August 26, 1996

MEMORANDUM TO: Charles Ader, Chief
Accident Evaluation Branch
Division of Systems Technology
Office of Nuclear Regulatory Research

THRU: Charles Tinkler
Accident Evaluation Branch
Division of Systems Technology
Office of Nuclear Regulatory Research

FROM: Richard Lee
Accident Evaluation Branch
Division of Systems Technology
Office of Nuclear Regulatory Research

SUBJECT: SUMMARY OF MEETING AUGUST 19-20, 1996 MEETING AT FAUSKE AND ASSOCIATES, INC., IN BURR RIDGE, ILLINOIS.

Enclosed is a summary of the meeting on the review of the SCDAP/RELAP5 code modeling of natural circulation under severe accident conditions, held on August 19-20, 1996, at Fauske & Associates, Inc., in Burr Ridge, Illinois.

Enclosure: As stated

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Enclosure

Meeting Summary

SCDAP/RELAP5 code modeling of natural circulation under severe accident conditions

August 19-20, 1996
Fauske & Associates, Inc.
16W07 West 83rd Street
Burr Ridge, IL

Participants:

R. Viskanta, Purdue University
M. Ishii, Purdue University
P. Griffith, Massachusetts Institute of Technology
C. Ader, C. Tinkler, R. Lee, J. Donoghue, J. Staudenmeier, USNRC
D. Knudson, E. Harvego, P. Bayless, Idaho National Engineering Laboratory
I. Catton, ACRS-NRC
M. Epstein, R. Henry, Fauske & Associates

Note: Reviewers are Viskanta, Ishii and Griffith (consultants of Energy Research, Inc.). Catton participated as an observer for ACRS. R. Henry participated as a representative of the nuclear industry.

Summary:

The meeting commenced at 9:00 a.m. on August 19, 1996, with the introduction of the agenda by R. Lee (Attachment 1). An introduction on the purpose of the meeting, and background on the SCDAP/RELAP5 (SR5) modelling of natural circulation under severe accident conditions was provided by C. Tinkler (Attachment 2). I. Catton stated his views on the scaling of the Westinghouse 1/7 scale natural circulation experiments, and how uncertainties should be treated in estimating risk for steam generator tube failures. P. Bayless presented the background on the development of the SR5 model (Attachment 3). Next, R. Henry gave a presentation on the Westinghouse 1/7 scale experiments (Attachment 4). After, Henry's presentation, Bayless returned to present the benchmarking of SR5 against the Westinghouse 1/7 scale experiments, and discussed the use of SR5 to analyze natural circulation in the Surry plant (Attachment 3). Next, D. Knudson presented the most recent SR5 analyses of the Surry plant (attachment 5). Throughout these presentations, discussions took place among the reviewers and participants on subjects presented. On August 20, 1996, J. Donoghue discussed how the SR5 analysis results are being used in estimating the risk associated with steam generator tube failure (Attachment 6). After some discussion, the reviewers caucused among themselves. Thereafter, P. Griffith provided their preliminary comments on the meeting. The reviewers stated their view that (a) the experimental (Westinghouse 1/7

scale experiments) data was good and that the experiment was well designed, and additional experimental data was not needed; and (b) the code (SR5) was certainly adequate for the job (i.e., to calculate natural circulation under severe accident conditions), the implementation of the code was good and the constitutive relationships used by SR5 were adequate. They recommended additional activit for the purpose of demonstrating more clearly the adequacy of the modeling {i.e., establishing a "figure of merit" by summarizing the experimental data and the analytical (SR5 calculations) results in a fashion (e.g., temperature vs. time} to show that the experimental data and the analytical results give the same systematic overall behavior in the reactor system being studied. P. Griffith went on to state his additional view that once a "figure of merit" was established, it would be worthwhile to perform a few sensitivity calculations for parameters which may vary widely {e.g., a factor of 2, affecting the heatup rates of different reactor components (e.g., surge line, hot leg, steam generator tubes)}. After a brief discussion to clarify some of the reviewers' comments, the meeting was adjourned around 1:00 p.m.

Attachments: As stated

ATTACHMENT 1

SCDAP/RELAP5 code modeling of natural circulation
under severe accident conditions

August 19-20, 1996
Fauske & Associates, Inc.
16W07 West 83rd Street
Burr Ridge, IL

AGENDA

August 19, 1996

a)	9:00 a.m.		Opening Remarks	NRC
b)			**SCDAP/RELAP5 (SR5) modeling of natural circulation and steam generator (SG) tube heating in a PWR: Model development and assessment, and scaling issues**	
	9:15 a.m.	•	development of SR5 model - in-vessel natural circulation - hot leg countercurrent flow - SG inlet plenum mixing - heat transfer modeling (in SG hot leg, surge line)	INEL
	10:45 a.m.		Break	
	11:00 a.m.	•	W 1/7 scale experiments	FAI
	11:45 a.m.		Discussion	Reviewers
	12:15 p.m.		Lunch	
	1:15 p.m.	•	SR5 assessment	INEL
	2:15 p.m.		Discussion	Reviewers
	3:00 p.m.		Break	
c)			Application of SR5 for PWR analyses	
	3:15 p.m.	• •	Surry modelling uncertainties	INEL INEL
	4:15 p.m.		Discussion	Reviewers
	5:00 p.m.		Adjourn	

August 20, 1996

d) Application of SR5 foı PWR analyses (continued)

9:00 a.m.	•	Most recent SR5 application for PWRs	INEL
10:00 a.m.		Discussion	Reviewers
10:30 a.m.		Break	
e) 10:45 a.m.	•	A perspective on the use of SR5 thermal hydraulic analyses (including affects of fission products transport)	NRC
11:15 a.m.		Discussion/Comments	Reviewers
12:00 p.m.		Lunch	
1:15 p.m.		Comments/Discussion	All
2:30 p.m.		Adjourn	

Review of SCDAP/RELAP5 Modelling for Assessment of Steam Generator Tube Integrity

August 19-20, 1996

ATTACHMENT 2

REVIEW OF SCDAP/RELAP5 MODELLING OF NATURAL CIRCULATION UNDER SEVERE ACCIDENT CONDITIONS

Objective:

Obtain independent assessment of the adequacy of SCDAP/RELAP5 modelling of natural circulation under severe accident conditions for the purpose of calculating the relative timing and failure of RCS components in order to evaluate the risk associated with thermally-induced steam generator tube ruptures

Reviewers:

Consultants to Energy Research Incorporated (ERI):

Raymond Viskanta (Purdue University)
Mamoru Ishii (Purdue University)
Peter Griffith (Massachusetts Institute of Technology)

Schedule:

Review Group to provide its conclusion to ERI by 8/30/96.

REVIEW OF SCDAP/RELAP5 MODELLING OF NATURAL CIRCULATION UNDER SEVERE ACCIDENT CONDITIONS

BACKGROUND

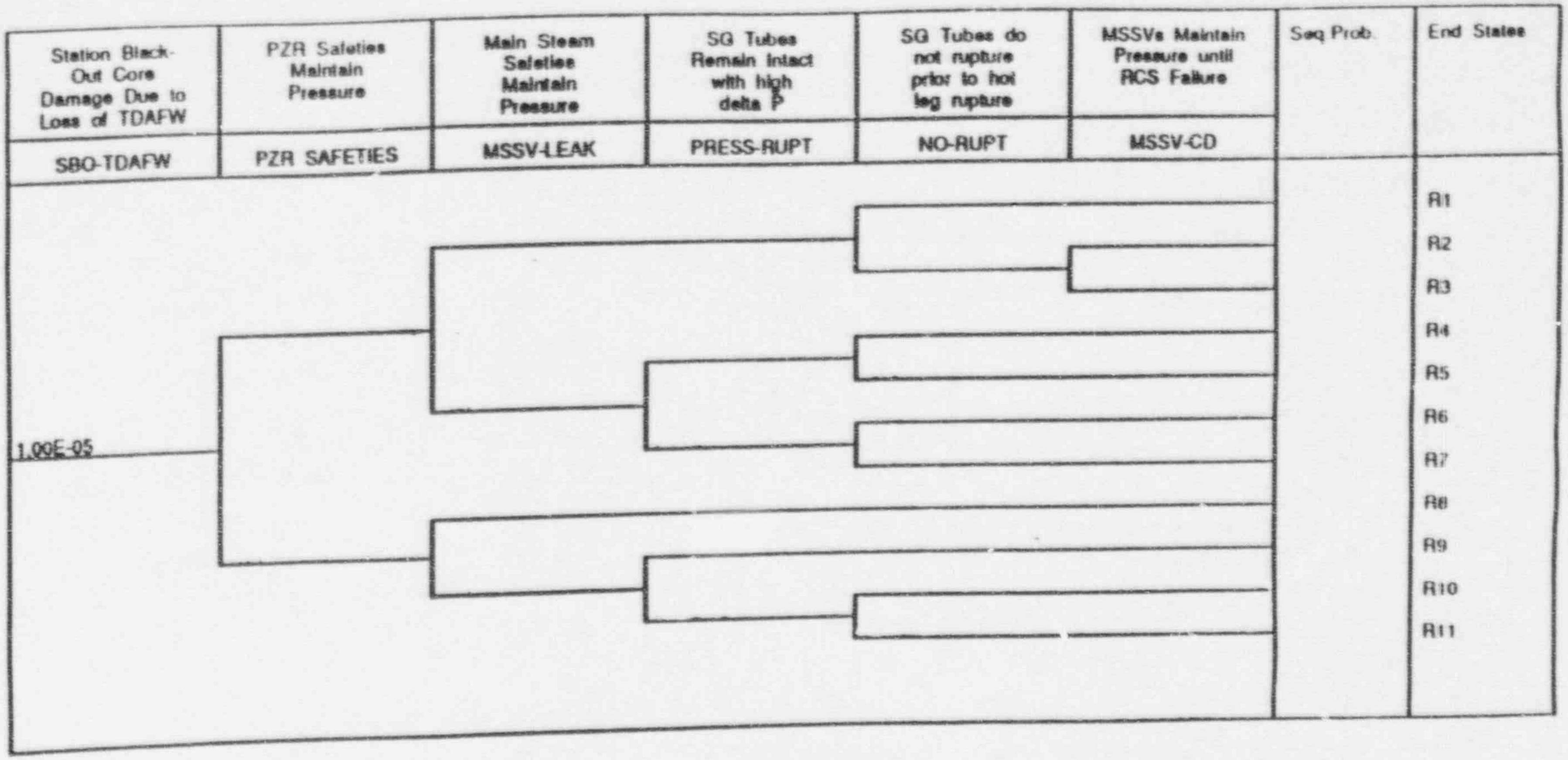
- As part of the rulemaking on steam generator tube integrity, an issue has been raised relative to the performance of flawed tubes and their likelihood of failure during a severe accident. The concern arises due to hot gases circulating through the steam generator tubes and inducing tube failure due to the elevated temperature of the tubes.

- Past SCDAP/RELAP5 (SR5) analyses performed in conjunction with DCH issue resolution assessed the relative heatup and failure (using creep rupture models) of RCS components (i.e., hot leg, surge line, unflawed SG tubes and the RPV lower head). The calculations considered a standard TMLB' sequence (with a pressurized secondary system).

- Current analyses focused on sequence with depressurized steam generator (on pressurizer loop). Greater challenge to tubes but a lower probability due to additional failures required.

- Additional analyses to address mixing/phenomenological uncertainties.

- SR5 analyses are to assist in addressing the thermally-induced SGTR. Spontaneous SGTR and pressure-induced SGTR are examined separately.



Event Tree for High-Pressure Station Blackout, Quantified to Account for Degraded Steam Generator Tubes

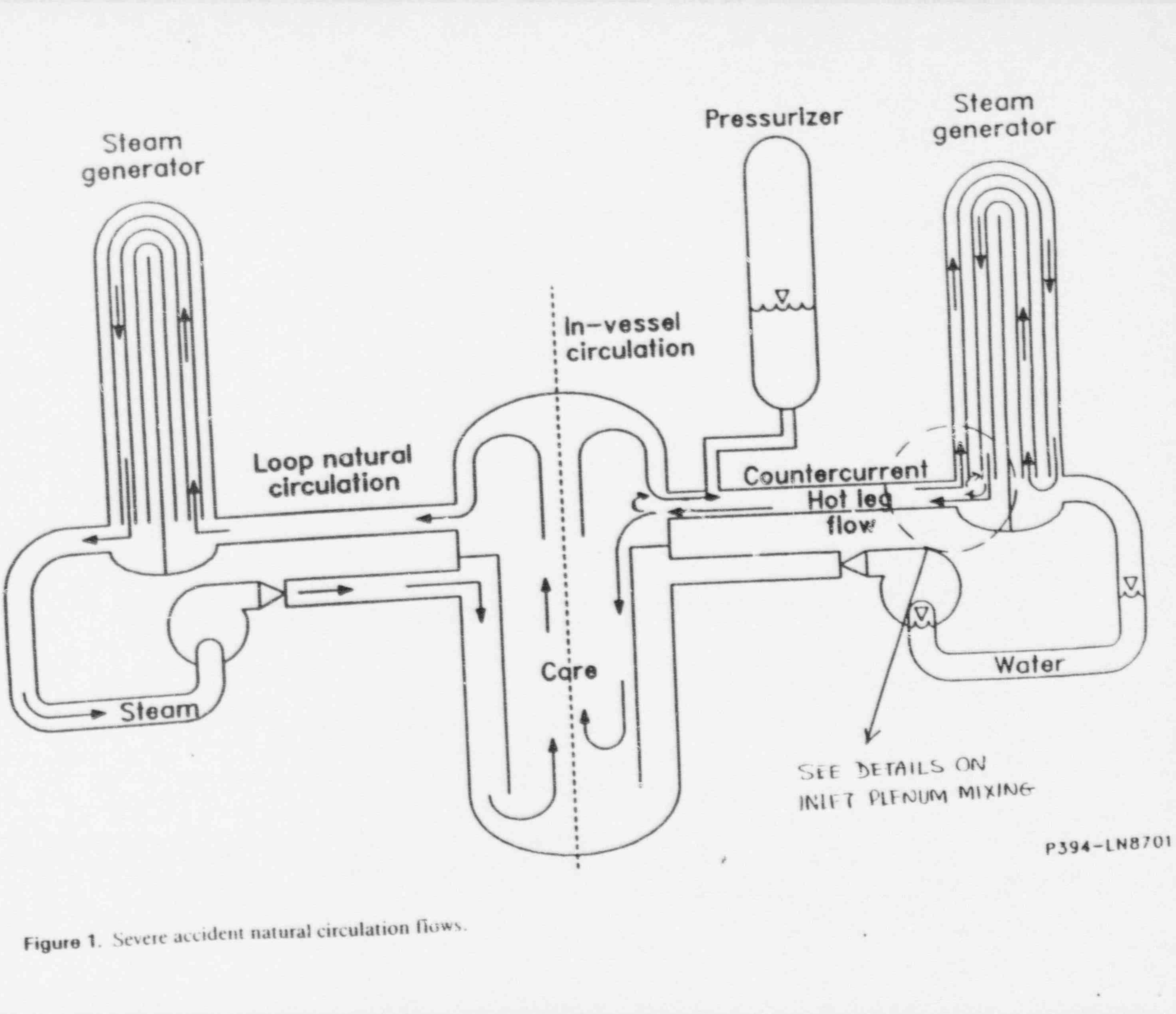


Figure 1. Severe accident natural circulation flows.

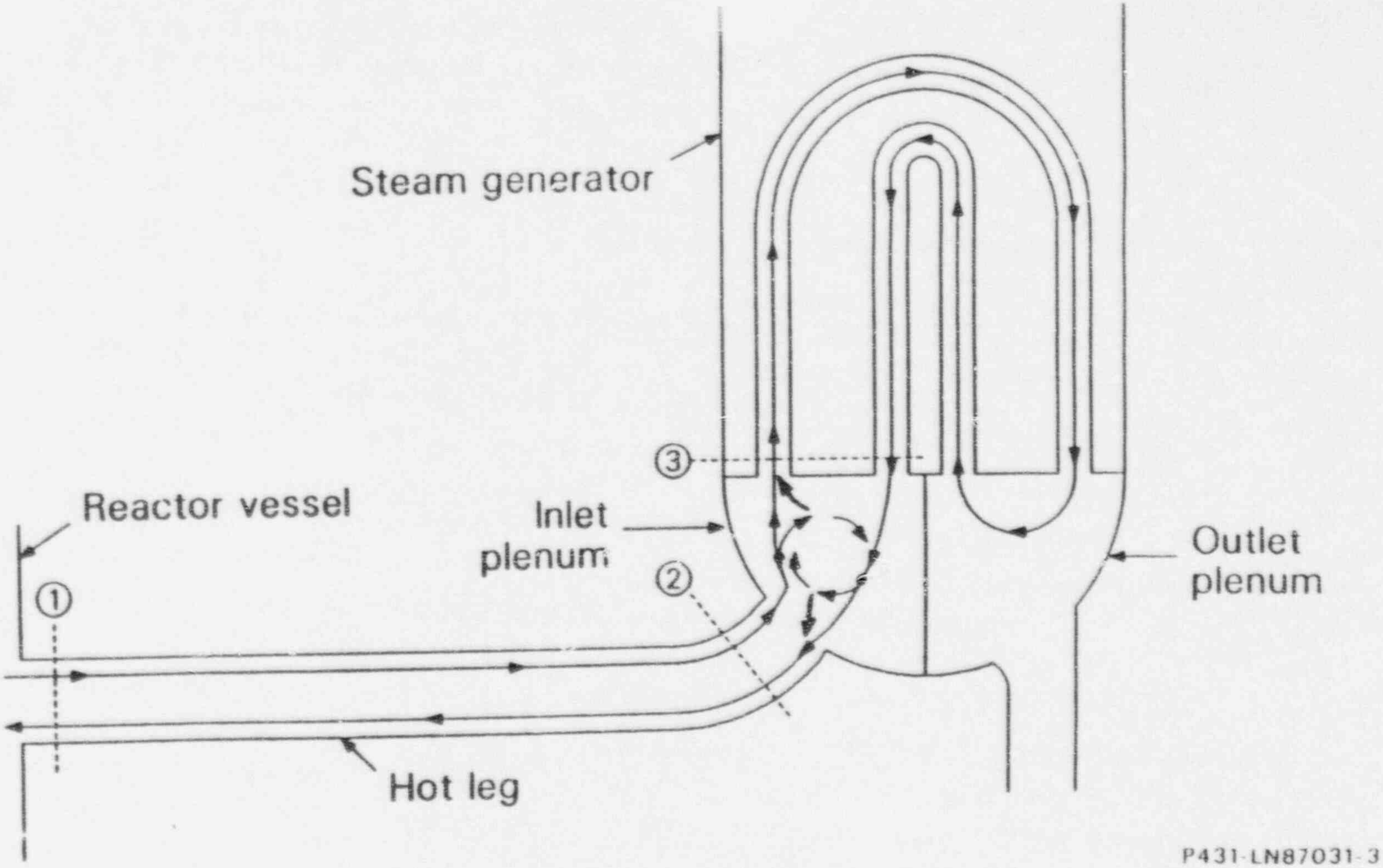


Figure 2. Hot leg natural circulation stream flows.

REVIEW OF SCDAP/RELAP5 MODELLING OF NATURAL CIRCULATION UNDER SEVERE ACCIDENT CONDITIONS

Past usage of SR5 calculations with natural circulation modelling:

- SR5 use to address natural circulation in the reactor system as one of the major areas of uncertainty identified in NURGE-0956 ("reassessment of the Technical Bases for Estimating Source Terms, 7/86). Specifically, SR5 analyses were performed for a Surry TMLB' accident (NUREG/CR-5214, 10/88)

- SR5 was also used to assess failures of ex-vessel components (hot leg, surge line, steam generator tube) vs. RPV lower head under the direct containment heating (DCH) issue resolution for PWRs. Results were peer reviewed.

  - Zion DCH issue resolution (NUREG/CR-6075, Supp. 1, 12/94),
    Surry DCH issue resolution (NURGE/CR-6109, 5/95),
    DCH issue resolution for Westinghouse plants with large dry containments or subatmospheric containments (NUREG/CR-6338, 2/96)

  - Peer reviewers: Levy-LA, Henry-FAI, Moody-GE, Modarres-Univ. of Md, Sheppard-Cal. Tech, Ishii-Purdue

- SR5 was also used to assist peer reviewers (Levy, Henry, Ishii, Moody. Corradini) in establishing initial conditions (e.g., melt mass and composition) for the CE DCH testing.

Summary of Results

- Plant analysis performed for representative designs
  - Surry (W)
  - ANO-2 (CE)
- Analyses consistently showed that for countercurrent flow severe accident conditions first failures occurred at surge line or hot leg nozzle
  - Surge line or hot leg failure occurred 20-40 minutes before SG tube failure
  - Sensitivities done on T-H modelling did not alter finding on tube integrity

SCDAP/RELAP5 Severe Accident Natural Circulation Modeling and Applications

Paul D. Bayless, INEL

NRC meeting on
SCDAP/RELAP5 code modeling of natural circulation
under severe accident conditions
August 19-20, 1996
Burr Ridge, IL

ATTACHMENT 3

Outline

Background

Severe accident natural circulation flow description

SCDAP/RELAP5 input model development

RELAP5 assessment with Westinghouse experiments

Plant applications

WHY INVESTIGATE SEVERE ACCIDENT NATURAL CIRCULATION FLOWS?

NUREG-0956 IDENTIFIED NATURAL CIRCULATION AS A SEVERE ACCIDENT ISSUE.

ANALYSES FOR ORIGINAL NUREG-1150 DID NOT CONSIDER NATURAL CIRCULATION.

WESTINGHOUSE WAS PERFORMING NC EXPERIMENTS UNDER EPRI SPONSORSHIP.

NRC WAS LOOKING AT THE PROBLEM ANALYTICALLY.

NATURAL CIRCULATION FLOWS TRANSFER HEAT FROM THE CORE TO OTHER RCS STRUCTURES.

UPPER PLENUM STRUCTURE MELTING

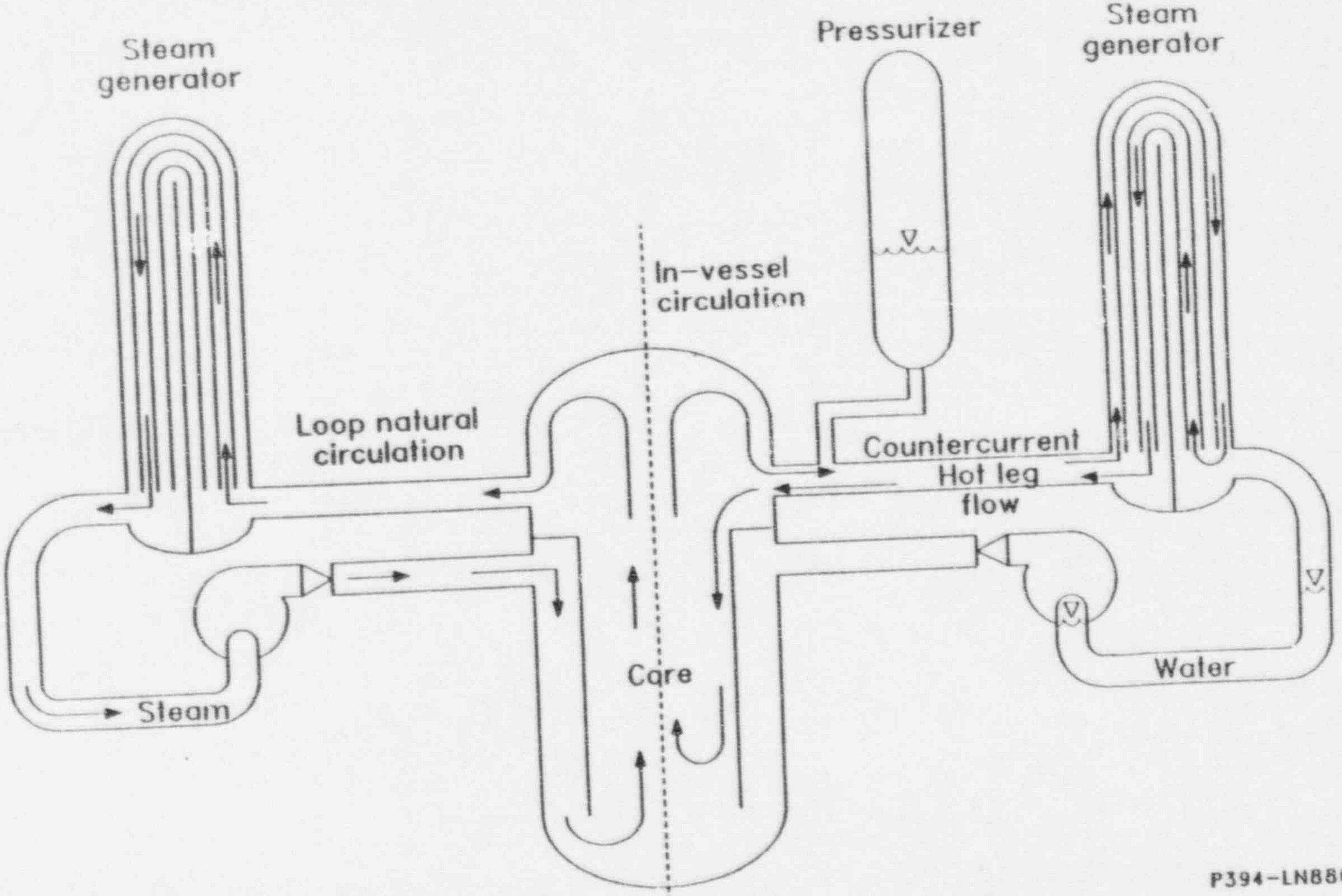
RCS PIPING FAILURE

- HPME OR LPME (CONTAINMENT INTEGRITY CONSIDERATION)
- ACCUMULATOR INJECTION

SG TUBE FAILURE

- CONTAINMENT BYPASS

Severe Accident Natural Circulation Flows



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NATURAL CIRCULATION FLOW CHARACTERISTICS

IN-VESSEL NATURAL CIRCULATION DRIVEN BY RADIAL POWER GRADIENT IN THE CORE.

- FOLLOWS THE CORE LIQUID LEVEL DOWN
- ADDITIONAL COLD RETURN PATH (IN CORE BYPASS PLANTS) WHEN LEVEL DROPS BELOW BOTTOM OF CORE FORMER PLATES

HOT LEG NATURAL CIRCULATION FLOW CONTROLLED BY MIXING IN THE SG INLET PLENUM.

- PORV OPENING INTERRUPTS FLOW, BUT IT IS QUICKLY RE-ESTABLISHED WHEN VALVES CLOSE

THE STANDARD RELAP5 HEAT TRANSFER PACKAGE IS USED.

- TURBULENT OR LAMINAR FLOW
- FORCED OR NATURAL CONVECTION
- CONVECTIVE HEAT TRANSFER BETWEEN THE FLUID AND STRUCTURES
- 1-DIMENSIONAL TREATMENT
- NO MODELING OF RADIATION OR FLUID-TO-FLUID HEAT TRANSFER

RELAP5 SINGLE PHASE HEAT TRANSFER CORRELATIONS

FORCED CONVECTION

- TURBULENT FLOW: DITTUS-BOELTER
- LAMINAR FLOW: SELLARS, NU = 4.36

FREE CONVECTION: CHURCHILL-CHU (VERTICAL), MCADAMS (HORIZONTAL)

CODE USES THE MAXIMUM OF THE FORCED AND FREE CONVECTION HEAT TRANSFER COEFFICIENTS.

(NEARLY ALL OF THE CALCULATIONS WERE IN TURBULENT FORCED CONVECTION.)

SCDAP/RELAP5 NODALIZATION DEVELOPMENT

THE HOT LEG WAS SPLIT INTO TOP AND BOTTOM HALVES TO MODEL THE COUNTERCURRENT FLOW, SINCE THAT FLOW IS NOT POSSIBLE WITHIN A CONTROL VOLUME OF A 1-DIMENSIONAL CODE.

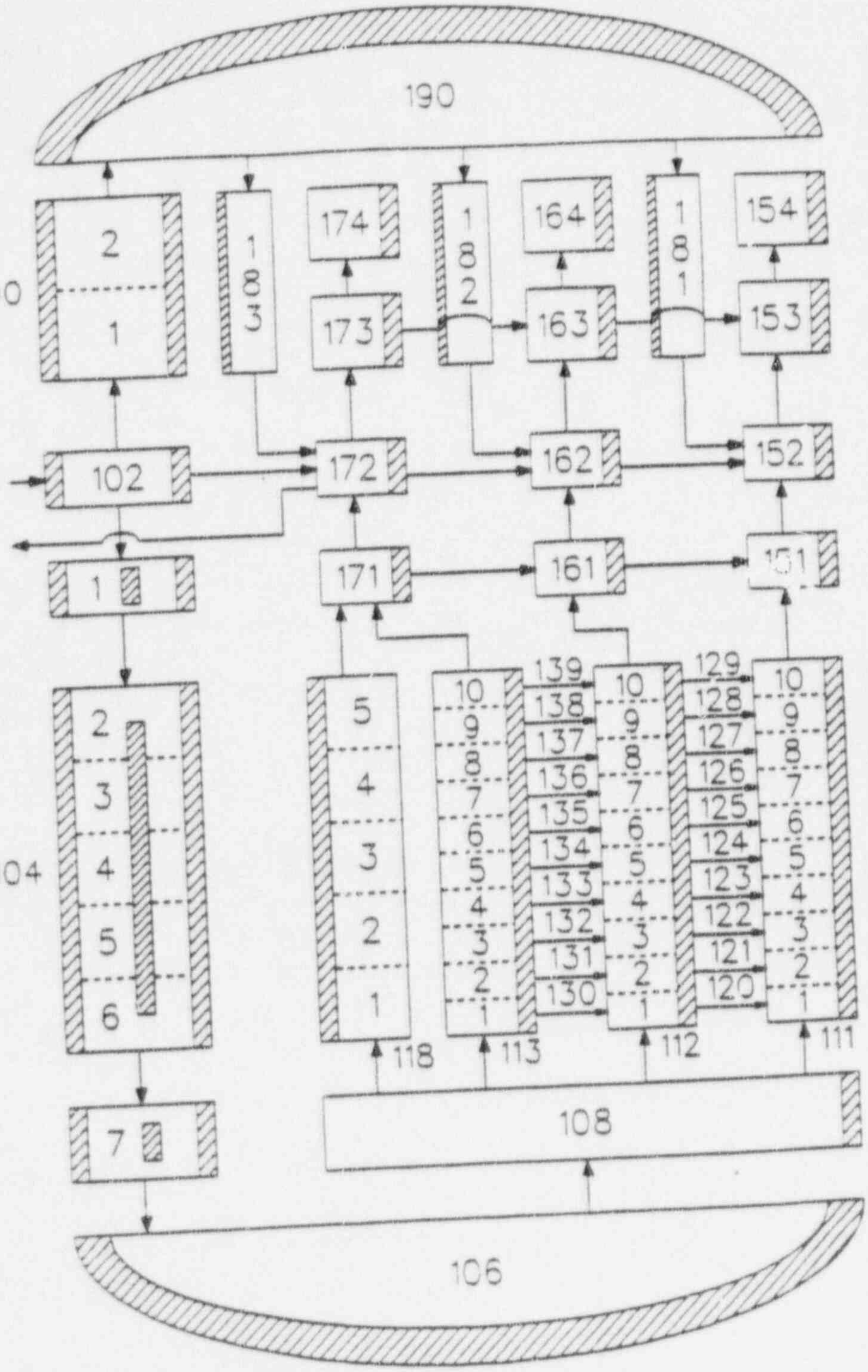
THE HOT/COLD FLOW STEAM GENERATOR TUBE SPLIT WAS SET TO 35/65%, BASED ON THE LOW PRESSURE EXPERIMENTS.

VARIOUS NODALIZATION SCHEMES WERE TRIED.

THE NODALIZATION USED PRESERVED THE CHARACTER OF THE FLOW PATTERN WHILE MINIMIZING UNPHYSICAL BEHAVIOR.

THREE RADIAL RINGS WERE USED IN THE CORE AND UPPER PLENUM, CONNECTED BY CROSSFLOW JUNCTIONS.

