February 10, 1988

Docket Nos. 50-266
and 50-301

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LICENSEE: WISCONSIN ELECTRIC POWER COMPANY

FACILITY: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

SUBJECT: SUMMARY OF FEBRUARY 4, 1988 MEETING

At the request of the licensee, a meeting was held on February 4, 1988 in the Rockville, Maryland offices of the NRC to discuss the licensee's accident reanalysis to support increased allowable core peaking factors at Point Beach 1 and 2. A list of meeting attendees is included in Enclosure 1. The slides used during the presentation are included in Enclosure 2. These slides discuss the goals of the reanalysis effort, the strategy involved in achieving these goals and the potential Final Safety Analysis Report/Technical Specification changes resulting from the reanalysis.

Original Signed By:

David H. Wagner, Project Manager
Project Directorate III-3
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Enclosures:
As stated

cc: See next page

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ENCLOSURE 1

ACCIDENT REANALYSIS TO SUPPORT INCREASED ALLOWABLE CORE PEAKING FACTORS AT POINT BEACH 1 AND 2

LIST OF ATTENDEES

NAME	AFFILIATION
David Wagner	NRC
Summer Sun	NRC
Lambros Lois	NRC
Y. Gene Hsii	NRC
Wayne Hodges	NRC
Randall Fieldhack	Wisconsin Electric (WE)
Harve Hanneman	WE
Bob LaMuro	Westinghouse (W)
Janelle Ivey	W
Beth Hall	W

ENCLOSURE 2

NRC/WE/WESTINGHOUSE MEETING

ON

ACCIDENT REANALYSIS

TO SUPPORT

INCREASED ALLOWABLE CORE PEAKING FACTORS

AT

POINT BEACH NUCLEAR PLANT

UNITS 1 AND 2

February 4, 1988

Harv Hanneman (WE)

Bob LaMuro (W)

Randy Fieldhack (WE)

MEETING AGENDA

INCREASED ALLOWABLE CORE PEAKING FACTORS

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

February 4, 1988

I. Introduction/Schedule — WE

II. Non-LOCA Transients

- A. Scope of Analyses & Evaluations — W
- B. Status of Generic WCAP Approval — NRC
  1. WCAP-11397 Revised Thermal Design Procedure (RTDP)
  2. WCAP-11394 WOG Dropped Rod Generic Methodology

III. Small-Break LOCA

- A. NOTRUMP Analysis — WE
- B. Reanalysis Strategy — WE

IV. Possible Technical Specification/FSAR Changes — WE

- A. Increased Peaking Factors
- B. Part-length Hafnium Absorber Assemblies
- C. Thimble Plug Removal
- D. Increased Steam Generator Tube Plugging
- E. Enhanced OFA Fuel Features

PURPOSE OF MEETING

- Present WE's Reasons, Strategy, and Schedule for Requesting a License Amendment to Increase Allowable Core Peaking Factors (F_Q and $F_{\Delta H}$) at Point Beach

- Obtain Feedback from the NRC Staff on Strategy and Schedule so WE Can Proceed With Desired Fuel Cycle Design and Implementation (Unit 1, Cycle 17 - Spring 1989 Refueling)

- Resolve Break-Spectrum Issue for Small-Break LOCA Analysis

ACCIDENT REANALYSIS AT INCREASED PEAKING FACTORS

GOALS:

- o Implement Low-Low Leakage Loading Pattern (L4P) Fuel Management Scheme with Part-Length Non-Burnable Absorbers in Peripheral Assemblies to Reduce Reactor Vessel Neutron Fluence
  - Enhance Ability for Additional Twenty (20) Years of Vessel Life
  - Address Current Reactor Vessel Safety Issues (PTS, Low Upper Shelf Material Toughness, Heatup and Cooldown Restrictions) and Uncertainty in Future NRC Requirements (e.g., Proposed Rev. 2 to R.G. 1.99)

- o Implement L4P and Enhanced OFA Fuel Features (e.g., Axial Blankets, High Burnup) to Reduce Fuel Cycle Costs

- o Improve Operational Flexibility
  - Removal of Thimble Plug Devices
  - Increased Steam Generator Tube Plugging

SCHEDULE: Implement L4P Reload Designs As Follows:

- o Unit 1 - Spring 1989 (Order Fuel - April 1988)
- o Unit 2 - Fall 1989 (Order Fuel - September 1988)

INCREASED PEAKING FACTOR STRATEGY

- Reanalyze Limiting Transients First (2/15/88)
  - Dropped RCCA ($F_{\Delta H}$)
  - Rod Ejection (F_Q)

- Complete Reanalysis/Evaluation Effort for Remaining Accidents Including SBLOCA (6/30/88)

- Coordinate with Upper Plenum Injection/Best-Estimate LOCA Analysis Project

- Request License Amendment (i.e., Tech Spec Revision) to Allow Increased Peaking Factors (8/31/88):
  1. $F_{\Delta H} = 1.70$
  2. $F_Q = 2.50$
  3. Certain RPS Settings
     - Overtemperature ΔT
     - Overpower ΔT
  4. Control Rod and Power Distribution Limits
  5. DNB Safety Limit Curves
  6. Non-Burnable Absorber Assemblies

- Reload Core 10CFR50.59 Safety Evaluation (April 1989)
  - Removal of Thimble Plug Devices
  - Insertion of Part-Length Non-Burnable Absorbers in Peripheral Fuel Assemblies

REQUIRED ANALYSES/EVALUATIONS FOR LICENSE AMENDMENT

- LOCA Reanalysis at Higher F_Q of 2.50
  - Large Break - Best-Estimate WCOBRA/TRAC
  - Small Break - NOTRUMP

