Westinghouse Non-Proprietary Class 3

Westinghouse Presentation
on
Westinghouse Fuel Performance Update Meeting
(Slide Presentation of June 21-24, 2004)
(Non-Proprietary)

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Westinghouse Fuel Criteria Evaluation Process (FCEP) - WCAP-12488-A & WCAP-12488-A, Addendum 1-A, Revision 1

NRC/Westinghouse Meeting
Monroeville, PA
June 2004

Agenda

- Overview of Westinghouse FCEP
- DNB Correlation Applicability using the FCEP
- Application of FCEP to 17x17 and 15x15 RFA fuel

Overview of Westinghouse FCEP

- FCEP allows for 10 CFR 50.59 conclusions to be reached by demonstrating that the criteria defined in WCAP-12488-A and Addendum 1 are used for the evaluation of the fuel mechanical change(s) and are shown to be met

- When applied to specific design changes, FCEP will also show compliance with the SAR acceptance limits since the design criteria specified in FCEP are the acceptance limits denoted in the SAR

Overview of Westinghouse FCEP

- Section 4.2 of the SRP indicates that the NRC reviews the vendor-established specified acceptable fuel design limits (SAFDLs) to provide assurance that:
  - the fuel system is not damaged as a result of normal operation and anticipated operational occurrences,
  - fuel system damage is never so severe as to prevent control rod insertion when it is required,
  - the number of fuel rod failures is not under-estimated for postulated accidents, and
  - coolability is always maintained

Overview of Westinghouse FCEP

- These design bases, or Specified Acceptable Fuel Design Limits (SAFDLs), and their respective evaluations comply with the "Acceptance Criteria" of SRP 4.2, "Fuel System Design," and as such provide assurance that the fuel system is mechanically designed to perform safely, consistent with General Design Criteria (GDC) 10

- GDC 10 states:

  "*Criterion 10* – Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

Overview of Westinghouse FCEP

- FCEP consists of the following sections:
  - 1.0 Introduction
  - 2.0 Background
  - 3.0 Fuel Criteria Evaluation Process Description/Evaluation
  - 4.0 Fuel Design Bases and Limits
  - 5.0 Lead Test Assemblies Application
  - 6.0 Evaluation of Methodology Changes
  - 7.0 Fuel Performance and Material Properties Models

- The presentations today will briefly discuss Sections 4.0 and 7.0 with the primary focus on Section 6.0

Overview of Westinghouse FCEP

- Section 4 of WCAP-12488-A discusses "Fuel Design Bases and Limits"

- The Fuel Design Bases and Limits are further categorized into three subsection, consistent with the SRP:
  - 1) Fuel System Damage and Fuel Rod Failure Criteria,
  - 2) Fuel Coolability Criteria, and
  - 3) Nuclear Design Criteria

- These subsections define all the design criteria that would need to be addressed to demonstrate that FCEP is applicable (less grid modification and correlation applicability which has its own section)

Overview of Westinghouse FCEP

Fuel System Damage and Fuel Rod Failure Criteria:

- DNB
- Clad Stress
- Clad Strain
- Clad Fatigue
- Clad Oxidation
- Clad Flattening
- Rod Internal Pressure
- Fuel Rod Axial Growth
- Fuel Clad Fretting Wear
- Fuel Pellet Overheating
- Fuel Rod Clad Rupture (Burst)
- LOCA Fuel Clad Temperature
- Zircaloy Clad Hydrogen Pickup
- Thermo-hydrodynamic Stability
- Fuel Assembly Holddown Force
- Non-LOCA Fuel Clad Temperature
- Fuel Structural Hydrogen Content replaced by Structural Component Stress (Addendum 1)

Overview of Westinghouse FCEP

Fuel Coolability Criteria:

- Clad Ballooning and Flow Blockage
- Violent Expulsion of Fuel (Rod Ejection)
- Fuel Assembly Structural Response to Seismic/LOCA Loads
- Clad Embrittlement During Locked Rotor/Shaft Break Accident

Nuclear Design Criteria:

- Stability
- Shutdown Margin
- Power Distribution
- Fuel Storage Subcriticality
- Reactivity Feedback Coefficients
- Maximum Controlled Reactivity Insertion Rate

Overview of Westinghouse FCEP

- As noted previously, Section 4 of WCAP-12488-A discusses "Fuel Design Bases and Limits" (less grid modification and correlation applicability which has its own section)

- Section 6.0 of WCAP-12488-A discusses "Evaluation of Methodology Changes", specifically addresses DNB Correlation Applicability

- This section will be addressed in more detail in the next presentation

Overview of Westinghouse FCEP

- The last major section of FCEP is Section 7.0, "Fuel Performance and Material Properties Models"
- This section of FCEP allows for fuel performance and materials properties model revisions to the extent that [ ]a,c and accepted by the NRC
- Note that with PAD 4.0, model changes were made and for [ ]a,c that allowed under FCEP

Overview of Westinghouse FCEP

- To understand the level of engineering effort involved in FCEP, the following information is provided:

a, c

[ ]

- The only difference in the level of effort involved in FCEP versus a traditional submittal is the actual topical report vice an FCEP Notification Letter

- The following slides list items that have been FCEP to the NRC

Overview of Westinghouse FCEP to date

NTD-NRC-94-4275	8/29/94	Westinghouse's Interpretation of Staff's Position on Extended Burnup
NSD-NRC-96-4694	4/22/96	Transmittal of Presentation Material from NRC/Westinghouse Fuel Design Change Meeting on April 15, 1996
NSD-NRC-97-5189	6/24/97	Transmittal of Response to NRC Request for Information on Wolf Creek Fuel Design Modifications
NPL 97-0538 CAW-97-1166	9/9/97	Slide Presentation "14x14, 0.422" OD VANTAGE + (422V+) Fuel Design, Application for Point Beach Units 1 & 2"
NSD-NRC-98-5618	3/25/98	Notification of FCEP Application for WRB-1 and WRB-2 Applicability to the 17x17 Modified LPD Grid Design for Robust Fuel Assembly Application

Overview of Westinghouse FCEP to date

NSD-NRC-98-5722	6/23/98	Fuel Criteria Evaluation Process Notification for the Revised Guide Thimble Dashpot Design for the 17x17 XL Robust Fuel Assembly Design
NSD-NRC-98-5796	10/13/98	Fuel Criteria Evaluation Process Notification for the 17x17 Robust Fuel Assembly with IFM Grid Design
NSD-NRC-99-5828	03/29/99	Notification of FCEP Application for DNB Testing for Revalidation of WRB-1 Applicability to the 15x15 VANTAGE + Fuel Design
NSBU-NRC-00-5960	01/24/00	Fuel Criteria Evaluation Process Notification for Axial Blanket Modification
LTR-NRC-01-44	12/19/01	Fuel Criterion Evaluation Process (FCEP) Notification of the RFA-2 Design

Overview of Westinghouse FCEP to date

LTR-NRC-02-2	01/15/02	Fuel Criterion Evaluation Process (FCEP) Notification of the Quick Release Top Nozzle (QRTN) Design, (Proprietary)
LTR-NRC-02-55	11/13/02	Fuel Criterion Evaluation Process (FCEP) Notification of the RFA-2 Design, Revision 1, (Proprietary)
LTR-NRC-04-8	2/6/2004	Fuel Criterion Evaluation Process (FCEP) Notification of the 15x15 Upgrade Design (Proprietary/Non-Proprietary)
LTR-NRC-04-22	4/19/2004	Fuel Criterion Evaluation Process (FCEP) Notification of the Westinghouse Integral Nozzle (WIN) (Proprietary/Non-Proprietary)

DNB Correlation Applicability using FCEP

Westinghouse FCEP Section 6.0 specifies the guidelines relevant to determining Correlation Applicability for either grid design modifications or a new grid design:

"An existing DNB correlation will be valid and will meet the above design basis* without reservation provided the new fuel assembly geometry is [ ] a, c." (of the licensed database for the DNB correlation of interest).

"If the new geometry is [ ] a, c of the test data, Westinghouse will evaluate the geometry [ ] a, c."

* DNB design basis is that the probability of the limiting fuel rod not being in DNB be greater than 95 percent at a 95 percent confidence level.

DNB Correlation Applicability using FCEP

"If additional test data are used, [ ] a, c. The new data will then be [ ] a, c.

"If the new data [ ] a, c it may be treated explicitly as a [ ] a, c may be developed. If this step is necessary, it would involve NRC review."

Thus, there are three approaches that can be used: [ ] a, c. If none of these approaches yield acceptable results, then a submittal to the NRC is required.

DNB Correlation Applicability using FCEP

As specified in the TER of FCEP:

> "An existing NRC approved DNB correlation is considered to be valid by Westinghouse for application to a new design when the new fuel assembly geometry is similar to or bracketed by the fuel assembly geometric parameters and correlation parameters of the critical heat flux (CHF) test data used to develop the approved DNB correlation."

DNB Correlation Applicability using FCEP

The relevant geometric parameters are:

a, c

[ ]

DNB Correlation Applicability using FCEP

The relevant correlation parameters are:

a, c

[ ]

DNB Correlation Applicability using FCEP

Example:

a, b, c

DNB Correlation Applicability using FCEP

The FCEP approach (WCAP-12488-A & WCAP-12488-A, Addendum 1-A, Revision 1) has been approved by the NRC. As stipulated in the SER/TER, Westinghouse must:

- Inform the NRC of each first time application of the FCEP process for each design change (this is handled through letters to the NRC of FCEP Notification for each change - see list), and

- Maintain auditable records for each application of the FCEP process.

