December 22, 1989

Docket Nos. 50-266
and 50-301

LICENSEE: Wisconsin Electric Power Company

FACILITY: Point Beach Nuclear Plant, Unit Nos. 1 and 2

SUBJECT: MEETING SUMMARY (TAC NOS. 71534/71535)

On August 10, 1989, a meeting was held among representatives of the Wisconsin Electric Power Company (WEPCO), Westinghouse, Babcock & Wilcox, and the NRC staff. The purpose of the meeting was to update the NRC staff concerning WEPCO's ongoing reactor vessel materials integrity program for the Point Beach Nuclear Plant (PBNP). Enclosure 1 contains the list of attendees. Enclosure 2 contains a copy of the briefing slides presented during the meeting by WEPCO, Westinghouse, and Babcock & Wilcox.

Since the meeting was informational and educational in nature, no conclusions or agreements resulted. WEPCO summarized the PBNP Reactor Vessel Integrity Program along with plans for reactor vessel life extension. Westinghouse and Babcock & Wilcox provided supporting data concerning the PTS scoping risk assessment and the low upper shelf energy fracture toughness issues. Technical Specification change requests concerning the Point Beach heatup/cooldown limitation curves and changes to the Unit 2 reactor vessel surveillance schedule were also discussed.

/s/

Warren H. Swenson, Project Manager
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures: As stated

cc: See next page

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Mr. C. W. Fay
Wisconsin Electric Power Company

Point Beach Nuclear Plant
Units 1 and 2

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ENCLOSURE 1

Meeting with Wisconsin Electric Power Company,

Westinghouse, Babcock & Wilcox and the NRC Staff Regarding:

Point Beach - Reactor Vessel Integrity

August 10, 1989

NAME	ORGANIZATION
A. Reimer	WEPCO
M. Moylan	WEPCO
J. Kohlwey	WEPCO
D. Schoon	WEPCO
A. Lowe, Jr.	B&W
K. Moore	B&W
K. Balkey	Westinghouse
W. Bamford	Westinghouse
P. Randall	RES/MEB
M. Mayfield	RES/MEB
A. Taboada	RES/MEB
R. Woods	RES/RPSIB
C. Cheng	NRR/EMTB
R. Hermann	NRR/EMTB
K. Wichman	NRR/EMTB
B. Elliot	NRR/EMTB
J. Tsao	NRR/EMTB
S. Lee	NRR/EMTB
L. Lois	NRR/SRXB
W. Swenson	NRR/PD33

ENCLOSURE 2

AGENDA

Point Beach Reactor Vessel Integrity Meeting

NRC/Wisconsin Electric Power Company
White Flint Office Building
Rockville, Maryland
August 10, 1989

Section	Topic	Time	Discussion Leader
I	Introduction	9:00	Moylan (WE)/ Elliott (NRC)
I	PBNP Reactor Vessel Integrity Program	9:10	Moylan (WE)
I	Flux Reduction Evaluation/ Implementation	9:30	Balkey (W), Moylan (WE)
II	Planned Technical Specification Changes	10:00	Moylan (WE)
	-Heatup/Cooldown Limitation Curves -Surveillance Capsule Schedules		
III	ISI Activities	10:15	Kohlwey (WE)
IV	PTS - Scoping Risk Assessment	11:00	Balkey (W)/ Bamford (W)
	LUNCH	12:15	
V	Master Integrated Surveillance Program	1:00	Moore (B&W)
VI	Low Uppershelf Toughness (Appendix G) Issue	1:30	Lowe (B&W)
VII	Appendix G Fracture Toughness	2:00	Bamford (W)
	Summary and Conclusion	2:30	Moylan (WE)

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POINT BEACH REACTOR VESSEL INTEGRITY MEETING

OBJECTIVES

- Present chronology of Wisconsin Electric's program to meet regulatory requirements for vessel integrity
- Characterize the critical weld materials in the Point Beach Nuclear Plant (PBNP) reactor vessels
- Show results of Point Beach flux reduction program and discuss its implementation status
- Show methodology and results of the Point Beach scoping risk assessment
- Discuss background information for pending PBNP Technical Specification change requests
- Show how the BWOG Integrated Reactor Vessel Surveillance Program (MIRVP) provides data applicable to the licensability of PBNP
- Discuss BWOG Materials Committee and licensee programs and plans for addressing Appendix G
- Present current status of ASME-Fracture Toughness Margins
    - Include relationship to ATWS event

REGULATORY REQUIREMENTS -
LOW UPPER SHELF FRACTURE TOUGHNESS

- INFORM NRC 3 YEARS PRIOR TO DROPPING BELOW 50 FT-LB LEVEL
- 3 REQUIREMENTS IF PREDICTED TO DROP BELOW 50 FT-LB (10 CFR 50 APPX G V.C.)

(1) PERFORM VOLUMETRIC EXAMINATION OF THE BELTLINE MATERIAL OF CONCERN, CHARACTERIZE ANY FLAW FOUND AS DESCRIBED IN ASME SECTION XI.

(2) OBTAIN ADDITIONAL EVIDENCE OF MATERIAL'S FRACTURE TOUGHNESS FROM SUPPLEMENTAL FRACTURE TOUGHNESS TESTS.

(3) PERFORM SAFETY ANALYSIS DEMONSTRATING CONTINUED SAFE OPERATIONS

SIGNIFICANT EVENTS
PBNP REACTOR VESSEL
INTEGRITY PROGRAM

November 2, 1970	Initial criticality for Unit 1.
May 30, 1972	Initial criticality for Unit 2.
September 1972	Unit 1 Capsule V removed.
March 13, 1974	WE submitted the Unit 1 Capsule V evaluation report to the NRC.
November 1974	Unit 2 Capsule V removed.
June 6, 1975	WE proposed new Unit 2 heatup and cooldown limits based on the preliminary surveillance Capsule V test results. Test specimens indicated a larger temperature shift than was predicted.
July 2, 1975	WE submitted the Unit 2 Capsule V evaluation report. The report noted that the average upper shelf energy level of the weld metal Charpy Impact specimens decreased to 42 ft-lbs. This report did not contain WOL testing results.
September 18, 1975	NRC requested that WE delay testing of WOL samples until additional testing guidance was available.
October 7, 1975	WE submitted the proposed Reactor Vessel Surveillance Program to meet the requirements of Appendices G & H. This also proposed removing 5 capsules rather than the 4 required by Appendix H.
March 4, 1976	WE, Westinghouse, and NRC met to discuss our proposed Appendix G Program.
April 22, 1976	NRC approved our reactor vessel fracture toughness evaluation program for Appendix G compliance.
March 4, 1977	WE submitted the results of our fracture toughness evaluation program thereby satisfying the analytical and procedural requirements of Appendix G.
August 18, 1977	WE submitted the Unit 1 reactor vessel in-service inspection results from the fall 1976 inspection.

SIGNIFICANT EVENTS
PBNP REACTOR VESSEL
INTEGRITY PROGRAM (CONTINUED)

December 13, 1977	NRC informed WE that detailed review of PBNP reactor vessel materials toughness submittals would not be conducted pending the NRC generic study of this issue.
May 4, 1978	WE submitted the Unit 2 reactor vessel in-service inspection results from the spring 1977 inspection.
November 8, 1978	WE submitted the Unit 1 Capsule R and Unit 2 Capsule T test reports. Unit 1 Capsule R was removed in October 1977. Unit 2 Capsule T was removed in March, 1977.
February 27, 1979	NRC indicated that WOL testing should be delayed until at least 1980 when further testing guidance would be available.
April 16, 1980	The Unit 2 Capsule R test report was submitted to the NRC. This capsule was removed in April 1979.

POINT BEACH NUCLEAR PLANT
ANALYSES SUBMITTED IN COMPLIANCE WITH APPENDIX G 1976-1977

UNIT 1

WCAP-8740 September 1976	Fatigue Crack Growth Evaluation of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel
WCAP-8741 September 1976	ASME III, Appendix G Analysis of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel
WCAP-8743 January 1977	Heatup and Cooldown Limit Curves for the Wisconsin Electric Power Company and the Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1
WCAP-8742 February 1977	Fracture Mechanics Evaluation of the Wisconsin Electric Power Company and the Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel
WCAP-8927 March 1977	ASME III, Appendix G Analysis of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Inlet Nozzle Corner

UNIT 2

WCAP-8735 February 1977	Fatigue Crack Growth Evaluation of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel
WCAP-8736 September 1976	ASME III, Appendix G Analysis of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel
WCAP-8738 January 1977	Heatup and Cooldown Limit Curves for the Wisconsin Electric Power Company and the Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 2
WCAP-8737 February 1977	Fatigue Mechanics Evaluation of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel
WCAP-8928 March 1977	ASME III, Appendix G Analysis of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel Inlet Nozzle Corner

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SURVEILLANCE CAPSULE TEST REPORTS

UNIT 1

Battelle Columbus Report - June 1973	Point Beach Nuclear Plant Unit No. 1 Pressure Vessel Surveillance Program: Evaluation of **CAPSULE V**
WCAP-8739 November 1976	Analysis of **CAPSULE S** from the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program
WCAP-9357 August 1978	Analysis of **CAPSULE R** from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program
WCAP-10736 December 1984	Analysis of **CAPSULE T** from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program

UNIT 2

Battelle Columbus Report - June 1975	Point Beach Nuclear Plant No. 2 Pressure Vessel Surveillance Program: Evaluation of **CAPSULE V**
WCAP-9331 August 1978	Analysis of **CAPSULE T** from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel Radiation Surveillance Program
WCAP-9635 December 1979	Analysis of **CAPSULE R** from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel Radiation Surveillance Program

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RELATED REACTOR VESSEL INTEGRITY REPORTS

WCAP-7513 June 1970	Wisconsin Michigan Power Co. Point Beach Unit No. 1 Reactor Vessel Radiation Surveillance Program
WCAP-7712 June 1971	Wisconsin Michigan Power Co. and the Wisconsin Electric Power Co. Point Beach Unit No. 2 Reactor Vessel Radiation Surveillance Program
SWRI Project 17-3076 September 1971	Point Beach Unit No. 2 Nuclear Power Plant Preoperational Inspection Report
WCAP-7924-A April 1975	Basis for Heatup and Cooldown Limit Curves
SWRI Project 17-3540 April 1977	1976 Inservice Examination of Selected Components at Point Beach Nuclear Plant, Unit No. 1
SWRI Project 17-4048 December 1977	1977 Inservice Examination of Selected Components at Point Beach Nuclear Plant, Unit No. 2
MUHN-1070 January 1985	Westinghouse Owners Group Reactor Vessel Materials Data Base User's Manual
WCAP-11477 June 1987	Handbook on Flaw Evaluation for Point Beach Units No. 1 and 2 Reactor Vessels
WCAP-11478 June 1987	Background and Technical Basis for the Handbook on Flaw Evaluation for the Point Beach Units No. 1 and 2 Reactor Vessels
NES/DYNACON Ref. 026-2510 July 1987	URDPS Report on Automated Data Acquisition and Processing for the Point Beach Nuclear Plant Unit 1 Ten Year Reactor Vessel Examination, Spring 1987 Outage for Wisconsin Electric Power Company
SWRI Project 1597 August 1987	1987 Inservice Examination of the Reactor Pressure Vessel at Point Beach Nuclear Plant, Unit 1
SWRI Project 2388 February 1989 (Draft)	Ultrasonic Indication Sizing Technique Development

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ADDITIONAL INSERVICE INSPECTION REPORTS

SWRI Project 17-3540 February 1982	1981 Inservice Examination of Selected Class I Components at Point Beach Nuclear Plant, Unit No. 1
SWRI Project 7222 May 1984	1983 Inservice Examination of Seleected Class I and Class 2 Components of the Point Beach Nuclear Plant, Unit No. 1
SWRI Project 7472 December 1984	1984 Inservice Examination of Selected Class I and Class 2 Components of the Point Beach Nuclear Plant, Unit No. 2
SWRI Project 17-7222-130 March 1984	Reactor Vessel Outlet Nozzle-to-Shell Weld Flaw Indication Fracture Mechanics Evaluation for Point Beach Nuclear Plant Unit 1

SUMMARY OF 1970'S REACTOR VESSEL INTEGRITY CORRESPONDENCE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

S. Burstein (WE) Letter to E. Case (NRC), January 23, 1975; "Reactor Vessel Heatup and Cooldown Limits."

S. Burstein (WE) Letter to B. Rusche (NRC), March 11, 1975; "Technical Specification Change Request 21 Revision to Unit 2 Heatup and Cooldown Limits."

S. Burstein (WE) Letter to B. Rusche (NRC), June 6, 1975; "Modification to Technical Specification Change Request 21 Revision to Unit 2 Heat-up and Cool-down Limits."

S. Burstein (WE) Letter to B. Rusche (NRC), July 2, 1975; "Pressure Vessel Surveillance Program."

S. Burstein (WE) Letter to B. Rusche (NRC), October 7, 1975; "Reactor Vessel Surveillance Program."

G. Lear (NRC) Letter to S. Burstein (WE), December 11, 1975; "Delay Testing of Unit 2 First Capsule WOL Specimens."

G. Lear (NRC) Letter to S. Burstein (WE), January 22, 1976; "Request for Additional Information."

S. Burstein (WE) Letter to G. Lear (NRC), March 10, 1976; "Reactor Vessel Surveillance Program."

G. Lear (NRC) Letter to S. Burstein (WE), April 22, 1976; "Reactor Vessel Surveillance Program."

S. Burstein (WE) Letter to B. Rusche (NRC), May 19, 1976; "Reactor Vessel Surveillance Program."

G. Lear (NRC) Letter to S. Burstein (WE), August 11, 1976; "Reactor Vessel Surveillance Program."

S. Burstein (WE) Letter to B. Rusche (NRC), September 17, 1976; "Interim Heatup and Cooldown Curve."

S. Burstein (WE) Letter to B. Rusche (NRC), March 4, 1977; "Reactor Vessel Fracture Toughness."

G. Lear (NRC) Letter to S. Burstein (WE), May 20, 1977; "Request for Additional Information Re: PBNP Reactor Vessel and Associated Specimen Materials."

S. Burstein (WE) Letter to G. Lear (NRC), July 18, 1977; "Reactor Vessel Materials Information."

S. Burstein (WE) Letter to G. Lear (NRC), August 18, 1977; "Reactor Vessel In-Service Inspection

G. Lear (NRC) Letter to S. Burstein (WE), December 13, 1977; "PBNP Reactor Vessel Fracture Toughness."

P. Wagner (NRC) Memorandum to G. Lear (NRC), February 3, 1978; "Summary of Meeting Held With WEPCO to Discuss Fracture Toughness."

S. Burstein (WE) Letter to A. Schwencer (NRC), May 4, 1978; "Reactor Vessel In-Service Inspection."

IE Bulletin No. 78-12, September 29, 1978; "Atypical Weld Material In Reactor Vessel Pressure Vessel Welds."

S. Burstein (WE) Letter to J. Keppler (NRC), October 26, 1978; "Reactor Vessel Weldment Information."

S. Burstein (WE) Letter to H. Denton (NRC), November 8, 1978; "Reactor Vessel Materials Surveillance Capsule Test Reports."

A. Schwencer (NRC) Letter to S. Burstein (WE), February 27, 1979; "Request for Additional Information Regarding Capsule Test Reports."

S. Burstein (WE) Letter to A. Schwencer (NRC), May 4, 1979; "Reactor Vessel Surveillance Programs."

C. W. Fay (WE) Letter to H. Denton (NRC), April 16, 1980; "Reactor Vessel Materials Surveillance Capsule Test Report (Unit 2)."

TABLE 1

POINT BEACH NUCLEAR PLANT
REACTOR VESSEL INTEGRITY PROGRAM
1984 - PRESENT

Project	Date Complete	Date Planned For Implementation
1. Neutron exposure evaluation of Point Beach reactor vessels.(1)	December, 1984	
2. Tested Unit 1 Surveillance Capsule T.(2)	December, 1984	
3. 10 CFR 50.61 - Pressured Thermal Shock (PTS) Submittal.(3)	January, 1986	
Correction to PTS submittal.(4)	March, 1986	
Safety evaluation report received from NRC.(5)	July, 1986	
4. Reactor Vessel Life Extension Study.		
Initiated study in May, 1986.		
Evaluation of fuel management techniques and internals modifications (shielding) to meet flux reduction goals.(6)	September, 1987	
Identification of critical components in NSSS, including the reactor vessel, and compilation of transient data associated with these components.(7)	October, 1987	
Comprehensive scoping risk assessment to examine Point Beach specific concerns and the propriety of the flux reduction goals.(8)	December, 1987	
Developed bases and specifications for a plantwide on-line fatigue monitoring system.(9)	December, 1987	

TABLE 1

POINT BEACH NUCLEAR PLANT
REACTOR VESSEL INTEGRITY PROGRAM
1984 - PRESENT

Project	Date Complete	Date Planned For Implementation
5. Inservice Inspection		
a. Second Unit 1 Reactor Vessel Ten-year Exam:		
Performed ASME Code exam utilizing SWRI standard data acquisition system, including 50/70 tandem near surface search units.(10)	May, 1987	
Performed exam using NES/Dynacon Ultrasonic Data Recording and Processing System (UDRPS) concurrent with ASME Code exam above.(11)	May, 1987	
b. Second Unit 2 Reactor Vessel Ten-year Exam:		
SWRIS EDAS or NES/Dynacon's UDRPS system will be utilized.		October, 1989
6. Joined Babcock and Wilcox Owner's Group (BWOG) Materials Committee.	August, 1988	
Full participant in BWOG Reactor Vessel Integrity Program (RVIP).	August, 1988	
Participant in BWOG Reactor Vessel Life Extension Surveillance Program (RVSP).		1989
Developing master integrated reactor vessel surveillance program to include Westinghouse utilities with Linde 80 welds in their reactor vessels. This document will describe the new Point Beach Nuclear Plant surveillance capsule schedule and will be referenced in the Point Beach Technical Specifications.		March, 1989

TABLE 1

POINT BEACH NUCLEAR PLANT
REACTOR VESSEL INTEGRITY PROGRAM
1984 - PRESENT

Project	Date Complete	Date Planned For Implementation
7. Installation of excore neutron dosimetry (radiometric monitors and solid state track recorders) over one octant of both unit's reactor vessels.		
Install mounting hardware and first set of dosimetry in Unit 2.	November, 1988	
Install mounting hardware and first set of dosimetry in Unit 1.		May, 1989
Analysis of sensor sets and correlation of cavity measurements with transport calculations will be performed after each fuel cycle until sufficient confidence exists to extrapolate neutron exposure data through expected plant life.		
8. Pilot project: On-line fatigue monitoring of Unit 2 pressure surge nozzle (related to reactor vessel life extension study fatigue evaluation).	November, 1988	
9. Implement super Low Leakage Loading Pattern (L4P) cores and axially-zoned hafnium inserts in the guide tubes of peripheral assemblies.		
Unit 1		May, 1989
Unit 2		November, 1989
10. Perform image enhancement of selected radiographs of important reactor coolant system components (reactor vessels, piping, steam generators, etc.) and retain radiograph image on more permanent media.		1989
11. Submit revised heatup and cooldown curves using the guidance of Regulatory Guide 1.99, Revision 2.		July, 1989

SUPPORTING DOCUMENTATION FOR TABLE 1

1. Anderson, S. L. and Balkey, K. R., "Adjoint Flux Program for Point Beach Units 1 and 2," WCAP-10638, December 1984.

2. Perone, V. A., et al, "Flux Reduction Evaluation for the Point Beach Units 1 and 2 Reactor Vessel Life Extension Study," WCAP-11535, September 1987.

2. Yanichko, S.E., et al, "Analysis of Capsule T From the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-10736, December 1984.

3. Letter C. W. Fay to H. R. Denton, "Docket Nos. 50-266 and 50-301 Response to 10CFR50.61 Protection Against Pressurized Thermal Shock (PTS) Point Beach Nuclear Plant, Units 1 and 2," January 1986.

4. Letter C. W. Fay to H. R. Denton, "Docket Nos. 50-266 and 50-301 Correction to Pressurized Thermal Shock (PTS) Submittal Dated January 20, 1986 Point Beach Nuclear Plant, Units 1 and 2," March 1986.

5. Letter USNRC, T. G. Colburn to C. W. Fay "Projected Values of Material Properties for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Point Beach Nuclear Plant Unit No. 1 and 2," July 24, 1986.

7. Bond, C. B. and Pepka, J. M., "Transient Monitoring Program for Wisconsin Electric Power Company Point Beach Units 1 and 2, Phase 1 Final Report," WCAP-11501 (Westinghouse Proprietary) October 1987.

8. Balkey, K. R., et al "Scoping Risk Assessment for the Point Beach Units 1 and 2 Reactor Vessel Life Extension Study," WCAP-11676 (Westinghouse Proprietary) December 1987.

9. Bond, C. B. and Pepka, J. M., "Point Beach Units 1 and 2 Transient and Fatigue Cycle Monitoring System Functional Requirements," WCAP-11706, December 1987.

10. Enoch, H. D., "1987 Inservice Examination of the Reactor Pressure Vessel at Point Beach Nuclear Plant Unit 1," Final Report (Volumes 1-5) SwRI Project 1597, August 1987.

11. Fong, R. and Martens, G., "UDRPS Report on Automated Data Acquisition and Processing for the Point Beach Nuclear Power Plant Unit 1, 10-Year Reactor Pressure Vessel Examination, Spring 1987 Outage," July 9, 1987.

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TECHNICAL SPECIFICATION CHANGE REQUESTS (TSCR)

POINT BEACH NUCLEAR PLANT (PBNP)

- DOCKETS 50-266 AND 50-301
  TSCR #126
  REVISION TO HEATUP AND COOLDOWN LIMIT CURVES
  PBNP, UNITS 1 AND 2

- DOCKET 50-301
  TSCR #134
  REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE
  PBNP, UNIT 2

FIGURE 1

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL
FOR THE POINT BEACH UNIT NO. 1 REACTOR VESSEL

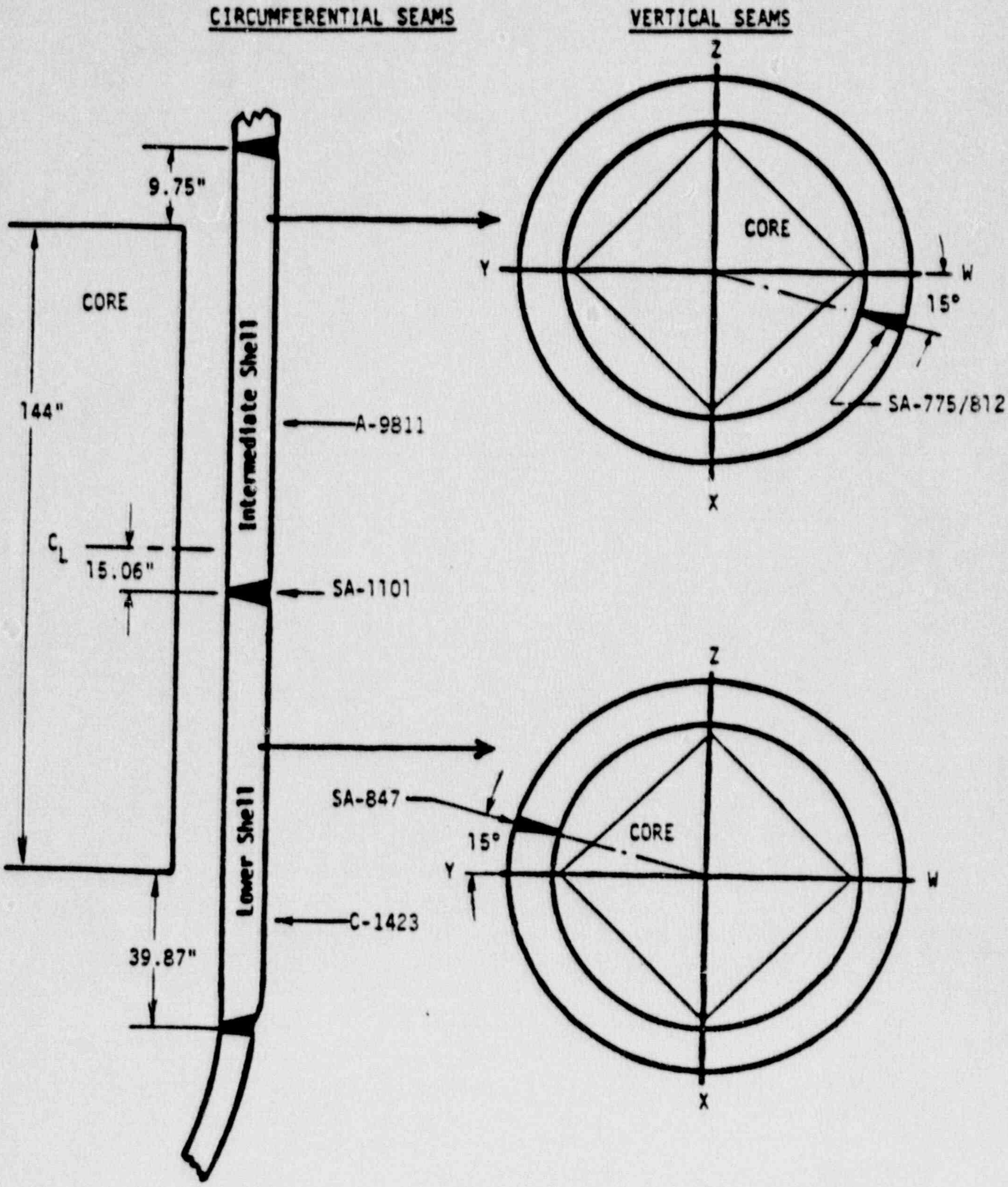


FIGURE 2

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE POINT BEACH UNIT NO. 2 REACTOR VESSEL

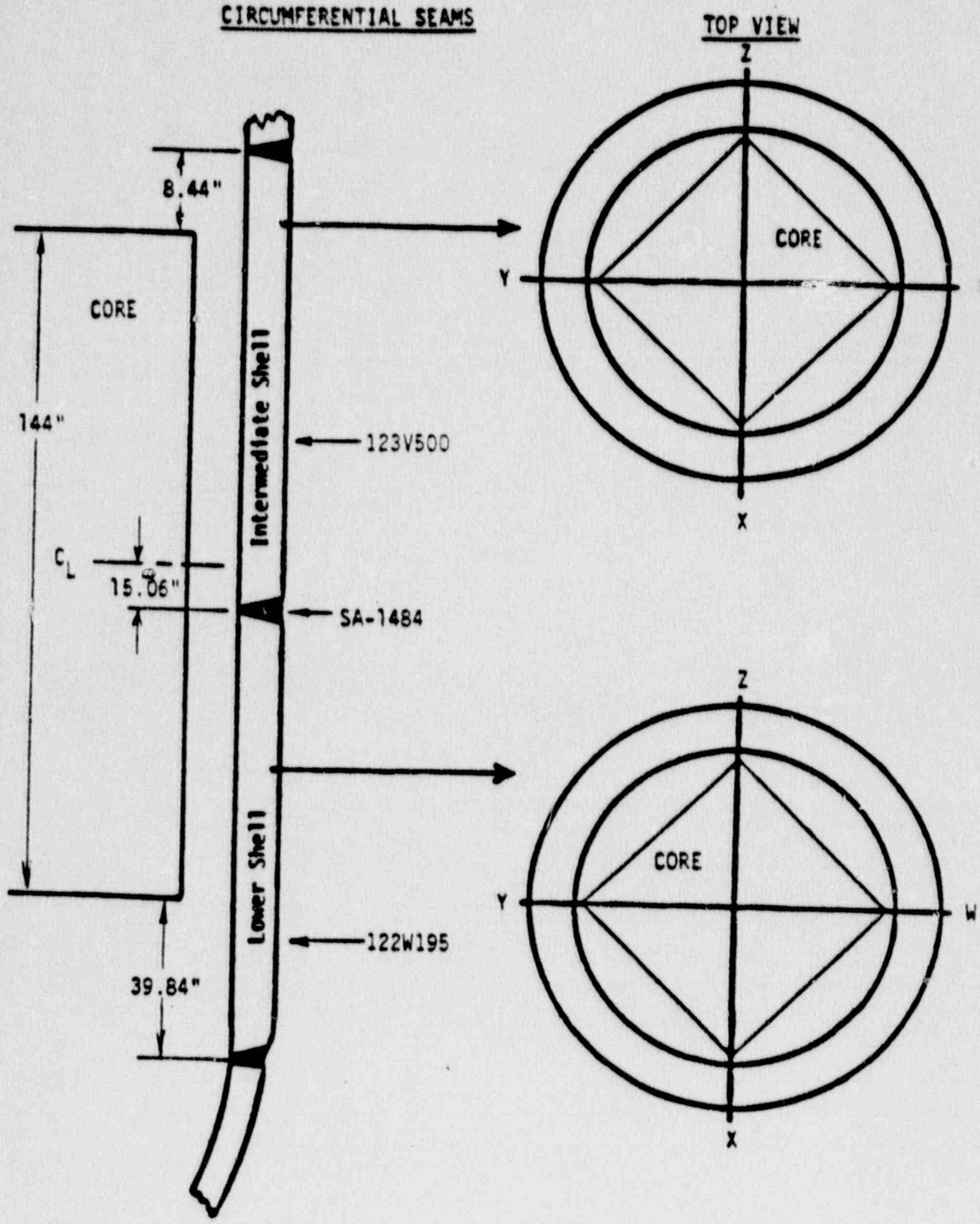


TABLE 1

POINT BEACH UNITS 1 AND 2 REACTOR VESSEL
BELTLINE REGION MATERIAL PROPERTIES

	Chemical Composition(a)		RT_{NDT} RG 1.99 Rev. 2[16]			RT_{PTS}* 10 CFR 50.61[13,14,15	
UNIT 1	Cu* (Wt.%)	Ni* (Wt.%)	Initial RT_{NDT}(°F)(b)	Chemistry Factor	Margin 2σ (°F)	Initial RT_{NDT}(°F)	Margin 2σ(°F)
Intermediate Shell Plate A-9811:[1,2]	0.20	0.056	-25[8,9]	87.6	34(e)	-2	48
Lower Shell Plate C-1423:[2,3]	0.12	0.065	-20[9,10]	54.85	34(e)	-20	48
Axial Weld - Inter. Shell SA-775/812:[4] Weld Wire Heat Nos. 1P0815/1P0661 Linde 80 Flux Lots 8350/8304	0.19(d)	0.63(d)	-6[11]	162.1	68(f)	0(c)	59
Axial Weld - Lower Shell SA-847:[4] Weld Wire Heat No. 61782 Linde 80 Flux Lot 8350	0.25	0.55	-6[11]	169.0	68(f)	0(c)	59
Circumferential Weld - Inter.:[4] to Lower Shell SA-1101 Weld Wire Heat No. 71249 Linde 80 Flux Lot 8350	0.20	0.55	-6[11]	152.25	68(f)	0(c)	59
UNIT 2							
Intermediate Shell Forging 123V500:[5,6]	0.09	0.70	+3[12]	58.0	69(g)	40	48
Lower Shell Forging 122W195:[6,7]	0.05	0.72	+3[12]	31.0	69(g)	40	48
Circumferential Weld - Inter.:[4] to Lower Shell SA-1484 Weld Wire Heat No. 72442 Linde 80 Flux Lot 8579	0.26	0.60	-6[11]	180.0	68(f)	0(c)	59

*The chemical composition and RT_{PTS} parameters are identical to that provided in our January 1986 PTS submittal.

( ) indicate notes; [ ] indicate documentation for Table 1.

NOTES TO TABLE 1

(a) The chemistry values for the shell plates and forgings were derived from vessel material test reports and surveillance capsule chemistry measurements. The chemistry values for welds were derived from searches in the WOG Materials Data Base, Rev. 0 and represent rounded, average values.

(b) Initial RT_{NDT} was determined according to the rules of the ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331. Additionally, where noted, data were treated statistically to obtain the mean value of initial RT_{NDT} and the corresponding standard deviation (σI).

(c) The initial RT_{NDT} values for weld are generic mean values defined by the PTS Rule at 10CFR50.61 (b)(2)ii.

(d) The chemistry data for SA-775 was utilized since this will result in a conservative calculation for this weld.

(e) The plate initial RT_{NDT} value was measured from material-specific drop weight and Charpy data. Hence, σI equals zero (0) as the test methods precisely determined initial RT_{NDT}. The margin added to obtain conservative, upper bound values of adjusted reference is therefore $2\sigma\Delta$ or 34°F.

(f) The statistical evaluation in BAW-1803[11] Table 3-5 concludes that the mean initial reference temperature for Linde 80 welds is -6°F and the standard deviation (σI) about this mean is 19°F. From Reg. Guide 1.99 Rev. 2[16], the standard deviation for ΔRT_{NDT} ($\sigma\Delta$) is 28°F for welds. Thus, the margin applied equals

$$M = 2\sqrt{(19)^2 + (28)^2} = 67.67 = \sim 68°F$$

(g) The statistical evaluation in BAW-1895[12] concludes that the average initial reference temperature for SA 508 CL.2 forgings is +3°F with a standard deviation of 30°F. From Reg. Guide 1.99 Rev. 2 [16], the standard deviation for ΔRT_{NDT} ($\sigma\Delta$) is 17°F for base metal. The margin applied is therefore:

$$\text{Margin} = 2\sqrt{(30)^2 + (17)^2} = 68.8 = \sim 69°F$$

DOCUMENTATION FOR TABLE 1

1. Lukens Steel Company Test Certificate No. RM12965-NS, January 3, 1966 for Babcock and Wilcox Company.

2. WCAP-10736, "Analysis of Capsule T from the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 1 Reactor Vessel Radiation Surveillance Program", December 1984.

3. Lukens Steel Company Test Certificate No. RM61766-BB, January 20, 1966 for Babcock & Wilcox Company.

4. Westinghouse Owner's Group (WOG), "Reactor Vessel Materials Data Base", Revision 0, March 1985.

5. Bethlehem Steel Corporation Test Report No. 911, July 15, 1968 for Babcock & Wilcox Company.

6. WCAP-7712, "Wisconsin Michigan Power Co. and the Wisconsin Electric Power Co. Point Beach Unit No. 2 Reactor Vessel Radiation Surveillance Program", June 1971.

7. Bethleham Steel Corporation Test Report No. 917, July 18, 1968 for Babcock & Wilcox Company.

8. Lukens Steel Company, A-9811 Materials Test Certificate - File No. 602, January 3, 1966.

9. WCAP-7513, "Wisconsin Michigan Power Co. Point Beach Unit No. 1 Reactor Vessel Radiation Surveillance Program", June 1970.

