UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 14, 2001

MEMORANDUM TO: ACRS Members

FROM: Med El-Zeftawy, Senior Staff Engineer M.E.

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON REACTOR FUELS, APRIL 4, 2001-ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued May 3, 2001, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: via e-mail:

J. Larkins
J. Lyons
H. Larson
R. Savio
S. Duraiswamy
ACRS Fellows and Technical Staff

MEMORANDUM TO: Med El-Zeftawy, Senior Staff Engineer
ACRS

FROM: Dana A. Powers, Chairman
Reactor Fuels Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON REACTOR FUELS, APRIL 4, 2001- ROCKVILLE, MARYLAND

I hereby certify that, to the best of my knowledge and belief, the Minutes of the subject meeting issued on May 3, 2001, are an accurate record of the proceedings for that meeting.

Dana A. Powers

Dana A. Powers, Chairman

11 \ May \ 2001

Date

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

May 3, 2001

MEMORANDUM TO: Dr. Dana A. Powers, Chairman
Reactor Fuels Subcommittee

FROM: Med El-Zeftawy, Senior Staff Engineer M.Z.
ACRS

SUBJECT: MINUTES OF THE ACRS REACTOR FUELS SUBCOMMITTEE MEETING, APRIL 4, 2001- ROCKVILLE, MARYLAND

A working Copy of the subject meeting minutes is attached. I would appreciate your review and corrections as soon as possible. Copies are being sent to all ACRS members for their information.

Attachments: As Stated

cc: ACRS Members
A. Cronenberg

cc via E-Mail:
J. Larkins
J. Lyons
H. Larson

DRAFT COPY- PREPARED FOR INTERNAL COMMITTEE USE

CERTIFIED

CERTIFIED: 5/11/2001 (D.A. Powers)
ISSUED: 5/3/2001

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REACTOR FUELS SUBCOMMITTEE MEETING MINUTES
APRIL 4, 2001
ROCKVILLE, MARYLAND

INTRODUCTION:

The ACRS Subcommittee on Reactor Fuels held a meeting on April 4, 2001 with representatives of the NRC staff, Framatome ANP, Westinghouse Electric Co., and Nuclear Control Institute. The purpose of the meeting was for the Subcommittee to review and discuss the safety issues associated with the use of high burnup and mixed-oxide fuels in commercial nuclear power plants. The meeting was open to the public. Dr. Med El-Zeftawy was the cognizant ACRS staff engineer and Designated Federal Official (DFO) for this meeting. The meeting was convened by the Subcommittee Chairman at 8:30 a.m, and adjourned on 2:30 p.m.

ATTENDEES

ACRS

D.Powers, Chairman
G. Apostolakis, Member
M. Bonaca, Member
T. Kress, Member
W. Shack, Member
R. Uhrig, Member
A. Cronenberg, ACRS Fellow
M. El-Zeftawy, ACRS Staff

NRC

R. Meyer, RES
J. Rosenthal, RES
S. Basu, RES
R. Lee, RES
H. Scott, RES
R. Caruso, NRR
M. Chatterton, NRR
R. Martin, NRR
U. Shoop, NRR
S. La Vie, NRR

FRAMATOME, ANP

G. Garner
M. Aldrich

WESTINGHOUSE

M. Nissley
W. Leech
E. Burns

NUCLEAR CONTROL INSTITUTE

E. Lyman

OTHERS

R. Coles, US/GAO
S. Copp, Duke Energy
R. Janati, PA DEP/BRP
D. Diamond, BNL
L. Ott, ORNL

Dr. Edward Lyman, Nuclear Control Institute, requested to make a brief presentation to the Subcommittee regarding the Mixed-Oxide Fuel Issues. The slides and handouts used during this meeting are available in the ACRS Office files. In addition, a list of public attendees is attached to the Office Copy of these Minutes.

Opening Remarks by the Subcommittee Remarks

Dr. Dana A. Powers, Subcommittee Chairman, convened the meeting at 8:30 a.m. on April 4, 2001. He stated that the purpose of the meeting was to discuss the safety issues associated with the use of high burnup and mixed-oxide fuels. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Office of Nuclear Regulatory Research Presentation

Dr. Ralph Meyer, RES, stated that the current NRC policy regarding high burnup fuel is to limit the peak rod burnups to 62 GWD/t. This policy partially stems from the irradiation experience gained from lead fuel rod testing programs conducted from the 1980's which involved the successful irradiation of several thousand fuel rods to burnups approaching this regulatory limit. In recent years, some problems have been revealed for high burnup fuel. Although the lead rod test experience has demonstrated the feasibility of achieving high burnup levels under normal operating conditions, there is not a demonstration of satisfactory performance of high burnup fuel for design basis or severe accidents. For design basis accidents, the available database extends to approximately 35 GWD/t. This regulatory strategy was based on a confidence that there was a thorough understanding of fuel behavior and that the database developed in the past experimental programs could be extrapolated to higher burnups and higher operating power densities.

Currently, the are some indications that the nuclear industry is interested in fuel burnup higher than 62 GWD/t. The RES program is constrained by the user need process to its confirmatory role to examine fuel only up to the currently permitted fuel burnups (62 GWD/t). There are both societal and economic advantages associated with use of fuel to higher burnups.

Previously, the ACRS recommended that RES to find ways to identify the types of data required to support applications for higher burnups. Especially important is to identify where experimental data must be used to substantiate predictions of analyses. RES responded with a well conceived program of Phenomena Identification and Ranking Tables (PIRTs). RES assembled a panel of world experts in fuel behavior to carry out the PIRTs effort.

The expert panel completed three PIRTs draft reports. These are 1) PWR rod ejection accidents, 2) BWR power oscillations without scram, and 3) Loss-of-Coolant accidents for PWRs and BWRs. These reports address the adequacy of regulatory criteria and evaluation models that are being used beyond their burnup range of validation. Confirmation of adequacy or revision of these criteria and models will ensure that the levels of safety that existed for low-burnup fuel is maintained for high-burnup fuel.

For the present PIRTs, the fuel with the highest burnup is assumed to have a burnup of 62 GWD/t. Although a specific plant and fuel have been selected, the experts panel recognizes the desirability of extending the applicability of the reactivity insertion accident PIRT for the specified plant and fuel. Accordingly, the panel selected to perform a preliminary screening of the phenomena identified for the selected plant, fuel and cladding to other plants (Westinghouse and Combustion Engineering), fuel types (mixed oxide fuel utilizing fissile plutonium), cladding types introducing niobium (NB) or having reduced tin (Sn) content (ZIRLO, Duplex, and M5), and burnups to 75 GWD/t. The PIRTs Reports are available on the NRC web site- www.nrc.gov/RES/pirt.

Dr. Meyer indicated that the main objective of fuel cladding alloy development is to reduce corrosion during operation. Zircaloy is zirconium with 1.2- 1.7% tin, ZIRLO is low-tin Zircaloy with approximately 1% niobium added, M5 is zirconium with approximately 1% niobium and no tin, and E-110 is zirconium with approximately 1.05% niobium. Evidence of embrittlement of E-110 at 6% oxidation versus 17% (limit in 10 CFR 50.46). However, M5 does not show such phenomenon!. The current status of RES program is as follows:

- Reactor operation to 62 GWD/t burnup approved for Zircaloy, ZIRLO, and M5
- Specific questions have been raised regarding criteria for accidents
- Confirmatory data and assessments of accident criteria for current burnup limit (62 GWD/t) to be provided by NRC
- PIRTs were developed to help focus Research Programs and find methods to resolve high burnup issues
- Data and assessment for extended burnup beyond 62 GWD/t to be provided by industry
- Dry storage to 45 GWD/t approved for Zircaloy-clad fuel only
- Dry storage criteria for higher burnups and other alloys to be developed by NRC.

Dr. Meyer stated that some implications from the LOCA PIRT indicated many thermal-hydraulic models were ranked as highly important and not well understood. In addition, the NRC's 10 CFR 50.46 embrittlement criteria were based on ring-compression ductility tests rather than quench tests. The cladding alloy type was found to be very important.

Dr. Meyer summarized the opinion of the regulatory staff regarding a recent Organization for Economic Cooperation and Development (OECD) topical meeting on LOCA fuel safety criteria. He stated that there is reluctance to neglect the effect of mechanical constraints on thermal-shock fragmentation. This has been justified later by JAERI constraint-quench tests and

Phebus LOCA-219 bundle test. Retention of ductility is the best guarantee against potential fragmentation under various types of loadings (thermal shock, hydraulic, seismic forces). Results from unconstrained quench tests were:

- considered only corroborative and reassuring.
- their use for regulatory purposes not accepted.
- later studies showed a large margin compared to 17% criteria

The 17% criteria are based on results from post-quench ductility test. Dr. Meyer noted that the effects of oxidation (before and during LOCA transient), hydrogen uptake (larger than a threshold amount), high burnup, and niobium addition appear inseparable. There is a need to understand the effect of small materials differences to avoid repeating all tests every time the manufacturer makes a small change in the alloy.

LOCA Ductility of M5 Cladding (Framatome ANP)

Mr. Garry L. Garner, Framatome ANP, presented a review of the M5 in-reactor operating experience and an overview of Framatome's testing of M5 cladding performance. Framatome presented data on the M5 manufacturing experience, including material on the chemical composition and the thermomechanical processing, as well as data on the irradiation experience. In addition, Framatome presented data which demonstrates that M5 exhibits low corrosion and low hydrogen pickup, and no acceleration of corrosion or hydrogen pickup at high burnups. In summary, Mr. Garner stated that the M5 has the following features:

- In-reactor operating performance is superior to Zircaloy-4
- LOCA and post- LOCA oxidation rates are equal to or slower than Zircaloy-4
- LOCA and post-LOCA mechanical performance is equivalent to Zircaloy-4
- LOCA and post-LOCA performance is acceptable and is equal to or better than Zircaloy-4 in events of equal duration
- LOCA and post-LOCA mechanical performance is superior to the Zr-1% Nb alloy tested by Bohmert. These tests included oxidation rate, quench embrittlement, ring compression, bending, and impact tests.

Westinghouse Electric Company Presentation

Mr. William Leech, Westinghouse, briefed the Subcommittee regarding Westinghouse efforts on ductility testing of Zircaloy-4 and ZIRLO cladding after high temperature oxidation in steam. The ductility measurements on Zircaloy oxidized in high temperature steam were used to establish cladding embrittlement criteria of 10 CFR 50.46. The testing consisted of quench tests from temperature and ring compression tests. Ring compression tests were conducted on Zircaloy-4, and quench tests of Zircaloy-2 and Zircaloy-4. The purpose of the criteria is to ensure that the cladding would remain sufficiently intact to assure an easily coolable geometry.

Testing of ZIRLO was performed to obtain data on material mechanical properties, density, thermal expansion, thermal conductivity, specific heat, phase changes, high temperature creep,

high temperature oxidation, and rod burst characteristics. Other than phase change characteristics, the properties are essentially equivalent to those of Zircaloy-4. Westinghouse expects that, because of the close similarity to Zircaloy-4, the 17% equivalent cladding reacted (ECR) criterion will be continued to apply to ZIRLO.

Mr. Leech outlined some of the results of tests on Alloy E-110 oxidized in high temperature steam (Bohmert-paper). He noted that ECR to cause complete embrittlement is about 1/3 the value for Zircaloy-4. A number of physical differences between the oxide layers of E-110 and Zircaloy-4 were observed. These are:

- E-110 displayed a heterogeneous appearance of the oxide scale
- E-110 formed two oxide layers (duplex oxide) that were frequently separated by cracks
- Multi-layer oxide layers tend to flake
- E-110 showed low hydrogen uptake only if firmly adherent crackless oxide layers were formed

Mr. Leech indicated that there are significant differences in the oxide layer structure reported for the E-110 alloy and those observed for either ZIRLO or Zircaloy-4. Westinghouse plans to perform more testing and meet with the NRC staff to discuss the results (possibly in May 2001).

MOX Fuel Program

Mr. Richard Lee, RES, discussed the RES program associated with the use of MOX fuel in commercial light water reactors. In March 1999, the Department of Energy signed a contract with Duke Power, Cogema, Stone & Webster, and Limited Liability Corporation (MOX Consortium) to provide fabrication, radiation services, and utilization of MOX fuels in commercial nuclear power plants. RES is responsible for developing the necessary data and tools to support NRR review of MOX fuels to ensure that the requirements in 10 CFR 50.46, Appendix A to 10 CFR Part 50, and Appendix K to 10 CFR Part 50 are met.

Weapons-grade MOX fuel has a different mix of plutonium isotopes than reactor-grade MOX fuel, which affects the neutronics properties of the fuel. In addition, the thermal and mechanical performance are different. These differences affect fabrication, transportation, storage, and the use of MOX fuel. The staff has grouped the technical issues associated with weapons-grade MOX into four areas: reactor physics; fuel behavior; source term; and fuel fabrication, storage, and disposal. The RES efforts are targeted to address the first three areas only.

Reactor Physics- the NRC uses neutronics code (PARCS) which will be modified to include MOX-specific models. Some of the models to be added include multiple energy groups with upscatter, a modeling capability to capture the steep gradients between the two types of fuel assemblies, improved delayed-neutron precursor calculations, and a method to handle local power peaking.

Fuel Behavior- Although MOX and UO_2 fuels behave in a similar fashion during normal steady-state operation, subtle differences caused by different physical properties have some effects on

fuel and cladding performance under accident conditions. The NRC fuel behavior codes (FRAPCON, FRAPTRAN) will be modified to account for altered material properties such as thermal conductivity, thermal expansion, and creep rates. The NRC is negotiating for participation in the Cabri (France) and in NSRR to obtain relevant reactivity insertion accident data for tests performed on MOX fuels.

Source Terms- The staff believes that the gap release (source term) may increase because of the elevated operating temperatures in MOX fuel. Because of the way MOX is fabricated , MOX porosity may be different from uranium-based fuel porosity. The NRC will benchmark the FRAPCON code against data from the NSRR reactor (Japan) and the Halden reactor. For severe accident conditions, the NRC staff will obtain fission product release data from the VERCORS (France) and the VEGA (Japan) tests under the NRC's current cooperative program. Additional fission product release test data may become available from the MADRAGUE (France) .

Mr. Lee noted that the MOX PIRT will be performed as part of the high burnup fuel program. The PIRT for LOCA and reactivity accident have been completed. PIRT for source term is being initiated. Dr. Kress noted that PIRT for Iodine spikes will be advised.

Recent Operational Issues and Experience with High Burnup Fuel

Ms. Margaret S. Chatterton, NRR, briefed the Subcommittee regarding the basic NRR's approach for burnup extension. Currently, the NRC is working with the industry to develop a strategy and a plan. For industry proposals to extend burnup beyond 62 GWD/t, the industry will be responsible for developing the plan and guidelines. In addition, the industry will perform the testing and develop the criteria. The staff objective is to endorse the industry's approach in a Regulatory Guide. The burnup extension guidelines will address the current licensing requirements including LOCA, ATWS, and reactivity insertion accidents. The guidelines will also be risk informed. The current emphasis from the staff is on the lead test assemblies (LTAs). The guidelines for the LTAs will be for prototypical that include the power history, type of cladding, flow conditions, and water chemistry. In addition, the LTAs guidelines will include the definitions, characterizations (pool side and hot cell examinations), number, placement, safety requirements, and reporting.

Ms. Chatterton noted that the current fuel reviews include duplex cladding for BWRs, and Zirlo for CE plants. Some of the recent fuel issues include oxidation higher than expected, axial offset anomalies, fuel failure due to high fuel duty , adverse effects of water chemistry, high crud buildup, and accelerated growth of rods and assemblies.

MOX Fuel Safety : A Need for Research

Dr. Edwin S. Lyman, Nuclear Control Institute, briefed the Subcommittee regarding his concerns on the MOX program. Dr. Lyman stated the following:

- No real choice of mission reactors
- Timetable dictated by international agreement and not by safety requirements

- DOE budget cuts will increase pressure for abbreviated MOX safety review

- Heavy reliance on proprietary foreign data
- Cancellation of immobilization track will increase burden on MOX program
- MOX might not meet Regulatory Guide 1.174 Criteria
- Compared to LEU core, the MOX core (with 40% MOX core fraction) at EOC contains approximately 2 times more Pu-239, 10% less Pu-238, 45% more Ru-106
- DOE's environmental impact statement inventory calculations are flawed
- Increased source term in MOX cores is important for severe accidents with early containment failure (25% increase in latent cancer fatalities, and 4% increase in prompt fatalities)
- Ice condensers are substantially more sensitive to early containment failure
- Pressurized thermal shock screening criteria for LEU cores may not be appropriate for MOX cores
- NRC ability to fully resolve MOX fuel safety issues is in jeopardy
- Current path for MOX fuel approval is not likely to engender public confidence

<u>Subcommittee Caucus</u>

The Subcommittee discussed the general status of the high burnup activities and agreed that it looks prescient now. Some Members indicated that the RES staff may consider developing PIRTs for Iodine spiking and source terms. Generally, the Subcommittee members believed that the high burnup fuel research needed additional resources to investigate a more diverse range of alloys.

<u>Background Material Provided to the Subcommittee Prior to This Meeting</u>

- Memorandum to Commissioners, form L. Joseph Callan, EDO, Subject: Agency Program Plan For High-Burnup Fuel, dated July 6, 1998.
- Letter to W. Travers, EDO, from D. Powers, ACRS, Subject: High Burnup Fuel Phenomena Identification And Ranking, dated March 24, 1999.
- Memorandum to ACRS Members, from A, Cronenberg, ACRS, Subject: ACRS Review Issues- MOX Fuel License Amendment Requests, dated March 16, 2001.
- Memorandum to the Commissioners, from W. Travers, EDO, Subject: Agency Plan For Confirmatory Research Associated with the Use of Mixed-Oxide Fuel in Commercial

Light Water Reactors, dated February 11, 2000.

- The PIRTs Reports/ Web site-www.nrc.gov/RES/pirt.

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NOTE: Additional details can be obtained from a transcript available for downloading or viewing on the Internet at www.nrc.gov/ACRSACNW, or can be purchased from Neal R. Gross & Co., 1323 Rhode Island Ave., N.W. Washington, D.C. 20005, (202) 234-443 (nrgross@nealgross.com).

MME

United States Nuclear Regulatory Commission

RESEARCH ACTIVITIES AND THE HIGH BURNUP PIRTS
(Phenomenon Identification and Ranking Tables)

Ralph Meyer
Office of Nuclear Regulatory Research

ACRS Reactor Fuels Subcommittee
April 4, 2001

FUEL CLADDING ALLOYS

(Main Objective of Alloy Development is to Reduce Corrosion during Operation)

- Zircaloy (Zirconium with 1.2-1.7% Tin)
- Low-Tin Zircaloy
- ZIRLO (Low-Tin Zircaloy with ~1% Niobium added)
- M5 (Zirconium with ~1% Niobium and no Tin)

FUEL-RELATED SAFETY CRITERIA

- Limited Fuel Damage during Postulated Accidents to Ensure Coolable Core Geometry and Avoid Core Melt
  - Criteria for Overpower Events (Reactivity Accidents)
  - Criteria for Undercooling Events (Loss-of-Coolant Accidents)

- Limited Fuel Damage during Dry Storage to Facilitate Removal from Storage
  - Criteria to Avoid Creep Rupture (Normal Storage Conditions)

- All Safety Criteria were developed for Low Burnup Fuel
  - It was thought that Early-life Conditions were more Limiting

- All Safety Criteria were developed for Zircaloy-clad Fuel
  - It was thought that Alloy Improvements for Operation would also be good for Accidents and Storage

STATUS

- Reactor Operation to 62 GWd/t Burnup Approved for Zircaloy, ZIRLO, and M5
- Specific Questions have been raised about Criteria for Accidents
- Confirmatory Data and Assessment of Accident Criteria for Current Burnup Limit (62 GWd/t) to be provided by NRC
- PIRTs were developed to Help Focus Research Programs and Find Methods to Resolve High Burnup Issues
- Data and Assessment for Extended Burnup beyond 62 GWd/t to be provided by Industry
- Dry Storage to 45 GWd/t Approved for Zircaloy-clad Fuel only
- Dry Storage Criteria for Higher Burnups and Other Alloys to be developed by NRC

PIRT SCENARIOS

(based on 1998 Agency Program Plan)

- PWR Rod Ejection Accident (assumed base case: TMI-1, 15x15 fuel, 62 GWd/t peak rod, hot zero power)
- BWR Power Oscillations without Scram (assumed base case: Lasalle-2, 8x8 fuel, 62 GWd/t, 84%power)
- Loss-of-Coolant Accident (no specific plant assumed, Zircaloy-clad fuel, 62 Gwd/t)

PIRT ACTIVITIES

- ~25 Fuel Experts from Industry, Labs, Universities, and Foreign Agencies
- 8 Meetings (total 25 Days) from August 1999 to October 2000
- 3 NUREG/CR Reports with PIRTS and Related Information (final Drafts)
- Staff Report with Interpretations and Suggestions (Draft)
- Web Site with all Reports and Transcripts (www.nrc.gov/RES/pirt)

PIRT FOR PWR ROD-EJECTION ACCIDENT
(280 cal/g Limit in Reg. Guide 1.77)

