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

January 12, 2001

MEMORANDUM TO: ACRS Members

FROM: Paul Boehnert, Senior Staff Engineer
ACRS

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON SEVERE ACCIDENT MANAGEMENT, NOVEMBER 15, 2000 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued November 30, 2000, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: via E-mail
J. Larkins
R. Savio
H. Larson
S. Duraiswamy
ACRS Staff Engineers
ACRS Fellows

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

MEMORANDUM TO: Paul Boehnert, Senior Staff Engineer
ACRS/ACNW

FROM: Thomas S. Kress, Chairman
Severe Accident Management Subcommittee

SUBJECT: CERTIFICATION OF THE SUMMARY/MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEE ON SEVERE ACCIDENT MANAGEMENT, NOVEMBER 15, 2000 - ROCKVILLE, MD

I hereby certify that, to the best of my knowledge and belief, the Minutes of the subject meeting issued November 30, 2000, are an accurate record of the proceedings for that meeting.

T. S. Kress Jan. 16, 2001
Thomas S. Kress, Chairman Date

GT 330

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

November 30, 2000

MEMORANDUM FOR: T. Kress, Chairman, Severe Accident Management Subcommittee

FROM: P. Boehnert, Senior Staff Engineer

SUBJECT: MINUTES OF THE ACRS SEVERE ACCIDENT MANAGEMENT SUBCOMMITTEE MEETING, NOVEMBER 15, 2000 - 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 for their information.

Attachment: As Stated

cc: ACRS Members
R. Savio

cc via E-Mail:
J. Larkins
J. Lyons
R. Savio
S. Duraiswamy
ACRS Staff Engineers
ACRS Fellows

DRAFT COPY - PREPARED FOR INTERNAL COMMITTEE USE

Certified by Thomas S. Kress
January 16, 2001

CERTIFIED

November 30, 2000

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SEVERE ACCIDENT MANAGEMENT SUBCOMMITTEE MEETING
NRC/NEI REPORT, NEI 99-03, ON CONTROL ROOM HABITABILITY/
NRC - RES SEVERE ACCIDENT MANAGEMENT RESEARCH PROGRAM
NOVEMBER 15, 2000
ROCKVILLE, MARYLAND

INTRODUCTION:

The ACRS Subcommittee on Severe Accident Management held a meeting on November 15, 2000 with representatives of the Nuclear Energy Institute and the NRC staff. The purpose of this meeting was to review the activities of the NRC staff and nuclear industry under the auspices of the Nuclear Energy Institute (NEI) pursuant to revision of the NEI guideline document, "Control Room Habitability Assessment Guidance", NEI 99-03, and review the NRC Office of Nuclear Regulatory Research (RES) severe accident management research program, pursuant to development of the ACRS annual report to the Commission on NRC safety research. 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 Subcommittee Chairman at 8:30 am, November 15, 1999, and adjourned at 5:17 p.m. that day.

ATTENDEES

ACRS Members/Staff:

T. Kress, Chairman
R. Seale, Member
P. Boehnert, DFO
J. Sieber (part time)
G. Wallis (part time)

NRC Staff:

F. Eltawila, RES
C. Tinkler, RES

NEI Representatives

K. Cozens, NEI
K. Taplett ()
R. Cambell ()
S. Schultz ()
J. Sims ()

There were approximately 5 - 10 other members of the public in attendance during this meeting. A listing of those attendees who registered is available in the ACRS office files. No members of the public participated in the meeting discussions.

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

CHAIRMAN'S COMMENTS

Dr. T. Kress, Subcommittee Chairman, convened the meeting. He noted that the first topic for discussion was the issue of control room habitability. He had no specific comments relative to the presentations on this topic.

NRC/INDUSTRY ACTIONS RELATIVE TO RESOLUTION OF ISSUES PERTAINING TO CONTROL ROOM HABITABILITY

NEI Presentations

Mr. K. Cozens, NEI, provided an overview of the revised draft NEI document, NEI 99-03, "Control Room Habitability Assessment Guidance". The status, key elements, and structure of NEI 99-03 were reviewed. A flow diagram of the implementation process was discussed. NEI believes that their approach achieves reasonable assurance of CR operator protection.

In response to questions from Dr. Kress, Mr. Cozens said that the industry is using deterministic methods to resolve this issue. Regarding the plants that failed initial tracer gas testing, NEI said that these plants made physical modifications to comply with regulatory inleakage requirements. In response to Mr. Sieber, Mr. Cozens said that licensees are exploring the option of using the revised source term.

Presentations were provided on the following issues as noted below:

- Baseline Testing - NEI proposes to require baseline testing using either integrated tracer gas testing, or a component test method. While tracer gas testing is effective in all CRs, component testing can only be used in pressurized CRs (~20 CRs are eligible for component testing). NEI favors use of component testing for these CRs, because of the potential problems that can be encountered with use of tracer tests.

  In response to questions, Mr, Taplett noted that no one has yet conducted component tests for determination of baseline inleakage. Dr. Wallis opined that the industry needs to develop and prove the component test method. Dr. Kress said that an assessment of the uncertainties associated with each test method would be desirable.

- Periodic Inleakage Assessment - NEI proposes periodic inleakage assessment to ensure CR integrity is maintained. Assessment frequency would be plant specific and a fixed retesting schedule is not recommended.

  The subcommittee posed questions regarding the performance measures to be applied for determination of when retesting will be required. Dr. Kress said that a rational decision making process for retest is needed, including development of objective criteria.

- CR Smoke Infiltration - NEI recommends a qualitative approach to assessment of the vulnerability of a plant's CR to smoke infiltration. It was argued that given the lack of relevant design criteria, a quantitative resolution approach was not recommended. Dr. Wallis argued that the adequacy of the preferred approach needs to be demonstrated.

- Toxic Gas - NEI proposes licensees review their existing toxic gas evaluation, given potential new sources and possible increased inleakage rates. Periodic re-assessment will be performed on a 5-year (plants near industrial areas) and 10-year (rural-site plants) basis.

- Analysis - The industry has proposed specific analysis improvements in light of the CR habitability resolution effort. Improvements include: use of the Alternative Source Term five rem Total Effective Dose Equivalent, assumption of (TID-14844) 50 Rem thyroid limit, elimination of the ECCS passive failure assumption, use of an iodine spiking factor of 335 for the steam generator tube rupture analysis, and modifications to assumptions for the fuel handling, meteorology, and dispersion modeling. NEI has also sought staff approval for use of enhanced containment mixing rates and assumption of an iodine spike duration of > 8 hours.

  Dr. Kress requested information relative to the bases for the choice of 335 for the iodine spiking factor.

- Technical Specifications - NEI argued that technical specifications for control of CR inleakage are not necessary. Since inleakage is a design parameter, it does not meet the criteria for a technical specification parameter; inleakage should, however, be controlled under an Appendix B-type program.

Mr. Cozens noted that a NEI survey shows that the guidance specified in the 99-03 document is supported by ~ 75% of licensees. NEI has requested staff comment on the

current 99-03 draft, in spite of the staff's initial refusal to do so, given NRC's intent to peruse a regulatory approach to resolution of this matter.

NRC Staff Presentation

Mr. J. Hayes discussed the staff's position on the CR habitability issue. He reviewed the background, history, and accomplishments of the NRC/NEI interactions. Regarding the remaining areas of concern, NRR noted the following:

- Baseline testing - NEI proposes to use component tests to determine baseline inleakage. NRC believes that an integrated test is appropriate for baseline leakage determination. Subsequently, component testing, with periodic integrated tests, would be acceptable.

- Radiological Analyses - The staff is looking for the industry to "do the right thing" here by providing realistic analysis; i.e., remove non- and over-conservatisms. In other words: "don't hide behind the licensing basis, and we (NRC) won't hide behind the SRP".

- Smoke and Toxic Gas - Mr. Hayes gave indication that the staff is basically satisfied with NEI's approach for addressing these matters, though NRC will need to explore the details further.

- Technical Specification Requirements - NRC believes that technical specification requirements are needed to ensure licensee compliance with inleakage limits.

R. Barrett, NRR, noted that the staff has elected to pursue a regulatory approach to resolution of this issue for two main reasons: (1) to engage the industry and stakeholders in a comprehensive discussion of this matter, and, (2) to avoid dictating a specific resolution approach(s). He also noted that "99-03" is a technical document, not a regulatory document, and its use by the industry will require some form of NRC endorsement. Seeking public comment on the open issues is seen as the next logical step. At this point, the staff is considering a regulatory guide(s) as the resolution vehicle.

Subcommittee Caucus

The Subcommittee recommended that this matter be brought to the ACRS for deliberation. Dr. Kress noted that the Committee scheduled a discussion session from 1-3 p.m. on December 6, 2000. The Chairman instructed the staff and NEI regarding the information to be presented to the Committee.

NRC - OFFICE OF NUCLEAR REGULATORY RESEARCH (RES) SEVERE ACCIDENT RESEARCH PROGRAM

Introductory Remarks

Dr. Eltawila noted that during last year's RES presentation on the severe accident (SA) research program to the subcommittee, he had stated that this program had decreased to a point he judged to be below minimum required core capabilities. He indicated that the situation has worsened in that the Agency is making more use of SA research information than ever before, absent any increase in resources. He also said that the Agency needs to take action, or RES will not be able to meet the NRC's needs in support of risk-informed regulation. Dr. Eltawila solicited that Committee's help in this regard.

Status of SA Research

Mr. C. Tinkler discussed the following issues regarding the RES SA research Program:

- SA Research Applications
  - Revised Source Term Implementation
  - Steam Generator (SG) Tube Integrity
  - Spent Fuel Pool Accident Analysis
  - Dry Cask Storage
  - Risk-Informed 10 CFR 50.44
  - Improved Containment T/H Analysis
- Improvement & Consolidation of Severe Accident Codes
  - OECD Lower Head Failure Program
  - PHEBUS
  - RASPLAV/MASCA (OECD)
  - MACE
- Cooperative Experimental Research
- MOX & High Burnup Fuel
- MAAP Code Review

Key points noted during the above discussion included:

- RES cannot obtain the needed technical information relative to the MOX & High Burnup fuel unless and until the industry files a license application/amendment, which has yet to occur.

- The SGTR tube integrity program objective is to determine the SA T/H boundary conditions to assess the likelihood of thermally induced tube rupture. To this end, a February 8, 2000 NRR User Need Letter asks RES to develop a confirmatory research program to address SG tube integrity. Work is underway at INEEL and is scheduled to be completed in March 2002.

- Regarding the spent fuel pool work, Dr. Kress opined that the staff should consider the impact of land contamination, total deaths, and injuries.

- In response to Dr. Kress, RES said that the DCH issue is considered resolved. Some follow-on work may be proposed for BWR plants. The CONTAIN code will be subject to a minimal maintenance program.

- RES has initiated a code consolidation program for its suite of SA codes, similar to what is being done for the T/H codes. MELCORE has been selected as the platform for building a consolidated code, and this new code will be the regulatory counterpart to the EPRI MAAP code. Dr. Kress asked if the staff is investigating the effect of charges on aerosol behavior. RES said that a Swiss research program (ARTIST) is investigating use of charged particles in future tests.

- A follow-on to the RASPLAV experimental program has begun. The three-year program to investigate the influence of chemical behavior on heat transfer pools in the lower RPV head under severe accident conditions was organized under OECD/NEA sponsorship and will end in June 2003. Dr. Kress asked what benefit is to be gained from the program. RES said that data from this program could impact SAM strategy. It was also noted that OECD severe accident research experts have stated that this work will be of high quality, it will help maintain necessary expertise, and should provide additional information relative to the ability to cool debris in-vessel by use of ex-vessel cooling.

- Dr. Kress also asked for justification for the OECD lower head vessel failure experimental program being conducted a Sandia Laboratories. RES said that this program should provide data pertinent to the issues associated with fuel-coolant interactions, specifically how rapidly the melt enters the water.

- A follow-on program of MACE testing has been proposed. A set of separate-effects tests would be run to investigate details associated with cooling of a molten mass overspread with water.

- The PHEBUS tests have essentially confirmed the provisions of NRC's revised source term

- Regarding issues associated with use of MOX and high-burnup fuel, it was noted that NRC has not been able to obtain relevant data to resolve source term issues for these items. Dr. Seale said that the ACRS may need to take up the issue of securing the necessary data in a timely manner from the consortium that is to build a MOX fuel facility at Savanna River. RES said they would welcome the Committee's help here.

- NRC has begun an initiative to review the latest version (4.0) of the EPRI MAAP code. A "kick-off" meeting is scheduled with EPRI for December 15, 2000.

- RES noted planned and potential future activities. Those activities not mentioned above include: perform a PRA for evaluation of SA risk for dry cask fuel storage, rebaselining generic risk analysis (Level 2) to account for advances in accident management, and potential research for proposed advanced reactor designs such as the pebble bed reactor proposed to be constructed in South Africa.

During Subcommittee discussion prior to meeting adjournment, Dr. Kress asked if the issues of the impact of air egression and atmospheric dispersion for spent pool fires is being addressed. RES said that both are being evaluated via the MOX/high burnup and MAACS code modeling, respectively. For the issue of fuel-coolant interactions, RES said that issues associated with long-term plant stabilization need to be addressed. Dr. Seale said that RES's development of a PIRT for high burnup fuel studies, was an example of forward-looking work and should be highlighted to the Commission. Dr. Eltawila indicated that RES will factor the comments of the Subcommittee into the work on MOX and high burnup fuel.

Subcommittee Caucus

The Subcommittee Chairman indicated that the information obtained at this meeting on the SA research program will be factored into the Committee's report on NRC safety research.

FOLLOW-UP ACTIONS

No specific follow-up actions were identified during this meeting.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING

1. Memorandum, dated October 13, 2000, from P. Boehnert, ACRS, to T. Kress, et al. ACRS transmitting:

   - Memorandum, dated July 13, 2000, to ACRS from P. Boehnert, T. Kress, R. Seale, Subject: "Cooperative Severe Accident Research Program Meeting (CSARP 2000), May 8-11, 2000 - Residence Inn Hotel, Bethesda, Maryland"

   - Memorandum to the Commission, dated November 9, 1998, from W. Travers, EDO, Subject: Schedule for Closure of Severe Accident Issues and Severe Accident Research Activities

   - Excerpt from Minutes of August 9-10, 1999 meeting of the SAM Subcommittee - NRC Office of Nuclear Regulatory Research Severe Accident Research Program, dated September 2, 1999

   - Memorandum from P. Boehnert to T. Kress, et al., dated April 19, 2000, Subject: NRR User Need Request Related to Steam Generator Severe Accident Response and Testing of Steam Generator Tubes

   - Letter from D. Modeen, NEI, to R. Barrett, NRC, transmitting Draft NEI Report, NEI 99-03, Control Room Habitability Assessment Guidance, dated October 13, 2000

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, Washington, D.C. 20006, (202) 634-3274, or can be purchased from Ann Riley & Associates, LTD., 1025 Connecticut Ave., NW, Suite 1014 , Washington, D.C. 20036, (202) 842-0034.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SEVERE ACCIDENT MANAGEMENT SUBCOMMITTEE MEETING:
NRC RES SEVERE ACCIDENT MANAGEMENT RESEARCH PROGRAM/
NRC/INDUSTRY ACTIVITIES RELATED TO CONTROL ROOM HABITABILITY
NOVEMBER 15, 2000
ROCKVILLE, MARYLAND

Contact: P. Boehnert (301/415-8065 - "pab2@NRC.GOV")

TOPICAL AGENDA

TOPIC	SPEAKER	TIME
I. Introduction	T. Kress, Chairman	8:30 a.m.
II. NRC/Industry Actions Related To Control Room (CR) Habitability	K. Cosins, NEI, et al.	8:40 a.m.
A. NEI/Industry Presentations		
1. Introduction/Background • History of NRC/Industry Actions Since 11/99 SAM Subcommittee Meeting		
2. Industry Positions and Bases on CR Habitability Assessment (NEI 99-03)		
3. Results of Recent NRC/NEI/ Industry Meeting • Resolution of Open Issues		
4. Future Milestones and Schedule		
	BREAK	10:30 - 10:45 a.m.
B. NRC Staff Presentations	J. Hayes, et al.	10:45 a.m.
1. Introduction/Background		
2. Status of NEI/NRC Interaction on Control Room Habitability		

TOPIC	SPEAKER	TIME
3. Future Actions		
C. Subcommittee Discussion		12:15 - 12:30 p.m.
	LUNCH	12:30 - 1:30 p.m.
III. NRC-RES Severe Accident Research Program		
A. Opening Remarks	F. Eltawila/ J. Flack, RES	1:30 p.m.
B. Severe Accident (SA) Research Program	C. Tinkler/et al., RES	
1. SA Research Applications		
2. Improvements & Consolidation of SA Codes		
3. Cooperative Experimental Research		
4. MOX and High Burnup		
5. MAAP Code Review		
C. Concluding Remarks	F. Eltawila/ J. Flack	
IV. Subcommittee Caucus		5:45 p.m.
■ Follow-on Items from this Meeting		
■ Future Actions		
■ Committee Action		
V. Adjourn		6:00 p.m.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON SEVERE ACCIDENT MANAGEMENT

