Brunswick Steam Electric Plant
Extended Power Uprate

Presentation to the
Advisory Committee On Reactor Safeguards
May 2, 2002

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A Progress Energy Company

Overview

Bob Kitchen
EPU Project Manager

CP&L
A Progress Energy Company

Overview
Implementation Approach

- Current BNP Operation
  - 105% Uprate Completed
  - 2-Year Operating Cycle
- EPU Request Is a 15% Increase
  - 120% From Original Licensed Thermal Power (OLTP)
  - No Reactor Pressure Increase
- Two Step Uprate Required
  - 1st Power Uprate = 112 to 115% (OLTP)
  - 2nd Power Uprate = 120% (OLTP)

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Overview
Parameter Comparison

Parameter	OLTP	CLTP	EPU
Reactor Thermal Power (MWt)	2436	2558	2923
Reactor Steam Flow (Mlb/hr)	10.470	11.089	12.781
Reactor Dome Pressure (psia)	1020	1045	1045

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Overview

Required Safety Significant Modifications

- Standby Liquid Control (SLC)
  - Increase amount of Boron injected
  - Change required to meet cold shutdown
- Unit Trip Load Shed
  - Reduce Challenge to Offsite Power
  - Ensures Adequate Post-Unit Trip Voltage
  - Selected Load Shed of Large Balance-of-Plant Motors

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Overview

Phase 1 – Balance-of-Plant Modifications

- Replace High Pressure Turbine
  - Modify Electro-Hydraulic Controls (EHC)
- Replace Reactor Feedwater Pump Turbines
- Replace Feedwater Heaters
- Improve Grid Stability
  - Power System Stabilizer
  - Out-of-Step Protection
- Improve Generator Isophase Cooling

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Overview
Phase 2 – Balance-of-Plant Modifications

- Replace Main Transformers
- Replace Feedwater Heaters
- Upgrade Moisture Separator Reheaters

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Overview
Impact on Plant Margins

Plant Modifications to Maintain Operational Margins

Plant Change	Margin Impact
SLC Boron Increase	Improve SLC Margin Require 1 Pump Versus 2
Stability Option III	Improve Operating Margin No Change to Safety Margin
Power Range Instrumentation	Improve Operator Interface, Improve Maintenance, Address Obsolescence
Upgrade Condensate	Maintain Standby Pumps
Power System Stabilizer	Upgrade Grid Stability

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Overview
Control of Phase 1 Operation

- Determination of Phase 1 Operating Level
  - Uprate Monitoring and Testing
  - Establishment of Interim Operating Power
- Supporting Analysis
- Procedural Controls

Overview
Exceptions to ELTR

- Thermal-Hydraulic Stability
- ECCS-LOCA
- Reactor Transients
- Large Transient Testing

Exceptions Consistent With Previously Approved EPU Submittals

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Overview
Unique Aspects

- Implement Actions to Enhance Grid Stability
- Maintain Hydrogen Water Chemistry
- Manage Cycle Energy Requirements
  - 2-Year Fuel Cycle
  - 97% Capacity

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Core Considerations

Tom Dresser

Lead Engineer - Nuclear Fuels

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Core Considerations
Equilibrium Cycle

Design Targets
- Cycle Energy
- Thermal Margin
- Discharge Exposure
- Hot Excess Reactivity
- SLC Margin
- Cold Shutdown Margin

Required Changes
- 10x10 GE14 Bundles
- Higher Enrichment
- Larger Reloads
- SLC Boron
- Thermal Limits Monitoring Threshold

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Core Considerations

BNP Unit 1 Cycle 14

- Design Goals Met
- Current Methods and Margin Expectations
- No SLC Modification
- Very Flat Radial Power Distribution
- SLMCPR Change

Adequate Margin Demonstrated for
BNP Initial EPU Through Equilibrium EPU Cores

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Core Considerations
ATWS - Analysis Results

Parameter	Event/ Conditions	OLTP Results	EPU Results	Limit
Peak Vessel Bottom Pressure (psig)	PRFO/BOC	1372	1487	< 1500
Peak Suppression Pool Temperature (°F)	MSIVC/EOC	194.8	195.5	< 207.7
Peak Containment Pressure (psig)	MSIVC/EOC	12.7	12.9	< 62
Peak Cladding Temperature (°F)	PRFO/EOC	1449	1309	< 2200

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Core Considerations

GE Proprietary Information Follows

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Reactor Vessel & Internals

Blane Wilton

Supervisor - Reactor Systems

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Reactor Vessel & Internals

Effects & Impacts

- Pressure/Temperature (PT) Curves
  - Current PT Curves Approved for Use With EPU Through March 2003
  - Will Submit New Curves With Updated Fluence Methodology Per RG 1.190 in June 2002
- Fluence
  - Fluence Increase Not Directly Proportional to Power Increase - Due to Core Configuration

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Reactor Vessel & Internals
Effects & Impacts

- Embrittlement
  - The Upper Shelf Energy (USE) Equivalent Margins Analysis (EMA) Values for the Reactor Vessel Beltline Materials Remain Within the Limits of 10 CFR 50 Appendix G for 32 EFPY of EPU Operation
- Fatigue
  - The ASME Code Fatigue Limits Are Met Through EOL + 20 Years for All Vessel Components for EPU Conditions

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Containment Response

Mark Grantham
Superintendent - EPU Design

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Containment Response
Key Results – DBA LOCA

Parameter	UFSAR Methods (102% CLTP)	Current Methods (102% CLTP)	Current Methods (102% EPU)	Acceptance Limit
Drywell Pressure (psig)	40.9	44.2	46.4	62
Drywell Air Space Temperature (°F)	286.7	290.4	293	340
Wetwell Pressure (psig)	14.0	30.5	31.1	62
Suppression Pool Temperature (°F)	189.4	197.9	207.7	220