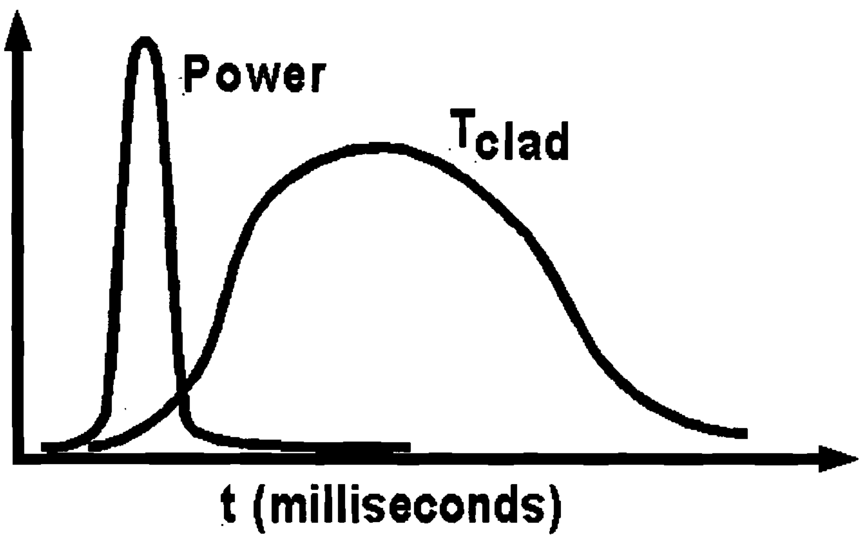


Fig. 1 Qualitative plot of fuel rod power and cladding temperature for a PWR rod-ejection accident

IMPLICATIONS FROM ROD-EJECTION PIRT

- Core Design Changes can Alter the Energy Deposited in the Accident
- Ejected Rod Worth might be used as a Substitute for a Fuel Enthalpy Limit
- Testing in Burnup Range of Interest is Important (Oxidation Phenomena alone will not determine Outcome)
- MOX Rod Testing is Important because of the Pu-rich Agglomerates
- It is Important to Test in Correct Coolant Environment (Water Loop)
- Effect of Different Cladding Alloys not very Important (extrapolate with Mech. Props.)
- High Temperature Ballooning and Rupture might occur for Some Cladding Alloys with high Ductility (i.e., no PCMI Failure)

A METHOD TO RESOLVE ROD-EJECTION ISSUES

- Improve Empirical Correlation with New Data from Cabri and NSRR
- Obtain Mechanical Properties Data for Zircaloy, ZIRLO, and M5 Cladding Alloys in
  Test 2002
- Use FRAPTRAN Fuel Rod Code to Adjust Correlation for Different Alloys (Uncertainty?)
- Use PARCS 3-D Neutron Kinetics Code for Generic Safety Analysis
  (sodium loop)

Target late 2003 for Confirmatory Resolution at 62 GWd/t using two Cabri Tests (ZIRLO and M5), Initial Tests from NSRR High Temperature Capsule, and Code Analysis.

in 2003 we (RES) -> criteria

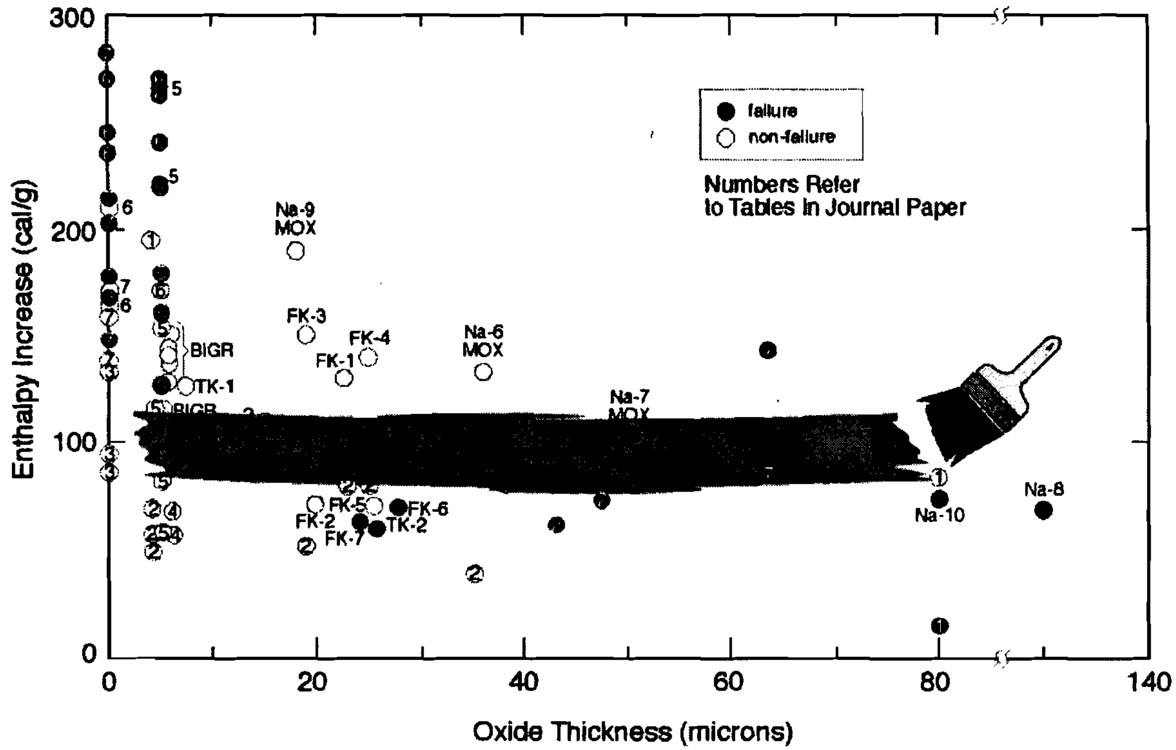


Fig. 2 Fuel enthalpy as a function of oxide thickness for tests described in Ref. 10 (solid symbols indicate cladding failure; open symbols indicate no failure)

PIRT FOR BWR POWER OSCILLATIONS WITHOUT SCRAM
(280 cal/g Limit used by GE)

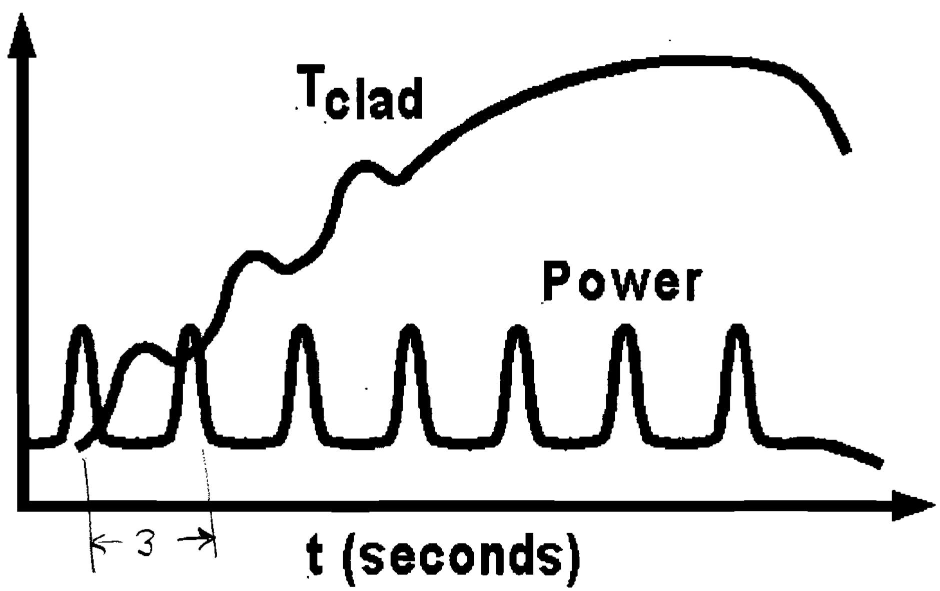


Fig. 3 Qualitative plot of fuel rod power and cladding temperature for BWR power oscillations without scram

IMPLICATIONS FROM POWER-OSCILLATION PIRT

(Because energy is small) -

- Pellet-Cladding Mechanical Interaction (PCMI) Cladding Failures are Not Expected
- LOCA-like Oxidation is Expected with possible Ballooning and Rupture
- Cladding Embrittlement will take place at Lower Temperature than Cladding Melting or Fuel Melting
- Runaway Oxidation is Not Expected
- LOCA-like Embrittlement Criteria appear to be Appropriate

A METHOD TO RESOLVE POWER-OSCILLATION ISSUES

Pellet cladding mechanical interaction

- Repeated-Pulse Test Capability in NSRR to address PCMI Failure
- High Temperature Dryout Test Capability in Halden Reactor
- Information from LOCA Work on Embrittlement Criteria
- Generic Calculations with FRAPTRAN-GENFLO (STUK Finland) Hot Channel Code to Compare with Embrittlement Criteria

Target 2004 for Confirmatory Resolution at 62 GWd/t. Depends on Testing that has not been Fully Planned and future Code Developments.

Peak cladding Temp.

PIRT FOR LOSS-OF-COOLANT ACCIDENTS
(1204°C PCT, 17% ECR Embrittlement Criteria in 10 CFR 50.46)
(Ballooning, Rupture, Oxidation EMs in Appendix K)

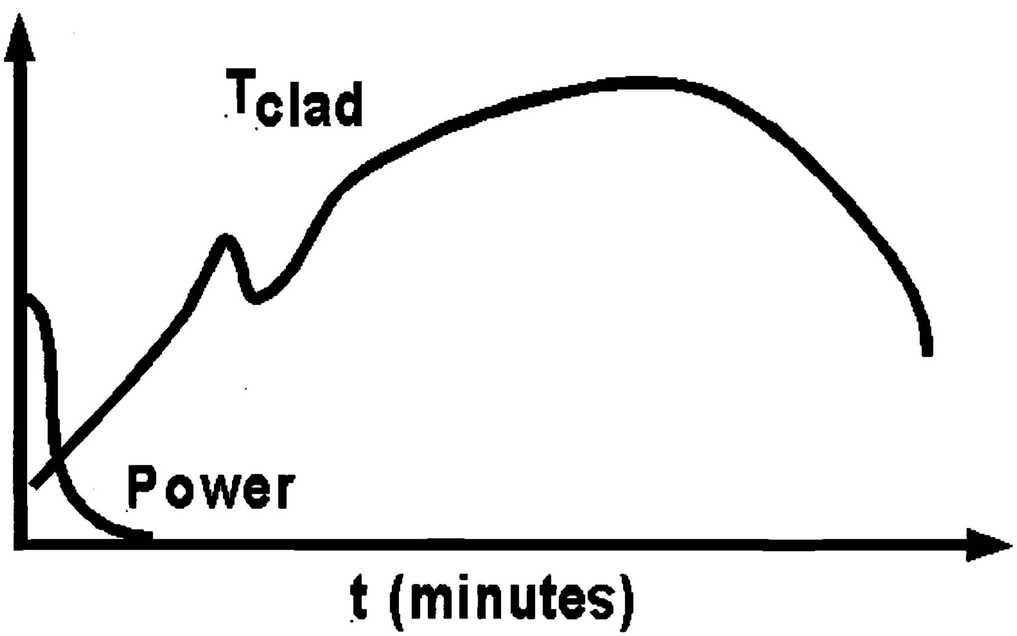


Fig. 4. Qualitative plot of fuel rod power and cladding temperature for a loss-of-coolant accident

IMPLICATIONS FROM LOCA PIRT

- Many Thermal-Hydraulic Models were ranked as Highly Important and Not Well Understood

- A Foreign Member of the PIRT Panel (G. Hache, France) reminded us that NRC's 10 CFR 50.46 Embrittlement Criteria were based on Ring-Compression Ductility Tests rather than Quench Tests (2200°)

- Cladding Alloy Type was found to be Very Important

A METHOD OF RESOLVING LOCA ISSUES

- Integral Testing at ANL with High Burnup Zircaloy-Clad Fuel (Ballooning, Rupture, Relocation, Oxidation, and Quenching)
- Integral Testing in Halden Reactor with High Burnup Zircaloy-Clad Fuel
- Separate-Effect Testing at ANL with High Burnup Zircaloy-Clad Fuel (Mechanical Properties, Oxidation, Post-Quench Ductility)
- Related Results from JAERI and RRC-Kurchatov Institute
- Limited use of FRAPTRAN Fuel Rod Computer Code for Design and Interpretation of Experiments
- Integral and Separate-Effect Testing of Advanced Cladding (ZIRLO and M5) at ANL

Target Resolution in 2002 for BWR with Zircaloy, 2003 for PWR with Zircaloy, 2004 for PWR with ZIRLO, and 2005 for PWR with M5 depending on Availability of Fuel Rods and Other Factors

*NRC do not fund these programs

"NRC" FUEL RESEARCH

- Argonne Nat'l. Lab. Hot Cells: LOCA-Related Research, Dry Storage Research, and General Mechanical Properties
- Penn. State University: Consulting and Subcontracting to Argonne
- Pacific Northwest Nat'l. Lab.: Fuel Rod Code Development for Steady State and Transients
- Brookhaven Nat'l. Lab.: Reactivity Accident Analysis with 3-D Plant Transient Code
- Halden (Norway) Materials Test Reactor: Steady-State and Transient Properties
- IPSN (France) Cabri Pulse Reactor and Hot Cells: Reactivity Accidents and Mechanical Properties
- JAERI (Japan) NSRR Pulse Reactor: Reactivity Accidents and LOCA-Related Research
- RRC-Kurchatov Institute (Russia) Pulse Reactors and Hot Cells: Reactivity Accidents and General Mechanical Properties

EPRI COOPERATION

- Successfully obtained High Burnup BWR (Limerick) and PWR (H. B. Robinson) Zircaloy-Clad Fuel
- Technical Assistance in Planning Integral and Separate-Effect Tests at ANL
- Expressed Interest in Continuing this Cooperation with NRC in the ANL Program with Advanced Alloys (ZIRLO, M5)

MME

United States Nuclear Regulatory Commission

SUMMARY OF OECD TOPICAL MEETING ON LOCA FUEL SAFETY CRITERIA

Ralph Meyer
Office of Nuclear Regulatory Research

ACRS Reactor Fuels Subcommittee
April 4, 2001

OECD/NEA/CSNI
SPECIAL EXPERT GROUP ON FUEL SAFETY MARGINS
Wolfgang Wiesenack (Halden, Norway), Chair

TOPICAL MEETING ON LOCA FUEL SAFETY CRITERIA
Georges Hache (IPSN France), Technical Program Chair
Aix-en-Provence, France
March 22-23, 2001

Proceedings to be Published

Post-Quench Ductility
- Background (G. Hache, IPSN)
- Hungarian Paper on E110 (L. Maroti, AEKI)
- 2 Russian Papers on E110 (L. Andreeva-Andrievskaya and N. Sokolov, VNIINM)
- French Paper on M5 (A. Lebourhis, Framatome)
- American Paper on ZIRLO (W. Leech, Westinghouse)

Effect of Axial Constraint during Quenching
- Japanese Paper on Experiments (Uetsuka, JAERI)
- French-American Paper on Calculations (Waeckel, EPRI & EdF)

Relocation of Fragmented Fuel into Ballooned Region
- French Paper on Calculations (M. Lambert, EdF)
- French Paper on Calculations (C. Grandjean, IPSN)

RATIONALE OF THE LOCA 10CFR50.46b CRITERIA FOR ZIRCALOY AND COMPARISON WITH E110 ALLOY

G. Hache (IPSN, France)
Introductory Presentation

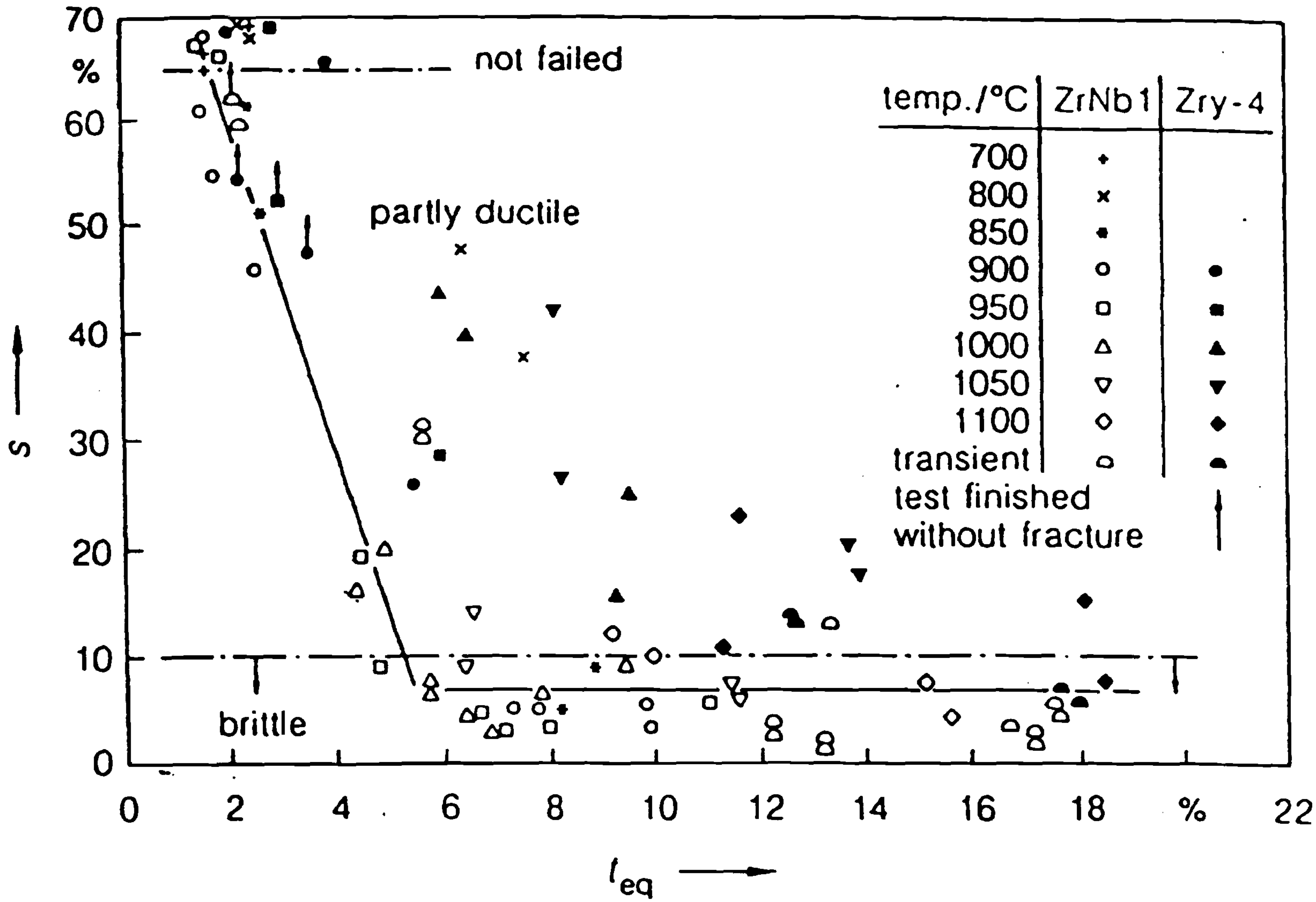
OPINION OF THE REGULATORY STAFF AND COMMISSIONERS (1)
(ECCS Rule - Making Hearing, 1973)

- Reluctance to neglect the effect of mechanical constraints on thermal-shock fragmentation
  - rod-rod interaction due to ballooning or bowing
  - rod-grid interaction due to differential shrinkage between fuel rods and guide tubes

- Justified later by JAERI constraint-quench tests and Phebus LOCA-219 bundle test

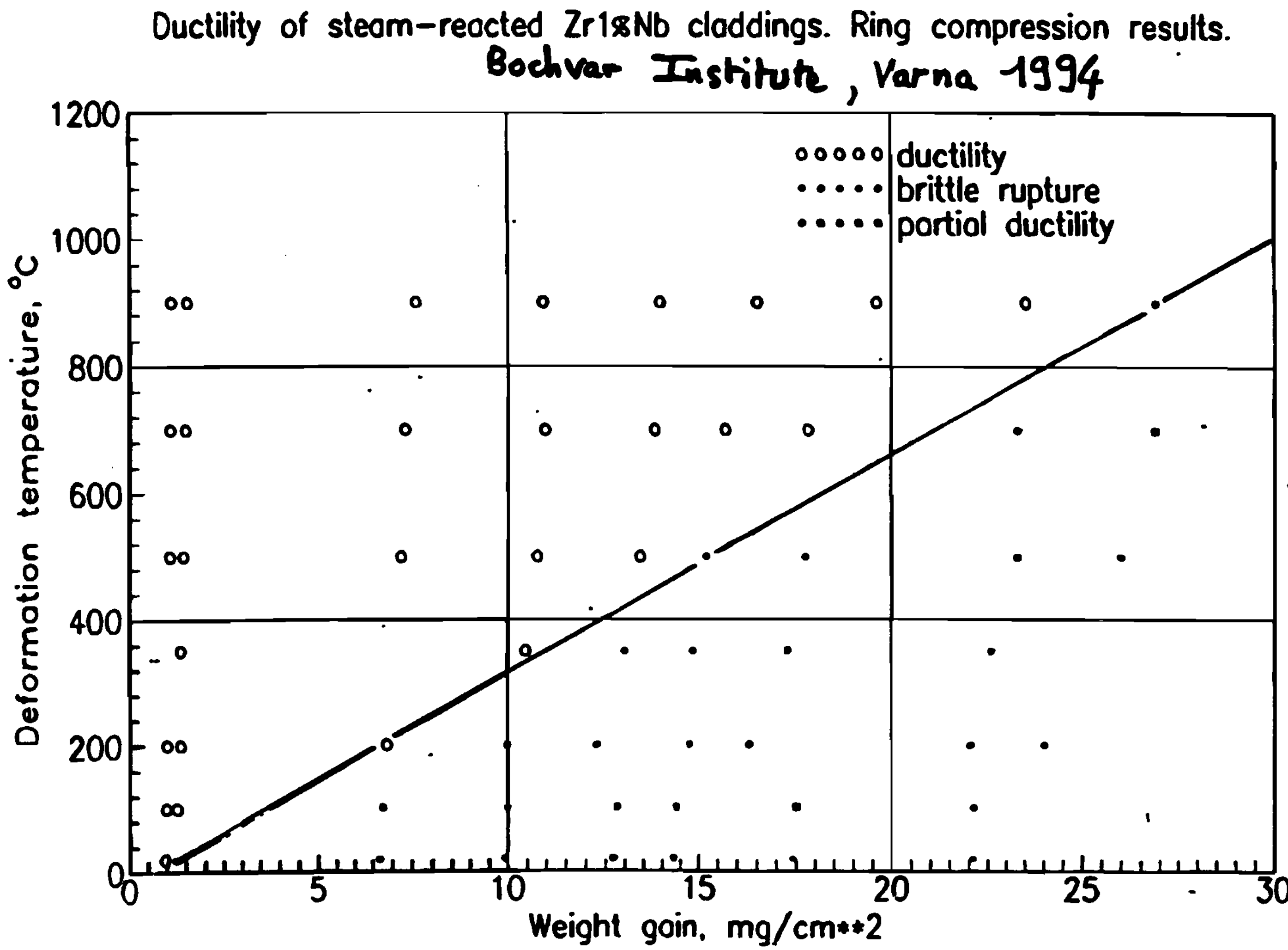
OPINION OF THE REGULATORY STAFF AND COMMISSIONERS (2)

- Retention of ductility is the best guarantee against potential fragmentation under various types of loadings (thermal shock, hydraulic, seismic forces).
- Results from unconstrained quench tests (simple thermal-shock test) were:
  - considered only corroborative and reassuring.
  - Their use for regulatory purposes not accepted.
  - Later studies showed a large margin compared to 17% -ECR and 2200°F -PCT criteria.
- 17% -ECR and 2200°F -PCT criteria are based on results from post-quench ductility test (Hobson's slow-ring-compression tests).



