration becomes more important for the success of the program.

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Session 1-5

Fuel Safety Activities of LOCA and RIA in Korea

Geun-Sun Auh, Bong-Hyun Kim, Hyun-Koon Kim
KINS
Jun-Hwan Kim, Yong-Hwan Jeong, Sun-Ki Kim, Chan-Bock Lee
KAERI

Fuel Safety Research Meeting 2005

March 2 – 3, 2005 at Toshi Center Hotel, Tokyo, Japan

Abstract

Regulatory requirements for fuel performance in Korea include:

- ✓ To limit peak cladding temperature in LOCA to 1204C, total cladding oxidation to 17% of the total thickness before oxidation, and to require that the core remain in a coolable geometry.
- ✓ Reactivity insertion accidents do not result in damage to the reactor coolant pressure boundary nor impair core coolability.
- ✓ To limit for peak enthalpy 280 cal/g for a Control Rod Ejection Accident.

Korea's current regulatory limit on peak rod average burnup is 60 GWD/MTU.

The current regulatory requirements for fuel performance were based on earlier test data of f resh or low burnup Zircaloy fuels of less than 40 GWD/MTU. Most countries have not chan ged the current regulatory requirements even if they are actively investigating the high burn up and new cladding alloy effects. Korea agrees with commonly accepted international conse nsus that although there are technical issues requiring resolutions, these issues do not consti tute immediate safety concerns. The high burnup fuel reactor performance experiences of Ko rea do not show any major problems.

A research project of High Burnup Fuel Safety Tests and Evaluations has started in 2002 un der a joint cooperation of KAERI/KNFC/KEPRI and KINS to obtain performance results of hi gh burnup fuel and to develop evaluation technologies of high burnup fuel safety issues.

From 1998, KINS has been closely monitoring and actively participating in international acti vities to reflect on regulatory requirements if needed:

- ✓ OECD/NEA CSNI Special Expert Group on Fuel Safety Margin.
- ✓ OECD/NEA CABRI Water Loop Program.
- ✓ International Conferences such as USNRC's NSRC.

KINS will closely monitor the high burnup fuel performances of Korea to strength the regula tory activities if needed. KINS judges that the Lead Test Assemblies have to be actively test ed to measure their performances and several full core loadings of new design fuels are to be quantitatively monitored by the industry.

The following research activities of LOCA and RIA being performed at KAERI with active su pports of the industry are shown in this presentation:

- ✓ Fresh Fuel LOCA Research
  - ➢ High Temperature Ballooning Test
  - ➢ Thermal Quench Test
  - ➢ Ring Compression Test
  - ➢ High Temperature Oxidation Test
- ✓ Fresh Fuel RIA Research
  - ➢ Ring Tensile Test
  - ➢ RIA Simulation Test
- ✓ Irradiated Fuel Research
  - ➢ Ring Tensile Test under RIA Condition
  - ➢ Ring Tensile Test under LOCA Condition
- ✓ Preparation of Irradiated Fuel Tests_

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Contents of Presentation

- Current Regulatory Approach
- Technical Basis supporting Regulatory Approach
- Present Status
- Fresh Fuel LOCA Research
  - High Temperature Ballooning Test
  - Thermal Quench Test
  - Ring Compression Test
  - High Temperature Oxidation Test
- Fresh Fuel RIA Research
  - Ring Tensile Test
  - RIA Simulation Test
- Irradiated Fuel Research
  - Ring Tensile Test under RIA Condition
  - Ring Tensile Test under LOCA Condition
- Appendix: Preparation of Irradiated Fuel Tests

2

Current Regulatory Approach

- **Regulatory requirements for fuel performance in Korea:**
  - To limit peak cladding temperature in LOCA to 1204C, total cladding oxidation to 17% of the total thickness before oxidation, and to require that the core remain in a coolable geometry.
  - Specified acceptable fuel design limits are not to be exceeded during normal operation and AOO's.
  - Reactivity insertion accidents do not result in damage to the reactor coolant pressure boundary nor impair core coolability.
  - To assume fuel failure when DNB occurs during a CEA Ejection Accident.
  - To limit for peak enthalpy 280 cal/g for a Control Rod Ejection Accident.

- **Korea's current regulatory limit on peak rod average burnup is 60 GWD/MTU.**

3

Technical Basis supporting Regulatory Approach

- **The current regulatory requirements for fuel performance were based on earlier test data of fresh or low burnup Zircaloy fuels of less than 40 GWD/MTU.**
- **Most countries have not changed the current regulatory requirements even if they are actively investigating the high burnup and new cladding alloy effects.**
- **Korea agrees with commonly accepted international consensus that although there are technical issues requiring resolutions, these issues do not constitute immediate safety concerns.**
- **The high burnup fuel reactor performance experiences of Korea do not show any major problems.**

4

Present Status

- A research project of High Burnup Fuel Safety Tests and Evaluations has started in 2002 under a joint cooperation of KAERI/KNFC/KEPRI and KINS:
  - To obtain performance results of high burnup fuel.
  - To develop evaluation technologies of high burnup fuel safety issues.

- From 1998, KINS has been closely monitoring and actively participating in international activities to reflect on regulatory requirements if needed:
  - OECD/NEA CSNI Special Expert Group on Fuel Safety Margin.
  - OECD/NEA CABRI Water Loop Program.
  - International Conferences such as USNRC's NSRC.

5

Present Status

- KINS will closely monitor the high burnup fuel performances of Korea to strength the regulatory activities if needed.

- KINS judges that the Lead Test Assemblies have to be actively tested to measure their performances and several full core loadings of new design fuels are to be quantitatively monitored by the industry.

- Research Activities of LOCA and RIA in Korea are being performed at KAERI with active supports of the industry.

6

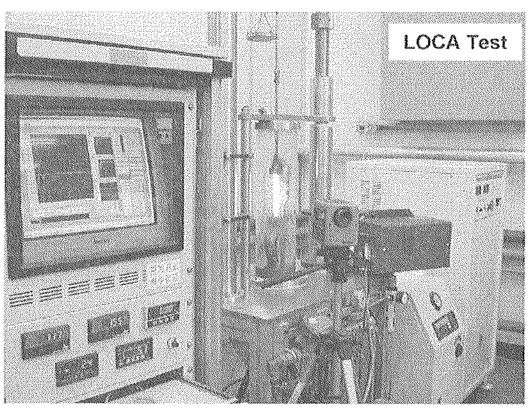
Fresh Fuel LOCA Research

- High Temperature Ballooning Test
  - Experimental variables
    - Hoop stress : 10~90MPa
    - Heating rate : 1~100℃/sec
  - Materials
    - Fresh cladding (Zry-4, Nb-contained cladding (A))
    - Pre-oxided cladding (Zry-4 : 20, 50㎛)
    - Pre-hydrided cladding (Zry-4 : 300, 600ppm of H)

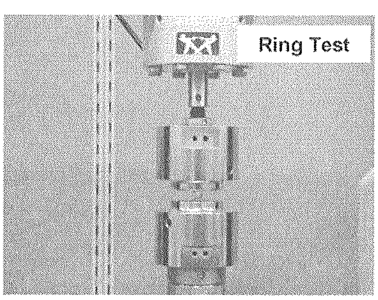
		2003	2004	2005	2006	2007~
A. LOCA : high temperature ballooning test						
A-1	Fresh cladding					
A-2	Effect of axial constraint					
A-3	Effect of pre-oxide					
A-4	Effect of pre-hydride					
A-5	Combined effect (oxide+hydride)					
A-6	Integral Test (Ballooning+Quenching)					

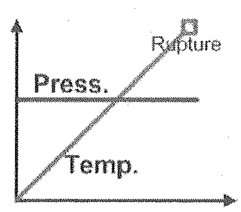
7

Fresh Fuel LOCA Research

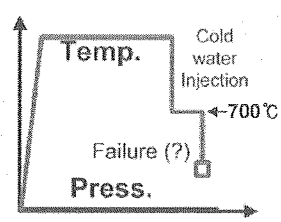


- Experimental Facilities
  - ✓ Temperature : ~1250℃ (by direct electrical heating)
  - ✓ Hoop stress : ~90MPa
  - ✓ Diametral differences can be measured by laser extensometer
  - ✓ Axial load can be controlled by moving crosshead
  - ✓ Two types of test
    - High temperature ballooning test
    - Quench test :





1) Ballooning test

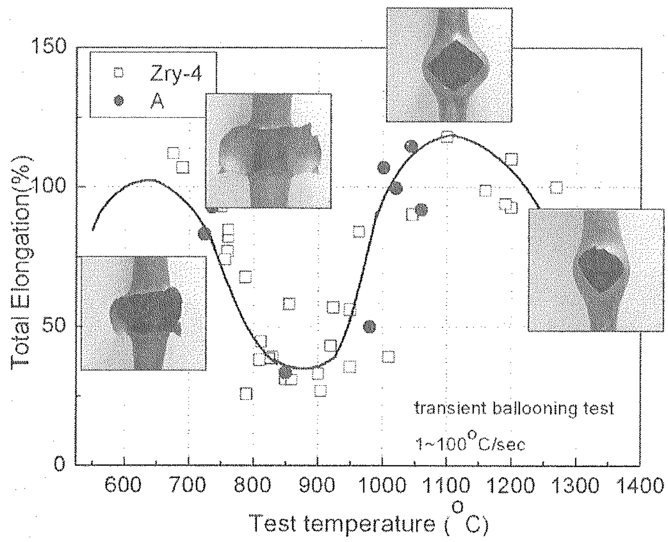


2) Quench test

8

Fresh Fuel LOCA Research

- **High Temperature Ballooning Test Results**



- **Initially pressurized cladding was heated up to rupture**
- **Circumferential burst elongation was measured with the burst temperature**
- **Elongation of the cladding deeply related to the phase transformation of the cladding tube.**

9

Fresh Fuel LOCA Research

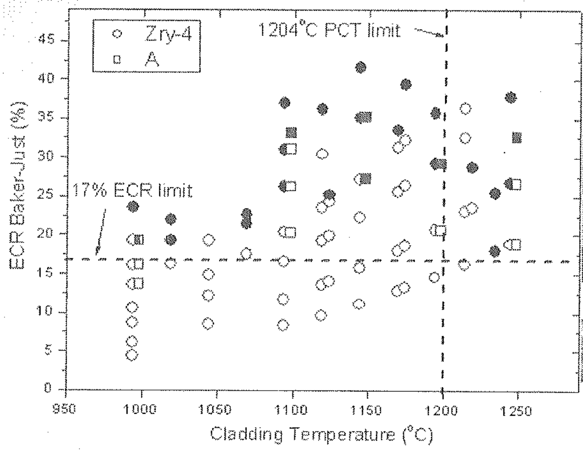
- **Thermal Quench Test**
  - **Experimental variables**
    - **Oxidation temperature : 1000~1250℃**
    - **Oxidation time : 500~10000sec**
    - **Ring compression test after thermal quench test**
  - **Materials**
    - **Fresh cladding (Zry-4, Nb-contained cladding (A))**
    - **Pre-oxided cladding (Zry-4 : 20, 50㎛)**
    - **Pre-hydrided cladding (Zry-4 : 300, 600ppm of H)**

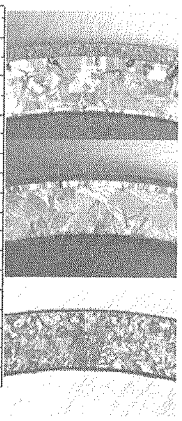
		2003	2004	2005	2006	2007~
B. LOCA : Quench test						
B-1	Fresh cladding					
B-2	Ring compression test					
B-3	Effect of axial constraint					
B-4	Effect of quenching rate					
B-5	Effect of pre-oxide					
B-6	Effect of pre-hydride					
B-7	Combined effect (oxide+hydride)					
B-8	3-point bend test					
B-9	Integral Test (Ballooning+Quenching)					

10

Fresh Fuel LOCA Research

- **Thermal Quench Test Results**



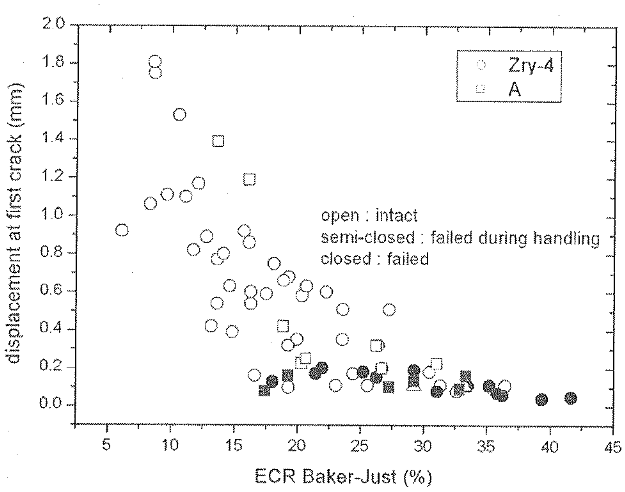


- Cladding was oxidized in a given temperature and time, followed by water quenching
- Intact, semi-intact (intact, but failed during handling), failed during water quenching was measured with respect to cladding temperature and ECR.

11

Fresh Fuel LOCA Research

- **Ring Compression Test**



- After quench test, cladding was cut into 15mm at the middle section, compressed at room temperature until fracture.
- Specimen ductility gradually decreased with the ECR.
- Ring compression test at 135℃ and three point bend test are planned

12

Fresh Fuel LOCA Research

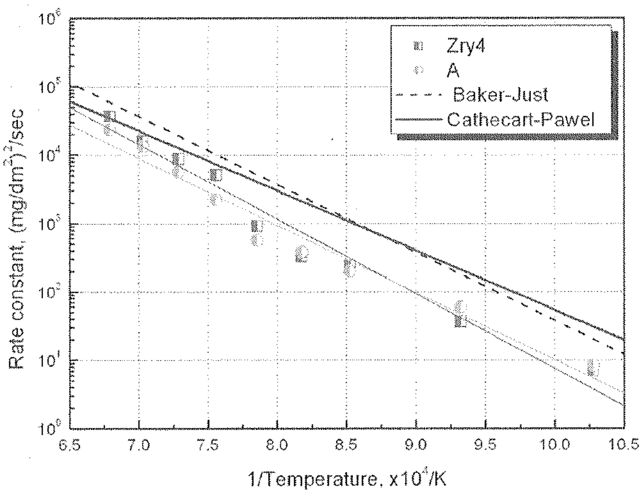
- High Temperature Oxidation Test
  - Experimental variables
    - Oxidation temperature : 900~1200℃
  - Materials
    - Fresh cladding (Zry-4, Nb-contained cladding (A))
    - Pre-oxided cladding (Zry-4, A)
    - Pre-hydrided cladding (Zry-4, A : 300, 600ppm of H)

		2003	2004	2005	2006	2007~
C. LOCA : High temperature oxidation						
C-1	Fresh cladding					
C-2	Alloying element (Sn, Nb)					
C-3	Effect of pre-oxide					
C-4	Effect of pre-hydride					
C-5	Combined effect (oxide+hydride)					

13

Fresh Fuel LOCA Research

- High Temperature Oxidation Test Results



- Cladding was oxidized in a given temperature and time. Changes of weight was measured by TGA (Thermo-Gravimetric Analyzer).
- Rate constant and oxidation equation were being made.

14

Fresh Fuel RIA Research

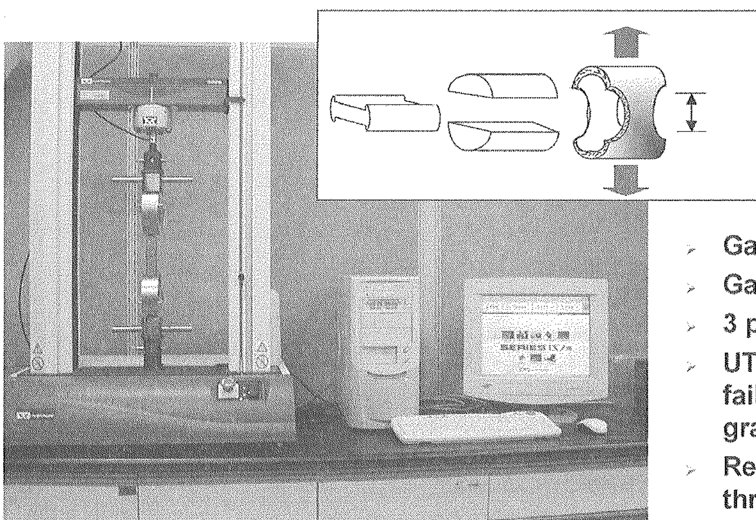
- **Ring Tensile Test**
  - **Experimental variables**
    - **Strain rate : 0.01, 0.1, 1/sec**
    - **Environment : RT, 350℃**
  - **Materials**
    - **Fresh cladding (Zry-4, Nb-contained cladding (A))**
    - **Pre-oxided cladding (Zry-4, A : 20, 50, 100㎛)**
    - **Pre-hydrided cladding (Zry-4, A : 300, 600, 1000ppm of H)**

		2003	2004	2005	2006	2007~
D. RIA : ring tension test						
D-1	Effect of strain rate					
D-2	Effect of pre-oxide					
D-3	Effect of pre-hydride					
D-4	Combined effect (oxide+hydride)					
D-5	Effect of test temperature					

15

Fresh Fuel RIA Research

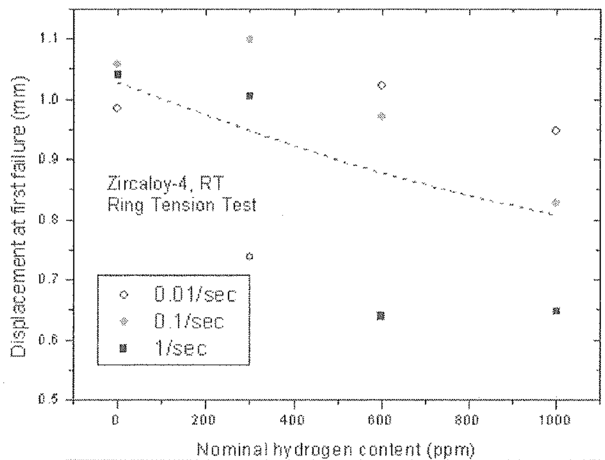
- Ring Tensile Test

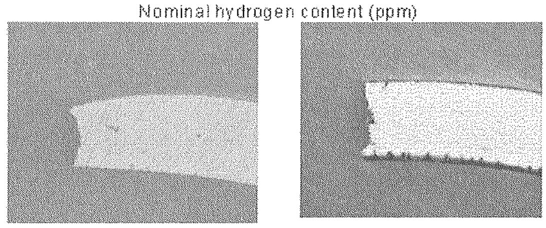


- **Gauge length: 2.11mm**
- **Gauge width: 1.7mm**
- **3 piece type**
- **UTS and displacement at the failure in load-displacement graph were measured**
- **Results were presented through averaging 3 identical tests**

16

Fresh Fuel RIA Research





- Ring Tensile Test Results
  - **As the hydrogen content increased, strength increased but ductility decreased, showing brittle fracture.**
  - **Brittle fracture was due to the hydrogen embrittlement of cladding, as well as the circumferential crack around hydride.**
  - **Ductility of cladding highly depends on the hydrogen contents rather than strain rate.**

<= (0 ppm, 1000 ppm)

17

Fresh Fuel RIA Research

- **RIA Simulation Test (TBD)**
  - **Candidate tests (TBD)**
    - **High speed burst test**
    - **EDC (Expansion Due to Compression) test**
    - **Temperature transient test**
  - **Materials**
    - **Fresh cladding (Zry-4, A)**
    - **Pre-oxided, pre-hydrided cladding**

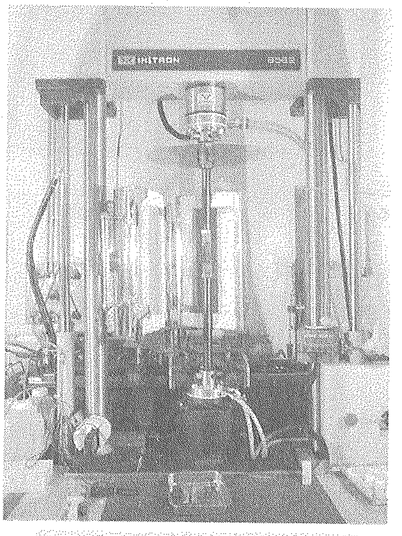
		2003	2004	2005	2006	2007~
E. RIA : simulated test (TBD)						
E-0	Experimental setup					
E-1	Effect of pre-oxide and pre-hydride					
E-2	Effect of axial stress state					

18

Irradiated Fuel Research

- Ring Tensile Test under RIA Condition
  - Temperature range = RT, 200, 400, 600, 800°C
  - Initial strain rate = 1/s
  - Oxide thickness = up to 100 μm
  - Burnup = 40,000~60,000 MWd/tU
  - Cladding Material: Zry-4, Nb-contained cladding (A)

- Axial Tensile Test (planned)
  - RT, 350, 600°C



Ring Tensile Test Machine

19

Irradiated Fuel Research

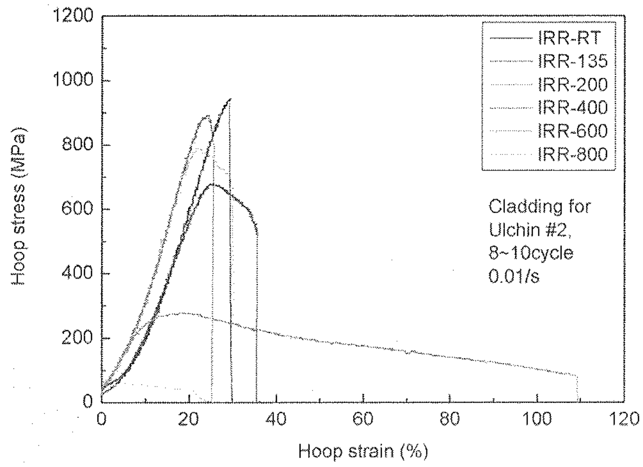
- Ring Tensile Test under LOCA Condition
  - Temperature range = RT, 135, 200, 400, 600, 800°C
  - Initial strain rate = 0.001/s ~ 0.01/s
  - Oxide thickness = up to 100 μm
  - Burnup = 40,000 ~ 60,000 MWd/tU
  - Cladding Material : Zry-4, Nb-contained cladding (A)

- Tube ballooning & Rupture Test (planned)
  - RT, 350°C

20

Irradiated Fuel Research

❑ Ring Tensile Test Results
(Strain Rate = 0.01/s)

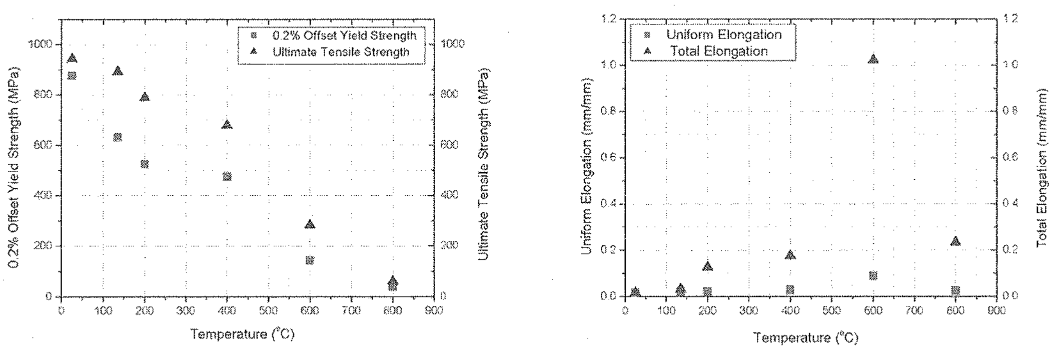


Hoop Stress-Strain Curve

21

Irradiated Fuel Research

❑ Ring Tensile Test Results



Mechanical Properties in the Temperature Range up to 800C

❑ **0.2% Offset YS and UTS decrease with temperature.**

❑ **UE and TE increase with temperature to 600C, but have a tendency to decrease beyond it.**

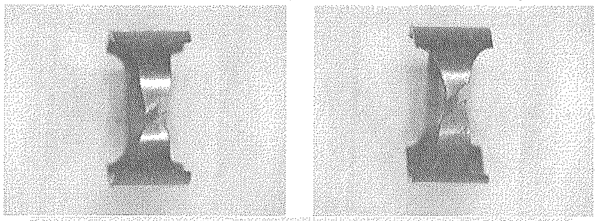
$\Rightarrow$ **Due to phase transformation of Zircaloy-4 material.**

22

Irradiated Fuel Research

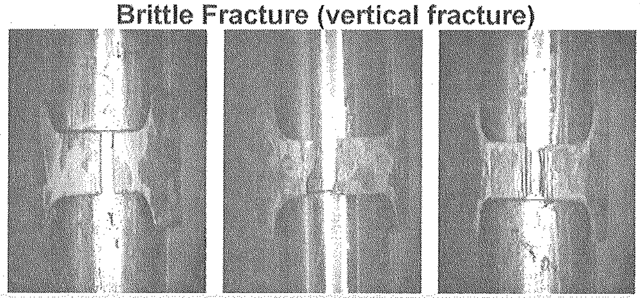
❑ Fracture Pattern After Ring Tensile Tests

Ductile Fracture (45° shear fracture)



Fracture Pattern of unirradiated specimen (RT)

Brittle Fracture (vertical fracture)



Fracture Pattern of irradiated specimen (RT, 400, 600C)

23

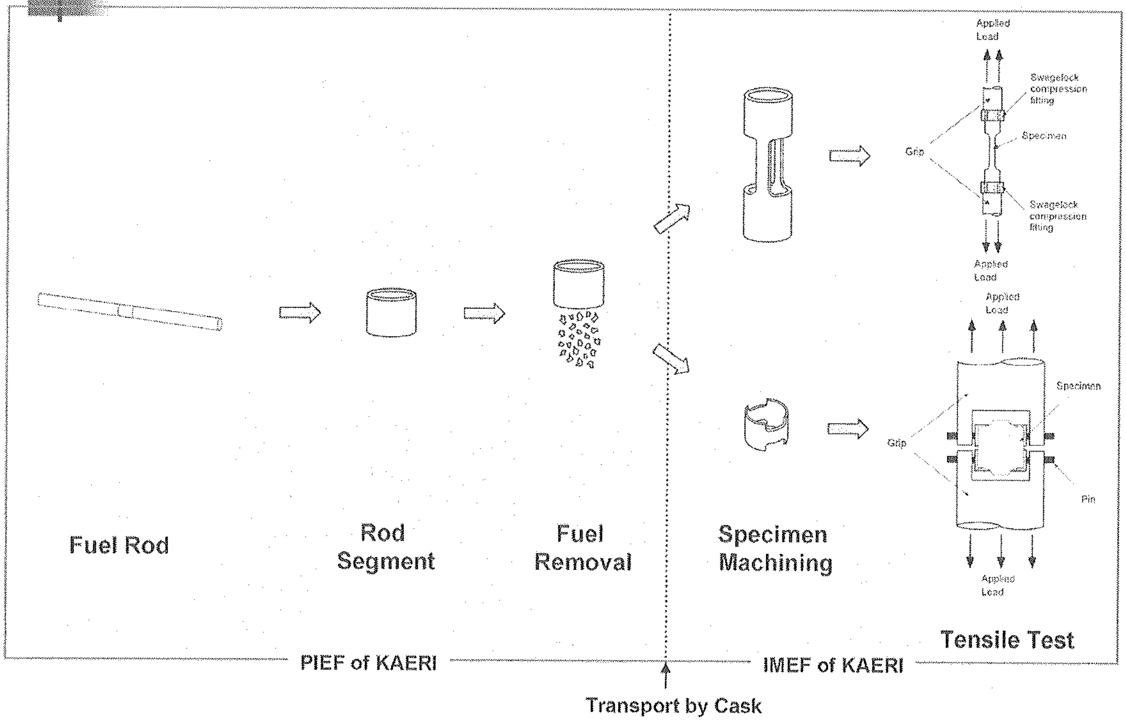
Irradiated Fuel Research

❑ Irradiated Fuel Test Schedule

	Mechanical Test	2004	2005	2006	2007~
1	Ring Tensile Test				
2	Axial Tensile Test				
3	Ballooning & Rupture Test				

24

Appendix: Preparation of Irradiated Fuel Tests



25

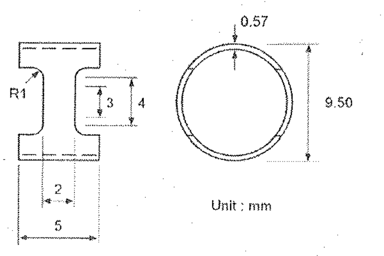
Appendix: Preparation of Irradiated Fuel Tests

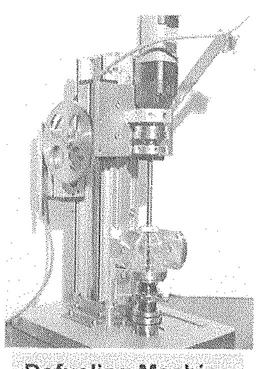
1. Removal of fuel in hot cell

- Mechanical drilling or grinding of the fuel is used to remove the fuel

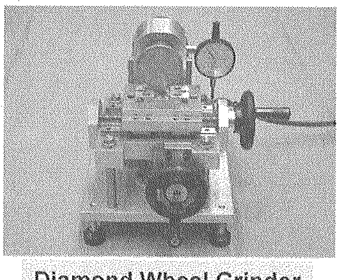
2. Specimen fabrication in glove box

- Diamond wheel grinder is used
- Precision : ± 0.05 mm

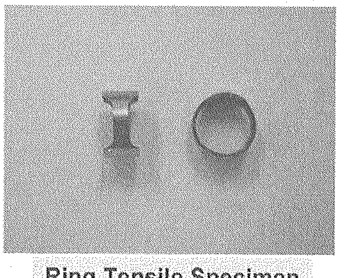




Defueling Machine



Diamond Wheel Grinder



Ring Tensile Specimen

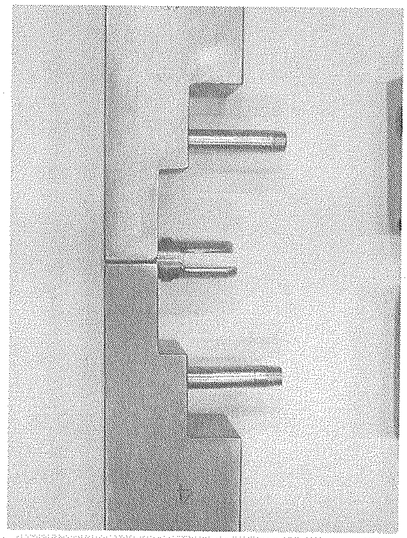
26

Appendix: Preparation of Irradiated Fuel Tests

3. Grip & Jig for Ring Tensile Test

❑ Specimen design:
- Hoop tensile specimen - Ring specimen (5 mm)
- Gauge length = 3 mm, gauge width = 2 mm

❑ Grip & Jig design:
- Hoop loading grip - Grip with 2-pieces of half-cylinder
  ⇒ The gauge section is toward up & down direction
- The diameter of half-cylinder = 8.08 mm

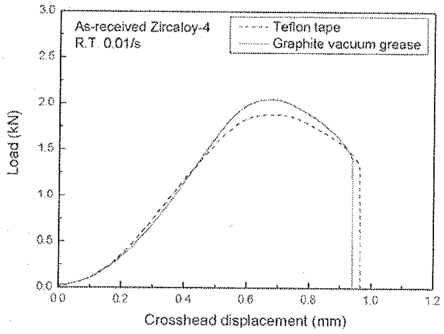


Grip & Jig for Ring Tensile Test

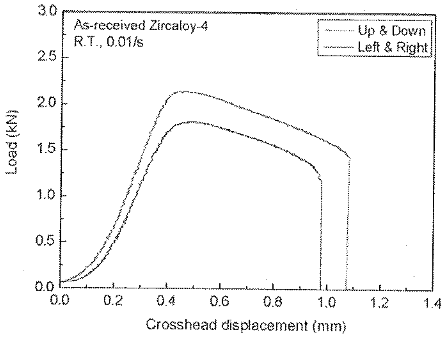
27

Appendix: Preparation of Irradiated Fuel Tests

4. Results of Preliminary Tests (Fresh Cladding)



Friction Effect



Gage Section Position Effect

❑ Graphite Vacuum Grease is used as lubricant because of its high allowable temperature of over 1,000C.

❑ Gauge Section is toward up & down in this study.

28

2.Session 2 Fuel behavior under RIA condition

Session 2-1

STUDIES ON HIGH BURNUP LWR FUEL BEHAVIOR UNDER RIA CONDITIONS

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JAERI's research activities on high burnup fuel behavior under reactivity initiated accident (RIA) conditions are presented.

Two pulse irradiation experiments simulating RIAs were performed at the NSRR on 60 GWd/t PWR UO_2 fuel rods. Test OI-10, which was performed with a peak fuel enthalpy condition of 435 J/g on large-grain (~28 μm) fuel pellets sheathed with an MDA cladding, showed that the large grain pellets reduced the amount of fission gas release (FGR) during a power transient by decreasing the amount of accumulated fission gas at grain boundaries. Cladding deformation was also lowered because of small increase of rod internal pressure due to the low FGR. Test OI-11 on conventional grain size (~10 μm) pellets sheathed with a ZIRLO cladding resulted in a fuel failure at a fuel enthalpy of 500 J/g due to the pellet-cladding mechanical interaction (PCMI). This result shows that the fuel claddings with improved corrosion resistance have a larger safety margin against PCMI failure than the conventional Zircaloy-2/4 claddings.

Improvement of cladding corrosion resistance, however, does not make advantage in safety evaluation with the current PCMI failure criterion in Japan, because the criterion is a function of the fuel burnup only and the cladding improvement is not considered. To establish better criterion reflecting the cladding performance, indices for fuel conditions and failure limit should be appropriately chosen on the basis of microscopic investigation of fuel state at failure. JAERI is going to proceed with the investigation using the NSRR experiments, post-irradiation examinations on the cladding samples, cladding mechanical property tests and a newly developed computer code which can analyze cladding local stress and strain during a power transient. Regarding the post-failure events, knowledge of mechanical energy generation during the PCMI failure has been accumulated. Reliable evaluation of the mechanical energy is prospected on the basis of heat transfer from fuel fragments to coolant.

Studies on high burnup LWR fuel behavior under RIA conditions

Tomoyuki SUGIYAMA

Fuel Safety Research Laboratory
Japan Atomic Energy Research Institute

Fuel Safety Research Meeting, Tokyo, March 2-3, 2005

Contents

- Objective
- Results from recent NSRR experiments
- Estimation of post-failure events
- PCMI failure criterion
- NSRR test schedule

Objective

Acquisition of knowledge to be used for the safety regulation covering advanced use of LWR fuels including:
- Further extension of fuel burnup
- Extensive use of MOX fuel in LWRs

Tasks
- Data accumulation of high burnup fuels
- Development of safety evaluation techniques for high burnup fuel behavior under accident conditions

Research items

NSRR experiment

- RIA-simulating experiments on LWR fuels
  Tests with high burnup fuels used at power plants
  Separate-effect tests with fresh fuels ➡(Tomiyasu)

Out-of-pile experiments

- Cladding mechanical property tests
  Ring tensile test, fracture toughness measurement, EDC test*, etc. ➡(Ikatsu)

* Test using Expansion Due to Compression, originated by Studsvik.

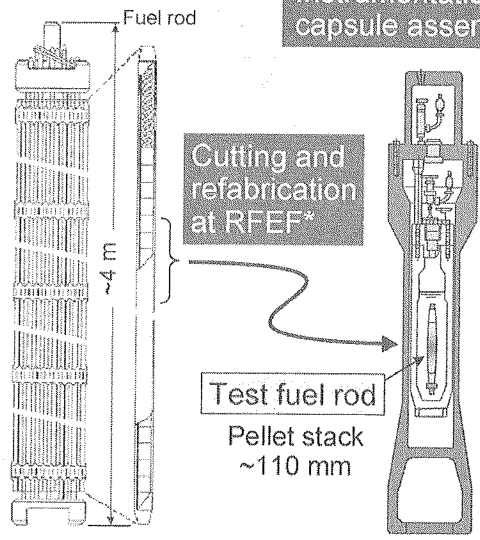
Code development

- RANNS code for fuel behavior analysis at RIA ➡(Suzuki)

NSRR experiment

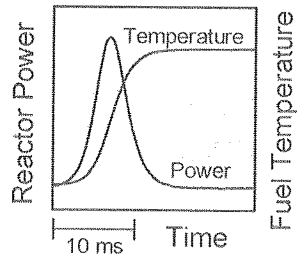
Instrumentation and capsule assembling

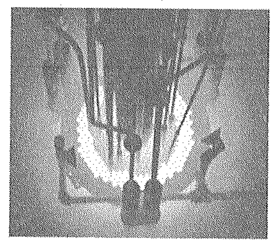
Pulse irradiation at NSRR



Fuel rod used in a power reactor

Test capsule





Transient measurement

- Temp. at clad surface and coolant
- Rod pressure, capsule pressure
- Clad surface strain (hoop, axial)
- Elongations of clad and pellet stack
- Water column velocity
- etc.

Detailed post irradiation examinations at RFEF

* RFEF: Reactor Fuel Examination Facility

Tests OI-10, -11 and -12 on high burnup PWR fuel

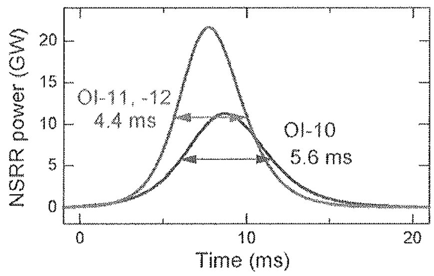
- PWR 55 GWd/t lead-use fuel (Ohi unit 4, KEPCO)
- OI-10 and -11 performed in July, 2003
- OI-12 to be performed in March, 2005

Test ID	OI-10	OI-11	OI-12
Fuel type		PWR 17x17	
Initial U-235 enrichment		4.5%	
Pellet density		95%TD	
Pellet grain size (μm)	~28	~10	7 - 8
Cladding material	MDA	ZIRLO	NDA
Cladding thickness (mm)	0.57	0.57	0.64
Operation period		4 cycles from Mar.97 to Mar.02	
Test rod sampling position		2nd span from the top	
Cladding oxide thickness (μm)	~27	~28	-
Test rod burnup (MWd/kgU)	60	58	61
Fission gas release in the PWR	0.57%	0.49%	-

MDA: Zr-0.8Sn-0.2Fe-0.1Cr-0.5Nb
ZIRLO: Zr-1.0Sn-0.1Fe-1.0Nb
NDA: Zr-1.0Sn-0.27Fe-0.16Cr-0.1Nb-0.01Ni

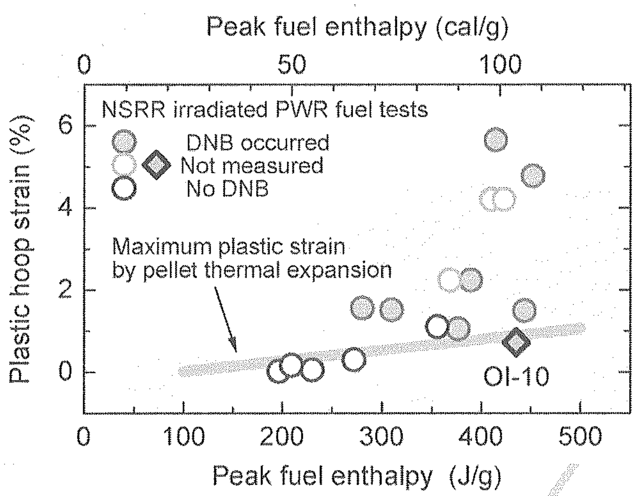
Tests OI-10, -11 and -12

	OI-10	OI-11	OI-12
Coolant conditions	Stagnant, ~20 deg C, 0.1 MPa		
Pulse irradiation			
Inserted reactivity($)	3.67	4.6	4.6
Peak fuel enthalpy			(preliminary)
(J/g)	435	657	~590
(cal/g)	104	157	~140
Main results	No failure No significant deformation	Failed at 120 cal/g Cladding axial crack Fuel fragmentation	-



Comparison of power histories.

Cladding deformation in test OI-10



DNB: departure from nucleate boiling at clad surface

Small deformation is produced by pellet thermal expansion.

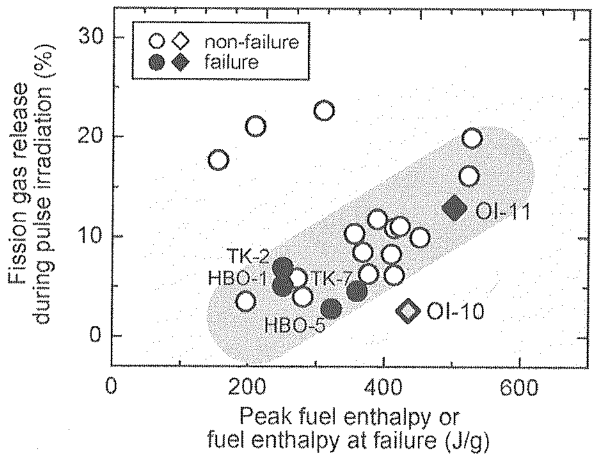
Conditions for large deformation:

- Cladding yield stress goes down with cladding temperature increase.
- Rod internal pressure becomes high due to fission gas release.

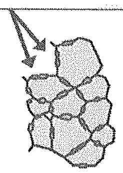
Considering the high peak fuel enthalpy in test OI-10, DNB probably occurred. *i.e.* Cladding temp. became high.

Small deformation in OI-10 suggests the rod internal pressure remained low. *i.e.* FGR was low.

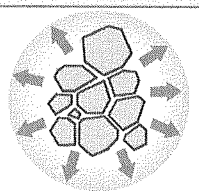
Fission gas release in test OI-10



Fission gas accumulated at grain boundaries



Burst release from grain boundary at power transient



Remarkably low FGR in test OI-10

Large grain pellet was developed to reduce fission gas release from grains during normal operations.

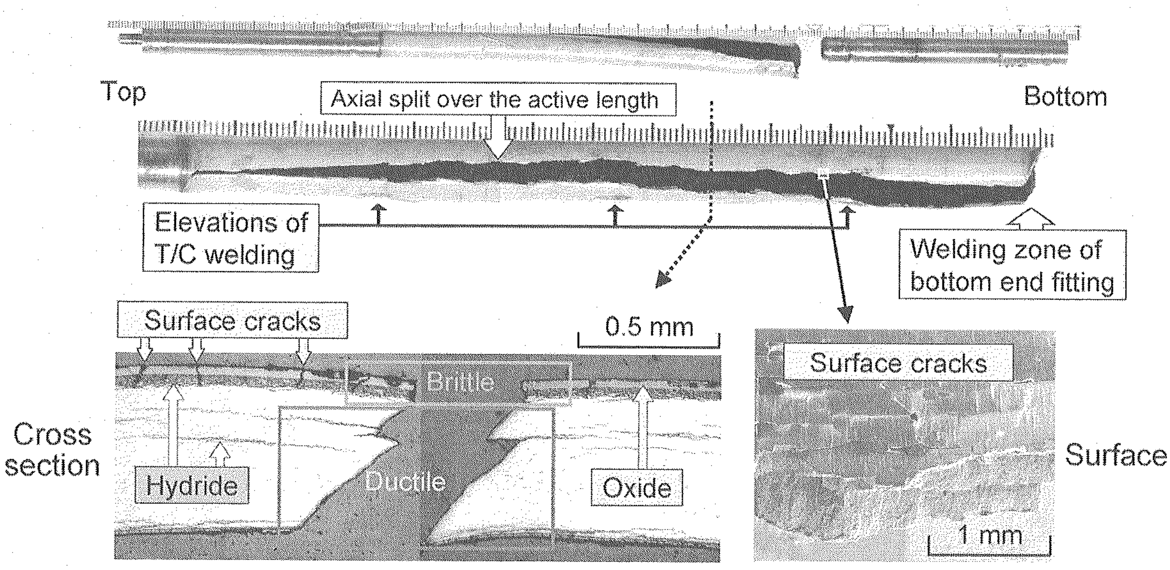
Reduction of the amount of gas accumulated at grain boundaries.

Reduction of fission gas release from grain boundary at power transient.

Large grain pellet is effective to reduce FGR at an RIA.

JAERI 9

Failed rod in test OI-11

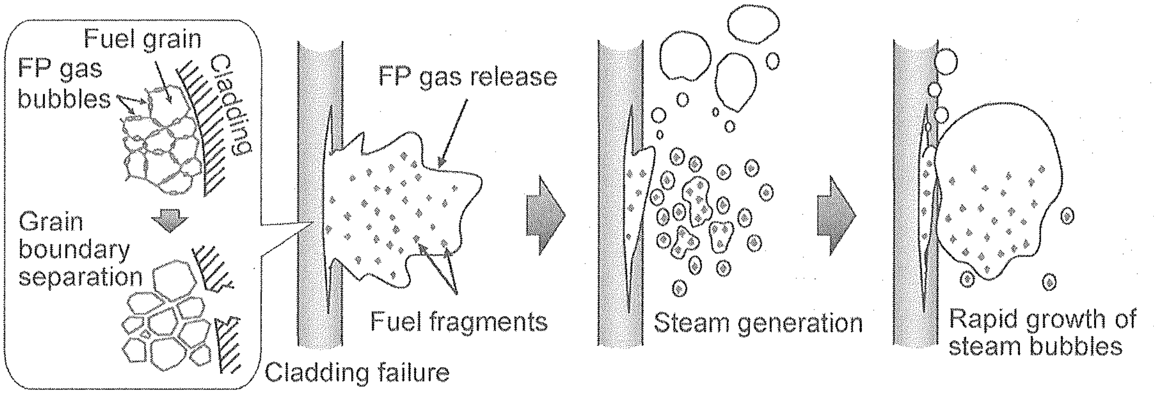


All fuel pellets were fragmented and dispersed into the coolant.

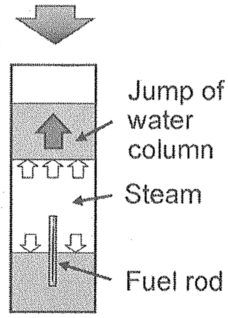
⇨ Mechanical energy

JAERI 10

Mechanical energy generation during PCMI failure

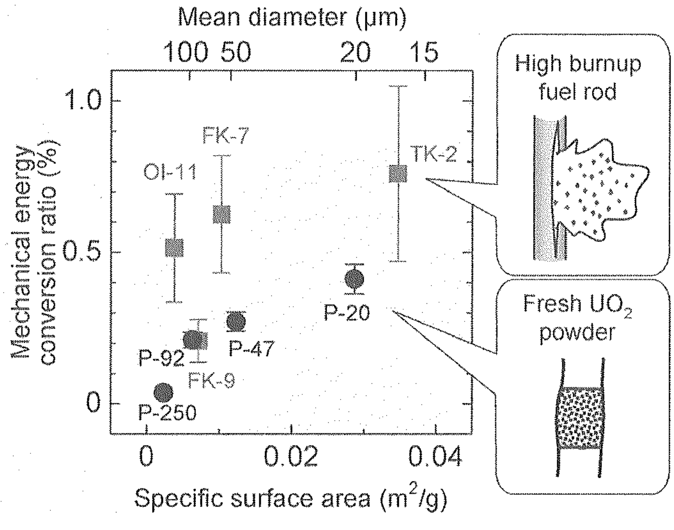


Mechanical energy is evaluated as a kinetic energy of jumping water column.

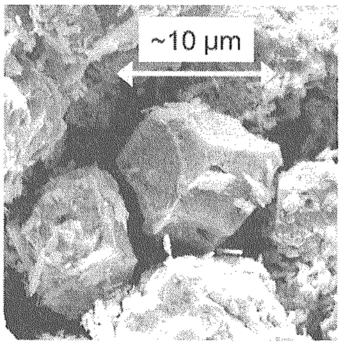


JAERI 11

Estimation of mechanical energy generated



Mean diameter: 152 μm



Smallest particles from OI-11 test rod

- Mechanical energy can be estimated by simple calculations based on heat transfer from fuel fragments to coolant.
- Differences in coolant conditions between NSRR and power reactors can be considered in the calculation.

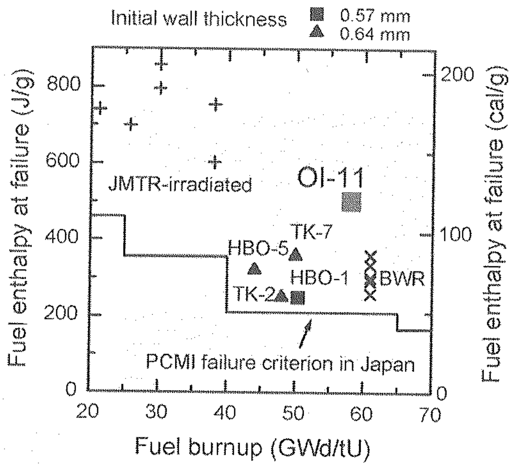
JAERI 12

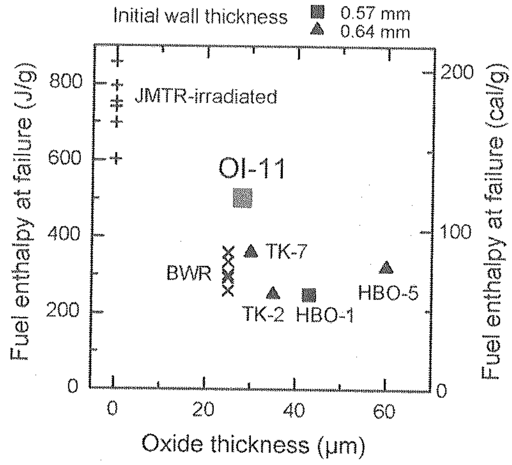
PCMI failure criterion

- OI-11 had less corrosion than conventional Zry-2/4 rods at a same burnup level.
- PCMI failure criterion in Japan uses only fuel burnup as an index for fuel condition.

⇨ Improvement of cladding corrosion-resistance does not make advantage in safety evaluation with the current criterion.

Attempts to describe the fuel condition with other parameters, such as cladding oxide thickness, are being made.





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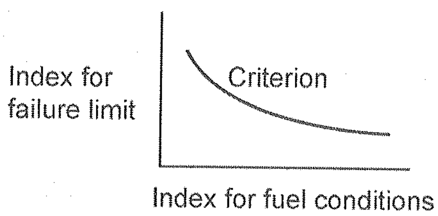
13

PCMI failure criterion (continued)

Problems in the use of oxide thickness:

- Oxide thickness must be estimated accurately (or conservatively) from fuel burnup and other irradiation conditions for each type of cladding.
- To confirm the estimated oxide thickness, cladding samples irradiated up to a target burnup under real conditions are needed.
- This approach is equivalent to defining different criteria for different claddings.



Alternative indices should be appropriately chosen for both fuel conditions and failure limit.

⇩

Better understanding of "failure" is necessary.

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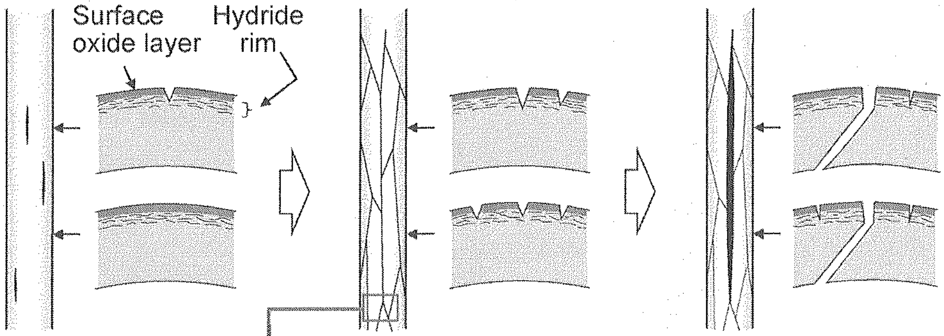
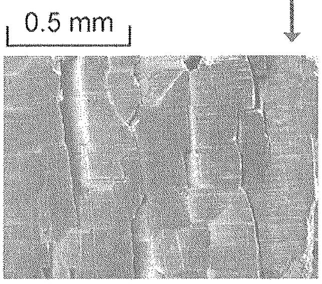
14

Anticipated process of PCMI failure



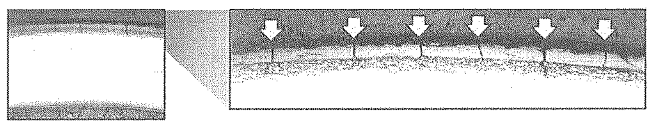
Initiation of surface cracks　　Surface crack propagation　　Penetration



OI-11 cladding surface

Radial cracks in the oxide and hydride layers were observed at the cladding surface of both OI-10 and -11 rods. In test OI-11, cladding penetration should have started at one of such surface cracks.

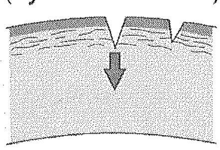
OI-10



Evaluation of microscopic fuel state at failure

"Failure" is defined as the occurrence of cladding penetration.

Outer surface: ~30 deg C
(by measurement)



Inner surface: >500 deg C
(by computation)

Cladding temperature
at PCMI failure in OI-11

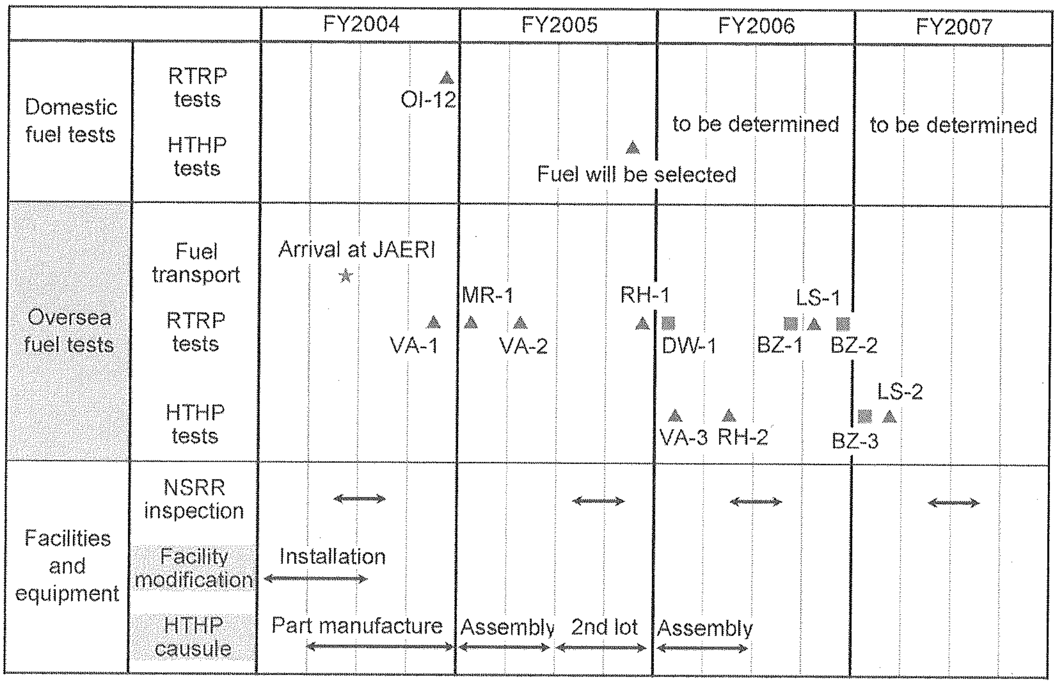
Evaluation of all aspects of fuel state at failure

- Fuel enthalpy at PCMI failure — Evaluated by NSRR tests
- Hoop/axial stress due to PCMI
- Stress concentration at the crack tip
- Thermal stress due to radial distribution of cladding temperature
- Radial distribution of mechanical properties (Young's modulus, etc.) due to temperature profile

  — by computer code

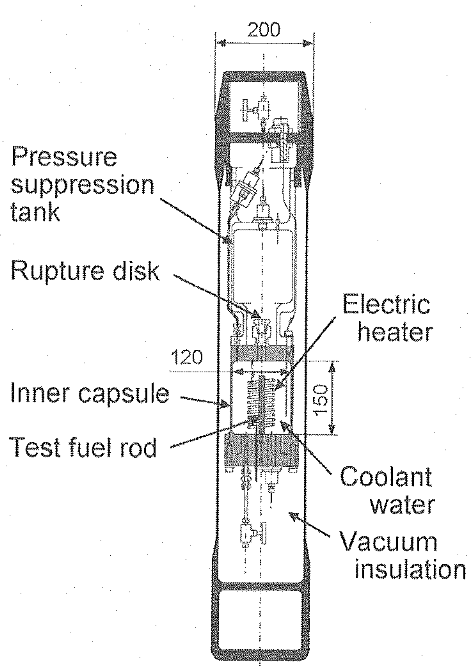
- Radial distribution of hydride precipitation — by clad PIEs
- Local fracture toughness as a function of temperature and hydrogen concentration — by mechanical property tests

NSRR test schedule until FY2007



RTRP : room temperature and room pressure
HTHP : high temperature and high pressure
▲ : UO_2 , ■ : MOX
[shaded box] : funded by METI / NISA (Nuclear and industrial safety agency)

High-temperature high-pressure test capsule



Test fuel rod
- Total length ~120 mm
- Active length ~50 mm

Coolant water
- Stagnant
- 286 deg C, 7 MPa (BWR conditions)

Main purpose is to clarify the influence of cladding ductility recovery at high temp. on the PCMI failure limit.