- FSAR Chapter 14 Accident/Transient Reanalysis/Evaluations at Higher $F_{\Delta H}$ of 1.70 (Rod Ejection at F_Q of 2.50)
  - Revised Thermal Design Procedure (RTDP) (WCAP-11397 Submitted 3/16/87)
  - WOG Dropped Rod Generic Methodology (WCAP-11394 Submitted 5/22/87)
  - Operational Considerations - Maintain 2000 psia, Up to 14% SG Tube Plugging, Eliminate Thimble Plugs
  - OFA Fuel Enhancements - Higher Burnup, Axial Blanket, IFBAs, DFBN

Point Beach

Increased Peaking Factor Effort

Non-LOCA Considerations

Areas of Impact

Increase in allowable $F_{\Delta H}$

DNB - criteria events

- o Core Limits/Setpoints
- o Flow Transients
- o Dropped Rod (WCAP 11394)

Increase in allowable F_Q

Fuel/clad temperature criteria events

Thimble Plug Removal

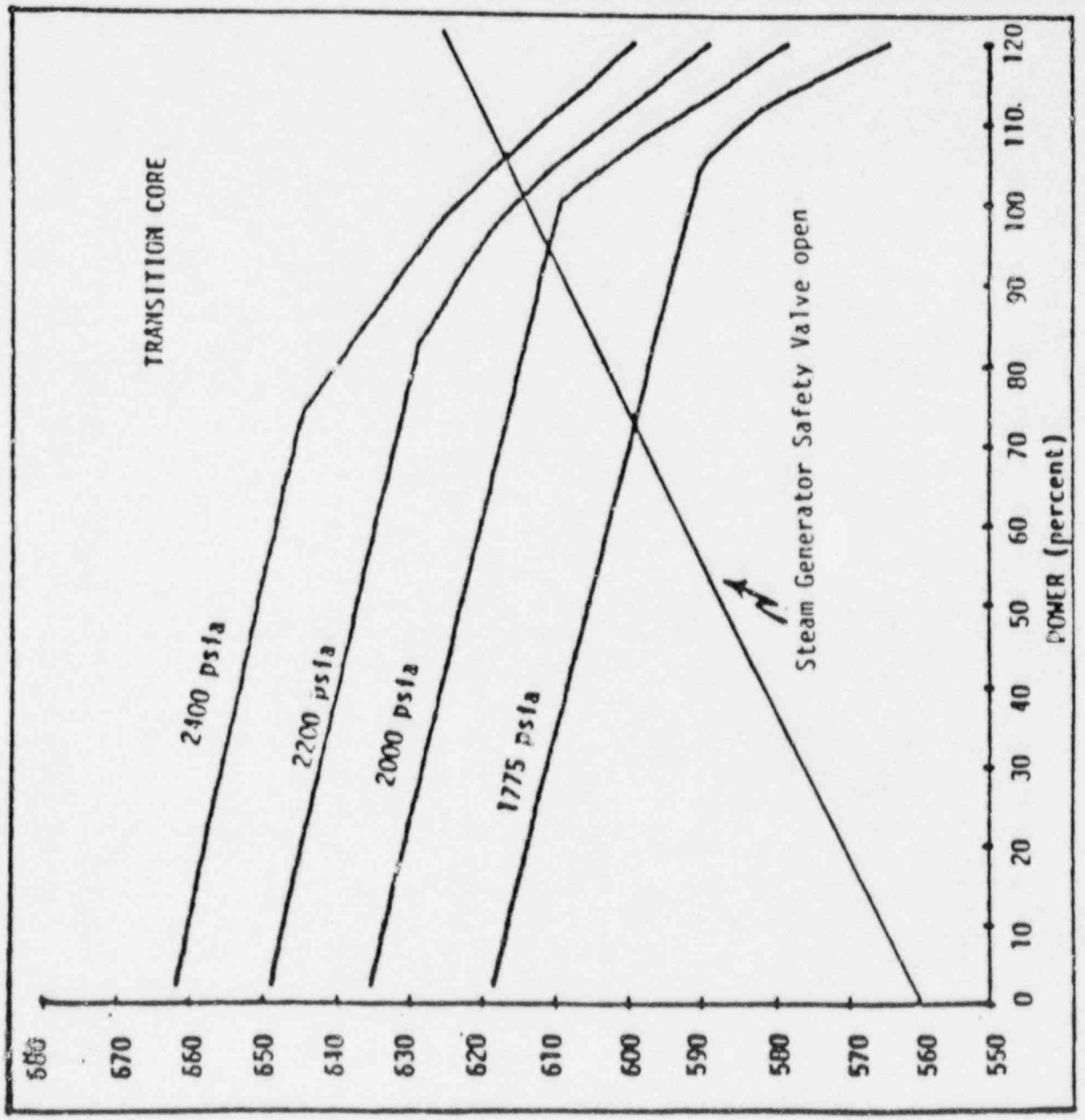
Increase core bypass flow

RTDP: Revised Thermal Design Procedure (WCAP 11397)