10. Lukens Steel Company, C-1423 Materials Test Certificate - File No. 602, June 20, 1966.

11. BAW-1803, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds", January 1984.

12. BAW-1895, "Pressurized Thermal Shock Evaluation In Accordance With 10CFR50.61 for Babcock and Wilcox Owners Group Reactor Pressure Vessels", January 1986.

13. U.S. NUCLEAR REGULATORY COMMISSION, 10CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", July 23, 1985.

14. Letter from C. W. Fay to H. R. Denton (NRC), "Response to 10CFR50.61 Protection Against Pressurized Thermal Shock (PTS) Point Beach Nuclear Plant, Units 1 and 2", January 20, 1986.

15. Letter from C. W. Fay to H. R. Denton (NRC), "Correction to Pressurized Thermal Shock (PTS) Submittal Dated January 20, 1986 Point Beach Nuclear Plant, Units 1 and 2", March 14, 1986.

16. U.S. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.

JUNE 1989
MFM/ANS

ASSUMPTIONS FOR NEW HEATUP/COOLDOWN CURVES

1. UNIT 2 CIRC WELD LIMITING: Cu = .26 Ni = .60
   (SA-1484) F = 2.05×10^{19} N/cm^2 (1/1/95)

2. ACTUAL FLUX REDUCTIONS WERE NOT TAKEN INTO ACCOUNT. L3P CORE FLUENCE PROJECTIONS WERE EXTRAPOLATED TO 1995.

3. R.G. 1.99 REV. 2 TREND FORMULA USED TO CALCULATE RT_{NDT}.

4. DPA ATTENUATION METHOD OF R.G. 1.99 REV. 2 WAS UTILIZED.

5. WCAPS 8743 (UNIT 1) AND 8738 (UNIT 2) WERE UTILIZED.

6. ADDITIONAL MARGINS FOR INSTRUMENT UNCERTAINTIES WERE NOT ADDED.

7.

UNIT 1	Cu	Ni	FLUENCE ($\times 10^{19}$ N/cm^2)	ART_{ID} (°F)	$ART_{1/4T}$ (°F)	$ART_{3/4T}$ (°F)
SA-847	.25	.55	1.30	243.3	224.9	188.1
SA-1101	.20	.55	2.05	244.1	228.1	195.2
UNIT 2						
SA-1484	.26	.60	2.05	277.3	258.4	219.5

TABLE 3

SIGNIFICANT PARAMETERS FOR
HEATUP AND COOLDOWN LIMIT CURVE REVISION
APPLICABLE TO JANUARY 1, 1995
POINT BEACH NUCLEAR PLANT

Method and Assumptions of Heatup/Cooldown Limit Curves

1. Controlling Welds: Unit 1 SA-847 Lower Shell Axial Weld
Unit 2 SA-1484 Inter.-to-Lower Shell Girth Weld

2. One set of heatup and cooldown curves will be calculated to be applicable to both units at Point Beach for clarity and simplification. From Table 1 the chemistry of Unit 2 weld SA-1484 is more limiting than that of Unit 1 weld SA-847. Additionally, the Unit 1 weld SA-847 is located 15° off the peak fluence (cardinal) axes, and thus, P-T curves calculated for Unit 2 are bounding for Unit 1.

3. Weld Chemistry: SA-1484 Cu = 0.26 wt.%
Ni = 0.60 wt.%

4. Trend and Fluence Attenuation Formulas: Regulatory Guide 1.99, Rev. 2

5. WCAP-10638 was used to extrapolate fluence. L3P cores are assumed to continue through 1995, and no credit for flux reduction measures is taken.

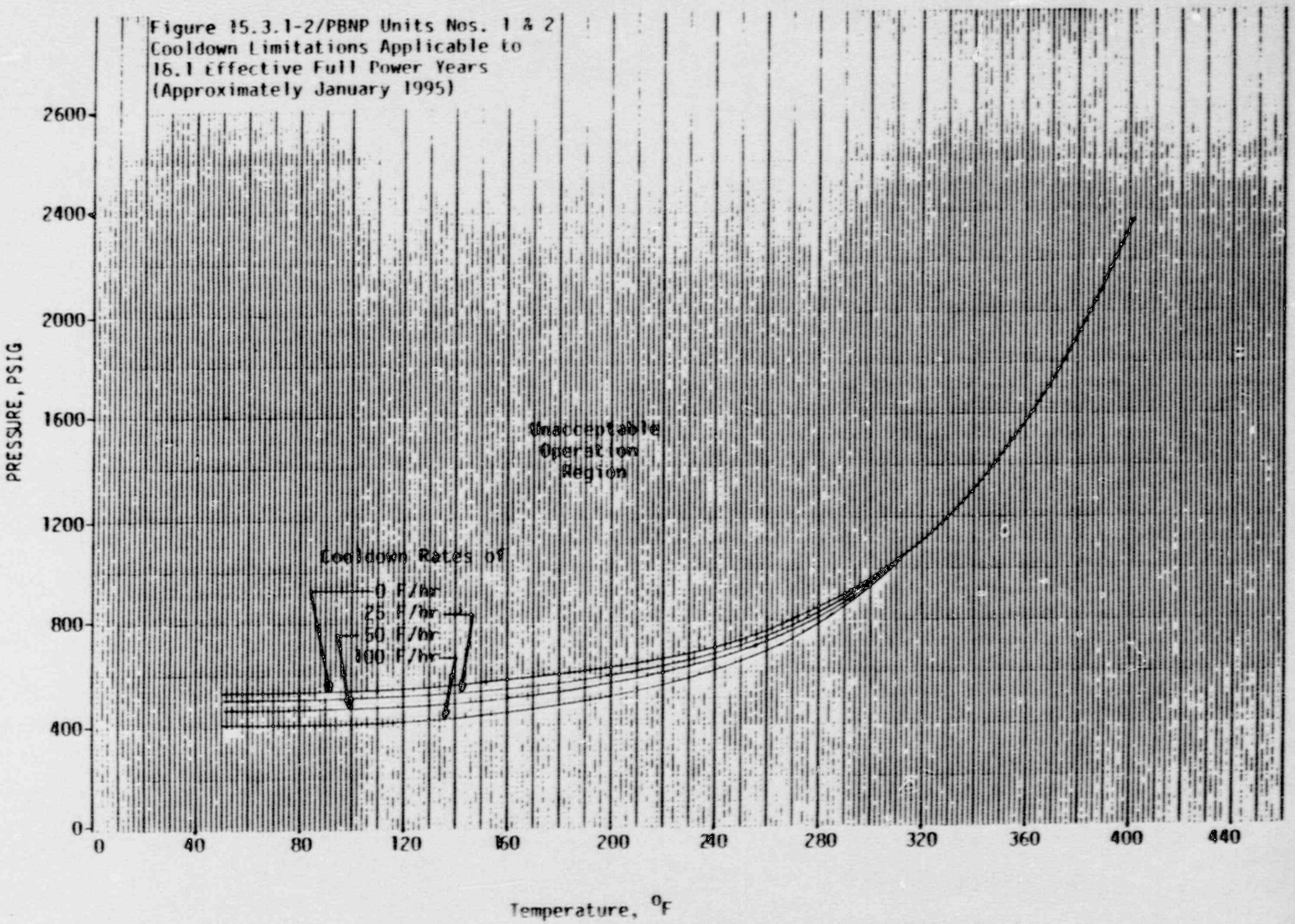
6.

Vessel Location	Chemistry Factor	Fluence ($x10^{19}$ n/cm^2)	Fluence Factor	ΔRT_{NDT}	ART
Wall	180	2.05	1.196	215.3	277.3
1/4 T	180	1.39	1.091	196.4	258.4
3/4 T	180	0.64	0.875	157.5	219.5

7. Heatup/Cooldown Limit Curve Calculational Procedure: WCAP-8738

8. Margins for instrumentation uncertainties are not applied in the calculation of the heatup and cooldown curves. These additional margins are not required by 10CFR50 Appendix G and ASME Section III Appendix G.

Figure 15.3.1-2/PBNP Units Nos. 1 & 2
Cooldown Limitations Applicable to
16.1 Effective Full Power Years
(Approximately January 1995)



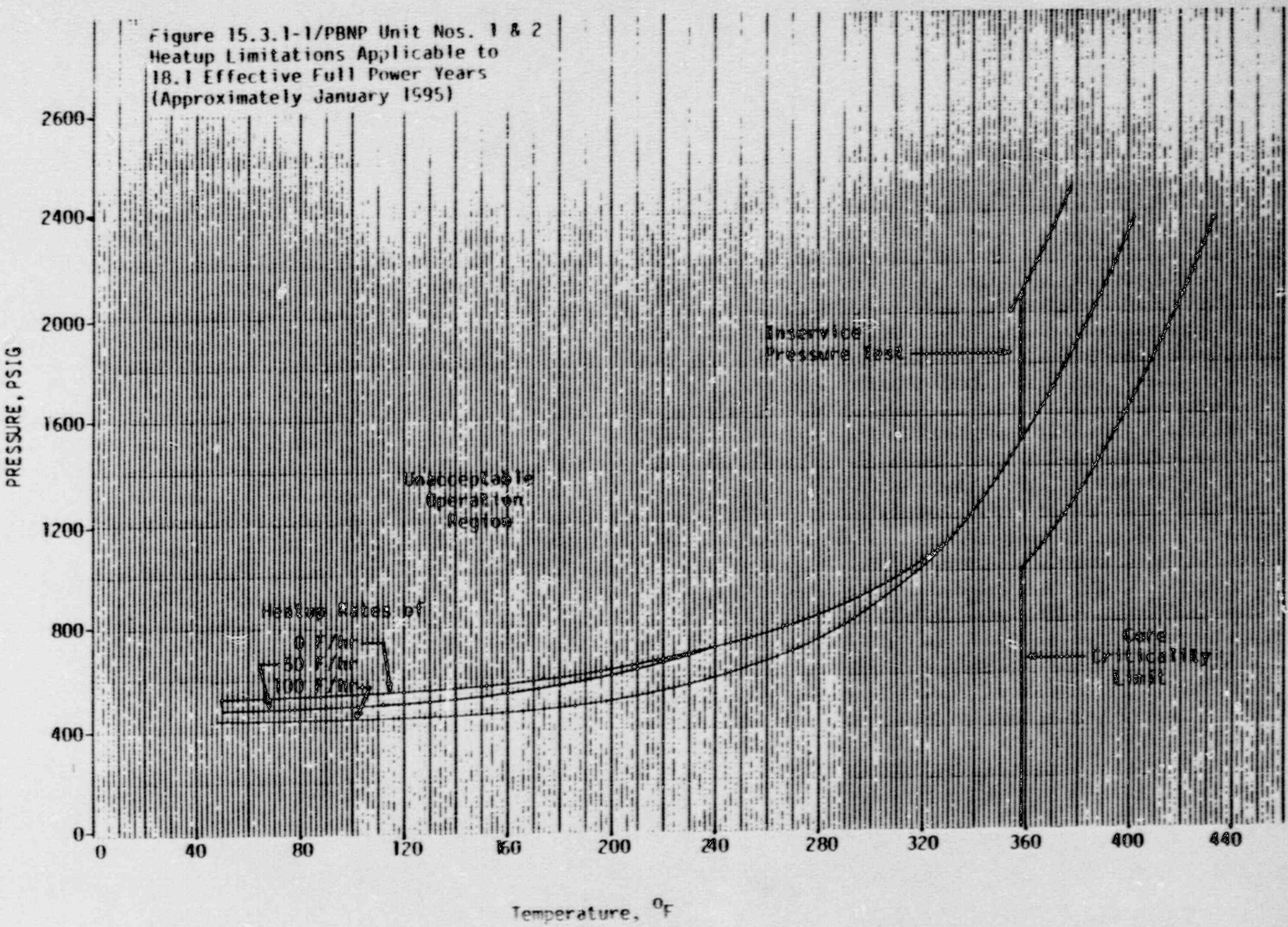


Figure 15.3.1-1/PBNP Unit Nos. 1 & 2
Heatup Limitations Applicable to
18.1 Effective Full Power Years
(Approximately January 1995)
PRESSURE, PSIG
2600
2400
2000
1600
1200
800
400
0
Inservice
Pressure Test
Unacceptable
Operation
Region
Heatup Rates of
0 F/hr
50 F/hr
100 F/hr
Core
Criticality
Limit
0
40
80
120
160
200
240
280
320
360
400
440
Temperature, ^{o}F

TABLE 2

REACTOR VESSEL ID FLUENCE (E > 1 MeV) PROJECTIONS[1]
POINT BEACH UNITS 1 AND 2

	Unit 1[2]			Unit 2[2]	
	EFPY	0°[1]	15°[1]	EFPY	0°[1]
1. Present: 05/31/89	13.6	1.60	1.02	13.6	1.60
2. P-T Curve Applicability Period:					
thru January 1, 1995 (w/L3P cores continued)	18.1	2.05	1.30	18.1	2.05
3. License Expiration:					
Unit 1 - 10/05/2010 Unit 2 - 03/08/2013 (w/L3P cores continued)	30.6	3.30	2.10	32.5	3.50
4. Implement Flux Reductions	13.5	1.59	1.00	13.9	1.63
Unit 1 - April 1989 Unit 2 - October 1989 (w/L4P cores and hafnium inserts)					
5. License Expiration: (See dates in No. 3 above.)	30.6	2.45	1.61[3] 1.79[4]	32.5	2.55
(w/L4P + Hf cores implemented when planned and continued to EOL)					

Notes

(1) Assumes cumulative capacity factor of 80%. Fluence values are $\times 10^{19}$ n/cm².
(2) WCAP 10638 "Adjoint Flux Program for Point Beach Units 1 and 2" dated December 1984, used to project fluence as a function of operating time for L3P cores.
(3) Lower shell axial weld: SA-847
(4) Intermediate shell axial weld: SA-775/812

TABLE 4

COMPARISON FOR PTS RULE PURPOSES
REFERENCE TEMPERATURES FOR REACTOR VESSEL BELTLINE WELD MATERIALS*
POINT BEACH NUCLEAR PLANT
RG 1.99 REV. 2 vs. 10CFR50.61

Unit 1

PERIOD	SA-775/812** Axial Weld Inter. Shell		SA-847** Axial Weld Lower Shell		SA-1101*** Circumferential Weld Inter. to Lower Shell	
	RG 1.99	RTPTS	RG 1.99	RTPTS	RG 1.99	RTPTS
Present (05/31/89)	225.0	180.8	231.9	215.5	234.0	198.1
License Expiration (10/05/2010) w/L3P Cores Continued	256.8	207.1	265.1	249.1	261.9	228.1
License Expiration (10/05/2010) w/Expected Flux Reductions From Planned L4P and Hf Insert Core Designs	250.0	200.8	253.2	236.0	251.0	215.0

Unit 2

PERIOD	SA-1484*** Circumferential Weld Inter. to Lower Shell	
	RG 1.99	RTPTS
Present (05/31/89)	265.4	248.4
License Expiration (03/08/2013) w/L3P Cores Continued	300.8	292.9
License Expiration (03/08/2013) w/Expected Flux Reductions From Planned L4P and Hf Insert Core Designs	287.2	273.8

*Predicted ^{RT}PTS values ass:me a cumulative lifetime capacity factor of 80%. All values are in °F.

**Applicable PTS screening criterion - 270°F.

***Applicable PTS screening criterion - 300°F.

TABLE 15.3.1-2

POINT BEACH NUCLEAR PLANT, UNIT NO. 2
REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL SCHEDULE

Capsule Letter	Approximate Removal Date*
V	November 1974 (actual)
T	March 1977 (actual)
R	April 1979 (actual)
S	Fall 1995
P	Fall 1998
N	Standby

*The actual removal dates will be adjusted to coincide with the closest scheduled plant refueling outage or major reactor plant shutdown.

III

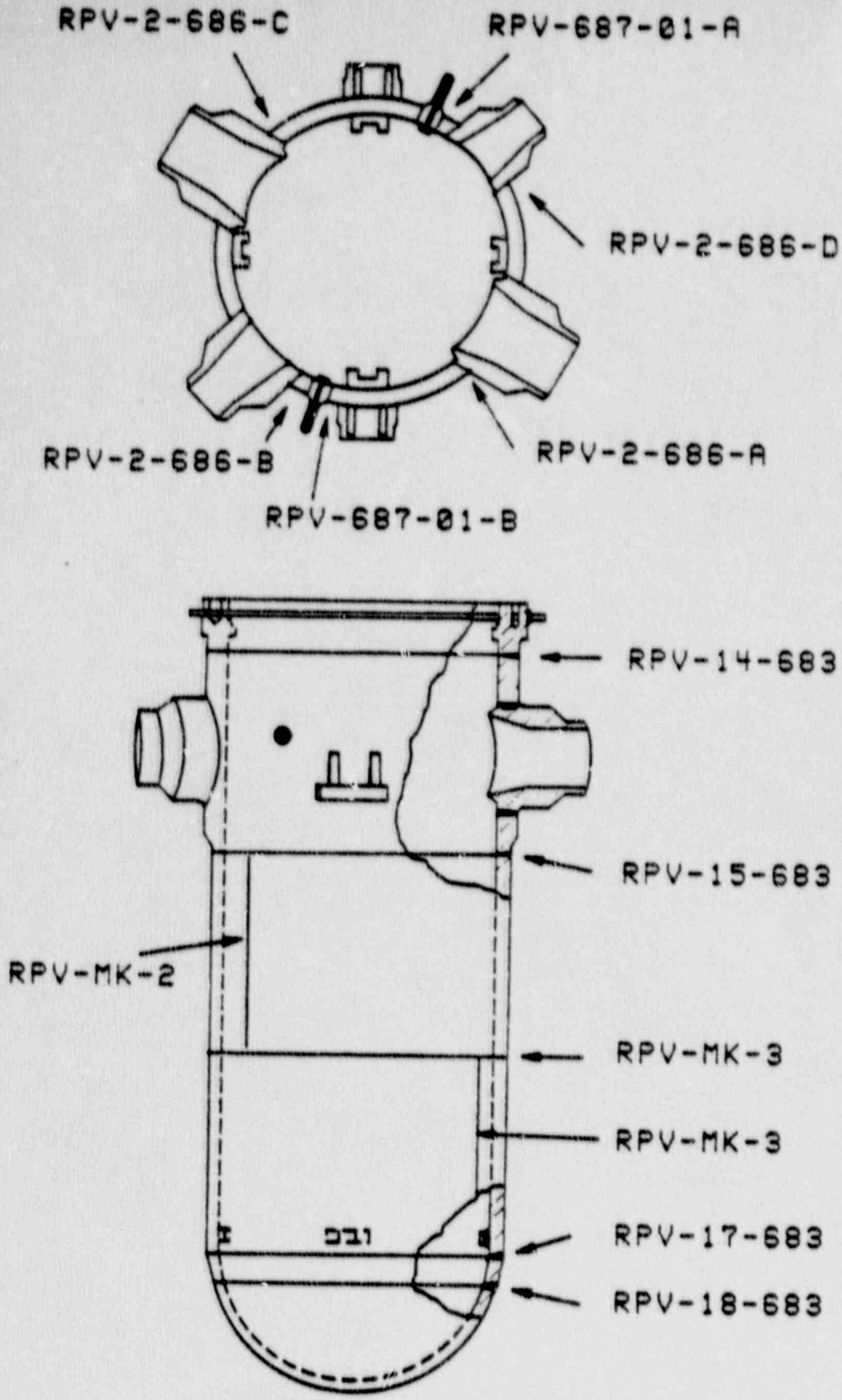
POINT BEACH

NUCLEAR PLANT

REACTOR VESSEL

INSERVICE INSPECTIONS

10 August, 1989



UNIT 1 REACTOR VESSEL

FIGURE A-1

FULL VESSEL EXAMINATION INCLUDES:

PRV-14-683	FLANGE TO UPPER SHELL
RPV-15-683	UPPER SHELL TO MIDDLE SHELL
RPV-MK-2	MIDDLE SHELL LONGITUDINAL WELD
RPV-16-683	MIDDLE SHELL TO LOWER SHELL
RPV-MK-3	LOWER SHELL LONGITUDINAL
RPV-17-683	LOWER SHELL TO LOWER HEAD RING
RPV-18-683	LOWER HEAD RING TO LOWER HEAD
RPV-2-686-A RPV-2-686-C	PRIMARY OUTLET NOZZLE TO SHELL - 'A' AND 'B' LOOPS
RPV-2-686-B RPV-2-686-D	PRIMARY INLET NOZZLE TO SHELL AND INSIDE RADIUS - 'A' AND 'B' LOOPS
RPV-687-01-A RPV-687-01-B	SAFETY INJECTION NOZZLES TO SHELL, INSIDE RADIUS, NOZZLE EXTENSION
LUG F (RPV-MK-16) LUG C (RPV-MK-17)	VESSEL INTEGRALLY WELDED SUPPORTS
# 1-4	CORE BARREL INTEGRALLY WELDED SUPPORTS GUIDES
	VESSEL INTERNALS
	VESSEL INTERNAL CLAD SURFACES

VESSEL EXAMINATION SCAN DIRECTIONS

ALL NOZZLES TO SHELL WELDS SCANNED 360° FROM SHELL WITH 45° AND 60° INCLUDING 1/2 OF BASE METAL.

VESSEL SHELL WELDS SCANNED FULL LENGTH WITH 0°, 45°, 45° T, 60°, 60° T.

INSIDE RADIUS SECTION OF INLET NOZZLES WITH SURFACE WAVE.

INLET NOZZLE TO SHELL ALSO SCANNED FROM NOZZLE BORE W/ 45° AND 15° RL.

SI NOZZLE TO SHELL FROM NOZZLE BORE WITH 0° L, 10° RL.

SI NOZZLE INSIDE RADIUS AND NOZZLE EXTENSION WITH 0° L, 10° RL, SURFACE WAVE.

WELDED SUPPORTS WITH 0°, 45°, 45° T, 60° FROM VESSEL INSIDE SURFACE.

SURFACES, CLAD PATCHES, INTERNALS, ETC. WITH REMOTE VISUAL.

VESSEL TO FLANGE WELD FROM SEAL SURFACE EXAMINED WITH 5° RL, 11° RL, 18° RL.

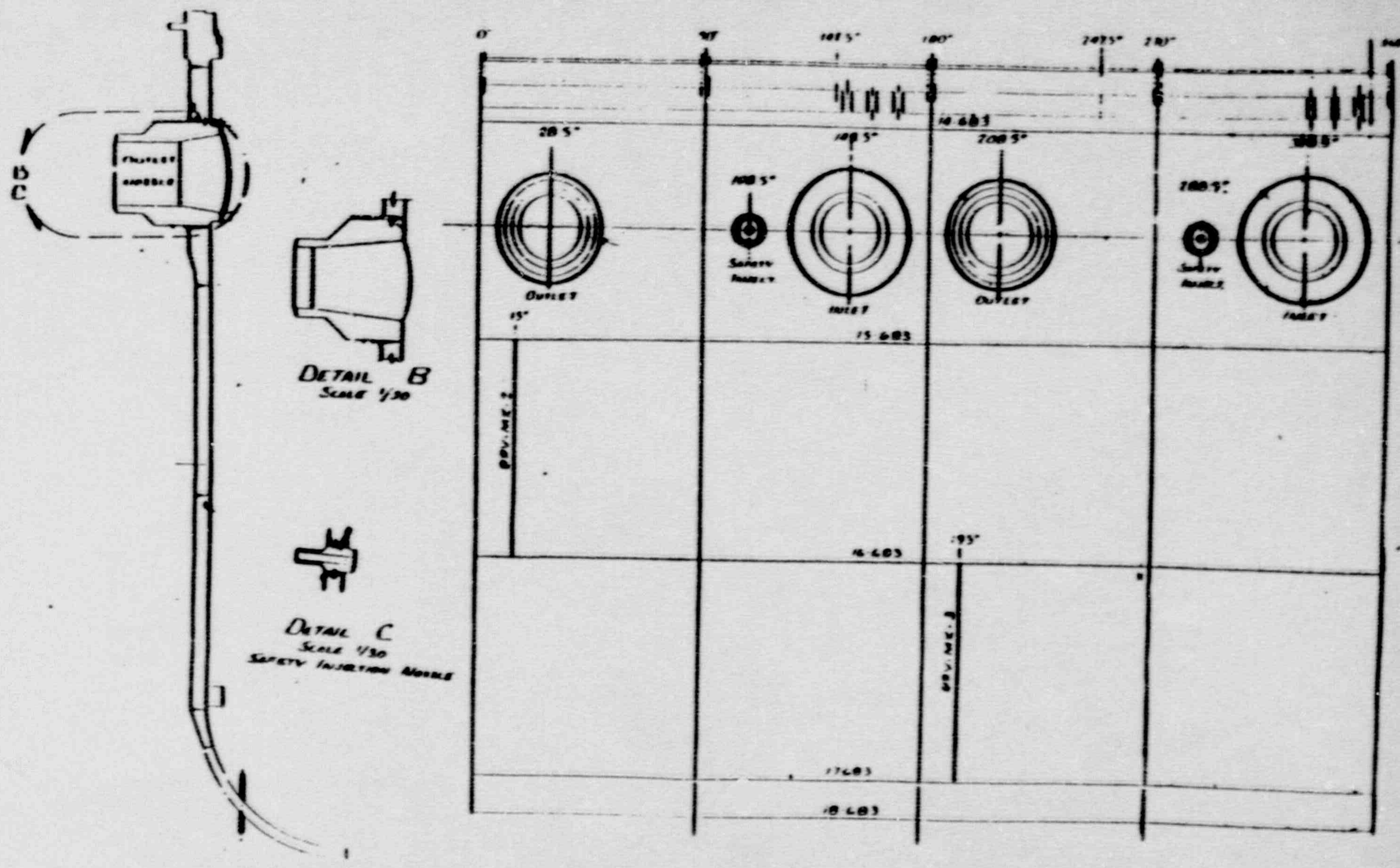
UNIT 1

EX INATION DATE	EXTENT OF VESSEL EXAM	CODE/STANDARD UTILIZED	RESULTS	NOTES
OCT-NOV 1976 (U1R4)	FULL VESSEL	APPENDIX I OF ASME SECT. XI 74/S 76	74 CODE ACCEPTABLE INDICATIONS IN RPV-2-686-A, RPV-2-686-C, RPV-2-686-B, RPV-2-686-D RPV-2-687-01B, RPV-NK-2 RPV-16-683 ALL SPOT REFLECTORS; NOT SIZED!	AUGMENTED EXAM TO INCLUDE INLET/OUTLET PIPING WELDS
OCT-DEC 1981 (U1R9)	1/3 VESSEL TO FLANGE OUTLET NOZZLES TO SHELL INCLUDING IRS AND NOZZLE TO PIPE WELDS (SAFE-ENDS)	ASME SECT XI 74/S75 AND REG GUIDE 1.150 DATED JUNE 1981	3 REFLECTOR SIZED AS ALLOWABLE IN RPV-2-686-A PER ASME SECT. XI 74/S75 AND REG. GUIDE 1.150 DATED JUNE 1981	
FEB-MAR 1984 (U1R11)	100% VESSEL TO FLANGE OUTLET NOZZLE TO SHELL INCLUDES IRS AND EXTENSION. INLET NOZZLE TO SHELL INCLUDES IRS. OUTLET NOZZLE TO PIPE WELDS	ASME SECTION XI 77/S79 REG. GUIDE 1.150 REV. 1 DATED FEB 1983	4 REFLECTORS IN RPV-2-686-A 7 REFLECTORS IN RPV-2-686-C 1 IN EACH NOZZLE EXCEEDED IWB-3512-1 AND EVALUATED FOR CONTINUED OPERATION BY FRACTURE MECHANICS PER IWB-3600	NOZZLES EXAMINED ONLY FROM THE BORE.
APRIL 1987 (U1R14)	FULL VESSEL EXCEPT WELDS RPV-15-683, RPV-NK-2 RPV-18-683	ASME SECT. XI 77/S79 REG. GUIDE 1.150 REV. 1 DATED FEB 1983	1) SIZES OF OUTLET NOZZLE FLOWS FROM 84 REMAINED UNCHANGED. 2) 1 REFLECTOR IN RPV-687-01-B WITHIN CODE ALLOWABLE SIZE. 8 REFLECTORS IN RPV-687-01-A 7 CODE ALLOWABLE; 1 REQUIRED FRACTURE MECHANICS AND SIZE CORRECTION FOR XDUCER BEAM SPREAD PER RG. 1.150 AND IWB-3600	UDRPS ALSO UTILIZED FOR DATA ANALYSIS AND RECORDING. NEAR SURFACE SUPPLEMENTED WITH 50/70 XDUCER

UNIT 2

EXAMINATION DATE	EXTENT OF VESSEL EXAM	CODE/STANDARD UTILIZED	RESULTS	NOTES
1971	FULL VESSEL	ASME SECT. III APPENDIX IX	-SPOT REFLECTOR IN LOWER SHELL TO HEAD RING, NO MEASURABLE SIZE. -SPOT REFLECTOR IN HEAD RING TO BOTTOM HEAD WELD, NO SIZE. -SPOT REFLECTOR IN OUTLET AND INLET NOZZLE TO SHELL WELD.	FIRST USE OF PAR DEVICE FOR VESSEL EXAM.
MARCH/APRIL 1979 (U1R3)	FULL VESSEL	TECH. SPEC. 15.4.2 (APPENDIX I) AND ASME SECT. XI 71/S71	7 CODE ALLOWABLE REFLECTORS IN WELDS RPV-15-683, RPV-2-686-01-A REFLECTOR IN RPV-15-683 WERE BASE METAL. EVALUATION AND SIZE WAS TO ASME SECTION XI 74/S76	SCAN SENSITIVITY WAS GREATER THAN AT PREOPERATIONAL EXAM.

Reactor Vessel Rollout



Some of the significant changes since the early 1970's

Unit 1 - 1976
Unit 2 - 1977

- Recognized ASME Section XI '74/5, '76 for the control of the examination.

- Scans were planned to include at least 1/2 T of base material.

- Calibration blocks were clad to more closely represent the actual materials being examined. This also resulted in increased scanning sensitivities.

- Transducer beam profiles were determined and considered for flaw size determinations.

- Nominal 60° transducers were added to the scan plans to improve the detection probability.

- Transverse scans of the base material in addition to the weld volume.

- Personnel met the qualification requirements of SNT-TC-1A (1975 edition).

- Significant improvements were made in the UT equipment and recording of data. All real time scans were recorded on video tape for future use.

Unit 1 - 1981 - Reg. Guide 1.150 was implemented at PBNP.

Unit 1 - 1987 - Additional scans were performed using 50/70 transducers for the near surface. These scans were calibrated on 1/16" diameter side drilled holes at depths from 1/4" to 2".

- Flaw recording level was set at 20% of DAC.

- Flaw sizing approach included a comparison of the flaw signal response to the response from planar reflectors of known area. The method for sizing included transducer energy profiles.

- NES/Dynacon's UDRPS was utilized to record scan data and aid in the analysis. UDRPS techniques are based on target motion and flaw tip diffraction signals. This provided a record of flaw distribution without consideration of an amplitude or size threshold.

EPRI was involved with PBNP during the 1987 examination to assist with UDRPS flaw detection and sizing. The report of the UDRPS analysis identified various 'SPOT' reflectors. These are reflectors that have essentially no dimensions.

The extent of required volume to be examined in 1987 was as follows:

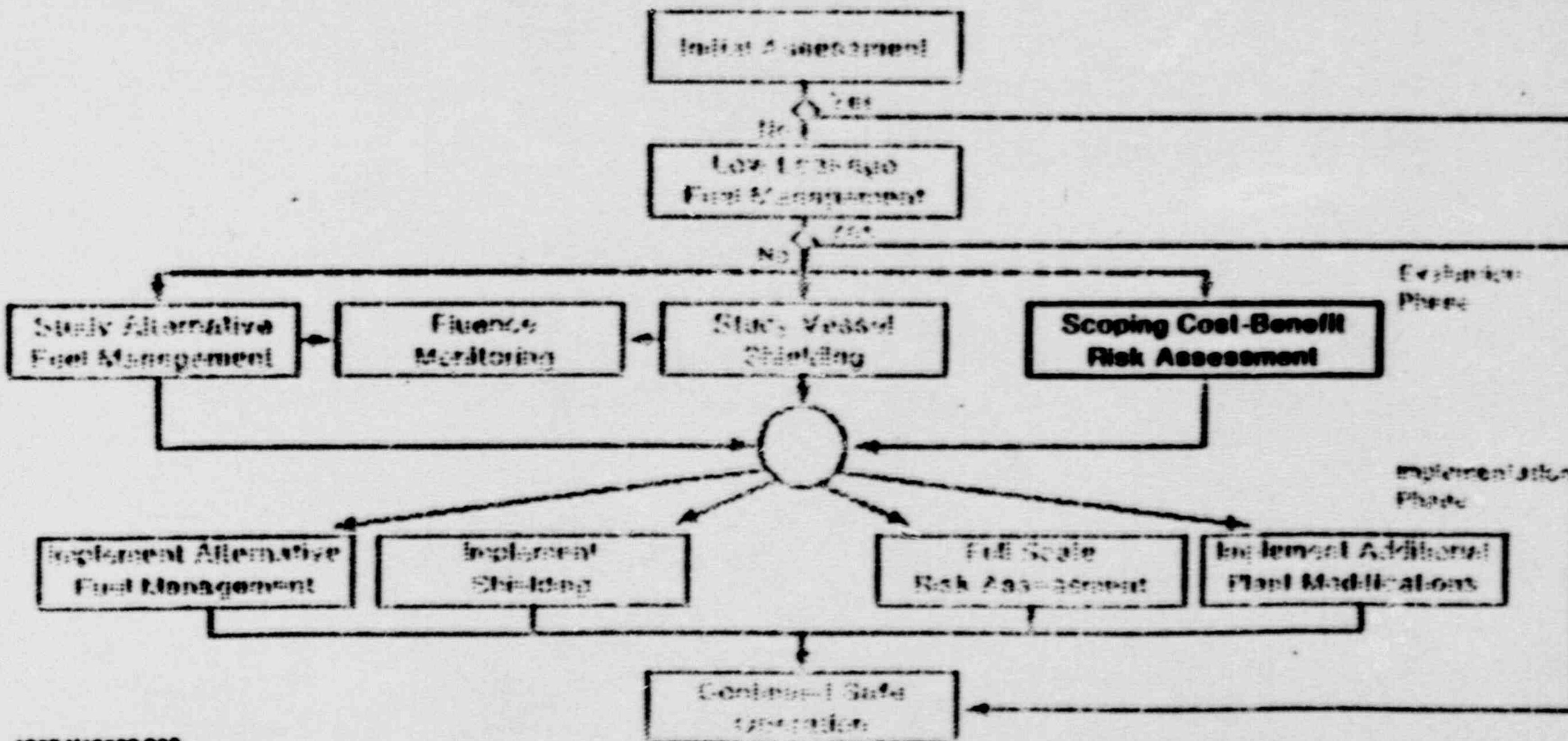
RPV-14-683	100%
RPV-16-683	100%
RPV-17-683	80%
RPV-MK-3	95%

Lower shell to lower head ring scans are limited by the core barrel support lugs and transducer lift-off caused by the weld transition angles. The shell seam is limited by one core barrel support lug located at 180°.

