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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 30 1981

Docket Nos. 50-269, 270, 289, 302, 312, 313 and 346

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LICENSEES: BABCOCK AND WILCOX REACTOR OWNER GROUP - DUKE POWER COMPANY METROPOLITAN EDISON CO., SACRAMENTO MUNICIPAL UTILITY DISTRICT, ARKANSAS POWER AND LIGHT CO., TOLEDO EDISON CO., and FLORIDA POWER CORPORATION

SUBJECT: SUMMARY OF MEETING WITH THE B&W OWNERS GROUP CONCERNING THE ABNORMAL TRANSIENT OPERATING GUIDELINES (ATOG) PROGRAM AND TMI ACTION ITEM II.K.3.30 SMALL BREAK LOSS-OF-COOLANT ACCIDENT MODELS (DECEMBER 16, 1980)

The purpose of the meeting was to continue discussions on the ATOG program for B&W reactors and to provide the Owners Group with a list of concerns in responding to the TMI Action Item II.K.3.30. A list of attendees is provided in Enclosure 1. The meeting agenda is provided in Enclosure 2.

Summary on ATOG Program

The staff stressed the point that the ANO-1 submittal needs to be suitable for a generic review of the ATOG program. In addition the TMI Action Plan scheduled is not changeable and the January 1, 1981 date for the formal submittal is to be met. The PTRB is currently drafting a set of guidelines to address the procedures and methodology for the vendor programs in response to TMI Action Plan Item I.C.1 and would like to incorporate the Owners Group ATOG program in the draft. A March 1 date is tentatively planned for the draft release.

The Owners Group has indicated that they would use 10CFR 50.59 to implement the ATOG program and are not particularly concerned as to what we actually review. Questions generated on the ANO-1 submittal should be addressed to ANO-1 or to all the licensees.

The ANO-1 draft ATOG program will be formally submitted, with a schedule for completion of the final report, on January 1, 1981 as required.* It will be noted that the draft is very much preliminary and that there is no guarantee the substantial changes will not be made in the final report. In addition it will be noted that Inadequate Core Cooling will be added to the ATOG program.

A formal presentation on the ATOG program was made by J. J. Kelly of B&W. The slides are provided in Enclosure 3.

* Each licensee will reference the ANO-1 submittal and supply a schedule.

Summary for II.K.3.30

The staff identified the following nine areas of concern which should be addressed by the Owners Group in future submittals:

1. Need to verify the current non-condensible model and the conservatism of the condensation heat transfer rate in the steam generator.

2. Need to verify the non-equilibrium model and to justify that the amount of ECCS water injected is conservative.

3. Need to discuss the pressurizer model and the effects of a non-equilibrium model.

4. Need to address the formation of a steam bubble in the hot leg "candy cane". (Is it a real or calculated phenomenon?) Experimental verification believed necessary.

5. The staff indicated that a mechanistic model of the steam generator heat transfer should be developed. A best estimate or verified conservative model would be acceptable.

6. As part of the additional systems verification needed, the following Semiscale and LOFT tests should be considered: Semiscale S-07-10D, LOFT L3-1, L3-5 and L3-6.

7. The overall thermal-hydraulic behavior of the core during uncovery should be verified against applicable experimental data, particularly the recent ORNL data.

8. The influence of metal heat on the system pressure response, particularly on the time of ECCS injection was identified as an area of concern and should be shown to be properly considered in the analysis models.

9. The break flow model needs to be confirmed. The use of combined models with various discharge coefficients applied to them needs to be compared to a best estimate model to demonstrate conservatisms.

A discussion of the impact of reactor coolant pump operations was held. The staff identified their present thinking regarding acceptance criteria for allowing manual pump trip.

1. If the operator is allowed to trip the pumps manually, the criteria of 10CFR 50.46 must be met using an acceptable model (Appendix K model with a best estimate verification analysis of LOFT L3-6) assuming RCP trip at 10 minutes or at the specified criteria, whichever is longer.

2. If pump trip is required in less than 10 minutes, using the Appendix K analysis model, then a best estimate analysis with pump trip assumed at worst time into the event must result in acceptable consequences. A minimum acceptable time for operator action will have to be identified and justified.

JAN 30 1981

A schedule and work scope for the II.K.3.30 item is to be submitted by January 30, 1981.

Edward D. Throm

Edward D. Throm
Reactor Systems Branch
Division of Systems Integration

Enclosures:
1. Attendance List
2. Meeting Agenda
3. ATOG Slides

ENCLOSURE 1

ATTENDEES:

NRC

E. D. THROM
J. GUTTMANN
F. ODAR, RES
D. BECKMAN
B.W. SHERON
G. S. VISSING
G. MEZETIS
R. PITMAN
J. CLIFFORD
R. URBAN
M. GOODMAN

AP & L

D. WILLIAMS
M. SMITH

SMUD

D. WHITNEY
R. A. DIETERICH

CPCO

W. J. HALL
L.S. GIBSON

B & W

J. J. KELLY
R.C. JONES
R. A. TURNER
J. J. CUDLIN
H. BAILEY
N. K. SAVANI

GPU

T.C. BRUUGHTON
C.W. SMYTH

DUKE POWER

R. L. GILL

TVA

F. A. KOONTZ, JR

FLORIDA POWER CO.

H. M. PERRY
A. F. FEGENDRE, JR.

WPPSS

A. HOSLER

TOLEDO EDISON

T. MYERS
S. JAIN

ENCLOSURE 2

B&W OWNERS GROUP
MEETING WITH NRC

TMI-2 & ATOG
SUBCOMMITTEES

December 16, 1980

Tuesday, December 16, 1980 (P 110 — 8:30)

TMI Action Plant Item I.C.1 (Short Term Accident & Procedures Review)

1. Inadequate Core Cooling Guidelines
2. Transients and Accidents

TMI Action Plan Item II.K.3.30 (SB LOCA Methods)