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Containment Response
Net Positive Suction Head

- Currently Committed to Safety Guide 1
  - No Credit for Containment Overpressure
- EPU Short Term NPSH
  - No Credit for Containment Overpressure Required
- EPU Long Term NPSH
  - Required Overpressure 3.1 psig
  - Requested Overpressure 5.0 psig
  - Available Overpressure 11.3 psig

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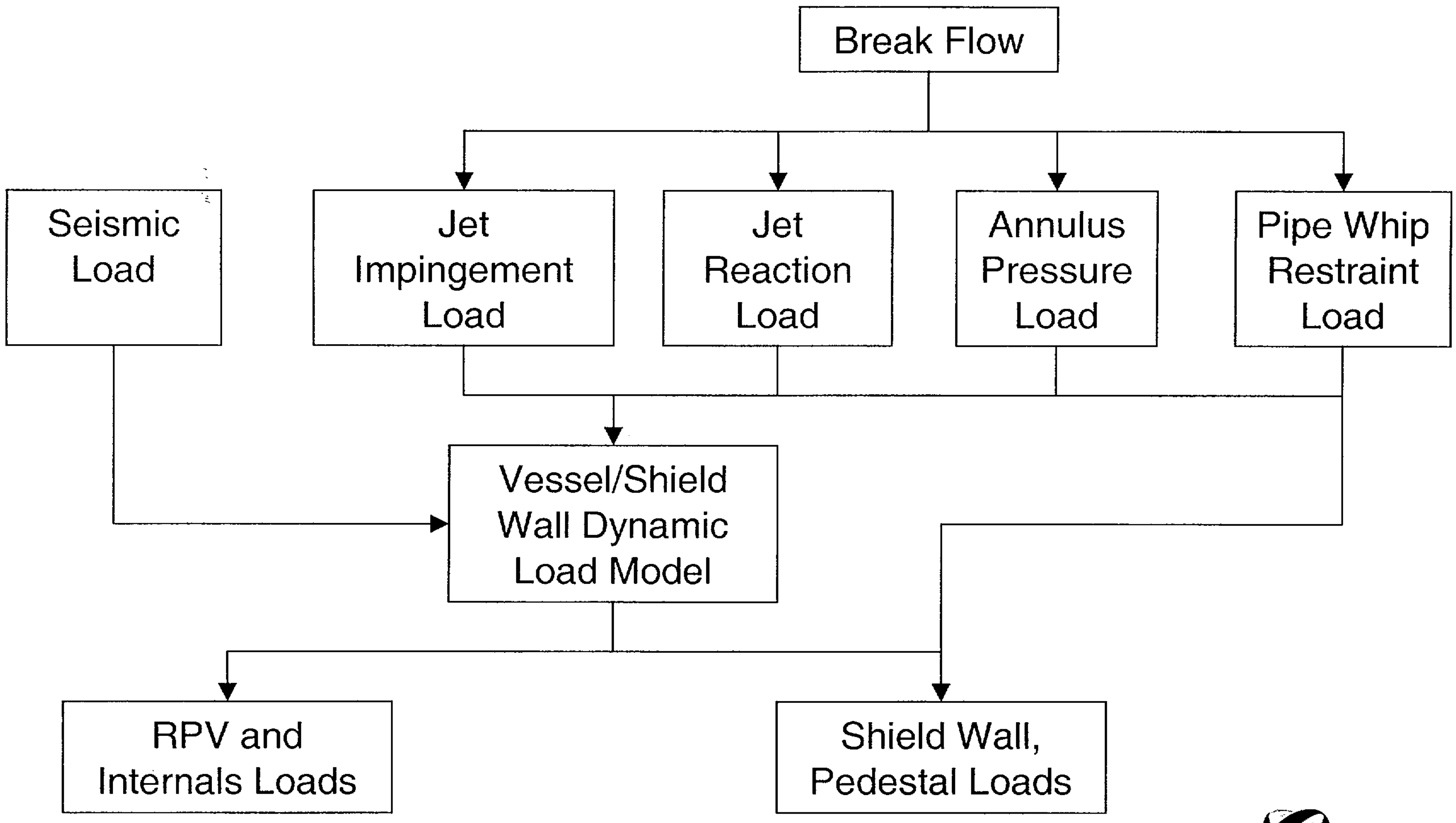
Response to ACRS Question
Feedwater/Recirculation Line Break

Dan Pappone
LOCA Process Lead - General Electric

Feedwater/Recirculation Line Break

Dynamic Loads on Vessel



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Feedwater/Recirculation Line Break
Acoustic and Flow-Induced Loads

- Recirculation Line Break Limiting
  - Subcooled Region
  - Narrow Space Between Vessel Wall and Shroud
  - Single Large Break Area for Pressure Wave Source
- Loads Evaluated for
  - Jet Pump
  - Core Shroud
  - Shroud Support (Acoustic Loads Only)

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Feedwater/Recirculation Line Break
EPU Approach

- Evaluate Change in Mass, Energy Release From Break
  - Reduce Conservatism in EPU Break Flow Analysis to Stay Within Original Hydrodynamic Load Definition If Possible
- Evaluate Increase in Hydrodynamic Load Definition, Consistent With Design Basis to Accommodate EPU
  - Use Margin Between Calculated and Allowable Stresses
  - Reduce Conservatism in Stress Calculations, Where Applicable

