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BWR SAFETY ANALYSIS VENDOR INDEPENDENCE PROGRAM
NRC/CECo MEETING MAY 5, 1994

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ENCLOSURE 2

Commonwealth Edison Company
BWR Safety Analysis Vendor Independence Program

NRC/CECo Meeting

May 5, 1994

Terry Rieck
John Freeman
Bob Tsai
Hossein Youssefnia

Nuclear Fuel Services
Commonwealth Edison Company

BWR SAFETY ANALYSIS VENDOR INDEPENDENCE PROGRAM
NRC/CECo Meeting - 5/5/94

AGENDA

1. Introduction
Review Agenda, Meeting Objectives

2. Update on Submittal Schedule
Transient, Core Thermal Limit & Reload Application Topicals

3. Methodology Overview
Approach, Computer Codes Used, Planned Topical Reports & Review of 9/23/93 Meeting

4. Training and Quality Assurance
Internal & External Trainings, Quality Assurance Program

5. Transient Analysis Topical Update
Preliminary Results on Core Thermal Hydraulics and System transients

6. Summary
CECo Summary

7. NRC Discussions
NRC Feedback on CECo BWR SA VIP

MEETING OBJECTIVES

To update NRC on CECo's Plan, approach & schedule for the CECo BWR Safety Analysis Vendor Independence Program (VIP).

To obtain NRC feedback on CECo plan for BWR SA VIP.

UPDATE OF SUBMITTAL SCHEDULE

- Transient Analysis Method Topical

• Methods, Models & Quad Cities, Dresden & Peach Bottom Benchmarking	12/94
• LaSalle Results Supplement	2Q/95
• NRC/CECo Meeting	2Q/95
• NRC Audit, If Necessary	3Q/95
• SER Requested	12/95

- Core Thermal Limit & Reload Application Topical

• Topical Submittal	06/96
• NRC/CECo Meeting	3Q/96
• NRC Audit, If Necessary	4Q/96
• SER Requested	06/97
• Start of D3C16 Analyses	06/97
• D3C16 Startup	06/98

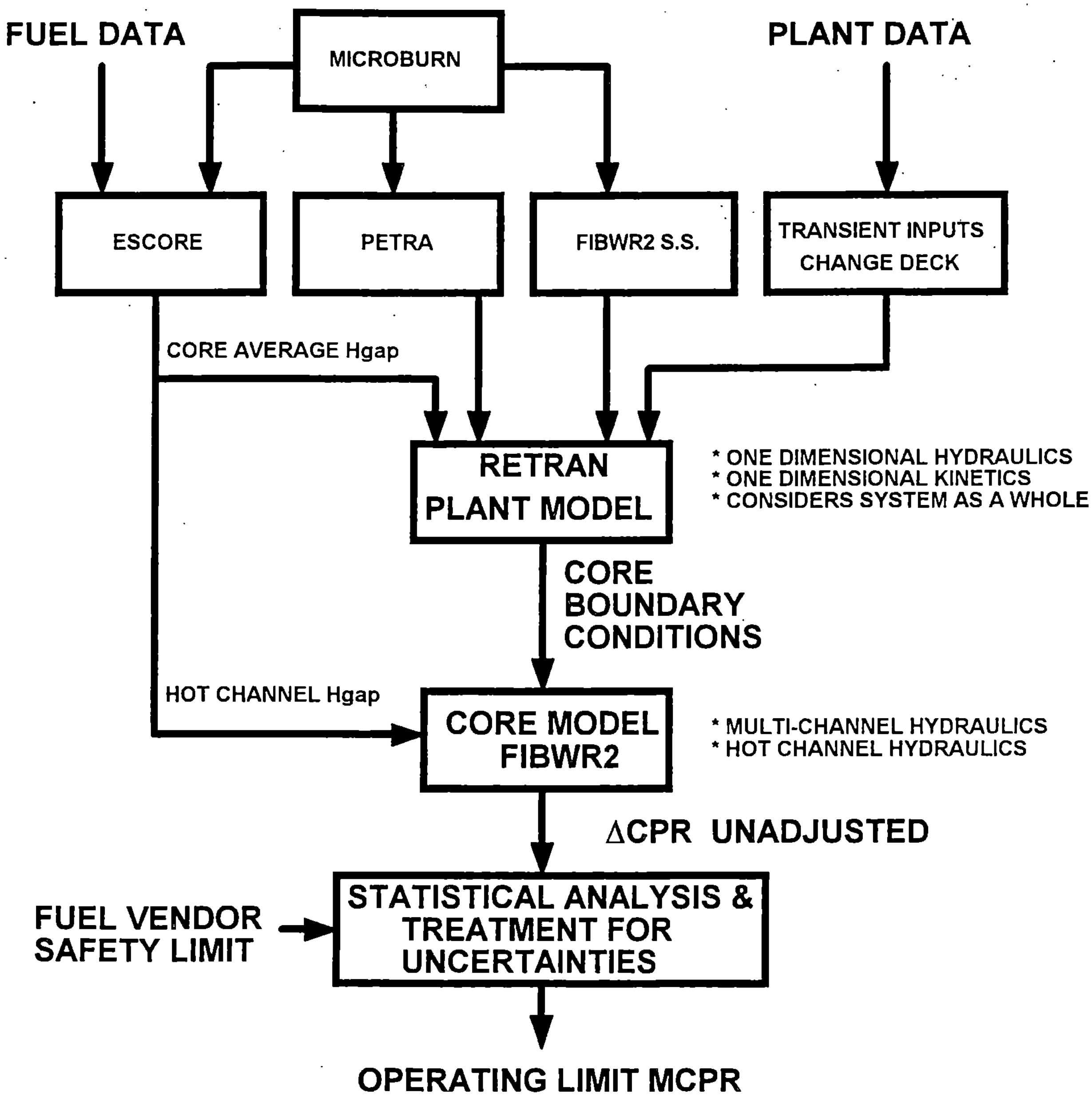
METHODOLOGY OVERVIEW

- In-House Neutronics Analysis: Vendor (SPC) Nuclear Design Methods
  - CASMO/MICROBURN
  - NRC Approved CECo Use in 03/93
- Safety Analysis: CECo/EPRI Methods
  - PETRA/RETRAN/FIBWR2
  - Other Utility Topicals Approved by NRC

    TVA, YAEC, GPUN, PECo, PP&L (GSU, WPSS)

BWR SAFETY ANALYSIS VENDOR INDEPENDENCE PROGRAM
NRC/CECo Meeting - 5/5/94
(Methodology Overview)

METHODOLOGY OVERVIEW



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9/23/93 NRC/CECo MEETING REVIEW

1. NRC/CECo Interaction on Submittal Plan & Schedule

2. Treatment of Uncertainties

3. Computer Code Applicability (conditions in SERs)

4. Thermal Limit Methodology Benchmarking

5. Mixed Core Effects in the Application Topicals

TRAINING & QUALITY ASSURANCE

- CECo Internal Training
    - Computer Code Applications
    - Reactor Systems
    - Nuclear Power Plant Operation & Simulator
- External Training
    - Design Participation Training Programs (fuel vendor assignments)
    - Fuel Vendor Reload Safety Analysis Courses
    - Computer Code Vendor Workshops
    - Industry Workshops
- Quality Assurance Program
    - Computer Code Certification
    - Perform, Review & Approval of Controlled Analysis per QP's & Department Procedures

- Core Thermal Hydraulics Model Development & Benchmarking

  - FIBWR2 Summary
  - CECo FIBWR2 Applications
  - Steady State Analysis Qualification
  - CPR Correlation Implementation & Benchmarking
  - Core Thermal Hydraulic Summary

FIBWR2 Summary

History

- Development Began in 1988
  Sponsored by the FIBWR2 Owner's Group

- Steady State: Extended Capabilities of FIBWR Code
  Funded by Yankee Atomic Electric Company
  First Review by NRC early 1980's (Part of YAEC's Submittal)

- Transient: BWR Core Model Under Non-LOCA Conditions

Some Important Features

- Predicts Flow Distribution for a Given Power Distribution
  Uses Total Core Flow or Pressure Drop Condition

- Axially Varying Flow Geometry

- Fuel Rod Model

- Bypass & Water Rod Model

- Transient Multi-Channel CPR Calculation

- Linkage to RETRAN Code
  Allows Time Varying Axial Power Shape

CECo FIBWR2 Applications

Steady State Core Thermal Hydraulics

- Calculates Inputs to Initialize RETRAN System Model
  - Core Equivalent Nodal Pressure Drop
  - Total Bypass Flow
  - Channel Dependent Flow

Transient Core Thermal Hydraulics

- Single Hot Channel Model
  - ΔCPR Calculation
  - Multi-Pin Heat Flux and LHGR Edits
- Multi-Channel CPR Calculations
  - Mixed Core Loading

Steady State Analysis Qualification

- Comparison to FIBWR
  - Quad Cities Cycle 1 Core Model
  - Dresden Cycle 1 Core Model
- Comparison to GE's Steady State Thermal Hydraulic Code (ISCOR)
  - Quad Cities Cycle 1 Core Model
  - Dresden Cycle 1 Core Model
- Coding of GE Correlation (GEXL) in FIBWR2
  - MCPR Prediction vs. GE's Method (Quad Cities Cycle 1)