Core Limits/Setpoints

Figure 15.2.1-1

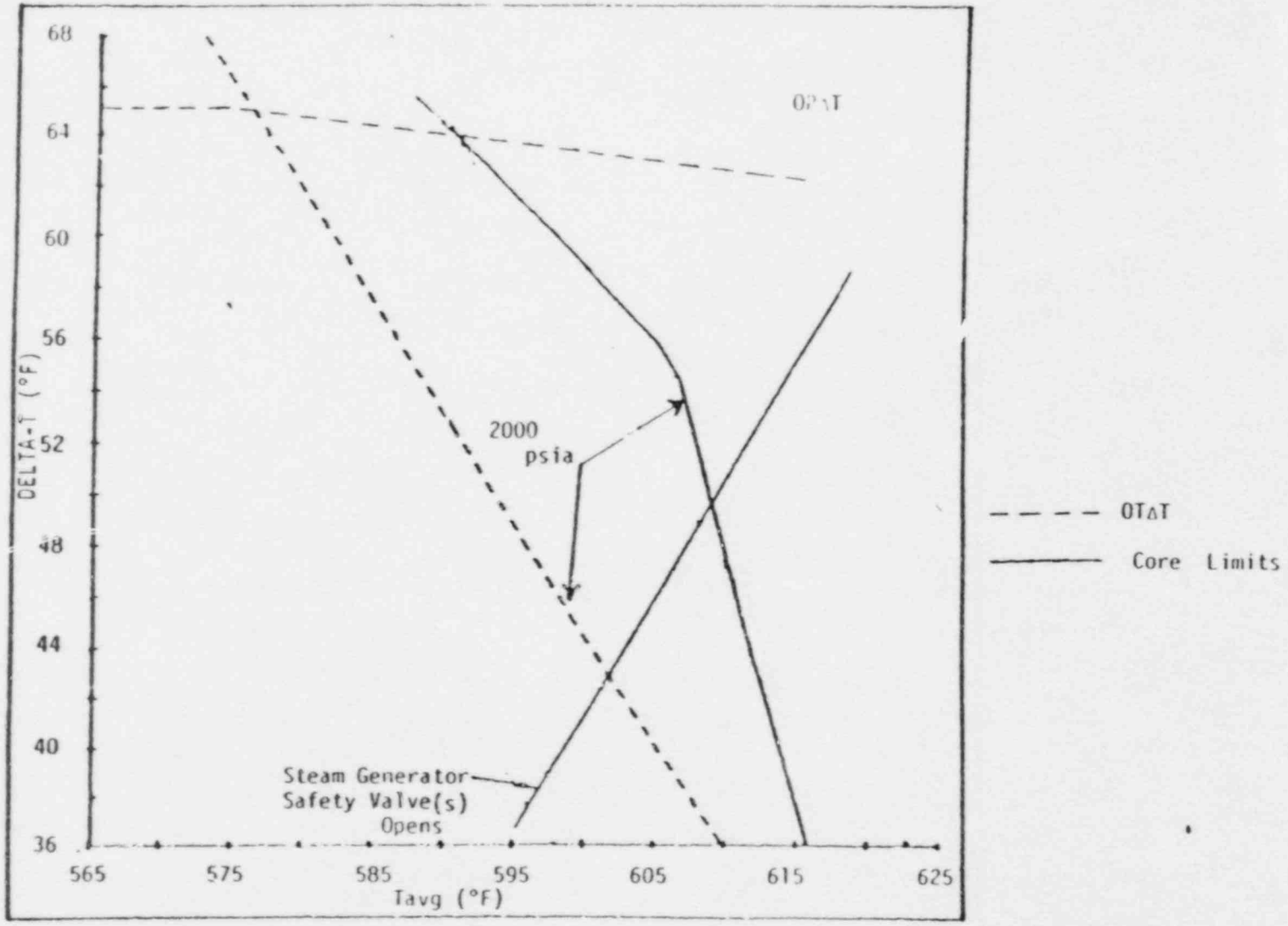
CORE DNB SAFETY LIMITS (TRANSITION CORE)
POINT BEACH UNITS 1 AND 2



T_{avg} - (°F)

FIGURE 14-1

ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER ΔT PROTECTION



EVENTS REQUIRING NEW ANALYSIS FOR INCREASED PEAKING FACTORS

EVENT	FSAR	Core Limits	FdH	Fq	Bypass
OTDT/OPDT Setpoint Calculation		X			X
Rod Withdrawal at Power	14.1.2	X			
Rod Withdrawal from Subcritical	14.1.1		X		
Dropped Rod	14.1.3	X			
Boron Dilution at Power at Startup at Shutdown at Refueling	14.1.4	X			
Startup of an Inactive Loop	14.1.5				X
Reduction in Feedwater Enthalpy	14.1.6				
Excessive Load Increase	14.1.7	X			
Loss of Flow	14.1.8		X		
Locked Rotor	14.1.8		X		X
Loss of Load	14.1.9	X			
Loss of Normal Feedwater	14.1.10				X
Station Blackout	14.1.11				X
Steamline Break, Core Response , Containment Response	14.2.5 14.2.5				
Rod Ejection	14.2.6			X	

NOTRUMP ANALYSIS

- o TMI Requirements
  - NOTRUMP Description

    WCAP - 10079-P-A, "NOTRUMP: A Nodal Transient Small Break and General Network Code"

    WCAP - 10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code"

  - Generic Study

    WCAP - 11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study With The NOTRUMP Code"

- o Wisconsin Electric Status

GENERIC STUDY - WCAP-11145-P-A

Assumptions: 2-Loop Plant

Downflow Barrel-Baffle Configuration

0% Steam Generator Tube Plugging

1709 MWt Core Power

2.32 F_Q

Results:

Break Size (inches)	NOTRUMP (PCT)	WFLASH (PCT)
3	NCU*	1132
4	796	1713
6	757	1547

* No Core Uncovery

WFLASH 74 EM

- Earlier Point Beach Analysis:
  - Assumptions: Standard Fuel
    1518 MWt Core Power
    2.32 F_Q
    Downflow Barrel-Baffle Configuration

  - Results:

Break Size (Inches)	WFLASH (PCT)
3	1469.
4	1688.
6	1167.

- Earlier Prairie Island Analysis:
  - Assumptions: Standard Fuel
    1709 MWt Core Power
    2.32 F_Q
    Downflow Barrel-Baffle Configuration

  - Results:

Break Size (Inches)	WFLASH (PCT)
3	1132.
4	1713.
6	1547.