Surry 3-Channel Reactor Vessel Nodalization



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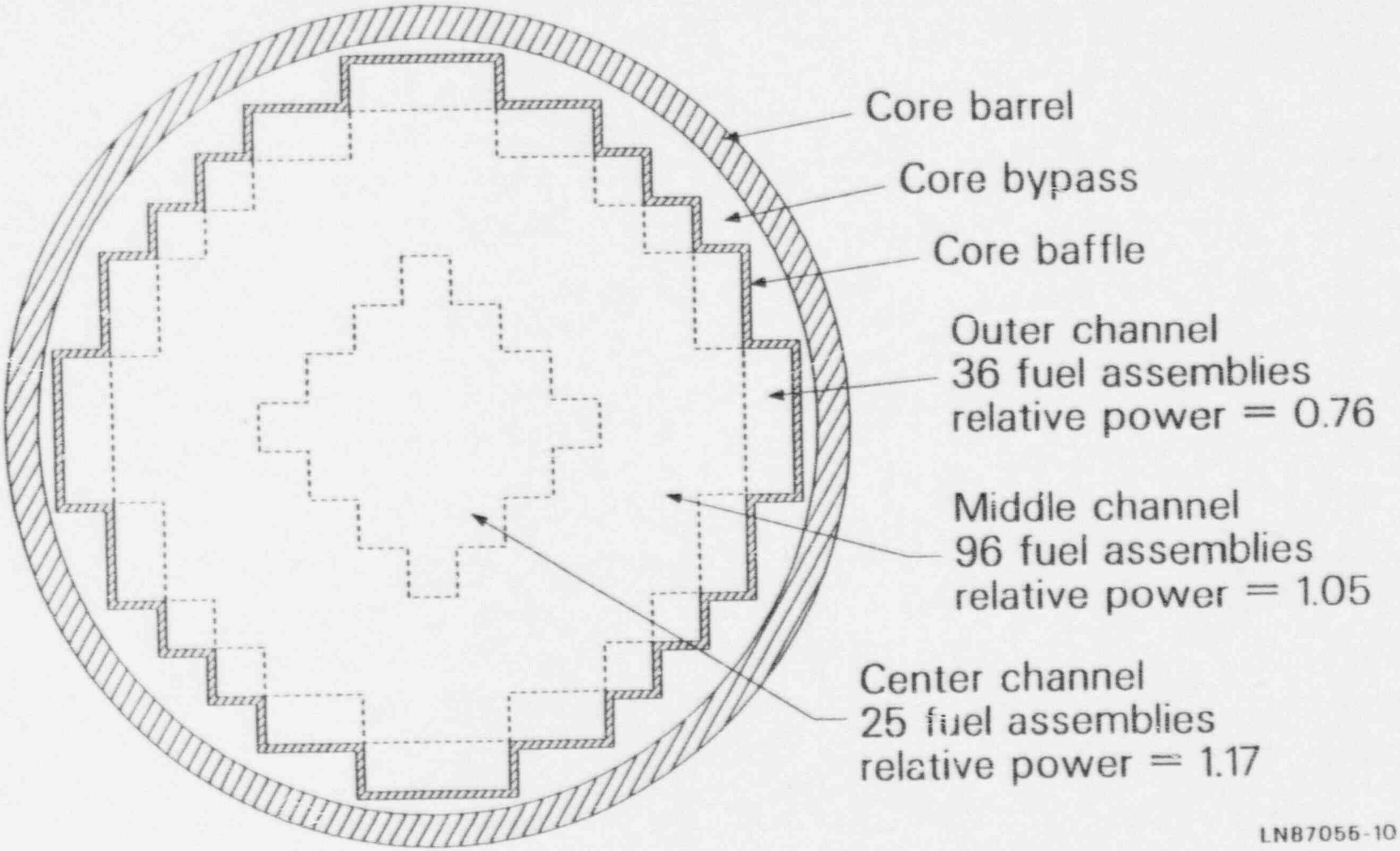
Cross Section of Three Channel Core Region



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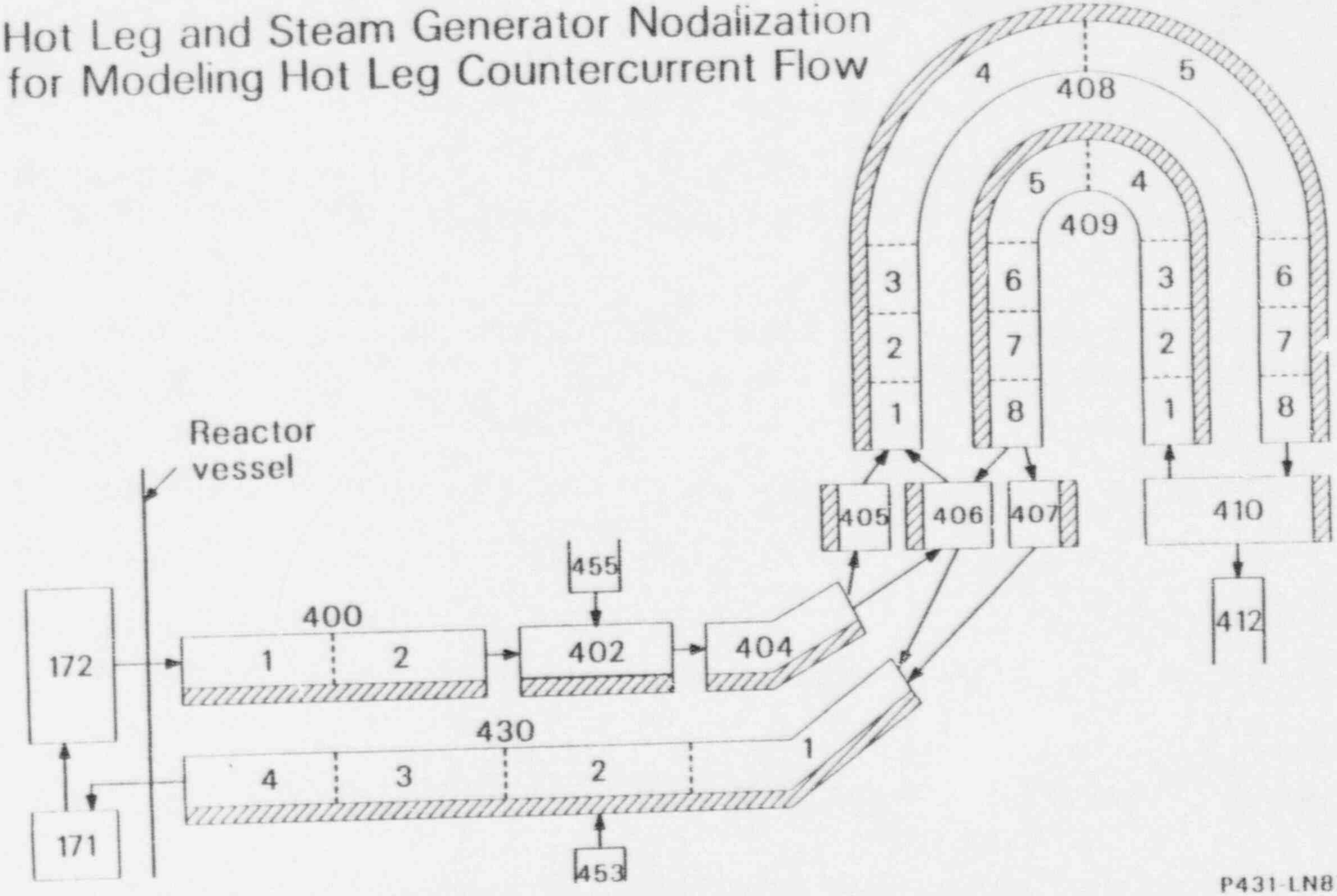
Hot Leg and Steam Generator Nodalization for Modeling Hot Leg Countercurrent Flow



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HOT LEG MODEL BENCHMARKING FOR NATURAL CIRCULATION FLOW MODELING

EPRI SPONSORED COMMIX CALCULATIONS OF THE LOW PRESSURE TESTS, WITH GOOD AGREEMENT BETWEEN THE CALCULATED AND MEASURED RESPONSES.

USING THE SAME MODELING APPROACH, TWO LOOPS OF THE SURRY PLANT WERE MODELED WITH COMMIX.

AN IDEALIZED BUT REPRESENTATIVE HEATUP TRANSIENT WAS RUN.

THE RELAP5 SURRY MODEL WAS MODIFIED TO MATCH THE COMMIX BOUNDARY CONDITIONS.

THE RELAP5 MODEL WAS ADJUSTED TO MATCH THE HEAT TRANSFER IN THE HOT LEGS AND STEAM GENERATORS FOR A GIVEN HOT VAPOR TEMPERATURE ENTERING THE HOT LEG.

- ALTERED THE SG INLET PLENUM VOLUME/FLOW AREA SPLIT, LOSS COEFFICIENTS THROUGH THE FLOW PATH

- DID NOT TRY TO MATCH FLOW RATES OR MIXING FRACTIONS

WHEN REASONABLE AGREEMENT WAS REACHED, THE HOT LEG/STEAM GENERATOR INPUT MODEL WAS "FROZEN" AND USED IN SCDAP/RELAP5 SEVERE ACCIDENT CALCULATIONS.

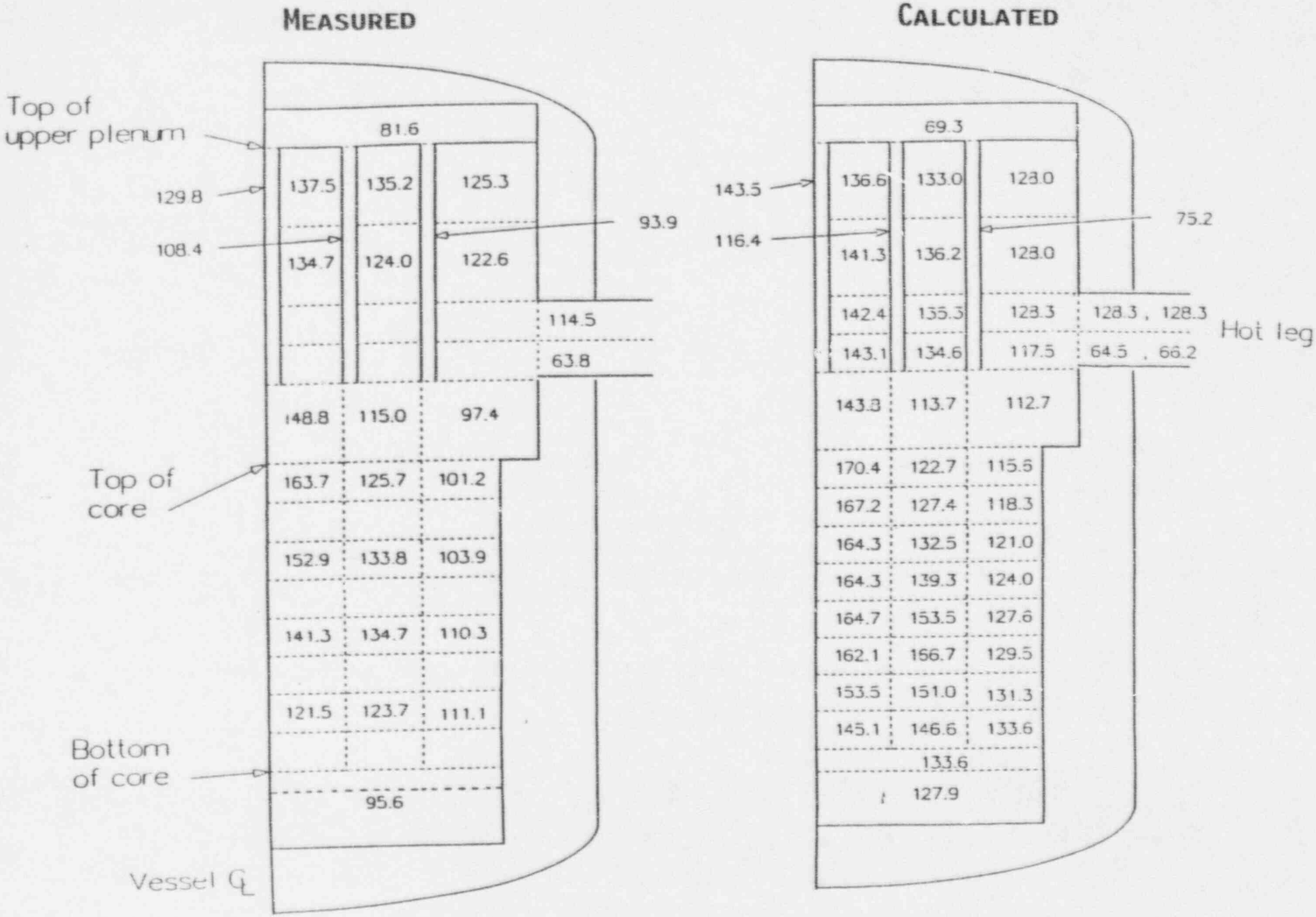
ASSESSMENT CALCULATIONS WERE PERFORMED USING TWO OF THE WESTINGHOUSE HIGH PRESSURE SF_6 TESTS.

STEADY STATE TESTS WERE SELECTED BECAUSE THEY HAD THE BEST ENERGY BALANCE.

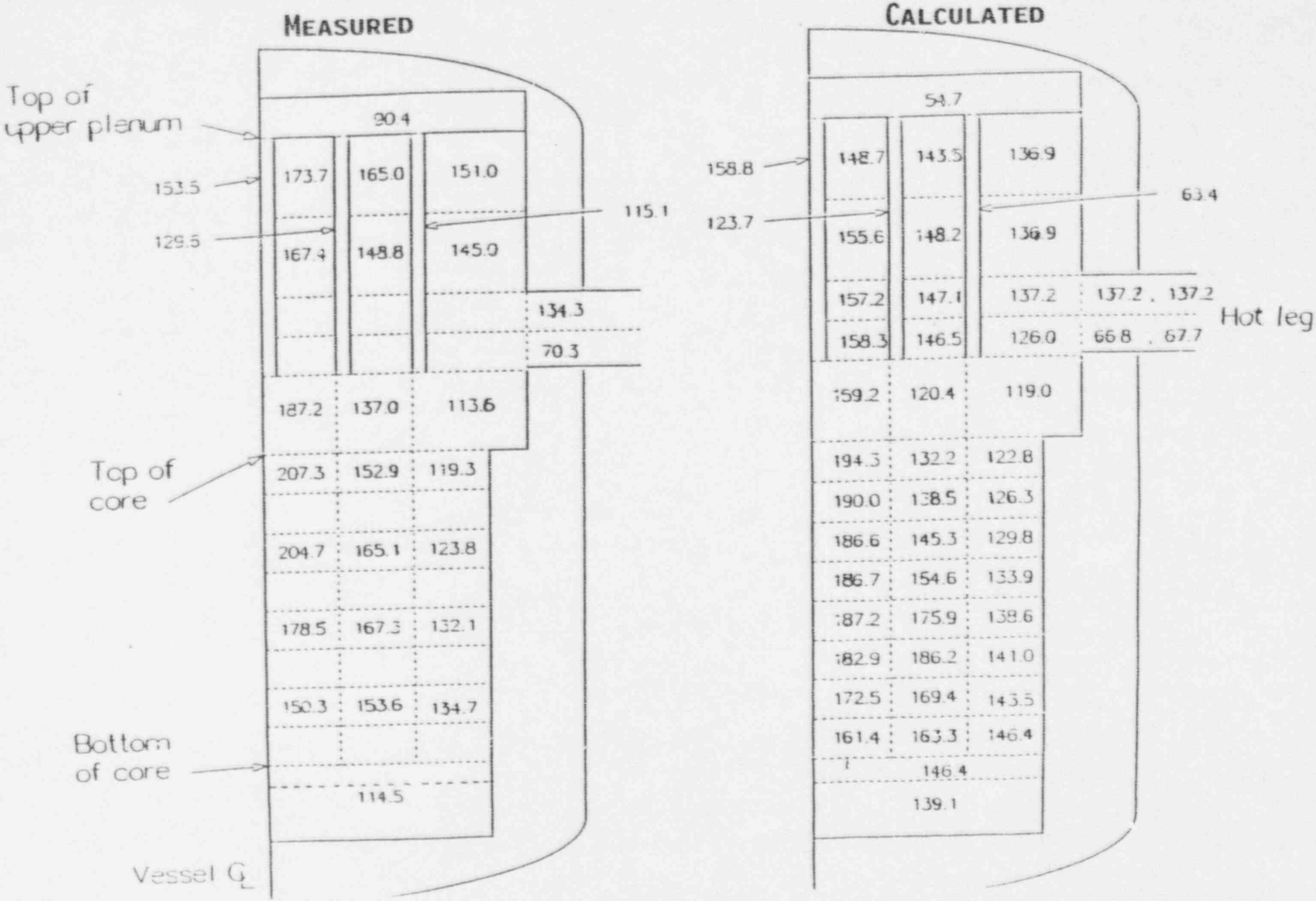
THE SAME MODELING APPROACH USED IN THE PLANT CALCULATIONS WAS USED TO MODEL THE FACILITY.

LOSS COEFFICIENTS FOR HOT LEG NATURAL CIRCULATION WERE ADJUSTED BASED ON THE RESULTS FROM ONE TEST, THEN WERE LEFT UNCHANGED TO MODEL THE SECOND TEST.

Reactor vessel vapor temperatures for Test S-7.



REACTOR VESSEL VAPOR TEMPERATURES FOR TEST S-6.



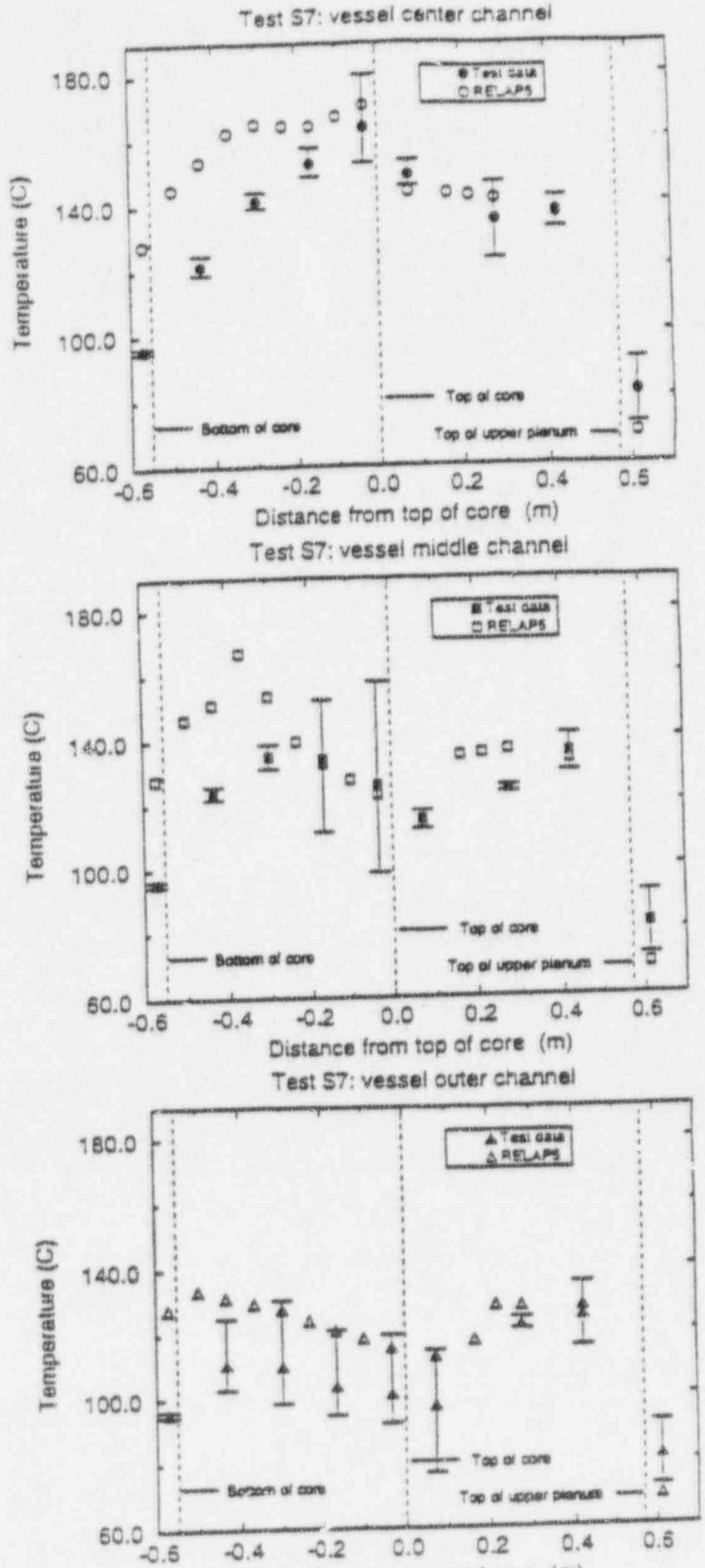


Figure D-11. Comparison of experimental and calculated SF_6 vapor temperatures in the core and upper plenum for Test S-7.

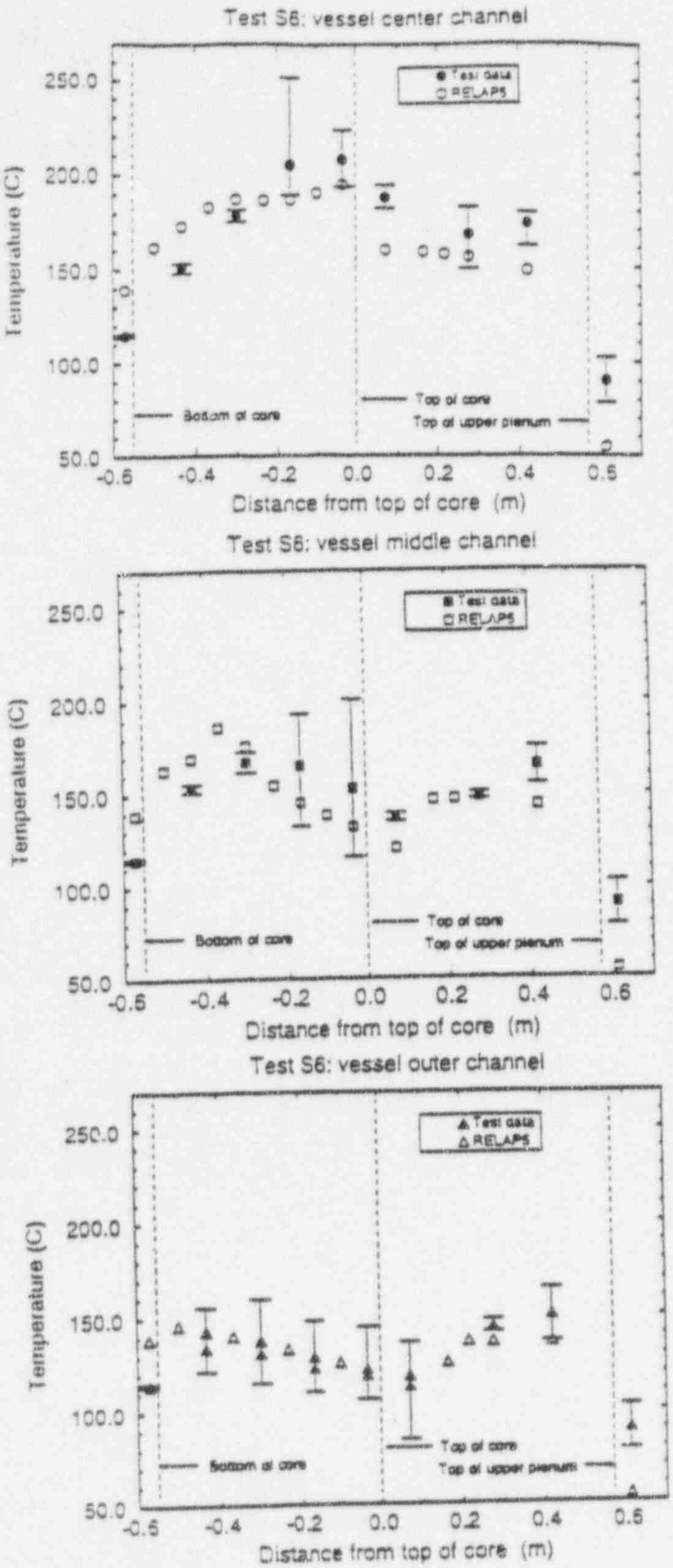


Figure D-12. Comparison of experimental and calculated SF_6 vapor temperatures in the core and upper plenum for Test S-6.

Hot leg natural circulation flow parameters for Test S-7.

Hot leg	Experiment Left	Experiment Right	Prediction Left	Prediction Right	Percent error Left	Percent error Right
Flow parameters						
$T_{h,out}$ (°C)	110.7	114.9	128.3	128.3	15.9	11.7
$T_{c,in}$ (°C)	60.8	60.3	66.2	64.5	8.9	7.0
$T_{h,out} - T_{c,in}$ (°C)	49.9	54.6	62.1	63.8	24.4	16.8
ρ_h (kg/m^3)	110.4	108.6	104.3	104.3	−5.5	−4.0
ρ_c (kg/m^3)	140.8	141.0	136.2	137.5	−3.3	−2.5
$\rho_c - \rho_h$ (kg/m^3)	30.4	32.4	31.9	33.2	4.9	2.5
q_{sg} (kW)	2.95	3.26	3.16	3.33	7.1	2.1
$\dot{m}_{hl}$ (kg/s)	0.071	0.072	0.0631	0.0647	−11.1	−10.1

HOT LEG NATURAL CIRCULATION FLOW PARAMETERS FOR TEST S-6.

Hot leg	Experiment		Prediction		Percent error	
	Left	Right	Left	Right	Left	Right
Flow parameters						
$T_{h,out}$ (°C)	129.7	134.9	137.2	137.2	5.8	1.7
$T_{c,in}$ (°C)	65.5	63.4	67.7	66.8	3.4	5.4
$T_{h,out} - T_{c,in}$ (°C)	64.2	71.5	69.5	70.4	8.3	−1.5
ρ_h (kg/m^3)	77.1	75.8	75.5	75.5	−2.1	−0.4
ρ_c (kg/m^3)	98.9	99.6	97.6	98.0	−1.3	−1.6
$\rho_c - \rho_h$ (kg/m^3)	21.8	23.8	22.1	22.5	1.4	−5.5
q_{sg} (kW)	2.43	2.67	2.49	2.55	2.5	−4.5
$\dot{m}_{hl}$ (kg/s)	0.0467	0.0461	0.0446	0.0451	−4.5	−2.2

STEAM GENERATOR NATURAL CIRCULATION FLOW PARAMETERS FOR TEST S-7.

Steam generator	Experiment		Prediction		Percent error	
	Left	Right	Left	Right	Left	Right
Flow parameters						
q_{sg} (kW)	2.95	3.26	3.16	3.33	7.1	2.1
Number of hot tubes	72	NA	72[a]	72[a]	—	—
Number of cold tubes	144	NA	144[a]	144[a]	—	—
m_t (kg/s)	0.147	NA	0.131	0.132	–10.9	—
m_{hl} (kg/s)	0.071	0.072	0.0631	0.0647	–11.1	–10.1
m_t/m_{hl}	2.06	NA	2.08[b]	2.05[b]	1.0	—
f_1 f_2	0.89	NA	0.89[b]	0.89[b]	0.0	—
T_{ht} (°C)	69.5	NA	72.9	71.4	4.9	—
T_{ct} (°C)	45.9	NA	44.8	42.3	–2.4	—
$T_{ht} - T_{ct}$ (°C)	23.6	NA	28.1	29.1	19.1	—
T_m (°C)	67.1	NA	72.2	70.9	7.6	—

a. Not predicted by code; input to code.

b. Not predicted by code; steam generator inlet plenum loss coefficients and junction areas adjusted to obtain mixing fractions and flow ratio.

NA = Data was not obtained in the experiments.

STEAM GENERATOR NATURAL CIRCULATION FLOW PARAMETERS FOR TEST S-6.

Steam generator	Experiment		Prediction		Percent error	
	Left	Right	Left	Right	Left	Right
Flow parameters						
q_{sg} (kW)	2.43	2.67	2.49	2.55	2.5	–4.5
Number of hot tubes	64	NA	72[a]	72[a]	—	—
Number of cold tubes	152	NA	144[a]	144[a]	—	—
m_t (kg/s)	0.0919	NA	0.0907	0.0891	–1.3	—
m_{hl} (kg/s)	0.0467	0.0461	0.0445	0.0451	–4.7	–2.2
m_t/m_{hl}	1.97	NA	2.04[b]	1.98[b]	3.6	—
$f_1 f_2$	0.85	NA	0.89[b]	0.89[b]	4.7	—
T_{ht} (°C)	77.5	NA	76.4	75.9	–1.4	—
T_{ct} (°C)	44.5	NA	42.8	40.9	–3.8	—
$T_{ht} - T_{ct}$ (°C)	33.0	NA	33.6	35.0	1.8	—
T_m (°C)	73.2	NA	74.5	74.0	1.8	—

a. Not predicted by code; input to code.

b. Not predicted by code; steam generator inlet plenum loss coefficients and junction areas adjusted to obtain mixing fractions and flow ratio.

NA = Data was not obtained in the experiments.

THE RELAP5 CALCULATIONS WERE IN REASONABLE AGREEMENT WITH THE MEASURED RESPONSE.

FLOW PATTERNS WERE THE SAME AS IN THE EXPERIMENT.

TEMPERATURE PROFILES WERE THE SAME AS IN THE EXPERIMENT.

CALCULATED HOT LEG AND STEAM GENERATOR MASS FLOW RATES WERE WITHIN 11% OF THE MEASURED VALUES.

VAPOR TEMPERATURES IN THE STEAM GENERATOR TUBES WERE WITHIN 5% OF THE MEASURED VALUES.

VAPOR TEMPERATURES ENTERING THE HOT LEGS WERE OVERPREDICTED BY UP TO 16%.

HEATING OF THE VAPOR IN DOWNFLOW IN THE CORE WAS OVERPREDICTED.

HEATING OF THE VAPOR IN UPFLOW IN THE CORE WAS UNDERPREDICTED.

SURRY STATION BLACKOUT (TMLB' SEQUENCE) CALCULATIONS

INITIAL CALCULATIONS WERE PERFORMED TO INVESTIGATE IF EX-VESSEL FAILURES MIGHT OCCUR, WHERE THEY WOULD OCCUR, AND WHEN THEY WOULD OCCUR IN RELATION TO THE CORE DAMAGE PROGRESSION

SCOPING CALCULATIONS

PROGRESSIVE ADDITION OF NATURAL CIRCULATION FLOWS

RESULTS AS EXPECTED: MORE NATURAL CIRCULATION LED TO SLOWER CORE HEATUP

SURGE LINE FAILURE PREDICTED IN BOTH NATURAL CIRCULATION CASES

SENSITIVITY CALCULATIONS

KNEW THERE WERE UNCERTAINTIES BECAUSE OF THE LIMITED DATA, CODE/MODEL RESTRAINTS, INABILITY TO BENCHMARK/ASSESS THE CODE AGAINST DATA.