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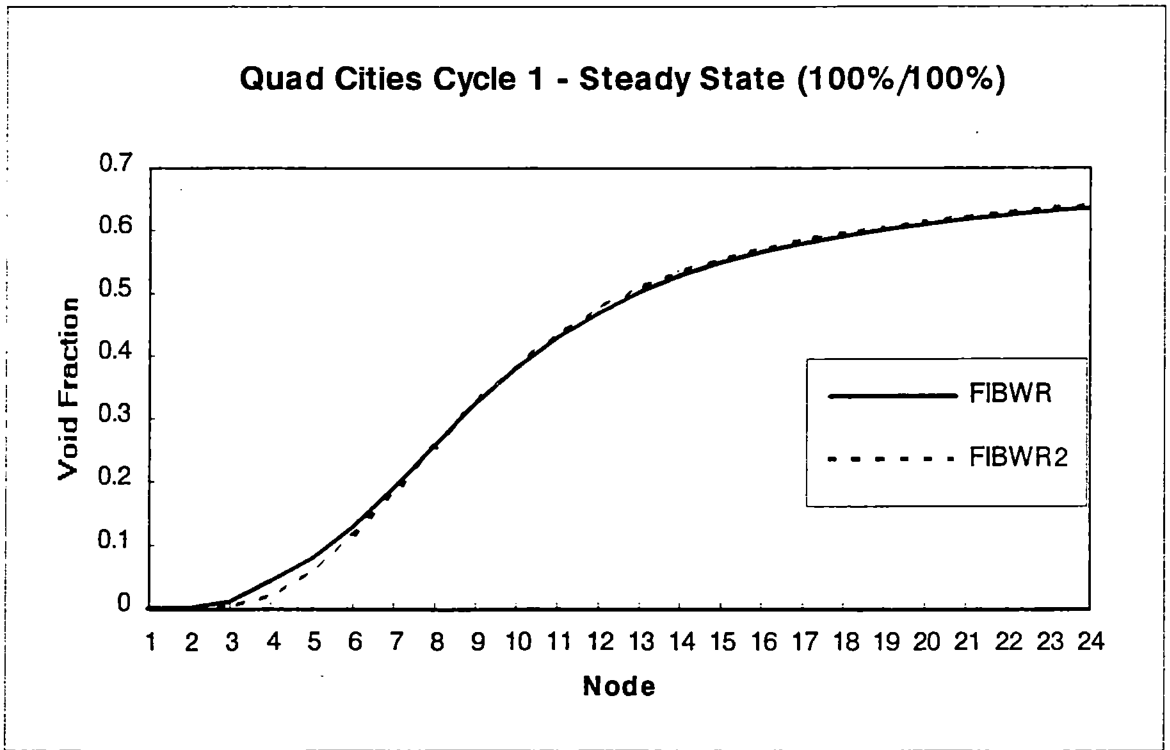
FIBRW2 VS. FIBWR RESULTS

Example 1 Quad Cities Cycle 1: 100% power/100% flow (Central Orificing)

	FIBWR2	FIBWR	DIFFERENCE
Core dp (psi)	21.5602	21.6978	-0.1376
Core Plate dp (psi)	16.9917	17.1455	-0.1538
Bypass Fraction	0.1119	0.1105	0.0014
Average Void	0.3839	0.3848	-0.0009
HEATED CHANNEL			
Friction dp (psi)	2.7746	2.7492	0.0254
Elevation dp (psi)	2.494	2.4854	0.0086
Local dp (psi)	3.8868	3.9009	-0.0141
Acceleration dp (psi)	0.8187	0.8135	0.0052
Total dp (psi)	9.9741	9.949	0.0251
Non-Boiling Length (in.)	9.1428	8.9942	0.1486



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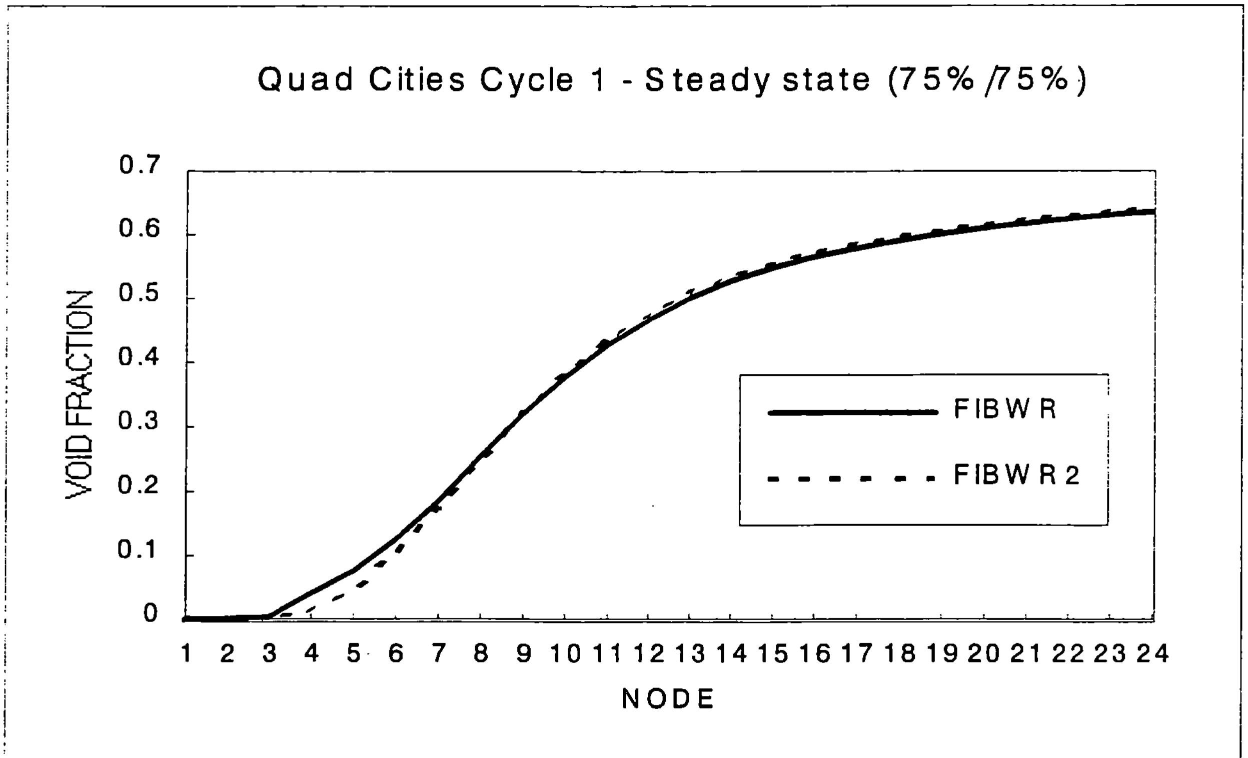
FIBRW2 VS. FIBWR RESULTS

Example 2 **Quad Cities Cycle 1: 75% power/75% flow (Central Orificing)**

	FIBWR2	FIBWR	DIFFERENCE
Core dp (psi)	13.7614	13.7855	-0.0241
Core Plate dp (psi)	9.1718	9.2154	-0.0436
Bypass Fraction	0.1009	0.0933	0.0076
Average Void	0.3797	0.3816	-0.0019
HEATED CHANNEL			
Friction dp (psi)	1.7574	1.7465	0.0109
Elevation dp (psi)	2.5146	2.5055	0.0091
Local dp (psi)	2.2298	2.2515	-0.0217
Acceleration dp (psi)	0.4717	0.47	0.0017
Total dp (psi)	6.9735	6.9735	0
Non-Boiling Length (in.)	12.4664	12.5414	-0.075



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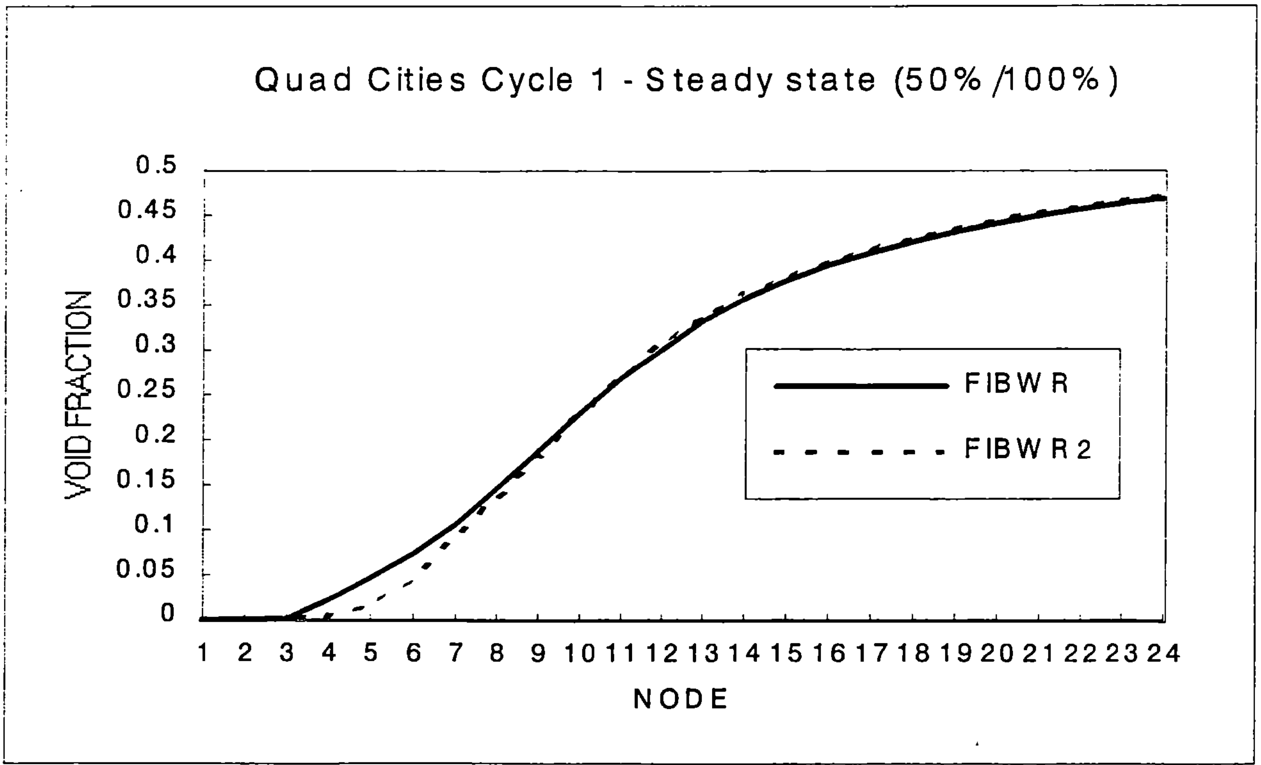
FIBRW2 VS. FIBWR RESULTS

