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Eighth Water Reactor Safety Research Information Meeting

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 27-31, 1980

U.S. Nuclear Regulatory Commission

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Office of Nuclear Regulatory Research
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Washington, D.C. 20555

FINAL AGENDA

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

AT THE

NATIONAL BUREAU OF STANDARDS
ADMINISTRATION BUILDING 101
GAITHERSBURG, MARYLAND

October 27-31, 1980

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Chairman: R. R. Sherry, NRC

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Chairman: L. H. Sullivan, NRC

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Chairman, W. S. Farmer, NRC

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Chairman: W. S. Farmer, NRC

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WEDNESDAY, OCTOBER 29, 1980

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Seismic Safety Margins Research Program

Chairman: C. W. Burger, NRC

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3:30 pm -	Component Fragility	M. Bohn, LLL
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THURSDAY, OCTOBER 30, 1980
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Chairman: R. F. Abbey, NRC

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Chairman: J. Harbour, NRC

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Irradiation Effects and Neutron Dosimetry

Chairman: C. Z. Serpan, NRC

10:30 am -	LWR Dosimetry Improvement Program Overview	W. N. McElroy, HEDL
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THURSDAY, OCTOBER 30, 1980
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Fracture Mechanics

Chairman: M. Vagins, NRC

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12:30 pm -	Crack Stability Analysis for Vessel Tests in the Upper Shelf Temperature Range	J. Merkle, ORNL
2:00 pm -	Validation of "Key Curve" Analysis of Elastic-Plastic Fracture Toughness	J. Joyce, USNA
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4:20 pm -	Crack Arrest Methodology and and Standard Test Methods for RPV Evaluations	G. Irwin/W. Fourney, UMD
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FRIDAY, OCTOBER 31, 1980
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Chairman: M. Vagins, NRC

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FRIDAY, OCTOBER 31, 1980
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Chairman: J. Muscara, NRC

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Chairman: J. Muscara, NRC

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FRIDAY, OCTOBER 31, 1980

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12:00 pm	- Evaluation of Dynamic Testing of NPP Structures	C. A. Kot, M. G. Srinivasan, ANL

STRUCTURAL ENGINEERING RESEARCH BRANCH

Chairman: G. Bagchi, NRC

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MECHANICAL ENGINEERING RESEARCH PROGRAM

Load Combinations

Chairman: J. O'Brien, NRC

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9:45 am	- Probabilistic Model and Computational Procedure	L. L. George, LLNL
10:20 am	- Fracture Mechanics Evaluation	R. D. Streit, LLNL
10:50 am	- Results and Conclusion	S. C. Lu, LLNL

FRIDAY, OCTOBER 31, 1980

MECHANICAL ENGINEERING RESEARCH PROGRAMS
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Chairman: D. Reiff, NRC

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3:15 pm -	Snubber Research and Testing	A. Onesto, ETEC
3:45 pm -	Experimental and Preliminary Analytical Results of Coupled Fluid-Structure Interactions During Blowdown of the HDR Vessel	U. Schumann, KfK (FRG)
4:10 pm -	Earthquake Simulation Experiments Performed at the HDR Facility	G. Katzemeier, KfK (FRG)
4:30 pm -	HDR Dynamic Tests of Late 1979	G. Howard, ANCO
5:00 pm -	Predictions of Recirculation Loop Response at HDR to Simulated Blast Excitation	R. Guenzler, INEL

HIGHLIGHTS OF WRSR ACHIEVEMENTS IN FY-1980 AND STATUS OF LOCA SAFETY EVALUATION

L. S. Tong

To be presented at the 8th WRSR Information Meeting

A. HIGHLIGHTS OF WRSR ACHIEVEMENTS IN FY-80

1. Instrumentation

   - PNA - to measure low-flow rate in natural circulation during SBLOCA
   - Heated T/C - to measure liquid level in reactor vessel

2. Code Development

   - TRAC/PD2 - published for multi-dimensional system analysis
   - RELAP 5 - published for one-dimensional system analysis
   - COBRA/TRAC - developed for detailed vessel T-H analysis with system feedback

3. Fuel Research

   - New clad oxidation embrittlement criteria established at ANL
   - Multi-rod burst effects on core cooling measured by flow resistance in 3 swelled rod bundles at ORNL
   - Fuel code to reliably calculate fuel rod gap conductance and stored heat
   - PBF/PCM-7 test has shown that DNB does not propagate in a rod bundle

4. LOCA Research

   - THTF - Core boiloff and recovery heat transfer data were obtained at high pressures during a small-break LOCA
   - CCTF - Upper plenum liquid de-entrainment data obtained for reducing steam binding
   - LOFT - Natural circulation data show adequate core cooling and smooth transition from single-phase to two-phase natural circulation. An advance diagnostic display system was operated under typical accident conditions during small-break tests.
   - Semiscale - Natural circulation data for core cooling was obtained and a two-phase mixture vortex was observed at the break during a SBLOCA; effects of pump on/off have been tested.
   - SSTF - Completed the test of BWR spray into steam environment with no up-drift.
   - TLTA - Completed two BWR small-break experiments

5. Operational Safety Research

- Halon tested successfully as a fire suppression agent
- Requirement for use of simulator in operator training and requalification established

6. Severe Accident Analysis

- Hydrogen handbook draft issued
- Steam explosion experiment conducted to evaluate the energy conversion ratio of the explosion
- Station blackout calculations performed

B. CURRENT STATUS OF LOCA SAFETY EVALUATION

1. Confirmation of Evaluation Models of ECCS (Appendix K)

- Decay heat is 7% less than ANS(1973) - ORNL, LASL, OSU
- Zr-Oxidation rate is 3/4 of Baker-Just-ORNL
- Rewet occurs at blowdown - LOFT, THTF, Semiscale
- Reflood heat transfer measured at rate>1 in/sec - FLECHT
- Rod swelling measured by flow resistance - ORNL

2. ECCS Effectiveness in Large-Break LOCA

- Current ECCS is adequate for core cooling - LOFT, Semiscale, TLTA
- Alternate ECCS could be even better - Semiscale, (CCTF, UPTF)
- Correlation ECC bypass during blowdown has been developed - BCL, CREARE, (CCTF, UPTF)
- Steam binding minimized by de-entrainment in U.P. - CCTF, (UPTF)

3. Core Cooling in Small-Break LOCA

- Natural circulation is adequate for core cooling with active steam generator - Semiscale, LOFT, PKL
- Makeup/letdown is adequate for core cooling in very small break with inactive steam generator - Semiscale, LOFT)
- Non-condensible effect on natural circulation - PKL, (Semiscale)
- Reflux boiler cooling - PKL, LOFT, (Semiscale, UPTF)

Reference: Attached paper on "USNRC LOCA Research Program" by L. S. Tong presented at IAEA Meeting in Stockholm.

INTERNATIONAL ATOMIC ENERGY AGENCY

INTERNATIONAL CONFERENCE ON CURRENT NUCLEAR POWER PLANT SAFETY ISSUES

Stockholm, 20–24 October 1980

IAEA-CN-39/ 99

USNRC LOCA RESEARCH PROGRAM

L. S. TONG

U.S. NUCLEAR REGULATORY COMMISSION

WASHINGTON, D. C. 20555

U.S.A.

IAEA-CN-39/99

USNRC LOCA RESEARCH PROGRAM

L. S. Tong

Abstract

This paper describes the characteristics of large-break and small-break LOCAs. It also identifies the differences in emphasis of their research. The results of NRC research on LOCA are reported in detail. Current status of LOCA evaluation is that a good understanding of LOCA/ECCS behavior, for both large- and small-breaks has been acquired. Future research efforts are recommended with emphasis on exploring plant response to anomalous transients which could be associated with LOCA, and on minimizing human errors during reactor operations.

1. DESCRIPTION OF LOCA AND SCOPE OF RESEARCH

A Loss-of-Coolant Accident (LOCA) in water reactors results from a break in the pressure boundary of the reactor cooling system, where the water inventory is reduced and radioactive fission products may be released into the containment. LOCA is commonly classified as large-break or small-break LOCA:

A large-break LOCA (LBLOCA) is caused by a broken large pipe in the reactor cooling system, and initiates a fast blowdown during which the reactor is shut down by excessive void. The licensing Design Basis Accident of LOCA is defined as a sudden severence of a large diameter cold leg pipe in a pressurized water reactor or a recirculation jet-pump inlet pipe in a boiling water reactor. The reactor core is cooled by emergency core cooling systems (ECCS), which are automatically activated during a fast depressurization, in a matter of tens of seconds. After the core is quenched, the low-pressure long-term core cooling relies on the Decay Heat Removal system for any size break in either pressurized water reactor (PWR) or boiling water reactor (BWR).

A small-break LOCA (SBLOCA) is caused by a broken small pipe or a stuck-open safety relief valve in the reactor cooling system, and initiates a slow blowdown during which the heat initially stored in the core will be readily transferred to the coolant, however, the core decay heat may not be entirely removed by the break flow, such as the Three Mile Island Accident. The primary system pressures in small-break LOCAs of various break areas are calculated to last for hours, as shown in Figure 1. During this long lasting pressure hang-up if ECCS does not work properly or it combines with anomalous transients, such as loss of all feedwater or station blackout, a SBLOCA may result in a prolonged core uncovery. This situation makes operator action crucial to the course of the accident, and it makes time in the anomalous transient vital to plant recovery.

The detailed characteristics of LBLOCA and SBLOCA are listed and compared in Table 1.

To ensure the safety of nuclear power plants during a LOCA, the concept of defense-in-depth is adopted, in which the primary assurance of safety is accident prevention. Quality assurance is applied to achieve this goal. Nevertheless, the occurrence of LOCA is postulated and engineered safety features (ESF) are installed to mitigate the consequences. The elements of ESF include:

- Reactor Trip - to stop major heat generation,
- Emergency Core Cooling (ECC) - to cool the core during LOCA,
- Decay Heat Removal - for long-term core cooling,
- Containment Heat Removal - to cool the atmosphere and reduce pressure in containment, and
- Radioactive Removal by Spray - to collect the fission products suspended inside containment.

Emergency Core Cooling Systems (ECCS) are designed to meet the NRC acceptance criteria[1], which limit peak clad temperatures, maximum clad oxidation, and maximum hydrogen generation, and which require coolable fuel geometry and long-term cooling.

In evaluating the adequacy of a water reactor design for conformity to the acceptance criteria for ECCS, the basic thermal-hydraulic behavior of the Reactor Coolant System must be well understood. Therefore, after the ECCS Rule Making Hearing in 1972-1973, various confirmatory research programs[2,3,4] were established to better understand reactor behavior and to confirm ECCS performance during LOCA.

In evaluating the relative importance of PWR-LOCA versus BWR-LOCA for establishing research priorities, it was found that LOCA contributions to the risk in PWR is much higher than that in BWR, while in BWR, the transient without scram contributes significantly to the risk. This statement is

FIGURE 1.

REACTOR COOLANT SYSTEM PRESSURE

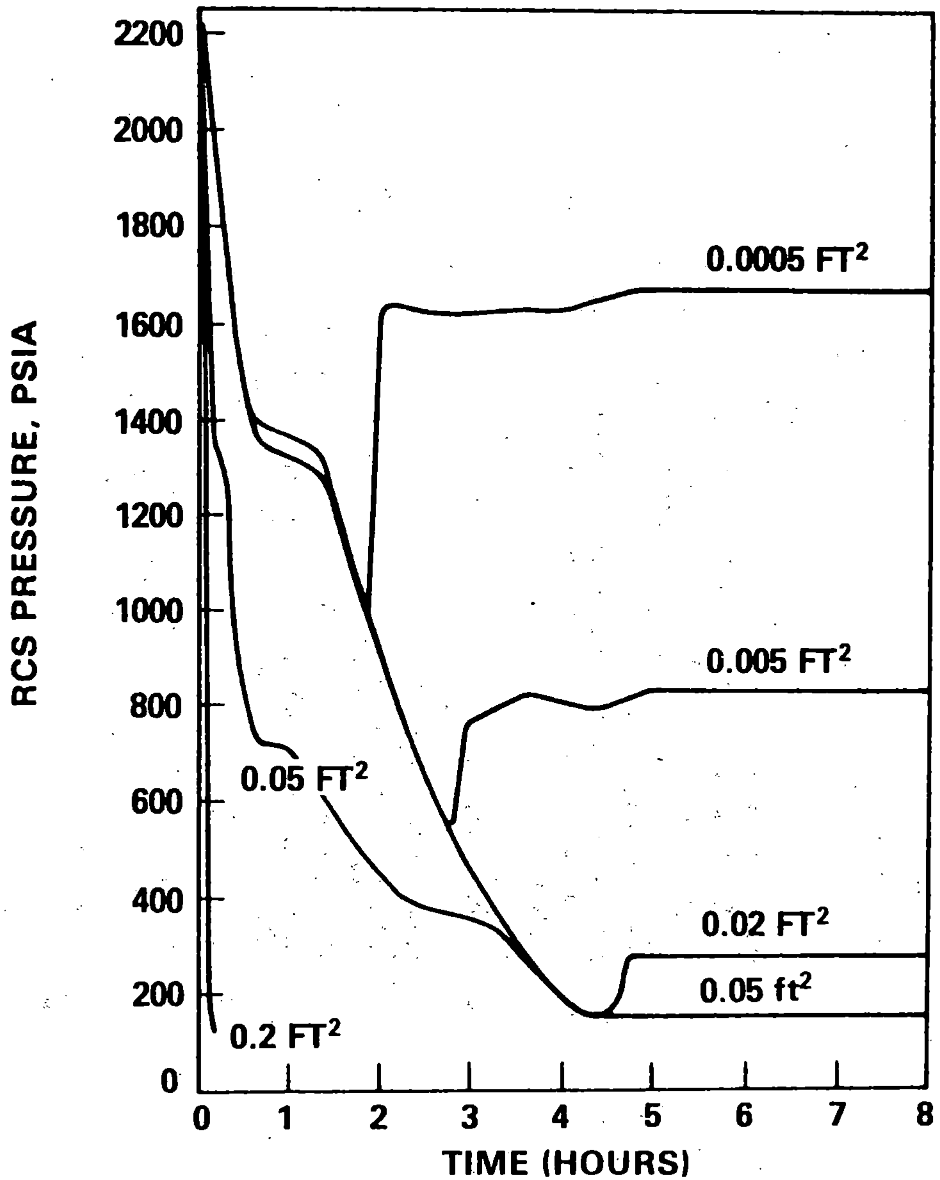


TABLE I.

COMPARISON OF SMALL BREAK AND LARGE BREAK LOCA CHARACTERISTICS IN A PWR

	Small Break LOCA	Large Break LOCA
Sample break size	0.02 ft^2	2 x 4.0 ft^2
Significant Heat Source	Decay heat (Stored heat only in early stage)	Stored and decay heat
Significant Heat Sink	Break Flow, Heat Transfer thru S. G. to Secondary side, and ECC water	Break flow and ECC water
Heat Transfer in Steam Generator (S. G.)	$P_{pri} \geqslant P_{sec}$ Auxiliary Feed Water (AFW) Significant	P_{sec} P_{pri} Auxiliary Feed Water (AFW) Insignificant
Primary Side Pressure	High pressure maintained because of slow draining	Fast depressurization by blowdown
Flow Behavior in Primary Side	1. Stratified flow 2. Separation of non-condensibles at high spot 3. Gravitational force control 4. Core may uncover by flashing and draining 5. Pressurizer effect significant	1. Bubbly or droplets dispersed flow 2. Homogeneous flow during blowdown 3. Momentum control 4. Core emptied and recovered quickly 5. Pressurizer has small effect
ECCS	1. Charging pump and HPSI 2. Effectiveness depends on the pressure for initiation of injection 3. In cold leg break LOCA, core may have to be partially uncovered to vent steam thru loop seal.	1. Accumulator most effective 2. Effectiveness depends on the initiation pressure and location of injection 3. In cold leg break LOCA, there may be steam binding and ECC bypass which slows down reflooding.
Plant recovery	1. AFW and natural circulation for wet S.G. 2. Manual opening all PORVs to lower the pressure for HPSI, Accumulator, LPSI and RHR when steam dump is not available	1. Accumulator and Reflooding 2. Continuous LPSI or RHR

based on the NRC Reactor Safety Study[2] in which the event sequences in PWR that contributed most to public risk are identified to be a break of pressure boundary caused by check valve rupture and a small-break LOCA with failure of containment spray heat removal, and the high risk event sequence in BWR is identified as a transient with failure to shut down reactor.

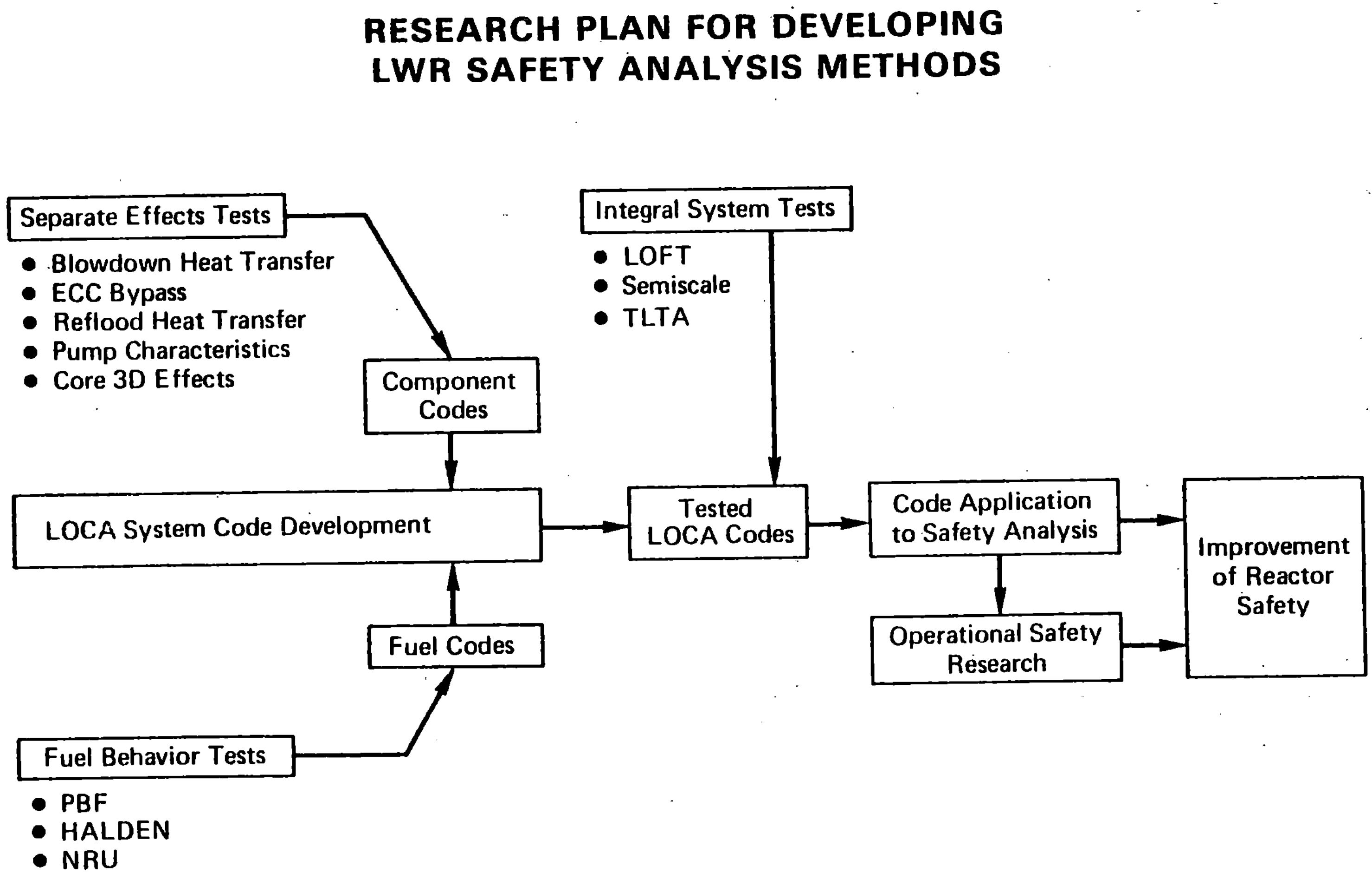
After the Three Mile Island accident, additional emphasis was given to the operational transients associated with LOCA, and also to the exploratory research for plant recovery from accidents with multiple failures and for design improvement.

The current LOCA research program elements include:

- PWR and BWR integral system effect tests, including reactor coolant system response and the balance-of-plant response,
- PWR and BWR thermal-hydraulic separate effects tests,
- Fuel behavior research,
- Computer codes development for analyzing the behavior of fuel, LWR systems and components, and
- Operational safety research for analyzing overall plant system behavior and operator actions during transients and LOCAs, and for enhancement of operator's performance and improvement of reactor safety.

The relationships between experimental research and analytical research are illustrated in Figure 2. Note that computer codes development is the backbone of the reactor safety research program because the computer codes provide the computational means of applying basic experimental and analytical information to commercial nuclear power plants. The research results of the LOCA research programs are described in the following sections.

FIGURE 2.


RESEARCH PLAN FOR DEVELOPING
LWR SAFETY ANALYSIS METHODS
Separate Effects Tests
• Blowdown Heat Transfer
• ECC Bypass
• Reflood Heat Transfer
• Pump Characteristics
• Core 3D Effects
Integral System Tests
• LOFT
• Semiscale
• TLTA
Component Codes
LOCA System Code Development
Tested LOCA Codes
Code Application to Safety Analysis
Improvement of Reactor Safety
Operational Safety Research
Fuel Codes
Fuel Behavior Tests
• PBF
• HALDEN
• NRU

2. LOCA RESEARCH RESULTS

The LOCA/ECCS research is being conducted in test facilities ranging in size from nearly full-scale to intermediate scale, small-scale, and bench scale. The research includes both integral system effects and separate effects, such as blowdown and reflood heat transfer, ECC bypass during blowdown and refill, pump and steam generator behavior, and upper plenum de-entrainment. The experimental program matrix for LOCA/ECCS research is shown in Table 2.

The nature of a large-break LOCA is quite different from that of a small-break LOCA. A comparison of scaling criteria for testing small-break and large-break LOCA in PWR is given in Table 3. It should be noted that the height is preserved in small-break LOCA testing facilities because of its importance in natural circulation.

2.1 Integral System Test Results

Integral systems tests measure the combined effects of various phenomena associated with small- and large-break LOCA, and provide data to validate LOCA codes.

Integral test facilities in the NRC LOCA program include two PWR facilities, the Loss of Fluid Test (LOFT) and Semiscale located at the Idaho National Engineering Laboratory, and a BWR facility, the Two-Loop Test Facility (TLTA) located at General Electric in California.

TABLE II.

LOCA ECC EXPERIMENTAL PROGRAM MATRIX

Model Size Linear Scale	Integral System	Separate Effects				
		Blowdown Heat Transfer	Reflood Heat Transfer	ECC Bypass Bldn & Refill	Pump and Steam Generator	Upper Plenum De-entrainment
Nearly Full Scale	Analysis	ORNL BDHT •CE/EPRI BDHT GE TLTA-1	Japanese SCTF W FLECHT-SEASET •PKL-340 rod bundle [BWR CCFL/Reflood]	FRG UPTF	•CE/EPRI 1/4 & 1/5 Scale	FRG UPTF Japan SCTF
Intermediate Scale	INEL LOFT Japan CCTF	INEL LOFT	INEL LOFT Japanese CCTF	INEL LOFT Japan CCTF	(CE LOFT Pump) INEL LOFT S.G.	Japan CCTF
Small Scale	INEL Semiscale •Japan ROSA GE TLTA-2	INEL Semiscale •ISPRA BDHT	INEL Semiscale	BCL 2/15 Model BCL 1/15 Model Creare 1/15 Model	(WCL Semiscale Pump) INEL Ser W FLECHT-SEASET S.G.	FLECHT SEASET Upper Plenum
Basic Model Study		ANL Freon Loop •(W tube BDHT) •(MIT Freon Loop) •(Hannover Freon) Lehigh Model Test	MIT Freon AECL Model Dev. RPI Model Test	DARTMOUTH Small Loop	•Creare/EPRI MPR Model	Harwell

( ) Completed Program. [ ] Proposed Program

• Non-NRC Funding

TABLE III.

SCALING CRITERIA FOR LOCA TESTING

Preservation of Physical Phenomena	Small Break LOCA	Large Break LOCA
Correct time of system pressure of flashing two-phase fluid	Power/volume = constant Break area/volume = constant	Power/volume = constant Break area/volume = constant
Correct time of relative pressure changes throughout the system	Flow length and resistance coefficient simulated	Volume distribution and resistance coefficient simulated
Reflood heat transfer	Core height maintained	Core height maintained
Natural circulation rate, fraction of core uncovery	Height of steam generator and loop geometry maintained	N/A
Reactor transient	Both primary and secondary sides of steam generator simulated. Secondary coolant temperature and level simulated.	N/A
ECC Water Bypass	N/A	Downcomer geometry maintained. Large size test preferred.
Steam bypass between downcomer and upper plenum	% bypass area should be approximately equal to large PWR for simulating loop seal clearing	% bypass area should be approximately equal to large PWR for simulating steam binding effect
Time constant of plant thermal transient	Wall heat transfer/volume = constant	Wall heat transfer/volume = constant
Time of core uncovery in S.B. and steam binding in L.B.	Correct upper plenum volume fraction and actual distance between the top of core to the top of exit nozzle nozzle maintained	Upper plenum and internals geometry and distance between the top of core to the top of exit nozzle maintained

2.1.1 Semiscale (PWR Facility)

Semiscale is a non-nuclear one-dimensional representation of a PWR facility. The first core (Mod 1) is 5.5 feet long and contains 40 electrical test rods. The second core (Mod 3) is 12 feet long and contains 25 electrically heated fuel pins. The primary coolant system has the same volume-to-power ratio as LOFT and commercial PWRs.

Findings of these Semiscale tests[5,6] are:

- Blowdown Heat Transfer - CHF was reached in 0.3-4.0 seconds in the Semiscale cold-leg LBLOCA which are comparable to delays observed in the ORNL BDHT runs.

- Reflood Heat Transfer - The data from the forced flooding tests were compared to the FLECHT data with good agreement, and the Mod 1 gravity flooding data were compared to the FLECHT-SET data, with some difference because of the differences in core length.

- Alternate ECC Injection Concepts - Lower plenum ECC injection tests showed substantially earlier core quench than the cold leg injection method (75 seconds versus 200 seconds). The combined upper plenum and cold leg injection tests also showed an earlier quench time. An improvement of performance of current ECCS seems feasible.

- Steam Generator Tube Rupture Tests - The test results showed that the potential for increased cladding temperature in tube rupture during a LOCA is largest for a range of ruptured tube numbers between 12 and 50. However, the computational methods currently used to analyze this transient appear to be overly conservative. The peak clad temperature observed in this test did not exceed 1366°K (2000°F) because of the core internal flow recirculation observed in the Semiscale tests.

- Small-Break Tests - Immediately after the TMI accident, ten tests were conducted at Semiscale in support of the recovery operation. Since then, tests have been run at Semiscale to show the cooling effectiveness of natural circulation in small-break LOCAs. The small-break tests for studying the effects of pump on or off at Semiscale provided visual indication of a vortex of steam-water mixture formed in the vicinity of a small break in a hot leg filled with stratified flow pattern. The small break was located either above or below the stratified liquid level. This phenomenon is analyzed in a NUREG report[7]. These small-break tests also revealed that the results are strongly effected by the heat leakage thru the walls of vessel and piping of the testing facilities.

2.1.2 LOFT (PWR FACILITY)

The LOFT facility uses an active nuclear core to simulate the transient behavior of a 1000 Mwe PWR under LOCA type conditions. The LOFT nuclear core is approximately 5.5 feet long and 2 feet in diameter, contains 1300 fuel pins and four control assemblies of typical PWR design. The ratio of primary coolant system volume to core power is similar to that of commercial PWRs. The unbroken PWR coolant loops are approximately simulated by the single unbroken circulating loop in the LOFT primary system, and the postulated broken PWR loop is simulated by the LOFT blowdown loop with passive components.

In 1976-1978 before loading nuclear fuel, 5 non-nuclear experiments were completed to provide operational experience and data to assess code capabilities prior to initiating nuclear tests. Then, two nuclear large-break LOCA experiments were conducted (L2-2 and L2-3) at 67% and 100% of equivalent PWR rated power. These two tests are unique and contribute greatly to the confidence in ECCS effectiveness and the ability of PWRs to stand the worst LBLOCA without fuel damage. Other major findings[8,9] are:

- The ECC water is delivered more quickly to the core, more reactor coolant remains in the core region and less ECC water flows from the break than is predicted by codes based on the conservative licensing ECC rule.

- Early in the accident, even before the ECCS is actuated, the core receives a flow of water which rewets the core and significantly lowers the temperature of the fuel during blowdown period.

- The hydraulic behavior of LOFT experiments is basically predictable by the "best-estimate" codes. No unexpected severe events were observed. This finding provides confidence in the current analytical methodology.

- The ECC systems work as expected. Peak cladding temperature in these LBLOCA tests was less than 980°K (1300°F).

After the TMI-2 accident, further large-break LOCA testing was postponed to permit testing small-break LOCAs and transients. The first small-break test involved a stuck-open power operated relief valve and demonstrated some of the events which occurred early in the TMI-2 accident. The second test, simulating a 4-inch break, caused a slow continuous depressurization and eventual activation of the ECC systems to refill the plant before core uncovery. The third test, simulating a 1-inch diameter break caused a very slow pressure reduction with stabilization at an intermediate value. Operator intervention then brought the pressure down sufficiently to actuate the ECC systems and the plant was recovered without uncovering the core. The steam generator appeared to transition from liquid natural circulation to liquid-vapor natural circulation, and possibly to reflux cooling (or condensate fallback) and then back again, with no evidence of instability. The fourth small-break test examined the effectiveness of various heat sinks available to PWRs. The results of these tests suggest that for larger small breaks (4" pipe and above), the break flow is sufficient to carry away all decay heat while for small breaks (1" pipe), the steam generator is the dominant heat sink and its pressure leads the primary system pressure very closely. It was also found that the bypass between the downcomer and upper plenum is an important factor in clearing the pump loop seal of water to allow a continuous steam flow path around the primary loop in a two-phase natural circulation. A bypass greater than 4% of total flow would stop this clearing action.

A computer-based diagnostic and display system has been in use in the LOFT control room during the past year and has been operated through several small breaks and transients. In addition, the reactor is serving as a test bed for instrumentation used to monitor accident conditions, for example, reactor vessel liquid level and the low flow rates during natural circulation.

2.1.3 TLTA (BWR Facility)

Studies of BWR LOCA behavior have been conducted in the Two Loop Test Apparatus (TLTA) at General Electric (GE) under the joint sponsorship of NRC, the Electric Power Research Institute, and GE, using a full-length 7 x 7 and 8 x 8 electrically simulated fuel bundle. Findings with the 7 x 7 bundle[10] include:

- The current licensing BWR LOCA evaluation method, when applied to the TLTA results, shows a substantial margin in the prediction of peak cladding temperature during blowdown.
- The system response was observed to be insensitive to large variations in the bundle power (3 to 6.5 MW).
- Boiling transition (BT or CHF) generally occurs after the lower plenum flashing surge due to rod uncovering as the two-phase mixture level is depleted. However, in the peak power bundle tests, boiling transition occurred due to exceeding the critical power during the non-typical TLTA core-flow coastdown while the mixture level remained above the bundle.
- The maximum measured cladding temperature was less than 1030°K (1400°F) for the peak power bundle tests.

Tests of an 8 x 8 bundle[11] show substantially lower temperatures than the 7 x 7 and intermittent rewetting attributed to the lower heat flux.

Recent tests (unpublished) investigated the influence of ECC injection during the blowdown on system response. Preliminary findings are:

- ECC injection results in significant bundle-cooling during the blowdown period.
- CCFL at the bundle inlet played a key role in preventing draining of the bundle to the lower plenum.

2.2 Separate Effects Test Results

Separate effects experiments have been designed to investigate key phenomena during blowdown, refill and reflood portions of LOCA under controlled boundary conditions. The major experiments and their interrelationship are shown in Table 2.

2.2.1 Heat Transfer During Blowdown and Reflood

The PWR blowdown test is being conducted at the Oak Ridge National Laboratory in the Thermal Hydraulic Test Facility (THTF)[12]. The THTF is a large non-nuclear pressurized-water loop incorporating a full-length bundle of 49 electrical rods that can be heated and cooled under conditions calculated to correspond to those in a nuclear power reactor.

Findings on transient CHF to date can be summarized as follows:

- Time to CHF (for peak power fuel regions) is generally found to occur at 0.4 to 1.0 seconds into blowdown in a cold-leg LBLOCA. This is collaborated by Semiscale findings.
- Unpowered rods delay the time to CHF in the adjacent powered rods, particularly in the upper half of the test bundle.
- Reduced bundle power and peak power density would delay CHF, and
- Steam heat transfer coefficients during boil-off in partially uncovered core were found to be 20 to 30 Btu/hr ft^2 °F at 400 to 1000 psia with exit steam temperatures around 930K (1200°F) and clad temperature around 1060°K (1450°F).
- Rewet after CHF during blowdown has been observed in LOFT, Semiscale, and THTF experiments. The transition boiling correlations can be used to predict rewet.
- Transient CHF has been tested for BWR on 7 x 7 and 8 x 8 geometries at TLTA, GE, and for PWR on 49-rod bundle at THTF, ORNL. Data show that modified void-CHF correlations[13,14] can be used in small-break LOCA. The steady-state CISE dryout correlation[15], when converted into average CHF form, is also applicable to transient conditions.

Effort is also being expended to validate transition boiling correlations. Current transition-boiling correlations being validated include the correlation of Tong and Young[16], as well as Condie and Bengston[17]. Film-boiling correlations include Groeneveld's[18] equations 5.7 and 5.9, Groeneveld's new nonequilibrium correlations[19], and Dougall-Rohsenow's[20] correlation for dispersed flow under equilibrium condition.

The rate of reflood heat transfer from electric heaters to ECC water during PWR reflood is obtained from Full-Length Emergency Cooling Heat Transfer Tests (FLECHT)[21,22,23] at Westinghouse, under cooperative effort with NRC and EPRI, for various flooding rates, system pressures, initial rod temperatures, powers and coolant inlet subcooling. Flooding rates below 1 inch/second were included, and a limited amount of flow blockage heat transfer information was obtained from several tests in which a horizontal plate was used to simulate "blockage" effects.

The conclusions obtained from FLECHT programs are as follows[22,23]:

- Reflood heat transfer, quench front velocity and liquid carryover fraction are correlated as functions of testing parameters.
- For flooding rates near or below 1.0 inch/second, trends of heat transfer, temperature rise, and quench times as functions of the various test parameters were essentially the same as in previous forced-flooding tests. The FLECHT correlation can be extended to the low flow rate region.
- Droplets entrained near the quench front were found to act as a heat sink for reflood rates around 1.0 inch/second and near a flow blockage.
- Steam heat transfer data in a partially uncovered core was obtained at 50 psia.

MIT has conducted experiments to provide better modeling of reflood phenomena, including heat transfer, carryover criterion, grid effect, etc. Reflood heat transfer models were established[24]. An inverse annular flow pattern is used for a high-flow reflood, and a droplets-dispersed flow pattern is used for low-flow reflood. These models are adopted in the NRC LOCA codes.

2.2.2 ECC Bypass and Refill Phenomena

Current NRC licensing calculations for LBLOCAs conservatively assume that: "For postulated cold-leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory." The research on ECC bypass and refill was conducted on small scale (1/15 and 2/15) experiments at Battelle Columbus and Creare to better understand ECC bypass through a cold-leg break and ECC penetration through the downcomer. The results are[25,26,27]:

- Experimental data support the Kutateladze (K*) scaling of flooding with saturated ECC fluid. However, the effect of subcooled ECC which enhances ECC penetration increases with scale size. This finding appears to be confirmed by preliminary results of 1/5 scale at Creare. The final scaling effect will be addressed in the large-scale downcomer tests of the international cooperative 2D/3D program.

- LOFT data shows that ECC water is delivered more quickly to the core than predicted by licensing calculations. A realistic correlation of ECC penetration data with steam condensation was developed[26] and can be used for best estimate.

- The limit of a countercurrent flow with condensation at perforated plate was developed at Northwestern University under contract with NRC. A general expression was developed[27] for CCFL using a new parameter, H*, to convert the j* to K* scaling as the size increases.

2.2.3 2D/3D Flow Distribution During Refill and Reflood in PWR

An internationally coordinated research effort is being conducted to study the multi-dimensional flow effects on ECCS performance during the refill and reflood for both large and small breaks, in large size test facilities to simulate a PWR. This research is coordinated among the Federal Republic of Germany (FRG/BMFT), Japan (JAERI) and USA (USNRC).

The BMFT is testing refill and reflood of large- and small-break LOCAs in PKL (340 heaters) at 500 psia, and is contracting to build at Manheim a full-scale PWR upper plenum test facility (UPTF), including internals, for the experimental study of deentrainment in upper plenum, water fallback into a PWR core, and ECC penetration in the downcomer during refill and reflood. Steam-water mixtures will be injected at the bottom of the upper core plate to simulate reflood carryover conditions. These data will be coupled with experiments performed in the Slab Core Test Facility (SCTF) in Japan. The NRC's TRAC code will be used to predict refill and reflood conditions and water deentrainment from the liquid carryover during the tests. Advanced two-phase flow instrumentation, provided by NRC, will be used to measure flow rates.

The Japanese Atomic Energy Research Institute (JAERI) has constructed the Cylindrical Core Test Facility (CCTF) (2000 heaters) and the aforementioned Slab Core Test Facility (SCTF) (2000 heaters) at Tokai to study the redistribution of steam and water flow within the core during refill and reflood. The SCTF has been designed to provide boundary conditions at the upper core plate compatible with those to be used in the BMFT upper plenum test facility.

The cooperative 2D/3D program will thus provide experimental information on ECCS behavior in large test facilities during the refill and reflood phases of a LOCA in PWR, and will allow assessment of the TRAC code for predicting the refill and reflood behavior of full-sized PWRs. The achievements to date are:

- The test of small-break LOCA at PKL Core I is completed, and data shows good core cooling behavior in natural circulation and reflux boiler modes,

- Testing of the first core of the cylindrical core test facility in Japan is completed (the data complements the results of FLECHT programs), shows a large deentrainment in upper plenum with simulated internals, and thus strongly reduced steam binding effect,

- Design work on the upper plenum test facility in FRG is completed, and

- Design of the slab core test facility in Japan has been completed and construction begun.

2.2.4 Primary Coolant Pump LOCA Behavior

The degradation of reactor circulation pumps during LOCA affects peak clad temperature. Semiscale pump tests[28,29] have shown rapid degradation of pump head to a few percent of the pump design head as the pump inlet void fraction was increased to 15-20 percent. Steady-state air water tests conducted on a 1/3 scale reactor coolant pump (EPRI/B&W research program) show similar behavior[30,31] but with less head degradation as void fraction increases. Most recently, results of a 1/5 scale pump operating in a steam-water loop, under both steady state and transient blowdown conditions (an EPRI/CE research program)[32] show that significant head degradation occurs at 25 to 30 percent void. The calculated peak clad temperature increases about 55°K (100°F) for the degraded model as compared with that of a non-degraded pump model[33,34].

Pump hydraulic torque data have also been obtained under two-phase flow conditions from the Semiscale and CE's 1/5 scale pump (steam-water) tests. These experiments confirm that pump torque decreases as void fraction increases. Preliminary results of EPRI/CE's 1/5 scale transient pump tests also confirm that pump speed is proportional to volumetric flow[35].

2.2.5 Two-Phase Flow Regime Modeling

Expressions were formulated by Ishii and his coworkers[36,37] for the interfacial area and drift velocity for various two-phase flow regimes. The agreement between calculated and measured drift velocities is within 20%.

The program conducted at the University of Houston is to provide experimental data for determining the interfacial friction factors and the interfacial area concentration in separated flows[38,39]. Experimental data and mechanistic models for predicting transitions between two-phase flow regimes in PWR and BWR rod bundle geometries have been obtained and used to establish a flow regime map applicable to rod bundle geometry[40]. Experiments are in progress to determine the effect of spacers.

2.3 Fuel Behavior Test Results

The elements of fuel behavior that affect the peak clad temperature and the cladding damage during accident can be shown in the chart in Figure 3.

2.3.1 Fuel Coolability During a LOCA

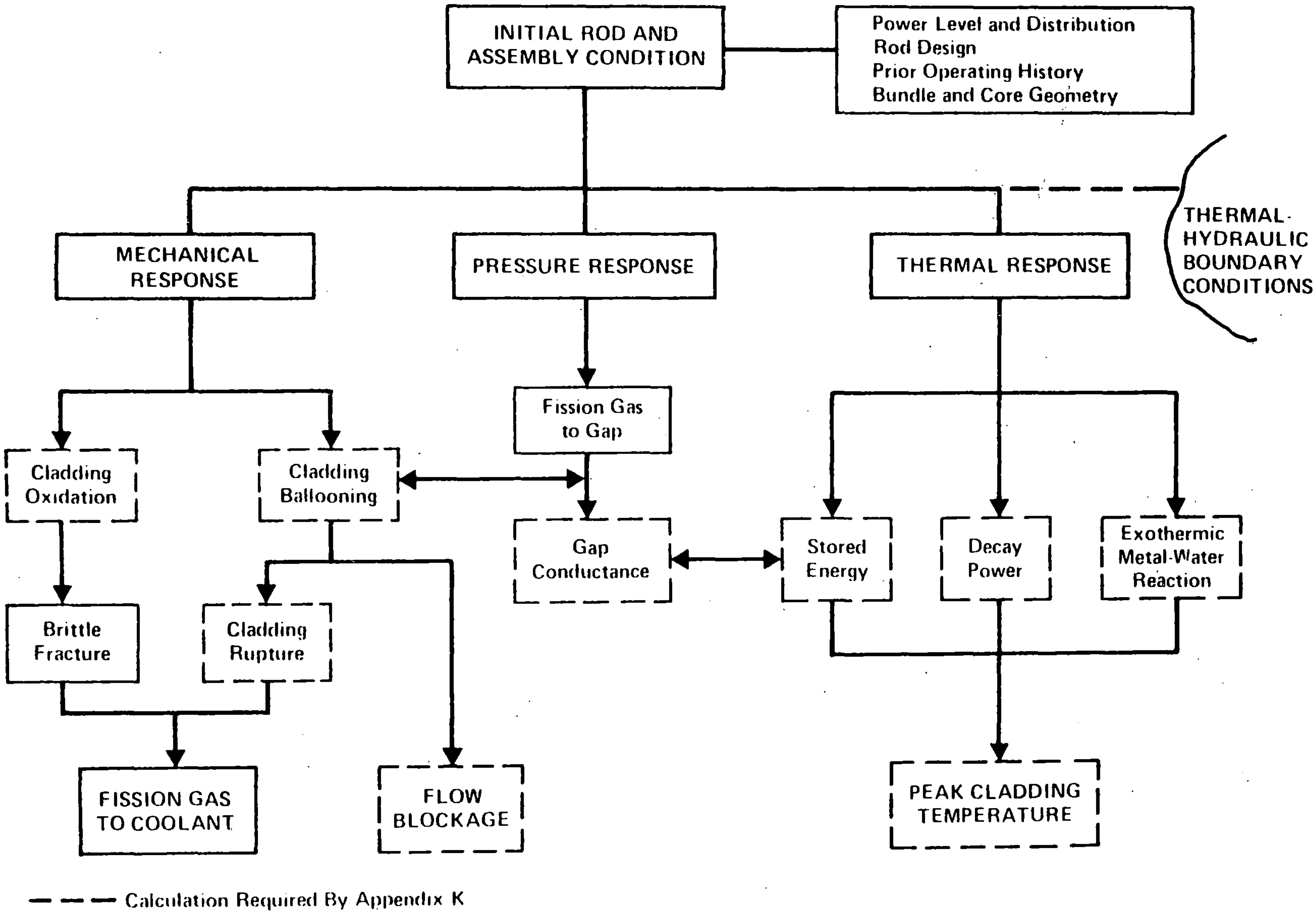
2.3.1.1 Decay Heat and Stored Energy

Fission product decay is an important heat source after reactor is shut down. Until recently, decay heat was represented by the proposed American Nuclear Society Standard ANS 5.1 (1973)[41], which was based on data from the late 1950s. The uncertainty in those data was judged to approach 15%, especially at cooling times of less than 100 seconds. The NRC has prescribed that a conservative value of 1.2 times the 1973 ANS value be used in evaluating the effectiveness of ECCS. Data obtained at LASL, ORNL and OSU[42] were approximately 7% below ANS value for time interval within 150 seconds after large LOCA. Furthermore, the uncertainties in the new data are nominally less than 5% for short transients and decrease as cooling time increases. On the basis of this data, a new standard has been developed by the American Nuclear Society. However, the old standard is still required for licensing calculations to date at NRC.

The calculation of the stored heat in the fuel rod requires knowledge of the gap conductance between the fuel and the cladding[43,44,45,46,47]. Current interpretation of the measurements suggests that fuel cracking is presumed to result in drastic (if not total) fuel relocation, accompanied by simultaneous large degradation of the fuel conductivity (stemming from resistance which cracks with nonradial components offer to radial heat flow). The overall consequences of this view, however, are that the cracked fuel model gives lower stored energy (due to the decrease in gap size) and this stored energy will be lost relatively faster in reactor shutdown (during LOCA) than would be predicted by a solid pellet model. To validate current calculation method[48], a complementary program to measure the gap conductance in the laboratory was completed at Battelle-Pacific Northwest Laboratory[49] over a range of conditions not previously measured, particularly high gas pressure in the gap between UO_2 and Zircaloy.

FIGURE 3.

FUEL BEHAVIOR DURING ACCIDENTS



The properties of cladding and the interaction of cladding and fuel in a fuel rod are mutually dependent. The properties and interactions affect the fuel-stored energy, which, in turn, affects the cladding temperature and its properties. Empirical and semiempirical correlations for many of these properties have been published as MATPRO[50].

2.3.1.2 Zr Oxidation and Embrittlement

Measurements have been made of the rate of Zircaloy oxidation in steam at temperatures between 1035 and 1700°K (1400 and 2600°F)[51]. The data have been used to predict oxidation under LOCA conditions. The maximum oxidation rate was about 76% of that predicted by extrapolation of the Baker-Just equation[52] to 1475°K (2200°F).

The mechanical properties of Zircaloy have been determined as functions of oxygen distribution and content, strain rate, biaxial stressing, microstructure, texture, and temperature over the range between 423 and 1700°K (300 and 2600°F). These factors determine the strength, ductility, and deformation of Zircaloy cladding during LOCA and power-cooling-mismatch accident[53].

The data allow consideration of new, quantitative embrittlement criteria to replace those presently in 10 CFR 50.46 and Appendix K. The new criteria are based on mechanical properties that are readily measured and are related to the mechanical loads that may be placed on damaged fuel cladding in postulated accident. The recommended criterion for surviving the thermal shock expected by quenching of hot oxidized fuel rod cladding during ECCS reflood, stipulates that there remain at least 0.1mm of wall thickness that contains a concentration of oxygen less than 0.9 wt%. The criterion recommended for an oxidized and damaged fuel bundle to survive a "bundle drop" accident during core disassembly is that at least 0.3mm of wall thickness remain that has a concentration of less than 0.7 wt% oxygen[54].

2.3.1.3 Clad Swelling and Rupture

Tests of clad burst at high temperatures are being performed with electrically heated rods to give flattened temperature gradients comparable to those in the middle of fuel rods in PWRs and BWRs, with internal pressures from 100 to 1800 psi and heating rates of up to 28K/sec (50°F/sec).

the findings from the bundle tests of ORNL[55,56] are:

- Clad swelling and burst are not coplanar because of the stochastic nature of the fuel pellets,

- Fuel rod swelling is, in average, smaller than the data obtained from hollow cylinders, partly because fuel rod clad is axially constrained by the fuel stack, and

- Ballooning is increased at low heating rates and greater local temperature uniformity.

The LOCA blowdown tests at the Power Burst Facility are showing that cladding deformation and oxidation in reactor are moderate and consistent with results from out-of-reactor tests using electrical heating. In the PBF LOCA test, PWR fuel rods survived thru three blowdowns in which clad temperature reached up to 1015°K (1370°F)[57].

The net effect of fuel rod swelling and rupture on the coolability of the fuel should be determined by measuring coolant flow resistance in fuel bundles during reflood. This flow resistance is measured by a separate hydraulic test of a swelled bundle.

2.3.2 Fission Product Release and Transport During LOCA

2.3.2.1 Fission Product Release Into Coolant

The release of the gaseous fission products from uranium dioxide during irradiation is a function of the operating history of the fuel rod. Fuel temperature is the principal determinant of gas release. As fuel temperatures exceed 1725°K (2592°F), migration of gas bubbles can be significant, and releases approaching 100% can be observed for molten fuel. Burnup has no discernible effect on gas release below 15 GWd/MTM. There is growing evidence for the enhancement of release above 30 GWd/MTM[58,59].

Mechanistic models[58], based on the growth and migration of intergranular bubbles, are emerging as most generally applicable in describing gas release. However, they are often uneconomical to use routinely in computer calculations. In their place, semiempirical correlations[60] are used in evaluations for licensing. Recently, however, efforts have been made to simplify the sophisticated mechanistic models[61].

Intermediate volatile fission products (iodine and cesium) are particularly important to safety. Although the migration mechanisms for iodine, cesium, and the less volatile fission products are not well characterized, it is known that substantial amounts of nongaseous fission products, such as cesium and iodine, can migrate to the fuel cladding gap during irradiation.[62,63].

A program at ORNL has measured fission product release from defected LWR fuel rod segments under prototypic LOCA conditions. Their results, when applied to a PWR LOCA, indicate that the release of cesium and iodine will be a factor of ten to one hundred less than the gap release assumptions for these species given in WASH-1400. Accelerated release of these fission products occurs at approximately 1625°K (2460°F). This study also presents strong evidence that the principal chemical species of iodine released from failed fuel into a steam environment is CsI which is highly soluble in water, thus reducing the amount of fission products released into the containment atmosphere. Out-of-reactor experiments[64,65], using irradiated fuel under more controlled conditions, are elucidating the mechanisms of release and transport. In-reactor experiments monitoring releases of activity during normal operating conditions as functions of power and of defect size for rods intentionally made defective are underway in Japan and France. The development of mechanistic models to trace the path of fission products once they leave the fuel is also in progress[66].

2.3.2.2 Fission Product Release and Transport Into Containment

Elemental vapor and chemically active particulate forms of iodine are readily removed from the air by chemical sprays and filtering systems. Methyl iodide, other organic iodides, and possibly hypoiodous acid have been identified as persistent airbone species but are conservatively believed to make up no more than 5% of the total iodine released into the containment atmosphere. Fission-product and fuel aerosols are effectively removed from the containment space by agglomeration and gravitational settling, as well as engineered safety features. Analytical models have been developed to predict the airborne concentration of fission products as a function of time for a given set of input data, including fission-product concentration, particle-size distribution, containment atmospheric conditions, and geometry[67].

2.4 Code Development and Assessment

The LOCA and transient codes being developed under the sponsorship of the NRC include: (1) systems codes, (2) component codes, and (3) fuel codes (see Figure 4).

2.4.1 System Codes

The system codes are developed to analyze both large- and small-break LOCAs and their associated transient. The systems included the reactor coolant system and the balance-of-plant.

The systems codes are further subdivided into reactor coolant system (RCS) codes and containment system codes. There are two versions of RCS codes. The evaluation model (EM) version employs conservative assumptions defined in the NRC's ECCS Acceptance Criteria for LOCA analysis[1]. The best estimate (BE) versions employ realistic modeling of the physical processes to predict plant response during LOCA and other transients, to evaluate safety margin, and to design test facilities and interpret test results.

Important system codes (illustrated in Figure 4) are RELAP and BEACON developed at INEL, TRAC developed primarily at LASL, and WRAP developed at Savannah River Laboratory.

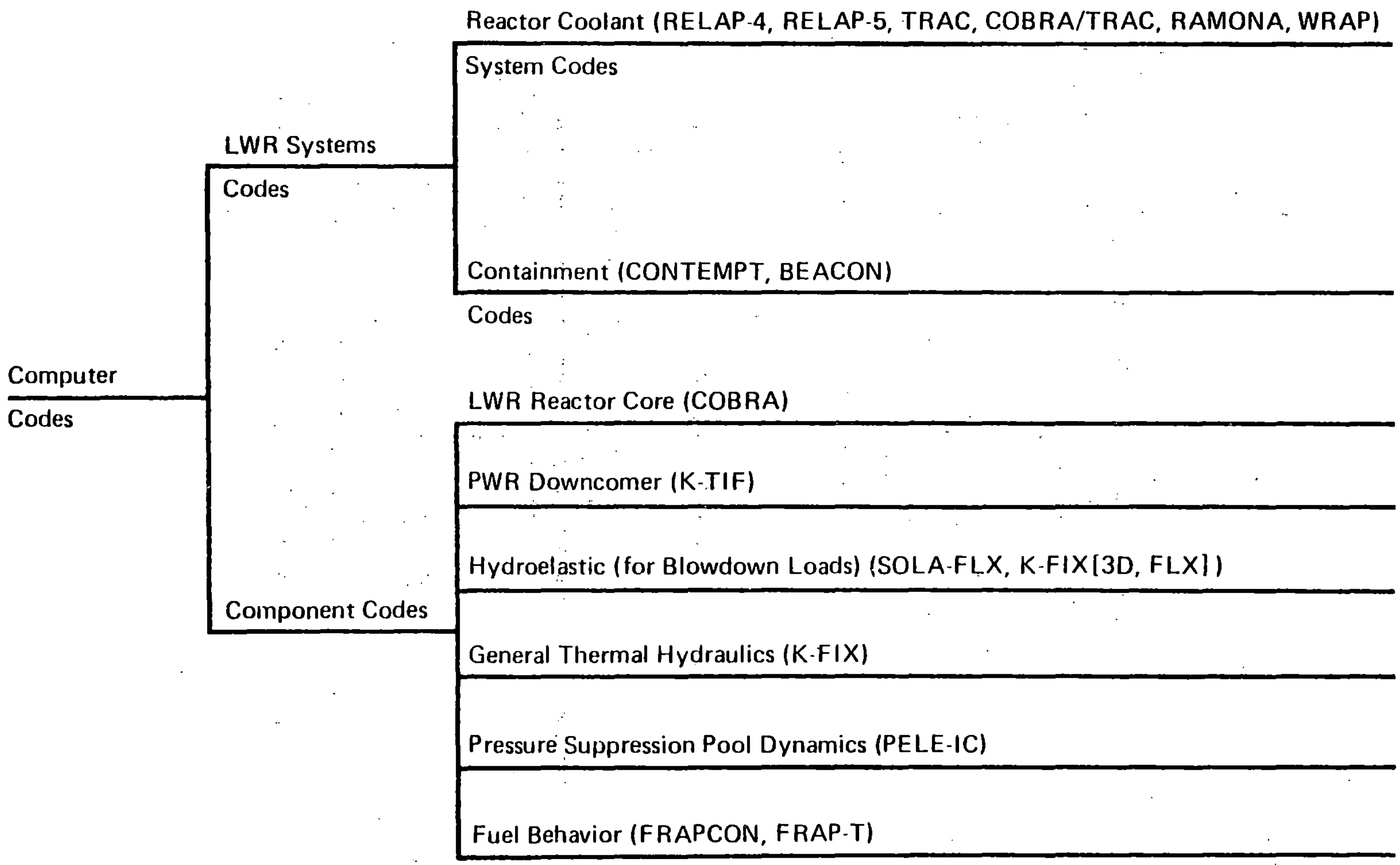
2.4.1.1 RELAP

The RELAP-4 series of reactor coolant system codes, developed at the Idaho National Engineering Laboratory (INEL), is based on assumptions of thermal equilibrium and equal velocity of the two-phases. The numerical solution of these equations is based on the lumped-parameter approximation, in which the system is represented by a collection of control volumes (nodes), interconnected by "junctions." The conservation of momentum is solved as a one-dimensional case for each flow junction.

Current versions of RELAP-4 are MOD6 and MOD 7[68,69]. MOD 6, released in 1978, extended the applicability of RELAP-4 to the PWR reflood regime. The purpose of RELAP-4/MOD7[69] is to further improve the heat transfer models and add nonequilibrium and nonhomogeneous explicit approximations into the RELAP-4/MOD6 code. Also, a limited BWR capability is added.

FIGURE 4.

COMPUTER CODES



To improve the Evaluation Model code package, the RELAP-4/MOD5[70,71] code was cast into the user-convenient version at the Savannah River Laboratory. By the end of CY 1979, additional codes had been interfaced with RELAP-4 to complete the WRAP-EM code package for analysis of LOCA in both BWR's and PWR's. The WRAP-EM package is currently undergoing a process of verification against test data and vendor calculations. Savannah River Laboratory has also completed a similar system for BWR evaluations.

The RELAP-5 code[72] is being developed by the Idaho National Engineering Laboratory (INEL) as a "parallel path" effort to ensure that a fast-running 1-D advanced systems code will be available on a timely basis. RELAP-5 is currently in use at INEL for interpretation of test data of LOFT and Semiscale, and is expected to be released later in 1980.

2.4.1.2 TRAC

The TRAC code[73] is the NRC's most advanced best estimate systems code. The TRAC-PWR code is being developed at Los Alamos Scientific Laboratory, and the TRAC-BWR code is being developed at the Idaho National Engineering Laboratory. Among the features of TRAC is the capability for multidimensional (r, Θ, z) representation of the reactor (PWR and BWR) vessel interior, employing the two-fluid model. This model uses separate conservation equations (for mass, momentum, and energy) for the liquid and the vapor, including the interfacial balance equations. Thus, fully nonequilibrium conditions, both thermal and velocity, are handled.

The BWR version of TRAC, which incorporates models specific to BWR geometry, is expected to be released early in 1981.

The PWR LOCA version (TRAC-PIA)[73] was released in 1979. An improved version, TRAC-PD2, will be released later in 1980.

Work is also continuing on restructuring of the TRAC code to produce a fast-running version, and to include noncondensible gases.

2.4.1.2 BEACON

BEACON-3 is a containment system code developed by INEL featuring multidimensional multifluid modeling with thermal nonequilibrium, suitable for the analysis of the first few minutes of LOCA involving hydraulic loads on intercompartment barriers and jet loads.

2.4.2 Component Codes

The component codes allow various reactor components to be nodalized and modeled separately. The model can be tested in various separate-effects tests, or in some instances, in integral system tests. These validated models of the components are then adopted in the system code, as illustrated in Figure 2.

Important component codes include 3-dimensional K-FIX[74] of Los Alamos National Laboratory and COBRA-TF[75] of Battelle Pacific Northwest Laboratories. These codes were developed for two-phase flow analyses in which flow regime is the key parameter.

During the subcooled portion of blowdown, structural/hydraulic coupling can affect the instantaneous pressures, and the hydraulic loads on the reactor core barrel and/or steam-generator. Hydroelastic codes are developed at Los Alamos National Laboratory as 2-dimensional SOLA-FLX and 3-dimensional K-FIX/FLX[76]. Similar hydroelastic effects could also take place in the wetwell of the pressure-suppression (BWR) containment during the steam-venting phase, such as PELE-IC code of Livermore National Laboratory.

The COBRA code was originally developed for reactor-core subchannel analysis by Battelle Pacific Northwest Laboratories. In addition, COBRA-IV-I can be applied to fuel assemblies and to core-wide analysis[75]. An advanced version of COBRA is being integrated with the TRAC code to provide loop system and boundary conditions to the vessel.

2.4.3 Fuel Codes

Fuel behavior codes analyze the thermal, mechanical, and internal gas response of fuel rods to predict rod behavior and integrity.

2.4.3.1 Steady-State Codes

Principal LWR steady-state fuel behavior codes include FRAPCON[77,78,48], COMETHE-IIIJ[79]. BEHAVE-4[80], LIFE-THERMAL-1[81], GAPCON-THERMAL-3[82], and CYGRO[83].

Gap conductance and fuel-stored energy models have been evaluated by tests at PBF and Halden. Results were used to validate capability of FRAPCON to adequately calculate stored energy.

2.4.3.2 Transient Codes

FRAP-T5[84,85] can analyze fuel transients in LOCA from blowdown through reflood using the FLECHT reflood correlation and several optional CHF correlations. It can also handle reactivity-initiated (RIA) transients and flow-reduction-caused transients. The code includes pellet relocation and uncertainty analysis.

In comparison with experimental data, the code predicts fuel centerline temperatures to within the data uncertainty of about 10%. The uncertainty increases with burnup but remains conservative. Better modeling of internal rod pressures and incorporation of a strain rate dependence on the burst models have improved predictions for time and strain at rupture. Assessment results indicate that the thermal models and predictions of FRAP-T5 are quite good.[86].

2.5 Operational Safety Research

The accident at Three Mile Island and the results of subsequent investigations have reemphasized the importance of reactor operators and the role they play in determining the level of safety associated with nuclear power. TMI has also shown us the need to improve our understanding of the physical response of the plant during accident sequences involving multiple failures of components and of how operator intervention or inaction might aggravate or mitigate the consequences of such sequences. With these needs in mind, NRC has initiated a series of analytical studies to explore the probable course of events and potential consequences of a spectrum of accidents extending well beyond the current design basis in terms of system failures, core damage and release of radioactivity to the environment. This program is called Severe Accident Sequence Analysis (SASA).

The objective of the program is to gain insights into the course of potentially severe accidents. Particular emphasis is placed on the perceptions and the needs of the operator, the alternative actions available under various combinations of component failures, and the consequences of those actions. The analyses consider the various recovery options available to the operator and the timing available for such actions.

These insights will be gained by applying best-estimate state-of-the-art codes (e.g., RELAP, TRAC, MARCH/CORRAL) to several specific plants. The accident sequences initially selected for analysis will be determined according to their perceived contribution to risk (e.g., station blackout, loss of feedwater) as suggested by documented risk assessments. Initial results on station blackout calculations are available. Four national laboratories (INEL, LASL, ORNL and SNL), each applying its particular strength, are participating in SASA.

3. CURRENT STATUS AND FUTURE LOCA RESEARCH

At present, a good understanding of LOCA/ECCS behavior, for both large- and small-breaks, has been acquired. The research results have provided technical bases for the licensing decisions, regulatory guides and standards at USNRC. Further research will emphasize exploring the plant response to anomalous transients and minimizing human errors. Details are given in the following:

3.1 Current Safety Status on LOCA

3.1.1 ECCS Effectiveness in Large-Break LOCA

The adequacy of existing ECCS's to cool the core during a large-break LOCA has been well demonstrated in LOFT, Semiscale and TLTA tests. The tests of alternative injection locations of ECCS at Semiscale show that the ECC effectiveness may be further improved in future design.

3.1.2 Core Cooling in Small-Break LOCA with Anamalous Transients

Natural circulation with an active steam generator, and makeup and letdown operation with an inactive steam generator proved adequate in various tests for recovery of small-break LOCA with anomalous transients. However, the effect of non-condensibles accumulated in the top of a once-through steam generator, the CCFL effect in the hot leg and steam generator tubing during reflux boiler mode cooling of a U-tube steam generator, and the variety of anomalous transients which could be associated with small-break LOCA are still under research.

3.1.3 Confirmation of Models in ECCS Evaluation

The thermal-hydraulic and fuel behavior models used in ECCS evaluation, as specified in Appendix K of 10 CFR 50, have been confirmed to be conservative by using the research results in previous sections. Some of the models, however, may be overly conservative, which may adversely influence the optimization of ECCS for both small- and large-break LOCA's. For example, the accumulator injection may be preferred at higher pressures during blowdown for a small-break LOCA in PWR, but the ECC bypass rule prevents accounting for its benefit. Similarly, the upper plenum injection may be more effective to cool the partially uncovered core during a small-break LOCA in PWR, but steam binding concern is against it. Further research and consideration are needed in this area.

3.2 Future LOCA Research

3.2.1 Operator Action Event Tree

To enhance the operator's performance, an "operator action event tree" is recommended for future research. The development of the event trees began with the trees as they appeared in the original risk analyses. The events in each sequence which involved operator action were identified, and in some cases, broken down into additional events in order to highlight individual operator tasks. In addition, the sequences were expanded (events added to the event tree) to include additional operator actions which could be performed to prevent core melt, but were not taken credit for in the original analysis. These additional events included "repair events," where the operator is given the opportunity to attempt to restore or replace a particular function, and "delay events," where the operator is called upon to delay an inevitable melt as long as possible or to perform some other consequence mitigating action.

For each accident sequence, the physical response of the plant is defined in terms of measurable parameters. The time-dependent variations and the interrelationships of these parameters generate an "accident signature," a uniquely characteristic array which can be used to evaluate the status of the plant.

Once the event logic and physical response of the plant are established, it is relatively straight forward to identify the key operator actions and the operator's information requirements. This is done by characterizing the status of the plant on each branch of the tree and associated appropriate actions in terms of physically measurable parameters.

This research introduces some important new concepts and technical approaches which, if properly developed and applied, could make significant contributions to accident analysis. It emphasizes the perceptions of the operator, the needs for information and the alternative successful actions one might take given various combinations of component failures. Beyond determining instrumentation requirements, the methods have important implications with respect to developing emergency procedures, generating training simulator exercises, and designing operational aids, including computerized disturbance analysis systems. Some of these features are being explored under the LOFT program.

3.2.2 Large- and Small-Break LOCA Tests with Multiple Failures

To mitigate the consequence of LOCA in existing reactor systems, many postulated multiple failures associated with various LOCAs should be tested. In large-break LOCA, a degraded ECCS should be tested for pressurized fuel rods and for a delayed injection pump operation.

In small-break LOCA, a station blackout and effects of large amount of non-condensibles should be tested further.

3.2.3 Optimization of ECC Systems

To enhance plant recovery from a partially uncovered core during a small-break LOCA, the effect of upper plenum injection of ECC should be tested for both large- and small-break LOCAs with well simulated upper plenum and internals, and the countercurrent flow limit should be tested in a large-size hot-leg.

To early terminate a small-break LOCA, the actuating pressure of accumulators should be tested to optimize the core cooling in SBLOCA and ECC bypass in LBLOCA for the existing cold-leg ECC injection system and for a future lower plenum ECC injection system. The lower plenum ECC injection system also minimizes the uncertainties in its performance calculations.

3.2.4 Code Assessment and Qualification

Reliable best-estimate computer codes are needed for predicting the system response of LOCA with anomalous transients. To ensure that significant physical phenomena in the reactor system and the balance-of-plant are properly modeled in these LOCA codes, the uncertainties caused by any simplification or approximation of these modelings must be qualified during their assessment.

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PRESENTED BY

J. H. LINEBARGER

MONDAY, OCTOBER 27, 1980

MORNING SESSION AT 11:20 a.m.

THE EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

GAITHERSBURG, MD

SMALL BREAK RESULTS FROM LOFT
L3-1, L3-2, L3-5/5A, AND L3-7

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-30, 1980
Gaithersburg, Maryland

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SMALL BREAK RESULTS FROM LOFT L3-1, L3-2, L3-5/5A, AND L3-7

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Following the Three Mile Island (TMI) accident, the Nuclear Regulatory Commission (NRC) focused their attention on the small break loss-of-coolant accidents (LOCAs). Computer codes are used, when preparing operating plant Safety Analysis Reports in conformance with NRC Regulatory Guide 1.70, to calculate plant response during small breaks. These calculations are used to determine the consequences of such events and to evaluate the capability built into the plant to control or accommodate such failures and situations. In particular, calculations are used to determine plant protection system and control system set points and to specify emergency operating procedures. However, no data base existed, other than the TMI accident itself, to verify these calculations and provide a realistic basis for understanding various accident scenarios. The LOFT small break experiment program was accelerated to meet this licensing need.

The NRC was specifically interested in general accident scenarios, the ability of the computer codes to predict system behavior, and the specifics of plant recovery including use of the steam generator and operator controlled and initiated actions. To satisfy these needs, LOFT decided to conduct simulated 0.10 m (4 in.) and 0.025 m (1 in.) cold leg small break experiments. The choice of break sizes and break location was based on calculated transient severity and scenario characteristics of a commercial pressurized water reactor (PWR).

To date, two 0.10 m (4 in.), L3-1[1,2,3] and L3-5/5A[4,5], and two 0.025 m (1 in.), L3-2[6,7,8] and L3-7[9,10,11], nuclear experiments have been conducted. In each instance the plant was recovered and sufficient coolant remained in the core so that fuel cladding temperatures decreased from steady state values throughout the transient. Operator controlled and initiated actions such as steam generator secondary feed-and-bleed and manually tripping the reactor primary coolant pumps (PCP) immediately after scram were evaluated. An additional experiment, L3-6 scheduled later this year will be a repeat of L3-5 except the PCPs will remain on throughout system depressurization. The two experiments, L3-5 and L3-6, will be used to determine the effect of PCP operation on system mass inventory and distribution during a small break LOCA, a specific issue of current interest to the NRC.[12]

The LOFT experiment scenarios were quite similar to those predicted for a four-loop Westinghouse commercial PWR, to which the LOFT system is scaled. The most notable exception is the partial core uncovery predicted during a 0.10 m (4 in.) commercial PWR break. As previously mentioned, core uncovery did not occur during the LOFT experiments.

Pretest calculations using RELAP5, a two-fluid, nonequilibrium, one-dimensional computer code, predicted the dominant phenomena in the proper sequence in most instances. Two particular problems encountered during preexperiment calculations were (a) predicting the gradual system repressurization late in the plant recovery process when the plant was refilling and vapor was being trapped and compressed in the highest parts of the primary system, (b) predicting the proper time scale during L3-5 depressurization.

LOFT experiment results indicate the steam generator is effective in removing decay heat from the system. Natural circulation, the system flow mechanism necessary to sustain decay heat removal by the steam generator, appears stable in all modes, reversible and stable between modes, and reestablishable after cessation. Reactor vessel fluid voiding, the introduction of cold emergency core coolant into the system, and noncondensibles, both accumulator nitrogen and those coming out of solution, have not measurably deterred natural circulation during LOFT experiments. When the steam generator is a system heat sink, operator controlled steam generator secondary feed-and-bleed expedites system recovery without opening an additional break in the primary system.

The current small break experiment series will be completed sometime during the next year. By that time, the LOFT Program will have furnished a realistic data base for a spectrum of conditions needed to verify licensing calculational techniques, to provide confidence in the method currently used to determine system protection and control system set points and specify emergency operating procedures, and to address specific licensing issues of current interest.

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2. V. T. Berta, "Modeling A Nuclear Reactor For Experimental Purposes", American Engineering Model Society Meeting, Pittsburgh, PA, May 1980.

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4. W. H. Grush, J. C. Lin, J. H. Linebarger, R. A. Nelson, Jr., "Results of Scaled, Nuclear Large Break Loss-of-Coolant Experiments", ASME Winter Annual Meeting, Chicago, IL, November 1980.

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SMALL BREAK RESULTS FROM L3-1, L3-2, L3-5/5A AND L3-7

by

J.H. LINEBARGER

EG&G Idaho, Inc.

CONTENTS

LICENSING CONCERN - PLANT RESPONSE

SCENARIOS

DATA PREDICTIONS

RECOVERY
- STEAM GENERATOR
- OPERATOR ACTIONS

JHL-2

SMALL BREAKS (SINGLE FAILURE)
WESTINGHOUSE PREDICTIONS

COLD LEG BREAK SIZE	PRESSURE	DECAY HEAT	ECC	CORE UNCOVERY
$\geq\sim$ 0.05m	CONTINUOUS	BREAK	HPIS AND ACCUMULATOR	PARTIAL
$\leq\sim$ 0.025m	STABILIZES ABOVE SECONDARY	S.G. AND BREAK	HPIS	NONE

JHL-3

REVIEW OF PROGRESS

COLD LEG BREAK SIZE (SIMULATED)	EXP.#	OPERATOR ACTIONS
0.10m	L3-1	S.G. F AND B
	L3-5	PUMP OPERATION
	L3-6	
0.025m	L3-2	S.G. F AND B
	L3-7	
	L3-5A	

JHL-4

RESULTS

SCENARIOS

TRANSIENT SIGNATURES

- L3-5/5A
- L3-7

SYSTEM PRESSURE DATA

RECOVERY

- L3-1, L3-5A
- L3-2, L3-7

SYSTEM DATA

JHL-5

LOFT SMALL BREAK

(SIMULATED 0.10 AND 0.025 m)

SCENARIOS

JHL-5A

SMALL BREAKS (SINGLE FAILURE)
LOFT EXPERIMENT RESULTS

COLD LEG BREAK SIZE (SIMULATED)	PRESSURE	DECAY HEAT	ECC	CORE UNCOVERY
0.10m	SHORT STABLE PERIOD-THEN CONTINUES	BREAK	HPIS AND ACCUMULATOR	NONE
0.025m	STABILIZES ABOVE SECONDARY	S.G. AND BREAK	HPIS	NONE

JHL-5B

TRANSIENT SIGNATURES

L3-5/5A

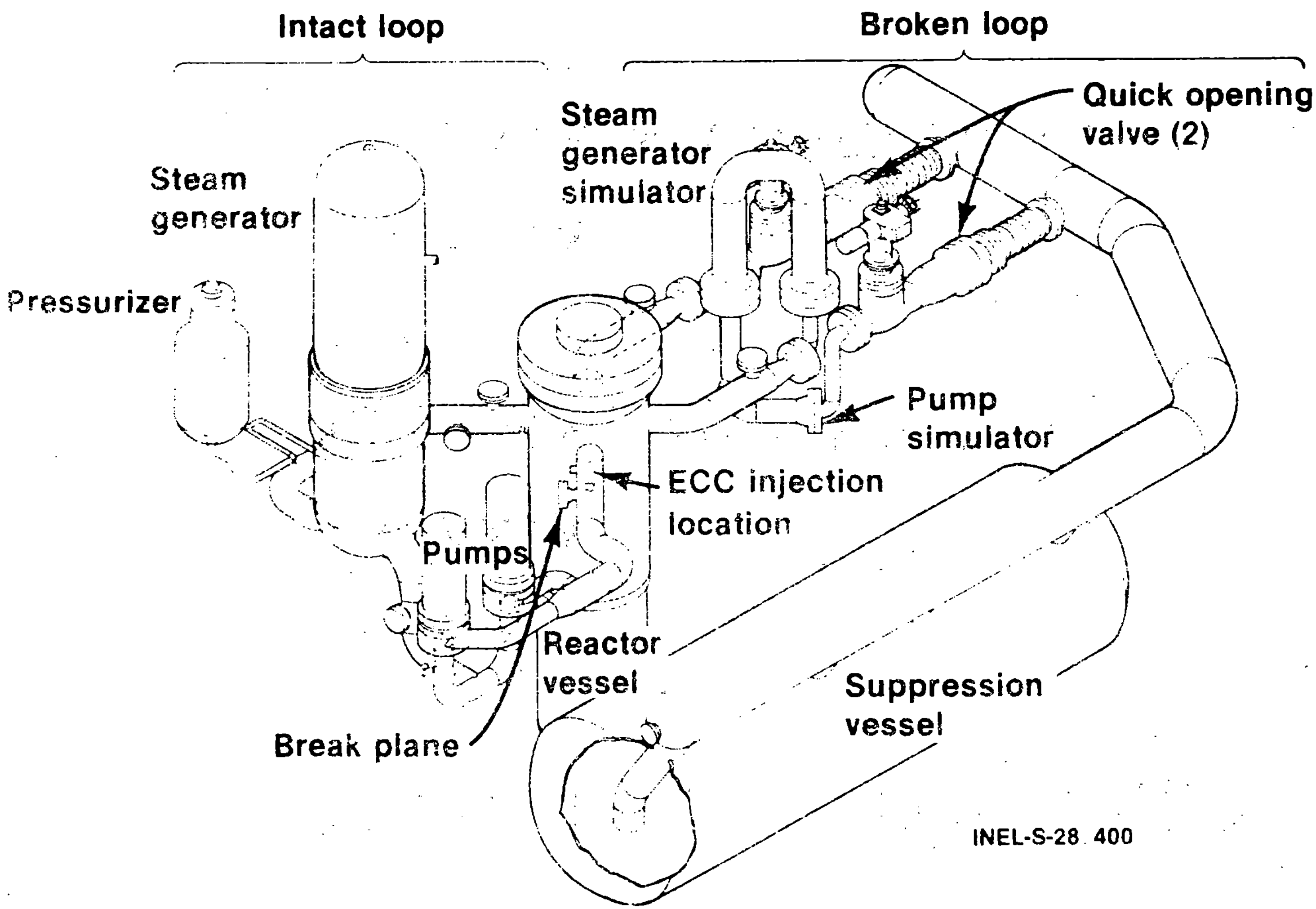
- DATA
- DATA AND PREDICTIONS

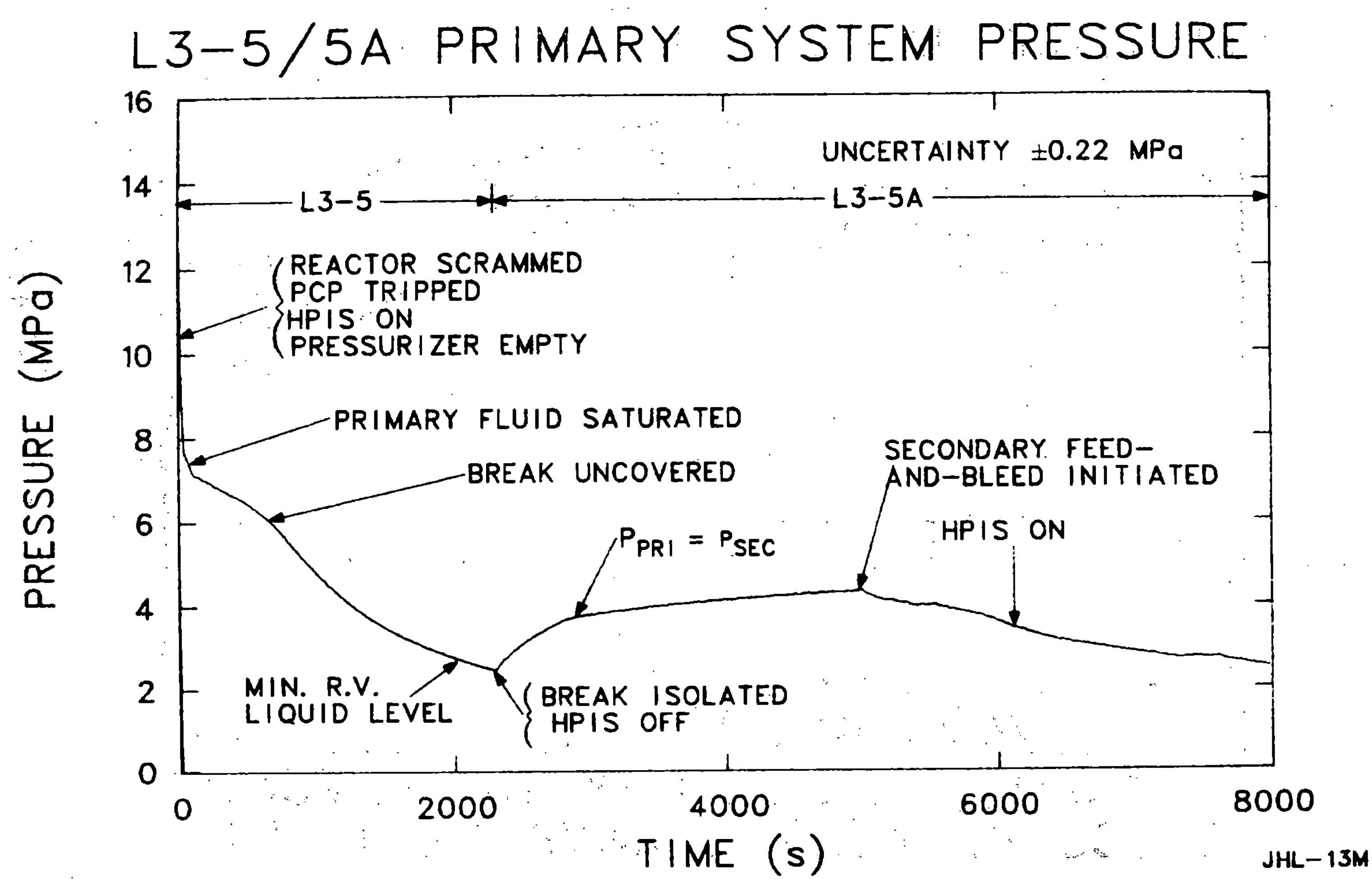
L3-7

- DATA
- DATA AND PREDICTIONS

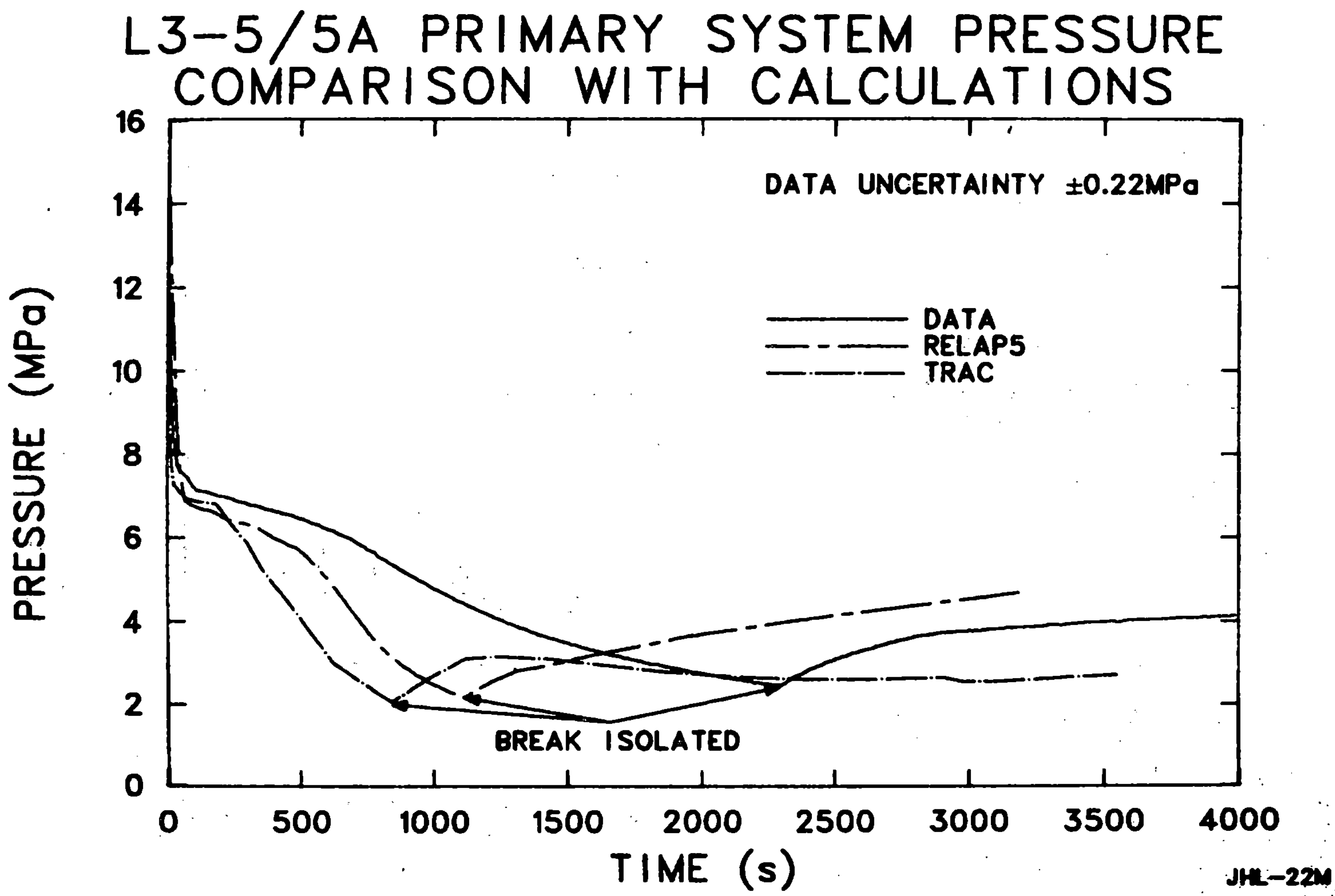
JHL-6

LOFT Primary System

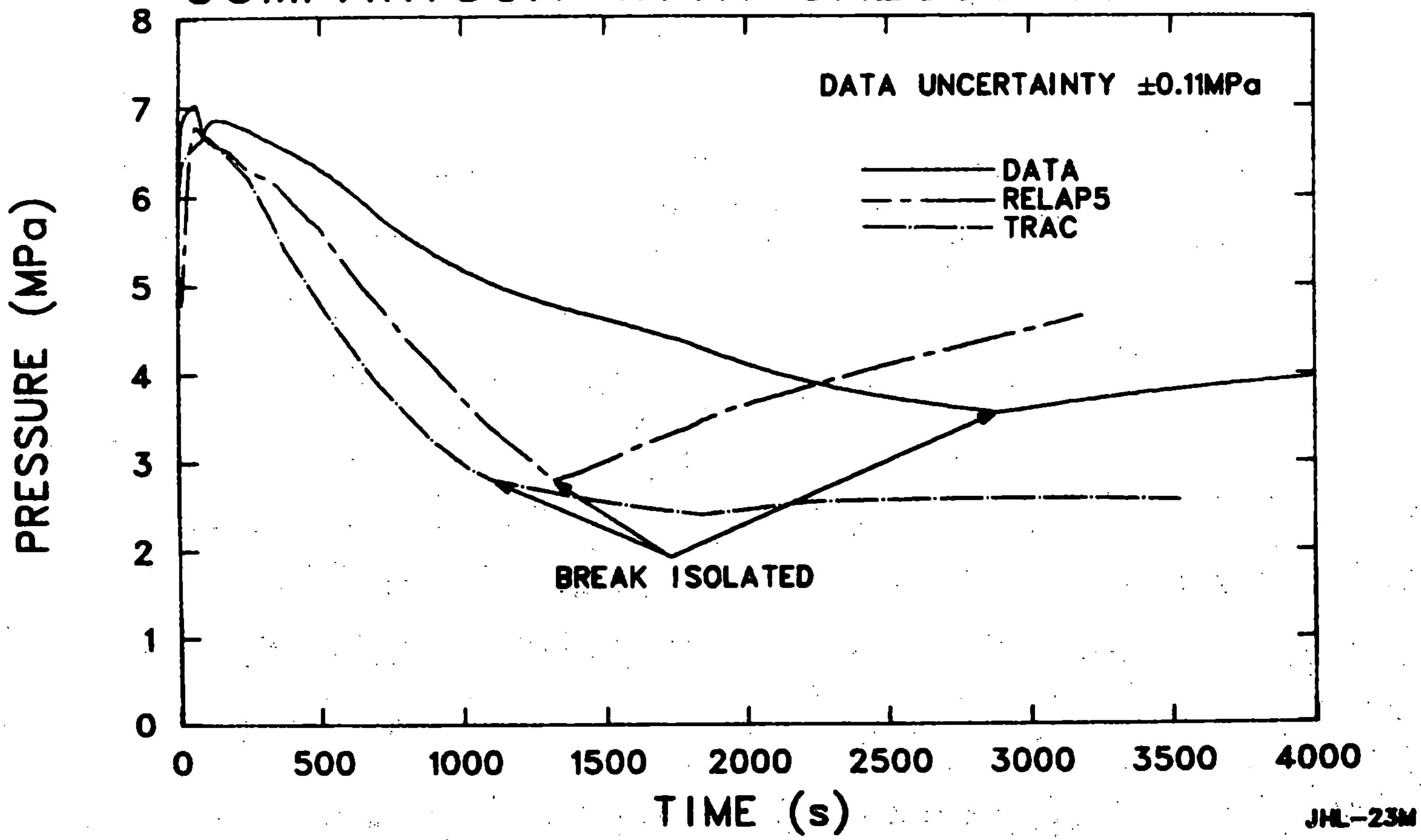




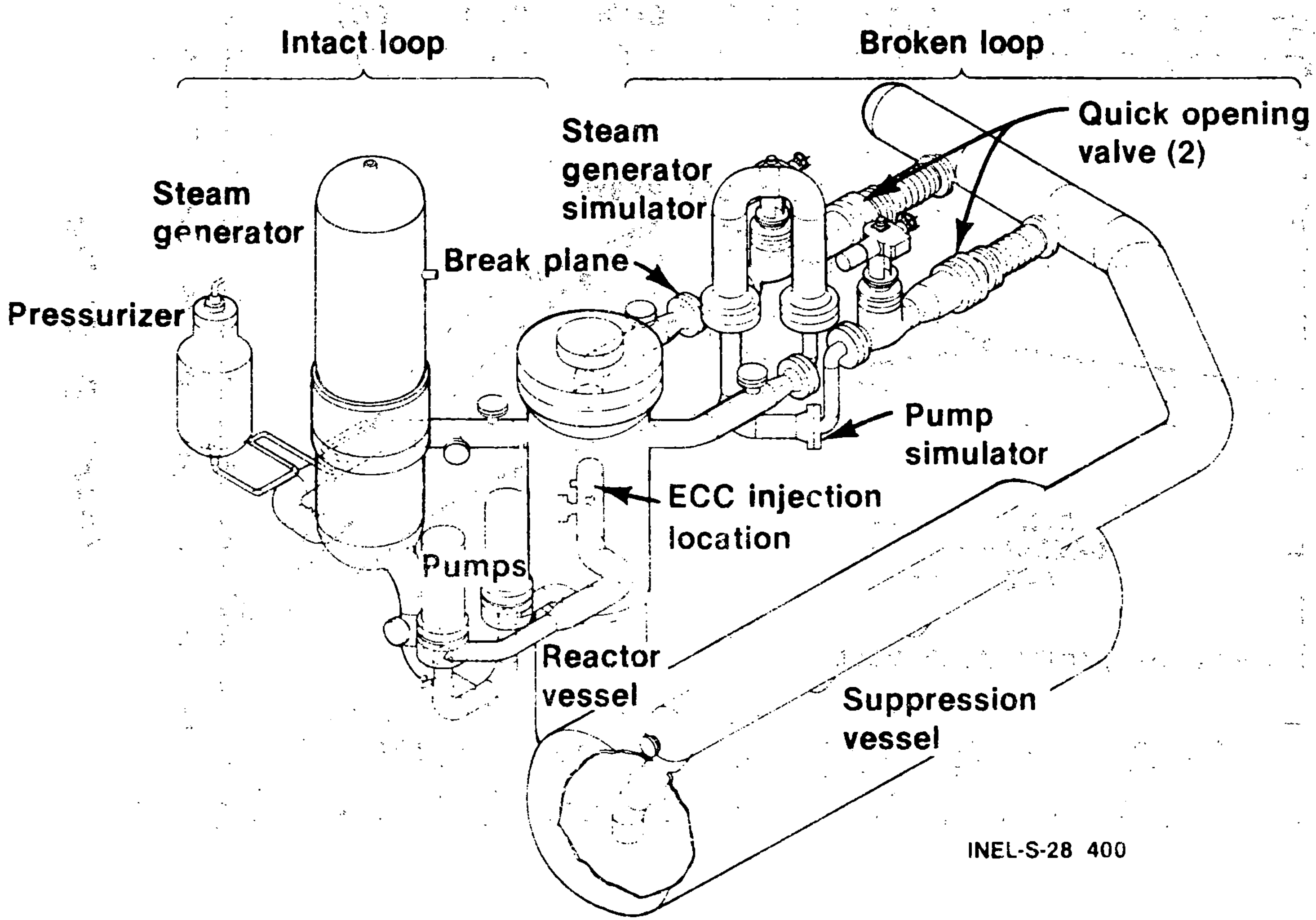
L3–5/5A PRIMARY SYSTEM PRESSURE
UNCERTAINTY ±0.22 MPa
L3–5
L3–5A
REACTOR SCRAMMED
PCP TRIPPED
HPIS ON
PRESSURIZER EMPTY
PRIMARY FLUID SATURATED
BREAK UNCOVERED
$P_{PRI} = P_{SEC}$
SECONDARY FEED–
AND–BLEED INITIATED
HPIS ON
MIN. R.V.
LIQUID LEVEL
BREAK ISOLATED
HPIS OFF
PRESSURE (MPa)
TIME (s)
0
2
4
6
8
10
12
14
16
0
2000
4000
6000
8000
JHL–13M


L3-5/5A PRIMARY SYSTEM PRESSURE
COMPARISON WITH CALCULATIONS
DATA UNCERTAINTY ±0.22MPa
DATA
RELAP5
TRAC
BREAK ISOLATED
PRESSURE (MPa)
TIME (s)
0
2
4
6
8
10
12
14
16
0
500
1000
1500
2000
2500
3000
3500
4000
JHL-22M

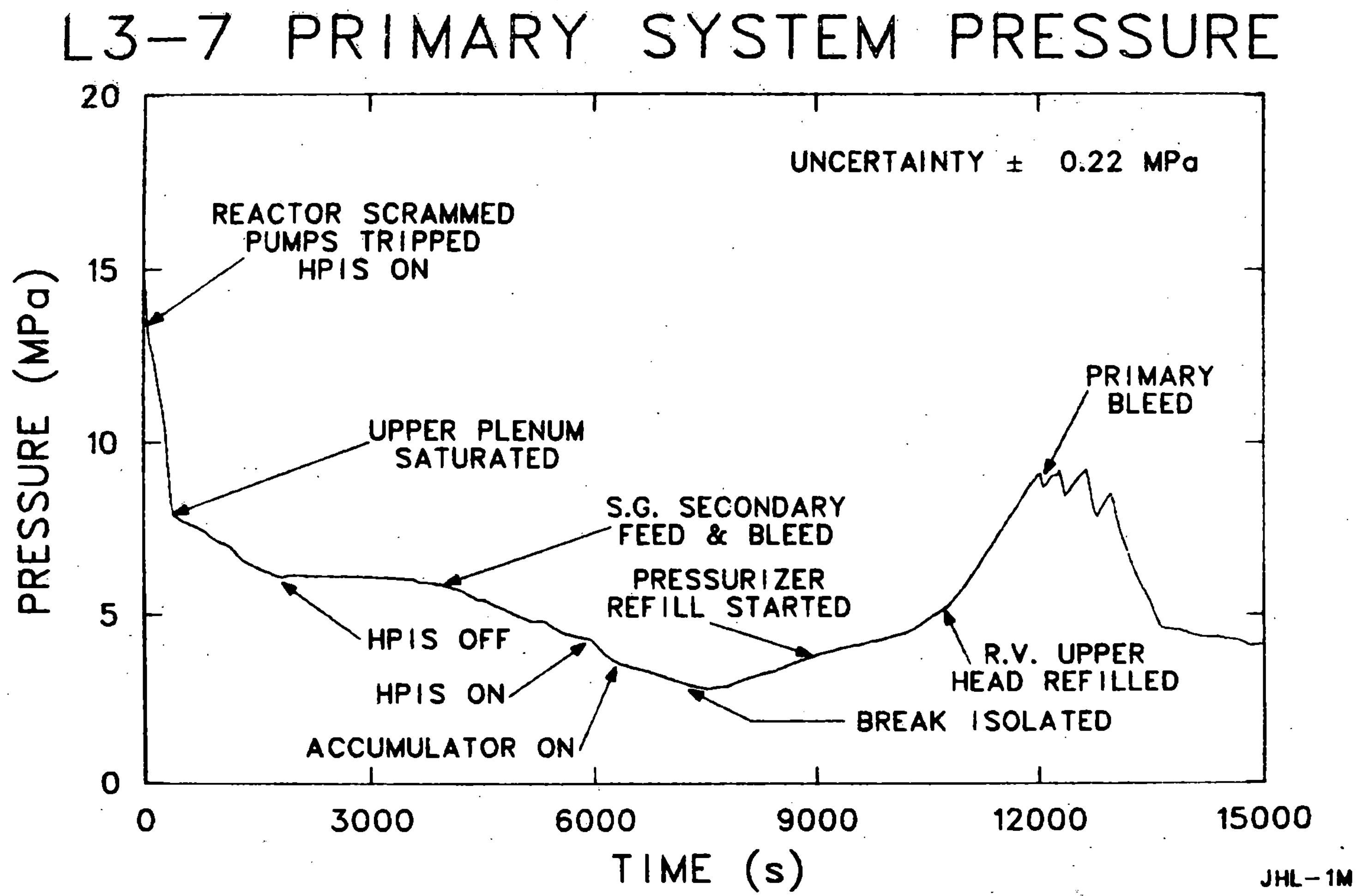
L3-5/5A SECONDARY SYSTEM PRESSURE
COMPARISON WITH CALCULATIONS

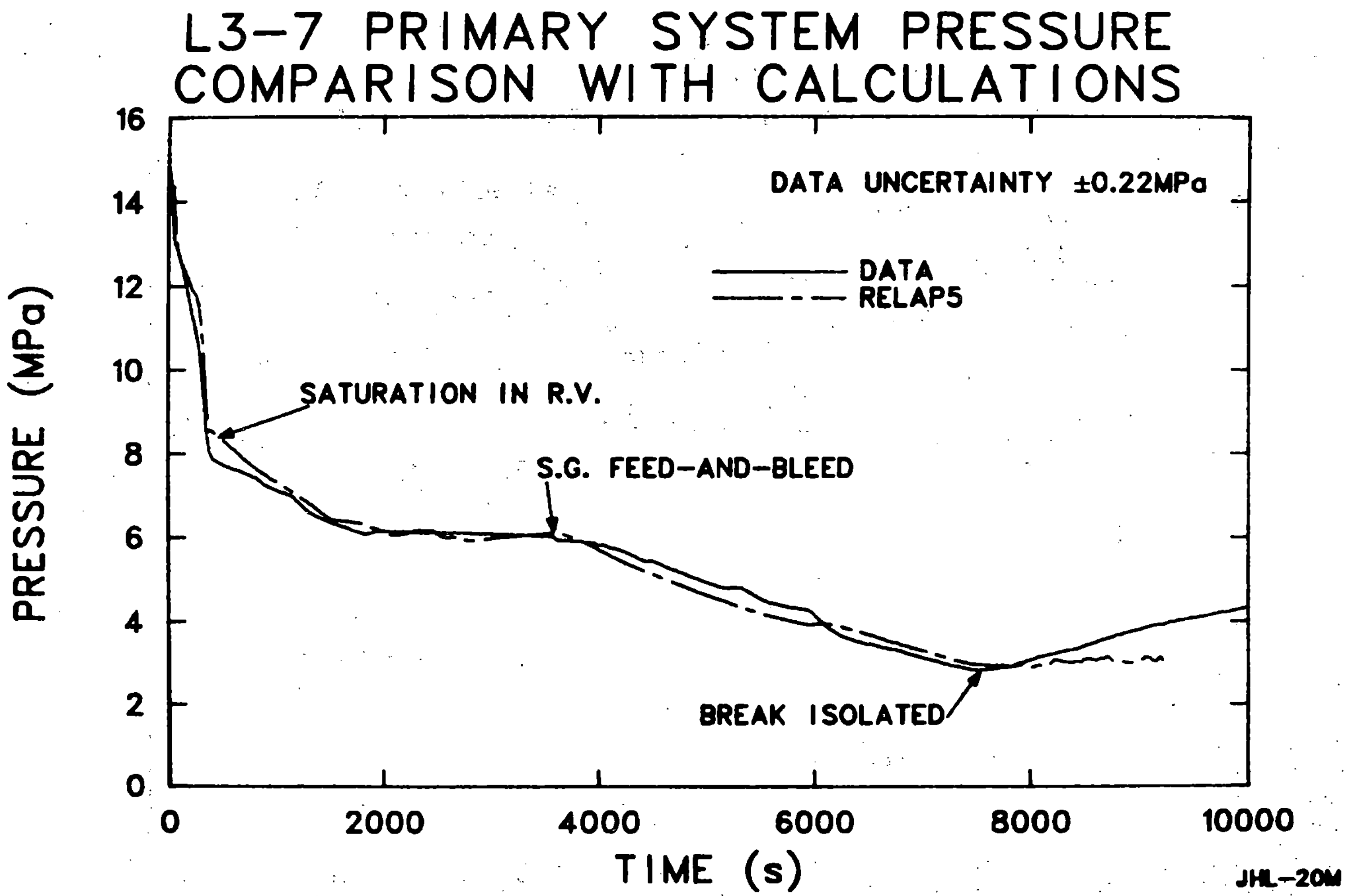


LOFT Primary System

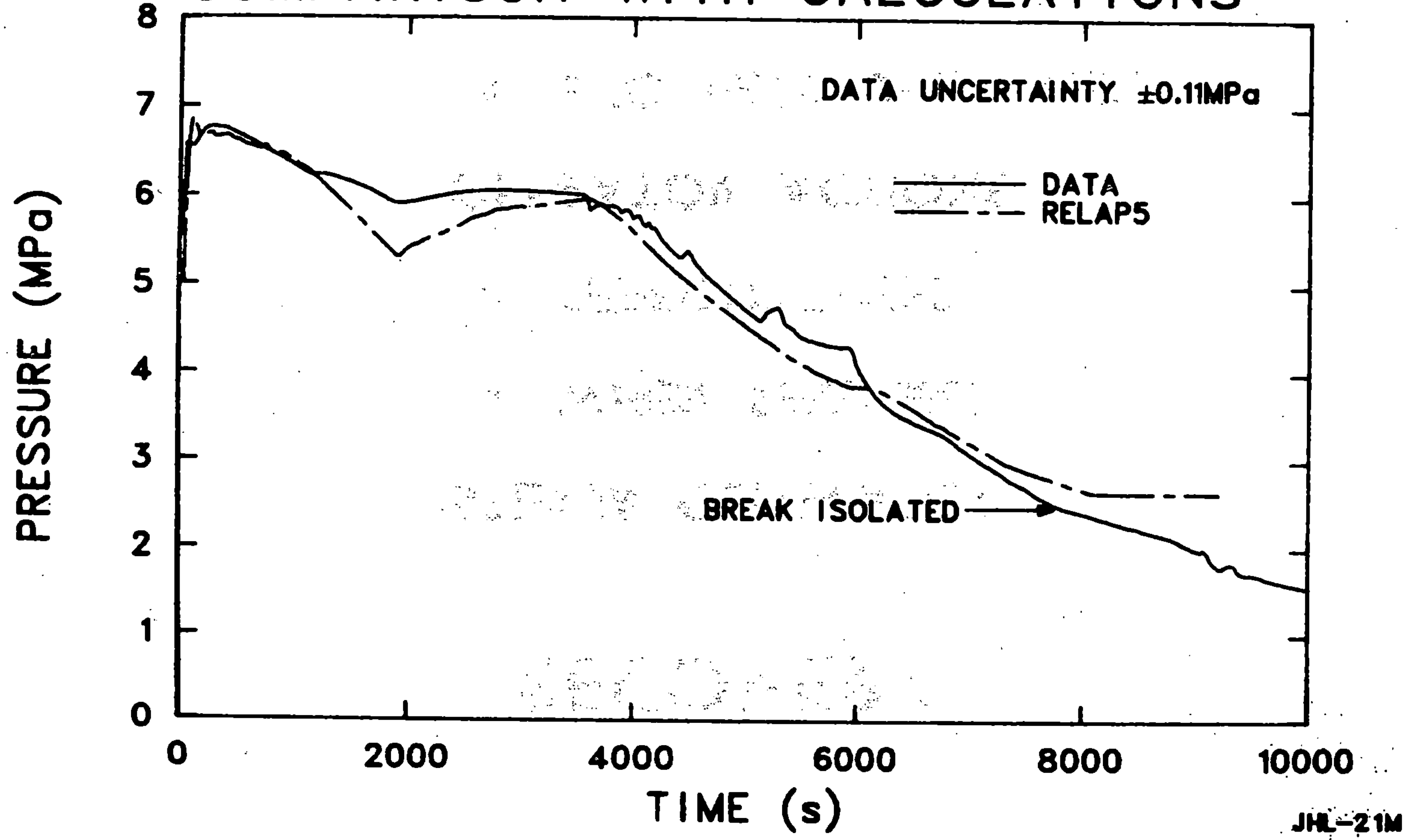


INEL-S-28 400


L3-7 PRIMARY SYSTEM PRESSURE
UNCERTAINTY ± 0.22 MPa
REACTOR SCRAMMED
PUMPS TRIPPED
HPIS ON
UPPER PLENUM
SATURATED
S.G. SECONDARY
FEED & BLEED
PRESSURIZER
REFILL STARTED
PRIMARY
BLEED
HPIS OFF
HPIS ON
ACCUMULATOR ON
BREAK ISOLATED
R.V. UPPER
HEAD REFILLED
PRESSURE (MPa)
0
5
10
15
20
0
3000
6000
9000
12000
15000
TIME (s)
JHL-1M


L3-7 PRIMARY SYSTEM PRESSURE
COMPARISON WITH CALCULATIONS
DATA UNCERTAINTY ±0.22MPa
DATA
RELAP5
SATURATION IN R.V.
S.G. FEED-AND-BLEED
BREAK ISOLATED
PRESSURE (MPa)
TIME (s)
0
2
4
6
8
10
12
14
16
0
2000
4000
6000
8000
10000
JHL-20M

L3-7 SECONDARY SYSTEM PRESSURE COMPARISON WITH CALCULATIONS



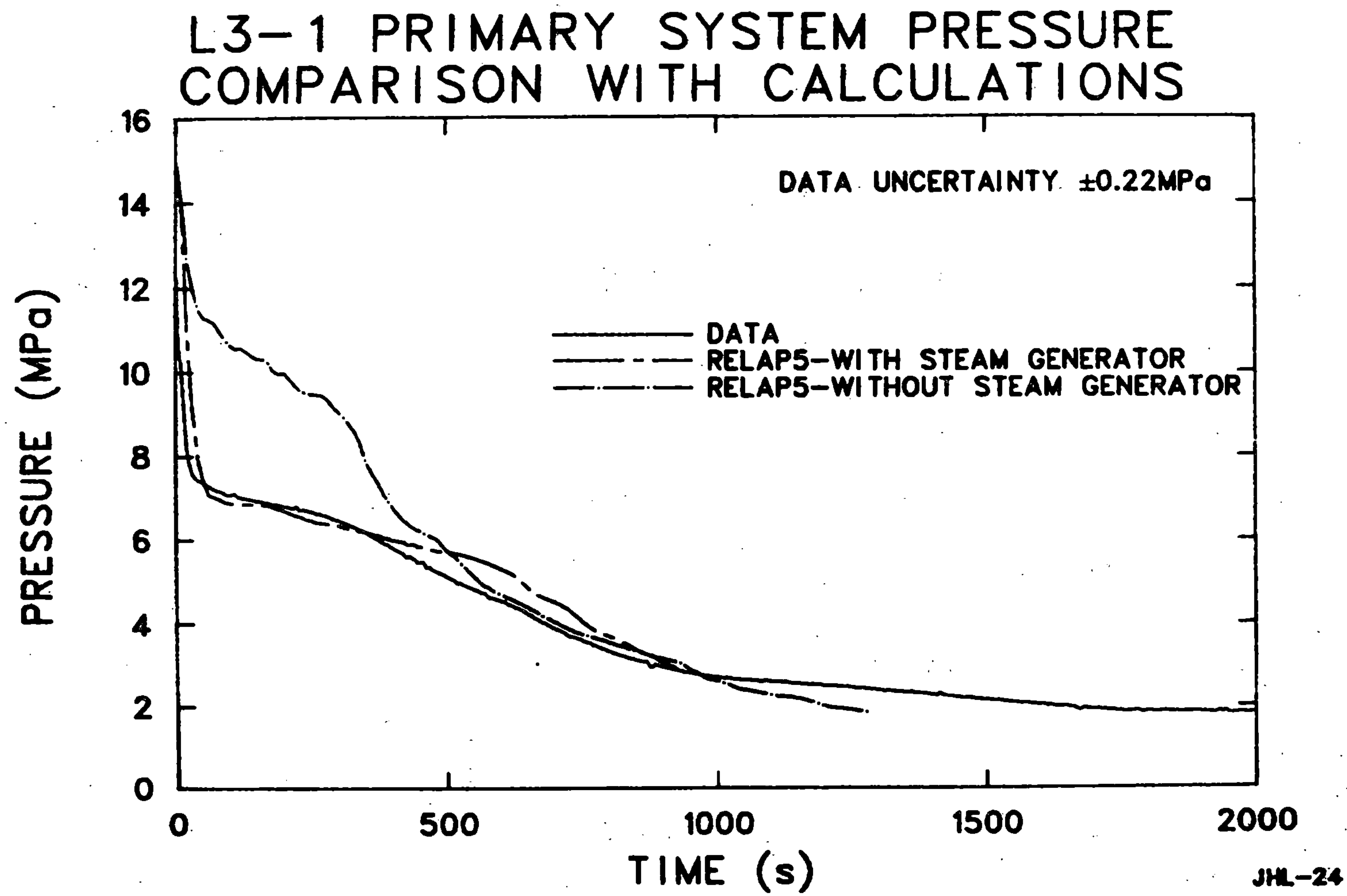
RECOVERY

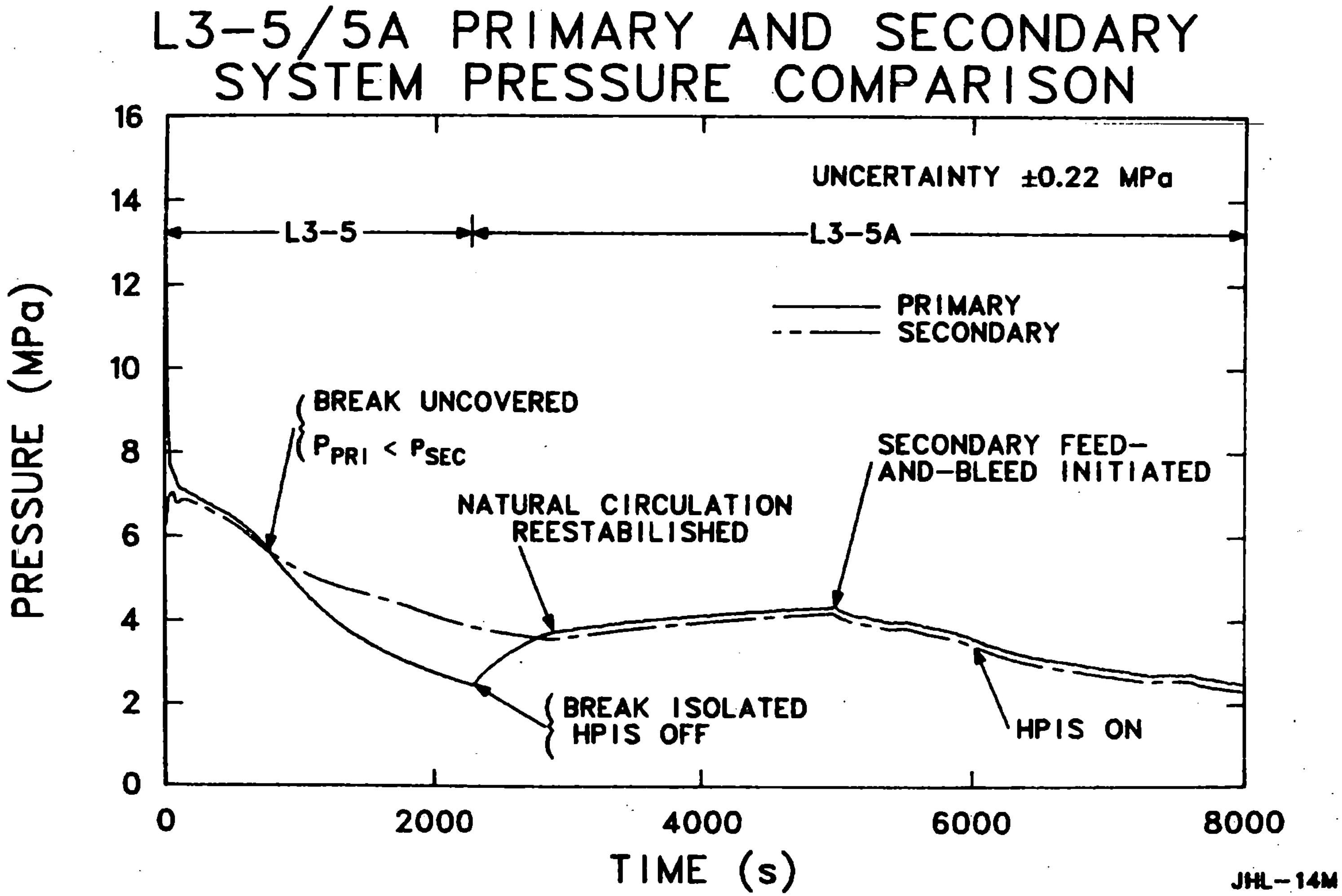
STEAM GENERATOR

- WHEN REQUIRED
- EFFECTIVENESS

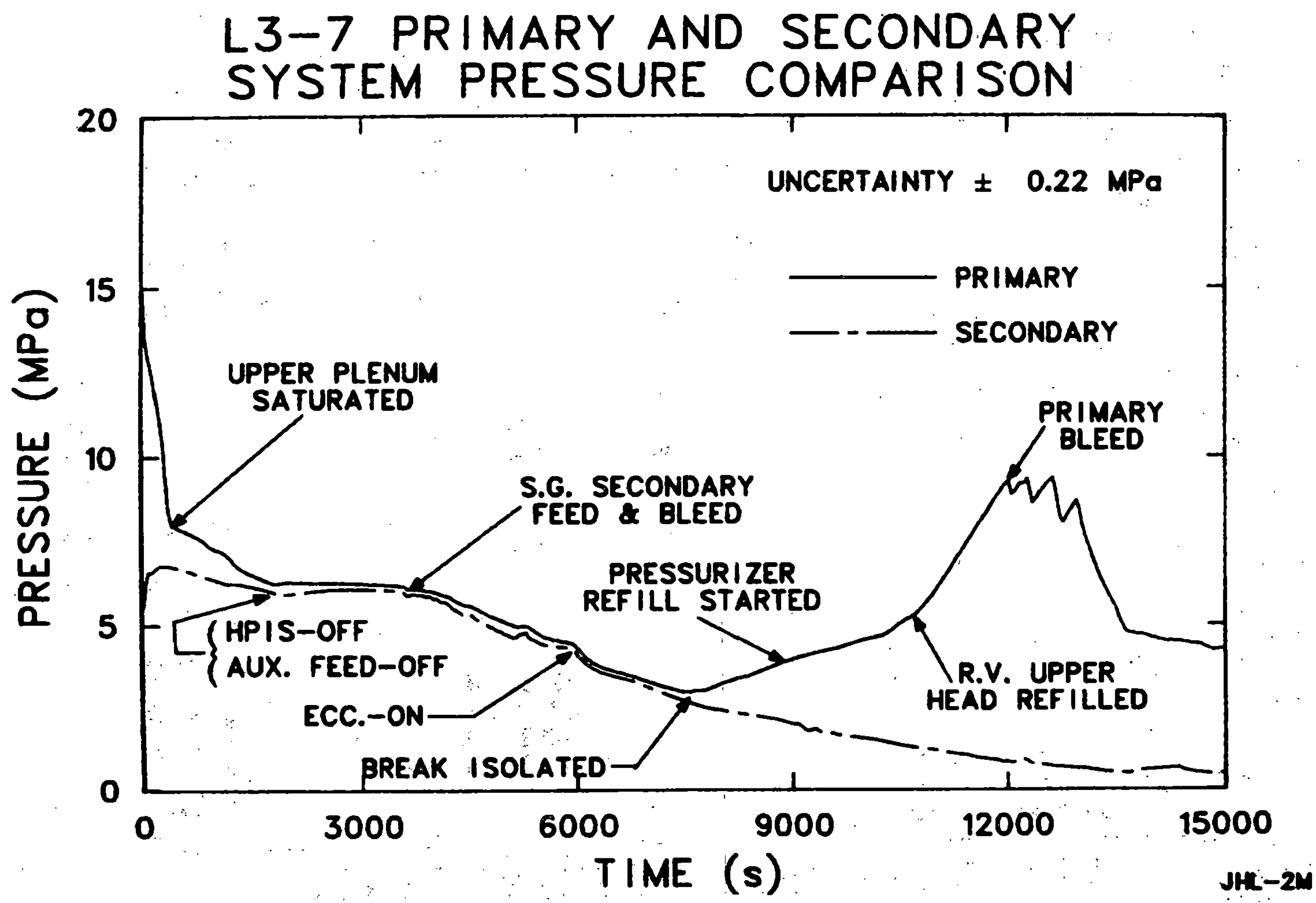
OPERATOR ACTIONS

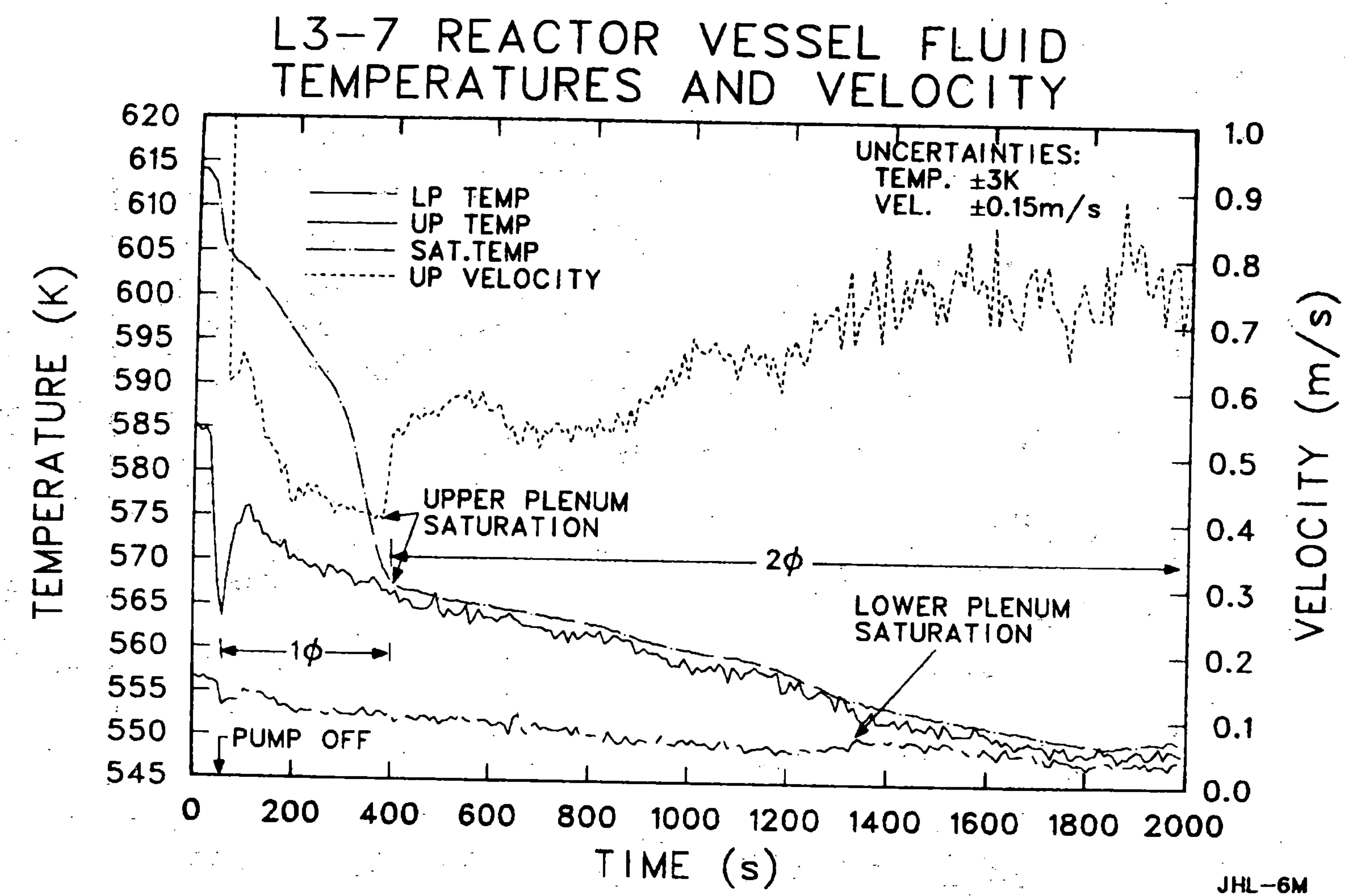
- S/G FEED AND BLEED


L3-1 PRIMARY SYSTEM PRESSURE
COMPARISON WITH CALCULATIONS
DATA UNCERTAINTY ±0.22MPa
DATA
RELAP5-WITH STEAM GENERATOR
RELAP5-WITHOUT STEAM GENERATOR
PRESSURE (MPa)
16
14
12
10
8
6
4
2
0
0
500
1000
1500
2000
TIME (s)
JHL-24

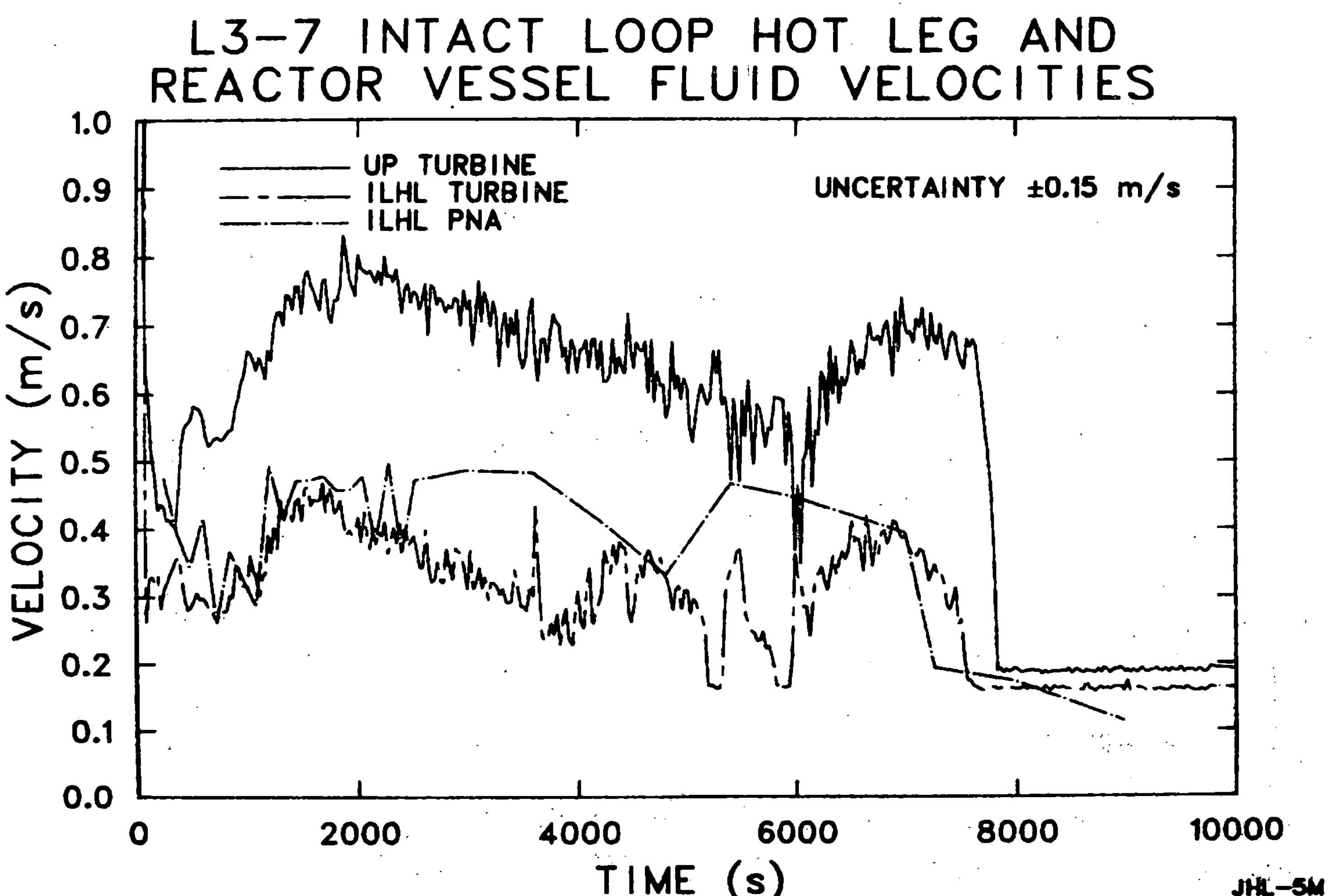


L3-5/5A PRIMARY AND SECONDARY
SYSTEM PRESSURE COMPARISON
UNCERTAINTY ±0.22 MPa
L3-5
L3-5A
PRIMARY
SECONDARY
BREAK UNCOVERED
$P_{PRI} < P_{SEC}$
NATURAL CIRCULATION
REESTABILISHED
SECONDARY FEED-
AND-BLEED INITIATED
BREAK ISOLATED
HPIS OFF
HPIS ON
PRESSURE (MPa)
TIME (s)
0
2
4
6
8
10
12
14
16
0
2000
4000
6000
8000
JHL-14M

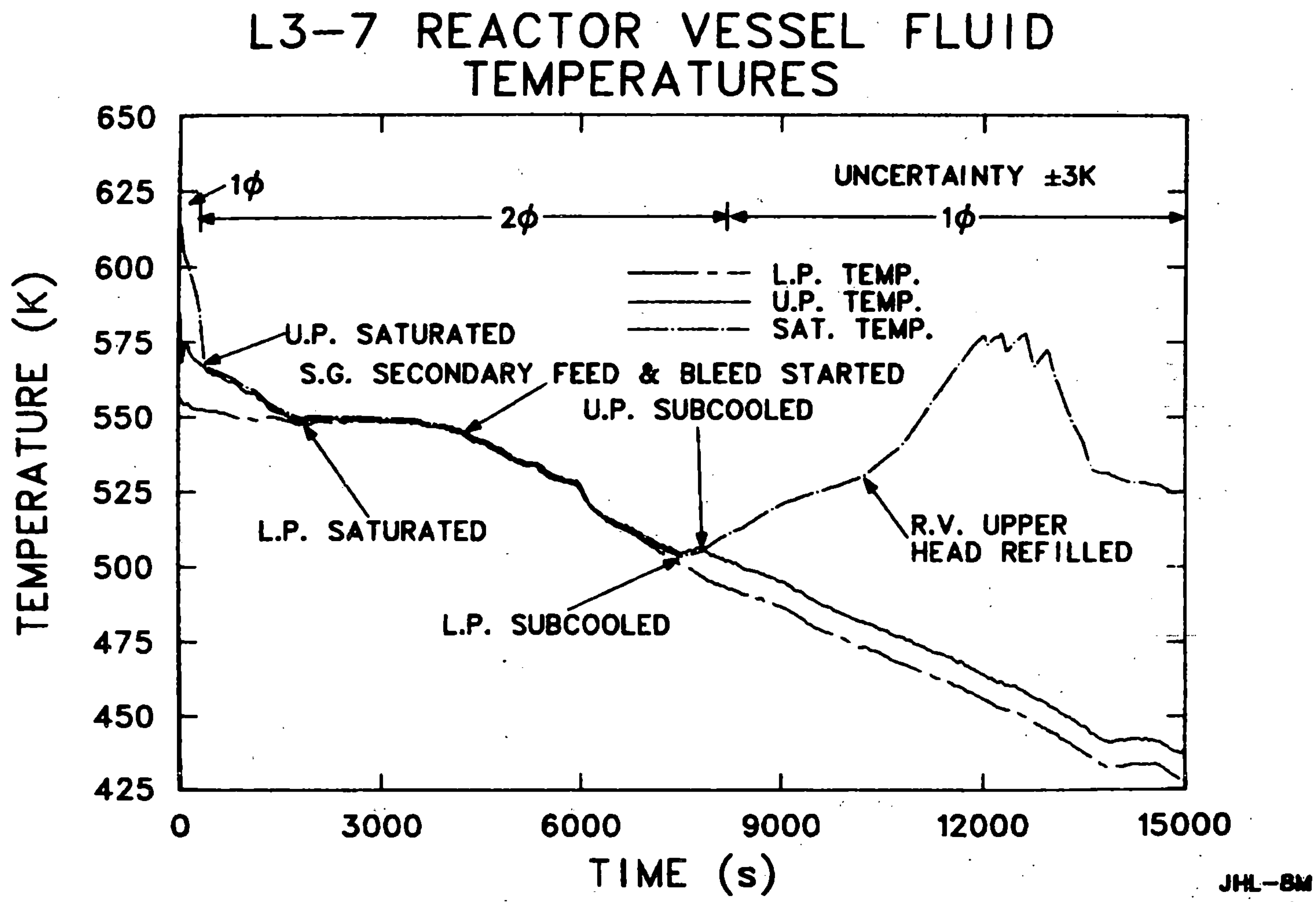

L3-7 PRIMARY AND SECONDARY
SYSTEM PRESSURE COMPARISON
UNCERTAINTY ± 0.22 MPa
PRIMARY
SECONDARY
UPPER PLENUM SATURATED
S.G. SECONDARY FEED & BLEED
PRIMARY BLEED
PRESSURIZER REFILL STARTED
HPIS-OFF
AUX. FEED-OFF
ECC.-ON
BREAK ISOLATED
R.V. UPPER HEAD REFILLED
PRESSURE (MPa)
TIME (s)
0
5
10
15
20
0
3000
6000
9000
12000
15000
JHL-2M



L3-7 REACTOR VESSEL FLUID
TEMPERATURES AND VELOCITY
LP TEMP
UP TEMP
SAT.TEMP
UP VELOCITY
UNCERTAINTIES:
TEMP. ±3K
VEL. ±0.15m/s
UPPER PLENUM
SATURATION
LOWER PLENUM
SATURATION
1ϕ
2ϕ
PUMP OFF
TEMPERATURE (K)
VELOCITY (m/s)
TIME (s)
620
615
610
605
600
595
590
585
580
575
570
565
560
555
550
545
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
0
200
400
600
800
1000
1200
1400
1600
1800
2000
JHL-6M

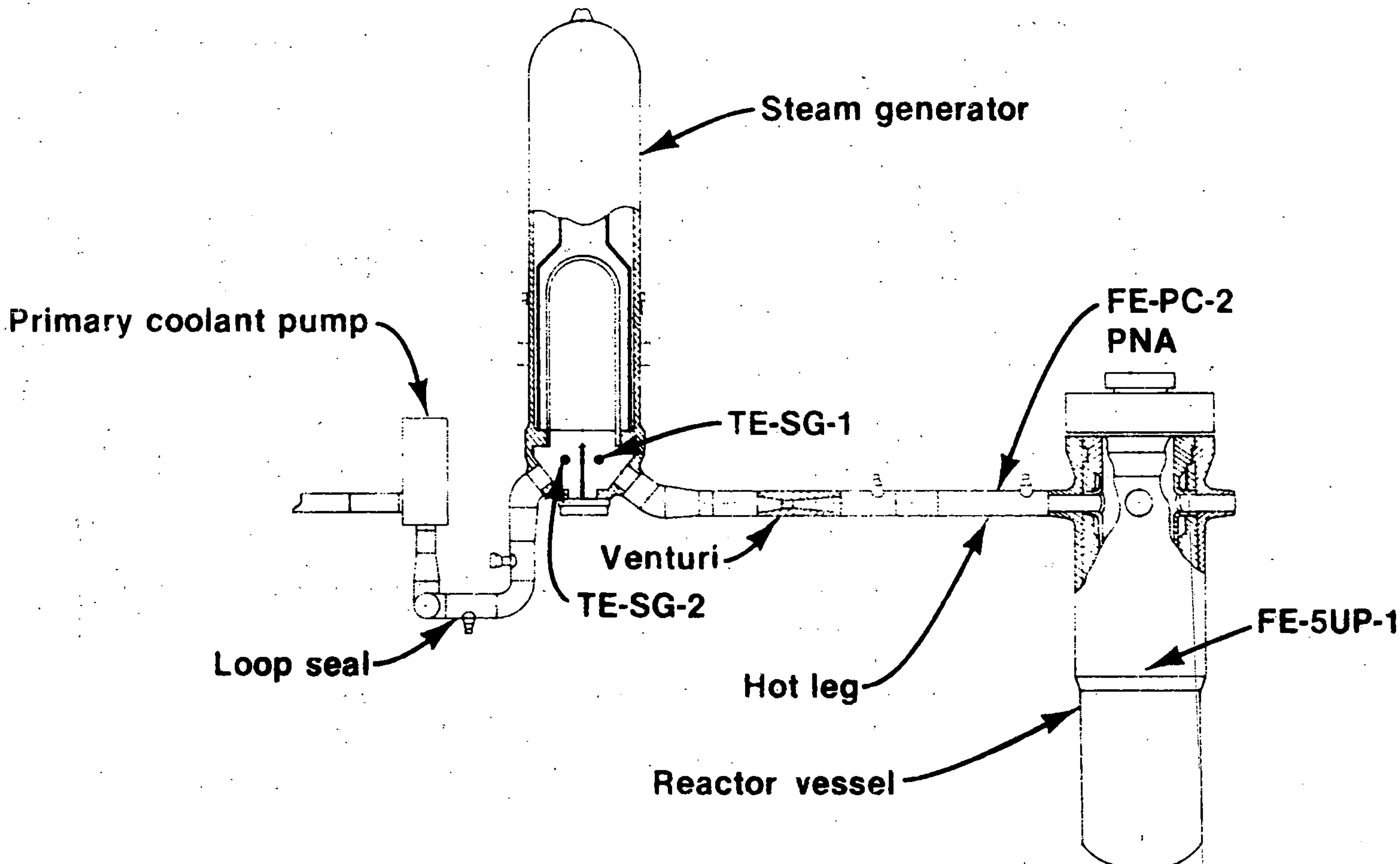


L3-7 INTACT LOOP HOT LEG AND
REACTOR VESSEL FLUID VELOCITIES
UP TURBINE
ILHL TURBINE
ILHL PNA
UNCERTAINTY ±0.15 m/s
VELOCITY (m/s)
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
0
2000
4000
6000
8000
10000
TIME (s)
JHL-5M

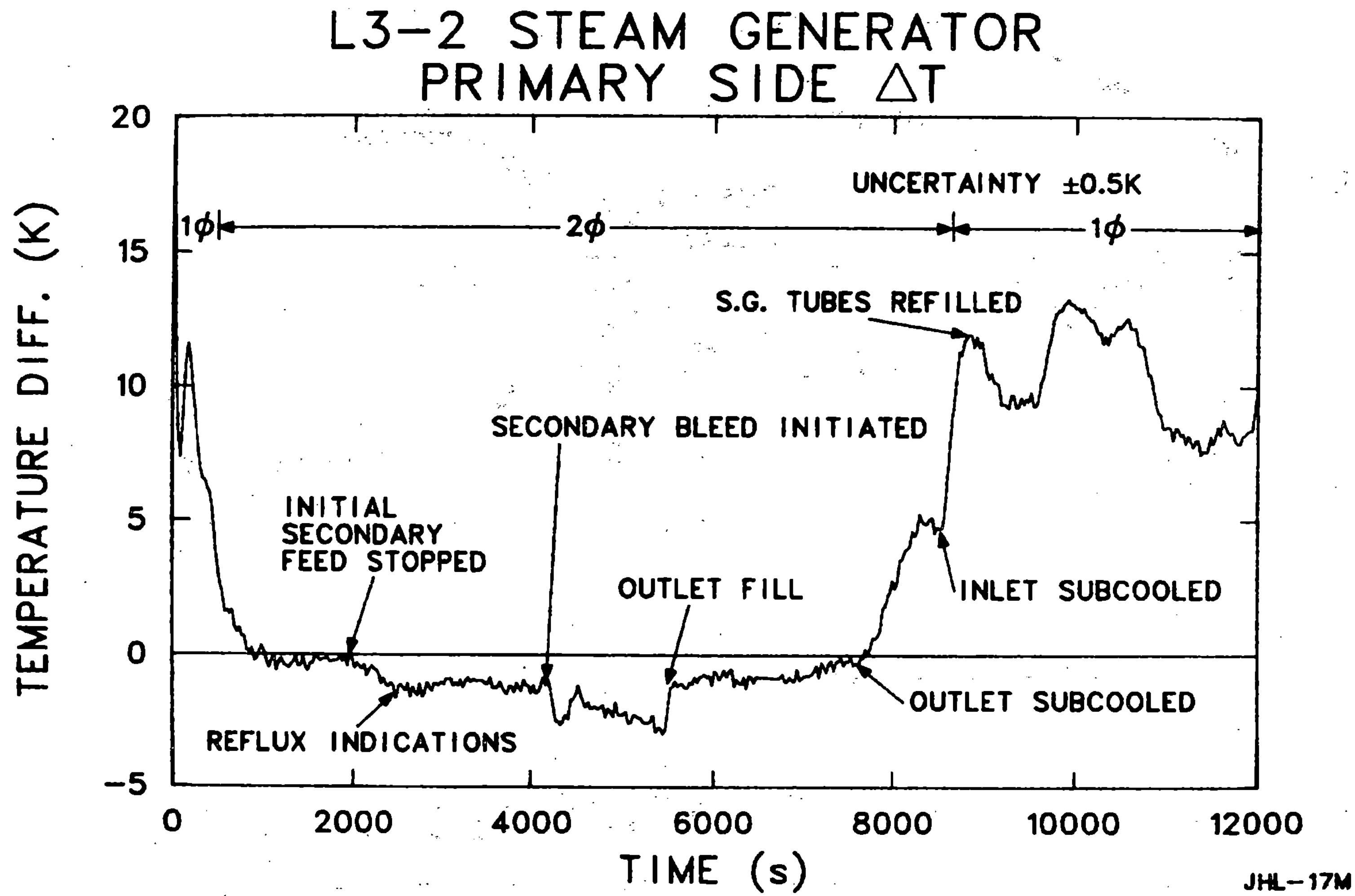


L3-7 REACTOR VESSEL FLUID TEMPERATURES
650
625
600
575
550
525
500
475
450
425
TEMPERATURE (K)
0
3000
6000
9000
12000
15000
TIME (s)
UNCERTAINTY ±3K
1ϕ
2ϕ
1ϕ
L.P. TEMP.
U.P. TEMP.
SAT. TEMP.
U.P. SATURATED
S.G. SECONDARY FEED & BLEED STARTED
U.P. SUBCOOLED
L.P. SATURATED
R.V. UPPER HEAD REFILLED
L.P. SUBCOOLED
JHL-BM

Selected Measurement Locations - LOFT Intact Loop

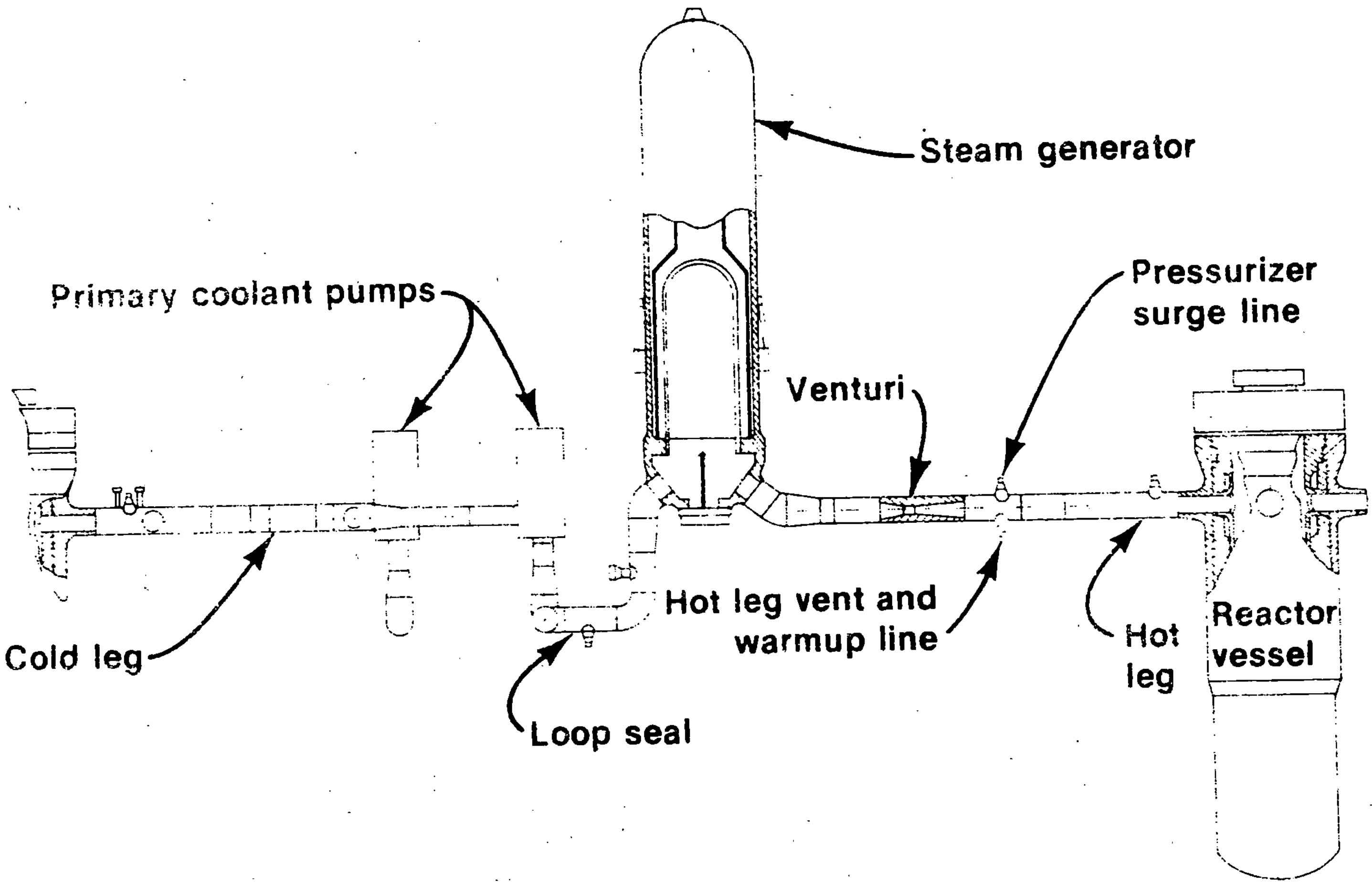


INEL-S-25 495-2

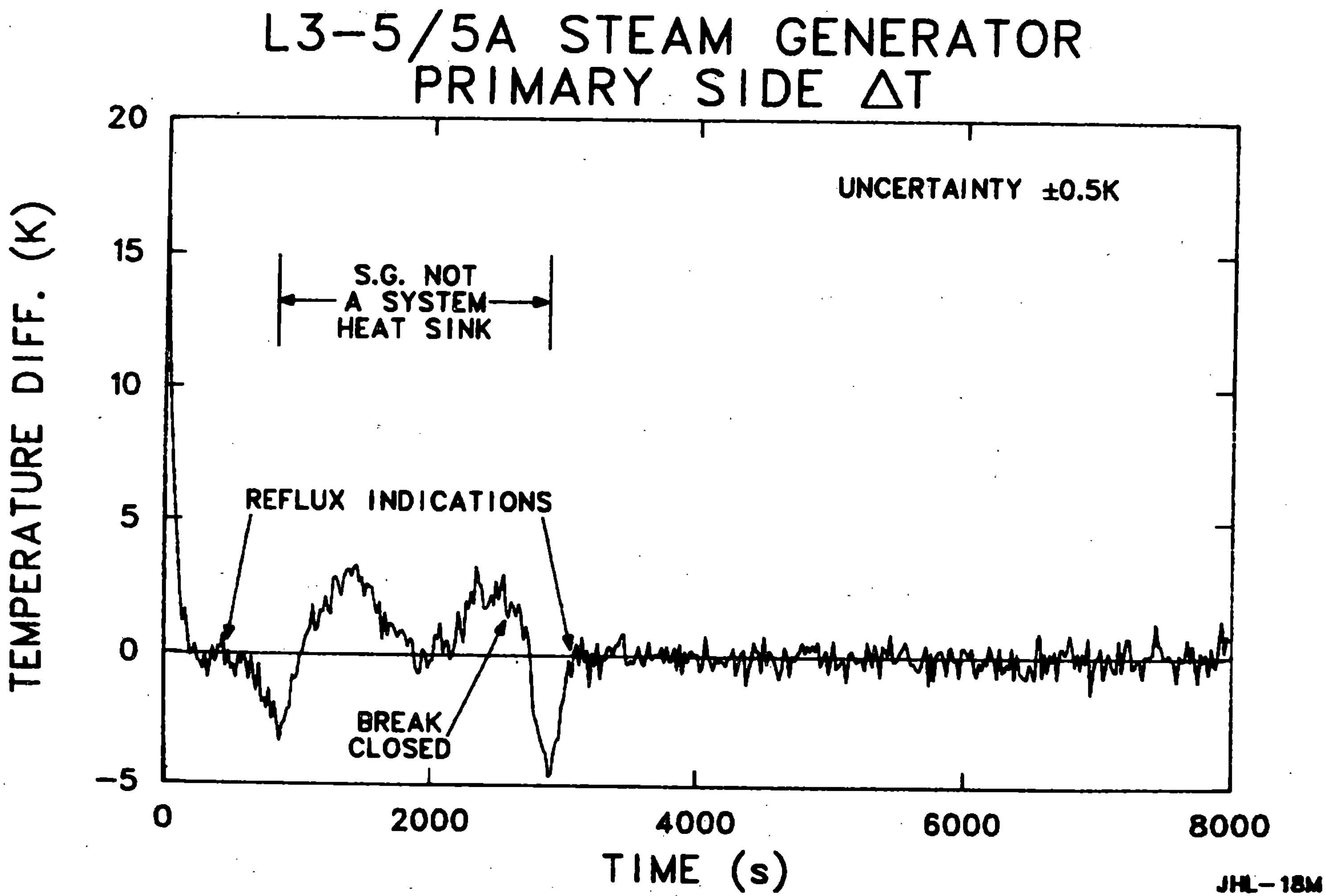


L3-2 STEAM GENERATOR
PRIMARY SIDE ΔT
UNCERTAINTY ±0.5K
1ϕ
2ϕ
1ϕ
S.G. TUBES REFILLED
SECONDARY BLEED INITIATED
INITIAL
SECONDARY
FEED STOPPED
OUTLET FILL
INLET SUBCOOLED
OUTLET SUBCOOLED
REFLUX INDICATIONS
TEMPERATURE DIFF. (K)
TIME (s)
20
15
10
5
0
−5
0
2000
4000
6000
8000
10000
12000
JHL−17M

LOFT Intact Loop

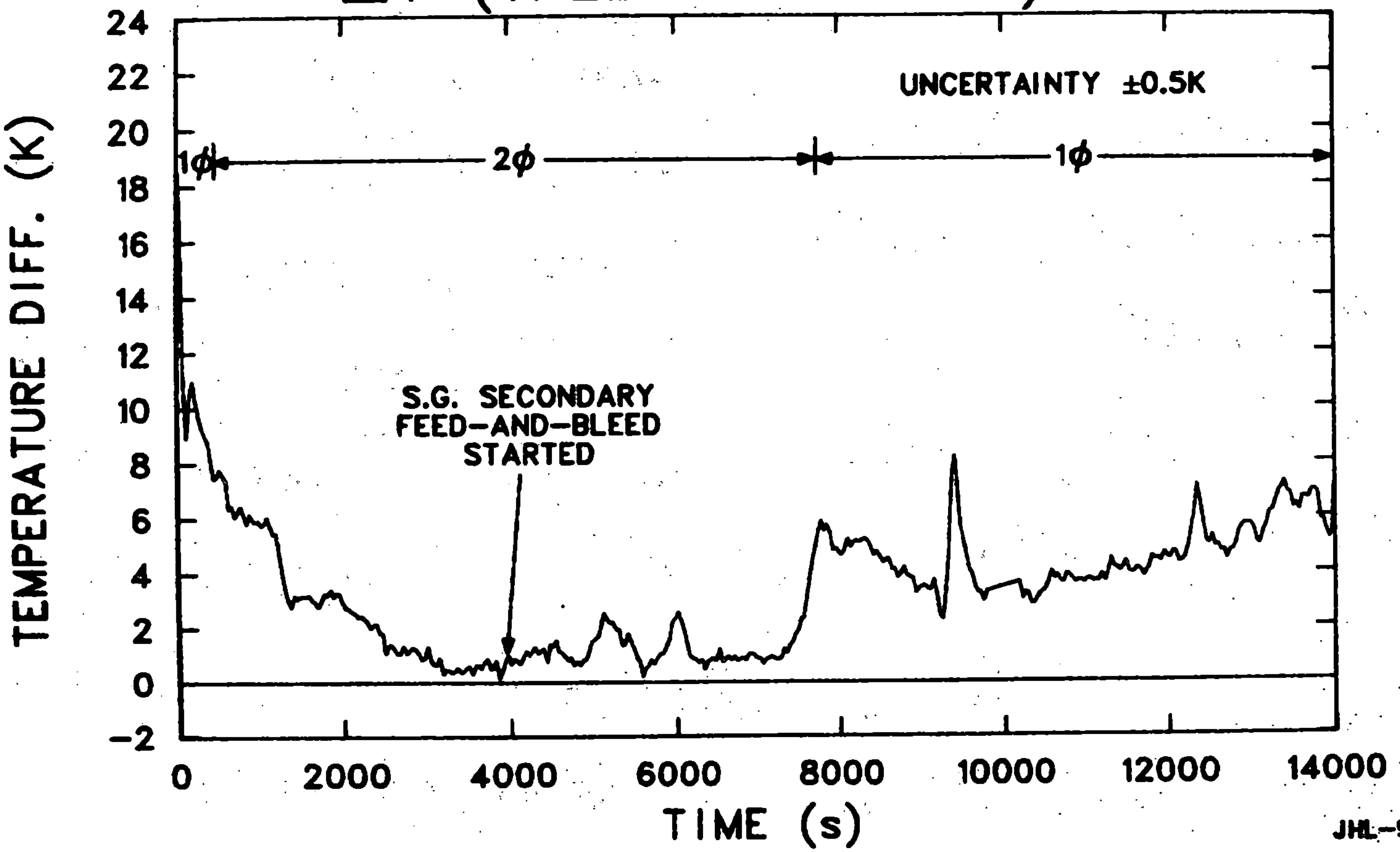


INEL-S-25 495-1



L3-5/5A STEAM GENERATOR
PRIMARY SIDE ΔT
UNCERTAINTY ±0.5K
S.G. NOT A SYSTEM HEAT SINK
REFLUX INDICATIONS
BREAK CLOSED
TEMPERATURE DIFF. (K)
20
15
10
5
0
−5
0
2000
4000
6000
8000
TIME (s)
JHL-18M

L3-7 STEAM GENERATOR PRIMARY COOLANT ΔT (INLET - OUTLET)



JHL-9M

NATURAL CIRCULATION CONCLUSIONS

- OCCURS IN SINGLE AND TWO-PHASE MODES
- STABLE IN AND BETWEEN MODES
- REVERSIBLE
- REESTABLISHABLE
- NOT DETERRED BY
  - ECC
  - R.V. VOIDING
  - NON CONDENSIBLES

LICENSING CONCLUSIONS

- PWR AND LOFT SCENARIOS COMPARABLE
- CALCULATIONS PREDICT DOMINANT TRANSITIONS AND ASSOCIATED PHENOMENA IN PROPER TIME SEQUENCE
- RECOVERY PROCESS IS CONVERGENT

JHL-9

LICENSING CONCLUSIONS (CONT'D)

- STEAM GENERATOR IS EFFECTIVE HEAT SINK

- OPERATOR INITIATED STEAM GENERATOR SECONDARY FEED-AND-BLEED EXPEDITES RECOVERY

JHL-9A

LOFT: A NUCLEAR PLANT PROVIDING REALISTIC ANSWERS TO PWR LICENSING ISSUES

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

C. W. Solbrig
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

LOFT: A NUCLEAR PLANT PROVIDING REALISTIC ANSWERS TO PWR LICENSING ISSUES

Charles W. Solbrig

EG&G Idhao, Inc.

The following topics are discussed in this presentation: (a) the LOFT mission, (b) LOFT results which have been used by NRC's office of Nuclear Reactor Regulation (NRR), (c) NRR current LOFT information needs, and (d) the channels of LOFT results dissemination.

LOFT is an operating nuclear power plant which is periodically subjected to severe accident conditions to determine the consequences of such conditions. The LOFT mission and program is determined by several sources with the primary source being NRR needs. NRR requests are used by Reactor Safety Research (RSR) to plan the basic program. The Code of Federal Regulations, such as 10 CFR 50 Appendix K, supply experimental requirements. Inspection and Enforcement bulletins issued by the NRC to nuclear power plant owners as well as other NRR information sources are used to supply input to the program. The LOFT review group, vendors, utilities, and other experts assist in the program development. And finally, the LOFT program continues to undergo reappraisal by the LOFT staff. All of this planning activity ensures that the most valuable experiments will be performed. At the current time, an event tree approach is being pursued to determine if the experiments in the LOFT schedule do, in fact, cover the entire range of experiments that are needed.

Various licensing requirements, issues and concerns have been answered by LOFT in the past. For example, LOFT experiments have shown it is justifiable to treat the core annulus region analytically in such a way that water can flow down one side of the annulus while steam flows up the other. This issue is important in the analysis of the double ended cold leg break. Another issue associated with this type of transient is fuel rod rewet. LOFT results showed that in a large break rewet occurs early in the transient thus causing a large amount of stored energy to be removed earlier than expected. LOFT results have also shown thus far that the temperatures observed during reflood are lower than the temperatures observed during the blowdown portion of the transient. Thus, these three examples of conservative requirements included in Appendix K for licensing were verified in the LOFT experiments to be very conservative.

Other issues investigated by LOFT include the following: LOFT has been on the forefront of measurement of water level in reactor vessels and continues to do experiments which help improve the instruments which are used for this purpose. LOFT has developed a thermocouple which can be used to measure the maximum cladding temperature on real fuel. Small break experiments have been run in LOFT in response to regulatory requests as a result of Three Mile Island. The different modes of natural circulation and transitions between these modes have been found to be reversible and stable. The different methods of heat removal in small breaks which show the trade-off between removing heat through the break, through the steam generator, or through the feed-and-bleed system have all been investigated.

Some licensing issues which LOFT is actively pursuing at the current time include the effect of whether pumps should be left on or turned off when a small break has occurred, the effect of pressurized fuel during a loss-of-coolant accident, and the investigation of anticipated transients with multiple failures.

Information from LOFT is transmitted in several ways. These include reports on experiments which are used as standard problems, answers to specific requests from NRR, and reports which are published on each experiment such as Quick Look Reports, Experimental Data Reports, and Posttest Analysis Reports. In addition, research information letters are issued which contain summaries of our research results. LOFT continues to be an important facility in the determination of information needed by NRR. The LOFT facility is an operating nuclear plant which has most of the problems and interfaces present in an operating power plant and thus it is capable of supplying information on operation, design, and accident situations in nuclear power plants.

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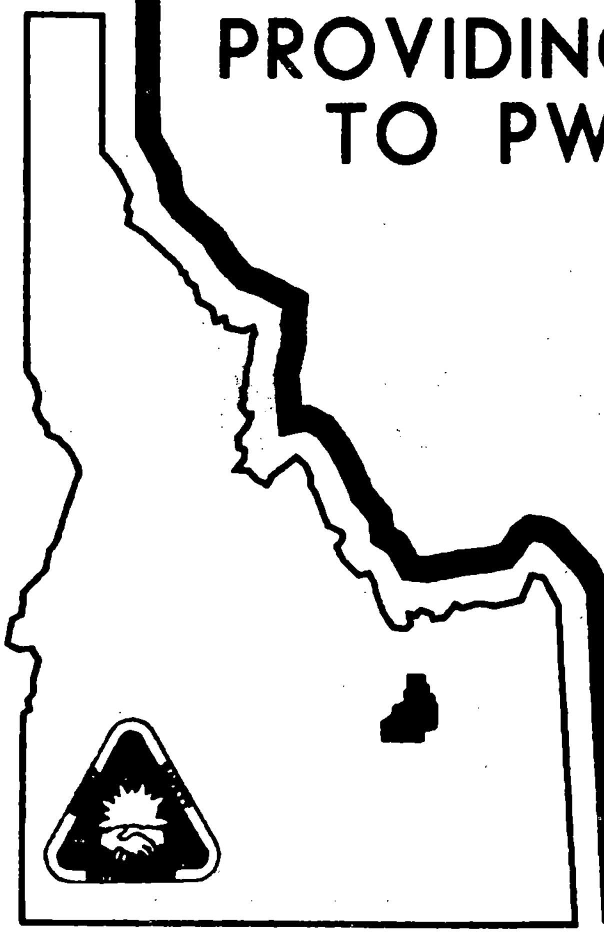
LOFT: A NUCLEAR PLANT PROVIDING REALISTIC ANSWERS TO PWR LICENSING ISSUES



by
DR. C.W. SOLBRIG

OUTLINE

- LOFT AND ITS MISSION
- PAST RESULTS USED BY LICENSING
- CURRENT LICENSING NEEDS
- DISSEMINATION OF LOFT RESULTS

CWS-1

RELEVANCE TO LICENSING

LOFT ADDRESSES IMPORTANT ISSUES

- LICENSING REQUESTS TO RESEARCH (RSR)
- 10 CFR 50 APPENDIX K
- I&E BULLETINS AND OTHER NRC INFORMATION SOURCES
- DIRECT CONTACT WITH THE VENDORS
- LOFT REVIEW GROUP
- DISCOURSE WITH THE UTILITIES
- CONTINUAL PROGRAM DEVELOPMENT (e.g. EVENT TREE APPROACH)

CWS-2

LOFT OBJECTIVES

- PROVIDE DATA TO VERIFY COMPUTER CODES
- EXPERIMENTALLY IDENTIFY IMPORTANT PHENOMENA
- PROVIDE DESIGN INFORMATION ON OPERATOR/PLANT INTERFACE

LOFT CHARACTERISTICS

- OPERATING NUCLEAR PLANT (50 MWt)
- PROTOTYPICALLY SCALED TO A PWR
- EXTENSIVELY INSTRUMENTED
- ONLY NUCLEAR PROTOTYPE IN THE WORLD
- TECHNICAL AND SUPPORT STAFF REPRESENTATIVE OF A NUCLEAR PLANT
- DESIGN, LICENSING, AND OPERATING PWR ISSUES

CWS-4

LICENSING REQUIREMENTS

APPENDIX K

- ALL ECC MUST BYPASS THE REACTOR VESSEL UNTIL DOWNWARD STEAM FLOW OCCURS IN THE ANNULUS

LOFT CONTRIBUTION

1. LIQUID CAN FLOW DOWN ONE SIDE OF THE ANNULUS AND STEAM UP THE OTHER

2. MEASUREMENT OF AMOUNT OF ECC BYPASS WATER

CWS-5

LICENSING ISSUE

- ALL EXPERIMENTAL FACILITIES WHICH ARE SCALED DOWN VERSIONS OF PWRS YIELD NON-USABLE RESULTS BECAUSE OF DELAY CAUSED BY STEAM GENERATED IN THE ANNULUS

LOFT CONTRIBUTION

THIS EFFECT (HOT WALL DELAY) IS NEGLIGIBLE IN LOFT

LICENSING REQUIREMENT

APPENDIX K

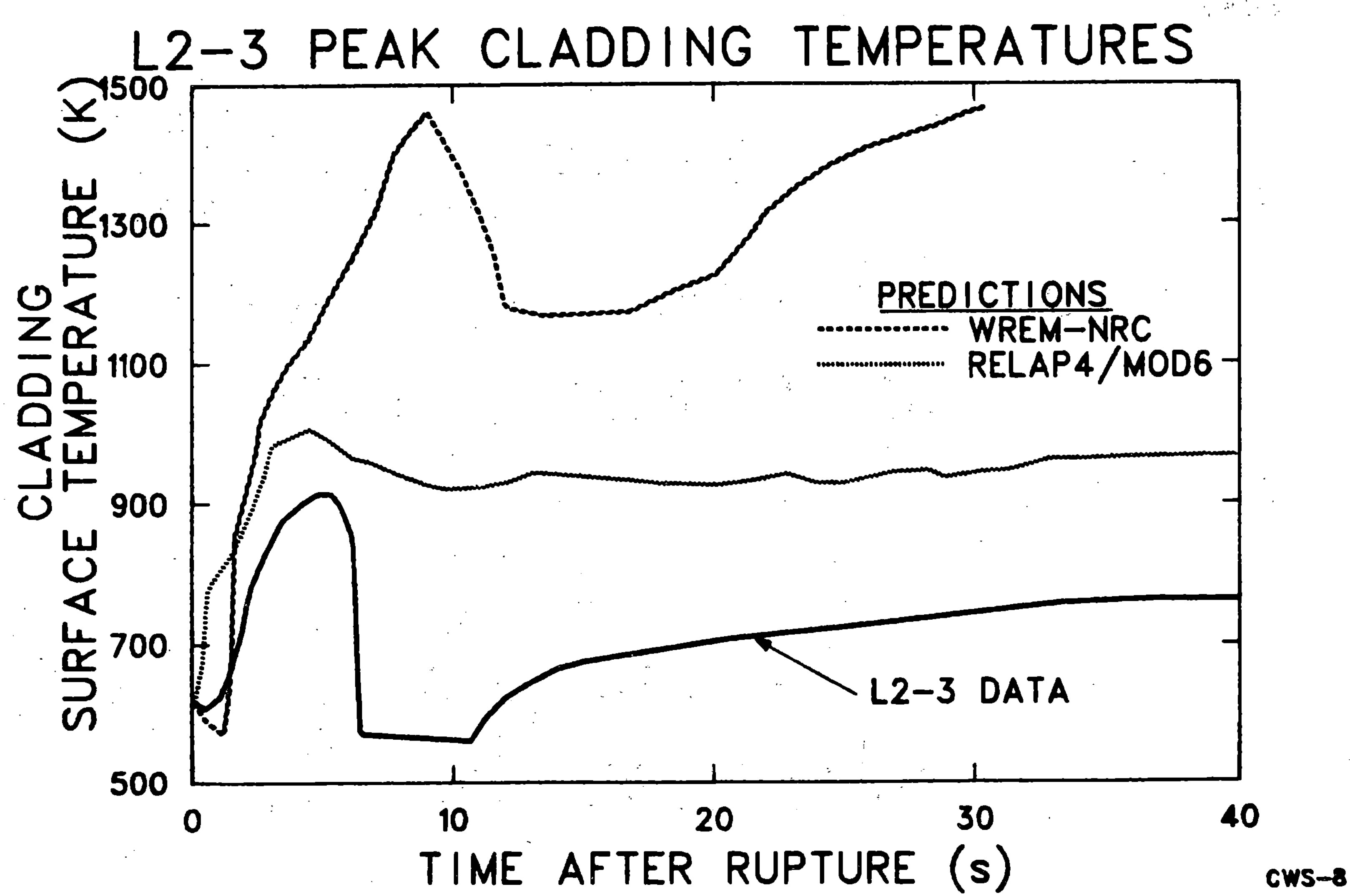
- LICENSING COMPUTER CODES CANNOT ALLOW FUEL ROD REWET. ALL REWET CONDITIONS ARE VERY CONSERVATIVE

LOFT CONTRIBUTION

1. REWET OCCURS QUITE EASILY AND QUICKLY

2. REWET PHENOMENA IS A STRONG HYDRAULIC EFFECT WHICH IS EVEN MORE PREDOMINANT IN A PWR

CWS-7



L2-3 PEAK CLADDING TEMPERATURES
CLADDING
SURFACE TEMPERATURE (K)
1500
1300
1100
900
700
500
0
10
20
30
40
TIME AFTER RUPTURE (s)
PREDICTIONS
WREM-NRC
RELAP4/MOD6
L2-3 DATA
CWS-8

IMPLICIT LICENSING REQUIREMENT

APPENDIX K

- THE MAXIMUM TEMPERATURE OCCURS DURING THE REFLOOD PORTION OF A LOCA WHICH ESTABLISHES THE CORE POWER LIMIT

LOFT CONTRIBUTION

1. EARLY REWETS DO NOT ALLOW A CORE DRYOUT AND ADIABATIC CONDITION TO OCCUR

2. IMPLIES THAT CURRENT LICENSING CODES ARE VERY CONSERVATIVE

CWS-9

LICENSING - I&E BULLETIN

- ALL REACTORS MUST HAVE A LIQUID LEVEL DEVICE INSTALLED ON THE REACTOR VESSEL BY JANUARY 1, 1981

LOFT CONTRIBUTION

1. LIQUID LEVEL TECHNOLOGY DEVELOPED BY LOFT REQUESTED BY NRC TO BE USED FOR EVALUATING SUCH DEVICES

2. LOFT WILL TEST A LIQUID LEVEL DEVICE

3. LOFT IS ONLY FACILITY FOR TESTING UNDER OPERATING PLANT CONDITIONS

CWS-10

LICENSING REQUIREMENT

- PRIMARY CRITERION IS CLADDING TEMPERATURE - NO OPERATING POWER REACTORS MEASURE CLADDING TEMPERATURE

- FEAR OF FUEL FAILURE

LOFT CONTRIBUTION

1. LOFT CLADDING THERMOCOUPLES DEVELOPED BY FUEL MANUFACTURER

2. POTENTIAL APPLICATIONS IN PWRS

3. USED IN ALL LOFT EXPERIMENTS - NO FUEL FAILURE

CWS-11

LICENSING - TMI CONCERNS

- WHAT ARE THE METHODS OF HEAT REMOVAL IN A SMALL BREAK?

LOFT CONTRIBUTION

STEAM GENERATOR HAS BEEN SHOWN TO BE AN EFFECTIVE MODE OF HEAT REMOVAL FOR SMALL BREAK SIZES

CWS-12

FUTURE WORK

LICENSING REQUIREMENTS

- WILL EARLY CORE REWET OCCUR UNDER ALL CONDITIONS?
- IS IT POSSIBLE TO ENTER INTO A CORE REFLOOD CONDITION?
- DETERMINE DEGREE OF CONSERVATISM IN APPENDIX K REFLOOD CALCULATIONS

LOFT WILL DETERMINE IF THESE CONDITIONS CAN OCCUR AND PERFORM APPROPRIATE EXPERIMENTS

CWS-13

FUTURE WORK

LICENSING REQUIREMENT

- CAN PRESSURIZED FUEL CAUSE AN UNCOOLABLE GEOMETRY OR EXCESSIVELY HIGH FUEL TEMPERATURES IN A DESIGN BASIS ACCIDENT OR OTHER CONDITION?

1. LOFT WILL TEST PRESSURIZED FUEL UNDER APPENDIX K REQUIREMENTS
2. A SEVERE CORE DAMAGE TEST SERIES WILL BE RUN IN LOFT TO DETERMINE IF A LARGE AMOUNT OF BLOCKAGE CAN OCCUR
3. PBF PROGRAM TESTS LIMITS
4. FEEDBACK BETWEEN LOFT AND PBF

CWS-14

FUTURE WORK

LICENSING I&E BULLETIN

- LICENSING CURRENTLY REQUIRES THE PUMPS TO BE TURNED OFF BY OPERATOR ACTION AT THE INITIATION OF A SMALL BREAK. DOES EARLY PUMP TRIP RETAIN MORE WATER IN THE REACTOR SYSTEM?

1. LOFT AND SEMISCALE COOPERATE TO ANSWER QUESTIONS

2. SEMISCALE HAS RUN SCOPING EXPERIMENTS

3. LOFT WILL RUN VERIFICATION EXPERIMENTS

CWS-15

FUTURE WORK

OPEN LICENSING ISSUES

- WHAT ARE THE EFFECTS OF MULTIPLE FAILURES SUCH AS ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) OR OTHER MULTIPLE FAILURES?

  LOFT WILL RUN A SERIES OF TESTS TO LOOK AT ATWS AND OTHER MULTIPLE FAILURES

CWS-16

FEEDBACK TO LICENSING

HOW DOES LOFT PROVIDE THE RESULTS TO LICENSING?

- STANDARD PROBLEMS L3-6
  QUALITY CONTROL FUNCTION
- LICENSING REQUESTS PUMPS/OFF
- REPORTS ON THE EXPERIMENTS EP, QLR, EDR, PAD
- RESEARCH INFORMATION LETTERS (RIL)

CWS-17

LOFT PROGRAM UNIQUENESS

- OPERATING, PROTOTYPICALLY SCALED PWR RESEARCH FACILITY
- SCIENTIFIC EXPERTISE COMBINED WITH EXPERIENCE IN OPERATING NUCLEAR PWR
- CAN BE SUBJECTED TO WIDE RANGE OF ABNORMAL CONDITIONS
- MAN/MACHINE INTERFACE CAN BE TESTED IN A NUCLEAR PLANT
- NO OTHER PROGRAM OR FACILITY IN THE WORLD PROVIDES THIS CAPABILITY FOR NRC

CWS-18

RESULTS OF THE LOFT ANTICIPATED TRANSIENTS EXPERIMENTS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

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RESULTS OF THE LOFT ANTICIPATED TRANSIENTS EXPERIMENTS

By

C. W. Solbrig

LOFT has recently completed four anticipated transient experiments. These experiments include (1) loss of load, (2) loss of flow, (3) excessive load increase, and (4) loss of feedwater. Each experiment was successfully completed and is briefly described in this presentation. The first part of this presentation describes why anticipated transient experiments are useful, and the second part describes the experimental results.

The anticipated transient experiments were performed primarily to provide a basis for calibrating the computer codes used to predict these types of transients. After the models in these codes are improved enough to describe these transients, they may then be used for predicting the course of anticipated transients with multiple failures, for example, ATWS, experiments to be performed in LOFT. The tests were non-trivial because several important phenomena were not predicted correctly in magnitude or time. These experiments will allow safety analysis report models for these type of transients to be evaluated. Anticipated transients are of interest because they are expected to occur in a power about once per year. The Three Mile Island incident has provided the need for increased simulator capability and which can only be met in the future with computer codes such as the RELAP5 and the RETRAN computer codes. These computer codes will have to represent operational transients, anticipated transients with multiple failures, small break and large break LOCAs and all transients in between. In order to accomplish this, all aspects of the plant must be represented including secondary side models, pressurizer heater and sprays and post-accident heat removal systems. Some of the questions which will have to be answered by these codes include determination of the correct operating procedure for a given situation, verification of current procedures, tech spec changes and information required in training programs for both operators and technical advisors in power plants.

LOFT is uniquely qualified to perform such experiments because it has most of the systems representative of a large nuclear plant. Small electrically heated systems usually do not represent multiple ECC trains, secondary side components and single failure proof components. In addition, small systems have large relative heat losses. Experiments which are performed in actual nuclear plants can be very helpful but the amount of information or instrumentation available in such a plant is usually insufficient for code verification. In addition, experiments performed in powerplants are usually not very severe and, therefore, do not test all aspects of the code adequately.

The dominant phenomena in these transients are related to primary coolant system (PCS) pressure response and the availability of a heat sink. The mass of the PCS is initially unchanging or increasing. The average temperature of the PCS results from the overall energy balance and determines the average specific volume. Changes in average specific volume determine pressurizer level which in conjunction with the automatic pressure control systems determines the PCS pressure.

The course of each transient was predicted prior to the experiments with the RETRAN computer code. Evaluation of the experimental results indicates the LOFT system response to these transients is not severe and that the LOFT automatic pressure and level control systems can effectively deal with the challenges issued by these transients. At all times during the experiments core cooling was sufficient to maintain the fuel rod cladding temperatures below the saturation temperature of the coolant. The operators were able to understand the course of the transients and respond appropriately in real time to return the plant to a stable controlled situation. Comparison of the experimental results with the RETRAN calculations revealed the major phenomena were predicted in the proper sequence, however, the magnitudes of some phenomena were not precisely calculated. Further analysis has shown the differences between the calculations and the data to come from the following sources: (1) steam generator secondary side feedwater and steaming flow rates, (2) pressurizer spray and heater operation, (3) thermal nonequilibrium between the pressurizer vapor and liquid during insurges and outsurges, and (4) main steam control valve leakage.

In summary, LOFT experiments have provided information useful to the understanding of anticipated transient behavior. The ability of the plant automatic control systems and the operators to recover the plant during transients not compounded by additional failures has been observed to be satisfactory in LOFT. Comparison of currently used analytical methods with the LOFT results has shown a generally good transient characterization with areas for improvement noted.

REFERENCES

1. R. P. Jordan, LOFT Experiment Operating Specification Non-LOCE Baseline Test Series L6, Rev. 1, October 5, 1980.

2. C. D. Keeler, Best Estimate Prediction for LOFT Nuclear Experiments L6-1, L6-2, L6-3, and L6-5, EGG-LOFT-5161, October 1980.