J. Böhmert, "Embrittlement of ZrNb1 at room temperature after high-temperature oxidation in steam atmosphere," Kerntechnik, Vol. 57 (1992) p. 56

DUCTILITY TESTS (2)



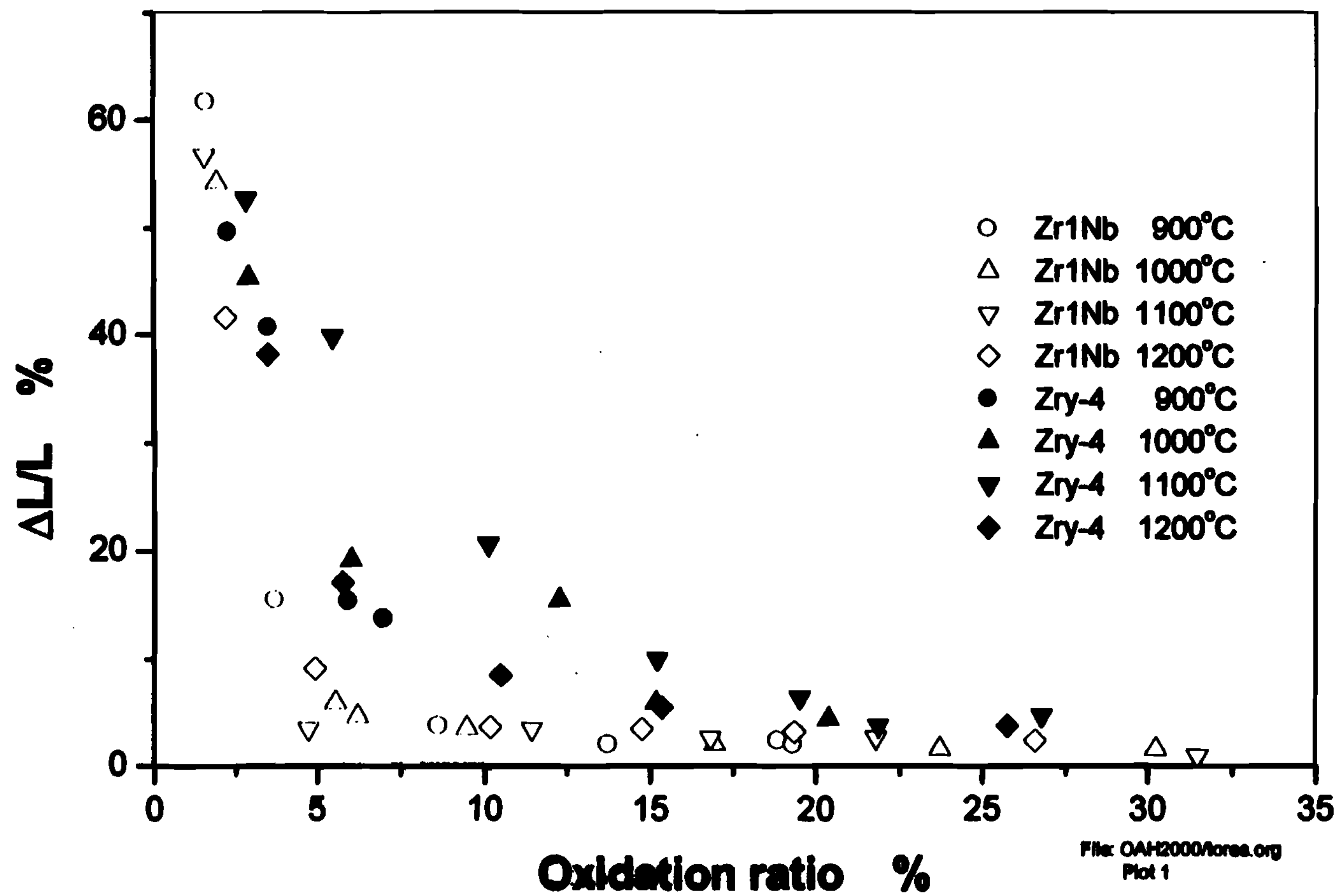
very good linear correlation for ZDT

The line reaches 275F (135°C) for a weight gain of 4.7 mg/cm² that is to say 6% ECR

The application of the hearing methodology leads to ZDT ≤ 275F or ECR ≤ 6%

PRESENTATIONS ON POST-QUENCH DUCTILITY

1. L. Maroti (AEKI, Hungary)
2. N. Sokolov (Bochvar, Russia)
3. A. Lebourhis (Framatome, France)
4. W. Leech (Westinghouse, USA)
5. H. Chung (Argonne, USA)



Ring compression tests with preoxidised Zr1%Nb and Zircaloy-4 claddings

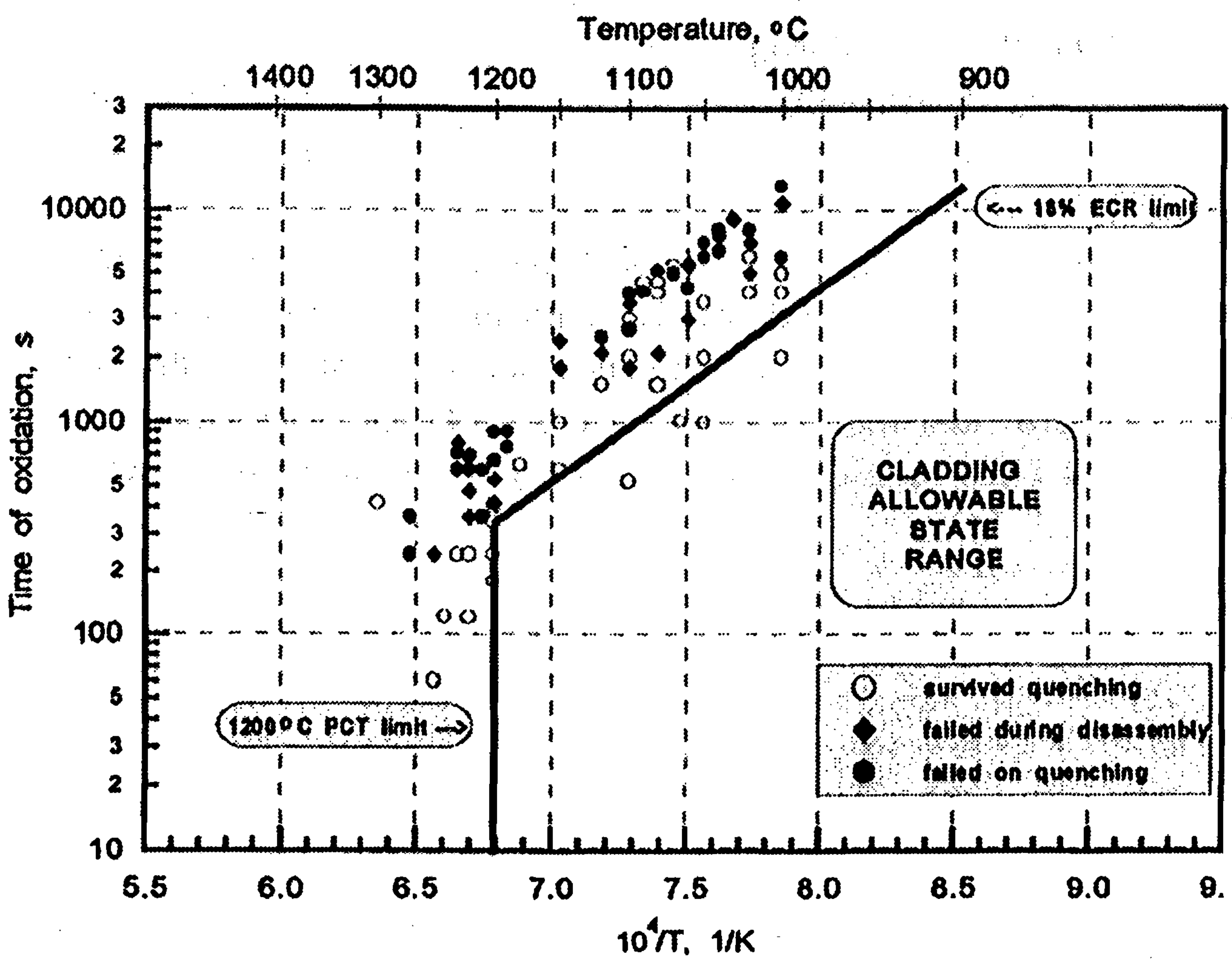
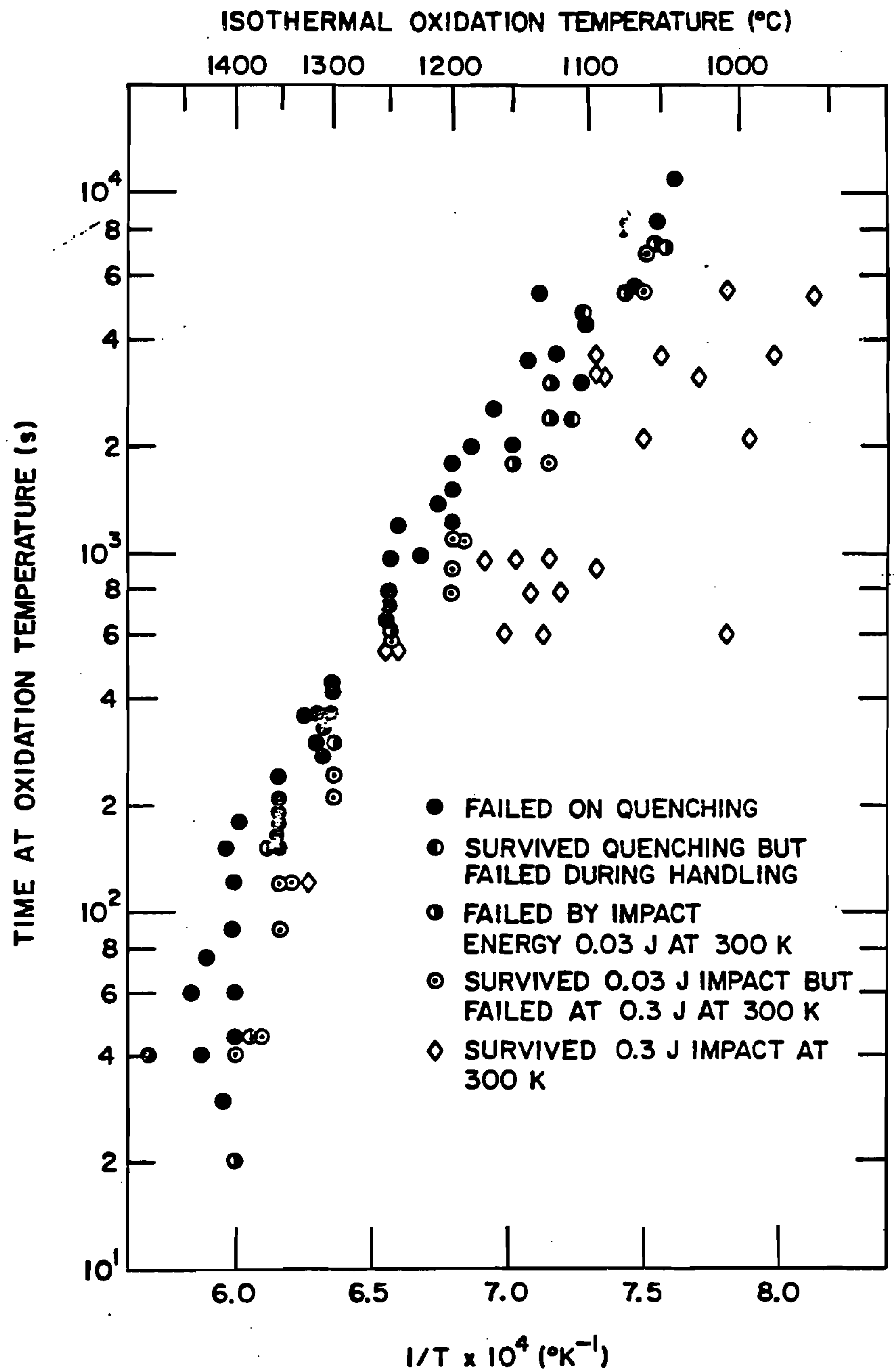
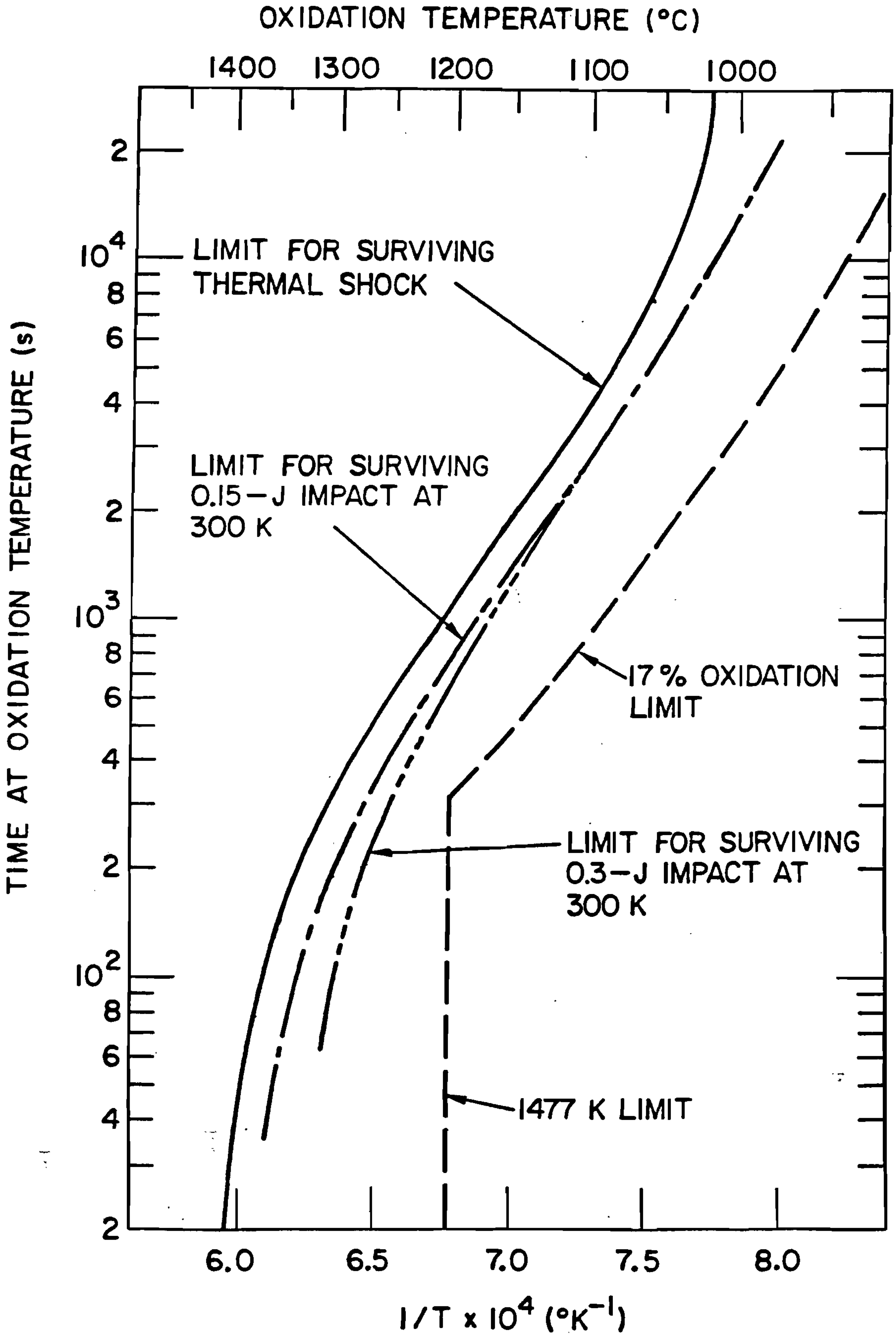


Fig.5 Post-test appearance and failure map.
Simulator of type 1



R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nuclear Boiling or Dryout, NUREG-0562 (1979) Fig. 2-A



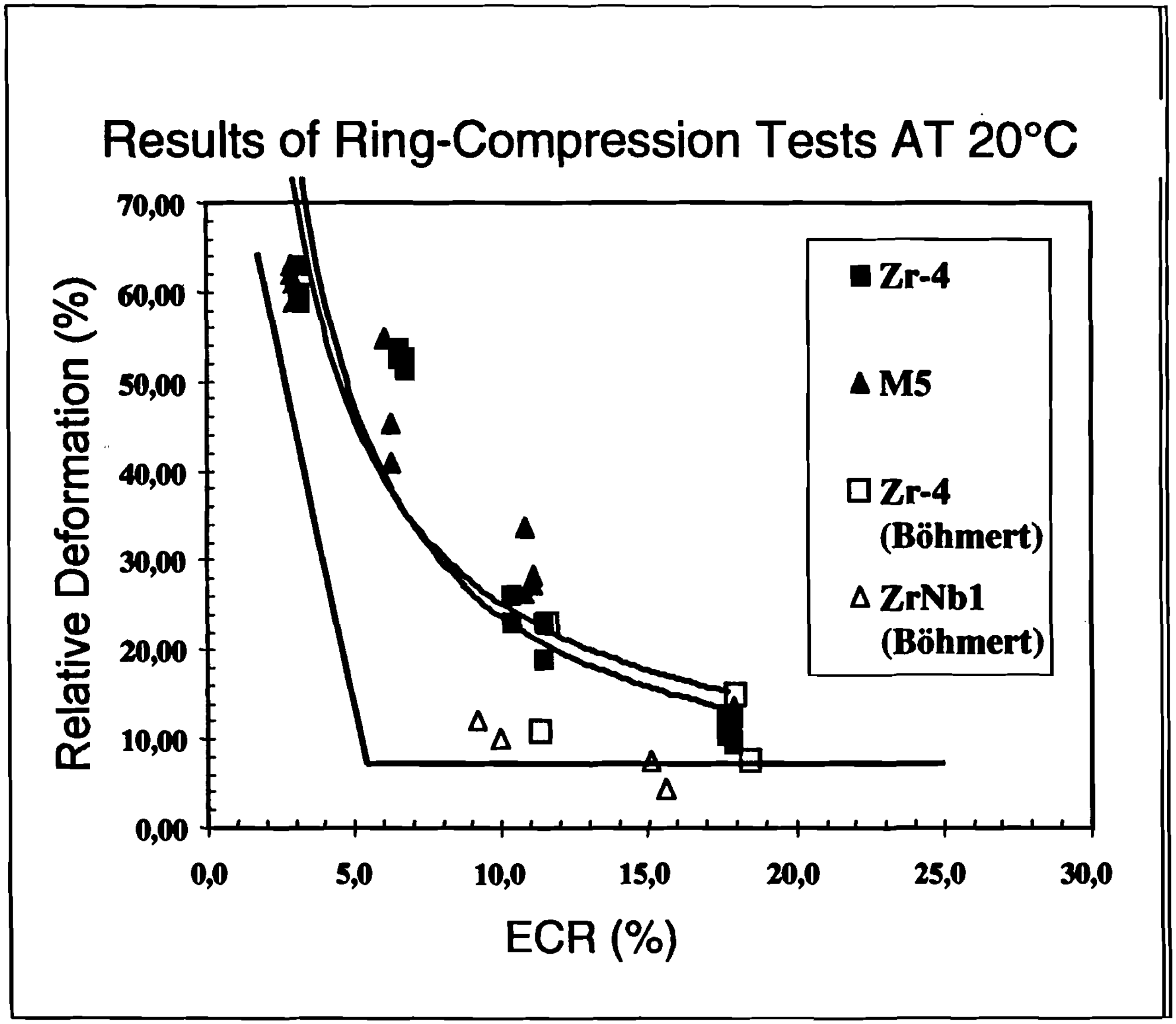
R. Van Houten, "Fuel Rod Failure as a Consequence of Departure from Nuclear Boiling or Dryout, NUREG-0562 (1979) Fig. 2-B