Example 3 Quad Cities Cycle 1: 50%power/100% flow (Central Orificing)

	FIBWR2	FIBWR	DIFFERENCE
Core dp (psi)	20.0475	20.1262	-0.0787
Core Plate dp (psi)	15.4736	15.5669	-0.0933
Bypass Fraction	0.095	0.095	0
Average Void	0.2621	0.2645	-0.0024
HEATED CHANNEL			
Friction dp (psi)	2.1246	2.1087	0.0159
Elevation dp (psi)	2.9614	2.9433	0.0181
Local dp (psi)	2.9655	2.9708	-0.0053
Acceleration dp (psi)	0.4431	0.4386	0.0045
Total dp (psi)	8.4946	8.4614	0.0332
Non-Boiling Length (in.)	14.8592	14.8177	0.0415



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FIBRW2 VS. FIBWR RESULTS

Example 1 Dresden Cycle 1: 100% power/120% flow (Central Orificing)

	FIBWR2	FIBWR	DIFFERENCE
Core dp (psi)	28.3621	28.5676	-0.2055
Core Plate dp (psi)	23.802	24.0342	-0.2322
Bypass Fraction	0.1152	0.1153	-0.0001
Average Void	0.3535	0.3542	-0.0007
HEATED CHANNEL			
Friction dp (psi)	3.4479	3.4163	0.0316
Elevation dp (psi)	2.6062	2.5958	0.0104
Local dp (psi)	4.9909	4.998	-0.0071
Acceleration dp (psi)	0.9928	0.9856	0.0072
Total dp (psi)	12.0378	11.9957	0.0421
Non-Boiling Length (in.)	8.7511	8.5863	0.1648

Example 2 Dresden Cycle 1: 100%power/75% flow (Central Orificing)

	FIBWR2	FIBWR	DIFFERENCE
Core dp (psi)	14.3506	14.3077	0.0429
Core Plate dp (psi)	9.7676	9.7456	0.022
Bypass Fraction	0.1014	0.101	0.0004
Average Void	0.4325	0.4356	-0.0031
HEATED CHANNEL			
Friction dp (psi)	2.074	2.0408	0.0332
Elevation dp (psi)	2.3188	2.2992	0.0196
Local dp (psi)	2.5061	2.5564	-0.0503
Acceleration dp (psi)	0.6156	0.6089	0.0067
Total dp (psi)	7.5145	7.5053	0.0092
Non-Boiling Length (in.)	10.0651	9.8424	0.2227

Example: FIBWR2 - ISCOR Comparison for Quad-Cities Cycle 1
(Full Power & Flow Conditions)

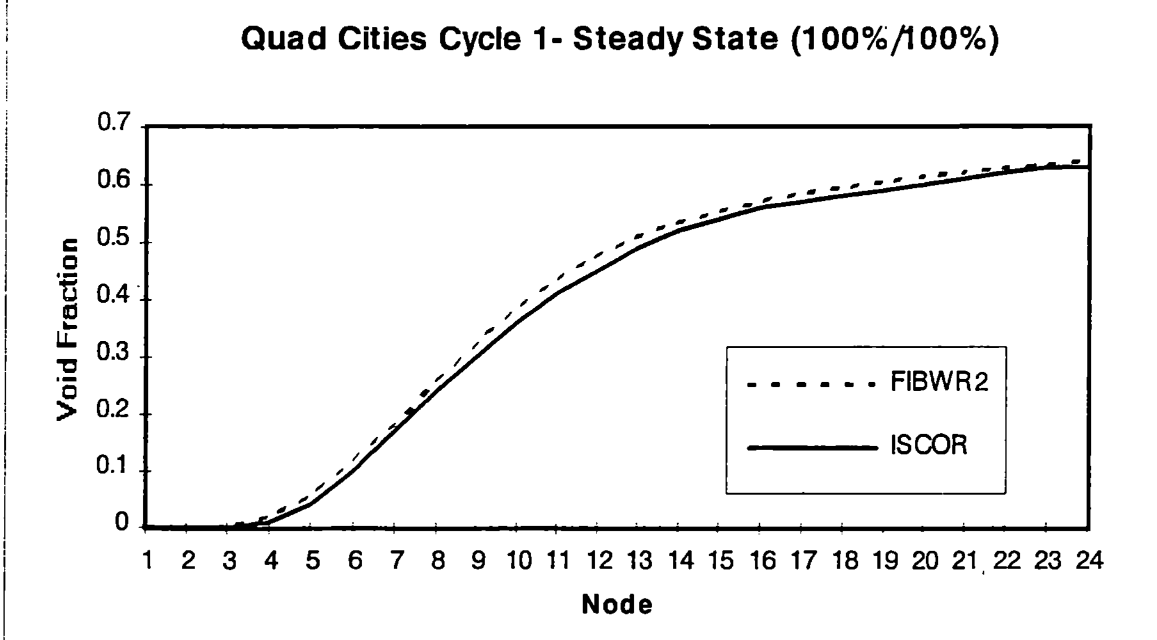
Variable	FIBWR2	ISCOR	Difference	Units
Total Bypass Flow	10.78	11.21	-0.43	Mlb/hr
Plenum to Plenum Press.	21.53	21.64	-0.11	psi
Total Exit Steam Flow	9.76	9.63	0.13	Mlb/hr
Core Avg Void Fraction	0.65	0.64	0.01	

Pressure Drop Components: Central Orificing

Variable	FIBWR2	ISCOR	Difference	Units
Friction	3.18	3.32	-0.14	psi
Elevation	3.07	2.70	0.37	psi
Local Losses	15.09	15.07	0.02	psi
Acceleration	0.18	0.55	-0.37	psi

Pressure Drop Components: Peripheral Orificing

Variable	FIBWR2	ISCOR	Difference	Units
Friction	1.36	1.33	0.03	psi
Elevation	2.80	2.47	0.33	psi
Local Losses	17.50	17.60	-0.10	psi
Acceleration	0.13	0.25	-0.12	psi



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Example: Steady State MCPR Calculation FIBWR2 vs. ISCOR (Quad Cities Cycle 1 Core)

	Average Channel (Central Orificing)		Difference
Power/Flow %	FIBWR2	ISCOR	FIBWR2-ISCOR
100/100	1.92	1.92	0.00
50/100	3.77	3.77	0.00
50/50	3.13	3.12	0.01
100/120	1.97	1.97	0.00
100/50	1.60	1.60	0.00
75/50	2.11	2.11	0.00
25/75	6.99	6.99	0.00

	Average Channel (Peripheral Orificing)		Difference
	FIBWR2	ISCOR	FIBWR2-ISCOR
100/100	2.24	2.24	0.00
50/100	4.43	4.41	0.02
50/50	3.46	3.43	0.03
100/120	2.36	2.37	-0.01
100/50	1.82	1.81	0.01
75/50	2.37	2.36	0.01
25/75	7.82	7.75	0.07

	Hot Channel (Central Orificing)		Difference
	FIBWR2	ISCOR	FIBWR2-ISCOR
100/100	1.32	1.31	0.01
50/100	2.64	2.62	0.02
50/50	2.15	2.13	0.02
100/120	1.37	1.36	0.01
100/50	1.06	1.05	0.01
75/50	1.42	1.41	0.01
25/75	4.90	4.88	0.02

CPR Correlation Implementation & Benchmarking

- Implementation of the SPC's Correlation (ANF-B)
  - Application within the Correlation Limits
- Comparison of Calculated MCPR vs. Vendors' Method
  - Steady State
  - Transient
- Benchmarking against Appropriate Test Data
- Qualification of the Vendors' Correlations in a Mixed Core Loading

Core Thermal Hydraulic Summary

- CECo FIBWR2 Qualified for RETRAN Initialization
  - Comparison to FIBWR
  - Comparison to GE's Steady State Thermal Hydraulic Method
- GE Correlation (GEXL) Coding Tested in FIBWR2
  - Comparison to GE's Steady State MCPR Predictions
- CPR Correlation Implementation and Benchmarking
  - Comparison to Vendors' Methods (Steady State & Transient)
  - Benchmarking against Appropriate Test Data
  - Qualification of a mixed Core Loading

Transient Analysis Model Development & Benchmarking

- **Model Development Status**
- **Benchmarking Results**
  - **-Quad-Cities Startup Test**
  - **-Peach Bottom Turbine Trip**
- **Contractor Review of CECo's One Dimensional Neutronics Collapsing Methods**

MODEL DEVELOPMENT STATUS

QUAD-CITIES RETRAN MODEL :

- Base Model with Cycle 1 Core Completed
- Cycle 1 Start Up Test Plant Data Collected
- Cycle 1 Start Up Test Benchmark Completed (sample results will be presented today)

DRESDEN RETRAN MODEL :

- Base Model with Cycle 1 Core Completed
- Cycle 1 Start Up Test Plant Data Collected
- Cycle 1 Start Up Test Benchmark in Progress

LASALLE RETRAN MODEL :

- Base Model in Progress
- Cycle 1 Start Up Test Plant Data Collection Nearly Complete