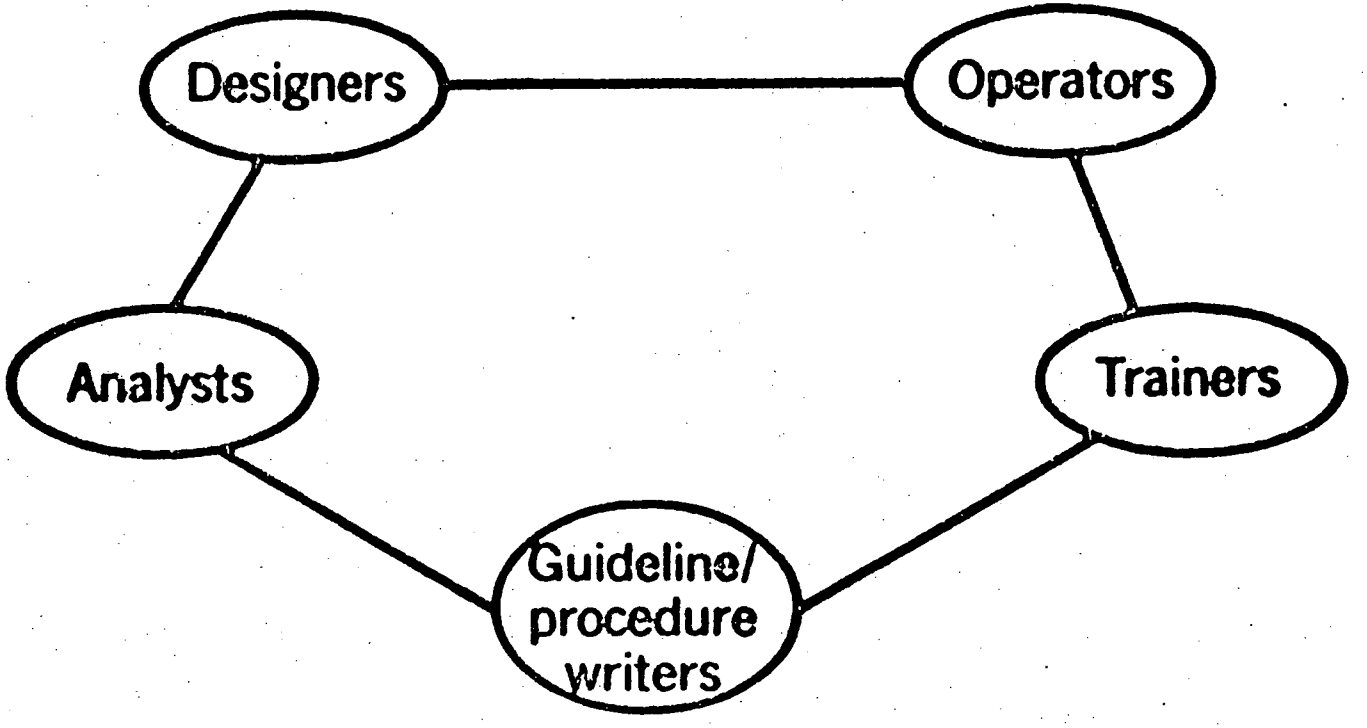
1. Introduction
2. Discussion of work to address NRC expectations in the following areas:
   - Areas of the model which will be upgraded
   - Areas of the model which will be further justified
   - Test data which will be used
3. Owners Caucus
4. Scope and schedule for completion of the above.

ATOG Objective

Simplify operator problem of identifying and treating abnormal transients

ENCLOSURE 3

Closing the loop



Methodology

Event trees

Analysis

Design basis/expected plant response

Simulation

Operator feedback

Event Trees

Purpose — Systematically determine various plant conditions which can evolve following a postulated initiating event

Assumptions
- Initial conditions
- Equipment failures
- Operator action

Transients Selected for Guideline Preparation

- Increase in heat removal by secondary system
  - Small steam leaks
  - Excessive feedwater flow
- Decrease in heat removal by secondary system
  - Loss of feedwater
  - Loss of station power
- Decrease in reactor coolant inventory
  - Steam generator tube rupture
  - Inadequate core cooling
  - Loss of coolant

Event tree bounding assumptions

- Initial conditions
  - —Power range.
  - —No equipment tagged out.
  - —Equilibrium core.
- Equipment failures
  - —Consequential failures.
  - —Non-safety system components only.
  - —Active failures only.
- Operator action
  - —Will be given the opportunity to act when required by procedure.
  - —His actions can be correct, incorrect or failure to act.
  - —Errors will not be random.
  - —Actions, including mistakes, will be complete within a system.
  - —No artificial time constraints.
- Prior plant transient experience

Analysis

Purpose: Realistically portray expected plant response

Analyze: Design success path
All single failure paths

Discuss subsequent failures:

Verify LOCA paths covered in small break guidelines

Design Basis/Expected Plant Response

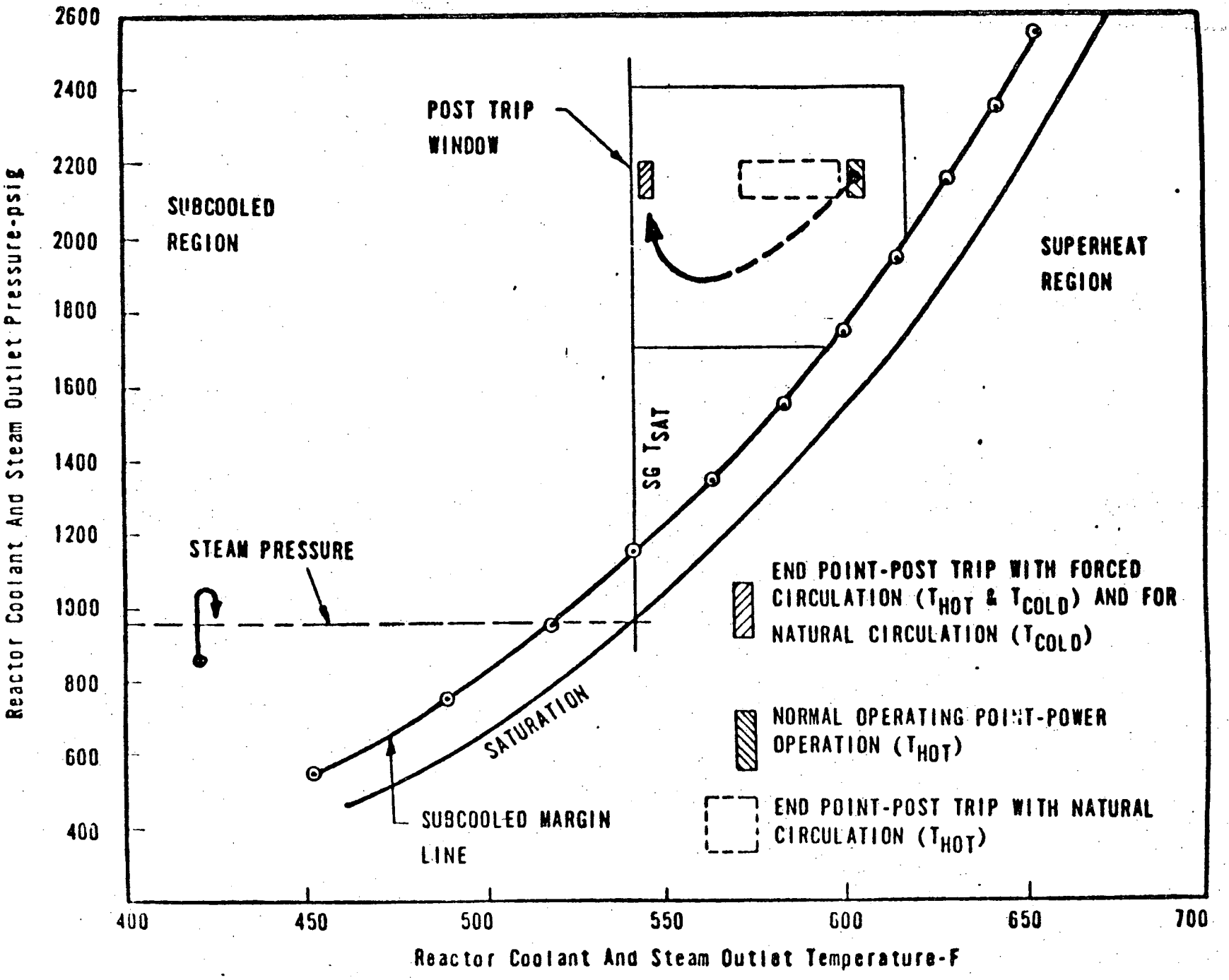
- Communication between designer and operator
- Supports operator action portion of guidelines
- Written for operator understanding