CASES CONSIDERED:

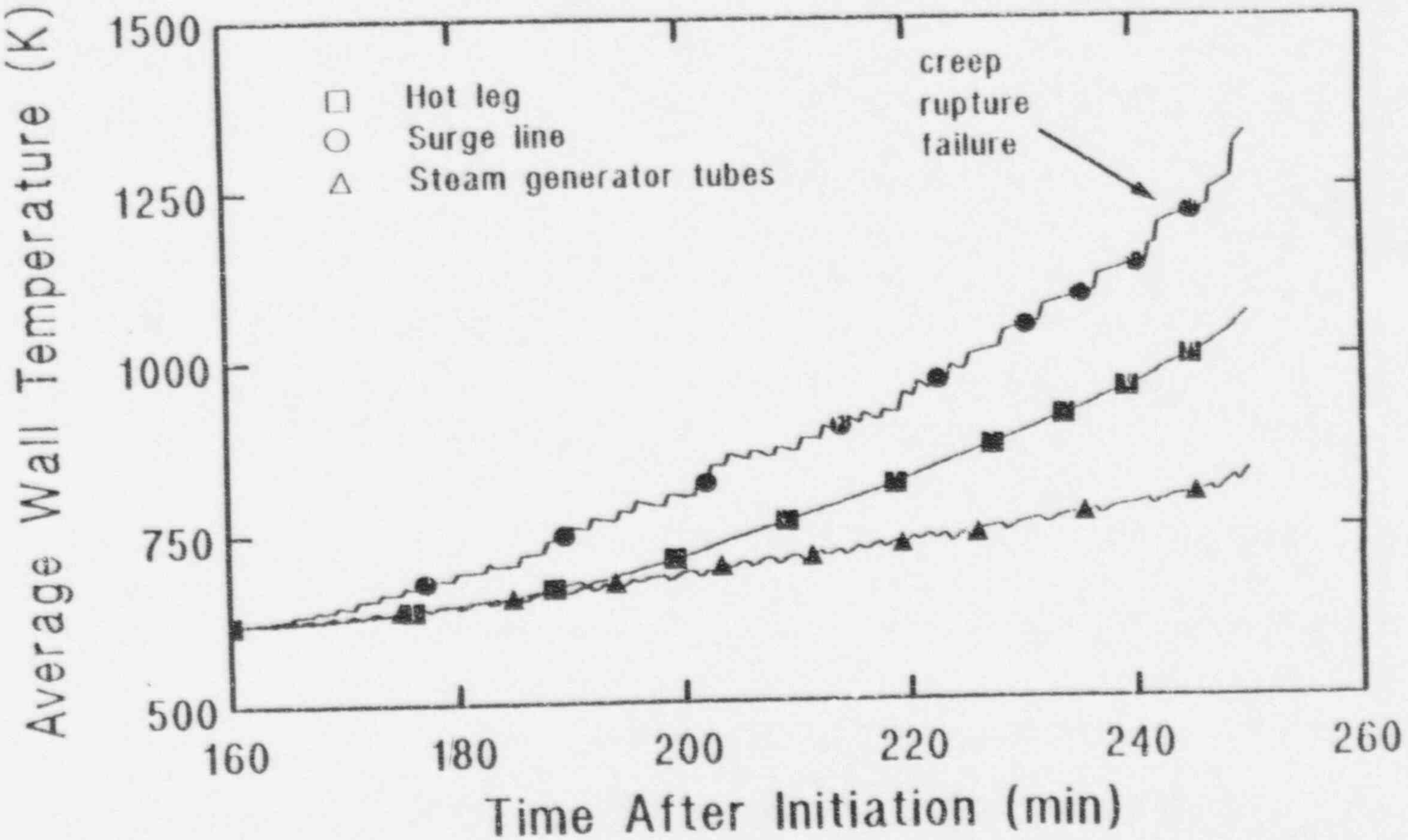
- AXIAL POWER PROFILE
- CORE AND UPPER PLENUM CROSSFLOW RESISTANCE
- STEAM GENERATOR INLET PLENUM MIXING
- HOT LEG/SURGE LINE PIPING HEAT LOSS
- HEAT TRANSFER COEFFICIENTS IN THE UPPER PLENUM, HOT LEG, AND STEAM GENERATOR TUBES
- SIMULATED RADIATION HEAT TRANSFER BETWEEN THE HOT LEG FLOW STREAMS

Base case used best-estimate values for the sensitivity parameters.

- Surge line failure at 246 min
- Initial fuel rod relocation at 248 min
- 75% of core heat removed by coolant
  - 4% to hot legs
  - 19% to steam generators

ECO00377

Steam generator tubes were much cooler than the surge line and hot legs.



ECO00461

Piping heat loss effects were primarily local.

	Convection	Convection and radiation
Surge line failure delay	7 min	13 min
Fuel rod relocation delay	2 min	4 min
Core heat removal	75%	76%
– To hot legs	5%	5%
– To steam generators	18%	17%
– To containment	3%	4%

EC000380

SIGNIFICANT REDUCTIONS IN THE STEAM GENERATOR INLET PLENUM MIXING HAD A SMALL IMPACT ON THE CALCULATED RESULTS.

MIXING FRACTION OF 0.7 IN THE PRESSURIZER LOOP, 0.3 IN THE OTHER TWO LOOPS

SURGE LINE FAILURE AT 255 MIN

INITIAL FUEL ROD RELOCATION AT 254 MIN

HOT LEG FLOW INCREASED 25% COMPARED TO BASE CASE

77% OF CORE HEAT REMOVED BY COOLANT

- 3% TO HOT LEGS
- 24% TO STEAM GENERATORS

CONCLUSIONS FROM THE SENSITIVITY CALCULATIONS

IN ALL OF THE CASES, EX-VESSEL PIPING FAILURES WERE PREDICTED TO OCCUR ABOUT THE TIME OF INITIAL FUEL ROD RELOCATION WITHIN THE CORE.

THE ONLY CALCULATIONS THAT HAD A NOTICEABLE DIFFERENCE FROM THE BASE CALCULATION WERE THE INLET PLENUM MIXING SENSITIVITIES.

SIGNIFICANTLY REDUCING THE MIXING FRACTION (TO 0.7 IN THE PRESSURIZER LOOP AND 0.3 IN THE OTHER TWO LOOPS, FROM 0.87 IN THE BASE CASE) BROUGHT THE STEAM GENERATOR TUBE TEMPERATURES HIGHER, BUT THE SURGE LINE STILL FAILED FIRST, WHEN THE MAXIMUM SG TUBE TEMPERATURE WAS ABOUT 360 K LOWER THAN THE SURGE LINE TEMPERATURE; IT WAS ABOUT 410 K LOWER THAN THE SURGE LINE TEMPERATURE IN THE BASE CASE.

ATTACHMENT 4

WESTINGHOUSE 1/7TH SCALE EXPERIMENTS

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and
M. Epstein

Presented for EPRI to
NRC Review Group on Natural Circulation

August 19, 1996

Approach to Scaling

NRC Severe Accident Scaling Methodology (SASM)

- Need to have an experiment which has all the physical processes even if the scaling is not perfect. This greatly aids the formation of Process Identification and Ranking Tables (PIRT). The W/EPRI experiments have all the dominant processes.

- Two types of considerations:

  - Top Down Scaling,
  - Bottom-Up Scaling.

- Both of these approaches are used to evaluate the W/EPRI experiments and to apply the understanding to the reactor system.

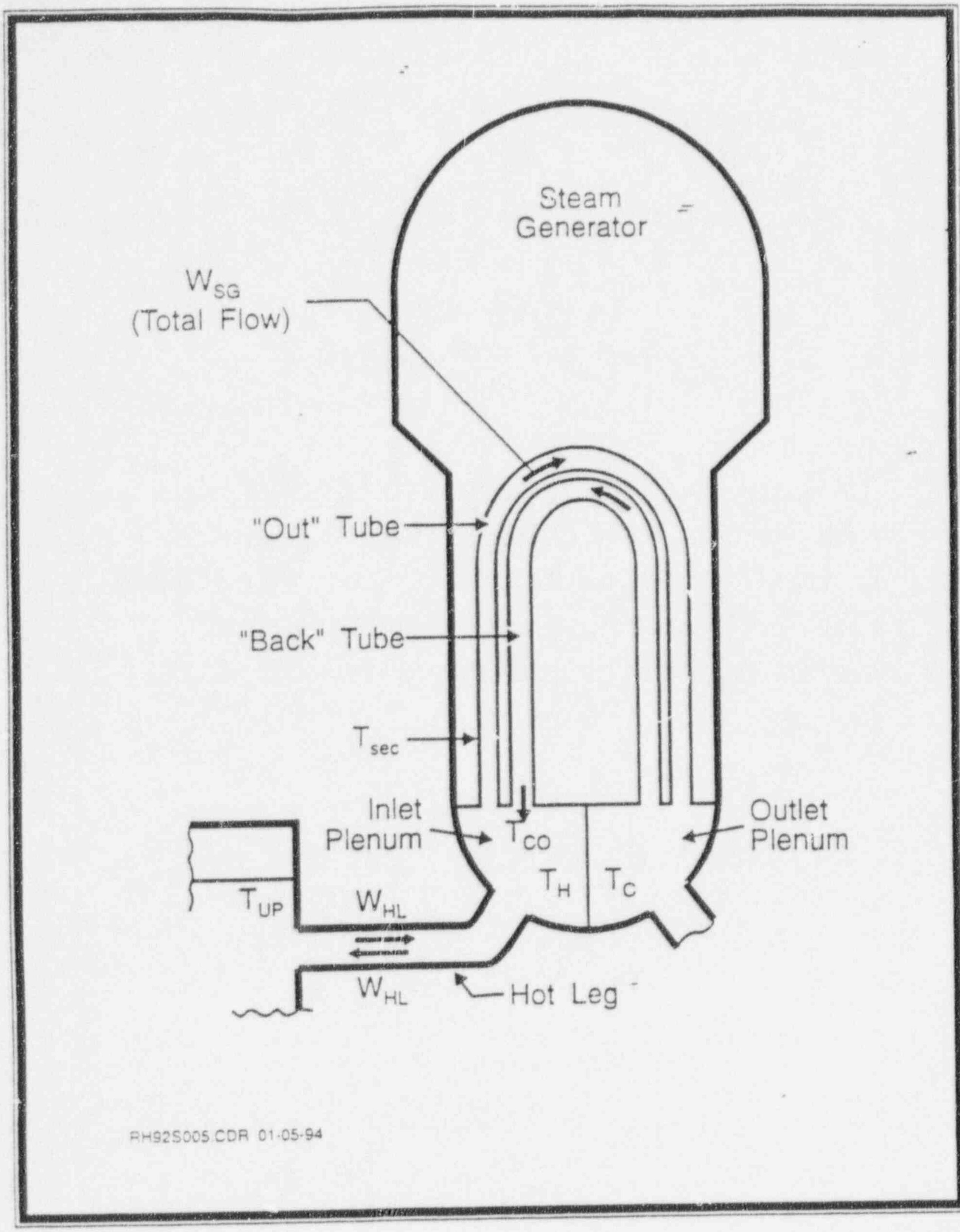


Figure 1 Hot leg and steam generator natural circulation flow model.

Top Down Scaling
Mixing in the SG
Inlet Plenum
Necessary Conditions

- To establish the appropriate mixing behavior in the inlet plenum, it is necessary to have the same geometry even though it may be in a scaled down system. The EPRI/W experiments represent the RPV, hot leg, surge line and SG geometry.

- The ratio of the flow through the steam generator tubes to the hot leg circulation flow should be preserved in the experiment.

Natural Convection Flow
In the Hot Legs

Countercurrent natural circulation flow through the hot leg

$$W_{HL} = C_{FC} \sqrt{g(\rho_H - \rho_{UP})\, \rho_{UP}\, D_{HL}^5}$$

Flow Through the SG Tubes

The turbulent momentum equation for flow through the tubes is given by

$$\Delta P = \Delta\rho \; g \; h_R = f\frac{L}{D_t} \; \frac{W_t^2}{2 \; \overline{\rho} \; A_t^2}$$

where h_R is a reference height.

Solving for the flow through the tubes results in

$$W_t = N\frac{\pi}{4} \; D_t^2 \left\{\frac{2D_t}{fL} \; \rho_H(\rho_c - \rho_H) \; g \; h_R\right\}^{1/2}$$

Ratioing the two flow rates results in

$$\frac{W_t}{W_{HL}} = N \left[\frac{\rho_H \left(\rho_c - \rho_H\right)}{\rho_{UP} \left(\overline{\rho}_H - \rho_{UP}\right)} \right]^{1/2} \left(\frac{\pi}{4\ C_{FC}} \right) \left(\frac{D_t}{D_{HL}} \right)^2 \left\{ \frac{2\ D_t\ h_R}{f\ L\ D_{HL}} \right\}^{1/2}$$

The density ratio term is of order unity.

The ratio of this flow should be preserved between the reactor system and the model, i.e. the left hand margin can be assumed to be a constant. Therefore:

$$N_{SG} \left(\frac{D_t}{D_{HL}} \right)^2_{SG} \left\{ \frac{2\ D_t\ h_R}{f\ L\ D_{HL}} \right\}^{1/2}_{SG} = N_m \left(\frac{D_t}{D_{HL}} \right)^2_m \left\{ \frac{2\ D_t\ h_R}{f\ L\ D_{HL}} \right\}^{1/2}_m$$

SG - Steam Generator
m - experimental model

It is further assumed that

1) the ratio of the effective height driving the natural circulation and the length of the tube is approximately the same in the scaled experimental model, and the steam generator,

2) the frictional coefficient is similar for the two systems, which implies a similar Reynolds number in the steam generator tubes. Finally we arrive at the expression

$$N_m = N_{SG} \frac{\left[\dfrac{D_t}{D_{HL}}\right]_{SG}^{2.5}}{\left[\dfrac{D_t}{D_{HL}}\right]_{m}^{2.5}}$$

Comparison

Steam Generator

- N_{SG} = 3260
- Tube inside diameter = 0.775 in.
- Diameter of the hot leg = 36 in.

Experimental Model

- Experimental model tube ID = 0.305 in.
- Hot leg diameter = 5.14 in.
- N_m = 258 (216 used in the model).

Hence, it is reasonable to assume that the mixing behavior observed in the Westinghouse experiments is representative of that which would be experienced in the reactor system.

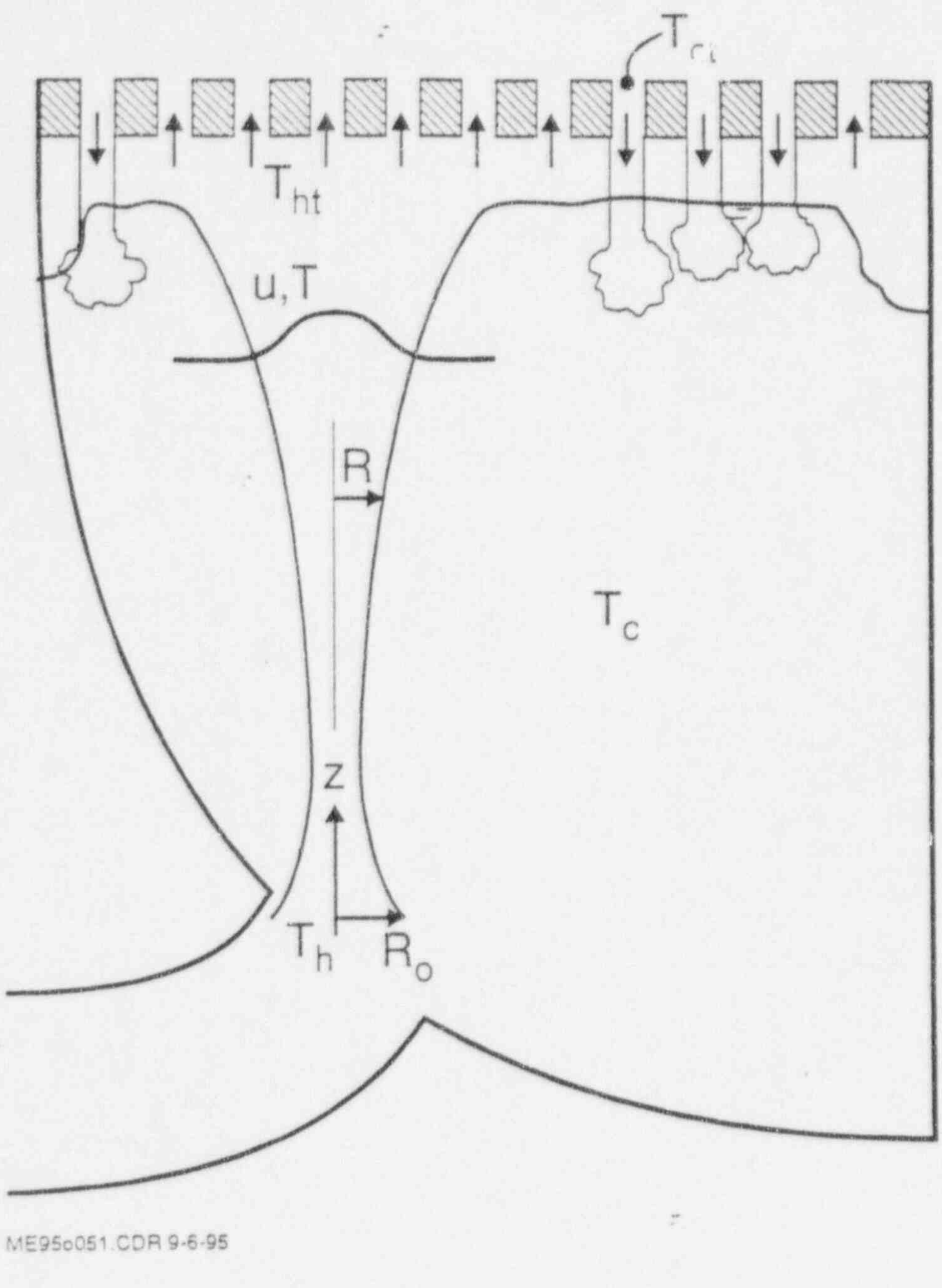
Bottom-Up Scaling
Major Elements of Plenum Mixing

1. Hot fluid rising through a scaled steam generator inlet plenum geometry. Experiments have this.

2. Entrainment of surrounding fluid with the appropriate (scaled) rise height. Experiments have this.

3. Spreading "ceiling plume" if the hot tube outflow cannot transmit the entire plume. Experiments have this.

4. Mixing of returning cold tube flow and the excess flow of the "ceiling plume". Experiments have this.



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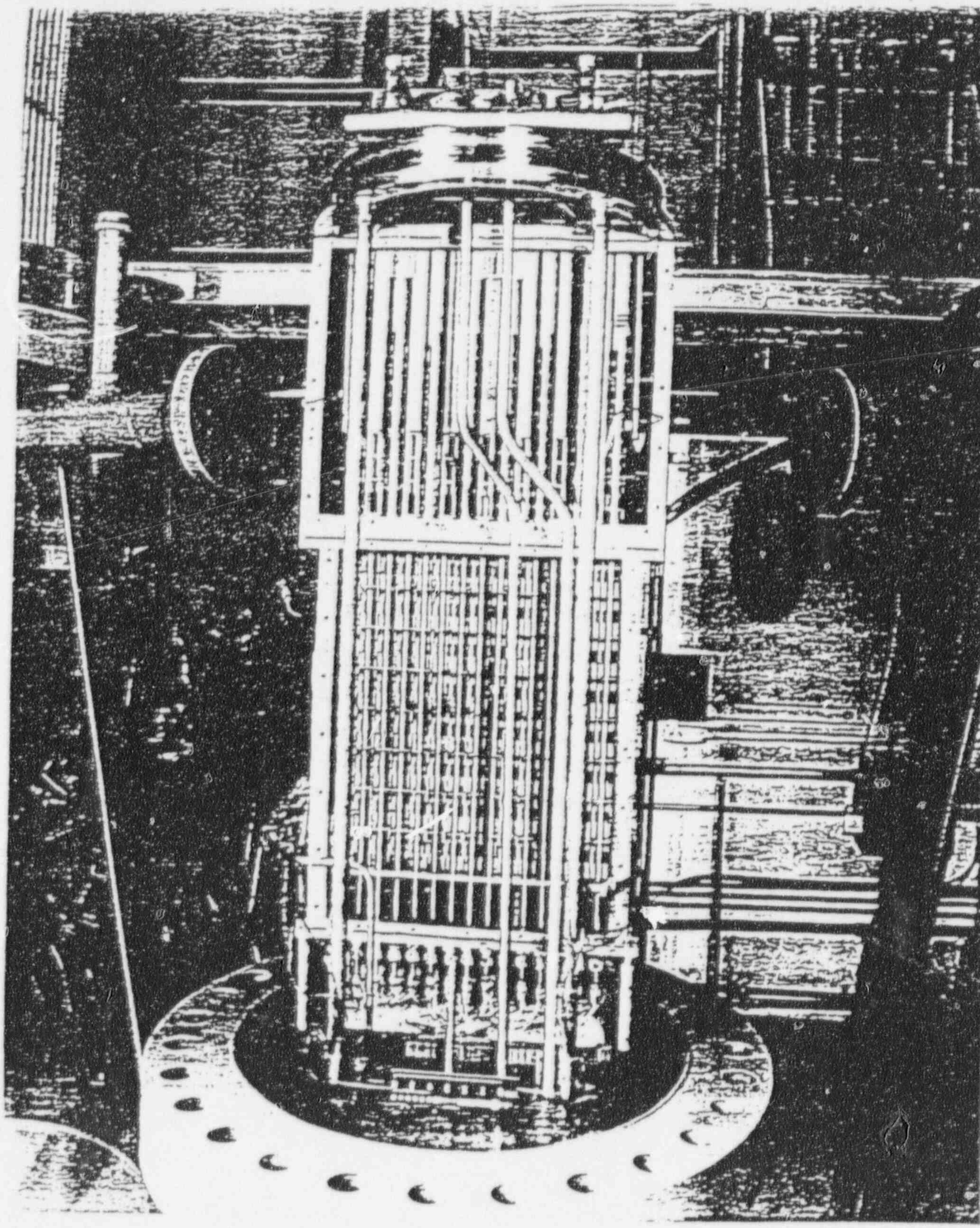


Figure 1 — One-seventh scale model of a Westinghouse four-loop reactor system. Model is one half section of reactor vessel (sliced through a vertical nearly diametral plane), hot legs, and two steam generators.

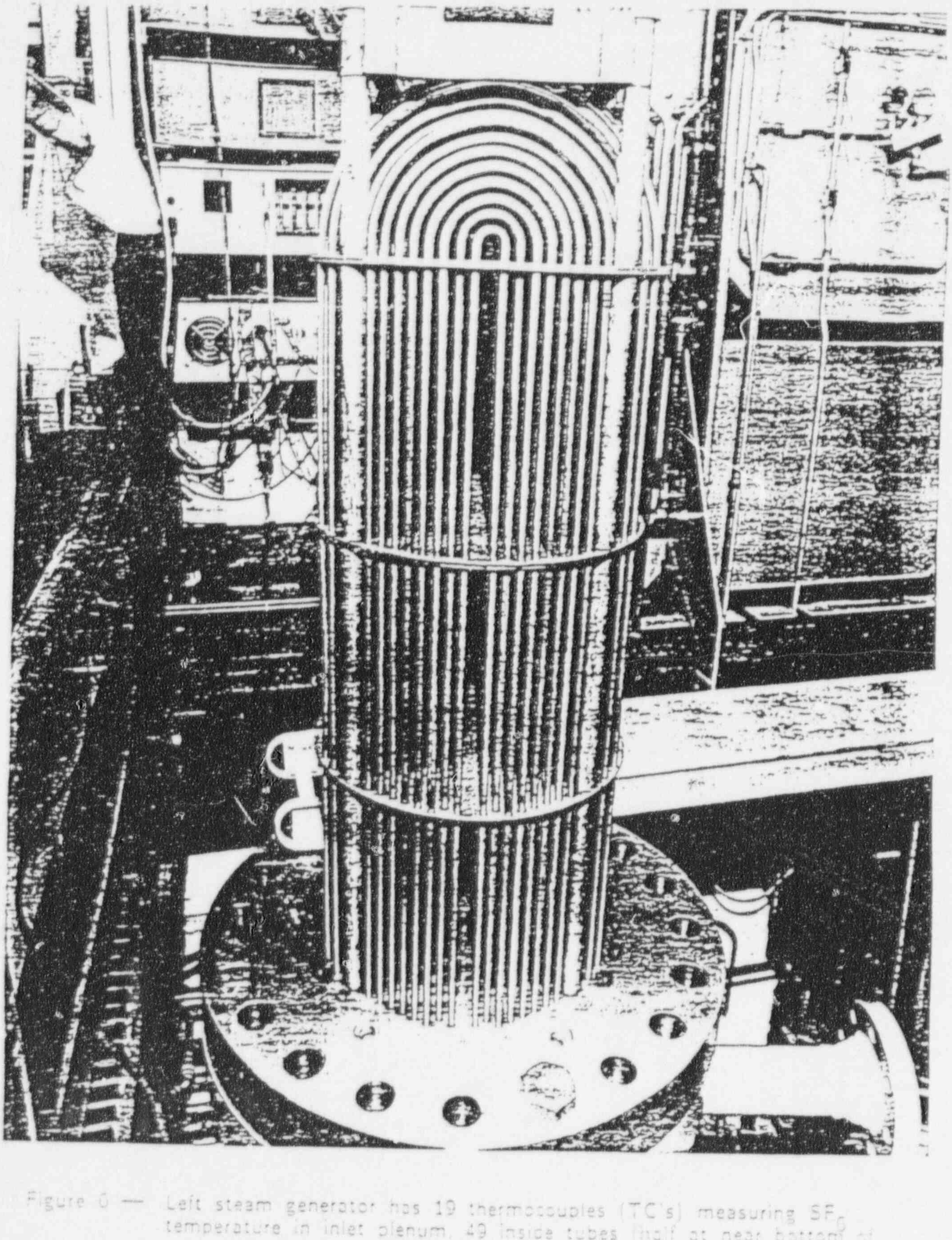
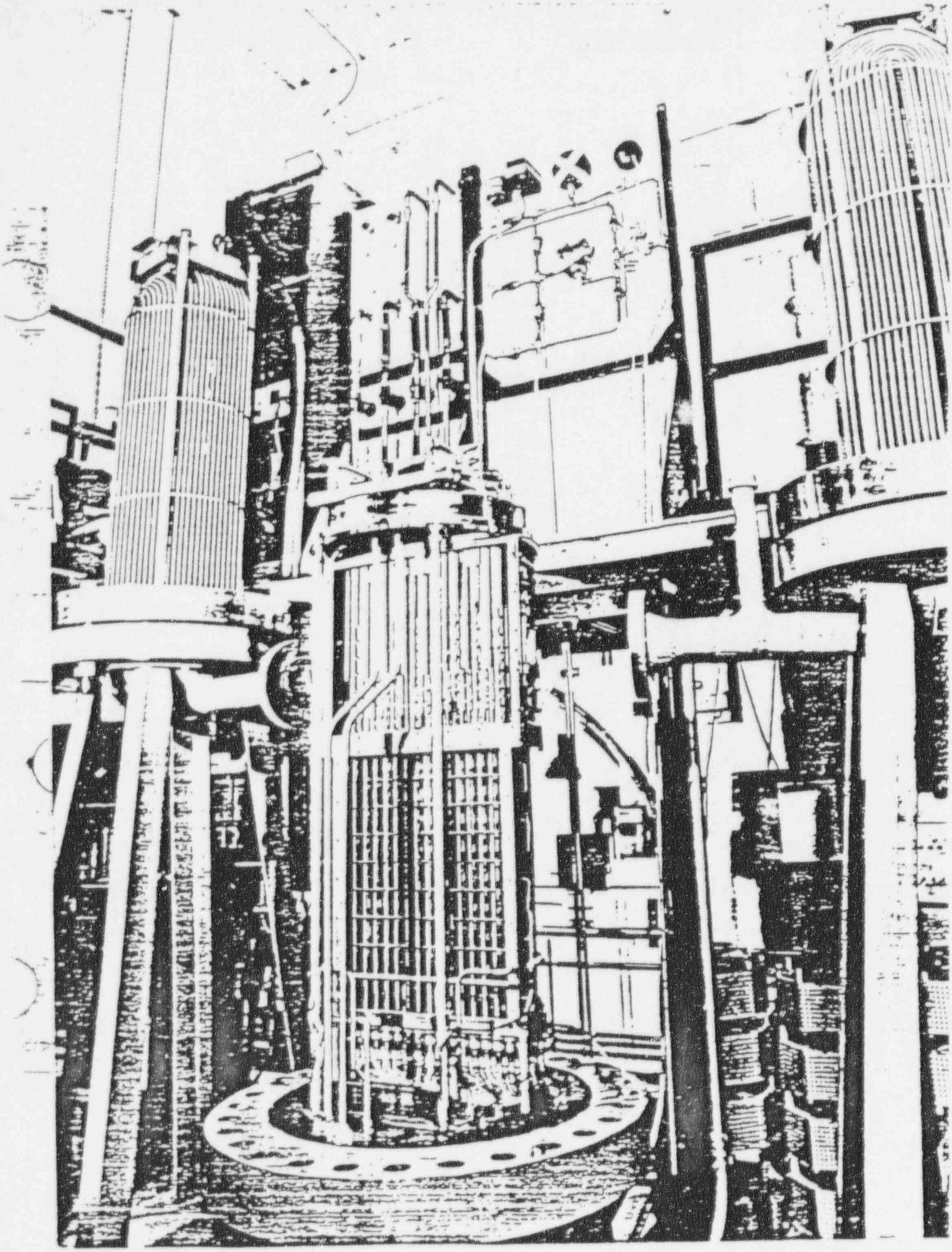
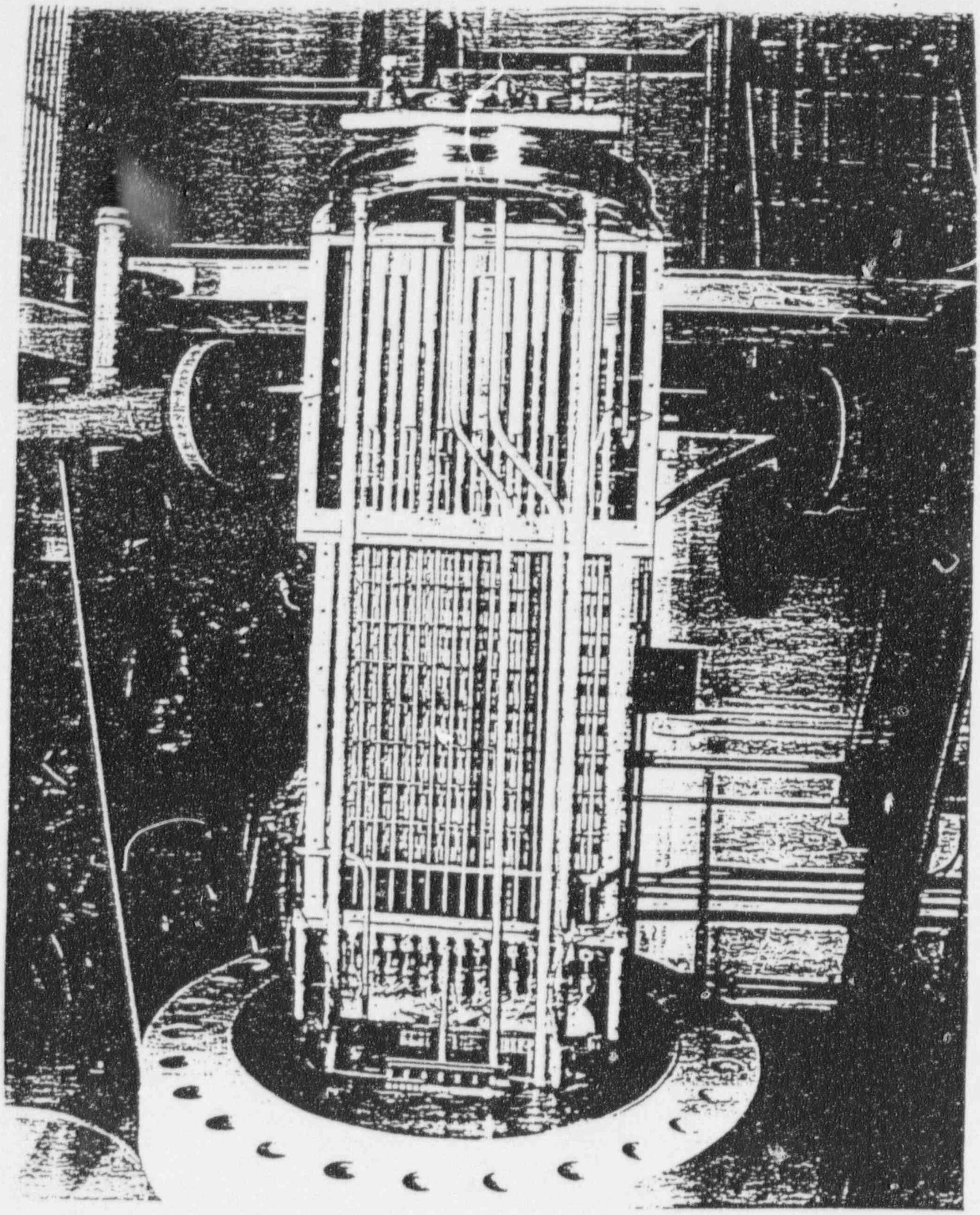


Figure 6 — Left steam generator has 19 thermocouples (TC's) measuring SF_6 temperature in inlet plenum, 49 inside tubes (half at near bottom of tube sheet, half near top), and 8 in outlet plenum. Three tubes have metal temperatures and gas temperatures measured along length. There are 25 TC's on tubesheet and channel head.





