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

November 13, 2002

MEMORANDUM TO: ACRS Members

FROM: Med El-Zeftawy, Senior ACRS Staff Engineer

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON REACTOR FUELS, OCTOBER 9, 2002 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued November 5, 2002, 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: J. Larkins
S. Bahadur
R. Savio
H. Larson
S. Duraiswamy
ACRS Staff and Fellows

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

November 5, 2002

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

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

SUBJECT: WORKING COPY OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON REACTOR FUELS, OCTOBER 9, 2002, ROCKVILLE-MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. Copies are being provided to each ACRS Member who attended the meeting for information and/or review.

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cc: ACRS Members
J. Larkins
S. Bahadur
ACRS Staff and Fellows

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FROM: Dr. Dana Powers, Chairman
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I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on October 9, 2002, are an accurate record of the proceedings for that meeting.

Dana A. Powers 9/Nov/2002

Dana A. Powers Date
Subcommittee Chairman

ISSUED; November 5, 2002
CERTIFIED BY: POWERS, 11/9/02

CERTIFIED

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
REACTOR FUELS SUBCOMMITTEE
MEETING MINUTES-- OCTOBER 9, 2002
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Reactor Fuels met on October 9, 2002, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2B3. The purpose of this meeting was to discuss and review the NRC's high burnup fuel research activities as well as the application of regulatory criteria for reactivity insertion accidents. The Subcommittee also heard a presentation by the Electric Power Research Institute (EPRI) representatives regarding a topical report on reactivity initiated accidents.

The Subcommittee received no written comments from members of the public regarding the meeting. The entire meeting was open to public attendance. Dr. Med El-Zeftawy was the cognizant ACRS staff engineer and the designated federal official for this meeting. The meeting convened at 8:30 a.m. and adjourned at 5:40 p.m.

ATTENDEES

ACRS Members

D. Powers, Subcommittee Chairman
M. Bonaca, Member
F. P. Ford, Member
G. Leitch, Member
S. Rosen, Member
M. El-Zeftawy, Staff Engineer

NRC Staff

R. Meyer, RES
J. Wermiel, NRR
M. Kowal, NRR
V. Klein, NRR
D. Tang, NMSS
S. Basu, RES
F. Eltawila, RES
J. Rosenthal, RES
H. Scott, RES
R. Caruso, NRR
J. Voglewede, RES
S. Wu, NRR
P. Wen, NRR
U. Shoop, NRR
R. Lee, RES

Industry

R. Yang, EPRI
R. Montgomery, EPRI
J. Rashid, ANATECH
L. Ott, ORNL
W. Slagle, W

A complete list of attendees is in the ACRS Office File and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the office copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. Dana Powers, Subcommittee Chairman, convened the meeting at 8:30 a.m. Dr. Powers stated that the purpose of the meeting is to discuss with the NRC's Office of Nuclear Regulatory Research (RES) representatives their confirmatory research program on high burnup fuel as well as research they do to support safety regulation of dry cask storage of spent fuel including high burnup fuel. Dr. Powers stated that the discussion will focus primarily on the behavior of high burnup fuel under design-basis accident conditions. The subcommittee will also discuss with representatives of the NRC's Office of Nuclear Reactor Regulation (NRR) regarding their plan to develop the regulatory criteria and review of the Electric Power Research Institute (EPRI) topical report on the response of high burnup fuel to reactivity insertion events. EPRI's representatives will brief the Subcommittee on such topical report. The Subcommittee will discuss with all representatives the development of fuel failure criteria and coolability criteria for high burnup fuel exposed to reactivity transients.

Dr. Powers indicated that there is economic and societal incentives to use nuclear fuel to higher levels of burnups. Burnup levels now approved exceed the data bases underlying the models that are used to predict fuel behavior under upset and design-basis accident conditions. French and Japanese tests of high burnup fuel have shown that cladding failure and even fuel dispersal can occur during reactivity insertions at energy levels below the current allowable criteria.

NRR Presentation

Ms. Undine Shoop outlined the reactivity insertion accidents (RIA) criteria history. She stated that the Commission in its memorandum of July 15, 1997, directed the staff to assess the adequacy of regulatory guidelines and licensing criteria for high burnup fuel. The original criteria of 280 cal/gm was developed in Regulatory Guide 1.77 (May 1974). On July 6, 1998, the NRC's Program Plan for high burnup fuel has been issued. In such program:

- The industry will have to provide the criteria , data base, and models for burnup greater than 62 GWd/t
- The industry will have to perform the research necessary to develop the data base to support extended burnup ranges greater than 62 GWd/t
- RES will confirm criteria for burnup up to 62 GWd/t.

EPRI developed a robust fuels program that includes an objective of providing industry wide criteria, data, analysis and methodology to achieve industry burnup extension greater than 62 Gwd/t. In addition, EPRI has recently developed RIA topical report. Such report is the first industry submittal to develop the criteria to support industry high burnup extension.

EPRI is proposing two criteria consistent with current R.G. 1.77 criteria. These criteria are:

- Criteria for long term cooling following an accident
- Criteria for radiological release following a cladding failure.

Currently, NRR is preparing a preliminary review plan to focus its resources and provide detailed review and identify all the elements needed to complete the review. These elements include data verification, fuel rod failure threshold, core coolability limit, strain energy density theory and model, FALCON code, fuel dispersal, uncertainty and conservatism, limitations of the criteria, safety evaluation conditions of acceptance, and revision of associated RG and SRPs. NRR plans to complete the final review Plan by December 31, 2002.

EPRI Presentation

Ms. Rosa Yang stated that the goal to achieve higher fuel burnup levels has produced considerable interest in the transient response of high burnup fuel. The data base on transient fuel behavior is limited at burnup levels beyond 40 GWd/t and is based on older fuel rod designs. Several experimental programs are currently underway to generate data on the behavior of high burnup fuel under transient conditions such as LOCA and RIAs. These programs include the RIA simulation experiments performed at the CABRI facility in France and the NSRR in Japan. The purpose of these experiments is to provide data that can be used to develop safety criteria for extended burnup levels and to validate analytical codes.

The CABRI REP Na-1 results raised concerns that the existing licensing criteria may be inappropriate. As a result, EPRI and the industry conducted an extensive review and assessment of the behavior of high burnup fuel under RIA conditions. The objective of this program was to conduct a detailed analysis of the data obtained from RIA simulation experiments and to evaluate the applicability of the data to commercial LWR fuel behavior during a rod ejection accident (REA) and rod drop accident (RDA). The industry assessment included a review of the fuel segments used in the tests, the test procedures, in-pile instrumentation measurements, post-test examination results, and a detailed analytical evaluation of several key RIA simulation. Major conclusions from the industry are:

- The RIA simulation test conditions are not representative of those expected during a postulated in-reactor REA or RDA. The pulses were considerably more rapid and narrower than anticipated LWR power pulses.

- The conditions under which the test rods were base-irradiated produced cladding corrosion and hydriding features that were not representatives of commercial LWRs.

- Analytical evaluations and separate effects data are required to understand the key mechanisms operative in RIA simulation.

- Loss of cladding ductility due to localized hydrides was the major cause of failure for high burnup test rods. The causes are more related to adverse hydride content and distributions resulting from outer surface cladding oxidation anomalies such as spallation. The primary effect of burnup is to increase PCMI by gap closure effects such as solid fission product swelling.

Mr. R. Montgomery, EPRI, stated that the approach used by EPRI to develop the revised licensing criteria combines three major elements:

- Establish the transient behavior of intermediate and high burnup fuel rods using well characterized RIA simulation tests. The RIA simulation experiments in the previous evaluation, and the more recent tests on rods with burnup levels ranging from 45-65

GWd/t in the CABRI, NSRR, and IGR/BIGR reactors, provide a data base of in-pile observations.

- Define the cladding mechanical properties using data from separate effects tests. The data base of Zircaloy cladding mechanical properties furnishes insights into the influence of irradiation damage, hydrogen content and distribution, and temperature .

- Benchmark the RIA analysis capabilities in the transient fuel behavior code FALCON using experimental data from the data base of RIA simulation tests. FALCON calculates the thermal and mechanical performance of a single fuel rod during power conditions.

Combined with the NRC Phenomena Identification and Ranking Tables (PIRTs) review conducted on the PWR REA, the industry believes this will establish a strong technical basis to develop a revised licensing criteria for RIAs. However, the development of additional RIA tests will slow for the next several years as the CABRI facility is modified to include a water loop.

EPRI developed a topical report that summarizes the technical bases for the revised fuel rod failure threshold criteria and core coolability criteria used in the licensing analysis of a PWR or BWR hot-zero power (HZP) and hot-full power (HFP), respectively. The primary RIA events considered in the topical report are the REA for PWRs and RDA for BWRs. The topical report is being developed to support the industry's effort to extend fuel rod average burnup levels beyond the current limit of 62 GWd/t.

For the fuel rod failure threshold, the radial average peak fuel enthalpy required to cause cladding failure by PCMI was calculated by FALCON as a function of rod average burnup using a cladding ductility model based on mechanical properties tests from irradiated low tin Zr-4 cladding material. The critical strain density (CSD) data formed the basis of the cladding ductility model. To account for the accumulation of outer surface corrosion, a conservative oxidation rate was used that bounded a large data base of low tin Zr-4 oxide thickness measurements. A maximum cladding outer surface oxide thickness of 100 microns was imposed and the impact of oxide layer spalling on the cladding mechanical properties was not considered.

For the core coolability criteria, recent RIA simulation experiments on rods with burnup levels greater than 30 GWd/t demonstrate a potential for dispersal of finely fragmented non-molten fuel material following cladding failure. In these cases, the tests were run with a power pulse width less than 10 milliseconds. The consequences from fuel-coolant interaction are much less for dispersal of finely fragmented non-molten material than for the dispersal of molten material. The failure threshold bounds the data for tests on non-spalled Zr-4 rods. This represents a conservative lower bound for modern, low corrosion cladding.

EPRI claims no experiments on high burnup fuel, that have been conducted, resulted in molten fuel dispersal. Consequently, an analytical evaluation was used to determine the maximum radial average peak fuel enthalpy that causes the local pellet temperature to reach the melting temperature. EPRI is concluding that no fuel dispersal leading to fuel-coolant interaction will occur following cladding failure for typical PWR REA power pulse widths, and in the unlikely event of fuel dispersal, the dispersed material will be below the UO_2 melting temperature. Therefore, there is a large margin between burnup at peak power location during REA and rod peak burnup used in UO_2 incipient melting calculation.

RES Presentation

Dr. R. Meyer, RES, stated that currently in the U.S. there are two types of regulatory criteria have been used in safety analyses to address RIAs. One is a limit of 280 cal/g fuel on peak fuel-rod enthalpy. The other regulatory criterion consists of several threshold values that are used to indicate cladding failure-that is, the occurrence of a breach in the cladding that would allow fission products to escape. This criterion is used in calculating radiological releases for comparison with other limits. For PWRs, a critical heat flux value related to departure from nucleate boiling (DNB) is used. For BWRs, a similar value is used for high-power accidents, but for low-power and zero-power accidents, a peak fuel-rod enthalpy of 170 cal/g fuel is used.

In the 1970s when the regulatory criteria and related analytical methods were being established, high burnup was thought to occur above 40 GWd/t (average for the peak rod). Data out to that burnup had been included in data bases for criteria, codes, and regulatory decisions, and it was believed that some extrapolation in burnup could be made. Fuel burnup in licensed reactors up to 62 GWd/t (average for the peak rod) were permitted. By the mid 1980s, however, unique changes in pellet microstructure had been observed from both vendor and international data at higher burnup along with increases in the rate of cladding corrosion. It thus became clear that other phenomena were occurring at high burnups and that continued extrapolation of transient data from the low burnup data base was not appropriate.

In late 1993, a test (REP Na-1) was run in the CABRI test reactor in France that produced cladding failure at a peak fuel-rod enthalpy of about 30 cal/g. Fragmented fuel particles were dispersed from the fuel rod in this test, and enhanced fission-product release was observed. In 1994, a similar test in NSRR in Japan produced cladding failure at a peak fuel-rod enthalpy of about 60 cal/g. These values were so far below the 280 cal/g coolability limit and the 170 cal/g fuel failure criterion that the NRC adopted in Regulatory Guide 1.77.

Currently the NRC has embarked on efforts to address two important needs. The first need is to identify the research to be done by the NRC and industry with respect to high burnup fuel issues. The original list of issues included cladding integrity and fuel design limits; control rod insertion problems; criteria and analysis for reactivity accidents; criteria and analysis for LOCA; criteria analysis for BWR power oscillations (ATWS); fuel rod and neutronic computer codes; source term and core melt progression; transportation and dry storage; and high enrichments (larger than 5%). The second need is to develop a new criterion to replace the current 280 cal/g coolability limit and the cladding failure criterion of RG 1.77. RES is proposing a single criterion of 100 cal/g enthalpy increase for cladding failure (Broad-brush) with no oxide spalling is allowed.

Mr. H. H. Scott, RES, briefed the Subcommittee on relevant LOCA research. He stated that Argonne National Laboratory (ANL) is conducting research on high burnup BWR and PWR fuel to provide data for assessing the licensing criteria (10 CFR 50.46) for LOCA. LOCA-relevant research includes fuel and cladding characterization, cladding high-temperature steam oxidation kinetics studies, LOCA integral testing of fueled segments, post-quench ductility testing of LOCA integral specimens and post-quench ductility testing of Zircaloy and advanced alloy unirradiated tubing. The work completed on samples from Limerick BWR fuel rods (about 57 GWd/t) and PWR fuel rods (about 67 GWd/t) is reported.

Limerick cladding is Zr-lined Zircaloy-2. The in-reactor formed outer-surface oxide layer is approximately 10μm. Axial variation of layer thickness is minimal for test sample regions compared to the circumferential variation. The inner-surface oxide layer is approximately 10-15 μm. Oxygen and hydrogen contents are approximately 0.7 wt.%.

Cathcart-Pawel (CP) model has been used to plan the LOCA integral test times-at-temperature to achieve desired equivalent cladding reacted (ECR) values. The tests have the following sequential steps: stabilization of temperature, internal pressure and steam flow at 300°C, temperature ramping through ballooning and burst to 1204 °C, hold at 1204 °C in flowing steam, slow cooling, and initiation of water quench. Four-point bend tests will be used to determine overall specimen ductility. Ring compression tests will be used for local ductility determination. Some future work include determination of the composition of dark deposit on quartz tube (gamma scanning) and the determination of the maximum ECR.

Mr. S. Basu, RES, briefed the Subcommittee regarding creep testing of spent fuel rods in dry storage. He stated that because of the limited storage capacity in spent-fuel pools, some spent fuel assemblies have to be relocated into dry casks for interim storage until long-term geological repositories are available. Upon discharge from the reactor, the internal pressure in the spent fuel rod can exert a significant stress loading on the fuel cladding. At elevated temperatures, these tensile stresses can induce significant outward thermal creep of the cladding. The vacuum drying operation can elevate the cladding temperature to 400-500 °C for many hours.

The U.S. Department of Energy (DOE) procured a Castor-V/21 dry-storage cask for testing at the Idaho National Environmental and Engineering Laboratory(INEEL). The primary purpose of the tests was to benchmark computer codes. The cask was loaded with irradiated assemblies from the Surry Nuclear Station and then tested in a series of configurations using a variety of cover gases. Subsequently, the cask sat on the storage pad at the INEEL for approximately 15 years with the fuel in an essentially inert atmosphere. Under the sponsorship the NRC, DOE, and EPRI, twelve rods were retrieved from the cask for post-storage characterization. Cladding from two of the rods was prepared for thermal creep testing.

The objective of the thermal creep tests is to evaluate residual creep ductility of the Surry cladding after the dry-cask storage. A significant residual creep strain (greater than 1%) would suggest that the rods may be suitable for further storage in the cask and may survive creep during transportation, reconsolidation and final repository conditions. As the Surry rods are not the limiting case for less than 45 GWd/t, demonstration of residual creep life can be used to argue that higher burnup rods with thicker oxide layers, higher hydrogen content and higher storage temperatures would also have survived 20 years of dry cask storage without creep failure.

General Subcommittee comments

The ACRS Subcommittee believe that RES has a well-organized and leveraged program of confirmatory research on high burnup fuel issues. RES is nearing resolution of the issues. However, the members remain concerned that the time-temperature conditions used in the study of high burnup design basis LOCA may not reveal phenomena unique to high burnup fuel.

SUBCOMMITTEE ACTION

This matter will be discussed during the ACRS meeting on October 10-12, 2002. The Committee expects to write a letter on this matter.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee meeting agenda
2. Subcommittee Status Report
3. Agency Program Plan for High Burnup Fuel.
4. Memorandum from S. Collins to A. Thadani, dated January 31, 2002.
5. ACRS letter, dated March 14, 2002.
6. EDO Response, dated June 11, 2002.
7. EPRI Topical Report (DRAFT)

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co., Inc. (Court Reporters and Transcribers) 1323 Rhode Island Ave., N.W., Washington, DC 20005 (202) 234-4433.

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NRC Staff

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EPRI claims no experiments on high burnup fuel, that have been conducted, resulted in molten fuel dispersal. Consequently, an analytical evaluation was used to determine the maximum radial average peak fuel enthalpy that causes the local pellet temperature to reach the melting temperature. EPRI is concluding that no fuel dispersal leading to fuel-coolant interaction will occur following cladding failure for typical PWR REA power pulse widths, and in the unlikely event of fuel dispersal, the dispersed material will be below the UO_2 melting temperature. Therefore, there is a large margin between burnup at peak power location during REA and rod peak burnup used in UO_2 incipient melting calculation.

RES Presentation

Dr. R. Meyer, RES, stated that currently in the U.S. there are two types of regulatory criteria have been used in safety analyses to address RIAs. One is a limit of 280 cal/g fuel on peak fuel-rod enthalpy. The other regulatory criterion consists of several threshold values that are used to indicate cladding failure-that is, the occurrence of a breach in the cladding that would allow fission products to escape. This criterion is used in calculating radiological releases for comparison with other limits. For PWRs, a critical heat flux value related to departure from nucleate boiling (DNB) is used. For BWRs, a similar value is used for high-power accidents, but for low-power and zero-power accidents, a peak fuel-rod enthalpy of 170 cal/g fuel is used.

In the 1970s when the regulatory criteria and related analytical methods were being established, high burnup was thought to occur above 40 GWd/t (average for the peak rod). Data out to that burnup had been included in data bases for criteria, codes, and regulatory decisions, and it was believed that some extrapolation in burnup could be made. Fuel burnup in licensed reactors up to 62 GWd/t (average for the peak rod) were permitted. By the mid 1980s, however, unique changes in pellet microstructure had been observed from both vendor and international data at higher burnup along with increases in the rate of cladding corrosion. It thus became clear that other phenomena were occurring at high burnups and that continued extrapolation of transient data from the low burnup data base was not appropriate.

In late 1993, a test (REP Na-1) was run in the CABRI test reactor in France that produced cladding failure at a peak fuel-rod enthalpy of about 30 cal/g. Fragmented fuel particles were dispersed from the fuel rod in this test, and enhanced fission-product release was observed. In 1994, a similar test in NSRR in Japan produced cladding failure at a peak fuel-rod enthalpy of about 60 cal/g. These values were so far below the 280 cal/g coolability limit and the 170 cal/g fuel failure criterion that the NRC adopted in Regulatory Guide 1.77.

Currently the NRC has embarked on efforts to address two important needs. The first need is to identify the research to be done by the NRC and industry with respect to high burnup fuel issues. The original list of issues included cladding integrity and fuel design limits; control rod insertion problems; criteria and analysis for reactivity accidents; criteria and analysis for LOCA; criteria analysis for BWR power oscillations (ATWS); fuel rod and neutronic computer codes; source term and core melt progression; transportation and dry storage; and high enrichments (larger than 5%). The second need is to develop a new criterion to replace the current 280 cal/g coolability limit and the cladding failure criterion of RG 1.77. RES is proposing a single criterion of 100 cal/g enthalpy increase for cladding failure (Broad-brush) with no oxide spalling is allowed.

Mr. H. H. Scott, RES, briefed the Subcommittee on relevant LOCA research. He stated that Argonne National Laboratory (ANL) is conducting research on high burnup BWR and PWR fuel to provide data for assessing the licensing criteria (10 CFR 50.46) for LOCA. LOCA-relevant research includes fuel and cladding characterization, cladding high-temperature steam oxidation kinetics studies, LOCA integral testing of fueled segments, post-quench ductility testing of LOCA integral specimens and post-quench ductility testing of Zircaloy and advanced alloy unirradiated tubing. The work completed on samples from Limerick BWR fuel rods (about 57 GWd/t) and PWR fuel rods (about 67 GWd/t) is reported.

Limerick cladding is Zr-lined Zircaloy-2. The in-reactor formed outer-surface oxide layer is approximately 10μm. Axial variation of layer thickness is minimal for test sample regions compared to the circumferential variation. The inner-surface oxide layer is approximately 10-15 μm. Oxygen and hydrogen contents are approximately 0.7 wt.%.