WFLASH - OCT 75 EM

- Current Point Beach Analysis
  - Assumptions:
    - OFA Fuel
    - 1518 MWt Core Power
    - 2.32 F_Q
    - 44F Steam Generators
    - Downflow Barrel-Baffle Configuration
  - Results:

Break Size (Inches)	WFLASH (PCT)
4	890.
6	992.
8	750.5

- Kori-1 Analysis
  - Assumptions:
    - OFA Fuel
    - 1723.5 MWt Core Power
    - 2.32 F_Q
    - Upflow Barrel-Baffle Configuration
  - Results

Break Size (Inches)	WFLASH (PCT)
4	1163.
6	1375.
8	1286.

- R.E. Ginna Analysis
  - Assumptions:
    - OFA Fuel
    - 1520 MWt Core Power
    - 2.32 F_Q
    - Downflow Barrel-Baffle Configuratio
  - Results:

Break Size (Inches)	WFLASH (PCT)
4	976
6	1092
8	758

NOTRUMP

- Generic Analysis:
  - Assumptions: 1709 MWt Core Power
    2.32 F_Q
    0% Steam Generator Tube Plugging
    Downflow Barrel-Baffle Configuration

  - Results

Break Size (Inches)	NOTRUMP (PCT)
3	NCU*
4	796.
6	757.

- Prairie Island Analysis:
  - Assumptions: 1650 MWt Core Power
    2.5 F_Q
    10% Steam Generator Tube Plugging
    Downflow Barrel-Baffle Configuration
    K(Z) Third Line Segment Removal

  - Results:

Break Size (Inches)	NOTRUMP (PCT)
3	NCU*
4	1000.
6	NCU*

*No Core Uncovery

COMPARISON OF POINT BEACH AND PRAIRIE ISLAND

NOTRUMP INPUT ASSUMPTIONS

POINT BEACH	PRAIRIE ISLAND
1518 MWt Core Power	1650 MWt Core Power
2.50 F_Q	2.50 F_Q
1.70 $F_{\Delta H}$	1.70 $F_{\Delta H}$
K(Z) Third Line Removal	K(Z) Third Line Removal
25% SGTP	10% SGTP
Upflow Configuration	Downflow Configuration

SBLOCA Reanalysis Strategy

- Select Expected Limiting Break Size
  - 4" Cold Leg Break
- Analyze This Break Using NOTRUMP
- Evaluate For Acceptable Results

  Criteria: 1. PCT $\leq$ 1600^{o}F

  2. Core Uncovery

BASIS OF PCT CRITERION

A 1600°F PCT Cutoff Will:

- Ensure PCT for All Break Sizes Fall Below the 2200°F Limit:

  Two-Loop Plant SBLOCA Analyses Results

Model	Max. ΔPCT Between Break Sizes
WFLASH 74 EM	600°F
WFLASH 75 EM	340°F
NOTRUMP	40°F

- Prevent Any Significant Zirc-Water Reaction

- Ensure Significant Margin to the 2200°F Limit

POTENTIAL FSAR/TECH SPEC CHANGES

- o Increased Peaking Factors
  - Technical Specification Changes
    1. Core Safety Limits
    2. OPΔT, OTΔT
    3. Rod Insertion Limits
    4. F_Q, $F_{\Delta H}$
    5. Third Line Segments, K(Z) Curve
    6. RAOC Band
  - Accident Analyses - Chapter 14 FSAR

- o Part-Length Absorber Assemblies - Tech Spec Design Features

- o Thimble Plugs - FSAR Change

- o Increased Steam Generator Tube Plugging - Future License Amendment

- o Enhanced OFA Fuel Features
  - IFBA/Axial Blankets - Add to Tech Specs, FSAR
  - Debris Filter Bottom Nozzles - 10CFR50.59 Review
  - Removable Top Nozzles/Higher Burnup - 10CFR50.59 Review