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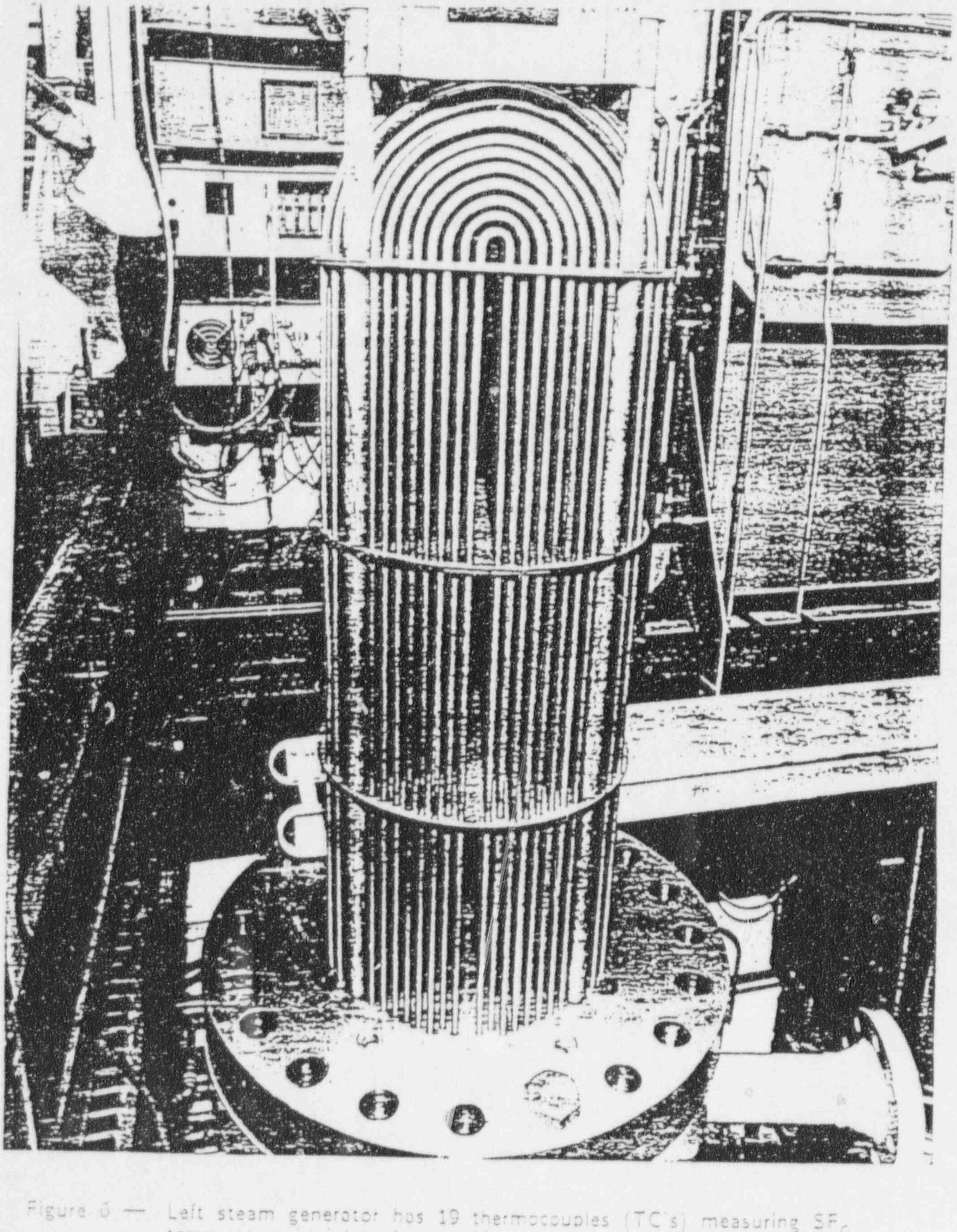

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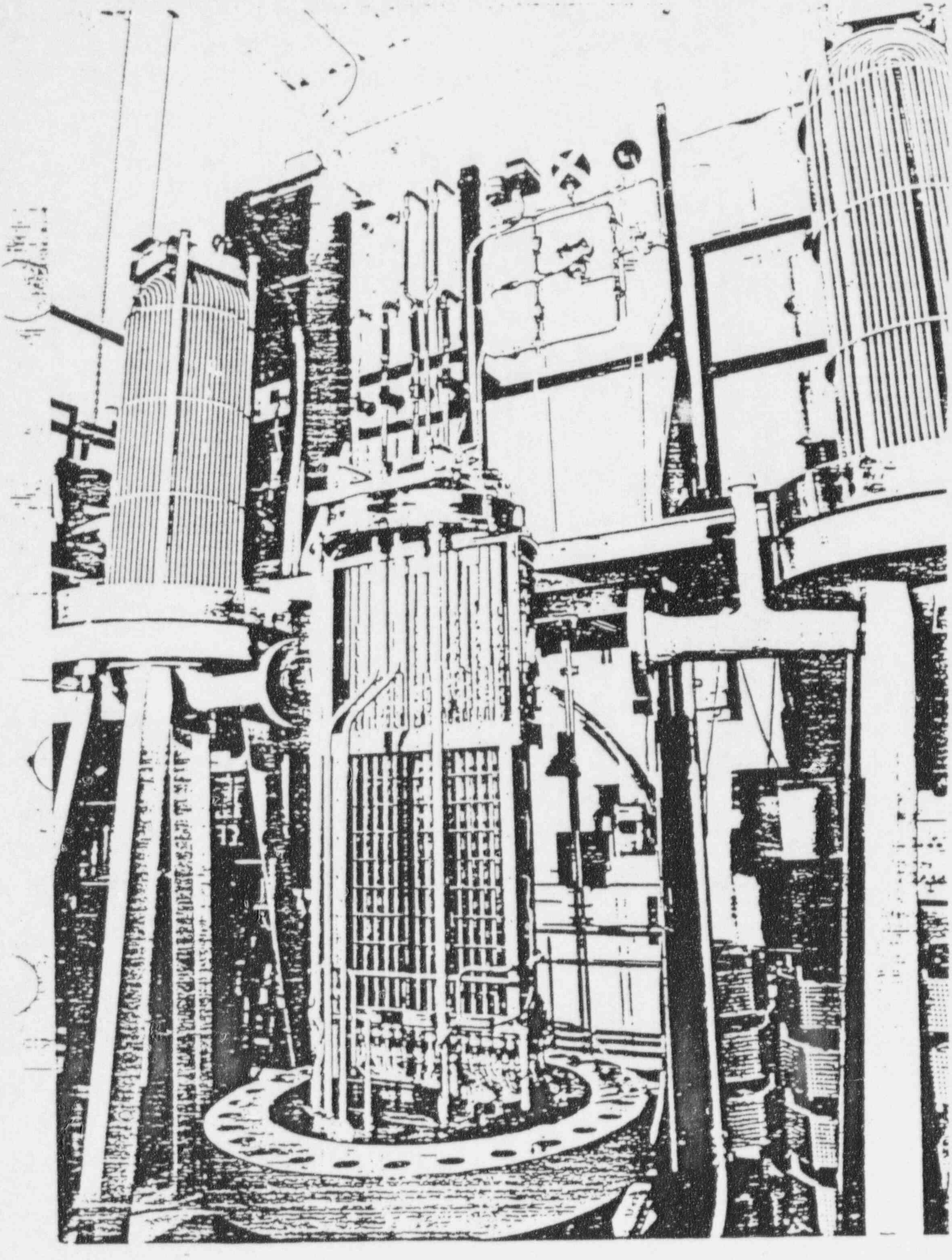

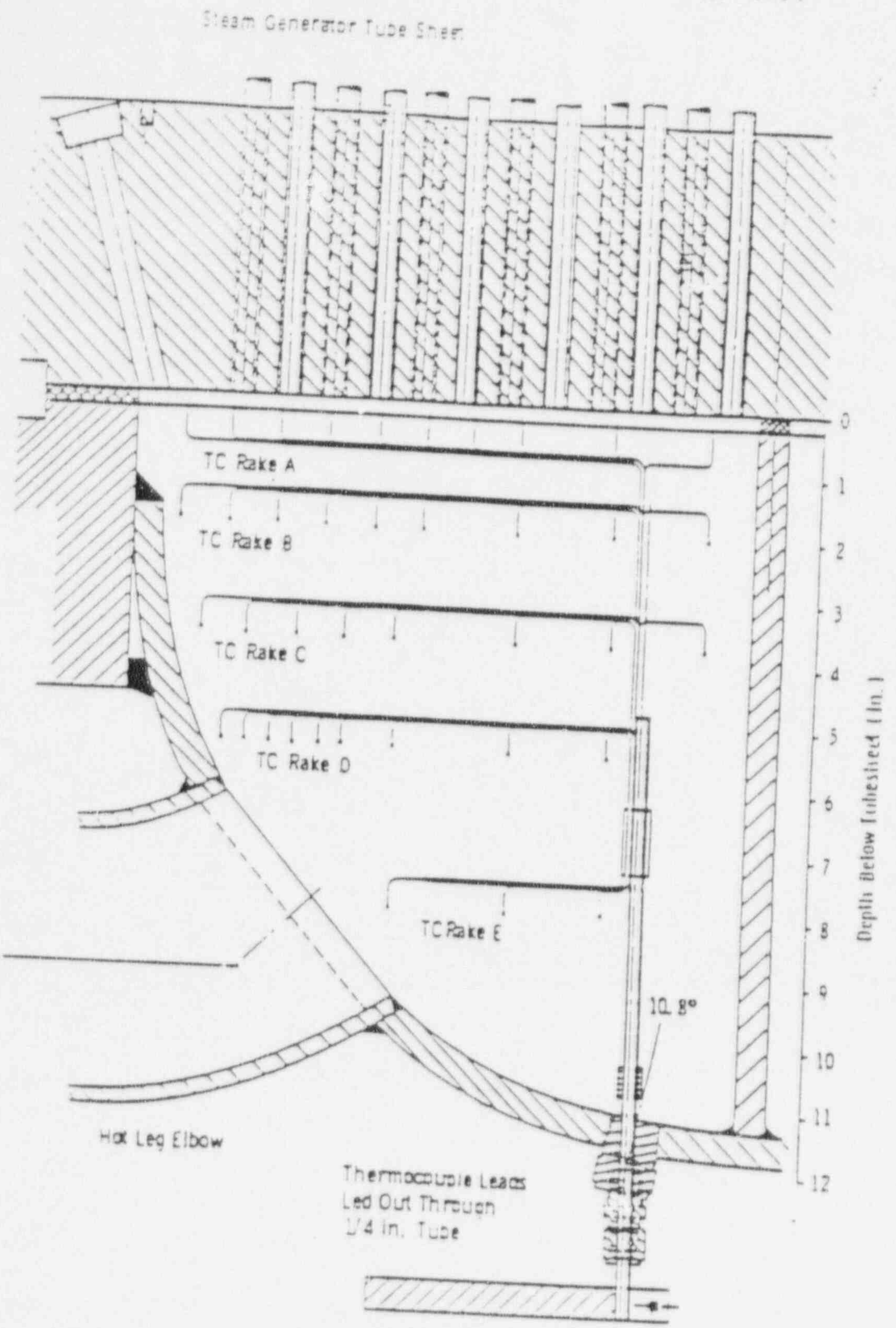

Figure 5 — Rotating thermocouple rake in right steam generator inlet plenum. Rake's arms are spaced 10° apart (except D and E).

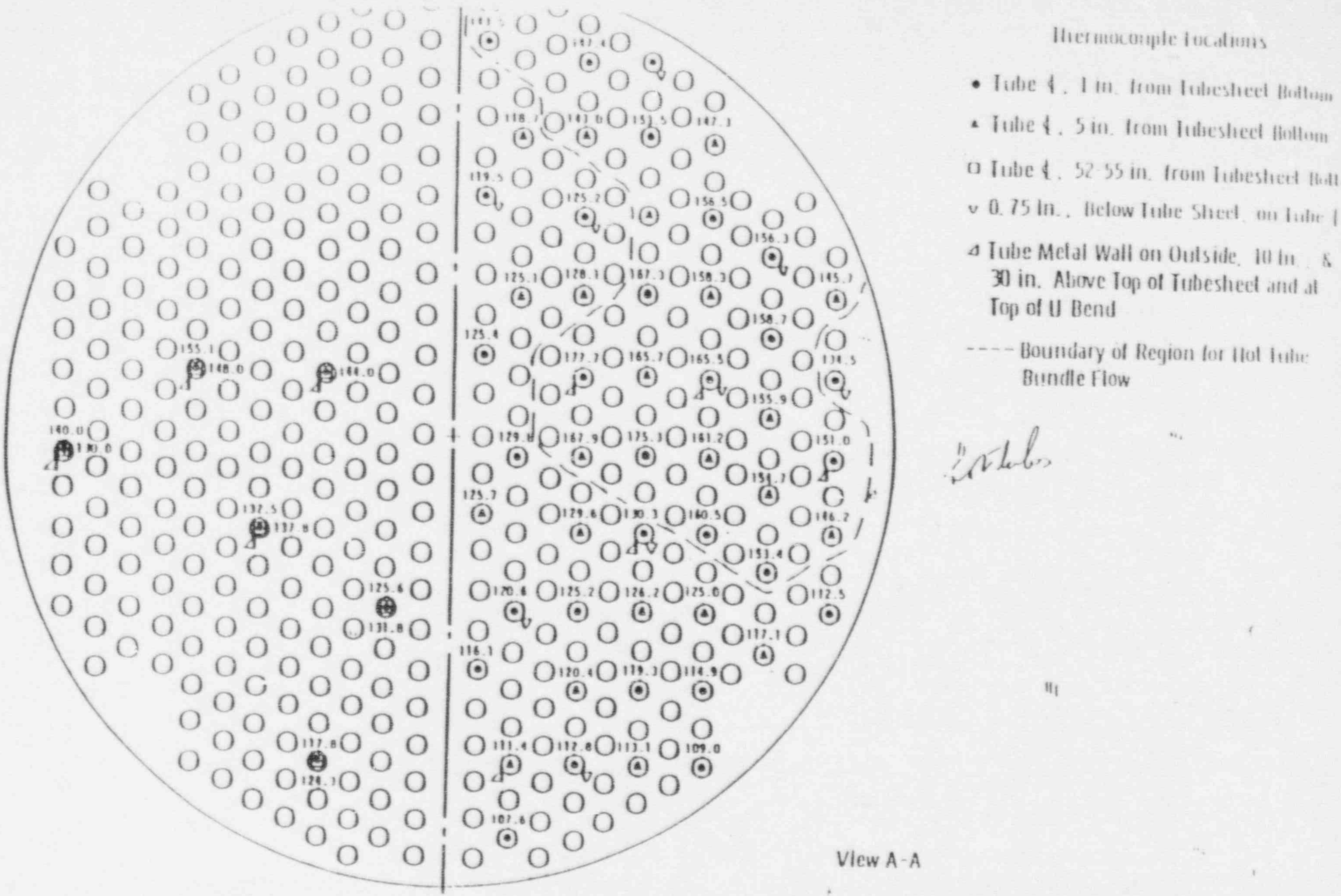

Figure 17 — Temperatures (°C) of SF_6 in Steam Generator Tubes Shown on Plan View. Transient Test SG-T4. Data at 3362 s.

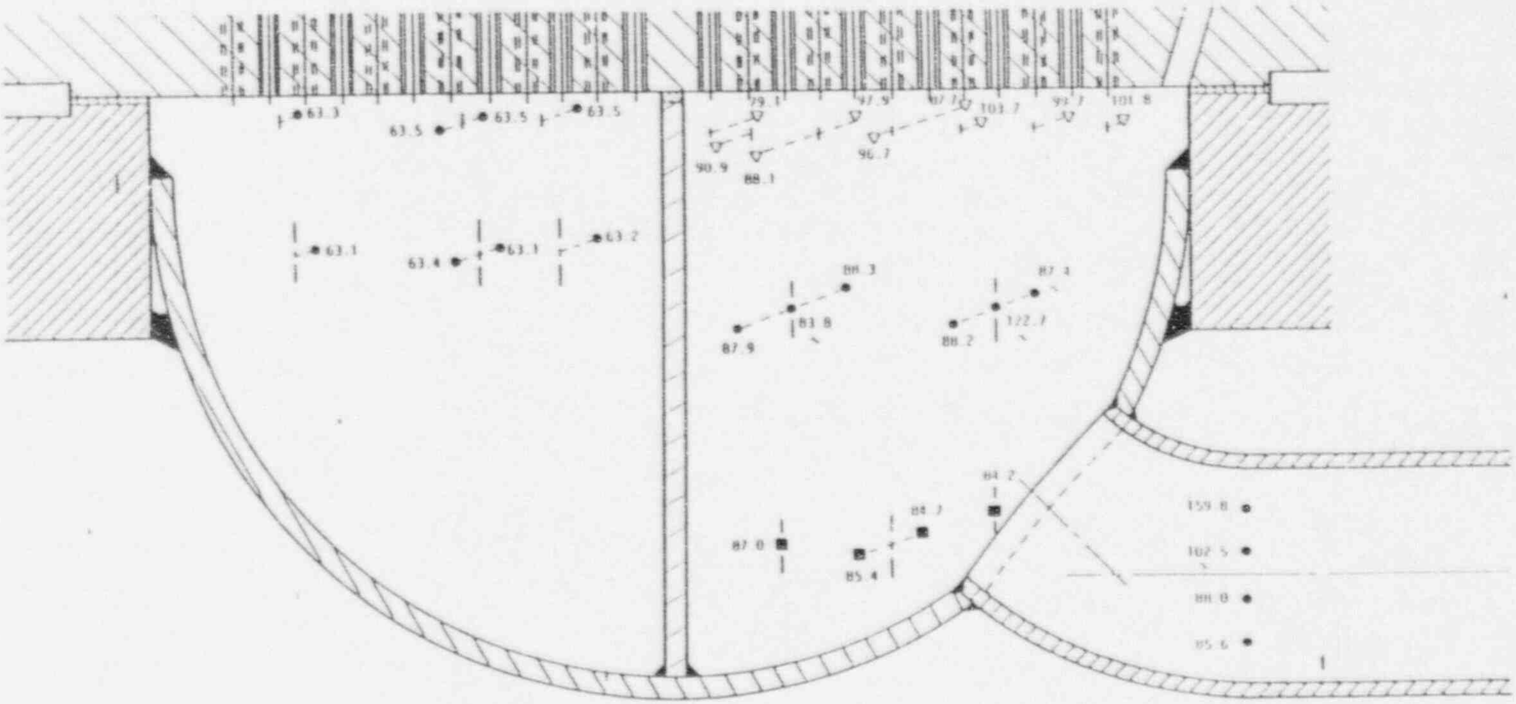

Figure 4-71. Temperatures (°C) of SF_6 in Steam Generator Channel Head and Hot Leg. Locations Shown Isometrically From Central Plane. Steady Cooling by Water on Secondary Side. Test SG-S3.

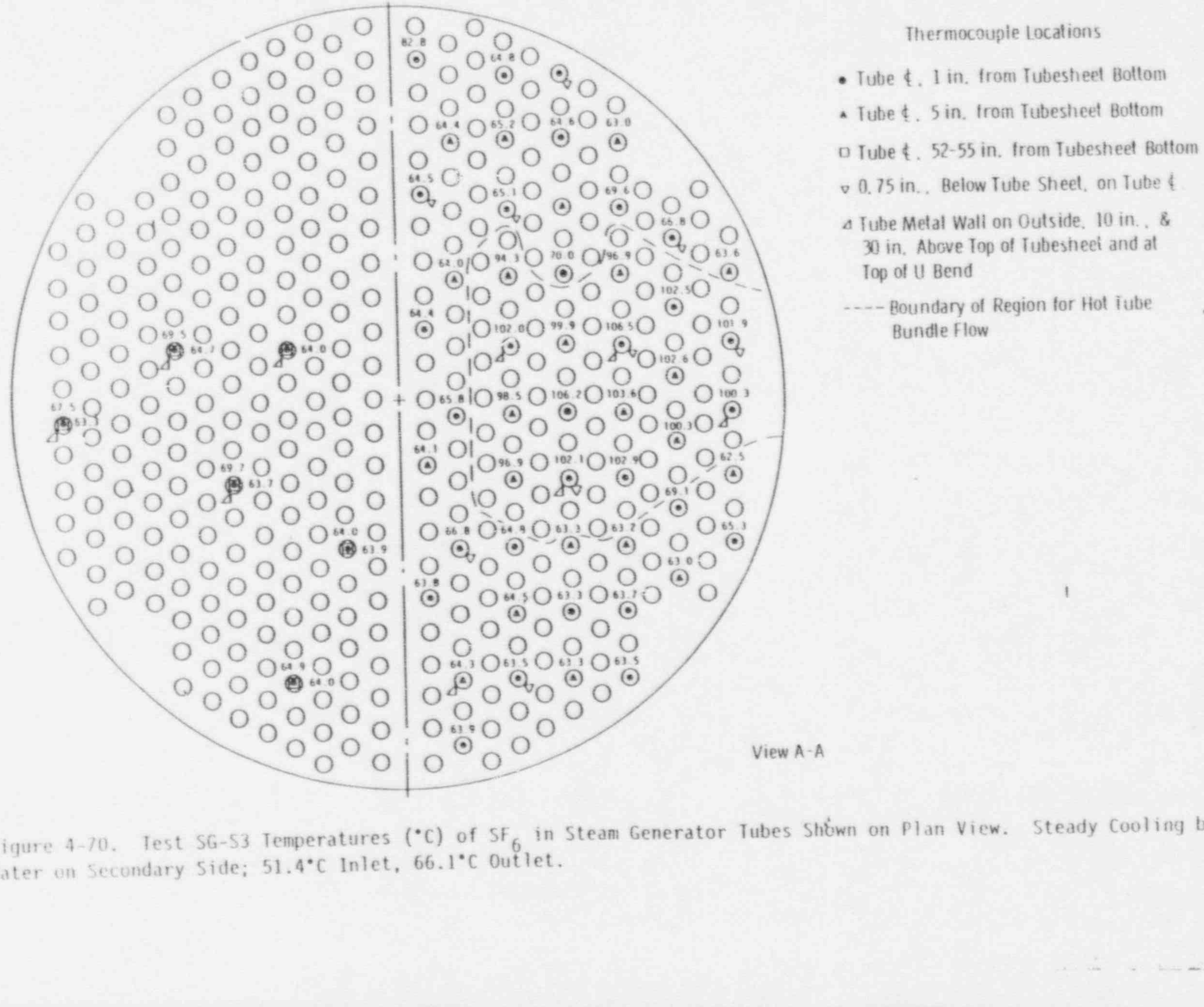

Figure 4-70. Test SG-S3 Temperatures (°C) of SF_6 in Steam Generator Tubes Shown on Plan View. Steady Cooling by Water on Secondary Side; 51.4°C Inlet, 66.1°C Outlet.

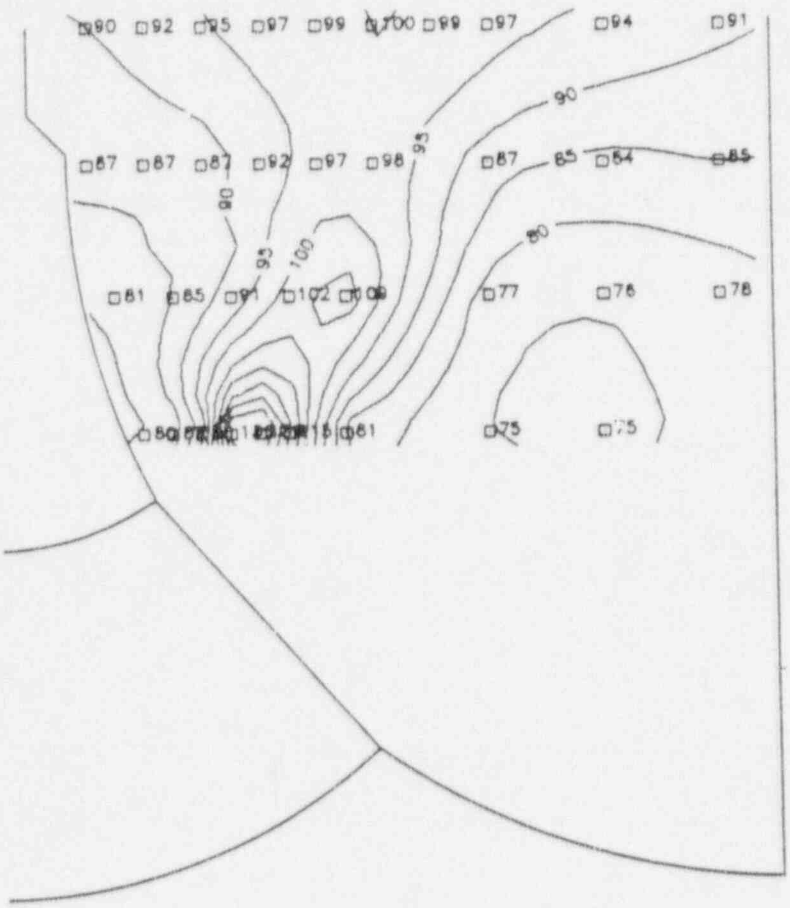

Figure 4-69. S. G. Inlet Plenum Mixing - Test SG-S3. Isotherms in a Vertical Plane Through the Hot leg Center Line.

"Top Hat" Model Comparison With Westinghouse SF_6 Test Data

Variable	Units	Test							
		SG-S1	SG-S2	SG-S3	SG-S4	SG-T1	SG-T2	SG-T3	SG-T4
Time	sec	-	-	-	-	6768	6700	3582	3362
Power	kw	22	22	30	30	22	22	30	30
Pressure	bar	20.7	27.6	20.7	27.6	20.7	27.6	20.7	27.6
T_h	°C	123.7	114.8	159.3	143.2	248.7	253.4	250.0	240.5
T_c	°C	70.3	72.1	86.8	86.2	142.4	171.7	126.4	146
T_{ht}	°C	79.7	80.2	100.8	98.4	165.8	185.8	153.4	159.2
$(T_{ht})_{max,data}$		85.4	83.3	106.5	101.6	181.3	199.4	170.9	177.7
$T_{ht,model}$ $E_o = 0.1$	°C	84.8	84.2	105.5	101.5	168	192	155	169
$T_{ht,model}$ $E_o = 0.116$	°C	83	83	103	100	164.5	190	151.5	166
$T_{ht,model}$ $E_o = 0.12$	°C	82.8	82.5	102.8	99.5	164	189	150.5	165.8

Model Comparison With Westinghouse Water Test Data		
Variable	H_2O -20	H_2O -21
Time	-	-
Power	29	29
Pressure	1.0	1.0
T_h	38.0	26.2
T_c	30.0	21.0
T_{ht}	32.9	23.5
$(T_{ht})_{max,data}$	32.9	23.5
$T_{ht,model}$ $E_o = 0.1$	-	-
$T_{ht,model}$ $E_o = 0.116$	32.5	22.8
$T_{ht,model}$ $E_o = 0.12$	-	-
$T_{ht,Gaussian}$ $E_o = 0.085$	32.9	23.2

Gaussian Plume Model

$$u = u_m \exp\left[-\frac{r^2}{R^2}\right]$$

$$\frac{T - T_\infty}{T_m - T_\infty} = \exp\left[-\frac{r^2}{\lambda^2 R^2}\right]$$

u_m - centerline velocity.
T_m - centerline temperature.

Entrainment coefficient = 0.082 for Gaussian distribution.

Boundary Conditions

$$R(o) = R'_o,\ u_m(o) = u_o,\ T_m(o) = T'_o$$

Where

$$T'_o = T_\infty + (T_o - T_\infty)\left[\frac{1 + \lambda^2}{2\lambda^2}\right]$$

$$R'_o = \sqrt{2}\ R_o$$

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"Gaussian" Model Comparison With Westinghouse SF_6 Test Data									
Variable	Units	Test							
		SG-S1	SG-S2	SG-S3	SG-S4	SG-T1	SG-T2	SG-T3	SG-T4
Time	sec	-	-	-	-	6768	6700	3582	3362
Power	kw	22	22	30	30	22	22	30	30
Pressure	bar	20.7	27.6	20.7	27.6	20.7	27.6	20.7	27.6
T_h	°C	123.7	114.8	159.3	143.2	248.7	253.4	250.0	240.5
T_c	°C	70.3	72.1	86.8	86.2	142.4	171.7	126.4	146
T_{ht}	°C	79.7	80.2	100.8	98.4	165.8	185.8	153.4	159.2
$(T_{ht})_{max,data}$		85.4	83.3	106.5	101.6	181.3	199.4	170.9	177.7
$T_{ht,Gaussian}$ $E_o = 0.085$	°C	89.5	88.0	112.0	106.4	176.5	199	165	176.5

PLUME VERSUS STEAM GENERATOR FLOW RATES

Test	Plume Flow* Just Below Tube Sheet, kg s^{-1}	Flow in Steam Generator, kg s^{-1}
SG-S1	0.250	0.114
SG-S2	0.345	0.146
SG-S3	0.243	0.120
SG-S4	0.330	0.137
SG-T1	0.208	0.036
SG-T2	0.239	0.083
SG-T3	0.220	0.101
SG-T4	0.283	0.136
H_2O-20	0.289	0.024

* Calculated with axisymmetric Gaussian plume model.