Cathcart-Pawel (CP) model has been used to plan the LOCA integral test times-at-temperature to achieve desired equivalent cladding reacted (ECR) values. The tests have the following sequential steps: stabilization of temperature, internal pressure and steam flow at 300°C, temperature ramping through ballooning and burst to 1204 °C, hold at 1204 °C in flowing steam, slow cooling, and initiation of water quench. Four-point bend tests will be used to determine overall specimen ductility. Ring compression tests will be used for local ductility determination. Some future work include determination of the composition of dark deposit on quartz tube (gamma scanning) and the determination of the maximum ECR.

Mr. S. Basu, RES, briefed the Subcommittee regarding creep testing of spent fuel rods in dry storage. He stated that because of the limited storage capacity in spent-fuel pools, some spent fuel assemblies have to be relocated into dry casks for interim storage until long-term geological repositories are available. Upon discharge from the reactor, the internal pressure in the spent fuel rod can exert a significant stress loading on the fuel cladding. At elevated temperatures, these tensile stresses can induce significant outward thermal creep of the cladding. The vacuum drying operation can elevate the cladding temperature to 400-500 °C for many hours.

The U.S. Department of Energy (DOE) procured a Castor-V/21 dry-storage cask for testing at the Idaho National Environmental and Engineering Laboratory(INEEL). The primary purpose of the tests was to benchmark computer codes. The cask was loaded with irradiated assemblies from the Surry Nuclear Station and then tested in a series of configurations using a variety of cover gases. Subsequently, the cask sat on the storage pad at the INEEL for approximately 15 years with the fuel in an essentially inert atmosphere. Under the sponsorship the NRC, DOE, and EPRI, twelve rods were retrieved from the cask for post-storage characterization. Cladding from two of the rods was prepared for thermal creep testing.

The objective of the thermal creep tests is to evaluate residual creep ductility of the Surry cladding after the dry-cask storage. A significant residual creep strain (greater than 1%) would suggest that the rods may be suitable for further storage in the cask and may survive creep during transportation, reconsolidation and final repository conditions. As the Surry rods are not the limiting case for less than 45 GWd/t, demonstration of residual creep life can be used to argue that higher burnup rods with thicker oxide layers, higher hydrogen content and higher storage temperatures would also have survived 20 years of dry cask storage without creep failure.

General Subcommittee comments

The ACRS Subcommittee believe that RES has a well-organized and leveraged program of confirmatory research on high burnup fuel issues. RES is nearing resolution of the issues. However, the members remain concerned that the time-temperature conditions used in the study of high burnup design basis LOCA may not reveal phenomena unique to high burnup fuel.

SUBCOMMITTEE ACTION

This matter will be discussed during the ACRS meeting on October 10-12, 2002. The Committee expects to write a letter on this matter.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Subcommittee meeting agenda
2. Subcommittee Status Report
3. Agency Program Plan for High Burnup Fuel.
4. Memorandum from S. Collins to A. Thadani, dated January 31, 2002.
5. ACRS letter, dated March 14, 2002.
6. EDO Response, dated June 11, 2002.
7. EPRI Topical Report (DRAFT)

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co., Inc. (Court Reporters and Transcribers) 1323 Rhode Island Ave., N.W., Washington, DC 20005 (202) 234-4433.

concentration below 4 percent is adequate in satisfying NRC Regulatory Guide 1.7. Accordingly, reactor operation with the TPBARs will not be a significant contributor to the post-LOCA hydrogen inventory, and will not have a significant impact on the total hydrogen concentration within the containment when compared to the values associated with the non-TPBAR core. The maximum containment hydrogen concentration can be maintained at less than the lower flammability limit of 4.0-volume-percent, with one recombiner train started at a 3-percent hydrogen concentration approximately 24 hours after an LBLOCA.

Summary

The Commission has completed its evaluation of the proposed action. The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential non-radiological impacts, the proposed action does not have a potential to affect historic sites. It does not affect non-radiological plant effluents and has no other environmental impact. Therefore, there are no significant non-radiological environmental impacts associated with the proposed action.

Environmental Impacts of the Alternatives to the Proposed Action

As an alternative to the proposed action, the staff considered denial of the proposed action (*i.e.*, the "no-action" alternative). Denial of the application would result in no significant change in current environmental impacts. However, because there are no significant environmental impacts associated with this action, and because PL 106–65 directs that DOE produce tritium at WBN or SQN, this is not considered a viable option.

Alternative Use of Resources

DOE evaluated alternatives to the proposed action, including completing construction of one or both of the Bellefonte Nuclear Plant Units and construction of an accelerator facility at the Savannah River site and concluded that the proposed action has the least environmental impact of the options considered. The NRC has no reason to disagree with DOE's decision.

Agencies and Persons Consulted

On September 16, 2002, the staff consulted with the Tennessee State official, Elizabeth Flannagan of the Tennessee Bureau of Radiological Health, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated September 21, 2001, as supplemented by letters dated June 11, July 19, August 9, August 30, September 5, and September 12, 2002. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, *http://www.nrc.gov/reading-rm/adams.html*. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1–800–397–4209 or 301–415–4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 17th day of September 2002.

For the Nuclear Regulatory Commission.

Ronald W. Hernan,

Senior Project Manager, Section 2, Project Directorate II, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 02–24152 Filed 9–20–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on October 9, 2002, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, October 9, 2002—1:30 p.m. until the conclusion of business.

The Subcommittee will discuss proposed ACRS activities and related matters. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Persons desiring to make oral statements should notify the Designated Federal Official named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301/415–7364) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: September 17, 2002.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 02–24148 Filed 9–20–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the Subcommittee on Reactor Fuels; Notice of Meeting

The ACRS Subcommittee on Reactor Fuels will hold a meeting on October 9, 2002, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Wednesday, October 9, 2002–8:30 a.m. until the conclusion of business.*

The Subcommittee will review the high burnup fuel research activities as well as the application of regulatory criteria for reactivity insertion accidents. The Subcommittee will also discuss the staff's review of the Electric Power Research Institute topical report on reactivity insertion accidents. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written standards will be accepted and made available to the Committee. Persons desiring to make oral statements should notify the Designated Federal Official named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff and its consultants, Electric Power Research Institute, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore can be obtained by contacting the Designated Federal Official, Dr. Medhat M. El-Zeftawy (Telephone 301/415–6889) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: September 17, 2002.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 02–24149 Filed 9–20–02; 8:45 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

Meeting of the Subcommittee on Plant License Renewal; Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on October 9, 2002, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows: *Tuesday, October 8, 2002–8:30 a.m. until the conclusion of business.*

The Subcommittee will review the Duke Energy Corporation's license renewal application for McGuire Nuclear Station Units 1 and 2, and Catawba Nuclear Station Units 1 and 2, and the associated Safety Evaluation Report with open items. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Persons desiring to make oral statements should notify the Designated Federal Official named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, Duke Energy Corporation, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore can be obtained by contacting the Designated Federal Official, Mr. Timothy Kobetz (telephone 301/415–8716) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact one of the above named individuals at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: September 17, 2002.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 02–24150 Filed 9–20–02; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Issuer Delisting; Notice of Application To Withdraw From Listing and Registration; (Canadian 88 Energy Corporation, Common Stock, No Par Value) From the American Stock Exchange LLC File No. 1–14752

September 17, 2002.

Canadian 88 Energy Corporation, a Canada corporation ("Issuer"), has filed an application with the Securities and Exchange Commission ("Commission"), pursuant to section 12(d) of the Securities and Exchange Act of 1934 ("Act")[1] and Rule 12d2–2(d) thereunder,[2] to withdraw its Common Stock, no par value ("Security"), from listing and registration on the American Stock Exchange LLC ("Amex" or "Exchange").

The Issuer stated in its application that it has met the requirements of Amex Rule 18 by complying with all applicable laws in effect in Canada, in which it is incorporated, and with the Amex's rule governing an issuer's voluntary withdrawal of a security from listing and registration.

The Board of Directors ("Board") of the Issuer unanimously approved a resolution on September 5, 2002 to withdraw the Issuer's Security from listing on the Amex. In making the decision to withdraw its Security from the Amex, the Board states that Issuer sought to reduce its general and administrative costs. The Issuer states that it will continue listing on the Toronto Stock Exchange. The Issuer's application relates solely to the withdrawal of the Security from listing on the Amex and registration under

[1] 15 U.S.C. 78*l*(d).

[2] 17 CFR 240.12d2–2(d).

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON REACTOR FUELS

OCTOBER 9, 2002
Date

NRC STAFF PLEASE SIGN BELOW
PLEASE PRINT

NAME	NRC ORGANIZATION
Ralph Meyer	RES
Jared Wermiel	NRR/DSSA/SRXB
Mark Kowal	NRR/DSSA/SRXB
Veronica Klein	NRR/DSSA/SRXB
DAVID TANG	NMSS/SFPO
SUDHAMAY BASU	RES
FAROUK ELTAWILA	RES/DSARE
Jack Rosenthal	RES/DSARE/SMSAB
Harold Scott	NRC-RES
RALPH CARUSO	NRR/SRXB
JOHN VOGLEWEDE	RES/DSARE
Shih-Liang Wu	NRR
PETER WEN	NRR
UNDINE SHOOP	NRR/DSSA/SRXB
RICHARD LEE	RES/DSARE/SMSAB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON REACTOR FUELS

OCTOBER 9, 2002
Date

PLEASE PRINT

ATTENDEES PLEASE SIGN BELOW

NAME	AFFILIATION
ROSA YANG	EPRI
Joe Rashid	ANATECH
William Slagle	Westinghouse
Robert Montgomery	EPRI / ANATECH
L. J. Ott	ORNL
~~Ch h L[illegible]~~	~~NRC~~

EPRI Topical Report on Reactivity Initiated Accidents

Undine Shoop
Office of Nuclear Reactor Regulation
October 9, 2002

RIA Criteria History

- RG 1.77 – May 1974
  - Original Criteria of 280 cal/gm
- NRR User Need Request – October 4, 1993
  - Evaluate Fuel Failure Thresholds for Normal Operation and RIA
- Commission Memorandum – July 15, 1997
  - Adequacy Assessment of Regulatory Guidelines and Licensing Criteria for High Burnup Fuel

RIA Criteria History - Continued

- Research Information Letter No. 174 – March 3, 1997
  - Proposed Changes to the RIA Criteria
- Agency Program Plan for High Burnup Fuel – July 6, 1998
  - Industry will have to provide the Criteria, Data base, and Models for Burnup > 62 GWD/MTU
  - Industry will have to perform the research necessary to develop the data base to support extended burnup ranges > 62 GWD/MTU
  - RES will confirm criteria for burnup < 62 GWD/MTU

Industry Response

- EPRI Robust Fuels Program
  - Included an objective of developing industry wide criteria, data, analysis and methodology to achieve industry burnup extension > 62 GWD/MTU
  - EPRI RIA topical report is the first industry submittal to develop the criteria to support industry high burnup extension

EPRI Criteria

- Two criteria approach proposed consistent with current RG 1.77 criteria
  - Criteria for long term cooling following an accident
  - Criteria for radiological release following a cladding failure

EPRI Topical Report on Reactivity Initiated Accidents – Part 2

Undine Shoop

Office of Nuclear Reactor Regulation

October 9, 2002

NRC Preliminary Review Plan Purpose

- To focus resources appropriately to provide a detailed review and identify all the elements needed to complete the review

NRC Preliminary Review Plan Elements

- Data Verification
  - Correct application in the methodology
  - Correct application in a manner consistent with the methods used to generate it
  - Statistically sound combination of the data sets
- SED/CSED Theory and Model
  - Investigation and verification of the equivalence of SED/CSED model to Rice's J/Jc formulation
  - FRAPTRAN independent verification
- Fuel Rod Failure Threshold
  - Validation of this application
  - Review of applicability to current and future proposed fuel types
- Core Coolability Limit
  - Application verification

NRC Preliminary Review Plan Elements – Cont.

- FALCON Code
  - Review of the code
- Fuel Dispersal
  - Review data for applicability of the phenomena to the proposed safety limit
- Uncertainty and Conservatism
  - Data uncertainty verification
  - Conservatism confirmation
- Limitations of the Criteria
  - Review data for limits of applicability which would create limitations of the methodology application
- Safety Evaluation Conditions of Acceptance
- Revision of associated RG and SRPs

Preliminary RES Assistance Needed

- Data Verification
- SED/CSED Theory and Model
- Fuel Dispersal

Future Activities

- Final Review Plan – December 31, 2002

EPRI

Bases for RIA Fuel Failure and Core Coolability Acceptance Criteria

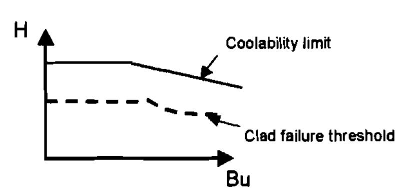
Robert Montgomery
Nicolas Waeckel
Rosa Yang

ACRS Reactor Fuels Subcommittee Meeting
NRC Offices
Washington, D.C.
October 9, 2002

EPRI

Presentation Outline

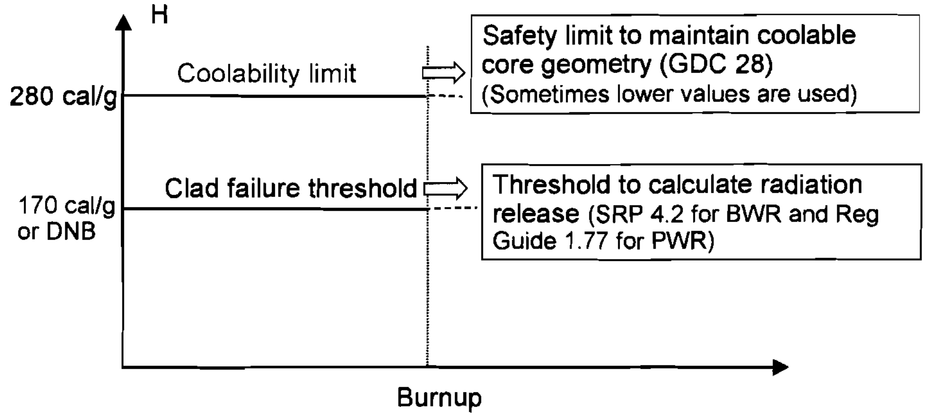
- Regulatory basis
- Database of RIA-simulation tests
  - integral test characteristics and test conditions
- Fuel Rod Failure
  - Clad failure mechanisms at low and high burnup
  - Clad failure model for PCMI
  - Revised fuel rod failure threshold
- Core Coolability
  - Core coolability issues
  - Revised core coolability limit
- Summary



EPRI

Regulatory background

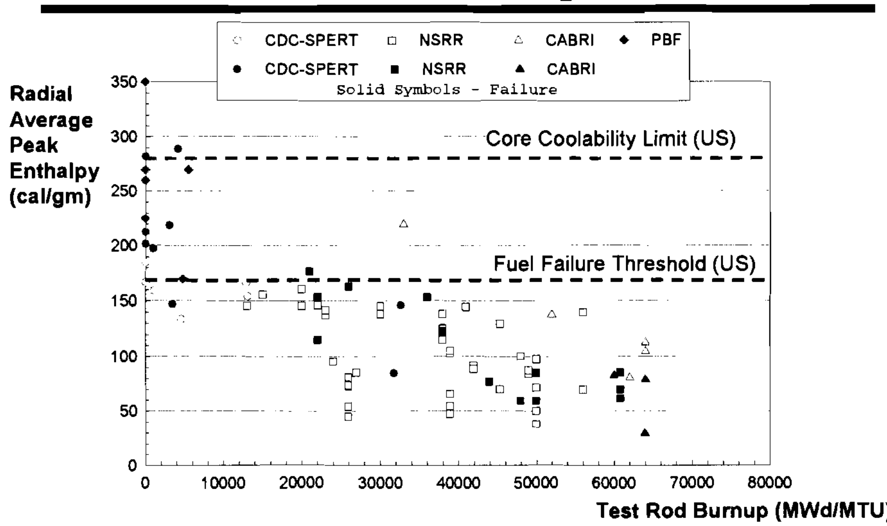
- Separate clad failure threshold and coolability safety limit



Robust Fuel Program

EPRI

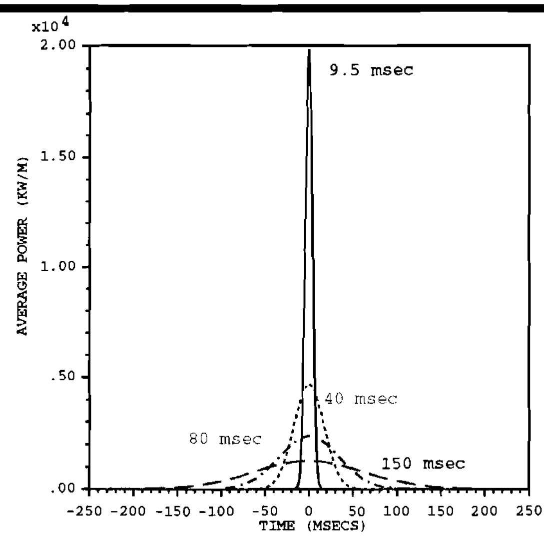
Database of RIA-Simulation Tests on Irradiated UO_2 Fuel



Robust Fuel Program

EPRI

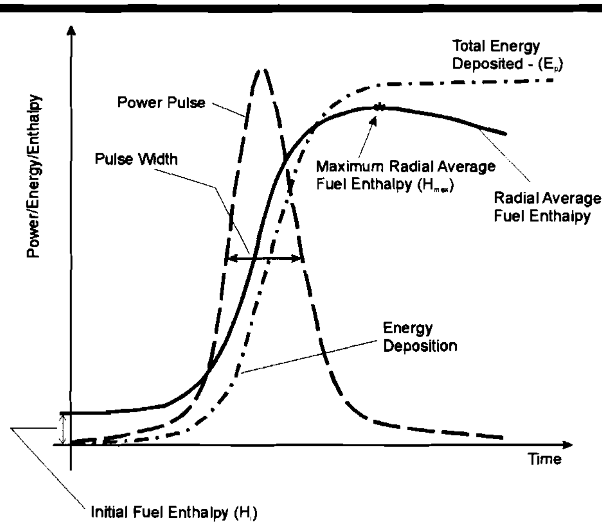
Comparison of RIA Power Pulse Shapes



Robust Fuel Program

EPRI

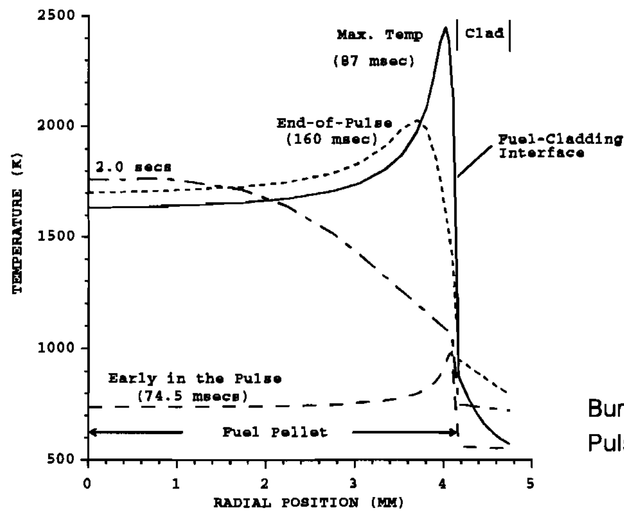
RIA Power Pulse Characteristics



Robust Fuel Program

EPRI

Fuel Rod Temperature Profiles



Burnup = 65 GWd/tU
Pulse Width 9.5 ms

EPRI

Test Conditions vs. LWR

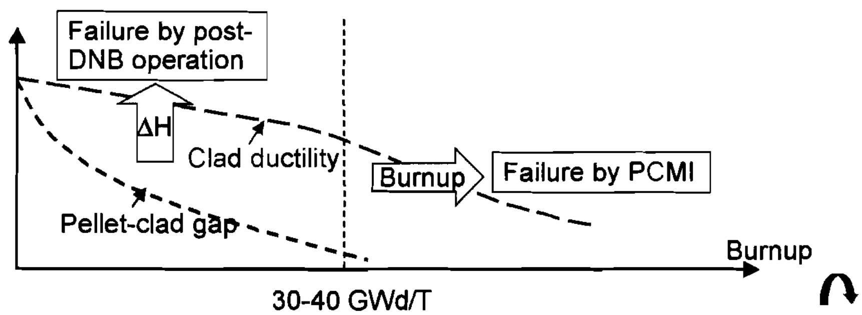
	SPERT-CDC	NSRR	CABRI	LWR
Number of Tests	> 15	> 50	12	
Coolant Conditions				
Type	Stagnant Water	Stagnant Water	Flowing Sodium	Flowing Water
Temp (°C)	25	25	280	280 - BWR 290 - PWR
Pressure (atm)	1	1	3	70 - BWR 150 - PWR
Pulse Characteristics				
Full-Width Half Max. (msec)	13 to 31	4.5 to 6.6	10 natural 30-80 pseudo	25 to 90
Deposited Energies (cal/gm)	160 to 350	20 to 200	100 to 200	TBD