The changes are validated against data

DNB Correlation Applicability using FCEP

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Application of FCEP to RFA Fuel

- The next several slides will discuss the evolution of the 17x17 design from V5H to RFA to RFA-2 (an illustration of the grid straps will show these changes)
- Following the 17x17 design changes is a discussion of the 15x15 design changes (again, an illustration of the grid straps will show these changes)

Application of FCEP to RFA Fuel

17x17 Design Evolution

- As noted on the previous slides, the evolution of the 17x17 RFA fuel assembly started with the development of the Modified V5H (MV5H) mid-grid and Modified IFM (MIFM) grid (refer to NSD-NRC-96-4694)

- This design change: [ ] a,c that had been seen at Salem and Beaver Valley and [ ] a,c

- The FCEP notification documented that [ ] a,c, the WRB-2 correlation was applicable

Application of FCEP to RFA Fuel

17x17 Design Evolution

- Around the same time period, the Incomplete Rod Insertion (IRI) issue surfaced
- The resolution for IRI was: [ ]a, c
- The combination of ZIRLO™ skeleton, thicker walled GTs and MV5H/MIFM grids was named Robust Fuel Assembly (RFA) (refer to NSD-NRC-97-5189)
- This FCEP notification showed that the WRB-2 correlation remained applicable for the combination of MV5H/MIFM grids with the thicker walled GTs
- It was noted that in demonstrating WRB-1 and WRB-2 correlation applicability, that the RFA design had [ ]a, c

Application of FCEP to RFA Fuel

17x17 Design Evolution

- Thus in 1998, Westinghouse submitted the WRB-2M correlation to the NRC for review and approval. The WRB-2M correlation [

  ][a, c]

- Refer to WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999

Application of FCEP to RFA Fuel

17x17 Design Evolution

- The original development of RFA covered only 12 foot cores that used V5H fuel with IFMs. To address 14 foot (XL) designs, a slight modification was required to the GT dashpot design. This modification eliminated the double dashpot configuration and introduced a tube-in-a-tube dashpot configuration (refer to NSD-NRC-98-5722)

- The original FCEP notification for MV5H/MIFM showed that WRB-2 was an applicable correlation. For the XL design, it was necessary to show that the WRB-1 correlation was applicable for designs that did not incorporate the MIFM grid (refer to NSD-NRC-98-5618)

Application of FCEP to RFA Fuel

17x17 Design Evolution

- This FCEP notification documented that WRB-1 and/or WRB-2 were applicable correlations for both 12 foot and 14 foot cores with RFA fuel that did not incorporate the MIFM grid

- The first FCEP notification for RFA fuel was specific to Wolf Creek LUA application

- Since this FCEP notification was specific to Wolf Creek, a follow-on FCEP notification that documented WRB-2 correlation applicability to RFA fuel with IFMs (MIFMs) for all 12 foot V5H plants was sent to the NRC staff (refer to NSD-NRC-98-5796)

Application of FCEP to RFA Fuel

17x17 Design Evolution

- The latest evolution of RFA fuel is called RFA-2. The principal difference between RFA and RFA-2 is a minor modification to the [ ][a,c] (see illustration on following slide). The change results in [ ][a,c] (refer to LTR-NRC-01-44)

Application of FCEP to RFA Fuel

a, b, c

[ ]

Application of FCEP to RFA Fuel

15x15 Design Evolution

- In the 1970's, Westinghouse performed a large number of CHF tests on rod bundles prototypical of the 15x15 (0.422 in OD rod) and 17x17 (0.374 in OD rod) Inconel "R-grid" mixing vane grid types. The WRB-1 CHF correlation was developed based on the results from these tests

- Westinghouse continued to develop new fuel products, including the Optimized Fuel Assembly (OFA) designs, which were "scaled" or extrapolated from the R-grid designs

- The 14x14 and 17x17 OFA designs were DNB tested and the results showed that WRB-1 accurately predicted CHF for OFA fuel designs

Application of FCEP to RFA Fuel

15x15 Design Evolution

- Based on the scaling method and the similarity of the 15x15 OFA design with the R-grid geometry, the WRB-1 correlation was extended to cover 15x15 OFA fuel without DNB tests. This extension was accepted by the NRC
- Similarly, when the 15x15 V5H product was first developed, it was again compared with the R-grid geometry and it was determined that the WRB-1 correlation could also be extended to cover the design again without DNB tests

Application of FCEP to RFA Fuel

15x15 Design Evolution

- In the process of getting a license amendment submittal approved for a licensee, the NRC staff insisted that Westinghouse perform DNB water tests on the 15x15 VANTAGE + fuel product (geometrically equivalent to the 15x15 V5H design with IFMs)

- As a result, Westinghouse performed a DNB test on the 15x15 VANTAGE + product with IFMs to confirm that the WRB-1 correlation would be applicable to this design (refer to the table below)

a, c

[ ]

Application of FCEP to RFA Fuel

15x15 Design Evolution

- Based on the DNB test performed for the 15x15 V5H/VANTAGE + fuel product with IFMs, Westinghouse submitted an FCEP Notification to the NRC to document the fact that the WRB-1 correlation was applicable to the design (refer to NSD-NRC-99-5828)

- Note: the 15x15 V5H/ VANTAGE + design still has the same original V5H LPD mid-grid configuration as the 17x17 V5H design incorporated (minor differences between array designs)

- To improve upon this 15x15 design and incorporate those features proven for the 17x17 design, a plan was established to implement upgrade features to the 15x15 design [

] a, c

Application of FCEP to RFA Fuel

a, b, c

[ ]

Application of FCEP to RFA Fuel

a, b, c

[ ]

Application of FCEP to RFA Fuel

15x15 Design Evolution

- Based on the assessment in the table on the preceding slide, it can be seen that the 15x15 RFA-2 and the 15x15 Upgrade design (I-spring configuration) both meet the geometric conditions for WRB-1

- Based on vibration testing, the 15x15 Upgrade product showed significant margin enhancement to reducing fuel rod vibration over the 15x15 RFA-2 design

- LTR-NRC-04-8, "Fuel Criterion Evaluation Process (FCEP) Notification of the 15x15 Upgrade Design (Proprietary/Non-Proprietary)," February 6, 2004

Application of FCEP to RFA Fuel

FCEP Process

- [

]a, c

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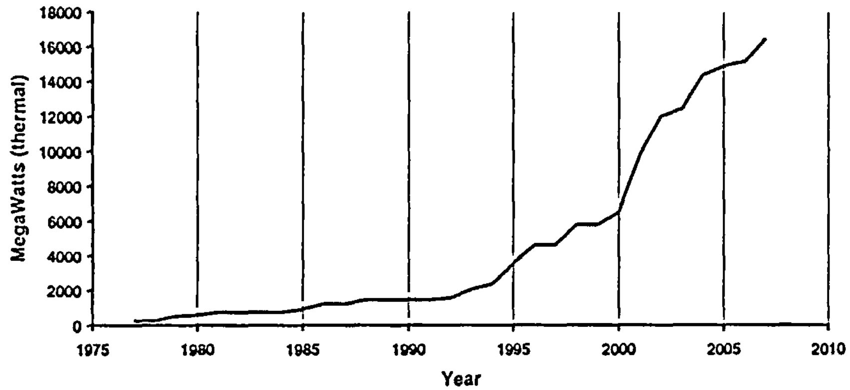
Flawless Fuel Performance Initiatives/ Fuel Performance Update

NRC/Westinghouse Meeting
Pittsburgh, PA
June 22, 2004

Our industry continues to improve and increase nuclear power generation

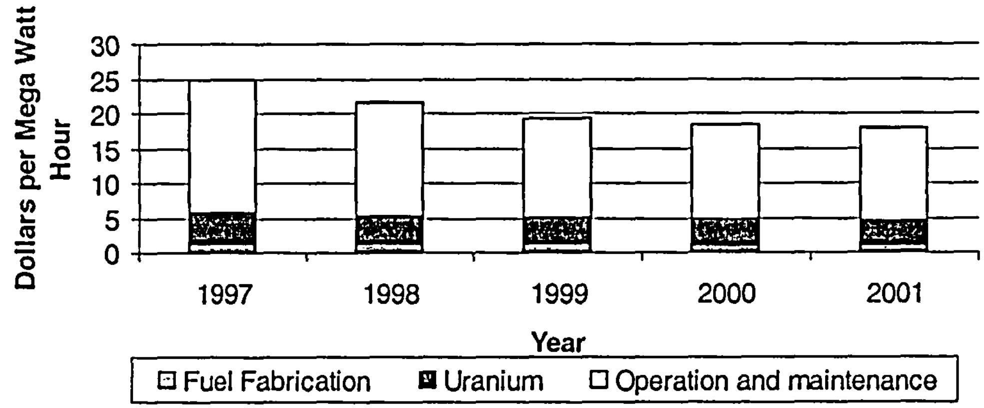
We have increased power output ..

Total Megawatts Added by Power Uprates



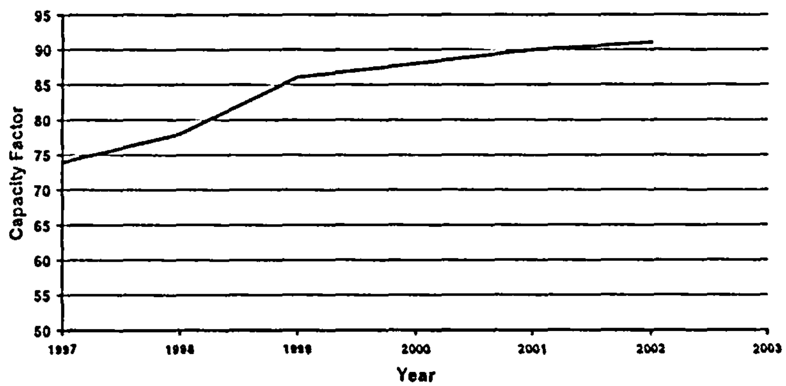
contributing to reduced $/Mwt..