Mixing of Excess Plume Flow and Returning Tube Flow

$$\left(\dot{m}_{Plume} - \dot{m}_{ht}\right)\left(T_{ht} - T_c\right) = \dot{m}_{ht}\left(T_c - T_{ct}\right)$$

$$R = \frac{\dot{m}_{ht}}{\dot{m}_{Plume} - \dot{m}_{ht}}$$

$$T_c = \frac{T_{ht} + R\,T_{ct}}{1 + R}$$

Comparison of Measured and Calculated Plenum Mixing Temperatures					
Test	R	T_{ht}	T_{ct}	T_c (expt.)	T_c (plume model)
SG-S1	0.838	83.0	55.4	70.3	70.4
SG-S2	0.734	83.0	57.3	72.1	72.1
SG-S3	0.976	103.0	64.7	86.8	84.1
SG-S4	0.71	100.0	65.2	86.2	85.5
SG-T1	0.209	164.5	115.8	142.4	156.1
SG-T2	0.532	190	152.0	171.7	176.8
SG-T3	0.849	151.5	103.3	126.4	129.4
SG-T4	0.925	166.0	120.9	146.0	144.3
H_2O-20	0.0906	32.9	10.1	30.6	30.0

SCDAP/RELAP5 SGTR Analyses

Idaho National Engineering Laboratory

D. L. Knudson

SCDAP/RELAP5 Natural Circulation Model Review
Burr Ridge, Illinois
August 19-20, 1996

ATTACHMENT 5

Presentation Overview

- SCDAP/RELAP5 calculation objectives
- SCDAP/RELAP5 Surry loop calculations
- Surry loop results
- SCDAP/RELAP5 Surry plant calculations
- Surry plant results
- Conclusions

SCDAP/RELAP5 Calculation Objectives

- Evaluate variations in hot leg countercurrent natural circulation with respect to SG tube temperatures using a Surry (stand-alone) loop model
- Evaluate the potential for natural circulation-induced RCS pressure boundary failures, including SGTRs, using a Surry (full) plant model

SCDAP/RELAP5 Surry Loop Calculations

- Based on a stand-alone model of the Surry primary coolant loop containing the pressurizer

- Boundary conditions to drive the loop model were extracted from Surry plant results for a TMLB' transient without recovery; without operator action; without oxidation; and with modeling provisions to allow development of in-vessel, full loop, and hot leg countercurrent natural circulation

- All loop calculations initiated at the onset of countercurrent flow (9200 s) and extended ~5000 s (corresponding with surge line failure in the plant calculation) with variations in hot leg countercurrent natural circulation conditions

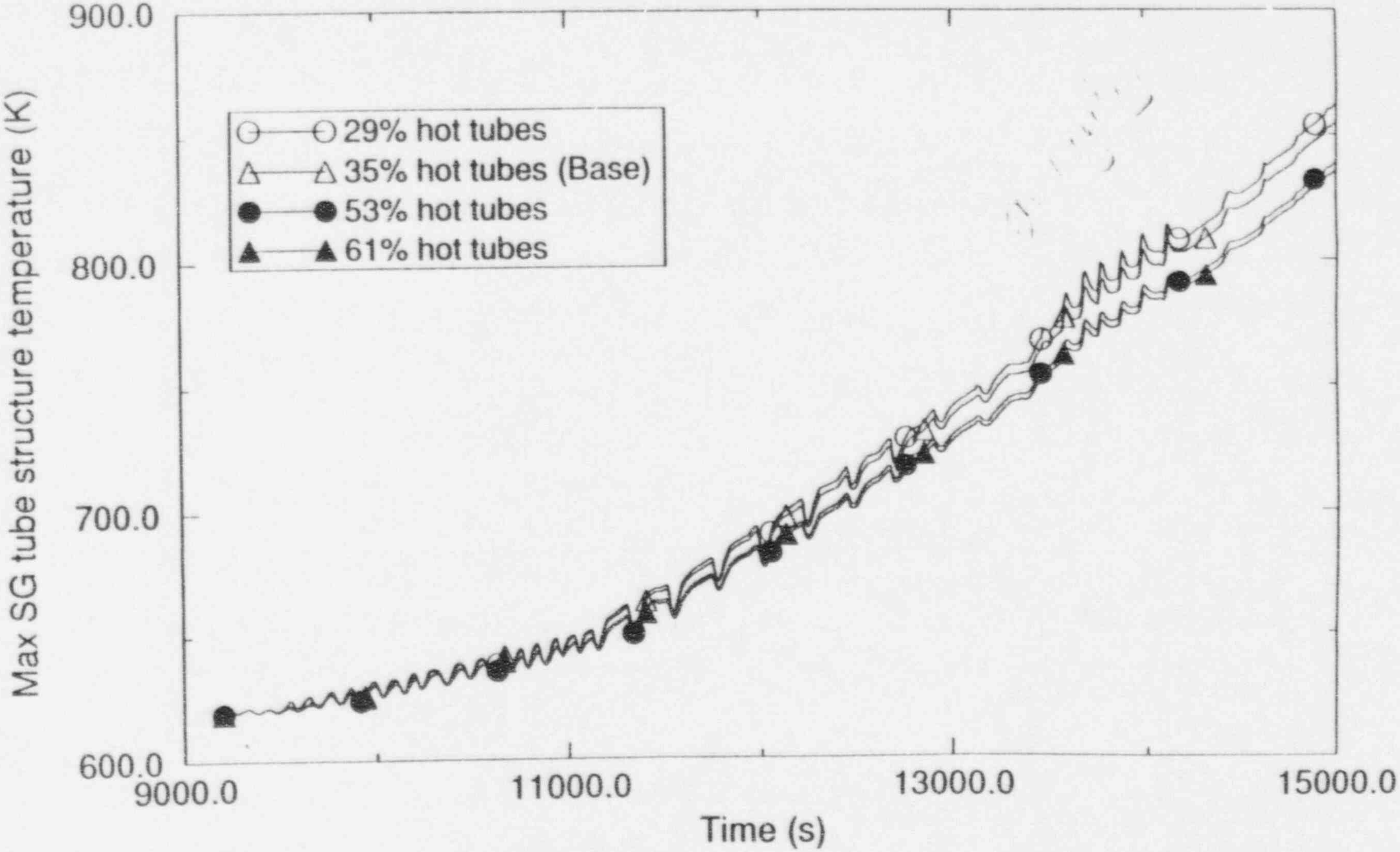
SCDAP/RELAP5 Surry Loop Calculations

- Variations considered
  - number of tubes participating in forward (hot) flow (measured between 29 and 61% of the SG tube bundle)
  - mixing fraction (measured between 0.76 and 0.89)
  - recirculation ratio (measured between 1.69 and 2.39)

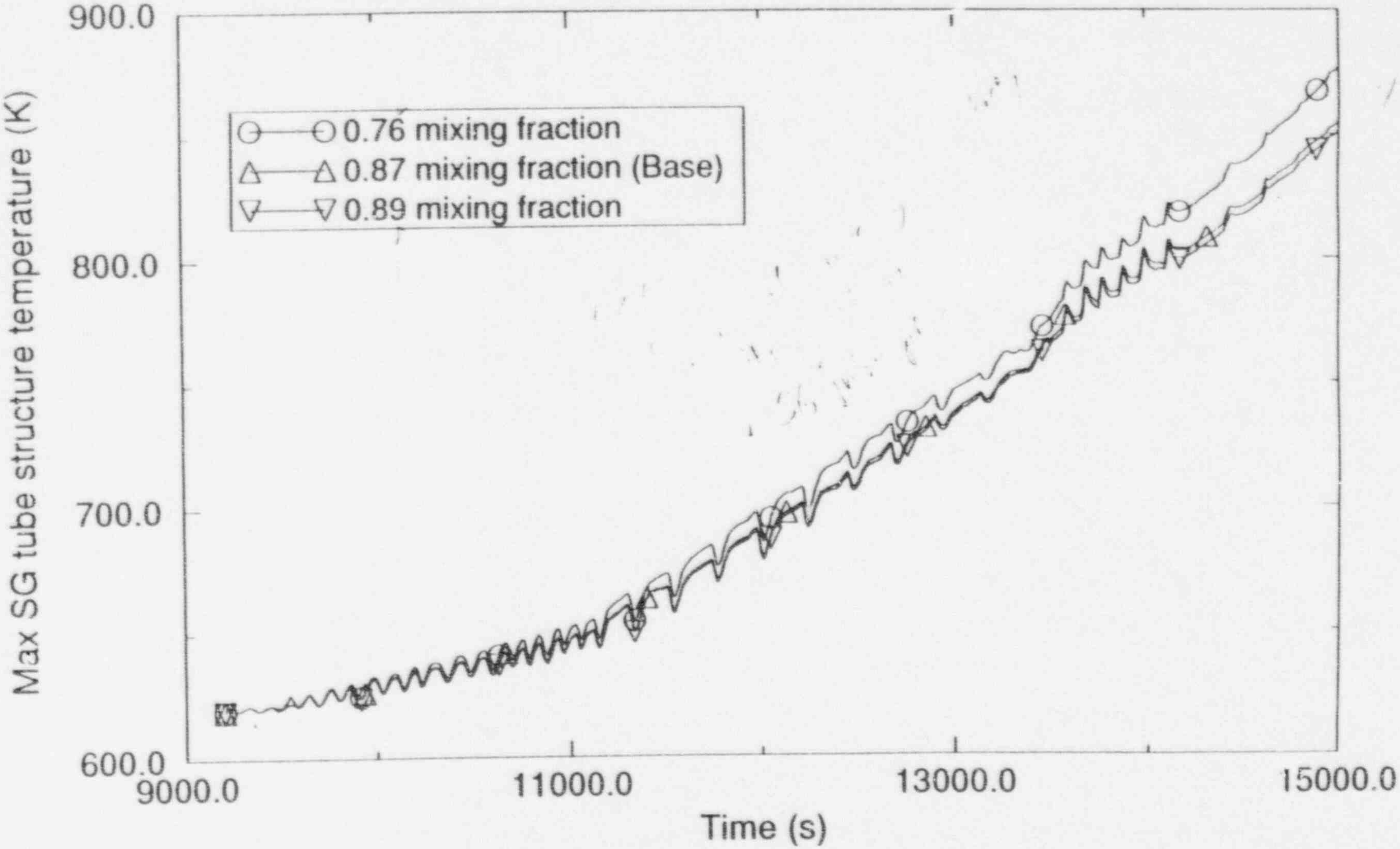
- Separate loop calculations were included so that SG tube temperatures could be evaluated over the measured range

- All loop calculations were completed with variation of only one parameter at a time (all other parameters were held constant at "average" conditions)

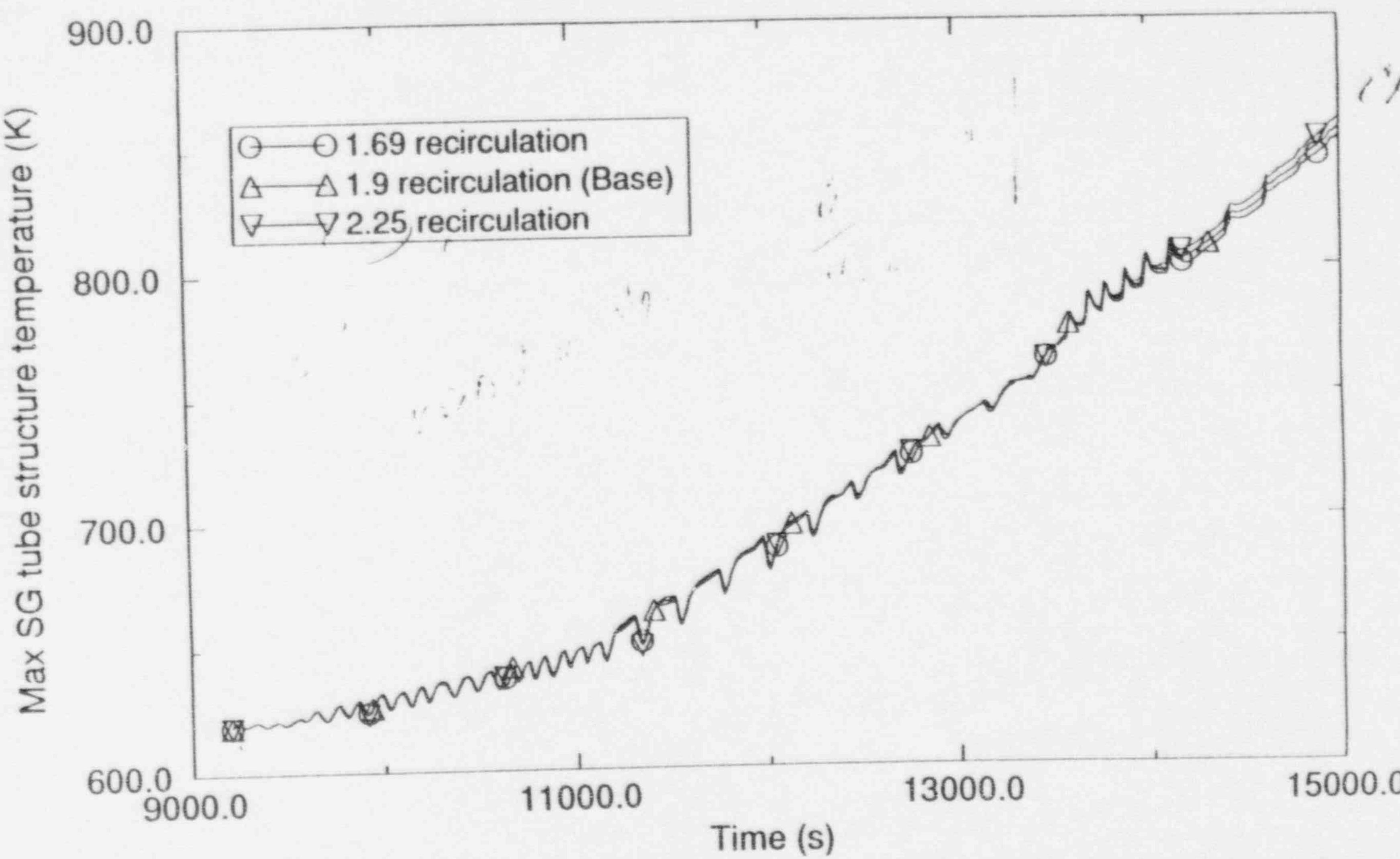
Surry Loop Results



Surry Loop Results



Surry Loop Results

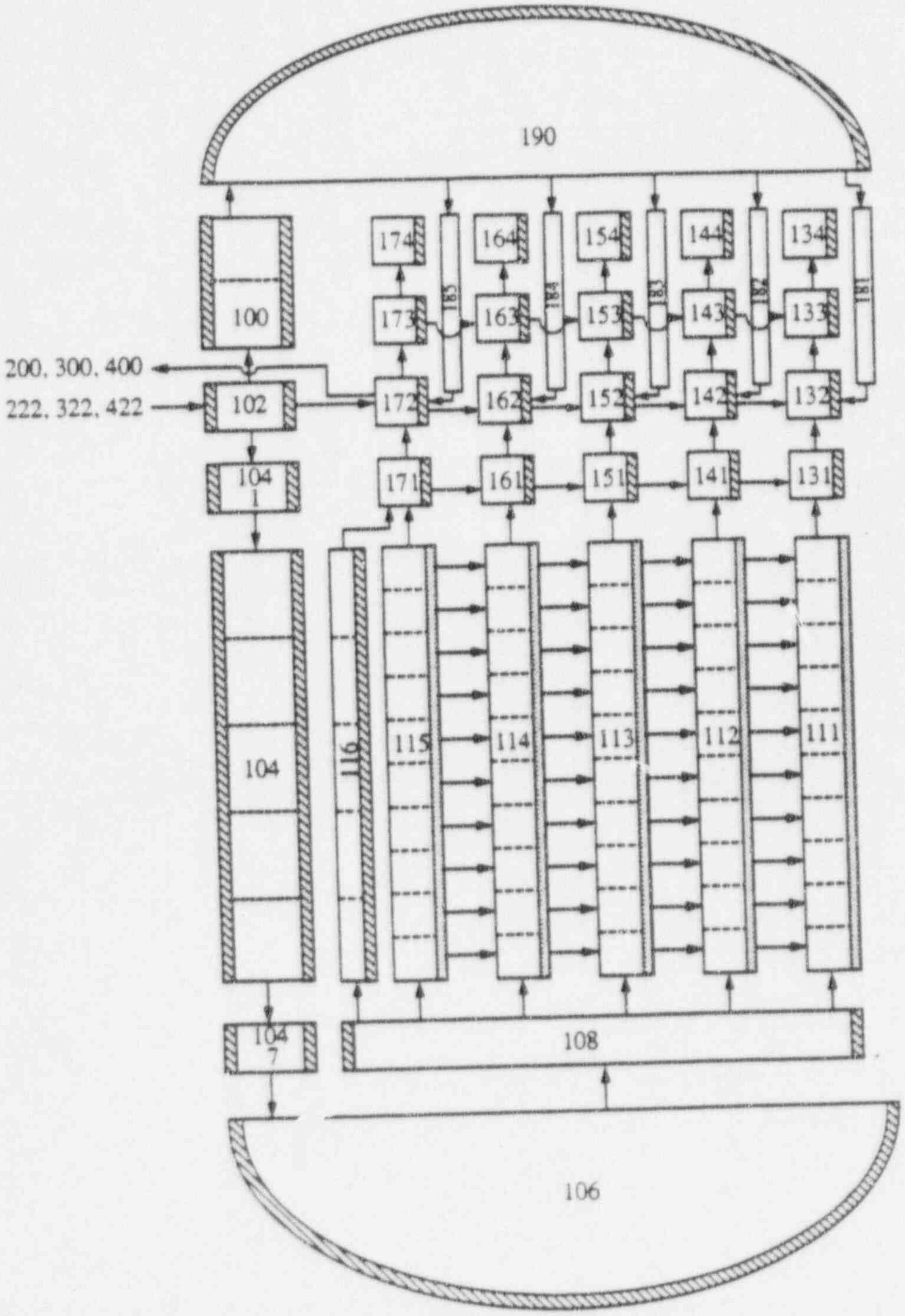


SCDAP/RELAP5 Surry Plant Calculations

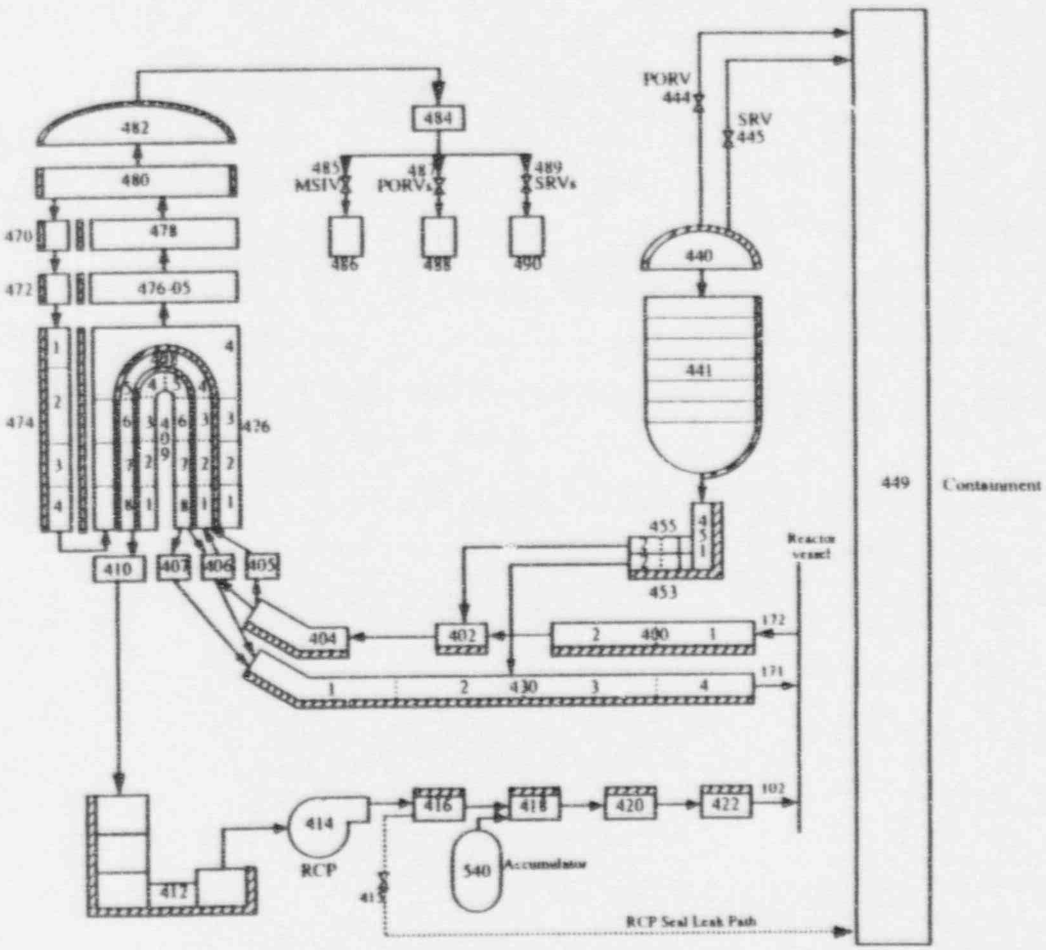
- TMLB' transient in all calculations
  - without recovery and without operator action
  - with creep rupture monitoring of the surge line, hot legs, and SG tubes
  - with modeling provisions to allow development of in-vessel, full loop, and hot leg countercurrent natural circulation

- Benchmarked with
  - ~9% of core energy deposited in structures in each loop
  - 35% of SG tube bundle available for forward (hot) flow
  - mixing fractions at ~0.87
  - recirculation ratio at ~1.9

Vessel Nodalization



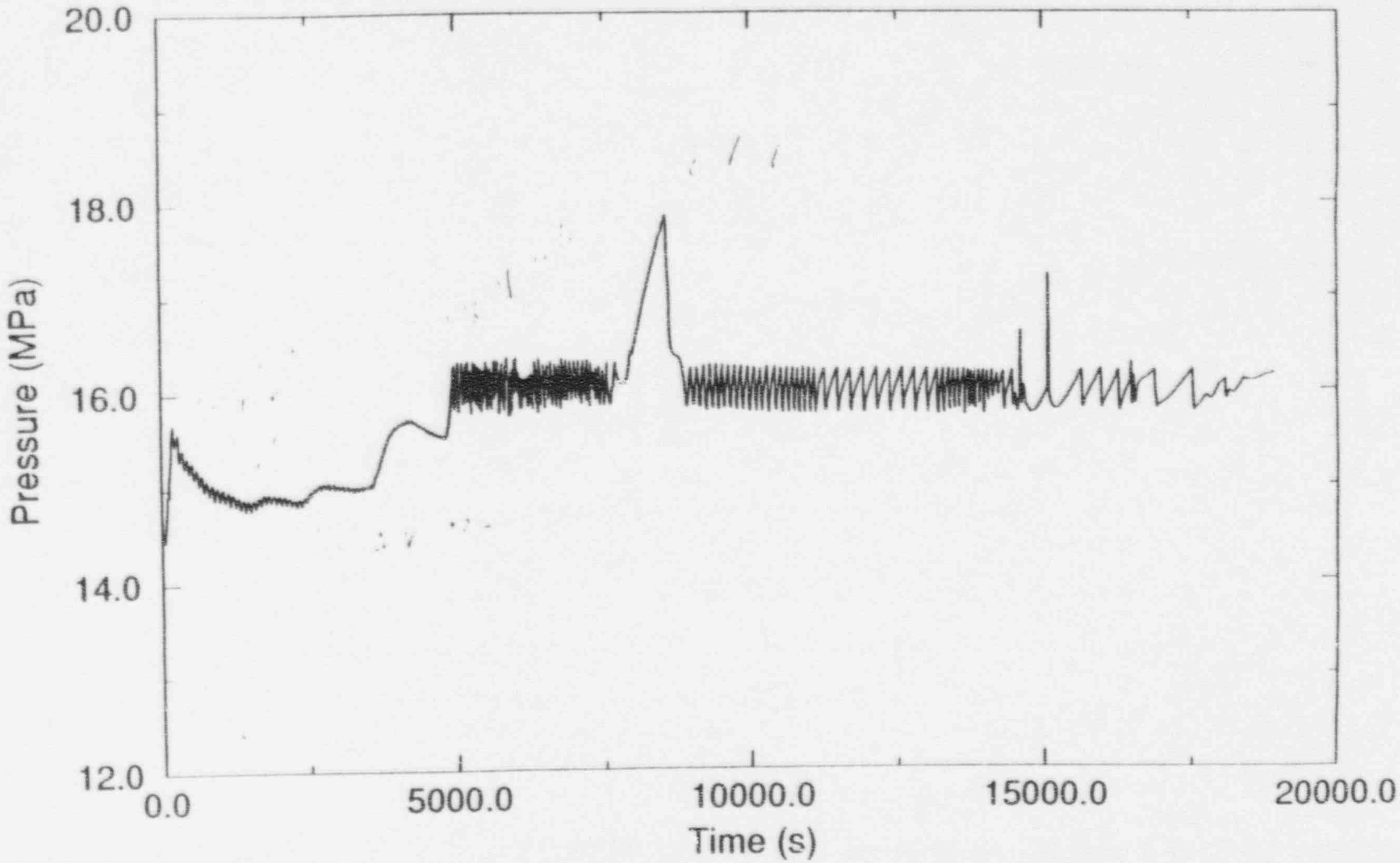
Pressurizer Loop Nodalization



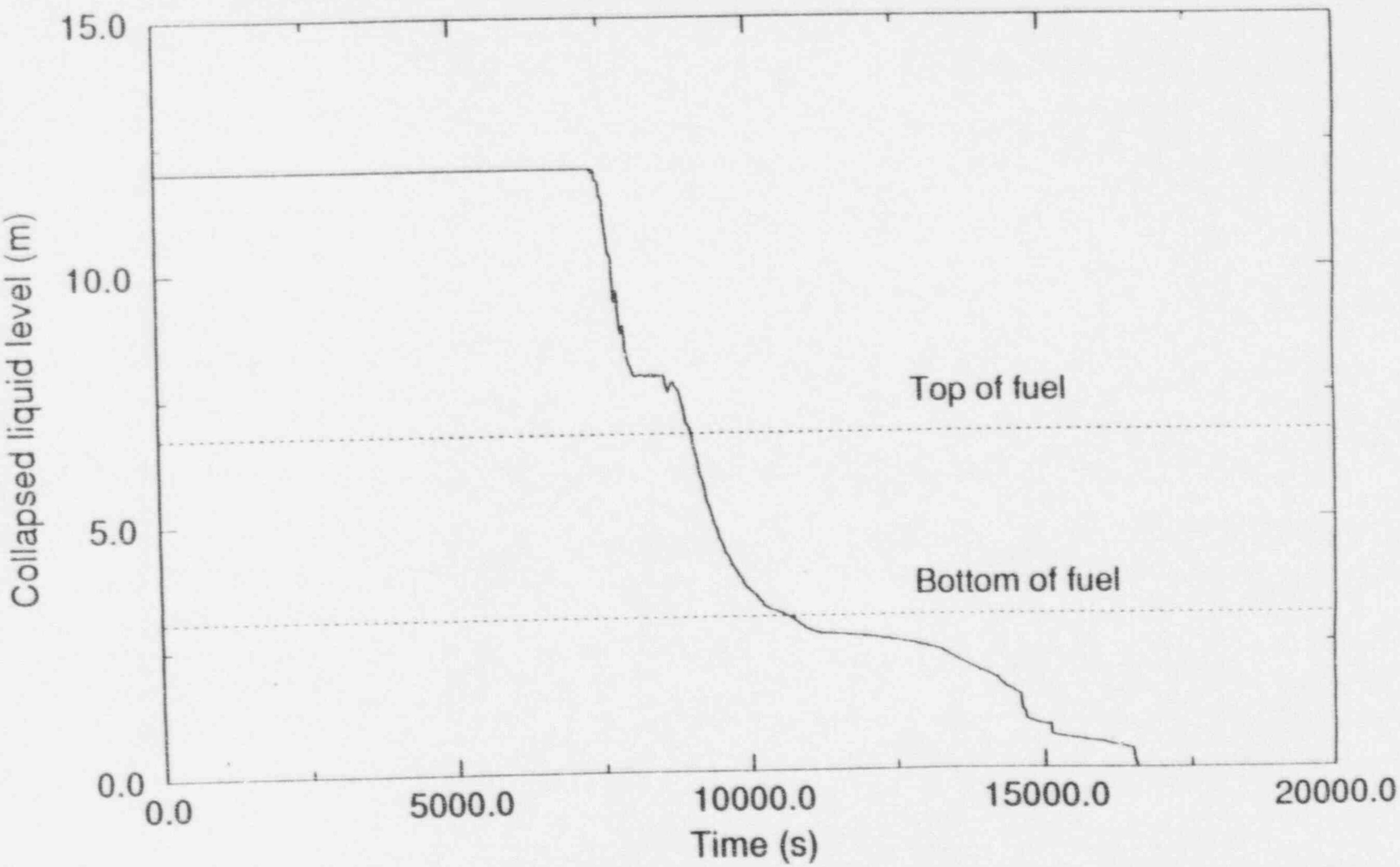
SCDAP/RELAP5 Surry Plant Calculations

Assumed Severe Accident Condition	Case						
	1	2	3	4	5	6	7
TMLB' w/o recovery and w/o operator actions	x	x	x	x	x	x	x
Depressurization of pzr loop SG secondary via failed ADV			x	x		x	
w/o RCS depressurization via pressure boundary failure	x		x		x	x	x
RCS depressurization via pressure boundary failure		x		x			
35% hot tubes/65% cold tube nodalization	x	x	x	x			x
53% hot tubes/47% cold tubes nodalization					x	x	
Depressurization of all SG secondaries via failed ADVs							x