⇒ Need analytical tools to assess tests results and compare to LWR conditions

EPRI

Clad failure mechanisms

- Based on over 100 RIA-simulation tests, the clad failure mechanisms are:
  - Low Burnup: high temperature failure caused by post-DNB operation (clad oxidation / embrittlement or clad ballooning)
  - High Burnup: Pellet Clad Mechanical Interaction (PCMI) combined with loss of clad ductility



Robust Fuel Program

EPRI

Clad failure mechanisms at high burnup

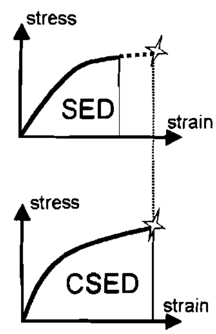
- Clad failure mechanism is PCMI resulting from fuel thermal expansion and fuel matrix fission gas swelling
  - ⇨ Cladding **ductility** is the key determining factor
  - ⇨ Conclusion of the PWR RIA PIRT Report (NUREG/CR-6742)
- Fuel rod failure depends mainly on cladding ductility NOT on burnup
  - Corrosion/hydriding and fuel duty define clad residual ductility
  - Spalled rods have significantly less ductility than non-spalled rods
    - » CABRI database shows NO failure up to 64 GWd/TU for non-spalled rods

Robust Fuel Program

EPRI

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
  - Addresses the effects of strain rate, temperature and stress biaxiality
- Critical SED is a measure of cladding failure potential or cladding residual ductility
  - CSED is determined from mechanical property tests
  - depends mainly on H level, temperature and materials
- Cladding failure occurs when SED reaches the CSED for a given clad material



Robust Fuel Program

EPRI

Extensive Database of Cladding Mechanical Properties

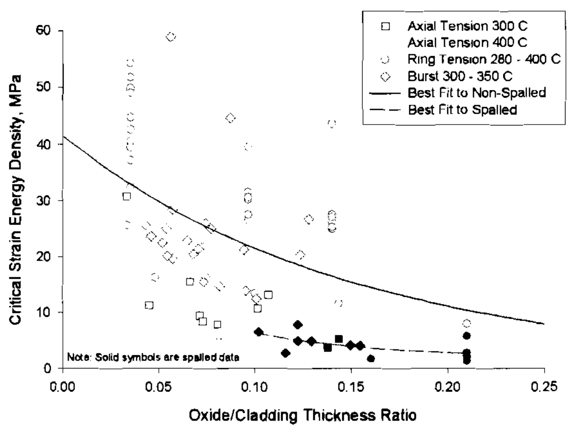
Program	Fuel Type	Max. Bu (GWd/tU)	Max. Fast Fluence (n/cm^2)	Range of Oxide Thickness (μm)	Temperature Range (K)	Strain Rate (/sec)
ESEERCO Hot Cell Program on Zion Rods						
Burst	15x15	49	$9.4x10^{21}$	15 - 25	588	$2x10^{-5}$
ABBCE-DOE Hot Cell Program on Fort Calhoun Rods						
Burst	14x14	53	$8x10^{21}$	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	$4x10^{-3}$
Ring Tension				24 - 115‡	573	$4x10^{-3}$
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	$7x10^{-3}$
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.

Robust Fuel Program

EPRI

Cladding CSED Database

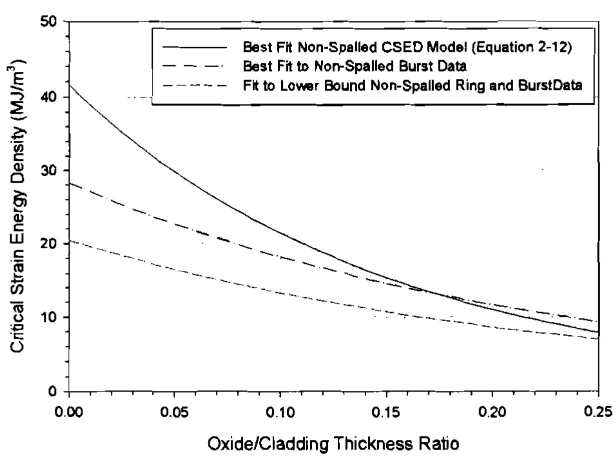


- Scatter is more related to test conditions and specimen design artifacts rather than to material variability
- Improved test designs will reduce the scatter
- Use of best-fit curves is justified when compared with failed-unfailed RIA database

Robust Fuel Program

EPRI

Different Data Evaluation Methods

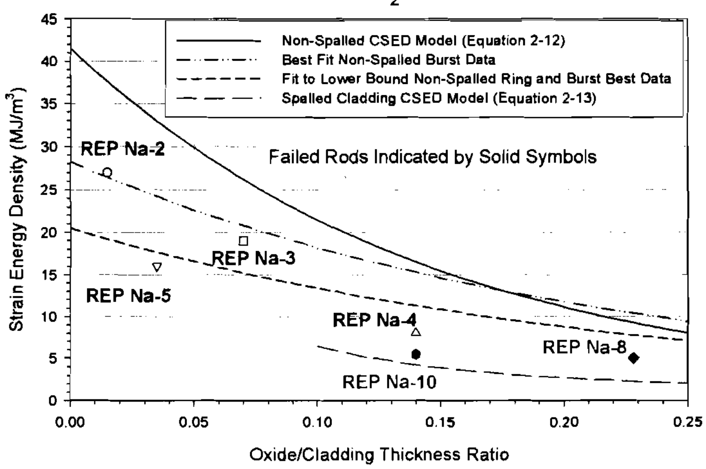


Robust Fuel Program

EPRI

Analysis of High Burnup RIA-Simulation Tests

CABRI REP Na Tests on UO_2 Rods in Sodium Coolant

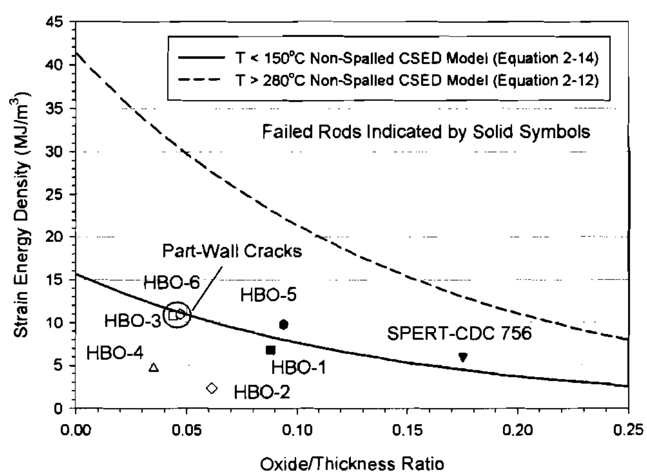


Robust Fuel Program

EPRI

Analysis of High Burnup RIA-Simulation Tests

NSRR Tests on UO_2 Rods in Ambient Water



Robust Fuel Program

EPRI

Development of Fuel Rod Failure Threshold

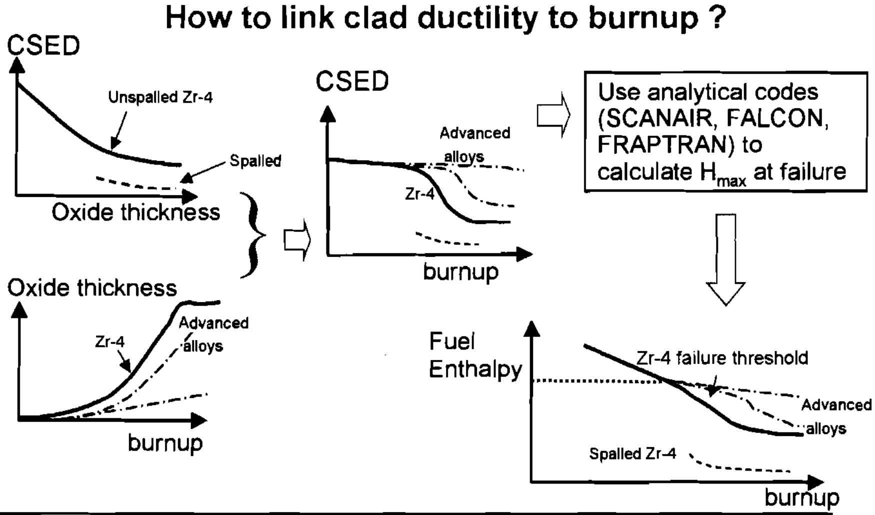
- Construct Fuel Rod Failure Threshold Consistent with Current Licensing Approach
  - Radial Average Fuel Enthalpy at Failure as a Function of Rod Average Burnup
  - Conservative Zircaloy-4 "Corrosion vs. Burnup" Correlation Used
    - » Relationship between cladding oxidation and rod average burnup

Robust Fuel Program

EPRI

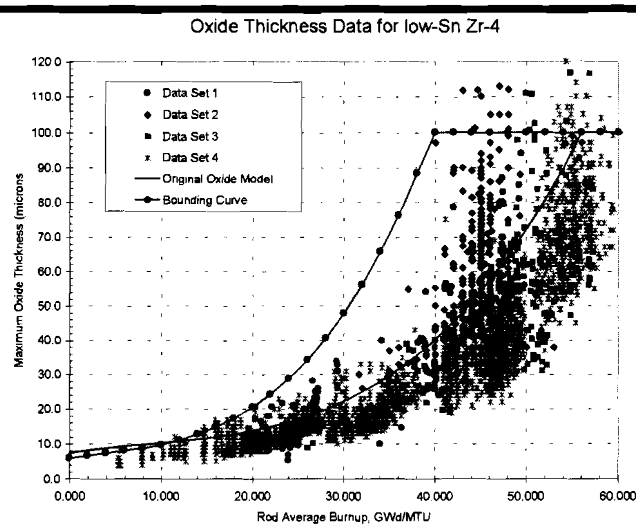
Approach to Develop Fuel Rod Failure Threshold

How to link clad ductility to burnup ?

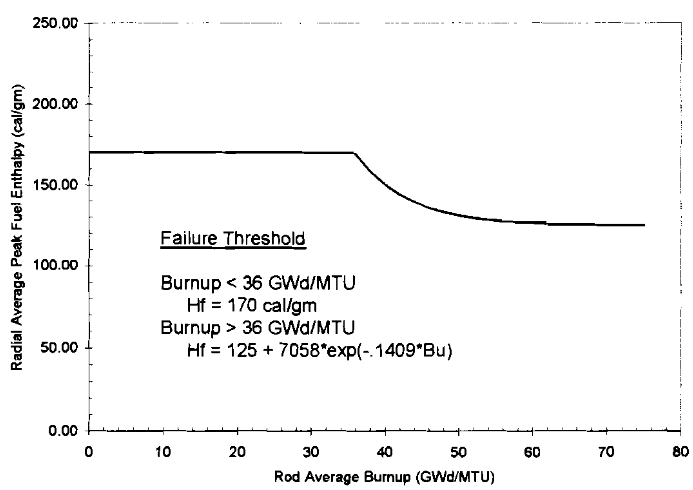


Robust Fuel Program

Maximum Oxide Thickness versus Burnup



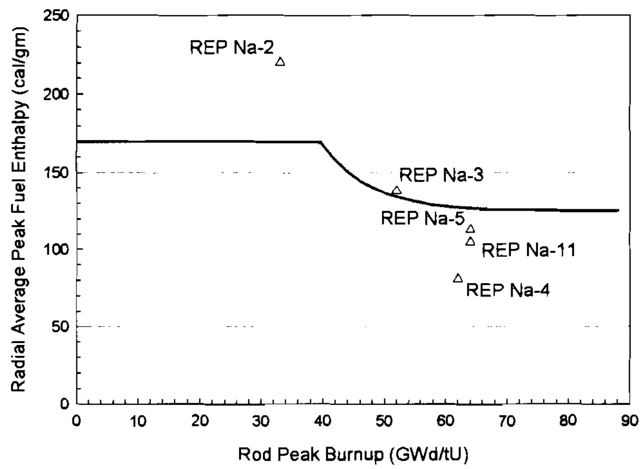
Revised Fuel Rod Failure Threshold



EPRI

Failure Threshold Bounds CABRI Test Data With Non-Spalled Oxide Layers

(CABRI Tests in Sodium Coolant - 280°C)



Note: Test Data Represent Rod Peak Burnup

EPRI

Fuel Rod Behavior Leading to Core Coolability Concerns

- Experimental Database
  - Past experiments in US and Japan focused on fuel enthalpy above 280 cal/gm
    - » Molten fuel dispersal kinetics
    - » Mechanical energy generation from fuel-coolant interaction
  - Recent experiments in France and Japan at fuel enthalpy levels below 220 cal/gm
    - » Some failures resulted in dispersal of a small amount of pellet material coming from the pellet periphery as finely fragmented solid particles
    - » Measurable mechanical energy generation

EPRI

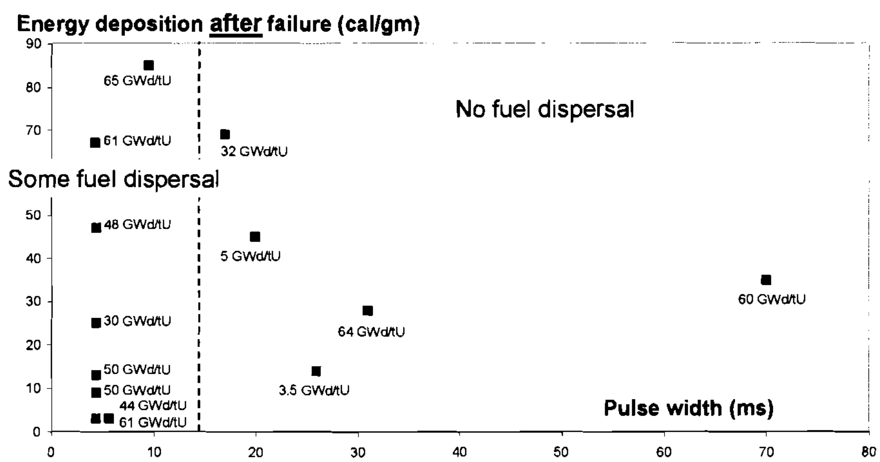
Current understanding of fuel dispersal and related core coolability issues

- Fuel particle dispersal during power pulse following cladding failure
  - Potential may increase above 40 GWd/T due to rim formation in fuel pellets
    - » Local peaking for burnup and fission density
  - Issues raised by fuel dispersal
    - » flow blockage and loss of rod geometry ?
    - » pressure pulse generation and threat on core geometry and pressure vessel integrity ?
- Data show that potential for fuel dispersal is a function of :
  - Energy deposition following cladding failure
  - Pulse width

Robust Fuel Program

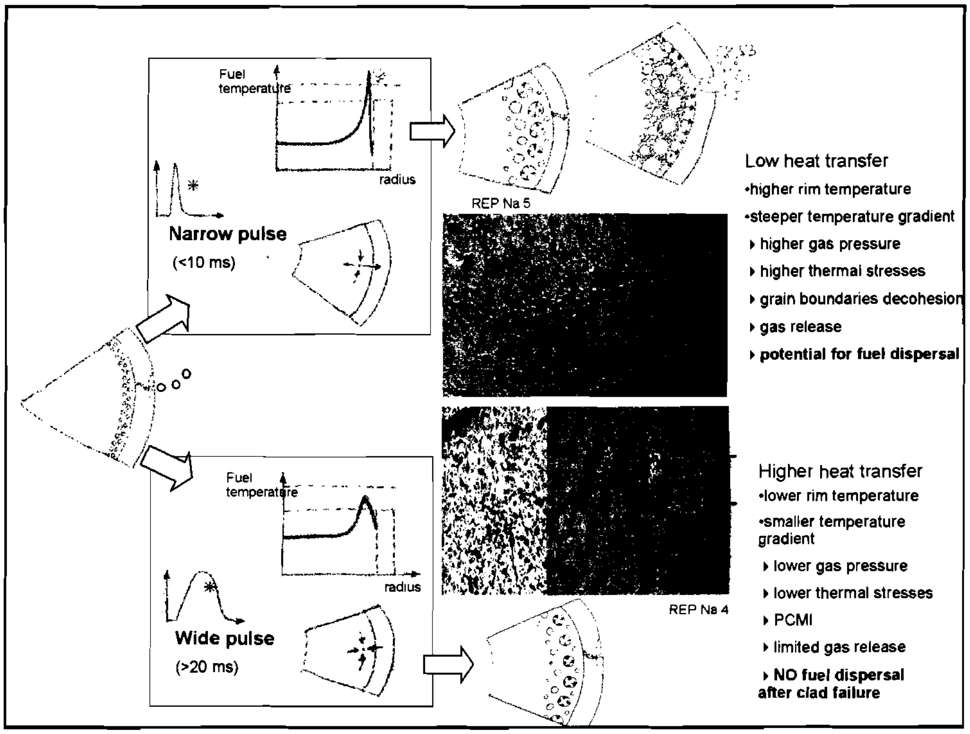
EPRI

Pulse Width Effect on Fuel Dispersal



Note: Fuel dispersal observed **only** below 10 ms

Robust Fuel Program



EPRI

Post-Failure Behavior of High Burnup Fuel

- No fuel dispersal is expected for prototypical pulse widths
- At high energy after failure, small amount of non-molten pellet material may be dispersed through failure opening but has low impact on:
  - Fuel rod geometry
    - » Experimental data (NSRR) show less then 10% of pellet material loss - mostly from rim region (1)
    - » Rod geometry is maintained in all cases (1) ➡
  - Fuel-coolant interaction (leading to pressure pulses)
    - » Tests exhibited low mechanical energy conversion (1)
      - temperature of dispersed material lower than UO_2 melting
      - involved limited amount of material (from rim region only)

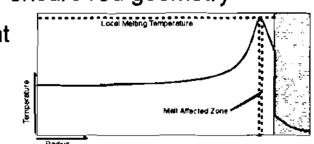
(1) T. Sugiyama and al. "Mechanical energy generation during high burnup fuel failure under RIA conditions". Journal of Nuclear Sciences and Technology, Vol 37, No. 10 October 2000

Robust Fuel Program

EPRI

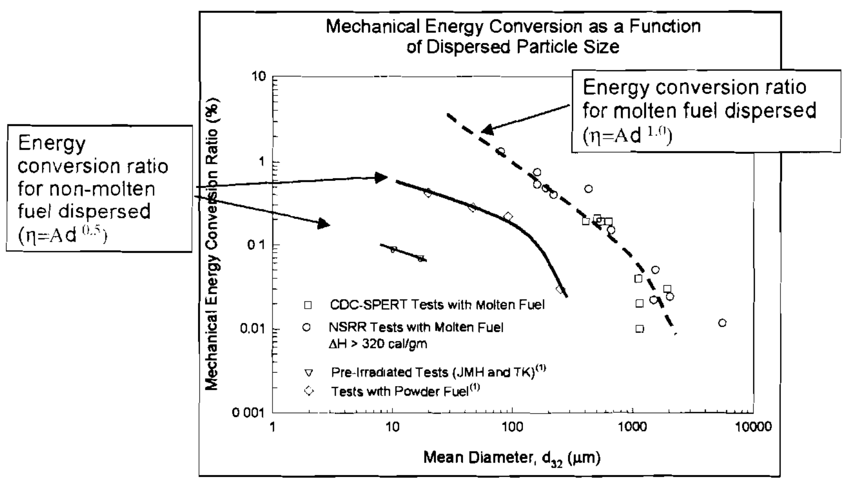
Basis for Coolability Limit

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



EPRI

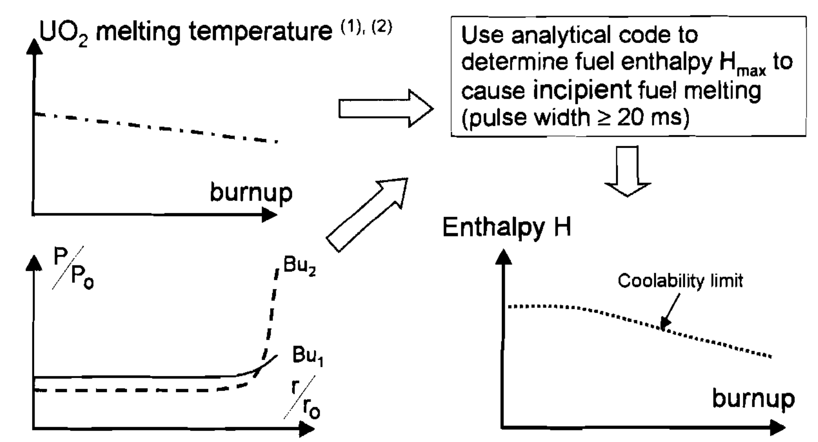
RIA Tests FCI Data



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

EPRI

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



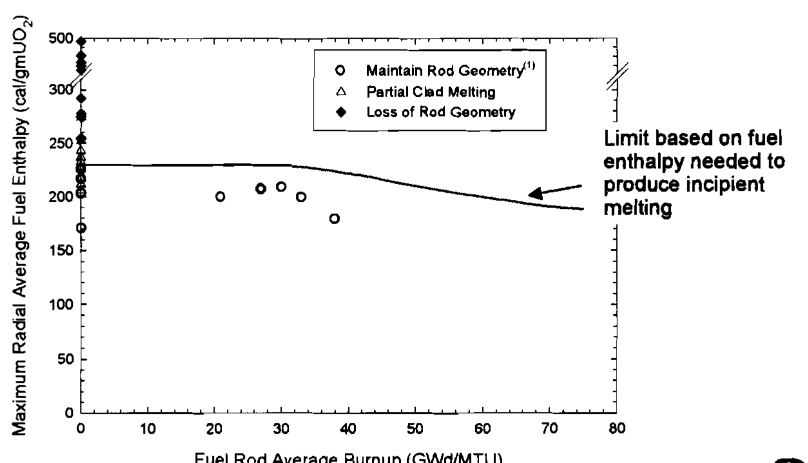
(1) Y. Philipponeau CEA technical Report LPCA n0 27