NOVEMBER 15, 2000
Today's date

[redacted] **SIGN IN FOR ACRS MEETING**

PLEASE PRINT

NAME	BADGE #	NRC ORGANIZATION
John J. Hayes	B6233	NRR/DSSA/SPSB
Rich Barrett	A7330	NRR/DSSA/SPSB
Jay Lee	B6664	NRR/DSSA/SPSB
Sud Basu	B7497	RES/DSARE/SMSAB
Chris Boyd	B8024	RES/DSARE
Steve Cahill	B8172	NRR/DSSA/SPSB
Jason Schaperow	B7363	RES
FAROUK ELTAWILA	A6364	RES/DSARE
STEVEN ARNDT	B-8396	RES/DSARE/SMSAB
Allen Notafan	D-6865	RES/DSARE/SMSAB
BOB PALLA	B-6891	NRR/DSSA/SPSB
Jocelyn Mitchell	B-6685	RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON SEVERE ACCIDENT MANAGEMENT

NOVEMBER 15, 2000
Today's Date

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
ROGER ZAVADOSKI	Defense Nuclear Facilities Safety Board.
[illegible]	[illegible]
KEN TAPLETT	STPNOC
John Wynn	CCNPP
JERRY BURFORD	EOI
KURT COZENS	NEI
Brad Harvey	DE+S
Stephen P. Schultz	Duke Energy
BERNIE JWASZKEWSKI	Vermont Yankee
Jerry G. Sims	Southern Company
Robert Campbell	TVA
Syed A. Ahmed	Dominion Generation.
JIM RILEY	NEI
STEVE LEONARD	NMPC.
Mike Ruby	RG&E
Deann Raleigh	SERCH Bechtel Power.
Herb Fontecchio	Dominion
Edwin Lyman	Nuclear Control Institute
George Vayssier	MSC Netherlands vayssier@hetnet.nl

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ACRS CRH Presentation
Industry Agenda

Topic	Presenter
Overview	Kurt Cozens
Baseline testing	Ken Taplett
Period Testing	Bob Campbell
Smoke	Bob Campbell
Toxic gas	Bob Campbell
Analysis	Steve Schultz
Tech Spec	Jerry Sims
Conclusions	Kurt Cozens

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Overview of Revised NEI 99-03

NEI

1

CRH Concerns

- Greater unfiltered in-leakage
- Limiting accident analysis
- Smoke ingress
- Licensing and design basis maintenance

NEI

2

Status of NEI 99-03

- Purpose
  - Provide an approach assuring adequate protection of CR operators (Radiological and Toxic gas)
  - Obtain generic NRC agreement with that approach
- October 2000 draft
  - Distributed to industry and NRC on 10/13/00
  - Extensive revision from original text
  - Comprehensive treatment of design, operation and maintenance facets

NEI

3

Key Elements

- CR in-leakage (baseline test and periodic assessment)
- Toxic gas (reassessment and period evaluation)
- Smoke infiltration -- qualitative assessment
- Uses existing licensing basis
- Limiting design basis accident assessment
- CR as-built configuration and operating procedures assessment
- Considers current radiological dose analysis methods
- Program to maintain CRH

Achieves reasonable assurance of CR operator protection

NEI

4

Document Structure

- Part 1, Background
  - Section 1, Introduction
  - Section 2, Regulatory Requirements and Guidance
  - Section 3, Industry Issues Associated with Control Room Habitability
- Part 2, Assessment Process
  - Section 4, Determining CRH Licensing Basis
  - Section 5, Comparing Existing Plant Configuration and Operations With Licensing Bases For CRH
  - Section 6, Industry Issue Applicability
  - Section 7, Air Inleakage
  - Section 8, Methodology for Dispositioning and Managing Discrepancies
- Part 3, Establishing and Maintaining CRH
  - Section 9, Long-Term CRH Program
- Appendices

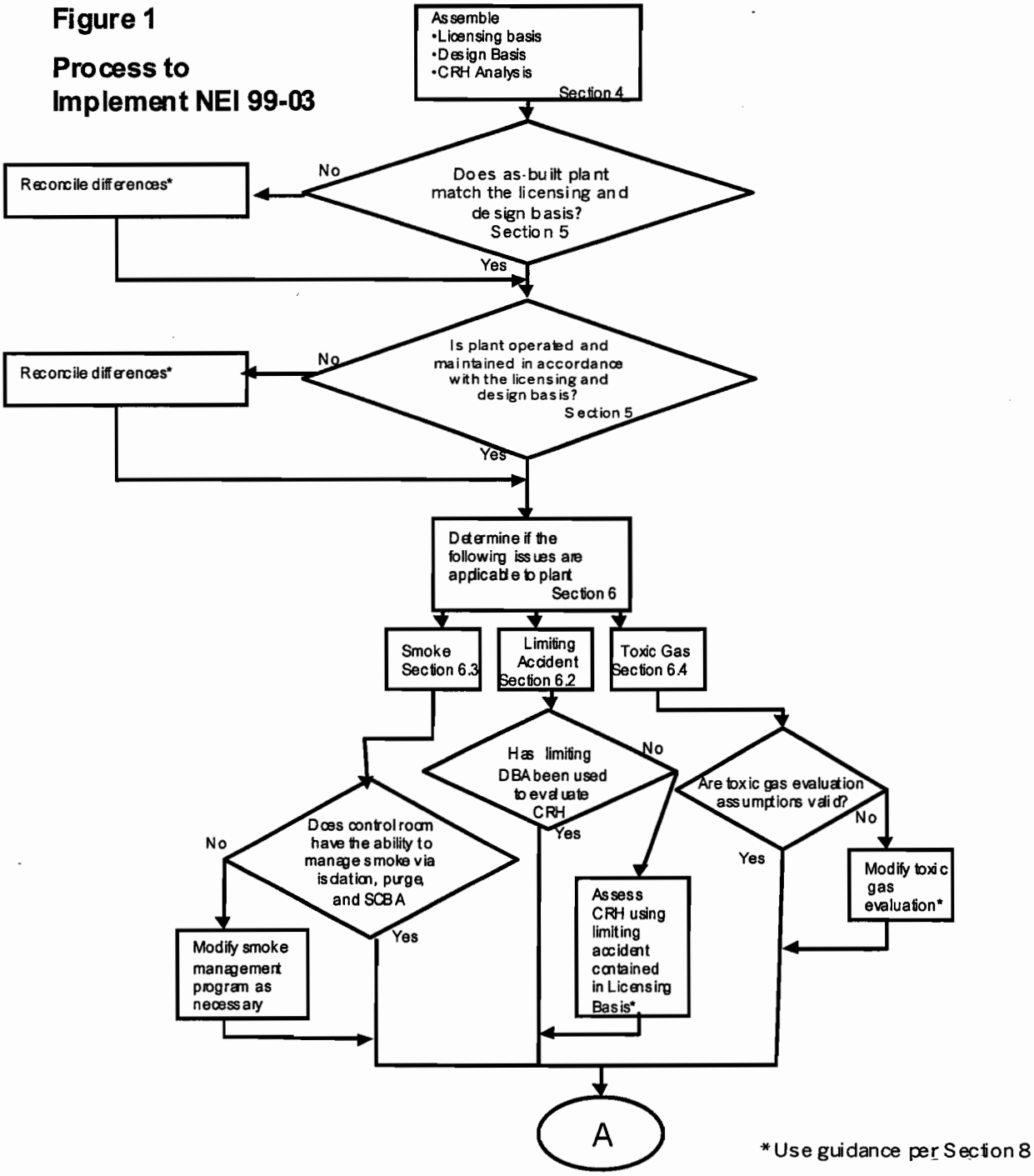
NEI

5

NEI 99-03 Process

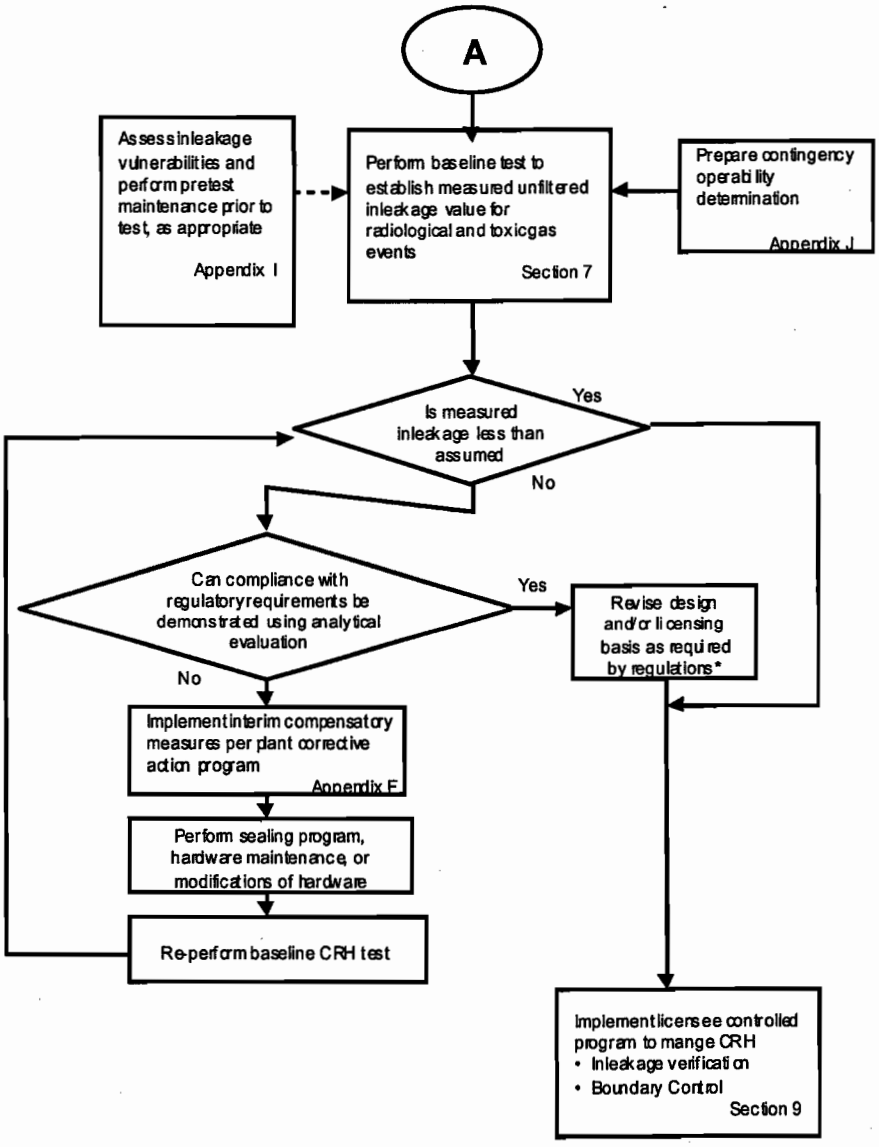
Figure 1

Process to Implement NEI 99-03



*Use guidance per Section 8

NEI 99-03 Process (continued)



* Use guidance per Section 8

Baseline Testing

NEI

8

Baseline Testing

- Baseline test to determine air in-leakage
- Baseline Test Attributes
  - Comprehensive
  - Reflects accident configuration lineup(s)
  - In accordance with recognized standards
- Acceptable baseline test methods
  - Integrated tracer gas testing
  - Component test method
  - Alternative test method(s)

NEI

9

Test Prerequisites

- CR envelope
  - Walkdowns
  - Sealing
  - Refurbishment
  - Repair
- HVAC system aligned and balanced
- Contingency Plans (radiological and toxic gas)
  - Operability
  - Preliminary calculations

NEI

10

Tracer Gas Method

- Per ASTM E741, *Standard Test Methods for Determining Air Change in a Single Zone by Means of Tracer Gas Dilution*
  - Valid for all CR designs
  - Recommended
    - Non-pressurized CR
    - CR with a large number of potential in-leakage sources
- Back calculates unfiltered in-leakage

NEI

11

Tracer Gas Method (Continued)

- Factors affecting accuracy
  - Concentration uniform throughout CR volume
    - Sampling techniques
    - Sampling locations
    - Injection location
    - Adequate mixing
  - Determination of CR volume
  - Environmental effects (wind, temperature, pressurization flows)

NEI

12

Component Test Method

- Three steps of test procedure
  - Demonstrate that CR spaces are positive pressure to all adjacent spaces
  - ID potential in-leakage sources (vulnerable)
  - Measure in-leakage of vulnerable components
    - Total in-leakage
- Only for pressurized CRs
  - Key -- CR positive pressure to all adjacent areas
  - Likely use -- CRs with small number of potential in-leakage vulnerabilities

NEI

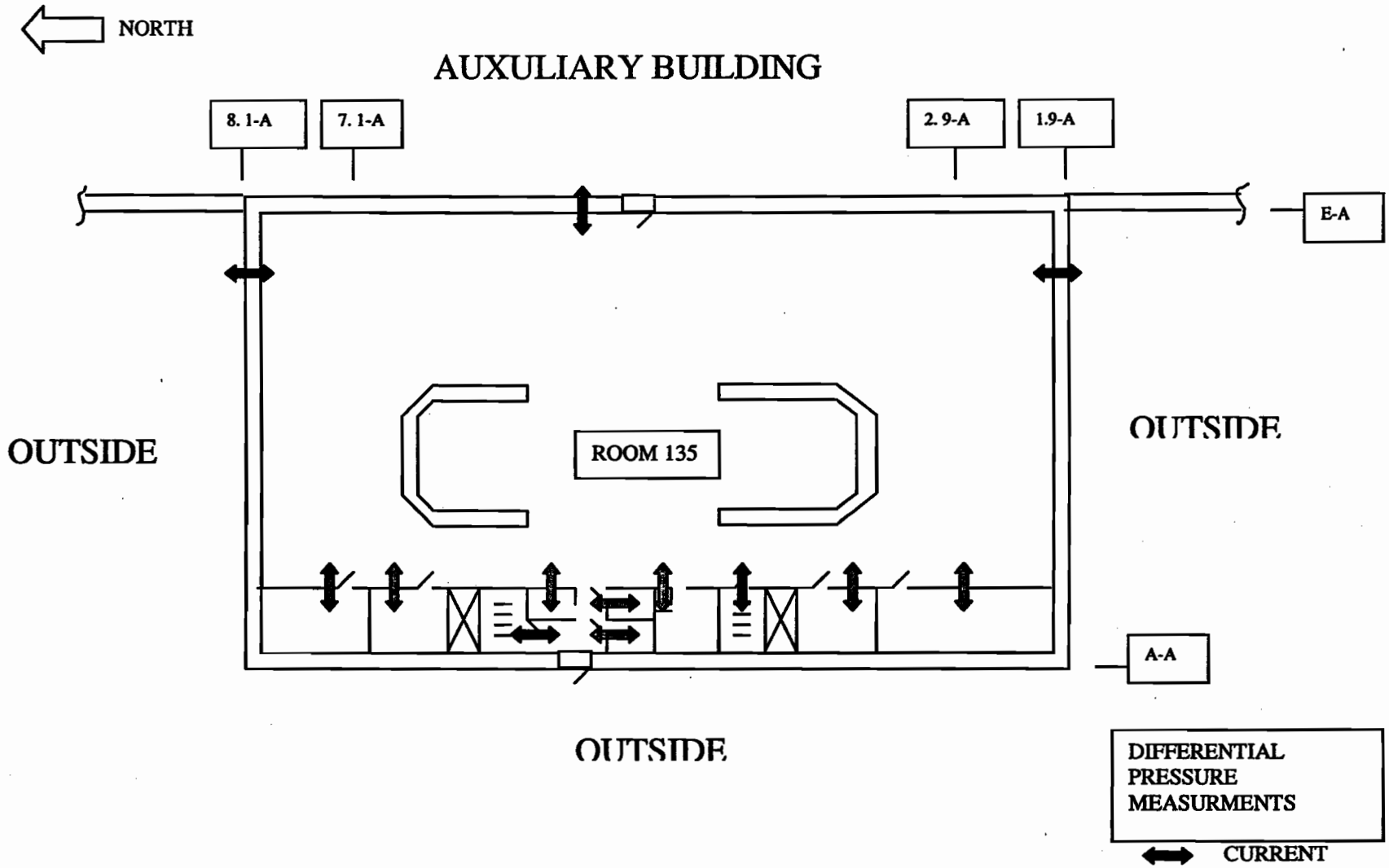
13

Component Test Method (Continued)