3. D. L. Reeder, Quick Look Report on LOFT Nuclear Experiment L6-5, EGG-LOFT-5165, June 1980.

4. D. L. Reeder, Quick Look Report on LOFT Nuclear Experiments 6-1, L6-2, and L6-3, EGG-LOFT-5270, October 1980.

LOFT ANTICIPATED TRANSIENT

EXPERIMENTAL RESULTS

C. W. SOLBRIG

EG&G Idaho

LOFT

ACCOMPLISHMENTS

- LOFT PERFORMED FOUR ANTICIPATED TRANSIENTS.
  - LOSS OF LOAD.
  - LOSS OF FLOW.
  - EXCESSIVE LOAD INCREASE.
  - LOSS OF FEEDWATER.
- THREE WERE PERFORMED IN ONE WEEK.
- EACH MET THE EXPERIMENTAL OBJECTIVE.

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LOFT

OUTLINE

- WHY ANTICIPATED TRANSIENT EXPERIMENTS ARE USEFUL.
- WHAT EXPERIMENTS LOFT HAS PERFORMED.
- EXPERIMENTAL RESULTS.

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LOFT

NEED FOR ANTICIPATED TRANSIENTS

- PROVIDE A BASIS FOR ATMF (E.G., ATWS).
- THE TESTS ARE NON-TRIVIAL. PREDICTIONS COULD HAVE BEEN BETTER.
- THE ADEQUACY OF MOST SAR ANALYSES HAVE NOT BEEN VERIFIED.
- SIMULATORS ARE GOOD ENOUGH FOR SET POINTS BUT NOT ATMF.
- ANTICIPATED TRANSIENTS PROBABILITY IS HIGHER.

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LOFT

REACTOR SIMULATORS

- SIMULATORS OF THE FUTURE WILL REQUIRE DIGITAL COMPUTER CODES AS A DRIVER.
- CODES DO NOT REPRESENT ALL ASPECTS OF NORMAL OPERATION AND ANTICIPATED TRANSIENTS.
- SECONDARY SIDE MODELS, PRESSURIZER HEATERS AND SPRAY, ETC., MUST BE IMPROVED.
- AT, ATMF, SMALL BREAKS, LARGE BREAKS, AND ALL TRANSIENTS IN BETWEEN MUST BE REPRESENTABLE.

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LOFT

OTHER REGULATORY ISSUES

- SIMULATORS CANNOT ANSWER OPERATION QUESTIONS.
- MANY QUESTIONS ARE POSED FOR THE OPERATION OF A NUCLEAR PLANT - BY PLANT OPERATIONS.
- TECH SPEC CHANGES.
- VERIFICATION OF CURRENT PROCEDURES (E.G., VALVING OUT ECC SYSTEMS AT 1000 PSI).
- TRAINING PROGRAMS - OPERATORS AND TECH ADVISORS.

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EXAMPLE - DIESEL GENERATOR LOADING TEST

- OPERABILITY CHECK REQUIRED OF DIESEL GENERATOR.
- THE HPI MUST BE BLOCKED FOR A SHORT TIME.
- MINIMUM DOWNTIME IS DESIRED.
- CAN THIS CHECK BE PERFORMED DURING HOT STANDBY CONDITIONS?
- IS THE PLANT ADEQUATELY PROTECTED AGAINST SMALL BREAK?

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LOFT MUST PERFORM SUCH EXPERIMENTS

- SMALL SYSTEMS DON'T HAVE REPRESENTATIVE EQUIPMENT:

  MULTIPLE ECC TRAINS.

  SECONDARY SIDE COMPONENTS.

  SINGLE FAILURE PROOF COMPONENTS.

- SMALL SYSTEMS HAVE LARGE HEAT LOSSES.

- A LARGE PLANT SUCH AS ARKANSAS NUCLEAR ONE OR SEQUOYAH DOESN'T RECORD ENOUGH INFORMATION - ARE NOT SEVERE.

- LOFT IS THE ONLY FACILITY CAPABLE OF PERFORMING REALISTIC EXPERIMENTS - OTHERS PERFORM CONSERVATIVE EXPERIMENTS.

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LOFT

LOFT ANTICIPATED TRANSIENT EXPERIMENT OBJECTIVES

- PHENOMENA UNDERSTANDING.
- THRESHOLD DETERMINATION.
- AUGMENTED OPERATOR PROGRAM.
- ENGINEERED SAFETY FEATURES/PLANT CONTROL SYSTEMS.
- CODE ASSESSMENT.

TRANSIENT CHARACTERISTICS

- COOLANT INVENTORY CONSTANT OR INITIALLY INCREASING.
- PCS ENERGY BALANCE IMPORTANT.
- PCS PRESSURE IS A FUNCTION OF ENERGY BALANCE AND PRESSURIZER DYNAMICS.

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LOFT ANTICIPATED TRANSIENTS CONSIDERATIONS

- BASELINE LOFT DATA.
- PCS AND SCS INITIATING EVENTS.
- CODE ASSESSMENT.
- MINIMUM SCHEDULE IMPACT.

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LOFT

LOFT ANTICIPATED TRANSIENT EXPERIMENTS COMPLETED

- LOSS-OF-STEAM LOAD.
- LOSS-OF-FORCED PCS FLOW.
- EXCESSIVE LOAD INCREASE.
- LOSS-OF-FEEDWATER

EACH WAS PREDICTED WITH RETRAN

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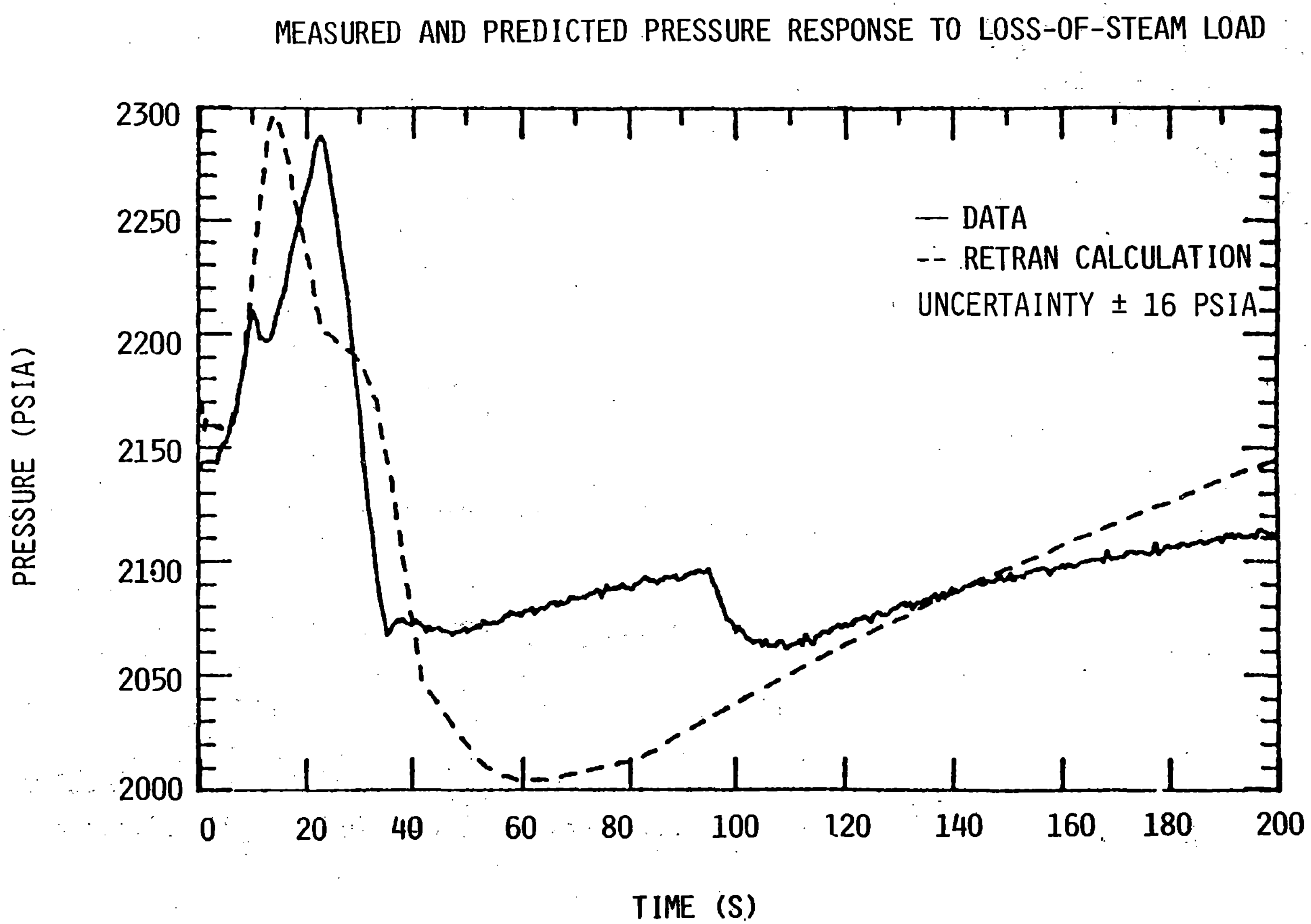
LOFT


MEASURED AND PREDICTED PRESSURE RESPONSE TO LOSS-OF-STEAM LOAD
PRESSURE (PSIA)
2300
2250
2200
2150
2100
2050
2000
— DATA
-- RETRAN CALCULATION
UNCERTAINTY ± 16 PSIA
0
20
40
60
80
100
120
140
160
180
200
TIME (S)

IMPROVEMENTS FROM L6-1

- SPRAY HEAT TRANSFER COEFFICIENTS.
- CONDENSATION EFFECTS.
- PRESSURIZER HEATER MODEL.

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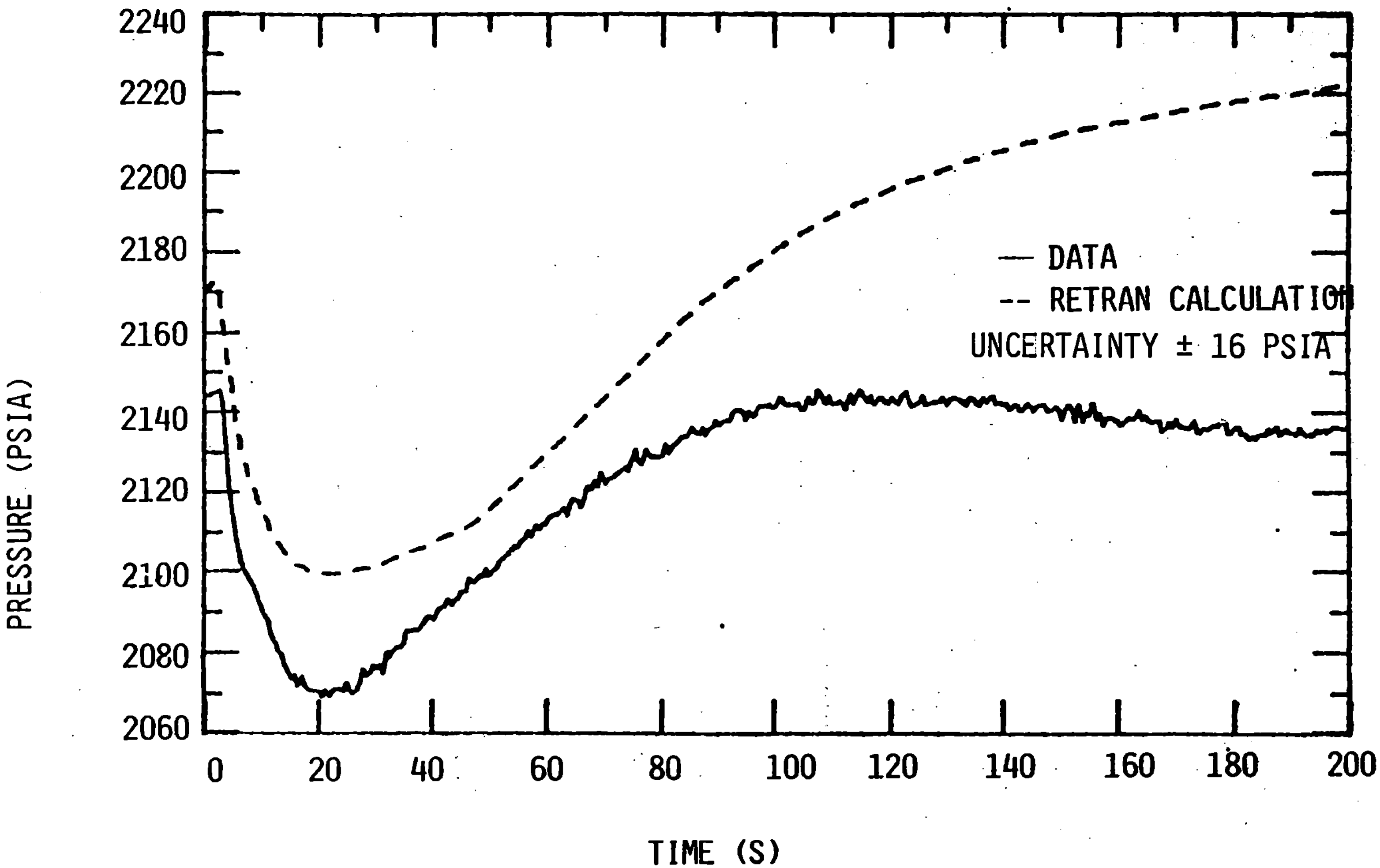
LOFT

MEASURED AND PREDICTED PRESSURE RESPONSE TO LOSS-OF-FORCED PCS FLOW



IMPROVEMENTS FROM L6-2

- NON-EQUILIBRIUM MODEL IN PRESSURIZER

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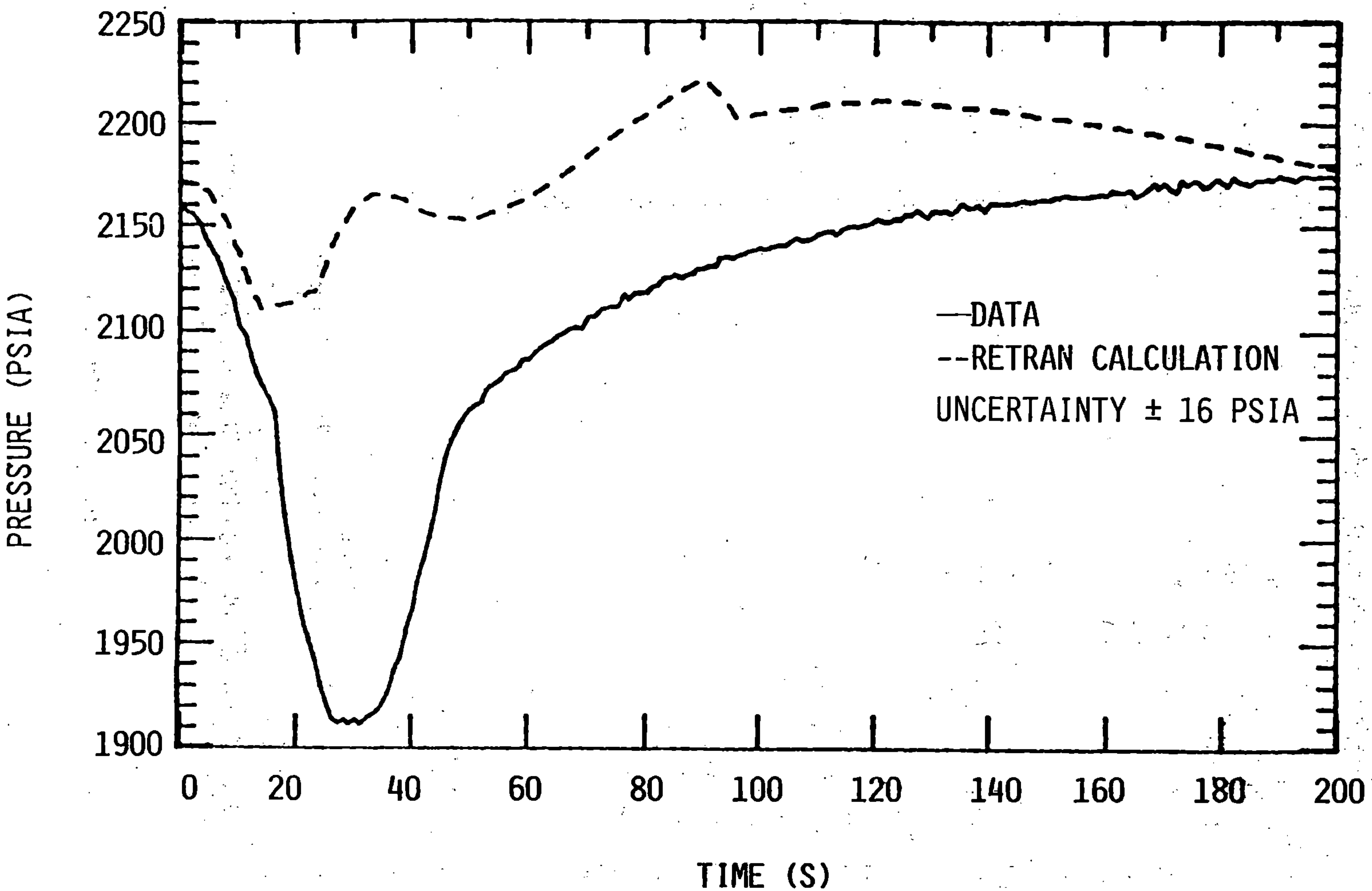
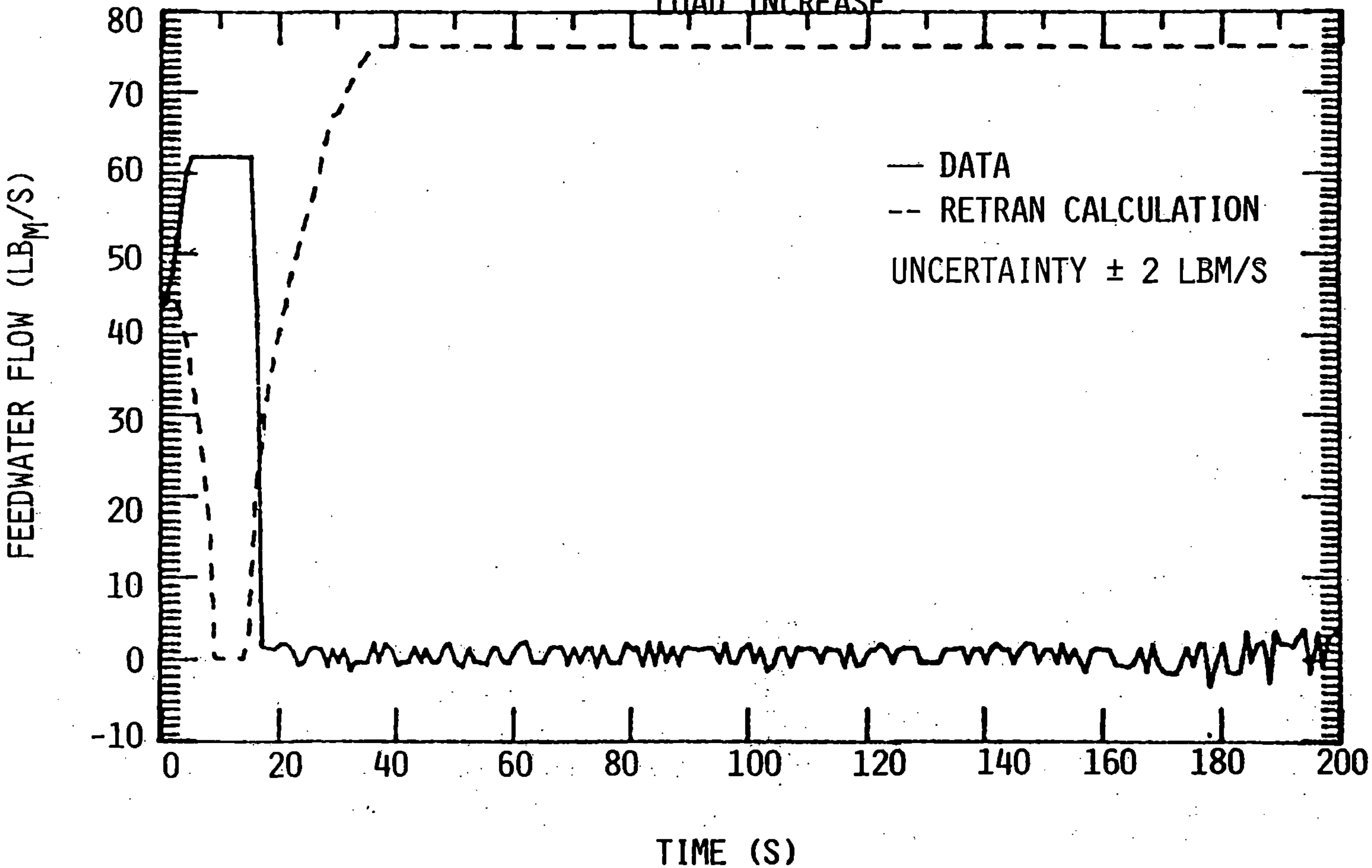
LOFT

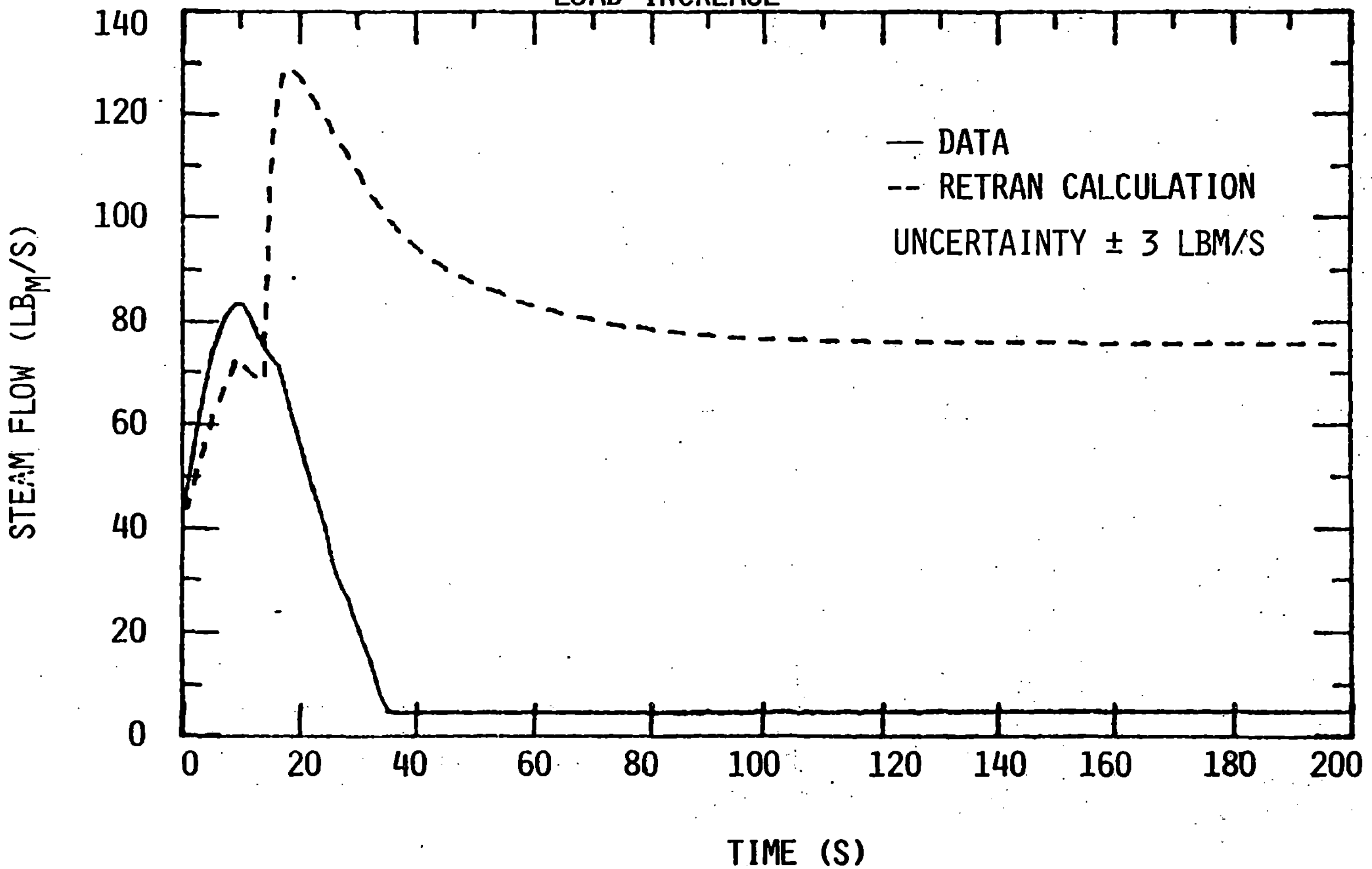
MEASURED AND PREDICTED PRESSURE RESPONSE TO AN EXCESSIVE LOAD INCREASE



MEASURED AND PREDICTED FEEDWATER FLOW RESPONSE TO AN EXCESSIVE LOAD INCREASE



MEASURED AND PREDICTED STEAM FLOW RESPONSE TO AN EXCESSIVE LOAD INCREASE



IMPROVEMENTS FROM L6-3

- SCRAM.
- FEEDWATER CONTROL.
- SECONDARY STEAM FLOW.

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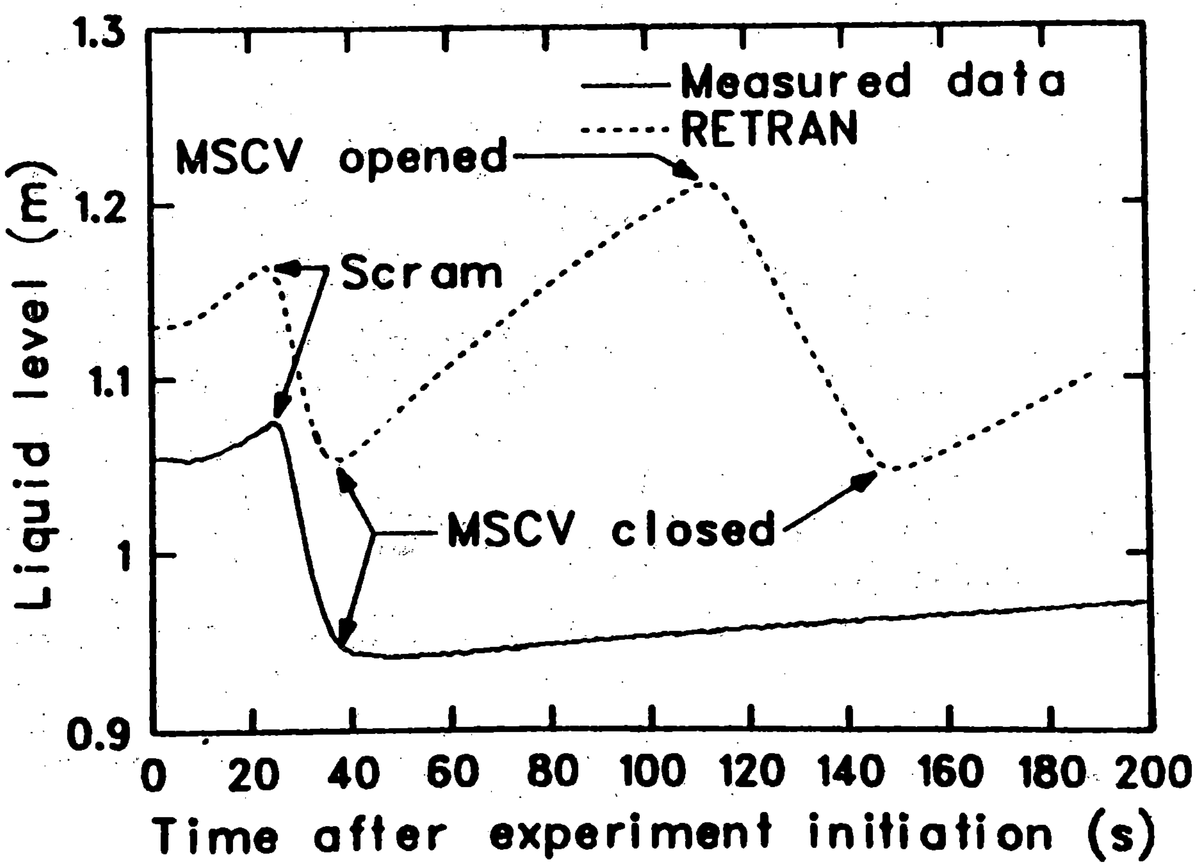
LOFT

PREDICTED AND MEASURED PRESSURIZER LEVEL

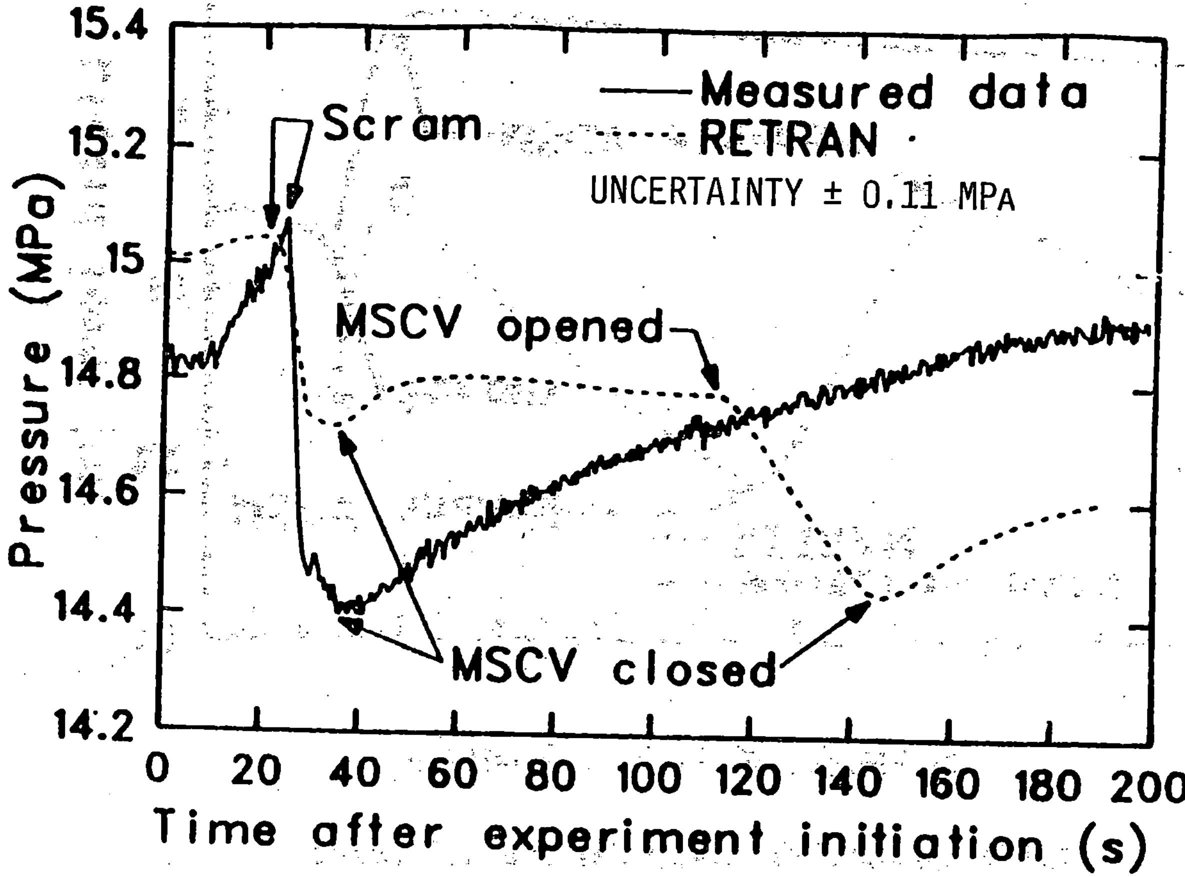
RESPONSE TO LOSS-OF-FEEDWATER



UNCERTAINTY ± 0.07 M

MEASURED AND PREDICTED PRESSURE RESPONSE

TO LOSS-OF-FEEDWATER



IMPROVEMENT FROM L6-5

- DECAY HEAT.
- SECONDARY SIDE LEAKAGE.

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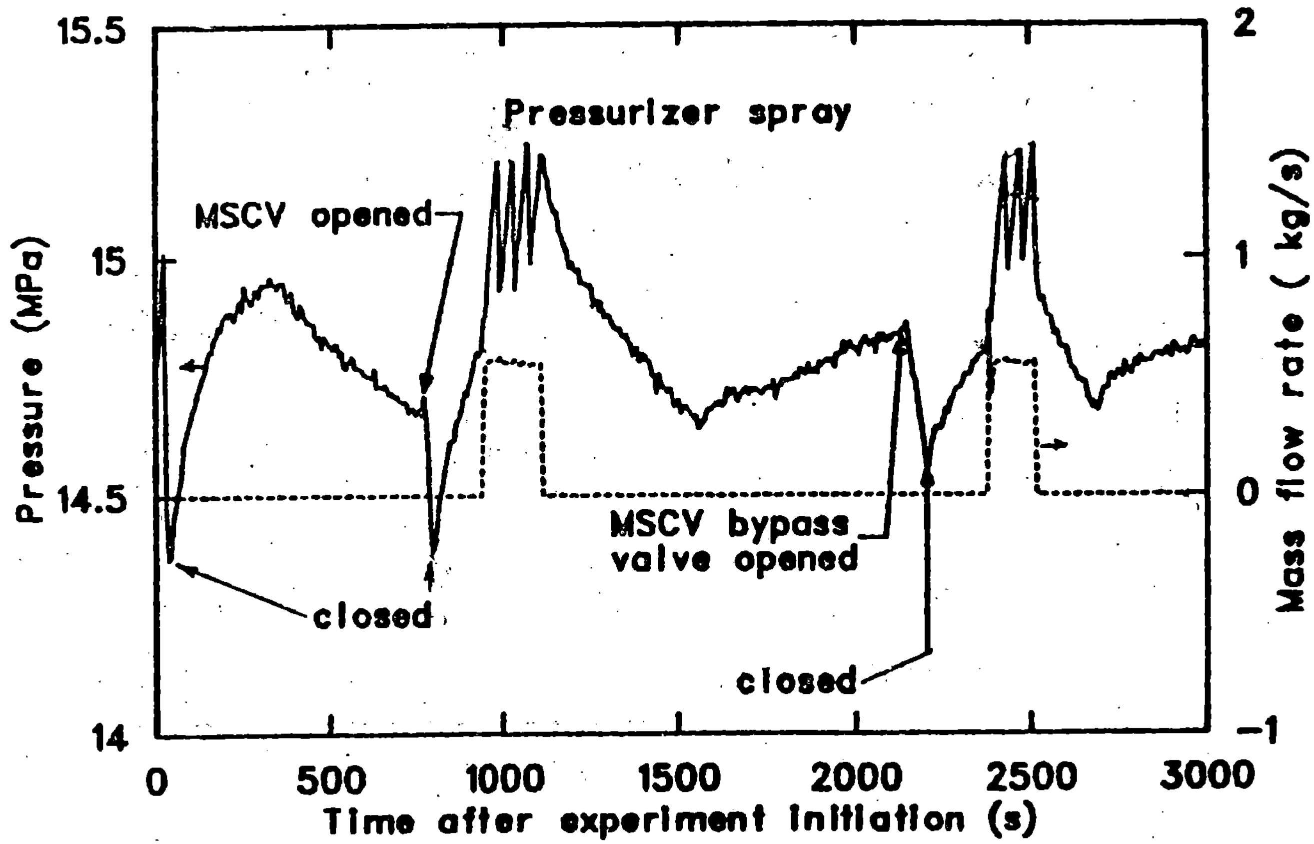
RECOVERY FOLLOWING LOSS-OF-FEEDWATER



UNCERTAINTY: PRESSURE ± 0.11 MPA
FLOW ± 0.1 KG/S

CONCLUSIONS

- FOUR SUCCESSFUL EXPERIMENTS WERE PERFORMED.
- CURRENT MODELS IN SAR'S SHOULD BE VERIFIED.
- RETRAN CAN PREDICT TRENDS AND EVENTS.
- SEVERAL AREAS FOR IMPROVEMENT HAVE BEEN DETERMINED.
- ESF/PPS AND OPERATOR ACTION WAS EFFECTIVE.

FLOW MEASUREMENT TECHNIQUES
IN LOFT SMALL BREAK EXPERIMENTS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

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FLOW MEASUREMENT TECHNIQUES IN LOFT SMALL BREAK EXPERIMENTS

D. J. Hanson
EG&G Idaho, Inc.

Knowledge of the system mass flow rates during small break loss-of-coolant experiments (LOCEs) is important for understanding the complex and inter-related phenomena which occur. It is particularly important to know the fluid conditions and mass flow rate at the break, as they strongly influence many aspects of the system behavior and the system natural circulation flow rates which in turn directly influence the methods of core heat removal and system recovery. To provide these fluid conditions and mass flow rates during Loss-of-Fluid Test (LOFT) experiments, new measurement devices were developed and applied at the break location and at two locations where the various modes of natural circulation can be easily distinguished. This summary briefly describes the measurement techniques and the data obtained to date.

The break location for the later small break tests to be conducted in LOFT will be in a 3-in. pipe which tees into the intact loop. Measurement of break mass flow in this configuration is difficult due to flow stratification upstream of the break plane and high fluid qualities and velocities downstream. To provide accurate break flow information under these conditions, a nuclear-hardened, low-energy gamma densitometer was placed upstream of the break and a full flow drag-device and turbine meter were placed downstream. The densitometer utilizes three parallel gamma beams to provide information on density distribution and average density immediately upstream of the break and a fourth beam provides an indication of the density near the break orifice. The momentum flux and velocity measured by the drag-device and turbine meter located downstream of the orifice are combined to provide the break mass flow. A homogenizer plate is used upstream of the drag-device to assure a more accurate measurement by minimizing flow stratification and interphase slip. The specially designed drag-device is located upstream of the turbine. Momentum flux is measured by the device utilizing a perforated plate attached to a cantilever beam and an eddy current coil to sense the beam movement. The turbine is a commercially available design.

Break flow was measured accurately using the described measurement techniques during the period of LOFT Test L3-5 when most of the mass was expelled from the system. The piping geometry upstream and downstream of the measurement spool affected the mass flow measurement during the very early portion and the latter portion of the test. The fluid upstream of the break was shown to be stratified for extended periods of time. Early break uncovery was measured by the horizontal density beam, and there is an indication that the stratified level near the break was somewhat depressed compared to the intact loop piping liquid level.

The different modes of natural circulation that are possible during a small break impose very different requirements on the measurement techniques. Single-phase natural circulation is characterized by low velocity; quasi-steady-state, unidirectional flow. However, two-phase natural circulation has an expanded velocity range and more transient flow conditions with either co-current or counter-current flow possible. To provide information in both modes of natural circulation, existing LOFT measurements were modified and pulsed neutron activation (PNA) was developed and applied for the first time in a nuclear environment. This PNA system used four neutron generators and a single sodium iodide detector to measure natural circulation velocities. Application of PNA in a nuclear environment required both neutron shielding to minimize activation of the detector and extensive gamma ray shielding.

Direct measurement of natural circulation in LOFT was made at a location directly above the core center fuel bundle using a turbine meter and in the intact loop hot leg piping using a multi-beam densitometer and a rake of three turbine meters. The measurement above the core is most meaningful during single-phase natural circulation whereas the measurements in the piping have the potential for measuring both single-phase and two-phase flow modes. The flow measurements using PNA are most indicative of the liquid velocity in the pipe.

The natural circulation measurement techniques developed for LOFT have provided accurate data during several LOFT experiments. These data are being used to aid in understanding the effectiveness of the various modes of natural circulation in removing energy from the core and the capability of other instrumentation to indicate natural circulation. As an example, during small break test L3-7 the data show natural circulation continuing for extended periods of time even though flow stratification had occurred in the piping.

REFERENCES

1. J. R. Fincke, Development of the LOFT Phototype Drag Screen-Transient Steam-Water Testing, NUREG/CR-0270, TREE 1241 (July 1979).

FLOW MEASUREMENT TECHNIQUES IN LOFT SMALL BREAK EXPERIMENTS

BY
D.J. HANSON

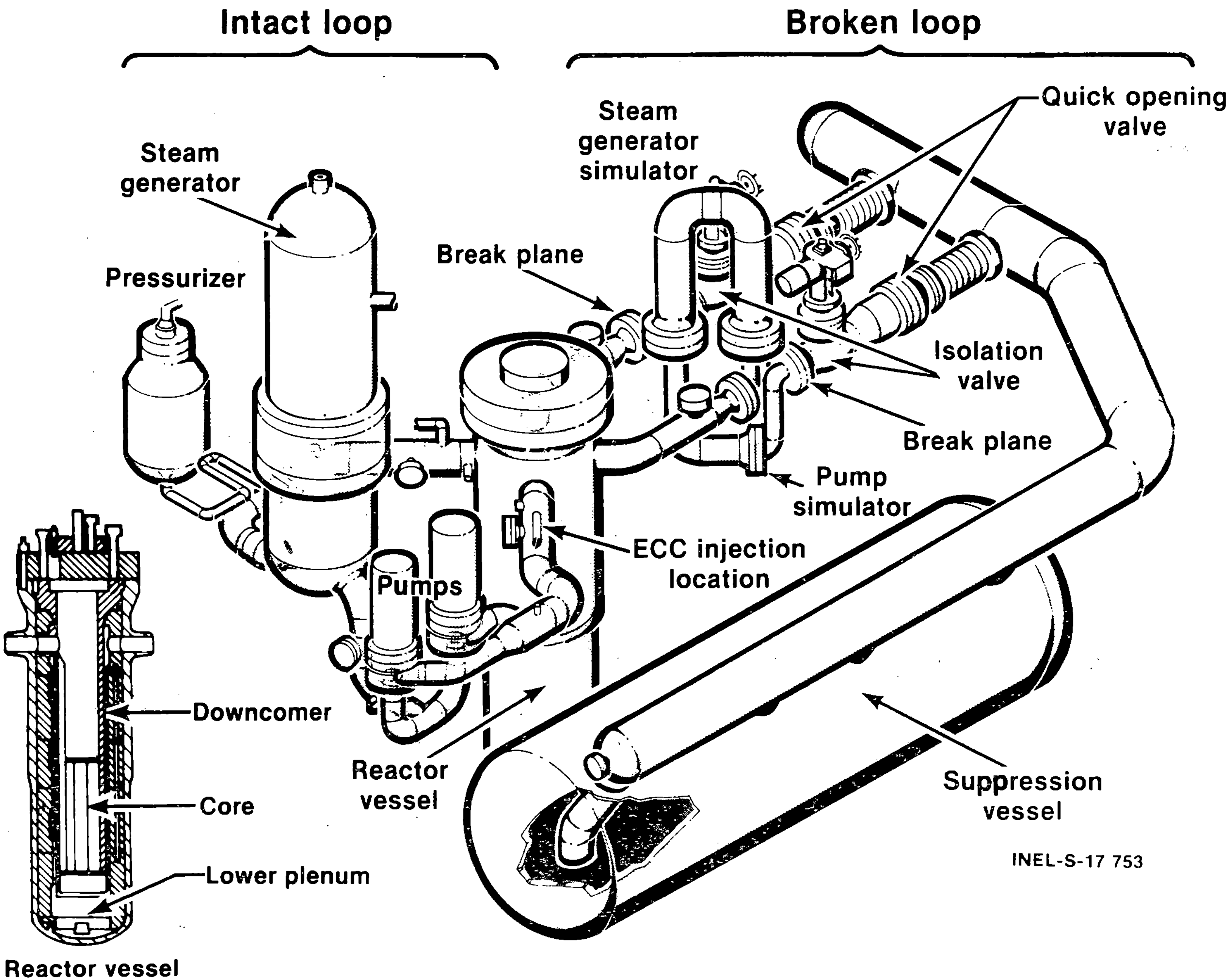
EG&G Idaho, Inc.

NEW MEASUREMENTS PROVIDE IMPORTANT SMALL BREAK DATA

- BREAK FLOW CONDITIONS
- BREAK MASS FLOW
- NATURAL CIRCULATION FLOW

DJH-2

LOFT System Configuration

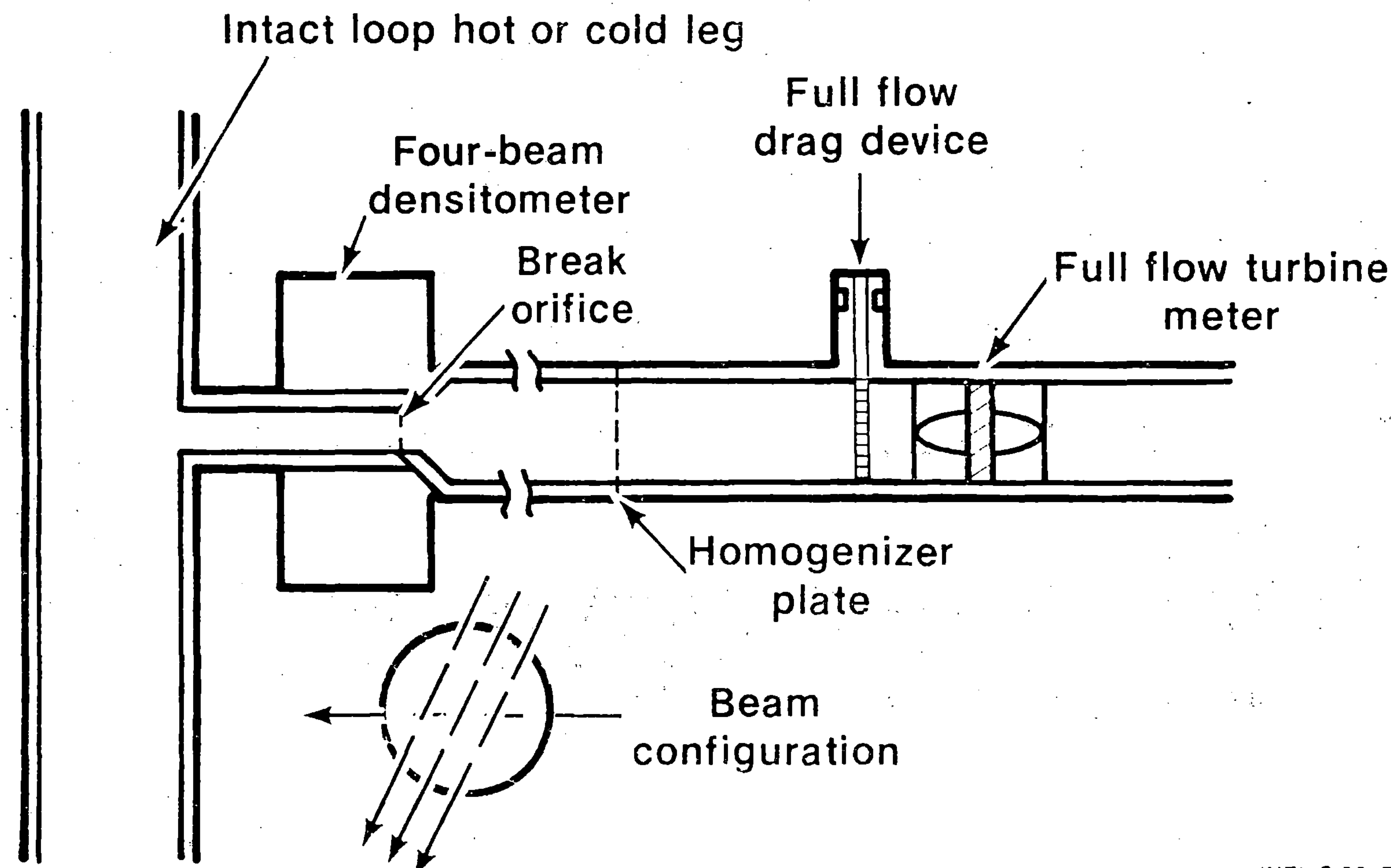


FLOW AND GEOMETRIC CONDITIONS MAKE BREAK MEASUREMENTS DIFFICULT

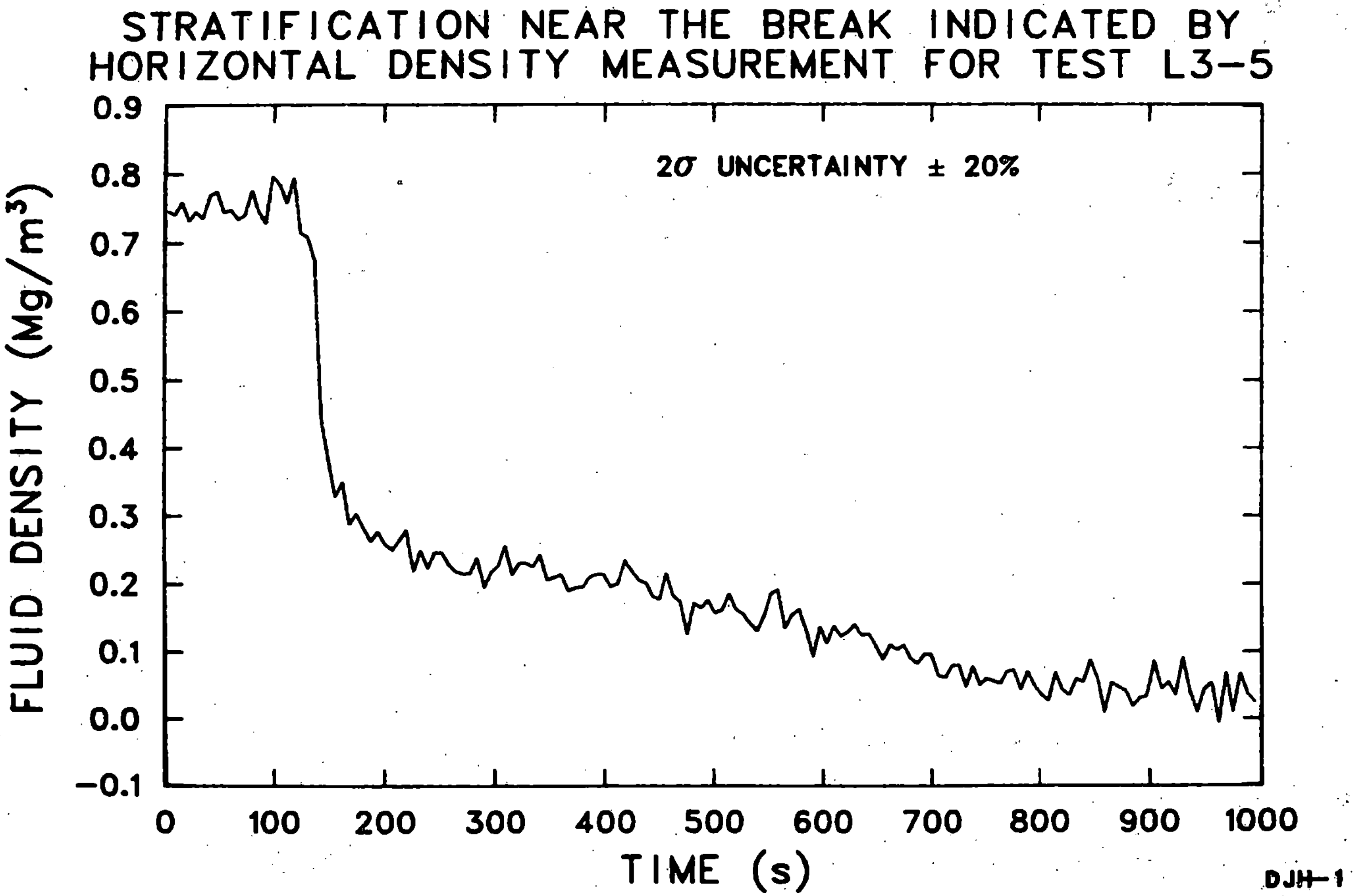
- STRATIFICATION OF FLOW UPSTREAM
- SMALL PIPE SIZE IN A NUCLEAR ENVIRONMENT
- HIGH QUALITY AND VELOCITY DOWNSTREAM

DJH-4

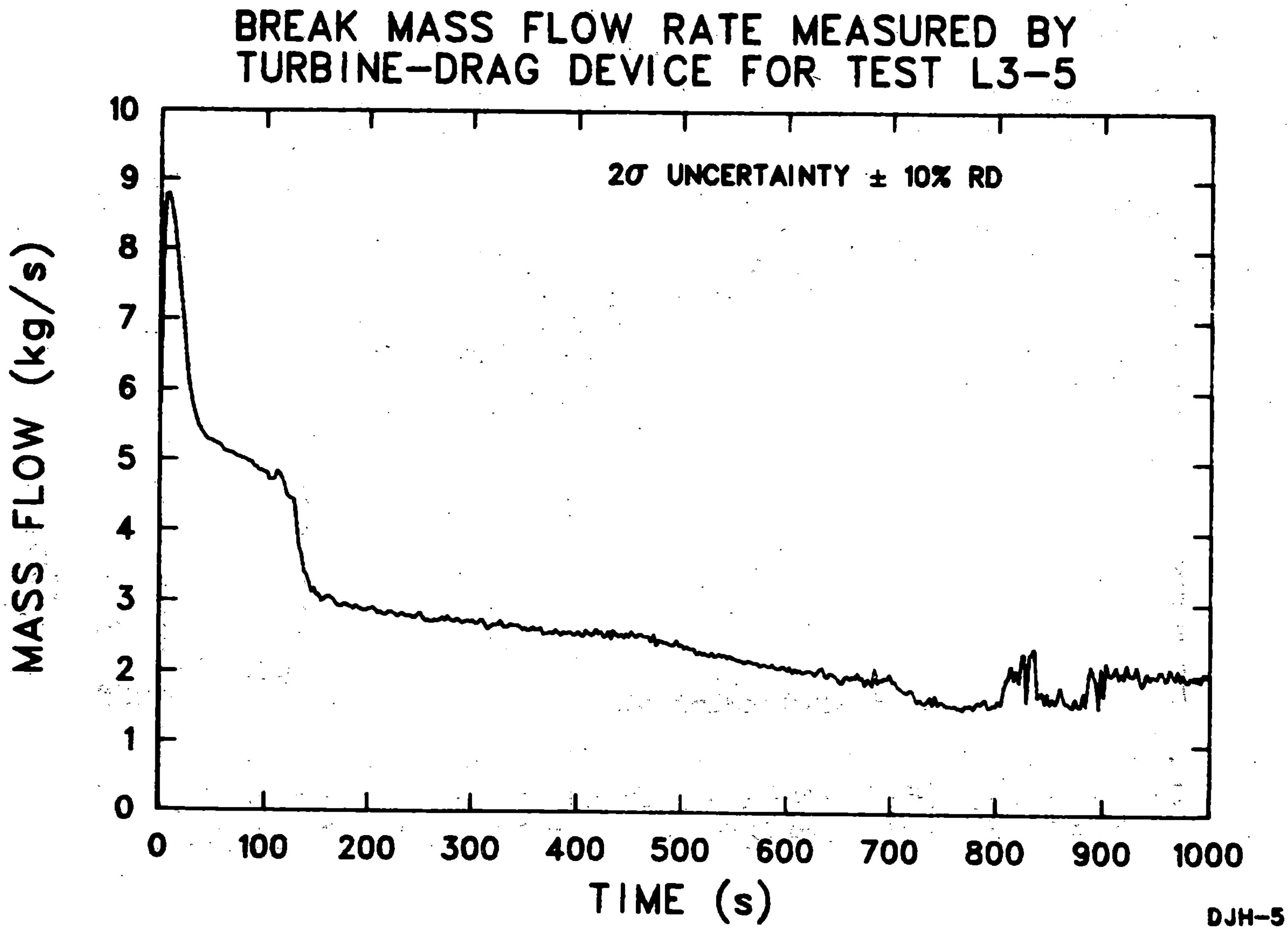
LOFT Break Flow Spool Arrangement



INEL-S-26 799



STRATIFICATION NEAR THE BREAK INDICATED BY HORIZONTAL DENSITY MEASUREMENT FOR TEST L3-5
2σ UNCERTAINTY ± 20%
FLUID DENSITY (Mg/m³)
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
-0.1
0
100
200
300
400
500
600
700
800
900
1000
TIME (s)
DJH-1



BREAK MASS FLOW RATE MEASURED BY
TURBINE-DRAG DEVICE FOR TEST L3-5
2σ UNCERTAINTY ± 10% RD
MASS FLOW (kg/s)
0
1
2
3
4
5
6
7
8
9
10
0
100
200
300
400
500
600
700
800
900
1000
TIME (s)
DJH-5

DIFFERENT NATURAL CIRCULATION MODES IMPOSE DIFFERENT REQUIREMENTS

- SINGLE-PHASE MODE
  - LOW VELOCITY
  - QUASI-STEADY STATE
  - UNIDIRECTIONAL
- TWO-PHASE MODE
  - EXPANDED VELOCITY RANGE
  - MORE TRANSIENT
  - CONCURRENT OR COUNTERCURRENT

DJH-8

TECHNIQUES TO MEASURE NATURAL CIRCULATION MODES IN LOFT

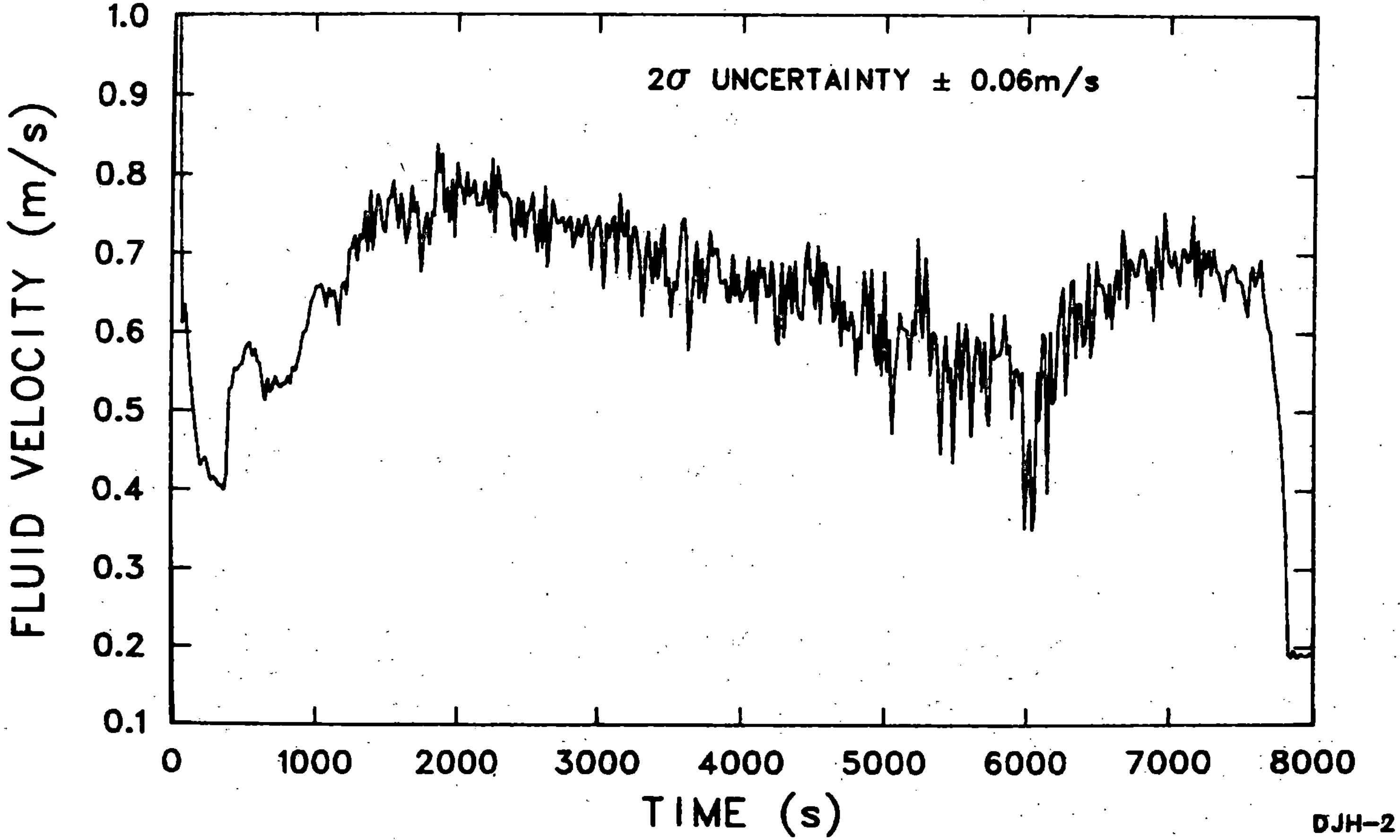
- UPPER CORE TURBINE METER
- PIPE DENSITOMETER AND TURBINE METER RAKE
- PULSED NEUTRON ACTIVATION (PNA)

DJH-9

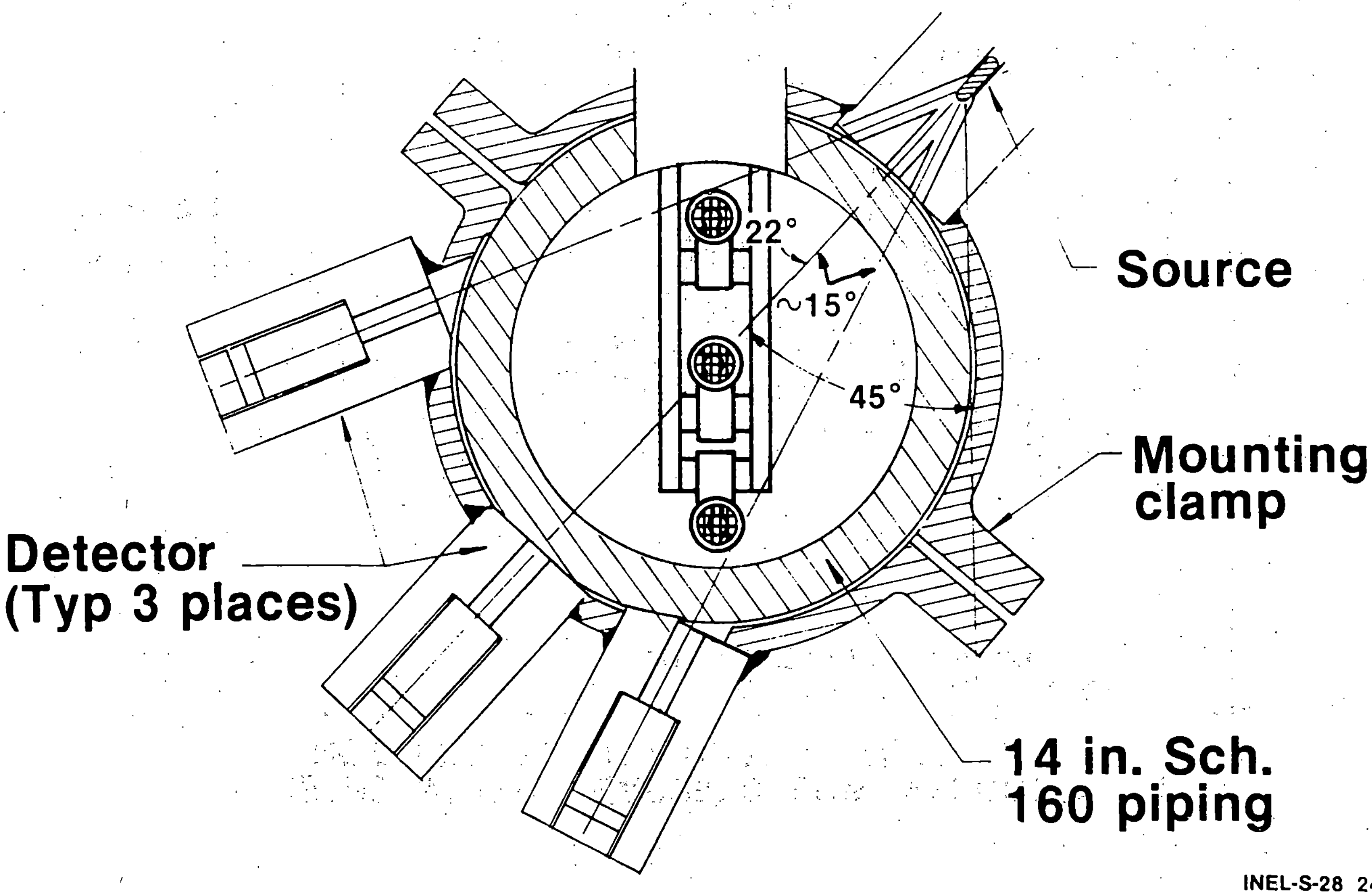
UPPER CORE TURBINE METER CAPABILITIES

- LOCATED DIRECTLY ABOVE CENTER FUEL BUNDLE
- PROVIDES DIRECT VELOCITY MEASUREMENT
- LIMITED CAPABILITY TO DISTINGUISH NATURAL CIRCULATION MODES

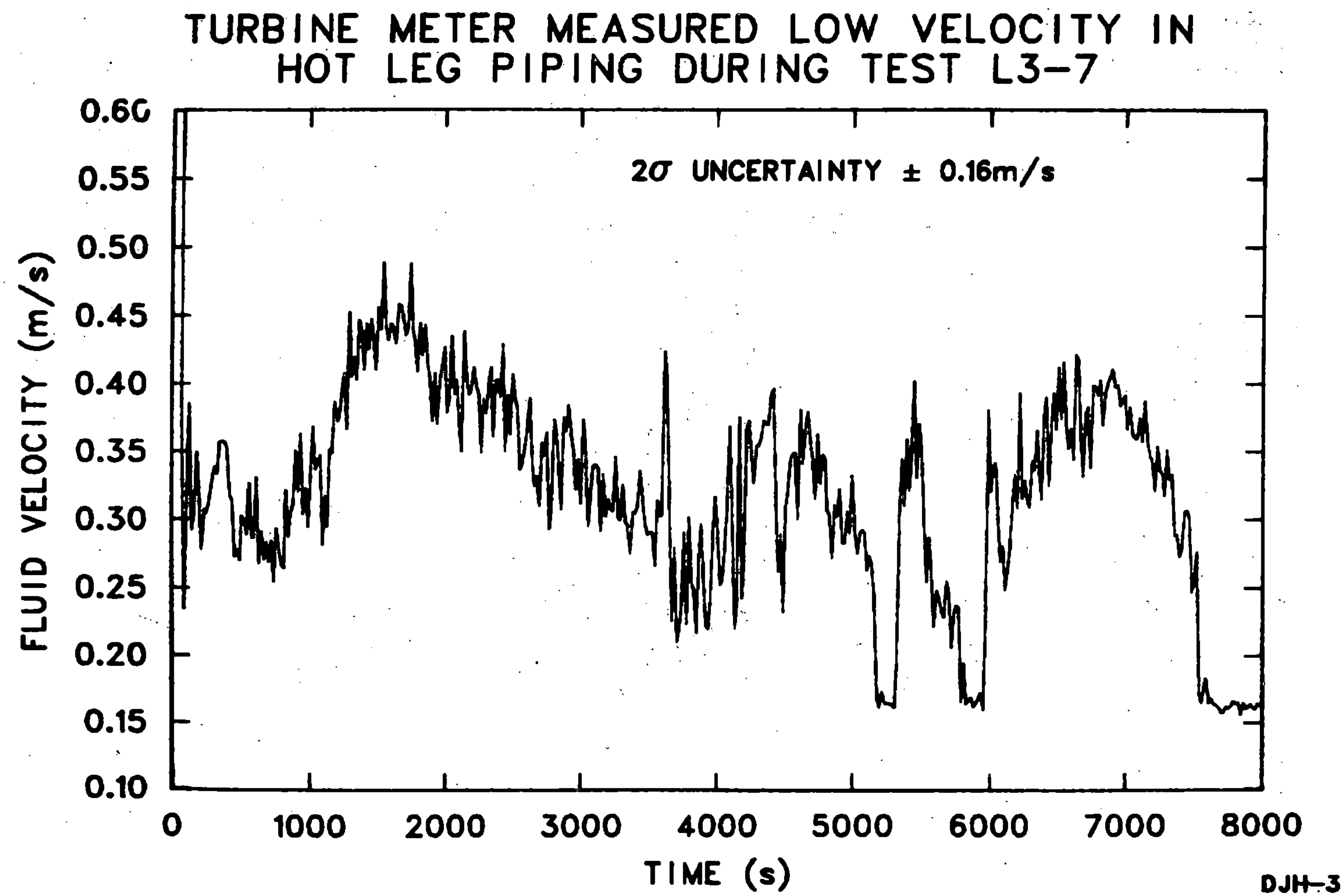
NATURAL CIRCULATION VELOCITIES MEASURED BY UPPER CORE TURBINE METER FOR TEST L3-7



Intact Loop Flow Measurement Technique

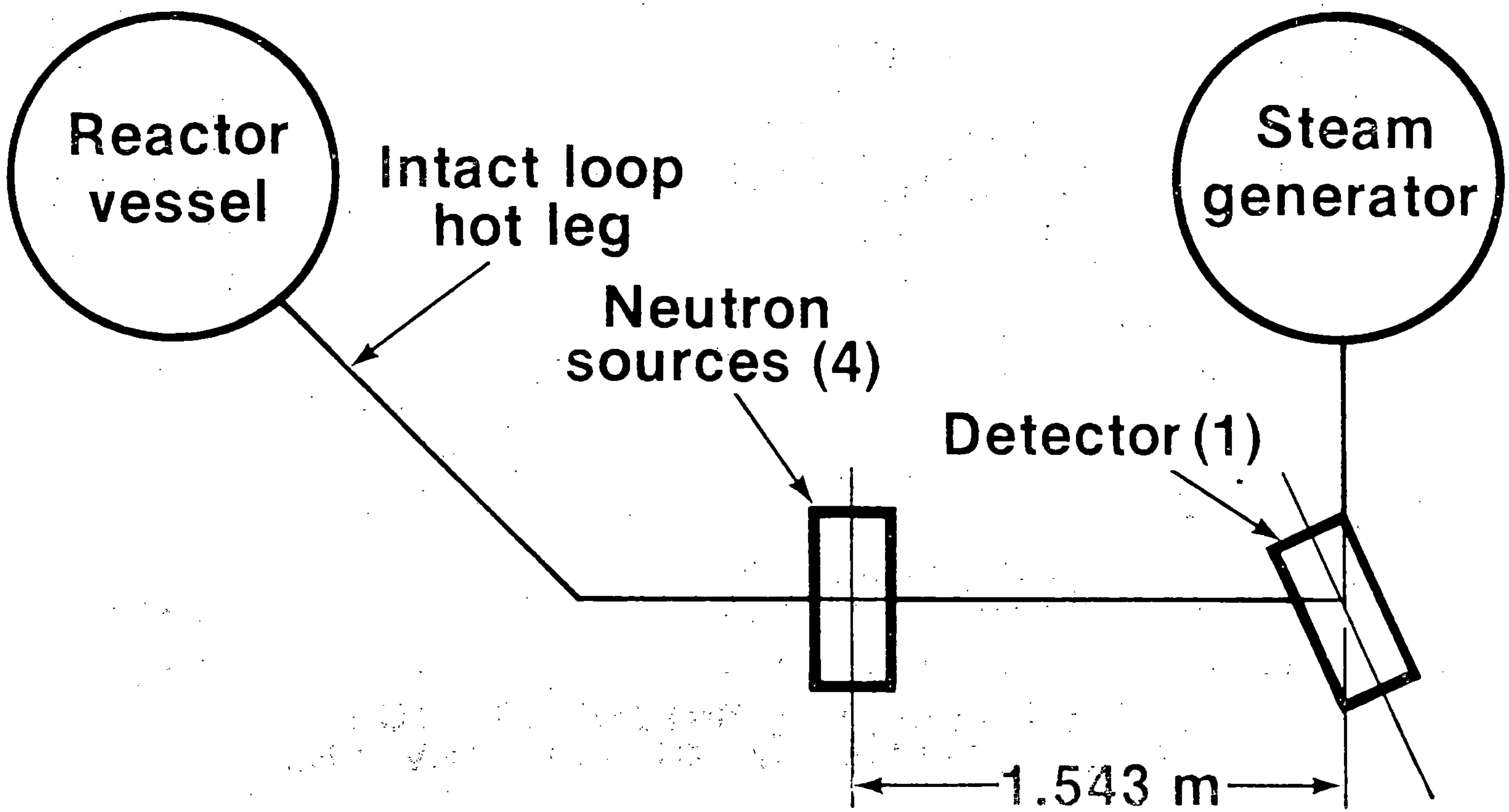


INEL-S-28 248

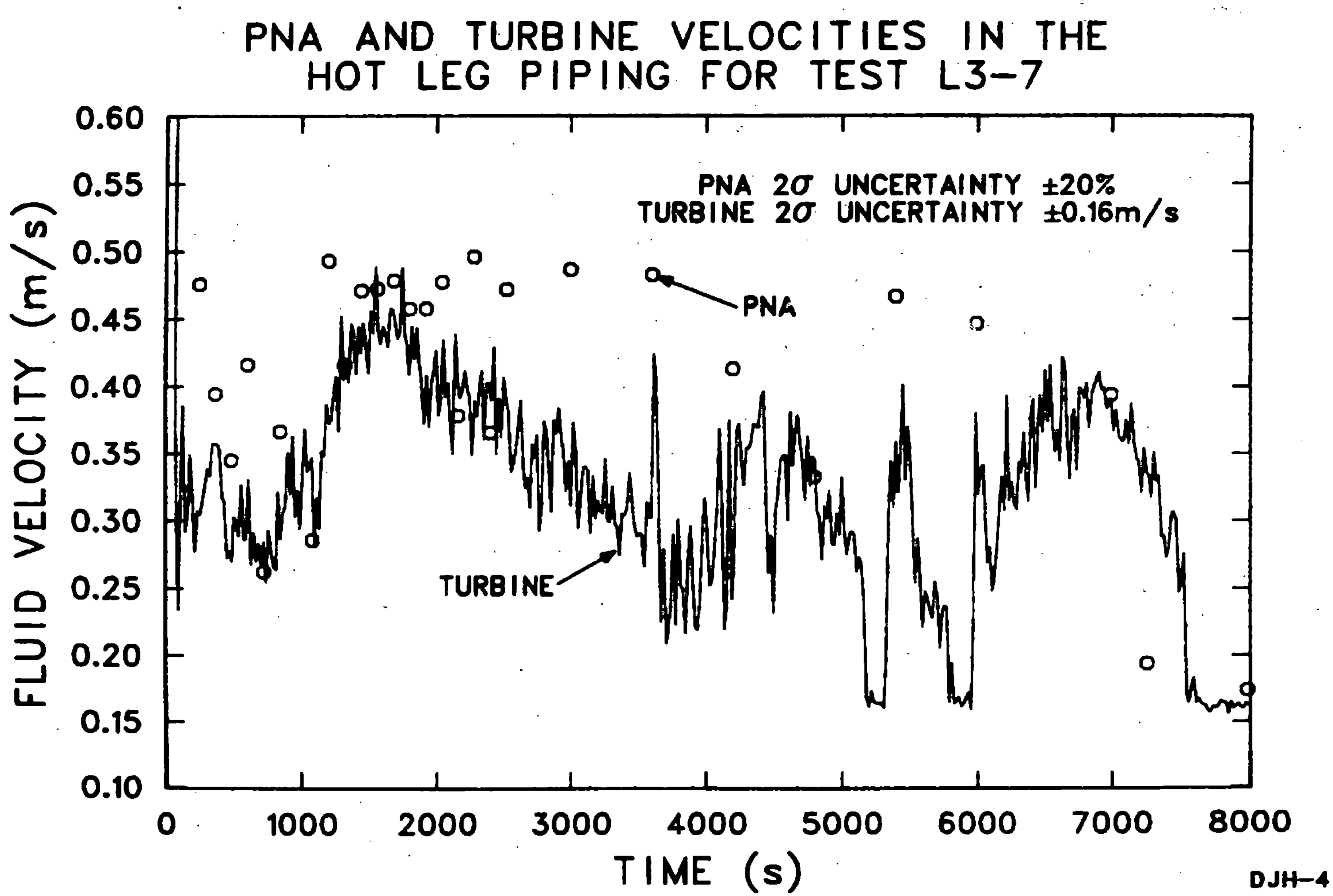


TURBINE METER MEASURED LOW VELOCITY IN
HOT LEG PIPING DURING TEST L3-7
2σ UNCERTAINTY ± 0.16m/s
FLUID VELOCITY (m/s)
0.60
0.55
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0
1000
2000
3000
4000
5000
6000
7000
8000
TIME (s)
DJH-3

Arrangement of PNA on LOFT

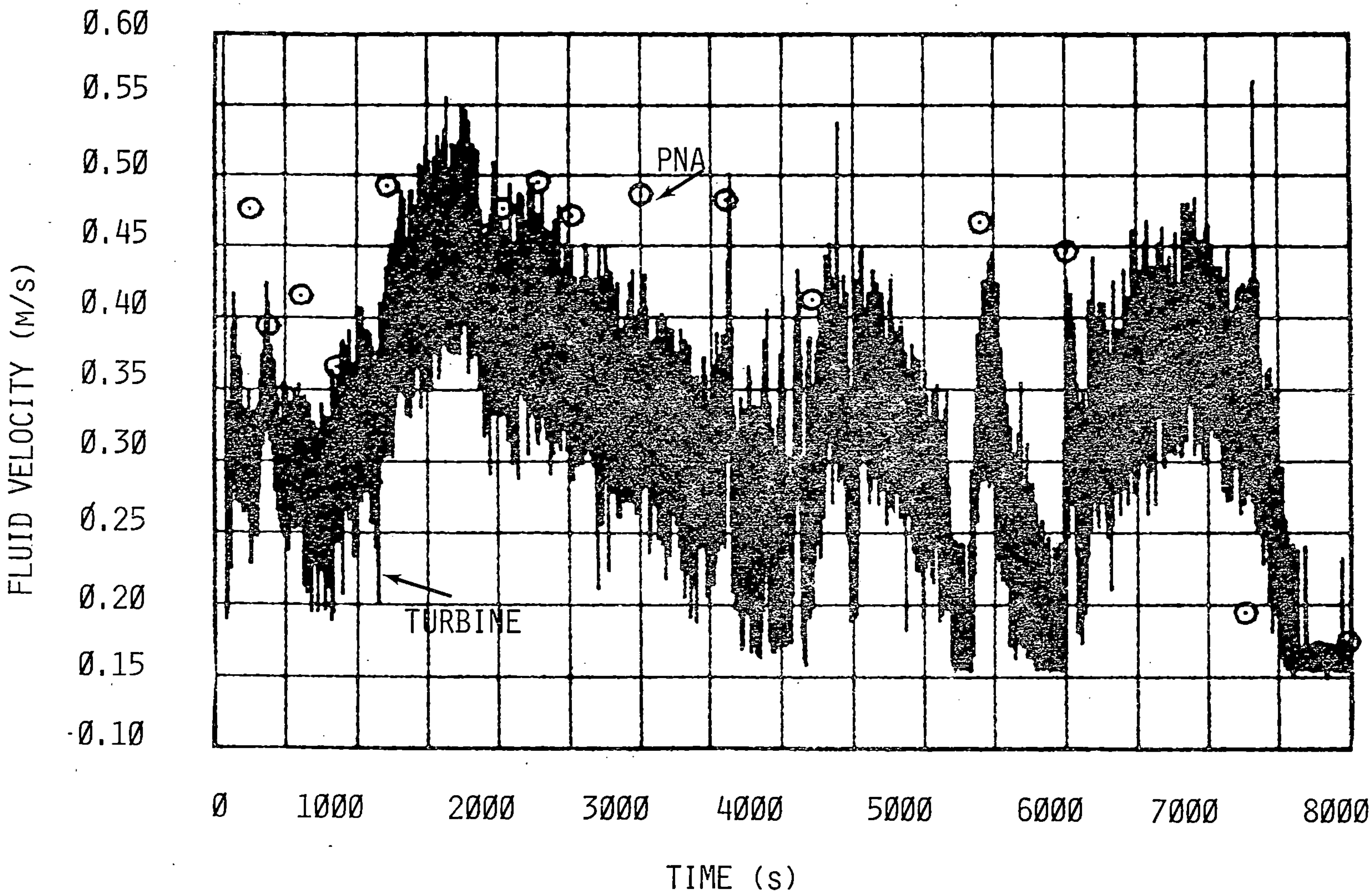


INEL-S-25 382


PNA AND TURBINE VELOCITIES IN THE
HOT LEG PIPING FOR TEST L3-7
PNA 2σ UNCERTAINTY ±20%
TURBINE 2σ UNCERTAINTY ±0.16m/s
PNA
TURBINE
FLUID VELOCITY (m/s)
0.60
0.55
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0
1000
2000
3000
4000
5000
6000
7000
8000
TIME (s)
DJH-4

UNDECIMATED TURBINE DATA AND PNA DATA SHOW GOOD AGREEMENT IN

HOT LEG VELOCITY FOR TEST L3-7



CONCLUSIONS

- NEW MEASUREMENTS DEVELOPED
- SUCCESSFULLY USED IN LOFT EXPERIMENTS
- DATA OBTAINED WILL AID IN UNDERSTANDING LOFT SMALL BREAK BEHAVIOR

DJH-16

AGENDA

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

MONDAY, OCTOBER 27, 1980

RED AUDITORIUM - ALL PARTICIPANTS

9:15 am -	Introductory Remarks	Commissioner
9:30 am -	Reactor Safety Research Program	T. E. Murley, NRC
10:00 am -	Highlights of WRSR Achievements in FY 80 and Status of LOCA Safety Evaluation	L. S. Tong, NRC
10:20 am -	Coffee Break	

MORNING SESSION - RED AUDITORIUM

LOFT PROGRAM

Chairman: G. D. McPherson, NRC

11:00 am -	Overview of the LOFT Program: Results of Small-Break LOCA and Operational Transient Test Program	G. D. McPherson, NRC
11:20 am -	Results of LOFT Small Break Experiments L3-1, L3-2, L3-5/5a and L3-7	J. H. Linebarger, INEL
12:10 pm	LOFT: A Nuclear Plant Providing Realistic Answers to PWR Licensing Issues	C. W. Solbrig, INEL
12:30 pm -	Results of Anticipated Transient Experiments	C. W. Solbrig, INEL
1:00 pm -	Lunch	

AFTERNOON SESSION - RED AUDITORIUM

LOFT PROGRAM (Continued)

2:00 pm -	Flow Measurement Techniques in LOFT Small Break Experiments	D. J. Hanson, INEL
2:35 pm -	LOFT Program Overview	N. C. Kaufman, INEL
3:05 pm -	Discussion	
3:15 pm -	Coffee Break	

LOFT PROGRAM OVERVIEW

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

N. C. Kaufman
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

LOFT PROGRAM OVERVIEW

N. C. Kaufman
EG&G Idaho, Inc.

During the last several months, the Loss-of-Fluid Test (LOFT) Program has been changing emphasis in order to remain responsive to the issues confronting the nuclear safety community. With these changes, the program will make maximum use of our unique combination of facility and personnel, which permit us to intentionally place an operating nuclear reactor into accident conditions, to measure responses, and to recover. The program value has been further reinforced by the participation in LOFT planning by representatives of several segments of the nuclear industry, including ten foreign countries, and by the recent concern with human factors and advanced display capability.

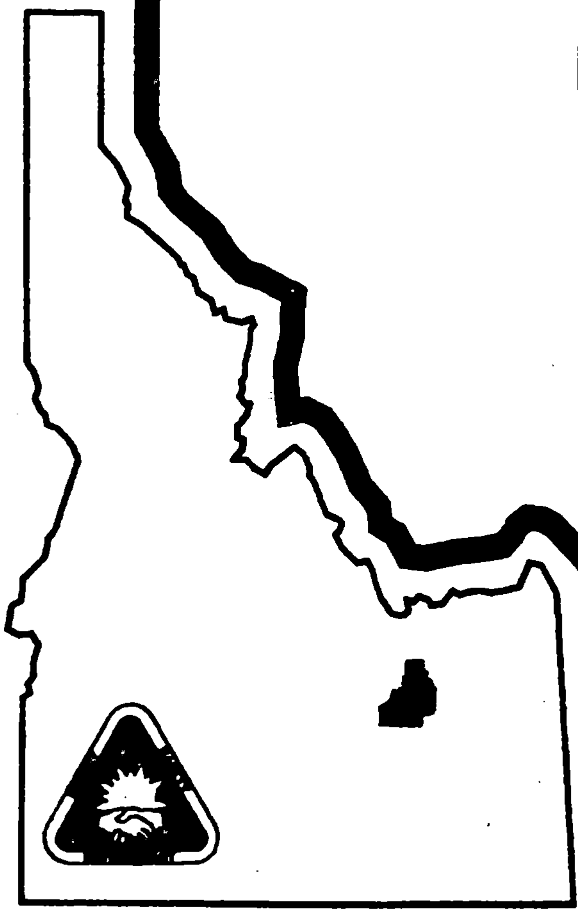
Today, we have two principal points of program emphasis. First we are creating an experimental data base that reflects a wide spectrum of reactor accident phenomena and plant states. Second, we are developing, using, and evaluating methods to recognize, characterize, control, and recover from accident conditions. Thus, the LOFT Program has shifted away from a large break, thermal-hydraulics focus and is now principally oriented toward improved plant safety by way of accident recognition and the selection of an appropriate combination of manual and automatic responses to insure stable plant recovery.

We believe that a reactor data base established for a range of accident types and severities is essential not only to the evaluation and development of analytical methods (such as computer codes) but to the development of a safety perspective relative to the several phenomena accompanying any accident. Additionally, such an integral data base is necessary to provide boundary conditions for tests of specific and separate effects and to relate transitions among effects and phenomena that have been considered separately. Finally, a data base obtained from a reactor with typical process instrumentation is necessary to the development of realistic accident simulations and to the evaluation of presumed relations between measured and critical variables.

Our efforts to develop, use, and evaluate methods for accident recognition and response currently have highest program priority. These efforts recognize that we at LOFT must routinely prepare our operators to enter into and recover from accident conditions. We must prepare also for additional compounding failures while conducting such tests. Therefore, we have previously developed methods for maintaining plant and process visibility while in accident conditions, quickly recognizing further degradation, and selecting appropriate responses. We are now systematically evaluating and extending these LOFT concepts for possible application to commercial reactor operations. As part of this activity, we have developed a program called Augmented Operator Capability to utilize color cathode ray tubes to display plant conditions accurately and quickly, and to provide feedback on the effectiveness of actions taken. The emphasis of this program will be on verification under accident conditions of concepts for information display and for operational suggestion, and on the development of schemes by which the quality (believability) of displayed information can be judged.

The specific tests performed and planned for LOFT are selected to satisfy specific NRC needs and to provide data important to completion of data base efforts and evaluation of operational methods. Specifically, we select tests designed to provide a wide spectrum of accident types and plant states. Thus, our current program includes small break tests (L3), anticipated transients (L6), intermediate break tests (L5), and large break tests (L2). Small break tests are selected to look at a range of controlled emergency core coolant (ECC) break flow conditions, heat sinks, cooldown techniques, and end states. Large break tests are selected to study the effects of power density, fuel pressurization, and automatic ECC delivery. Anticipated transients have been selected based on transient probability and relation to PWR startup testing in order to establish a useful base for multifailure accident studies. The intermediate breaks will be selected to test the transition from small, communicative breaks to large, double-ended breaks. A number of tests have been scoped for each of these test series. However, the specific tests performed and their sequence reflect continual reevaluation of these scoped tests with modifications added to reflect the understanding gained in the previous tests and the needs of NRC. Thus, the current LOFT Program is, and has been, flexible and heuristic.

As part of the LOFT efforts to assure a relevant program, three new test series are being planned: L8, degraded core tests; L9, transients with multiple failures; and L10, accident override transients. The degraded core tests will look at core uncovery and fuel response to severe transients with various prepressurizations. The transients with multiple failures will look at compounded accidents deriving from relatively probable initiating transients. The accident override transients will look at possible schemes for effective, predictable accident intervention to permit rapid plant system convergence to a safe end condition. As for all test series, specific tests will be selected in accordance with their importance to the licensing process and safety significance.



OVERVIEW OF LOFT PROGRAM

by

N.C. KAUFMAN

EG&G Idaho, Inc.

LOFT MISSION

- ESTABLISH NUCLEAR REACTOR CONDITIONS CHARACTERISTIC OF POSTULATED LPWR ACCIDENTS
- DEVELOP METHODS
  - ANALYTICAL DESCRIPTION
  - ACCIDENT RECOGNITION
  - MANUAL/AUTOMATIC PLANT STABILIZATION AND RECOVERY

NCK-1

PRIMARY PROGRAM EMPHASIS

- CREATE AN EXPERIMENTAL DATA BASE REFLECTING A WIDE SPECTRUM OF ACCIDENT PHENOMENA AND PLANT STATES

- USE AND EVALUATE METHODS FOR RECOGNITION, CONTROL, AND RECOVERY FROM ACCIDENT PHENOMENA

DATA BASE IS IMPORTANT TO IMPLEMENTATION OF NRC ACTION ITEMS

- EMERGENCY TRAINING OF SHIFT TECHNICAL ADVISORS AND OPERATORS (I.A.1.1, I.A.2.1)
- ANALYSIS OF SMALL BREAK LOCA, INADEQUATE COOLING (I.C.1)
- CHARACTERIZATION OF COOLANT INVENTORY AND NATURAL CIRCULATION (I.C.1)

DATA BASE IS IMPORTANT (CONT'D)

- EMERGENCY PROCEDURE UPGRADE, NSSS VENDOR REVIEW, NRC REVIEW (I.C.5, I.C.7, I.C.8)
- TRAINING FOR MITIGATION OF CORE DAMAGE (II.B.4)
- DEVELOPMENT OF INSTRUMENTS FOR ACCIDENT MONITORING, DETERMINATION OF INADEQUATE CORE COOLING (II.F.1, II.F.2)

NCK-

DATA BASE IS IMPORTANT (CONT'D)

- LOCATION AND NEEDED CHARACTER OF COOLANT SYSTEM VENTS (II.B.1)
- DEVELOPMENT AND APPROPRIATE LOCATION OF POST ACCIDENT SAMPLING SYSTEM (II.B.3)
- DESIGN OF AUXILIARY FEEDWATER INITIATION AND INDICATION SYSTEM (II.E.1)
- DEGRADED CORE RULEMAKING (II.B.8)

NCK-

OTHER IMPORTANT DATA BASE USES

- RESOLVE SPECIFIC NRC CONCERNS (PUMP ON, OFF)
- DEVELOP ANALYTICAL METHODS THAT CHARACTERIZE PLANT ACCIDENT RESPONSE
- BOUNDARY CONDITIONS AND PERSPECTIVE TO ASSESS SEPARATE EFFECTS AND NONNUCLEAR TESTS

NCK-

OPERATIONAL METHODS EFFORT IMPORTANT TO IMPLEMENTATION OF NRC ACTION ITEMS

- CONTROL ROOM STAFFING REQUIREMENTS (I.A.1.3)
- EMERGENCY PROCEDURE UPGRADE, NSSS VENDOR REVIEW, NRC REVIEW (I.C.5, I.C.7, I.C.8)
- ESTABLISH UPGRADE PLANS FOR CONTROL ROOMS AND NRC AUDIT OF PLANS (I.D.1)

NCK-

OPERATIONAL METHODS EFFORT (CONT'D)

- TRAINING FOR CORE DAMAGE MITIGATION (II.B.4)
- DEVELOP INSTRUMENTS FOR MONITORING ACCIDENTS AND INADEQUATE CORE COOLING (II.F.1, II.F.2)
- DEVELOPING AND UPGRADING EMERGENCY SUPPORT FACILITIES (III.A.1.2)

NCK-

OTHER IMPORTANT OPERATIONAL METHODS USES

- DEVELOP METHODS FOR ACCIDENT RECOGNITION AND EFFECTIVE RESPONSES
- DEVELOP METHODS FOR REAL TIME INFORMATION VALIDATION DURING ACCIDENTS
- DEVELOP PREFERRED COURSES OF ACCIDENT RESPONSE

NCK-

LOFT TEST PROGRAM

SERIES	
L1	ISOTHERMAL CONDITIONS (NON-NUCLEAR)
L2	LARGE BREAKS
L3	SMALL BREAKS
L4	ALTERNATE ECC SYSTEMS
L5	INTERMEDIATE BREAKS

NCK-3

LOFT TEST PROGRAM (CONT'D)

SERIES	
L6	ANTICIPATED TRANSIENTS
L7	STEAM GENERATION TUBE RUPTURES
L8	CORE UNCOVERY AND FUEL DAMAGE
L9	TRANSIENT WITH MULTIPLE FAILURES
L10	ACCIDENT OVERRIDE TRANSIENT

NCK-4

TEST SELECTION AND SEQUENCE

- NRC/INDUSTRY INFORMATION NEED
- UNIQUENESS OF LOFT CONTRIBUTION
- RISK OF EVENTS TO BE STUDIED
- PLANT EQUIPMENT/ANALYSES PREREQUISITES

NCK-5

INVOLVEMENT OF INDUSTRY FACTIONS

- IMPORTANCE OF WIDELY BASED EVALUATION, PARTICIPATION, DISSEMINATION IN OPEN FORUMS

- METHODS IN USE
  - REVIEW GROUPS
  - FOREIGN PARTICIPANTS
  - UTILITY GROUP PRESENTATIONS
  - GOVERNMENT AGENCIES
  - UNIVERSITY CONTRACTS AND PRESENTATIONS
  - WORKSHOPS AND SEMINARS
  - TECHNICAL SOCIETY PARTICIPATION

NCK-10

SUMMARY

LOFT OFFERS UNIQUE FACILITY AND PERSONNEL CAPABILITIES

CAPABILITY IS RELEVANT TO NEEDS FOR UNDERSTANDING OF A WIDE RANGE OF ACCIDENT SEQUENCES AND THEIR RECOGNITION AND CONTROL

CAPABILITY IS OPERATIONAL WITH OUTSTANDING RECORD OF ACHIEVEMENT

NCK-11

SEMISCALE INTEGRAL AND

SEPARATE EFFECTS PROGRAM

OCTOBER 27, 1980

AFTERNOON SESSION - RED AUDITORIUM

PRESENTED BY:

WARREN C. LYON

U.S. NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND

The objective of the Semiscale program is to provide information regarding pressurized water reactor (PWR) transients and loss of coolant accidents (LOCAs) to aid the U.S. Nuclear Regulatory Commission (NRC) in investigating, understanding and improving light water reactor safety. To achieve this objective, personnel experience, test data, analysis techniques, instrumentation and analyses developed in this program are applied to:

1. Provide a data base for transient and LOCA code development and assessment;

2. Evaluate the adequacy of NRC regulations pertaining to thermal hydraulic PWR behavior during transient conditions, including accidents;

3. Provide information and expertise to aid in regulation improvement;

4. Relate PWR instrument readings to the true system state for a wide range of transient conditions to assess and improve operation of PWR's, particularly under off normal and accident conditions;

5. Provide technical advice and support to the NRC on a routine basis;

6. Immediately assist the NRC in the event of an accident or other high priority support need; and

7. Support the United States and International Standard Problem programs.

Support to NRC from the Semiscale program has been provided for many years. Tests which have been conducted are summarized in Table 1. Vessel blowdown testing was initiated in 1965. Since then, there have been many changes in the test facility, including:

1968: Single loop plus vessel blowdown capability with and without core heat and ECC.

1974: MOD1 - Two loop volume scaled to LOFT with LOFT height major components.

1978: MOD3 - Two loop with vessel and broken loop having PWR elevations, volume scaled to PWR, unbroken loop roughly as in MOD1.

1980: MOD 2A - Two loop, volume scaled to PWR, major components actively simulated and having PWR elevations.

TABLE 1

SEMISCALE TEST SUMMARY

Test Series	Dates	No. of Tests	Description
500-700	65 - 68	∿100	Vessel Blowdown
800	69 - 71	∿ 40	Vessel and Single Loop Blowdown and ECC
900	71	One Test	As Above, Longer Core
1000	72	9	Vessel and Two Loops Blowdown and ECC
-	72 - 74	∿ 90	Countercurrent Flow and Hot Wall Tests
1	08/74 - 07/75	8	Isothermal Blowdown
2	05/75 - 11/75	8	Blowdown Heat Transfer
3	01/76 - 03/76	12	Reflood Heat Transfer
4	05/76 - 10/76	9	Alternate ECC Concepts
5	10/76 - 03/77	9	Alternate ECC Locations
6	04/77 - 08/77	6	Integral LOCA
7	06/78 - 05/80	10	MOD3 Baseline (LOCA Series)
28	07/77 - 09/77	13	Tube Rupture, Power Control
TMI	03/79 - 07/79	10	Three Mile Island
SB	11/79 - 05/80	10	Small-Break
TR	05/80 - 06/80	2	Preliminary Station Blackout Transients

Each change in general provided more detail and a better simulation of PWR characteristics. The most recent (MOD 2A) version is shown in Figure 1. It consists of a pressure vessel with complete internals; an intact loop with an active steam generator, pump and pressurizer; a broken loop with an active steam generator, pump and rupture assembly; a pressure suppression system with a suppression tank, heater and steam supply system; coolant injection accumulators for the intact and broken loop cold legs and an accumulator for the intact loop hot leg; and high and low pressure injection pumps for each loop. The pressure vessel consists of an upper head that allows upper head ECC injection, an upper plenum, heated core region, lower plenum and an external downcomer pipe and inlet annulus. The reactor core is simulated by an electrically heated core containing 25 heater rods. Total core power is two megawatts. All major components and subcomponents are one to one elevation scaled to a four loop PWR. External heaters and additional thermal insulation have been added to control heat losses, and a comprehensive program to control leaks has been initiated to enhance PWR typicality for long term transients.

There are essentially four restrictions regarding application of the Semiscale facility to provide thermal hydraulic related data:

1. Semiscale is in some respects a small scale facility. Although all major component elevations are identical to those of a four loop PWR, power and volume are scaled. Reasonable care is taken to reproduce PWR event timing and phenomena, but distortions occur due to the one dimensional nature of the facility and other non-typicalities. Normally, the data must be carefully extrapolated to PWR conditions, generally via computer programs developed in part from Semiscale and other data.

2. Semiscale is non-nuclear, and core heat is electrically simulated. Although feedback from selected parameters is available for core heat control, distortions will occur due to the fixed heating rate distribution within the simulated core and due to the non-typical behavior of electrically heated rods.

3. Available electrical power limits the experimental facility to test simulation in the vicinity of PWR full power or below. Power transients significantly above full power cannot be investigated.

4. The PWR secondary system is not simulated on a component basis. The facility is designed to simulate the conditions that are "seen" by the steam generator secondary. Therefore, PWR secondary conditions to be simulated must be specified prior to the experiment. Feedback from the primary or steam generator secondary can be used as control for the simulated secondary parameters.

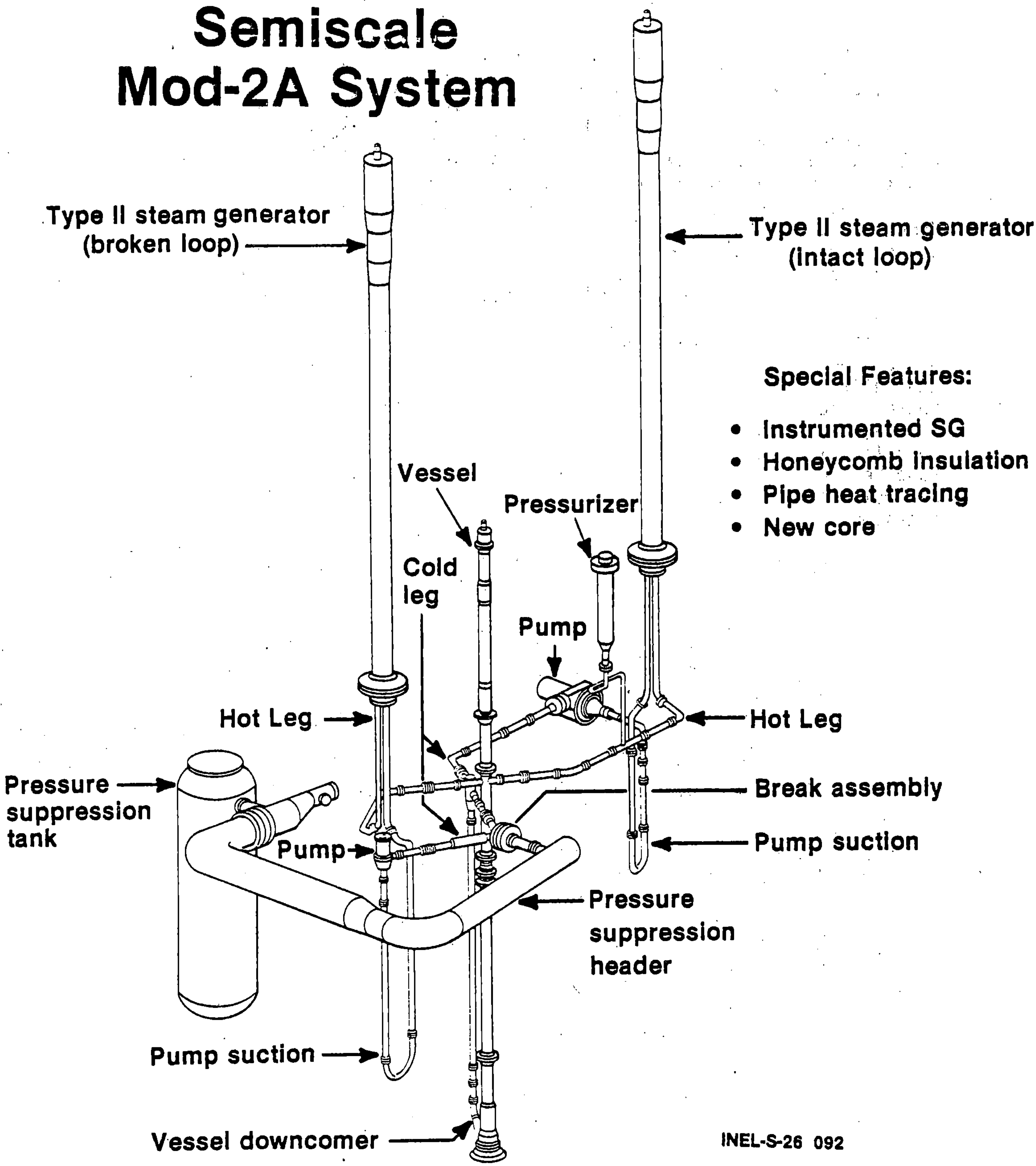


Figure I. Semiscale MOD 2A System.

Tests conducted during the past year and planned tests are outlined in the enclosed slides. The emphasis is on transient and small-break behavior, with specialization in areas where new data are desired for PWR events identified as having the highest risks.

The test results during the past year are summarized in the following two papers:

(1) Leach, L. P., "Key Results of Semiscale Transient Thermal Hydraulic Tests," EG&G Idaho, Inc.

(2) Johnsen, G. W., "Results of Semiscale Pumps On/Off Experiments," EG&G Idaho, Inc.

SEMISCALE FUTURE

COMPLETE PWR MOD 2A LOCA AND TRANSIENTS INVESTIGATION

2 X 4 CONFIGURATION?

TWO LOOP TEST APPARATUS (TLTA)

* MODEL OF A BWR UTILIZING A SINGLE

SEMISCALE INTEGRAL AND SEPARATE EFFECTS PROGRAMS

WARREN C. LYON

OCTOBER 27, 1980

AFTERNOON SESSION - RED AUDITORIUM

U. S. NUCLEAR REGULATORY COMMISSION

EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

GAITHERSBURG, MARYLAND

Enclosure

SEMISCALE

OBJECTIVE:

PROVIDE SEPARATE EFFECT AND INTEGRAL FACILITY THERMAL HYDRAULIC DATA PERTINENT TO PWR OPERATIONAL TRANSIENTS AND LOCAS.

SEMISCALE HISTORY

1965 - 67 100 VESSEL BLOWDOWNS

1968 - 72 50 SINGLE AND TWO LOOP WITH VESSEL BLOWDOWN

- CORE HEAT CAPABILITY
- ECC CAPABILITY

1972 - 74 90 COUNTERCURRENT FLOW AND HOT WALL TESTS

1974 - 77 65 MOD I TESTS

- TWO LOOPS
- VOLUME, POWER SCALED TO LOFT
- LOFT ELEVATIONS
- VESSEL COMPONENTS
- ACTIVE INTACT LOOP STEAM GENERATOR
- ACTIVE INTACT LOOP PUMP
- COMPLETE ECC

1978 - 80 32 MOD 3 TESTS

- TWO LOOPS WITH ACTIVE COMPONENTS
- VESSEL, BROKEN LOOP VOLUME SCALED TO PWR WITH PWR ELEVATIONS
- VESSEL COMPONENTS, INCLUDING UHI
- COMPLETE ECC

1980 TESTS

- SMALL-BREAK LOFT COUNTERPART
- SMALL-BREAK AUDIT SUPPORT
- PUMPS ON/OFF
- PRELIMINARY STATION BLACKOUT

1980 INITIATE MOD 2A TESTS

- TWO LOOPS WITH ACTIVE COMPONENTS
- VOLUME AND POWER SCALED TO PWR
- PWR ELEVATIONS
- VESSEL COMPONENTS INCLUDING UHI
- COMPLETE ECC
- HEAT LOSS CONTROL
- FLUID LOSS CONTROL

MOD 2A TEST PLAN

- SYSTEM CHARACTERIZATION
- 10 PERCENT BREAK
- ST. LUCIE OVERCOOLING?
- 2.5 PERCENT BREAK
- NATURAL CONVECTION, INCLUDING REFLUX AND TWO COMPONENT CONDITIONS
- STATION BLACKOUT
- STEAM GENERATOR TUBE BREAK

MOD 2A TEST PLAN (CONTINUED)

- STEAM LINE BREAK
- FEED LINE BREAK
- LOSS OF FEED
- OVERCOOLING TRANSIENT
- UHI
- ALTERNATE ECC
- OFF DESIGN TRANSIENTS

KEY RESULTS OF SEMISCALE TRANSIENT THERMAL-HYDRAULIC TESTS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

L. P. Leach
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

KEY RESULTS OF SEMISCALE TRANSIENT THERMAL-HYDRAULIC TESTS

L. P. Leach
EG&G Idaho, Inc.

Thirteen tests were performed in the Semiscale facility during the past year to aid NRC in licensing evaluations. Specific licensing issues addressed include:

1. How appropriate are the requirements of 10 CFR 50.46, and Appendix K, in providing conservatism for small break LOCA analysis?
2. How well can the computer models used for small break safety analysis predict the observed small break phenomena?
3. How complete is our understanding of the controlling phenomena in a station blackout?

Eleven of these tests were PWR small break loss-of-coolant accident (LOCA) simulations, and two were scoping tests to evaluate key features of PWR behavior following a loss of ac and dc power (station blackout). This paper presents key results from seven of these tests; six "pump effect" tests are addressed in a companion paper, "Results of Semiscale Pumps On/Off Experiments."

All of the tests described herein were performed in the Semiscale Mod-3 facility. This facility operates at normal PWR pressure and temperature (15.5 MPa, 555 K), and contains a 25-rod electrically heated core simulator. There are two loops, an intact loop which represents three loops in a commercial four loop PWR, and a broken loop which simulates the loop in which the LOCA occurs. The core and broken loop steam generator are full height simulations of a PWR core and steam generator and the intact loop steam generator is scaled to the LOFT steam generator.

The results of Tests S-SB-4 and S-SB-4A, which were performed prior to LOFT Test L3-1, were used to confirm the safety of the LOFT test. Comparison of data from S-SB-4 and L3-1 shows excellent agreement, confirming scaling of Semiscale to LOFT and indicating that the dominant small break phenomena in a 2.5% break are relatively insensitive to hardware variations. Comparison of results from S-SB-4 to S-SB-4A, in which core power was increased to offset Semiscale system heat losses, indicate this method of offsetting the heat losses was applicable only prior to uncovering of the core.

Data from Test S-07-10D, a 10% communicative cold leg break experiment, was compared to calculations provided by reactor vendors and national laboratories participating in the NRC Standard Problem program. Comparisons indicated significant differences between all the calculated results and the measured results. In none of the calculations was both the core uncovering time and the peak cladding temperature predicted accurately.

The results of the station blackout scoping tests, S-TR-1 and S-TR-2, will be used in defining more typical future tests and to identify modifications to the Semiscale system to improve the simulation capability. The key result of interest from these scoping tests was the formation of superheated steam in the vessel and hot legs prior to excessive core heatup. This result indicates that primary system pressure/temperature design limits must be taken into account in evaluating this type of transient.

Although the Semiscale system has been designed to replicate transient behavior of a PWR system as closely as possible, the small size of the system, design compromises, and differences between PWR designs make it inappropriate to relate the data from Semiscale tests directly to a PWR. Rather, the approach used is to assess and improve the analytical models used to predict PWR performance by way of the Semiscale tests.

Overall, the tests have provided an improved understanding of the phenomena important in slow transients, and have therefore aided the licensing process for small break LOCA safety analysis. Since the analysis performed to date has been on a "best estimate" basis, the degree of conservatism in 10 CFR 50 requirements have not been quantified. The best estimate calculations have provided significant insight regarding the accuracy, and appropriate use of the models for small break analyses.

REFERENCES

1. Semiscale Program, Appendix SB - Experimental Operating Specification - S-SB-2 through S-SB-5 - Semiscale Mod-3 Small Break Test Series, EG&G Idaho, Inc., SEMI-TR-011 (August 1979).
2. S. E. Dingman, Experimental Operating Specification (EOS) for Small Break Test S-SB-2A, D. J. Olson letter to R. E. Tiller (DJO-3-80), EG&G Idaho, Inc., (January 4, 1980).
3. B. W. Murri, D. M. Snider, S. E. Dingman, and C. P. Fineman, Test Prediction for Semiscale Mod-3 Test S-SB-2 - Small Break Series, EG&G Idaho, Inc., EGG-SEMI-5025 (September 1979).
4. Semiscale Program, Test Prediction for Semiscale Mod-3 Small Break Test S-SB-2A, D. J. Olson letter to R. E. Tiller (DJO-2-80), EG&G Idaho, Inc., (January 4, 1980).
5. J. L. Steiner and D. M. Snider, Test Prediction for Semiscale Mod-3 Test S-SB-4 - Small Break Test Series, EG&G Idaho, Inc., EGG-SEMI-5047 (October 1979).
6. J. M. Cozzuol, Quick Look Report for Semiscale Mod-3 Small Break Test S-SB-2, EG&G Idaho, Inc., EGG-SEMI-5073 (December 1979).
7. T. J. Fauble, Quick Look Report for Semiscale Mod-3 Small Break Test S-SB-2A, EG&G Idaho, Inc., EGG-SEMI-5113 (March 1980).
8. T. J. Fauble and J. M. Cozzuol, Quick Look Report for Semiscale Mod-3 Small Break Tests S-SB-4 and S-SB-4A, EG&G Idaho, Inc., EGG-SEMI-5062 (November 1979).
9. R. G. Hanson, Analysis Report for Semiscale Mod-3 Station Blackout Tests S-TR-1 and S-TR-2, EG&G Idaho, Inc., EGG-SEMI-5227 (August 1980).
10. M. N. Arevalo and K. E. Sackett, Experiment Data Report for Semiscale Mod-3 Small Break Test Series (Tests S-SB-4 and S-SB-4A), NUREG/CR-1293, EGG-2021, April 1980.
11. D. H. Miyasaki, Experiment Data Report for Semiscale Mod-3 Small Break Test Series, (Tests S-SB-2 and S-SB-2A), NUREG/CR-1459, EGG-2038, June 1980.

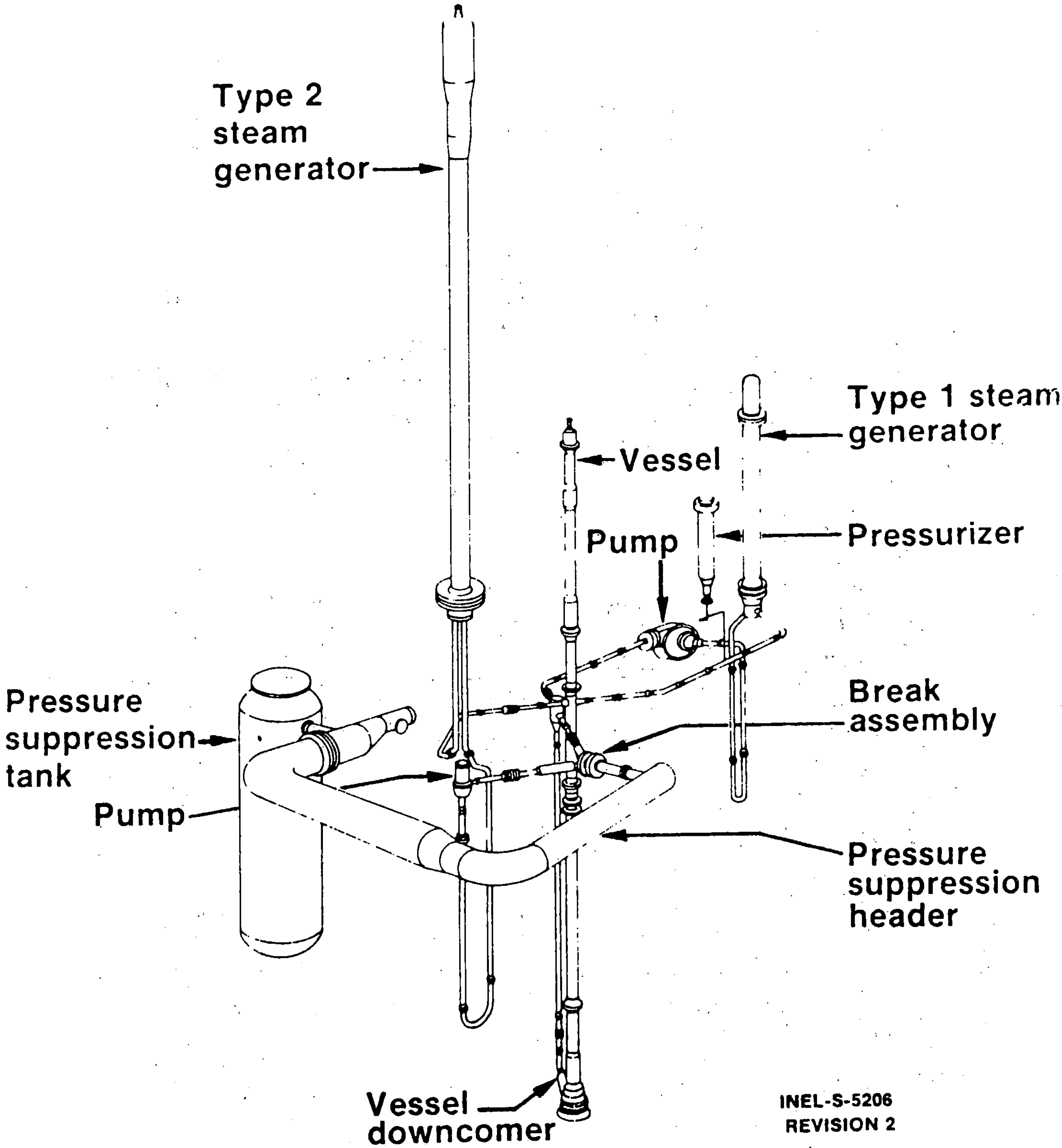
KEY RESULTS OF SEMISCALE TRANSIENT THERMAL-HYDRAULIC TESTS

by

T.K. LARSON

EG&G Idaho, Inc.

Semiscale Mod-3 System



INEL-S-5206
REVISION 2

SEMISCALE MOD-3 TESTING
(FY-1980)

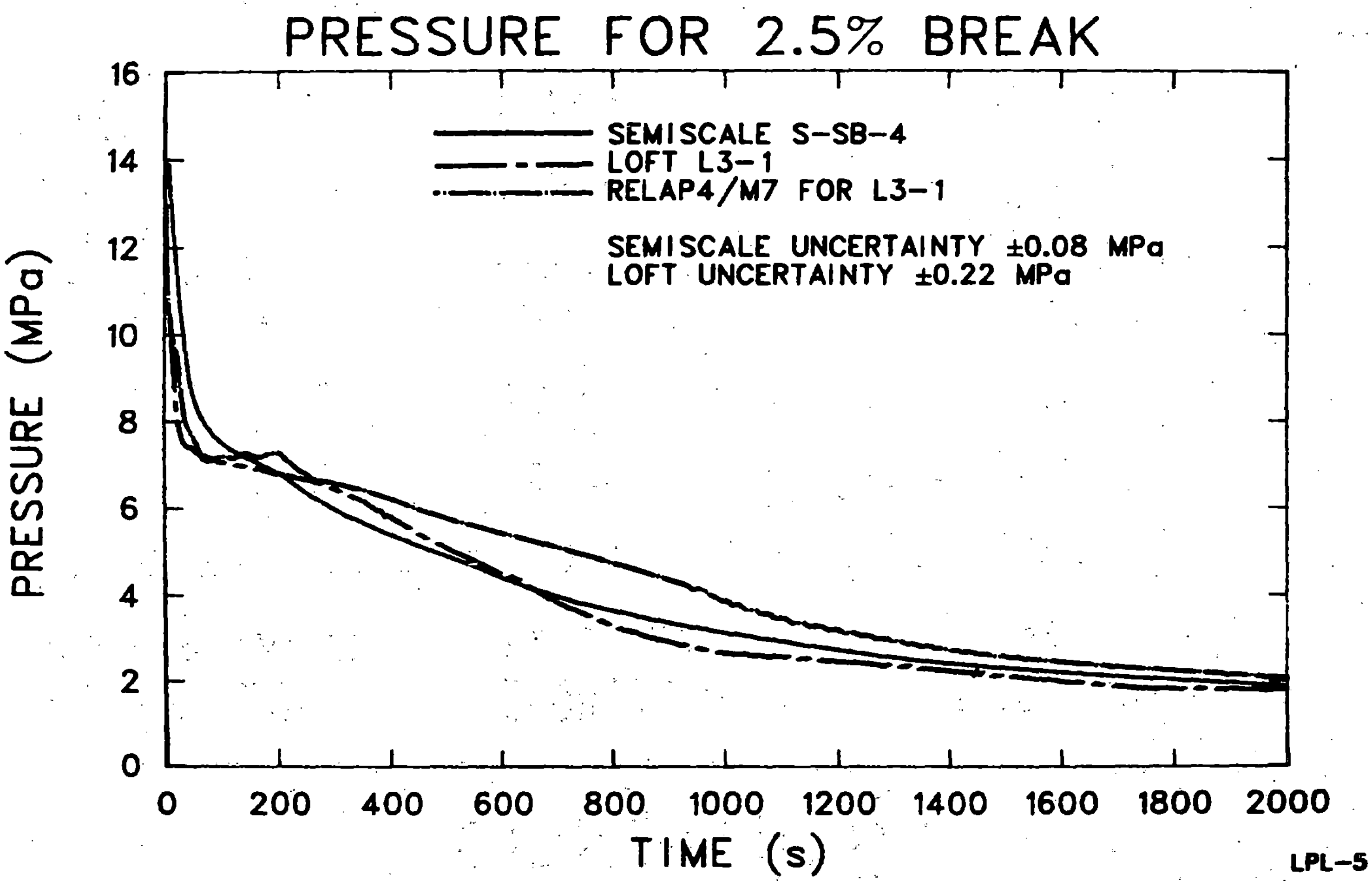
TEST	TYPE	OBJECTIVE(S)
S-SB-2.2A, 4.4A	2.5% COLD LEG BREAK	LOFT TEST L3-1 AUDIT CALCULATIONS
S-TR-1.2	STATION BLACKOUT	SYSTEM OPERATION THERMAL HYDRAULIC BEHAVIOR
S-07-10D	10% COLD LEG BREAK	NRC STANDARD PROBLEM

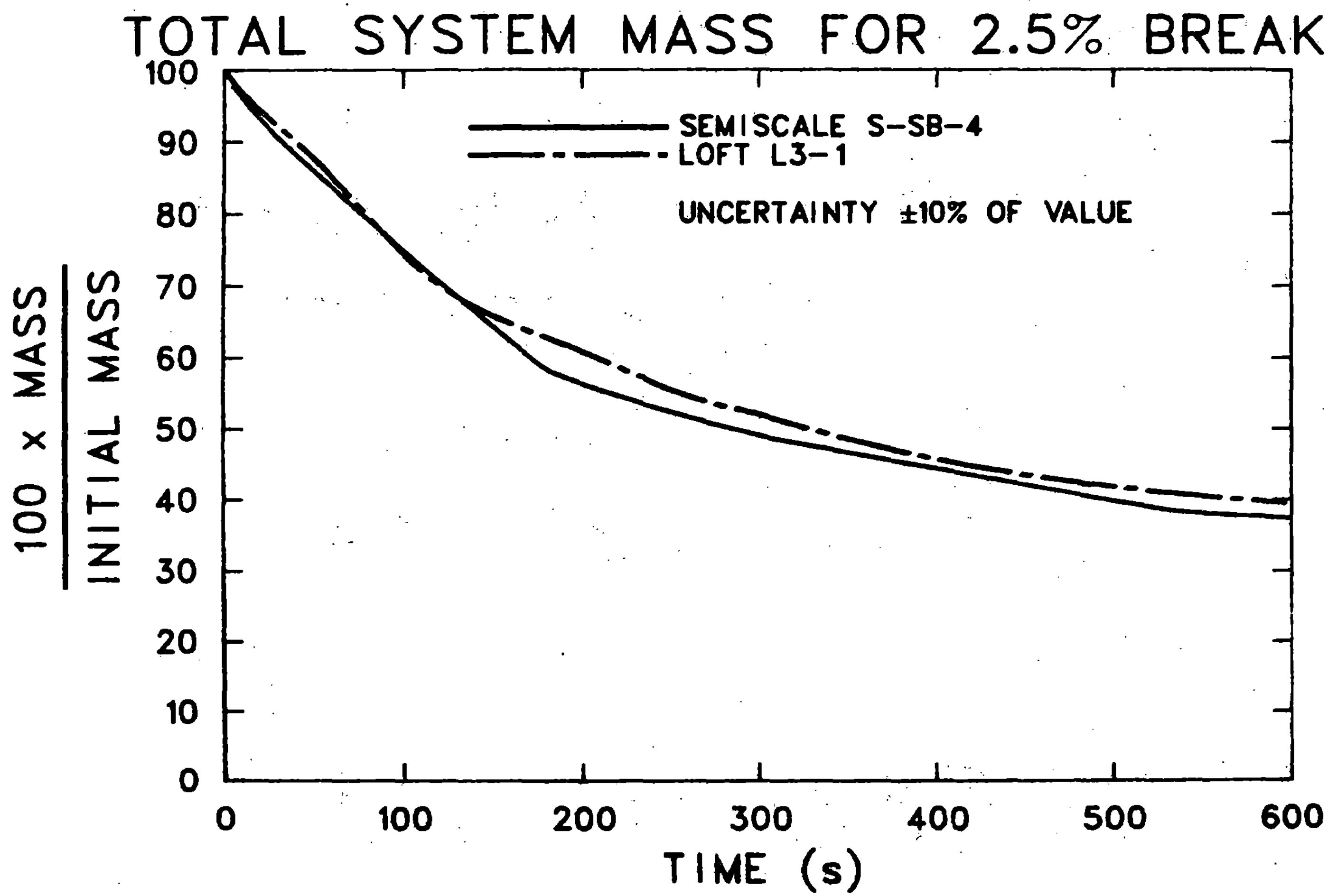
LPL-3

INITIAL AND OPERATING CONDITIONS

PARAMETER	S-SB-4/4A	LOFT L3-1
POWER (MW)	1.2 / 1.2	49
PRESSURE (MPa)	14.8 / 15.1	15.0
COLD LEG T (K)	558.2 / 558.7	554
ΔT (K)	20.0 / 19.2	20
BREAK SIZE, LOCATION	2.5% COLD LEG	2.5% COLD LEG
HPIS	1 TRAIN	1 TRAIN
LPIS	1 TRAIN	1 TRAIN

LPL-4


PRESSURE FOR 2.5% BREAK
SEMISCALE S-SB-4
LOFT L3-1
RELAP4/M7 FOR L3-1
SEMISCALE UNCERTAINTY ±0.08 MPa
LOFT UNCERTAINTY ±0.22 MPa
PRESSURE (MPa)
0
2
4
6
8
10
12
14
16
0
200
400
600
800
1000
1200
1400
1600
1800
2000
TIME (s)
LPL-5


TOTAL SYSTEM MASS FOR 2.5% BREAK
SEMISCALE S-SB-4
LOFT L3-1
UNCERTAINTY ±10% OF VALUE
100 x MASS / INITIAL MASS
0
10
20
30
40
50
60
70
80
90
100
0
100
200
300
400
500
600
TIME (s)

LPL-6

TEST S-07-10D

INITIAL AND OPERATING CONDITIONS

- INITIAL POWER - 1.94 MW
- INITIAL PRESSURE - 15.7 MPa
- CORE ΔT - 35 K
- CORE FLOW - 9.72 kg/s

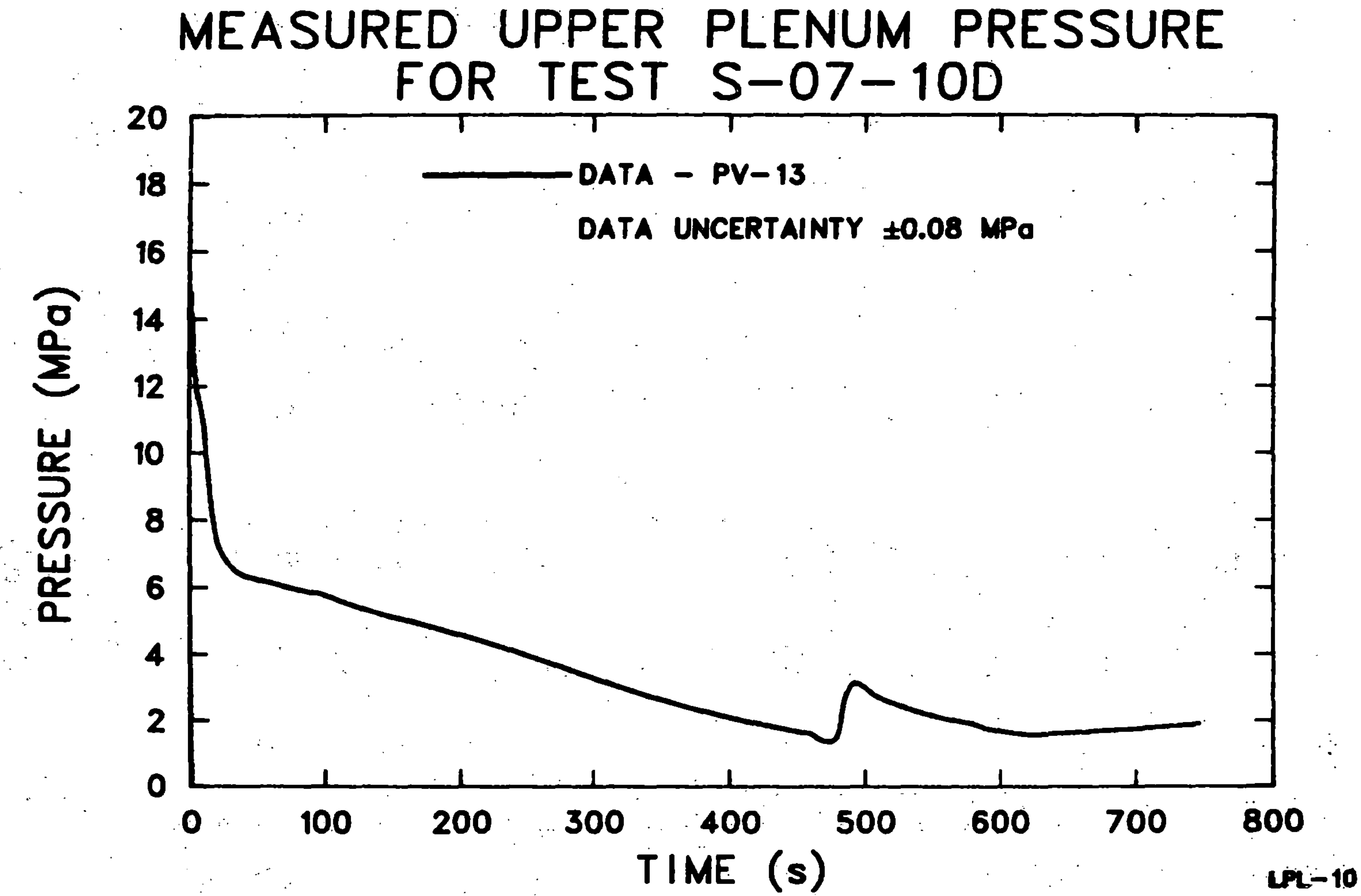
LPL-9

TEST S-07-10D (CONT.)

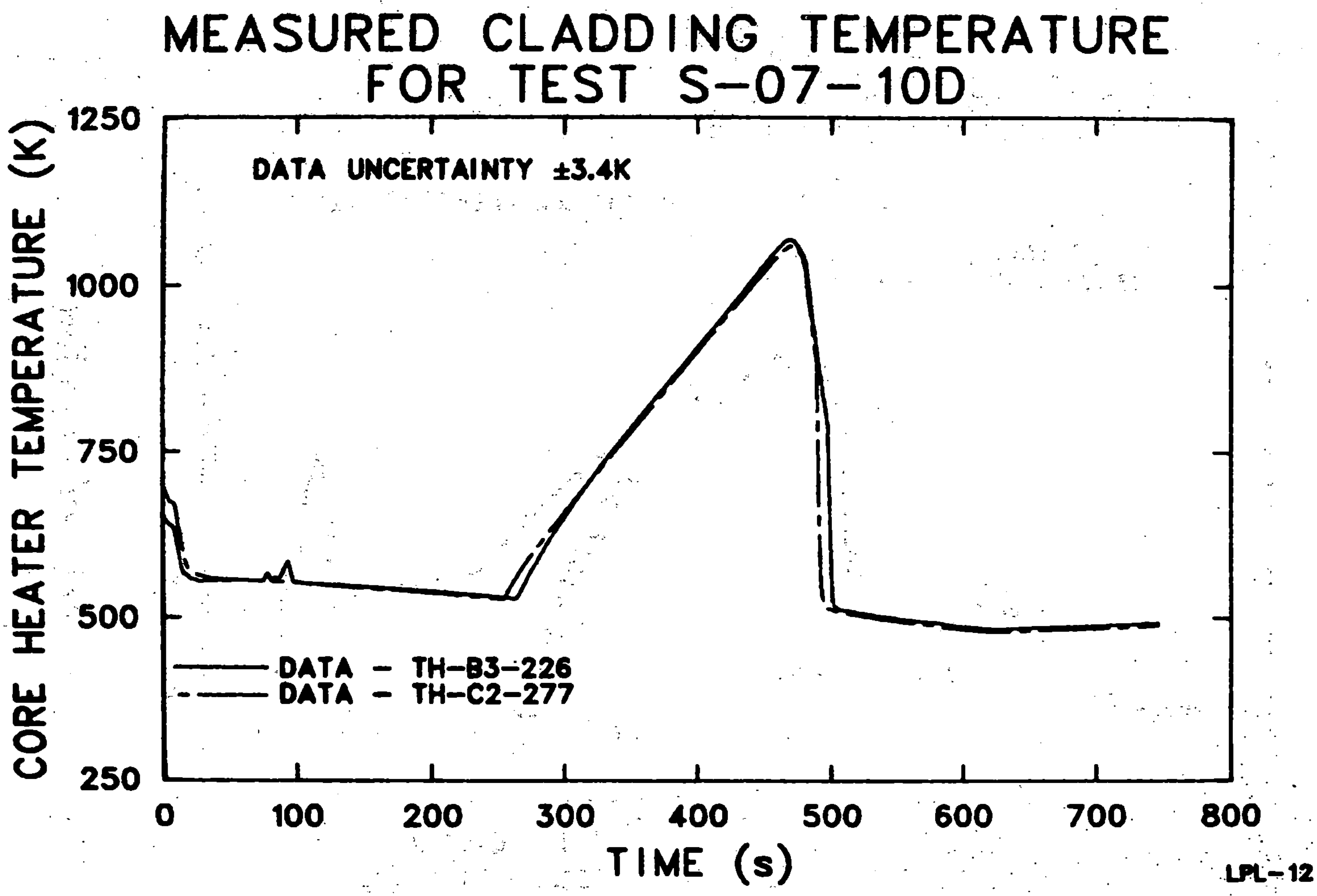
INITIAL AND OPERATING CONDITIONS

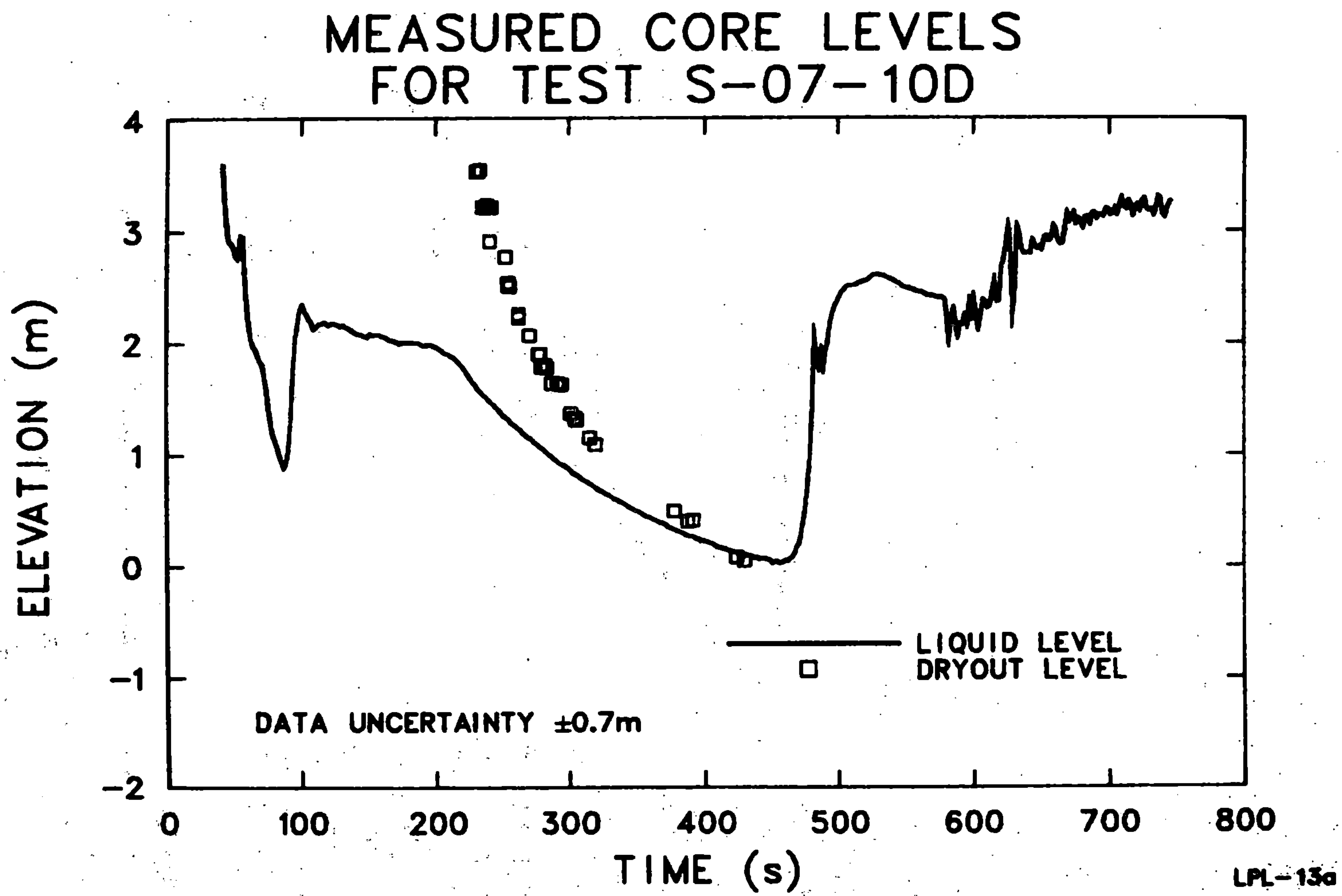
- CORE SCRAM WHEN PRESSURIZER PRESSURE REACHED 12.41 MPa
- PUMP TRIP AFTER PRESSURIZER PRESSURE REACHED 12.41 MPa
- FEEDWATER AND STEAM VALVE TRIP AFTER PRESSURIZER PRESSURE REACHED 12.41 MPa

LPL-9A



MEASURED UPPER PLENUM PRESSURE
FOR TEST S-07-10D
DATA - PV-13
DATA UNCERTAINTY ±0.08 MPa
PRESSURE (MPa)
TIME (s)
0
2
4
6
8
10
12
14
16
18
20
0
100
200
300
400
500
600
700
800
LPL-10


MEASURED CLADDING TEMPERATURE
FOR TEST S-07-10D
DATA UNCERTAINTY ±3.4K
CORE HEATER TEMPERATURE (K)
1250
1000
750
500
250
0
100
200
300
400
500
600
700
800
TIME (s)
DATA - TH-B3-226
DATA - TH-C2-277
LPL-12


MEASURED CORE LEVELS
FOR TEST S-07-10D
ELEVATION (m)
TIME (s)
LIQUID LEVEL
DRYOUT LEVEL
DATA UNCERTAINTY ±0.7m
4
3
2
1
0
−1
−2
0
100
200
300
400
500
600
700
800
LPL-13a

BLACKOUT SIMULATIONS
TESTS S-TR-1, S-TR-2

INITIAL AND OPERATING CONDITIONS

- INITIAL POWER - 1.97 MW
- CORE ΔT - 34 K
- CORE FLOW - 11.7 kg/s
- CORE POWER DECAY BEGINNING AT 3.4 s
- FEEDWATER VALVE CLOSED AT 5 s

LPL-14

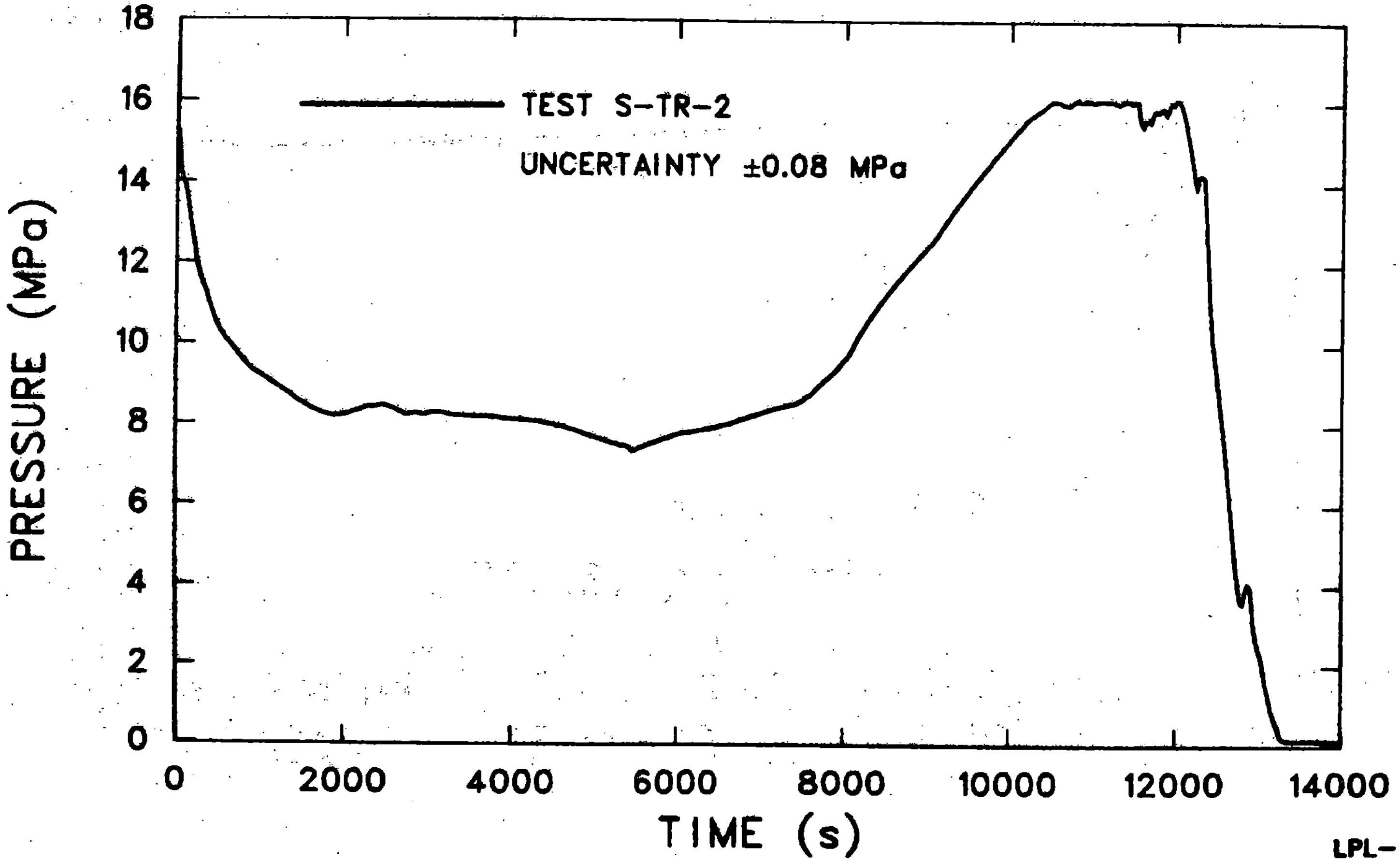
BLACKOUT SIMULATIONS
TEST S-TR-1, S-TR-2 (CONT.)

INITIAL AND OPERATING CONDITIONS

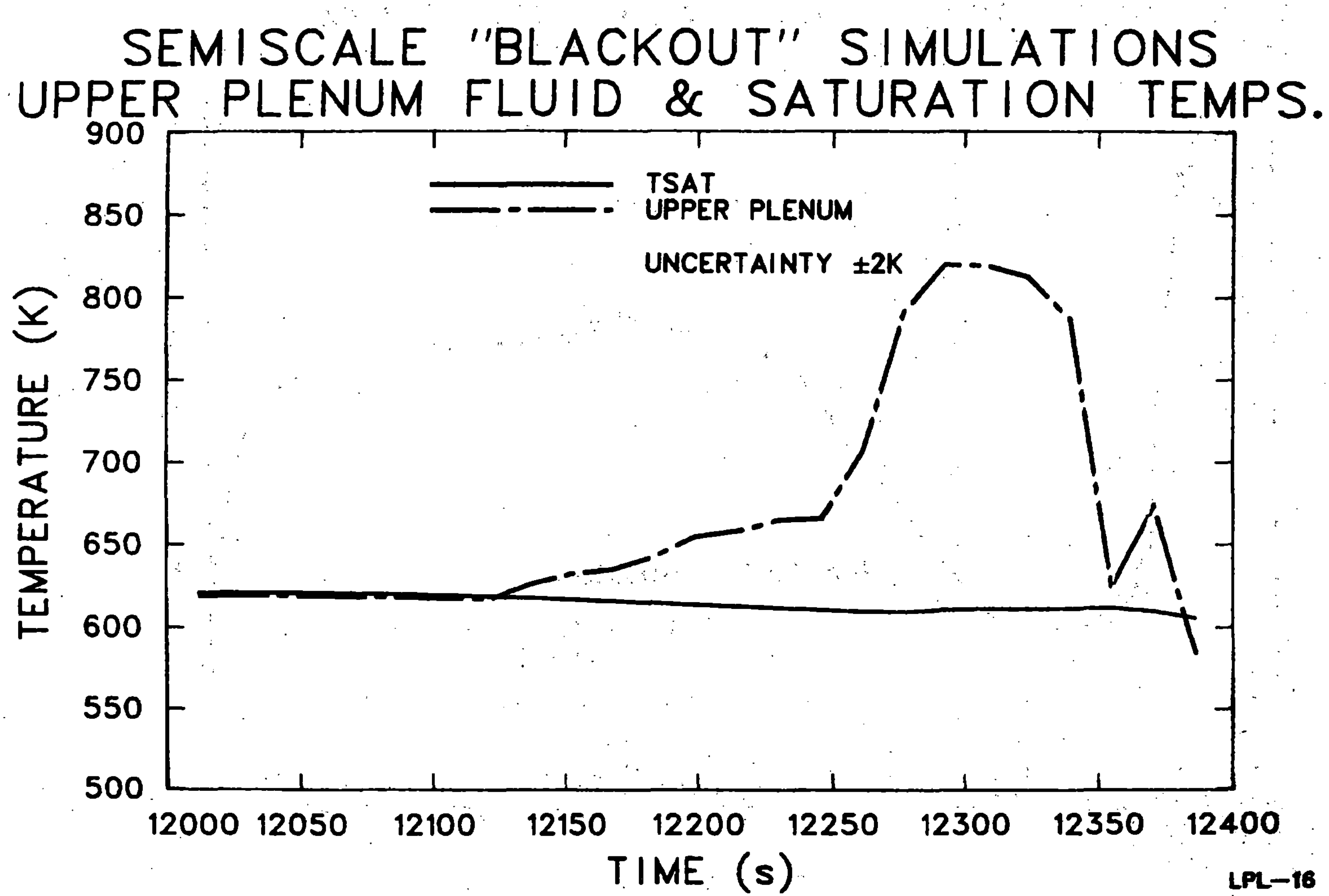
- PRIMARY PUMPS TRIPPED AT 60 s
- STEAM GENERATOR SECONDARY VOLUMES DRAINED AT 57 MIN
- CORE POWER REDUCED TO DECAY HEAT AFTER CORE UNCOVERS

LPL-14A

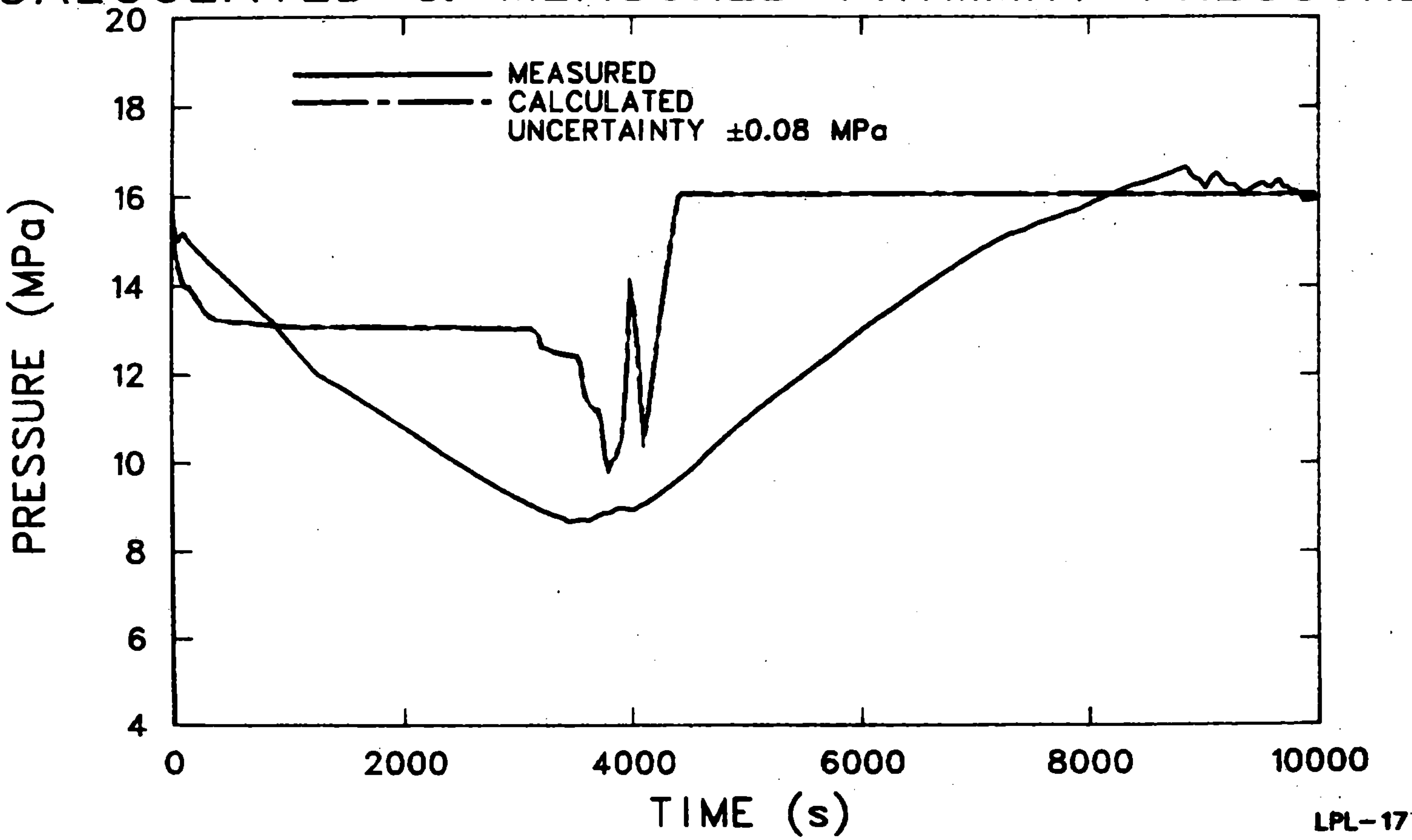
SEMISCALE "BLACKOUT" SIMULATIONS



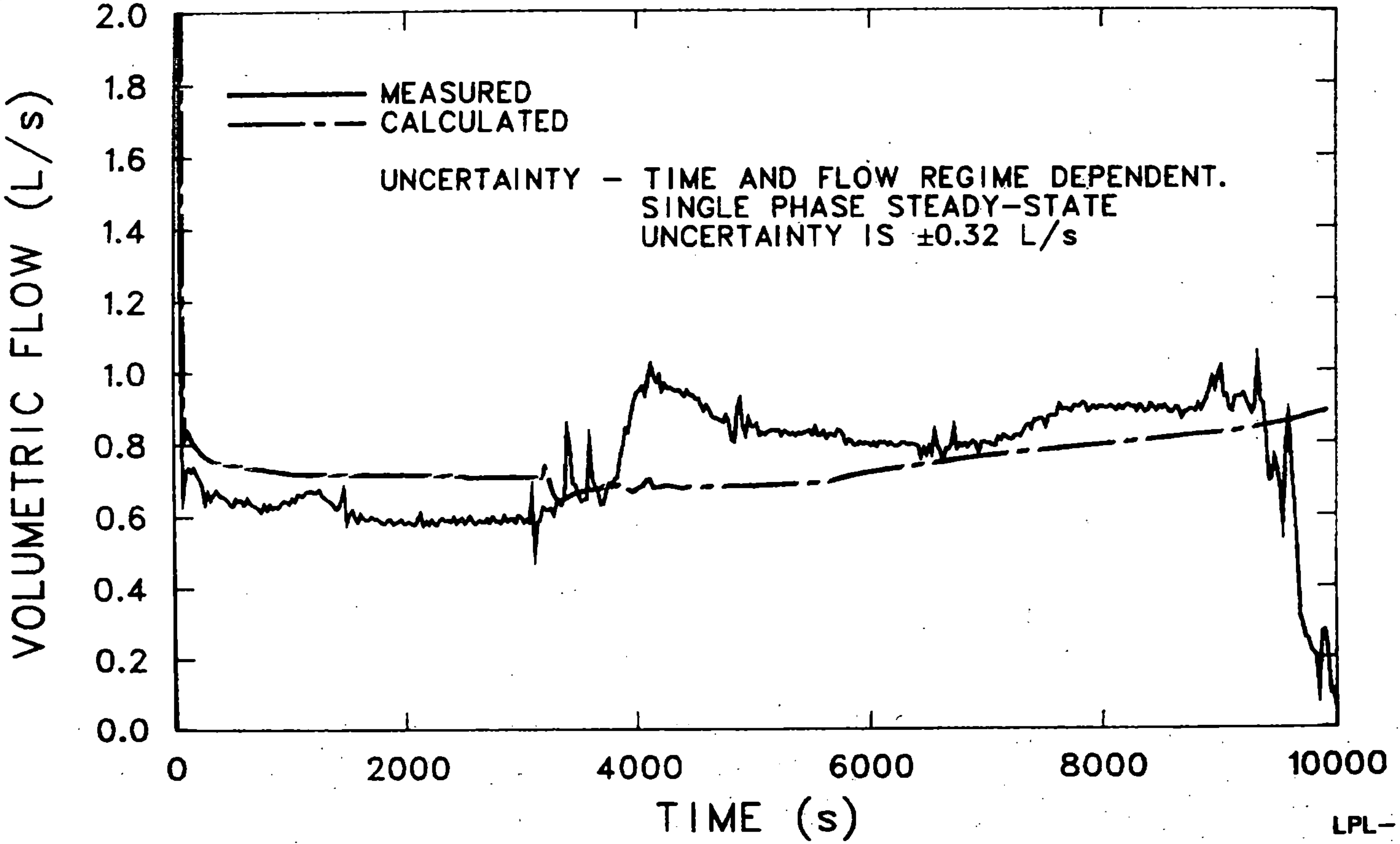
LPL-15


SEMISCALE "BLACKOUT" SIMULATIONS
UPPER PLENUM FLUID & SATURATION TEMPS.
TSAT
UPPER PLENUM
UNCERTAINTY ±2K
TEMPERATURE (K)
900
850
800
750
700
650
600
550
500
12000 12050 12100 12150 12200 12250 12300 12350 12400
TIME (s)
LPL—16

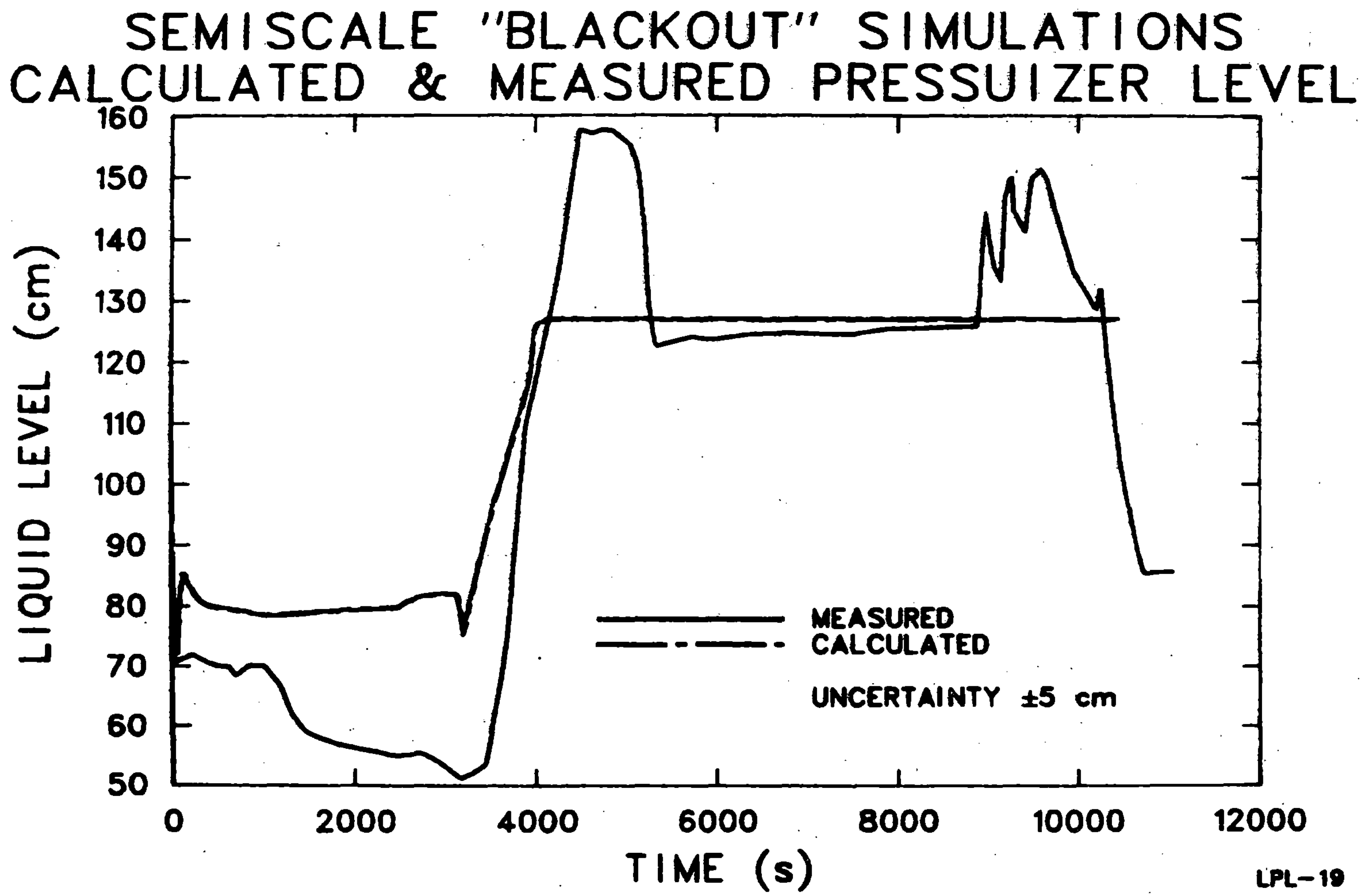
SEMISCALE "BLACKOUT" SIMULATIONS
CALCULATED & MEASURED PRIMARY PRESSURE



SEMISCALE "BLACKOUT" SIMULATIONS
CALCULATED & MEASURED INTACT LOOP FLOW



LPL-18


SEMISCALE "BLACKOUT" SIMULATIONS
CALCULATED & MEASURED PRESSUIZER LEVEL
LIQUID LEVEL (cm)
160
150
140
130
120
110
100
90
80
70
60
50
0
2000
4000
6000
8000
10000
12000
TIME (s)
MEASURED
CALCULATED
UNCERTAINTY ±5 cm
LPL-19

CONCLUSIONS

- LOFT TEST L3-1 SAFE (CONFIRMED)
- S-SB-4 COMPARED WELL TO L3-1 $\Longrightarrow$ GOOD SCALING
- LARGE HEAT LOSS IN SEMISCALE AFFECTS NATURAL CIRCULATION

CONCLUSIONS (CONT.)

- INCREASED CORE POWER
  TO OFFSET HEAT LOSS
  ONLY PART SATISFACTORY

- SIGNIFICANT DISAGREEMENT
  BETWEEN CALCULATIONS AND
  DATA FROM S-07-10D

LPL-21

CONCLUSIONS (CONT.)

- SMALL BREAK ANALYSIS CAPABILITY IMPROVED
- STATION BLACKOUT EVALUATIONS REQUIRE NON-EQUILIBRIUM ANALYSIS; EVALUATION OF P/T LIMITS

RESULTS OF SEMISCALE PUMPS ON/OFF EXPERIMENTS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

G. W. Johnsen
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

RESULTS OF SEMISCALE PUMPS ON/OFF EXPERIMENTS

G. W. Johnsen
EG&G Idaho, Inc.

Six experiments were conducted in the Semiscale Mod-3 system to investigate the effect of primary coolant pump operation on thermal-hydraulic behavior during a small break loss-of-coolant accident (LOCA). The impetus for these experiments stemmed from licensing concerns based on analyses conducted as a result of the accident at the Three Mile Island nuclear power plant. PWR vendor computer code analyses predicted that continued pump operation might cause more severe coolant depletion, thereby jeopardizing core coolability if the pumps had to be shut down at some intermediate point during the LOCA.

The Semiscale experiments were designed to evaluate the effect of pump operation on primary coolant mass inventory and distribution. Three cold leg break and three hot leg break experiments were conducted. The break size simulated was 2.5% (of cold leg pipe flow area), representing a circular opening in the side of a PWR pipe of approximately 11 cm. For each of the experiments, emergency core coolant (ECC) was injected into the cold legs at scaled flowrates corresponding to the availability of a single high pressure injection system train. The accumulators and low pressure injection system were not used in these experiments so as to improve the experimental determination of coolant inventory. A condenser and weigh tank arrangement was connected downstream of the break to provide an accurate measurement of break discharge. Three different pump operation scenarios were imposed for both the cold and hot leg break cases; pump trip at scram, delayed trip, and continuous pump operation.

For the cold leg break, early pump trip caused greater primary coolant system mass depletion than observed in either the continuous pump operation or delayed trip cases. The difference in minimum transient coolant inventory was small, however, amounting to a difference of only 8% between the early trip and continuous pump operation cases. It was found that early pump trip caused highly subcooled ECC liquid to pool in the vicinity of the break, resulting in a greater break discharge rate early in the transient. Pump operation tended to homogenize primary coolant, thereby resulting in less cold leg fluid subcooling. Approximately 350 s into the transients, break flow was higher with the pumps running, but not to the extent that the difference which developed early in the transient was reversed.

The system hydraulic behavior for the hot leg break experiments was similar. When the coolant pumps were left running, higher density fluid was delivered to the hot portions of the system (hot legs, core, and upper vessel regions). However, because the break was located in the hot leg, more mass was lost out the break when the pumps were left running than when the pumps were tripped early. This led to greater system mass depletion when the pumps were left on. The minimum coolant inventory was approximately 27% lower in the pumps-running case than in the early pump trip case.

Continuous pump operation in both the cold leg and hot leg break cases caused a greater percentage of system coolant to reside in the vessel, since the pumps continued to deliver liquid. Moreover, in the delayed pump trip experiments, liquid stored in the hot legs drained into the vessel when the pumps were tripped, thus augmenting vessel inventory. Consequently, delayed pump trip actually proved beneficial in Semiscale.

RELAP4/MOD7 computer code calculations correctly predicted the differential effect the pump operation had in the hot and cold leg break cases. However, the computer code failed to adequately predict several hydraulic aspects of these transients, notably fluid density in the hot legs where the existence of countercurrent flow was evident.

In summary, these experiments have provided useful data that will contribute toward an ultimate resolution of the pump operation issue. Effects of scale preclude a direct extrapolation of these results to expected behavior in a full-size PWR. Nevertheless, the results have uncovered effects heretofore not considered in previous analyses and furthermore suggest that the differential system response caused by pump operation may be situational dependent. In addition, the data are providing a basis for refining computer code models so that more accurate predictions of PWR response under various scenarios are possible.

REFERENCES

1. B. Sheron, Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors, Nuclear Regulatory Commission, NUREG 0623, November 1979.

2. J. M. Cozzuol, Experimental Operating Specification (EOS) for Tests S-SB-P1 and S-SB-P2, G. W. Johnsen letter to R. E. Tiller (GWJ-8-80), EG&G Idaho, Inc., February 13, 1980.

3. L. P. Leach, Experimental Operating Specification (EOS) for Small Break Test S-SB-P7, L. P. Leach letter to R. E. Tiller (LPL-33-80), EG&G Idaho, Inc., March 13, 1980.

4. L. P. Leach, Experimental Operating Specification (EOS) for Small Break Tests S-SB-P3, S-SB-P4, and S-SB-P6, L. P. Leach letter to R. E. Tiller (LPL-46-80), EG&G Idaho, Inc., April 3, 1980.

5. G. W. Johnsen, Test Prediction for Semiscale Mod-3 Small Break Test S-SB-P2, G. W. Johnsen letter to R. E. Tiller (GWJ-10-80), EG&G Idaho, Inc., February 13, 1980.

6. C. P. Fineman, Test Prediction for Semiscale Small Break Tests S-SB-P3 and S-SB-P4, EG&G Idaho, Inc., EGG-SEMI-5139 (April 1980).

7. S. E. Dingman, T. J. Fauble, J. R. Hewitt, Quick Look Report for Semiscale Mod-3 Small Break Tests S-SB-P1, S-SB-P2, and S-SB-P7, EG&G Idaho, Inc., EGG-SEMI-5137 (April 1980).

8. J. M. Cozzuol, Quick Look Report for Semiscale Mod-3 Small Break Tests S-SB-P3 and S-SB-P4, EG&G Idaho, Inc., EGG-SEMI-5158 (May 1980).

RESULTS OF SEMISCALE PUMPS ON/OFF EXPERIMENTS

by
GARY JOHNSEN

EG&G Idaho, Inc.