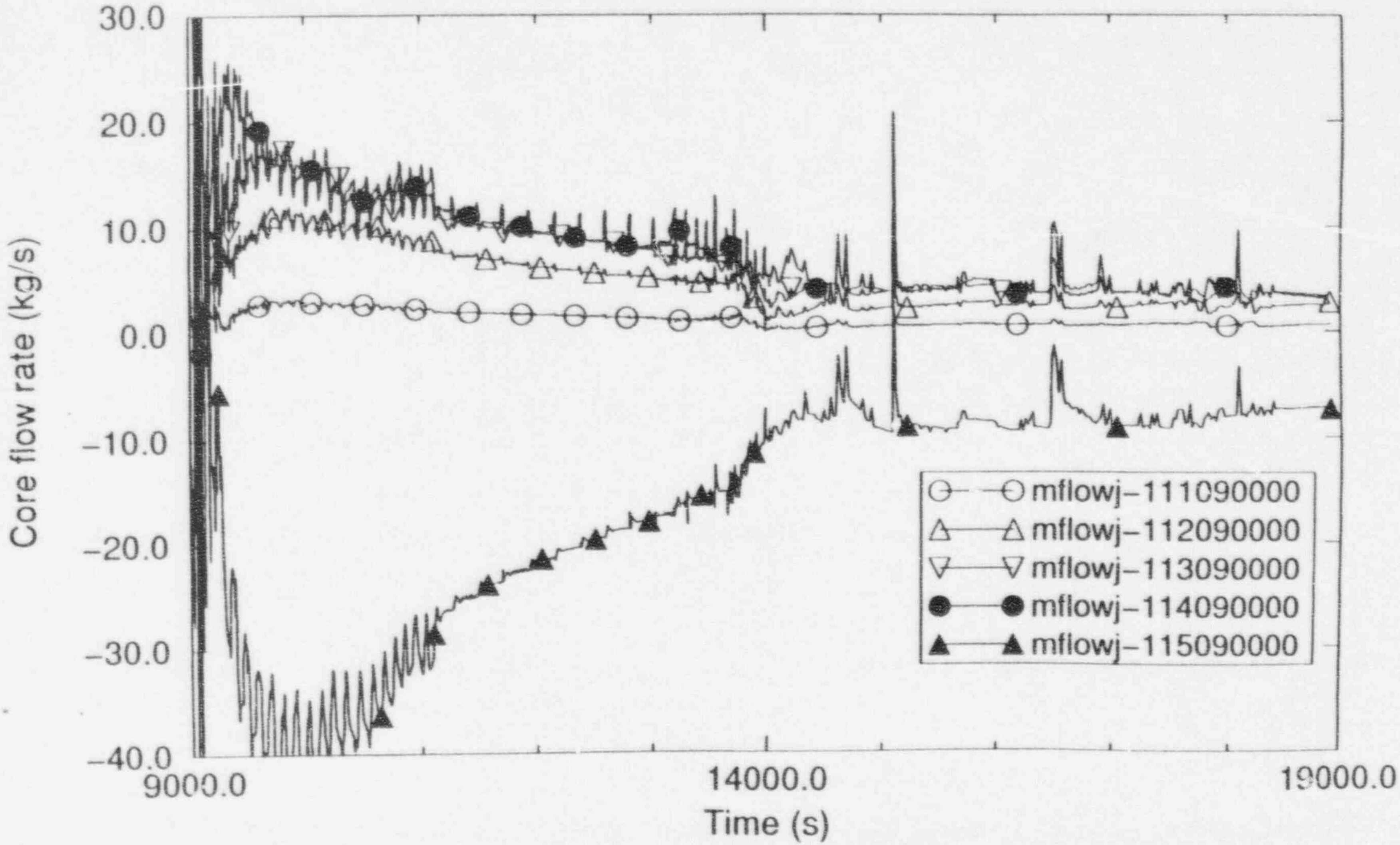
Surry Plant Results - Case 1



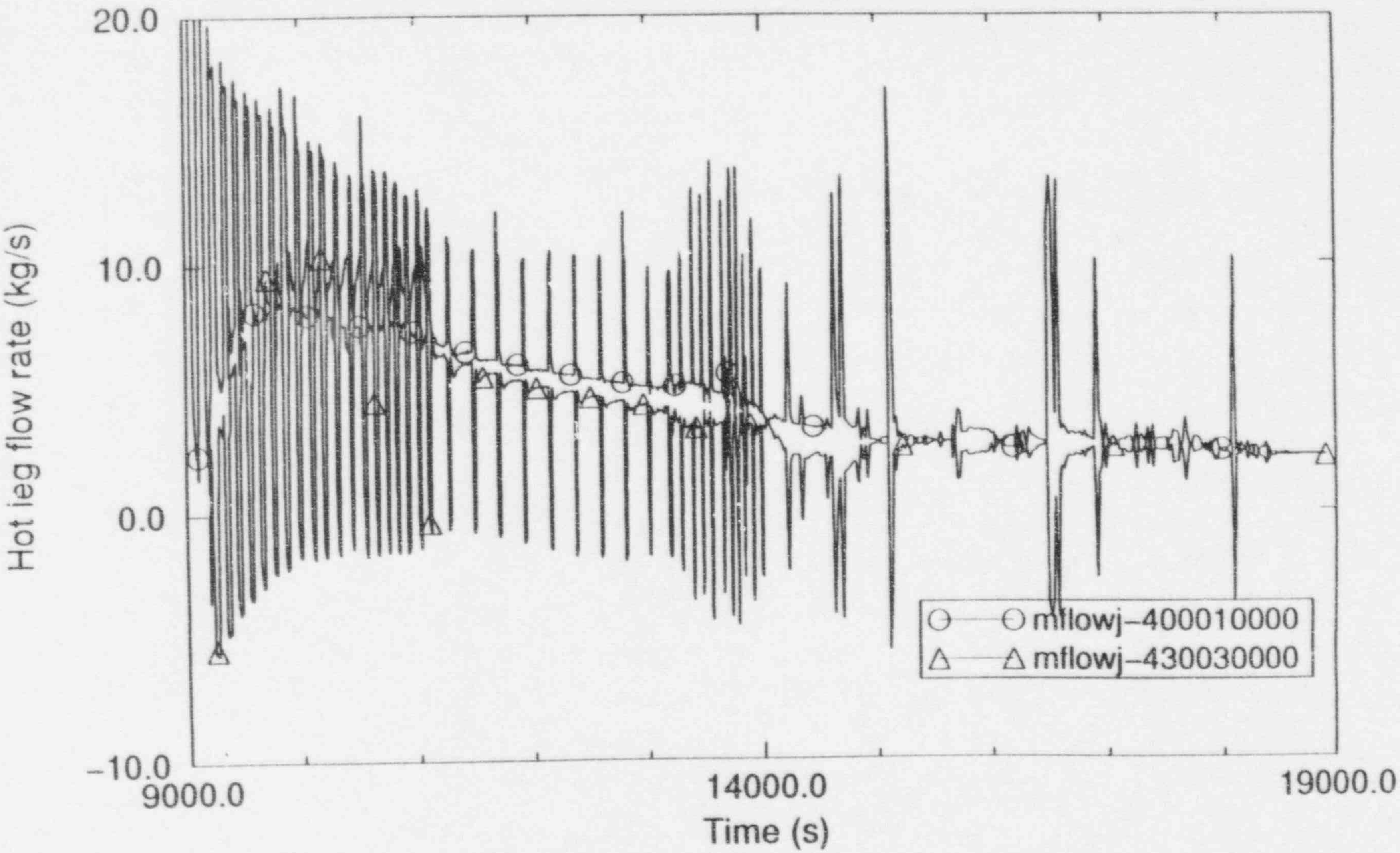
Surry Plant Results - Case 1



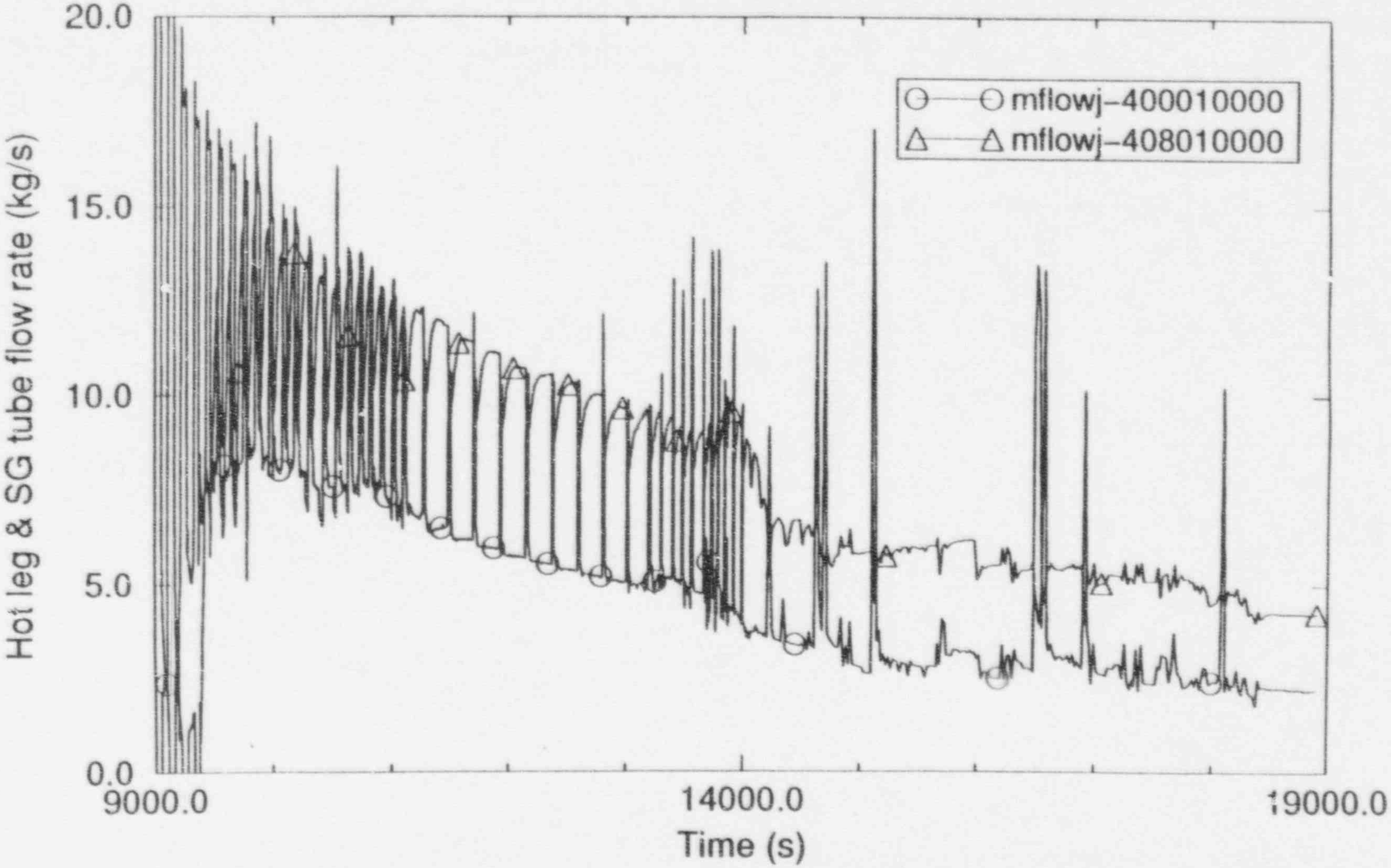
Surry Plant Results - Case 1



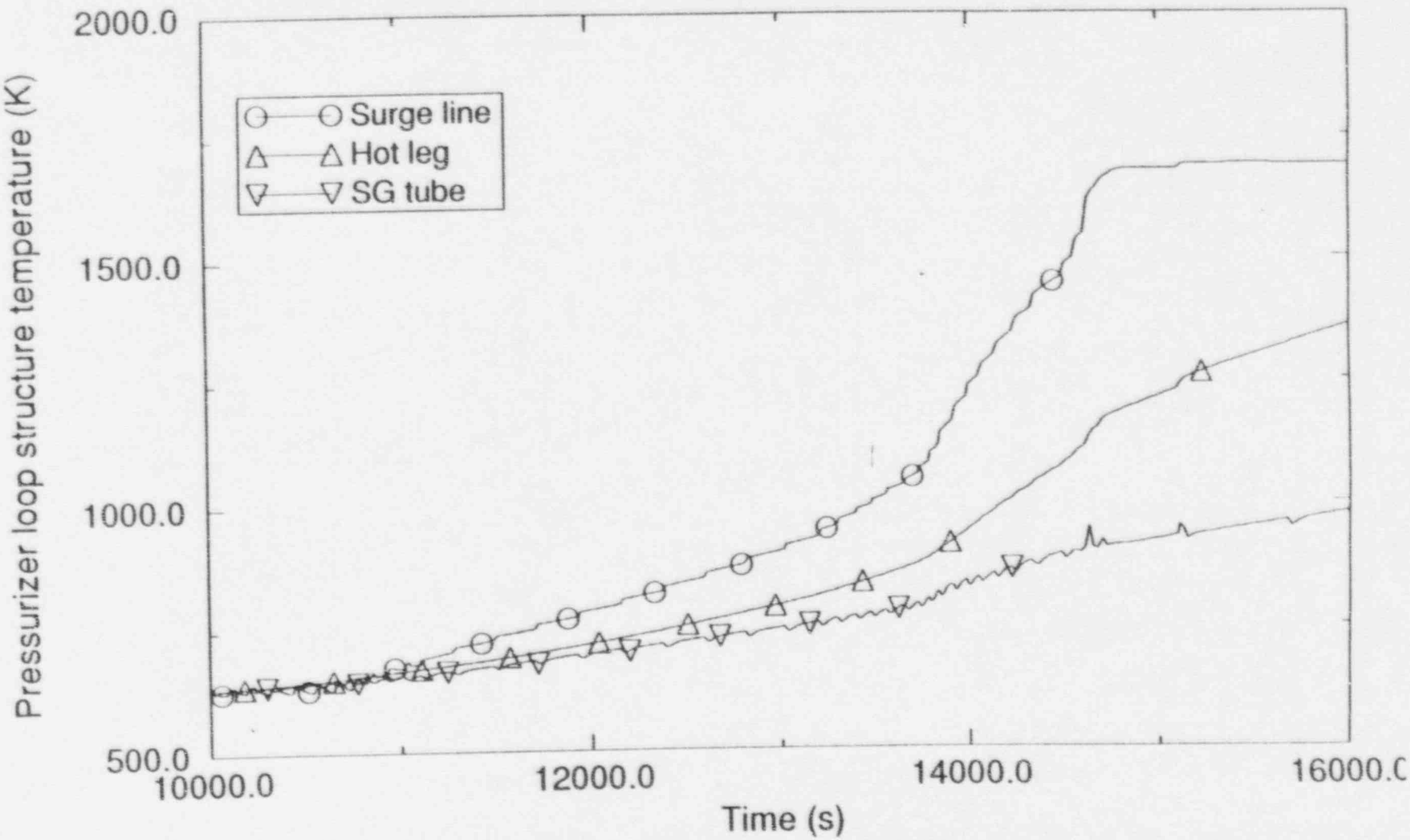
Surry Plant Results - Case 1



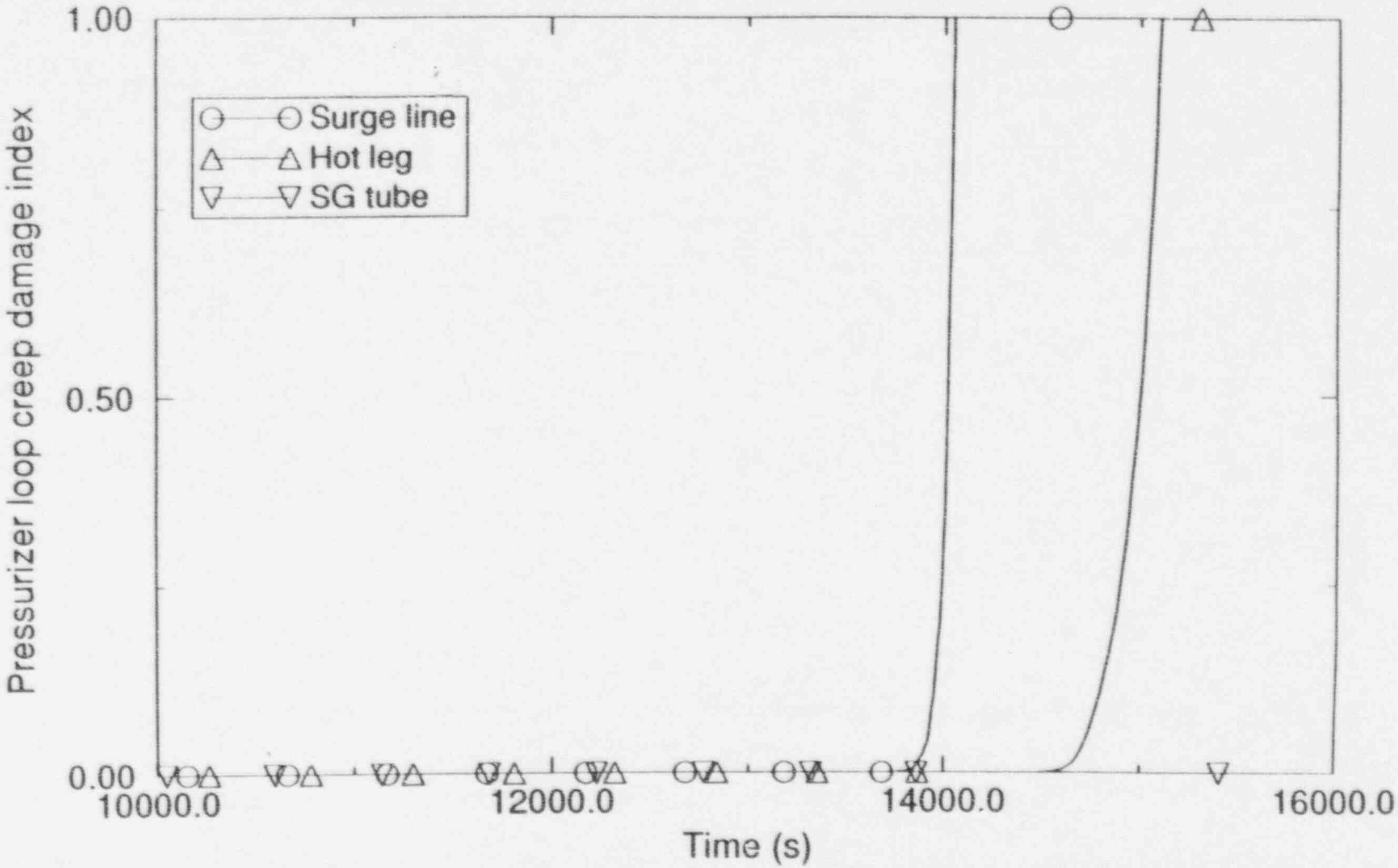
Surry Plant Results - Case 1



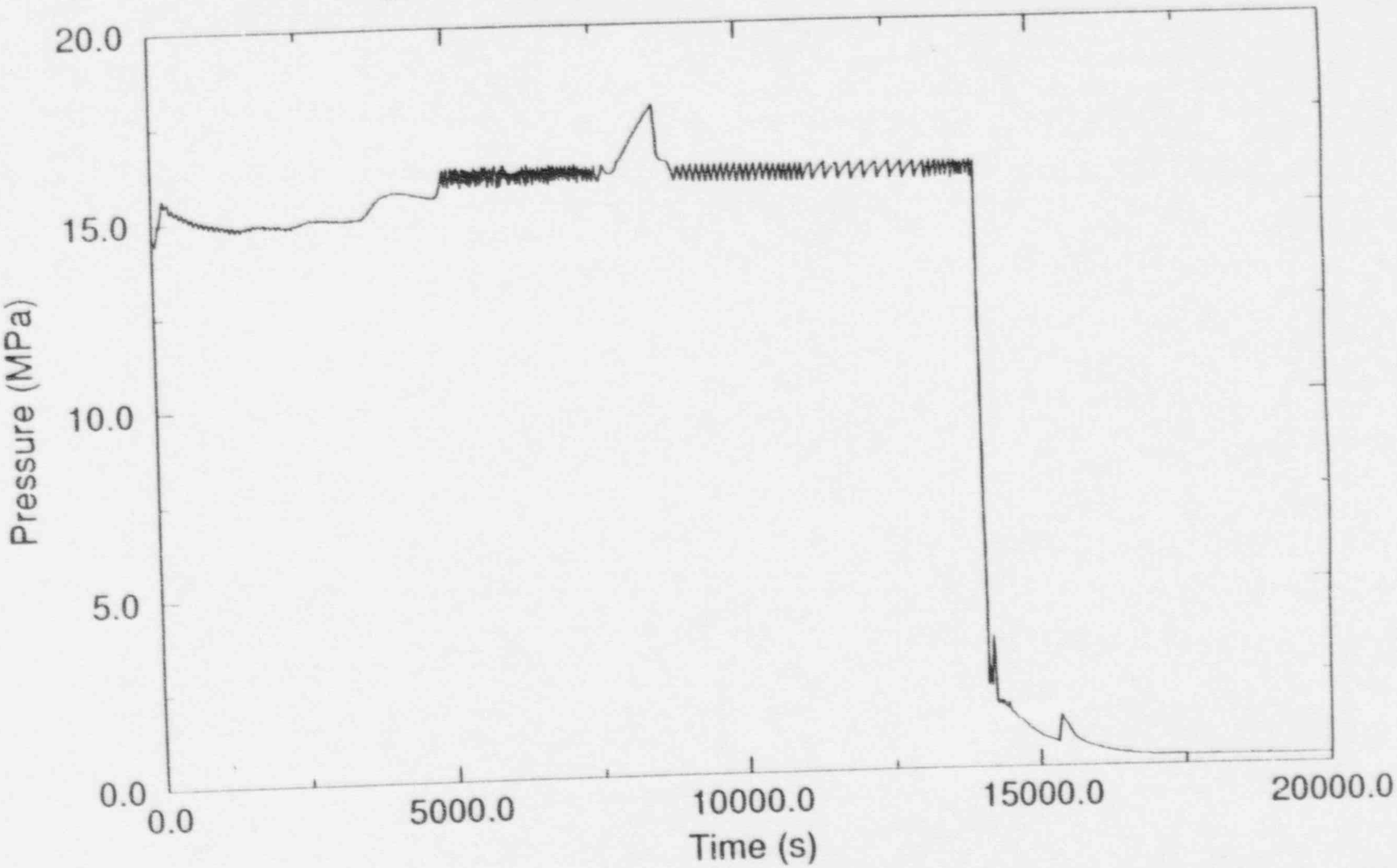
Surry Plant Results - Case 1



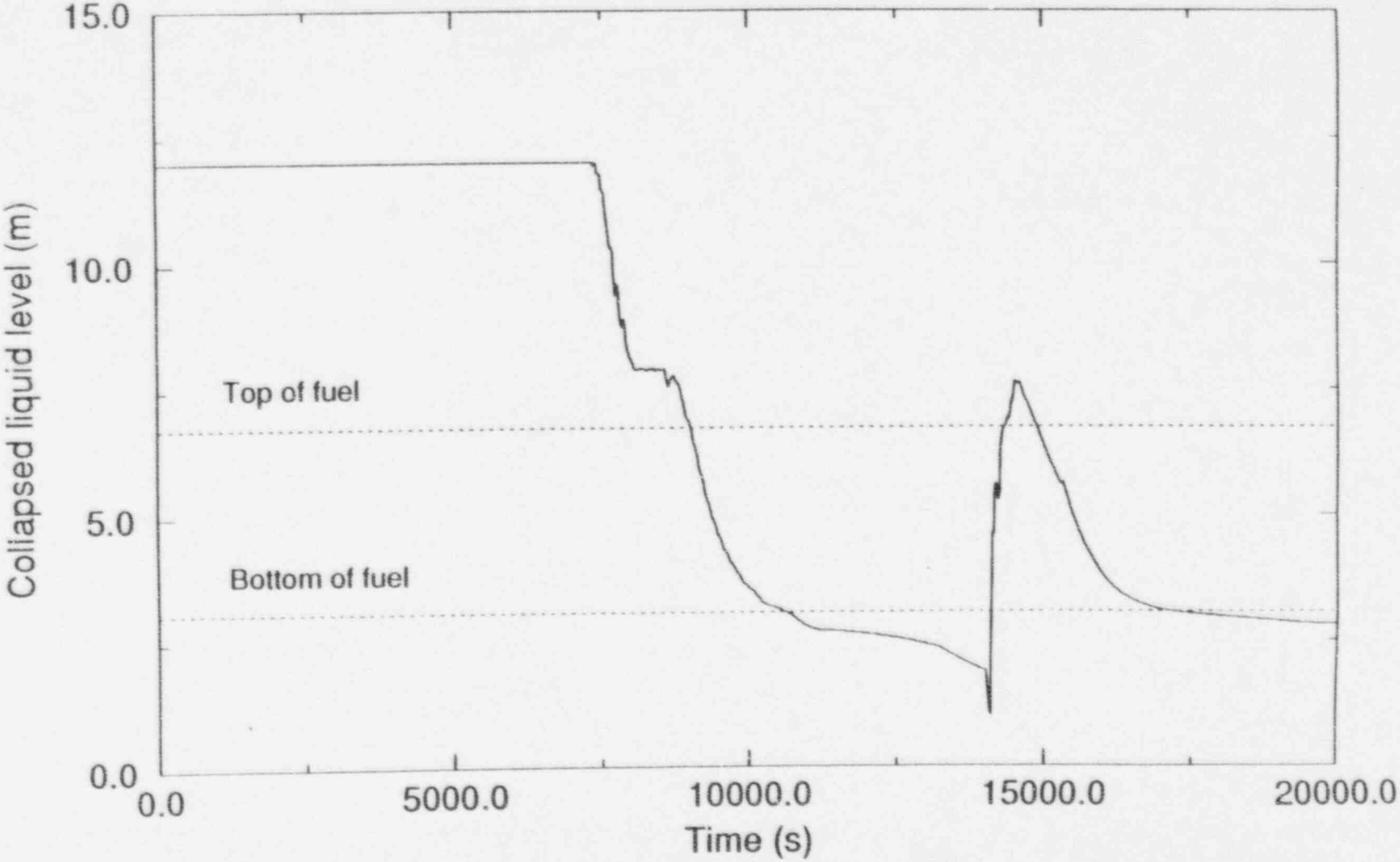
Surry Plant Results - Case 1



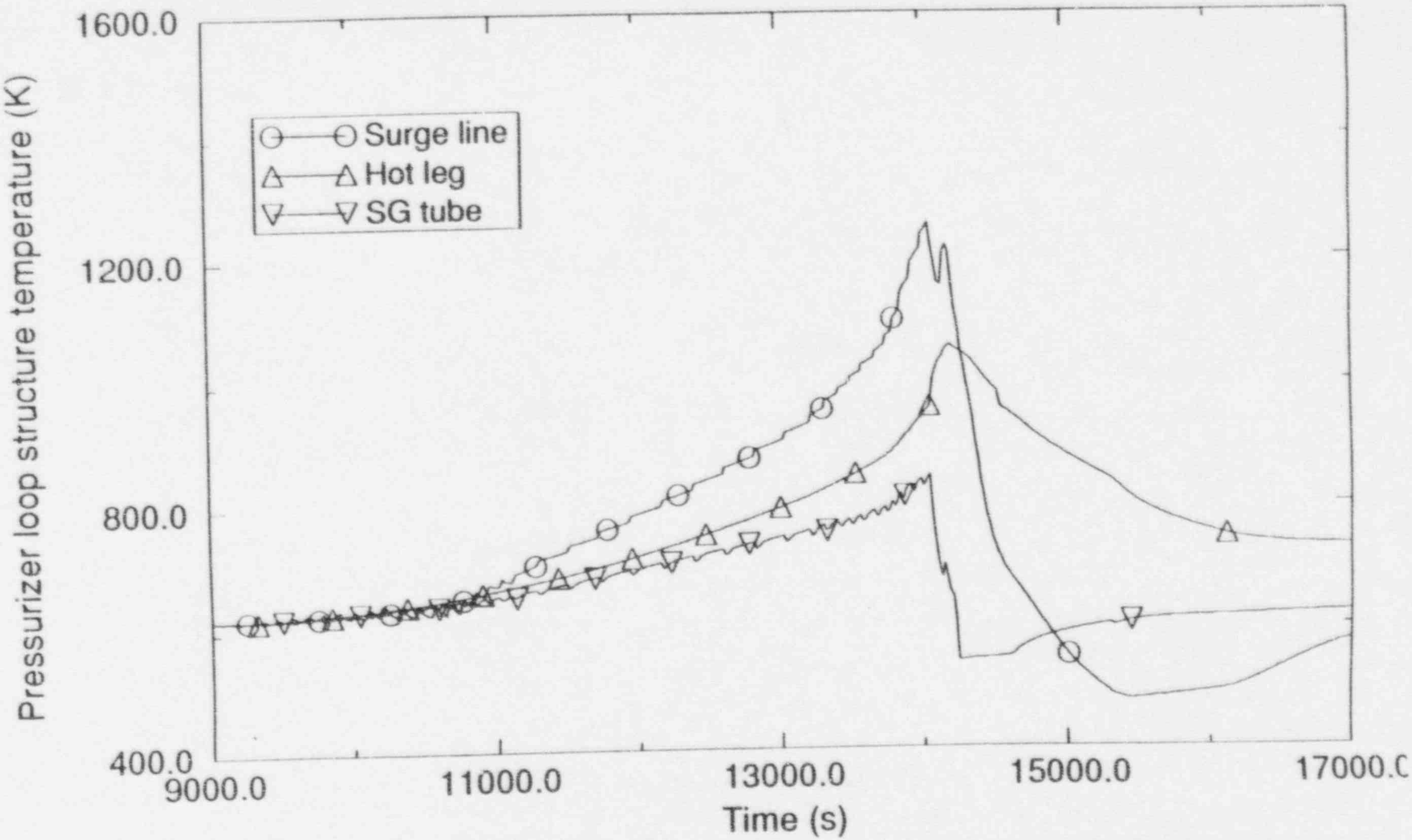
Surry Plant Results - Case 2



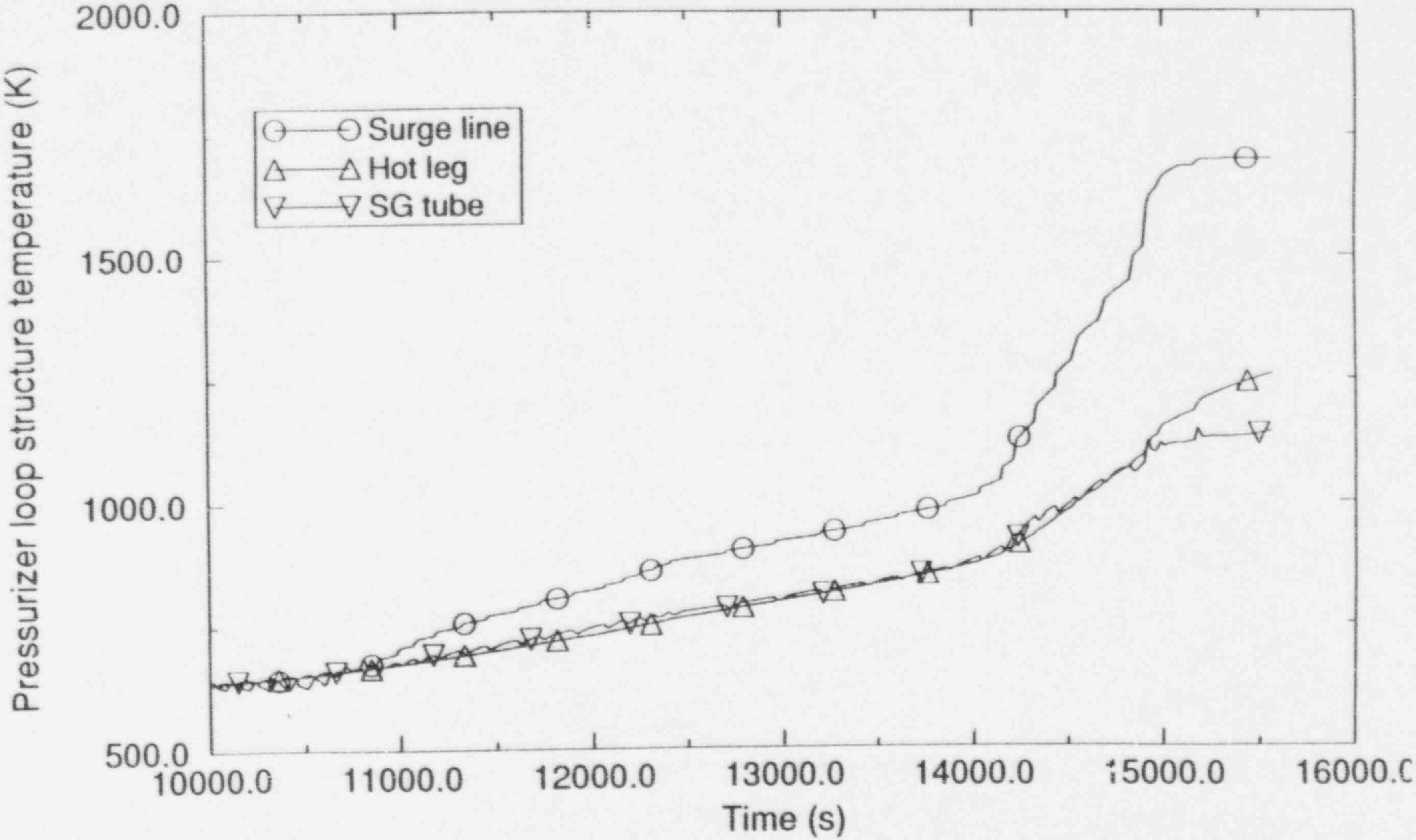
Surry Plant Results - Case 2



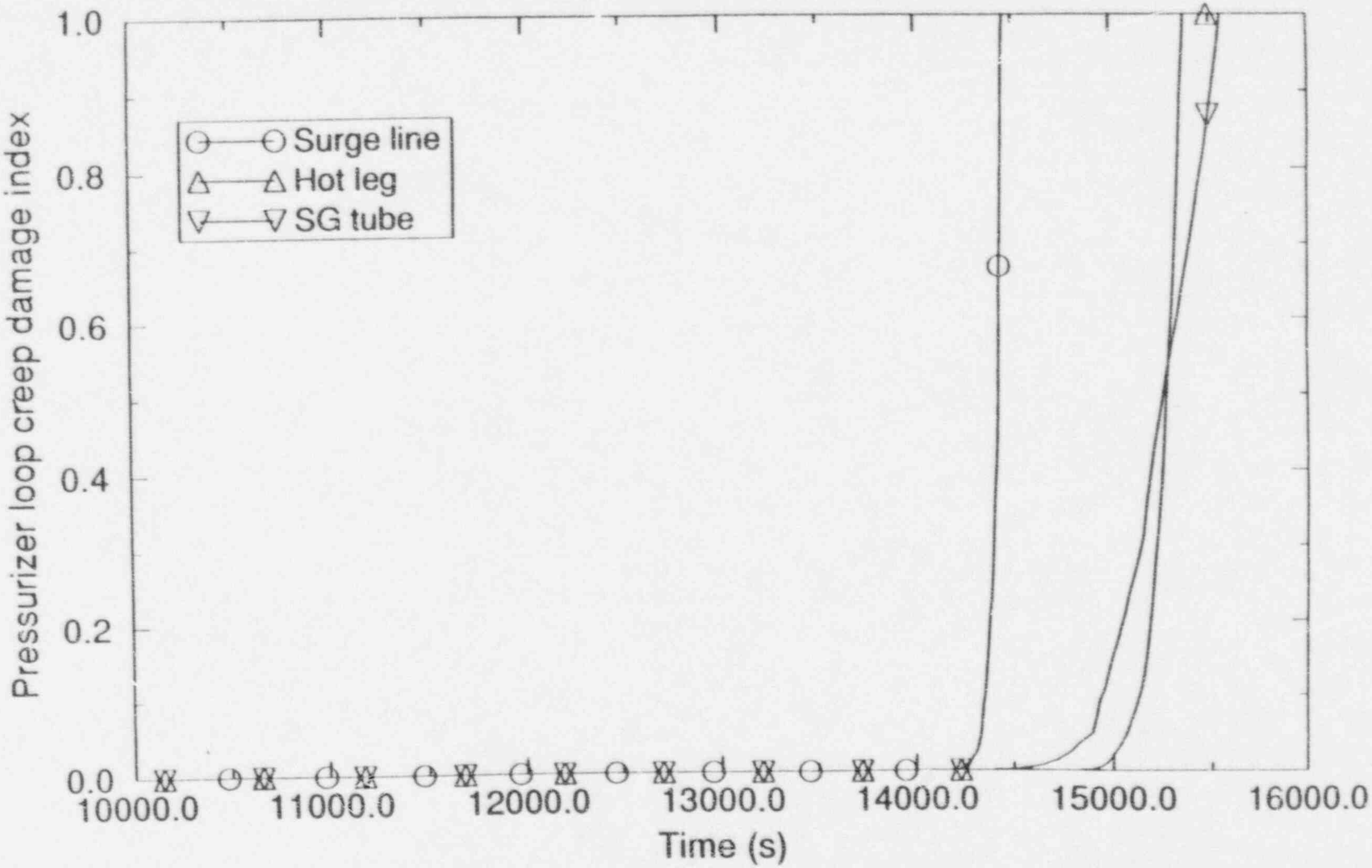
Surry Plant Results - Case 2



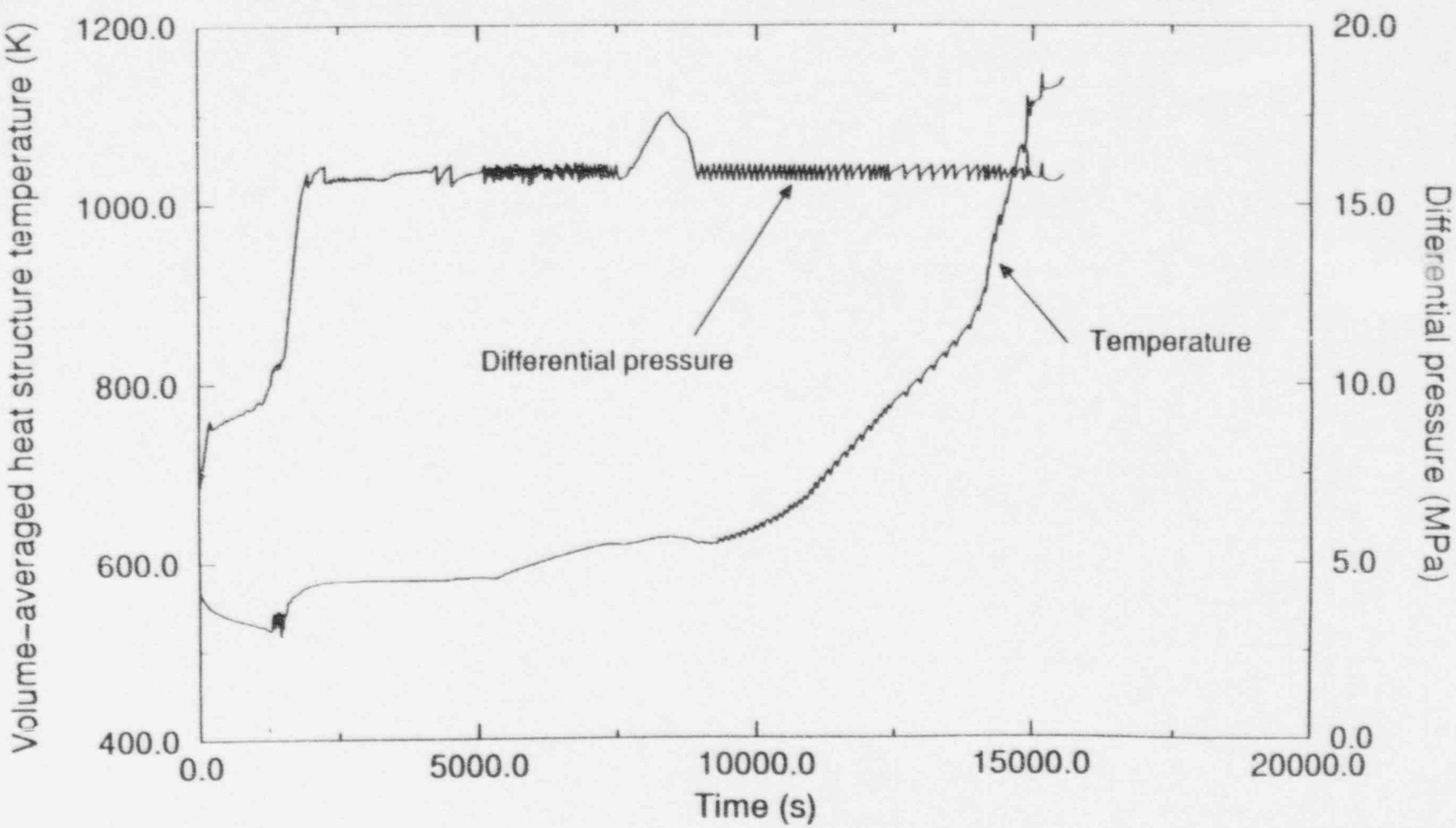
Surry Plant Results - Case 3



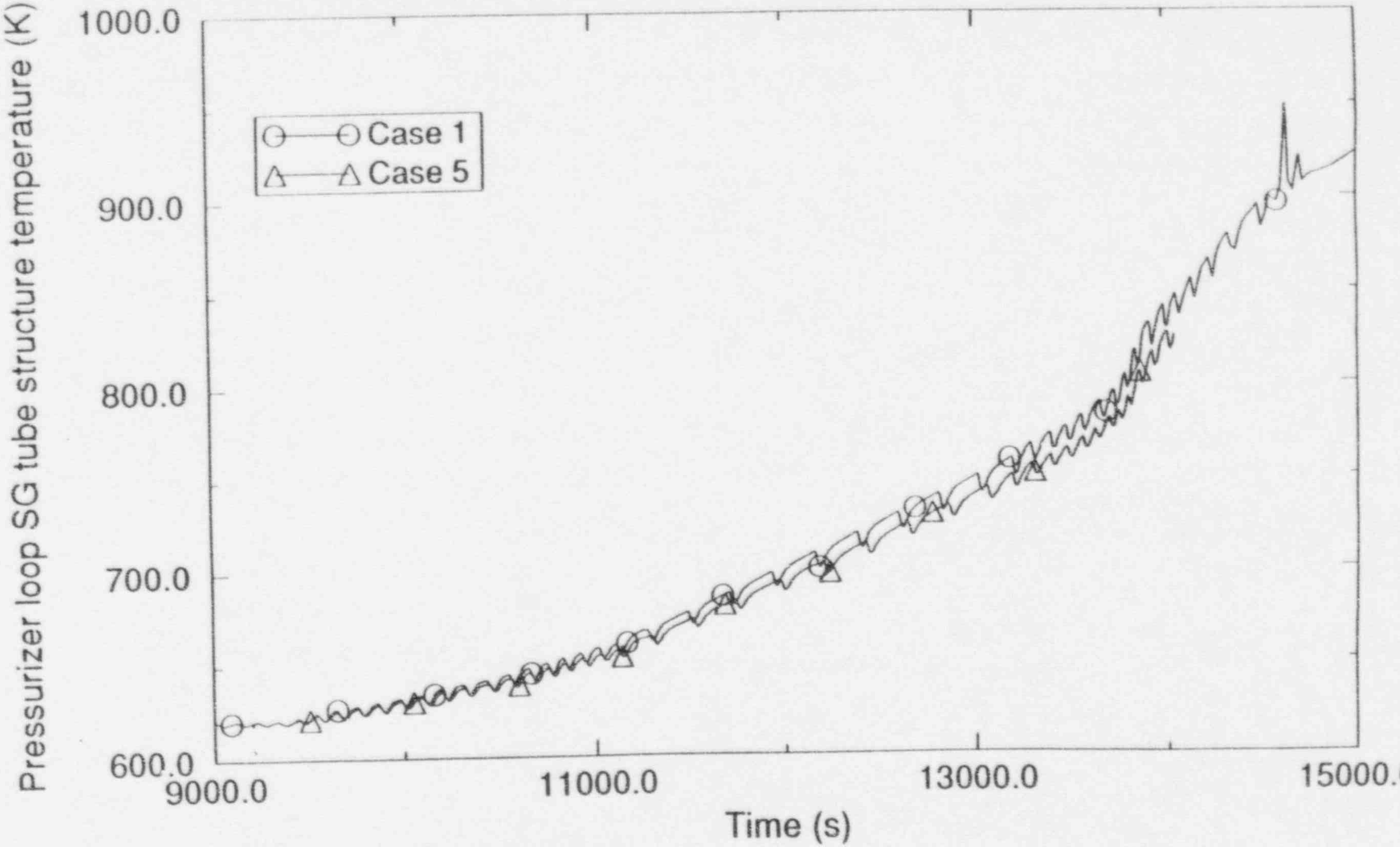
Surry Plant Results - Case 3



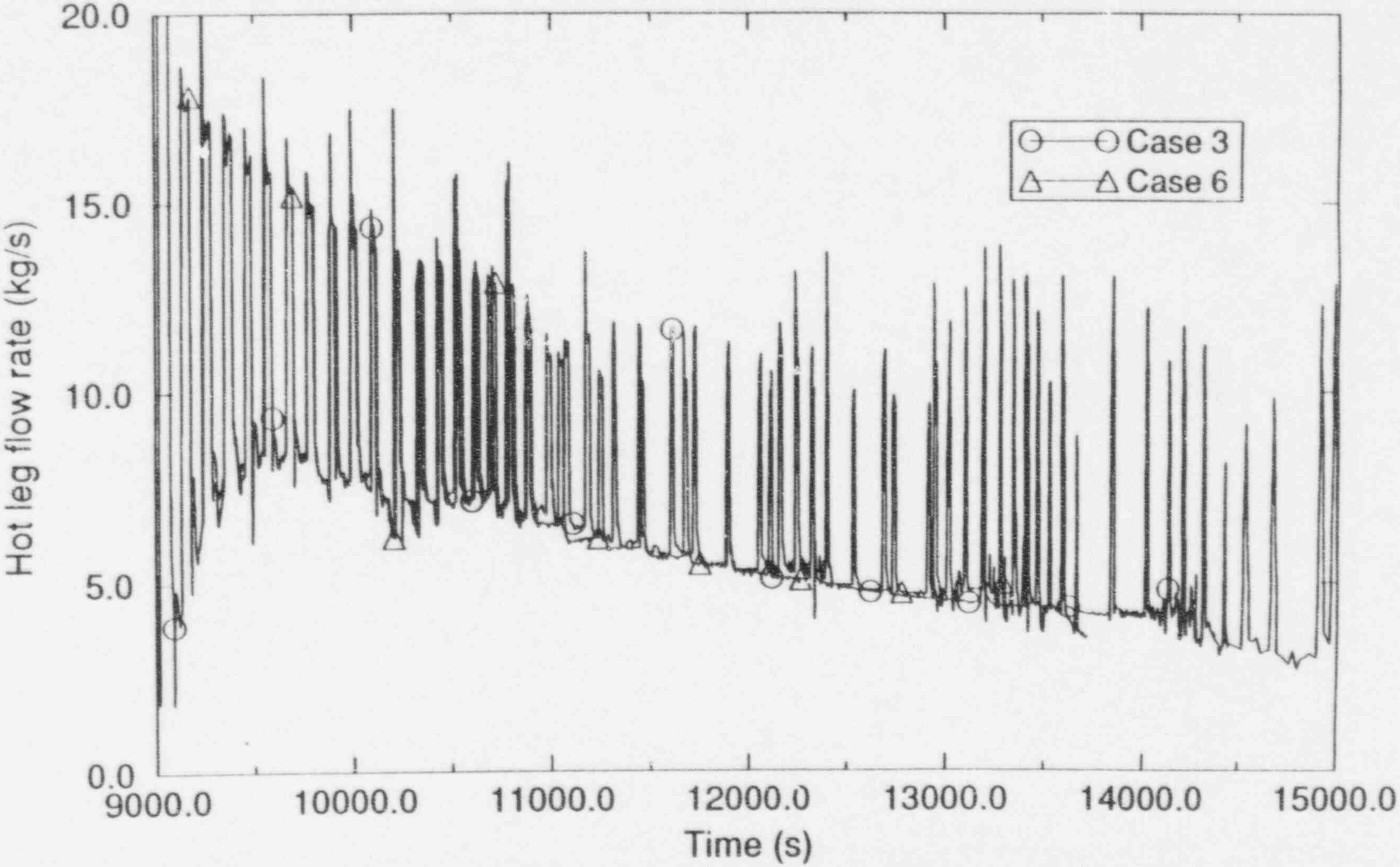
Surry Plant Results - Case 3



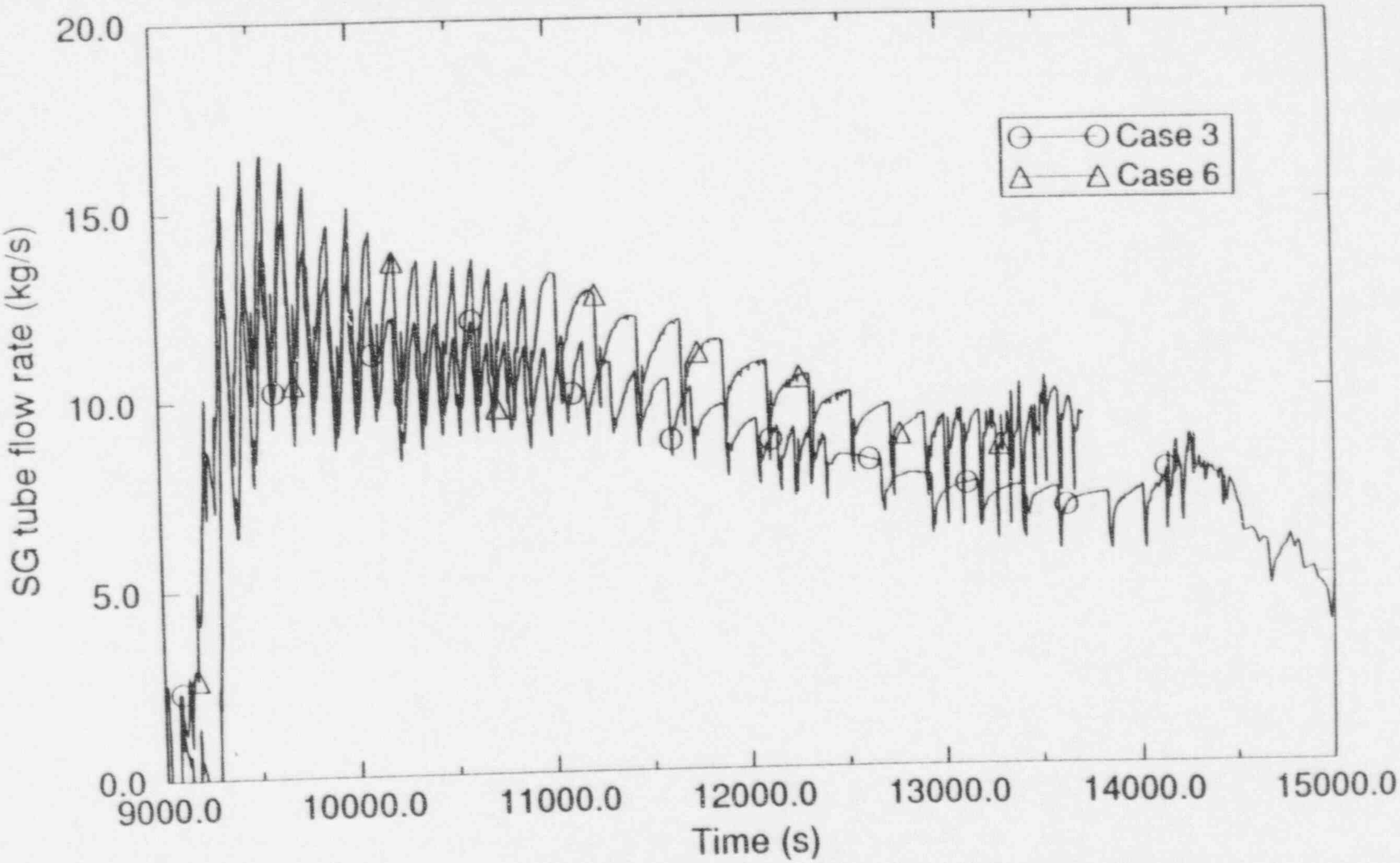
Surry Plant Results - Case 5



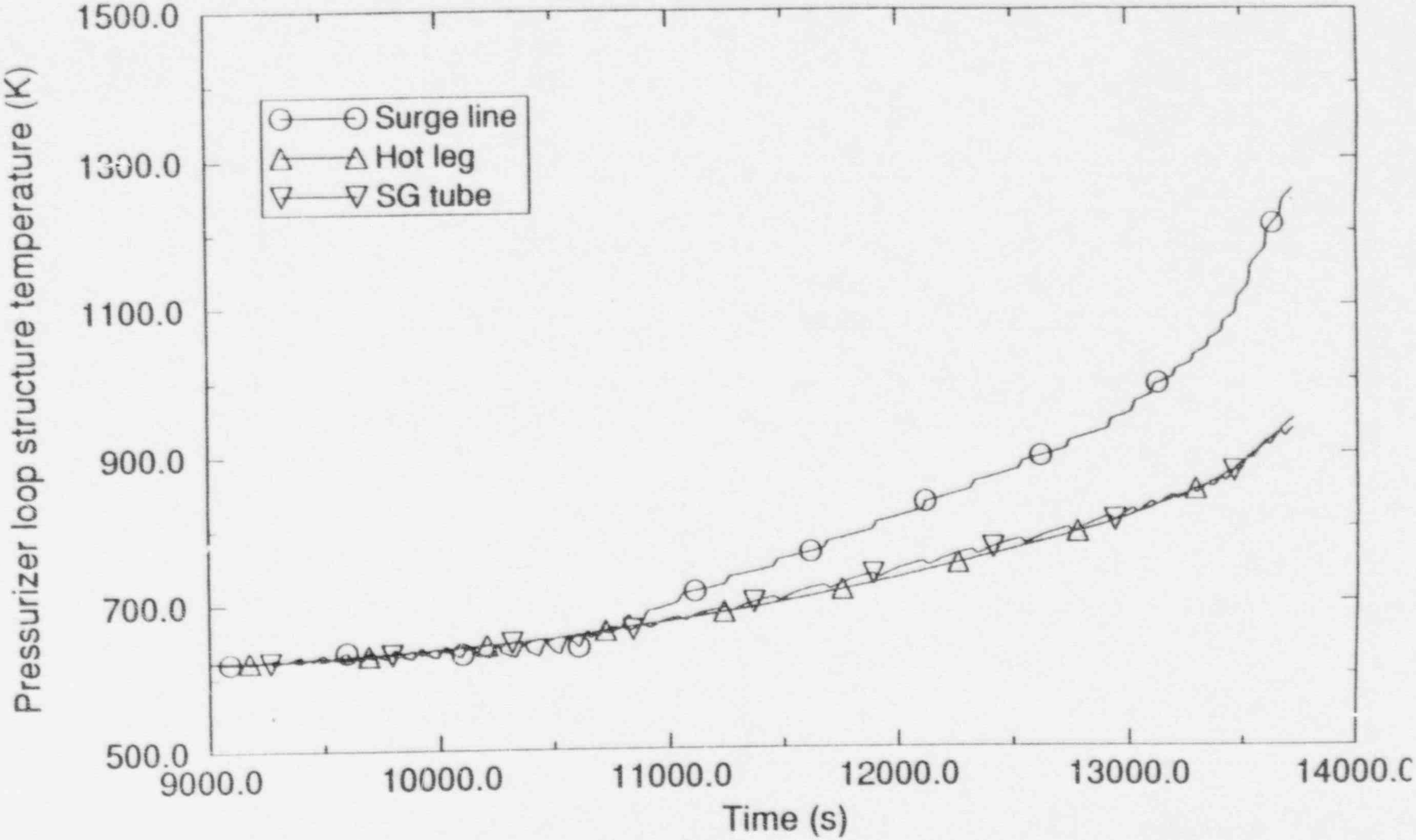
Surry Plant Results - Case 6



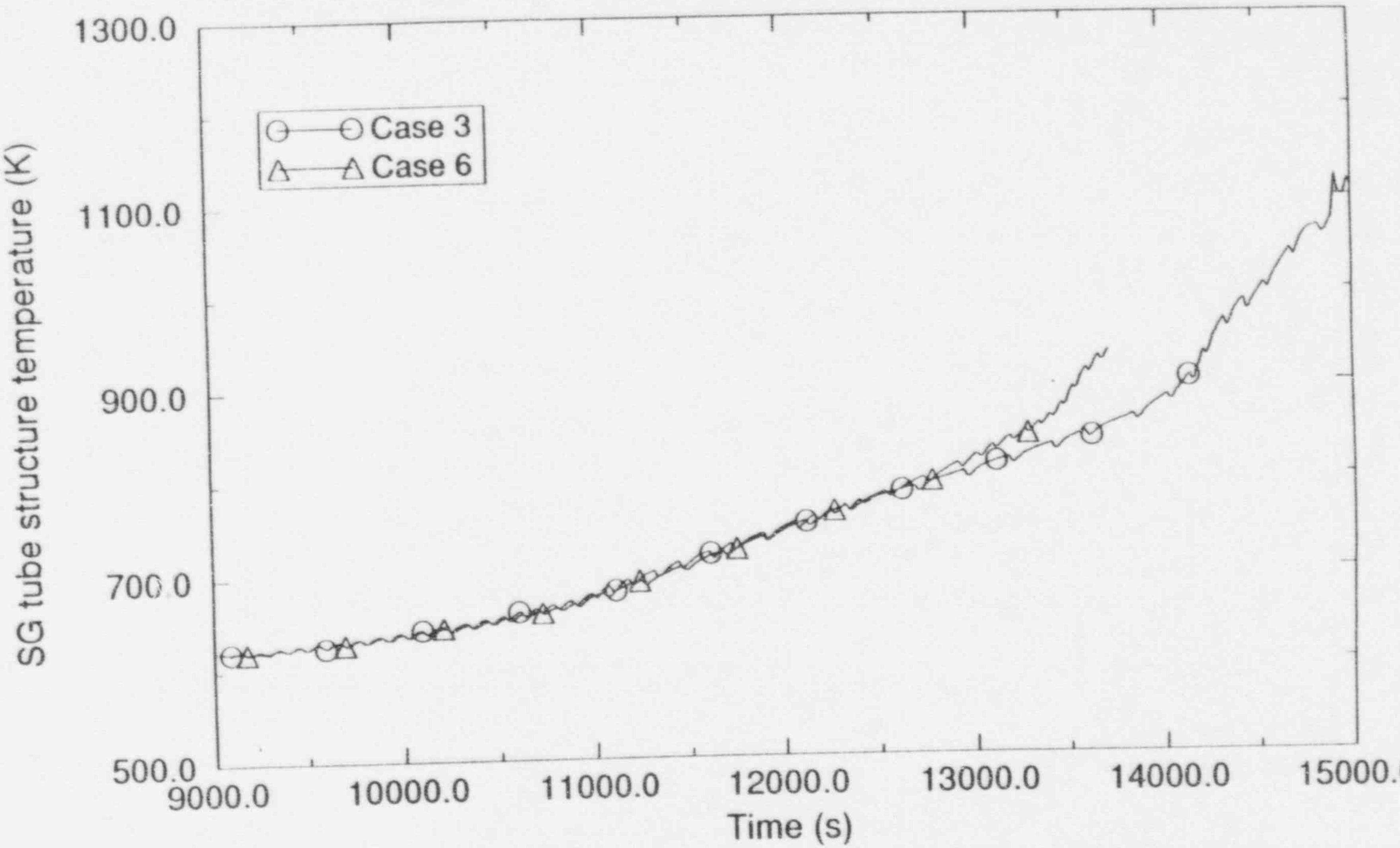
Surry Plant Results - Case 6



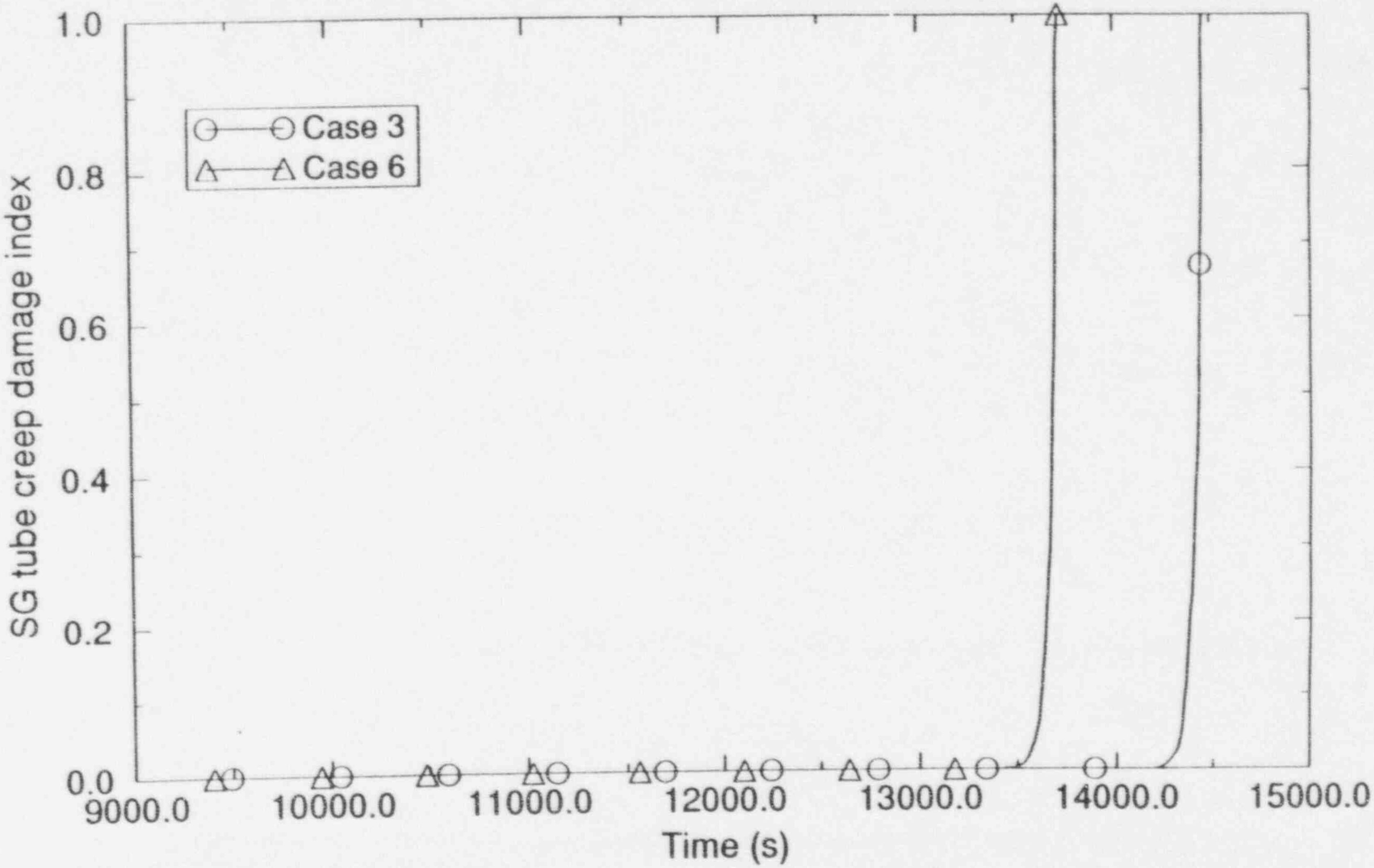
Surry Plant Results - Case 6



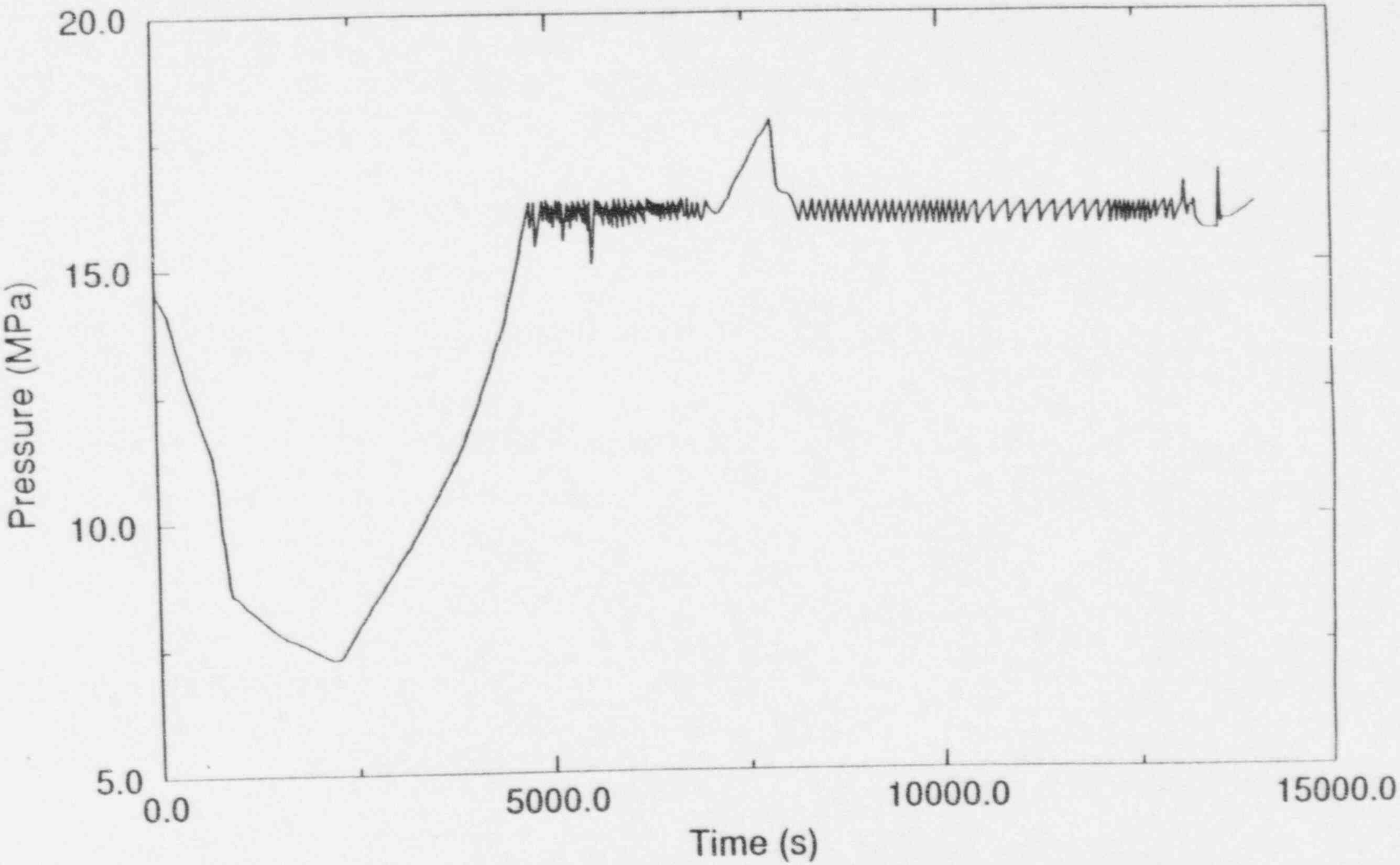
Surry Plant Results - Case 6



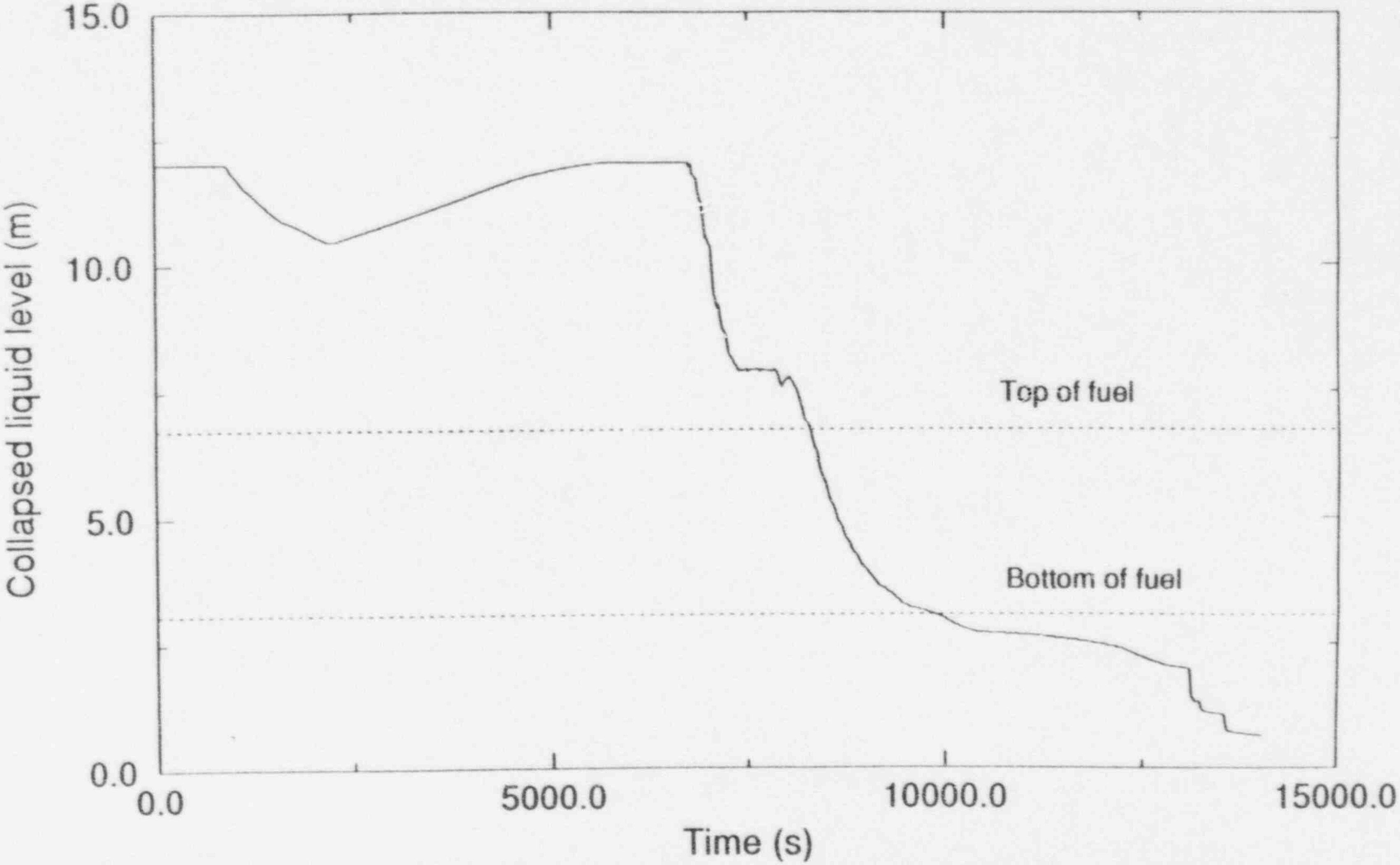
Surry Plant Results - Case 6



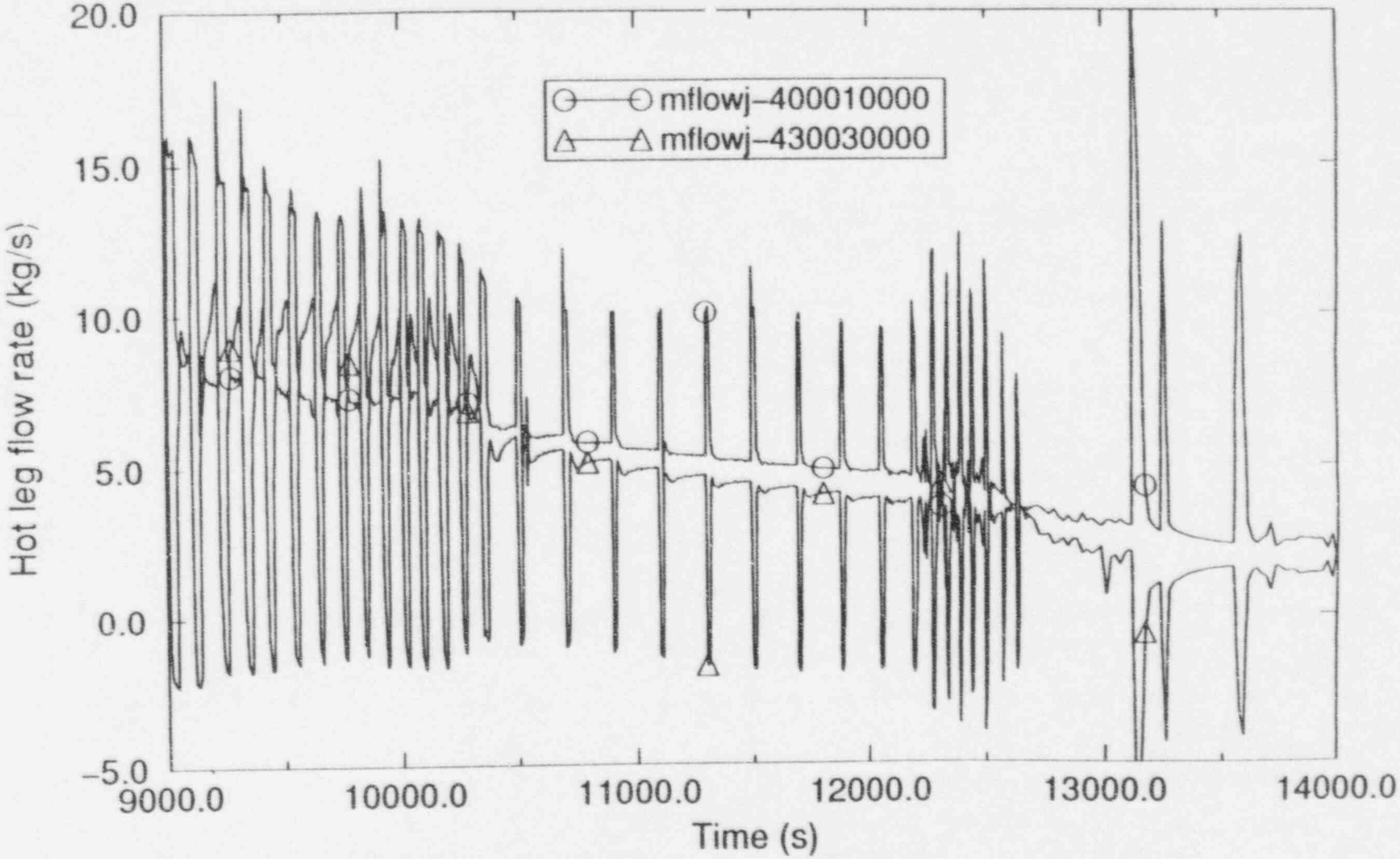
Surry Plant Results - Case 7



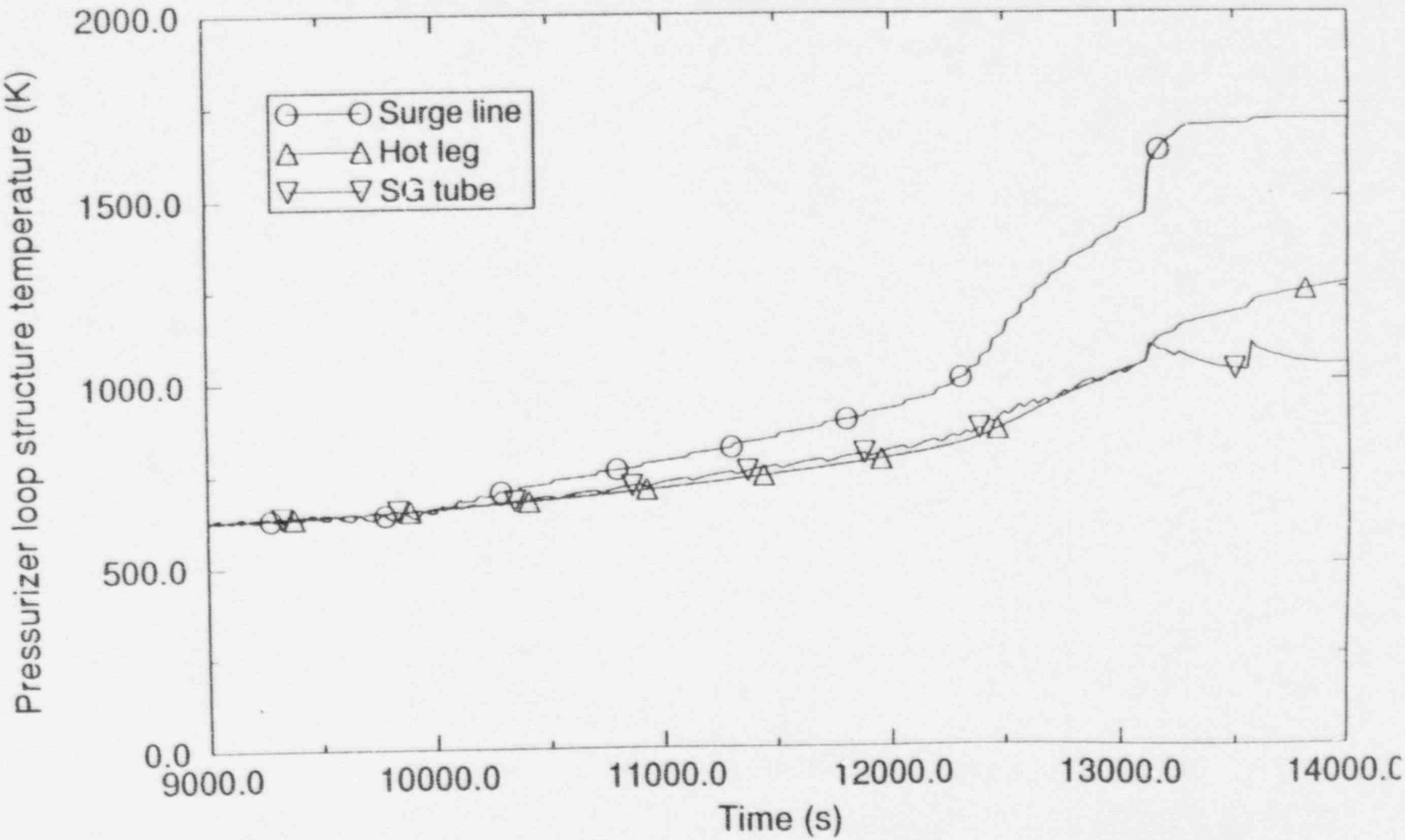
Surry Plant Results - Case 7



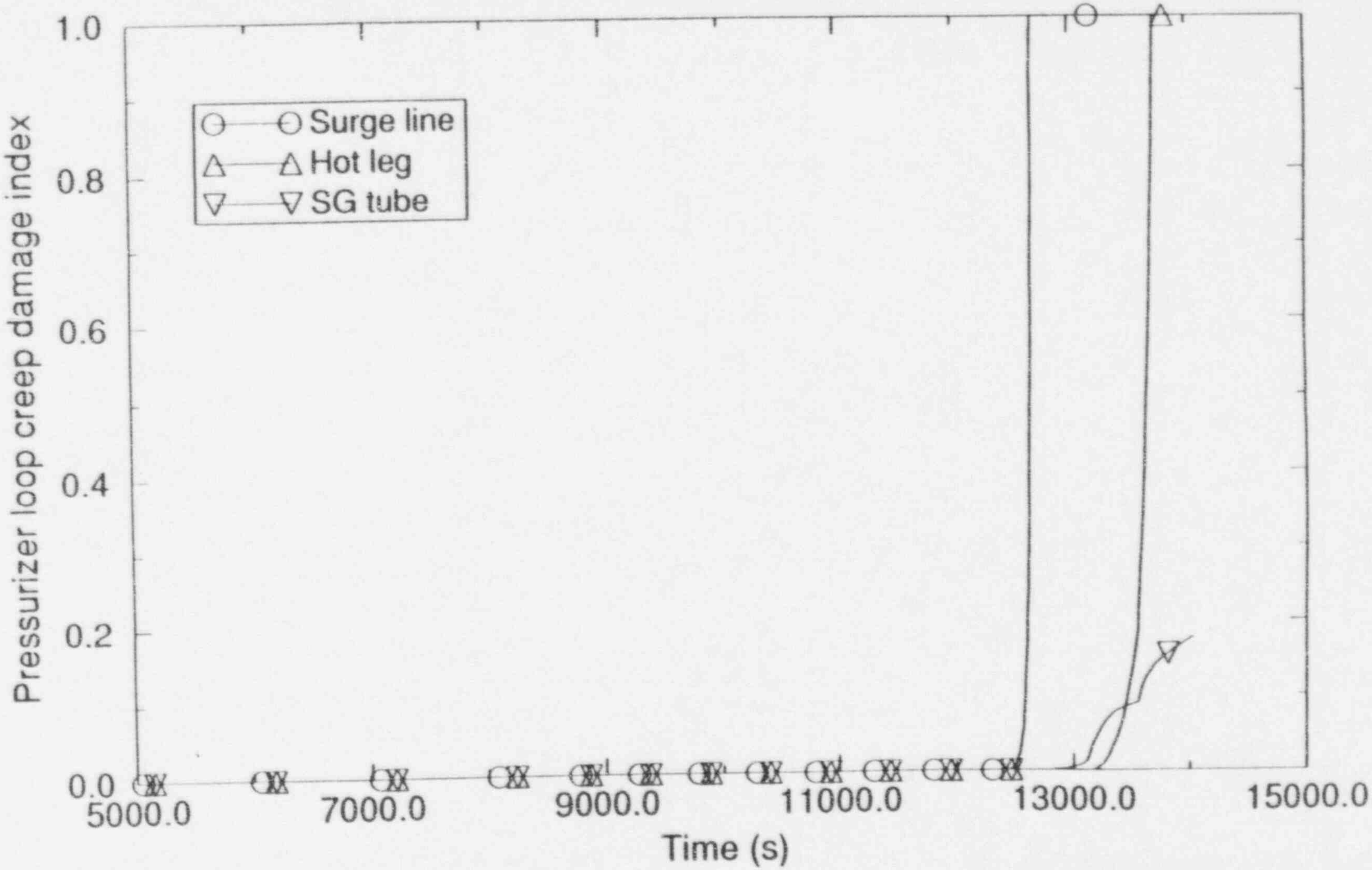
Surry Plant Results - Case 7



Surry Plant Results - Case 7



Surry Plant Results - Case 7



Conclusions

- Maximum SG tube temperatures increase with
  - a decrease in the number of tubes participating in forward (hot) flow
  - a decrease in the mixing fraction
  - an increase in the recirculation ratio
- SCDAP/RELAP5 results are consistent with the expected SG tube temperature effects for variations in hot leg countercurrent natural circulation
- Variations in hot leg countercurrent natural circulation (within the experimental range) have a negligible impact on SG tube temperatures

Conclusions

- Pressurizer surge line creep rupture was the first RCS pressure boundary failure in all Surry plant calculations considered
- The pressurizer surge line failed in the early phase of core damage (before the onset of fuel melting) in all Surry plant calculations considered
