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

July 10, 2002

MEMORANDUM TO: ACRS MEMBERS

FROM: Paul Boehnert, Senior Staff Engineer

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING, JUNE 26, 2002 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on July 9, 2002, have been certified as the official record of the proceedings for that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: ACRS Members
S. Banerjee
V. Schrock
R. Savio

cc via e-mail:
ACRS Members
J. Larkins
S. Bahadur
R. Savio
H. Larson
S. Duraiswamy
S. Banerjee
V. Schrock
ACRS Staff Engineers

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

July 9, 2002

MEMORANDUM TO: Graham Wallis, Co-Chairman
Thermal-Hydraulic Phenomena Subcommittee

FROM: Paul A. Boehnert, Senior Staff Engineer
ACRS/ACNW

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON THERMAL-HYDRAULIC PHENOMENA MEETING, JUNE 26, 2002, ROCKVILLE, MD

I certify that, based on my review of these minutes, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

Graham Wallis 07/09/02

Graham Wallis, Co-Chairman Date

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

July 8, 2002

MEMORANDUM FOR: G. Wallis, Chairman, Thermal-Hydraulic Phenomena Subcommittee

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: MINUTES OF THE ACRS THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING, JUNE 26, 2002 - ROCKVILLE, MARYLAND

A Working Copy of the subject meeting minutes is attached. I would appreciate your review and corrections as soon as possible. Copies are being sent to all ACRS members and to the Subcommittee Consultants for their information.

Attachment: As Stated

cc: ACRS Members
S. Banerjee
F. Moody
V. Schrock
R. Savio

cc via E-Mail:
ACRS Members
S. Banerjee
F. Moody
V. Schrock
J. Larkins
S. Bahadur
R. Savio
H. Larson
S. Duraiswamy
ACRS Staff Engineers

DRAFT COPY - PREPARED FOR INTERNAL COMMITTEE USE

ISSUED JULY 9, 2002
CERTIFIED BY G. WALLIS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
THERMAL HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING MINUTES
STATUS OF RES "ATLATS" PHASE SEPARATION PROGRAM /
RESOLUTION OF GSI-185 - RECRITICALITY CONTROL FOR SBLOCAs IN PWRs
JUNE 26, 2002
ROCKVILLE, MARYLAND

INTRODUCTION:

The ACRS Subcommittee on Thermal-Hydraulic Phenomena held a meeting on June 26, 2002 with representatives of the NRC Staff. The purpose of this meeting was to review portions of the Office of Nuclear Regulatory Research's Thermal-Hydraulic Research Program. Specific topics to be discussed included: (1) the phase separation test program being conducted in the Air-Water Test Loop for Advanced Thermal-Hydraulic Studies ("ATLATS") experimental facility located at Oregon State University (OSU); (2) the status of the TRAC-M code consolidation and documentation effort; and (3) the Rod Bundle Heat Transfer test program being conducted at the Pennsylvania State University. The Subcommittee also reviewed the proposed resolution of Generic Safety Issue 185: "Control of Reactivity Following Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors". The entire meeting was open to the public. Mr. P. Boehnert was the cognizant ACRS staff engineer and Designated Federal Official (DFO) for this meeting. The meeting was convened by the Chairman at 8:30 a.m, June 26, 2002, and adjourned at 5:22 p.m. that day.

ATTENDEES

ACRS Members/Staff:

G. Wallis, Chairman
S. Banerjee, Consultant
V. Ransom, Member
V. Schrock, Consultant
P. Boehnert, DFO

NRC RES Staff

J. Rosenthal
S. Bajorek
J. Kelly
F. Odar
Q. Wu, OSU*
D. Diamond, BNL*
H. Scott
M. DiMarzo, U. MD.*

* NRC Contractor

A list of public attendees is attached to the Office Copy of these Minutes.

The presentation slides and handouts used during the this meeting are attached to the Office Copy of these Minutes. The presentations to the Subcommittee are summarized below.

INTRODUCTORY REMARKS

Dr. Wallis convened the meeting. He had no specific comments regarding the day's discussion topics.

NRC OFFICE OF NUCLEAR REGULATORY RESEARCH PRESENTATIONS

"ATLATS" Phase Separation Research Program

The following topics were discussed regarding the ATLATS Phase Separation Research Program:

- Overview
- ATLATS Facility Description & Recent Experimental Results
- Entrainment Onset Modeling
- Model for Entrainment Rate

In opening remarks, S. Bajorek noted that this program is designed to develop improved models and correlations for characterizing entrainment in a horizontal pipe with an upward oriented branch line. Referring to the Subcommittee's last meeting on this matter at OSU on July 17, 2001, he said that the Program has been re-directed in response to the Subcommittee's concerns expressed during that meeting.

Dr. Q. Wu, OSU, discussed the progress made since last year's Subcommittee meeting. Specific work on the database review and ATLATS tests were noted, including development of an entrainment onset correlation and an entrainment rate model.

RES noted that tests will be conducted in the OSU APEX experimental facility in support of the AP1000 design certification effort. Two series of tests are planned: tests supported by DOE - NERI and a set of tests sponsored by NRC. These two test programs will run concurrently, and are scheduled to commence in October 2002. Some of these tests will address concerns associated with upper plenum entrainment.

Subcommittee Comments on the above presentations included:

- In response to Dr. Ransom, RES noted that the results of the ATLATS Program will be used to improve the TRAC-M and RELAP5 codes' calculations of entrainment.

- In response to Dr. Wallis, RES said that the entrainment modeling will be applicable to the AP600/1000 cases.

- In response to Dr. Banerjee, Dr. Wu indicated that OSU did not study the impact of surface tension effects. Professor Schrock noted that experiments conducted at Berkeley showed a liquid viscosity effect for the case of submerged breaks.

- Dr. Wallis said that the entrainment onset correlation development shows progress. He said that for this correlation, OSU needs to assess the effects of scale-up to the AP1000 plant design. For the entrainment rate model development, he cited a concern with the modeling approach, noting a lack of rigor for the equations shown. He also said that no model exists for the case of a closed pipe (oscillating plugs) and suggested that additional experiments be conducted. Professor Schrock also expressed concern with the modeling approach, stating that the approach lacked coherence.

TRAC-M Code Development and Documentation

Messrs J. Kelly and F. Odar discussed the status of the TRAC-M code development and documentation efforts. Key points noted were:

- The TRAC-M code architecture uprgrade is complete. Work is on-going to retain investment in the legacy input models (RELAP5 & TRAC-B). The legacy work has delayed in release of the TRAC-M code (an alpha-version) until January 2003. Public release of the first code version will be in January 2004.

- Work is proceeding on improvements to the interfacial drag and reflood models. These models will be included in the alpha-code version.

- Long-term development work includes experimental programs at UCLA (subcooled flow boiling), OSU (phase separation), Pennsylvania State Univ. (rod bundle heat transfer), and Purdue/Univ. of Wisconsin (interfacial area transport). RES's goal is to have the results of code assessments drive future experimental needs.

- The existing and planned documentation associated with TRAC-M was reviewed. Most of the current suite of documents will be revised on a yearly basis, in conjunction with the release of new code versions.

Comments noted by the Subcommittee regarding the above discussions included:

- Dr. Wallis requested that the Subcommittee be briefed in detail on the TRAC-M Consolidation Program at a future meeting (**Note**: a meeting has been tentatively planned in December to discuss this matter).

- Drs. Ransom and Banerjee recommended that the work at Purdue/U. of Wisconsin be reviewed by the Subcommittee in the near future (**Note**: a meeting on this matter has been tentatively scheduled for January, 2003).

RBHT Program

The status of the Rod Bundle Heat Transfer Program, underway at Pennsylvania State University was discussed. The long-term objective of this work is the development of mechanistic models for reflood heat transfer, entrainment, and interfacial drag for use in such codes as TRAC-M. Results of test performed to date were briefly discussed. RES proposed that the Subcommittee hold a meeting at Pennsylvania State University to review this work in detail, in the October/November 2002 timeframe. Dr. Banerjee requested that RES provide documentation on the facility instrumentation, prior to the meeting. In response to Professor Schrock, RES said that the phenomena of precursor cooling will be addressed in this test program.

Resolution of Generic Safety Issue (GSI) -185, "Control of Reactivity Following Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors"

GSI-185 addresses the issue that for certain small-break LOCA scenarios, deborated water may accumulate in parts of the RCS (due to condensation cooling in the steam generators). If natural circulation restarts, or the RCP pumps are "bumped", a slug of deborated water may result in a recriticality event, due to dilution of boron in the vessel. B&W-designed reactors are particularly vulnerable to this event.

Two interconnected work products were discussed: (1) development of a "back mixed volume transfer function" by M. DiMarzo to determine the extent of boron dilution seen in the vessel, and (2) a neutronic/thermal-hydraulic analysis, performed by D. Diamond, et al., Brookhaven National Laboratories, to determine the fuel enthalpy seen for the dilution/power excursion event. The BNL calculations show for the case of natural circulation restart, a peak fuel enthalpy of ~37 cal/g.

Comments by the Subcommittee included:

- RES noted, in response to questions, that they will perform a calculation to determine the impact of "pump bump" on fuel enthalpy.

- Dr. Wallis suggested that Dr. DiMarzo perform additional analyses to provide more rigor to his boron dilution analyses, particularly for the flow/mixing seen in the reactor coolant pump. The details of the slug morphology and origin also need to be made available.

- Dr. Wallis requested that RES provide another presentation to the Subcommittee prior to full Committee review to address the above issues.

SUBCOMMITTEE CAUCUS

- The Subcommittee agreed additional work is needed for the ATLATS Phase Separation Program. Written comments on this matter will be provided to RES by the Subcommittee and its Consultants.

- The Subcommittee requested that RES provide a follow-on presentation regarding resolution of GSI-185, prior to review by the ACRS.

FOLLOW-UP ACTIONS

- The Subcommittee will provide written comments regarding the ATLATS Phase Separation research Program for RES's consideration.

- The Subcommittee will hold meetings with RES to discuss the status of key experimental research programs, as well as the status of the TRAC-M Code Consolidation Program. These meetings will be scheduled for later this year and for early 2003.

- The Subcommittee will meet with RES to continue its review of the GSI-185 resolution effort. This meeting is currently scheduled for August 28-29, 2002.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE PRIOR TO THE MEETING

1. Memorandum, P. Boehnert, ACRS, to ACRS Members and T/H Phenomena Subcommittee Consultants, Subject: ACRS Thermal-Hydraulic Phenomena Subcommittee Meeting, June 26-27, 2002 - Status of Selected NRC Office of Nuclear Regulatory Research Programs, dated May 22, 2002, and transmitting:

    - "TRAC-M/Fortran 90 (Version 3.0) Theory Manual", J. W. Spore, et al., Los Alamos National Laboratory, J. H. Mahaffy, C. Murray, Pennsylvania State University, July 2000.

    - "TRAC-M/Fortran 90 (Version 3.0), User's Manual", R. G. Steinke, et al., Los Alamos National Laboratory, February 2000

    - TRAC-M/F77, Version 5.5 Developmental Assessment (DA) Manual, Volumes I & II, B. E. Boyack, et al., Los Alamos National Laboratory, April 2001

    - TRAC-M Programmer's Manual

2. Memorandum, P. Boehnert, ACRS, to ACRS Members and T/H Phenomena Subcommittee Consultants, Subject: ACRS Thermal-Hydraulic Phenomena Subcommittee Meeting, June 26, 2002 - Status of NRC Office of Nuclear Regulatory Research Programs/ Resolution of GSI-185: "Control of Recriticality Following SBLOCAs in PWRs, dated June 11, 2002, and transmitting:

- "Improvement and Evaluation of Models for Liquid Entrainment at an Upward Oriented Vertical Branch Line from a Horizontal Pipe", Progress Report, Q. Wu, et al., Oregon State University, undated

- "Review of Studies on Liquid Entrainment at an Upward Oriented Vertical Branch Line from a Horizontal Pipe (Draft)", Q. Wu, et al., Oregon State University, dated May 8, 2002

- Memorandum to A. Thadani, RES, from F. Eltawila, RES, Subject: GSI-185, Control of Reactivity Following Small-Break LOCAs in PWRs, dated July 7, 2000

- Memorandum to P. Boehnert, ACRS, from J. Rosenthal, RES, Subject: Transmittal of Technical Report for GSI-185, Control of Reactivity Following Small-Break LOCAs in PWRs, dated June 6, 2002.

- B&W Owners Group Analysis Committee Report, 47-5006624-00, Evaluation of Potential Boron Dilution Following Small Break Loss-of-Coolant Accidents, Final Report, Framatome Technologies, Inc., dated January 2000.

••

NOTE: Additional details of the open portions of this meeting can be obtained from a transcript of this meeting available for downloading or viewing via the ADAMS document management system, or can be purchased from Neal R. Gross & Co., Inc., 1323 Rhode Island Ave., NW, Washington, D.C., 20005, (202) 234-4433 (Voice), 387-7330 (Fax), E-Mail: "nrgross@nealrgross.com".

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING:
STATUS OF RES "ATLATS" PHASE SEPARATION TEST PROGRAM/
RESOLUTION OF GSI-185 - RECRITICALITY CONTROL FOR SBLOCAs IN PWRs
JUNE 26, 2002
ROCKVILLE, MARYLAND

PRESENTATION SCHEDULE

Contact: P. Boehnert (301/415-8065) ("pab2@nrc.gov")

TOPIC	PRESENTER	TIME
I. Introduction	**G. Wallis, Chairman**	8:30 a.m.
II. NRC-RES Presentations - Status Of T/H Research Programs		
A. "ATLATS" Phase Separation Test Program		
1. Overview	**S. Bajorek - RES**	8:40 a.m.
2. ATLATS Facility Description & Recent Experimental Results	**Q. Wu - OSU**	9:00 a.m.
3. Entrainment Onset Modeling	**Q. Wu**	9:30 a.m.
BREAK		10:30 a.m.
4. Model for Entrainment Rate	**Q. Wu**	10:45 a.m.
LUNCH		12:00 p.m.
B. TRAC-M Code Development Status	**J. Kelly - RES**	1:00 p.m.
C. TRAC-M Documentation	**F. Odar - RES**	1:15 p.m.
D. PSU- RBHT Program Status	**S. Bajorek**	1:30 p.m.

TOPIC	PRESENTER	TIME
III. Resolution of GSI-185: Recriticality Control for SBLOCAs in PWRs		
A. Scenario Description/ Issue Summary	**H. Scott, RES**	2:00 p.m.
B. Boron Mixing & Concentration	**M. DiMarzo, U. Md.**	2:30 p.m.
	BREAK	3:15 p.m.
C. Neutronics Analysis	**D. Diamond, BNL**	3:30 p.m.
IV. Subcommittee Caucus 1. Comments on Meeting Presentations 2. Follow-on Actions 3. Decision to Bring Review to ACRS/ Instructions to Presenters		4:30 p.m.
V. Recess		4:45 p.m.

SUMMARY: The U.S. Nuclear Regulatory Commission will convene a telephone conference meeting of the Advisory Committee on the Medical Uses of Isotopes (ACMUI) on July 8, 2002. The meeting will take place at the address provided below. At this meeting, the ACMUI will discuss the recommendations from the June 21, 2002, ACMUI subcommittee meeting. The ACMUI subcommittee is charged with formulating recommended changes to the training and experience requirements of authorized users in the revised 10 CFR part 35, Medical Use of Byproduct Material.

DATES: ACMUI will hold a public meeting on Monday, July 8, 2002, from 1 to 5 p.m.

ADDRESS FOR PUBLIC MEETING: U.S. Nuclear Regulatory Commission Auditorium, Two White Flint North Building, 11545 Rockville Pike, Rockville, MD 20852–2738.

FOR FURTHER INFORMATION CONTACT: Linda M. Psyk, telephone (301) 415–0215; e-mail *lmp1@nrc.gov* of the Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

Conduct of the Meeting

Manuel D. Cerqueira, M.D., will chair the meeting. Dr. Cerqueira will conduct the meeting in a manner that will facilitate the orderly conduct of business. The following procedures apply to public participation in the meeting:

1. Persons who wish to provide a written statement should submit a reproducible copy to Linda M. Psyk, U.S. Nuclear Regulatory Commission, Two White Flint North, Mail Stop T8F5, 11545 Rockville Pike, Rockville, MD 20852–2738. Submittals must be postmarked by June 21, 2002, and must pertain to the topics on the agenda for the meeting.

2. Questions from members of the public will be permitted during the meeting, at the discretion of the Chairman.

3. The transcript and written comments will be available for inspection on NRC's Web site (*www.nrc.gov*) and at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852–2738, telephone (800) 397–4209, on or about August 30, 2002. Minutes of the meeting will be available on or about September 9, 2002.

This meeting will be held in accordance with the Atomic Energy Act of 1954, as amended (primarily Section 161a); the Federal Advisory Committee Act (5 U.S.C. App); and the Commission's regulations in Title 10, U.S. Code of Federal Regulations, part 7.

Dated: June 5, 2002.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 02–14622 Filed 6–10–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee Meeting on Thermal-Hydraulic Phenomena; Notice of Meeting

The ACRS Subcommittee on Thermal-Hydraulic Phenomena will hold a meeting on June 26, 2002, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, June 26, 2002—8:30 a.m. until the conclusion of business

The Subcommittee will review portions of the Office of Nuclear Regulatory Research's Thermal-Hydraulic Research Program. Specific topics to be discussed include the Phase Separation Test Program being conducted in the Air-Water Test Loop for Advanced Thermal-Hydraulic Studies ("ATLATS") test facility, and the status of the TRAC-M code consolidation and documentation effort and of the Reflood Test Program being conducted at Pennsylvania State University. The Subcommittee will also review the proposed resolution of Generic Safety Issue (GSI)–185, "Control of Recriticality Following Small-Break LOCAs in PWRs". The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

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

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

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

Further information regarding topics to be discussed, the scheduling of sessions open to the public, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the Designated Federal Official, Mr. Paul A. Boehnert (telephone 301–415–8065) between 7:30 a.m. and 5 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda that may have occurred.