MODEL ADJUSTMENTS :

- Final modeling techniques from completed Peach Bottom Benchmark will be incorporated into the Quad-Cities, Dresden and LaSalle RETRAN models

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Quad-Cities Preliminary Results

List of Startup Test Initial Conditions used to Benchmark Quad-Cities RETRAN Models

Initial Condition	Units	PRSC	RWLSC	BPVC
Reactor Power	(%)	22.5	91.5	68.0
Core Flow	(MLb/hr)	36.5	98.0	55.0
Reactor Pressure	(PSIG)	958.0	998.0	972.0
Feed Water Flow	(MLb/hr)	2.21	8.8	6.40
Reactor Water Level	(In.N.R.)	29.0	32.5	30.0
Steam Flow	(MLb/hr)	2.64	8.8	0.00
Duration for Benchmark	(sec)	20	70	70

Start Up Test Acceptance Criterion for RETRAN Model Benchmark

	Rating		
Parameter	(+)	(0)	(-)
Steam Dome Pressure	<10psi	<20psi	>20psi
Downcomer Level	<5in	<10in	>10in
Steam Flow Rate	<5%	<10%	>10%
Feedwater Flow Rate	<5%	<10%	>10%
Recirculation Loop Flow Rate	<5%	<10%	>10%
Core Flow Rate	<5%	<10%	>10%
Reactor Power	<3%	<6%	>6%
Turbine Control, Valve Position	<0.5%	<1.0%	>1.0%
Main Steam Flow Bypass Valve Position	<10%	<20%	>20%

Instrument Accuracy Data

Plant Measured Parameter	Full Scale Span	Accuracy	Accept. Criteria
Reactor Power (APRM/LPRM)	0-125%	#	<3% of Rated
Core Flow	0-80 Mlb/hr per loop	#	<5% of Rated
Dome Pressure	0-1200 PSIG	±2%FS	<10 PSI
Reactor Water Level, NR	0-60 INWC	±2%FS	<5 Inches
Reactor Water Level, WR	-42 to 358, -340 to 60 INWC	±2%FS	<5 Inches
Feed Water Flow	0-6 Mlbm/hr	#	<5% of Rated
Main Steam Flow	0-3Mlb/h	±8.5%FS	<5% of Rated
Main Steam Bypass Valve Position	0-5V	±10mV i.e.±0.2%	N/A
Turbine Control Valve Position	0-5V	±10mV i.e.±0.2%	N/A

 # Indicates GE proprietary data omitted

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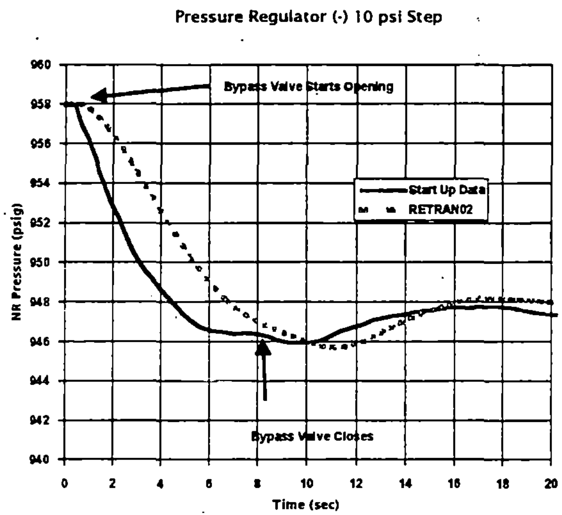
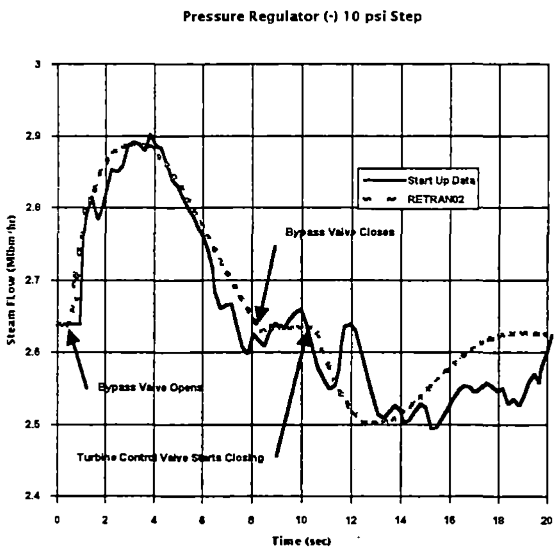
Quad-Cities Preliminary Results

Pressure Regulator Setpoint Change (Case 1)

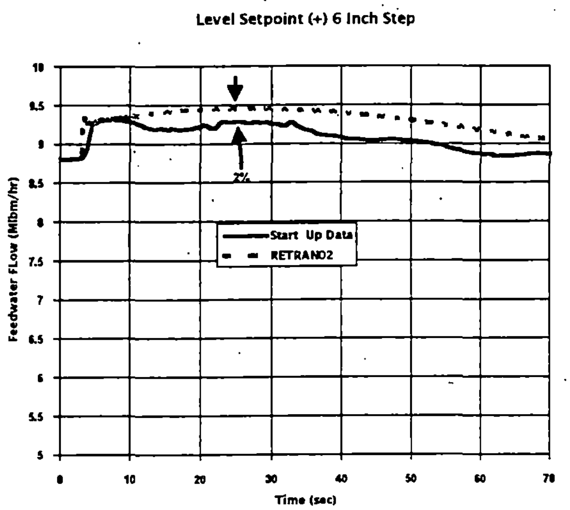


Reactor Dome Pressure for PRSC

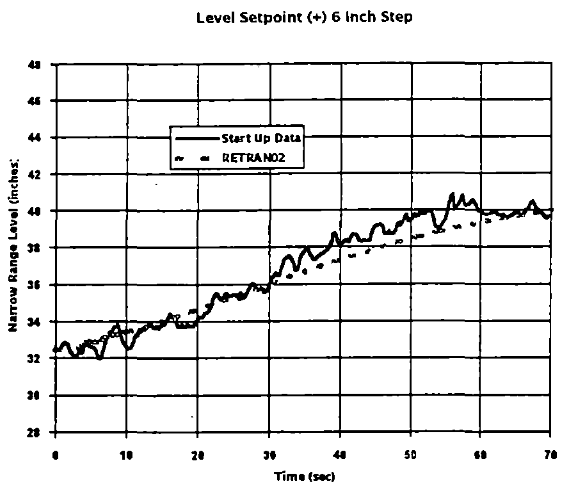


Main Steam Flow for PRSC

Reactor Water Level Setpoint Change (Case 2)



Feed Water Flow for RWLSC



Reactor Water Level for RWLSC

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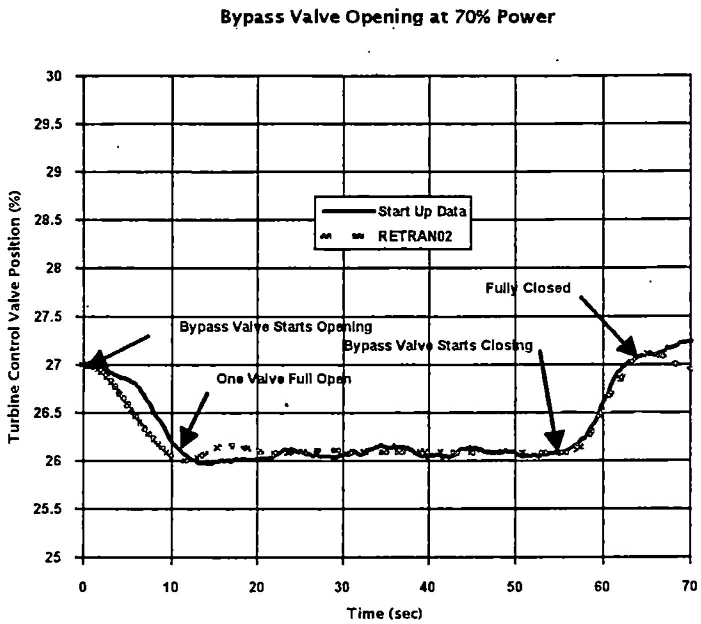
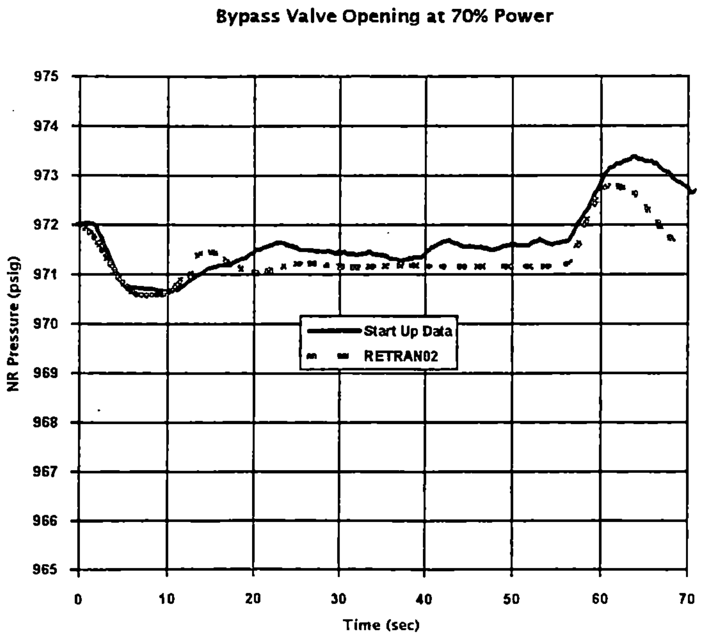
Quad-Cities Preliminary Results