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

Robust Fuel Program

EPRI

Comparison to High Energy Tests

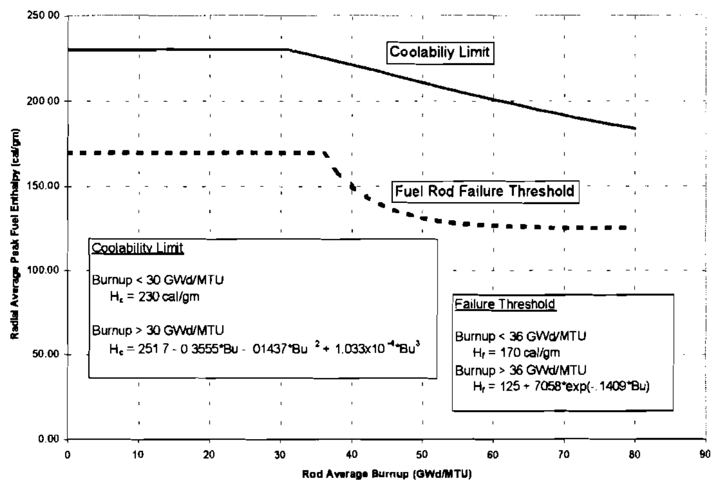


(1) T. Sugiyama and al. Journal of Nuclear Science and Technology, Vol 37, No 10, Oct 2000

Robust Fuel Program

EPRI

Revised RIA Acceptance Criteria



Robust Fuel Program

EPRI

Summary (1)

- Revised clad failure threshold and core coolability limit as a function of burnup
  - Incorporates key controlling parameters
    - » Corrosion/hydriding evolution with burnup
    - » Burnup impact on UO_2 melting
- Criteria are given in terms of radial average peak fuel enthalpy
  - Applicable to HZP RIA
  - Use directly in core reload designs
  - Consistent with current practice
- DNB limit remains an acceptable criterion for at-power REA

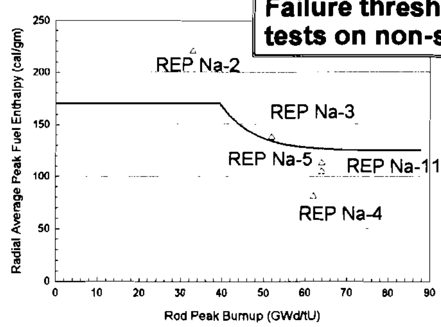
Robust Fuel Program

EPRI

Summary (2)

- Fuel Failure Threshold
  - Based on integral test results, mechanical property test data, and analytical approach
  - Represents a conservative lower bound for modern, low-corrosion cladding

Failure threshold bounds the data for tests on non-spalled Zr-4 rods

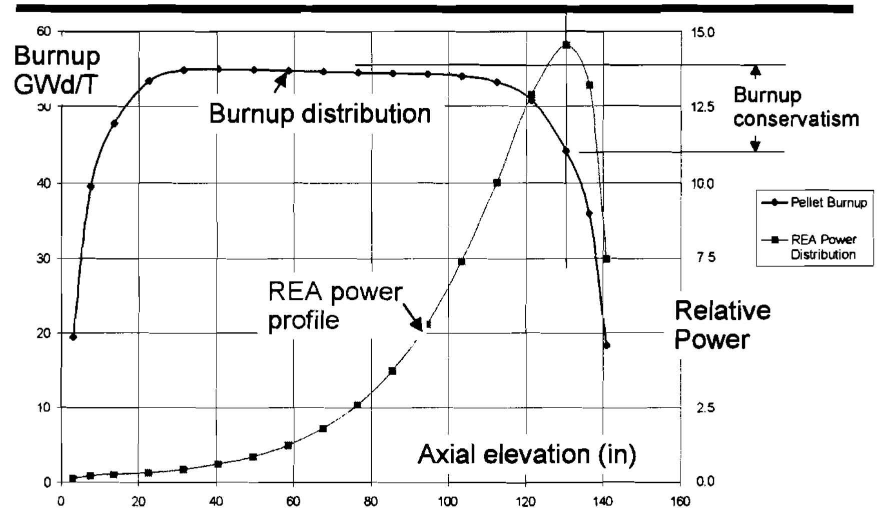


EPRI

Summary (3)

- Core Coolability Limit
  - No fuel dispersal expected under typical LWR conditions
  - However, fuel enthalpy limit established to minimize mechanical energy generation if fuel dispersal is assumed
    - » Limit peak fuel enthalpy to preclude incipient fuel melting
      - function of burnup
      - The limit is supported by data from both loss of rod geometry and mechanical energy release issues
    - » the limit is conservative
      - Small amount of fuel material involved (< 10%)
      - Large margin between burnup at peak power location during rod ejection and rod peak burnup used in UO_2 incipient melting calculation

Conservatism

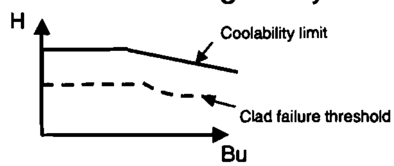


EPRI

Outline

- Industry effort for preparing the RIA (Reactivity Initiated Accident) Topical- Yang
  - Experimental and analytical effort
  - RepNa-1 is an outlier
  - CABRI Water Loop Project

- Bases for RIA Fuel Failure and Core Coolability Acceptance Criteria - Montgomery



EPRI

Lower RIA Limits For High Burnup Fuel ?

- CABRI RepNa-1 test (November, 1993) raised concerns about RIA fuel failure limits and fuel dispersal for high burnup fuel

Materials

- High burnup (64 GWD/T) Zr-4 cladding
- Oxide=80 μm with extensive spallation

Test Conditions

- Narrow (9.5 ms) pulse width
- Low pressure Na-loop

Test results

- Reported failure enthalpy ~30 cal/g- low failure level
- Fuel dispersal observed

EPRI

Significant Progress Made Since 1994

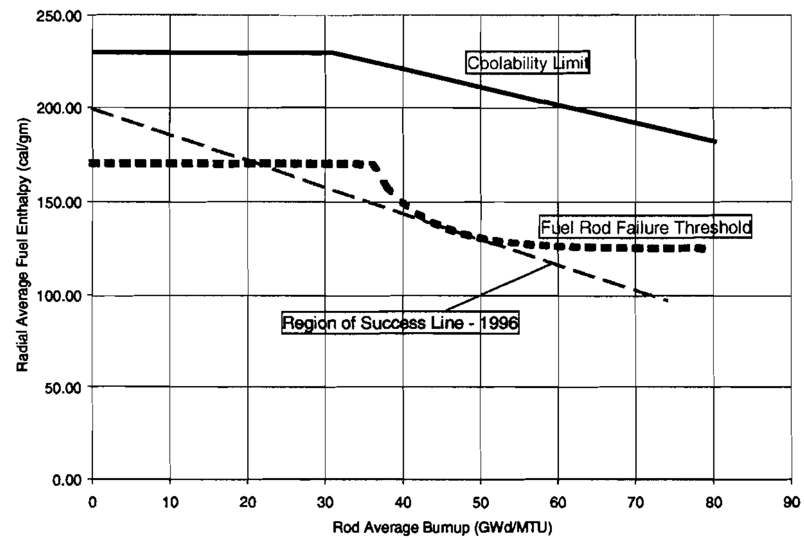
- Many RIA-simulation tests performed since 1994
  - 11 CABRI tests from France
  - 36 tests NSRR tests from Japan
  - **RepNa-1 results never duplicated**
- Considerably more knowledge and data now available
  - Good understanding and agreement from conferences and published papers on the RIA failure mechanisms
    - Data are consistent if differences in key experimental parameters are accounted for
      - Cladding ductility, temperature, pulse width
  - Analytic tools capable of predicting RIA response are available
    - FALCON, SCANAIR, and FRAPTRAN
    - Model calculations are consistent with experimental results, except RepNa-1

EPRI

Significant Progress Made Since 1994 (cont'd)

- First industry evaluation of RIA *(EPRI report, 1996)*
  - Recognized core coolability limit of 230 Cal/g
  - Proposed burnup-dependent failure limit based on "Region of Success"
    - Based entirely on RIA simulation tests
  - Many countries have used the "Region of Success"
    - A Very conservative approach
- As the knowledge base increases, new, more realistic approach is appropriate. The industry has:
  - Used FALCON, mechanical property data and RIA simulation tests to develop a revised **failure limit**
  - Adopted "no incipient melting" to ensure **coolability**
- New failure limit is consistent with experimental data and is similar to "region of success"
  - Supported by mechanical property data and RIA-simulation tests

RIA Criteria



RepNa-1 Task Force Formed

- RepNa-1 is an outlier
  - Much lower failure enthalpy compared to other RepNa tests
  - Failure did not initiate at peak power location
  - None of the codes can explain the test results
- Concerns raised:
  - Pre-existing defects
  - Accuracy of the timing of failures (interpretation of signals)
    - Narrow pulse
    - Failure occurred during the steep rise of the pulse
  - Unique pre-conditioning conditions
  - Microstructure
- RepNa-1 Task Force formed within the CABRI International Project in October, 2000
  - To perform an objective investigation of RepNa-1

EPRI

Technical Reasons To Revisit Rep Na - 1

	Burnup (Gwd/t)	Oxide (micron)	Pulse width (ms)	H at failure (cal/g)	Comment
Rep Na -1	65	80-100 (spalled)	9.5	30	Fuel dispersal
Rep Na –5	64	20	9.1	No Failure (Peak H=113)	1% strain
Rep Na –8	60	130 (spalled)	75	82	No fuel dispersal
Rep Na – 10 (Sibling of RepNa-1)	64	80 (spalled)	31	79	No fuel dispersal

EPRI

Two Major Areas Investigated By The RepNa-1 Task Force

- Uncertainties in signal analysis: microphones, different recording systems: flow meters and pressure sensors, have been used to record the timing (and enthalpy level) for rod failures & fuel dispersal
  - The reported low value was based on microphone signals
    - The acoustic signals could come from events other than failures, as demonstrated in RepNa-8
  - Significant uncertainties exist for pressure sensors and flow meters
    - Conflicting failure time from different recording systems
    - Very small volume displacement involved
  - Difficult to retrieve detailed data (generated long time ago)

Current conclusion based on signal analysis: the failure occurred between 30-50 cal/g (NOT the 30 cal/g reported)

EPRI

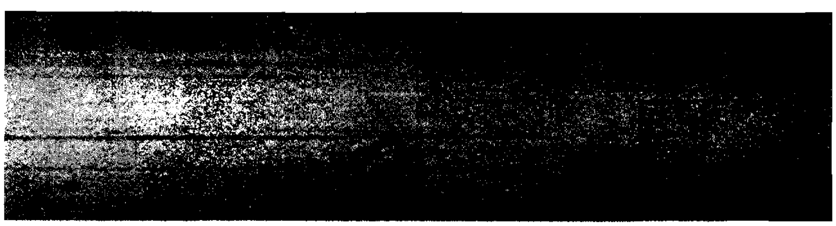
Two Major Areas Investigated By The RepNa-1 Task Force (Cont'd)

- Microstructures investigations
  - Artifact found after the re-fabrication
  - Pre-conditioning of RepNa-1 may have embrittled the cladding (Hee Chung hypothesis-LWR Fuel performance, April, 2000, Park City, Utah)
    - 380C for 14 hours (RepNa-1) vs. 310C for 12 hours
  - Cladding ductility and failure modes of RepNa-1, 8 and 10
- Current status
  - **Work in progress**, final report expected in 2003
  - Failure initiation site (90 ± 20 mm) identified by IRSN is partly ductile, peak power node (280 mm) is entirely brittle
    - PIE indicated multiple cracks with fuel loss
    - The "artifact" could not be found after the test
    - Failure could have been initiated at other locations
  - Currently reviewing mechanical property tests (PROMETRA) data and fractography for relevance to RepNa tests

EPRI

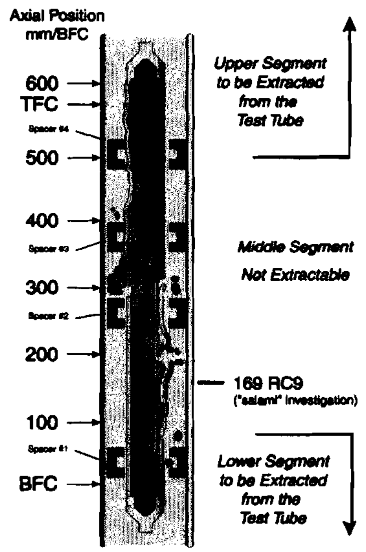
Artifact Observed After Re-Fabrication

(prior to test)



EPRI

Schematic Of RepNa-1 After The Test



EPRI

RepNa-1 Not Included In Deriving The Criteria

- Concerns investigated by RepNa-1 Task Force are significant:
  - Inconsistent timing of failure from different recording systems
  - Relevance of preconditioning temperatures
  - Artifact introduced during re-fabrication
  - Microstructures
- Sufficient number of more representative RIA-simulation tests form a consistent data base
  - RepNa-1 has significant spallations
    - Modern PWR claddings have better corrosion performance
      - M5, Zirlo and low-tin Zr-4
  - RepNa-1 has very narrow pulse (9.5 ms)
    - Typical PWR pulse is around 30 ms

EPRI

RIA Evaluation Is A Key Component Of The Robust Fuel Program (RFP)

- RFP Vision: High performance fuel for a competitive world.
- Utilities take charge to ensure
  - No operational surprises (fuel performs as advertised)
  - No regulatory surprises
  - Burnup extension

EPRI

Effort For Burnup Extension

- For burnup extension, NRC has mandated
  - The industry to propose a consistent set of criteria
  - Provide the data necessary to develop the criteria and to demonstrate compliance
- Three major RFP focus for burnup extension
  - Industry Guide
    - Framework for burnup extension
  - RIA
  - LOCA
- Robust Fuel Program has conducted/planned programs to confirm margins of current high-duty fuel designs and establish the bases for burnup extension
  - Poolside and hotcell exams, lab tests

EPRI

Recent Industry Effort

- Conducted poolside and hot cell campaigns
  - **BWR**
    - Limerick rods at 57 GWD/T
    - Limerick rods at 70 GWD/T with and without NMCA (Noble Metal Chemical Addition)
  - **PWR**
    - North Anna Zirlo at 70 GWD/T
    - North Anna M5 at 70+ GWD/T (2004)
  - Will obtain high burnup data under high-duty conditions
    - Fission gas release, corrosion, hydriding, mechanical property and others
  - Rods have also been used for safety research
    - LOCA and RIA

EPRI

Recent Industry Effort (Cont'd) – RIA

- Developed RIA Topical
- Actively participating in CABRI International Water Loop Project
  - Additional 12 tests in prototypical PWR loop planned
  - Will provide
    - RIA-simulation tests of fuel rods with advanced alloys (in 2002)
    - Tests with higher burnup fuel (>70-80 GWD/T)
    - Data on fuel/coolant interaction above the proposed failure limit
    - Mechanistic understanding on the effects of pulse width, microstructure, etc.

Cabri international project

EPRI

Proposed Test Matrix/Schedule Cabri Project

- CIP-0 series: Two tests in the Na-loop in 2002
- CIP-Q :Qualification test for the water loop in 2005
- CIP-1 : Tests in water loop, comparison tests of CIP-0 tests, 2006+
- CIP-2: High burnup UO2 fuel, >80 GWD/T
- CIP-3: Mechanistic understanding on effects of pulse width, fuel microstructure, etc
- CIP-4 Study of high burnup MOX fuel, > 60 GWD/T
- CIP-5 To be defined

EPRI

CIP0 Tests Will Determine Future Scope Of RIA

- RIA criteria proposed was based on Zircaloy clad
- Two additional RIA tests in CABRI Na-loop in 2002
  - **CIP0-2**
    - M5 rod (~ 20μm, ~73 GWd/T)
    - Test will be performed in 10/02
    - 30 ms, with enthalpy of ~95 cal/g (based on calculations)
  - **CIP0-1**
    - ZIRLO rod (~ 100μm, ~73 GWd/T)
    - Test will be performed in 11/02
    - 30 ms, with enthalpy of ~90 cal/g (based on calculations)
- New parameters involved
  - Higher burnup, 63 GWD/T → 73 GWD/T
  - New alloys, M5 and Zirlo

EPRI

Industry Has Submitted The RIA Topical

- Based on extensive data coupled with analytical evaluations
  - Over 80 RIA-simulation tests using irradiated rods
  - Extensive corrosion and mechanical property tests
  - Analysis and experiments on fuel/coolant interaction
- RIA tests to be performed in 2002 using high burnup LWR rods with advanced alloys
  - Confirm the conservatism in proposed criteria
  - Can be used to develop criteria for advanced alloys
- Data from the Cabri Water Loop Project will NOT change conclusions of the current RIA Topical
  - Na-loop test results are conservative (lower clad temperature)
  - DNB-induced failure mechanisms are NOT expected at the proposed failure limits
  - Will provide margins and enhanced understanding of post-CHF rod behavior

United States Nuclear Regulatory Commission

UPDATE ON ISSUES IN 1998 AGENCY PROGRAM PLAN FOR HIGH-BURNUP FUEL

Ralph Meyer

Office of Nuclear Regulatory Research

ACRS Subcommittee

October 9, 2002

ORIGINAL LIST OF ISSUES

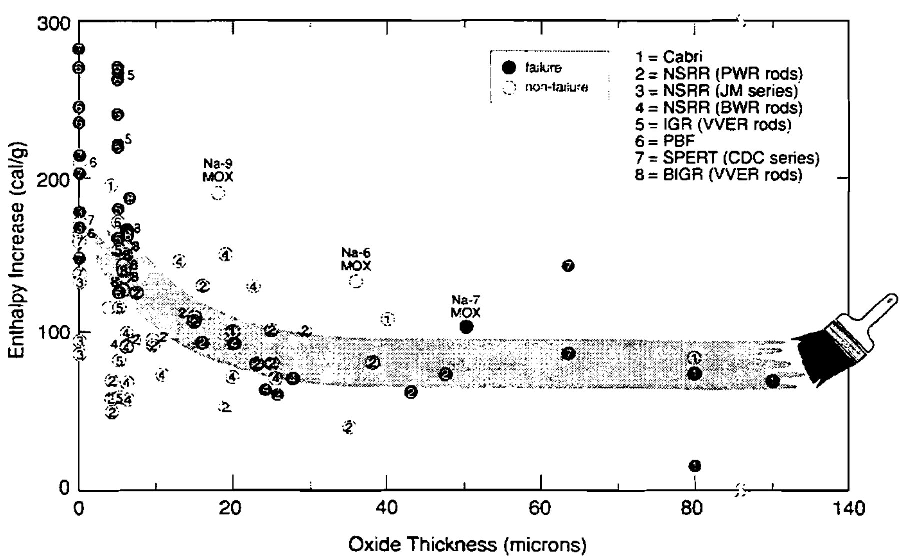
1	Cladding Integrity and Fuel Design Limits	Resolved in original plan (no further discussion)
2	Control Rod Insertion Problems	Resolved in original plan (no further discussion)
3	Criteria and Analysis for Reactivity Accidents	NRC confirmatory assessment at 62 GWd/t, early 2005. Revision of Reg. Guide 1.77, TBD.
4	Criteria and Analysis for Loss-of-Coolant Accidents	Zircaloy criteria and models at 62 GWd/t, 2004. New performance-based criteria possible.
5	Criteria and Analysis for BWR Power Oscillations (ATWS)	Schedule to be determined
6	Fuel Rod and Neutronic Computer Codes for Analysis	Resolved
7	Source Term and Core Melt Progression	Technical issues essentially resolved. Revision of Reg. Guide 1.183, TBD.
8	Transportation and Dry Storage	Research Information Letter, 2004
9	High Enrichments (>5%)	No activity needed now (no further discussion)

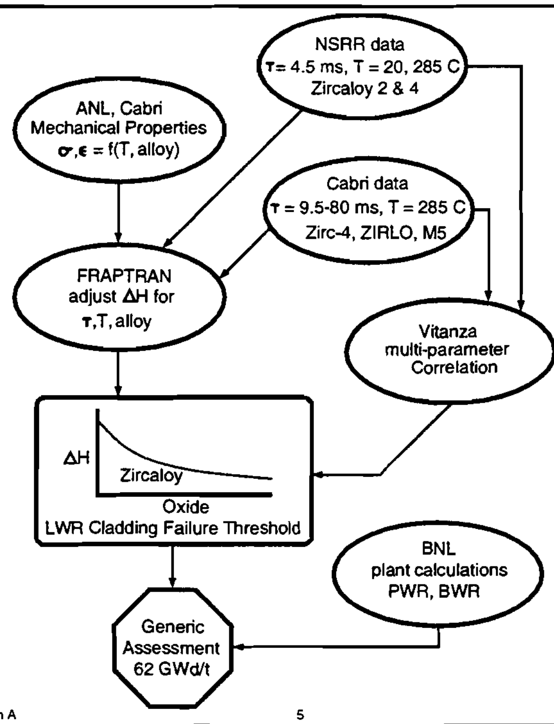
CRITERIA AND ANALYSIS FOR REACTIVITY ACCIDENTS

ISSUE: 280 cal/g regulatory limit in Reg. Guide 1.77 is not adequate for high-burnup fuel. New limit needed.

