LOCA Equivalent Clad Reacted (ECR) Criteria

NRC/Westinghouse Meeting
Rockville, MD
August 17, 2005

LOCA Equivalent Clad Reacted (ECR) Criteria

ACRS subcommittee meeting on Reactor Fuels met on July 27, 2005

Ralph Meyer of NRC/RES presented a summary of the Argonne National Labs (ANL) program and their proposed LOCA criteria which was:

- ECR including both operational corrosion and transient oxidation < 17% with the transient oxidation calculated by Cathcart-Pawel (C-P)
- Total time for transient < 2700 sec (45 minutes)
- Peak Cladding temperature (PCT) < 2200 ^{0}F

This was unexpected since ANL had earlier issued an embrittlement correlation which was more phenomenological based

LOCA Equivalent Clad Reacted (ECR) Criteria

ANL presented the results of LOCA simulation testing performed at ANL

ANL program plan is to wrap-up the program following the completion of the irradiated ZIRLO™ and M5 tubing tests and the integral test of the HBR rod segments

EPRI and ANATECH both made presentations which claimed that there are still unanswered questions from the ANL testing and that the ANL results do not correlate with the results from other programs

LOCA Equivalent Clad Reacted (ECR) Criteria

FANP presented joint EDF/CES/FANP data which showed much greater reduction in post test ductility as a function of hydrogen compared to ANL data

Rationale for this appears to be the direct quench used in the French tests

Apparently this lock in the high temperature morphology in the β-layer, where slower cooling provided time for segregations of oxygen and hydrogen in the β-layer providing greater ductility

NRC/RES stated that EPRI was not doing enough work to develop new limits, but only enough to verify the existing interpretation of the LOCA limits for high burnup fuel

LOCA Equivalent Clad Reacted (ECR) Criteria

ACRS question related to "What is the impact of up and down temperatures variations during high temperature oxidation on ECR and post test ductility?

Reactivity Initiated Accident (RIA) Criteria

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Reactivity Initiated Accident (RIA) Criteria

NRC/RES presented a summary of their analysis of RIA tests and his proposed criteria. The most limiting aspect of the proposed criteria was the collapse of the coolability limit onto a low cladding failure limit. The cladding failure limit was given as a function of maximum fuel rod corrosion

EPRI and ANATECH presented the industry proposed criteria and the methods used to produce it

Westinghouse presented summary of comments on proposed RIA criteria along with a sample analysis to demonstrate how limited the volume was of the core close to peak power and how unlikely conditions of high rod worth were

Reactivity Initiated Accident (RIA) Criteria

ACRS indicated that they thought the NRC/RES proposed criteria was very conservative

ACRS indicated that although they thought separate coolability and clad failure limits were reasonable, they were skeptical that the onset of fuel melt was the best limit and a lower one might be easier to justify

Reactivity Initiated Accident (RIA) Criteria

ACRS was skeptical of the methods ANATECH used to treat cladding test data to develop critical strain energy density (CSED) relationships as a function of temperature and oxide thickness.

ACRS though the method non-conservative, and the overall method using FALCON too obscure to easily understand

ACRS was skeptical that oxide spalling could be ruled out and thought that tests with spalled cladding should be included in developing the limit as was done by NRC/RES

AGENDA
Westinghouse Semi-Annual Fuel Performance Update
August 16, 2005
Westinghouse Office
Rockville, MD

Tuesday, Aug 16	BWR Fuel Update	
8:00 – 8:10 am	Welcome	[ ]a,c
8:10 – 8:20 am	BWR Organization & Overview	[ ]a,c
8:20 – 10:15 am	Fuel Performance Update	[ ]a,c
10:15 – 10:30 am	Break	
10:30 – 11:00 am	Application of European Experience Base to U.S. Plants	[ ]a,c
11:00 – 11:20 am	[ ]a,c Nuclear Benchmark Results	[ ]a,c
11:20 – 11:40 am	Westinghouse BWR Short & Long Term Interactions with the USNRC	[ ]a,c
11:40 – 11:50 am	Wrap-up	
11:50 – 1:00 pm	Lunch/Informal Discussion between NRC, Customers & Westinghouse	
	PWR Fuel Update	
1:00 – 3:00 pm	Fuel Performance Update	[ ]a,c
3:00 - 3:15 pm	Break	
3:15 – 3:35 pm	Oden CHF Loop Update	[ ]a,c
3:35 – 4:00 pm	[ ]a,c Creep/Growth Test	[ ]a,c
4:00 – 4:30 pm	Reactivity Insertion Accident Feedback	[ ]a,c
4:30 – 4:50 pm	Update on APA Development Activities	[ ]a,c
4:50 – 5:00 pm	Wrap-up	

DRESS IS BUSINESS CASUAL

Westinghouse Presentation
on
Westinghouse Fuel Performance Update Meeting
BWR / PWR Fuel Update
(Slide Presentation of August 16, 2005)

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Westinghouse

A BNFL Group company

BWR Fuel Performance Update

NRC/Westinghouse Meeting
Rockville, MD
August 2005

Outline of Presentation

1. Statistics
   - Deliveries
   - Burnup
   - Failures
2. In-Reactor Performance
   - Pellet
   - Cladding
     - Liner
     - Outer Component
   - Channel
3. Development
4. Secondary Fuel Degradation Program
5. Fuel Performance Program
6. Summary

BWR Fuel Deliveries

a, c

BWR Fuel Burnup Experience, 2004

a, c

All Fuel Failures in Westinghouse 10x10 fuel

a, c

Primary Fuel Failures
Westinghouse BWR Experience

a, c

Age Distribution of Debris Fretting Failures

a, c

Axial Distribution of Debris Fretting Failures

a, c

Outline of Presentation

❶ Statistics
- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance
- Pellet
- Cladding
  - Liner
  - Outer Component
- Channel

❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

What is ADOPT?

a, c

Technical Objectives

a, c

Development of ADOPT

a, c

ADOPT Deliveries

a, c

ADOPT – Next Steps

a, c

Outline of Presentation

❶ Statistics
- Deliveries
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- Failures

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- Pellet
- Cladding
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❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

Liner Cladding – Background

a, c

Three Ramp Tests Performed in 2004-2005

a, c

PIE after Stair-Case Ramp of 62 MWd/kgU Segment

a, c

PIE after Stair-Case Ramp of 62 MWd/kgU Segment

a, c

Secondary degradation

Mono and Liner Fuel by Degradation Type

a, c

Liner Cladding – Summary

a, c

Outline of Presentation

1. Statistics
   - Deliveries
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2. In-Reactor Performance
   - Pellet
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3. Development
4. Secondary Fuel Degradation Program
5. Fuel Performance Program
6. Summary

Westinghouse BWR Cladding

a, c

Cladding Outer Component

[ ] a,b,c

- Improve high burnup performance
  - ➢Development of LK3
- Verify high burnup performance
  - ➢Pool-side and hot-cell examinations of leading fuel rods
    - –Corrosion
    - –Rod growth
    - –Hydriding

LK3 - Achieved Burnup

a, c

Two-Life Rods
Rod positions

a, c

Two-Life Rods
Power history

a, b, c

Cladding Corrosion

Midspan Oxide Thickness by Cladding Type

[ ] a, b, c

Rod Growth
By Cladding Type

[ ] a, b, c

Two-Life Rods
Planned hot-cell PIE

a, c

Clad Hydrogen Pick-Up

a, c

5-6 Cycle LK3 and 6 Cycle LK2 & LK2+ Secondary Phase Particle Size Distribution

a, c

Cladding Outer Component – Summary

a, c

Outline of Presentation

❶ Statistics
- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance
- Pellet
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  - Liner
  - Outer Component
- Channel

❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

BWR Fuel Performance
Channel Material Evolution

a, c

Channel Corrosion

a, b, c

Hydrogen Pick-Up

a, b, c

Hydrogen Pick-Up
Outer Zry-2 Channel at 45 MWd/kg U

a, b, c

Channel Bow & Irradiation Induced Growth

a, c

Channel Growth

a, b, c

Channel Bow in Symmetric Lattice

a, b, c

Channel – Summary

a, c

Outline of Presentation

❶ Statistics
- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance
- Pellet
- Cladding
  - Liner
  - Outer Component
- Channel

❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

Alternative Cladding Alloys Modified Zircaloy-2

a, c

Alternative Cladding Alloys
ZIRLO™

a, c

Alternative Cladding Alloys ZIRLO™

a, c

ZIRLO^{TM} Channels

a, c

Outline of Presentation

❶ Statistics
- Deliveries
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- Pellet
- Cladding
  - Liner
  - Outer Component
- Channel

❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

Background & Time Schedule

a, c

Outline of Presentation

❶ Statistics
- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance
- Pellet
- Cladding
  - Liner
  - Outer Component
- Channel

❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

Fuel Performance Program
Aims and Goals

a, c

Fuel Performance Program
Support to other research programs

a, c

Fuel Performance Program
Summary of Rods Used for PIE

a, b, c

Fuel Performance Program Continuation

a, c

Outline of Presentation

❶ Statistics
- Deliveries
- Burnup
- Failures

❷ In-Reactor Performance
- Pellet
- Cladding
  - Liner
  - Outer Component
- Channel

❸ Development

❹ Secondary Fuel Degradation Program

❺ Fuel Performance Program

❻ Summary

Summary

a, c

Summary, cont.

a, c

Application of European Experience Base to U.S. Plants

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Topics – Application of European Experience

- Background
- Overall Approach
- Application Confirmation
  - Thermal-Hydraulic
  - Nuclear
  - Mechanical
  - Dynamic
    - AOOs, CRDA, Stability, LOCA

Background

- Recent 10 Year Reload Experience

a, c

- Illustrates need for robust, flexible, and portable methods

Overall Approach - Processes

- Flexible and Robust methods versus validation of tuned models to each application
- Phenomenological methods applicable to intended applications
- Generalized methodology applicable to intended applications
- Formulation in terms of analyses input which capturing plant-specific requirements
- Application of performance data in an applicable manner
- Application of methods based on test data within data range or conservative

Overall Approach – Fuel/Plant Data

- Major data transfer to support application of generic methods to a specific plant – Steady-state
    - Mechanical – core and legacy fuel data
    - T/H – core, legacy fuel hydraulic and CPR data, core heat balance
    - Nuclear – legacy fuel description, previous cycle core follow, LPRM system, etc.

Overall Approach – Fuel/Plant Data

- Major data transfer to support dynamic applications
  - Plant geometries and volumes, water levels and trips/alarms, Safety/relief/isolation/bypass/control valve data (e.g. pressures, timing, tolerances), recirculation/jet pump data, scram and RPS data, LPRM/APRM/RBM/OPRM data, containment, suppression pool, drywell data, safety systems (LPCS, HPCS, HPCI, LPCI, ADS, etc) description and logic, seismic data

Application Confirmation

- Steady-state Thermal-hydraulic modeling
  - Confirmation that core pressure drops and flow splits obtained from utility accurately predicted in Westinghouse T/H models
  - T/H Compatibility evaluation for mixed Westinghouse fuel/Legacy fuel cores
  - T/H models embedded in 3D core simulator (POLCA7)

Application Confirmation

- Nuclear Model Verification
    - Analyses of cycles prior to initial loading of Westinghouse fuel to confirm:
        - Acceptable hot reactivity performance (keff)
        - Acceptable cold reactivity predictions
        - Acceptable power distribution predictions (comparison with TIP data)

Application Confirmation

- Assembly/Rod performance verification
  - Application of assembly and fuel rod corrosion, growth, etc.