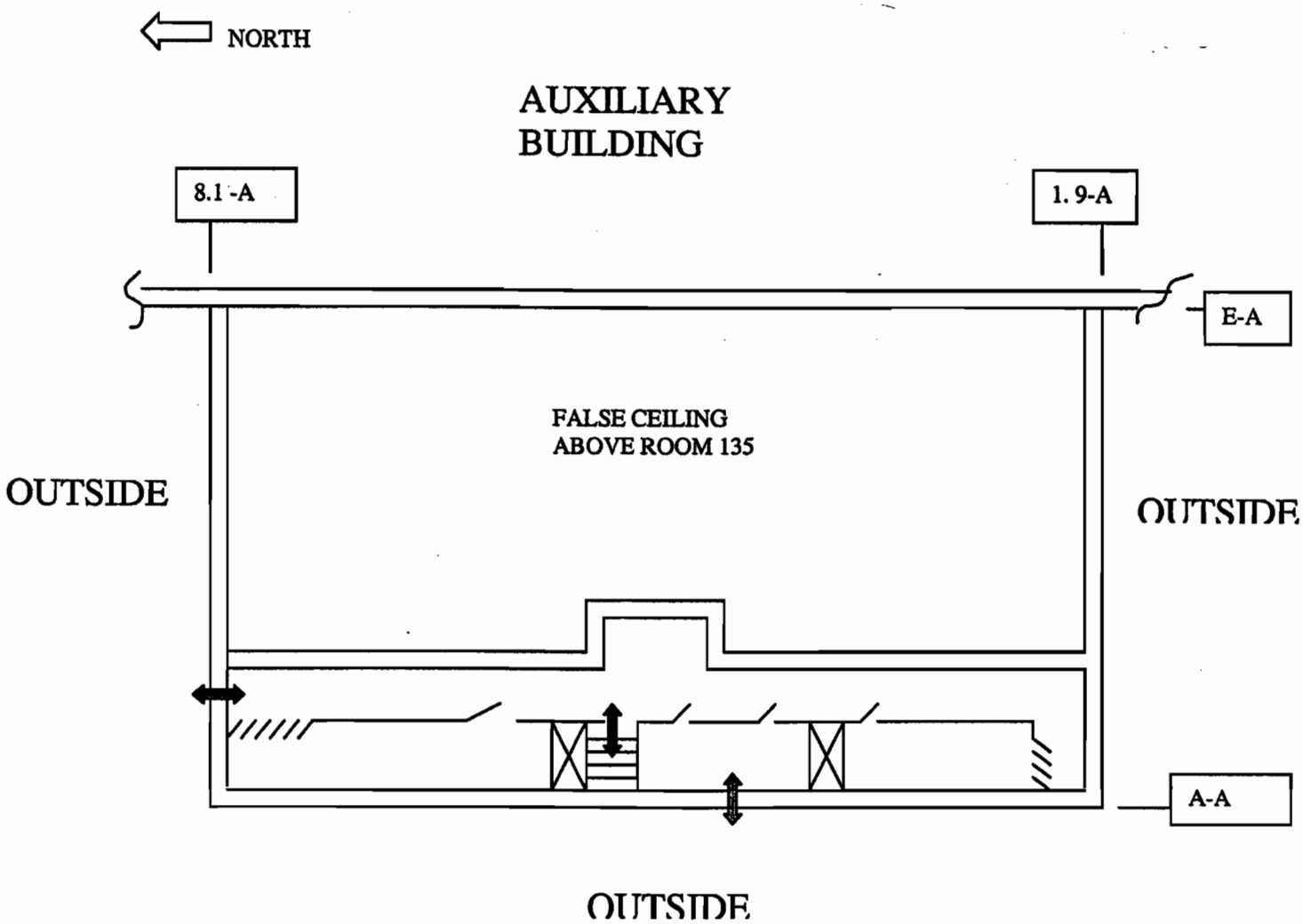
- Desirable characteristics
  - Majority of CR HVAC equipment internal
  - Minimal non-CR ventilation ducting or air systems penetrate CRE
  - Seam welded ventilation ducting
- Test requirements
  - Demonstrate CR pressure positive to all adjacent areas
    - High precision equipment
    - Multiple measurements
  - Identify CR design and unfiltered in-leakage locations (vulnerabilities)
  - Use proper pressure or vacuum decay test method (ASTM, ANS, ANSI, ASME) for each component

NEI

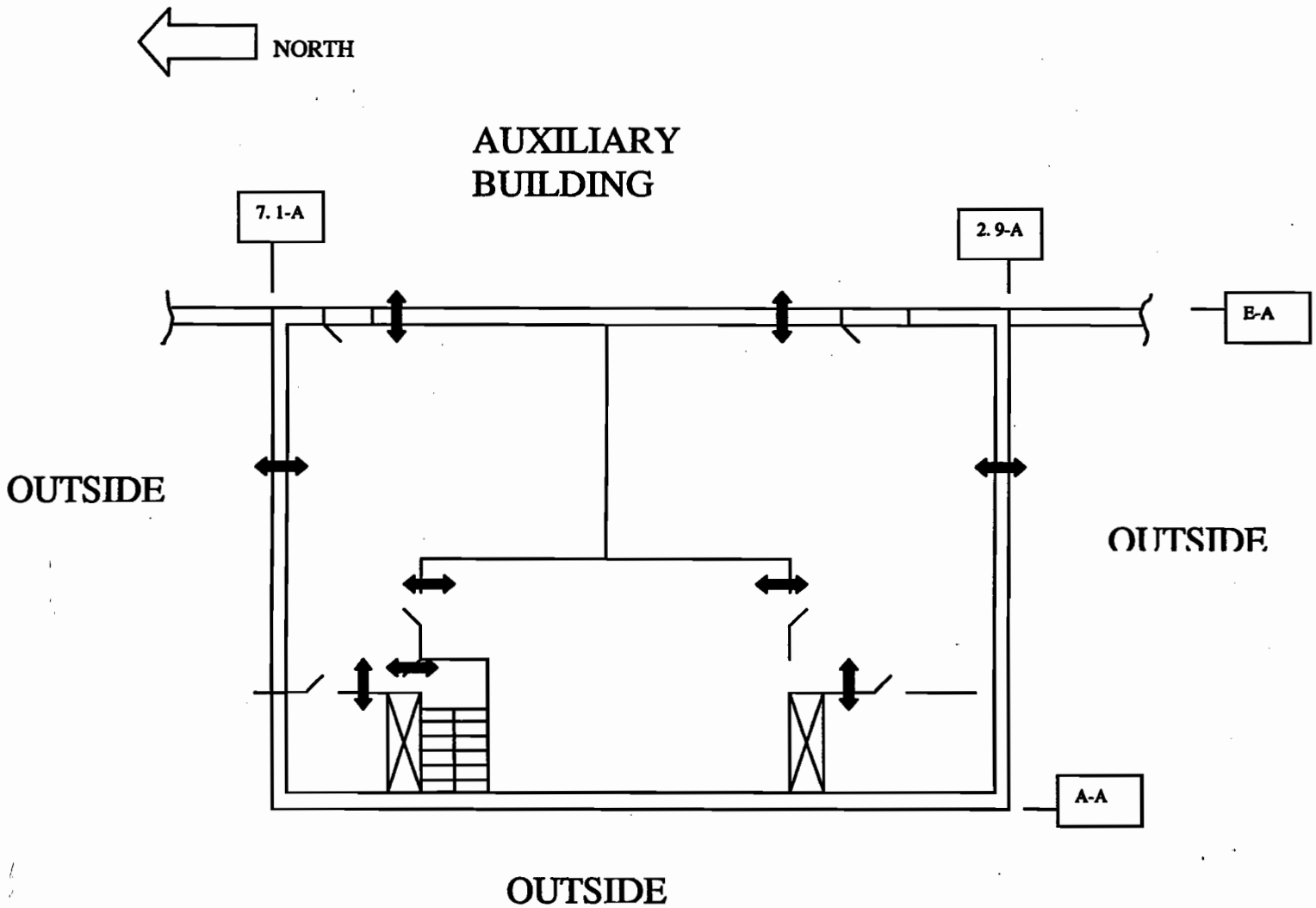
14



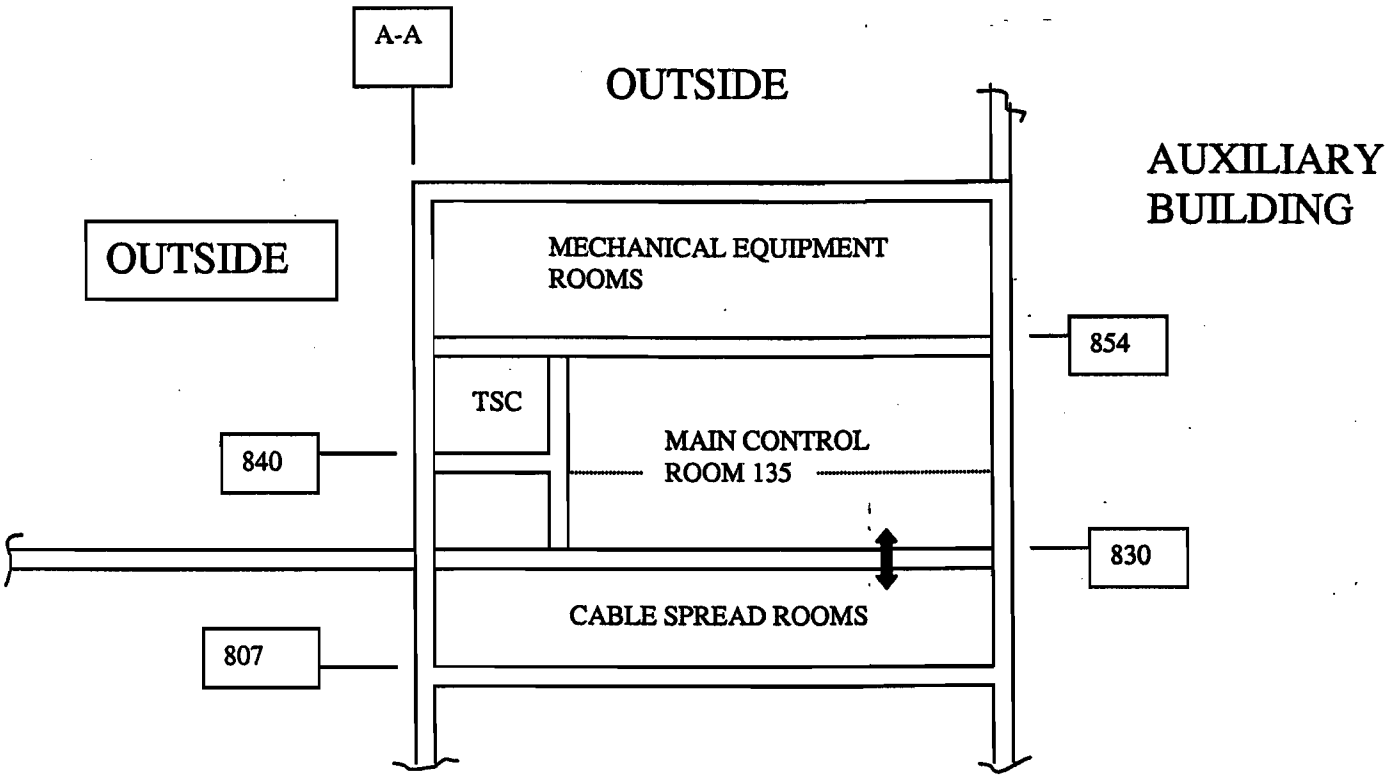
PLAN VIEW AT EL. 830' 0''



PLAN VIEW AT ELEVATION 840
(TSC ELEVATION)

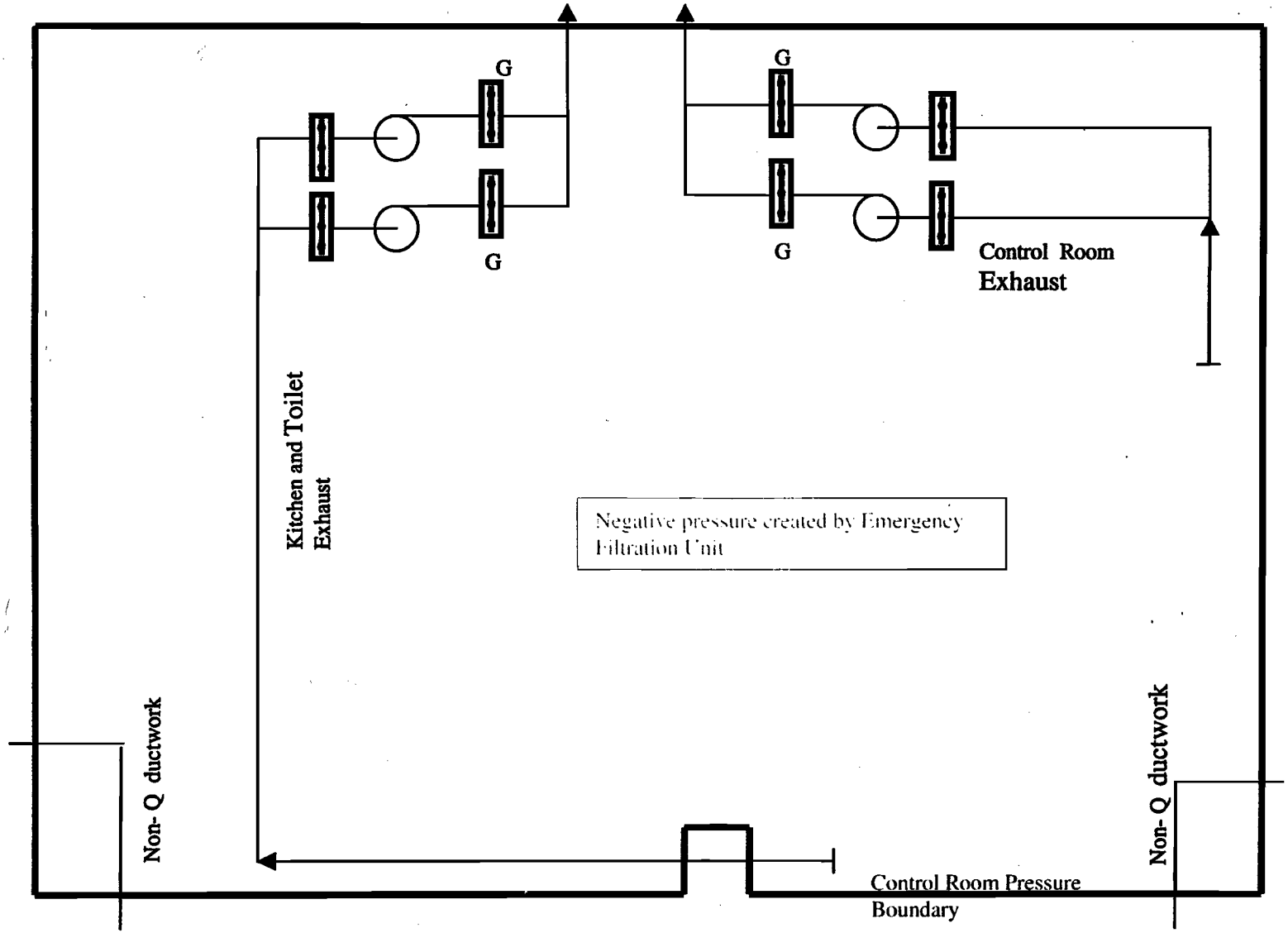


PLAN VIEW AT ELEVATION 854

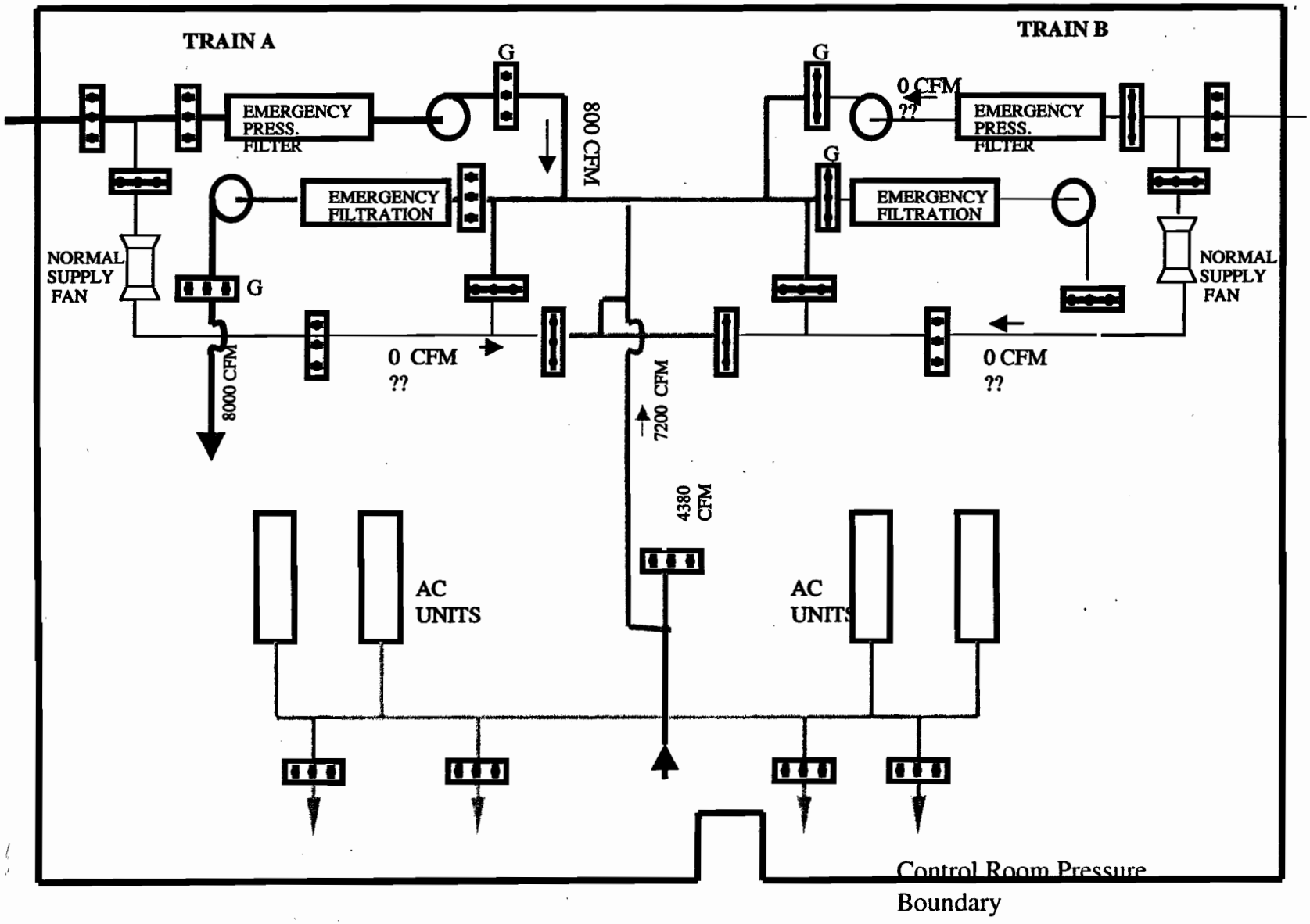


SECTION VIEW OF CONTROL BUILDING
VIEW LOOKING NORTH

Train A in Emergency Recirculation Train B OFF (All Exhaust Fans OFF)