Nuclear Generation Production Costs



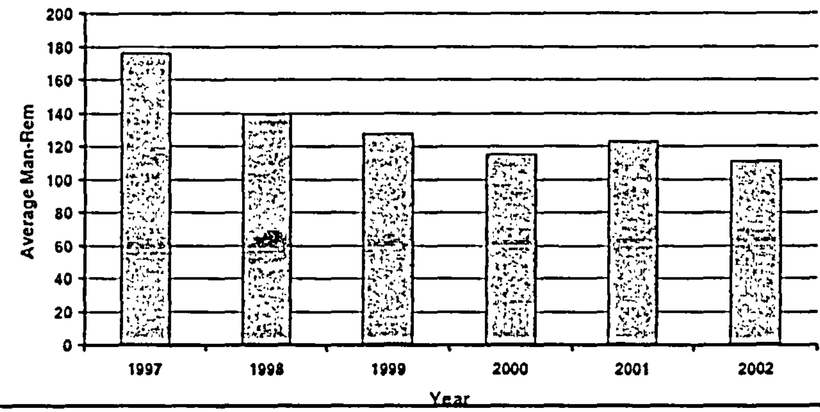
and added capacity ..

Average Capacity Factor



while improving safety ..

Collective Radiation Exposure



Westinghouse PWR fuel reliability has plateaued

a, c

Westinghouse PWR Fuel Reliability

(50 to 60 Plants per Year, W and CE-NSSS Fuel)

a, c

Leakage Mechanisms in PWR Fuel in 2003

a, c

Where Are We Today?

a, c

PWR Grid-Rod Fretting

a, c

Current Status of RFA

a, c

One Key Initiative (Proactive Approach):

[ ] a, c

[ ] a, c

Essential Elements of [ ]a, c

[ ]a, c

Fuel Performance [ ] a,c

[ ] a, c

"ZIRLO™ trademark property of Westinghouse Electric Company LLC"

Fuel Performance [ ] a,c (Cont.)

a, c

17 OFA Root Cause Investigation Update

Fall 2003 PIE Results

a, c

Proactive Approach to Identify Failure Mechanisms in 17 OFA Fuel

a, c

Potential Failure Mechanisms

a, c

Immediate Actions to Address Current OFA Failure Mechanisms

a, c

[ ][a, c] EOC 13 Inspections and Preliminary Evaluations

EOC 13 Fuel Inspection Results

a, c

EOC13 Leaking Assembly Core Locations

a, c

EOC 13 Visual Fuel Inspection Results

[ ] a, c

EOC 13 Visual Fuel Inspection Results (con't)

a, c

Suspected Leaking Rods in [ ] a, c

a, c

Appearance of [

]a, c

a, c

Preliminary Failure Mechanisms of [ ] a, c Leaking Rods

[ ] a, c

Summary of Lower Core Plate Flow Anomaly Issue

US Experience of Grid-to-Rod Fretting due to Flow Anomaly

a, c

Core Locations of Leakers in [

] a, c

a, c

Example Location of Leaking Fuel Rods
[ ]a,c

[ ]a, c

Design Modifications To Improve Fuel Rods Resistance to Flow Induced Vibration

a, c

Fuel Assembly with [ ] a, c

[ ] a, c

Conclusions

a, c

Future Inspection Plans

Planned Inspections

a, c

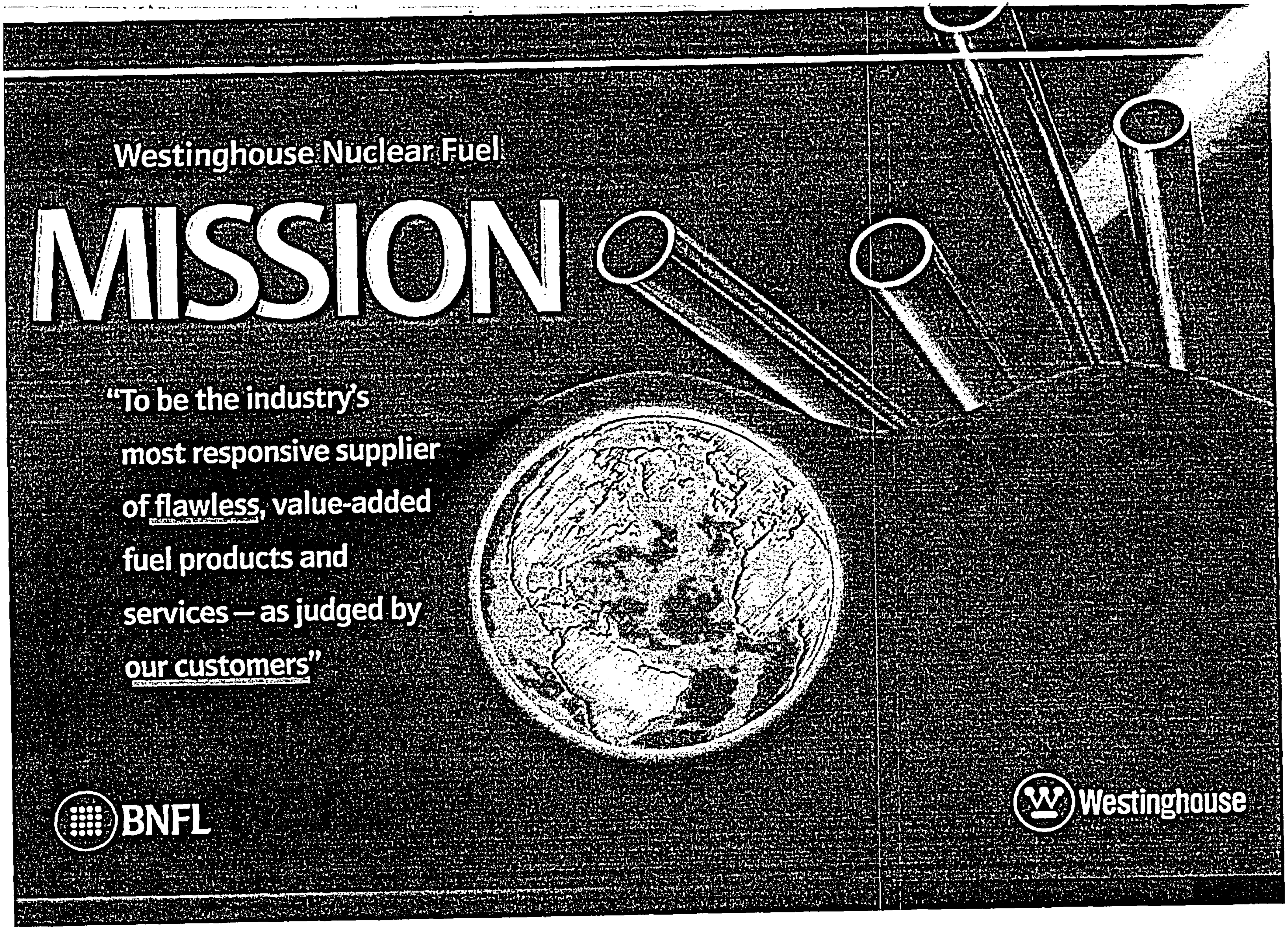
Potential Future Inspections

a, c

Summary

- Fuel performance improvement has reached plateau
- Action plans in place and being implemented to achieve goal of flawless fuel performance

[ ] a, c


Westinghouse Nuclear Fuel
MISSION
"To be the industry's most responsive supplier of flawless, value-added fuel products and services – as judged by our customers"
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General Fuel and Core Component Issues

NRC/Westinghouse Meeting

Monroeville, PA

June 22, 2004

Discussion Topics

- Grid-to-Rod Gaps in CE SYSTEM 80™ Assemblies
- Assembly Growth Issue for CE 16x16 Assemblies Utilizing Inconel Top Grid
- 10CFR21 Issue with Cracked Hybrid B_4C RCCAs
- Separated RCCA Rodlet

Grid-to-Rod Gaps in CE SYSTEM 80 Assemblies

Grid-to-Rod Gaps in CE SYSTEM 80 Assemblies

- During the fabrication of SYSTEM 80 fuel assemblies, a higher than normal occurrence of fuel rods were detected to have gaps between the grid support features and cladding at the top grid elevation after rod pushing
- Gaps were also discovered in assemblies at the plant site from the same region that previously passed inspection in Columbia
- Previous incidence of gaps was low
  - Four SYSTEM 80 design regions produced in Columbia prior to current region
  - Total of 8 loose rods in 7 assemblies in four previous Columbia regions
  - One loose rod observed in most recent region
- Primary concern for gaps is increased risk of grid-to-rod fretting failures

Background

- SYSTEM 80 design is unique 16X16 design
  - Expanded section in the upper guide tube requires expansion process in the top grid guide tube cell
- Top grid fretting fuel performance generally good

[–] a, c

a, c

Corrective Actions

a, c

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Long Term Actions

- Skeleton and assembly loading process strengthened
- 100% inspection and reset of top grid cells implemented after skeleton fabrication
- Revised design and fabrication processes under evaluation

Summary – Grid-to-Rod Gaps in CE SYSTEM 80 Assemblies

- Gaps that increase risk of fretting leakers during operation detected at top grid locations during region fabrication
- Corrective actions implemented to mitigate design and process weaknesses identified by Root Cause Investigation
- Assemblies reworked, rebuilt and new assemblies fabricated to complete region delivery
- Region successfully delivered without gaps
- Design and additional process modifications under evaluation to further reduce potential for gap formation

Assembly Growth Issue for CE 16x16 Assemblies Utilizing Inconel Top Grid

Inconel Top Grid

- Purpose
  - Provide improved grid-to-rod fretting margin compared to Zircaloy top grid designs in CE units
    - Benefit from reduced relaxation of Inconel material
    - Reduce the potential for grid-to-rod gap formation at the top grid elevation
- Current Status
  - One CE unit is operating with the Westinghouse Inconel top grid design
  - Deployment in other CE units is planned