4. QUENCH EMBRITTLEMENT : MAIN RESULTS (1/2)

ALLOY	OXIDATION TEMPERATURE(°C)	TIME TO FAILURE (S)	ECR FAILURE (%)
Zy4	1000	6500	22*
	1100	2970	30
	1200	950	29
	1300	390	29
M5™	1000	13500	16 *
	1100	2959	28
	1200	1200	30
	1300	495	31

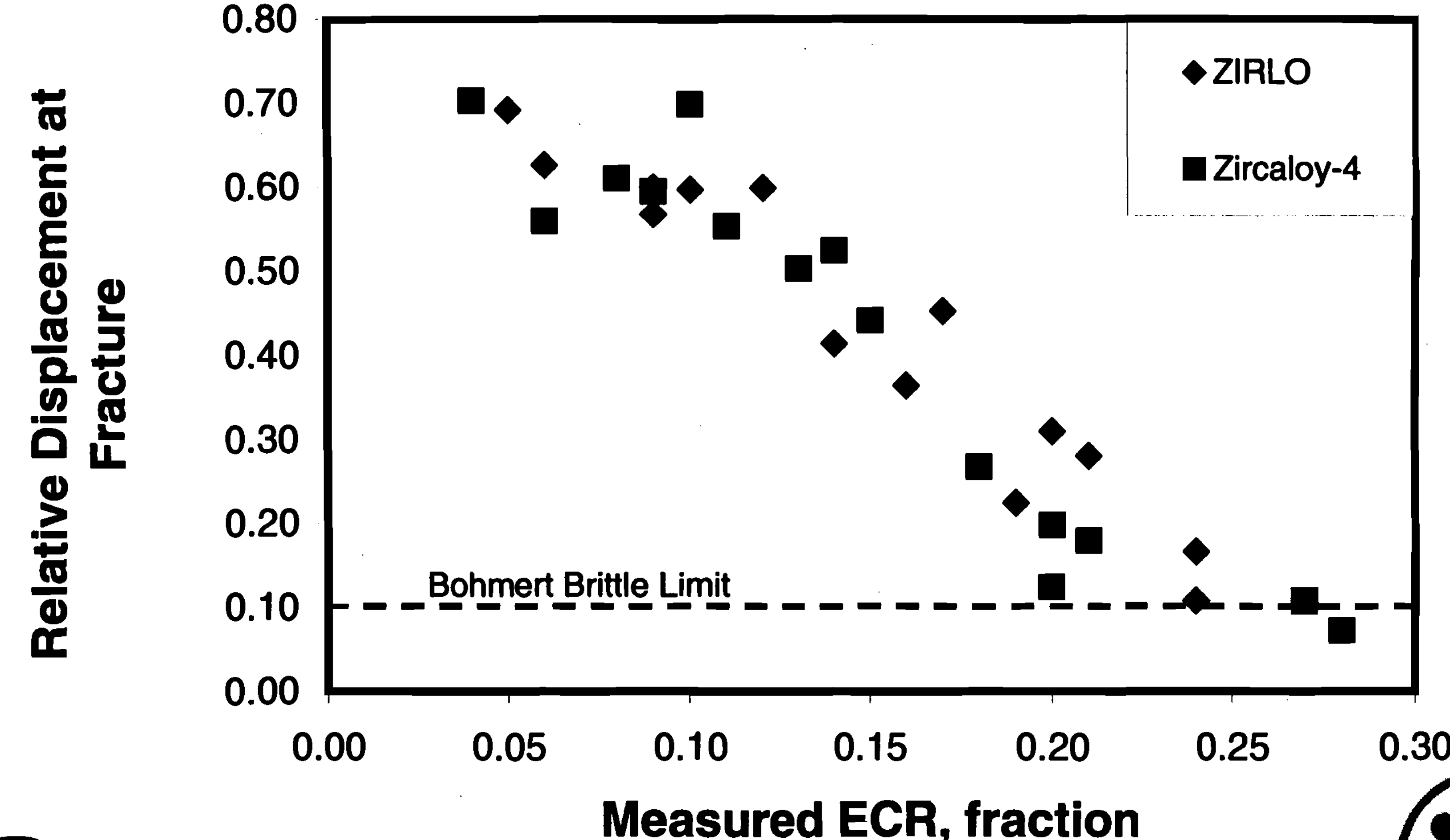
* conservative value

COMPARISON WITH BÖHMERT'S RESULTS AT 1100°C



March 22, 2001
Slide #6

Relative Displacement at Fracture vs Measured ECR at a Temperature of 275F (PRELIMINARY)



BNFL

CONCLUDING OBSERVATION

We need to understand the effect of small materials differences so we don't have to repeat all tests every time the manufacturer makes a small change in the alloy.

G. Hache
(more or less)

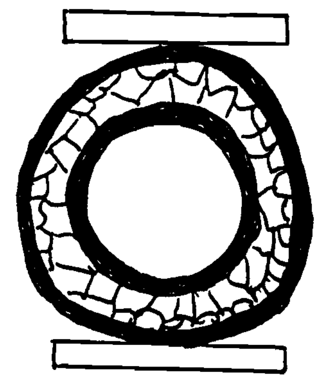
Comparison of M5 and E110 Composition
(Both are recrystallized)

Element	M5 Composition wt%	E110 Composition wt%
Zr	~99 (balance)	~99 (balance)
Nb	0.95 (0.80-1.20)*	0.9-1.1 (0.95-1.05)**
O	0.114 (0.09-0.18)*	<0.1 (0.05-0.07)**
Fe	0.054 (0.015-0.060)*	<0.05 (0.006-0.012)**
Cr	0.0029	<0.02
Si	<0.003	<0.02
C	0.0026	<0.02
Ni	<0.005	<0.02

Data from Halden report HWR-636
* ASTM Toronto, p. 506
** ASTM Garmisch-Partenkirchen, p. 787

Background
Post-Quench Ductility

- Key rationale for LOCA embrittlement criteria--1204°C (2200°F) PCT and 17% ECR limits:
  - **avoid zero-ductility in cladding**
  - ensure coolable core geometry



- Primarily based on Hobson's test 1972-73:
  - Zircaloy-4 tube oxidized at 1100-1315°C on two sides
  - short ring cut, compressed 3.8 mm slowly
  - crack-free adherent oxide, **H uptake low, <150 wppm**
  - **reflects O-induced embrittlement only**
  - **H-induced ductility degradation negligible--unknown in 1973.**

Background (Continued)
Post-Quench Ductility

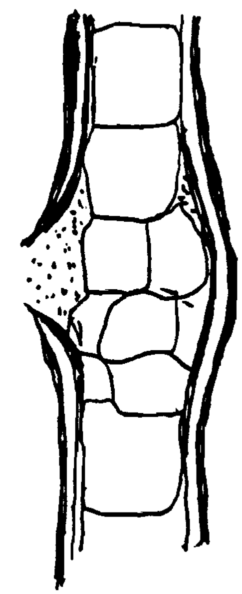
- H-induced embrittlement at H contents higher than about 600-700 wppm:
  - observed in 1980-1983, ANL & JAERI
  - local regions near burst opening, Zircaloy-4 tube
  - H alone (low O in beta layer) not much deleterious

- Significant effect of H uptake in E110 Zr-1Nb:
  - Boehmert 1992, Griger et al. 1999
  - at H contents higher than about 150-200 wppm

- Effects of 4 factors appear inseparable:
  - oxidation (before and during LOCA transient)
  - H uptake (larger than a threshold amount)
  - high burnup
  - Nb addition (E110, M5, Zirlo, Alloy A)

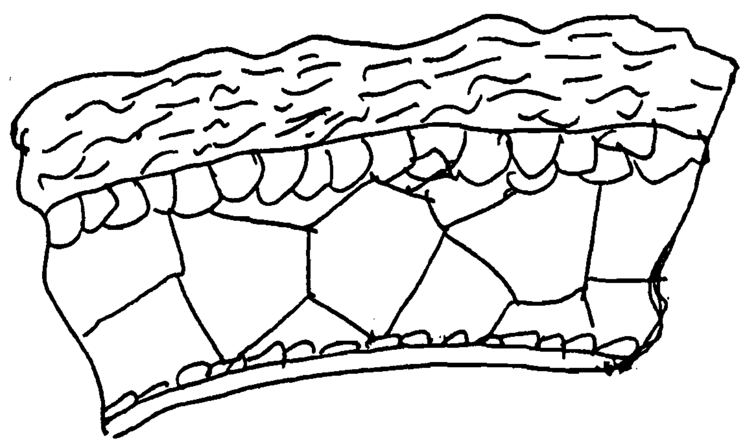
3 Routes for Large Hydrogen Uptake

#1 During normal operation to high burnup (≈62 MWd/kgU)
- standard Zircaloy-4 up to ≈700-800 wppm
- low-Sn Zircaloy-4, Zirlo
- M5



#2 Through "unprotected" ID surface near burst opening

#3 Through "high-temperature breakaway" oxides on the OD surface
- H uptake through normal high-temperature oxide (crack-free, tetragonal, protective) is limited to <150 wppm.



heechung@anl.gov

SCOPE OF WORK ON HIGH-BURNUP ISSUES AT ARGONNE
(PIRT Adjusted, EPRI Cooperation)

- Testing in Current ANL Program for Zircaloy-2 and Zircaloy-4 (Target 2003)
  - Integral Test (Ballooning, Rupture, Oxidation, Quench — with Fuel)
  - Oxidation
  - Thermal Shock (to be determined)
  - Phase Relations
  - Mechanical Properties (including Post-Quench Ductility)
  - Post-LOCA Seismic Loading
  - Fuel Relocation (limited to Observation during Integral Test)

- NRC is Interested in Conducting Confirmatory Tests on ZIRLO and M5

- May only need Subset of Tests for Other Cladding Types like ZIRLO and M5
  - Oxidation
  - Thermal Shock (to be determined)
  - Phase Relations
  - Mechanical Properties (including Post-Quench Ductility)

PROPOSED WORK ON UNIRRADIATED ZIRLO AND M5
(Target 2001)

- Review All Test Methods to Determine Test Conditions (Zircaloy Specimens first)
- Agreement on Test Conditions will involve EPRI, Westinghouse, and Framatome
- Post-Quench Standard Test (perhaps Axial Tensile Test) on Unirradiated Cladding
- Post-Quench Ring-Compression Tests (probably also) on Unirradiated Cladding
- Oxidation Rate and Phase Relations as needed to interpret Ductility Results
- No Mechanical Properties or Other Testing at this Time (later in High Burnup Program)
- Proprietary Treatment of Data may be arranged if Requested

PROPOSED COOPERATION

- Pattern after Current ANL Program with EPRI Cooperation
- Westinghouse and Framatome would be Included in all Test Planning
- EPRI is also Interested in further Cooperation (Subject to Approval of RFP)
- Once Agreement is Reached, Start Unirradiated Testing in 2001 and Irradiated Testing in 2003

NRC PROGRAM FOR ADDRESSING EFFECTS OF HIGH BURNUP AND CLADDING ALLOY ON LOCA SAFETY ASSESSMENT

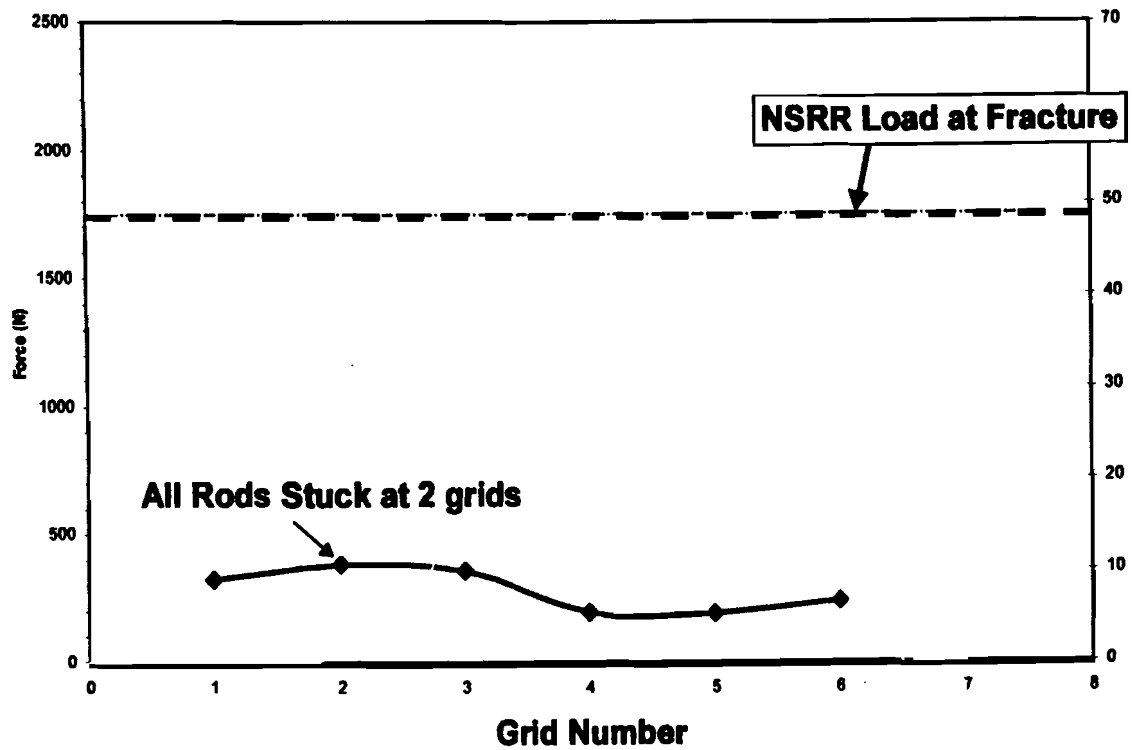
R. Meyer
NRC Office of Nuclear Regulatory Research