Instrumentation
- Cladding surface thermocouple
- Coolant thermocouple
- Capsule pressure sensors

Fuel rods to be tested until FY2007

Type: Fuel	Type: Reactor	Type: Spec	Reactor (Country)	Burnup GWd/t	Cladding	NSRR test ID: RTRP	NSRR test ID: HTHP
UO_2	PWR	17x17	Oi-4 (Japan)	61	NDA	OI-12	-
			Vandellos (Spain)	78	MDA	VA-1	VA-3 (MDA or ZIRLO)
				79	ZIRLO	VA-2	
			McGuire(USA) R2(Sweden)	69	NDA	MR-1	-
			Ringhals (Sweden)	67	M5	RH-1	RH-2
	BWR	10x10	Leibstadt (Switzerland)	73	Zry-2	LS-1	LS-2
MOX	PWR	14x14	Beznau (Switzerland)	59	Zry-4	BZ-2	BZ-3
				48	Zry-4	BZ-1	-
	BWR	8x8	Dodewaard (Netherland)	45	Zry-2	DW-1	-
					Total	9	4

JAERI 19

Summary

Tests OI-10 and -11 on 60GWd/t PWR fuel rods respectively showed:

- Large grain pellet reduces fission gas release during a power transient, and consequently lowers the cladding deformation driven by the rod internal pressure.
- Fuel claddings with improved corrosion resistance have a larger safety margin against PCMI failure than the conventional Zircaloy-2/4 claddings.

Improvement of cladding corrosion-resistance does not make advantage in safety evaluation with the current PCMI failure criterion. To establish better criterion reflecting the cladding performance, indices for fuel conditions and failure limit should be appropriately chosen on the basis of microscopic investigation of fuel state at failure.

Knowledge of mechanical energy generation during the PCMI failure has been accumulated. Reliable evaluation of the mechanical energy is prospected on the basis of heat transfer from fuel fragments to coolant.

JAERI 20

Session 2-2

FISSION GAS RELEASE FROM HIGH BURNUP PWR FUEL DURING AN RIA

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The expansion and release of fission gas have important role on the fuel behavior under RIA conditions. We have analyzed fission gas released from the high burnup fuel during pulse-irradiation experiments in the nuclear safety research reactor (NSRR) in order to identify radial position in the pellet which mainly contributes to the release.

Relatively larger number of Pu fission gives a higher ratio of xenon to krypton in fuel pellet peripheral region with a higher local burnup. Therefore, data of the Xe/Kr ratio obtained from puncturing of post-test rod give information regarding whether the gas is released from pellet center, periphery, or entire region.

The ratio of xenon and krypton (Xe/Kr ratio) depends also on pellet specification and base-irradiation conditions. The Xe/Kr ratio was also calculated with RODBURN code in each fuel and burnup, and compared with the data. In order to estimate the radial position of fission gas release in the pellet, a normalized Xe/Kr ratio, R, is introduced. A higher value of R corresponds to significant release from peripheral region of pellet.

Test fuel rods were obtained from 14x14 and 17x17 PWR fuel rods base-irradiated in the reactors up to burnup of approximately 50 GWd/t. Fission gas release (FGR) during the base-irradiation is evaluated to be less than 1% for each type fuel rod. Therefore, most of fission gas generated during the base-irradiation can be estimated to remain in the fuel pellet. Pulse irradiation tests with burnup fuels were conducted at peak fuel enthalpy ranging from 155 J/g to 530 J/g in the NSRR.

The present evaluation revealed that radial position of fission gas release in the pellet is highly dependent on peak fuel enthalpy during the pulse-irradiation. In the tests with low enthalpies, contribution of fission gas in grain boundaries of peripheral region is not dominant. A higher value of R is obtained for the middle enthalpy condition; hence a contribution of fission gas in grain boundaries of peripheral region becomes significant. Then, in the tests with the high enthalpies, fission gas is released due to grain boundary separation in entire region of pellet.

Fission Gas Release from High Burnup PWR Fuel during an RIA

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Fuel Safety Research Laboratory
Department of Reactor Safety Research
JAERI

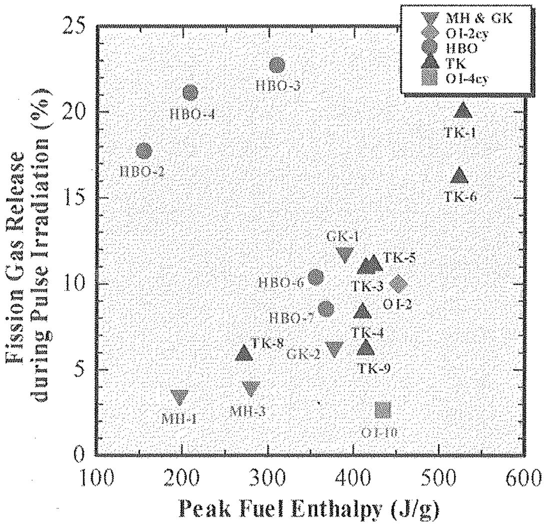
Prepared for Fuel Safety Research Meeting 2005
on March 2-3, 2005 at Tokyo, Japan

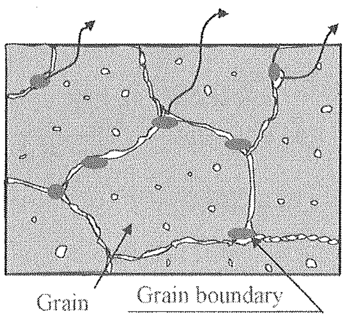
Outline

The expansion and release of fission gas play an important role on the fuel behavior under RIA conditions. We have analyzed fission gas released from the high burnup fuel during pulse-irradiation experiments in the NSRR in order to identify radial position in the pellet which mainly contributes to the release.

2

Fission gas release during pulse-irradiations

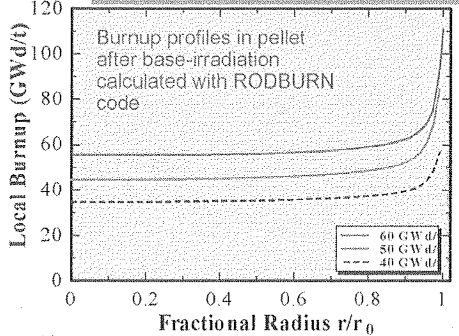


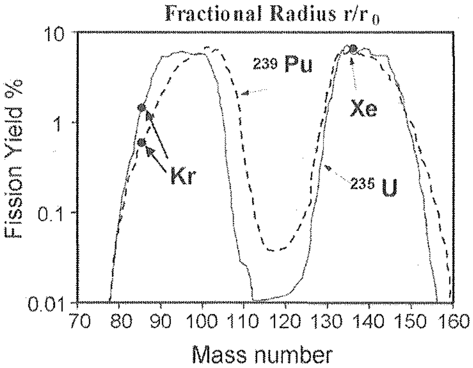


- The higher peak fuel enthalpy brought the larger amount of fission gas release.
- During the pulse-irradiation, fission gas accumulated in the grain boundaries would be released.
- In the tests HBO-2, -3 and -4, 18 to 23% of FGR was measured in spite of lower enthalpy.

3

Evaluation method





- The fission yields of Kr is smaller for ^{239}Pu than that for ^{235}U.
- Xe/Kr ratio becomes higher as the burnup increases and plutonium builds up

- **FP gas compositions**

  : Xe/Kr ratio

  Measured by rod puncture and gas analysis
- **Change of Xe/Kr ratio**

  with burnup in radial direction of the pellet

  : Calculated with burning analysis code, RODBURN

Comparison

Radial distribution in the pellet which mainly contributes to fission gas release will be known

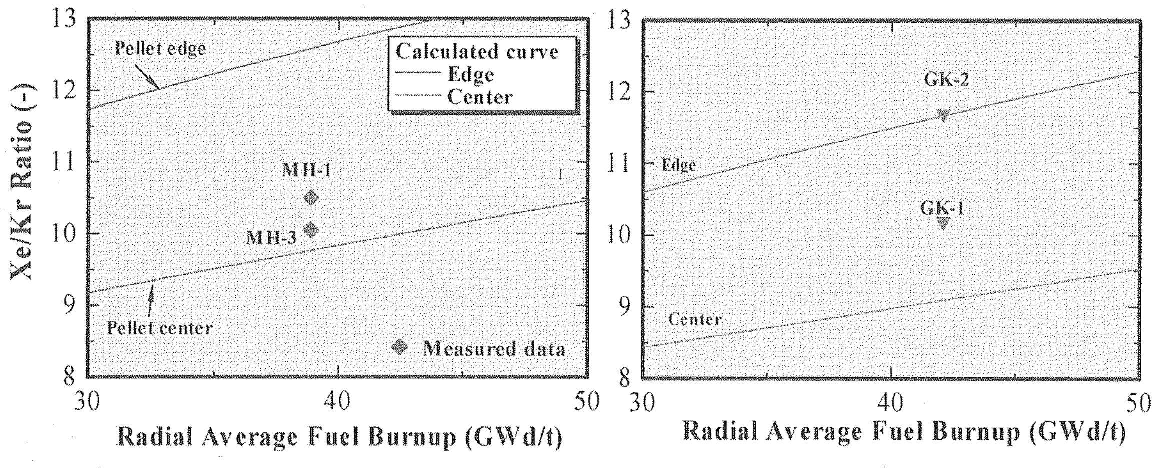
4

Test fuels

Fuel ID	Fuel type	Reactor	Fuel Burnup, GWd/t	FGR during base-irradiation, %
MH	14x14	Mihama-2	38.9	‹ 0.2
GK	14x14	Genkai-1	42.1	‹ 0.4
HBO	17x17	Ohi-1	44 - 50.4	‹ 0.8
TK	17x17	Takahama-3	37.8 - 50	‹ 0.4

5

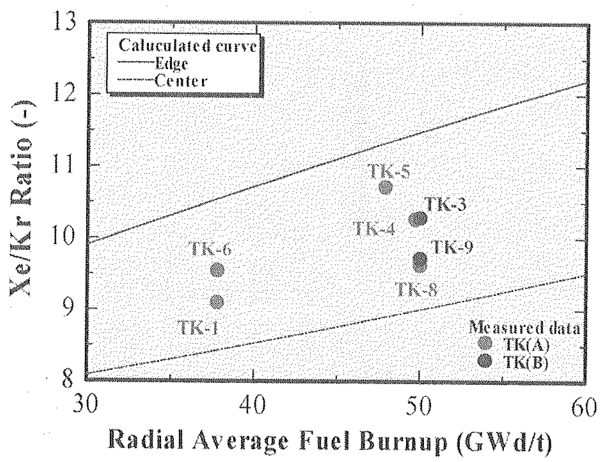
Measured Xe/Kr ratio compared to calculated results

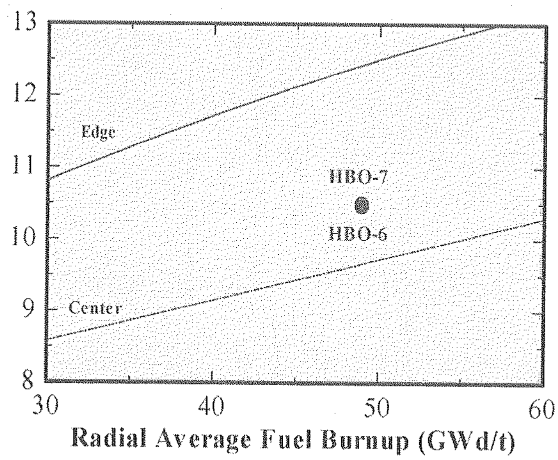


- The calculated curves depend on initial enrichment, outer diameter, and base-irradiation conditions such as liner heat generation late.
- Each fuel has different calculated curve.

6

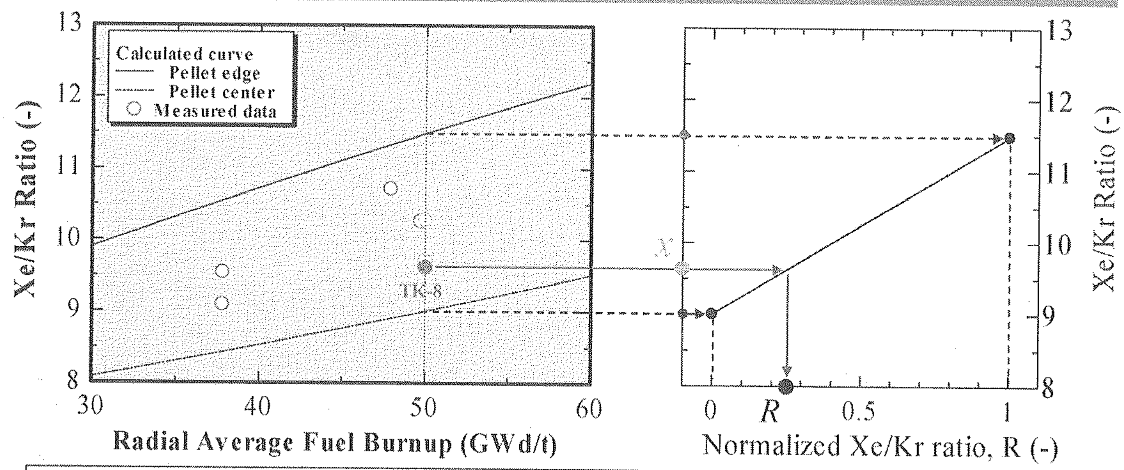
Measured Xe/Kr ratio compared to calculated results





7

Normalization of Xe/Kr ratio



- Define the calculated Xe/Kr ratio in pellet center and edge as 0 and 1 in the R value.
- Apply the linear relationship between the calculated Xe/Kr ratio and R values
- Convert the measured Xe/Kr ratio to the R value.

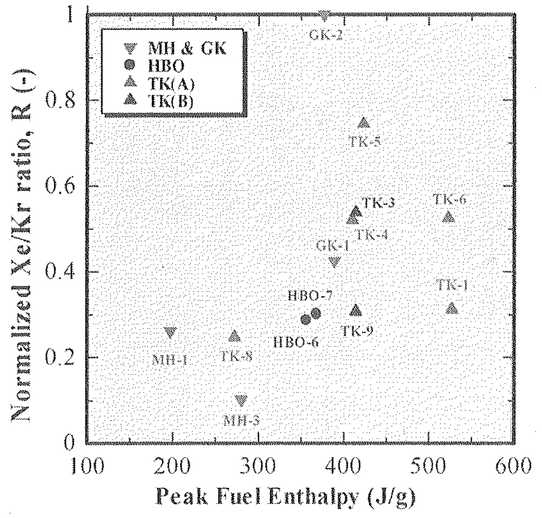
$$R = \frac{x - \mathrm{Xe/Kr}_{center}}{\mathrm{Xe/Kr}_{edge} - \mathrm{Xe/Kr}_{center}}$$

e.g. R = 0.25 in Test TK-8

A higher value of R corresponds to significant release from outer region of pellet.

8

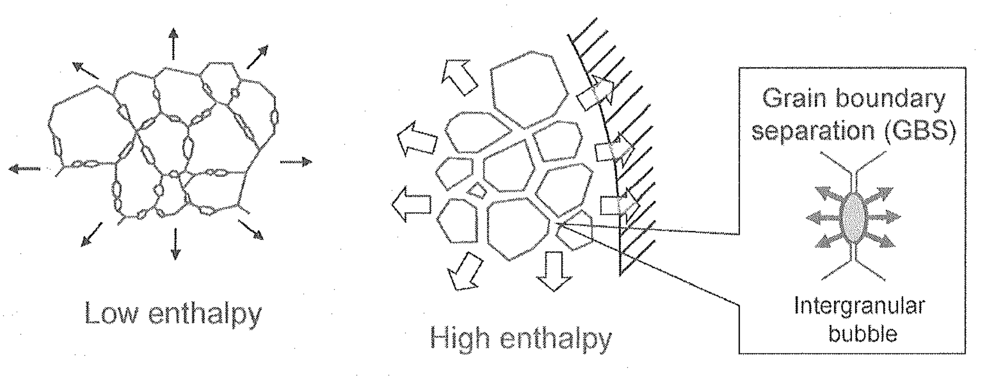
Dependence of normalized Xe/Kr ratio on enthalpy



- The higher enthalpy brought higher value of normalized Xe/Kr ratio.
  → Contribution of fission gas existing in grain boundary of the outer region is highly dependent on enthalpy.

9

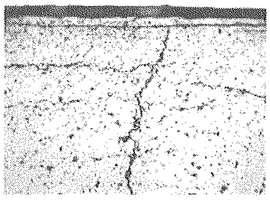
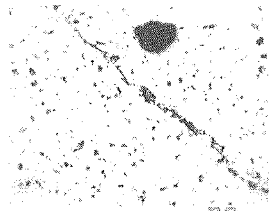
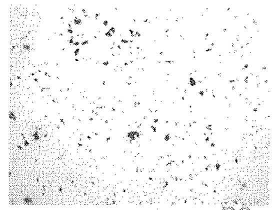
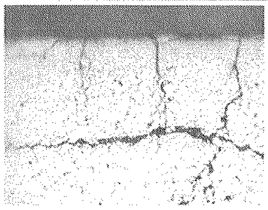
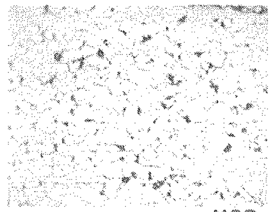
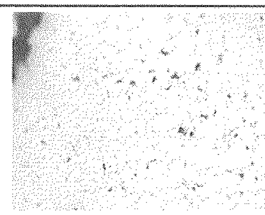
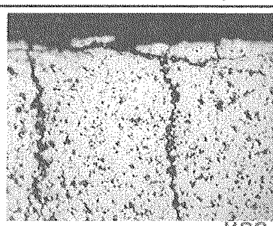
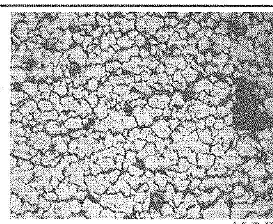
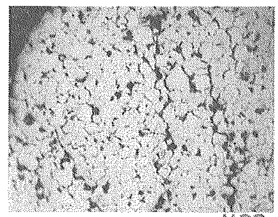
Concept of mechanism on fission gas release during RIA conditions



- Low enthalpy condition
  - Small amount of fission gas accumulated in grain boundary is released.
- High enthalpy condition
  - Increased pressure of intergranular gas bubble.
  - Burst release of fission gas accumulated in grain boundary occurred.
  - Grain boundary separation (GBS) is generated, consequently.

10

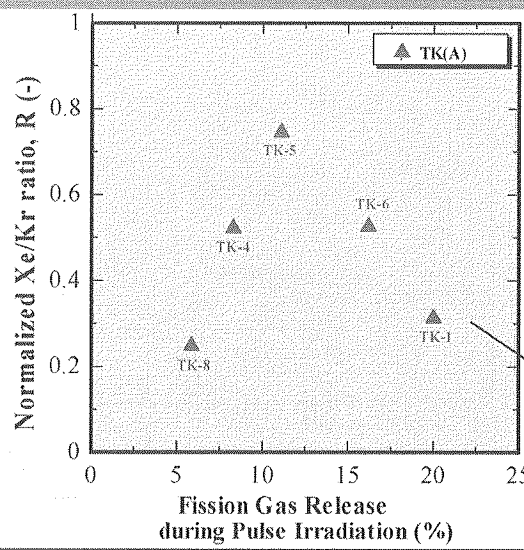
Metallography of radial cross section of pellet after pulse

Polished	Outer Edge	Intermediate	Center
TK-8 272 J/g 50GWd/t	 no	 no	 no
TK-4 410 J/g 49GWd/t	 yes	 yes	 no
TK-1 527 J/g 38GWd/t 0.1 mm	 yes	 yes	 yes

no: GBS was not observed yes: GBS was observed

11

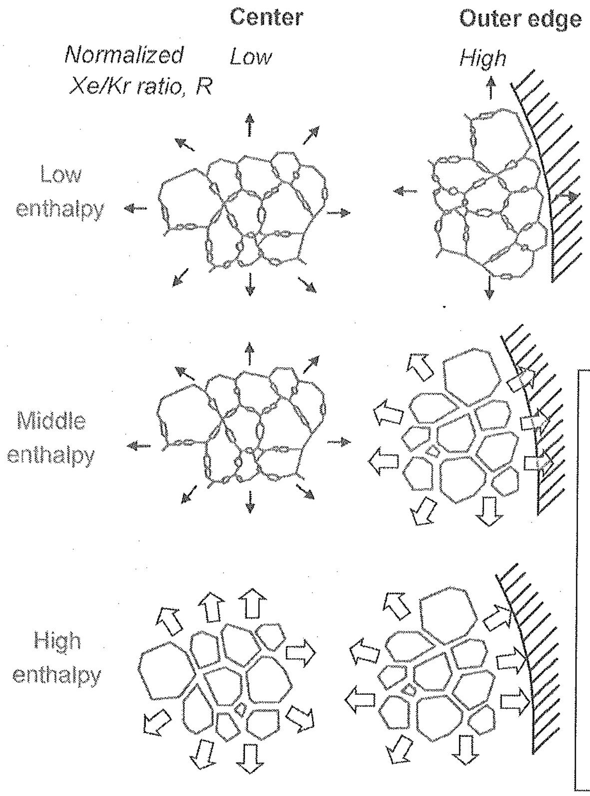
Normalized Xe/Kr ratio corresponding to fission gas release from entire region

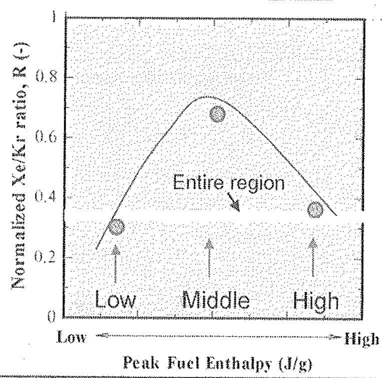


- For Test TK series
  - FGR of TK-1: ~20%(all FP gas in grain boundary)
  - Accordingly, FP gas in grain boundary would be released from entire region.
  - Normalized Xe/Kr ratio is 0.31 in TK-1 corresponding to the value of fission gas released from entire region.

12

Release mechanism and Xe/Kr ratio

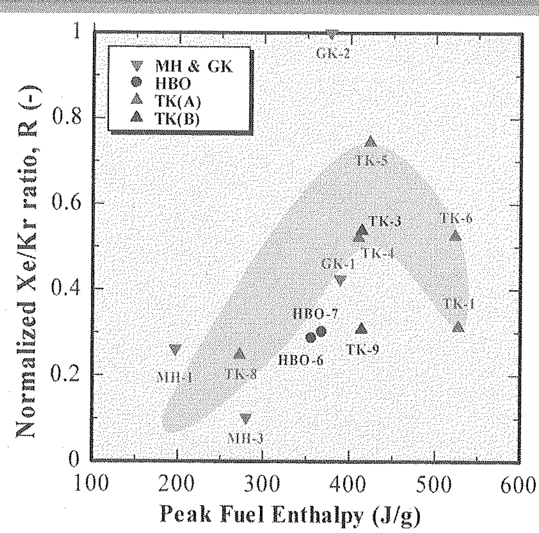




- **Low enthalpy**
  - GBS does not occur in entire region.
  - ⇨ Small amount of fission gas is released uniformly from the entire region.
- **Middle enthalpy**
  - GBS occurs only in outer region.
  - ⇨ Higher value of R is obtained Contribution of fission gas in grain boundaries of the outer region becomes significant.
- **High enthalpy**
  - GBS occurs in entire region.
  - ⇨ Fission gas is released due to the GBS in the entire region. R value becomes close to 0.31 in TK series.

13

Normalized Xe/Kr ratio



- Normalized Xe/Kr ratio, R
  - The higher enthalpy brought the higher value of the Xe/Kr ratio up to middle enthalpy.
  - Contribution of fission gas existing in grain boundary of the outer region would be remarkably significant with the middle enthalpy.
  - R value increases with peak fuel enthalpy and then, decreases to that corresponding to entire release.

14

Summary

- ➢ Data of the Xe/Kr ratio obtained from puncturing of post-test rod give information regarding whether the gas is released from pellet center, outer or entire region.
- ➢ Radial distribution of fission gas release in the pellet is highly dependent on peak fuel enthalpy.
  - ◆ Low enthalpy
    - Grain boundary separation dose not occur in entire region.
    - Small amount of fission gas is released uniformly from the entire region.
  - ◆ Middle enthalpy
    - Grain boundary separation occurs only in outer region.
    - A higher value of R is obtained for the middle enthalpy condition; hence a contribution of fission gas in grain boundaries of the outer region becomes significant.
  - ◆ High enthalpy
    - Grain boundary separation occurs in entire region.
    - Fission gas is released in the entire region.

15

Session 2-3

RIA-Simulation Experiments on Fresh Fuel Rods with Hydride Rim

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A characteristic failure on high burnup fuels, called hydride-assisted PCMI failure, has been observed in the NSRR experiments. In this failure, cladding embrittlement due to hydrogen absorption has a particular importance. Hence RIA-simulation experiments were carried out on fresh fuel rods with hydride rim as a separate-effect test to investigate an influence of the hydride rim on high burnup fuel behavior. The hydride rim was produced at the cladding peripheral region by localizing the hydrogen precipitation. Fourteen experiments, H-1 through H-14, were performed with the peak fuel enthalpy conditions of 241 to 967 J/g (57 to 231 cal/g). To simulate cladding creep down and pellet swelling in high burnup fuels, radial pellet/cladding gap was narrowed to be 11 to 20 μm. Three levels of hydrogen concentration, 400, 800 and 1200 ppm in a radial average of the cladding, were applied.

In seven experiments, H-2 through H-8, cladding failure occurred at cladding surface temperatures below 50 degree C, and fuel enthalpy at failure ranged from 190 to 445 J/g (45 to 106 cal/g). A long axial crack was generated over the fuel stack region of the cladding. Post-test metallography of the horizontal cross section showed brittle fracture in the outer region of the cladding and ductile one in the inner region. Since these features are similar to those observed in high burnup PWR fuels, the hydride-assisted PCMI failure was well simulated with fresh fuel rods with hydride rim. While high burnup fuel tests showed lower failure limits at higher hydrogen concentrations, clear dependence of failure limit on hydrogen concentration was not observed in the present study.

A simple comparison of the cladding deformation was applied between the measured cladding strain and the estimation based on free thermal expansion of the UO_2 pellet. As a result, the estimation was in good agreement with the measurement, especially in deforming rate. It is accordingly concluded that thermal expansion of the pellet is a dominant driving force at PCMI and a role of the fission gas is very limited to the PCMI failure.

RIA-Simulation Experiments on Fresh Fuel Rods with Hydride Rim

Kunihiko TOMIYASU

Fuel Safety Research Laboratory
Japan Atomic Energy Research Institute

Fuel Safety Research Meeting, Tokyo, March 2-3, 2005

①

OUTLINE

- INTRODUCTION
  - Background & Purpose
- EXPERIMENTAL CONDITION
  - Pulse Condition
  - Cladding with Hydride Rim
  - Instrumentation
- RESULTS & DISCUSSIONS
  - Transient Behavior
  - Comparison with High Burnup Fuel PWR Tests
  - Evaluation of Thermal Expansion of the Pellet
- CONCLUSIONS

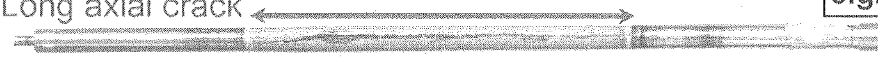
②

INTRODUCTION

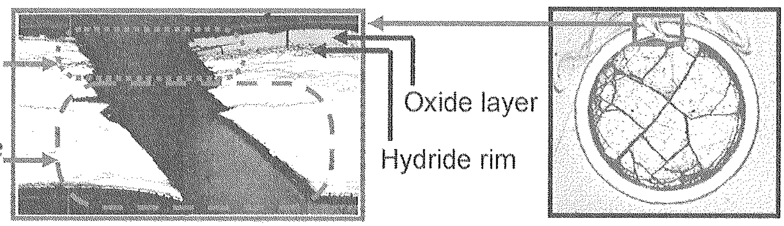
Hydride-assisted PCMI failure has been observed
in the NSRR Experiments on high burnup PWR fuels.

e.g. HBO-5

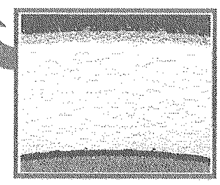
- Long axial crack



- Brittle fracture (outer region)
- Ductile fracture (inner region)



RIA-simulation experiments were carried out
on fresh fuel rods with hydrided rim.



to Investigate ...

- Hydride-assisted PCMI failure occurs?
- Failure threshold depends on hydrogen concentration?

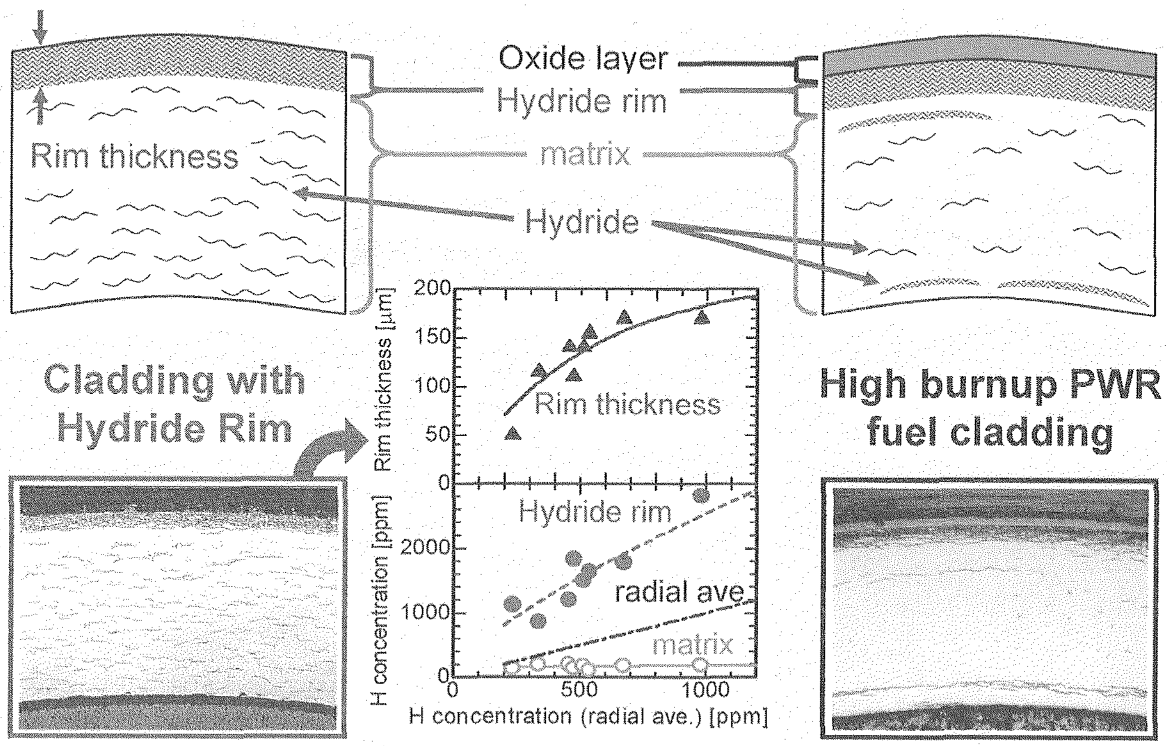
3

EXPERIMENTAL CONDITION

Experiment ID	H-1 —	H-14	(14 in total)	
Peak fuel enthalpy	241 —	967	J/g	
	(57 —	231	cal/g)	
Radial P/C gap width	11 —	20	μm	
Hydrogen concentration				
radial average	400,	800,	1200	ppm
in hydride rim	1300,	2200,	2900	ppm
Rim thickness	120,	170,	190	μm
Failure	H-2 —	H-8	(7 in total)	

4

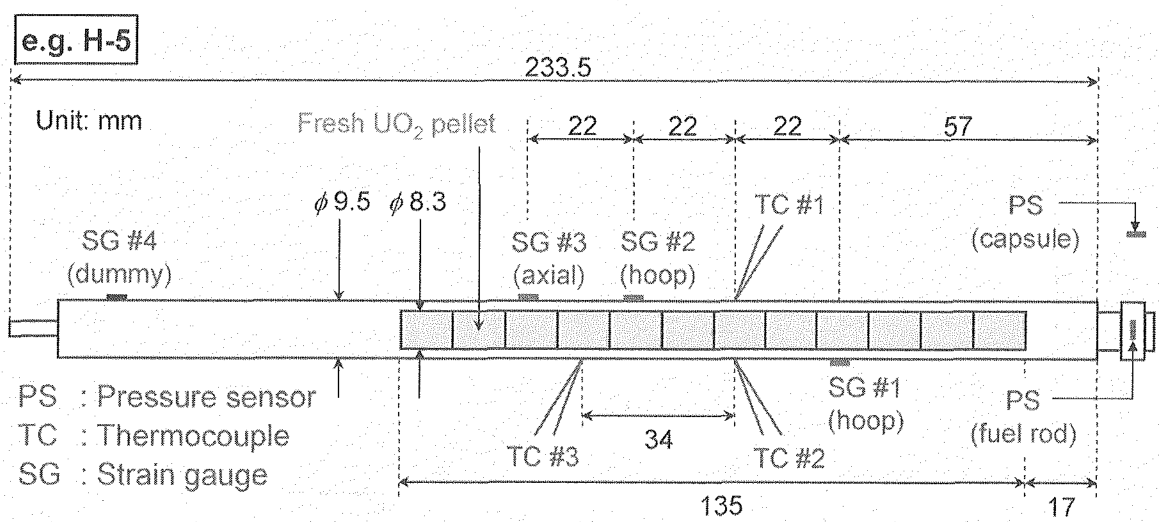
Cladding with Hydride Rim



* Nagase, et al., JAERI-Res. 2000-046 (2000)

5

Instrumentation

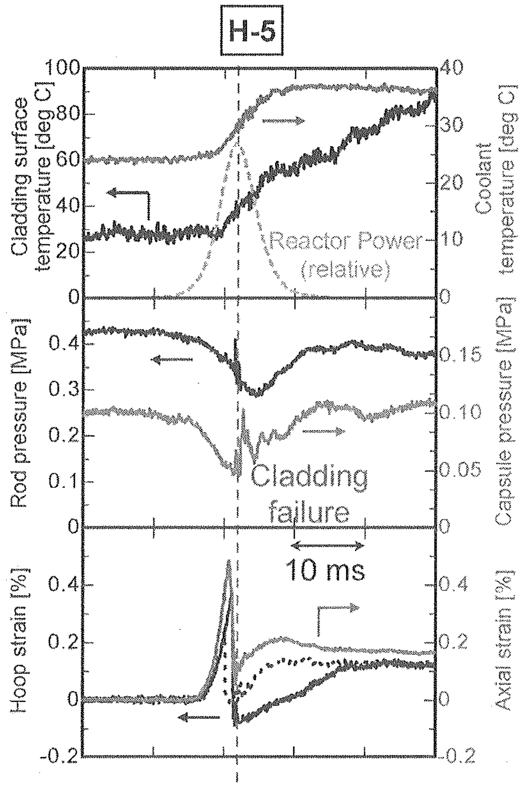


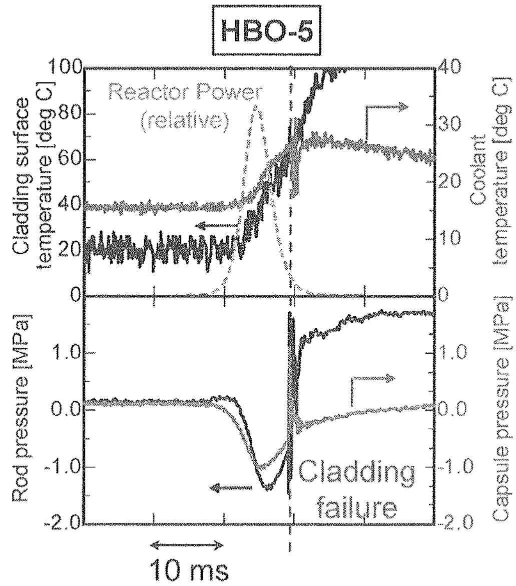
Transient measurement:

- Capsule internal pressure
- Rod internal pressure
- Cladding Surface Temperature
- Hoop and axial strain

6

Transient History





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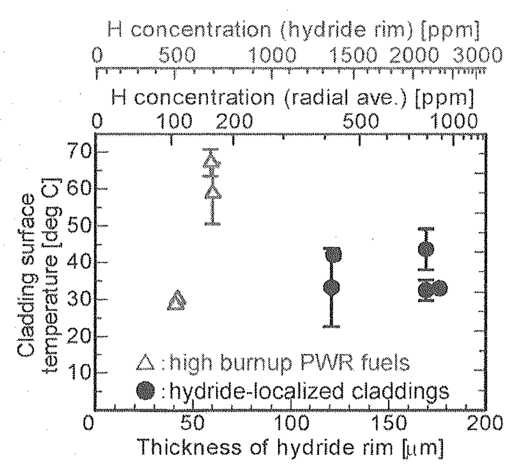
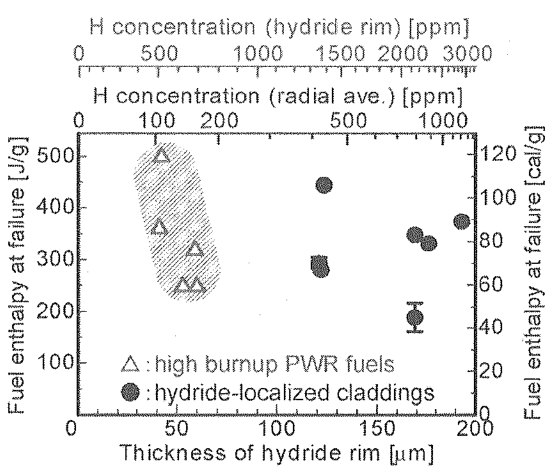
Comparison with High Burnup Fuels

Cladding Surface Temperature at Failure



Fuel Enthalpy at Failure

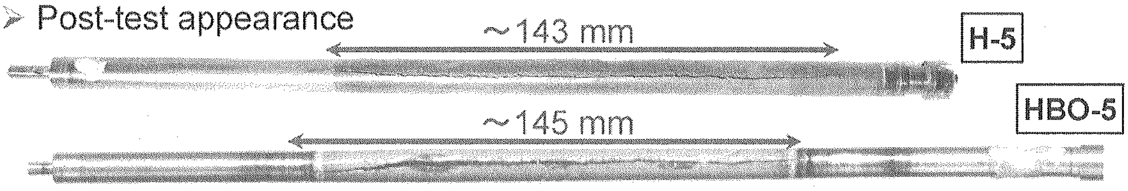


- Cladding surface temperature at failure: < 50 C
- Fuel enthalpy at failure: 190~445 J/g (45~106 cal/g)
  → **Comparable with those at PCMI failure on high burnup fuels**
- **No dependence of the failure threshold on H concentration**

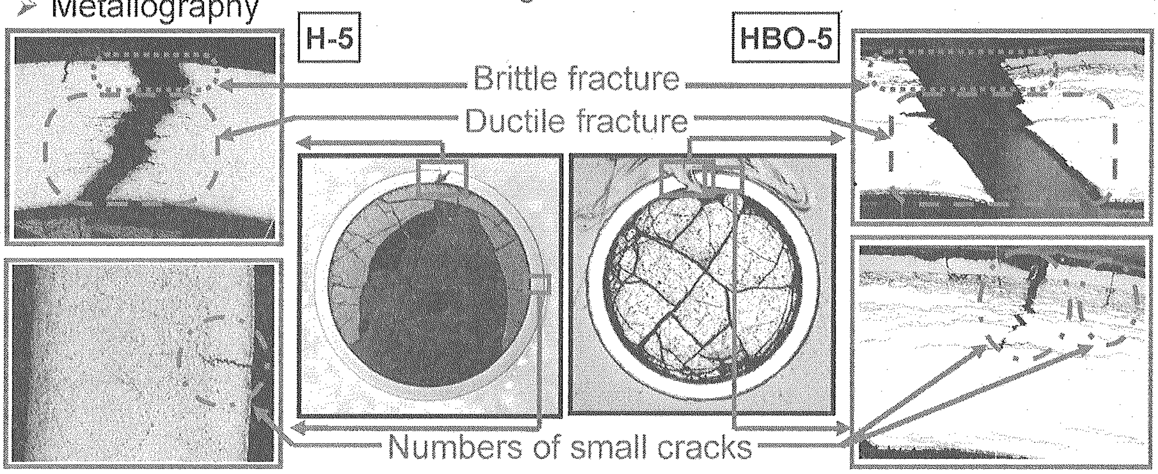
⑧

PIE Results

➢ Post-test appearance



➢ Metallography



Similar configuration to that of PCMI failure on high burnup fuels

(9)

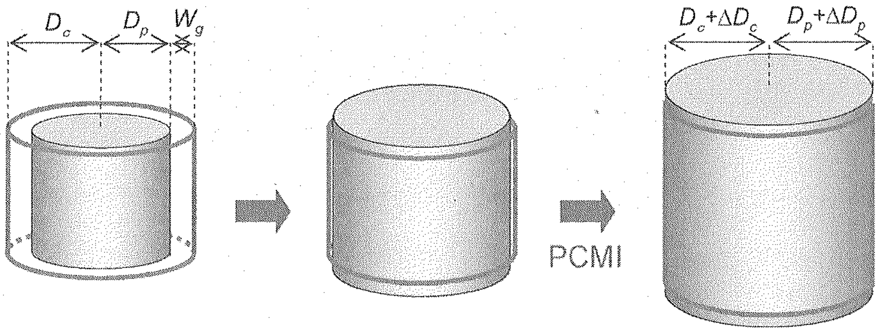
Thermal Expansion of Pellet

Simple comparison was applied between a measured strain and an estimation based on free thermal expansion of the pellet.

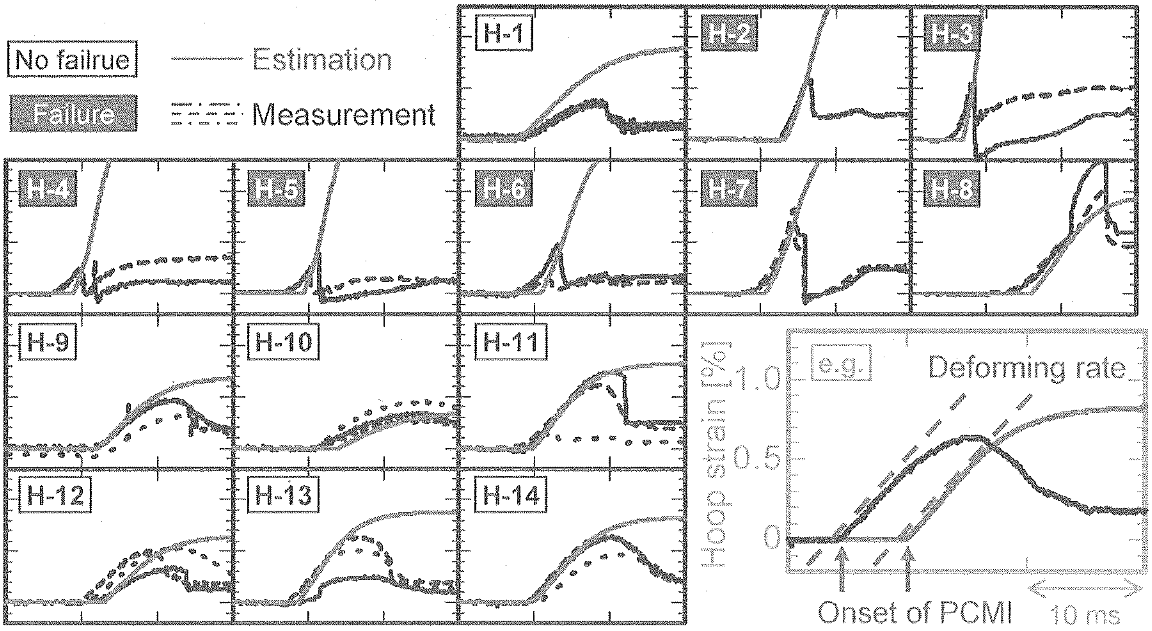
$$\frac{\Delta D_p}{D_p} = 1\times10^{-5}T - 3\times10^{-3} + 4\times10^{-2}\exp\left(\frac{6.9\times10^{-20}}{kT}\right) \quad \text{* MATPRO}$$

$$\frac{\Delta D_c}{D_c} = \frac{\Delta D_p - 2W_g}{D_c} = \varepsilon$$

D_p : pellet diameter [mm]
T : pellet temperature [K]
k : Boltzmann's constant (=1.38x10^{-23} J/K)
D_c : cladding outer diameter [mm]
W_g : P/C gap width [mm]
ε : cladding strain



(10)



- Onset of PCMI ... Disagreement
  → **Uncertainty of the estimated P/C gap width**
- Deforming rate ... Good agreement
  → **Thermal Expansion of the pellet as a dominant driving force at PCMI**

(11)

CONCLUSIONS

1. The failure on fresh fuel rods with hydride rim were **similar** to those observed in high burnup PWR fuels.

- Cladding temp. at failure : < 50 C
- Fuel enthalpy at failure : 190~445 J/g (45~106 cal/g)
- Configuration of fracture : Long axial crack, Brittle fracture (outer region), Ductile fracture (inner region)

→ Hydride-assisted PCMI failure on high burnup fuels was well simulated.

2. Estimated cladding deformation based on free thermal expansion of the pellet was in good agreement with the measured strain.

→ Thermal expansion of the pellet is a dominant driving force at PCMI.

1, 2 → suggest a small contribution of Fission Gas at PCMI

3. Failure threshold did not depend on hydrogen concentration in the present study.

(12)

Session 2-4

Development of the RANNS code for high burnup fuel behaviors in RIA conditions

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Objectives of the RANNS code are investigation of the mechanism underlying the observations in the NSRR experiments, analysis on the failure condition, and prediction of fuel behaviors in RIA in commercial reactors.

Model and method of the recent RANNS code, and results of experimental analyses are described in detail and discussed in terms of PCMI and model validity.

Mechanical and thermal analyses were performed for the PCMI behaviors of high burnup BWR rod (FK-10, 61GWd/t) and PWR rod (OI-10, 60GWd/t), which were pulse-irradiated under cold start-up conditions in the NSRR experiments.

In the analysis of the FK-10 rod, thermal expansion in the inner region of cladding counteracts the tensile stress induced by pellet instantaneous swelling, resulting in relatively higher stress in the outer region of cladding. This can successfully account for the observation that a failure crack was initiated at the outer surface in the experiment.

In the analysis of the OI-10 rod, calculated permanent (residual) strain of cladding is compared with the measured data. While the calculated strain is overall in good agreement with the data, locally enhanced strain, or bulging expansion which appeared in measured profile raises another point for RANNS modeling.

On the basis of the above results, a new PCMI model for local bulging is being developed as one of the evolutions of the RANNS code.

1

Fuel Safety Research Meeting, March 2-3, 2005, Tokyo.

Development of the RANNS code for high burnup fuel behaviors in RIA conditions

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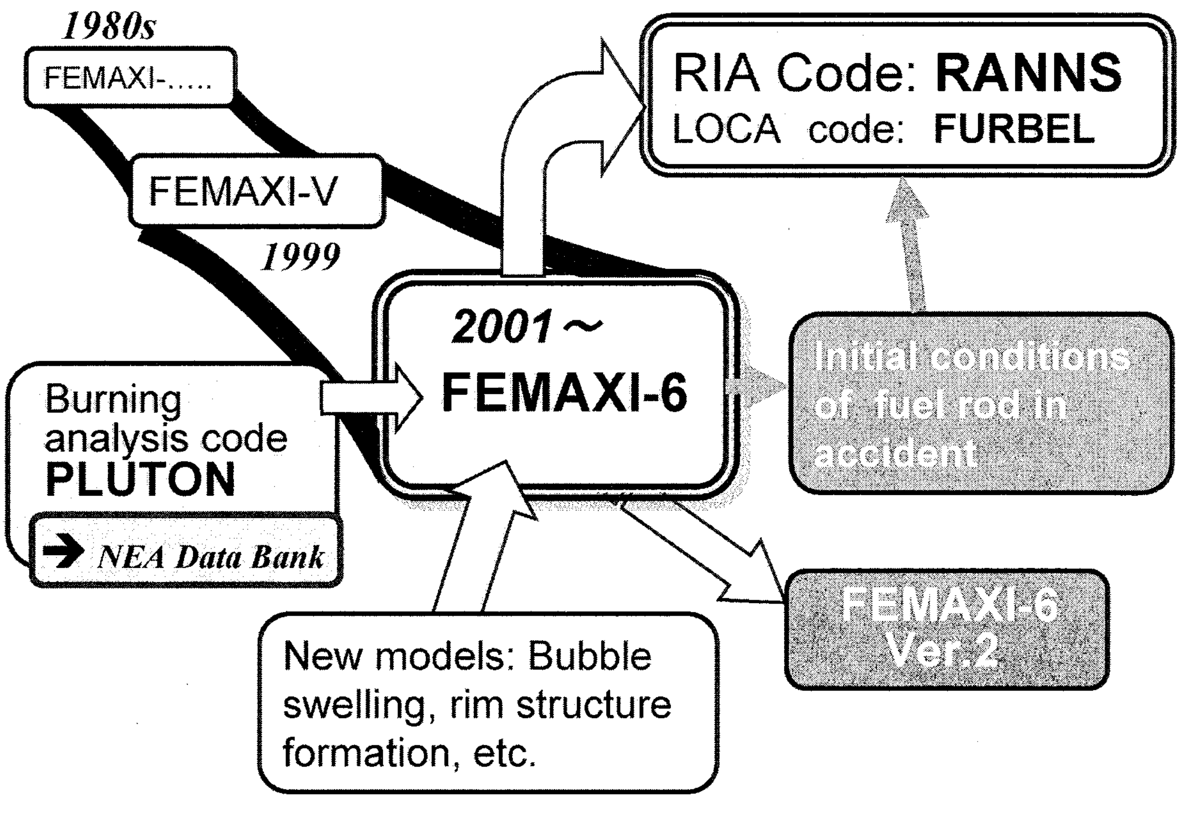
2

Code Development Progress



3

Development of the RANNS code

◆Objectives

1. To investigate the mechanism underlying the observations in the NSRR experiments.

   ←→ Limited amount of test rods and data.

2. To analyze and predict failure condition.
3. To predict RIA behaviors in commercial reactors.

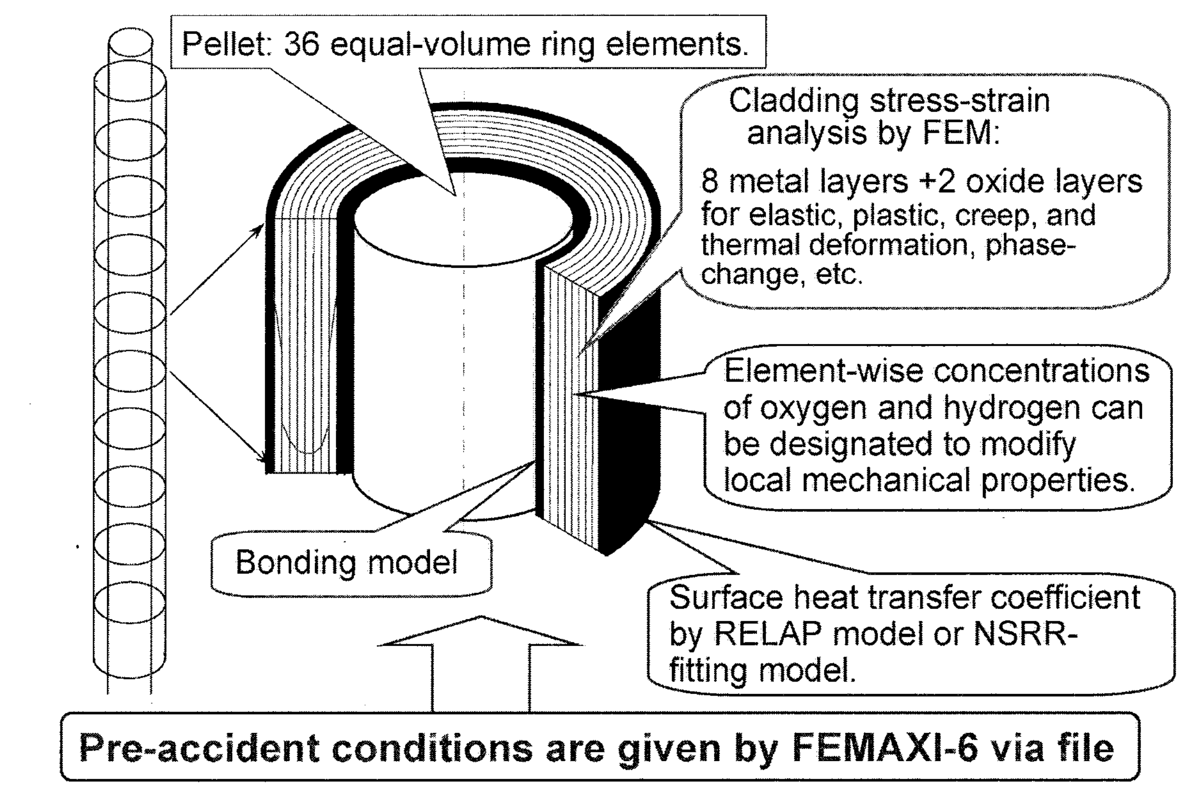
◆Features of model and method.

◆Progress of experimental analysis.

◆Evolution with new model.