Performance Aspects

- Potential for increased:

[a, c

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]

  – assembly length change

[a, c

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Performance Aspects

a, c

- The model was updated and now functions properly

a, c

Updated Assembly Length Prediction

a, c

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Assembly Length Change

- Resolution for Operating assemblies

a, c

Summary - Inconel Top Grid

- No adverse effect on co-resident fuel assemblies of an older design or fuel rod bowing
- Improvement in top grid fretting margin remains an important advantage of the Inconel top grid
- Additional assembly length measurements will be acquired and the information will be factored into both current and future fuel regions for the CE fleet to accommodate assembly length change

10CFR21 Issue on Cracked Hybrid RCCAs

10CFR21 Issue on Cracked Hybrid RCCAs

- Westinghouse Hybrid B_4C RCCAs used in only 1 US PWR

[a, c

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10CFR21 Issue on Cracked Hybrid RCCAs

a, c

10CFR21 Issue on Cracked Hybrid RCCAs

a, c

- Potential exists for some of the RCCAs in that plant to have cracks

10CFR21 Issue on Cracked Hybrid RCCAs

- Environment in RCCA after cracking is uncertain
  - Geometry of crack and interference with cladding may limit water ingress
  - Static flow conditions prevail
  - Dissolution of Boron in B_4C pellets measured to be low in autoclave testing, but data and time of exposure are limited
- Conservative position and Westinghouse recommendation is to avoid reinsertion of Hybrid RCCAs with known cladding cracks

Actions Taken for Hybrid RCCA Cracking

- Westinghouse 10CFR21 process initiated
  - NRC notified on April 16, 2004 of potential 10CFR21 issue if cracks go undetected for extended time
- Risk assessment performed for utility regarding affect of cracking on continued operation through current cycle
  - Risk of affecting shutdown margin and RCCA worth due to cracking is low based on BOC rod worth measurements and static flow environment in rodlets
  - Recommendations provided to utility to mitigate risks if mid-cycle reactor cool down required
- Impact of potential cracking on other plants with Westinghouse RCCAs and CEAs assessed and found to be none
- Nuclear Safety Advisory Letter Issued for widespread customer notification
- Utility preparing to replace RCCAs at end of current cycle

Summary – Hybrid RCCA Cracking

- Westinghouse's review of operating experience data for Hybrid B_4C RCCAs design identified crack initiation time as low as 5.6 EFPY
- 1 US PWR currently operating in Cycle 6 with RCCA exposures greater than earliest cracking observations
- Potential exists for some of the RCCAs in that plant to have cracks
- NRC notified of 10CFR21 Issue if cracks go undetected for extended time
- Utility planning to replace affected RCCAs at end of current cycle
- Westinghouse developing internal process to improve communications of RCCA operating life expectations

Separated RCCA Rodlet

Separated RCCA Rodlet

- One of twenty-four rodlets separated from a single 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 or plant shutdown at EOC
- RCCA had been in service for ~11 EFPY
- Fuel assembly and RCCA were discharged, spare RCCA used as replacement

Separated RCCA Rodlet

a, c

Separated RCCA Rodlet

a, c

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Actions Taken for Separated Rodlet

- Videotape inspection performed to characterize appearance
- Field Anomaly Report generated and Risk Assessment performed

[
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-
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] a, c

- Root Cause Investigation initiated by Westinghouse

[
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] a, c

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Summary – Separated RCCA Rodlet

- Separated rodlet observed for 1 RCCA
- Rodlet located inside thimble tube of host assembly
  - no affect on RCCA insertion or plant shutdown
- Root Cause investigation in progress

[–

–] a, c

- Westinghouse will keep NRC and customers informed of RCA conclusions

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Optimized ZIRLO™

NRC/Westinghouse Meeting

Monroeville, PA

June 22, 2004

Optimized ZIRLO

- ZIRLO - Excellent performance, low growth, low corrosion, licensed to 62000 MWD/MTU, FDI ~ 800 -900
  - Extremely large experience base - over 2.5 million ZIRLO fuel rods are in operation or discharged

[ ] a, b, c

ZIRLO™ trademark property of the Westinghouse Electric Company LLC

Optimized ZIRLO

- ZIRLO with lower tin levels improves corrosion resistance while preserving the mechanical properties and margin for abnormal operating conditions

- Higher Burn-UP Capability (>70,000 vs. 62000 MWD/MTU)

- Higher Rod Internal Pressure Margin

ZIRLO Chemistry

Standard and Optimized ZIRLO Chemistry :

Alloy	Sn%	Nb%	Fe%	Oxygen
ZIRLO	[			] a, b, c
Optimized ZIRLO				]

The tin level in ZIRLO is lowered to improve the corrosion resistance but still maintain the other excellent characteristics of ZIRLO. Similar to what was done for Zircaloy-4 in prior years.

ZIRLO AND OPTIMIZED ZIRLO PEAK OXIDE THICKNESS COMPARISON

a, b, c

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Status of Optimized ZIRLO LTA Programs

a, b, c

ZIRLO Licensing History/Issues

- For region use

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- NRC has approved several exemptions for Optimized ZIRLO

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Optimized ZIRLO

SUMMARY:

- ZIRLO - Excellent performance, low growth, low corrosion, licensed to 62,000 MWD/MTU, capable of 70000 MWD/MTU depending on the fuel duty, and FDI ~ 800 - 900

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[ ]a, c LTA Plans

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What is the Need for a New Alloy

- Potential Driving Factors
  - Ultra high burn up designs - 75 GWD +
  - Higher uranium loadings
    - higher densities
    - different fuels
  - Uprates
  - New Regulatory Provisions
    - Long term storage
    - LOCA margin improvements
- **It takes many years to bring a new alloy to full licensing**
  - **We can not wait for a specific need to be identified before beginning development**

[ ][a, c] is Needed for Very High Fuel Duties

[ ][a, b, c]

"ZIRLO™ trademark property of Westinghouse Electric Company LLC"

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[ ] Development Program

- Alloy development team formed with Westinghouse experts from US and Sweden

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What Is [ ]?

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[ ] a, b, c

[ ]a, c Show Excellent Potential for Very Low Corrosion Rates

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Material Property Summary

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Strong Technical Justification for LTA License Exemption

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- Westinghouse core design indicated no significant concerns
- Westinghouse licensing group has participated from initiation

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Solid Performance Basis

- *Corrosion performance*

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Typical LTA Configuration

- *Use limited number of assemblies and rods*

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Operational Considerations

- *Reactor Coolant contamination*
  - Corrosion rates will be monitored
  - Limited number of rods
- *Rod Fabrication*

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LTA Licensing Plan

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Core Design Considerations

- Typical Core Design

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Inspection Plans

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For Typical LTA in Spring 05

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Latest Overall LTA Schedule

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Latest Overall LTA Schedule

- LTAs Used for these reasons
  - Demonstrate new materials, designs or design features
  - Used to evaluate changes in operating conditions
  - Used to obtain high burnup data

Westinghouse High Burnup ZIRLO™ LTA Summary

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Other Test Programs

a, b, c

Creep and Growth Program - Status

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ZIRLO Hotcell Exams

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Summary of LTA Programs

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Summary of LTA Programs (cont.)

a, b, c

Summary of LTA Programs (cont.)

a, b, c

Summary of LTA Programs (cont.)

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Summary of LTA Programs (cont.)

[ ] a, b, c

Summary of LTA Programs (cont.)

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Low Tin ZIRLO LTA Programs

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Status of Low Tin ZIRLO LTA Programs

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[ ][a, c] LTAs

- [ ][a, c] LTAs being proposed for several US 17x17 OFA plants
- Expect submittal of exemption requests in Summer of 2004
- Need to complete exemption review within 6 months to fit within fabrication schedules for Spring 2005 refueling outages

Westinghouse perspective on RES's RIS on RIA

- Proposed Limits are excessively restrictive
  - –Expansion due to contraction tests on ZIRLO show good ductility under rapid deformation even at high levels of corrosion
  - –CIP01 test performed well at energy levels well above those proposed in RIS
  - –Reducing energy to account for impact of 11 to 15 ms vs 30 ms pulse not justified

Westinghouse perspective on RES's RIS on RIA (cont'd)

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Westinghouse perspective on RES's RIS on RIA (Cont'd)

- Westinghouse proposed 100 cal/g energy addition with approval of 3 D methods for RIA analysis.
- Cabri water loop conversion will provide means to perform prototypical test under 10 ms pulse width.
- Expansion due to contraction (EDC) tests provide good cost effective means to test cladding under RIA like conditions. Materials are available at Studsvik hotcell to test if the test results would be useful.