Feedwater/Recirculation Line
Break Analysis Results

- BNP EPU Results
  - Components in the Core Flow/Steam Flow Path Saw Increase in Stresses
  - Refined Analysis Performed for Core Shroud, Shroud Support, and Jet Pump Diffuser
  - All EPU Stresses on RPV Internals Are Within Allowable Limits

Response to ACRS Question
Appendix R - PCT Evaluation

Dan Pappone
LOCA Process Lead - General Electric

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Appendix R - PCT Evaluation

Assumptions

- Loss-of-Offsite-Power
- FW Ramps Down to Zero in 5 Seconds
- High Pressure System Not Credited for Vessel Makeup
- Nominal Power
- Nominal Decay Heat
- 3 SRVs Start Blowdown at 40 Minutes
- 1 Low Pressure Coolant Injection Pump Available

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Appendix R - PCT Evaluation
Conservatisms & Results

- Conservatisms
  - 90% of SRV Capacity for Blowdown
  - ECCS Performance (Minimum Flow Rates As Functions of Vessel Pressure, Maximum Valve Stroke Times, etc.)
  - High Pressure Injection Not Available for Makeup
- Results
  - 1468°F Versus 1500°F Limit

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Operational Aspects of Extended Power Uprate

Bob Kitchen
EPU Project Manager

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Operational Aspects
Summary of Operator Impacts

- EPU Test Plan per ELTR1
  - Exception – Large Transient Testing
- Operational Changes
  - New Approach to Instability
  - More Power Reductions
  - Small Reduction in Operator Response Time

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Operational Aspects
Probabilistic Safety Analysis - Results

- No Change in System Success Criteria
- No New Accident Sequences Identified
- No Significant Impact Due to Procedural Changes
- No Significant Impact Due to Hardware Changes (e.g., Replaced "In-Kind" With Like Equipment)
- Slight Decrease in Time Available for Four Operator Actions

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Operational Aspects
Probabilistic Safety Analysis - Results

	CDF (delta)	LERF (delta)
Base	2.55E-5	4.27E-6
EPU	2.59E-5 (+1.6%)	4.46E-6 (+4.5%)
With SLC Modification	2.32E-5 (-9%)	3.07E-6 (-28%)

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Core Considerations
Thermal-Hydraulic Stability

GE Proprietary

- Option III Stability Solution
- Exception to ELTR
- GE Recommended Validation of Setpoints
- Impact of EPU
  - Met All Limits
  - Current Methods and Margin Expectations
  - No Change in Regions

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Core Considerations

ECCS LOCA - Impact of EPU

GE Proprietary

Criteria	CLTP	EPU	Limit
Nominal PCT (°F)	1052	1049	N/A
Appendix K PCT (°F)	1537	1530	N/A
Upper Bound PCT (°F)	**< 1490**	**< 1487**	**< 1600**
Licensing Basis PCT (°F)	**< 1560**	**< 1557**	**< 2200**
Maximum Local Oxidation	< 1%	< 1%	< 17%
Core-Wide Metal-Water Reaction	< 0.1%	< 0.1%	< 1%
Coolable Geometry, Long-Term Cooling	Maintained	Maintained	Maintained

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Core Considerations
Transients Overview

GE Proprietary

- ELTR Exception
- EPU Analysis
  - Overpressurization
  - Loss of Feedwater
- Reload Analysis
  - Load Reject Without Bypass

Core Considerations
Transient Analysis Results

GE Proprietary

Limiting Event	CLTP	EPU	Limit
Overpressurization (psig Peak Vessel Pressure)	1324	1335	< 1375
Loss of Feedwater (Feet Above TAF)	Acceptable	7.26	> TAF
Load Reject Without Bypass (OLMCPR)	1.52	1.51	N/A

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BRUNSWICK STEAM ELECTRIC PLANT
Units 1 & 2
Extended Power Uprate

Brenda Mozafari

NRR Senior Project Manager
Division of Licensing Project Management

May 2, 2002

Overview

- BWR4/Mark I
- 20 percent power uprate from OLRTP
- Constant reactor dome operating pressure
- 5 percent stretch uprate approved Nov 1996
- 2 part implementation (7% and 8%)
- BOP modifications
- GE14 fuel

Application

- Mostly follows ELTR1 and ELTR 2
- Exceptions to ELTR1 and ELTR 2 are identified
- Non-risk-informed submittal
- Experience from previous EPU reviews

SCOPE OF REVIEW

- REACTOR CORE AND FUEL PERFORMANCE
- REACTOR COOLANT SYSTEM
- CONTAINMENT ANALYSES
- ECCS / LOCA EVALUATION
- SPECIAL EVENTS/LIMITING OPERATIONAL TRANSIENTS
- RADIOLOGICAL CONSEQUENCES
- HARDWARE CAPABILITIES/SYSTEMS AND COMPONENTS
- VESSEL / NSSS PIPING
- INSTRUMENTATION AND CONTROLS
- ELECTRICAL POWER AND EQUIPMENT QUALIFICATION
- HUMAN PERFORMANCE (OPERATOR RESPONSE TIME)
- PSA

BRUNSWICK UNITS 1 & 2 EXTENDED POWER UPRATE

ACRS Meeting

May 2, 2002

Reactor Systems Branch
BWRs and Fuel Performance Section

Ralph Caruso: Section Chief

Zena Abdullahi: Lead Reviewer

Scope and Depth of Staff Reviews

Code Applicability

- Staff review scope
  - Experience from previous uprates
  - Guidance documents (SRP, RG, topical reports)
  - Other ongoing licensing reviews
  - Operating experience and reports
  - Knowledge/experience of reviewers

- Codes continue to be used within limits of applicability - no new phenomena

- No new codes being used - no Maine-Yankee scenario

- Staff has recent experience reviewing some of these codes
  - SAFER/GESTR
  - ODYSY
  - TRAC-G