METHOD: (see following slides)

SCHEDULE:
Cabri test(s) late 2002 (early 2003)
ANL Zircaloy mechanical properties 2003
NSRR Zirc. tests in high-temp. capsule late 2004
NRC confirmatory assessment 62 GWd/t early 2005



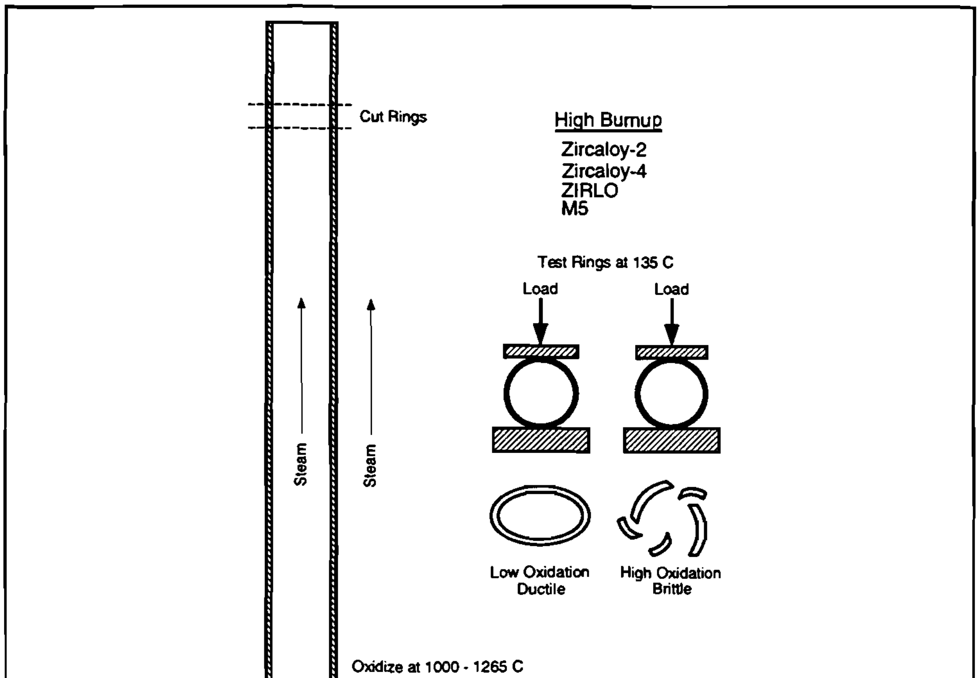


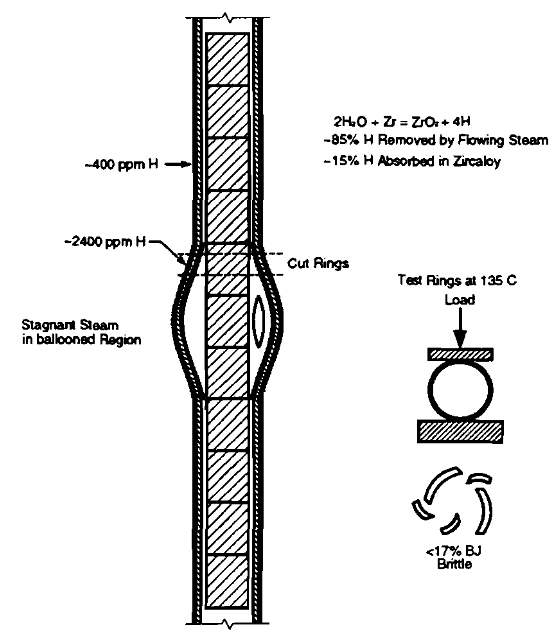
CRITERIA AND ANALYSIS FOR LOSS-OF-COOLANT ACCIDENTS

ISSUE: **Embrittlement criteria in 10 CFR 50.46 and related evaluation models are probably affected by burnup and alloy. Check and revise if necessary.**

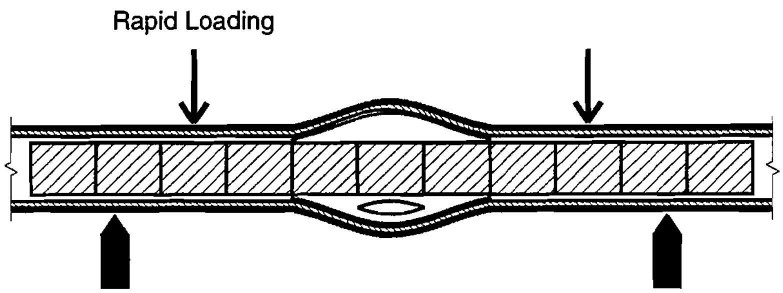
METHOD: **(see following slides)**

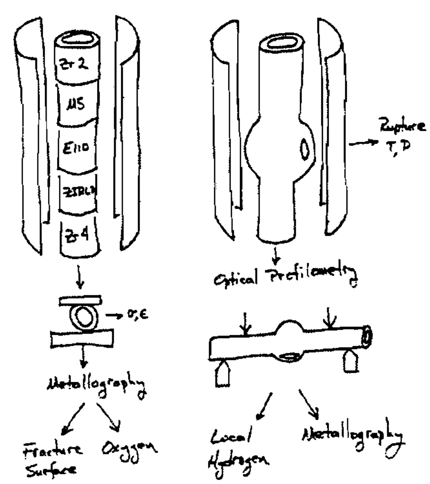
SCHEDULE: **Zircaloy criteria and models at 62 GWd/t in 2004**





Hydrogen Effect Discovered ~1980


Rapid Loading


Zr 2
MS
E110
ZIRLO
Zr4
σ,ε
Metallography
Fracture Surface
Oxygen
Rupture
T,P
Optical Profilometry
Local Hydrogen
Metallography

CRITERIA AND ANALYSIS FOR BWR POWER OSCILLATIONS (ATWS)

ISSUE:	280 cal/g limit currently used may not be adequate to ensure benign result in PRA for "successfully" terminated oscillations
METHOD:	Analytical + some experimental separate effects
SCHEDULE:	TBD

FUEL ROD AND NEUTRONIC COMPUTER CODES FOR ANALYSIS

ISSUE:	NRC codes did not have high-burnup capability and were needed to help review vendor codes for high-burnup applications.
METHOD:	Develop, assess, peer review
SCHEDULE:	Resolved

SOURCE TERM AND CORE MELT PROGRESSION

ISSUE: Applicability of NUREG-1465 source terms to high-burnup fuel

METHOD: Expert elicitation, more data

SCHEDULE: Expert elicitation completed in June 2002
VERCORS, PHEBUS, VEGA data as available
Revision of Reg. Guide 1.183 TBD

TRANSPORTATION AND DRY STORAGE

ISSUE: What is the effect of burnup on fission product inventory (shielding, heat source, activity) and cladding degradation (removal from storage)?

METHOD: Direct tests and measurements

SCHEDULE: ANL tests on Zircaloy in 2003
Research Information Letter in 2004

United States Nuclear Regulatory Commission

ANALYSIS OF RIA AND ATWS EVENTS

Ralph Meyer
Office of Nuclear Regulatory Research

ACRS Subcommittee
October 9, 2002

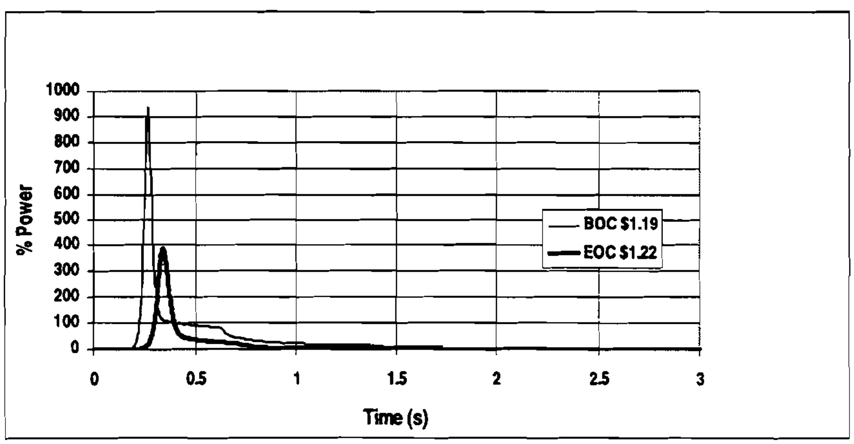
REACTIVITY-INITIATED ACCIDENTS (RIA) [Unassembled Pieces of the Puzzle]

- Summarize Pulse Width Situation
- Show Vitanza Correlation
- Describe Method for Making Temperature Corrections

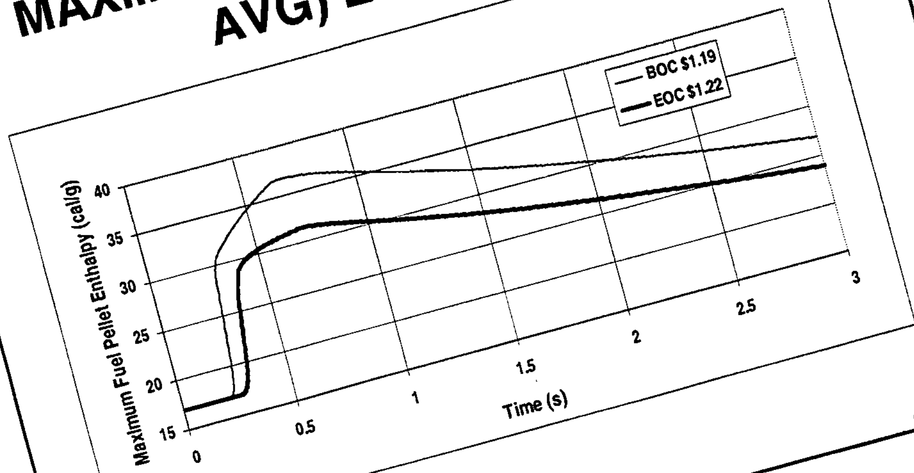
BNL CALCULATIONS OF PULSE WIDTH

- PWR Rod-Ejection Accident (REA)
- PWR Boron-Dilution Accident
- BWR Rod-Drop Accident (RDA)

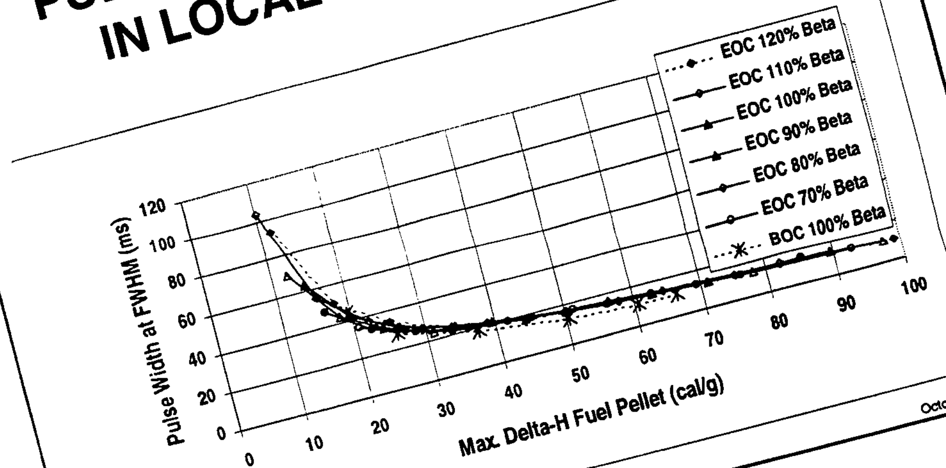
POWER DURING AN REA



MAXIMUM LOCAL FUEL (PELLET AVG) ENTHALPY



PULSE WIDTH VS MAX CHANGE IN LOCAL FUEL ENTHALPY



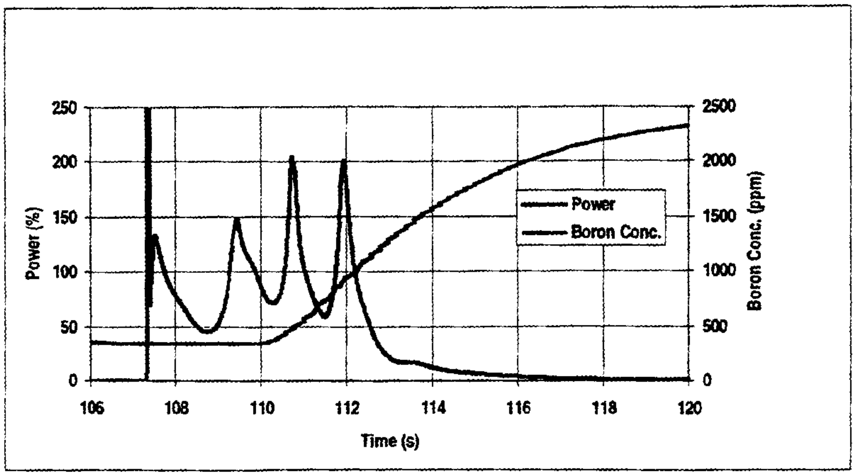
CONCLUSIONS FOR REA

- General trends are in agreement with analytical model
- Pulse width is 25-100 ms as energy deposition goes from 30 to 10 cal/g; range for most likely prompt-critical REAs
- Pulse width is 10-15 ms for energy depositions (fuel enthalpy change) of 60-100 cal/g
  - If testing limits of fuel, use these short pulse widths

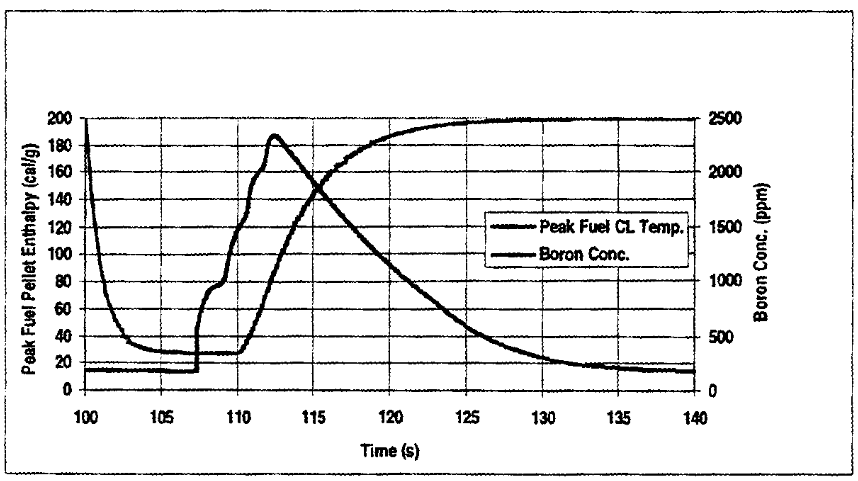
PULSE WIDTH FOR BORON DILUTION EVENT

- Worst case considered at BNL for 25% pump start
- Initial power spike most severe
  - Pulse width 20-40 ms corresponding to peak enthalpy increase of 30-15 cal/g
  - Additional power spikes with much longer pulse widths

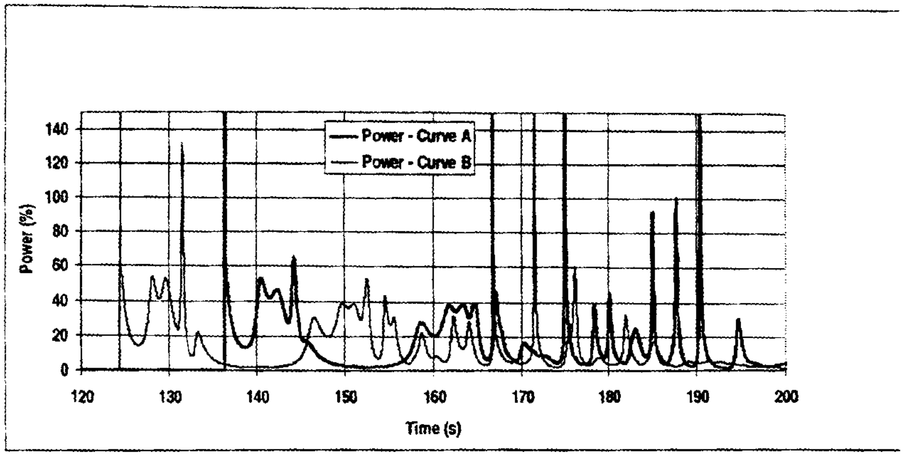
BORON DILUTION WITH PUMP ON



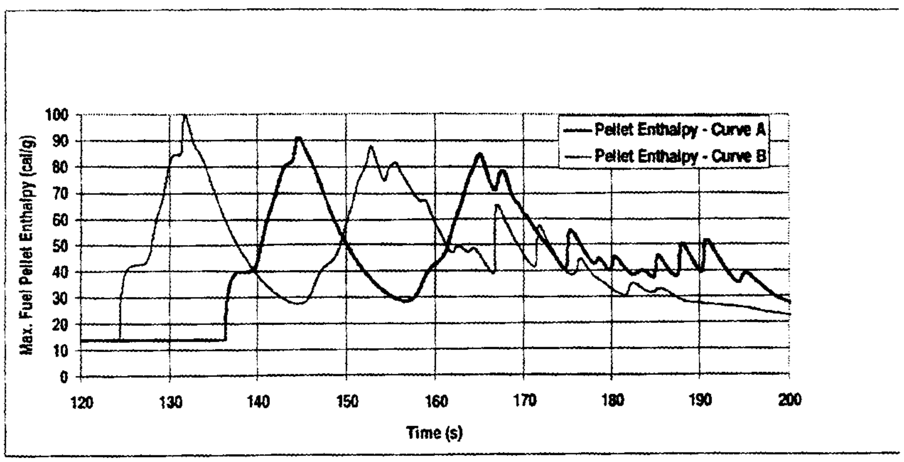
PEAK PELLET-AVERAGE ENTHALPY



BORON DILUTION - POWER UNDER NATURAL CIRCULATION CONDITIONS



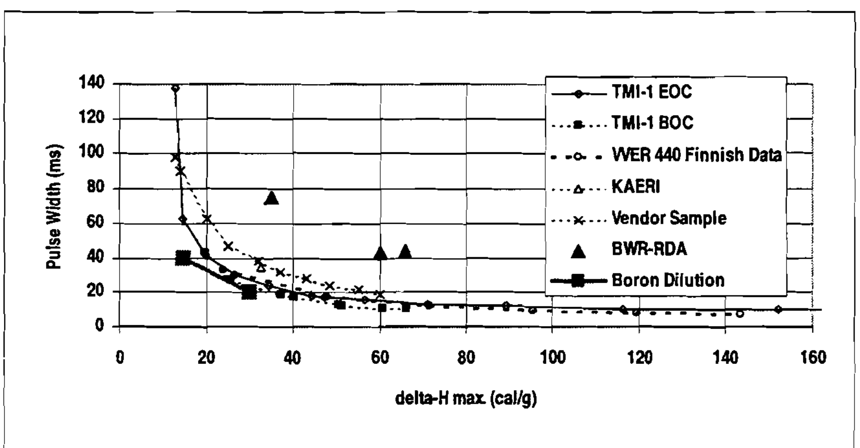
BORON DILUTION – PEAK FUEL ENTHALPY UNDER NATURAL CIRCULATION CONDITIONS

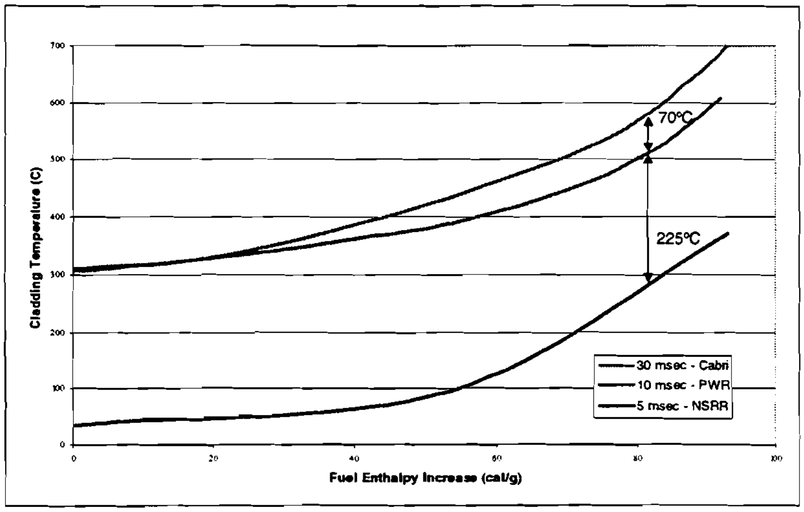


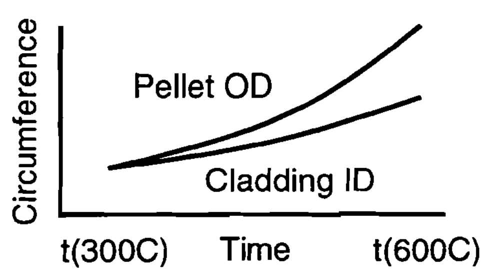
BWR ROD DROP ACCIDENT (RDA)