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[ ]

Application Verification

a, b, c

Application Verification

a, b, c

Application Verification

a, b, c

Application Verification

a, b, c

Application Confirmation

- Fast AOO Models (BISON)

a, c

[ ]

Application Confirmation

a, b, c

[ ]

Application Confirmation

a, b, c

Application Confirmation

- CRDA Models (RAMONA)

a, c

Application Confirmation

- Stability Models (RAMONA)

a, c

Application Confirmation

a, b, c

Application Confirmation

- LOCA Models (GOBLIN/DRAGON/CHACHA)

a, c

[ ]a,c Nuclear Design Benchmark

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Introduction

Overview:

BWR Nuclear Design Code System

Nuclear Benchmark:

[ ]a,c Cores

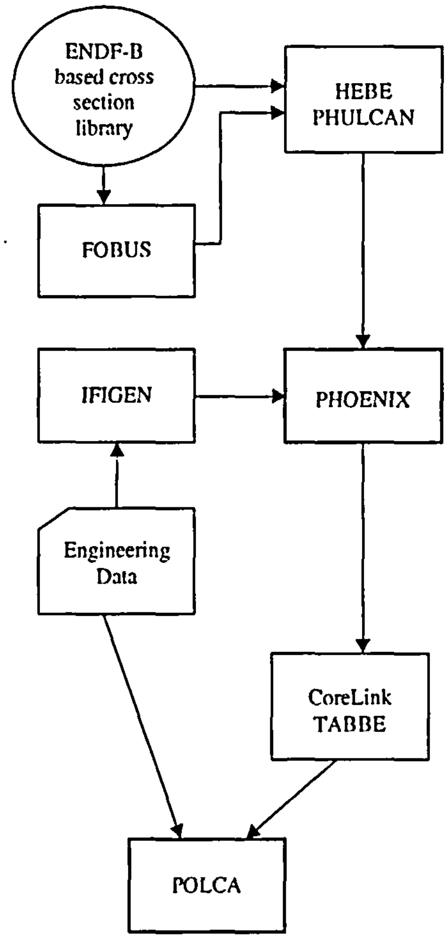
Summary and Conclusions

Overview

- Neutronic Codes for BWR steady State Nuclear Design
  - PHOENIX4: 2D multi-group transport theory code used to calculate lattice physics constants
  - POLCA7: 2-group nodal code used for 3-dimensional simulation of nuclear and thermal-hydraulic conditions in BWR cores

Overview

Figure 1.1: Westinghouse Code System for BWR Nuclear Design and Analysis



FOBUS:	Monte Carlo transport theory code for burnable absorber cross section generation
HEBE & PHULCAN:	Library processing codes for nuclear cross section data and depletion chains
IFIGEN:	Input data generation for PHOENIX
PHOENIX:	Two-dimensional multi-group transport theory code for lattice physics constants
CoreLink & TABBE:	PHOENIX output processing codes for nuclear data generation
POLCA:	Three-dimensional, two-group nodal code for steady-state reactor core simulation

Overview

- Approved Topical Reports for PHOENIX/POLCA :
  - BR 91-402: ABB Atom Nuclear Design and Analysis Programs for Boiling Water Reactors: Programs Description and Qualification, May 1991
  - **CENPD-390-P-A:** The Advanced PHOENIX and POLCA7 Codes for Nuclear Design of Boiling Water Reactors, December 2000.

Overview

a, c

Overview

a, c

Overview

a, c

Overview

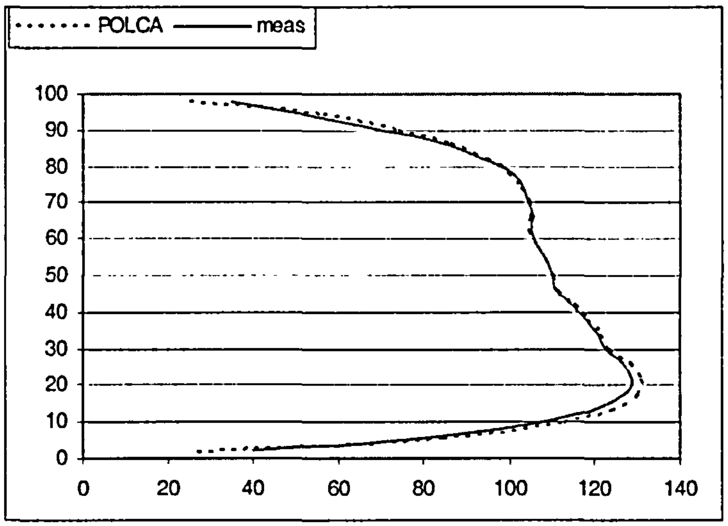
- The hot target k_{eff} determined from the core follow is used for the design cycle to predict
  - ✦ Cycle length
  - ✦ Number of fresh assemblies
  - ✦ Enrichment level
  - ✦ Hot excess reactivity
  - ✦ Control rod patterns

Overview

- The cold target k_{eff} determined from the past cycles measurement is used for the design cycle to predict
  - Cold shutdown margin
  - Burnable absorber design
  - Standby Liquid Control System (SLCS) Verification
  - Startup prediction

Overview

- TIP and other nuclear benchmark comparisons are used to verify how well the axial and radial power distributions are computed
  - ✦ Used as a guide in establishing the design thermal margins



Overview

- In Summary, the Nuclear Benchmark
  - ✦ gives confidence that BWR core is correctly modeled
  - ✦ forms the basis for cycle nuclear design
  - ✦ provides starting point for licensing analyses

Overview
PHOENIX4/POLCA7 Codes Application

PHOENIX4/POLCA7 have been used in:

Plant Name	Reactor Class	Size (Bundle)
[ ]a,c	BWR-6	624
[ ]a,c	BWR-6	648
[ ]a,c	BWR-4	764
[ ]a,c	BWR-3	724
	KWU Designs	
	Westinghouse-Atom Designs	

Benchmark Results

[ ]^{a,c} Nuclear Benchmark

- Hot k_{eff} results from core follow
- Cold critical measurements k_{eff} results
- TIP comparison results
  - ✦Nodal RMS
  - ✦Radial RMS

Benchmark Results: General Information

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Core Reactivity

a, c

Benchmark Results: Cold Critical

a, c

Benchmark Results: Cold Critical

a, c

Benchmark Results: Cold Critical

a, c

Benchmark Results: TIP Comparison

a, c

Benchmark Results : TIP Comparison

a, c

Benchmark Results : TIP Comparison

a, c

Benchmark Results : TIP Comparison

a, c

Summary and Conclusions

- Hot k_{eff} Core Follow Results
  - Post Uprate Cycles consistent and stable, good confidence on hot target k_{eff} curves to be selected for designs
- Cold Critical Results
  - Few results, but considerably consistent
- TIP Comparison Results
  - Larger differences are as expected in the first few cycles of simulation
  - Large variability on RMS differences is consistent with neutron TIP experience

USNRC – Westinghouse BWR Short and Long Term Interactions with the USNRC

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Topics – USNRC/Westinghouse Strategical Discussion

- Planned Submittals
- Application Implementations
- Exelon Related Issues
- Open

Planned Submittals

a, c

Planned Submittals

a, c

Application Implementations

a, c

Exelon Related Issues

a, c

- Safety Limit Submittal
  - –Format or guideline requirements from the USNRC?

Open

- Any advisement from USNRC on Sump Screen LOCA issue specific for BWR application
  - Westinghouse is keenly aware of PWR concerns and some impending BWR concerns, what would the USNRC like for Westinghouse to be doing soon or to be prepared for concerning BWR?
- Other?

PWR Fuel Performance Update

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Outline

- Fuel Reliability Overview
- 17 OFA Root Cause Investigation Update and recent PIE results
- Future PIE & Hot Cell Plans
- Status of LTA Programs
- RCCA Update

Westinghouse PWR Fuel Reliability

a, c

Leakage Mechanisms in Westinghouse PWR Fuel: 2004

a, c

Leakage Mechanisms in Westinghouse PWR Fuel: 2005 YTD (May)

a, c

Fuel Performance Trend

a, c

2004 – 2005 Leaking Rods by Major Product Family

a, c

Grid-Rod Fretting Solutions Being Implemented

a, c

17x17 RFA/RFA-2 Experience

a, c

Current Status of RFA

a, c

CE Improved Designs Implementation

a, c

Summary

a, c

Outline

- Fuel Reliability Overview
- 17 OFA Root Cause Investigation Update and recent PIE results
- Future PIE & Hot Cell Plans
- Status of LTA Programs
- RCCA Update

17 OFA Fuel Performance

a, c

[ ]

Recent 17 OFA PIE Results - [ ]a,c

a, c

Recent 17 OFA PIE Results - [ ][a,c]

a, c

17x17 OFA Leaking Fuel Since 2002

a, c

Proactive Approach to Identify Leakage Mechanisms in 17 OFA Fuel

a, c

Potential Leakage Mechanisms

a, c

Summary of Actions to Address Most Likely OFA Leakage Mechanisms

a, c

Revised pellet chip criteria status

a, c

Outline

- Fuel Reliability Overview
- 17 OFA Root Cause Investigation Update and recent PIE results
- Future PIE & Hot Cell Plans
- Status of LTA Programs
- RCCA Update

Planned Inspections for 17x17 OFA

a, c

Leaking Rod Hot Cell Program

a, c

Outline

- Fuel Reliability Overview
- 17 OFA Root Cause Investigation Update and recent PIE results
- Future PIE & Hot Cell Plans
- Status of LTA Programs
- RCCA Update

Westinghouse High Burnup ZIRLO™ LTA Summary

a, c

Other Test Programs

a, c

Status of Optimized ZIRLO™ LTA Programs

a, c

Outline

- Fuel Reliability Overview
- 17 OFA Root Cause Investigation Update and recent PIE results
- Future PIE & Hot Cell Plans
- Status of LTA Programs
- RCCA Update

[ ]a,c - Separated RCCA Rodlet

- One of twenty-four rodlets separated from a single EP-RCCA
  - rodlet was located in the thimble tube of host fuel assembly
  - event occurred during Cycle 10 and was discovered at EOC-10
  - no affect on RCCA insertion during plant shutdown at EOC

- RCCA was manufactured prior to 1995

Root Cause Status

- Final Westinghouse CARB review performed in January, 2005
- Additional corrective actions completed 2005 YTD
  - [ ] a, c
  - [ ] a, c
  - [ ] a, c
- Longer-term corrective actions
  - Evaluate current RCCA design

[ ]^{a,c} – Possible Separated RCCA Rodlet(s)

- No further RCCA performance issues

- Poolside PIE scheduled for the next outage in Fall 2005

[ ]a,c Incomplete Rod Insertion

- [ ]a,c has a non-standard core comprising non-Westinghouse 14x14 fuel design with an 8 ft active length
- The RCCAs are the Westinghouse EP design and are close to their design life of 12 EPFY
- A single RCCA stuck in the fuel assembly dashpot
- Swelling of the absorber was the most probable cause
- Westinghouse is working to ensure this experience is integrated into its RCCA operating guidelines

Summary

- Fuel performance has improved in some areas, but deteriorated in others
- Programs and action plans are in place and being implemented
- Fuel designs susceptible to grid to rod fretting being replaced with improved products that are performing well
- Most pressing issue at this time is resolution of 17X17 OFA leakers
  - [ ] a, c
  - [ ] a, c