TEST OBJECTIVES

ASSIST IN THE RESOLUTION OF

NUREG-0623 ISSUES:

- DETERMINE THE DIFFERENTIAL RESPONSE CAUSED BY CONTINUOUS PUMP OPERATION VERSUS EARLY PUMP TRIP DURING A SMALL BREAK

- PROVIDE RELEVANT INTEGRAL SYSTEM DATA TO ENABLE ASSESSMENT OF COMPUTER CODES

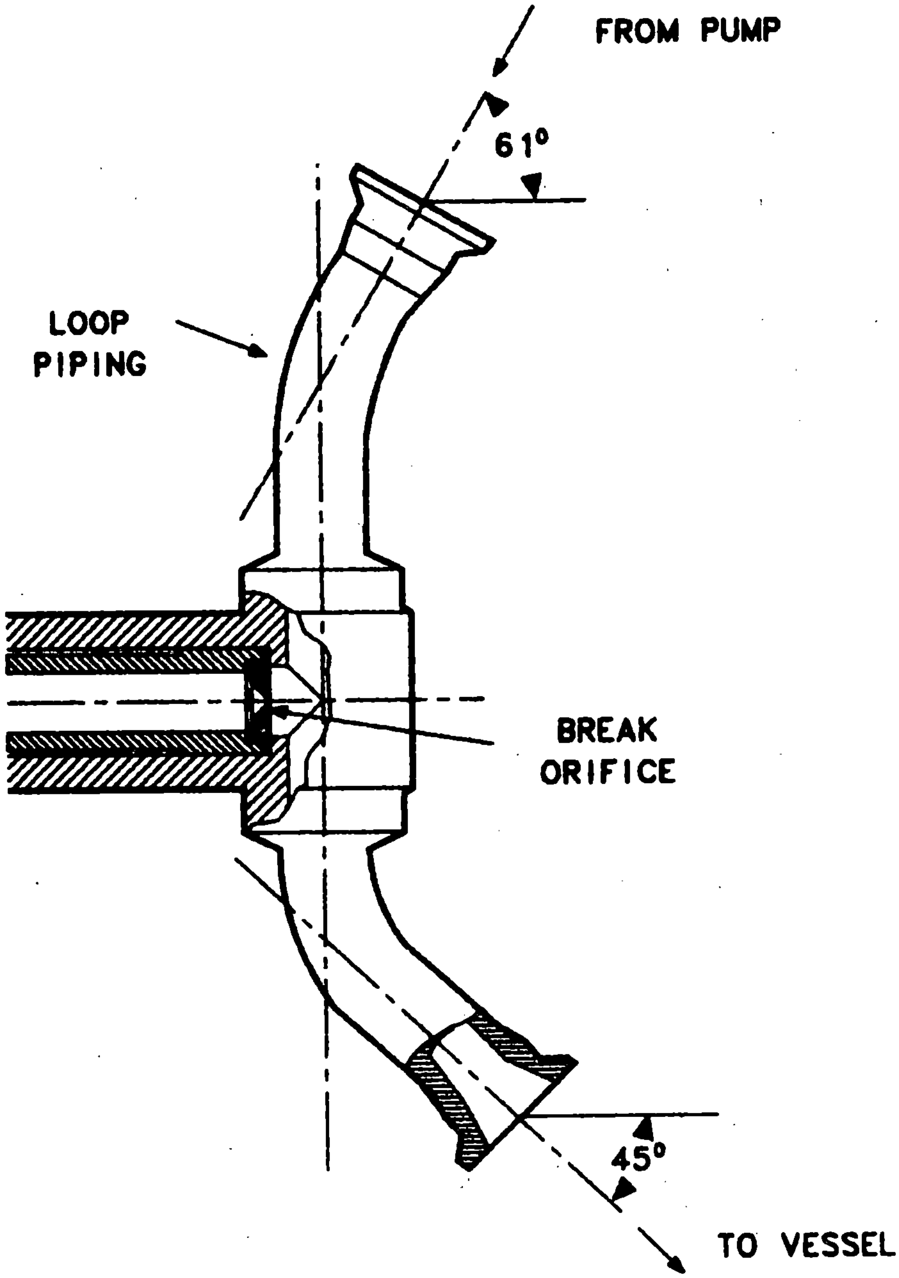
GHJ-2

TEST MATRIX

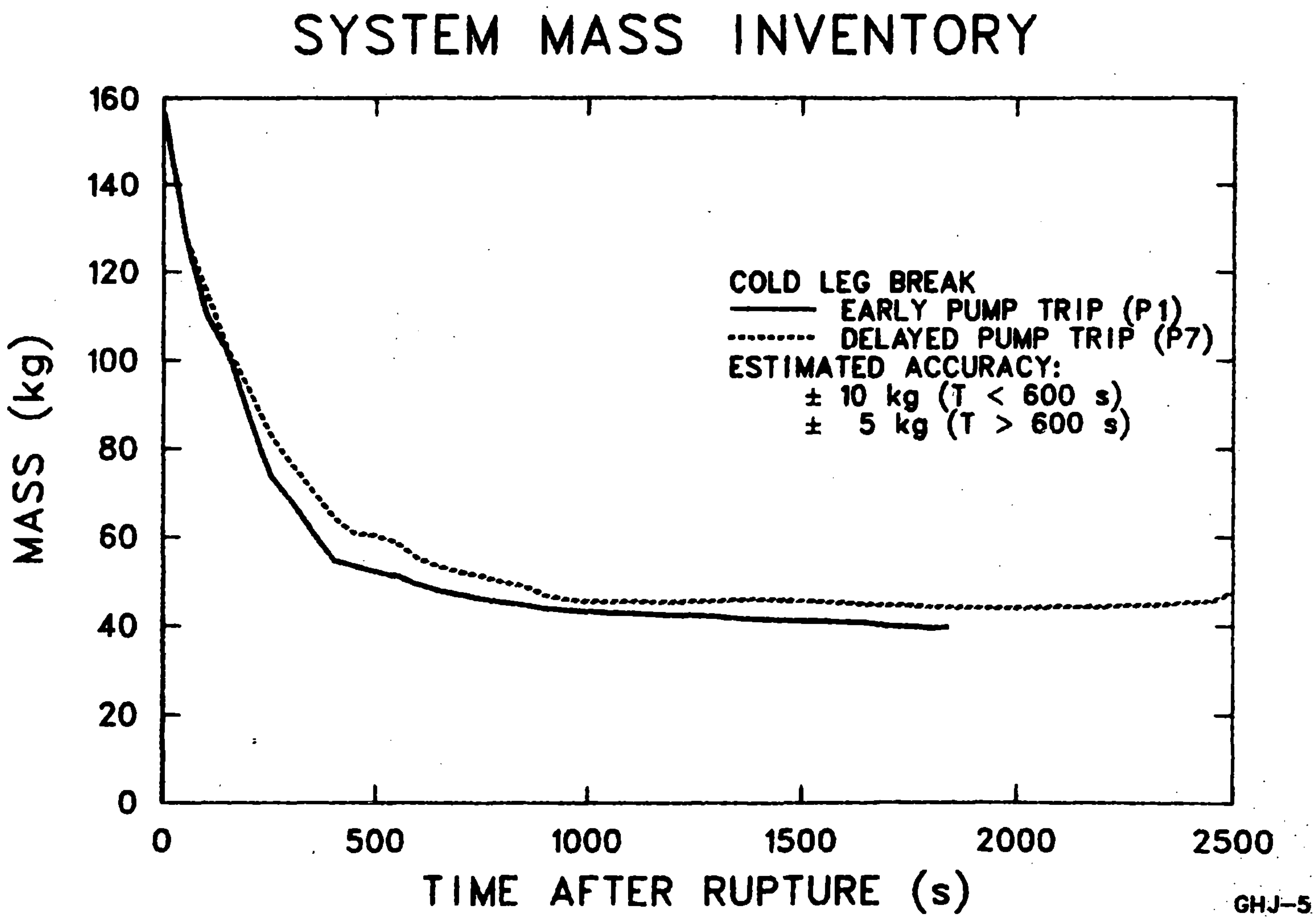
TEST	BREAK/LOCATION	PUMP OPERATION
S-SB-P1	2.5% COLD LEG	TRIP AT SCRAM
S-SB-P2	2.5% COLD LEG	CONTINUOUS
S-SB-P7	2.5% COLD LEG	TRIP AT 3.3 MPa
S-SB-P3	2.5% HOT LEG	TRIP AT SCRAM
S-SB-P4	2.5% HOT LEG	CONTINUOUS
S-SB-P6	2.5% HOT LEG	TRIP AT 3.3 MPa

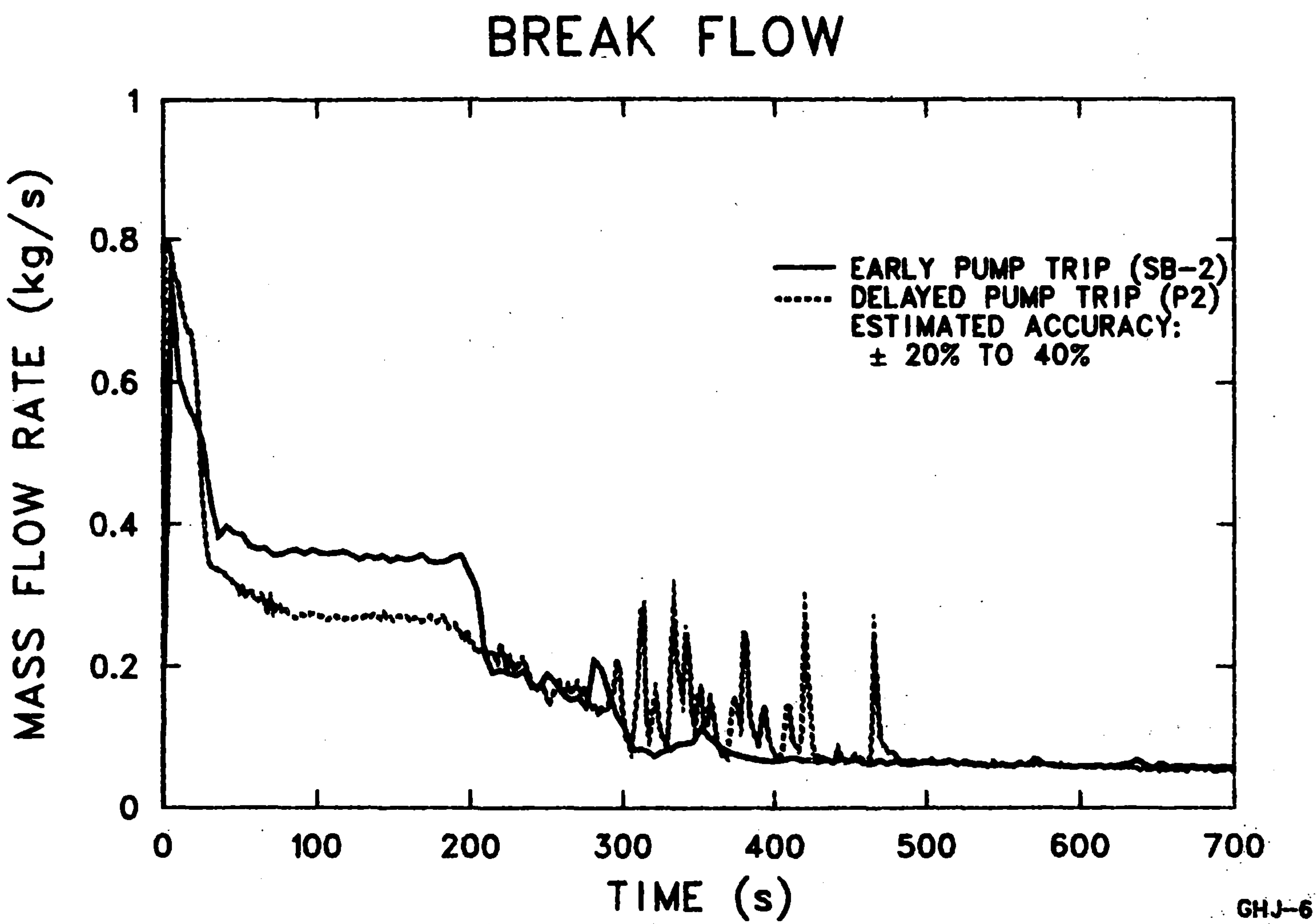
GHJ-3

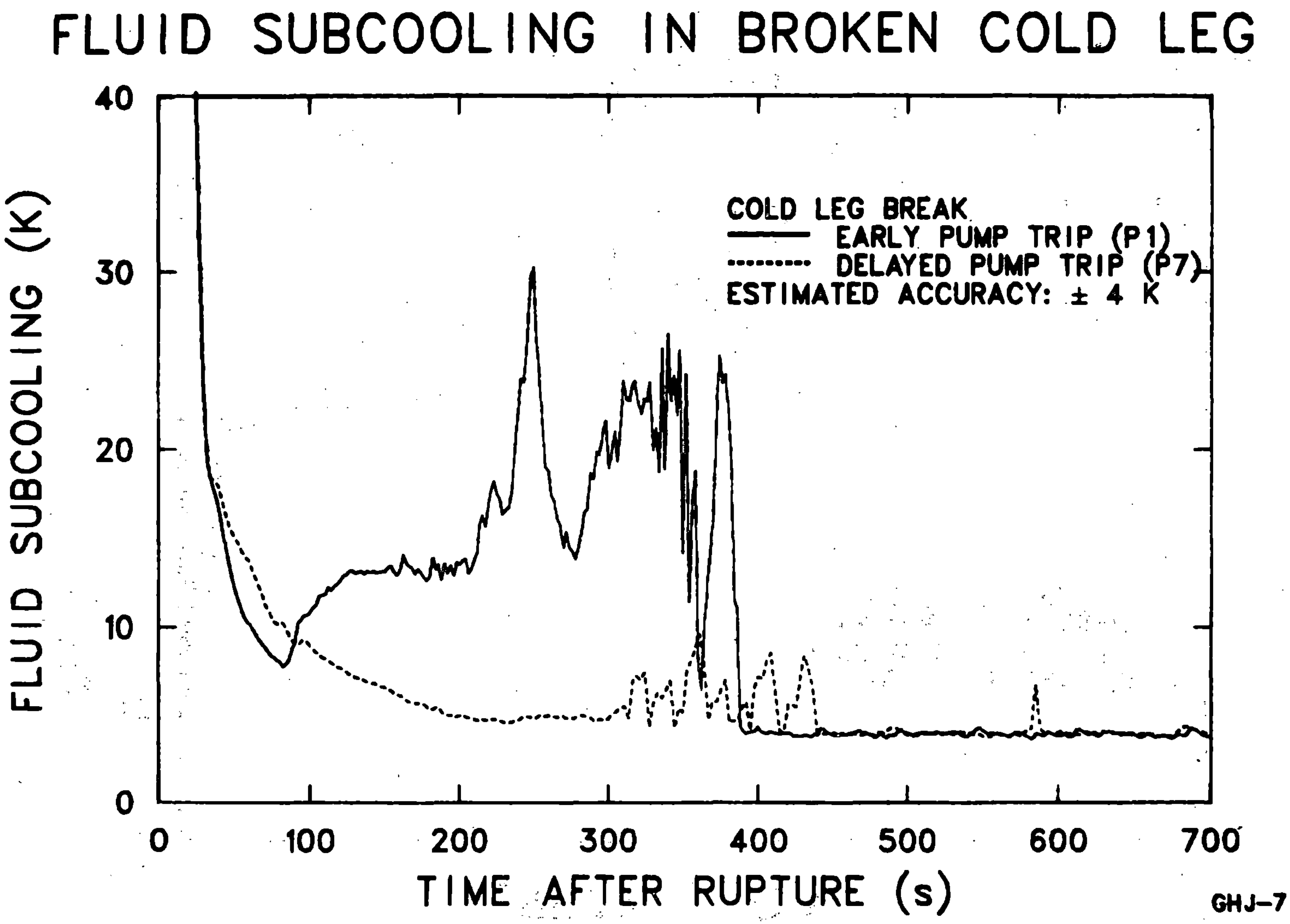
COLD LEG BREAK
CONFIGURATION - PLAN VIEW

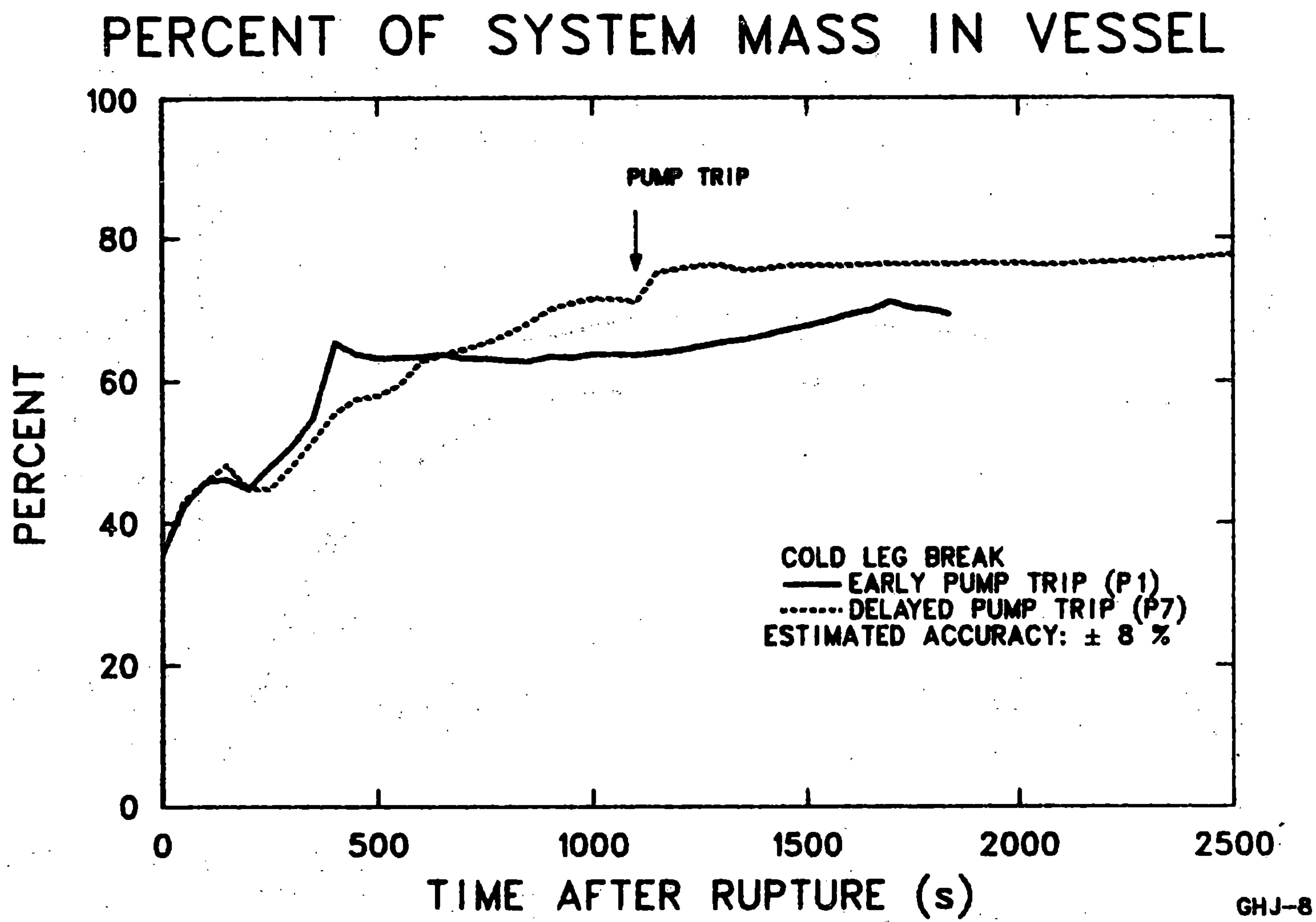


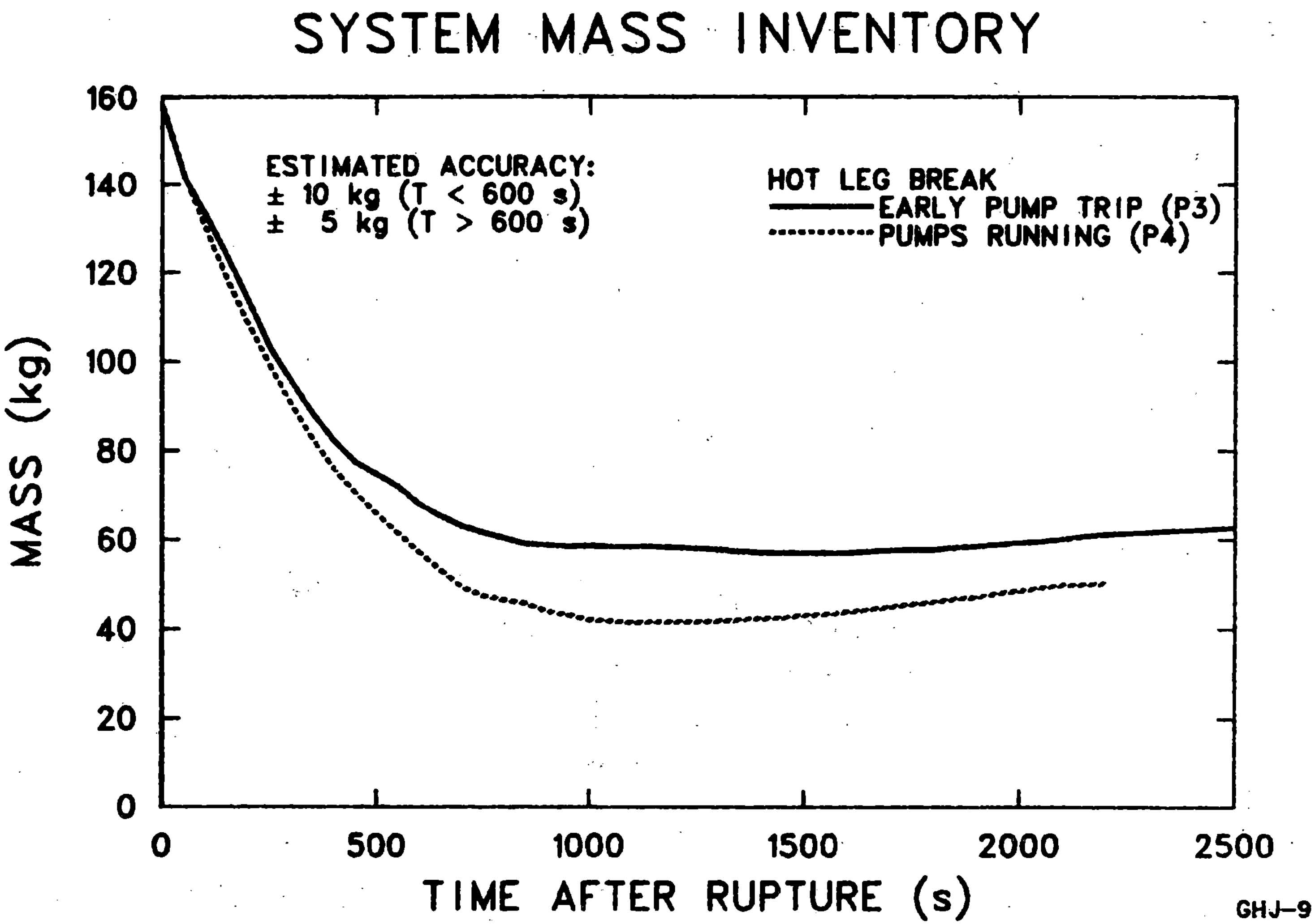
GHJ-4


SYSTEM MASS INVENTORY
MASS (kg)
160
140
120
100
80
60
40
20
0
0
500
1000
1500
2000
2500
TIME AFTER RUPTURE (s)
COLD LEG BREAK
EARLY PUMP TRIP (P1)
DELAYED PUMP TRIP (P7)
ESTIMATED ACCURACY:
± 10 kg (T < 600 s)
± 5 kg (T > 600 s)
GHJ-5

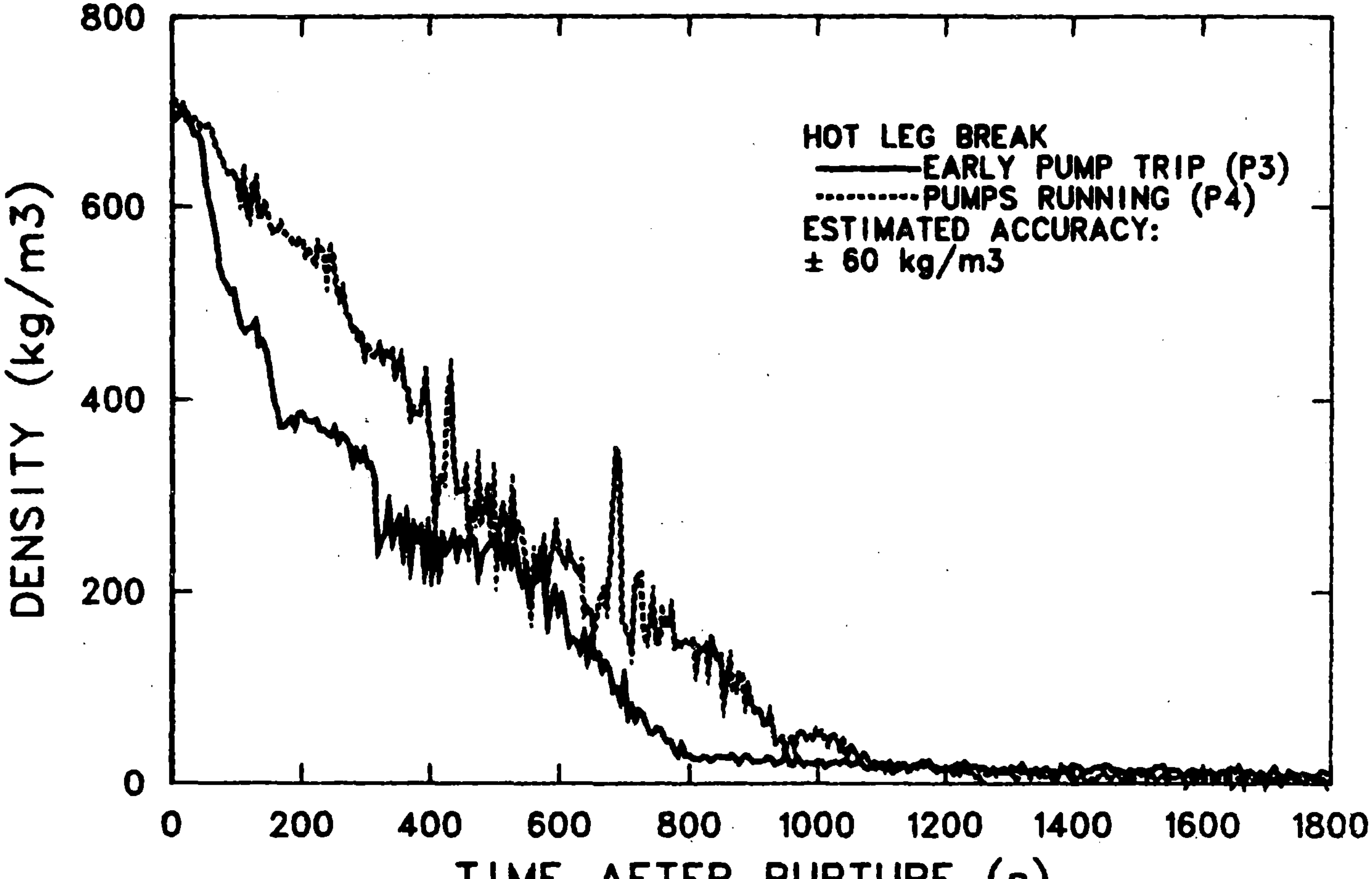

BREAK FLOW
MASS FLOW RATE (kg/s)
1
0.8
0.6
0.4
0.2
0
0
100
200
300
400
500
600
700
TIME (s)
EARLY PUMP TRIP (SB-2)
DELAYED PUMP TRIP (P2)
ESTIMATED ACCURACY:
± 20% TO 40%
GHJ-6


FLUID SUBCOOLING IN BROKEN COLD LEG
COLD LEG BREAK
EARLY PUMP TRIP (P1)
DELAYED PUMP TRIP (P7)
ESTIMATED ACCURACY: ± 4 K
FLUID SUBCOOLING (K)
0
10
20
30
40
0
100
200
300
400
500
600
700
TIME AFTER RUPTURE (s)
GHJ-7

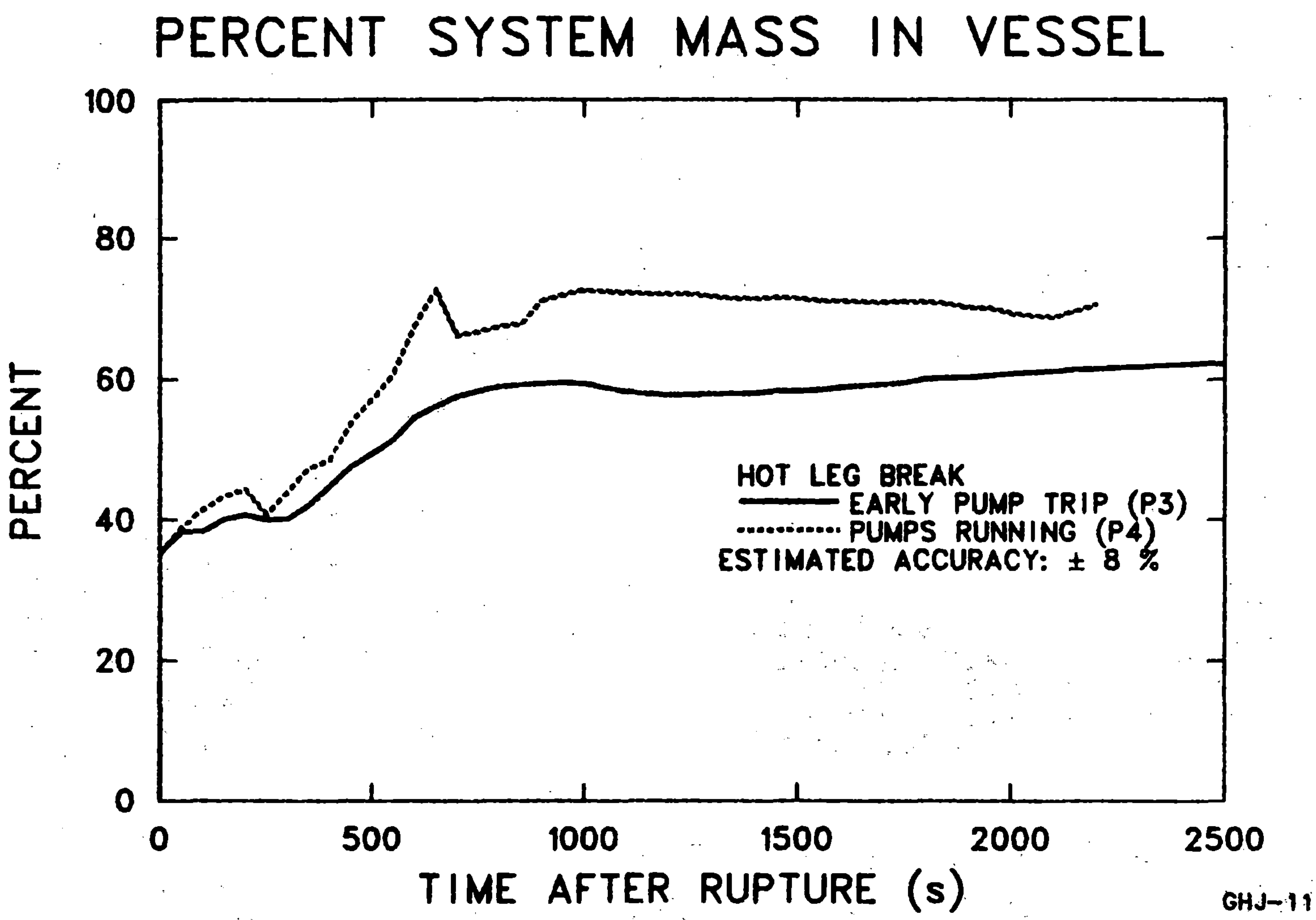

PERCENT OF SYSTEM MASS IN VESSEL
100
80
60
40
20
0
PERCENT
0
500
1000
1500
2000
2500
TIME AFTER RUPTURE (s)
PUMP TRIP
COLD LEG BREAK
EARLY PUMP TRIP (P1)
DELAYED PUMP TRIP (P7)
ESTIMATED ACCURACY: ± 8 %
GHJ-8

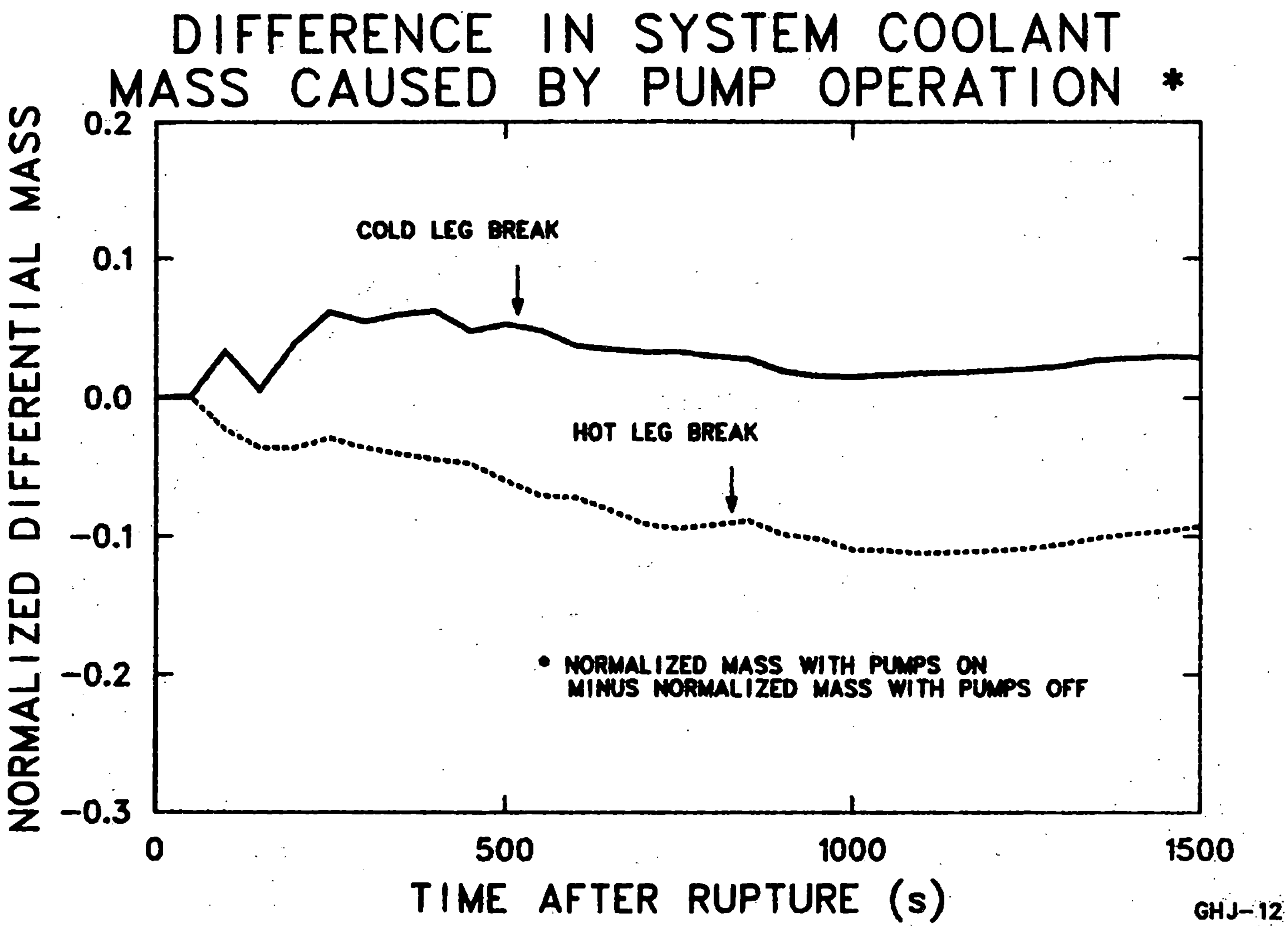

SYSTEM MASS INVENTORY
ESTIMATED ACCURACY:
± 10 kg (T < 600 s)
± 5 kg (T > 600 s)
HOT LEG BREAK
EARLY PUMP TRIP (P3)
PUMPS RUNNING (P4)
MASS (kg)
160
140
120
100
80
60
40
20
0
0
500
1000
1500
2000
2500
TIME AFTER RUPTURE (s)
GHJ-9

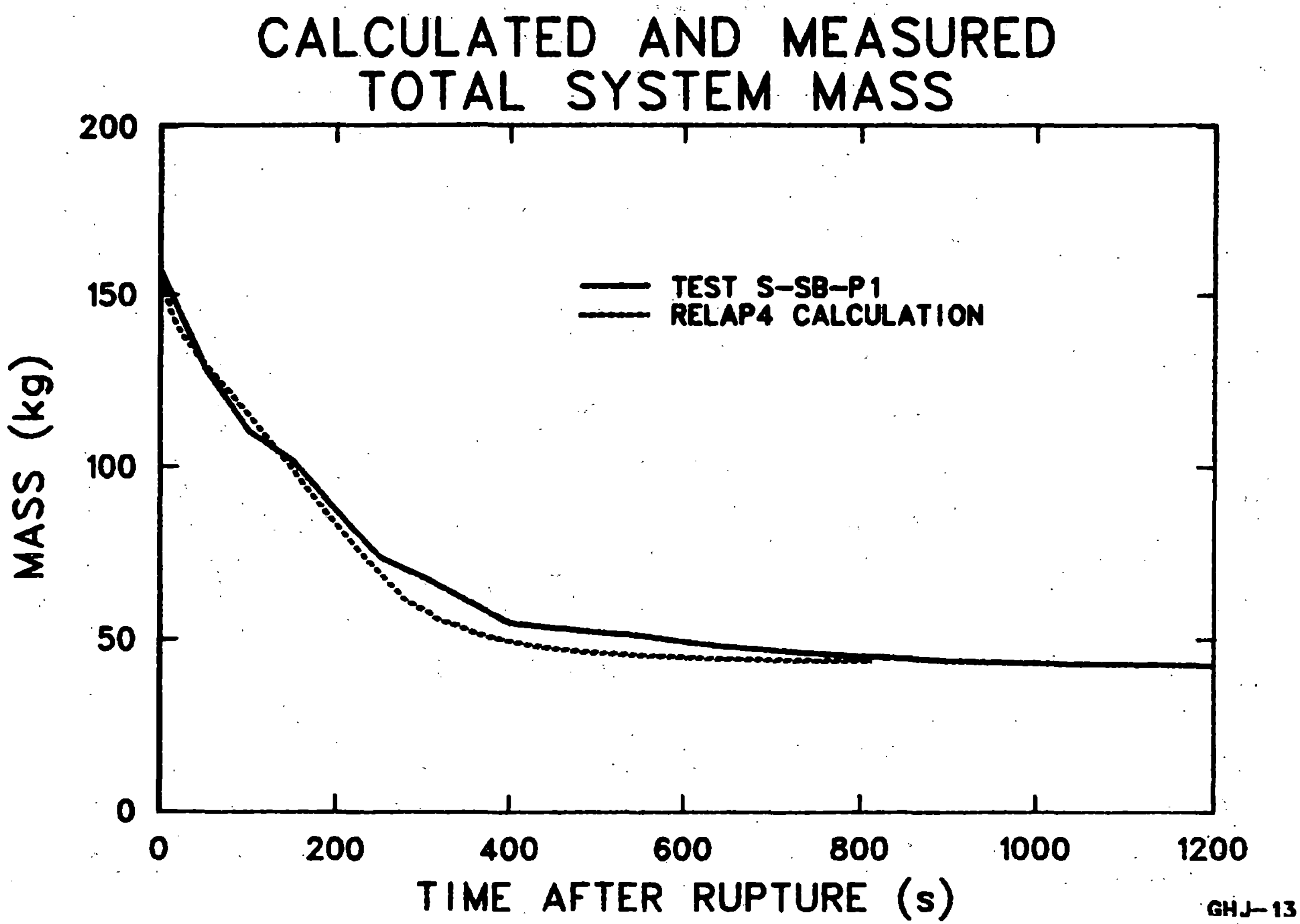
BROKEN LOOP HOT LEG DENSITY



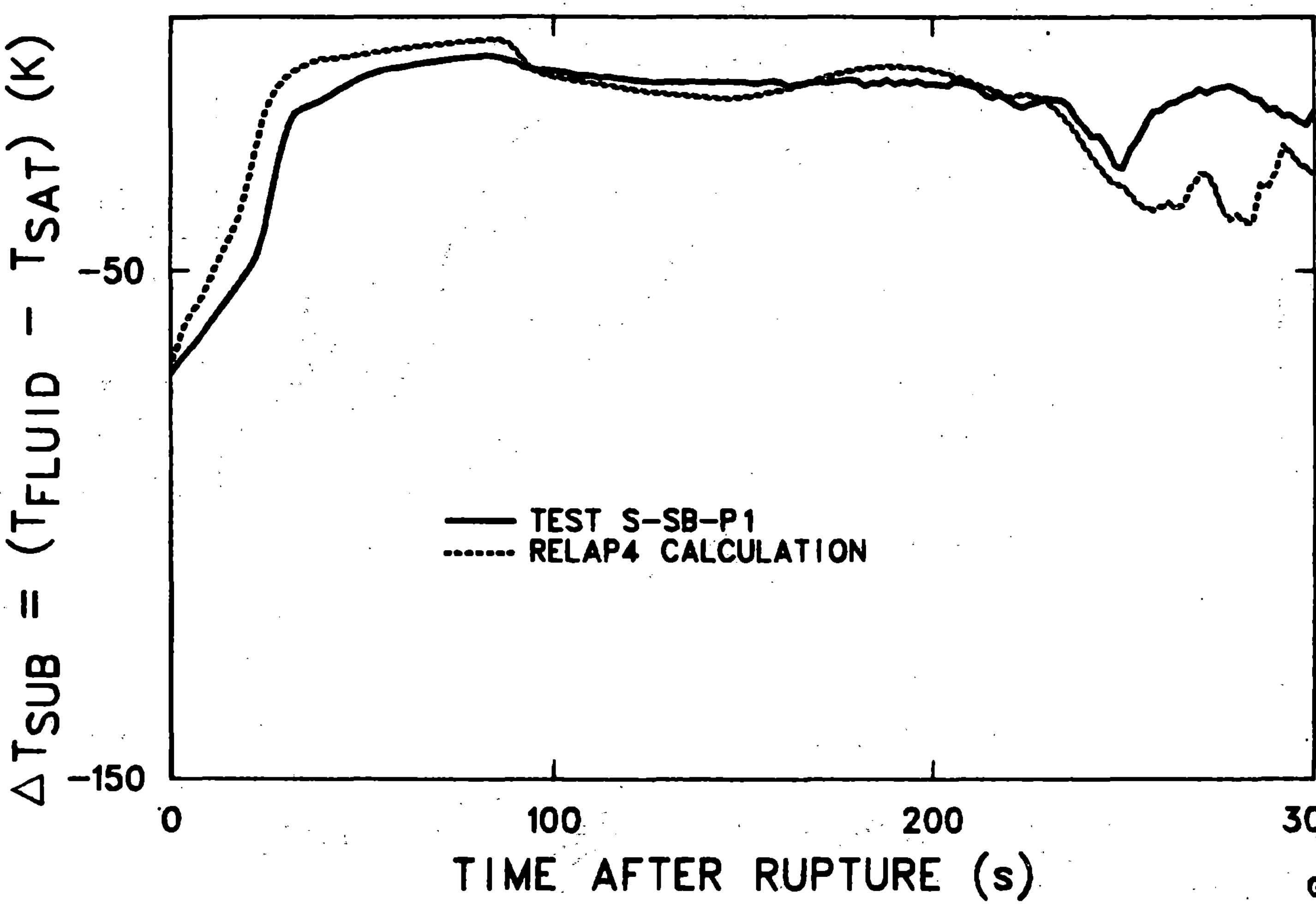
GHJ-10


PERCENT SYSTEM MASS IN VESSEL
100
80
60
40
20
0
PERCENT
0
500
1000
1500
2000
2500
TIME AFTER RUPTURE (s)
HOT LEG BREAK
EARLY PUMP TRIP (P3)
PUMPS RUNNING (P4)
ESTIMATED ACCURACY: ± 8 %
GHJ-11

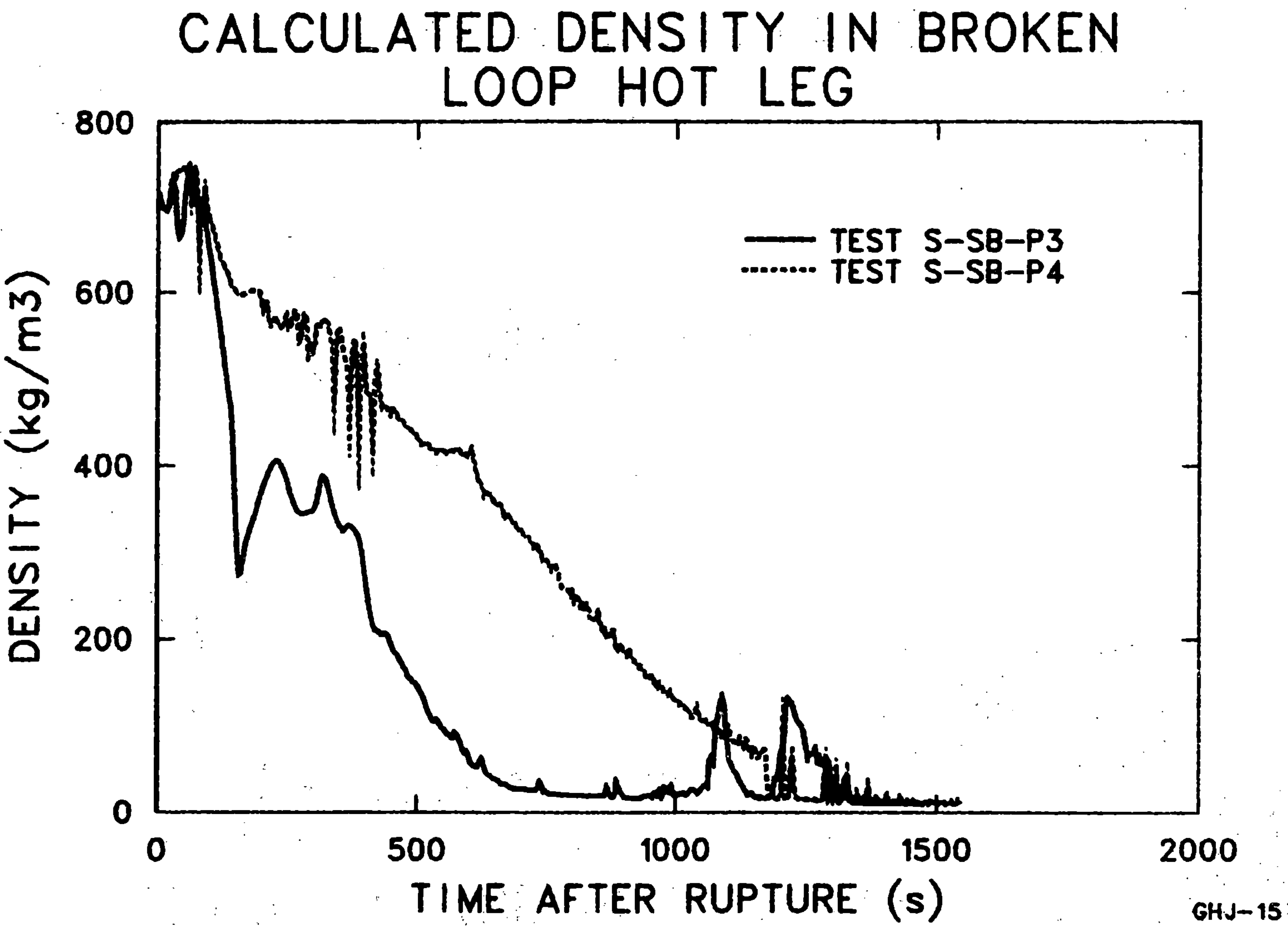

DIFFERENCE IN SYSTEM COOLANT
MASS CAUSED BY PUMP OPERATION *
NORMALIZED DIFFERENTIAL MASS
0.2
0.1
0.0
−0.1
−0.2
−0.3
COLD LEG BREAK
HOT LEG BREAK
* NORMALIZED MASS WITH PUMPS ON
MINUS NORMALIZED MASS WITH PUMPS OFF
0
500
1000
1500
TIME AFTER RUPTURE (s)
GHJ−12


CALCULATED AND MEASURED
TOTAL SYSTEM MASS
TEST S-SB-P1
RELAP4 CALCULATION
MASS (kg)
200
150
100
50
0
0
200
400
600
800
1000
1200
TIME AFTER RUPTURE (s)
GHJ-13

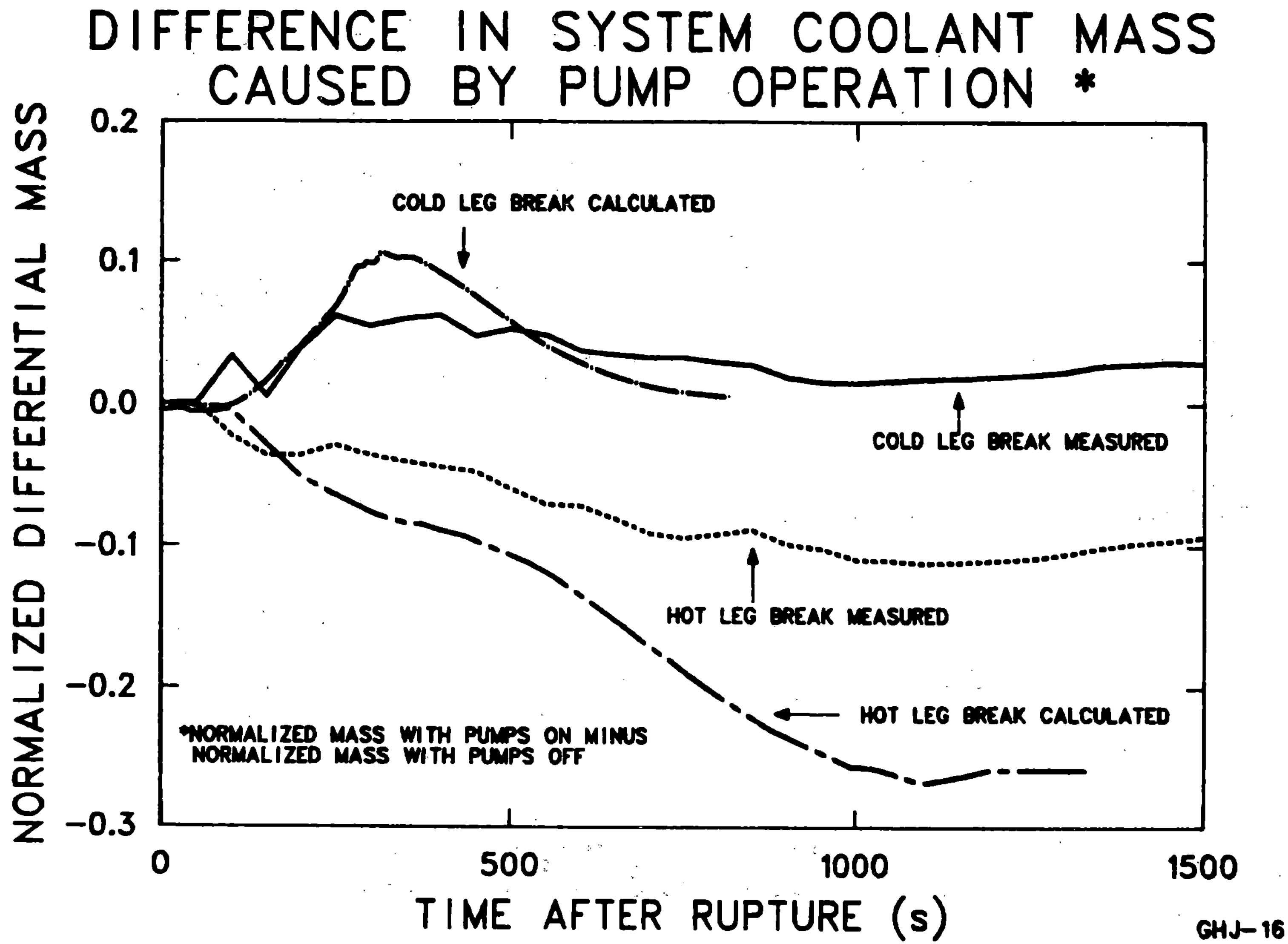
SUBCOOLING IN THE BROKEN LOOP COLD LEG



GHJ-14



CALCULATED DENSITY IN BROKEN
LOOP HOT LEG
DENSITY (kg/m3)
TIME AFTER RUPTURE (s)
TEST S-SB-P3
TEST S-SB-P4
800
600
400
200
0
0
500
1000
1500
2000
GHJ-15



DIFFERENCE IN SYSTEM COOLANT MASS
CAUSED BY PUMP OPERATION *
NORMALIZED DIFFERENTIAL MASS
0.2
0.1
0.0
−0.1
−0.2
−0.3
COLD LEG BREAK CALCULATED
COLD LEG BREAK MEASURED
HOT LEG BREAK MEASURED
HOT LEG BREAK CALCULATED
*NORMALIZED MASS WITH PUMPS ON MINUS
NORMALIZED MASS WITH PUMPS OFF
0
500
1000
1500
TIME AFTER RUPTURE (s)
GHJ-16

CONCLUSIONS

- CONTINUED PUMP OPERATION INFLUENCES BREAK DISCHARGE
  - LESS MASS DEPLETION FOR COLD LEG BREAK
  - GREATER MASS DEPLETION FOR HOT LEG BREAK
- PUMP OPERATION CAUSES COOLANT REDISTRIBUTION FROM COLD TO HOT PORTIONS OF SYSTEM

GHJ-17

CONCLUSIONS (CONT'D)

- HOT LEG BREAK LESS SEVERE THAN COLD LEG BREAK
- RELAP4 CODE CORRECTLY PREDICTS DIFFERENTIAL TRENDS CAUSED BY PUMP OPERATION
- OVERALL RESULTS SUGGEST SENSITIVITY TO ASSUMED BREAK CONFIGURATION AND SCENARIO

GHJ-18

PNL-SA-9017

Frapcon-2 Structure and Results

D. D. Lanning
W. N. Rausch

October 1980

Presented at the Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

Battelle

PNL-SA-9017

PNL-SA-9017

FRAPCON-2 STRUCTURE AND RESULTS

D. D. Lanning
W. N. Rausch

October 1980

Presented at the Eighth Water Reactor Safety Research Information Meeting
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FRAPCON-2 STRUCTURE AND RESULTS

I. INTRODUCTION

The NRC-sponsored steady-state fuel rod modeling work at INEL and PNL have been integrated since the production of FRAPCON-1[1] in 1979. FRAPCON-1 is a steady-state single-rod code which proved during "independent assessment" comparison to in-reactor data[2] to have the same shortcomings as its predecessors (FRAPS-3, GAPCON-3). Among these were overprediction of measured fuel temperatures and underprediction of fuel rod axial elongation.

The Fuel Behavior Research Branch authorized and sponsored a second version, FRAPCON-2, to be developed jointly by PNL and INEL. The objectives were at least two fold:

1. To establish an improved version, which might better account for available fuel performance data;
2. To provide a "Forum" - a common calculational framework - within which several major models could be evaluated.

The original FRAPCON-1 included the following major features:

FRACAS-I (rigid fuel stack) mechanical model
MacDonald-Weisman gas release model
MacDonald-Broughton or GAPCON-1 gap conductance model
Coleman relocation/fuel thermal conductivity degradation
(used only with GAPCON-1 gap conductance)

FRAPCON-2 includes a much larger variety of both mechanical and thermal models. The INEL mechanical options now include both FRACAS-I and FRACAS-II, the latter being a deformable pellet model in which fuel stress relaxation is incorporated via hot-pressing. The fuel is not infinitely rigid, and radial fuel cracks are allowed to form and to heal. The Coleman fuel relocation function is still utilized. A fully interconnected finite element (1/2 pellet) model called AXISYM can also be accessed to assess localized deformation.

PNL has contributed another option to the mechanical models. The cracked fuel model RADIAL, based on the assumption of hydrostatic stress in the fuel, provides estimates of gap size, fuel conductivity degradation, and fuel elastic moduli for each axial region. These are fed into the chained axial-radial finite element models of the PELET package from GAPCON-3,[3] which provides axial stress/strain and incremental creep stress/strain.

There have also been changes and additions to the thermal models. A single common model for gap conductance is used (the GAPCON-2 model),[4] but there are now five gas release options (described later). Also, whereas the Coleman relocation and fuel effective conductivity functions are retained with FRACAS-1 or FRACAS-II, an entirely different rationale for these parameters is attendant to the PELET/RADIAL mechanical option. This also is explained in the next section.

Status of the code development and documentation is as follows: developmental assessment is complete and the code is "frozen." Collections of corrections will take the form of successive revision numbers. The code document and user's manual is in publication; we anticipate the code to be available from the Argonne code* center by January, 1981.

*I.e., the National Energy Software Center at Argonne National Laboratory, Argonne, Illinois.

II. CODE DESCRIPTION

FRAPCON-2 is composed of a driver routine and a series of sub-codes, some of which are optional. Table 1 lists these subcodes, and Figure 1 shows their interrelationship. The driver controls multiple-case runs (with statistically varied input) to accommodate the automated uncertainty analysis. The main subcode is FRPCON, which connects all the other subcodes. FRPCON is essentially FRAPCON-1 in structure; as such it contains the FRACAS-1 mechanical package, several gas release options, the temperature calculator, failure prediction and the output routines. Depending on user input it may use the FRACAS-II or PELET/RADIAL mechanical packages in place of FRACAS-I. There is a passive link to the AXISYM localized finite element model for assessment of ridge formation. Active links exist to the ANS 5.4 and to the GRASS/FASTGRASS gas release options, and of course to the MATPRO subcode for material properties. Only the subcodes FRPCON and MATPRO are essential to execution.

For completeness, we list in Table 2 all the major models used in the code. The mechanics and gas release options will be described below in greater detail.

FLOWCHART AND SOLUTION SCHEME

Figure 2 shows a simplified flow chart for FRAPCON-2 which is superficially quite standard. Within the time step loop is an iteration of gas pressure/release; during each of those iterations fuel temperatures and incremental gas release are recalculated. Internal to the gas pressure loop is an iteration loop on fuel temperature and gap conductance--carried out at each axial node.

Figure 3 shows major models superimposed on the basic structure. We should emphasize that, whereas the INEL models (FRACAS) follow the conventional route of recalculating gap size <u>within</u> the gap conductance iteration loop, the PNL model (PELET/RADIAL) assesses gap size and fuel conductivity <u>ahead</u> of the temperature loop (but within the gas release loop). Both mechanical options assess final elastic deformation and creep after gas release convergence.

TABLE 1. Sub-code Description

Section	Description
FRPCON	The main section of the code, including all of the thermal models. Also includes the uncertainty analysis routines and the INEL failure package, and the FRACAS-I mechanical package.
MATPRO	The INEL developed material properties package.
FRACAS-2	Contains all of the routines that make up INEL's FRACAS-2 mechanical modeling subcode.
PELET	Contains all of the routines that make up PNL's PELET mechanical modeling subcode.
AXISYM	Contains the routines composing INEL's detailed finite element mechanical modeling package.
GRASS	Contains ANL's detailed fission gas release routines.
FAST-GRASS	A faster executing, less detailed, version of GRASS.

TABLE 2.

Generic Model Name or Function	Model Used in FRAPCON-2	Reference Number
Fuel cladding average elastic deformation	FRACAS-I (rigid pellet)	1
	FRACAS-II (deformable pellet)	5
	PELET/RADIAL (cracked pellet)	3
Gap conductance	GAPCON-2	4
Gas release	Booth	1
	MacDonald-Weisman	1
	Beyer-Hann	6
	GRASS/FAST-GRASS	7
	ANS 5.4	8
Fuel conductivity adjustment	Coleman	1
	PNL (RADIAL)	9
Cladding creep	Pankaskie (PNL) rule	3
	MATPRO-11	10
Cladding failure predictions	INEL fuel failure models	1
Fuel Relocation	Coleman Model	1
	Constant contact	9
	Col.2, Table 2	-
	Carlson Relocation (FRACAS-II)	

THE PELET/RADIAL MECHANICAL PACKAGE

The operation of the PELET/RADIAL mechanical package is sketched in Figure 4. The cracked fuel is assumed to be in contact with the cladding at all times. Iteration between gap/crack closure and fuel hydrostatic stress converges to yield an estimate of the gap size and crack volume, and interfacial pressure. This in turn permits an estimate of the fuel thermal conductivity degradation derived from empirical fits to instrumented test fuel data. The gap size, interfacial pressure, and conductivity factor then go to the axial node loop where fuel temperatures and fission gas release are calculated. The fuel temperatures are fed back to RADIAL, and the process is repeated to convergence on gas pressure. Then RADIAL can pass its estimates of fuel elastic modulus on to PELET, which (once per time step) updates the stress state in the cladding and calculates the current creep correction.

GAS RELEASE OPTIONS

The gas release model options are listed in Table 3, together with their analytical or empirical basis. Of these, perhaps the most empirically based is the Beyer-Hann model, which considers fixed release rates from four different temperature zones within the fuel. These temperature zones correspond roughly to regions of fuel melting, columnar grain growth, equiaxed grain growth, and no restructuring.

The MacDonald-Weisman, Booth, and ANS 5.4 models are all based on various applications of diffusion theory. The MacDonald-Weisman model considers the interacting probabilities of release and entrapment and uses empirically-derived constants to express these probabilities as functions of fuel temperature and density. This is applied separately to 10 equal-volume annuli of fuel at each axial region, as is the Booth model.

The ANS 5.4 model considers species, time, temperature, and burnup dependence of the diffusion coefficients, and has separate high- and low-temperature release expressions for both stable and radioactive species. The model always calculates both high- and low-temperature release and uses the higher of the two values. This model also applies to rings of fuel within each axial region.

TABLE 3.

Gas Release Model	Model Basis
Beyer-Hann	Empirical (4 temperature zones)
Booth	Diffusion-Empirical constants
MacDonald-Weisman	Release/trapping probability-Empirical constants
ANS 5.4	Detailed diffusion; empirical constants
GRASS/FAST-GRASS	Mechanistic approach

The GRASS subcode is a version of the GRASS code developed at ANL by J. Rest. FAST-GRASS is a faster-running, simplified version of the same. The GRASS code is truly mechanistic: it considers the basic mechanisms of bubble formation, migration, coalescence, channeling, and eventual release, applied across small subregions of the fuel and laboriously summed for the entire rod. More detailed discussion is beyond the scope of this presentation, and the reader is directed to the references. Note that because of the computing time involved in the GRASS subcode it is called only once per time step.

III. CODE RESULTS

Some examples of thermal and mechanical predictions of the code will be shown here versus in-reactor data.

SMALL-GAP HELIUM-FILLED ROD (IFA-431, Rod 3)[11]

A good thermal "target" for any code is the fuel temperature from a small-gap helium-filled rod at beginning-of-life. The uncertainties in heat transfer are minimized in such a situation. Rod 3 from the NRC/PNL Halden Assembly IFA-431 had a nominal 50-μm (2-mil) diametral gap, helium fill 10% enriched (95% theoretical density) pellets, with an outer diameter of 10.89 mm. The cladding was Zircaloy-2, 10.90 x 12.79 mm ID x OD. Beginning-of-life data (with 95% confidence limits) for centerline temperature versus power is shown in Figure 5, together with prediction using both the FRACAS-II and PELET subcodes. The divergence of the predictions from the data and from each other is slight.

Figure 6 shows the measured axial elongation of Rod 3 on the first rise to power, along with the predictions of both subcodes. FRACAS-II does not predict the onset of enhanced elongation as well as PELET in this case.

DIAMETRAL ROD STRAIN (IFA-508)[13,14]

Figures 7, 8, and 9 show, respectively, fuel center temperature, axial elongation, and diametral strain (at ridges) from the beginning-of-life for JAERI/Halden assembly IFA-508 rod 11. This rod contained 10.5% enriched UO_2 pellets (95% theoretical density), dished, with a pellet diameter of 11.31 mm. The rod had a 100 μm (4 mil) diametral gap, and was helium filled. The zircaloy cladding was quite thin (12.19 x 11.41 mm ID x OD). Figures 7, 8, and 9 show predictions of both major subcodes.

Figure 7 shows close agreement between both subcodes and the fuel temperature data. This is consistent with the previous rod. Similarly, Figure 8 shows PELET in better overall agreement with the axial elongation data than FRACAS-II.

Figure 9, however shows a different trend with respect to diametral strain. PELET is overpredicting the diametral strain, and FRACAS-II does also, but not as badly.

HIGH-TEMPERATURE FISSION GAS RELEASE (Studsvik Rod S150)[14]

Figure 10 shows both measured and predicted temperature and measured/predicted fission gas release from a high-power, high-temperature irradiation of a BWR-type rodlet in the Studsvik Reactor, Sweden. The peak powers ranged from 55 to 69 kW/m (with a peak/average ratio of ~1.2), and the average burnup attained was 4,560 MWd/MTU (79 days of irradiation). The predictions shown in Figure 10 are made using two different gas release models (Beyer-Hann and ANS 5.4) and the PELET mechanical option. Although the gas release fraction is overpredicted, the temperatures are underpredicted. In contrast, Table 4 shows the results of combining the ANS 5.4 model with the FRACAS-I mechanical model for this rod. Both the gas release and the temperatures are far overpredicted in that case. This emphasizes that gas release models in particular cannot be evaluated independent of the associated thermal and mechanical models used in simulating in-reactor experiments.

TABLE 4. Comparison of Model Combinations Applied to the Studsvik S150 Rod

Gas Release Model	Mechanical Option	Peak Centerline Temp. at 62.5 kW/m, °C (Lifetime Peak)		Gas Release Fraction, %	
		Measured	Predicted	Measured	Calculated
ANS 5.4	PELET	2320	2190	10	44
ANS 5.4	FRACAS-I	2320	2800	10	100
Beyer-Hann	PELET	2320	2180	10	20
McDonald-Weisman	FRACAS-II	2320	2090	10	25

LONG-TERM FISSION GAS RELEASE (IFA-432, Rod 1)[15]

The NRC/PNL Halden Assembly IFA-432 began operating in 1975, and features six heavily instrumented rods, five of which are still under irradiation presently. Rod 1 of that assembly had 95% dense, flat-ended 10.68 mm dia pellets, with a 230 μm diametral gap. The cladding was Zircaloy-2, 10.90 X 12.79 mm ID

X OD, and the fill was helium at 1 ata. The fuel center temperature and gas pressure rose simultaneously after about 5,000 MWd/MTM average burnup, providing a good cross check on the fission gas release deduced from the pressure data. The inferred cumulative gas release fraction is shown in Figure 11, together with preliminary puncture data from a sister rod. The temperature at a constant 32.0 kW/m is shown on the same figure. In Figure 12 we show the fuel temperature versus burnup predictions of both FRACAS-II and PELET (with Beyer-Hann fission gas release) relative to the data. The fission gas release results are summarized in Table 5. The FRACAS model predicts similar initial temperatures as PELET, but PELET then predicts higher gas release and still higher temperatures as burnup progresses.

TABLE 5. Predicted Gas Release for Rod 1-IFA-432 at 23,000 MWd/MTM

Source	Gas Release, %
Inferred from Pressure Transducer	5-10
Measured from Rod 8 Puncture	9
Predicted Using PELET (with Beyer-Hann Model)	31.0
Predicted using FRACAS-II (with McDonald Weisman Model)	22

IV. CONCLUSIONS

The FRAPCON-2 code is completed and currently undergoing independent assessment. The code document will be issued by December, 1980 and the code will be available from the Argonne code center by January, 1981. The code contains redundancy of major models for evaluation, but is constructed so that core space is only allocated to models actually used for a given run. The solution scheme is standard for steady-state thermal performance codes.

Code results indicate reasonable agreement between the models and in-reactor data for fuel temperature and rod strains. Gas release appears to be generally overpredicted.

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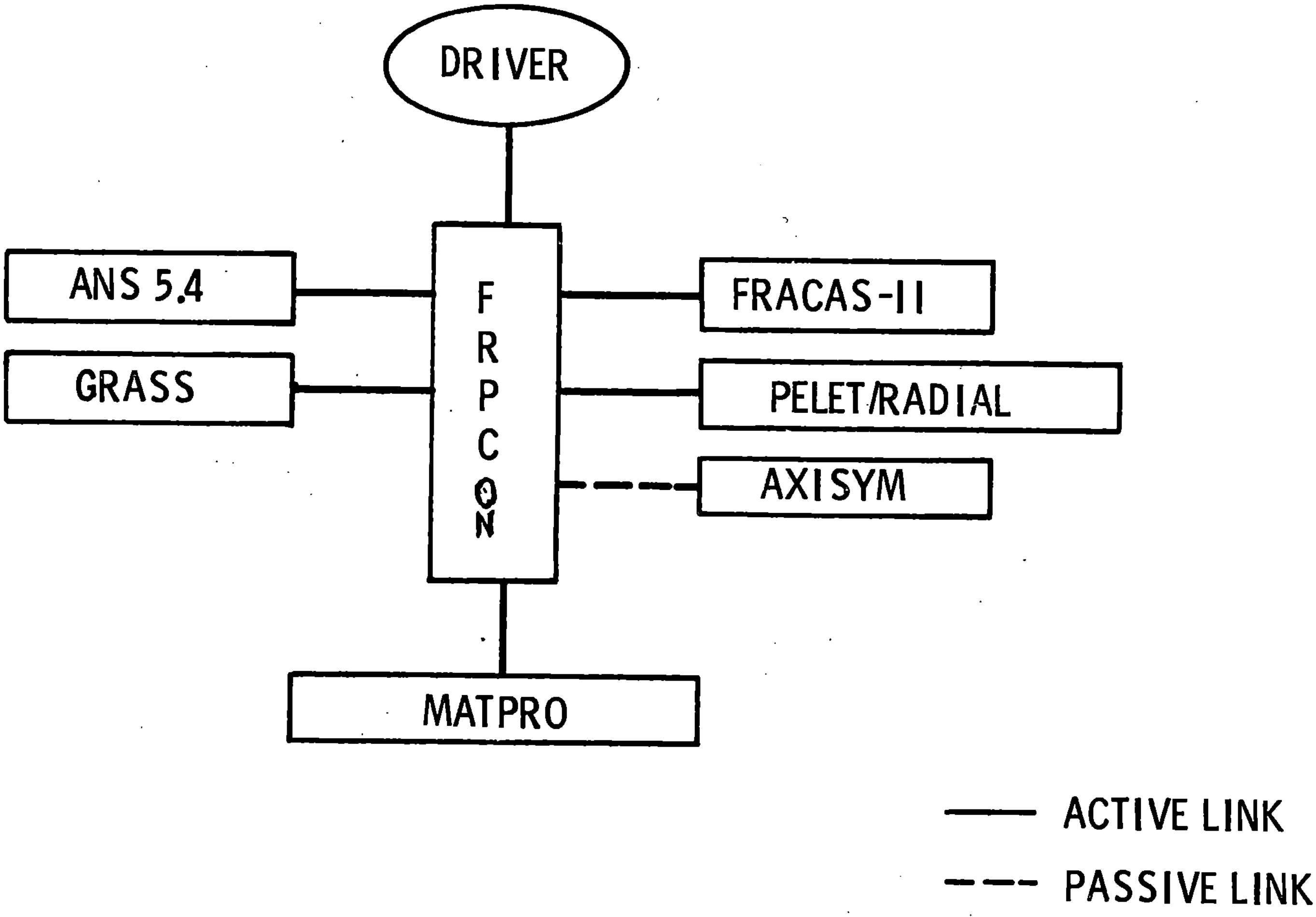

FIGURE 1. Relationship among Primary FRAPCON-2 Subcodes

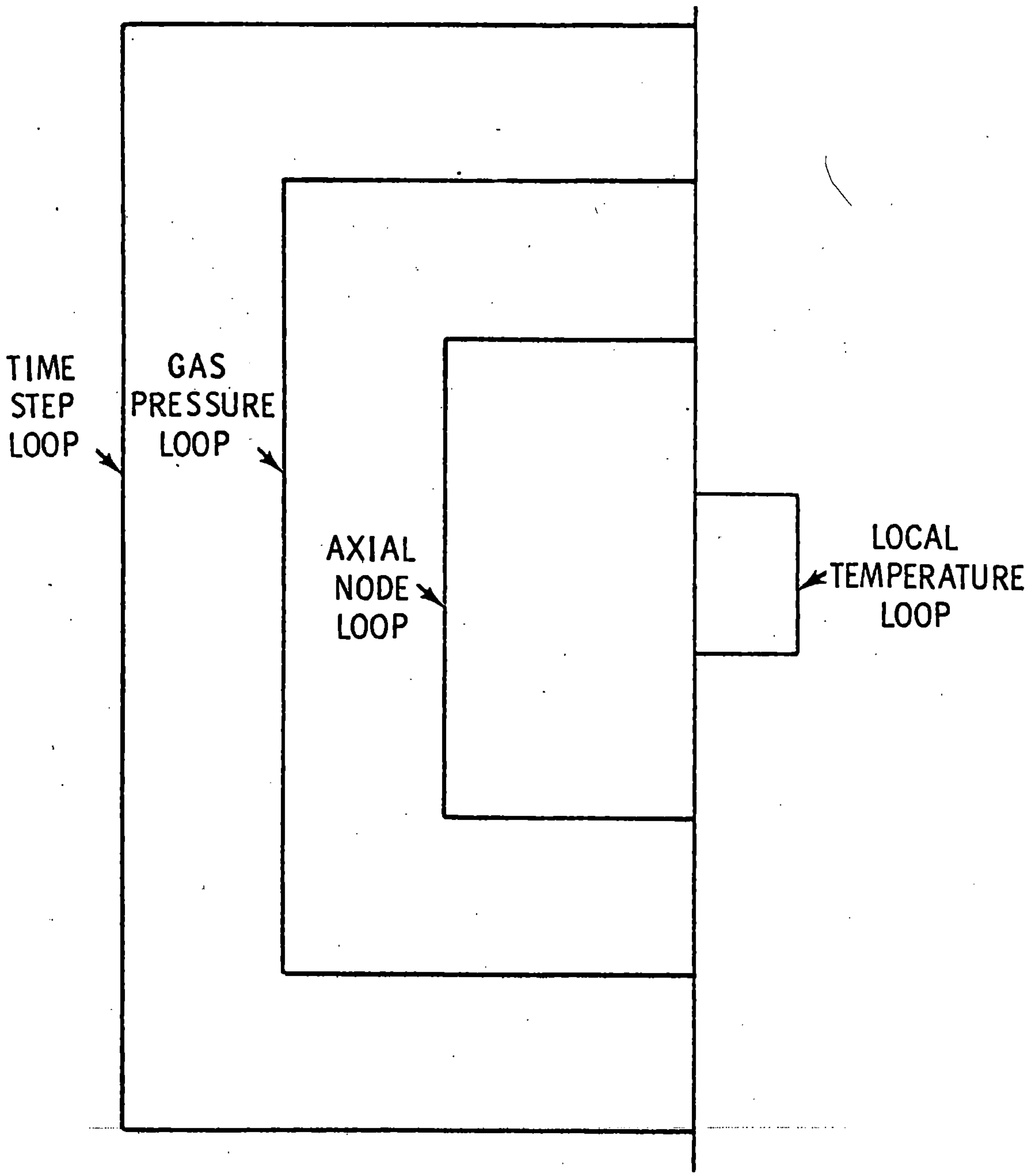

FIGURE 2. FRAPCON-2 Code Structure

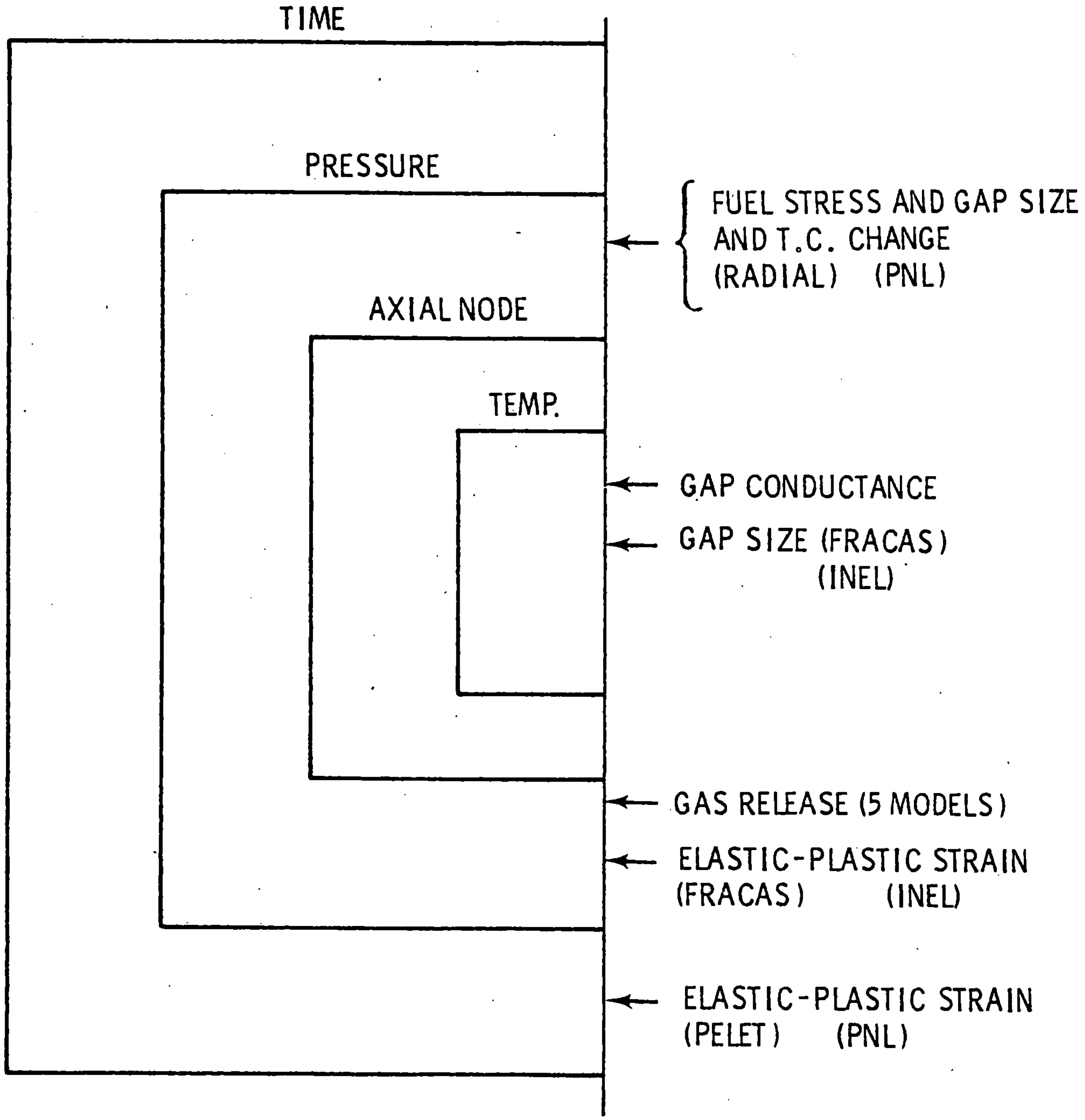

FIGURE 3. Major Models in FRAPCON-2

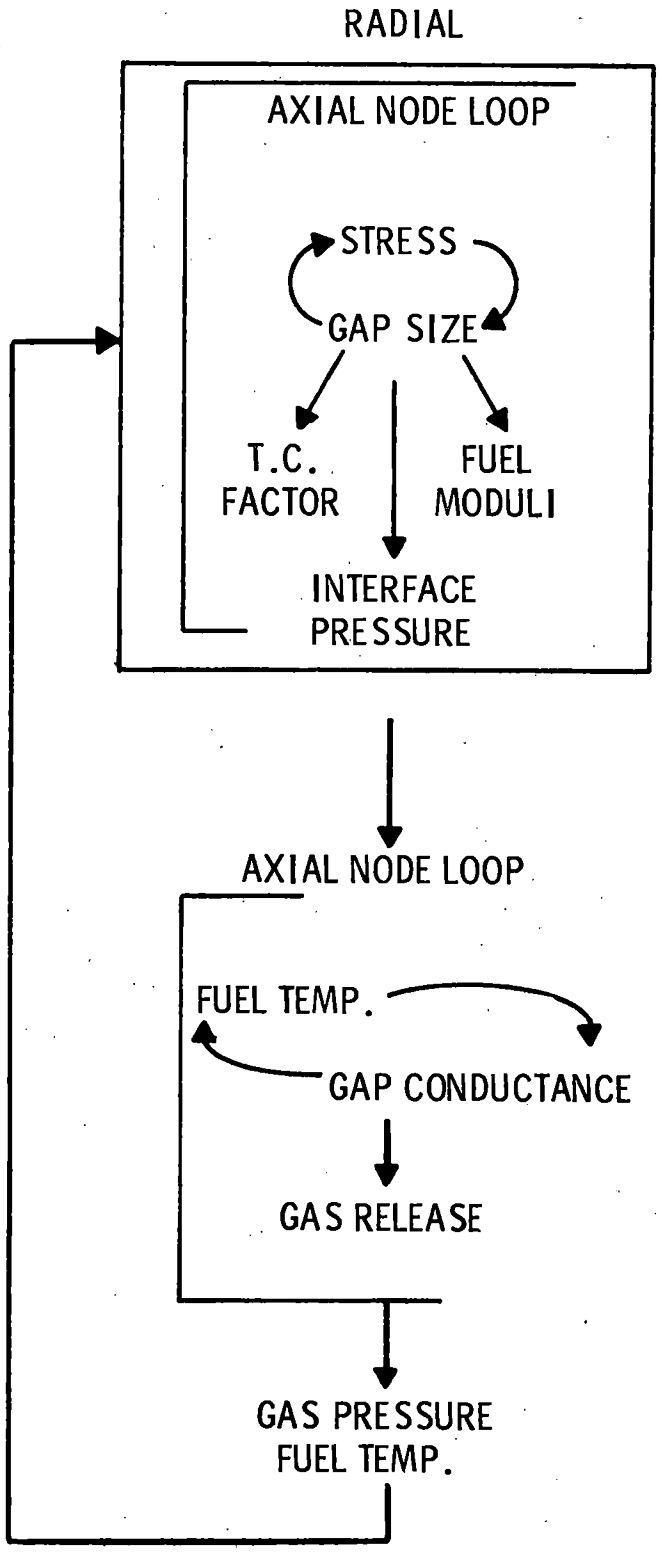

FIGURE 4. Integration of the RADIAL Subroutine Within FRAPCON-2

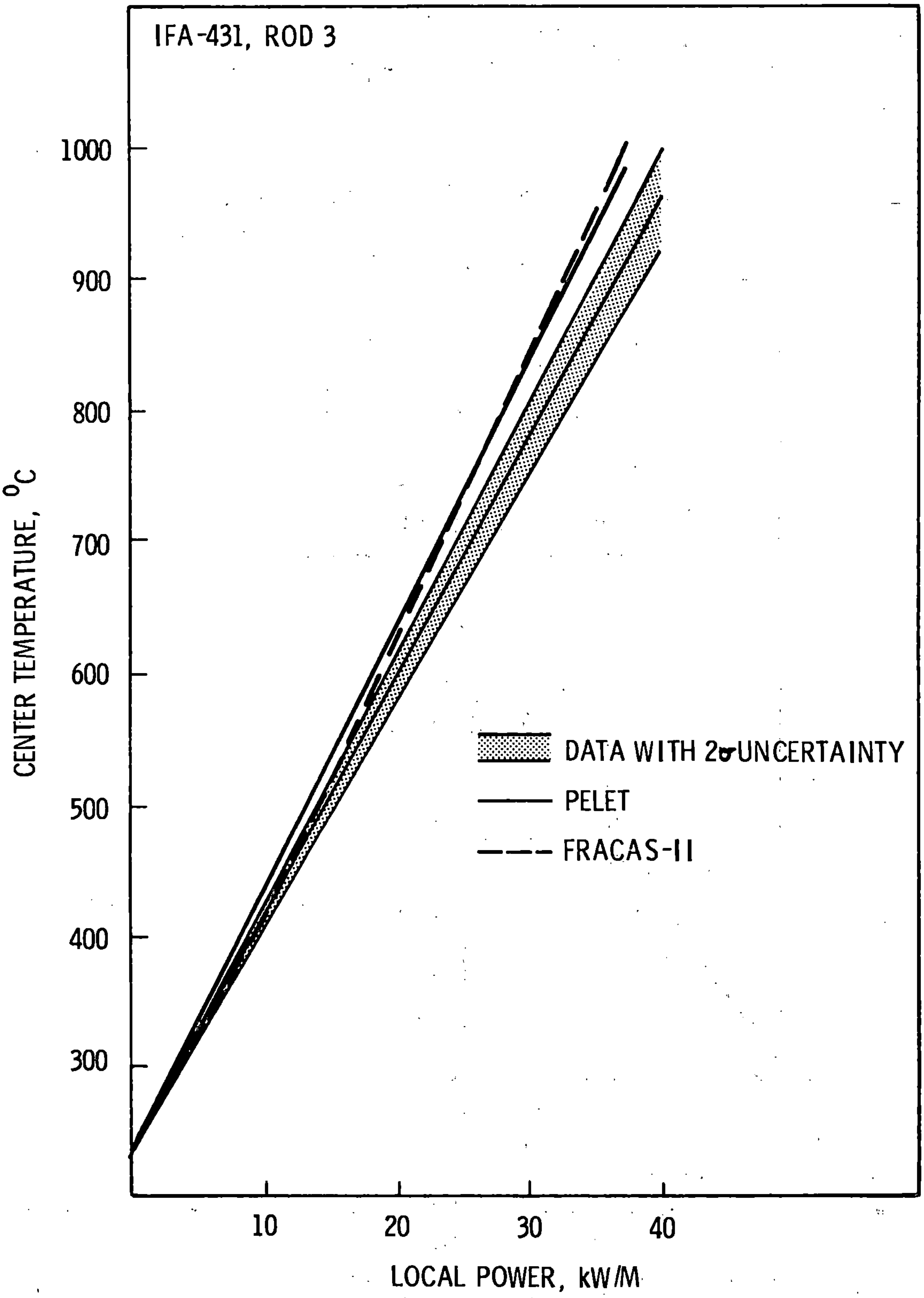

FIGURE 5. Measured and Calculated Centerline Temperatures for IFA-431, Rod 3 at Beginning-of-Life

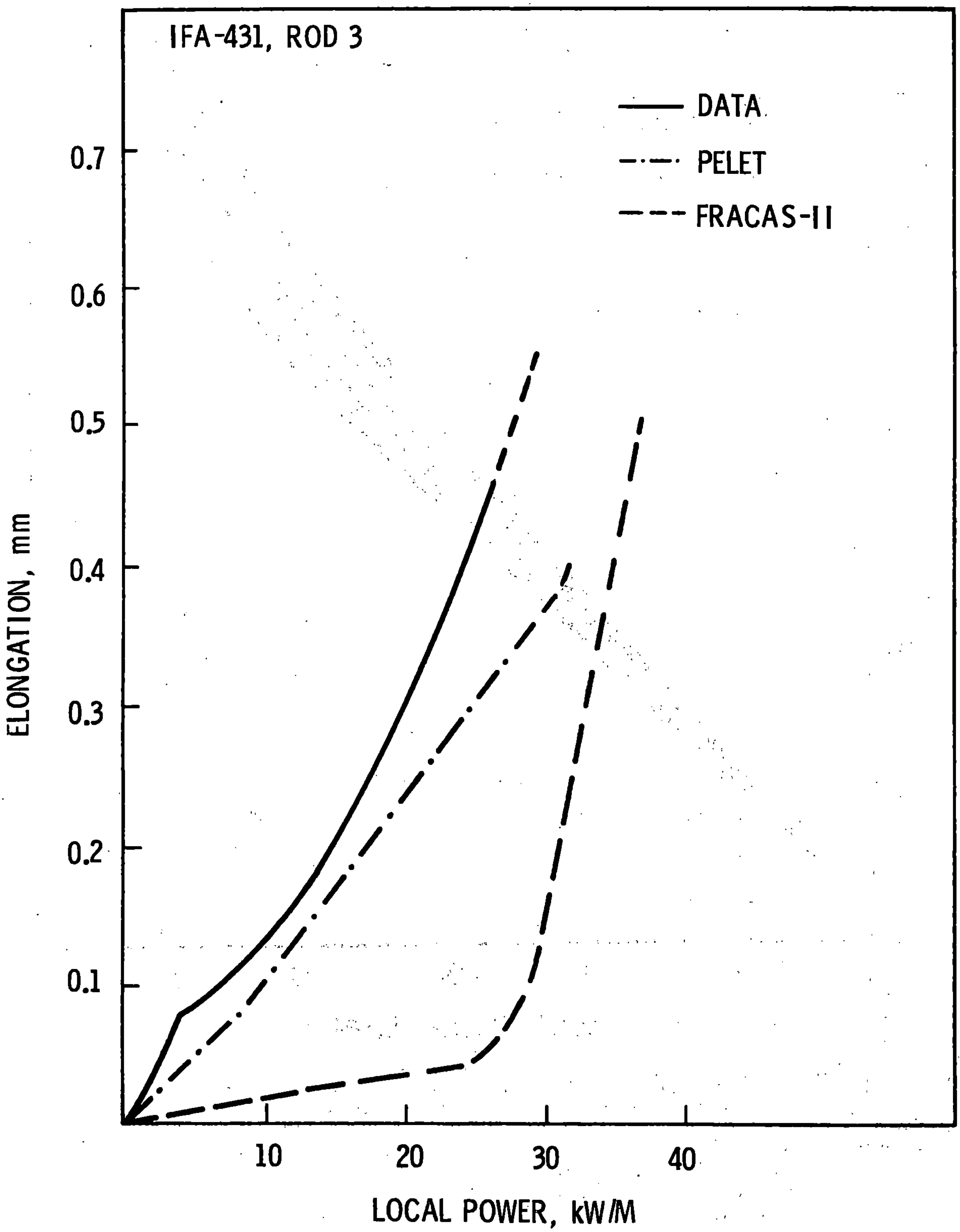

FIGURE 6. Measured and Calculated Axial Rod (Elongation from IFA-431, Rod 3 (First Rise to Power)

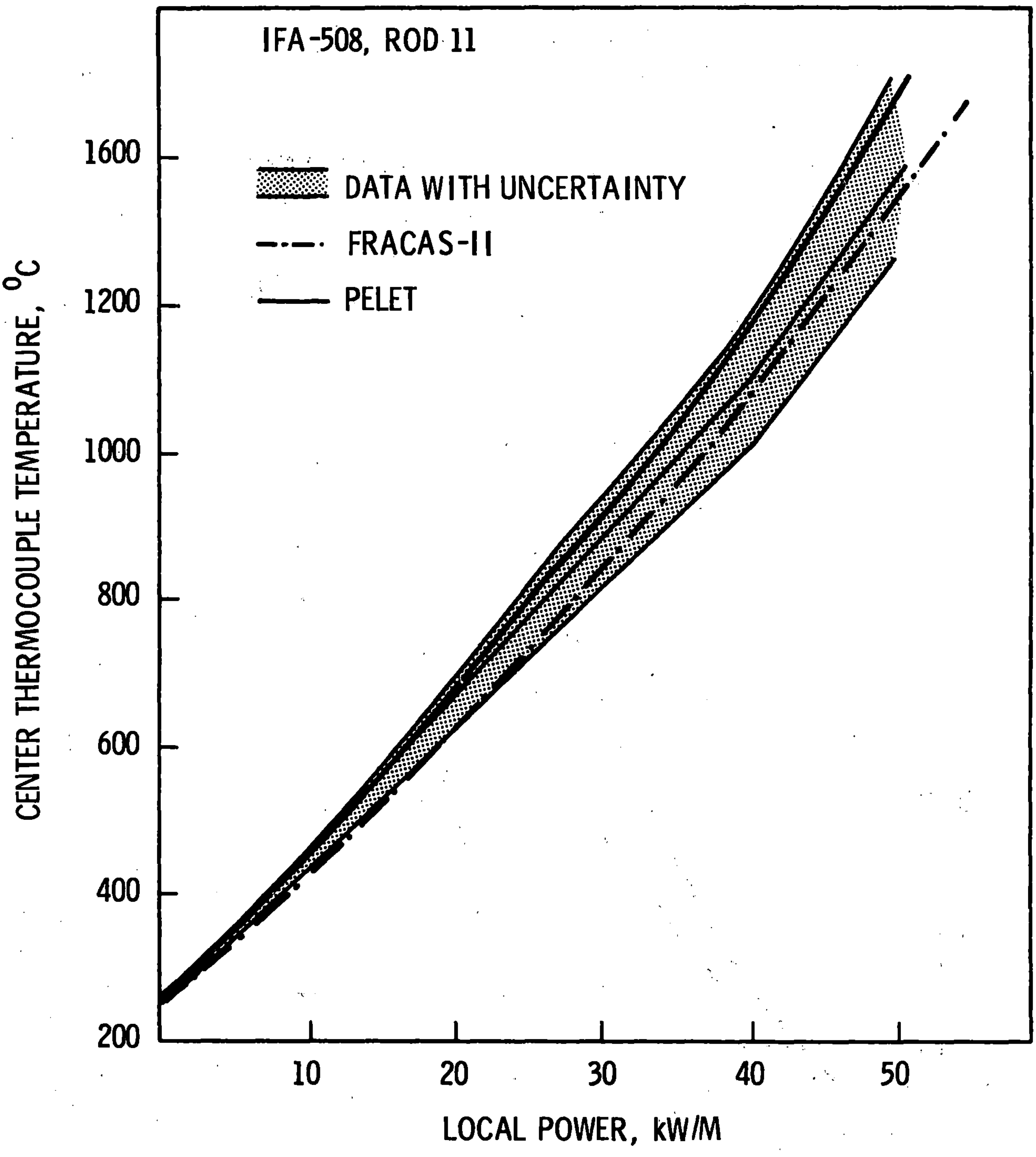

FIGURE 7. Measured and Calculated Centerline Temperature for IFA-508, Rod 11, at Beginning-of-Life

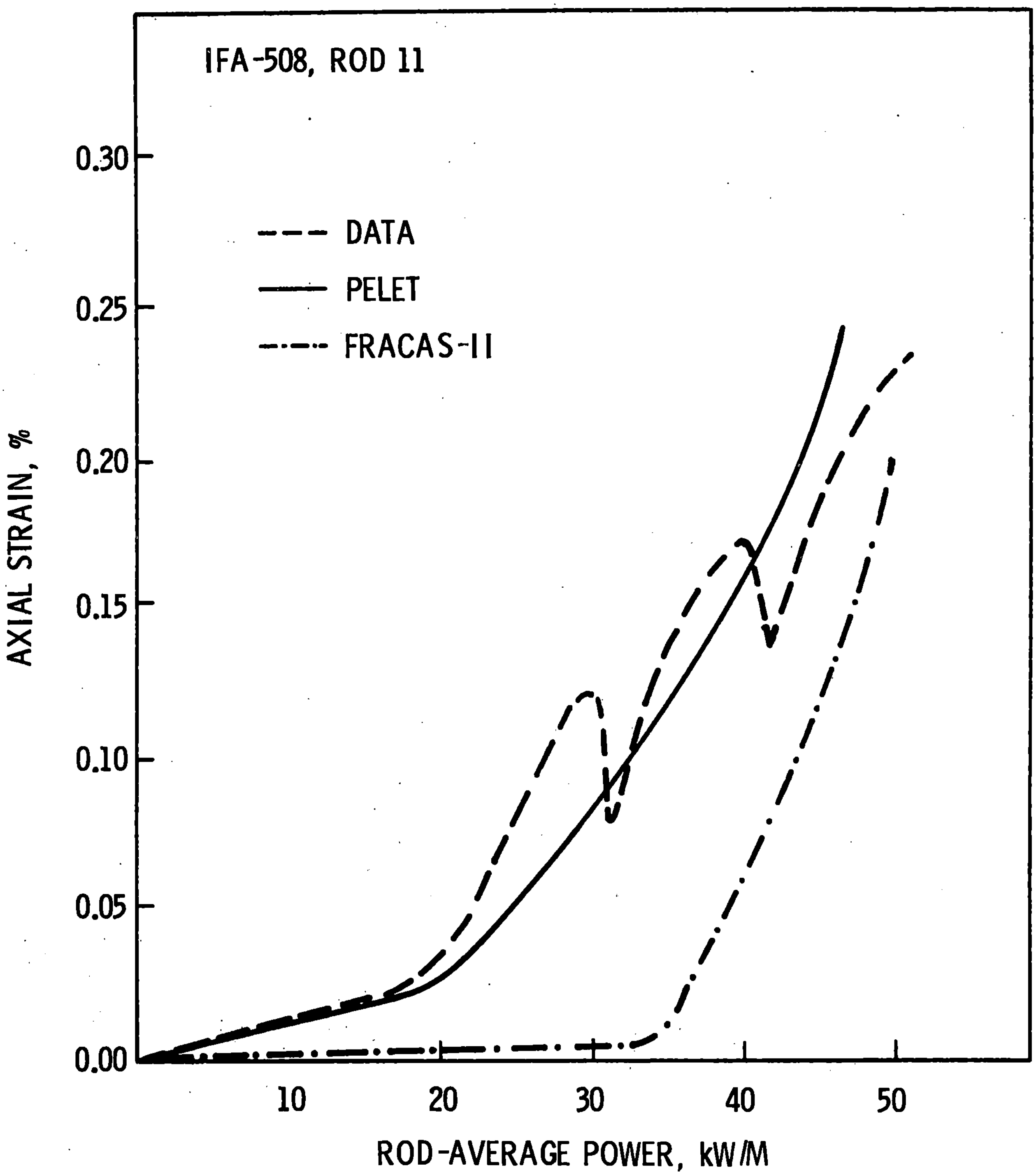

FIGURE 8. Measured and Calculated Rod Axial Elongation for IFA-508, Rod 11 at Beginning-of-Life (Burnup = 0.07 GWd/MTM)

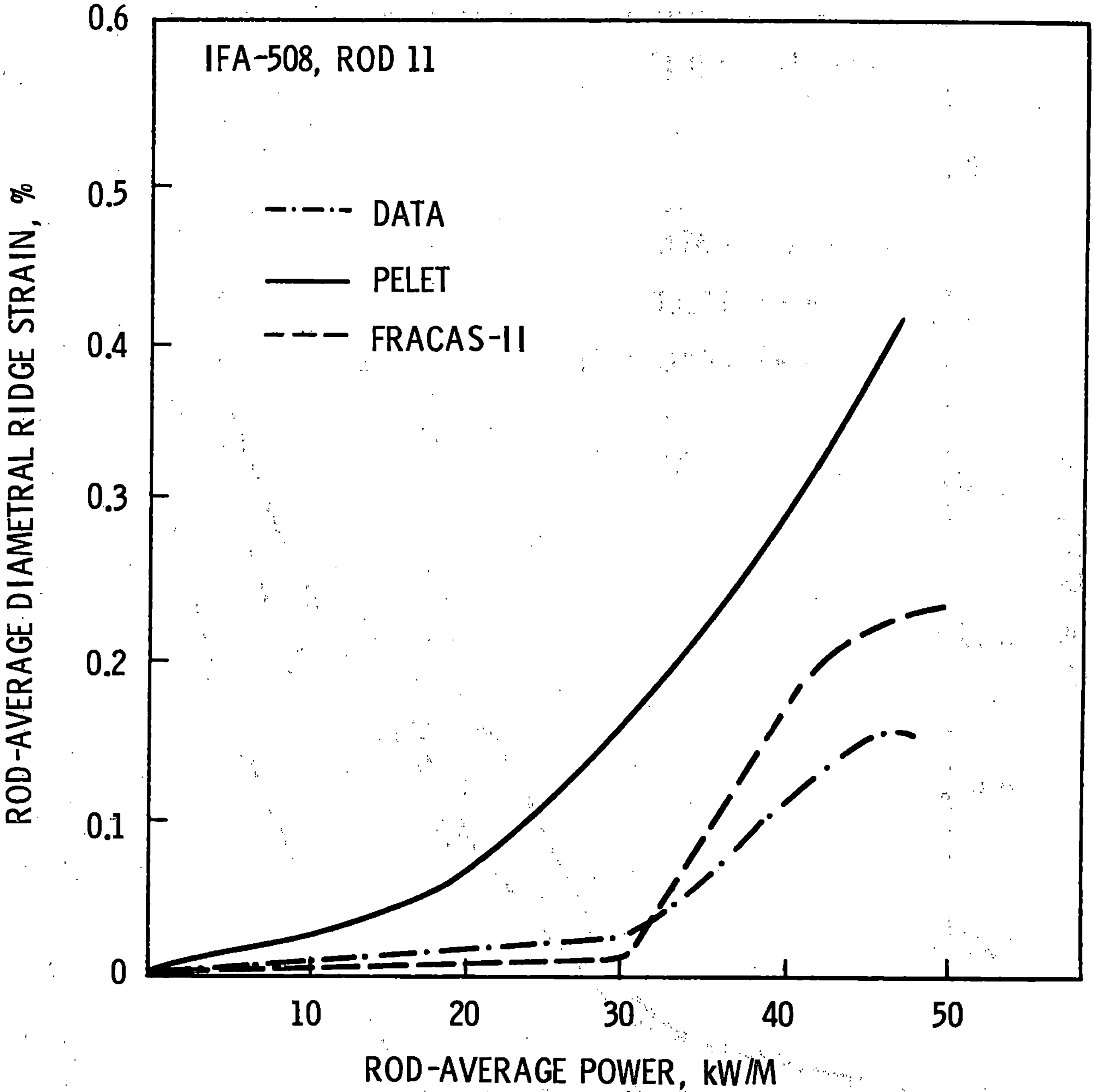

FIGURE 9. Measured and Calculated Rod-Averaged Diametral Strain (at Ridges) for IFA-508, Rod 11 at 0.07 GWd/MTM

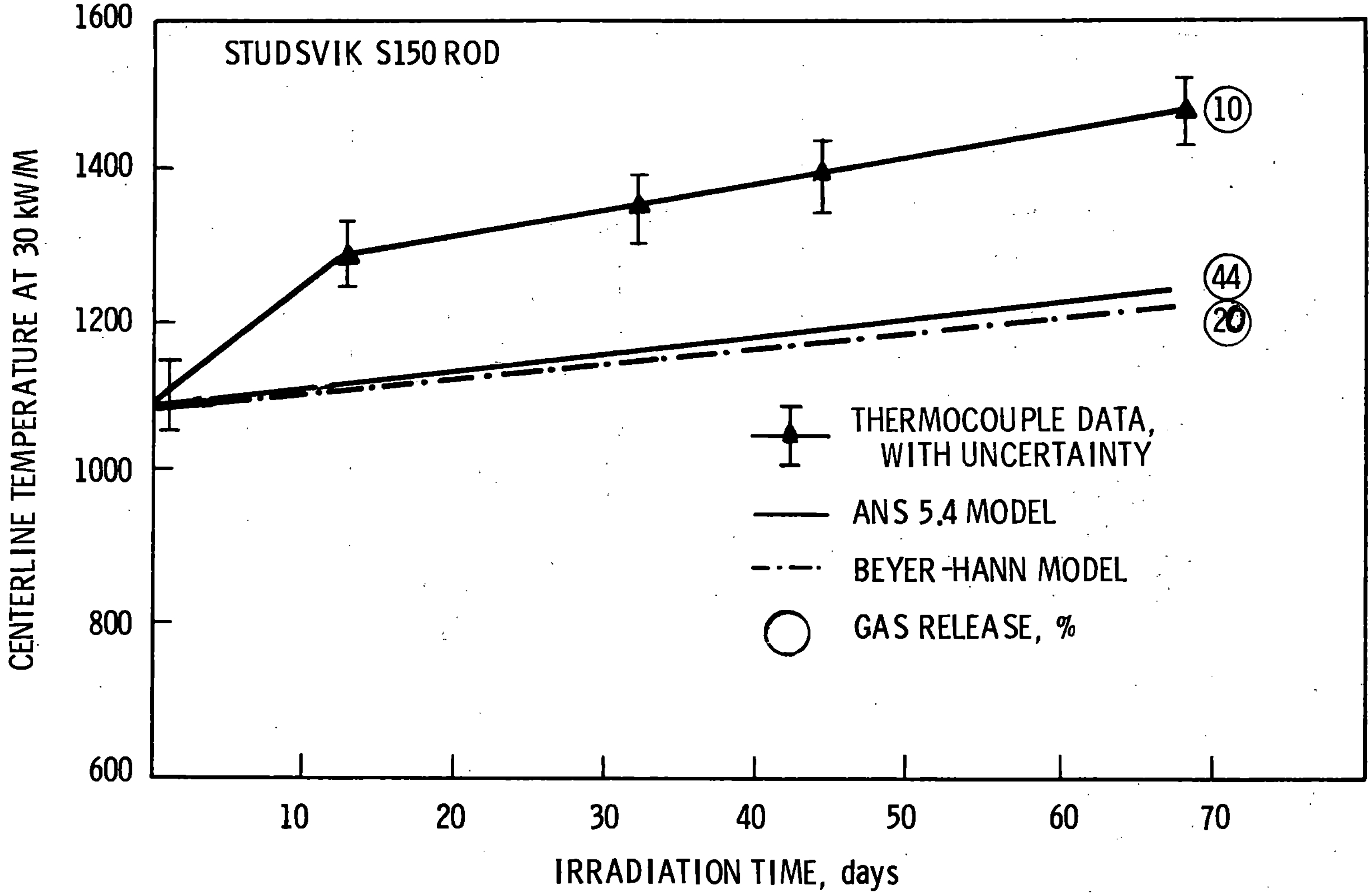

FIGURE 10. Measured Versus Calculated Center Temperature and Fission Gas Release for Studsvik S150 Rod (Both predictions utilizing PELET Mechanical Option)

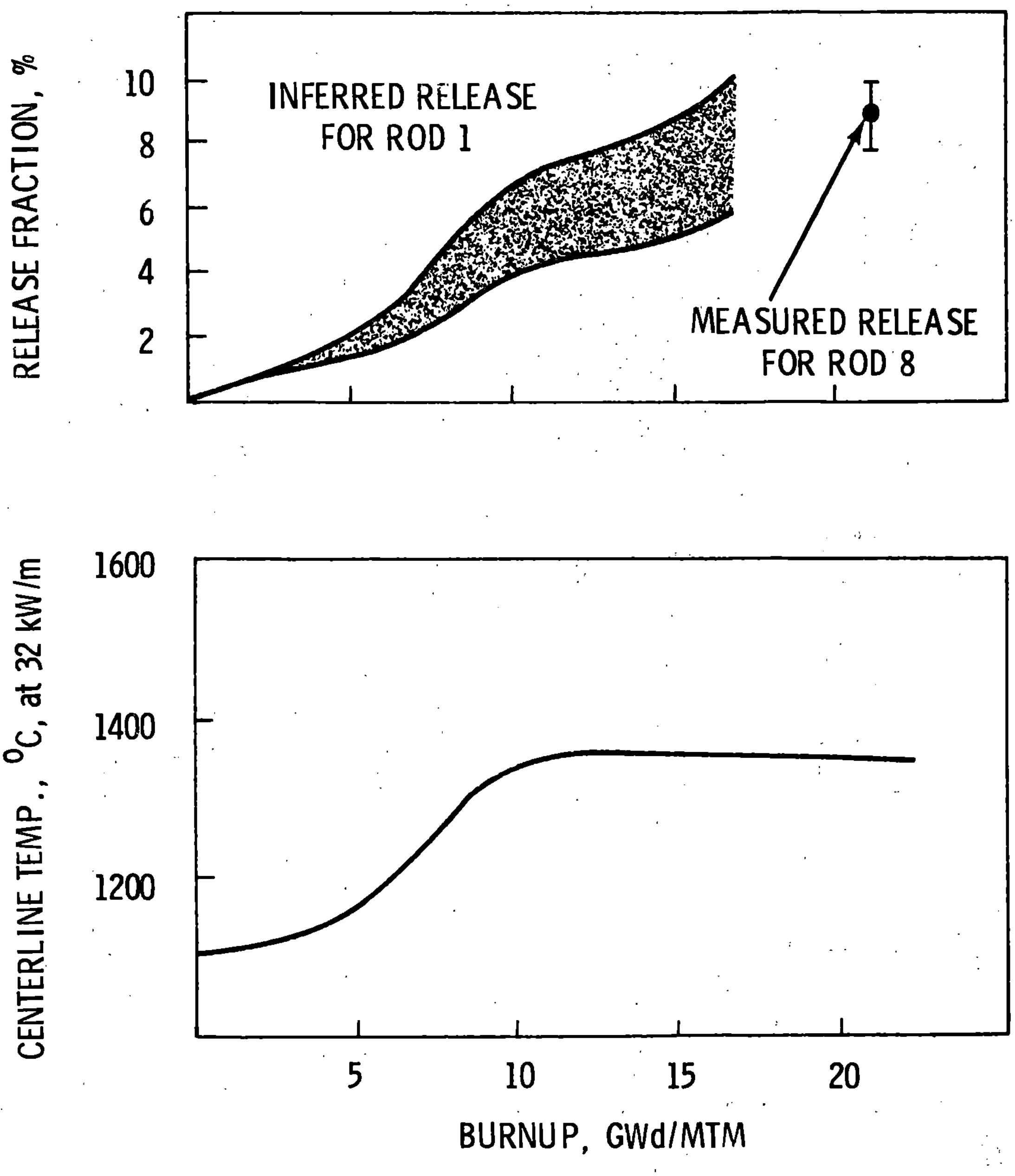

FIGURE 11. Measured Temperature and Gas Release Data for Rod 1, IFA-432, Plus Rod Puncture Data from Sister Rod 8, IFA-432.

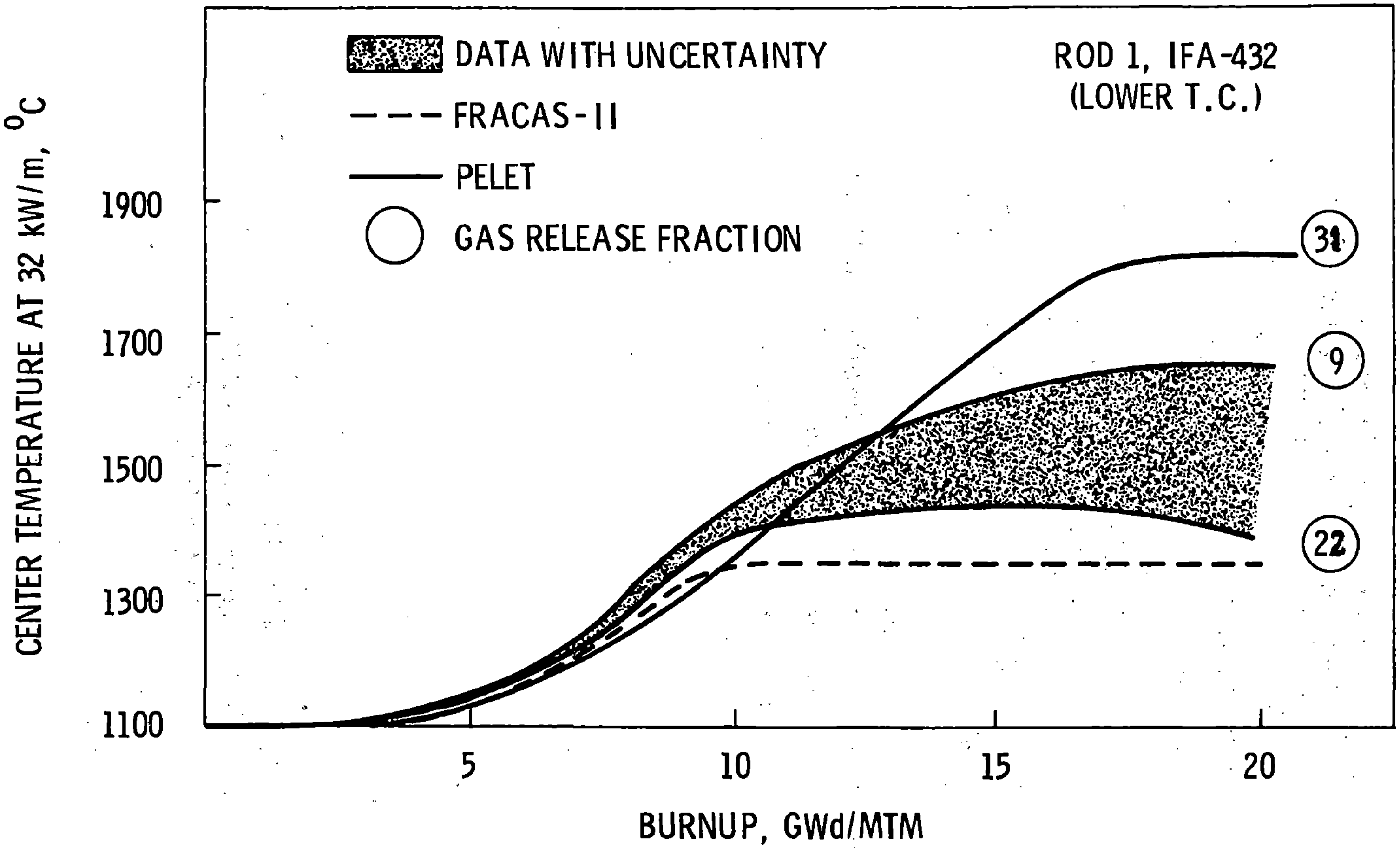

FIGURE 12. Temperature and Gas Release Data and Predictions for Rod 1, IFA-432. The Uncertainty in Temperature Represents Possible Thermocouple Decalibration of up to -1%/GWd/MTM.

RECENT PNL STUDIES ON GAP CONDUCTANCE AND FUEL STORED ENERGY

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RECENT PNL STUDIES ON GAP CONDUCTANCE AND FUEL STORED ENERGY

INTRODUCTION

A coordinated series of in-reactor fuel temperature measurements and ex-reactor "direct" gap conductance measurements have been conducted over the past 5 years for NRC by Pacific Northwest Laboratory (PNL). The focus of these PNL studies has been evaluation of gap conductance and fuel stored energy, as a function of fabricated gap size, power level, burnup, interfacial pressure, and fill gas pressure and composition. The studies have extended to include effects of fuel cracking and relocation and pellet-cladding mechanical interaction.

This report updates three areas of study:

1. Effects of fill gas pressure upon gap conductance
2. Re-analysis of in-reactor fuel temperature data, assuming cracked fuel in equilibrium with the cladding.
3. Impact of the above model on cladding temperature predictions for a loss-of-coolant accident (LOCA).

It is concluded that fuel cracking results in higher and more constant values of conductance versus power than heretofore supposed; but offsetting effects of degraded fuel conductivity mitigate the impact on LOCA calculations.

I. EX-REACTOR MEASUREMENTS OF PRESSURE EFFECTS OF GAP CONDUCTANCE; CORRESPONDENCE TO IN-REACTOR MEASUREMENTS

Measurements of pressurization effects on gap conductance have been compiled at PNL.[1] This completes a series of ex-reactor gap conductance measurements conducted over 3 years by John Garnier and Stefan Begej. The measurement technique used was the laser-flash "modified pulse design" (MPD) system described in detail by Garnier.[2,3] The measurement system consisted of thin UO_2 and zircaloy discs within a pressure chamber; a laser pulse coming through a quartz window would impinge on the UO_2, and the heat transmission across the UO_2-Zircaloy gap was monitored by a thermocouple on the back side of the zircaloy disc. From analysis of the shape of the temperature trace versus time, the gap conductance was determined for various combinations of gap separation, fill gas composition, and gas pressure. Figure 1 shows a sketch of this system.

The pressure at which conductance was measured varied from 0.1 to 7 MPa (1 to 70 atmospheres). The gap separations were 5.9 and 21.3 microns; temperature varied from 283 to 673K, and 4 gas compositions, ranging in conductivity from pure He to pure Ar, were used.

The classical expression for gap conductance is

$$h_{gap} = \frac{K_{gas}}{d_{eff} + g_1 + g_2} = \frac{K_{gas}}{d_{eff} + \frac{C_{gas}}{P}}$$

where h_{gap} = conductance
d_{eff} = effective gap separation
$g_1 g_2$ = temperature jump distances
P = gas pressure
C_{gas} = constant, dependent on gas composition and temperature
K_{gas} = gas conductivity

This expression clearly predicts that as pressure increases, h_{gap} increases to some constant value. The accepted values for C_{gas}(4,5) are such that for gaps in the order of 5 to 50 microns, the maximum is approached as P goes to 1 MPa for He, and is attained at even less pressure for gas mixtures/compositions having lower thermal conductivity than helium.

Results from the MPD apparatus are plotted in Figure 2. Gap conductance versus gas pressure is shown at fixed gap separation and gas temperature for helium, argon, and 50:50 mixtures of He-Xe and He-Ar. We definitely see qualitative agreement with our expectations. Comparison to quantitative estimates of conductance are shown in Figures 3 and 4 for helium and argon, respectively. The quantitative estimates of conductance were obtained using the nominal gap size (5.9 μm) and the Kennard expression for the temperature jump distances in the form (5)

$$(g_1 + g_2) = \left[0.2174 \frac{K_{gas}\sqrt{T}}{P}\left(\frac{2 - a}{a}\right)\sqrt{m}\right] \times 2$$

where $g_1 + g_2$ = total temperature jump distance (meters)
k_{gas} = gas conductivity (W/m-K)
T = gas temperature (K)
P = gas pressure (Pa)
M = kilogram-molecular weight.

The accommodation coefficient "a" is temperature dependent, but was taken as roughly 0.3 and 0.5 for the helium and argon, respectively.

The major conclusion that we draw from these two figures are:

1. The qualitative behavior of conductance versus pressure is predicted by the models for argon, but not totally for helium. The "hump" in the helium curve at 473K is quite repeatable and real, but not predicted by current models.

2. No simple adjustment of gap size or accommodation coefficients will account for all of the data.

3. We suspect the "effective gap" between two rough surfaces in close proximity is a function of both the temperature and the composition of gas gap, because subregions of the gap have markedly different separations and contribute differently to the heat transfer in different situations. In short, "the gap size" cannot be uniquely defined.

4. Conductances predicted by standard models using nominal gap sizes are correct in order of magnitude but generally low compared to measured conductances.

It is encouraging that at least the magnitude and qualitative features of the pressure dependence of conductance are also confirmed by a recent in-reactor experiment, wherein gas pressure was varied at constant power and the resulting changes in fuel centerline temperature were noted. This experiment is the subject of another report, Reference (6), so we only briefly show major results. In Figure 5, the inferred surface temperature changes are plotted as a function of gas pressure, for the case of a helium-filled gap at an estimated gap size of ~14 microns and a gas temperature of approximately ~600K. Note that the major effect occurs between 0.1 and 1.0 MPa, just as it did with helium in the ex-reactor tests. The magnitude of the effect is approximately the same also; these surface temperaure decrements represent an approximate 50 to 100% increase in the conductance.

II. PNL ESTIMATES OF GAP CONDUCTANCE, FUEL CRACKING, AND RELOCATION FROM IN-REACTOR DATA

The NRC through PNL has sponsored a series of irradiated Halden reactor tests in which gas composition and gap size were variables. These included instrumented fuel assemblies IFA-513, IFA-432, IFA-431, and IFA-527.[a] A common result of comparison between calculated and measured fuel centerline temperatures from these tests is sketched in Figure 6. Fuel performance codes considering fuel expansion only (no relocation) overpredict the measured temperatures by about 250 to 300°C at, say, 35 kW/m. On the other hand, putting the fuel in intimate contact with the cladding results in a underestimate of fuel temperatures by about the same amount, assuming no fuel thermal conductivity degradation. Since observed cracks in irradiated fuel do have nonradial components, we presume that fuel cracking does result in some degradation of the effective UO_2 conductivity. However, steady-state fuel centerline temperature data gives no clue as to the true amount of fuel thermal conductivity degradation. A whole spectrum of gap conductance enhancement and conductivity degradation will yield the same temperature versus power relationship for a given rod. Fortunately, transient fuel centerline temperature is more sensitive to gap/fuel thermal resistance partition. From numerical studies we have concluded the following:

1. The transient centerline temprature is progressively more sensitive to resistance partition variations for rods with progressively more resistance in the fuel-cladding gap; e.g., it is most sensitive for xenon-filled rods.[7] This is why all rods in IFA-527 are xenon-filled.)

2. Transient fuel centerline temperatures are quite insensitive to the spatial distribution of thermal conductivity degradation given distributions of equal volume-average value. Similarly, transient fuel temperatures are insensitive to azimuthal variations of thermal resistance or of power generation.[7,8]

(a) A listing of these rods and their design parameters appears as Appendix A.

From the above, we believe the examination of transient temperature behavior in mixed-gas and xenon-filled test rods will give enhanced estimates of the real partition of resistance between fuel and gap. Given an estimate of the gap resistance (conductance) we can estimate the physical gap size. Such an estimate is most accurate for xenon-filled rods, wherein the gas composition stays essentially stable for long burnup, and the temperature jump effects are minimized.

Our examination of definitive transient behavior has so far been limited to the mixed gas rod of IFA-513 (23% Xe-He) the fission-gas saturated, high burnup rods of IFA-432, and scram data (only) from the xenon-filled IFA-505 rods built by the Halden Project. Several common threads have run through all these examinations:

1. Given a choice between model predictions involving a power/fuel temperature-dependent gap size as opposed to a constant gap size, transient temperature data consistently favor the constant gap size. The fuel seems to be revealing itself incapable of shrinking back away from the cladding upon power decrease. This is a logical consequence of fuel cracking.

2. The assigned conductance value of rods that best fits the transient data is usually significantly higher than that inferred from the centerline temperature assuming laboratory (solid-piece) UO_2 conductivity. The only exception has been rod sections at powers high enough to produce firm fuel-cladding contact. Thus the transient data appears to confirm concomitancy of pellet cracking, relocation, and thermal conductivity degradation.

Some examples of calcuated/measured transient temperatures are given in Figures 7 through 9. In Figure 7 we show the results of a 20% rapid drop in power, for Rod 1 of IFA-432 at ~16,000 MWd/MTM peak burnup. The thermocouple measured/calculated response is plotted as $-\ln (T_N-a)$ versus time, where T_N is the normalized temperature and "a" is the ratio of initial to final power. This parameter was chosen because the curvature (slope) of this curve can be shown to be relatively insensitive to the choice of thermocouple time constant. In Figure 7 we see the data favoring the cracked fuel constant conductance model, which might be expected for medium burnup fuel.

In Figure 8 we see the same behavior from Rod 6, IFA-513 at beginning-of-life. The constant conductance model is again favored, even at though this fuel had seen only a few days' irradiation prior to the test.

Finally in Figure 9, we see the same behavior, this time from a full scram, for a similar rod from IFA-505. It is apparent that the constant conductance model is relatively more successful, not only for 20% drops but also for full scrams.

We have taken the "cracked-fuel, constant conductance" idea into a complete re-evaluation of steady-state temperature data. This reevaluation is discussed in a document by Williford.[9] In brief, we assume a hydrostatic stress in the fuel and across the gap, assume a total crack length of (equal width) cracks within the fuel, and then relate stress to crack/gap closure in order to find effective values of fuel thermal conductivity, elastic moduli, and gap size/conductance. The adjustable parameters in this model are the assigned crack and gap roughnesses, which we selected as functions of fabricated gap size to get resistance partitions commensurate with transient data. Resulting estimated gap sizes are shown in Figure 10 for Rod 1, IFA-513, (230 micron fabricated gap, He-filled). Also shown on this plot are Miller's[6] estimates for the gap size of a nearly identical IFA-430 rod, based on temperature changes noted during gas exchange. Note the correspondence of these two totally independent estimates, and their mutual divergence from the "uncracked, expanding pellet" model (shown across the top of the figure).

To further emphasize the divergence of this new model from other versions, we present Figures 11 through 13. These figures show hot gap size, gap conductance and fuel stored energy <u>as deduced from the same set of centerline temperature/power data</u> by both the conventional and the "new" views. These parameters are plotted against fabricated gap size, for He-filled rods at beginning-of-life at a power level of 20 kW/m. The divergence of the stored energy values in particular suggest that the new view of fuel rod mechanics and heat transfer may have consequences to LOCA calculations. This is the subject of the next section.

III. SENSITIVITY OF LOCA CALCULATIONS TO VARIATIONS IN MODELS

This section briefly discusses the sensitivity of LOCA heat transfer calculations to variations in the major models. The results shown should be viewed only for the relative variations portrayed. The calculational procedure is so simplified that little meaning can be attached to the absolute values of predicted temperatures. This is explained below.

A. CALCULATIONAL PROCEDURE

The small transient code MWRAM was used to make all the calculatiuons presented here. The code solves the coupled fuel and cladding nonlinear radial heat transfer equations. The initial steady-state temperature distribution, and gap conductance, is found from input geometry, flux depression, power and centerline temperature. Then the transient calculation proceeds.

The code was modified to have a more realistic gap conductance estimation and to accept the film coefficient and coolant temperature as a function of time. The specific form of the gap conductance chosen was

$$h_{gap} = \frac{k_{gas}}{\Delta r_c - \Delta r_f + d_{gap}}$$

where Δr_c and Δr_f are changes in cladding and fuel radius due to thermal expansion, respectively, and d_{gap} is the design gap separation.[a]

The particular history of power, coolant temperature, and film coefficient input to MWRAM are shown in Figure 14. These functions do not correspond to any particular LOCA, but are simply representative of the variations that are predicted to occur. Also, there is no feedback among the fuel rod heat flux, coolant conditions, and film coefficient. For these

(a) Temperature jump distance is ignored because, for the two cases to which the modified code will be applied (high pressure helium and fission gas-saturated fill gas) the temperature jump distance is negligible.

reasons, the results of the MWRAM calculations are not correct in absolute value. Nevertheless, the relative variations in the results due to variations in models are probably representative of much more sophisiticated LOCA calculations.

Two rod types were input: a PWR-sized pressurized helium-filled rod, and a BWR-sized rod filled with fission gas. Because of the relative nature of the calculational results, the specific numerical values of input are not very important. They correspond to test rod steady-state data.

B. CALCULATIONAL RESULTS

The two extreme models discussed earlier can be tested by MWRAM. For the conventional model, the gap conductance is allowed to vary with temperature as indicated in the previous section. For the cracked fuel model, the gap conductance is fixed at the conventional-model initial value, and the fuel conductivity is modified by a temperature dependent crack factor so as to produce the ~~conventional-view centerline temperature at various power levels. A table~~ of crack factors and corresponding fuel volume average temperatures is produced, and MWRAM interpolates with this table as the transient proceeds.

Figures 15 and 16 show predictions of LOCA cladding temperatures for the PWR and BWR cases, respectively. The difference in absolute value of cladding temperature between the two cases is a result of differences in design and initial condition, and should be disregarded. But the relative differences between the model predictions in the two cases is of significance. In the PWR (He-filled) case, there is very little difference in the model predictions. For the BWR case (fission gas filled), there is some difference—about 50K in the peak cladding temperature. This would indicate that the model differences only become significant for LOCA calculations applied to unpressurized rods of high burnup (>20,000 MWd/MTM) experiencing significant gas release (>2%). It is only for these conditions that the gap conductance degrades significantly due to fission gas. Even in these conditions, the impact of the model differences on the peak cladding temperature is still only about 50K.

IV. CONCLUSIONS

Recent PNL measurements and studies relative to gap conductance and stored energy have been reviewed. Major conclusions are as follows:

1. Ex-reactor measurements of gap conductance vs. pressure show a 50-100% conductance increase as the pressure increases from 0.1 to 1.0 MPa, but little change with further pressure increase. This behavior is corroborated by in-reactor measurements and qualitatively predicted by current gap conductance models. Quantitative and detailed comparison between models and the well-controlled ex-reactor experiments, however, is poor, especially for helium.

2. In-reactor measurements of fuel temperature have shown inferred effective gap conductance consistently higher than those calculated by standard fuel performance codes, when these include no fuel relocation. Steady-state measurements at PBF, and transient measurements at Halden, have further shown that fuel relocation and gap conductance enhancement is accompanied by fuel thermal conductivity degradation. Additionally, gap conductance does not appear to be a strong function of fuel power/temperature.

3. The impact of fuel cracking/relocation/thermal conductivity degradation upon peak LOCA cladding temperature is only significant for a high-resistance, fission-gas saturated rod. Even for these conditions, the increase in calculated peak cladding temperature due to adopting a cracked pellet model is probably less than 50K.

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2. John Garnier and Stefan Begej, Ex-Reactor Determination of Thermal Gap and Contact Conductance Between Uranium Dioxide: Zircaloy-4 Interfaces. Stage I: Low Gas Pressure, NUREG/CR-0330, PNL-2696, April 1979.

3. John Garnier and Stefan Begej, "Ex-Reactor Determination of Thermal Gap Conductance between Uranium Dioxide: Zircaloy-4 Interfaces," presented 15th Annual Thermoconductivity Conference, Ottawa, August 24-27, 1977.

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8. Letter, D. D. Lanning to H. H. Scott (NRC/RSR), December 11, 1979, "Changes in Calculated Fuel Temperatures Caused by Explicit Radial Dependence of the "Crack Factor" in Fuel Thermal Conductivity."

9. R. E. Williford, et al., The Analysis of Fuel Relocation for the NRC/PNL Halden Assemblies IFA-431, IFA-432, and IFA-513, NUREG/CR-0588, PNL-2709, 1980.

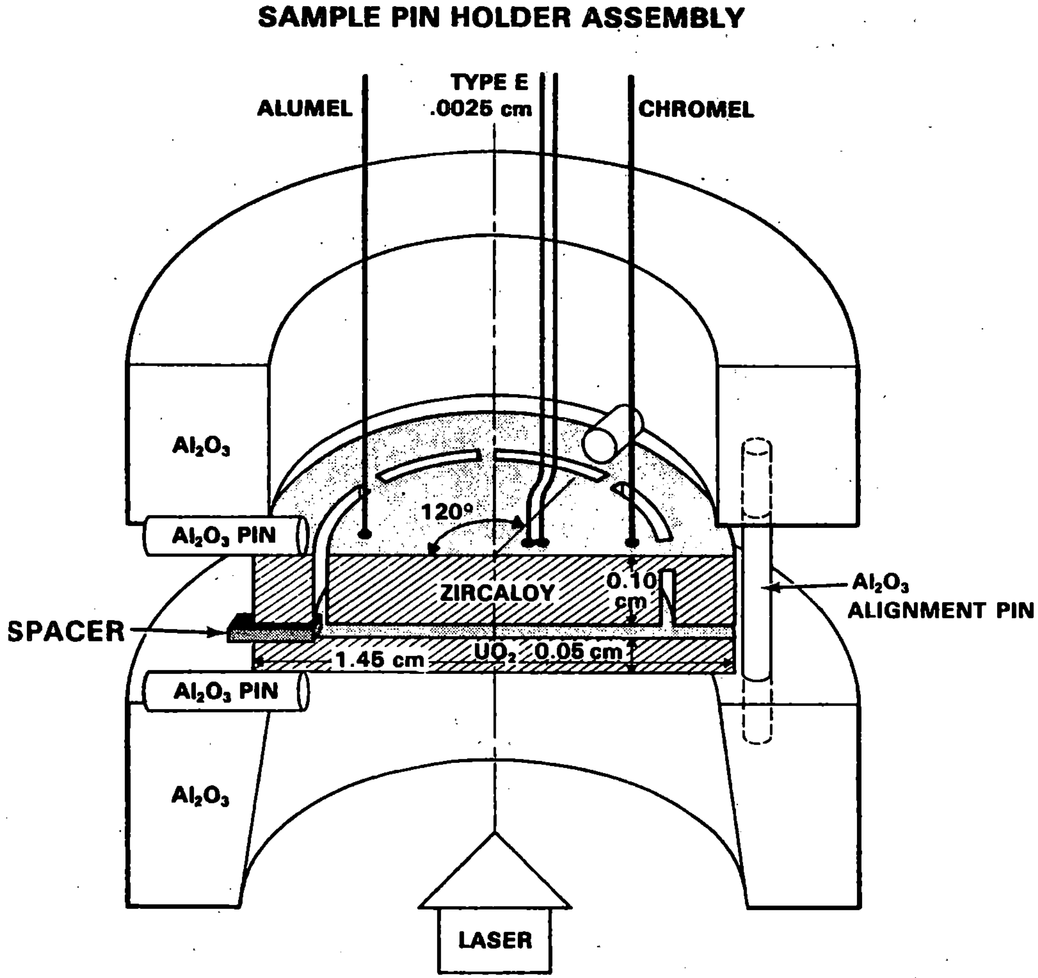

FIGURE 1. Sketch of the Measurement System For Gap Conductance

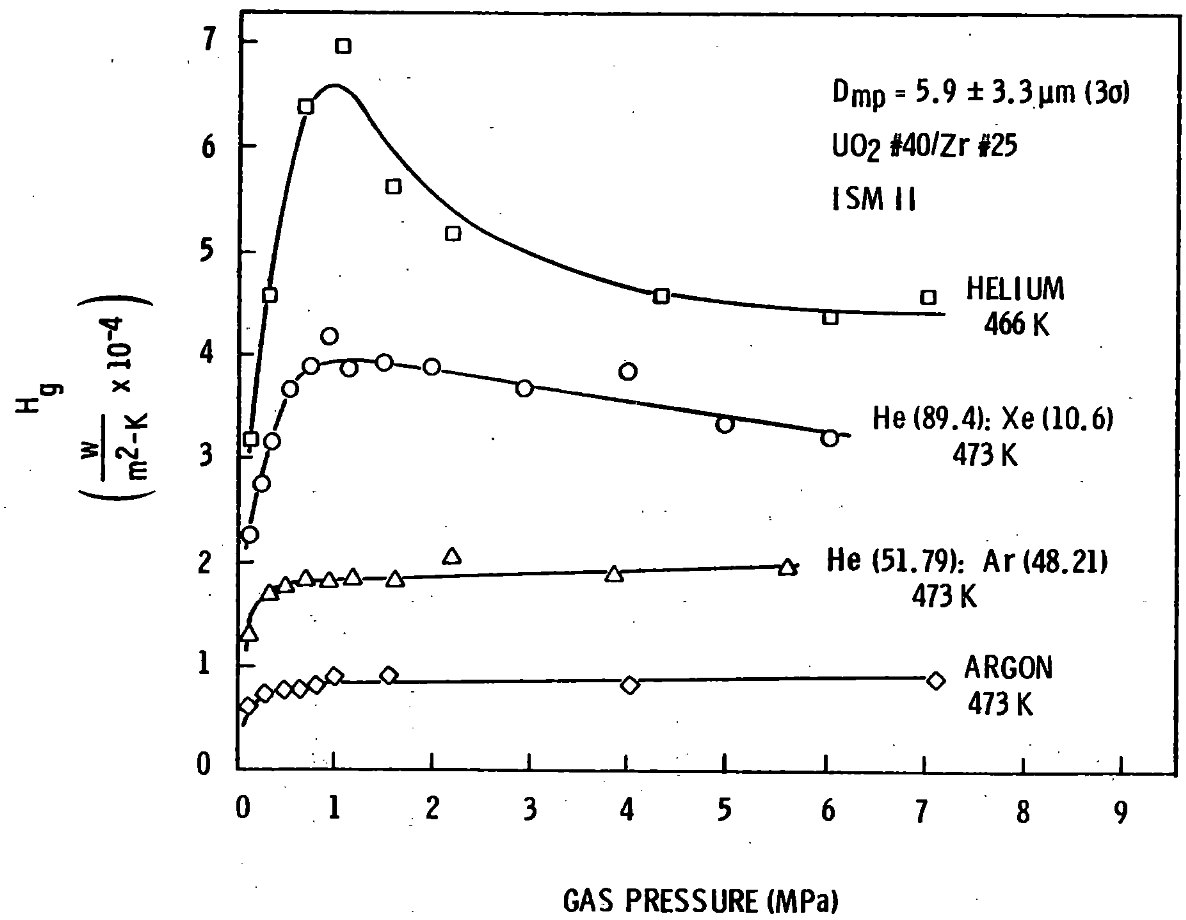

FIGURE 2. Measured Gap Conductance, Hg, vs. Gas Pressure for Various Gas Mixtures at 473°K at a Gap Separation of 5.9 ± 3.3 µm

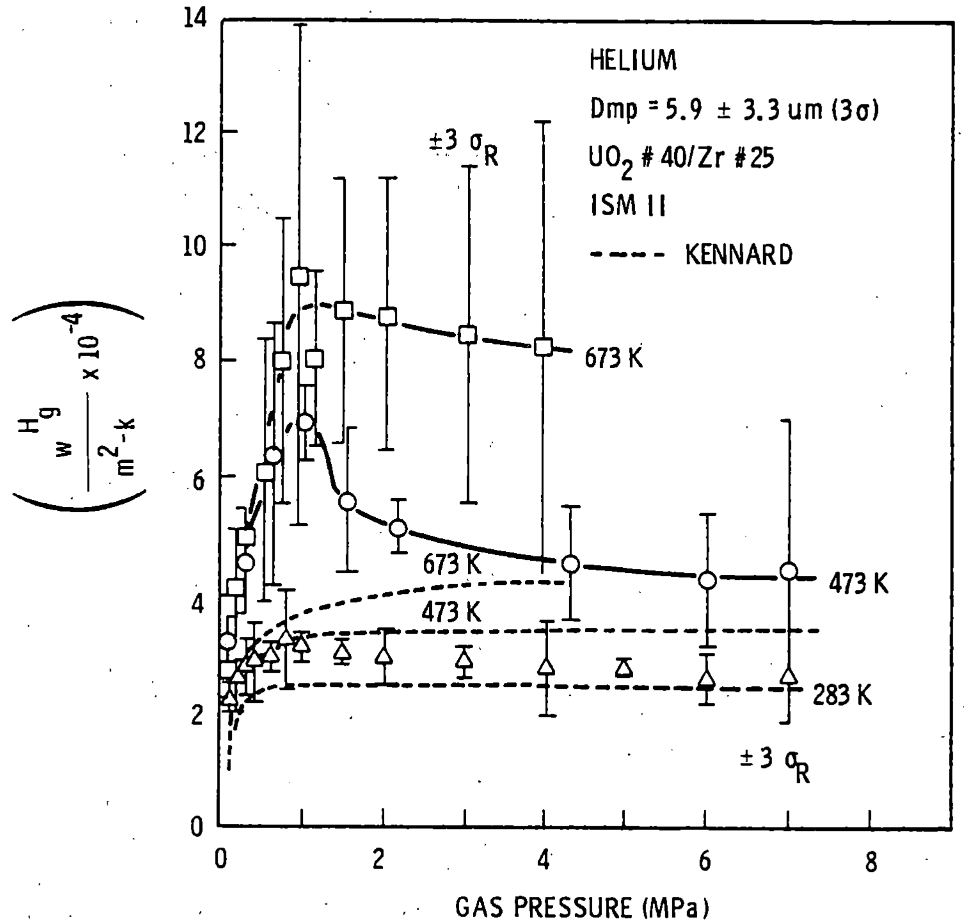

FIGURE 3. Measured Gap Conductance vs. Pressure in Helium at Various Gas Temperatures. The dashed lines represent calculated conductances using the Kennard expression for temperature jump distance (5).

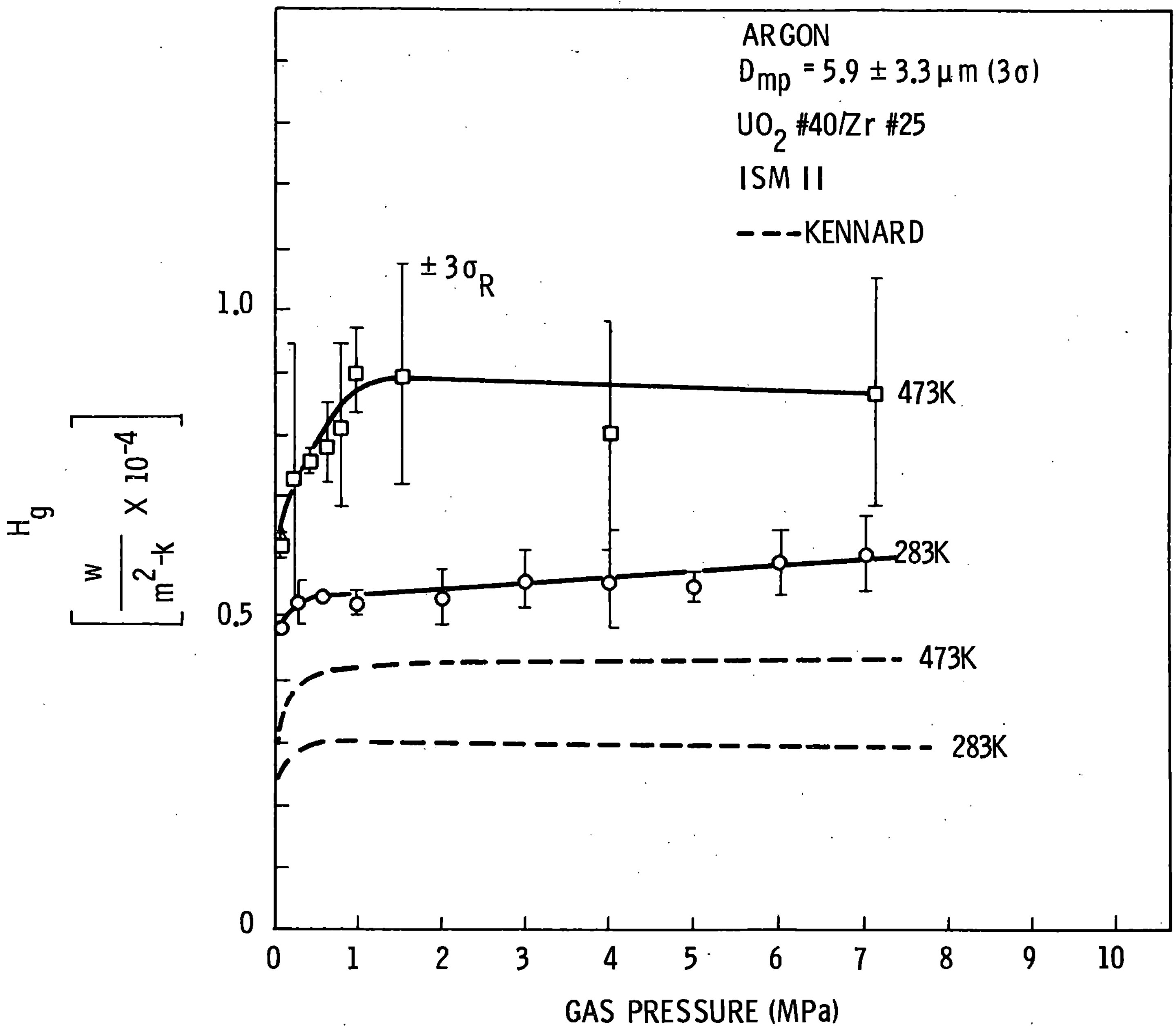

FIGURE 4. Measured Gap Conductance vs. Pressure in Argon at Various Gas Temperatures. The dashed lines represent calculated conductances using the Kennard expression for temperature jump distance (5).

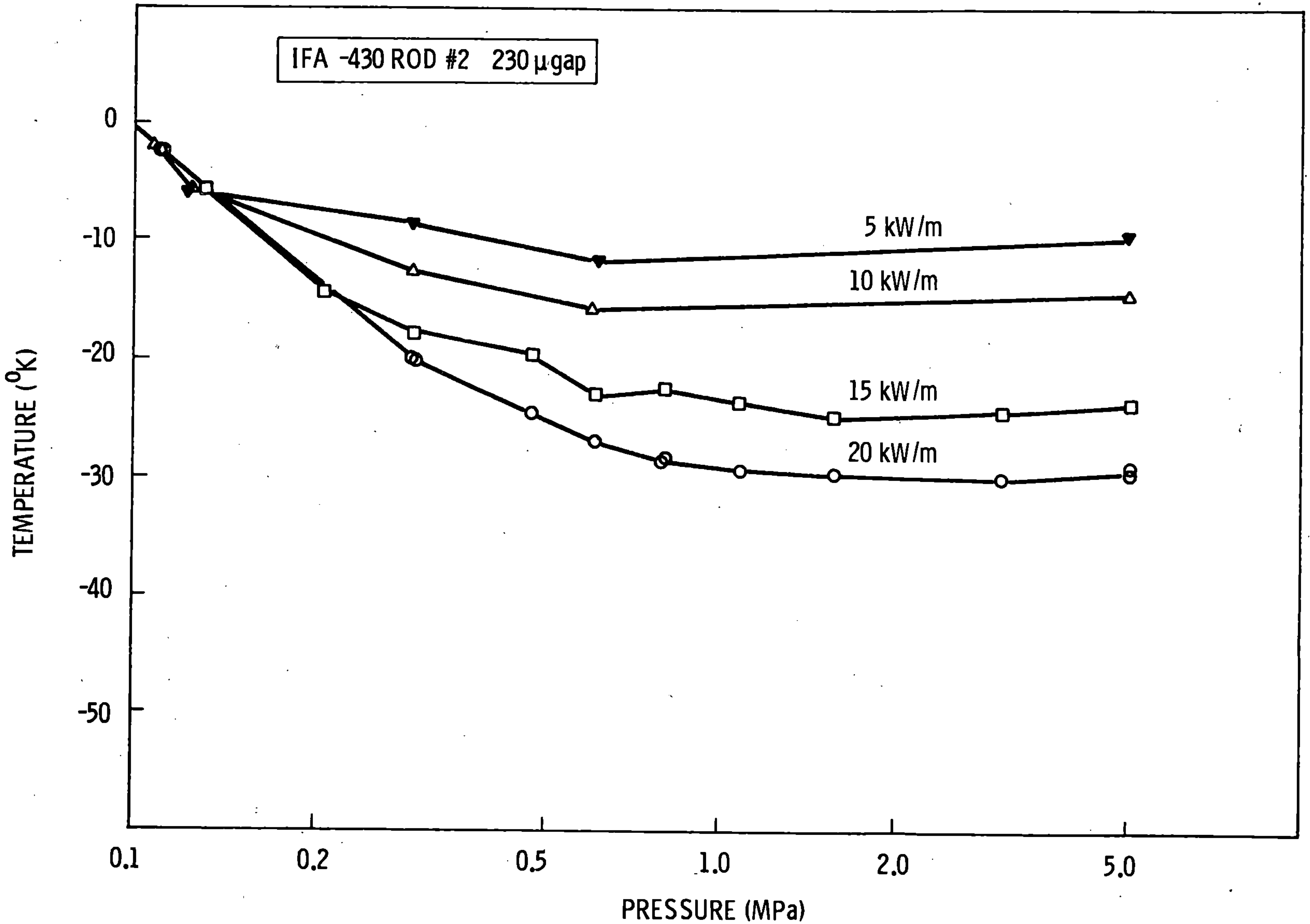

FIGURE 5. Inferred Decrements of Fuel Surface Temperature vs. Increasing Helium Gas Pressure, from IFA-430, rod 2, at Indicated Powers

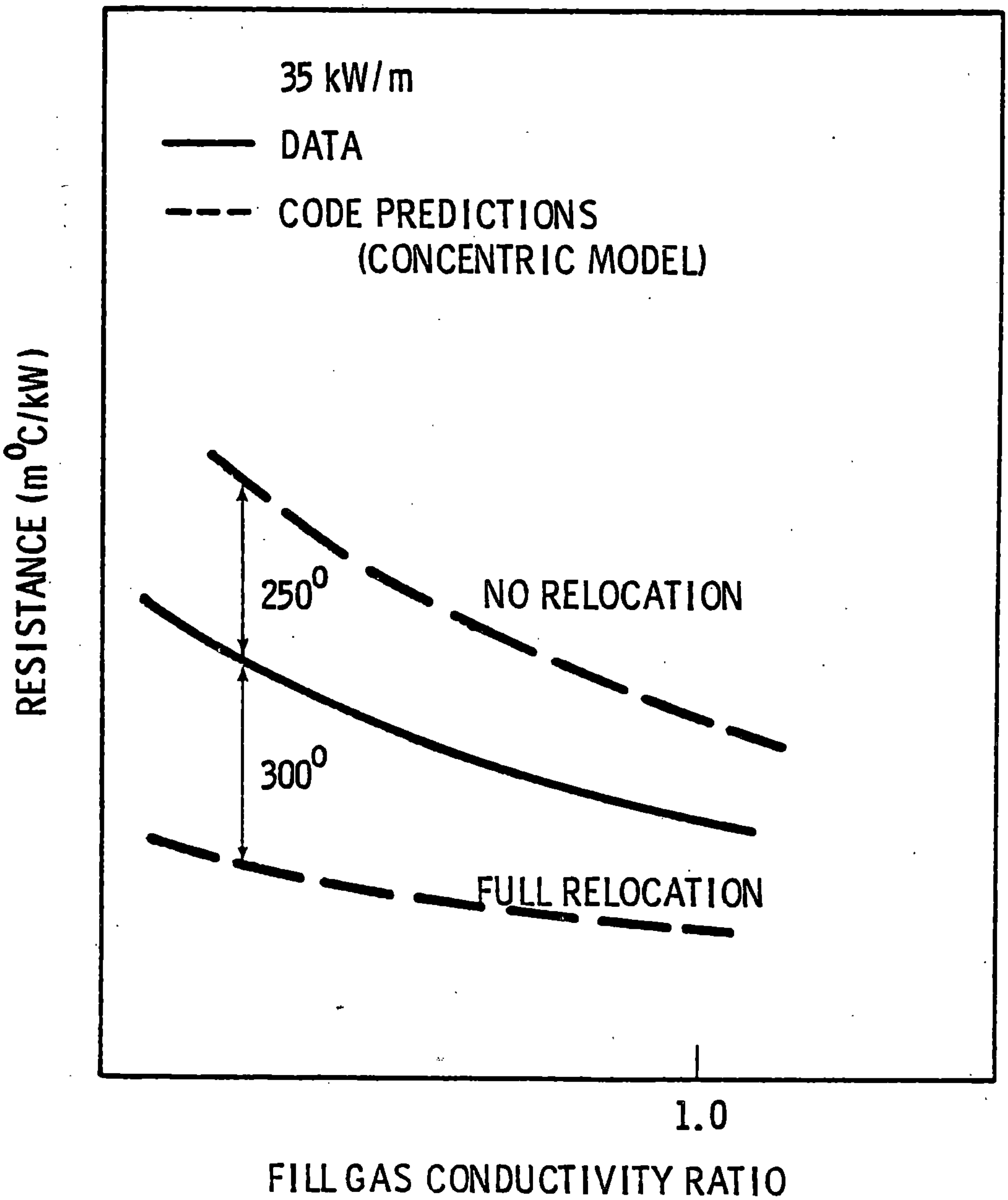

FIGURE 6. Typical Fuel Rod Resistance Trend vs. Gas Conductivity, Expressed as Ratio of Conductivity to that of Pure Helium

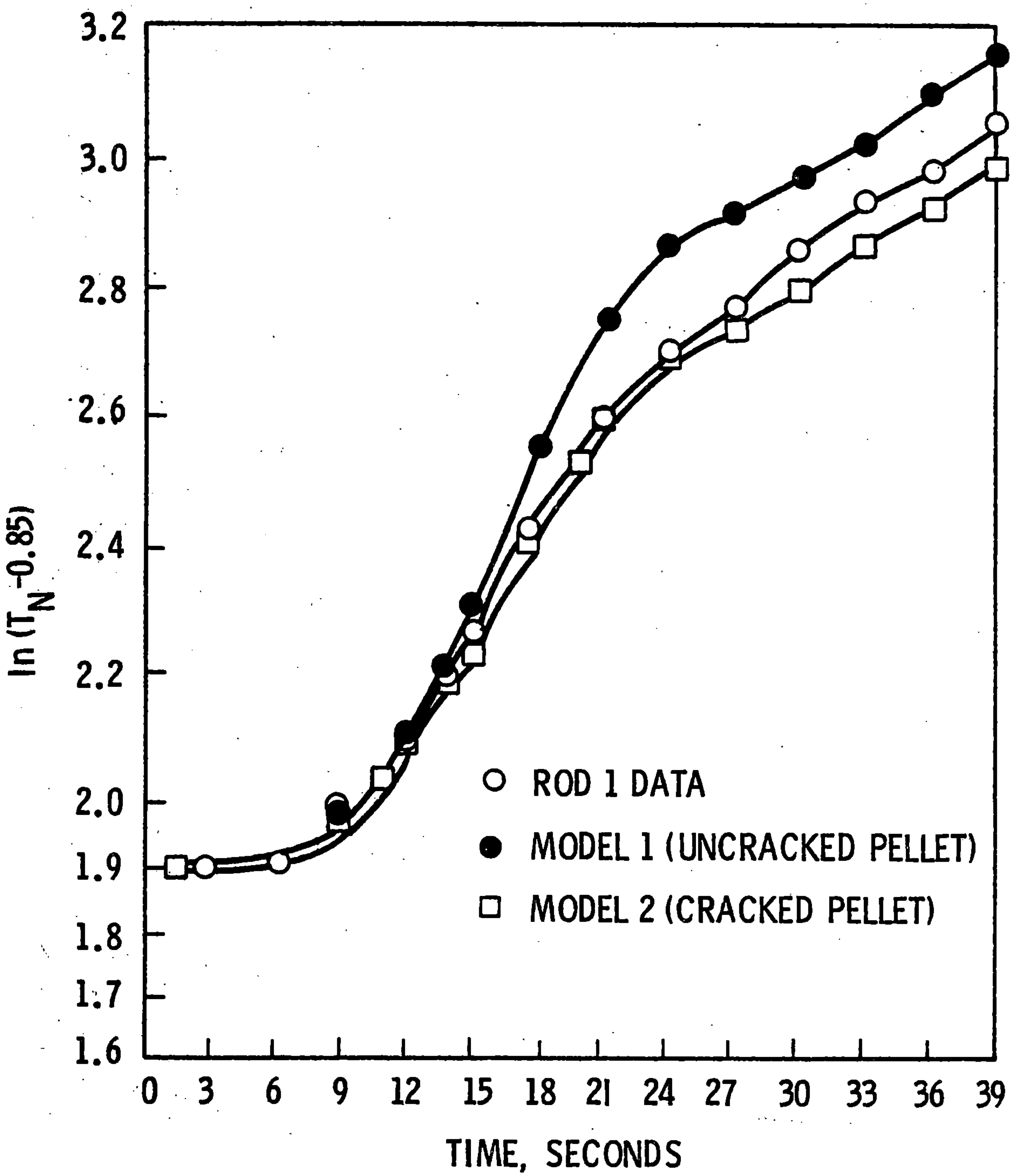

FIGURE 7. Model 1 and Model 2 Predictions Compared to Rod 1, IFA-432 at 17,000 MWd/MTU (230 μm fabricated gap) During a 20% Power Decrease

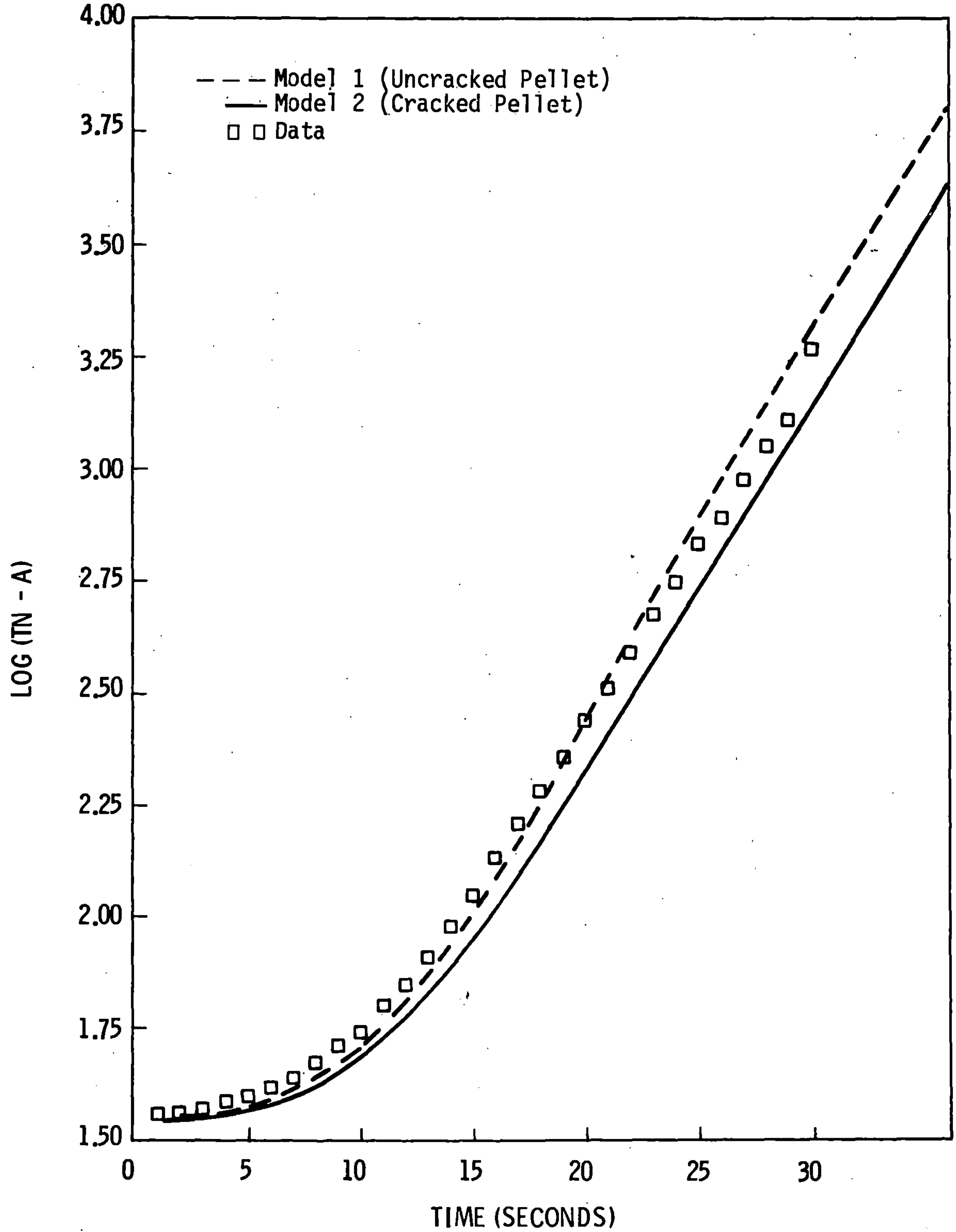

FIGURE 8. Data and Model Predictions for Rod 6, IFA-513 (Mixed Gas Rod) at Beginning-of-Life

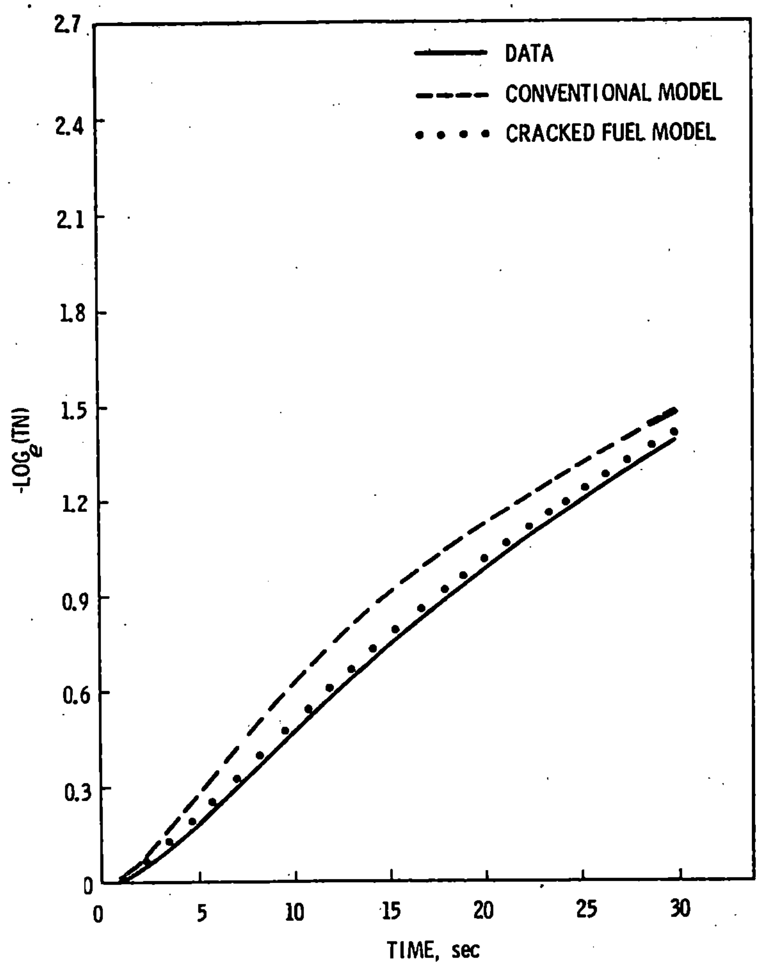

FIGURE 9. Scram Data from a Xenon-Filled Rod, Relative to Predictions by Solid Pellet (conventional) Model, and Cracked Pellet Model

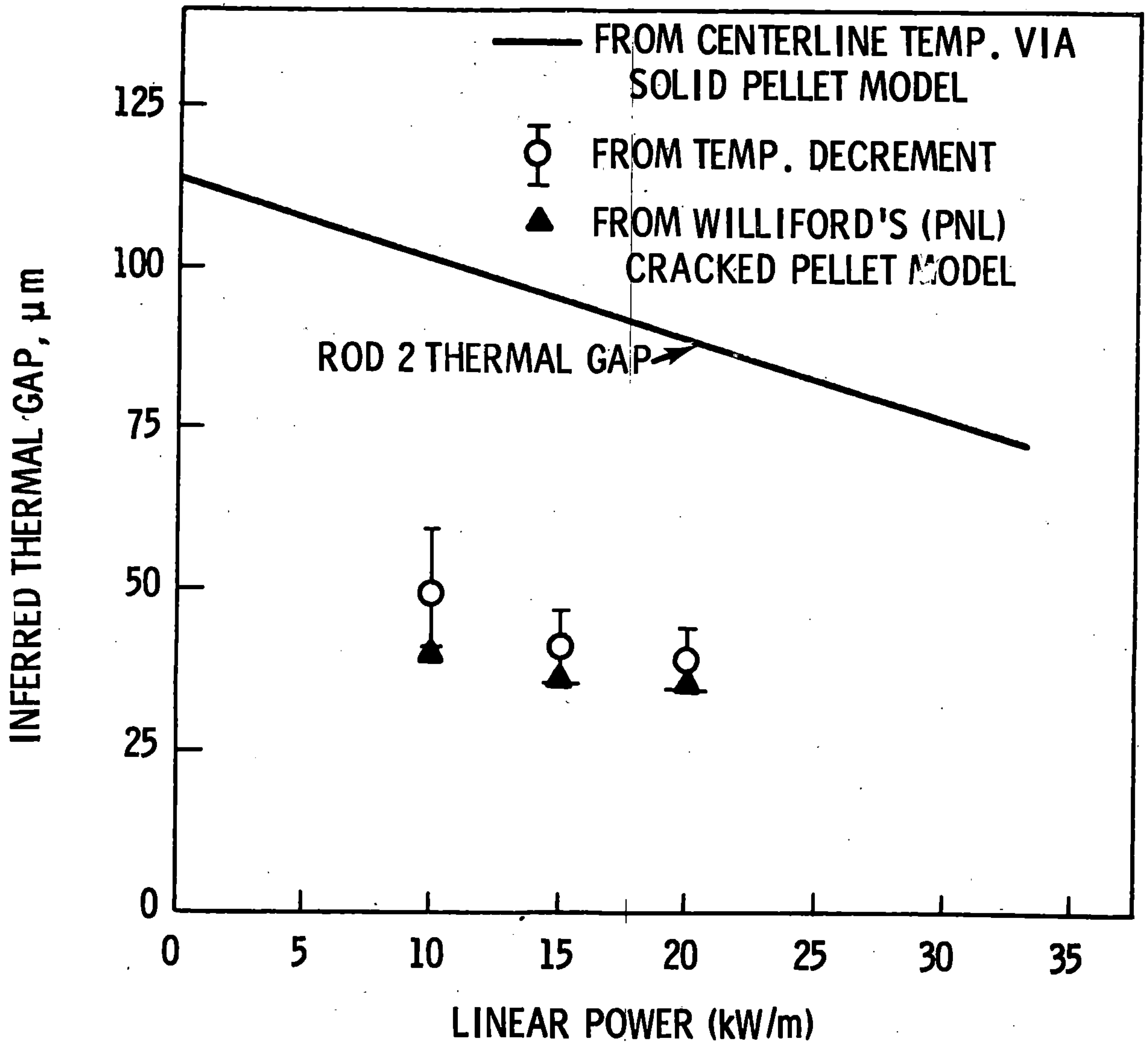

FIGURE 10. Inferred Gap Sizes from: 1) Thermal Expansion of Solid Pellet; 2) Pressure Effects on Fuel Temperature; and 3) Williford's Cracked-Pellet Model (Reference 9)

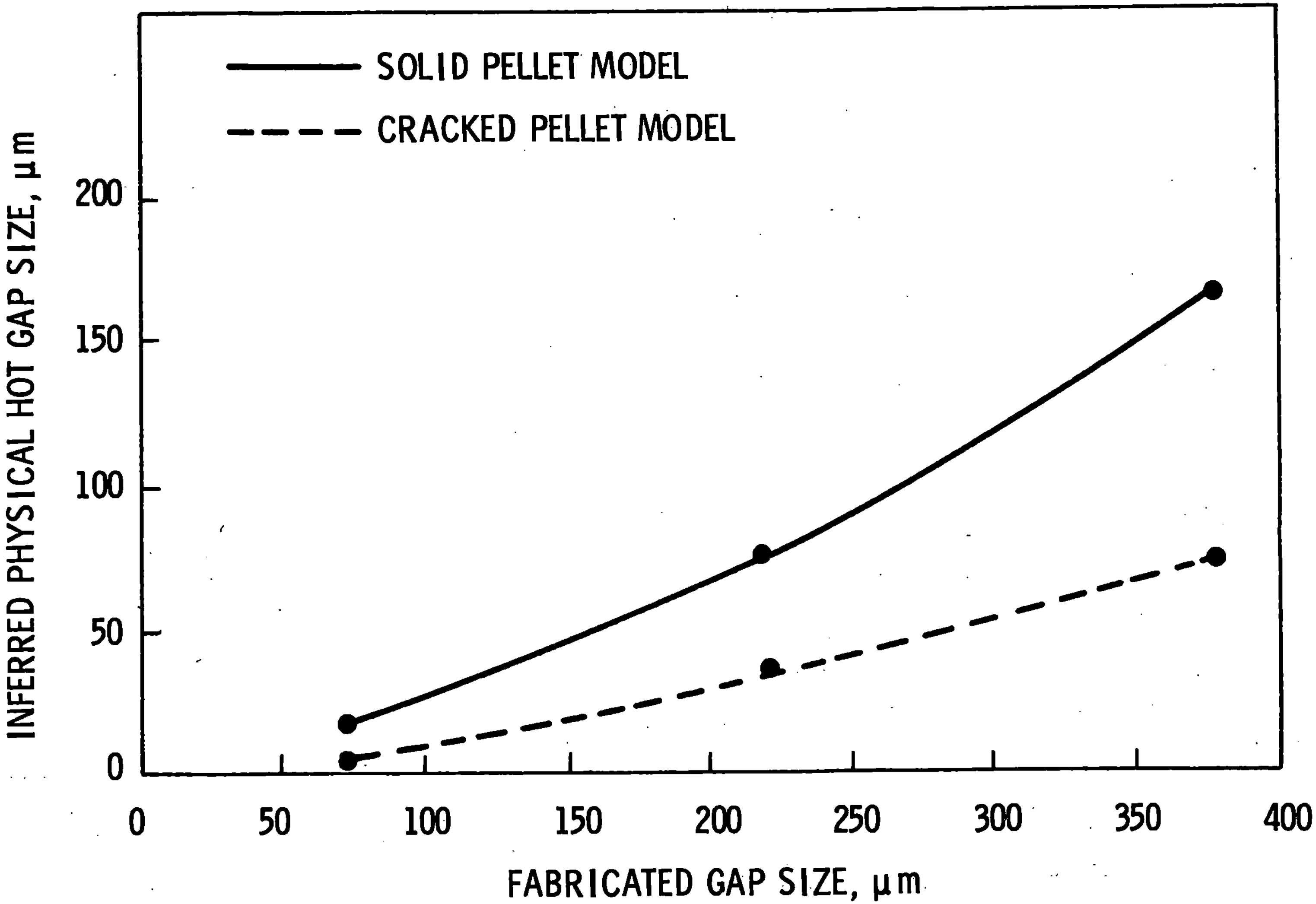

FIGURE 11. Inferred Gap Size at 35 kW/m at Beginning-of-Life for IFA-432 Helium-Filled Rods

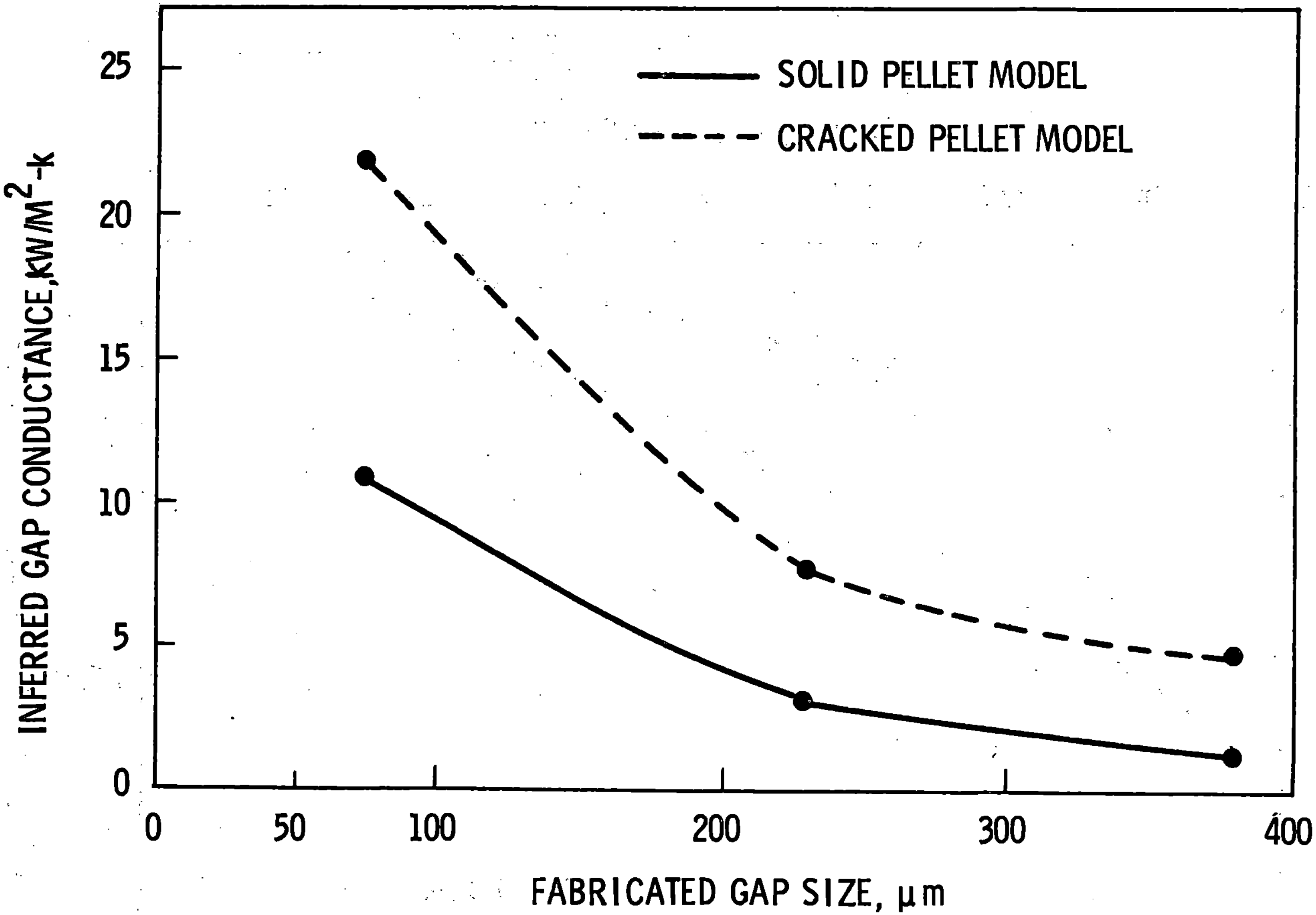

FIGURE 12. Inferred Gap Conductance at 35 kW/m for IFA-432 Helium-Filled Rods at Beginning-of-Life

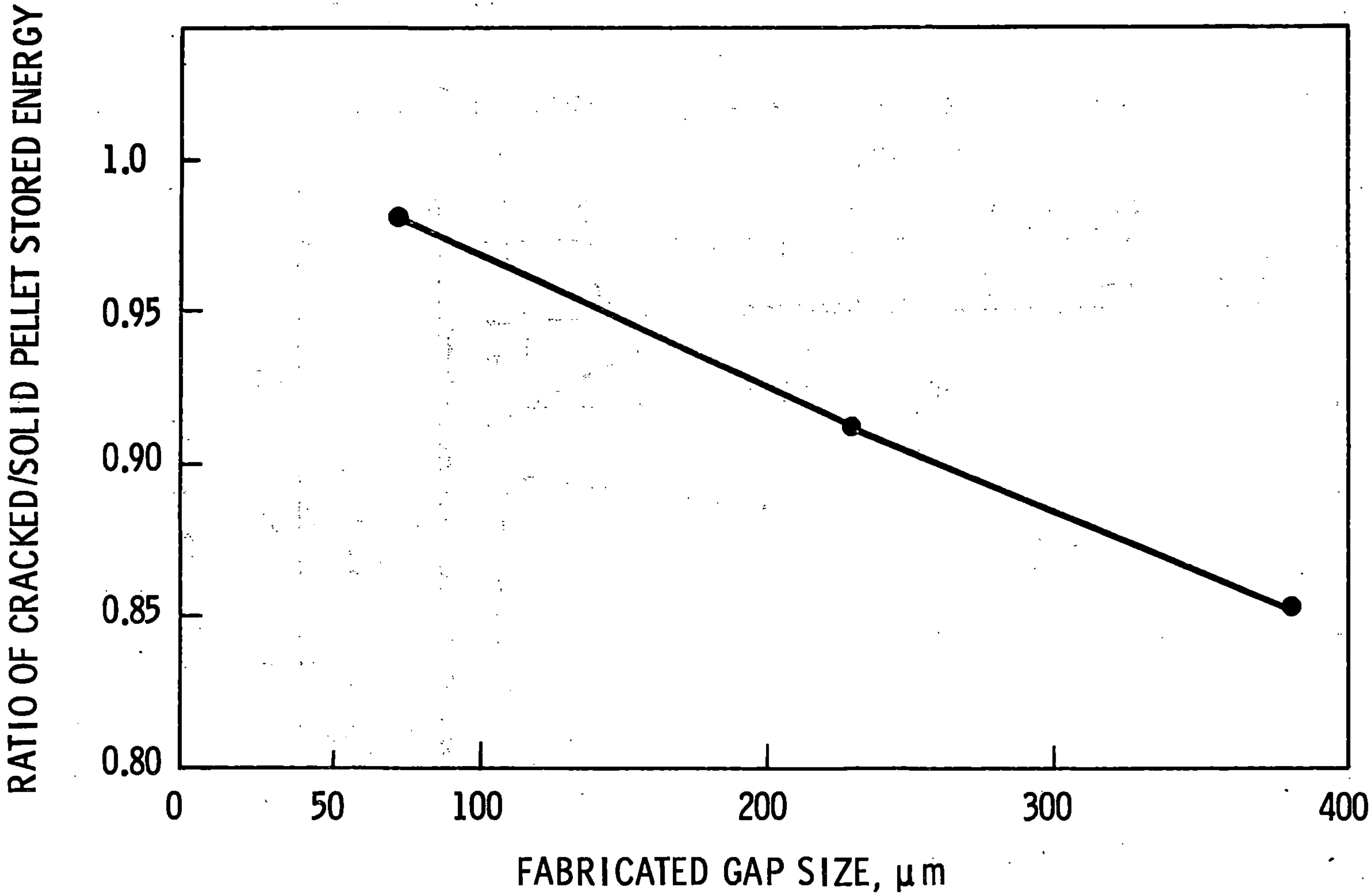

FIGURE 13. Ratio of Stored Energy Estimated by Cracked and Solid Pellet Models for IFA-432 Helium-Filled Rods at Beginning-of-Life

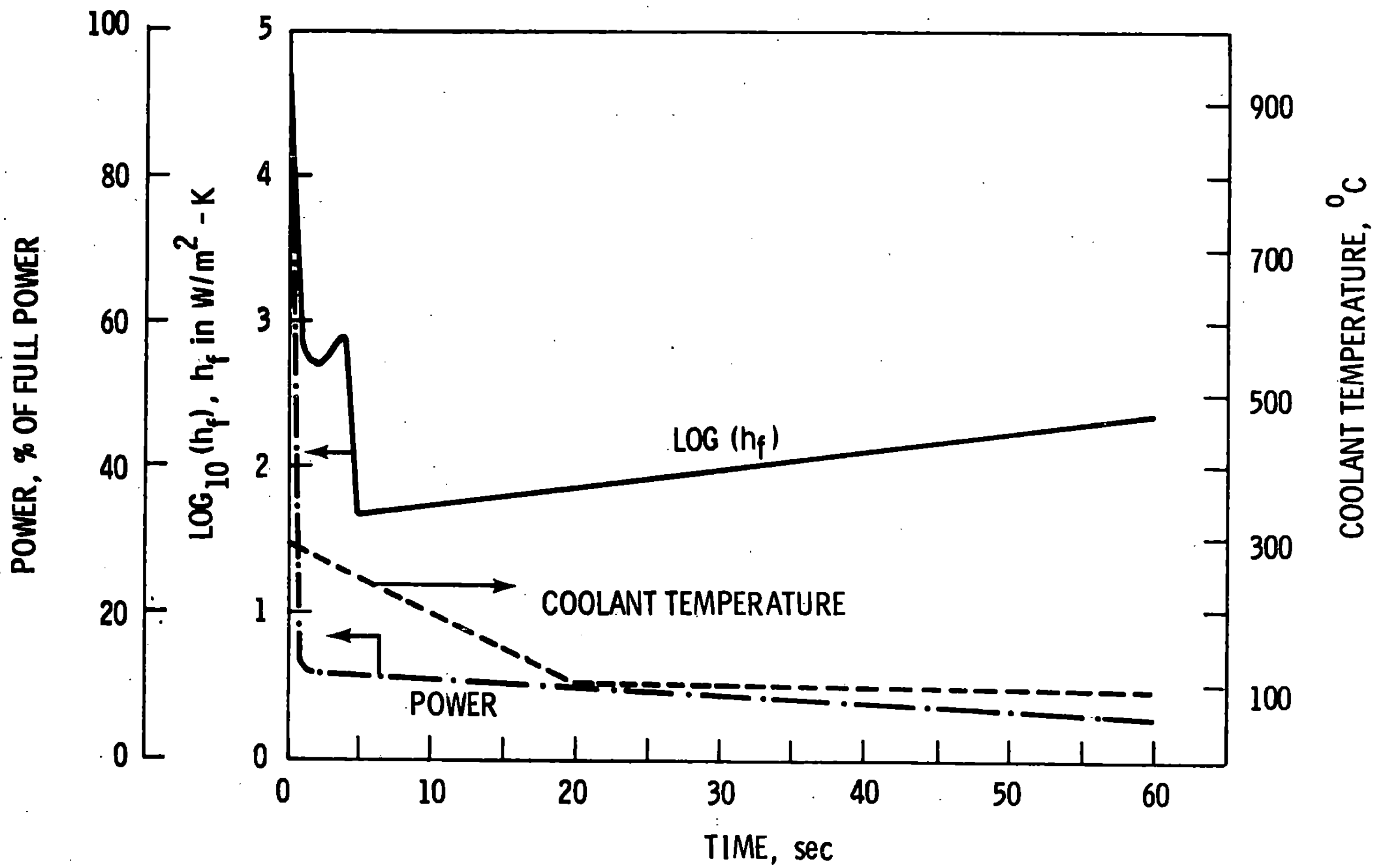

FIGURE 14. Input for Transient Cladding Temperature Calculation

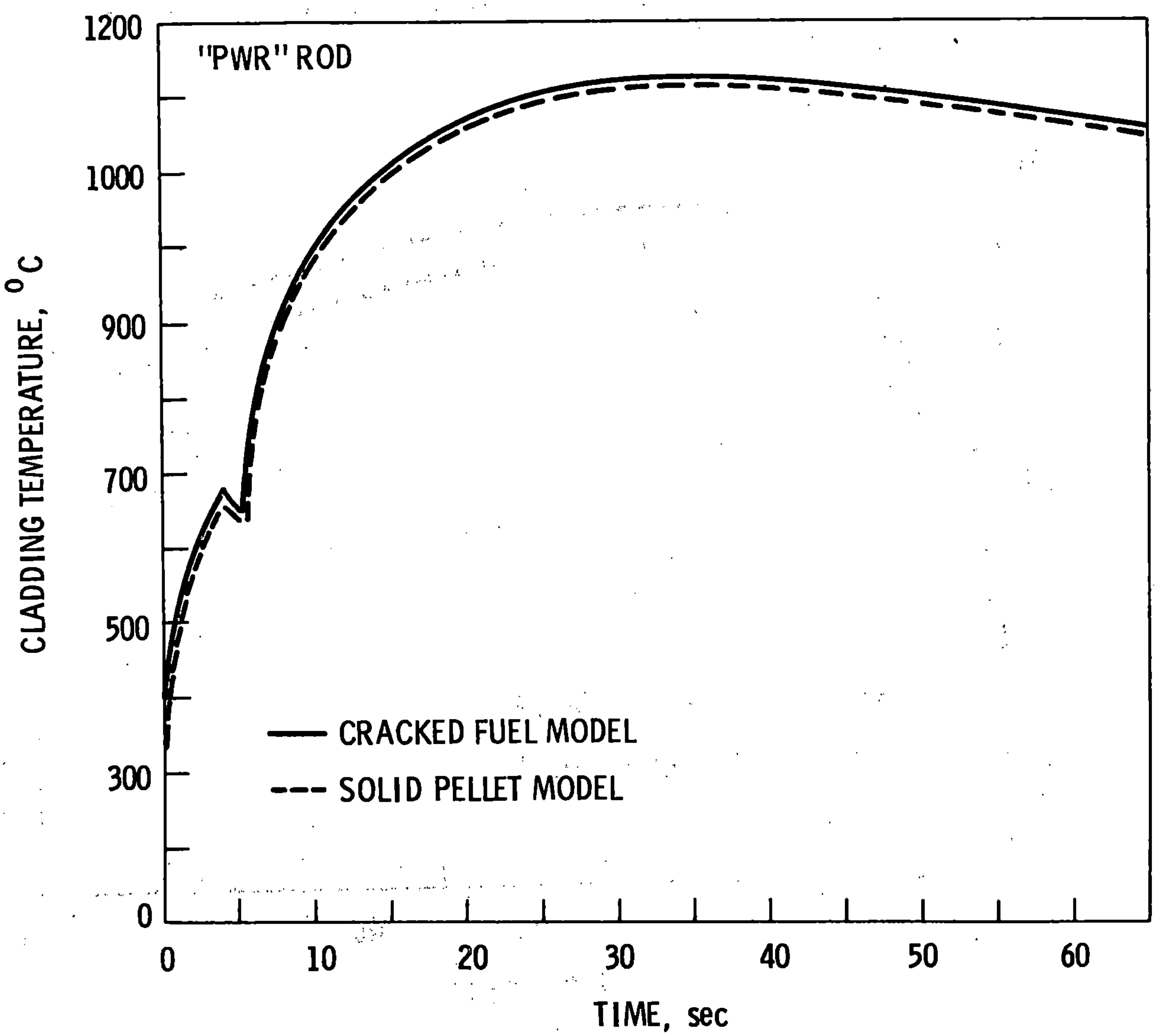

FIGURE 15. Effect of Model Assumptions on Peak Cladding Temperature for a PWR (Helium-Filled) Rod

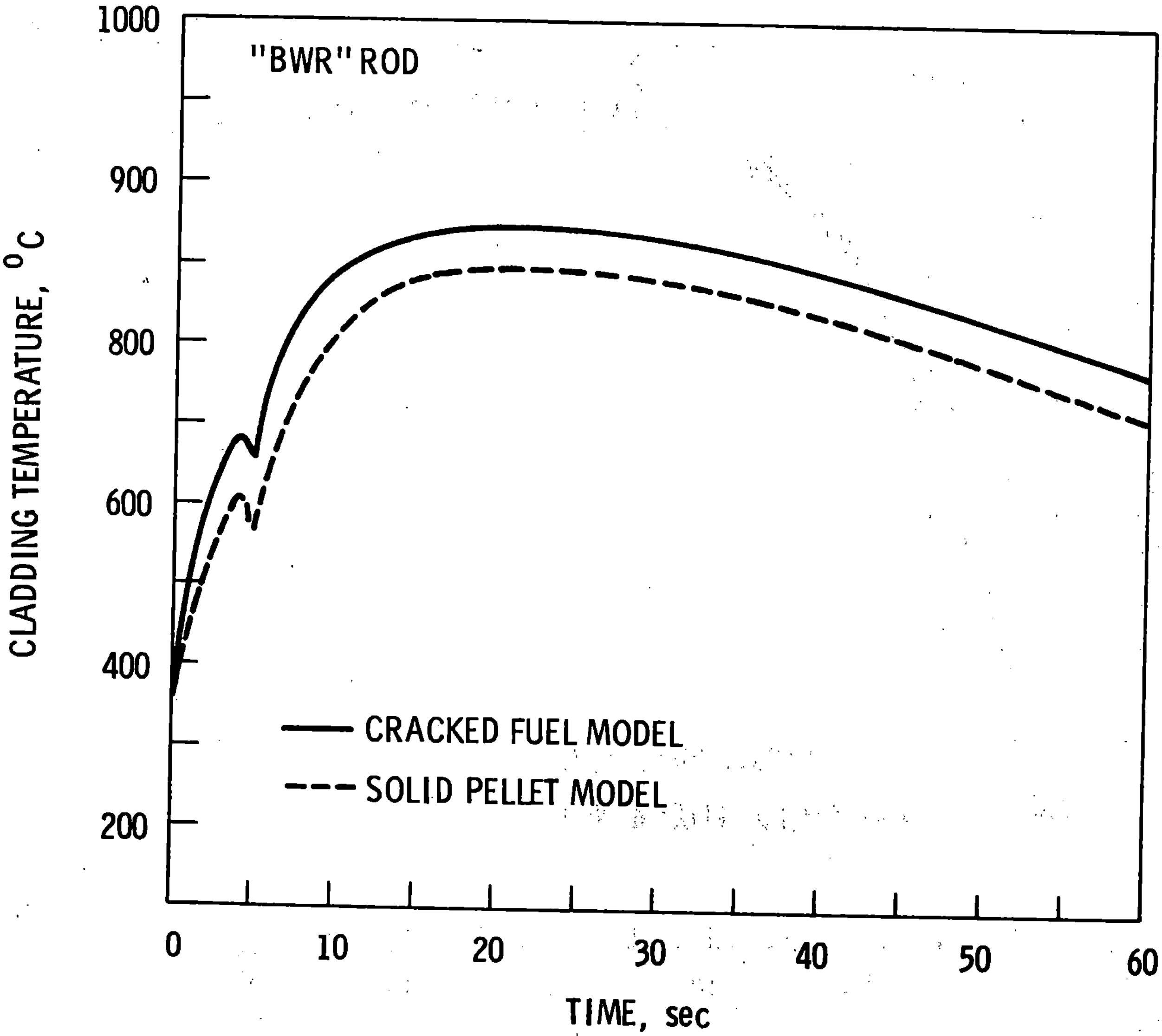

FIGURE 16. Effect of Model Assumptions on Peak Cladding Temperatures for a "BWR" Rod Filled with Fission Gas

APPENDIX A

Experimental Matrix of PNL/NRC Haden Tests

EXPERIMENTAL MATRIX

Assembly &Rod	Power (KW/m)	Fuel Type	Diametral Gap 2-3 mil	9 mil	15 mil	Initial Fill Gas 100%He 1 atm	100%He >1atm	%Xe 1 atm
IFA-431	35/25							
1		95S		X		X		
2		"			X	X		
3		"	X			X		
4		"		X				100%
5		92S		X		X		
6		92U		X		X		
IFA-432	50/35							
1		95S		X		X		
2		"			X	X		
3		"	X			X		
4		"		X				100%
5		92S		X		X		
6		92U		X		X		
IFA-513	40/28							
1		95S		X		X		
2		"		X			3 atm	
3		"		X		X		
4		"		X				8%
5		"		X		X		
6		"		X				23%
IFA-527	14/10							
1		95S		X				100%
2		"		X				"
3		"		X				"
4		"		X				"
5		"		X				"
6		"	X					"

MULTIROD BURST TEST PROGRAM STATUS REPORT*

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Presented at the
Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

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MULTIROD BURST TEST PROGRAM STATUS REPORT

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The Multirod Burst Test (MRBT) Program, in progress at Oak Ridge National Laboratory, is investigating LWR cladding deformation in single- and multi-rod test arrays under conditions representative of refill and reflood phases of a large break LOCA. This research addresses pertinent licensing issues related to the requirements for ECCS Acceptance Criteria by providing test data for development and verification of deformation models in computer codes. The data are also used by the Nuclear Regulatory Commission to generate audit models for assessment and approval of licensing applications.