4

Fig.1 Overview of "Cylinder expansion" model



Ring elements for thermal and mechanical analyses in the "Cylinder expansion" model 5

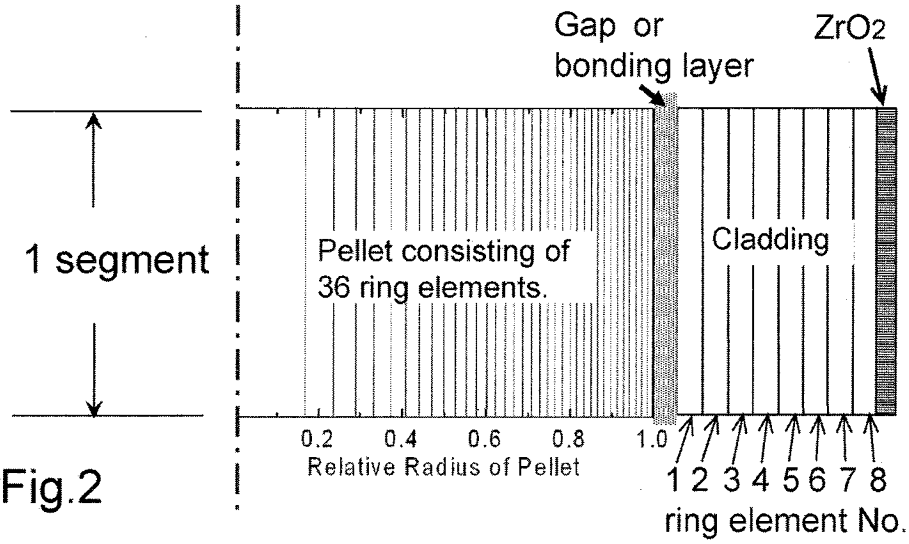


Fig.2

1. Pellet: 36 equal-volume ring elements. Cladding: 8 metal + 2 oxide rings.
2. Pellet instantaneous swelling is induced by solid thermal expansion only.
3. Oxidation is calculated with an independent mesh and results are given to re-size the ring elements.

PCMI analyses of NSRR experiments by "Cylinder expansion" model 6

1) FEMAXI-6 gives pre-accident conditions.

2) BWR rod: FK-10 (61GWd/t), failed by PCMI.
Cladding temperature,
Stress and strain vs. strain gauge data.

3) PWR rod: OI-10 (60GWd/t), survived.
Cladding diameter profile indicates local enhancement of PCMI strain.

Diameter changes of FK-10 rod by FEMAXI-6

7

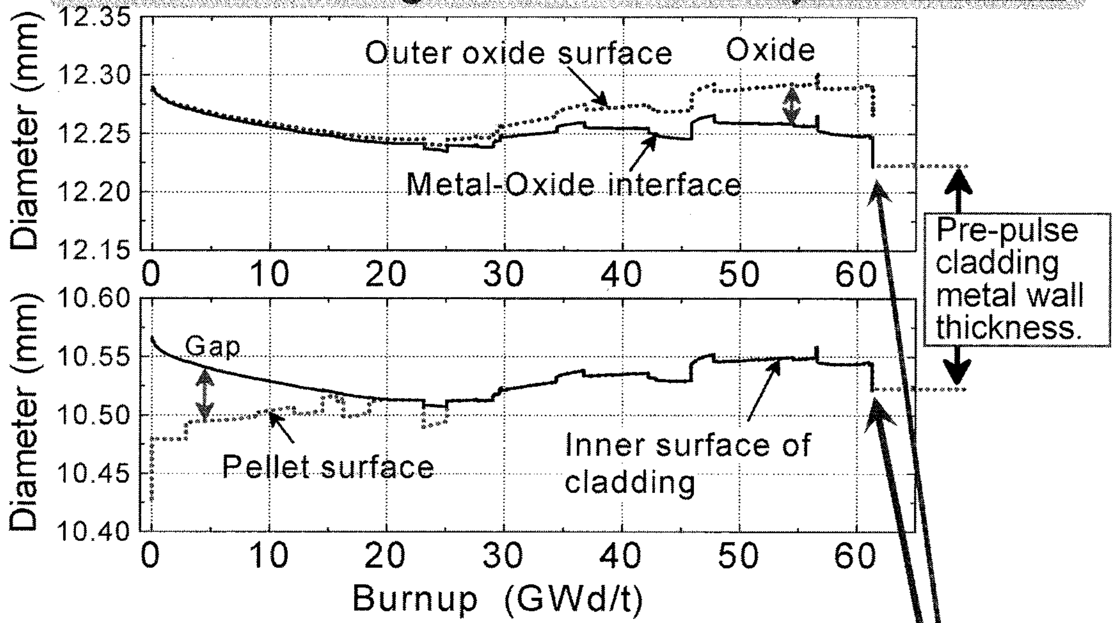


Fig.3 Diameter and cladding thickness change calculated by FEMAXI-6 during the base-irradiation of FK-10 rod in BWR.

◆ FEMAXI-6 results: stress-strain of pellet and cladding, burnup and heat generation density profiles, etc. are fed to RANNS.

Analysis of **FK-10** experiment in the NSRR

8

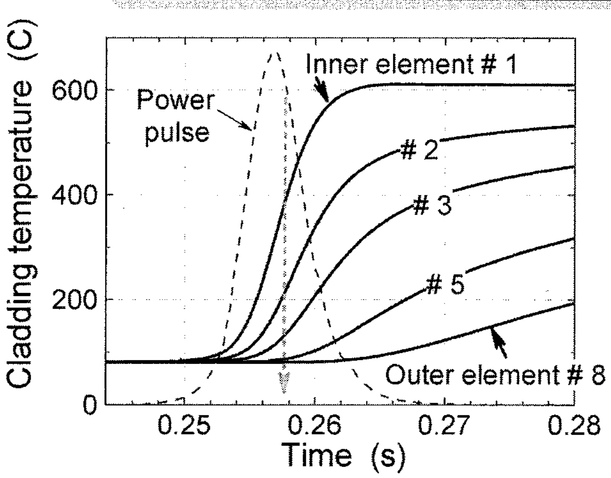


Fig.4 Cladding temperature in the five ring elements.

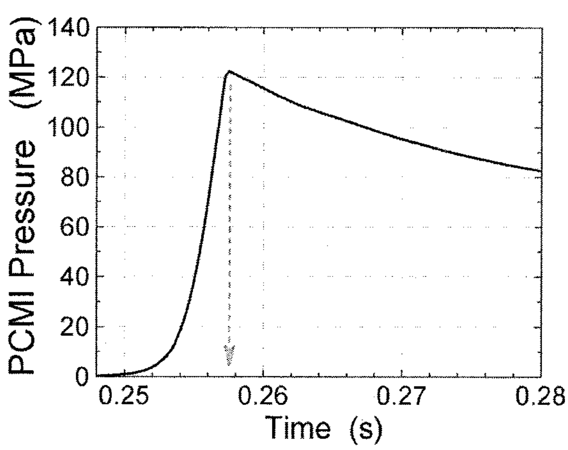


Fig.5 Pellet-clad PCMI contact pressure.

1. Large temperature difference across the wall is induced.

2. PCMI pressure peaks and gradually attenuates.

9

Cladding stress in the **FK-10** rod failed by PCMI

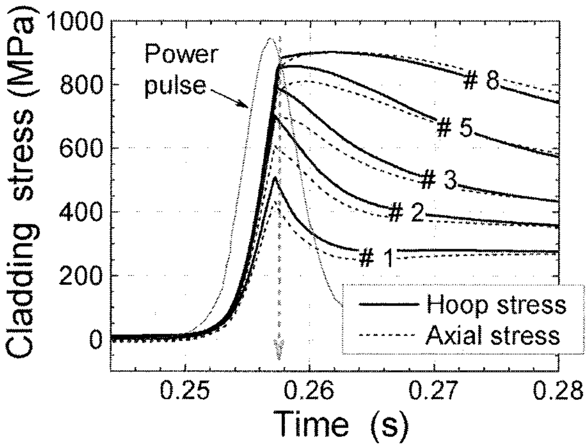


Fig.6 Calculated cladding hoop and axial stresses.

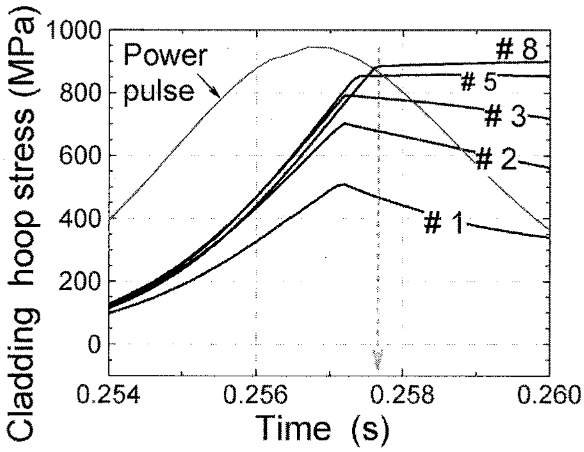
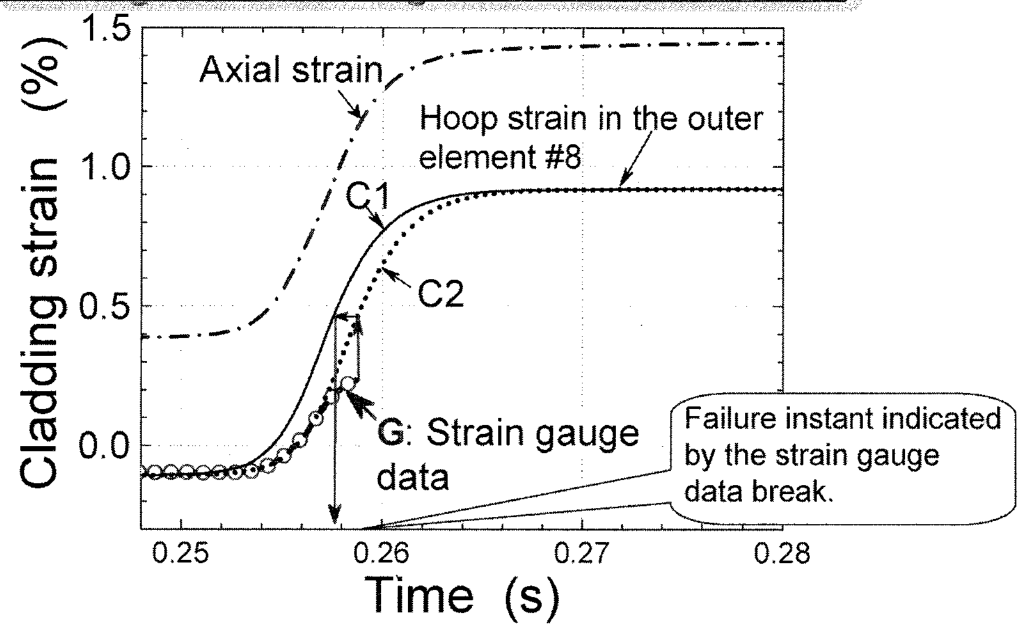


Fig.7 Cladding hoop stress in zoomed time scale.

1. Thermal expansion counteracts the tensile stress in the inner region.
   ➔ The failure crack is generated in the outer surface of cladding.
2. Failure stress at the failure instant is estimated to be 890 MPa.
3. Temperature and stress gradients across the cladding wall is essential in the PCMI analysis of RIA of high burnup fuel rod.

10

Fig.8 Cladding strains in **FK-10**



1. "1 ms shift" of the calculated curve C1 gives C2 with a better agreement.
2. In the "1 ms" period, cracks in the pellet is closed and PCMI begins.
3. Failure stress and failure instant can be estimated by the strain gauge data break.

Analysis of **OI-10** experiment in the NSRR

11

Cold start-up; the rod survived with permanent strain of cladding.

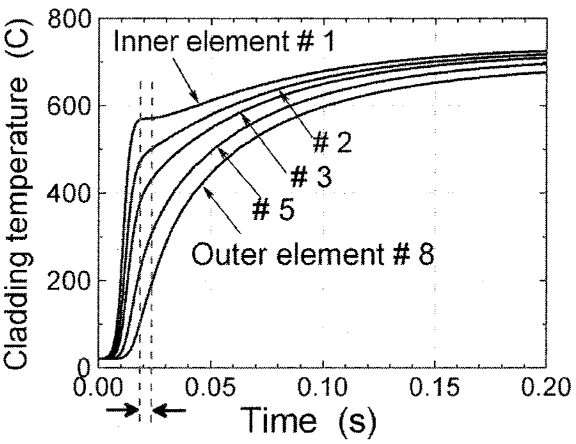


Fig.9 Cladding temperatures in the five ring elements

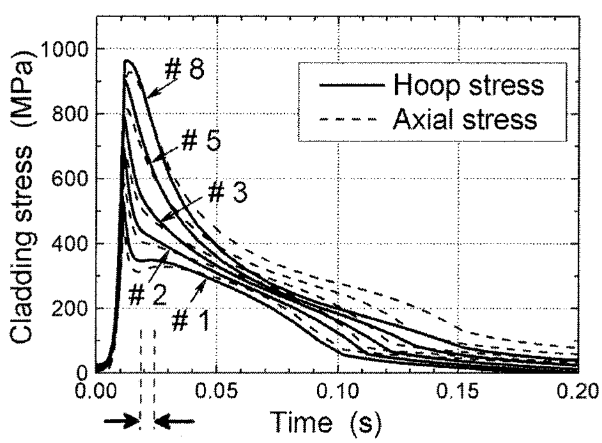


Fig.10 Cladding hoop and axial stresses.

1. Large temperature gradient; stress in the outer region is relatively high.
2. NB and DNB are predicted, but heat transfer after onset of DNB is uncertain.

Total strain of cladding calculated in the **OI-10**

12

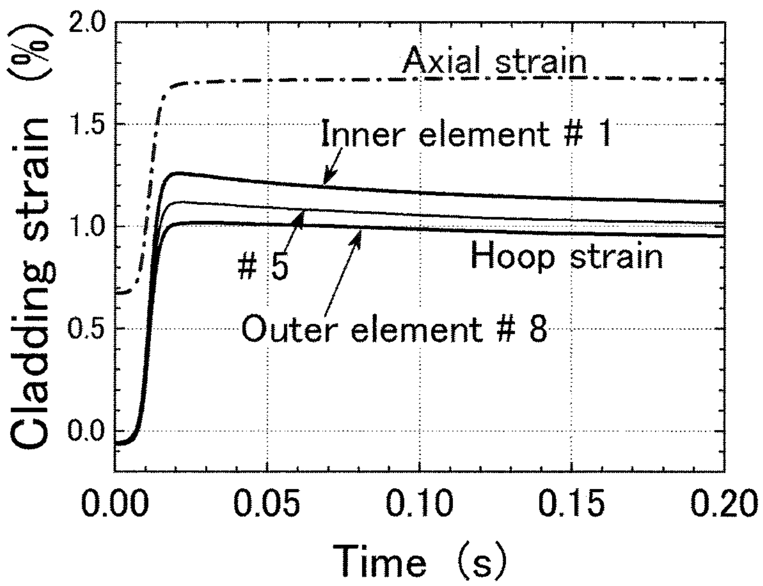


Fig.11 Cladding total strain progress during PCMI.

Total strain of cladding consists of elastic, thermal, plastic, and creep components. ➔Permanent strain?

13

Plastic and creep strains of cladding **OI-10**

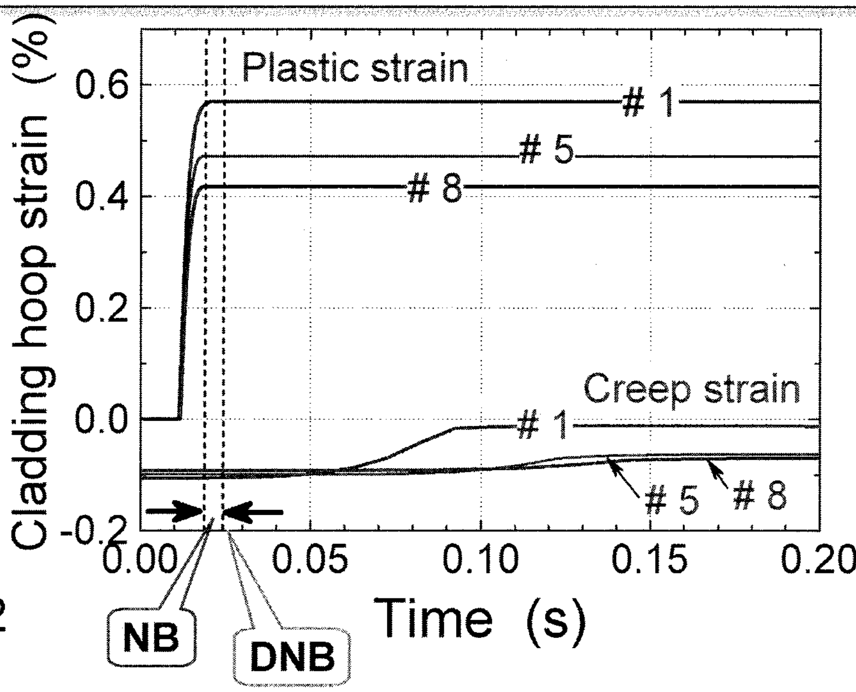
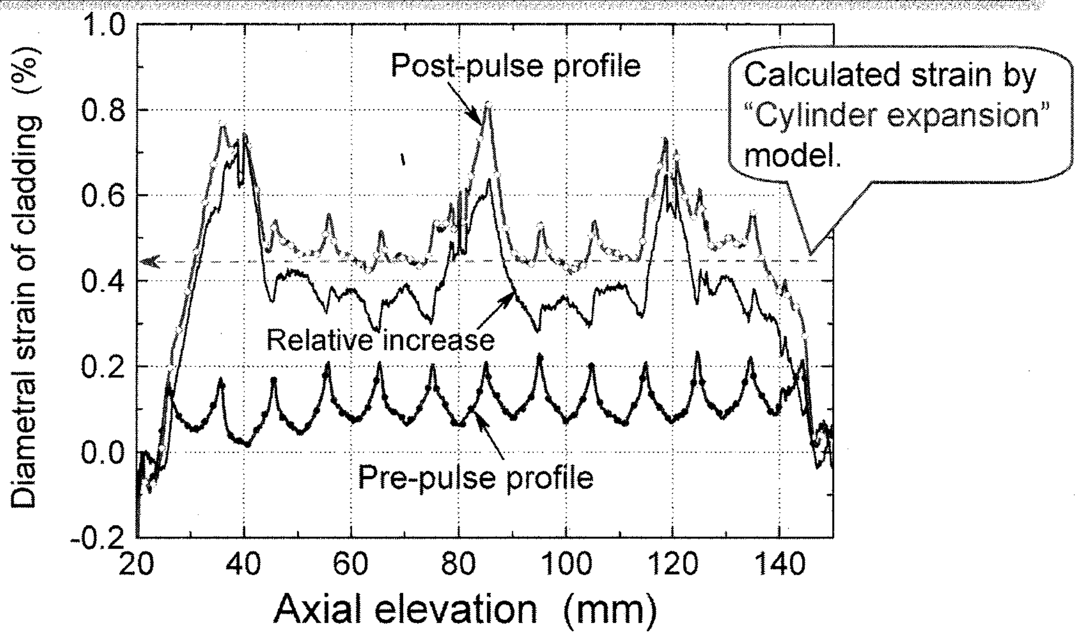


Fig.12

Bulk of the permanent strain of cladding is plastic strain before the onset of nucleate boiling. Creep strain after DNB is slight.

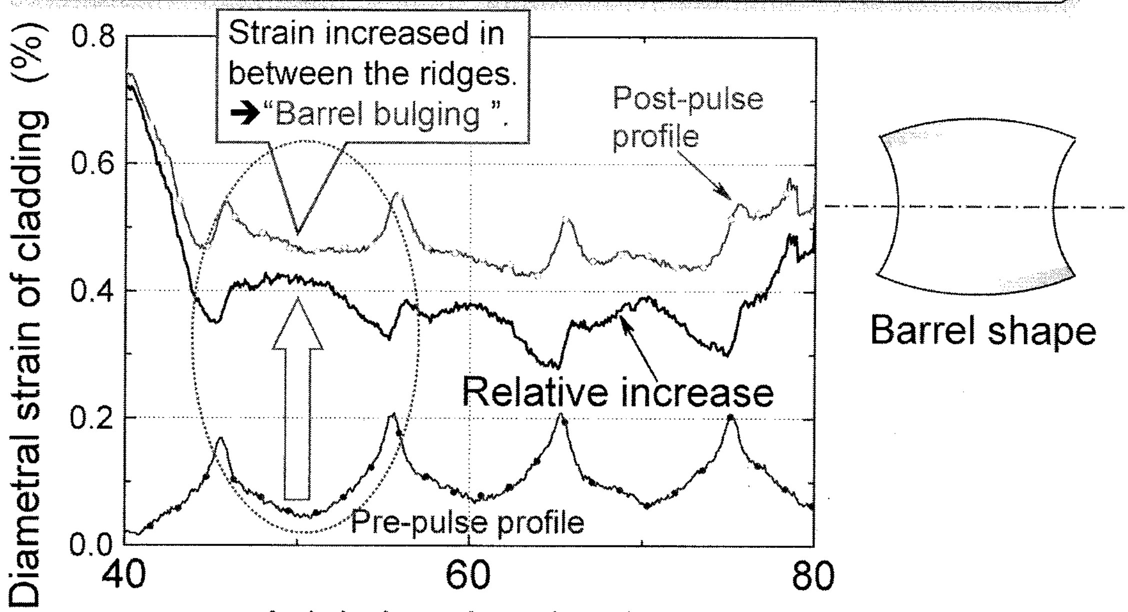
14

Fig.13: Cladding strain profiles in **OI-10**



- The predicted permanent strain has some overestimate.
- The relative strain increase indicates uneven local strain during the pulse-irradiation.

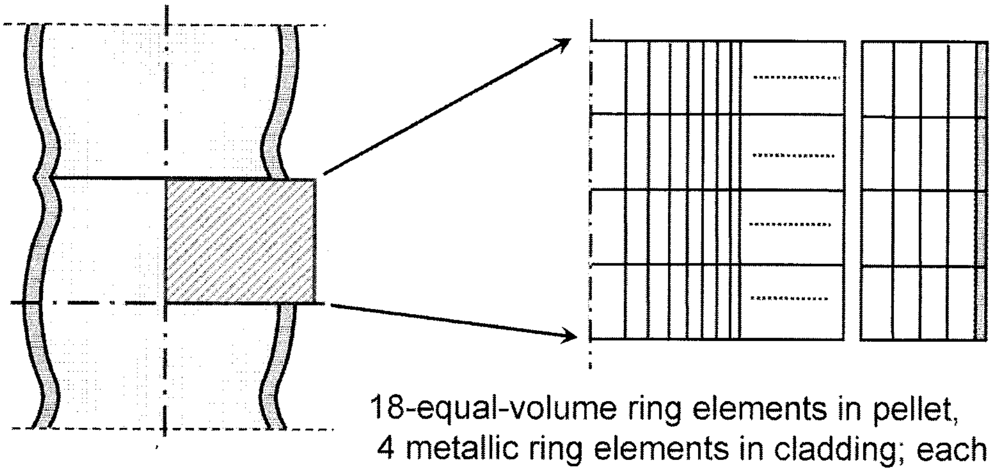
Fig.14: Cladding local strain profile in **OI-10** 15



Locally enhanced strain of cladding induced by "Barrel bulging" of pellet are found in the NSRR experiment with high burnup rods.

"Barrel-bulging" expansion of pellet under modeling 16

Local non-uniformity of stress and strain
➔failure crack initiation;
➔ criteria for failure-non-failure prediction.



18-equal-volume ring elements in pellet,
4 metallic ring elements in cladding; each element has 8-node and quadric displacement function.

Fig.15 FEM geometry for Local PCMI analysis

Summary

1. Experimental analysis by RANNS:

- PCMI analysis by "Cylinder expansion" model:
  Thermal expansion counteracts the tensile stress,
  → Stress in the outer region is relatively high.
- Cladding is subjected to a bi-axial stress state.
  Deformation is dominated by pellet thermal expansion.
- Locally enhanced straining of cladding is one of the important analytical targets in terms of failure crack initiation.

2. Evolution of RANNS

- Local PCMI model is being developed.
- Failure prediction model on the basis of mechanical test results, and Grain separation model will be developed.

3.Session 3 High burnup fuel behavior

Session 3-1

Fuel Behavior under Power Oscillation

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To investigate the fuel behavior during power oscillation of BWRs, three irradiated fuel rods were tested under simulated power oscillation conditions in the Nuclear Safety Research Reactor(NSRR). One is high burnup BWR fuel (56GWd/t) test, with 4 power oscillation cycles, to clarify the behavior of high burnup fuel. The second and third ones are high enriched fuel(19.5%,16-25GWd/t) test, with 7 power cycles, to perform the test under high power conditions. The fuel behavior data, such as cladding elongation, fuel stack elongation, cladding temperature, etc. were obtained in these tests. DNB occurred only in one test. The fuel rod integrity was confirmed under the tested conditions without DNB, and the plastic deformation was very small.

In case DNB occurred, small deformation of cladding was observed due to the temperature increase of cladding during film boiling.

PCMI was observed by cladding elongation measurement in the tests. The cladding deformation was roughly proportional to estimated fuel enthalpy rather than to the power. The cladding deformation in the power oscillation conditions was comparable to those observed in the RIA tests at similar fuel enthalpy levels.

The fission gas release was quite smaller in the power oscillation tests than in the RIA tests. The slower heatup rate during power oscillation could be one reason for the smaller release from the grain boundaries. The amount of the fission gas release is comparable to that of diffusion release from fuel grains during power oscillation calculated with FASTGRASS code.

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Fuel Behavior under Power Oscillation

Jinichi NAKAMURA
Japan Atomic Energy Research Institute

Fuel Safety Research Meeting

March 2-3, 2005, Tokyo

1

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Background

- Fuel behavior during ATWS such as power oscillation in BWR is one of the concerns in the fuel safety research area, as pointed out in PIRTs (the Phenomenon Identification and Ranking Tables) by USNRC. However, frequency of the power oscillation in BWRs is about 0.5Hz and it is difficult to perform such a power oscillation tests in test reactors.
- JAERI has tried to perform a simulation tests of power oscillation in BWR by means of the NSRR, and succeeded to perform power oscillation test at the NSRR.
- Three tests with irradiated fuel have been conducted to clarify the fuel behavior during power oscillation conditions.

2

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Fuel failure during power oscillation pointed out in PIRTs

- Low temperature phase

  PCMI failure of low ductility cladding could result in a degraded coolable geometry.

- High temperature phase

  Critical heat flux may be exceeded, and the high temperature may lead to oxidation, ballooning, rupture and fragmentation.

3

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Test matrix for power oscillation tests

	Test conditions		Coolant conditions	Instrumentation
	Number of power peaks	MAX LHR (kW/m)		
High burnup BWR fuel(56GWd/t) (FK-11)	4	50	Room temperature	CE FE CT
JMTR preirradiated high enriched (20%) fuel (25GWd/t) (JMH-6)	7	96	Smaller subcooling conditions with heater ～80℃	CE CT
JMTR preirradiated high enriched (20%) fuel (16GWd/t) (JMH-8)	7	89	Smaller subcooling conditions with heater ～90℃	CE CT

CE:cladding elongation, FE:fuel stuck elongation,

CT:cladding temp., FT:fuel center temp.

4

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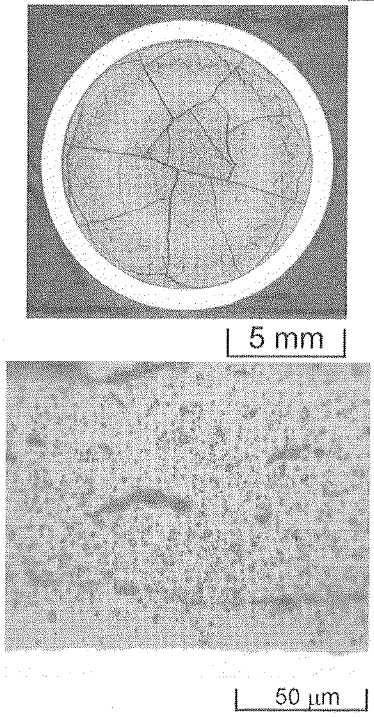
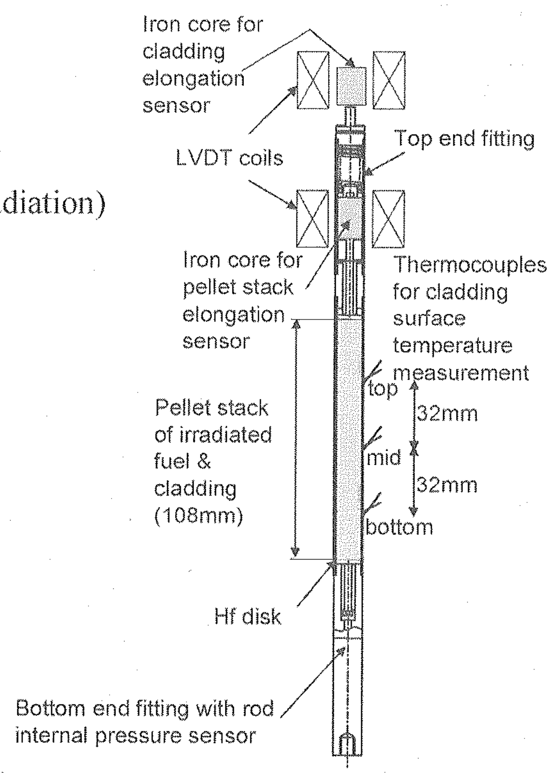
Fuel rod for FK-11 test



Sibling rod

4 cycle irradiation)

Bonding at pellet cladding gap



5

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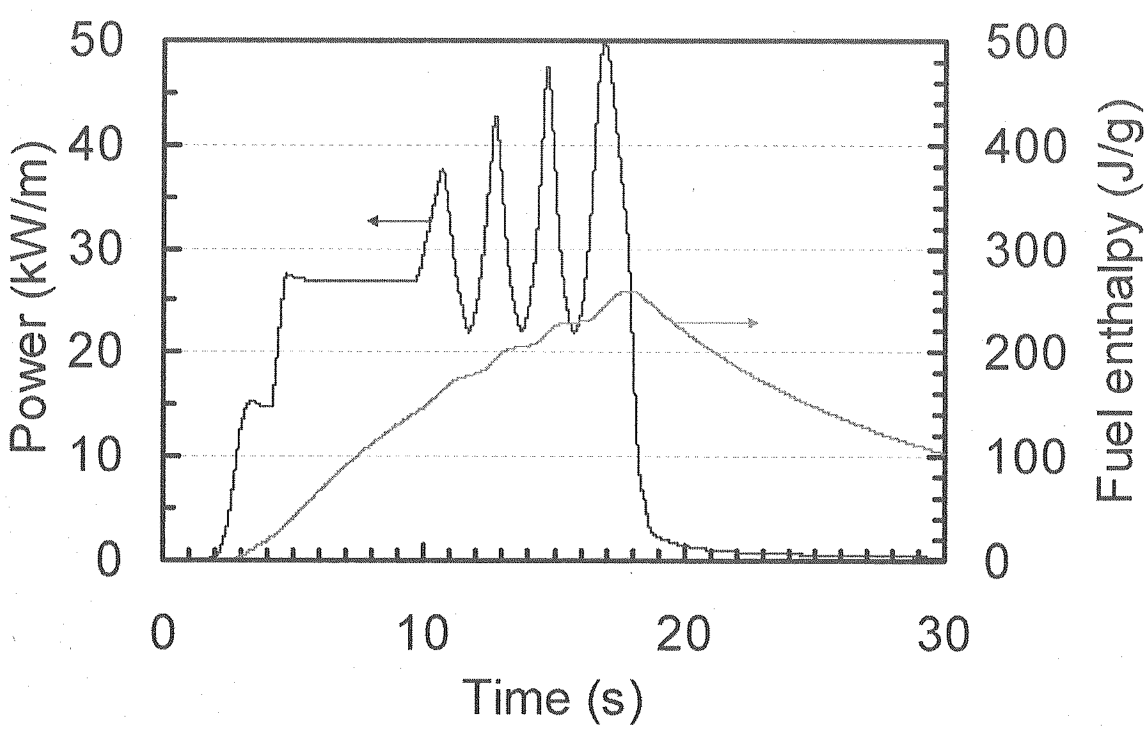
Rod Power and Fuel Enthalpy of High burnup BWR Fuel (FK-11)



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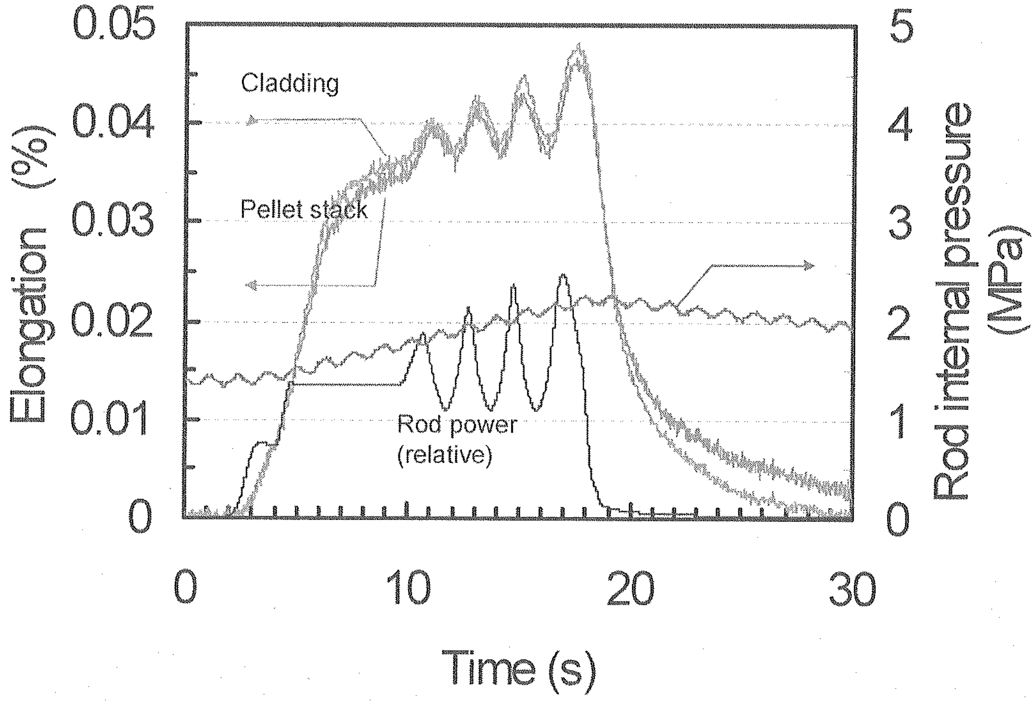
Elongation, Power and Rod pressure of High burnup BWR fuel(FK-11)



7

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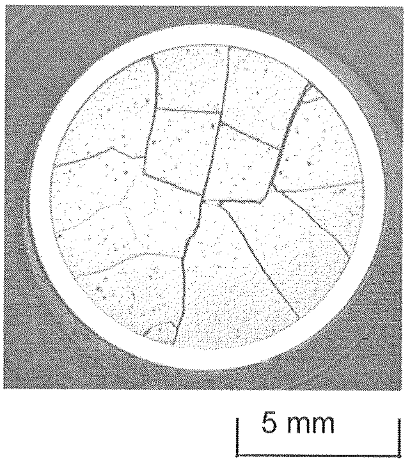
Fuel rod for JMH-6/-8 test

- To investigate fuel behavior under high power conditions.
- 14x14 type PWR fuel rod with 20% enrichment

- Pre-irradiated in JMTR up to 25GWd/t(JMH-6), 16GWd/t(JMH-8)
- Initial gap :190 μ m, some gap remains after pre-irradiation in JMTR



Sibling rod

8

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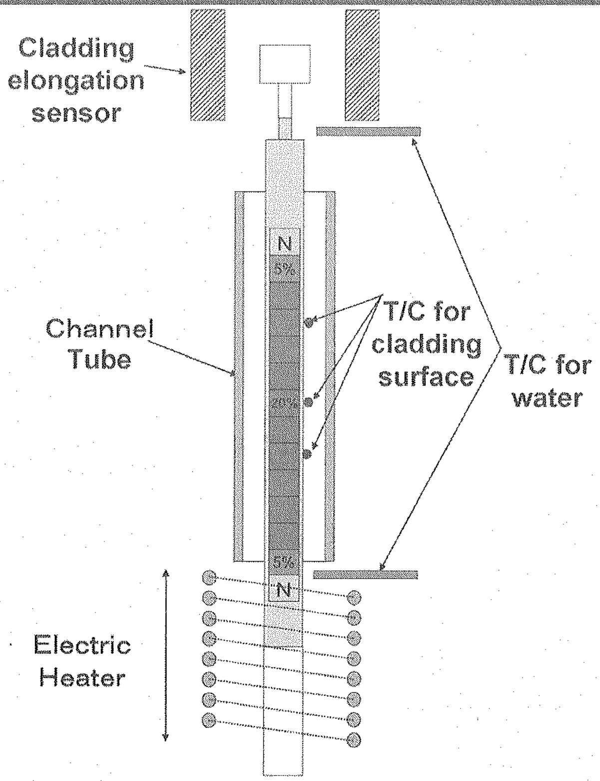
Configuration around fuel rod (JMH-6/JMH-8)



- Smaller subcooling with electric heater
- Reduced amount of available coolant with channel tube

9

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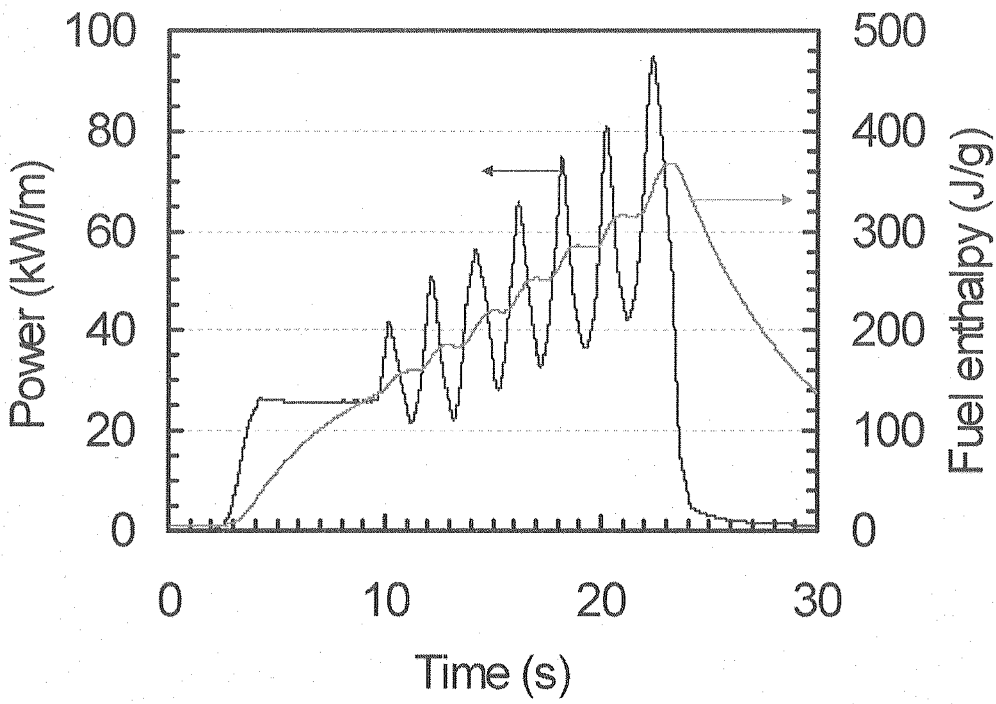
Rod power and fuel enthalpy of JMH-6 test



10

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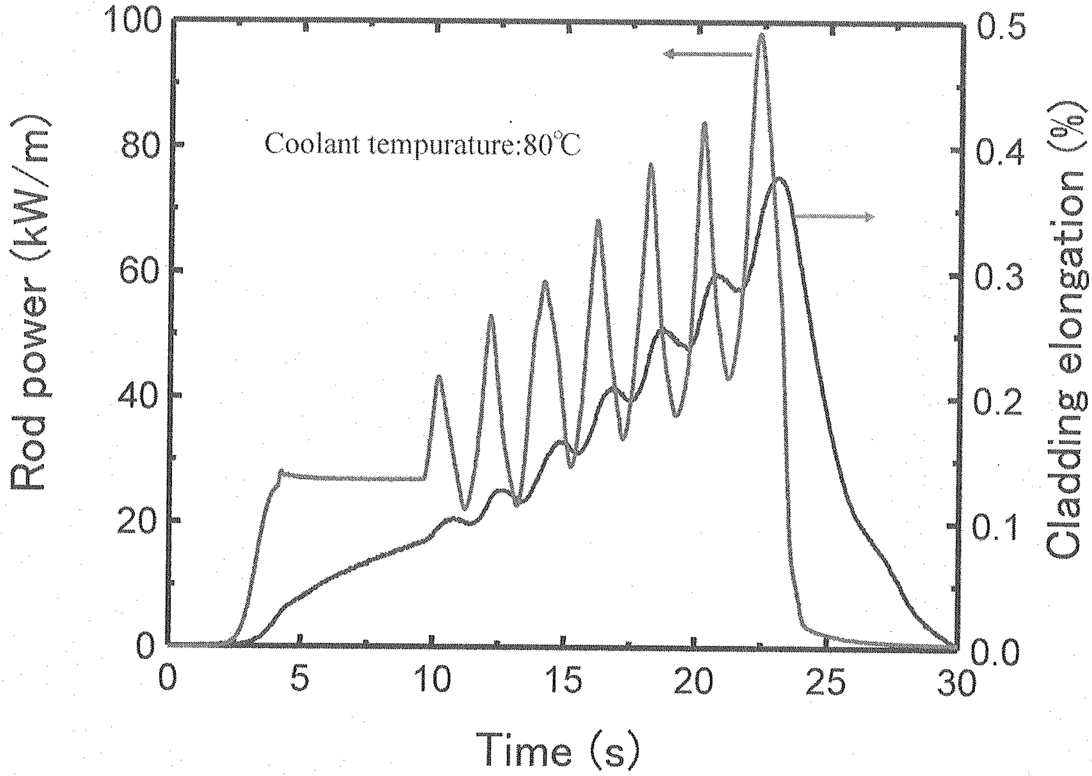
Rod power and cladding elongation of JMH-6 test



11

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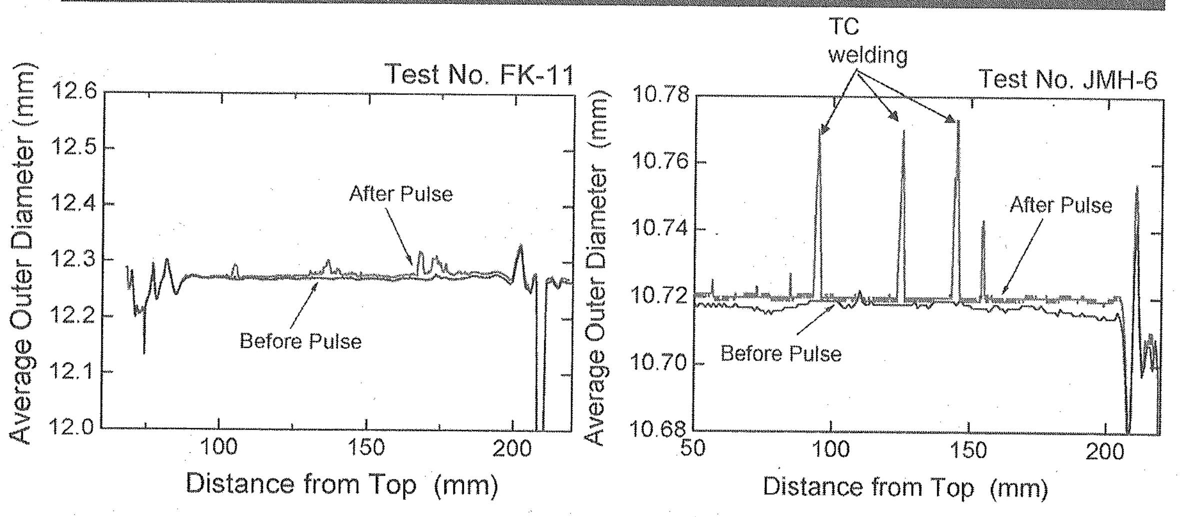
Rod diameter measurement before/after power oscillation test. (without DNB)



Almost no permanent deformation

12

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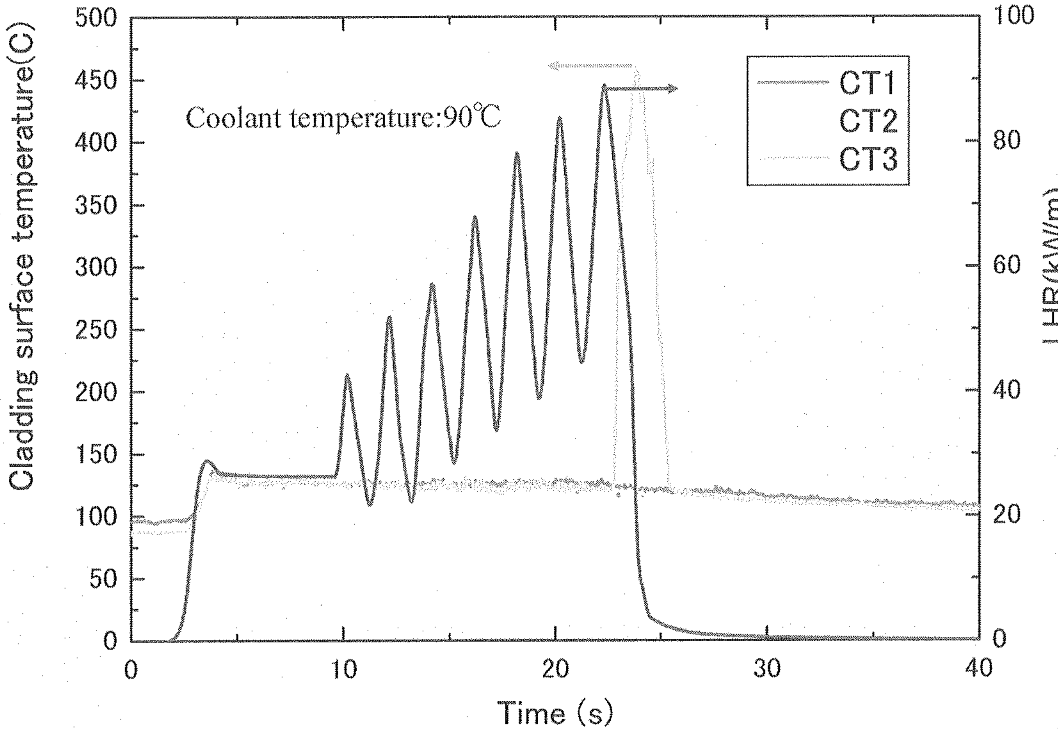
Cladding surface temperature and LHR of JMH-8 test



13

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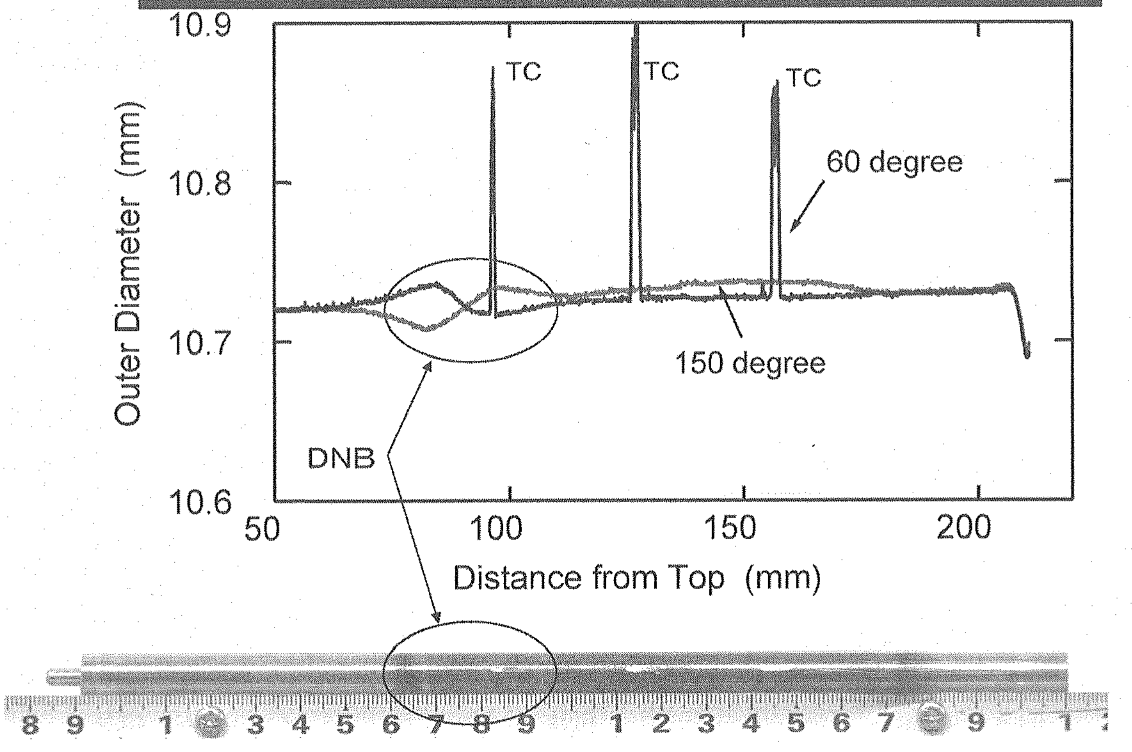
Rod profile and appearance of JMH-8 test rod



14

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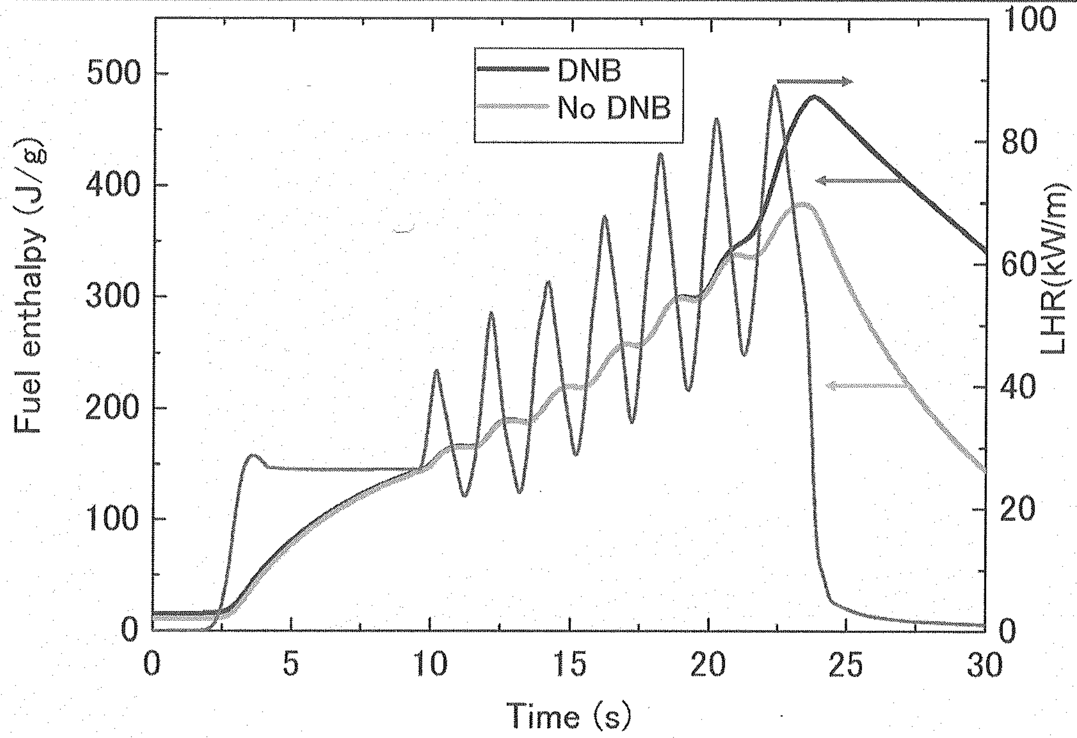
Fuel enthalpy and LHR of JMH-8 test (by FRAP-T6)



15

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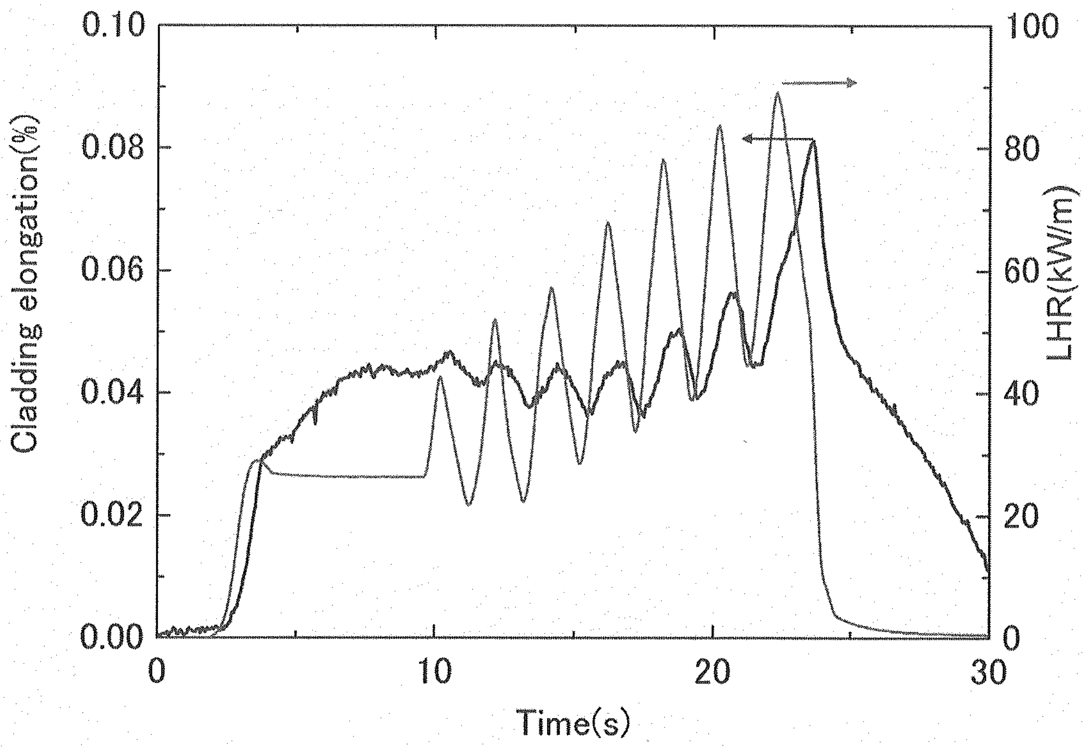
Cladding elongation and LHR of JMH-8 test



16

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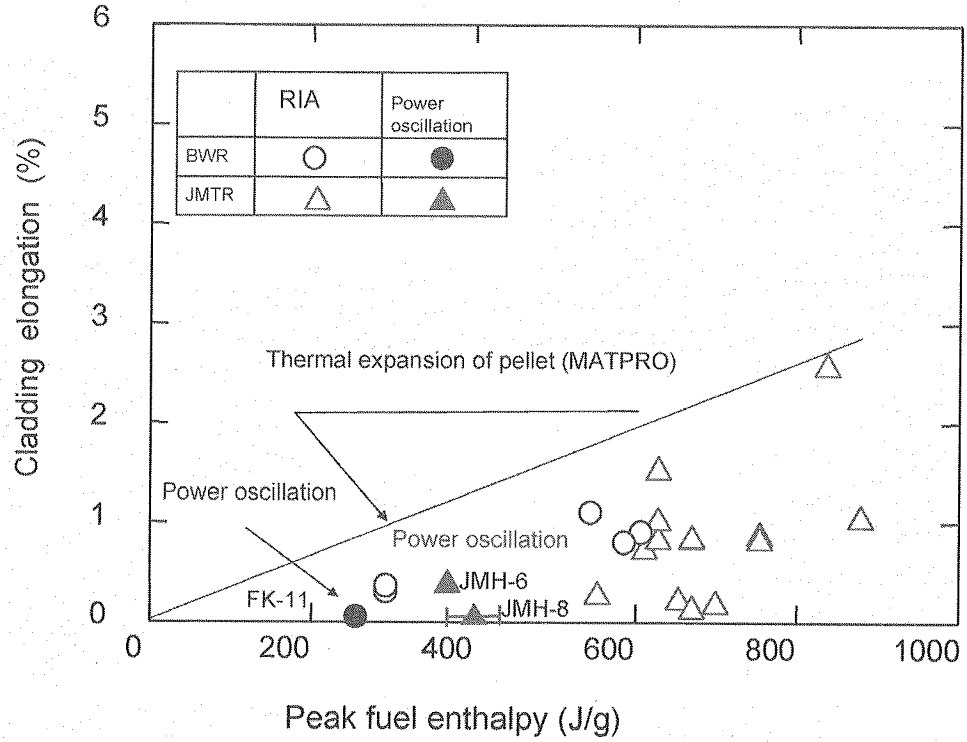
Cladding elongation in power oscillation tests and RIA tests



17

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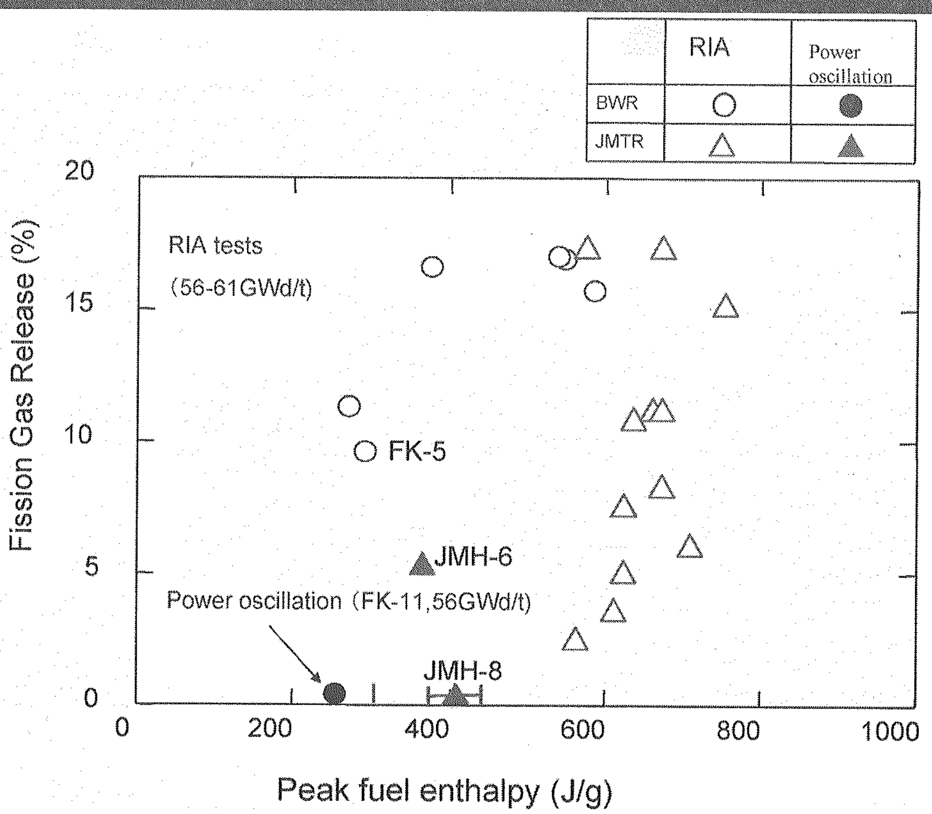
Fission gas release in power oscillation tests and RIA tests



18

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Fission gas release in power oscillation tests and RIA tests

	FK-11	FK-5 (RIA test)	JMH-6	JMH-8
Burnup(MWd/kg)	56	56	25	16
Fission gas release(%) during NSRR test	0.4	9.6	5.8	0.2
(base irradiation)	(12.5)	(12.5)		
Heat up rate (℃/s)	About 10^2	About 10^5	About 10^2	About 10^2
Calculated peak fuel enthalpy, J/g (cal/g)	256 (61)	293 (70)	368 (88)	~ 430 (~ 103)

19

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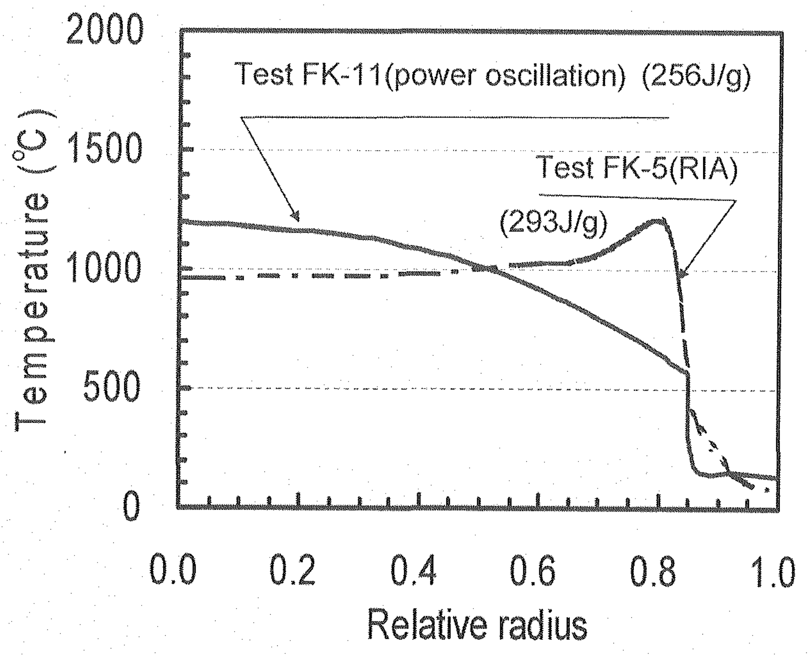
Comparison of radial temperature profiles in the power oscillation test FK-11 and in the RIA test FK-5.(at peak fuel enthalpy)



Difference of temperature profile : at periphery

⇩

Xe/Kr ratio is similar for both cases.