Bypass Valve Change (Case 3)



Turbine Control Valve Position for BPVC



~~Reactor Dome Pressure for BPVC~~

MODEL DEVELOPMENT STATUS

PEACH BOTTOM RETRAN MODEL :

- Base Model at Nominal Conditions Completed

- Benchmark in Progress

  -Cycle 2 Startup Turbine Trip Test # 1
  -Cycle 2 Startup Turbine Trip Test # 2*
  -Cycle 2 Startup Turbine Trip Test # 3
  -NRC Licensing Problem, Turbine Trip W/out Bypass

  (sample results will be presented today)

- Final Adjustments to models pending

- Sample Results Follow

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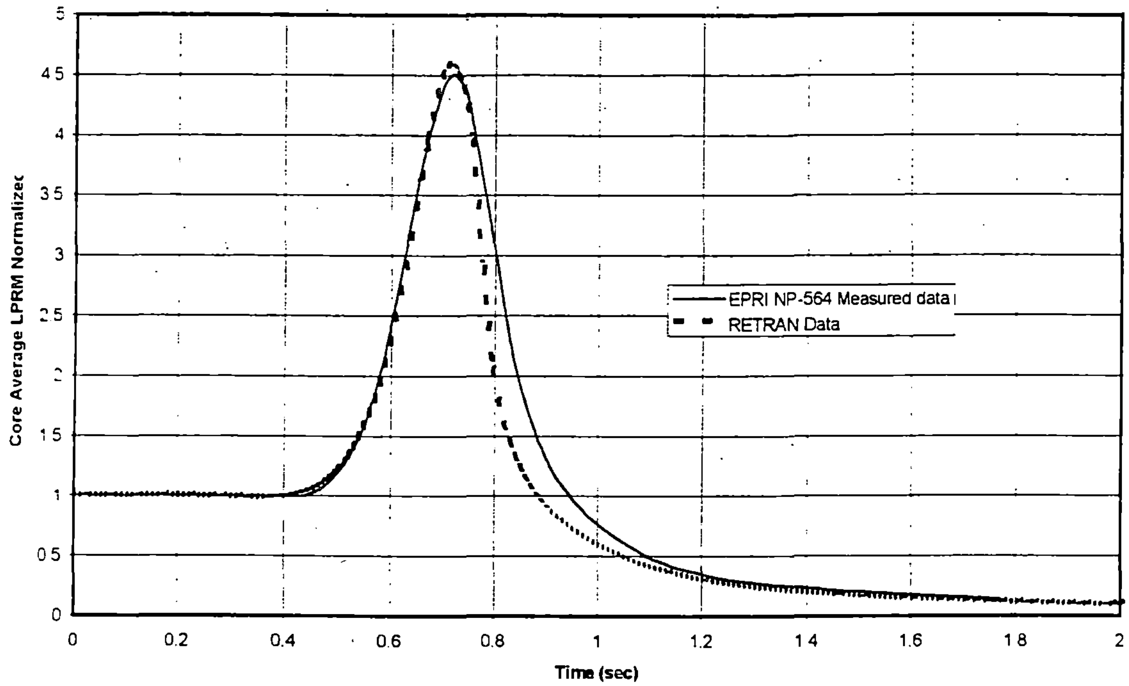
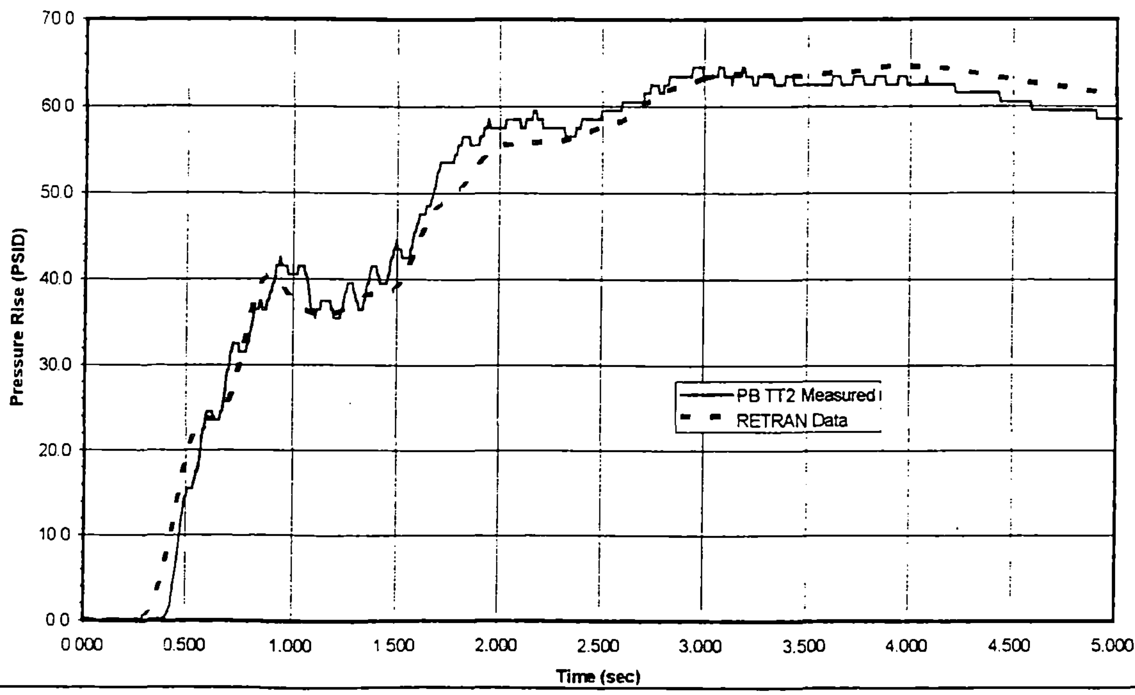
Peach Bottom Preliminary Results

Peach Bottom Turbine Trip #2, Core Avg LPRM



Peach Bottom Turbine Trip #2, Pressure Rise Reactor Dome



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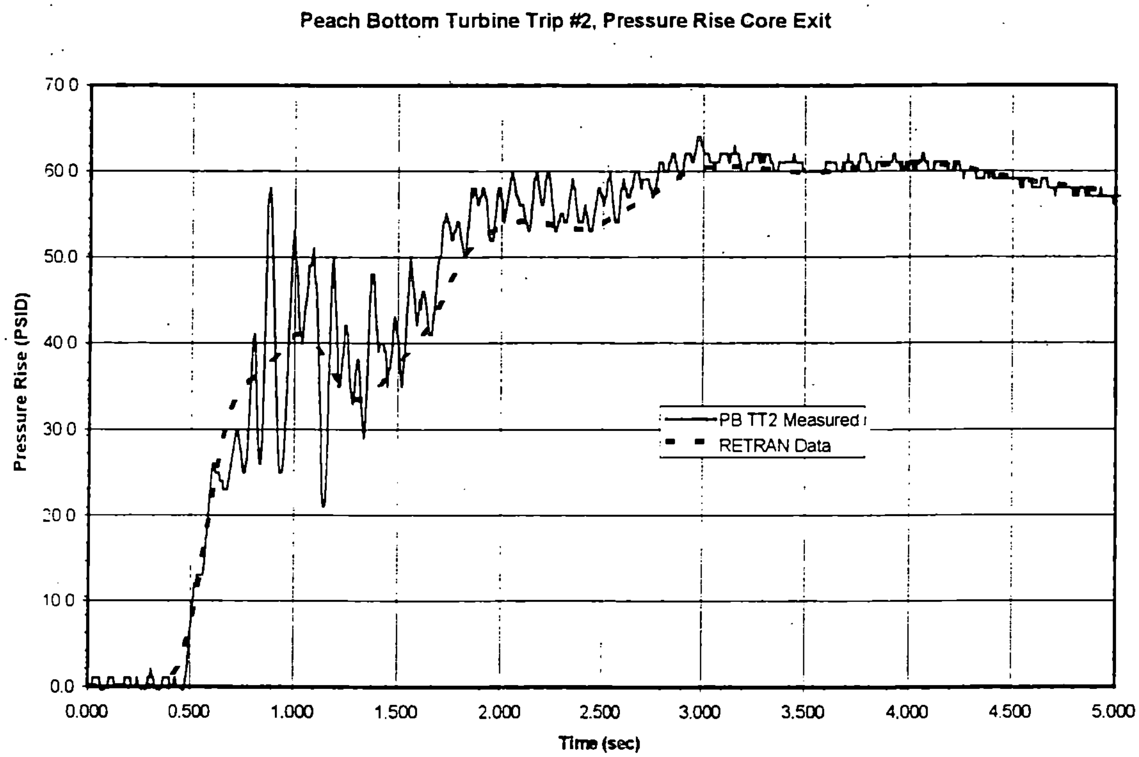
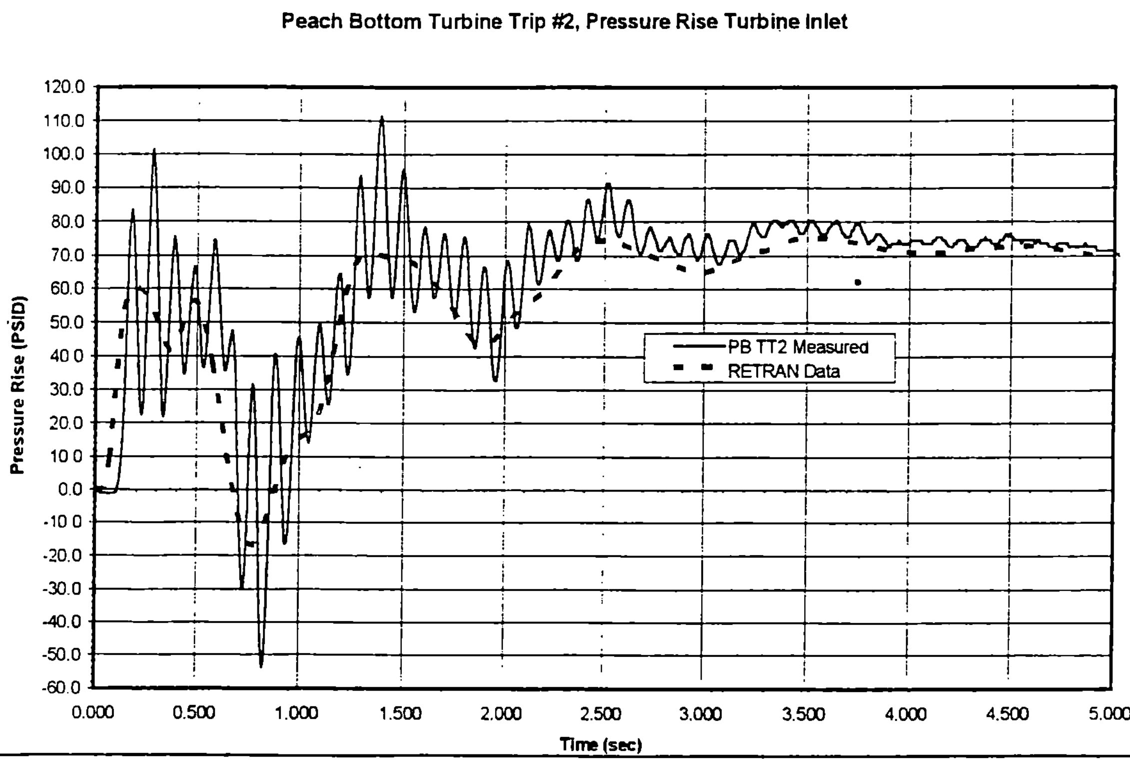
Peach Bottom Preliminary Results