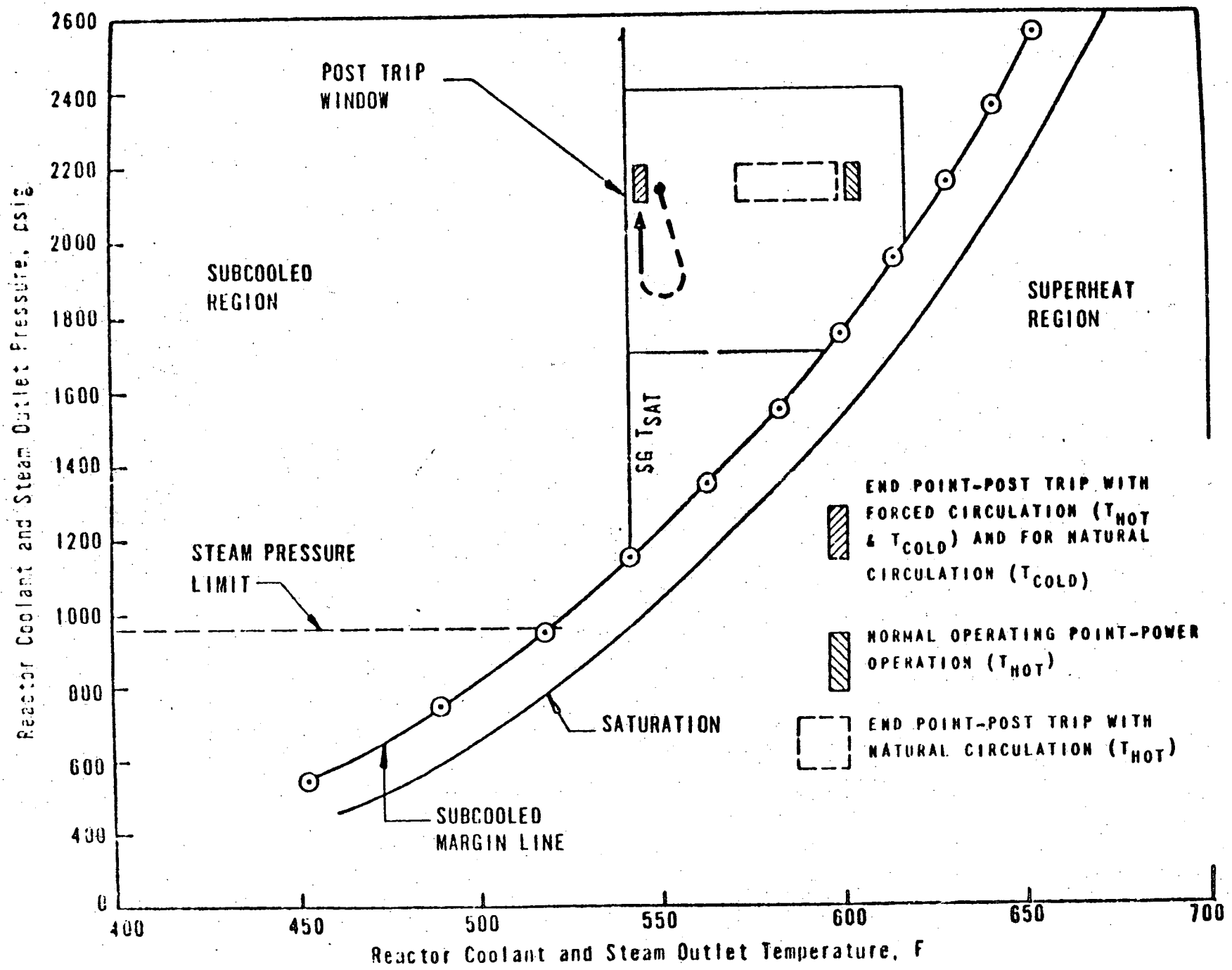
Training Simulator

- Test various methods of approach to guidelines
- Verify final product
- Train operator

Operator Feedback

- Detailed review of event trees
- Input to guideline format
- Plant walk through
- Training


2600
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
Reactor Coolant And Steam Outlet Pressure-psig
POST TRIP WINDOW
SUBCOOLED REGION
SUPERHEAT REGION
SG TSAT
STEAM PRESSURE
SATURATION
SUBCOOLED MARGIN LINE
END POINT-POST TRIP WITH FORCED CIRCULATION (THOT & TCOLD) AND FOR NATURAL CIRCULATION (TCOLD)
NORMAL OPERATING POINT-POWER OPERATION (THOT)
END POINT-POST TRIP WITH NATURAL CIRCULATION (THOT)
400
450
500
550
600
650
700
Reactor Coolant And Steam Outlet Temperature-F