Figure 15.2.1-1

CORE DNB SAFETY LIMITS (TRANSITION CORE)
POINT BEACH UNITS 1 AND 2

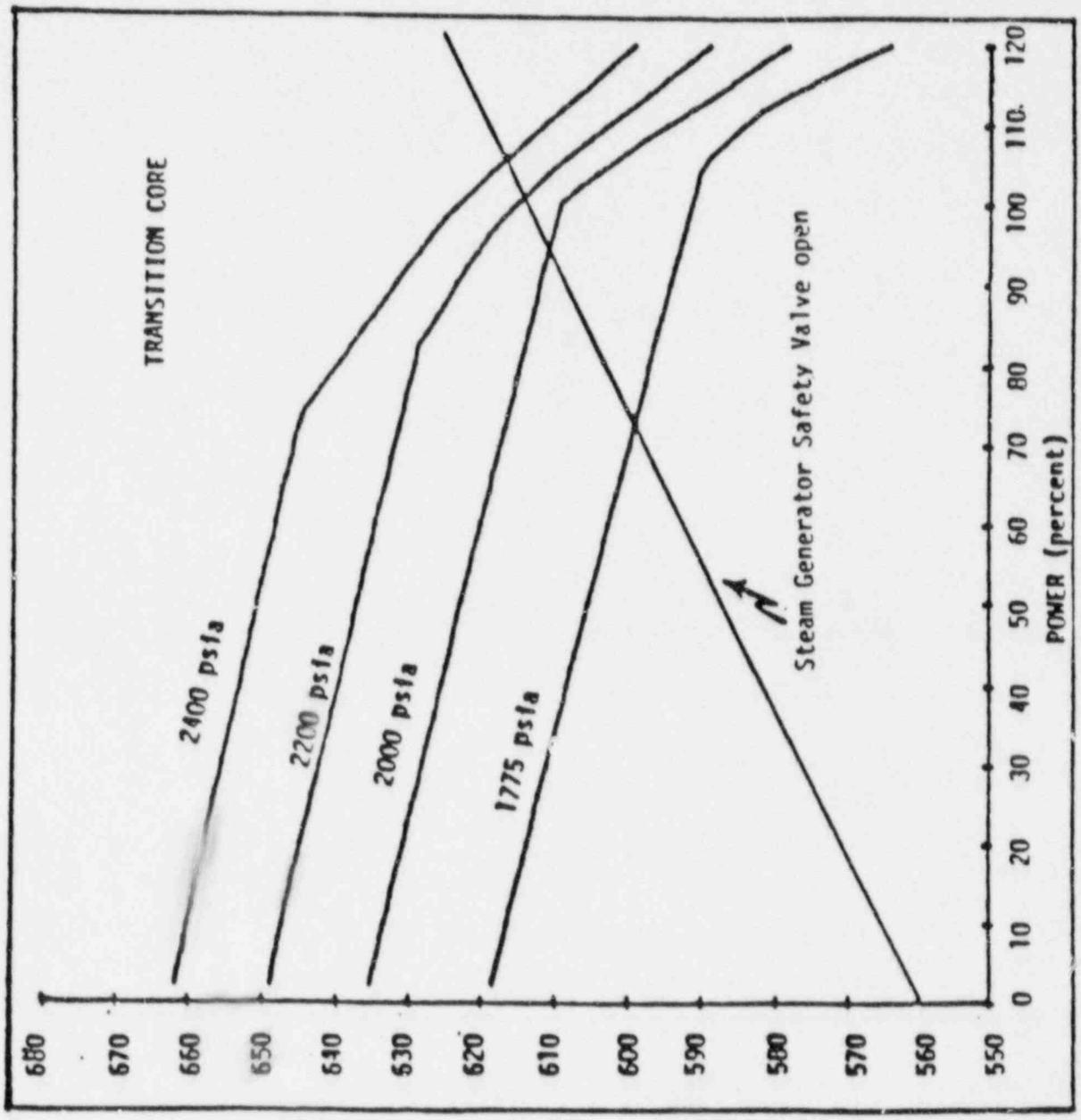


FIGURE 14-1

ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER ΔT PROTECTION

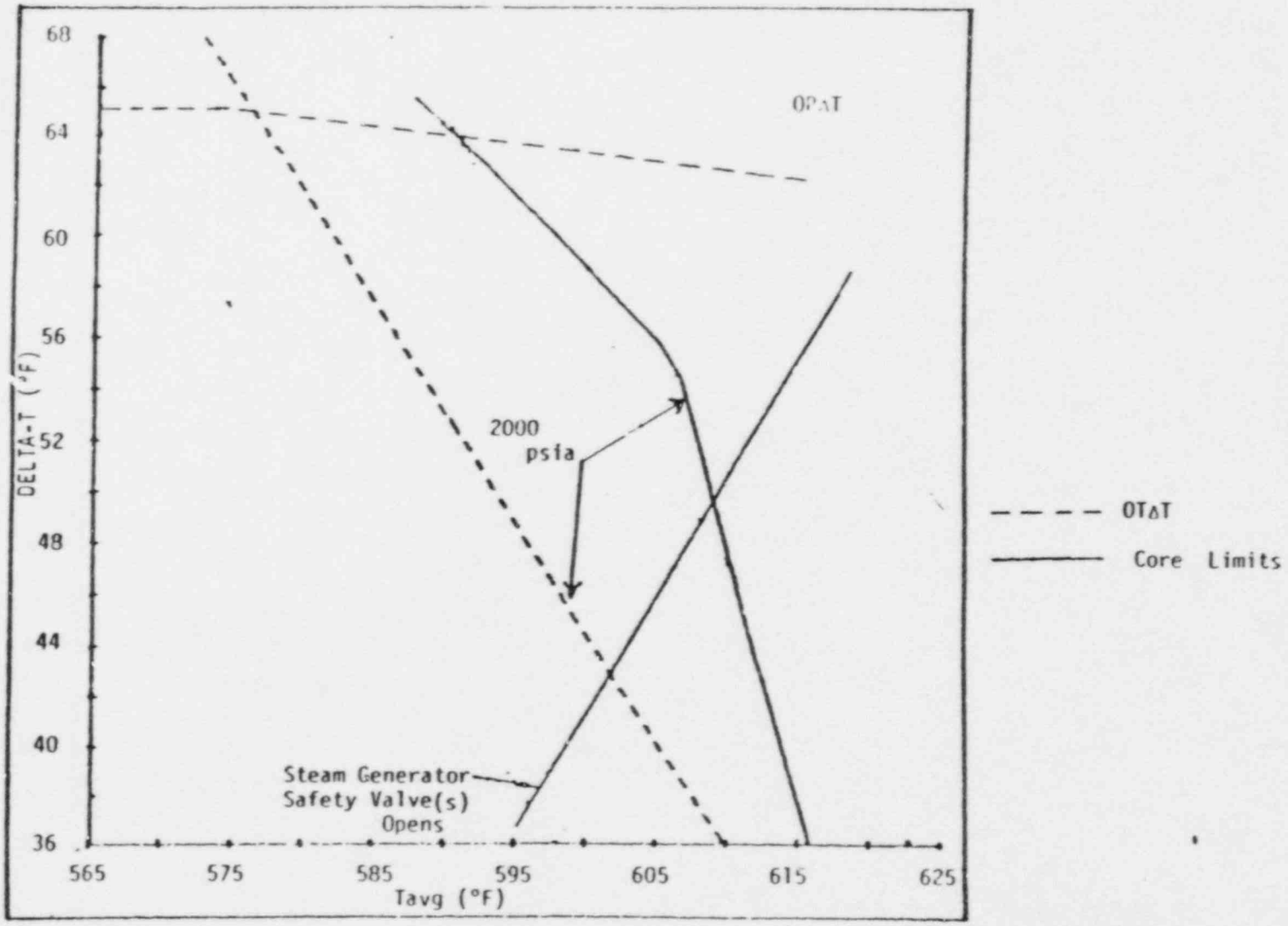
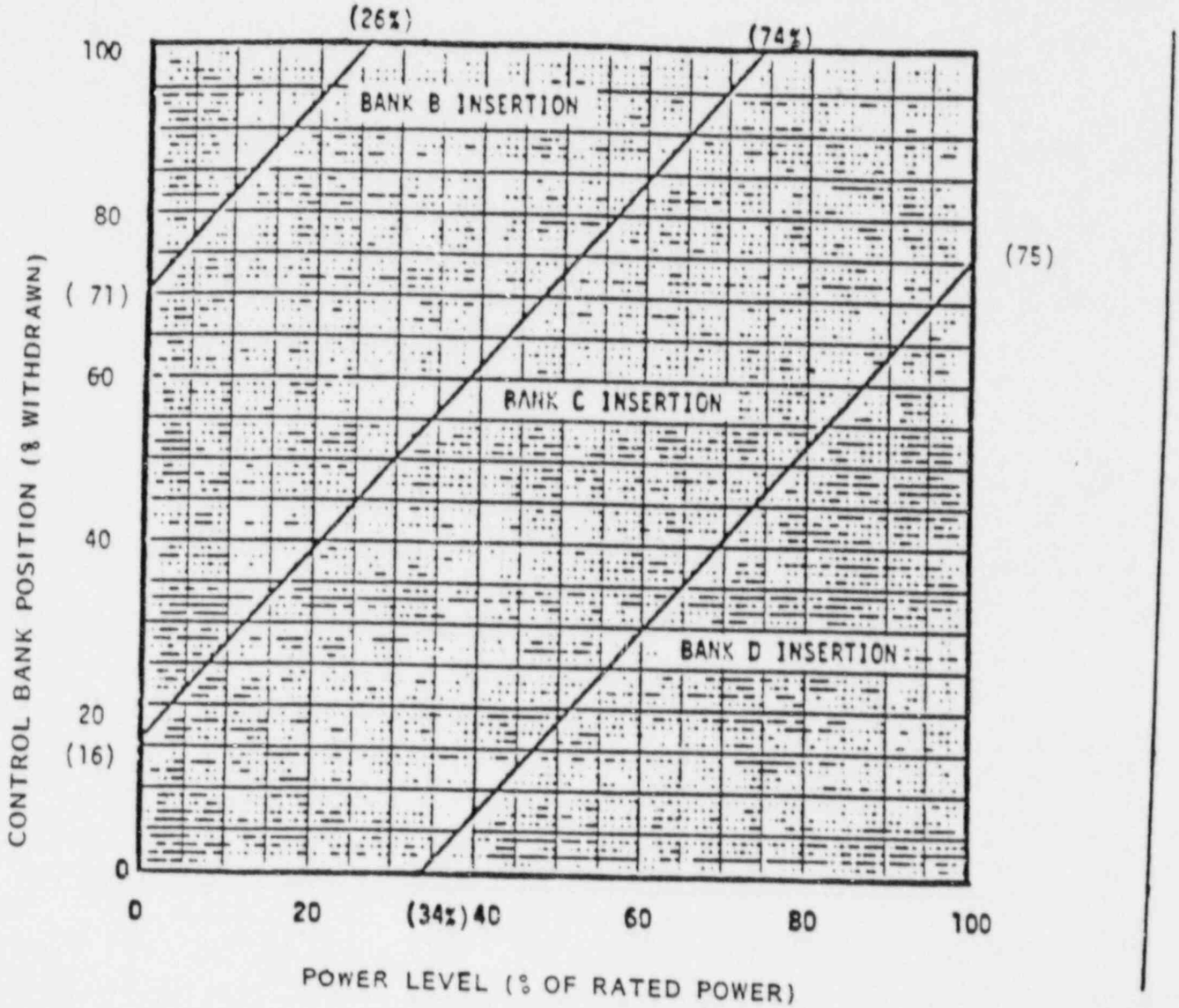