Oden CHF Loop Update

NRC/Westinghouse Meeting
Rockville, MD
August 2005

Outline

- Overview of Oden Facility
- Status of Loop Design/Construction
- Test Loop Description and Characteristics
- Qualification Test Plan
- Schedule

Västerås Fuel T/H-Testing Facility

a, c

Loop Design & Construction (1) - schematic

a, c

Loop Design & Construction (2)

a, c

Loop Design & Construction (3)

a, c

Loop Description, Characteristics

[ ] a, c

Qualification Test Plan (1) - Objectives

1. Characterize Oden loop response to changing conditions
2. Demonstrate Oden repeatability
3. Benchmark Oden data to HTRF data
4. Develop experience base

Qualification Test Plan (2) - Overview

a, c

Qualification Test Plan (3) – Current Test Geometry Selection

[ ] a, c

Schedule

a, c

[ ]^{a,c} Creep/Growth Test

Westinghouse/NRC Meeting
Rockville, MD
August 16, 2005

Presentation Outline

- Test overview and status
- Optimized ZIRLO™ irradiation growth and creep
- Determined sample stresses
- Tensile and compressive stress irradiation creep
- Completion of NRC commitments

Test Status

- Test assembly A1
  - Completed irradiation (1 cycle)
  - Completed PIE/NDE measurements
  - Evaluation of NDE data is in-progress
    - Optimized ZIRLO™ $\Delta D/D_0$ data are available
    - ZIRLO™ $\Delta D/D_0$ tensile and compressive stress data are available
  - Destructive examination may be performed after the NDE evaluation is complete

- Irradiation is continuing for test assemblies A2, A3, A4 & A5

Irradiation Schedule

a, c

Data Acquisition Measurement Methodology

a, b, c

Irradiation Growth and Irradiation Creep Data (1/2)

- Irradiation growth measurements were made on tube samples open to coolant water flow
  - Ensured a stress-free condition

- Irradiation creep measurements were made on internally Helium pressurized tube samples
  - Irradiation creep, $\Delta D/D_0(ic)$, was calculated from the total diameter change, $\Delta D/D_0(total)$, and the irradiation growth, $\Delta D/D_0(ig)$, according to, $\Delta D/D_0(ic) = \Delta D/D_0(total) - \Delta D/D_0(ig)$

Irradiation Growth and Irradiation Creep Data (2/2)

- OD accuracy is outstanding
  - [ ]a, b, c laser OD measurements on each sample
  - 95% confidence interval is [ ]a, b, c
  - Pre and post-test measurements were performed with the same facility – minimizes measurement errors

- Each data point in the following graphs represents one sample

ZIRLO™ and Optimized ZIRLO™ Irradiation Growth

a, b, c

ZIRLO™ and Optimized ZIRLO™ Irradiation Creep

a, b, c

ZIRLO™ and Optimized ZIRLO™ Results

a, b, c

Creep and Growth Sample Design

[ ] a, b, c

- Evaluated the sample temperatures, the internal Helium gas pressure and hoop stress using the actual gamma-heat rate
  - All test parameters are based on experimental measurements when the dosimetry analysis is finalized ($\Delta D/D_o$, T, σ_Θ & Φt)

Creep and Growth Sample Design

a, b, c

Determination of the Gamma-Heat Rate & Hoop Stress

- Performed parametric calculations of the sample hoop stress as a function of the gamma heat rate

[ ]a, b, c

- Actual gamma-heat rate is given by the maximum regression R^2 coefficient associated with $\Delta D/D_o$(ic) versus hoop stress
  - Corresponds to the minimum deviation of the data from the regression line
  - (Perfect regression fit is associated with an R^2 coefficient of 1.0)

Hoop Stress Variation with the Gamma-Heat Rate

a, b, c

Regression R^2 Coefficient Versus Gamma-Heat Rate

a, b, c

Irradiation Creep versus Hoop Stress

a, b, c

Tension and Compression Irradiation Creep Results

- Gamma-heat rate is [ ][a, b, c]
- Replicate sample-to-sample consistency is excellent
- Irradiation creep is the same in tension and compression for Westinghouse ZIRLO™ and Optimized ZIRLO™
  - $\Delta D/D_o$(ic) versus σ_θ is [ ][a, b, c]
  - [ ][a, b, c]
  - [ ][a, b, c] hoop stresses: [ ][a, b, c] compression and [ ][a, b, c] tension

Completion of NRC Commitments

- Completed PAD 4.0 SER commitment in Section 2.1 to initiate a clad irradiation growth and creep test to provide a more accurate measurement of irradiation creep under tension and compression stresses and share the data with the NRC

- Completed Optimized ZIRLO™ SER commitment in Section 5.0, Item 7 a & b to report the [ ]a,c Creep/Growth Optimized ZIRLO™ data and show Optimized ZIRLO™ irradiation creep is consistent with ZIRLO™

- Confirmed that tension and compression irradiation creep of Westinghouse ZIRLO™ and Optimized ZIRLO™ are equal
  - No impact on Westinghouse rod pressure analysis (Optimized ZIRLO™ SER Section 5.0, Item 7)

Conclusions

- Different processing methods such as PRXA or SRA may be used to fabricate Optimized ZIRLO™ with the same irradiation creep as ZIRLO™
  - [ ]a,c data confirms creep behaviour for Optimized ZIRLO™
- Irradiation creep is the same in tension and compression for Westinghouse ZIRLO™ and Optimized ZIRLO™
  - [ ]a,c data confirms the Westinghouse model for tensile creep
- Westinghouse commitments to the NRC concerning the [ ]a,c Creep/Growth test are complete

Future Data Currently In-Process

- [

  ][a, b, c]