IV

Reactor Vessel Life Extension

A Cost-Effective Approach



1029 N10230 038

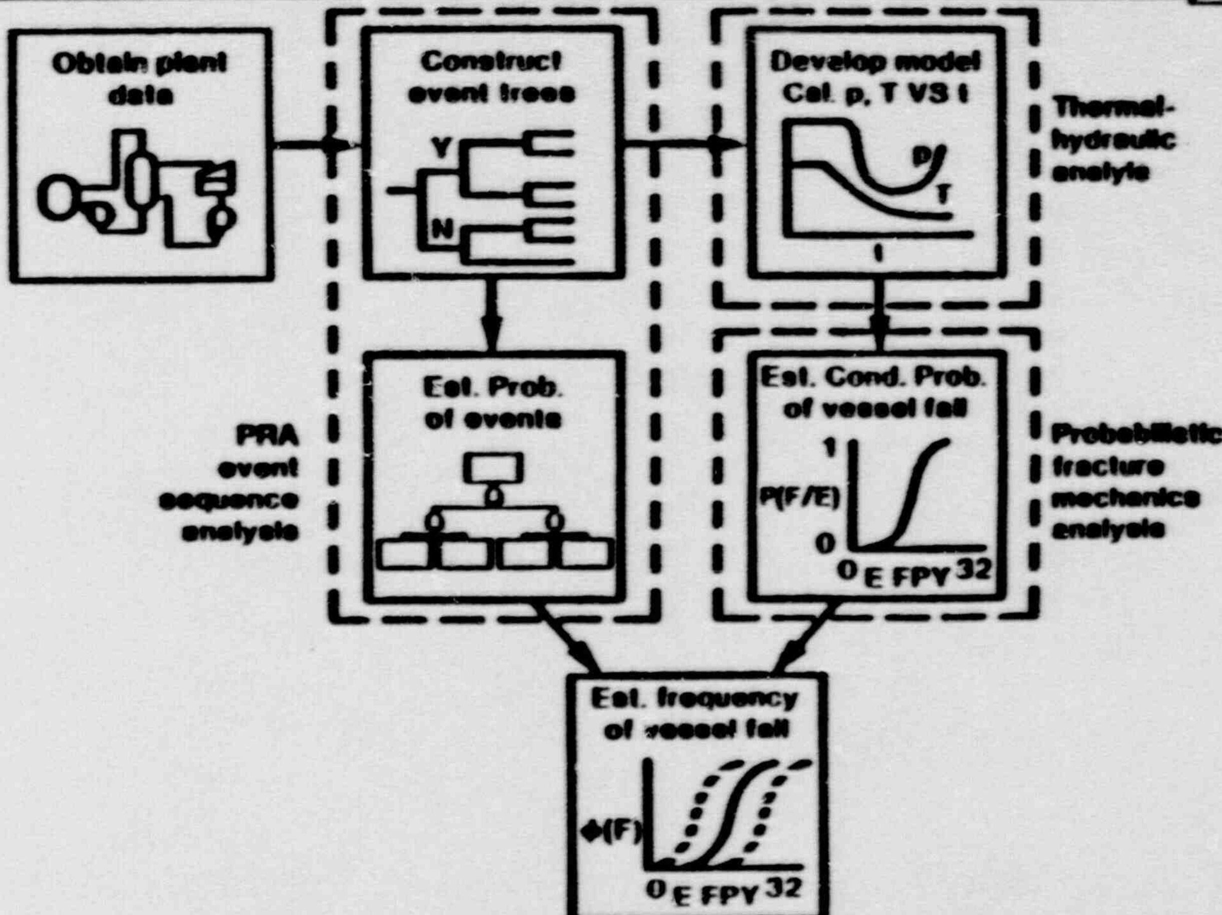
Point Beach Nuclear Plant
Scoping Cost-Benefit Risk Assessment

Uses

- Demonstrate that prior design basis accident analyses (large LOCA & large SLB) results are insignificant contributors to vessel failure risk
- Check whether initial flux reduction goals were appropriate
- Proactively address low upper-shelf fracture toughness issue
- Proactively address the effect of R.G. 1.99 Rev. 2 on the PTS screening criteria
- Define plant specific RT_{PTS} screening criteria
- Vessel integrity and PTS decision-making for contemplated plant modifications (narrow scope - high resolution risk analysis only)

1187 023433 005

Flow Chart to Evaluate the Risk of Reactor Vessel Failure



1020 N10230 020

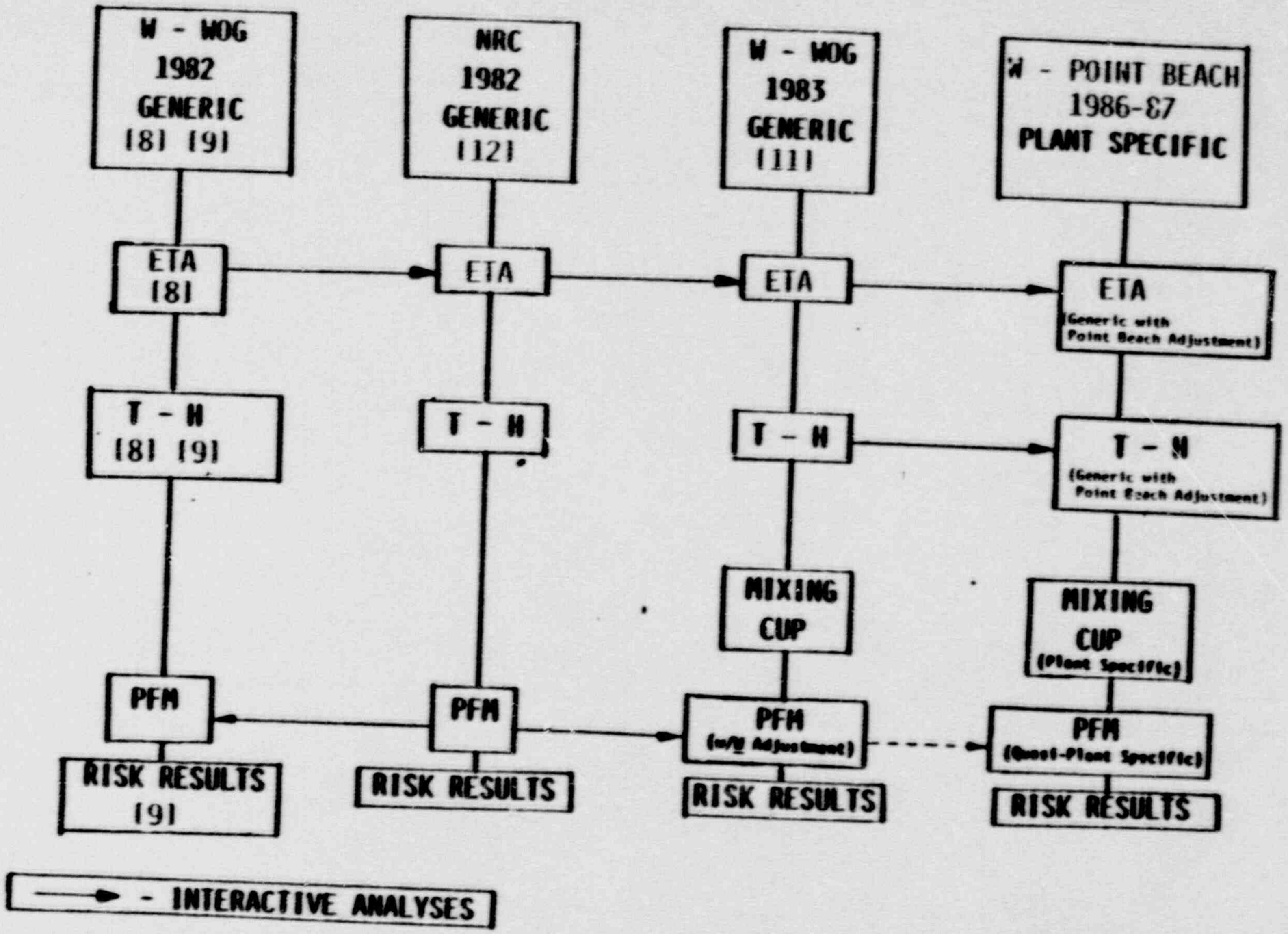
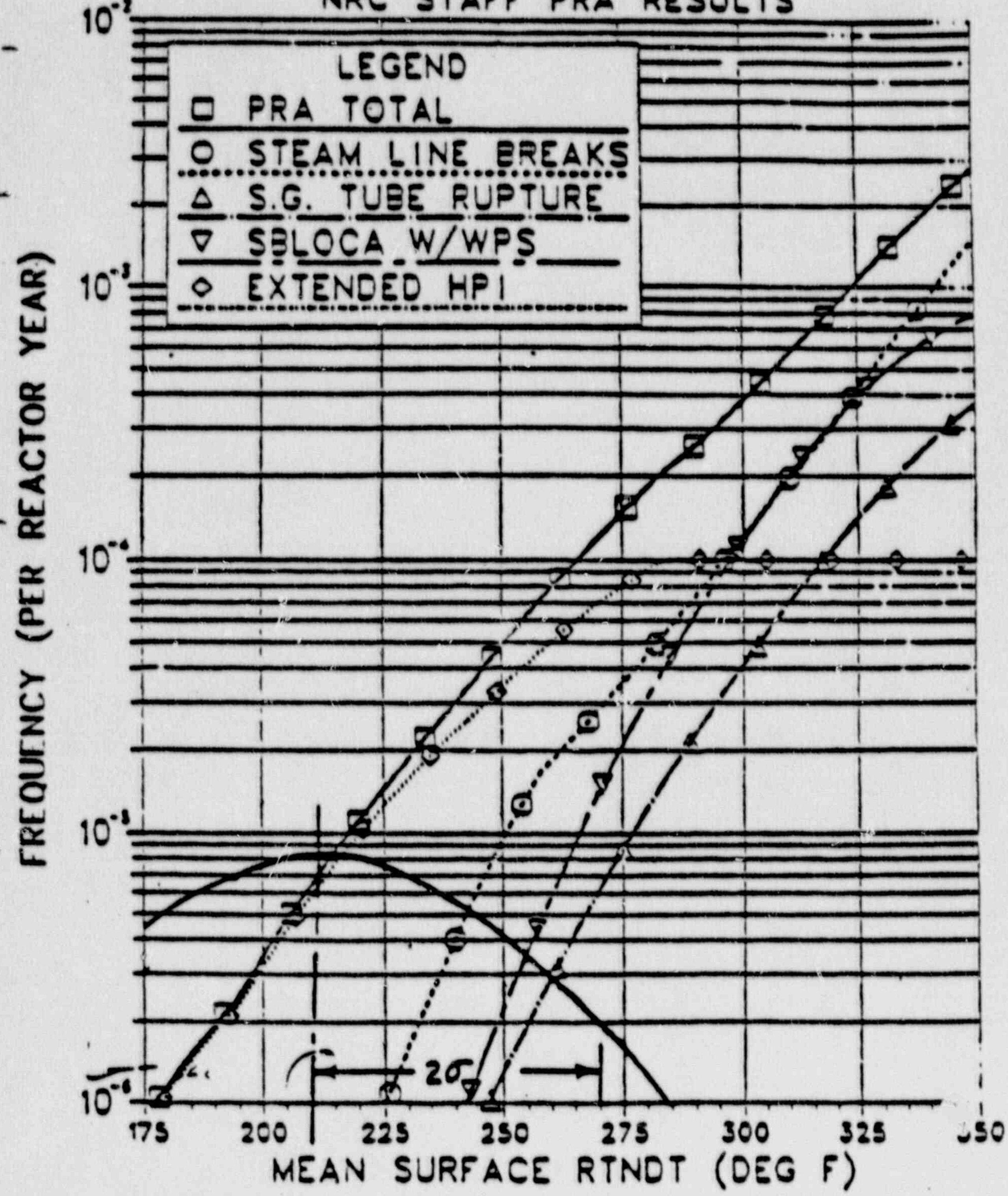
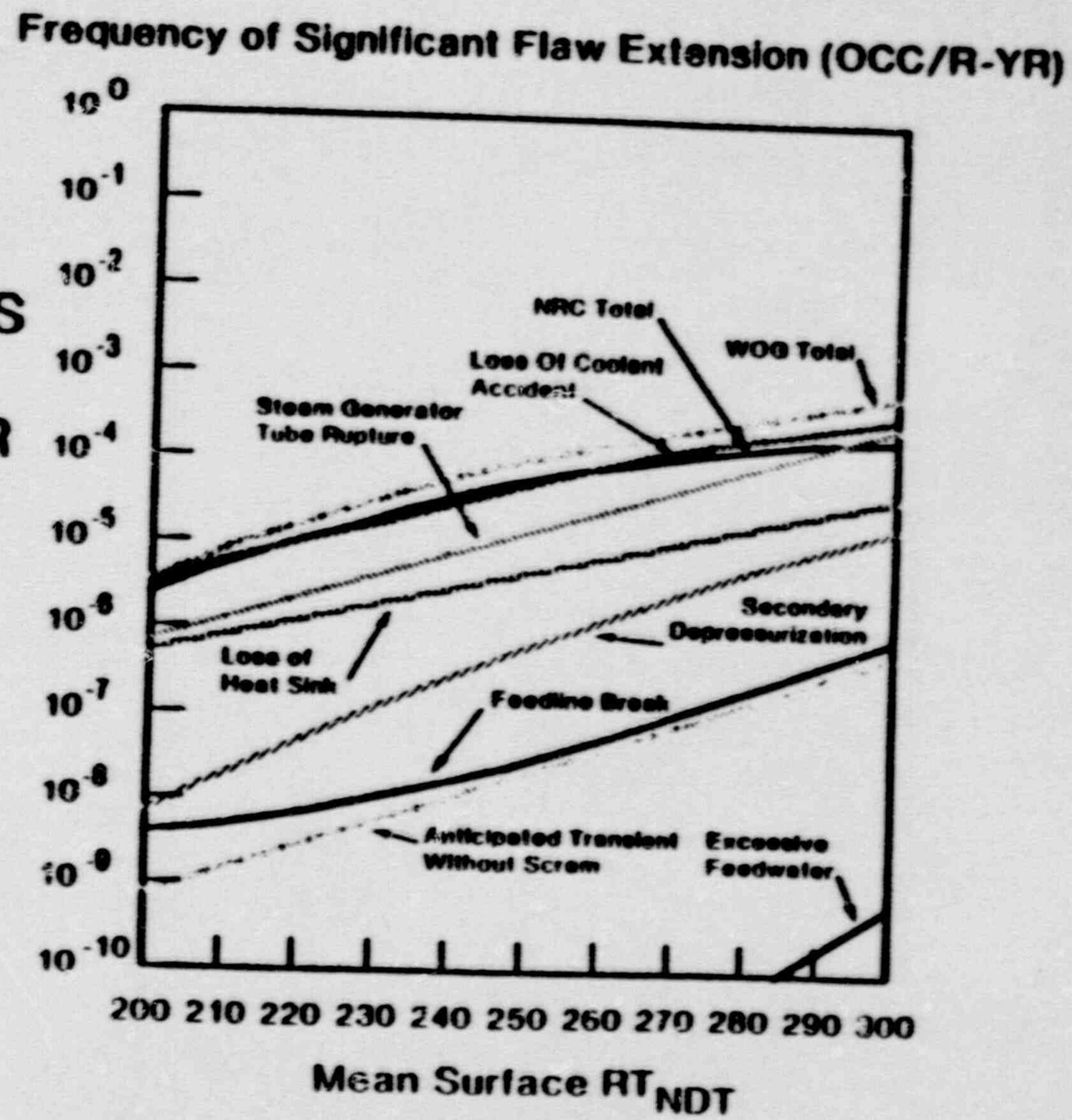


Figure III.a-1: RELATIONSHIP OF POINT BEACH RISK STUDY TO GENERIC PTS EVALUATIONS

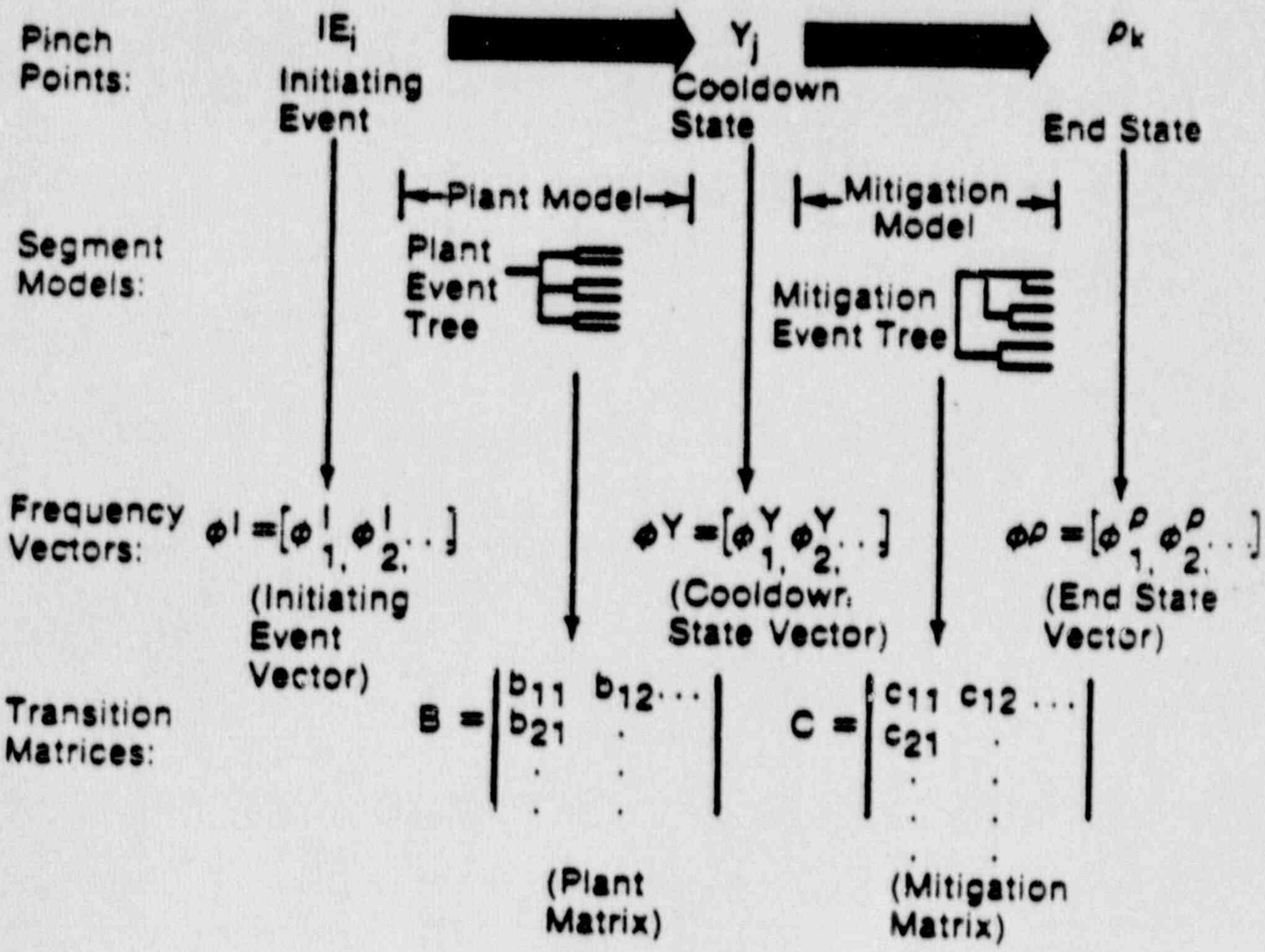
LONGITUDINAL CRACK EXTENSION NO ARREST
NRC STAFF PRA RESULTS



WOG RESULTS FOR LONGITUDINAL WELDS IN A TYPICAL WESTINGHOUSE PWR



OVERVIEW OF THE ASSEMBLY PROCESS, SHOWING RELATIONSHIP OF PINCH POINTS, FREQUENCY VECTORS, EVENT TREES, AND TRANSITION MATRICES



Assembly Process:

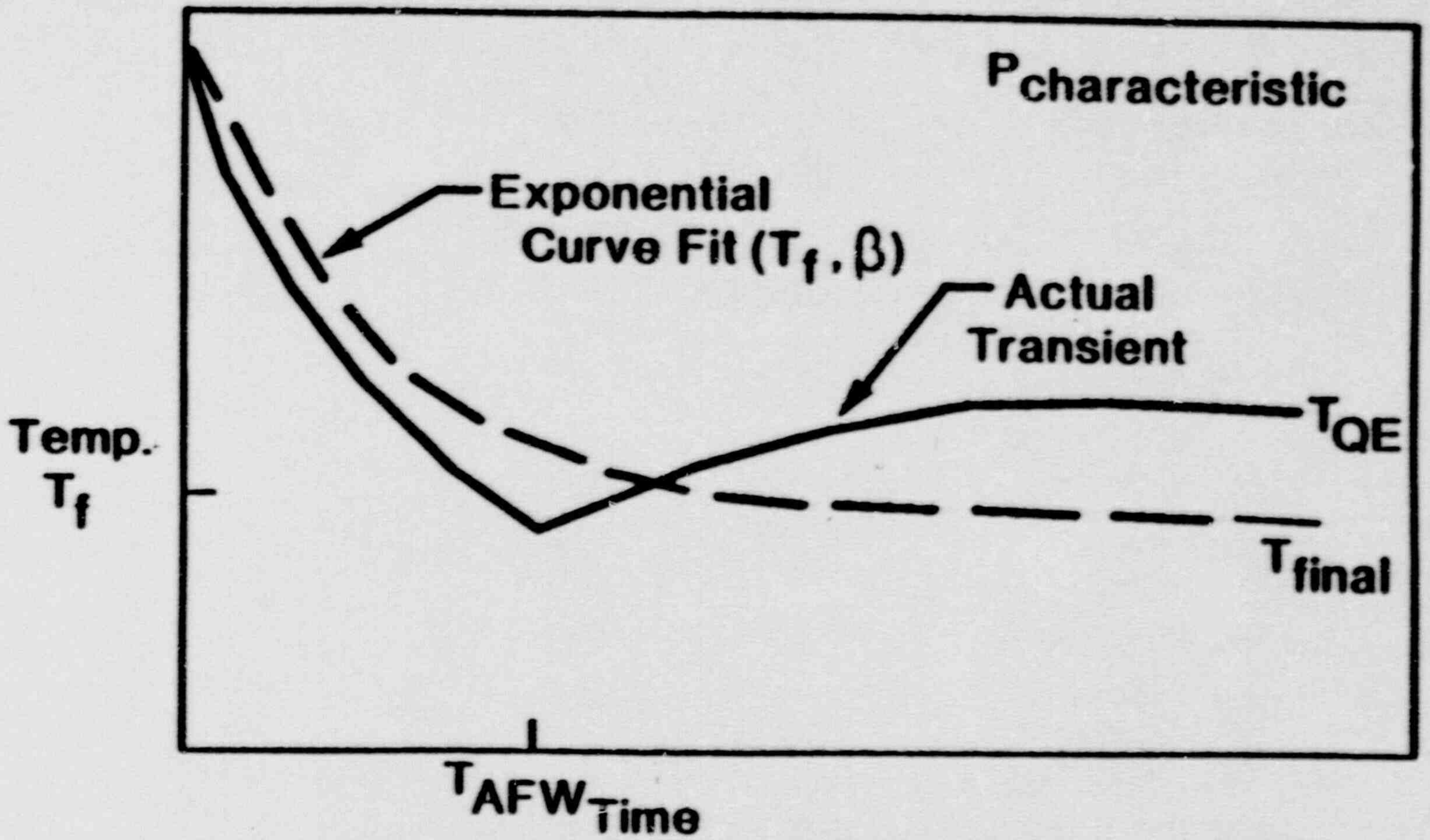
$$\phi^Y = \phi^I B$$

$$\phi^\rho = \phi^Y C = \phi^I BC$$

TABLE III.b.4-1

SUMMARY OF INITIATING EVENT FREQUENCIES

Event	Point Beach
1. Loss of Main Feedwater (LOFW)	2.50E+00
2. Closure of one MSIV	1.70E-01
3. Loss of Primary Flow (LOPF)	3.10E-01
4. Core Power Increase (POWIN)	1.20E-02
5. Turbine Trip (TT)	1.89E+00
6. Spurious Safety Inj. Activation (SSI)	5.90E-02
7. Reactor Trip (RT)	2.82+00
8. Turbine Trip due to Loss of Offsite Power (TT/LOOP)	1.50E-01
9. Steam Generator Tube Rupture (SGTR)	3.90E-02
10. Small LOCA, <1.5 in. dia. (LOCA-1)	5.50E-02
11. Small LOCA, 1.5-6.0 in. dia (LOCA-2)	6.11E-04
12. Large LOCA, >6 in. dia. (LOCA-3)	1.00E-07
13. Excessive Main Feedwater (EX FW)	2.10E-01
14. Steamline Rupture Inside Containment (STM BRK IN)	6.00E-03
15. Steamline Rupture Outside Containment (STM BRK OUT)	9.20E-02
TOTAL	8.32E+00



T_{AFW} = Time to terminate uncontrolled auxiliary feedwater flow to the steam generators

T_{QE} = Final Quasi-Equilibrium temperature

Reactor Vessel Life Extension

Significant design differences from WOG Generic Results

- Frequency of secondary power operated relief valves to close
- Safety injection system characteristics
- Safety injection/reactor coolant loop thermal mixing volume
- Reactor Vessel
  - Geometry
  - Circumferential weld is limiting location
  - Low upper shelf fracture toughness

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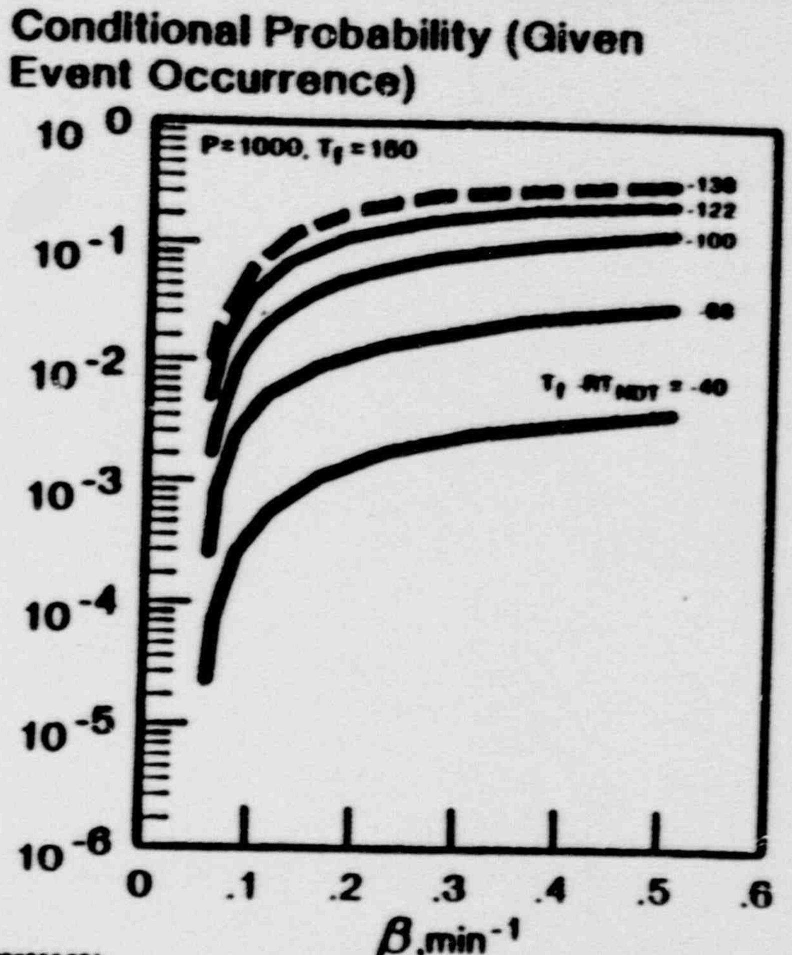
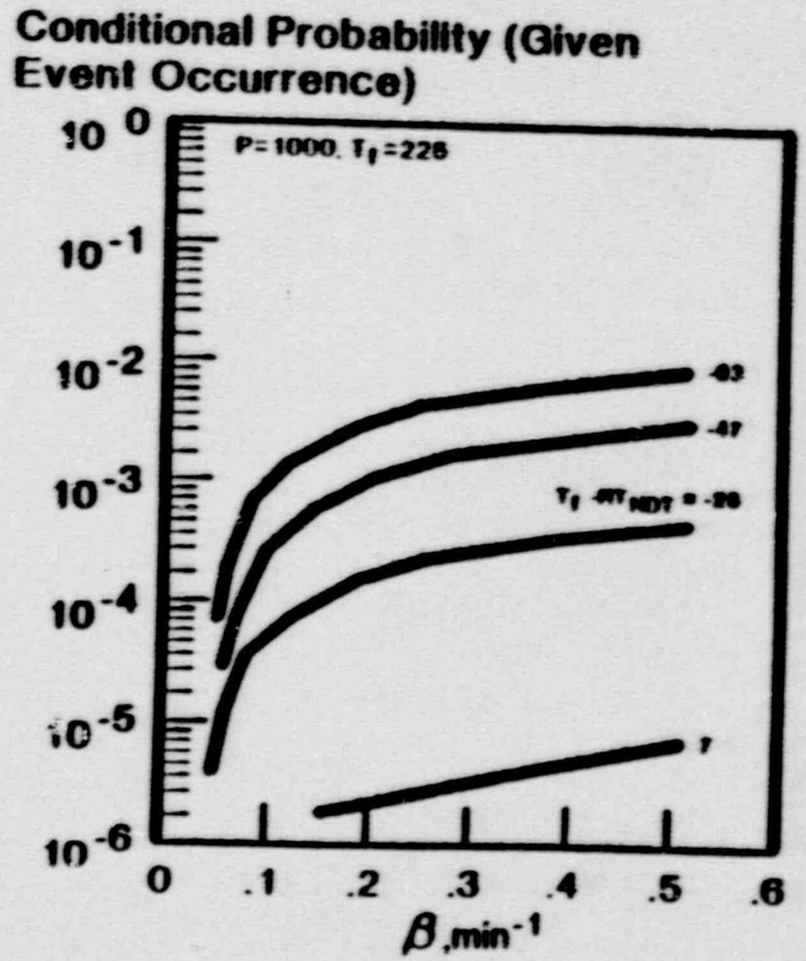
CONDITIONAL PROBABILITY OF SIGNIFICANT FLAW EXTENSION FOR A SINGLE LONGITUDINAL BELTLINE WELD (SECY - 82 - 465)





DETERMINATION OF CRACK INITIATION AND ARREST

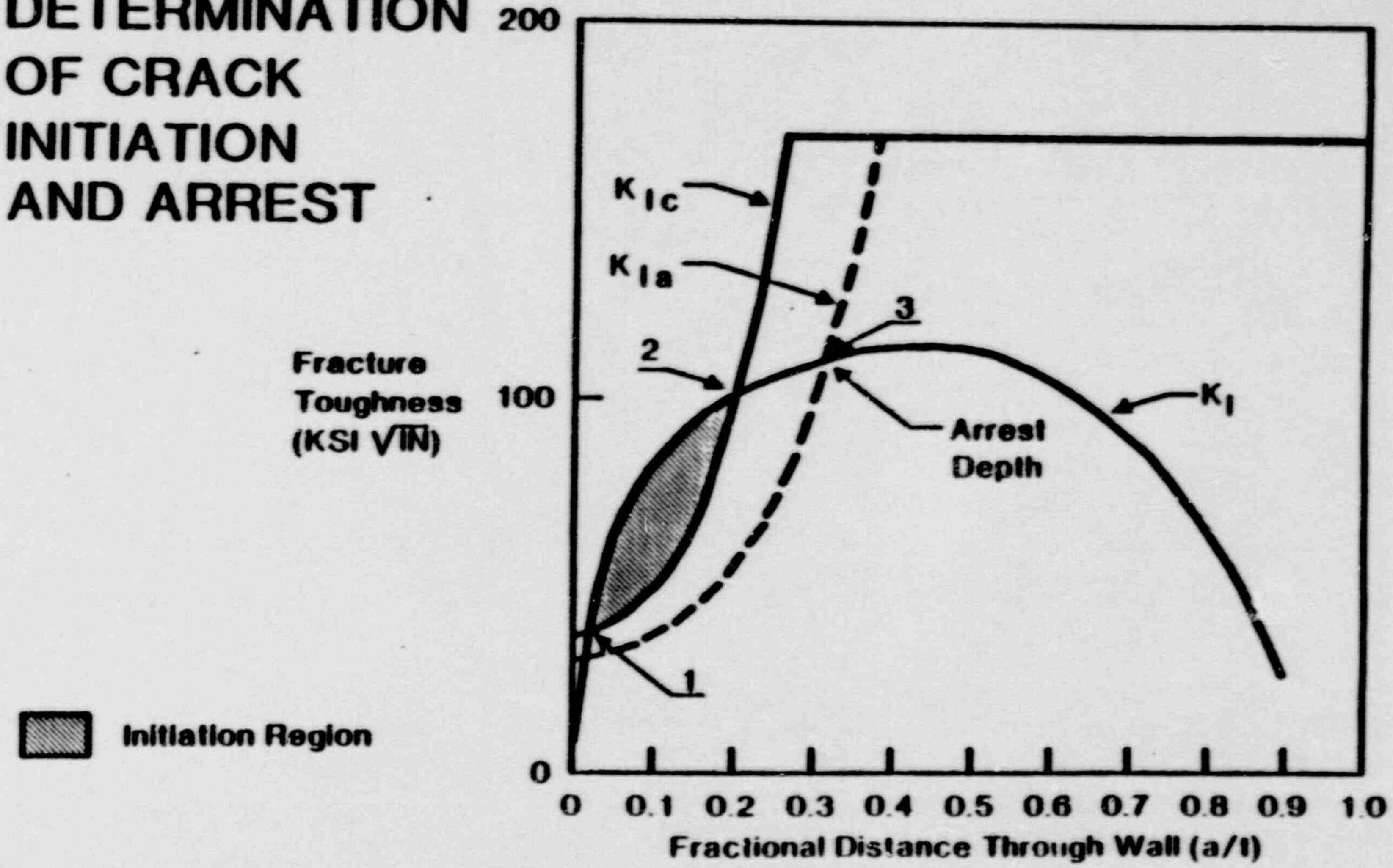


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Table III.d.3-3

POINT BEACH REFINEMENTS BEYOND NRC RESULTS

TRANSIENT: $T_f = 150^{\circ}F$, $\beta = 0.15/MIN$, $P = 1{,}000$ PSI

CASE	VESSEL MODEL	$\bar{h}(t)$	FLAW ORIENT.	TREND CURVE	UPPER SHELF	FAILURE CRITERION	FLUENCE ATTENUATION	FLAW SHAPE	FLAW SIZE DIST.
A	NRC	$\bar{h} = 320$	LONG	GUTH-REV 0	200 TR	ARREST $\geq$ 1.0a/t	NRC-NO dPa	CONT.	OCTAVIA (PTS)
B	POINT BEACH								
C1		(W) FREE				ARREST $\geq$ 0.75a/t	POINT BEACH W/ dPa	2-FLAW	
C2			CIRC						
D1			LONG	RG 1.99-REV 2					
D2			CIRC						
E1			LONG	GUTH-REV 0	200 TR; 75→200 DISTRIB.				OCTAVIA (Post PTS)
E2			CIRC						