Scope and Depth of Staff Reviews

Assurance of Appropriate Code Usage

- Vendors use codes in rigorous fashion
  - Limits described in topical reports
  - Methods described in detail and performed in accordance with written procedures
  - Licensees audit calculations
  - Appendix B Quality Assurance
  - Inputs maintained in controlled database

- Staff has performed audits of selected BWR calculations and is comfortable that they are being done properly

- Staff will continue to perform audits of BWR calculations

Review Agenda

- Example of Review Focus
  - ATWS
  - Standby Liquid Control Relief Valve Margin Evaluation
    - License Condition
- Conclusion

Example of Review Focus

ATWS

- PUSAR peak vessel pressure of 1492 psig (1500 psig allowable)

- Due to low margin, staff performed more in-depth review
  Staff determined
  - limiting Unit not analyzed for (PRFO),
    - Unit 2 with larger bypass capacity (80.6 % v.s. 20.26% EPU steam flow)
    - ATWS analysis based on Unit 1 (limiting for LOOP and MSIVC)

- CP&L reanalyzed
  - PRFO ATWS event based on Unit 2 bypass capacity.
  - Reanalyzed PRFO event based on plant-specific data
  - Yielded lower peak vessel pressure of 1487 psig

Example of Review Focus

Standby Liquid Control

- No SLC relief valve margin evaluation in PUSAR
- Initial BSEP SLC relief valve evaluation resulted in negative margin.
  - Evaluation based on GE data.
- CP&L re-evaluated the SLC relief valve margin based on
  - Predicted dome pressure
  - Two pump system losses based on plant-specific tests
    - Original system losses based on 1984 GE evaluation
  - Plant-specific elevation head calculation
- Staff concluded
  - margin low but acceptable
- CP&L
  - acknowledges low margin
  - plans to modify the SLC system to improve the margin
    - Will replace the relief valves to obtain 50 psid margin

SLC License Condition

- No SLC Amendment Request Submitted
  - SLC shutdown capability
    - Phase I, Unit 1, Cycle 14
      - 660 ppm sufficient to meet cold shutdown concentration.
    - Phase II
      - Increase boron (660 ppm to 720 ppm) required to meet cold shutdown
      - Due to core design changes (i.e. larger batch fractions, higher enrichment in the GE14 fuel, driven by EPU and energy requirement to meet cycle length)
- License Condition
  - Requires amendment request
    - Changes to TS 3.1.7,"Standby Liquid Control (SLC) System"
    - 6 months before implementation
  - Single pump/squib valve commitment
    - Will start both pumps to avoid changing EPGs and operator retraining

Conclusions

- Licensing analyses are based on NRC-approved methods, codes and acceptance criteria.

- BSEP EPU SAR is consistent with NRC-accepted guidelines and generic analysis for evaluating the impact of the extended power uprate on safe operation of the plant.

- Deviations were presented to the Committee during the Clinton and the CPPU topical report meetings. (NEDC-32989P and NEDC-33004P)

- The staff finds that CP&L provided sufficient evidence to support approvable for operation of BSEP Units 1 and 2 at the proposed power level of 2923 MWt.

Appendix R Evaluation

- 10 CFR 50 Appendix R requires that tne train of systems necessary to achieve & maintain safe shutdown be maintained free of fire damage
- Brunswick used 1500 °F PCT as the fuel design limit for satisfying Appendix R
- The staff finds the use of 1500 °F PCT as the limit for a fire event to be acceptable

BRUNSWICK UNITS 1 & 2
EXTENDED POWER UPRATE

ACRS Meeting

May 2, 2002

MECHANICAL AND CIVIL ENGINEERING BRANCH
CIVIL AND ENGINEERING MECHANICS SECTION

Kamal Manoly: Section Chief

May 2, 2002

Feedwater And Recirc Line Break Effects on Reactor Internals

- In General, Loads Resulting From Pipe Break Dependent on Line Pressure and Temperature; But, Not Flow Rate Increase Due To Power Uprate
- Slight Increases in RIPDs For Some Internal Components as a Result of EPU
- Acoustic and Flow-Induced Loads Evaluated For Governing Recirc Line Break
- Blowdown Loads Governed by Main Steam Line Break
- Calculated Stresses and CUFs For Various Internal Components Either Bounded by Original Analysis or Show Slight Increase; However, Within Allowable Limits of ASME Code, Section III

OVERALL CONCLUSIONS

- Analyses are based on NRC-approved analytical methods and codes

- On-site audit confirmed compliance with staff approved methodology

- EPU SAR is consistent with NRC-accepted guidelines and generic evaluations

- Thermal limits and the applicable safety analyses would be reanalyzed or reconfirmed using NRC approved core reload analyses methodology

Applicability of Revised Fission Product Source Term (NUREG-1465) for High Burnup and MOX Fuels

Presentation to the Advisory Committee on Reactor Safeguards

Jason Schaperow
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

May 2, 2002

Overview

Objective

Assess applicability of revised source term to high burnup and MOX fuels

Approach

Hold a series of expert panel meetings, including experts who developed basis for the revised source term

Results

Experts suggested source term values for high burnup and MOX fuels, identified issues, and recommended research

Revised Source Term

Source term is the fission product release into containment atmosphere which is available for release to the environment

RES published revised source term (aka alternative source term) in NUREG-1465 in 1995

More realistic than earlier TID-14844 source term

Aerosol except for 5% of iodine which is vapor

Four-phase release: gap, early in-vessel, ex-vessel, late in-vessel

A few differences in release timing and magnitude between PWR and BWR (main difference is release timing for I)

Regulatory Applications

Gap and early in-vessel phases of revised source term used for LOCA design basis accident analyses

- Exclusion Area Boundary, Low Population Zone, and control room doses

- containment isolation valve closure time (start time of gap release)

- integrated dose used to qualify equipment in containment

- post accident shielding, sampling, and access

- hydrogen generated by radiolytic decomposition of water

All four phases of source term may be used for severe accident risk assessment

Regulatory Applications (cont.)