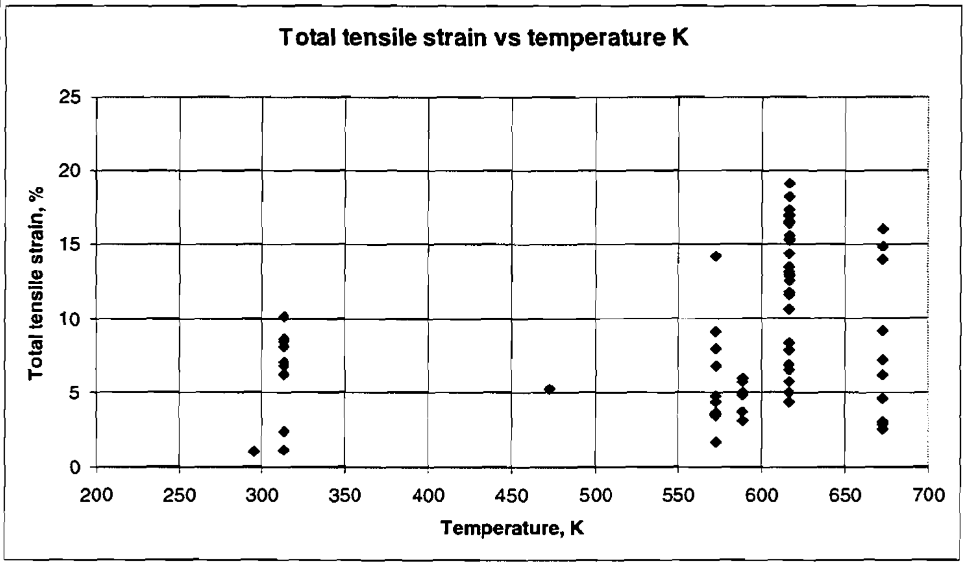
- Previous analysis by BNL et al. substantial but pulse width not usually given or measurable from power vs time paper plots
- Data points (limited number) suggest pulse widths longer than for PWRs
  - In part due to longer neutron lifetime (average time needed for a fission neutron to cause another fission—P ~ $P_0 \exp(at/\ell)$ where a depends on reactivity and ℓ is the lifetime)

PULSE WIDTH FROM PWR AND BWR ANALYSIS OF DIFFERENT RIAs

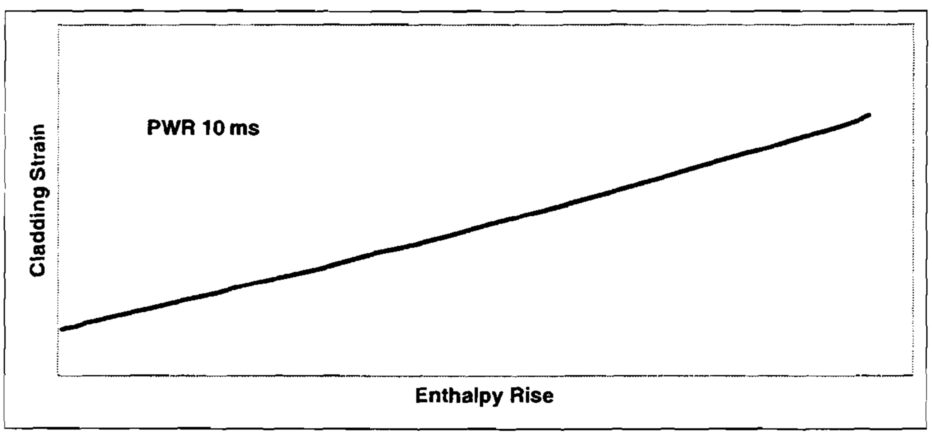



Cladding Temperature (C)
700
600
500
400
300
200
100
0
0
20
40
60
80
100
Fuel Enthalpy Increase (cal/g)
70°C
225°C
30 msec - Cabri
10 msec - PWR
5 msec - NSRR


Circumference
Pellet OD
Cladding ID
t(300C)
Time
t(600C)



Total tensile strain vs temperature K
Total tensile strain, %
0
5
10
15
20
25
200
250
300
350
400
450
500
550
600
650
700
Temperature, K



PWR 10 ms
Cladding Strain
Enthalpy Rise

SOME PROGRESS ON ANALYZING FUEL BEHAVIOR DURING BWR POWER OSCILLATIONS

- PIRT Implications (reminder)
- Data from Japan (new)
- Codes from Finland (progress)

IMPLICATIONS FROM POWER-OSCILLATION PIRT

(ACRS Subcommittee, April 4, 2001)

- 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

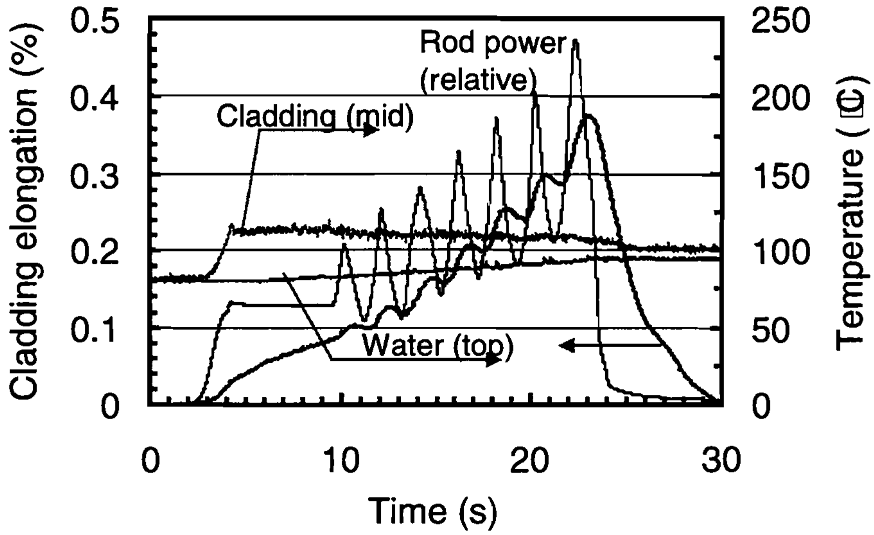
(ACRS Subcommittee, April 4, 2001)

- 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.

NEW JAERI DATA TO BE PRESENTED AT NSRC-2002

- Two tests performed on BWR rods with 25 and 56 GWd/t burnup
- PCMI not enhanced by cyclic loads (i.e., no ratchet effect)
- Results not fully analyzed (by NRC, anyway)



FRAPTRAN-GENFLO CODE ANALYSIS

- **Coupled codes installed at PNNL in early September 2002**
- **Sample cases have been run by PNNL and NRC staff**
- **Analytical plan to be developed in 2003**

United States Nuclear Regulatory Commission

CREEP TESTING OF SPENT FUEL RODS IN DRY STORAGE

By
Sudhamay Basu
Office of Nuclear Regulatory Research

Presentation for
ACRS Fuel Subcommittee Meeting
October 9, 2002

Program Scope

- Post-storage characterization of spent fuel rods
  - profilometry
  - fission gas analysis
  - oxide and hydride, hydrogen content
  - mechanical properties (tensile, microhardness)
- Creep testing of fuel rods
- Post-creep mechanical properties
  - tensile
  - ductility
- Medium (≤45 GWd/MTU) and high burnup cladding
- Focus of presentation - testing of PWR (Surry) fuel rods
  - average rod burnup of 36 GWd/MTU
  - rods in spent fuel pool ~5 years
  - rods stored in dry cask since 1985

Regulatory Issues

- License renewal of existing dry casks for storing spent nuclear fuel
  - applications expected as early as 2004
  - cask integrity for continued storage (additional 20 to 100 years) important for safe storage under normal and accident conditions
- Licensing new casks for storage and transport of high burnup fuel
  - power plants to discharge more high burnup (>45 GWd/MTU) fuel
  - spent fuel pools to loose full core reserve capacity
  - timely licensing important for safe storage and transport
- Spent fuel in dry casks must be protected from degradation that leads to gross ruptures (10CFR Part 72)
- Creep and mechanical properties data required for spent fuel cladding in long-term storage
- For high burnup fuel, technical basis required to demonstrate validity of Part 72 requirements

Post-Storage Characterization

- Profilometry data
  - diameter changes ~0.6%, very little variation
  - thermal creep during storage <0.1%
- Gas analysis data
  - fission gas release 0.4 to 1.0% - within range
  - internal gas pressure ~3.5 MPa - within range
- Metallography data
  - OD oxide layer thickness ~20 - 40 μm
  - hydrogen content ~200 - 300 wppm
  - no hydride reorientation
- Mechanical properties data
  - post-storage microhardness ~240 DPH (no annealing)
  - creep tests
  - tensile tests

Surry Creep Test Matrix

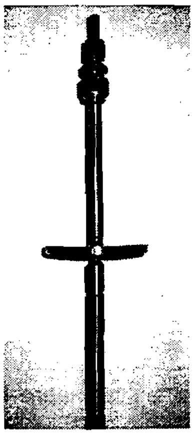
Test No.	Temp. (°C)	Stress (MPa)	Purpose
1	380	220	primary/secondary creep, residual creep strain
2	380	190	primary/secondary creep
3	400	190	primary/secondary creep
4	400	250	primary/secondary creep, residual creep strain
5	360	220	primary/secondary creep
6	400	160	primary/secondary creep
7	400	220	primary/secondary creep, residual creep strain

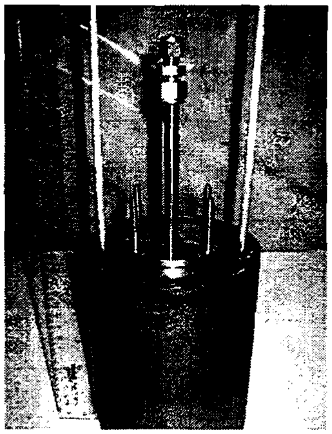
Test Description

- Specimen Configuration
  - 3.0 in. cladding segments, defueled, refilled with Zr pellets, welded ends
  - specimens pressurized with argon gas up to 6000 psi (330 MPa)
  - Pressure regulated to ±10 psi
  - five specimens loaded in furnaces for concurrent creep testing
- Measurements
  - temperature and pressure measured for control
  - diameter measurements at multiple axial and azimuthal locations by laser profilometry
  - length measurements for possible creep anisotropy
- Derived data
  - hoop strain from diameter measurements
  - strain rate from strain-time history

Thermal Creep Tests
Specimen Test Chamber

- **Purposes**
  - **preclude oxidation**
  - **mitigate contamination spread in case of rupture**





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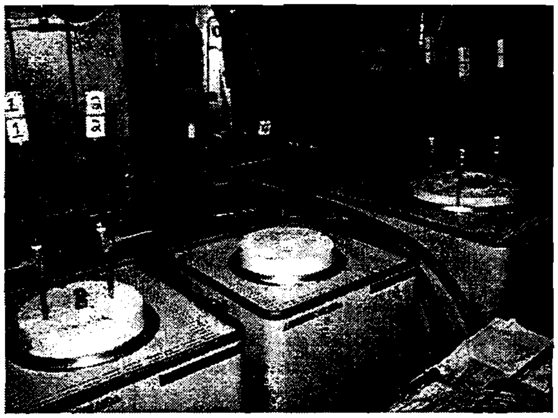
Thermal Creep Tests
Furnaces

- **One sample each in the 2 small furnaces**
- **Three samples in the large furnace**



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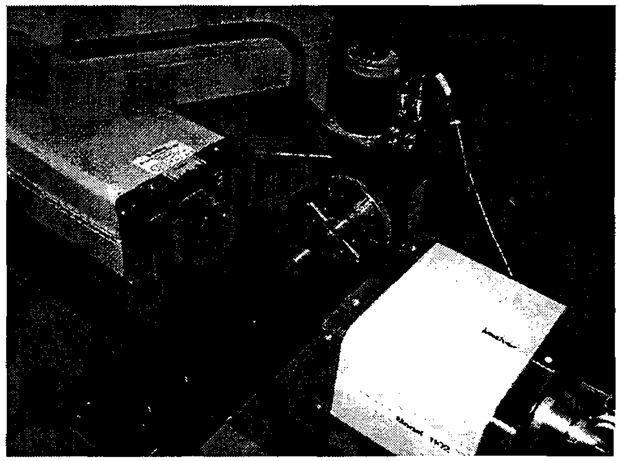
Thermal Creep Tests
Laser Profilometry

Diametral measurement intervals:
- **9° azimuthal**
- **0.3 in. axial**

Length is measured by profiling the bottom end



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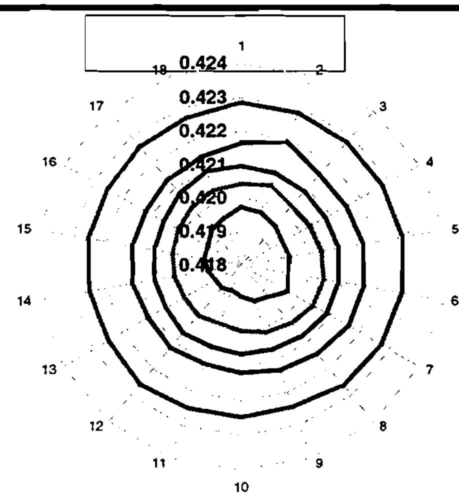
Laser Profilometry
- Data Reduction

- **Cross-sectional profile of a sample after**
  - **0 h,**
  - **335 h,**
  - **671 h,**
  - **1028 h, and**
  - **1820 h,**



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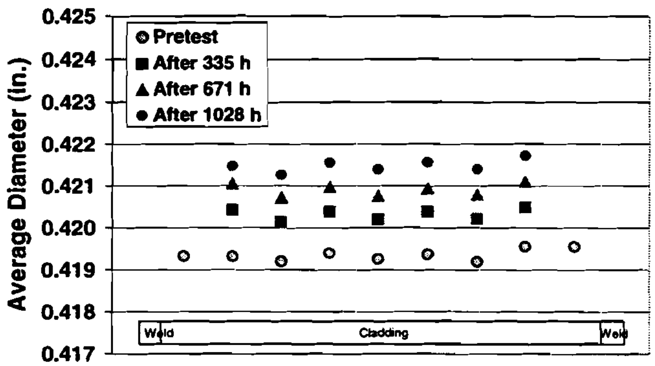
Laser Profilometry
- Data Reduction

- Middle 5 axial readings are used to produce the specimen's average diameter



Surry Thermal Creep Tests
- Summary Results

Test No.	Temp. (°C)	Stress (MPa)	Duration (hrs)	Avg. Strain	Failure	Strain Rate (%/hr)
1	380	220	2180	1.10	No	4.5×10^{-4}
2	380	190	2348	0.35	No	8.8×10^{-5}
3	400	190	1873	1.03	No	4.9×10^{-4}
4	400	250	693	5.83	No	$>4.9 \times 10^{-3}$
5	360	220	3305	0.22	No	4.2×10^{-5}

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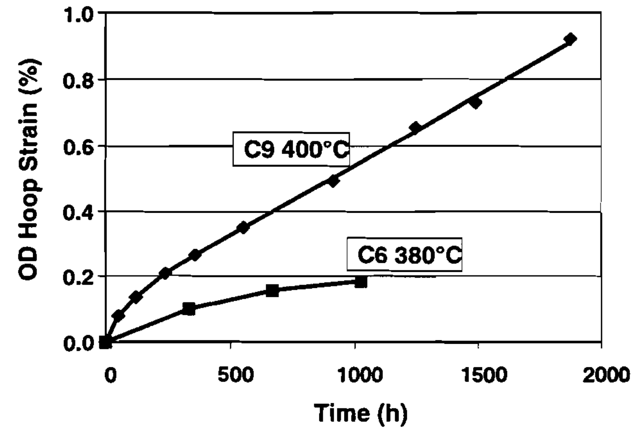
Summary Test Results

Temperature Dependence

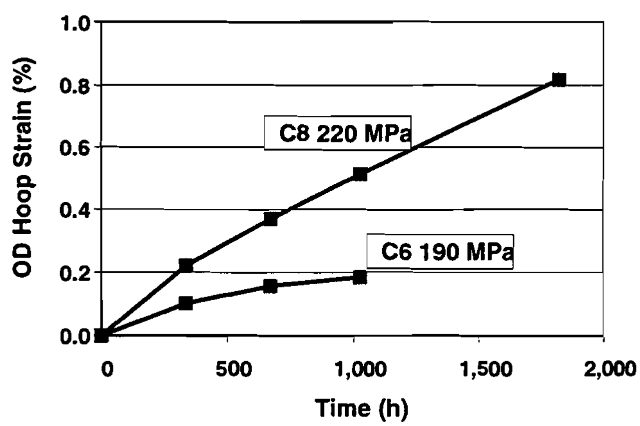
Both at 190 MPa



Summary Test Results

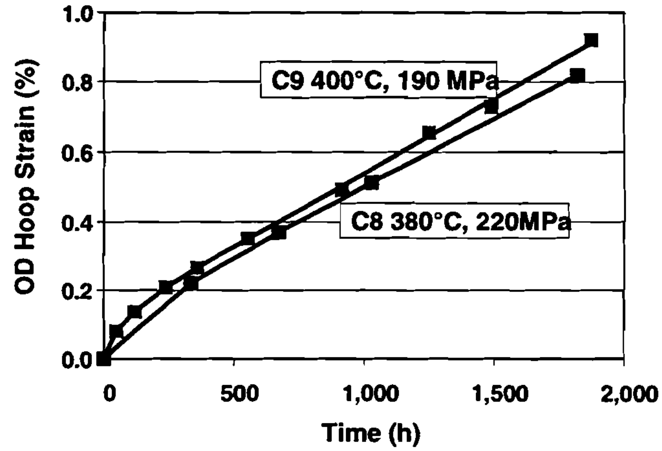
Stress Dependence

Both at 380°C



Summary Test Results

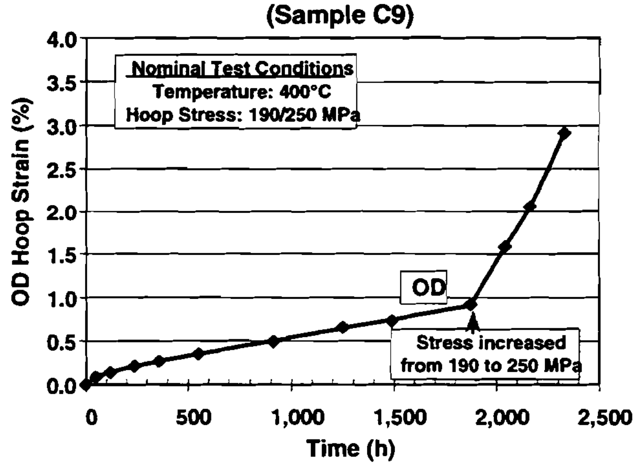
Combined Effects



Summary Test Results

Effect of Increased Stress

(Sample C9)



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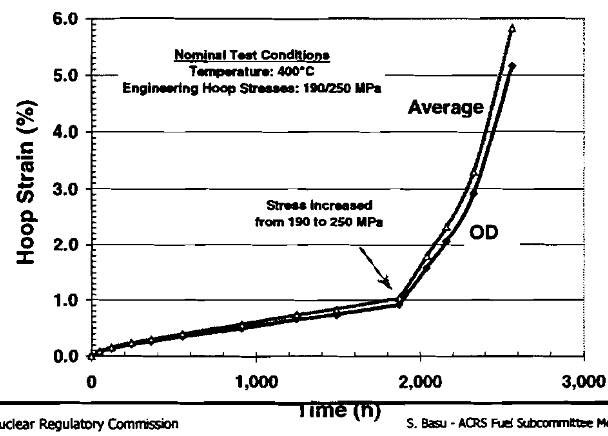
Surry Thermal Creep Results (C9)