Train A in Emergency Recirculation Train B OFF



Reasons for Component Test

- Large pressurization air flow can lead to significant uncertainties in tracer gas results
- Few vulnerabilities
  - Component test more practical than tracer gas, easier to control
  - Errors associated with tracer gas test may be larger than actual in-leakage
- Strengths of component test occur in CR designs where tracer gas limitations exist

NEI

20

Summary

- NEI 99-03 recommends quantifiable baseline testing
- Tracer gas and component testing are acceptable methods for measuring CR in-leakage

Reasonable assurance of accurate in-leakage measurement

NEI

21

Periodic Inleakage Assessment

NEI

22

Periodic Inleakage Assessment

- CRH Integrity Program
- Approach - Generic
  - Evaluate
    - Degree of available margin
    - Significant degradation
    - Magnitude of vulnerability
  - Assessment frequency – plant-specific
    - Baseline assessment findings
    - Baseline test result vs. analysis
    - Program effectiveness
- Retest if appropriate
  - Fixed testing schedule - not recommended
    - Unnecessary activity and cost
    - Distraction of the operator

NEI

23

CR Smoke Infiltration

NEI

24

CR Smoke Infiltration

- Qualitative approach
  - Smoke infiltration vulnerability assessed
    - Plant layout
    - Potential fire scenarios
    - Procedures
    - Ventilation lineups
    - SCBAs
    - Training
  - Mitigation of smoke infiltration
    - A progression of
      - Actions
      - Decision making logic
  - Corrective actions, if necessary

NEI

25

CR Smoke Infiltration

- Quantitative approach - not recommended
  - No design criteria for evaluating smoke infiltration, such as
    - Characteristics
    - Concentrations
    - Location
    - Duration
    - Transport properties
- Qualitative approach adequately addresses the smoke issue

NEI

26

Toxic Gas Assessment

NEI

27

Toxic Gas Assessment

- Typical
  - Evaluated per RG 1.78 and 1.95
  - Isolate only
- Plant alignment is important when determining in-leakage
- Review existing toxic gas evaluation
  - Increased in-leakage
  - New sources
- Periodic toxic gas re-assessment
  - Frequency
    - Industrial areas 5 years, rural areas 10 years
    - Increased in-leakage
    - New sources

NEI

28

Analysis

NEI

29

ANALYSIS IMPROVEMENT OPPORTUNITIES

- Alternative Source Term insights (timing, chemical form)
- Iodine spike modeling
- Fuel Handling Accident input assumptions
- LOCA ECCS passive failure assumption
- Meteorology and dispersion modeling

NEI

30

Improvements

- Alternate Source Term (AST)
  - 5 Rem TEDE
- 50 Rem Thyroid Limit (TID)
  - 30 Rem per SRP 6.4.2
- Elimination of ECCS passive failure
  - 50 gpm for 30 minutes at 24 hours after Maximum Hypothetical Accident

NEI

31

AST POSITIONS VALID FOR TID

- **Fuel Handling Accident**
  - **Revision to Isotopic Gap Fractions**
  - **Decontamination Factor of 200 for 23 ft of water**
- **SGTR Spiking factor of 335**
- **SGTR and MSLB**
  - **Spike duration less than 8 hrs**
- **Lower BWR Containment Leak Rates at time greater than 24 hours**
- **Enhanced containment mixing rates**

NEI

32

METEOROLOGY AND DISPERSION MODELING

- Use of ARCON96 acceptable for χ/Q calculations
  - Advanced Modeling Technique
  - Critical Component for Analysis
  - Does Not Handle All Release Scenarios
    - Free Standing Stack Releases
    - High Energy Steam Line Valve Releases
- MSSV and ADV χ/Q should consider elevated release
- ARCON96 upgrade
  - Joint NRC/NEI Effort
  - NRC Contribution
    - Enhancement of ARCON96
  - NEI Contribution
    - Technical Inputs
    - Benchmarking

NEI

33

Technical Specification

NEI

34

Technical Specifications

- Options considered
  - No change from existing requirements
  - Commitment to a CRH Program as described in NEI 99-03
  - Admin TS committing to a CRH program
  - Moving CR HVAC TS requirements into a TRM and adoption of periodic in-leakage assessment requirements through a CRH program
  - Adoption of an in-leakage surveillance in the current TS

NEI

35

Technical Specifications

- Observations
  - Commitment to reassess CRH in-leakage periodically is the right thing to do
  - Current TS surveillances adequately address the operability of CR HVAC systems
  - Technical Specification should focus on parameters and indications observable and controllable by the operator
  - In-leakage is a design input parameter and does not meet the criteria for a TS parameter but should be controlled under an Appendix B program

NEI

36

Technical Specifications

- NEI 99-03
  - Recommends establishing a CRH Program
    - 85% of survey respondents agreed to some form of commitment to a CRH program
  - Provides guidance for a CRH Program

NEI

37

Conclusions

NEI

38

CRH Industry Survey

- NEI survey
  - Understand utility perspective on key variation in design and operating practices and issues addressed in NEI 99-03
  - 90% response
- Survey responses generally support NEI 99-03 recommendations

NEI

39

Future Activities

- Address outstanding issues
  - Alternate test to tracer gas test
  - Technical specifications (need, AOT and test frequency)
  - Change to common licensing basis - accidents analyzed, methods, and assumptions
  - Analysis conservatisms vs. non-conservatisms
- NRC staff comments on NEI 99-03 requested
- Issue revised NEI 99-03 to Industry for final review -- TBD
- Issue final NEI 99-03 and conduct industry workshop -- TBD

NEI

40

R

Control Room Habitability

Jack Hayes

Senior Health Physicist

Probabilistic Safety Assessment Branch

U.S. Nuclear Regulatory Commission

Unfiltered Inleakage Testing

- About 25% control rooms tested
- None satisfied analyses assumed value
- All satisfied Technical Specifications SR
- Each licensee was able to recover
  - New analyses
  - Restored boundary
- No plants shut down

Recent History

Mar 1998	NRC invited Industry Interface
Jul 1998	Workshop
Sep 1998	Initial Interface Meeting
Aug 1999	First draft NEI 99-03 issued
Nov 1999	Commenced revision effort
Jan – Jun 2000	Monthly meetings
Oct 2000	Second draft NEI 99-03

Accomplishments

- Open Dialogue
- Improved Awareness of Issue
- Concentration of Information
- Significant Areas of Agreement

Agreements to Move Forward

- Industry complete NEI 99-03 on own
- NRC issue Regulatory Guidance
  - Resolve remaining issues in public comment process

Remaining Discussion Areas

- Testing method for unfiltered inleakage
- More realistic radiological analyses:
  - Most Limiting Design Basis Accident
  - Remove Over Conservatisms
  - Remove Non-Conservatisms
  - Appropriate Acceptance Criteria
- Smoke & Toxic Gas
- Technical Specifications

Presentation to ACRS
Severe Accident Management Subcommittee

Subcommittee Meeting

November 15, 2000

Charles G. Tinkler
Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SEVERE ACCIDENT RESEARCH

OUTLINE

- SEVERE ACCIDENT RESEARCH APPLICATIONS
  - Revised Source Term Implementation
  - Steam Generator Tube Integrity
  - Spent Fuel Pool Accident Analysis
  - Dry Cask Storage
  - Risk Informed 10 CFR50.44
  - Improved Containment T/H Analysis
- Improvement & Consolidation of Severe Accident Codes

SEVERE ACCIDENT RESEARCH OUTLINE (CONT.)

- Cooperative Experimental Research
  - OECD LHF PROGRAM
  - PHEBUS
  - RASPLAV/MASCA
  - MACE
- Mox & High Burn up Fuel
- MAAP Code Review

SEVERE ACCIDENT RESEARCH APPLICATIONS

- **IMPLEMENTATION OF REVISED SOURCE TERM (NUREG-1465) FOR DBA LICENSING ANALYSIS**
  - New source term significantly more mechanistic than original (TID-14844)
  - Safety improvements & cost savings
  - RADTRAD Code developed for plant analysis
    - Improved models for ESF and natural deposition mechanisms
    - RADTRAD version 3.02a available August 2000
    - Widely requested by industry (~ 30 utilities/consultants)
  - Maintenance activities planned (including possible user group activities)
  - Potential code development for dry cask applications
  - Potential code development for E.Q. analysis

SEVERE ACCIDENT RESEARCH APPLICATIONS (Cont.)

STEAM GENERATOR TUBE INTEGRITY

OBJECTIVE: Determine the severe accident thermal hydraulic boundary conditions seen by SG tube in order to assess likelihood of thermally induced tube rupture given a severe accident scenario

- Analysis using SCDAP/RELAP5 have been performed for representative plants for potentially risk-significant scenarios (high pressure TMLB' sequences with depressurized secondary side) to estimate effects of high temp fluid circulation.
  - SR5 analyses predict failure of hot leg or surge line before unflawed SG tubes.
  - Sensitivities on T-H Modeling did not alter conclusion on tube integrity but margins are relatively small.

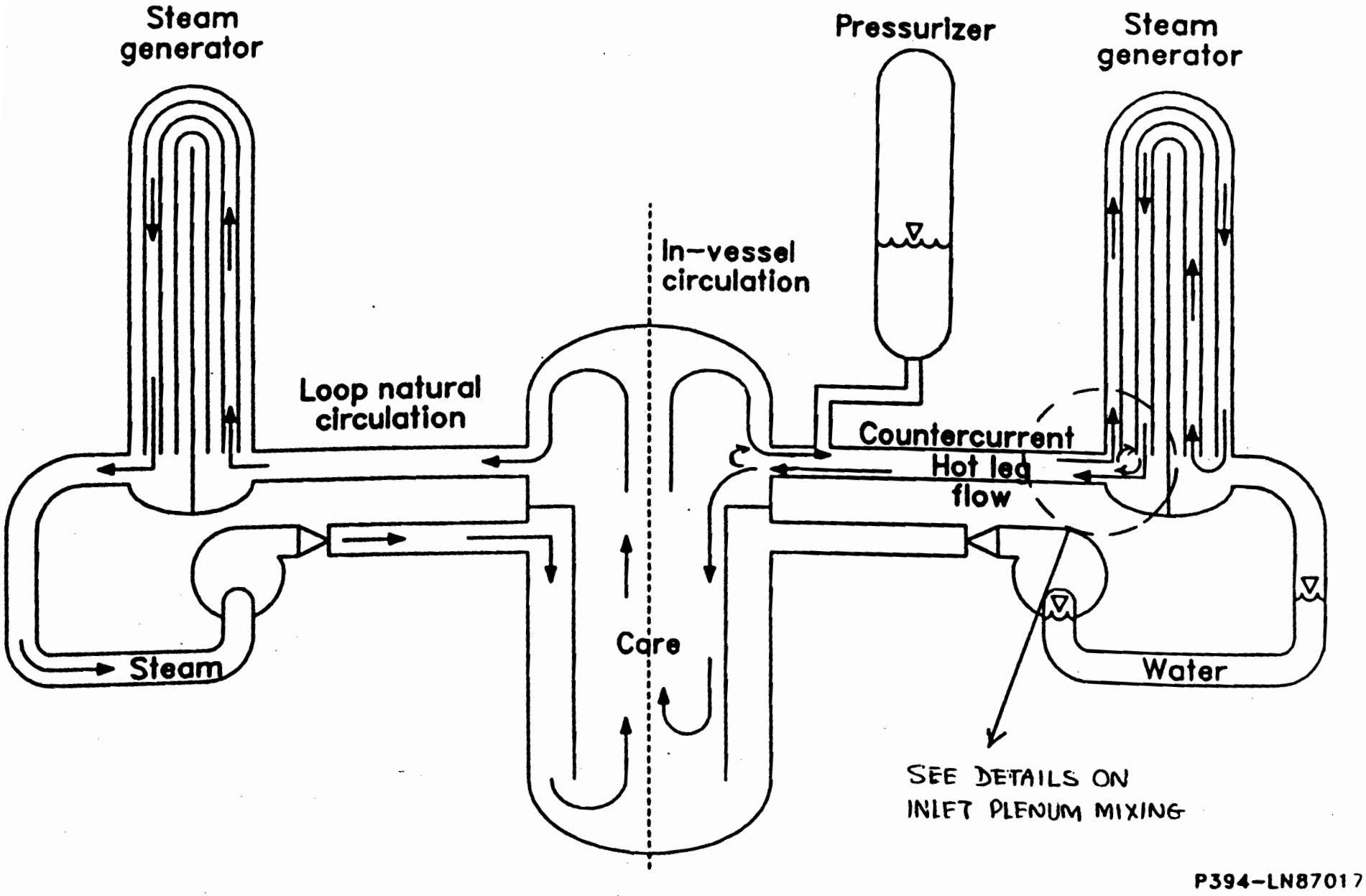


Figure 1. Severe accident natural circulation flows.