- Beneficial effect of hydrogen on reducing irradiation growth and creep is under evaluation
  - Important for dry storage

Reactivity Insertion Accident Feedback

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Status

- Tests in '90s indicated need to reexamine the limit for reactivity insertion accidents (RIA)

- Using more accurate 3-D analyses provide significant margin compared to older 1-D based analyses
  - So no safety concern in operating plants

- EPRI, representing the industry, submitted report recommending new limit

- Westinghouse submitted 3-D rod ejection methodology for review and received SER

- NRC-NRR rejected EPRI report and is proposing new limits

Driver

- Would like to close out this issue
  - Creates uncertainty in licensing space
  - Is diverting resources
  - Permit licensing of high burnup fuel

- But with acceptable limits

NRC-NRR Proposed Criteria

Select one of the following:

1. Ejected rod worth of $2.20 or less with oxide thickness of 70 microns, or $1.70 if greater oxide

2. Reactivity excursion should not exceed cladding failure threshold curve in RES RIL-0401 Figure 1

3. Dose calculations for rods exceeding limit, plus coolability limit based on limiting pressure pulse resulting from fuel dispersal

RES RIL-0401 Figure 1

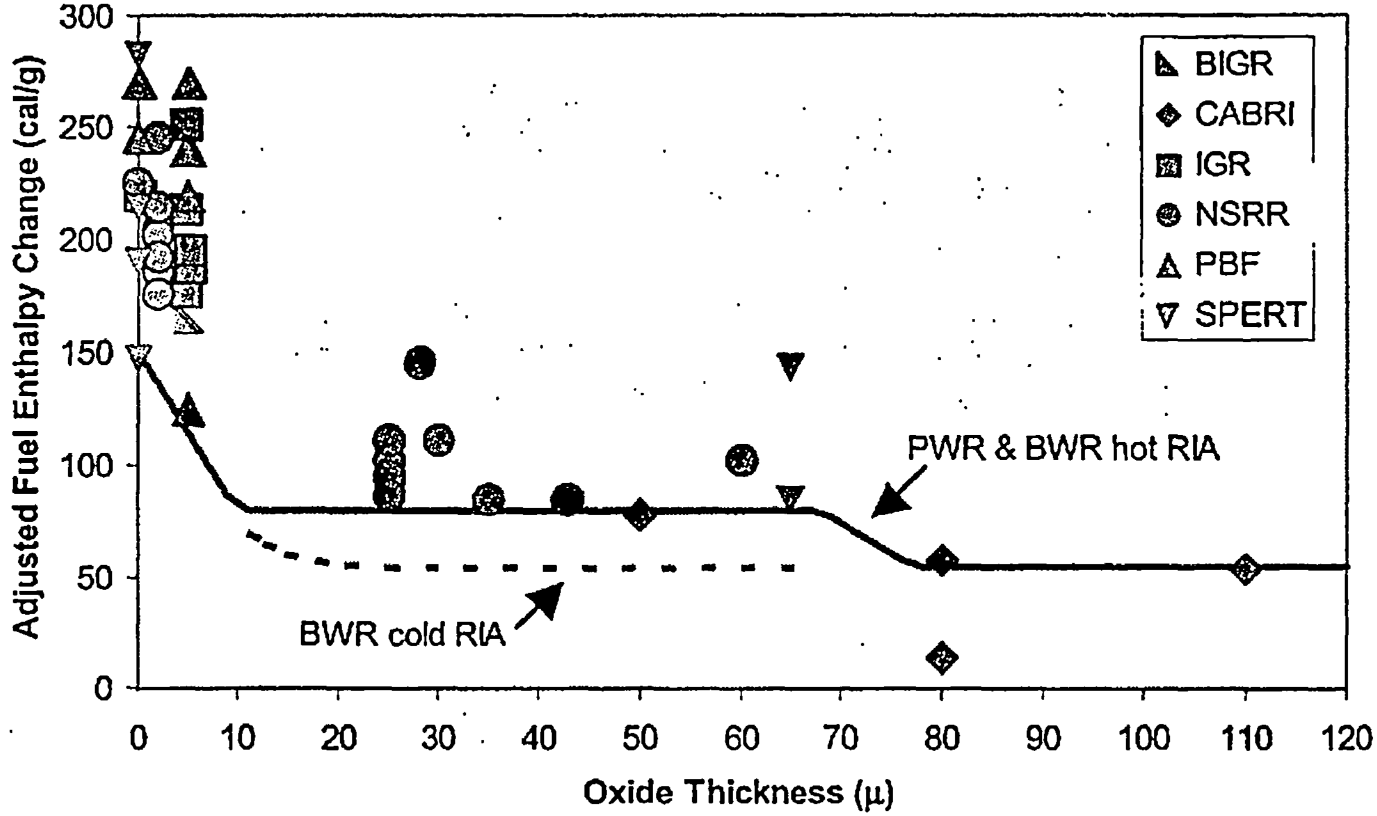


Figure 1. Cladding failure data with adjustments from the scaling analysis and lower-bound failure correlations. The lowest point at 80 microns of oxide thickness is for a test that has been discredited.