PRESENTATIONS ON AXIAL CONSTRAINT DURING QUENCHING

1. H. Uetsuka (JAERI, Japan)
2. N. Waeckel (EPRI-EdF, USA-France)

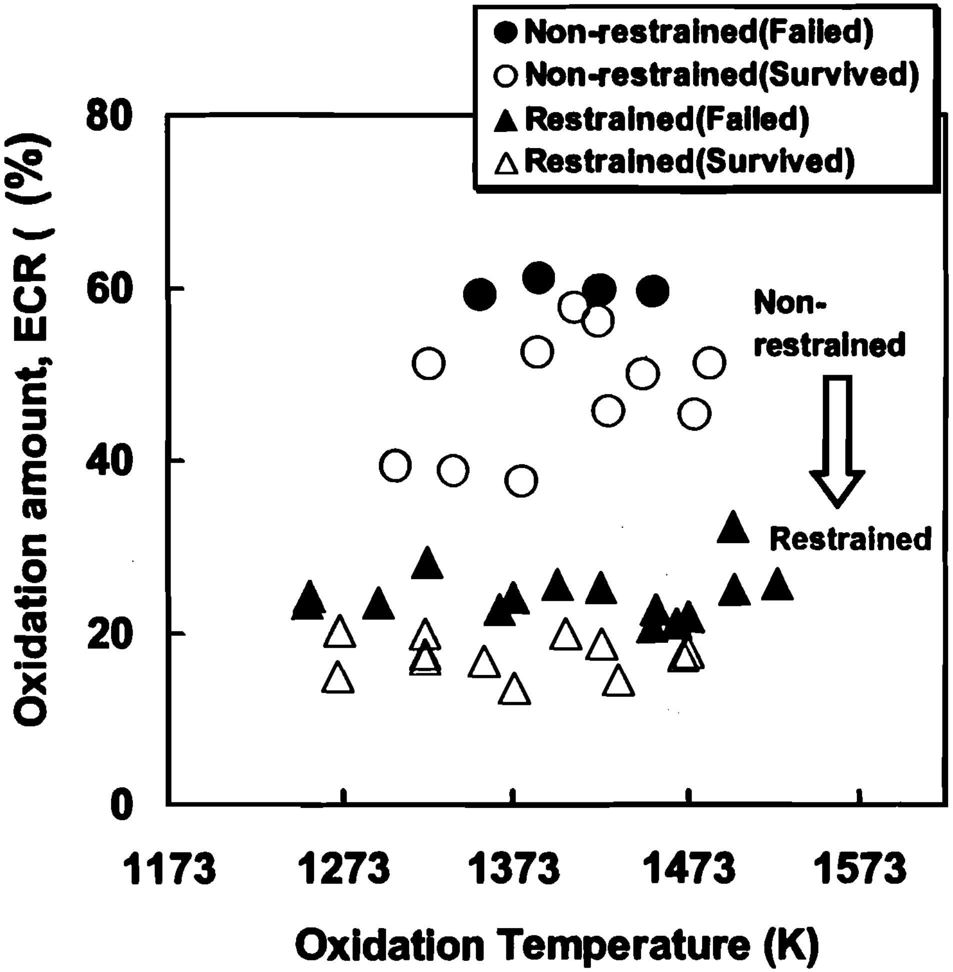
EPRI EDF

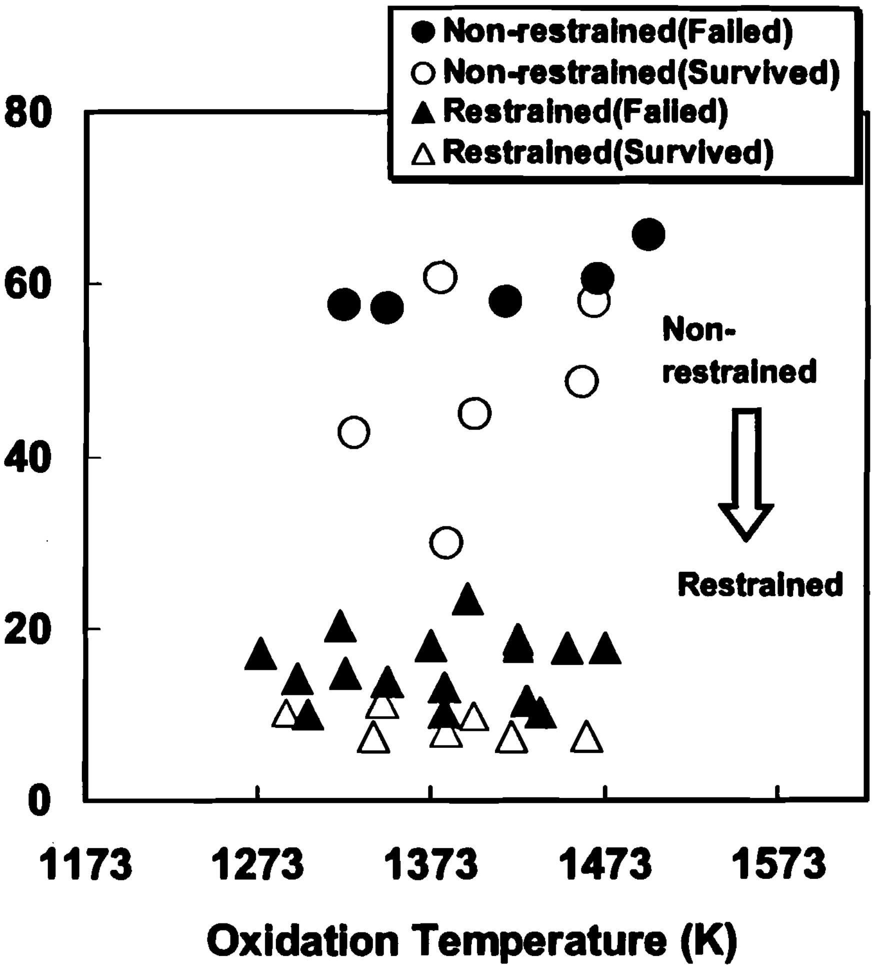
Fuel Rod Axial Force Distribution



Robust Fuel Program

Failure map(2/2) -Restraint condition-





JAERI

PRESENTATIONS ON FUEL RELOCATION INTO BALLOONS

1. M. Lambert (EdF, France)
2. C. Grandjean (IPSN, France)

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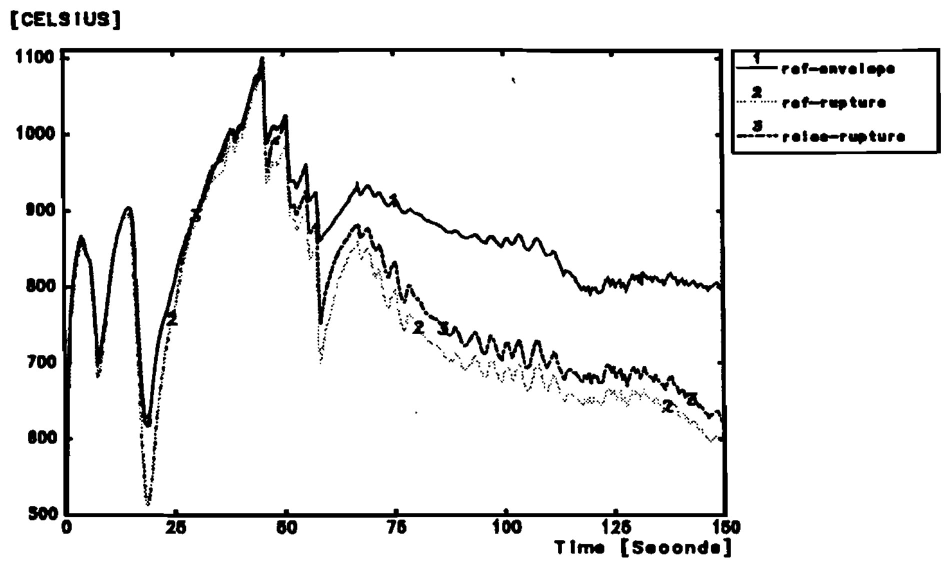
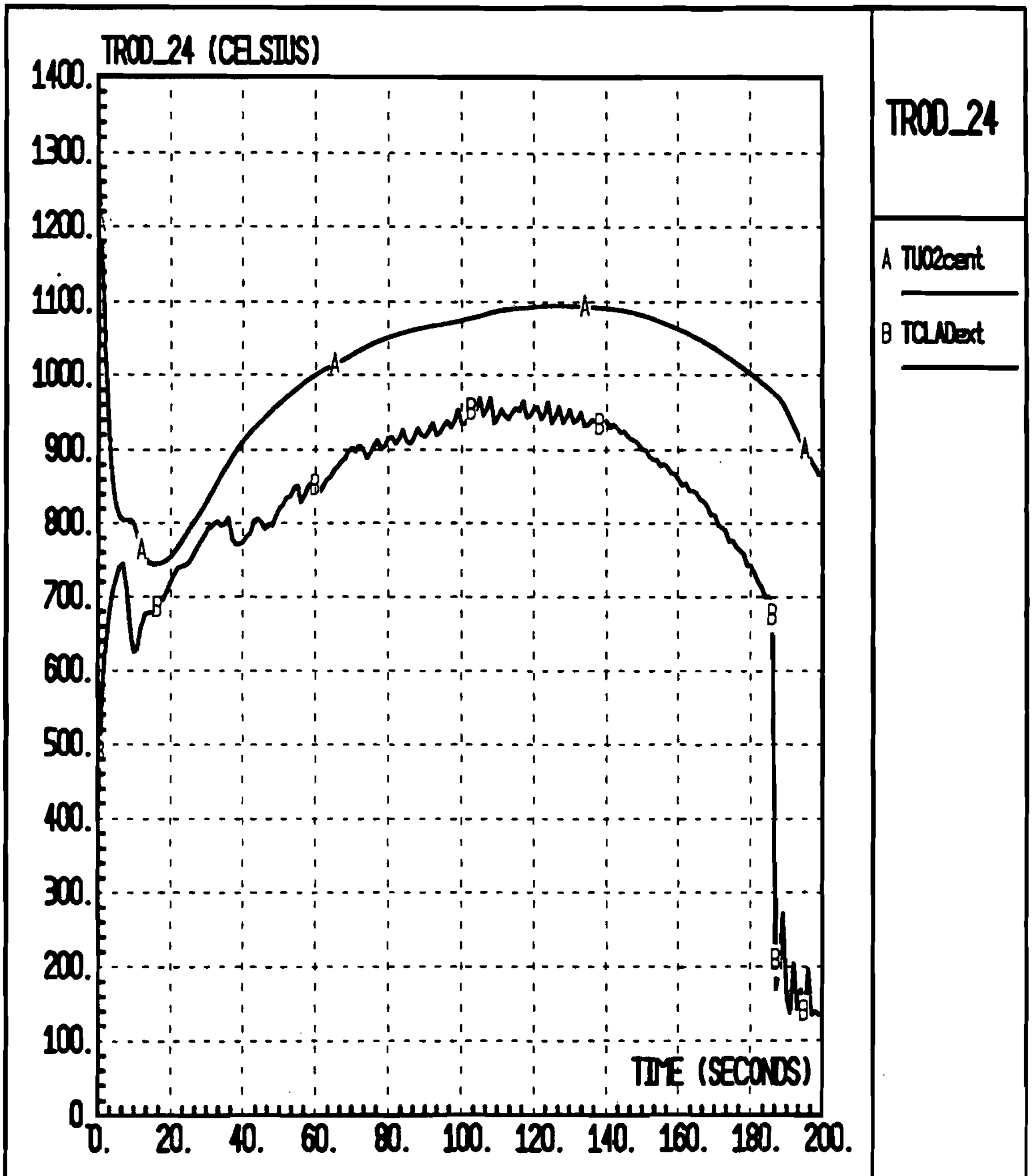
Relocation in Large Break LOCA

Calculation



cladding temperatures

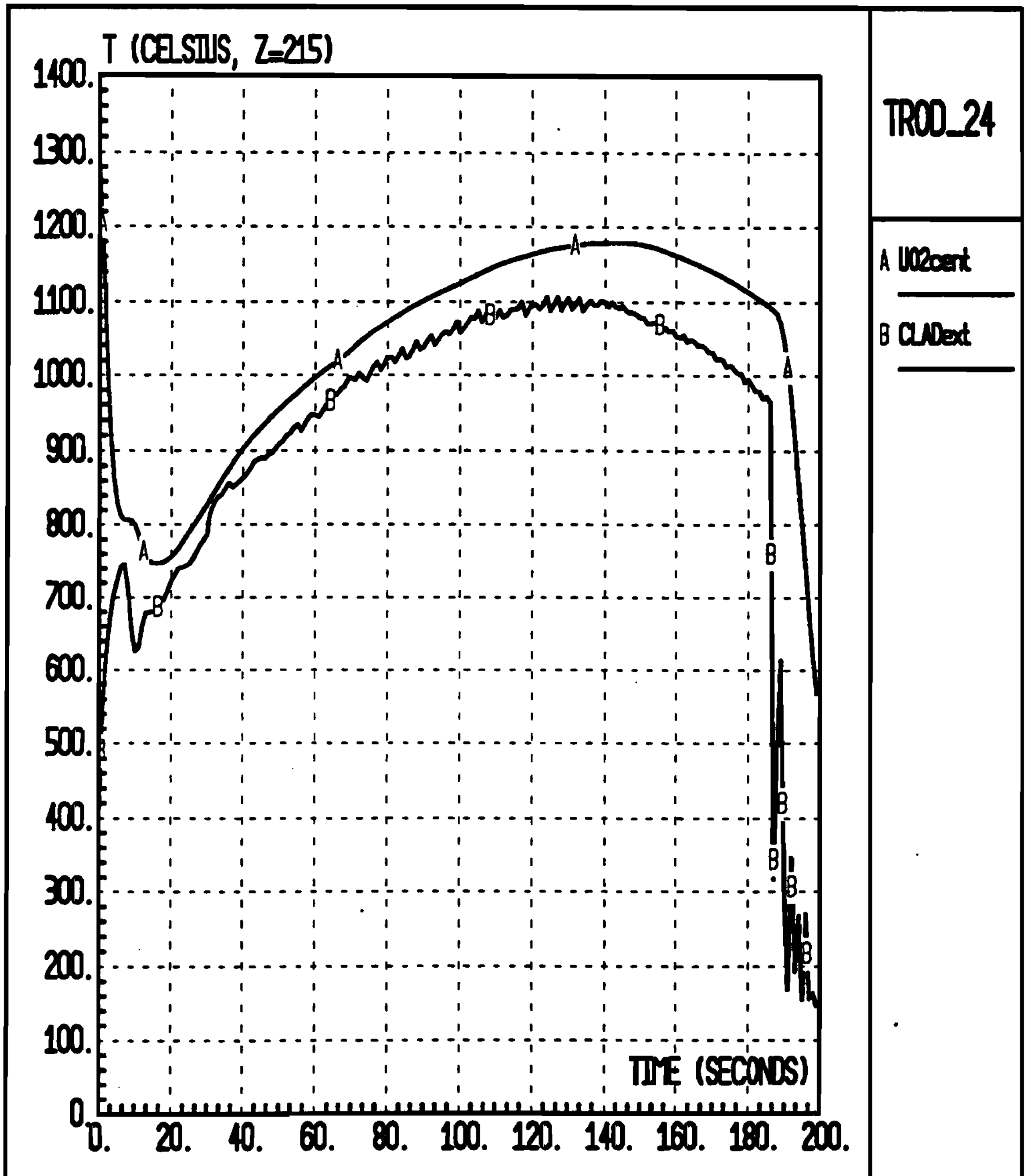


Large Break LOCA. Hot Rod. BU = 57 GWj/tU

No fuel relocation. No protective effect of initial oxide

CATHARE2

V1.3L priv



LB LOCA. Hot Rod. Burnup = 57 GWd/tU

Fuel relocation in rupt. mesh : 61.5% of balloon volume

CATHARE2

V1.3L priv

MME

LOCA Ductility of M5™ Cladding

Garry L. Garner
Framatome ANP
Lynchburg, Virginia

Outline

- Review of In-Reactor Operating Experience
  - Alloy composition, fabrication parameters
  - Corrosion/hydrogen properties

- Review of High Temperature Testing
  - Oxidation Tests
  - Quench Tests
  - Post-Quench Mechanical Testing

- Conclusions and Summary

In-Reactor Performance

Alloy M5™

- Composition:
  - Sn: An impurity in M5™
  - Fe: Target 250 - 500 ppm (improve corrosion)
  - O: Target value 1250 - 1450 ppm (improve creep)
  - S: Maintain consistent creep behavior

- Thermomechanical processing
  - Low temperature annealing to insure stable microstructure

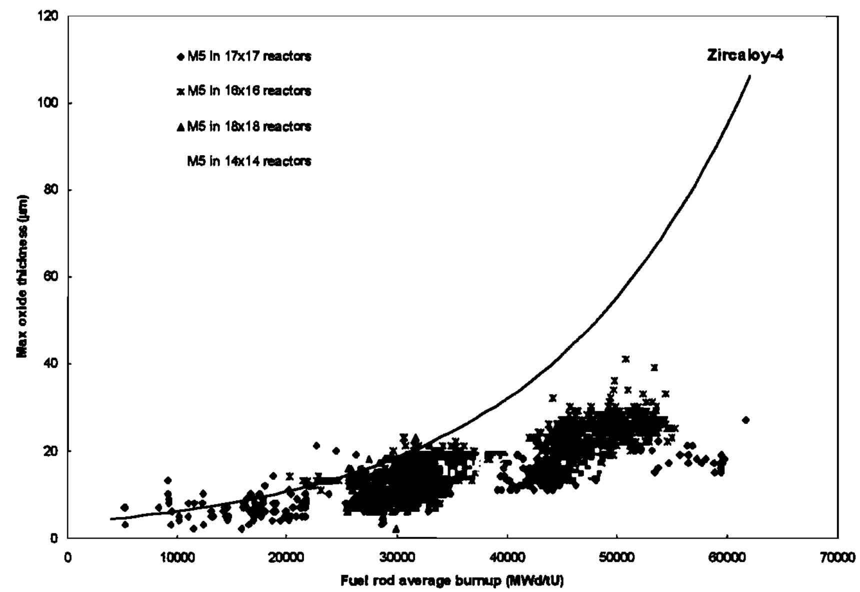
M5™ PWR Corrosion Performance

- ➢ Additional data in BU range 50-60 GWd/tU

- ➢ Excellent corrosion behavior of M5™
  - ➢ for all designs and for all operating conditions

- ➢ Thickness < 40 μm for BU up to 63 GWd/mtU

Corrosion behavior of Zirconium alloy claddings



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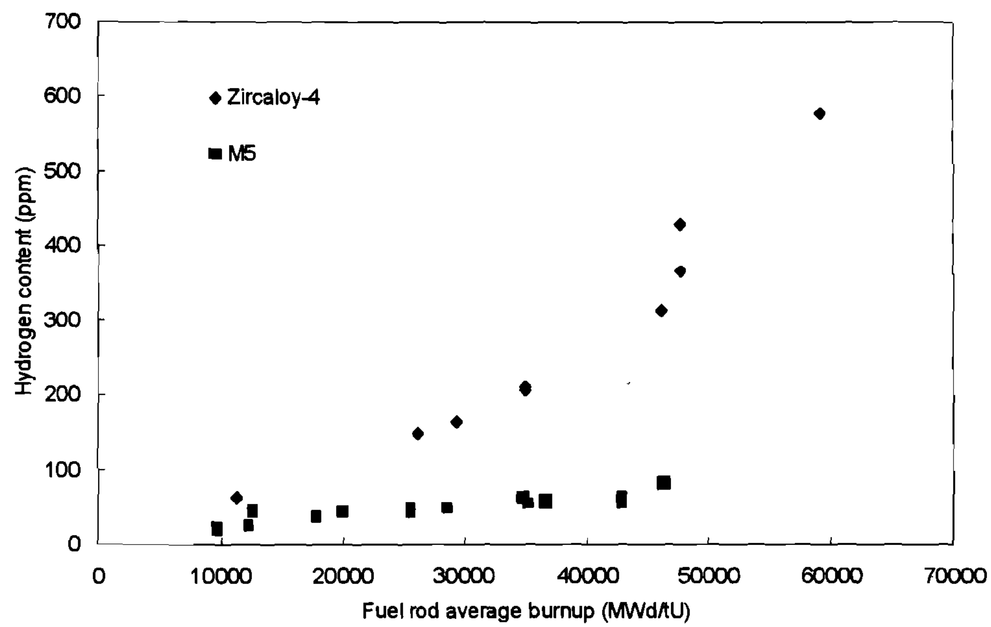
PWR Hydrogen Performance Of M5™

- Significant reduction of clad hydrogen content

- Additional M5™ data at high burnup planned in 2001



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Summary: M5™ In-Reactor Performance

- Low oxidation rate
- No increase in rate to burnups of 63 GWd/mtU
- Lower sensitivity to temperature and rod power than Zr-4 (reactor duty)
- **Low oxidation rate + low hydrogen absorption = low hydrogen content at high burnup**

CINOG High Temperature Testing

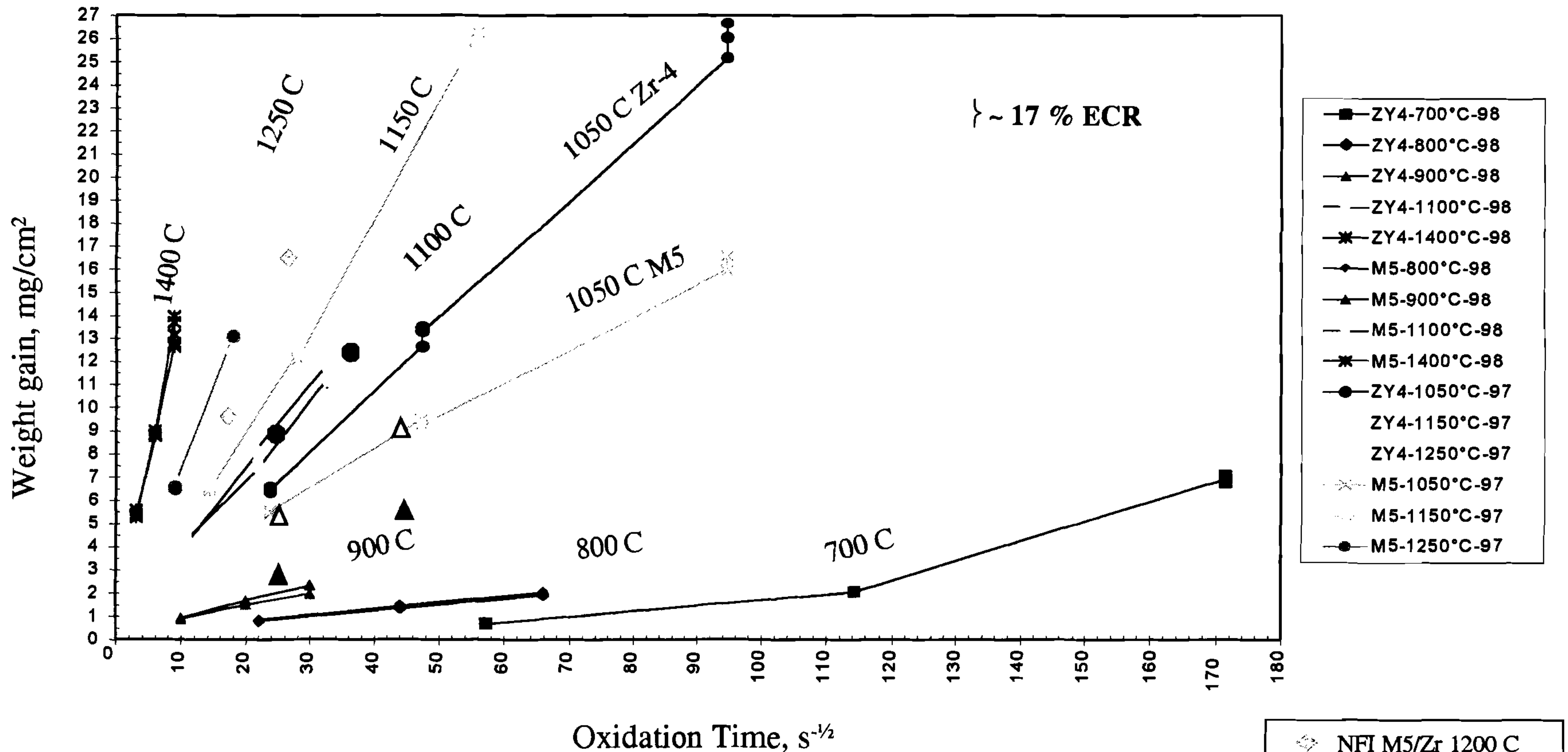
Facility

FRAMATOME ANP

High Temperature Oxidation (CINOG) Test Matrix

- Oxidation Tests (M5™ and Zr-4)
  - Double-Sided Oxidation (L = 20 mm)
  - T = 700, 800, 900, 1050, 1100, 1150, 1250, 1400 C
    (as manufactured cladding)
  - T = 1200 C for Pre-Hydrided Cladding
    (200 ppm for M5, 200 and 450 ppm for Zr-4)
  - 3 Oxidation times/Test Temperature → (50, 100, and 200 μm/side)
  - 3 Samples/Test Condition (2 repeat tests)

Oxidation 700 TO 1400°C
Zr-4 and M5™



- M5™ behaves better than Zr-4 at 1050°C
- Zr-4 values are consistent with literature
- M5™ values are consistent with independent Japanese tests

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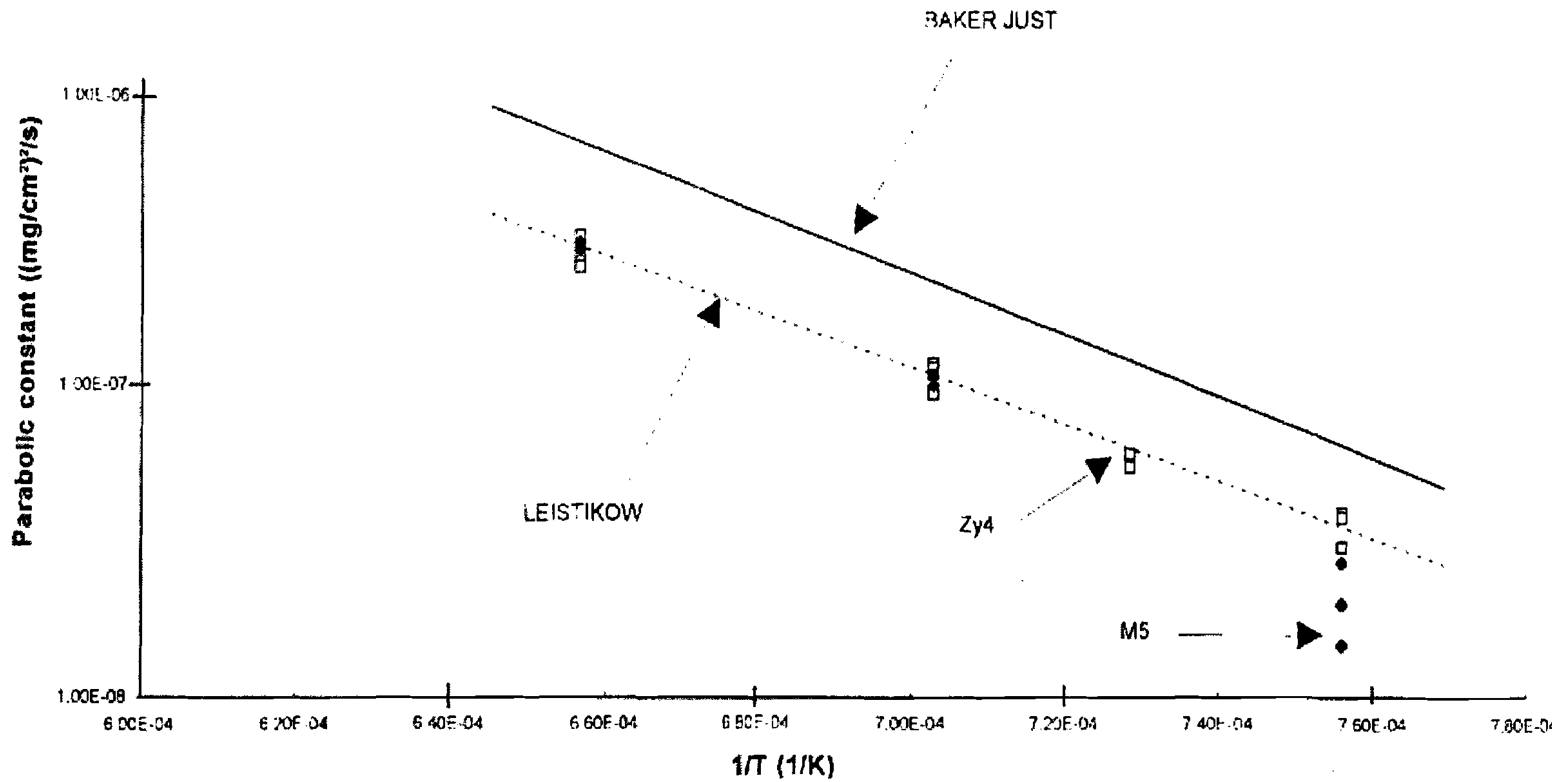
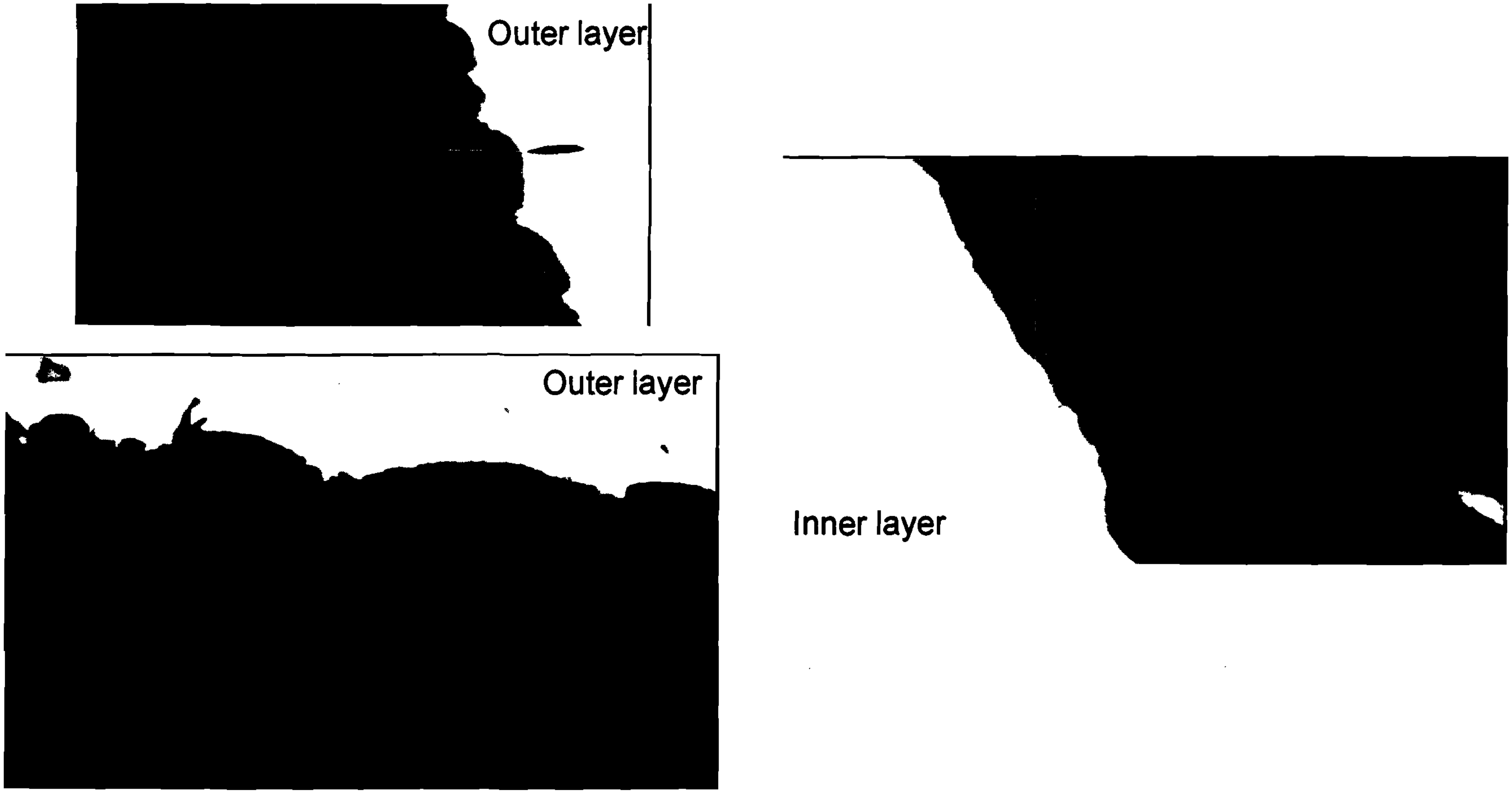
Oxidation Kinetics - CINOG
Comparison with literature results