Revised source term being implemented voluntarily because of safety and cost benefits

License amendments issued:	**Perry**	**Fort Calhoun**
	Grand Gulf	**Three Mile Island 1**
	Indian Point 2	**Hope Creek**
	Duane Arnold	**Surry 1 & 2**
	Crystal River	
Applications under review:	**Oyster Creek**	
	Brunswick	
	Columbia (WNP2)	
	Oconee 1, 2, & 3	
	Kewaunee	

Approach

Held a series of expert panel meetings (Sep 2001 - Feb 2002)

Panel members were requested to judge applicability of each aspect of the revised source term, and if judged not applicable, to propose alternative

As part of this effort, panel members...

- considered recent data from international tests
- discussed physical phenomena affecting source term for high burnup and MOX fuels
- identified and prioritized source term research

Approach (cont.)

Panel of International Experts

Bernard Clement (IPSN, France)
James Gieseke (consultant)
Thomas Kress (consultant)
David Leaver (Polestar Applied Technology)
Dana Powers (Sandia National Laboratories)

Others

Principal Investigator: Mohsen Khatib-Rahbar (Energy Research)
Panel Facilitator: Brent Boyack (Los Alamos National Laboratory)
Consultant: Hossein Nourbakhsh (Energy and Environmental Science)

Applicability of Revised Source Term for High Burnup Fuel

Panel assessment based on:

Maximum assembly burnup of 75 Gwd/t

Core average burnup of 50 GWd/t

Zirlo cladding (PWR), Zircaloy cladding (BWR)

Low pressure scenario (minimizes RCS retention)

Table 3.1 PWR Releases Into Containment (High Burnup Fuel)

	Gap Release	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.4 (0.5)[1]	1.4 (1.3)	2.0 (2.0)	10.0 (10.0)
Noble Gases	0.05; 0.07; 0.07; 0.07 (0.05)	0.63; 0.63; 0.63; 0.65 (0.95)	0.3 (0)	0 (0)
Halogens	0.05 (0.05)	0.35 (0.35)	0.25 (0.25)	0.2 (0.1)
Alkali Metals	0.05 (0.05)	0.25 (0.25)	0.35 (0.35)	0.1 (0.1)
Tellurium group	0.005 (0)	0.10; 0.30; 0.30; 0.30 (0.05)	0.40 (0.25)	0.20 (0.005)
Barium, Strontium	0 (0)	0.02 (0.02)	0.1 (0.1)	0 (0)
Noble Metals	(0)	(0.0025)	(0.0025)	(0)
Mo, Tc	0	0.15; 0.2; 0.2; 0.2; 0.7TR[2]	0.02; 0.02; 0.2; 0.2; TR	0; 0; 0.05; 0.05; TR
Ru, Rh, Pd	0	0.0025; 0.0025; 0.01; 0.01; 0.2TR	0.0025; 0.02; 0.02; 0.02; TR	0.01; 0.01; 0.01; 0.10; TR
Cerium group	(0)	(0.0005)	(0.005)	(0)
Ce	0	0.0002; 0.0005; 0.01; 0.01; 0.02TR	0.005; 0.005; 0.01; 0.01; TR	0
Pu, Zr	0	0.0001; 0.0005; 0.001; 0.002; 0.002TR	0.005; 0.005; 0.01; 0.01; TR	0
Np	0	0.001; 0.01; 0.01; 0.01; 0.02TR	0.005; 0.005; 0.01; 0.01; TR	0
Lanthanides (one group)	0; 0; 0; 0; 0 (0)	0.0005; 0.002; 0.01 (0.0002)	0.005; 0.01; 0.01 (0.005)	0 (0)
La, Eu, Pr, Nb		0.0002; 0.02TR	0.005; TR	
Y, Nd, Am, Cm		0.0002; 0.002TR	0.005; TR	
Nb		0.002; 0.002TR	0.005; TR	
Pm, Sm		0.0002; 0.002TR	0.005; TR	

1 The numbers in parenthesis are those from NUREG-1465, Accident Source Terms for PWR Light-Water Nuclear Power Plants (Table 3.13).
2 TR = total release. The practice in France is to assign all releases following the gap release phase to the early in-vessel phase.

Results of Panel Assessment for High Burnup Fuel

Physical/chemical forms expected to be applicable.

Only small changes in release-phase duration and release fraction expected.

Burnup-independent issues identified based on recent tests

- **potential for enhanced Te release**
- **continued uncertainty in releases of noble metals, Ce, La groups**
- **recent data suggests subdividing noble metals, Ce, La groups**

Related issues

- **BWR power uprates**
- **BWR fuel design**

Tellurium Release

Revised source term specifies early in-vessel Te release of 0.05

> **ORNL tests indicate Te gets sequestered in the Sn in Zircaloy cladding and not released until high fraction of cladding is oxidized**

More recent French tests (VERCORS, PHEBUS-FP) indicate that Te release could be similar to I (i.e., 0.30)

For PWRs, this was a contentious issue among panel members.

For BWRs, panel members specified release fractions similar to revised source term

> **BWR zircaloy fuel channels tend to limit cladding oxidation**

Other Source Term Issues Related to High Burnup

BWR Power Uprates

One expert saw no basis for significant effect on fission product release

Another expert stated that flux-profile flattening associated with power uprates could increase the release rate for the outer assemblies.