Ejected Rod Worth

- Highly dependent on the control rod pattern, loading pattern and bank insertion limits

- Based on most adverse allowable operational conditions

- Ejected rod worth has been increasing
  - Longer cycles (higher enrichments, axial burnup effects)
  - IRI concerns places feed fuel in most control rod locations

- Operation can impact ejected rod worth
  - Control rod shadowing
  - CIPS

- Occurrence of operational restrictions even with $ 2.15

Feedback on ER Worth Limit

- Would assume corrosion thickness is nominal value

- Rod ejection is a local event
  - Fuel enthalpy dependent on local peaking and core power
  - Significant local peaking only in area of ejected rod
  - Only neighborhood of ER sees significant fuel enthalpy increase
  - Fuel failure, if it occurred, would be local to ejected rod

- High oxide on high burnup rods
  - Low reactivity, lower peaking factor
  - Could restrict use of fuel from spent fuel pool

- Failure limit should reflect local fuel enthalpy

Fuel Failure Limit

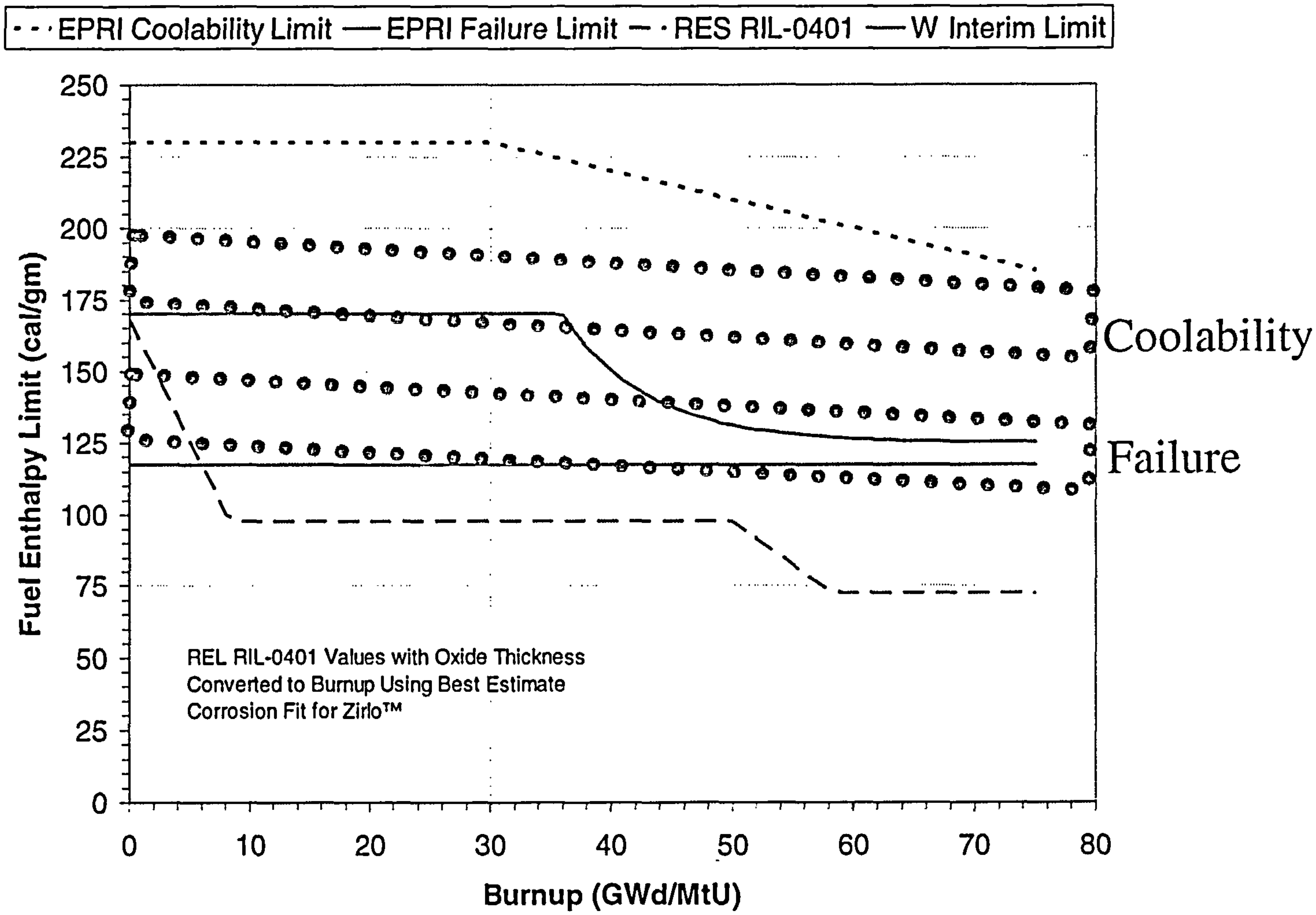
- Discussed in NRC/EPRI meeting on July 7, 2004 and ACRS Fuel Subcommittee meeting on July 28, 2005
  - Difference in failure limits due to different methods of adjustment of experimental data
  - Difference of opinion on need to include spalled cladding data

- Compromise position may include
  - Limit on fuel enthalpy <u>increase</u> of 100 -125 cal/gm
  - Some reduction with burnup to reflect corrosion impact
  - Separate penalty for spalled cladding, if expected (~25 cal/gm)

Coolability Limit

- Separate coolability limit is appropriate
- Limit can be set to prevent fuel melting
  - Pressure pulse would require much higher energy input
  - Dose calculation assumptions valid for whole transient
- Compromise position could include:
  - Initial limit of 175-200 cal/gm
  - Small reduction with burnup to reflect melting point change

Comparison of Potential Limits



Realistic Conservatism

- PWR rod ejection most limiting over very small and unlikely operational space
  - EOC, HZP, rods at insertion limit and critical
- Conservatism applied to peaking factors and ejected rod worth
- The EPRI curves would be appropriate limits
- Some compromise position may be acceptable

Summary

- Westinghouse believes NRC-NRR proposed limits should be revised
- Westinghouse will continue to work with EPRI to define industry position
- Westinghouse believes it is in the best interests of everyone to bring this issue to closure

Update on APA Development Activities

NRC/Westinghouse Meeting
Rockville, MD
August 2005

NEXUS Project - Description

a, c

ANC and Related Technology Development

a, c

ANC 9 / NEXUS Project – Status and Actions

a, c

ANC 9 Production Rollout Project – Description

a, c

ANC 9 Production Rollout Project– Status and Actions

a, c

ANC 9 / SPNOVA Merge Project – Status and Actions

a, c

Westinghouse

A BNFL Group company

AGENDA
Westinghouse Semi-Annual Fuel Performance Update
August 17, 2005
Westinghouse Office
Rockville, MD