- If pressurizer surge line and hot leg failures are ignored AND a SG secondary ADV fails open, SGTR could occur ~15 to 20 min after the first RCS pressure boundary failure

USE OF SEVERE ACCIDENT THERMAL-HYDRAULIC ANALYSIS RESULTS IN RISK ESTIMATE FROM STEAM GENERATOR TUBE FAILURE

AUGUST 20, 1996

Joseph Donoghue, NRC Telephone: (301) 415-1131

ATTACHMENT-6

BACKGROUND

- SGTR risk contributions from spontaneous and induced tube failures

  - Spontaneous failures often due to unknown mechanisms
  - Induced Failures:

    Mechanical - Circumferential Crack failures
    Pressure - ATWS, Secondary Depressurization
    Thermal - Severe Accident

- Previous studies concluded that thermally induced tube failure not a significant concern during severe accidents (e.g, NUREG 1150, DCH studies)

  - Same conclusions reached for degraded and pristine tubes (NUREG/CR-4551, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," December 1990)

SEVERE ACCIDENT TUBE CHALLENGE

- Event tree developed to include thermally induced SGTR resulting from core damage events

- Considerations:
  - Events leading to high tube temperature
  - Events resulting in high or intermediate RCS pressure and dry SGs

- Frequency range of E-5 for thermal challenge from NUREG 1150 analysis for Surry and comparisons with available information in IPE Database

- Starting an order of magnitude from surrogate safety goal

- Applicability to other designs considered:
  - No PORVs in some CE plants
  - Potential for RCP seal LOCAs, loop seal clearing
  - Differences in severe accident progression

SEVERE ACCIDENT TUBE CHALLENGE

Thermal-Hydraulic Analyses:

- Use representative plants to analyze most likely thermal challenge scenario
- Surry Base Case: SBO, loss of AFW, One SG depressurized
- Arkansas Nuclear One, Unit 2
  - CE design without PORVs
- Other cases:
  - RCP seal LOCA
  - Primary-to-secondary leakage
  - PORV fails open
  - All SGs depressurized

TUBE FAILURE PROBABILITY

- Calculate probability of SG tube failure PRIOR TO surge line or hot leg for any given crack size. Result is <u>conditional tube failure probability</u>

- Steps:

  - SCDAP/RELAP5 analysis provides temperature and pressure histories

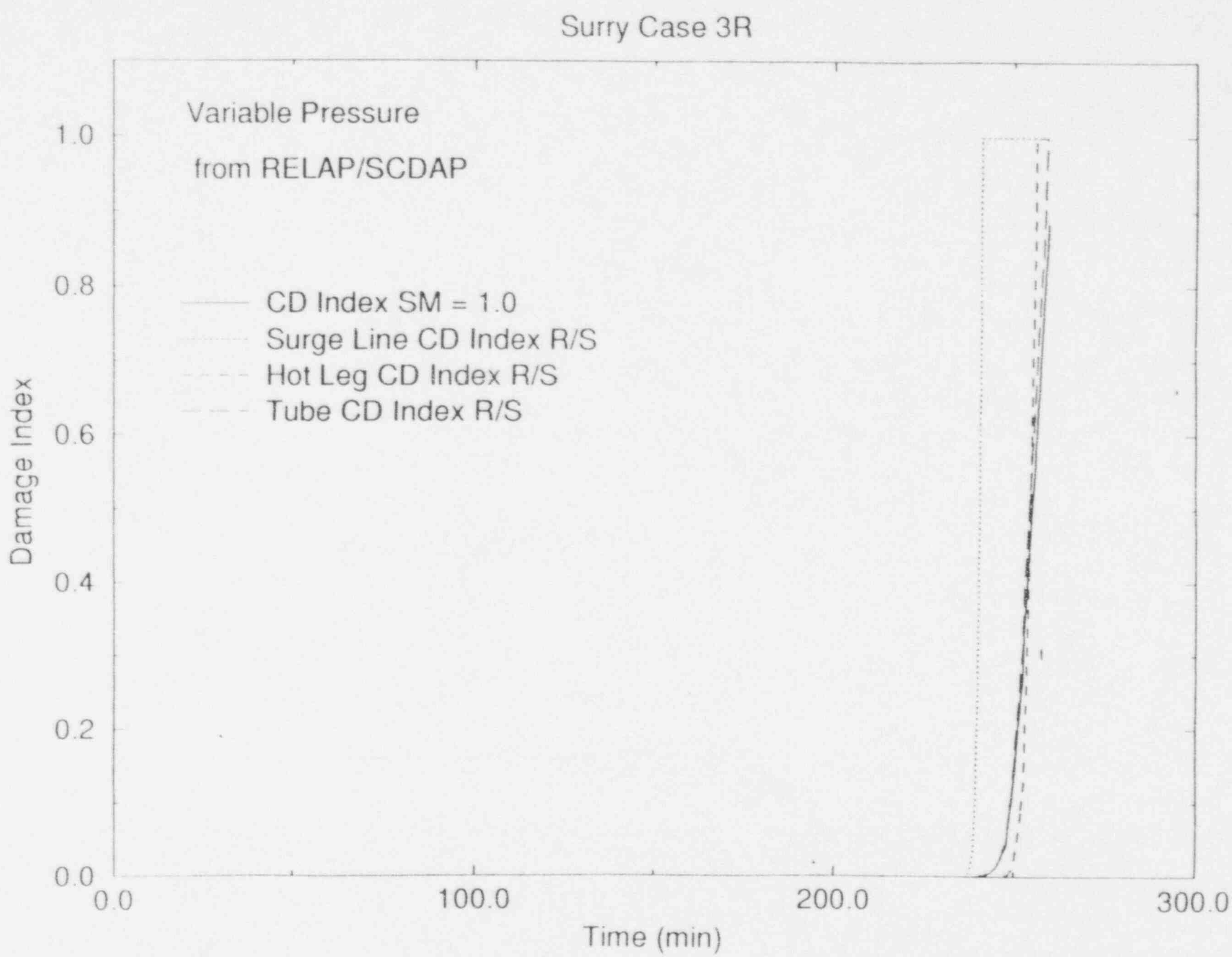
  - Calculate creep failure times for each component

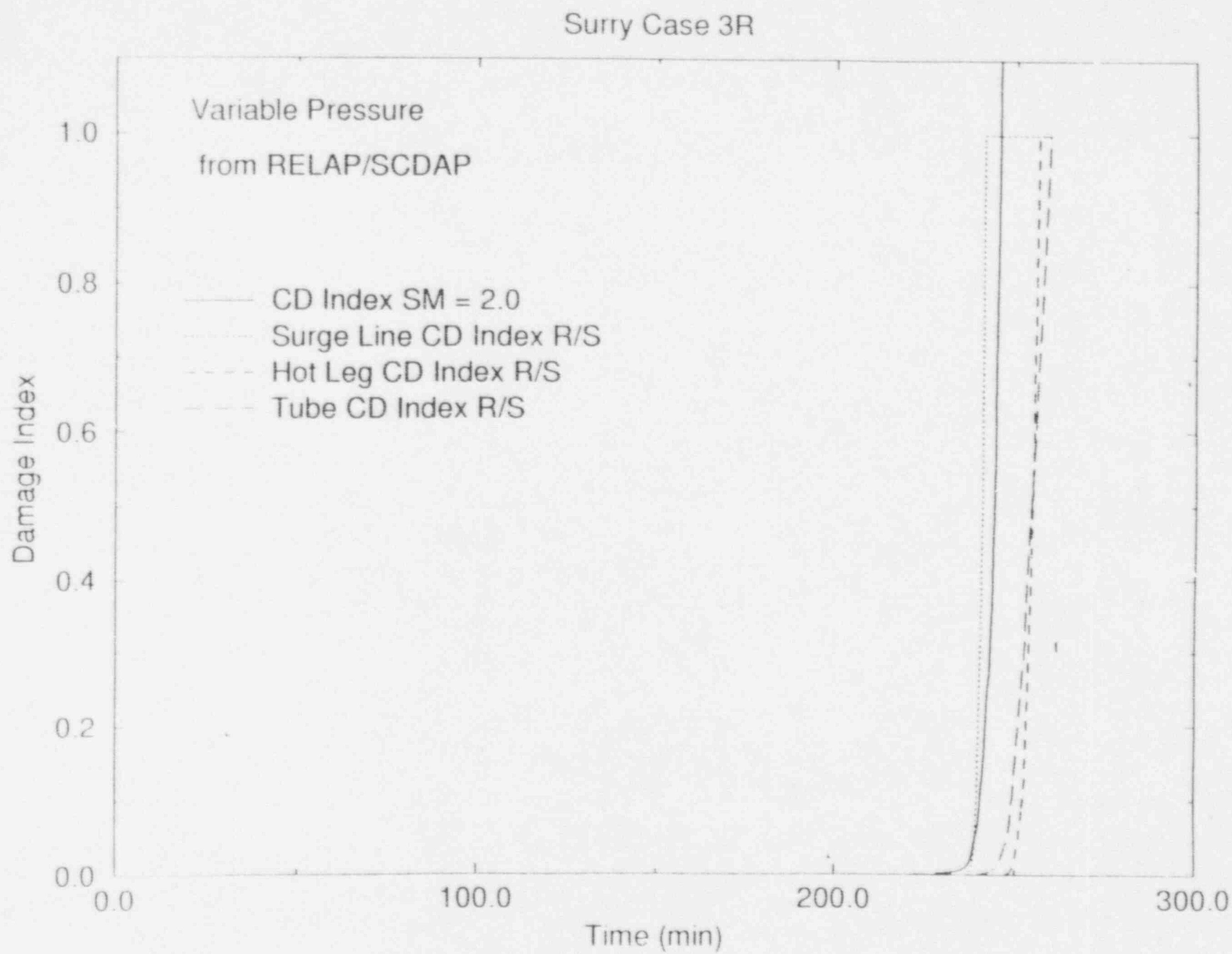
  - Include material uncertainties