- 400°C, 190/250 MPa engineering hoop stress, 2566 h
- 5.8% average hoop strain, no rupture



Conclusions of Surry Creep Testing Program

- Significant residual creep strain demonstrated for Surry cladding after 15-y dry-cask storage
- Creep data show strong temperature and stress dependency in the regime tested
- Two additional tests at 400°C and 160 MPa and 220 MPa, respectively, planned to expand the database
- Data on possible hydride reorientation and post-creep ductility to be generated
- Lower temperature tests for permanent repository applications may also be carried out in future

High Burnup Spent Fuel Testing for Dry Cask Storage

- **Program Program Scope**
  - fuel and cladding characterization
  - isotopic analysis
  - annealing tests
  - thermal creep tests
  - mechanical properties tests
- **Cladding Material**
  - H.B. Robinson PWR rods: 2 rods (2.9% enrichment, 67 GWd/MTU); one rod (1.9% enrichment, 10% Gd_2O_3, 47 GWd/MTU)
  - OD oxide thickness ~ 60 µm to 110 µm
  - hydrogen content ~ 580 wppm to 750 wppm
- **Program Status**
  - characterization and isotopic analysis in progress
  - annealing tests completed
  - creep test matrix developed; lead test started
  - mechanical properties testing planned

Robinson Cladding Annealing Test Results

- Using microhardness as the figure-of-merit, determined percent recovery of radiation damage in cladding
- Preliminary findings – annealing can produce significant recovery

Robinson Zry-4	H wppm	Annealing		Vickers DPH	% Recovery
		Temp.°C	Time, h		
Nonirrad.	<10	---	---	203	100
As-irrad.	≈600	---	---	252	0
As-irrad. & Annealed	≈600	420	20 72	226 215	54 75
As-irrad. & Annealed	≈600	450	2 10	224 217	58 71
As-irrad. & Annealed	≈600	500	2 48	218 206	69 94

High Burnup Thermal Creep

- **Testing Strategy**
  - **Conduct two lead tests duplicating Surry test conditions to determine effects of Robinson's higher hydrogen content and fast fluence**
    - **One of the lead tests has been started**
  - **Establish test matrix based on lead test results**
    - **Simple and flexible**
    - **Emphasizing 400°C**
  - **Duplicate Surry creep testing techniques**
    - **5 additional systems being built**

Preliminary HBR Creep Matrix
(07/12/02 Version)

H-content wppm	Temp. °C	Stress MPa	Time h	Predicted Strain, %
650±50	400	220	TBD	TBC
650±50	400	190	TBD	TBC
650±50	400	160	TBD	TBC
650±50	420	160	TBD	TBC
650±50	380	220	TBD	TBC
650±50	380	190	TBD	TBC
650±50	380	160	TBD	TBC
650±50	360	220	TBD	TBC
650±50	360	190	TBD	TBC

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United States Nuclear Regulatory Commission

ANL LOCA-RELEVANT RESEARCH

H. H. Scott, NRC-RES

Y. Yan and M. C. Billone, ANL

NRC-ACRS Meeting
Rockville, MD
October 9, 2002

HIGH BURNUP LOCA FEATURES

- **BWR Fuel Rods (Limerick at ≈57 GWd/MTU, ≈10 μm OD Oxide)**
  - Effect of tight fuel-cladding bond and restricted gas flow on ballooning, burst, inner-surface-oxidation/hydrogen-pickup
  - Effect of irradiation on high temperature oxidation in steam
  - Effect of fuel-cladding mechanical interaction on fragmentation resistance during water quench; post-quench ductility

- **PWR Fuel Rods (HBR at ≈67 GWd/MTU, ≤ 100 μm OD Oxide)**
  - Similar fuel-cladding features as for BWR
  - Effect of in-reactor oxide layer on oxidation kinetics and ECR.
  - Effect of hydrogen pickup on oxidation kinetics, fragmentation-resistance during water quench and post-quench ductility

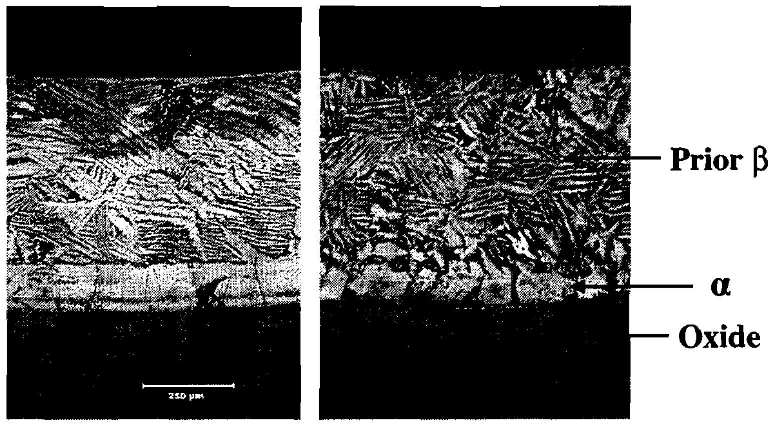
ANL LOCA-RELEVANT TESTS FOR HIGH BURNUP FUEL CLADDING

- **Steam Oxidation Kinetics Studies**
  - 900-1300°C, emphasis on 1204°C for 5-20 minutes
  - Kinetics of weight gain, (oxide + α) layer growth rate, effective β layer thickness vs. ECR
- **LOCA Integral Tests**
  - Test adequacy of 10CFR50.46 ECCS licensing criteria (ECR $\leq$ 17%, T $\leq$ 1204°C) for high burnup fuel
  - Determine ECR thresholds for thermal quench fragmentation and loss of post-quench ductility
- **Post-Quench Ductility Tests (Bend & Ring Compress.)**

SUMMARY OF HIGH-TEMPERATURE STEAM OXIDATION KINETICS RESULTS

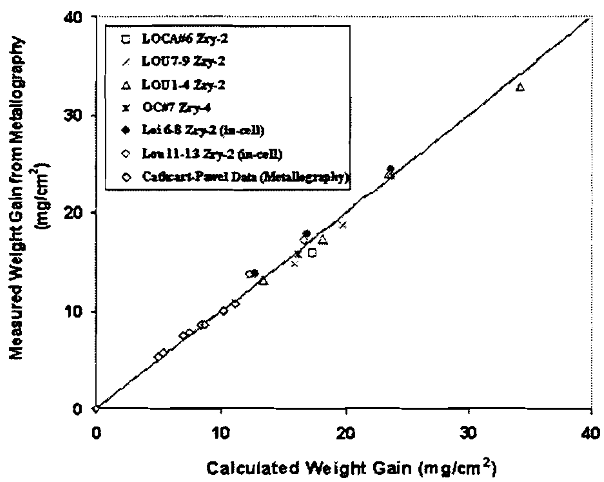
- **Metallographic Results for 1200°C Tests**
  - No difference in measured weight gain (Δw_m) for unirradiated and irradiated (10-µm pre-test oxide layer) Zry-2 and unirradiated Zry-4
  - Excellent agreement between measured Δw_m and Cathcart-Pawel (CP) model predictions (Δw_p)
  - CP Δw_p is good "best-estimate" correlation for Zry-2, Zry-4, ZIRLO, M5 and E110 at 1100-1500°C
- **Metallographic Analysis for 1000-1100°C Test Samples (in progress)**

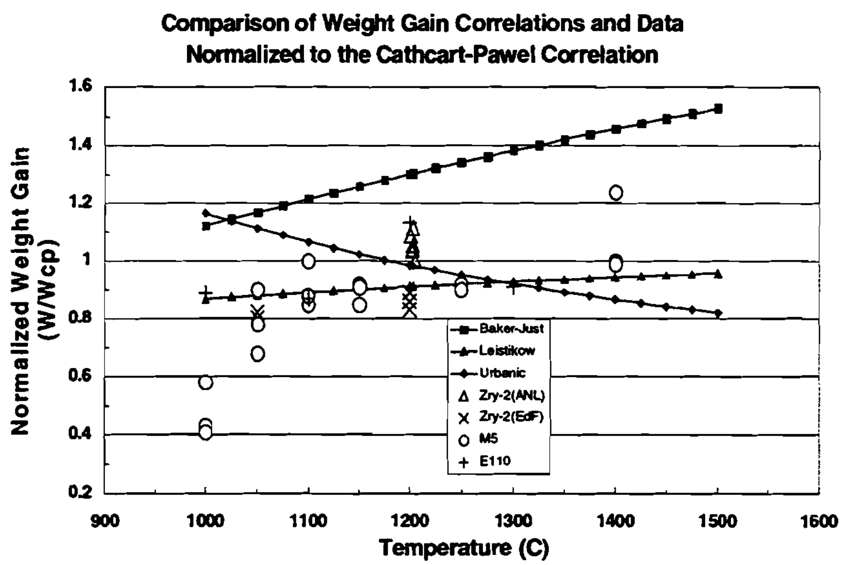
Oxide, α and β Layer Characteristics (in Steam at 1204°C for 10 Minutes)



unirradiated Zry-2 irradiated Zry-2

Measured Weight Gain from Metallography for Irradiated and Unirradiated Zry-2 and Zry-4





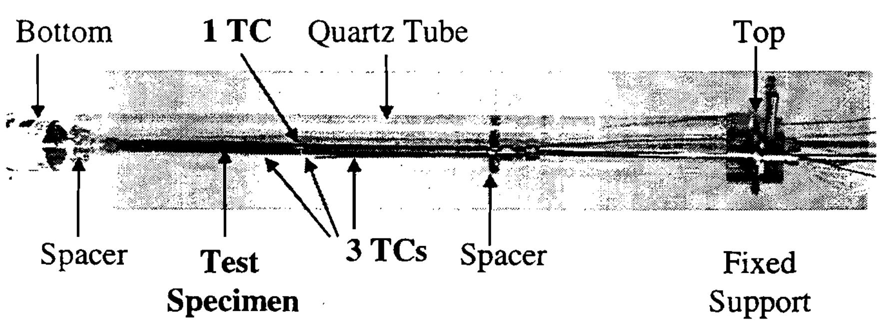
SUMMARY OF STEAM OXIDATION KINETICS RESULTS (Cont'd)

- **Assessment of Cathcart-Pawel Models**
  - CP model based on very rapid heating and cooling rates
  - Weight gain correlation is good even for slow ramp rates
  - Underpredicts α-layer and overpredicts β-layer thickness for LOCA-relevant cooling rates (1-8°C/s) due to oxygen diffusion from β to α phases during 1200°C → 800°C
  - ANL results at ≈5°C/s cooldown from 1200°C to 800°C
  - Impact is TBD as "ductility" increases with reduction in oxygen and decreases with thickness reduction

LOCA INTEGRAL TESTING SCOPE

- **Parameters Common to BWR and PWR Tests**
  - Fuel-cladding samples = 305-mm long; fueled region = 270 mm
  - PCT = 1204±20°C, temperature ramps relevant to SB-LB LOCA
  - Internal pressure P_i <1.3×system pressure, plenum V = 5 to 10 cc
  - Best-estimate 17% ≤ ECR < ≈30% → oxidation time ≈2-10 min.
- **High Burnup BWR Rods (Limerick)**
  - Temperature ramp rate = 5°C/s (2.5-7°C/s for SB-to-LB LOCA)
  - Cladding $\Delta P = P_i - P_s \leq$ 8.6 MPa [6.7 MPa (SB)- 8.6 MPa (LB)]
- **High Burnup PWR Rods (H. B. Robinson)**
  - Temperature ramp rate = 5°C/s (1-2°C/s for SB, 7-10°C/s for LB)
  - Cladding $\Delta P = P_i - P_s <$ 20 MPa [P_s = 3.4 → 0.2 MPa (SB → LB)]

LOCA TEST TRAIN ASSEMBLY



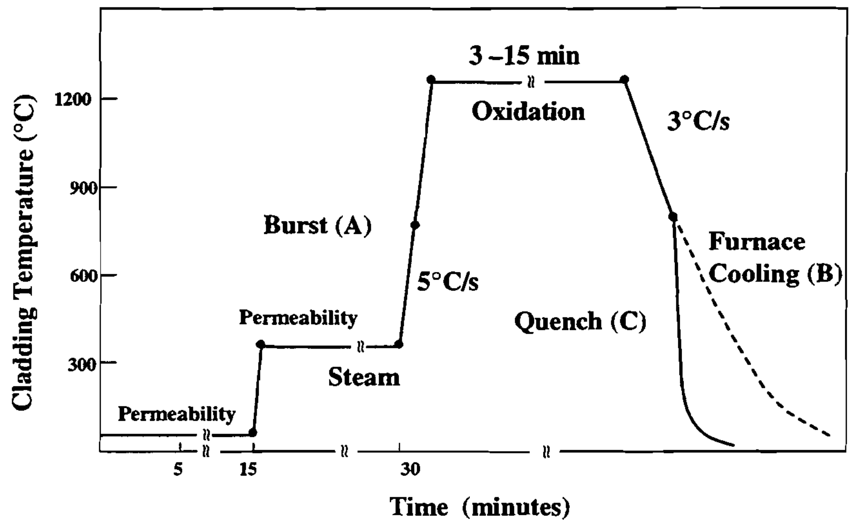
LOCA INTEGRAL TESTING SCOPE (Continued)

- **Steam and Quench Water Flow-rates/Volume**
  - Steam flow = 5-10 g/minute
  - Cool-down rate = 3°C/s from 1204°C to 800°C (1-8°C/s for BWR)
  - Quench water velocity = 5 mm/s (initiated at 800°C)
- **Test Times at 1204°C**
  - Maximum ECR depends on wall thinning and extent of double-sided oxidation
  - First test will be run for 5 minutes at 1204°C

LOCA INTEGRAL TEST SEQUENCE FOR FIRST SERIES OF BWR TESTS

- **Phase A: Fuel Permeability, Ballooning and Burst**
  - Permeability at 30°C and 300°C
  - Ramp (5°C/s) to burst in high purity argon
  - Slow furnace cool from burst temperature
- **Phase B: Above Plus Oxidation**
  - Permeability (30°C and 300°C); ramp to 1204°C in steam
  - Hold (5 min.) at 1204°C; cool to 800°C at 3°C/s
  - Slow furnace cool from 800°C to RT
- **Phase C: Above Plus Quench at 800°C**
  - Repeat B through cooling to 800°C; quench at800°C

LOCA INTEGRAL TEST SEQUENCE



SUMMARY OF OUT-OF-CELL LOCA INTEGRAL TEST RESULTS

- **Test Specimens and Conditions**
  - Specimens: GE-11 (9×9) Zry-2 cladding (0.71-mm wall), zirconia pellets with 0.1-mm radial gap, 10-cm^3 void volume above pellets
  - Conditions: cladding $\Delta P = 8.62$ MPa at RT

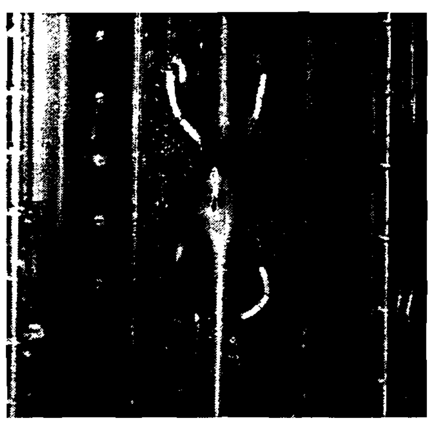
- **Test #3 Results (10 min. in steam at 1204°C)**
  - Peak $\Delta P = 9.31$ MPa, burst $\Delta P \geq 8.41$ MPa, burst $T \approx 760$°C
  - "Dog-bone-shaped" burst opening; ≈13-mm long
  - Peak $\Delta D/Do \approx 45\%$; axial extent of balloon ≤130 mm
  - Specimen survived thermal quench & post-quench handling

SUMMARY OF OUT-OF-CELL LOCA INTEGRAL TEST RESULTS (Cont'd)

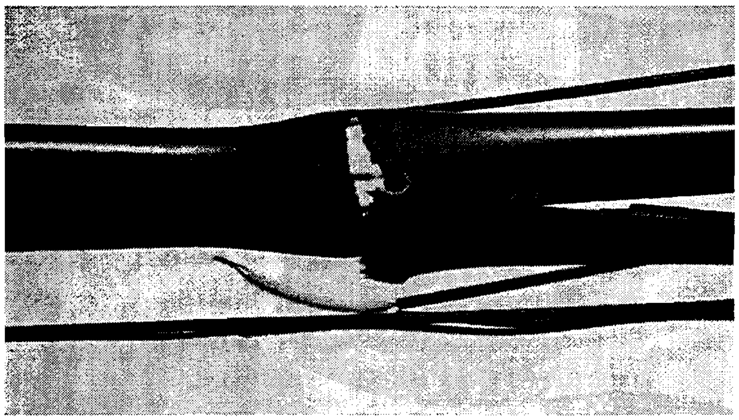
- **Test #4 Results(10 min. in steam at 1204°C)**
  - Peak ΔP = 10.28 MPa, burst $\Delta P \geq$ 9.42 MPa, burst T $\approx$ 720°C
  - Similar burst opening and ballooning strain as in Test #3
  - Sample failed across mid-burst region at 100°C after quench
  - Based on results, future specimens will be pressurized at 300°C and time at 1204°C will be < 10 min.
- **Test #5 Results(ramped to burst in Ar)**
  - Peak ΔP = 8.95 MPa, burst $\Delta P \geq$ 8.61 MPa, burst T $\approx$ 732°C
  - "Dog-bone-shaped" burst opening; $\approx$13-mm long; 2-mm wide
  - Peak $\Delta D/Do \approx$ 44%; axial extent of balloon $\approx$100-mm long

Out-Cell Test 3: 10 min. at 1204°C, C-P ECR = 38%

(Survived quench & post-quench handling)



Out-of-cell Test 4: 10 min. at 1204°C, C-P ECR = 38%
(Survived quench; fractured at 100°C under dead-weight load)



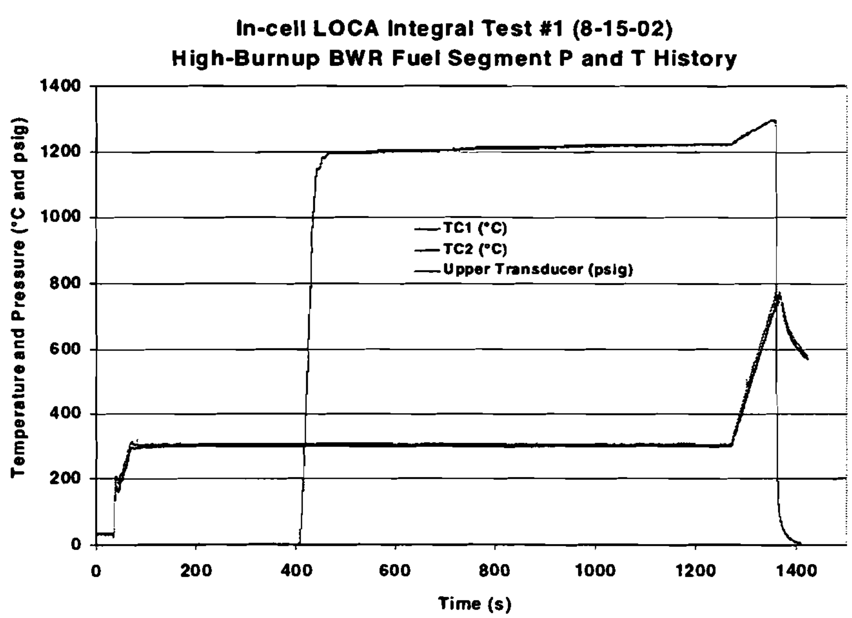
1st LOCA INTEGRAL TEST RESULTS
LIMERICK HIGH-BURNUP BWR PHASE A

- **Limerick Specimens Prepared**
  - Phase A: middle of Grid Span #5; 0.46-0.76 m above fuel MP
  - Phase B: middle of Grid Span #6; 0.94-1.24 m above fuel MP
  - Phase C: to be prepared from GS #5 & 6 of different rod
- **Phase A Test (Completed on 08-15-02)**
  - Calibration of top pressure transducer at RT from 0-10 MPa
  - Pressurize top of specimen with He to 8.38 MPa at 300°C
  - Stabilize (pressure rose to 8.56 MPa over 15 min) at 300°C
  - Ramp temperature to burst in Ar; slow furnace cool