9.4 (FK-11) 10.2(FK-5)

The fission gas release of FK-11 test(0.4%) is comparable to that of diffusion release during power oscillation calculated with FASTGRASS code.

Difference of heatup rate could be responsible for the difference of fission gas release.

20

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Conclusions

- To examine fuel performance under power oscillation conditions in BWRs, three tests of irradiated fuels under simulated power oscillation conditions were conducted in the NSRR. DNB occurred only in one test. The fuel rod integrity was confirmed under the tested conditions without DNB, and the plastic deformation was very small.

- In case DNB occurred, small deformation of cladding was observed due to the temperature increase of cladding during film boiling.

- PCMI was observed by cladding elongation measurement in the tests. The cladding deformation was roughly proportional to estimated fuel enthalpy rather than to the power. The cladding deformation in the power oscillation conditions was comparable to those observed in the RIA tests at similar fuel enthalpy levels.

- The fission gas release was quite smaller in the power oscillation tests than in the RIA tests. The slower heatup rate during power oscillation could be one reason for the smaller release from the grain boundaries. The amount of the fission gas release is comparable to that of diffusion release from fuel grains during power oscillation calculated with FASTGRASS code.

21

Fuel Safety Research Meeting, March 2-3, 2005
Toshi Center Hotel, Tokyo, Japan

Session 3-2

Performance of Fuel and Cladding Material at High-Burnup

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Magnus Limbäck, Westinghouse Atom AB, SE-721 63 Västerås, Sweden

Abstract

A strategy to reduce the fuel cycle costs pursued by the Swiss utilities seeks minimization of the reload batch size by using advanced fuels capable of achieving high discharge burnups (BUs). Assemblies designed to reach BUs larger than 50 MWd/KgU have been operated in the Leibstadt Nuclear Power plant (KKL). Since 1994, continued operation for seven cycles allowed attainment of bundle average BUs above 60 MWd/kgU for this fuel.

The fuel was designed and fabricated by Westinghouse and adopts Zircaloy-2 cladding material (LK3). The irradiation conditions in the KKL reactor featured linear heat rates ranging from 250 W/cm down to 100 W/cm in the last cycle and normal water chemistry with zinc injection. Selected rods have been extracted at both intermediate and final irradiation stages and hot cell examinations have been carried out. Specifically, clad oxidation measurements using both Eddy current methods and destructive testing indicate that the clad corrosion is limited to ~ 50 microns. Furthermore detailed characterization of axial and radial nuclide distributions, rod profilometry as well as of the chemical composition of the fission products using both gamma measurements and second ion mass spectroscopy have been concluded. Results indicate little fission product migration and production of a tenacious crud with formation of two distinct phases. The preliminary assessment yields favourable features with respect to the possibility of operating this fuel until local BUs of 80 MWd/kgU.

In this contribution we bring attention to additional material characterizations focused on fuel/clad EPMA and SIMS analyses. Specifically, chemical burnup evaluations were carried out for three selected rods that operated for 7 cycles. These measurements yielded radial distributions of Uranium, Plutonium, Neodymium and Cesium-137 isotopic vectors along two directions across the rods, specifically, one following the maximum burnup gradient and the other along the perpendicular direction. The analysis of the cladding was centred on measurement of the hydrogen content and distribution, using the Hot Extraction method, which allows separation between hydrogen in the oxide layer and in the metal.

Selected rod segments have been provided to the ALPS programme for dedicated tests with regard to RIA and LOCA behaviour of such high burnup fuel. Quantification of the concentration of hydrogen in the cladding and the hydride precipitates morphology are provided for selected rods of interest for ALPS.

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PAUL SCHERRER INSTITUT

Fuel Safety Research Meeting 2005, Tokyo March 3rd, 2005

Performance of Fuel and Clad Material up to High Burnup

Status Report

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Kürzel, Datum

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Motivations and Objectives

- Assess Performance of SVEA-96 Fuel Assembly above Bundle average Burnups of 60 MWd/kgU
- Perform Comprehensive NDT and DT tests from hot-cell examinations for high burnup rods (up to 75 MWd/kgU and above the limit imposed by 5% U235 enrichment)
- Generate Experimental data for validation of Neutronic and Fuel Performance Codes (Actinides, Gadolinium, FGR, Fuel and Cladding Behaviour)
- Provide Fully Characterized Fuel and Clad Material for ALPS and other Safety Related Tests

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Fuel Irradiation at Leibstadt NPP

Reactor: GE BWR-6, 3600 MW thermal power with 648 Fuel Assemblies
(32 kW/kg, 62 kW/ l)

SVEA96+ Bundle from Westinghouse, 10x10 FA with four subbundles

Lead Test Assemblies (LTA):

Zy-2 Cladding (LK3, Fe+Cr+Ni ~ 3300 ppm); thickness~0.63 mm;
liner containing Fe~ 200 ppm;

Rod: D ~ 9.62 mm, H ~ 4152 mm (with 7 Spacers);

Start Irradiation in 1994 for 3, 5, 6, 7 cycles

Peak Rod Burnup ~ 73.5 MWd/kgU (Bundle ave. 60 MWd/kgU)
and LHR~ 250 W/cm

FA AGB108 with two-life Rods:

5 rods operated for 9 cycles up to a peak local BU ~92 MWd/kgU

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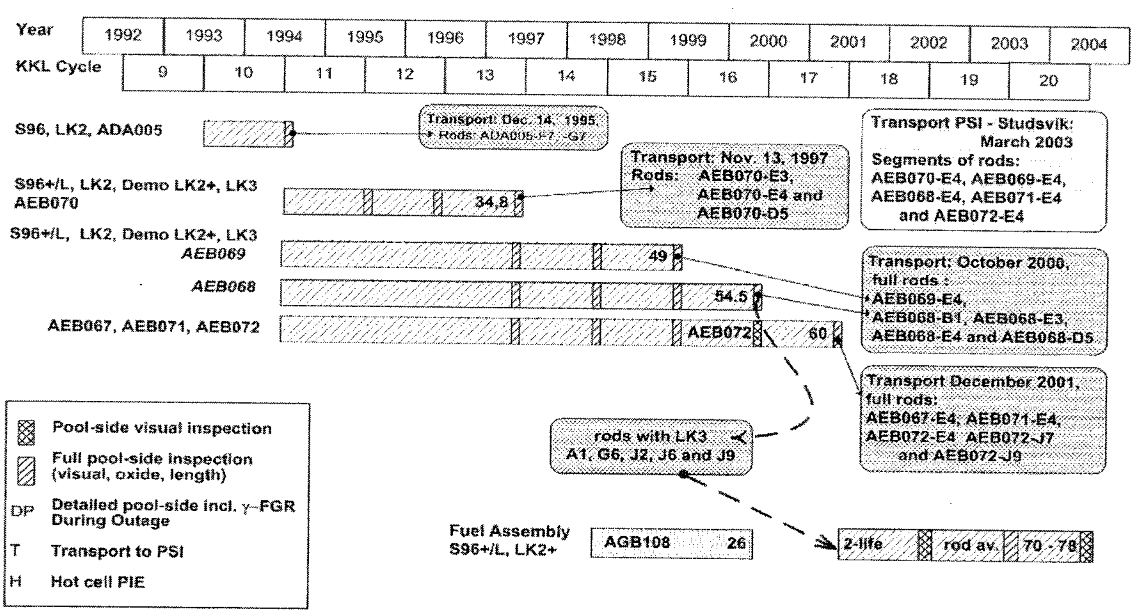
LTA Fuel Irradiation at Leibstadt NPP



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KKL Support to Fuel Test Programs

ALPS - LOCA and RIA Testing

Three Assembly Rods Provided from LTA:

AEB071-E4

AEB072-E4

AEB072-J9

Assembly Rod	Initial Enrichment [% U235]	Av. Rod / Local Burnup [MWd/kgU]	N. of Cycles in Core	Cladding Material	Unloaded Cycle - Year
AEB071-E4	4.46	57.6/67.8	7	LK3/L	17 - 2001
AEB072-E4	4.46	63.1/73.5	7	LK3/L	17 - 2001
AEB072-J9	3.71	63.1/72.3	7	LK3/L	17 - 2001

Other Programs:

Halden Experiments:

LOCA (IFA-650)

FGR (IFA-629)

Cladding Lift-off (IFA-610)

CABRI

Test of BWR Fuel in PROMETRA (proposal of Swiss partnership, material in Studsvik)

SCIP (Studsvik Cladding Integrity Program)

Segments for operational transients and RIA simulation tests on LK3 material

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Fuel/Clad Characterization of Rods for ALPS

(WA-KKL-PSI Fuel Performance Programme)

❖ *Non-Destructive Tests (Completed)*

Visual Inspection, Length Measurement and Profilometry and γ Spectrometry

Oxide Thickness / Crud Measurements

Fission Gas Release (Rod Free Volume, Composition and Release)

❖ *Destructive Tests (nearly Completed)*

Cladding Characterization (Metallography, Hydrogen Content, SPP, micro-hardness) on all three rods

Fuel Characterization (Chemical Burnup, ceramography, density, distribution of selected fission products (EPMA), distribution of selected isotopes (SIMS)) on AEB072-E4

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Overall Characterization at PSI

(WA-KKL-PSI Fuel Performance Programme)

Fuel Characterization (AEB072-E4 Rod):

- Well Characterized Fuel Profilometry;
- Nearly no Migration of Volatile Fission Products (from Radial Profiles of Plutonium and FPs at high BU for);
- Cs Precipitation in HB region;
- Over 100 MWd/kgU local BU in Fuel Rim

Clad Characterization (all three rods)

- Low Clad Oxidation, below 30 microns, eddy current measurements in the lower part of the rod deteriorated ;
- CRUD layer < 5 microns, composition hematite and spinel;
- Hydrogen Content and morphology analyses completed for E4 and J9 rods

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Fuel Characterization (AEB072-E4 rod)

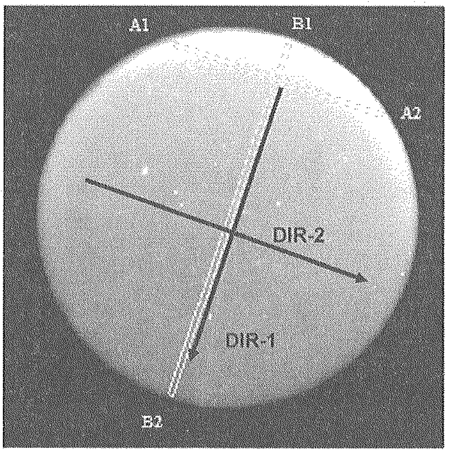
EPMA Analyses (Both Qualitative & Quantitative)

SIMS Measurements (Isotopic Composition)

Two Fuel Directions Considered

(Dir - 1 = Maximum BU Gradient;

Dir - 2 Perpendicular to Dir. 1)



Radiograph Image of AEB072-E4

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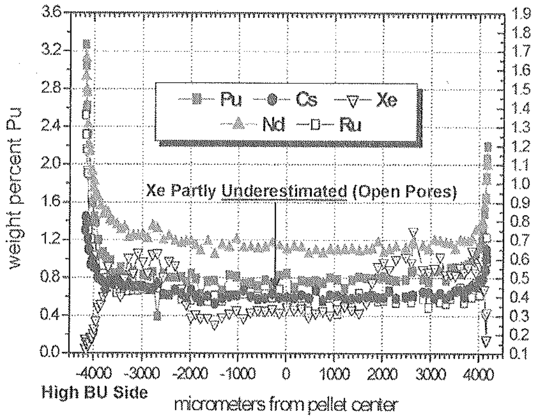
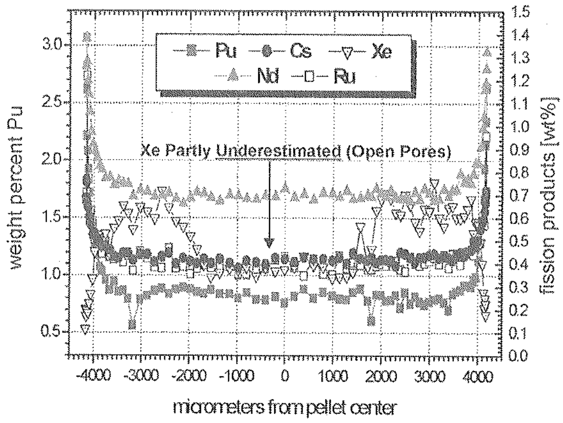
Quantitative EPMA (AEB072-E4 Rod)

Radial Profiles, Pu and FPs (AEB072-E4)





Dir 1: Highest BU Gradient **Dir 2: Perpendicular to Dir.1**

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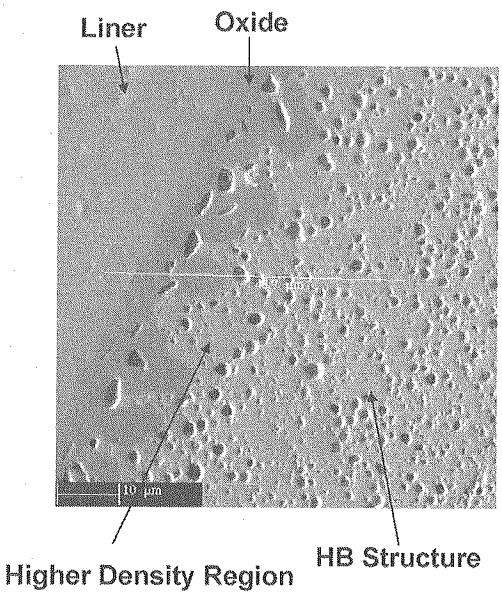
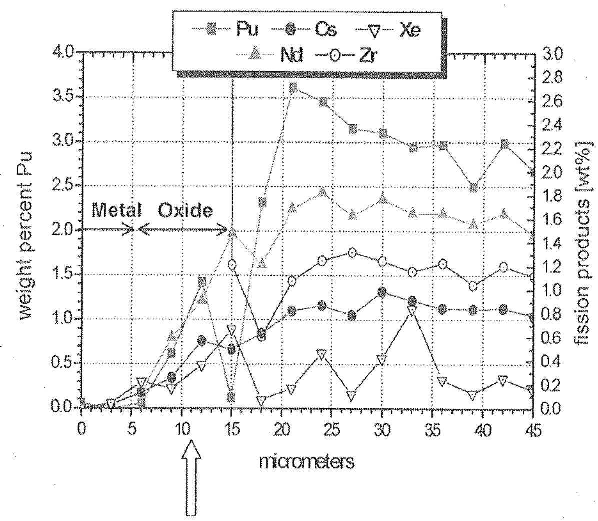
Fuel Rim Examination (AEB072-E4 Rod)



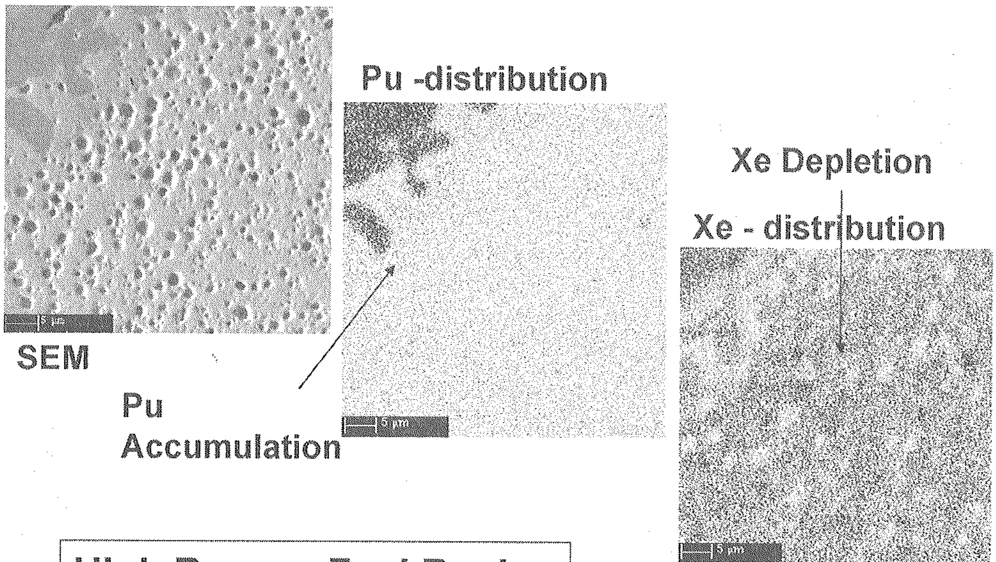


FPs Clad Penetration due to Recoil

Kurzel, Datum

PAUL SCHERRER INSTITUT

Qualitative EPMA (AEB072-E4 Rod)



High Burnup Fuel Region

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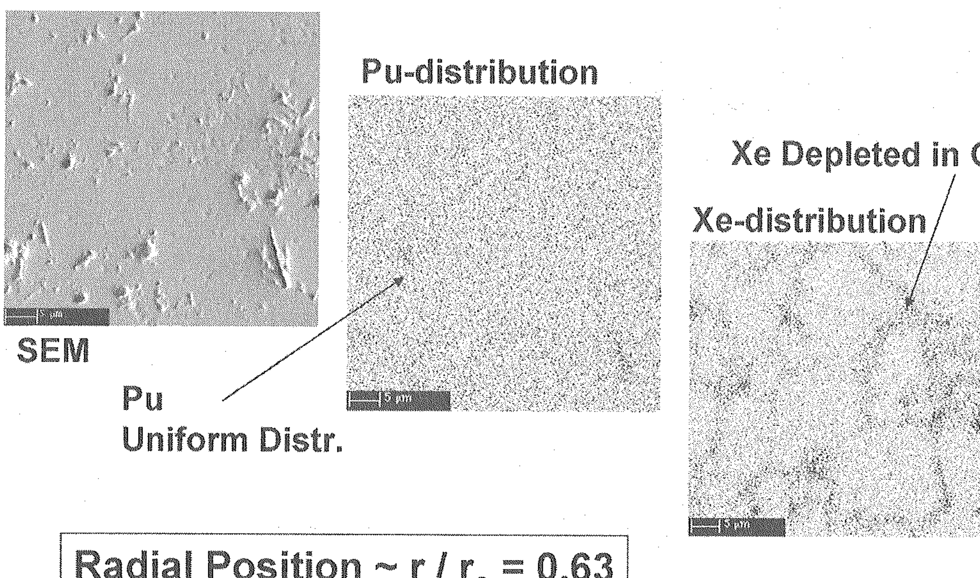
Qualitative EPMA (AEB072-E4 Rod)



Radial Position ~ $r / r_o = 0.63$

Kürzel, Datum

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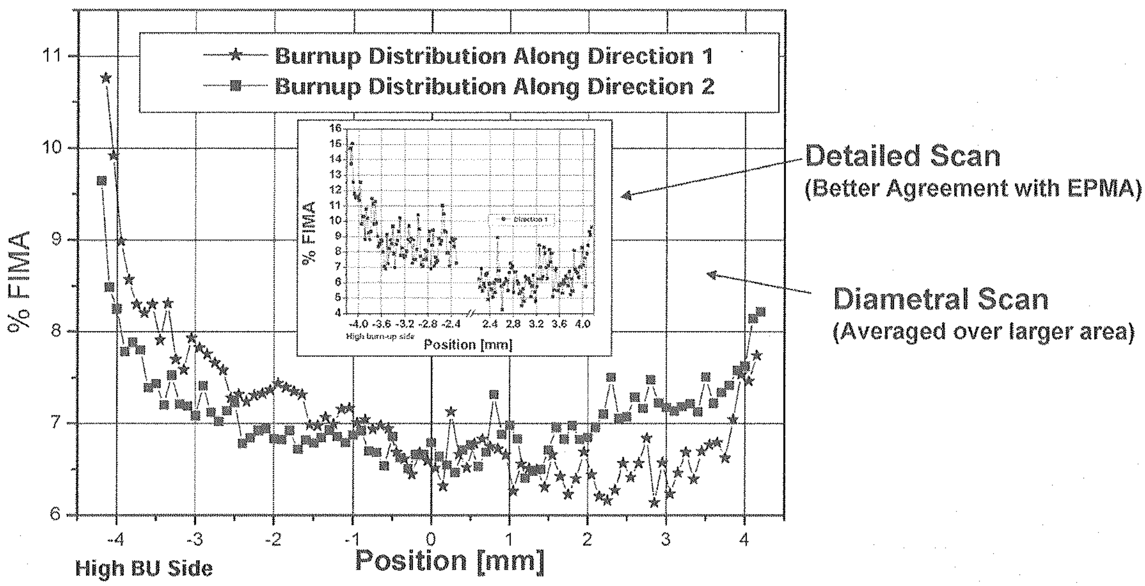
SIMS Analyses: BU & Isotopic U, Pu and FPs Distributions (AEB072-E4)



Kürzel, Datum

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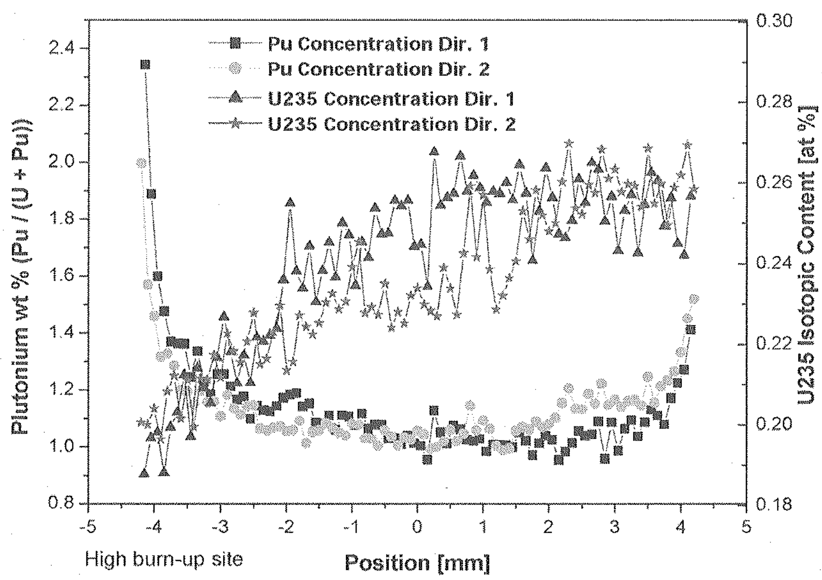
SIMS Analyses: U, Pu Distributions (AEB072-E4)



Kürzel, Datum

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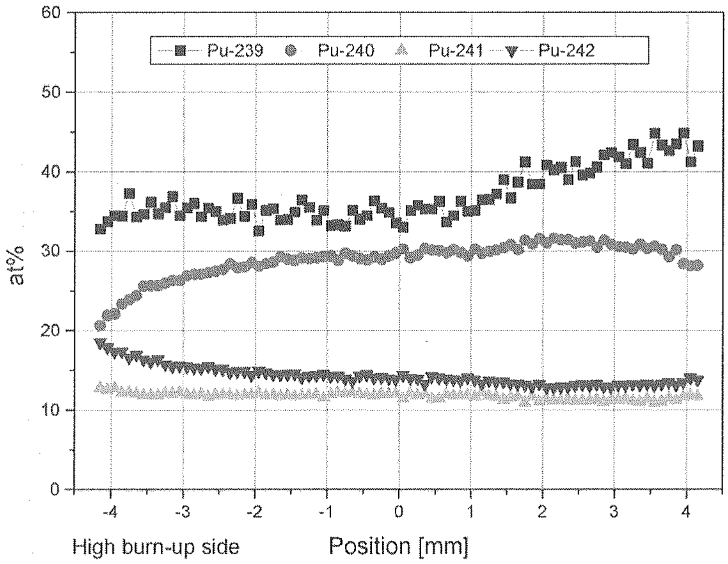
SIMS Analyses: Isotopic Pu Distribution (AEB072-E4)



Kürzel, Datum

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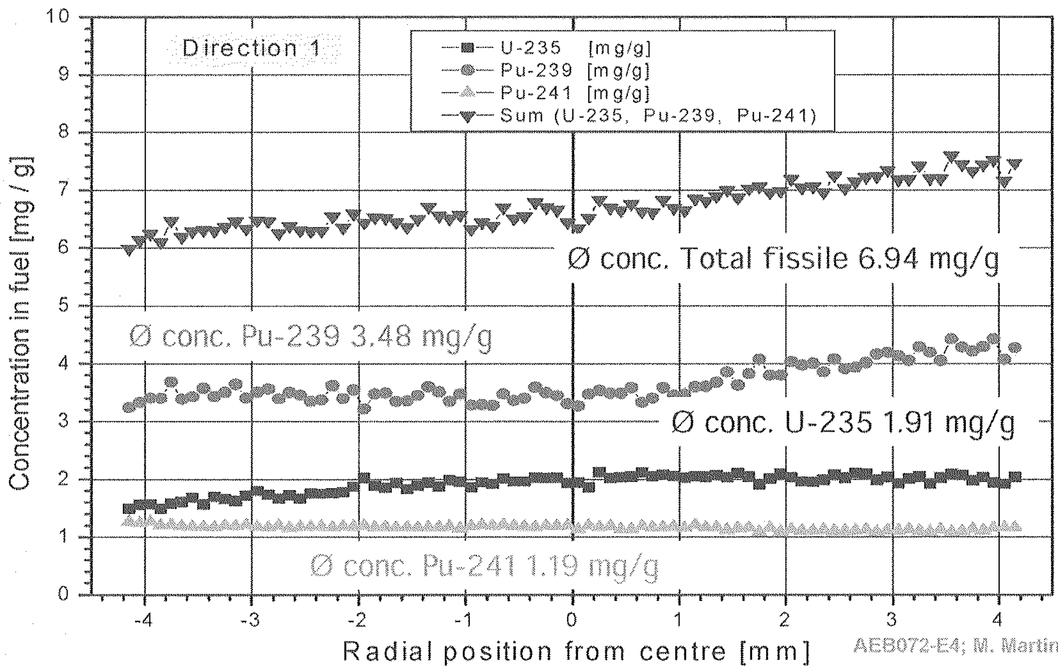
SIMS and burnup Analyses: Distribution of fiss. material (AEB072-E4)



AEB072-E4; M. Martin et al. PSI

Kurzel, Datum

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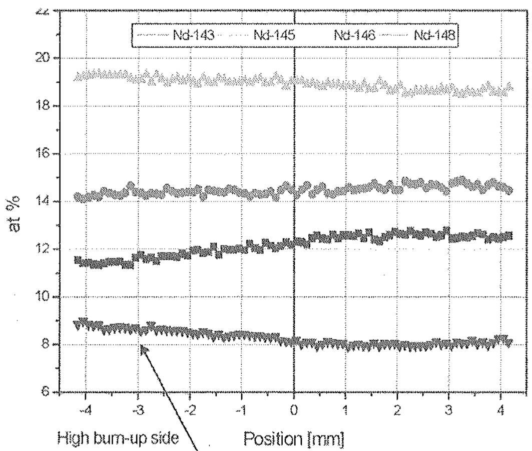
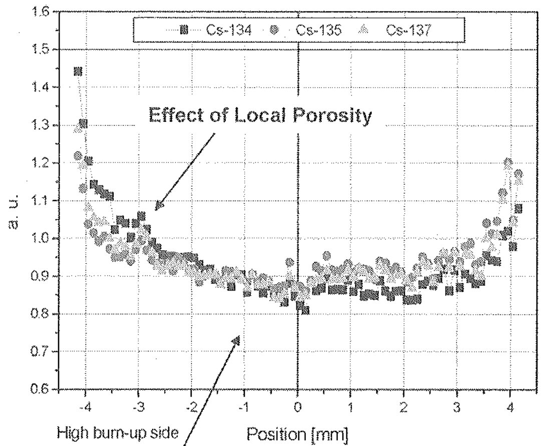
SIMS Analyses: Selected FPs Distributions (AEB072-E4)





Follows BU Gradient

No Indication of Cs Diffusion

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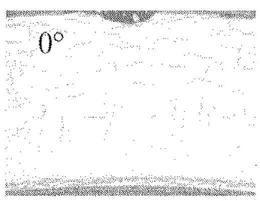
Clad Characterization: Hydrides Morphology

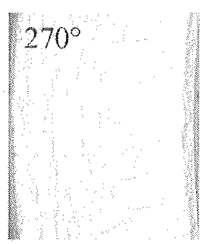
Hydride etched
Rod AEB072-J9
Segment GB

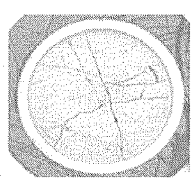


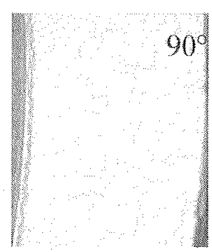
Oxide thickness: 20 μm

H_2 in metal: 160 [ppm]

H_2 in metal: 195 [ppm]





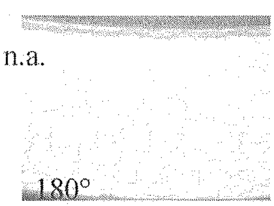


H_2 in metal: 195 [ppm]

Oxide thickness: n.a.

Oxide thickness: 24 μm

H_2 in metal: 205 [ppm]



Oxide thickness: 10 μm

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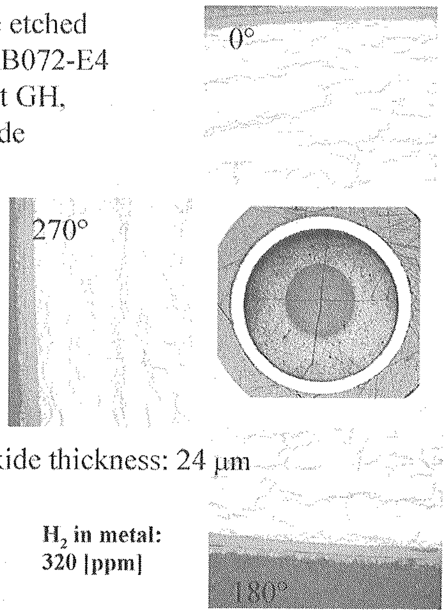
Clad Characterization: Hydrides Morphology

Hydride etched
Rod AEB072-E4
Segment GH,
water side



Oxide thickness: 18 μm

H_2 in metal:
260 [ppm]

H_2 in metal:
300 [ppm]

H_2 in metal:
290 [ppm]

Oxide thickness: 24 μm

Oxide thickness: 18 μm

H_2 in metal:
320 [ppm]

Oxide thickness: 20 μm

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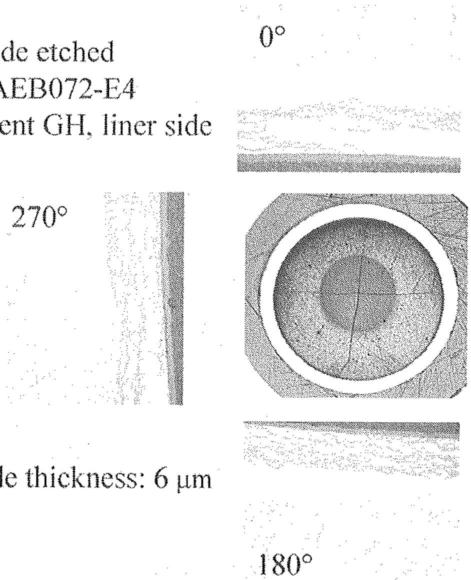
Clad Characterization: Hydrides Morphology

Oxide thickness: 8 μm

Hydride etched
Rod AEB072-E4
Segment GH, liner side



Oxide thickness: 6 μm

Oxide thickness: 7 μm

Oxide thickness: 6 μm

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Conclusions

- KKL is participating as key fuel deliverer to several international fuel testing programmes
- Fuel Performance Programme: provides material property data for fuel enriched up to 5%
- ALPS Fuel: operated up to 72 MWd/kg in KKL, shows very good behaviour (limited PCI uniform isotopic distributions, well defined rim region)
- Comprehensive characterization of candidate fuel for ALPS: AEB072-E4
- Characterization of other two rods focused on Clad only
- W.r.t. other rods more data available for AEB072-E4 thus more proficient utilization for experiments in the perspective of computational modeling

Kürzel, Datum

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Future Work

- Neutonic modelling work with regards to uranium, plutonium and neodymium isotopic composition with Helios / Presto-2 started (NOK / KKL)
- Modelling of thermal-mechanical fuel response on selected RIA & LOCA tests foreseen (use of FALCON, FALCON/TRACE integration for LOCAs)

Kurzel, Datum

Session 3-3

Grain subdivision and strain distribution between subdivided grains in high burnup UO_2 pellet

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It is well-known that microstructural changes called "rim structure" occur at high burnup region in LWR fuel pellet, such as pellet periphery. Rim structure is characterized by increase of porosity and formation of sub-divided grain and coarsened bubble. Since the rim structure has special features like this, it is considered that rim zone, at which rim structure is formed, is possible source of fission gas release during normal or transient condition. Therefore, it is important to grasp the threshold burnup of rim structure formation and formation mechanism for safety of high burnup fuel.

Samples were prepared from fuel pellets of STEP-II type-segmented fuel rod irradiated in a commercial BWR in order to investigate microstructural changes in UO_2 pellet. Pellet-averaged burnups are about 60GWd/t. For cross section of these irradiated fuel pellets, Optical microscope observations and micro-X-ray diffractions were carried out. From the obtained ceramographs, zone widths of rim structure formation were evaluated in each sample. Based on the micro-x-ray diffraction results, lattice parameters and diffraction peak broadenings (increase of FWHM of diffraction peak) of irradiated UO_2 pellets were measured. Crystallite diameter, which corresponds to subdivided grain diameter, and strain distribution between crystallites can be evaluated from the increase of FWHM of diffraction peak by using Willamson-Hall method.

While lattice parameters of samples increase in burnup region up to about 70GWd/t, the lattice parameters fall and tend to be leveled off above this burnup due to rim structure formation. Accordingly, the threshold burnup of rim structure formation would be about 70 GWd/t. Crystallite diameter gradually decreases with increasing burnup up to about 67 GWd/t and the diameter tends to saturate at 100 - 200 nm above this burnup. The saturated value is comparable to that of《 recrystallized grain' in rim structure. This tendency can be explained by dislocation behavior in UO_2 pellet during irradiation.

This work was operated by one of new-cross-over projects conducted under the Atomic Energy Committee of Japan and was sponsored by Ministry of Education, Culture, Sports, Science and Technology.

Grain subdivision and strain distribution between subdivided grains in high burnup UO_2 pellet

M. Amaya

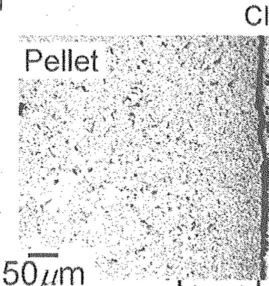
Fuel Safety Research Lab., JAERI

Fuel Safety Research Meeting 2005
March 2-3, 2005
Tokyo, Japan

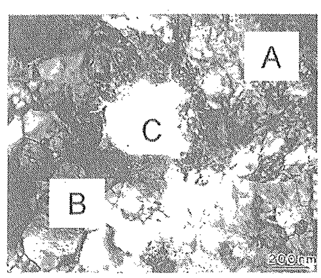
1

Background

- Burnup extension of LWR fuel
- Microstructural changes at periphery of fuel pellet due to local burnup increase ("rim structure formation")



- Characteristics of rim structure
  - (1)Increase of porosity
  - (2)Formation of
    - ·Sub-divided grain ("Recrystallized grain")
    - ·Coarsened bubble



A: defect-clusters-accumulated region
B: recrystallized grain
C: coarsened bubble

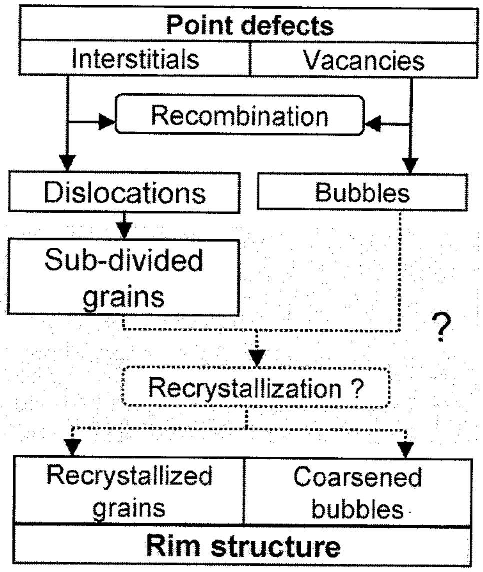
(Nogita et al.) 200 nm

TEM image of rim structure

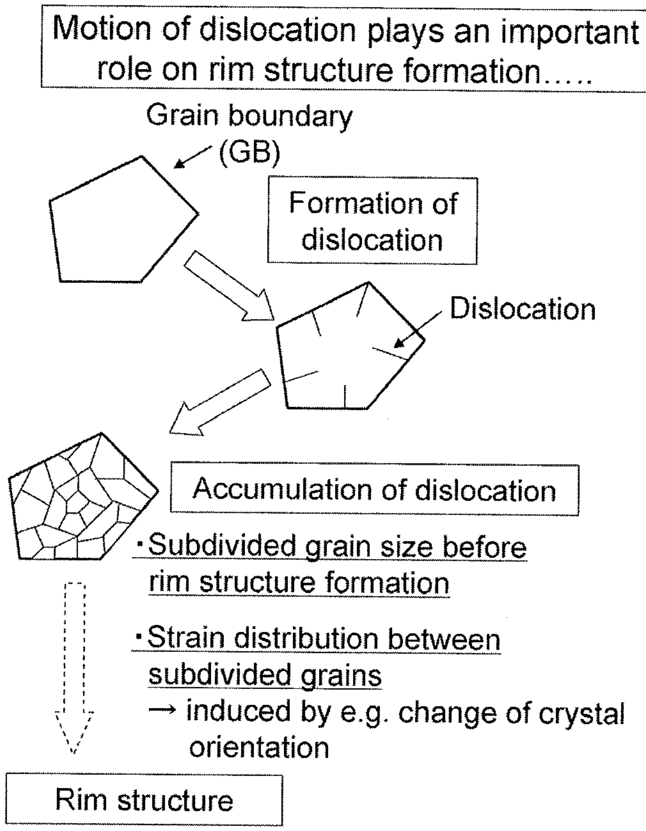
·Possible release source of fission gas during normal/ transient condition.
→ For safety of high burnup fuel, it is important to grasp the threshold burnup of rim structure formation and formation mechanism

2

Previous studies concerning rim structure formation



Mechanism proposed by Nogita et al.



3

Objectives

- To evaluate the threshold burnup of rim structure formation in UO_2 pellet.
- To investigate burnup dependence of grain subdivision and strain distribution between subdivided grains, from the view point of formation mechanism of rim structure.

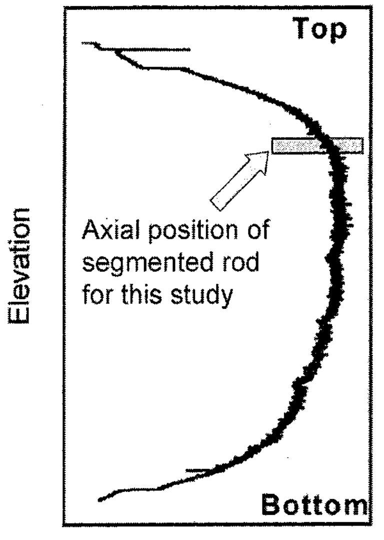
4

Sample characteristics

Sample ID	1	2
Fuel type	BWR-STEP II (high burnup 8x8)	
Pellet density (nominal, %TD)	97	
Irradiation cycle*	4	5
Pellet-averaged burnup (GWd/t)	56	61
FGR (%)	12.5	12
Peak linear heat rate during irradiation (kW/m)	35	

* irradiated in a commercial BWR.

Gammma-ray intensity



Burnup profile of fuel rod.

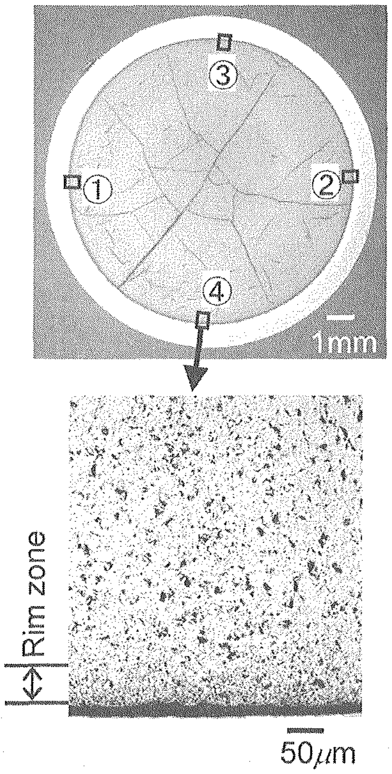
5

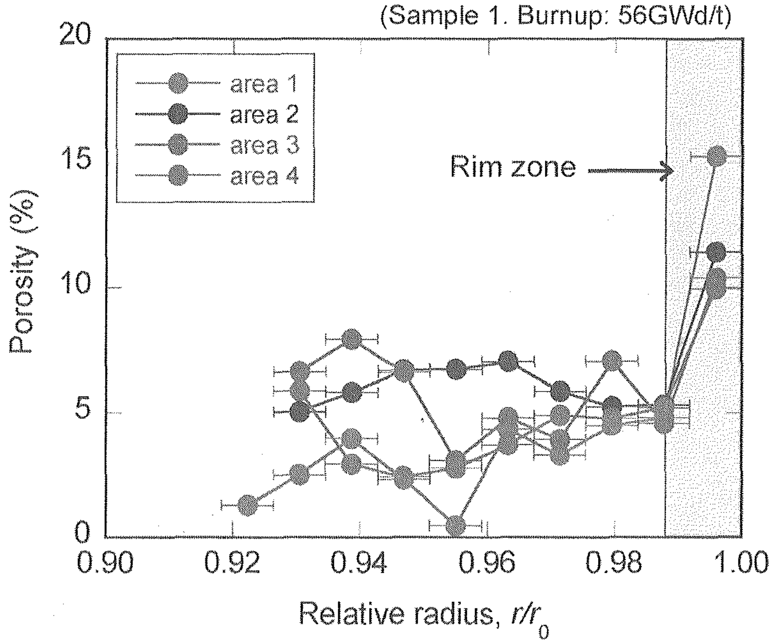
Experimental

- For cross section of irradiated fuel pellets,
  - Optical microscope observation
    → evaluation of rim zone width
  - Micro-X-ray diffractometry
    1) Lattice parameter change
    → evaluation of uniform strain in crystal lattice of irradiated UO_2 and threshold burnup of rim structure formation.
    2) Increase of FWHM (Full Width at Half Maximum) of diffraction peak
    → evaluation of crystallite (subdivided grain) diameter and strain distribution between crystallites

6

Radial porosity profile of irradiated UO_2 pellet





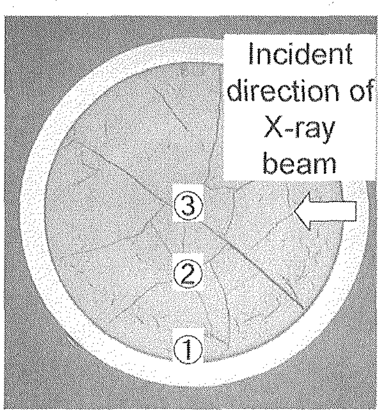
Radial porosity profiles at peripheral region

Width of rim zone: ~50 μm (sample 1)
~ 200 μm (sample 2)

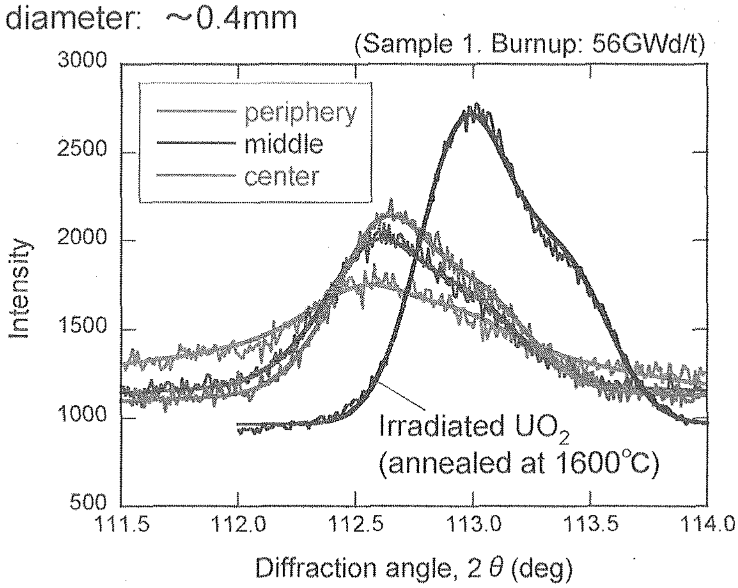
7

X-ray diffraction results of irradiated UO_2 pellet

- Radial position: periphery, middle and center of pellet
- Incident X-ray beam diameter: ~0.4mm



position



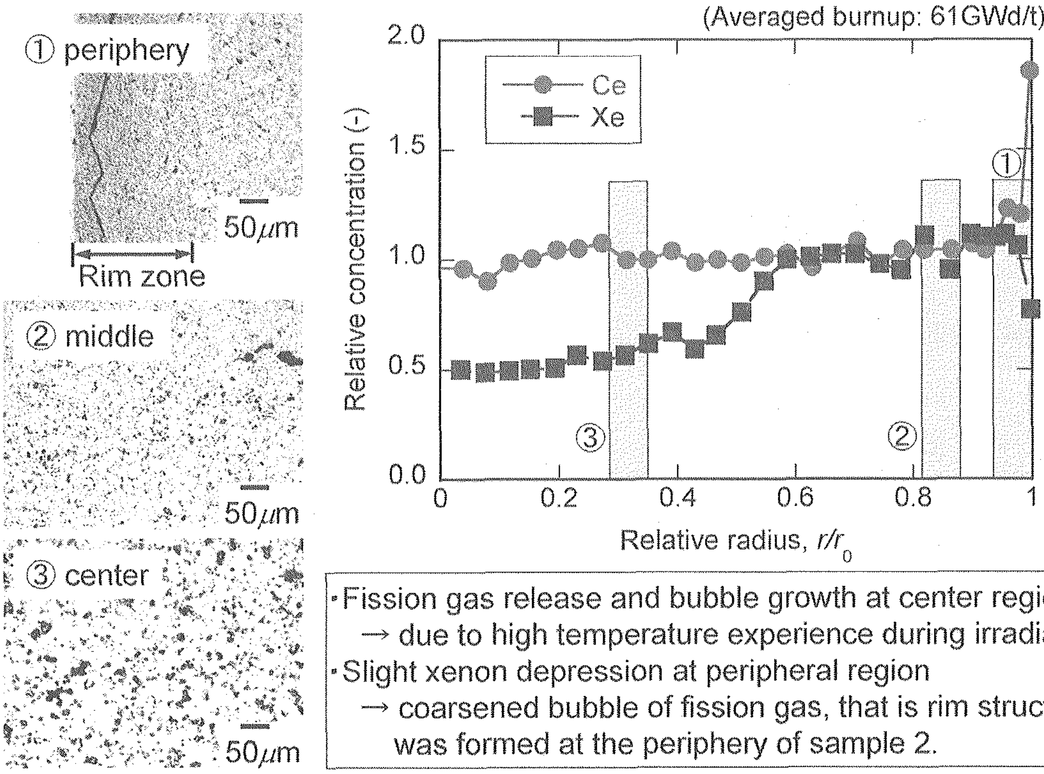
Comparison of diffraction peaks of (531) plane

·Peak broadening → subdivision of grain and/or strain distribution between grains
·Peak shift to lower angle side→ crystal lattice dilatation

8

Radial profiles of burnup and fission gas concentration

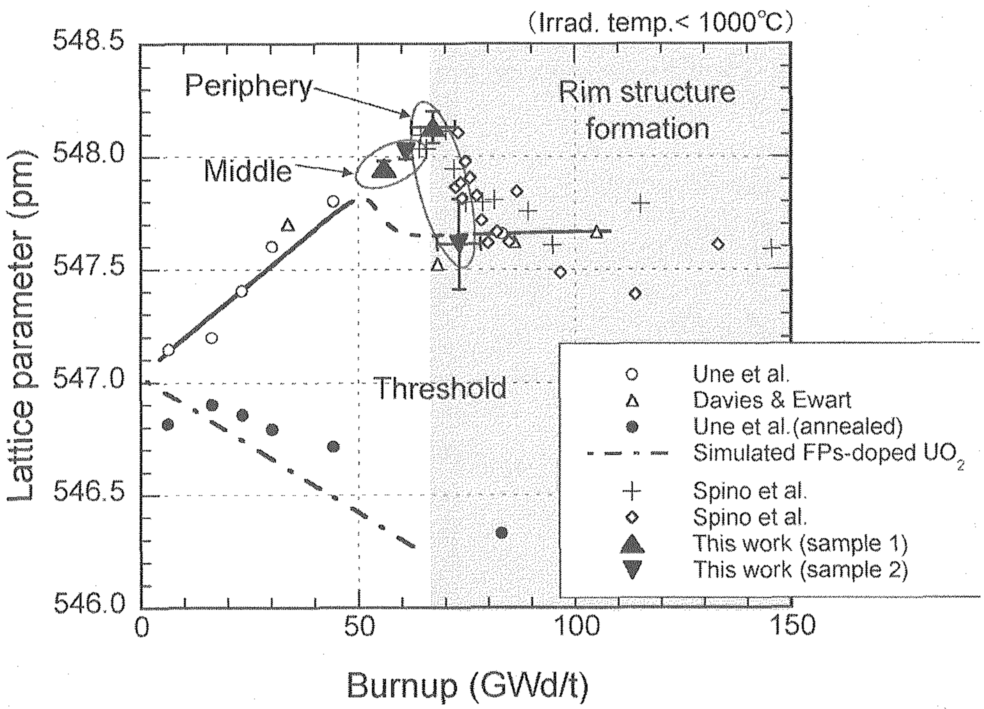
●Radial profiles of burnup(Ce) and fission gas concentration (Xe) in sample 2



- Fission gas release and bubble growth at center region
  → due to high temperature experience during irradiation.
- Slight xenon depression at peripheral region
  → coarsened bubble of fission gas, that is rim structure, was formed at the periphery of sample 2.

9

Lattice parameter in irradiated UO_2 pellet



- Lattice parameter falls at ~70GWd/t and saturates due to rim structure formation.
  → Threshold burnup of rim structure formation would be ~ 70 GWd/t.

10

Crystallite size and strain distribution between crystallite

● Williamson-Hall method

・Strain distribution between crystallites and crystallite diameter are evaluated from FWHM increase of diffraction peaks.