In these tests internally pressurized, unirradiated Zircaloy-4 tubes containing electrically heated fuel simulators are tested to failure in a low-pressure, superheated-steam environment. The tubes are "uniformly" heated over a 915-mm length; the simulator pressure, due to the small enclosed gas volume, also varies with temperature (and deformation) during the test. The tests are conducted at very low steam flow rates ($\sim$300 $<Re \sim <$800) under either constant power or constant heating rate modes of control, as desired.

Approximately 100 single rod tests have been performed to date, covering a burst pressure range from 0.77 to 19.2 MPa; the corresponding burst temperatures range from 1170 to 690°C. Heating rates ranging from 0 to 28 K/s have been used. The shroud surrounding the test simulator was unheated in approximately two-thirds of the tests (i.e., in the early phase of the program) and heated to the same temperature as the simulator in the remainder of the tests.

Two 4 X 4 multirod tests (B-1 and B-2), one with and one without shroud electrical heating, have been conducted with a bundle heating rate of $\sim$29 K/s; initial pressure conditions for these tests were selected to cause failure at about 860°C. An additional 4 X 4 array (B-3) was tested using a bundle heating rate of $\sim$10 K/s; the shroud was also electrically heated in this test. Initial conditions were adjusted to cause failure at $\sim$760°C. A 8 X 8 array (B-5) was recently tested under B-3 test conditions to determine the effect of array size on deformation. The B-5 shroud (unheated) was spaced one-half of a coolant channel thickness away from the outer ring of simulators to provide lateral constraint typical of a fuel rod bundle (Fig. 1).

Posttest examination (including flow tests) of the three 4 X 4 bundles has been completed and reported. Some quick-look results of the 8 X 8 array are included in this presentation. Flow tests on this bundle are now in progress; detailed deformation data will not be available until late next summer.

Deformation in the 4 X 4 bundle tests was greater than would have been expected from the single rod unheated shroud tests, due to temperature uniformity effects. It was shown subsequently that deformation in single rod heated shroud tests is comparable to that in relatively unconstrained bundles tested under the same conditions. Preliminary data indicate greater deformation in the 8 X 8 bundle than in the comparable 4 X 4 test. These observations accentuate the importance of both thermal and mechanical boundary conditions in single rod and small bundle tests.

A correlation relating burst temperature to pressure and heating rate was derived from the single rod unheated shroud tests. The recent heated shroud test data tend to be underpredicted by the correlation, particularly for heating rates of 10 and 28 K/s. This tendency is believed to be related to local strain rate effects.

Current test plans include approximately 12 additional single rod heated shroud tests to explore the effect of this important parameter in the beta temperature range and to provide additional data for improving the predictive capability of the burst temperature correlation. Two additional bundle tests are planned. The first of these, a 6 X 6 array (B-4) similar in design to the 8 X 8 (B-5) array, will be performed next February and will explore the effects of unheated rods in a constrained bundle tested under conditions known to produce large deformation, i.e., at ∿800°C with a heating rate of 5 K/s. The second, a 4 X 4 array (B-6) with a heated shroud, will be tested in September of next year. Test conditions of 900°C and 10 K/s have been tentatively selected for the test to define bundle deformation in the two-phase region. Flow tests are not planned for these tests.

Work on this program is routinely reported in a series of reports entitled *Multirod Burst Test Program Progress Report*. Significant results appear in the following reports of this series:

NUREG Report No.	ORNL Report No.	Period Covered
	ORNL/NUREG/TM-36	January-March 1976
	ORNL/NUREG/TM-74	April-June 1976
	ORNL/NUREG/TM-77	July-September 1976
	ORNL/NUREG/TM-95	October-December 1976
	ORNL/NUREG/TM-108	January-March 1977
	ORNL/NUREG/TM-135	April-June 1977
NUREG/CR-0103	ORNL/NUREG/TM-200	July-December 1977
NUREG/CR-0225	ORNL/NUREG/TM-217	January-March 1978
NUREG/CR-0398	ORNL/NUREG/TM-243	April-June 1978
NUREG/CR-0655	ORNL/NUREG/TM-297	July-December 1978
NUREG/CR-0817	ORNL/NUREG/TM-323	January-March 1979
NUREG/CR-1023	ORNL/NUREG/TM-351	April-June 1979
NUREG/CR-1450	ORNL/NUREG/TM-392	July-December 1979

Published topical reports and papers of interest include:

1. R. H. Chapman, et al., Effect of Creep Time and Heating Rate on Deformation of Zircaloy-4 Tubes Tested in Steam with Internal Heaters, ORNL/NUREG/TM-245, October 1978.

2. J. F. Mincey, Steady-State Axial Pressure Losses Along the Exterior of Deformed Fuel Cladding: Multirod Burst Test (MRBT) Bundles B-1 and B-2, NUREG/CR-1011 (ORNL/NUREG/TM-350), January 1980.

3. R. H. Chapman, et al., Zircaloy Cladding Deformation in a Steam Environment with Transient Heating, in Proceedings of Fourth International Conference on Zirconium in the Nuclear Industry, held June 26-29, 1978, at Stratford-on-Avon, England, ASTM STP 681 (1979).

The following limited-distribution data reports have been published:

1. R. H. Chapman, et al., Bundle B-1 Test Data, ORNL/NUREG/TM-322, June 1979.

2. R. H. Chapman, et al., Bundle B-2 Test Data, ORNL/NUREG/TM-337, August 1979.

3. R. H. Chapman, et al., Bundle B-3 Test Data, ORNL/NUREG/TM-360, January 1980.

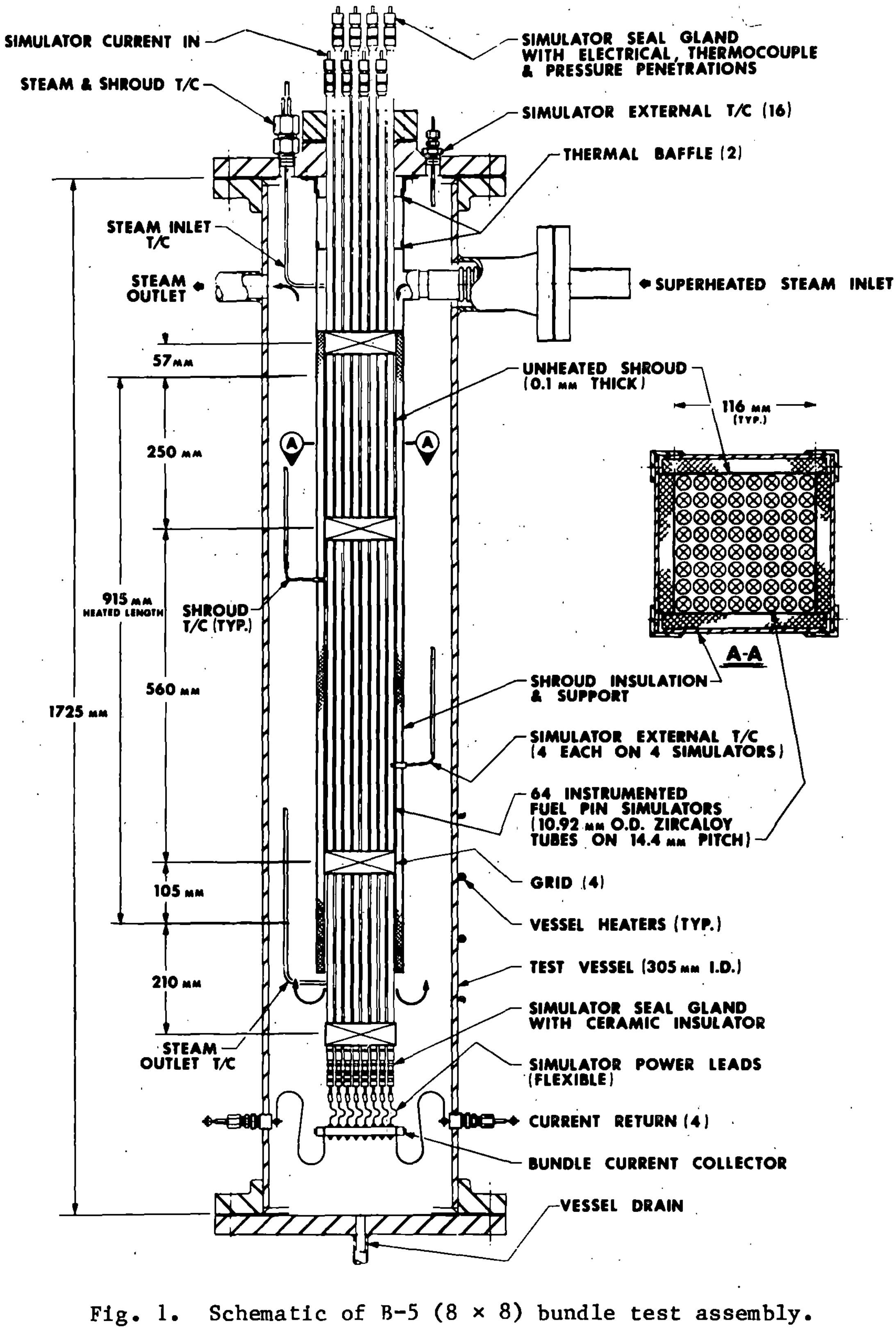

Fig. 1. Schematic of B-5 (8 × 8) bundle test assembly.

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R. H. CHAPMAN, MANAGER
MULTIROD BURST TEST PROGRAM
OAK RIDGE NATIONAL LABORATORY

MRBT STATUS REPORT

PRESENTED AT

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND
OCTOBER 27, 1980

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MULTIROD BURST TEST PROGRAM ADDRESSES LICENSING ISSUES RELATED TO ECCS ACCEPTANCE CRITERIA

- CRITERIA ARE SIGNIFICANTLY INFLUENCED BY DEGREE OF SWELLING AND NUMBER OF TUBE RUPTURES
- COMPLIANCE IS DETERMINED BY EVALUATION MODELS THAT MUST NOT UNDERESTIMATE SWELLING AND TUBE RUPTURES
- LICENSING ISSUES
  - APPLICABLE DATA
  - VALIDITY OF EVALUATION MODELS
  - MAINTENANCE OF A COOLABLE GEOMETRY

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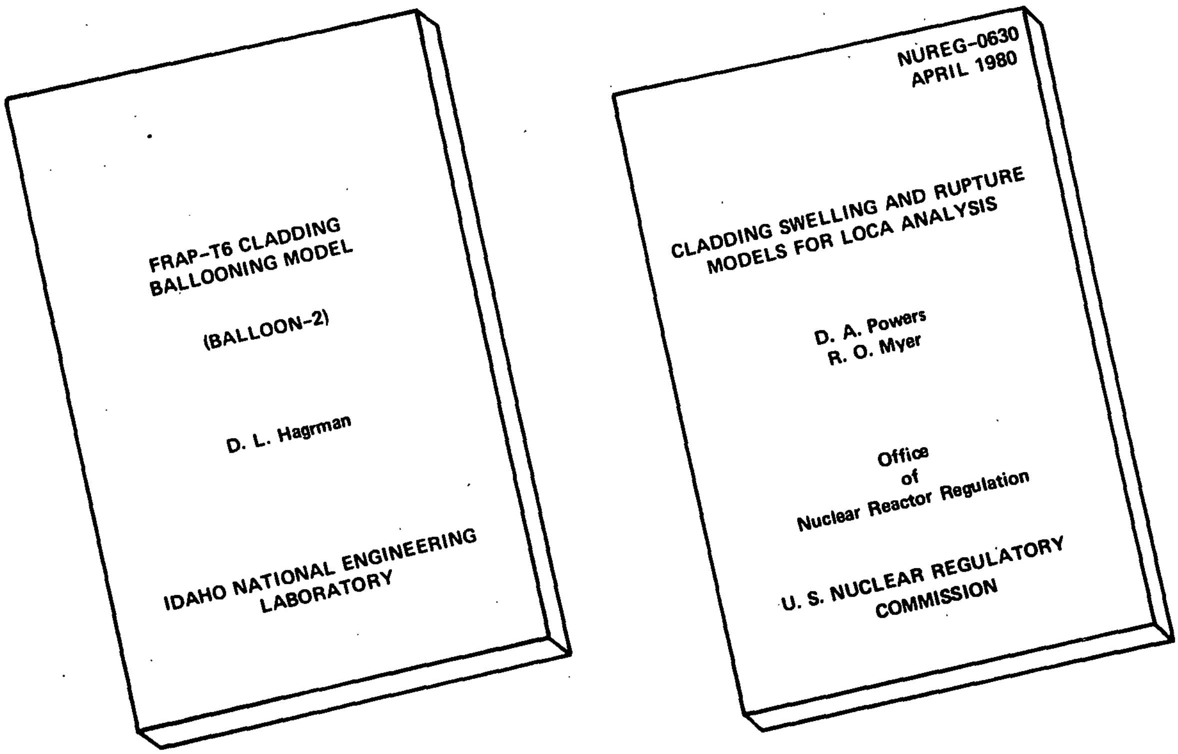
MRBT PROGRAM ADDRESSES THESE ISSUES BY PROVIDING DATA FOR DEVELOPMENT OF DEFORMATION MODELS AND NRC AUDIT MODELS FOR LOCA ANALYSIS



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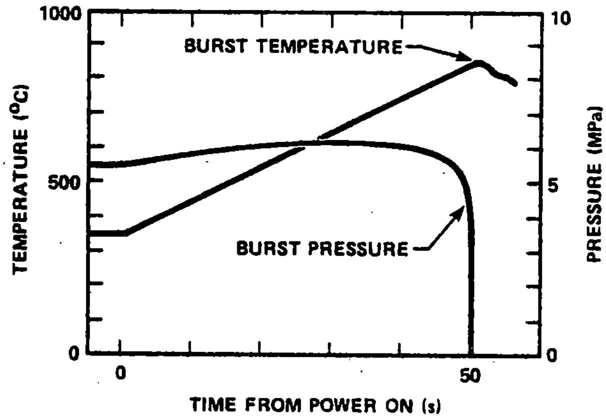
MULTIROD BURST TEST PROGRAM IS A STUDY OF CLADDING DEFORMATION UNDER CONDITIONS REPRESENTATIVE OF REFILL AND REFLOOD PHASES OF A LOSS-OF-COOLANT ACCIDENT

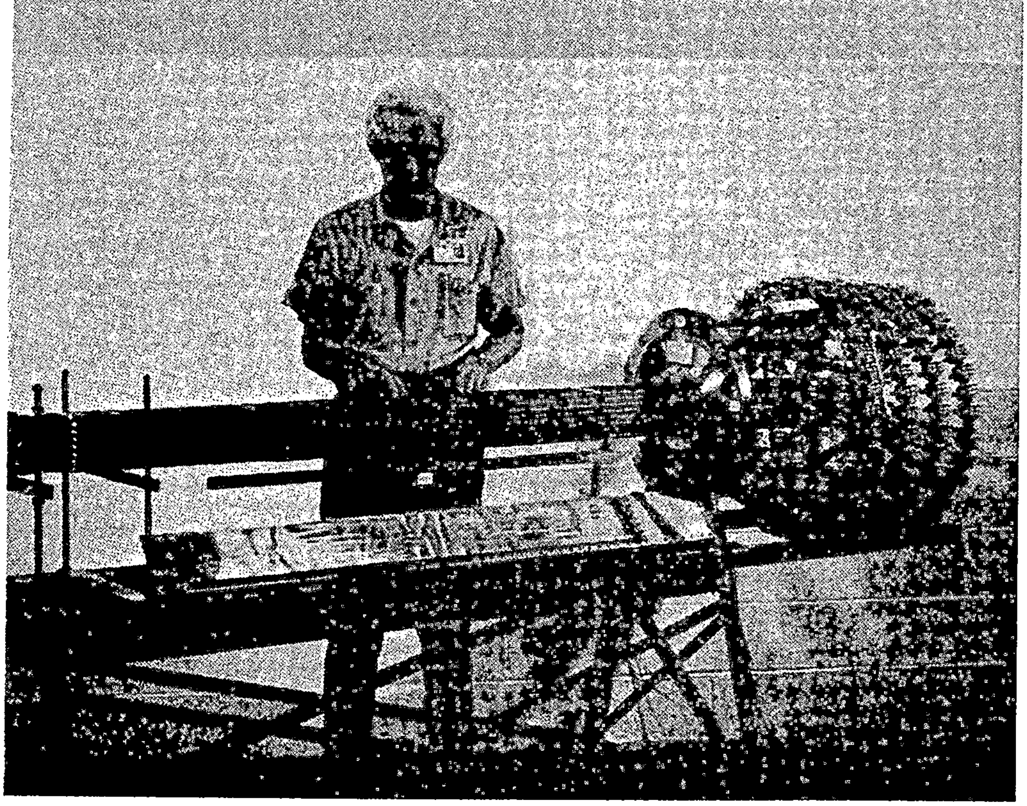
- ELECTRICALLY HEATED FUEL ROD SIMULATORS IN LOW PRESSURE SUPERHEATED STEAM ENVIRONMENT
- SINGLE ROD TESTS FOR PARAMETRIC EFFECTS
- MULTIROD TESTS FOR ROD-TO-ROD INTERACTIONS AND FLOW RESISTANCE EFFECTS

RESULTS ARE OF IMPORTANCE IN DETERMINING IF CORE GEOMETRY REMAINS AMENABLE TO COOLING

BURST TESTS INVOLVE BOTH TEMPERATURE AND PRESSURE TRANSIENTS



BUNDLE TESTS ARE NECESSARY TO REPRESENT THERMAL AND MECHANICAL BOUNDARIES OF REACTOR FUEL ASSEMBLY



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A LARGE AND RELIABLE DATA BASE IS BEING PRODUCED

TESTS COMPLETED

- 65 SINGLE ROD UNHEATED SHROUD TESTS
- 33 SINGLE ROD HEATED SHROUD TESTS
- 2 4 X 4 BUNDLE TESTS WITH HEATED SHROUDS
- 1 4 X 4 BUNDLE TEST WITH UNHEATED SHROUD
- 1 8 X 8 BUNDLE TEST WITH UNHEATED SHROUD

TESTS PLANNED

- 12 SINGLE ROD HEATED SHROUD TESTS
- 1 6 X 6 BUNDLE TEST WITH UNHEATED SHROUD
- 1 4 X 4 BUNDLE TEST WITH HEATED SHROUD

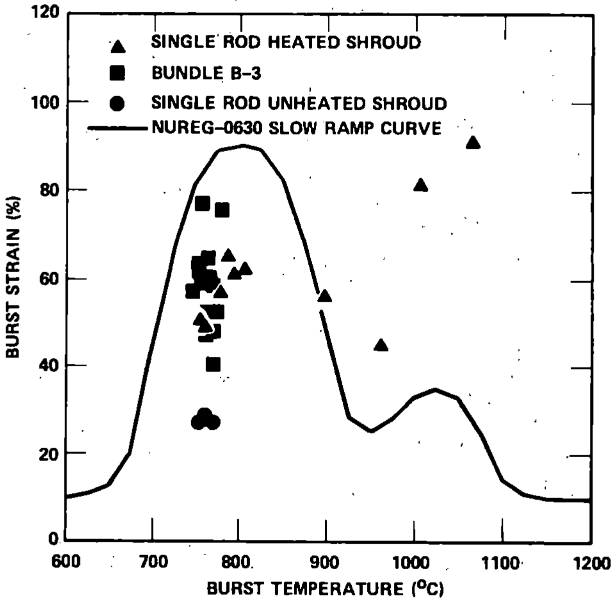
UNION CARBIDE ORNL

DEFORMATION CHARACTERIZATION ALMOST COMPLETE

- DEVELOPED BURST TEMPERATURE CORRELATION
- DEFORMATION DEPENDENT ON TEMPERATURE GRADIENTS
- DEFORMATION IN SINGLE ROD UNHEATED SHROUD TESTS LESS THAN IN BUNDLE TESTS
- DEFORMATION IN SINGLE ROD HEATED SHROUD TESTS COMPARABLE TO BUNDLE TESTS WITHOUT INTERACTIONS
- MRBT SINGLE ROD HEATED SHROUD DATA AND RECENT KFK DATA AND MODEL ARE IN SUBSTANTIAL AGREEMENT
- DEFORMATION IN 4 X 4 BUNDLES PROBABLY NOT REPRESENTATIVE OF LARGE BUNDLES WITH RESPECT TO ROD-TO-ROD INTERACTIONS
- COBRA-IV CORRECTLY PREDICTS MEASURED PRESSURE LOSSES USING GEOMETRIC INPUT DATA FROM DEFORMED BUNDLE

UNION CARBIDE ORNL

SINGLE ROD HEATED SHROUD BURST DATA ARE CONSISTENT WITH BUNDLE DATA FOR 10 K/s TEST CONDITIONS



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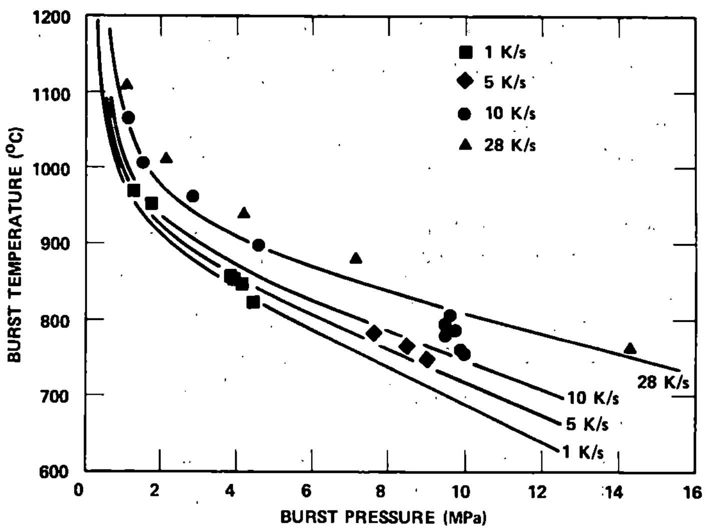
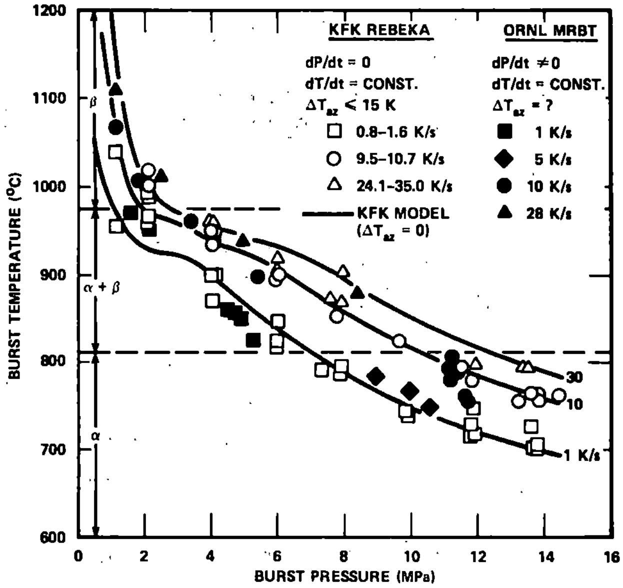
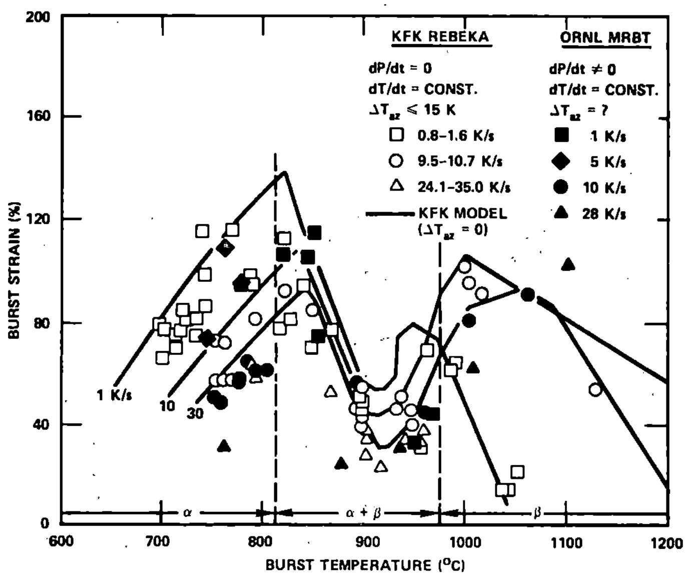
LOCAL STRAIN RATE EFFECTS BELIEVED CAUSE FOR UNDERPREDICTION OF HEATED SHROUD DATA BY CORRELATION DERIVED FROM UNHEATED SHROUD DATA



ORNL SINGLE ROD HEATED SHROUD TEST DATA CORRECTED FOR TUBE SIZE AND RECENT KFK RUPTURE DATA AND MODEL ARE IN AGREEMENT FOR SIMILAR TEST CONDITIONS



ORNL SINGLE ROD HEATED SHROUD TEST DATA AND RECENT KFK STRAIN DATA AND MODEL ARE IN AGREEMENT FOR SIMILAR TEST CONDITIONS



B-5 (8 X 8) BUNDLE TEST CRITERIA

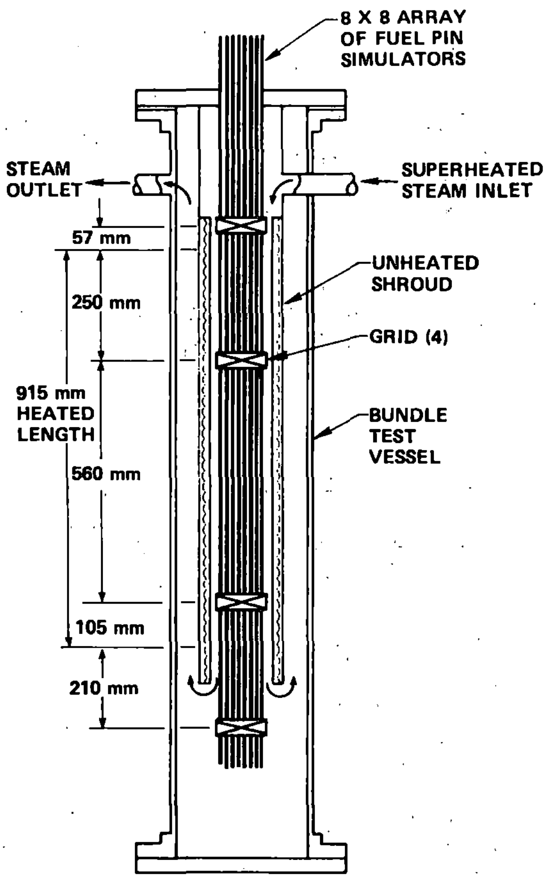
OBJECTIVE

- TEST CONDITIONS SAME AS B-3 (4 X 4) TO DETERMINE EFFECTS OF ARRAY SIZE ON DEFORMATION

FEATURES

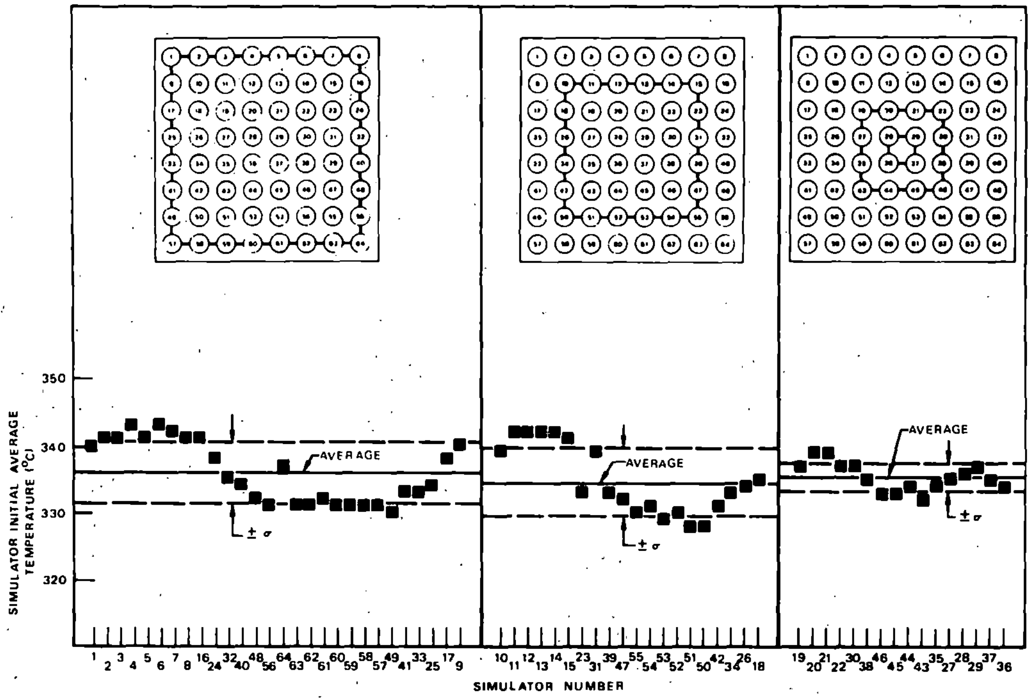
- TEST AT ~770°C WITH 10 K/s HEATING RATE
- ALL SIMULATORS PRESSURIZED AND HEATED
- OUTER RING OF SIMULATORS RESTRAINED BY SHROUD
- HIGHLY REFLECTIVE, UNHEATED SHROUD
- FLOW TEST REFERENCE AND DEFORMED BUNDLES

SCHEMATIC OF 8 X 8 BUNDLE TEST



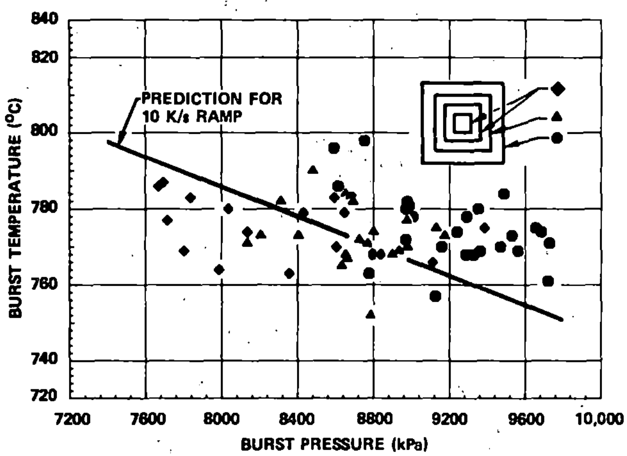
UNION CARBIDE ORNL

B-5 INITIAL TEMPERATURE DISTRIBUTION SHOWS NORTH-TO-SOUTH TEMPERATURE GRADIENT



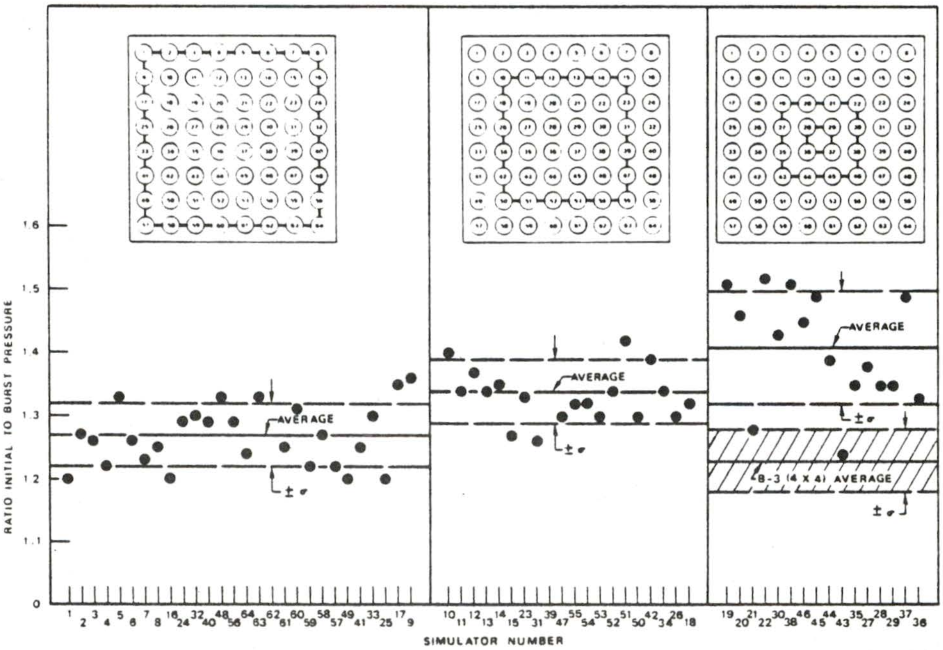
UNION CARBIDE ORNL

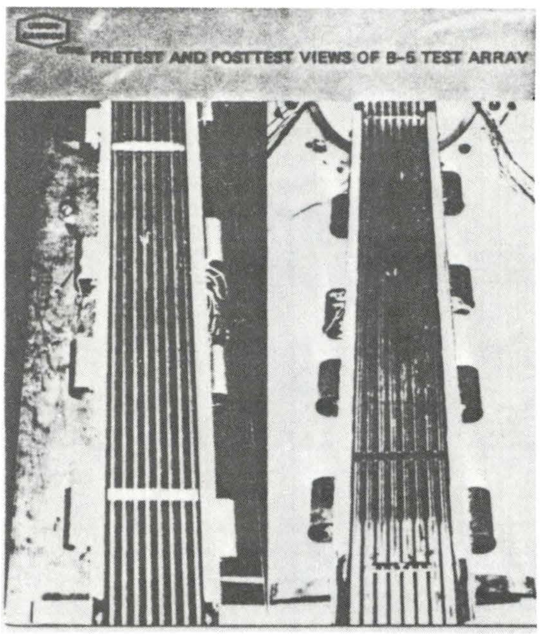
COMPARISON OF B-5 EXPERIMENTAL BURST DATA BY ZONES WITH CORRELATION PREDICTION

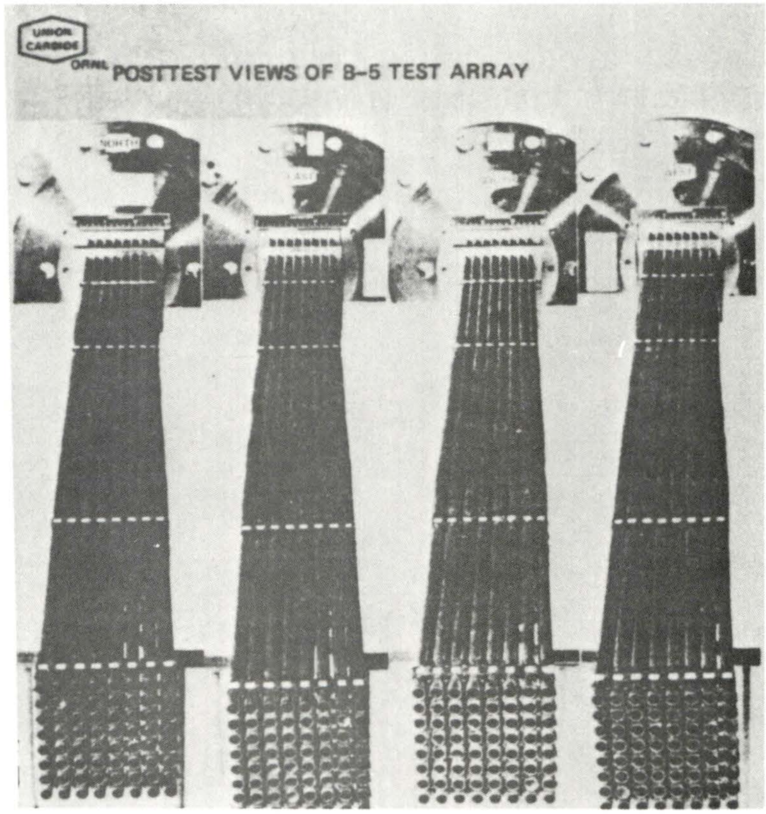


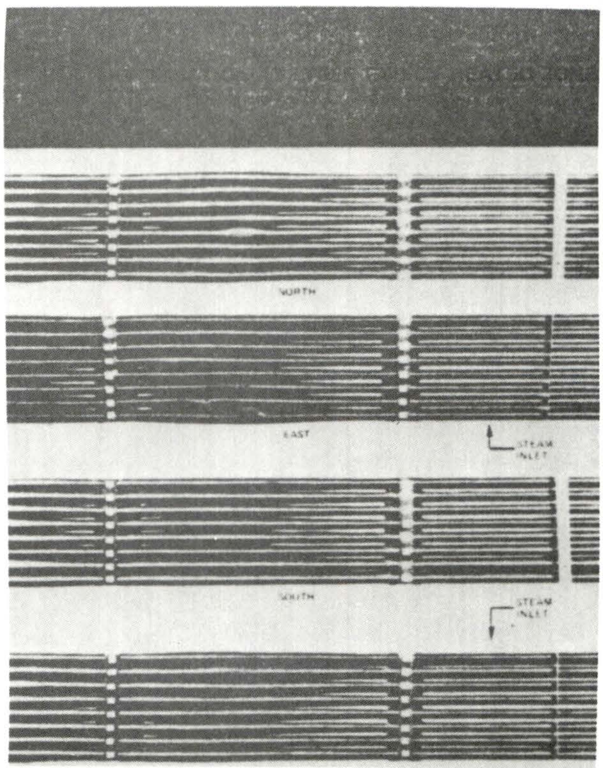
UNION CARBIDE ORNL

DEFORMATION GREATER IN B-5 INTERIOR SIMULATORS THAN IN EXTERIOR SIMULATORS AND GREATER THAN IN B-3 (4 X 4) TEST



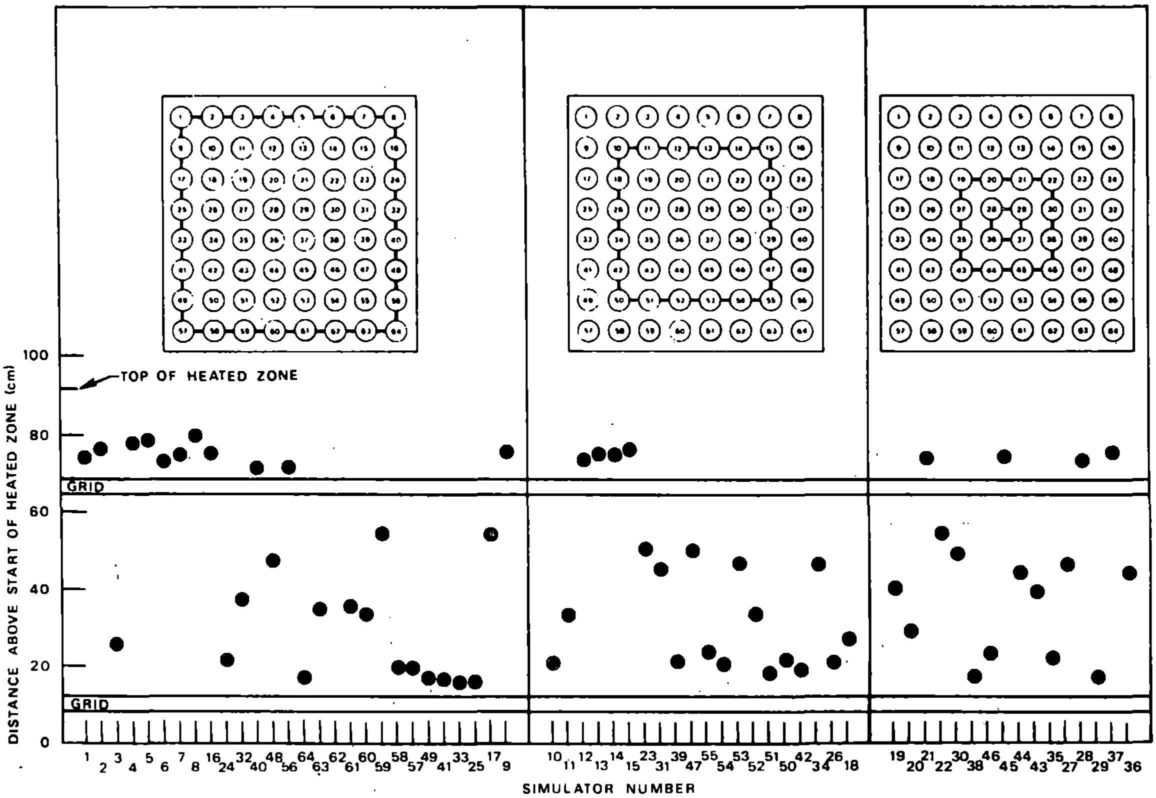



UNION CARBIDE
ORNL
POSTTEST VIEWS OF B-5 TEST ARRAY
NORTH


NORTH
EAST
STEAM INLET
SOUTH
STEAM INLET

UNION CARBIDE
ORNL

BURST ELEVATIONS IN B-5 TEST REFLECT INITIAL NORTH-TO-SOUTH TEMPERATURE GRADIENT



CLAD EMBRITTLEMENT CRITERIA--APPLICATION TO AN ASSESSMENT OF THE MARGIN OF PERFORMANCE OF ECCSs IN LWRs*

by

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FOR PRESENTATION AT THE

EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

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Gaithersburg, MD

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SUMMARY

Evaluation models, which compute the time-temperature conditions and the extent of oxidation of the fuel cladding during postulated loss-of-coolant-accident (LOCA) transients, are used to assess the margin of performance of emergency core cooling systems (ECCSs) in light-water reactors (LWRs). During the blowdown stage of a LOCA transient, ballooning and rupture of the internally pressurized cladding, accompanied by wall thinning of the tube, can occur. The thickness of the cladding after deformation and rupture in steam is an important parameter, in addition to the time and temperature, in evaluations of maximum permissible oxidation to prevent fracture of the cladding by thermal shock and impact loads during the later stages of the transient and to ensure capability for fuel handling and transport of the oxidized assemblies.

To assess the extent of embrittlement of the cladding during a LOCA transient, one must evaluate the circumferential and radial (thickness) strains along the length of the rod, i.e., the strains in the rupture location as well as in regions of lesser deformation which encompass most of the overall length. Multirod tube-burst results from ORNL, JAERI, and KfK suggest that the maximum circumferential strain and the average rod strain in the plane of maximum blockage are linearly related for reductions in flow area in the bundle to $\sim$ 90%.[1] Since the maximum circumferential strains in the bundle tests are similar to those in ANL single-rod tests on pellet-constrained cladding[2] for the same heating rate and pressure differential, a relationship between the circumferential strain and the average wall thickness or wall-thickness ratio of the cladding from the latter experiments can be applied to Zircaloy deformation in multirod arrays. Consequently, when the initial heating rate and pressure differential or hoop stress on the cladding are specified, the maximum and average circumferential strains, the reduction in flow area, and the minimum and average wall-thickness ratios in the plane of maximum blockage can be obtained from curves based on single and multirod burst-test results (e.g., Figs. 1-3). However, the frequency distribution of the circumferential strain and wall-thickness ratio, depicted in Fig. 4 for ORNL multirod experiment B-3, will vary depending upon the thermal-hydraulic conditions in the fuel rod assembly during deformation.

Fuel-element parameters and the time-temperature transients in Figs. 5 and 6 for a double-ended guillotine break in the pump discharge leg of two PWRs (obtained from Final Safety Analysis Reports) were used to determine the minimum and average wall-thickness ratios after cladding deformation, i.e., $\sim$ 0.65 and 0.51 for regions of average ($\sim$ 38%) and maximum ($\sim$ 70%) circumferential strain, respectively, in Table 1. The influence of wall-thickness ratio on the

*Work supported by the Division of Water Reactor Safety Research, U. S. Nuclear Regulatory Commission.

equivalent-cladding-reacted (ECR) parameter and the thickness of the transformed β-phase layer ($L_{0.9}$) for the peak-temperature and rupture nodes is shown in Figs. 7-8 and 9-10 during two- and one-side oxidation of the cladding, respectively.

Table 2 shows that for an average circumferential strain of ∿ 38% and two-side oxidation of the cladding at the peak-temperature node, the ECR parameter and the β-layer-thickness values are 26% and 0.25 mm and 22.5% and 0.28 mm for the respective plants. For these conditions, the total oxidation exceeds the 17% ECR limit in the present acceptance criteria for ECCSs in LWRs; however, the β-layer thicknesses are considerably larger than the $L_{0.9} \geq 0.1$ mm value required for the cladding to survive thermal shock.[1] The β-layer thicknesses are somewhat smaller than the $L_{0.7} \geq 0.3$ mm criterion, which ensures that the cladding can withstand a 0.3 J impact energy at room temperature.[1] For the case of one-side oxidation at the peak-temperature node of the cladding, the total oxidation, as indicated by the ECR parameter, is ∿ 11 to 14% and the thickness of the transformed β-phase layer is ∿ 0.35 mm for the two transients. The relatively low temperature at the rupture node of the cladding in Fig. 6 leads to an ECR value of 4% total oxidation and a β-layer thickness of 0.3 mm, primarily due to wall thinning during ballooning deformation.

For the transients corresponding to the peak-temperature node, compliance with the oxidation limit in the present acceptance criteria for ECCSs, based upon this analysis, occurs for two-side oxidation of cladding with virtually no wall thinning due to deformation and for conditions in which the peak-temperature node of the cladding (with $\lesssim$ 60% circumferential strain) is located at a sufficient distance from the rupture region to justify one-side oxidation on the basis of limited steam access to the inner surface. One-side oxidation of the cladding is invoked in present licensing considerations for axial distances of $\gtrsim$ 38 mm on either side of the rupture opening. Relative to the proposed embrittlement criterion of $L_{0.9} \geq 0.1$ mm, the values for the margin of performance of the ECCSs in Table 2 indicate adequate resistance to thermal-shock failure. For these transients, fracture by thermal shock would occur after simultaneous oxidation of the inner and outer surfaces of cladding with an average wall-thickness ratio of $\lesssim$ 0.45 (circumferential strain of $\gtrsim$ 120%).

Recent results that quantify the oxidation kinetics, deformation behavior, and embrittlement characteristics of Zircaloy fuel cladding can be used to assess the margin of performance of ECCSs in a straightforward manner provided that (a) evaluation models properly account for the effect of flow area reduction on the time-temperature transient and (b) the applicability of one- or two-side oxidation of the cladding away from the rupture region is based on the deformation behavior and the thermal-hydraulic conditions during reflood.

<u>References and Recent Publications</u>

1. H. M. Chung and T. F. Kassner, "Embrittlement Criteria for Zircaloy Fuel Cladding Applicable to Accident Situations in Light-Water Reactors: Summary Report," NUREG/CR-1344, ANL-79-48, January 1980.

2. H. M. Chung and T. F. Kassner, "Deformation Characteristics of Zircaloy Cladding in Vacuum and Steam under Transient Heating Conditions: Summary Report," NUREG/CR-0344, ANL-77-31, July 1978.

3. A. M. Garde and T. F. Kassner, "Instrumented Impact Properties of Zircaloy-Oxygen and Zircaloy-Hydrogen Alloys," NUREG/CR-1408, ANL-80-14, April 1980.

ASSESSMENT OF ECCS MARGIN OF PERFORMANCE

I. FUEL ELEMENT PARAMETERS

- Zircaloy Cladding Dimensions
- Internal Pressure f (Burnup)
- Decay Heat f (Power Level)

II. CLAD DEFORMATION RELATIONSHIPS

- Hoop Stress vs Maximum Circumferential Strain
- Maximum vs Average Circumferential Strain
- Circumferential Strain vs Average Wall Thickness
- Average Wall Thickness vs Axial Position

ASSESSMENT OF ECCS MARGIN OF PERFORMANCE

III. ZIRCALOY CLADDING EMBRITTLEMENT CRITERIA

Present Criteria

- Peak Cladding Temperature of 1477 K
- Total Oxidation 17% of Wall Thickness

Proposed Criteria

- Thermal-Shock Failure Limit

  β-Layer Thickness $L_{0.9} \geq 0.1$ mm

- 0.3 J Impact Failure Limit

  β-Layer Thickness $L_{0.7} \geq 0.3$ mm

ASSESSMENT OF ECCS MARGIN OF PERFORMANCE

IV. LOCA OR PCM ACCIDENT SCENARIO

- Time-Temperature Transient for Cladding

  Peak Temperature Region
  Rupture Location
  Maximum Oxidation Node

- One- or Two-side Oxidation of Cladding

V. DEFINE MARGIN OF PERFORMANCE OF ECCS

- Compare Calculated Oxidation Characteristics of Zircaloy under Transient Conditions with

  Present (17% ECR, 1477 K)
  and
  Proposed ($L_{0.9} \geq 0.1$ mm, thermal shock)
  ($L_{0.7} \geq 0.3$ mm, impact)

  Cladding Failure Limits

ZIRCALOY DEFORMATION AND REDUCTION IN FLOW AREA CORRELATIONS BASED ON SINGLE AND MULTIROD BURST-TEST RESULTS FROM ANL, JAERI, KfK, AND ORNL

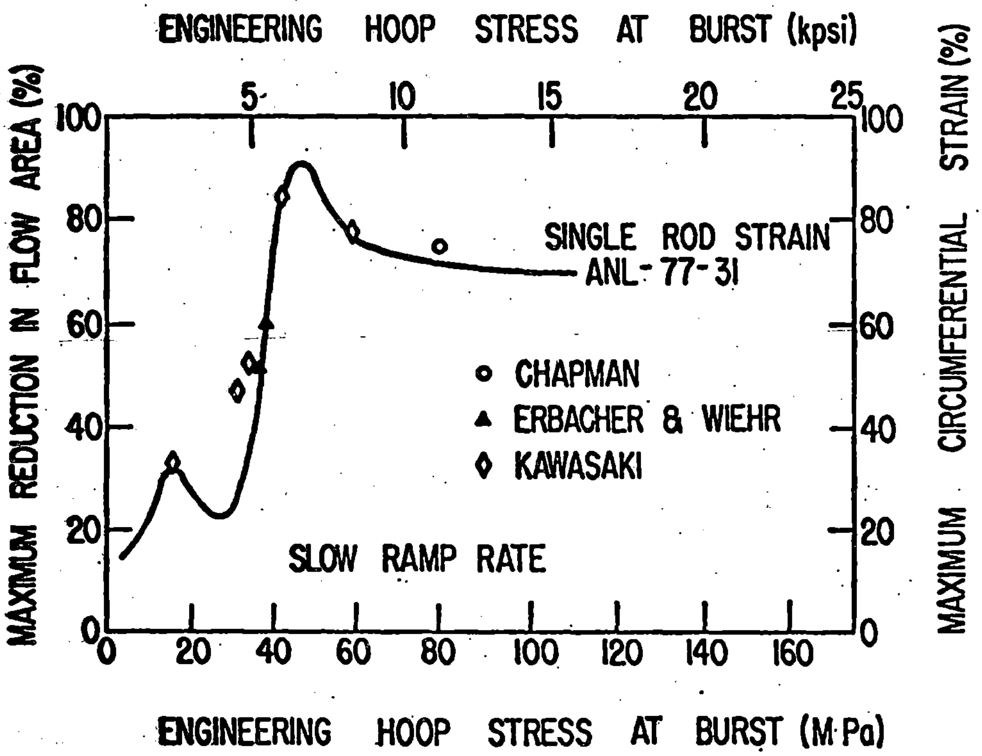

Fig. 1. Dependence of Maximum Reduction in Flow Area (from JAERI, KfK, and ORNL Bundle Tests) and Maximum Circumferential Strain (from ANL Single-rod Tests on Pellet-constrained Cladding) on the Engineering Hoop Stress at Rupture during Transient Heating at Low Ramp Rates ($\lesssim$ 10 K/s).

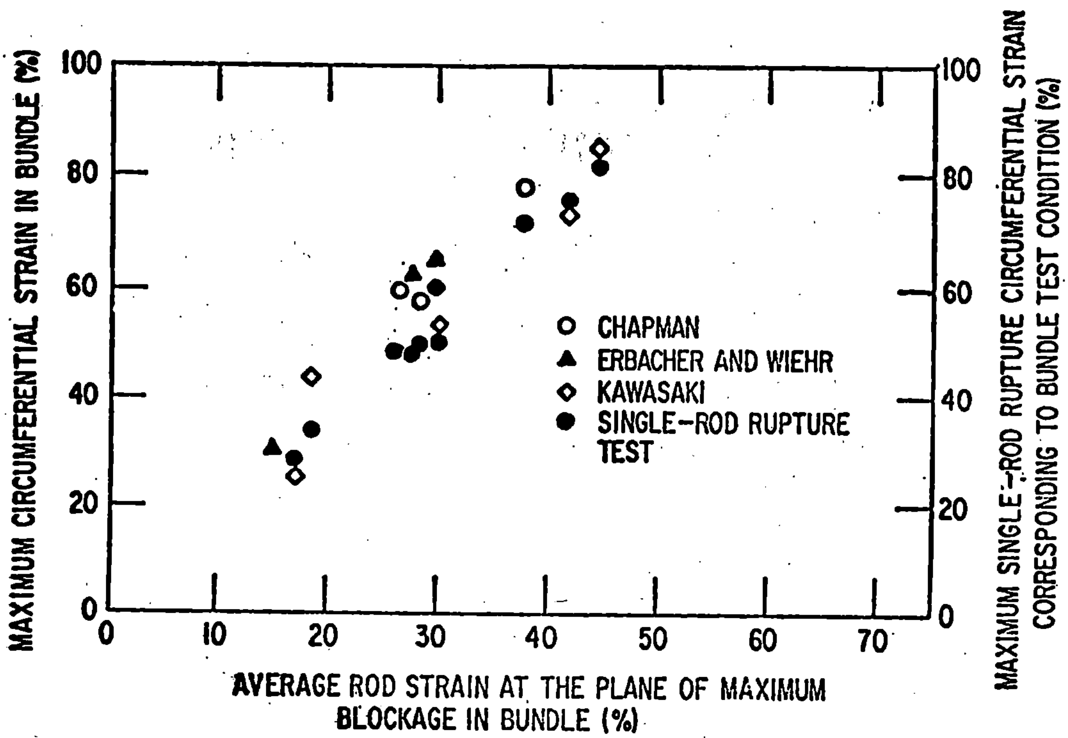

Fig. 2. Maximum Circumferential Strain vs Average Rod Strain at the Plane of Maximum Flow Reduction from Multirod Burst Tests. For comparison, the maximum circumferential strains from single-rod tests on pellet-constrained cladding are shown for the same conditions of heating rate and cladding hoop stress as in each of the multirod tests.

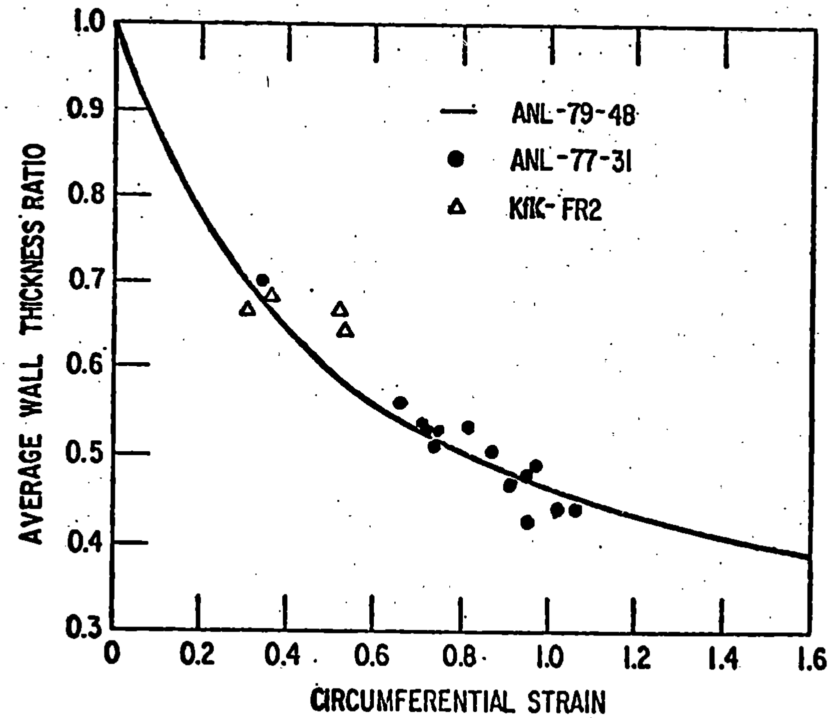

Fig. 3. Average Wall-thickness Ratio for Zircaloy-4 Cladding as a Function of Circumferential Strain after Deformation and Rupture in Steam at Temperatures of $\lesssim$ 1100 K at Heating Rates of $\leq$ 10 K/s.

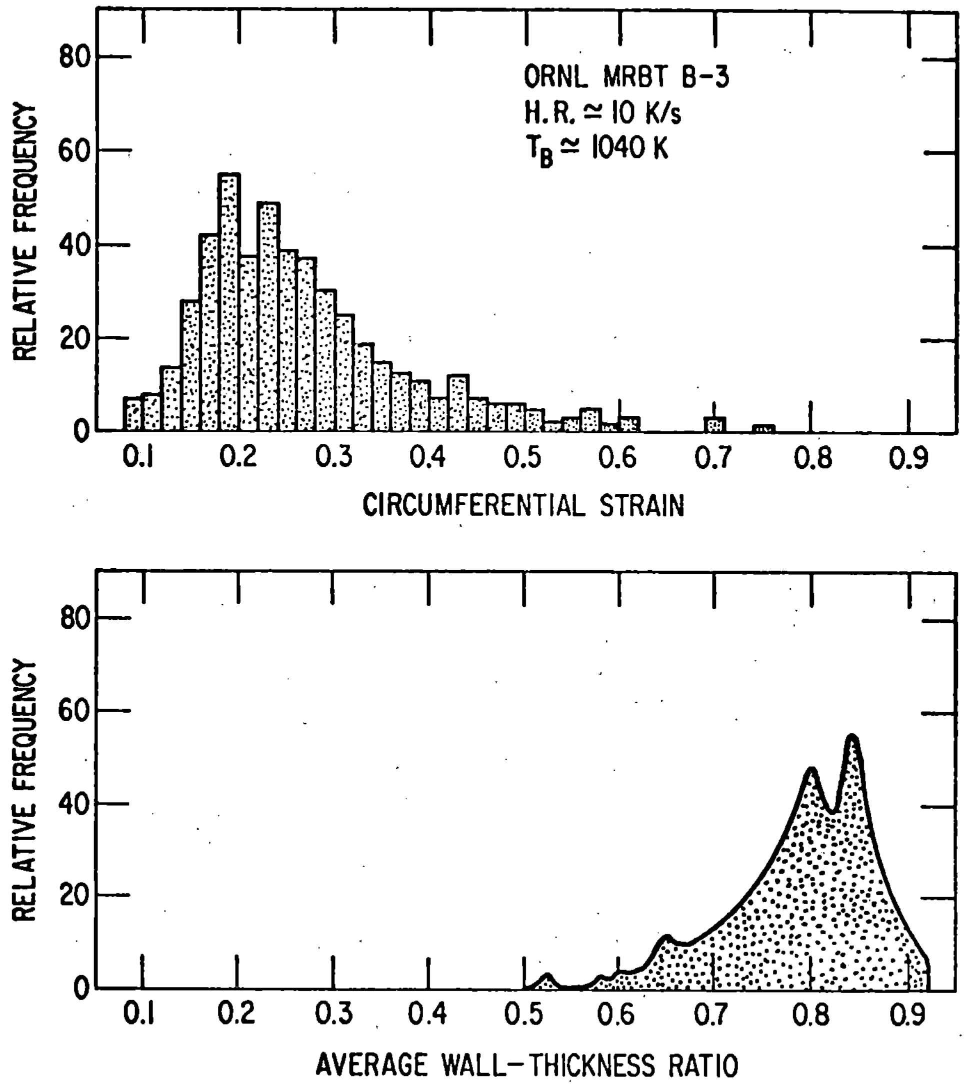

Fig. 4. Frequency Distribution of Circumferential Strain in Zircaloy Fuel Cladding from the ORNL B-3 Experiment Based upon 26 Cross Sections of a 4 x 4 Array of Rods between the Grid Locations. The frequency distribution of the wall-thickness ratio was obtained from these data and the curve in Fig. 3.

ESTIMATE OF ZIRCALOY CLADDING DEFORMATION AFTER A DOUBLE-ENDED GUILLOTINE BREAK IN THE PUMP DISCHARGE LEG OF TWO PWRs BASED UPON INPUT PARAMETERS FROM THE FINAL SAFETY ANALYSIS REPORTS (FSAR)

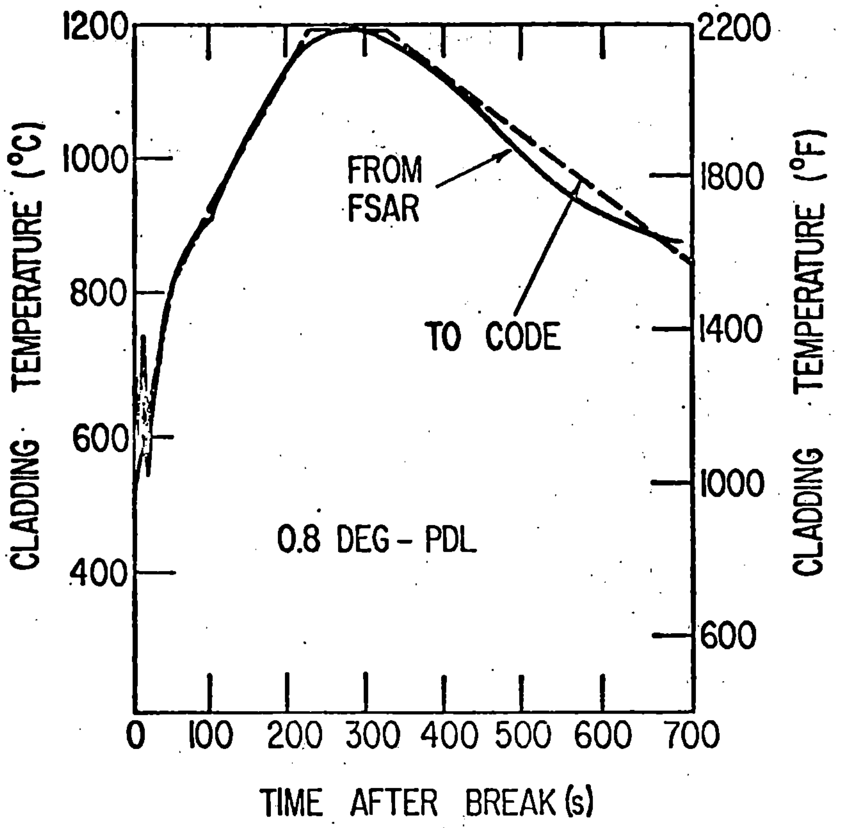

Fig. 5. Temperature vs Time Transient (Solid Line) for the Peak Temperature Node of Zircaloy Cladding during a 0.8 Double-ended Guillotine Break in the Pump Discharge Leg of a PWR. Transient represented by the dashed line was used with the computer code in ANL-79-48, NUREG/CR-1344 (Ref. 1) to calculate various oxidation parameters for the cladding.

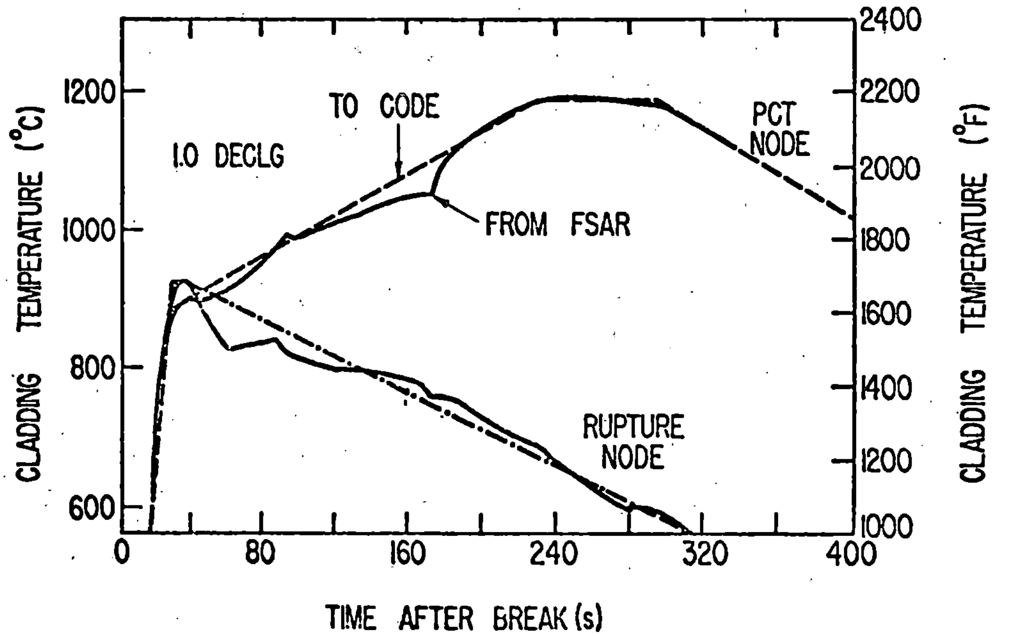

Fig. 6. Temperature vs Time Transients (Solid Lines) for the Peak Temperature and Rupture Nodes of Zircaloy Cladding during a 1.0 Double-ended Cold Leg Guillotine Break in a PWR. Transients represented by the dashed lines were used with the computer code in ANL-79-48, NUREG/CR-1344 (Ref. 1) to calculate various oxidation parameters for the cladding.

TABLE 1. Initial Clad Dimensions and Summary of Clad Deformation Parameters after Ballooning and Rupture during a Double-ended Guillotine Break in the Pump Discharge Leg of Two PWRs.

EVALUATION OF ECCS MARGIN OF PERFORMANCE

Reactor Plant	Design Basis Accident	Clad Dimensions			Clad Deformation Parameters[a]					
		IWT, mm	ID, mm	OD, mm	Node	H.R., K/s	σ_H, MPa	ϵ_θ	AWT, mm	WTR
OS	0.8 DEGPLD	0.635	8.43	9.70	PCT	5	76	0.38	0.41	0.65
CF	1.0 DECLG	0.64	9.86	11.34	PCT	10	69	0.38	0.41	0.64
					RUP	10	69	0.70	0.33	0.51

[a]Estimated values.

Definition of symbols

H.R. = heating rate
σ_H = hoop stress at rupture
ϵ_θ = circumferential strain

IWT = initial wall thickness
AWT = average wall thickness
WTR = wall thickness ratio, AWT/IWT

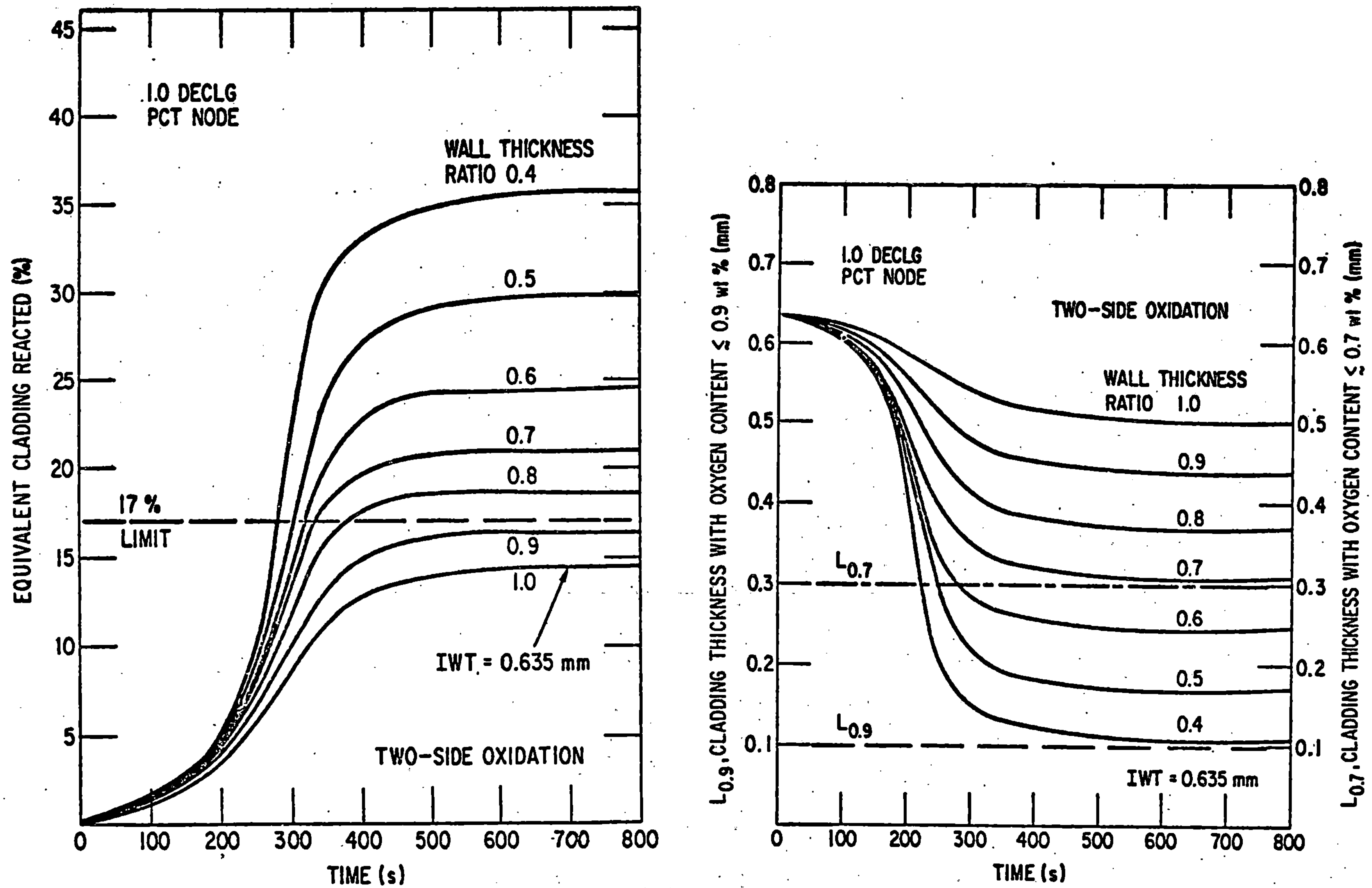

Fig. 7. Equivalent-cladding-reacted Parameter and Thickness of the Transformed β-phase Layer ($L_{0.9}$ and $L_{0.7}$) as Functions of Wall-thickness Ratio (Initial Wall Thickness of 0.635 mm) and Time for the Transient Corresponding to the Peak Temperature Node of Zircaloy Cladding after a 1.0 DECLG Break in a PWR. Curves are based on two-side oxidation of the cladding and the computer code in ANL-79-48, NUREG/CR-1344 (Ref. 1).

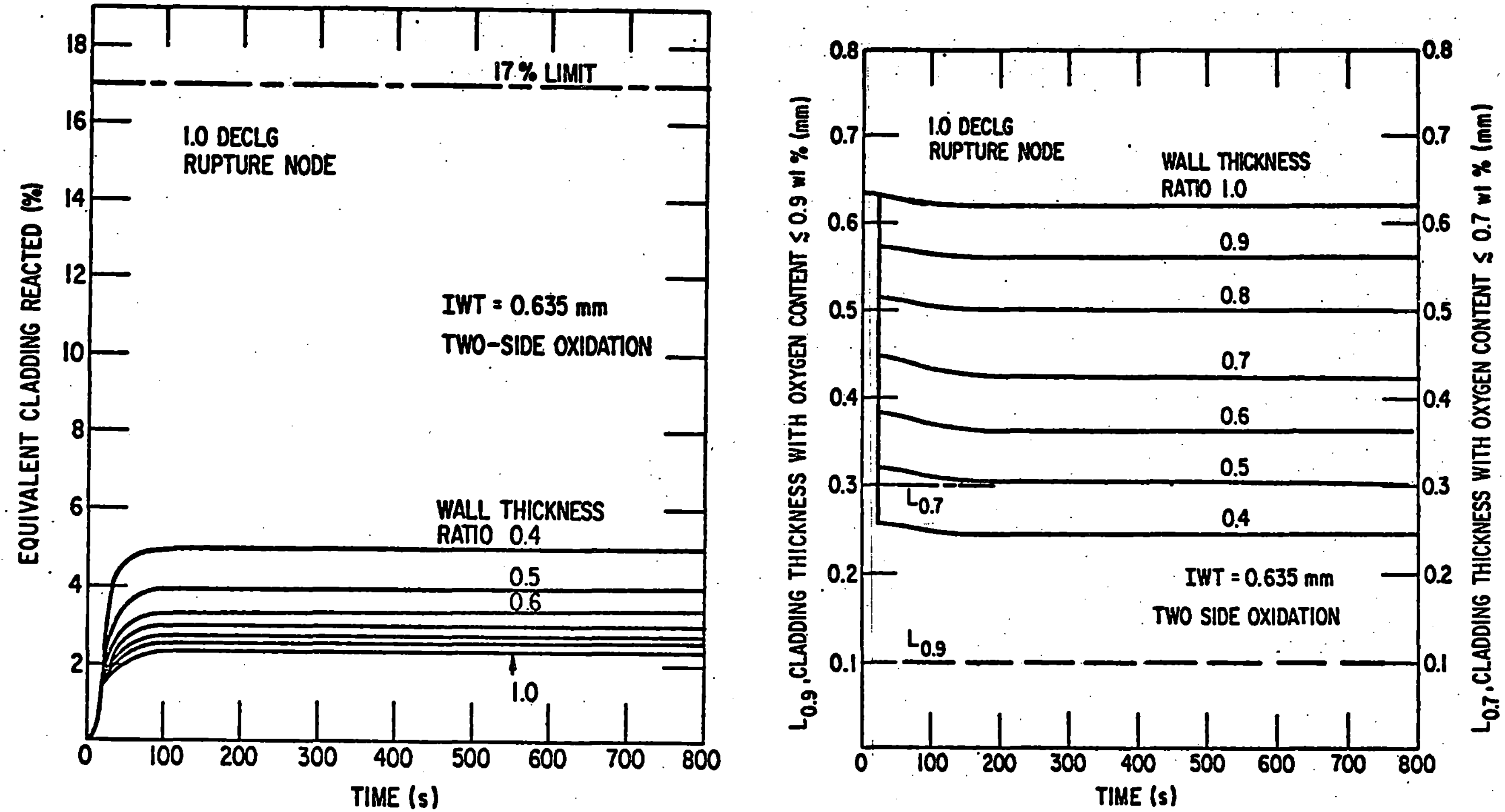

Fig. 8. Equivalent-cladding-reacted Parameter and Thickness of the Transformed β-phase Layer ($L_{0.9}$ and $L_{0.7}$) as Functions of Wall-thickness Ratio (Initial Wall Thickness of 0.635 mm) and Time for the Transient Corresponding to the Rupture Node of Zircaloy Cladding after a 1.0 DECLG Break in a PWR. Curves are based on two-side oxidation of the cladding and the computer code in ANL-79-48, NUREG/CR-1344 (Ref. 1).

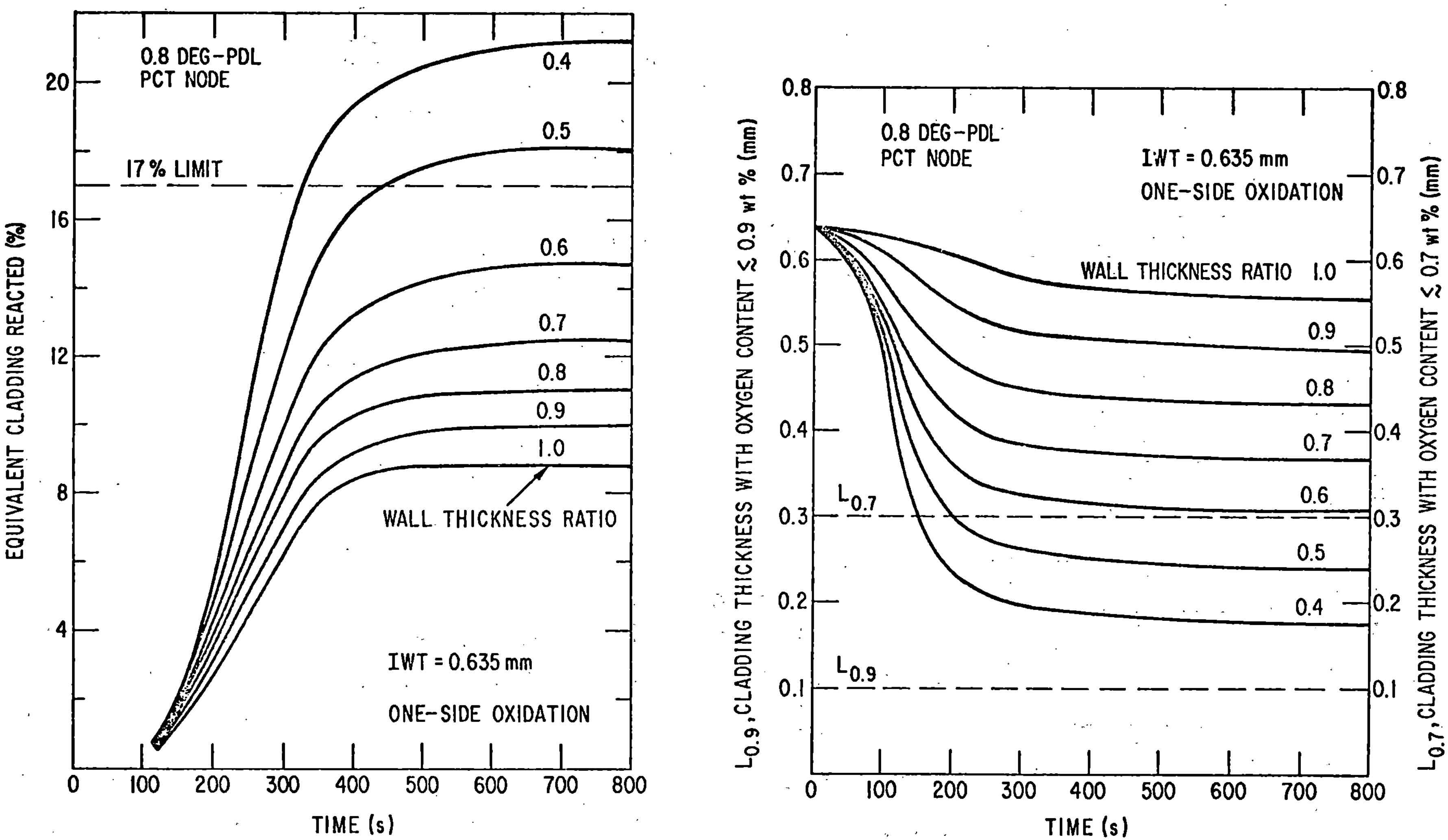

Fig. 9. Equivalent-cladding-reacted Parameter and Thickness of the Transformed β-phase Layer ($L_{0.9}$ and $L_{0.7}$) as Functions of Wall-thickness Ratio (Initial Wall Thickness of 0.635 mm) and Time for the Transient Corresponding to the Peak Temperature Node of Zircaloy Cladding after a 0.8 DEG-PDL Break in a PWR. Curves are based on one-side oxidation of the cladding and the computer code in ANL-79-48, NUREG/CR-1344 (Ref. 1).

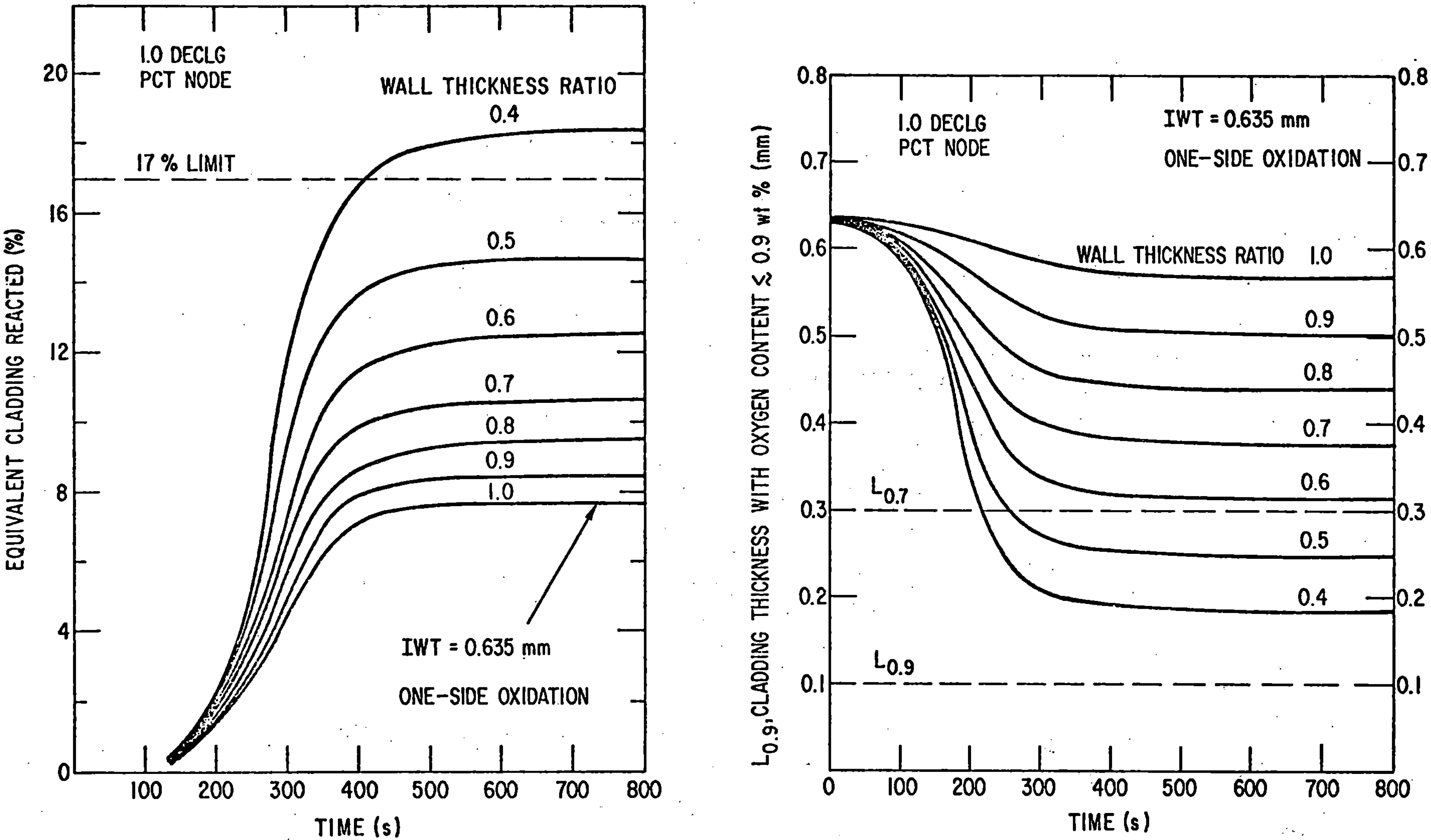

Fig. 10. Equivalent-cladding-reacted Parameter and Thickness of the Transformed β-phase Layer ($L_{0.9}$ and $L_{0.7}$) as Functions of Wall-thickness Ratio (Initial Wall Thickness of 0.635 mm) and Time for the Transient Corresponding to the Peak Temperature Node of Zircaloy Cladding after a 1.0 DECLG Break in a PWR. Curves are based on one-side oxidation of the cladding and the computer code in ANL-79-48, NUREG/CR-1344 (Ref. 1).

TABLE 2. Summary of Cladding Oxidation Parameters and Performance Limits for the ECCS in Two PWRs for a Double-ended Guillotine Break in the Pump Discharge Leg

EVALUATION OF ECCS MARGIN OF PERFORMANCE

		Clad Oxidation Parameters[a]						Performance Limits[b]		
Reactor Plant	Design Basis Accident	Node	17% ECR, s	t_f, s	ECR, %	$L_{(0.7)}$, mm	$L_{(0.9)}$, mm	Total Oxidation, $\frac{17\%}{ECR}$	0.3 Joule Impact, $\frac{L_{(0.7)}}{0.3}$	Thermal Shock, $\frac{L_{(0.9)}}{0.1}$
TWO-SIDE OXIDATION										
OS	0.8 DEGPLD	PCT	275	500	26.0	0.25	0.25	0.65	0.83	2.5
CF	1.0 DECLG	PCT	325	475	22.5	0.28	0.28	0.75	0.93	2.8
		RUP	-	100	4.0	0.30	0.30	4.20	1.00	3.0
ONE-SIDE OXIDATION										
OS	0.8 DEGPLD	PCT	-	500	13.5	0.34	0.34	1.26	1.10	3.4
CF	1.0 DECLG	PCT	-	475	11.5	0.35	0.35	1.48	1.17	3.5

[a]Values computed from the oxidation model reported in ANL-79-48, NUREG/CR-1344.

[b]Value of ≥ 1 indicates performance limit is met.

SUMMARY AND CONCLUSIONS

- A Method for Evaluating the Margin of Performance of ECCSs in PWRs, Which Incorporates Recent Information on the Ballooning and Embrittlement Characteristics of Zircaloy Cladding, Was Illustrated for Several LOCA Transients.

- For the DECLG Breaks, Compliance with the Oxidation Limit in the Present Acceptance Criteria for ECCSs Occurs for Two-side Oxidation of Cladding with Virtually No Wall Thinning Due to Deformation, and for Regions of the Rod, with ≤ 60% Circumferential Strain, Located far enough from the Rupture Area to Justify One-side Oxidation of the Cladding.

- Relative to the Proposed Embrittlement Criterion (i.e., β-Layer Thickness $L_{(0.9)} \geq 0.1$ mm), the Margin of Performance of the ECCSs Indicated Adequate Thermal-shock-failure Resistance for Ballooned and Ruptured Cladding after Two-side Oxidation during the Large-Break LOCA Transients.

RECOMMENDATIONS

- Applicability of One- or Two-side Oxidation of the Cladding Should Be Established on the Basis of Location of the Peak-Temperature Node in Relation to the Rupture Position and Factors That Influence Steam Access to the Inner Surface of the Cladding.

- Evaluation Models Used in ECCS Performance Analyses Must:

    - Properly Account for the Effect of Reduction in Flow Area on the Time-Temperature Transient.

    - Incorporate a More Quantitative Oxidation Code That Will Not Overestimate the Time Period Associated with the Extent of Cladding Oxidation near the Thermal-shock-failure Boundary.

AN ASSESSMENT OF THE INFLUENCE OF SURFACE THERMOCOUPLES ON THE BEHAVIOR OF NUCLEAR FUEL RODS DURING A LARGE BREAK LOCA

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-30, 1980
Gaithersburg, Maryland

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Idaho Falls, Idaho 83415

AN ASSESSMENT OF THE INFLUENCE OF SURFACE THERMOCOUPLES ON THE BEHAVIOR OF NUCLEAR FUEL RODS DURING A LARGE BREAK LOCA

T. R. Yackle
EG&G Idaho, Inc.

A second series of thermocouple effects tests[1] (TC-3) is being conducted in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory (INEL). These tests are designed to evaluate the influence of external surface thermocouples on the behavior of nuclear fuel rods during a large break loss-of-coolant accident (LOCA) experiment, which includes partial or total rod rewet during blowdown. Questions have been raised with regard to possible effects[2-4] of surface thermocouples on the thermal behavior of in-pile and out-of-pile test rods. This issue was highlighted by the recent large break LOCA test results obtained at the Loss-of-Fluid Test[5] (LOFT) facility at the INEL where measured cladding temperatures were significantly lower than expected and a core rewet was measured early in the system depressurization. The key questions are: (a) did the external thermocouples accurately measure cladding temperatures, and (b) did the external thermocouples cause the core rewet early in the depressurization?

The TC-3 test series, currently being conducted at the PBF, is an extension of the TC-1[1] tests with four LOFT-type fuel rods contained in individual flow shrouds. The fuel rods are symmetrically positioned within a test train in the PBF in-pile tube in an environment similar to the LOFT test environment. Two rods are each instrumented with four LOFT cladding outside surface thermocouples, with junctions located near the high power region of the fuel rods. All four rods are instrumented with internal thermocouples, with junctions at the same axial level as the external thermocouples. By comparing the response of the internal and external thermocouples, the behavior of fuel rods during a LOCA with and without external thermocouples is being examined.

Surface thermocouples were found to influence fuel rod cladding temperatures during both the blowdown and reflood phases of the first test of this series, TC-3A. During blowdown, the TC-3A results verify the earlier TC-1 results with the surface thermocouples (a) slightly delaying the onset of critical heat flux (CHF) and (b) slightly improving cladding surface heat transfer. Peak temperatures measured during the TC-3A blowdown were reduced by about 10% (about 100 K) due to the presence of external thermocouples. Similarly, peak temperatures measured during the TC-1 blowdowns were generally about 100 K lower due to the combined thermocouple effects of delayed CHF and improved cladding surface heat transfer. The surface thermocouples also influenced the cladding quench and rewet times during the reflood phase of both the TC-1 and TC-3A tests. In general, the fuel rods with external thermocouples quenched 3 to 12 seconds before the fuel rods without external thermocouples. In addition, some external thermocouples did not properly measure the cladding temperature response as was evident by momentary quenching and reheating of the surface thermocouples prior to the actual rod quench.

The TC-3 test series extends the results of TC-1 by providing data for evaluation of thermocouple effects during the apparent two-phase rewet that occurred early in the LOFT blowdowns. Based on TC-3A data, the external thermocouples did not significantly influence the fuel rod thermal response during the blowdown quench. The cooling rates of both the fuel and cladding during the blowdown quench were similar for the rods with and without surface thermocouples. However, the external thermocouples did not exactly measure the cladding surface temperature. The preliminary test data during the TC-3 blowdown quench indicates that the external thermocouples quenched about four times faster than the cladding quenched. The results also indicate that thermal decoupling of the cladding and fuel was apparently significant, allowing the cladding to rapidly quench during the blowdown, regardless of the presence of external thermocouples. This thermal decoupling of fuel and cladding demonstrates the importance of in-pile experiments or out-of-pile experiments where the fuel-to-cladding gap is properly simulated.

Future PBF thermocouple tests will be designed to further quantify the influence of surface thermocouples during a blowdown quench with variations in the degree of quench which should encompass the range of conditions that occurred in the LOFT L2 test series.

REFERENCES

1. T. R. Yackle, et al., Loss-of-Coolant Accident Test Series Test TC-1 Test Results Report, EGG-TFBP-5068, May 1980.

2. G. Class, et al., "Concerning the Pretests Conducted at COSIMA for the Influence of Outside Mounted Thermocouples", KfK Report 06.01.07P03A, PNS-Nr., August 1979 pp. 402-79.

3. R. C. Gottula, LOFT Transient (Blowdown) Critical Heat Flux Tests, TREE-NUREG-1240, April 1978.

4. G. L. Shires, et al., An Experimental Study of the Effect of External Thermocouples in Rewetting During Reflood, AEEW-R 1357, April 1980.

5. D. L. Reider, Quick Look Report on LOFT Nuclear Experiment L2-3, QLR-L2-3, Project No. P394, May 1979.

AN ASSESSMENT OF THE INFLUENCE OF SURFACE THERMOCOUPLES ON THE BEHAVIOR OF NUCLEAR FUEL RODS DURING A LARGE-BREAK LOCA (TC-3 RESULTS)

by
T. R. YACKLE

EG&G Idaho, Inc.

OUTLINE

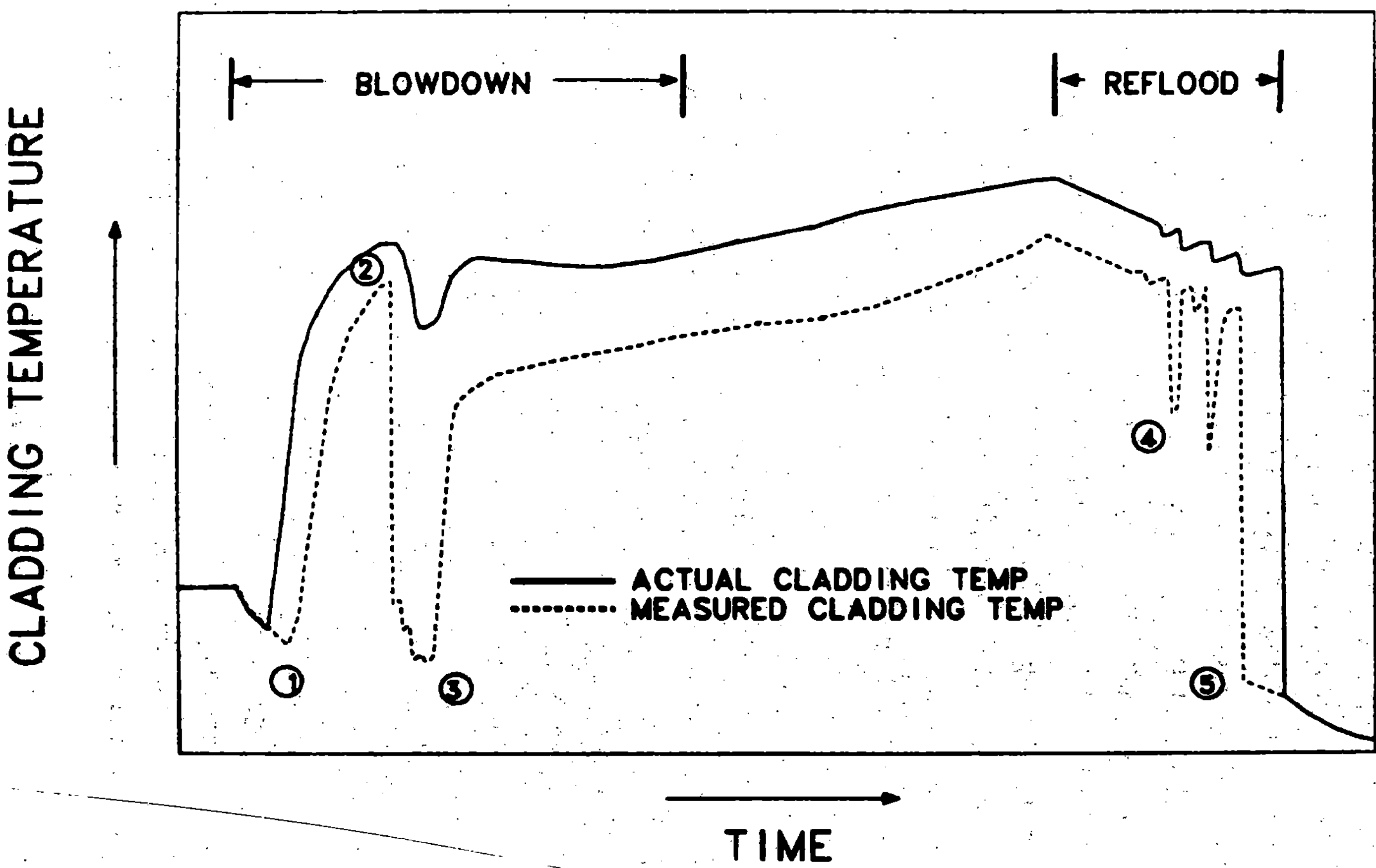
- OBJECTIVES
- TEST DESIGN AND CONDUCT
- TC-3 THERMOCOUPLE EFFECTS
- CONCLUSIONS

TC-3 TEST OBJECTIVES

- DO CLADDING SURFACE THERMOCOUPLES INFLUENCE FUEL ROD THERMAL BEHAVIOR DURING A LOCA?
- DO CLADDING THERMOCOUPLES ACCURATELY MEASURE CLADDING TEMPERATURES?

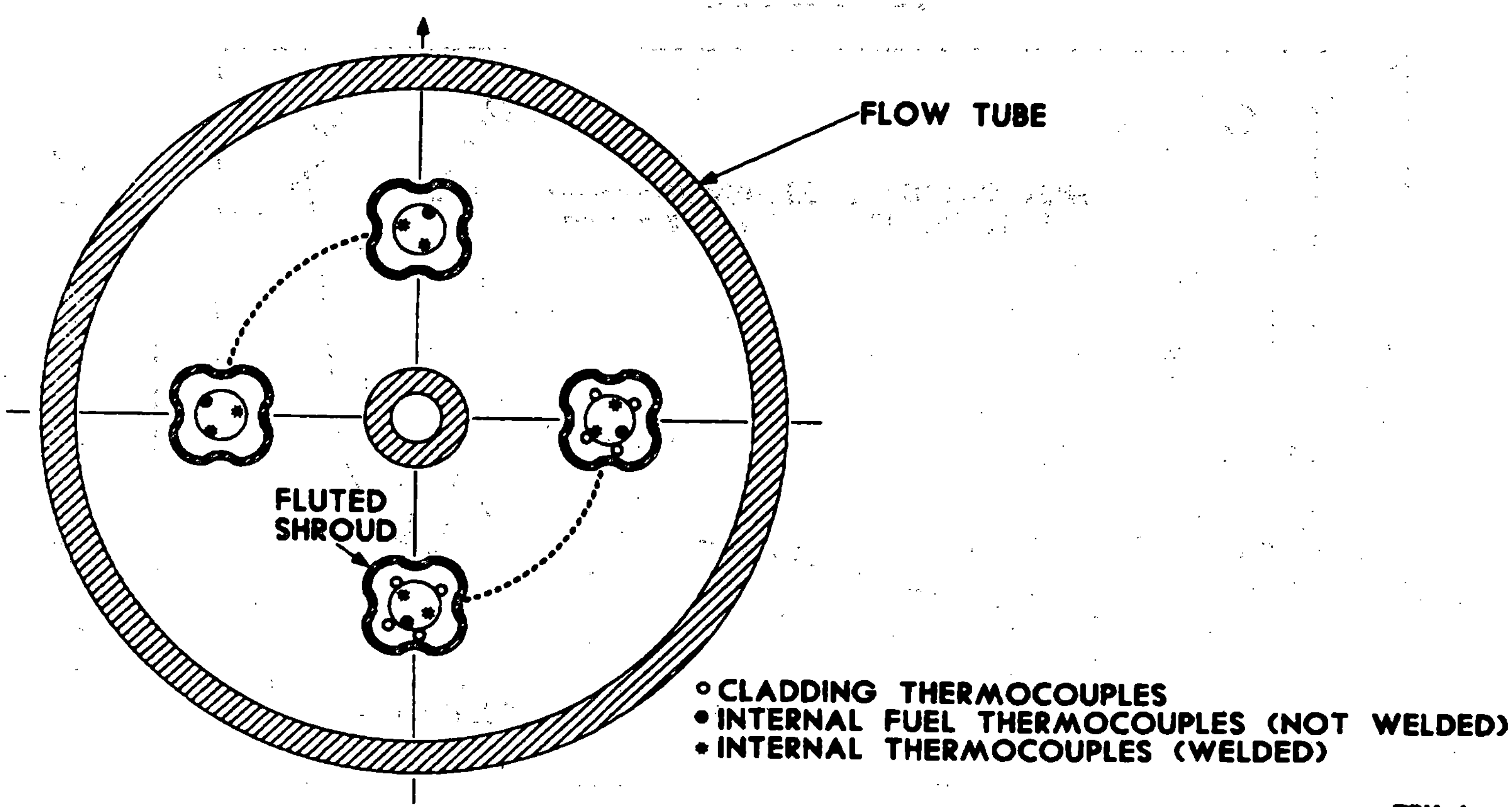
TRY-3

LOCA THERMOCOUPLE EFFECTS



TRY-5

TC-3 GEOMETRY

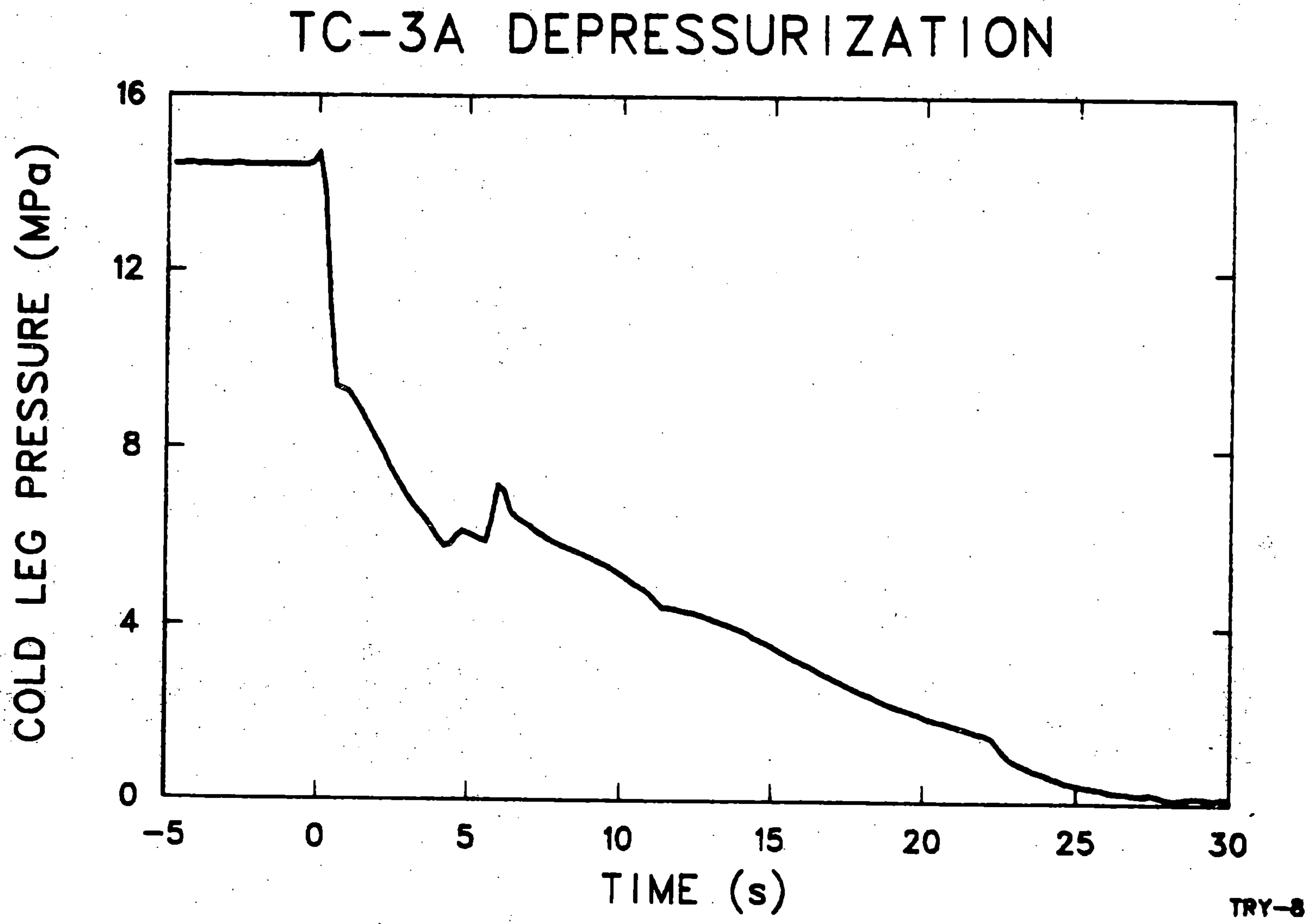


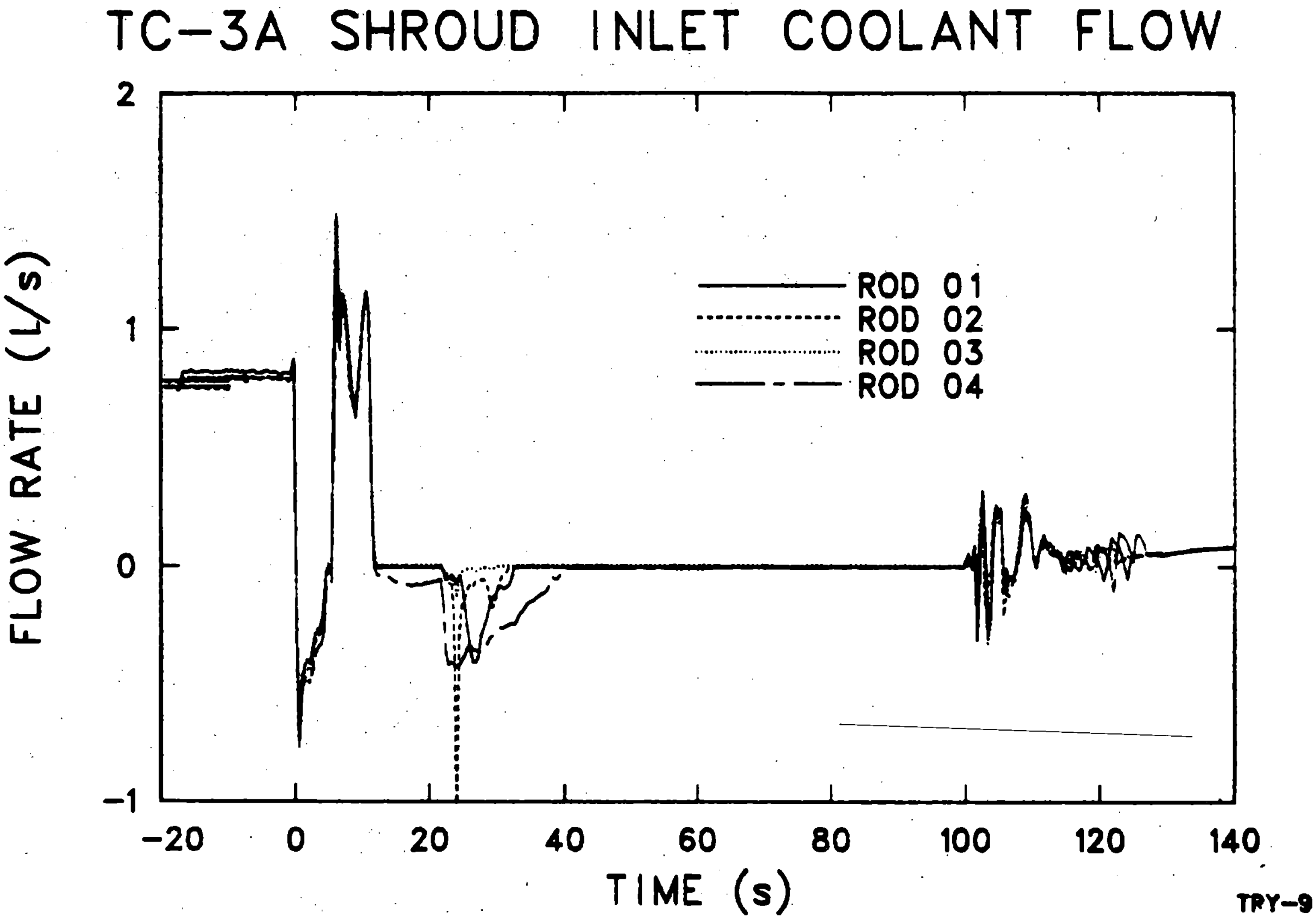
TRY-6

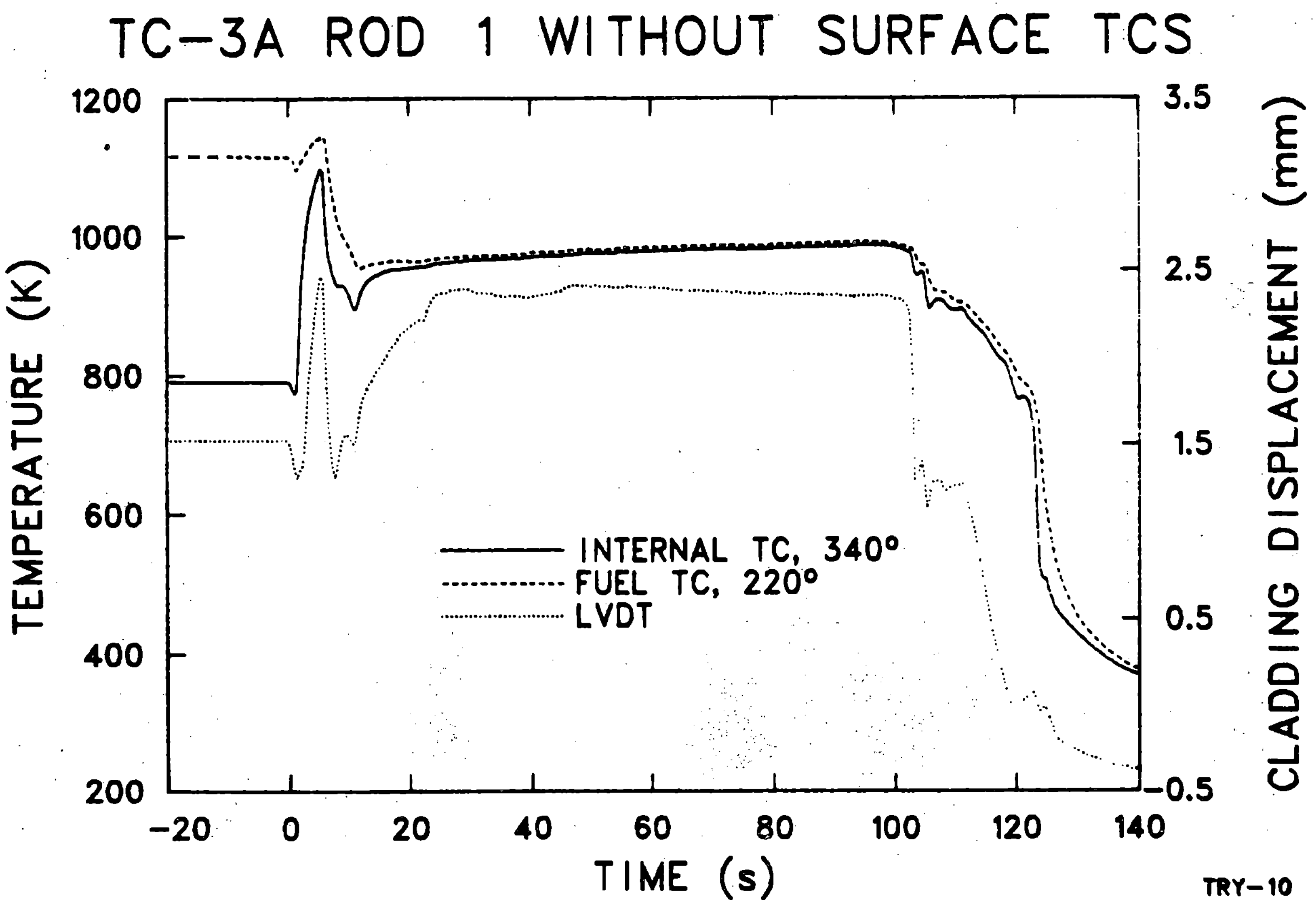
TC INITIAL CONDITIONS

POWER	~50 kW/m
INLET TEMPERATURE	600 K
SYSTEM PRESSURE	15.5 MPa
COOLANT FLOW	0.8 l/s

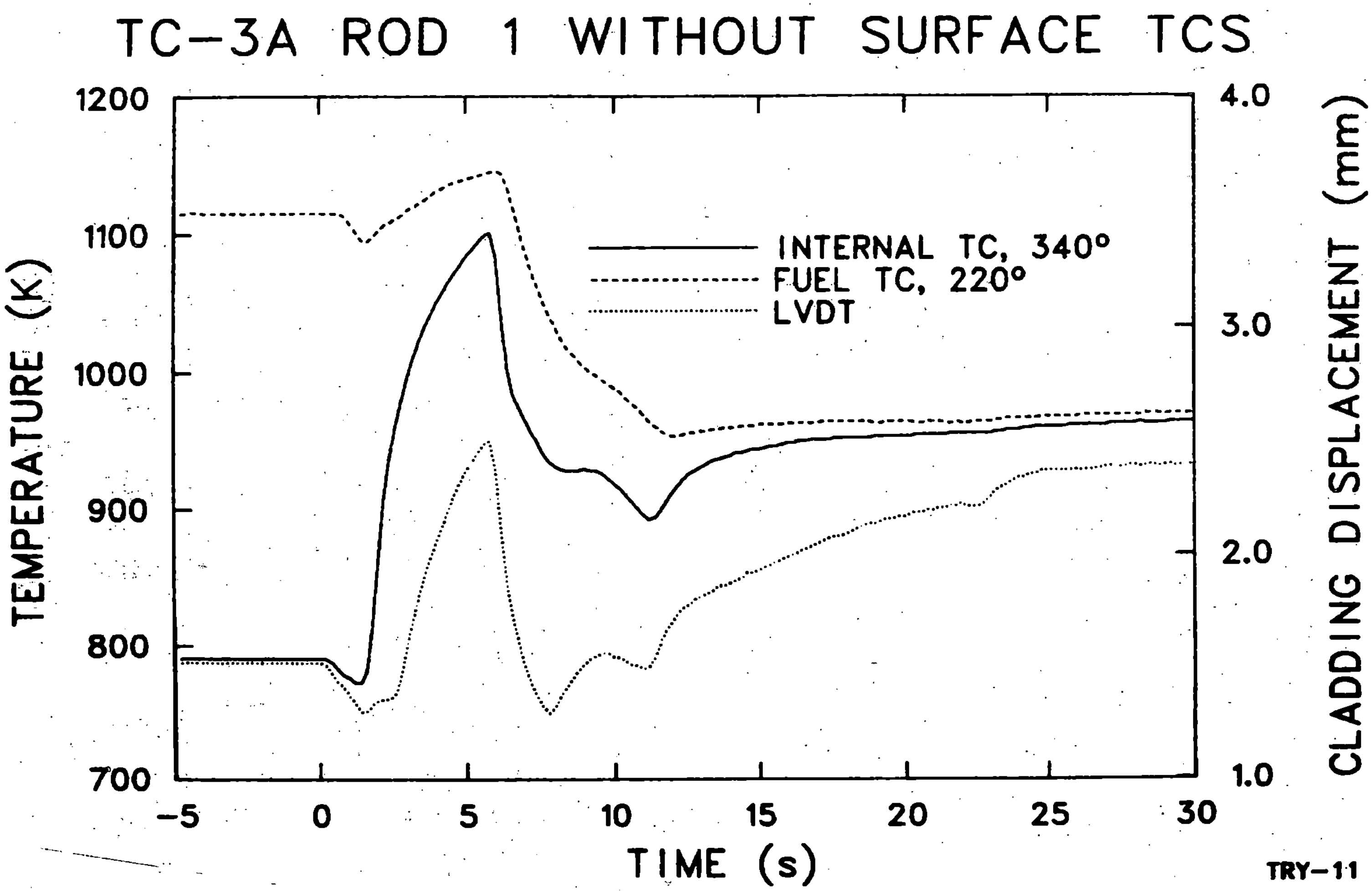
TRY-7

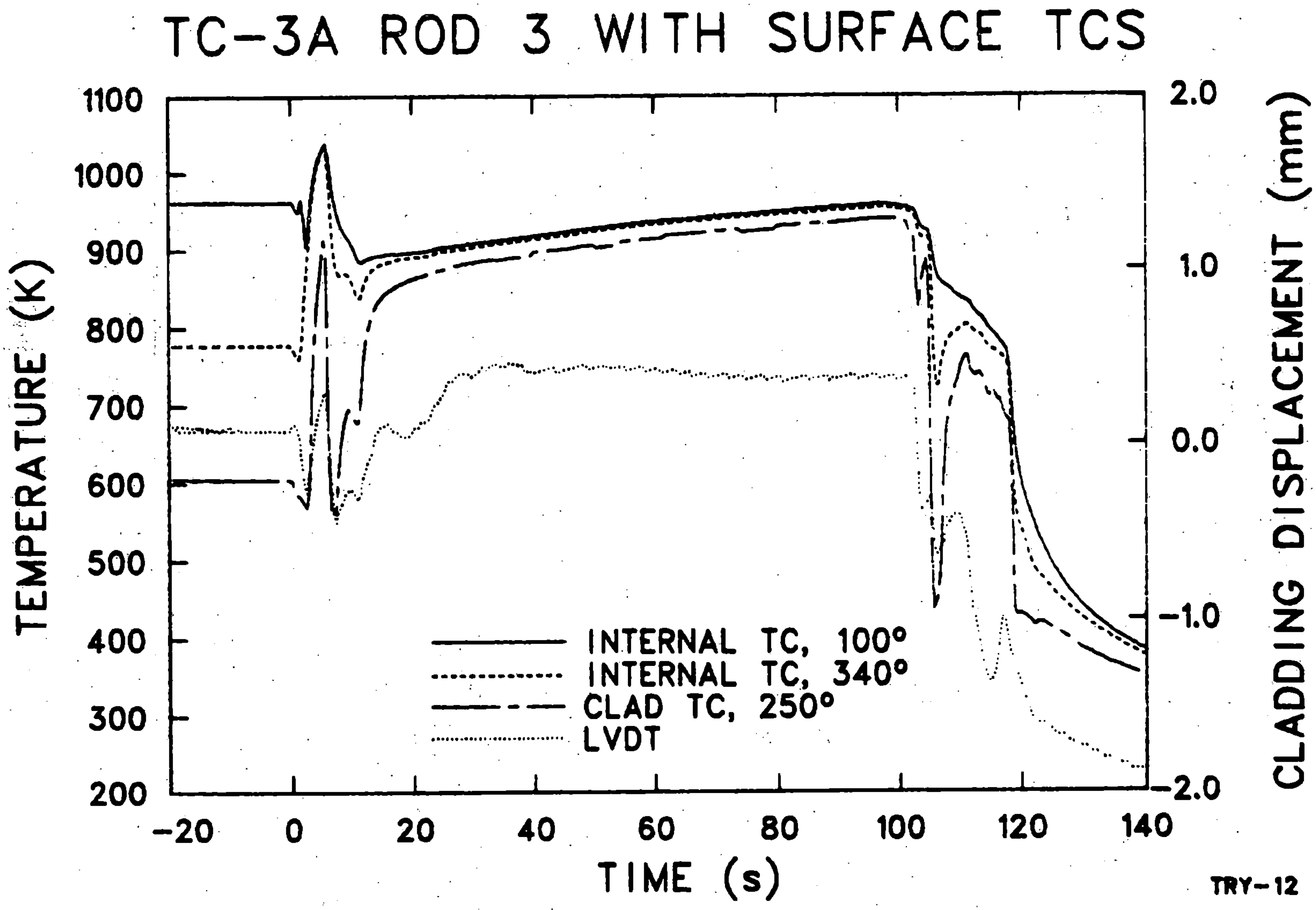

TC-3A DEPRESSURIZATION
COLD LEG PRESSURE (MPa)
16
12
8
4
0
-5
0
5
10
15
20
25
30
TIME (s)
TRY-8


TC-3A SHROUD INLET COOLANT FLOW
FLOW RATE (L/s)
2
1
0
-1
-20
0
20
40
60
80
100
120
140
TIME (s)
ROD 01
ROD 02
ROD 03
ROD 04
TRY-9

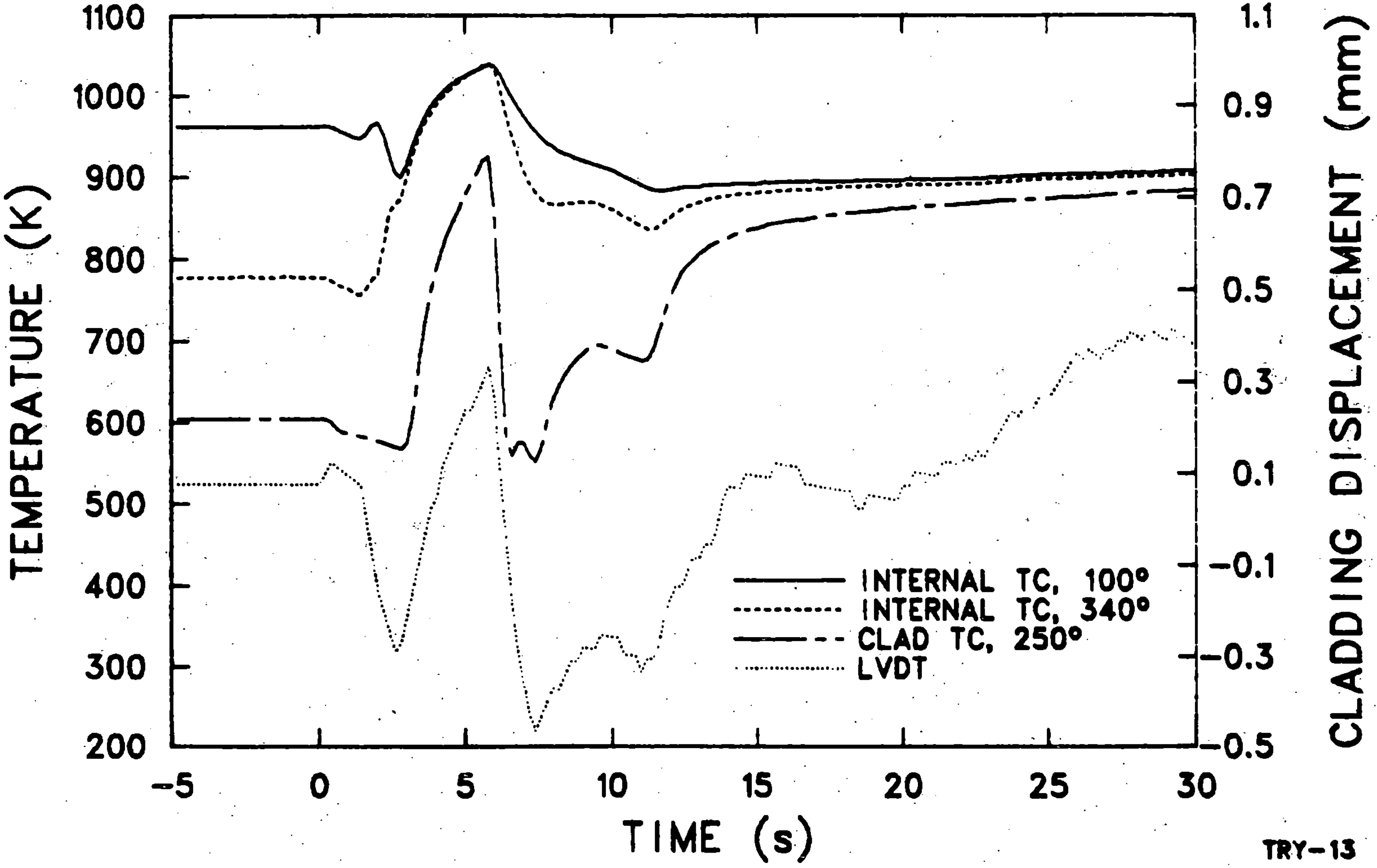


TC-3A ROD 1 WITHOUT SURFACE TCS
TEMPERATURE (K)
1200
1000
800
600
400
200
CLADDING DISPLACEMENT (mm)
3.5
2.5
1.5
0.5
-0.5
INTERNAL TC, 340°
FUEL TC, 220°
LVDT
-20
0
20
40
60
80
100
120
140
TIME (s)
TRY-10

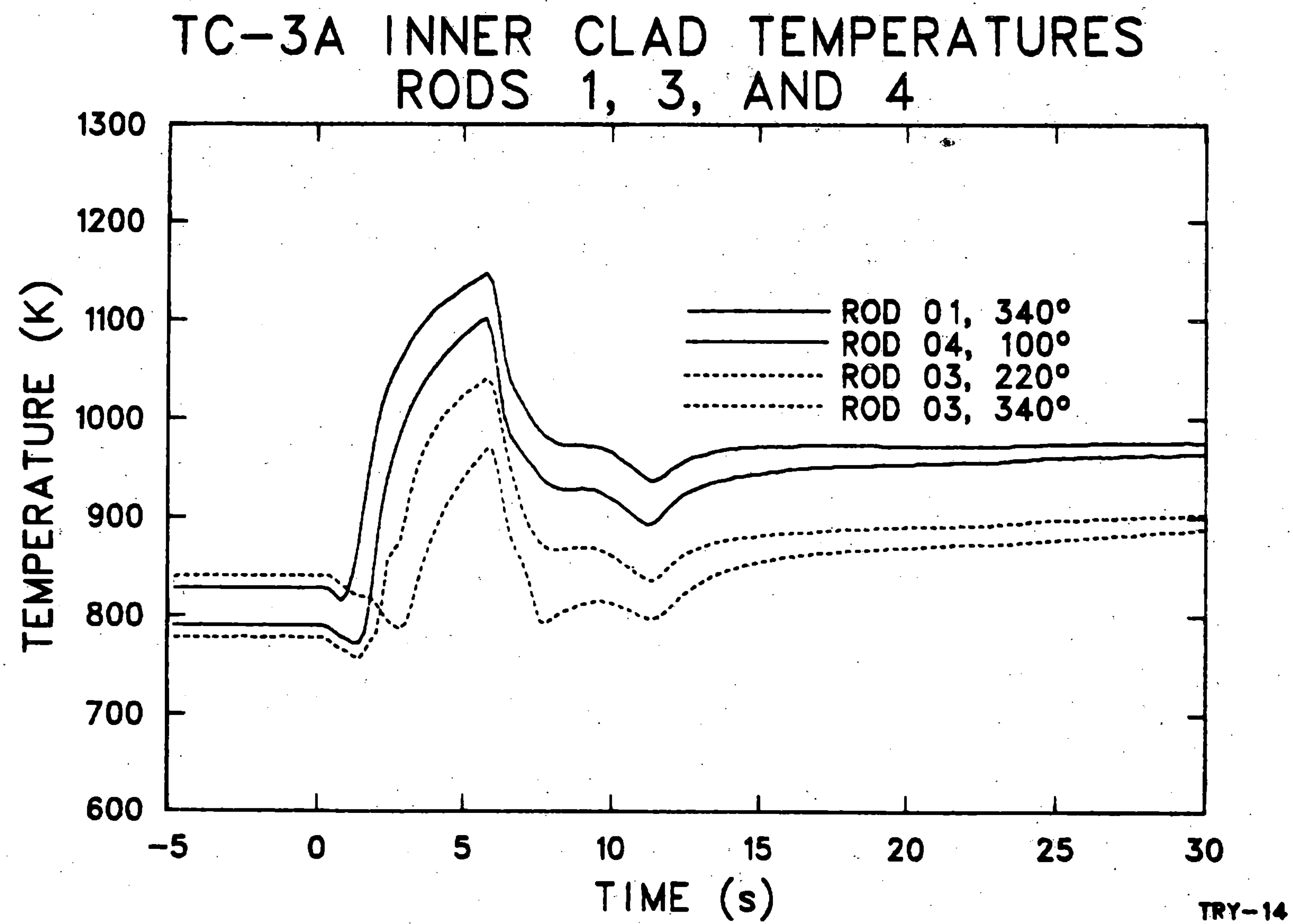

TC-3A ROD 1 WITHOUT SURFACE TCS
INTERNAL TC, 340°
FUEL TC, 220°
LVDT
TEMPERATURE (K)
1200
1100
1000
900
800
700
CLADDING DISPLACEMENT (mm)
4.0
3.0
2.0
1.0
-5
0
5
10
15
20
25
30
TIME (s)
TRY-11

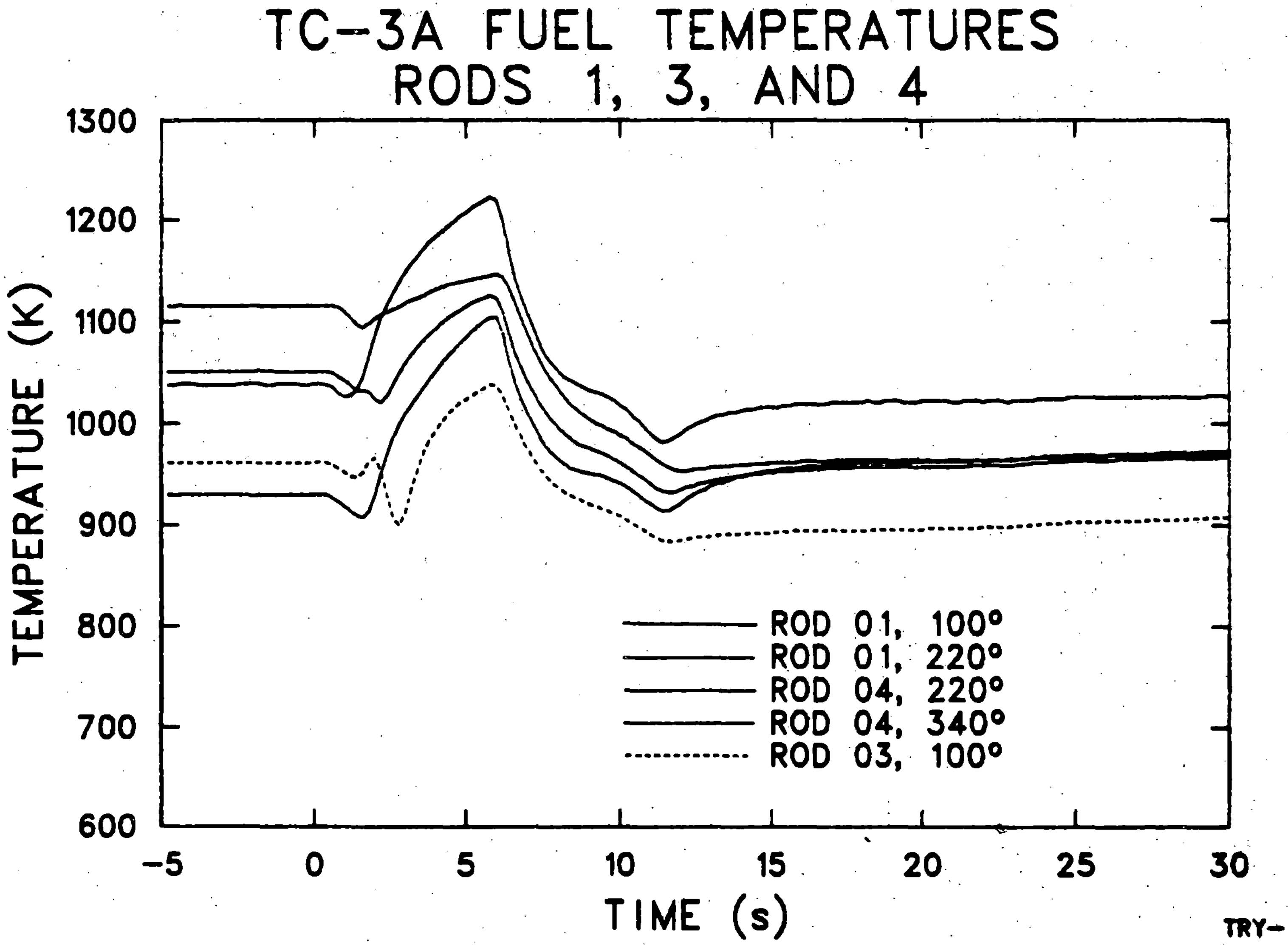

TC-3A ROD 3 WITH SURFACE TCS
TEMPERATURE (K)
CLADDING DISPLACEMENT (mm)
TIME (s)
INTERNAL TC, 100°
INTERNAL TC, 340°
CLAD TC, 250°
LVDT
1100
1000
900
800
700
600
500
400
300
200
2.0
1.0
0.0
−1.0
−2.0
−20
0
20
40
60
80
100
120
140
TRY-12

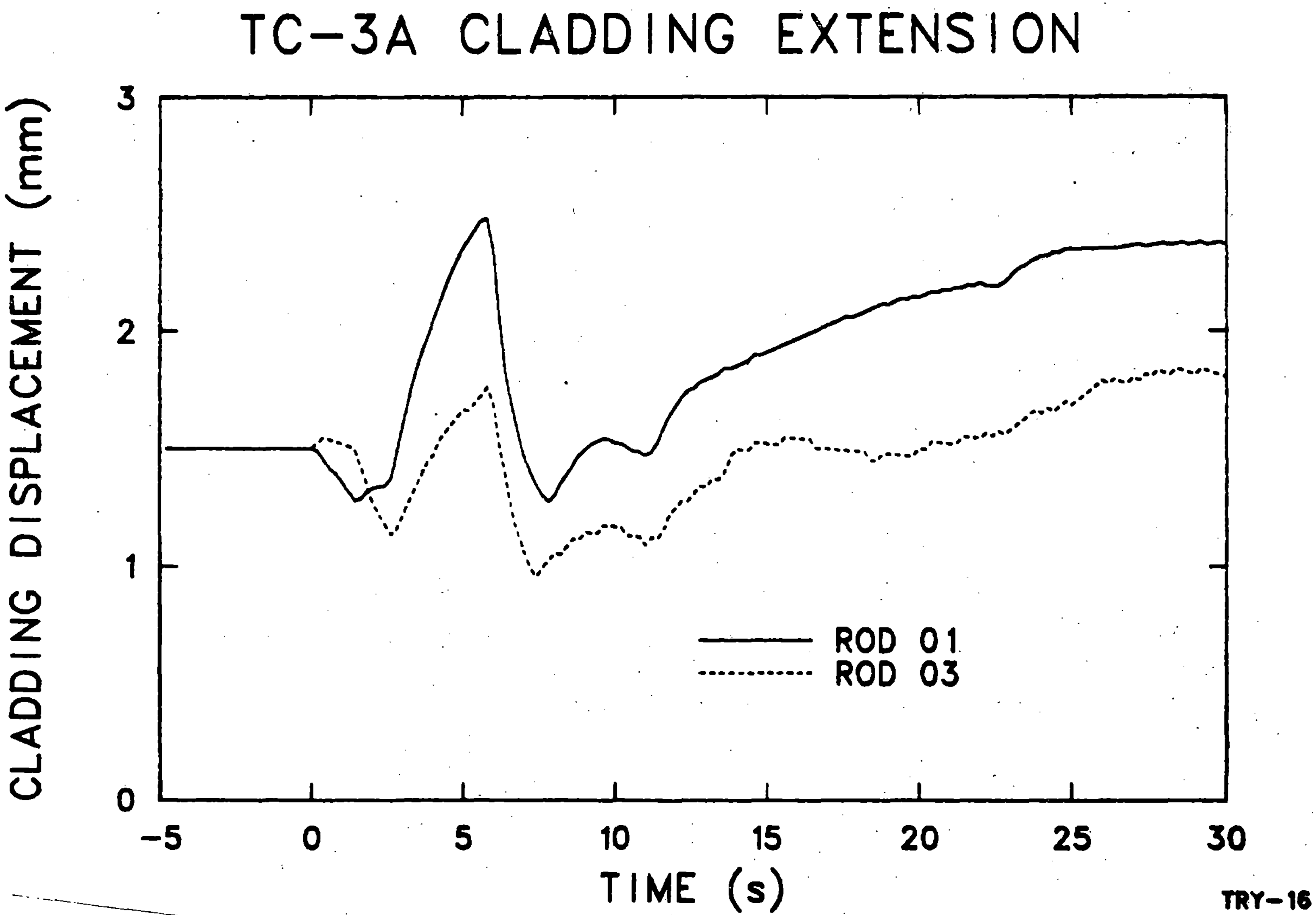
TC-3A ROD 3 WITH SURFACE TCS



TRY-13


TC-3A INNER CLAD TEMPERATURES
RODS 1, 3, AND 4
TEMPERATURE (K)
1300
1200
1100
1000
900
800
700
600
-5
0
5
10
15
20
25
30
TIME (s)
ROD 01, 340°
ROD 04, 100°
ROD 03, 220°
ROD 03, 340°
TRY-14


TC-3A FUEL TEMPERATURES
RODS 1, 3, AND 4
TEMPERATURE (K)
1300
1200
1100
1000
900
800
700
600
−5
0
5
10
15
20
25
30
TIME (s)
ROD 01, 100°
ROD 01, 220°
ROD 04, 220°
ROD 04, 340°
ROD 03, 100°
TRY-15

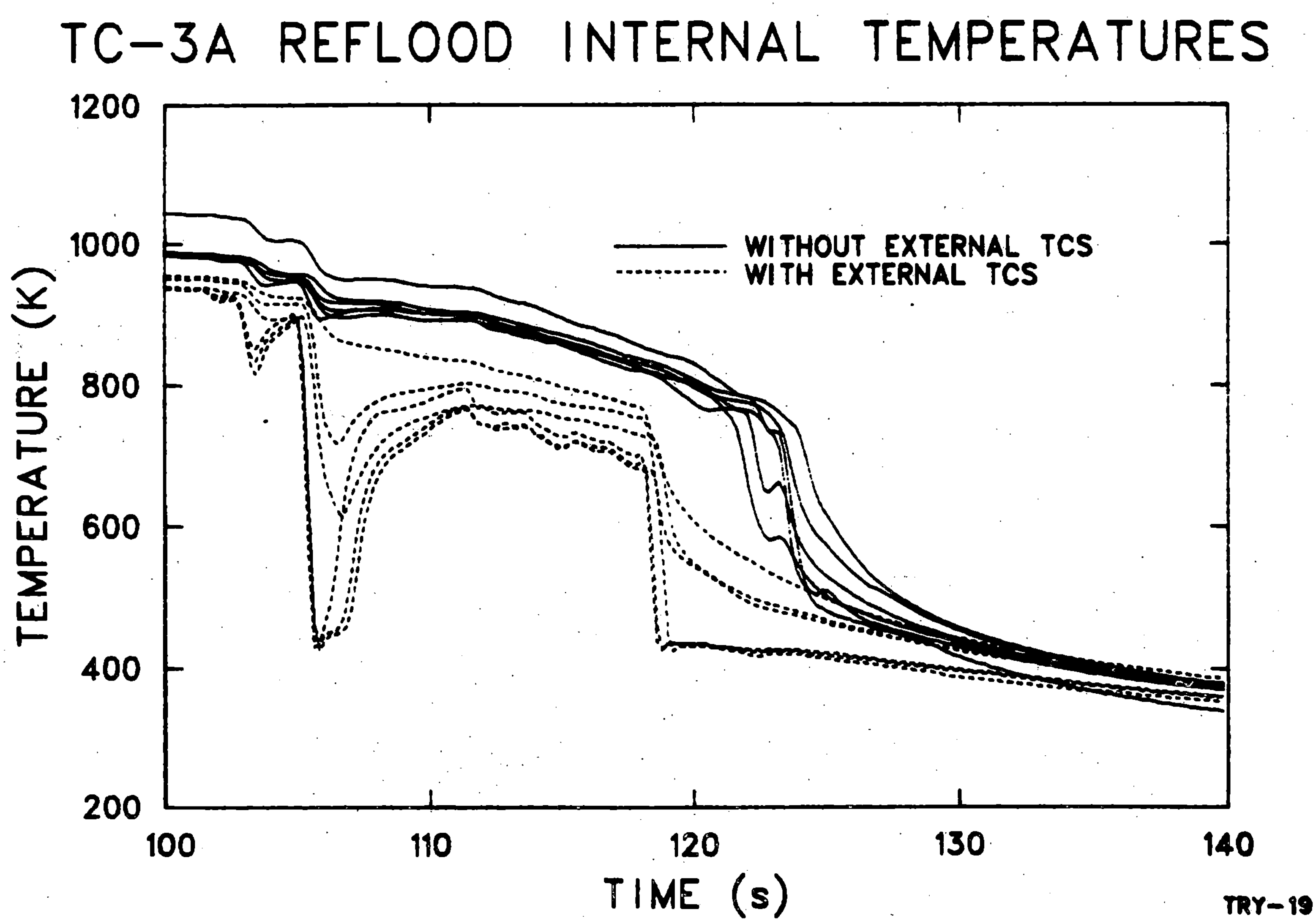


TC-3A CLADDING EXTENSION
CLADDING DISPLACEMENT (mm)
TIME (s)
ROD 01
ROD 03
0
1
2
3
-5
0
5
10
15
20
25
30
TRY-16

ROD COOLDOWN RATES DURING THE TC-3A BLOWDOWN QUENCH

	CLADDING TCs (k/s)	CLADDING LVDT (k/s)	FUEL TCs (k/s)
ROD 01	NONE	145	70
ROD 03	570	125	80
ROD 04	NONE	---	85

TRY-17


TC-3A REFLOOD INTERNAL TEMPERATURES
TEMPERATURE (K)
1200
1000
800
600
400
200
100
110
120
130
140
TIME (s)
WITHOUT EXTERNAL TCS
WITH EXTERNAL TCS
TRY-19

CONCLUSIONS

- PEAK CLADDING TEMPERATURES DURING BLOWDOWN REDUCED ABOUT 10% DUE TO EXTERNAL THERMOCOUPLES INFLUENCE ON
    - CHF
    - SURFACE HEAT TRANSFER
- NO MAJOR DIFFERENCES IN ROD THERMAL RESPONSE OCCURRED DURING BLOWDOWN QUENCH DUE TO EXTERNAL THERMOCOUPLES

TRY-2C

CONCLUSIONS (CONT'D)

- EXTERNAL THERMOCOUPLES DID NOT EXACTLY MEASURE CLADDING SURFACE TEMPERATURE
  - QUENCHED ABOUT 4 TIMES FASTER THAN CLADDING
- EXTERNAL THERMOCOUPLES INFLUENCED CLADDING QUENCH DURING REFLOOD
  - MOMENTARY REWETS
  - EARLY QUENCH (3 - 12 SEC)

TRY-21

CONCLUSIONS (CONT'D)

- THERMAL DECOUPLING OF FUEL AND CLADDING IS SIGNIFICANT DURING A LOCA ALLOWING CLADDING TO COOL ABOUT 2 TIMES FASTER THAN FUEL DURING THE BLOWDOWN QUENCH
  - DEMONSTRATES CONSERVATISM BETWEEN SOME OUT-OF-PILE AND IN-PILE EXPERIMENTS

RESPONSE OF PREIRRADIATED FUEL ROD BUNDLE DURING REACTIVITY INITIATED ACCIDENT TEST 1-4

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

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Z. R. Martinson

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Idaho Falls, Idaho 83415

RESPONSE OF PREIRRADIATED FUEL ROD BUNDLE DURING REACTIVITY INITIATED ACCIDENT TEST 1-4

P. E. MacDonald
Z. R. Martinson
EG&G Idaho, Inc.

The rapid insertion of excess reactivity into a light water reactor (LWR) core has long been recognized as an accident mechanism with the potential for failure of the fuel rod cladding. Extensive cladding failure and subsequent dispersal of fuel into the coolant could disrupt the core such that the postaccident cooling capability of the core would be significantly impaired. Light water reactors operated within the United States must be designed such that a worst-case Reactivity Initiated Accident (RIA) will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod. This NRC design requirement is meant to ensure minimal fuel rod damage and maintain the core in a coolable geometry in the event of an RIA. The NRC criterion[1] is based on RIA experimental data[2] obtained prior to 1974 from the behavior of single or small clusters of fuel rods tested under near room temperature and atmospheric (or near atmospheric) pressure conditions, no forced coolant flow, and zero initial power. Only a few irradiated fuel rods were tested.

A new RIA fuel behavior experimental program sponsored by the NRC in the Power Burst Facility (PBF) reactor at the Idaho National Engineering Laboratory has now been completed by EG&G, Idaho, Inc.,[3,4]. The program was focused on the behavior of irradiated fuel rods tested under coolant conditions typical of hot-startup conditions in a commercial boiling water reactor (BWR). The PBF-RIA test series was comprised of seven tests: four single-rod tests, two four-rod tests, and the latest test, RIA 1-4 -- a bundle test with nine previously irradiated fuel rods.

The results from the first six tests with separately shrouded fuel rods indicate that failure and loss of geometry will occur when unirradiated rods are subjected to fuel enthalpies of 250, 260, and 285 cal/g UO_2. Previously irradiated fuel rods tested at 285 cal/g UO_2 failed due to extensive fission product induced molten fuel swelling. The swelling of the molten or nearly molten fuel within the irradiated rods resulted in complete blockage of the coolant flow channels. The principal fuel rod and cladding damage mechanisms, in chronological order, included: pellet-cladding mechanical interaction of previously irradiated cladding due to high strain rate deformation, variations in cladding wall thickness due to cladding plastic deformation associated with partial or total melting of the cladding, fuel swelling and cladding rupture of previously irradiated fuel rods, cladding oxidation and embrittlement during film boiling and subsequent fragmentation of the cladding upon quench, and fuel grain boundary separation and fuel powdering during quench.

The objective of Test RIA 1-4 was to evaluate the coolability of bundle of previously irradiated fuel rods subjected to a radial average peak fuel enthalpy near the present NRC licensing criterion of 280 cal/g UO_2. The 3 x 3 array of fuel rods irradiated to burnups of about 5300 MWd/t was positioned within a zircaloy flow shroud by a series of Inconel grid spacer. Preliminary data indicated that the power burst resulted in a radially averaged axial peak fuel enthalpy of 280 cal/g for the corner fuel rods, 255 cal/g for the side rods, and 235 cal/g for the center rod. The final value of the peak fuel enthalpy will be determined after fuel burnup analyses are completed.

A rapid increase in the coolant pressure from an initial value of 6.45 to 8.4 MPa occurred due to the direct moderator heating by the extremely high neutron and gamma flux during the power burst. The pressurization within the shroud expelled about 30% of the coolant from both ends of the flow shroud. As the pressure pulse decreased to the initial system pressure, normal upward flow was restored within 0.5 s. Heat transfer from the fuel rods then resulted in a second, gradual pressure increase to 8 MPa for about 10 s.

All nine rods failed as a result of the power burst, however, severe fuel rod fragmentation and loss of rod-like geometry did not occur. Fission gas induced fuel swelling, combined with cladding thickening and thinning, caused numerous cladding ruptures and the swelling of the eight peripheral rods of the bundle led to approximately 10% decrease in the coolant flow area. The center rod failure was apparently due to failure propagation. Expulsion of molten or nearly molten fuel fragments from an adjacent peripheral rod apparently induced local cladding melting of the center rod.

Conclusions regarding the test results are listed below. Since determination of the peak fuel enthalpy value is not final, all conclusions are tentative.

1. The fuel rod bundle geometry resulted in better cooling and possibly more relevant data than for separately shrouded fuel rods. Fuel swelling, cladding damage, and flow blockage were less severe than previously observed for separately-shrouded, previously irradiated fuel rods (where complete flow blockage occurred with otherwise the same test conditions and fuel enthalpy). However, fuel swelling out into the coolant channel was observed during Test RIA 1-4 dispite the relatively low burnup of the test rods.

2. Rod-to-rod failure propagation due to fuel expulsion was observed. Failure progagation and additional swelling in higher burnup rods could be a cause for concern - should a 280 cal/s RIA ever occur.

3. Prompt moderator heating followed by direct fuel rod-coolant heat transfer increased the coolant pressure by about 2 MPa. The initial pressure increase will inhibit steam void formation and reduce negative reactivity feedback during a RIA event in a BWR at hot startup condition.

4. The Experimental results from the RIA program indicate that the 280 cal/g radial average peak fuel enthalpy NRC limit for a RIA event may be nonconservative.

4. The Experimental results from the RIA program indicate that the 280 cal/g radial average peak fuel enthalpy NRC limit for a RIA event may be nonconservative.

Light water reactor control systems are presently designed such that if a reactivity initiated accident does occur, the resulting peak fuel enthalpy will be below 110 cal/g. The PBF results indicate that there is no safety problem with respect to loss-of-coolable geometry, fuel failure propagation, or molten fuel-coolant interaction as a result of RIA in a commercial power plant.

REFERENCES

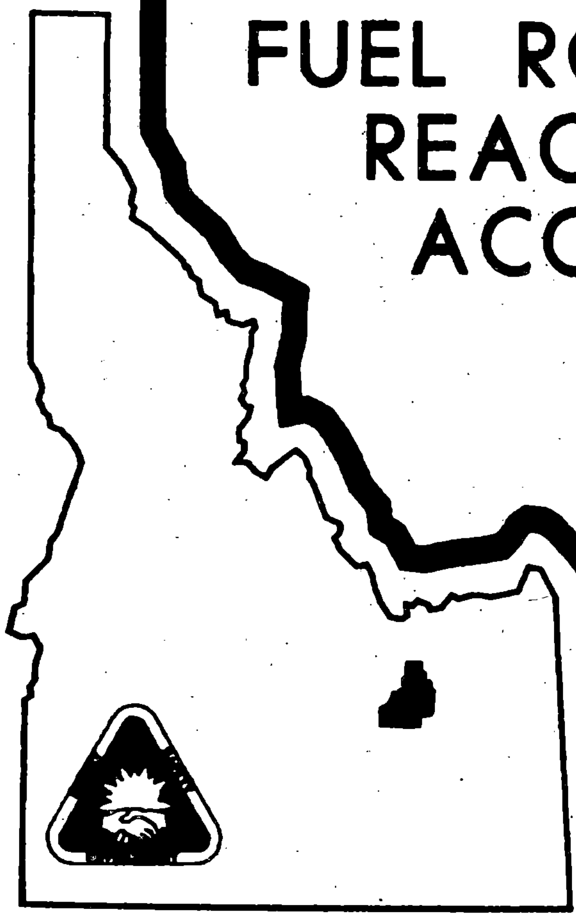
1. Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, NRC Regulatory Guide 1.77, May 1974.

2. Toshio Fujishiro et al., Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments, NUREG/CR-0269, TREE-1237 August 1978.

3. P. E. MacDonald et al., "Assessment of Light Water Reactor Fuel Damage During a Reactivity Initiated Accident," Proceedings of CSNI Specialist Meeting on Safety Aspects of Fuel Behavior in Off-Normal and Accident Conditions, Espoo, Helsinki, Finland, September 1-4, 1980.

4. R. S. Semken et al., Reactivity Initiated Accident Test Series, RIA Scoping Tests Fuel Behavior Report, NUREG/CR-1360, EGG-2024, April 1980.

RESPONSE OF PREIRRADIATED FUEL ROD BUNDLE DURING REACTIVITY INITIATED ACCIDENT TEST 1-4



Presented by

P.E. MacDONALD

EG&G Idaho, Inc.