<u>Wednesday, Aug 17</u>	<u>Licensing Review (Westinghouse & NRC)</u>	
8:00 – 9:00 am	Brief Overview of Westinghouse Organization	[ ]^{a,c}
	PWR/BWR Topicals and Schedule	[ ]^{a,c}
	[ ]^{a,c}	
9:00 – 11:50 am	General Licensing Concerns & Issues	All
	[ ]^{a,c}	
11:50 – noon	Wrap-up Next meeting	
noon – 1:00 pm	Lunch/Informal Discussion between NRC & Westinghouse	

<u>DRESS IS BUSINESS CASUAL</u>

Westinghouse Presentation
on
Westinghouse Fuel Performance Update Meeting
Management Licensing Overview
(Slide Presentation of August 17, 2005)

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

Westinghouse

A BNFL Group company

Westinghouse Organization
(NRC Interface)

a, c

Topical Report Status

Westinghouse/NRC Meeting
Rockville, MD
August 16, 2005

PWR Topical Reports Under Review

a, c

PWR Topical Reports Planned

a, c

BWR Topical Reports Under Review

a, c

BWR Topical Reports Planned

a, c

BWR Topical Reports Planned

a, c

IRI Update

NRC/Westinghouse Meeting
Rockville, MD
August 17, 2005

IRI Update - Background

- NRC Bulletin 96-01 and Draft Supplement regarding IRI for Westinghouse NSSS fleet
- WOG Program initiated to respond

[ ] a, c

- WOG Program Successful; NRC canceled plans to issue Supplement 1 to Bulletin 96-01
  - Last WOG communication in January 2001 to NRC, and 2002 to WOG utilities

IRI Update – Burn-up Threshold Summary

- IRI Burn-up Threshold

a, c

IRI Update – Recent Activities

- Overtures from several utilities to extend burnup threshold for fuel assemblies in rodded locations
  - Core Designs challenging threshold
  - No recent IRI issues
  - Preclude IRI, susceptibility evaluations
  - Extend BU threshold requiring IRI susceptibility evaluations

IRI Update – Summary

- Positive Aspects
  - No IRI for instances with rodded assembly > burn-up threshold
  - Improved fuel design features to resist IRI
  - Continued low risk of IRI at current approved limits

LOCA Equivalent Clad Reacted (ECR) Criteria

NRC/Westinghouse Meeting
Rockville, MD
August 17, 2005

LOCA Equivalent Clad Reacted (ECR) Criteria

ACRS subcommittee meeting on Reactor Fuels met on July 27, 2005

Ralph Meyer of NRC/RES presented a summary of the Argonne National Labs (ANL) program and their proposed LOCA criteria which was:

- ECR including both operational corrosion and transient oxidation < 17% with the transient oxidation calculated by Cathcart-Pawel (C-P)
- Total time for transient < 2700 sec (45 minutes)
- Peak Cladding temperature (PCT) < 2200 ^{0}F

This was unexpected since ANL had earlier issued an embrittlement correlation which was more phenomenological based

LOCA Equivalent Clad Reacted (ECR) Criteria

ANL presented the results of LOCA simulation testing performed at ANL

ANL program plan is to wrap-up the program following the completion of the irradiated ZIRLO™ and M5 tubing tests and the integral test of the HBR rod segments

EPRI and ANATECH both made presentations which claimed that there are still unanswered questions from the ANL testing and that the ANL results do not correlate with the results from other programs

LOCA Equivalent Clad Reacted (ECR) Criteria

FANP presented joint EDF/CES/FANP data which showed much greater reduction in post test ductility as a function of hydrogen compared to ANL data

Rationale for this appears to be the direct quench used in the French tests

Apparently this lock in the high temperature morphology in the β-layer, where slower cooling provided time for segregations of oxygen and hydrogen in the β-layer providing greater ductility

NRC/RES stated that EPRI was not doing enough work to develop new limits, but only enough to verify the existing interpretation of the LOCA limits for high burnup fuel

LOCA Equivalent Clad Reacted (ECR) Criteria

ACRS question related to “What is the impact of up and down temperatures variations during high temperature oxidation on ECR and post test ductility?

Reactivity Initiated Accident (RIA) Criteria

NRC/Westinghouse Meeting
Rockville, MD
August 16, 2005

Reactivity Initiated Accident (RIA) Criteria

NRC/RES presented a summary of their analysis of RIA tests and his proposed criteria. The most limiting aspect of the proposed criteria was the collapse of the coolability limit onto a low cladding failure limit. The cladding failure limit was given as a function of maximum fuel rod corrosion

EPRI and ANATECH presented the industry proposed criteria and the methods used to produce it

Westinghouse presented summary of comments on proposed RIA criteria along with a sample analysis to demonstrate how limited the volume was of the core close to peak power and how unlikely conditions of high rod worth were

Reactivity Initiated Accident (RIA) Criteria

ACRS indicated that they thought the NRC/RES proposed criteria was very conservative

ACRS indicated that although they thought separate coolability and clad failure limits were reasonable, they were skeptical that the onset of fuel melt was the best limit and a lower one might be easier to justify

Reactivity Initiated Accident (RIA) Criteria

ACRS was skeptical of the methods ANATECH used to treat cladding test data to develop critical strain energy density (CSED) relationships as a function of temperature and oxide thickness.

ACRS though the method non-conservative, and the overall method using FALCON too obscure to easily understand

ACRS was skeptical that oxide spalling could be ruled out and thought that tests with spalled cladding should be included in developing the limit as was done by NRC/RES