Summary

- Obtaining data from various plants with different operating conditions and heats of material
- Data obtained is sufficient to obtain high burnup licensing
- RES proposed RIA limits are too conservative and restrictive on core design. Not justified by either performance data or accident consequence

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Next Generation Fuel Update

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17x17 NGF LTA Status

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Key Objectives of NGF Program

- Flawless fuel (zero defects, no hassle)
  - mechanical margin
- Improved design margin for extended upratings or improved fuel management
  - Crud / CIPS (aka AOA)
  - DNB
  - Rod internal pressure
  - Corrosion
  - LOCA
  - hi- burnup
  - high fuel duty
- Improved fuel handling
  - allow incore shuffles
- Design for manufacturability
  - strong focus on quality and cost

17 x17 NGF Fuel Assembly Overview

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17x17 NGF Schedule

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17x17 NGF Critical Heat Flux Testing

- Five test series completed
- Data analyzed to demonstrate that WRB-2 and WRB-2M correlations are applicable and /or conservative for NGF LTA application

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Critical Heat Flux Testing

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Licensing for Region Implementation

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CE 16x16 Next Generation Fuel Update

NRC/Westinghouse Meeting
Pittsburgh, PA
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Agenda

- CE 16x16 NGF Design Expectations
- CE 16x16 NGF Design
- NGF LTAs
- General Program Schedule
- Licensing Plan

CE 16x16 NGF Design Expectations

- Three (3) issues that we expect to resolve through implementation of a 16x16 NGF design:

  - Increase margin to Grid-to-Rod fretting failures

  - Increase thermal margin to accommodate further power uprates in the future

  - Increase margin to core crudding resulting from hot spots in the core

CE 16x16 NGF Assembly

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NGF LTAs to be Implemented at [ ]a, c

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General Program Schedule

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Licensing Plan

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CE Physics Methods and Code Integration with APA

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Overall Plan for Integrated Neutronics System

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Neutronics code integration – Strategy

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Neutronics code integration - Project Plan and Status

- Transition will require some changes to APA software

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Neutronics code integration - Project Plan and Status

- Transition will require some changes to APA software

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Neutronics code integration - New Methodology

- PARAGON

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Licensing - Strategy

- Obtained approval for PARAGON, ZrB2 for CE methods, and ZIRLO™ cladding in CE cores
- Provide required reload submittals well in advance

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- Perform reload under 10 CFR 50.59
  - New methodology already accounted for in COLR references which will include PARAGON/PHOENIX-P and ANC

Neutronics code integration - Licensing Issues

- Based on current information (e.g. results of benchmarking, implementation strategy, review of SERs) Westinghouse believes

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Neutronics code integration - Summary

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BEACON™-COLSS Integration

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BEACON-COLSS Project

- Project started in 2002 to merge COLSS and BEACON™ core monitoring system
  - –Combine the best parts of BEACON & COLSS to develop a functional upgrade to COLSS
  - –Develop a BEACON-COLSS product line for CE plants
  - –Significant benefits to current COLSS plants
  - –Back fit potential to CE analog and W plant designs

BEACON™ trademark property of the Westinghouse Electric Company LLC

BEACON-COLSS Project

- COLSS is a licensed core monitoring system on CE digital plants
  - –Operating at 12 CE plants
  - –NRC Review & approval of COLSS uncertainty methods in 1987
- BEACON is a licensed PWR core monitoring system with movable and fixed incore detectors (MIDs & FIDs)
  - –Operating at ~50 plants
  - –NRC review and approval of BEACON methods in 1994
  - –NRC approval of BEACON Addendums in 1999 and 2001

Design Bases

- COLSS design bases
  - Assist operations in maintaining LCOs
  - Assist operations in maintaining core power below licensed limit
  - Assist operator during normal operation
- BEACON-COLSS design bases

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Functional Design

- BEACON-COLSS Functional Design Requirements

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BEACON-COLSS Overview

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Uncertainty Methodology

- COLSS licensed uncertainty methodology
  - Compares COLSS synthesized results with "true" 3-D results

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Uncertainty Methodology

- BEACON licensed uncertainty methodology
  - Compares BEACON 3-D results with "true" 3-D results

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Uncertainty Methodology

- BEACON-COLSS uncertainty methodology
  - Merge COLSS and BEACON methods

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Licensing Submittal

- Submit licensing document as Addendum 3 to the BEACON Topical Report for COLSS application

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Status / Schedule

- New uncertainty methodology developed and test results generated for one plant
- Production version of analysis software being developed
- Development of BEACON licensing addendum in progress
- Uncertainty analysis being performed on second plant
- Formal licensing submittal in September

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BWR
Fuel Performance Update

NRC/Westinghouse Meeting
Monroeville, PA
June 22, 2004

Presentation overview

- BWR Fuel Deliveries
- SVEA-96 Optima2
- Fuel Performance Experience
  - Burnup experience
  - Channels, Cladding and Liner performance
  - Primary and Secondary Fuel Failures
    - Mitigating programs

BWR Fuel Deliveries

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SVEA-96 Optima2

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Channels

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Channel Bow
Symmetric Lattice

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Cladding

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Cladding Corrosion

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Rod Growth

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Rod Growth

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Primary Fuel Failure Causes

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Fuel Failures in Westinghouse 10x10 fuel

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BWR Fuel Performance
Monthly Values per Million Operating Rods

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Secondary Fuel Failures & Fuel Washout

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Fuel Failures in Westinghouse 10x10 Fuel Secondary Degradation

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Summary

- SVEA-96 Optima2 is now being introduced to US customers

- Good performance at higher burnup verified

- Many development programs have been successful implemented to improve reliablity

- Continued effort to achieve our goal:

  Flawless Performance !!!

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Westinghouse Nuclear Fuel Licensing Update

NRC/Westinghouse Meeting
Monroeville, PA
June 2004

Overview

- Update on Fuel Related Activities Discussed in December:
  - WCAP-10125 Addendum 1 "Transient Stress" - Final SER Issued
  - WCAP-10125 Addendum 1 Rev 1 "Transient Stress" - Final SER Issued
  - WCAP-16072 "CE ZrB_2" - Final SER Issued
  - WCAP14565 Add 1 "ABB CHF Correlation with VIPRE" - Final SER Issued
  - WCAP-16045 "PARAGON" - Final SER Issued

- Current NRC Submittal List
  - Topicals (see attached table)

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- Other Regulatory Topics

PWR Topical Update

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BWR Topical Update

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Other Regulatory Topics

- Need a resolution on the ZIRLO™ topical:
  - Addendum 1 to WCAP-12610 and CENPD-404
- Implementation of [ ] a,c Reload Methodology
- Implementation of ANC to CE NSSS
- SER for [ ] a,c change to Westinghouse Methodology
- Topical reviews and RAIs
  - Process
  - Basis for RAIs
  - schedule
- Open Discussion

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Power Up-rates Westinghouse Experiences from European Plants

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Power Up-rates - Background

- Westinghouse Electric Sweden (WSE) has delivered eleven BWRs
  - Nine are located in Sweden
  - Two are located in Finland

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- WSE has participated in all up-rates (hardware/software)
- WSE has delivered or delivers fuel to most of the plants.

Power Up-rates - Background

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Power Up-rates – General Approach

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Power Up-rates – Core Related Aspects

- During the feasibility study an equilibrium core and the cycle length are defined as well as a preliminary power/flow map.
- Estimation and verification of the core are performed for normal operation and limiting transients and design basis accidents.
- Final verification of the core and the recirculation system is performed during the detail design & licensing phase.

Power Up-rate – FSAR Topical Reports

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Power Uprate Licensed Safety

- Core design
- Operating domain layout
- General methodology to identify limiting transients
- Transients

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Core design

- Design an equilibrium licensing core at specified power
- Meet requirements of regulatory body and customer
- Fulfill various limits and demands for the fuel
- Consider transient criteria, i.e. dryout, stability

Operating domain layout

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General methodology to identify limiting transients

- Consider:
  - –Existing valid SAR
  - –Documentation that specifies identified events
  - –Plant updates at power uprating
  - –Updated operating domain
- Considering given information, limiting transients will be specified.
- Perform verification calculations, if necessary.

Stability

- Will mainly be considered at:

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- Used code: RAMONA, POLCA-T

Transients - Dryout

- Limiting transient within following areas:
  - –Pressure increase
  - –Pressure decrease
  - –Recirculationflow increase
  - –Recirculationflow decrease
  - –Feedwaterflow increase
  - –Feedwaterflow decrease
- Used code: BISON

Integrity of Reactor Coolant Pressure Boundary (RCBP)

- Limiting transient within following areas:
  - –Anticipated events
  - –Postulated accidents
  - –Independency SCRAM / Safety Relief Valve
  - –Common Cause Failure in Safety Relief Valve
  - –ATWS

- Used code: BISON

LOCA

- Limiting transient within following areas:
  - –Large break LOCA
  - –Small break LOCA
- Used code: GOBLIN

Reactor containment

- Limiting transient within following areas:
  - –Large steam line break
  - –Large feedwater line break
  - –ATWS
- Both containment pressure and temperature as well as suppression pool temperature will be verified
- Used code: COPTA

  ATWS - Input taken from GOBLIN calculation

ATWS

- Will be considered for:
  - –Integrity of RCPB – According to prior slides
  - –Reactor Containment – According to prior slides
  - –Dryout / Core uncovery

ATWS – Dryout / Core uncovery

- Two types of transients:

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- Two types of codes: BISON
  GOBLIN

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BWR Control Rod Update

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Topics

- Design Evolution
- Experience
- Licensing History
- NRC Interactions

Westinghouse Design Evolution

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Experience - US Deliveries Details

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CR-99 Deliveries

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Experience - Inspections

- All Westinghouse control rod designs are easily inspected
  - Simple visual inspection
  - Low cost inspection
  - No special tools needed
  - Inspection verifies the condition of the control rods
- Easily inspected control rod a clear advantage
  - Lower risk of costly future inspection programs

Experience - Inspections

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Experience - Lessons Learned from Inspections

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Licensing History

- Westinghouse (Atom/ABB) Licensing History
  - –TR UR 85-225, CR 82 use in D-Lattice, approved 1986
  - –Supplement 1, CR 82 use in C-Lattice, approved 1988
  - –Supplement 2, CR 82 use in S-Lattice, approved 1989
  - –Supplement 3, CR 82 use of High Worth rods, submitted in 1988
- CR 82 is currently NRC licensed
  - –CR 82M-1is same as CR except for 316 ss vs 304 for CR 82
  - –Used via a 50.59 approached (Discussed with NRC previously)

NRC Interactions

- Westinghouse BWR Control Rod Experience Update Meeting
  - Summer 2003

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Recent BWR Licensing Initiatives

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BWR EPU in the U.S.