OUTLINE

- SAFETY ISSUES
- PREVIOUS RIA TEST RESULTS
- TEST RIA 1-4 OBJECTIVES
- TEST CONDUCT
- TEST MEASUREMENTS
- POSTTEST CONDITION
- CONCLUSIONS

KEY REACTIVITY INITIATED ACCIDENT SAFETY ISSUES

- FUEL FAILURE THRESHOLD
- LOSS OF COOLABLE CORE GEOMETRY
- OVERSTRESS OF PRESSURE VESSEL

RESULTS FROM REACTIVITY INITIATED ACCIDENT TEST PROGRAM

- MODE AND CONSEQUENCES OF ROD FAILURE ARE SIGNIFICANTLY AFFECTED BY PRIOR IRRADIATION
  - IRRADIATED RODS FAIL DUE TO PELLET-CLADDING INTERACTION AT LOW ENERGY DEPOSITIONS
  - EXPANSION OF GASEOUS AND VOLATILE FISSION PRODUCTS INDUCE EXTENSIVE SWELLING OF MOLTEN IRRADIATED FUEL

RESULTS FROM REACTIVITY INITIATED ACCIDENT TEST PROGRAM (cont'd)

- COOLANT FLOW BLOCKAGE OCCURRED FOR IRRADIATED RODS TESTED AT 285 cal/g

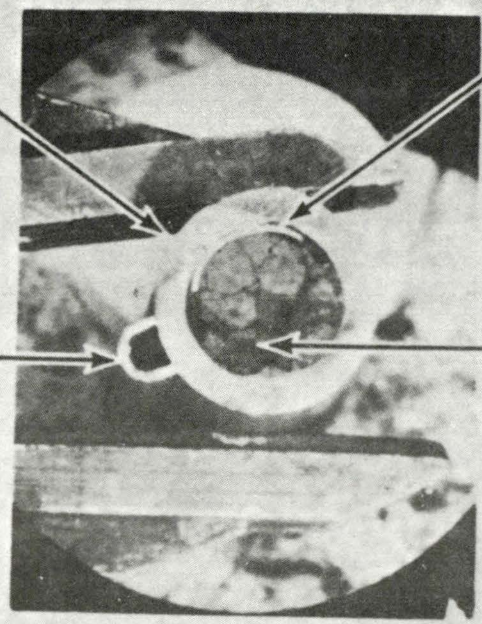
- CLADDING WALL THINNING, OXIDATION AND FRACTURE RESULTED IN LOSS OF ROD-LIKE GEOMETRY FOR UNIRRADIATED RODS TESTED AT 260 cal/g

RESULTS FROM REACTIVITY INITIATED ACCIDENT TEST PROGRAM (cont'd)

- COOLANT FLOW BLOCKAGE OCCURRED FOR IRRADIATED RODS TESTED AT 285 cal/g

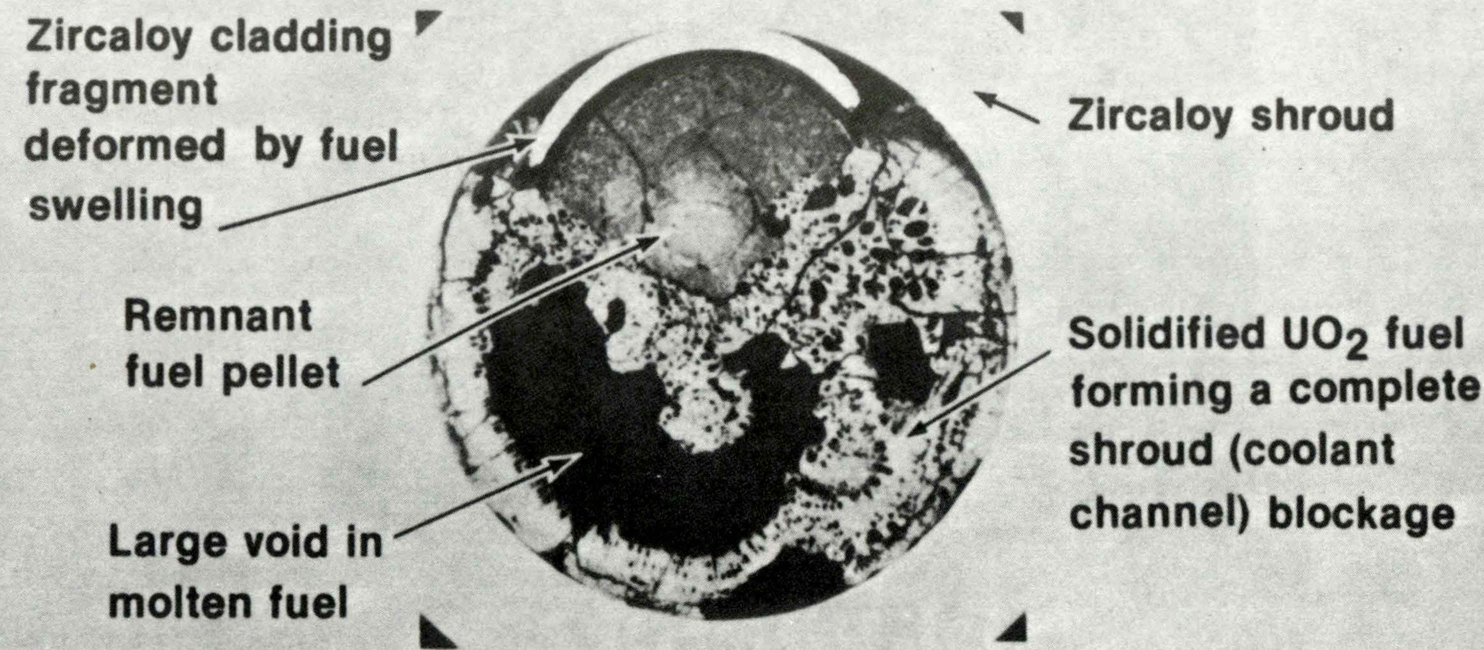
- CLADDING WALL THINNING, OXIDATION AND FRACTURE RESULTED IN LOSS OF ROD-LIKE GEOMETRY FOR UNIRRADIATED RODS TESTED AT 260 cal/g

Test RIA 1-1 Rod 801-1
(285 cal/g)



INEL-S-18 208

Test RIA 1-1 Rod 801-1



Peak Flux Region of RIA ST-2
(260 cal/g)

INEL-S-14 097

PROGRESSION OF FUEL ROD DAMAGE EVENTS

PELLET-CLADDING INTERACTION

CLADDING PLASTIC DEFORMATION

FUEL SWELLING AND THERMAL FAILURE

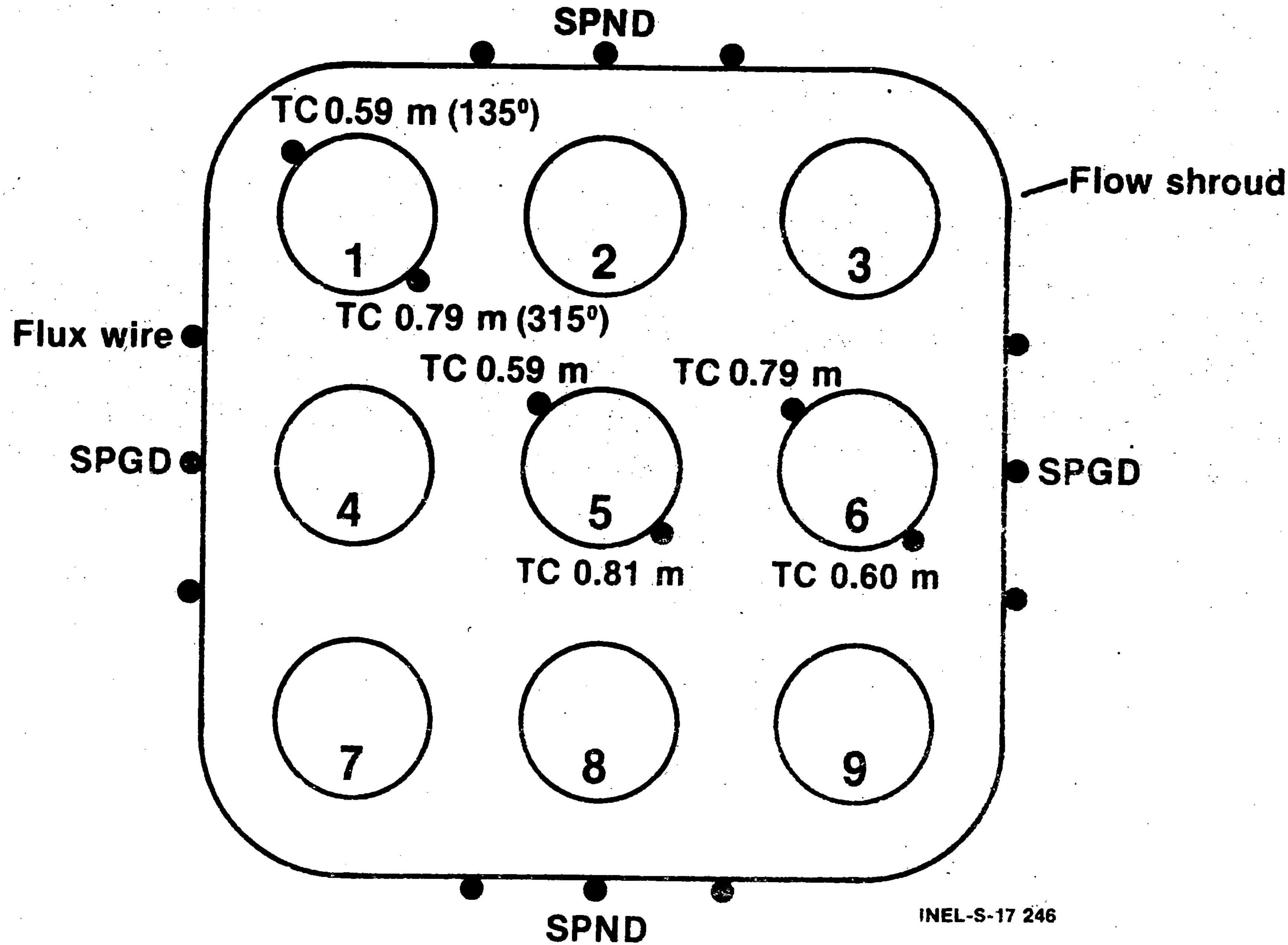
OXIDATION AND EMBRITTLEMENT

ROD FRAGMENTATION

TEST RIA 1-4 OBJECTIVE

DETERMINE FUEL COOLABILITY AND CHANNEL BLOCKAGE FOR NINE PREIRRADIATED FUEL RODS EXPOSED TO A PEAK FUEL ENTHALPY OF 280 cal/g UO_2

Test RIA 1-4 Rod Instrumentation



INEL-S-17 246

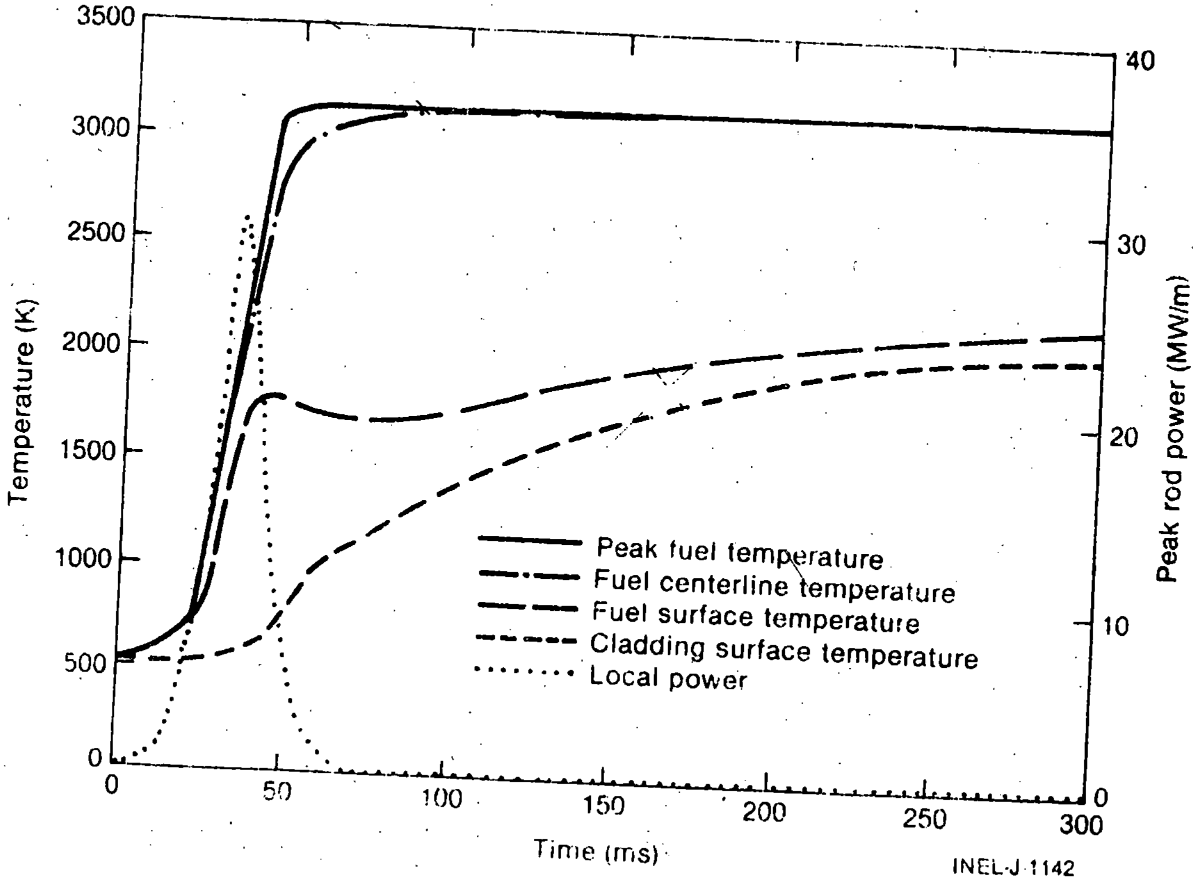
INITIAL CONDITIONS

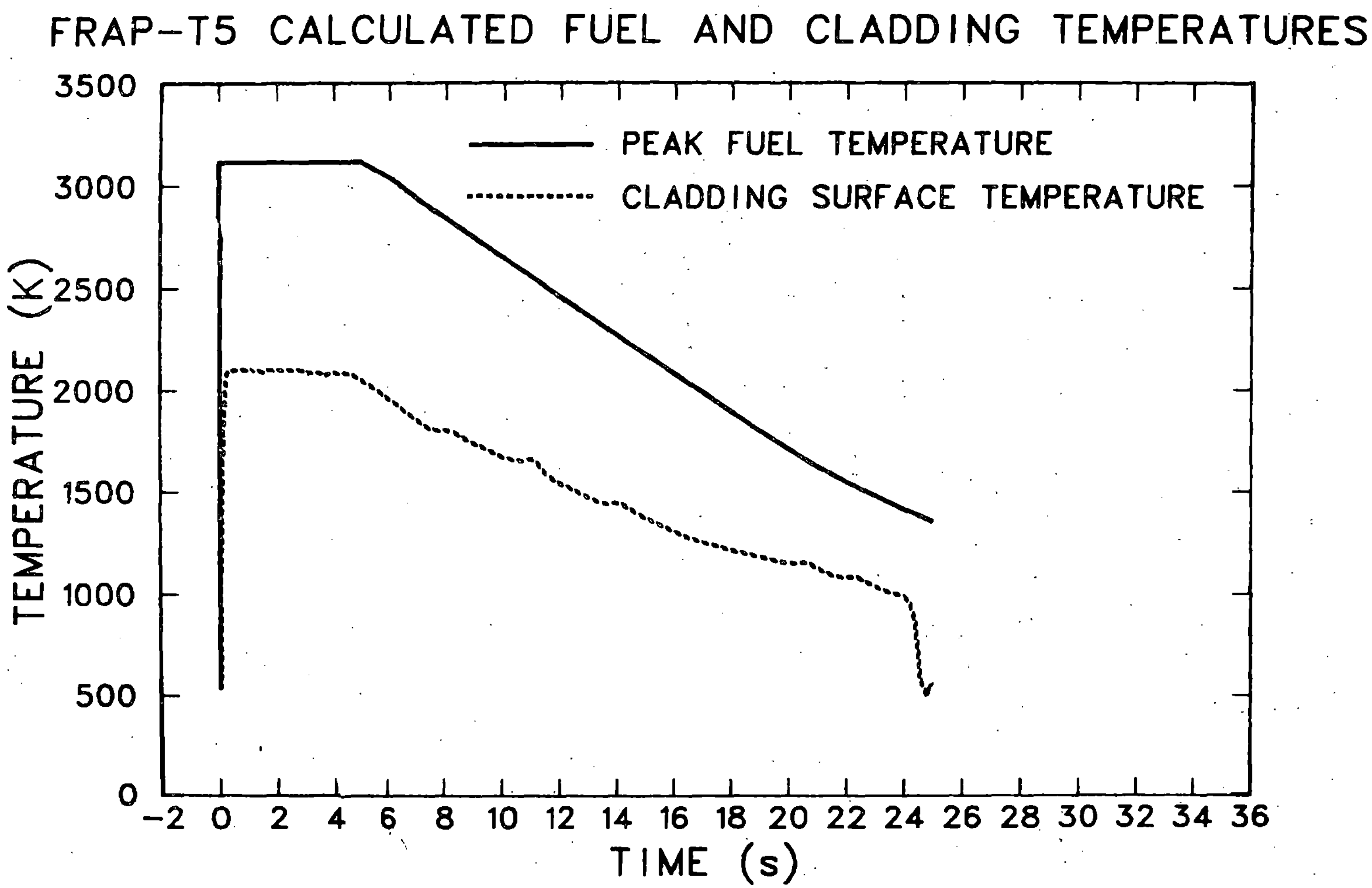
COOLANT TEMPERATURE	538 K
COOLANT PRESSURE	6.45 MPa
SHROUD FLOW	800 cm^3/s
ROD POWER	0

TEST RIA 1-4 ENTHALPY DISTRIBUTION

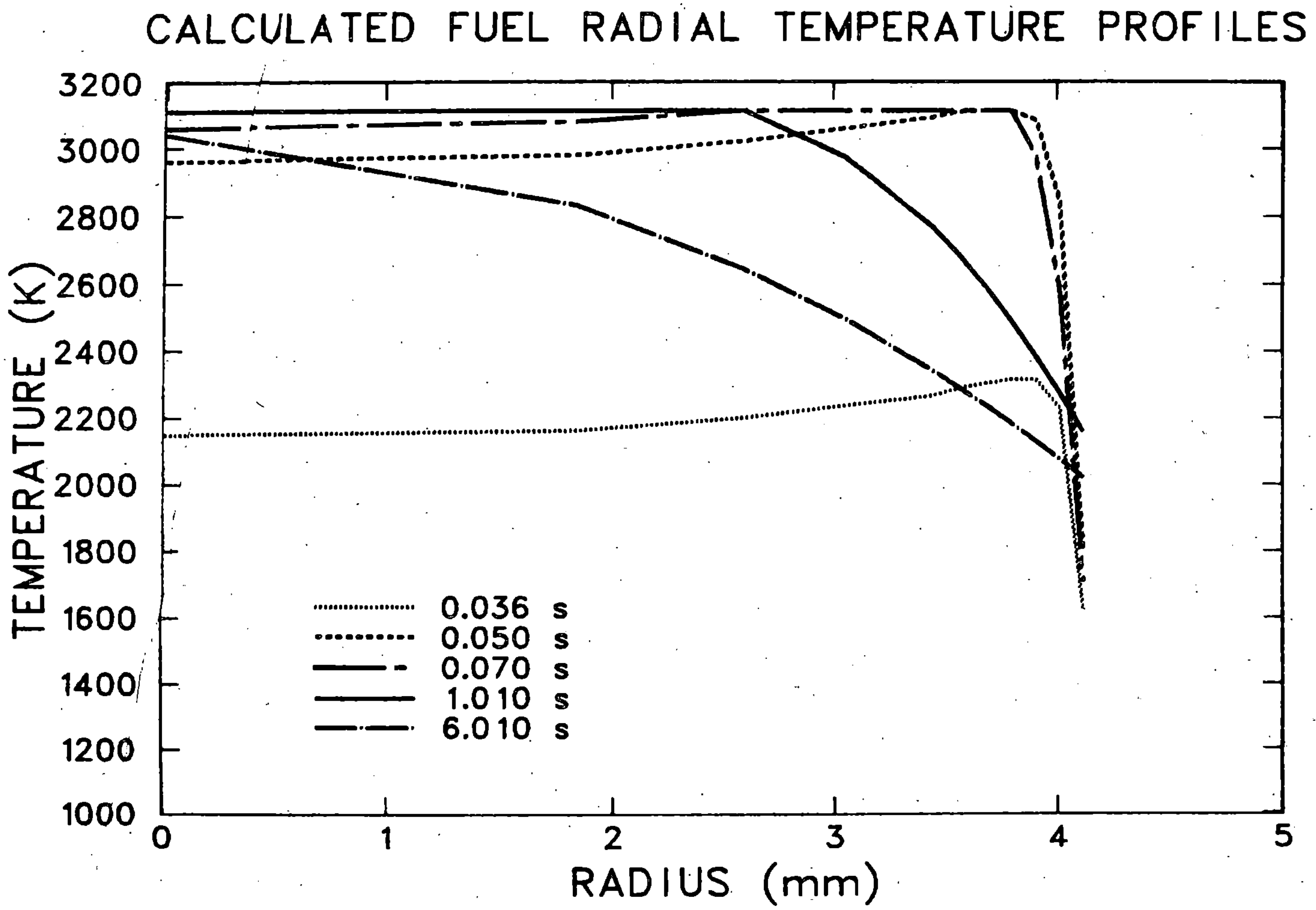
ROD POSITION	ROD AVERAGE PEAKING FACTOR	RADIALLY-AVERAGED PEAK ENTHALPY (cal/g)	MAXIMUM PELLET SURFACE ENTHALPY (cal/g)
CORNER	1.06	280	340
SIDE	0.97	255	300
CENTER	0.88	235	265

FRAP-T5 Calculated Fuel and Cladding Temperatures

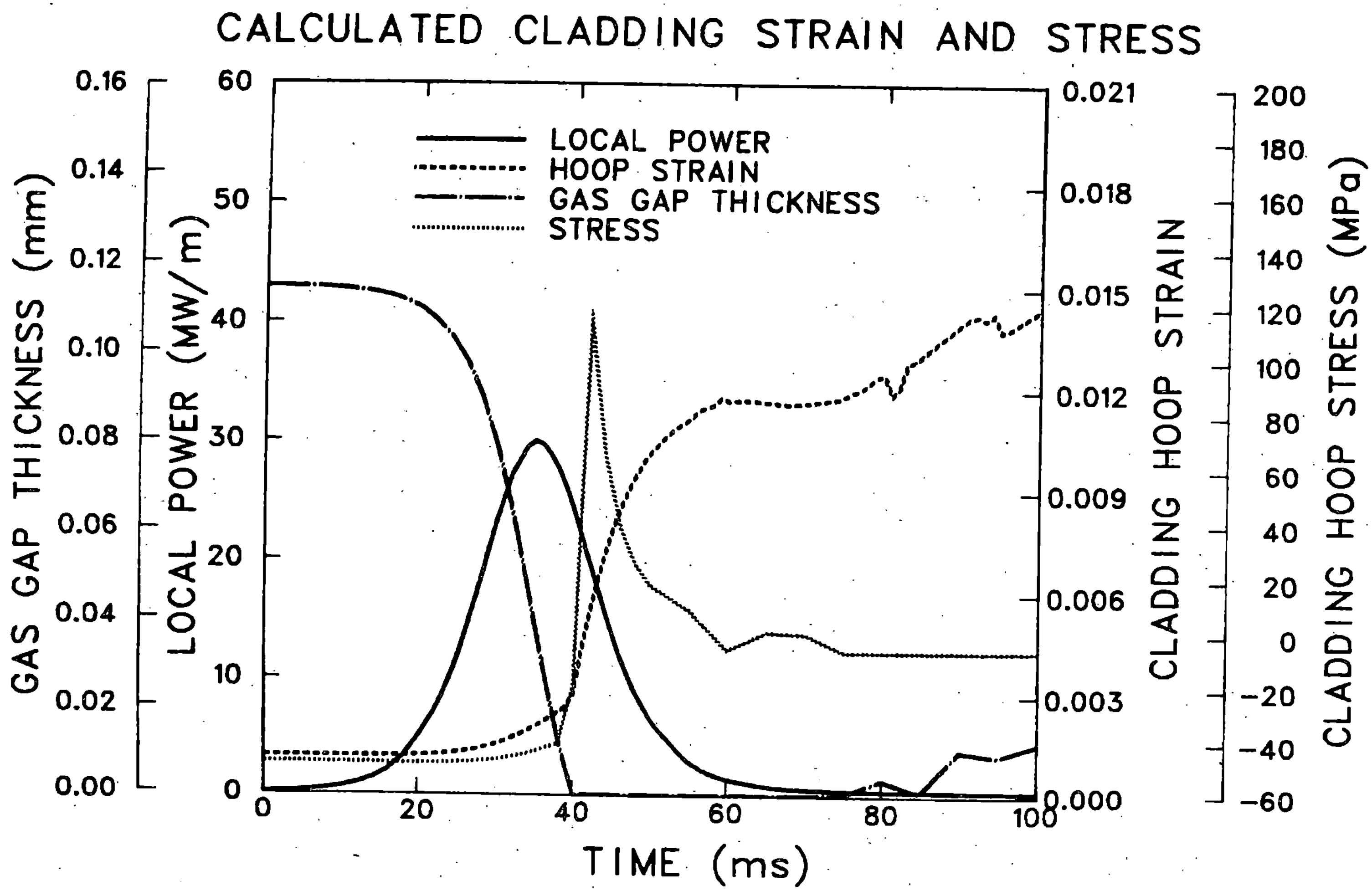


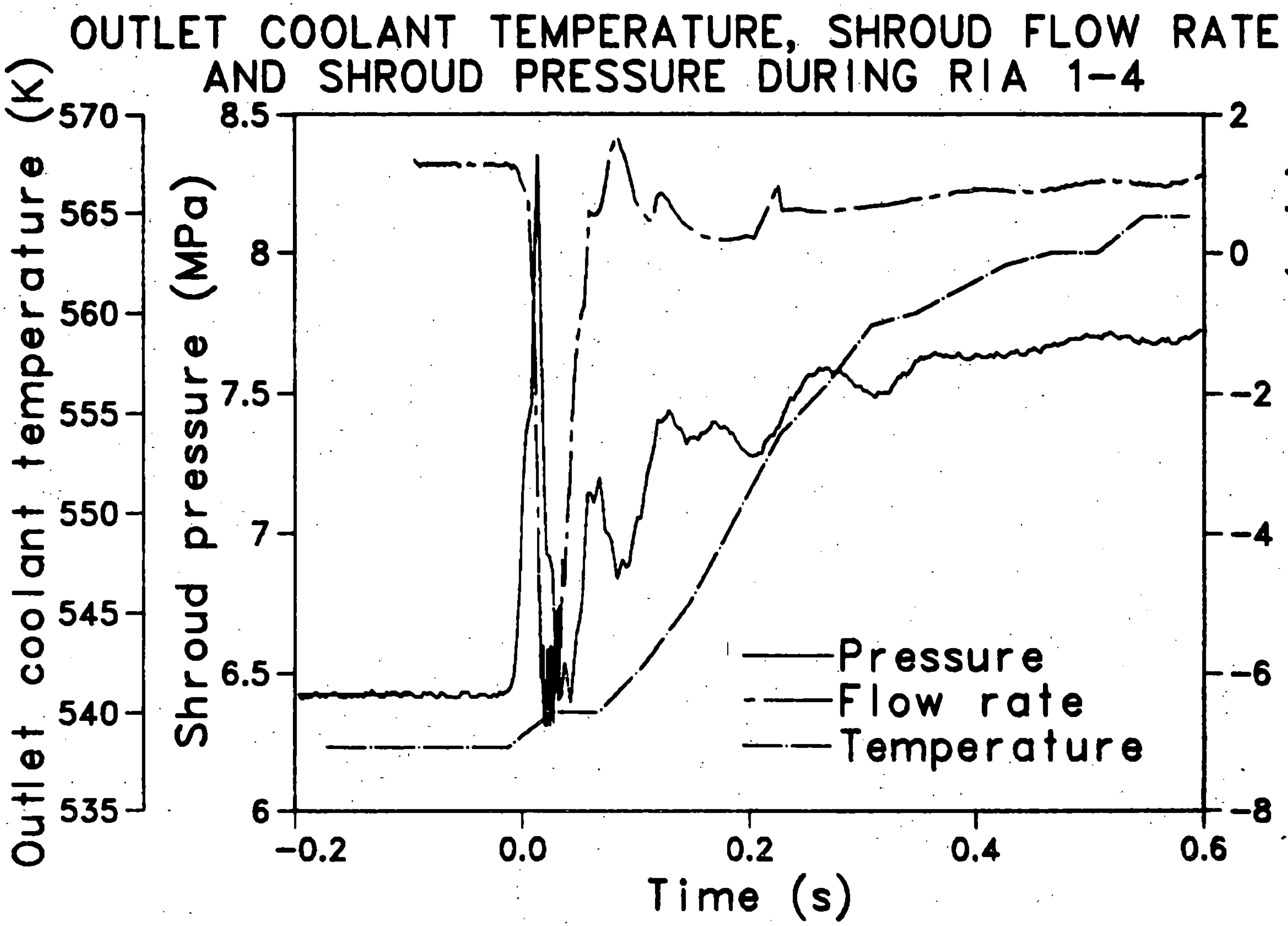


FRAP-T5 CALCULATED FUEL AND CLADDING TEMPERATURES
PEAK FUEL TEMPERATURE
CLADDING SURFACE TEMPERATURE
TEMPERATURE (K)
3500
3000
2500
2000
1500
1000
500
0
−2 0 2 4 6 8 10 12 14 16 18 20 22 24 26 28 30 32 34 36
TIME (s)



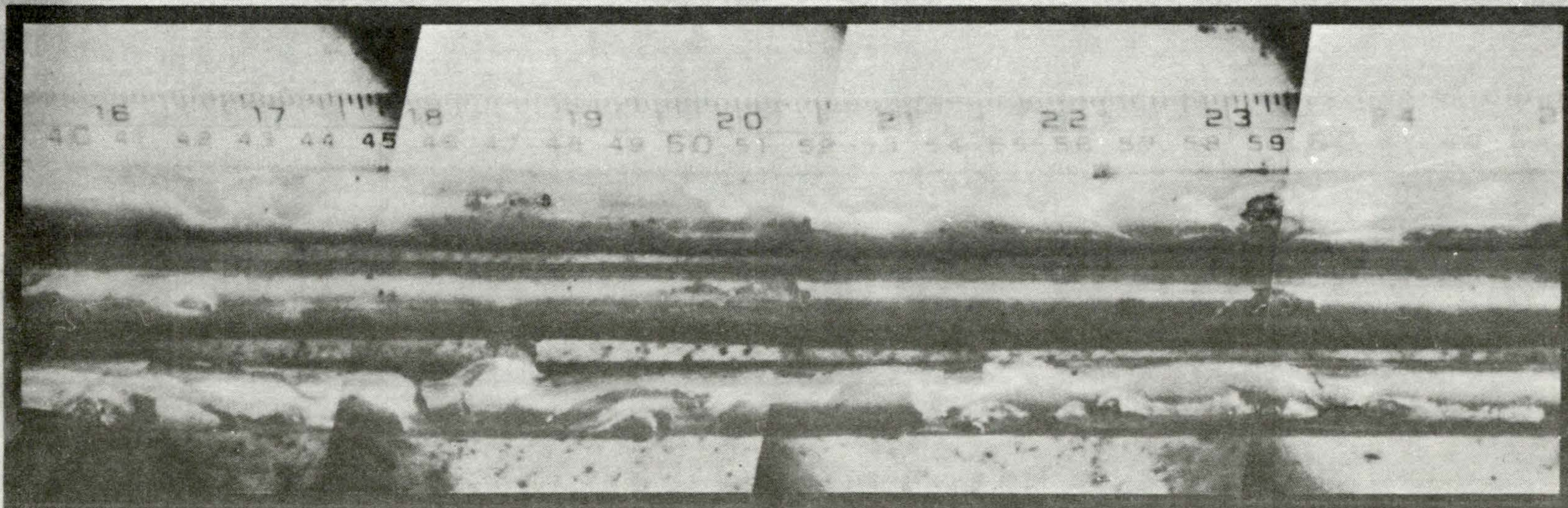
CALCULATED FUEL RADIAL TEMPERATURE PROFILES
3200
3000
2800
2600
2400
2200
2000
1800
1600
1400
1200
1000
TEMPERATURE (K)
0
1
2
3
4
5
RADIUS (mm)
0.036 s
0.050 s
0.070 s
1.010 s
6.010 s


CALCULATED CLADDING STRAIN AND STRESS
LOCAL POWER
HOOP STRAIN
GAS GAP THICKNESS
STRESS
GAS GAP THICKNESS (mm)
LOCAL POWER (MW/m)
CLADDING HOOP STRAIN
CLADDING HOOP STRESS (MPa)
TIME (ms)
0.16
0.14
0.12
0.10
0.08
0.06
0.04
0.02
0.00
60
50
40
30
20
10
0
0.021
0.018
0.015
0.012
0.009
0.006
0.003
0.000
200
180
160
140
120
100
80
60
40
20
0
-20
-40
-60
0
20
40
60
80
100



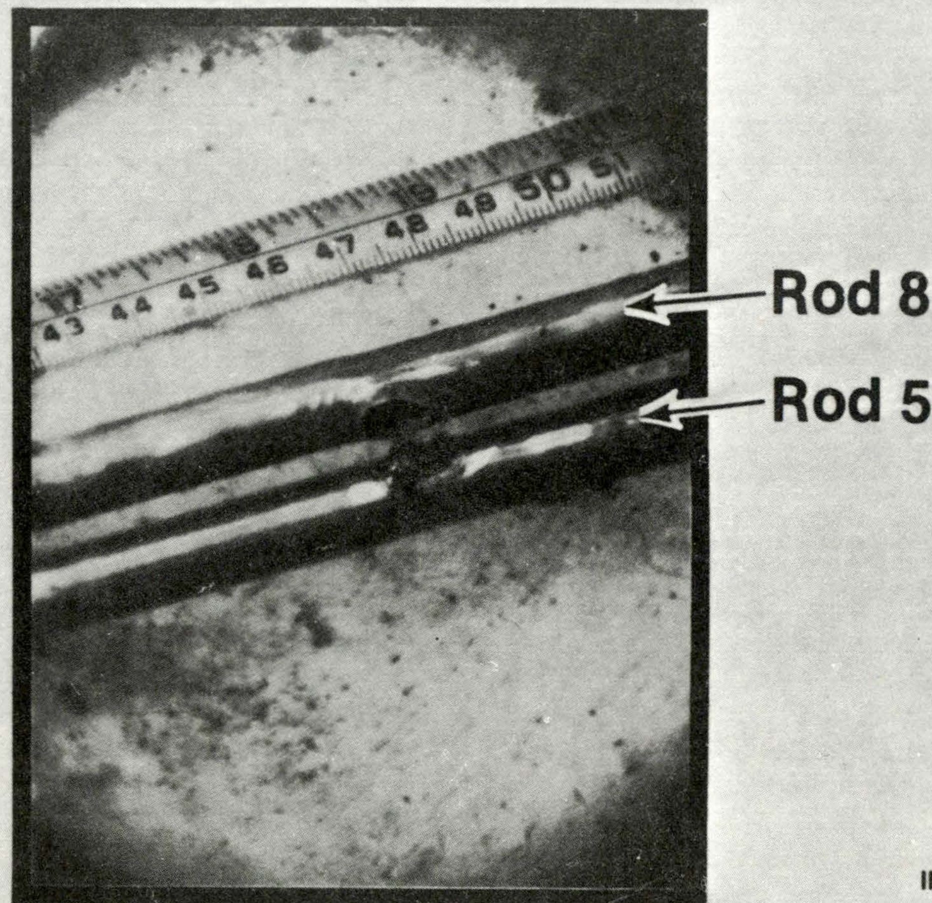
OUTLET COOLANT TEMPERATURE, SHROUD FLOW RATE
AND SHROUD PRESSURE DURING RIA 1-4
Outlet coolant temperature (K)
570
565
560
555
550
545
540
535
Shroud pressure (MPa)
8.5
8
7.5
7
6.5
6
Shroud flow rate (L/s)
2
0
−2
−4
−6
−8
−0.2
0.0
0.2
0.4
0.6
Time (s)
Pressure
Flow rate
Temperature

Post Test Photograph of RIA 1-4 Fuel Bundle (Rods 7, -8, -9)

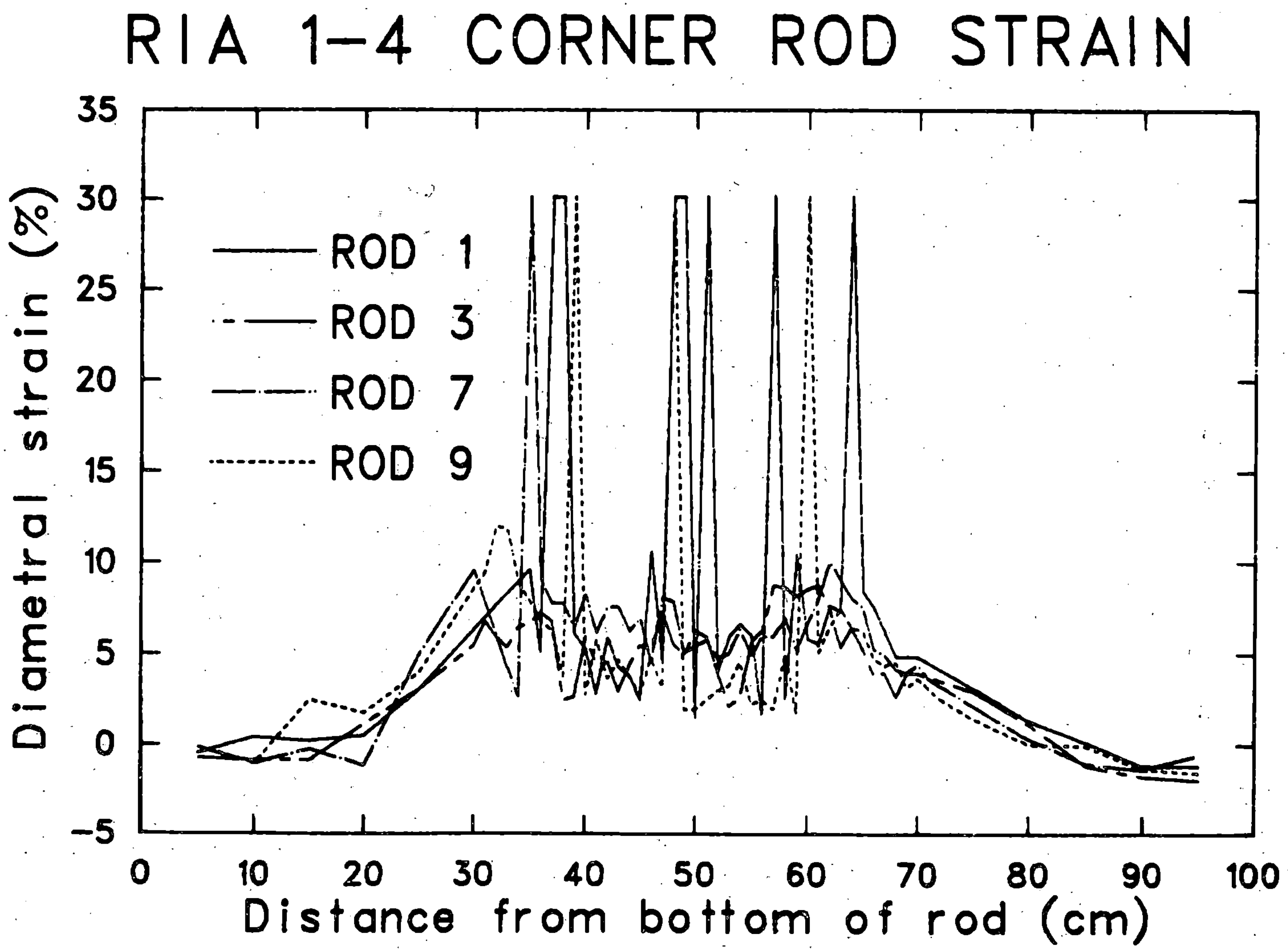


INEL-S-26 615

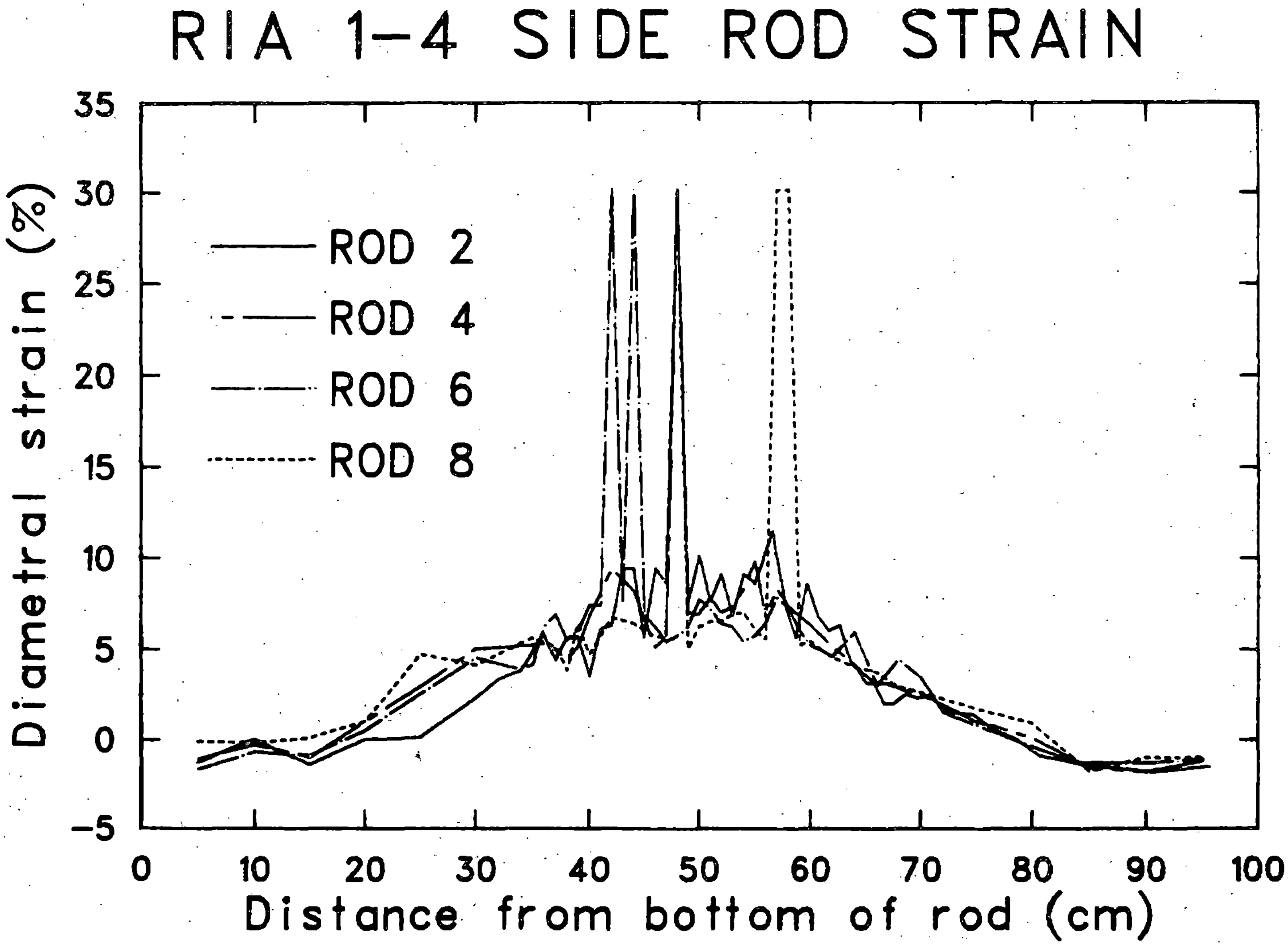
Test RIA 1-4 Fuel Rod Failure Propagation



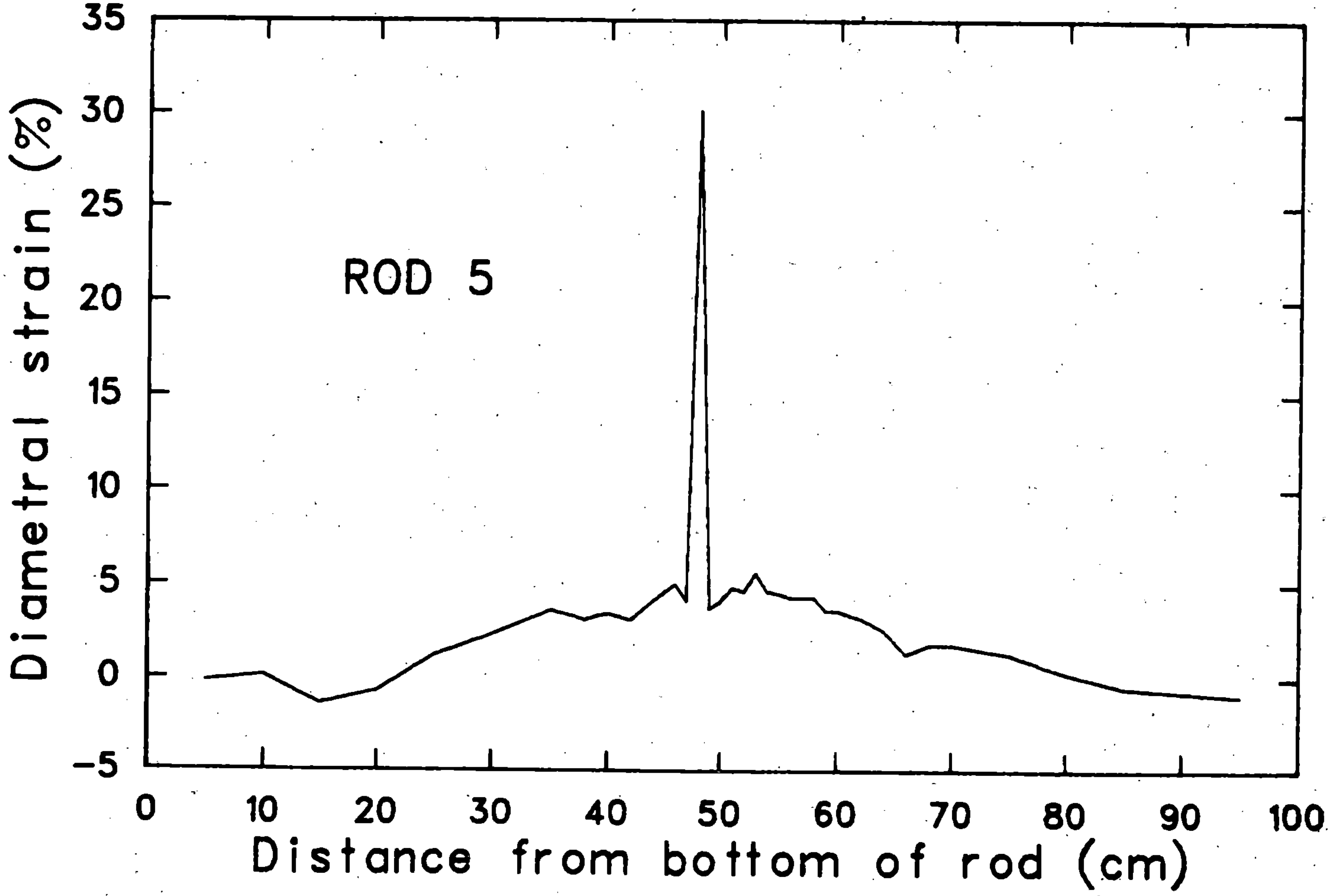
INEL-S-26 616



RIA 1-4 CORNER ROD STRAIN
ROD 1
ROD 3
ROD 7
ROD 9
Diametral strain (%)
-5
0
5
10
15
20
25
30
35
0
10
20
30
40
50
60
70
80
90
100
Distance from bottom of rod (cm)


RIA 1-4 SIDE ROD STRAIN
Diametral strain (%)
Distance from bottom of rod (cm)
ROD 2
ROD 4
ROD 6
ROD 8
35
30
25
20
15
10
5
0
-5
0
10
20
30
40
50
60
70
80
90
100

RIA 1-4 CENTER ROD STRAIN



SUMMARY OF PRELIMINARY RESULTS

- FUEL ROD BUNDLE GEOMETRY RESULTED IN BETTER COOLING THAN FOR SEPARATELY SHROUDED FUEL RODS

- PROMPT MODERATOR HEATING AND HEAT TRANSFERRED FROM BUNDLE TO COOLANT INCREASED COOLANT PRESSURE BY 2 MPa

SUMMARY OF PRELIMINARY RESULTS (cont'd)

- NO FUEL ROD FRAGMENTATION
- MODEST FISSION GAS INDUCED FUEL SWELLING
- ROD-TO-ROD FAILURE PROPAGATION DUE TO MOLTEN FUEL EXPULSION
- THE 280 cal/g NRC LICENSING CRITERIA MAY NOT BE CONSERVATIVE

FLOODING EXPERIMENTS IN BLOCKED ARRAYS

F E B A

Recent Results & Future Plans

by

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Federal Republic of Germanv

Presented at the
Eighth Water Reactor Safety Research Information Meeting
October 27 - 31, 1980
Gaithersburg , Maryland

SUMMARY

Introduction

The FEBA program is placed between two fields of reactor safety research: (1) Fuel Behavior in Water Reactors and (2) Emergency Core Cooling. It aims at providing data to understand and predict thermohydraulic phenomena in the core during the reflood phase of a LOCA under the condition that the core geometry is partly disturbed. Local subchannel blockages formed e.g. by ballooning of some fuel rod claddings influence the fluid flow and the local heat transfer during reflood. And, vice versa: If the deformation of the zircaloy cladding continues during emergency core cooling the final size and the shape of the blockages are influenced by the local cooling conditions.

The results obtained up to now have provided information about the following topics: (1) Difference between plate type and sleeve type blockages used in simulating ballooned fuel rods, (2) influence and extent of the influence of the blockages, (3) flow blockage effects with bypass, (4) influence of spacer grids, (5) axial temperature distributions, (6) azimuthal fluctuations in the cooling conditions, (7) steam superheat and presence of water in subchannels, (8) development and test of special devices, e.g. suitable for simulation of ballooned fuel rods, for measurement of transient two-phase flow conditions, and, rod instrumentation.

Program, Status and Future Plans

Three major steps characterize the separate effects test program: (1) Experiments with a 5-rod row, all subchannels blocked by the same blockage ratio including variation of size and shape of blockages. The test series are completed. They served mainly for qualitative studies of the cooling and the two-phase flow conditions and development of special devices. (See topics 1,2,7,8 mentioned above). (2) Experiments with a 5x5 rod bundle, partly blocked including variation in blockage size. Test series I, II and III are completed (base line, spacer grid effect, 90% partial blockage), series IV is under way (62% partial blockage). For investigation of blockage and spacer grid effects superimposed, series V will follow early 1981. (3) Experiments with a 10x5 rod bundle, partly blocked including variation in the ratio of blocked to unblocked area. All blockages used in the different arrays of the steps 1 to 3 are coplanar. The program, performed in the framework of the Nuclear Safety Project (PNS) at Karlsruhe is scheduled to end in 1982 after completion of these three steps.

Test Facility

The facility is designed for simulation of idealized reflood conditions in a PWR core using forced feed, system pressure and geometry as constant parameters during each test. System effects and interaction between cooling and deformation are excluded. Fuel rod simulations of German PWR dimensions, i.e. 3.9 m heated length with chopped cosine axial rod power distribution and 10.75 mm outer diameter are used. The rods are heated electrically by embedding heating elements in MgO. In the NiCr-cladding of 1 mm thickness thermocouples are embedded completely to avoid any coolant channel disturbance. Temperatures, pressures, pressure differences, flow rates, water carry over and rod power are recorded digitally at a scan frequency of 10 cycles/s.

Recent Results

Analysis of the data obtained from 46 tests with the 25 rod bundle concerns mainly: (1) Influence of a 90% as well as a 62% coplanar sleeve blockage on a 3x3 rod cluster, (2) spacer grid effects, (3) azimuthal fluctuations in cooling conditions at a rod within the bundle, (4) transient two-phase flow conditions.

The following reflood conditions have been used for the 4 test series mentioned: system pressures of 2, 4.1 and 6.2 bar, flooding velocities of 3.4 and 5.2 cm/s, initial clad temperatures of about 780 C, and, initial housing temperatures of about 650 C.

(1) Influence of a partial blockage, 90% local blockage at 3x3 rod cluster in the 5x5 bundle: Increase of the maximum clad temperatures 20 K (average) or 50 K (maximum) 10 mm downstream of the blockage top end. The values apply to a system pressure (p) of 4 bar and a flooding velocity (v) of 3.4 cm/s. The decrease of heat transfer downstream of the blockage is time-dependent. The ratio of the heat transfer coefficients (HTC) blocked to HTC unblocked decreases during flooding from about 1.0 at the very beginning of flooding to about 0.65 a short time before quenching of the corresponding region. A 62% local blockage in the same 3x3 rod cluster shows similar, but much smaller effects. In this case quenching of the sleeves occurs earlier than quenching of the rods in the bypass area. Influenced by the blockages different quenching phenomena occur due to different precursory cooling and conduction controlled rewetting.

(2) Spacer grid effects: There is no steady increase in the axial temperature profile. Due to the spacer grids the axial temperature profile shows local temperature decrease downstream of each spacer grid. At the time of maximum temperature at the corresponding level the deviations from a "mean" axial temperature profile are: about + 15 K 150 mm upstream of the spacer, about - 30 K 100 mm downstream of the spacer. The local increase of heat transfer downstream of a spacer grid is time-dependent. At the beginning of flooding the ratio of HTC with to HTC without spacer grid is between 1.15 and 1.5 depending on the system pressure and the flooding velocity (2 - 6 bar, 3.4 - 5.2 cm/s). This ratio decreases down to 1.0 at the onset of film boiling.

(3) Azimuthal fluctuations of cooling conditions at a rod within the bundle are due to flow instabilities. The instantaneous local heat transfer coefficients fluctuate up to ± 10% about the mean value which is identical for the whole rod circumference. The frequencies vary between 0.1 and 0.5 cycles/s. In few tests synchronous flow oscillations occur for a period of up to one minute in the whole bundle. The frequency range of such flow pulsations is the same as mentioned above. The maximum clad temperatures, and the quenchtime are not influenced by these effects, but, local clad temperatures are affected over a certain period.

(4) Transient two-phase flow conditions have been analyzed investigating local steam superheat and presence of water. Steam temperatures are approximately 200 K lower than the clad temperatures in the middle portion of the bundle. Droplets are entrained through the total bundle a few seconds after initiation of reflood. At low flooding velocities the uppermost portion of the bundle is heated up by superheated steam rising from the middle portion of the bundle.

Conclusion

Up to now results show that even a severe flow blockage does not lead to significantly higher clad temperatures and, the bundle size is relatively small. The flow diversion around blocked rod clusters of larger extent may lead to somewhat different results. Moreover, due to e.g. the quenching behavior of a fuel rod different from that of a simulated rod usually used for reflood experiments the cooling history might be slightly different for identical reflood conditions. Nevertheless, taking into account the boundary conditions defined for the FEBA tests, the results show the order of magnitude of flow blockage effects with bypass. The data contribute to a better understanding of the transient two-phase flow phenomena and, they contain quantitative information to check and improve reflood code models.

Acknowledgment

Significant support was provided by W. Goetzmann, G. Hofmann, K. Kreuzinger, S. Malang, M. Politzky, K. Rust and H. Schneider discussing the program, designing the test facility, performing the tests, reducing and analyzing the data and, discussing the results. The cooperation is greatfully acknowledged.

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1. Ihle, P.; Rust, K.; "Einfluß der Stababstandshalter auf den Wärmeübergang in der Flutphase eines DWR-Kühlmittelverluststörfalles". Jahrestagung Kerntechnik 80. Reaktortagung 1980. Berlin, 25. - 27. März 1980. Kerntechnische Ges. e.V. Deutsches Atomforum e.V. Eggenstein-Leopoldshafen: Fachinformationszentrum Energie, Physik, Mathematik 1980. S. 145 - 48

2. Ihle, P.; Müller, St.; "Transient two-phase flow conditions in heated rod bundles". ANS Topical Meeting on Thermal Reactor Safety, Knoxville, Tenn., April 8-11, 1980

3. Ihle, P.; Politzky, M.; Rust, K.; "FEBA-Flooding experiments with blocked arrays. Heat transfer in partly blocked 25-rod bundle". 19. National Heat Transfer Conf., Orlando, Fla., USA. July 27 - 30, 1980, ASME HTD-Vol: 7, pp 129 - 138

4. Rust, K.; Ihle, P.; "Heat transfer and fluid flow during reflooding of blocked arrays". Topical Meeting on Nuclear Reactor Thermal Hydraulics, Saratoga, N.Y., October 7 - 9, 1980

5. Ihle, P.; Müller, St.; "Experimence with steam temperature and water detection probes for transient mist flow in hot rod bundles". Topical Meeting on Nuclear Reactor Thermal Haydraulics, Saratoga, N.Y., October 7 - 9, 1980

FLOODING EXPERIMENTS IN BLOCKED ARRAYS
F E B A

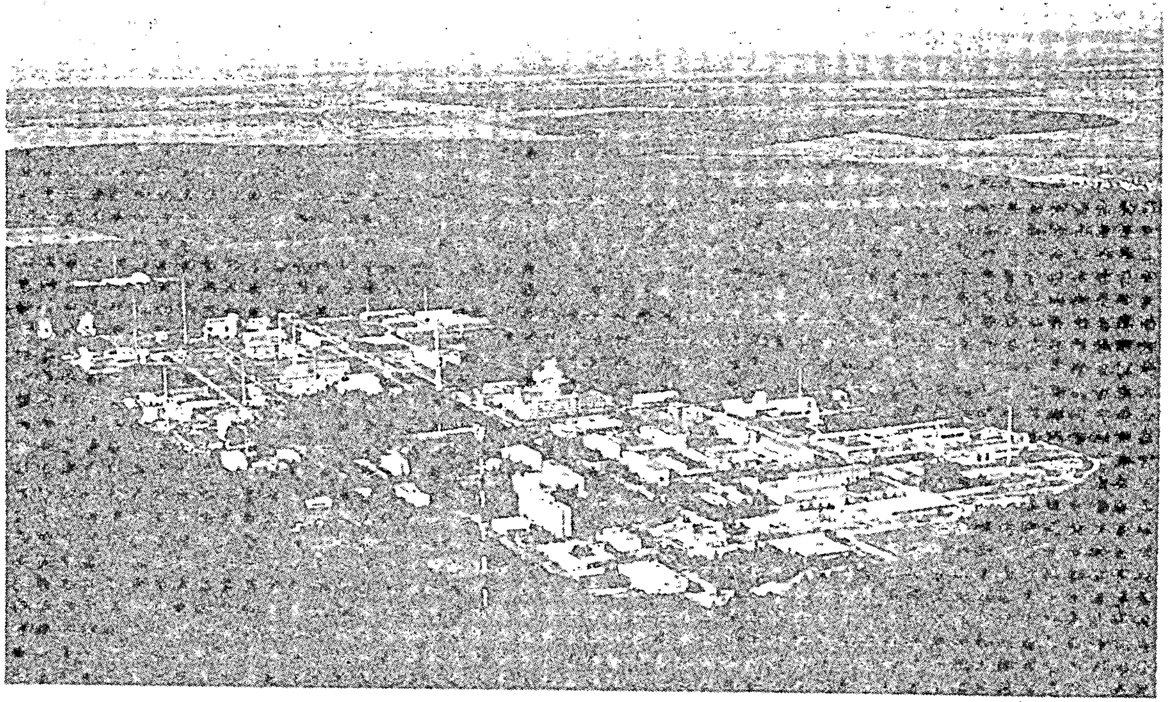
presented by:

PETER IHLE

Kernforschungszentrum

Karlsruhe

Germany



KfK

RECENT RESULTS and FUTURE PLANS

F E B A contributes to the topics:

* Difference between plate and sleeve blockages simulating "balloons"

* Flow blockage effects with by-pass

* Influence of spacer grids

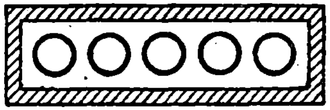
* Axial temperature distributions

* Azimuthal fluctuations of the cooling conditions

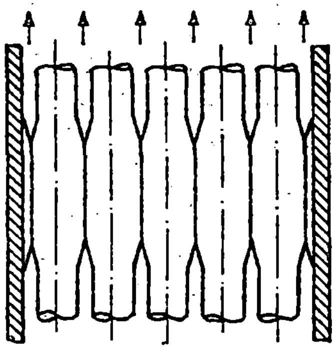
* Steam superheat and presence of water

* Water carry over during reflood

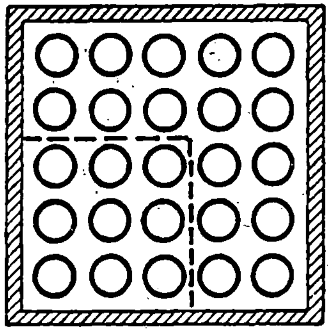
* Development & test of special devices
  - sleeves to simulate "balloons"
  - rod instrumentation
  - steam superheat and
  - water detection probes



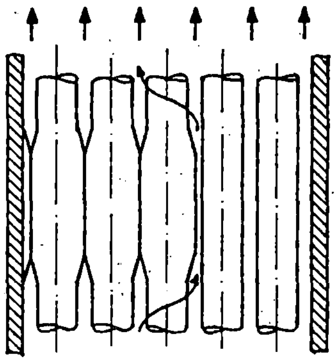
tests completed



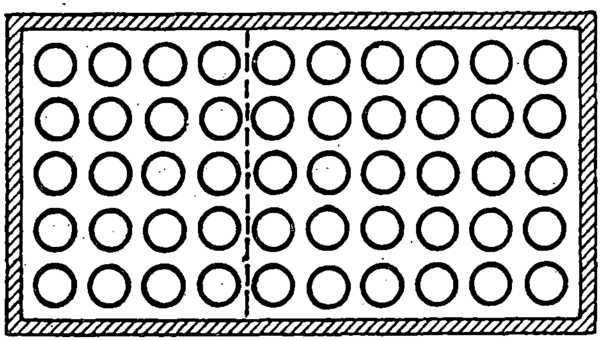
- all subchannels unblocked
- all subchannels partly blocked
  variables: blockage shape
  subchannel blockage ratio



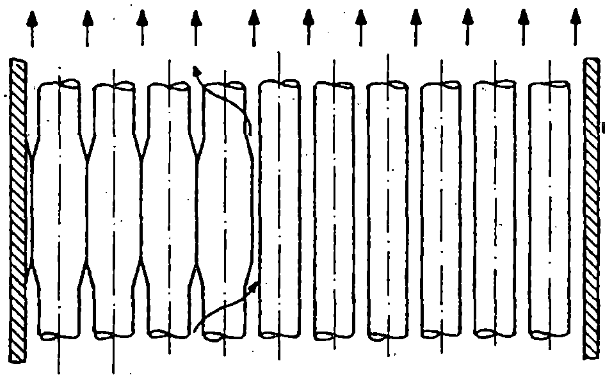
tests under way



- all subchannels unblocked
- effect of a spacer grid
- effect of a partial blockage
  variable: subchannel blockage ratio



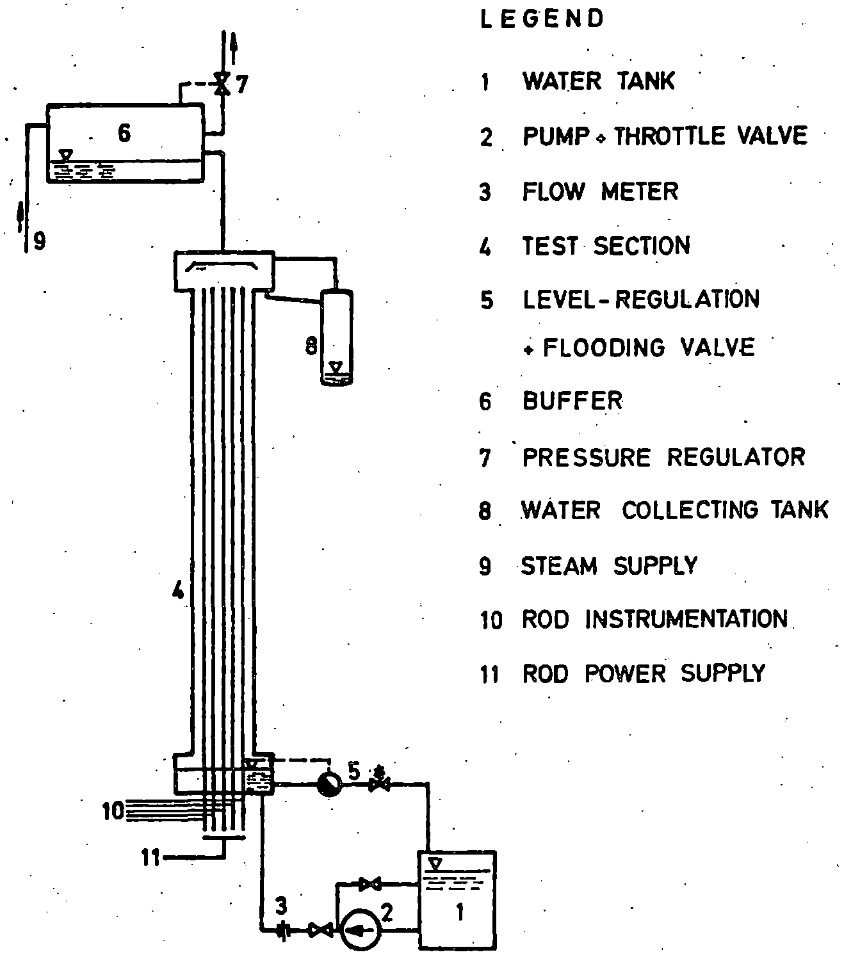
tests under preparation

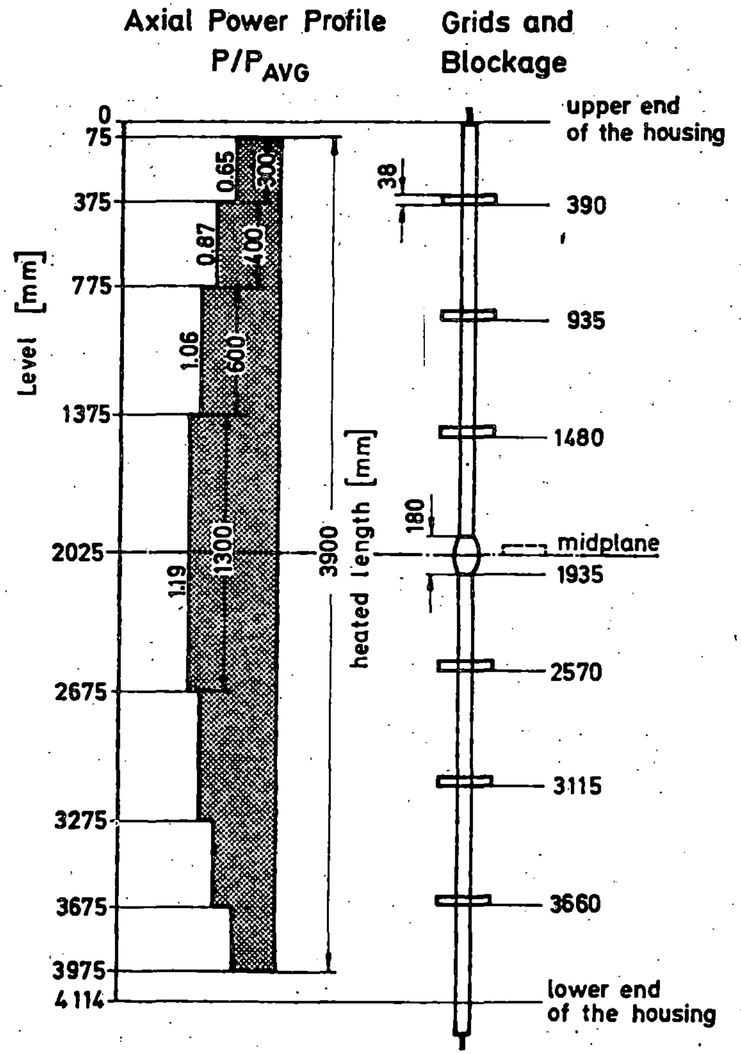


- all subchannels unblocked
- effect of a partial blockage
  variables: ratio blockage / bypass
  subchannel blockage ratio

KfK IRB

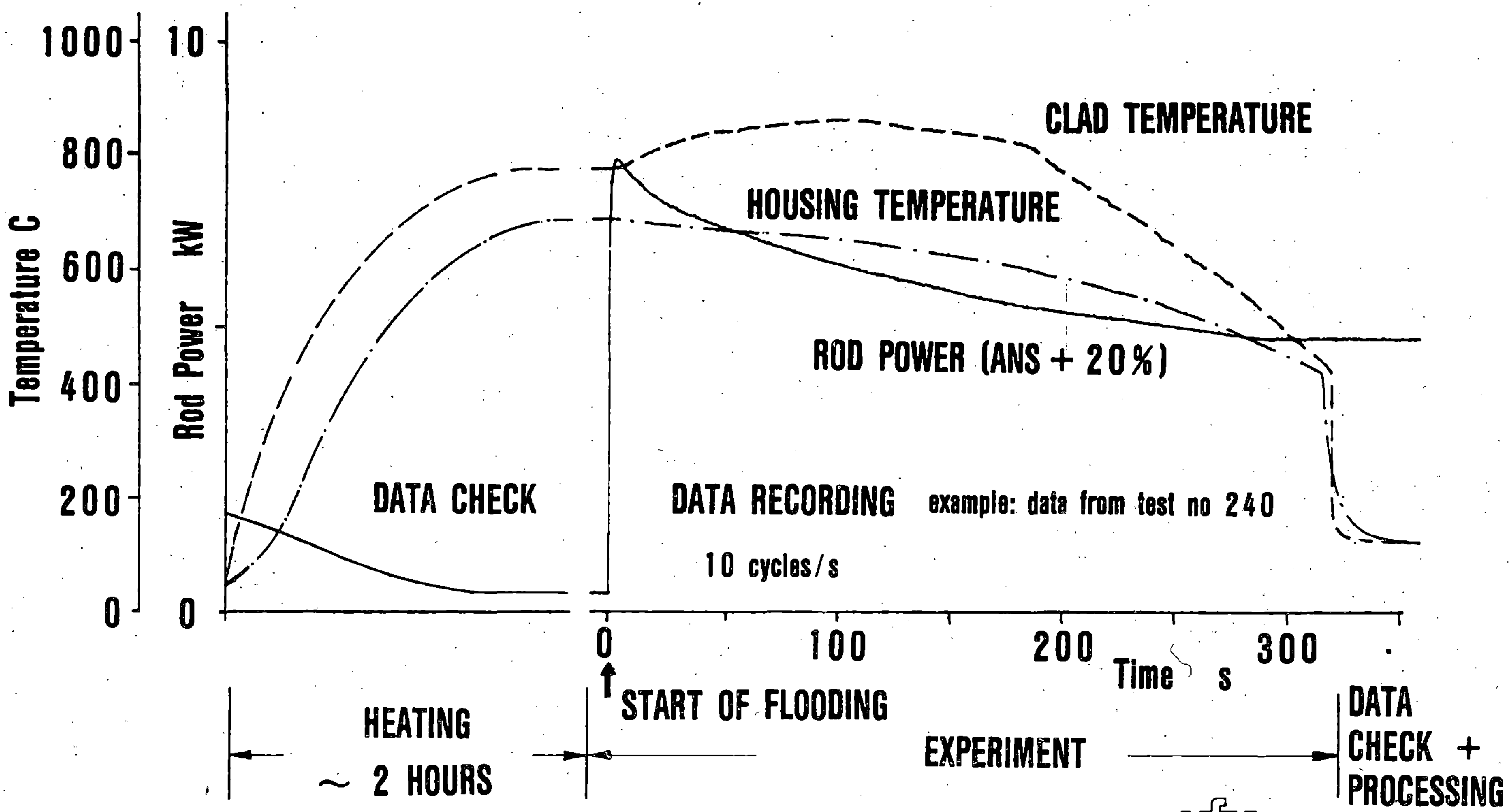
FEBA main features of bundle tests





F E B A TEST RIG SCHEMATIC AXIAL HEATER ROD LAYOUT

for full length rod bundles of 5 to 50 rods



Operation of the Flooding Experiments with Blocked Arrays

MEASURED VALUES:

CLAD TEMPERATURE

FLOODING DATA

FLUID DATA

WATER PRESENCE

STEAM SUPERHEAT

CALCULATED VALUES:

CLAD SURFACE TEMPERATURES

SURFACE HEAT FLUX

HEAT TRANSFER COEFFICIENTS

LINEAR ROD POWER

STORED HEAT

COMPARISON HEATER / FUEL ROD

SIMULATION QUALITY

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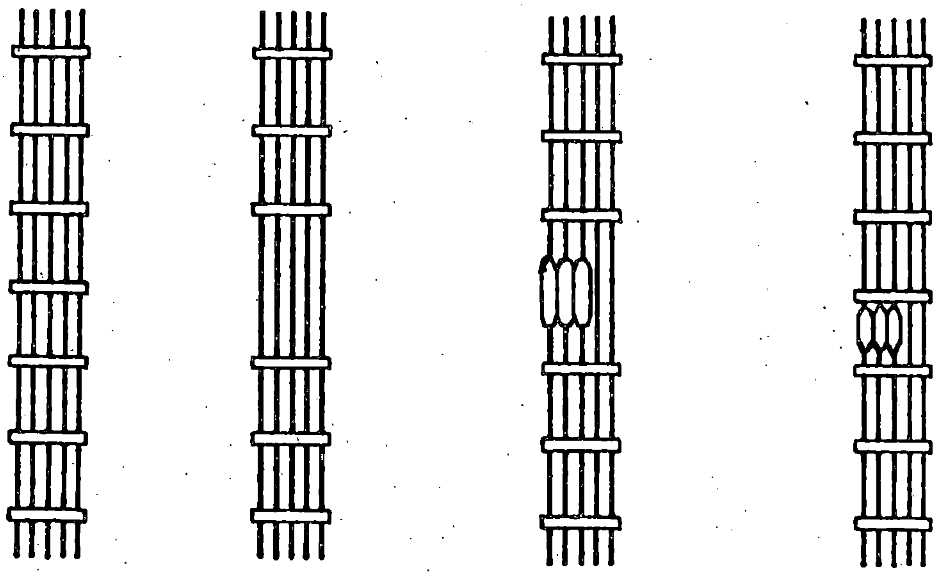
EVALUATION

GEOMETRY

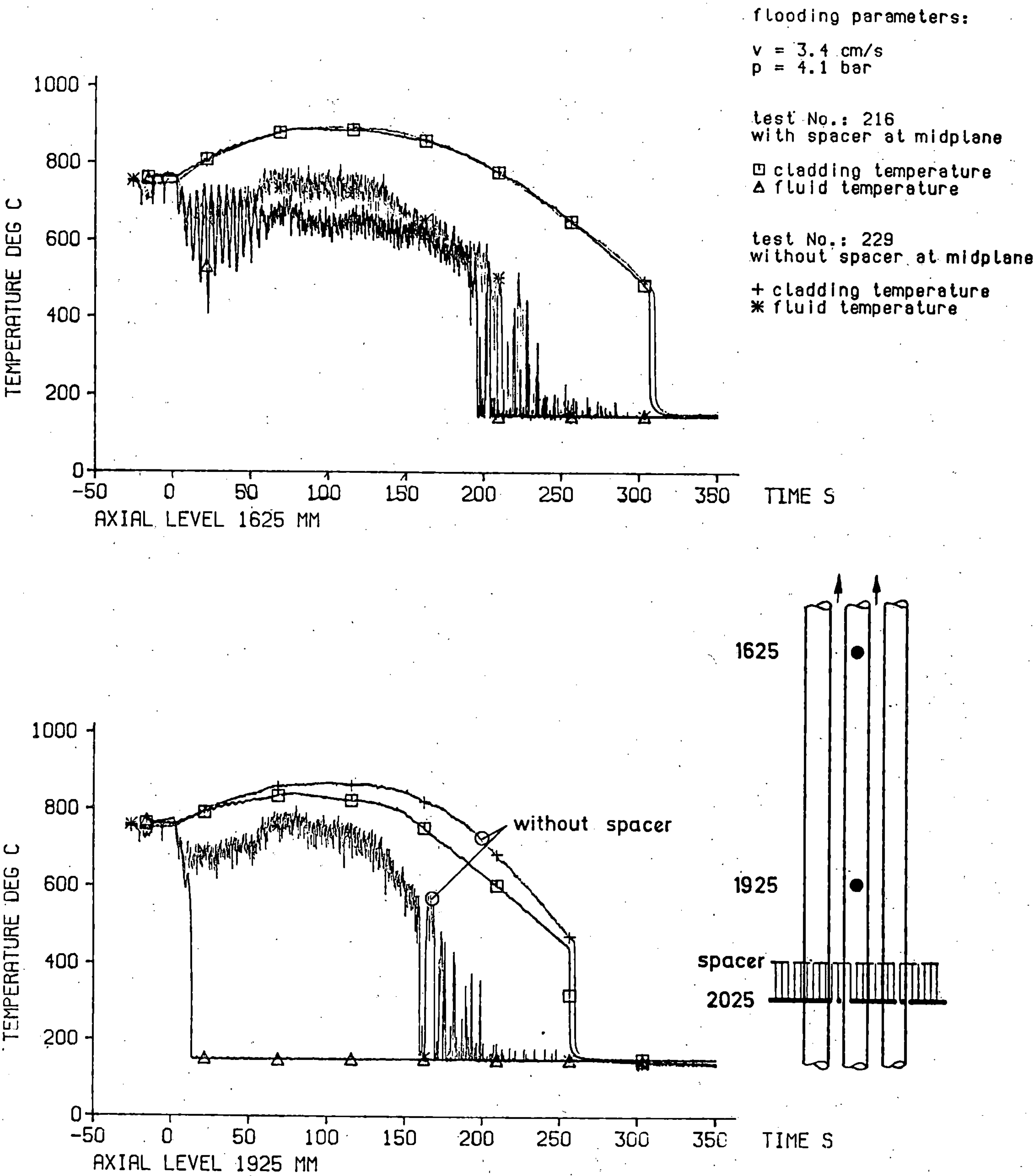


Series	I	II	III/IV	V
	base line	spacer grid	90 % bl. 62 % bl. at 3x3	spacer +90% bl. at 3x3

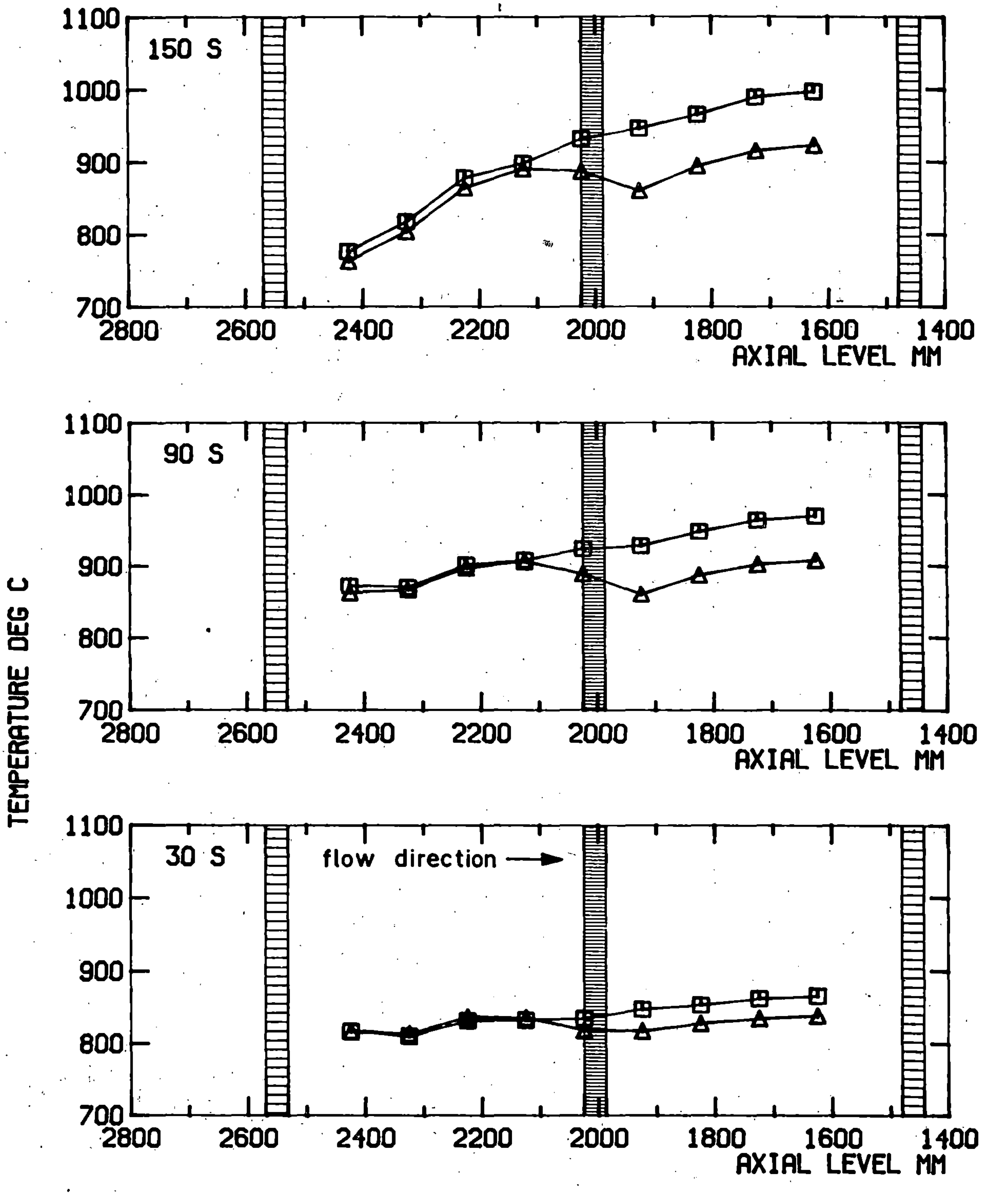
FLOODING CONDITIONS

System pressures: 2, 4, 6 bar
Flooding velocities: 3.4, 5.2 cm/s
(in the cold test section)
Initial clad temperatures: 700 to 800 C
Initial housing temp.: 600 to 700 C
Water inlet temperature: 40 C
Rod power: decay heat transient 120 % ANS

25-Rod Bundle (5x5) Test Parameters



5x5 ROD BUNDLE: INFLUENCE OF A SPACER GRID ON THE TRANSIENT CLADDING AND FLUID TEMPERATURES



flood parameters: v = 3.4 cm/s, p = 2.0 bar

△ test 223,   (with spacer grid at midplane)
⊡ test 234, (without spacer grid at midplane)

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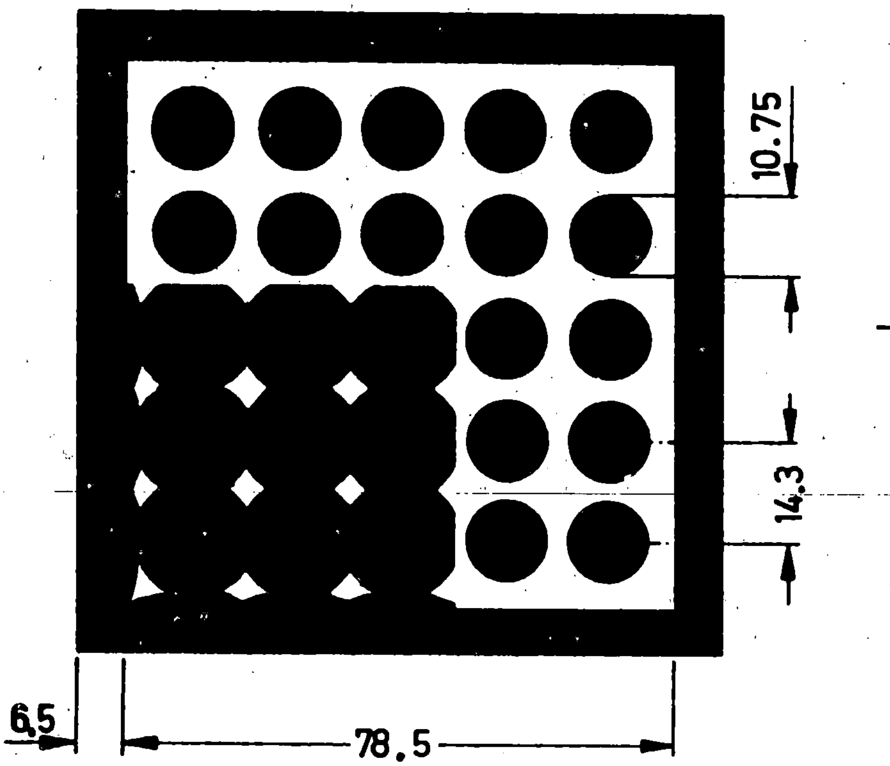
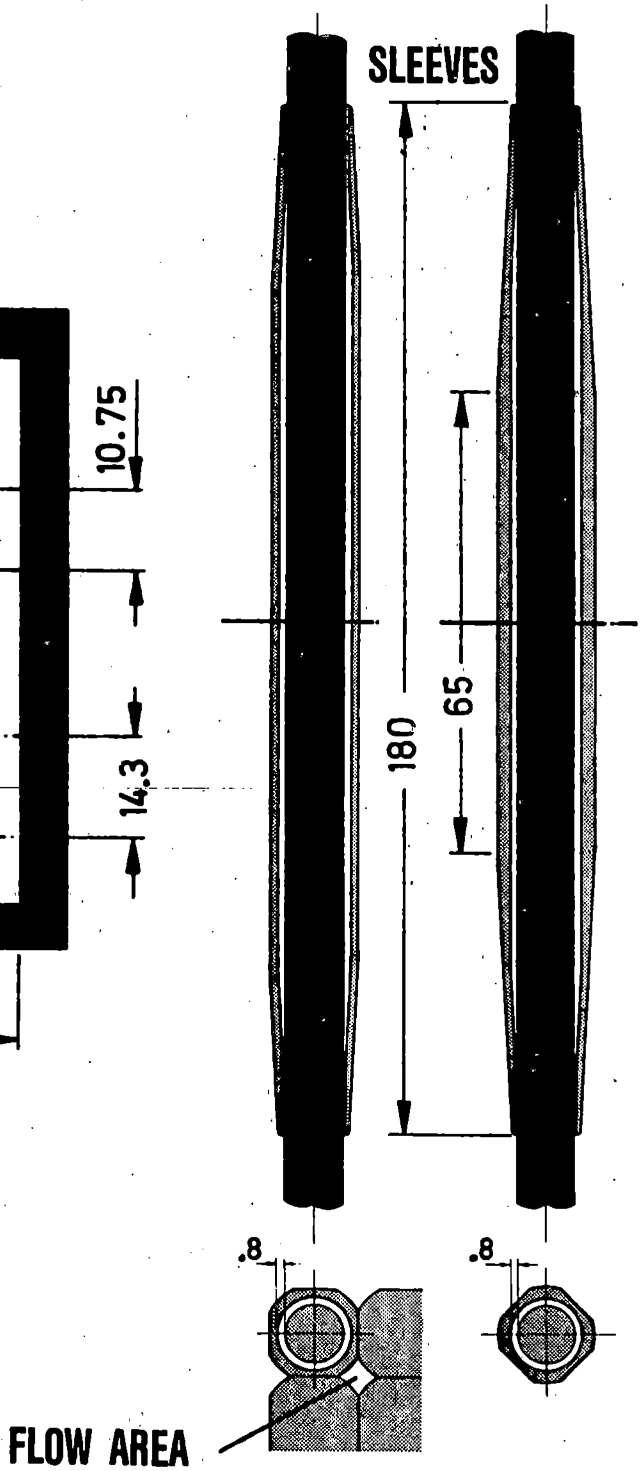
INFLUENCE OF A SPACER GRID ON THE AXIAL TEMPERATURE PROFILE

Cross Section at Midplane of the Bundle

local blockage ratio	90%
overall blockage ratio	31%





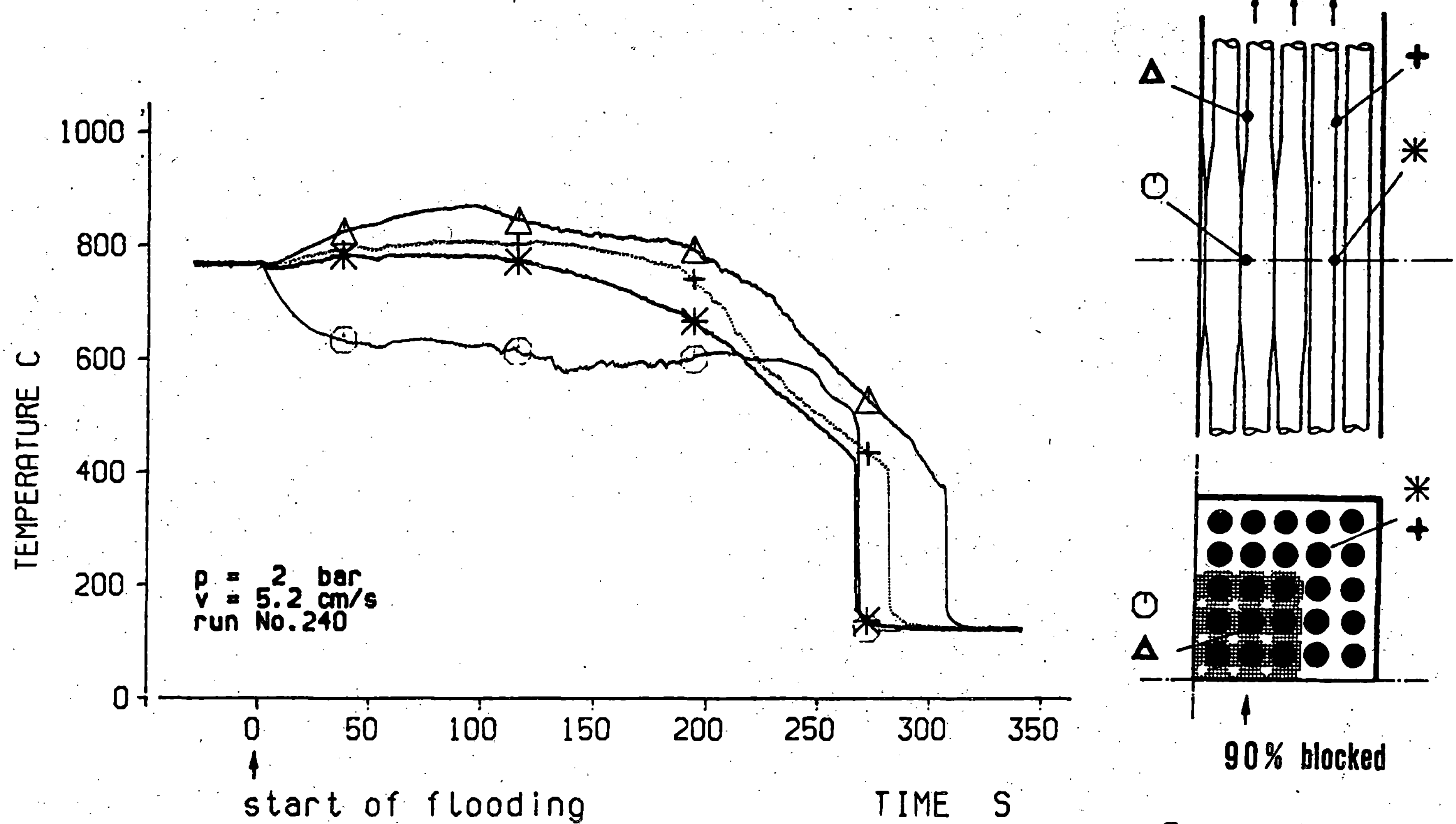
Bundle Data:

pitch	14.3 mm
rod diameter	10.75 mm
heated length	3900 mm

X: rods for comparison blocked / unblocked

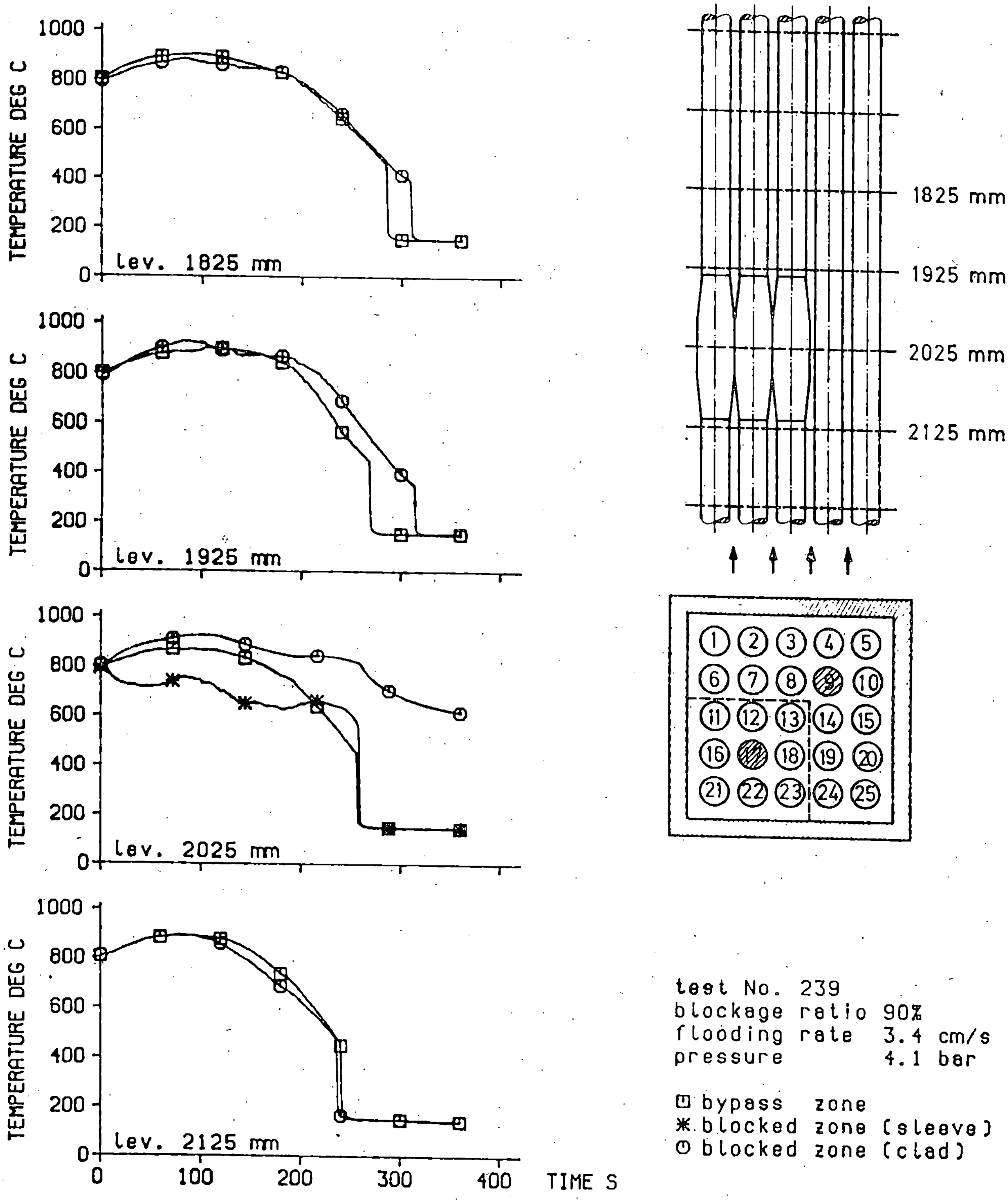
ARRAY OF THE 90% BLOCKAGE

achieved with sleeves



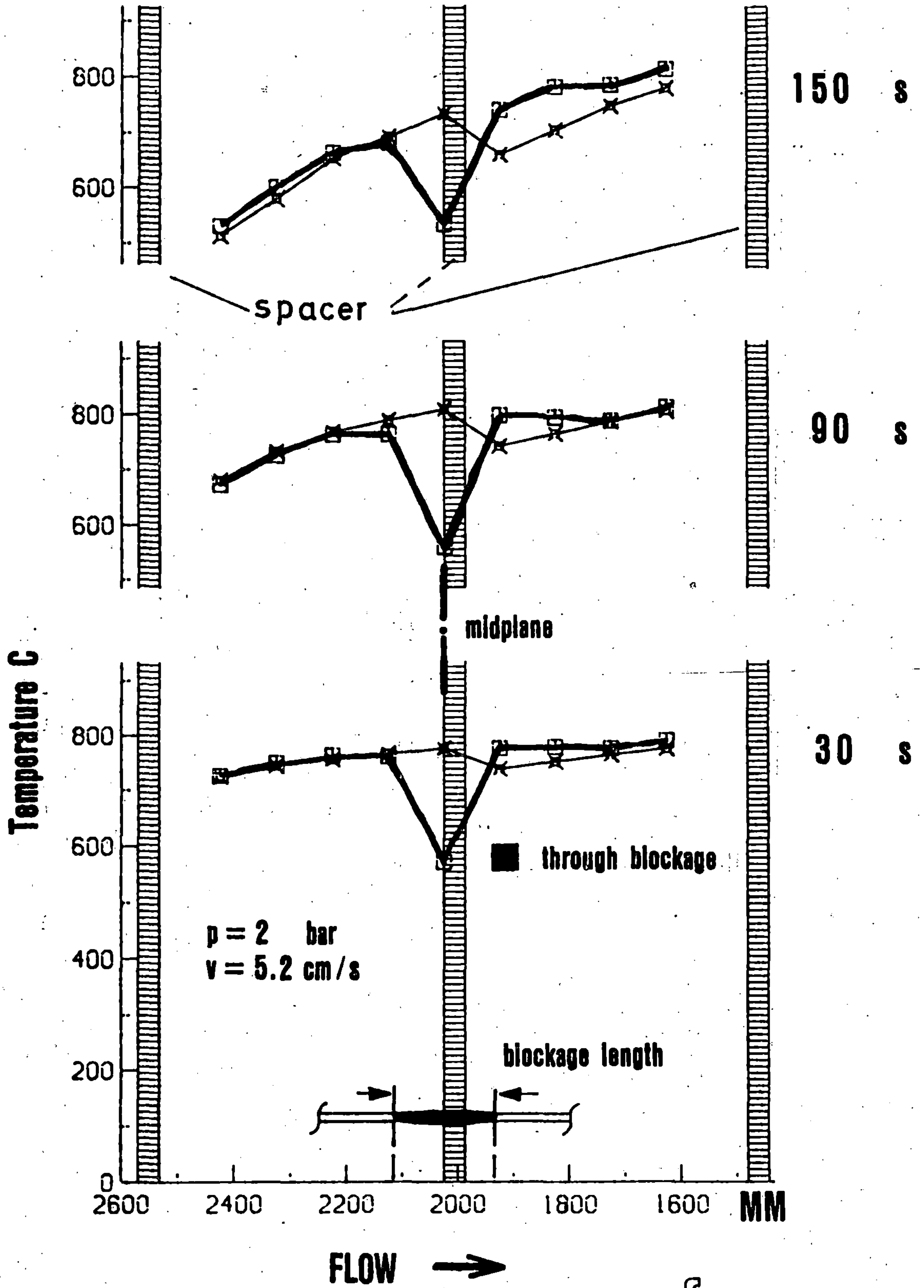
FEBA KfK IRB

CLADDING TEMPERATURES IN PARTLY BLØCKED 25 - RØD BUNDLE



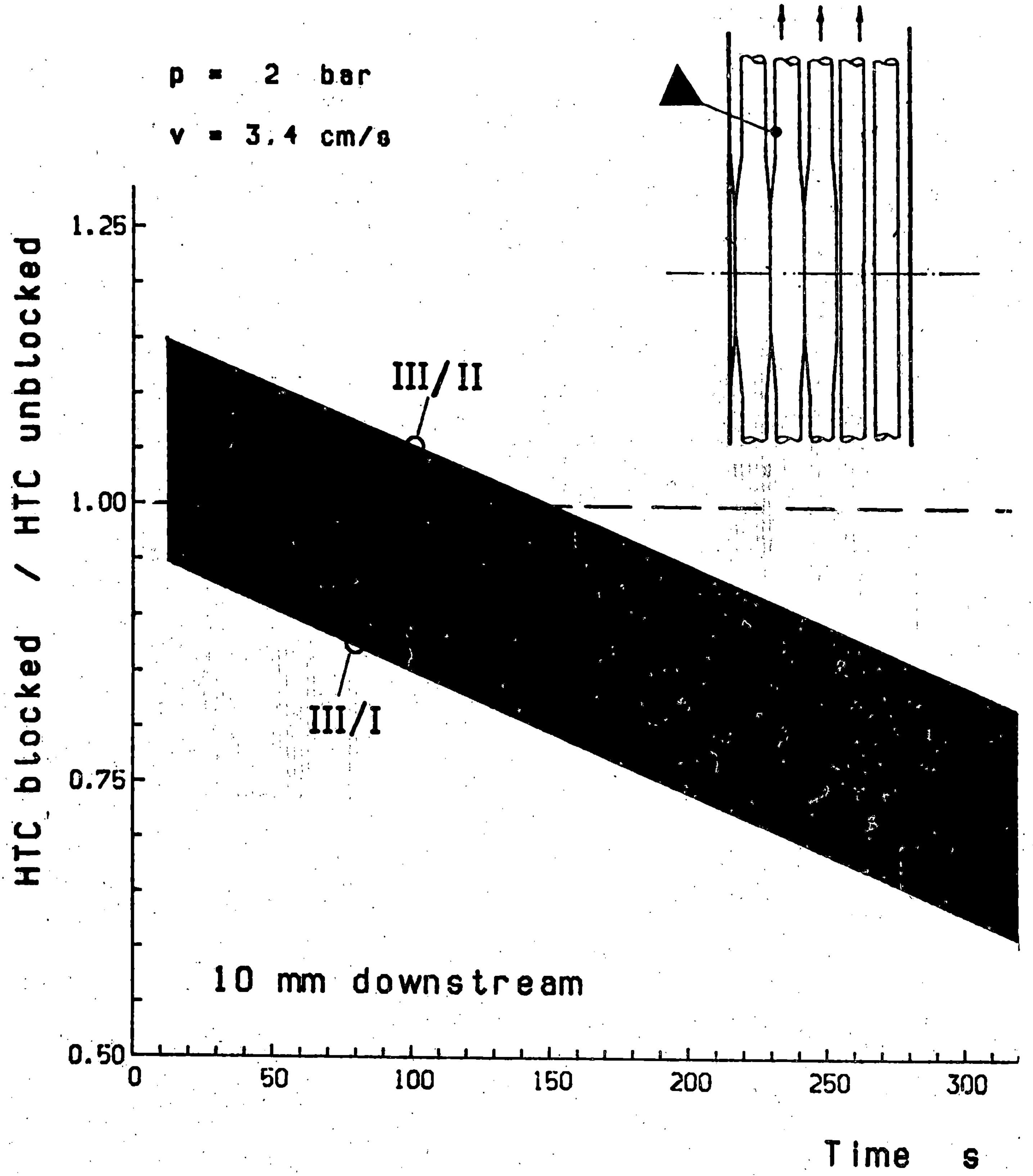
KfK IRB

5x5 RØD BUNDLE: CLADDING TEMPERATURES BYPASS ZØNE - BLØCKAGE ZØNE



COMPARISON OF AXIAL TEMP. PROFILES

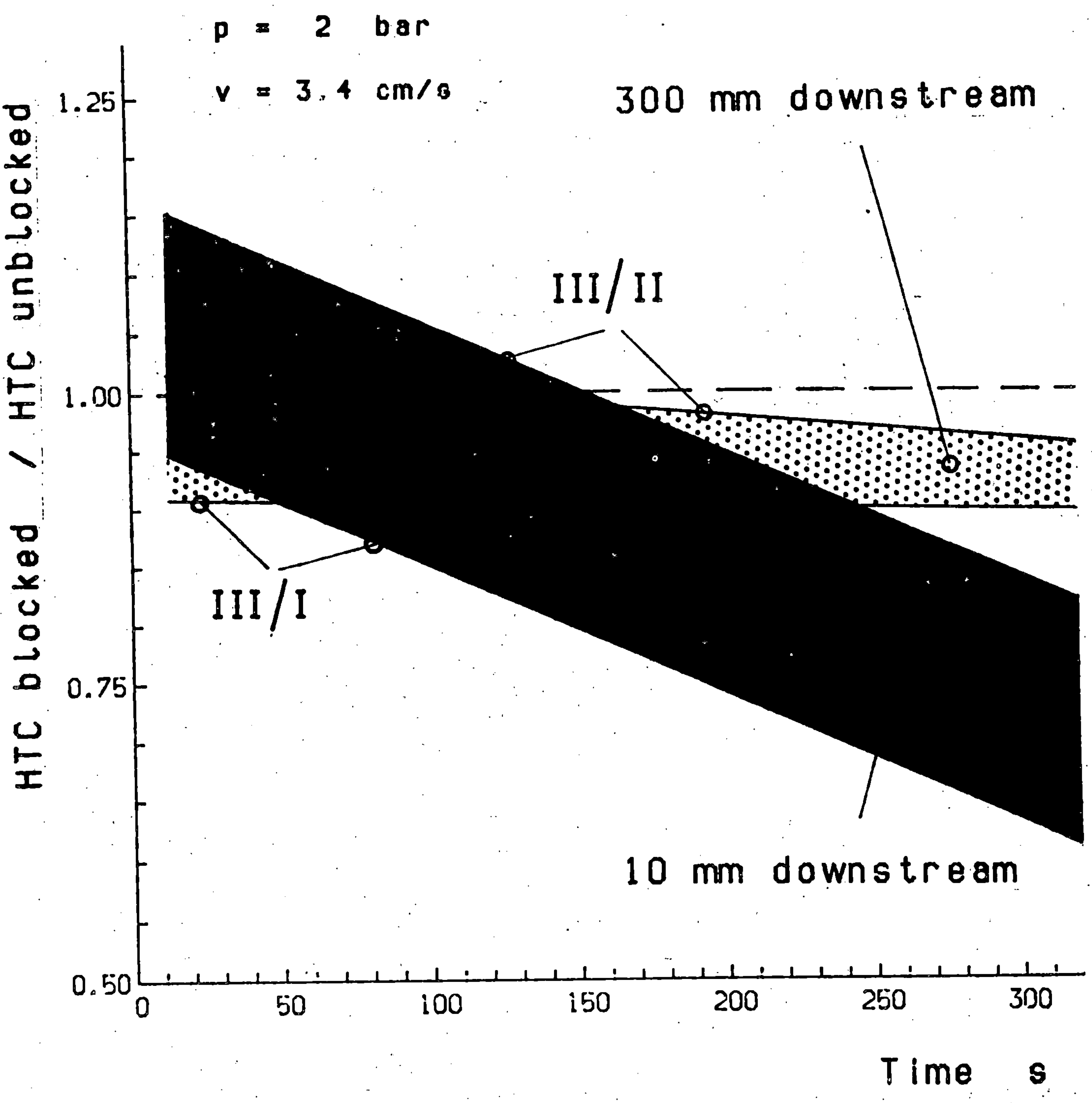
(III) through blockage - (I) through spacer



NØRMALIZED HEAT TRANSFER

10 mm downstream of blockage top end

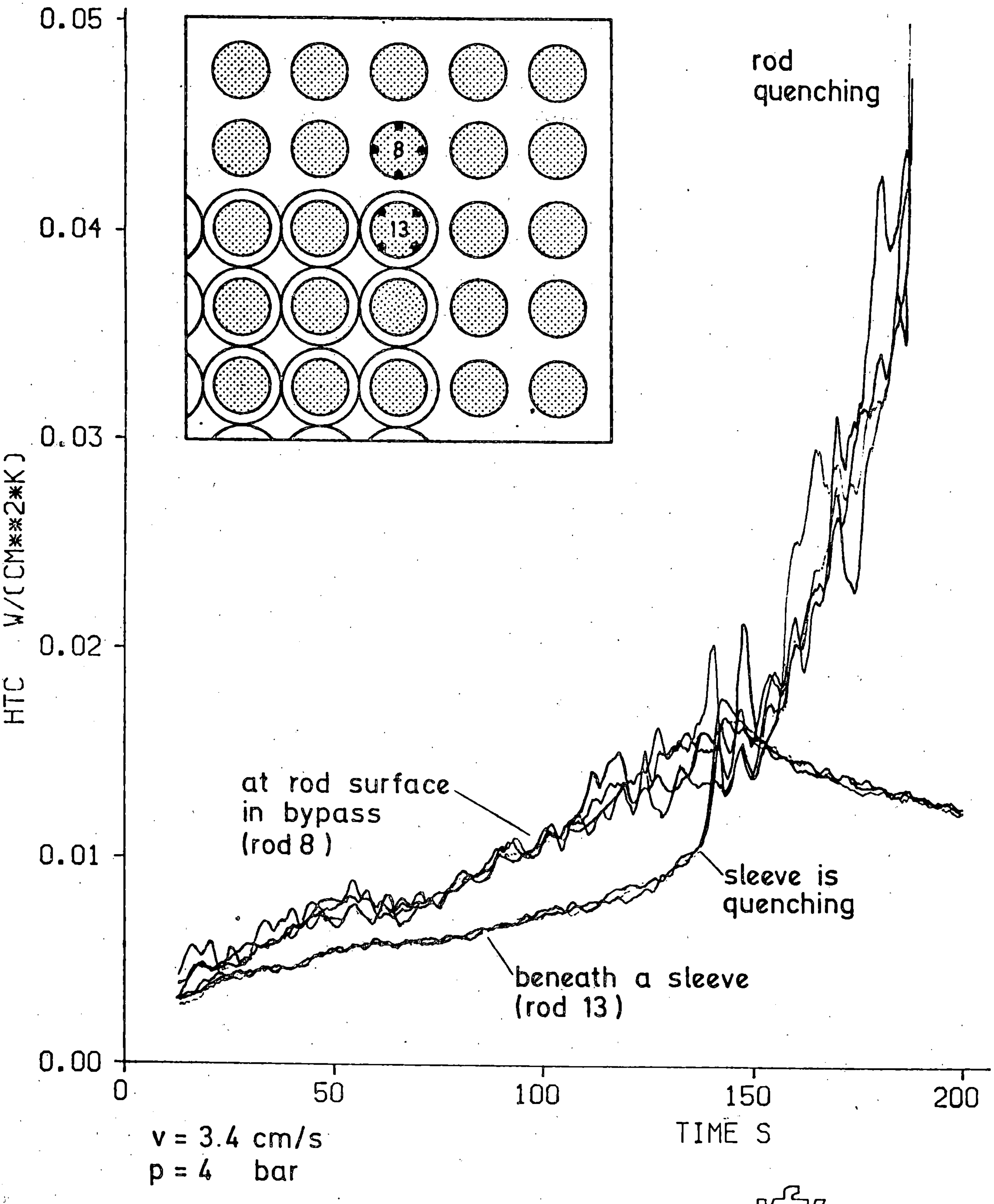
(III) blocked / (I) and (II) unblocked



NORMALIZED HEAT TRANSFER

10 and 300 mm downstream of the blockage

90 % blocked at 3x3 cluster in 5x5 bundle



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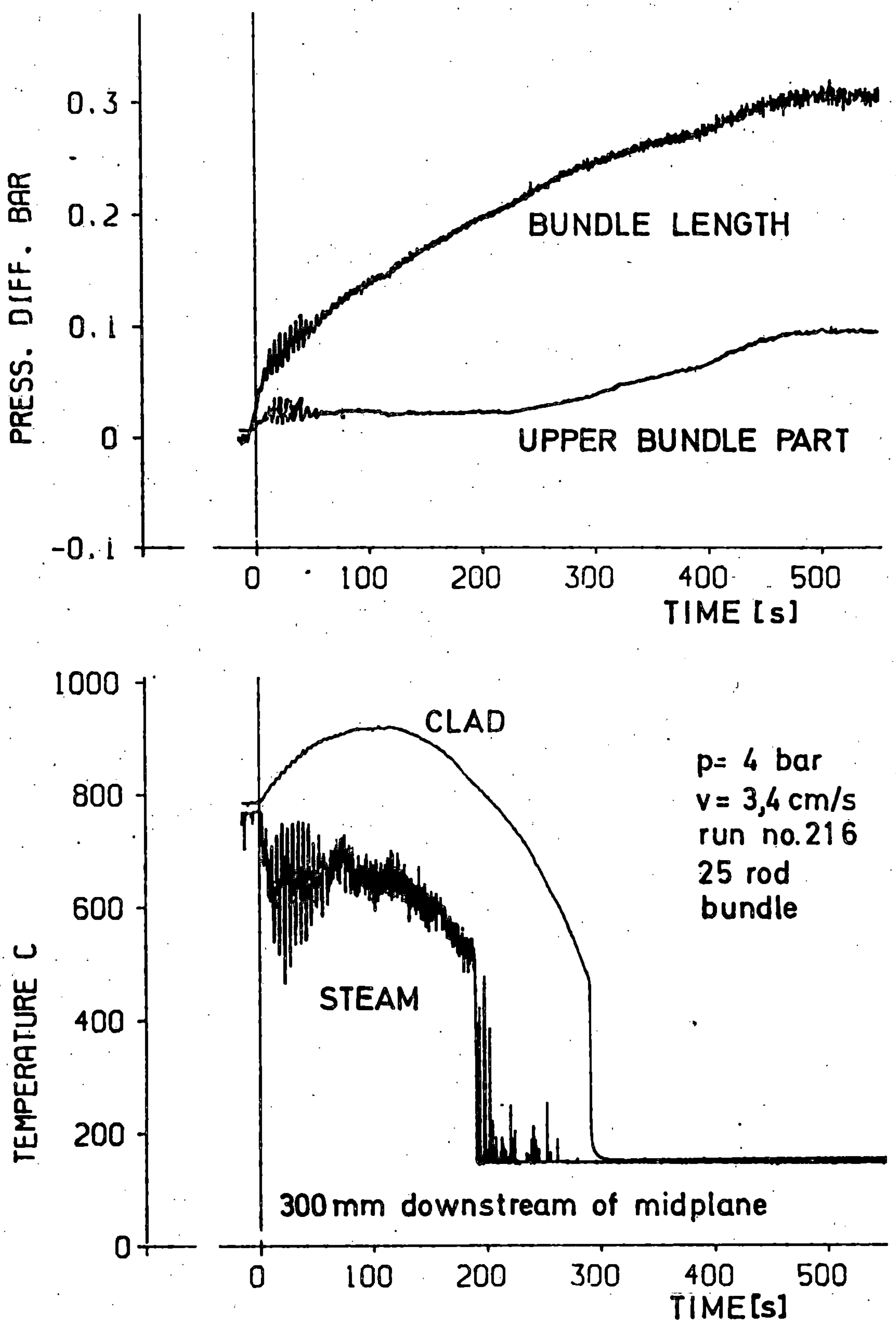
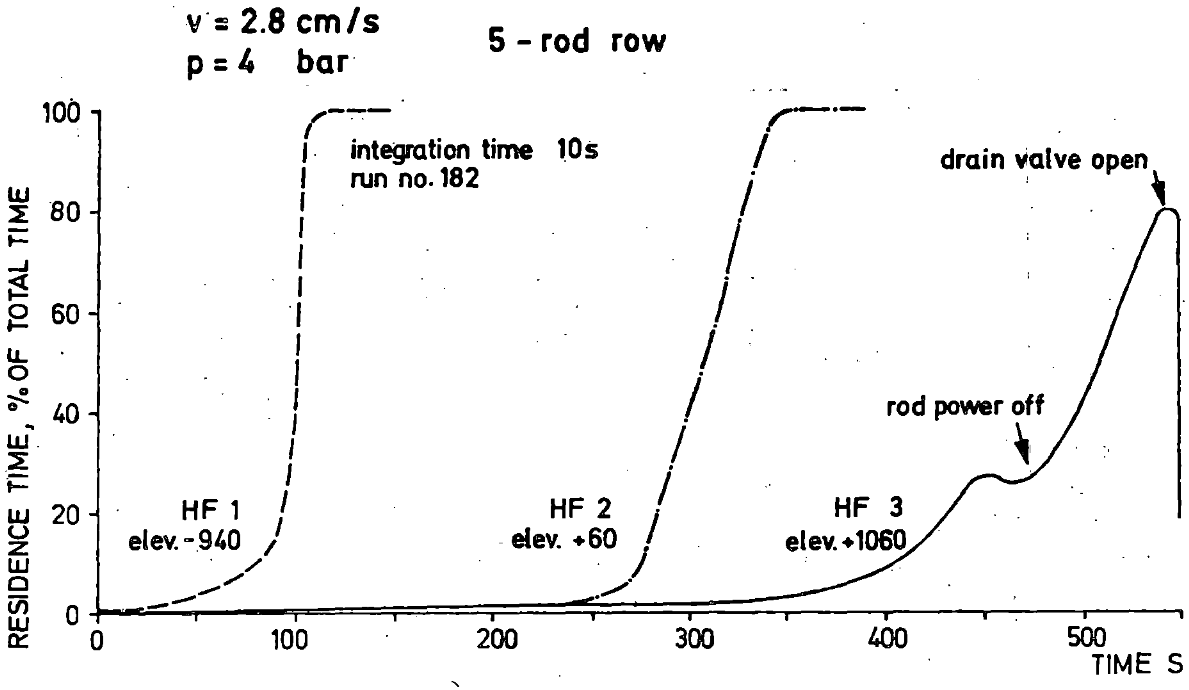
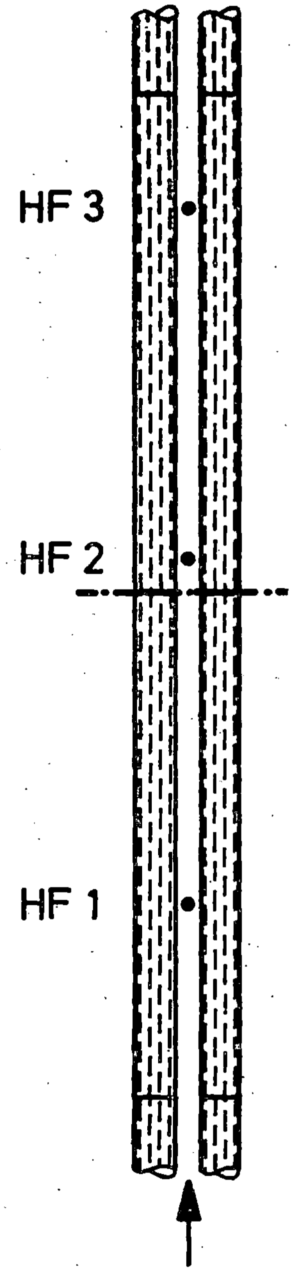
AZIMUTHAL HEAT TRANSFER COEFFICIENTS
IN THE BYPASS AND BENEATH A 62% SLEEVE



KfK

Influence of Flow Pulsations on Pressure Difference, Clad and Fluid Temperature Signals





RESIDENCE TIME OF WATER AT HIGH FREQUENCY PROBES

CONCLUSIONS from recent FEBA results

90 % blockage at 3x3 rods in 5x5 bundle influences flooding conditions locally:

* Lower temperatures at "balloons"
* Higher clad temperatures only downstream
* av. 20 K and max. 50 K were observed before turnaround point
* Blockage effect is time-dependent: HTC blocked / HTC unblocked decreases during flooding
* Steam temperatures about 200 K lower than corresponding clad temperatures

FISSION-GAS RELEASE FROM SEVERELY DAMAGED LWR FUEL

S. M. Gehl

Materials Science Division
Argonne National Laboratory

presented at

Eighth Water Reactor Safety Research Information Meeting

October 27, 1980

SAFETY IMPLICATIONS OF FISSION-GAS RELEASE

Public Safety and Health:

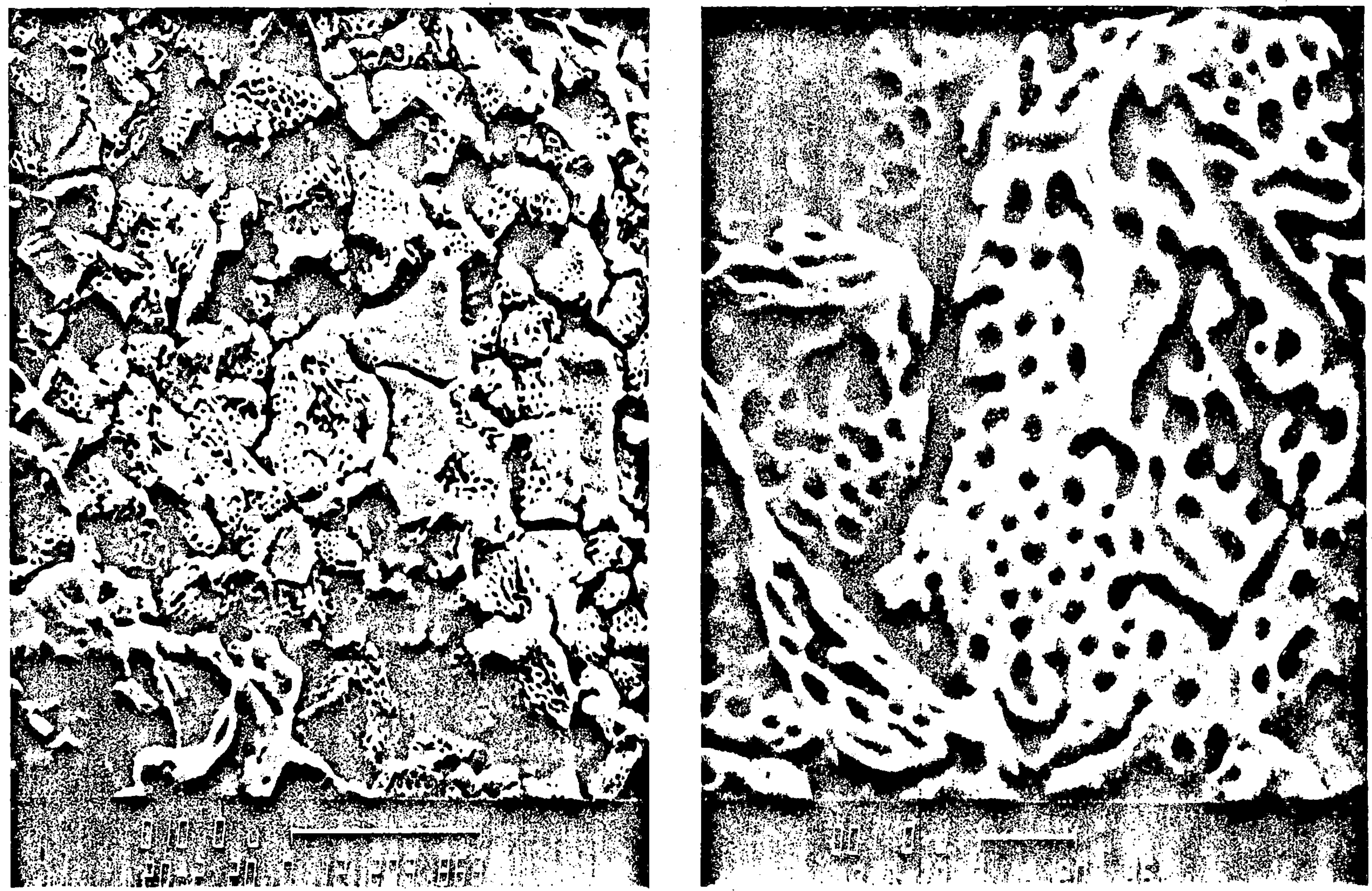
- Radiological Source Term

Accident Event Sequence:

- Decrease in Fuel Thermal Conductivity
- Decrease in Fuel Mechanical Strength
- Swelling

MECHANISMS OF FISSION-GAS RELEASE

- Grain-edge Tunnel Interlinkage
- Intergranular Microcracking
- Fuel Melting
- UO_2-Zircaloy Reaction



INTERLINKAGE

Grain Surface Bubbles and Channels Produced by DEH

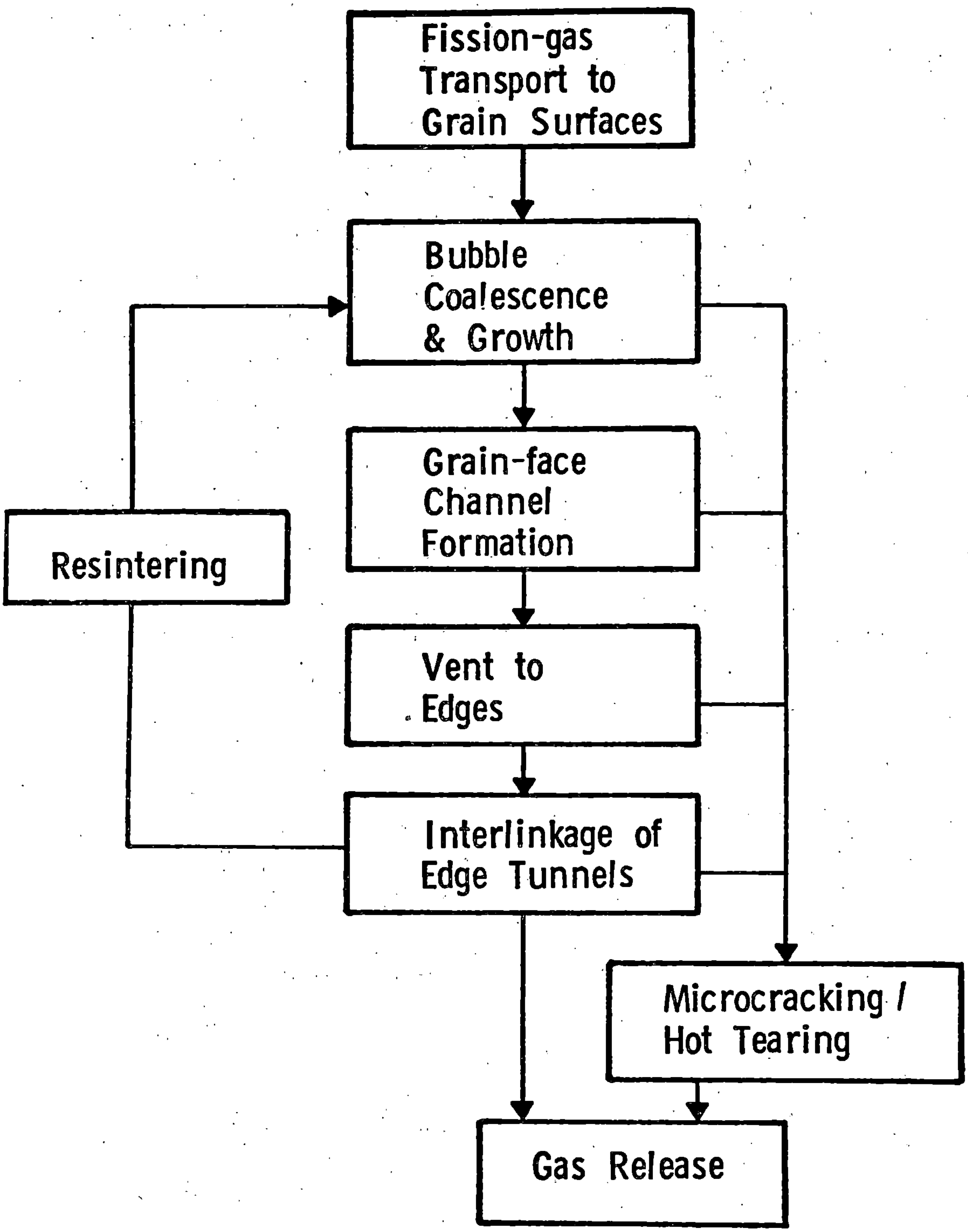
MICROCRACKING PRODUCED BY DEH AND IN-REACTOR TRANSIENTS

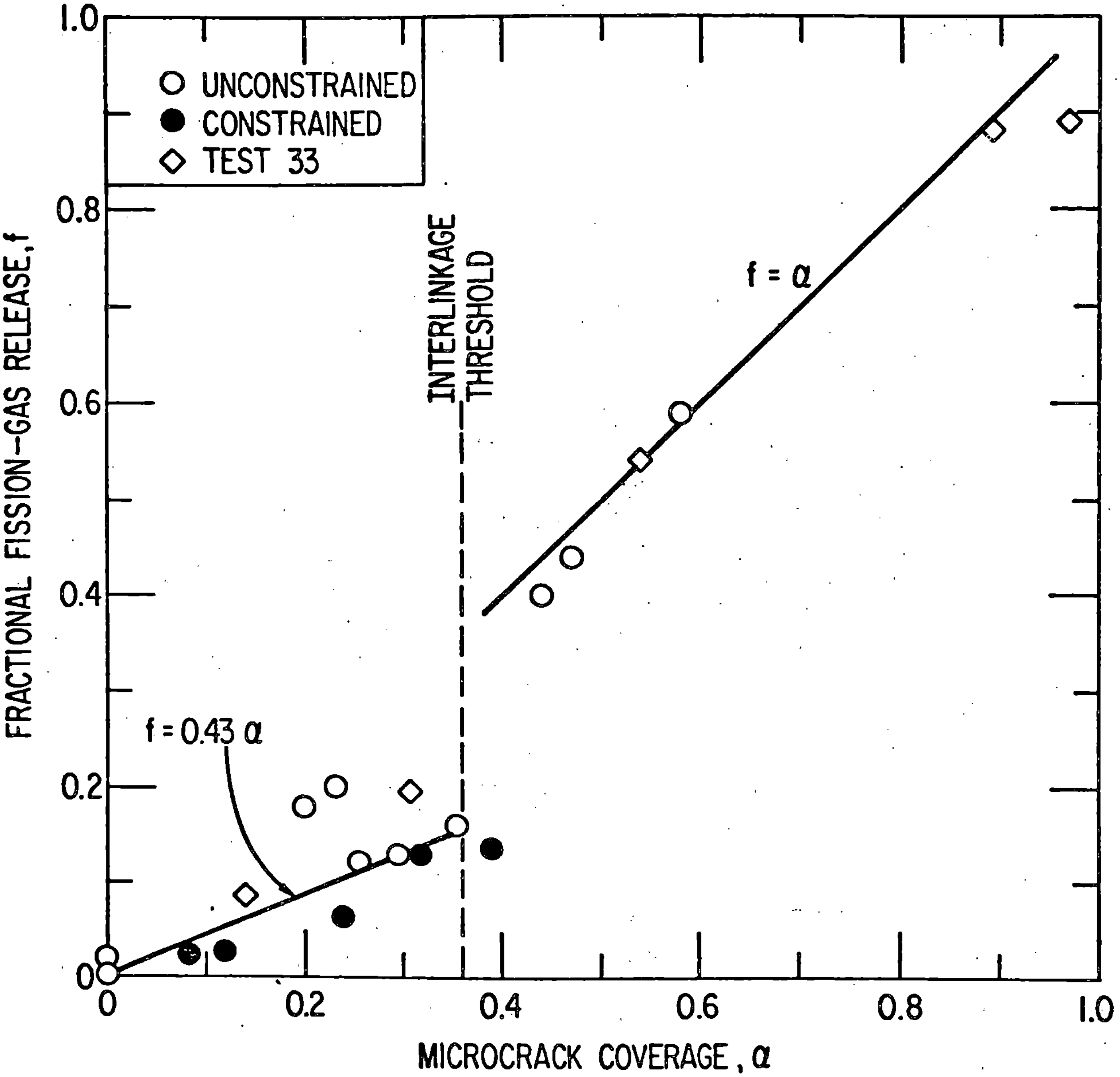
DEH-TESTED ROBINSON

DRESDEN-3

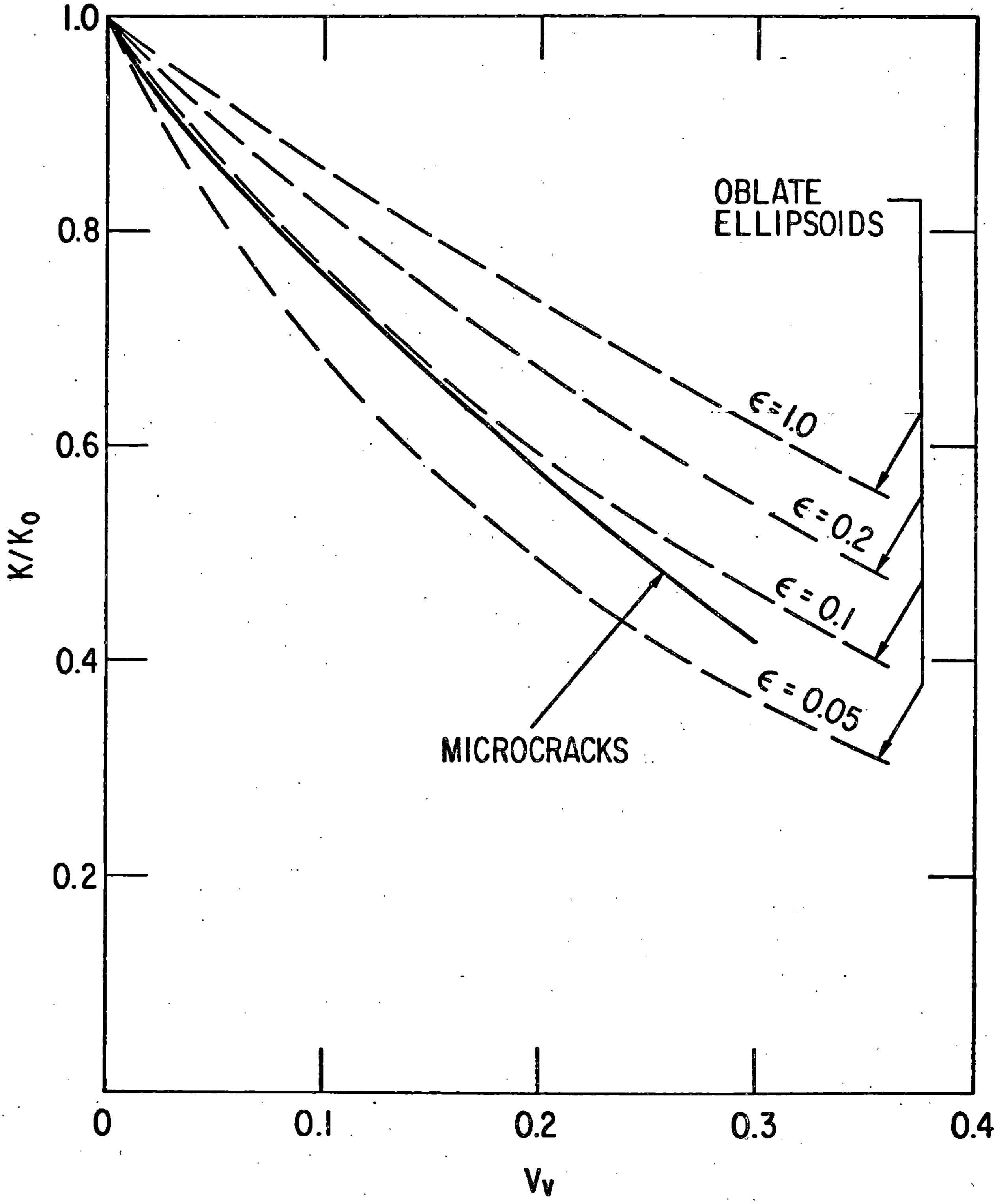
100 μm

GAS RELEASE PROCESSES ON GRAIN SURFACES

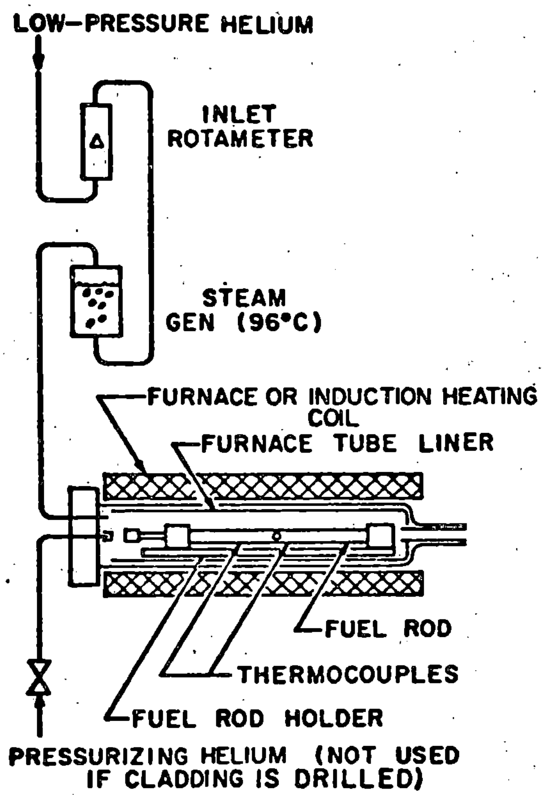




1.0
0.8
0.6
0.4
0.2
0
UNCONSTRAINED
CONSTRAINED
TEST 33
FRACTIONAL FISSION-GAS RELEASE, f
INTERLINKAGE THRESHOLD
f = α
f = 0.43 α
0
0.2
0.4
0.6
0.8
1.0
MICROCRACK COVERAGE, α



1.0
0.8
0.6
0.4
0.2
K/Ko
0
0.1
0.2
0.3
0.4
Vv
OBLATE
ELLIPSOIDS
ϵ=1.0
ϵ=0.2
ϵ=0.1
ϵ=0.05
MICROCRACKS



IRRADIATED FUEL FROM H. B. ROBINSON NO. 2

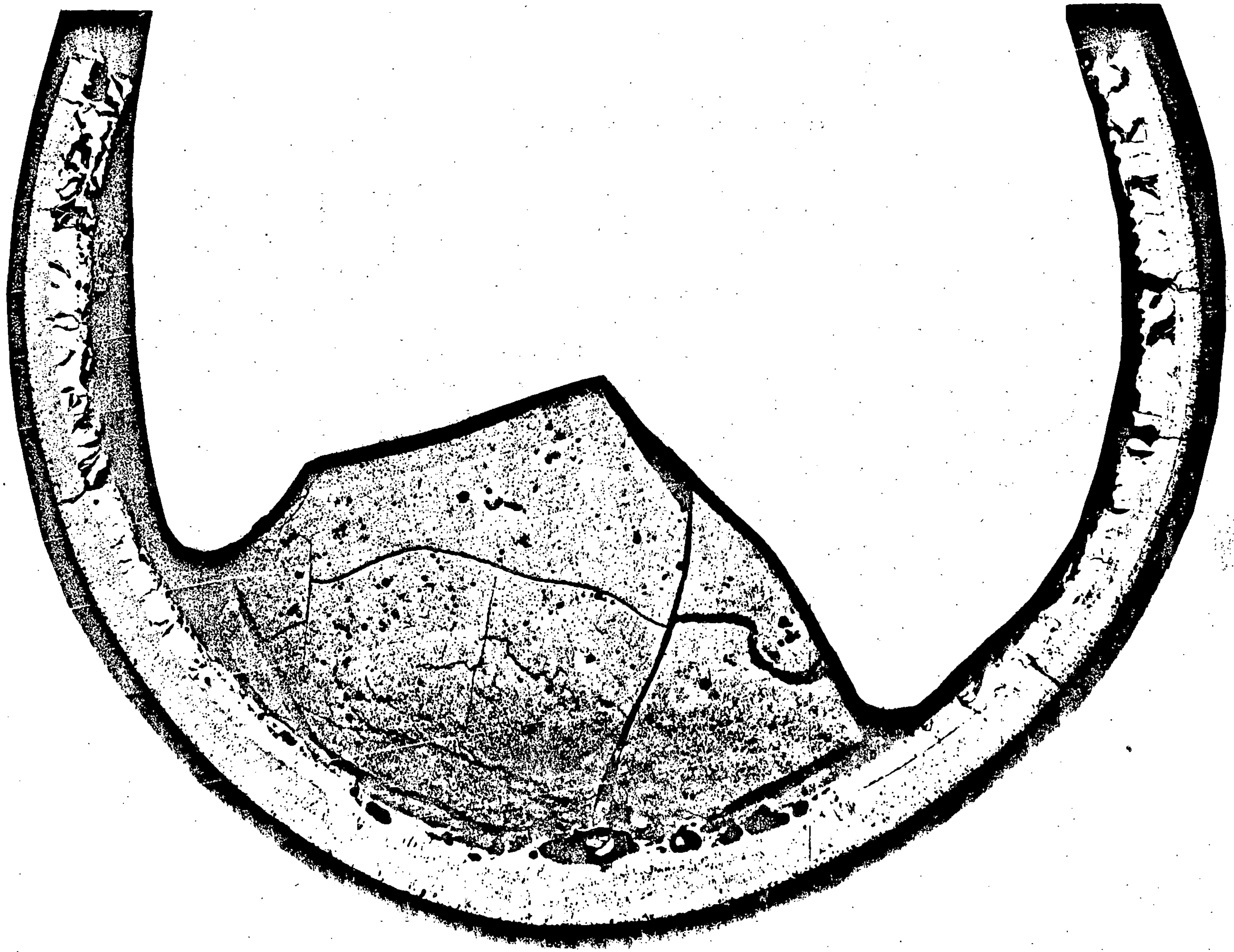
28000 MWd/t Burnup

22.4 and 17.7 kW/m LHGR

0.2% Fission Gas Released from Fuel

Fission Gas Retained in Fuel
- 0.68 cm^3 Xe/g Fuel
- 0.05 cm^3 Kr/g Fuel

Appearance of Test HT-2 Specimen As-received at ANL



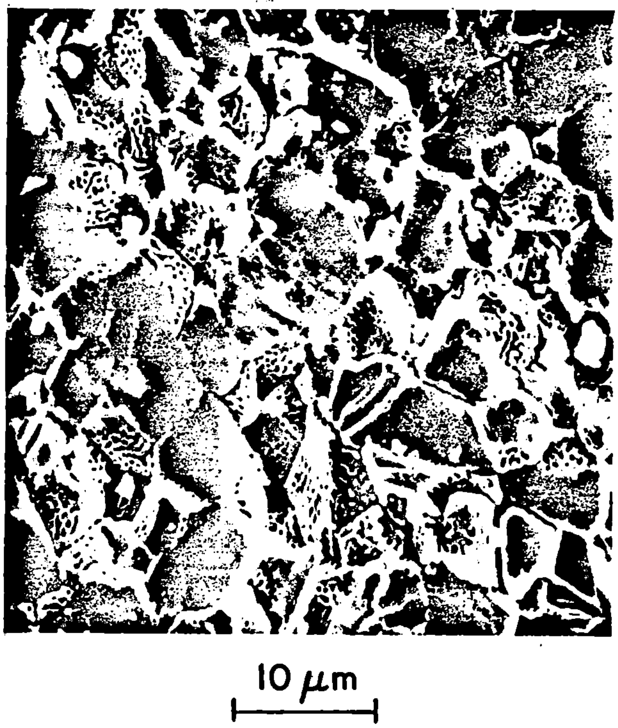
Transverse Section Through Cladding and Adherent Fuel (Test HT-2)

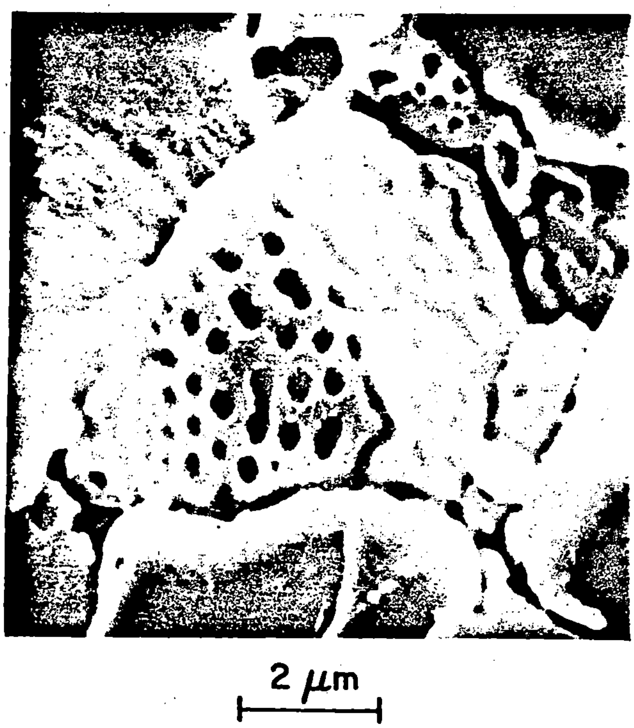
Reaction Zone at Fuel-Cladding Interface

Microcracking Pattern in Fuel Chunk from ORNL Test HT-2

Robinson Fuel ORNL HT 1

Robinson Fuel ORNL HT 2





Grain Surface Bubbles and Channels Formed in Tests HT-1 and HT-2

SUMMARY--GAS RELEASE DURING SEVERE ACCIDENTS

LOW HEATING RATES

Process	Fission-gas Release	Additional Effects
Clad Ballooning	--	Loss of Fuel Constraint
Clad Rupture	Gap Inventory	--
Diffusion to Boundaries	Low	Strength Reduction
Interlinkage/ Microcracking	up to 90%	Strength Reduction Thermal Cond. Decrease
UO_2-Zircaloy Liquation	Remainder	
UO_2 Melting	Remainder	

MEASURED RELEASE OF RADIOACTIVE XENON, KRYPTON, AND IODINE FROM UO_2 DURING NUCLEAR OPERATION AND A COMPARISON WITH RELEASE MODELS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

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a. On assignment at the Halden Reactor Project from the CEGB Berkeley Nuclear Laboratories, U.K.

MEASURED RELEASE OF RADIOACTIVE XENON, KRYPTON AND IODINE FROM UO_2 DURING NUCLEAR OPERATION AND A COMPARISON WITH RELEASE MODELS

A. D. Appelhans
J. A. Turnbull[a]
EG&G Idaho, Inc.

The amount of radioactive fission products in the fuel-cladding gap of light water reactor (LWR) fuel rods, the gap inventory, which is available for release should the cladding fail, is one of the Nuclear Regulatory Commission's (NRC's) current licensing issues. The current Regulatory Guides 1.25, 1.3, and 1.4 assumptions of 100% release of noble gases and 25% release of iodine for loss-of-coolant type accidents and 10% release of noble gases and iodine during fuel handling accidents, have been estimated by the NRC [1,2] to be over-conservative by factors of 100 and 10, respectively.

Direct measurement of the gap inventory during irradiation provides the data necessary to assess the conservatism of the Regulatory Guides and check the predictions of models being developed to calculate the gap inventory. As part of the NRC sponsored Safety Researach Program, EG&G Idaho, Inc., is conducting fuel behavior experiments in the Halden Reactor in Norway. The Instrumented Fuel Assembly-430 (IFA-430) operated in that facility is designed to provide data on the release of xenon (Xe), krypton (Kr), and iodine (I) fission products to the fuel-cladding gap during irradiation.

This summary report presents the preliminary results of the initial measurements of short-lived Xe, Kr, and I release fractions, compares the measured data with release fractions predicted by the proposed American

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a. On assignment at the Halden Reactor Project from the CEGB Berkeley Nuclear Laboratories, U.K.

Nuclear Society Standard ANS 5.4 model for fission gas release and by the diffusion model of Turnbull and Friskney, and compares the measured and calculated release fractions with the NRC Regulatory Guide assumptions.

The IFA-430 test assembly contains four 1.28-m-long fuel rods loaded with 10% enriched UO_2 pellet fuel. Two of the rods are used in fission gas release experiments; each is instrumented with a centerline thermocouple and three axially spaced pressure sensors. These two rods are of typical LWR design with diametral gap sizes of 0.23 and 0.10 mm representing beginning-of-life and end-of-life conditions, respectively. The rods are connected to a gas flow system which permits the fission gases released to the gap to be swept out of the fuel rods to a gamma spectrometer where the isotopic content is quantitatively measured.

The steady state equilibrium release rate of Xe and Kr isotopes and ^{135}I were measured at an average fuel rod heating rate of $\sim$23 kW/m, with fuel centerline temperatures of $\sim$1350 K, at a burnup of 4500 MWd/t. The release fraction of ^{131}I was measured for similar conditions at burnup of 5000 MWd/t. Table 1 shows the measured release fractions for several isotopes of Xe, Kr, and I.

TABLE 1. MEASURED RELEASE FRACTIONS

Xenon		Krypton	Iodine	
^{135}Xe	^{133}Xe	^{85m}Kr	^{135}I	^{131}I
1.2×10^{-4}	3.3×10^{-4}	8.0 x 10-5	4.2×10^{-5}	2.3×10^{-4}

The ANS 5.4 predicted release fractions for ^{135}Xe and ^{133}Xe are within the uncertainty band of the data; however, for shorter lived (15 min) isotopes the predicted release is low by a factor of about 2. For krypton the ANS 5.4 predicted release fraction is low by a factor of 2 to 4. The iodine release fraction calculated with ANS 5.4 is within a factor of 2 of the measured values for ^{135}I and ^{131}I.

The diffusion model of Turnbull and Friskney, which takes precursor effects into account, predicts noble gas release fractions which show very good agreement with the data when the fuel pellets are assumed to contain about seven radial cracks. More significantly, the model predicts the detailed relationship between the different isotopes indicating the model probably includes the correct mechanisms accounting for precursor effects and element dependent diffusion coefficients.

A comparison of the order of magnitude of the measured and calculated release fractions, generally 10^{-4} to 10^{-3}, with the NRC Regulatory Guide assumptions of 0.1 for fuel handling type accidents confirms the NRC estimate that the Regulatory Guide assumption is overly-conservative. For successfully terminated loss-of-coolant type accidents for which fuel temperatures remain below 1500 K the data and calculations also support the NRC estimate that the Regulatory Guide assumed release fraction of 1.0 is high by a factor of 100.

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CALCULATED AND MEASURED FISSION GAS RELEASE DURING IRRADIATION

by

A.D. APPELHANS

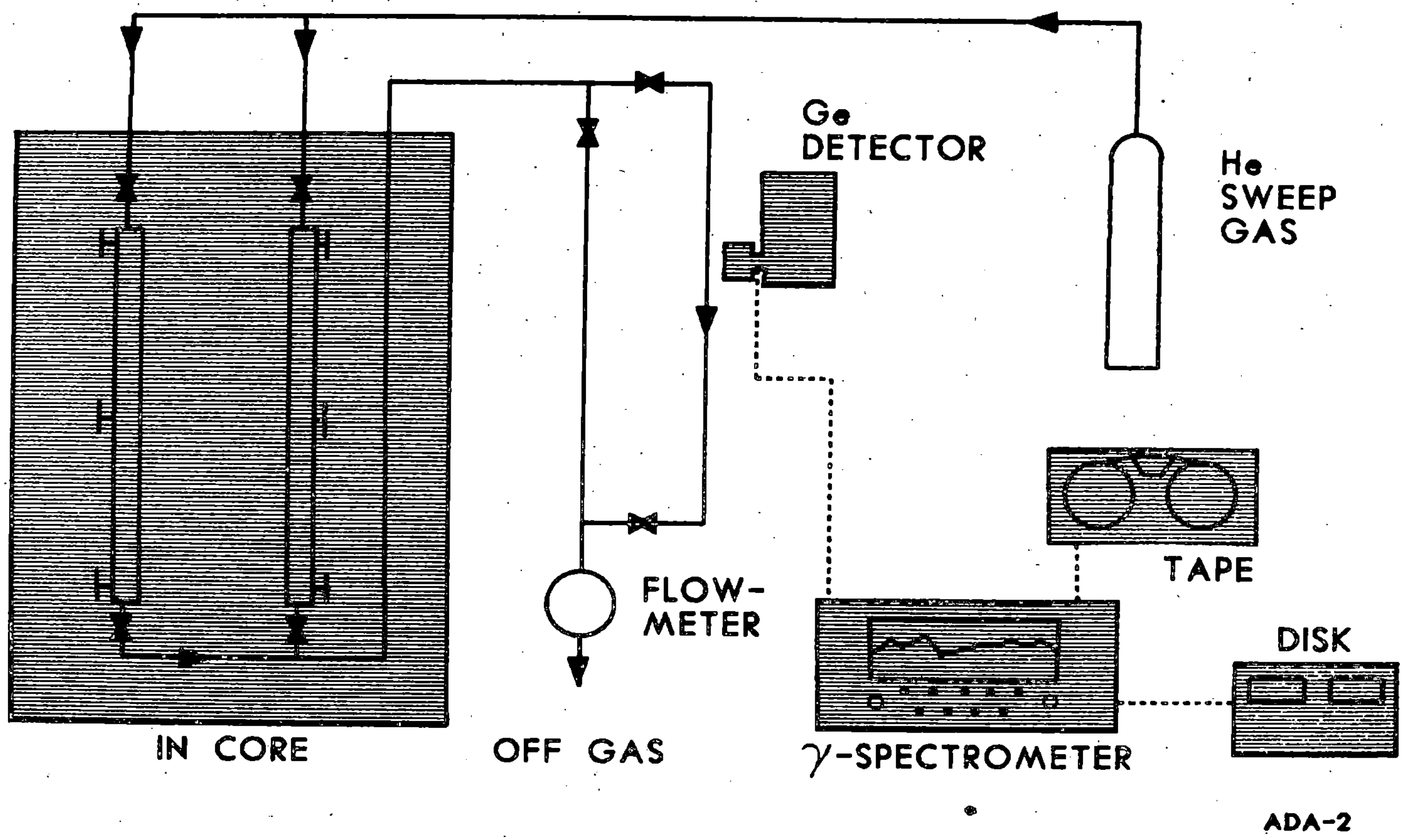
EG&G Idaho, Inc.

OUTLINE

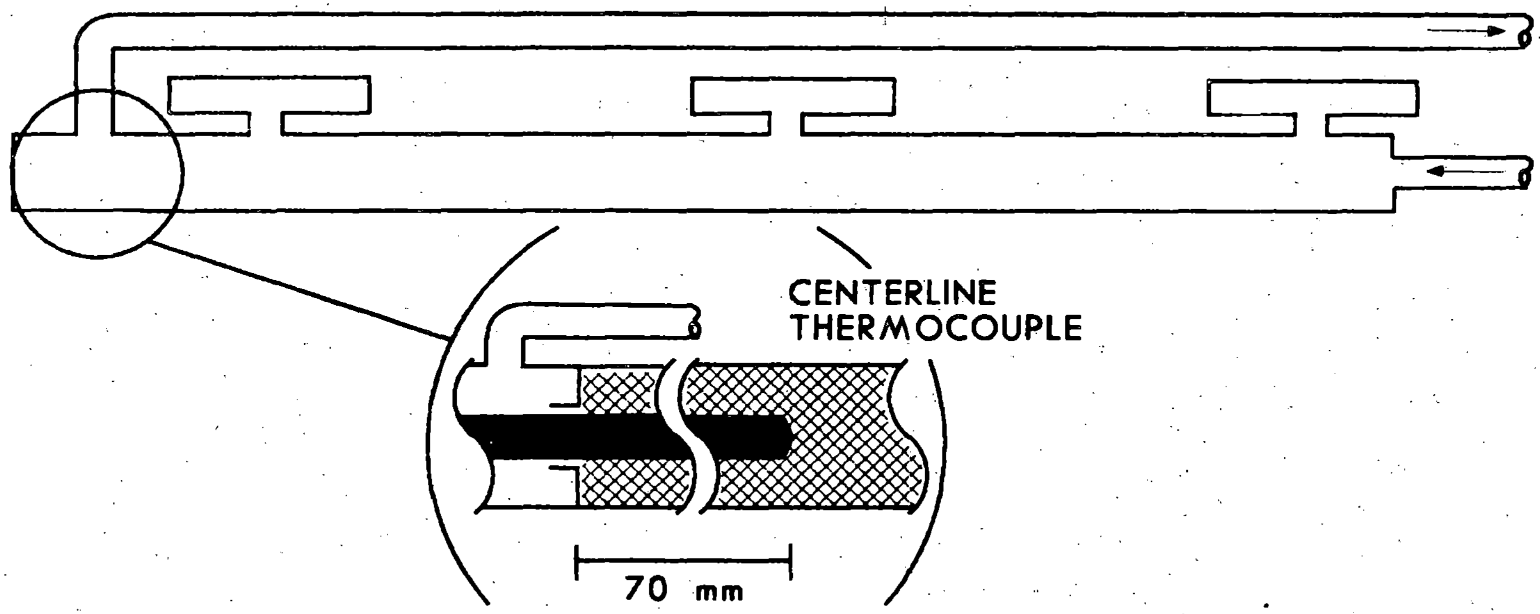
- DESCRIBE IFA-430 TESTS
- PRESENT MEASURED RELEASE FRACTIONS
- COMPARE WITH ANS 5.4 PREDICTIONS
- COMPARE WITH DIFFUSION MODEL PREDICTIONS
- COMPARE MEASURED AND CALCULATED RESULTS WITH NRC REGULATORY GUIDES

ADA-1A

FISSION PRODUCT MEASUREMENT SYSTEM



GAS FLOW ROD



CLADDING	FUEL	DIAMETRAL GAP
12.8 mm OD	10wt% U-235	0.1 mm
1.28 m LONG	95% TD	0.23 mm

ADA-3

EXPERIMENT CONDITIONS

ROD POWER: 22-24 kW/m (25 W/gm)

FUEL TEMPERATURE: 1350 K PEAK
850 K BULK AVERAGE
< 1560 K DURING FUEL LIFETIME

BURNUP: ~ 4500 MWd/t

ADA-4

RELEASE-TO-BIRTH RATIO
RELEASE FRACTIONS

$$R/B = \frac{\text{RELEASE RATE}}{\text{BIRTH RATE}}$$

$$F = \frac{\text{UNDECAYED INVENTORY IN GAP}}{\text{TOTAL UNDECAYED INVENTORY}}$$

AT EQUILIBRIUM:

$$R/B = F$$

ADA-5

MEASURED R/B NOBLE GASES

ISOTOPE	$(R/B)_m$	ISOTOPE	$(R/B)_m$
^{137}Xe	1.6×10^{-5}	^{89}Kr	1.5×10^{-5}
^{138}Xe	2.0×10^{-5}	^{87}Kr	3.8×10^{-5}
^{135m}Xe	4.7×10^{-5}	^{88}Kr	4.7×10^{-5}
^{135}Xe	1.2×10^{-4}	^{85m}Kr	8.0×10^{-5}
^{133}Xe	3.3×10^{-4}		

ADA-6

IODINE RELEASE

$R(Xe) = R_{fuel} + \lambda I_{gap}$

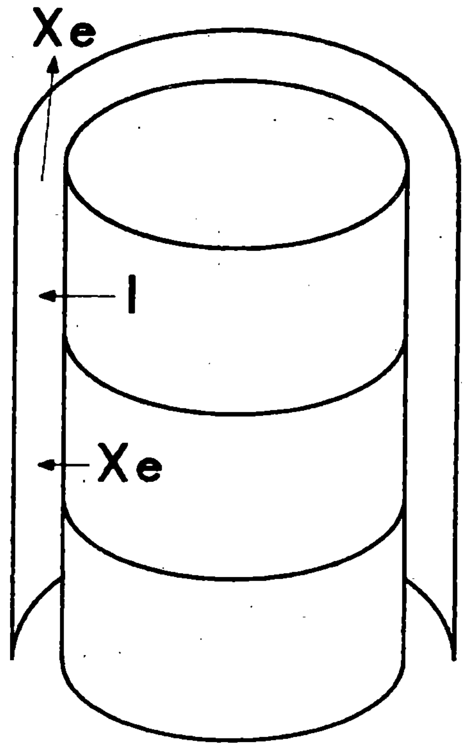
AT EQUILIBRIUM :

$\lambda I_{gap} = R(I)$

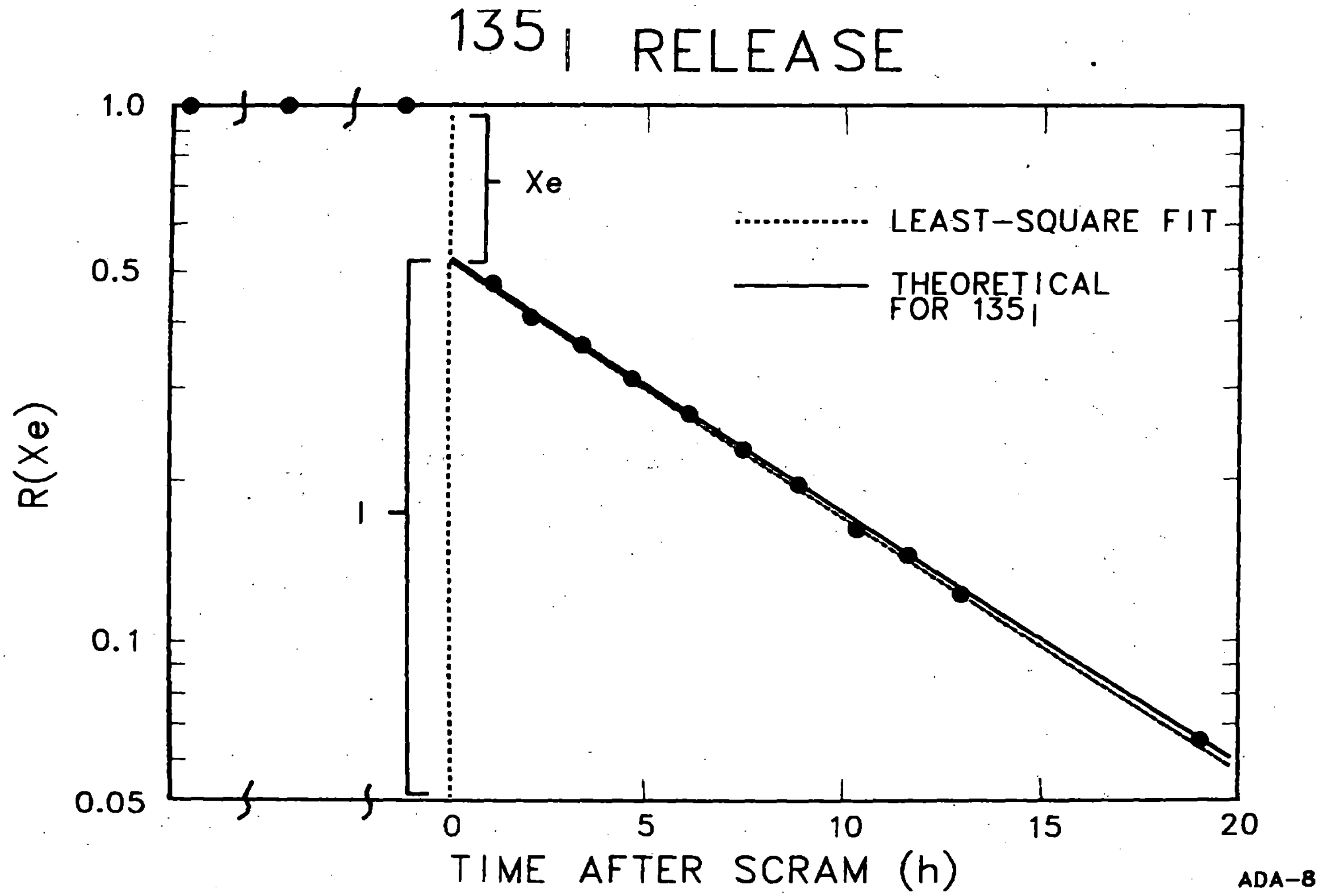
AT SCRAM :

$R_{fuel} = 0$ THUS

$R(Xe) = \lambda I_{gap}$



ADA-7

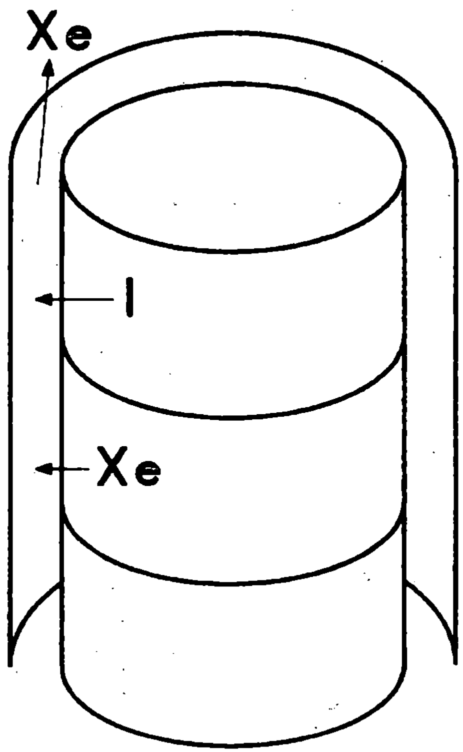


135I RELEASE
1.0
0.5
0.1
0.05
R(Xe)
Xe
I
LEAST-SQUARE FIT
THEORETICAL FOR 135I
0
5
10
15
20
TIME AFTER SCRAM (h)
ADA-8

IODINE RELEASE FRACTION

$$F = \frac{\text{GAP INVENTORY}}{\text{TOTAL INVENTORY}}$$

$$Fm(^{131}I) = 2.3 \times 10^{-4}$$



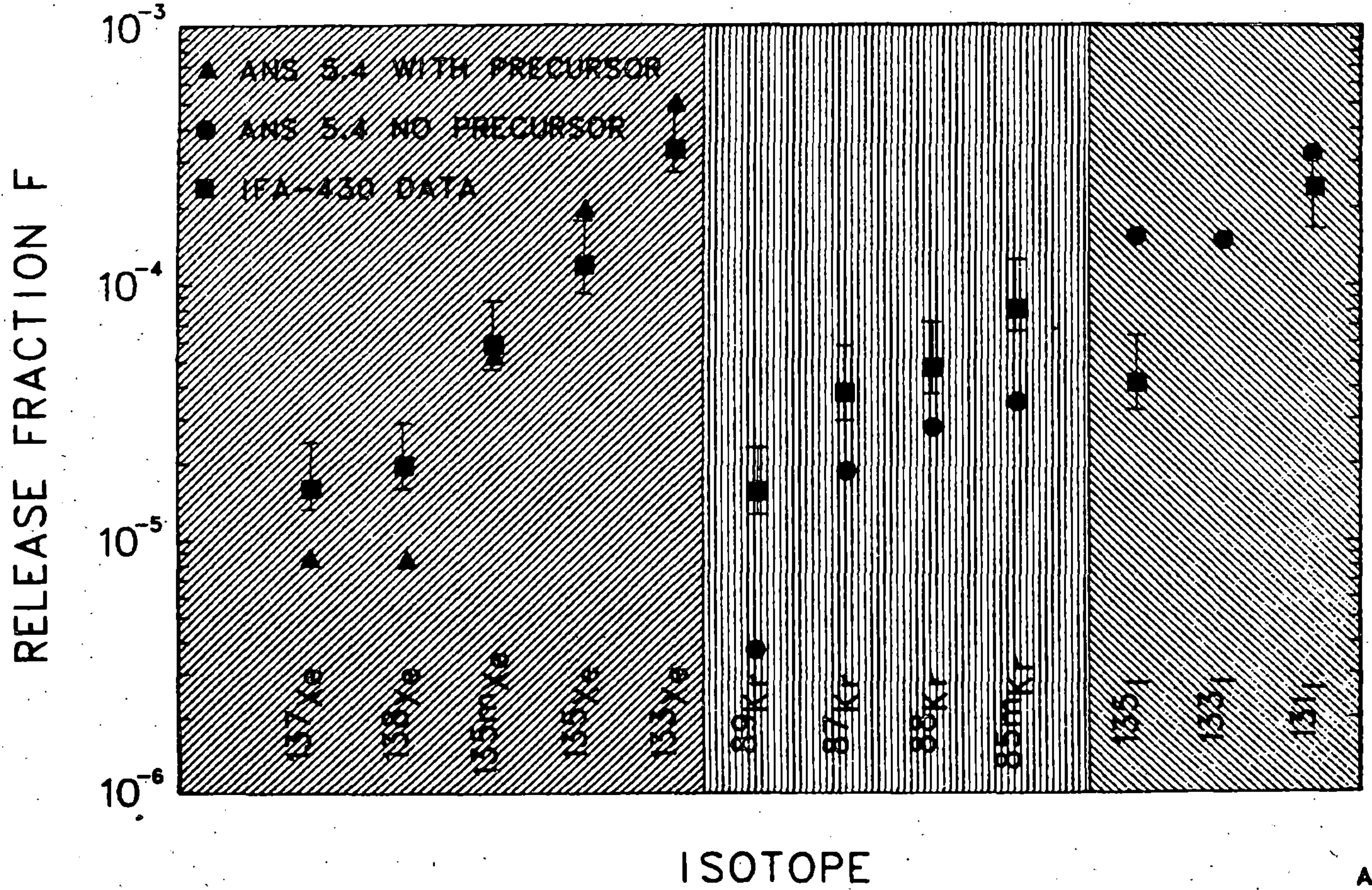
ADA-9

ANS 5.4 MODEL

- HIGH AND LOW TEMPERATURE MODELS
- STABLE AND RADIOACTIVE FISSION PRODUCTS
- IN FRAPCON-2 FUEL BEHAVIOR CODE
- MODIFIED TO INCLUDE SPECIFIC ISOTOPES OF Xe, Kr, I

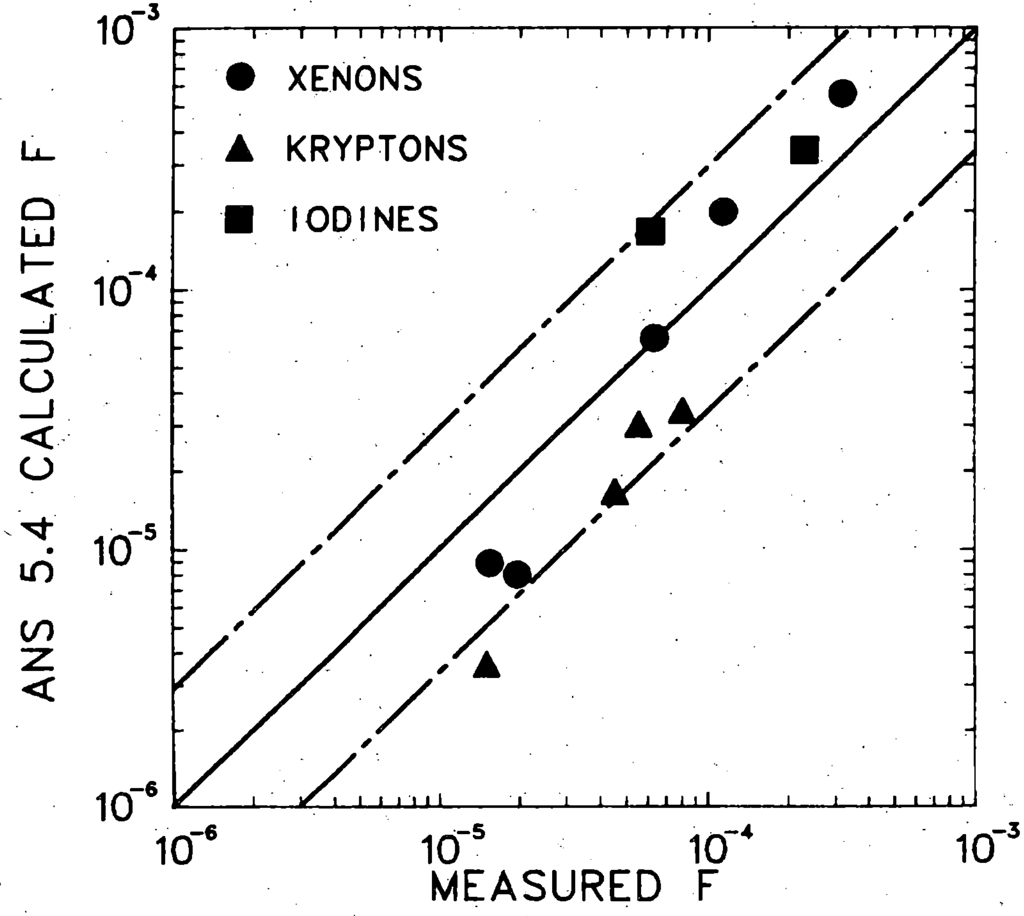
ADA-10

COMPARISON OF RELEASE FRACTIONS



ADA-11

COMPARISON OF RELEASE FRACTIONS



ADA-12

TURNBULL AND FRISKNEY
DIFFUSION WITH PRECURSORS

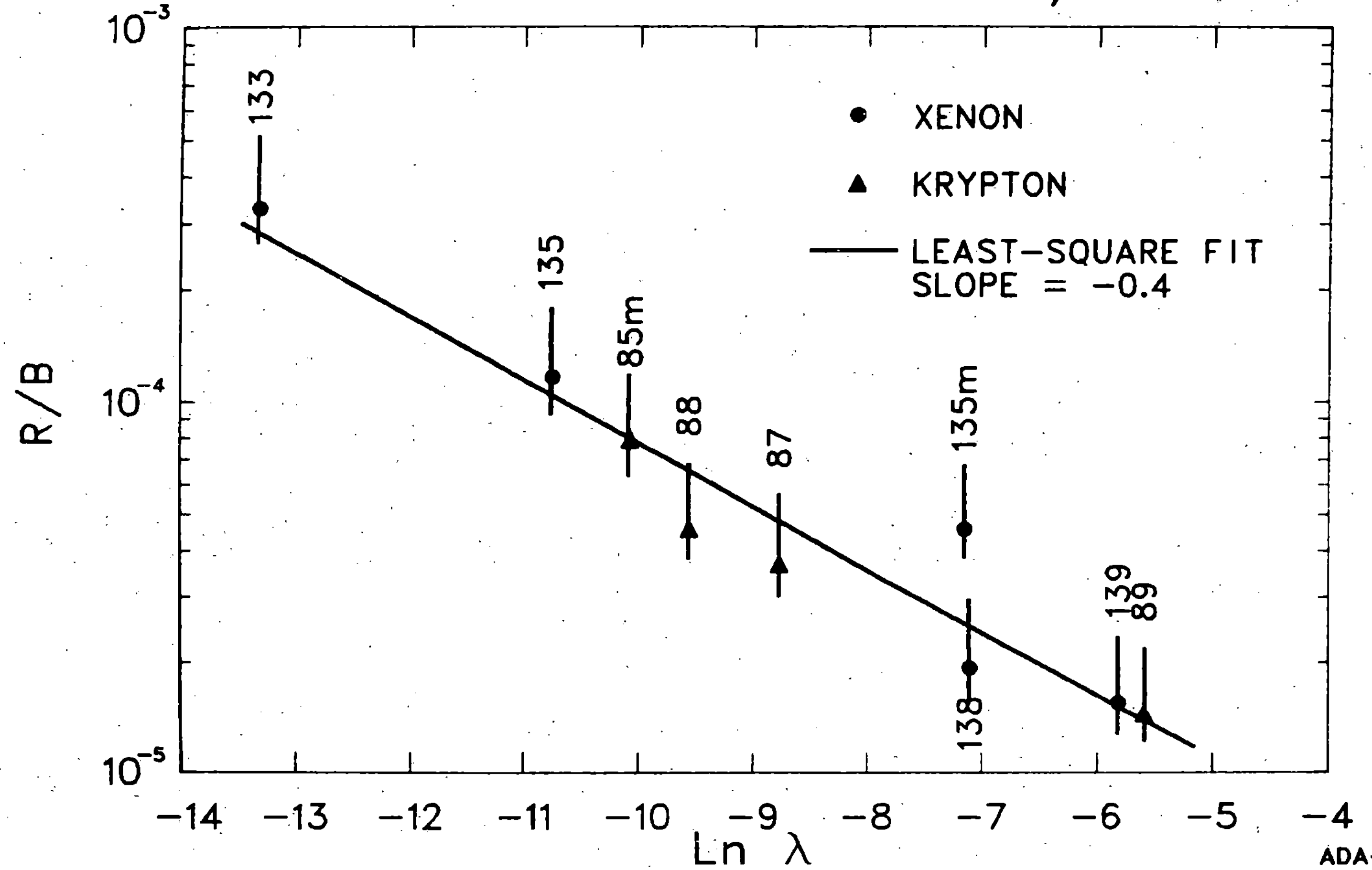
$$Ri/Bi = \frac{S}{V}\sqrt{\frac{KiDi}{\lambda i}}$$

Ki = CONSTANT RELATING PARENT-DAUGHTER

$$D(T,\phi) = A\ \exp[-B/T] + \sqrt{\phi}\ C\ \exp[-D/T] + E\phi$$

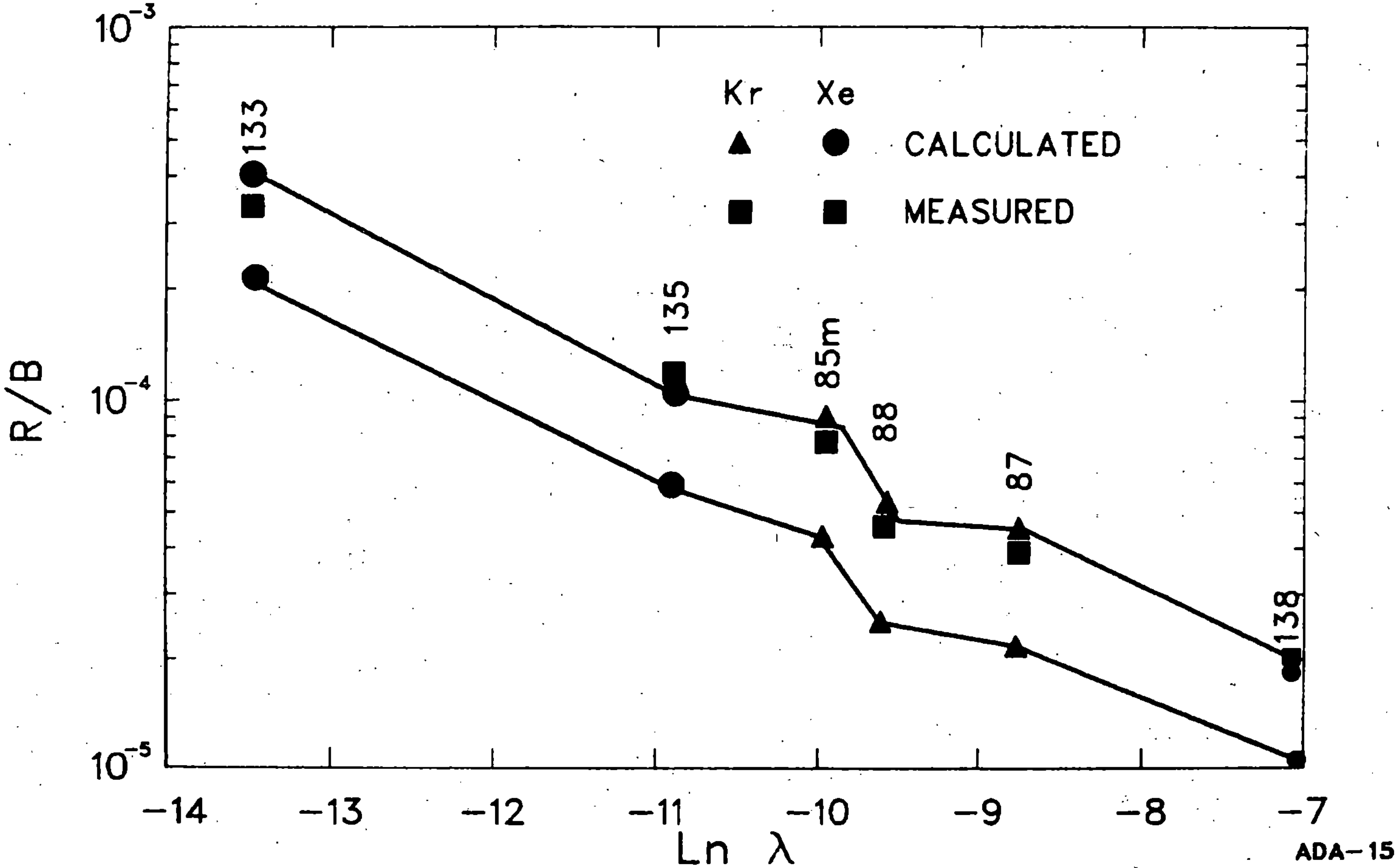
ADA-13

λ DEPENDENCE OF R/B



ADA-14

COMPARISON WITH DIFFUSION MODEL



COMPARISON WITH REGULATORY GUIDES

REGULATORY GUIDE ASSUMPTION	MEASURED AND CALCULATED
LOCA:	**LOCA:**
$F(Xe) = 1.0$	$F(Xe) \leq 0.5\times10^{-3}$
$F(Kr) = 1.0$	$F(Kr) \leq 0.8\times10^{-4}$
$F(I) = 0.25$	$F(I) \leq 0.4\times10^{-3}$
FUEL HANDLING:	**FUEL HANDLING:**
$F(Xe) = 0.10$	$F(Xe) \leq 0.5\times10^{-3}$
$F(Kr) = 0.10$	$F(Kr) \leq 0.8\times10^{-4}$
$F(^{85}Kr) = 0.30$	$F(I) \leq 0.4\times10^{-3}$
$F(I) = 0.10$	

ADA-16

CONCLUSIONS

- RELEASE FRACTIONS MEASURED DURING IRRADIATION
- ANS 5.4 MODEL GENERALLY WITHIN A FACTOR OF 2 TO 4
- DIFFUSION MODEL PREDICTS QUANTITATIVE AND RELATIVE BEHAVIOR
- DATA SUPPORTS NRC ESTIMATE OF OVERCONSERVATISM IN NRC REG. GUIDES

ADA-17

LA-UR-80-2890

TITLE: LASL TRAC PROGRAM OVERVIEW AND TRAC-PD2 DESCRIPTION

AUTHOR(S): John C. Vigil, Q-DO/RS

SUBMITTED TO: Eighth Water Reactor Safety Research Information Meeting
October 28, 1980
Gaithersburg, Maryland

University of California

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DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

LASL TRAC PROGRAM OVERVIEW

AND TRAC-PD2 DESCRIPTION

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Presented at the
EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING
October 28, 1980
Gaithersburg, Maryland

LASL TRAC PROGRAM OVERVIEW

The Los Alamos Scientific Laboratory (LASL) is developing, testing and applying methods for the analysis of loss-of-coolant accidents (LOCAs) and other transients in light-water reactors (LWRs). This effort is focused on the development, assessment, and application of the Transient Reactor Analysis Code (TRAC). TRAC contributes to LWR accident risk and overall safety evaluation through its ability to provide detailed analyses of system transients (Fig. 1). The scope (Fig. 2) of the TRAC project at LASL includes model and code development, developmental and independent assessment, and application activities. This paper presents a brief overview of the TRAC program at LASL with emphasis on the most recently completed TRAC version (PD2) and its developmental assessment. Other papers [1-4] at this meeting focus on the remaining aspects of the TRAC project. In particular, progress on a fast-running small-break version of TRAC is described by Liles and Mahaffy,[1] the status of TRAC-PD2 independent assessment is summarized by Knight,[2] applications to the 2D/3D program are described by Williams,[3] and applications to loss-of-feedwater transients are discussed by Burns.[4]

Some of the distinguishing TRAC features are listed in Fig. 3. The underlying philosophy in TRAC development has been the use of best-estimate mechanistic models (to the extent possible within computer limitations) designed to yield realistic solutions over a broad range of applications. This approach requires a basic treatment of two-phase thermal-hydraulics and detailed geometric models of system components. User-selected options are minimal and hence the code must determine local flow topology and supply appropriate constitutive relations. Development of flow-regime-dependent constitutive relations is a continuing process; improvements are based on further code testing and new experimental data. The modular feature has made it possible to use the LASL PWR version in the development of a BWR version (TRAC-BD1) at INEL and the COBRA/TRAC program at PNL. These codes are described in the next two papers of this session by Thurgood[5] (COBRA/TRAC) and Aguilar[6] (TRAC-BD1).

Work on the TRAC project began at LASL in April 1975 (Fig. 4). The first documented[7] version, TRAC-P1, was completed in December 1977 and was made available to various organizations following a TRAC Workshop held at LASL in March 1978. Models in this initial version are directed towards pressurized-water reactors (PWRs) and the analysis of large-break LOCAs. An improved version, TRAC-P1A, was released through the National Energy Software Center (NESC) in March 1979 together with a user's manual.[8] Reports[9,10] containing detailed results of TRAC-P1A developmental and independent assessment analyses have also been published. TRAC-PD2, the latest detailed PWR version, was released at the request of NRC to a limited number of organizations in July, 1980 prior to completion of the code documentation and developmental assessment. A public release of TRAC-PD2 was made this month along with draft documentation.[11] Developmental assessment results for TRAC-PD2 are being documented in a separate report.[12]

Progress during the past year was highlighted by the release of TRAC-PD2. LASL personnel who contributed to the development and developmental assessment of TRAC-PD2 are listed in Fig. 5. Some of the more significant improvements of PD2 over P1A are listed in Fig. 6. The major improvements are in the areas of reflood and heat transfer models, interfacial relations, and numerical methods. In the reflood area, quench front motion is now determined explicitly from a two-dimensionl (R-Z) conduction solution in the clad using a dynamic fine-mesh rezoning technique in the axial direction. Much of the development effort since P1A was spent on improving the constitutive relations and numerical methods. Needed improvements in those areas became evident as TRAC was applied to a wider variety of problems both at LASL and elsewhere. One important example is mass conservation, which was a problem for long-term transients in P1A. This is no longer a significant problem in PD2 which can run TMI-type small-break calculations with very good mass conservation. In summary, PD2 is a much more reliable code than P1A and is faster-running for most large problems.

Other improvements (Fig. 7) have been made since the last Information Meeting but are not included in the official PD2 version. Some of these improvements have not been fully implemented but many are already in use at LASL and have received limited testing. All of these improvements will eventually be included in future code versions (i.e., PD2 mods, PF1, or PD3). Features to be included in the initial fast-running small-break code, TRAC-PF1, are described in a separate paper.[1]

The basic developmental assessment set (Fig. 8) was completed in July 1980 although sensitivity calculations continued beyond that date for some of the problems. The PD2 assessment set is greatly expanded over that used for P1A; it includes additional integral, reflood, and heat transfer tests. In summary, PD2 yields basically the same results as P1A for blowdown and refill but reflood results are considerably improved. The improved reflood results are illustrated by Fig. 9 which shows the clad temperature at the 8 ft elevation for FLECHT Test 17201. This test is from the low forced-flooding-rate skewed test series.[13] TRAC-PD2 predictions of the FLECHT quench times are generally late whereas P1A tended to quench early. Radiative heat transfer was found to be significant in these tests. When accounted for in the PD2 calculation,[14] the calculated results were significantly improved as shown in Fig. 9. Another example of improved performance is shown in Fig. 10. Without resorting to a special T_{min} correlation, PD2 predicts the early core rewet that occurred in LOFT Test L2-2. The developmental assessment problems for TRAC-PD2 did not include gravity-feed reflood tests. However, the results for the CCTF system reflood test indicate that further work is needed to improve entrainment and liquid carryover under gravity reflood.

Fig. 1. TRANSIENT REACTOR ANALYSIS CODE (TRAC)

OBJECTIVE: Validated LWR System Analysis Capability

APPROACH: Advanced Best-Estimate Mechanistic Models
Versatility and Flexibility
Extensive Testing

SAFETY ISSUES: Multidimensional and System Effects
Safety Margins During System Transients
Conservatism in Licensing Codes
Safety Impact of Design Features and Operating Procedures

Fig. 2. TRAC SCOPE

- Model and Methods Development

  Field Equations, Interfacial Relations, Heat Generation and Transfer, Numerical Methods, Solution Algorithms, etc.

- Code Development

  Code Architecture, Programming and Debugging, Data Storage and Management, Preprocessors and Postprocessors, etc.