ONE DIMENSIONAL METHODOLOGY REVIEW

- CECo has initiated an Independent review of its 3-D to 1-D cross section collapsing methods (MICROBURN-->PETRA-->RETRAN)

  -Contract with Computer Simulation & Analysis (Developers of RETRAN)

  -Utility Code MICPET Documents reviewed

  -Utility Code WIDE Documents reviewed

  -Methodology Calculation Note Review in Progress

  -Sample Peach Bottom model Results Review in Progress

  -Final comments from CSA are pending completion of CECo Peach Bottom Studies

SUMMARY

- Update on Submittal Schedule
- Training & Quality Assurance Program
- Update on Methods & Progress
- NRC Feedback on CECo Plan & Schedule on Submittals

CECo Nuclear Fuel Services Department Overview

NRR Presentation May 5,1994

By Terrance A. Rieck, NFS Manager

ENCLOSURE 3

1:00PM CECO NUCLEAR FUEL SERVICES

• NFS OVERVIEW	TERRY RIECK
• VENDOR INDEPENDENCE PROGRAM	KEN KOVAR
• ENGINEERING AND OPERATIONAL SUPPORT	KEVIN RAMSDEN
• NUCLEAR FUEL AND REACTOR ENGINEERING SUPPORT	JACK DOLTER

2:15PM BREAK

2:30PM BWR SAFETY ANALYSIS PROGRAM UPDATE

• INTRODUCTION	TERRY RIECK
• SUBMITTAL SCHEDULE • METHODS OVERVIEW • TRAINING & QA	BOB TSAI
• TOPICAL PROGRESS	HOSSEIN YOUSSEFNIA JOHN FREEMAN
• SUMMARY	TERRY RIECK

NUCLEAR FUEL SERVICES OVERVIEW

A. Vision

B. Key Expectations

C. History

D. Organization

NFS VISION

- Professional Partner on NOD Team
- Proactive Response
- Excellence
- Impeccable Nuclear Engineering
- Stimulating Work Environment Centered Around Our People

KEY EXPECTATIONS

VENDOR INDEPENDENT RELOADS

Development and use of analytical methods for reactor neutronic, thermal hydraulic, and transient analyses in order to safely and efficiently design, license, and operate reload cores.

IN-HOUSE ENGINEERING AND OPERATIONAL SUPPORT

Application of in-house analytical tools and expertise to support plant design changes, equipment problems, and other engineering and operational needs of the nuclear stations.

RESPONSIBLE DESIGN AUTHORITY FOR FUEL AND CORE COMPONENTS

Implementing safe, economic, and reliable fuel and core component designs which meet changing station needs, improve product performance, and reduce product and fuel cycle costs by working with the Company's vendors and site engineers.

DIRECTION AND MONITORING OF ON-SITE FUEL ACTIVITIES

Directing, monitoring, and assessing on-site fuel related activities, including reactivity management, fuel reliability, and core component performance.

KEY EVENTS IN DEVELOPMENT of NUCLEAR FUEL SERVICES at CECO

1966
- -Corporate Nuclear Fuel Committee formed
- -Four Production Dept. people sent to Purdue Fuel Management course

1967
- -Task Force on Nuclear Fuel Management Planning officially formed
- -United Nuclear (UNC) computer programs obtained

1968
- -Production Nuclear Reactor Analysis (PNRA) formed

1970
- -Approximately 10 people in PNRA

1974
- -PNRA name changed to Nuclear Fuel Servies (NFS)
- -Major Dresden-3 7x7 fuel failure event (departure from preconditioning rules)

1975
- -Major Quad Cities-2 7x7 fuel failure (departure from preconditioning rules)
- -BWR Qualified Nuclear Engineer (QNE) Program Initiated by NFS

1977
- -Parallel of vendor fuel management on large BWR's using UNC codes
- -Full scope Dresden-1 fuel management by NFS using UNC codes
- -Completed development and implemented a computer based nuclear material accountability system called the Nuclear Fuel Data Bank

1978
- -Carroll County contract provides rights to use W neutronics methods for PWR's
- -Exxon contract provides rights to use Exxon neutronics methods for BWR's
- -Design Participation Program at W (3 engineers and supervisors for 1 year)

1980
- -Approximately 25 people in NFS
- -Began development of safety analysis methods using EPRI codes
- -Intensive effort to develop and implement POWERPLEX advanced core monitoring system for Dresden transition to Exxon fuel supply
- -Intensive effort to implement W codes on CECO IBM and benchmark/validate for NRC Topical Report

1983
- -NRC approval obtained for NFS to use W neutronics methods

1988
- -INPO push for utility Reactivity Management Program
- -PWR Qualified Nuclear Engineering Training Program begins

1990
- -Approximately 70 people in NFS

1989
- -NRC approval of Zion DNB limit for PWR in-house safety analysis methods

1992

-NRC approval obtained for NFS to use GE neutronics methods for BWR's

1993

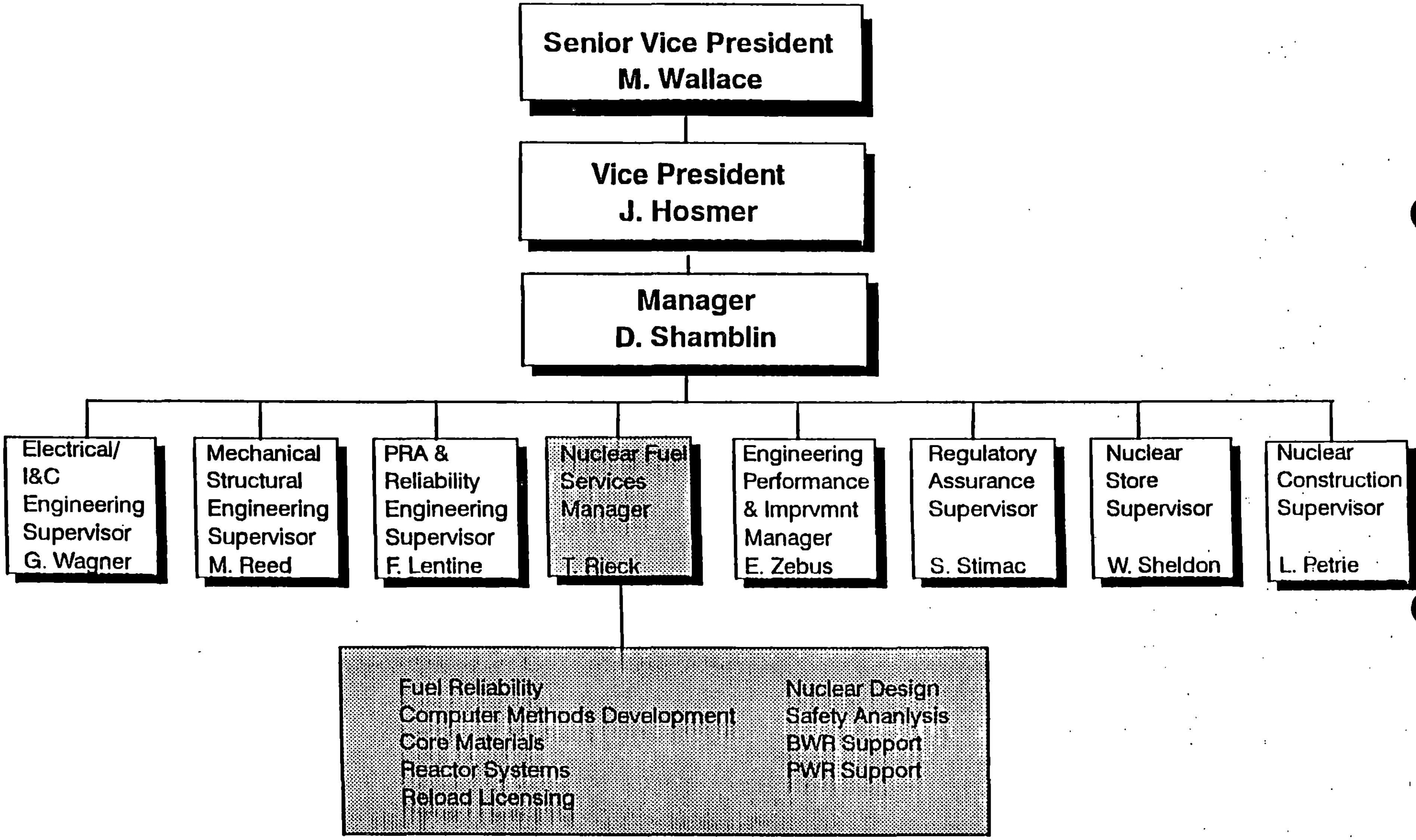
-NRC approval obtained for NFS to use SPC neutronics methods for BWR's