BWR Fuel Design

NUREG-1465 specifies a different source term for a BWR than a PWR

Characteristics of more recent BWR fuel rod designs are closer to PWR fuel rod characteristics (e.g., pellet diameter, cladding thickness)

Panel indicated that similar rod designs tend to result in similar source terms.

<u>**Applicability of Revised Source Term for MOX Fuel**</u>

Panel assessment based on:

Using MOX in PWR (about ½ of core)

Typical MOX assembly burnup of 42 GWd/t

M5 cladding

Low pressure scenario (minimizes RCS retention)

<u>**Results of Panel Assessment for MOX Fuel**</u>

Physical/chemical forms expected to be applicable.

Only small changes in release-phase duration and noble gas, I, and Cs release fractions expected.

Same Te issue as for high burnup fuel.

Some of the experts did not recommend release fractions for Ba/Sr, noble metals, cerium, and lanthanum groups, because of the lack of test data.

> **Only data was a VERCORS test result for Cs with an arbitrary scale on the y-axis**

Panel-Recommended Research

High Priority Research

Validate severe-accident analysis codes against recent source term tests

Investigate in-vessel core degradation following vessel failure (air ingress)

Acquire any available data on fission product releases for high burnup and MOX fuels

Perform fission product release tests for high burnup fuels using modern cladding designs (Zirlo and M5)

Perform revaporization tests

Panel also recommended several medium and low priority source term research efforts.

Status

Panel members recently provided comments on draft panel report.

Final panel report to be issued by June 2002.

Results of expert panel assessment to be used to help address reactor safety issues

> **applications for high burnup and MOX fuels**
>
> **severe accident risk assessment**
>
> **other applications (e.g., vulnerability assessement)**

High Burnup Fuel
Research and Regulatory Issues

Ralph Caruso
Office of Nuclear Reactor Regulation

Ralph Meyer
Office of Nuclear Regulatory Research

May 2, 2002

Background

- NRR User Need Request - October 4, 1993
  - Updates to NRC Fuel Performance Models
  - Revise Models for Stored Energy during LOCA, and Evaluate Impact on LOCA Analyses
  - Evaluate Fuel Failure Thresholds for Normal Operation and RIA

- Commission Memorandum - November 25, 1996
  - Low Enthalpy Fuel Failures
  - Incomplete Control Rod Insertion

- Commission Memorandum - July 15, 1997
  - Adequacy Assessment of Regulatory Guidelines and Licensing Criteria for High Burnup Fuel
  - High Enrichment
  - Spent Fuel Storage and Transportation

Background (continued)

- Research Information Letter No. 174 March 3, 1997
  - Proposed Changes to RIA Criteria
    - Vendors and NRC calculations Below Proposed Limits
    - Uncertainties may be Large

- Agency Program Plan for High Burnup Fuel - July 6, 1998

  - Cladding Integrity and Fuel Design Limits
  - Control Rod Insertion Problems
  - Criteria and Analyses for RIA, LOCA, ATWS
  - Update Fuel Rod and Neutronic Codes
  - Source Term and Core Melt Progression
  - Transportation and Dry Storage
  - High Enrichments

  - NRC Confirmatory Research for Burnup <62 GWD/MTU
  - Industry Responsibility for Criteria/Data/Models for Burnup >62 GWD/MTU

RIA Regulatory Criteria

- **General Design Criterion 28**

  "The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effect of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core."

- **Standard Review Plan Section 4.2 - Coolable Geometry**

  "... [retaining a] rod-bundle geometry with adequate coolant channels to permit removal of decay heat."

- **Standard Review Plan Section 4.2 - Violent Expulsion of Fuel**

  " ... To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing wide-spread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 280 cal/g should be observed. This 280 cal/g limit should also be used in BWRs."

Revisions to Regulatory Criteria

- Current Criteria Challenged by Experience from CABRI and NSRR
- High Burnup Fuel Cladding Failed at Unexpectedly Low Enthalpy
  - Highly Corroded
  - Fabrication/Preconditioning Issues (REP/Na1)
- Interim Criteria Described in July 6, 1998 Commission Memorandum
  - Based on RIL 174
  - Oxide Spalling: None Allowed
  - Cladding Failure: 100 cal/g (enthalpy increase)
  - Coolability: 280 cal/g (enthalpy limit) for <30 GWD/MTU
No Cladding Failure for > 30 GWD/MTU

Current Situation

- Detailed Analyses Performed by Vendors and by NRC
  - 3-D Neutronics instead of 1-D
  - Limiting Fuel (30GWD/MTU) Enthalpy Values less than 100 cal/g
    - Therefore, No Fuel Cladding Failures Expected
  - Radiological Consequences well within 10CFR100
  - New Methods Under Review

- New Fuel Cladding Materials
  - Much Lower Corrosion
  - No Spallation at Current Burnup

- PWR Operational Practices
  - All Rods Out at Full Power - No Control Rods to Eject
  - Hot Zero Power Cases Analyzed
  - LWR Rod Worths are Small
  - High Burnup Fuel has Limited Reactivity

- No Evidence From CABRI or NSRR of Molten Fuel Ejection

NRR User Needs

- Proposed User Needs are Evaluated against Four Criteria
  - Maintain Safety
  - Improve Efficiency and Effectiveness
  - Reduce Unnecessary Regulatory Burden
  - Improve Public Confidence

- No Licensing Action Under Review which Requires the Results of RES High Burnup Program
  - Current Plants meet Interim Criteria