→ BAKER-JUST MODEL IS BOUNDING IN ALL ENCOUNTERED CONFIGURATION

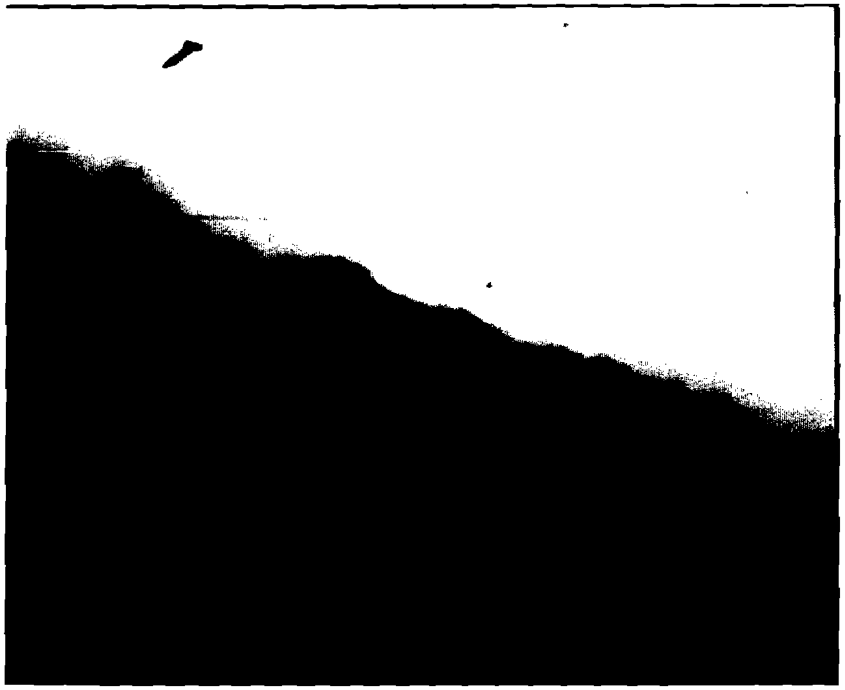
→ LESTIKOW MODEL ACCURATELY PREDICTS ZY4 RESULTS AND IS BOUNDING FOR M5™

Zy-4 Metallographic Observations After Oxidation at 1000°C for 3,270 Seconds

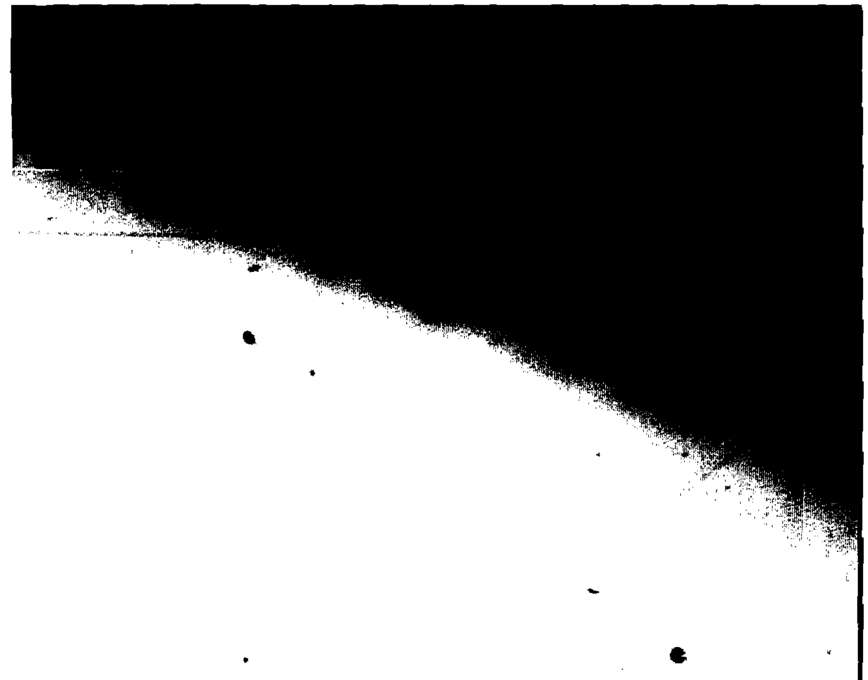


→ Trace of delamination in inner and outer zirconia layers

M5™ Metallographic Observations After Oxidation at 1000°C for 3,270 Seconds



Outer layer



Inner layer

- The inner and outer zirconia layers are homogeneous
- No trace of delamination

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Metallographic Observations M5™ and Zr-4 After Oxidation at 1000 °C for 3,270 seconds

	Zr-4	M5™
External Zirconia Layer (μm)	55.2 to 61	18.9 to 20.3
External α Zr-O Layer (μm)	53.9 to 71.6	53.9 to 61.9
β_{Zr} Layer (μm)	351 to 379	394 to 409
Internal α Zr-O Layer (μm)	53.5 to 70.4	47.4 to 57.3
Internal Zirconia Layer (μm)	48.7 to 55.8	19.0 to 21.7

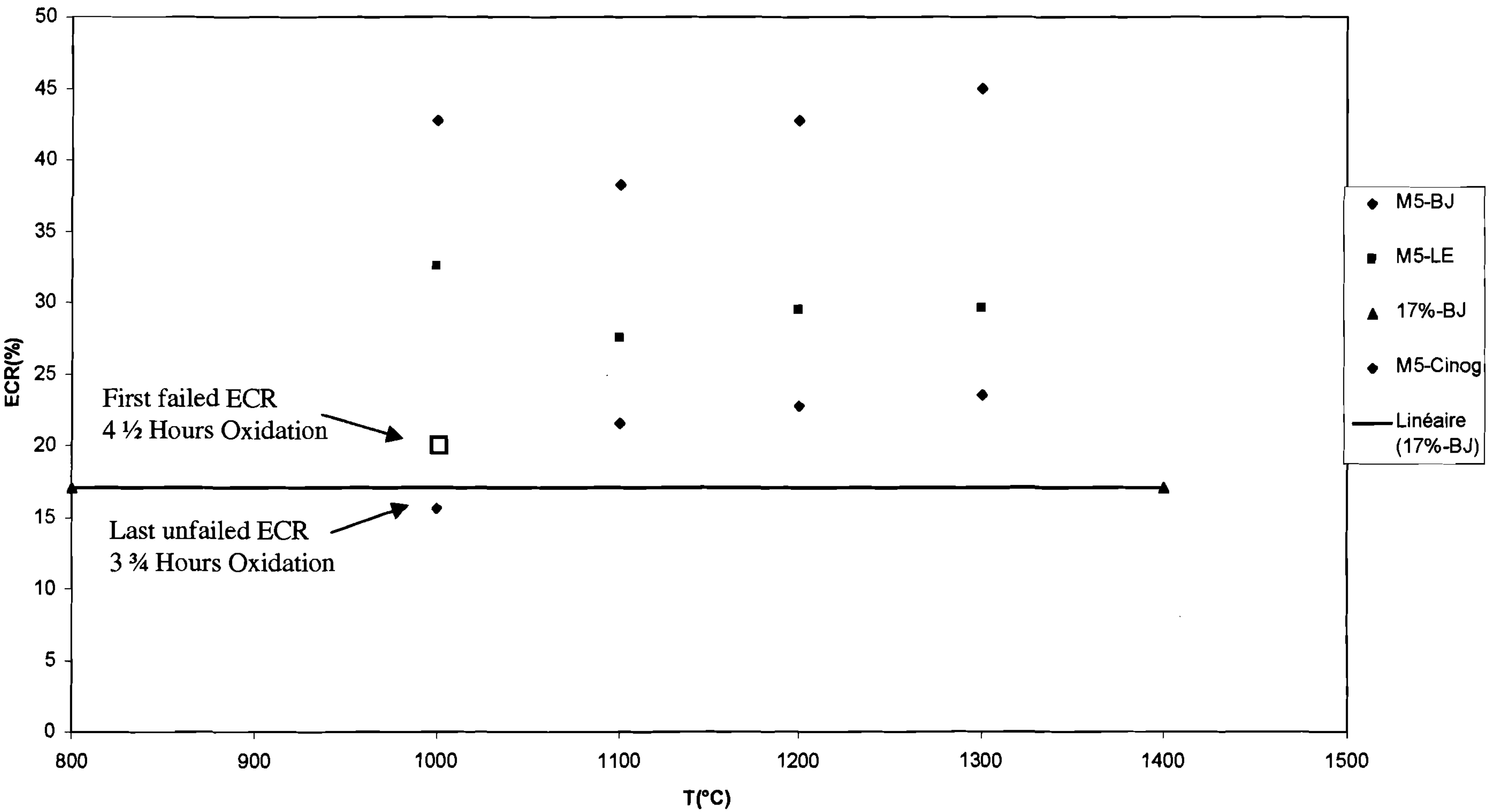
CINOG Quench Test Matrix

- Quench Embrittlement Tests (M5™ and Zr-4)
  - Double-Sided Oxidation (L = 100 mm)
  - Cladding Failure when Cladding Leaks Air Under Slight Overpressure
  - T = 1000, 1100, 1200, 1300 C for as manufactured cladding
  - T = 1200 C for Pre-Hydrided Samples
    (200 ppm for M5, 200 and 450 ppm for Zr-4)
  - Generally 5 or more Tests to Establish Cladding Failure Threshold
- Post-Test Metallography and Hydrogen Analysis

Quench Test Results

Alloy	Oxidation Temperature (^{o}C)	Time to Failure (seconds)
Zr-4	1000	6,500
	1100	2,970
	1200	950
	1300	390
M5™	1000	13,500
	1100	2,959
	1200	1,200
	1300	495

CINOG Quench Test
ECR versus temperature



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CINOG Quench Test

HYDROGEN CONTENT IN Zy-4 AND M5™ AFTER QUENCH

Alloy	Oxidation temperature (°C)	Duration (sec)	(H2) (ppm)
Zy4	1100	2970	24-32-22
ZY4	1200	950	21-22-22
Zy4	1300	390	26-25-25
M5™	1100	2959	18-18-20
M5™	1200	1200	16-19-17
M5™	1300	495	21-24-21

→ Maximum oxidation duration before embrittlement similar or higher for M5™

→ Slight hydrogen pickup, practically temperature-independent

CINOG Test Results Summary

- High Temperature Oxidation Performance of M5™ is Equivalent or Superior to Zr-4
- M5™ Hydrogen Uptake is Low
- M5™ Accident Survival is Superior to Zr-4

  T > 1100 C M5™ and Zr-4 Have Similar Survival Ability
  T < 1100 C M5™ Survives up to 2 Times Longer than Zr-4

- M5™ Does Not Exhibit Delamination of Oxide
- Using Baker/Just to Establish ECR M5™ Always Meets the 17 % Criterion
- At Moderate Temperatures (1100 C > T > 900 C) M5™ Requires Excessive Oxidation Times to Achieve ECRs near 17 %
- Because M5™ Actually Performs Better During an Accident, The LOCA Criterion Should Remain 17 % Local Oxidation as Calculated by Baker/Just

Post-Quench Mechanical Tests

Post-Quench Mechanical Tests
Test Matrix

➢ Oxidation

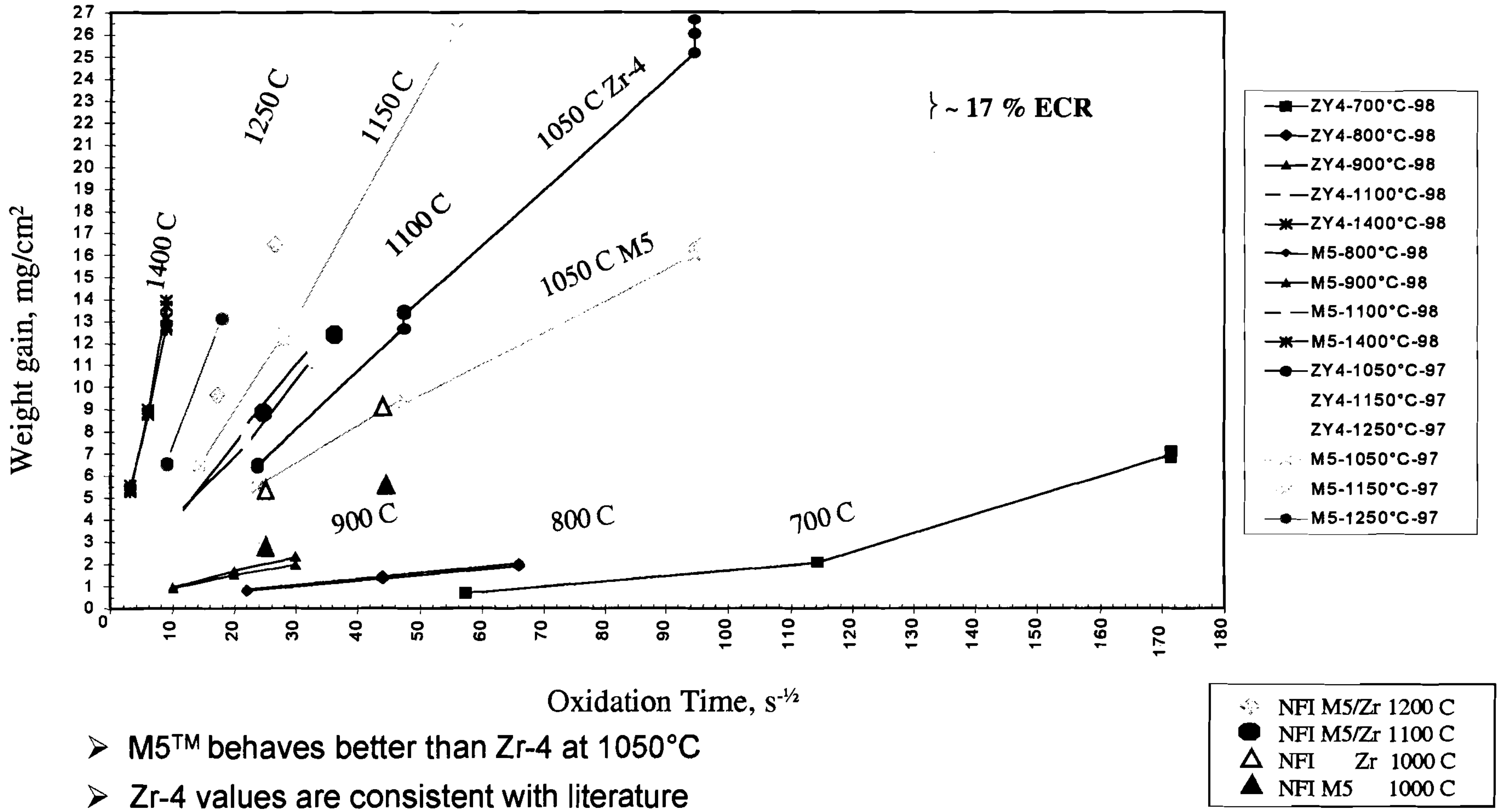
- T=1100 °C
- t → ECR=3, 6, 10 and 17% (Lestikow law)
- Single face oxidation
- As-fabricated M5™ and Zr-4 cladding tubes

➢ Water Quench

➢ Mechanical tests

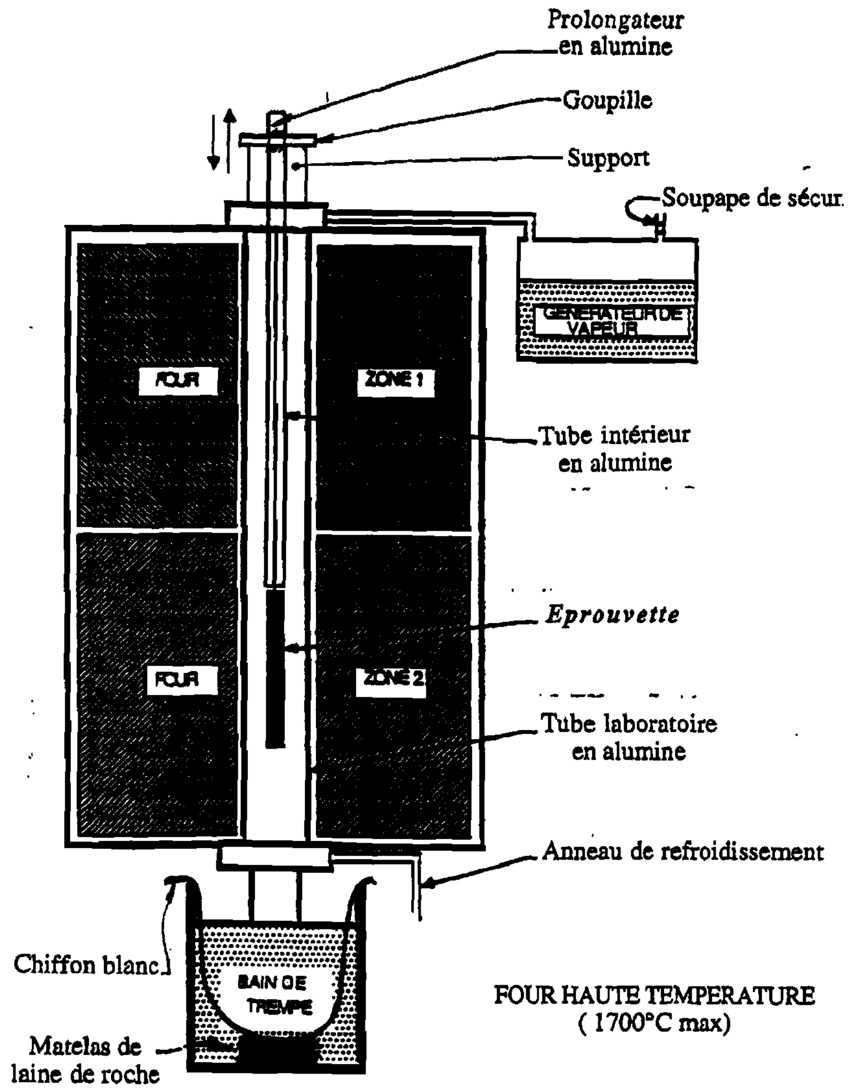
- Three point bend
- Impact
- Ring compression