Dated: June 5, 2002.

Sher Bahadur,

Associate Director for Technical Support.

[FR Doc. 02–14620 Filed 6–10–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued, from May 17, 2002, through May 30, 2002. The last

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

JUNE 26, 2002
Date

PLEASE PRINT

ATTENDEES PLEASE SIGN-IN FOR THE MEETING

NAME	NRC ORGANIZATION
J. M. Kelly	RES/DSARE/SMSAB
JAMES HAN	RES/DSARE/SMSAB
Frank Odar	RES/DSARE/SMSAB
Akira Tokuhiro	RES/DSARE/SMSAB
Jack Rosenthal	RES/DSARE/SMSAB
Steve Bajorek	RES/DSARE/SMSAB
David Diamond	BNL
Chris Murray	RES/DSARE/SMSAB
Harold Scott	RES - SMSAB
MARINO DIMARZO	RES
Harold VanderMolen	RES/DSARE/REAHFB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON THERMAL-HYDRAULIC PHENOMENA

JUNE 26, 2002
Date

PLEASE PRINT

ATTENDEES PLEASE SIGN-IN FOR THE MEETING

NAME	AFFILIATION
QIAO WU	OREGON STATE Univ.

Thermal-Hydraulic Test Programs at OSU for Code Development & Validation

Presentation to the ACRS Subcommittees on Thermal-Hydraulic Phenomena

June 26, 2002

Stephen M. Bajorek
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

OBJECTIVES

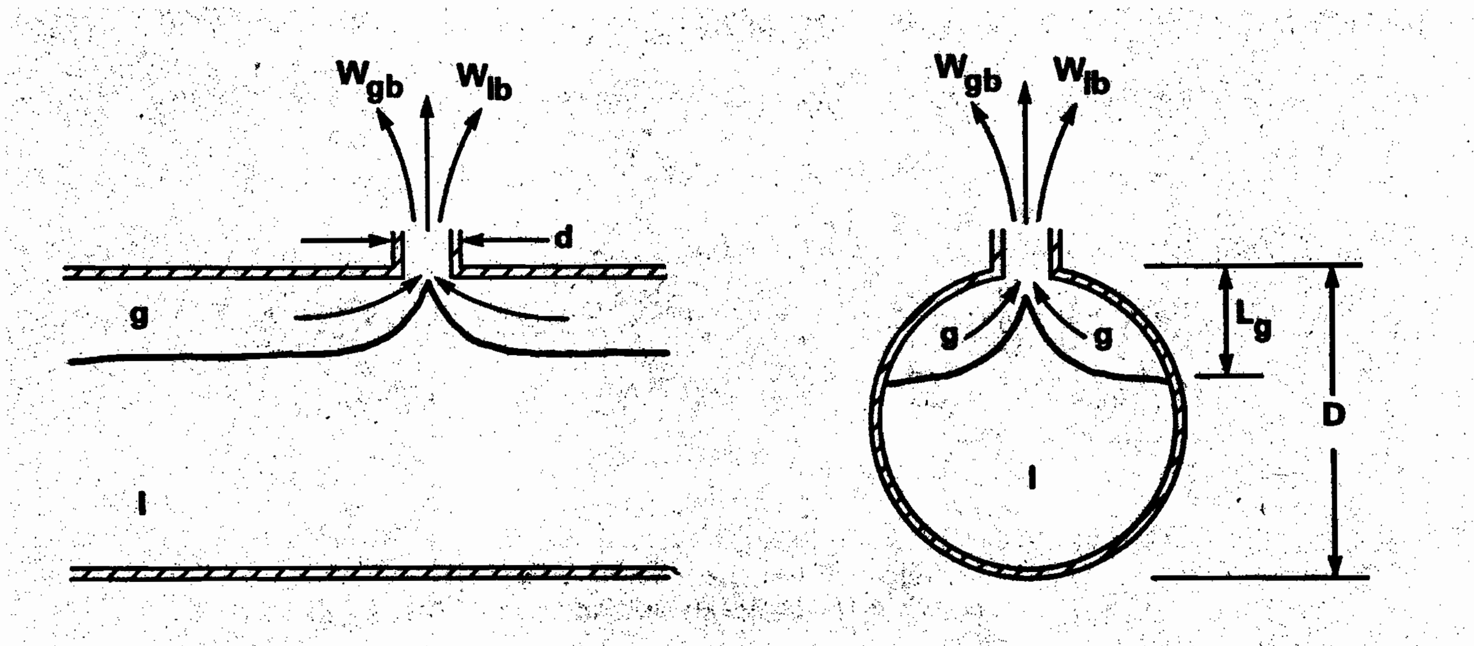
1. Update the Subcommittee on efforts to develop improved models & correlations characterizing entrainment in a horizontal pipe with upward oriented branch line.

2. Outline confirmatory experimental work being planned for OSU facilities (APEX and ATLATS) to address entrainment phenomena.

3. Obtain comments from the Subcommittee on value of the test programs, and suggestions on model development.

Background

- The ATLATS facility was constructed in 1999 and was designed to investigate phase separation at the tee formed by a large diameter pipe and a "small" branch line.



- Initial tests examined geometry applicable to AP600 due to concerns on predicting phase separation in the hot leg at the ADS-4 branch line junction.

AP1000 Scaling & Database Issues

- Entrainment in the hot leg & carryover to the ADS-4 remains an open issue in AP1000:

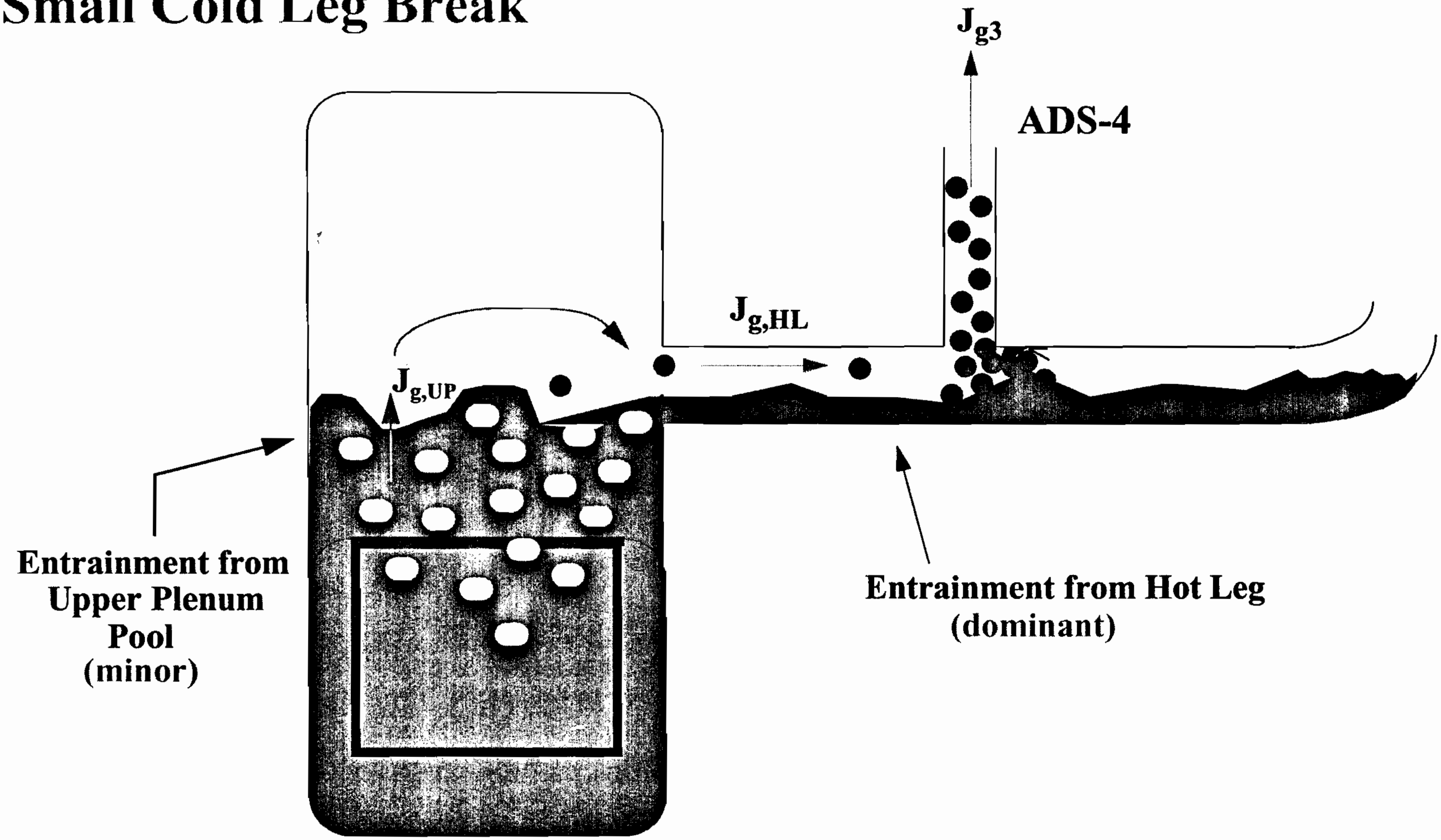
  - Higher Jg in AP1000 hot leg expected to result in earlier onset, and higher entrainment rates

  - Value of $\left.\frac{d}{D_{HL}}\right|_{AP1000}$ is larger than ratio is AP600 or in test facilites used previously to develop phase separation models & correlations.

- Upper plenum pool entrainment & carryover

Hot Leg Entrainment

Small Cold Leg Break



Summary of Main Issues from July 2001 Meeting

- **Literature Review: Not focussed on upward oriented branchline, and prior work improperly referenced.**
- **Flow Patterns: The flow patterns observed in the hot leg were highly oscillatory. Typical flow pattern descriptions from co-current horizontal flows do not apply.**
- **Model Development: Models were preliminary. Based primarily on horizontal-stratified flow assumptions even though the flow was oscillatory.**

Program Re-Direction

1. Revise & reduce the literature survey and corresponding database. Focus only on prior work for upward oriented branch line.

2. Modeling efforts should assume a physical situation similar to that observed in ATLATS and expected to occur in APEX and AP600/AP1000: Flow patterns and entrainment dominated by a coherent "oscillating plug" between branch line and SG inlet plenum.

3. Objective is to develop models to predict the onset of entrainment, and the net "global" entrainment rate.

Phase Separation at an Upward Oriented Vertical Branch in a Horizontal Pipe

Q. Wu, K.B. Welter, Y. Yao, J.N. Reyes
Advanced Thermal Hydraulics Research Laboratory

Presentation to the ACRS of NRC
June 26, 2002

Progress Since Last ACRS Review Meeting on July 17, 2001

- Database Review Update
- ATLATS Test Facility
  - Liquid Level Probe Sampling Rate Evaluation
- Entrainment Onset Study
  - Tests (air injection from the vessel top)
  - Entrainment Onset Correlation Development
- Entrainment Rate Study
  - Tests
  - Entrainment Rate Model Development

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Oregon State University

Database Review Update

Nuclear Engineering & Radiation Health Physics

Oregon State University

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Database Review Update

- Focus on liquid entrainment at an upward oriented vertical branch in a horizontal pipe
- Database includes:
  - Description of each test facility
  - Test conditions
  - Instrumentation
  - Model Development
  - Cross Comparison

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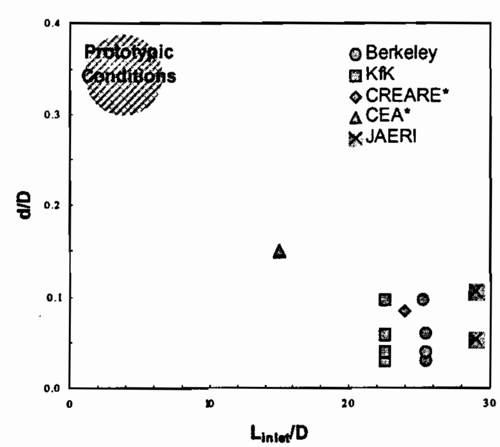
Database Review Update

Date Published	Authors and Institution	Experiment		Correlations
		Fluids	D/d*	
1980	Zuber (NRC)	N/A	N/A	Scaling, correlations
1981	Crowley and Rothe (Creare Inc.)	Air-water	12	Entrainment Onset
1984	Reimann, Khan and Smoglie (KfK)	Air-water	34, 26, 17, 10	Entrainment Onset Entrainment Rate
1986	Schrock, Revenkar, and Mannheimer (Berkeley)	Air-water Steam-water	31, 25, 17, 10	Entrainment Onset Entrainment Rate
1989	Maciaszek and Micaelli (CEA)	Steam-water	6.8	Entrainment Onset Entrainment Rate
1991	Yonomoto and Tasaka (JAERI)	Air-water	~19, 8.5	Entrainment Rate

Database Review Update
(Summary of Geometric Conditions)



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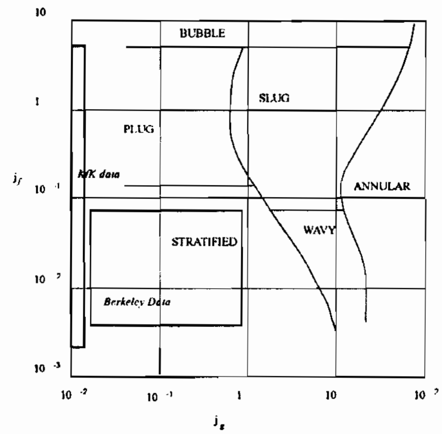
Database Review Update

(Range of the Entrainment-Rate Test Data)



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Database Review Update

- For advanced plants, d/D_{AP1000}=0.47 and d/D_{AP600}=0.34, significantly greater than those of previous investigations
- The gas superficial velocity range in previous tests is much lower than that in the advanced plant (>1 m/s)
- Previous experimental investigations were applicable to co-current stratified flow. Traditional flow regime map may not be adequate for this investigation.

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Database Review Update

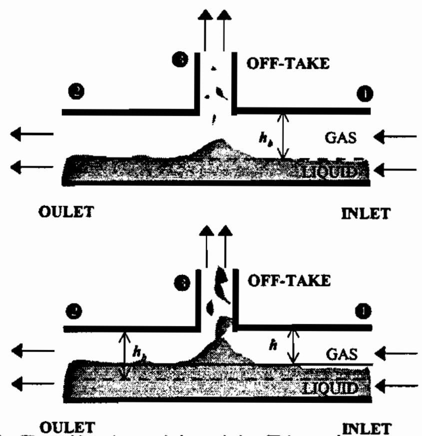
Existing Correlations

Onset of Entrainment

$$\left(\frac{h_b}{d}\right) \sim \left(\frac{V_{g3}^2}{gd}\right)\left(\frac{\rho_{g1}}{\Delta\rho}\right)$$

Steady-State Entrainment

$$x_3 \sim (h / h_b)$$



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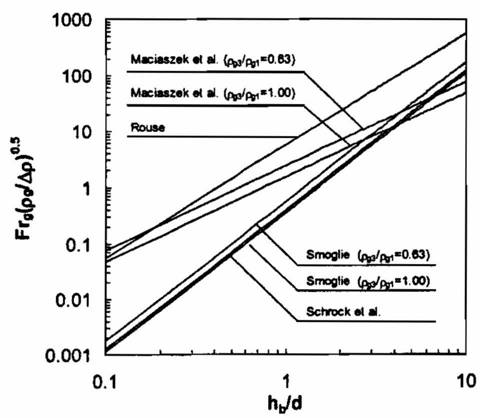
Database Review Update

Existing Entrainment Onset Correlation



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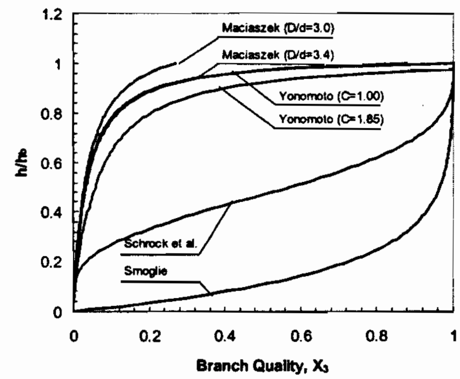
Database Review Update

Entrainment Rate Correlation Cross Comparison



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ATLATS Test Facility

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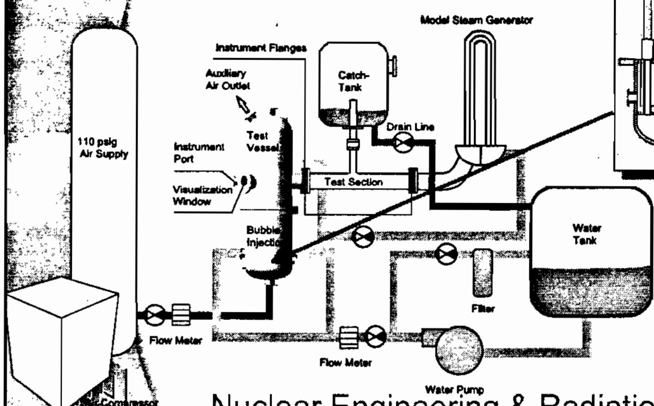