Figure III.d.3-4

PFM CONDITIONAL PROBABILITIES OF FAILURE
BENCHMARKING NRC,HBR,AND PBNP PROBABALISTIC RESULTS

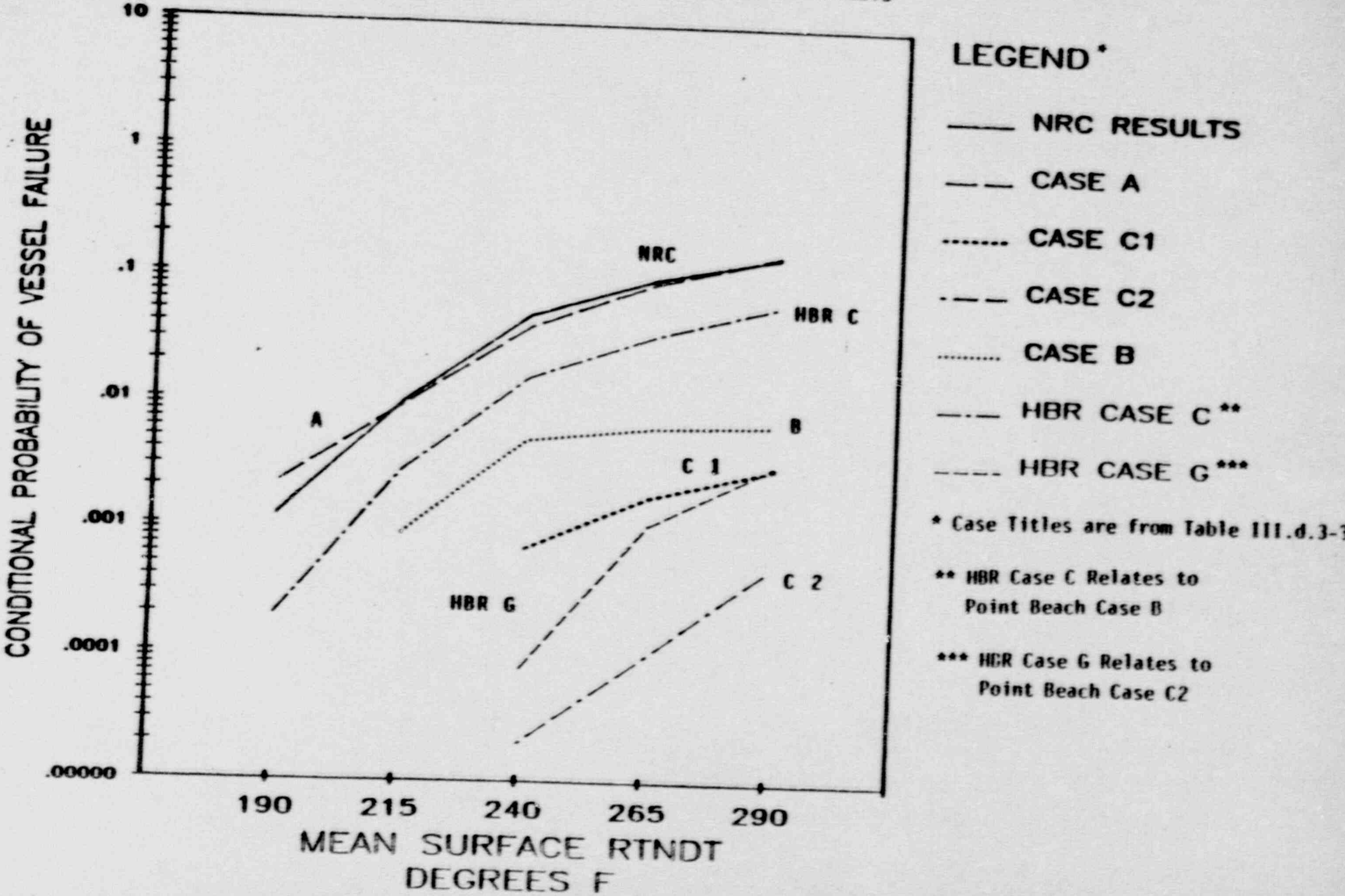


Figure III.d.3-5

PFM CONDITIONAL PROBABILITIES OF FAILURE

SENSITIVITY STUDY OF GUTHRIE VERSUS R.G. 1.99 REV 2 TREND CURVES

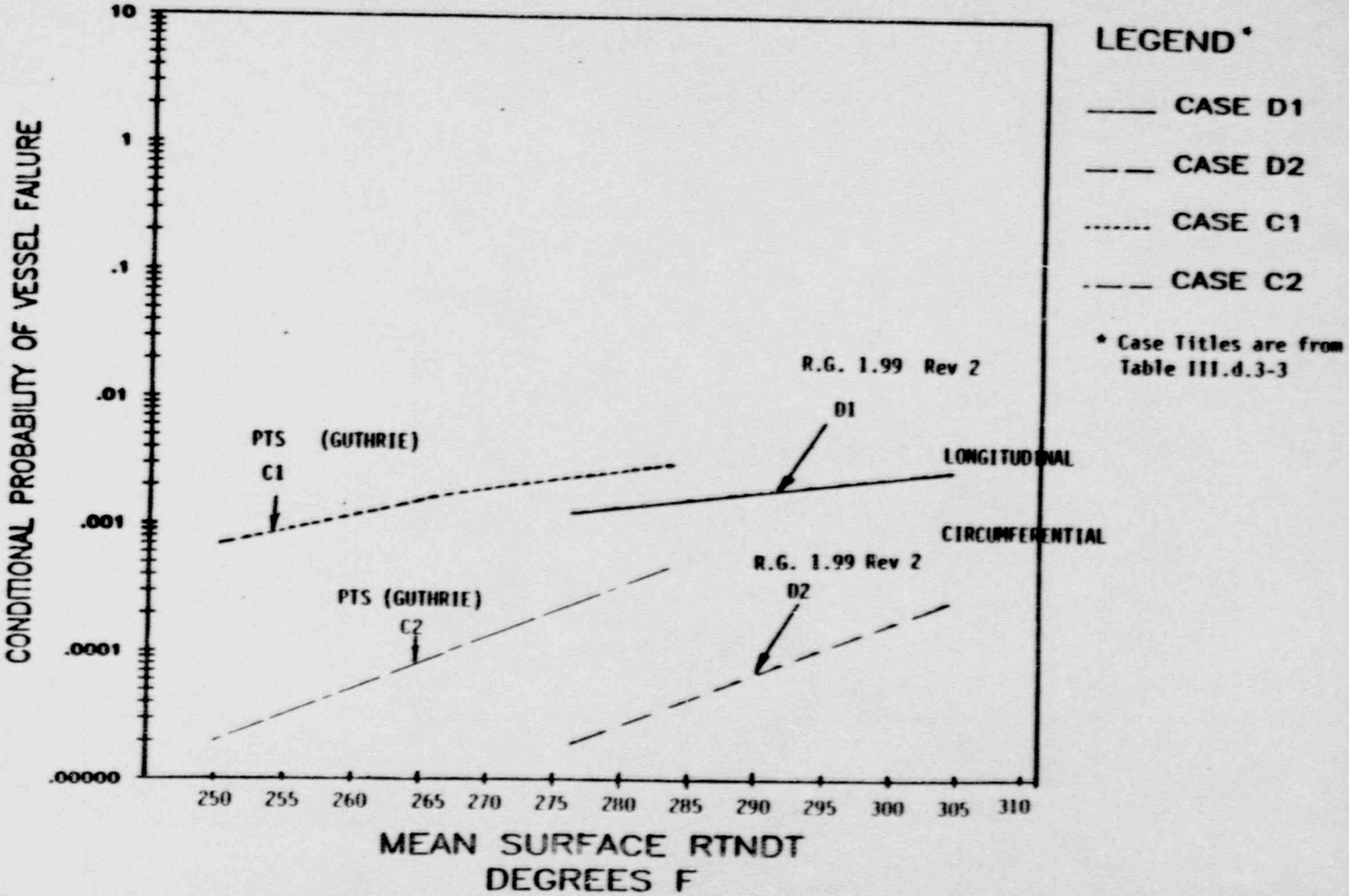
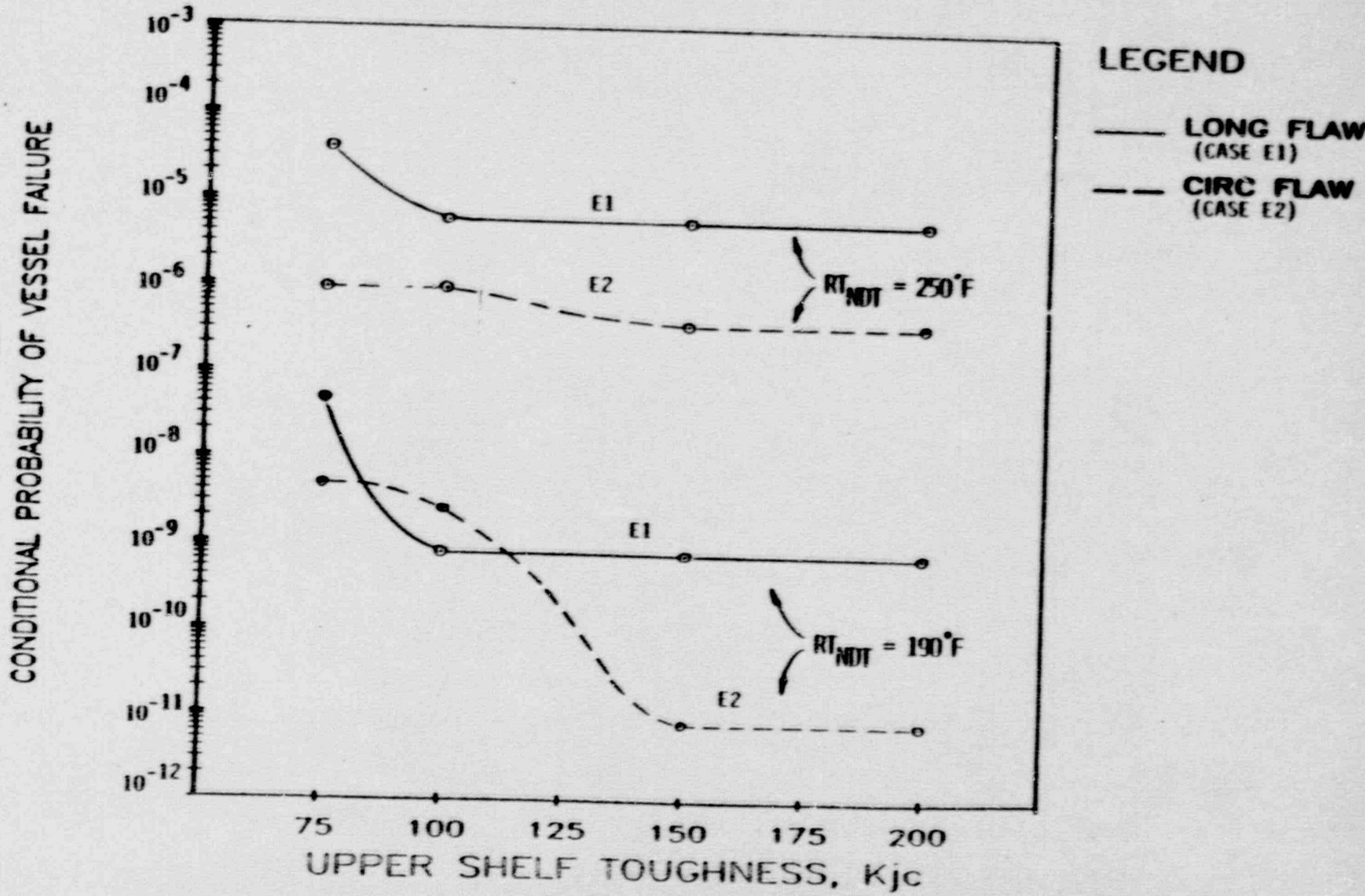


Figure III.d.3-6

PFM CONDITIONAL PROBABILITIES OF FAILURE

SENSITIVITY TO DECREASING UPPER SHELF TOUGHNESS AND INCREASING RT_{NDT}



Approaches Used:

- Deterministic
- Probabilistic

1328 D26067.008

Deterministic Fracture Sensitivity Study

Transient: Representati[illegible] small LOCA
Pressure = 10[illegible]

Vessel shell thickness: 6.5, 8.7 inches

Flaw orientation: Axial & circumferential

Upper shelf toughness: 75-200 ksi$\sqrt{in}$

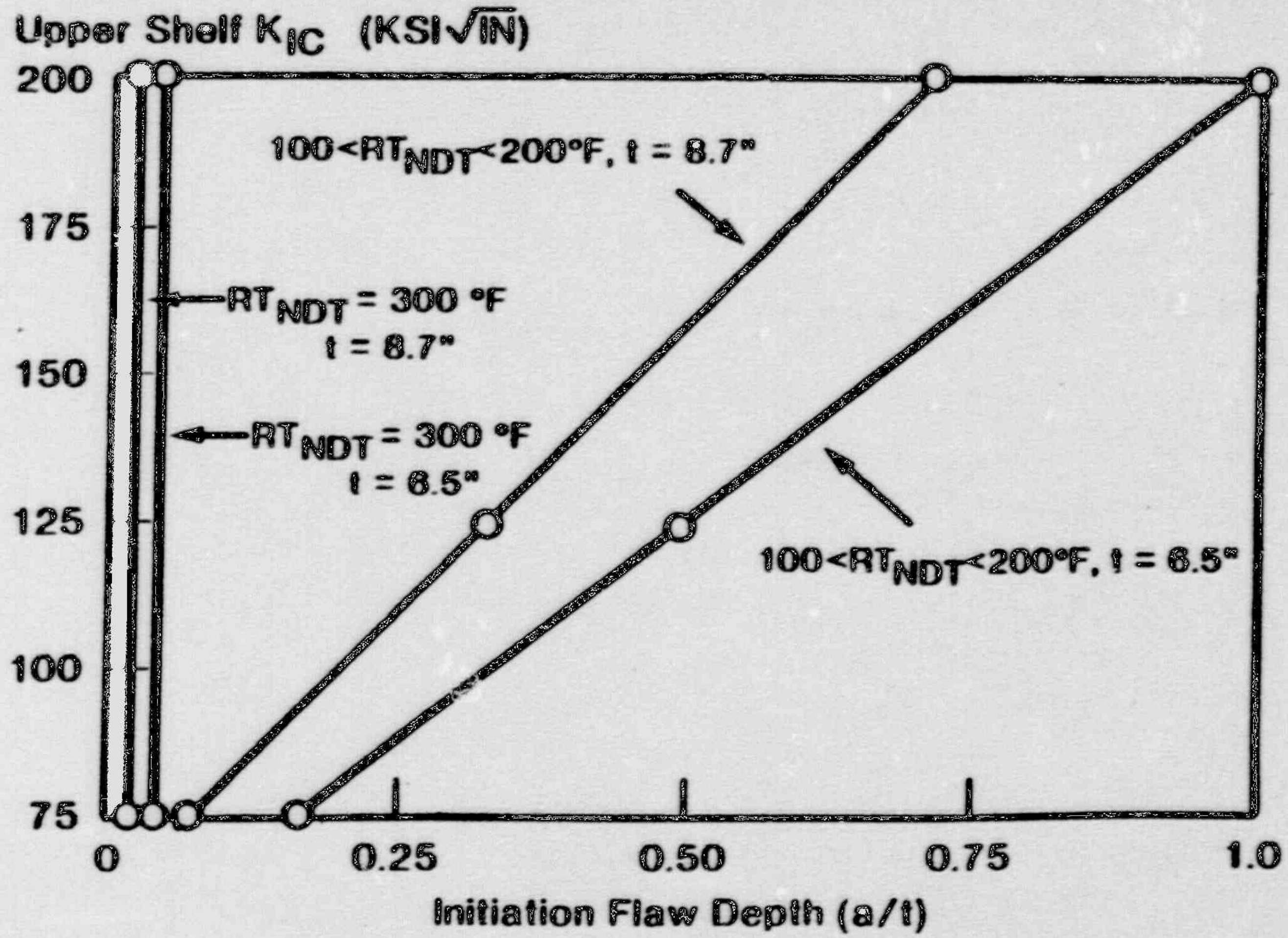
No warm prestressing: Results are conservative

Deterministic Results:

- Significant interaction between upper shelf toughness and PTS for low, intermediate RT_{NDT}
- Little or no interaction at high RT_{NDT}
- Similar trends for axial and circumferential flaws

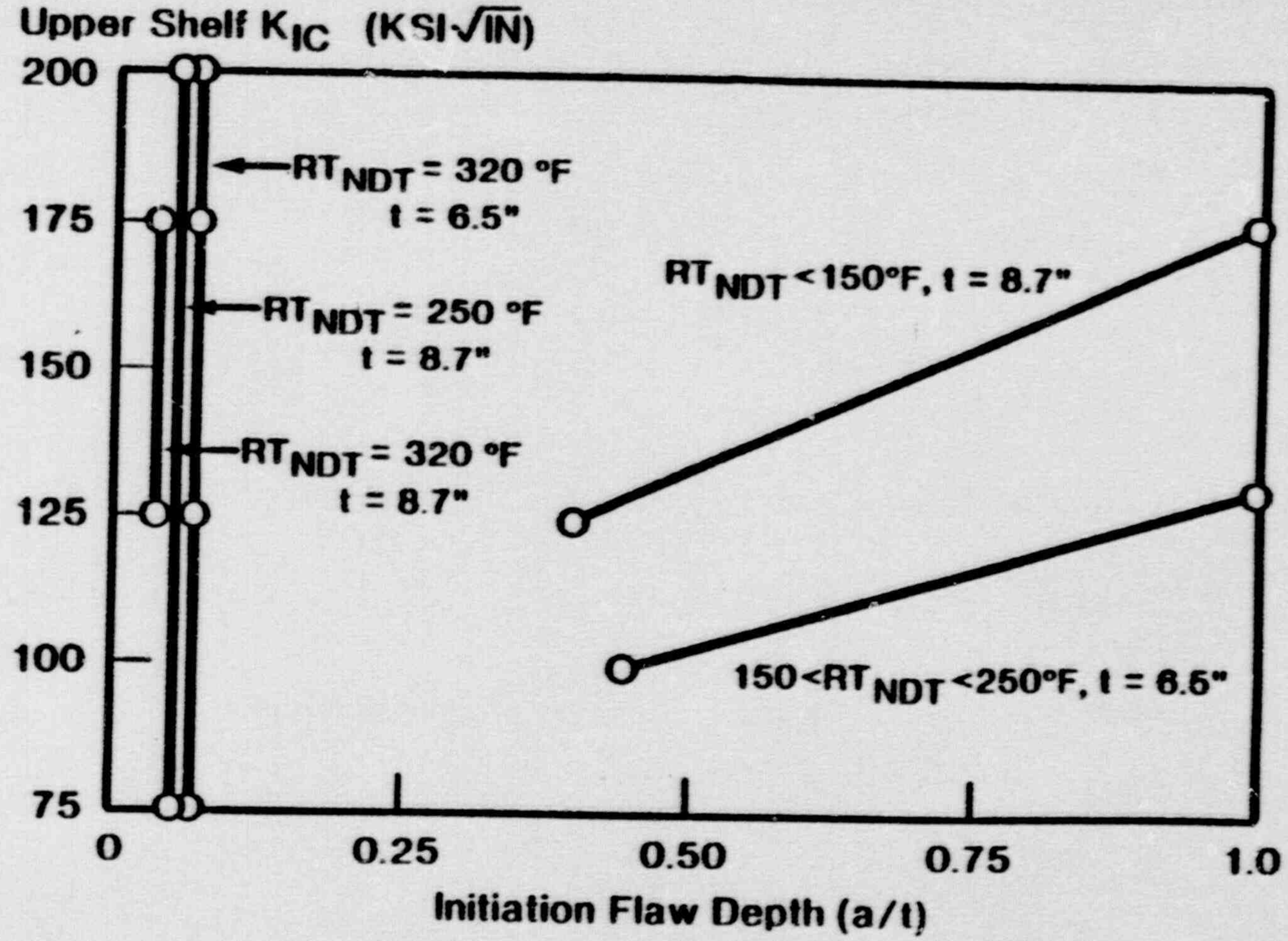
1320 b26057.012

EFFECTS OF UPPER SHELF TOUGHNESS AND RT_{NDT} ON INITIATION FLAW SIZE FOR A RANGE OF VESSEL SIZES - LONGITUDINALLY ORIENTED FLAWS



1226 D24624 012

EFFECTS OF UPPER SHELF TOUGHNESS AND RT_{NDT} ON INITIATION FLAW SIZE FOR A RANGE OF VESSEL SIZES - CIRCUMFERENTIALLY ORIENTED FLAWS



1225 D24824.008

Probabilistic Study:

Interaction now identified, now what is the impact on risk of vessel failure?

- Assumptions and procedure generally follow NRC approach of SECY-82-465
- Study was more limited than deterministic work:
  - Single thickness: 6.5 in.

1326 D26067 013

Additional Inputs, Different from NRC Study

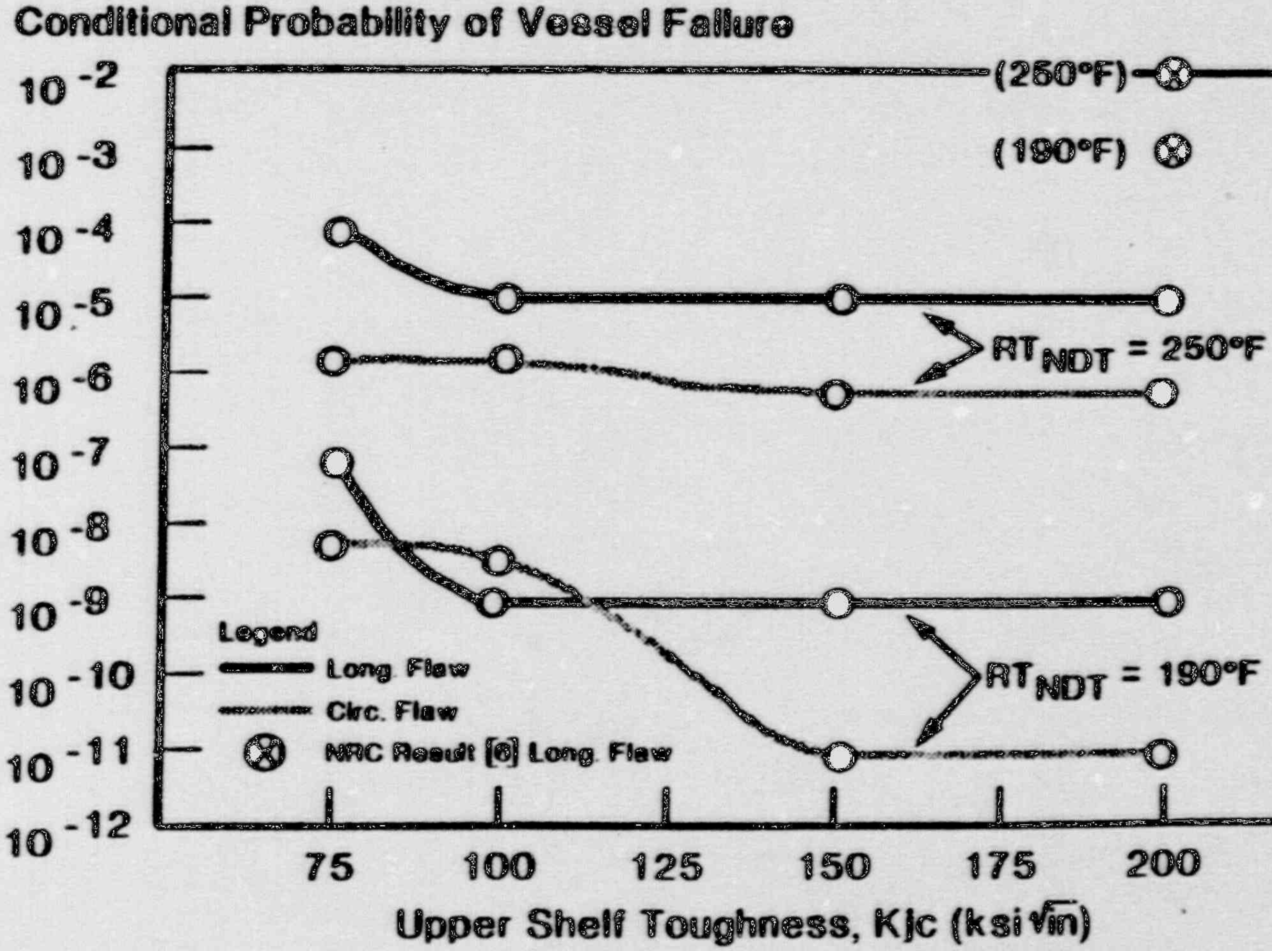
- Flaw distribution: Updated octavia distribution
- Fluence attenuation:

$$F = F_{ID} \cdot (1 - .207a + 0.014a^2 - 0.0003a^3)$$

- Failure criterion: Fracture criteria only
- Shelf toughness: 75-200 ksi $\sqrt{in}$