FIGURE 15.3.10-1
CONTROL BANK INSERTION LIMITS
POINT BEACH UNITS 1 AND 2



Unit 1 Amendment No. 86
Unit 2 Amendment No. 90

May 22, 1985
October 5, 1984

B. Power Distribution Limits

1. a. Except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$$F_Q(Z) \leq \frac{(2.21)}{P} \times K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq 4.42 \times K(Z) \qquad \text{for } P \leq 0.5$$

$$F^N_{\Delta H} < 1.58 \times [1 + 0.3\,(1-P)]$$

Where P is the fraction of full power at which the core is operating, K(Z) is the function in Figure 15.3.10-3 and Z is the core height location of F_Q.

b. Following a refueling shutdown prior to exceeding 90 percent of rated power and at effective full power monthly intervals thereafter, power distribution maps using the moveable incore detector system shall be made to confirm that the hot channel factor limits are satisfied. The measured hot channel factors shall be increased in the following way:

(1) The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

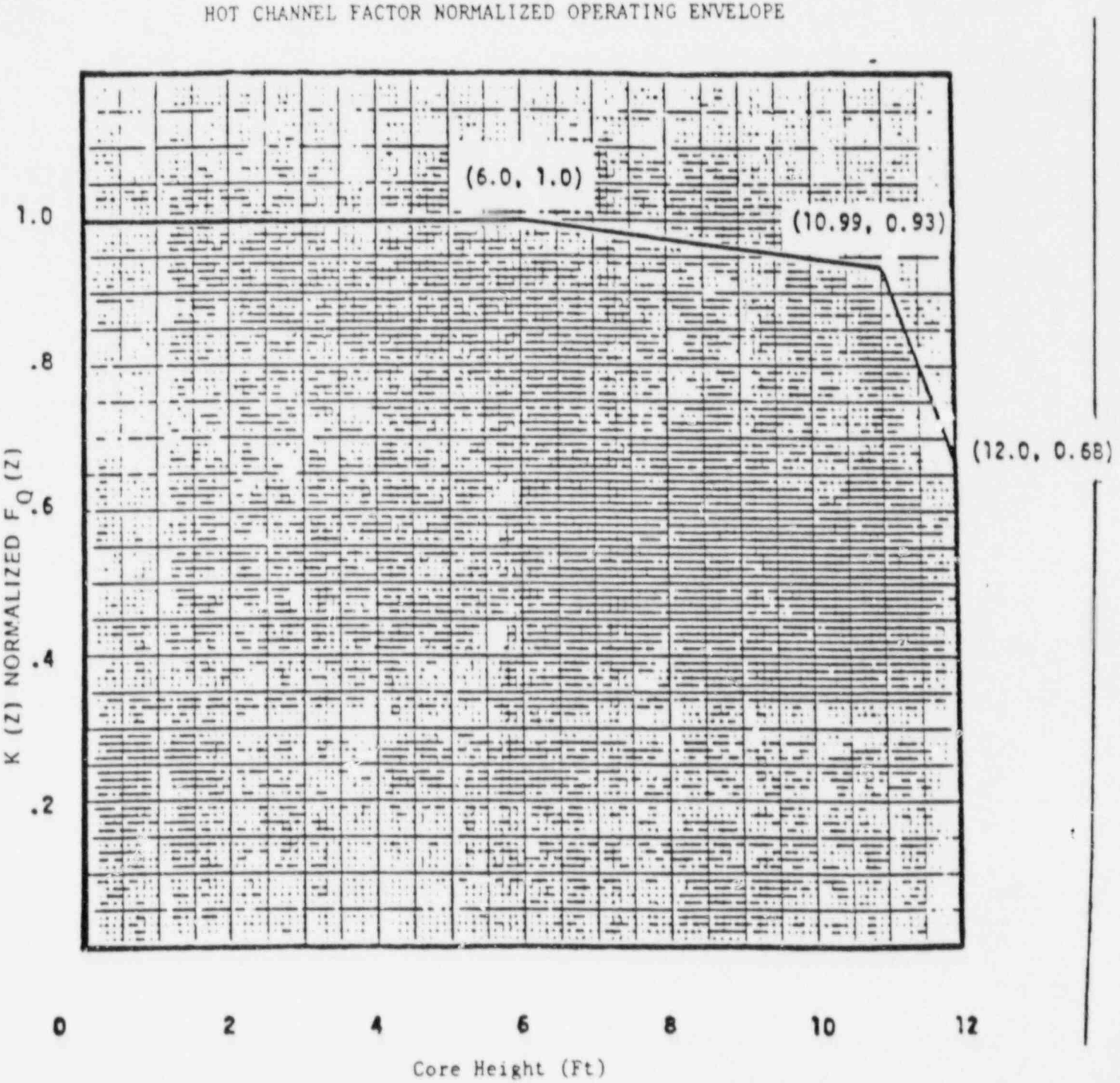
(2) The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error.

c. If a measured hot channel factor exceeds the full power limit of Specification 15.3.10.B.1.a, the reactor power and power range high setpoints shall be reduced until those limits are met. If subsequent flux mapping cannot, within 24 hours, demonstrate that the full power hot channel factor limits are met, the overpower

FIGURE 15.3.10-3

POINT BEACH UNITS 1 AND 2

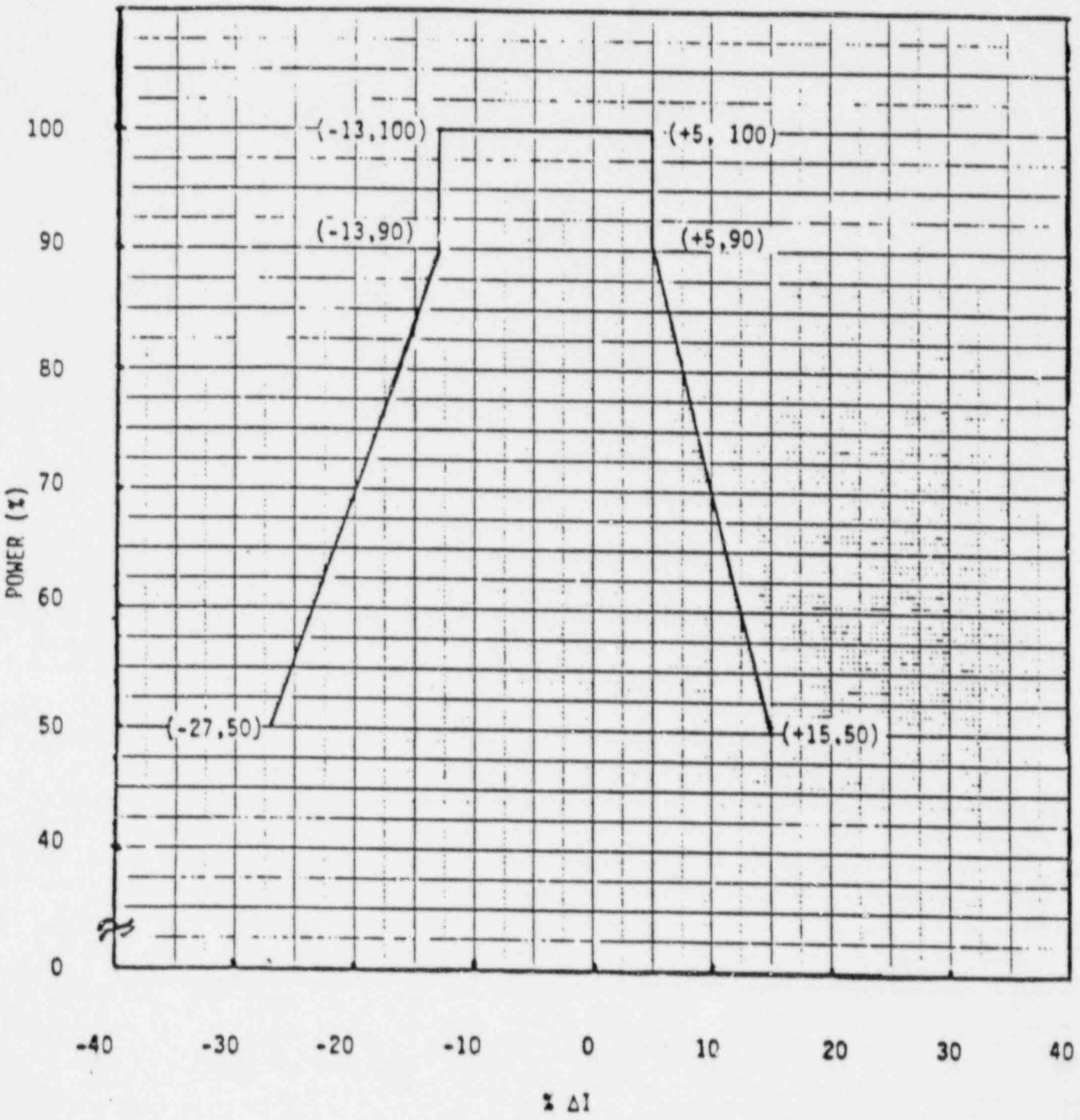
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