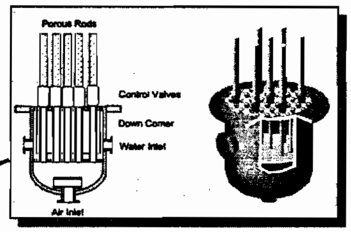

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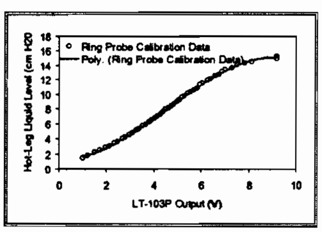
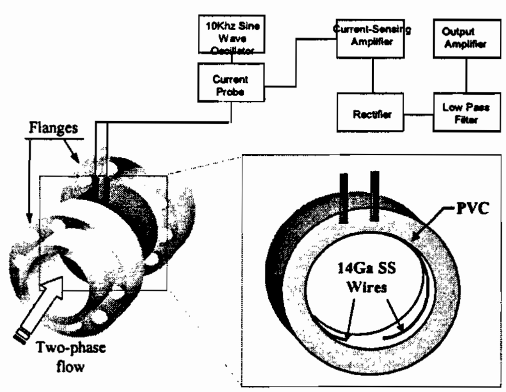
Liquid Level Probe Sampling Rate Evaluation

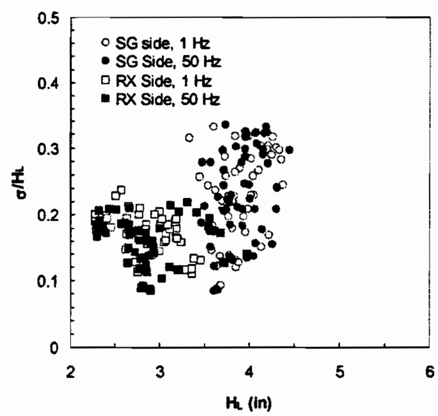




Liquid Level Probe Sampling rate Evaluation in Entrainment Rate Tests

(4 minute Duration)

- The scattering is due to actual liquid level fluctuations
- 1 Hz vs. 50 Hz sampling rate, similar scattering range:
  - SG side, 10% ~ 34%
  - RX side, 10% ~ 24%
- H_L (SG) > H_L (RX)



Entrainment Onset Studies

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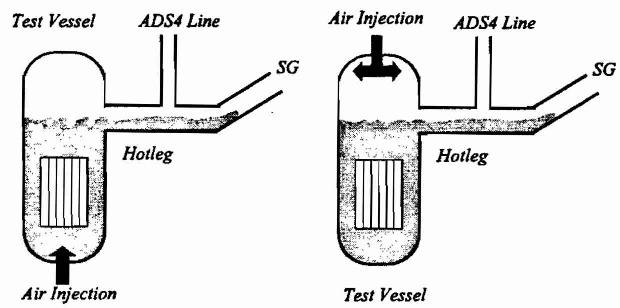
Entrainment Onset Test with Air Injection from the Vessel Top



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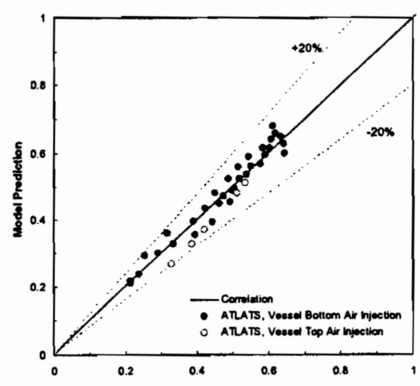
Entrainment Onset Test with Air Injection from the Vessel Top



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Entrainment Onset Criterion Development

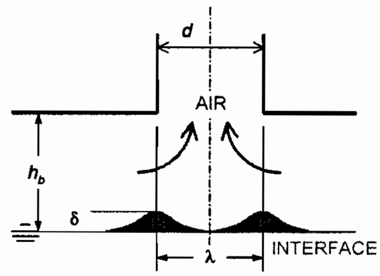
- Original Approach

$$\rho_1 V_1 \pi \lambda (h_b - \delta) = \rho_z V_3 \pi d^2 = w_{3g}$$

$$\Delta P = \Delta \rho g \delta = \frac{1}{2} \rho_1 V_1^2$$

$$\lambda^2 \delta (h_b - \delta)^2 = \frac{1}{2\pi} \frac{w_{3g}^2}{\rho_1 \Delta \rho g}$$

$\delta = 1/3\ h_b$.



Maciaszek/Bharathan et al. approach: $\lambda = d$

$$h_b = \frac{3}{2\pi^{2/3}} \left(\frac{w_{3g}^2}{\rho_g \Delta \rho g d^2} \right)^{1/3}$$

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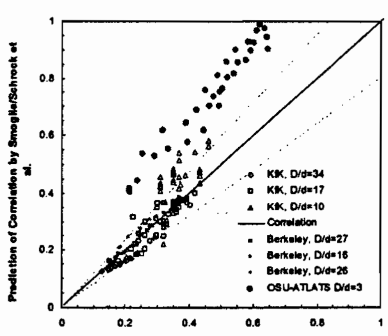
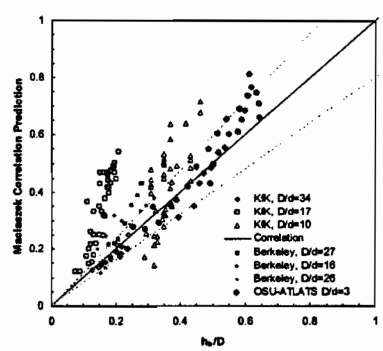
Entrainment Onset Criterion Development

KfK/Berkeley Correlation



Correlation for small d/D ratio

CEA Correlation

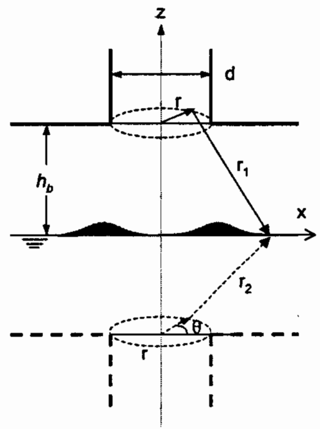


Correlation for large d/D ratio

Entrainment Onset Criterion Development

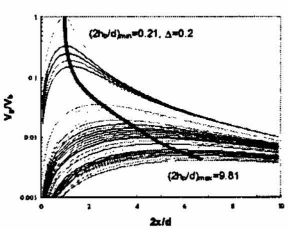
- Modification

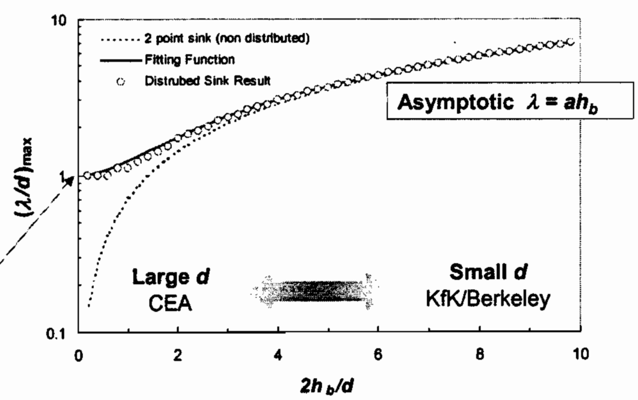
  To find λ, potential flow of 2 distributed mirror sinks is considered.



$$V_x = \left(\frac{4w_{3g}}{\pi^2 \rho_g d^2}\right) \int_0^{d/2}\int_0^{2\pi} \frac{(x - r\cos\theta)}{\left[h_b^2 + (r\sin\theta)^2 + (x - r\cos\theta)^2\right]^{3/2}} r\,d\theta\,dr$$

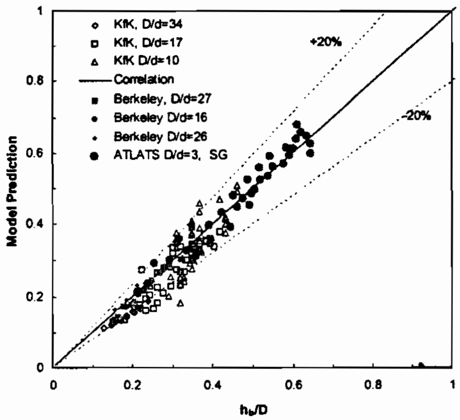
Entrainment Onset Criterion Development





Wave crest spacing minimum $d \leq \lambda$

Entrainment Onset Criterion Development



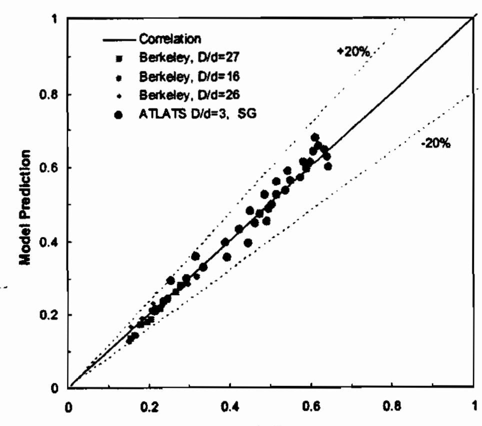
$$\Delta P = \Delta\rho g\delta = \frac{1}{2}\rho_1\left(V_{1,crest}^2 - V_1^2\right)$$

λ/d

$$\left(w_{3g}^2\right)^* = K\left(\frac{h_b}{d}\right)^3 \left[a\left(\frac{h_b}{d}\right)+1\right]^2 \left[1-\left(\frac{h_b}{D}\right)^2\right]^{-1}$$

$$\left(w_{3g}^2\right)^* \equiv \frac{w_{3g}^2}{d^5 \rho_g \Delta\rho g}$$

Entrainment Onset Criterion Development



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Entrainment Rate Studies

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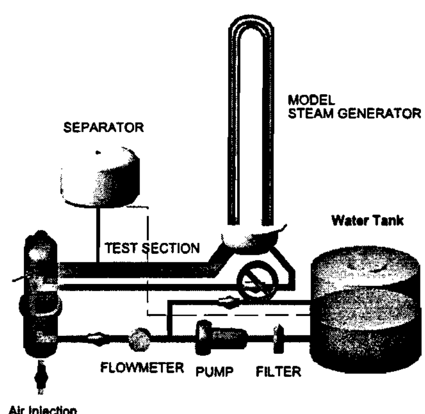
Entrainment Rate Tests



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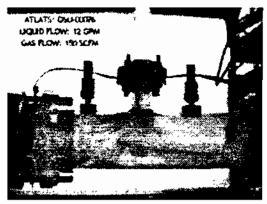
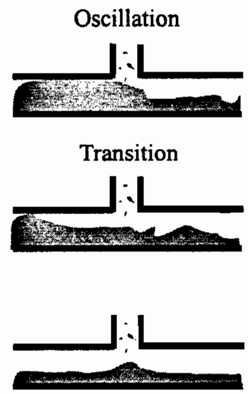
Flow Patterns





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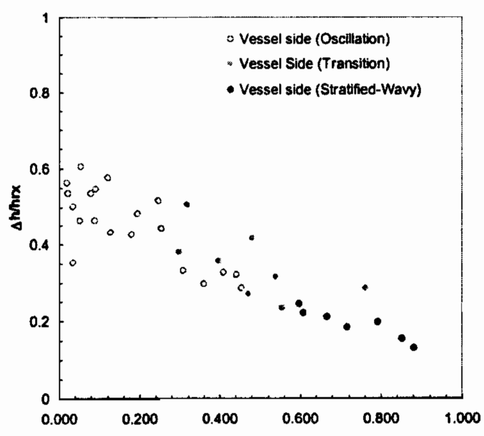
Flow Patterns



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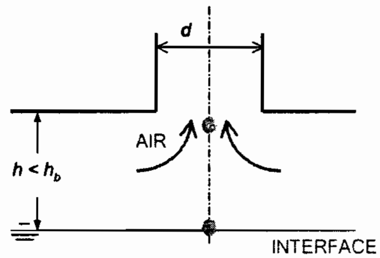
Entrainment Rate Model Development

When $h < h_b$, liquid entrainment occurs



$$\frac{\rho_f V_{f3}^2}{2} \propto \left(\frac{\rho_{g1} V_{g1}^2}{2}\right) - C\Delta\rho g h$$

$$w_{f3} \propto (1-\alpha_3) A_3 \sqrt{2\Delta\rho\rho_f g h} \sqrt{\frac{C'}{2h(\alpha_1 A_1)^2}\left(\frac{w_{g3}^2}{\Delta\rho\rho_g g}\right) - 1}$$

$$C' = 2(\alpha_{1b} A_1)^2 h_b \frac{\Delta\rho\rho_g g}{w_{g3}^2}$$

Entrainment Rate Model Development

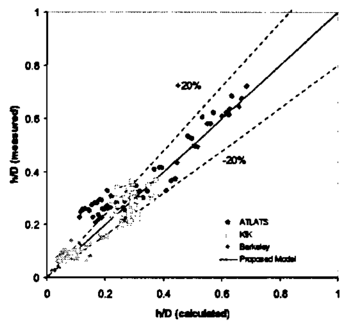
$$x_3 = \frac{1}{1 + w_{f3}/w_{g3}}$$

$$\frac{w_{f3}}{w_{g3}} = C_1 \frac{\left(1 - \left(\frac{h}{h_b}\right)\right)\sqrt{\left(\frac{\rho_f}{\rho_g}\right)\left(\frac{h}{h_b}\right)\left(1 - \left(\frac{h_b}{D}\right)^2\right)\left[\left(\frac{h_b}{h}\right)\left(\frac{\alpha_{1b}}{\alpha_1}\right)^2 - 1\right]}}{\left(\frac{h_b}{d}\right)\left(a\frac{h_b}{d} + 1\right)}$$

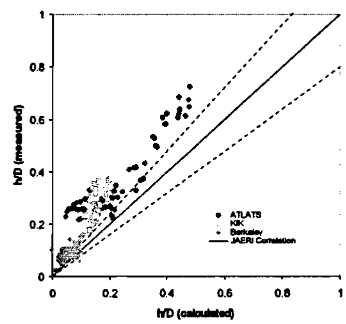
- Function of D, h_b, h, d, and ρ_f/ρ_g
- Relies on accurate estimation of h_b at the given gas flow rate

Entrainment Rate Model Development

Present Model

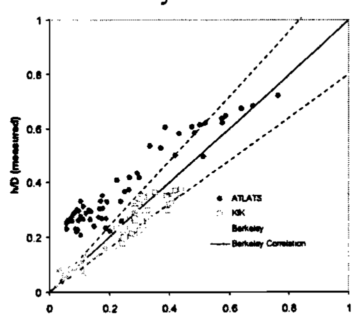


JAERI Correlation

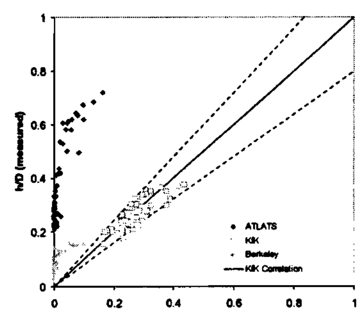


Entrainment Rate Model Development

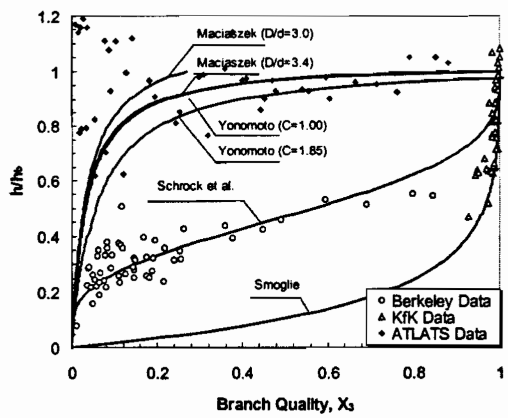
Berkeley Correlation



KfK Correlation



Entrainment Rate Model Development



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Summary

- Database Improvement
  - The new Database focuses solely on previous investigations of liquid entrainment in an upward oriented vertical branch of a horizontal pipe.
- Entrainment Onset Experiments
  - Air injection from the vessel top did not have much affect on the entrainment onset condition (<10%)
  - Data sampling rate for the liquid level probe was appropriate for a duration of ~4 min

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Summary (continued)

- Entrainment Onset Model Development
  - Simplified formulation
  - Considered gas velocity effects in the main pipe
  - The new model agreed with available test data of different geometry, scale and fluid properties (±20% of accuracy)
- Entrainment Rate Experiments
  - Entrainment rate tests were focused on cases with steam generator (oscillatory, transition and stratified-wavy flows)

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Summary (continued)

- Entrainment Rate Model Development
  - Proposed a model based on a mechanical energy balance approach
  - The mechanistic model predicted the trends of different data sets with a reasonable accuracy (an improvement compared to other correlations)

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Related & Future Activities

■ Integration and Validation with TRAC-M

- ◆ Model development & data analysis to continue. Need to establish conditions for hot leg flow pattern transitions in ATLATS, APEX and AP600/AP1000.
- ◆ RELAP model of ATLATS developed. Future efforts will impliment and validate new models by TRAC-M simulations of ATLATS and APEX.

■ Confirmatory Intergal Effects Testing

- ◆ Cooperated with DOE defining DOE-NERI test matrix for advanced plant studies at OSU using the APEX facility.
- ◆ DOE-NERI test matrix to include several tests useful for investigating upper plenum entrainment processes.
- ◆ NRC Confirmatory Test Matrix assumes DOE/NERI tests performed. Focus of NRC tests on Beyond-Design-Basis conditions and to assist in NRC code development & analysis activities.