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10x10 SVEA Design Evolution

- SVEA-96

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- SVEA-96+

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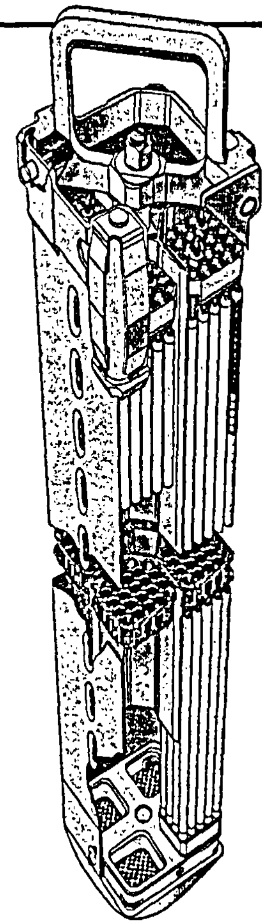
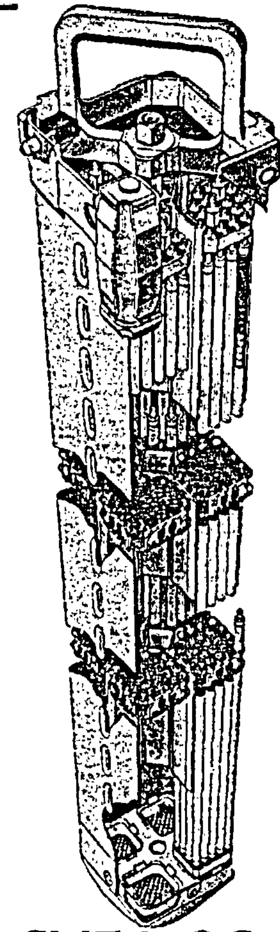


SVEA-96/96+



SVEA-96 Optima2

Support Introduction SVEA-96 Optima2 Fuel into the U.S.

Report No	Title	Submittal Date	Approval Date
WCAP-16078	ECCS Evaluation Model Code Sensitivity for SVEA-96 Optima2	[	] a,c

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- Responses to three Sets of RAI's Submitted

Support Introduction SVEA-96 Optima2 Fuel into the U.S.

Report No	Title	Submittal Date	Approval Date
WCAP-16081	10x10 SVEA-96 Optima2 Critical Power Correlation		

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- Responses to two Sets of RAI's Submitted

Improvements to Licensing Analysis Methodology

Report No	Title	Submittal Date	Approval Date
WCAP-15836	Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1	[ ]a,c	

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- Responses to two Sets of RAI's Submitted

Improvements to Licensing Analysis Methodology

Report No	Title	Submittal Date	Approval Date
WCAP-15942	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1	[	] a,c

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Improvements to Licensing Analysis Methodology

Report No	Title	Submittal Date	Approval Date
WCAP-00000	Three-Dimensional Core Simulator-Based System Analysis Code, POLCA-T: Application to Stability Evaluations and Control Rod Drop Accident	[ ]^{a,c}	

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Improvements to Licensing Analysis Methodology

What is POLCA-T?

- POLCA-T is a computer code system which will replace existing codes for nuclear, T/H, and transient 3D BWR analyses.

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Improvements to Licensing Analysis Methodology

- POLCA-T could also potentially replace the following functions currently performed in Europe

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Improvements to Licensing Analysis Methodology

One code system for steady-state and dynamic analyses:

- No interfaces, no transformation/condensation/normalization required.
- A single data base
- Powerful models and a modern, well structured code
- Reduced maintenance
- Improved QC

Improvements to Licensing Analysis Methodology

POLCA-T Status

- Released 1.0.0 February 2003 for production application to stability calculations in Europe.

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Improvements to Licensing Analysis Methodology

- POLCA-T Status – Used for European Engineering Studies
- Powerful tool for studying special events, particularly when asymmetries and/or local effects are important.

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Improvements to Licensing Analysis Methodology

Typical POLCA-T Verification and validation

- Analytical solutions:

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Improvements to Licensing Analysis Methodology

Typical POLCA-T Verification and validation efforts

- Separate effects:

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Improvements to Licensing Analysis Methodology

Typical POLCA-T Verification and validation efforts

- Integral tests, Stability:

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Improvements to Licensing Analysis Methodology

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Improvements to Licensing Analysis Methodology

Typical POLCA-T Verification and validation efforts

- Integral tests, Transients:

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Improvements to Licensing Analysis Methodology

POLCA-T Plans for the near future

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Improvements to Licensing Analysis Methodology

POLCA-T Plans for the future

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- Submittal dates to be determined.

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US BWR Columbia Manufacturing and Shipping Update

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Topics

- Fabrication Split and Schedule
- Shipping/Transport
- Facilities Changes
- Licensing Activities

Major Component Sources for US BWR Fabrication

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Current Manufacturing Schedule

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Shipping/Transport

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Facilities Changes - Physical

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Facilities Changes - Physical

- Status

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Facilities Changes - FME

- Why are we focusing on fuel rod FME?

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Facilities Changes - FME

- Objective
  - Implement all necessary steps at SMP and Columbia to produce final fuel rods free from hydrogenous material contamination

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Facilities Changes - FME

- Why decide now?

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Facilities Changes - FME Prevention

- Multiple options were evaluated against several criteria such as cost, implementation risk, impact, etc. prior to final recommendation

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Facilities Changes - FME Prevention

- Summary of Overall strategy

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Licensing Activities

- Columbia Facility
  - Finalize Layout Plan – Complete
  - Complete Integrated Safety Assessment – Jan-05
    - No NRC License Submittal Needed

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Current Industry issues

NRC/Westinghouse Meeting
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June 23, 2004

Current BWR Industry Issues

- BWR Channel Bow
- BWR Debris Fretting
- BWR Coolant Chemistry
- BWR Secondary Failure Mitigation
- BWR Stability: Plant-Specific Detect and Suppress (Option 3) Solution
- NRC Perspective

Industry BWR Channel Bow

- Historical Emphasis on
  - Interference with plant components – particularly control rods
  - Channel Reuse – no longer practiced
  - Degradation of CPR performance
    - Oskarshamn 2 dryout in 1988
    - Now CPR Performance treated explicitly (R-factor modification and/or treated in SLMCPR evaluation)
  - Reactivity Loss
    - Implicitly accommodated by following critical $k_{effective}$ based on core follow

Industry BWR Channel Bow

- Recent reimergence of Interference problems – our impression
  - Primarily BWR/6 with some occurrence in C-lattice BWR/4 plants
  - Associated with BWR Shadow corrosion
    - High energy cycles – control blades inserted next to fresh assemblies
    - Increased hydriding on control blade side causes substantial bowing

Westinghouse BWR Channel Development

- Major emphasis on channel material and process development

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BWR Channel Bow – Symmetric Lattice

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PWR assembly growth

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BWR Channel Bow – ZIRLO™ Potential

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BWR Channel Bow – Shadow Corrosion

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BWR Channel Bow – SVEA Channel Hot Cell Exam

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BWR Channel Bow – Summary

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Debris Fretting - Summary

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Debris Fretting – Spacer Concept

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Debris Fretting – Oxide Protection

- Recognized that an oxide layer provides wear/fretting protection

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Debris Fretting – Cladding Corrosion

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BWR Secondary Failure Mitigation

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BWR Stability: Option 3 – PS-DIVOM

- Recent events illustrate need for Detect and Suppress (DSS)
  - Oskarshamn 2, NMP2
- Application of NEDO-32465
  - Protects SLMCPR
  - Plant-specific (PS) DIVOM since Generic DIVOM can be inadequate
  - BWROG, GNF-GE, AREVA and Westinghouse jointly developed Guideline Procedure to assure uniform application of the licensed methodology
  - Application process to be established by each vendor

BWR Stability: Option 3 – PS-DIVOM

- Westinghouse will use codes currently licensed for reload licensing stability and hot channel transient CPR calculations:

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BWR Water Chemistry

- Enhanced Spacer Shadow Corrosion
  - FE/(ZN+NI) ratio
- Accelerated Corrosion – Enhanced Nodular Corrosion
  – Condenser Leak: Specific Chemistry cause not established
- Crud Induced Failures
  – High concentrations of Zn and Cu
- Chemical Incursion during first cycle
  – Alloy content, high duty after event

BWR Water Chemistry - Lessons

- Careful control – avoid chemical incursions
- Eliminate Cu. Control Zn, Fe, Ni
- As much fuel surveillance as possible
  - Data to support understanding
- Try to understand – phenomenological models
- Focus on corrosion resistant cladding
- Avoid excessive crud
- Encourage as much cross cooperation as possible
  - Vendors, utilities, EPRI, European Experiences, etc.

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Westinghouse BWR Nuclear Design Tools

NRC/Westinghouse Meeting
Monroeville, PA
June 23, 2004

Westinghouse BWR Nuclear Design Tools

Contents

- Background
- Code Package
- Validation effort

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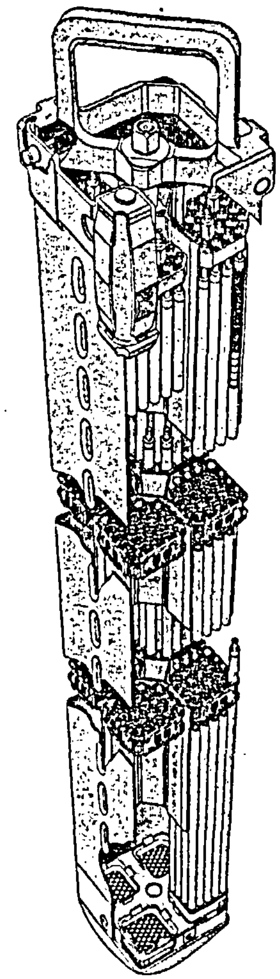
- Experience from core follow
- Summary & conclusions

Background

The introduction of the SVEA-96 Optima generation of fuel designs implies a further increment in complexity and heterogeneity, challenging the capabilities of the reload design and core monitoring tools.