- Extended Burnup Criteria are Responsibility of Industry
  - EPRI Topical Report on RIA Criteria is First of Several Expected Reports
  - NRR/NMSS/RES Technical Advisory Group on Fuels will Provide Input to Help Identify Need for Research Information in Reviews

- Agency Prioritization Process is Under Review
  - Research "Needs" Process will be Revised in Integrated Fashion
  - Revision will consider how parts of the High Burnup Fuel Program Plan fit into Agency Strategic Plan

Research Activites on High-burnup Fuel

Agency Program Plan

- 1993 User Need memo of limited scope (codes and reactivity transients)
- 1994 memo to Commissioners on reactivity transients and high-burnup
- 1998 User Need from NMSS on dry storage and transportation
- 1998 Agency Program Plan (Commission paper)
  - identified 4 reactor issues for confirmatory research
  - said why okay to wait 3-5 years for resolution
  - outlined research that would do the job
- Research expanded to cover dry storage as well as reactor transients
- Research activities broadened to include ZIRLO and M5
- All experimental programs have industry cooperation

PWR Rod Ejection

- Regulatory Guide 1.77 needs revision of criteria at high burnups
- 1997 Research Information Letter (RIL-174) with 100 cal/g interim criterion
- 1999 Research Information Letter (RIL-177) on spalling
- Heightened interest because of corrosion (Bulletins 2001-01, 2002-01)
- Confirmatory assessment in 2-3 years with minimum effort

LOCA

- 50.46 criteria and ECCS models were derived for unirradiated Zircaloy
- Validation for unirradiated ZIRLO and M5 to be confirmed
- Adequacy of corrosion adjustment for high burnup fuel to be confirmed
- Significant hot-cell work left to complete confirmatory assessment
- This research supports the performance-based approach to 50.46 revision

BWR ATWS Instability

- Higher risk than the postulated rod drop accident that is usually analyzed
- Cladding temperature appears to be more important than mechanical loads
- Improvement in analytical methods needed to complete confirmatory assessment
- Value of related future testing being considered

Analysis

- Needed for resolution in all areas
- Improvements in steady-state code completed in 1997
- Additional transient code development needed to complete resolution in above areas

Description of Research on High-burnup Fuel

Research on PWR Rod Ejection

- Cabri (France) test reactor (NRC is a paying participant)
- NSRR (Japan) test reactor (no cost to NRC)
- IGR and BIGR (Russia) test reactors (small cost to NRC)
- ANL measurement of mechanical properties (small part of larger program)
- PNNL and BNL code analysis (small part of larger programs)
- RES staff effort to document confirmatory assessment

Research on LOCA

- ANL LOCA tests and mechanical properties measurements in hot cell
- Halden (Norway) LOCA tests in test reactor (small part of larger program)
- JAERI (Japan) unirradiated test data (no cost to NRC)
- Kurchatov (Russia) test in hot cell (small cost to NRC)
- PNNL code analysis (small part of larger program)
- RES staff effort on 50.46 and Reg. Guide

Research on BWR ATWS

- STUK and VTT (Finland) analysis with VTT-NRC codes (no cost to NRC)
- Halden (Norway) test reactor data (small part of larger program)
- PSU heat-transfer data
- PNNL code analysis (small part of larger program)
- RES staff effort on code analysis

Research on Analysis

- PNNL code development, analysis (FRAPTRAN, FRAPCON, MATPRO)
- Halden (Norway) test reactor data
- RES staff effort on code development and analysis

Research on Dry Cask Storage

- ANL hot cell tests on creep and other mechanical properties
- INEEL storage data
- RES staff effort to assess regulatory requirements