Table 1: Tentative DOE - NERI Test Matrix

	Comment
1	**Double-ended guillotine break of direct vessel injection line with ADS-4 valve**: Design basis DEDVI with single ADS-4 failure.
2	**2-inch break in bottom of cold leg 3 (CMT side) with 3/4 ADS-4**: Design basis 2-inch break case.
3	**Double-ended DVI break with failure of intact side accumulator**: Beyond design basis case.
4	**Primary Loop Characterization Single-Phase Natural Circulation**: Provides assessment of loop pressure drop.
5	**"No Reserve" Test**. Using AP1000 expected initial conditions for ADS blowdown.
6	**"No Reserve" Test**. Using AP1000 expected initial conditions for ADS blowdown.
7	**SS Entrainment Test**: 3 open ADS-4 valves, 0 psig containment back pressure, UP internals in.
8	**SS Entrainment Test**: 3 open ADS-4 valves, 25 psig containment back pressure, UP internals in.
9	**SS Entrainment Test**: 4 open ADS-4 valves, 0 psig containment back pressure, UP internals in.
10	**SS Entrainment Test**: 2 open ADS-4 valves (both on one side), 0 psig containment back pressure, UP internals in.
11	**SS Entrainment Test:** 4 open ADS-4 valves, 0 psig containment back pressure, UP internals out.

Upper Plenum Entrainment

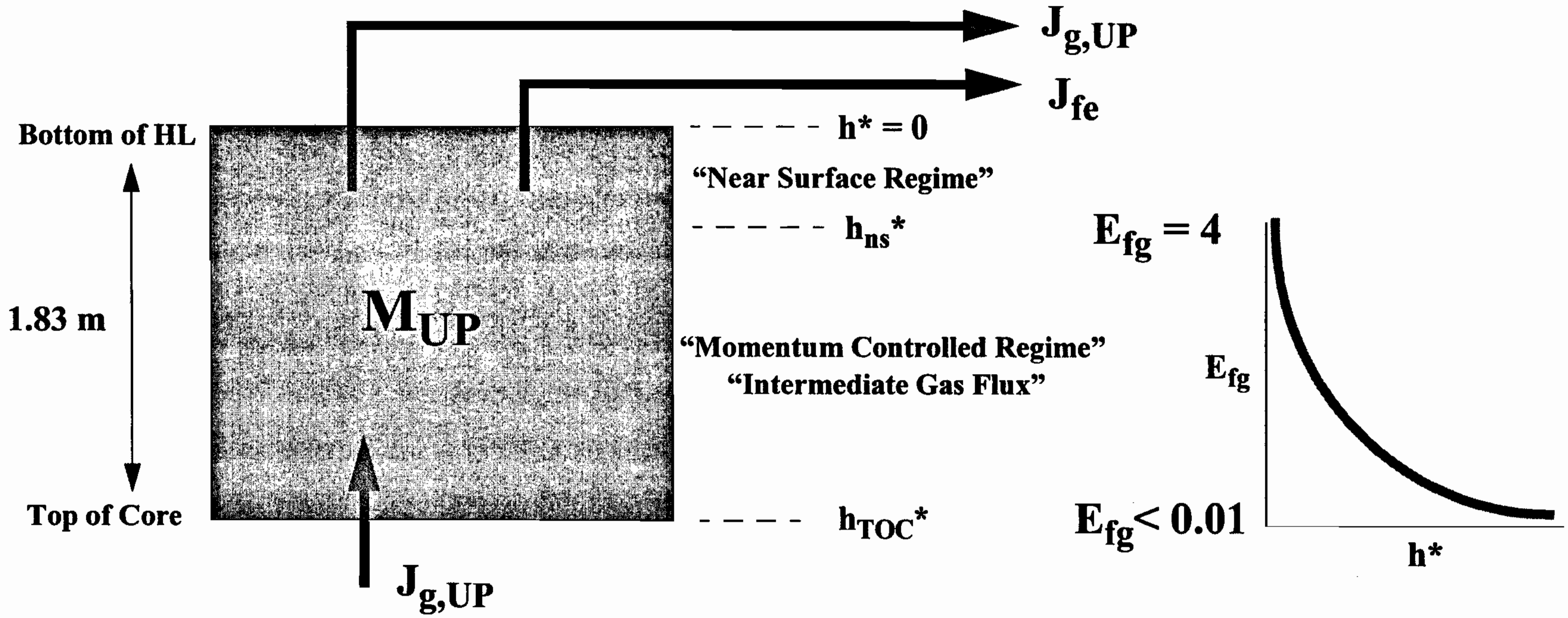


Table 2: Proposed NRC Confirmatory Test Matrix

Test ID	Priority	Comments
NRC-AP1000-1	M	**Double-ended guillotine break of direct vessel injection line with failure of ADS-1/2/3:** DEDVI with failure of ADS-1/2/3, forcing ADS-4 to provide blowdown.
NRC-AP1000-2	H	**Mode 5 (Cold Shutdown) Operation with Loss of RNS Cooling:** Determines if RCS would repressurize sufficiently to prevent IRWST injection and if CCFL prevented pressurizer drainage. (Note: New surge line being added to APEX-AP1000 facility.)
NRC-AP1000-3	M	**Double-ended guillotine break of direct vessel injection line with failure of PRHR, 50% of ADS4-2:** Beyond design basis test.
NRC-AP1000-4	H	**1-inch cold leg break with degraded sump:** Similar test for AP600 used to showed level at which liquid in sump was no longer able to support recirculation.
NRC-AP1000-5	M	**Double-ended guillotine break of non-pressurizer loop direct vessel injection line with failure 50% of ADS4-2:** Beyond design basis test with variation of break location.
NRC-AP1000-6	H	**2-inch DVI break with single ADS-4 valve failure:** Design basis type case not previously considered. Provides effect of break size.
NRC-AP1000-7	M	**Station Blackout.** Addresses PRA, and may also provide information on PRHR performance.
NRC-AP1000-8	H	**SS Entrainment Test:** 2 open ADS-4 valves (both on one side), 60 psig containment back pressure (or max.allowable, UP internals in.
NRC-AP1000-9	H	**SS Entrainment Test:** 2 open ADS-4 valves (both on one side), 60 psig containment back pressure (or max. allowable pressure), UP internals out.
NRC-AP1000-10	H	**SS Entrainment Test:** 2 open ADS-4 valves (both on one side), 0 psig containment back pressure, UP internals out. Comparison to one of the DOE-NERI tests provides low pressure de-entrainment sensitivity.
NRC-AP1000-11	H	**"No Reserve" Test.** Initial pressure = 100 psia, containment backpressure = 25 psia and with core power = 1000 kW. Corresponds to NRC-6425.

R

United States Nuclear Regulatory Commission

TRAC-M
Code Consolidation and Development

Presented to the ACRS Thermal-Hydraulic and Severe Accident Subcommittee

by

Joseph M. Kelly

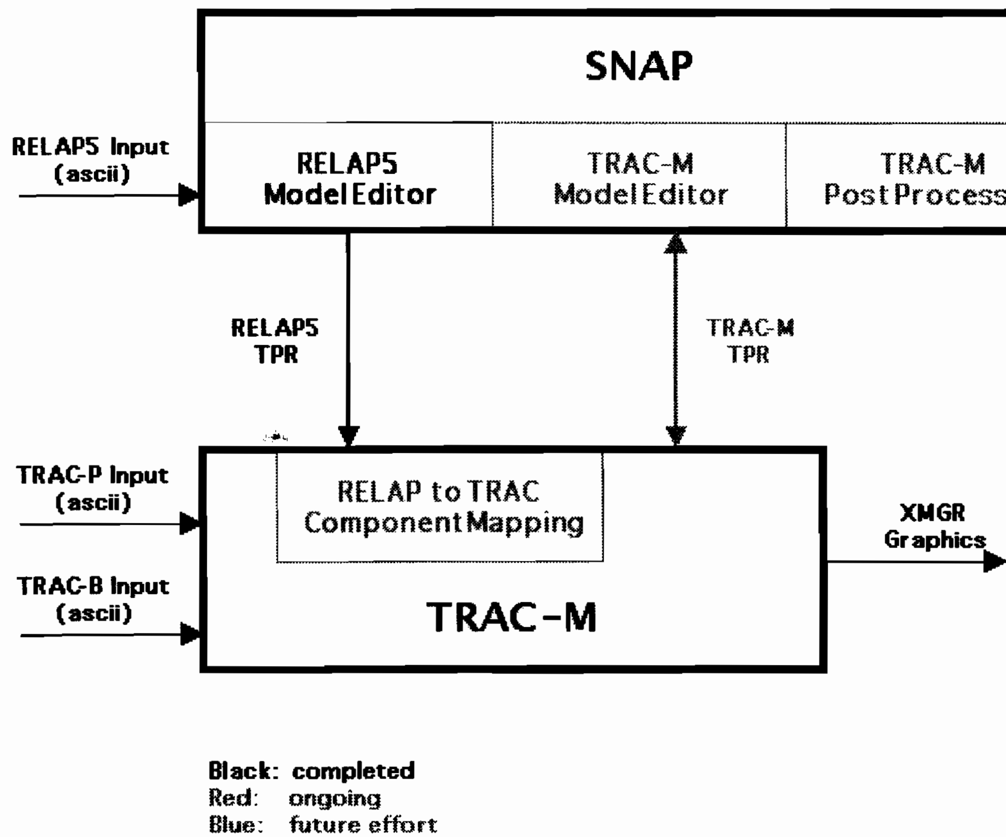
June 26, 2002

TRAC-M: Code Consolidation and Development

- TRAC-M Development Objectives
  - Modern Architecture
  - Code Consolidation:
    - Recover modeling capabilities of predecessor codes (Ramona,TRAC-P, TRAC-B, RELAP5), and
    - Retain investment in legacy input models (RELAP5 & TRAC-B).
    - ➔ Success Metric: simulation fidelity must be equal to or better than that of predecessor codes for their targeted application.
  - Ease of Use
  - Accuracy
  - Numerics

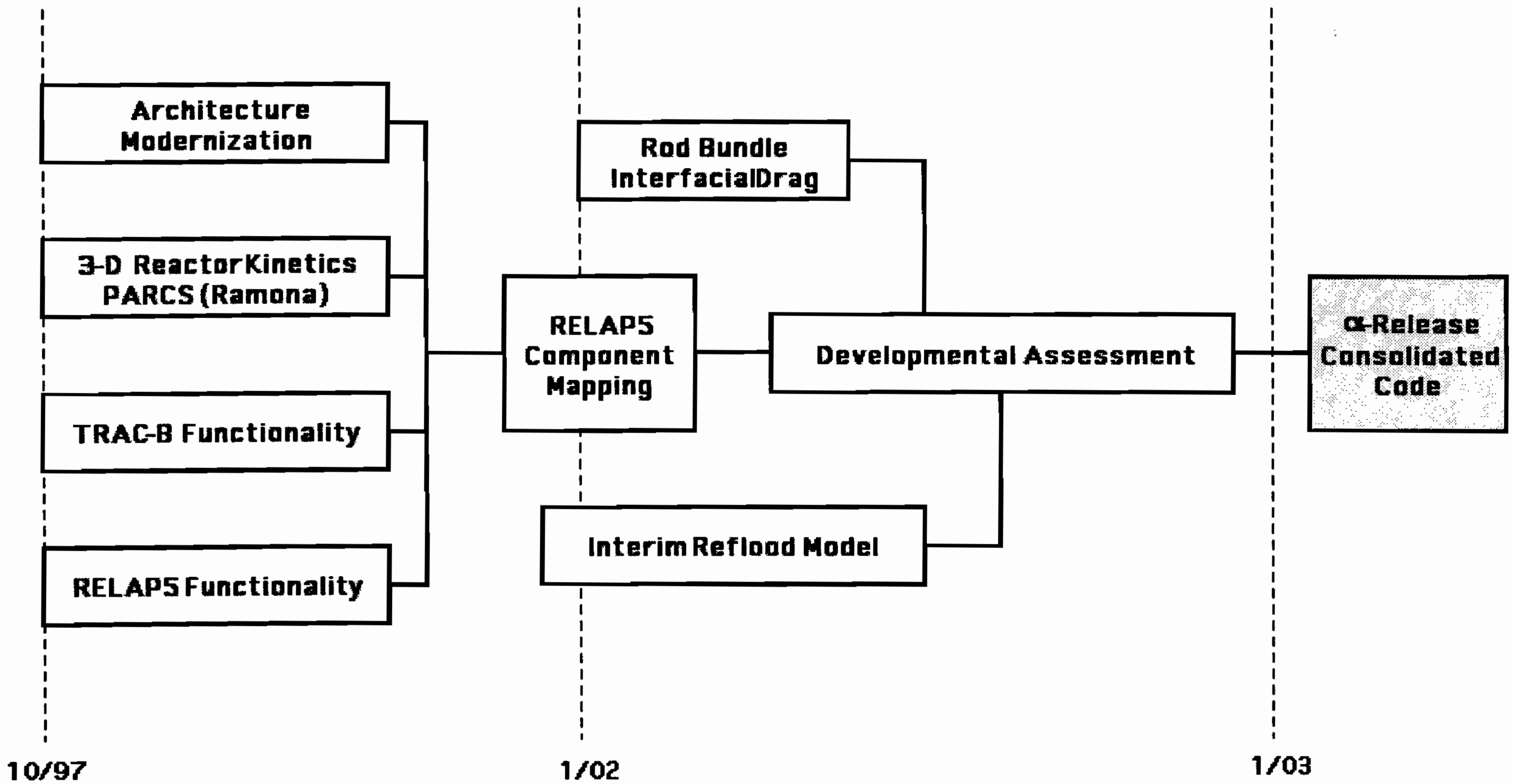
TRAC-M: Code Consolidation and Development

- Legacy Input Models



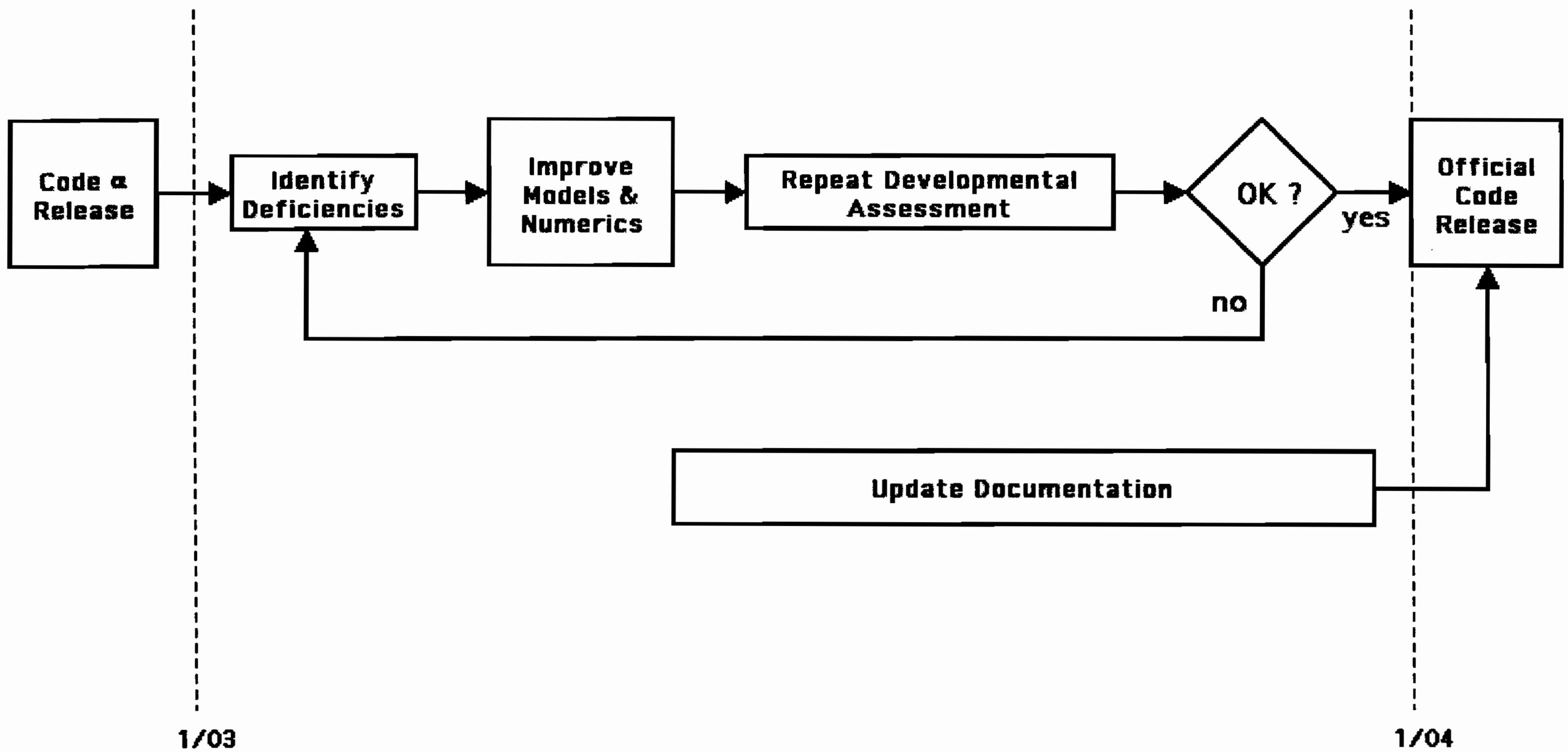
TRAC-M: Code Consolidation Plan & Status

- Calendar Year 2002 Activities:



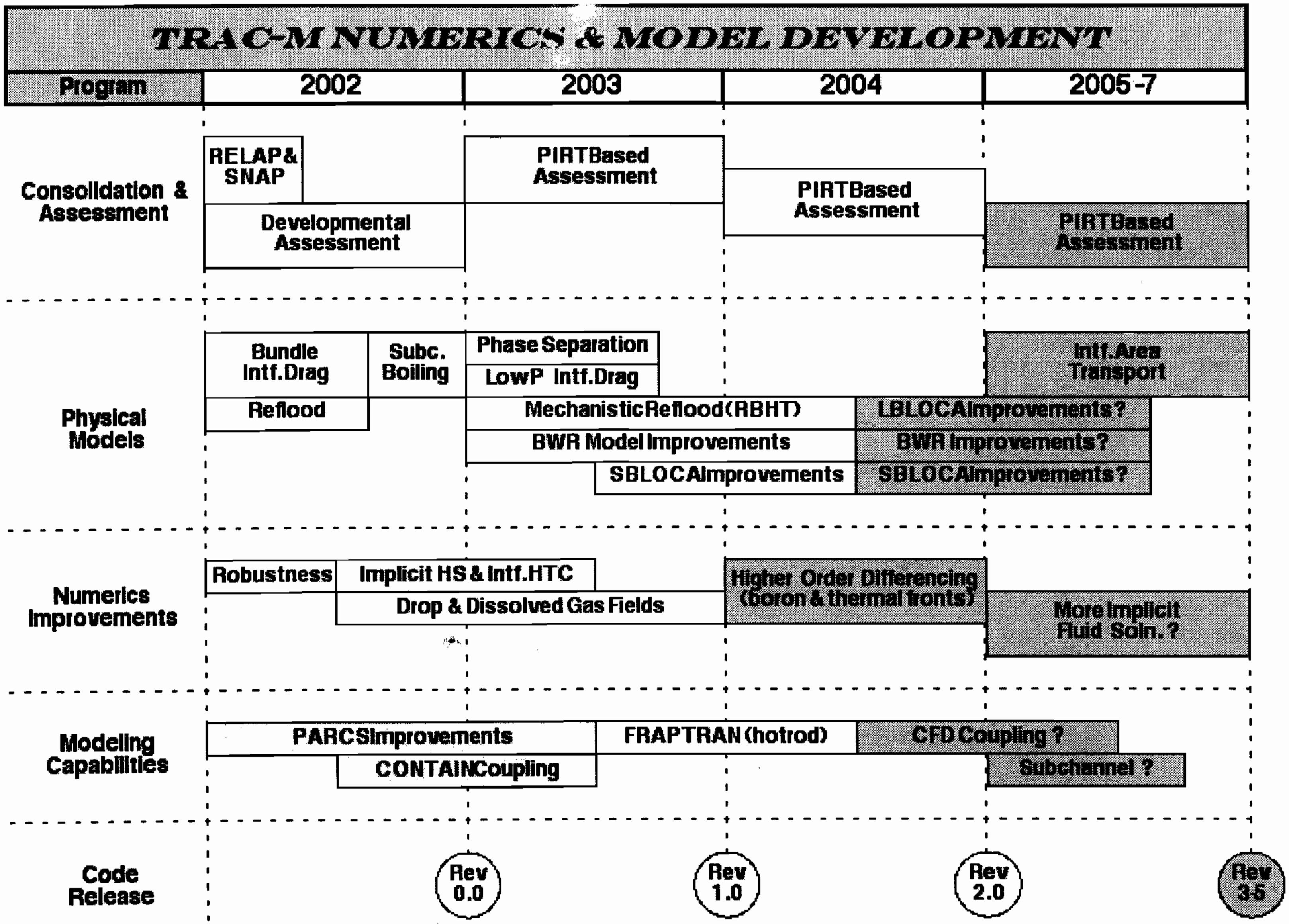
TRAC-M: Code Consolidation Plan & Status

- Calendar Year 2003 Activities:



TRAC-M Current Model Development

- Bundle Interfacial Drag:
  - Necessary for Peach Bottom Turbine Trip benchmark.
    - Implement TRAC-B interfacial drag and heat transfer models.
      - Apply to CHAN component (BWR fuel assembly).
      - Apply to 3-D Vessel core region.
    - Implement low-level modularization of interfacial drag package.
    - In-house effort: Joe Staudenmeier & Tony Ullses
- Reflood Model (interim)
  - Necessary for realistic auditing calculations of AP-1000.
      - Current model has unacceptably large oscillations and is highly conservative for separate effects tests.
    - Physical models and fine-mesh rezoning numerical scheme.
    - In-house effort: Weidong Wang & Joe Kelly



TRAC-M NUMERICS & MODEL DEVELOPMENT
Program
2002
2003
2004
2005-7
Consolidation & Assessment
RELAP& SNAP
Developmental Assessment
PIRTBased Assessment
PIRTBased Assessment
PIRTBased Assessment
Physical Models
Bundle Intf.Drag
Subc. Boiling
Reflood
Phase Separation
LowP Intf.Drag
Intf.Area Transport
MechanisticReflood(RBHT)
LBLOCAImprovements?
BWR Model Improvements
BWR Improvements?
SBLOCAImprovements
SBLOCAImprovements?
Numerics Improvements
Robustness
Implicit HS & Intf.HTC
Drop & Dissolved Gas Fields
Higher Order Differencing (boron & thermal fronts)
More Implicit Fluid Soln.?
Modeling Capabilities
PARCSImprovements
FRAPTRAN (hotrod)
CFD Coupling ?
CONTAINCoupling
Subchannel ?
Code Release
Rev 0.0
Rev 1.0
Rev 2.0
Rev 3-5

TRAC-M Long-Term Development

- Incorporation of Experimental Results
  - UCLA Subcooled Boiling
    - Targeted to known code deficiency, implementation in late 2002.
  - OSU Phase Separation
    - Extension of data base to larger off-take diameter ratio and non-stratified regimes.
    - Targeted to known code deficiency, implementation in 2003
  - PSU Rod Bundle Heat Transfer
    - Designed to provide detailed measurements for model development.
      - Reflood tests to be conducted in 2002-2003.
      - Steam cooling/drop injection tests in 2003.
      - Data analysis & model development in 2003-2004.
  - Purdue/UW Interfacial Area Transport
    - Exploratory research program with the potential for a revolutionary improvement in two-phase flow modeling capability.
      - Implementation to begin in 2005, data can be used for model assessment.
  - Code assessment results => future experimental programs.

TRAC-M: Code Consolidation and Development

- Summary
  - Code development associated with consolidation will be completed by the end of summer 2002.
  - Developmental assessment will be conducted in the second half of 2002.
  - Both interfacial drag and reflood models will be improved for inclusion in the consolidated code.
  - Initial α-release of the consolidated code at end of 2002.
  - Initial public release of the consolidated code at end of 2003.
  - Long-term code development and experimental programs to be driven by assessment results and user needs.

R

STATUS AND PLANS FOR TRAC-M DOCUMENTATION

FRANK ODAR

ACRS T&H SUBCOMMITTEE MEETING

JUNE 26, 2002

EXISTING DOCUMENTATION

- TRAC-M/FORTRAN 90 (VERSION 3.0) THEORY MANUAL, NUREG/CR-6724, JULY 2001
- TRAC-M/FORTRAN 90 (VERSION 3.0) USER'S MANUAL, NUREG/CR-6722, MAY 2002
- TRAC-M/F77, VERSION 5.5, DEVELOPMENTAL ASSESSMENT MANUAL, NUREG/CR-6730, JULY 2001
- TRAC-M/FORTRAN 90 (VERSION 3.0) PROGRAMMER'S MANUAL, NUREG/CR-6725, MAY 2001
- ASSESSMENT OF MODERNIZATION AND INTEGRATION OF BWR COMPONENTS AND SPACIAL KINETICS IN THE TRAC-M, VERSION 3690, CODE, NUREG-1752, DECEMBER 2001
- ASSESSMENT OF TRAC-M CODES USING FLECHT-SEASET REFLOOD AND STEAM COOLING DATA, NUREG-1744, MAY 2001
- SOFTWARE QUALITY ASSURANCE PROCEDURES FOR NRC THERMAL HYDRAULIC CODES, NUREG 1737, DECEMBER 2000

PLANNED DOCUMENTATION

- TRAC-M, (VERSION ____), USER'S GUIDE-DRAFT(1) (TARGET DATE: 12/31/02)
- TRAC-M, (VERSION ____), THEORY MANUAL-DRAFT(1) (TARGET DATE: 12/31/02)
- TRAC-M, (VERSION ____), USER'S GUIDE- DRAFT(2) (TARGET DATE: 03/31/03)
- TRAC-M, (VERSION ____), THEORY MANUAL-DRAFT(2) (TARGET DATE: 03/31/03)
- TRAC-M, (VERSION ____), DEVELOPMENTAL ASSESSMENT MANUAL-DRAFT(1) (TARGET DATE: 03/31/03)
- TRAC-M, (VERSION ____), USER'S GUIDE (TARGET DATE: 12/31/03)
- TRAC-M, (VERSION ____), THEORY MANUAL (TARGET DATE: 12/31/03)
- TRAC-M, (VERSION ____), DEVELOPMENTAL ASSESSMENT MANUAL (TARGET DATE: 12/31/03)
- TRAC-M, (VERSION ____), PROGRAMMER'S MANUAL (TARGET DATE: 12/31/04)
- ASSESSMENT REPORTS ON INDIVIDUAL TEST CASES (REPORTS TO BE PUBLISHED AS WORK IS COMPLETE)

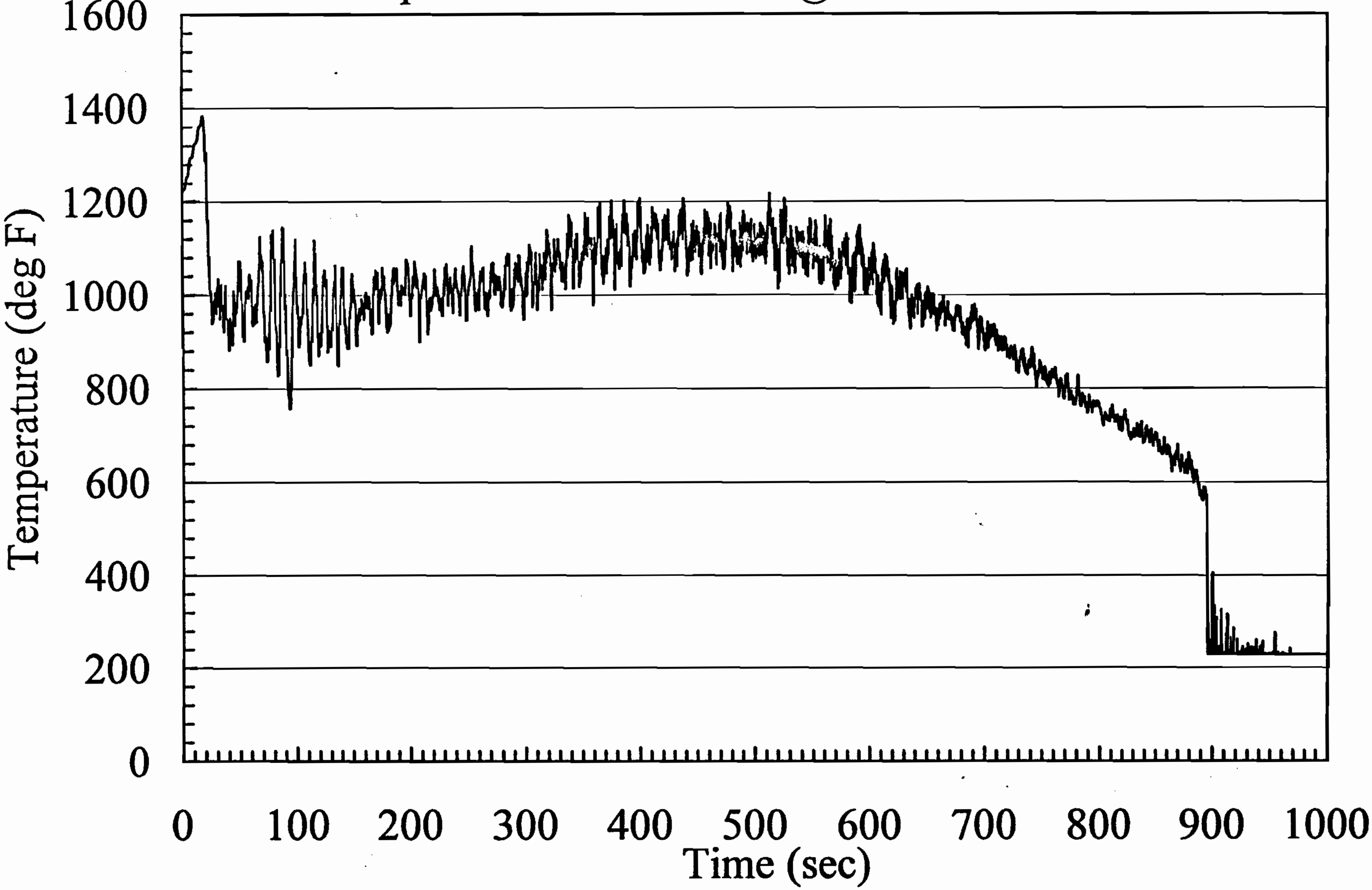
Rod Bundle Heat Transfer (RBHT) Program Status

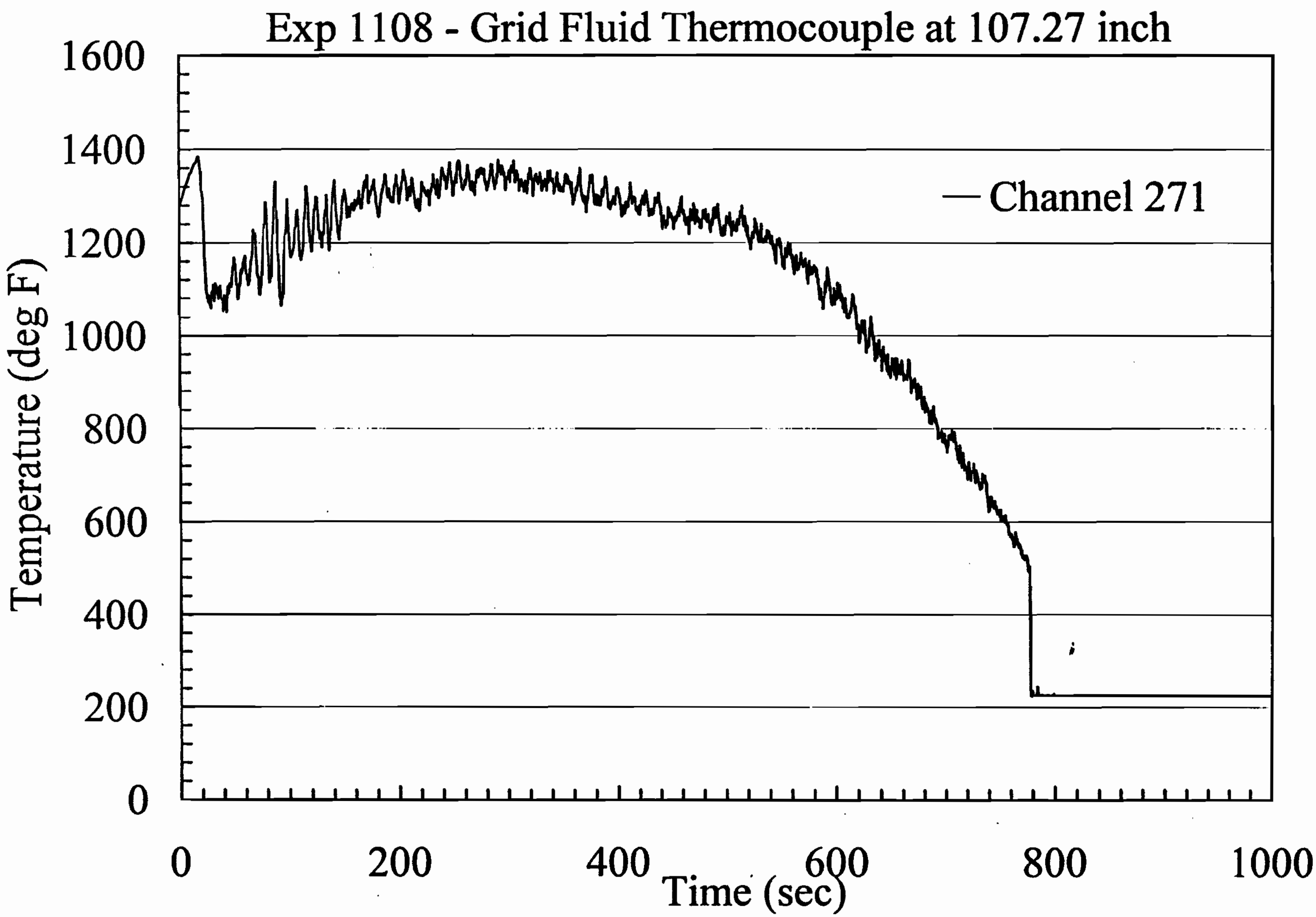
Presentation to the ACRS Subcommittees on Thermal-Hydraulic Phenomena