-NFS completes 40th in-house reload design

1994

-NRC approval obtained for NFS to use EPRI transient analysis methods (RETRAN/VIPRE) for Zion

Nuclear Engineering & Technology Services



Nuclear Fuel Services Department

T. A. Rieck
Manager

P. C. LeBlond
Chief Nuclear Engineer

- Direction of BWR/PWR Common Issues
- Reactivity Management Oversite
- Technical Direction of Reactor Engineers

K. B. Ramsden
Reactor Systems Engineer

- Transient Analysis Consultant
- Containment Performance

D. R. O'Boyle
Core Materials Engineer

- Core Materials
- Corrosion
- Fuel and Core Component Performance
- In-Core Fuel / Component Tests

E. A. Armstrong
Fuel Reliability Engineer

- Fuel Reliability
- Failed Fuel Action Plan
- Fuel Fabrication Reviews
- Fuel Tests & Inspections
- Review New Fuel Designs

J. A. Silady
Reload Licensing Engineer

- Reload Design / Safety Licensing Issues Coordination
- Fuel Bid Evaluation
- Fuel Contract Input

L. Gutierrez
Office Supervisor

- Clerical Support
- Departmental Training and Trip Coordination
- New Employe Orientation
- Custodian of Confidential and/or Proprietary Data

R. J. Chin
Nuclear Design Supervisor

- Nuclear Fuel Management
- Neutronic Reload Licensing Calculation
- Nuclear Core Operational Data

16 Staff Members

H. S. Kim
Computer Methods Development Supervisor

- NFS Software System Development & Support
- Station Core Monitoring Sys. Implementation & Maintenance
- NFS Local Area Network/Workstation Management

11 Staff Members

K. N. Kovar
Safety Analysis Supervisor

- Reactor Safety Analysis
- NOD Safety Technical Support
- Fuel Vendor Safety Technical Review

16 Staff Members

R. W. Tsai
BWR Safety Analysis Development Supervisor

E. H. Young
PWR Support Services Supervisor

- Station NE Support & Assistance
- Reload Licensing Package Development
- Outage & Startup Support
- Core Monitoring & Operation Software Support
- Core Component Acquisition
- PWR QNE Training

11 Staff Members

J. M. Dolter
BWR Support Services Supervisor

- Station NE Support & Assistance
- Reload Licensing Package Development
- Outage & Startup Support
- Core Monitoring & Operation Software Support
- Core Component Acquisition
- BWR QNE Training

9 Staff Members

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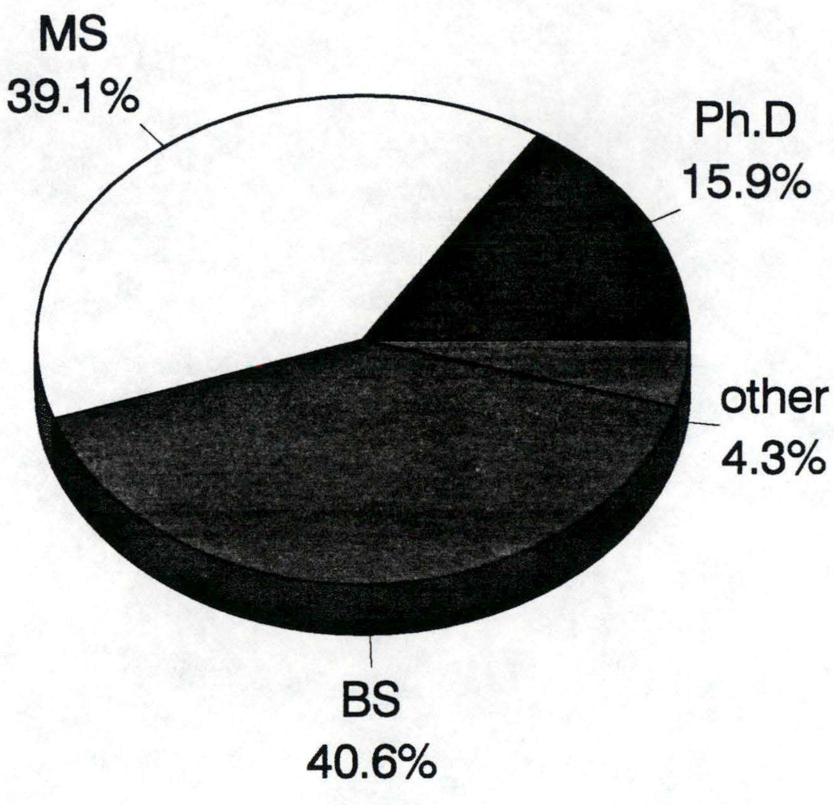
PERSONNEL EDUCATION

by degree



Nuclear Fuel Services

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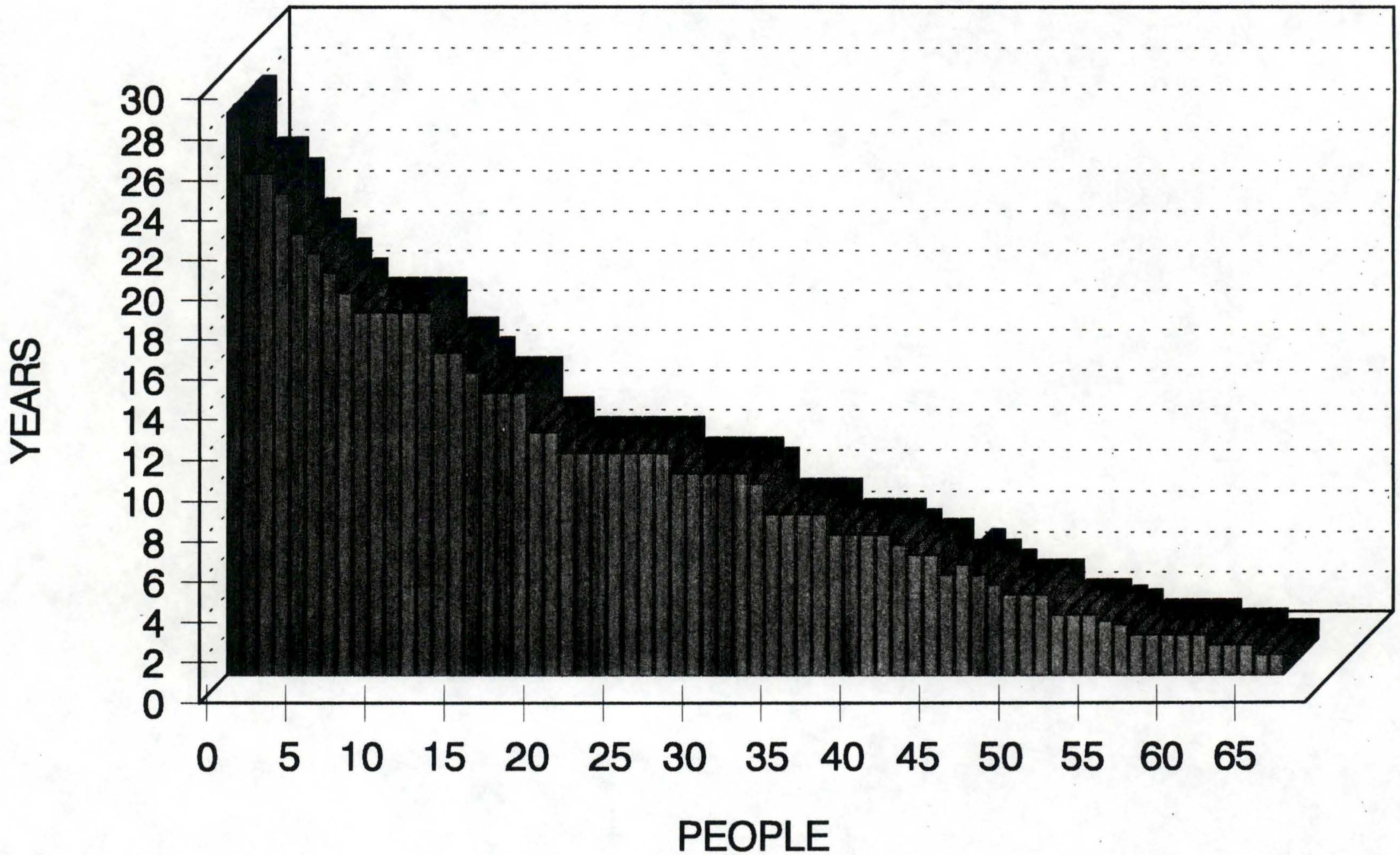
NUCLEAR EXPERIENCE

in years



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Vendor Independence Program (VIP)

NRR Presentation May 5, 1994

Ken Kovar, Safety Analysis Supervisor

What is it?

Development of CECo reload neutronic and safety analysis capabilities

Production of the analyses of record in these areas

Why do this?