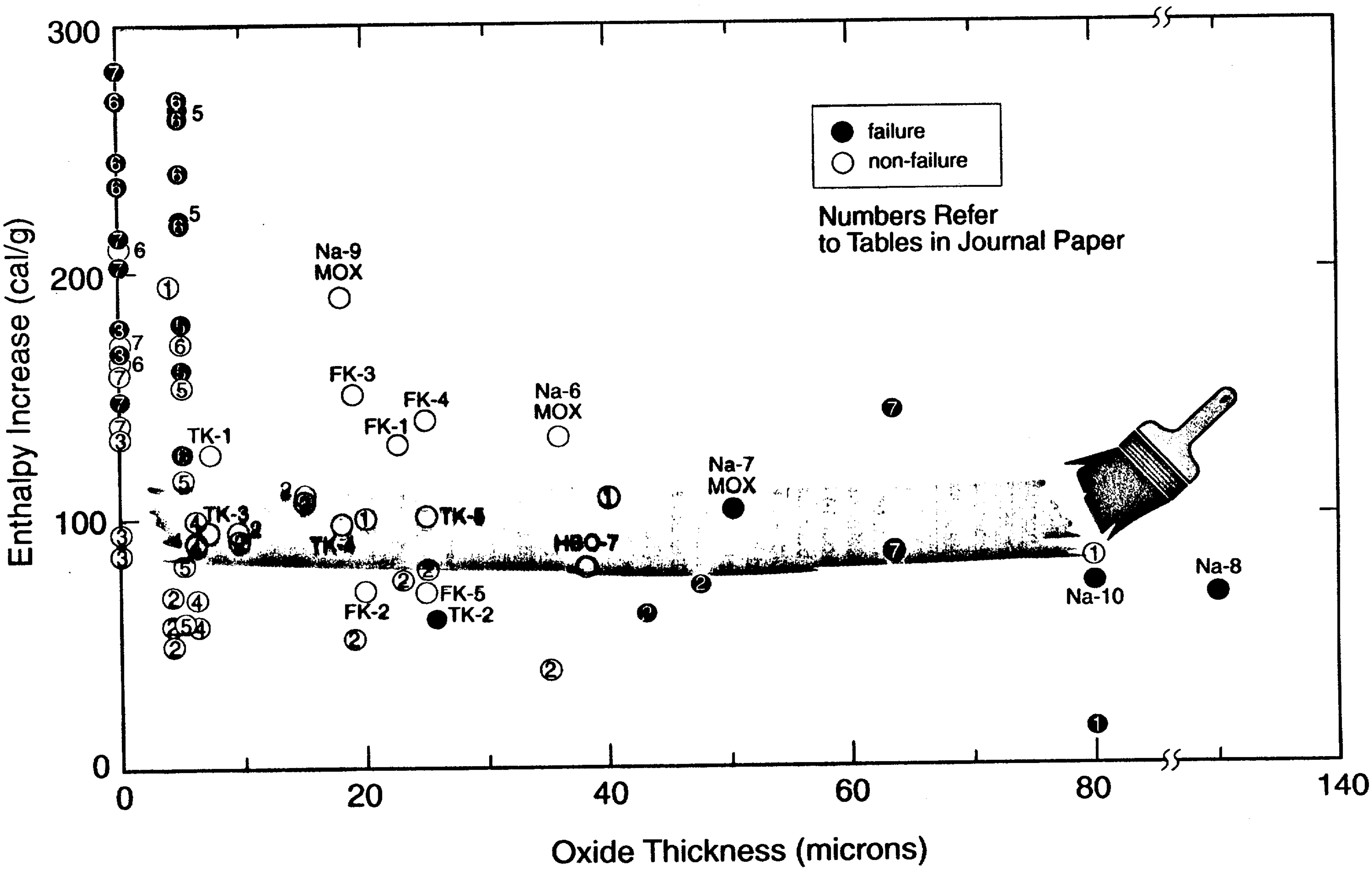
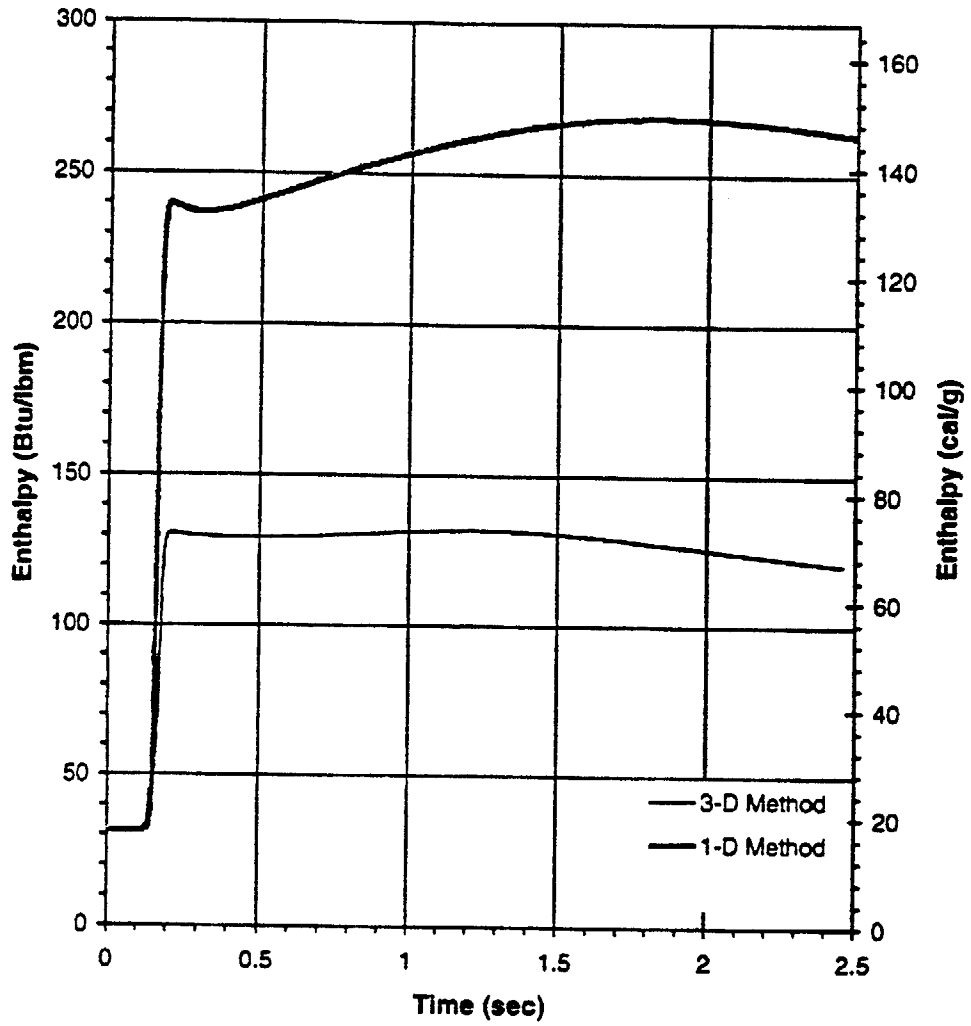

failure
non-failure
Numbers Refer
to Tables in Journal Paper
Enthalpy Increase (cal/g)
Oxide Thickness (microns)
300
200
100
0
0
20
40
60
80
140
Na-9
MOX
FK-3
FK-4
FK-1
Na-6
MOX
TK-1
TK-3
TK-4
TK-5
HBO-7
Na-7
MOX
FK-5
FK-2
TK-2
Na-10
Na-8

Figure D.3 3-D and 1-D Fuel Enthalpy vs. Time



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PULSE WIDTH VS MAX CHANGE IN LOCAL FUEL ENTHALPY