June 26, 2002

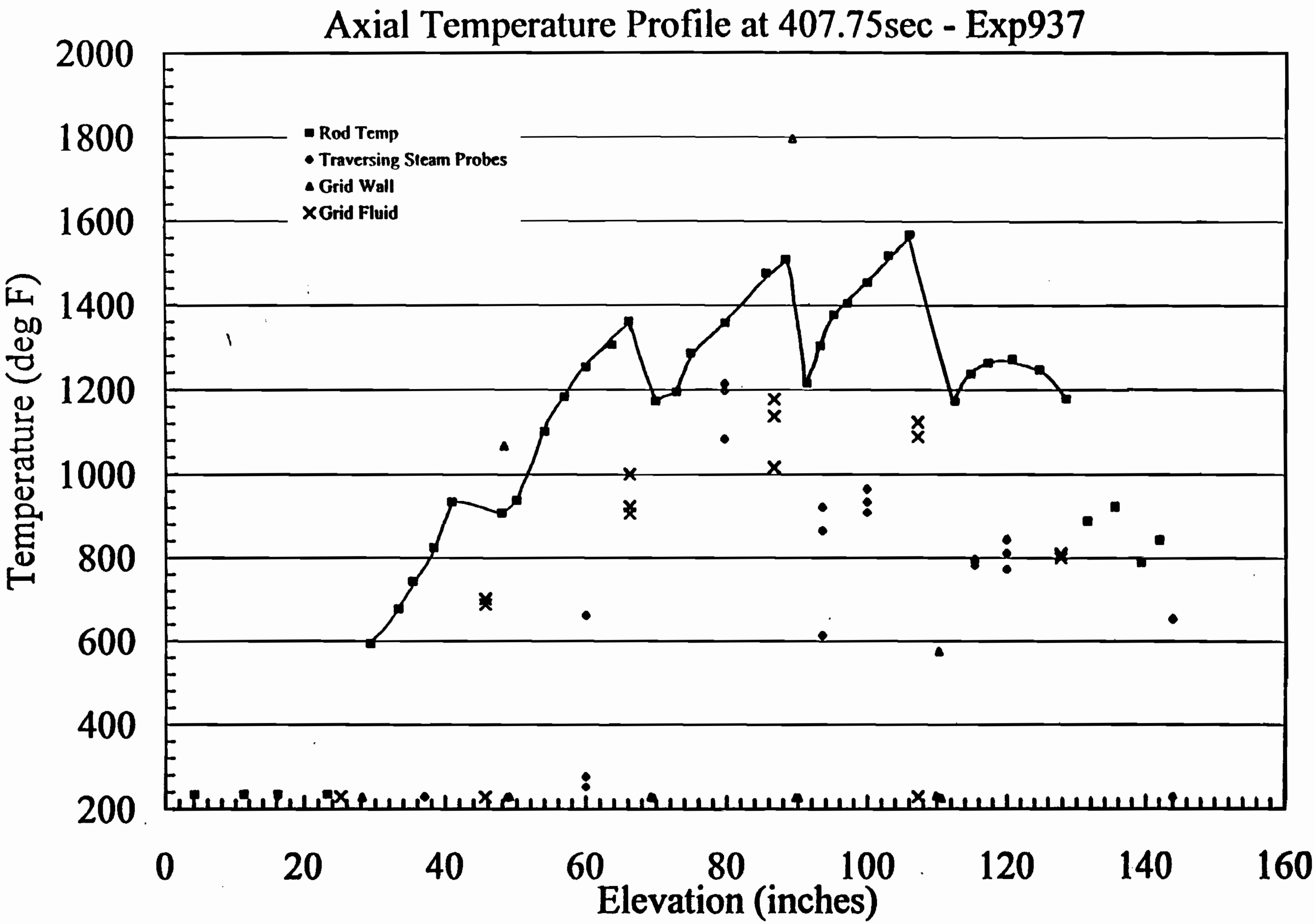
Stephen M. Bajorek
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

Exp 1108 - Steam Probe @ 120 inch - A



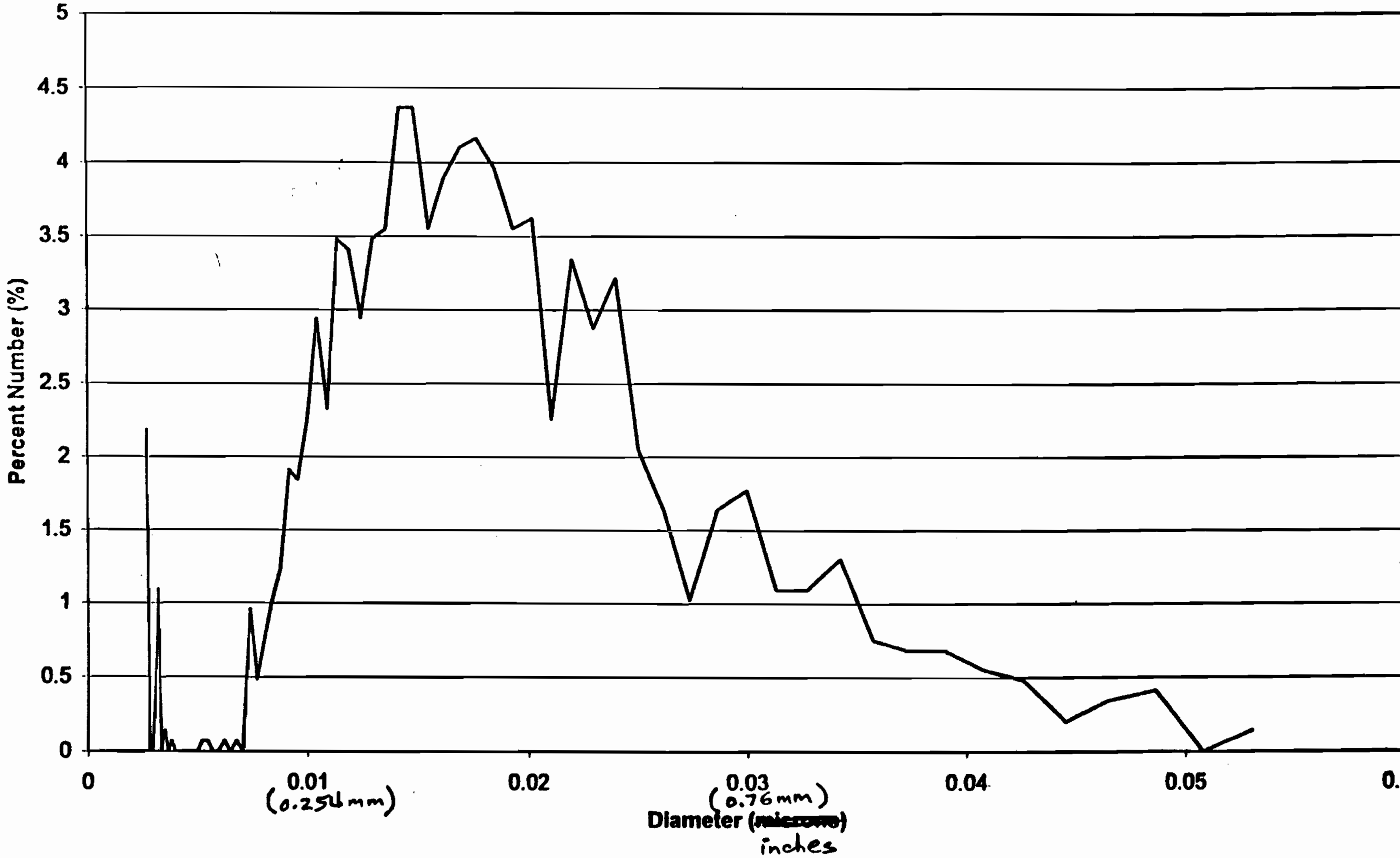


Exp 1108 - Grid Fluid Thermocouple at 107.27 inch
Channel 271
Temperature (deg F)
1600
1400
1200
1000
800
600
400
200
0
0
200
400
600
800
1000
Time (sec)



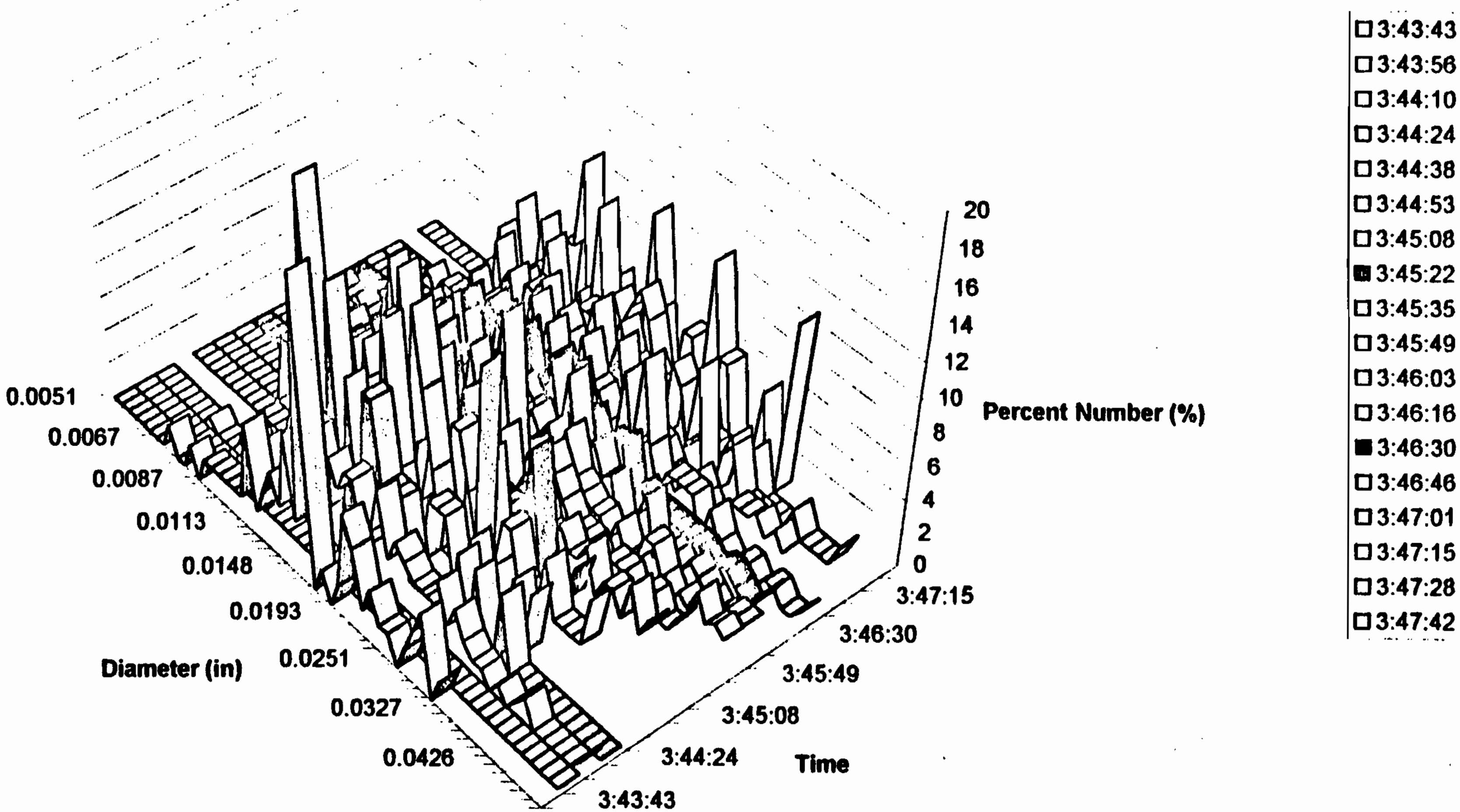
Axial Temperature Profile at 407.75sec - Exp937
Rod Temp
Traversing Steam Probes
Grid Wall
Grid Fluid
Temperature (deg F)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
20
40
60
80
100
120
140
160
Elevation (inches)

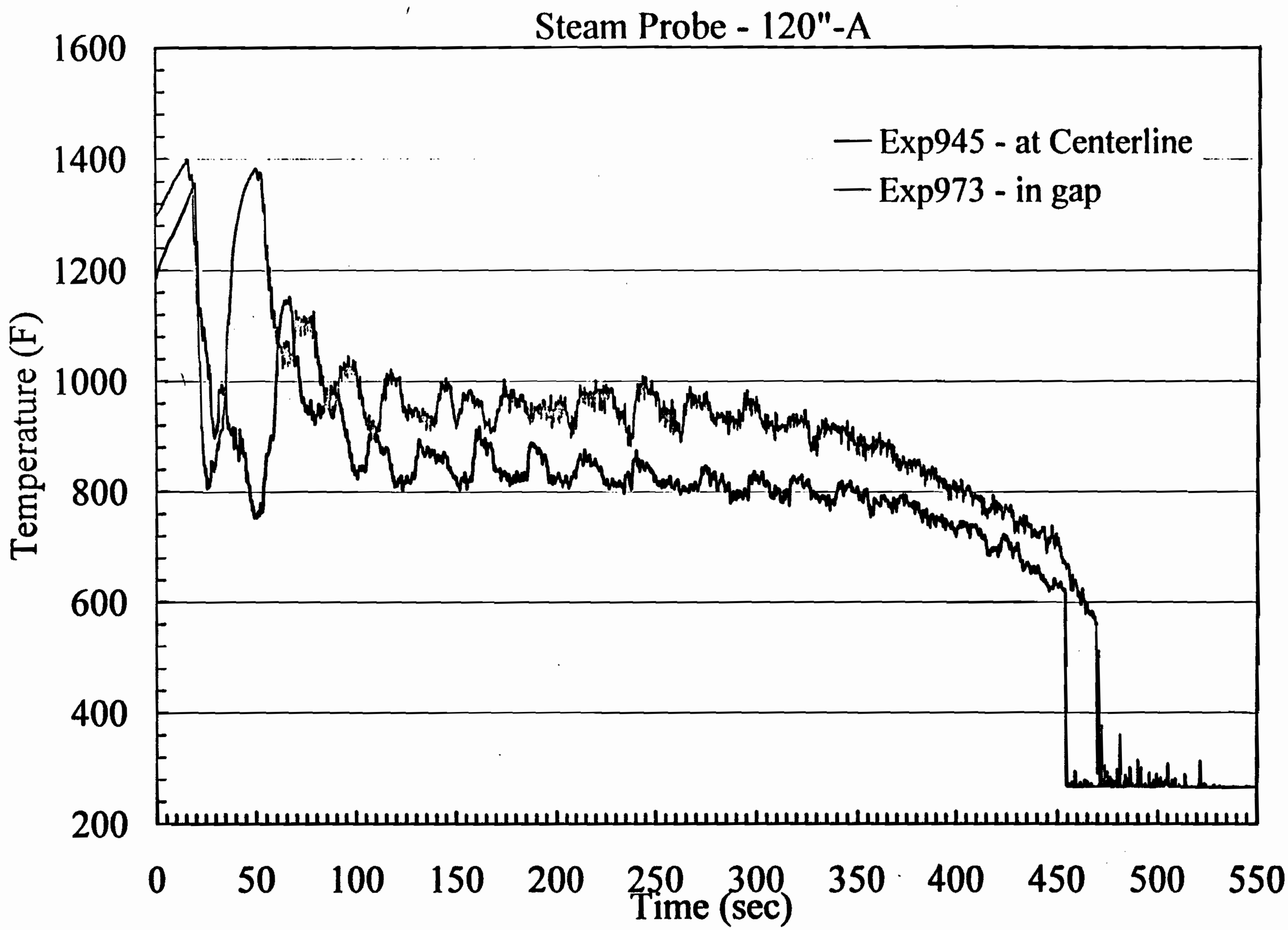
Run 954
Cumulative Droplet Distribution (Data taken from downstream of grid at the highest power)



Droplet Distributions - 5/15/02

(Data taken downstream of grid at the highest power)




Steam Probe - 120"-A
Exp945 - at Centerline
Exp973 - in gap
1600
1400
1200
1000
800
600
400
200
Temperature (F)
0
50
100
150
200
250
300
350
400
450
500
550
Time (sec)

- Testing began in May 2002. Initial tests and preliminary data have been reviewed & test matrix revised accordingly.

- Initial observations:

  - Spacer grids are a first order effect
  - Relatively uniform planar rod temperature profile
  - Traversing steam probes show lateral profile, subchannel effects
  - Reasonable drop size distribution / rapid processing
  - Acceptable mass balance & facility heat loss

- Reflood testing to continue through September 2002

GSI - 185 Recriticality

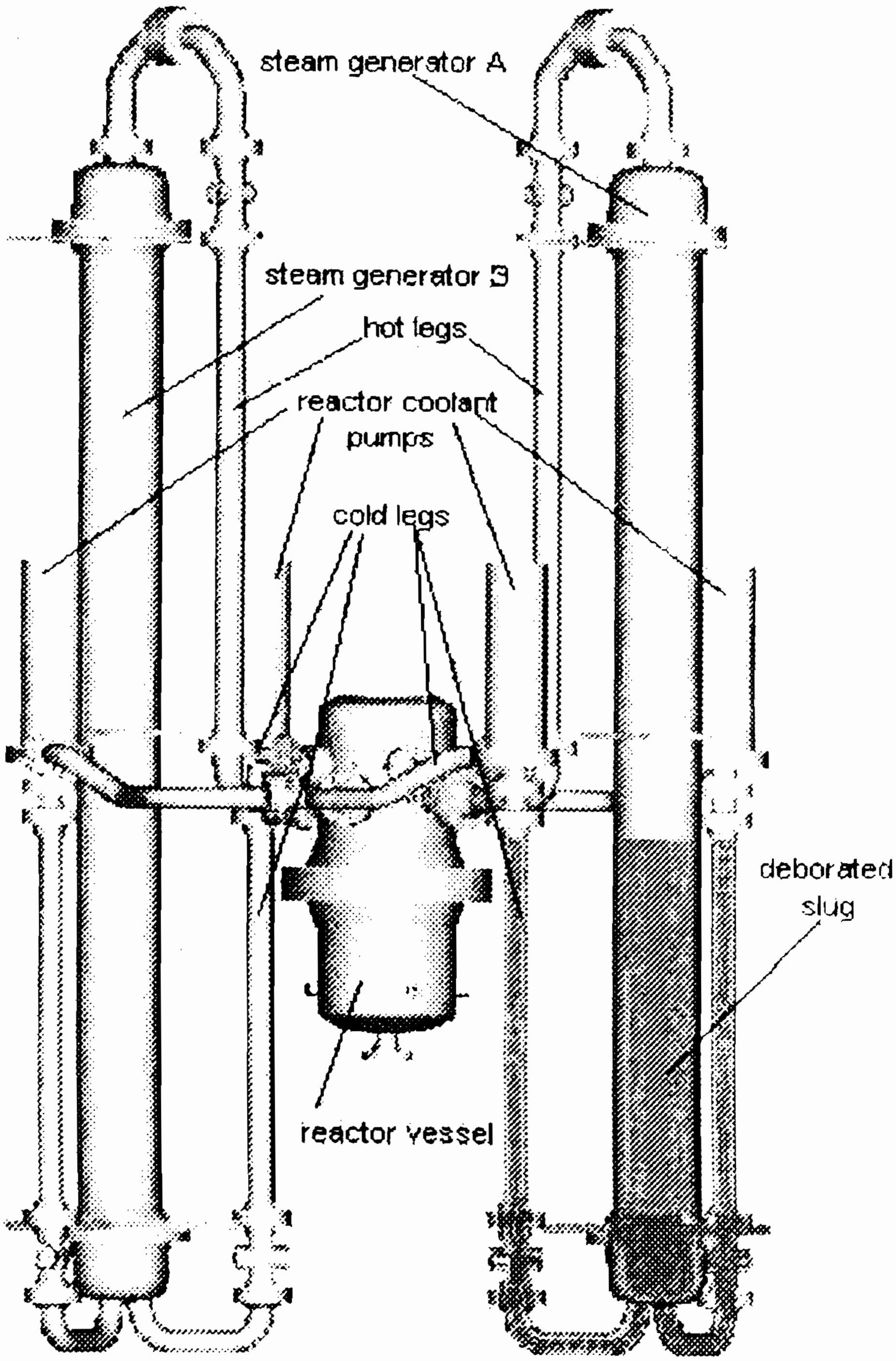
Via Boron Dilution during SBLOCA in PWRs

Harold Scott   Marino diMarzo   David Diamond

June 26, 2002

- Overview of GSI-185 Harold Scott
- Thermal Hydraulic Analyses Marino diMarzo
- Neutronics Analyses David Diamond

Generic Safety Issue (GSI) 185 addresses those SBLOCA scenarios in PWRs that involve steam generation in the core and condensation in the steam generators, causing deborated water to accumulate in part of the RCS. Restart of RCS circulation may cause a recriticality event (reactivity excursion) by moving this deborated water into the core.


steam generator A
steam generator B
hot legs
reactor coolant
pumps
cold legs
deborated
slug
reactor vessel

BNL FINDING TO DATE

The boron dilution transient caused fuel to:

- go above prompt-critical
- reach peak reactivity as high as $1.02
- reach 80% of nominal power
- undergo peak fuel enthalpy change of 37 cal/g.