To save $$$

- Fuel savings
- Avoided analysis cost

To provide in-depth technical support to the stations

How do we do it?

General plan - Four phases:

- PWR Neutronic Analysis
- PWR Safety Analysis
- BWR Neutronic Analysis
- BWR Safety Analysis

Common Approach:

- Vendor or EPRI methods and software
- On-the-job training at vendors (Design Participation Training)
- Software installed at CECo
- Topical reports submitted for NRC approval

PWR Neutronic Analysis

- Westinghouse methods and software
- Started in '78
- Eight engineers trained at Westinghouse
- Total project scope - thirty-five person years
- NRC approval in '83
- NRC approval of methods upgrade in '91
- Thirty-two (32) reload designs
- Technical support (e.g. Byron bent s-pin reanalysis)
- Savings: $15.5 million

PWR Safety Analysis

- Westinghouse methods and EPRI software
- Started in early '80's
- Three engineers trained at Westinghouse
- Total project scope - sixty person years (estimated)
- NRC approval of Zion thermal limit topical in '89
- NRC approval of Zion transient application in '94
- NRC review of Byron/Braidwood thermal limit topical in progress
- Byron/Braidwood transient application to be submitted '95
- Technical support (e.g. ECCS flow calculations)
- Savings: $4.4 million

BWR Neutronic Analysis

- GE and SPC methods and software
- Started in early '80's
- Eleven engineers trained at GE and SPC
- Total project scope - fifty person years
- NRC approval for GE applications in '92
- NRC approval for SPC applications in '93
- Eleven (11) reload designs
- Technical support (e.g. Quad Cities high flux trip evaluation)
- Savings: $19.3 million

BWR Safety Analysis

Details on methods and schedule to be presented later today

- **SPC methods and EPRI software**
- **Started in '90**
- **One engineer trained at GE**
- **Total project scope - sixty person years (estimated)**
- **Topicals to be submitted starting later this year**
- **Technical Support (e.g. Post-LOCA suppression pool temperature monitoring)**
- **Savings: $4.1 million**

What's in the future?

CECo rejoining EPRI (nuclear)

- Left EPRI in '92 due to financial situation
- Remained as members of code maintenance groups
- Access to research and development

Joint development with fuel vendors

- Transition from GE to SPC fuel at LaSalle and Quad Cities
- PWR multi-dimensional kinetics (Rod Eject)
- Unified neutronic methods

Continuing ties with Nuclear Engineering Schools

- CECo/DOE matching grants
- Parallel projects
- Summer intern program

Engineering and Operational Support

NRR Presentation 5 May, 1994

Kevin B. Ramsden, Reactor Systems Engineer

Introduction

NFS Performs a Wide Variety of Support Functions

Some are highly visible

Some are transparent

This discussion will highlight the more important functions and services NFS performs

Tools

System Transient Analysis

RETRAN02

RELAP5

TRAC

Vendor tools (ODYN/LOFTRAN/TWINKLE)

Core T/H

VIPRE

COBRAIIICMIT

FIBWR2

CONTAINMENT

CONTEMPT4M5

GOTHIC

Tools, contd.

SPECIAL

COMMIX1B
HEATING7
RELAP4M6
MATHCAD

ANALYSIS PLATFORMS

IBM Mainframes
HP 735 Workstations
Vax
Prime
PCs

NFS "Customers"

Nuclear Stations

- Nuclear Groups
- Site Engineering
- Systems Engineers
- Operations
- Regulatory Assurance
- Training

Nuclear Licensing

Offsite Review

AEs and Consultants

Vendors

NFS Design Ownership

Reload Neutronic Analyses

LOCA Analyses

Vendor performs/NFS controls

Containment Analyses

Vendor or NFS may perform

Reload Transient Analyses

NFS Design Ownership, contd.

EQ Compartment Analyses

B/B Steam Tunnel Superheat

Review/performance of new analyses

Special Analyses

PWR ECCS Flow Balance

AFW Performance

B/B SG Tube Rupture

B/B UHS analysis

BWR ECCS room cooling requirements

Safety Evaluation/Support

- **Perform Input Reviews (OPL3/4)**
- **Review MODS/Changes with Sites**
- **Close Relationship with Vendors/Consultants**
- **Review of Vendor/AE analysis**
- **Generate/support preparation of 50.59 for Sites**
- **Maintain understanding of margins**
  - LOCA rackups
  - MCPR/DNBR penalties

Operability/Plant Problems

Assist System Engineers and SEC Personnel

Develop JCO/BCOs when appropriate

Perform Operability Evaluations

Support Safety Significance Determination

- LERS
- Enforcement Conferences

"Informal Support"

- AEOD
- Licensing

On-Site Investigations

NFS Provides Direct Support

Analytical Support Provided

- QC HPCI Exhaust Failure Event
- LaSalle RCIC Exhaust Failure Event
- Undervoltage Issues
- Battery Loading Profiles
- MOV concerns
- Dresden SORV
- Dresden FW Transient

Generic Issue Support

Station Blackout Coping Studies

- Dresden
- Quad Cities
- Containment Analysis for D/QC/B&B
- Review of Zion submittal

BWR Stability

- Owners Group Activities
- Analysis Subcommittee

MOV Issues

- Review of Generic BCOs
- Review of Valve Priority Assignments

ECCS Strainers

Operations Support

EOPs

Simulator Support

Startup support(Physics/criticality etc.)

Tech Spec Interpretation/Improvement

Direct Assistance of Site NEs

GSEP

Observations

Need to shift more work from "Reactive" to "Preventive"

Earlier engineering involvement may prevent/reduce "fire drills"

"Excuse Engineering" is challenging, but it doesn't improve the plant.

Communications

Welcome more early interaction with NRR to prevent misunderstandings, particularly on older plant design basis questions.

Summary

NFS Provides a Variety of Services

Much of our work is "behind the scenes"

NFS growth has been deliberately controlled

Nuclear Fuel and Reactor Engineering Support

NRR Presentation May 5, 1994

By: Jack M. Dolter

BWR & PWR Support Services

Purpose: Support Services provides technical support to the Site Reactor Engineers and others for the nuclear fuel and core components.

Key Responsibilities:

- Reactivity Management
- Station Nuclear Engineer Training and Qualification
- Reload Licensing
- Core Monitoring Codes
- Fuel Reliability
- Core Components
- Nuclear Material Control and Accountability
- Spent Fuel and Criticality Analysis
- Operations Support
- Safety Evaluations for Plant Transients and Accidents

Personnel → 17 Full-time Engineers
3 Engineering Assistants

Reactivity Management

Policy: The Nuclear Fuel must be operated and handled in such a way that unplanned criticalities and fuel failure from operation beyond design or operational limits can never be permitted. All planned reactivity changes shall be conducted in a controlled manner, the effects of reactivity changes are known and monitored and any anomalous indication is met with conservative action.

Activities:

- Develop Policy
- Review Station Procedures
- Review License Training Lesson Plans
- Event Investigation
- Develop Guidelines/Procedures
- Review Industry Events
- Recommend SNE Staffing Levels
- BWR & PWR Owner's Groups Committees (GE & W)
- ANS Standards Committees

Reactor Engineering Support

Training and Qualification of site NE's

- Ensure the station NE's can perform their routine duties and can respond appropriately to any credible core problem.

- BWR & PWR NE Qualification Program
  - * Vendor Nuclear Engineering Course
  - * On-the-job training
  - * CECo SNE Course
  - * Qualification Oral Board
    - *Tech Specs/Bases*
    - *Job Responsibilities*
    - *Situations*
      - *Normal (Xenon)*
      - *Abnormal*

On-Going Communication

- Lead Nuclear Engineer Meetings
- Weekly Conference Calls
- Site Visits

Reload Licensing

Goal: Provide coordination of reload licensing among stations/vendors/NFS groups.

- Develop Schedule
- Submit Anticipatory Tech Spec Changes
- Develop Safety Evaluation Report
- Prepare Reload Package
- Develop or Review COLR
- Answer On-Site and Off-Site Review Questions
- GE/SPC Transition

Fuel Reliability

Goal: No defective fuel assemblies in any of Edison's Units and operate the fuel unrestricted to its end of life.

Focus on technical aspects of design, manufacture and operation of nuclear fuel.

- Manufacturing Plant Reviews
- New Fuel Designs
- Fuel Design Changes
- Parts List Reviews
- Site Inspections (FME Control)
- Reactor Water Chemistry
- Lost Parts Analysis
- Failed Fuel Action Plan
- Failed Fuel Inspections
- Vendor Technical Review Meetings

Core Monitoring

Goal: Provide the site with the most accurate and useful information to support operation of the units.

- Pioneer in core monitoring code development
  - *early 80's POWERPLEX (with Exxon Nuclear)*
  - *Currently BEACON (Westinghouse)*
- Initial Testing at NFS
- Site Testing/Parallel Run
- Software Problem Reports
- Revisions to Core Monitoring Codes
- Cycle Data Updates

Core Components

Goal: Provide the sites with the best core components at the most reasonable prices. Encourage Innovation - Non-Original Equipment Control Blades, & RCCAs.

- BWR's - Control Blades, LPRM's, Channels, and Channel Fasteners
  PWR's - RCCA's
- Bid Specification
- Bid Technical Evaluations/Recommendations
- Technical Issues
- End Of Life Tracking
- Vendor Meetings

Nuclear Material Control & Accountability

- Technical Content of CECo's Nuclear Material Control Procedures
- Technical Consultant to Stations
- Changes/Process Improvements
- Nuclear Fuel Data Bank

Conclusion

Duties - Diverse, Point of Contact for Stations & Vendors

Staff - Mostly Former Station NE's

Goal - Support Nuclear Fuel & Reactor Engineering Staffs