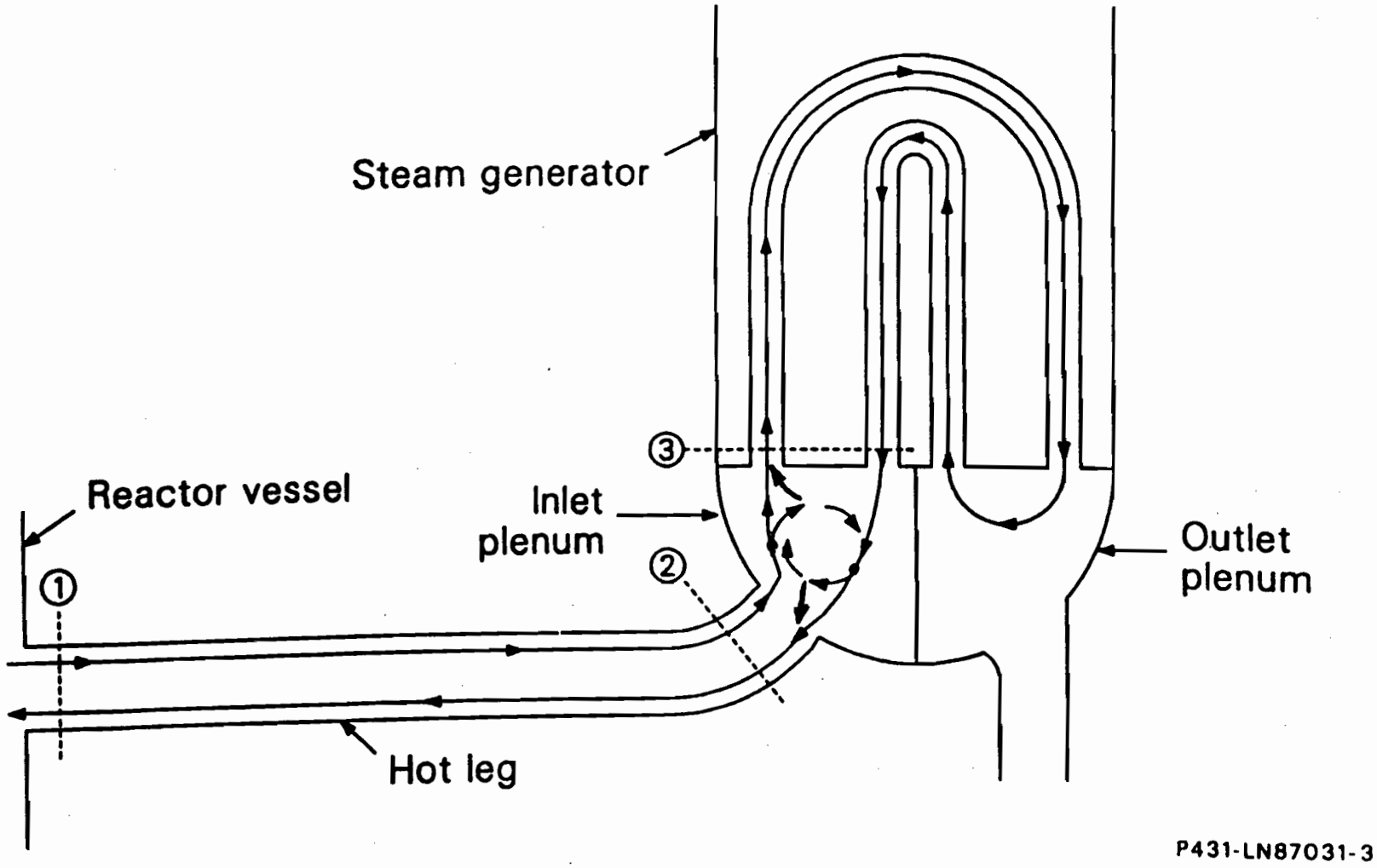
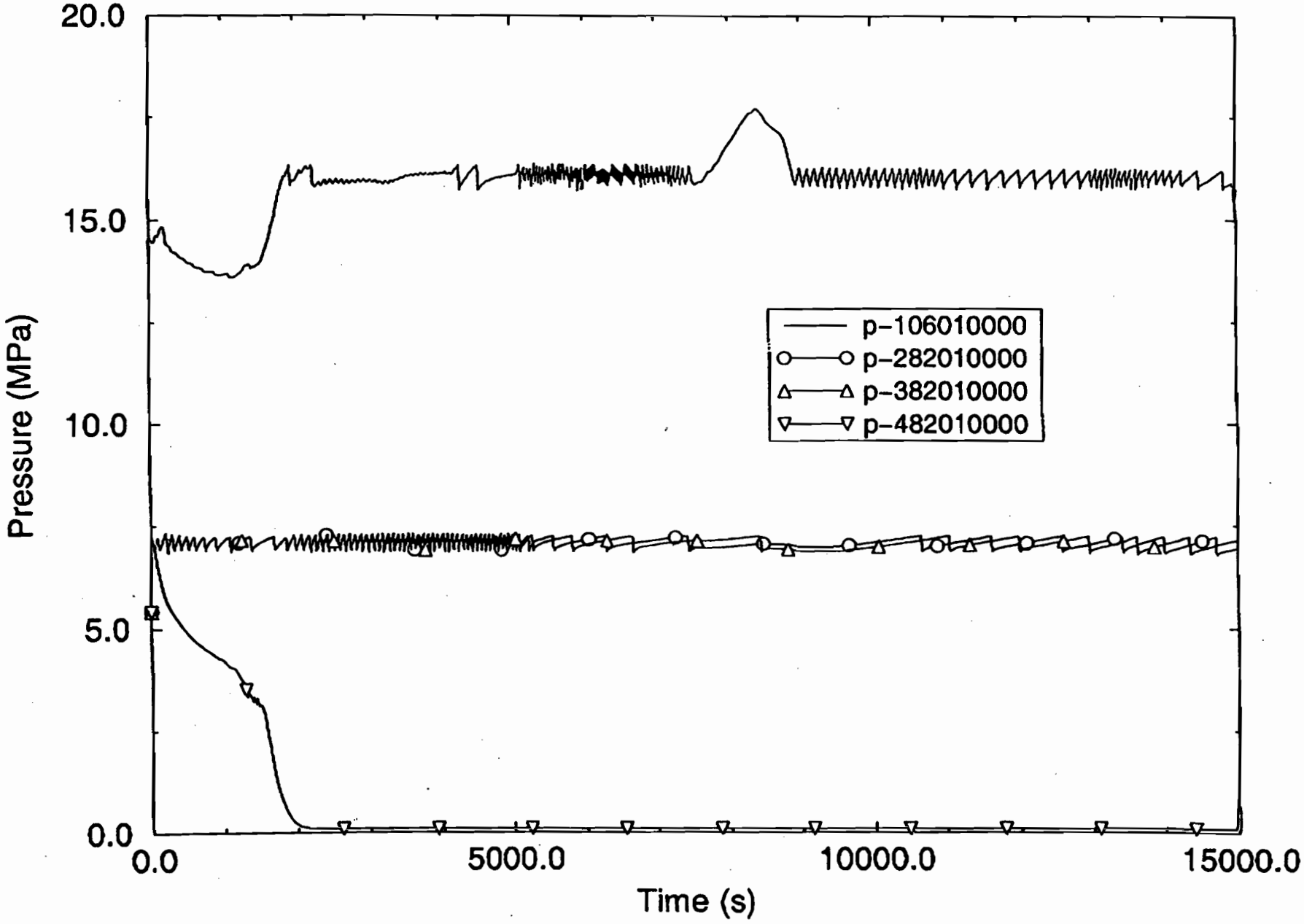
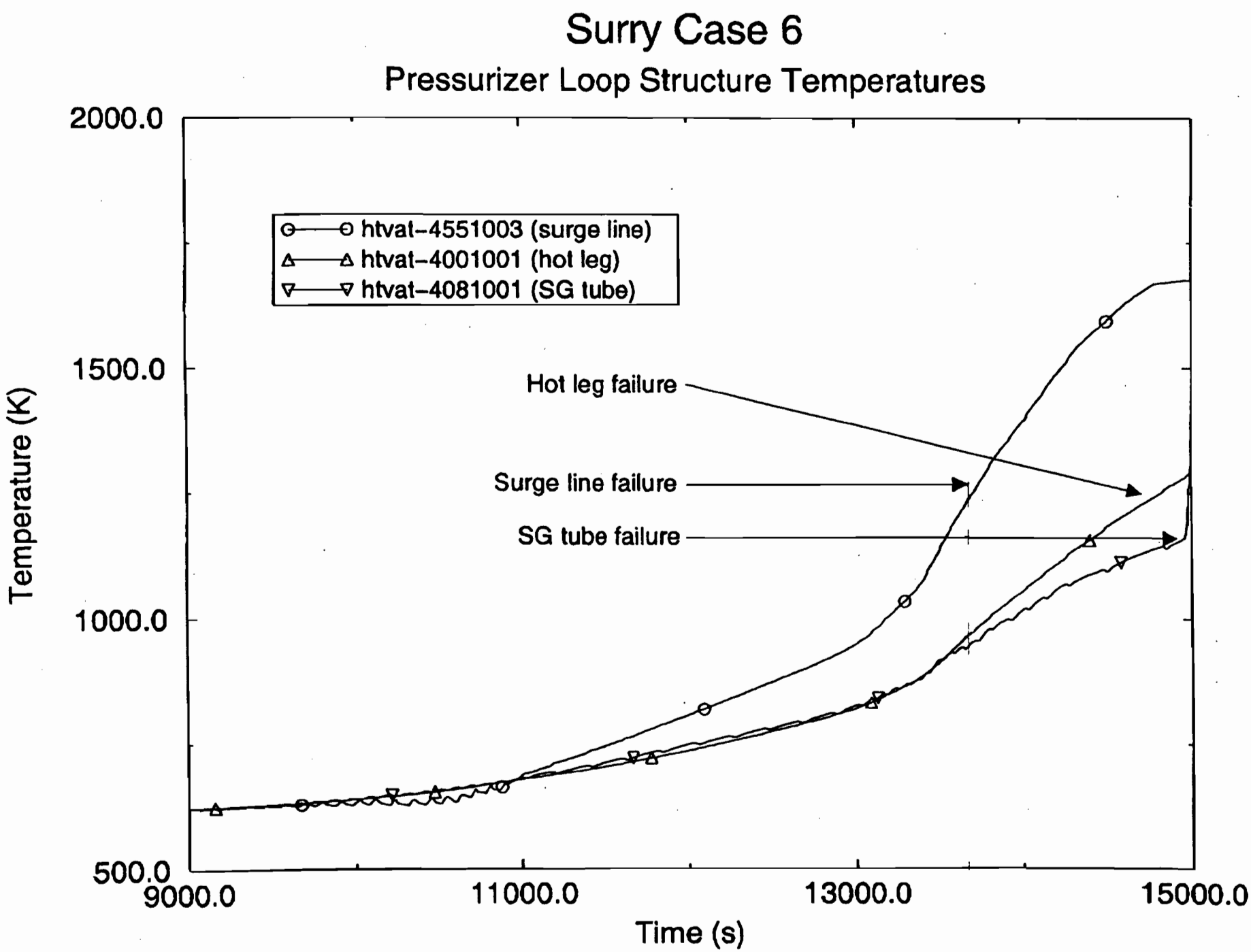


Figure 2. Hot leg natural circulation stream flows.

Surry Case 6

RCS and SG Secondary Pressures




Surry Case 6
Pressurizer Loop Structure Temperatures
htvat-4551003 (surge line)
htvat-4001001 (hot leg)
htvat-4081001 (SG tube)
Hot leg failure
Surge line failure
SG tube failure
Temperature (K)
2000.0
1500.0
1000.0
500.0
9000.0
11000.0
13000.0
15000.0
Time (s)

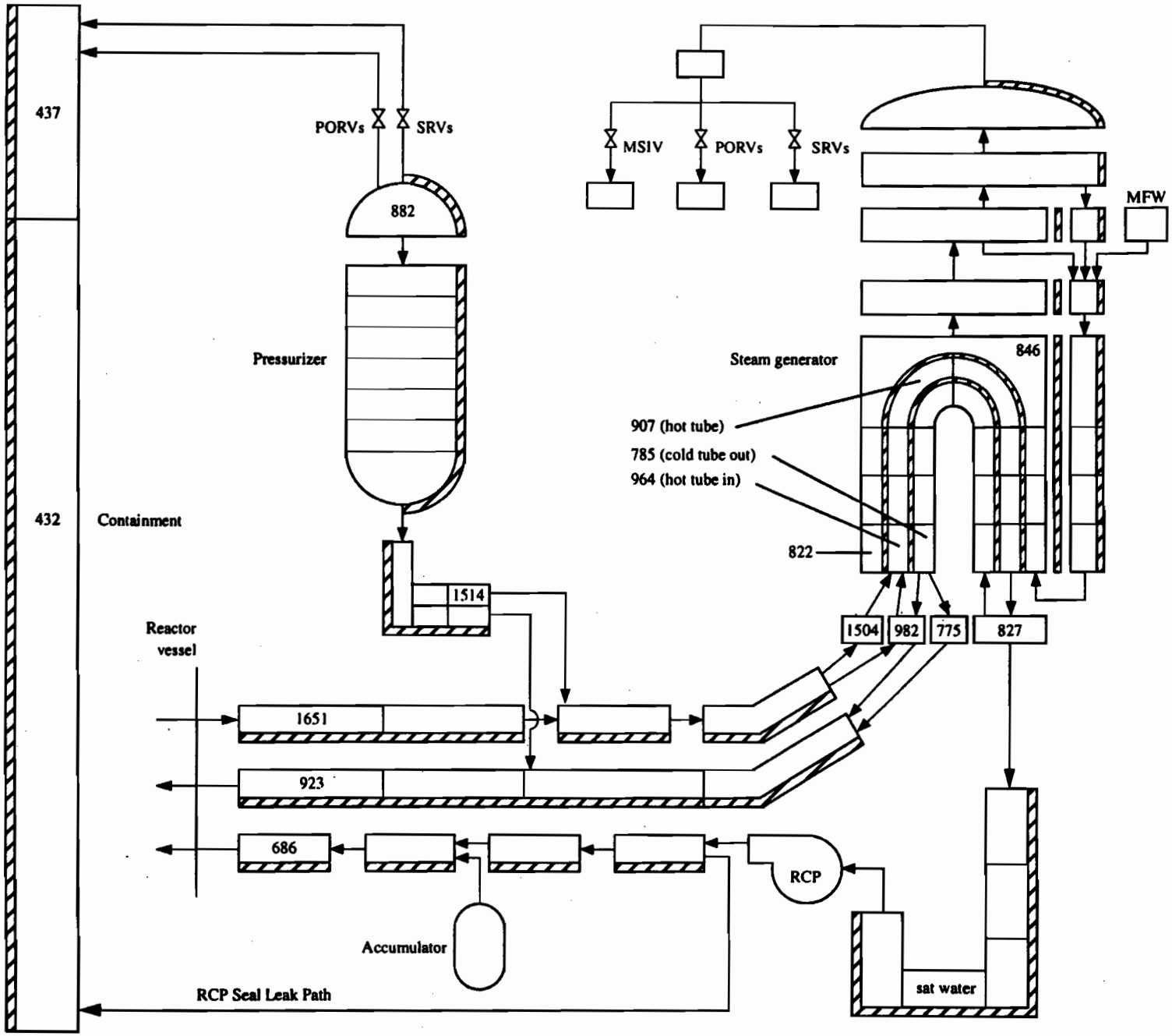


Figure 1. Surry Case 6 vapor temperatures (K) near the time of surge line failure (13,730 s).

STEAM GENERATOR TUBE INTEGRITY

- SG tube T/H boundary conditions are most directly influenced by variations in the accident sequence which determine pressurization/depressurization of the primary and secondary system
  - Failure of primary and secondary relief valves
  - Primary side PORV operation (Accident Management)
  - Pump seal leakage
- SG tube T/H boundary conditions are also influenced by phenomenological issues associated with counter current natural circulation and RCS T/H.
  - SG inlet plenum mixing
  - heat transfer modeling
  - loop seal clearing
- Variations in the treatment of these issues within reasonable ranges did not significantly worsen SG tube boundary conditions.

Conclusions

- Tube heating during severe accidents has been analyzed using benchmarked models (validated against scaled experimental data), undergone peer review, and sensitivities examined through parametric variations.

- Variation in tube temperatures estimated at 20-50 °K.

- Evaluation of tube performance during severe accidents would benefit from resolution of uncertainty regarding T/H conditions.

- Confirmation of temperature variation/uncertainty underway.

  – More rigorous consideration of uncertainties in SG inlet plenum mixing.

  – Additional sequence/plant variation.

  – More detailed CFD modeling.

New Research in the Area of Severe Accident-Induced Steam Generator Tube Rupture

- NRR user need letter, February 8, 2000, requested RES to develop a confirmatory research program to address SG tube integrity.

- The T/H part of this program (already started) includes additional work in the following area:

  - Accident sequence variations
  - Plant design differences
  - Inlet plenum mixing
  - Tube to tube variations
  - Core melt progression

Inlet Plenum Mixing

- The SCDAP/RELAP5 code will continue to be used as the principal tool for analysis of the tube T/H boundary conditions.
- Phenomenological uncertainty in the natural circulation calculation has centered on the issue of mixing in the SG inlet plenum.
- Additional uncertainty relates to heat transfer assumptions
- Earlier work considered single and multiple simultaneous variations in inlet plenum mixing characteristics as well as variations in heat transfer coefficients.

Inlet Plenum Mixing (Continued)

- New analysis performed under this plan will involve a more rigorous treatment of uncertainties.

- Distributions will be developed for the individual mixing parameters and heat transfer coefficients and sampled using Monte-Carlo techniques. SCDAP/RELAP5 analysis will then be performed for sampled points to develop a probabilistically weighted picture of SG tube temperatures.

- The current plan is to use the following parameters for the analysis: mixing fraction, recirculation ratio, number of tubes carrying forward (hot) flow, hot leg and surge line heat transfer coefficients.

Effects of Leakage and Tube to Tube Variations

- SCDAP/RELAP5 analysis limited in ability to resolve variation in T/H effects among tubes.
- To estimate tube-to-tube variations, re-examine the experimental basis for the modeling (i. e., the 1/7th scale test data) to determine the appropriate variability for plant conditions.
- Use computational fluid dynamics (CFD) code to predict inlet plenum mixing and tube to tube variations, including the effects of leakage.
- CFD code will need benchmarking against experimental data, but fundamentally, CFD codes have greater inherent capabilities for solving this type of fluid flow problem.

Current Status of Work

- SCDAP/RELAP5 work will be done at INEEL. The contract is in place and the work was started in July 2000.

- The work to date has included establishing a new baseline calculation using the newest version of SCDAP/RELAP5, Mod 3.3, and beginning the development of parameter distributions for the inlet plenum mixing task.

- Work is now planned to be complete in March 2002, with intermediate products available as work on individual tasks are completed.

- The CFD work will begin with a validation of the code to the experimental data and will begin in the first quarter of FY 2001.

Severe Accident Research Applications
Spent Fuel Pool Accident Analysis

- Evaluation of temperature criteria for risk analysis
  - "Ignition" temperature (threshold for temperature escalation leading to significant fuel damage)
  - Temperature threshold for fuel failure and significant fission product release

- CFD analysis of convective cooling by air flow to evaluate critical decay time

- Consequence analysis to evaluate sensitivity of source term, plume behavior, evacuation.

Spent Fuel Pool Accidents (continued)

- Practically, the temperature criteria was used in draft generic study:

  1) Signal onset of significant fuel pool release for evaluating time for ad hoc evacuation.

  2) For determination of decay heat level and corresponding time ("critical decay time") at which equilibrium temperature could be maintained, <u>precluding large release</u> (~ 5 years).

- NRC has reevaluated appropriateness of temperature criteria considering:
  - Zr reaction kinetics
  - Hydriding/autoignition
  - Fuel damage testing
  - Fission product release data (ruthenium)
  - Materials interactions

Summary

	Adequacy of 10 hrs for Evacuation	Precluding Large Release Fuel <5yrs	Precluding Large Release Fuel >5yrs
Dominant Air Environment	900 °C	600 °C	800 °C
Dominant Steam Environment	1200 °C	N/A	N/A

- Use of temperature criteria must be supported by analysis of all significant heat generation and loss mechanisms.
- Determination of an acceptable long term condition requires confirmation of equilibrium temperature condition.
- Integrated modeling of thermal hydraulics, cladding reactions and fuel heatup and fission product release would provide consistent consideration of conditions for sequence specific analysis. Would provide means for more realistic estimates.