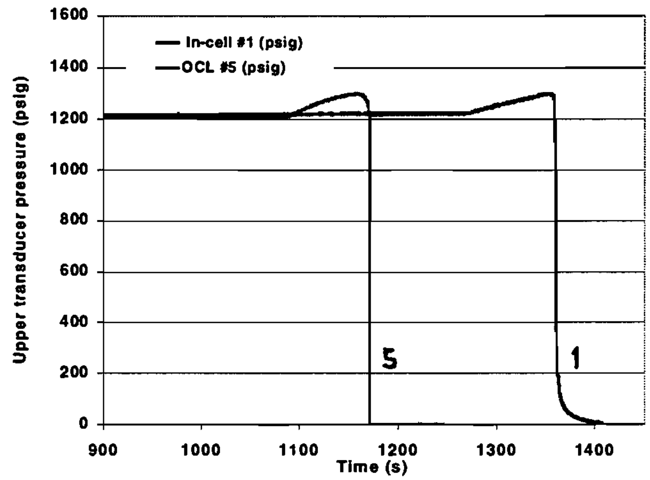
1st LOCA INTEGRAL TEST RESULTS LIMERICK HIGH-BURNUP BWR PHASE A (Cont'd)

out cell

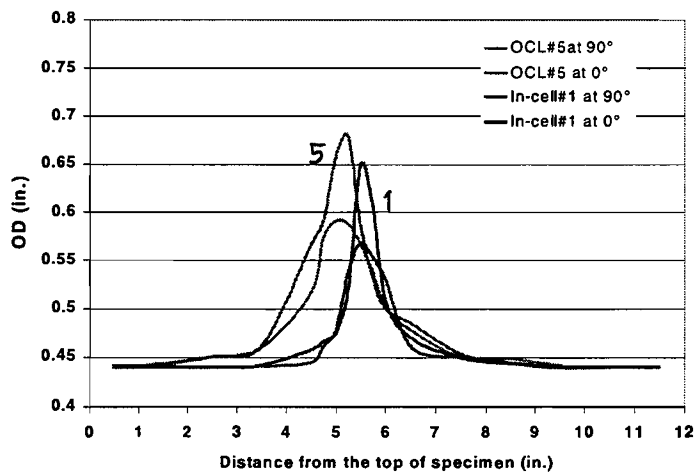
- **Burst Conditions for Phase A vs. OCT#5** Test
- **Peak ΔP = 8.95 MPa for both tests**
  - Burst ΔP=8.61 MPa at 755°C (vs. 8.26 MPa at 732°C for OCT#5)
  - Burst shape is oval (vs. dog-bone for OCT#5)
  - Burst length (≈12-13 mm) and max. opening (2-3 mm) for both
- **Balloon Characteristics for Phase A vs. OCT#5**
  - Average $\Delta D/D_o$ at burst center = 38% (vs. 44% for OCT#5)
  - Axial extent of balloon = 50 mm (vs. 100 mm for OCT#5)
  - Note: $\Delta T_\theta \approx$ 30°C (vs. ≈10°C for OCT#5)



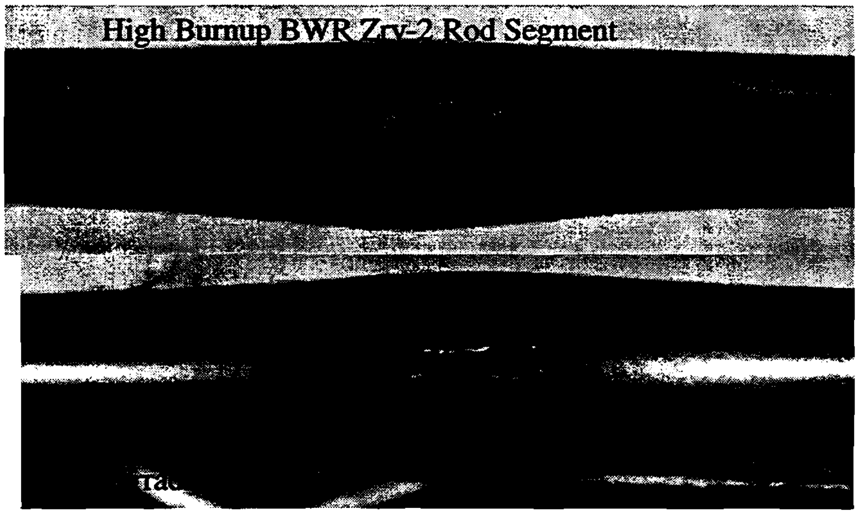
PRESSURE HISTORIES FOR IN-CELL TEST #1 AND OUT-OF-CELL TEST #5



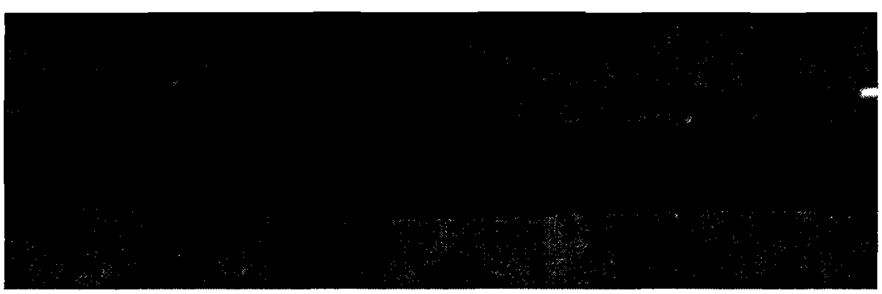
BALLOONING COMPARISON IN-CELL TEST #1 vs. OUT-OF-CELL TEST#5



BURST OPENING COMPARISON



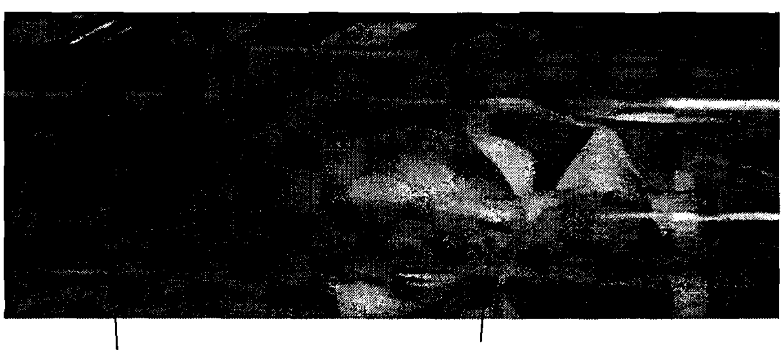
SIDE VIEW OF HIGH-BURNUP BWR ROD SEGMENT AFTER LOCA PHASE A TEST



FUEL BEHAVIOR DURING AND AFTER HIGH-BURNUP BWR LOCA TEST #1

- **Dark Deposit on Quartz Tube**
  - Black deposit on tube (will be gamma-scanned, Cs??)
  - Probably occurred during burst
  - Extends from burst region to about 50 mm above burst
- **Fuel Particle Fallout during Post-Test Handling**
  - Test train was moved from vertical position in furnace to horizontal position at a different workstation
  - Large number of small fuel particles (5.2 g) fell out of burst opening during rotation of specimen from vertical to horizontal and about longitudinal axis

FUEL DEPOSIT AND PARTICLES WITHIN QUARTZ TUBE

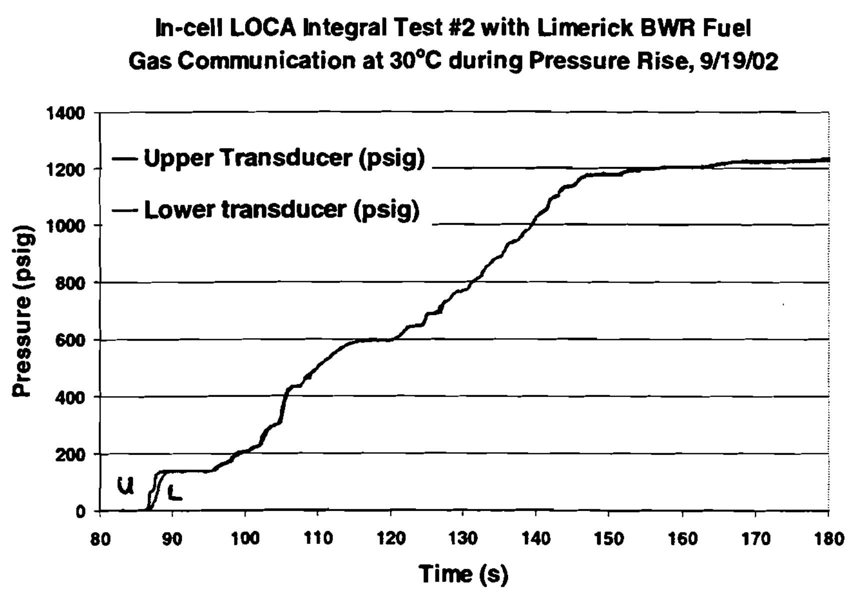


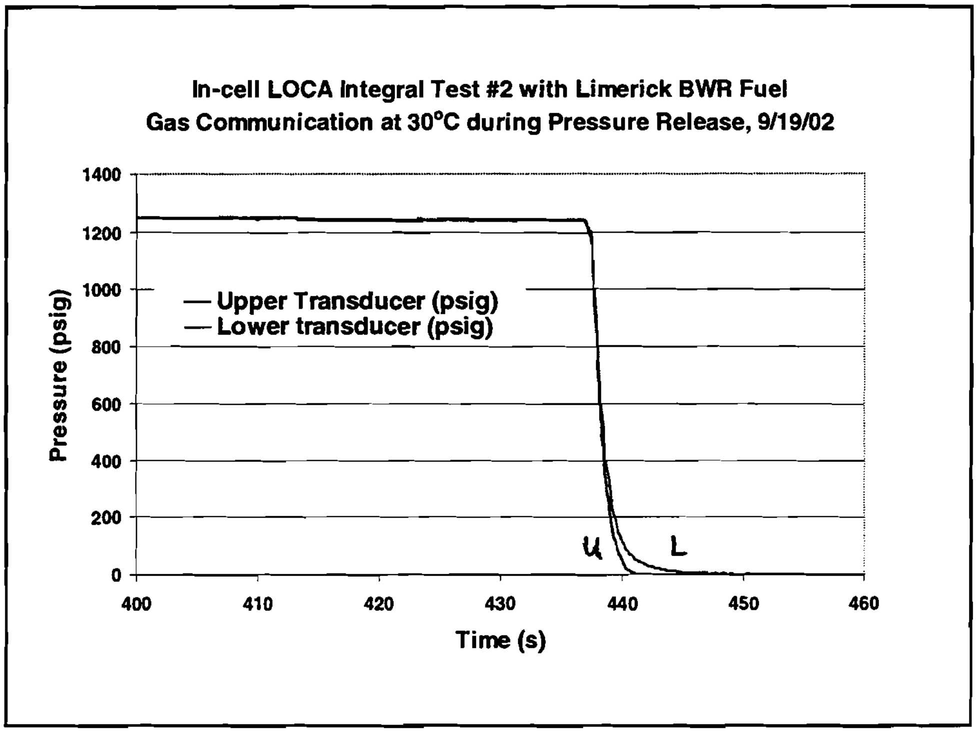
Black Deposit
Cs Compound??

Fuel Particles

2nd LOCA INTEGRAL TEST RESULTS HIGH-BURNUP BWR PHASE B

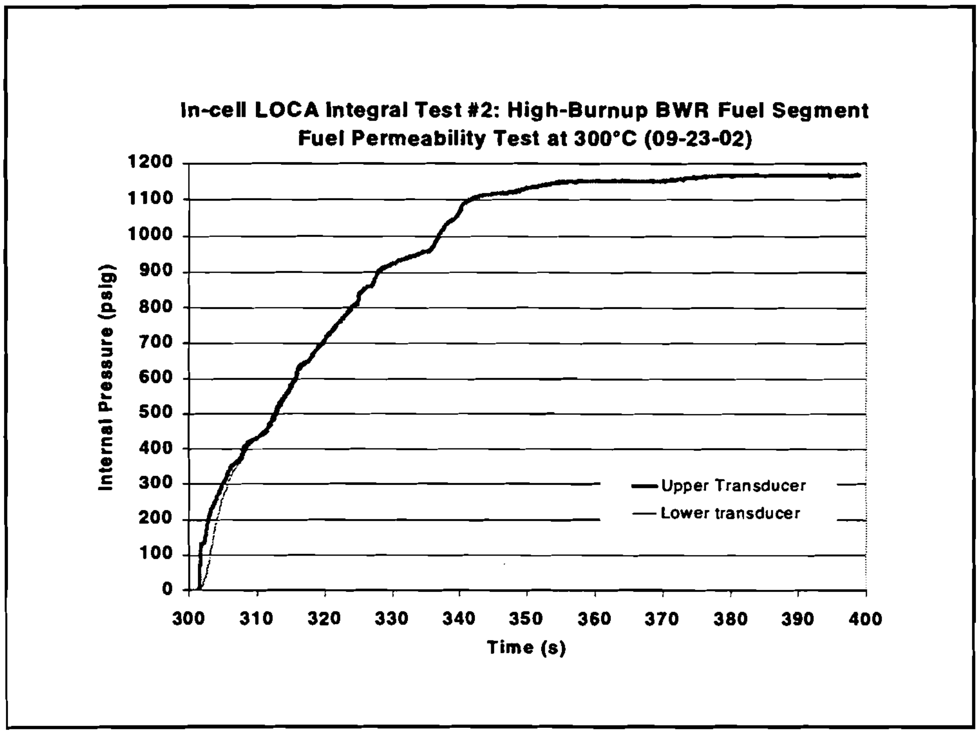
- **Permeability Results at 30°C**
  - Pressurization ramp at top of specimen to 8.7 MPa
    Excellent gas communication from 1 to 8.7 MPa
    Small axial pressure drop ($\Delta P_z \leq 0.5$ MPa) for 0-4s
  - Rapid pressure release at top of specimen (valve open)
    Lag in lower pressure response ($\Delta P_z \leq 0.6$ MPa)
    Slow release from bottom transducer from 2→0.1 MPa
  - Results are consistent with fuel microstructure
    Macrocracks; extensive microcracks in outer fuel zone
    Note: 20% fission gas release during irradiation

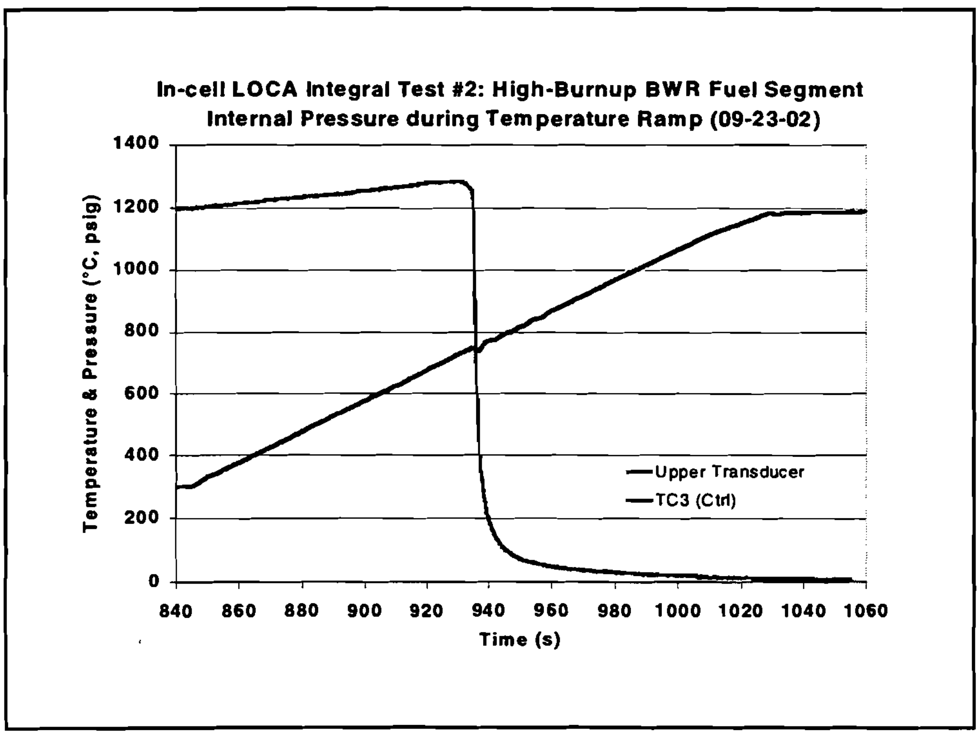




2^{nd} LOCA INTEGRAL TEST RESULTS HIGH-BURNUP BWR PHASE B (Cont'd)

- **Permeability Results at 300°C**
  - Pressurization ramp at top of specimen to 8.0 MPa
  - Excellent gas communication from 2 to 8 MPa
  - Some axial pressure drop ($\Delta P_z \leq 0.9$ MPa) for 0-4s
  - Pressure increases to 8.4 MPa during 300°C hold
- **Temperature Ramp to 1204°C**
  - Pressure peaks at 9.0 MPa at 728°C
  - Burst at 750°C and ≈8.4 MPa (1200 psig)
  - Rapid drop to 3.5 MPa; slow drop from 3 → 0.1 MPa

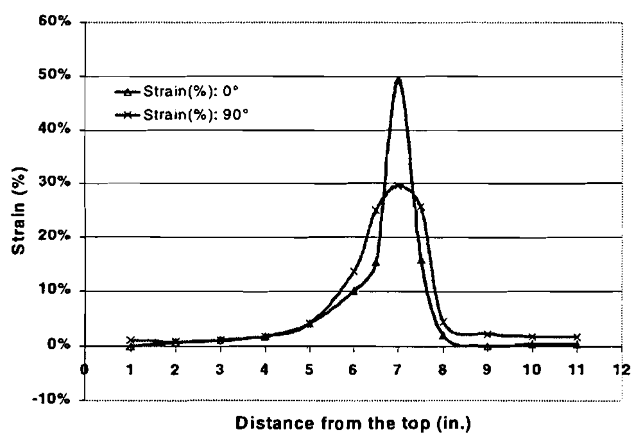

In-cell LOCA Integral Test #2: High-Burnup BWR Fuel Segment
Fuel Permeability Test at 300°C (09-23-02)
Internal Pressure (psig)
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
300
310
320
330
340
350
360
370
380
390
400
Time (s)
Upper Transducer
Lower transducer


In-cell LOCA Integral Test #2: High-Burnup BWR Fuel Segment
Internal Pressure during Temperature Ramp (09-23-02)
Temperature & Pressure (°C, psig)
1400
1200
1000
800
600
400
200
0
840
860
880
900
920
940
960
980
1000
1020
1040
1060
Time (s)
Upper Transducer
TC3 (Ctrl)

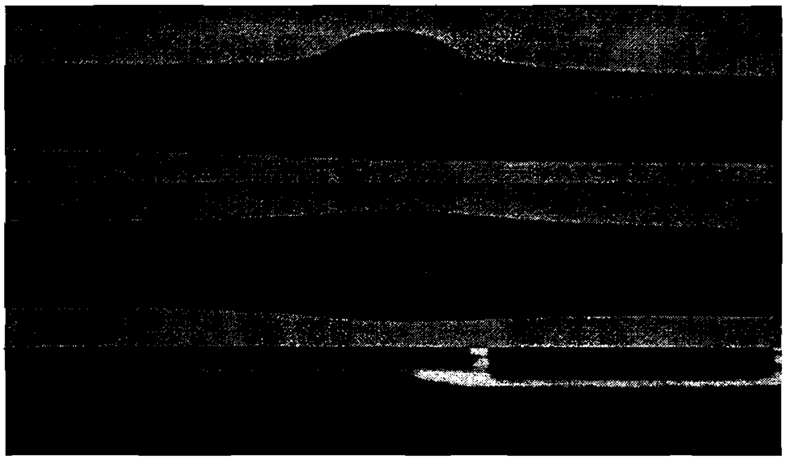
2nd LOCA INTEGRAL TEST RESULTS HIGH-BURNUP BWR PHASE B (Cont'd)

- **Ballooning**
  - Axial extent ≈100 mm, peak at 25 mm below midplane
  - Max. $\Delta D/Do$ = 49%; max. average strain = 39%
  - Uncorrected for oxide thickness
- **Burst Opening**
  - Oval-shaped
  - 14-mm long; 3.5-mm maximum width

2ND LOCA INTEGRAL TEST WITH HIGH-BURNUP BWR ROD: PROFILOMETRY



LOCA INTEGRAL TEST (PHASE B) HIGH-BURNUP BWR BALLOON & BURST



FUEL BEHAVIOR DURING AND AFTER HIGH-BURNUP BWR LOCA TEST #2

- **Dark Deposit on Quartz Tube (same as in Test 1)**
  - Black deposit on tube (will be gamma-scanned, Cs??)
  - Probably occurred during burst
- **Fuel Particle Fallout during Post-Test Handling**
  - Fuel particles (<1 g) ejected during test were collected
  - Bottom of test train was capped to trap fuel fallout during transfer and handling
  - Total of 4 grams of fuel were collected

LOCA INTEGRAL TEST (PHASE B)
HIGH-BURNUP BWR FUEL PARTICLES

Fuel Particles (4 g)
≈15% Released
during Test;
≈85% Released
during Transfer

30×30 mm Jar
Cross-section

NEAR TERM LOCA WORK

- **Verify Specimen Preparation Techniques**
  - Six-inch "practice"sample and bottom of Test #1 sample
  - Metallographic examinations
- **Determine Composition of Dark Deposit on Quartz Tube (Gamma Scanning)**
- **Determine Max. ECR and H Distribution for 5-min. Tests (in-cell & out-of-cell) at 1204°C**
- **Move Quench System In Cell and Run Full LOCA Sequence (11-02)**