2600
2400
2200
2000
1800
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1400
1200
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400
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Reactor Coolant and Steam Outlet Pressure, psia
POST TRIP WINDOW
SUBCOOLED REGION
SUPERHEAT REGION
SG TSAT
STEAM PRESSURE LIMIT
SATURATION
SUBCOOLED MARGIN LINE
END POINT-POST TRIP WITH FORCED CIRCULATION (THOT & TCOLD) AND FOR NATURAL CIRCULATION (TCOLD)
NORMAL OPERATING POINT-POWER OPERATION (THOT)
END POINT-POST TRIP WITH NATURAL CIRCULATION (THOT)
400
450
500
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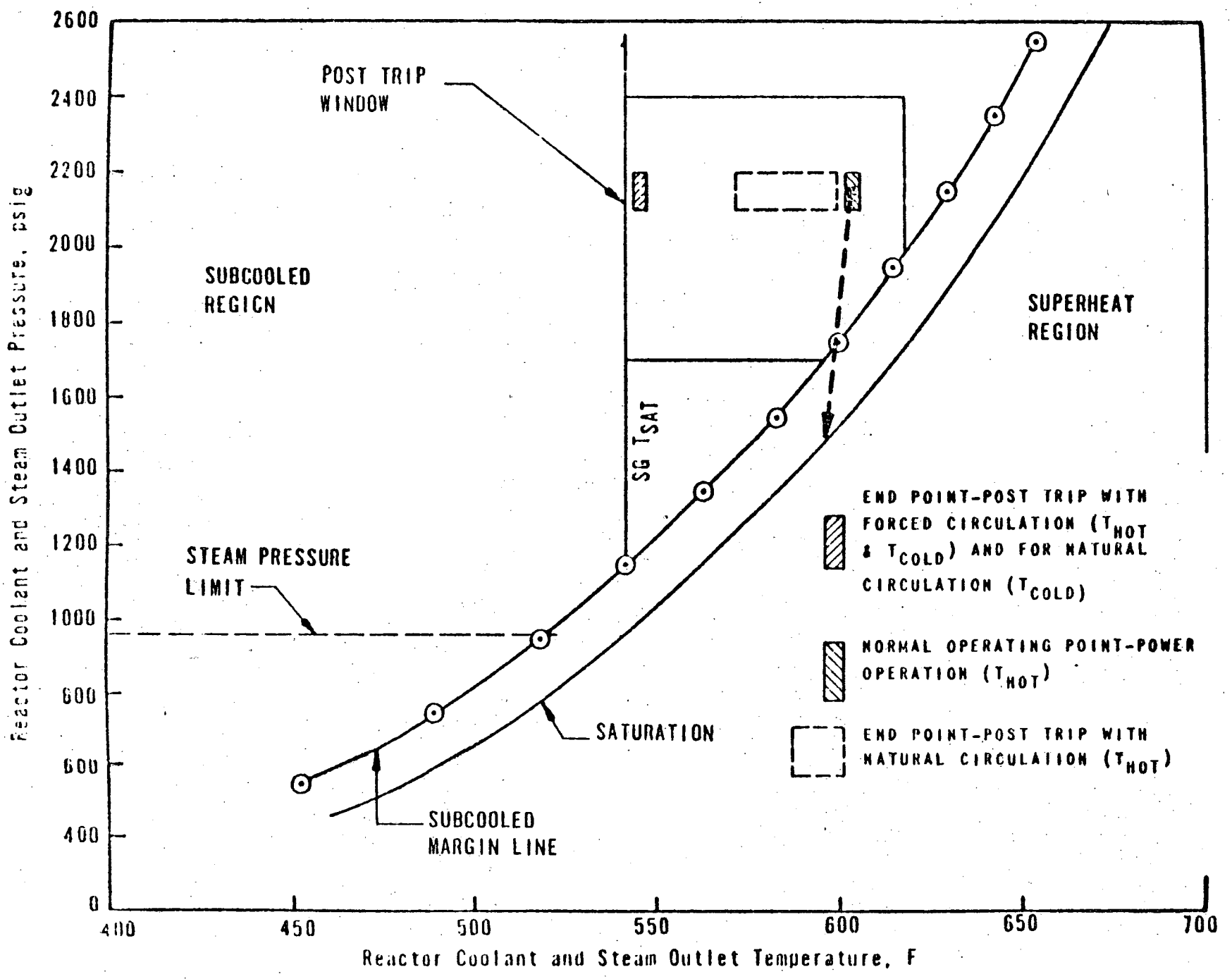
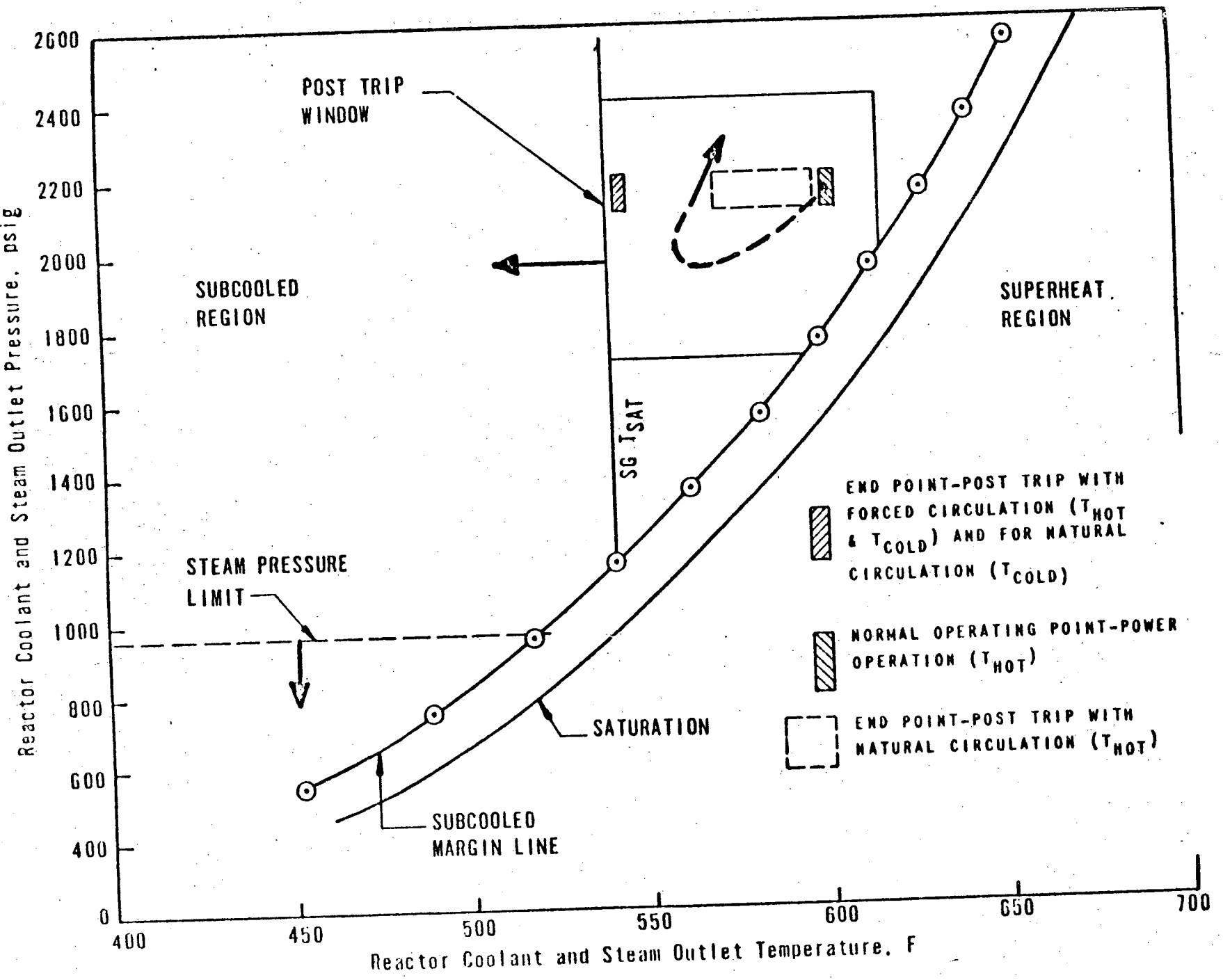
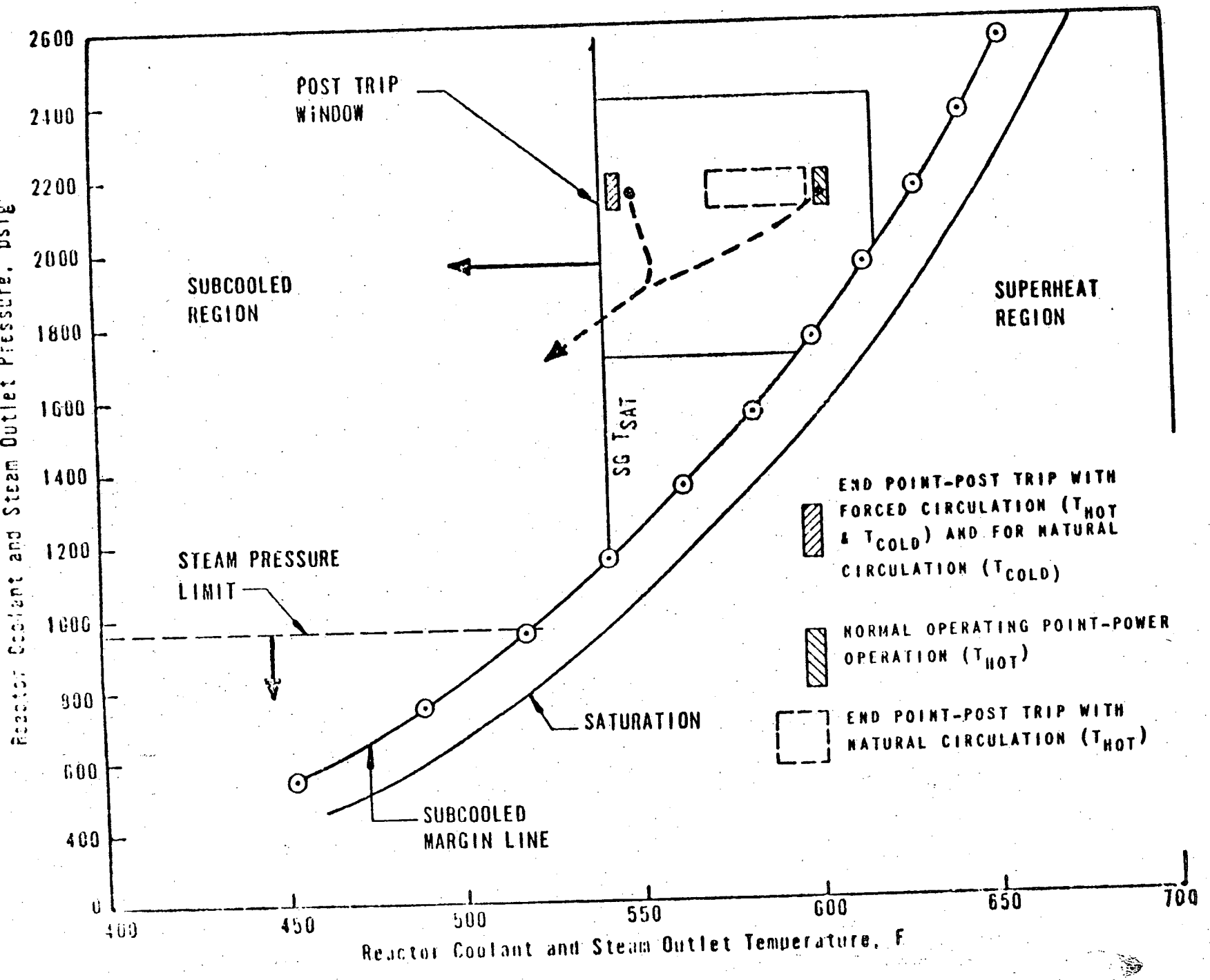
600
650
700
Reactor Coolant and Steam Outlet Temperature, F