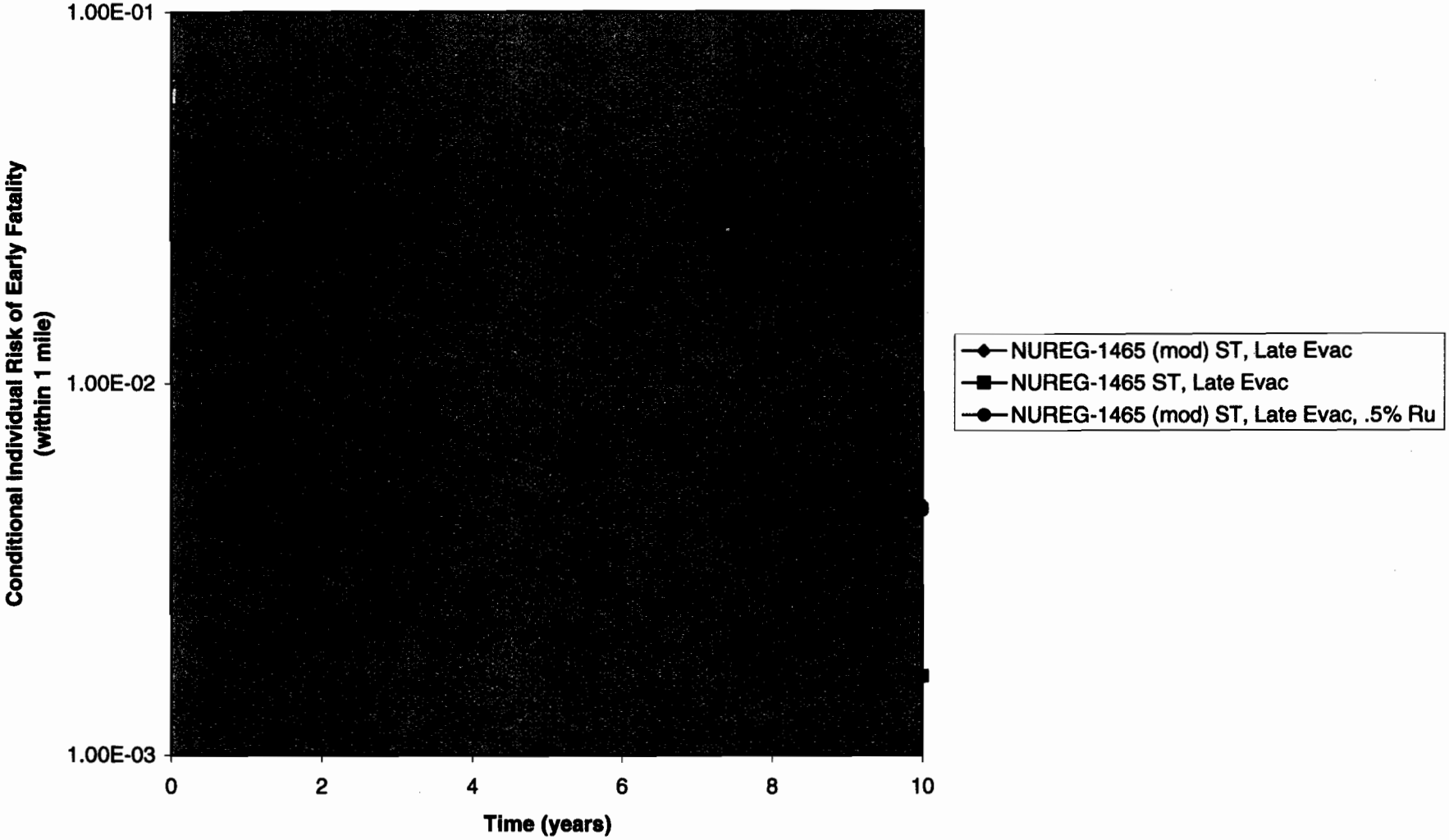
Consequence Assessment

Issue examined

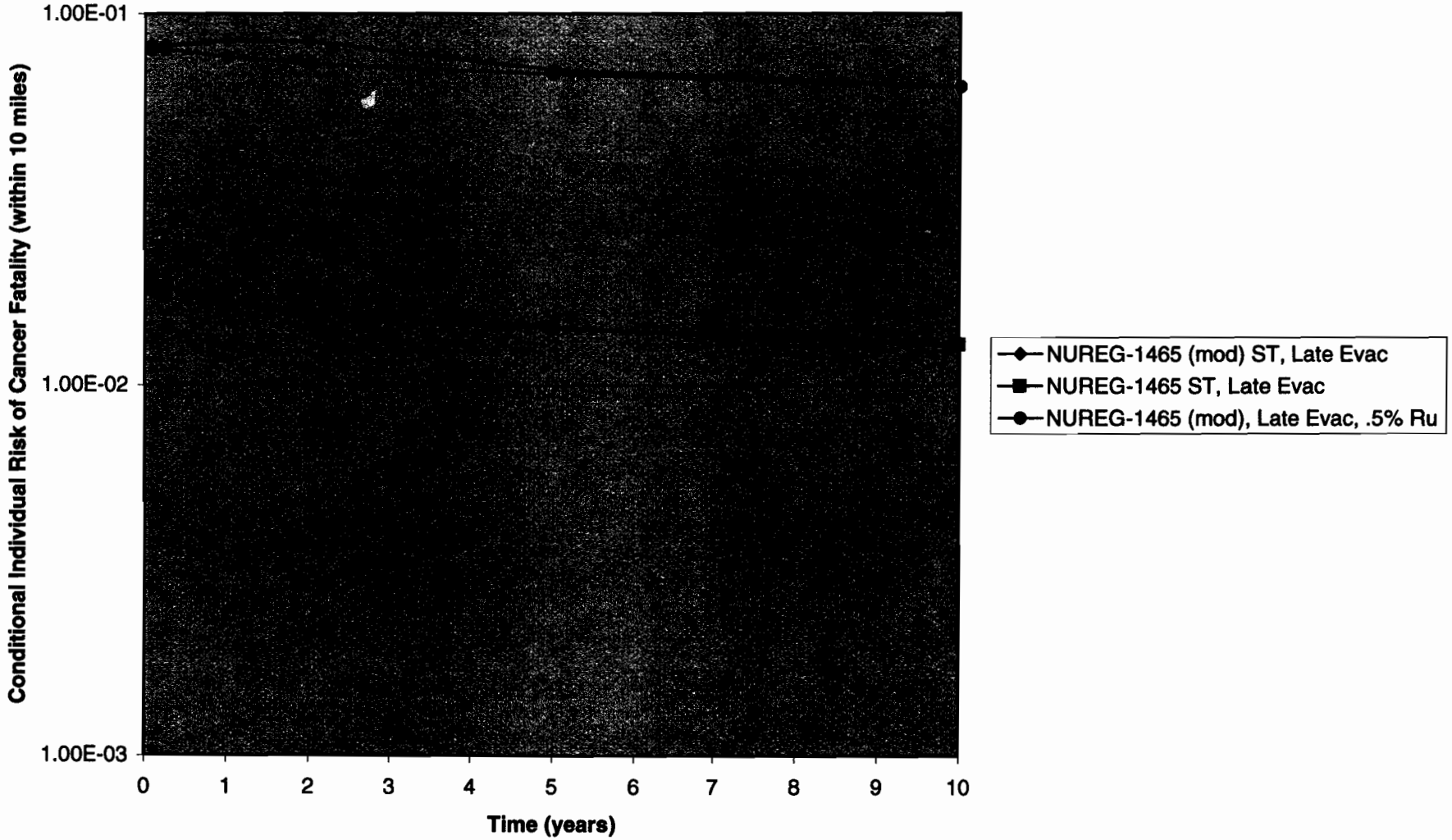
- reassessment of source term and release fractions of fission products
  - ruthenium
  - cesium
  - fuel fines
- reduced inventory for different decay time
- Plume spreading
- Plume heat content
- early vs. late evacuation

Results of large number of MACCS calculations were used to understand decommissioning risk in staff's generic study.

Mean Consequences
(Accident frequencies about 1E-6/year or less)



Mean Consequences
(Accident frequencies about 1E-6/year or less)



Severe Accident Research Application
Dose Assessment for Dry Storage

Object of the analysis

Provide more realistic quantification, with uncertainty bounds, of offsite doses associated with dry storage cask leakage

Summary of approach

Used RADTRAD code with isotopic inventories for spent fuel after 5 years of decay to calculate individual offsite dose

- Focus of more realistic modeling was aerosol deposition in cask

Conclusion

Modeling aerosol deposition in cask reduces dose by a factor of 400

Dose Modeling for Individual Offsite Dose from Dry Storage Cask Leakage
(From HI-STORM Safety Analysis Report)

Parameter	Value of parameter for...			
	Accident		Off-normal	Normal
Fraction of crud released	1		.15	.15
Fraction of fuel assemblies releasing fission products	1		.1	.01
Fraction of fission product inventory released from each fuel assembly	fission product gas	.3	same	same
	volatile fission products	$2x10^{-4}$	same	same
	actinides, non-volatile fission products (fines)	$3x10^{-5}$	same	same
free volume of the cask	$6.0x10^{6}$ cm^3		same	same
leak rate of the cask	$1.3x10^{-5}$ cm^3/sec		$9.5x10^{-6}$ cm^3/sec	$9.5x10^{-6}$ cm^3/sec
dilution factor (i.e., X/Q)	$8.0x10^{-3}$ sec/m^3		$1.6x10^{-4}$ sec/m^3	$1.6x10^{-4}$ sec/m^3
breathing rate	$3.3x10^{-4}$ m^3/sec		same	same
dose conversion factors	Federal Report Guidance Reports 11 and 12 using most conservative clearance class		same	same
release duration	30 days		1 year	1 year
dose limit	TEDE	5 rem	25 mrem	25 mrem

Approaches to Determine Deposition

NUREG/CR-6189

- Provides rough estimate of λ
- based on dimensions and T-H conditions for a reactor containment

NUREG/CR-6189 adjusted for cask dimensions

- provide insight into effect of dimensions (containment vs. cask)
- not directly applicable, because do not know how much deposition due to each mechanism

Stand-alone calculation of settling using distributions for aerosol density, diameter, and shape factor from reactor accident studies

- ignores additional deposition due to thermophoresis

Stand-alone calculation of settling using distribution for aerosol density, diameter, and shape factor for a spent fuel cask

- provide best estimate of λ

Comparison with MELCOR Results

MELCOR accident analyses performed for TN-125 cask with a 4 mm^2 hole (SAND98-1171/7, Data and Methods for the Assessment of the Risks Associated with Maritime Transport of Radioactive Materials, Results of the SeaRAM Program Studies, May 1998).

Calculated accident dose using RADTRAD for HI-STORM using MELCOR-predicted deposition rate constants from the TN-125 cask study.

- RADTRAD accident doses using MELCOR-predicted deposition rate constants were .070 to .11 mrem.

- RADTRAD accident dose using settling rate constant (case 5b) was .096 mrem.

- Excellent agreement because settling was dominant mechanism in MELCOR analyses.

Doses for Accident Conditions

Case	Deposition Modeling	TEDE for an Individual at the Site Boundary (mrem)		
		lower bound	best estimate	upper bound
1	none	N/A	N/A	44
2a,2b,2c	NUREG/CR-6189	.037	.059	.097
3a,3b,3c	NUREG/CR-6189 with cask dimensions	.0088	.014	.024
4a,4b,4c	settling only, based on reactor containment conditions	.027	.077	.35
5a,5b,5c	settling only, based on cask conditions	.031	.096	.24

Risk Informing 10CFR50

- As part of risk informing 50.44 NRC drew on severe accident research insights of recently completed DCH research on ice condensers (NUREG/CR-6427)
  - DCH, per se, not a threat to ice condensers, but resolution considered other failure modes
  - H_2 accumulation; uncontrolled in certain accident sequences, where igniters are inoperable, could be important.
- Specification of hydrogen release for risk informed regulation

IMPROVED CONTAINMENT T/H

- CONTAIN, originally developed as a more detailed containment severe accident T/H code, was also used by NRC to evaluate ALWR containment response to DBA's
  - Superior capability to traditional DBA single volume codes (CONTEMPT LT)
- NRR user need requested RES qualify CONTAIN for DBA confirmatory analyses of operating plants
- RES has completed this work, documented in a series of reports

CONTAIN Code Qualification Report/User Guides for Auditing Design Basis Calculation;

- PWR contaiments; large drys; ice condensers; subatmospherics
- BWR containments; Mark I, II, IIIs
- Subcompartment analysis

IMPROVED CONTAINMENT T/H (cont.)

- "Transitioning" to Audit Existing Plants Using CONTAIN
  - Consulting with NRR
  - Currently developing NRC-wide training course; December 2000

Severe Accident Code Improvements and Consolidation

- All major NRC severe accident codes have been peer reviewed and subsequent improvements made to address major comments (MELCOR, SCDAP/RELAP, CONTAIN, VICTORIA)
- At this point code improvements are linked to specific applications and long term goals, including consolidation
- Recent major code updates/releases
  - MELCOR 1.8.5 released May 2000
  - SCDAP/RELAP5 3.3 released Sept 2000

SEVERE ACCIDENT RESEARCH
SEVERE ACCIDENT CODE CONSOLIDATION

- Evolution of regulatory and research environments makes it appropriate to consolidate severe accident analysis code capabilities
  - More mature understanding of phenomenological issues
  - Improved computing technology
  - Confluence of modeling between integrated and more detailed codes
  - Need for resource efficiency
  - Greater demands of best estimate predictive capability and uncertainty quantification
  - More emphasis on integrated analysis - through level 3.

SEVERE ACCIDENT RESEARCH
SEVERE ACCIDENT CODE CONSOLIDATION (cont.)

- Targeted application for consolidated code
  - Regulatory application decision making, and risk informed regulation
    - Licensing issues (e.g. SG tube integrity)
    - Risk informing 10CFR50
  - Level 2 & 3 PRA insights for safety goal comparison
  - Analysis of experimental data
  - Non reactor applications
  - Regulatory counterpart to MAAP

SEVERE ACCIDENT CODE CONSOLIDATION (cont.)

- Capability of consolidated code should generally be comparable to capabilities of more detailed codes,
- MELCOR identified as natural platform for building consolidated code since as integrated code it has already incorporated many of the features of more detailed codes.

- One of the steps to consolidation is the assessment of MELCOR's parity with more detailed codes

- Assessment of state of parity with other severe accident codes in progress or planned

  - CONTAIN - containment phenomena (completed)
    - comparative analyses against test problems
  - SCDAP/RELAP5- core and in-vessel degradation (upcoming)
    - RCS natural circulation, TMI-like core melt progression, plant sequence comparisons

SEVERE ACCIDENT CODE CONSOLIDATION (cont.)

- VICTORIA - fission product chemistry and transport (planned)
  - fission product speciation, deposition in steam generator secondary, experiments

Parity Assessment Matrix

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	CONTAIN/Assessment Comparison	MELCOR/Assessment Comparison	MELCOR/CONTAIN Comparison
LACE-LA-4							✔	✔	✔						2	2	2
PNL Ice Condenser													✔		2	2	2
NUPEC M-4-3			✔				✔								3	2	2
NUPEC-M-5-5							✔							✔	2	2	3
NUPEC-M-7-1	35		✔				✔							✔	3	2	2
NUPEC-M-8-1			✔				✔								3	2	2
NUPEC-M-8-2			✔				✔							✔	2	2	2
HDR V-44	16		✔			✔	✔								2	2	2
VANAM-M3	37						✔	✔	✔						2	3	3
SNL/IET 1-7				✔	✔	✔									2	2	3
SNL/IET9-11 (Surtsey)				✔	✔	✔									2	2	2
RTF 0.2 Iodine Test	41										✔				N/A	3	N/A
NTS H_2 Burn Tests				✔		✔	✔								2	2	2
GE Supp. Pool Tests						✔						✔			2	2	2
CSE A-9 Spray Scrub								✔		✔				✔	N/A	N/A	3
Grand Gulf LBLOCA						✔						✔			N/A	N/A	3
SURRY CCI		✔				✔									N/A	N/A	3

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Conclusions from Study

- *Overall MELCOR and CONTAIN models perform very similarly*
- MELCOR User-Specification of intra-cell phenomena could be improved
- Hybrid flow solver not considered needed in MELCOR
- MELCOR hygroscopic aerosol model improved for mixed aerosol
- MELCOR might benefit from an impaction model for aerosol removal models
- Spray scrubbing appears reasonable - additional assessment against new data recommended
- MELCOR includes an iodine chemistry model not present in CONTAIN
- MELCOR's DCH models inferior to CONTAIN - cost/benefit ?
- *Often more variability observed between different users of the same code than between MELCOR and CONTAIN*
- *MELCOR models appear not to be lacking in any important ways from those of CONTAIN*