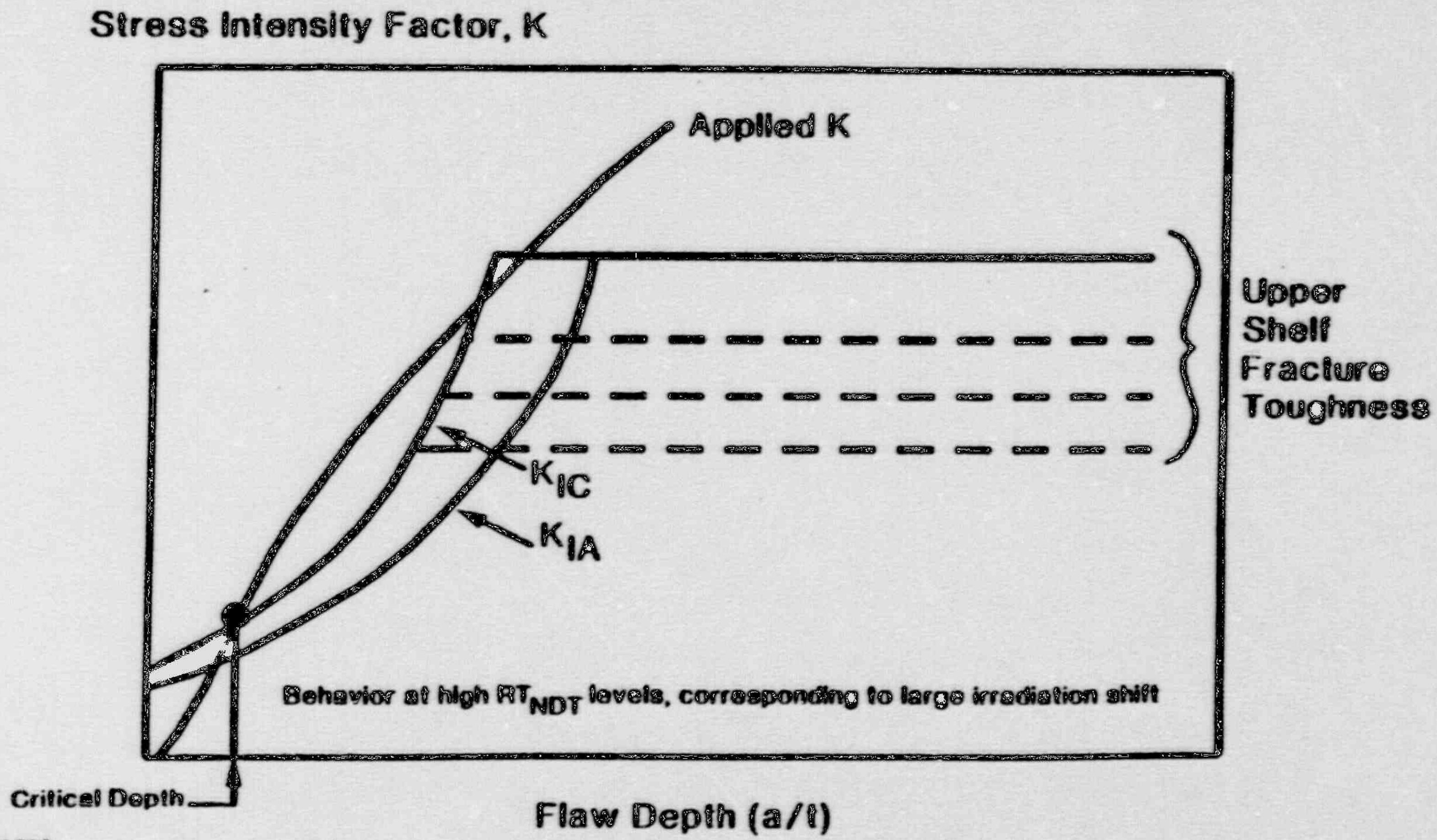
1326 D26057.014

CONDITIONAL PROBABILITIES OF FAILURE, SHOWING SENSITIVITY TO DECREASING UPPER SHELF TOUGHNESS AND INCREASING RT_{NDT}



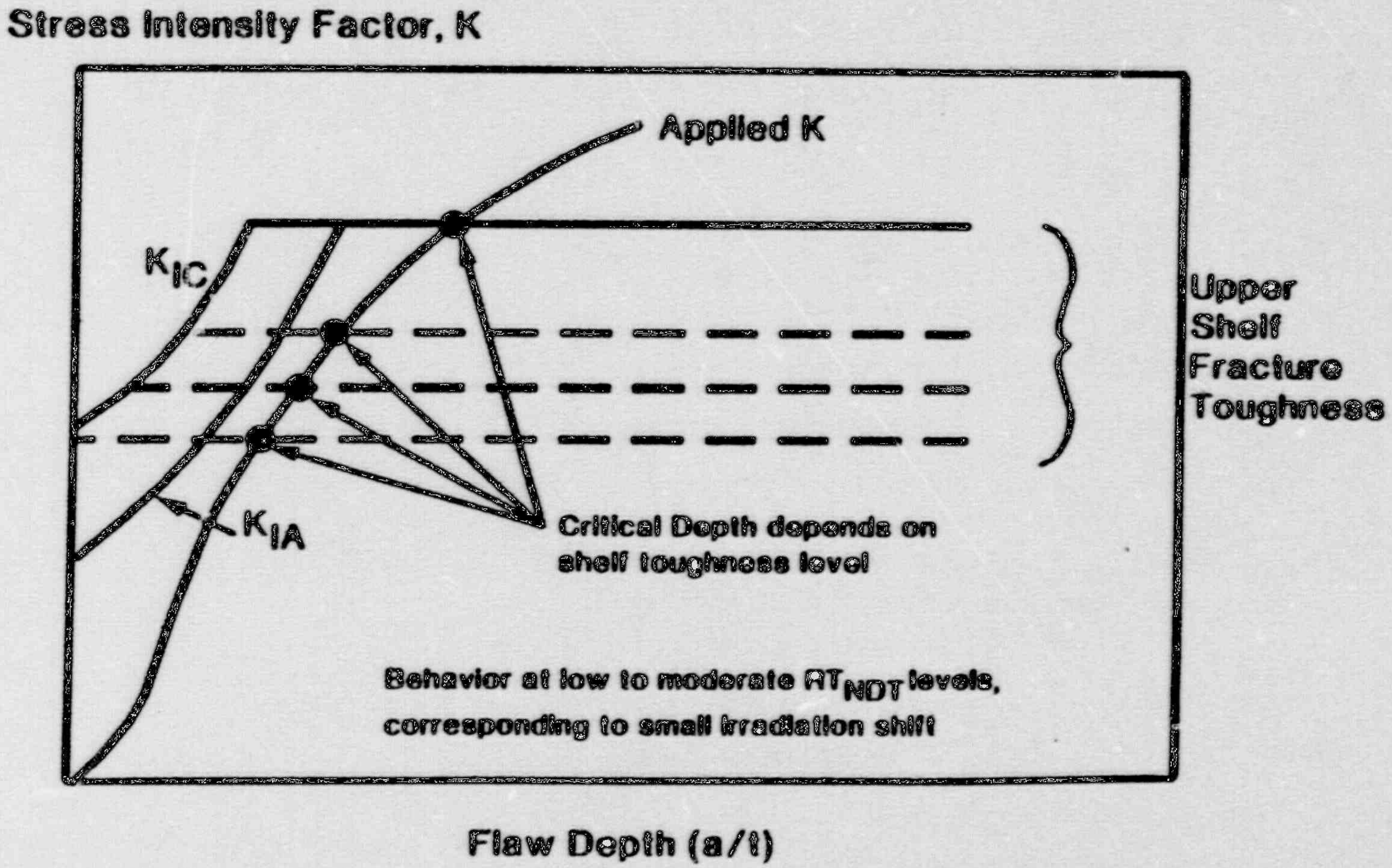
1226 D24824.011

SCHEMATIC OF FRACTURE EVALUATION FOR A LONGITUDINALLLY ORIENTED FLAW UNDER PRESSURIZED THERMAL SHOCK CONDITIONS



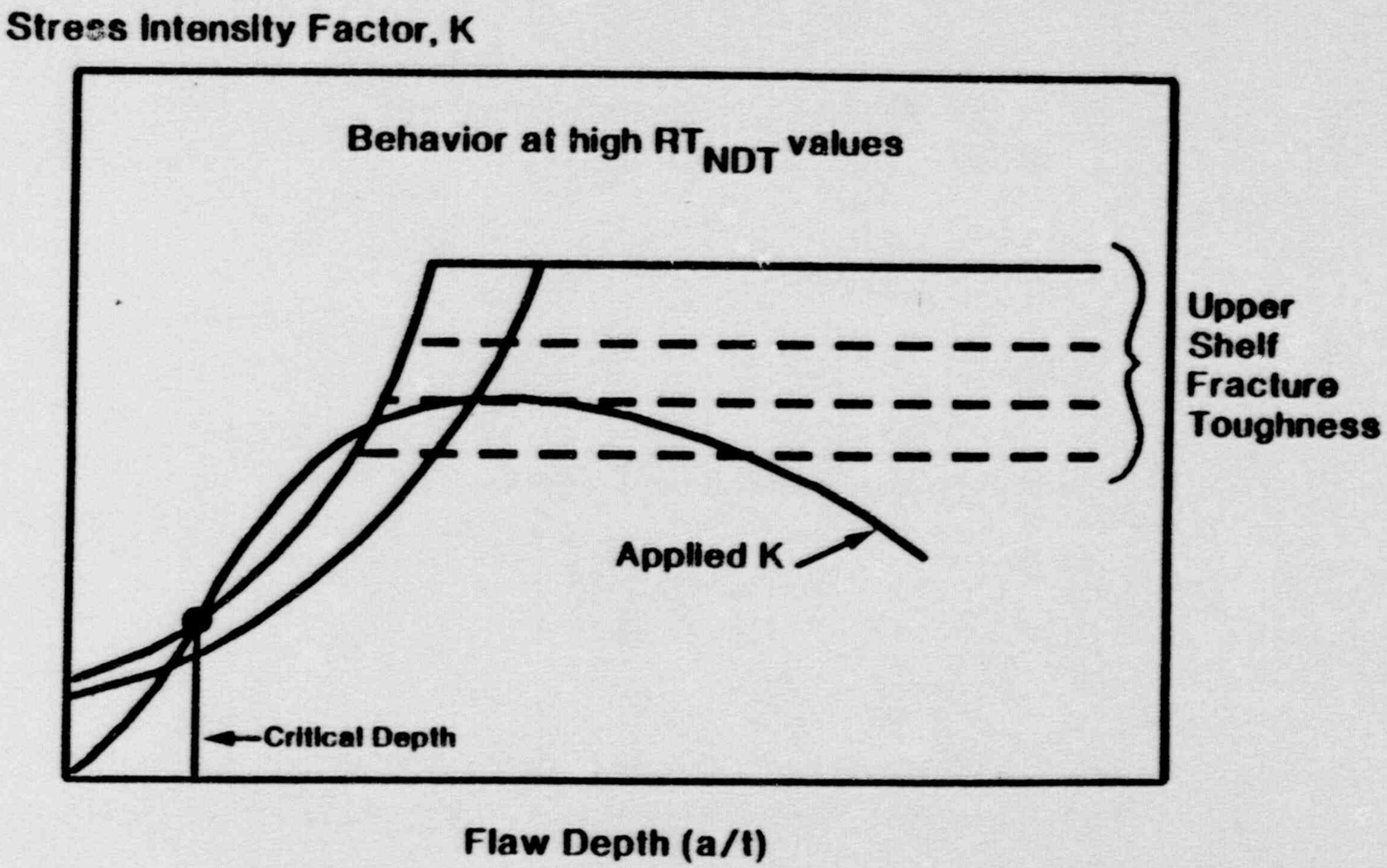
1226 D24824 008

SCHEMATIC OF FRACTURE EVALUATION FOR A LONGITUDINALLLY ORIENTED FLAW UNDER PRESSURIZED THERMAL SHOCK CONDITIONS



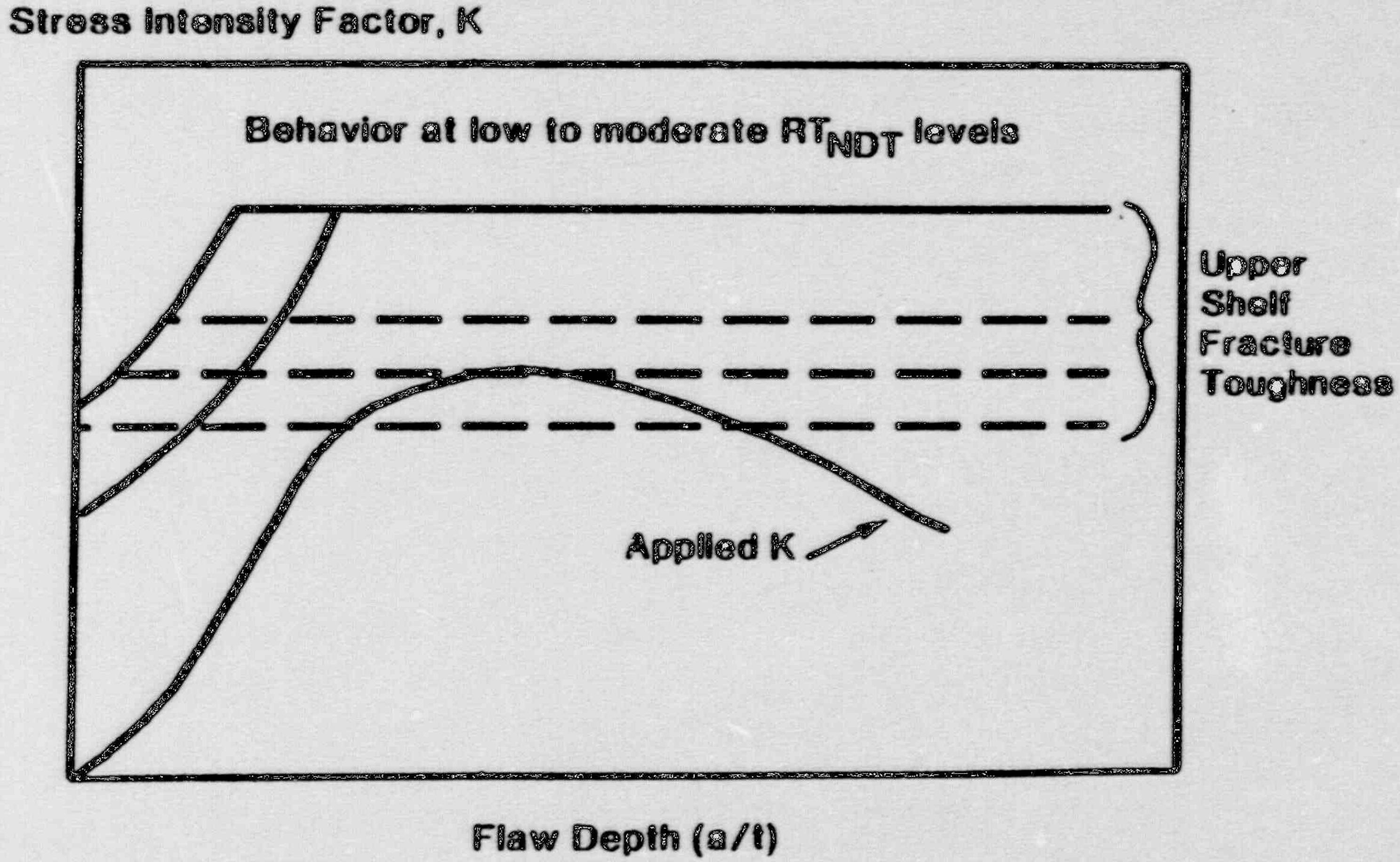
1226 D24824.007

SCHEMATIC OF FRACTURE EVALUATION FOR A CIRCUMFERENTIALLY ORIENTED FLAW UNDER PRESSURIZED THERMAL SHOCK CONDITIONS



1225 D24824.009

SCHEMATIC OF FRACTURE EVALUATION FOR A CIRCUMFERENTIALLY ORIENTED FLAW UNDER PRESSURIZED THERMAL SHOCK CONDITIONS



1226 D24824 010

- Low shelf toughness CAN adversely affect reactor vessel integrity under some conditions
- Severity of the effect depends on:
  - Irradiation damage
  - System transient behavior
  - Wall thickness
  - Shelf toughness level
- Such effects must be considered in integrity assessments
- Careful treatment of shelf fracture toughness is needed

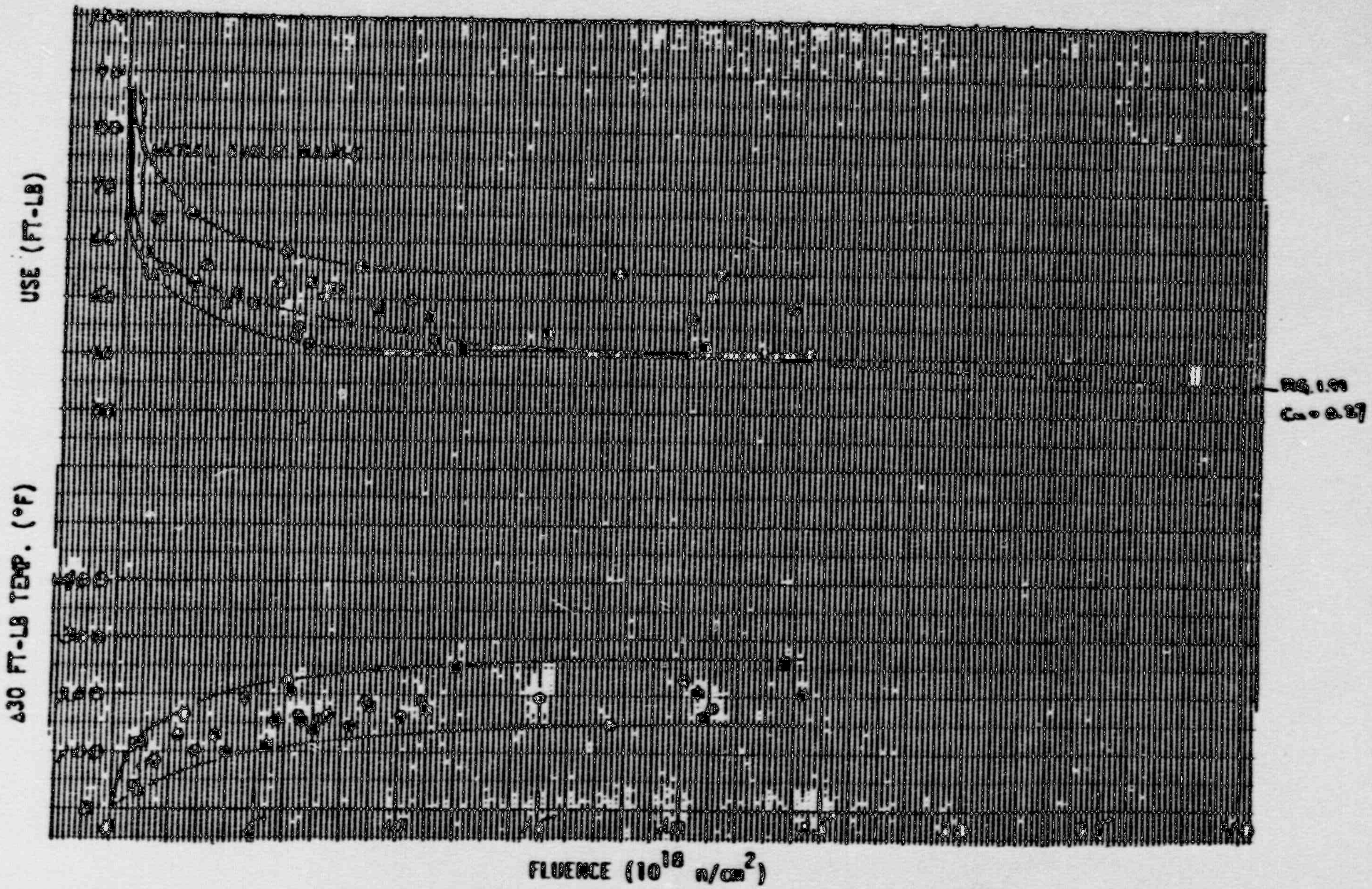
1320 D26067.016

DEVELOPMENT OF FRACTURE PROPERTIES FOR ANALYSIS

- LOW TOUGHNESS IS PRIMARILY A WELD CONCERN FOR B&W VESSELS
- COMPILED AVAILABLE DATA
  - CHARPY
  - FRACTURE TOUGHNESS

COMPILATION OF CHARPY DATA

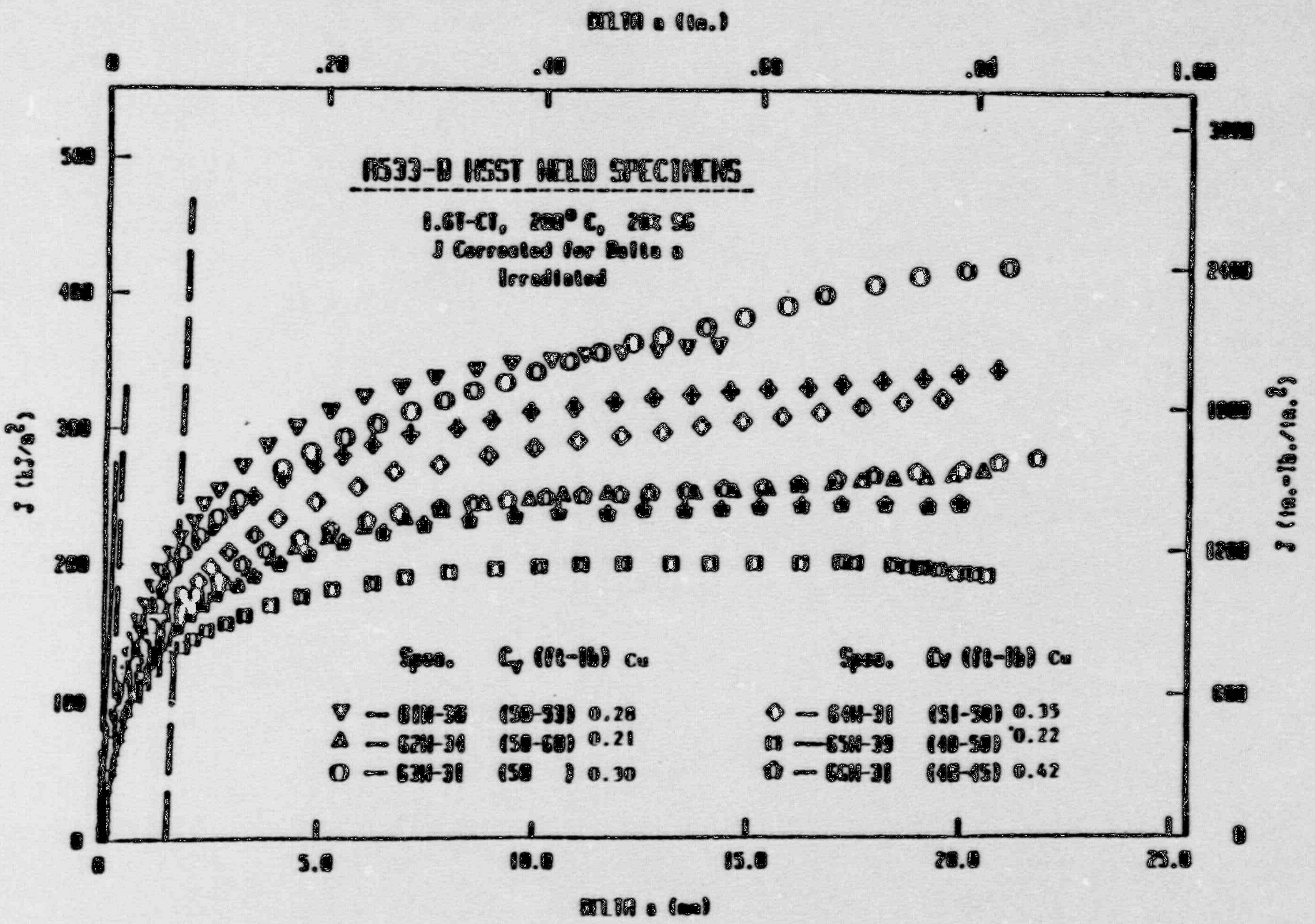
- LARGE AMOUNT OF DATA AVAILABLE
- SIMILAR BEHAVIOR FOR ALL HEATS, REGARDLESS OF CHEMISTRY
- IRRADIATION DAMAGE SATURATION OCCURS AT 7 - 9 x 10^{18} N/CM² FOR BOTH UPPER SHELF ENERGY AND TRANSITION SHIFT



B&W SUBMERGED ARC WELDS MADE WITH LINDE 80 FLUX

COMPILATION OF FRACTURE TOUGHNESS DATA

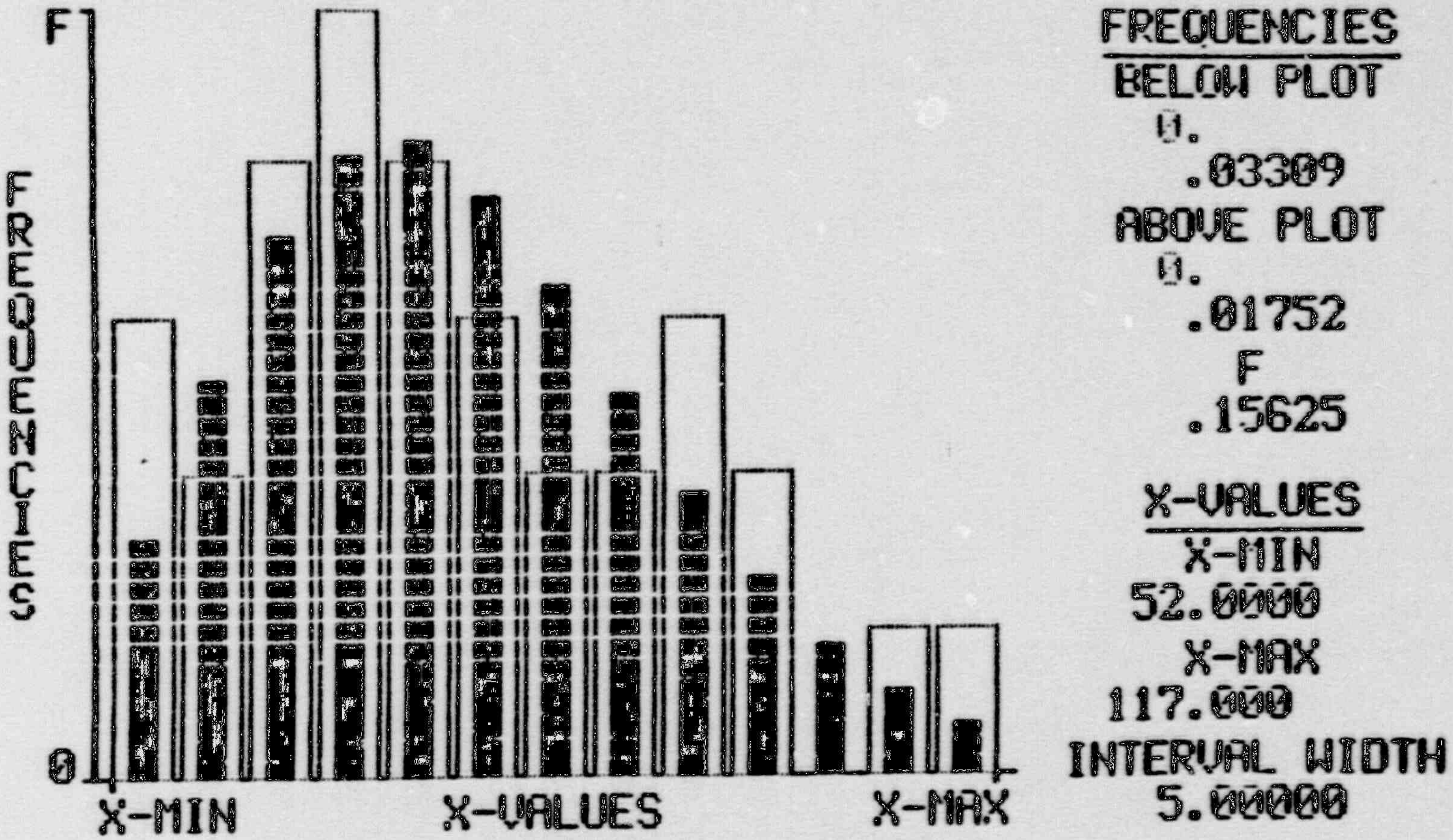
- SOURCES:
  - WESTINGHOUSE
  - NRC
  - EPRI
- TEST TEMPERATURES RANGE FROM 250 - 550F
- TEMPERATURE EFFECT ON TOUGHNESS IS MORE IMPORTANT THAN HEAT TO HEAT VARIATION
- TEMPERATURE RANGE OF INTEREST DURING TRANSIENTS: 300 - 400F
- DATA AT 390F CHOSEN FOR CHARACTERIZATION
  - SHELF TOUGHNESS INCREASES AT LOWER TEMPERATURES
  - LARGEST NUMBER OF TEST RESULTS AVAILABLE



J-R CURVES FOR SIX HEATS OF LINDE 80 WELDS

UPPER SHELF FRACTURE TOUGHNESS DETERMINATION

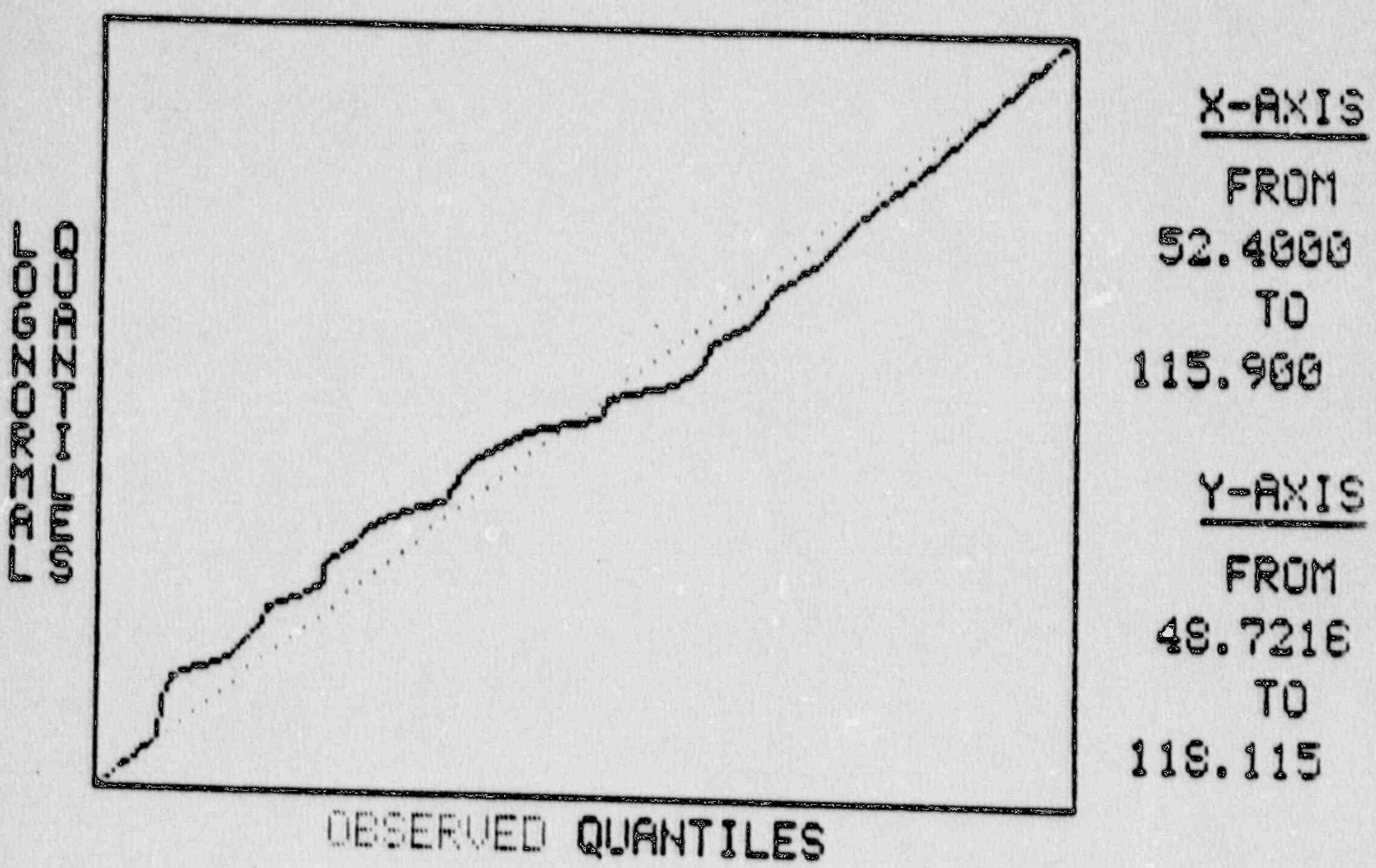
- CONSERVATIVE MEASURE OF TOUGHNESS USED : K_{JC}, BASED ON J_{IC}
- DISTRIBUTION OF K_{JC} DETERMINED STATISTICALLY : LOGNORMAL
- FIT DETERMINED IS VERY GOOD
- RESULTING TOUGHNESS IS PRIMARILY 90 - 140 KSI$\sqrt{\text{IN}}$, WITH MORE VALUES ABOVE THIS RANGE THAN BELOW.



FREQUENCY PLOT FOR LOGNORMAL DISTRIBUTION

Key: ■ Calculated

□ Actual



PROBABILITY PLOT TO CHECK ACCURACY OF LOGNORMAL DISTRIBUTION

KEY INPUT FOR POINT BEACH PROBABILISTIC FRACTURE MECHANICS ANALYSIS

- POINT BEACH VESSEL MODEL AND MATERIAL PROPERTIES
- o $\bar{h}$ = CONSTANT 300 BTU/HR-FT^2-^{o}F
- o LONGITUDINAL AND CIRCUMFERENTIAL FLAW ORIENTATIONS
- OCTAVIA FLAW-SIZE DISTRIBUTION*
- o 6:1 FINITE FLAW FIRST INITIATION/CONTINUOUS FLAW FOR ARREST AND SUBSEQUENT REINITIATIONS
- o POINT BEACH FLUENCE ATTENUATION W/DPA
- PROPOSED REGULATORY GUIDE 1.99 - REVISION 2 TREND CURVE
- UPPER SHELF TOUGHNESS - LOGNORMAL DISTRIBUTION FOR LINDE 80 WELD
- o FAILURE CRITERION - $a_{ARREST} \leq 0.75\ a/t$
- o SIMULATION TECHNIQUE: MONTE CARLO W/IMPORTANCE SAMPLING

* POINT BEACH FLAW INDICATIONS WILL NOT BE ACCOUNTED FOR AT THIS TIME

DETERMINATION OF RISK (OR FREQUENCY) OF SIGNIFICANT FLAW EXTENSION

F_S	Frequency of a specific transient scenario (reactor years^{-1})	
β	Beta (min^{-1})	Stylized Transient
T_f	Final Temperature (°F)	$T_{(t)} = T_f + (T_{initial} - T_f)\, e^{-\beta t}$
CP	Conditional probability of significant flaw extension given that the transient scenario occurs	
F_{SFE}	Frequency of significant flaw extension	

$$(F_S) \quad \times \quad (CP) \quad = F_{SFE}$$

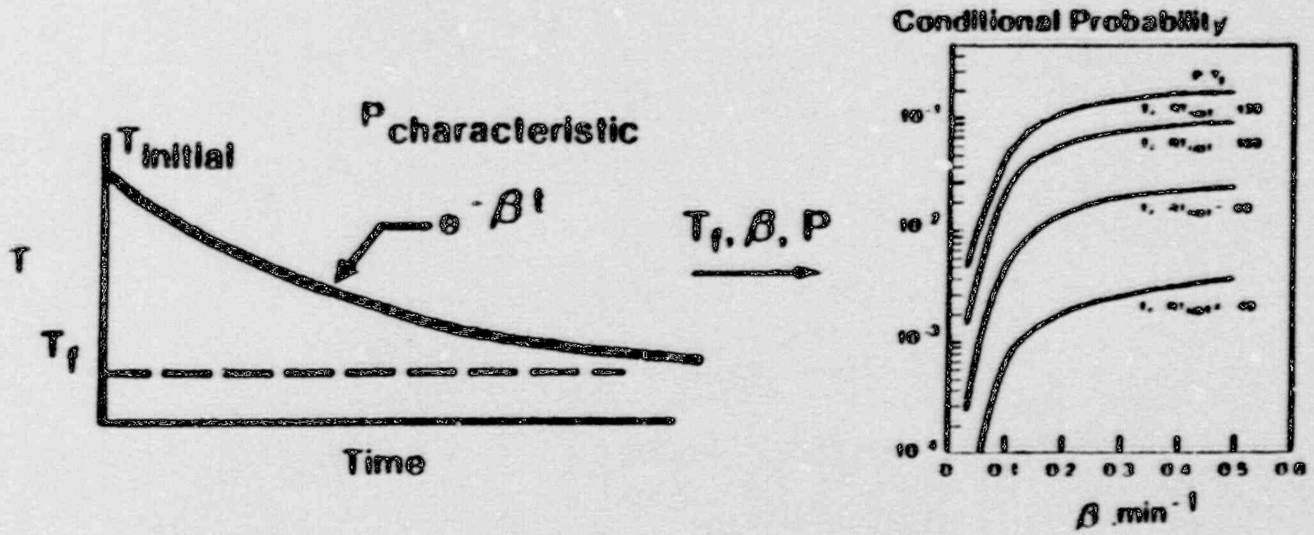


Figure III.e.2-1

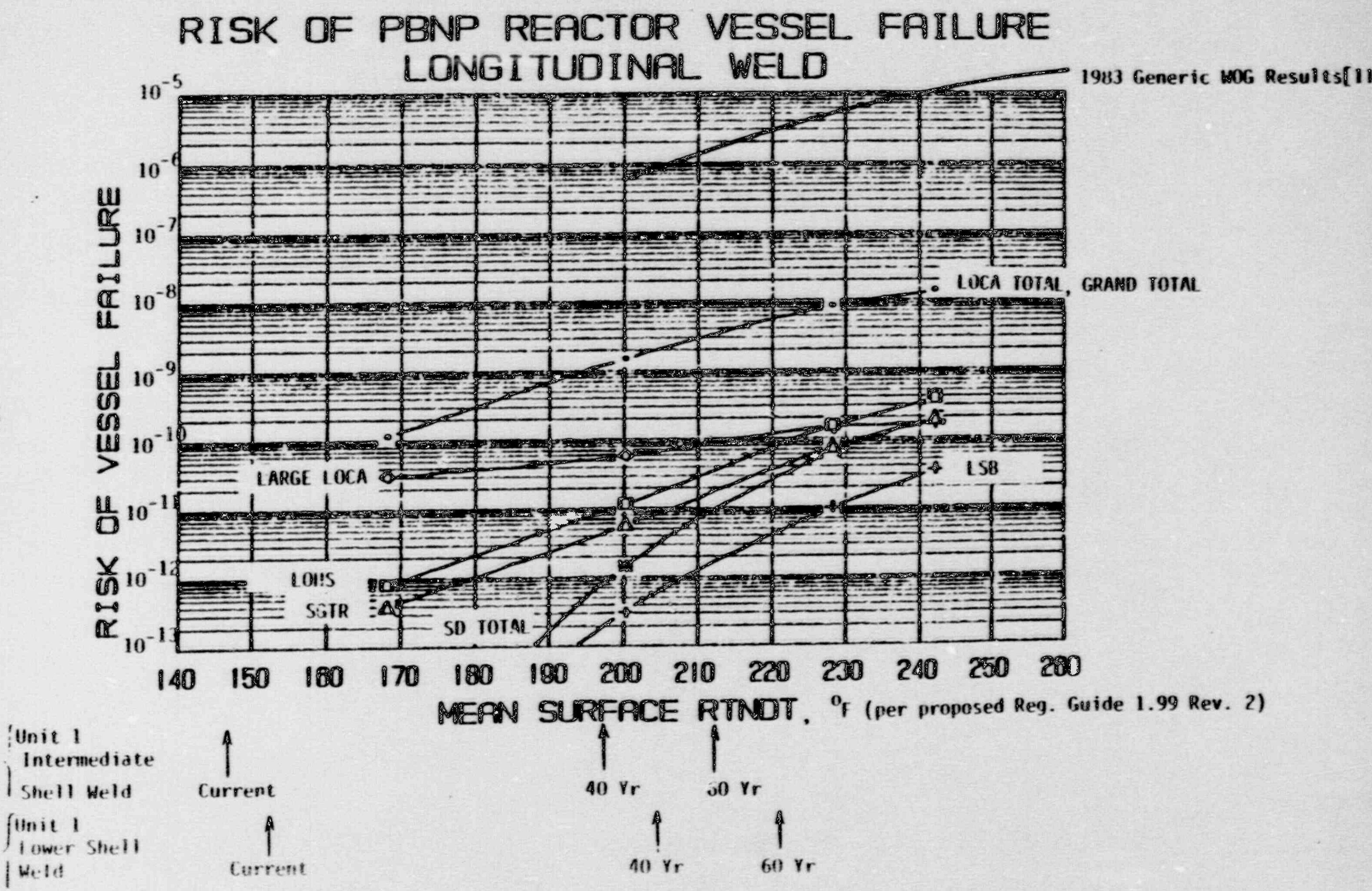
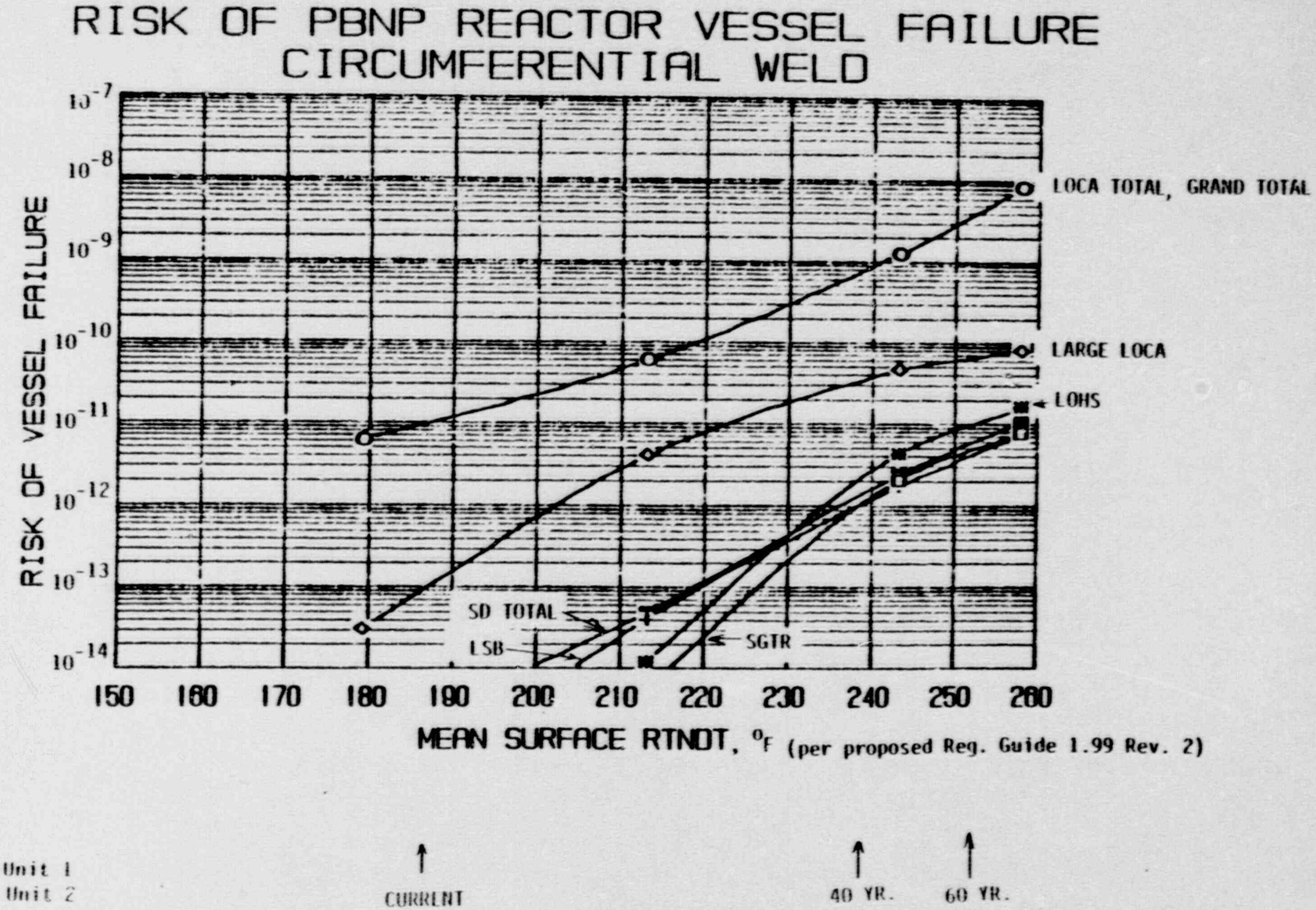