OXIDATION 700 TO 1400°C
ZIRCALOY-4 and M5™



- M5™ behaves better than Zr-4 at 1050°C
- Zr-4 values are consistent with literature
- M5™ values are consistent with independent Japanese tests

Post-Quench Mechanical Test
Oxidation - Device



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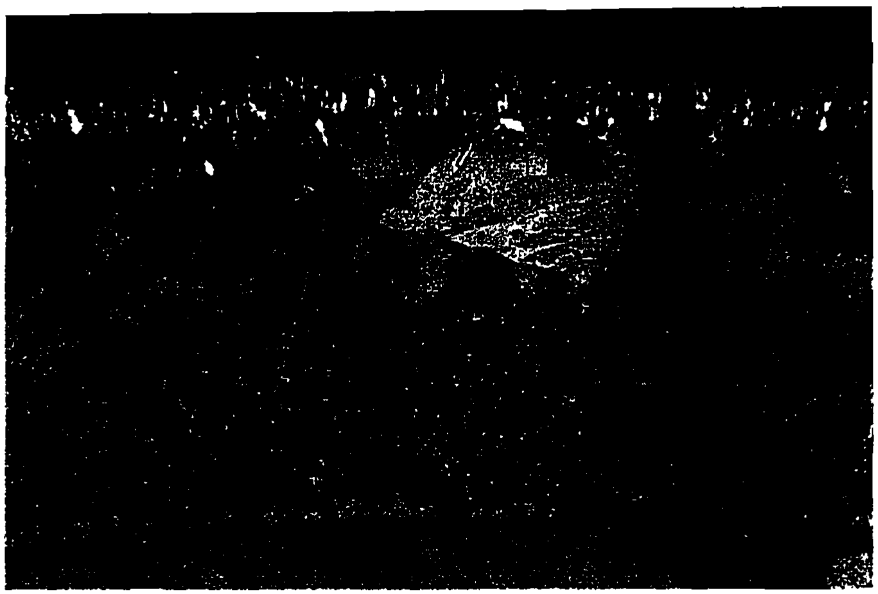
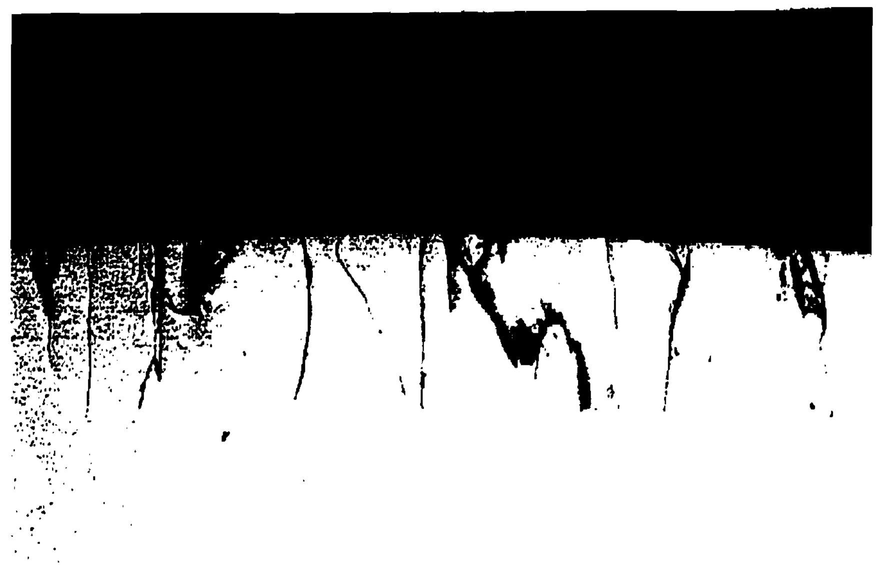
Post-Quench Mechanical Test

Percent of spalled oxide after oxidation at 1100 °C and quench for the longest exposure time (3098 - 3800 s)

Alloy	Test Number	Weight Gain During Oxidation (g)	Oxide Spalled (g)	Oxide Spalled (%)
Zr-4	71	1.0839	0.7215	66.9
	74	1.0799	0.6975	64.6
	77	1.0919	0.9088	83.2
M5	73	1.1634	0.0230	2
	76	1.1544	0.0259	2.2
	79	1.1696	0.0458	3.9

Metallographic Observations Of Low-Tin Zr-4 After Oxidation At 1100°C t = 1349 s and Quenched





- α-Zr(0) layer: large α-grains
- α-Zr(0) layer: cracks

FRAMATOME ANP

Metallographic Observations Of M5™ After Oxidation At 1100°C t = 3600 s and Quenched

X100

X200

- α Zr (O) layer : Linear distribution of niobium particles in α platelets
- α Zr (O) layer : no cracks

FRAMATOME ANP

High - Temperature Oxidation
Russian Alloy E-110 Cladding

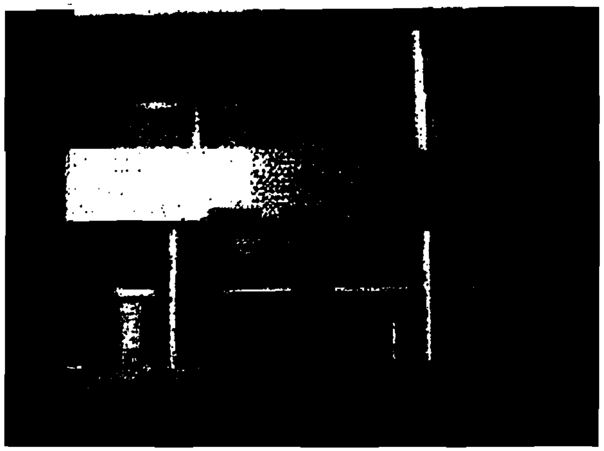
- ➢ Stratified and cracked oxide layer
- ➢ Different morphology than M5™

"… at an early stage, multilayer oxide scales are formed which tend to flake."

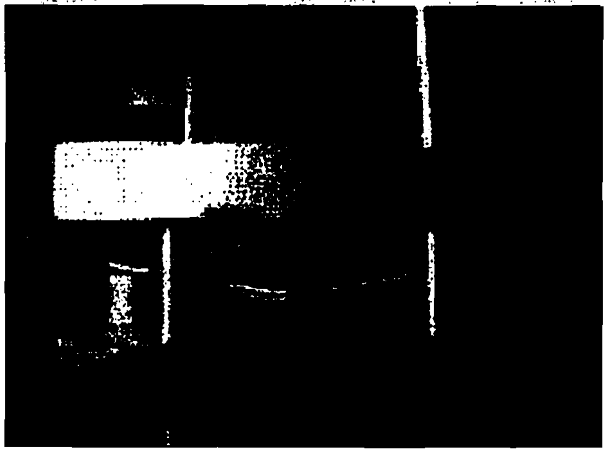
Böhmert et al. on Russian alloy E110

- M5™ has not exhibited multilayered oxide scale
- M5™ did not flake in quench tests

Post-Quench Mechanical Test
3 Point Bend Test Apparatus

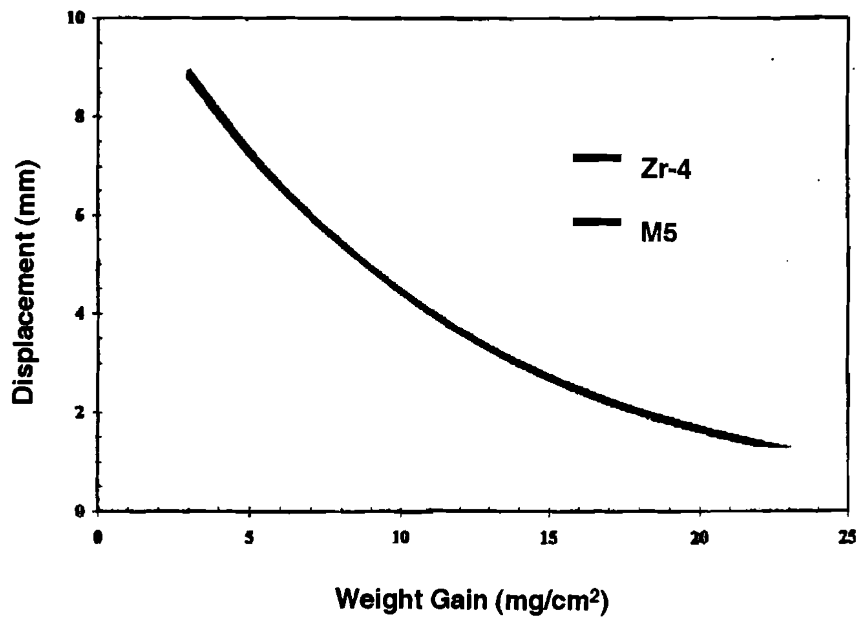


Starting position



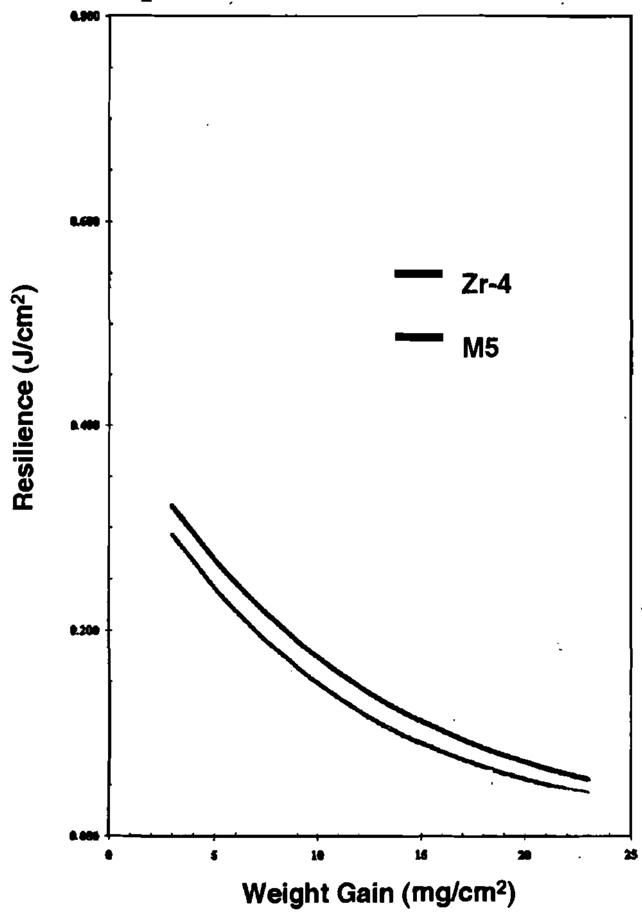
7.5 mm displacement

Post-Quench Mechanical Test
3 Point Bend Test Results



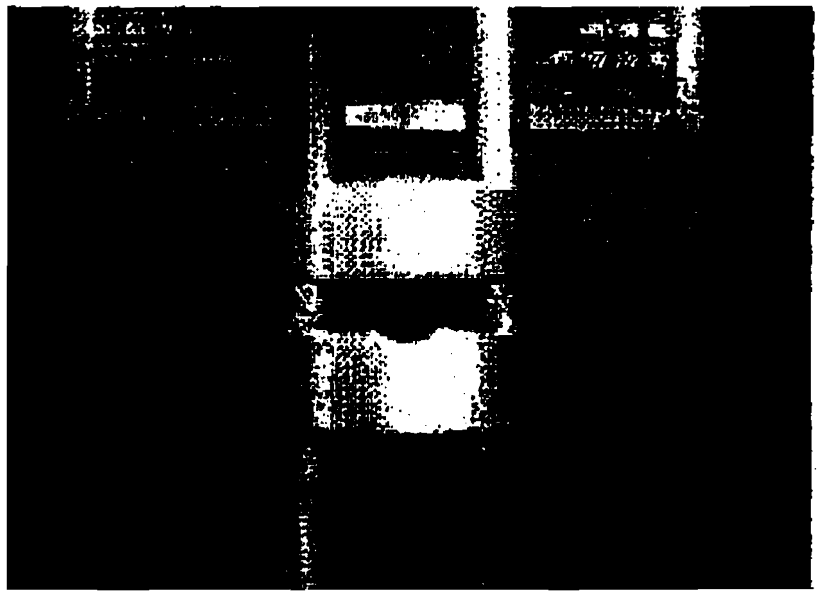
- M5™ and Zy4 behave similarly

Post-Quench Mechanical Test
Impact Test Results



- M5™ behaves slightly better than Zr-4
- Zy4: ductile rupture in ex α-β phase and brittle fracture in α-Zr(0)
- M5™: ductile rupture in ex-α-β phase and quasi-ductile in α-Zr(0) layer

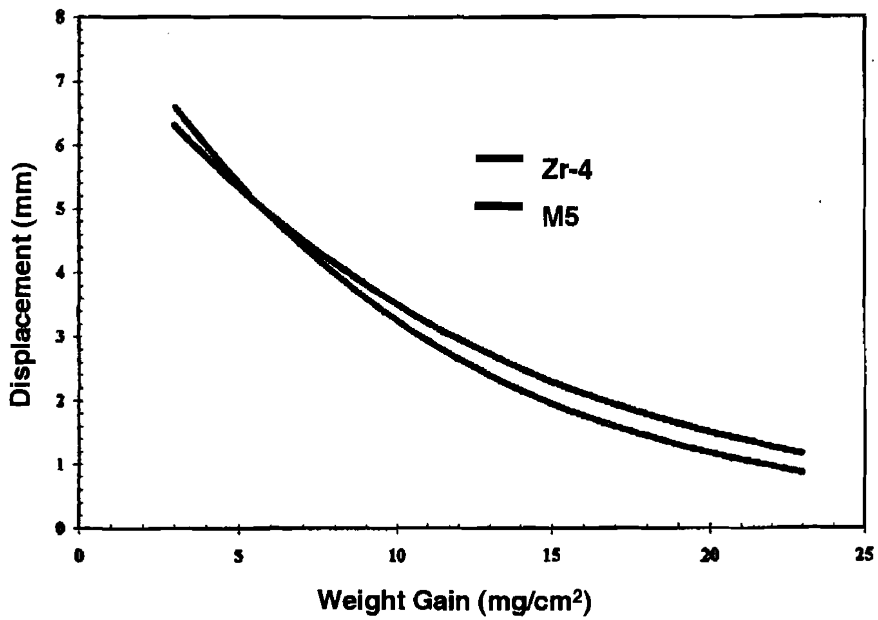
Post-Quench Mechanical Test
Ring Compression Test



Starting position

FRAMATOME ANP

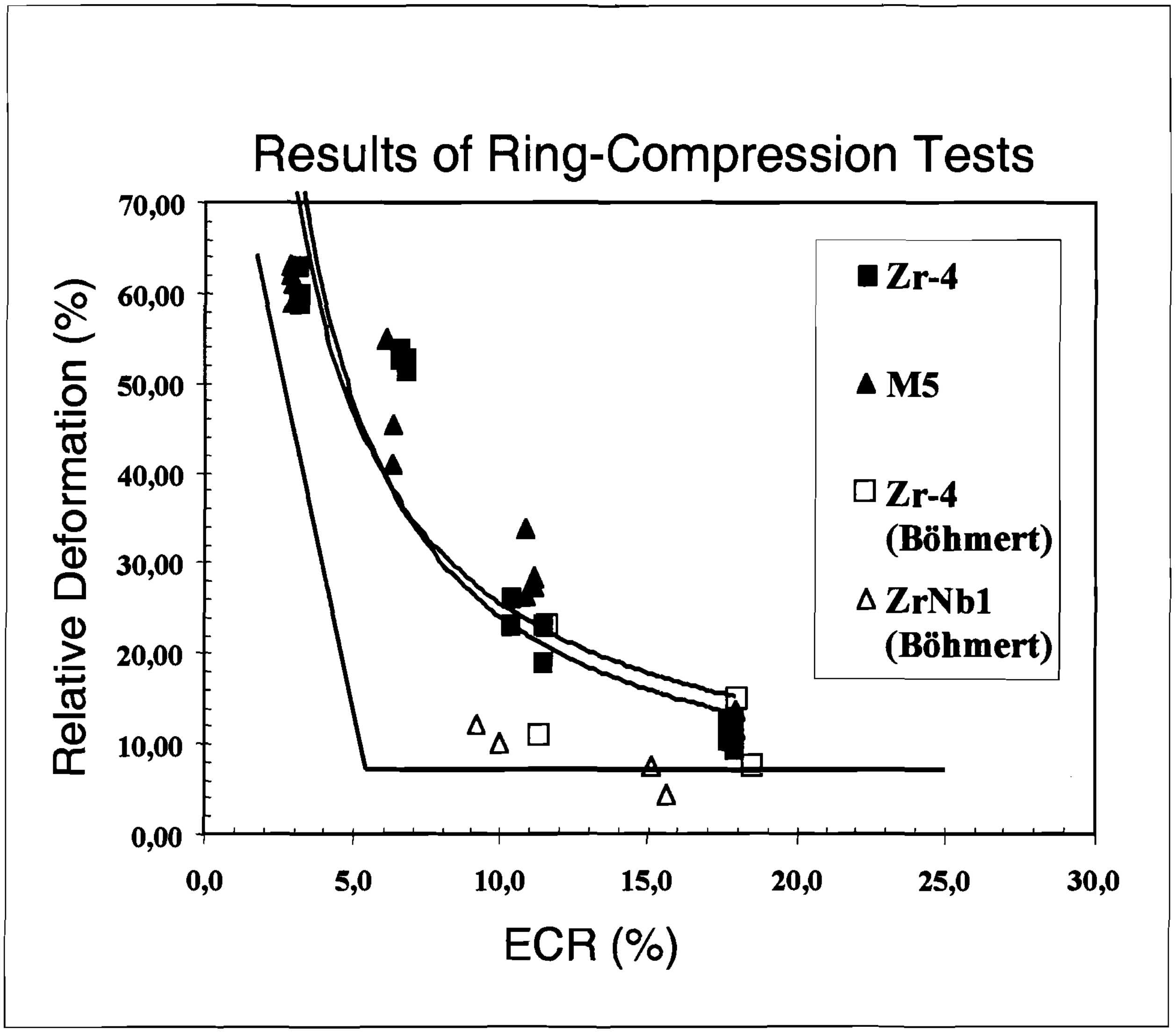
Post-Quench Mechanical Test Ring Compression Test Results



- M5[TM] behaves slightly better than Zr-4

FRAMATOME ANP

Comparison With Böhmert's Results at 1100°C



FRAMATOME ANP

Conclusions
Post-Quench Mechanical Tests

- M5™ Tested in the Böhmert range with results different than E110
  - Order of magnitude less hydrogen uptake
  - Completely different oxide morphology

- M5™ Performed better than or similar to Zr-4
  - No delamination
  - Similar bend test results
  - Slightly better impact test results
  - Slightly better than Zr-4 and much better than E110 in ring compression tests

- Böhmert's conclusions regarding Zr-1Nb alloy performance may be valid for Russian alloy E110 tested in 1992, but are not valid for M5™
  - Significantly different composition and processing parameters

Summary

- M5™ in-reactor operating performance is superior to Zr4

- M5™ LOCA and post-LOCA oxidation rates are equal to or slower than Zr4

- M5™ LOCA and post-LOCA mechanical performance is equivalent to Zr4

- M5™ LOCA and post-LOCA performance is acceptable and is equal to or better than Zr4 in events of equal duration

- M5™ LOCA and post-LOCA mechanical performance is superior to the Zr-1%Nb alloy tested by Böhmert

MME

22, 2001
1

Ductility Testing of Zircaloy-4 and ZIRLO™ Cladding After High Temperature Oxidation in Steam

Westinghouse Electric Company

Advisory Committee on Reactor Safeguards
Reactor Fuels Subcommittee

April 4, 2001
Rockville, MD

Zircaloy Ductility after High Temperature Oxidation in Steam

- Ductility measurements on Zircaloy oxidized in high temperature steam were used to establish cladding embrittlement criteria of 10 CFR 50.46
  - Peak Cladding Temperature no greater than 2200F
  - Equivalent Cladding Reacted (ECR) no greater than 17%
- Testing consisted of quench tests from temperature and ring compression tests
  - Ring Compression Tests conducted on Zircaloy-4
  - Quench Tests of Zircaloy-2 and Zircaloy-4
- The purpose of the criteria is to ensure the cladding would remain sufficiently intact to assure an easily coolable geometry

BNFL

Information Supplied for ZIRLO™ Licensing

- Testing of ZIRLO™ was performed to obtain data on the following areas
  - Material mechanical properties, density, thermal expansion, thermal conductivity, specific heat, phase changes,high temperature creep, high temperature oxidation, and rod burst characteristics.
- Other than phase change characteristics, the properties are essentially equivalent to those of Zircaloy-4
- It was argued that because of the close similarity to Zircaloy-4, the17% ECR criterion continued to apply
- The NRC agreed that the 17% criterion for Zircaloy also applied to ZIRLO™ and 10 CFR 50.46 was amended to state that the acceptance criteria applied to ZIRLO™

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March 22, 2001
Slide #4

Results of Tests on Alloy E110 Oxidized in High Temperature Steam (Bohmert, Kerntechnik 57)

- ECR to cause complete embrittlement is about 1/3 the value for Zircaloy-4
- A number of physical differences between the oxide layers of E110 and Zircaloy-4 were observed
  - E110 displayed a heterogeneous appearance of the oxide scale
  - E110 formed two oxide layers that were frequently separated by cracks
  - Multi-layer oxide layers tend to flake
  - Zircaloy-4 always had a glossy black firmly adherent single layer relatively free from mechanical failures
  - E110 showed low hydrogen uptake only if firmly adherent crackless oxide layers were formed
- High temperature steam oxidation tests of ZIRLO™ and Zircaloy-4 produce similar dark adherent oxide layers