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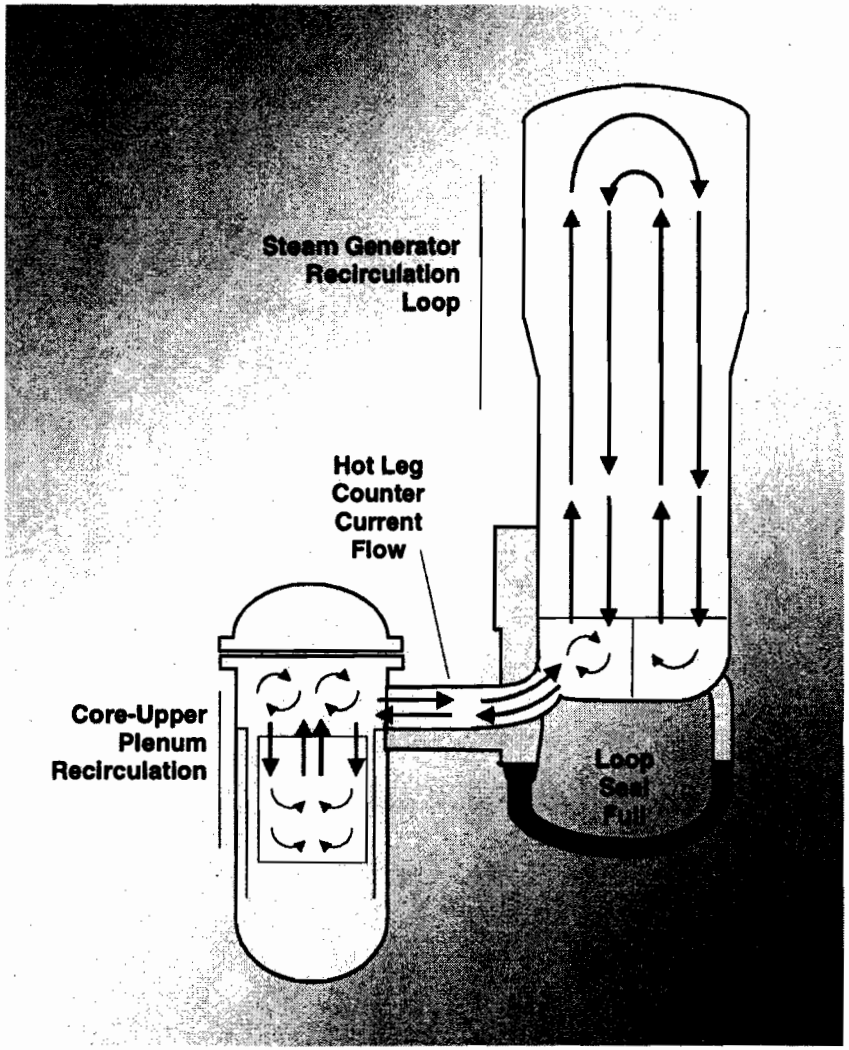
Consolidation of In-Vessel Phenomena in MELCOR : RCS Thermal Hydraulics



- Natural circulation effects are important at high system pressure
- 3 Principal circulation loops implied by Westinghouse experiments
- Core heatup delayed (*relative to neglecting natural circulation*)
- Hydrogen generation delayed
- Fission product retention in RCS enhanced
- Hot leg / surge line failure likely first RCS failure location

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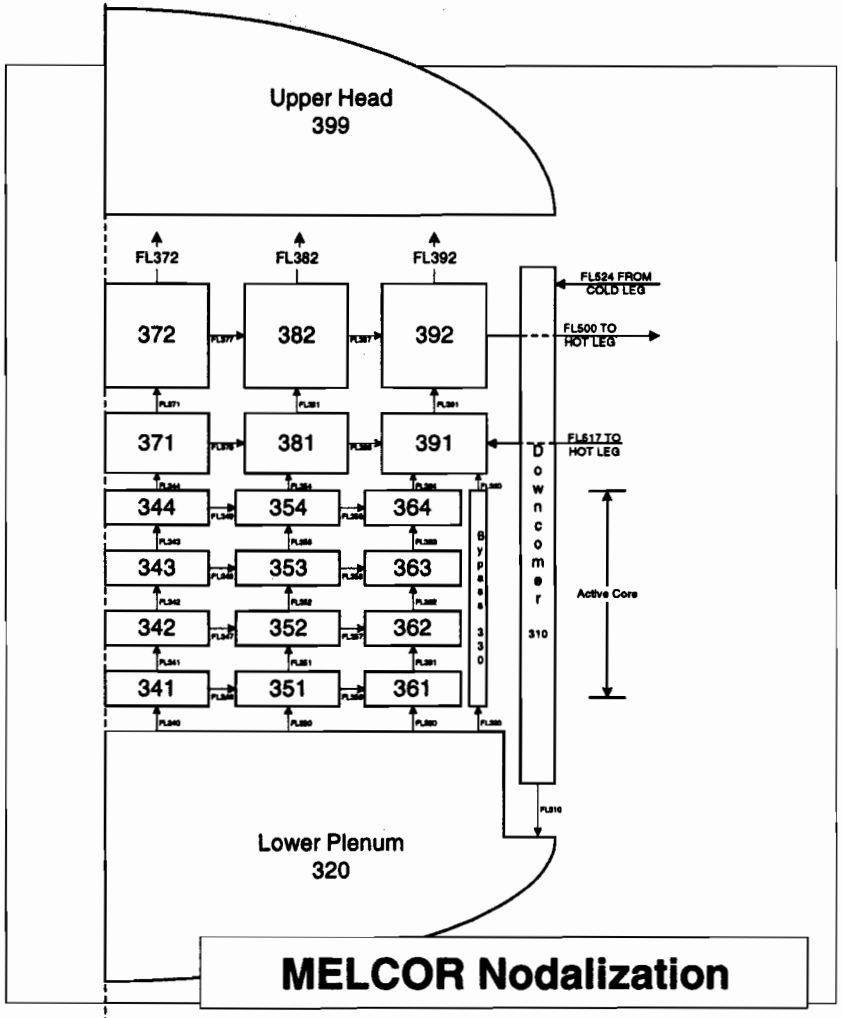
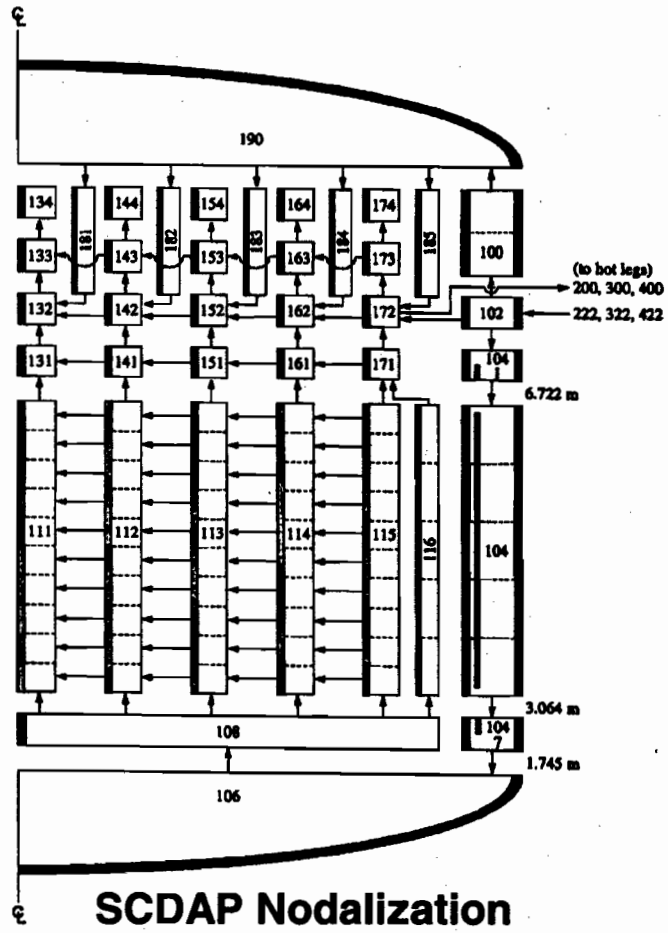
MELCOR Vessel Nodalization for Recirculation Analyses

Current Progress



MELCOR Nodalization



SCDAP Nodalization

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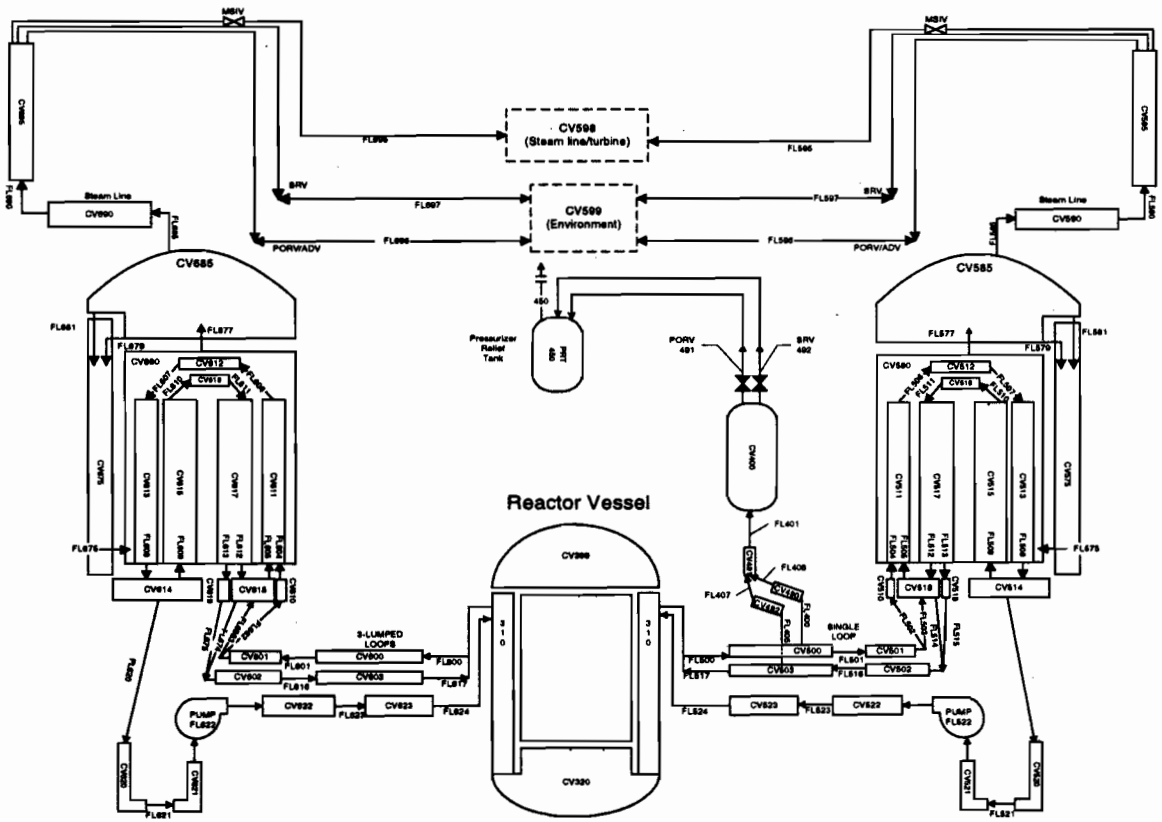
MELCOR RCS Nodalization for Recirculation Analyses

Current Progress



- Hot leg and surge line "split"
- One nodalization treats entire analysis
  - boildown
  - transition to counter current natural circulation
  - RCS breach of hot leg or surge line
  - long tern accident progression

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Consolidation of In-Vessel Phenomena in MELCOR : RCS Thermal Hydraulics

- MELCOR RCS Natural circulation models to be assessed against Westinghouse 1/7th scale experiments
- MELCOR Plant analyses with detailed nodalization will be compared against comparable SCDAP/RELAP-5 analyses
- TMI-2 Analysis including core-upper plenum natural circulation effects

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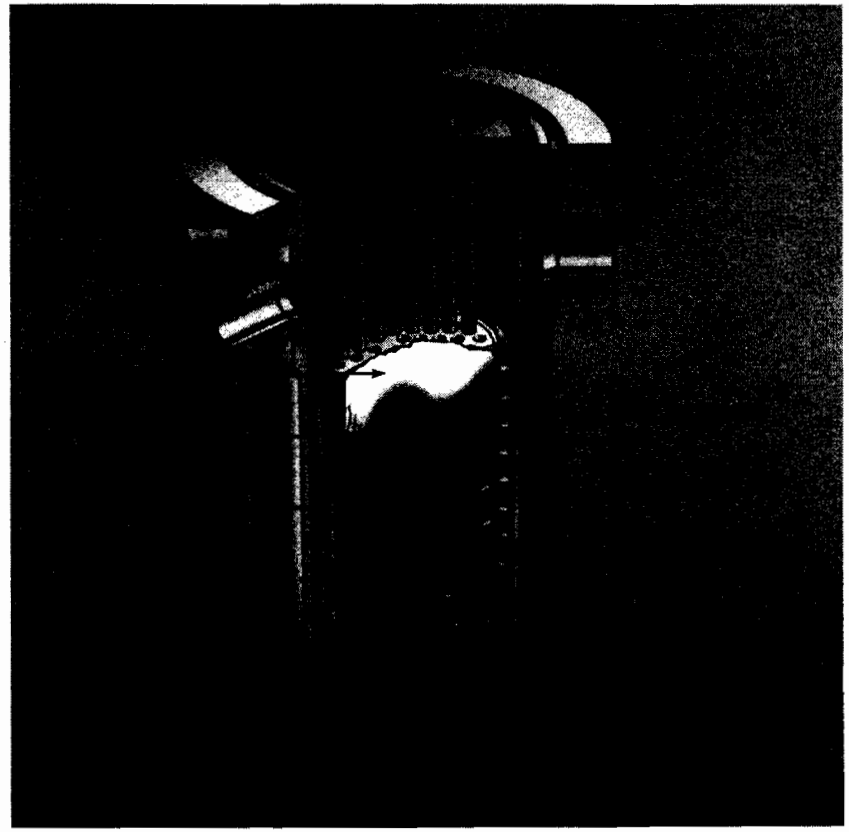
Consolidation of In-Vessel Phenomena in MELCOR : Core Melt Progression



- Improve COR models for
  - in-core molten pool formation
  - release of molten pool materials to lower head
  - lower head pool behavior
    - natural convection
    - vessel head loading
- Assessment of parity with SCDAP/RELAP-5

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Consolidation of In-Vessel Phenomena in MELCOR : Fission Product Release and Transport

- VICTORIA assessment planned for out years
  - fission product release models
    - CORSOR, Booth, degraded fuel, new data (VERCOR)
  - fission product speciation
    - CsOH, CsI, Cs-molybdate ? (Phebus)
  - transport and retention models
    - (retention in steam generator secondary for tube faults) (PSI Artist Experiments)
- MOX and high burnup issues (Cabri Tests)
- Future reactor designs

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Severe Accident Research
OECD MASCA Program

- Investigate the influence of chemical behavior on heat transfer pools in the lower RPV head under severe accident conditions
- Investigate Fission Product behavior in molten pool.
  - Partitioning of FP between layers in case of stratification.
  - Partitioning of FP between phases during melting and solidification.
- cooperative project under OECD/NEA sponsorship with participation of US and 16 other countries (Russian Research Center Kurchatov Institute)
- started July 2000 and will be completed in June 2003

Project Activities

- Experimental investigation of prototypic core material
  - Small to medium scale (0.5-15kg) tests
  - Material properties
  - Large scale (150 Kg), need for this test will be decided in December 2001
- Experiments with simulant fluids (salt)

Severe Accident Research
OECD Lower Head Failure Program

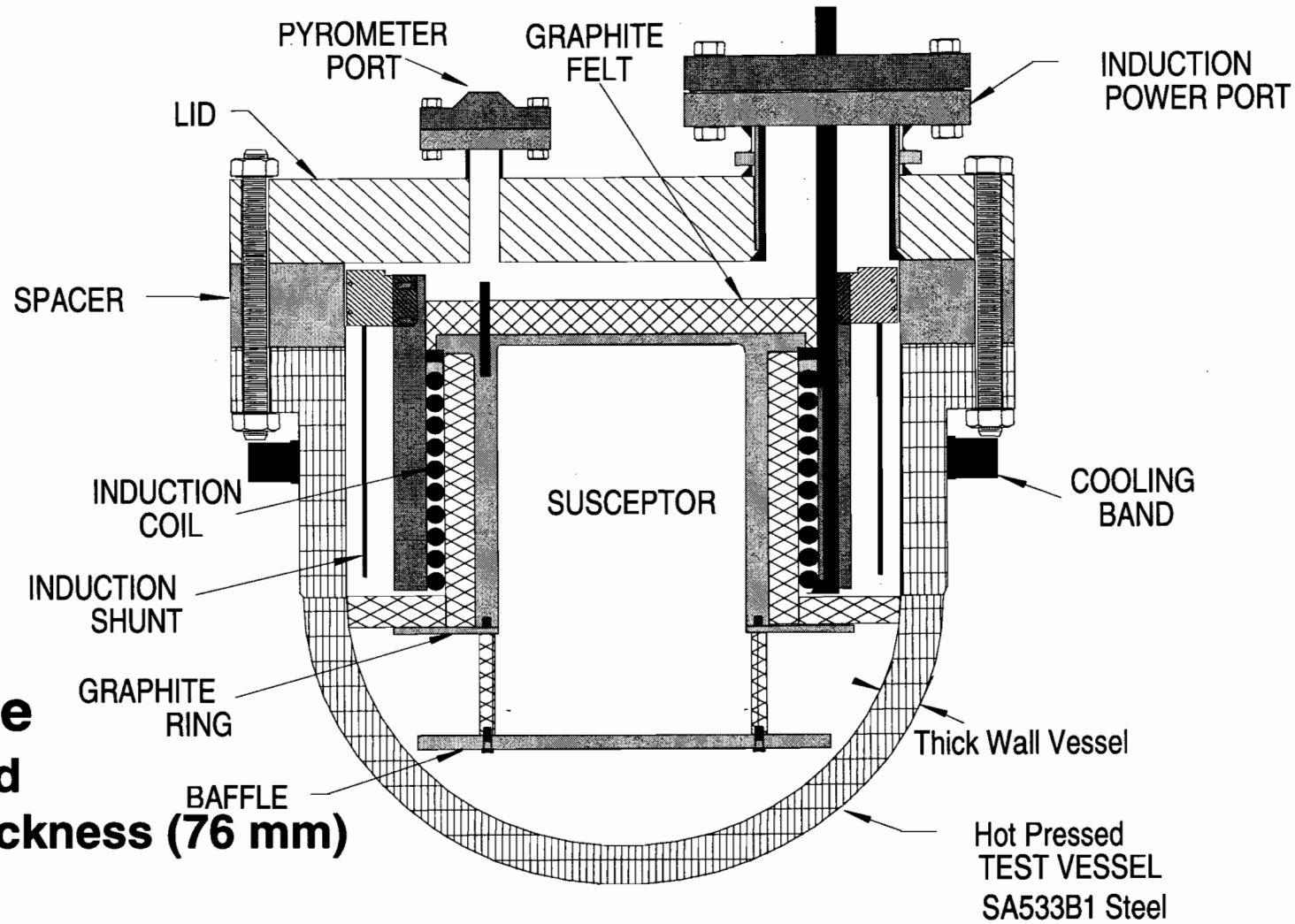
Objective of the OLHF Program

- The objectives of this project are to characterize the mode, timing and size of reactor pressure vessel (RPV) lower head failure for conditions of
  - low reactor coolant system pressure with
  - large differential temperatures across the lower head wall, and
  - for pressure transients