- Developmental Assessment

  Posttest Analyses, Model Improvement

- Independent Assessment

  Test Predictions, Frozen Models

- Applications

  LWR Transients, Licensing Questions, New Facilities

Fig. 3. TRAC FEATURES

- Multidimensional Fluid Dynamics
- Nonhomogeneous, Nonequilibrium Models
- Flow-Regime-Dependent Constitutive Equations
- Comprehensive Heat Transfer (Including Mechanistic Reflood Quench Front Capability)
- Transient and Steady-State Calculations
- All LOCA Phases
- Modular Programming
- Efficient Solution Strategies
- Trips, Graphics, Restart

Fig. 4. TRAC RECORD

4/75 - Project Initiated
10/75 - Programming Started
7/76 - Rudimentary 1-D Version Completed
4/77 - Rudimentary 3-D Version Completed
12/77 - TRAC-P1 Released to NRC
4/78 - First Complete PWR LOCA Calculation
1/79 - First Pretest Prediction (L2-2)
3/79 - TRAC-P1A Released to NESC
10/79 - Successful Application to TMI Accident
1/80 - New Reflood Model Installed
7/80 - Limited TRAC-PD2 Release
10/80 - TRAC-PD2 Released to NESC

Fig. 5. TRAC-PD2 DEVELOPMENT CREDITS

Models and Methods:	D. R. Liles J. H. Mahaffy F. L. Addessio D. A. Mandell S. B. Woodruff
Code Development:	J. M. Sicilian R. P. Harper M. R. Turner J. R. Netuschil
Developmental Assessment:	T. D. Knight J. M. Sicilian J. S. Gilbert J. K. Meier T. F. Bott
Project Management:	R. J. Pryor

Fig. 6. TRAC-PD2 IMPROVEMENTS OVER TRAC-P1A

- REFLOOD Model
  Quench Front Motion, Dynamic Fuel Gap
- VESSEL SOLUTION STRATEGY
  Direct Inversion, Coarse-Mesh Rebalance, Relinearization
- VESSEL CONSTITUTIVE PACKAGE
  Entrainment, Interface Sharpener, Interfacial Drag and H.T., Wall Shear
- HEAT TRANSFER PACKAGE
  Boiling Curve, Wall H.T. Correlations
- MASS CONSERVATION
  Vessel Numerics, Slip Package
- WATER PACKING
  Detection, Prevention
- OTHER
  Thermal and Material Properties, Hydrogen Generation, Time-Step Control, Graphics, Break and Fill Models, Error Corrections, etc.

Fig. 7. OTHER TRAC DEVELOPMENT PROGRESS

- ADDITIONAL CAPABILITIES
  Noncondensable Gas Field, Droplet Field, Horizontal Flow Map, Reactivity Feedback, Break Flow, Delayed Nucleation, 1-D Core, Vessel Vent Valve, Radiative H.T.
- IMPROVED MODELS
  1-D 2-Fluid Hydodynamics, Distributed Heat Slab, Pressurizer, Steam Generator, Valve
- IMPROVED NUMERICAL METHODS
  2-Step Stability-Enhancing Numerics, Fully Implicit Wall H.T., Second-Order Time-Differencing
- CODE IMPLEMENTATION
  CRAY and VAX, FRAPCON, FRAP-T, QUANDRY

Fig. 8. TRAC-PD2 DEVELOPMENTAL ASSESSMENT SET

EDWARDS (SP 1)
CISE (TESTS 4 and R)
MARVIKEN (TESTS 4 and 24)
BENNETT (TESTS 5336, 5431, 5442)
SEMISCALE (S-02-8 and S-06-3)
CREARE (2 FLOODING CURVES)
LOFT (L1-4 and L2-2)
FLECHT-SET (TESTS 4831 and 17201)
FLECHT-SEASET (TEST 4)
THTF (TEST 177)
CCTF (TEST C1-1)

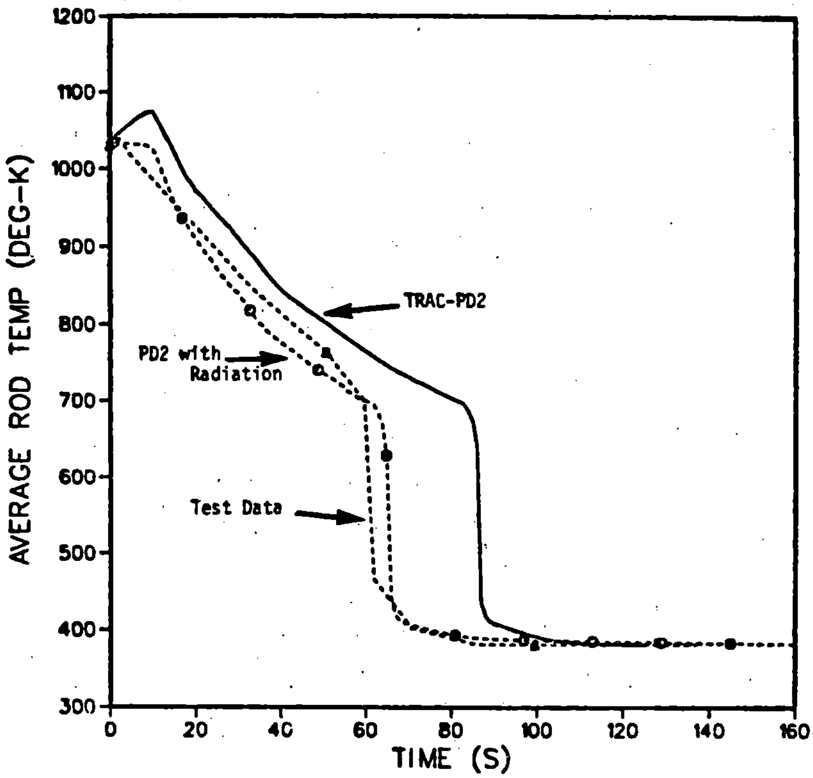

Fig. 9. FLECHT Test 17201 8 ft elevation rod clad temperatures.

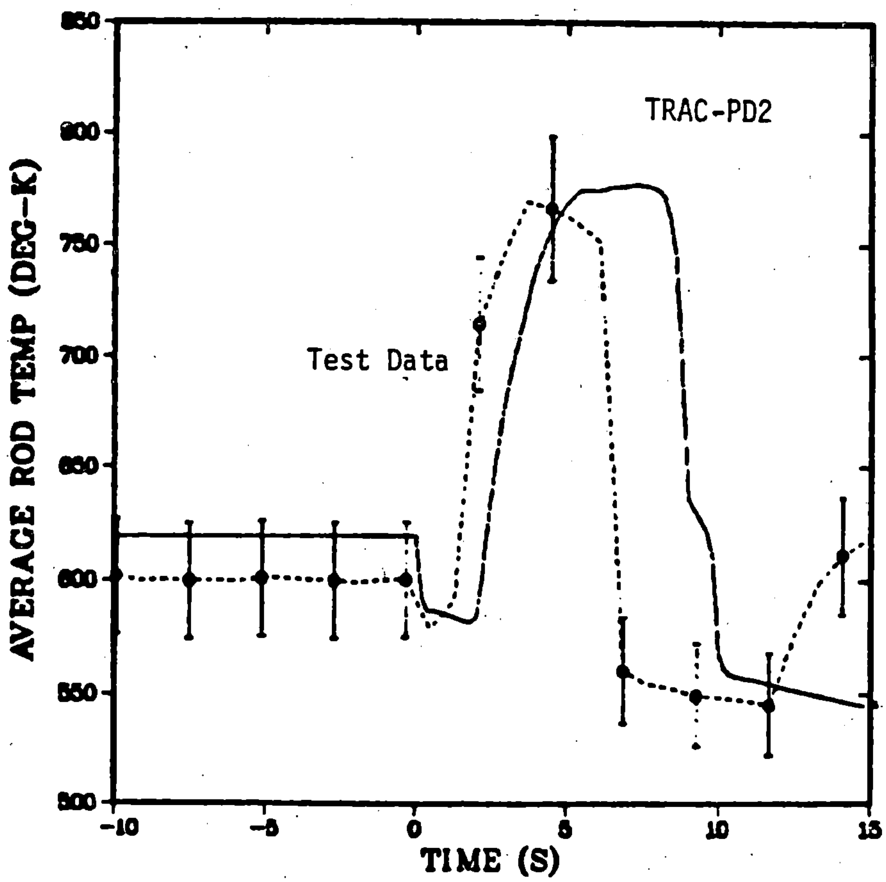

Fig. 10. Midplane clad temperature comparison for LOFT Test L2-2.

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3. K. A. Williams, "TRAC Analysis Support for the 2D/3D Program," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland (October 27-31, 1980).

4. R. D. Burns, "Loss of Feedwater Transients in PWRs," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland (October 27-31, 1980).

5. M. Thurgood, "COBRA/TRAC Description and Application," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland (October 27-31, 1980).

6. F. Aguilar, "TRAC-BD1 Description and Application," Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland (October 27-31, 1980).

7. J. C. Vigil, R. J. Pryor, and J. F. Jackson (compilers), "TRAC-P1: An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," Los Alamos Scientific Laboratory report LA-7279-MS, NUREG/CR-0063 (June 1978).

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TRAC-BD1, TRANSIENT REACTOR ANALYSIS CODE FOR BOILING WATER SYSTEMS

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

F. Aguilar
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Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

TRAC-BD1, TRANSIENT REACTOR ANALYSIS CODE FOR BOILING WATER SYSTEMS

F. Aguilar
J. W. Spore
EG&G Idaho, Inc.

The TRAC-BD1 code is under development at the Idaho National Engineering Laboratory (INEL) with the objective of providing the Nuclear Regulatory Commission (NRC) and the nuclear industry with a best estimate capability for the detailed analysis of loss-of-coolant accidents (LOCA) in boiling water reactor (BWR) systems. Another objective is to provide analytical support to experimental BWR safety programs sponsored by NRC. The TRAC-BD1 project is a very new development effort, and it is based on a version of TRAC[3] supplied by the Los Alamos Scientific Laboratory (LASL) which uses two-fluid hydrodynamics for one-dimensional flow (TF1D model). In this first year of TRAC-BD1 development, a new approach to modeling BWR systems with TRAC has been proposed. The feasibility of the analytical scheme has been demonstrated with a system calculation. Finally, new models and code improvements have been implemented in the TF1D version of TRAC that now provides a basic capability for BWR LOCA analysis. The good progress achieved to date is attributable in large measure to the active participation of the General Electric Company. General Electric has contributed a number of models, mainly in the areas of constitutive relations and heat transfer. This summary describes the model content of TRAC-BD1 and the results of some developmental assessment calculations.

TRAC-BD1 provides distinct models for the hardware components that distinguish BWR systems: shrouded fuel bundles, jet pumps, and separators and dryers. The code has two-fluid hydrodynamics, a new model to limit countercurrent flow (CCFL), and an upgraded constitutive package. These improve the capability to calculate countercurrent flow and CCFL breakdown, hydrodynamic phenomena of critical importance to BWR safety analysis. TRAC-BD1 also contains a number of features that are not BWR-specific: a choked flow model and a decay heat model that incorporates the recent ANS/ANSI Standard.[4] The capability of the heat "slab" model in the TRAC pressure vessel component (VESSEL) has been upgraded so that the slab may exchange heat on each surface with VESSEL hydrodynamic cells. This permits the calculation of heat transfer through a BWR core shroud or through the reactor vessel itself. Similarly, the wall heat transfer model for the TRAC pipe component (PIPE) has been generalized to allow coupling to the VESSEL component. This permits the calculation of stored energy release from control rod guide tubes which are modeled with PIPE components.

The INEL approach to modeling a BWR system with TRAC is based on a new component called CHAN, that simulates a fuel bundle and cannister assembly. CHAN components are connected across the usual core region of the VESSEL component. Three-dimensional core flow in the bypass region is handled by the usual VESSEL hydrodynamics. The TF1D hydrodynamics of CHAN are borrowed from the PIPE component. CHAN has a detailed radiation model which permits heat exchange between several rod groups, the coolant, and the cannister wall. CHAN can calculate top-down and bottom-up quenching of individual rod groups and the cannister. The modeling of heat transfer and leakage flow from the cannister to the core bypass region is also possible with CHAN.

The CHAN component has been designed so that it may be used independently of the VESSEL component. The Thermal Fuels Behavior Program at INEL has taken advantage of this versatility to do scoping analyses for the Severe Fuel Damage Test. Calculations of bundle heating in a low-velocity steam environment have been performed four times faster than real time. CHAN has also been used to analyze a radiation heat transfer experiment on the Swedish Gota Test Loop[2,5] with excellent results.

Developmental assessment of the new jet pump component and the CCFL and choked flow models is in progress. The jet pump component has been found to agree well with experimental data including the 1/6-scale data obtained for abnormal operating conditions.[6] The CCFL and choked flow models have also been found to agree satisfactorily with data.[7,8,9] Areas of continued development by both INEL and GE are upper plenum spray mixing, CCFL breakdown, level swell, and jet pump uncovery.

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INEL TRAC-BWR DEVELOPMENT
TRAC-BD1

by

J.W. SPORE

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TRAC-BD1 OBJECTIVES

- PROVIDE BASIC CAPABILITY FOR MODELING OF BWR LOCA
- DESIGN NEW BWR MODELING APPROACH
- IDENTIFY ADDITIONAL MODELING DEVELOPMENT REQUIREMENTS
- PROVIDE ANALYSIS CAPABILITY FOR SUPPORT OF BWR EXPERIMENTAL PROGRAMS

JWS-1

TRAC-BD1 DEVELOPMENT

- BWR LOCA MODELING
- TRAC-BD1 MODELS
- TRAC-BD1 APPLICATIONS

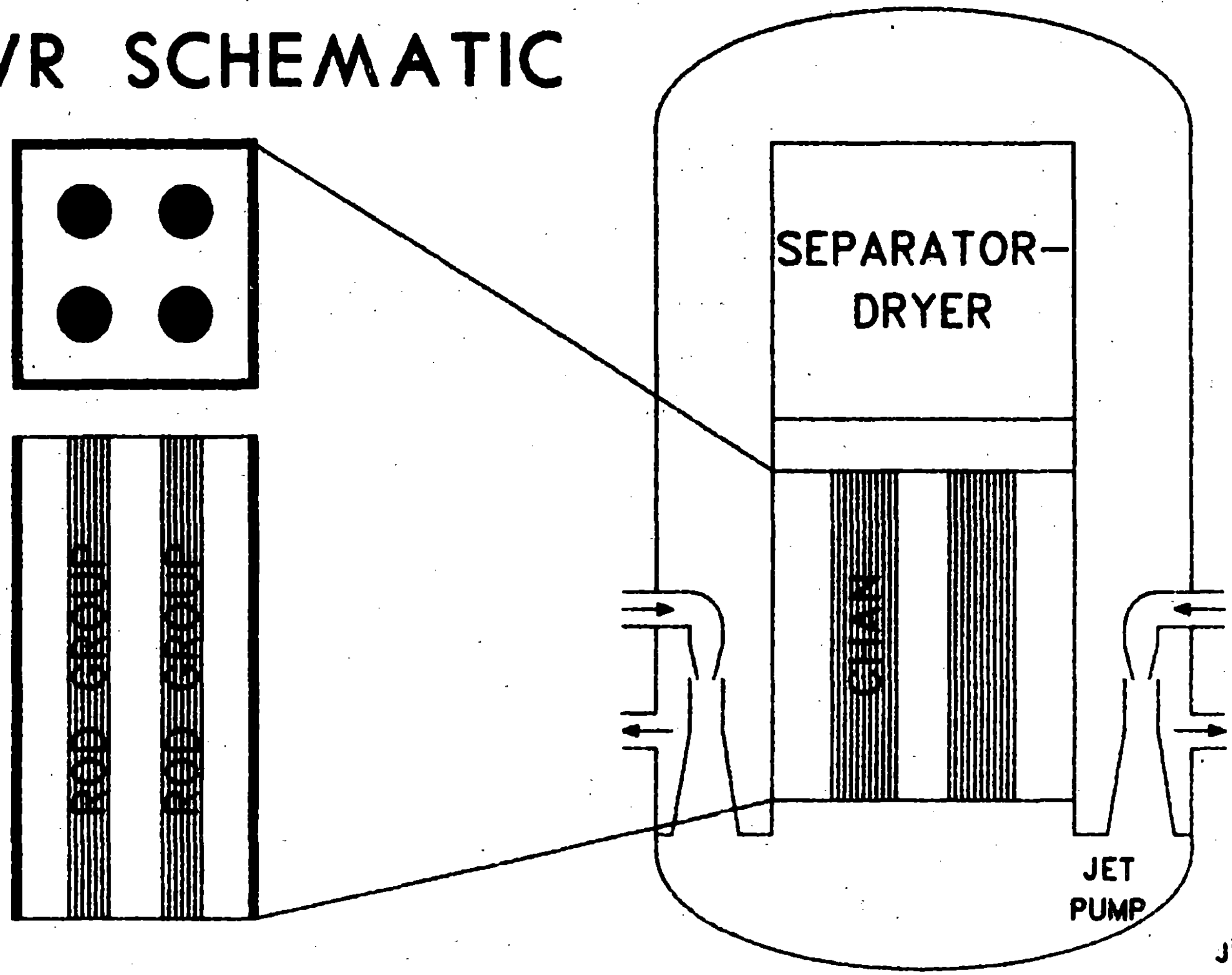
JWS-2

MODELING CONSIDERATIONS FOR BWR LOCA MODEL

- GEOMETRY
- BWR COMPONENTS
- HEAT TRANSFER
- HYDRODYNAMICS

JWS-3

BWR SCHEMATIC



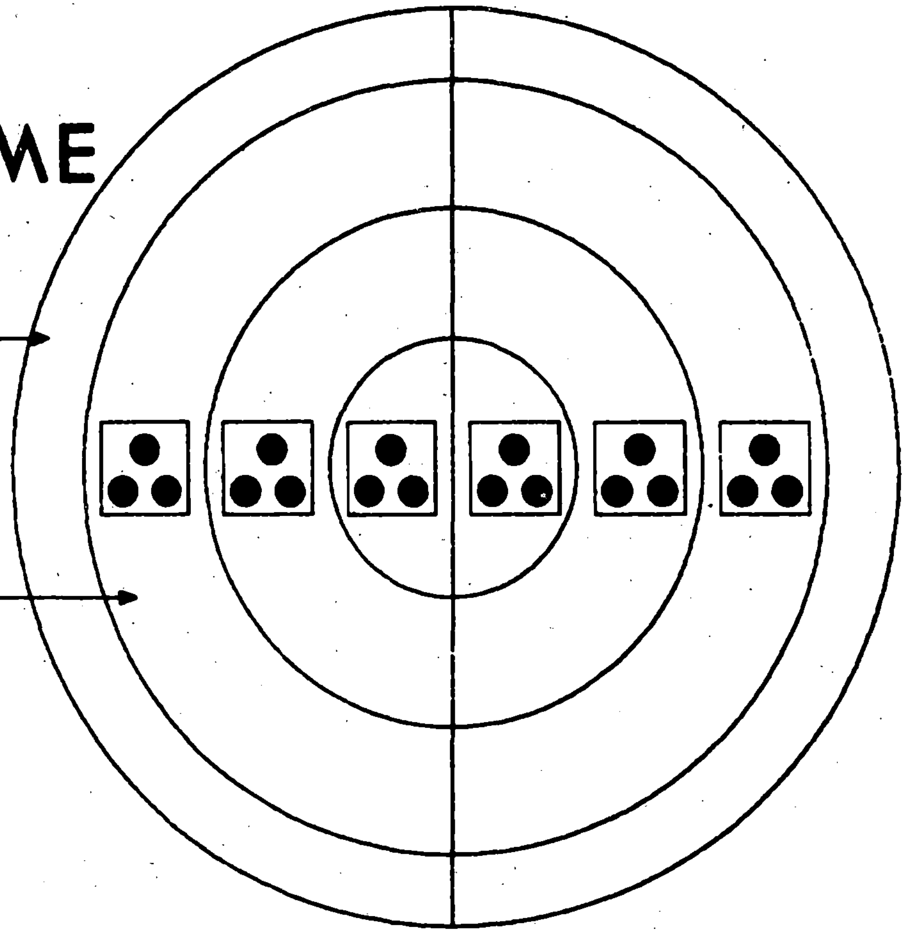
JWS-4

TRAC-BD1 MODELS

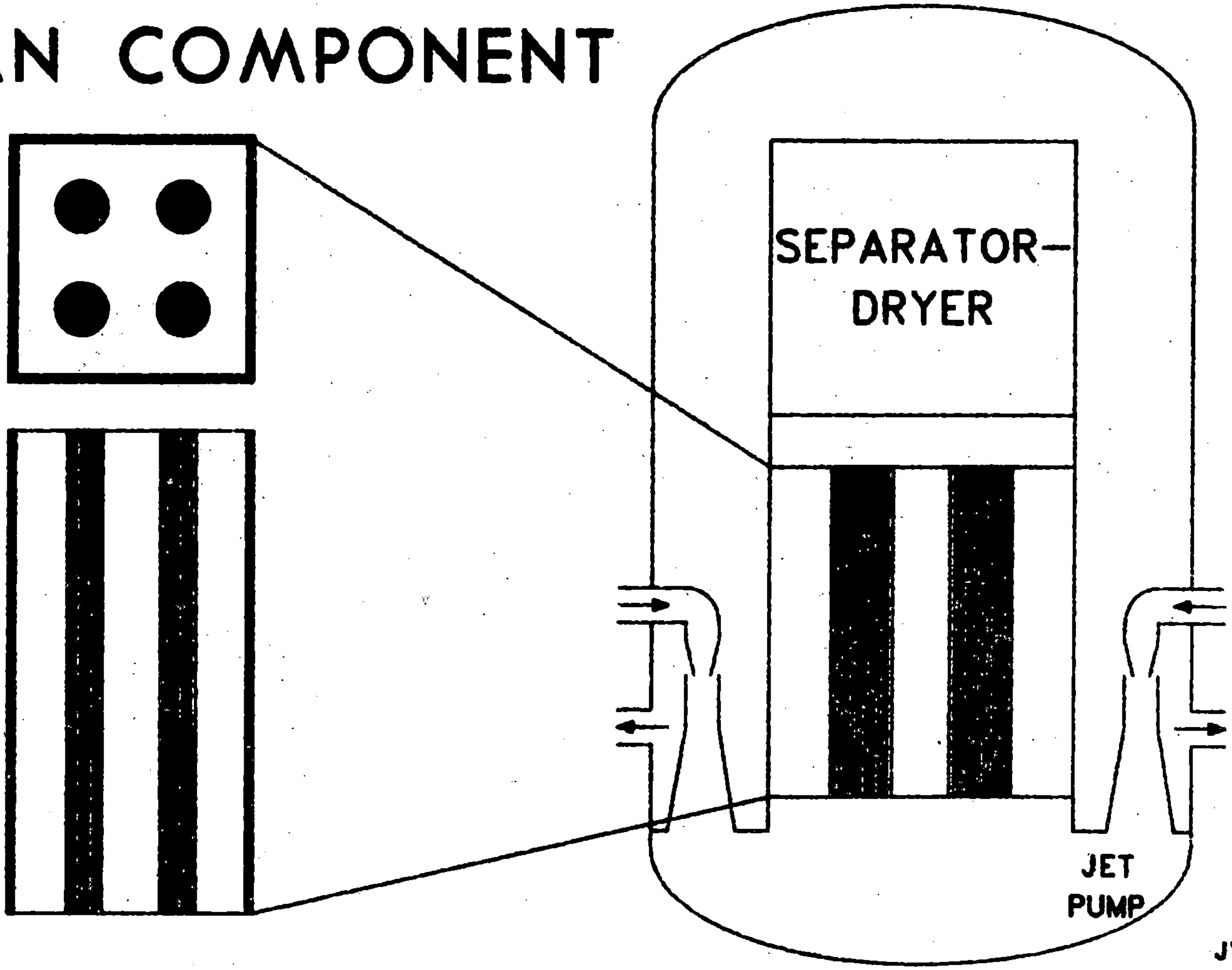
- CHAN COMPONENT
- JET PUMP COMPONENT
- SEPARATOR-DRYER
- COUNTER CURRENT FLOW LIMITING MODEL
- CHOKING MODEL
- DOUBLE-SIDED HEAT SLAB
- GENERALIZED PIPE HEAT TRANSFER
- 1979 ANS DECAY HEAT STANDARD
- MOMENTUM CONVECTIVE TERM

JWS-5

TRAC-BD1
NODING SCHEME

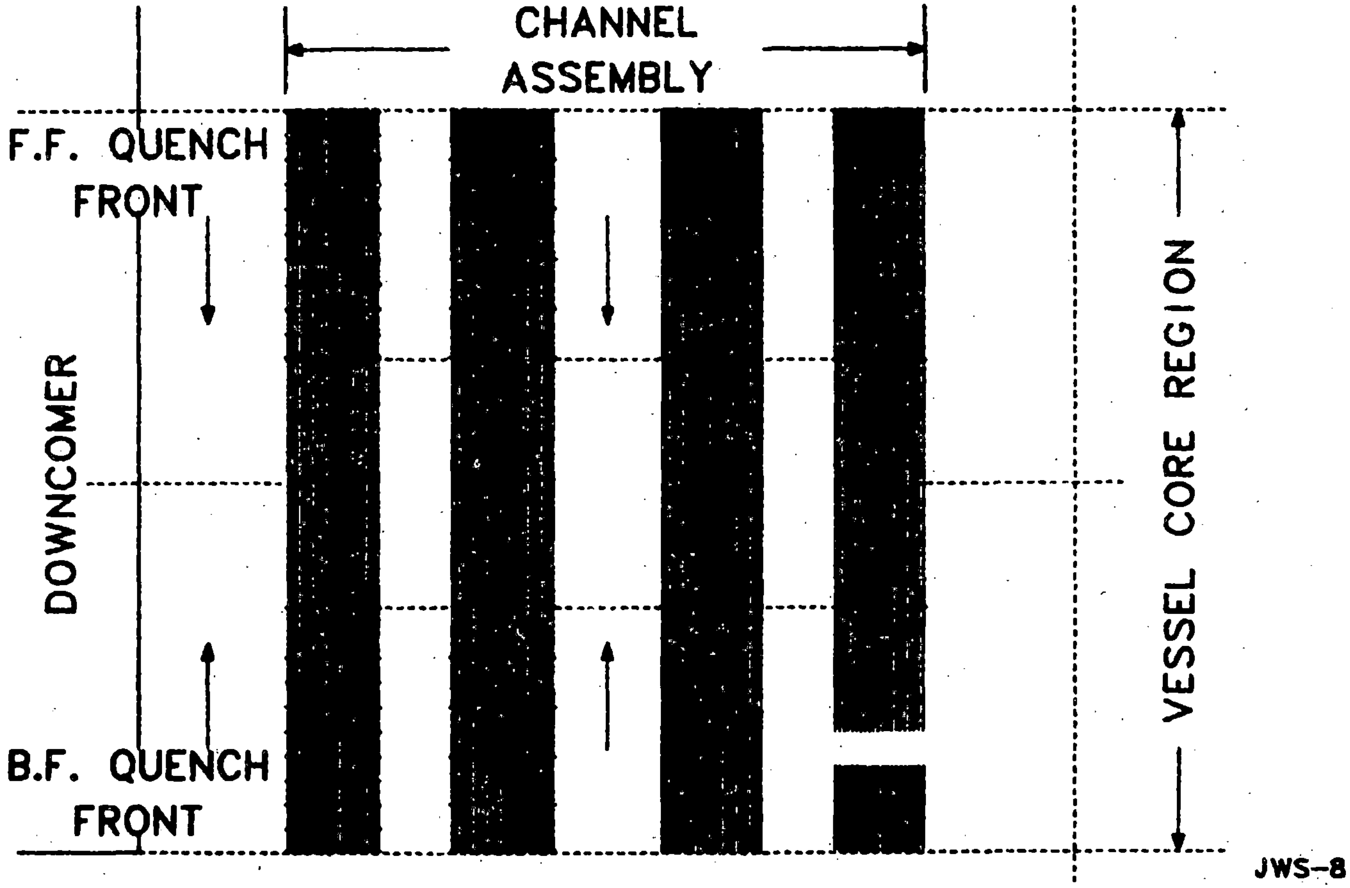


CHAN COMPONENT

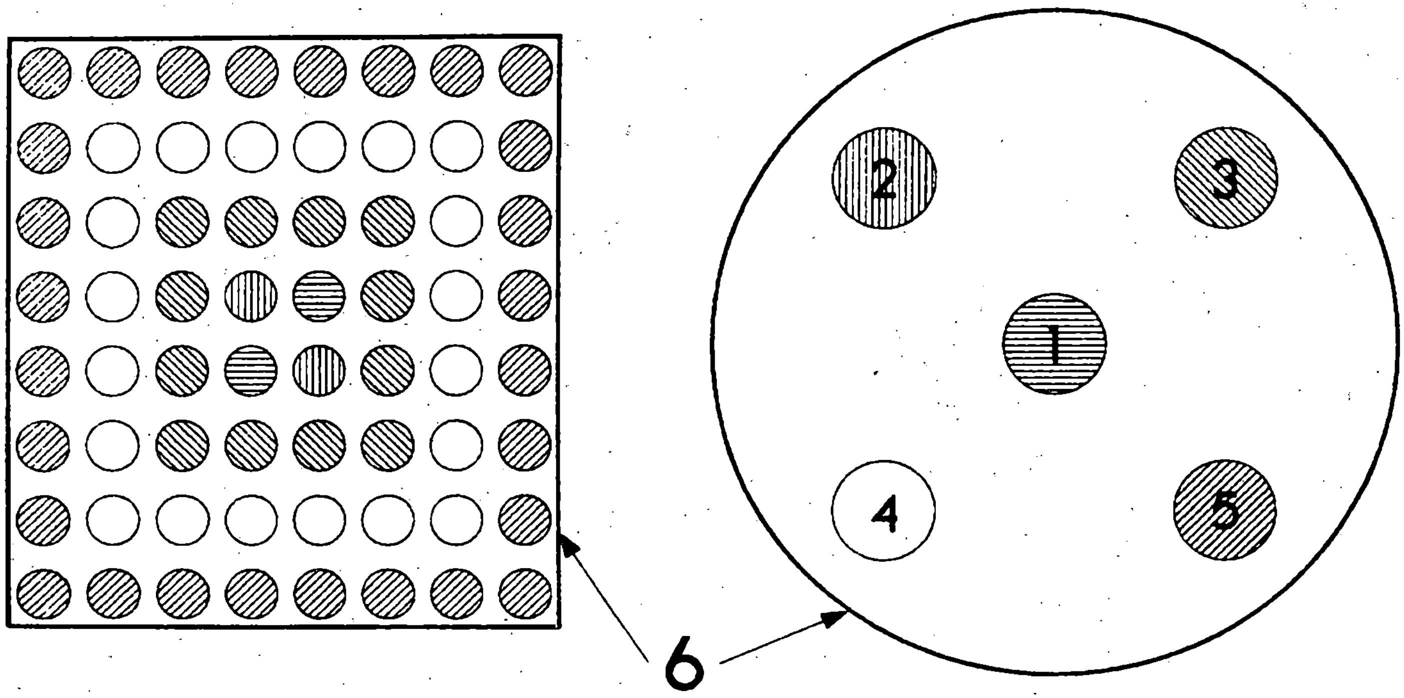


JWS-7

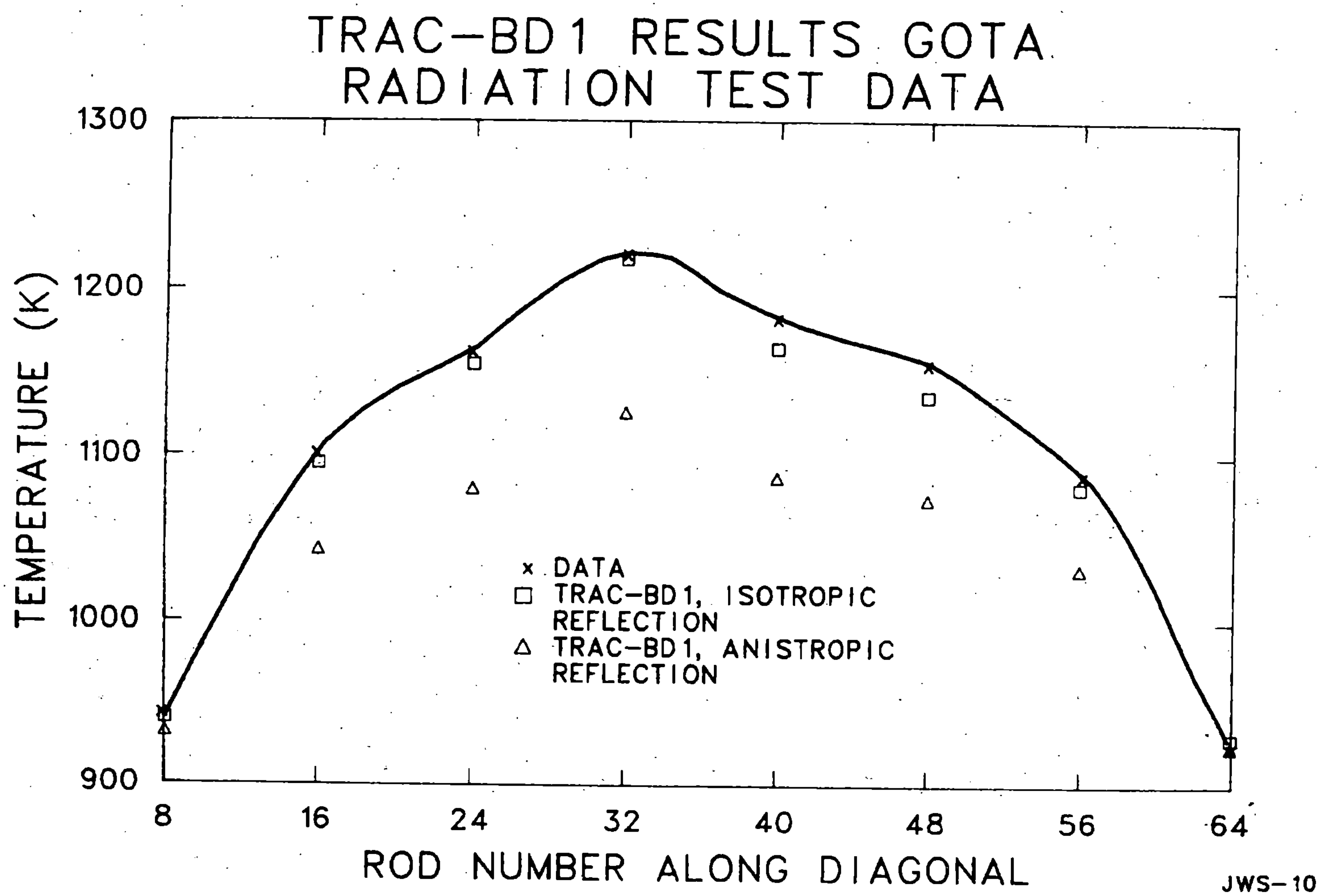
CHAN COMPONENT



ROD GROUPS FOR BWR6 8x8

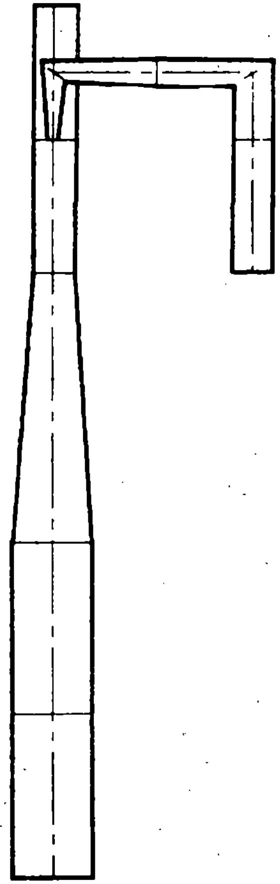


JWS-9



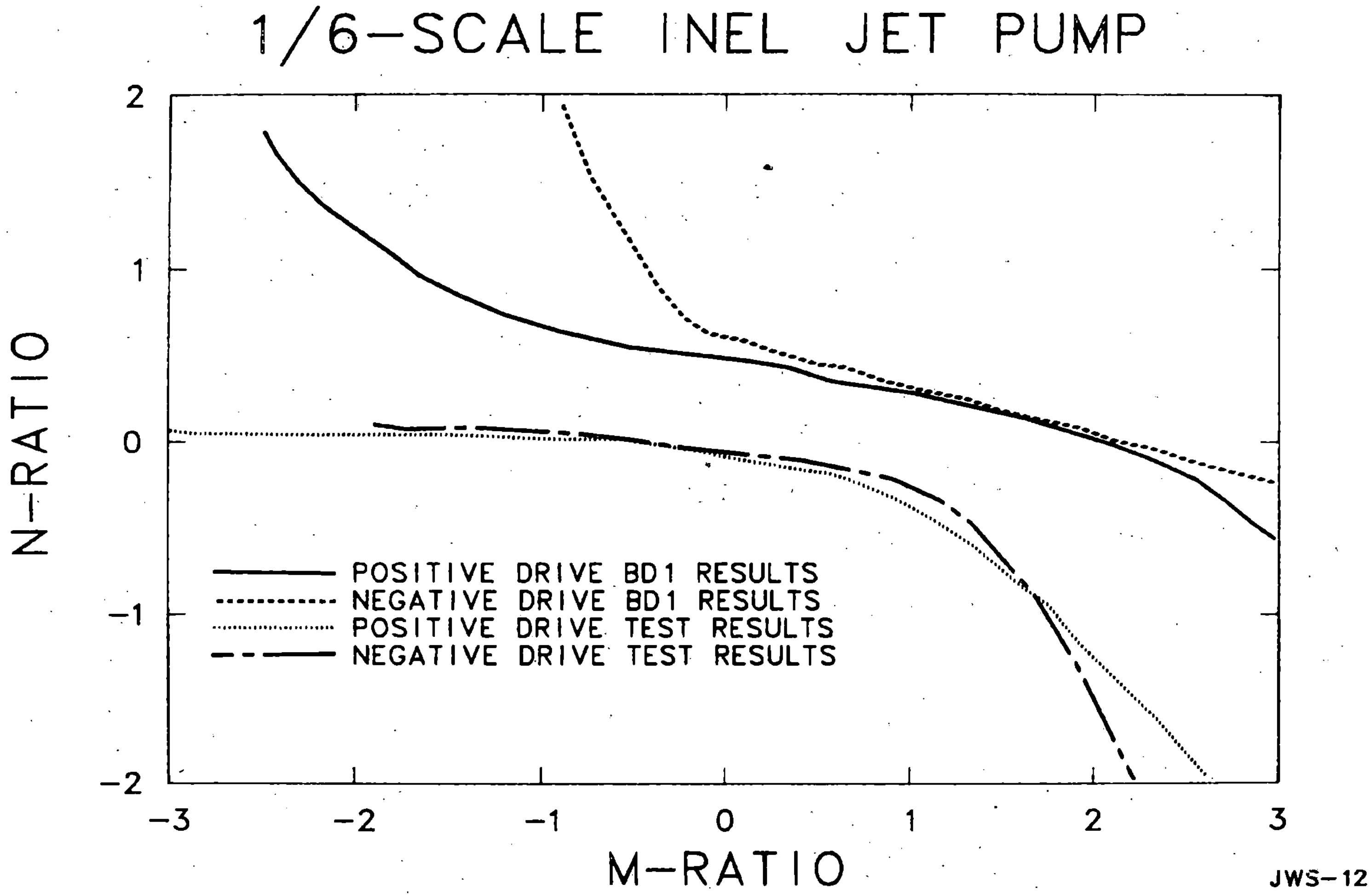
TRAC-BD1 RESULTS GOTA
RADIATION TEST DATA
1300
1200
1100
1000
900
TEMPERATURE (K)
× DATA
□ TRAC-BD1, ISOTROPIC REFLECTION
△ TRAC-BD1, ANISTROPIC REFLECTION
8
16
24
32
40
48
56
64
ROD NUMBER ALONG DIAGONAL
JWS-10

JET PUMP MODEL

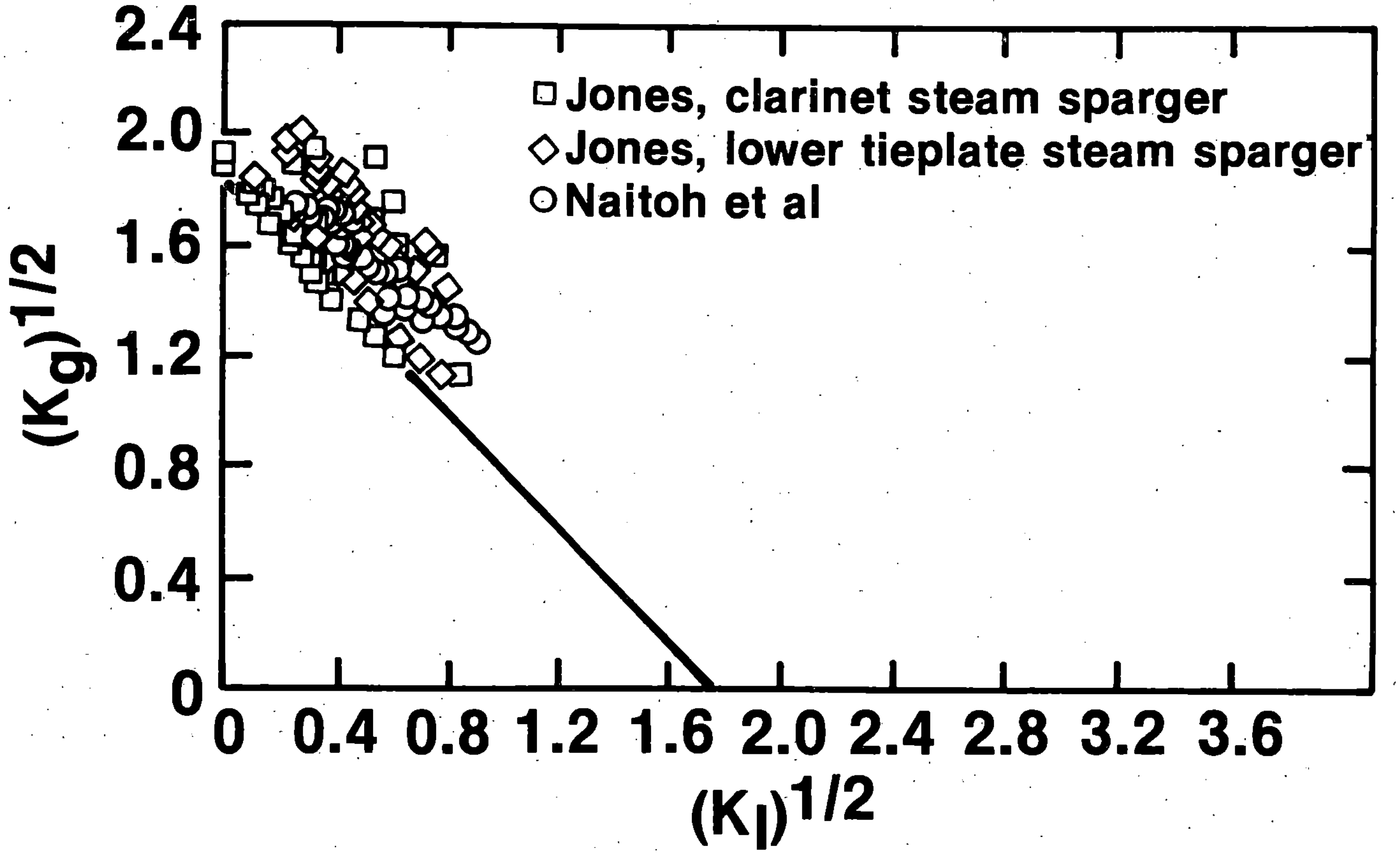


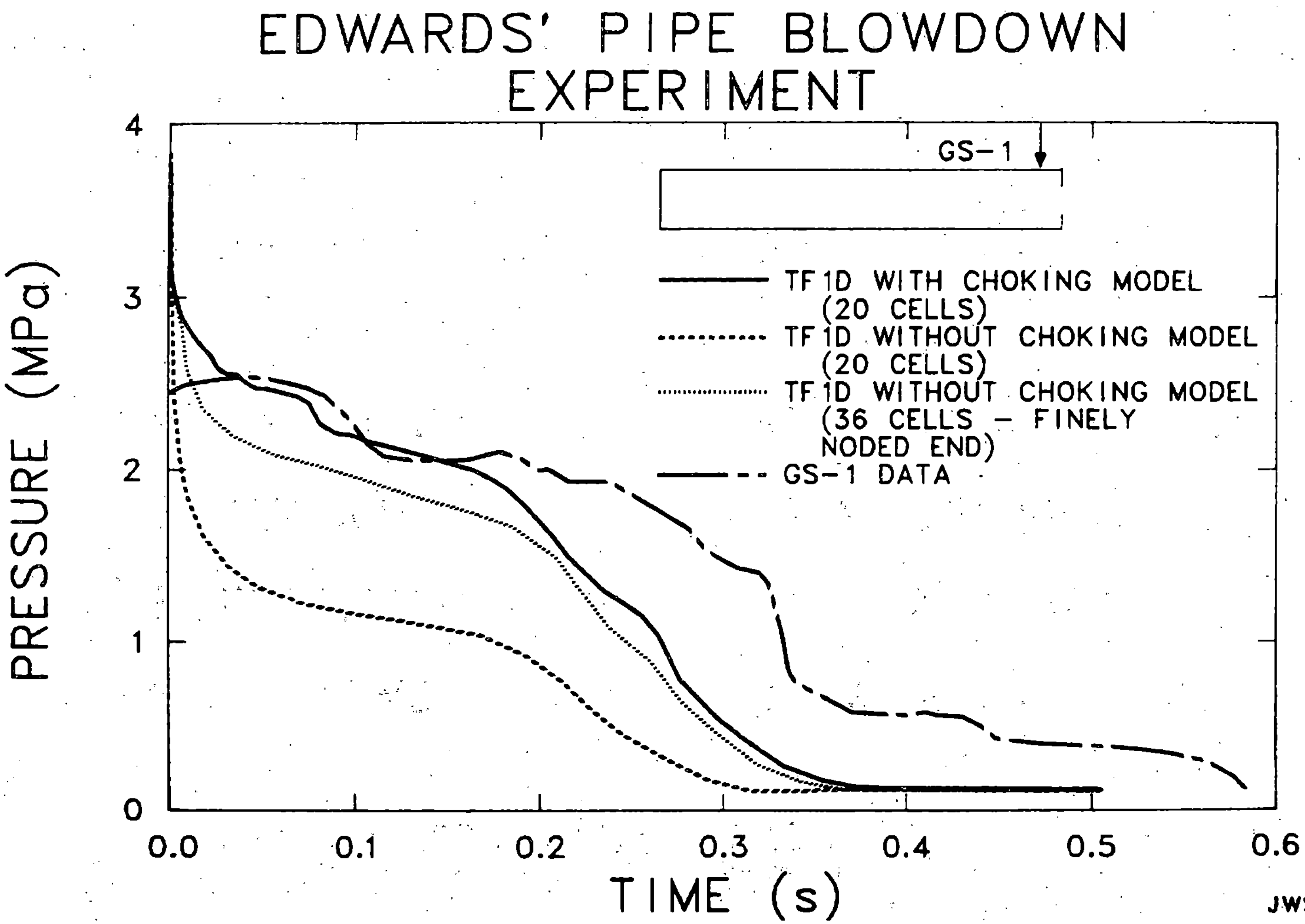
- REDUCED NUMBER OF CELLS
  20 TO 8
- CORRECTED MOMENTUM
  SOLUTION
- USER CONVENIENT
  INPUT

JWS-11


1/6-SCALE INEL JET PUMP
N-RATIO
M-RATIO
2
1
0
-1
-2
-3
-2
-1
0
1
2
3
POSITIVE DRIVE BD1 RESULTS
NEGATIVE DRIVE BD1 RESULTS
POSITIVE DRIVE TEST RESULTS
NEGATIVE DRIVE TEST RESULTS
JWS-12

Comparison of Data with TRAC Counter Current Flow Limit Line





EDWARDS' PIPE BLOWDOWN
EXPERIMENT
GS-1
TF1D WITH CHOKING MODEL (20 CELLS)
TF1D WITHOUT CHOKING MODEL (20 CELLS)
TF1D WITHOUT CHOKING MODEL (36 CELLS - FINELY NODED END)
GS-1 DATA
PRESSURE (MPa)
0
1
2
3
4
0.0
0.1
0.2
0.3
0.4
0.5
0.6
TIME (s)

JWS-14

TRAC-BD1 TEAM

F. AGUILAR	R. W. SHUMWAY
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S. T. FREE	J. W. SPORE
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G. A. JAYNE	J. E. TOLLI
C. M. MOHR	K. TRAMBAUER
R. E. PHILLIPS	W. L. WEAVER

JWS-15

TRAC-BD1

- BASIC CAPABILITY FOR DETAILED DBLOCA
- TWO-FLUID APPROACH 1- AND MULTI-D COMPONENTS
- BWR COMPONENT MODELS
  - FUEL ASSEMBLY (RODS AND CHANNEL)
  - BYPASS LEAKAGE PATHS
  - JET PUMP
  - SEPARATORS AND DRYERS

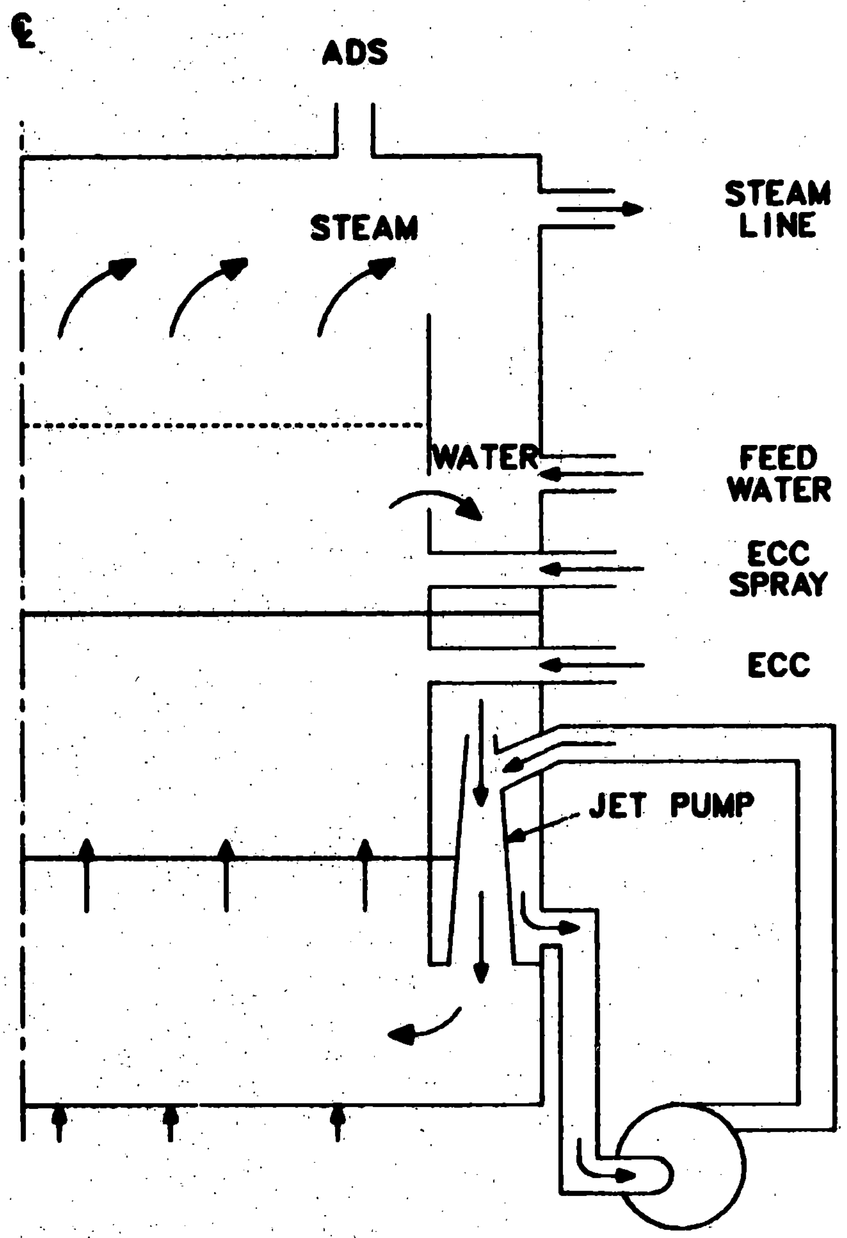
JWS-16

TRAC-BD1 (CONTD.)

- COUNTER CURRENT FLOW LIMITING MODEL
- BEST-ESTIMATE DECAY HEAT (ANSI/ANS-5.1-1979)
- CRITICAL FLOW MODEL
- GENERALIZED HEAT SLAB AND PIPE WALL HEAT TRANSFER
- TRIP ON DOWNCOMER LEVEL
- IMPROVEMENTS TO INPUT STREAM, GRAPHICS, AND TRACEBACK CAPABILITY

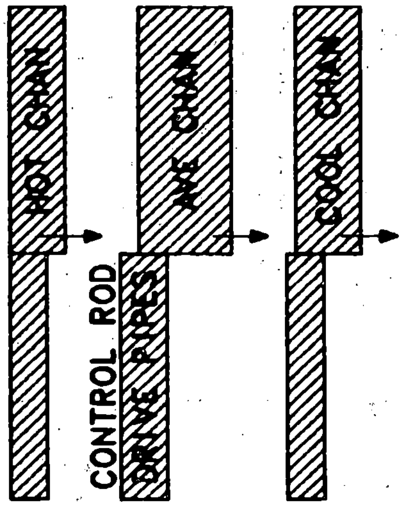
JWS-17

TRAC-BD1 BWR/6 MODEL



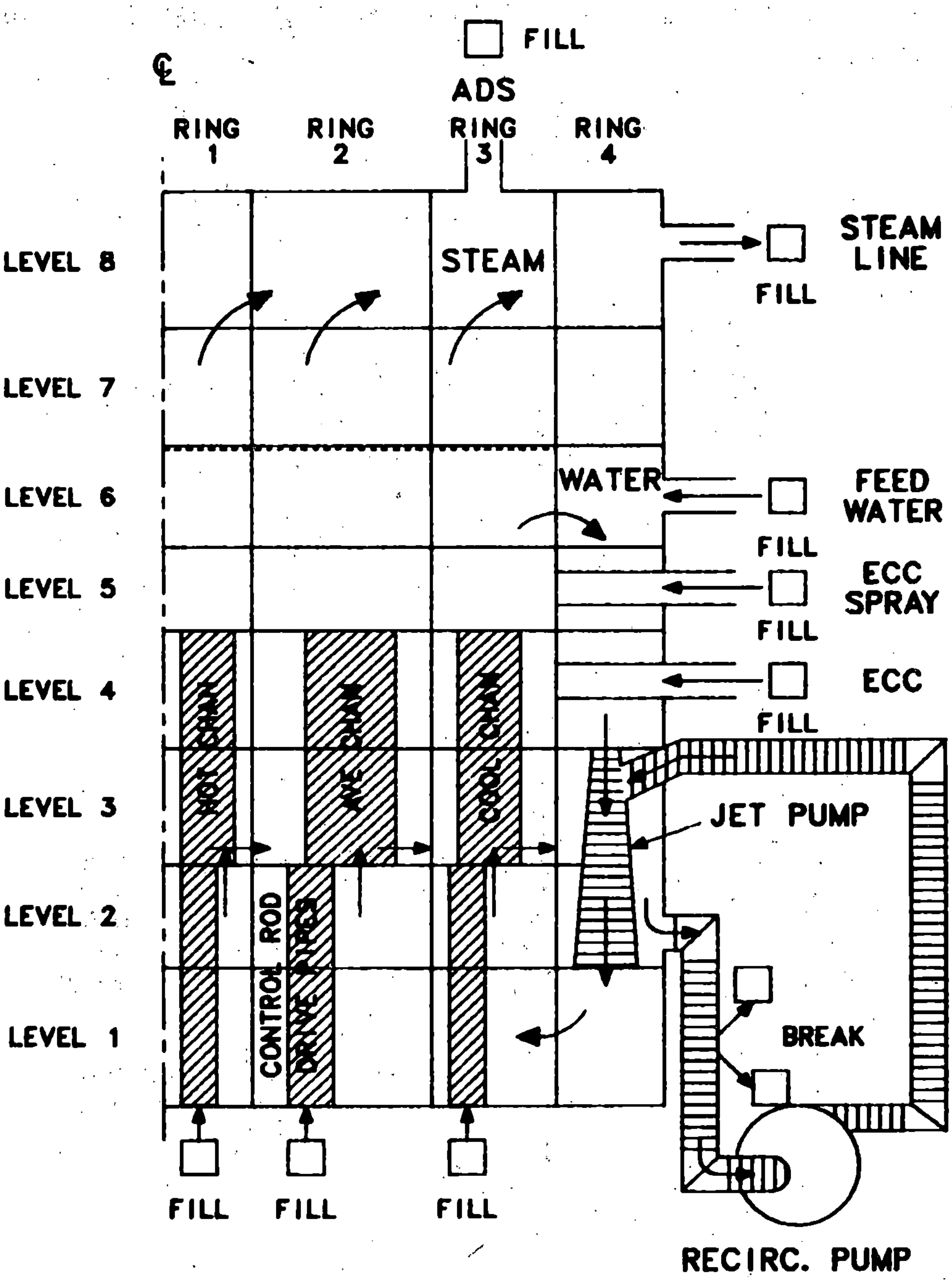
JWS-18

TRAC-BD1 BWR/6 MODEL



JWS-19

TRAC-BD1 BWR/6 MODEL



JWS-20

LA-UR 80-3008

TITLE: LASL TRAC SMALL-BREAK MODELING

AUTHOR(S): Dennis R. Liles, Q-8
John H. Mahaffy, Q-8

SUBMITTED TO: Eighth Water Reactor Safety Research Information Meeting
October 29, 1980
Gaithersburg, Maryland

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DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

LASL TRAC SMALL-BREAK MODELING

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Presented at the
EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

October 28, 1980
Gaithersburg, Maryland

TRAC Code Development

(PF1)

Summary

The accident at Three Mile Island has increased the need for a best-estimate fast-running small-break code. LASL is responding to this challenge with the accelerated development of the first TRAC small-break version (TRAC - PF1). This code which is due for completion in May, 1981 will have greatly improved 1-D numerics, a full two-fluid model, and other constitutive changes appropriate for small breaks. It will be capable of running TMI-type transients in a full 1-D mode in a fraction of real time.

The numerics have been modified so that the material Courant condition can be violated in the 1-D components. A core model has been added to the pipe module to permit the calculation of transients in a fully one-dimensional mode. The three-dimensional vessel module will be retained to permit mixed 1-D, 3-D analyses. A noncondensible gas field is also slated for incorporation in PF1.

Numerous constitutive changes should improve the physical modeling in the code. A horizontal flow regime with appropriate interfacial exchange terms will be included. A choking model will permit a less costly calculation of the broken pipes.

In general PF1 will provide versatile, accurate modeling of small-break LOCAs. It will allow both rapid one-dimensional simulations and more expensive detailed calculations to be performed.

TRAC Small Break Modeling

(by D. Liles and J. Mahaffy)

Purpose: To provide a best-estimate fast-running computer code which will enable the NRC to efficiently assess the consequences of different types of small breaks and multiple fault transients.

GOAL: TO PRODUCE A CODE WHICH WILL ACCURATELY PREDICT TMI-TYPE TRANSIENTS AT LEAST THREE TIMES FASTER THAN REAL TIME.

TRAC-PF CHARACTERISTICS

- Full Two-Fluid Equations
- Noncondensable Gas Field
- Improved Thermal-Hydraulic Models
  - Horizontal Flow Map
  - Entrainment
  - Interfacial Interaction Terms
  - Wall Heat Transfer
  - Delayed Nucleation
- Small-Break Features
  - 1-D Core Model
  - Break-Flow Model
  - Reactivity Feedback
- Improved Methods
  - Two-Step Numerics
  - More Implicit Wall H.T.
- Improved Component Models
  - Valve
  - Steam Generator
  - Pressurizer

TRAC PF1 NUMERICS

presented by

John Mahaffy

LOS ALAMOS SCIENTIFIC LABORATORY

NUMERICAL IMPROVEMENTS

- Velocities in 1–D components no longer restrict the time step size
- More implicit treatment of the pump momentum source terms
- More stable method for treating friction force terms
- Heat transfer made more implicit

1–D MASS CONSERVATION

$$\partial\rho/\delta \mathrm{t} \quad + \quad \partial(\rho \mathrm{v})/\partial \mathrm{x} \quad = \quad 0$$

TWO STEP DIFFERENCE EQUATIONS

$$(\tilde{\rho}_j^{n+1} - \rho^n_j) / \Delta t \ +$$
$$(\rho_j^n \ v^{n+1}_{j+1/2} - \rho^n_{j-1} v^{n+1}_{j-1/2}) / \Delta x$$
$$= 0$$

$$(\rho_j^{n+1} - \rho^n_j) / \Delta t \ +$$
$$(\rho_j^{n+1} \ v^{n+1}_{j+1/2} - \rho^{n+1}_{j-1} v^{n+1}_{j-1/2}) / \Delta$$
$$= 0$$

RELAP5 - AN ADVANCED FAST RUNNING LWR TRANSIENT ANALYSIS CODE

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

V. H. Ransom, R. J. Wagner, K. E. Carlson, D. M. Kiser,
H. H. Kuo, D. L. Slegel, H. Chow
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

RELAP5 - AN ADVANCED FAST RUNNING LWR TRANSIENT ANALYSIS CODE

EG&G Idaho, Inc.

V. H. Ransom, R. J. Wagner, K. E. Carlson, D. M. Kiser,
H. H. Kuo, D. L. Slegel, H. Chow

RELAP5 is an advanced, fast running, one-dimensional, transient, light water reactor (LWR) system transient analysis code for use in LWR safety analysis. RELAP5 is faster running and includes more complete physical models than the present best estimate licensing code. This capability will provide more accurate and responsive analytical support to safety related licensing efforts. Several RELAP5 calculations for small break loss-of-coolant accidents (LOCAs) have been performed faster than real time and the development of RELAP5 to operate as a real time simulator is being undertaken. A simulation tool such as this can contribute significantly to reactor safety as a training aid, as a licensing support tool, and eventually as a faster-than-real-time predictor, which can assist in operational decisions to mitigate accident consequences.

The RELAP5 code has been used extensively this past year in cooperation with the LOFT and Semiscale experimental programs for pre and posttest predictions, posttest data analysis, and pretest planning. This cooperative effort between the code developers and the experimentors has been a very beneficial liaison in which improved code models have evolved, and greater insight has been gained about the physical processes of the experiments. Improved understanding of the relative importance of the various physical phenomena present in LWR systems and the experiments has thus been established.

The RELAP5 hydrodynamic model is a one-dimensional, nonequilibrium, nonhomogeneous, two-fluid model of the two-phase flow process. This model includes LWR component models and is integrated into a user convenient modular code framework suitable for future growth of the code. This past year several new modeling capabilities have been added to the code. These capabilities include: a polytropic accumulator model, a noncondensible gas field, point neutronics model, horizontal as well as vertical component flow regime maps, stratified flow and break flow models, a general control valve, feedback control modeling components, a steam separator, and a generalized restart feature for initialization. In addition to the new models, several improvements have been made to the code for faster running and increased user convenience. These improvements include: a more general nonequilibrium subcooled critical flow model, semi-implicit coupling of the critical flow model, smoothed interphase drag constitutive relations, general input for the internal plotting feature and revised major and diagnostic edits.

The new models and improvements are included in the RELAP5/MOD1 version of the code which was completed August 31, 1980. A comprehensive set of checkout problems have been run with the new code and it will be released to the National Energy Software Center during November 1980. A code workshop will be held for NRC contractor personnel the second week of November. This initial workshop will be followed by an open workshop for other RELAP5 users. As a further service, the newsletter "RELAP5 NEWS" is published periodically to provide code update information and to pass on general interest information. The first issue of "RELAP5 NEWS" was published in August of 1980.

Future development efforts will be directed to completion of an automatic steady state initialization feature, extension of the control system simulation model, extension of the thermal-hydraulic and thermal models to all reflood conditions, and exploratory development on several ideas for even faster running capability. The ultimate objective of the RELAP5 code is to obtain an advanced, one-dimensional LWR system simulation code which can execute in real or less than real time for most transients.

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2. H. H. Kuo, V. H. Ransom, and D. M. Snider, "Calculated Thermal Hydraulic Response for Semiscale Mod-3 Test S-07-6 Using RELAP5--A New LWR Sykstem Analysis Code," ASME Heat Transfer Division Nuclear Reactor Thermal-Hydraulic 1980 Topical Meeting, Saratoga, New York.

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4. J. F. Jackson, V. H. Ransom, L. J. Ybarrondo, D. R. Liles, "Light-Water Reactor Safety Analysis Codes", Nuclear Reactor Safety Heat Transfer 1980 Summer School and International Seminar August 25-September 5, 1980, Dubrovnik, Yugoslavia (to be published in the Proceedings of the International Centre for Heat and Mass Transfer).

RELAP5 - AN ADVANCED FAST-RUNNING LWR TRANSIENT ANALYSIS CODE

by
V.H. RANSOM

EG&G Idaho, Inc.

RELAP5

- RELAP5/MOD1
- SMALL BREAK MODELING
- APPLICATIONS

RELAP5

- MAJOR FEATURES

  1-D NONHOMOGENEOUS - NONEQUILIBRIUM
  FLOW REGIME DEPENDENT CONSTITUTIVE
  RELATIONS
  CRITICAL FLOW MODEL
  ABRUPT AREA CHANGE MODEL
  EXTENSIVE INPUT DIAGNOSTICS
  MODULAR STRUCTURE
  NO DIALS

- VERSIONS AVAILABLE

  RELAP5/MOD0 5/79

VHR-3

RELAP5/MOD1 DEVELOPMENT

NEW FEATURES

- POLYTROPIC ACCUMULATOR
- NONCONDENSIBLE MODEL
- POINT KINETICS
- HORIZONTAL AND REVISED VERTICAL FLOW MAPS
- HORIZONTAL STRATIFICATION MODELS
- CONTROL VALVE

VHR-4

RELAP5/MOD1 DEVELOPMENT (CONTINUED)

NEW FEATURES

- FEEDBACK CONTROL SYSTEM
- IDEAL STEAM SEPARATOR
- GENERALIZED RESTART
- EULERIAN BORON TRACKING

VHR-5

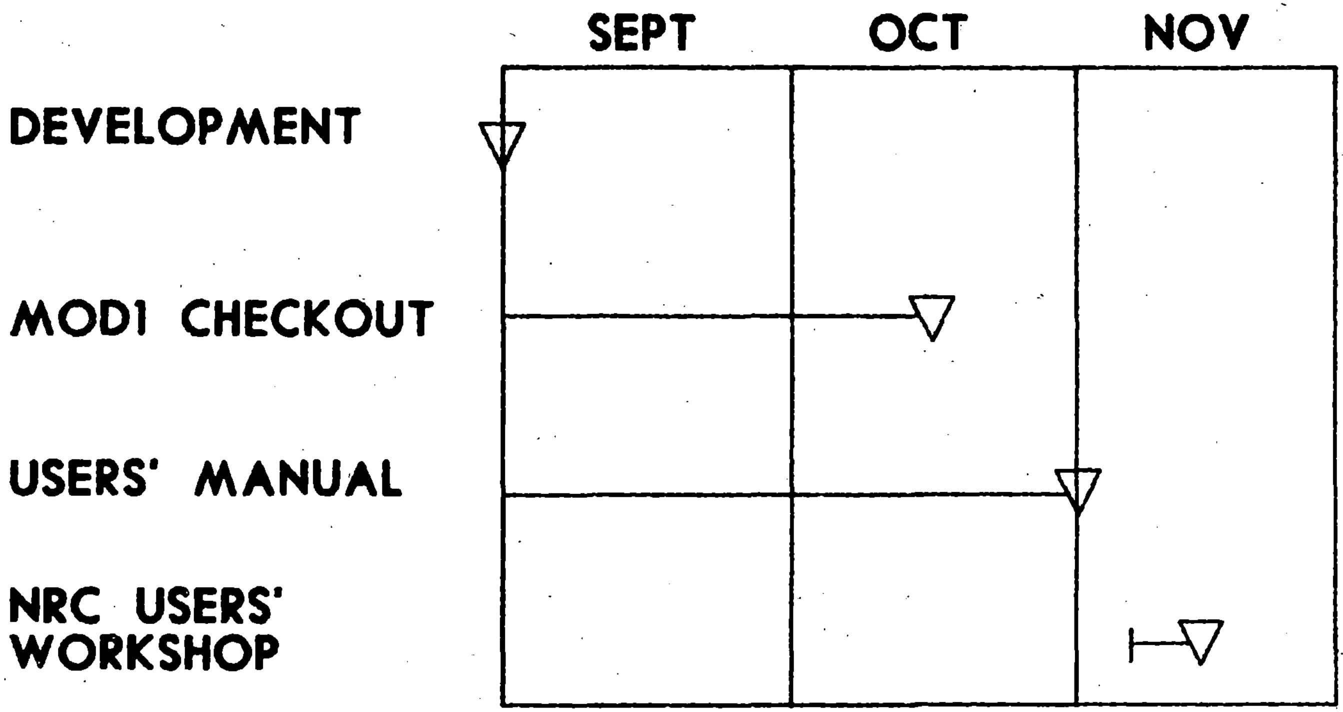
RELAP5/MOD1 DEVELOPMENT (CONTINUED)

IMPROVEMENTS

- NONEQUILIBRIUM SUBCOOLED CRITICAL FLOW
- SEMI-IMPLICIT CRITICAL FLOW
- INTERNAL PLOTTING ROUTINE
- CODE EDIT IMPROVEMENTS

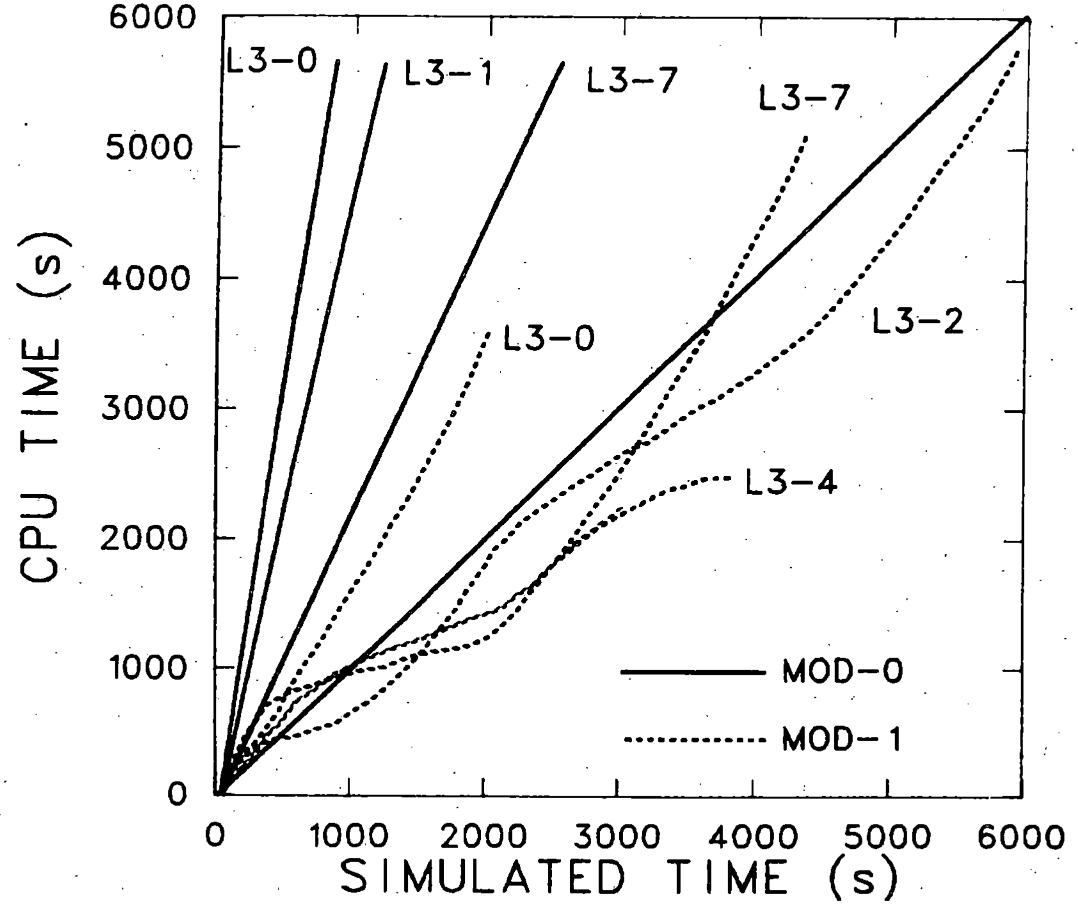
RELAP5/MOD1

- BASED ON RELAP5/MOD0 (RELEASED 5/17/79)
- DEVELOPMENT SCHEDULE



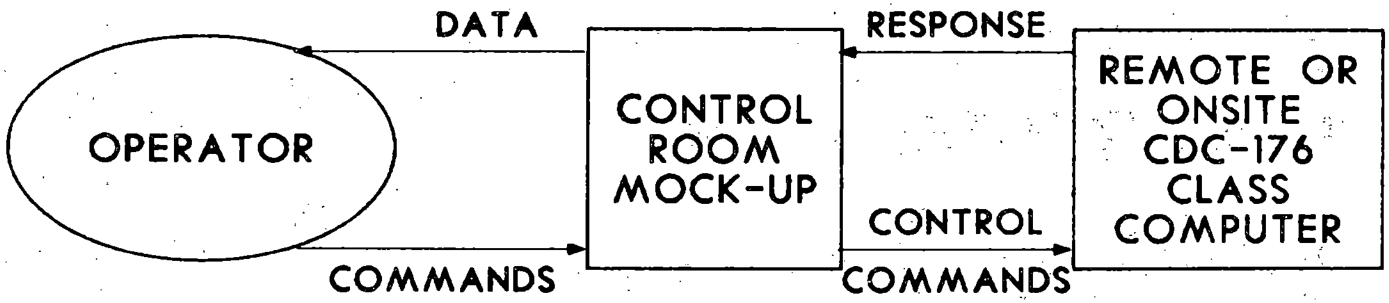
VHR-7

RELAP5 RUNNING TIMES



VHR-8

RELAP5 SIMULATOR



VHR-9

RELAP5 SMALL BREAK MODELS

- HORIZONTAL FLOW
- BREAK FLOW
- HEAT TRANSFER

VHR-10

RELAP5 STRATIFIED FLOW MODEL

- HORIZONTAL AND VERTICAL FLOW REGIME MAPS
- STRATIFIED REGION IN HORIZONTAL MAP
- HYDROSTATIC EFFECT ON AREA AVERAGE PRESSURE

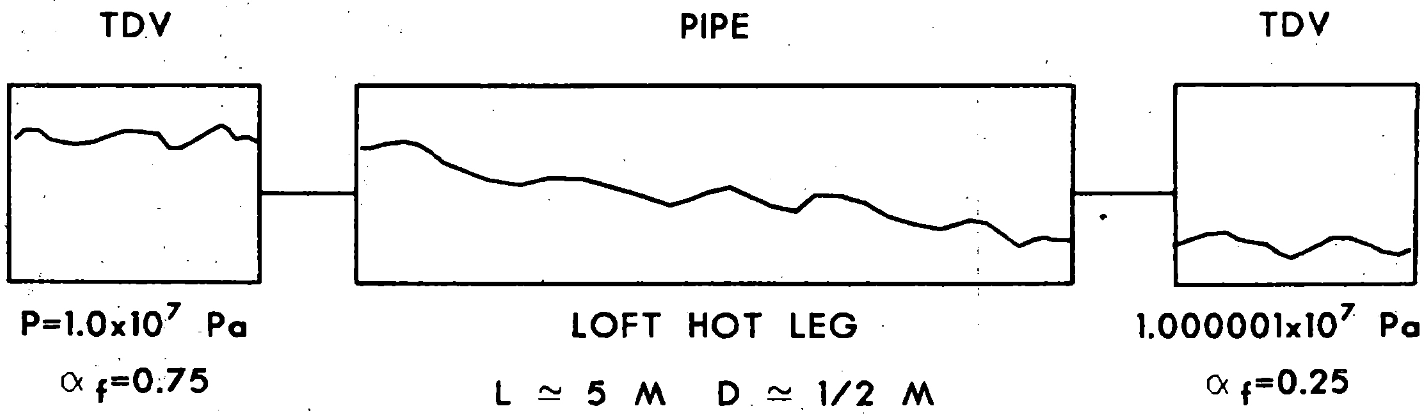
  MIXTURE EQUATION NOTHING

  DIFFERENCE OF PHASIC MOMENTUMS WHEN STRATIFIED

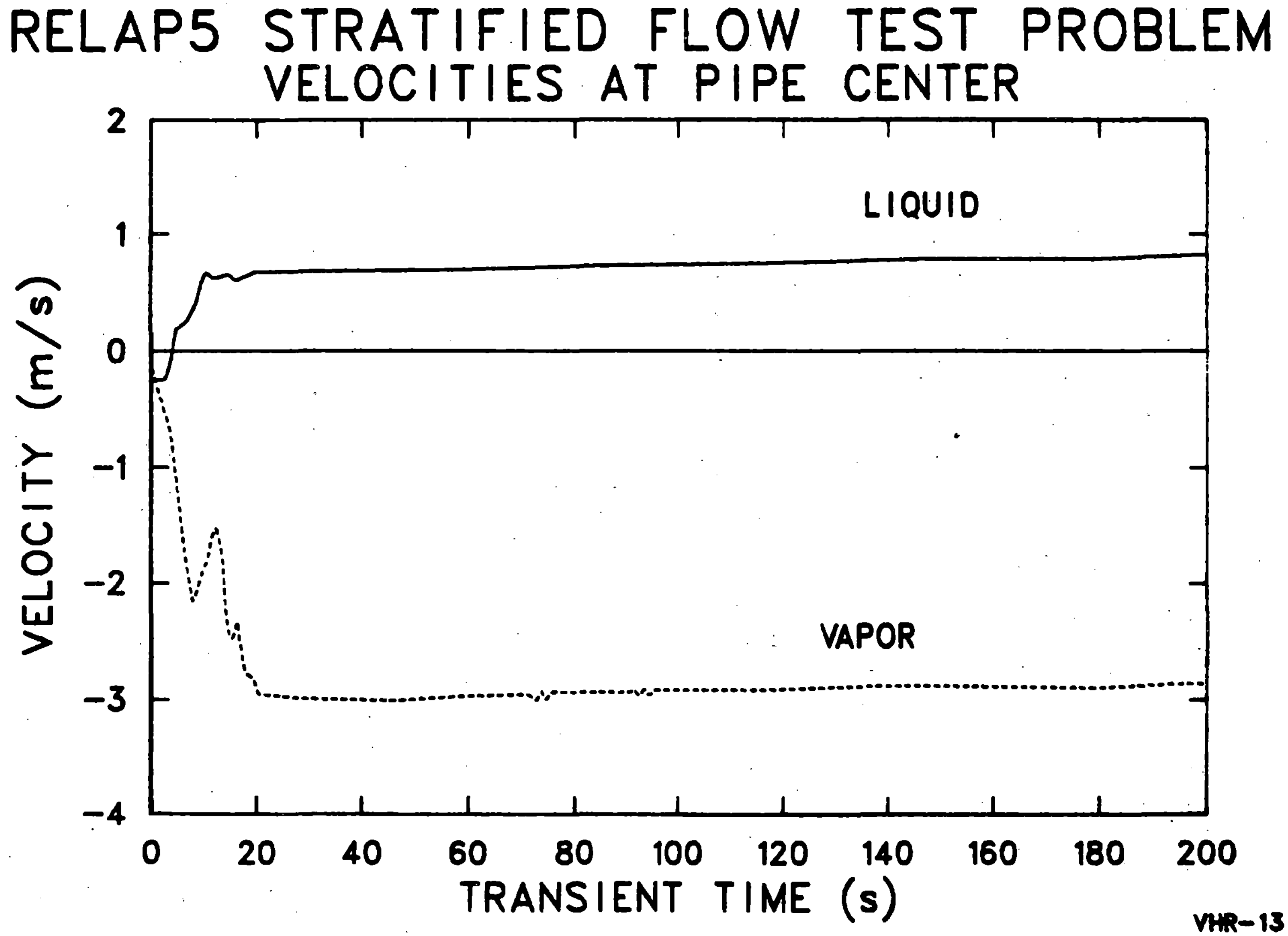
$$-Dg\ (\rho_g - \rho_f) \frac{\bar{\rho}}{\rho_g \rho_f}\ \Delta\ \alpha_g\ \Delta t$$

VHR-11

RELAP5 STRATIFIED FLOW TEST PROBLEM



VHR-12


RELAP5 STRATIFIED FLOW TEST PROBLEM
VELOCITIES AT PIPE CENTER
LIQUID
VAPOR
VELOCITY (m/s)
2
1
0
-1
-2
-3
-4
0
20
40
60
80
100
120
140
160
180
200
TRANSIENT TIME (s)
VHR-13

RELAP5 GENERAL CRITICAL FLOW MODEL

- ALAMGIR-LIENHARD NUCLEATION MODEL (SUBCOOLED)

$$P_{sat} - P_n = 0.258 \frac{\sigma^{3/2} \, T_r^{13.76} \sqrt{1 + 13.25 \, \Sigma^{0.8}}}{\sqrt{kT_c} \, (1 - v_f/v_g)}$$

- LINEAR IMPLICIT COUPLING

  SUBCOOLED AND TWO-PHASE

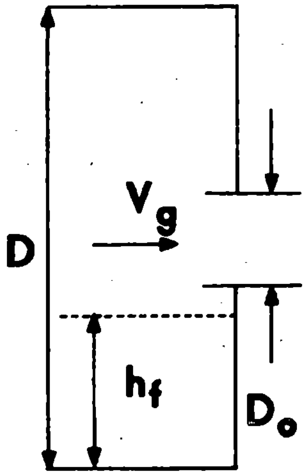
  ELIMINATES NEED FOR UNDER-RELAXATION

  RUNS FASTER

VHR-14

RELAP5 STRATIFIED BREAK MODEL

STRATIFIED IF: $V_g \leq V_{gL} = \left(\frac{(\rho_f - \rho_g) g \alpha_g A}{\rho_g D \sin\theta}\right)^{1/2} \left(1 - \frac{h_f}{D}\right)$



I. $h_f > (D+D_o)/2;\ \alpha_g(\text{out}) = \alpha_g (V_g/V_{gL})^{1/2}$

II. $h_f < (D-D_o)/2;\ \alpha_f(\text{out}) = \alpha_f (V_g/V_{gL})^{1/2}$

III. $(D+D_o)/2 \geq h_f \geq (D-D_o)/2;\ \alpha_g(\text{out}) =$ LINEAR INTERPOLATION FROM I TO II.

VHR-15

RELAP5 SMALL BREAK HEAT TRANSFER

- CONDENSATION

  HORIZONTAL:
  CHATO - FILM CONDENSATION

  VERTICAL:
  COLLIER - MOD. NUSSELT THEORY

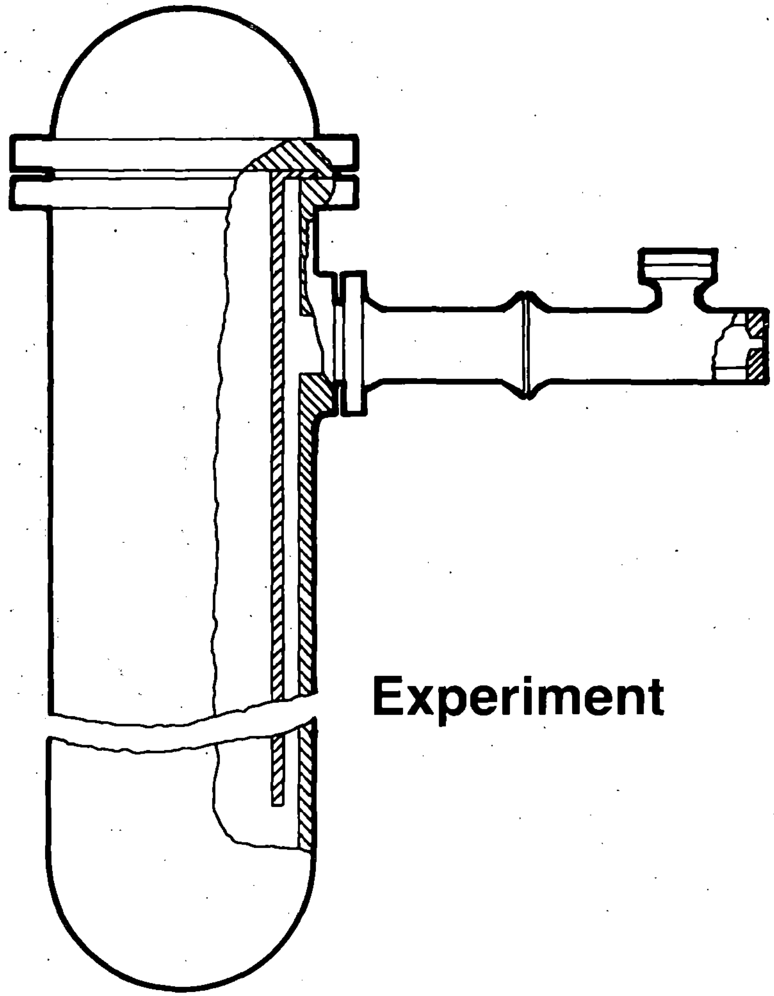
- NATURAL CONVECTION

  COOLING:
  $$Nu = MAX\left[0.555(GrPr)^{1/4}, 0.130(GrPr)^{1/3}\right]$$

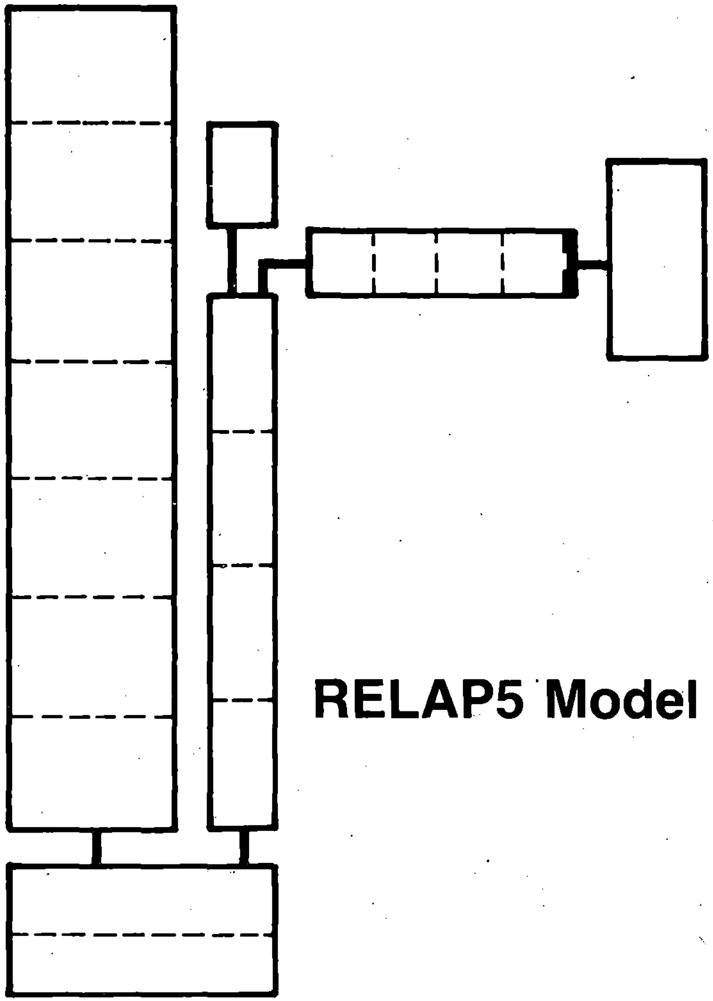
  HEATING:
  MAX [POOR BOILING, COOLING]

VHR-16

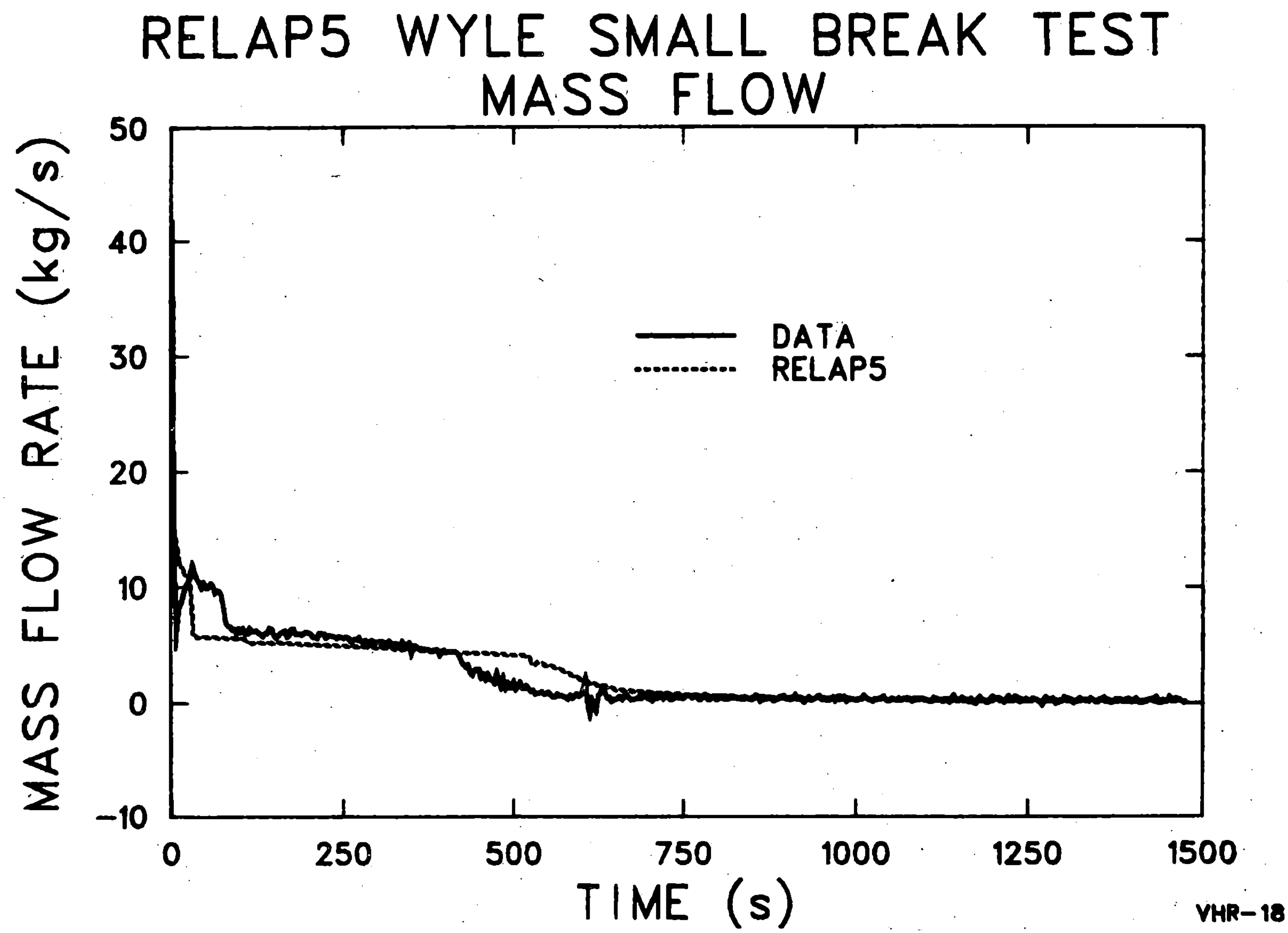
Wyle Small Break Test



Experiment

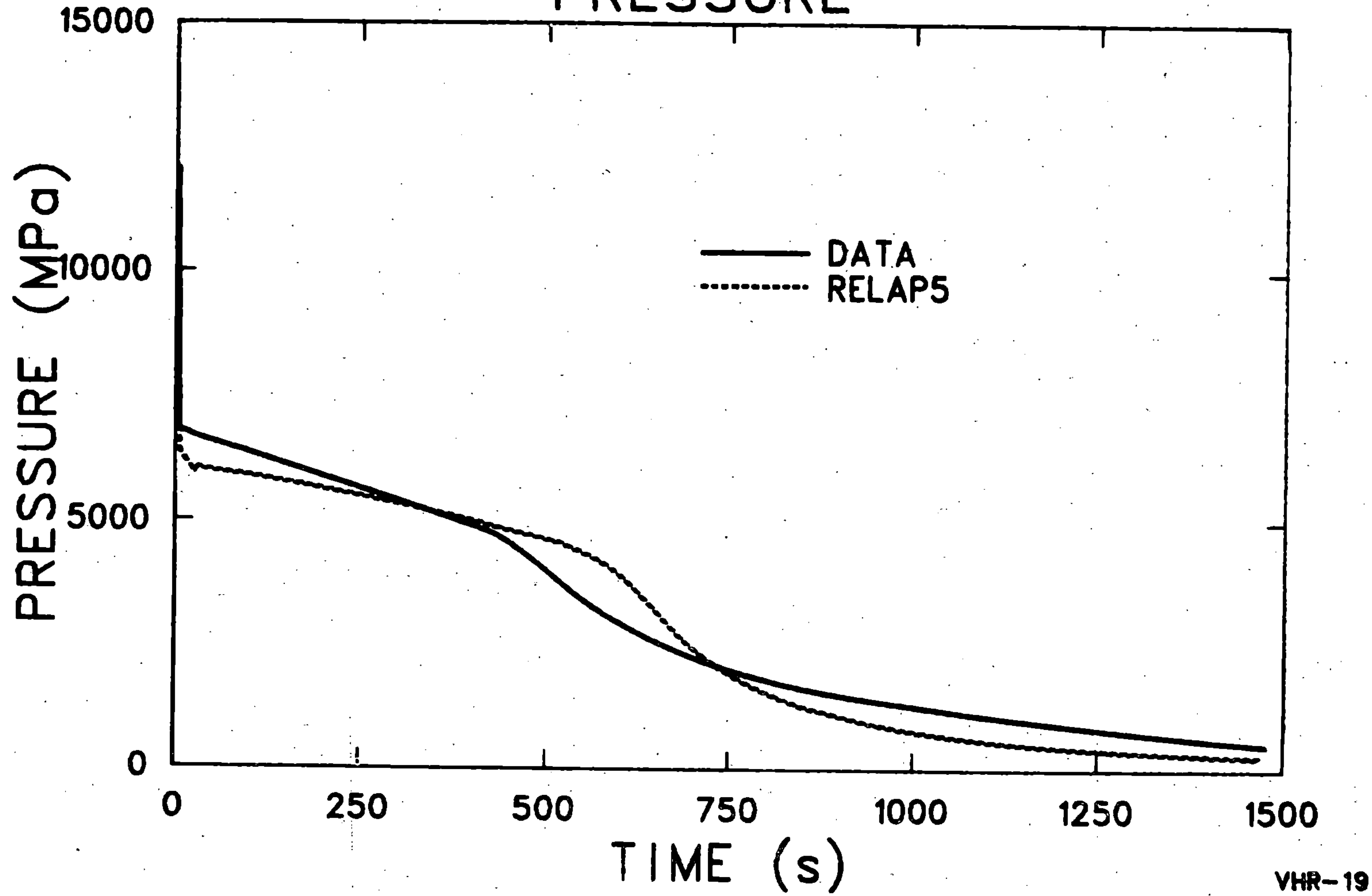


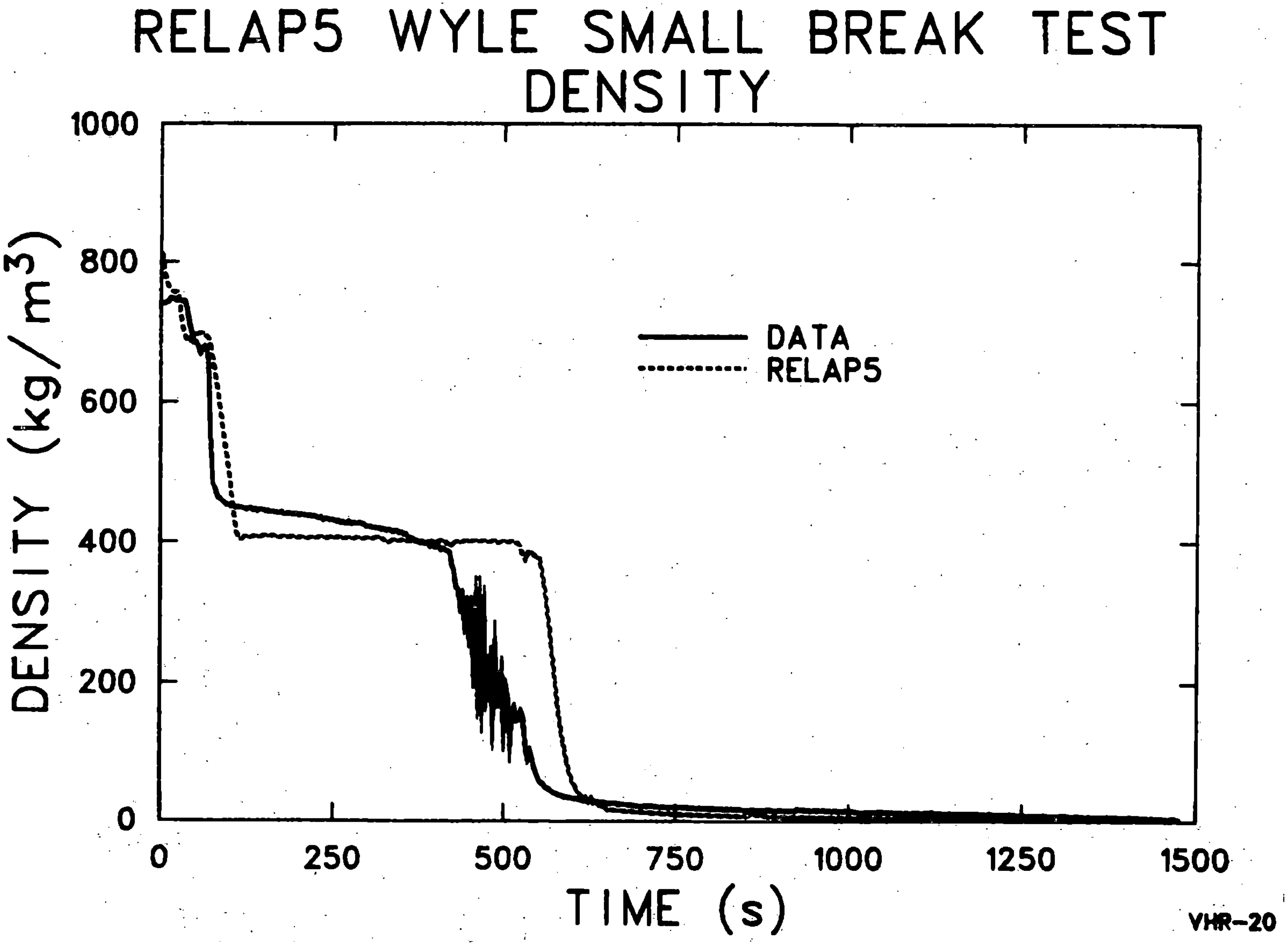
RELAP5 Model



RELAP5 WYLE SMALL BREAK TEST
MASS FLOW
MASS FLOW RATE (kg/s)
50
40
30
20
10
0
−10
0
250
500
750
1000
1250
1500
TIME (s)
DATA
RELAP5
VHR−18

RELAP5 WYLE SMALL BREAK TEST PRESSURE




RELAP5 WYLE SMALL BREAK TEST
DENSITY
DENSITY (kg/m³)
1000
800
600
400
200
0
DATA
RELAP5
0
250
500
750
1000
1250
1500
TIME (s)
VHR-20

RELAP5/MOD1 APPLICATIONS

- SEMISCALE

    TEST S-SB-P1

- LOFT

    TEST L3-0

    TEST L3-7

VHR-21

RELAP5 SEMISCALE MOD3 MODEL

- FEATURES

  TWO ACTIVE STEAM GENERATORS
  UPPER HEAD BYPASS LINE
  POLYTROPIC ACCUMULATOR
  HEAT LOSS TO THE ENVIRONMENT
  SINGLE CHANNEL CORE
  (HIGH AND LOW POWER RODS)

- DISCRETIZATION

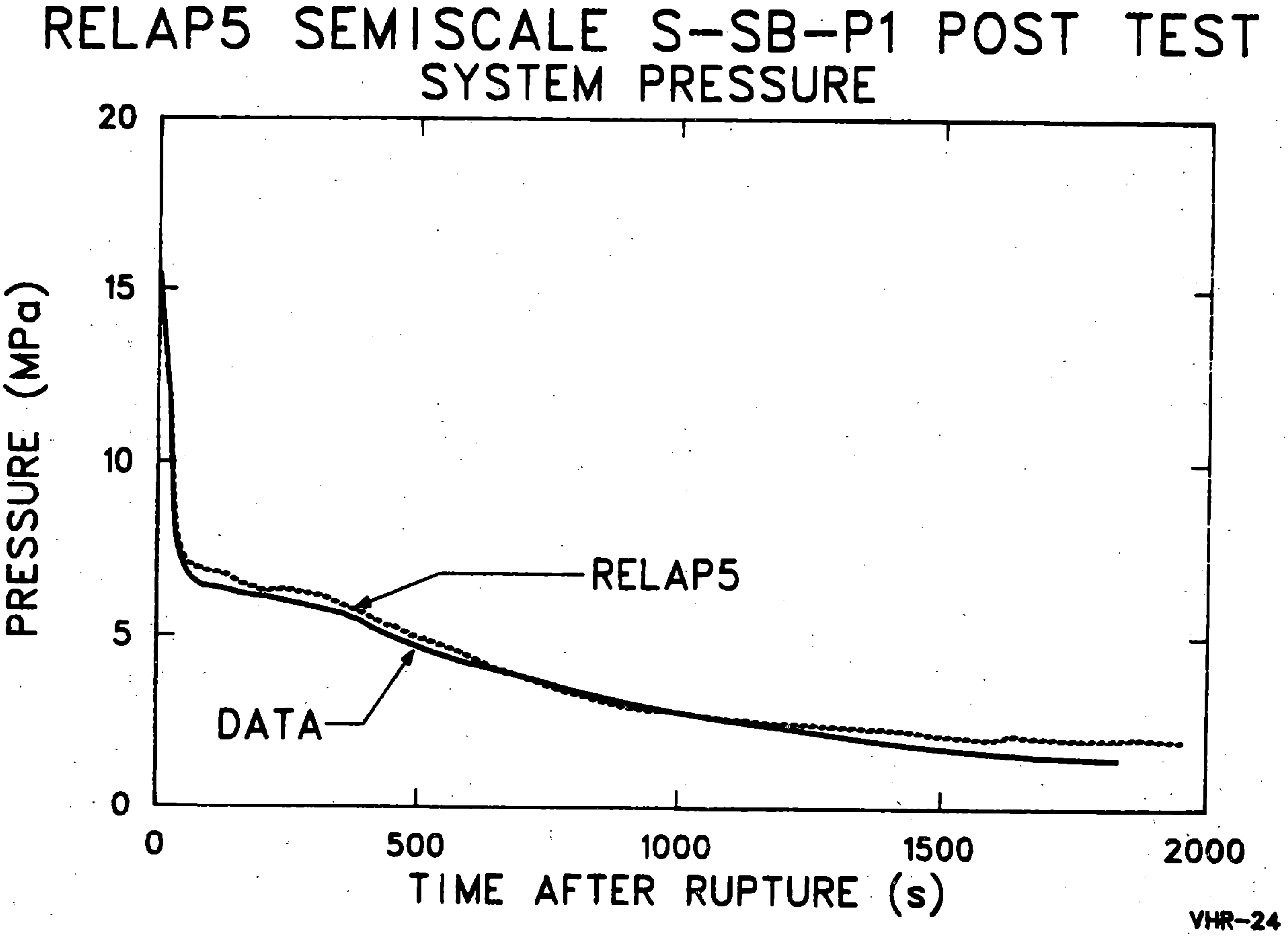
  132 CONTROL VOLUMES, 142 JUNCTIONS
  96 HEAT STRUCTURES

VHR-22

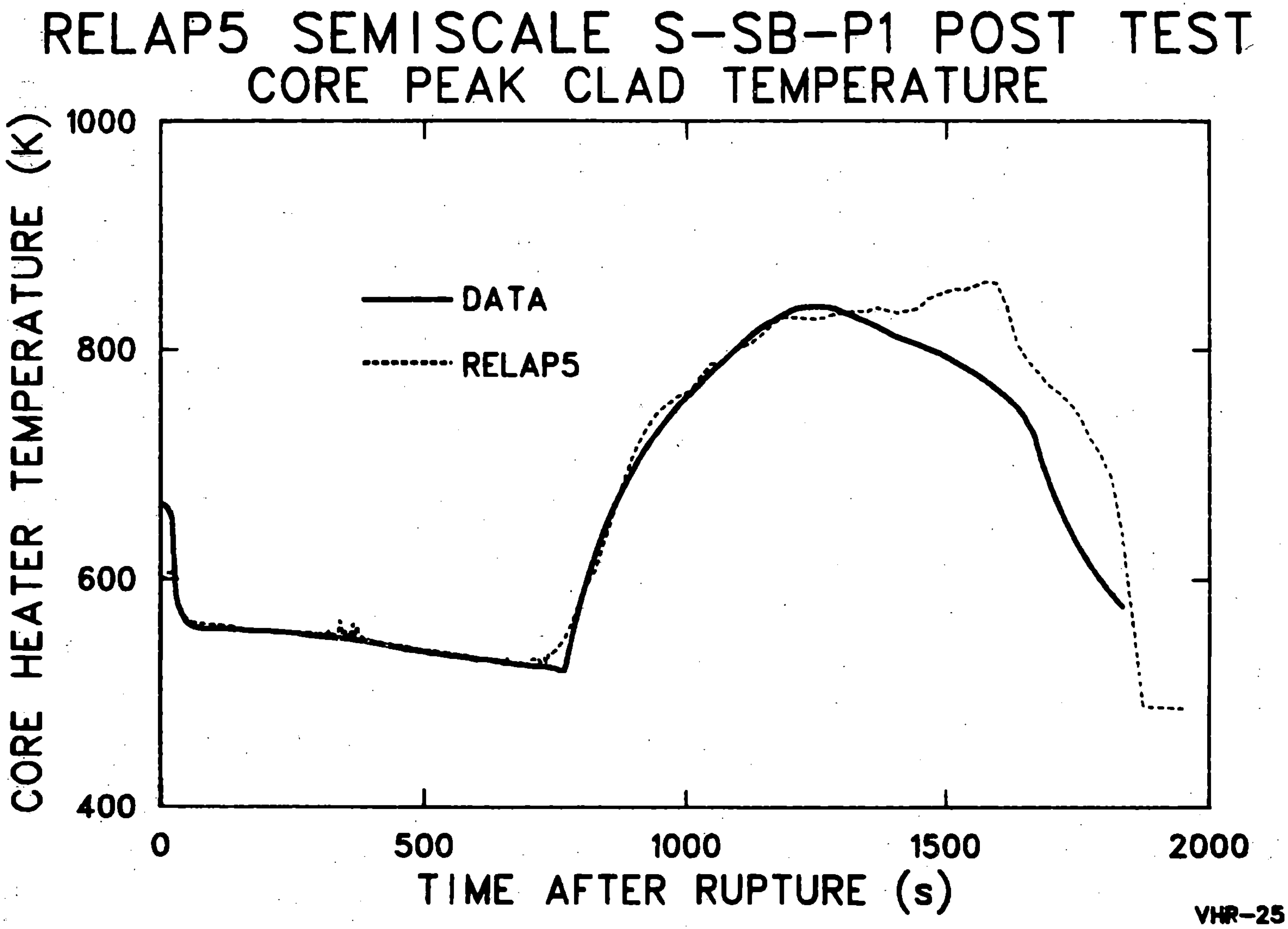
SEMISCALE TEST S-SB-P1

PRESSURE	15.51 MPa
CORE POWER	2.1 MW
BREAK	
SIZE	2.5%
TYPE	COMMUNICATIVE
LOCATION	COLD LEG
ECC INJECTION (HPIS)	
INTACT LOOP	28.4 s AFTER PRESSURE REACHES 12.58 MPa
BROKEN LOOP	SAME
PUMP COASTDOWN	
STEAM VALVE ISOLATION	3.9 s AFTER PRESSURE REACHES 12.58 MPa

VHR-23



RELAP5 SEMISCALE S-SB-P1 POST TEST
SYSTEM PRESSURE
PRESSURE (MPa)
20
15
10
5
0
500
1000
1500
2000
TIME AFTER RUPTURE (s)
RELAP5
DATA
VHR-24



RELAP5 SEMISCALE S-SB-P1 POST TEST
CORE PEAK CLAD TEMPERATURE
CORE HEATER TEMPERATURE (K)
1000
800
600
400
DATA
RELAP5
0
500
1000
1500
2000
TIME AFTER RUPTURE (s)
VHR-25

RELAP5 LOFT STANDARD MODEL

- FEATURES

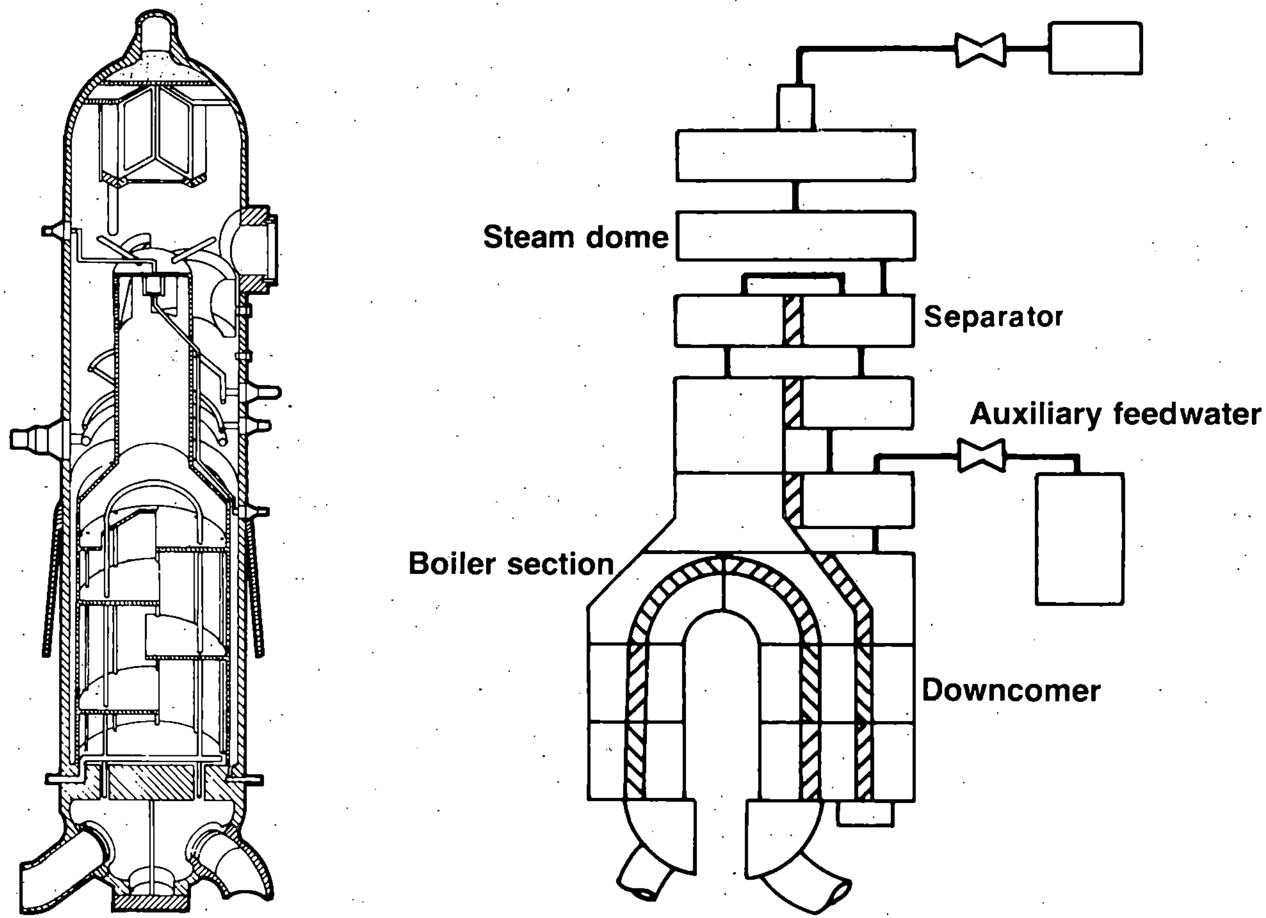
  ACTIVE STEAM GENERATOR
  ANNULAR DOWNCOMER
  UPPER HEAD LEAKAGE
  POLYTROPIC ACCUMULATOR
  HEAT LOSS TO CONTAINMENT

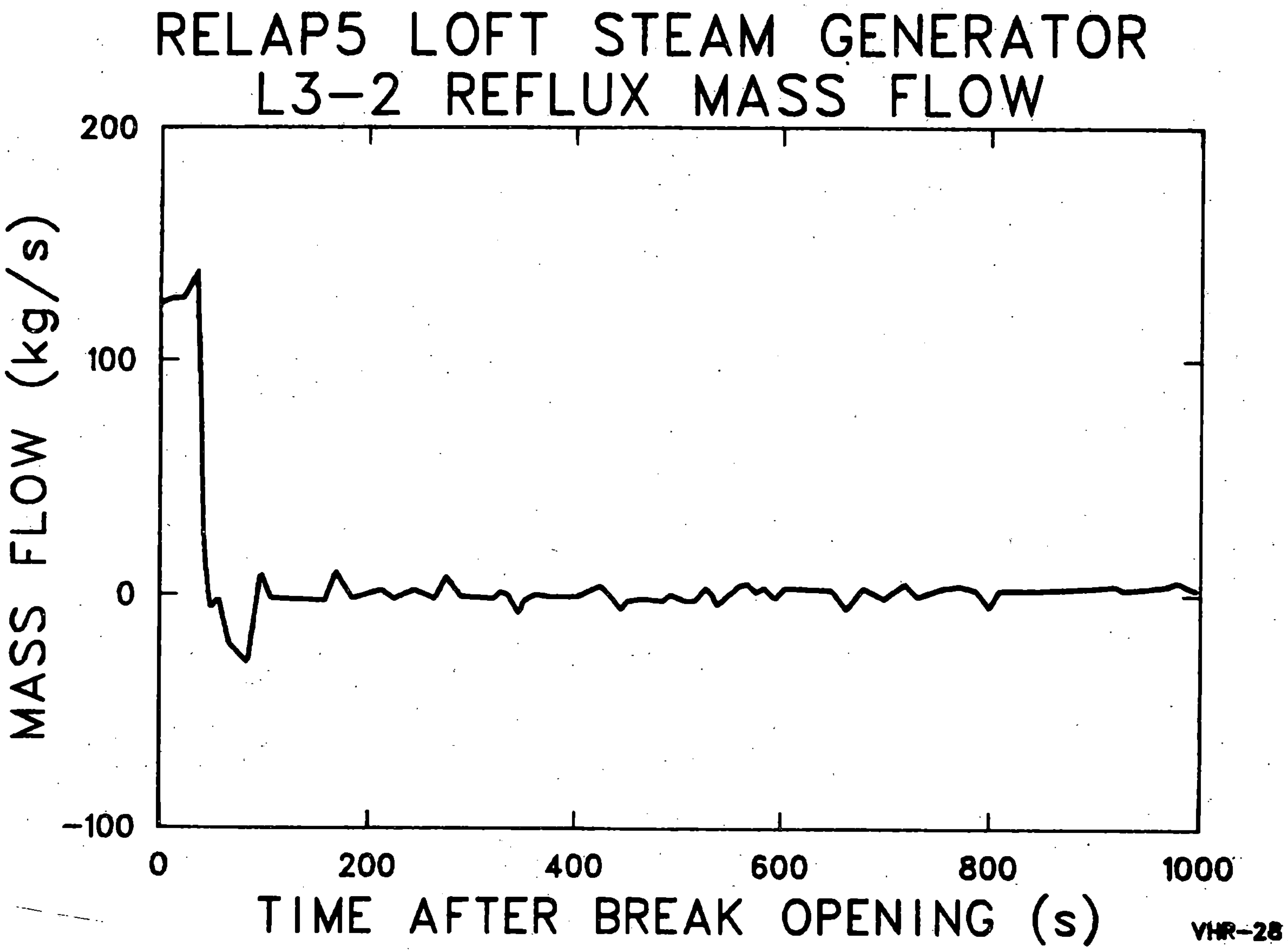
- DISCRETIZATION

  113 VOLUMES, 122 JUNCTIONS
  43 HEAT STRUCTURES

VHR-26

RELAP LOFT Steam Generator




RELAP5 LOFT STEAM GENERATOR
L3-2 REFLUX MASS FLOW
MASS FLOW (kg/s)
200
100
0
-100
0
200
400
600
800
1000
TIME AFTER BREAK OPENING (s)
VHR-28

LOFT TEST L3-0

NONNUCLEAR EXPERIMENT

PRESSURIZER BREAK (PORV OPENED)

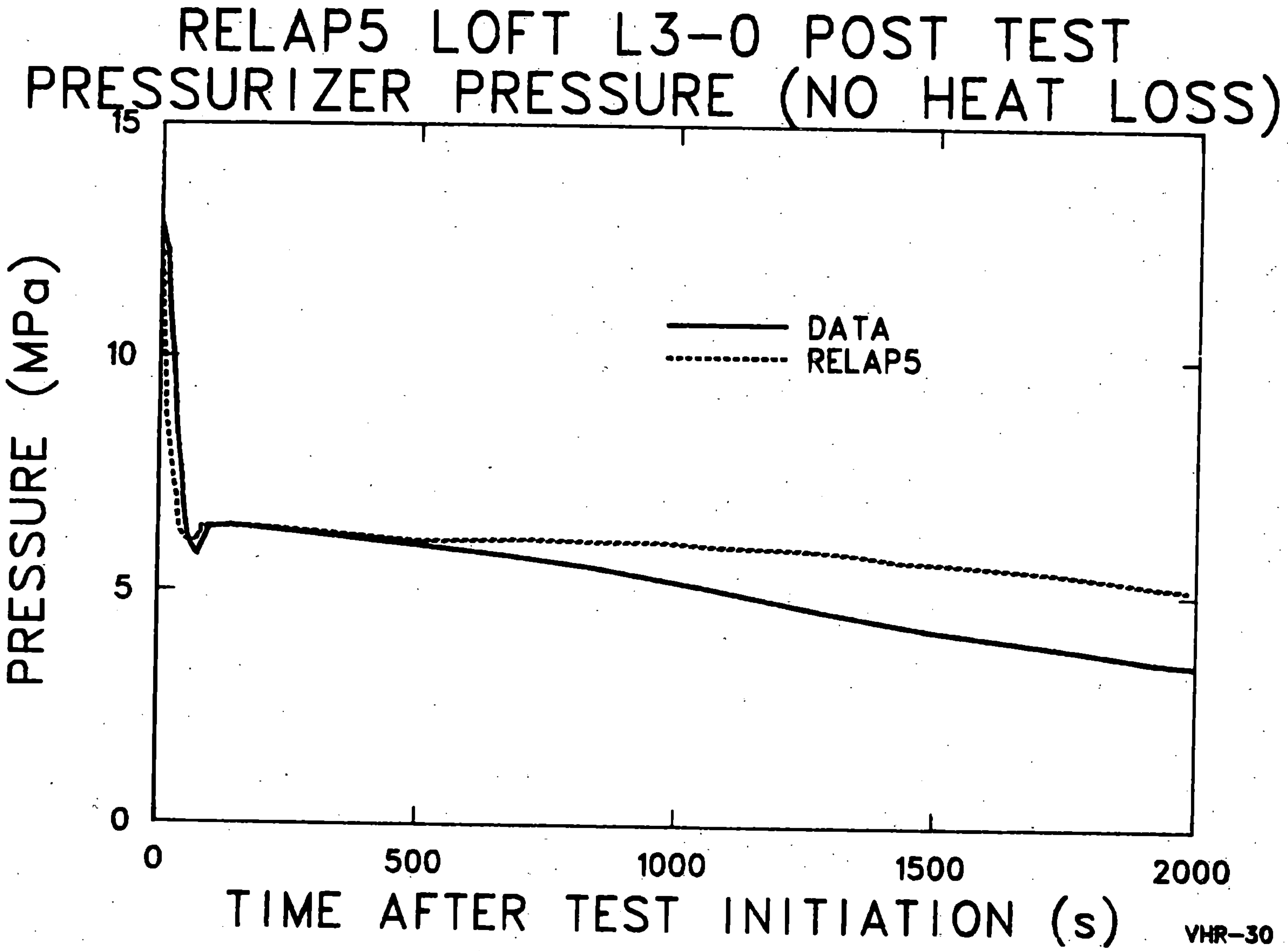
PRESSURE 14.74 MPa

TEMPERATURE 557.0 K (HOT LEG)

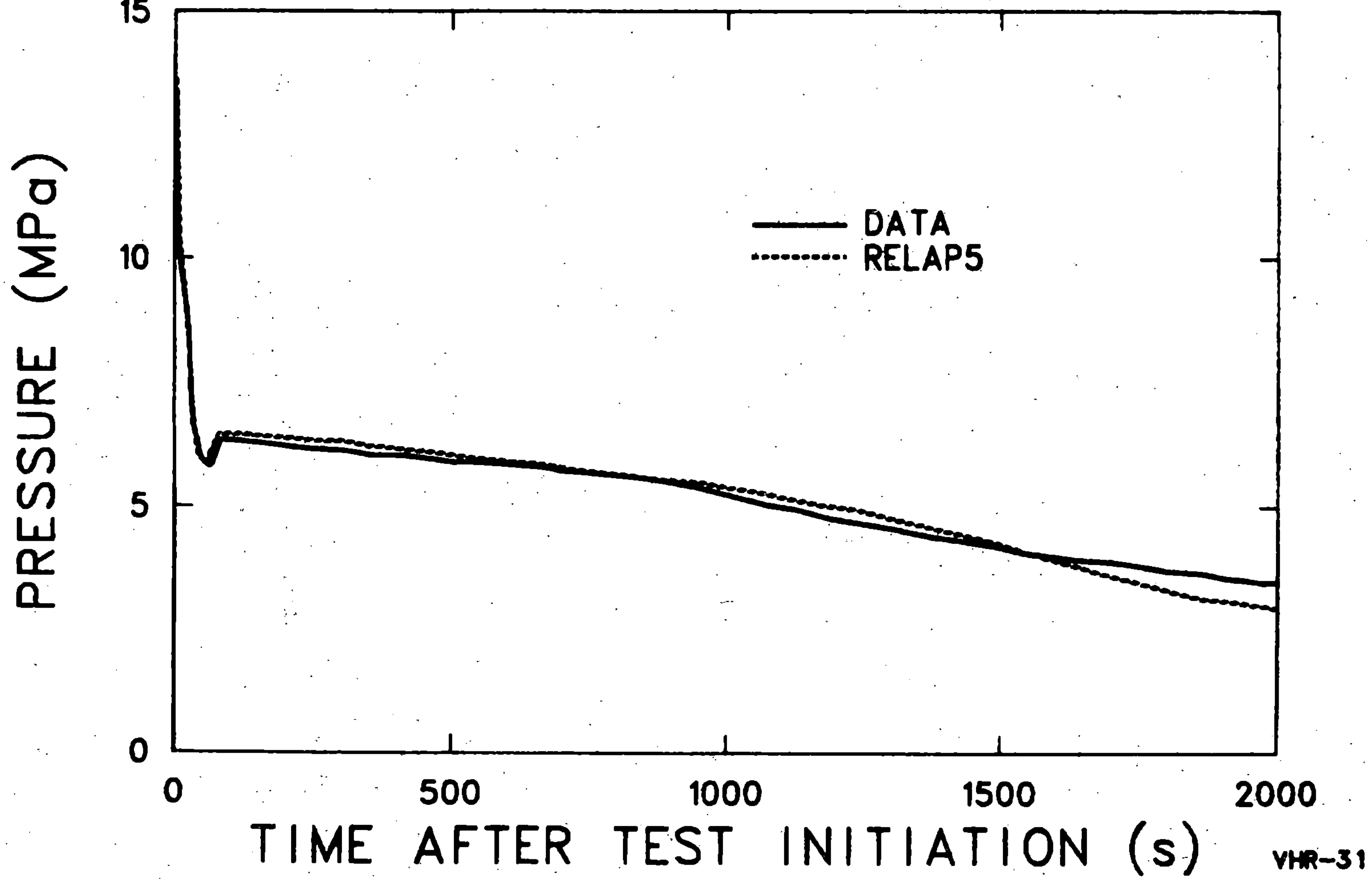
PUMPS TRIPPED AND NO ECC

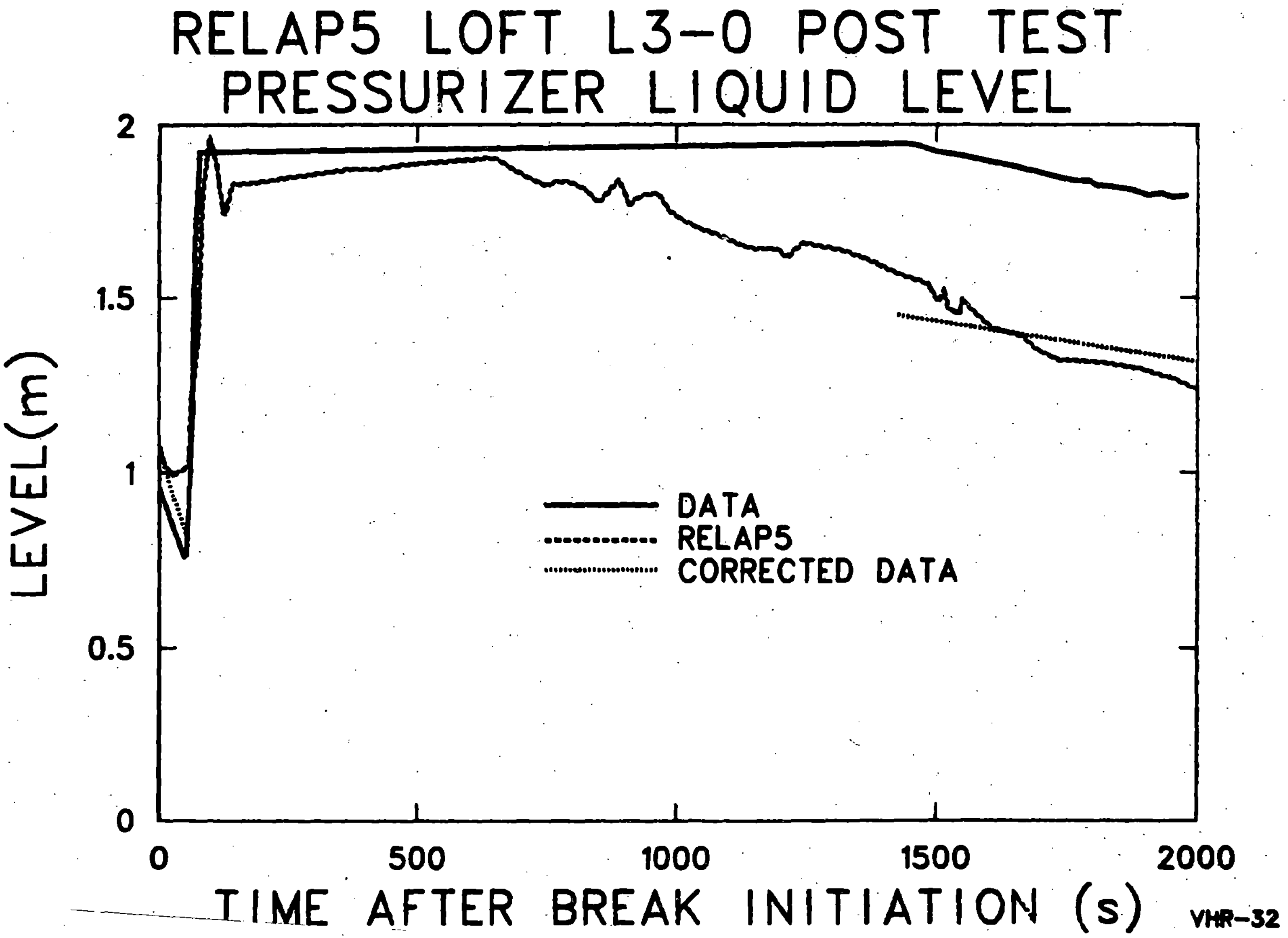
SYSTEM RECOVERY AT 2460 s

VHR-29


RELAP5 LOFT L3-0 POST TEST
PRESSURIZER PRESSURE (NO HEAT LOSS)
PRESSURE (MPa)
15
10
5
0
DATA
RELAP5
0
500
1000
1500
2000
TIME AFTER TEST INITIATION (s)
VHR-30

RELAP5 LOFT L3-0 POST TEST
PRESSURIZER PRESSURE





RELAP5 LOFT L3-0 POST TEST
PRESSURIZER LIQUID LEVEL
LEVEL(m)
2
1.5
1
0.5
0
0
500
1000
1500
2000
TIME AFTER BREAK INITIATION (s)
DATA
RELAP5
CORRECTED DATA
VHR-32

LOFT TEST L3-7

NUCLEAR FULL POWER EXPERIMENT (50 MW)

SMALL COLD LEG BREAK 0.16%

PRESSURE 14.95 MPa

TEMPERATURE 576.1 K (HOT LEG)

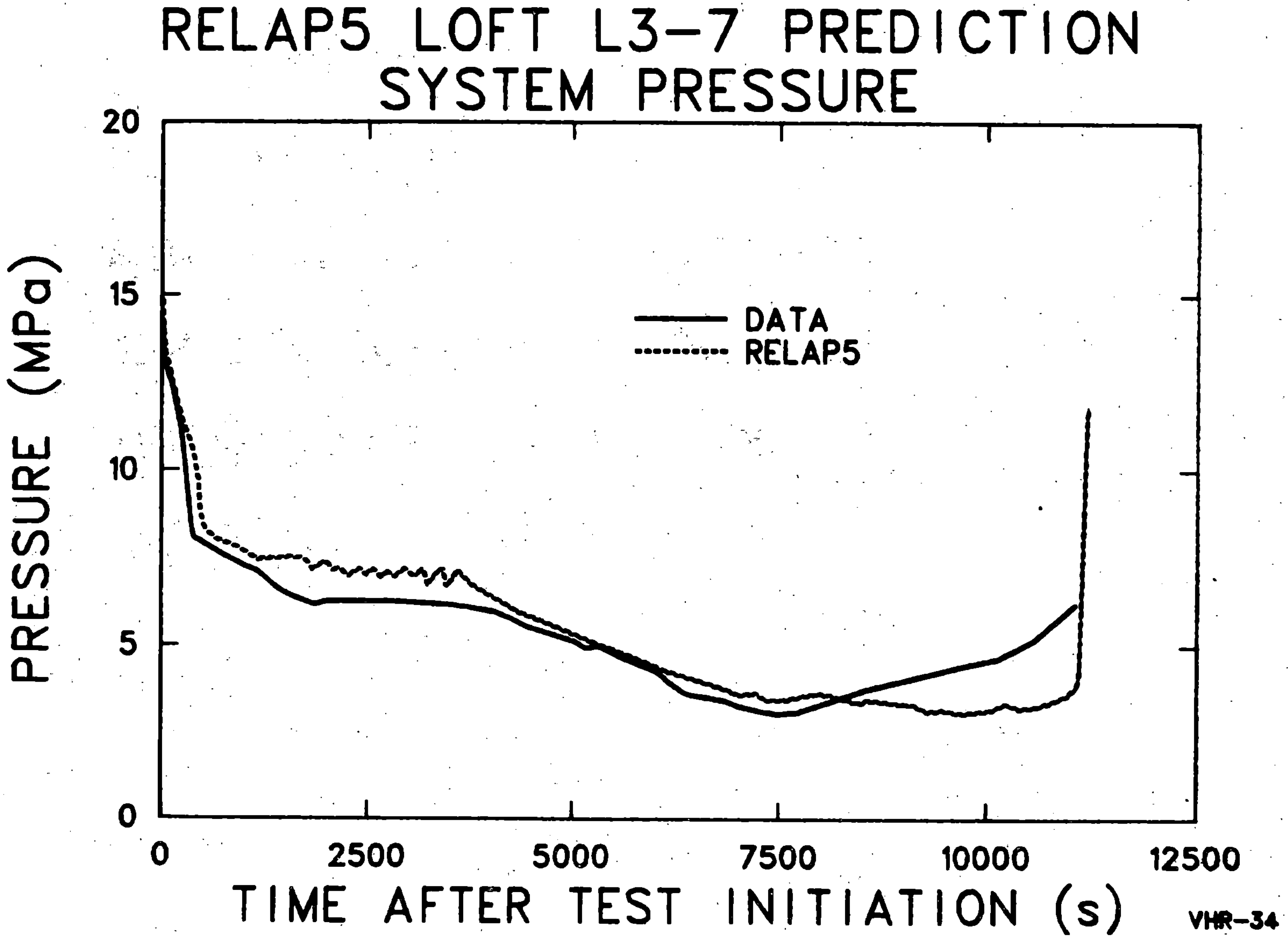
REACTOR SCRAM ON LOW PRESSURE 36 s

PUMPS TRIPPED 10 s AFTER SCRAM

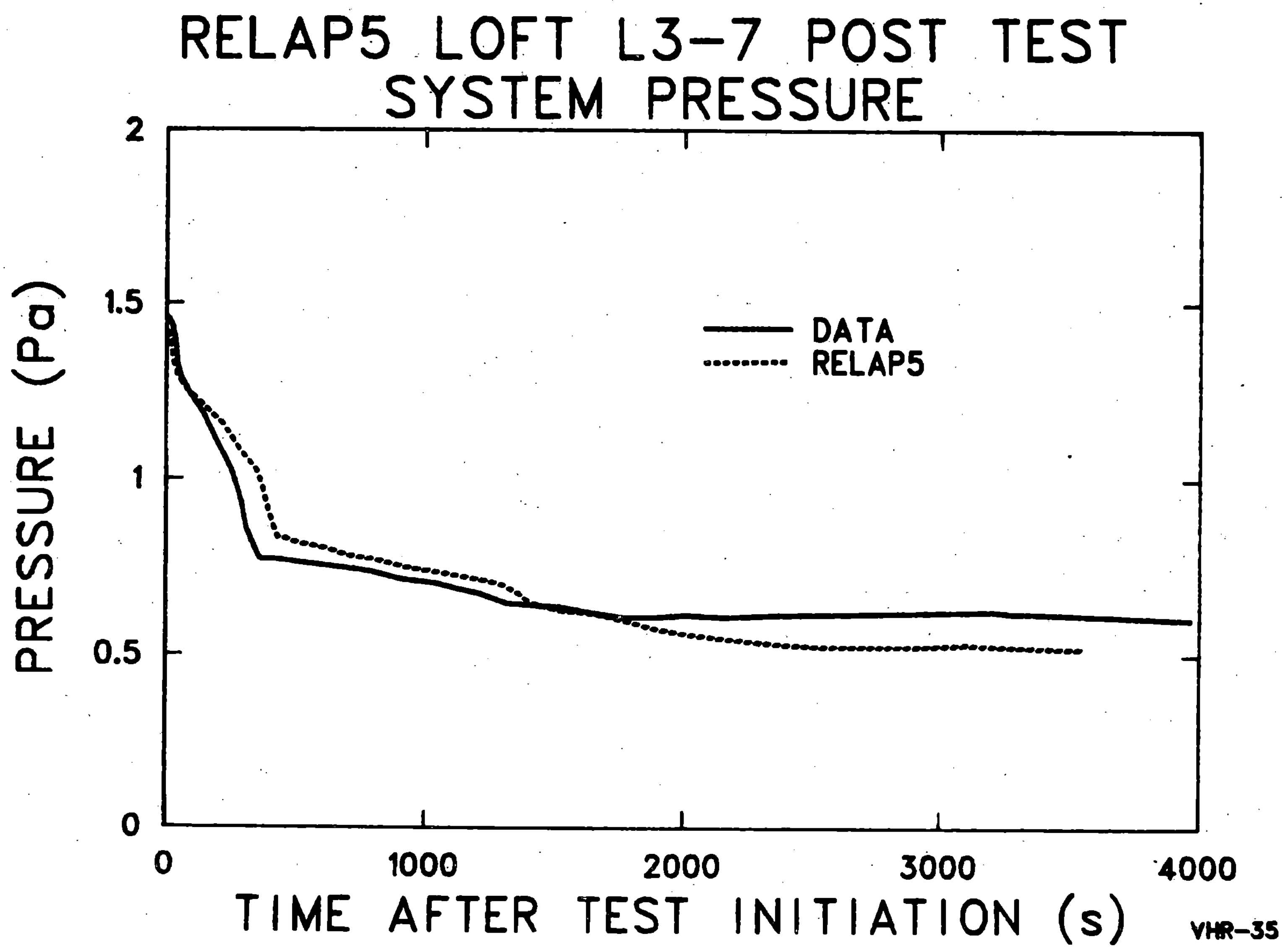
HPIS ON AT 13.16 MPa

BREAK CLOSED AT 3600 s

VHR-33



RELAP5 LOFT L3-7 PREDICTION
SYSTEM PRESSURE
PRESSURE (MPa)
TIME AFTER TEST INITIATION (s)
DATA
RELAP5
0
5
10
15
20
0
2500
5000
7500
10000
12500
VHR-34



RELAP5 LOFT L3-7 POST TEST
SYSTEM PRESSURE
PRESSURE (Pa)
2
1.5
1
0.5
0
0
1000
2000
3000
4000
TIME AFTER TEST INITIATION (s)
DATA
RELAP5
VHR-35

SUMMARY

- RELAP5 - INEL EXPERIMENTS
  - BETTER MODELING
  - INCREASED PHYSICAL INSIGHT
  - RELIABLE PREDICTIONS
- REAL-TIME SIMULATION
  - FEASIBLE FOR MOST TRANSIENTS
  - DEVELOP EXISTING AND NEW IDEAS

LA-UR -80-2658

TITLE: SOME RECENT APPLICATIONS OF THE K-FIX CODE

AUTHOR(S): Bart J. Daly
William C. Rivard
Martin D. Torrey

SUBMITTED TO: Eighth Water Reactor Safety Research Information Mtg.
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University of California

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DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

SOME RECENT APPLICATIONS OF THE K-FIX CODE

Bart J. Daly, William C. Rivard, and Martin D. Torrey

In this paper, we report on two recent applications of the K-FIX[1] code that demonstrate its ability to describe complex flow and fluid-structure interaction in PWR geometries. The first of these is a three-dimensional study of hot-leg ECC injection into the upper plenum of a PWR,[2] and the second is a pretest prediction of the HDR[3] blowdown test V31.1.

Full-scale tests[4] of water injection into an air-filled upper plenum have been made at Kraftwerk Union in Germany. A horizontal cross section of this facility is shown in Fig. 1. Also shown is the K-FIX computing region, which encompasses the five control rod guide tubes closest to the point of ECC injection. Figures 2 and 3 show the steady-state liquid velocity field and void fraction contours from the K-FIX air-water study. These figures show the water splitting around guide tubes 2 and 4 and converging into two liquid sheets that exit on the left side of the computing region. As a result of flow impingement on the lead columns, there has been an appreciable transfer of horizontal to vertical momentum. This is demonstrated in Fig. 4, which shows a comparison of the calculated and experimental measurements of the liquid flow rate distribution at the bottom boundary of the test facility. Downward flow near the columns may be overpredicted in the calculations because of the stair-step column cross section. On the other hand, the downward flow is likely to be underestimated by the experimental measurement technique, which cannot accurately detect film flow down the columns.

Also included in the computational study, although not in the experiments, is the effect of condensation on the flow distribution. The bottom plot in Fig. 4 shows the water flow rate at the bottom when the air is replaced by saturated steam. Notice the enhanced vertical flow that results from condensation. The subcooling of the water that reaches the bottom is approximately 107°C, compared to 139°C at the inlet.

Figures 5-7 show contours of liquid mass flux and comparisons between the calculated and experimental mass flow rates for the open vertical faces in the air-water study. These faces are labeled in the upper plot of Fig. 4. The contour plots of Figs. 5 and 6 indicate that the water flow across these vertical edges is in the form of liquid sheets, as in also observed in the experiments. Also included in Figs. 5-7 are mass flux contours and mass flow rates for the steam-water study. The flow across these vertical sides is greatly reduced compared to the air-water study because of the enhanced vertical flow as shown in Fig. 4. Table I shows a summary of the liquid mass flow rates across the bottom and the open vertical sides for the air-water and steam-water calculations and the experiment.

TABLE I

TOTAL MASS FLOW RATE (kg/s)

	Waterfall Data	Experiment Calculation	Steam-Water Calculation
Side A	40.9	45.8	3.0
Side B	184.2	143.4	52.6
Side C	21.1	11.5	0.0
Bottom	80.1	131.3	330.7

Pretest predictions for transducer responses in the HDR blowdown test V31.1 were furnished in January 1980. The calculations were made with the three-dimensional K-FIX code[1] coupled to the three-dimensional elastic shell code FLX[5] to describe the fluid-structure interaction dynamics of the core barrel. Flow through the discharge pipe was calculated in one dimension and coupled with the K-FIX vessel flow. The break was modeled as an instantaneous rupture that exposed the full pipe cross-sectional area to ambient conditions. Flashing was calculated according to a nonequilibrium rate[6] that has been tested against steady and transient data from large and small scale tests. The calculation used 3774 cells in the vessel, 612 cells in the core barrel, and 16 cells in the discharge pipe. The calculation was carried to 100 ms with a time step of 0.04 ms in 66 minutes on a CDC-7600 computer. The calculation for the elastic motion of the core barrel in FLX and for the flashing flow in the discharge pipe were subcycled at smaller time steps, typically 2-5 times smaller. Implicit and explicit computation in K-FIX yielded essentially identical results.

Comparison of the predictions with data are shown in Figs. 8-10. Core barrel displacement histories for two locations are given in Figs. 8 and 9. Referred to the centerline of the discharge pipe, these locations are 30° around and 107 cm and 330 cm below, respectively. Figure 10 compares the predicted and measured pressure differential across the core barrel at the angular and elevation locations of the discharge pipe centerline. Similar good agreement with measurements of strain, absolute pressure, and flow rate was also achieved.

REFERENCES

1. W. C. Rivard and M. D. Torrey, "K-FIX: A Computer Program for Transient, Two-Dimensional, Two-Fluid Flow," Los Alamos Scientific Laboratory report LA-NUREG-6623, (1979); "THREED: An Extension of the K-FIX Code for Three-Dimensional Calculations," Los Alamos Scientific Laboratory report LA-NUREG-6623, Suppl. II (1979).

2. W. C. Rivard and M. D. Torrey, "The Deflection by Upper Plenum Columns of a Water Jet Issuing from an Outlet Pipe," presented at the 2d/3d Multinational Coordination Meeting, Munich, Germany, July 1980 (LA-UR-80-1941).

3. W. C. Rivard, "Numerical Simulation of Hydroelastic Motion with Application to the Full Scale HDR Tests," Seventh Water Reactor Safety Research Information Meeting, Gaithersburg, MD, November 1979 (LA-UR-79-2651).

4. H. Kiehne, "Luft-Wasser-Versuche in Oberen Plenum," Kraftwerk Union report R-11-1002-79 (1979).

5. J. K. Dienes, C. W. Hirt, W. C. Rivard, L. R. Stein, and M. D. Torrey, "FLX: A Shell Code for Coupled Fluid-Structure Analysis of Core Barrel Dynamics," Los Alamos Scientific Laboratory report LA-7927 (NUREG/CR-0959) (1979).

6. W. C. Rivard and J. R. Travis, "A Nonequilibrium Vapor Production Model for Critical Flow," Nucl. Sci. and Engr. 74, 40 (1980).

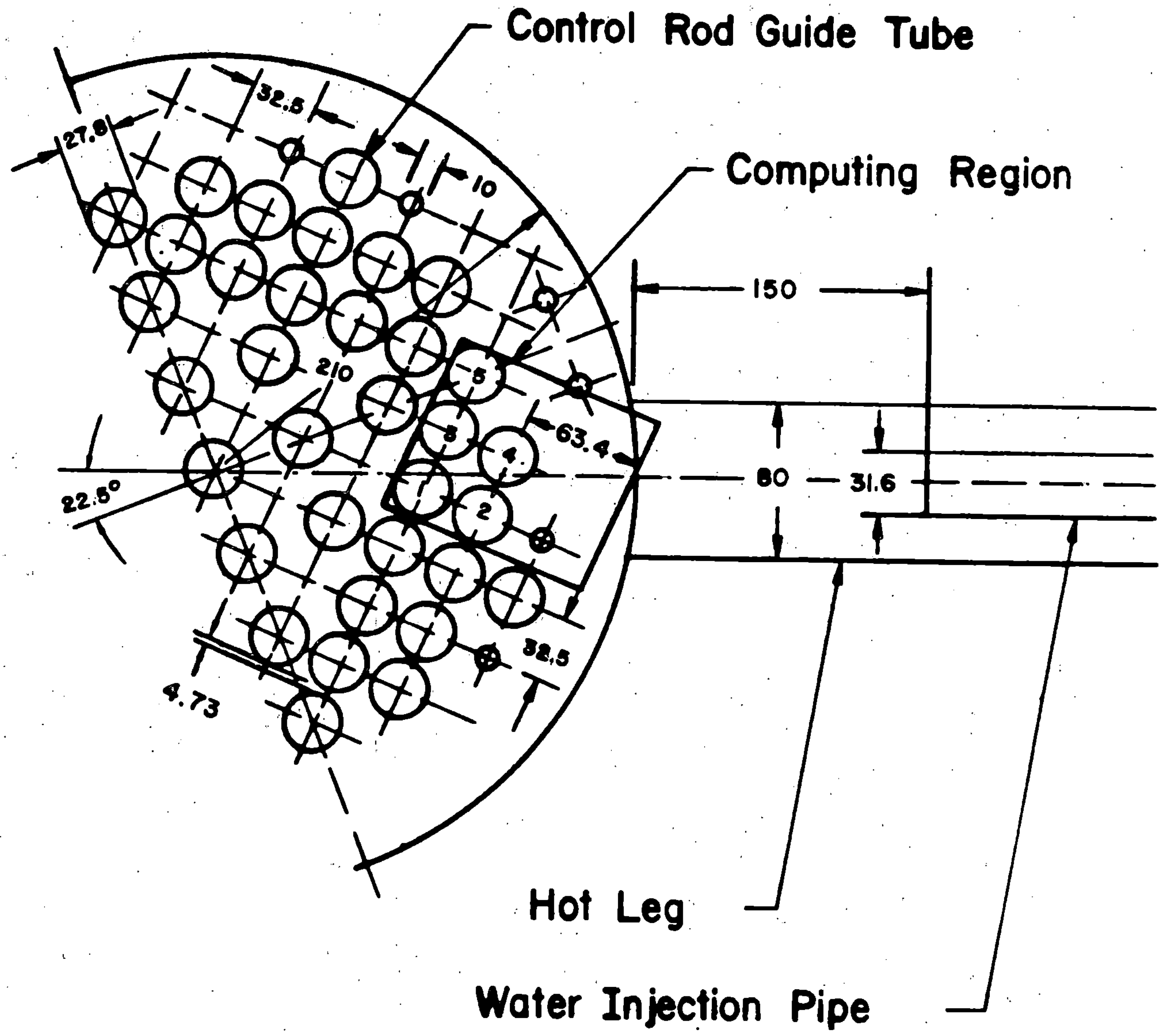


Fig. 1. Horizontal section of the air-water test facility and the computing region boundaries. Dimensions are in cm.

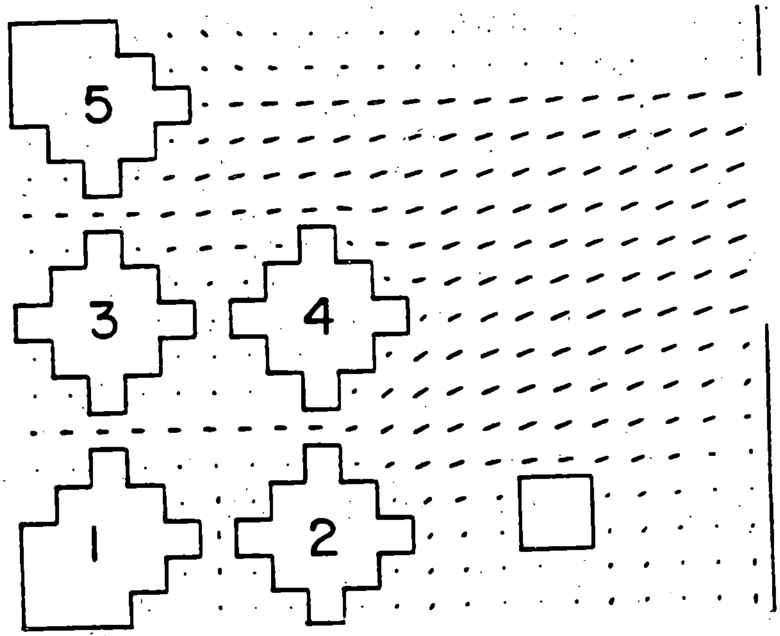


Fig. 2. Steady-state liquid velocity field for the air-water calculation in a horizontal plane at the elevation of the water injection. The speed of the incoming water is 790.4 cm/s.

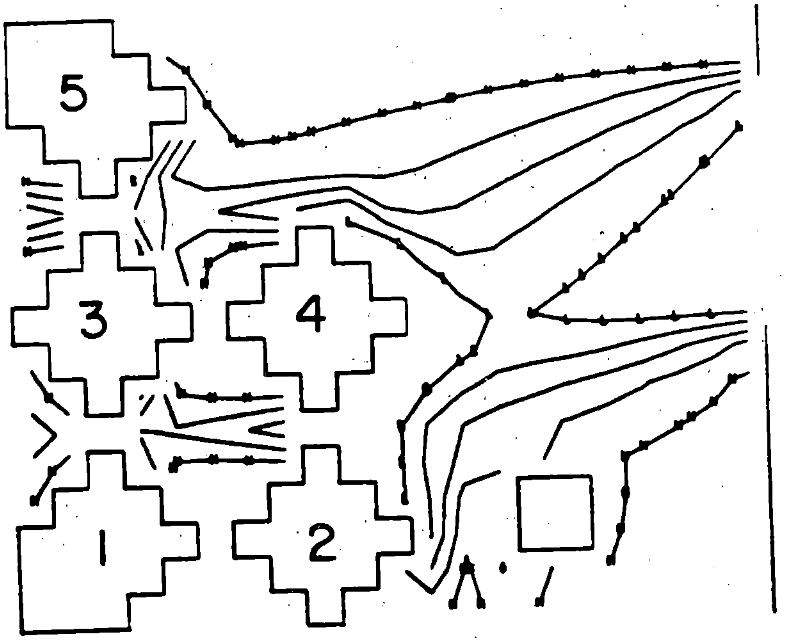


Fig. 3. Steady-state contours of void fraction in a horizontal plane at the elevation of the water injection. The high contour value indicated with an H is 0.9, the low contour value indicated with an L is 0.1, and the interval between contours is 0.2.

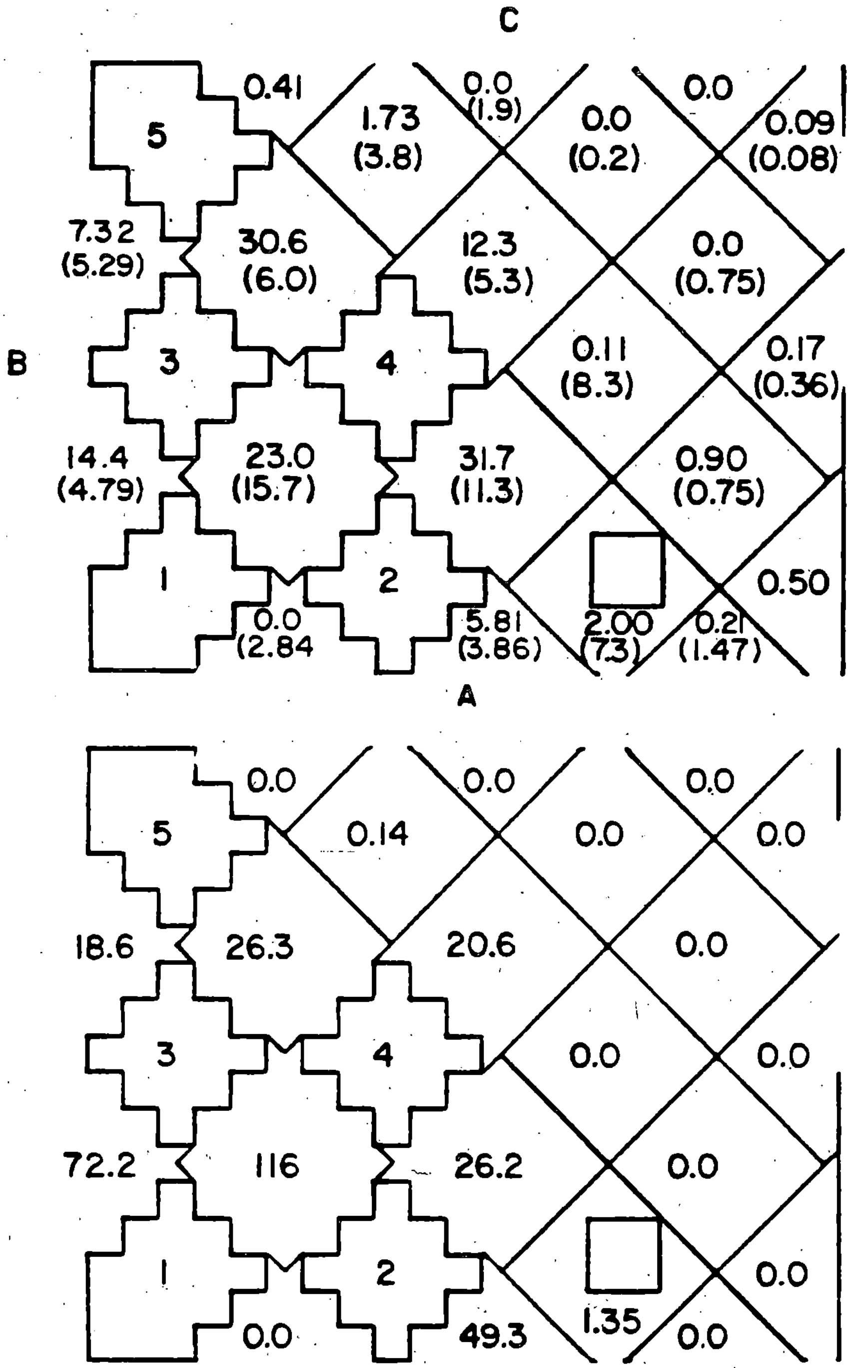


Fig. 4. Flow rate distribution in kg/s for water flowing out of the computing mesh across its bottom boundary for the air-water (upper) and steam-water (lower) calculations. The values in parenthesis are experimental values.

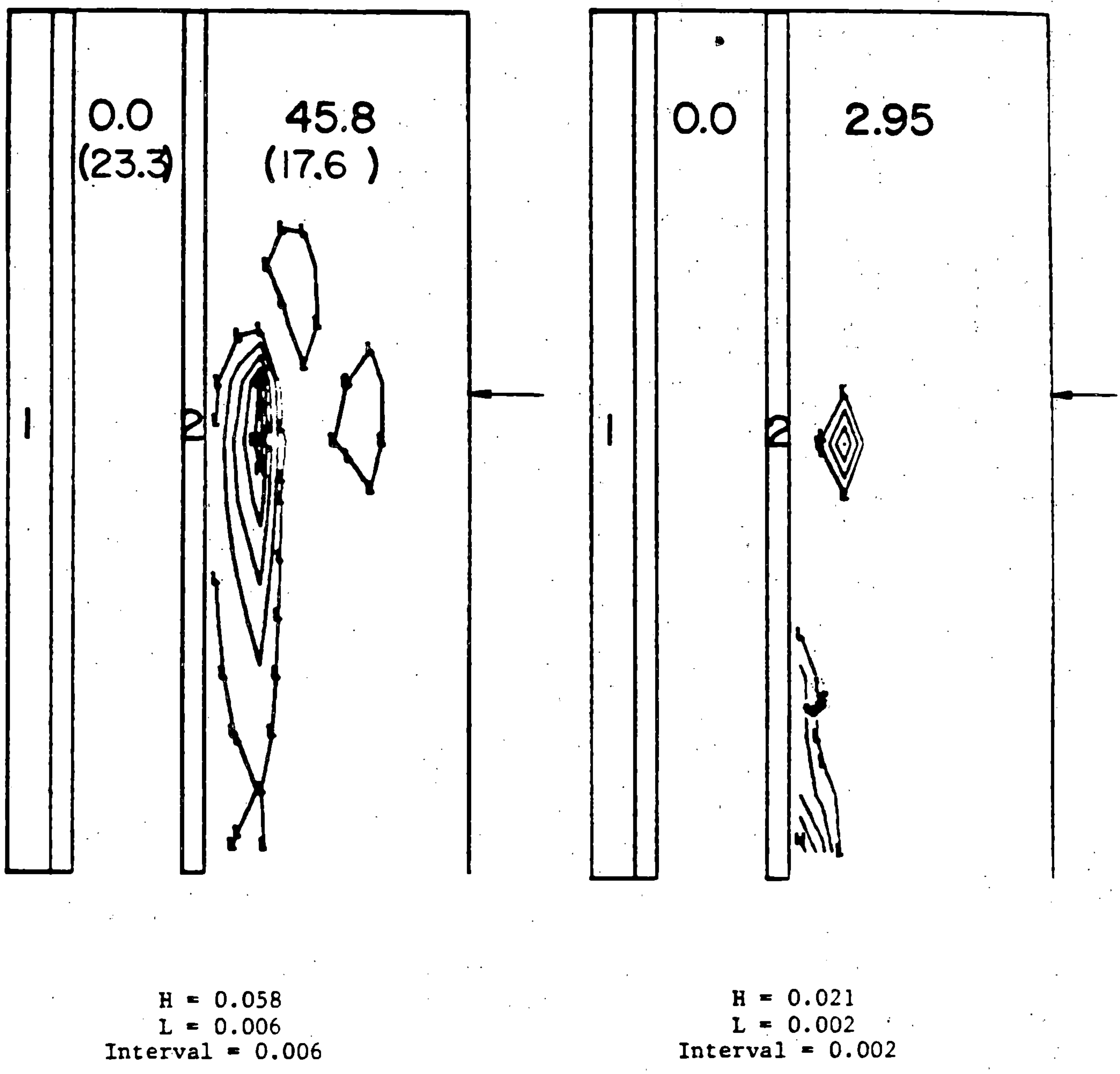
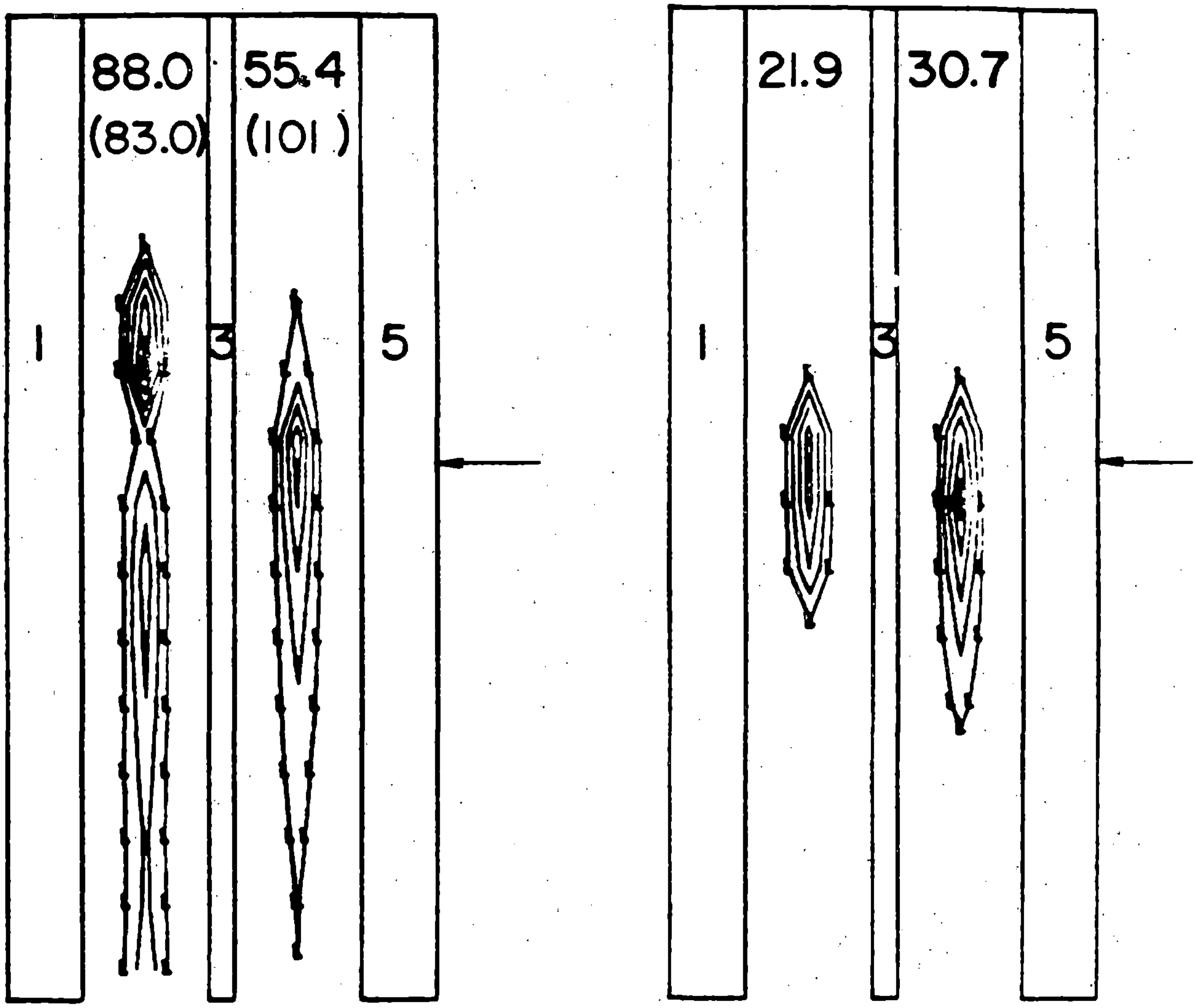


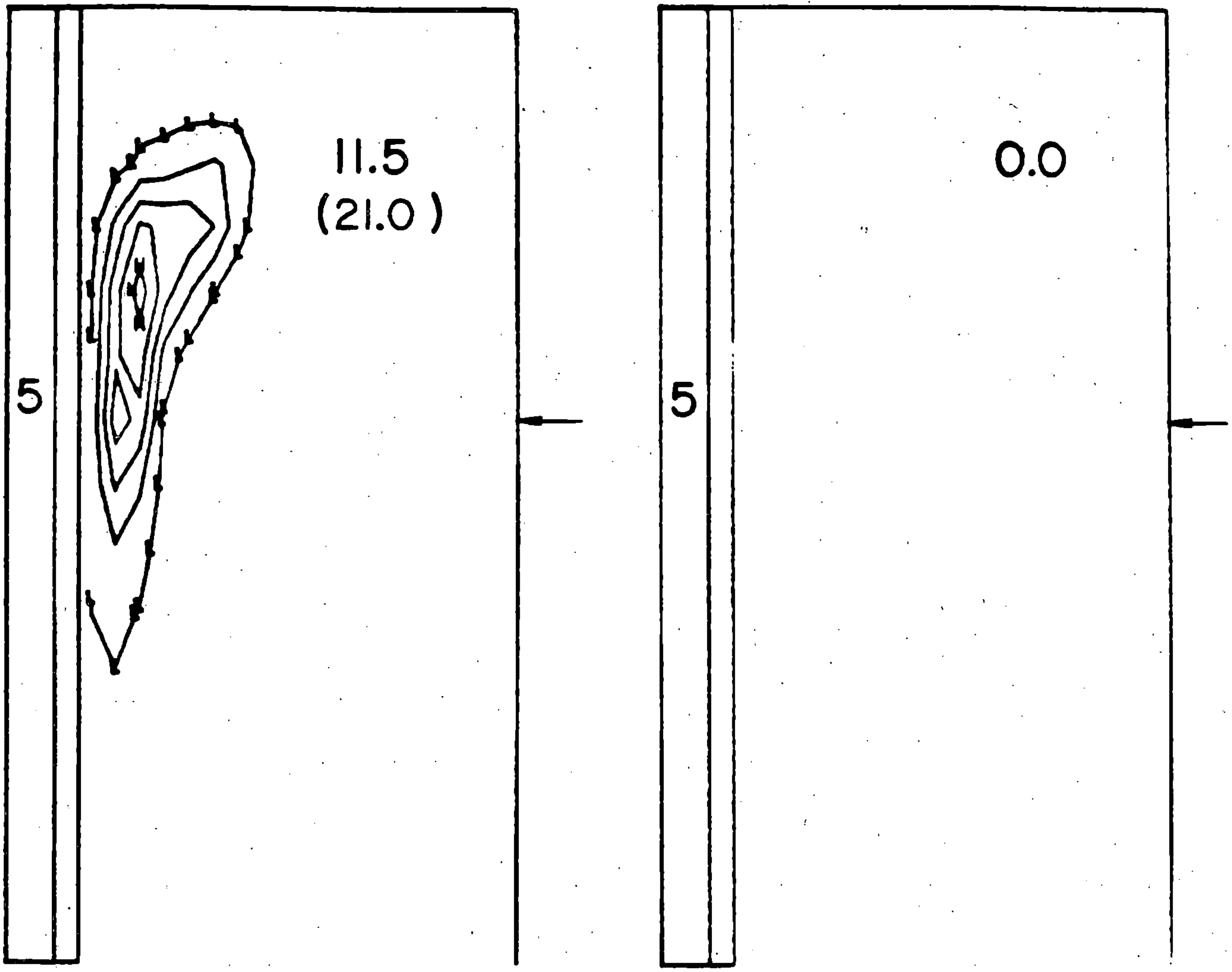
Fig. 5. Contours of mass flux in kg/s·cm^2 flowing across vertical face A (see Fig. 4, top) for the air-water (left) and steam-water (right) calculations. The values shown between the columns are flow rates in kg/s. Those in parenthesis are the experimental values. The arrows show the elevation at which water is injected.



H = 0.197
L = 0.022
Interval = 0.022

H = 0.155
L = 0.017
Interval = 0.017

Fig. 6. Contours of mass flux in kg/s•cm^2 flowing across vertical face B (see Fig. 4, top). The orientation of the plots and the meaning of the values is the same as in Fig. 5.



H = 0.012
L = 0.001
Interval = 0.001

H = 0.
L = 0.
Interval = 0.

Fig. 7. Contours of mass flux in $kg/s \cdot cm^2$ flowing across vertical face C (see Fig. 4, top). The orientation of the plots and the meaning of the values is the same as in Fig. 5.

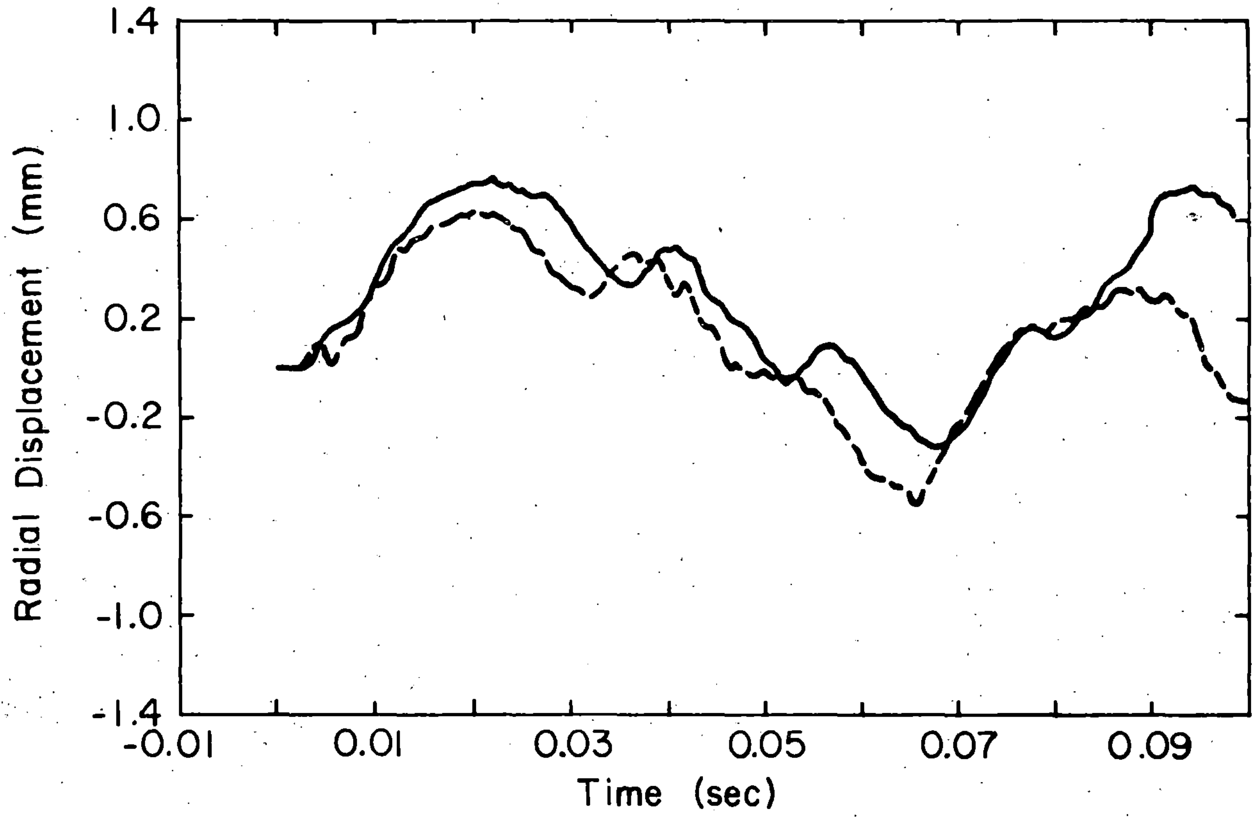


Fig. 8. Comparison of calculated (dashed line) and experimental (solid line) measurements of the core barrel displacement history at a location 107 cm below and 30° azimuthal angle from the centerline of the discharge pipe.

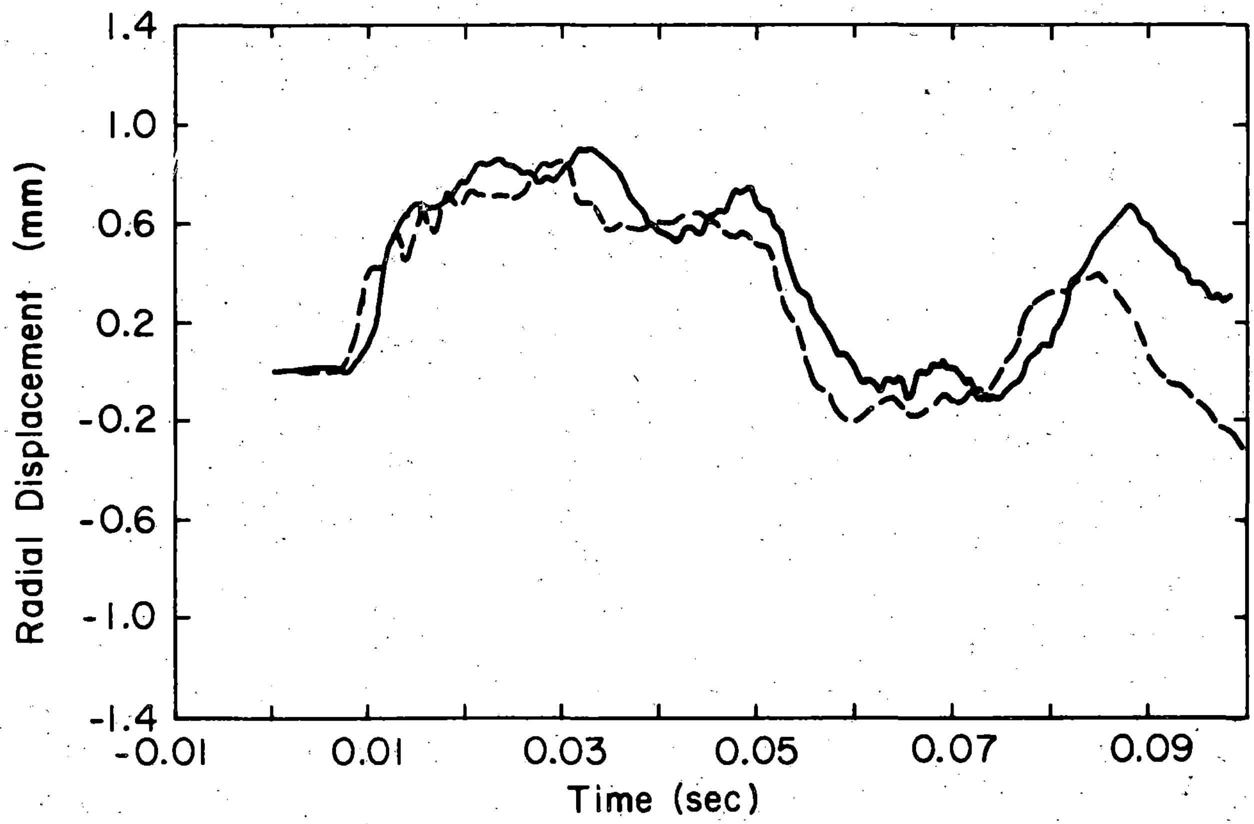


Fig. 9. Comparison of calculated (dashed line) and experimental (solid line) measurements of the core barrel displacement history at a location 330 cm below and 30° azimuthal angle from the centerline of the discharge pipe.