INADEQUATE SUBCOOLING MARGIN

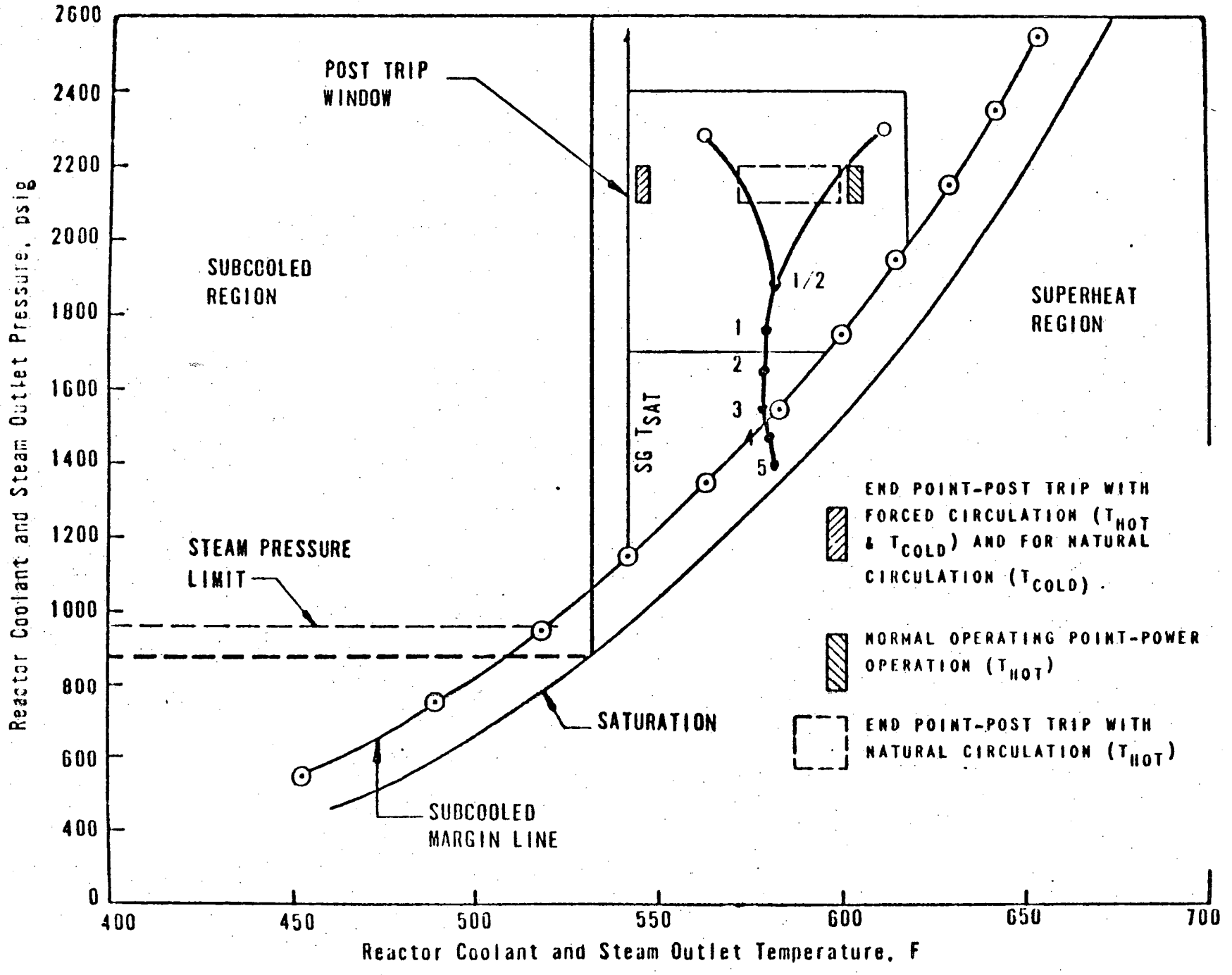


LOSS OF PRIMARY TO SECONDARY HEAT TRANSFER



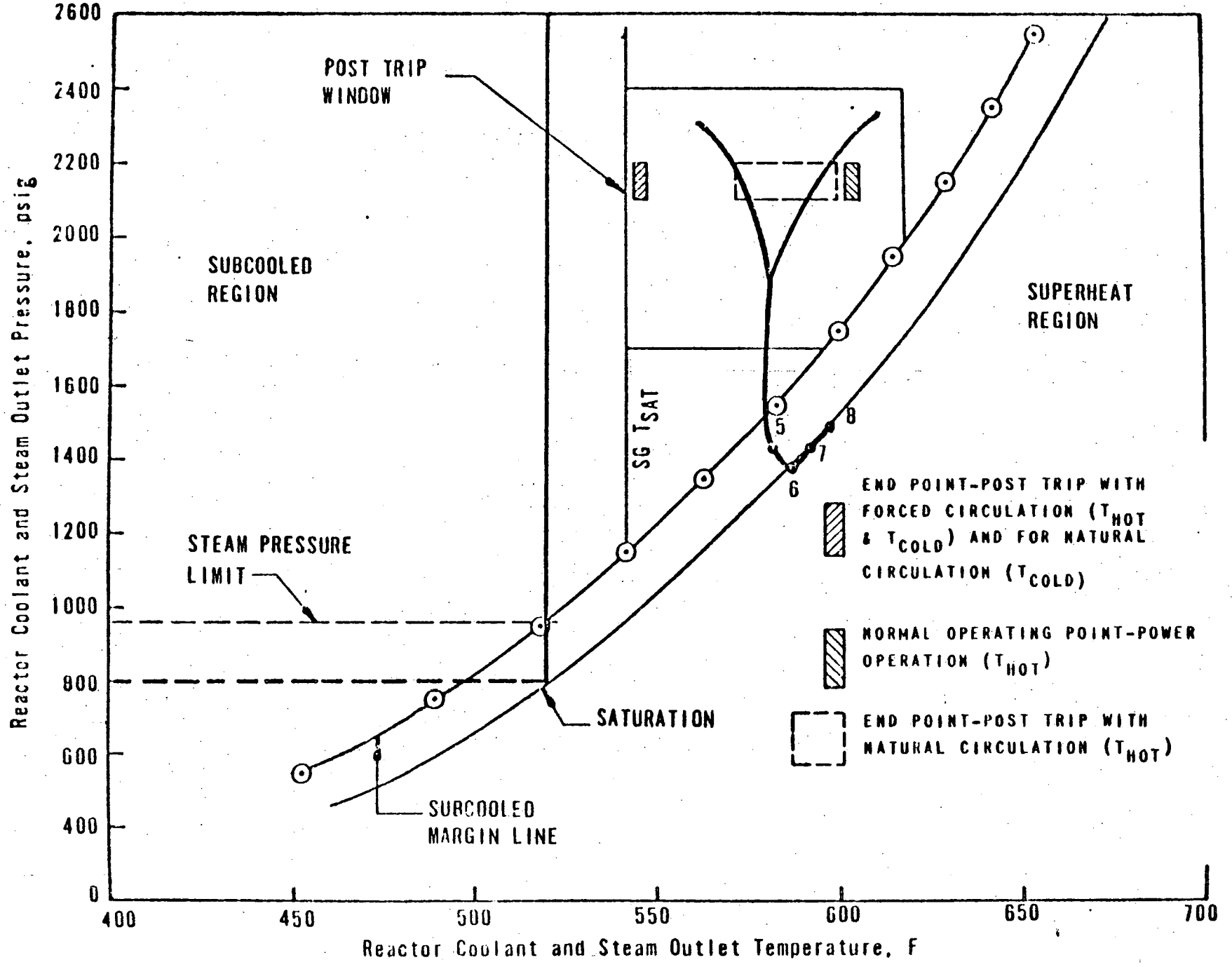
EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER

0 TO 5 MINUTES



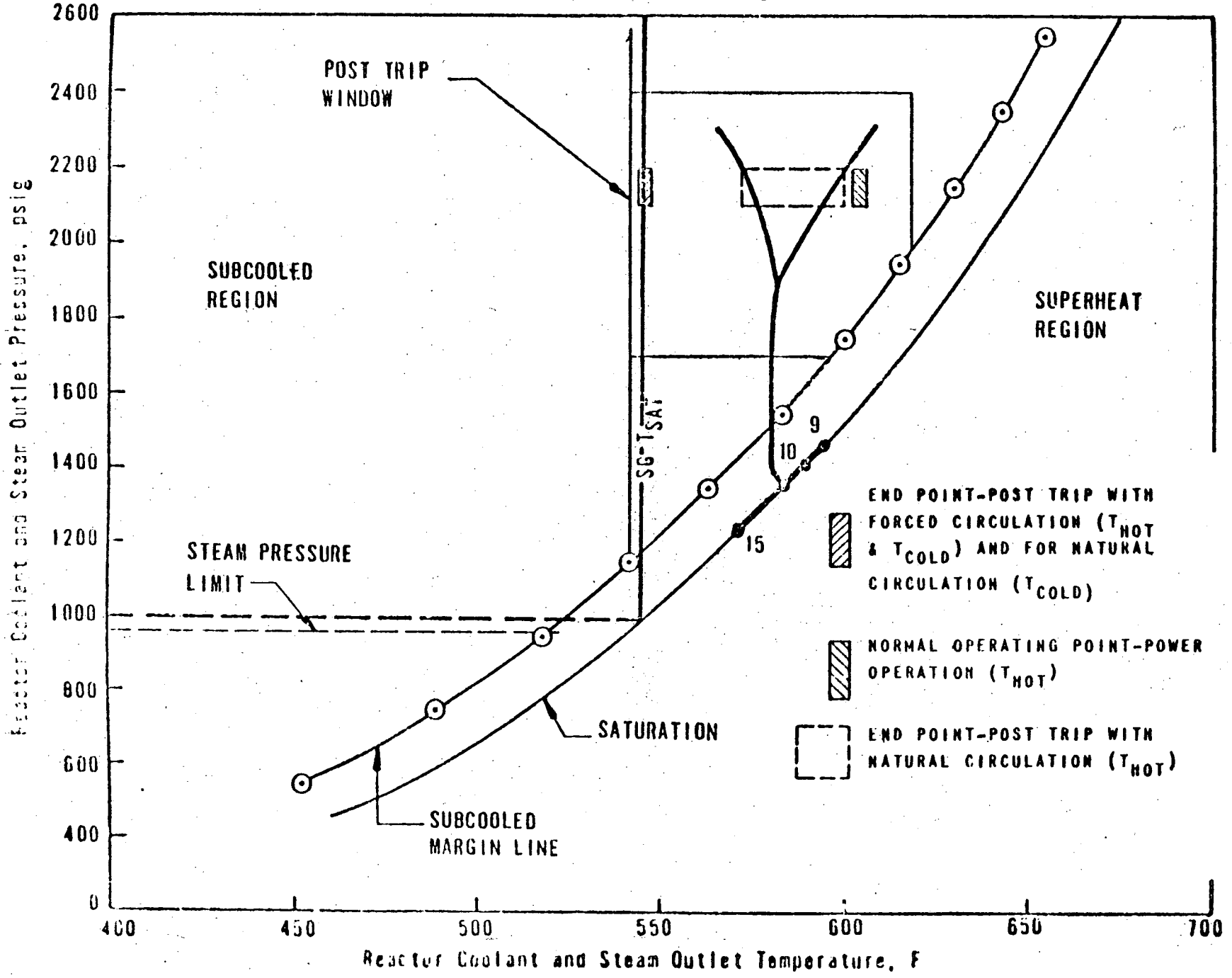
TMI-2 ACCIDENT

5 TO 8 MINUTES



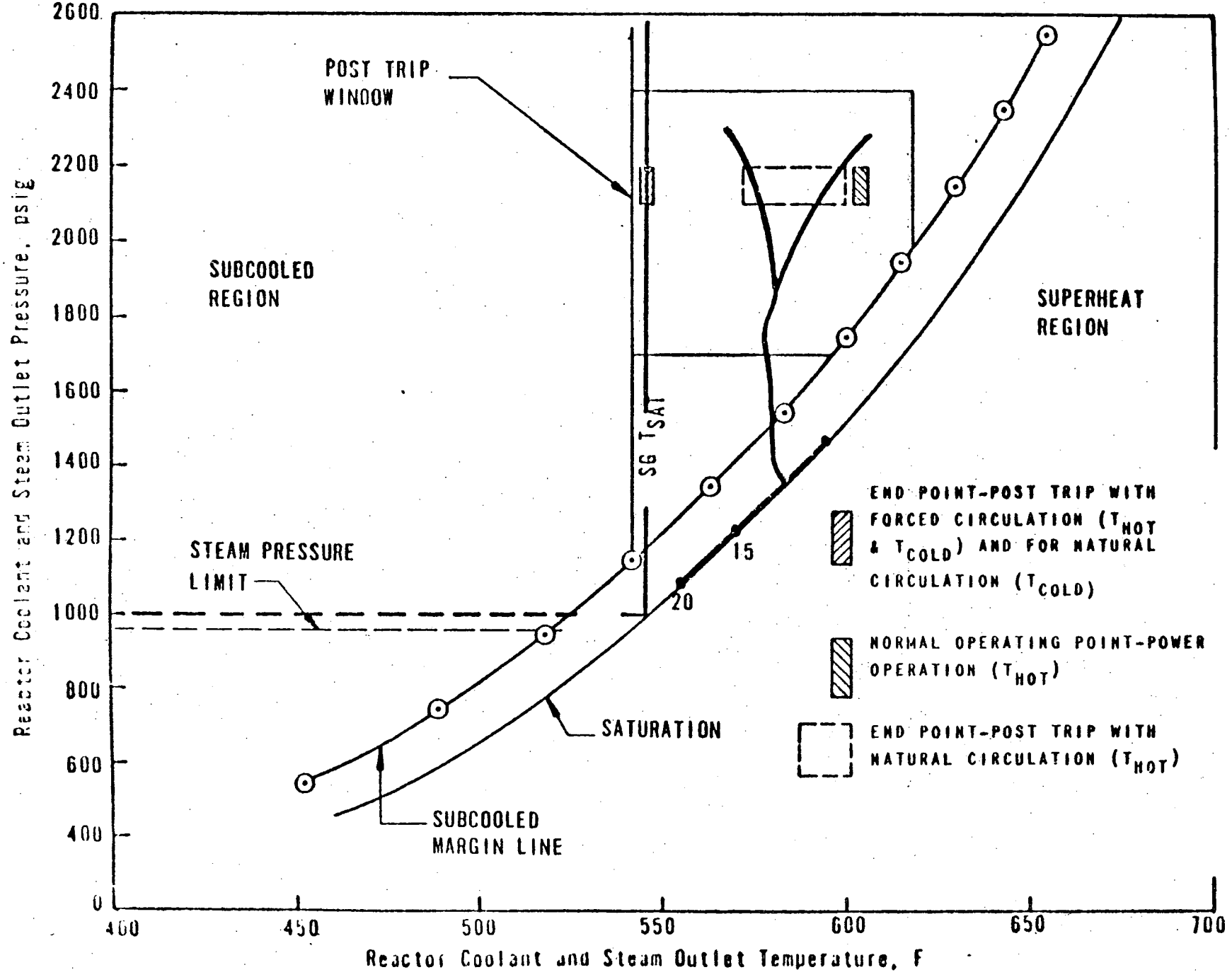
TMI-2 ACCIDENT

8 TO 15 MINUTES



TMI-2 ACCIDENT

15 TO 20 MINUTES



TMI-2 ACCIDENT

Part I. Organization

SECTION I. Immediate actions

SECTION II. Vital system status verification

SECTION III.