Thus do not expect fuel damage for this case.

OTHER SCENARIOS

GSI-22 Inadvertent boron dilution

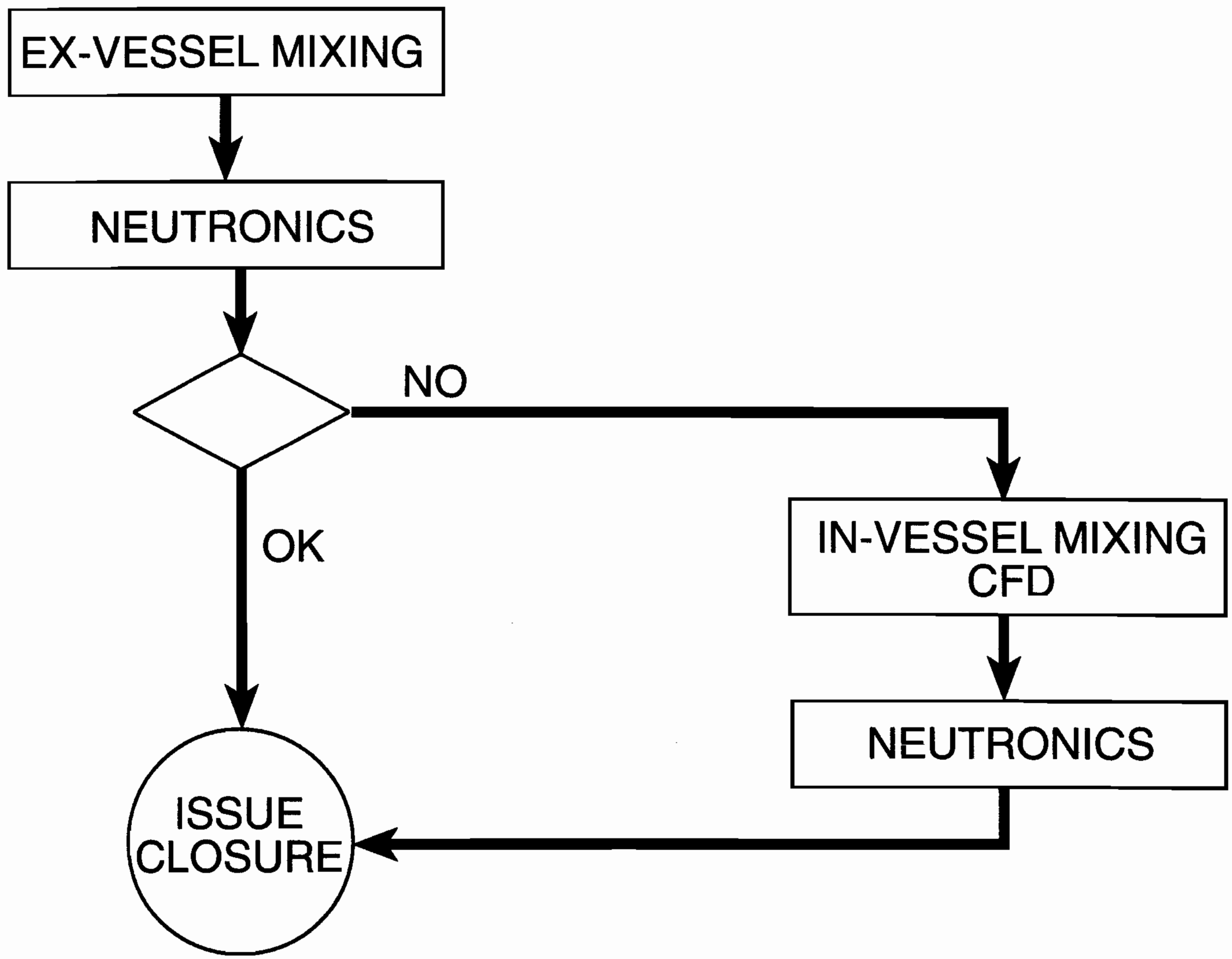
Reactor startup with LOOP ["French" Scenario]

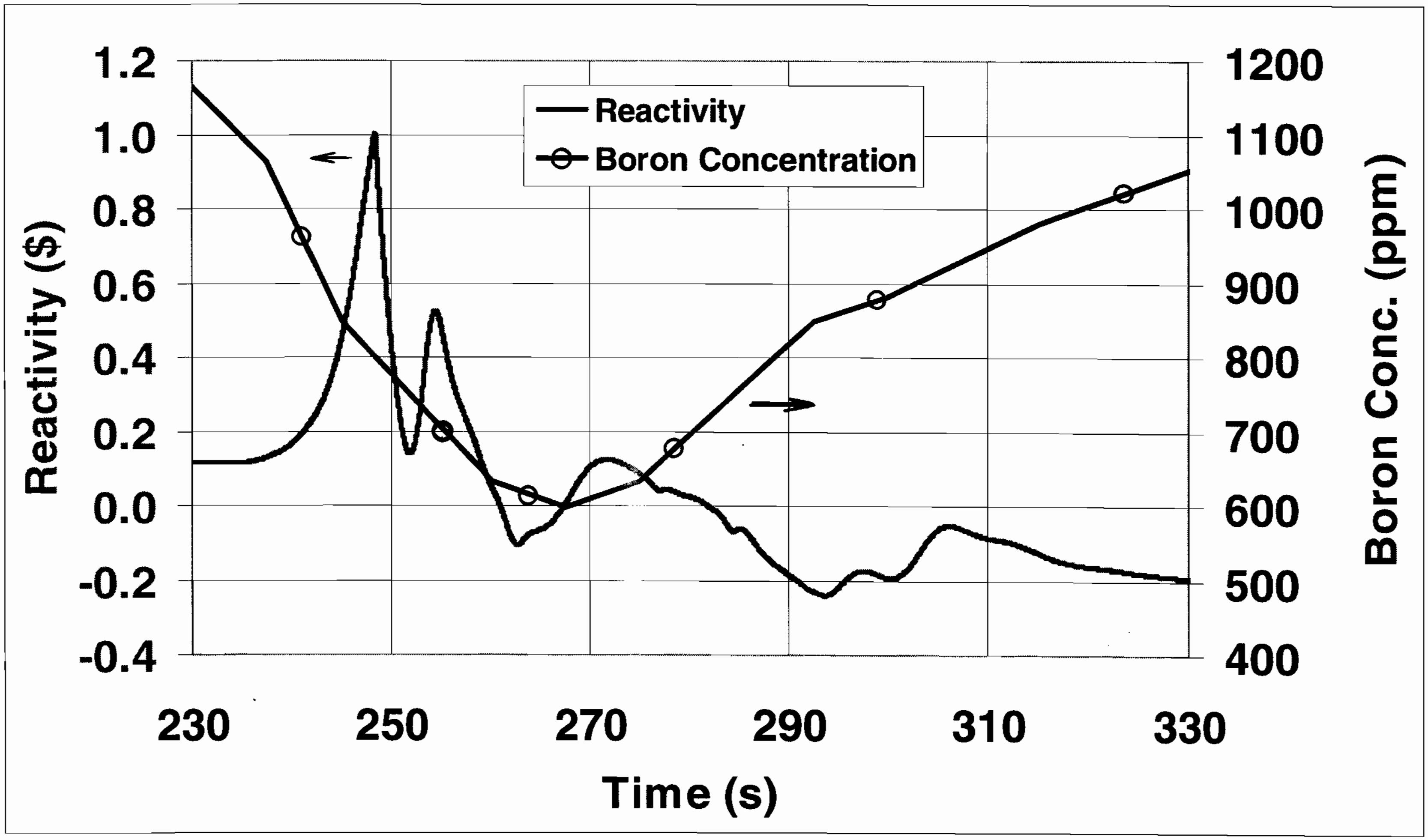
Leakage of secondary water via SG tube leak followed by RCP start ["Swedish" scenario]

GSI-185 SBLOCAs with reflux or boiling - condensation resulting in boron dilution

During shutdown with loss of decay heat removal and consequent reflux or boiling -condensation resulting in boron dilution and subsequent pump restart

GSI - 185


EX-VESSEL MIXING
NEUTRONICS
NO
OK
IN-VESSEL MIXING
CFD
NEUTRONICS
ISSUE
CLOSURE



1.2
1.0
0.8
0.6
0.4
0.2
0.0
-0.2
-0.4
Reactivity ($)
Reactivity
Boron Concentration
1200
1100
1000
900
800
700
600
500
400
Boron Conc. (ppm)
230
250
270
290
310
330
Time (s)

CLOSING THE GENERIC SAFETY ISSUE

An additional calculation assuming RCP bump

Preparation of RES report

Review of RES report and BNL analyses by NRR

Review by full ACRS

Closeout Memo to EDO

PUBLIC AVAILABILITY OF DOCUMENTS

- B&W Owner Group Report ML003686545
  January 2000
- Prioritization Report ML003730563
  July 2000
- Brookhaven Report ML021700591
  February 2002

R

EX-VESSEL MIXING

Consider two extreme idealized conditions [1]:

- plug flow
- backmixed flow

The ex-vessel mixing is evaluated with the following assumptions [2]:

1. In the primary system we identify two backmixed volumes
   - steam generator outlet plenum
   - reactor coolant pump

2. The cold legs are considered plug flow volumes

This model is validated against UM data

[1] Levenspiel, O., 1962. Chemical Reaction Engineering, Wiley
[2] diMarzo, M., 2001. Ex-vessel transport and mixing of a deborated slug in a PWR primary geometry, NED 210, pp.169-175

MODEL

Slug transit time:

$$\tau = V_{SLUG} / \dot{V}$$

Non-dimensional variables:

$$\theta = t / \tau \qquad N = V_{SLUG} / V_{COMPONENT}$$

Backmixed volume
transfer function:

$$C(\theta) = N \int_0^{\theta} \left[C(\lambda) - C_0 \right] e^{N(\lambda - \theta)} d\lambda + C_0$$

VALIDATION

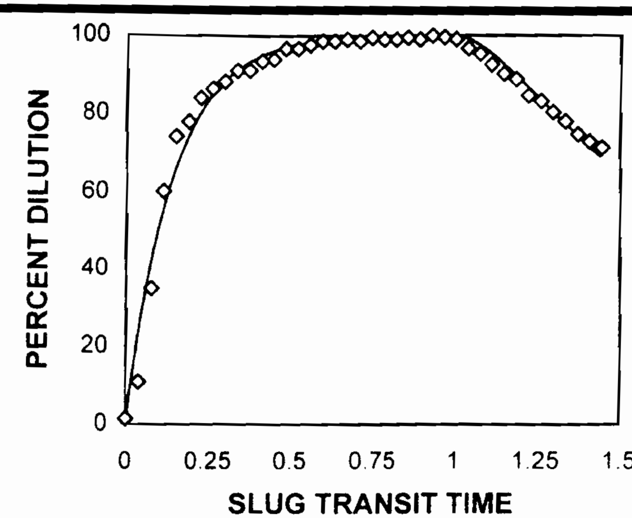


Fig. 5. Comparison of the calculated and measured concentrations at the entrance into the reactor vessel as a function of the transit time [2]

R

FUEL ENTHALPY DURING A RAPID BORON DILUTION EVENT IN A PWR

Presented to

The NRC Advisory Committee on Reactor Safeguards
Subcommittee on Thermal-Hydraulic Phenomena

Work performed by

D.J. Diamond, B.P. Bromley, and A.L. Aronson
Brookhaven National Laboratory
Energy Sciences and Technology Department

June 26, 2002

INTRODUCTION

- Boron Dilution Accident (BDA) an NRC Generic Safety Issue (185)
- Previous "conservative" study of BDA by Framatome
  - Estimate of boron concentration as a function of time
  - Lumped thermal-hydraulic/point-kinetics model
  - Lacked significant spatial effects, including boron transport
- What can a 3-D coupled neutronic/thermal-hydraulic analysis tell us?
  - Radial and axial distribution of boron changes significantly
  - Checkerboard pattern of control rods inserted
  - Radial and axial power distribution also complicated
- What are the fuel enthalpy increases?
- Calculations done as part of reactor core analysis project at BNL

BORON DILUTION EVENT

- Small Break Loss of Coolant Accident (SBLOCA)
  - Boiling in core
  - Natural circulation ends
  - Deborated steam condenses in steam generator
- Cooling system refills and natural circulation starts
- Slug of diluted water is pushed into core
- Even with all control rods in, reactor can go critical
- Positive Reactivity Insertion ⇒ Power Pulse ⇒ Energy Deposition

PWR CORE MODEL FOR BDA ANALYSIS

- TMI-1 Core Model at Beginning-of-Cycle
  - Babcock & Wilcox design, 177 15x15 FAs, 2772 MW_{th}
- Starting point for calculations
  - Hot zero power (2772 W, 1.0E-6 of full power)
  - Fuel, Moderator at 551 K; 1700 ppm boron
  - Banks 5-7 inserted (checkerboard pattern)
  - Equilibrium Xe from hot full power
- Starting point for BDA
  - All banks inserted (control, and shutdown)
  - Fuel, Moderator at 500 K, 1165 ppm boron
  - Reactivity ~ 0.0

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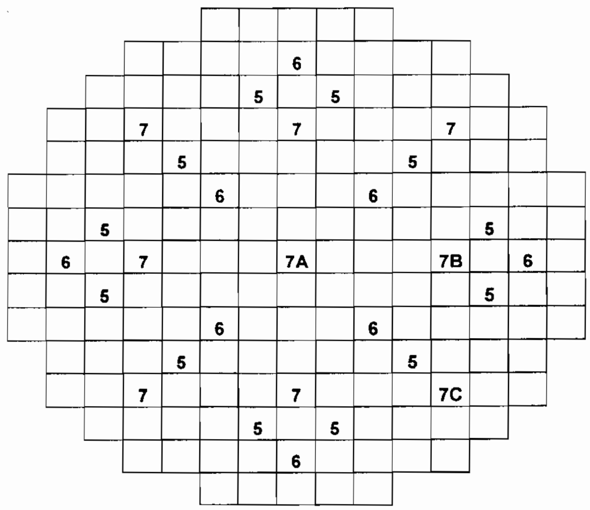
TMI-1 CORE LAYOUT WITH CONTROL BANKS



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FUEL ASSEMBLY AVERAGE BURNUP AT BOC

1 30.69	*2* 0.16	*3* 29.50	*4* 0.18	*5* 24.53	*6* 0.16	*7* 36.51	*8* 48.20
	9 32.26	*10* 0.17	*11* 29.30	*12* 0.17	*13* 29.25	*14* 0.15	*15* 40.34
		16 31.69	*17* 0.18	*18* 30.12	*19* 0.17	*20* 0.14	*21* 39.62
			22 24.52	*23* 0.18	*24* 31.73	*25* 26.73	
				26 24.89	*27* 0.17	*28* 32.22	
					29 24.82		

Control& SCRAM Banks

High Fuel Enthalpy

TH Channel

Burnup (GWD/T)

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3-D NEUTRONICS ANALYSIS

- PARCS (Purdue Advanced Reactor Core Simulator)
- Three-dimensional neutron kinetics via nodal method
- Two neutron energy group diffusion theory
- Feedback from fuel , moderator, boron ppm, control rod movement
- CASMO-3 ⇒ Homogenized FA cross section data for TMI-1 at BOC
- Cross Section Data not reliable below 500 K (limitation)