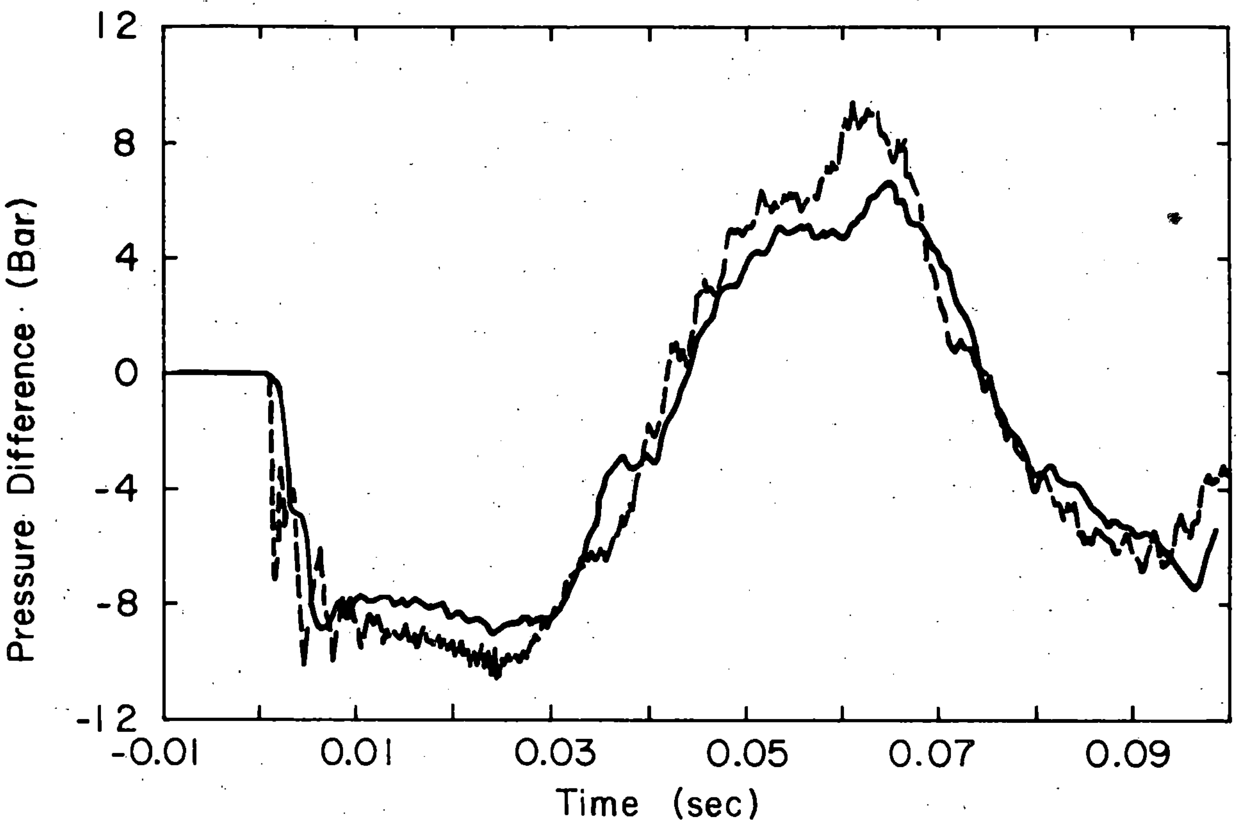


Fig. 10. Comparison of calculated (dashed line) and experimental (solid line) measurements of the pressure differential across the core barrel at the angular and elevation locations of the discharge pipe centerline.

CODE ASSESSMENT FOR NUCLEAR REACTOR ACCIDENT ANALYSIS PROGRAMS

S. Fabic, NRC

Abstract

Systems codes play a very important role in evaluation of safety of nuclear power plants. The so-called Best Estimate systems codes, in contrast to the Evaluation Model codes which intentionally assume the worst, most pessimistic combination of events and processes, are amenable to an in-depth assessment through confrontation with experimental evidence.

That confrontation has many facets, each admitting different approaches or "strategies." The paper describes various considerations, decision points, and alternative strategies that can be employed in the course of code assessment. Those currently pursued by the NRC/RES are indicated. This sizeable effort will take several years to complete, involving four national laboratories.

1. Code Assessment Goals

These can be summarized as follows:

(a) To ascertain whether the code can address the geometries, the system configurations, and all the important thermo-hydraulics and reactor physics-related processes that can occur in those postulated LWR accidents or transients which the code was designed to handle.

(b) To ascertain whether the relevant physical processes are modeled well enough to justify application to LWRs.

(c) To establish the degree of accuracy with which the code can predict its key results.

For example, one of the key results of a systems thermal-hydraulics code is its prediction of the clad temperature distribution in time and space. This information may or may not be utilized in the same code for calculating the peak local clad oxidation, core-wide clad oxidation, hydrogen generated, and the degree of core damage.

Quantification of the degree of accuracy (of a systems code) may also be required for prediction of the coolant mass and energy released to containment, or for prediction of hydraulic forces acting on system components.

The work related with item (a) above is quite straightforward and non-controversial. It merely serves to (1) check the code ability to actually "walk through" the various accident scenarios, without aborting, and (2) establish whether the code yields reasonable results.

The bulk of this paper will therefore focus on the difficult and controversial issues involved in the assessment goals stated in (b) and (c) above.

2. Issues Addressed During Code Assessment

A computer code solves a set of conservation equations that model the pertinent physical processes, utilizing a numerical analysis technique. All that involves structured programming to facilitate storage and transfer of data.

The issues that need to be addressed during code assessment must therefore involve:

(a) The Conservation Equations: These come in many different forms and degrees of sophistication. It needs to be established whether the related form and number has the ability to account for all the fluids and their phases that can be encountered in LWR, for their steady and transient transport in the needed degrees of freedom, and for the local and the global interactions between the fluids (and/or their phases) and between the fluids and the wetted solids. Since interactions involve exchange of mass, momentum and energy, the energy conservation equations must also be addressed for all those solids which play a part in the energy exchange.

(b) Models and Correlations: The right-hand sides of all thermal-hydraulic conservation equations contain all the important terms that model the intricate physical phenomena involving the exchange processes, as functions of the local or global flow and heat transfer regimes; this is illustrated in Figure 1. The most challenging part of code assessment is to ascertain whether the selected models are complete enough, whether they are sufficiently mechanistic to contain a scale-up potential and whether the empiricism built into the models has a wide enough data base to expect applicability to LWR conditions.

(c) Numerical Analysis: Stability, convergence, truncation errors, numerical diffusion, economy vs. accuracy.

During the course of code development, and as part of the so-called "developmental assessment" phase, the NRC contractors involved in code development are required to perform studies of the above aspects of the numerical analysis, including comparisons of code results against analytical solutions, wherever possible. The results of that effort are only audited during the so-called "independent assessment" phase which is performed by personnel who were not involved in code development.

Comparisons of code results against analytic solutions are of very limited value primarily because these are possible only for extremely restricted cases which do not challenge the treatment of non-linearities and of the implicitness of the coupling terms and feedback. A recent international attempt to examine the numerical solution techniques through comparison of results of different codes (for a fairly complex benchmark problem void of test data) was not successful because the experts could not agree as to which calculation result ought to be used as a yardstick.

Clearly, more research needs to be done in this area of qualification of the numerical solution technique. For the time being, we rely on numerous comparisons of code results against test data, for both simple and complex situations. The drawback of this approach is that, when it is established that code improvements are needed, the usual tendency is to start modifying the physical models, thereby absolving the numerics from any culpability.

(d) User Convenience and Quality Control: Code accuracy is a necessary but not sufficient attribute. The code must be designed in such a way that it could be used by engineers who were not involved in its development; the labor necessary for code input generation must be minimized; multiple input options, of the kind that cause two people to get different results for the same case, should be avoided. These issues need also to be addressed during code assessment. We find it helpful to ask, from time to time, two engineers from different institutions to use the code in making comparisons against data for the same test, to see if different results are obtained and why.

Quality control has to be built into the code development procedure to safeguard against coding errors and to guarantee traceability.

The issues mentioned in (d) above will not be further discussed in this paper.

3. Assessment Strategies

3.1 Common Denominator

The heart of the code assessment process involves comparisons of code results with test data, regardless of the manner in which the assessment proceeds; i.e., regardless of the adopted strategy.

Most workers in the reactor safety field are familiar with the overall classification of the experiments which produce the test data base:

(1) The Integral Systems Tests are designed to reproduce, as closely as possible, the overall reactor coolant system thermo-hydraulic behavior under conditions duplicating various postulated accidents and transients, including the effects of parametric variations. Test facilities that fall in this category exist in the United States and abroad, in different geometric scales.

(2) The Separate Effects Tests are designed to produce much more detailed information on the behavior of the individual system components or parts of the overall system, subject to the imposed set of initial and boundary conditions which are parametrically varied to cover the ranges expected during postulated accidents and transients. These types of tests feature the widest range of geometric scales (a number of them in full-scale) and the widest range of parametric variations.

(3) The Basic Tests, sometimes referred to as the "model development tests," are used to study thermo-hydraulic interactions on a very idealized and basic level, and/or to collect empirical information needed to define various constitutive relations.

Further information about the available test data base in all three categories can be found in References (1) and (2).

Parameters that define the individual strategies for code assessment include:

- Emphasis placed on the type of test data selected from the available test data base

- Rate of "sampling" from different categories of the selected test data

- Choice of the computed results for which the code accuracy is to be determined

- Method of quantifying the code accuracy

- Type of the adopted code acceptance criteria

These parameters are discussed in more detail in the sections that follow.

3.2 Emphasis on the Type of Data Selected From the Available Test Data Base

It will be taken for granted that the selected cases for code assessment will involve all three major areas: Integral Systems Tests, Separate Effects Tests, and the Basic Tests. The issue here is whether a given strategy chooses to emphasize some areas over the others.

As an example, the code assessment matrix defined by NRC/RES for assessment of TRAC code involves about 50% of all the available Integral Systems Tests cases, about 30% of the available Separate Effects Tests, and a small fraction of the available Basic Tests.

Stated differently, this choice involves about 90 Integral Systems Tests, about 170 Separate Effects Tests, and about 60 Basic Tests, indicating the fact that the total number of tests available in each category is the smallest for the Integral Systems Tests and the largest for the Basic Tests.

The parameters that influenced our selection were as follows:

(a) Integral Systems Tests:

- Coverage of accident/transient scenarios. Those scenarios associated with a greater risk to public are given more prominence.

- Facility design (PWR, BWR). Facilities featuring nuclear core are given more weight.

- Facility scale - not only involving the volumetric scale but also the number of active loops, core length, steam generator height. The larger the test scale the more emphasis in the selection.

- Quality, quantity and diversity of measurements.

- "Virginity": Tests scheduled on a new facility or new and significantly different tests with an existing facility play a very prominent part in the selected test data base, offering the best opportunity to examine the predictive capability of the code.

(b) Separate Effects Tests:

- Coverage of system components, except for those, such as a centrifugal pump, which the code describes purely empirically. Whether the adopted empiricism for such a component is adequate can be determined from the Integral Systems Test data.

- Design: The more faithful the geometric simulation and the greater the capability to reproduce the processes expected in the particular LWR component, the stronger the candidacy.

- Scale: The larger the scale the stronger the candidacy.

- Quality, quantity and diversity of measurements.

- Virginity.

- Age: Newer tests featuring better controls and measurements take precedence. Example: FLECHT vs. FLECHT-SEASET, etc.

- Potential for studying the code capability to serve as a scaling tool. This includes potential to study the validity of physical models in the code.

- Diversity of the initial and boundary conditions.

(c) Basic Tests:

- One of the biggest contributors to the reduction of the number of candidate cases came from the decision to exclude all tests that were utilized for the development of correlations utilized by the code, and other empiricism embedded in the constitutive relations.

- Tests utilizing fluids other than air, water and steam were eliminated because they would require development of specific equations of state, and the empiricism built into the code involving the fluids of interest in reactor safety, may not be valid for other fluids (freon, etc.).

- The emphasis was on tests, in the same facility, where gradual introduction of complexities could help in assessment of "physics" of fluid flow and heat transfer built into the code. Tests featuring simple geometry and two-phase fluid transients are also useful for assessment of the numerical solution technique.

- Tests sponsored by the NRC have a priority because the analytic support is, in most cases, required to ascertain that test results can be assimilated by the calculation tools (systems and component codes) used in the reactor safety work.

Other choices could be made concerning the types and number of the test cases selected for code assessment. The advent of TMI accident introduced a significant perturbation in the emphasis contained in our first code assessment matrix. New surprises found in the course of research lead to new and important tests that need to be considered. Hence, a degree of flexibility is always required in code assessment, including the awareness of the available resources (funds, manpower, computer access), and of "customers'" needs.

3.3 Rate of Sampling From Test Data Categories

By the "rate of sampling" it is meant the rate with which comparisons between test data and code results are to be made.

The obvious influencing factors have to do with the availability of resources and the progress made in acquiring test data. For example, a number of the important tests belonging to the Integral Systems Tests and the Separate Effects Tests categories are scheduled over the next 4 years. This, incidentally, greatly improves the chances for enlarging the fraction of the "virgin tests" which are extremely important in assessing the true predictive capability of the code. Therefore, if it is agreed that data from all of the important present and future tests must be utilized in the course of code assessment, then the job of code assessment cannot be completed before all the important test data are at hand, regardless of the availability of other resources.

There are, however, other nuances or parameters that can influence the adopted strategy concerning the rate of sampling:

- Sequential rather than parallel approach, in sampling from different test categories. Some strategies may choose sequential completion of the selected cases, starting with the Basic Tests category and leaving the Integral Systems Tests for the last task.

  This strategy will appear most attractive to those who appreciate a bottom-up approach. It should be recalled however that, before the code is ready for an independent assessment, the code developers had to resort to "sampling" from all test categories in order to ensure that the code will be able to at least address the accident's transients with a reasonable chance of success.

  An important disadvantage of this strategy is the fact that the sponsor would have no intermediate information about the code capability to handle its missions, until all code assessment process is completed. The regulatory agencies and institutions involved in the design of the Integral Test facilities - who need best estimate systems codes to help them in the design and conduct of tests and interpretation of test results - could hardly afford this assessment strategy.

- In the parallel approach, one strategy may require accelerated sampling from the most important test category (Integral Systems Tests) to facilitate gathering of information concerning the code capability and accuracy at yearly intervals. Conversely, a strategy could be devised which assigns the largest fraction of sampling from the Integral Systems Tests category, to the final year of code assessment. The primary goal of the latter strategy could be accomplished only if test data for the cases to be sampled in the last year are also made inaccessible to code developers. However, locking-up of test data, especially from the important test facilities, appears to be an impractical task. Therefore, a less extreme strategy would be desirable that allows for both the intermediate information on code accuracy and for the opportunity to apply the code to "virgin" tests.

- One of the important parameters in any strategy involving the rate of sampling involves the fact that improved code versions are being released to the public, a number of times during the period required for the complete sampling of the selected test data base.

  It does not appear feasible to repeat, with every new code version, all of the test cases previously considered. One possibility is to select for repetition (a) one test case from each integral test facility, and (b) those test cases which identified particular weaknesses in the last code version. This strategy is currently pursued by the NRC/RES.

3.4 Choice of the Computed Results Used in Determination of Code Accuracy

The choices of results selected for code assessment could involve:

- Global and local, single-valued results,

- Time histories of results, and

- Statistical measure of fit of time histories.

It is certainly necessary to select those key results which reflect the basic mission of the code and for which information on code accuracy needs to be obtained and compared against code acceptance criteria. In addition, those results must be identified that provide information concerning the code ability to model the relevant physical processes.

(a) Global and Local, Single-Valued Results:

For code comparisons against test data from the Basic and the Separate Effects Tests local, single-valued results need to be determined on a case-by-case basis, depending on the individual processes or system components. Examples are too numerous to be listed here.

The following global and local, single-valued results pertain to comparisons with Integral System Test data:

- Results defining the reactor core clad temperature "signatures." These are described in detail in Appendix A, and are applicable to all types of accidents and transients. Incidentally, they are also applicable to Separate Effects Tests featuring single fuel bundles or bundle arrays.

Other results particularly applicable to Small Break LOCAs, may involve:

- The minimum liquid or froth level (whichever measured) reached in the reactor vessel.

- Amount of heat removed by a specified steam generator, during a specified length of time, t*. The latter may be the final core quench time or the time at which some operator action is initiated, etc.

- Amount of coolant mass lost through the break during time, t*.

- Amount of coolant energy released through the break during time, t*.

- Times (secs) when coolant pressure in the upper plenum reaches 10 and 5 MPa, respectively. These quantities provide the pressure "signature" for certain types of small break LOCA. For types featuring very small break sizes the time and magnitude of the minimum pressure may be more relevant.

Other global results, pertaining to large break LOCA and to non-LOCA transients may involve:

- times to start and end the discharge of ECC accumulators in the intact loop(s)

- time to start LPIS

- time of the minimum coolant inventory within the lower plenum

- time when the lower plenum liquid inventory first exceeds 90% of maximum

- time when the minimum liquid inventory is reached on the secondary side of a specified steam generator

- time of the first activation of the steam generator relief valve

- time of the first activation of the pressurizer safety valve

- the minimum coolant pressure reached, etc.

Examples of various plotting formats for the quantity Ø representing any of the above global results are illustrated in Figure 2.

(b) Time Histories of Results:

Plotted overlays of time histories of the predicted and the measured results provide the most useful information regarding the code capability and, in particular, regarding the consistency of the calculated trends.

In the case of the Basic and the Separate Effects Tests, it is important to plot all results for which measurements are made, as well as other results that shed light on consistency of trends.

For comparisons against data from Integral Systems Tests, examples include the time histories of clad temperature, the mass of liquid within the lower and the upper plenum, upper plenum pressure, local void fractions within regions of the reactor vessel (where measured), the froth or liquid level positions within the vessel for Small Break cases, and results for all the important measurements recorded in the loop spool pieces (local void fraction, coolant temperature, fluid velocity, pressure differentials, metal temperature) and within other system components (steam generator, pressurizer, etc.).

(c) Statistical Measure of Fit of Time Histories:

The overlays of time histories are not amenable to condensation of results and to application of the acceptance criteria. Some researchers have therefore proposed using statistical means of quantifying the discrepancies between the calculated and the measured results. For example, the shaded areas in Figure 3, indicating the amount of discrepancy, could be weighted differently for different time segments of the transient (e.g., blowdown, refill, reflood), for different results (e.g., flows, pressures, temperatures), and even for different regions of the system. The idea is to produce "statistics" of the code accuracy expressible by a figure of merit related to the sum of all the weighted areas of discrepancies, perhaps normalized by the number of terms in the sum. Other "statistical" approaches can be concocted, with an endless variety of weighting factors and figures of merit.

The final aim would be to compare the figure of merit, for each test case, against some acceptable bound and counting the percentage that remained within. The main disadvantage of this approach is that it obscures the information regarding the validity of physical models and the computed trends. In addition, it may be extremely difficult to specify various weighting factors and other assumptions that would be widely acceptable.

(d) Current Approach at NRC:

The current approach being tried out at NRC/RES is to utilize the global and local key results enumerated above for the case of Integral Systems Tests, and confront them with acceptance criteria if and when such become available. In addition, the current approach relies heavily on numerous overlays of time histories and subjective judgement of their validity, for all test categories.

3.5 Characterization of Code Accuracy

Measurement of some physical property, Ø, is reported in terms of its best estimate (or mean, or nominal) value and its uncertainty band, supplemented (in some cases) by the information on the confidence level. Measurement uncertainty is caused by imperfections in the measuring instrument, in signal processing, and in the models through which certain indirect measurements are combined to define the physical property Ø. The narrower the uncertainty band the more accurate is the measurement. In other words, the measurement accuracy is characterized by the magnitude of the measurement uncertainty.

The best estimate code predictions also contain uncertainties. In addition, the nominal or the best estimate value of the code prediction may differ from the nominal or best estimate value of the measurement. As pointed out in Reference 3, the smaller that difference (or the offset) and the narrower the uncertainty "band" of the code prediction, the more accurate is the code.

In what follows, the causes of the prediction uncertainty will be described, together with two methods for its quantification. The preferred method will be indicated.

(a) Sources of Code Prediction Uncertainty:

The uncertainty in the prediction of key single-valued results for LWR can be viewed as being the result of:

1. Uncertainty in the plant condition at the onset of any given accident scenario. The plant condition may include fuel burnup, peaking factors, core power, water levels in ECC accumulators and in the steam generator secondary side, etc. These uncertainties are not considered in the course of code assessment, for obvious reasons.

2. Uncertainties in modeling of reactor fuel rod's thermal and mechanical properties, such as UO_2 thermal conductivity, gap conductance as affected by the gap size, gap gas composition and pressure, pellet deformation, clad deformation, etc. Information concerning the nuclear fuel rod modeling uncertainty is obtained from a separate assessment program involving fuel behavior codes. That information is pertinent to systems codes since, eventually, the latter will include all models and correlations which were found to be important.

   Even though the majority of the test data base used for assessment of systems codes feature electric heaters for simulation of nuclear fuel rods, uncertainties related to their modeling are still present. For example, electric heaters may contain non-uniformities in properties of their materials, non-uniformities in centering of the heater coils or tubes, plus uncertainties in heater coil spacing and installation of clad thermocouples. Effects of these non-uniformities and the shadow and/or fin cooling effects of clad thermocouples are not accounted for in mathematical/ physical models of fuel simulators. Their effects should, however, not be ignored in forming conclusions about the code accuracy.

3. Uncertainties in modeling of the reactor primary and secondary coolant systems thermal hydraulics. Their causes are listed below:

   (1) Code input uncertainties related to physical properties or to those coefficients whose specification is left to code user's discretion. Current trend in design of advanced codes is to eliminate, as much as possible, input choices left to user discretion.

(2) Coefficients embedded in the code that are related to physical models and/or correlations.

(3) Degree of system geometry discretization used for numerical solution.

(4) Upper limits on time steps and on convergence criteria.

(5) Adequacy, or inadequacy, of the set of conservation (field) equations solved in the code.

(6) Adequacy, or inadequacy, of the thermo-hydraulic models for interphasic and fluid/wall interactions, and for the flow/heat transfer regime recognition criteria.

(7) Truncation and numerical diffusion errors inherent in the numerical solution strategy.

(8) Inability to address phenomena of stochastic nature.

(9) Coding (programing) errors.

(b) Quantification of Code Uncertainty Through Statistical Code Uncertainty Study:

A summary of this approach is given in Appendix B. This method is applicable only to quantification of code results uncertainties caused by the uncertainties in items listed in (1) through (4) above. This is a very important limitation of the method. The second limitation is that the information obtained on code uncertainty is tied to the particular accident scenario used in the Study. The third limitation is that the method requires knowledge of the uncertainty range and the probability distribution function for each of the "input" parameters. Only a limited number of such parameters can be considered since very significant expenditures in computing resources are involved. Consequently, prior knowledge of the importance of each parameter is needed to select only those that are judged to significantly affect the uncertainty of the final result. Finally, the sampling strategy and its amount may be affected by the choice of the final result for which the code uncertainty is sought.

One of the important key results is the global peak clad temperature, GPCT. Various code uncertainty studies, utilizing the techniques summarized in Appendix B, have shown that, for the case of the so-called Design Basis LOCA, the peak clad temperature is normally distributed about its mean or the "best estimate" value. Hence, if its standard deviation is denoted σ_{TOT}, then, in the absence of plant condition uncertainty,

$$\sigma_{TOT} = \sqrt{\sigma^2_{T-H} + \sigma^2_{FB}}$$

where: σ_{T-H} is the contribution to the total standard ~~deviation, caused by the uncertainties in the~~ modeling of thermal-hydraulics (T-H)

σ_{FB} is the contribution due to the fuel behavior modeling uncertainty and/or due to uncertainties associated with the electric heaters and clad thermocouples mentioned above.

The smaller the standard deviation σ and the smaller the offset between the predicted mean (best estimate) value and the measured mean value, the more accurate is the code in predicting the selected key result. Offsets usually indicate some sort of systematic error which can be traced and removed, leaving only the random error.

Information at hand indicates that for the case of the design basis LOCA, prediction uncertainty concerning the logarithm of the core-wide amount of clad oxidation and, by inference, of the local amount of clad oxidation (logarithm), is also normally distributed. Therefore, the use of the standard deviation is applicable for description of code accuracy for, at least, the peak clad temperature and the logarithm of clad oxidation.

(c) Information on Code Uncertainty From Scatter Plots:

In the statistical code uncertainty study, input parameters are varied around their best estimate or nominal values. If, on the other hand, code predictions are made of many test situations, using only the nominal or best estimate input values, plots of the predicted minus the measured (nominal or best estimate) values of key, single-valued results will exhibit scatter, as illustrated in Figure 2. That scatter will not only reflect the uncertainties associated with the nominal (BE) values of the code input and the embedded coefficients but, in fact, it will account for all of the effects listed under (1) through (9) in (a) above.

Through proper normalization of the ordinate in the scatter plot the abscissa can account for a large variety of test conditions, in different geometric scales.

Scatter plots are amenable to quantification of the code uncertainty probability distribution function and of the offset, providing sufficient number of entries are present to provide for a statistically meaningful count.

This approach is currently pursued by NRC/RES and it greatly influences the selection strategy for the number and type of cases to be considered.

(d) Summary:

Any best estimate type analysis is associated with an "uncertainty band" reflecting the code accuracy. The narrower the band (or scatter) the more accurate is the code. For this assertion to be valid the prediction accuracy must be tested for a variety of key results that characterize the important thermo-hydraulic processes, in different geometric scales and with different boundary conditions. We at NRC also insist that overlay plots of a variety of calculated time histories (measured and predicted, or just predicted if the measurements are not available) be obtained to give indications of the computed trends and consistency. Plots of the spatial distribution of the local results (predicted minus measured) are also used to infer whether the physical models are adequate.

3.6 Acceptance Criteria

Code acceptance criteria are aimed at providing a yardstick for judging whether the code is accurate enough to fulfill its intended mission. If the criteria are met, further efforts in code development would be either terminated or greatly diminished. There is no universal agreement as to the need for code acceptance criteria since advances in sciences and in the applied research, will continue regardless of whether someone, at some time, believes that the current knowledge is adequate. On the other hand, it is very important for the regulatory agencies to know whether and when a code can be relied upon. Some might argue that the answer to this question cannot be obtained in the absence of a yardstick; others may feel that a subjective judgement could serve equally well.

Most people would agree that predicted results which lie between the measurement uncertainty bounds, are automatically acceptable. However, some measurements are poor enough to provide little challenge even for simple codes. In other instances, the computational mesh is made coarse to such a degree that many measurements are taken within a computational cell. Their combined scatter is then used to define such a wide "uncertainty band" that most codes would pass the accuracy test. A typical example may be a very coarse nodalization of the reactor core and comparison of clad temperature signatures. This approach serves no useful purpose in code assessment, unless trying to prove that, due to special conservatisms used in the code, the computed results upper-bound the measured temperatures. On the other hand, some of the measurements (temperatures, pressures, pressure differentials) are so accurate that the code acceptance criteria based on their measurement uncertainty bands are unnecessarily stringent.

Realizing, therefore, that such easy acceptance criteria are not going to be helpful, let us see what could be done for the selected global results. It should again be pointed out that a global result is represented either by the difference between the predicted and the measured value, or by that difference divided by the measured value. The former is more amenable to temperatures and certain other single-valued results. On the other hand, the global results involving time are better represented by the latter method, owing to wide ranges of transient durations.

The acceptance criteria may be connected to some regulatory requirements or may originate from an accuracy goal which is thought to be achievable.

The current regulatory requirements for conservative analyses of the design basis LOCA prohibit the peak clad temperature from exceeding 2200°F. The best estimate analyses of the design basis LOCA yield much lower peak clad temperatures. How accurate should be such best estimate analyses? Having concluded in the preceeding section, that the best estimate analysis is associated with a probability distribution - which appears to be "normal" in the case of peak clad temperature - it may be possible to define an acceptable standard deviation as function of the regulatory limit. For example, it could be required that the standard deviation, σ_{TOT}, be of such magnitude that the probability of the peak clad temperature exceeding the regulatory limit (2200°F) be equal to or less than, say 5% or less. As illustrated in Figure 4, such a criterion would tolerate fairly large uncertainty (for the peak clad temperature) if the best estimate (or the mean) value were much lower than 2200°F. Conversely, if the best estimate value of GPCT happened to be much closer to the regulatory limit, the required standard deviation would be so small as to be unattainable.

It is very unlikely, however, that best estimate predictions of the key results that directly affect reactor safety, are going to be very much smaller than the regulatory limit, for every conceivable type of the accident or transient. Considerations of multiple failures and operator actions may, in some cases, lead to cases in which the regulatory limit is not only reached but even exceeded. It appears, therefore, that the above described prescription for code acceptance may not be very useful.

This leads us to acceptance criteria that are based on an accuracy goal which is thought to be achievable. However, the proof of the code accuracy must come from an in-depth assessment of the code involving many comparisons with test data. The available, or the achievable, test data base may, therefore, in itself impose a limitation on the code accuracy which could be substantiated as pertinent to LWRs.

These thoughts lead us to believe that a reasonable accuracy goal, reflecting the current state-of-the-art in code development, could only be posed after a good deal of experience has been gained in the assessment of the current generation of codes. From what is known today and based on the experience gained thus far in the assessment of the advanced systems code (TRAC), a reasonable accuracy goal for the peak clad temperature may amount to 150-200°F in σ_{T-H} and 200-250°F in σ_{FB} (nuclear fuel), resulting in σ_{TOT} of about 250-320°F for calculation of accidents and transients which do not involve any significant core damage.

Twenty percent accuracy on times t_{PCT}, and t_{LO} may be achievable. No experience exists so far to forecast the achievable accuracy on $I_{\Delta T}$ or on ΔR_{OX} (defined in Appendix A).

One may be tempted to invoke the code sensitivity studies for prioritization of various systems effects on the clad temperature signatures, since other key results measure the code ability to describe these effects; more stringent accuracy would be required for those systems effects that affect more strongly the core thermo-hydraulics. Bearing in mind, however, that the same code would be used to analyze different accidents and transients, it appears that such prioritization efforts would lead to conflicting requirements. For example, good description of the steam generator thermal hydraulics plays a minor role for large break LOCAs. Yet, very good description of steam generator behavior is extremely important for certain small break LOCAs. Similarly, very good description of the pressure-time history is not essential for the large break LOCA, yet it is very important for small break LOCAs and for certain non-LOCA transients.

Are we again arriving at the conclusion that best achievable accuracy is needed for all key results, regardless of the event being analyzed (by the same code), or should separate acceptance criteria be written for different classes of events?

The author believes that towards the end of 1981 enough code assessment experience will be gained to be able to specify the achievable goals (acceptance criteria) for the key results defined in this paper.

It would take much longer to specify acceptance criteria for the derived results based on statistical manipulation of the predicted vs. measured time histories. This approach is not being pursued by the NRC.

In the meantime, emphasis is being placed on the displays (overlays) of the measured and predicted time histories of all results listed in Section 3.4, to ascertain whether correct trends are predicted and to make subjective judgements about the code adequacy.

4. Summary and Conclusions

Conclusions were reached that the most important aspect of code assessment involves numerous comparisons of code prediction results, with measurements obtained in various domestic and foreign test facilities. The measurements made in full-scale LWR plants are also to be used where available.

It was shown that the code assessment process involves great many selections from among alternative approaches, for each step of the way. The selections are not all black-and-white; many are subject to valid criticism and some are controversial.

Five parameters were identified as influencing the code assessment strategy.

The first parameter deals with the type of data selected from among the available test data base. Rationale employed at the NRC for this selection has been described.

The second parameter has to do with the rate of sampling from among different categories of the selected test data base. Preference was indicated for the parallel rather than sequential sampling, and without accelerated sampling from among one particular test data category.

The third parameter involves the choice of the computed results for which the code accuracy is to be determined. It was shown that overlays of time histories of the predicted and the measured results are most informative concerning the code validity in general and validity of the physical models in particular. These overlays form the backbone of the code assessment sponsored by the NRC. However, such comparisons are only amenable to qualitative rather than quantitative assessment of code accuracy. Means were described for condensing the information so that quantitative assessment could also be made, indicating those currently pursued in the NRC program.

The fourth parameter has to do with the adopted method for quantifying the code accuracy. It is shown that, for a variety of reasons, results of code predictions involve an uncertainty band and maybe an offset between the measured mean and the predicted nominal (best estimate) value of each key result. Hence, the adopted measure of the code accuracy is expressed by the width of the uncertainty band and by the amount of offset. For reasons discussed in the paper, preference was indicated for finding this information by means of scatter plots rather than through statistical studies of code uncertainty.

The fifth parameter involves the code acceptance criteria. It was shown that the approach (previously advocated by this author) in which the acceptance criteria for few of the key results are tied to the current regulatory limits may not be generally applicable. The other approach discussed, and recommended, in the paper identifies the accuracy goal for each key result, consistent with what is judged to be achievable with the current state-of-the-art. It is projected that, by the end of 1981, sufficient experience in code assessment would be gained to allow these goals, hence code acceptance criteria, to be stated.

Every currently-adopted strategy will find its supporters and opponents. It is anticipated, however, that as experience is gained in this field the advantages and disadvantages of various approaches will become more discernable, hopefully leading towards a strategy acceptable to most of the technical community.

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APPENDIX A

Quantification of Clad Temperature Signatures

The measured time behavior signature of fuel clad temperature reflects the local core hydraulics. For example, the time when the clad temperature commences its first excursion in the case of large cold leg break LOCA (signature illustrated in Figure A-1) denotes the condition of an increase in the local void fraction and a decrease in the local mass velocity. The temperature decrease after its first peak is a consequence of an enhanced local cooling caused by a surge of coolant either from above or from below. The last local quench, at time t_{LQ} is caused by the reflood process. A small break LOCA clad temperature signature is illustrated in Figure A-2 where the onset of temperature excursion indicates the local increase in void fraction caused by the falling liquid or froth level or the depletion of the vessel coolant inventory.

Given the fact that (a) the majority of experiments are conducted employing electric simulators of the nuclear fuel rods, (b) most tests feature fairly extensive measurement locations for the clad temperature, and (c) very few tests provide indication of the local void fraction in the core while measurements of the local fluid velocities in the core interior are extremely rare, the information provided by the clad temperature signatures presents the only feasible way of evaluating the code performance where it matters most.

A computer code calculates such signatures for each computational cell representing a given core region. There may be more than one measured signature within individual computational cells. Some weighted mean - to diminish the influence of thermo-couples facing the unpowered rods or the control rods - must be employed in making comparisons with test data. In the comparisons of the computed and measured clad temperature histories, for qualitative assessment, the upper and the lower bounds of the measured histories should also be shown.

The issues at hand are: (1) how to represent a signature, (2) how to quantify the difference between the measured and the predicted signatures, and (3) how to specify and apply the acceptance criteria.

Three single-valued parameters are indicated in Figures A-1 and A-2 that collectively aid in identifying the signature: The local peak clad temperature, $\overline{LPCT}$, the local time of peak clad temperature, t_{LPCT}, and the local final quench time, t_{LQ}. If these three parameters are insufficient to identify the signature, one may also consider some forms of the weighted integral.

One such integral may be of the form:

$$I_{\Delta T_{sat}} = \int_0^{t_{LQ}} [T_{CL}(t) - T_{sat}(t)]^n dt$$

where n>1 (say, n=2) emphasizes the peaks above T_{sat}. The quantities in the integral and the upper limit of integration would come either from the code (for $I_{\Delta Tsat,calc}$) or from measurements (for $I_{\Delta Tsat,meas}$), utilizing the weighted average mentioned above.

Another type of the signature integral may assume the same form as that used in the computation of the local amount of clad oxide penetration:

$$\Delta R_{OX} = \sqrt{2A \int_0^{t_{LQ}} \exp\,[B/T_{CL}(t)]dt}$$

where it is assumed that no oxide existed at t=0. A and B are given constants featured in the Cathcart-Pawel model.

The author prefers this form because it is relatable to the current regulatory limit for the maximum local clad oxidation. In addition, a summation of all ΔR_{ox} (times the cummulative clad surface within a computational cell), over all computational cells in the reactor core, can be related to the global (core-wide) amount of clad oxidation and, therefore, hydrogen generation. Allowable upper bounds for both are currently specified in the Appendix K acceptance criteria.

Admittedly, the oxidation thickness is not directly measured and the peak clad temperatures reached in experiments may not be high enough to give a significant contribution to ΔR_{ox}. Nevertheless, the difference $\log (\Delta R_{ox})_{calc} - \log (\Delta R_{ox})_{meas}$ is a meaningful representation of the code ability to calculate the clad temperature signature.

The knowledge of how well the local signatures are predicted sheds light on the code's ability to calculate multi-dimensional behavior where it matters most. Even in one-dimensional calculations it is important to know whether the code calculates well the axial distribution of signatures. If these comparisons are not adequate then it is questionable whether the code has predictive capabilities, even it the core-wide properties - such as the global peak clad temperature (GPCT) and the global t_{GPCT} and $(t_{LQ})_{max}$ are well predicted.

Should all of the above parameters (LPCT, t_{LPCT}, t_{LQ}, $I_{\Delta Tsat}$ or ΔR_{ox}) be used in quantifying the prediction accuracy or only some of them? Some strategies may ignore the local signatures and only quantify the accuracy for the global parameters, such as GPCT (= the largest PCT anywhere), and the summation of $I_{\Delta Tsat}$ (or of ΔRox) over all cells. One should bear in mind that the differences between the measured and the predicted times, t_{LPCT} or t_{LQ} or t_{GPCT} or $(t_{LQ})_{max}$ may differ greatly for the large and the small break LOCAs, or for the short vs. long duration transients. For such situations it may, therefore, be more convenient to feature the differences in predicted and the measured times, divided by the measured time, to fit many comparisons on the same scale. It appears that condensation of results of comparisons with many test cases could only be made for the global parameters. The local parameters can be plotted as illustrated in Figure A-3.

The most informative way of displaying the calculated vs. measured signatures is shown in Figure A-4 pertaining to a vertical stack of computational cells at a given azimuthal location illustrated in Figure A-5. Such a display is useful for a qualitative rather than a quantitative assessment of the code and is not amenable to confrontation with acceptance criteria.

APPENDIX B

Code Uncertainty Studies

Computer codes contain many empirical correlations. Each such correlation has its own uncertainty "band" and/or probability distribution. The so-called scatter plots that feature the measured vs. the predicted parameter (e.g., heat transfer coefficient) provide information as to the uncertainty range and probability distribution.

Code sensitivity studies performed during code development are designed to indicate which of the many parameters (coefficients) contained in the code are important with respect to the calculated key result. Those that are deemed to be important are then utilized in the statistical analysis of code uncertainty. Suitable sampling procedures such as the "experimental design" and "latin hypercube" are devised for the purpose of developing a response surface with a minimum number of computer code runs.

Many runs with a best estimate computer code are performed, featuring a value of each important parameter, X_i, selected within the range of its particular uncertainty. Results of all these runs are fitted into the Response Surface, which is an algebraic expression defining the effect of variation of each parameter, X_i, on the key result (such as peak clad temperature, or the % of clad oxidation in the core) predicted by the given systems code, and for a particular accident/ transient scenario.

Other effects that can influence the key results include system discretization (nodalization), time step control, the choice of "user options" in selecting models, and the uncertainties concerning the plant condition at the start of the accident. Advanced LOCA codes are developed in such a manner as to preclude or minimize User Options. Optimum nodalization schemes for LWR plants should be prescribed during the so-called developmental assessment stage of the code, including the recommendations for the optimum time step control. Nevertheless, these variables may still be present and can influence the total uncertainty in the key results. Other, less tangible phenomena can also affect the code results, as discussed in the text.

The next step is to perform Monte Carlo calculations with the response surface to obtain the probability distribution for the key result. This step requires the knowledge of the probability distribution function, PDF (X_i), for each of the variable parameter, X_i, defining the response surface. Many of those are not sufficiently known; however, effects of different distributions could easily be tested since the Monte Carlo calculations with the response surface are fast and economical.

Figure B-1 illustrates the steps in the code uncertainty study.

Code uncertainty studies are very expensive and their results are restricted to the selected accident scenario. In the past, when the main emphasis was placed on the large break loss-of-coolant accident, it was reasonable to argue for the need for such studies to (a) obtain a feel for the probability distribution of the computed key result (the peak clad temperature, etc.), (b) prioritize the efforts in code development, and (c) prioritize the experimental programs.

The new research direction which does not focus on one particular accident scenario may refrain from extensive use of code uncertainty studies unless very fast running and economical tools for analyses become available.

Figure 1

Relationship Between System Region Geometry, Flow Regime, and Physical Models in Field Equations

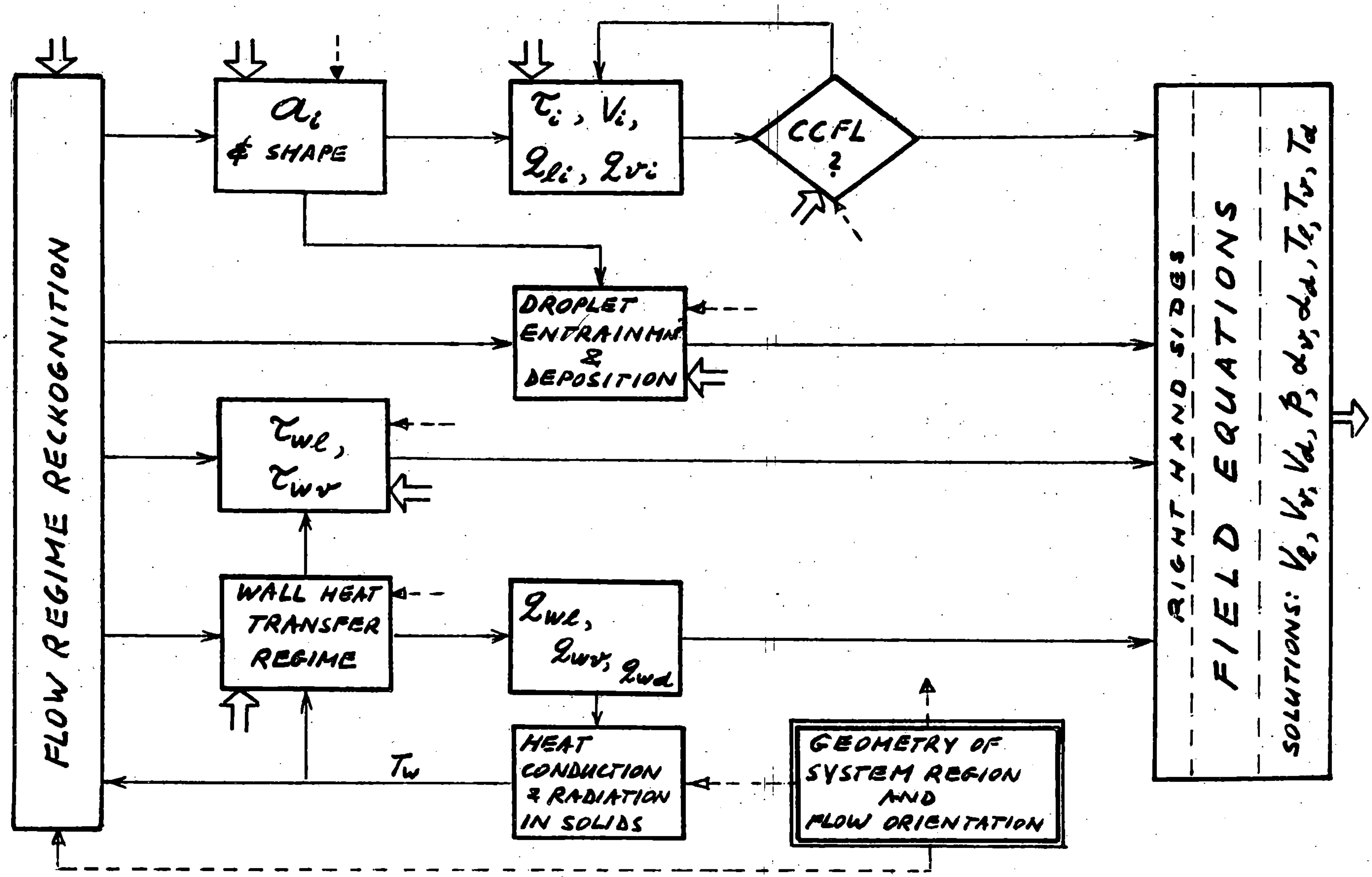


Figure 1 - Continued

Legend:

a_i = interfacial surface area

τ_i, V_i, q_{li}, q_{vi} = interfacial shear, velocity, and heat fluxes, from the liquid and the vapor sides, respectively.

CCFL = empirical correlation for counter-current flow limitation

τ_{wl}, τ_{wv} = shear between wall and liquid or vapor, respectively.

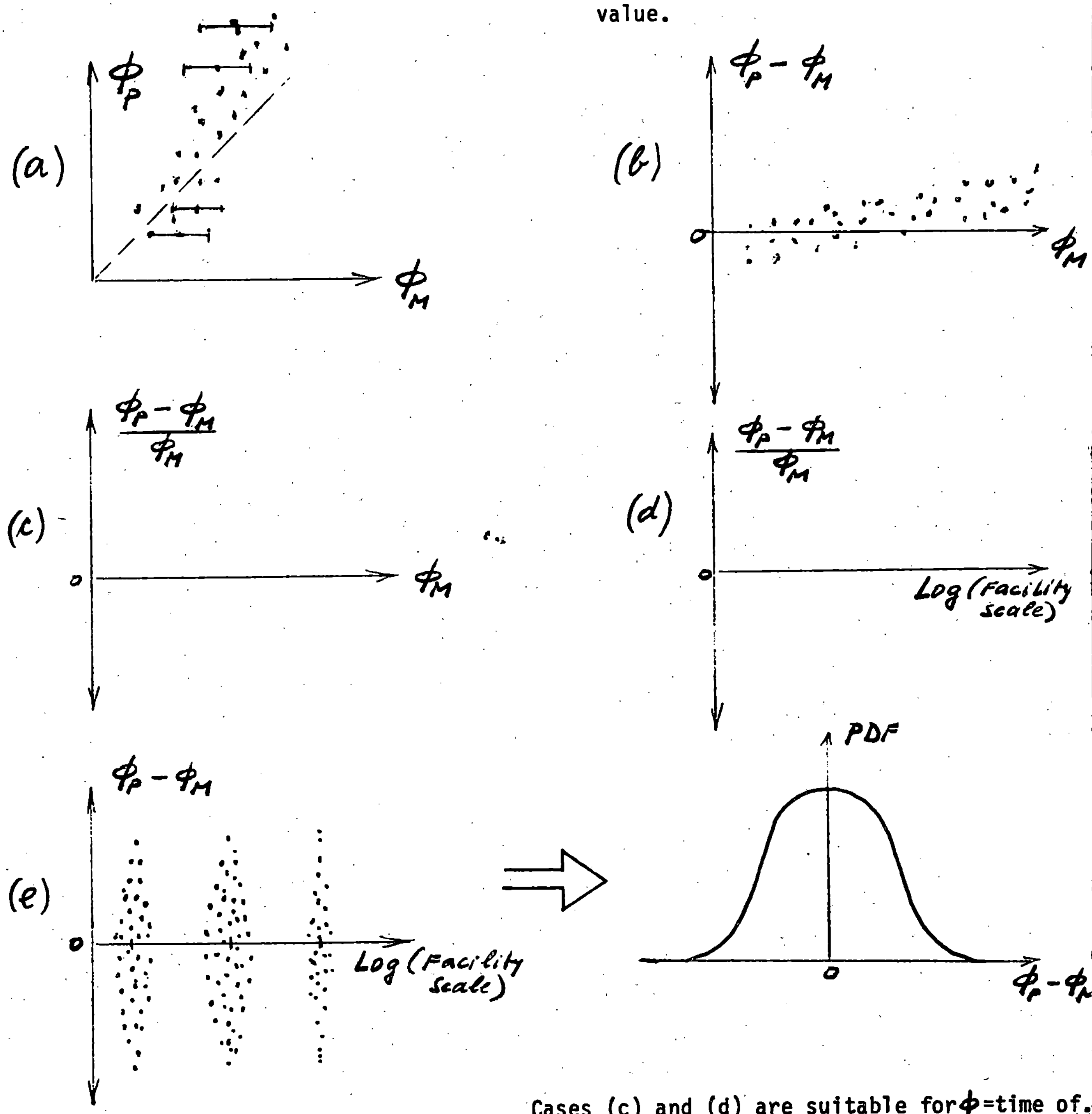
q_{wl}, q_{wv} = heat flux from the wall to liquid and vapor, respectively.

The main point is to illustrate that each of the above models is affected by the flow and heat transfer regimes which, in turn, can be strongly dependent on the geometry of the region of the system being analyzed. The short arrows indicate the dependence on the results of the field equations solutions and on the local geometry. The latter may not be resolvable by the adopted spatial discretization.

Figure 2

Alternative Plots of Global Results

ϕ =single-valued key result
Subscript P indicates predicted value
M indicates measured (best estimate) value.

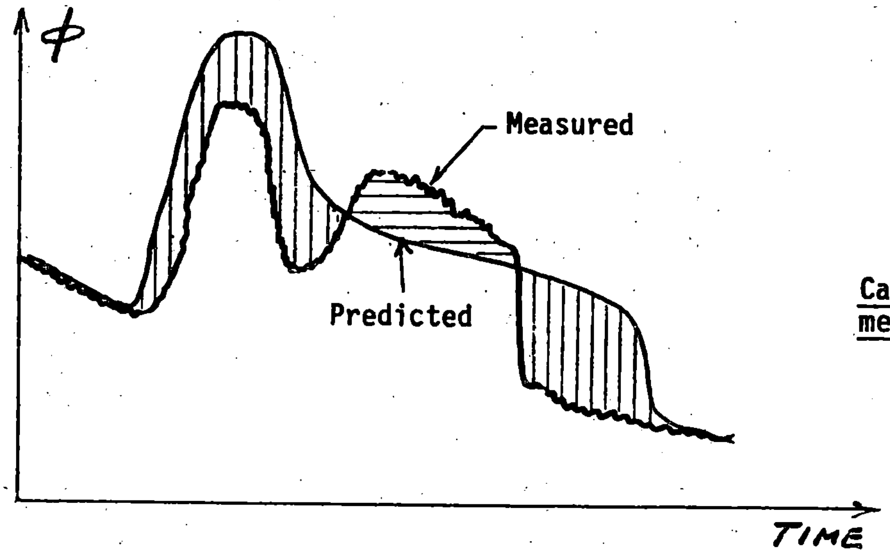


Cases (c) and (d) are suitable for ϕ=time of.
Case (e) illustrates prediction uncertainty which is scale-independent.
If scale effects exhibit clear trends it may possible to extrapolate to LWR.

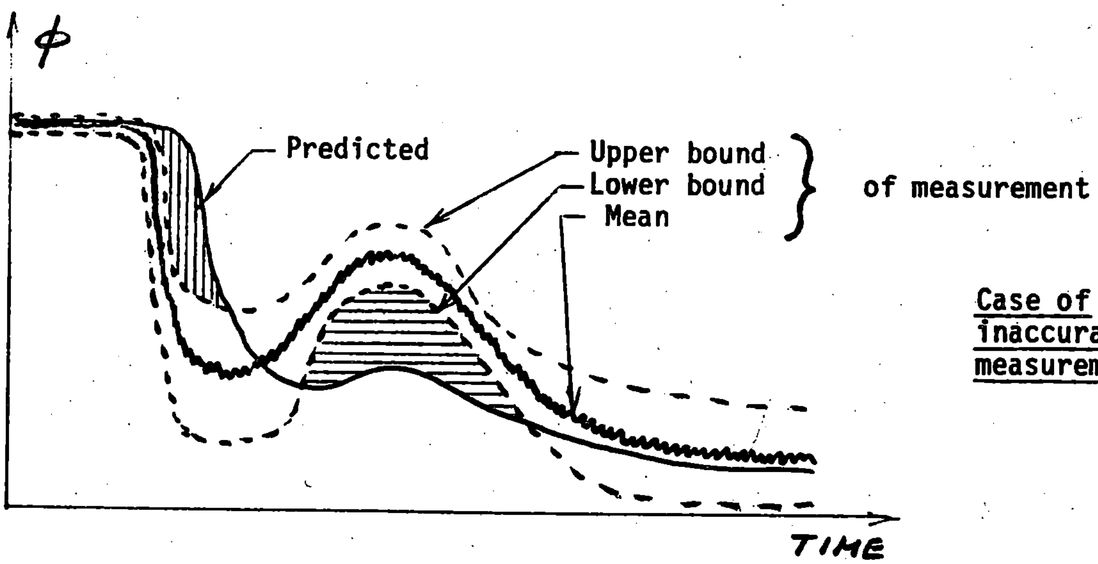
Figure 3

Information for statistical manipulation of overlay plots of the predicted and measured time histories.

Shaded areas indicate zones of disagreement.



Case of accurate measurement



Case of inaccurate measurement

Figure 4

Illustration of two probability distribution functions for peak clad temperature, both obeying the limitation on the probability of PCT exceeding some regulatory limit (EM).

The case featuring the Best Estimate (BE) value of PCT which is closer to the regulatory limit demands a more accurate code, i.e. smaller standard deviation.

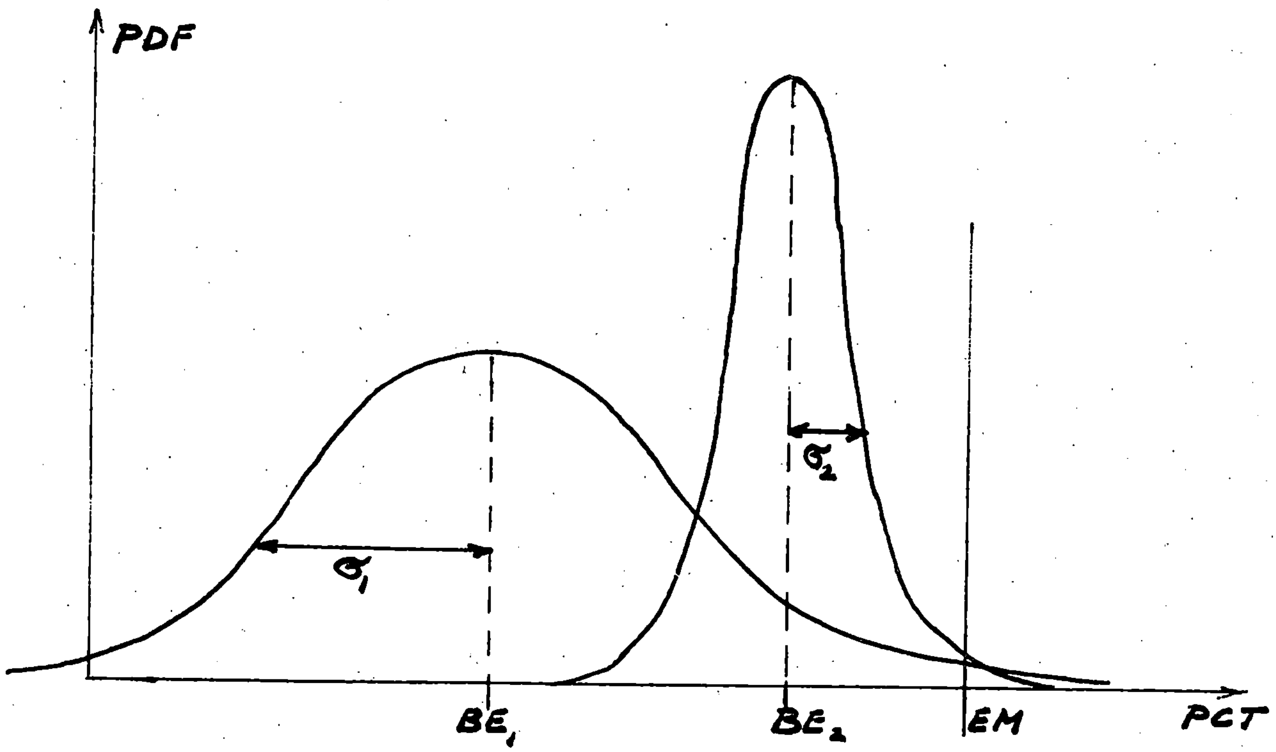


Figure A-1

Sample T_{CL} signature for large break LOCA

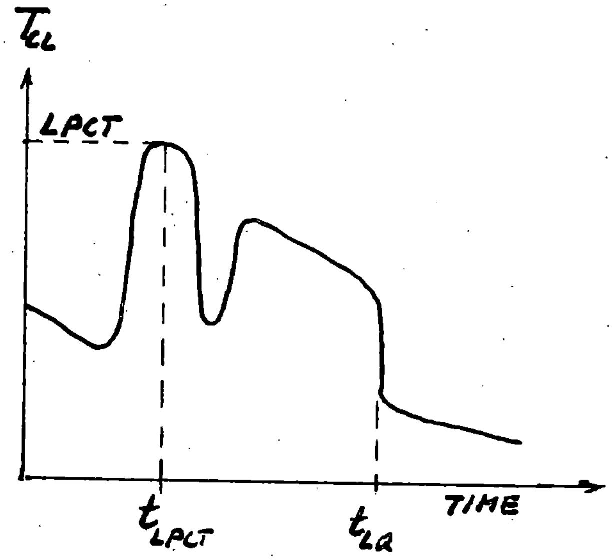


Figure A-2

Sample T_{CL} signature for small break LOCA

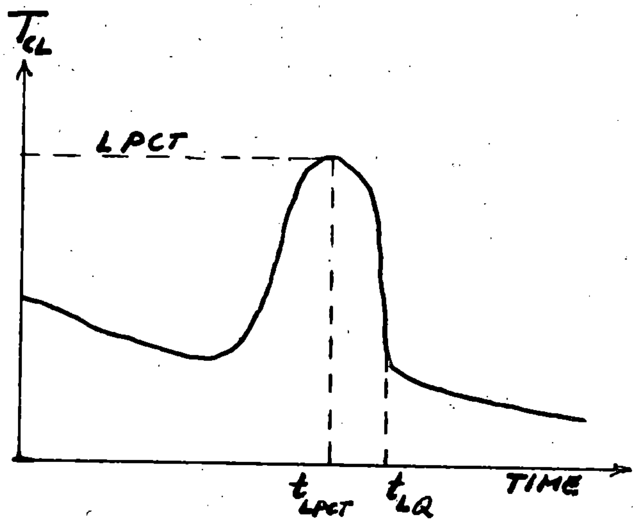
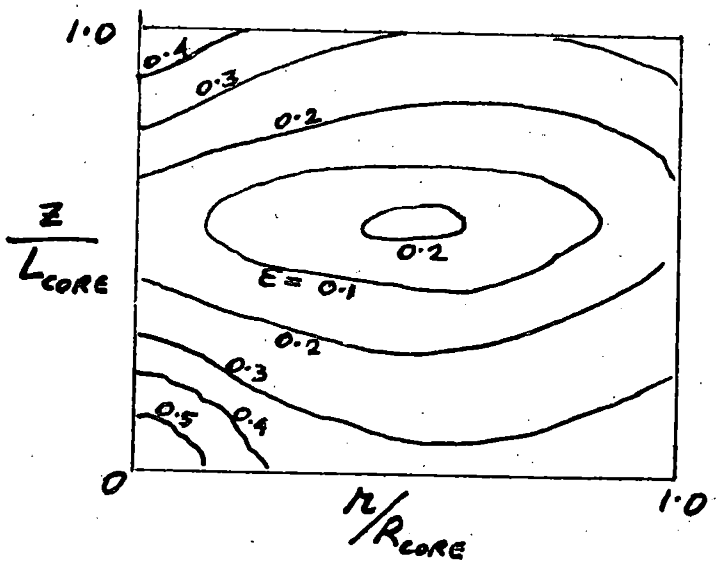


Figure A-3

Contours of equal values of ε for local parameters



$$\varepsilon = \left| \frac{\phi_P - \phi_M}{\phi_M} \right|$$

EXAMPLES FOR ϕ:

LPCT, t_{LPCT}, t_{LQ}, $I_{\Delta T}$, $\log \Delta R_{ox}$ etc.

Figure A-4

Clad. Temperature Histories in the Stack of Cells Pertaining to a Specific Circumferential Zone,

(Used For Qualitative Assessment)

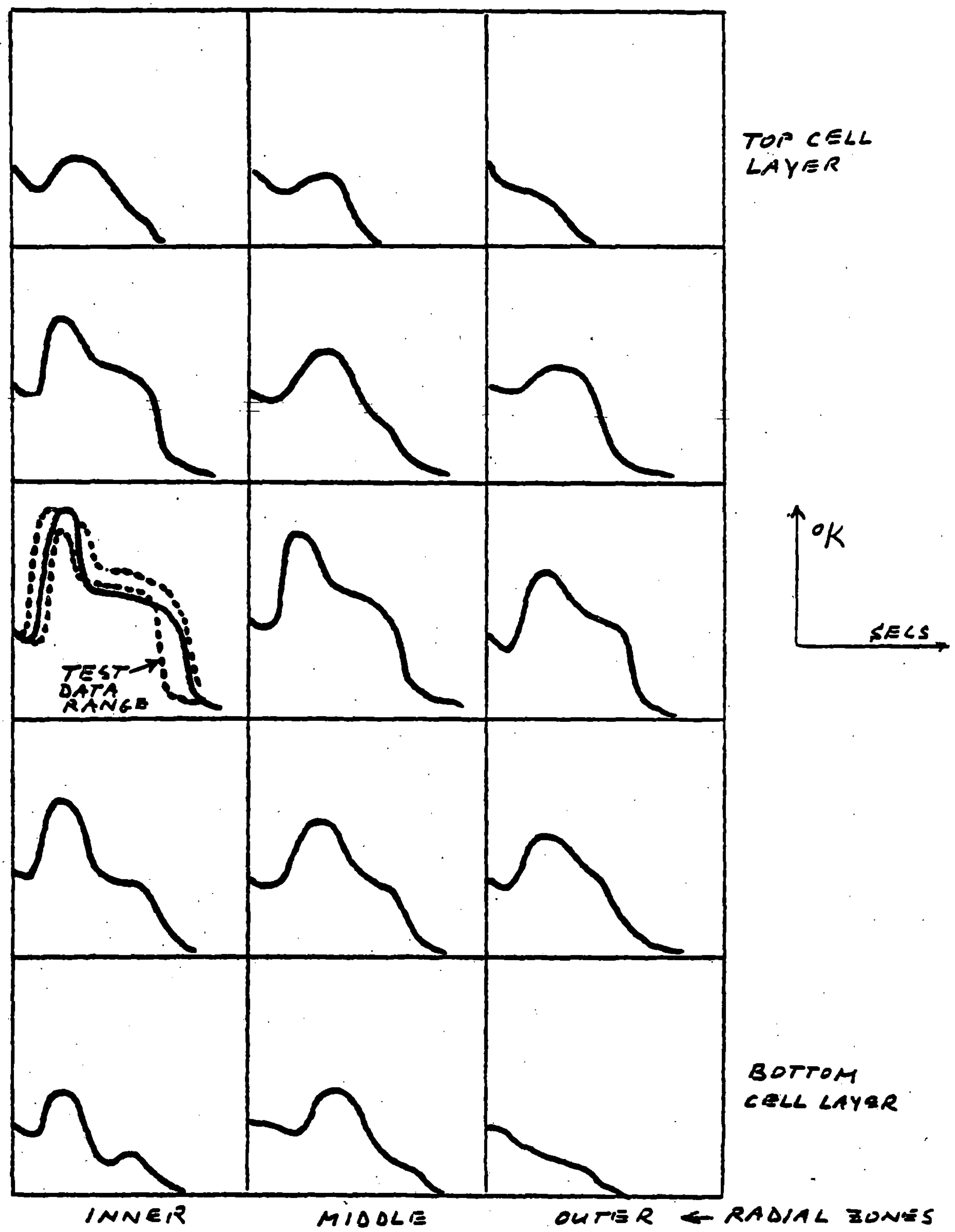


Figure A-5

Example of Three-dimensional Noding of Reactor Core With Illustration of a Stack of Cells For Which the Temperature Histories are Plotted, as in Figure A-4

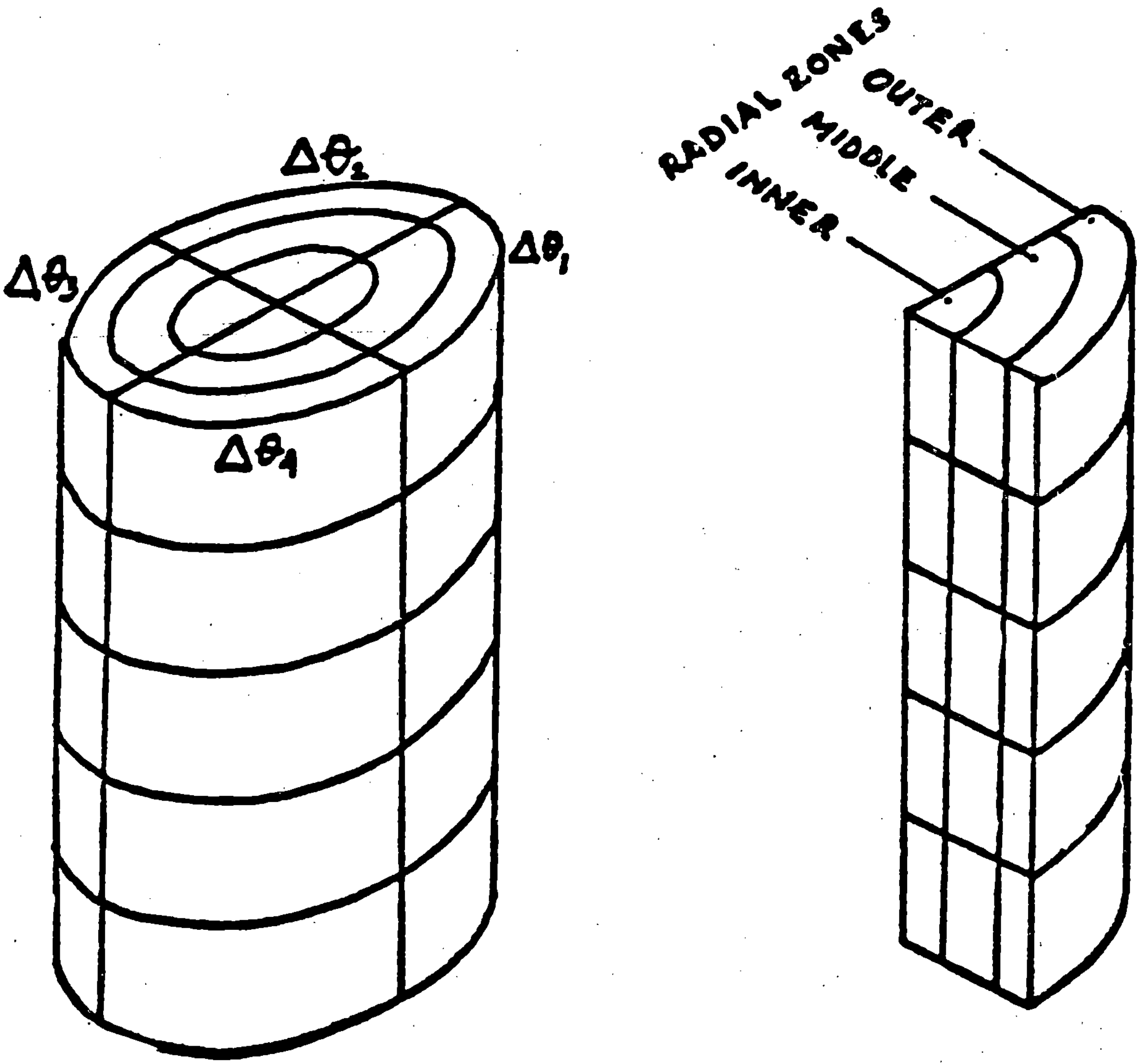
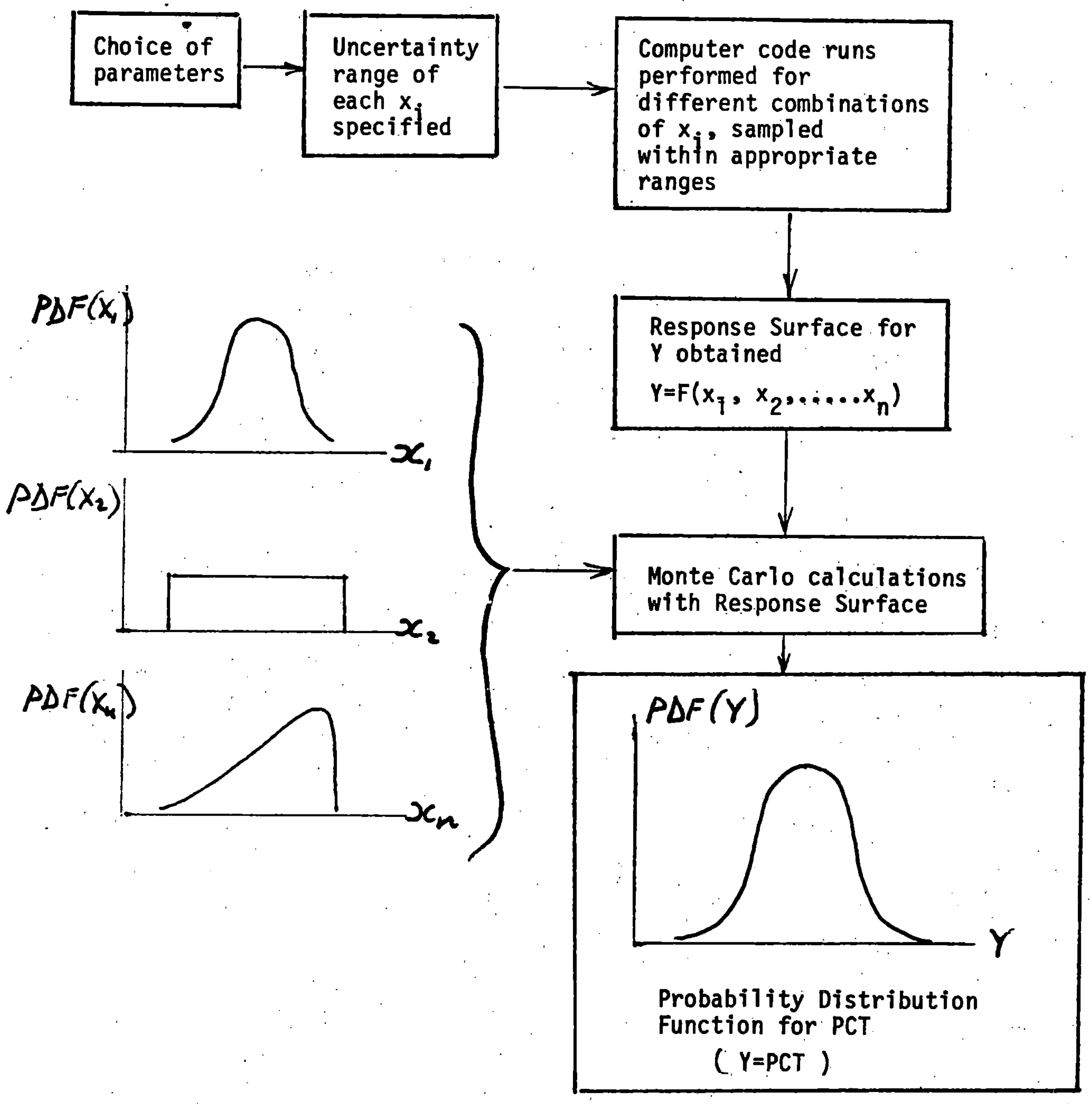


Figure B-1

Illustration of Procedure For Statistical Study of Code Uncertainty



LA-UR-80-3079

TRAC-PD2 INDEPENDENT ASSESSMENT AT LASL*

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LOS ALAMOS, NM 87545

LOS ALAMOS SCIENTIFIC LABORATORY

*RK PERFORMED UNDER THE AUSPICES OF
E US NUCLEAR REGULATORY COMMISSION.

TRAC-PD2 INDEPENDENT ASSESSMENT AT LASL*

by

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Energy Division
Los Alamos Scientific Laboratory
Los Alamos, NM 87545

We are assessing the TRAC code at LASL by comparing the calculated results to data from various experimental facilities. TRAC is being developed at LASL and incorporates state-of-the-art nonhomogeneous, nonequilibrium thermal-hydraulic modeling. During 1980, the TRAC-PD2 development was completed, and we converted the assessment process to the new code.

TRAC represents a best-estimate modeling capability for thermal-hydraulic transients and, as such, the code does not represent a licensing code. However, the code is useful in analyzing both hypothetical and real light-water-reactor transients that are pertinent to the licensing and regulatory processes. These analyses provide an in-depth understanding of the phenomena taking place and thus provide a background for possible licensing decisions. Currently, TRAC is being used by both the regulatory and research sides of the NRC for just such analyses. Independent assessment attempts to quantify the predictive capability of the code for various applications through comparisons to data and sensitivity studies; this assessment supports the code application to hypothetical and real transients when the supporting data is either nonexistent or very limited.

Since the last information meeting, independent assessment has consisted of analyses of LOBI, Semiscale, FLECHT-SEASET, and LOFT tests. All of this work is ongoing. Pretest and posttest predictions of LOBI test A1-04 have been conducted with TRAC-P1A. The pretest prediction of Semiscale test S-07-10D that was made with TRAC-P1A is being repeated with TRAC-PD2, and analyses of tests S-SB-P1 AND S-SB-P7, the pumps-off and pumps-on counterpart small-break tests, are underway with TRAC-PD2. Posttest, blind predictions for FLECHT-SEASET tests 31701 and 31805 were made with a preliminary version of TRAC-PD2 for U. S. Standard Problem No. 9. Pretest predictions of the LOFT small-break tests L3-1, L3-2, L3-7, and L3-5 were made. TRAC-P1A was used for L3-1, and preliminary versions of TRAC-PD2 were used for L3-2 and L3-7. The test L3-5 prediction was completed with TRAC-PD2. Posttest analyses of tests L3-1 and L3-7 have been conducted with TRAC-PD2.

For small-break tests in LOFT, the performance of the secondary system is more important than it is in the large-break tests. For the LOFT small-break analyses the secondary system, consisting of the steam-generator secondary and steam line to the isolation valve, has been represented with more fluid cells.

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* This work is performed under the auspices of the U. S. Nuclear Regulatory Commission.

The new nodalization scheme provides the recirculation path in the steam-generator secondary through the steam-generator downcomer. The more detailed nodalization results in a more representative secondary liquid inventory. A new valve type was added to the code to describe the opening and closing characteristics of the steam flow control valve.

The constitutive relations in TRAC do not permit the accurate calculation of critical flow when the upstream fluid conditions are subcooled, but this period of the blowdown is important to the overall depressurization transient. For the LOFT small-break tests a coarse nodalization scheme represents the break orifices. The additive friction factor or the wall friction is then used to adjust the flow based on a critical flow model.

The overall LOFT system model has been modified to represent known leakage paths. In particular, the reflood-assist-bypass lines have been connected, and the steady-state flow is approximately 6% of the total intact-loop flow. Also, the heat losses from the primary system to the environment have been included in the input model.

The calculated results for tests L3-1 and L3-7 have been compared to data. The system depressurization is good, and the pressurizer empties at the correct time. For test L3-1 the accumulator injection begins and ends at the correct times. The comparisons for the steam-generator secondary pressures are sensitive to, among other things, the amount of leakage permitted through the steam-flow-control valve. The calculated break flow is, on average, representative of the experimental data, and the system-pressure comparisons indicate that the proper magnitude of the flow is calculated. The liquid level in the vessel always remains above the core, and the core remains well cooled.

TRAC-PD2 represents a useful small-break modeling capability for predicting most thermal-hydraulic phenomena during slow transients. Improved critical flow modeling is needed, either in improved constitutive relations or in a critical flow model. Both of these paths are being explored as a part of the current TRAC development. Finally, small-break analysis requires a detailed definition of the flow paths, including the leakage paths, and it is desirable to have measurements for all important flows.

TRAC-PD2

INDEPENDENT ASSESSMENT

AT LASL

T. D. Knight

Energy Division
Los Alamos Scientific Laboratory

LICENSING AND SAFETY IMPACT

Provides an assessed, state-of-the-art thermal-hydraulic code for analyzing a large variety of hypothetical and real transients in light water reactors

INDEPENDENT ASSESSMENT TASKS

LOBI Test –
A1-04

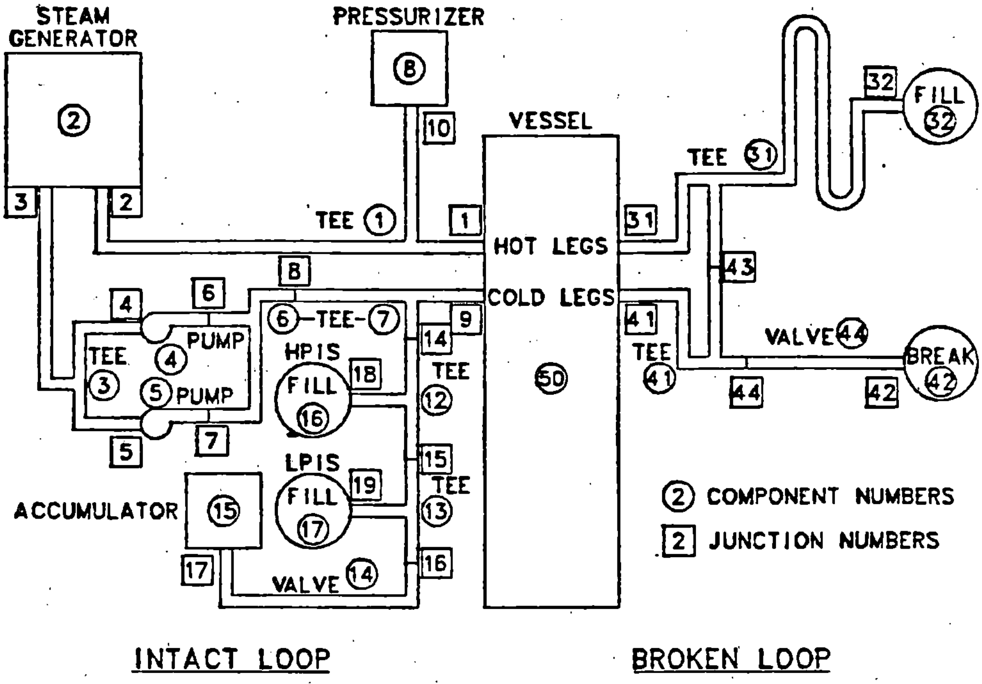
Semiscale Tests –
S-07-10D, S-SB-P1, S-SB-P7

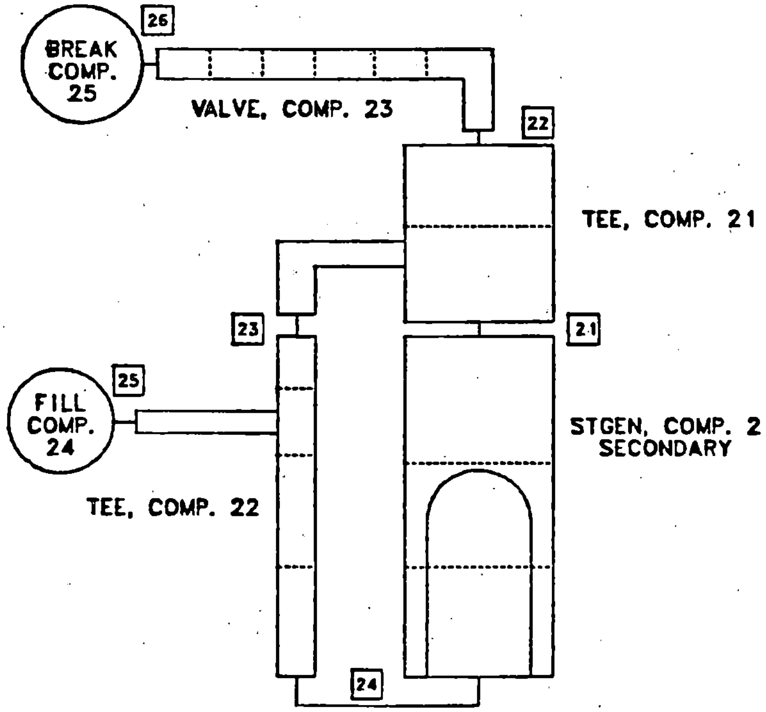
FLECHT-SEASET Tests –
31701, 31805

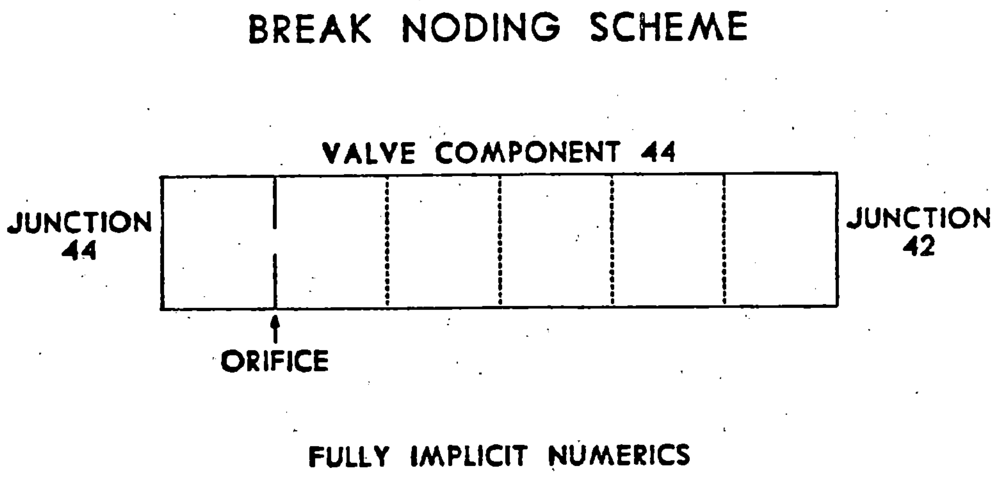
LOFT Small Break Tests –
L3-1, L3-2, L3-7, L3-5

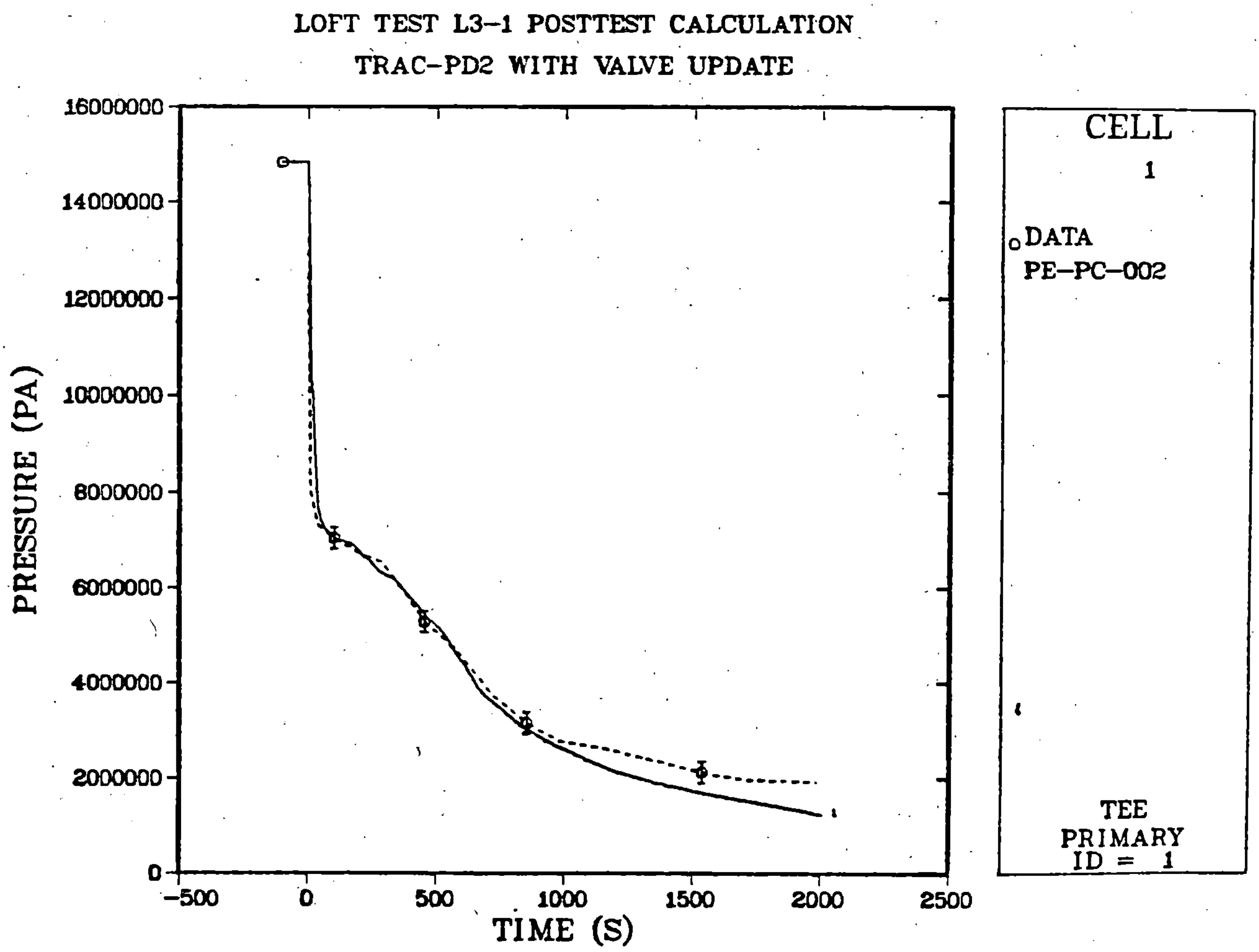
INITIAL CONDITIONS

	L3-1	L3-7
Power (MW_t)	48.9	49.0
PCS Flow (kg/s)	484.0	481.3
PCS Pressure (MPa)	14.85	14.90
Hot Leg Temperature (K)	574.0	576.1
Break Diameter (single-ended cold leg) (mm)	16.2	4.1
Scram (s)	-2.1	36.0


STEAM GENERATOR
2
PRESSURIZER
8
10
VESSEL
TEE 1
1
3
2
HOT LEGS
31
TEE 31
32
FILL 32
43
COLD LEGS
41
8
4
6
6 — TEE — 7
9
14
TEE
PUMP
4
HPIS
FILL
16
18
3
5 PUMP
TEE
12
VALVE 44
BREAK 42
TEE
41
44
42
50
5
7
LPIS
15
19
TEE
13
ACCUMULATOR
15
FILL
17
17
VALVE 14
16
2 COMPONENT NUMBERS
2 JUNCTION NUMBERS
INTACT LOOP
BROKEN LOOP

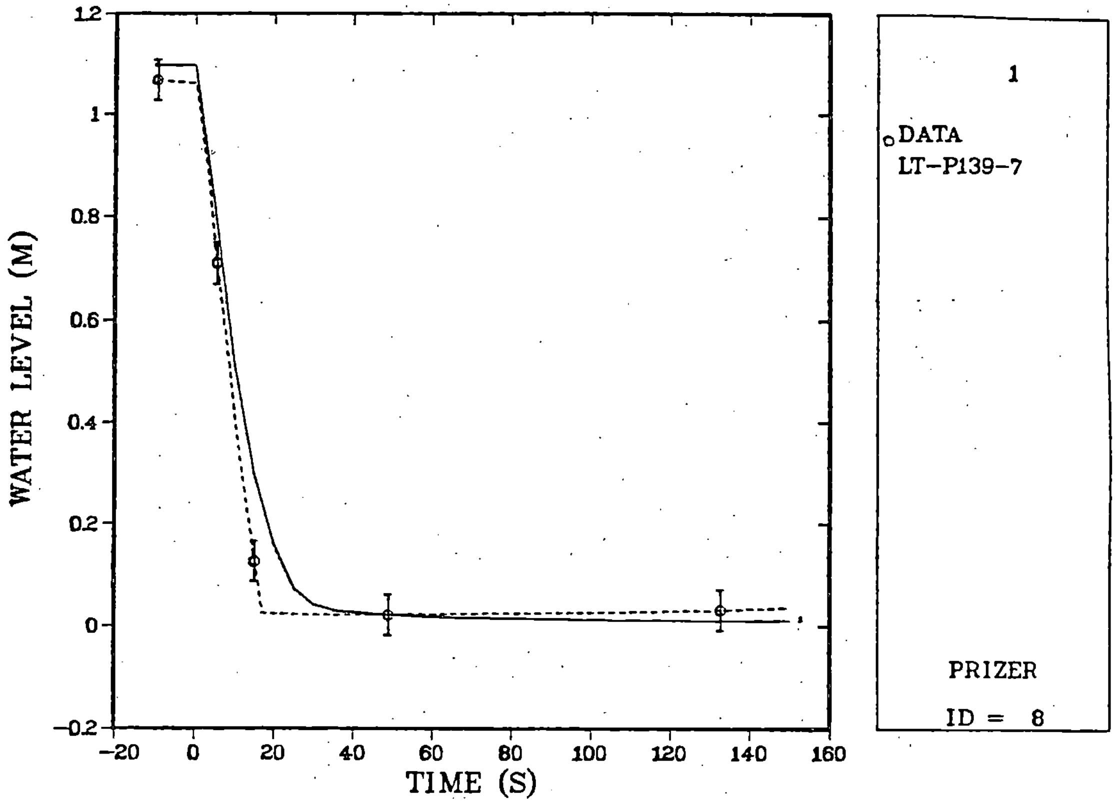

BREAK COMP. 25
26
VALVE, COMP. 23
22
TEE, COMP. 21
23
21
FILL COMP. 24
25
STGEN, COMP. 2 SECONDARY
TEE, COMP. 22
24





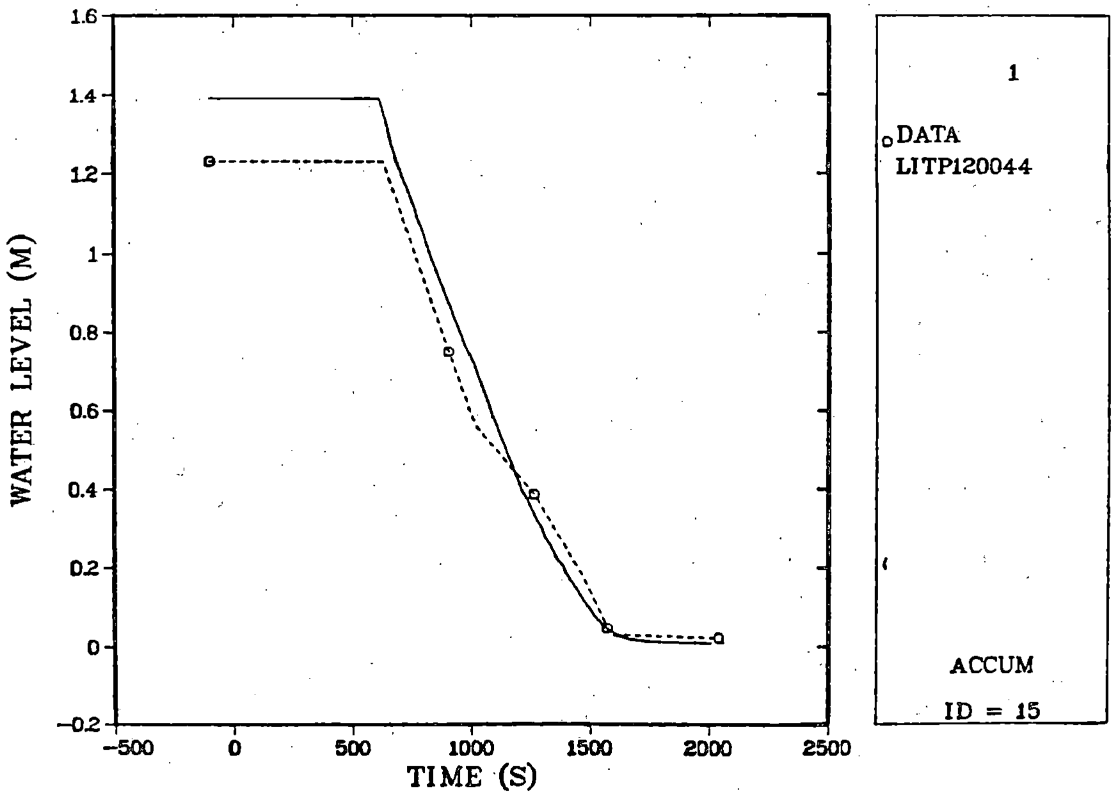
L3-1 intact loop hot-leg pressure.

LOFT TEST L3-1 POSTTEST CALCULATION
TRAC-PD2 WITH VALVE UPDATE

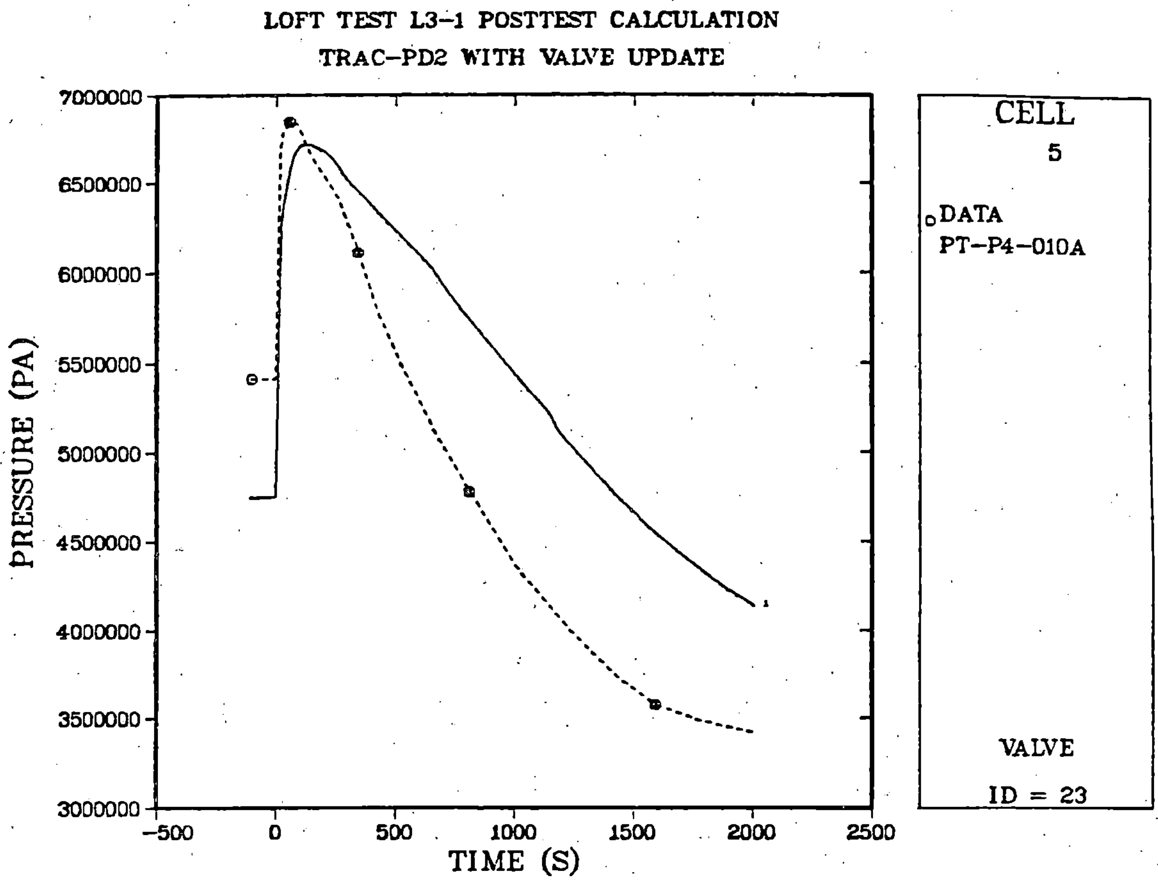


L3-1 pressurizer liquid level.

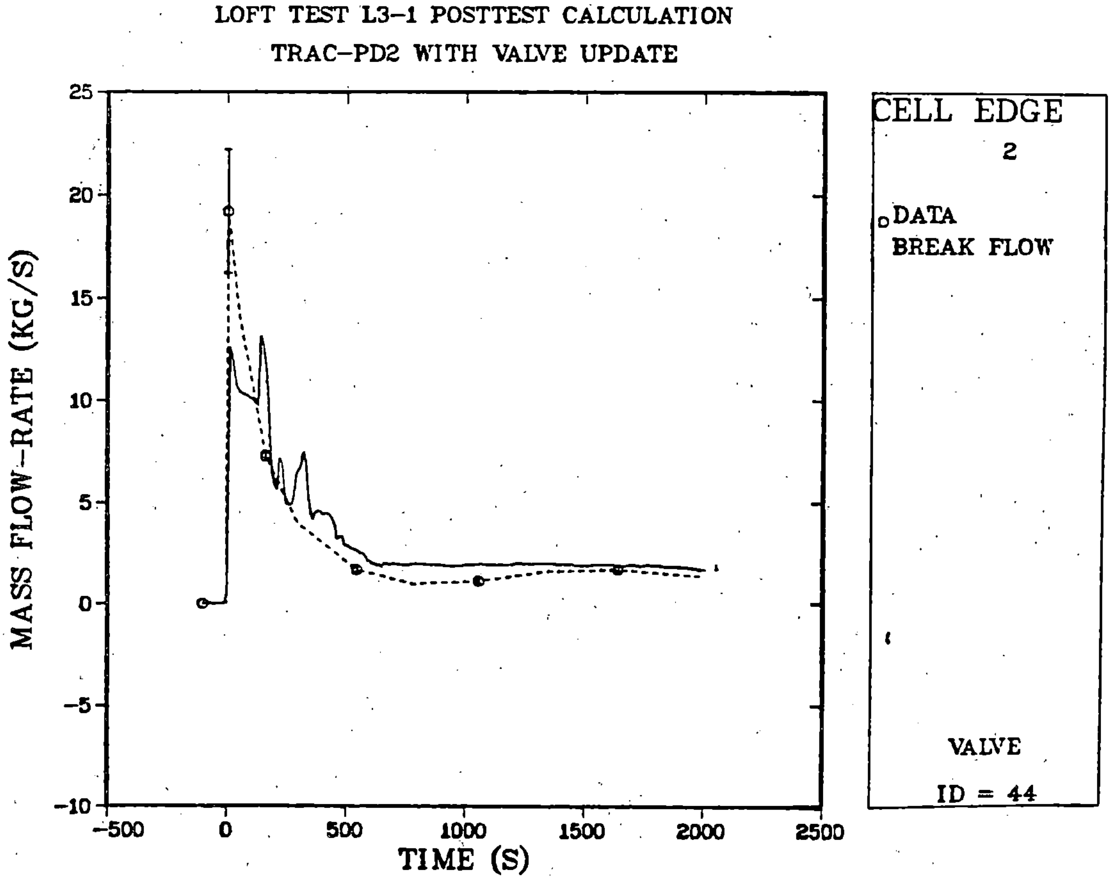
LOFT TEST L3-1 POSTTEST CALCULATION
TRAC-PD2 WITH VALVE UPDATE



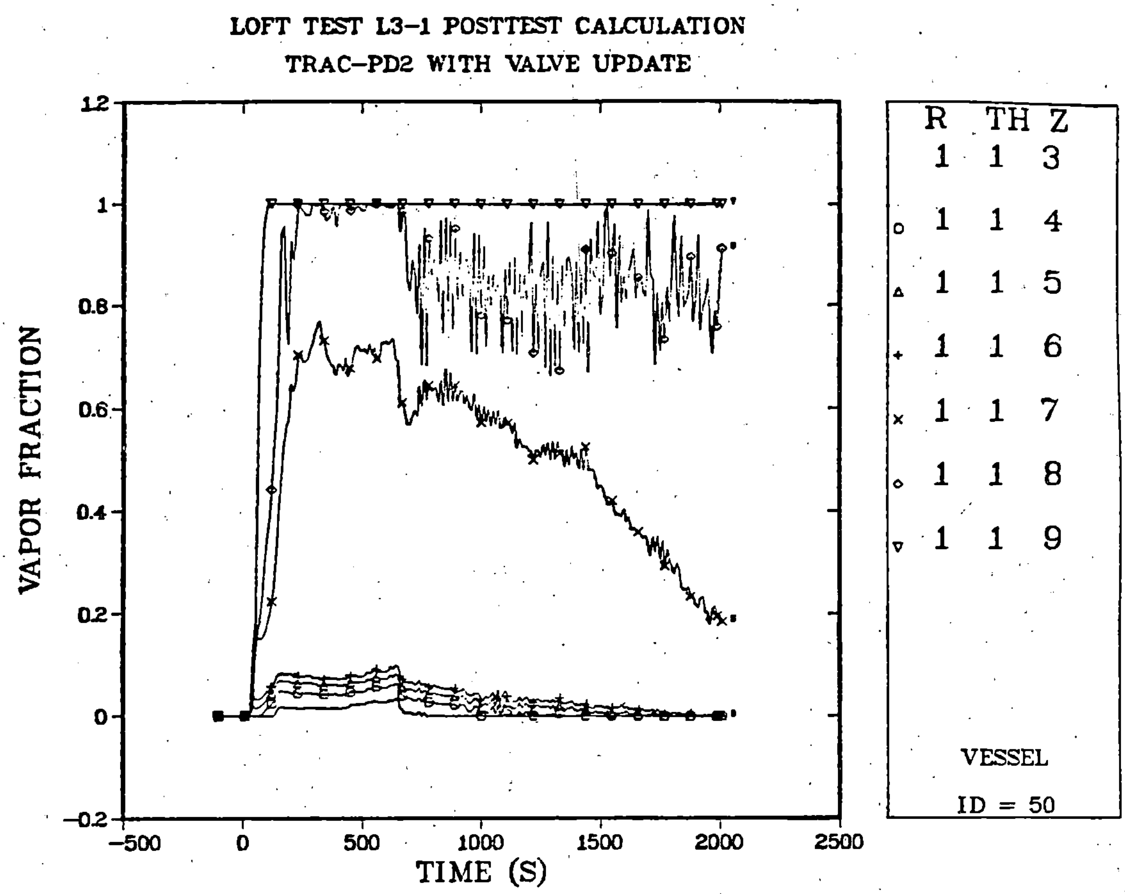
L3-1 accumulator liquid level.



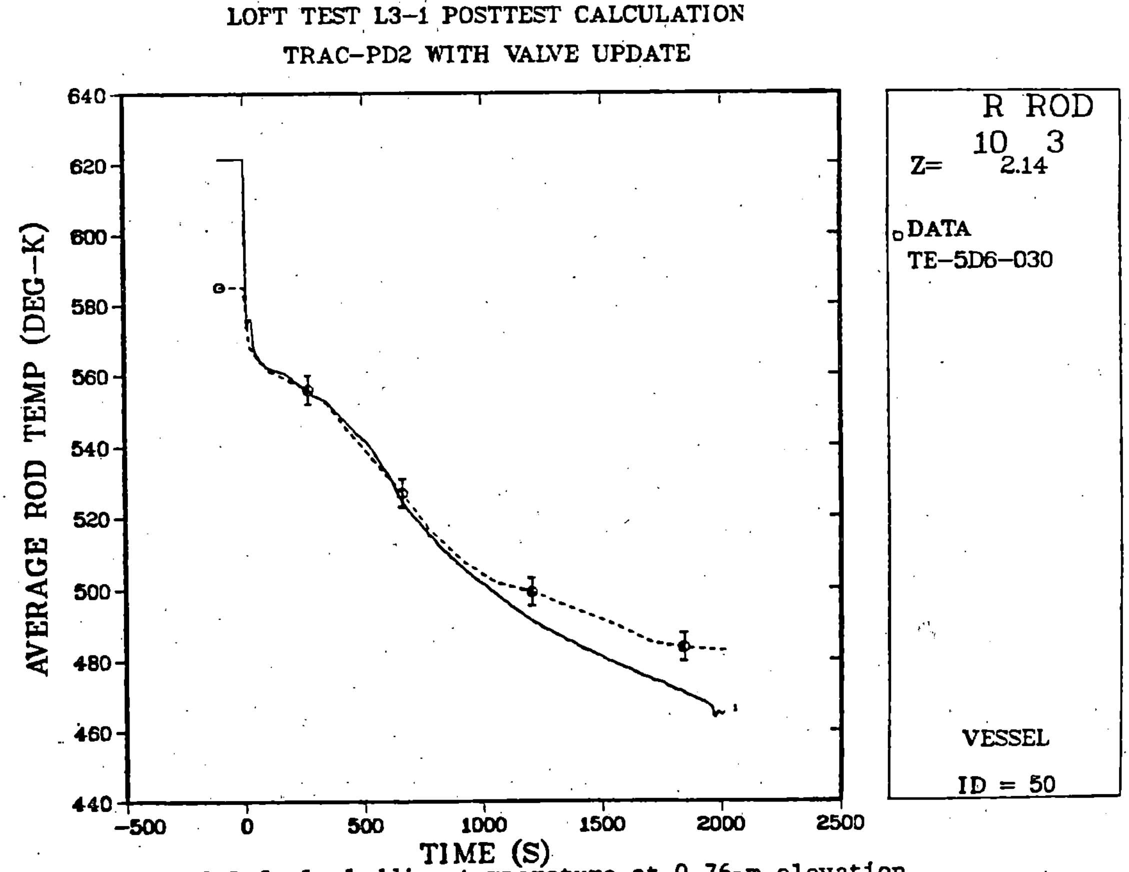
L3-1 steam line pressure.



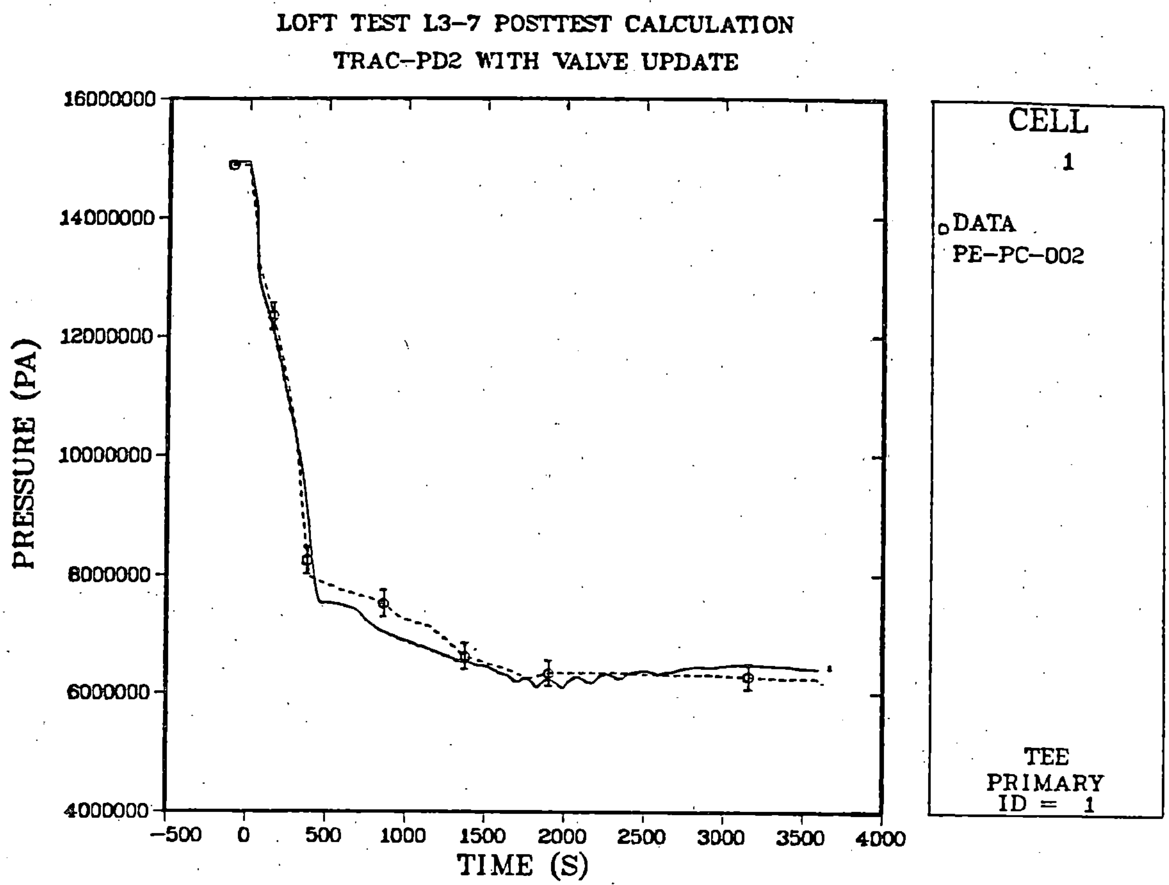
L3-1 break mass flow.



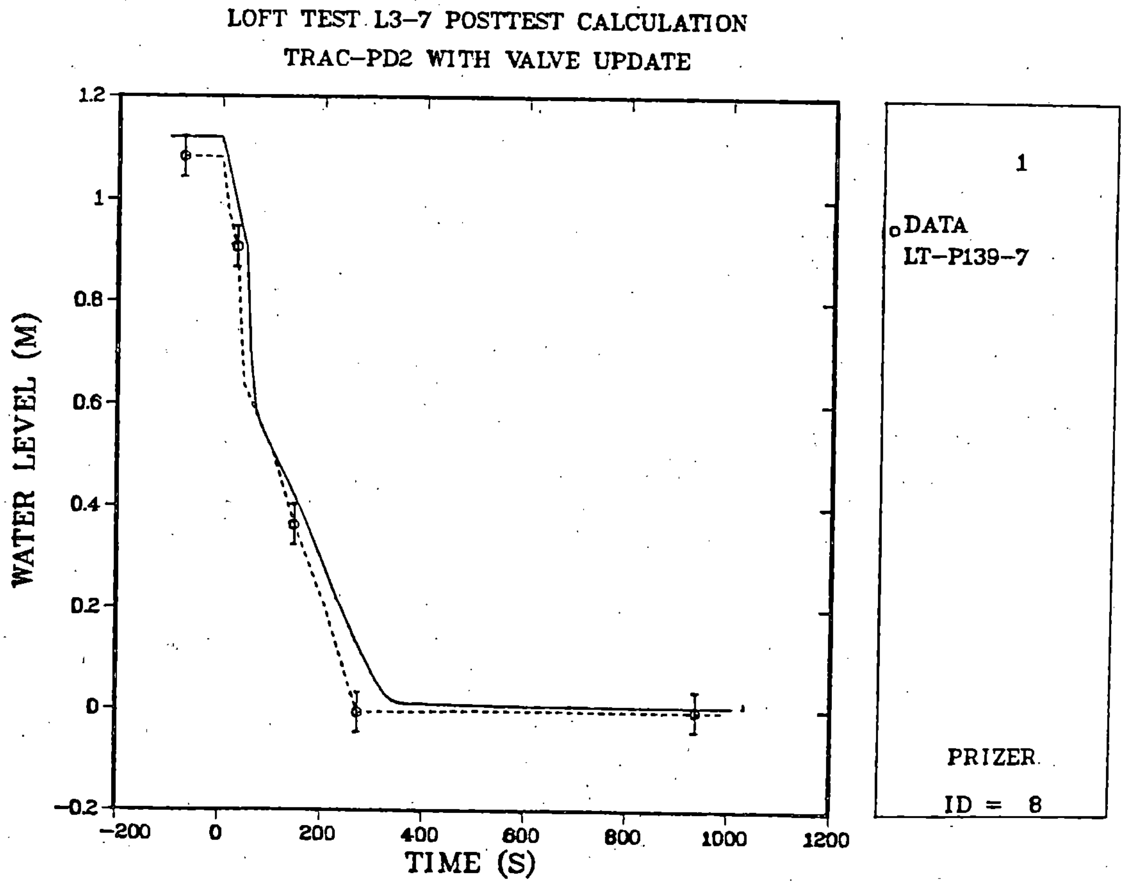
L3-1 core and upper plenum void fractions.



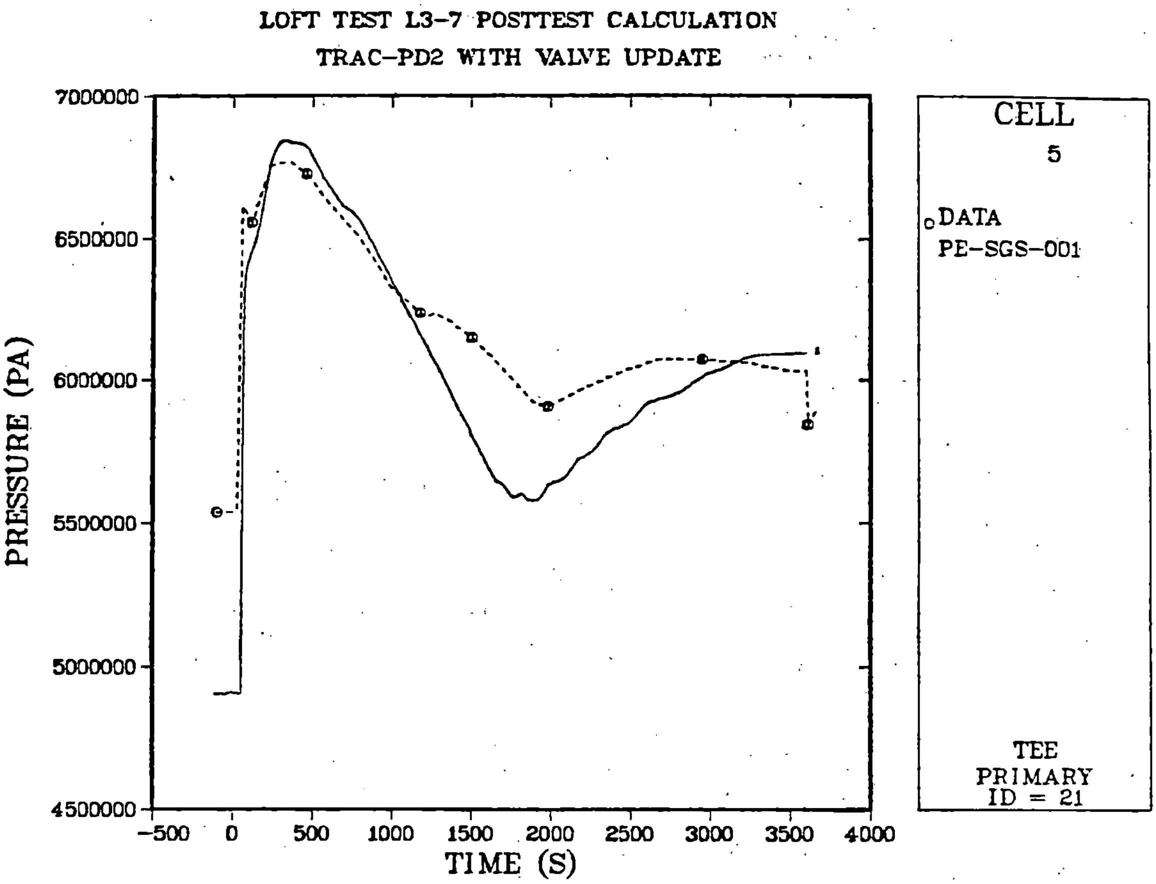
L3-1 fuel cladding temperature at 0.76-m elevation.



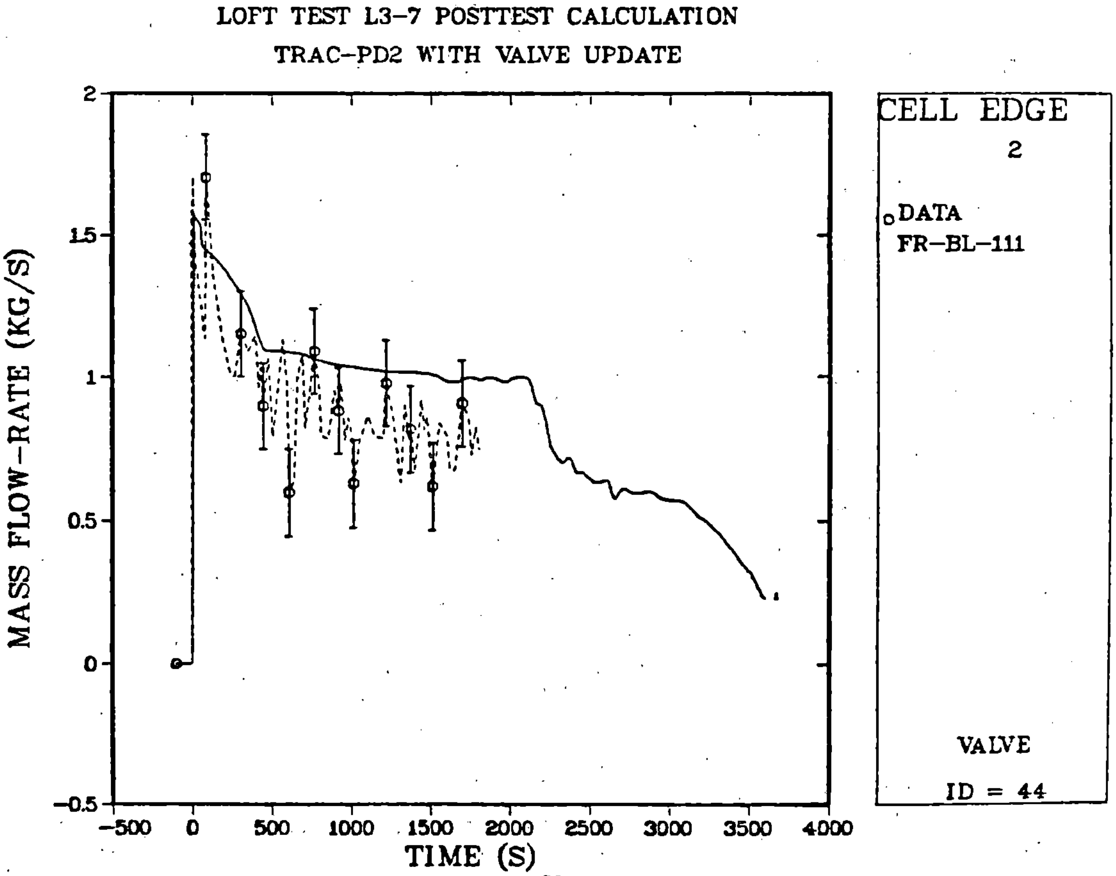
L3-7 intact loop hot-leg pressure.



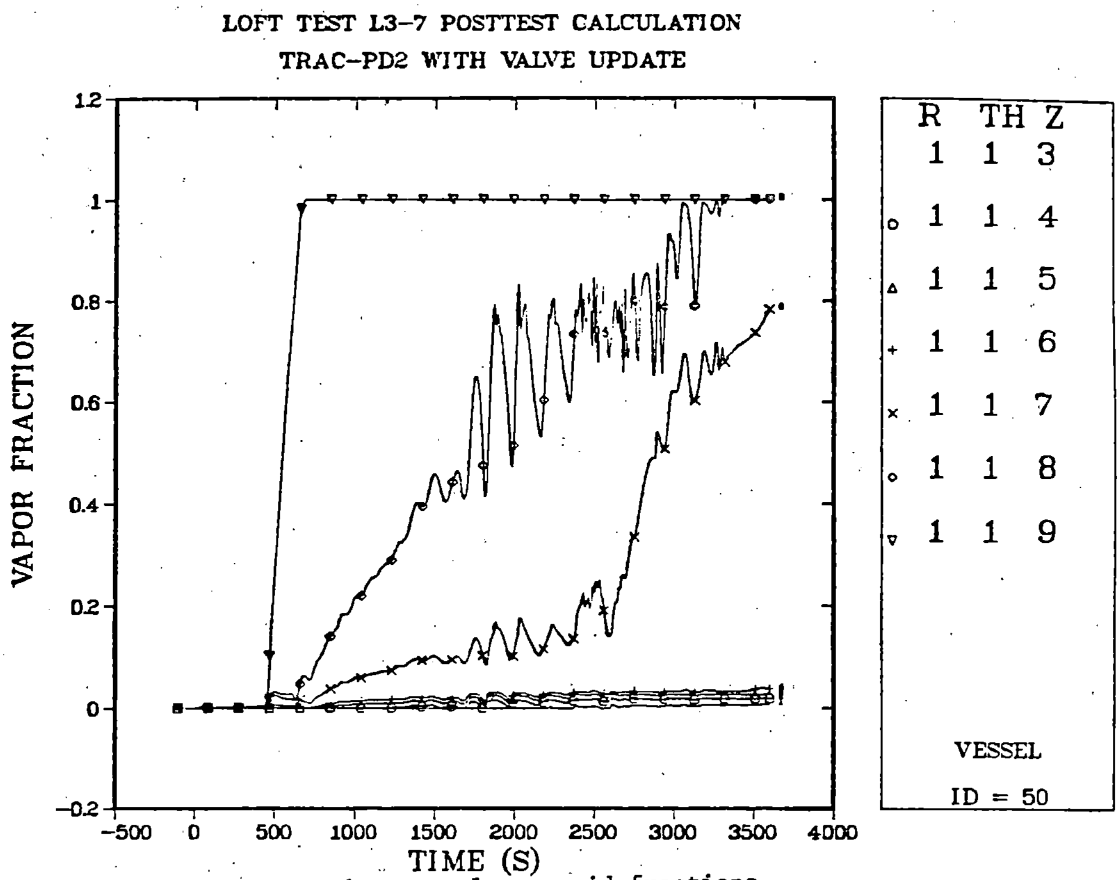
L3-7 pressurizer liquid level.



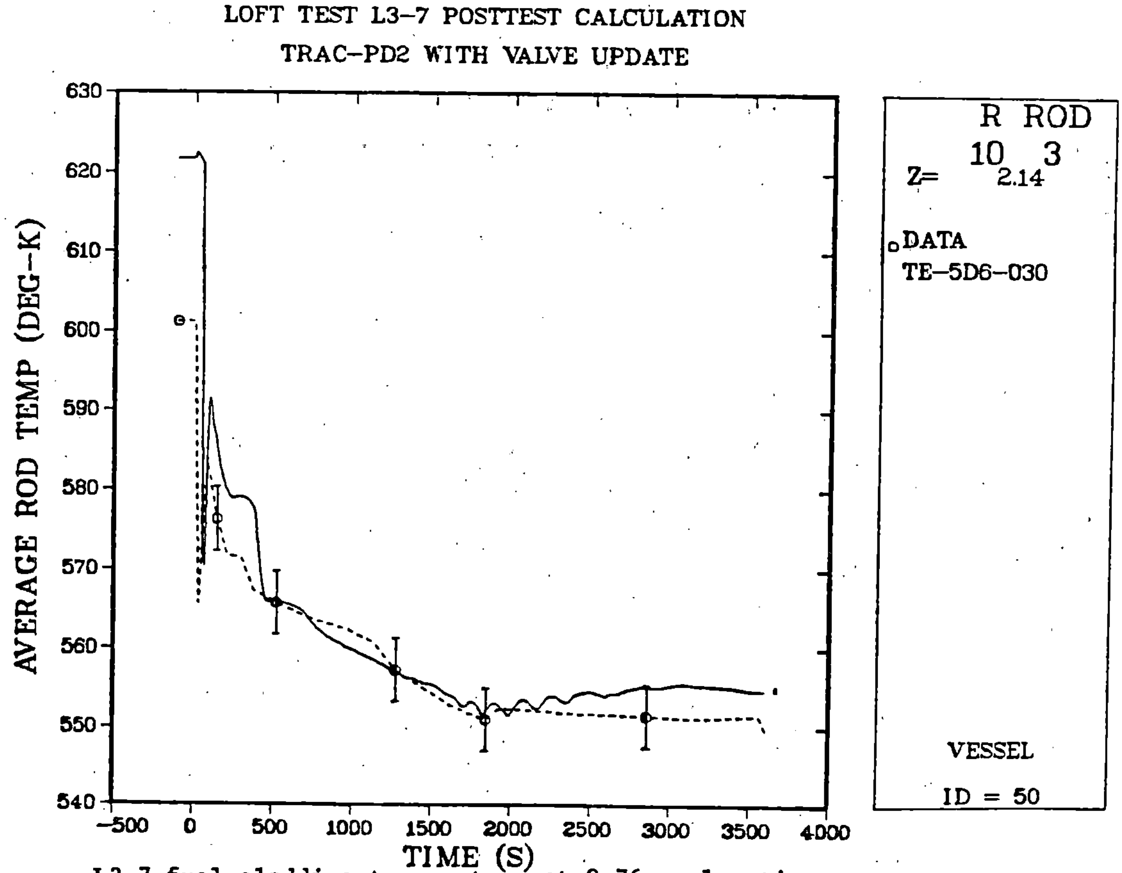
L3-7 steam-generator steam-dome pressure.



L3-7 break mass flow.

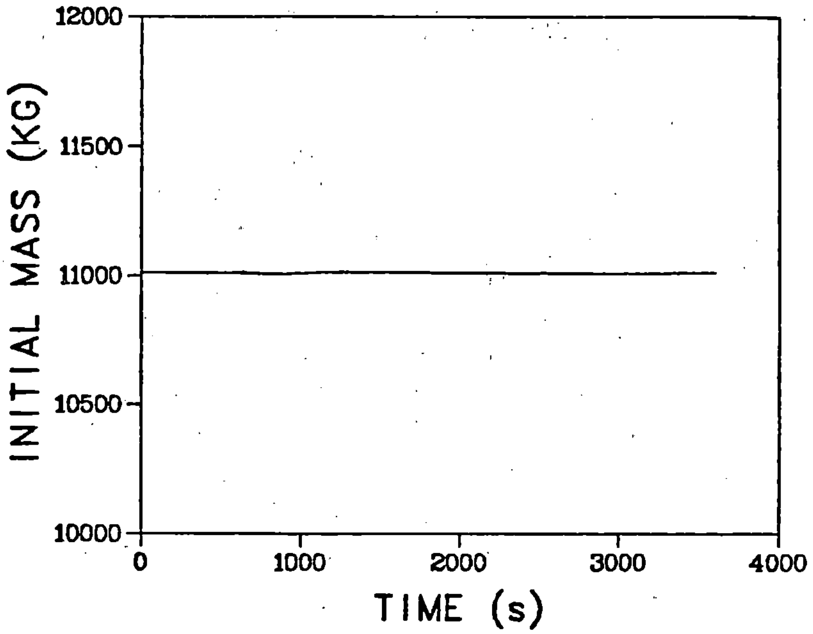


L3-7 core and upper plenum void fractions.



L3-7 fuel cladding temperature at 0.76-m elevation.

LOFT L3-7 POSTTEST



L3-7 computed system initial mass.

L3-1/L3-7 CONCLUSIONS

Comparisons of the calculation to thermal-hydraulic parameters are good

Method for calculating critical flow is adequate but cumbersome

New secondary system modeling is good

GENERAL CONCLUSIONS

TRAC-PD2 provides a useful small break modeling capability for predicting most thermal-hydraulic phenomena during slow transients

Improved critical flow modeling is desirable - critical flow model or improved constitutive relations

- Detailed definition of the system is necessary, including small leakage paths

ASSESSMENT OF TRAC-P1A USING CALCULATIONS FOR INTEGRAL FACILITIES

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

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EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

ASSESSMENT OF TRAC-P1A USING CALCULATIONS FOR INTEGRAL FACILITIES

A. C. Peterson
EG&G Idaho, Inc.

A calculation of the performance of the emergency safety features during a loss-of-coolant accident (LOCA) is required for each licensed pressurized water reactor (PWR). The TRAC-PIA computer code is one of a series of TRAC computer codes being developed for the NRC to provide best estimate calculational capability for both LOCA and non-LOCA analysis. To determine the capabilities and limitations of TRAC-P1A to calculate transient phenomena associated with LOCAs, comparisons of calculated results with data from scaled experimental facilities were performed. The capabilities and limitations were investigated further by performing calculations for seven LOCAs in a PWR. Selected results from these calculations are discussed in this paper.

TRAC-P1A calculations were performed for two integral experimental facilities of different scale: Semiscale Mod-1 and LOFT. Comparison of the calculated results with experimental data provided a basis for evaluating the capability of TRAC-P1A to calculate phenomema associated with LOCAs.

The results of the comparisons of the calculated and measured response were similar for each facility. A somewhat higher system pressure was calculated than was measured for the initial 5 s of the system depressurization. A lower system pressure was calculated than was measured from 5 to 25 s of the transient. From 25 s after the initiation of the experiments, the system pressure response was satisfactorily calculated. The difference between the calculated and measured pressure response during the early portion of the transient was partly a result of the calculation of too low a mass flow rate in the broken loop cold leg in each facility during the early period of the transient.

The comparison of the measured and experimental core thermal response was different for the two facilities during the blowdown and refill periods. The capability of TRAC-P1A to calculate the measured rod thermal response was dependent on the location in the core. The measured rod cladding temperature was satisfactorily calculated in Semiscale Mod-1 at locations where critical heat flux (CHF) occurred between 0.5 and 1.0 s. At core locations where CHF occurred after 2 s, the calculated rod cladding temperature was higher than the measured temperature. The rod thermal response for the LOFT facility was not satisfactorily calculated. A rewet of the entire LOFT nuclear core was measured at about 6 s after rupture. This core rewet was not calculated by TRAC-P1A.

TRAC-P1A calculations were performed for seven selected LOCAs in a PWR. The calculations consisted of: 200%, 0.25 m diameter, and 0.10 m diameter cold leg break; 200% hot leg break; 200% hot leg break simultaneous with the rupture of 16 steam generator tubes; and 200% cold leg break simultaneous with the rupture of 16 steam generator tubes, and with the rupture of 60 steam generator tubes. Since there are no data to compare directly with these calculations and no criteria for code acceptance, the capability of TRAC-P1A was judged from the consistency of the calculations and a qualitative interpretation of similar experiments in scaled facilities.

The results of these calculations indicate that for large hot and cold leg breaks the calculated system hydraulic response was reasonable. For large hot and cold leg breaks with steam generator tube ruptures, the calculated effects were consistent with expected results. For an intermediate cold leg break, the calculated system response was reasonable. For a small cold leg break, excessive stratification of the secondary side of the steam generator can occur, which strongly influence the results.

In summary, the dominant hydraulic phenomena during a LOCA were satisfactorily calculated by TRAC-P1A, however, significant differences between some measured and calculated local phenomena occurred.

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ASSESSMENT OF TRAC-P1A FROM CALCULATIONS FOR INTEGRAL FACILITIES

by
A.C. PETERSON

EG&G Idaho, Inc.

TRAC-P1A ASSESSMENT

LICENSING AND SAFETY ISSUE

CAPABILITY TO CALCULATE PERFORMANCE OF EMERGENCY SAFETY FEATURES DURING A LOCA

ACP-1

OUTLINE

- COMPARISON OF CALCULATIONS WITH DATA FROM INTEGRAL EXPERIMENTS
- RESULTS OF LOCA CALCULATIONS FOR PWRs
- CONCLUSIONS

INTEGRAL EXPERIMENT DATA COMPARISONS

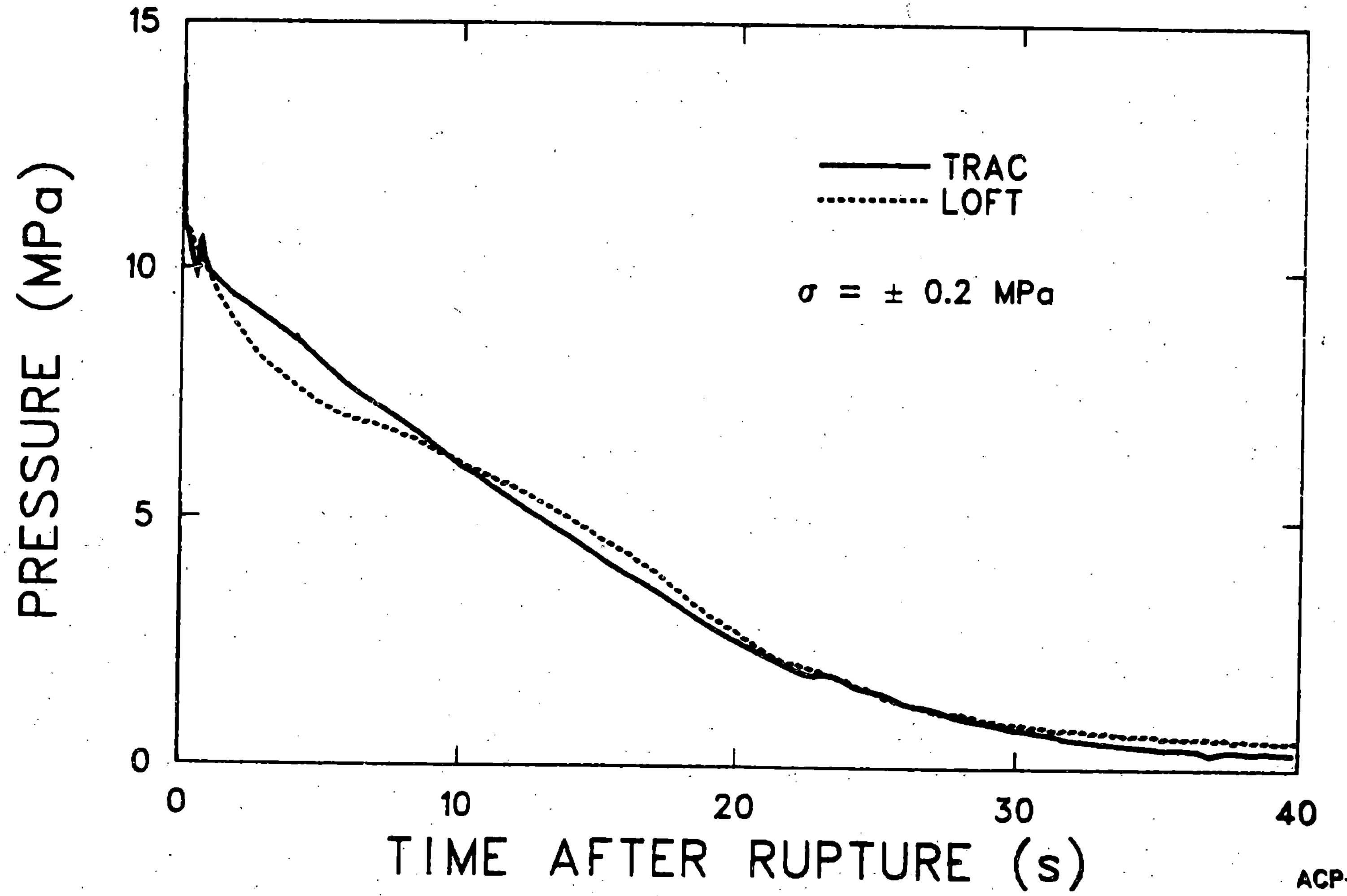
- LOFT TEST L2-3
- SEMISCALE MOD-1 TEST S-04-6

ACP-3

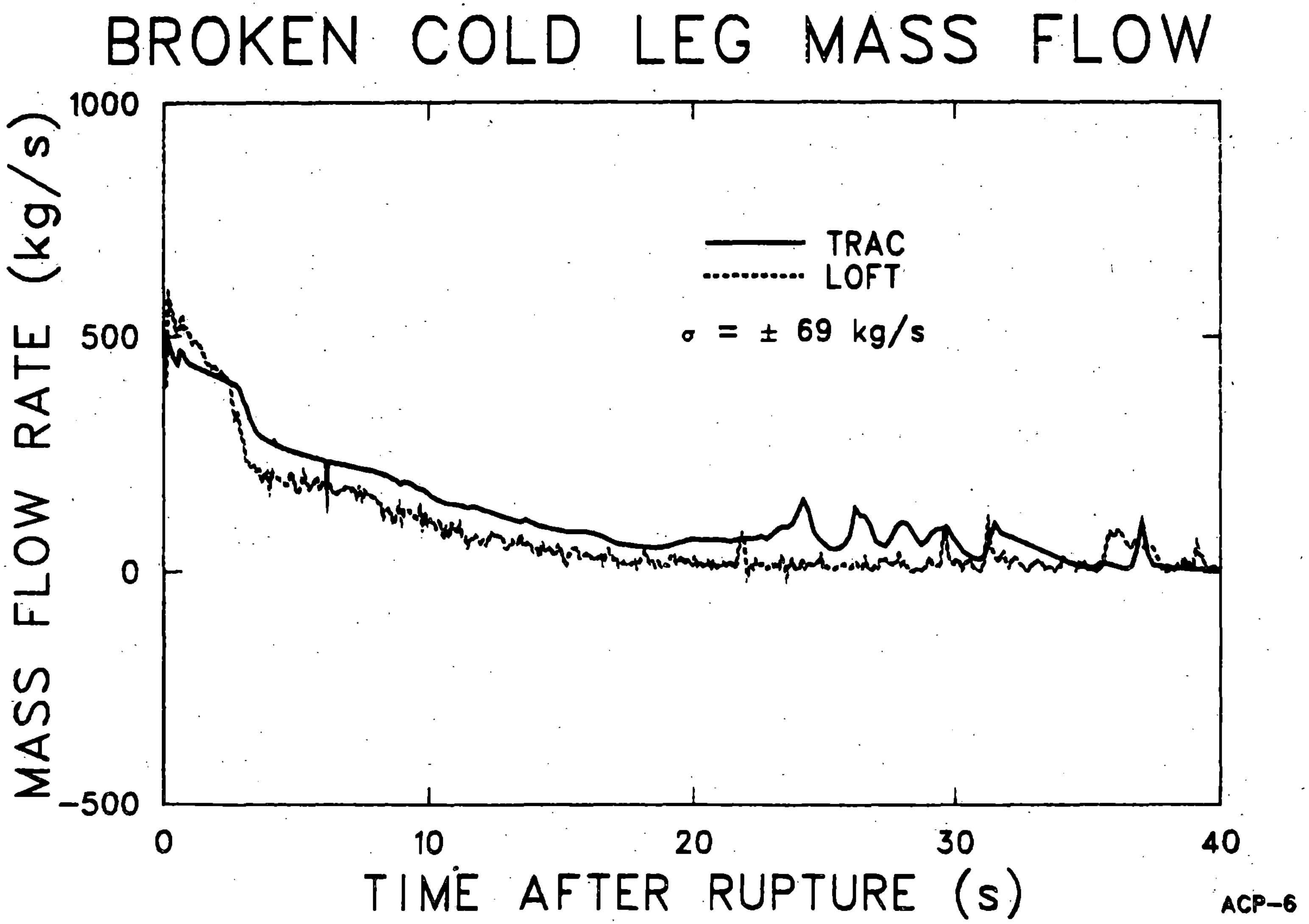
SUMMARY OF LOFT TRAC MODEL

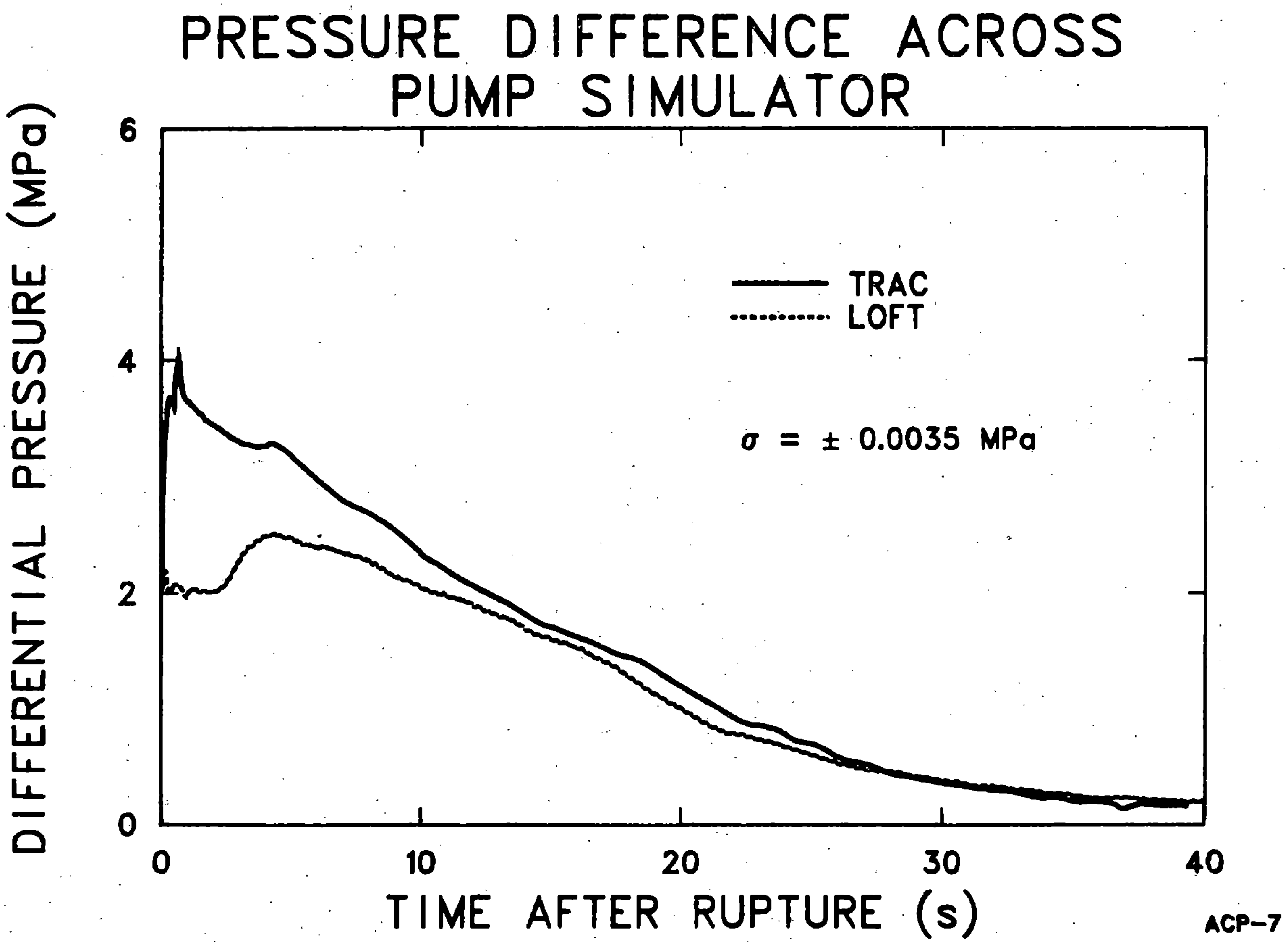
LOOPS	2
VESSEL RINGS	5
AZIMUTHAL SECTIONS	4
VESSEL CELLS	280
COMPONENTS	27
TOTAL CELLS	444

SYSTEM PRESSURE RESPONSE

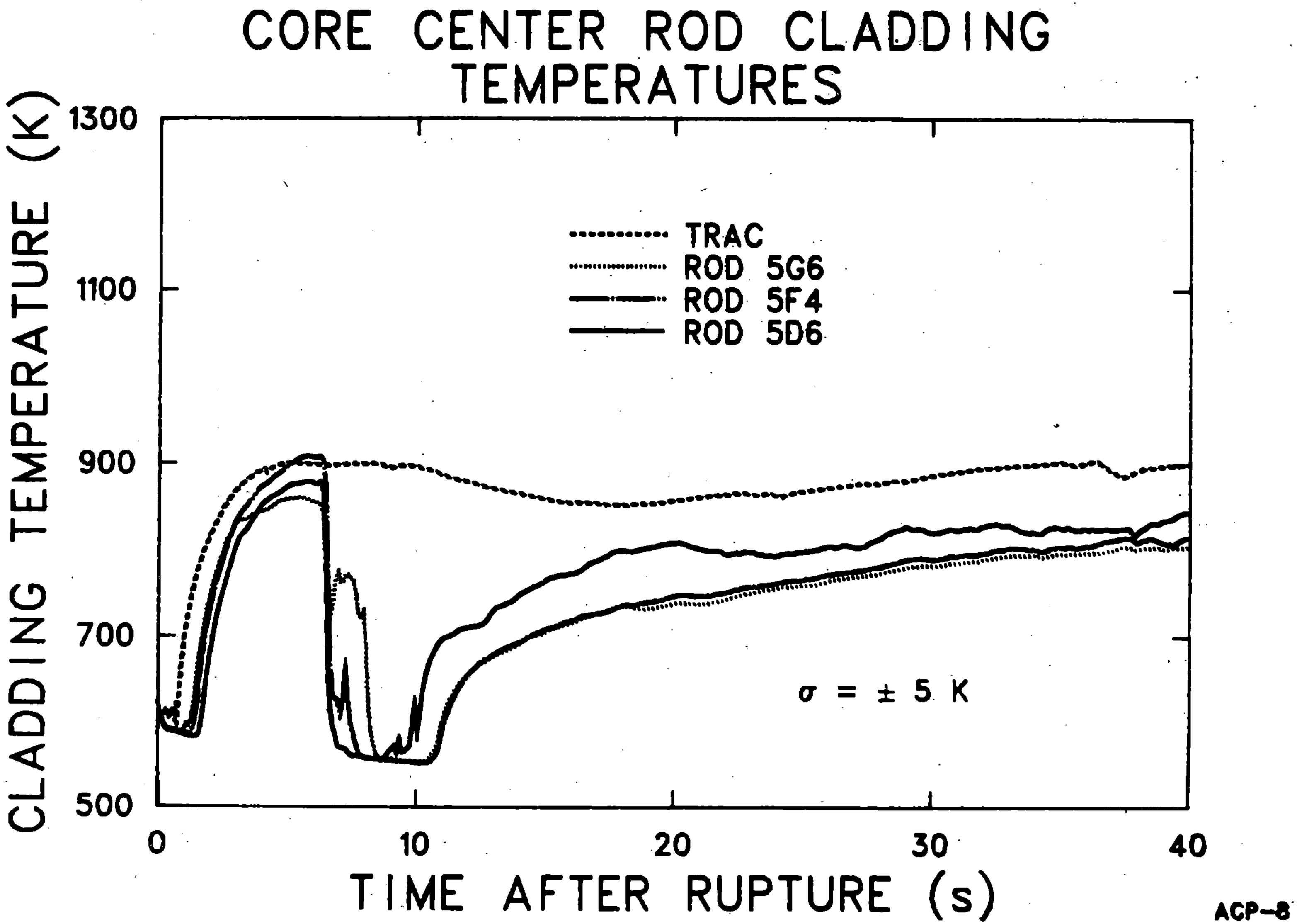


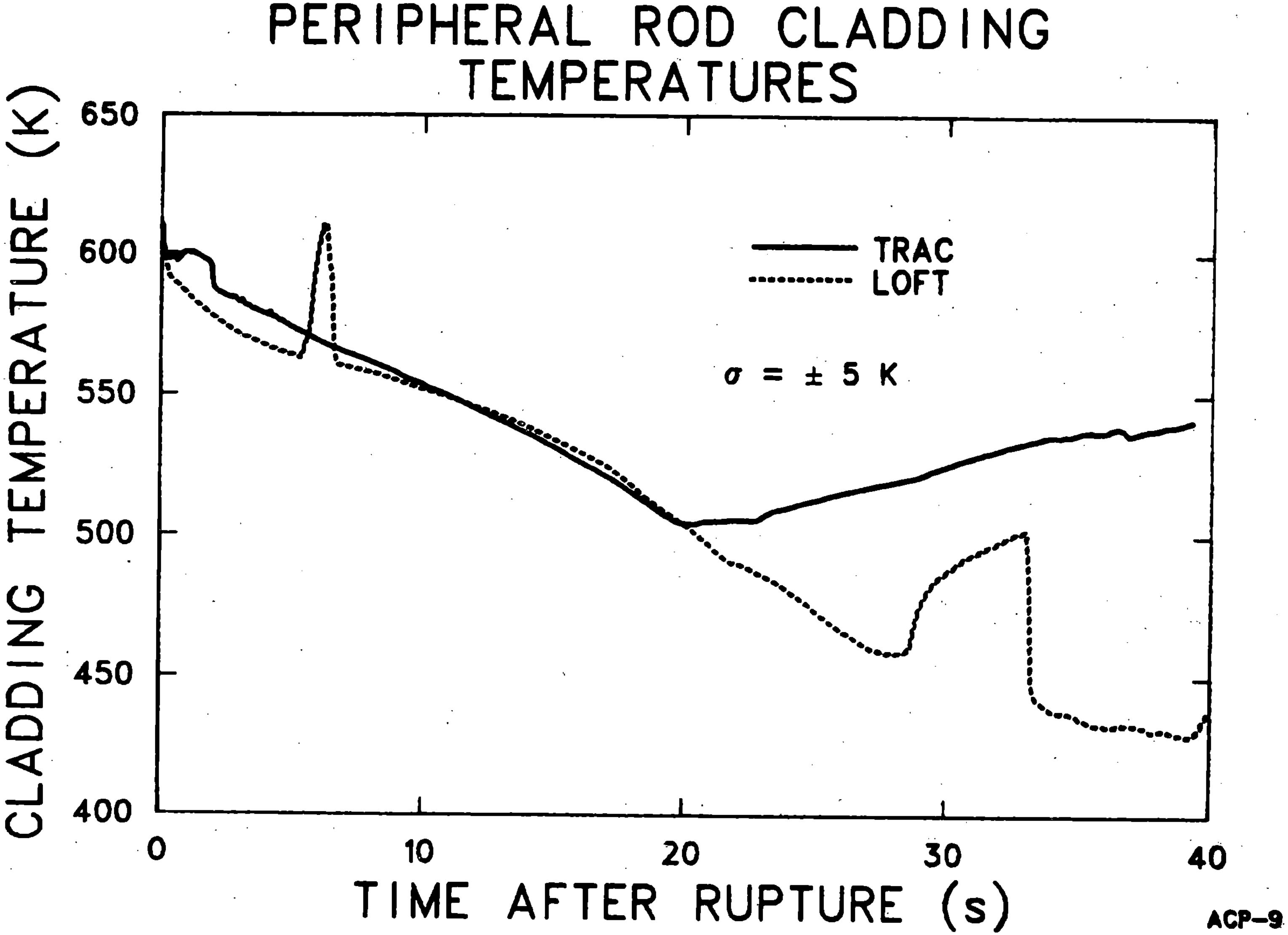
ACP-5


BROKEN COLD LEG MASS FLOW
MASS FLOW RATE (kg/s)
1000
500
0
−500
0
10
20
30
40
TIME AFTER RUPTURE (s)
TRAC
LOFT
σ = ± 69 kg/s
ACP-6

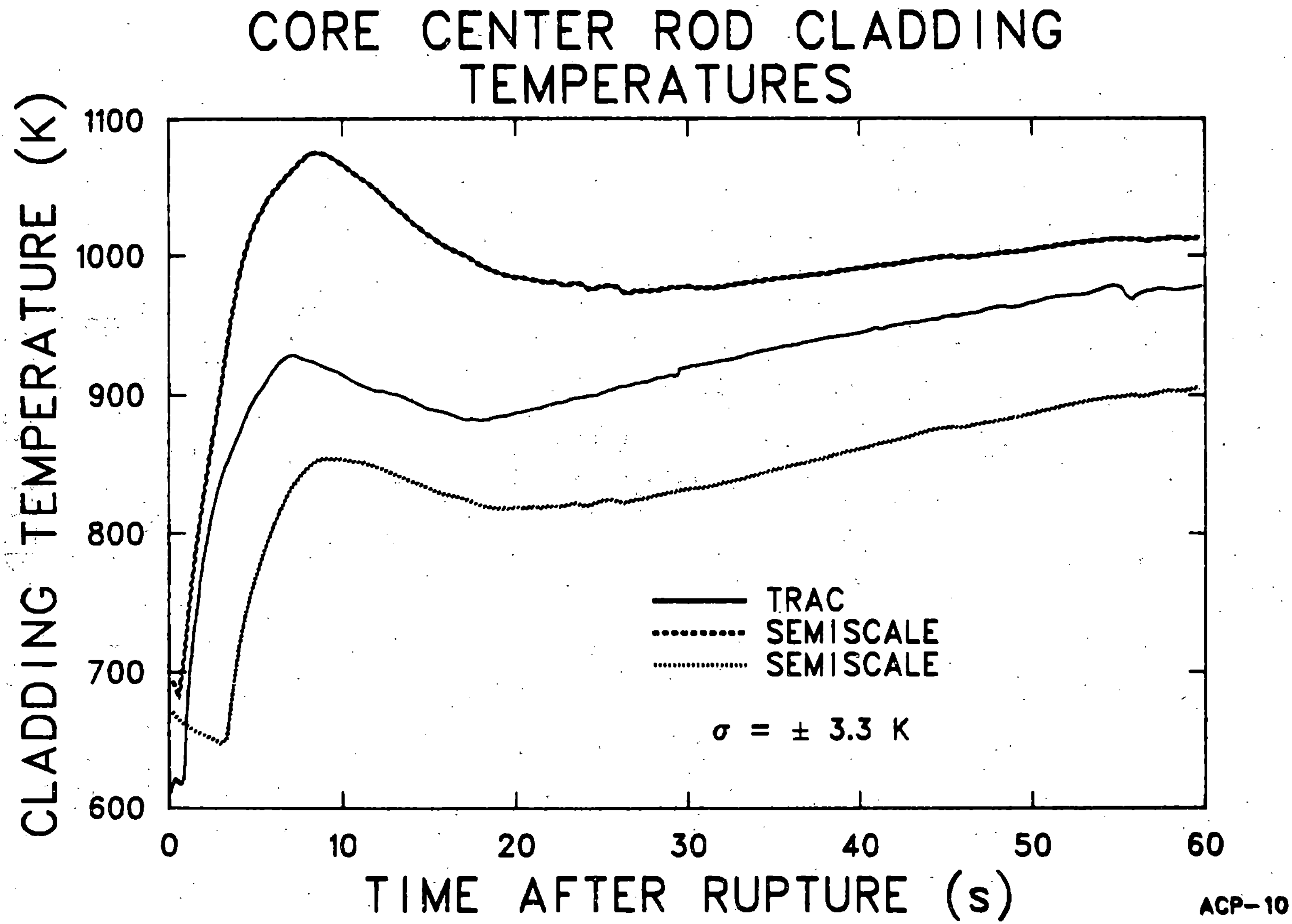


PRESSURE DIFFERENCE ACROSS
PUMP SIMULATOR
DIFFERENTIAL PRESSURE (MPa)
TIME AFTER RUPTURE (s)
TRAC
LOFT
σ = ± 0.0035 MPa
0
2
4
6
0
10
20
30
40
ACP-7

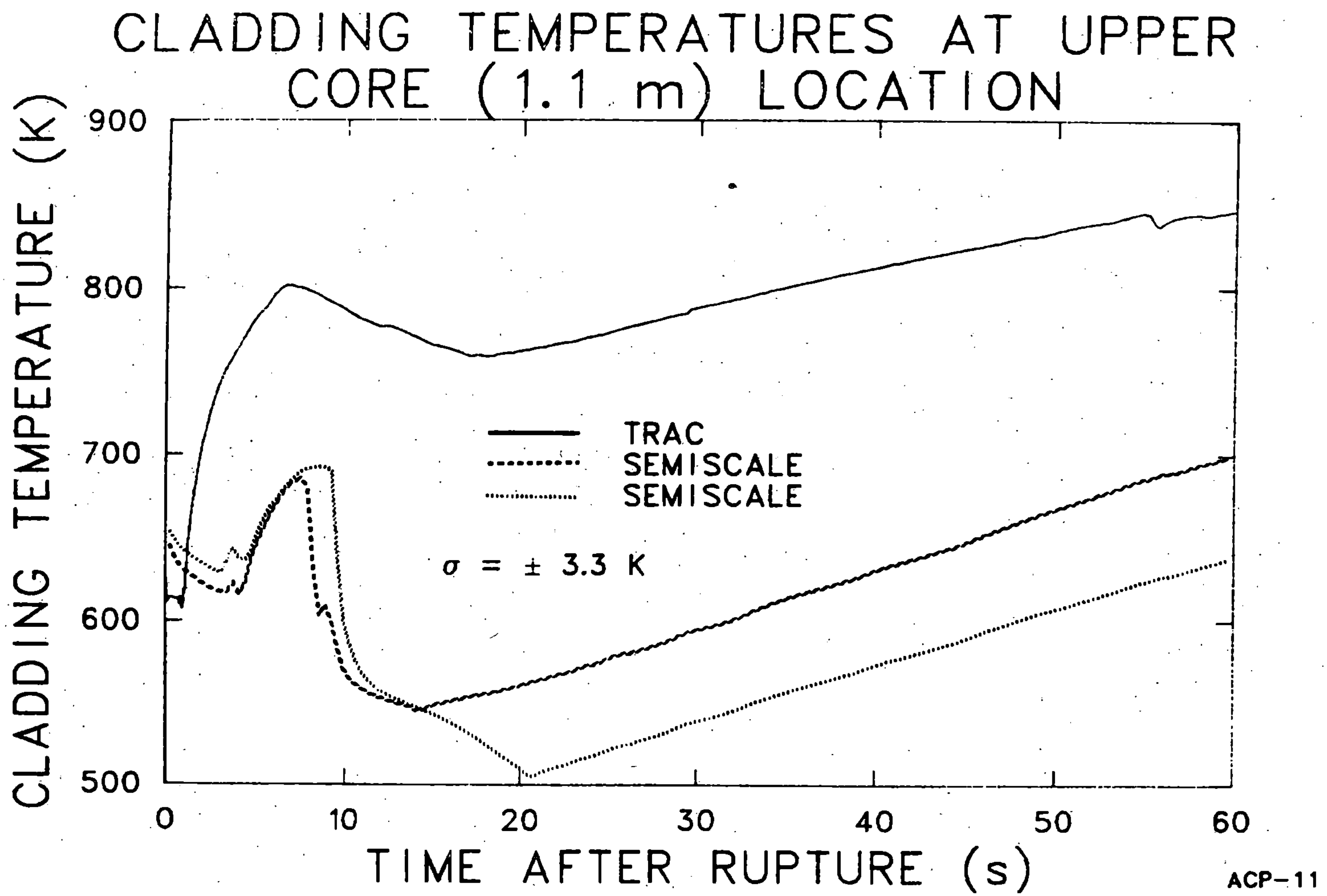

CORE CENTER ROD CLADDING TEMPERATURES
CLADDING TEMPERATURE (K)
1300
1100
900
700
500
0
10
20
30
40
TIME AFTER RUPTURE (s)
TRAC
ROD 5G6
ROD 5F4
ROD 5D6
σ = ± 5 K
ACP-8



PERIPHERAL ROD CLADDING TEMPERATURES
CLADDING TEMPERATURE (K)
650
600
550
500
450
400
TRAC
LOFT
σ = ± 5 K
0
10
20
30
40
TIME AFTER RUPTURE (s)
ACP-9



CORE CENTER ROD CLADDING TEMPERATURES
CLADDING TEMPERATURE (K)
1100
1000
900
800
700
600
0
10
20
30
40
50
60
TIME AFTER RUPTURE (s)
TRAC
SEMISCALE
SEMISCALE
σ = ± 3.3 K
ACP-10



CLADDING TEMPERATURES AT UPPER
CORE (1.1 m) LOCATION
CLADDING TEMPERATURE (K)
900
800
700
600
500
0
10
20
30
40
50
60
TIME AFTER RUPTURE (s)
TRAC
SEMISCALE
SEMISCALE
σ = ± 3.3 K
ACP-11

SUMMARY OF TRAC-P1A PWR CALCULATIONS

CALCULATION	TRANSIENT TIME (s)
200% COLD LEG BREAK	207
0.25 m COLD LEG BREAK	212
0.10 m COLD LEG BREAK	1224 850

ACP-12

SUMMARY OF TRAC-P1A PWR CALCULATIONS

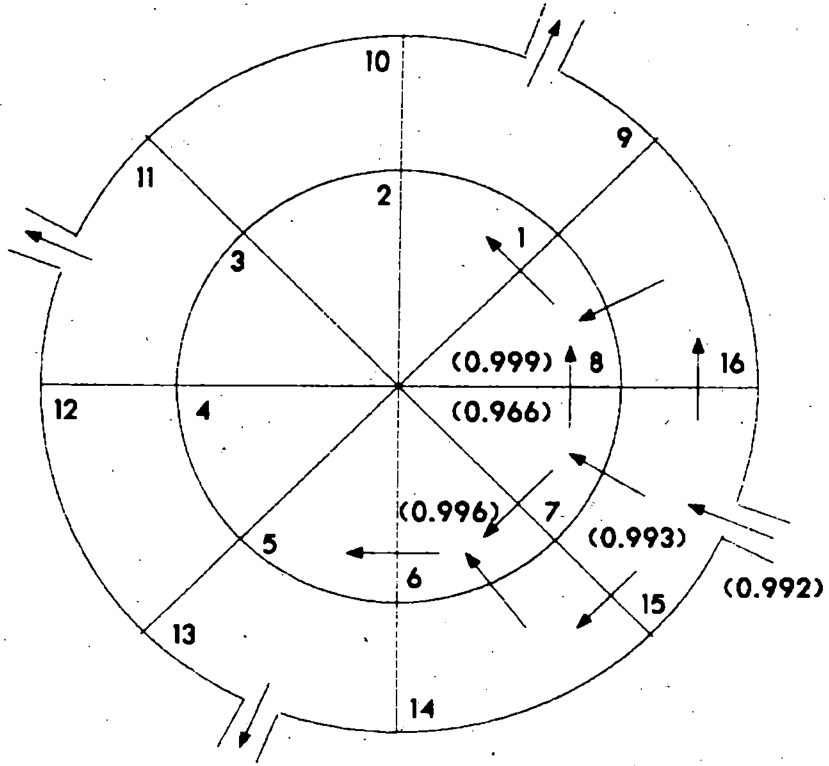
CALCULATION	TRANSIENT TIME (s)
200% COLD LEG BREAK, 16 RUPTURED S/G TUBES	150
200% COLD LEG BREAK, 60 RUPTURED S/G TUBES	148
200% HOT LEG BREAK	29
200% HOT LEG BREAK, 16 RUPTURED S/G TUBES	25

ACP-12A

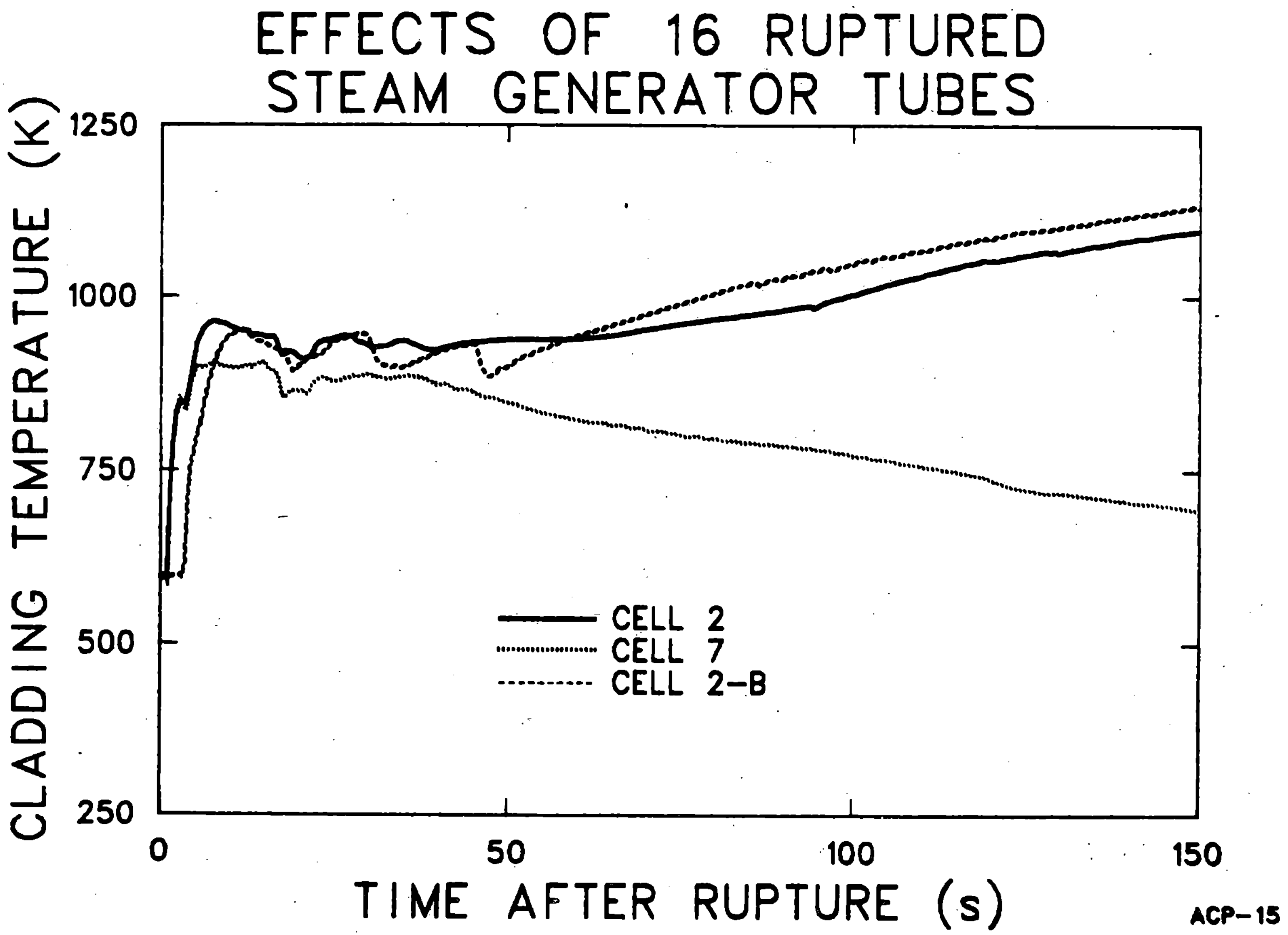
SUMMARY OF DETAILED MODEL

LOOPS	4
VESSEL RINGS	3
THETA SECTIONS	8
VESSEL CELLS	288
COMPONENTS	57
TOTAL CELLS	550

UPPER PLENUM VELOCITIES AND VOID FRACTIONS



APC-14

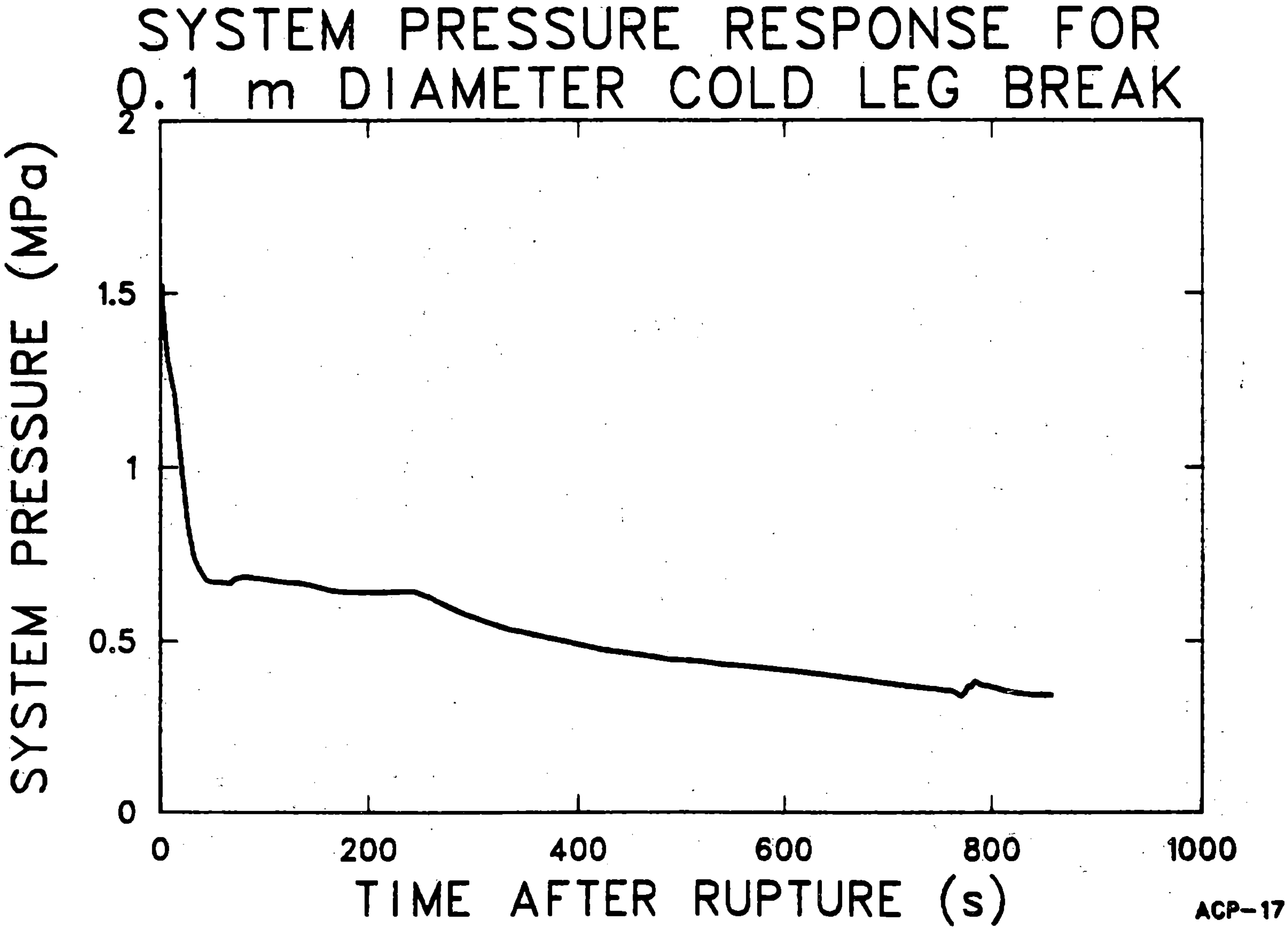


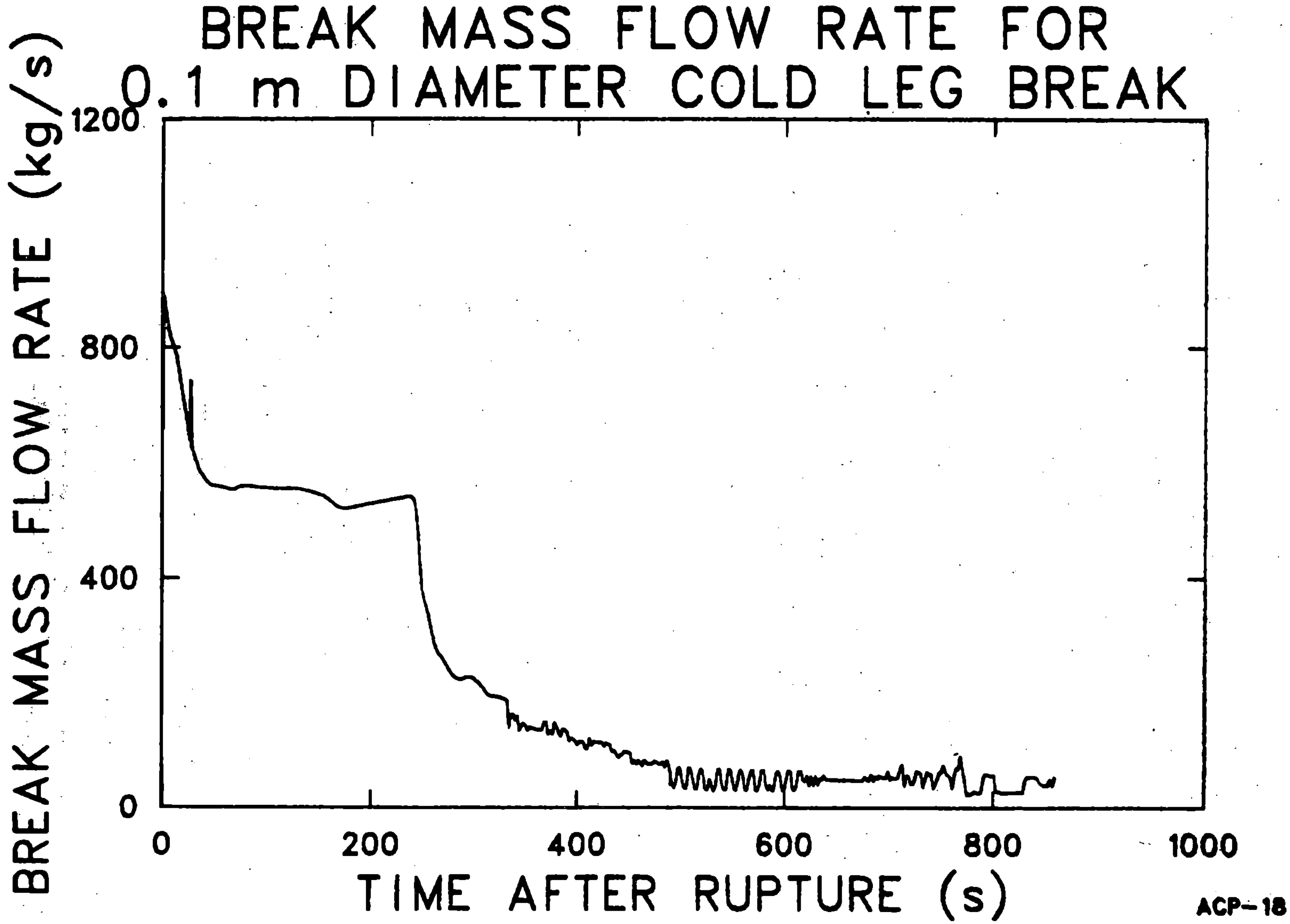
EFFECTS OF 16 RUPTURED
STEAM GENERATOR TUBES
CLADDING TEMPERATURE (K)
1250
1000
750
500
250
0
50
100
150
TIME AFTER RUPTURE (s)
CELL 2
CELL 7
CELL 2-B
ACP-15

SUMMARY OF COARSE MODEL FOR SMALLER BREAKS

LOOPS	2
VESSEL RINGS	2
THETA SECTIONS	2
VESSEL CELLS	72
TOTAL COMPONENTS	32
TOTAL CELLS	208

ACP-16


SYSTEM PRESSURE RESPONSE FOR
0.1 m DIAMETER COLD LEG BREAK
SYSTEM PRESSURE (MPa)
2
1.5
1
0.5
0
0
200
400
600
800
1000
TIME AFTER RUPTURE (s)
ACP-17



BREAK MASS FLOW RATE FOR
0.1 m DIAMETER COLD LEG BREAK
BREAK MASS FLOW RATE (kg/s)
1200
800
400
0
0
200
400
600
800
1000
TIME AFTER RUPTURE (s)
ACP-18

CONCLUSIONS

- OVERALL SYSTEM RESPONSE WAS CALCULATED SATISFACTORILY
- BREAK MASS FLOW RATE WAS NOT CALCULATED ACCURATELY
- CAPABILITY TO CALCULATE CORE THERMAL RESPONSE DEPENDENT ON FACILITY

CONCLUSIONS (CONTD.)