A. Treatment of lack of adequate subcooling margin

B. Treatment of lack of primary to secondary heat transfer

C. Treatment of too much primary to secondary heat transfer

D. Follow up actions for OTSG Tube rupture

COOLDOWN PROCEDURES

- Large LOCA
- Normal
- Saturated RCS
- HPI cooling
- Solid water cooldown

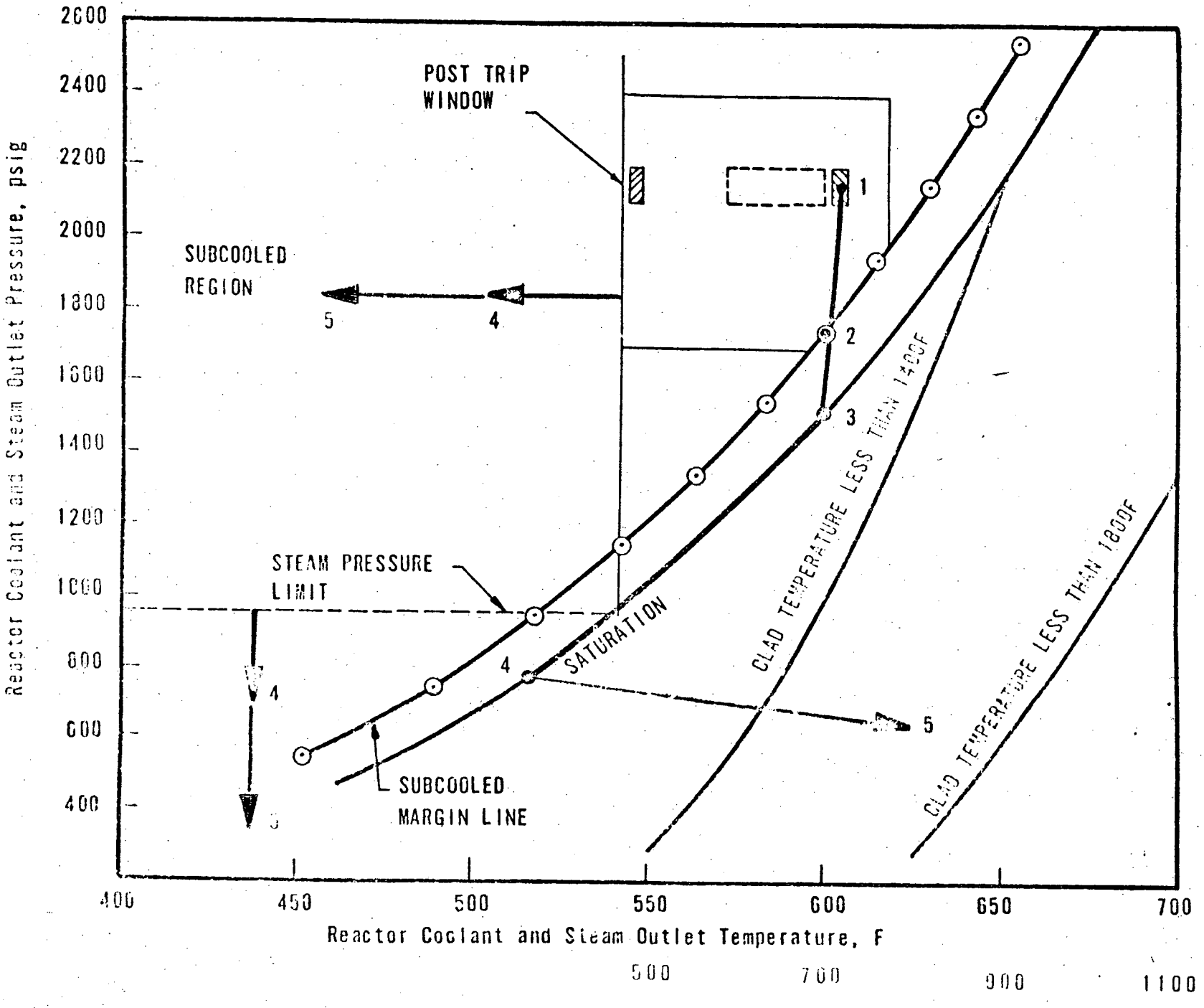
Part II. Organization

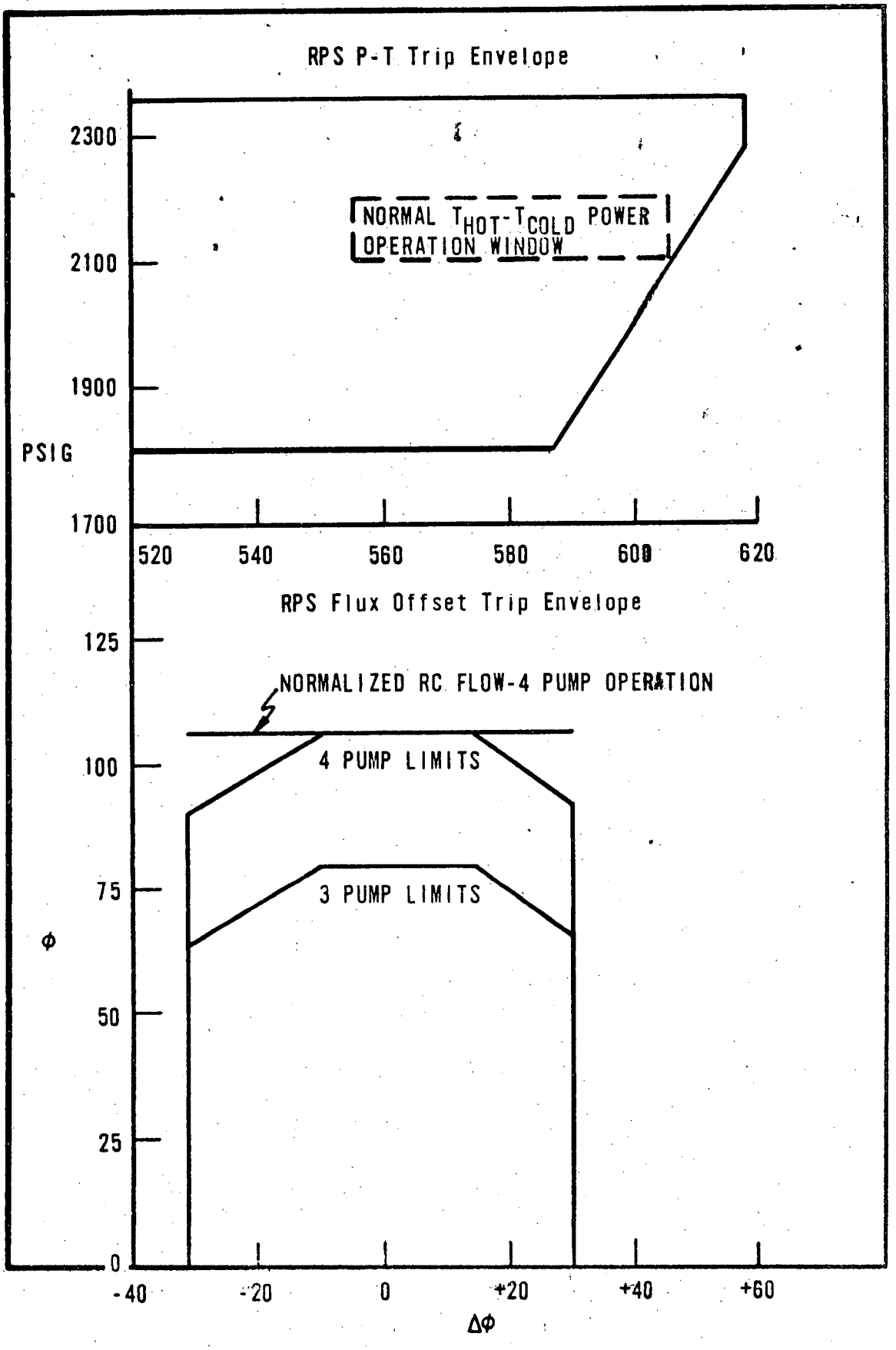
VOLUME 1: Fundamentals of reactor control for abnormal transients

A. Heat transfer
B. Use of P-T diagram
C. Abnormal transient diagnosis and mitigation
D. Backup cooling methods
E. Best methods of equipment operation
F. Stability determination

VOLUME 2: Appendices - Selected transients

A. Excessive feedwater
B. Loss of feedwater
C. Steam generator tube rupture
D. Loss of off-site power
E. Small steam line break
F. LOCA


POST TRIP
WINDOW
SUBCOOLED
REGION
STEAM PRESSURE
LIMIT
SATURATION
SUBCOOLED
MARGIN LINE
CLAD TEMPERATURE LESS THAN 1400F
CLAD TEMPERATURE LESS THAN 1800F
1
2
3
4
5
Reactor Coolant and Steam Outlet Pressure, psig
2600
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
Reactor Coolant and Steam Outlet Temperature, F
400
450
500
550
600
650
700
500
700
900
1100


RPS P-T Trip Envelope
2300
2100
1900
PSIG
1700
NORMAL T_{HOT}-T_{COLD} POWER OPERATION WINDOW
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540
560
580
600
620
RPS Flux Offset Trip Envelope
125
100
75
Φ
50
25
0
NORMALIZED RC FLOW-4 PUMP OPERATION
4 PUMP LIMITS
3 PUMP LIMITS
-40
-20
0
+20
+40
+60
ΔΦ

ATOG Full Scale Display