Key Program Elements

- Vessel Failure Experiments
  - Design and construction of experimental apparatus
  - 5 lower head failure experiments
- Material testing
- Test analysis and Interpretation
  - constitutive model development
  - FEM simulation of experiment
- Status
  - 2 tests run
  - Program completion Dec 2001

Test Vessel Assembly



~ 1/5 scale

1. **Distorted Wall Thickness (76 mm)**
2. **Large q**
3. **No wall insulation**

Proposed Test Matrix for OLHF Tests

- **Design goals: (1) ΔT_w>200 K – Stress Redistribution**
  (2) Partial Depressurization: 2 to 5 MPa

Two *Baseline* Tests: OLHF-1 and OLHF-2

Test	Heat Flux	Pressure	Features	Issues
OLHF-1	Uniform	5 MPa	None	Tie Back to USNRC LHF-7 Test Direct Assessment of the Effects of Stress redistribution (Due to Large ΔT_w)
OLHF-2	Uniform	2 MPa	None	Accident Mgmt Depressurization Possibly post-test Benchmark

Three *Issue Specific* Tests: OLHF-3 to OLHF-5 (not finalized)

Test	Heat Flux	Pressure	Features	Issues
OLHF-3	Uniform	2 MPa + P Transient	None	Accident Management Scenario with Repressurization Transient
OLHF-4	Uniform	2 MPa	Penetration	Global Deformation vs. Local Weld Failure
OLHF-5	Side-peaked	2 MPa + P Transient	Vessel Ablation	Vessel Erosion due to Melt Convection

OLHF-1 and OLHF-2 Baseline Tests

OLHF-1

- $\Delta T_w \sim 200$ K
- RCS Pressure: 5 MPa
  Test Pressure: 12 MPa
- Uniform Heat Flux

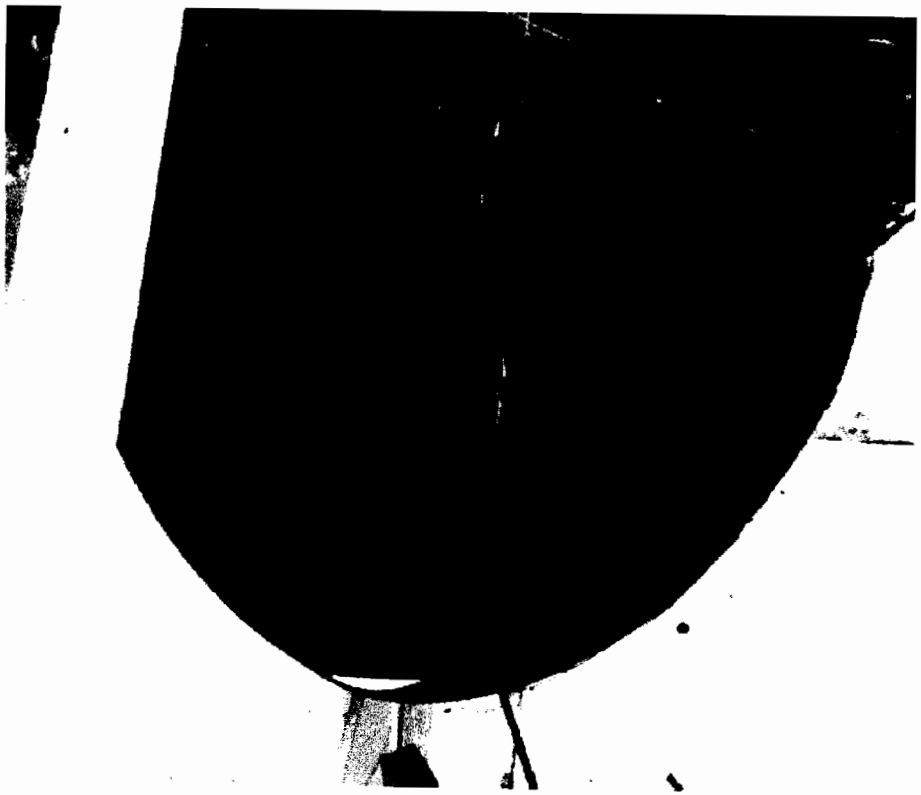
OLHF-2

- $\Delta T_w \sim 200$ K
- RCS Pressure: 2 MPa
  Test Pressure: 4.8 MPa
- Uniform Heat Flux

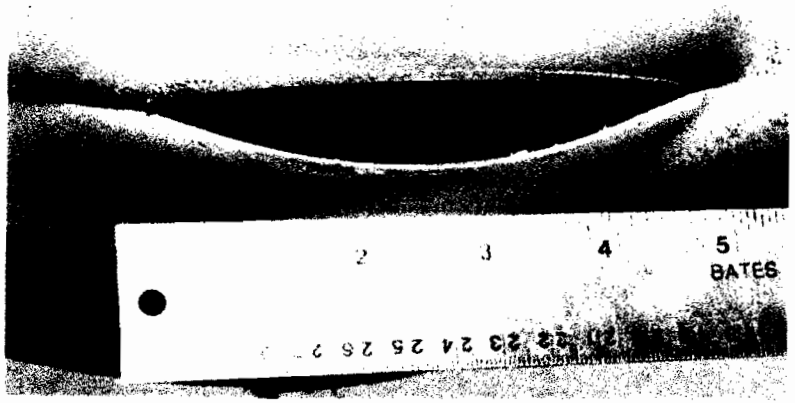
Justifications

1. **2 MPa is an expected end-state for depressurization**
2. **5 MPa test will provide tieback to USNRC LHF-7**
   - can assess the effect of stress redistribution
3. **5 MPa test as OLHF-1 because it is less challenging**

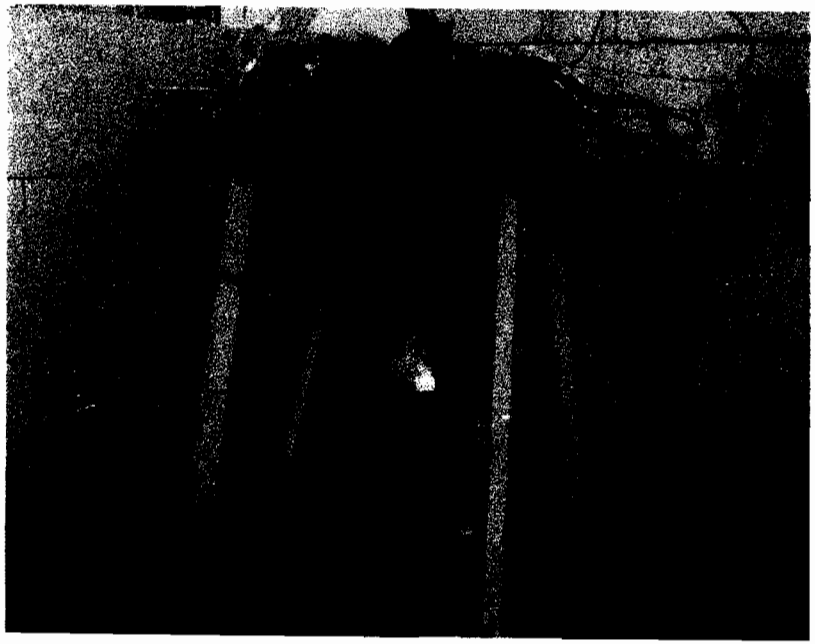
OLHF-1 Test Vessel and Rupture



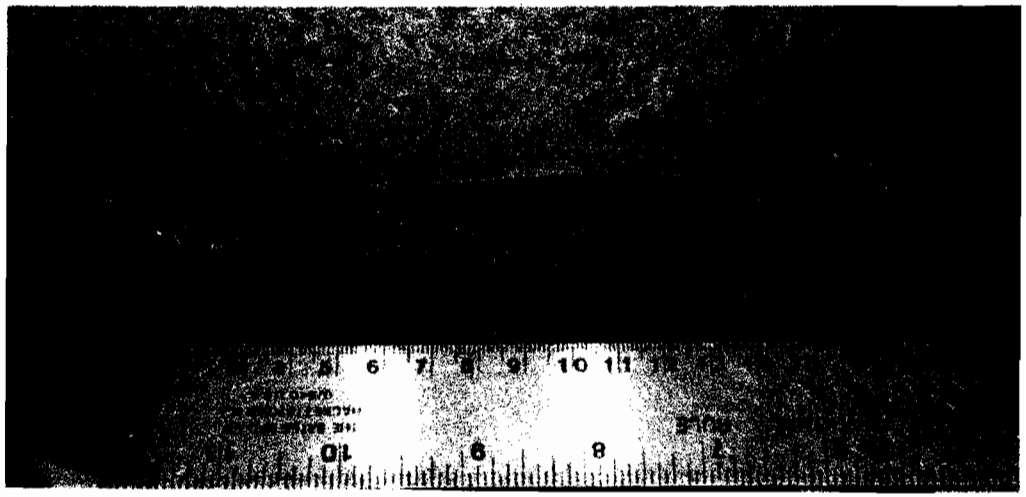
- Latitudinal rip, 14 cm × 2.1 cm
- FES ~ 0.22 m
- Vessel deformation 14.6 cm
- Failure condition:
  T_{int}~1500K, ΔT_W > 350K



OLHF-2 Test Vessel and Rupture



- Latitudinal rip, 18 cm × 3 cm
- FES ~ 0.32 m
- Vessel deformation 11.2 cm
- Failure condition:
  T_{int}~1750K, ΔT_W ~ 500K



Ex-Vessel Debris Coolability
Melt Attack and Coolability Experiments (MACE)

Objective

- To investigate the potential for an overlying water pool to cool ex-vessel core debris interacting with concrete basemat in a reactor cavity or pedestal region of the containment; ex-vessel coolability is important in assuring a successful accident management strategy.

Current Status

- Four large scale integral MACE experiments (100 kg to 2000 kg) conducted using prototypic materials and internal heat generation rates.

- Initial high bulk cooling observed in all tests for a very short period, however, not prolonged enough to cool the debris.

- Stable crust formed at melt-water interface in all experiments and anchored on sidewalls thereby limiting coolability.

Ex-Vessel Debris Coolability
Melt Attack and Coolability Experiments (MACE) (cont.)

- Analysis of MACE data led to identification of various debris cooling mechanisms: bulk cooling, melt eruptions or dispersal, water ingression, and crust breach.

- Need for separate effect tests recognized to investigate cooling mechanisms in detail. Two MACE separate effect tests (MSET) planned in current program; preparation for the first test (melt eruption test) in progress.

Ex-Vessel Debris Coolability
Melt Attack and Coolability Experiments (MACE)

Future Plan

- An ANL OECD/MACE proposal for MACE extension was presented at Severe Accident Technical Experts Meeting held at NEA in March 2000, a revised proposal was presented to NEA in September 2000.

- The proposed program - International Accident Management Analysis and Testing (IAMAT) Program - under the OECD sponsorship in intended to:

  Provide evidence and test data for coolability mechanisms identified in integral tests
  Respond to specific SAM needs in relation to current and future generation reactors

Ex-Vessel Debris Coolability
Melt Attack and Coolability Experiments (MACE) (cont.)

- For debris coolability mechanisms, two types of separate effect tests are proposed: melt eruption tests with inert basemat and gas sparging, and water ingression tests with inert basemat and no gas spraging.

- Also simulant experiments are proposed to investigate crust breach phenomena at large scale.

PHEBUS-FP Project (Research Centre of Cadarache, France)

Objective:

To obtain integral experimental data on severe accident fission product release, transport, deposition, resuspension, and chemistry in light water reactors for validation of models used for source terms

Approach:

- Perform integral effects experiments in an in-pile test facility located at the Research Centre of Cadarache, France
- Tests are to be conducted under sufficiently prototypical conditions, on the processes governing the transport, retention, and chemistry of fission products under light water reactor severe accident conditions
- Comparison of data to predictions of mechanistic models

PHEBUS-FP Project Status

PHEBUS-FP Test Matrix:

Test	Objective
FPT-0	Degradation and fission product release from fresh fuel
FPT-1	Same as FPT-0 but with pre-irradiated fuel (23GWD/tU)
FPT-2	Same as FPT-1, Reducing environment (steam starved) and boric acid
FPT-3	Effect of B_4C control rod on source term
FPT-4	Late phase core configuration, rubble bed
FPT-5	Air ingress

PHEBUS-FP Project Status (cont.)

- RES has transmitted a Research Information Letter (RIL) to NRR recently (8/21/00) summarizing results obtained so far.

- Highlights from the RIL (8/21/00):

  - Cofirmation of the NRC's Revised Source Term

    - Iodine is released predominantly as an aerosol with allowance for a small fraction (<5%) in gaseous form.
    - Low release of refractory fission products (e.g., ruthenium, plutonium).
    - Both radionuclides deposition in the reactor coolant system (RCS) and revaporization from the RCS to the containment have to be considered.
    - Cesium (Cs) was not transported as CsOH, but as compound of cesium and molybdenum (Cs-Mo). Prediction of correct chemical form of Cs is important in determining the pH of the containment sump.

PHEBUS-FP Project Status (cont.)

- NRC continuing its participation in the PHEBUS-FP tests and analysis. Plan to perform analysis of FPT-1 and FPT-4 with NRC codes

- NRC will also assist IPSN in planning for the FPT-5 (air-ingress) test.

- Post Test Analysis of Debris Bed Test (FPT-4) commences in October 2000, and experimental results from laboratories are expected in September 2001.

MOX and High Burnup Fuel

- Severe accidents involving MOX and High Burnup Fuel can potentially impact fission product source term releases.

- Effects of fuel on core melt progression not believed to be significant based on minimal effect on thermal properties.

- Higher burnup fuel fission product inventories can be adequately accounted for (e.g., ORIGEN analysis)

  - Uncertainty relates to FP release rates and release fractions.
  - Current source term regulations (revised source term) applicable to average burnups of ~40GWd/Mt
  - Effect on volatiles release fraction is limited by already high release fractions assumed for existing burnup

MOX and High Burnup Fuel (cont.)

- MOX fuel impact on fission product source term is primarily the change in fission product isotopic spectra (i.e., greater inventory of certain actinides)
  - Minor effects on gap releases and volatile release
- Effect of shift in fission product inventory should largely be influenced by magnitude of release fractions of those low volatile actinides. Variability in release fractions of low volatiles may be greater than change in inventory.
- Data is needed to resolve fission product source term issues on high burnup and MOX fuel.

MOX and High Burnup Fuel (cont.)

- Current plan is to assess available data on impacts of high burnup and MOX fuel on fission product source term
  - VERCORS (FRANCE)
  - VEGA (JAPAN)
  - PHEBUS
- Effect of high burnup and MOX fuel to be addressed by PIRT panel in early 2001
- Confirmatory work on integral core melt and fission product release
- Additional data is needed on release of fission products due to core degradation in air environments
  - Integral and separate effects data

MAAP Code Review

- Reliance on MAAP code by industry for wide array of analysis
  - Design basis containment analysis
  - Level 2 PRA
  - Risk impacts, e.g., SG tube integrity
- Past review of MAAP 3.0B completed in 1992
  - Issues raised over use for T/H success criteria
  - Additional sensitivities recommended reflecting uncertainty over core melt progression and certain ex-vessel phenomena (e.g.,CCI, DCH)
- Proposed initiative to undertake review of MAAP 4.0

MAAP Code Review (cont.)

- Scope of review yet to be agreed
  - 1st meeting scheduled for December 15, 2000
  - Targeted applications
  - Comparative analyses with NRC codes (MELCOR, SCDAP/RELAP5)

Severe Accident Research Program
Alignment with Strategic Goals

Maintain Safety:
- Application to SGTR
- Spent Fuel Pool
- Dry Cask

Effectiveness & Efficiency:
- Participation in International programs
- MELCOR code consolidation

Reduce Unnecessary Conservatism:
- Support RI Part 50
- Revised Source Term (RADTRAD)

Future Activities
Planned & Potential

- SG tube integrity
- Risk informing Part 50
- Dry cask PRA
- Improved dose analysis (RADTRAD)
  - reactors
  - dry cask
- Code consolidation
- MOX / High burnup
- International collaboration

Future Activities
Planned & Potential (cont.)

- MAAP code review
- Rebaselining generic risk analysis (Level 2)
  - accident management
- Advanced reactors