- OVERALL SYSTEM HYDRAULIC RESPONSE FOR LARGE HOT AND COLD LEG BREAKS CALCULATED SATISFACTORILY
- SOME HYDRAULIC EFFECTS CALCULATED SATISFACTORILY FOR LARGE PIPE BREAKS WITH RUPTURED STEAM GENERATOR TUBES
- OVERALL SYSTEM HYDRAULIC RESPONSE CAN BE SATISFACTORILY CALCULATED FOR SMALL PIPE BREAKS

CODE (TRAC-P1A) ASSESSMENT RESULTS AT BNL

by

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Gaithersburg, Maryland

Summary

The primary objective of the code assessment activity at Brookhaven National Laboratory is to verify the TRAC-P1A and the future released versions of TRAC with basic and separate-effects thermal-hydraulic tests. Emphasis is placed on identifying the deficiencies, if any, in the basic thermal-hydraulic models and correlations that are used in TRAC for the wall friction, wall heat transfer, interfacial momentum transfer, and the interfacial heat and mass transfer. The final objective is, of course, to recommend improved models and/or correlations for the future versions of TRAC, where deemed necessary.

A number of basic and separate-effects experiments have been simulated with TRAC-P1A at BNL during the last fiscal year. They can be grouped in the following categories:

1. One-dimensional steady-state experiments which are simulated by the one-dimensional drift-flux formulation of TRAC-P1A. These are: (a) Moby-Dick nitrogen-water tests [1], (b) BNL flashing flow tests [2], and KFK-IRE nozzle flow tests[3]. Attempts were made to simulate two other experiments--namely, the Moby-Dick steam-water tests[4] and the University of Houston flooding tests [5]. However, in both cases, the code failed to converge to a steady-state.

2. One-dimensional transient experiments, also simulated by the one-dimensional drift-flux formulation of TRAC-P1A. These are: (a) Super-CANON tests [6], (b) Marviken critical flow tests [7], and (c) Battelle-Frankfurt top blowdown test [8].

3. Multi-dimensional steady-state experiments--namely, (a) RPI two-dimensional phase separation tests [9], and (b) FRIGG loop tests [10].

We shall now summarize the results obtained at BNL during FY 1980.

1. One-Dimensional Steady-State Experiments

Both the Moby-Dick nitrogen-water tests and the KFK-IRE nozzle flow tests indicated that the TRAC-P1A predictions for mass flow rates are quite sensitive to the two-phase friction factor options available in TRAC-P1A. Further examination revealed that the use of one single-phase friction factor correlation, rather than two different correlations as available in TRAC, reduces this sensitivity significantly. Therefore, it is recommended that only the Colebrook-type single phase friction factor be retained with the relative wall roughness parameter as an input.

Simulation of the BNL flashing flow tests in a converging-diverging nozzle indicates that the TRAC-P1A underpredicts the mass flow rates significantly for low flow rates with subcooled water inlet condition. The reason for this discrepancy is believed to be the lack of a nucleation or flashing delay model in TRAC at the inception point of flashing.

The relative velocity for the horizontal pipes in TRAC-P1A is specified by a constant value, 1.1, for the void distribution parameter, C_o. However,

this does not appear to be correct since an air-water experiment in the KFK-IRE horizontal nozzle indicates that the void distribution parameter decreases along the length of the test section--i.e., the flow becomes more homogeneous downstream of the throat.

Sometimes, the TRAC-P1A code does not converge to a steady-state for flows in vertical channels with high void fractions ($\alpha \geq 0.75$). This is due to the large difference in relative velocity between the slug and the annular flow regimes. Thus the code failed to converge for the high void fraction cases of the Moby-Dick nitrogen-water and steam-water experiments. Also the slip correlation for the annular flow regime cannot take into account the counter-current flow situations; and, therefore, the University of Houston flooding tests could not be simulated by the one-dimensional drift-flux formulation of TRAC-P1A.

2. One-Dimensional Transient Experiments

Three different blowdown experiments have been simulated with the one-dimensional drift-flux formulation of TRAC-P1A. These are: (a) blowdown of initially subcooled high pressure water from a horizontal pipe--Super-CANON experiments, (b) blowdown of water and two-phase mixtures through the bottom of a large vessel fitted with large diameter nozzles of varying length-to-diameter ratios--Marviken experiments, and (c) blowdown of steam and two-phase mixtures through the top of a vessel--Battelle-Frankfurt test. The TRAC-P1A predictions for all of these tests are in qualitative agreement with the data. However, there are a number of quantitative discrepancies which need further discussion.

For the Super-CANON tests, TRAC-P1A overpredicts the pressure during the early part of the transient. Thereafter, the calculated pressure drops rather sharply, and it underpredicts the experimental data. It can, therefore, be inferred that the code tends to overpredict the discharge flow rate, at least during the early part of the transient. Inclusion of a nucleation delay model--i.e., the model of Alamgir and Lienhard [11], did not improve the code prediction because after an initial dip, the pressure recovered to the original code value within a millisecond. However, a sensitivity study, with the void distribution parameter C_0 and the initial water temperature as parameters, showed the direction for achieving improved agreement.

For the Marviken tests simulated at BNL, two different nozzle diameters (300 and 500 mm) with varying length-to-diameter ratios (1.7 to 0.33) were used. The code tends to underpredict the discharge flow rates more as the nozzle length-to-diameter ratio is decreased. No clear influence of the initial water subcooling was found in the discrepancies between the measured and the predicted discharge flow rates. However, the error in the predicted pressure at the early stage of the transient, due to the lack of a delayed nucleation or flashing model, increased as the initial water subcooling was increased. Inclusion of the Alamgir-Lienhard correlation for delayed flashing improved the early stage pressure prediction considerably; however, the effect disappeared after a few seconds, and the predictions for the flow rate did not improve.

For the Battelle-Frankfurt top blowdown test, steam is discharged during the early part of the transient. A two-phase mixture level swells during this period. When this mixture level reaches the discharge nozzle, a jump in the discharge flow rate is observed due to the sudden increase in the fluid density. The TRAC-P1A underpredicts the time of arrival of the mixture level at the discharge nozzle. In other words, according to the TRAC calculation, the mixture level swells faster than in the experiment. An increase in the TRAC relative velocity--i.e., an increase in the degree of phase separation, produced results in closer agreement with the data. However as expected, TRAC-P1A could not predict the pressure dip observed during the early stage of the transient.

Based on the above discussion, it is clear that further examination of the TRAC models and the constraints[12] placed in the code for the non-equilibrium phase change and relative velocity, is required to explain the discrepancies between the TRAC-P1A predictions and the experimental data. Inclusion of a delayed flashing model is not enough.

3. Multi-Dimensional Steady-State Experiments

Simulation of the RPI two-dimensional air-water tests with or without rods was not successful with the 2-D version of the vessel module of TRAC-P1A because of the code's inability to reach a steady-state. For three out of four tests with one outlet the code stopped because of indefinitive or overflow conditions. For the remaining case, the code reached a condition close to a steady-state. For the tests with two outlets, the code reached a stable, but not steady-state condition. Moreover, the solution was asymmetric, although symmetric boundary conditions were imposed. Even the run with all air flow did not produce a symmetric solution, and an additional large dummy cell at the left hand side of the 2-D slab produced an oscillatory solution. Based on these results, it may be concluded that in the 2-D option of TRAC-P1A there might be some errors during transformation from the cylindrical (3-D) to the slab (2-D) geometry.

The heated rod bundle test in the FRIGG loop was first attempted with the three-dimensional vessel module of TRAC-P1A. However, the code was unable to reach a steady-state even though a wide range of time steps and convergence criteria was attempted. The one-dimensional option of the vessel module was then tried without success. A number of scoping runs with air-water flows and with one-dimensional drift-flux formulation led to the belief that in the two-fluid formulation the fluctuations in the relative velocity calculation caused the oscillations in the vapor generation, void fraction, and eventually the pressure calculations. To alleviate the problem, the relative velocity used for the vapor generation calculation after an arbitrarily large number of time steps, say N_0, was taken to be an average value of relative velocity--i.e., $\overline{V}_r = \sum_{i=N_0}^{n} V_{r,i} \Big/ (n - N_0 + 1)$, rather than just the current value as was done in TRAC-P1A. This modification resulted in a steady-state solution for a FRIGG rod bundle test using the one-dimensional two-fluid formulation of TRAC. However, the agreement between the calculation and the data is poor because of (i) the lack of subcooled boiling model in the code and (ii) a lower rate of vapor generation in the two-fluid model.

Finally, it can be concluded that further improvements are needed in the TRAC-P1A models for the interfacial area, heat, mass and momentum transfer. In addition, a delayed flashing correlation should be added, and improved relative velocity correlations for both the horizontal and the vertical channels are needed for the one-dimensional drift-flux formulation of the code.

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Saha, P., and Sanborn, Y., "Independent Assessment of TRAC-P1A with Super-CANON Blowdown Tests," to be presented at the ANS/ENS International Conf., Washington, D.C., November 17-21, 1980.

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CODE (TRAC-P1A) ASSESSMENT RESULTS AT BNL

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EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING
OCTOBER 27 - 31, 1980
GAITHERSBURG, MARYLAND

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

OBJECTIVES

- INDEPENDENTLY ASSESS THE RELEASED VERSION OF TRAC FOR SPECIFIC PHENOMENA, SUCH AS
  - CRITICAL FLOW
  - ECC BY-PASS
  - REFLOOD

  AS APPLIED TO LARGE AND SMALL BREAK LOCA'S

- SIMULATE BASIC AND SEPARATE-EFFECTS TESTS WITH TRAC

- IDENTIFY THE DEFICIENCIES (IF ANY) IN THE BASIC THERMAL-HYDRAULIC MODELS AND CORRELATIONS USED IN TRAC
  - WALL FRICTION
  - WALL HEAT TRANSFER
  - INTERFACIAL MOMENTUM TRANSFER
  - INTERFACIAL HEAT AND MASS TRANSFER

- RECOMMEND IMPROVED MODELS AND/OR CORRELATIONS FOR FUTURE VERSIONS OF TRAC

ASSESSMENT CALCULATIONS PERFORMED IN FY 1980 WITH TRAC-P1A

- **1-D STEADY-STATE EXPERIMENTS**

MOBY-DICK NITROGEN-WATER TESTS . . . . DONE
BNL NOZZLE TESTS . . . . DONE
KFK-IRE NOZZLE FLOW TESTS . . . . DONE
MOBY-DICK STEAM-WATER TESTS . . . . CODE FAILED TO CONVERGE
UNIVERSITY OF HOUSTON FLOODING TESTS . . . . CANNOT BE SIMULATED

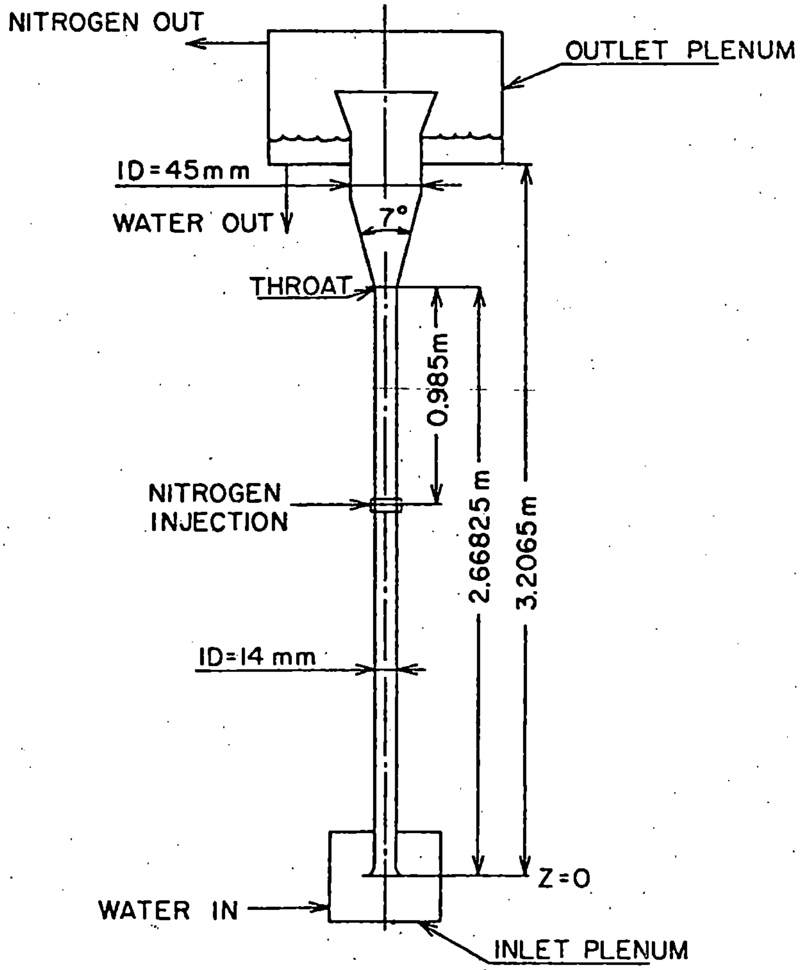
- **1-D TRANSIENT EXPERIMENTS**

SUPER-CANON BLOWDOWN TESTS . . . . DONE
MARVIKEN CRITICAL FLOW TESTS . . . . DONE
BATTELLE-FRANKFURT TOP BLOWDOWN TEST . . . . DONE

- **MULTI-DIMENSIONAL STEADY-STATE EXPERIMENTS**

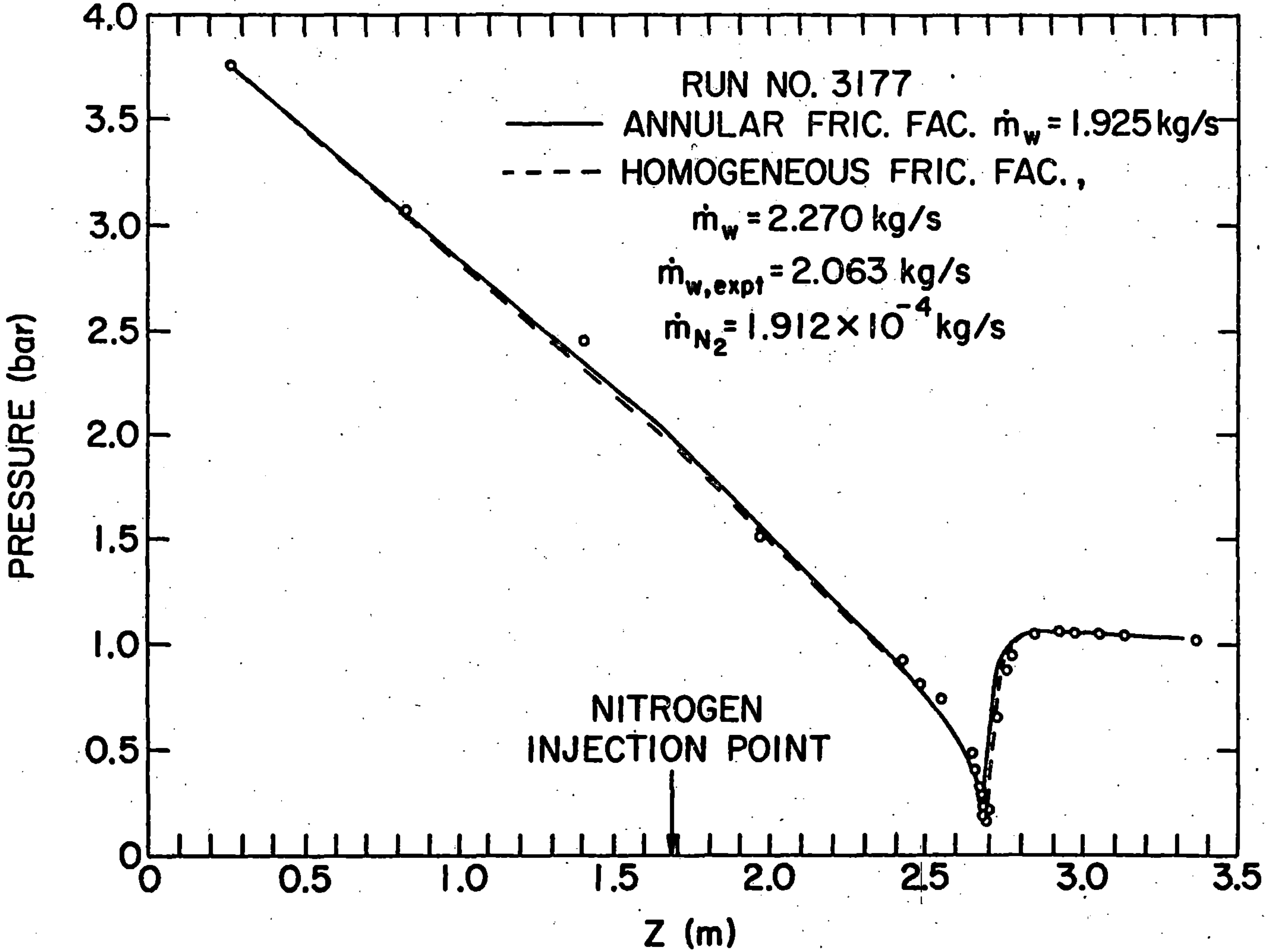
RPI 2-D PHASE SEPARATION TESTS
FRIGG LOOP TEST
. . . . CODE FAILED TO CONVERGE

MOBY-DICK NITROGEN-WATER TESTS

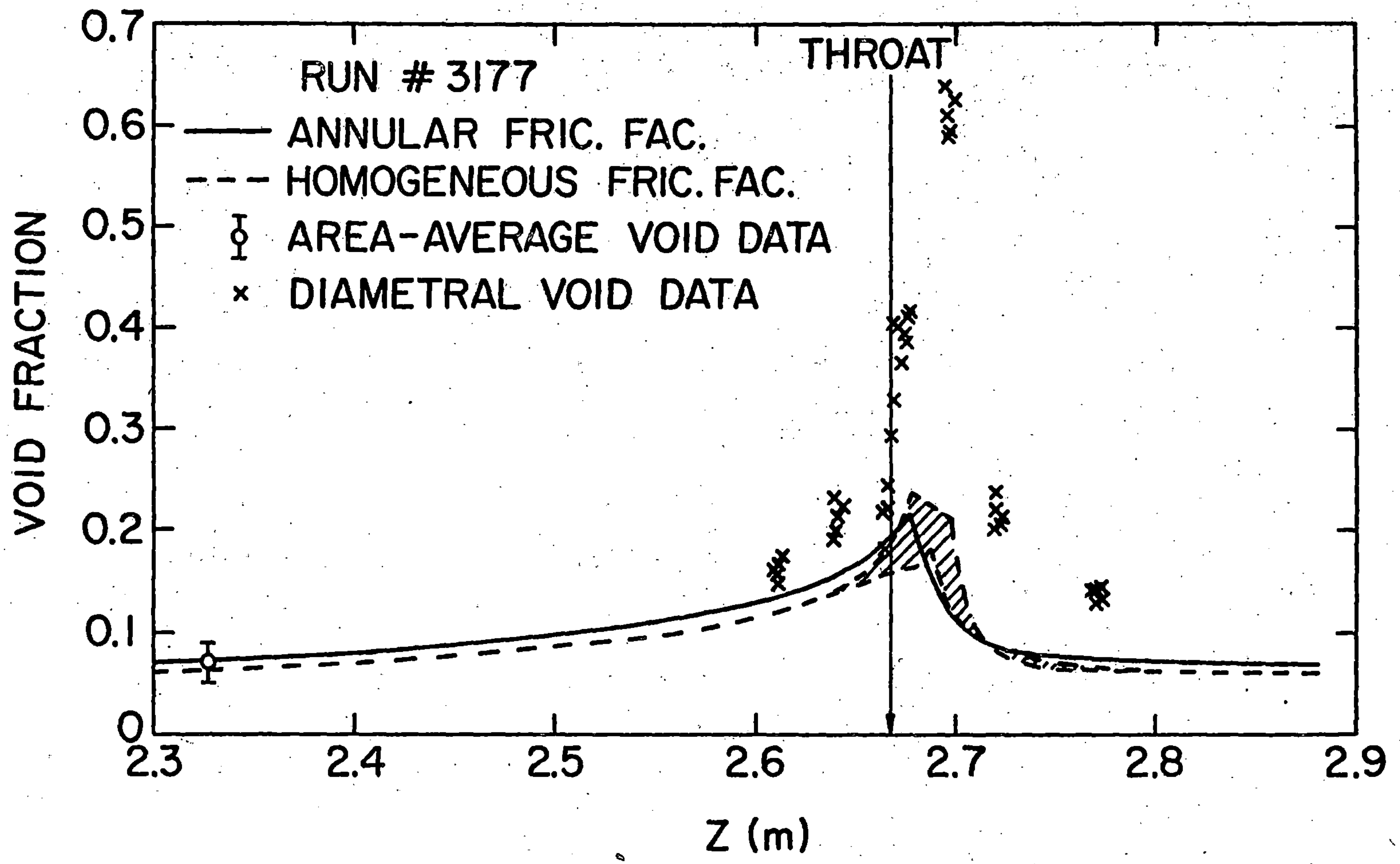


SUMMARY OF MOBY-DICK NITROGEN-WATER TEST RESULTS

RUN NO.	MEASURED FLOW QUALITY (X)	WATER MASS FLOW RATE (KG/S)				
		EXPT.	TRAC CALC. (ANNULAR F.F.)	ERROR (%)	TRAC CALC. (HOMOGENEOUS F.F.)	ERROR (%)
3095	0	1.912	1.697	-11.2	2.138	+11.8
3176	0.94×10^{-4}	2.057	1.829	-11.1	2.259	+ 9.9
3177	0.93×10^{-4}	2.063	1.925	- 6.7	2.27	+10.0
3087	5.91×10^{-4}	1.915	1.822	- 4.9	2.251	+17.5
3089	5.90×10^{-4}	1.918	1.849	- 3.6	2.253	+17.5
3091	5.95×10^{-4}	1.915	1.846	- 3.6	2.250	+17.5
3141	51.3×10^{-4}	1.222	............DID NOT CONVERGE TO A STEADY-STATE.............			
3167	0.75×10^{-4}	2.634	2.390	- 9.3	2.81	+ 6.7
3052	8.72×10^{-4}	1.929	1.865	- 3.3	2.289	+18.5



RUN NO. 3177
ANNULAR FRIC. FAC. $\dot{m}_w = 1.925$ kg/s
HOMOGENEOUS FRIC. FAC., $\dot{m}_w = 2.270$ kg/s
$\dot{m}_{w,expt} = 2.063$ kg/s
$\dot{m}_{N_2} = 1.912 \times 10^{-4}$ kg/s
NITROGEN INJECTION POINT
PRESSURE (bar)
Z (m)
0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0

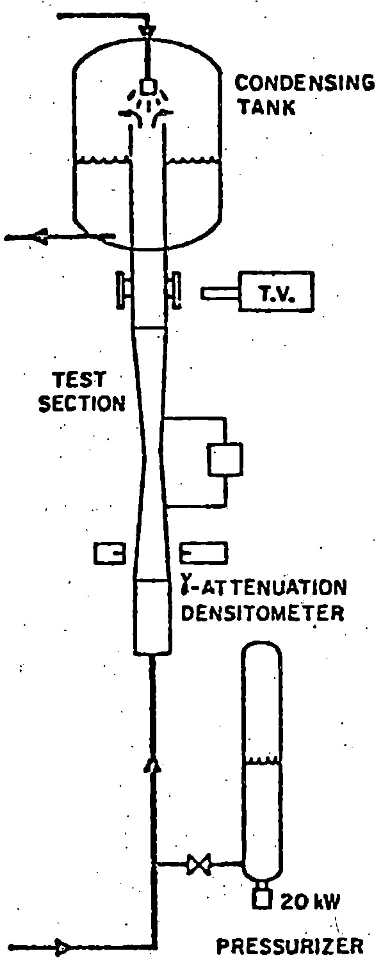


THROAT
RUN # 3177
ANNULAR FRIC. FAC.
HOMOGENEOUS FRIC. FAC.
AREA-AVERAGE VOID DATA
DIAMETRAL VOID DATA
VOID FRACTION
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
2.3
2.4
2.5
2.6
2.7
2.8
2.9
Z (m)

SUMMARY OF MOBY-DICK RESULTS WITH "CORRECTED" SINGLE-PHASE FRICTION FACTORS

RUN NO.	MEASURED FLOW QUALITY (X)	WATER MASS FLOW RATE (KG/S)				
		EXPT.	TRAC CALC. (ANNULAR F.F.)	ERROR (%)	TRAC CALC. (HOMOGENEOUS F.F.)	ERROR (%)
3095	0	1.912	1.932	1.1	2.045	6.9
3176	0.94×10^{-4}	2.057	2.053	-0.2	2.184	6.2
3177	0.93×10^{-4}	2.063	2.109	2.2	2.213	7.3
3087	5.91×10^{-4}	1.915	2.019	5.4	2.198	14.8
3089	5.90×10^{-4}	1.918	2.027	5.7	2.200	14.7
3091	5.95×10^{-4}	1.915	2.024	5.7	2.196	14.7
3141	51.3×10^{-4}	1.222	............DID NOT CONVERGE TO A STEADY-STATE..............			
3167	0.75×10^{-4}	2.634	2.661	1.0	2.790	5.9
3052	8.72×10^{-4}	1.929	2.044	6.0	2.235	15.8

BNL FLASHING FLOW EXPERIMENTS IN A CONVERGING-DIVERGING NOZZLE



THROAT DIA. = 25 MM

INLET &
EXIT DIA. = 50 MM

TEST SECTION LENGTH = 600 MM

COMPARISON OF 121 C INLET TEMPERATURE RUNS
WITH TRAC PREDICTIONS

RUN NO.	P_{IN} (kPa)	P_{CT} (kPa)	EXP. MASS FLOW RATE KG/S	TRAC MASS FLOW RATE KG/S	% DEVIATION
141	239.7 ± 4.9	236.2 ± 0.6	5.98	4.84	-18.9
144	242.5 ± 4.8	237.3 ± 0.8	5.96		
145	306.2 ± 0.7	234.1 ± 0.4	7.46	7.04	- 5.6
148	304.1 ± 0.6	232.6 ± 0.4	7.46		
133	350.3 ± 0.7	232.9 ± 0.6	8.93	8.37	- 6.4
136	347.9 ± 0.6	233.2 ± 0.6	8.95		
140	465.2 ± 2.2	234.2 ± 0.4	11.93	11.71	- 1.6
137	462.8 ± 1.5	233.3 ± 0.5	11.87		

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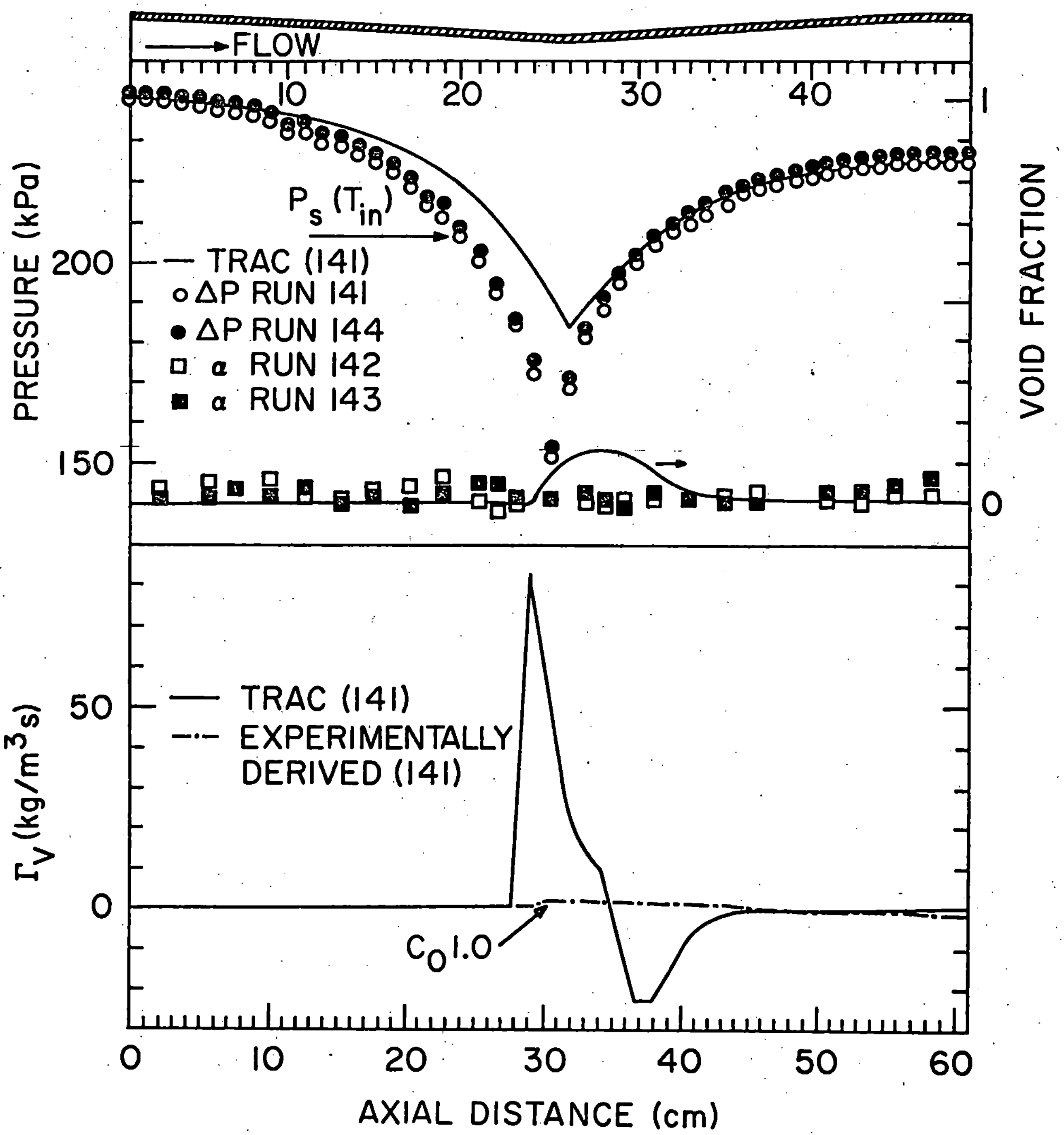
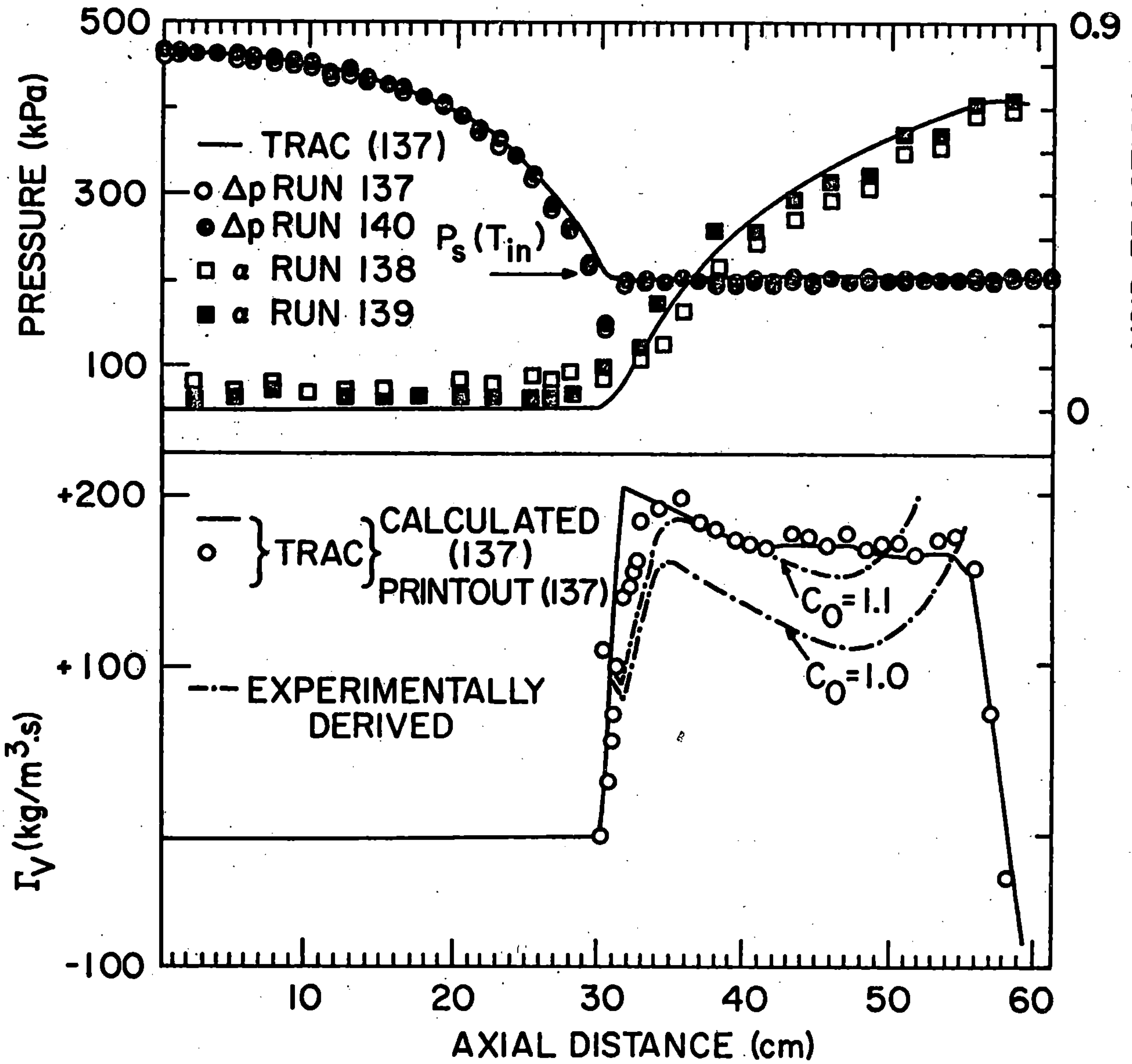
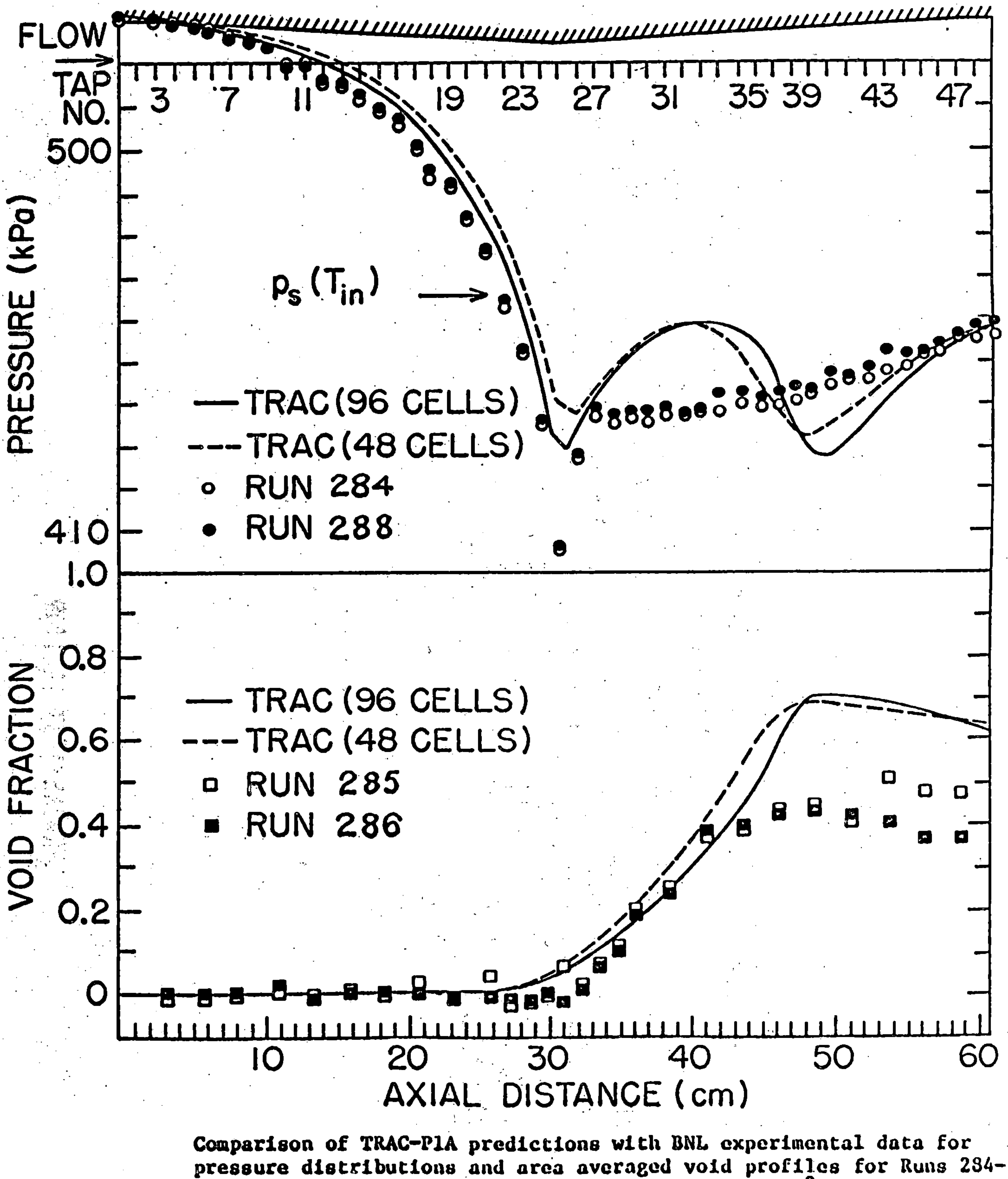


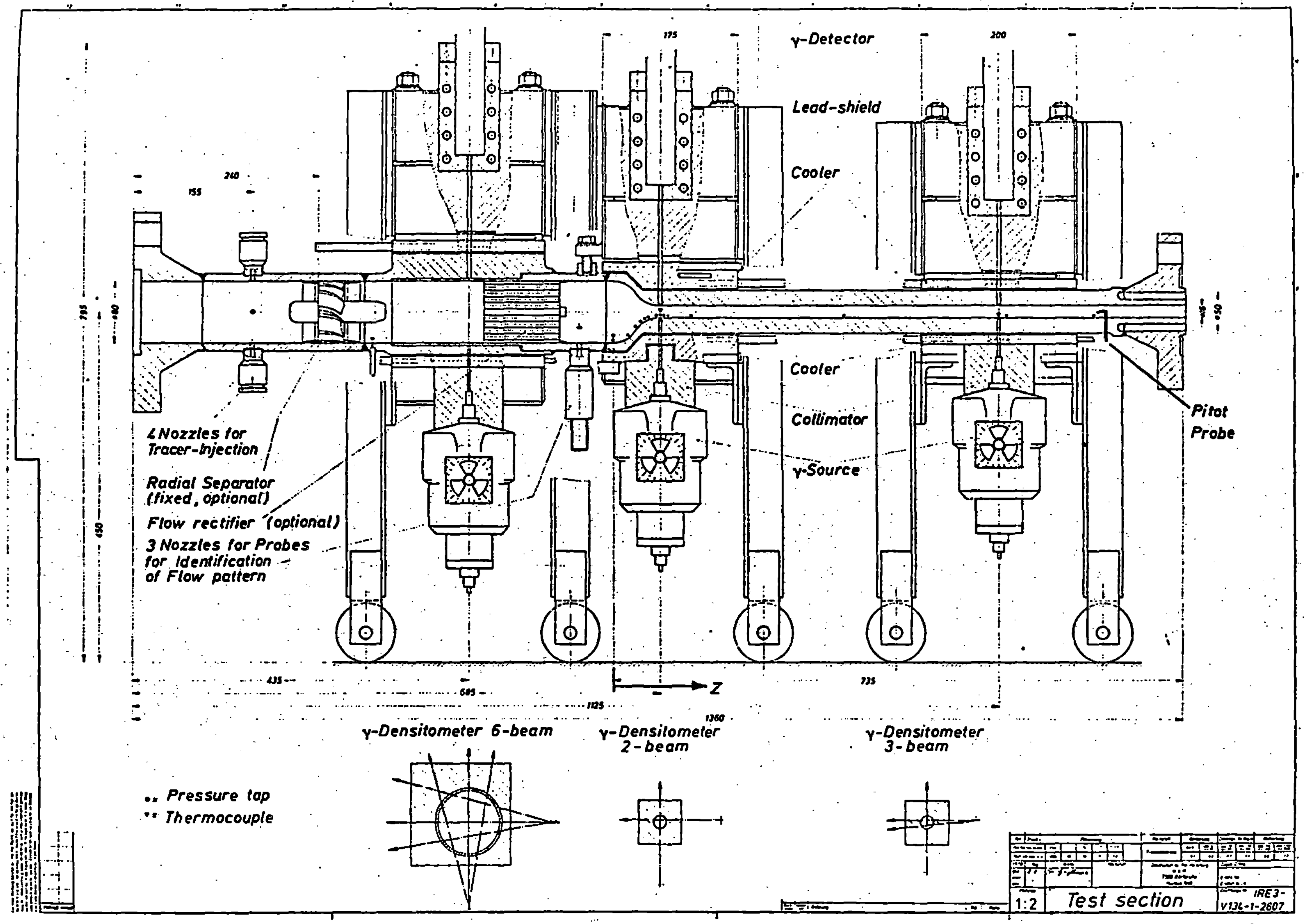
FLOW
PRESSURE (kPa)
VOID FRACTION
$P_s\ (T_{in})$
TRAC (141)
ΔP RUN 141
ΔP RUN 144
α RUN 142
α RUN 143
$\Gamma_V\ (kg/m^3 s)$
TRAC (141)
EXPERIMENTALLY DERIVED (141)
C_0 1.0
AXIAL DISTANCE (cm)



500
300
100
0.9
0
PRESSURE (kPa)
VOID FRACTION
TRAC (137)
Δp RUN 137
Δp RUN 140
α RUN 138
α RUN 139
$P_s(T_{in})$
+200
+100
-100
}TRAC{CALCULATED (137)
PRINTOUT (137)
EXPERIMENTALLY DERIVED
C_0=1.1
C_0=1.0
Γ_V (kg/m^3.s)
10
20
30
40
50
60
AXIAL DISTANCE (cm)



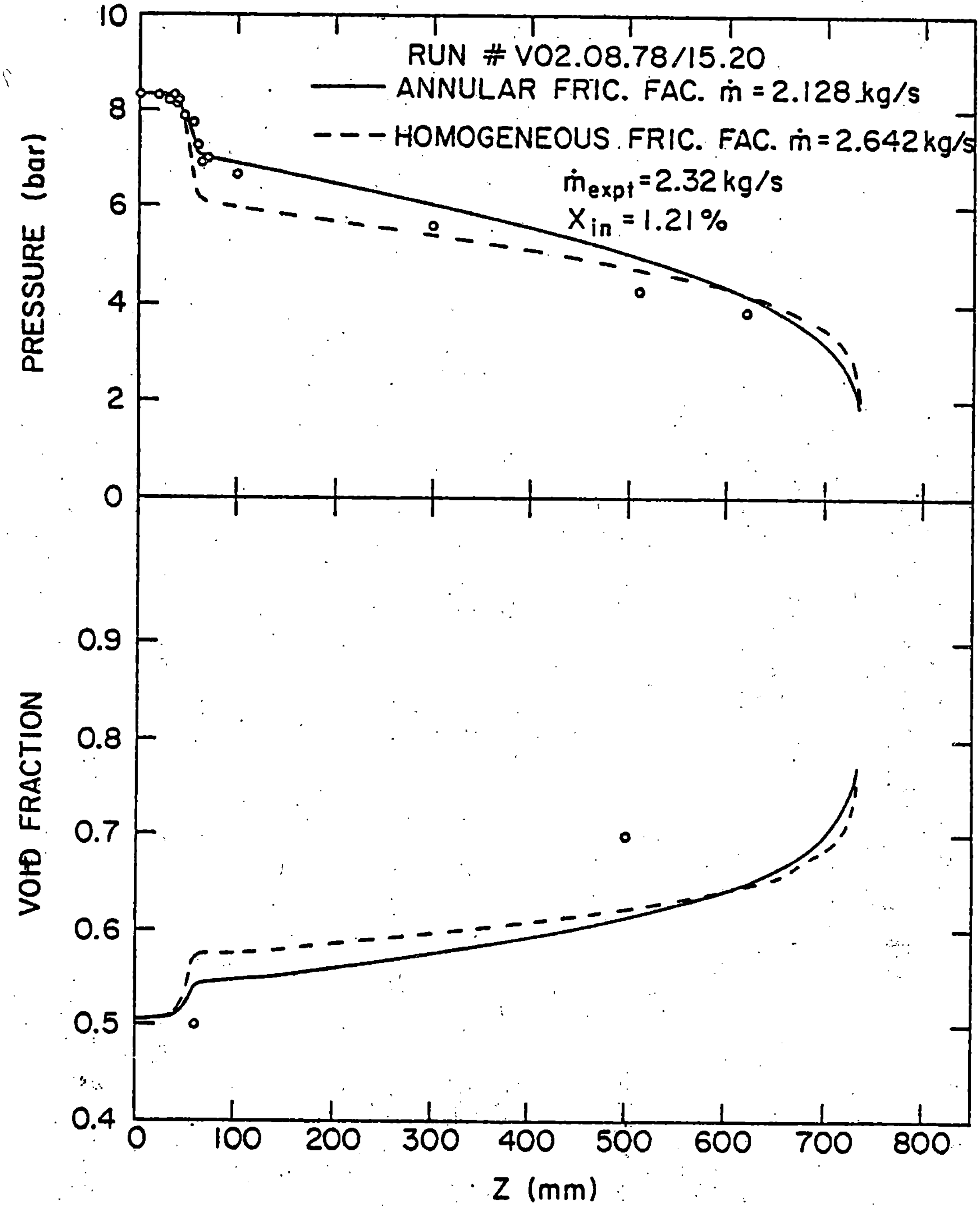
Comparison of TRAC-P1A predictions with BNL experimental data for pressure distributions and area averaged void profiles for Runs 284-288. p_{in} = 530 kPa, T_{in} = 149.2 C, G_{in} = 3580 kg/m^{2}s, p_{ct} = 456 kPa, T_{ct} = 149.2 C. (BNL Neg. No. 9-642-80).



KFK-IRE TEST SECTION

SUMMARY OF KFK-IRE TEST RESULTS

RUN NO.	INLET FLOW QUALITY	MASS FLOW RATE (KG/S) EXPT.	TRAC CALC. (ANNULAR F.F.)	ERROR (%)	TRAC CALC. (HOMOGENEOUS F.F.)	ERROR (%)
V02.08.78/ 13.59 (COLD WATER)	0	6.53	6.216	-5.1	6.694	+ 2.5
V02.08.78/ 15.20 (AIR-WATER)	0.0121	2.32	2.128	-8.3	2.642	+13.9
V15.09.78/ 11.11 (STEAM-WATER)	0.083	3.065	2.816	-8.1	3.771	+23.0



RUN # V02.08.78/15.20
ANNULAR FRIC. FAC. ṁ = 2.128 kg/s
HOMOGENEOUS FRIC. FAC. ṁ = 2.642 kg/s
ṁexpt = 2.32 kg/s
Xin = 1.21 %
PRESSURE (bar)
10
8
6
4
2
0
VOID FRACTION
0.9
0.8
0.7
0.6
0.5
0.4
0
100
200
300
400
500
600
700
800
Z (mm)

ANALYSIS OF VOID DATA FOR KFK-IRE AIR-WATER TEST V02.78.78/15.20

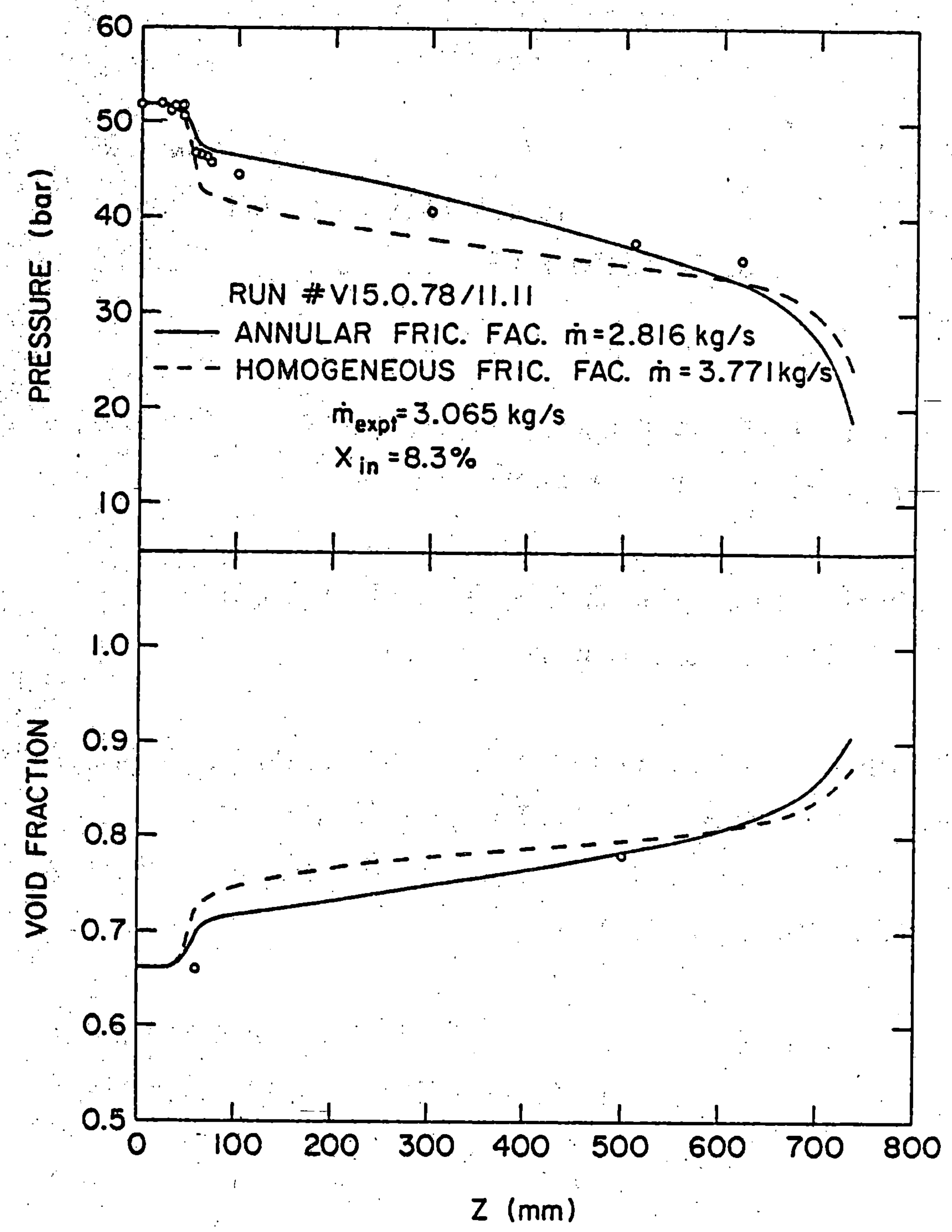
	Z = 60 MM		Z = 500 MM	
ITEM	Co	S	Co	S
1. EXPERIMENTAL VALUE	1.190	1.470	1.019	1.067
2. TRAC - P1A	1.10	1.247	1.10	1.349
3. ISHII	1.182	1.450	1.186	1.649

NOTE: $X = 0.0121$

$\rho_\ell = 996$ KG/M^3

ρ_V (Z = 60) = 8.3 KG/M^3

ρ_V (Z = 500) = 4.9 KG/M^3



PRESSURE (bar)
60
50
40
30
20
10
RUN # V15.0.78/11.11
ANNULAR FRIC. FAC. ṁ = 2.816 kg/s
HOMOGENEOUS FRIC. FAC. ṁ = 3.771 kg/s
ṁexpt = 3.065 kg/s
X in = 8.3%
VOID FRACTION
1.0
0.9
0.8
0.7
0.6
0.5
0
100
200
300
400
500
600
700
800
Z (mm)

CONCLUSIONS FROM THE 1-D STEADY-STATE TESTS

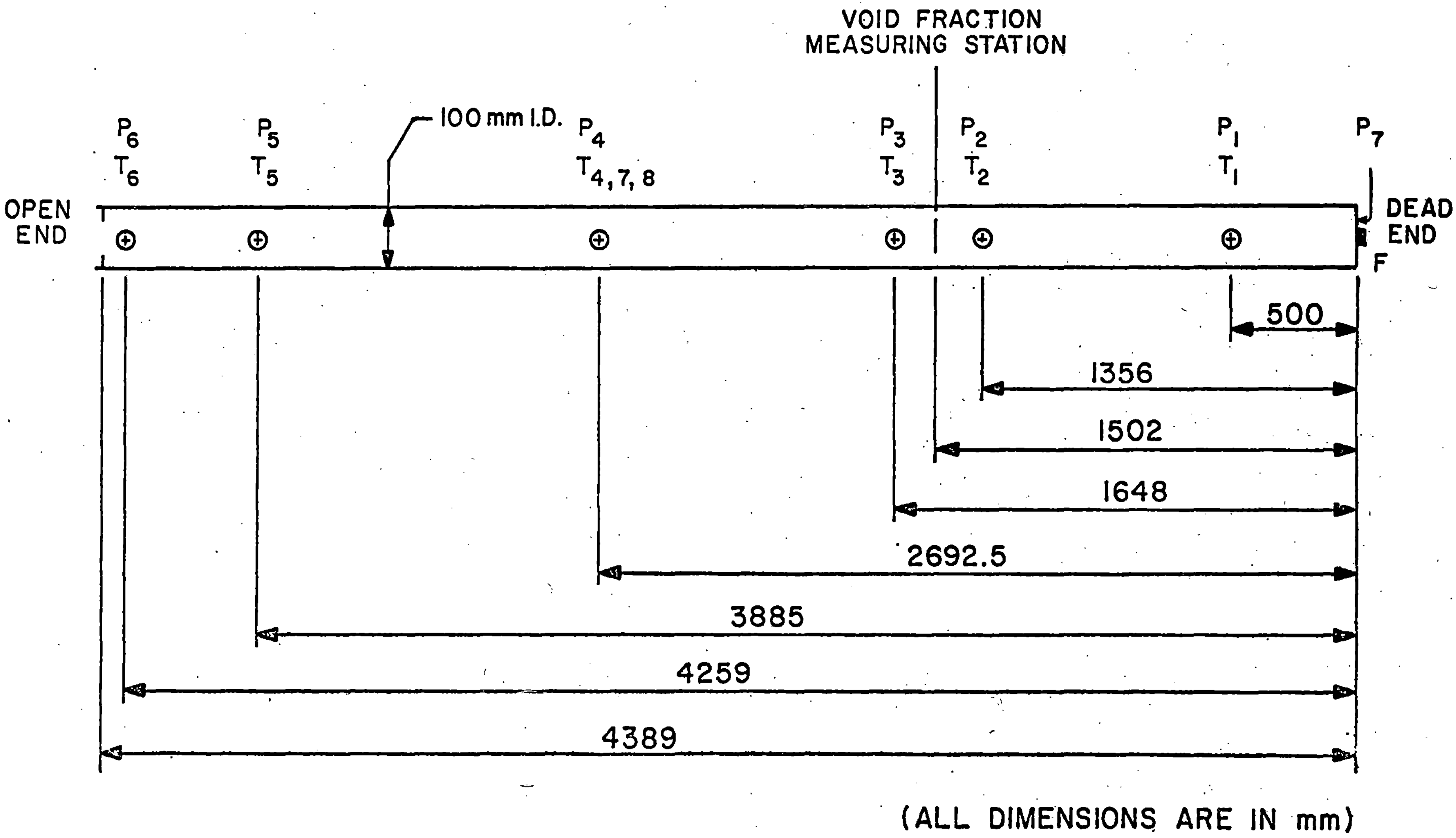
- FOR FRICTION DOMINATED FLOWS, TRAC-P1A PREDICTIONS ARE SENSITIVE TO TWO-PHASE FRICTION FACTOR OPTIONS. USE OF ONE SINGLE-PHASE FRICTION FACTOR CORRELATION REDUCES THIS SENSITIVITY.

  - A COLEBROOK-TYPE CORRELATION WITH INPUT ROUGHNESS PARAMETER IS RECOMMENDED.

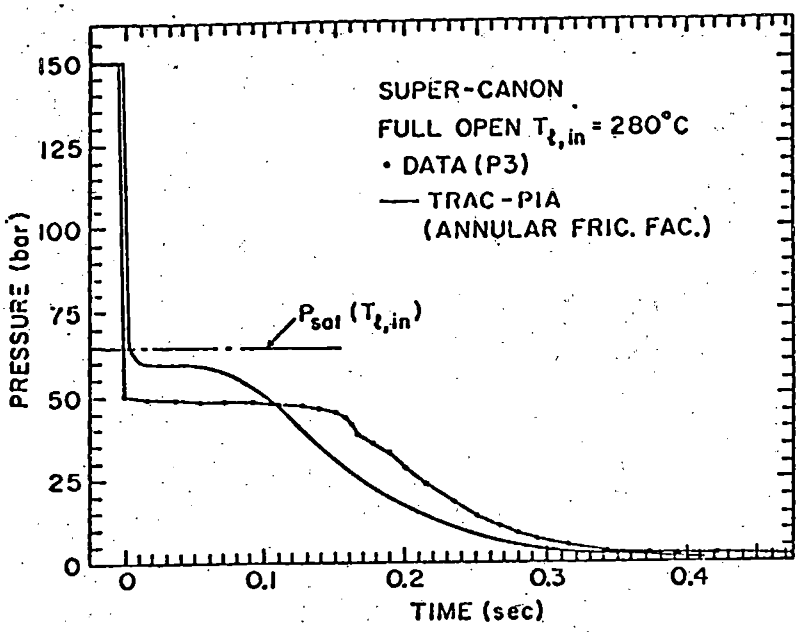
- AT LOW FLOW RATES AND SUBCOOLED INLET, A DELAYED NUCLEATION MODEL SHOULD IMPROVE TRAC PREDICTION.

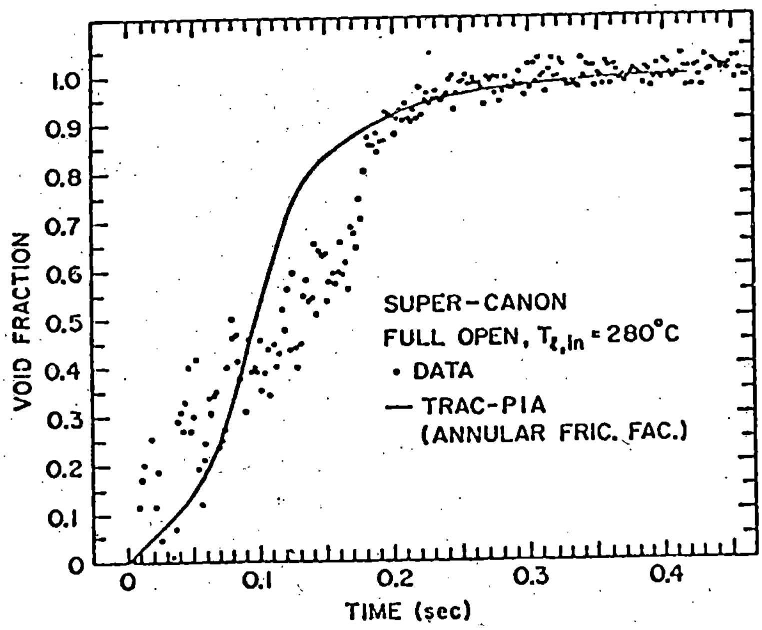
- SOMETIMES THE CODE (TRAC-P1A) DOES NOT CONVERGE TO A STEADY-STATE, PARTICULARLY FOR VERTICAL PIPES WITH HIGH VOID FRACTIONS. THIS IS DUE TO LARGE DIFFERENCE IN SLIP BETWEEN THE SLUG AND THE ANNULAR FLOW REGIME.

- TRAC-P1A SLIP CORRELATION FOR HORIZONTAL PIPES MAY NOT BE ADEQUATE. TRAC-P1A SLIP CORRELATION IS INADEQUATE FOR COUNTER-CURRENT ANNULAR FLOW SITUATIONS.

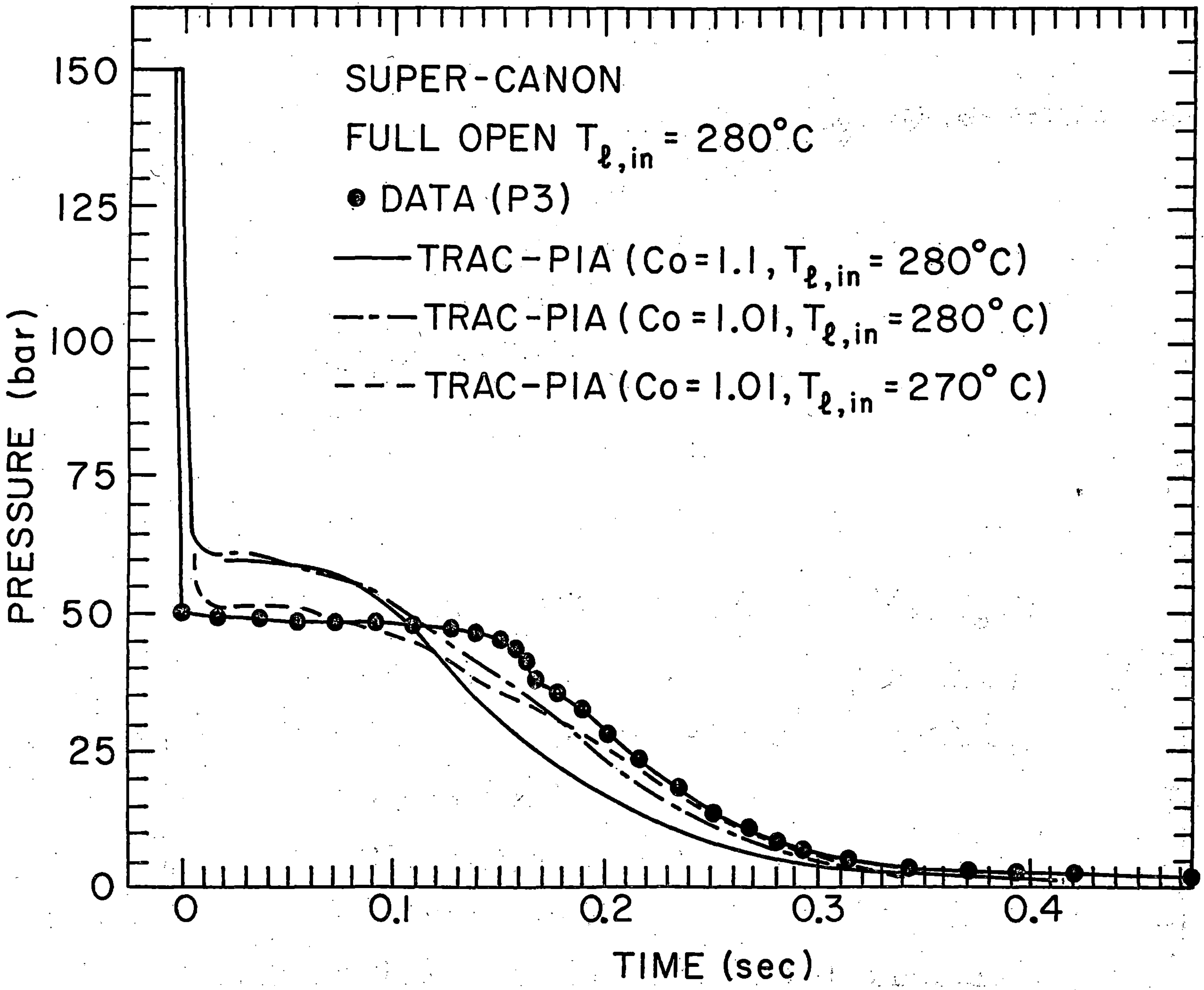


SUPER-CANON TEST SECTION

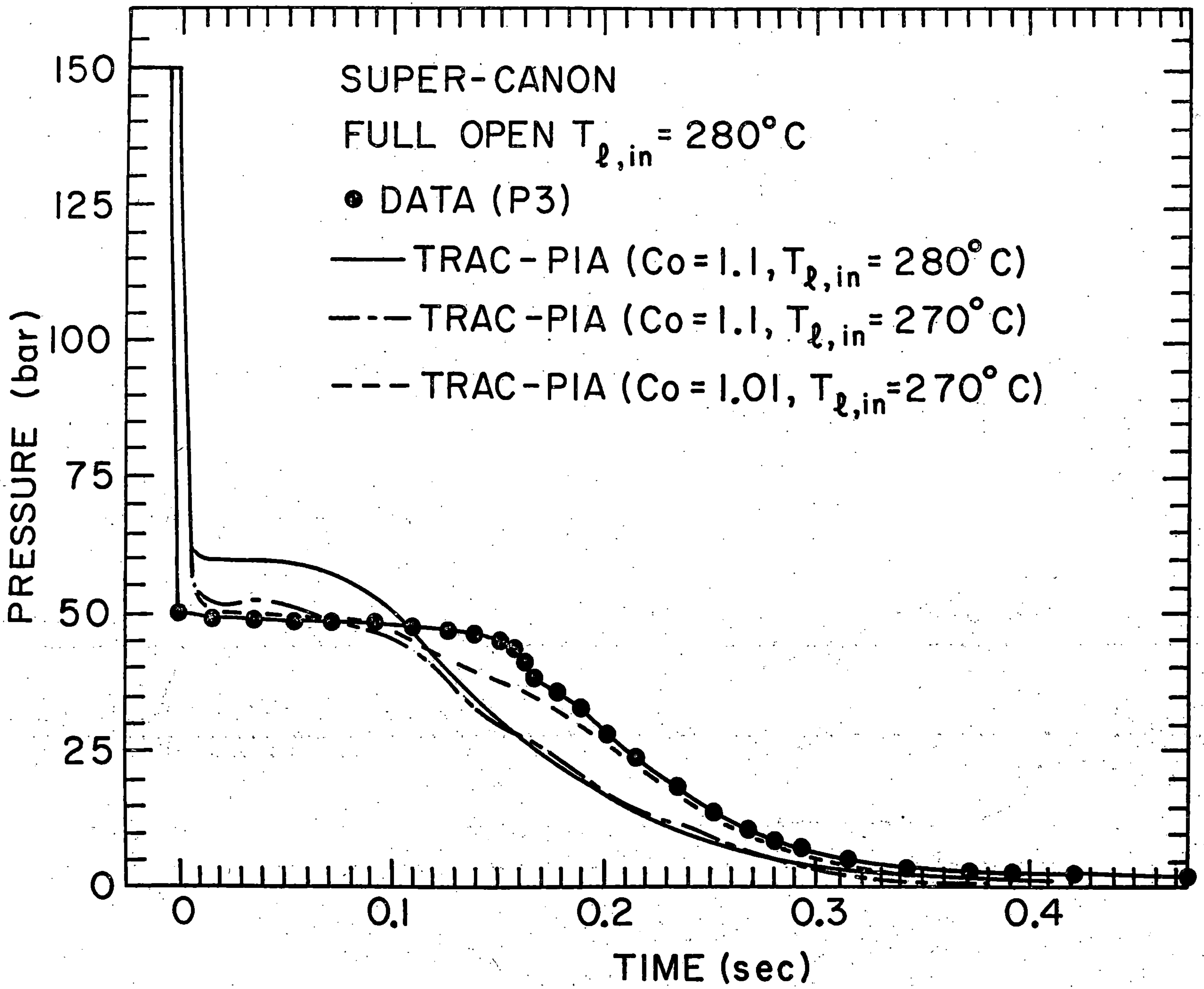




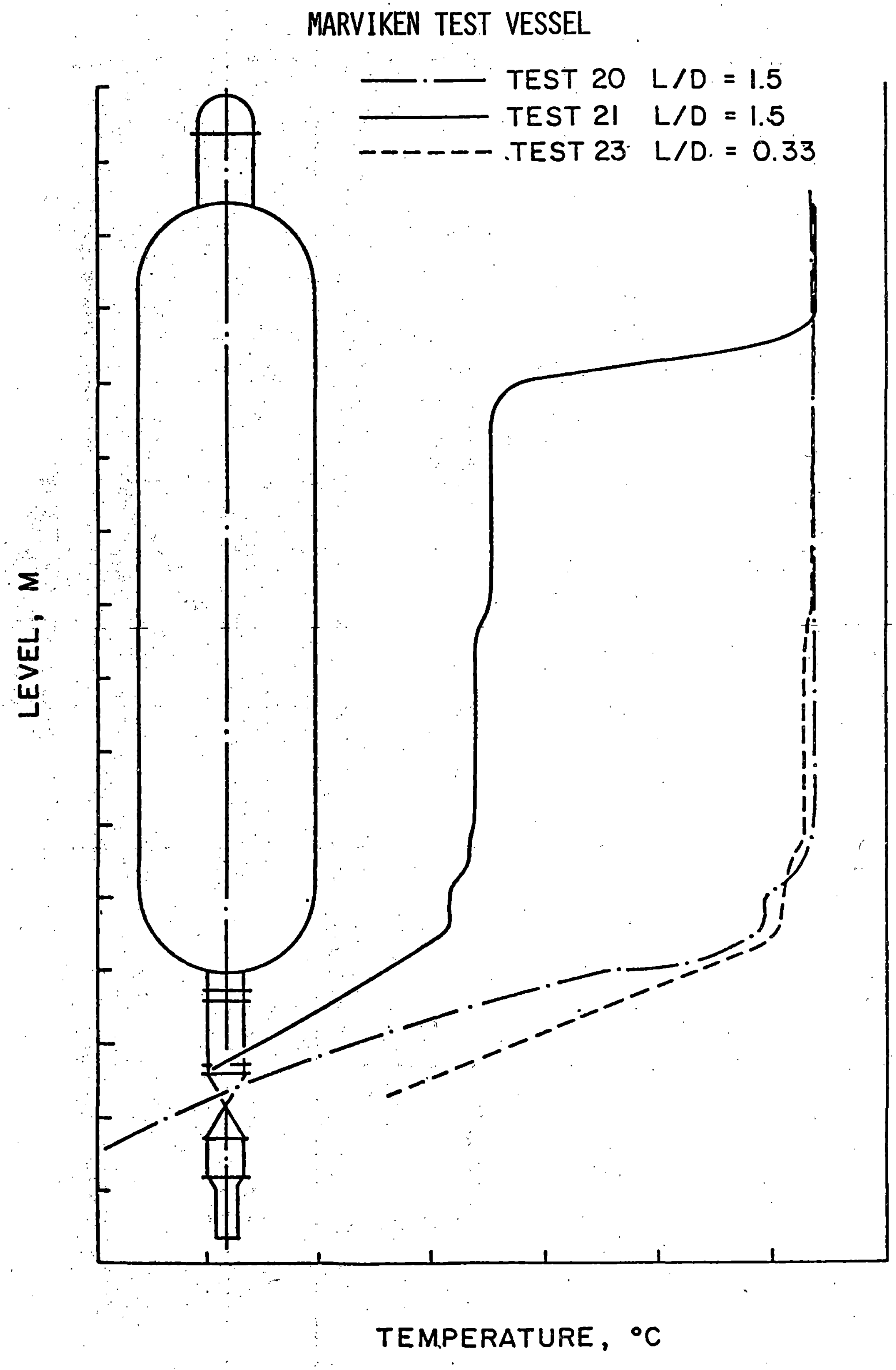
Comparison of TRAC-P1A Prediction of the Pressure and Void Fraction with the Experimental Data of a Super-CANON Test with Full Open Break and Initial Water Temperature of 280° C.



SUPER-CANON
FULL OPEN $T_{\ell,in}$ = 280°C
DATA (P3)
TRAC-PIA (Co=1.1, $T_{\ell,in}$ = 280°C)
TRAC-PIA (Co=1.01, $T_{\ell,in}$ = 280°C)
TRAC-PIA (Co=1.01, $T_{\ell,in}$ = 270°C)
PRESSURE (bar)
150
125
100
75
50
25
0
0
0.1
0.2
0.3
0.4
TIME (sec)



SUPER-CANON
FULL OPEN $T_{\ell,in}$ = 280°C
DATA (P3)
TRAC-PIA (Co=1.1, $T_{\ell,in}$= 280°C)
TRAC-PIA (Co=1.1, $T_{\ell,in}$= 270°C)
TRAC-PIA (Co=1.01, $T_{\ell,in}$=270°C)
PRESSURE (bar)
150
125
100
75
50
25
0
0
0.1
0.2
0.3
0.4
TIME (sec)



MARVIKEN TEST VESSEL
TEST 20 L/D = 1.5
TEST 21 L/D = 1.5
TEST 23 L/D = 0.33
LEVEL, M
TEMPERATURE, °C

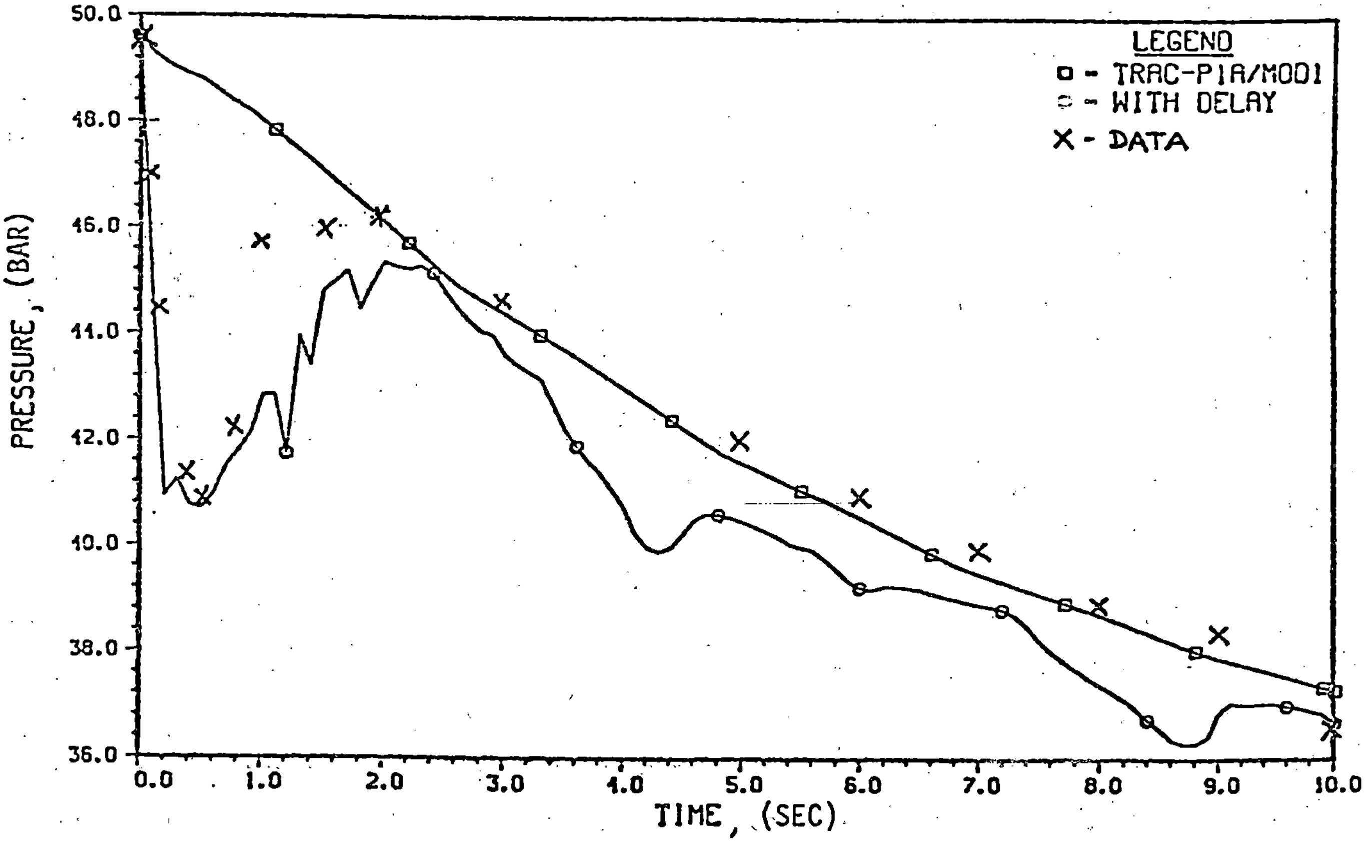
MARVIKEN CRITICAL FLOW TESTS

TEST NUMBER	L/D	$(\Delta T)_{sub}$ IN ^{0}C	MAX ERROR IN P, BAR 0-10 SEC	% MAX ERROR IN P 0-10 SEC	MAX ERROR IN MASS FLUX IN KG/M^2S 0-20 SEC	% MAX ERROR IN MASS FLUX 0-20 SEC
20 *	1.5	7	2.0	4%	-7500	-25%
21 *	1.5	33	7.0	15.5%	-8000	-25%
22 *	1.5	52	8.0	15.5%	-9000	-20%
23 *	0.33	5	3.8	8%	-22400	-33%
24 *	0.33	33	8.4	19.5%	-14000	-34%
25 †	1.7	6	1.9	4%	-4000	-32%
26 †	1.7	30	3.0	7%	-2500	-16%

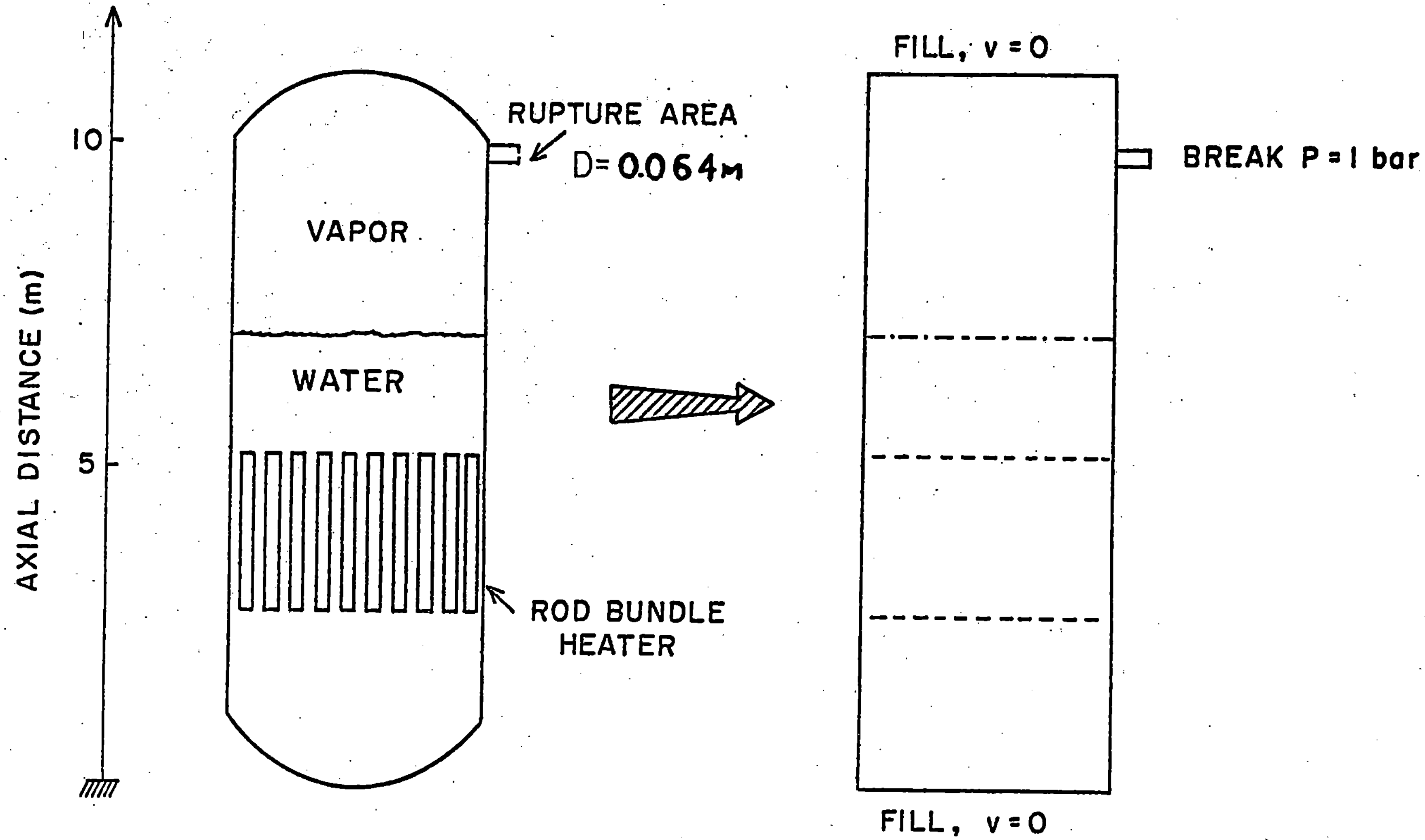
*D = 500 MM

†D = 300 MM

MARVIKEN TEST 24, VESSEL TOP PRESSURE

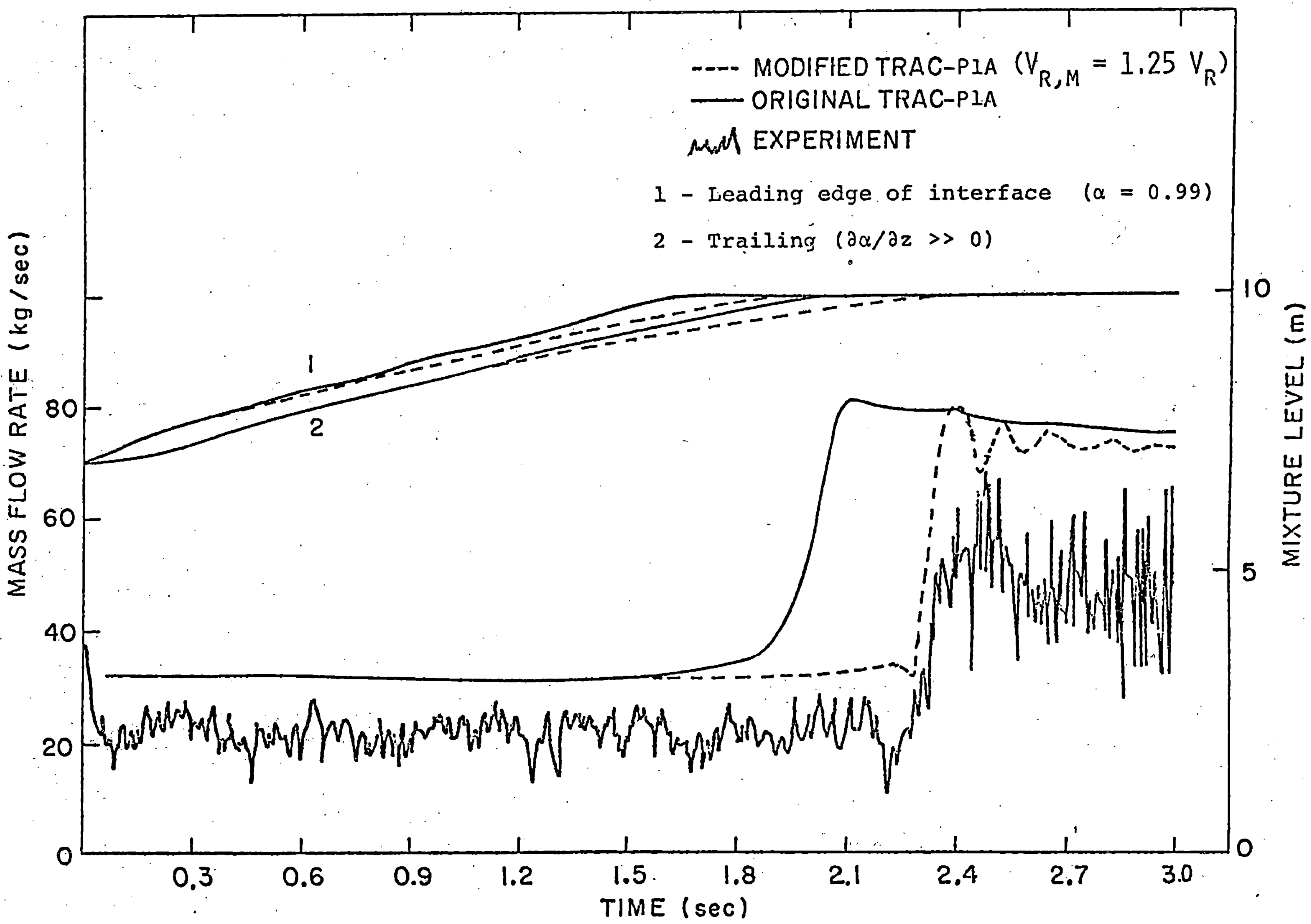


BATTELLE-FRANKFURT TOP BLOWDOWN TEST

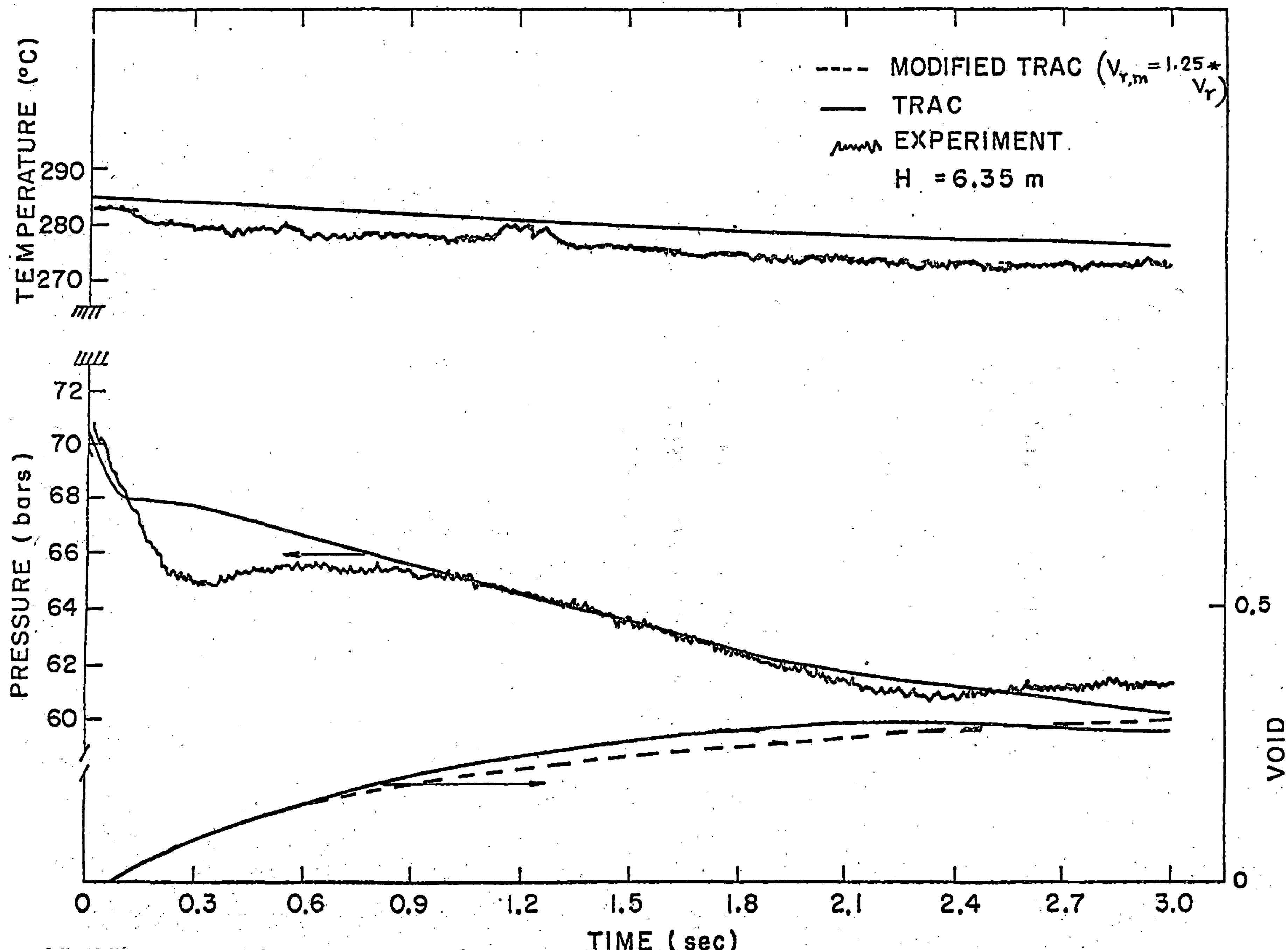


a. BATTELLE VESSEL
(RADIAL DIMENSION
NOT IN SCALE)
D≈0.78m

b. TRAC TEE COMPONENT



---- MODIFIED TRAC-P1A ($V_{R,M}$ = 1.25 V_R)
—— ORIGINAL TRAC-P1A
EXPERIMENT
1 - Leading edge of interface (α = 0.99)
2 - Trailing (∂α/∂z >> 0)
1
2
MASS FLOW RATE (kg/sec)
0
20
40
60
80
MIXTURE LEVEL (m)
0
5
10
0.3
0.6
0.9
1.2
1.5
1.8
2.1
2.4
2.7
3.0
TIME (sec)



MODIFIED TRAC ($V_{r,m}$ = 1.25 * V_r)
TRAC
EXPERIMENT
H = 6.35 m
TEMPERATURE (°C)
290
280
270
PRESSURE (bars)
72
70
68
66
64
62
60
VOID
0.5
0
0
0.3
0.6
0.9
1.2
1.5
1.8
2.1
2.4
2.7
3.0
TIME (sec)

CONCLUSIONS FROM THE 1-D TRANSIENT TESTS

- TRAC-P1A SHOWS QUALITATIVE AGREEMENT WITH THE TRANSIENT EXPERIMENTS.

- FOR SUPER-CANON TESTS, THE CODE TENDS TO OVERPREDICT THE DISCHARGE FLOW RATE.

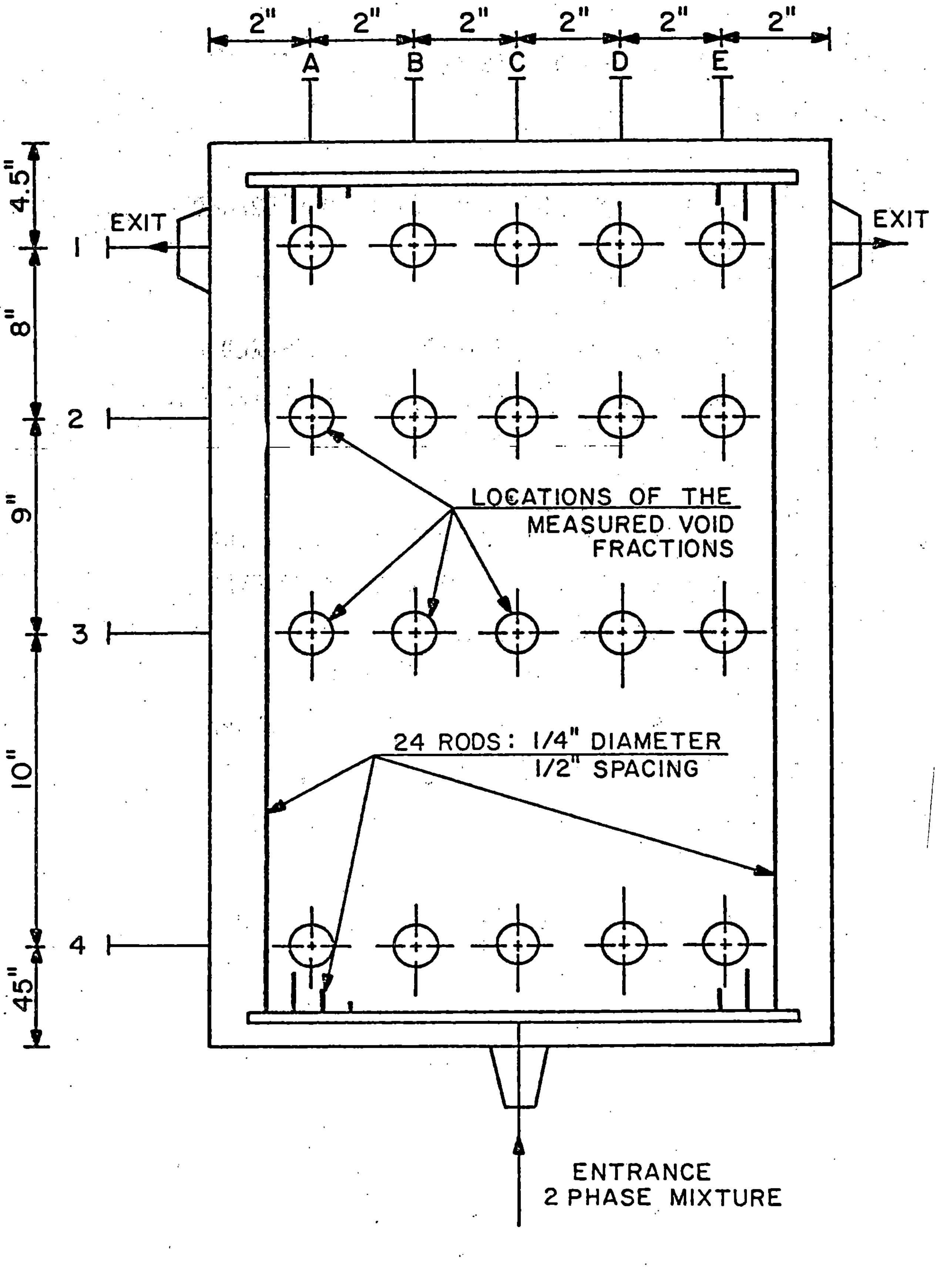
- FOR MARVIKEN TESTS, THE CODE TENDS TO UNDERPREDICT THE MASS-FLUX MORE AS THE NOZZLE LENGTH-TO-DIAMETER RATIO IS DECREASED.

- A DELAYED NUCLEATION MODEL (ALAMGIR-LIENHARD CORRELATION) IMPROVED TRAC'S SHORT-TERM PREDICTIONS. HOWEVER, IT DID NOT IMPROVE THE LONG-TERM RESULTS.

- TRAC PREDICTION FOR BATTELLE-FRANKFURT TOP BLOWDOWN TEST LOOKS REASONABLE. MORE DATA (E.G. VOID FRACTION) AND MORE TESTS ARE NEEDED TO EXPLAIN THE DISCREPANCIES.

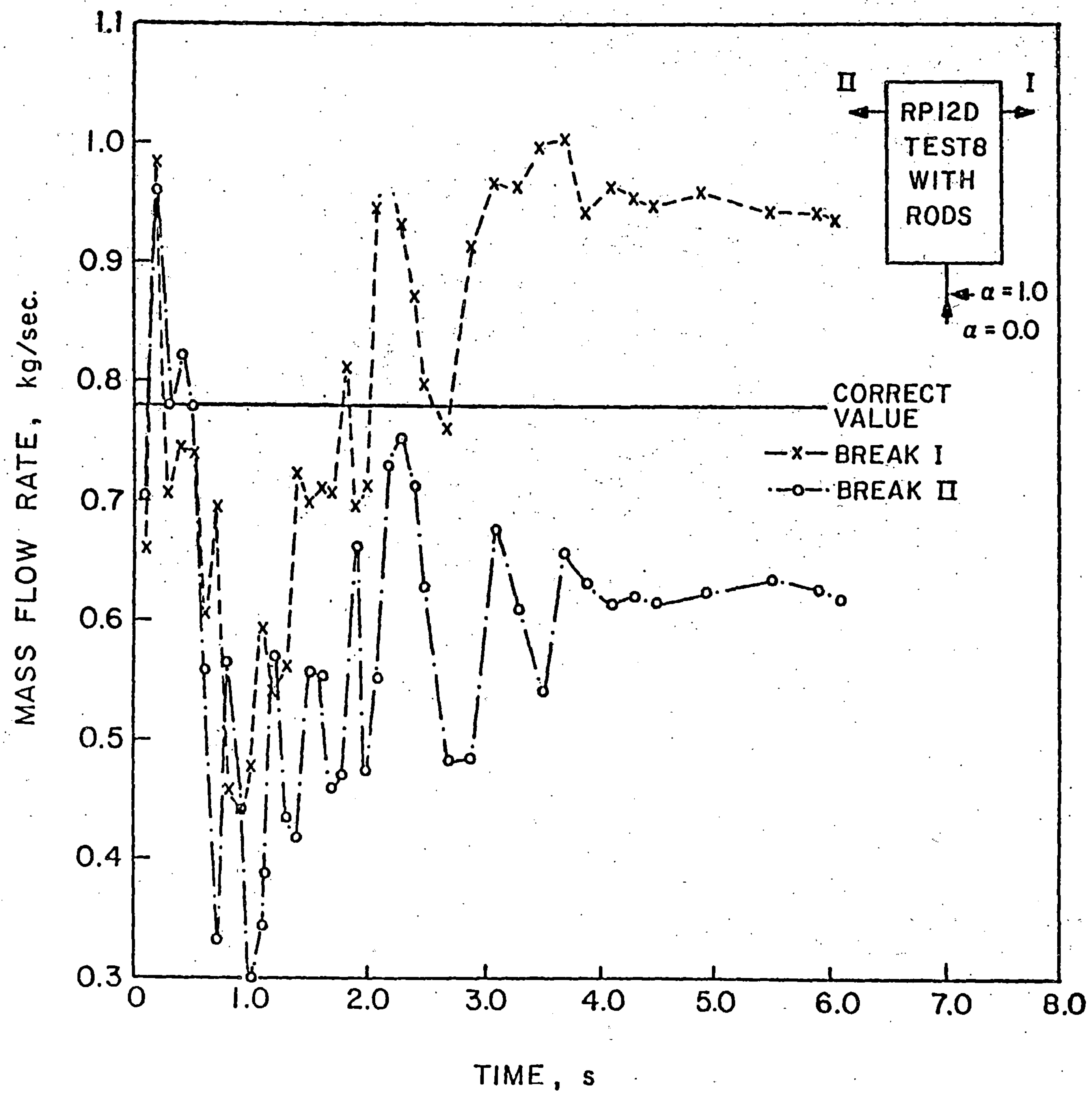
- FURTHER EXAMINATION OF TRAC'S PHASE CHANGE AND RELATIVE VELOCITY MODELS IS NEEDED.

RPI 2-D TEST SECTION



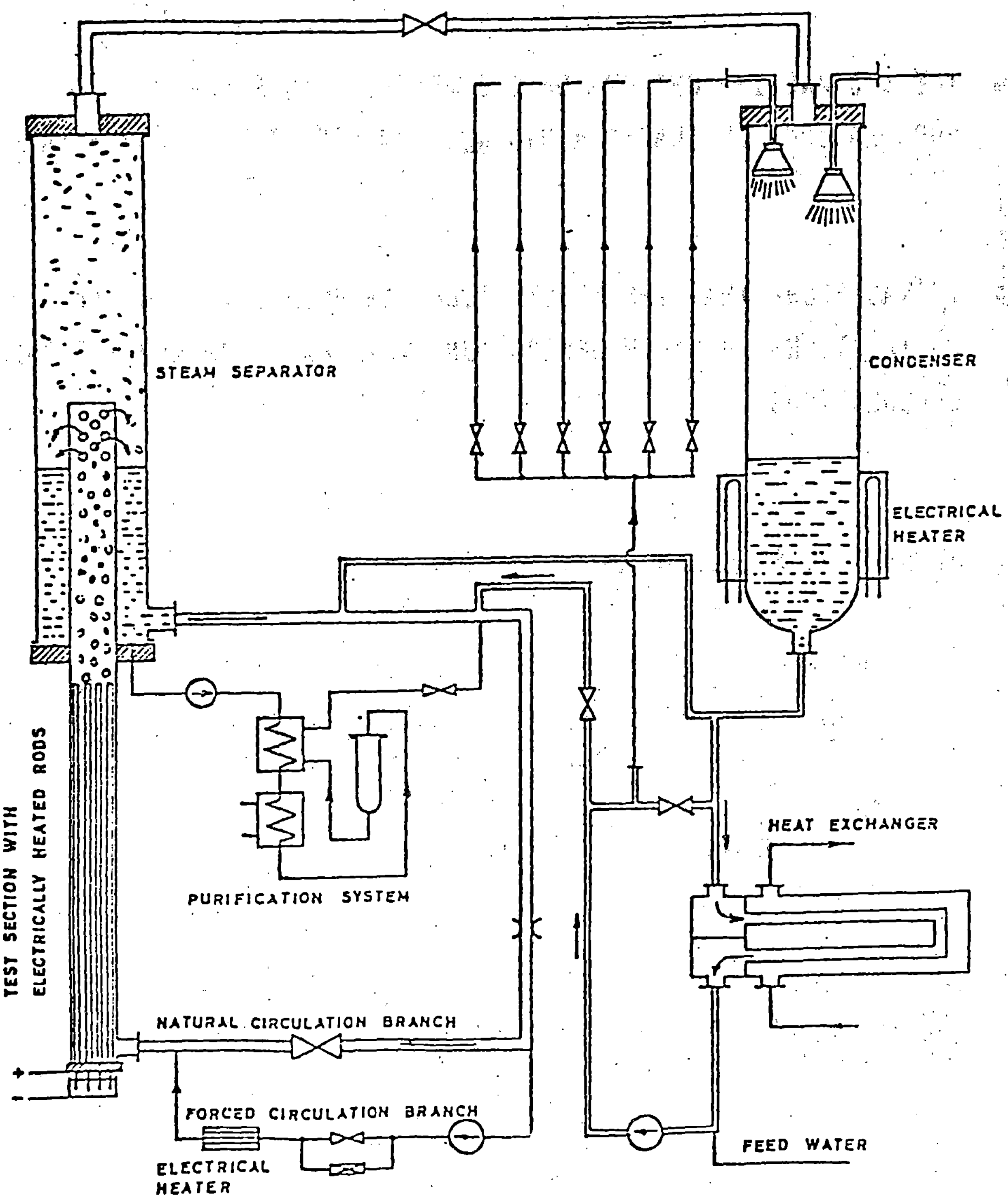
SUMMARY OF TRAC-P1A RESULTS FOR RPI TESTS

TEST	RODS	OUTLET	REMARKS
1	NO	ONE	STOPS, INDEFINITE CONDITIONS
3	NO	TWO	FAILED TO REACH STEADY-STATE STABLE SOLUTION
6	YES	ONE	FAILED TO REACH STEADY-STATE CLOSE TO STEADY-STATE
8	YES	TWO	FAILED TO REACH STEADY-STATE STABLE SOLUTION
11	NO	ONE	STOPS, OVERFLOW CONDITIONS
14	NO	TWO	FAILED TO REACH STEADY-STATE STABLE SOLUTION
15	YES	ONE	STOPS, OVERFLOW CONDITIONS
18	YES	TWO	FAILED TO REACH STEADY-STATE STABLE SOLUTION



RP12D
TEST8
WITH
RODS
II
I
α = 1.0
α = 0.0
CORRECT VALUE
BREAK I
BREAK II
MASS FLOW RATE, kg/sec.
TIME, s
1.1
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0

SIMPLIFIED FLOW DIAGRAM FOR THE FRIGG LOOP



CONSTRUCTION MATERIAL : CARBON STEEL

COOLING CAPACITY : 8 MW

MAX. PRESSURE : 100 bars

RESULTS OF FRIGG TEST SIMULATION

- THE 3-D AND 1-D OPTIONS OF THE VESSEL MODULE (TWO-FLUID FORMULATION) OF TRAC-P1A DID NOT CONVERGE TO A STEADY-STATE.

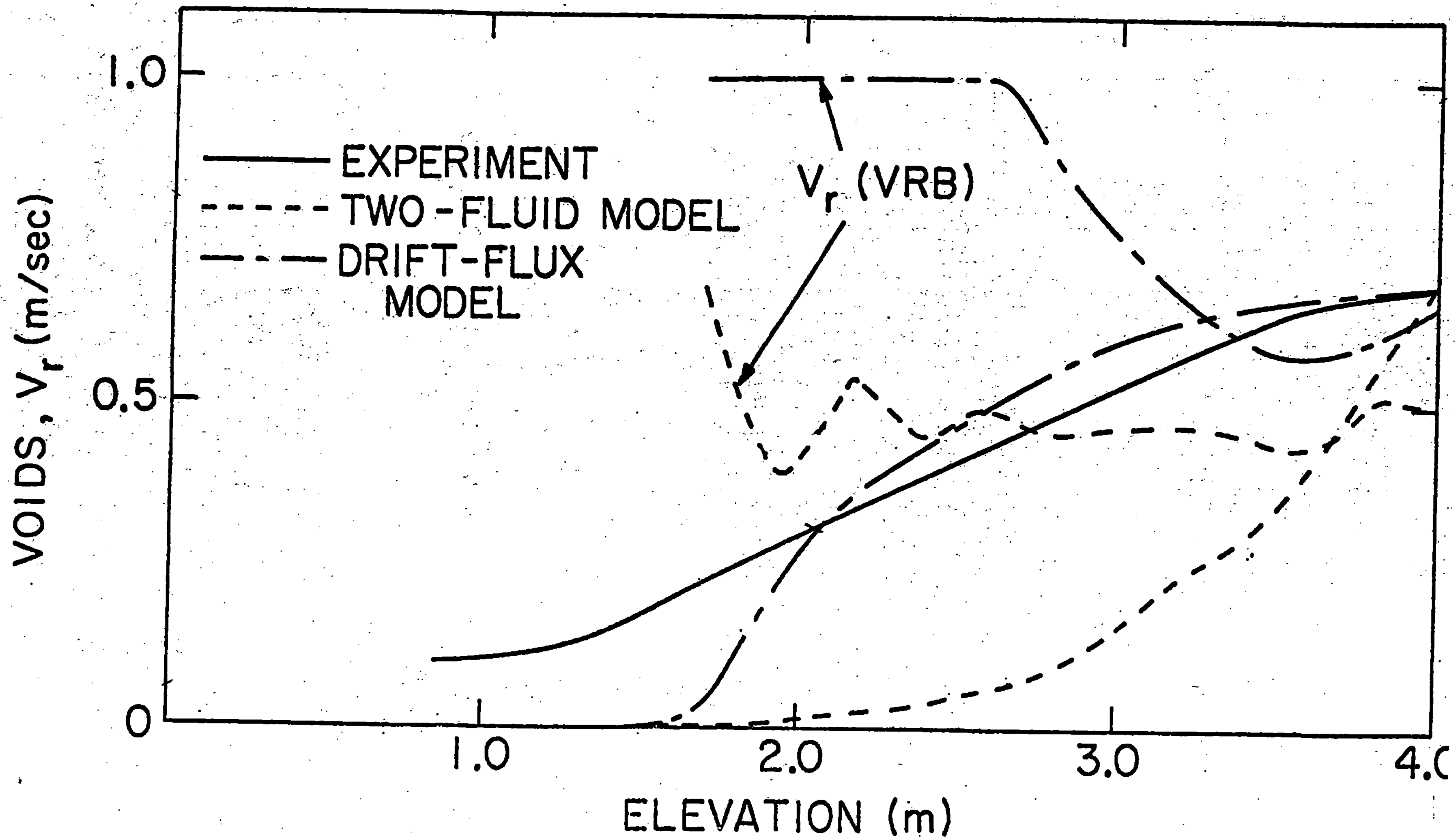
- IT WAS FOUND THAT THE FLUCTUATIONS IN RELATIVE VELOCITY CAUSED OSCILLATIONS IN VAPOR GENERATION, VOID FRACTION AND PRESSURE CALCULATIONS.

- AT BNL THE FOLLOWING MODIFICATION WAS DONE:

  - AFTER N_0 TIME STEPS, RELATIVE VELOCITY FOR VAPOR GENERATION CALCULATIONS IS TAKEN AS,

$$\bar{V}_R = \sum_{i=N_0}^{n} V_{R,i} \Big/ (n - N_0 + 1)$$

- THIS MODIFICATION RESULTED IN A STEADY-STATE SOLUTION FOR THE 1-D TWO-FLUID FORMULATION.



TRAC's Predictions of the Voids and Relative Velocity for FRIGG Test (313020).

CONCLUSIONS FROM THE MULTI-DIMENSIONAL STEADY-STATE EXPERIMENTS

- FOR 2-D VESSEL OPTION OF TRAC-P1A, THERE MIGHT BE SOME ERRORS DURING TRANSFORMATION FROM THE CYLINDRICAL (3-D) TO SLAB (2-D) GEOMETRY.

- THE 3-D AND 1-D FRIGG TEST SIMULATIONS WITH THE VESSEL MODULE OF TRAC-P1A DID NOT REACH A STEADY-STATE.

- WITH BNL MODIFICATION, THE 1-D VESSEL OPTION OF TRAC-P1A PRODUCED A STEADY-STATE FOR THE FRIGG TEST. HOWEVER, POOR AGREEMENT IS DUE TO:

  - LACK OF SUBCOOLED BOILING MODEL IN TRAC-P1A

  - LOWER RATE OF VAPOR GENERATION IN THE 2-FLUID MODEL OF TRAC-P1A.

MODELS NEEDED IN TRAC

(BASED ON BNL ASSESSMENT OF TRAC-P1A)

- IMPROVED MODELS FOR INTERFACIAL MASS, MOMENTUM AND HEAT TRANSFER

- A DELAYED FLASHING (NUCLEATION) MODEL

- IMPROVED RELATIVE VELOCITY CORRELATIONS FOR THE HORIZONTAL AND VERTICAL CHANNELS (FOR 1-D DRIFT-FLUX FORMULATION)

- SUBCOOLED BOILING MODEL

SEPARATE EFFECTS THERMAL HYDRAULIC MODELING PROGRAM

OCTOBER 28, 1980

AFTERNOON SESSION - RED AUDITORIUM

PRESENTED BY:

L. HAROLD SULLIVAN

U.S. NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND

SEPARATE EFFECTS THERMAL HYDRAULIC MODELING PROGRAM

The Separate Effects Thermal Hydraulic Modeling program has the overall purpose of providing experimental data for specific licensing issues, development of analytical models and correlations to be incorporated in the U.S. Nuclear Regulatory Commission's (NRC) accident analysis codes and providing futher understanding of phenomena that occurs during system transients. This program consists of approximately 10 small projects being performed by several national laboratories and universities (see enclosure). Because of the time restriction for this section, only results from four of these programs will be covered. If there are specific interests in the scope or results from any other projects, information can be obtained from the Separate Effects Research Branch at NRC or from individual project managers directly.

The first talk covers steam generator flow pattern and modeling under conditions similar to those expected to occur during a small-break loss of coolant accident (LOCA) event. Since steam generator heat transfer has a dominate effect on small-break transients, its behavior during this event is important. This program was initiated to provide understanding of flow patterns in multiple tube passages to aid both licensing review and code development efforts. Results to date indicate that the steam generator will be effective in removing decay heat during small-break LOCA events.

Another talk from the Massachusetts Institute of Technology will be on penetration of water from the upper plenum into core during reflood tests. It will provide dp-criteria for such penetration. It is a cooperative program between the Japanese Atomic Energy Research Institute and the United States.

The third talk covers the condensation heat transfer work being performed to support the NRC advance code development area. Preliminary version of these codes have been released and are being used by both licensing and experimental personnel. The purpose of this program is to determine condensation heat transfer coefficients for various flow configurations to serve as input to codes. Products of this program are a group of correlations for condensation coefficients in the form of a Stanton number, which is a function of the Reynolds numbers, as well as criteria for upper plenum penetration. This presentation will cover the parallel flow in both vertical and horizontal directions, penetration of flow across the tie plate and spray condensation. Correlations will be presented for code use. They will also be used by the Office of Nuclear Reactor Regulation (NRR) directly.

The fourth talk covers phase separation and parallel channel flow instabilities work being performed by Rensselaer Polytechnic Institute. This project was initiated as a result of concerns about the ability of the present version of computer codes to calculate these effects. The presentation will cover the phase separation in various geometries and flow instabilities in light water reactors. Data and models developed for phase separation are to be used by codes in predicting void distribution in the core as well as in other parts of the reactor. This kind of information will be of practical importance to flow distribution calculation in small-break LOCA analyses. Flow instabilities might determine the core thermal hydraulic behavior during BWR transients. The possible existence and consequence of core instabilities will be used by NRR for technical review of licensing applications.

PROJECT	INSTITUTION	APPLICATION
1. Steam Generator Behavior and Reflood Modeling	MIT	Small-Break and Reflood
2. Post-CHF Heat Transfer	ANL/LEHIGH/AECL	Reflood and Blowdown Heat Transfer
3. Condensation Phenomena	NWU	Steam Generation and ECC Injection
4. Phase Separation and Parallel Channel	RPI	Code Development for Two-Phase Flow
5. Two-Phase Flow Constitative Equation	ANL	Code Development for Two-Phase Flow
6. Heat Transfer Coordination	ANL	LOCA Heat Transfer
7. BWR Heat Transfer	INEL	Code Development
8. Void Fraction Generation	BNL*	Blowdown
9. Droplet Flow	SUNY-SB*	Reflood

* Project Completed

Enclosure

THERMAL HYDRAULIC TOPICS TO BE PRESENTED IN THIS SESSION

1. STEAM GENERATOR MODEL - GRIFFITH, MIT
2. WATER DUMPING FROM UPPER PLENUM - SUDO, MIT/JAERI
3. STEAM-WATER CONDENSATION PHENOMENA - BANKOFF, TANKIN, YUEN, NORTHWESTERN UNIV.
4. PHASE DISTRIBUTION/SEPARATION AND THERMAL HYDRAULIC INSTABILITIES - LAHEY, RPI

STEAM GENERATOR FLOW MODEL - MIT

- OBJECTIVE: DETERMINATION OF FLOW PATTERN IN STEAM GENERATOR
- APPLICATION: MODEL FOR TWO-PHASE FLOW IN STEAM GENERATOR DURING SMALL-BREAK FOR USE IN CODE
- RESULTS
  - FLOW PATTERNS IN STEAM-GENERATOR
  - PRESSURE DROP AS FUNCTION OF FLOW RATE
  - DETERMINATION OF LONG-TERM COOLING CAPABILITY OF STEAM GENERATOR DURING SMALL-BREAK

WATER DUMP FROM UPPER PLENUM - MIT/JAERI

- OBJECTIVES: ESTABLISH FLOW MAP TO DETERMINE THE ONSET OF DUMPING OF WATER FROM UPPER PLENUM
- APPLICATION: 2D/3D REFLOOD TESTS TO SET CRITERION FOR WATER DUMPING FROM UPPER PLENUM
- RESULTS: DUMPING CRITERION IN THE FORM OF PRESSURE DIFFERENCE

CONDENSATION HEAT TRANSFER - NORTHWESTERN UNIVERSITY

- OBJECTIVES: STEAM WATER CONDENSATION RATE FOR VARIOUS FLOW CONFIGURATIONS
- APPLICATION: CONDENSATION COEFFICIENT EQUATIONS AS INPUT TO THE CODES FOR STEAM GENERATOR, ECCS AND TIE-PLATE
- RESULTS
  - HEAT TRANSFER CORRELATIONS FOR VERTICAL FLOW
  - HEAT TRANSFER CORRELATIONS FOR HORIZONTAL FLOW
  - WATER FLOW RATE THROUGH TIE-PLATE
  - BREAKUP OF SPRAY IN STEAM

PHASE DISTRIBUTION AND THERMAL HYDRAULIC INSTABILITY - RPI

. OBJECTIVES: 1. ESTABLISH MODELS FOR PHASE DISTRIBUTION/ SEPARATION FOR VARIOUS GEOMETRIES OF FLOW CHANNELS

2. ESTABLISH THERMAL HYDRAULIC INSTABILITY MODEL

. APPLICATION: 1. PHASE DISTRIBUTION/SEPARATION MODELS AND DATA FOR ADVANCED CODES (COBRA-TF, TRAC, RELAP5)

2. INSTABILITY BOUNDARY DURING TRANSIENTS FOR LICENSING CODE DURING TRANSIENTS

. RESULTS

- PHASE DISTRIBUTION DATA FOR TRIANGULAR AND 2D CHANNEL
- PHASE SEPARATION DATA FOR TEE
- NON-LINEAR MODEL FOR INSTABILITY ANALYSIS

OVERVIEW OF THE SEISMIC SAFETY MARGINS RESEARCH PROGRAM

Presented By:
Dr. Richard G. Dong

Lawrence Livermore National Laboratory

Presented At:
8th Water Reactor Safety Research
Information Meeting
Sponsored By:
U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Reactor Safety Research
Gaithersburg, Maryland
October 30, 1980

SUMMARY

This overiew presentation describes the Seismic Safety Margins Research Program (SSMRP), why it is needed, the computational procedure used, the technical disciplines involved, and the outline of presentations at this session covering the research undertaken to date. The program is sponsored by the U. S. Nuclear Regulatory Commission (NRC), Division of Reactor Safety Research, Office of Nuclear Regulatory Research. The SSMRP was started in mid-1978, and Phase I is scheduled to be completed January 1, 1981.

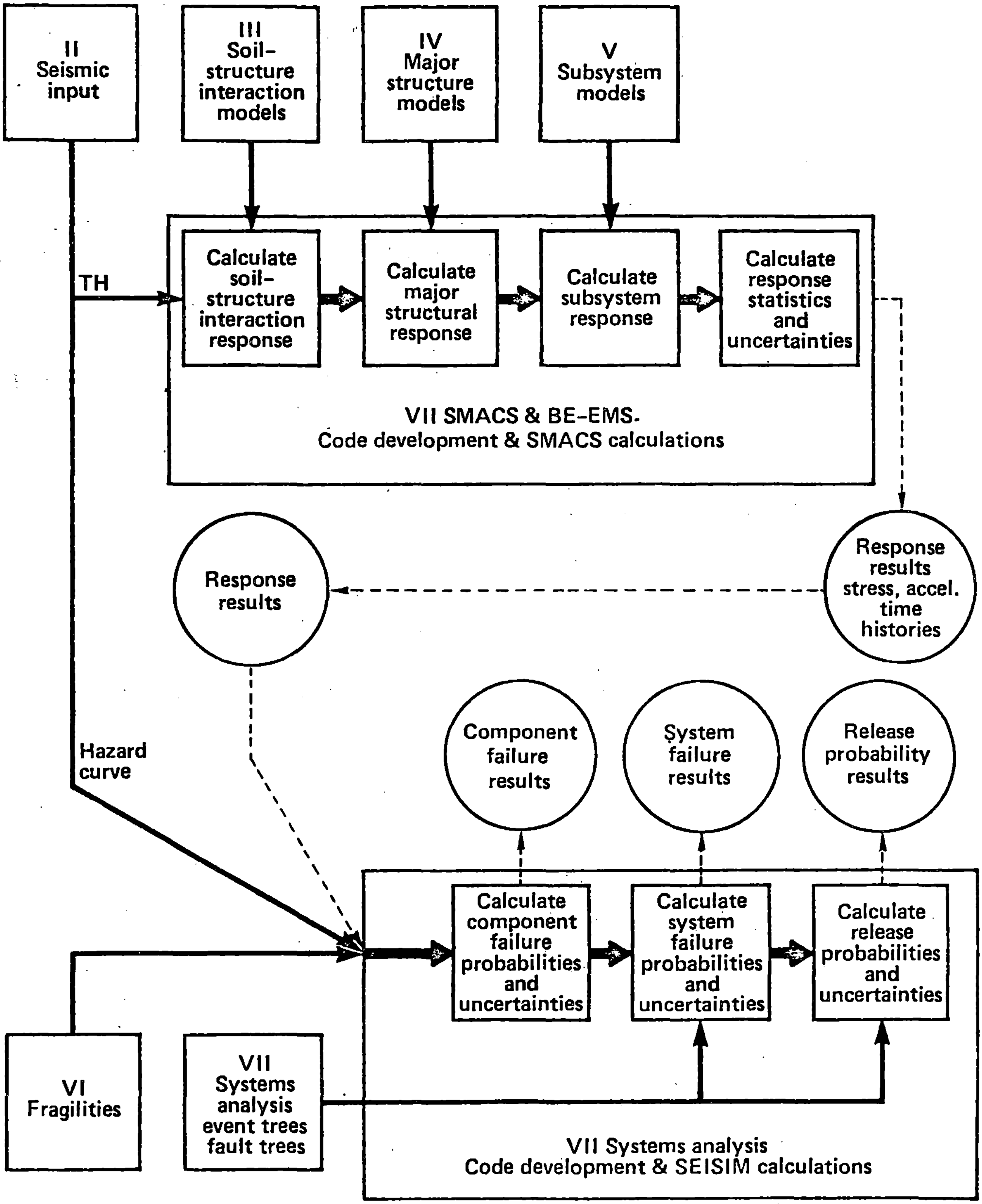
Technical areas involved with the seismic design of nuclear power plants include seismic input, soil-structure interaction, major structural response, subsystem response, and strength capacity determinations. In the usual approach these are handled separately with individual factors of safety. In the SSMRP these are addressed together in a systems analysis approach, and the figure of merit for safety evaluation is risk rather than factors of safety. This research was needed to address several important issues: (1) conventional design procedure is believed to lead to compounding conservatisms, (2) risk may ultimately be a more useful figure if merit for safety evaluation, (3) a systems approach can identify and rank the important contributors to public risk, (4) a more rational basis for allocating resources for future research is possible through a systems approach, and (5) probabilistic assessment methods do not imply design methods are not deterministic.

The SSMRP consists of eight technical projects. The NRC technical monitors and the Lawrence Livermore National Laboratory (LLNL) program staff members are:

		NRC	LLNL
Program Manager		J. E. Richardson	P. D. Smith
Deputy Program Manager		C. W. Burger	R. G. Dong
Projects:			
I	Plant/Site Selection and Data Collection	G. Bagchi	T. Y. Chuang
II	Seismic Input	R. J. Brazee	D. L. Bernreuter
III	Soil-Structure Interaction	J. F. Costello	J. J. Johnson
IV	Major Structural Response	C. W. Burger	J. J. Johnson
V	Subsystem Response	J. J. Burns	T. Y. Chuang
VI	Fragilities	J. J. Burns	M. P. Bohn
VII	Systems Analysis	J. J. Burns	G. E. Cummings/ J. E. Wells
VIII	SMACS and BE-EMS	C. W. Burger	J. J. Johnson

The attached flow chart of the probabilistic computational procedure identifies the role of each project. The sequence of presentations at this session is aimed at providing a cohesive picture of how the diversed technical areas come together to form the computational procedure. Accordingly, we begin with a presentation on systems analysis, the technical effort which directly evaluates risk. Feeding into systems analyses are seismic hazard curves, fragilities and responses. Consequently, presentations on these three areas follow. Feeding into response calculations are analytical models for soil-structure interaction, major structures, and sybsystems, and seismic time histories. Therefore, presentations on these areas follow. Finally, the session ends with closing comments on results produced by the program as a whole.

A FLOW CHART OF THE PHASE I PROBABILISTIC COMPUTATIONAL PROCEDURE IDENTIFIES THE ROLE OF EACH PROJECT



Steam Generator Flow Patterns and Modeling

Peter Griffith
Professor of
Mechanical Engineering
MIT

Christopher Calia
Research Assistant
MIT

Der-Yu Hsia
Former Student
MIT

Present Address

Institute of Nuclear Energy
Tiawan

Water Reactor Safety Information Meeting

Gaithersberg, Md.

October 1980

Introduction - Predicting how nuclear steam generators will operate during loss-of-coolant accidents is important for two reasons:

1) The steam generator can provide a flow resistance from the core to the break which can alter the re-flooding rate.

2) The steam generator can serve as a heat sink by carrying away the decay heat.

For this reason, it is important to know how to predict both the flow and the pressure drop through nuclear steam generators for all pressures and water inventories at steam flow rates corresponding to decay heat levels. Prediction is complicated by four factors.

1) The amount of, and behavior of, non-condensible gases is not known. (We can make some recommendations here).

2) The multiple inverted U tube array is subject to a pressure drop versus flow rate instability. (We know how to calculate this).

3) How to model the two phase flow in the inlet plenum of the steam generator is still not known.

4) The possibility exists that a reactor system with two or more loops might suffer a loop-to-loop pressure-drop versus flow rate instability. (An experiment is needed here).

Reference (1) reports on a study of a small four tube inverted U tube condenser in which condensation occurred on the primary side. The condenser was constructed out of glass tubes of 6.4 mm ID of about 2.4 m total length. Fig. 2.2 shows the apparatus. Fig. 3.3e shows the flow configuration with one tube stalled. Figure 3.4 shows the four modes of operation for the array and the accompanying pressure drops. Figure 4.7 shows the effect of non-condensibles on the pressure drop across the array, while

Fig. 4.9 shows how well this pressure drop can be calculated. Conclusions are as shown following Fig. 4.9.

Reference (2) reports a study of pressure drop vs. flow rate instabilities in a four tube inverted U tube array passing air and water. The objective of this work was to develop a method of calculating the effective heat transfer area and pressure drop across a steam generator when heat was being transferred from the secondary to the primary as might occur during a large break LOCA.

A schematic of the experimental apparatus is shown in Figure 13. Figure 15 shows how the two phases enter the inlet plenum. The variety of flow regimes observed in the four tubes is shown in Figure 21. Pressure drop to the array is shown in Figure 29. Liquid distribution for the four tubes is shown in Figure 30. The several flow regimes are illustrated in Figure 40. One of the overall pressure drops vs. flow rate curves is shown in Figure 45. The recommended method of calculating the pressure drop - flow rate curve is illustrated in Figure 48. The unstable (some tubes not flowing regions are illustrated in Figure 49 along with a wide variety of unsuccessful calculations. The conclusions are as shown.

References

1) Calia, Christopher, "Modes of Circulation in an Inverted U-Tube Array with Condensation", MS Thesis in Mechanical Engineering, 1980, (to appear as an NRC report with P. Griffith as co-author).

2) Hsia, Der-Yu, "Steam Generator Flow Instability Modeling", PhD Thesis in Nuclear Engineering, MIT, May 1980 (to appear as an NRC report with P. Griffith as co-author).

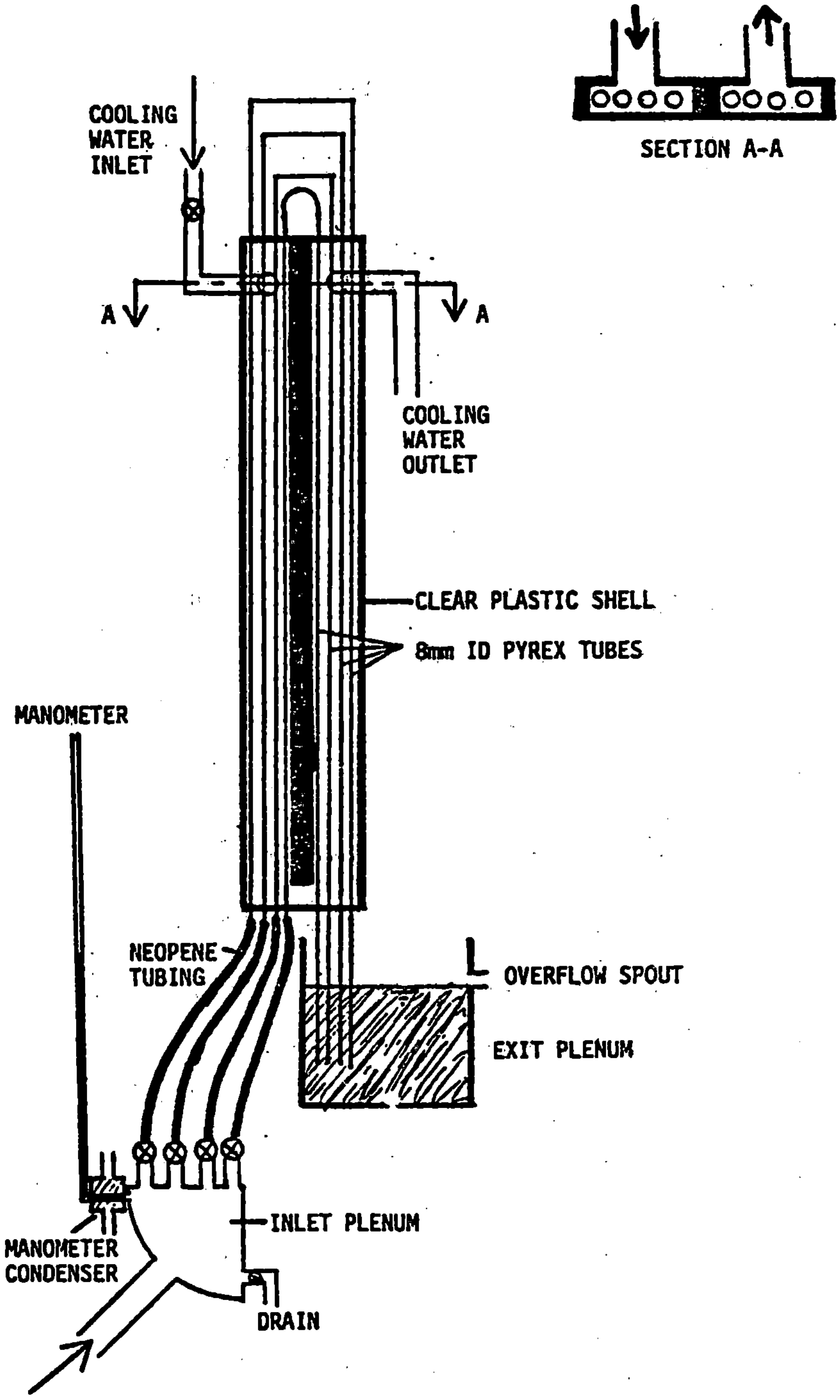


FIGURE 2.2 - THE FOUR TUBE CONDENSER

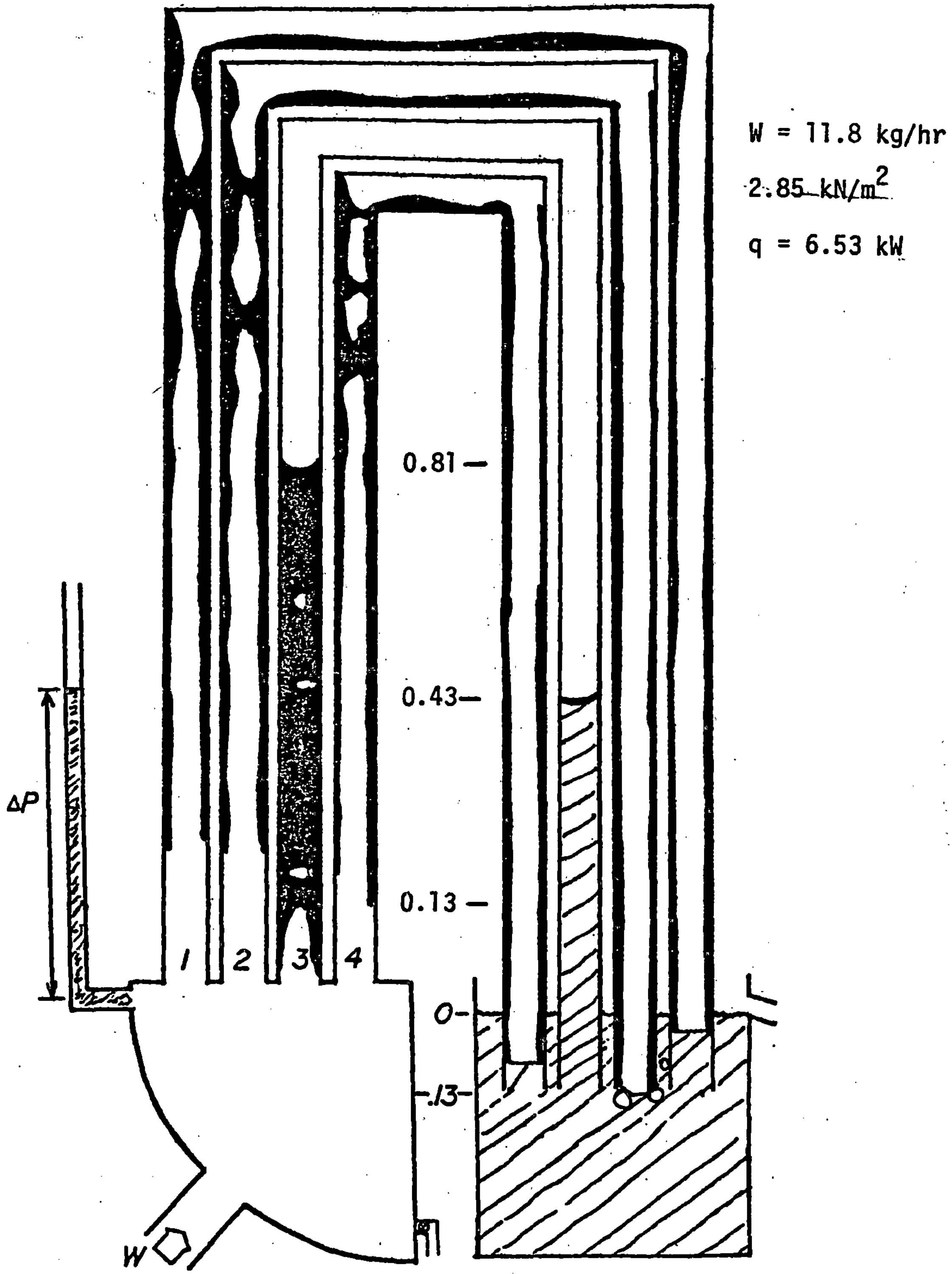


Figure 3.3e - Tube 3 has stalled. The other tubes remain in the churn flow mode.

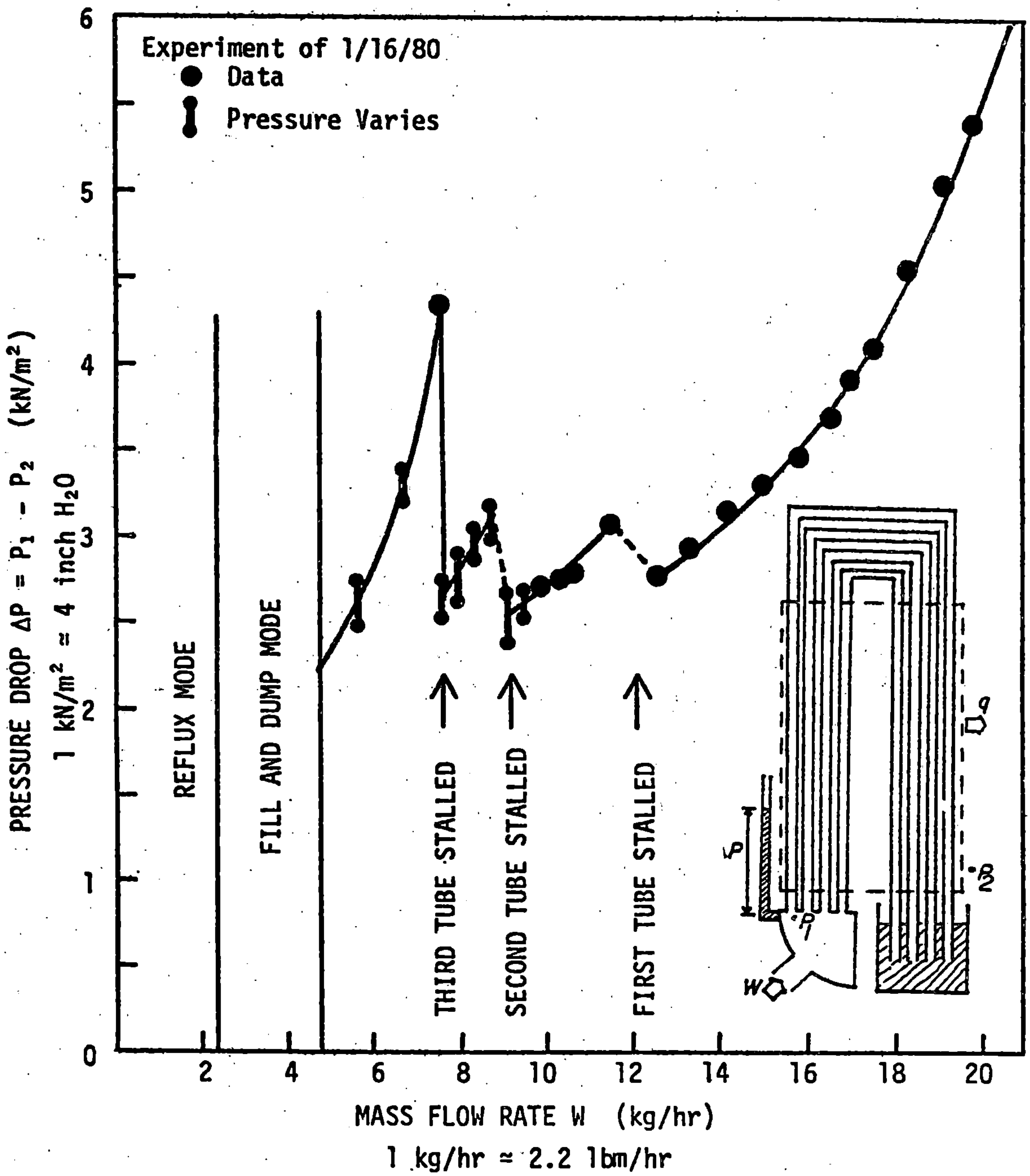


Figure 3.4- Pressure drop vs. flow rate for the four tube inverted U-tube condenser. Flow rate was started at a high value and reduced slowly.

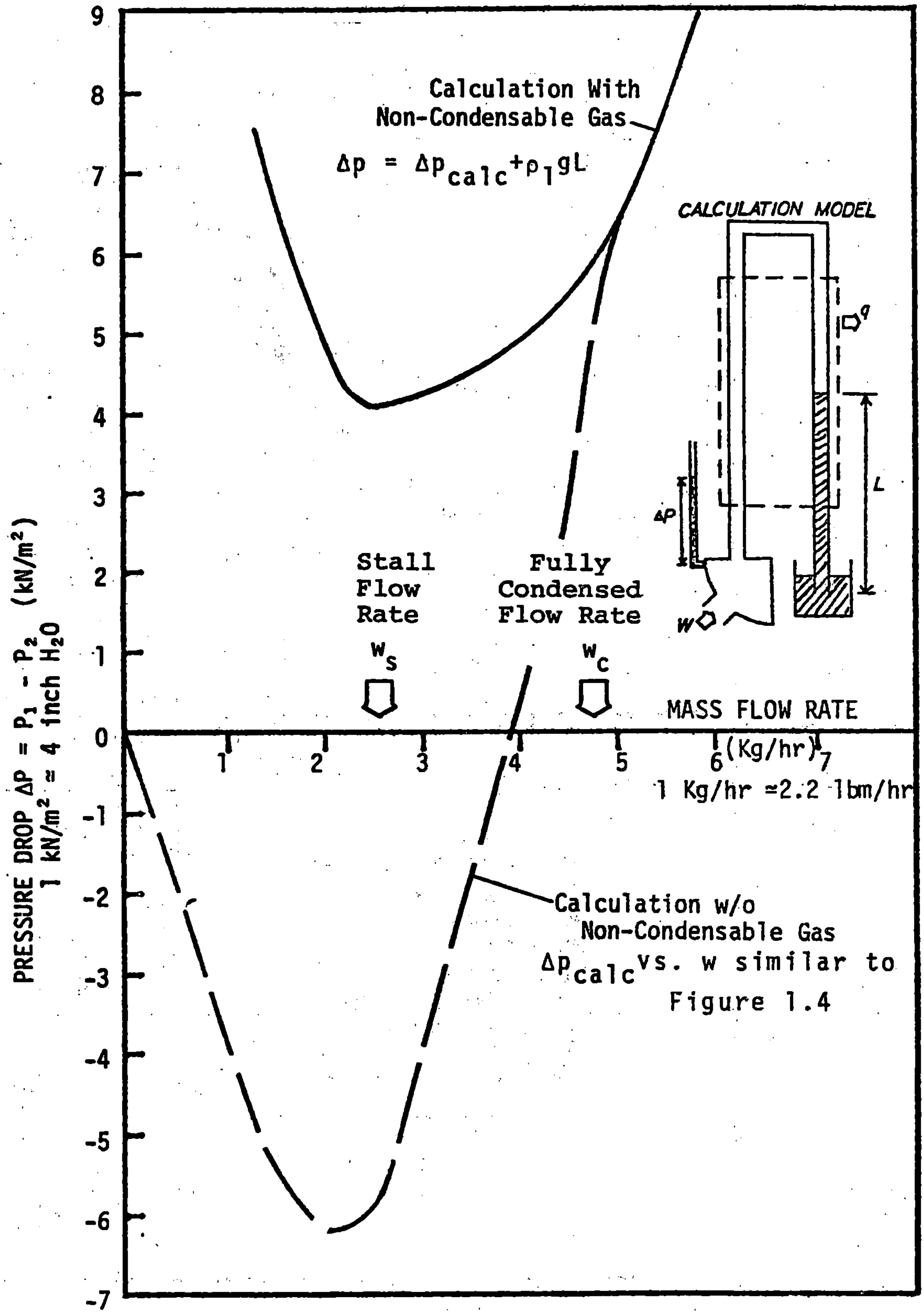


Figure 4.7 - Calculated pressure vs. flow rate characteristic for a single tube in the four tube inverted U-tube condenser. The dotted line represents the characteristic for a system with no non-condensable gas. The solid line line accounts for the non-condensables replacing the water column in the downcomer.

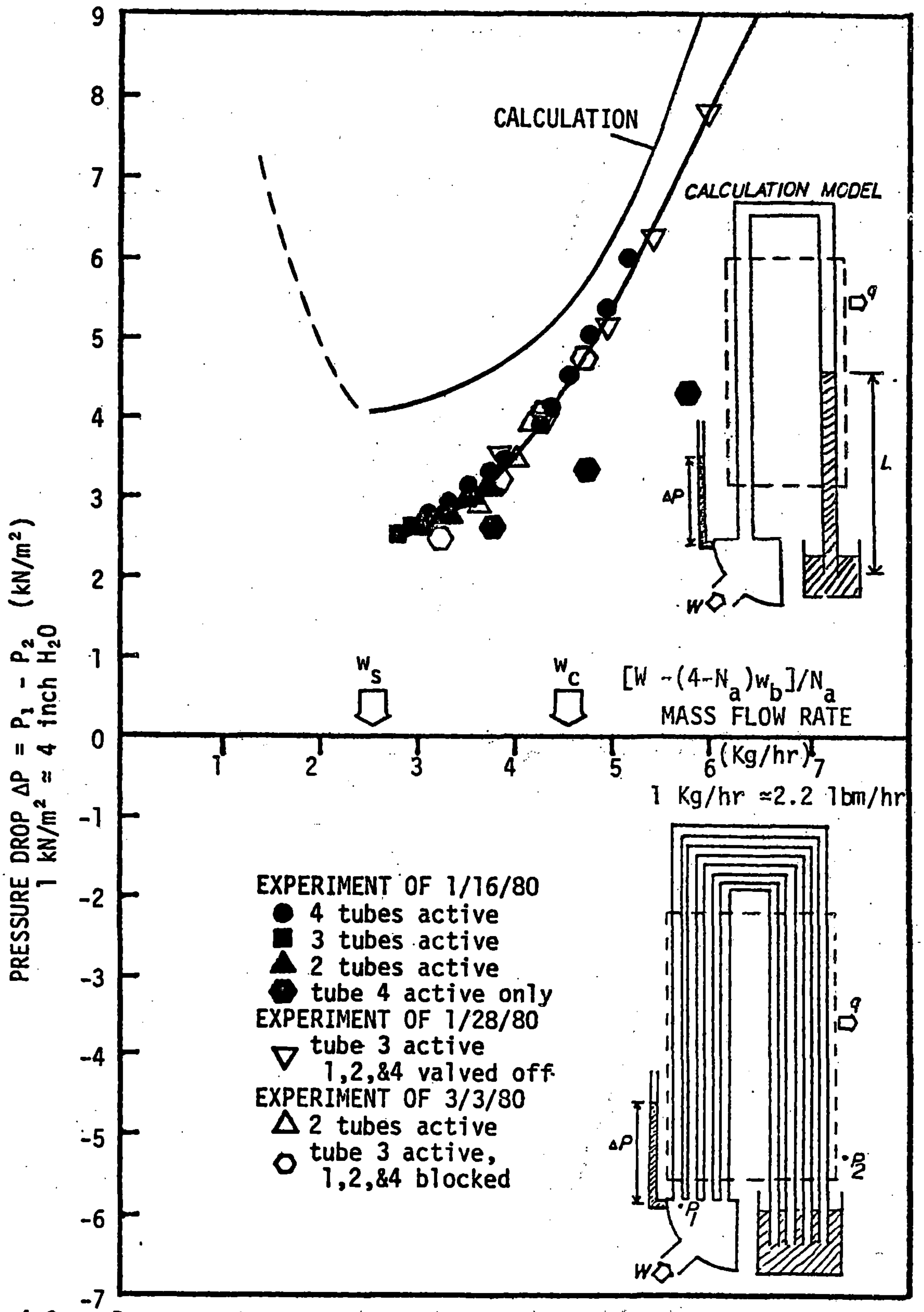


Figure 4.9 - Pressure drop vs. average tube flow rate per active tube with flow rate adjusted to account for blocked tube flow as in equation (23).

CONCLUSIONS- REFERENCE (1)

(1) FOUR MODES OF CONDENSATION ARE POSSIBLE IN AN INVERTED U-TUBE ARRAY.

A) ALL TUBES HAVE FLOW

B) SOME TUBES HAVE FLOW, SOME ARE BLOCKED

C) ONE OR MORE TUBES OPERATE IN A FILL AND DUMP CYCLE

D) THE TUBES ARE IN THE REFLUX MODE

(2) PRESSURE DROP AND HEAT TRANSFER FOR ALL THESE REGIONS CAN BE CALCULATED WITH THE RECOMMENDED TOOLS.

(3) THE SIZE OF THE REFLUX REGION IS LARGE ENOUGH SO THAT FROM 1 TO 6% OF RATED POWER (DEPENDING ON PRESSURE) CAN BE REJECTED FOR CE AND W RE- ACTORS WHEN ALL STEAM GENERATORS ARE USED.

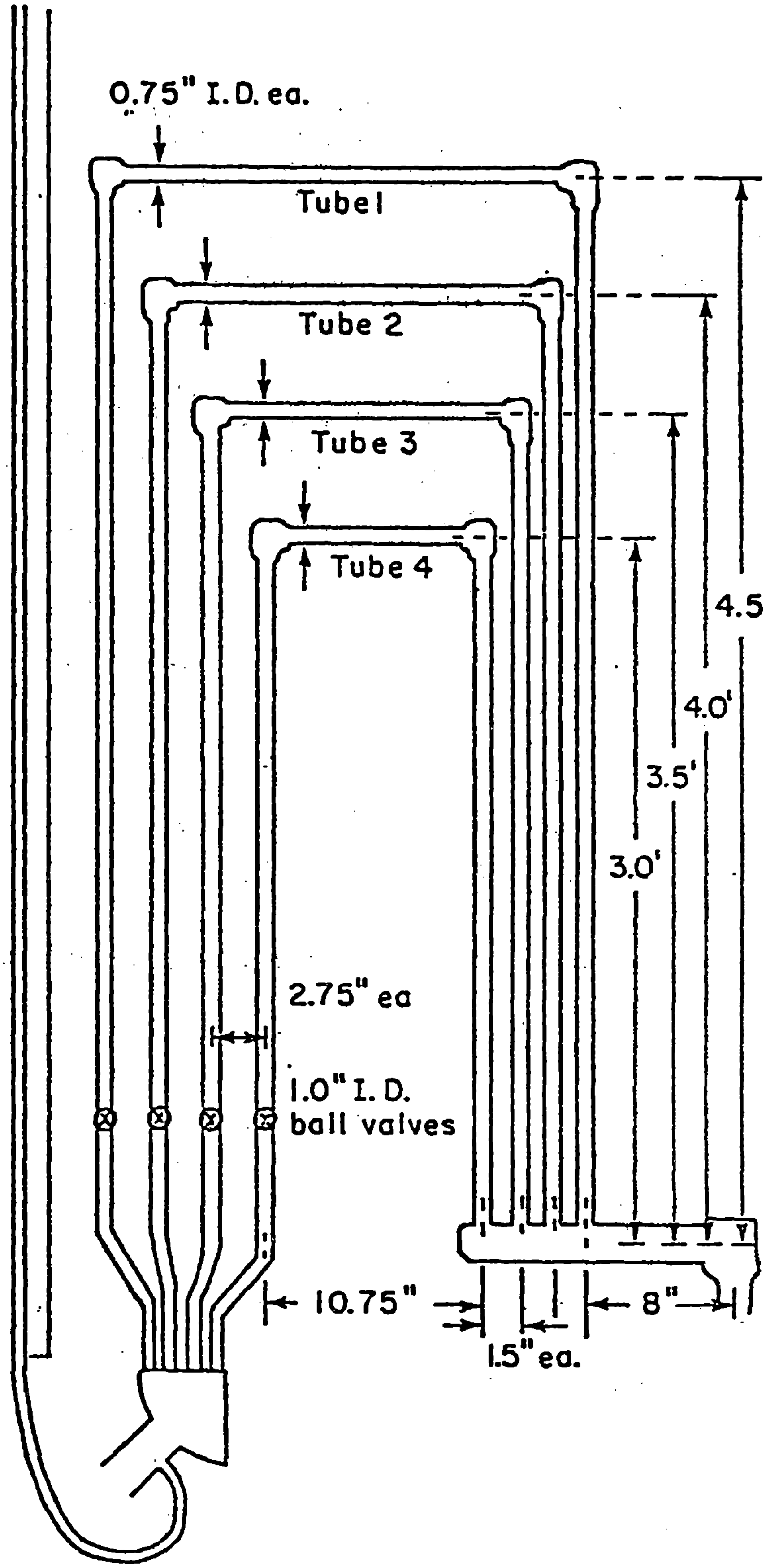


Fig. 13. Schematic of the Experimental Rig

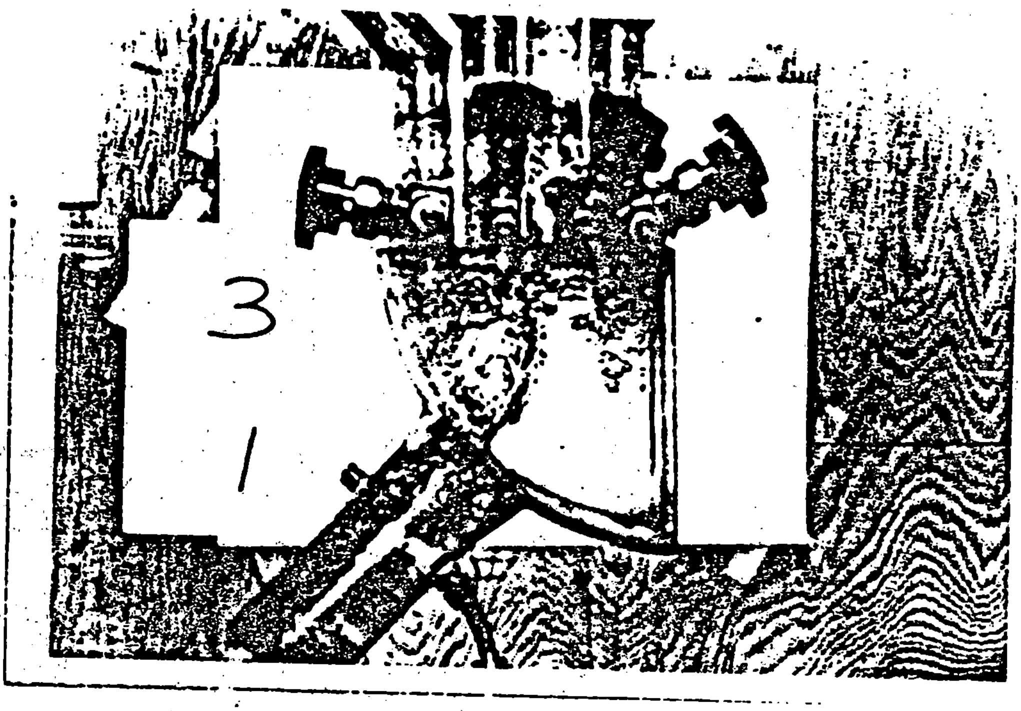
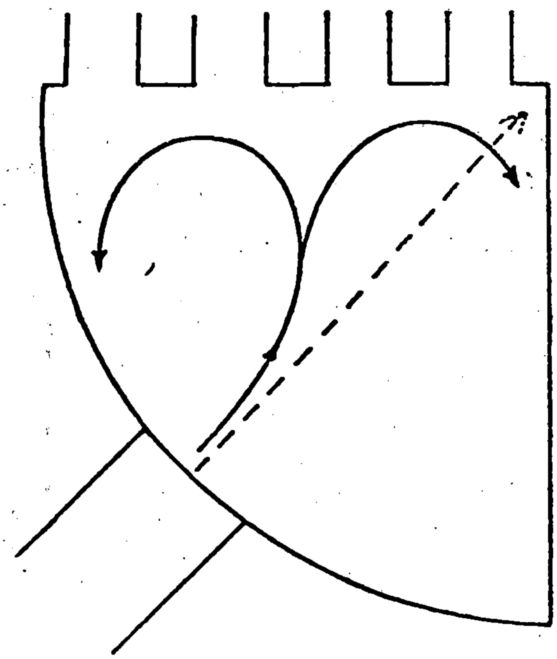


Fig. 15. Photograph(above) and sketch(below) of the flow distribution in the plenum.



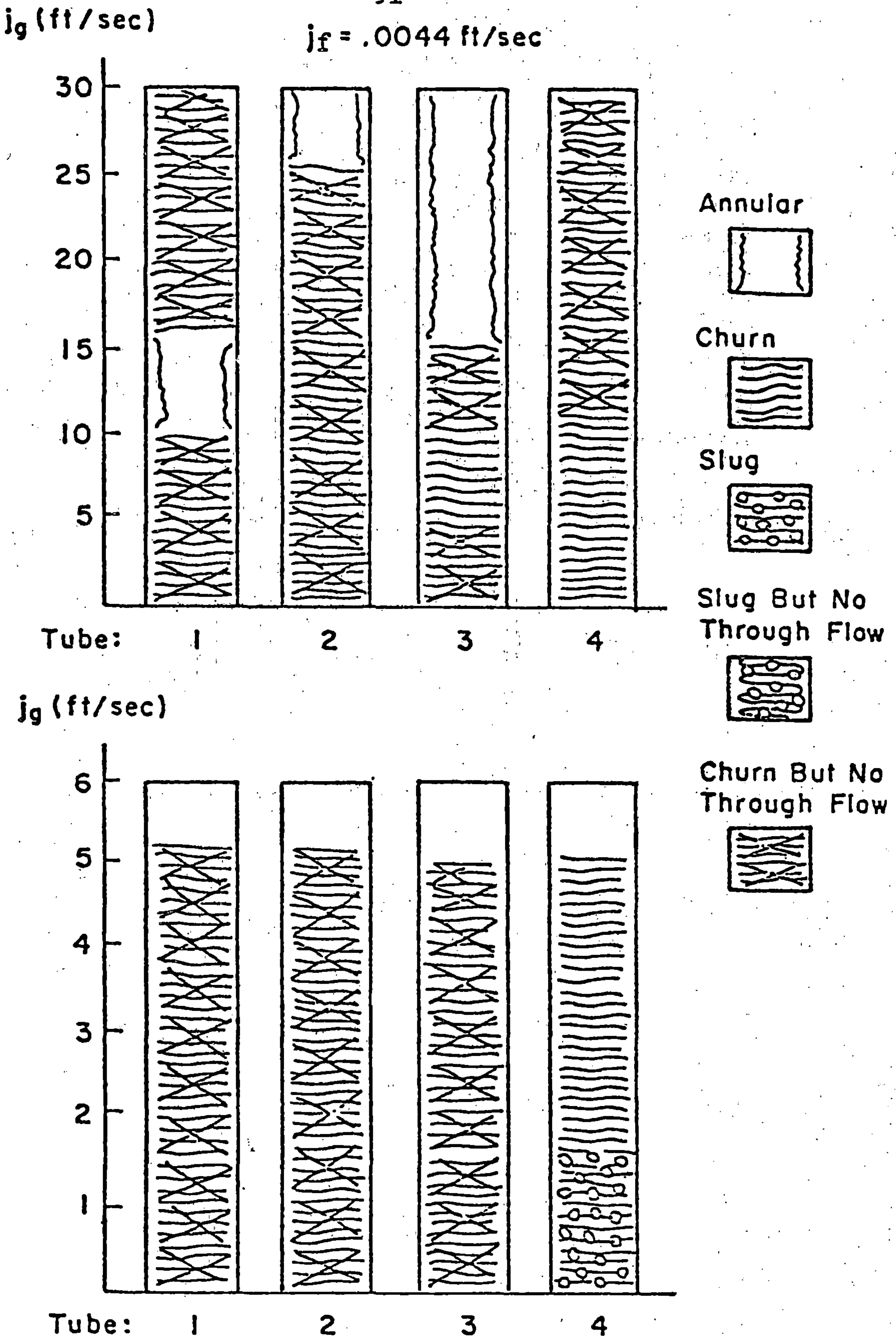


Fig.21. Flow Regime Mapping

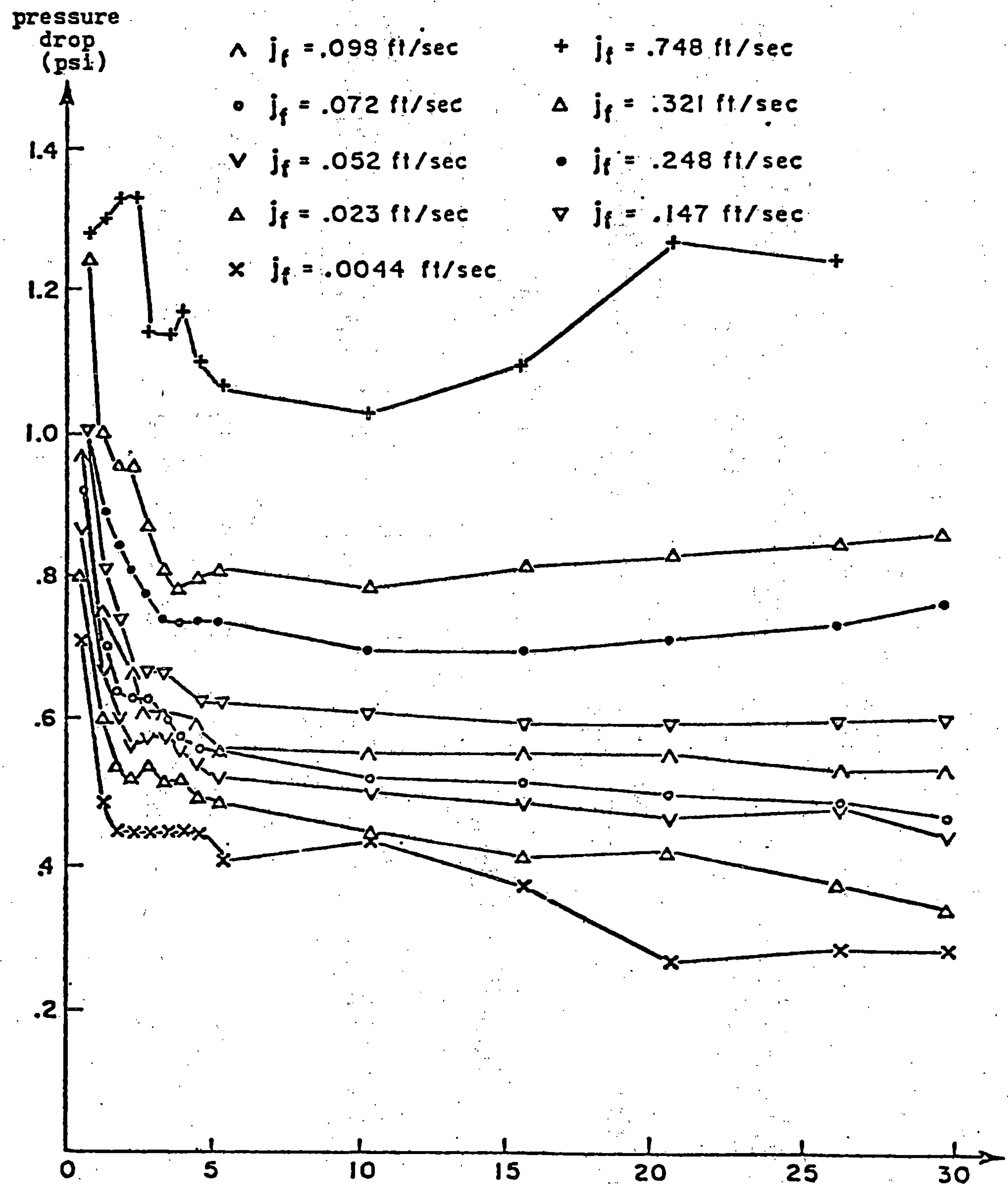


Fig.29. The Overall Pressure Drop for Four Tubes On Top Of a Inlet Plenum. Inlet j_g and j_f Were Used.

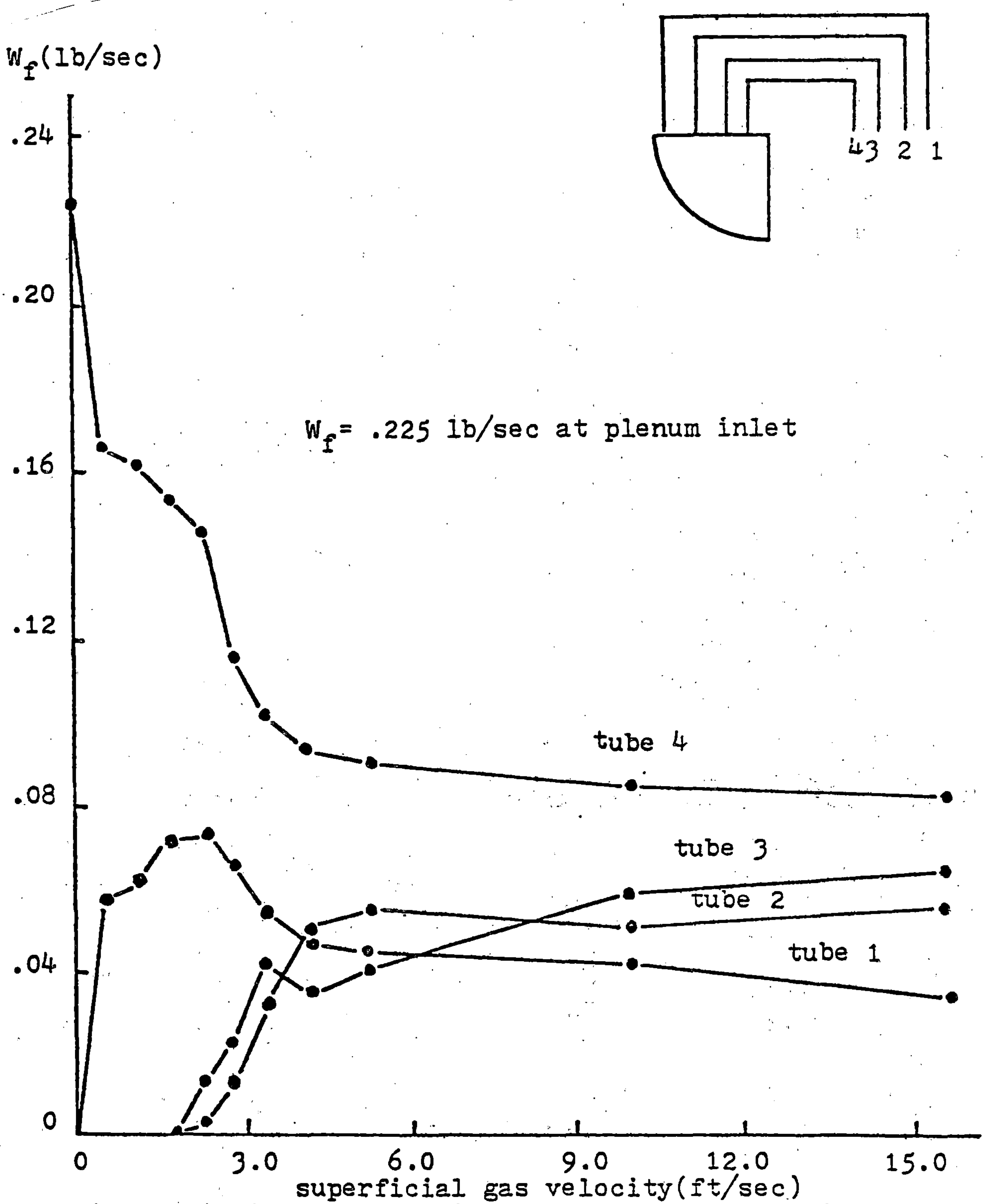


Fig.30 Liquid Flow Rate in Each Tube As a Function of Plenum Inlet Superficial Gas Velocity

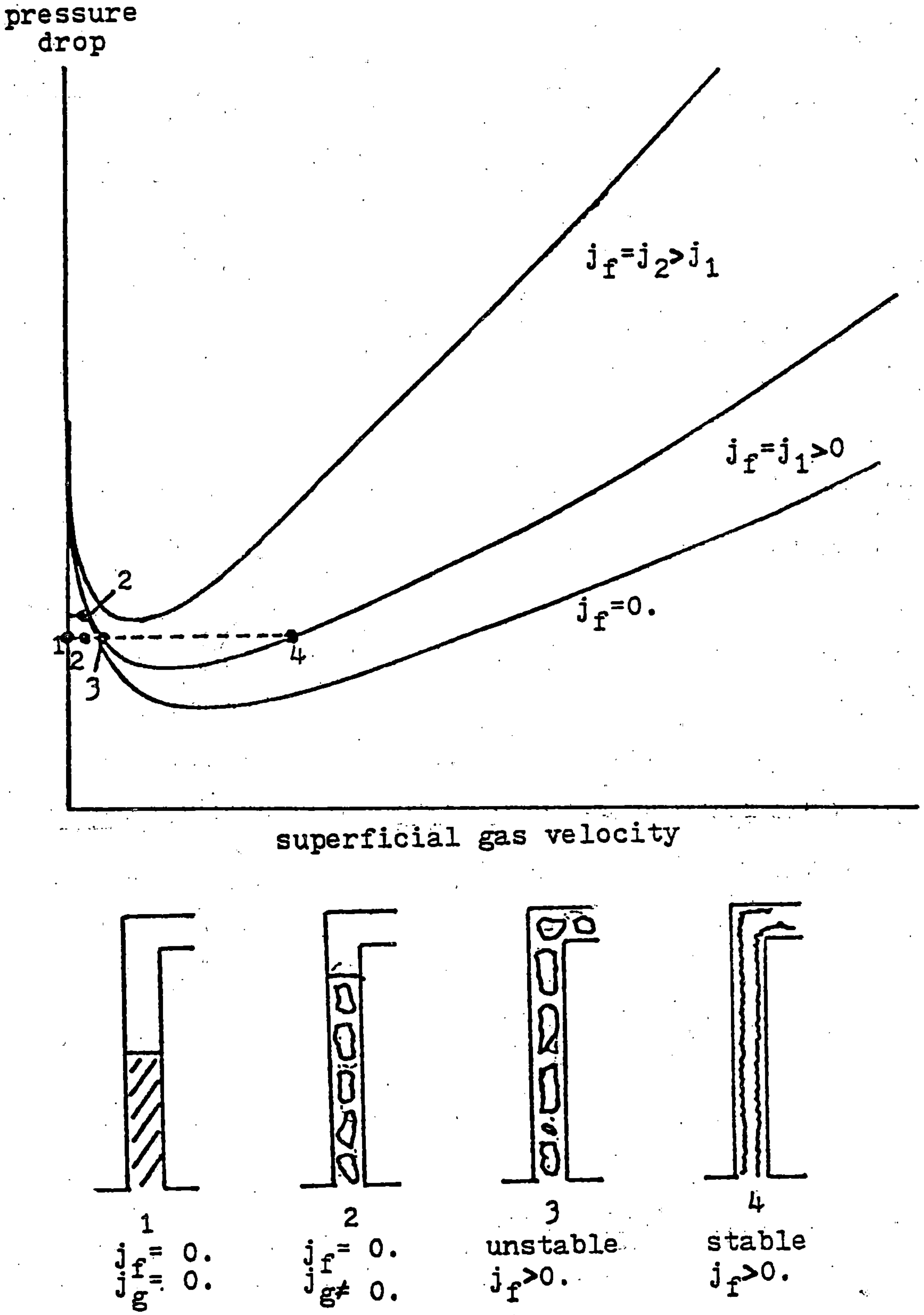


Fig.40. Illustration of the existence of several possible flow patterns at fixed pressure drop.

CONCLUSIONS - REFERENCE (2)

1. THE EFFECT OF FLOW BLOCKAGE IN THE STEAM GENERATOR IS TO DECREASE (NOT INCREASE) THE PRESSURE DROP ACROSS IT.

2. OVERALL PRESSURE DROP IN THE STEAM GENERATOR CAN BE PREDICTED AS WELL AS SINGLE TUBE PRESSURE DROPS USING THE INDICATED METHODS.

3. CRITERIA FOR DETERMINING WHEN FLOW MAL-DISTRIBUTION WILL OCCUR ARE COMPLETELY INADEQUATE.

4. NO CODE CAN PREDICT PLENUM PHASE DISTRIBUTIONS.

5. THE LOSS OF AREA IN THE STEAM GENERATOR DUE TO BLOCKAGE OF SOME TUBES IS OF LITTLE OR NO CONSEQUENCE AS SO MUCH AREA REMAINS ANYWAY.

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16. ABSTRACT *(200 words or less)*

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