Unit 1 Amendment No. 86 May 22, 1985

Unit 2 Amendment No. 90 October 5, 1984

FIGURE 15.3.10-4
FLUX DIFFERENCE
OPERATING ENVELOPE
POINT BEACH UNITS 1 AND 2

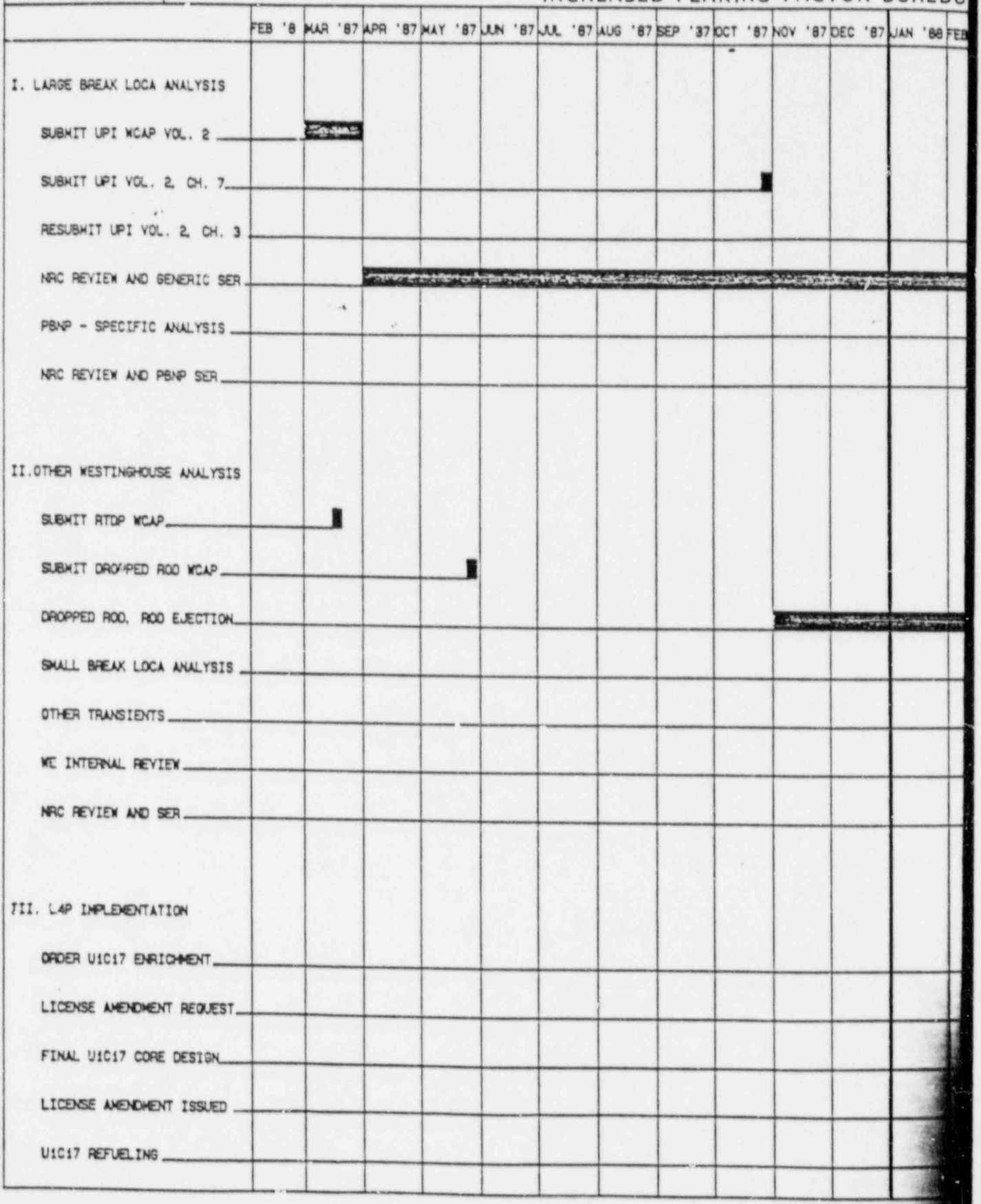


Unit 1 Amendment 86 May 22, 1985
Unit 2 Amendment 90 October 5, 1984

PFA 12/17/87

GANTT CHART

INCREASED PEAKING FACTOR SCHEDU



LE - POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

REVISION 7