Figure III.e.2-2



151

Figure III.e.3-1

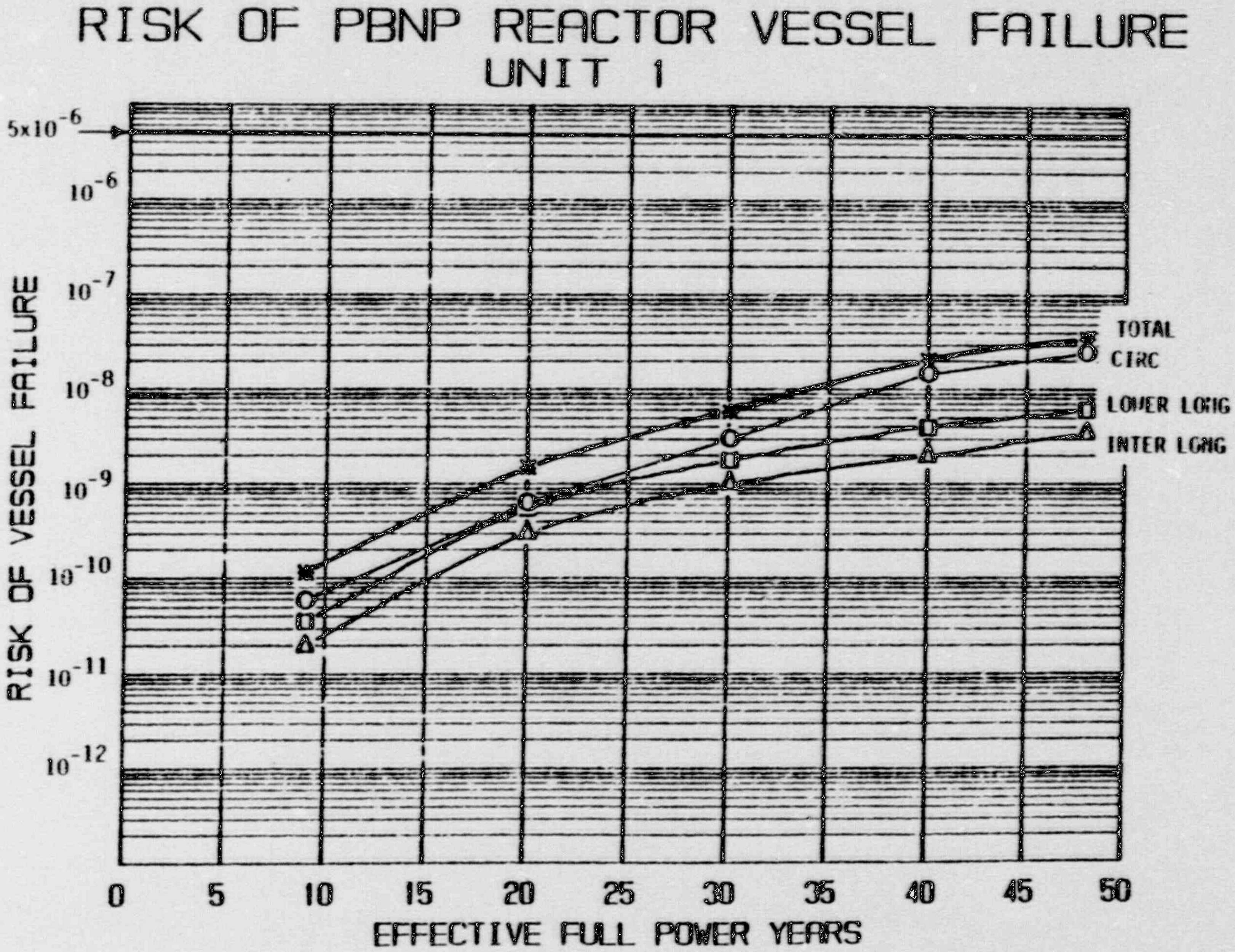
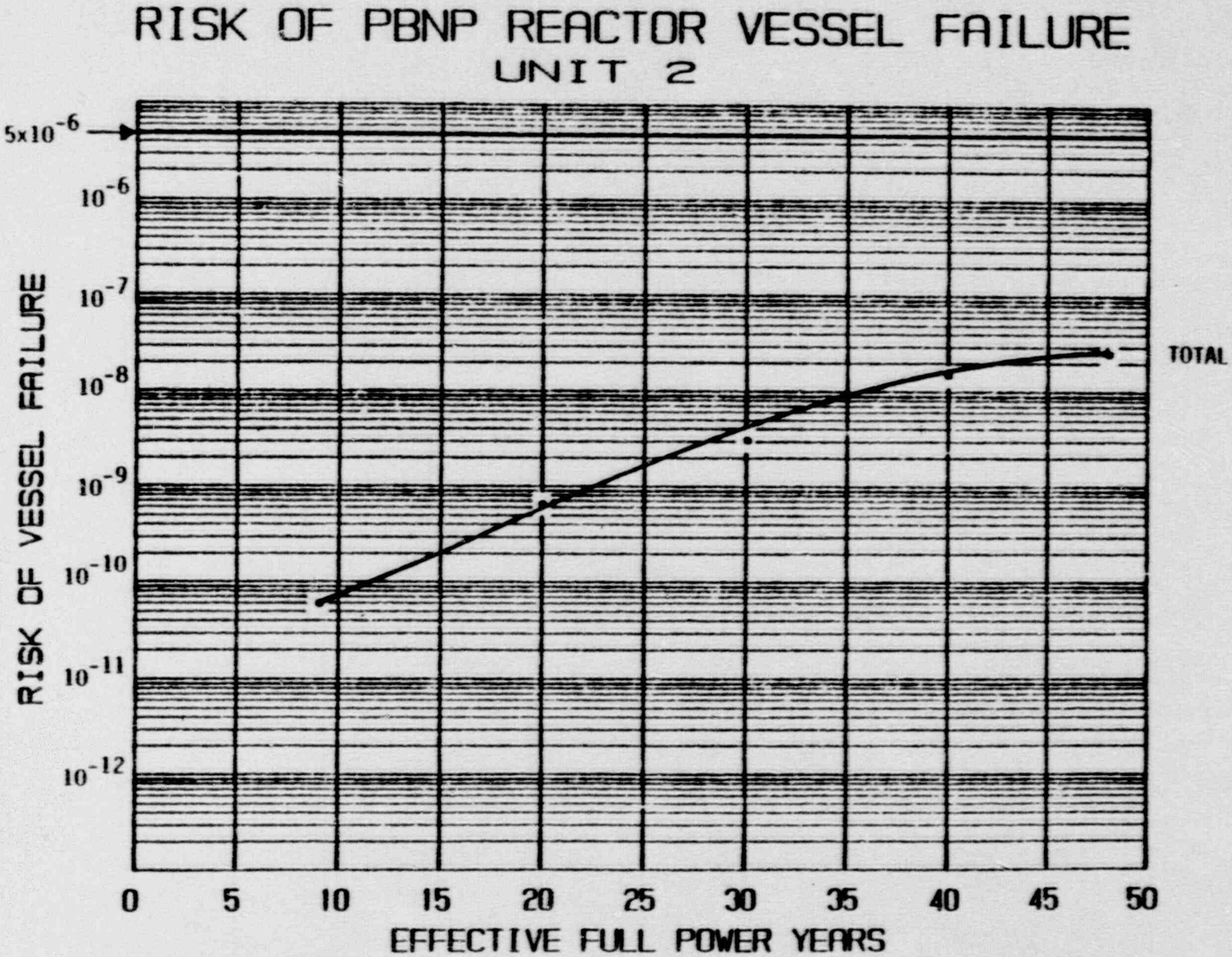
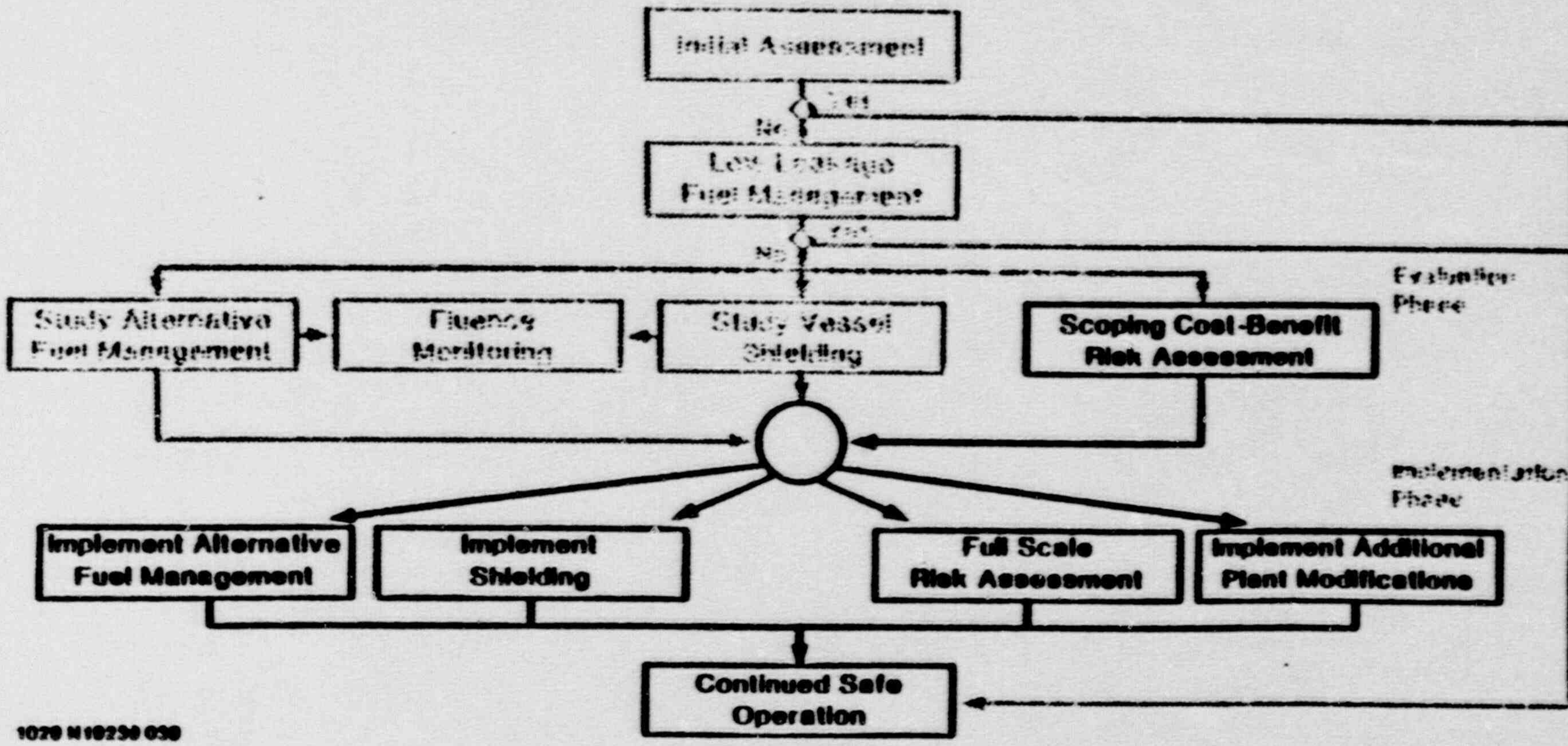


Figure III.e.3-2



Reactor Vessel Life Extension

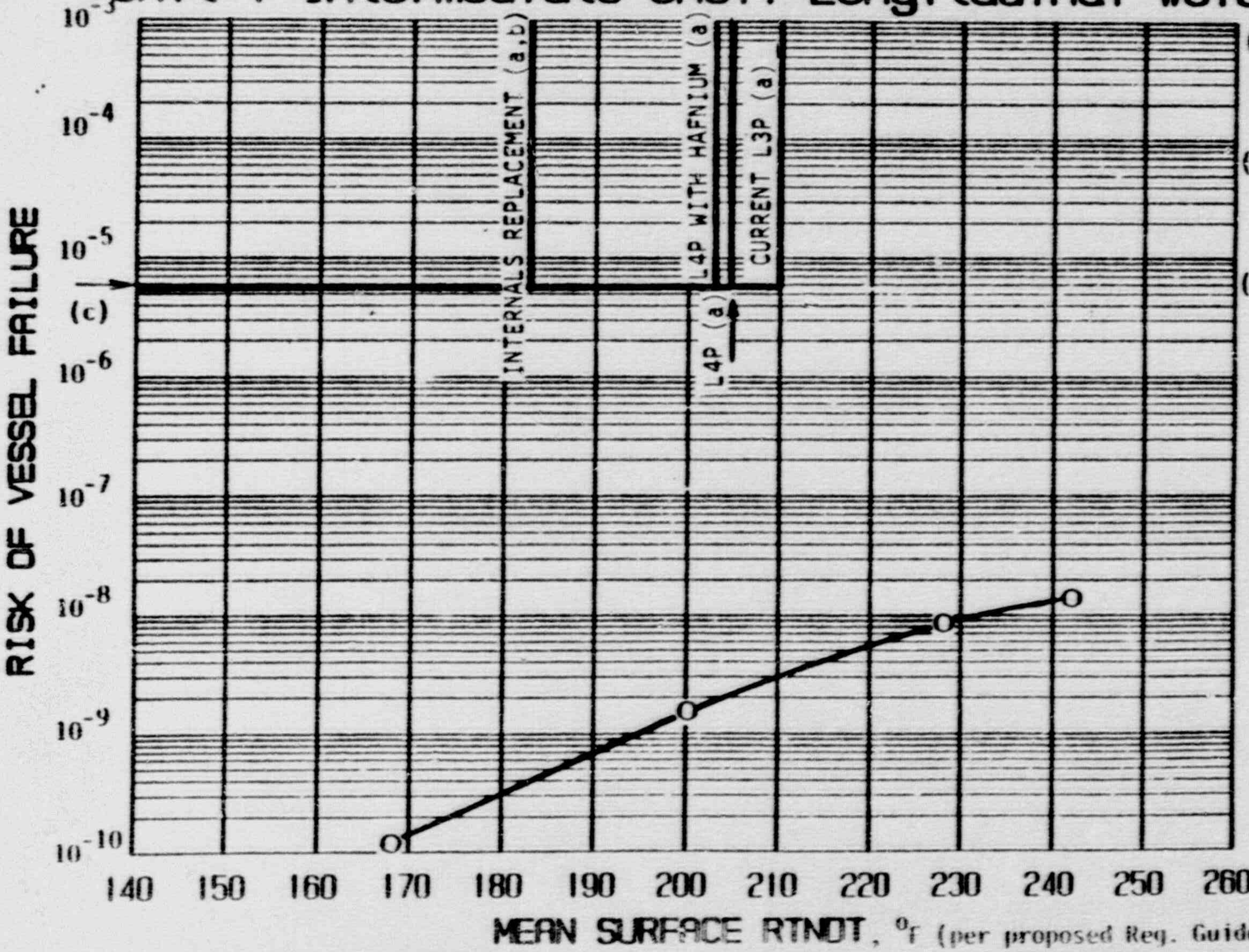
A Cost-Effective Approach



1020 N10230 030

FIGURE III.e.4-1

Risk of PBNP Versus Flux Reduction Measures
Unit 1 Intermediate Shell Longitudinal Weld



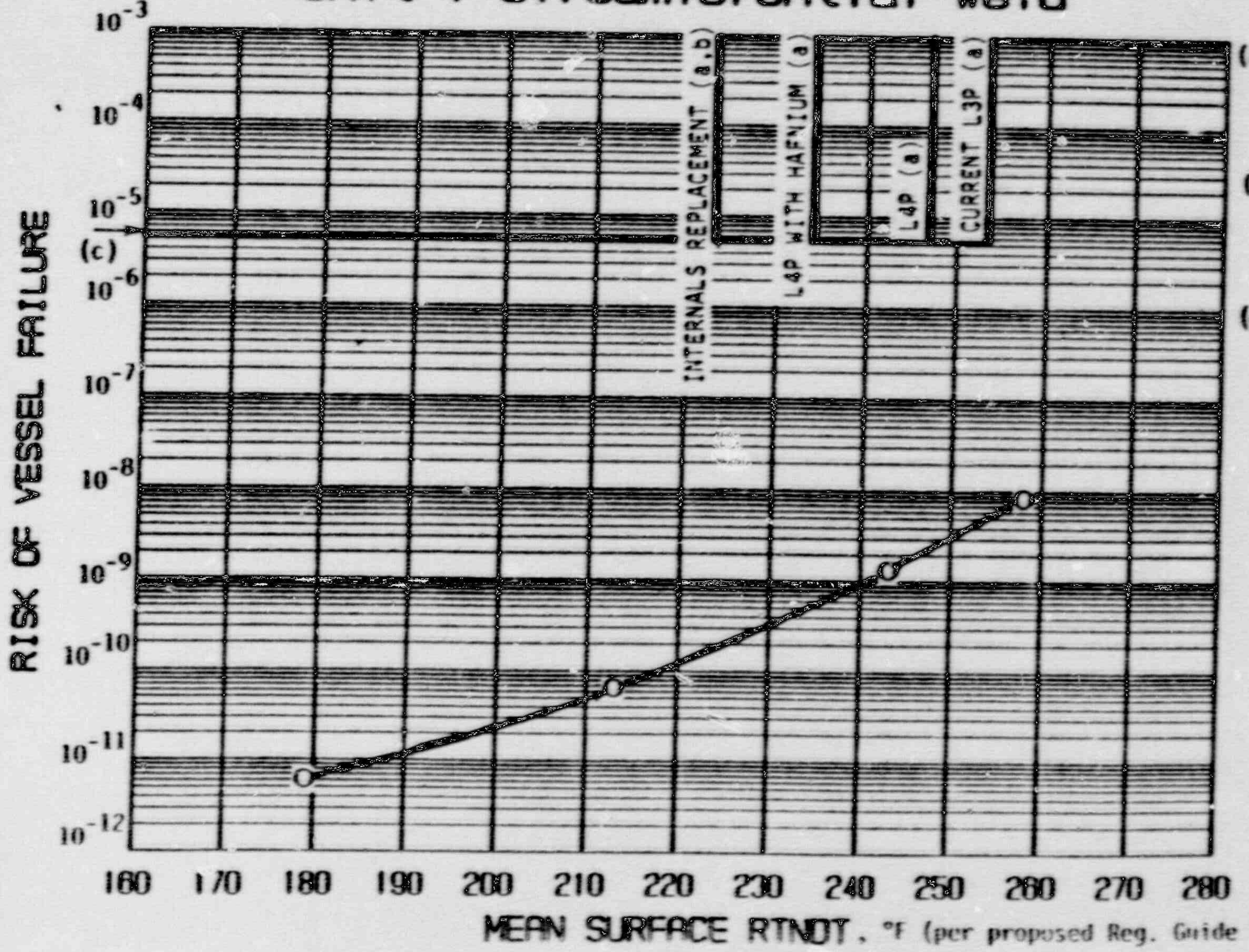
(a) Estimated RT_{NDT} values at 48 EFPY, assuming implementation at cycle 17

(b) Difference between Internals Replacement with L3P and L4P is negligible

(c) Approximate goal for reference only; All welds must be considered for comparison against safety goal of $5x10^{-6}$/R-yr [15]

FIGURE III.e.4-3

Risk of PBNP Versus Flux Reduction Measures
Unit 1 Circumferential Weld



(a) Estimated RT_{NDT} values at 48 EFPY, assuming implementation at cycle 17

(b) Difference between Internals Replacement with L3P and L4P is negligible.

(c) Approximate goal for reference only; All welds must be considered for comparison against safety goal of 5×10^{-6}/R-yr [15]

Reactor Vessel Life Extension

Goal of Point Beach Study

- Identify, evaluate, and rank actions that, if appropriately implemented, will maintain a viable life extension option for the vessels beyond forty operating years
- Add 20 years to reactor vessel life
- Do not adversely impact ongoing plant reliability and capacity

1C20 N10230 003

Components of Study

Neutron Embrittlement

- Flux reduction evaluation
  - Fuel management
  - Vessel shielding via internals modification/ replacement
  - Improved neutron dosimetry (fluence monitoring)
- Scoping cost-benefit risk assessment

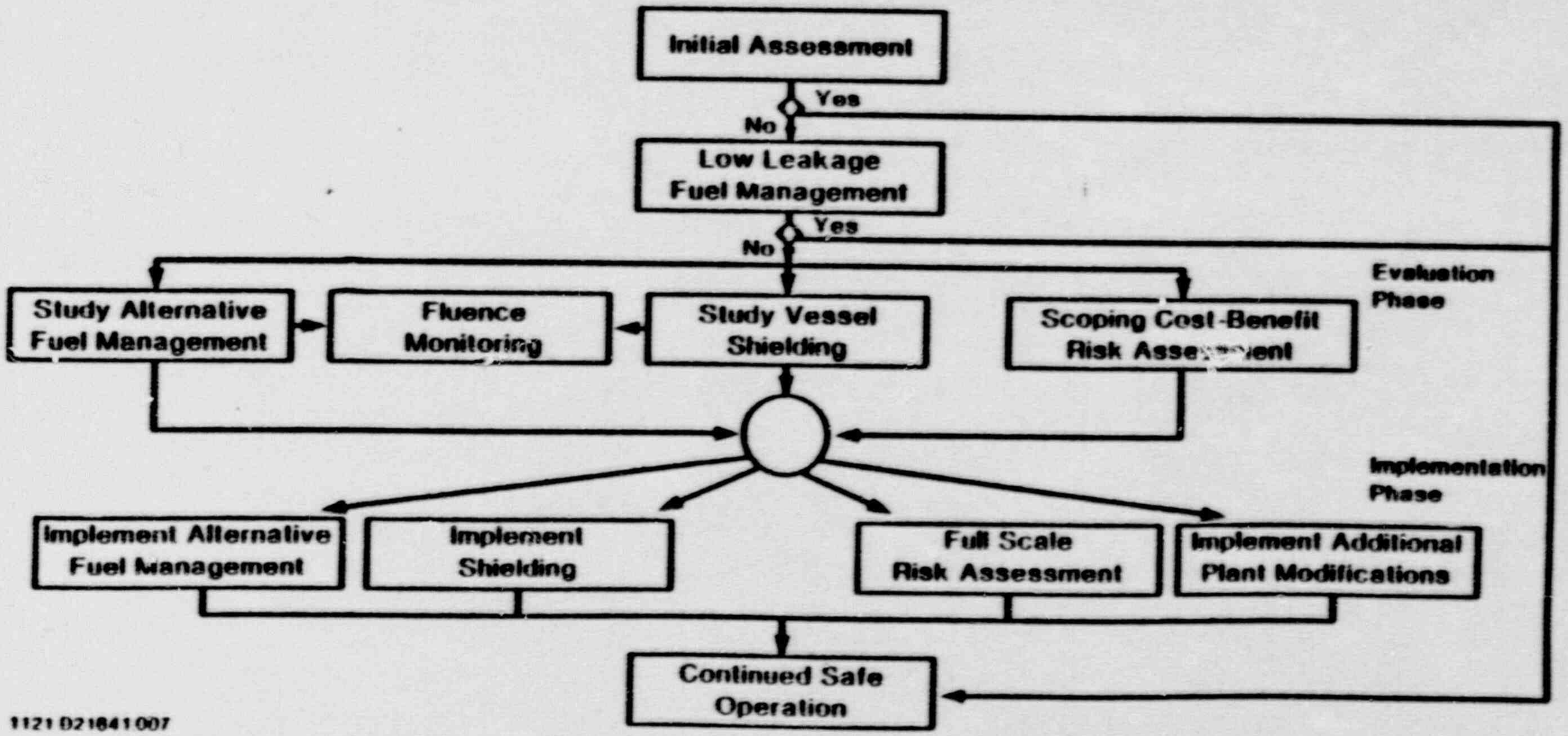
Fatigue

- Transient monitoring program for reactor vessel and other critical plant components

1121 D21641.008

Reactor Vessel Life Extension

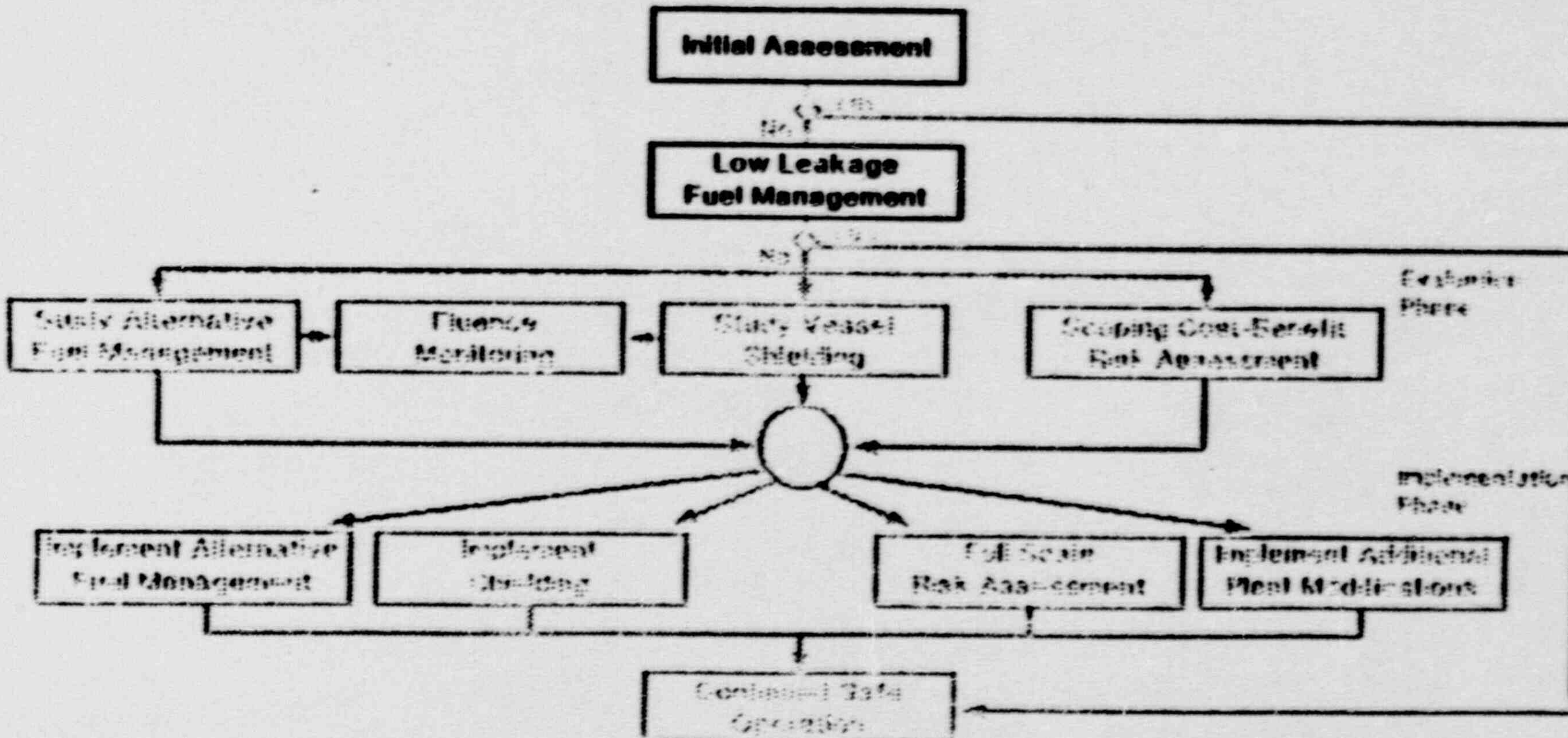
A Cost-Effective Approach to Address Neutron Embrittlement