Westinghouse is engaged in a continuous improvement process of its PHOENIX-4 / POLCA7 calculation package to support the most advanced BWR fuel designs.

Code Package

Main Components

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- Lattice code: PHOENIX4, 2D transport code with burnup
- Core Simulator: POLCA7, advanced 3D steady-state nodal code for nuclear design & core monitoring
- Work environment: Core Master 2 (CM2), integrated system for core calculations and analyses.

Code Package

- Stability/RIA calculations: RAMONA - 3D core analysis
- Transient calculations: BISON - plant analysis (1D core model)

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Code Package

PHOENIX-4

- 2D Lattice calculations
  - Multi-group collision probabilities + discrete ordinates
  - Depletion & branches

POLCA7

- 3D Core Calculations
  - 2-group standard Analytical Nodal Model
  - Axial homogenization (1D calculation)

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Code Package

POLCA7 (Cont.)

- Pin Power Reconstruction
- Advanced XS representation

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Code Package

POLCA7 (cont.)

- BWR Thermal Hydraulics

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Validation effort

Predictions at fuel rod level

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- LWR-PROTEUS experiments (at PSI, Switzerland):
  - Unique experimental set-up for validation of 2D codes
  - LWR-PROTEUS Phase 1 project: NOK, PSI, W

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  - Measurements at different conditions

LWR-PROTEUS Critical Experiments

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LWR-PROTEUS Critical Experiments

Configurations evaluated:

- Reference (unperturbed)
- Control blades in central position (Hf/B_4C)

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Evaluations presented (from fuel rods gamma-scan)

- (Ftot) Total fission rate (= power dist.) ~ Thermal range
- (C8) Capture rate U238 ~ Epithermal range

LWR-PROTEUS Critical Experiments

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LWR-PROTEUS Critical Experiments

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Validation effort

Predictions at individual fuel bundle level

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  –
  – Representative of the last 7-8 weeks of operation
  – Validation of bundle power predictions beyond the limitations of the TIP-based comparisons

Predictions at fuel bundle level

Latest gamma-scan campaigns particularly interesting

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Predictions at fuel bundle level

Evaluation conditions

- Calculations
  - Standard PHOENIX4/POLCA-7 calculations
  -
  -

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- Comparisons
  - Measurements & calculations normalized to same value
  - Nodes affected by structural materials excluded

Gamma-scan at [ ]a, c

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Gamma-scan at [ ][a, c]

[ ][a, b, c]

Gamma-scan campaigns

Overall Statistics

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Experience from Core Follow

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Cycle averages:

- Hot reactivity level
- Cold reactivity level
  - global and local reactivity measurements
  - local reactivity vs. fraction of bundles of new design

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Experience from core follow

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Experience from core follow

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Experience from core follow

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Experience from Core Follow

a, b, c

Experience from Core Follow

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- Hot reference level
- Cold reference level

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Reliable Tools to Model Advanced SVEA Fuel Designs

Summary

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-
  - demonstrates the reliability of PHOENIX-4 and points out areas for further modeling improvements.

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-
  - prove satisfactory performance of POLCA7 under quite different conditions.
- Overall core follow experience
  - demonstrates the reliability of PHOENIX4/POLCA7 to predict conditions during transition cycles.

Reliable Tools to Model Advanced SVEA Fuel Designs

Conclusions

- The Westinghouse methods and methodology fully support the transition to the new BWR fuel designs.
- For monitoring purposes, the very same methods are utilized.
- The improved performance of the SVEA-96 Optima fuel designs , as originally predicted by these modeling tools, is actually achievable.

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CHF Test Facility Status Update

NRC/Westinghouse Meeting
Monroeville, PA
June 2004

Westinghouse DNB Testing

Requirements of DNB Testing:

- Westinghouse uses NRC-approved FCEP (WCAP-12488-A) for evaluating fuel design changes

- Fuel changes not covered by FCEP may require DNB tests
  - [ ]a, c
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  - [ ]a, c

Westinghouse DNB Testing

Westinghouse Qualification Process:

- Over the years, Westinghouse has developed a process for qualifying test facilities and data for safety analyses
  - [ ]a,c
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  - [ ]a,c
- The process was applied to Columbia HTRF and SKODA test loop for VVER-1000 applications
- The process will include initial benchmarking of an existing data set
- This same process will be applied to the new test facility in [ ]a,c
- NRC participation during qualification DNB testing of the facility is welcomed

Westinghouse DNB Testing

Westinghouse New DNB Test Facility:

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[ ]

Oden DNB Loop – Västerås

Westinghouse New DNB Test Facility:

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PWR Core Technologies (PCT) Training Update

NRC/Westinghouse Meeting
Waltz Mill, PA
June 24, 2004

Agenda

- PCT Internal Training Initiatives
  - Motivation
  - Ongoing Programs
- Current Internal/External Course Offerings
  - Course Catalog and Customized Courses

Training Initiatives in PCT -- Why

a, c

Current PCT Qualification Matrix (sample)

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Training Initiatives in PCT -- TAC

- The Training Advisory Committee (TAC) was constituted in PCT to oversee all training to ensure consistency and effectiveness of training.

- TAC will enforce the use of the Systematic Approach to Training (SAT) for all training products developed by PCT.

- TAC will review upcoming training (internal seminars, for example) to ensure it is provided effectively and consistently.

- TAC is sponsoring initiatives for developing new training system for qualifying and re-qualifying designers.

Systematic Approach to Training

Essential aspects of the SAT Methodology

1. Analysis Phase

   Systematic analysis conducted to determine job requirements and need for training

2. Design Phase

   Training objectives derived from the results of systematic analysis which describes desired performance after training

3. Development Phase

   Training programs are designed to meet the specified objectives selected for training

4. Implementation Phase

   Student progress and mastery of objectives evaluated during training

5. Evaluation Phase

   Revise/upgrade training programs based on analysis of students' ability to perform successfully on the job after training

Training Initiatives in PCT – Qual Cards

- First training initiative is to develop qualification tracking cards, "qual cards," for all disciplines in PCT, including Nuclear Design, T/H Designer, Fuel Rod Designer, Licensing, and Software.
- "Qual cards" track competencies achieved by trainees and individual competencies are signed off by qualified designers upon completion of activities by trainees.
- US Navy and many US utilities use "qual cards" for qualification tracking.
- Pilot program underway to develop Job Task Analysis (JTA) for Nuclear Designers, to be followed by other disciplines. JTAs list Knowledges required by trainees and Activities they must perform to be qualified. In following examples, "K" indicates a Knowledge and "A" an Ability.

Core Designer – High Level JTA

CORE DESIGN (W PWR)
•**Core Design Basics** •**Core Reload** • Design Initialization • Loading Pattern Development • Reload Safety Evaluation • Nuclear Design Report (NDR) and Operational Data • Core Follow •**Uprating/New Core** •**Software Support** •**Licensing Support**

Job Task Analysis (example)

QUAL 6: Reload Safety Analysis Checklist (RSAC) Shutdown Margin Analyses

- K6.1.1 - Understand the following METCOM Sections:
  - Section 6.9 Control Rod Operational Limits
  - Section 6.10 Shutdown Margin
  - Section 6.11 Trip Reactivity
- K6.1.2 - Know how to run shutdown margin (SDM) and trip worth simulations in ANC and/or APOLLO.
- A6.1.1 - Perform, document, and verify Reload Safety Analysis Checklist (RSAC) calculations for rod insertion allowance and/or shutdown margin in accordance with METCOM.
- A6.1.2 - Perform, document, and verify Reload Safety Analysis Checklist (RSAC) calculations for trip reactivity and trip shape, in accordance with METCOM.

Core Designer – Qualification Card (example)

Qualification Card Evaluation Criteria

Evaluation Criteria	Skill Demonstrated
1	Reading
2	Discussion
3	Demonstration
4	Simulation/Shadowing
5	Training
6	Testing
7	Other

Core Designer – Qualification Card (example)

QUAL6:	RSAC	Perform engineering job performance requirements associated with Reload Safety Analysis Checklist (RSAC) Shutdown Margin Analyses.

Step	Description	Activities Performed by Trainee to Demonstrate Knowledge or Ability	Evaluator Initial/Date
1	K 6.1.1 – Understand METCOM Sections 6.9, 6.10, and 6.11. *Evaluation Criteria [1,2,3, or 4]*		____/____
2	K 6.1.2 – Know how to run simulations in ANC and APOLLO. *Evaluation Criteria [1,2,3, or 5]*		____/____
3	A 6.1.1 – Perform, document, and verify RIA and/or SDM calculations. *Evaluation Criteria [1,2,3, or 6]*		____/____
4	A 6.1.2 – Perform, document, and verify trip reactivity and trip shape calculations. *Evaluation Criteria [1,2,3, or 5]*		____/____
5	**Evaluator Justification for Signoff:** *The evaluator must use the suggested evaluation criteria from Steps 1 through 4 along with any other criteria deemed appropriate to document justification for approving qualification for the candidate.*		

EVALUATED BY:____________________________ **DATE:**__________
(EVALUATOR)

The Evaluator confirms the trainee has successfully demonstrated the knowledges and abilities associated with the specified qualification number and is qualified to independently author all calculations.