Measured FWHM increase can be expressed by the sum of following terms:

$$\beta = \frac{k\lambda}{D_X \cdot \cos\theta} + 2\eta \tan\theta$$

Effect of crystallite size (Scherrer's formula)

Effect of strain distribution

β : Measured FWHM increase of peak (corrected the peculiar broadening caused by X-ray diffraction system)
k : Constant (=0.9)
θ : Diffraction angle
D_x : Crystallite (=subdivided grain) diameter
η : Strain distribution between crystallites
λ : X-ray wave length for diffraction

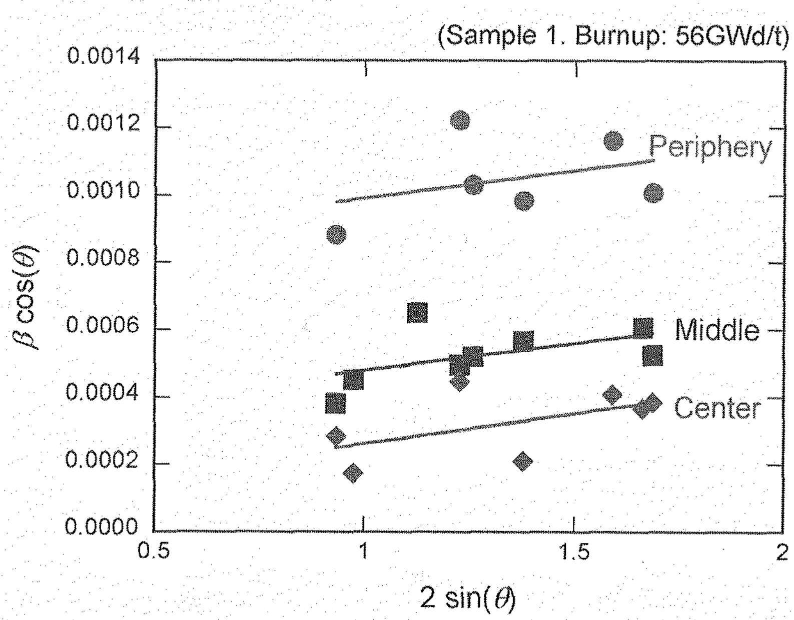
$$\beta \cos\theta = \frac{k\lambda}{D_X} + 2\eta \sin\theta$$

From plots of $\beta\cos\theta$ against 2 $\sin\theta$,

・slope → strain distribution (η)
・intercept → crystallite diameter (D_x)
can be evaluated.

11

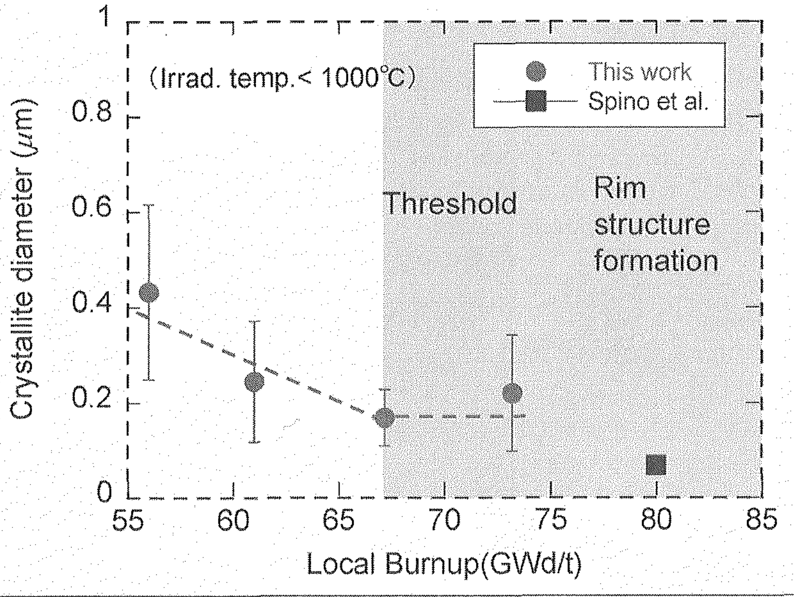
Relationship between $\beta\cos\theta$ and 2 $\sin\theta$



・Linear relationship is observed.
→ strain distribution (η) and crystallite size (D_x) are evaluated.

12

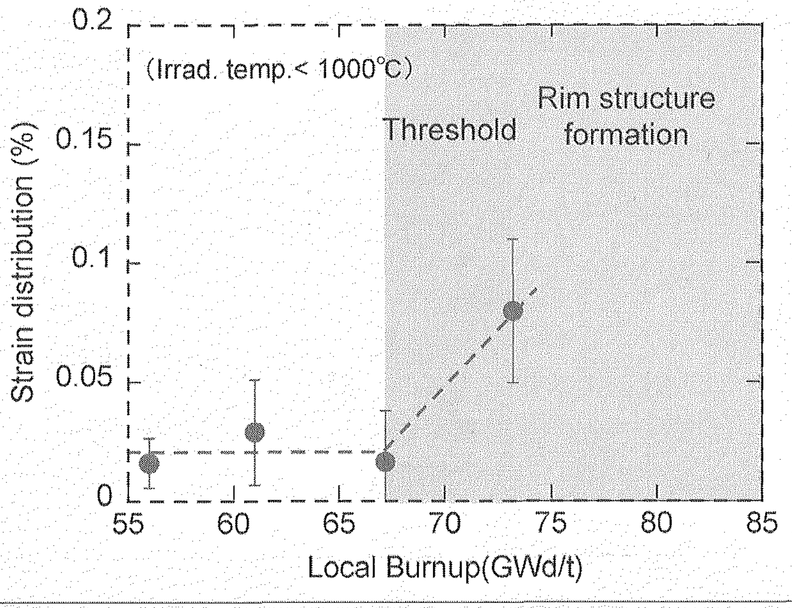
Burnup dependence of crystallite diameter



- Crystallite diameter tends to decrease with increasing burnup up to ~67 GWd/t.
- Crystallite diameter saturates in burnup region above ~67 GWd/t and the saturated values lies at ~200 nm.
  → comparable to the diameter of "recrystallized grain" in rim structure.
- Dislocation formed by irradiation moves to grain boundary and accumulates.
  → Grain may be divided into subdivided grains from grain boundary.

13

Burnup dependence of strain distribution between crystallites

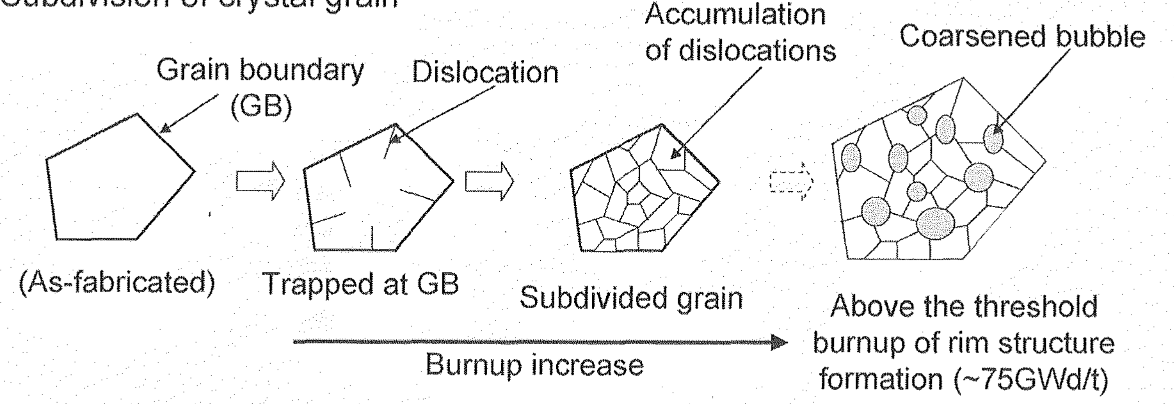


- Strain distribution between crystallites is hardly dependent on burnup up to ~67GWd/t and tends to increase in burnup region above ~67GWd/t.
- Strain distribution is related to crystal orientation of each crystallite.
  → Accumulated dislocations would change crystal orientation of crystallites and the strain distribution may be induced by the orientation change between crystallites.

14

Formation mechanism of rim structure

● Subdivision of crystal grain



Formation of dislocation ↓ Grain subdivision → Generation of strain distribution (e.g. crystal orientation change) → Coarsened bubble formation? (induced by strain distribution?)

● Motion of dislocation in grain would affect rim structure formation.

- reduction of trapping site (e.g. grain boundary) of dislocation,
- pinning of dislocation motion in grain (precipitates, etc.),

may be effective to suppress rim structure formation.

15

Conclusions

● Crystallite sizes and strain distributions between crystallites were evaluated from the broadenings of X-ray diffraction peaks of irradiated UO_2.

● Threshold burnup of rim structure formation would be ~70 GWd/t.

● Crystallite size decreases with increasing burnup up to 65GWd/t and tends to saturate in burnup region of rim structure formation. On the other hand, strain distribution between crystallites is hardly dependent on burnup up to 70GWd/t and tends to increase in burnup region of rim structure formation.

● It is suggested that motion of dislocation in grain affects rim structure formation. Reduction of trapping site of dislocation and pinning of dislocation motion may be effective to suppress rim structure formation.

16

Session 3-4

FRACTURE TOUGHNESS TEST OF UNIRRADIATED PWR CLADDING

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Fracture toughness is an effective index for resistance against crack propagation in brittle materials. JAERI has developed a test method to measure fracture toughness of high burnup fuel cladding in a cooperative research between JAERI and Nuclear Fuel Industreis, ltd. In this study, the fracture toughness test was applied to unirradiated PWR claddings with excess hydrogen concentrations up to 800ppm for the temperature range up to 400°C. The fracture toughness was evaluated as a J-integral value obtained from a load-displacement curve of the test sample.

Fracture toughness of the highly hydrided claddings decreases with the hydrogen concentration increase at temperatures below 100°C, while it is nearly constant irrespective of hydrogen concentration at above 100°C. It is considered that the higher ductility of Zircaloy matrix controls the fracture toughness of hydrided cladding at higher temperatures. In addition fracture toughness increases linearly with displacement at maximum load irrespective of hydrogen content. Consequently, these two results indicate that the fracture toughness strongly depends on the plastic deformation at the crack tip.

Fracture Toughness Test of Unirradiated PWR cladding

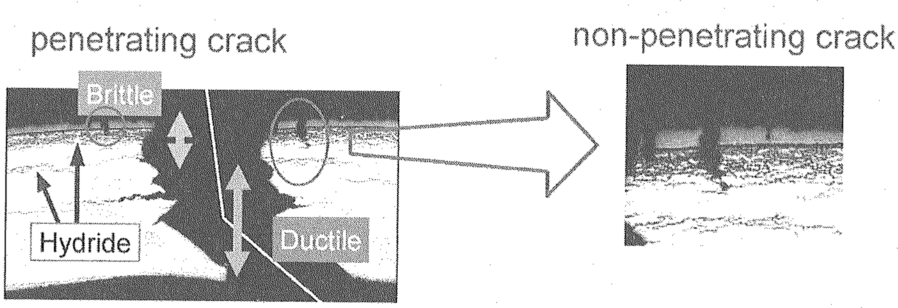
N. Ikatsu

Japan Atomic Energy Research Institute

Fuel Safety Research Meeting 2005
March 3, 2005, Tokyo

1

Background



One of sevaral cracks of outer surface is propagating into claddings. The resistance against crack propagation is important properties for estimation of cladding integrity.

⇩

Fracture toughness is effective indicator for resistance to cracking.

In this study, the fracture toughness of unirradiated cladding is measured in order to evaluate the influence of hydride precipitation.

2

TEST PROCEDURE

Fracture Toughness is evaluated with J-integral

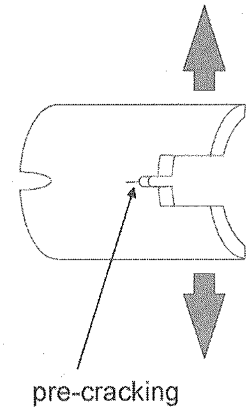
PWR cladding with hydrogen absorbed
hydrogen content (20, 200, 400, 800ppm)

Cladding was cut off as CT-shape specimen

Pre-cracking by fatigue

Fracture toughness test
temperature (20, 50, 70, 100, 150, 200, 300, 400 °C)
All pair of Hydrogen content and test temperature
Measurement of Load and Displacement

Evaluation of J-integral by Load and Displacement



3

Calculation of Fracture Toughness

calculation of J-Integral
ASTM-E1820-01

$J_{max} = J_{el} + J_{pl}$

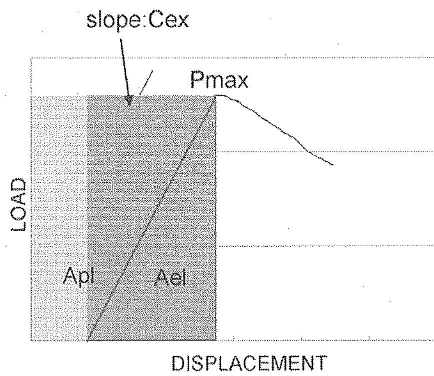
$J_{el} = K_i^2 (1 - \nu^2) / E$

$J_{pl} = \eta \cdot A_{pl} / (B_N \cdot b_0)$

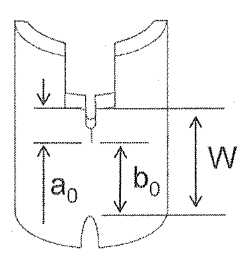
$K_i = P_{max} \cdot f(a_0/W) / (B_N \cdot W)^{0.5}$

$A_{pl} = A_{total} - A_{el}$

$A_{el} = (P_{max})^2 / (2 \cdot C_{ex})$



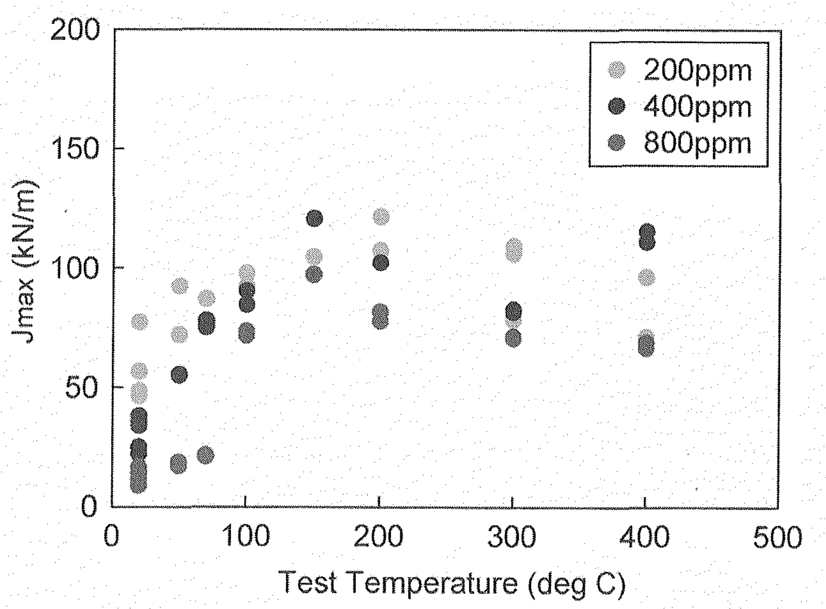
K_i: stress intensity factor
ν: Poisson ratio
E: young module
B_N: thickness of cladding
b_0: ligament length
$f(a_0/W)$: function of sample dimension



In this study, J-integral was calculated as J_{max} at the maximum load.

4

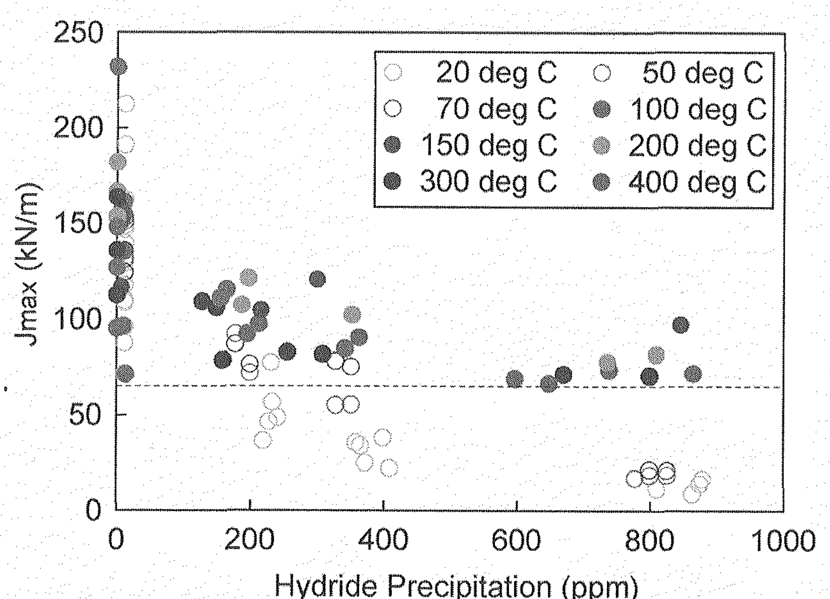
Temperature dependence of Jmax



Jmax becomes larger with higher temperature up to 150°C.
Jmax is leveled off with higher temperature over 200°C.

5

Hydrogen content dependence of Jmax



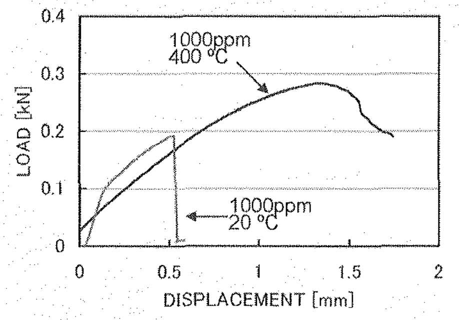
Jmax becomes smaller with higher hydride precipitation (< 70°C)
Jmax is leveled off with higher hydride precipitation (>100°C)

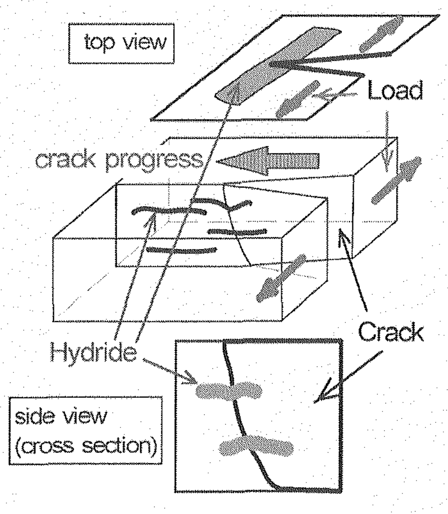
hydride precipitation was estimated from hydrogen content and theoretical solid solubility limit 6

Discussion

High hydrogen content specimen

- at low temperature
  displacement at max. load is small
- at high temperature
  displacement at max. load is large





Hydride: low ductility

Matrix:

high ductility at high temperature
(matrix endure by plastic deformation)

low ductility at low temperature
(matrix easily crack)

$J_{max} = J_{el} + J_{pl}$

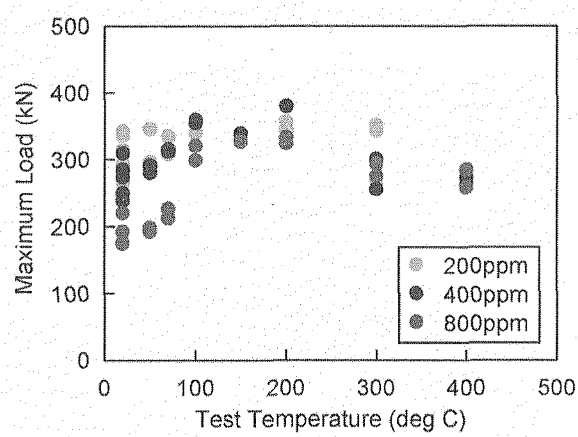
$J_{el} \propto P_{max}^2$

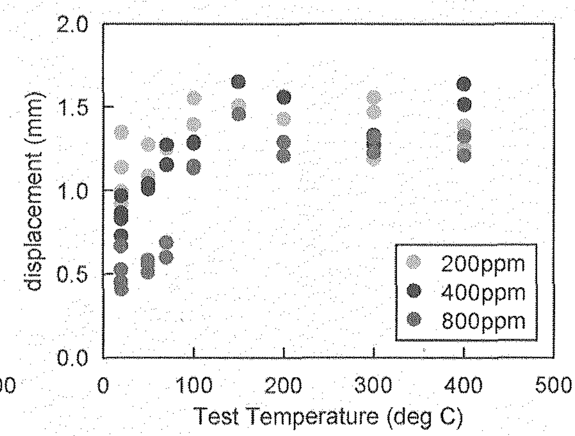
$J_{pl} \propto A_{pl}\ (= A_{total} - A_{el})$

$A_{total} \propto \text{Displacement} \times P_{max}$

7

Temperature dependence of Load and Displacement



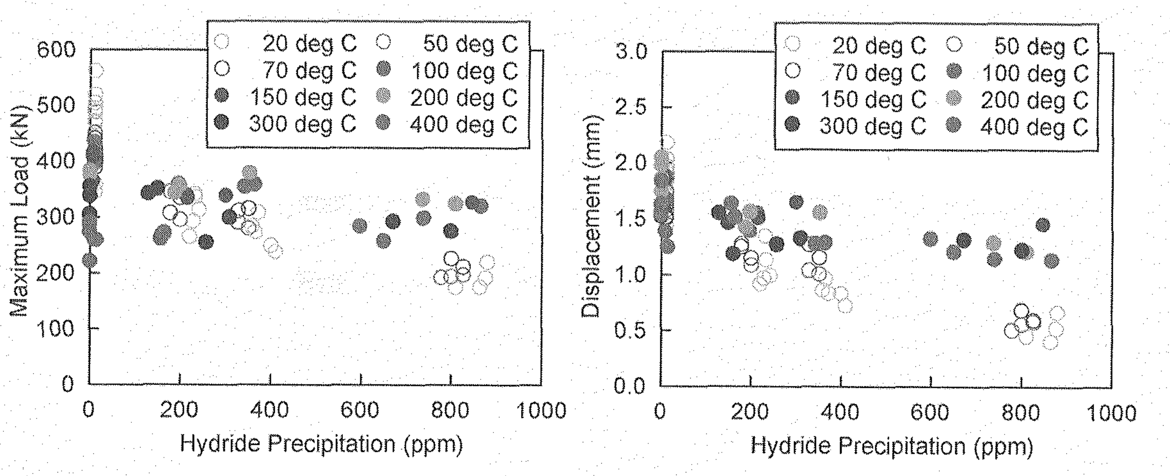


◊Temperature < 200°C
Maximum load and displacement become larger with higher temperature

◊Temperature > 200°C
Maximum load become smaller with higher temperature
Displacement is leveled off with higher temperature

8

Hydrogen content dependence of Load and Displacement

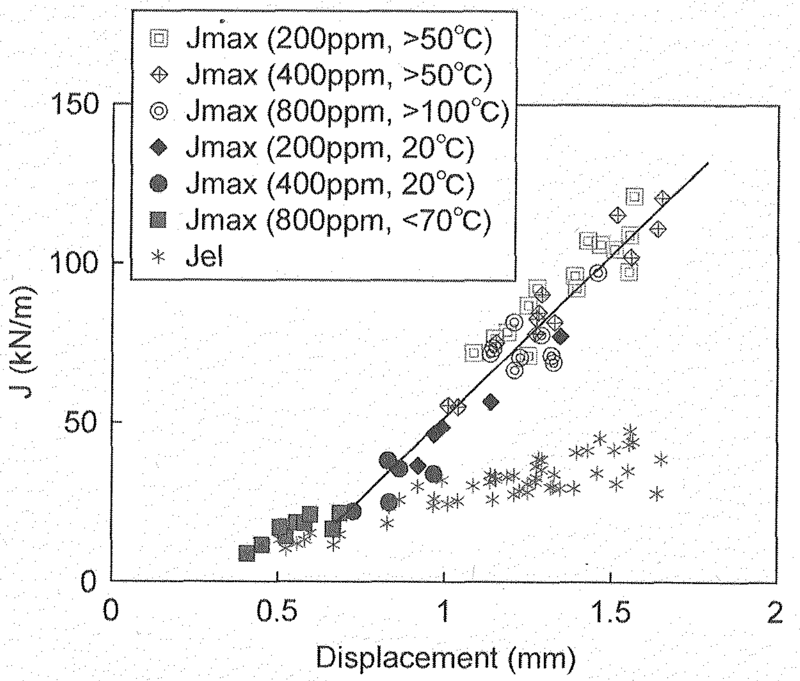


open mark : temperature < 70 °C

- Temperature > 100°C
  Maximum load and displacement is leveled off with higher hydride precipitation
- Temperature < 70°C
  Displacement decrease is much larger than maximum load decrease

9

Displacement dependence of Jmax



Jmax of high hydride precipitation and high temperature seems to increase linearly with displacement

10

Summary

The results of fracture toughness test of unirradiated PWR cladding with high hydrogen content are listed in below.

- J_{max} does not decrease at high temperature.
- J_{max} has good linear correlation with displacement.

Fracture toughness is influenced by the deformation of non hydrided region.

11

4.Session 4 Fuel behavior under LOCA condition
Session 4-1

"The Fuel Safety Research Meeting 2005", Tokyo, Japan. March 2-3, 2005.

LOCA TESTING AT Halden

Second in-pile test in IFA-650 and preliminary PIE

E. Kolstad[1], W. Wiesenack[1], V. Grismanovs[1], B. Oberländer[2]

[1] OECD Halden Reactor Project, P.O.Box 173, 1751 Halden, Norway

[2] Institute of Energy Technology, P.O.Box 40, 2027 Kjeller, Norway

Abstract

The safety criteria for loss-of-coolant accidents (LOCA) are defined to ensure that the core will remain coolable. Since the LOCA experiments performed in the 1970s, largely with fresh fuel, changes in fuel design, the introduction of new cladding materials and in particular the move to high burnup have generated a need to re-examine these criteria and to verify their continued validity. The Halden reactor is suitable for integral in-pile tests on fuel behaviour under LOCA conditions. It is aimed to utilize BWR and PWR fuel rods irradiated in commercial reactors to burnup levels over 80 MWd/kg, with a thorough characterization of the cladding and its bonding with the fuel. There is an intention to include medium burnup fuel (40-46 MWd/kg) in the test series in order to bridge the gap between the low and high burnup fuels.

The second trial experiment on LOCA was successfully carried out in May 2004. The test was performed with a fresh pressurized PWR rod and consisted of a blow down phase, a heat-up, a hold at target peak clad temperature (PCT) and termination by a reactor scram. The main objective was to achieve ballooning and cladding failure to gain experience that will be used to run later experiments with pre-irradiated fuel rods. The PCT of 1050°C was achieved and rod rupture occurred at 800°C as evidenced by rod pressure and elongation measurements, as well as by gamma monitoring of the blow down line to the dump tank. The hold time above 900°C was ~6 min.

Pre-test calculations were carried out by VTT using the FRAPTRAN/GENFLO code. The code predicted the maximum cladding temperature with good accuracy. Also the timing and temperature of the rod failure were well predicted.

The rod with its capsule was gamma scanned at Halden and then shipped to the Kjeller hot cells for detailed post-irradiation examinations (PIE). The gamma scanning showed pellet relocation slightly below the rod axial mid-height (power peak) position. The PIE at Kjeller (on-going) has revealed ballooning and burst along the same area, uniform clad deformation above and below the burst region and appreciable clad oxidation.

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LOCA TESTING AT HALDEN

Second in-pile test in IFA-650 and preliminary PIE

E. Kolstad, W. Wiesenack, V. Grišmanovs, B. Oberländer
OECD Halden Reactor Project

Fuel Safety Research Meeting
Tokyo, March 2nd-3rd, 2005

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2

CONTENT OF PRESENTATION

- Background and objectives
- Test facility and instrumentation
  - Test rig and test rod
  - Outer loop
- Pre-test code calculations
- Test execution and results
- Preliminary PIE
- Summary
- Further plans

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NEED FOR LOCA STUDIES

The move to high burnup and the introduction of new cladding materials have generated a need to re-examine the safety criteria for LOCA and to verify their continued validity.

The in-pile tests in the Halden reactor will address LOCA issues using ex-LWR high burnup fuel segments.

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RESEARCH ON THERMAL-HYDRAULICS, CRITICAL HEAT TRANSFER AND LOCA AT HALDEN REACTOR

1963-68	Experiments on natural convection flow instabilities and dry-out limit
1965-72	Dry-out experiments in natural convection flow channels
1979-83	Safety-related tests: Blow-down, heatup & quench behaviour of nuclear rods and electric simulators IFA-511: Thermal response studies
1982-85	Safety-related tests: Blow-down, heatup & quench behaviour of nuclear rods and electric simulators IFA-54x: Ballooning and rod-to-rod interaction studies
1996-98	IFA-613: Short-term dry-out test series

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BASIC CONCEPTS OF LOCA

- **3 Phases:**
  - Blowdown (fuel/core uncovered), de-pressurisation
  - Refill (ECCS systems start)
  - Reflood (water level above core top)
- **Timing:** uncovery, quenching, long-term cooling
- **After-effects:**
  fuel temperature rises - - > cladding oxidation and hydriding - - > embrittlement (melting) of cladding - - > fuel fragmentation

→ **Safety criteria:**
  - Peak Clad Temperature (PCT) < 2200 F (1204 C)
  - Oxidation of cladding < 17% of its thickness
- Rod swelling/ballooning may alter core geometry/coolability

→ **Safety requirement:**
  calculated geometry changes must still warrant core cooling

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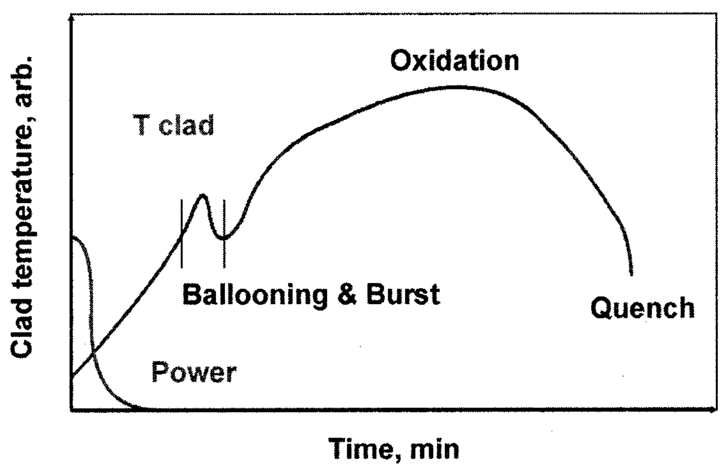
PHASES OF LOST OF COOLANT ACCIDENT



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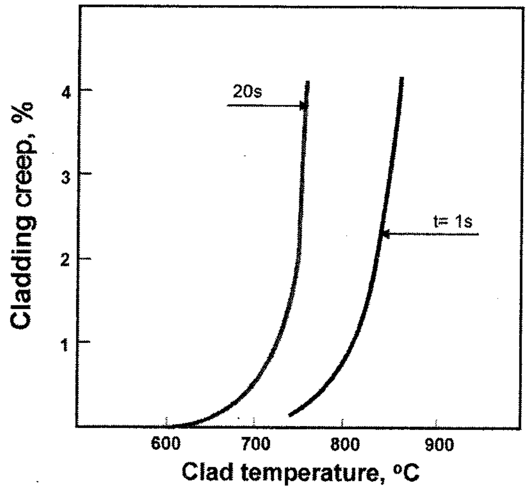
CREEP OF ZIRCALOY (MATPRO)



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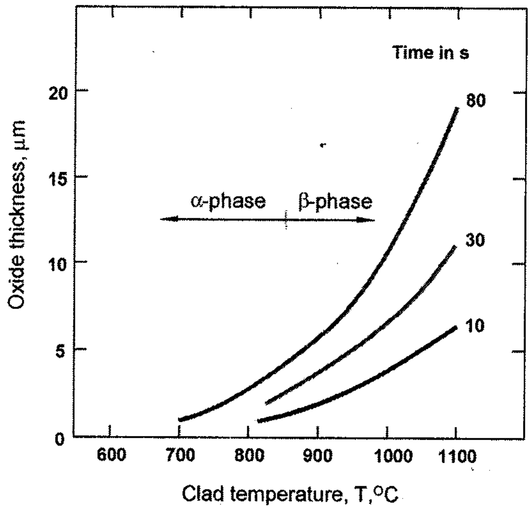
OXIDATION OF ZIRCALOY IN STEAM (MATPRO)



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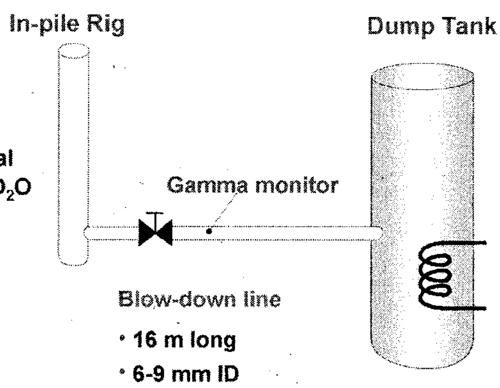
HALDEN LOCA EXPERIMENT (IFA-650.2)

(USNRC/EPRI/IRSN/EDF/FRAMATOM-ANP/GNF)

Primary objectives

- Measure the extent of fuel (fragment) relocation into the ballooned region and evaluate its possible effect on cladding temperature and oxidation
- Investigate the extent of – "secondary transient hydriding"- on the inner side of the cladding above and below the burst region

SETUP OF THE LOCA TEST



- 1 liter
- 70 bar
- Forced / Natural circulation of D_2O
- Heat rate: 15-30 W/cm

Blow-down line
- 16 m long
- 6-9 mm ID
- Gamma monitor
- Blow-down rate control

- Shielded
- Cooled
- Overpressure: 2-3 bar
- Level gauge

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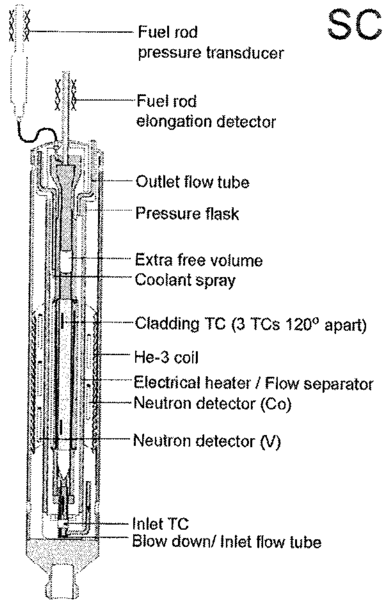
SCHEMATIC OF IFA-650.2



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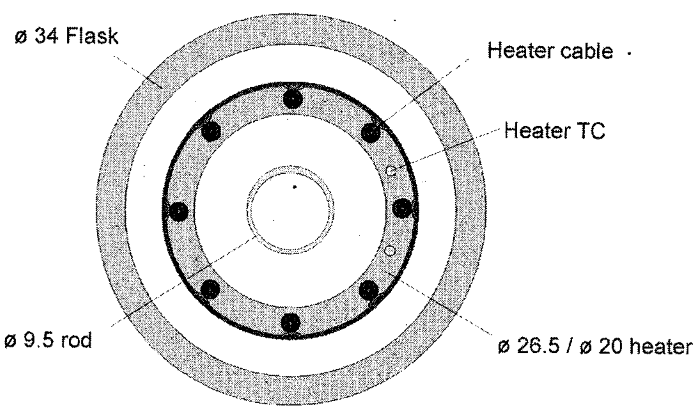
12

CROSS SECTION OF IFA-650.2



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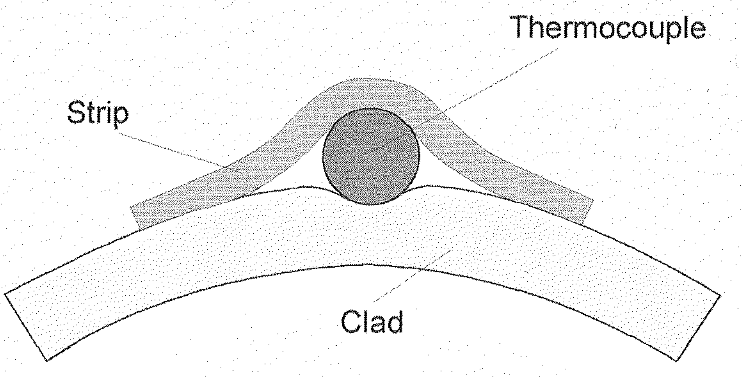
ATTACHMENT OF CLAD THERMOCOUPLES



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PRE-TEST CODE CALCULATIONS

- TRAC-BF1 code (PSI)
- FRAPTRAN – GENFLOW (VTT)
- SCTEMP and ALGOR (Halden)

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LOCA TRIAL RUNS: IFA-650.1

Test rod and instrumentation

Rod (Zry-4)		**Instrumentation**
Length:	50 cm	3 Clad thermocouples
OD / ID:	9.50/8.36 mm	1 Clad extensometer
Gap size:	70 μm	1 Fuel thermocouple
Enrichment:	4 wt% U-235	1 Rod pressure sensor
Fill pressure:	2 bar He	2 Heater thermocouples
Free volume:	15 cm^3	
Dished pellets		

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LOCA TRIAL RUNS: IFA-650.1

- Blow-down tests at zero power
- Power calibrations (reactor power ~18 MW)
- Trial runs (reactor power 5-6 MW)

Six test runs:

- 14 W/cm (rod) + 6 W/cm (heater) (~ 830°C)
- 14 W/cm (rod) + 6 W/cm (heater) (~ 830°C)
- 14 W/cm (rod) + 12 W/cm (heater) (~ 900°C)
- 25 W/cm (rod) + 6 W/cm (heater) (~ 930°C)
- 25 W/cm (rod) + 18 W/cm (heater) (~1030°C)
- 30 W/cm (rod) + 20 W/cm (heater) (~1120°C)

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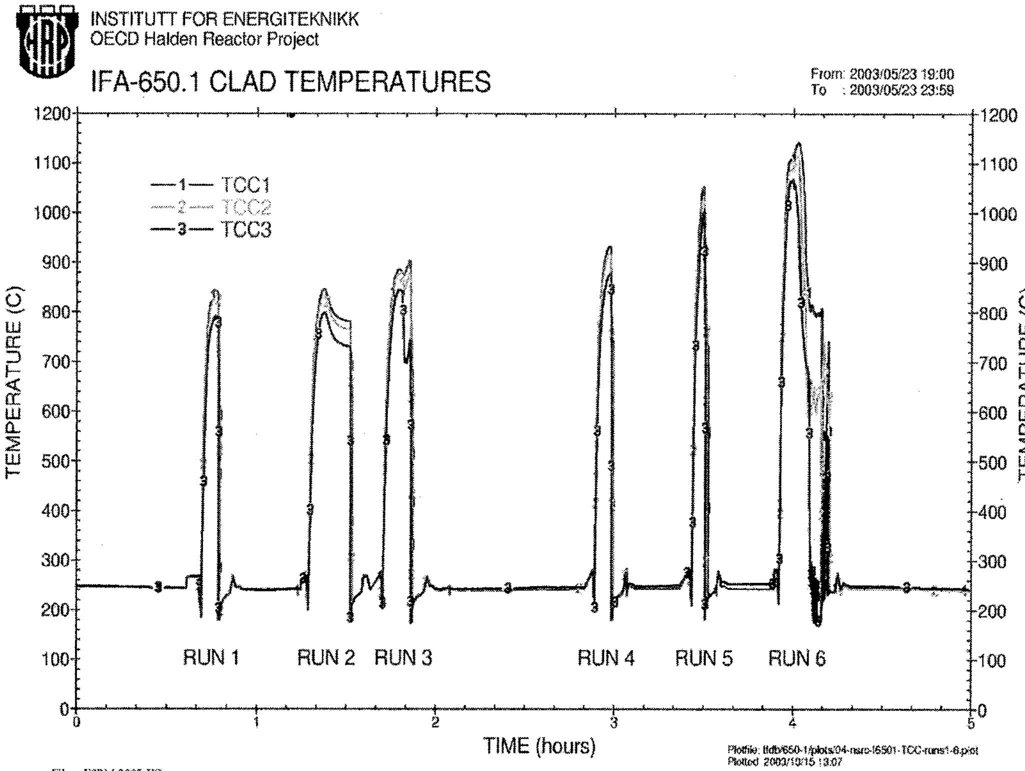

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18

LOCA TRIAL RUN: IFA-650.2

Test rod and instrumentation

Rod (Zry-4)		**Instrumentation**
Length:	50 cm	4 Clad thermocouples
OD / ID:	9.50/8.36 mm	1 Clad extensometer
Gap size:	70 μm	1 Rod pressure sensor
Enrichment:	2 wt% U-235	2 Heater thermocouples
Fill pressure:	40 bar He (RT)	
Free volume:	15 cm^3	
Dished pellets		

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19

LOCA TRIAL RUN: IFA-650.2

- Blow-down tests at zero power
- Power calibration (reactor power ~17 MW)
- Trial run (reactor power 8.6 MW)

One run:

22 W/cm (rod) + 17 W/cm (heater) (~1050°C)

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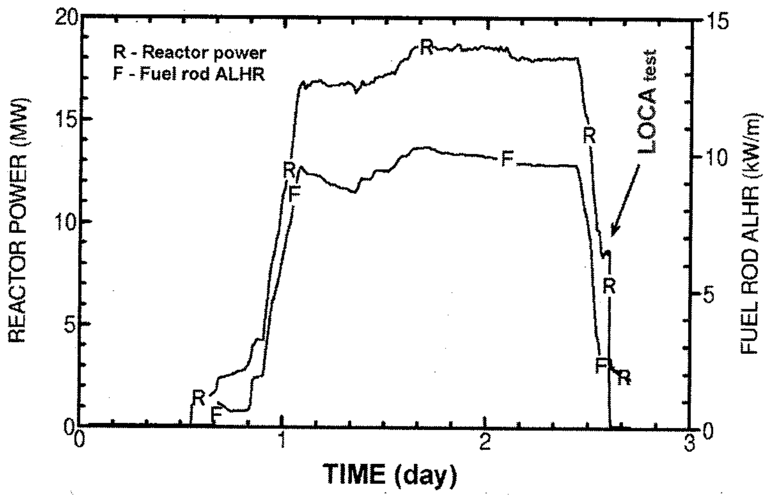
20

IFA-650.2 Fuel rod power history



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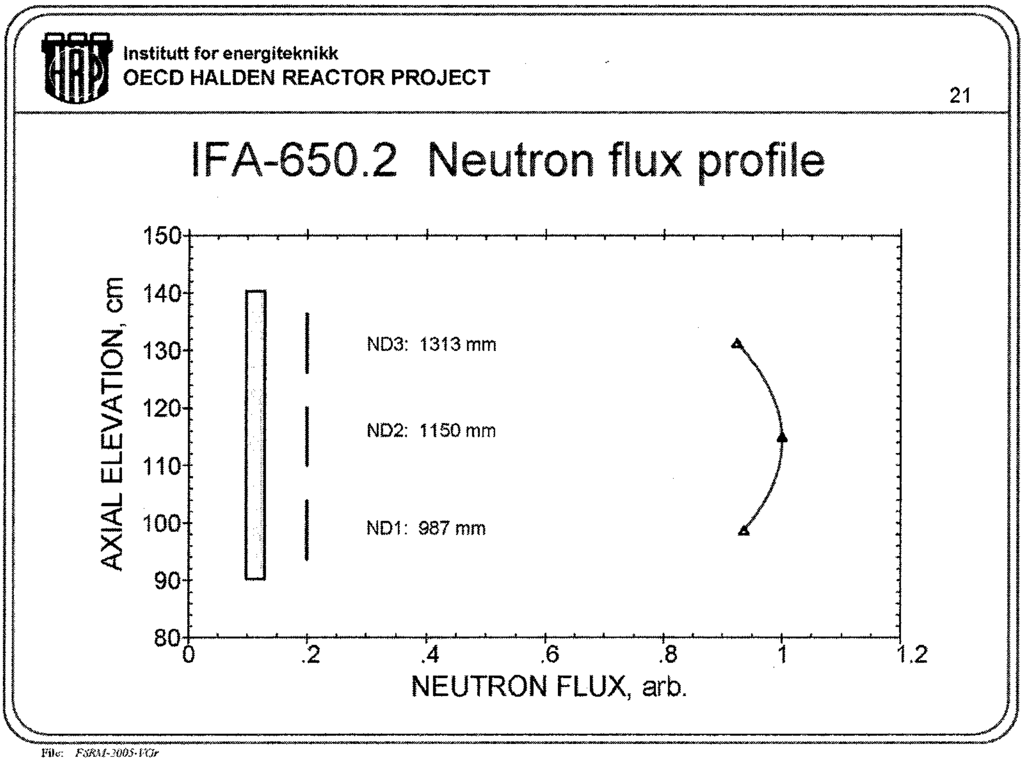
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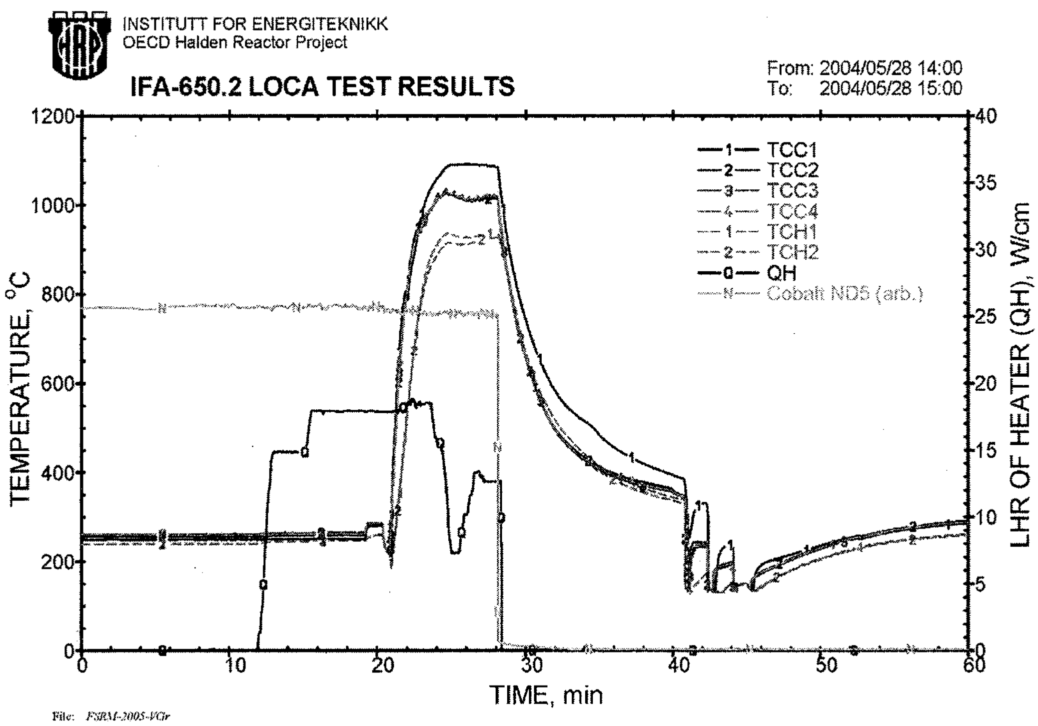
IFA-650.2 Neutron flux profile
AXIAL ELEVATION, cm
ND3: 1313 mm
ND2: 1150 mm
ND1: 987 mm
NEUTRON FLUX, arb.


INSTITUTT FOR ENERGITEKNIKK
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IFA-650.2 LOCA TEST RESULTS
From: 2004/05/28 14:00
To: 2004/05/28 15:00
TCC1
TCC2
TCC3
TCC4
TCH1
TCH2
QH
Cobalt ND5 (arb.)
TEMPERATURE, °C
LHR OF HEATER (QH), W/cm
TIME, min

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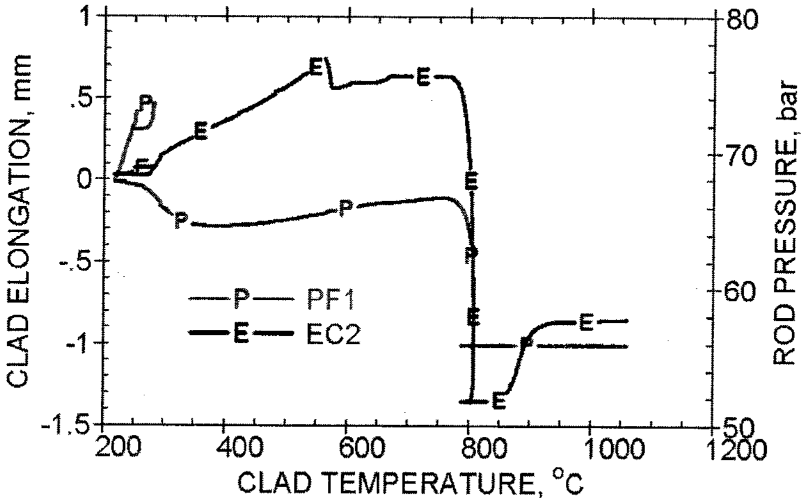
23

IFA-650.2 Rod pressure and clad elongation



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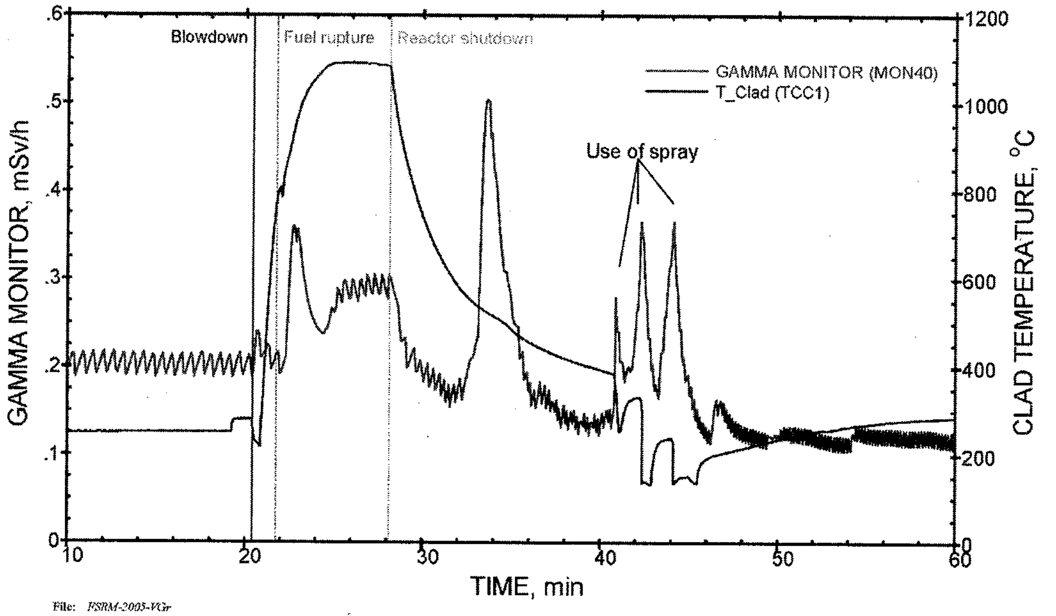
IFA-650.2 LOCA TEST RESULTS

From: 2004/05/28 14:00
To: 2004/05/28 14:59



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Summary of 2nd in-pile test in Halden (1)

- Loop and rig, and rod instrumentation worked well
- Target PCT ~1050 ^{O}C was achieved
- Rod rupture was detected at 800 ^{O}C
- Hoop stress at failure: ~55 MPa
- Clad temperature increase rate: 7.8 ^{O}C/s
- Small azimuthally temperature variation: +/- 3 ^{O}C
- Holding time above 900 ^{O}C: 6.5 minutes
- Water spray was applied intermittently above 900 ^{O}C
- Test was terminated by reactor scram

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IFA-650.2 Preliminary PIE (On-going)

- Gamma-scanning at Halden
- Detail PIE at Kjeller hot cells
  - characterization of balloon (dimensions, shape)
  - oxide thickness (axial distribution: inside/outside)
  - hydrogen content in balloon area and its vicinity

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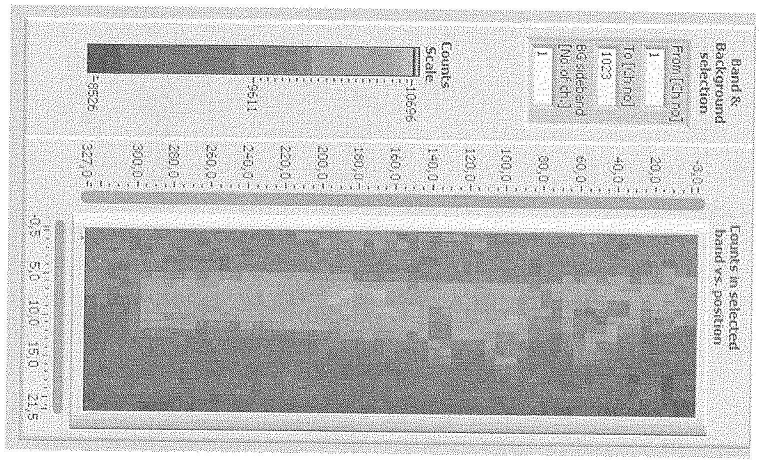
GAMMA SCANNING AT HALDEN



Sum of total gamma spectrum of fuel rod 650-2 (lower scan).