1121 D21841.007

Reactor Vessel Life Extension

A Cost-Effective Approach



1070 N10230 041

Point Beach Unit 1
Reactor Vessel Beltline Region

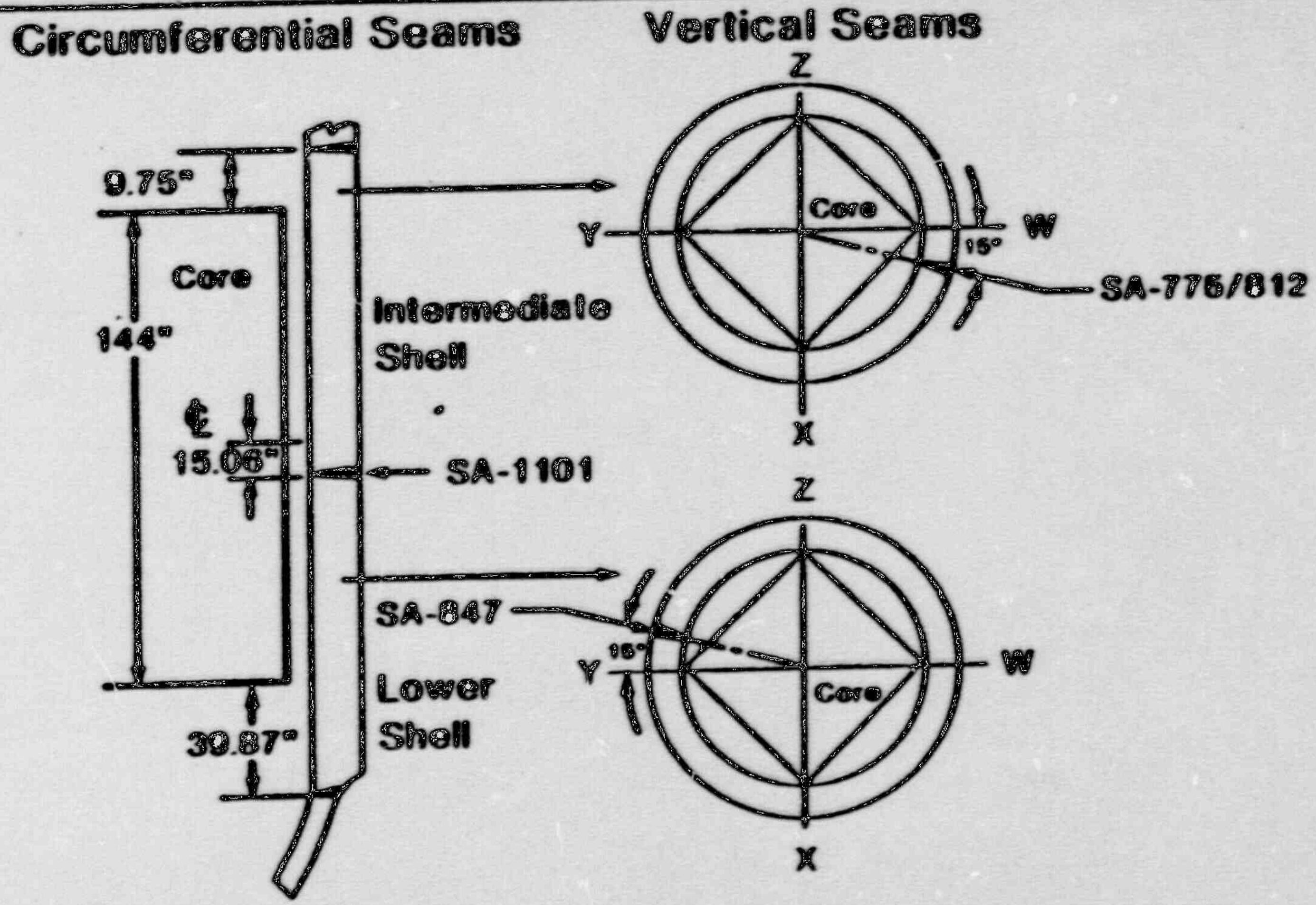
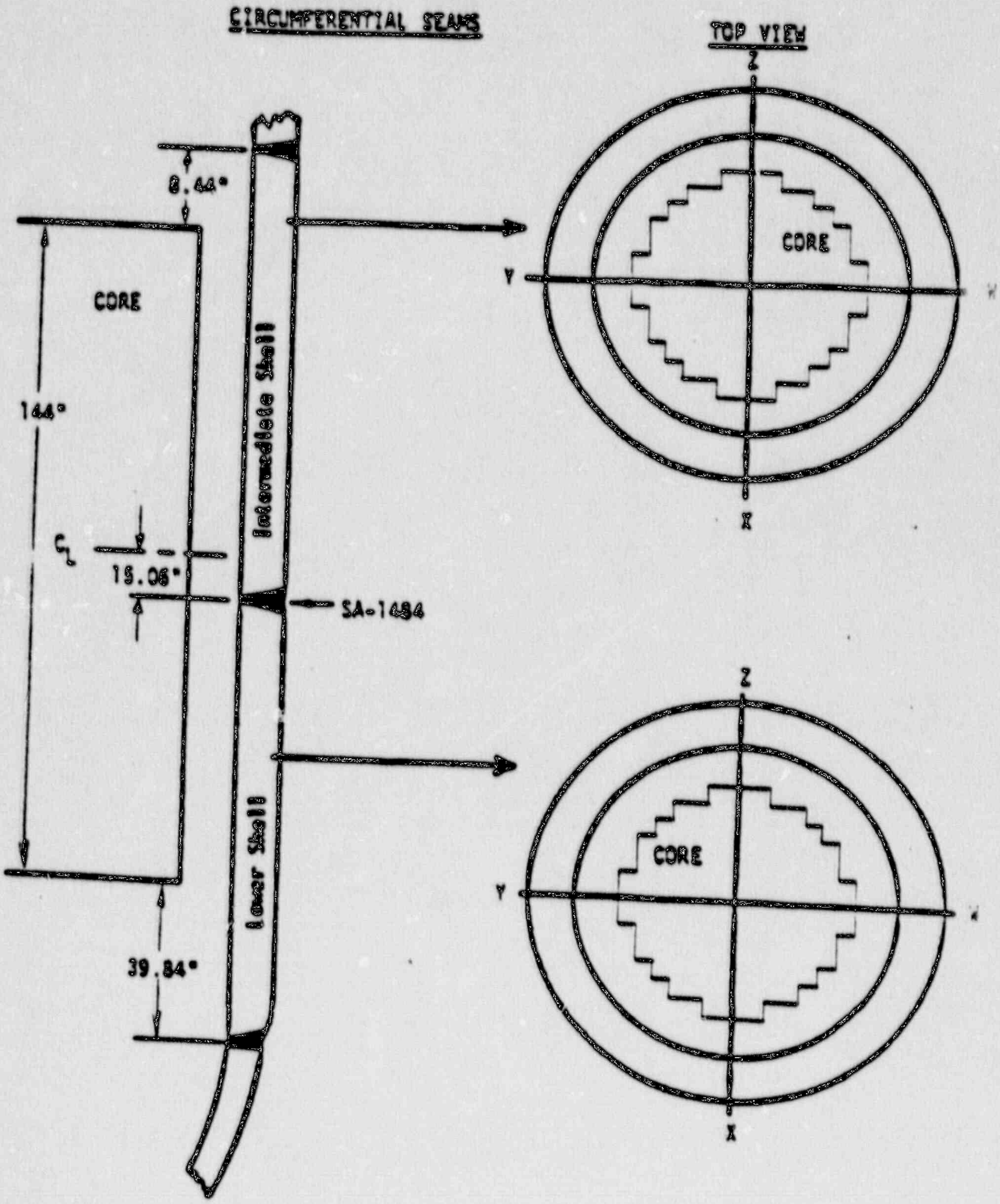


Figure II.a.1-2

IDENTIFICATION AND LOCATION OF BELTLINE REGION MATERIAL FOR THE POINT BEACH UNIT NO. 2 REACTOR VESSEL



Initial Assessment (Including Low Leakage Fuel Management)

- <u>Pressurized thermal shock</u>
  - 10CFR50.61 screening criteria will be approached only after 40 years of operation for the critical welds
  - Reference nil-ductility transition temperature projections could change in future
- <u>Low upper shelf charpy toughness</u>
  - 10CFR50 Appendix G 50 ft-lb (67.8J) limit is expected to be challenged before 40 years of operation for critical welds
  - 50 ft-lb is not of real concern, but fracture toughness degradation below 40 ft-lb is
- <u>Heatup and cooldown operational limits</u>
  - Narrow "operating windows" expected for operation beyond 40 years
  - Procedural and other operating changes could be implemented to overcome this situation

1121 021641 009

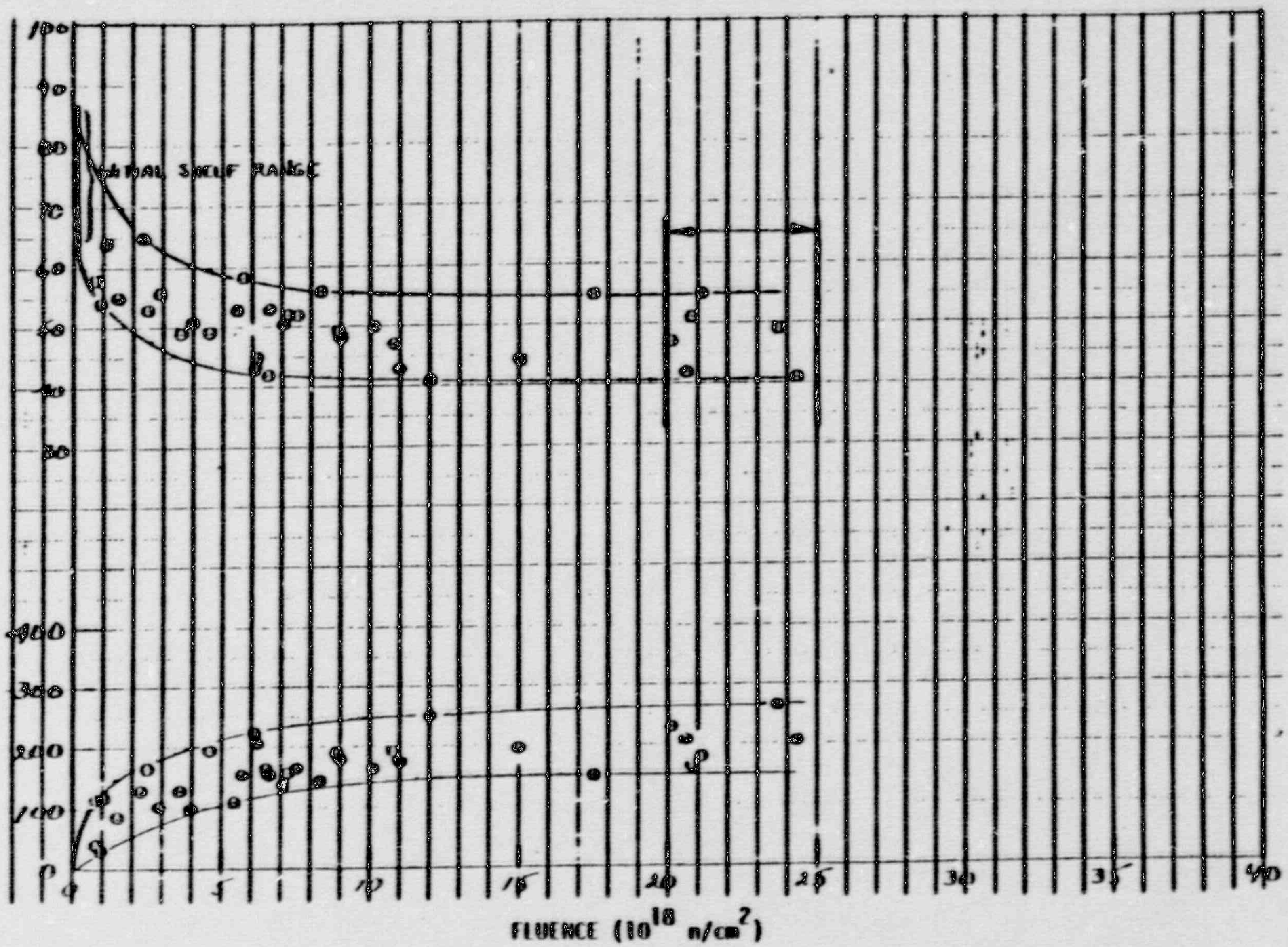


Figure II.d.2-1 SUMMARY OF LINDE 80 WELD IRRADIATION BEHAVIOR

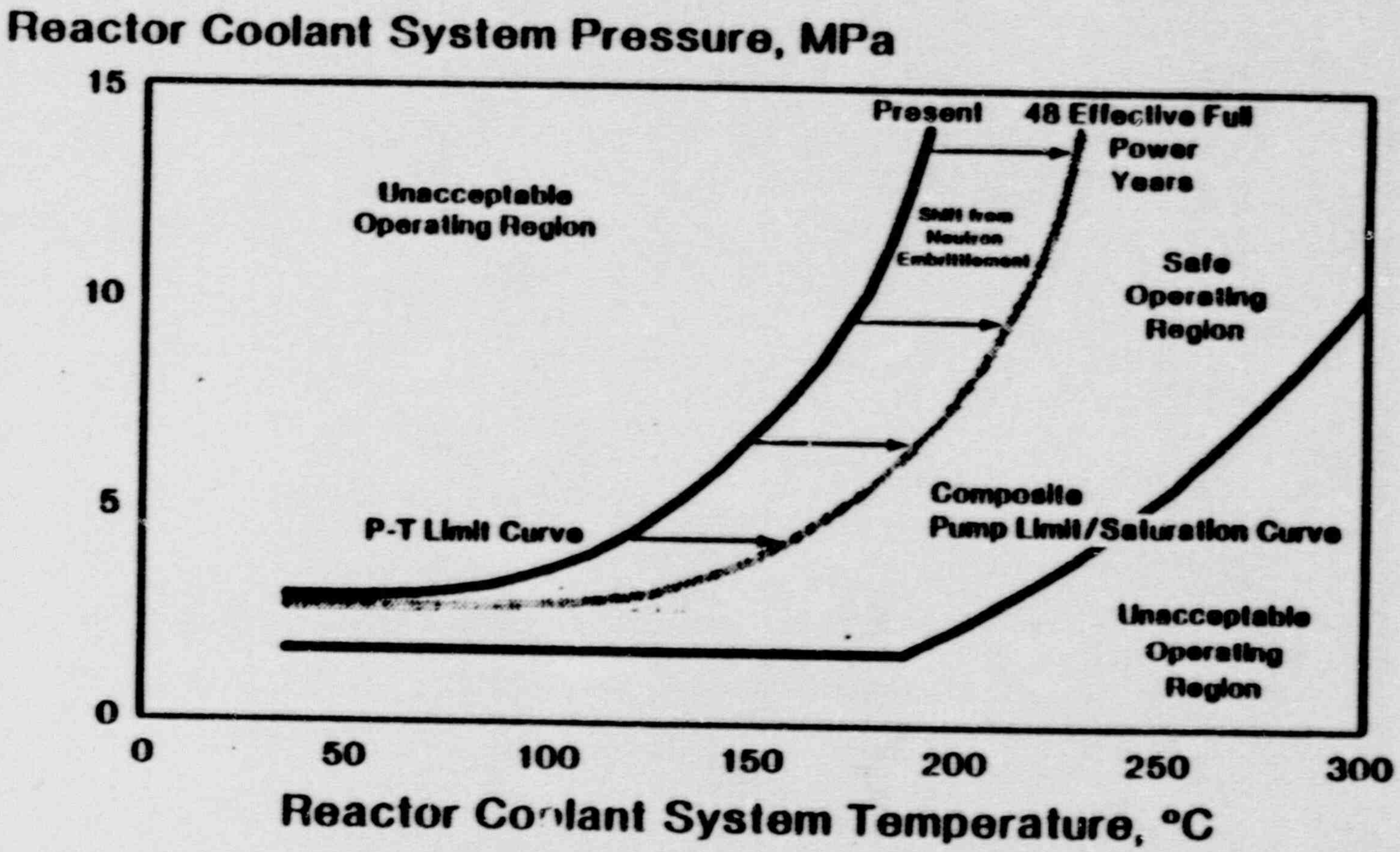


Figure II.a.2-2 Pressure - Temperature Heatup and Cooldown Operational Limit Curves

1084 D20903 006

Determination of Flux Reduction Goals

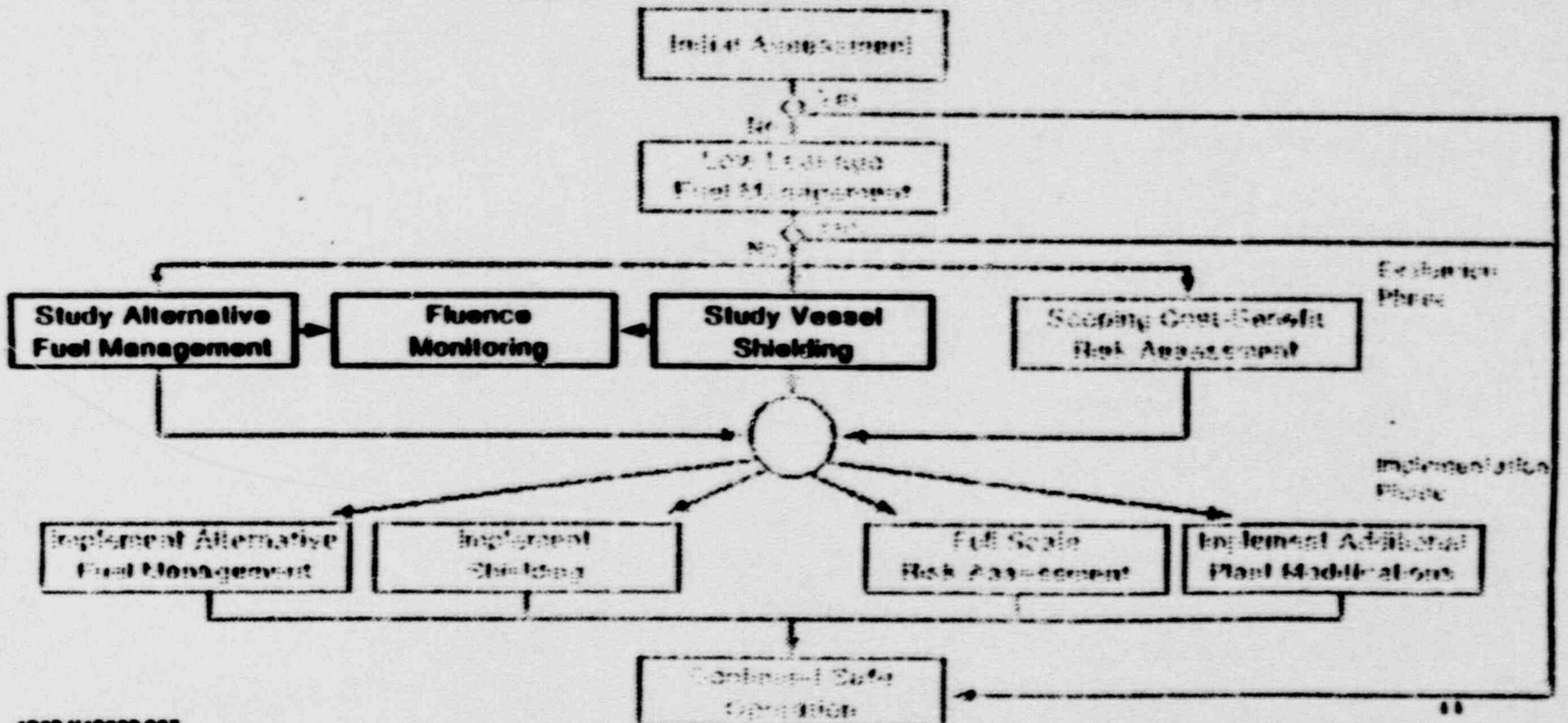
Weld	Flux Reduction Goal*
Unit 1 - Girth Weld	1.5 - 2.0
- Axial Weld in Intermediate Shell	1.0 - 1.2
- Axial Weld in Lower Shell	1.5 - 1.8
Unit 2 - Girth Weld	1.5 - 2.0

*Relative to Low Leakage Loading Pattern (L^3P)

1020 N10230 012

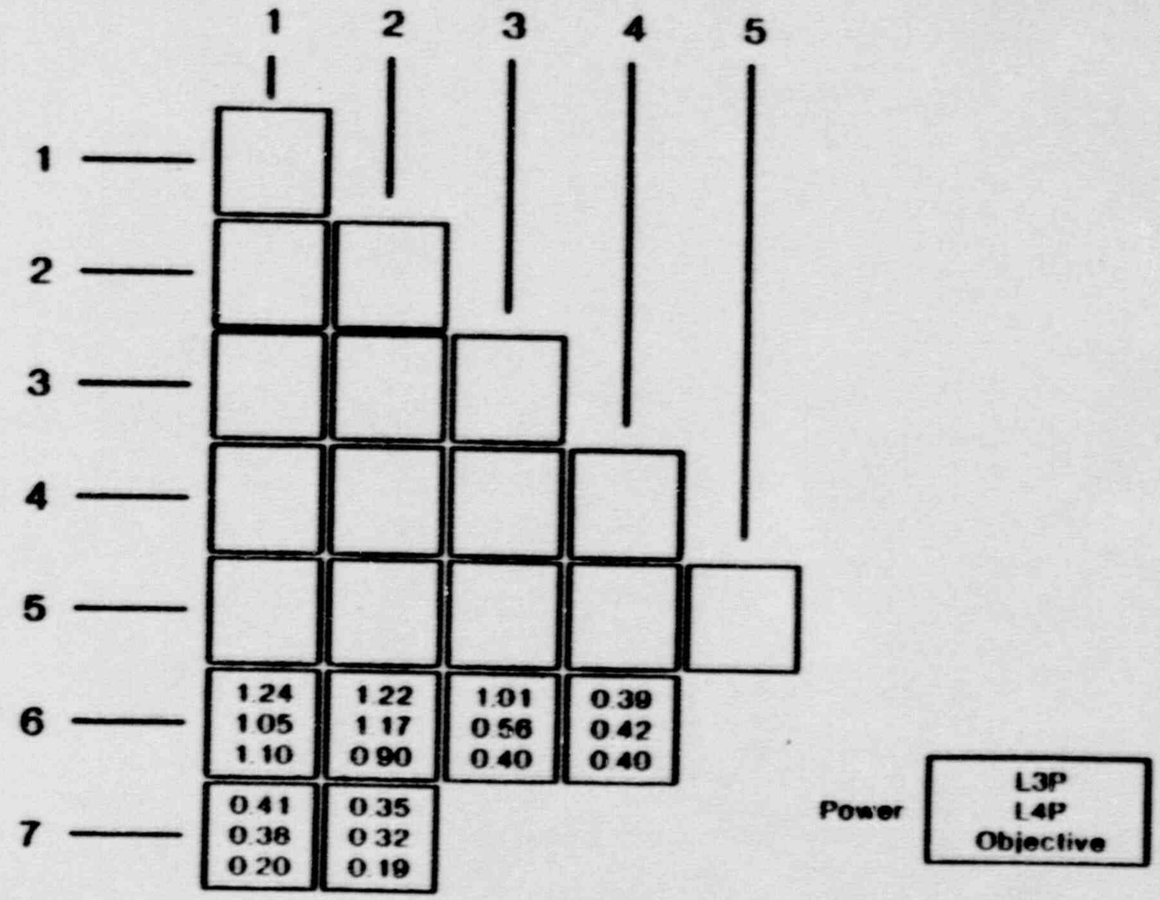
Reactor Vessel Life Extension

A Cost-Effective Approach



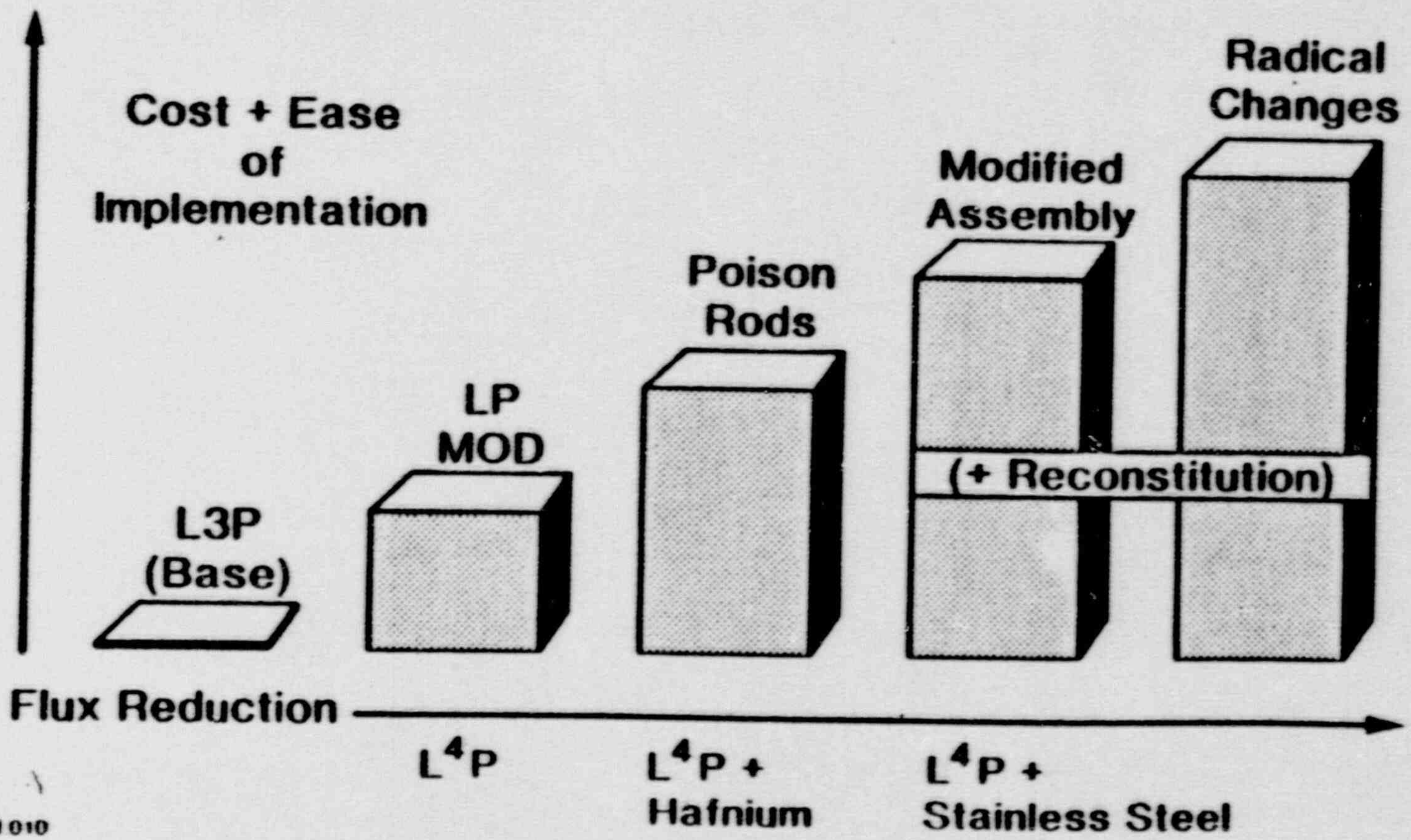
1020 N19230 037

Average Assemblywise Powers of L^3P and L^4P Patterns and the Unit 1 Target for Maximum Flux Reduction



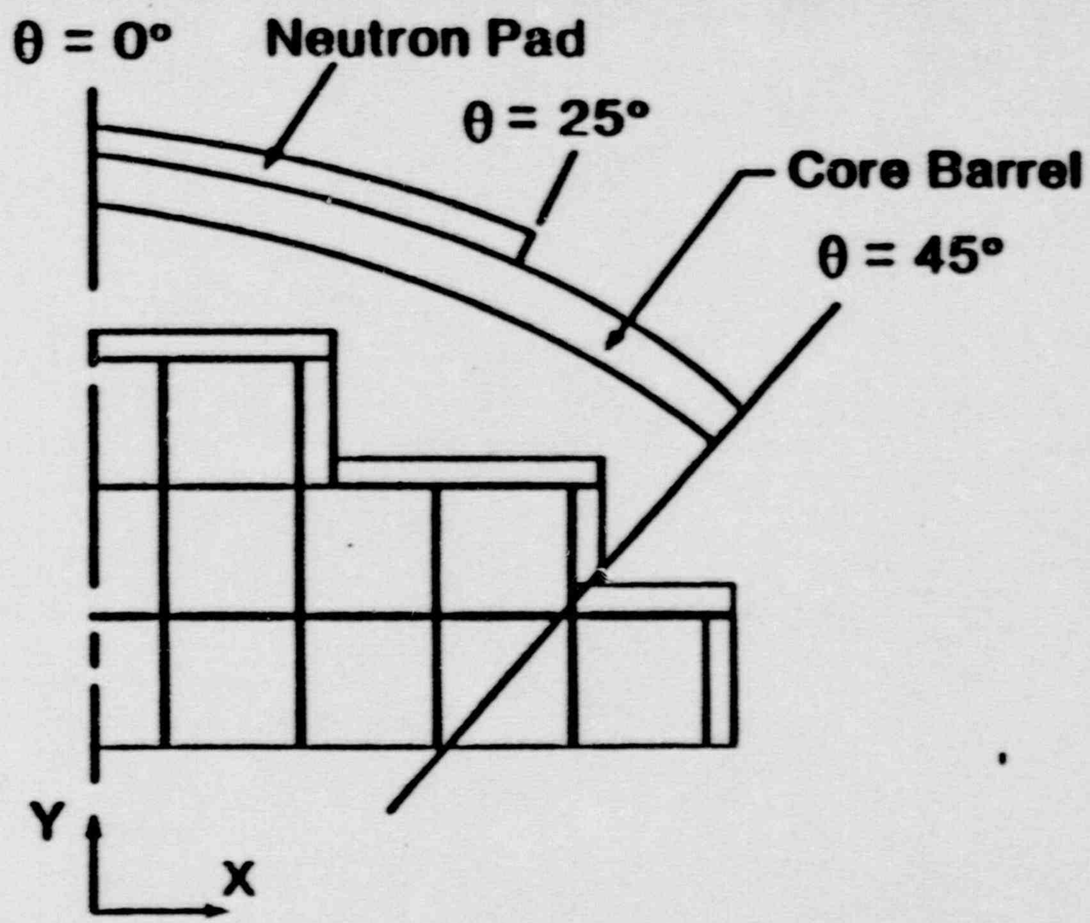
1187 D23433003

Flux Reduction



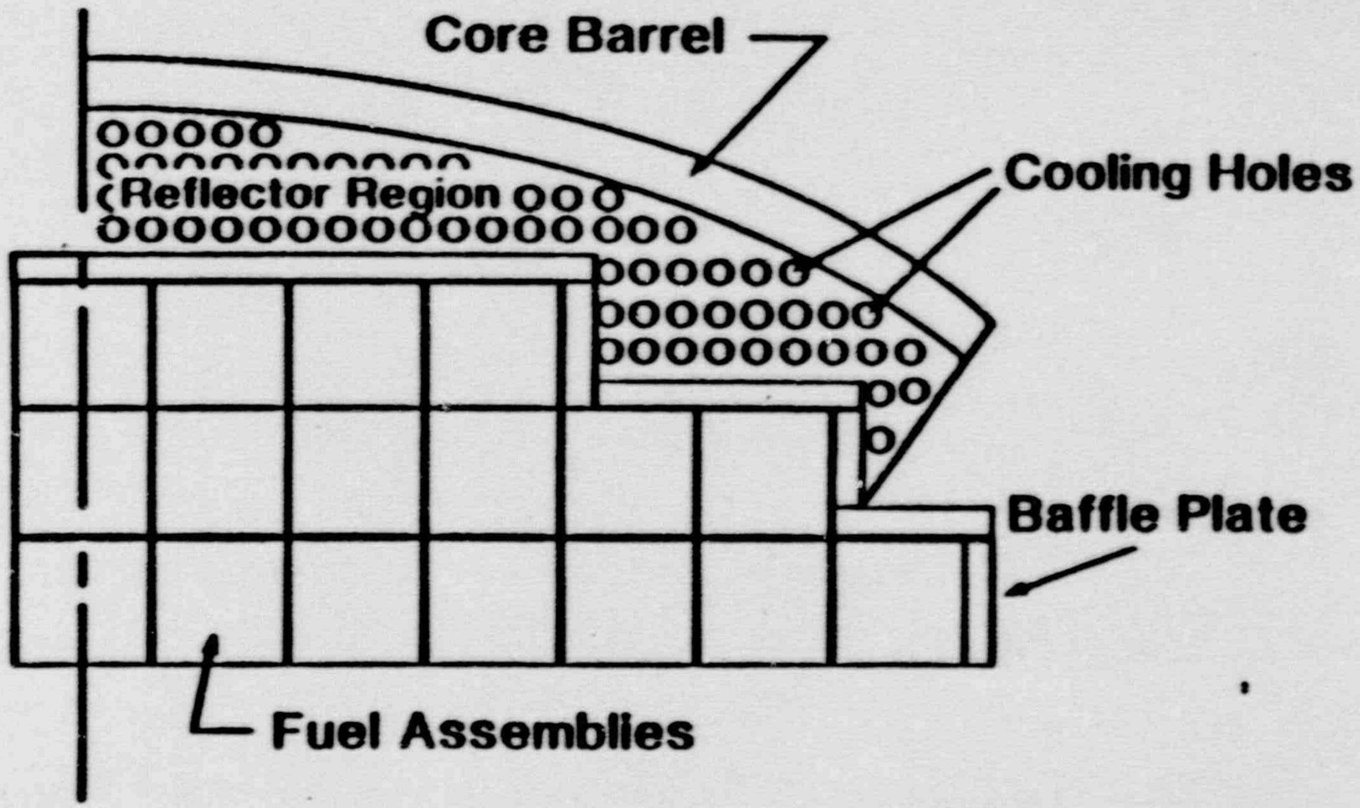
1121 021841 010

One-Eighth Core/Lower Internals Cross-Section for Standard Plants



1020 N10280 024

Typical Neutron Reflector Concept



1020 N10230 023

Results of Transport Analysis

Case	FRF Achieved 0°	FRF Achieved 15°
L^4P	Slightly < 1.3	Slightly < 1.3
L^4P + 4 Rows Stainless Steel Rods	2.1	1.6
L^4P + Part-Length HF Absorbers	2.0	1.4 (Intermed. Shell) 1.8 (Lower Shell)
Internals Replacement	3.1	2.9

FRF Goals

	0° (Circ. Weld)	15° (Long. Welds)
Unit 1	1.5 - 2.0	1.0 - 1.2 (Intermed. Shell) 1.5 - 1.8 (Lower Shell)
Unit 2	1.5 - 2.0	N/A

1187 D23433.004

Core Accident Analyses

Purpose: Raise $F_{\Delta H}$ and F_Q
Relax reactor protection system setpoints

Address: Large LOCA and UPI Model
Small LOCA w/NOTRUMP Methodogy
FSAR Chapter 14 Accidents
Miscellaneous Safety Analyses

Impacts: Fuel Cycle Economics
Plant Life Extension
Regulatory
Operational Enchancements

1187 D23433 006

Economic Evaluation of Fuel Strategies (1988 - 1994)

Case	Savings over Current L3P - %
L3P (base)	-
L4P w/stainless steel rods (contract fuel vendor)	1%*
L4p w/Hafnium inserts	3%**
L4P w/stainless steel rods (Lease MFRS)	5%*
L4P	7%**

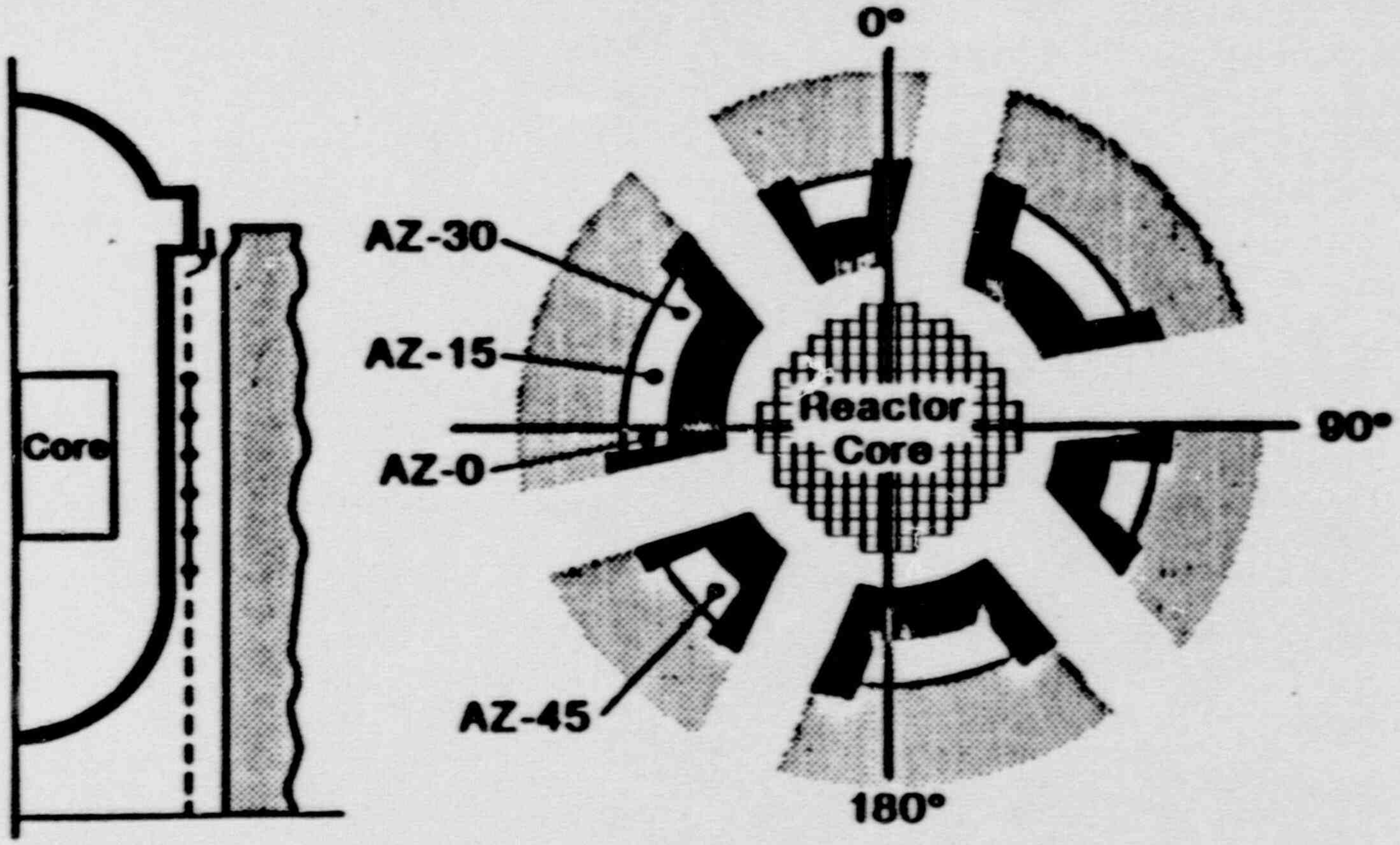
* Estimated by approximating costs in a generic L4P cycle plan.