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EVENT ANALYSIS

- RELAP5
  - Octant symmetry (177 + 64) ⇒ (29 + 1) parallel channels
  - Lower inlet and upper exit plena connect channels
  - Inlet flow and temperature fixed; exit pressure fixed
- Boron Dilution Transient
  - Adapted from previous analysis for 6.5 cm^2 SBLOCA
  - $3.44 maximum boron reactivity insertion over 40 s
  - 3% mass flow rate at lower inlet plenum
  - 200 s simulation to bring TMI-1 core to conditions before BDA at ~1 hr into SBLOCA

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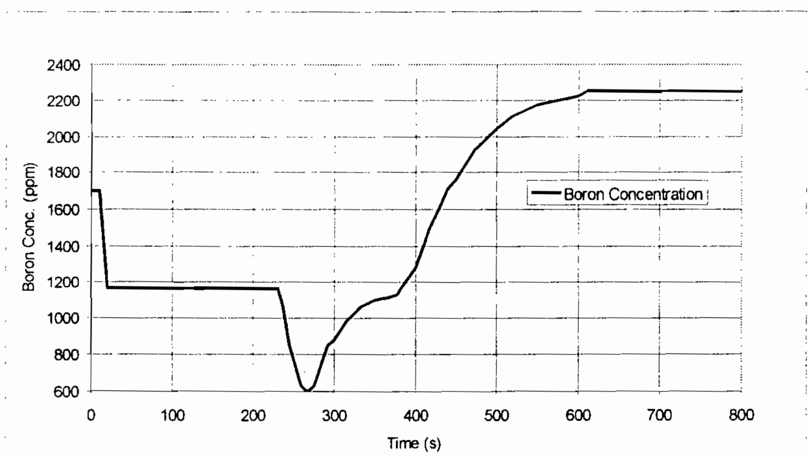
LOWER INLET PLENUM BORON DILUTION CURVE
(Adapted from Framatome Analysis of 6.5-cm² SBLOCA)



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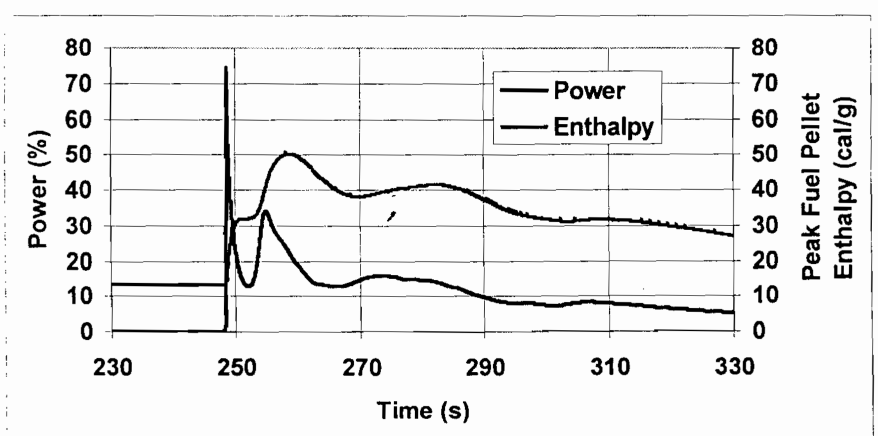
POWER AND PEAK FUEL PELLET ENTHALPY



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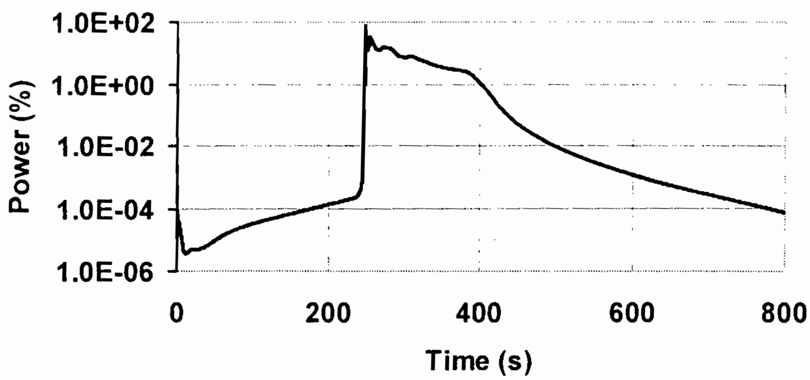
POWER VS TIME



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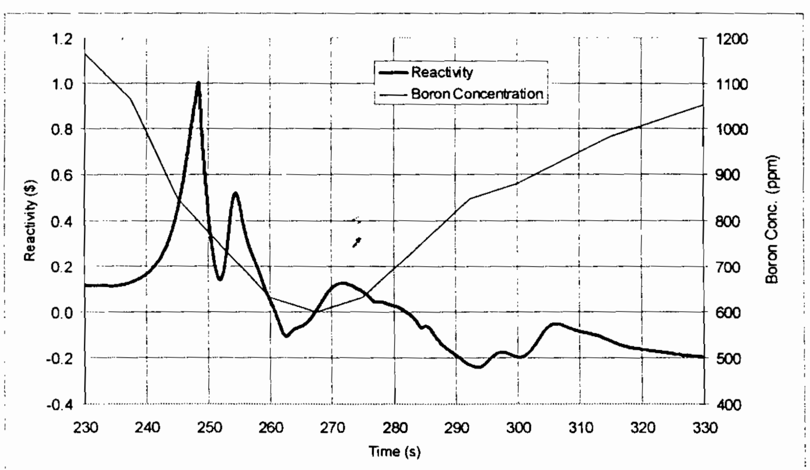
REACTIVITY AND BORON CONCENTRATION



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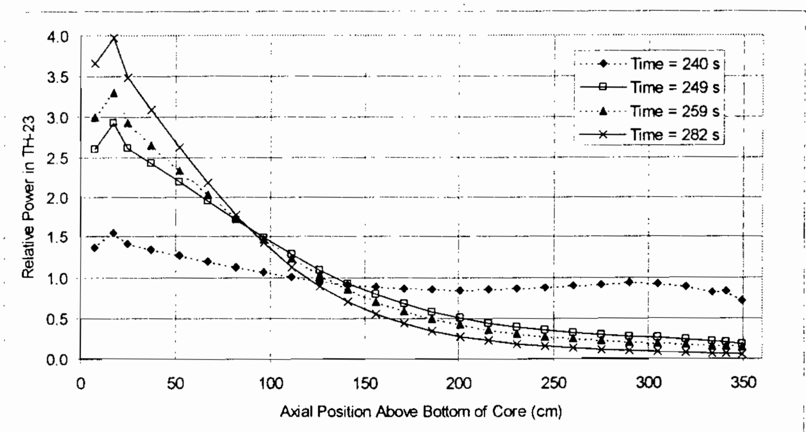
POWER PEAKING NEAR BOTTOM OF CORE



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RADIAL POWER DISTRIBUTION AT ~ 260 s

0.510	0.896	0.670	1.197	0.807	1.031	0.436	0.201
	0.616	1.122	0.881	1.415	0.773	0.822	0.246
		0.879	**1.607**	1.555	1.329	0.494	0.183
			1.210	**1.925**	0.986	0.645	
				1.460	**2.119**	0.663	
					1.191		

High Power FA's

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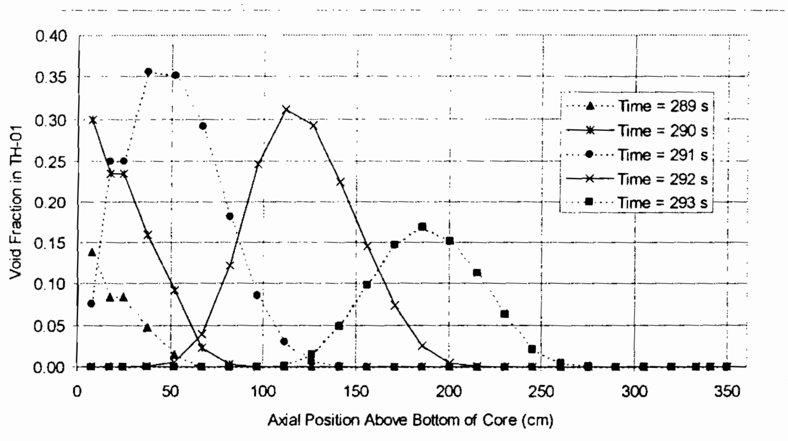
SPORADIC VOID FORMATION



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AVERAGE BORON CONCENTRATION AT ~ 260 s

1147	967	1049	894	1024	931	1150	1151
	1107	907	985	868	1037	1006	1151
		992	**859**	861	874	1148	1151
High Power FA's			893	**852**	933	1107	
				862	**851**	1086	
					885		

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COMPARISON:
3-D VS LUMPED POINT-KINETICS

- LUMPED POINT KINETICS
  - Peak reactivity ~ $1.2 (β~0.0065)
  - Peak power ~ 83%, 6 seconds after dilution starts
  - Peak enthalpy increase ~ 69 cal/g
  - Sporadic voids every 5 s, peak void 26%
  - Core returns sub-critical 45 s after prompt

- 3-D PARCS/RELAP5 - TMI-1
  - Peak reactivity ~ $1.002 (β=0.006323)
  - Peak power ~ 74%, 13 seconds after dilution starts
  - Peak enthalpy increase ~ 37 cal/g
  - Sporadic voids every 5 s, peak void 41%
  - Core returns sub-critical 24 s after prompt

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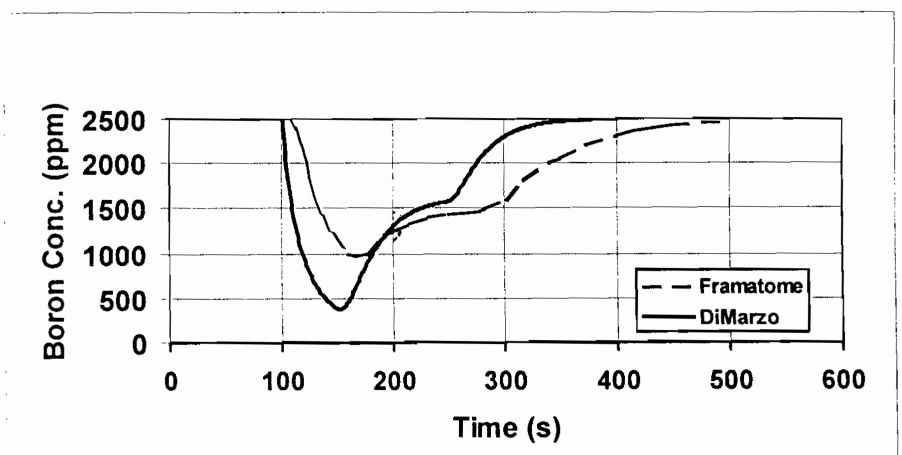
INLET PLENUM BORON CONCENTRATION



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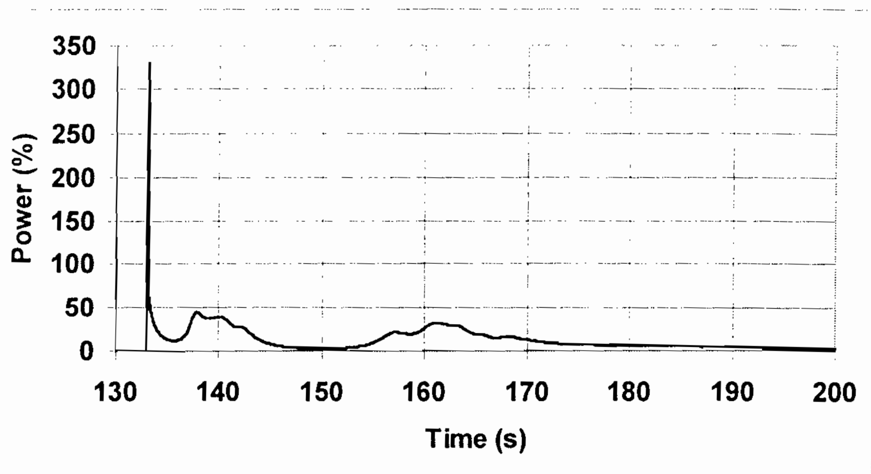
POWER VS TIME (DIMARZO CURVE)



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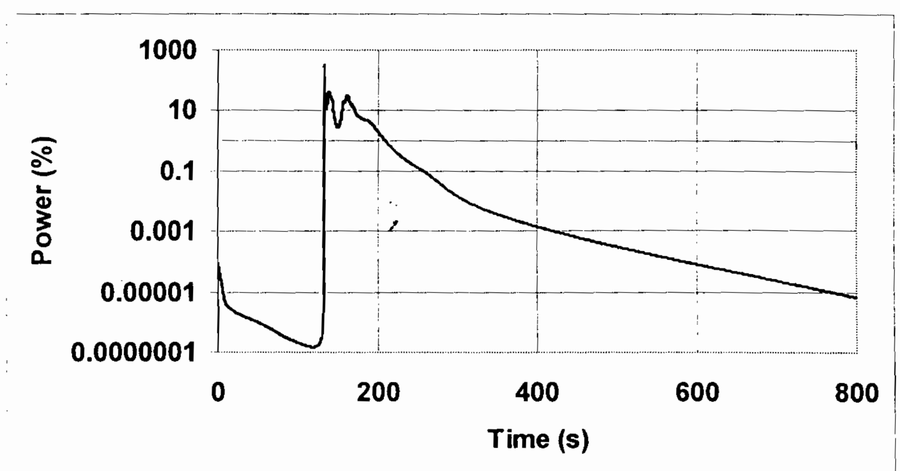
POWER (WITH DIMARZO CURVE)



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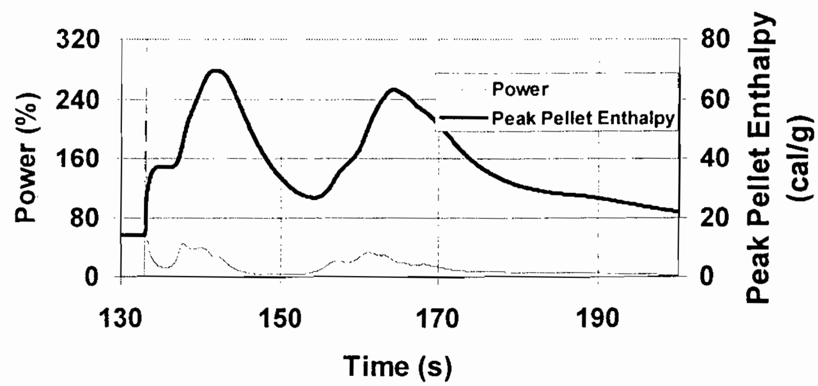
POWER AND ENTHALPY (WITH DIMARZO CURVE)



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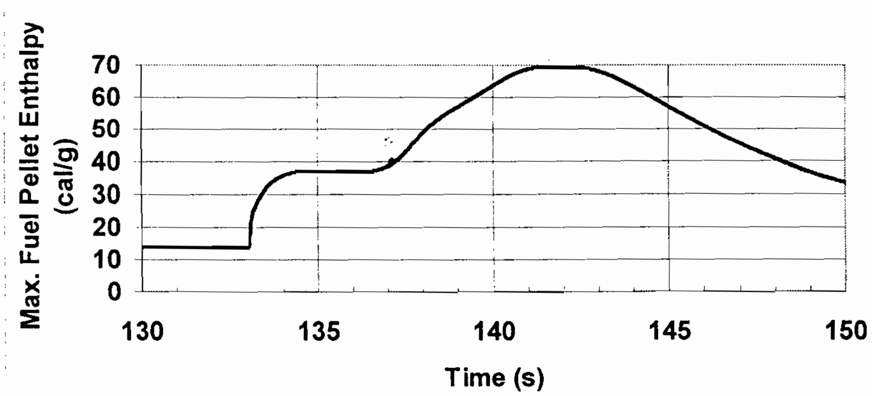
FUEL ENTHALPY DURING FIRST 20 S



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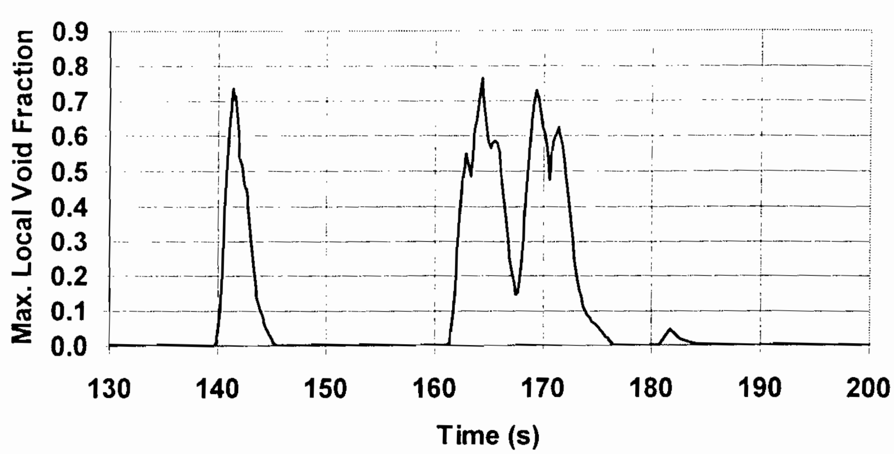
MAXIMUM LOCAL VOID FRACTION (DIMARZO CURVE)



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CONCLUSIONS AND RECOMMENDATIONS

- 3-D analysis gives lower energy deposition relative to point kinetics
- Evolution of energy deposition slower than an REA
- Thermal-hydraulic feedback limits fuel enthalpy during BDA
  - Initial enthalpy increase < 25 cal/g for cases considered
- Void formation sporadic, but DNB may be possible in more severe cases
- Preliminary comparisons with BARS/RELAP5 good (not shown)
- Results could use refinement/extension
  - Mixing in core
  - Radial/azimuthal non-uniform boron concentration at inlet
  - Effect of turning on pump

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