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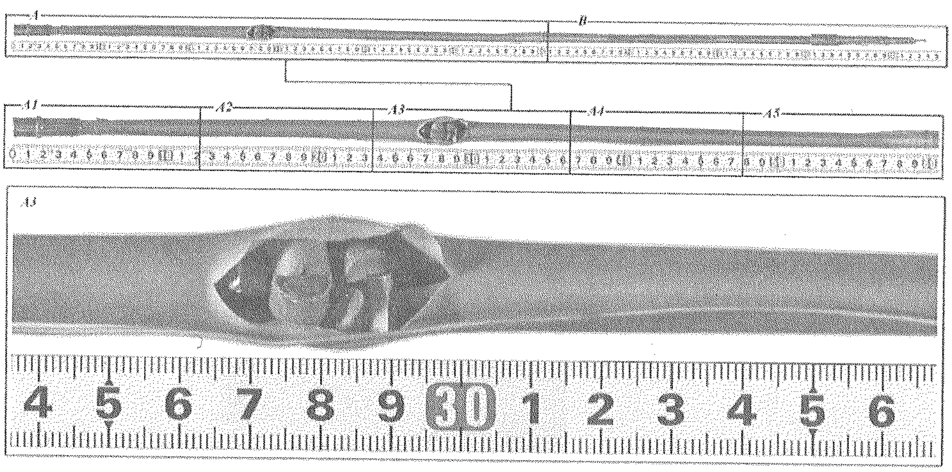
28

2nd LOCA TEST - VISUAL EXAMINATIONS



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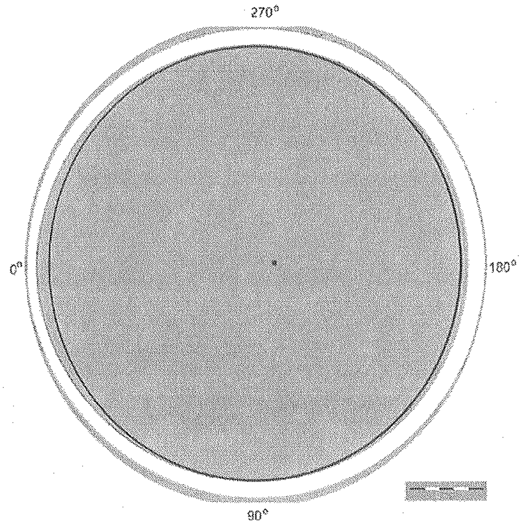
SECOND LOCA TEST - PIE

Cladding cross-section above the burst area



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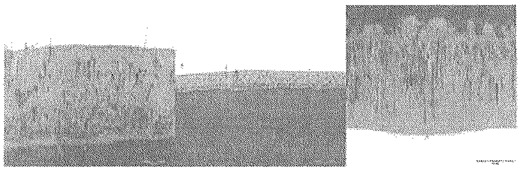
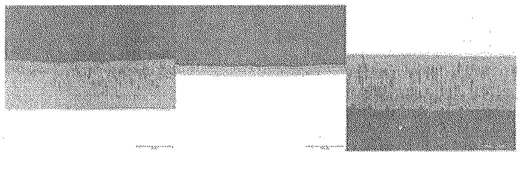
SECOND LOCA TEST - PIE

Oxide thicknesses above the burst region



404 0 fuelside, 100x   404 0 fuelside, 20x   404 0 waterside, 100x



404 180 fuelside, 100x   404 180 fuelside, 20x   404 180 waterside, 100x

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Summary of 2nd in-pile test in Halden (2)

Preliminary PIE results

- Fuel relocation detected by gamma scanning (Halden)
- Visual exams revealed balloon and burst at around rod mid-height (peak power) position
- Length of clad burst ~ 35 mm
- Uniform deformation prior to burst 35-40% close to the the burst opening
- Fuel fragments had fallen out. (In-pile/handling/ transport?)
- Oxide thickness: 25-40 μm

Somewhat larger at OD. Decreased a little away from the burst

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FURTHER PLANS AND TEST OBJECTS

- Pairs of rods (50 - 80 MWd/kg, 40-50 cm length) from commercial LWRs to be tested in PWR and BWR conditions, suitable to address possible effect of axial fuel fragment relocation
  - when does it occur (heat-up, quenching)?
  - does bonding prevent the movement of fragments?
- Include medium burnup fuel (~40 MWd/kg, less or no bonding) to bridge the gap between low and high burnup
- VVER fuel envisaged for testing at a later stage

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Session 4-2

ANL LOCA Research Results for Cladding Alloys: Zircaloy-2, Zircaloy-4, ZIRLO, and M5

M. C. Billone, Y. Yan, T. Burtseva, and H. M. Chung

Argonne National Laboratory (ANL)
Argonne, Illinois, USA

The ANL test program is directed toward providing LOCA-relevant data for unirradiated advanced-alloy cladding, unirradiated Zircaloy-2 (Zry-2) and Zircaloy-4 (Zry-4) cladding, defueled high-burnup Zry-4, and fueled high-burnup Zry-2 and Zry-4. The purpose of the advanced-alloy program is to determine the post-quench ductility of ZIRLO and M5 alloys, as compared to Zry-4 oxidized and quenched under the same time-temperature conditions. The Zry-4 and Zry-2 data also serve as baseline data for the performance of fueled high-burnup segments subjected to LOCA integral and post-quench ductility tests. The data are generated for assessment of the adequacy of USNRC licensing criteria (10 CFR 50.46) for loss-of-coolant accident (LOCA) events. The criteria assume that cladding oxidized to ≤17% ECR at ≤1204°C PCT will retain enough ductility to resist fragmentation during and following quench.

Room-temperature (RT) post-quench ductility results were presented at FSRM-2004 for 17×17 low-tin Zry-4, ZIRLO, and M5 PWR cladding. Samples were exposed to two-sided steam oxidation at 1000, 1100, and 1200°C for test times corresponding to calculated ECR values of 5-20%. Oxidation was followed by cooling at ≈10°C/s to 800°C and rapid quench to 100°C. The RT results indicated good post-quench ductility up to >17% ECR for samples oxidized at 1000 and 1100°C. However, samples oxidized at 1200°C exhibited RT embrittlement at <13% Cathcart-Pawel (CP) ECR (<17% Baker-Just ECR). These 1200°C-oxidized alloys were retested at 135°C, which is assumed to be the temperature of the core immediately following quench cooling. All three alloys exhibited significant enhancement in ductility at 135°C to >17% CP-ECR. The post-quench ductility of ZIRLO and M5 was higher than for Zry-4 at both RT and 135°C. Thus, non-deformed, as-fabricated PWR cladding does retain enough post-quench ductility to ensure ductile behavior during quench. However, irradiated cladding contains hydrogen, which is known to enhance embrittlement by increasing the oxygen concentration in the prior-beta layer.

In order to study the embrittling effects of hydrogen, prehydrided (150-800 wppm) 17×17 Zry-4 samples were oxidized at 1200°C to measured ECR values of ≈8.5 and ≈10.5%. At 135°C, the 8.5%-ECR samples embrittled at ≈400 wppm, while the 10.5%-ECR samples embrittled at about 300 wppm. Clearly, more tests are needed to map out embrittlement vs. oxidation level (ECR) and hydrogen content.

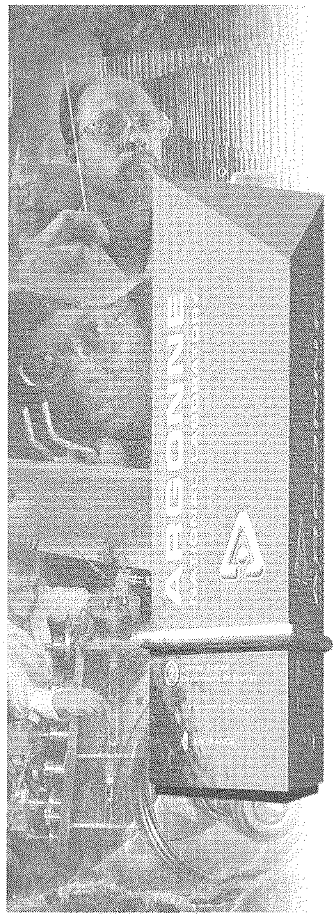
Four LOCA integral tests have been completed with high-burnup (≈56 GWd/MTU) BWR fueled segments. The 4th test, completed just after FSRM-2004, included the full test sequence: pre-pressurization, ramp in steam through burst to 1204°C, hold for 5 minutes at 1204°C, slow cool at 3°C/s to 800°C and quench to 100°C. Test #2 (without quench) and Test #3 (partial quench to 470°C) were also run for the same oxidation time-temperature. These samples were sectioned to generate samples for hydrogen and oxygen determination. The oxygen concentration peaked near the axial location at the center of the burst and decreased from the burst center to the balloon necks, as expected. The secondary hydrogen pickup was high in the burst region, peaked just beyond the burst region and decreased toward the balloon neck regions. These results are consistent with JAERI results for irradiated and defueled PWR cladding exposed to similar integral tests. However, they are qualitatively different from tests conducted on fresh cladding filled with alumina or zirconia pellets. Prior to sectioning the Test #3 sample, it failed at 3 locations during handling. SEM fractography of 2 of the failure surfaces indicated that brittle failure occurred at multiple locations during the room-temperature handling.

Post-quench ductility tests were also performed using pre-hydrided (300-800 wppm) 15x15 Zry-4 and defueled high-burnup Zry-4. Both sets of samples were oxidized (two-sided) in steam at ≈1200°C, cooled at ≈8°C/s to 800°C and either quenched (all pre-hydrided samples and one high-burnup sample) or slow cooled (five high-burnup samples). For the pre-hydrided samples, the hydrogen content was varied and the ECR values were fixed at two levels: 5% and 7.5% CP-ECR. At 135°C, the 5%-ECR samples embrittled at ≈600 wppm, while the 7.5%-ECR samples embrittled at ≈400 wppm. Irradiated cladding samples were prepared from near the midplane (-40 to 100 mm) of H. B. Robinson Rod F07 (≈64 GWd/MTU), oxidized in steam to 3% CP-ECR (Tmax = 1140°C), 5% CP-ECR (Tmax = 1185°C), 7% CP-ECR (Tmax = 1204°C), and 10% CP-ECR (Tmax = 1206°C), and cooled without quench. The corrosion layer thickness at the fuel midplane was 70 μm and the hydrogen content of the cladding was measured to be 550±90 wppm. Ring-compression tests at 135°C indicated that the 3 and 5% CP-ECR samples were highly ductile, the 7% CP-ECR sample was marginally ductile, and the 10% CP-ECR sample was brittle. Metallography of these samples indicated that the corrosion layer was partially protective giving an oxygen pickup and ECR less than predicted by the CP model for bare cladding. However, it should be noted that low ductile-to-brittle ECR values are dependent on the ramp rate to 1204°C and the transient ECR accumulated at 1204°C. Only 2 of the samples tested reached 1204°C.

Additional samples were prepared from 320 mm above the midplane (≈80-μm corrosion layer; 545±80 wppm H) and from 660 mm above the midplane (≈95-μm corrosion layer and 800±100 wppm H). Both samples were oxidized to 8% CP-ECR (Tmax = 1206°C). The lower hydrogen-content sample was slow cooled, while the higher hydrogen-content sample was quenched from 800°C to 100°C. The slow-cooled sample was marginally ductile, while the quenched sample was brittle. It is reasonable to assume that the ductility difference was more a function of hydrogen content than quench vs. slow-cooling. Quantitative metallography of the slow-cooled sample showed a thinner outer-surface steam-oxide layer (20 μm) than inner-surface steam-oxide layer (27 μm) and a lower-than-predicted weight gain leading to a measured ECR = 5.8%. Although additional testing is planned, the preliminary conclusion is that the PWR corrosion layer is partially protective with respect to steam oxidation and that post-quench ductility results from prehydrided bare-cladding samples may be overly pessimistic for low ductile-to-brittle transition ECR values.

LOCA integral tests are planned for fueled high-burnup PWR cladding to characterize ballooning and burst, inner- and outer-surface oxidation in the balloon region at 1204°C, secondary hydrogen pickup in the balloon region, oxidation in the beyond-balloon region, post-quench ductility and strength of the balloon region (4-point-bend test) and post-quench ductility of the beyond-balloon region (ring-compression tests). With the high hydrogen pickup during irradiation and the expected high secondary hydrogen pickup during the LOCA event, it is unlikely that the balloon region will retain post-quench ductility except at very low exposure times to 1204°C (e.g., very low ECR values). Additional one-sided oxidation tests of defueled, high-burnup Zry-4 cladding are planned to generate post-quench ductility data vs. ECR at 1000°C, 1100°C and 1200°C. These data will be directly relevant to the beyond-balloon region of high-burnup PWR fuel rods with Zry-4 cladding.

High-burnup defueled ZIRLO and M5 cladding will soon become available for use in the ANL test program, as well as high-burnup fueled M5 (≈74 GWd/MTU) and intermediate-burnup fueled ZIRLO (≈50 GWd/MTU). The one-sided oxidation tests and the LOCA integral tests described for high-burnup PWR Zry-4 will be repeated for these advanced alloys.



ANL LOCA Research Results for Cladding Alloys

M. Billone, Y. Yan, T. Burtseva and H. Chung
Energy Technology Division

Fuel Safety Research Meeting (FSRM-2005)
Tokyo, Japan
March 2-3, 2005

Argonne National Laboratory

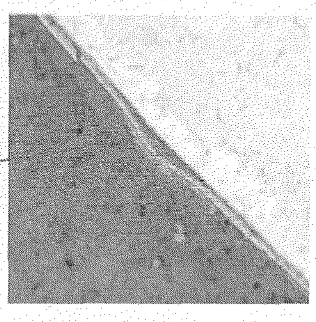
Office of Science U.S. Department of Energy

A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

Background Information

- **ANL LOCA Program: Focused on Post-Quench Ductility**
  - Embrittlement ↑ as oxygen content ↑ in metal (prior-beta layer)
    As-fabricated Zry-4: 0.12-0.14 wt.% O (similar for ZIRLO and M5)
    RT-embrittlement occurs at ≈0.5 wt.% oxygen for Zry-4
    135°C-embrittlement occurs at ≈0.6-0.7 wt.% oxygen for Zry-4
  - Solubility of oxygen in prior-beta layer of as-fabricated cladding (low H)
    0.24 wt.% at 1000°C, 0.38 wt.% at 1100°C, **0.58 wt.% at 1204°C**
  - Solubility of oxygen in prior beta vs. hydrogen (300-600 wppm)
    >0.24 wt.% at 1000°C
    >0.38 wt.% at 1100°C, **≈0.9 wt.% at 1200°C & ≈320-wppm-H (CEA)**
- **Ring-Compression Ductility Screening Tests**
  - Permanent strain (pre-test Do – post-test Do)/pre-test Do
  - Offset strain: derived from load-displacement curve

E110 after 300 s and 1400 s at 1000° C Ductile or Brittle?



Delaminated
Oxide Layer
after 300 s

5% ECR
≈300 s at 1000° C
120±50 wppm H

10% ECR
≈1400 s at
1000° C
>4000 wppm H

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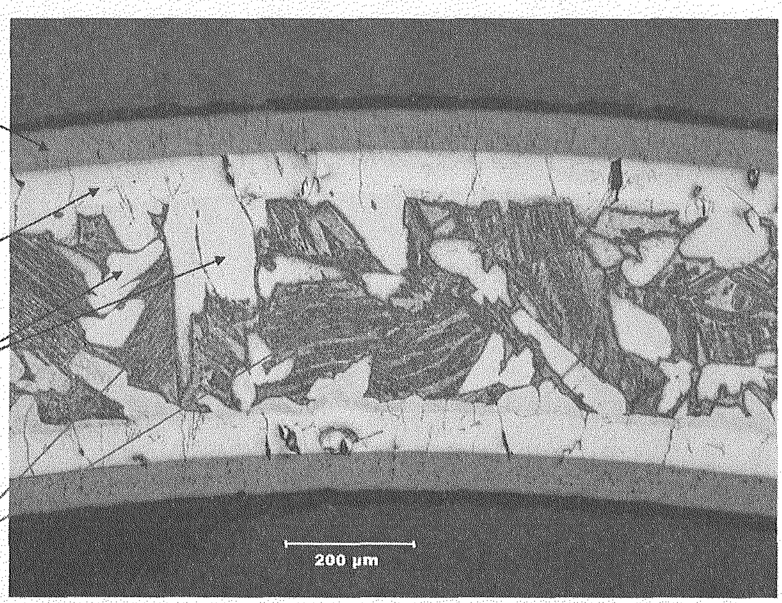
Post-Quench Zry-4: 20% ECR at 1100°C Ductile or Brittle?

Brittle
Oxide

Very Brittle Metal
≈2-3 wt.% O
≈1000 DPH

Brittle Metal
≈0.6-2 wt.% O
≈500 DPH

Ductile Metal
<0.4 wt.% O
<300 DPH



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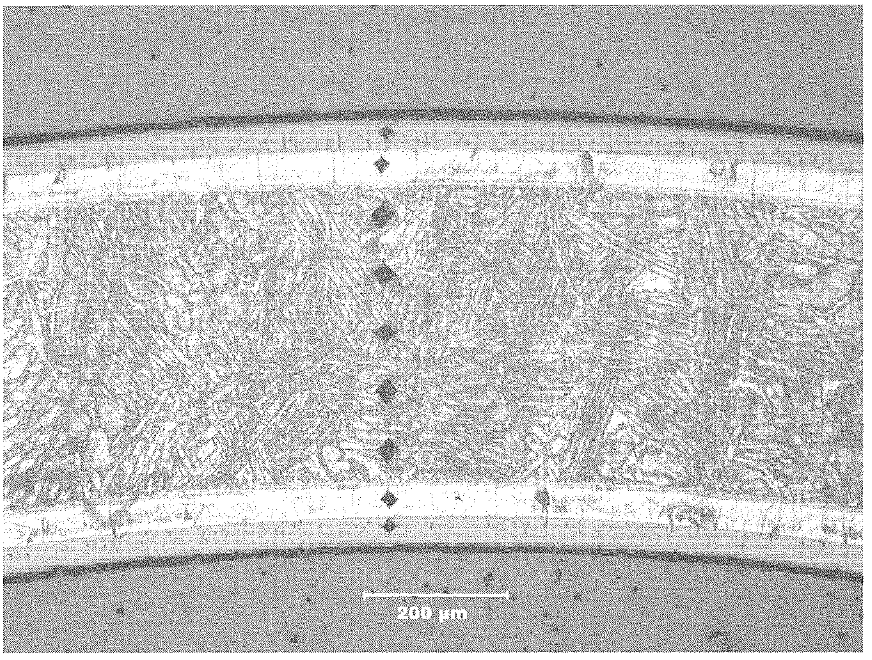
Post-Quench Zry-4: 13% ECR at 1200°C Ductile or Brittle????



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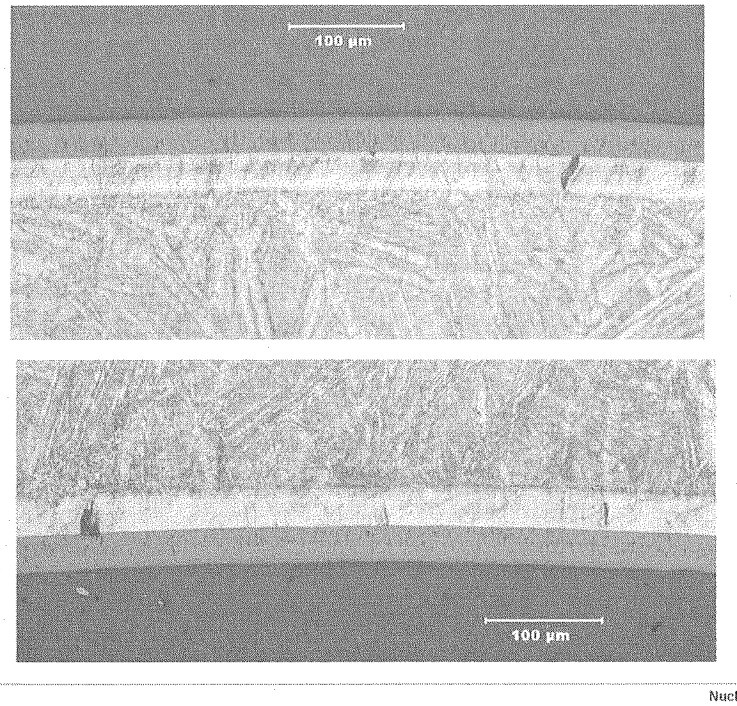
Post-Quench Zry-4: 7.5%-ECR at 1204°C (600-wppm-H) Ductile or Brittle??



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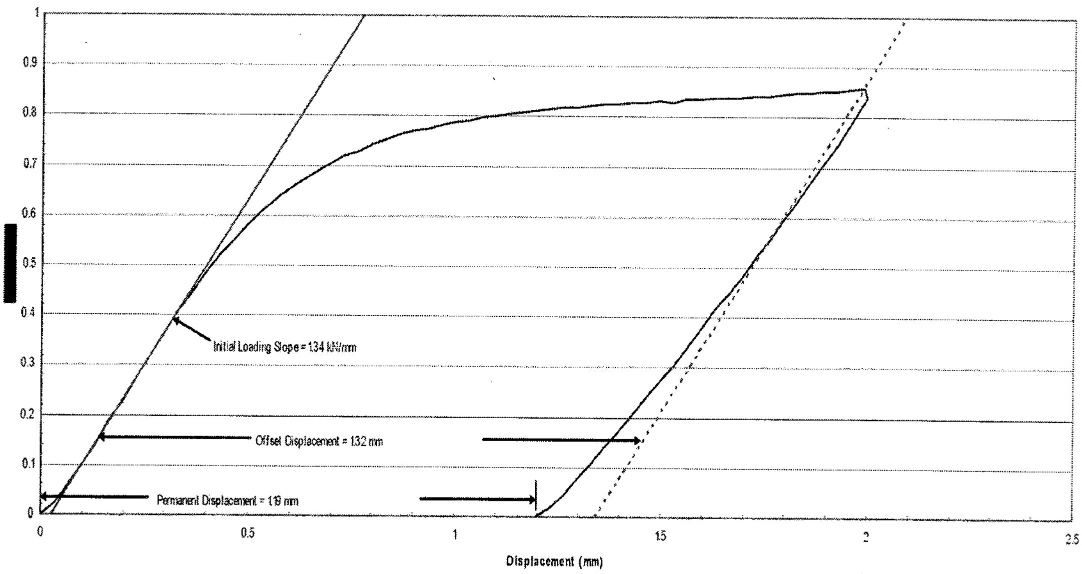
Definitions of Offset and Permanent Displacements and Strains for As-Fabricated 15×15 Zry-4



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Scope of LOCA-Relevant Research

- **Licensing Issues Addressed**
  - 10 CFR 50.46 embrittlement criteria for maintaining residual ductility in Zircaloy (Zry) cladding; temperature limit: PCT ≤ 1204°C, oxidation limit: effective cladding reacted ECR ≤ 17%
  - Confirm embrittlement criteria for high-burnup Zry-2 and Zry-4
  - Compare post-quench ductility of ZIRLO and M5 to Zry-4 vs. ECR
- **High-Burnup Phenomena Investigated**
  - Fuel behavior and effects of fuel on cladding during a LOCA sequence
  - Effects of corrosion, hydriding and irradiation on cladding: Ballooning, burst, high-temperature steam oxidation, Secondary hydriding, quench behavior and post-quench ductility
- **Advanced-Alloy Cladding Phenomena Investigated**
  - ZIRLO and M5 oxidation kinetics (vs. Zry-4); E110
  - ZIRLO and M5 post-quench ductility (vs. Zry-4); E110

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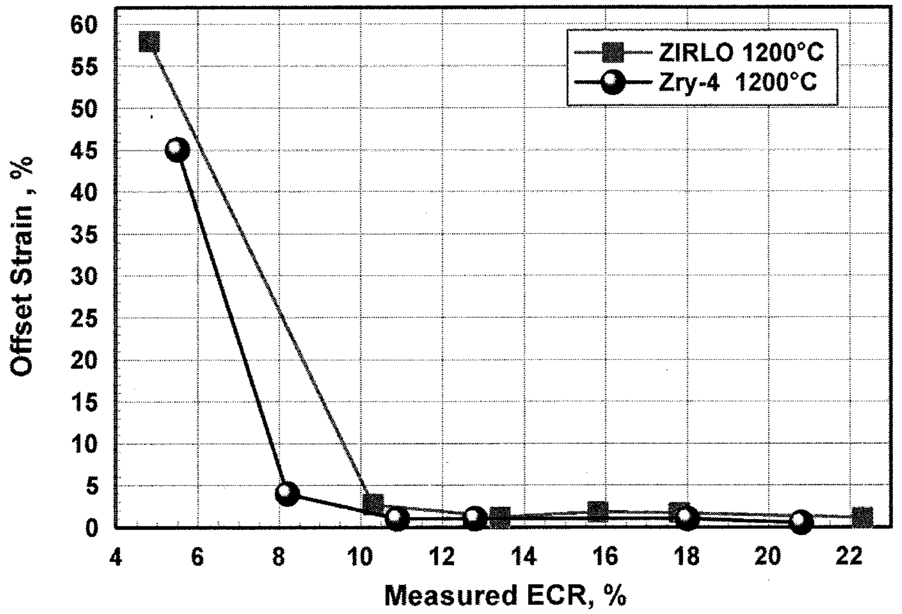
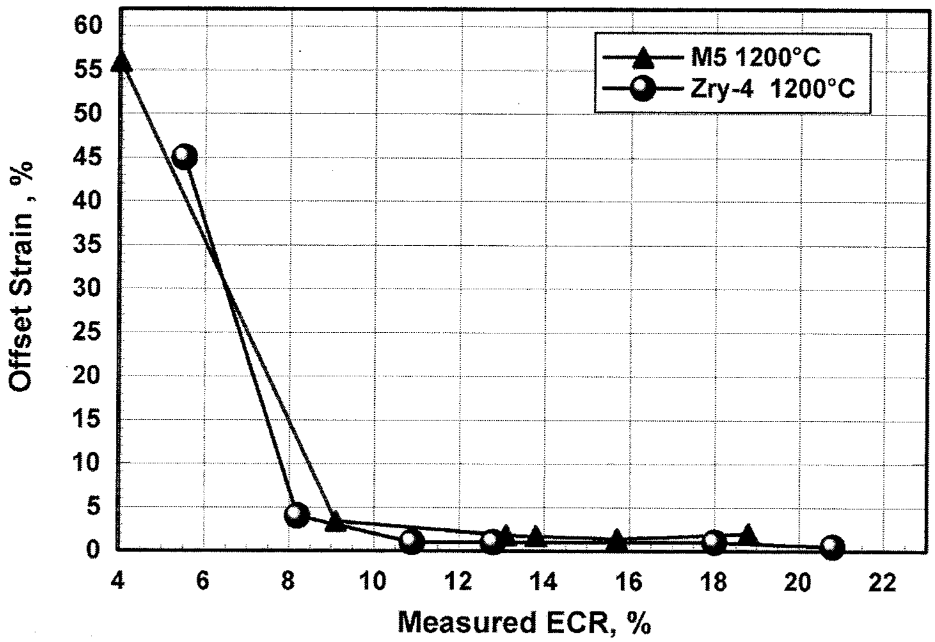
Summary of ANL Results

- **Advanced Alloy Program (NSRC-2004, Billone)**
  - 1000 & 1100°C oxidation, slow-cooled to 800°C & quenched
    Good RT ductility to >17% Cathcart-Pawel (CP)-ECR (Zry-4, ZIRLO, M5)
  - 1200°C-oxidized, slow-cooled to 800°C & quenched (Zry-4, ZIRLO, M5)
    RT embrittlement at ≈9% ECR (Zry-4) & ≈12% ECR (ZIRLO & M5)
    Significant ductility improvement at 135°C (embrittlement ECR>17%)
    Severe embrittlement (135°C) at <10% ECR for >300 wppm H (pre-H)
  - Plan to test high-burnup ZIRLO and M5
- **LOCA Integral Test Results at 1204°C (NSRC-2004, Yan)**
  - High-burnup BWR Zry-2 (significant embrittlement in balloon region)
    Non-uniform wall-thinning, 2-sided oxidation, high secondary H pickup
  - High-burnup PWR Zry-4 (test details to be determined from data below)
    Baseline ductility data for as-fabricated & **prehydrided** 15×15 Zry-4 rings
    High-burnup Zry-4 data: rings oxidized at ≤1206°C to 3-10% CP-ECR

RT Offset Strain for 1200°C-Oxidatized ZIRLO vs. Zry-4



RT Offset Strain for 1200°C-Oxidatized M5 vs. Zry-4



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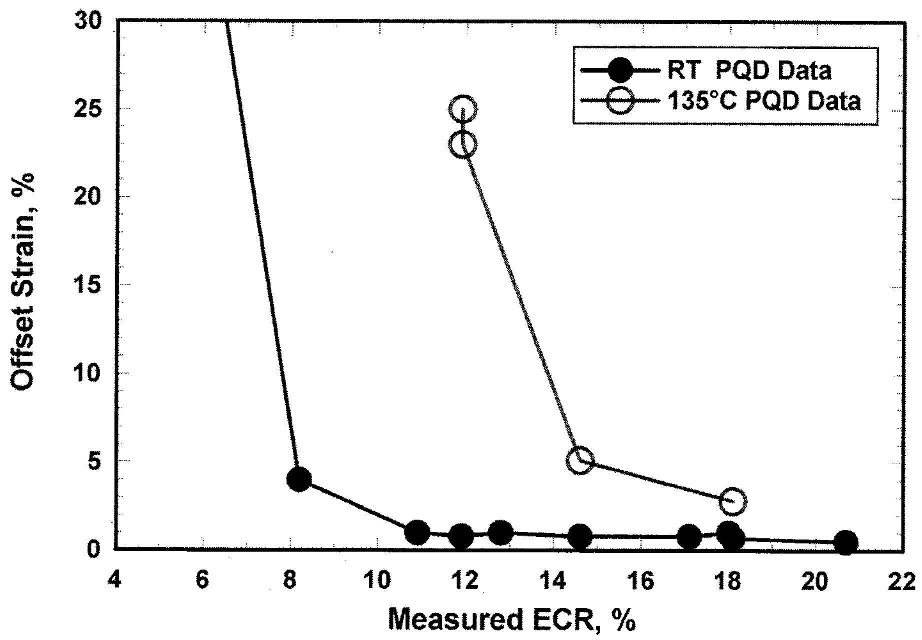
135°C Offset Strain for 17x17 Zry-4 Oxidized at 1200°C



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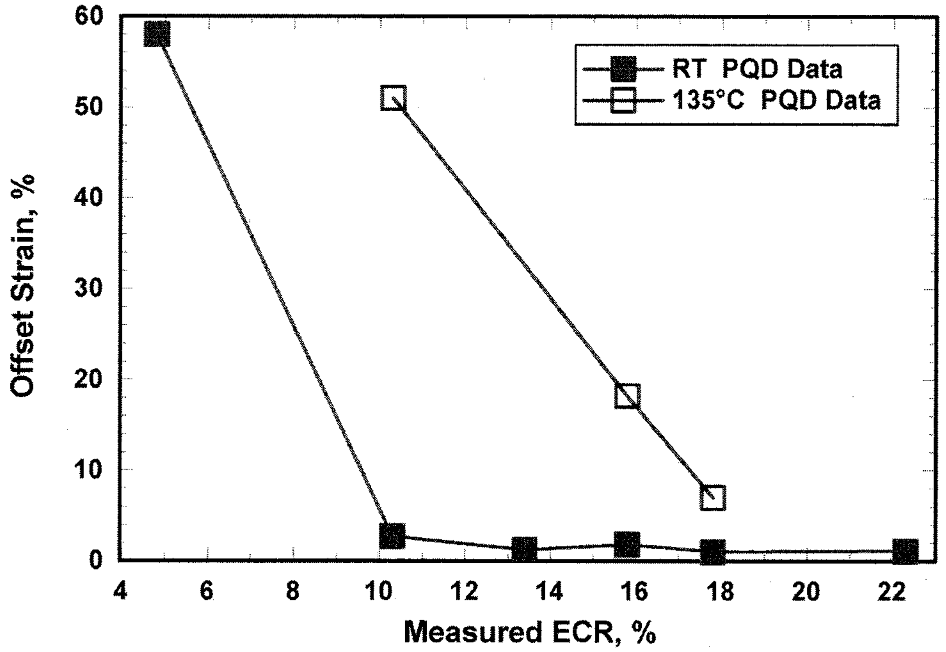
135°C Offset Strain for 17x17 ZIRLO Oxidized at 1200°C



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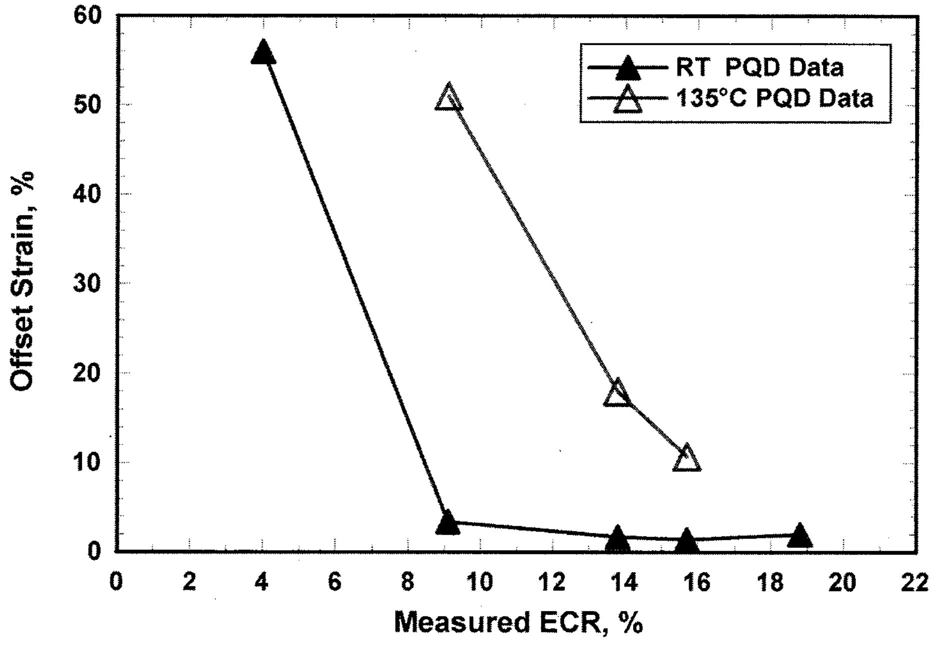
135°C Offset Strain for 17x17 M5 Oxidized at 1200°C



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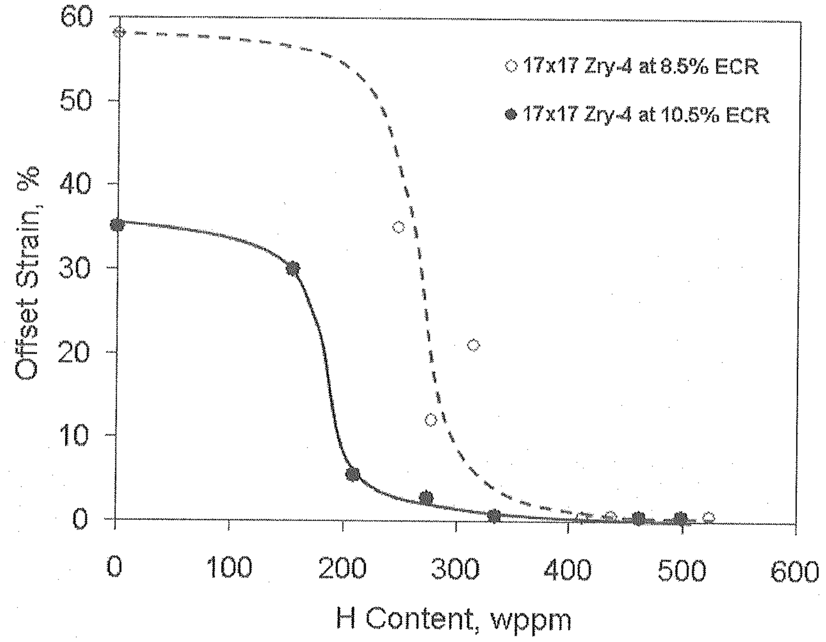
135°C Offset Strain for Prehydrided 17x17 Zry-4 Oxidized at 1200°C



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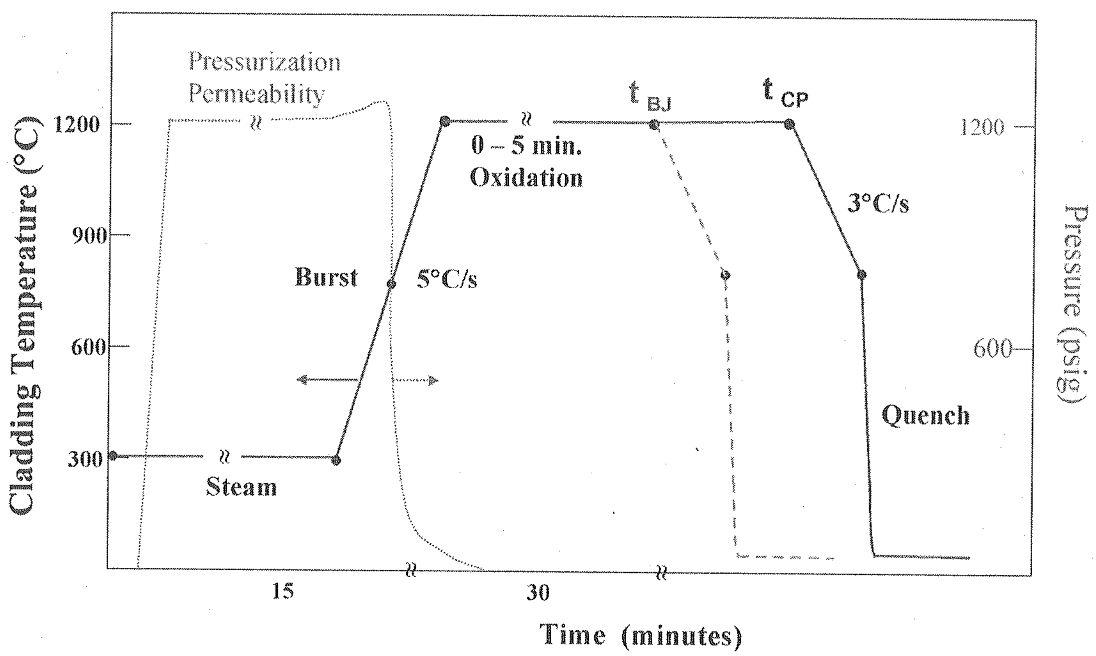
LOCA Integral Test Sequence & Time



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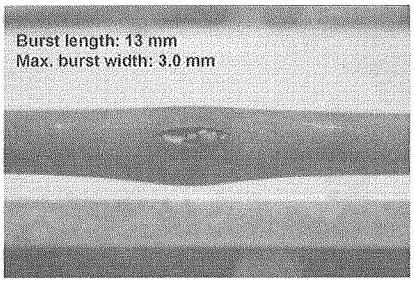
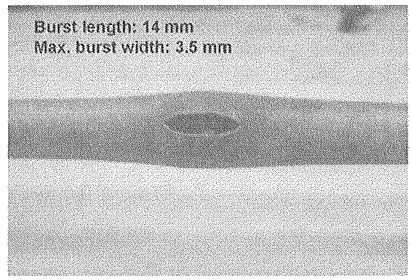
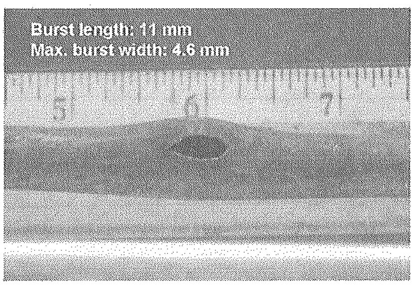
Balloon and Burst Regions for High-burnup Tests



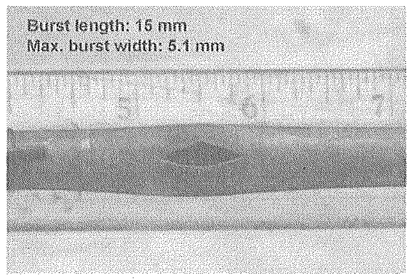
ICL#1: Ramp-to-Burst test conducted in argon; slow furnace cooling



ICL#2: LOCA sequence with 5-minute oxidation at 1204°C and slow-furnace cooling from 800° C



ICL#3: 5-min. oxidation at 1204°C followed by quench at 800° C (quartz tube failed at 480° C)



ICL#4: Full LOCA sequence (5-minute oxidation at 1204°C) with quench at 800° C

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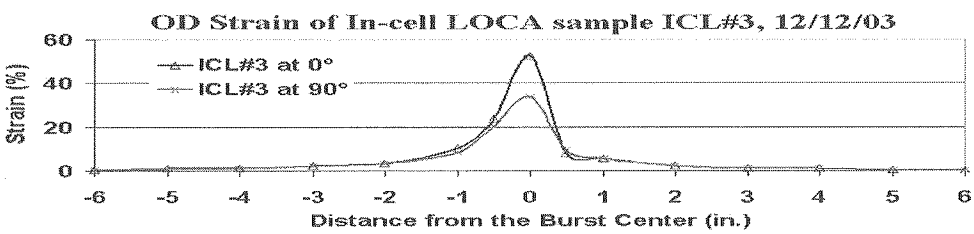
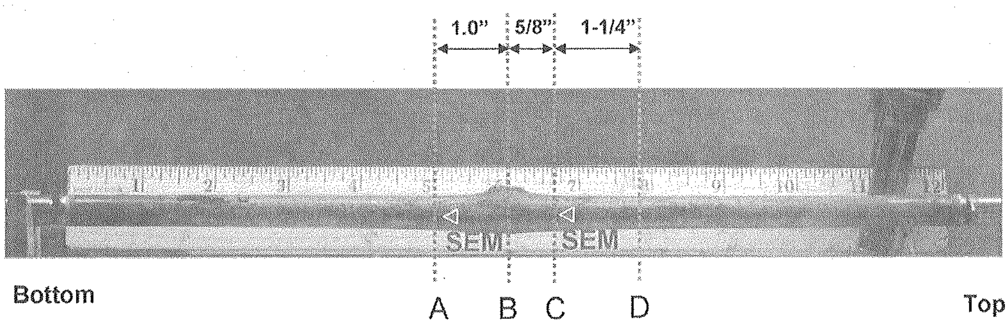
Post-test Characterization for ICL#3 Specimen





Sample ICL#3 was broken at locations A, B and C during the sample handling before the sectioning was performed at location D.

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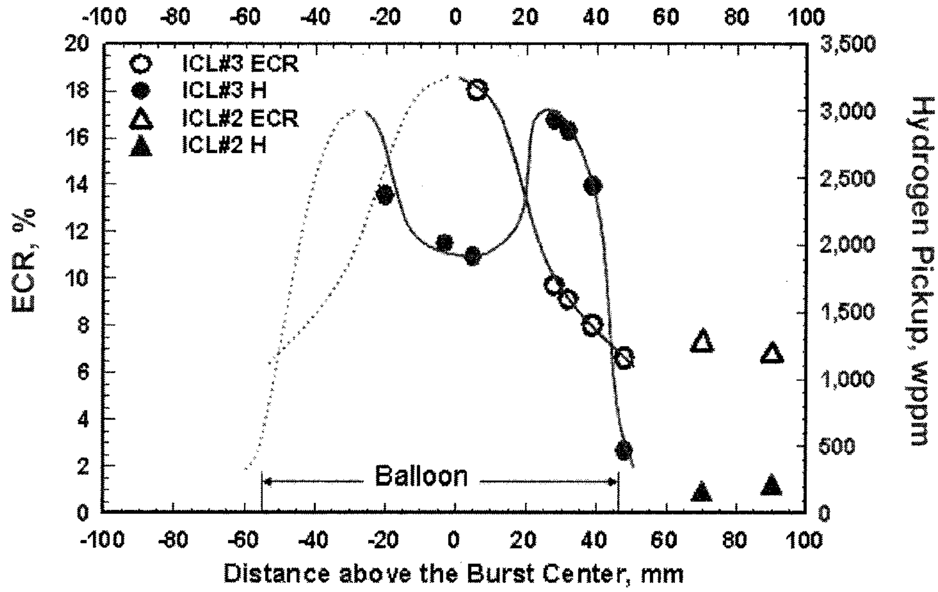
H and O Analyses of ICL#2 and ICL#3



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Summary of ANL Testing Program for Non-Deformed 15 × 15 Zry-4 Rings

- **Baseline Data for Non-Irradiated 15 × 15 Zry-4 Cladding**
  - Weight gain (1204°C); PQD vs. ECR, H-content & test temperature
- **HBR Rod Selection (F07) and Characterization**
  - Gamma scanning, corrosion thickness, hydride morphology
  - Hydrogen- and oxygen-content profiles (axial and circumferential)
- **HBR F07 Oxidation Test Samples and Results**
  - 25-mm-long samples cut and defueled from F07 midplane (-40 to 100 mm)
  - Two-sided steam oxidation tests conducted at ≈1204°C T_{max} to 3, 5, 7, 8, 10% ECR without quench (550±90 wppm H) **8% ECR with quench (800±110 wppm H)**
- **Post-Oxidation (POD) and Post-Quench Ductility (PQD)**
  - POD offset & permanent strains for ring-compression tests at 135°C
  - PQD offset & permanent strains for ring-compression tests at 135°C

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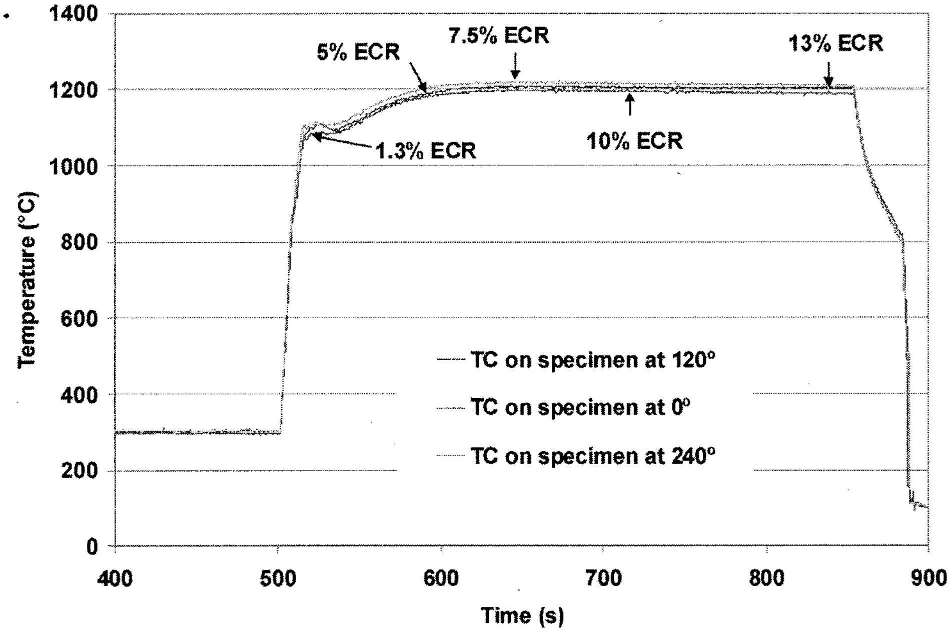
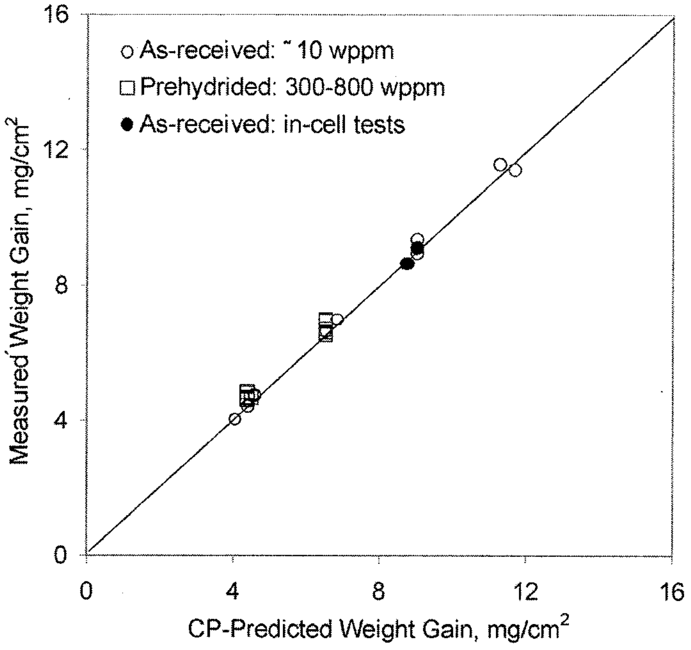
Baseline Data for Non-irradiated 15 × 15 Zry-4 Cladding

- **Weight Gain due to Steam Oxidation at 1204°C**
  - Thermal benchmark results with 3 TCs welded onto sample and holder
    Subsequent tests conducted with 3 TCs welded onto holder above sample
  - Data are in excellent agreement with Cathcart-Pawel (CP) correlation
    Hydrogen content and quench have no significant effect on weight gain
    Note 2 data points at about 8.7 mg/cm^2 : one with & one without quench
- **Post-Quench Ductility of 1204°C-Oxidized/Quenched Zry-4**
  - Nonirradiated 15 × 15 Zry-4 cladding
    Offset & permanent strains vs. ECR based on measured weight gain
    Ductile-to-brittle trans. ECR: 8% at RT, 11.5% at 100°C, 14% at 135°C
    Offset strain vs. ECR calculated with CP correlation: trend curves
  - Pre-hydrided, non-irradiated 15 × 15 Zry-4 cladding; 135°C tests
    5% CP-ECR: ductile-to-brittle transition H-content ≈ 600 wppm
    7.5% CP-ECR: ductile-to-brittle transition H-content ≈ 400 wppm

Temperature History for 1204±10 ° C Steam Oxidation of 15 × 15 Zry-4 Post-Quench-Ductility Samples



Weight-Gain Results for Nonirradiated 15 × 15 Zry-4 (0604 and 1204 Test Trains)



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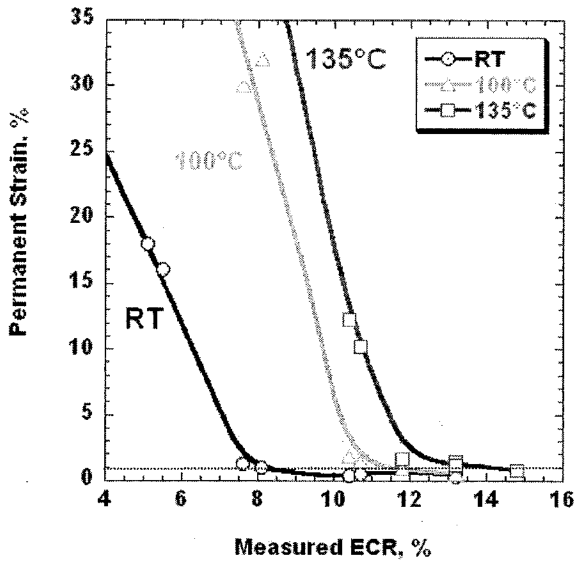
Ring-Compression Permanent Strain vs. Measured ECR for 15x15 Zry-4 Oxidized at 1204 ± 10°C



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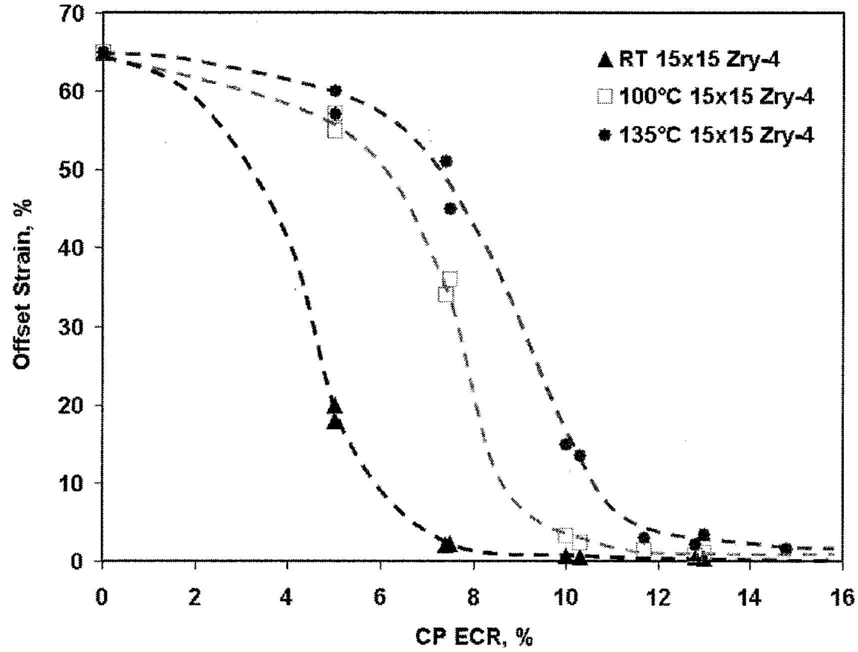
Offset Strain vs. CP-ECR for As-Received 15x15 Zry-4 Oxidized at 1204±10°C



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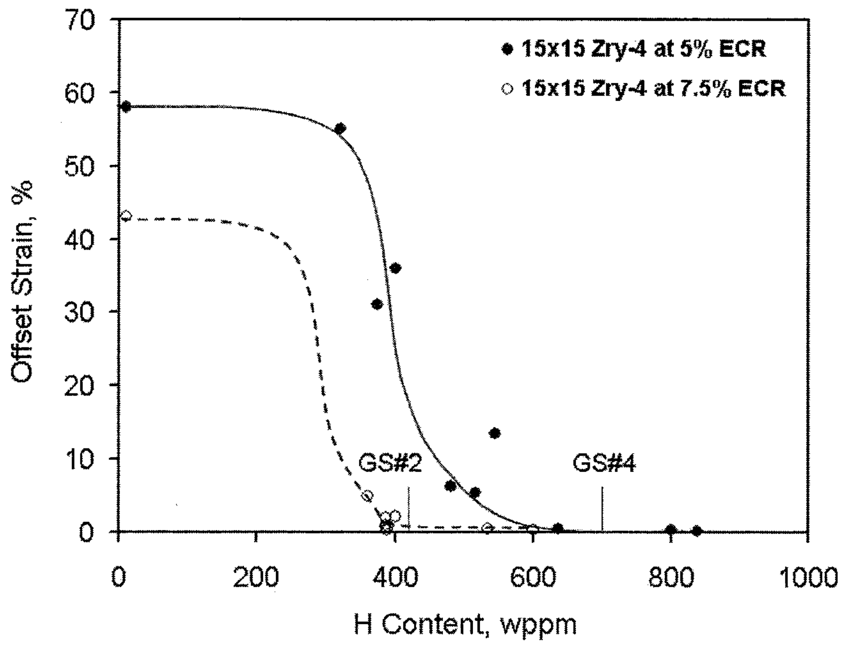
Post-Quench Ductility at 135°C vs. Hydrogen Content for Prehydrided 15x15 Zry-4 Oxidized at 1204±10°C



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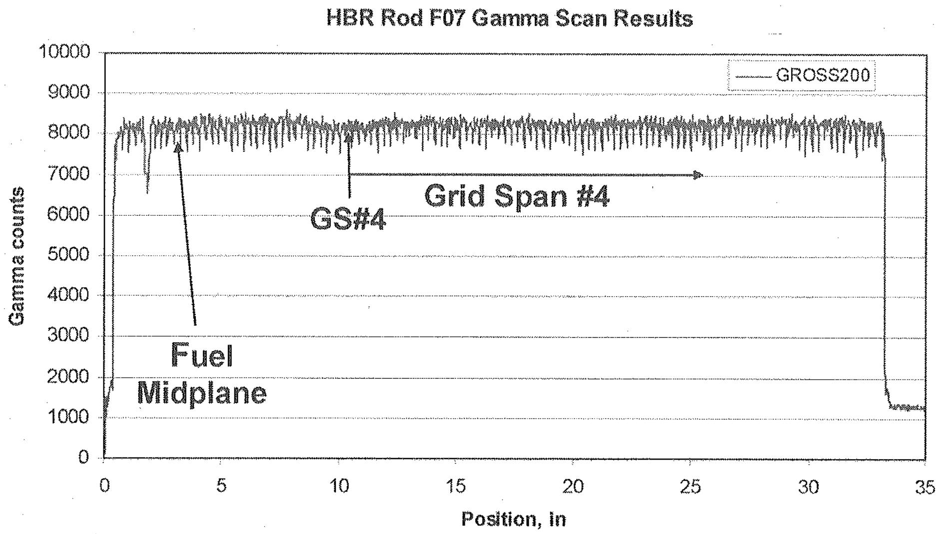
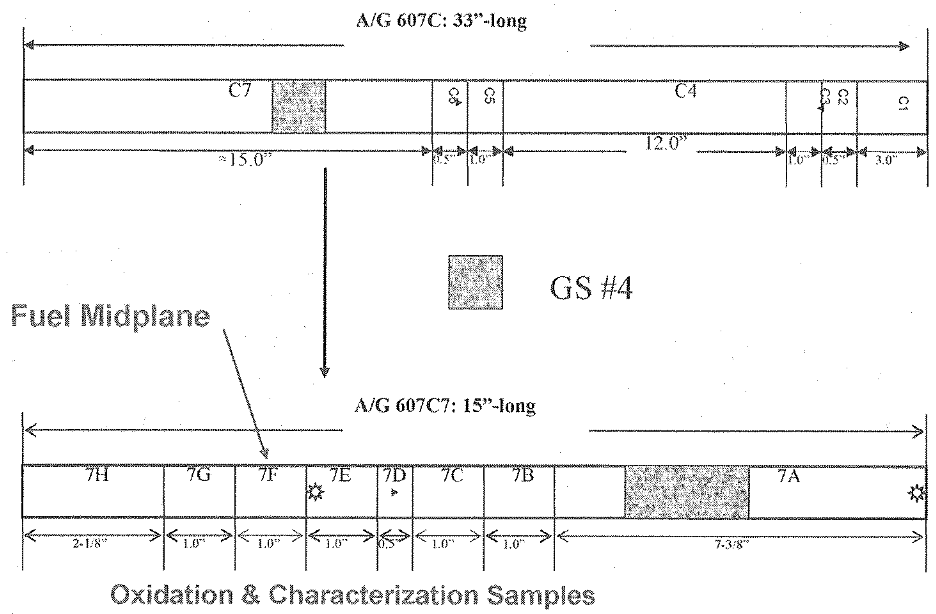
Characterization of HBR Rod F07