Core Designer – Qualification Card (example)

Qualification Summary

QUAL NO.	Title	Page	Date
D.I.			
1	External Design Initialization		
2	Internal Design Initialization		
LP			
1	Loading Pattern Scoping Fundamentals and Fuel Cycle Economics		
2	Cross-Section Development and ANC Model		
3	Provide Fuel Management Scoping with ALPS		
4	Generate a Loading Pattern, Cross-Sections, and ANC Model with CoreXpert		
5	Perform Loading Pattern Risk Assessment		
6	Perform ALFRED Analysis (Fuel Rod Design Data)		
RSAC			
1	Reload Safety Analysis Checklist (RSAC) Plan Development		
2	Reload Safety Analysis Checklist (RSAC) Core Model Development		
3	Reload Safety Analysis Checklist (RSAC) Reactivity Coefficient Calculation		
4	Reload Safety Analysis Checklist (RSAC) Beta-Effective Calculation		
5	Reload Safety Analysis Checklist (RSAC) Peaking Factors		
6	Reload Safety Analysis Checklist (RSAC) Shutdown Margin Analyses		
7	Reload Safety Analysis Checklist (RSAC) Constant Axial Offset Control (CAOC) Analysis		
8	Reload Safety Analysis Checklist (RSAC) Relaxed Constant Axial Offset Control (RAOC) Analysis		
9	Reload Safety Analysis Checklist (RSAC) Rod Accident Analyses		
10	Reload Safety Analysis Checklist (RSAC) Boron Dilution and Steamline Break Analyses		
11	Reload Safety Analysis Checklist (RSAC) LOCA and BORDER Analyses		
12	Reload Safety Analysis Checklist (RSAC) CXRSAC Execution		

Training Courses

- Training Catalog Lists Standard Courses Offered
  - Courses scheduled based on demand
  - Advertise courses once tentative dates established
  - Require minimum number of students to justify course offerings

- Development of New Courses
  - Customer specific training requests
    - Course curriculum developed jointly with utility
    - Plant specific focus
    - Training provided at Westinghouse or Utility site
  - New Technology or Industry Needs
    - BOB Training
    - SOER training
    - BWR Core Design Methodology

Training Courses - External

2004 NUCLEAR FUEL TRAINING COURSE SCHEDULE

	Course Number	Course Title	Dates Offered
Advanced Technology Courses	AFM100	Advanced Fuel Management	Scheduled upon request
	• AFM110	Module 1: Code Review and Model Preparation (2 days)	
	• AFM120	Module 2: Fuel Cycling Evaluation and Economics (2 1/2 days)	
	• AFM130	Module 3: Fuel Management Analysis (3 1/2 days)	
	• AFM140	Module 4: Loading Pattern "Tool Box" (4 days)	
	SOI100*	Implications of Design and Economic Decisions on Safety Analyses and Operational Issues (5 days)	Scheduled upon request
	MET100	Advanced Applications Methodology (5 days)	Scheduled upon request
Nuclear Design	ND200*	Nuclear Design Technology and Methods (15 days)	June 7-25, 2004
	• ND210	Module 1: Nuclear Design Code Systems and Models (5 days)	June 7-11, 2004
			Feb. 9-12 in Windsor
	• ND220	Module 2: Fuel Management (2 1/2 days)	June 14-16, 2004
	• ND230	Module 3: Nuclear Design Safety Evaluation Process (4 days)	June 16-22, 2004
	• ND240	Module 4: Plant Operations Calculations (3 1/2 days)	June 22-25, 2004
	ND300*	Operations Core Analysis Package (OCAP) (4 days)	Jan. 6-9, 2004

* Courses may help in meeting requirements of INPO SOER 03-2.

PCT Training Courses - External

2004 NUCLEAR FUEL TRAINING COURSE SCHEDULE (continued)

	Course Number	Course Title	Dates Offered
Thermal- Hydraulic Design	TH200*	Thermal-Hydraulic Design Methods (5 days)	Aug. 9-13, 2004
Fuel Rod Design	FR200*	Fuel Rod Design (3 days)	June 15-17, 2004
Nuclear Fuel Design	INT100	Introduction to Nuclear Fuel Design (3 days)	Scheduled upon request
Coolant Activity Analysis	CA200*	Coolant Activity Analysis (4 days in Columbia, SC)	April 26-29, 2004
Quality Assurance	QA100	Product and Software Design Control Process (1 day)	Scheduled upon request
Reactivity Management	REA200*	Reactivity Management During Refueling Operations (2 days)	Scheduled upon request
	REA300*	Reactivity Management During Plant Operations (5 days)	Scheduled upon request
Spent Fuel Management	SF100	Dry Spent Fuel Management (2 days)	Scheduled upon request
Reactor Engineering	ENG100*	Reactor Site Engineer Course (5 days)	Scheduled upon request
Software Engineering	SE100	Basic HP-UX Turnkey System Administration (2 days)	Scheduled upon request
	SE200	Overview of HP-UX Workstation and User Environment (2 days)	
	SE300	Software Engineering Methodology Course (2 days)	
	SE400	Fundamentals of Shell Programing (3 days)	
Station Nuclear Engineering	SNE594	Station Nuclear Engineer Applications	July 12-30, 2004

* Courses may help in meeting requirements of INPO SOER 03-2.

PCT Training Courses - External

2004 NUCLEAR FUEL TRAINING COURSE SCHEDULE (continued)

	Course Number	Course Title	Dates Offered
Core Design Training Center	NDTC1000*	Nuclear Core Design Training Center	
	NDTC1010	Fuel Management Scoping (5 days)	Scheduled upon request
	NDTC1020	Loading Pattern Scoping (5 days)	Scheduled upon request
	NDTC1030	Cross-Section Development (5 days)	Scheduled upon request
	NDTC1040	Preliminary Model Generation (5 days)	Scheduled upon request
	NDTC1050	Final Loading Pattern Search (15 days)	Scheduled upon request
	NDTC1060	Nuclear Design Safety Evaluation Models (15 days)	April 12-30, 2004
	NDTC1070	Nuclear Design Safety Evaluation Calculations (35 days)	May 3 – June 21, 2004
	NDTC1080	Nuclear Design Calculations (20 days)	June 22 – July 20, 2004
	NDTC1090	Core Follow Calculations (5 days)	Scheduled upon request
	THDTC1000**	Thermal-Hydraulic Design Training Center	Scheduled upon request
	THDTC1010	Hydraulics (5 days)	
	THDTC1020	Core Components Design (5 days)	
	THDTC1030	Operating Limits (15 days)	
	THDTC1040	Safety Analysis (15 days)	
	FRDTC1010**	Fuel Rod Design Training Center	Scheduled upon request
	FRDTC1010	Rod Internal Pressure (5 days)	
	FRDTC1020	Stress/Strain/Fatigue (9 days)	
	FRDTC1030	Corrosion (4 days)	
	FRDTC1040	Rod Growth (2 days)	
	FRDTC1050	Fuel Temperatures (10 days)	

Core Monitoring and Surveillance	BCN100*	BEACON System (up to 10 days)	Scheduled upon request
	BCN200	BEACON Model Generation (up to 11 days)	Scheduled upon request
	MUN100	Measurement Uncertainty Training (1 day)	Scheduled upon request
	BCN300*	BEACON Special Topics	Scheduled upon request
	TRK200*	TracWorks System (2 days)	Scheduled upon request

PCT Training Courses - External

2004 NUCLEAR FUEL TRAINING COURSE SCHEDULE (continued)

	Course Number	Course Title	Dates Offered
Overview Courses	ND100	Nuclear Design Overview (5 days)	Scheduled upon request
	TH100	Thermal-Hydraulic Design Overview (2 days)	Scheduled upon request
	FR100	Fuel Rod Design Overview (1 day)	Scheduled upon request
	CA100	Introduction to Coolant Activity Analysis (1 day at Columbia)	Scheduled upon request
	DI100	Reload Design Interface Process (3 days)	Scheduled upon request
	NFC100*	Nuclear Fuel Cycle (4 days)	Scheduled upon request
	RSE100*	Reload Safety Evaluation Process (4 days)	Scheduled upon request

*Courses may help in meeting requirements of INPO SOER 03-2.

PCT Training - Integration with Nuclear Services

- PCT works jointly with Westinghouse Nuclear Services (NS) in the determination of acceptable fuel operation from a Chapter 15 Transient Analysis perspective.
- NS is working closely with PCT to determine best training practices for implementation.
- Similar to PCT, NS is seeing an increase in training needs as a result of work force demographics.
- NS has a number of widely-used training courses available for the industry.

Transient Analysis Training Courses - External

- Non-LOCA Overview Training
  - 3-Day Course (up to 2 weeks if hands-on Code Instruction is requested)
  - Overview of Westinghouse Transient Analysis Methods & Codes
  - Overview of each non-LOCA accident with discussions on:
    - Acceptance criteria
    - Cases analyzed
    - Key parameters
    - Protection
    - Analysis assumptions
    - Results (Including transient plots and a sequence of events)

Transient Analysis Training Courses - External

- General LOCA Training
  - 1 or 2 Day Course
  - Overview of Westinghouse Methods & Codes
    - Large Break/Small Break LOCA
  - Overview of LOCA Accident Analyses
    - Acceptance criteria
    - Cases analyzed
    - Key parameters
    - Protection
    - Analysis assumptions
    - Results (Including transient plots and a sequence of events)
  - Course can be tailored to specific interests

Transient Analysis Training Courses - External

- Best – Estimate LOCA College
  - Semester 1 - W Best Estimate LOCA Fundamentals
    - 10-12 Modules
    - 3 or 4 days
  - Semester 2 – Two-Phase Flow and Boiling Heat Transfer
    - 6 Modules
    - 2 or 3 days
  - Semester 3 – WCOBRA-TRAC Validation Results
    - 6-8 Modules
    - 3 days
  - Semester 4 – Numerical Methods / Concepts
    - 6 Modules
    - 2 or 3 days

Transient Analysis Training Courses - External

- Non-LOCA and LOCA Overview Training
  - Performed for plant personnel from:

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