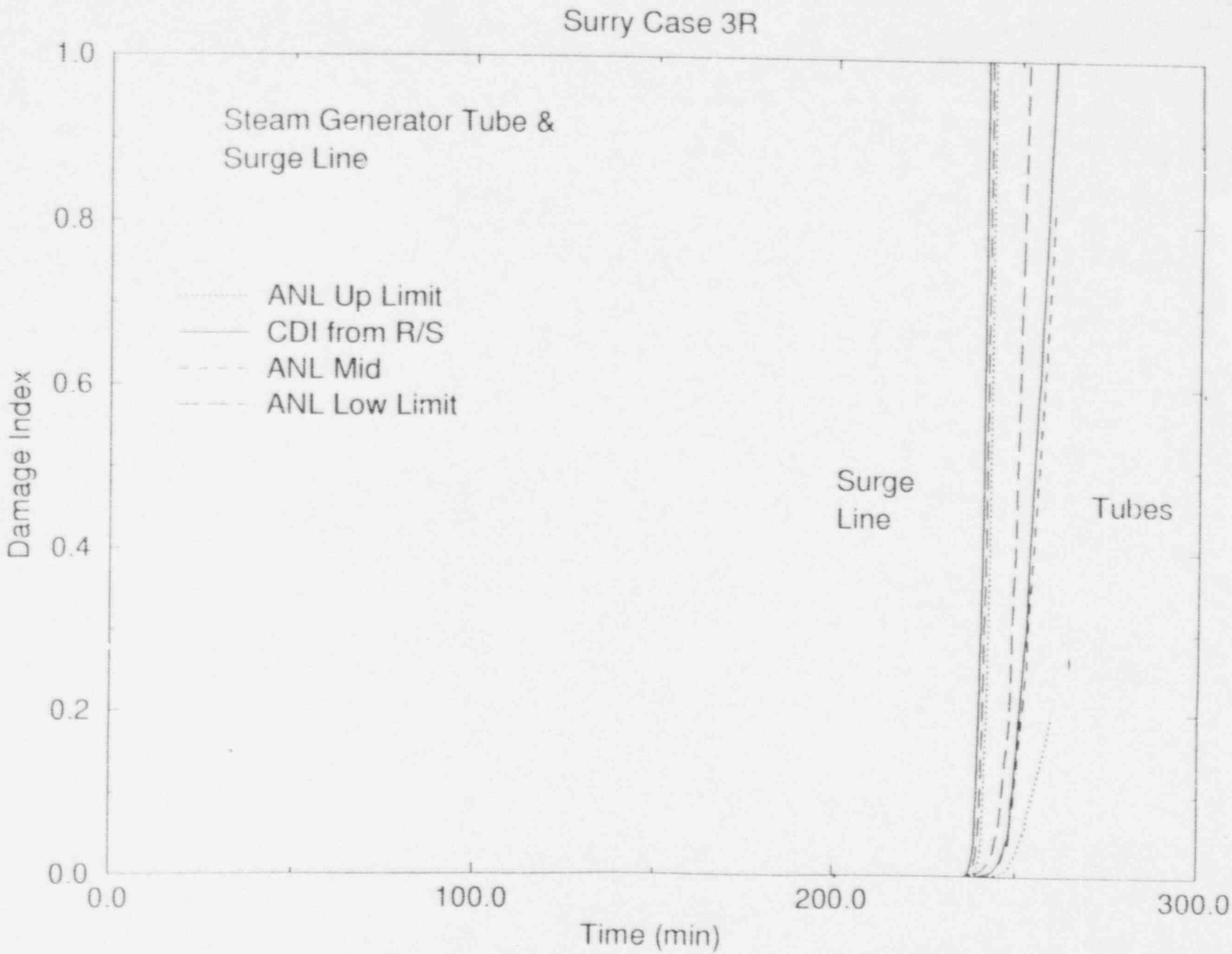
  - Include effects of tube flaws

  - Include thermal-hydraulic uncertainties:
    - Relative time to failure
    - Conditions leading to tube failure

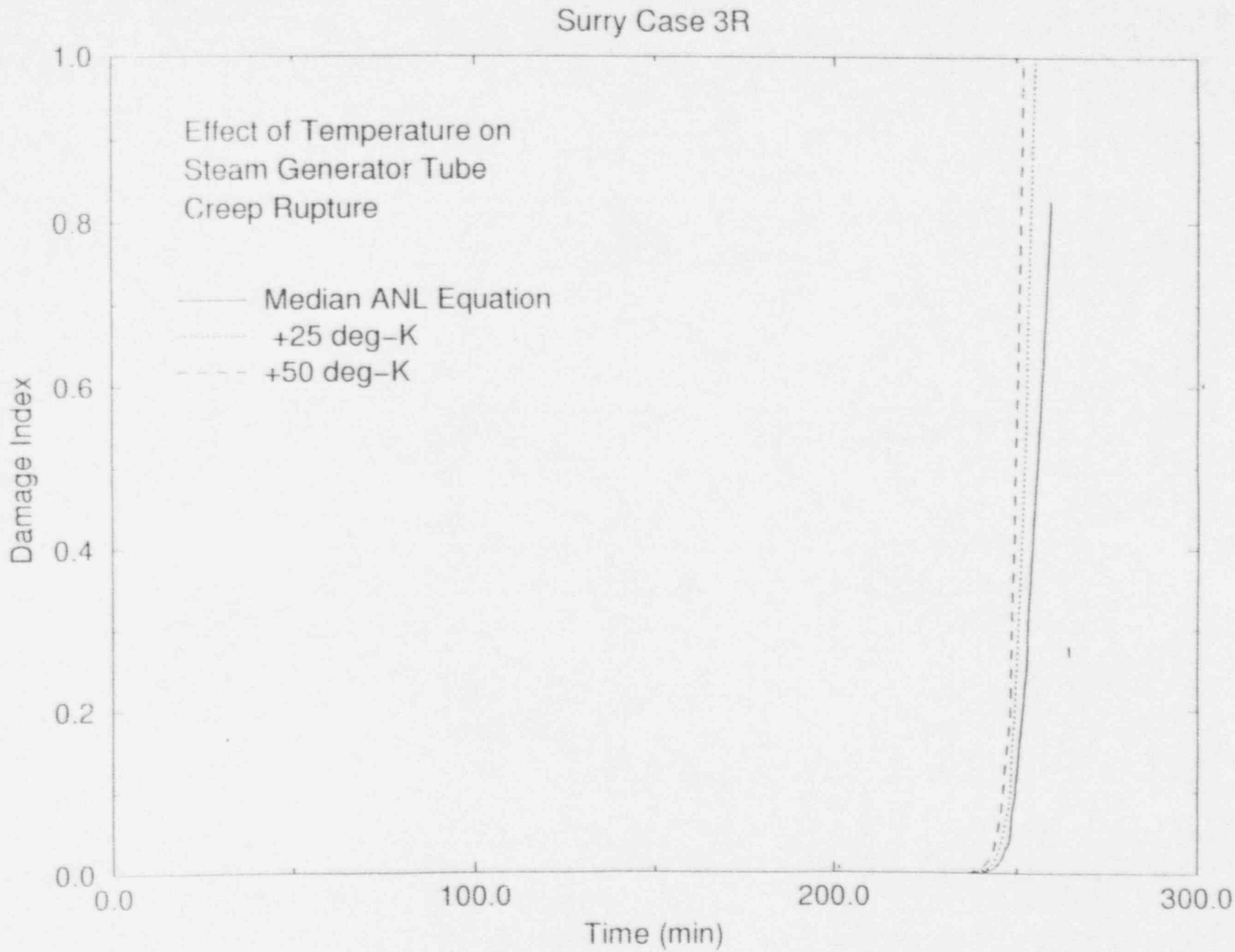
  - Other factors:
    Tube dimension uncertainties, Crack parameter variability


Surry Case 3R
Variable Pressure
from RELAP/SCDAP
CD Index SM = 1.0
Surge Line CD Index R/S
Hot Leg CD Index R/S
Tube CD Index R/S
Damage Index
0.0
0.2
0.4
0.6
0.8
1.0
0.0
100.0
200.0
300.0
Time (min)


Surry Case 3R
Variable Pressure
from RELAP/SCDAP
CD Index SM = 2.0
Surge Line CD Index R/S
Hot Leg CD Index R/S
Tube CD Index R/S
Damage Index
0.0
0.2
0.4
0.6
0.8
1.0
0.0
100.0
200.0
300.0
Time (min)



Surry Case 3R
Steam Generator Tube &
Surge Line
ANL Up Limit
CDI from R/S
ANL Mid
ANL Low Limit
Surge
Line
Tubes
Damage Index
0.0
0.2
0.4
0.6
0.8
1.0
0.0
100.0
200.0
300.0
Time (min)



Surry Case 3R
Effect of Temperature on
Steam Generator Tube
Creep Rupture
Median ANL Equation
+25 deg-K
+50 deg-K
Damage Index
0.0
0.2
0.4
0.6
0.8
1.0
0.0
100.0
200.0
300.0
Time (min)

CONTAINMENT BYPASS PROBABILITY

- Apply conditional failure probabilities for each crack size to flaw distribution covering range of crack sizes

- Flaw distributions generated for three plant categories:
  Good, Average, Poor

- Preliminary results for "Average" plant:

  - Assuming nominal tube temperatures,
    Tube failure probability about 6%, Bypass $<$E-6

  - Assuming higher tube temperatures,
    Tube failure probability about 20%, Bypass $\approx$E-6

- "Poor" plant distribution being revised, but initial result yielded
  Tube failure probability of 100%, Bypass $\approx$E-5

RISK ESTIMATE KEY ISSUES

- Event Tree Quantification:
  - Event frequency in E-5 range is consistent with range in IPE survey
- Thermal-Hydraulic Modeling:
  - Creep failure prediction dependent on understanding temperature and pressure time histories
- Representative Flaw Distribution:
  - Effort ongoing to revise poor plant distribution
  - Also working to understand uncertainties
- Tube Performance Model:
  - Based on high temperature tube tests
  - Includes material uncertainties
  - Working to understand flaw characterization uncertainties
- RCPB Weak Points: Only qualitative treatment for potential of other component failures (other than hot leg, surge line, tubes)

SG SEVERE ACCIDENT RISK ESTIMATE