- **Gamma Scanning of Middle Segment (C)**
- **Characterization, Oxidation, LOCA Sample Locations**
  - Metallography at 50 mm, 300 mm & 660 mm above fuel midplane
  - H & O analysis at 25 mm, 320 mm and 650 mm above fuel midplane
  - Oxidation samples at -40 to 100 mm above fuel midplane
- **Characterization Results**
  - **Fuel morphology: heat generation and T are not axisymmetric**
  - **Metallography at +50 mm : oxide layer thickness = 71±5 μm**
    Cladding thickness = 712 μm
    Hydride morphology: relatively uniform in hoop (θ) direction
  - **H-content: 550±90 wppm at -65 to -40 mm below midplane**
  - O-content: 2.08±0.19 wt.% at +35 mm above midplane
    in good agreement with corrosion layer thickness for PB = 1.75

Gamma Scan Results: Mid-segment (C) of HBR Rod F07 Note: Dips at Pellet-Pellet Interfaces Every ≈7 mm



Sectioning Diagram for HBR Rod F07 Segment C



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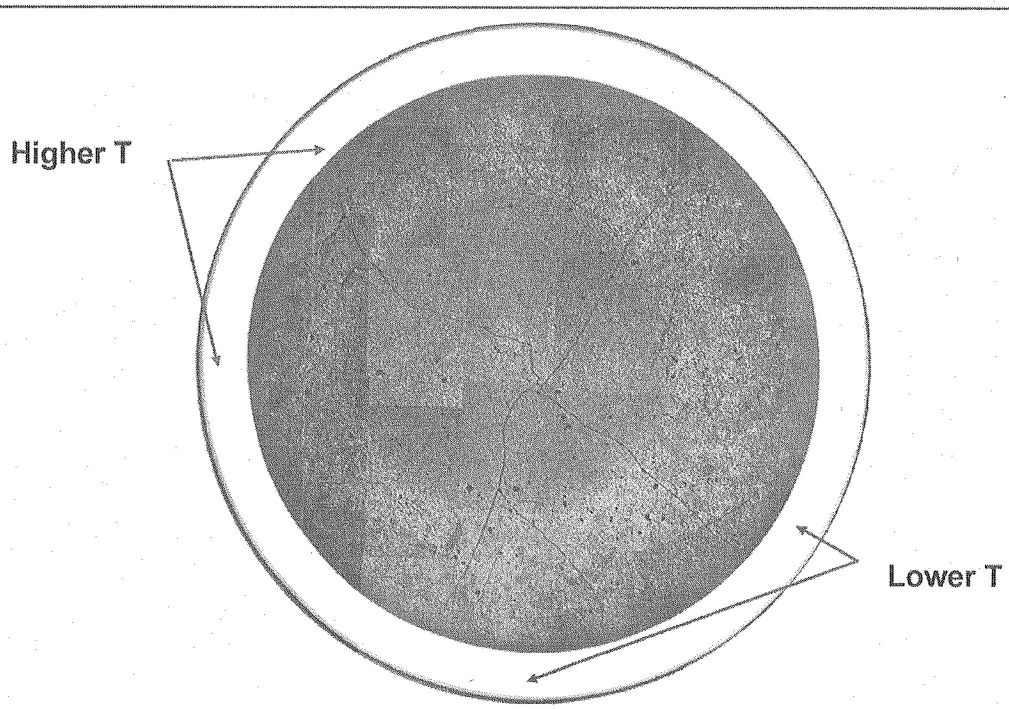
Fuel Mosaic for HBR Rod F07 Cladding at ≈50 mm above Fuel Midplane



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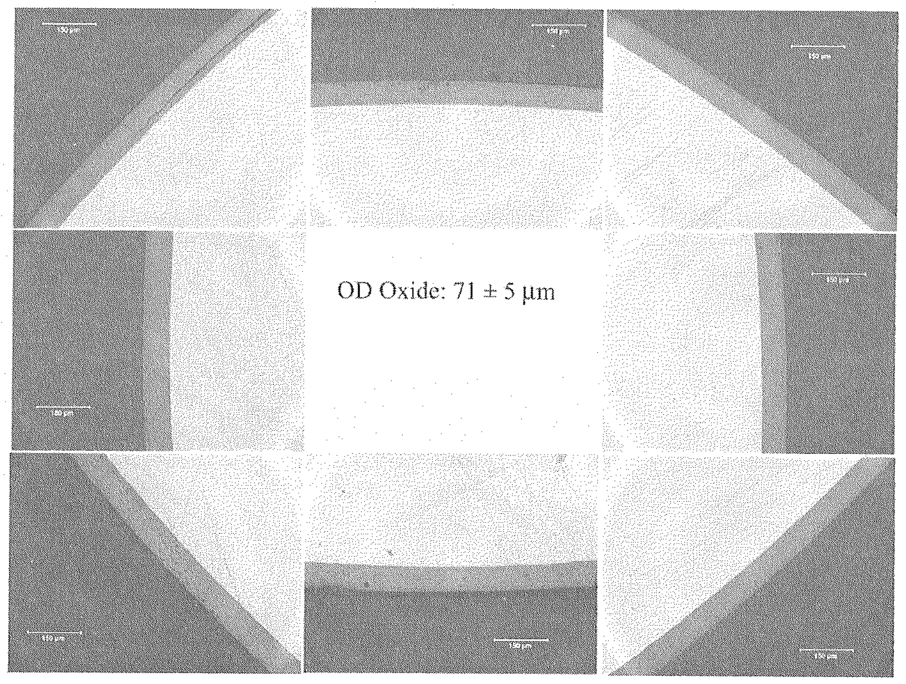
Oxide Layer Thickness for HBR Rod F07 Cladding at ≈50 mm above Fuel Midplane



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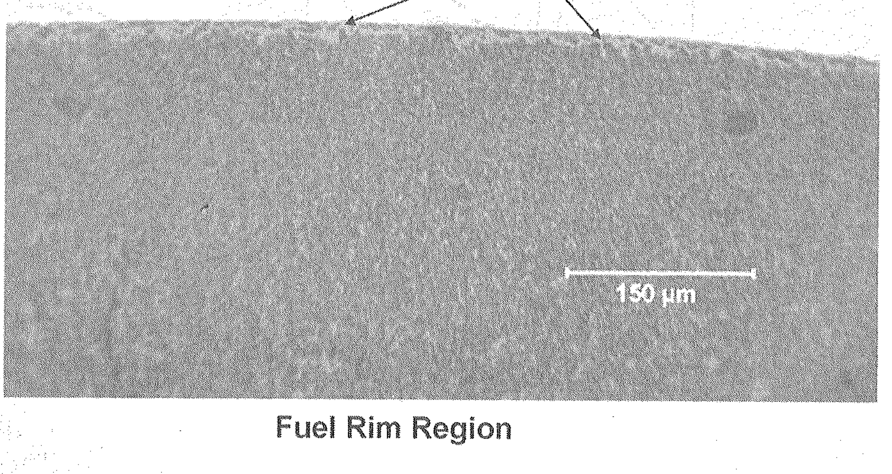
Fuel-Cladding Interface for HBR Rod F07 Cladding at ≈50 mm above Fuel Midplane

Fuel-Cladding Bond: ≈11±4 μm
(Remains after Defueling in RT HNO_3)

Cladding



Fuel Rim Region

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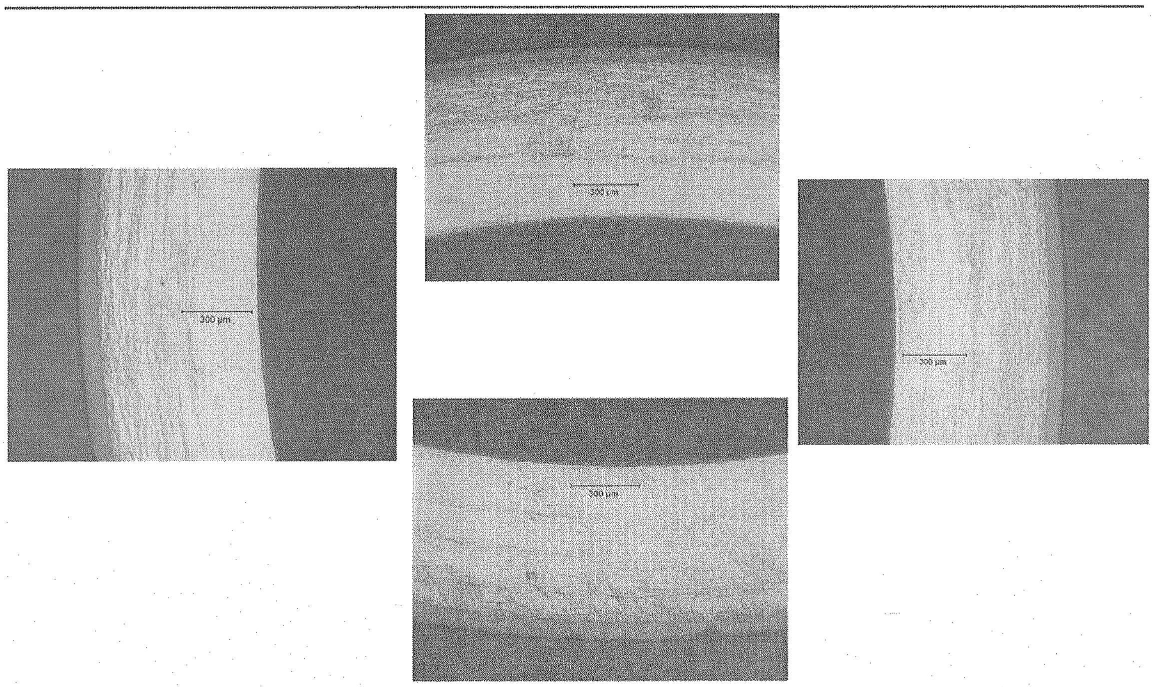
Hydride Morphology for HBR Rod F07 Cladding at ≈50 mm above Fuel Midplane (Low Mag.)



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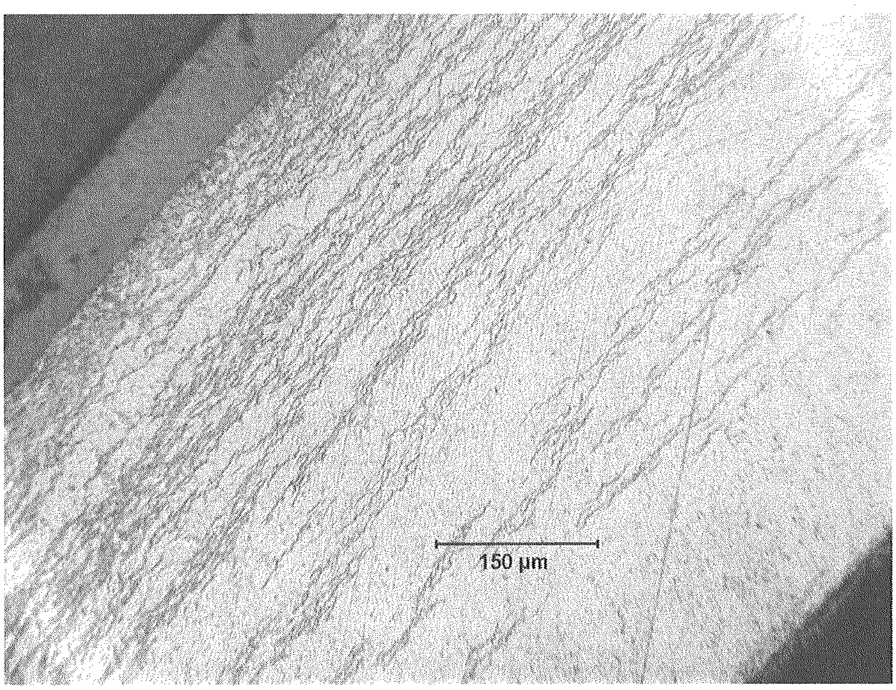
Hydride Morphology for HBR Rod F07 Cladding at ≈50 mm above Fuel Midplane (High Mag.)



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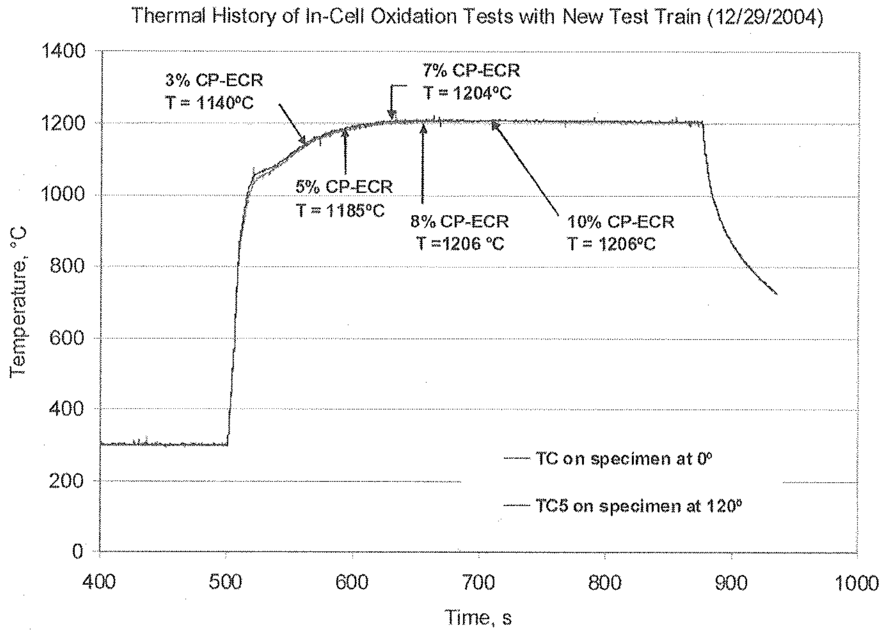
New Test Train: Thermal Benchmark Results for Nonirradiated 15 ×15 Zry-4



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Post-Oxidation Ductility of HBR Cladding at 135°C: 8% CP-ECR Sample

- **8% CP-ECR with T = 1206°C Prior to Cooling**
  - Note: sample from ≈320 mm above midplane (74 μm corros. layer)
  - Post-test hydrogen: **545±80 wppm** (2-mm rings sides of sample)
  - Metallography

    Outer-surface: steam oxide layer ≈20 μm, alpha layer ≈27 μm

    Inner-surface: steam oxide layer ≈26 μm, alpha layer ≈25 μm

    Quant. Met. → WG = 4.7 mg/cm^2, ECR = 5.8% vs. 8% CP-ECR

    Compare to pre-H (600 wppm) Zry-4: 6.3 mg/cm^2, 7.5% ECR

    Clearly establish partial protection of corrosion layer
  - Ring-compression ductility = 3.8% offset strain, 2.9% permanent strain
  - **Assessment: low ductility**

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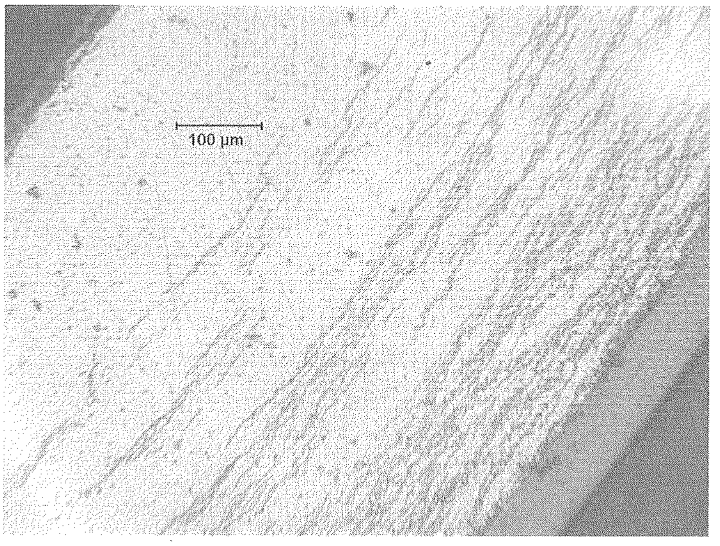
Hydride Morphology of Slow-Cooled 8% CP-ECR Sample Prior to Steam Oxidation



74-µm Corrosion Layer
545±80 wppm H (post-test)

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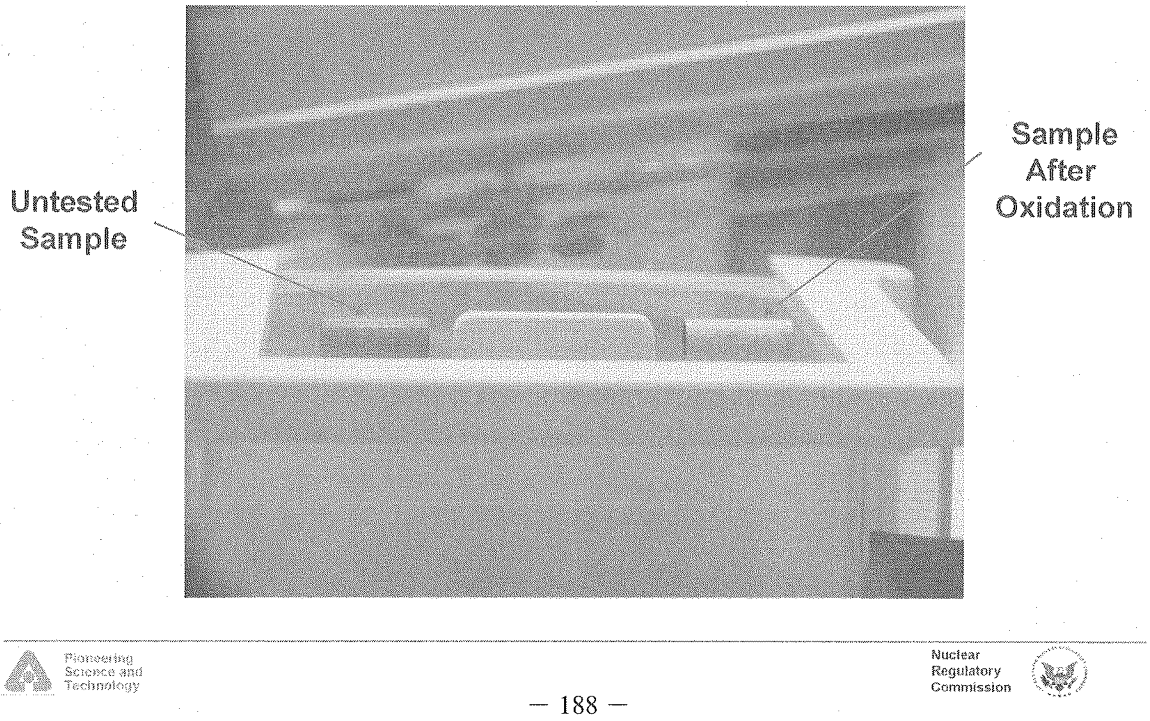
Post-Test vs. Pre-Test Visual Appearance of Sample



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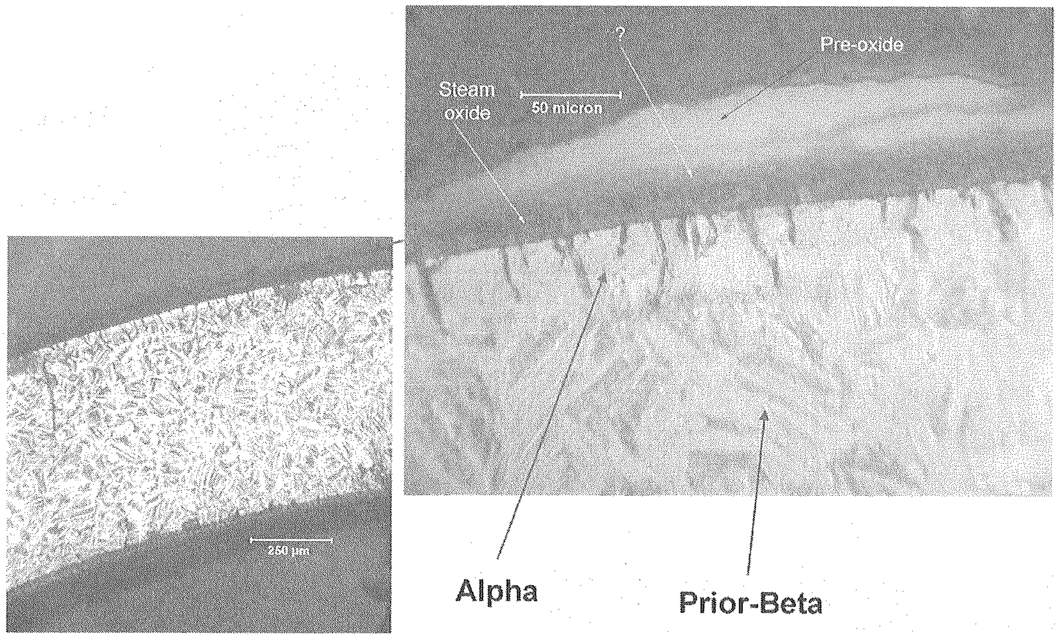
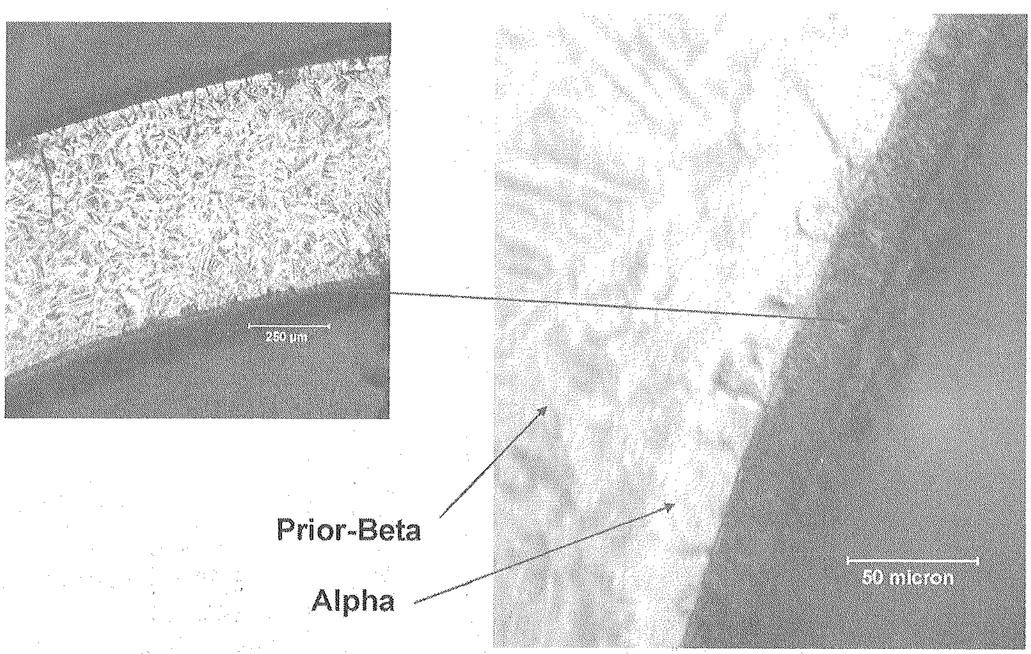
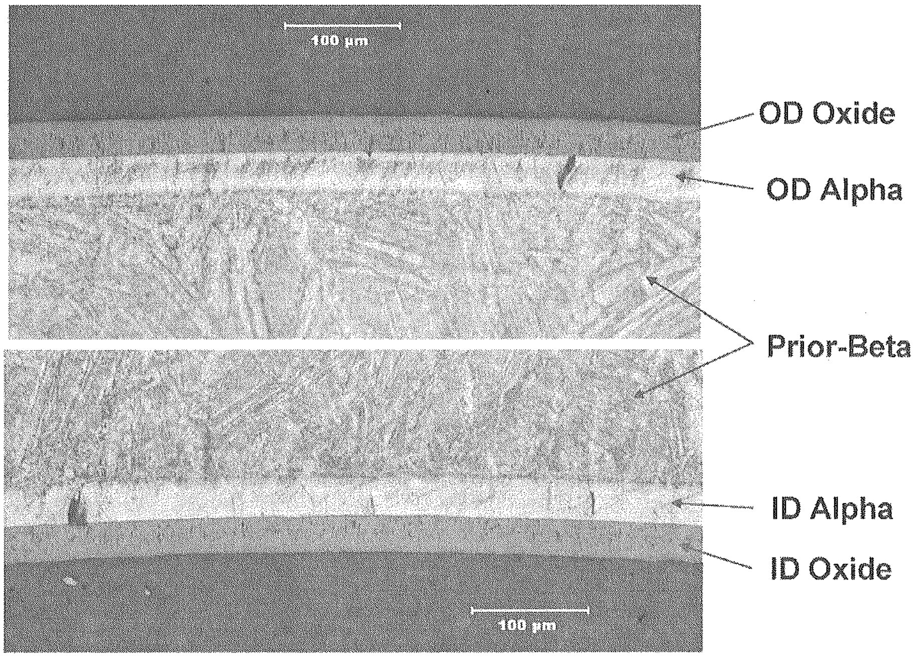
Metallography of 8% CP-ECR Sample – Outer Surface



Metallography of 8% CP-ECR Sample – Inner Surface



Metallography of 7.5%-ECR Pre-Hydrided (600 wppm) Zry-4 Sample



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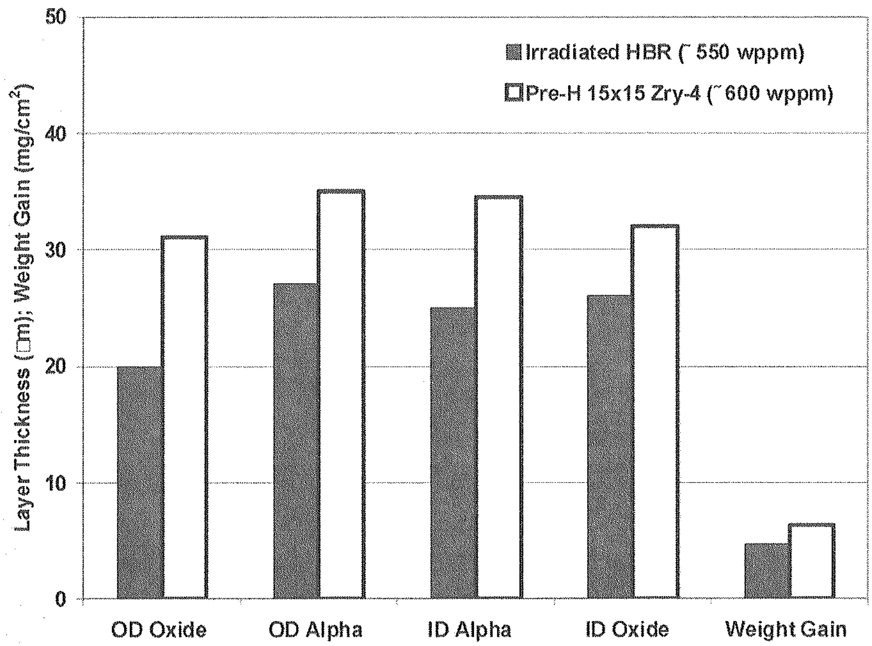
Comparison of Metallography for 8% CP-ECR HBR and 7.5%-ECR Pre-Hydrided Zry-4



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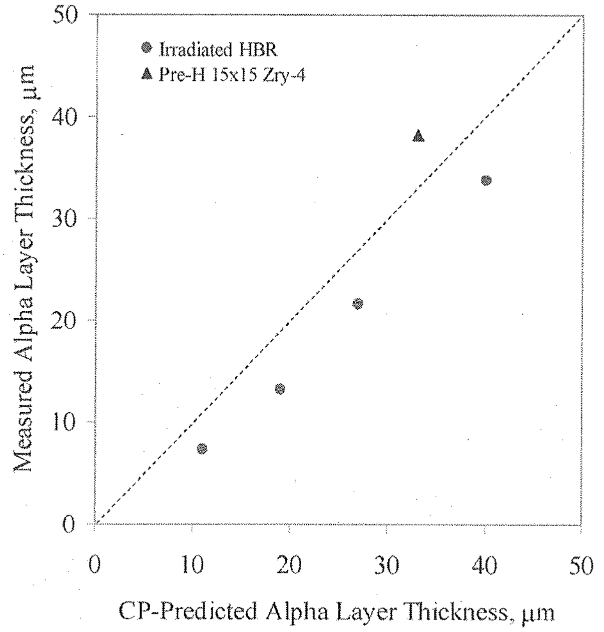
Measured vs. Predicted Average Alpha Layer Thickness Preliminary Results for (OD+ID)/2 Alpha Layer



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Post-QUENCH Ductility of HBR 8% CP-ECR Sample

- **Axial Location of Sample ≈660 mm above Midplane**
- **Corrosion Layer = 95 μm**
- **Post-Quench Hydrogen Content = 800±110 wppm**
  - Consistent with hydrogen contents for A02 (750±90) & R01 (770±125)
  - Consistent with hydride morphology
- **Post-Quench Ductility ≈0%**
- **Comparison to Slow-Cooled 8% CP-ECR Sample**
  - Higher hydrogen content sufficient to embrittle sample at 5.8% measured ECR
  - Currently, no mechanism can be identified to decrease ductility by quenching from 800°C
  - Slow cooling from 800°C should enhance embrittlement by allowing more time for hydrides to precipitate at prior-beta interfaces

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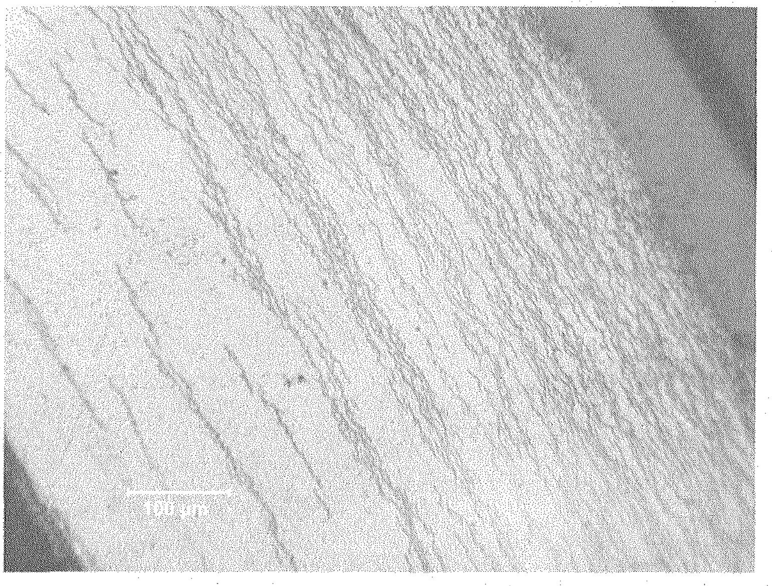
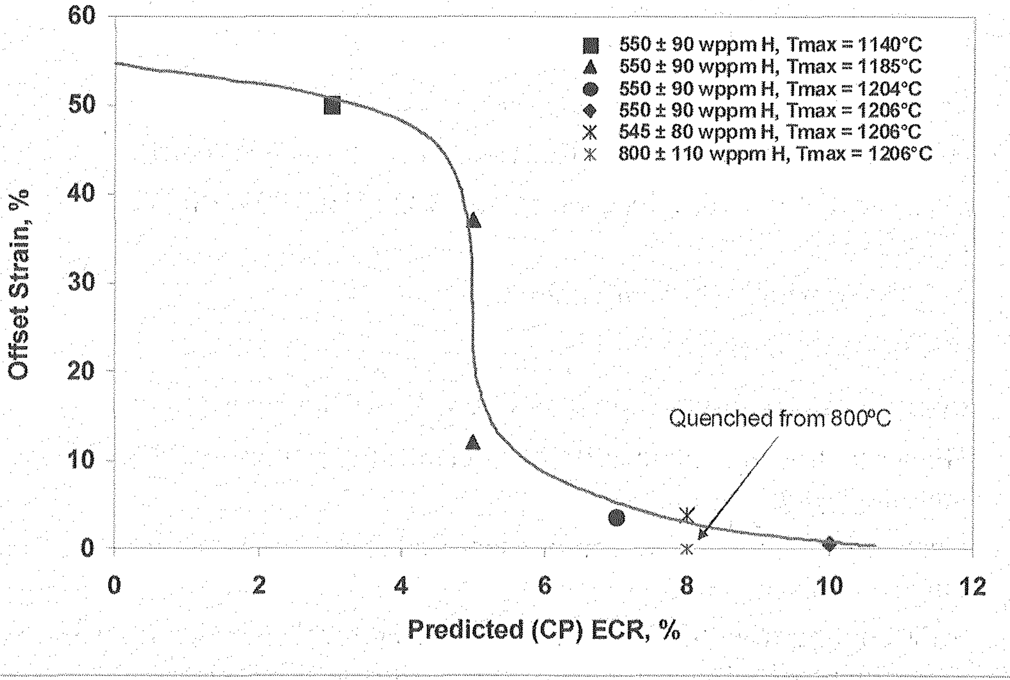
Hydride Morphology of Quenched 8% CP-ECR Sample Prior to Steam Oxidation

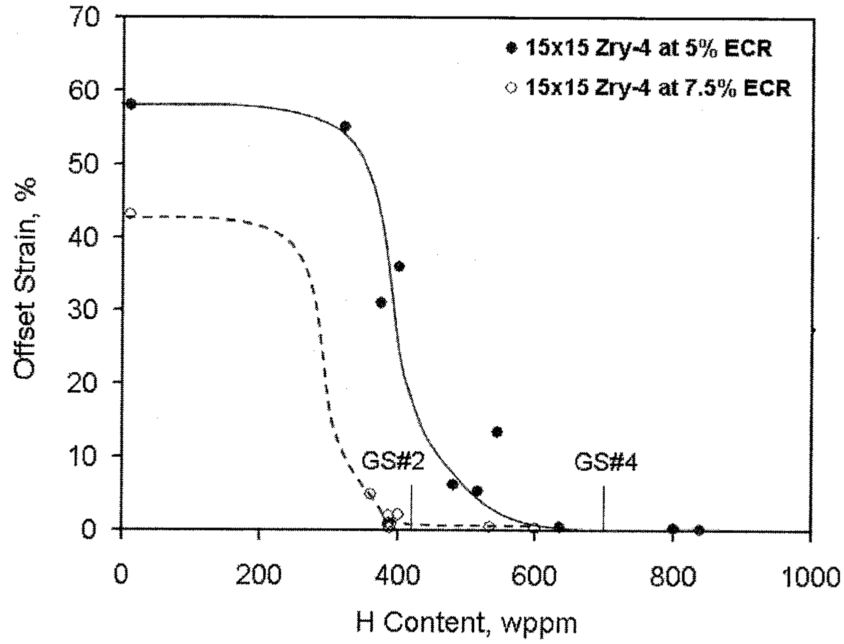


95-μm Corrosion Layer
800±110 wppm H

Summary of Post-Oxidation Ductility (Offset Strain) at 135°C vs. CP-ECR for High-Burnup Zry-4 Cladding



Post-Quench Ductility at 135°C vs. Hydrogen Content for Prehydrided 15x15 Zry-4 Oxidized at 1204±10°C



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Discussion of Results for High-Burnup HBR Cladding

- **Characterization of As-Irradiated Rod F07 Cladding**
  - Reasonable to assume ductility data correspond to ≈550±90 wppm H for 3, 5, 7 and 10% CP-ECR samples cut from near the midplane
  - 8% CP-ECR samples

    Slow-cooled sample from 320 mm above midplane (545±80 wppm)

    Quenched sample from 660 mm above midplane (800±110 wppm)
- **Effects of Corrosion Layer on Steam Oxidation Kinetics**
  - Corrosion layer

    Source of oxygen for growth of alpha layer & O-increase in beta layer

    Partial protection from steam oxide growth (ID > OD oxide layer)
  - Metallography of 8% CP-ECR sample → 5.8% measured ECR
- **Post-Oxidation & Post-Quench Ductility Results**
  - Appear consistent with pre-H Zry-4 in terms of measured ECR

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Future Plans

- **Post-Quench Ductility of High-Burnup Zry-4**
  - 1-sided oxidation tests
    T_h = 1204° C, 1100° C & 1000° C
    Fast ramp to T_h - 50° C, slower ramp to T_h
    CP-ECR = 3, 5, 7.5, 10% at 1204° C; 5, 7.5, 10% for 1000 & 1100° C
    Run tests without quench at 800° C; repeat one test with quench
    Ring-compression tests at 135° C & 100° C
    (CEA-EdF-FANP baseline data for prehydrided Zry-4 are desirable)
  - LOCA integral tests
    1 test to burst with slow cooling → balloon-burst characterization
    3 tests at 3, 5, 7.5% CP-ECR in non-ballooned region (TBD)
    4-point-bend test (RT), ring-compression tests at 135° C & 100° C

Future Plans (Cont'd)

- **Post-Quench Ductility of High-Burnup ZIRLO**
  - Perform oxidation-quench-compression test matrix used for Zry-4
    ZIRLO from North Anna → Studsvik → ANL
    1204° C, 1000° C & 1100° C tests
    (Baseline data on nonirradiated, prehydrided ZIRLO?????)
  - Perform LOCA integral test matrix
    Fuel at ≈50 GWd/MTU (<62 GWd/MTU)
- **Post-Quench Ductility of High-Burnup M5**
  - Perform oxidation-quench-compression test matrix used for Zry-4
    European M5 → Studsvik → ANL
    1204° C, 1000° C & 1100° C tests
    Baseline data on nonirradiated, prehydrided M5 desirable (135° C)
  - Perform LOCA integral test matrix
    Fuel at ≈50 GWd/MTU

Session 4-3

HIGH BURNUP FUEL BEHAVIOR UNDER LOCA CONDITIONS

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A systematic research program is being conducted at the Japan Atomic Energy Research Institute (JAERI) in order to evaluate the influence of burnup extension on fuel behavior under LOCA conditions. Various data have been obtained regarding oxidation rate, ballooning/rupture behavior and fracture during quench. Pending questions, however, remain in studies on high burnup fuel behavior during LOCA. This paper summarizes the information which has been clarified in the JAERI's program and the subjects to be solved.

It has been shown that both rupture strain and temperature are affected by hydrogen absorption. These hydrogen effects can be explained by alterations of phase transformation temperature and cladding strength with hydrogen concentration. On the other hand, neutron irradiation and corrosion have negligible effect on ballooning and rupture behavior.

Corrosion layer formed during the reactor operation has a protective effect on high temperature oxidation under LOCA conditions. The effect becomes smaller with increase in both the oxidation temperature and time. In addition, the protective effect may not be expected in ballooned claddings since corrosion layer cracks during the ballooning.

The Japanese criterion on fuel safety for LOCA is based on a fracture/no-fracture threshold of oxidized cladding which was experimentally determined under simulated LOCA conditions. In order to evaluate the fracture threshold of high burn-up fuel rods, JAERI performs integral thermal shock tests simulating LOCA conditions. The numerous tests with unirradiated claddings showed that fracture threshold in terms of the oxidized fraction decreases as both the initial hydrogen concentration and axial restraint load increase. The threshold is higher than 20% cladding oxidation, and is irrespective of the hydrogen concentration when the restraint load is below 535 N. Six tests were recently performed with PWR fuel rods, irradiated to 39 and 44GWd/t (rod average). It was revealed that the fracture threshold is not reduced so significantly by irradiation to the examined burnup level.

The following subjects are to be investigated in order to promote a better understanding of high burnup fuel behavior and to evaluate safety of the high burnup fuel properly.

- Transient secondary hydriding in irradiated claddings,
- Effect of further burnup extension and use of new alloys, and
- Test methodology to evaluate cladding performance and safety in LOCA.

High burnup fuel behavior under LOCA conditions

Fumihisa Nagase

Japan Atomic Energy Research Institute

Fuel Safety Research Meeting
Tokyo, Japan
March 2-3, 2005

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JAERI's program on high burnup fuel behavior under LOCA conditions

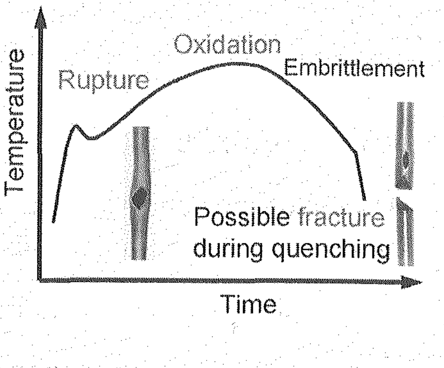
Objective

promote a better understanding of high burnup fuel behavior during LOCA for regulatory judgment

Approach

investigate influence of burnup extension (increasing corrosion, hydrogen absorption and irradiation) on

- Oxidation kinetics
- Ballooning / Rupture
- Embrittlement of cladding (Fracture during quenching)



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Conducted tests

- Integral test with short test rod
  - simulate the LOCA sequence
  - examine all the behavior including ballooning, rupture, oxidation and fracture

- Separate-effect tests with small specimen
  - on oxidation rate
  - on cladding embrittlement

with non-irradiated cladding (pre-hydrided or pre-corroded) and irradiated cladding from spent fuel

2

This presentation will try to discuss

✓ What has been clarified?

✓ What should be investigated?

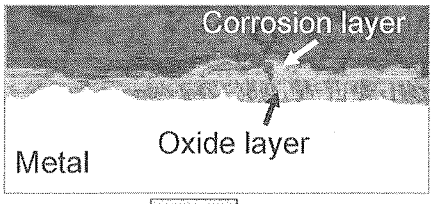
- on high burnup fuel behavior under LOCA conditions
- with information obtained from JAERI's program.

3

Oxidation kinetics (1)

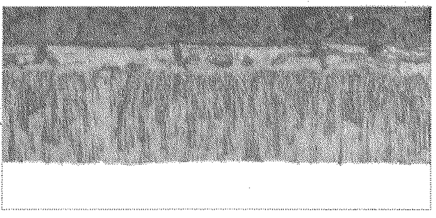
Protective effect of corrosion layer pre-formed in reactor
at lower temperatures for shorter time range
(Local oxide growth at cracked position)

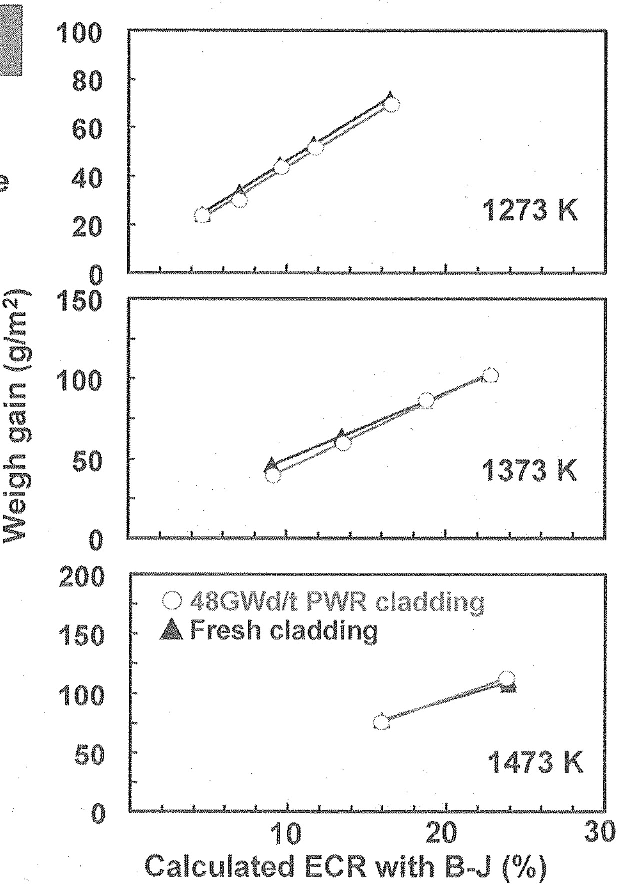
1273 K
7% ECR



Decrease of protective effect
as temperature and time increase
(Uniform growth of oxide layer)

1373 K
19% ECR





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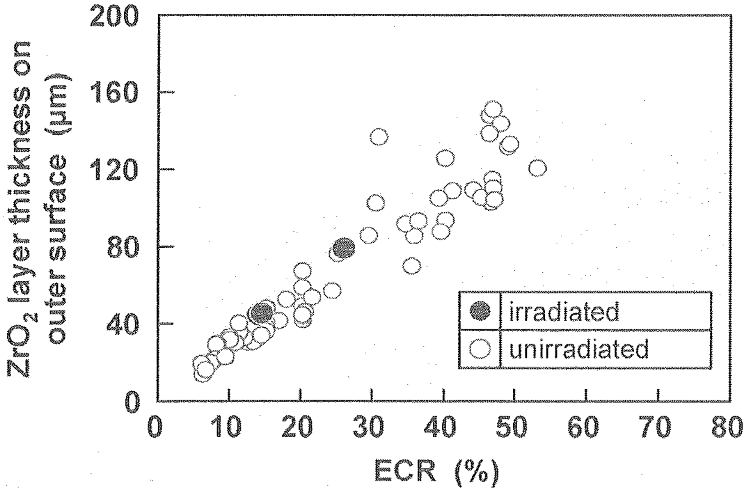
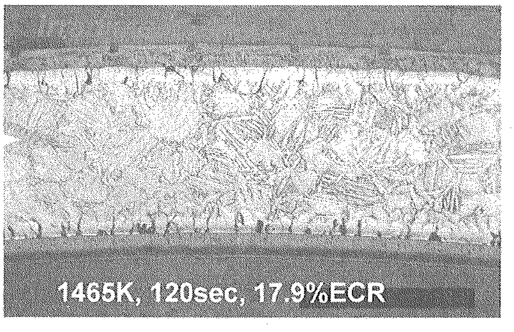
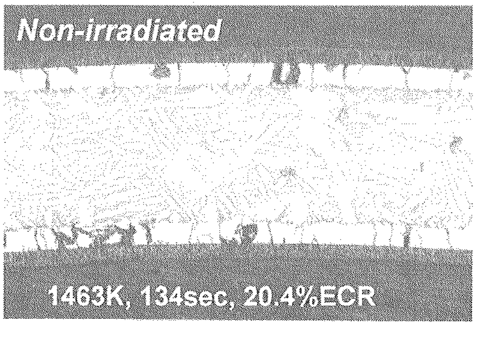
Oxidation kinetics (2) – Oxidation after ballooning -



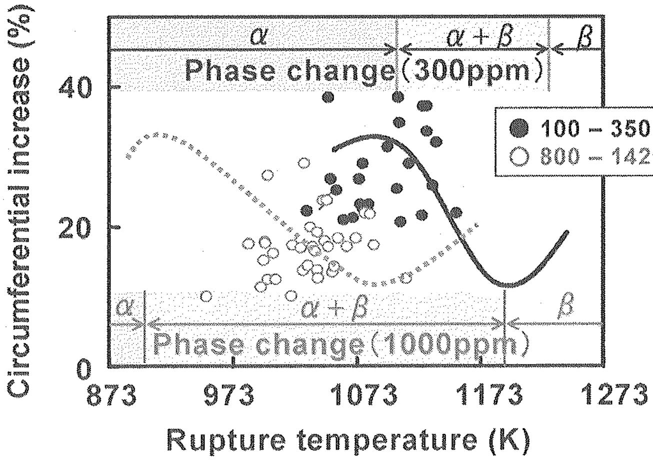




- Oxide thicknesses are in the same range in unirradiated and irradiated specimens. Effect of corrosion layer was not observed.
- Small cracks can be seen in the pre-formed oxide layer. The cracks may reduce the protective effect of corrosion layer.

5

Rupture Behavior (1) – unirradiated cladding -

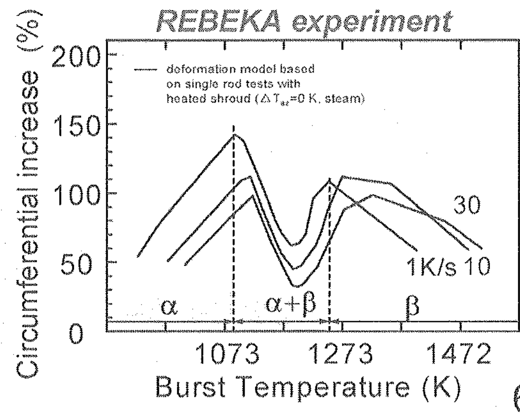


- Data from tests with unirradiated specimens in two different hydrogen ranges (Other conditions including initial pressure are the same)

➠Higher circumferential increase in less hydrided cladding

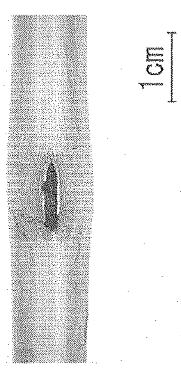
- The Increase depends on rupture temperature, hence, on phase structure at rupture.
- Phase transformation temperatures change with hydrogen concentration.

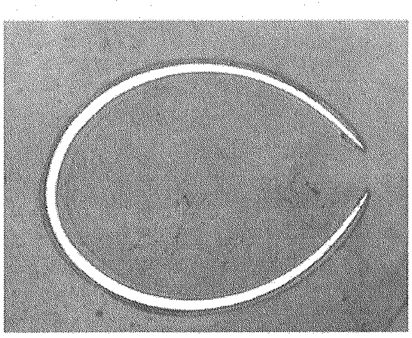
➠Rupture behavior, increase of circumference, depends on hydrogen concentration.



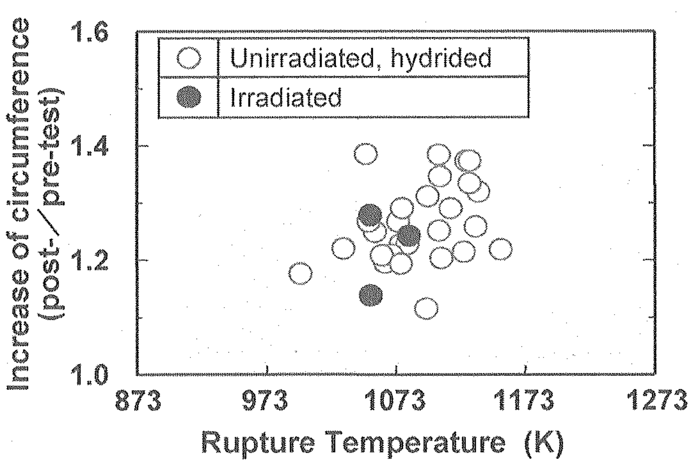
6

Rupture Behavior (2) – Irradiated cladding -





Radial cross section at rupture position



- Visual appearances are similar to those in unirradiated specimens.
- Increase of circumference depends on rupture temperature. Results from irradiated specimens are equivalent to those from unirradiated samples with similar hydrogen concentration.

7

Obtained information on oxidation kinetics and ballooning/rupture behavior

Oxidation kinetics

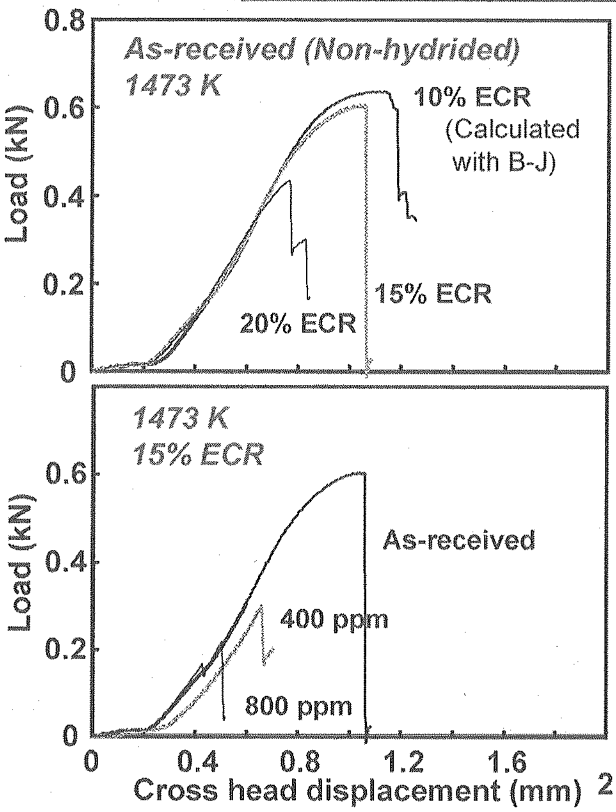
- Corrosion layer has protective effect for oxidation under LOCA conditions, but it may be negligible in safety analysis.
- Effect of hydrogen absorption is smaller than 5% for the concerned conditions (<800 ppm, <1800 s)

Ballooning and Rupture behavior

- Increase of circumference due to ballooning is affected by hydrogen absorption.
- Rupture temperature may be somewhat decreased with increase in hydrogen concentration, because strength of Zircaloy is reduced by hydrogen absorption.

8

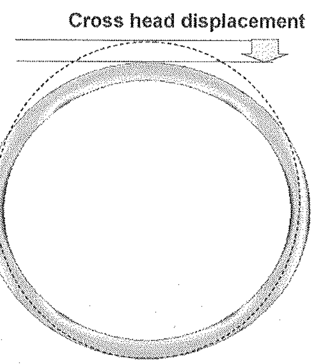
Embrittlement – Hydrogen effect -



As- received → Oxidized and quenched under LOCA conditions

As- received → Hydrided → Oxidized and quenched under LOCA conditions

Oxidized and quenched under LOCA conditions → Ring compression test



Embrittlement of oxidized cladding is enhanced by pre-hydriding

➡ Possible reduction of fracture condition

9

History of LOCA criteria and study in Japan

1970 ······ 1980 ······ 2000 ······

- Establishment of criteria ('75)
  (Limits for cladding oxidation, 1200°C and 15%)
- Revision of criteria* ('81)

Burnup extension

- Ring compression test of oxidized cladding
  ⇩
  Zero-ductility criteria

- Oxidation test of burst cladding
  ⇩
  Finding of "*Inner surface oxidation*" accompanied by significant hydrogen absorption
- Ring compression test of burst and oxidized cladding
  ⇩
  Cladding embrittlement enhanced by hydrogen absorption
- Integral thermal shock test
  - Fresh cladding
  - Ruptured, oxidized and quenched
  - Considering *axial restraint* during quench (Fully restrained)

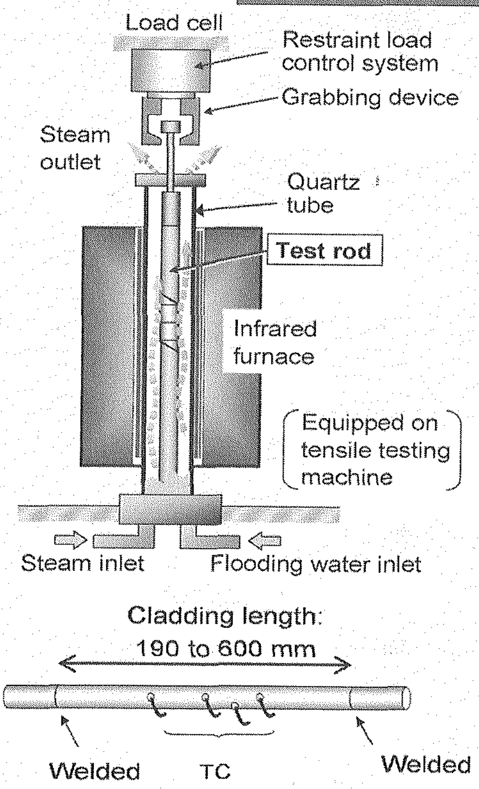
  ⇩
  Confirmed safety margin

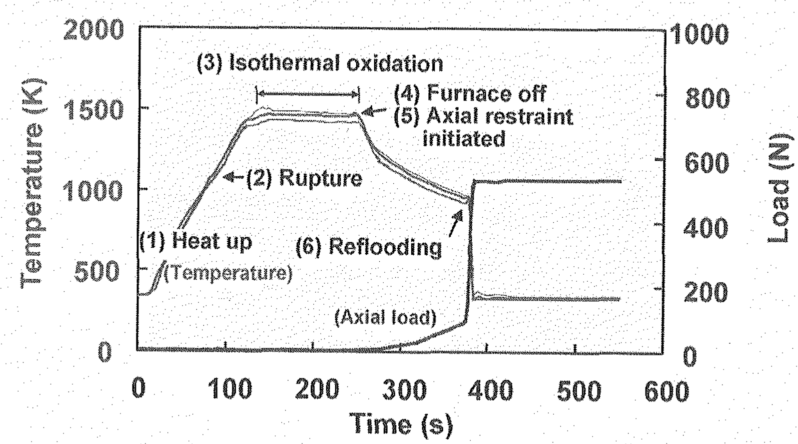
- Integral thermal shock test ('99-)
  - Pre-hydrided
  - High burnup cladding
  - Realistic restraint condition

* Consequently, limits for cladding oxidation remained unchanged.