BNFL

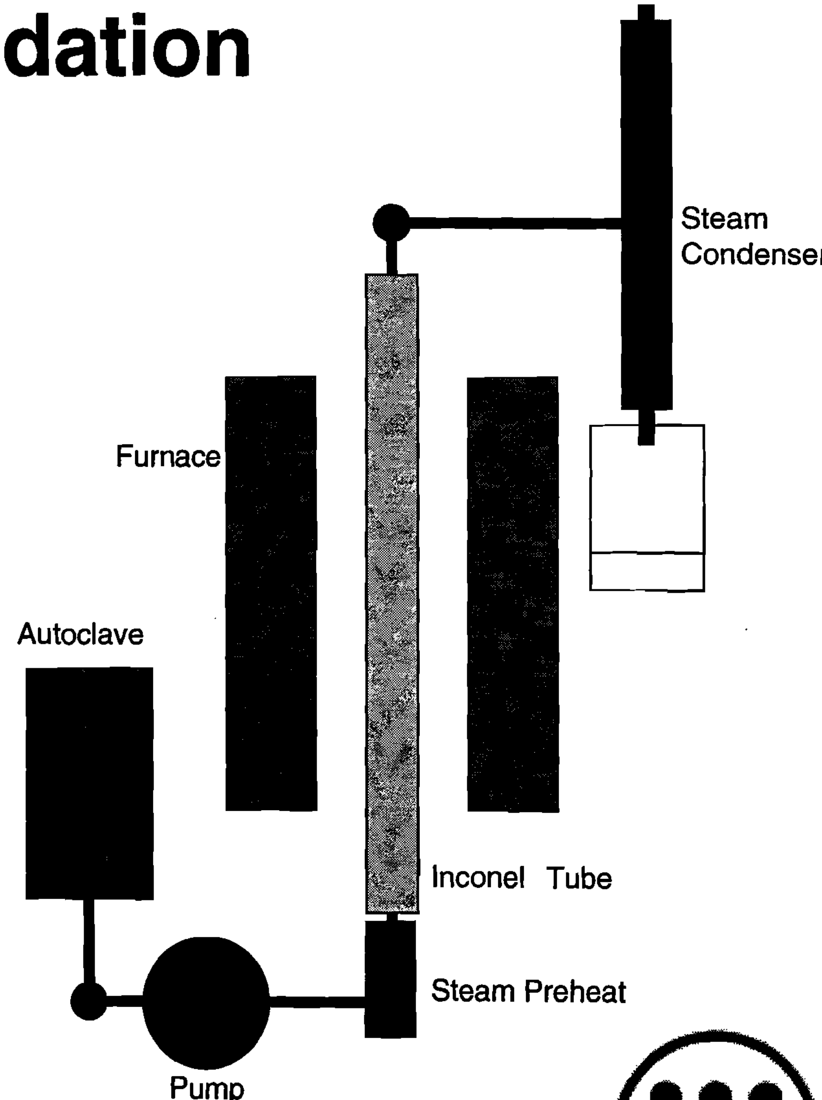
ZIRLO™ and E110 Are Not Equivalent

- Both alloys contain 1% niobium
- ZIRLO™ also contains
  - Sn
  - O
  - Fe
- Sn and oxygen are alpha phase stabilizers and raise the transition temperature relative to Zr-Nb binary alloys.
- There are significant differences in the oxide layer structure reported for the E110 alloy and those observed for either ZIRLO™ or Zircaloy-4

BNFL

Steam Oxidation

- Clam shell resistance furnace.
- Specimens placed in Inconel tube.
- Deaerated water from autoclave pumped into Inconel tube.
- Exit steam condensed by water cooling jacket.



March 22, 2001
Slide #7

Specimen Evaluations (<u>In Progress</u>)

- Specimen Evaluations
  - Oxide Layer Characteristics
  - Ring compression tests
    - Assess cladding ductility. (17% criteria was set)
    - Room temperature and 275°F.
    - Test performed similar to Hobson & Rittenhouse (ORNL Report 4758) and Böhmert.
  - Optical metallography
    - Oxide thickness, α-stabilized layer, transformed-β layer.
    - Microhardness to assess oxygen penetration.
  - Hydrogen and oxygen concentrations

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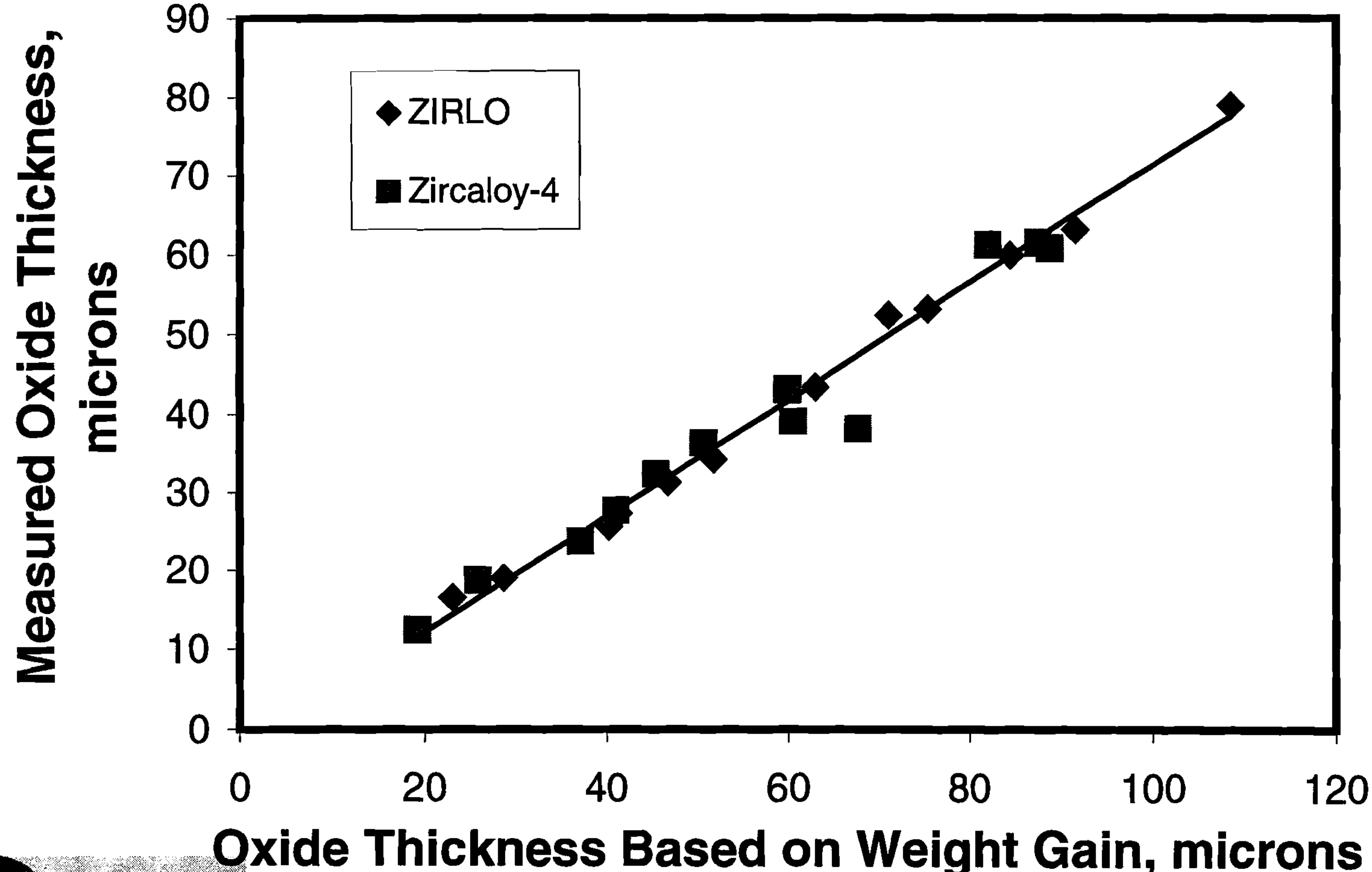
Slide #8

Measured Oxide Thickness vs. Oxide Thickness Based on Weight Gain



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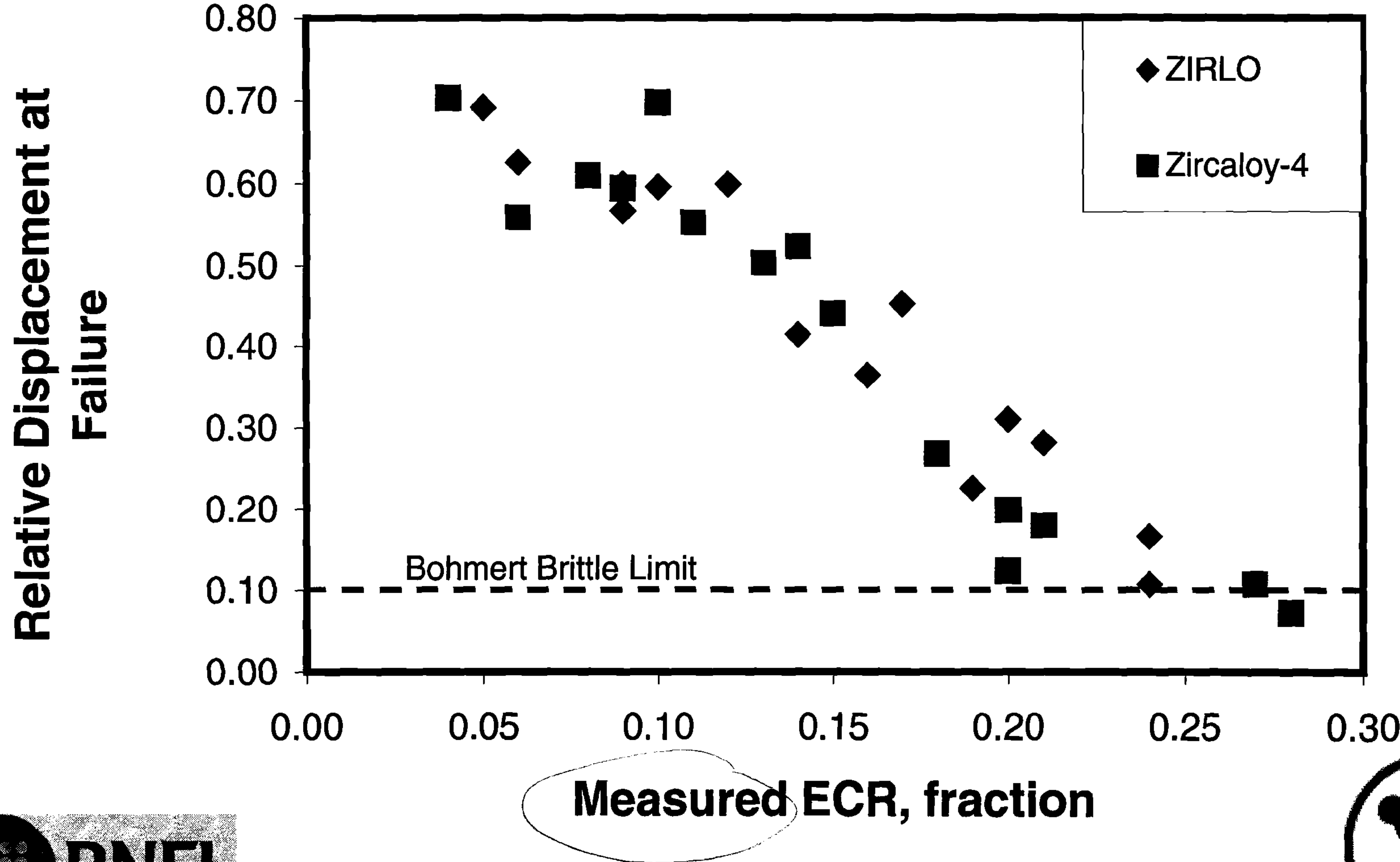
Slide #9

Relative Displacement at Failure vs Measured ECR at a Temperature of 275F (PRELIMINARY)



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Comparisons of ZIRLO™ and Zircaloy-4

- Both oxide layers were dark, adherent, and with no laminations
- Both have similar fractions of oxygen in the oxide layer and in the metal
- Ring compression tests show similar values of displacement at failure versus the measured Equivalent Cladding Reacted
- ZIRLO™ and Zircaloy-4 exhibit similar behavior

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Slide #11

Plan for Project Completion

- Perform remaining sample preparation
- Complete all planned tests
- Document and Review the results
- Meet with the NRC to discuss the results (May)

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MOX RESEARCH

Presented to the
ACRS Reactor Fuel Subcommittee
April 4, 2001

by

Richard Lee
Office of Nuclear Regulatory Research
(301) 415-6795

MIXED-OXIDE FUEL

ISSUE

- Utilization of weapons-grade mixed oxide fuel (MOX) in specific U.S. Pressurized Water Reactors (PWRs)

BACKGROUND

- U.S. Department of Energy issued Record of Decisions (1/14/97 and 1/4/00) to pursue a hybrid approach to safely and securely dispose of up to 50 metric tons of surplus plutonium from the U.S.

- The hybrid approach allows for the immobilization of approximately 17 metric tons of surplus plutonium and the use of 33 metric tons in MOX fuel.

- Savannah River Site has been selected for weapons-grade MOX fuel fabrication.

- Weapons-grade MOX are to be used in selected U.S. PWR commercial reactors (McGuire and Catawba).

MIXED-OXIDE FUEL

RES ACTIVITIES:

- NRC/RES is to provide technical support in licensing review of weapons-grade MOX use in PWRs

- Technical support: Improvement to Analysis Codes and Assessment of Environmental Impact of MOX fuel use   up to 40% in core

  - Neutronics: develop models for MOX, benchmark against critical experiments, computational benchmarks, and plant data

  - Fuel: revised model for MOX, assessment of fuel behavior under normal and abnormal conditions

  - Source Terms: validate model(s) against relevant experimental data, and perform consequence analysis

MIXED-OXIDE FUEL

STATUS:

- Conduct Phenomena Identification and Ranking Tables (PIRTs) for MOX http://www.nrc.gov/RES/PIRT/)

  - PIRT for LOCA and reactivity accident completed
  - PIRT for source term is being initiated and expects to complete by this year

- Neutronics:

  PARCS code development at the Purdue University
  - initiated in November 2000
  - implement and assessment of multi-group, P1 and P3 for X-sections representations
  - collaboration with France - Saclay, comparison of CRONOS vs. PARCS
  - development of a "theoretical" benchmark for reactivity transient for MOX under discussion with OECD/NEA

- Neutronics: (continued)

  Brookhaven National Laboratory

  - independent assessment of PARCS
  - provide feedback to code developer
  - assist in assembling benchmark/assessment problems for PARCS analysis of MOX cores
  - assist NRR in review of technical issues related to MOX licensing as needed (e.g. MOX fuel qualification program)

  Oak Ridge National Laboratory

  - Initiated the development of the NEWT lattice physics code

- Fuel:

  Pacific Northwest National Laboratory

  - initiated modifications of fuel codes for MOX analysis
  - assess code against MOX fuel behavior (e.g., Halden) completed
  - * also CABRI –

- Source Terms:

  - Initiated effort to obtain relevant experimental data (e.g., VERCORS, France; VEGA, Japan) for the assessment of fission products release models for MOX fuel

  - Additional experimental data may be available from the IPSN MAGRAGUE program at Cadarache, France

- Assist in licensing review of technical issues as they arise

RECENT OPERATIONAL ISSUES AND EXPERIENCE WITH HIGH BURNUP FUEL

Margaret S. Chatterton
ACRS Reactor Fuels Subcommittee Meeting
April 4, 2001

Outline of Presentation

- Burnup Extension Activities
- Lead Test Assembly Guidelines
- Recent Fuel Issues
- Current Fuel reviews

Basic Approach for Burnup Extension

- NRC Working with Industry to Develop a Strategy and a Plan
- Industry is Developing the Plan and Guidelines
- Industry Will Do the Testing and Develop the Criteria
- Objective is to Endorse Industry Approach in a Regulatory Guide

Burnup Extension Guidelines

- Address Current Licensing Requirements including LOCA, ATWS and RIA
- Be Risk Informed
- Emphasis on Lead Test Assemblies
- Fuel Performance Monitoring Program

Recent Fuel Issues

- Oxidation Higher than Predicted
- Axial Offset Anomalies
- Fuel Failures Due to High Fuel Duty
- Adverse Effects of Water Chemistry
- High Crud Buildup
- Accelerated Growth of Rods and Assemblies

Current Fuel Reviews

- Duplex Cladding — Siemens
  - Extensive Testing and Use in Europe up to 62 GWd/t
  - Beginning Review
- Zirlo for CE Plants
  - Reason for Request
  - Timetable for Review
  - Issues to be Examined

MMG

MOX FUEL SAFETY: A NEED FOR RESEARCH

Edwin S. Lyman, PhD

Nuclear Control Institute

Presentation to the ACRS Reactor Fuels Subcommittee

April 4, 2001

OVERVIEW

- MOX Program Concerns
- MOX Source Term Impact on Severe Accident Consequences and Risk
- MOX Impact on Transients
  - Overcooling and PTS
  - Reactivity Insertion Accidents
- MOX Fuel Qualification Issues

MOX PROGRAM CONCERNS

- No real choice of mission reactors
- Timetable dictated by international agreement and not by safety requirements
- DOE budget cuts will increase pressure for abbreviated MOX safety review
- Heavy reliance on proprietary foreign data
- Cancellation of immobilization track will increase burden on MOX program

MOX SOURCE TERM FOR SEVERE ACCIDENTS

- Compared to LEU core, DCS MOX core (40% MOX core fraction, Am removal) at EOC contains approximately
  - 2 times more Pu-239, Am-241, Cm-242
  - 10% less Pu-238
  - 45% more Ru-106 (important for PTS events, spent fuel pool accidents?)
- DOE EIS inventory calculations flawed

MOX IMPACT ON SEVERE ACCIDENT CONSEQUENCES

- Increased TRU source term in DCS core is important for severe accidents with early containment failure (ECF):
  - 25% increase in latent cancer fatalities (LCFs)
  - 4% increase in prompt fatalities (PFs) (E.Lyman, *Science and Global Security*, forthcoming)
  - Both LCFs and PFs increase by about 30% for high-Ru release fraction

MOX IMPACT ON RISK

- Assuming all initiator frequencies remain the same, average LCF population risk (< 1 mi) also increases by 25% for DCS core
- Risk of MOX use can be assessed using RG 1.174 methodology by defining

$$\Delta LERF_{eff} \equiv LERF \times \Delta R/R$$

- Also useful for evaluation of extended power uprates

RISK IMPACT OF MOX IN ICE CONDENSER PLANTS

- Ice condensers "substantially more sensitive to early containment failure" than other PWRs (NUREG/CR-6427, April 2000)
- Precisely the class of accidents in which additional MOX source term impact is felt
- McGuire IPE LERF (int+ext): 4.7×10^{-6}
  - $\Delta LERF_{eff} = 1.2 \times 10^{-6}$
  - $>$ RG 1.174 1×10^{-6} threshold

RISK IMPACT OF MOX IN ICE CONDENSER PLANTS (cont.)

- Estimate does not take into account Sandia finding that McGuire IPE underestimates ECF frequency by a factor of 7
- If taken into account, McGuire IPE LERF would exceed 1×10^{-5}: no LERF increase greater than 10^{-7} allowed (RG 1.174)
- MOX risk increase may be unacceptable
- Implications for extended power uprates

MOX IMPACT ON TRANSIENTS: OVERCOOLING EVENTS AND PTS

- PTS screening criteria for LEU cores may not be appropriate for MOX cores:
  - reduced decay heat leads to more rapid RCS temperature decrease
  - greater actinide and Ru inventory implies air oxidation source term is more severe
- faster embrittlement from greater fast flux
  - Duke Power not planning to consider MOX use in license renewal TLAAs

MOX IMPACT ON TRANSIENTS: REACTIVITY INSERTION

- Increased vulnerability of MOX fuel to RIAs is a concern (Cabri REP Na-7 test)
- Key consideration is fuel homogeneity and size distribution of Pu agglomerates:
  - Westinghouse (1994) recommended to DOE that "adherence to limits on Pu agglomerates in the range of 10 to 15 μm" be required
  - Yet DCS appears to be proposing a relaxation of the French specification!

PU PARTICLE DISTRIBUTIONS

- Cogema MIMAS Pu particle distribution:
  - mean size 20-40 μm
  - max. 2% of clusters > 100 μm
  - max. size about 140 μm
- DCS specification:
  - mean size < 50 μm
  - max. 5% of clusters > 100 μm
  - max. size 400 μm

MOX QUALIFICATION ISSUES

- Schedule for fuel qualification and licensing is very aggressive:
  - Oct. 2003: commencement of LTA irradiation
  - Oct. 2006: discharge of twice-burnt LTAs
  - Oct. 2007: first MOX reload batch
- Where will the LTAs be made?
  - if in Europe, may not be representative
  - if at the U.S. MOX plant, will cause delay

QUALIFICATION ISSUES (cont.)

- NRC ability to fully resolve MOX fuel safety issues is in jeopardy:
  - Time for post-irradiation LTA characterization and testing is insufficient
  - may force over-reliance on proprietary foreign data without confirmation --- Framatome/M5 experience should give pause
  - DOE uncooperative --- has rejected RES request for access to spent LTA rods

CONCLUSIONS

- Timetable and staff resources for MOX safety issue resolution should be based on NRC and not DOE needs
- Cancellation of the immobilization track will increase pressure on NRC not to be “obstructionist” in MOX licensing
- Current path for MOX fuel approval is not likely to engender public confidence