10

Integral Thermal Shock Test

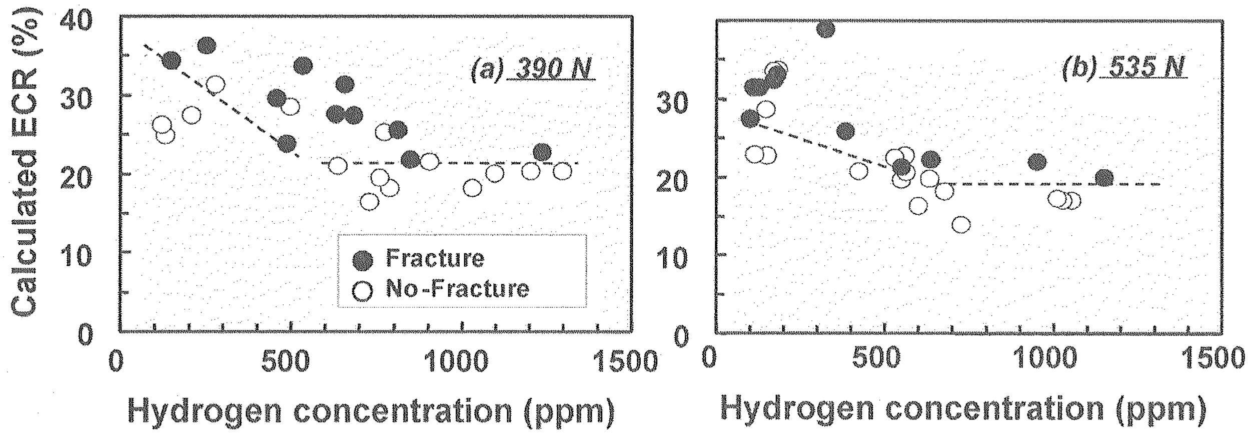




- Test rod is quenched with axial restraint to represent a possible condition of fuel rods between grid positions.
- Maximum tensile load is limited up to ~540 N. The "540 N" is based on measurement of resistant load between deformed or chemically interacted cladding and spacer grid (K. Homma et al., 2001).

11

Oxidation condition for facture on quench as a function of hydrogen concentration
From tests with non-irradiated cladding

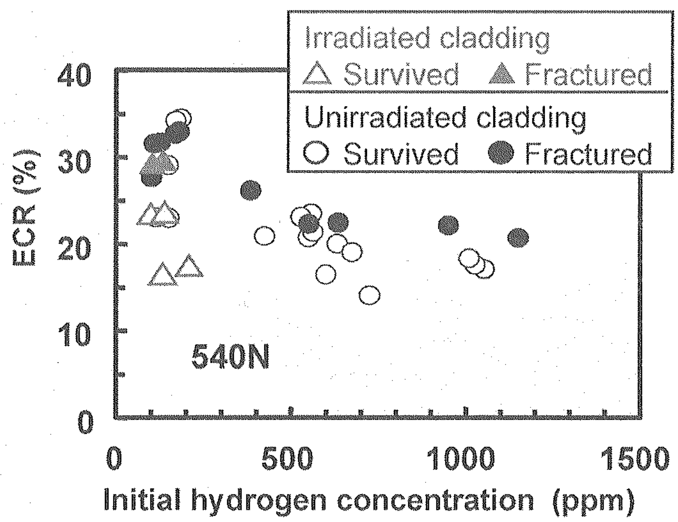


- The fracture threshold decreases as both the initial hydrogen concentration and axial restraint load increase.
- Under the realistic restraint load condition, the threshold is higher than 20% ECR irrespective of the hydrogen concentration.

ECR was calculated with the Baker-Just equation, based on reduced cladding thickness after ballooning measure at rupture position

12

Fracture / No-fracture threshold Relevant to ECR and Initial Hydrogen Concentration
From tests with non-irradiated cladding



17 × 17 Type PWR cladding

Cladding material : Low tin Zircaloy-4

Rod average burnup: 39 – 44 GWd/t

Estimated corrosion layer thickness: 15 – 25 μm

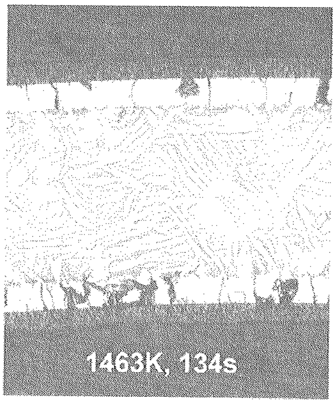
Estimated initial hydrogen content: 120 - 210 ppm

Test rods were quenched under restrained condition. Maximum restraint load was about 540 N.

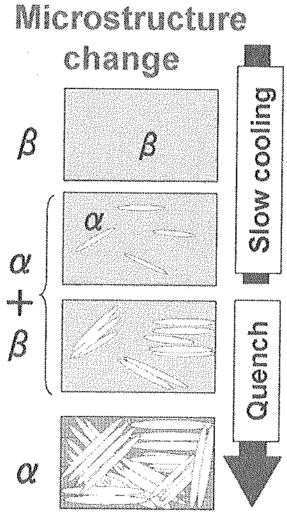
The present result indicates that failure boundary is not reduced significantly by PWR irradiation in the examined burnup level.

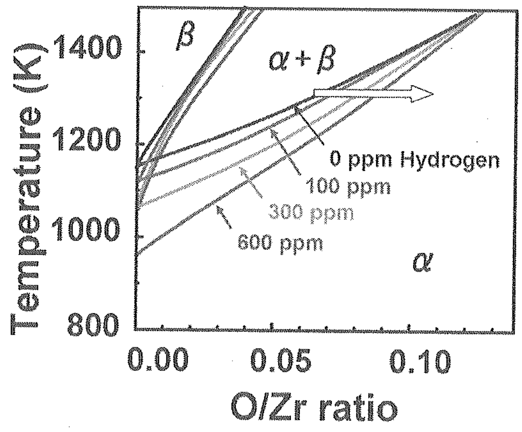
13

Enhanced embrittlement of oxidized cladding by hydriding



Accumulation of hydrogen in beta phase. (Much higher solubility)





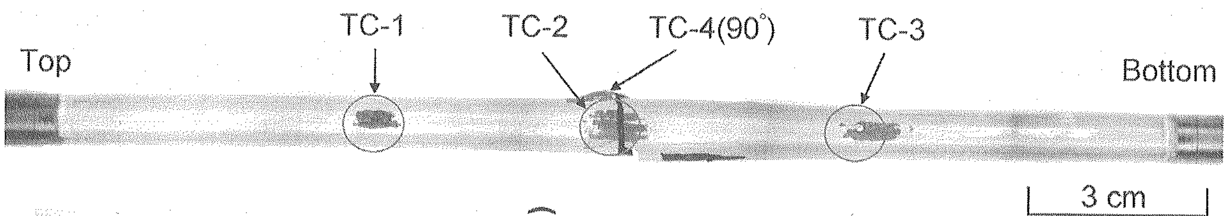
Phase boundary shift to higher oxygen concentration with increase in hydrogen content
(Cooperative research with Osaka Univ.)

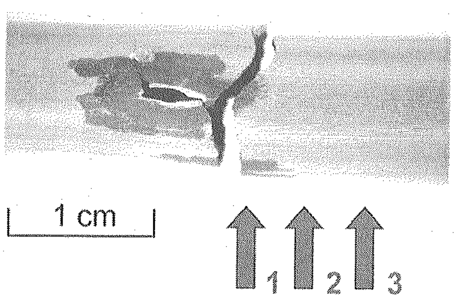
Direct influence
Distribution of very fine hydrides in prior-beta layer

Indirect influence
Increase of oxygen concentration in alpha phase
→ Reduction of ductility

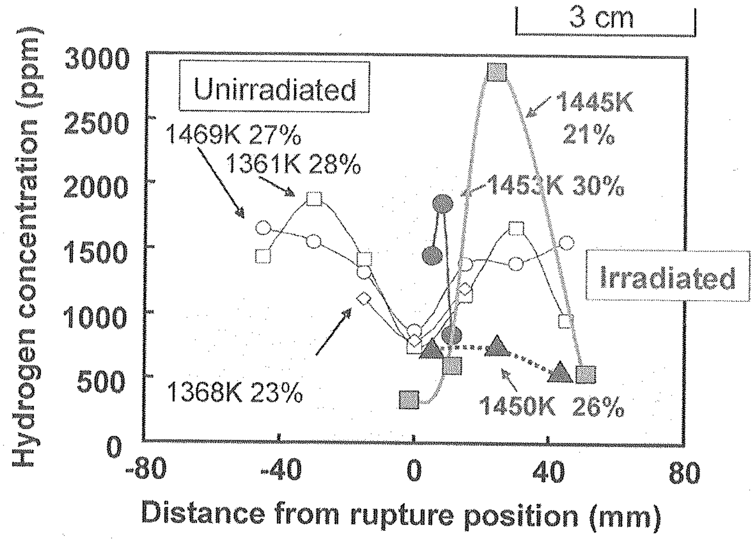
14

Fracture Morphology and Secondary Hydriding





Hydrogen concentration
1. 1450 ppm
2. 1850 ppm
3. 840 ppm



- The transient secondary hydriding is localized at 30 to 50 mm from the burst location in unirradiated claddings. More towards the burst location in irradiated claddings?

15

What should be investigated further?

- **investigate transient secondary hydriding in irradiated claddings**
- investigate effect of further burnup extension and use of new alloys
- develop or modify test methodology to evaluate cladding performance and safety in LOCA

16

What should be investigated further?

- investigate transient secondary hydriding in irradiated claddings
- **investigate effect of further burnup extension and use of new alloys**
- develop or modify test methodology to evaluate cladding performance and safety in LOCA

17

Future plan

Integral tests with

- **63 to 79 GWd/t PWR cladding (ZIRLO™ , M5, MDA, NDA)**
- **63 GWd/t BWR cladding (Zircaloy-2)**

Burnup:	Rod average
MDA:	Zr-0.8Sn-0.2Fe-0.1Cr-0.5Nb developed by Mitsubishi Heavy Industries, Ltd
NDA:	Zr-1.0Sn-0.27Fe-0.16Cr-0.1Nb-0.01Ni developed by Nuclear Fuel Industries, Ltd. and Sumitomo Metal Industries, Ltd.

18

What should be investigated further?

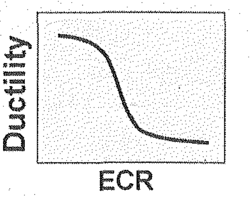
- investigate transient secondary hydriding in irradiated claddings
- investigate effect of further burnup extension and use of new alloys
- **develop or modify test methodology to evaluate cladding performance and safety in LOCA**

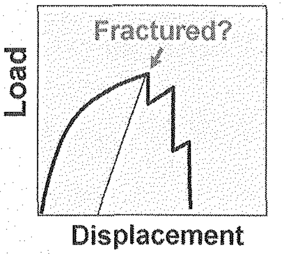
19

Test methodology

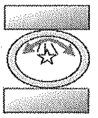
- **Integral test is best to evaluate the fuel behavior as well as safety. However, it needs much work and time.**
- **Are mechanical test substitute for integral test?**
- **Ring compression test is simple and can produce numerous data, but**
  - ▶ **Well-defined 'Zero-ductility', and**
  - ▶ **Standardization of test methodology**

  are needed.
- **Stress condition should be considered on adopting mechanical tests for predicting cladding embrittlement.**
  For example, ring tensile test cannot predict 'Zero-ductility' above 15% ECR.



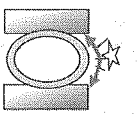


With flat surfaces



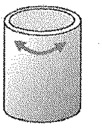
Crack initiates from internal surface

With dented surfaces



Crack initiates from external surface

At quench



Compression at inner surface and tensile at outer surface (Circumferentially uniform)

20

Summary

✓ A systematic research program is being conducted at JAERI for evaluating high burn-up fuel behavior under LOCA conditions.

✓ Data are accumulated for the influence of burnup extension on oxidation kinetics, rupture behavior and cladding fracture during quench.

✓ Recent tests with irradiated PWR cladding (39 to 44 GWd/t) indicated that failure boundary is not reduced significantly by PWR irradiation in the examined burnup level.

✓ Unresolved questions and subjects remain regarding transient secondary hydriding, effect of further burnup extension, test methodology.

21

Acknowledgment

This presentation contains results from a cooperative research program between JAERI and Japanese PWR utilities.

22

5.Session 5 Radionuclides release under severe accident condition

Session 5-1

VEGA Program; Experimental Study on Radionuclides Release from Fuel

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To promote a better understanding of radionuclides release from reactor fuels during severe accident, an experimental program VEGA (Verification Experiments of radionuclide Gas/Aerosol release) has been conducted at Japan Atomic Energy Research Institute (JAERI). In the VEGA program, both of UO_2 and MOX fuels were subjected to experiments with high temperature conditions up to 3150 K and with an elevated pressure up to 1.0 MPa. Ten tests have been performed in order to study and clarify effects of pressure, high temperature resulting in fuel melting, atmosphere, and MOX.

- Release at elevated pressure

  The cesium (Cs) release is suppressed at elevated pressure condition in comparison with atmospheric pressure. The reason for this suppression of release is that diffusion velocity of Cs in open pore decreased at elevated pressure comparing with atmospheric pressure. The decrease of diffusion velocity causes the increase of the Cs concentration in open pore. Then, the concentration gradient in the grain became small and it suppressed the release.

- Release at high temperature

  The Cs release at temperature above 2800 K was enhanced in comparison with that at lower temperature. Although melting point of UO_2 is about 3100 K, the enhanced release of Cs occurs at temperature below that.

- Release in oxidizing atmosphere

  The release of Cs from the fuel in steam condition was slightly higher than that in the He inert condition. Since concentration of interstitial oxygen in the fuel is elevated under the steam condition, resulting higher concentration of vacancies may cause faster diffusion of Cs in a fuel grain. Release of ruthenium (Ru) under the steam condition was obviously greater than that under the He inert condition. The reason for this release enhancement is formation of volatile Ru oxide.

- Release from MOX fuel

  Pu release at above 2800 K was higher by nearly three orders than that in lower temperature. The enhancement of Pu release could be connected with softening and melting of the MOX fuel.

VEGA Program; Experimental Study on Radionuclides Release from Fuel

Tamotsu KUDO

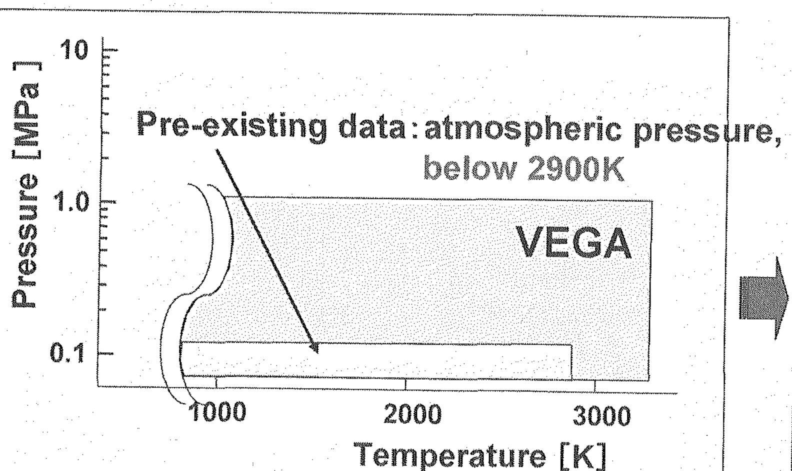
Japan Atomic Energy Research Institute

Presented at Fuel Safety Research Meeting 2005
March 3, 2005, Tokyo

Introduction

Radionuclides release from fuel during severe accident

- Technical bases for safety evaluation



Data for radionuclides release from MOX or influence of atmosphere: limited

VEGA

To promote a better understanding of radionuclides release from reactor fuels

Pressure: - 1.0MPa

Temp.: - 3150K

Atmosphere: He、Steam

Fuel: UO_2、MOX

1

Schematics of VEGA facility

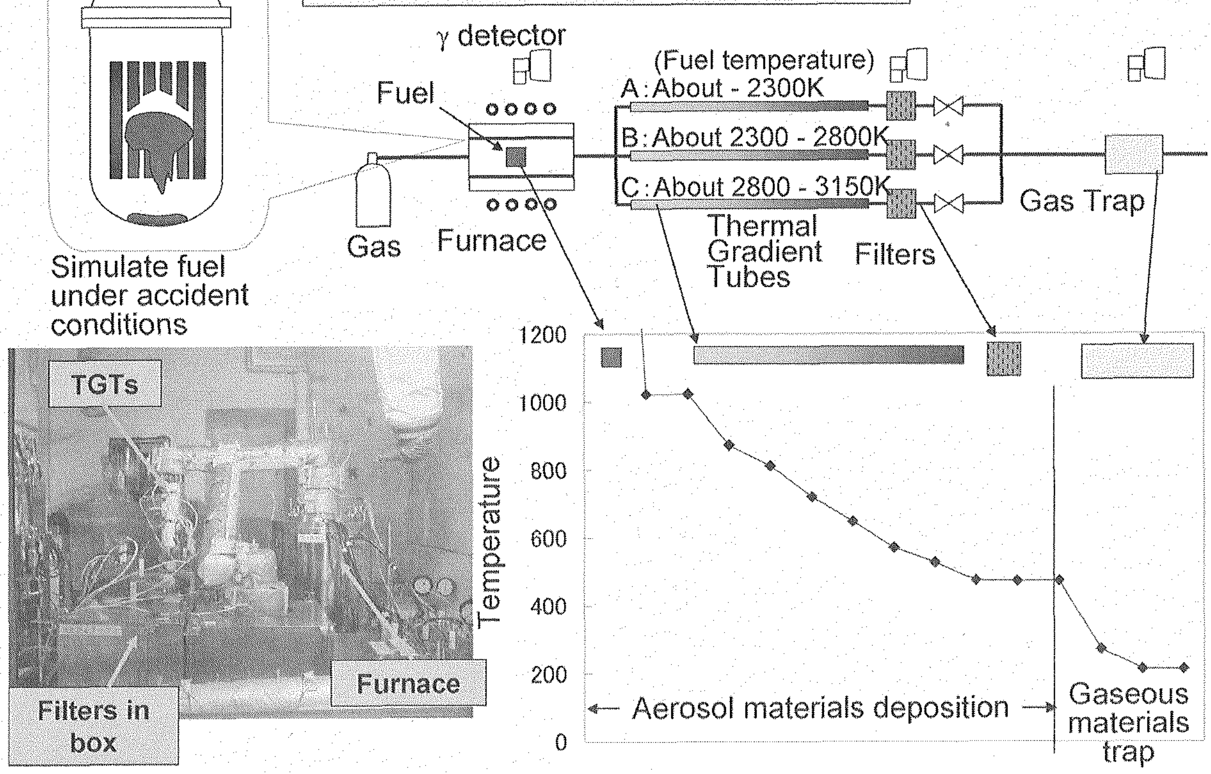


Photo. of VEGA Facility

2

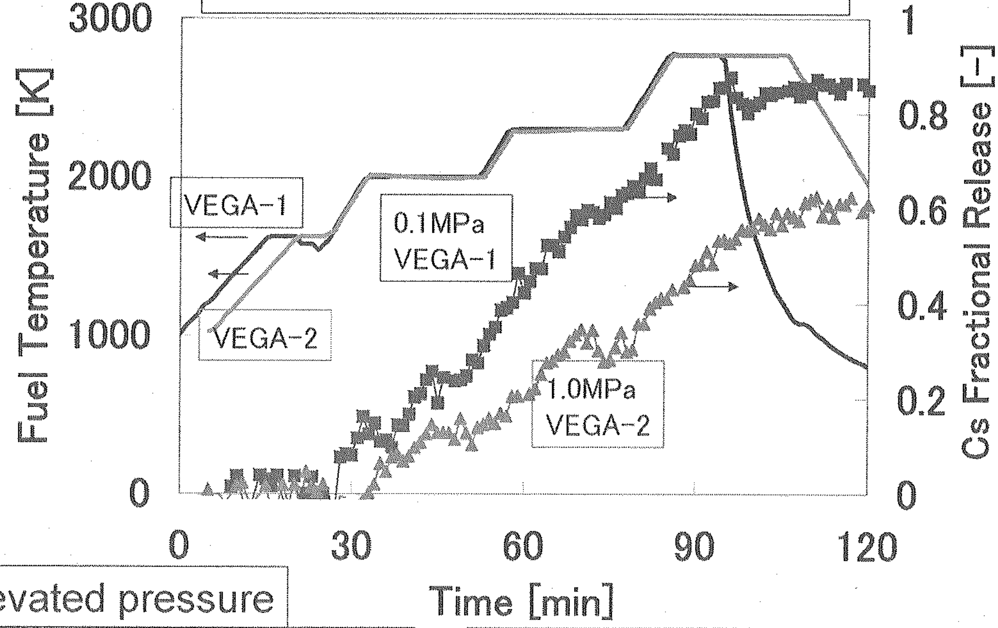
VEGA test conditions

		Pressure	
		0.1MPa	1.0MPa
Max. Temp. of Fuel	2800K	VEGA-1 ← Press. → (Atmosphere) VEGA-4 [steam] VEGA-6 [steam re-irradi.]	VEGA-2 VEGA-5 [re-irradi.] VEGA-7 [steam re-irradi.]
	3150K	VEGA-3 — Temp. VEGA-8 (MOX) VEGA-M1 [MOX]	VEGA-M2 [MOX]

Tests without notes were performed using no re-irradiated UO_2 fuel under He inert condition.

3

Release at Elevated Pressure

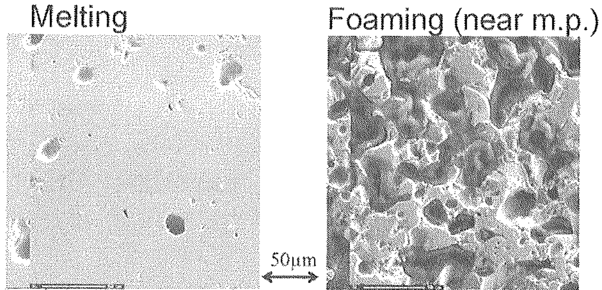


Elevated pressure

Open pores: decrease of diffusion velocity causes the increase of Cs concentration ⇨ Concentration gradient in the grain become small ⇨ Cs release was suppressed

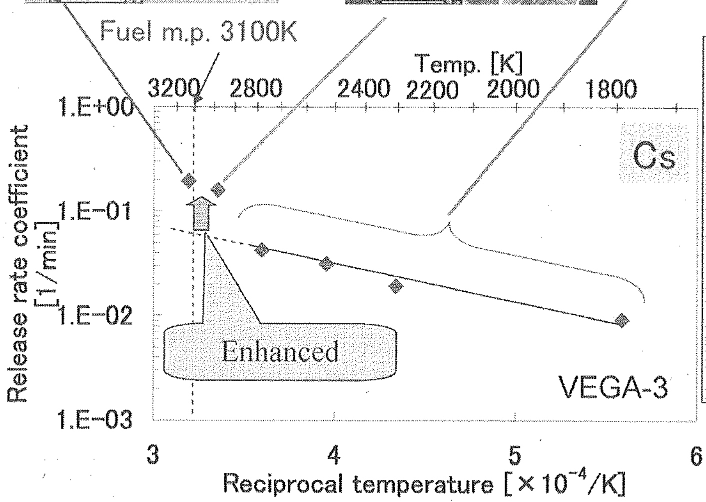
4

Acceleration of FP Release on High Temperature



VEGA-3: Temp. 3150K, Helium

1800K - 2800K:
Release is controlled by diffusion in solid

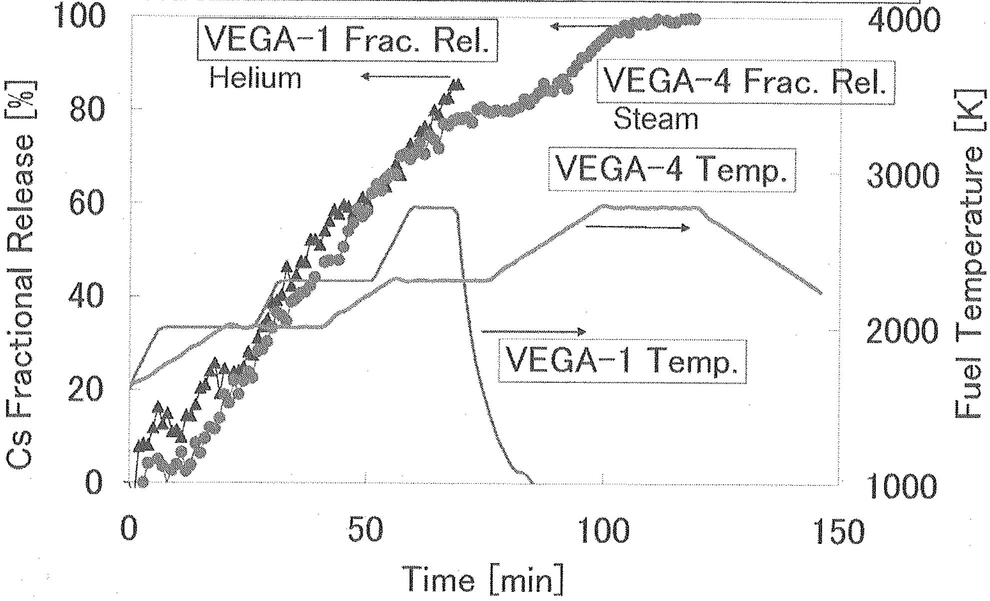


Above 2800K:
Acceleration of release rate

- Change of release mechanism
- Although melting point of UO_2 is about 3100K, the enhanced release of Cs was confirmed at temperature just below that.

5

Release in oxidizing atmosphere 1/2

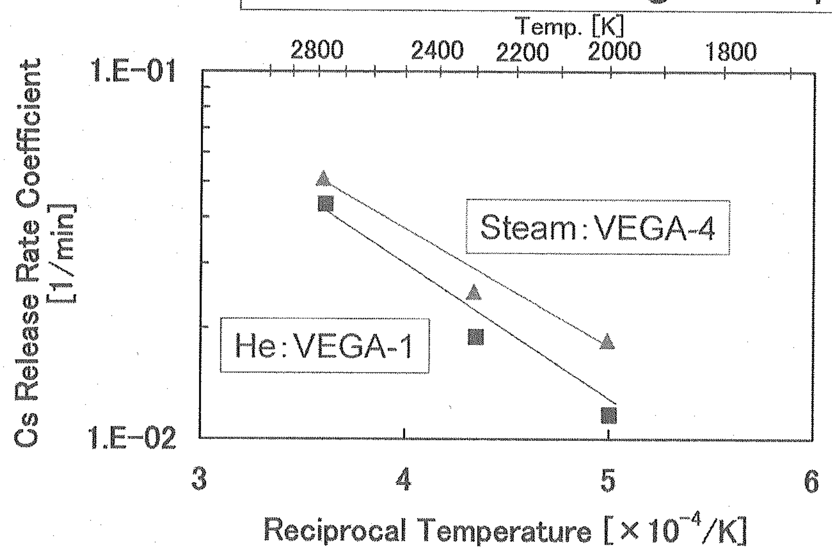


	Atmosphere	Crucible material
VEGA-1	Helium	W
VEGA-4	Steam+ Helium	ThO_2

In the steam condition, heat up rate was decreased, because thoria is weak in thermal shock.

6

Release in oxidizing atmosphere 2/2



Cs

Steam condition
⇩
Elevate conc. of interstitial oxygen
⇩
Higher conc. of vacancy
⇩
Faster diffusion of Cs

Nuclide	γ-ray energy [keV]	Helium			Steam	
		VEGA-1 2800K 0.1MPa Frac. Rel. [%]	VEGA-2 2800K 1.0MPa Frac. Rel. [%]	VEGA-3 3150K 0.1MPa Frac. Rel. [%]	VEGA-4 2800K 0.1MPa Frac. Rel. [%]	VEGA-6 2800K 0.1MPa Frac. Rel. [%]
Cs-137	662	86	61	100	93	93
Ru-106	512	5	0	0	18	14

Ru

Steam condition
⬇
Ru oxide (volatile)
⬇
Increase of Ru release

7

Release from MOX fuel 1/2

Conditions

Test No.	VEGA-M1
Fuel	ATR/MOX about 25g without cladding
Burnup End of irradiation	43GWd/tHM Jan., 1997
Initial Pu enrichment	5.7wt%
Characteristic of fuel	Volume rate of Pu spot: small
Maximum temperature	About 3150K (20 min)
Heat up rate	60K/min
Atmosphere	He, 0.1MPa

Evaluation of fractional release

γ-ray measurable elements: **γ-ray measurement of fuel before/after heat up test**

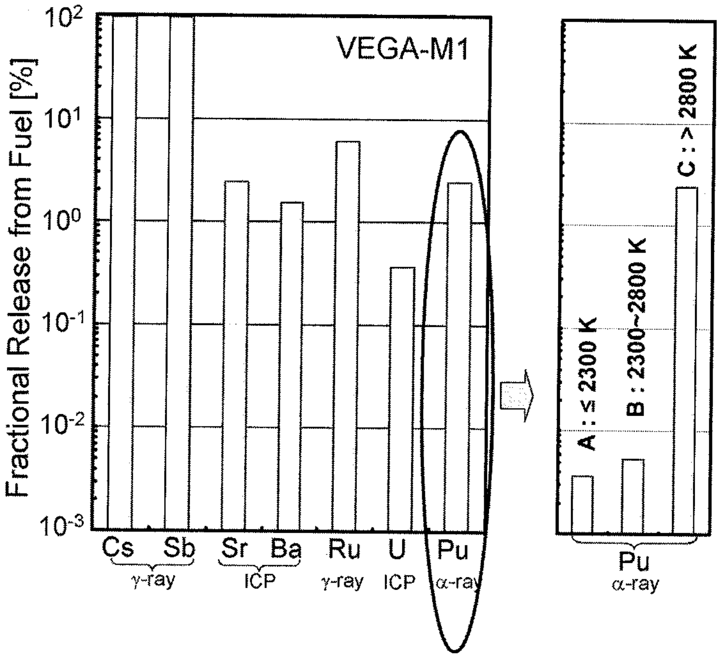
γ-ray non-measurable elements :

Acid leaching of piping → **Cs removal** → **ICP-AES**

Acid leaching of piping → **Pu separation** → **α-ray measurement**

8

Release from MOX fuel 2/2



Pu fractional release

・Previous data: 0.09% (ORNL、- 2440K)

・VEGA: 2%

Pu fractional releases at A(≦2300K), B(2300 - 2800K), C(>2800K) were estimated

Pu release above 2800 K was higher by nearly three orders than that in lower temp.

Are there any differences in radionuclides release between MOX and UO_2 ?

Compare with VEGA-8 (UO_2) → ICP-AES and α-ray measurements will be performed

9

Conclusions 1/2

- Results from VEGA tests contribute to improvement in the precision of the source term.
  - ▷ Effect of pressure: Cs release is suppressed at elevated pressure condition in comparison with atmospheric pressure
  - ▷ Release on high temperature: Cs release at temperature above 2800 K was enhanced in comparison with that at lower temperature. Cs release is accelerated from foaming fuel which is not reached melting point.
  - ▷ Effect of atmosphere: Release of Cs from fuel in steam condition was slightly higher than that in He inert condition. Higher concentration of vacancies by fuel oxidation may cause faster diffusion of Cs in a fuel grain.

    Release of Ru under steam condition was obviously greater than that under the inert condition, because volatile Ru oxides were formed.

10

Conclusions 2/2

- ▷ Release from MOX: The fractional release of Pu above 2800 K was enhanced in comparison at low temperature

  → In order to confirm whether there are any differences in radionuclides release between UO_2 and MOX, we will analyze the acid solution leached from piping and filter of VEGA-8 (UO_2, almost same condition with VEGA-M1) using ICP-AES and α-ray measurement.

- Since the original purpose has been achieved, the VEGA program is finished in this fiscal year.

11

Session 5-2

Post-VERCORS needs for analytical experiments on fission-product release

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Abstract

The final tests of the fission-product release experiments VERCORS RT and HT were performed in 2002. These tests represent the culmination of fission-product release experiments that started in 1983 with the HEVA programme[1], continued with the VERCORS programme[2] before the HT then the RT series started in the mid-1990s. HEVA and the three VERCORS series represent a total of 28 tests. The results of all these experiments (carried out by CEA with the support of IRSN and EDF) and their interpretation (by IRSN) have made a considerable contribution to the overall knowledge of fission-product release in severe-accident conditions. This can be combined with the efforts of a number of other significant experimental campaigns that have been performed, notably at ORNL[3,4], at AECL/CRL[5,6] and by JAERI[7]. Despite this extensive knowledge base, there remain gaps and little-explored areas where IRSN intends to pursue its acquisition of the information needed for it to fulfil its objectives in relation to accident evaluation, management and mitigation.

The motivation for further experiments can be summarized as arising from the following aims:

- fill in the remaining gaps for existing fuels (e.g., TU2-type MOX) and accident conditions (e.g., air ingress);
- reduction of uncertainties and overly-conservative assumptions with respect to level 2 PSAs;
- improve the understanding of fission-product release processes by confirming or improving conclusions reached in experiment interpretation (most notably with the MFPR code)[8,9];
- based on the existing, detailed-modelling capability, produce a tool capable of predictive applications in order to anticipate the consequences of evolutions in fuel design and fuel-cycle management.

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1 J.P. Leveque, B. André, G. Ducros, G. Le Marois, G. Lhiaubet, "The HEVA experimental program", Nucl. Technol. 108 (1994), p33.

2 G. Ducros, P.P. Malgouyres, M.P. Kissane, D. Boulaud, M. Durin, "Fission product release under severe accidental conditions: general presentation of the program and synthesis of VERCORS 1-6 results", Nucl. Eng. Des. 208(2) (2001), pp 191-203

3 M.F. Osbourne, R.A. Lorenz, "ORNL studies of fission product release under LWR severe accident conditions", Nucl. Safety 33(3) (1992)

4 R.A. Lorenz, M.F. Osborne, "A summary of ORNL fission product release tests with recommended release rates and diffusion coefficients", NUREG/CR-6261, July 1995.

5 D.S. Cox, Z. Liu, P.H. Elder, C.E.L. Hunt, V.I. Arimescu, "Fission-product release kinetics from CANDU and LWR fuel during high-temperature steam oxidation experiments", *IAEA Technical Meeting on Fission Gas Release and Fuel Rod Chemistry Related to Extended Burnup*, Pembroke, Canada, 28 Apr. – 1 May 1992

6 F.C. Iglesias, B.J. Lewis, P.J. Reid, P. Elder, "Fission product release mechanisms during reactor accident conditions", J. Nucl. Mater. 270(1-2) , 21-38 (1999)

7 A. Hidaka, "Recent progress of VEGA program: radionuclide release from MOX and release model with pressure effect", in *Summary of Fuel Safety Research Meeting*, Oct. 2004, Tokyo, Japan, JAERI-Review 2004-021 (2004)

8 R. Dubourg, J.M. Berniolles, G. Nicaise, M. Kissane, "Development of mechanistic code MFPR for modelling fission products release from irradiated UO_2 fuel. Part 2: application to integral tests VERCORS 4/5, PHEBUS FPT0/1", ENS TOPFUEL 2003, March 2003, Würzburg, Germany.

9 G. Nicaise, V. Ozrin, "Analysis of accidental sequence tests and interpretation of fission product release: interdependence of Cs, Mo and Ba release", *8th International Conference on CANDU Fuel*, Honey Harbour, Canada, Sept. 2003, Canadian Nuclear Society (2003)

This will not only improve confidence in results of detailed analyses but also in analysis of accident sequences. This will be achieved through substantiation or improvement of assumptions used in the simplified release modelling of the ASTEC code[10].

It is concluded that, as a priority, a limited number of experiments are required in the following specific areas:

- micro-characterization of fuel (SEM, EPMA, etc.) before and after annealing in inert, oxidizing and reducing atmospheres;
- FP release (rate and amount) experiments on MOX fuel, especially TU2-type, in a variety of reducing/oxidizing conditions;
- FP release (rate and amount) experiments in air ingress conditions for UO_2 and MOX fuels;
- FP release (primarily the amount) from high burn-up UO_2 and MOX fuels in design-basis LOCA conditions, this not being an immediate concern but necessary in the mid-term.

IRSN is pursuing the goal of obtaining the majority of results from such experiments through its so-called "Source Term Programme" (which includes fission-product release and covers several other issues). This programme is being established as a partnership already involving IRSN, CEA and EDF.

[10] W. Plumecocq, M.P. Kissane, H. Manenc, P. Giordano, "Fission-product release modelling in the ASTEC integral code: the status of the ELSA module", *8th International Conference on CANDU Fuel*, Honey Harbour, Ontario, Sept. 2003, Canadian Nuclear Society (2003)

IRSN

Fuel Safety
Research Meeting
Tokyo
2-3 March 2005

Post-VERCORS needs for analytical experiments on fission-product release

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Major Accident Prevention Division (DPAM)
www.irsn.org

- The status of the VERCORS programme
- Remaining needs
- Plans at IRSN for further experiments
- Conclusion

1/13

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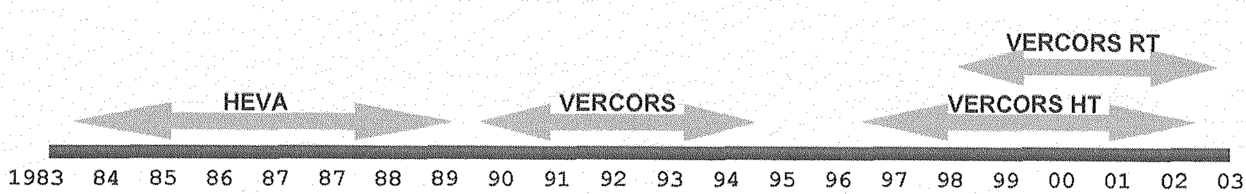
Post-VERCORS needs

VERCORS status

- **Test series**
  - **HEVA: 8 tests, release of volatile FPs**
    re-irradiation for 4 tests $1900 \leq T_{max} \leq 2370K$
  - **VERCORS: 6 tests, release of volatile + low volatility FPs**
    re-irradiation for all tests $2070 \leq T_{max} \leq 2620K$
  - **VERCORS HT: 3 tests, release of volatile + low volatility FPs + transport**
    re-irradiation for all tests $2600 \leq T_{max} \leq 2900K$
  - **VERCORS RT: 8 tests, release of low volatility (priority) + volatile FPs**
    re-irradiation for 5 tests $2400 \leq T_{max} \leq 3000K$



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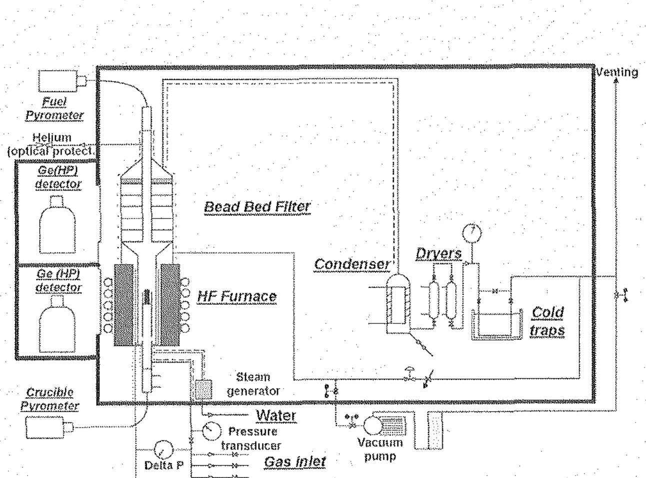
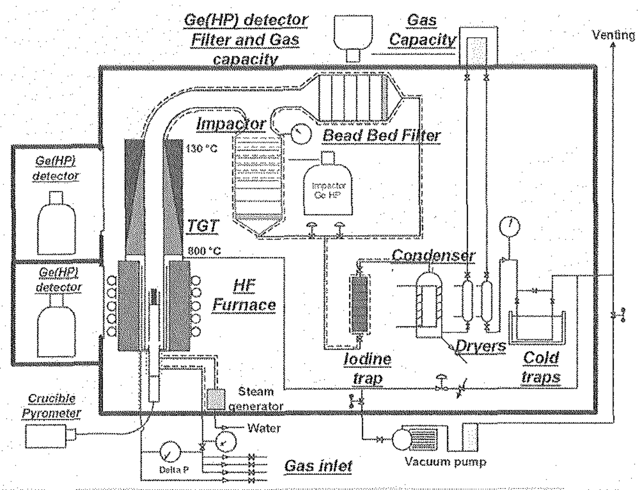
Post-VERCORS needs

VERCORS status

- **Test apparatus**

  - RT series:
    - release of low-volatility FPs (priority)
    - release of volatile FPs
  - HT series:
    - release of volatile FPs
    - release of low-volatility FPs
    - FP transport (incl. Ag, In, Cd, B)





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Post-VERCORS needs

VERCORS status

- **VERCORS RT series: final test matrix**

RT	1	2	3	4	5	6	7	8
date of test	03/1998	04/1998	11/1999	06/1999	12/1998	09/2002	04/2000	11/2002
burn-up GWd/t_U	4 cycles 47	3 cycles 46	3 cycles 38	3 cycles 33	5 cycles 60	6 cycles 70	3 cycles 43	6 cycles 70
re-irradiation	no	no	yes	no	yes	yes	yes	yes
fuel	UO_2	MOX-AUC	UO_2	UO_2	UO_2	UO_2	MOX-AUC	UO_2
atmosphere at high temperature	H_2O/H_2	H_2O/H_2	H_2/H_2O	H_2O/H_2	$H_2^{\varnothing}$	H_2O/H_2	He/H_2	H_2O/H_2, He, He/air
temperature ramp rate, $K.s^{-1}$	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
max. T_{fuel} K	2570	2440	~3000	2500	2600	2470	2890	2670
other	clad fuel	clad fuel	UO_2 debris	UO_2 debris+ oxidized Zrly	clad fuel	clad fuel	clad fuel	clad fuel

IRSN

Post-VERCORS needs

VERCORS status

- **VERCORS HT series: final test matrix**

HT	1	2	3
date of test	06/1996	04/2002	06/2001
burn-up GWd/t_U	4 cycles 49	4 cycles ~50	4 cycles 49
re-irradiation	yes	yes	yes
fuel	UO_2	UO_2	UO_2
atmosphere at high temperature	He/H_2	H_2O/H_2	He/H_2
temperature ramp rate, $K.s^{-1}$	0.5	0.2	0.2
max. T_{fuel} K	2900	liquefaction	2680
other	clad fuel	clad fuel; boric oxide + SIC	clad fuel; boric oxide + SIC^{Ψ}

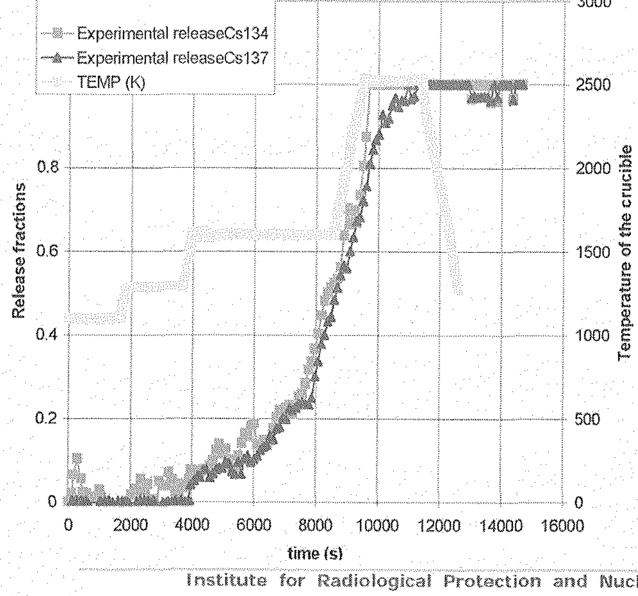
- All tests started with a H_2O/H_2 phase that fully oxidized cladding
- HT2 + HT3: absorber materials introduced at intermediate temperature to explore effect on transport, i.e., phase without then phase with.
- Ag, In and Cd activated for better measurement.

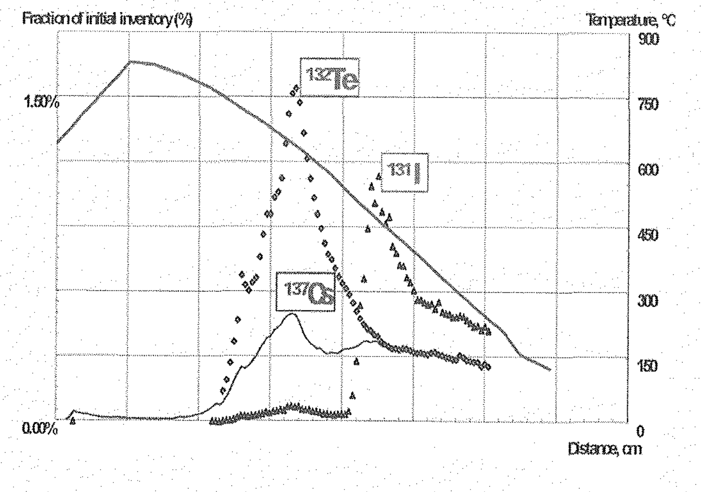
IRSN

Post-VERCORS needs

VERCORS status

- **Two examples of results**
  - **Test VERCORS 5, release:** Cs kinetics
  - **Test HT1, transport:** I, Cs, Te post-test deposition profile along TGT





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Post-VERCORS needs

Remaining needs

- **Evolution of fuel in France**
  - UO_2
    - average burn-up per assembly, current: 47 GWd/tU (52 max. in assembly)
    - average burn-up planned by 2008-10: 62 GWd/tU
  - MOX
    - average burn-up per assembly, current: 39 (41 max. in assembly)
    - average burn-up planned by 2008-10: 49-53
    - a new manufacturing process for MOX around 2007
  - Gadolinium-doped UO_2
    - used in CYCLADES and GEMMES fuel-management cycles (higher ^{235}U enrichment, longer cycles)

IRSN

Post-VERCORS needs

Remaining needs

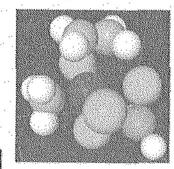
- **Evolution of safety issues at IRSN (recent internal action)**
  - PSA level 2
    - desirable to reduce uncertainties and over-conservatism
  - Modelling needs
    - confirm current interpretation where data on FP location & chemical species (e.g., Cs in solid solution, uranate, molybdate, iodide?) are lacking
    - detailed-modelling capability (MFPR code) → exploratory FP-release tool
  - Air ingress
    - increased awareness of potential consequences
    - inadequate data on release (and transport) especially with respect to Ru
  - Gadolinium-doped UO_2
    - very few data relevant to impact on source term
  - MOX
    - few data relevant to impact on source term, i.e., 2 VERCORS tests (also 2 VEGA tests) for AUC, nothing for TU2

IRSN

Post-VERCORS needs

Remaining needs: further analytical experiments (ideally)

- **Micro-characterization**
  - **Knowledge/modelling needs**
    - as a function of temperature and oxidation, require data on FP location and chemical speciation
    - analyses (SEM, TEM, EPMA, XRD,...) before and after annealing
    - atmosphere either inert, oxidizing or reducing, $1200 \leq T \leq 2500$K



- **FP release (rate and amount)**
  - **Air ingress**
    - require more data on release (and transport) especially with respect to Ru
  - **MOX**
    - TU2
  - **Design-basis LOCA**
    - high burn-up UO_2
    - MOX

IRSN

Post-VERCORS needs

Remaining needs: further analytical experiments

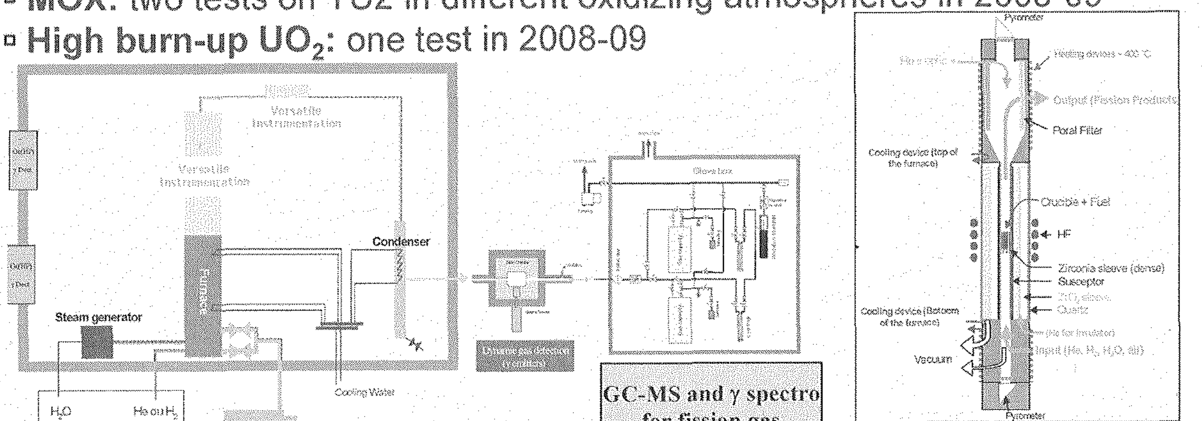
- **Context: the Source Term Programme**
  - **IRSN-CEA-EDF joint-research programme**
    - open to and being proposed to other partners, e.g., CEC
    - based on separate-effect experiments
    - four axes of research:
      1. iodine
      2. boron-carbide
      3. air-ingress
      4. fission-product release
    - programme duration: 2005 – 2010, 6 years

IRSN

Post-VERCORS needs

Remaining needs: priority experiments

- **Micro-characterization**
  - **Knowledge/modelling needs:** analyses (SEM, EPMA,...) of VERCORS samples including pellets from source rods (covers initial and final state)
- **FP release : VERDON facility (CEA/Cadarache)**
  - **Air ingress:** one test including transport aspects in 2008-09
  - **MOX:** two tests on TU2 in different oxidizing atmospheres in 2008-09
  - **High burn-up UO_2:** one test in 2008-09



IRSN

Post-VERCORS needs

Conclusions

- **VERCORS very successful**
  - **rich database on FP release and, to a lesser extent, transport**
    - wide range of burn-ups, reducing/oxidizing conditions & final temperatures
    - fuel types and physical configurations explored to some extent
- **Require now...**
  - **Knowledge of microscopic evolution of FPs in fuel**
    - confirm IRSN's understanding of FP behaviour in fuel and release process
    - creation of an IRSN tool for exploratory analyses, i.e., predictive capability
  - **More data on current, outstanding issues**
    - air ingress
    - high burn-up UO_2
    - MOX
- **Much to be gained from Source Term Programme**

IRSN **Post-VERCORS needs**

Acknowledgements

- IRSN's nuclear safety activities are sponsored by the French government
- The VERCORS tests were funded by IRSN in association with EDF (Electricité de France)
- The VERCORS experiments were carried out by the
  Commissariat à l'Energie Atomique,
  Direction de l'Energie Nucléaire
  whose experimental team included G. Ducros, P.P. Malgouyres, Y. Pontillon and M. Prouvé.
- VERCORS test conditions were specified by IRSN in consultation with CEA and EDF.

Appendix 1 Program of FSRM

Fuel Safety Research Meeting 2005
Program

Wednesday, March 2

Opening Address (9:40-)	M. Nomura	JAERI

Session 1 General report (9:50-14:10)

Fuel safety research in EPRI	R. Yang	EPRI
Fuel safety research at EDF	N. Waeckel	EDF
IRSN R&D studies on high burnup fuel behavior under accidental conditions	J. Papin	IRSN
Fuel safety research at JAERI	T. Fuketa	JAERI
Lunch (12:30-13:30)		
Fuel Safety Activities of LOCA and RIA in Korea	G-S. Auh	KINS

Session 2 Fuel behavior under RIA condition (14:10-17:05)

Studies on high burnup LWR fuel behavior under RIA conditions	T. Sugiyama	JAERI
Fission gas release from high burnup PWR fuel during an RIA	H. Sasajima	JAERI
Coffee break (15:30-15:45)		
RIA-simulation experiments on a fresh fuel rods with hydride rim	K. Tomiyasu	JAERI
Development of the RANNS code for high burnup fuel behaviors in RIA conditions	M. Suzuki	JAERI

Adjourn (17:05)

Thursday, March 3

Session 3 High burnup fuel behavior (9:30-12:10)

Fuel behavior under power oscillation	J. Nakamura	JAERI
Performance of fuel and cladding material at high-burnup	A. Romano	PSI
Grain subdivision and strain distribution between subdivided grains in high burnup UO2 pellet	M. Amaya	JAERI
Fracture toughness test of unirradiated PWR fuel cladding	N. Ikatsu	JAERI

Lunch (12:10-13:10)

Session 4 Fuel behavior under LOCA condition (13:10-15:10)

LOCA Testing at Halden	V. Grismanovs	Halden
ANL LOCA research results for cladding alloys	M. Billone	ANL
High burnup fuel behavior under LOCA conditions	F. Nagase	JAERI

Cofee break (15:10-15:25)

Session 5 Radionuclides release under severe accident condition (15:25-16:45)

VEGA program; experimental study on radionuclides release from fuel	T. Kudo	JAERI
Post-VERCORS needs for analytical experiments on fission-product releas	M. Kissane	IRSN

Session 6 JAERI's future program (16:45-17:10)

JAERI's future program	T. Fuketa et al.	JAERI
Closing remarks (17:10-)	K. Ishijima	JAERI

Adjourn (17:30)

Appendix 2 List of Participants

FSRM Fuel Safety Research Meeting 2005

March 2-3, 2005 at Toshi Center Hotel, Tokyo, Japan

List of Participants

Sadaaki Abeta	Mitsubishi Heavy Industries, Ltd.	JAPAN
Chris Merrill Allison	Innovative Systems Software	U.S.A.
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Geun-Sun Auh	Korea Institute of Nuclear Safety	KOREA
Jan In de Betou	Swedish Nuclear Power Inspectorate	SWEDEN
Michael C. Billone	Argonne National Laboratory	U.S.A.
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Jose M. Conde	Consejo de Seguridad Nuclear	SPAIN
Soichi Doi	Mitsubishi Heavy Industries, Ltd.	JAPAN
Toshio Fujishiro	Research Organization for Information Science and Technology	JAPAN
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Shigeru Fukuda	Nuclear Safety Research Association	JAPAN
Hirotaka Furuya	Nuclear Safety Technology Center	JAPAN
Viktors Grismanovs	OECD Halden Reactor Project	NORWAY
Eizo Hamanishi	Japan Atomic Energy Research Institute	JAPAN
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Akihide Hidaka	Nuclear Safety Commission	JAPAN
Kunio Higashi	Nuclear Safety Commission	JAPAN
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Jun Hwan Kim	Korea Atomic Energy Research Institute	KOREA
Motoyasu Kinoshita	Central Research Institute of Electric Power Industry	JAPAN
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Takehiko Nakamura	Ministry of Economy, Trade and Industry	JAPAN
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