** Calculated using corporate economic models and specific cycle plans.

1187 023433 007

Example of Fluence Monitoring

Typical Cavity Dosimetry Arrangement



V

B&W Owners Group Integrated Reactor Vessel Surveillance Program

12

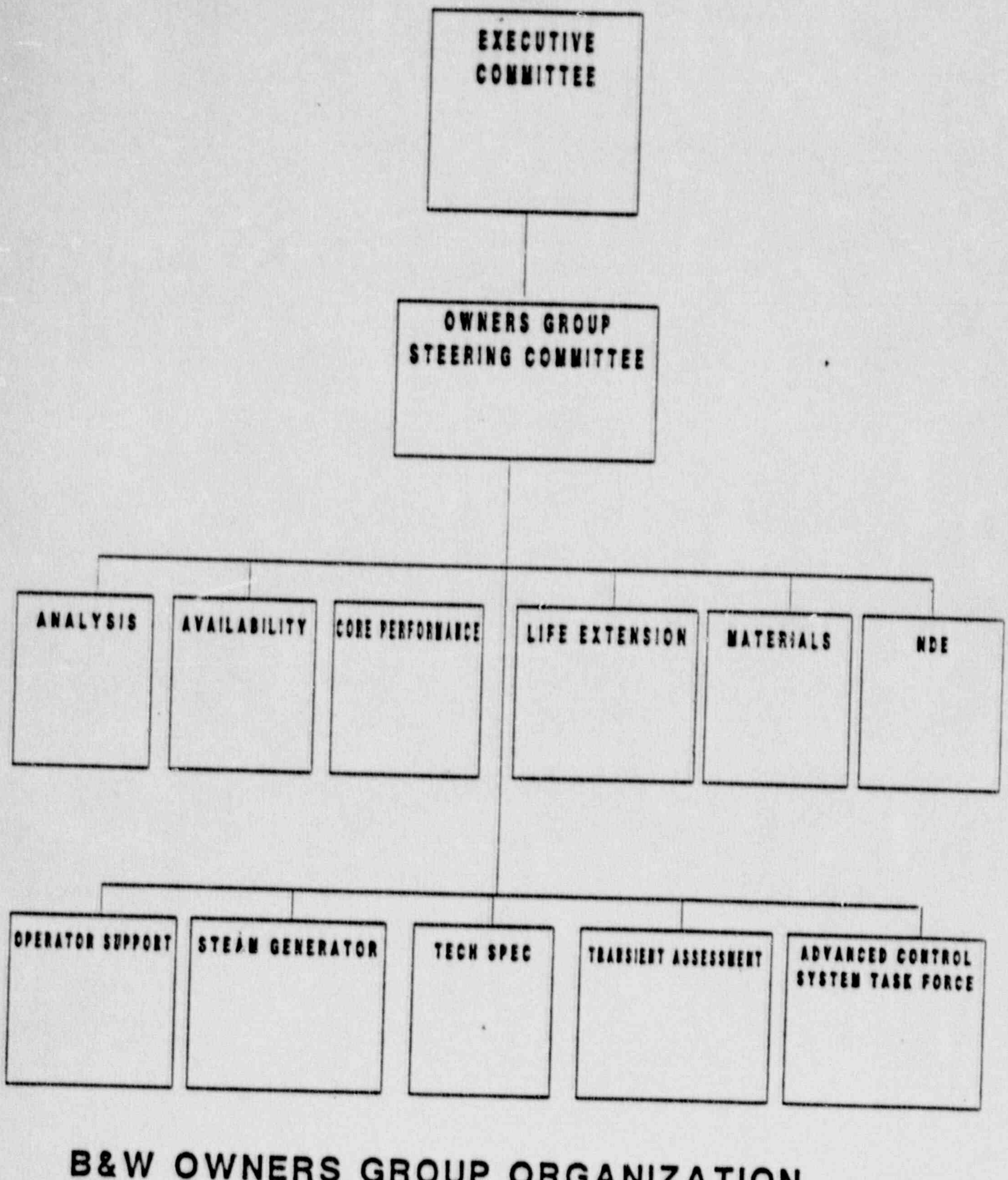
Discussion Outline

- Owners Group Reactor Vessel Working Group — Membership and Purpose
- Reactor Vessel Integrity Program — Objectives and Scope
- Master Integrated Reactor Vessel Surveillance Program — Objectives, Scope, Content

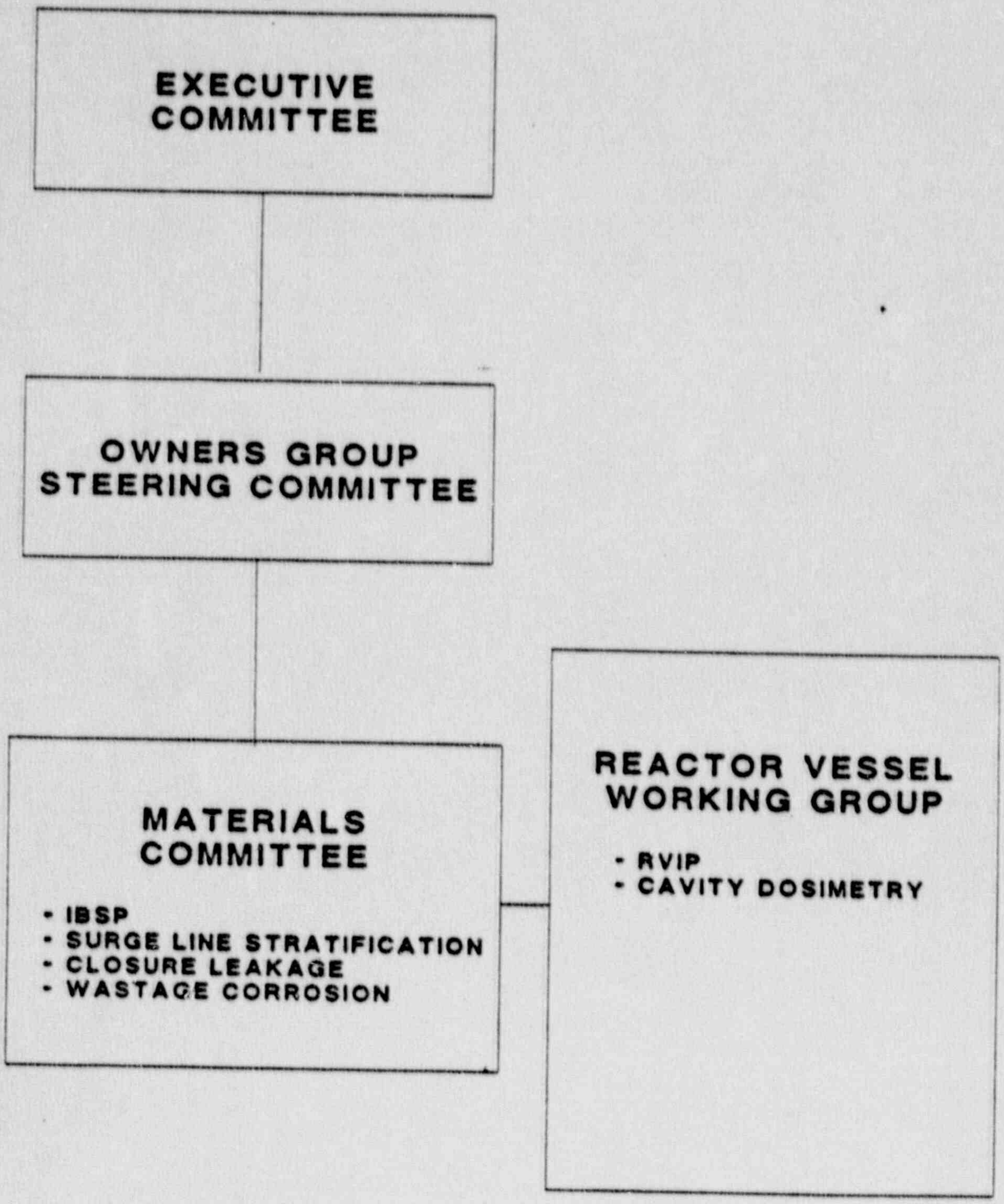
  Data Sharing for Plant Specific Application

Reactor Vessel Working Group Representatives

Arkansas Power & Light	ANO-1
Commonwealth Edison Company	Zion 1, 2
Duke Power Company	Oconee 1, 2, 3
Florida Power Corporation	Crystal River 3
Florida Power & Light	Turkey Point 3, 4
GPU Nuclear Corporation	Three Mile Island 1
Rochester Gas & Electric	Ginna
Toledo Edison Company	Davis-Besse
Virginia Power Company	Surry 1, 2
Wisconsin Electric Company	Point Beach 1, 2
Babcock & Wilcox	



B&W OWNERS GROUP ORGANIZATION


EXECUTIVE COMMITTEE
OWNERS GROUP STEERING COMMITTEE
MATERIALS COMMITTEE
- IBSP
- SURGE LINE STRATIFICATION
- CLOSURE LEAKAGE
- WASTAGE CORROSION
REACTOR VESSEL WORKING GROUP
- RVIP
- CAVITY DOSIMETRY

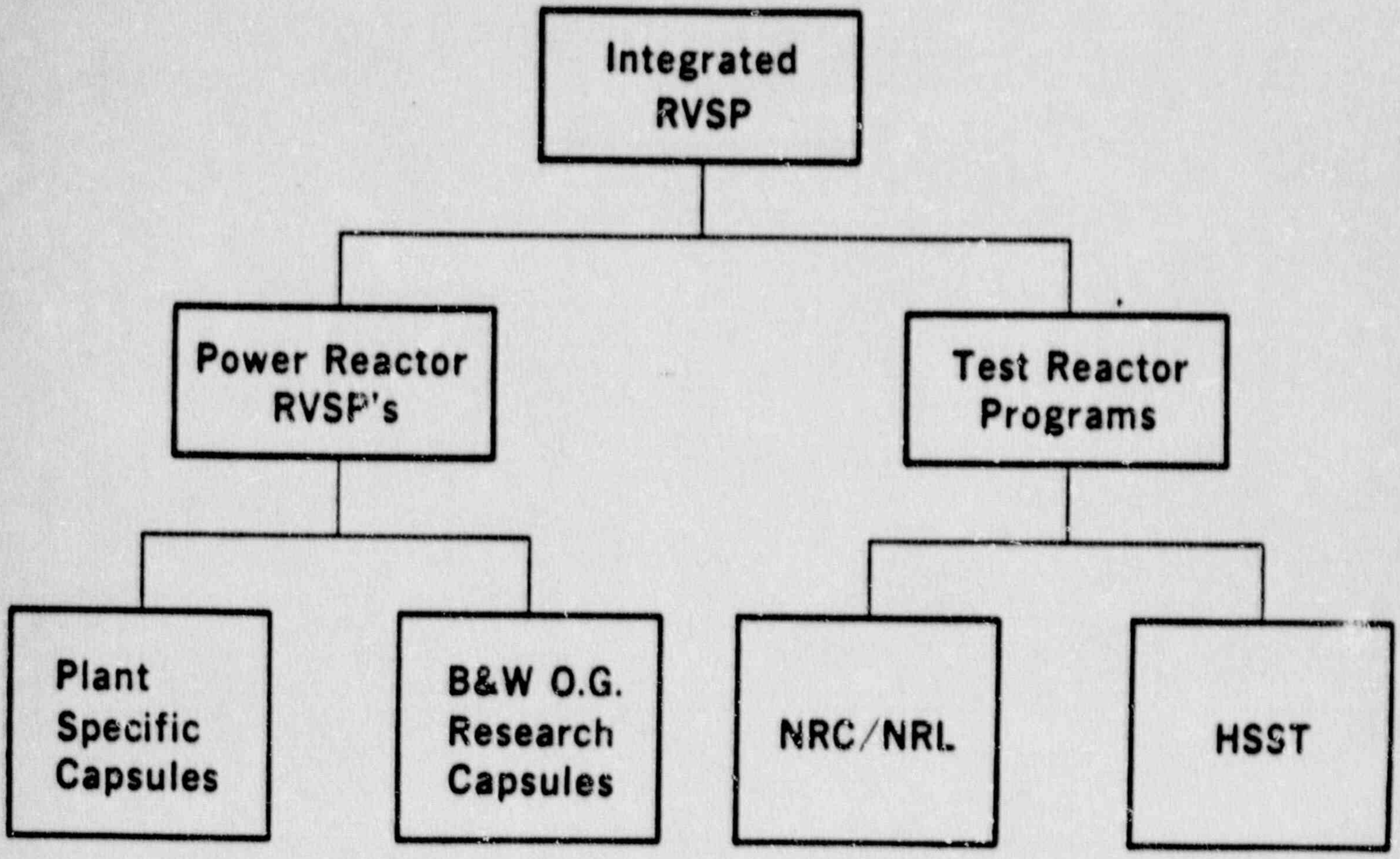
B&W OWNERS
REACTOR VESSEL MATERIALS PROGRAM

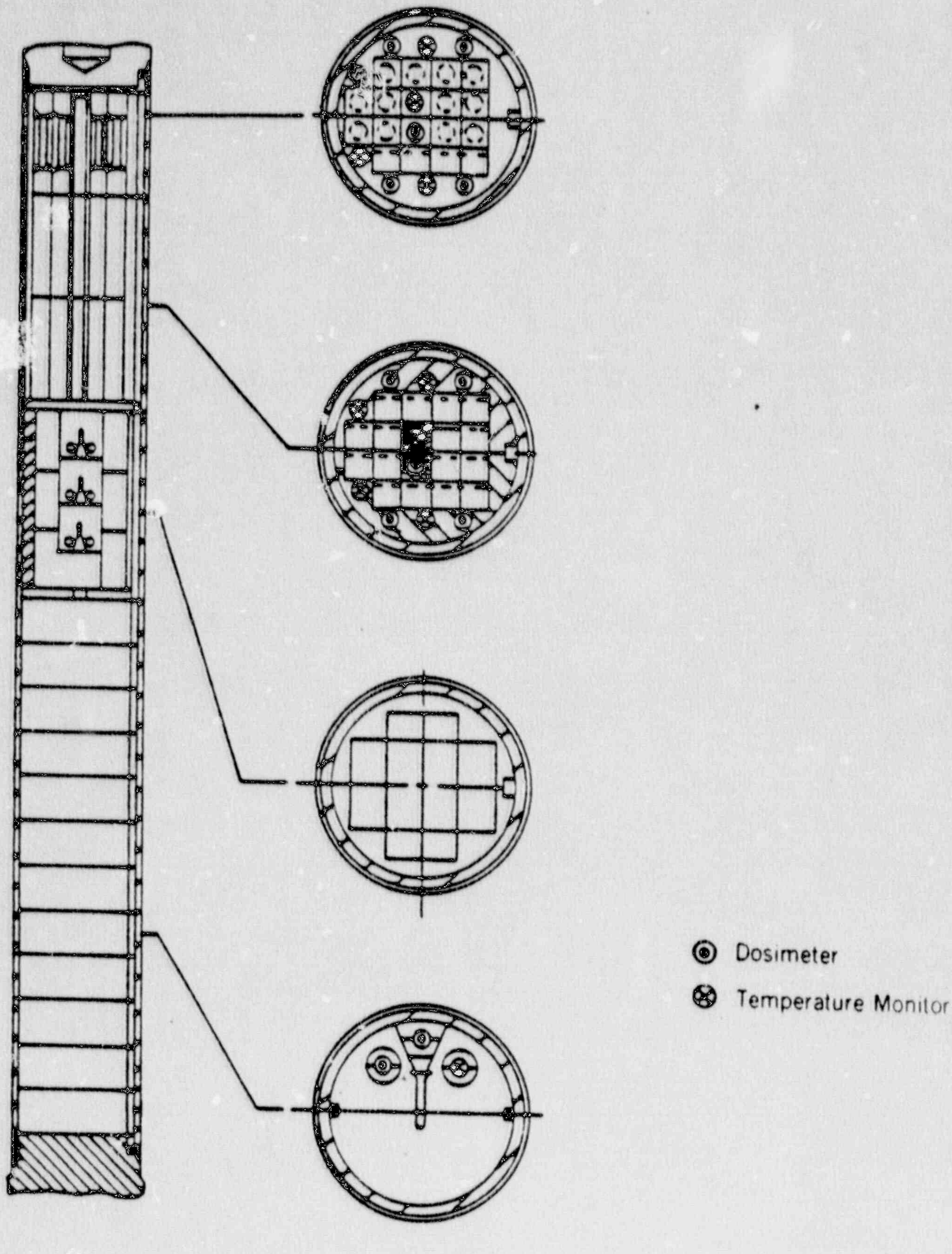
INITIATED IN 1976
INTEGRATED RVSP BASIS FOR 10CFR50 APP. H

SHORT & LONG TERM FEATURES ADDRESSING 50 FT-LB AND INTEGRITY REQUIREMENTS OF 10CFR50 APP. G

ADDED PHASES ON ATYPICAL WELD, NDE, FLUENCE ANALYSIS & DOSIMETRY, STRESS RELIEF TIME EFFECTS AND OTHER AS NEEDED

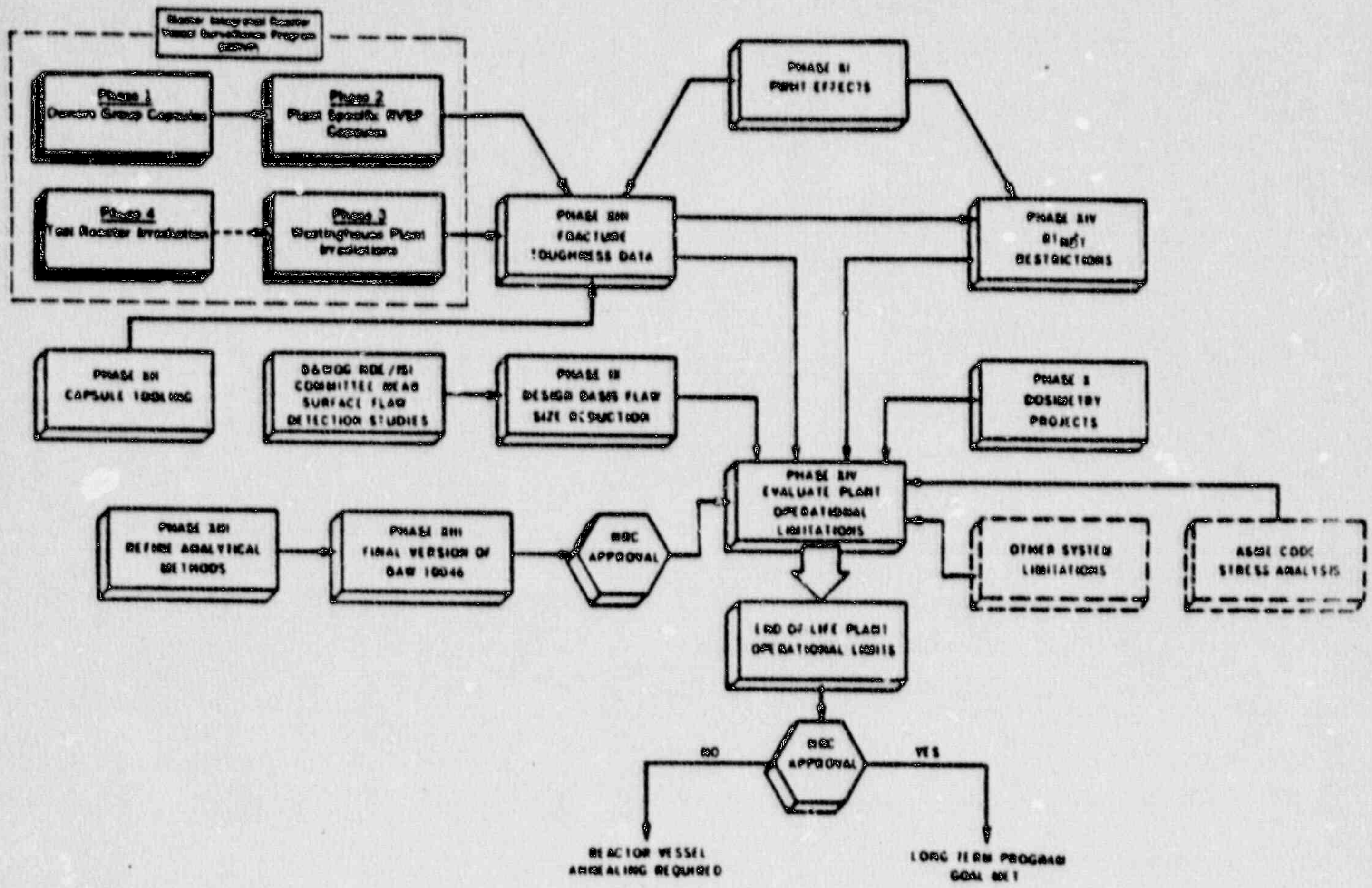
ADDED W UTILITIES WITH B&W MANUFACTURED REACTOR VESSELS


Integrated RVSP
Power Reactor RVSP's
Test Reactor Programs
Plant Specific Capsules
B&W O.G. Research Capsules
NRC/NRL
HSST



B&WOG Surveillance Capsule

Owners Group Reactor Vessel Integrity Program



PROGRAM GOALS & OBJECTIVES
FOR CONTINUATION OF
REACTOR VESSEL INTEGRITY PROGRAM

PROGRAM GOAL

TO ASSURE THE CONTINUED LICENSABILITY, REDUCE OPERATIONAL RESTRICTIONS, AND EXTEND THE USEFUL LIFE OF THE B&W FABRICATED REACTOR VESSELS.

PROGRAM OBJECTIVES

1. MINIMIZE IMPACT OF PLANT OPERATIONAL RESTRICTIONS RESULTING FROM LOW UPPER SHELF FRACTURE TOUGHNESS OF REACTOR VESSEL BELTLINE WELDS.

2. REDUCE PLANT OPERATIONAL RESTRICTIONS DUE TO IRRADIATION INDUCED CHANGES IN REFERENCE TEMPERATURE (RT_{NDT}).

3. MAINTAIN ACCEPTANCE OF THE INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM.

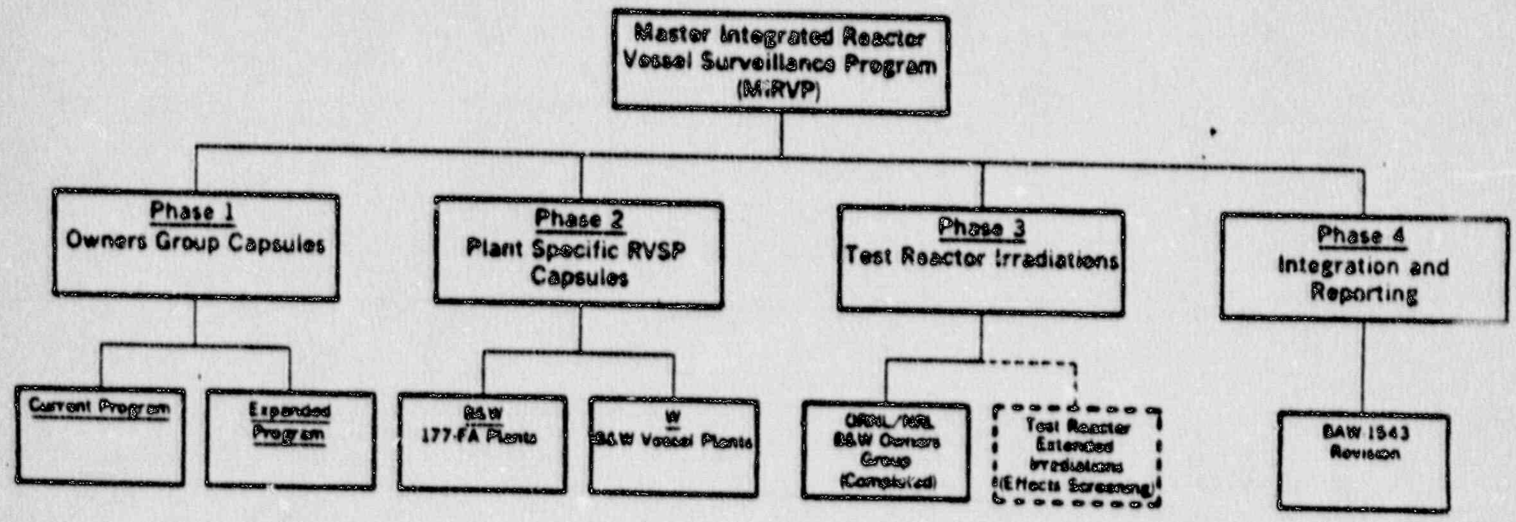
4. PROVIDE THE DATA REQUIRED TO EXTEND THE LIFE OF THE REACTOR VESSELS.

MIRVP

The Master Integrated Reactor Vessel Surveillance Program

- To assure continued licensability of 17 reactor vessels fabricated by B&W and containing "Linde 80" high copper beltline welds

Elements of Master Integrated Reactor Vessel Surveillance Program



Family of Materials

ASA Process Welds
Mn-Mo-Ni Filler Wire
Linde 80 Flux

15 Heats — Filler Wire
23 Lots — Flux
32 Total Combinations

BAW-1803
Surrogate Materials

Weld Materials in "High" Flux Locations

Weld Wire Heat	Weld Material	Plant(s)
61782	SA-847	Ginna, Point Beach-1
71249	SA-1101	Point Beach-1, Turkey Point-3 & 4
	SA-1229	Oconee-1
	SA-1769	Crystal River-3, Zion-2
72102	WF-29	Rancho Seco, Zion-2
72105	WF-70	Crystal River-3, Rancho Seco, Zion-1 & 2
72442	SA-1484	Point Beach-2
	WF-67	Oconee-3
72445	SA-1585	Oconee-1, Surry-1 & 2
1P0661	SA-775	Point Beach-1
1P0815	SA-812	Point Beach-1
8T1554	SA-1484	Surry-1
8T1762	WF-4	Surry-2, Zion-1
	WF-8	Crystal River-3, TMI-1, Zion-1
	WF-18	Crystal River-3, ANO-1
	SA-1426	Oconee-1
	SA-1430	Oconee-1
	SA-1493	Oconee-1
	SA-1580	Crystal River-3
209L44	SA-1526	Surry-1
	WF-25	Oconee-2, TMI-1
406L44	WF-112	ANO-1
	WF-154	Oconee-2, Rancho Seco
821T44	WF-182-1	Davis Besse
	WF-200	Oconee-3

Reactor Environment Considerations

- Temperature differences
  Lower T produces increased damage
  Parameter in neutronics
- Spectral differences between plants
  Flux
  Flux spectrum

Resolution:

1. Use benchmarking concept
2. Review/document plant differences
3. Use output from 1&2 to validate/correlate data base for any beltline weld

	ORIGINAL CAPSULES						ADDED CAPSULES							
Capsule Identification	C1	C2	D1	D2	T1	T2	A1	A2	A3	A4	A5	L1	L2	W1
Host Reactor	CR	CR	DB	DB	CR	CR	CR or DB							Surry
	Fluence, XE19						Fluence, XE19							
	0.6	1.7	0.8	1.7	0.8	1.7	3.0	3.0	1.7	3.0	1.7	1.7 + A	1.7 + A + R	1.0*
WF-25**	X9				X6	XX	X			X		X	X	
WF-67	X	X					X			X		X	X	
WF-70(B)									X					
WF-70(N)		X	X	X	X		X			X		X	X	X
WF-209-1(S)											X			
WF-112			X	X										
WF-182-1									X					
SA-1101								X						
SA-1101(S)											X			
SA-1135			X	X				X						
SA-1263(S)											X			
SA-1484									X					
SA-1526					X	X		(X)						X
SA-1585	X	X						X						X

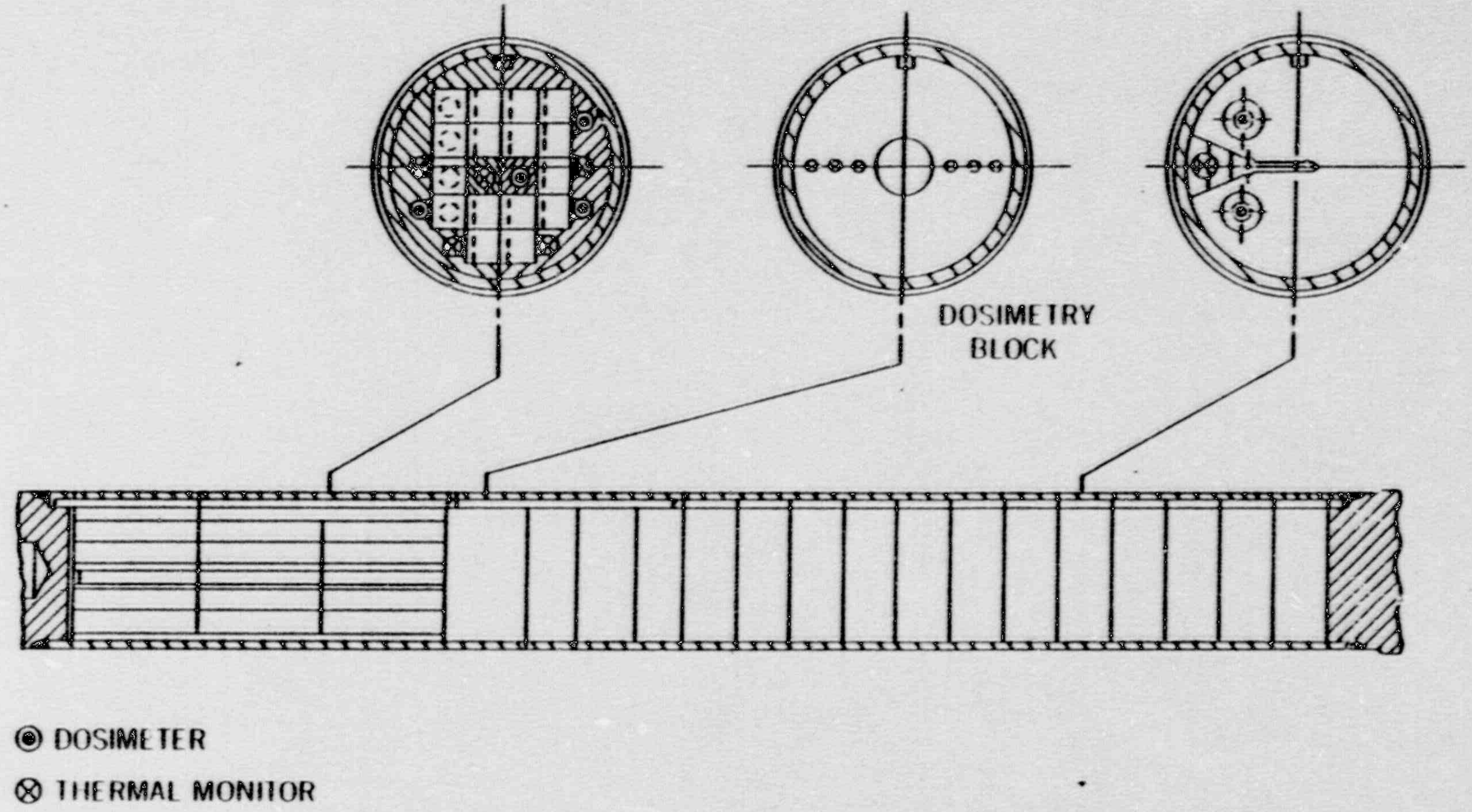
B = Beltline
N = Nozzle dropout
S = Surveillance weld

A = Anneal
R = Reirradiate
() = Material will be included in capsule if space is available

*Fluence in this irradiation only. Reconstituted specimens receive greater (total) exposures.

**WF-25 material taken from two sources: NSS-6 and NSS-9. Source noted by "6" and "9."

Figure 3-19a HUPCAP Capsules A1, A2, A3, A4, L1, and L2



Owners Group Program

Capsule	Material	Existing Program Fluence	Existing Program Test Date	Proposed Change Fluence	Proposed Change Test Date
CR3-LG1	SA-1585 WF-25 WF-67	0.6E19	Complete	—	—
CR3-LG2	SA-1585 WF-67 WF-70	1.4E19	1992	1.7E19	1994
DB1-LG1	SA-1135 WF-70 WF-112	0.8E19	Complete	—	—
DB1-LG2	SA-1135 WF-70 WF-112	1.4E19	1993	1.7E19	1995
TMI2-LG1	SA-1526 WF-25(6) WF-70	0.8E19	1992	—	—
TMI2-LG2	SA-1526 WF-25(6) WF-25(9)	1.4E19	1997	—	—

Additional Owners Group Capsules

Capsule	Material	Fluence	Test Date
A1	WF-25 WF-67 WF-70(N)	3.0E19	2008
A2	SA-1101 SA-1135 SA-1585 (SA-1526)	3.0E19	2008
A3	SA-1101 SA-1484 WF-70(B)	1.7E19	2000
A4	WF-25 WF-67 WF-70(N)	3.0E19	2008
A5 (Irrad.)	SA-1101 SA-1263 WF-209-1	1.7E19	2000
L1	WF-25 WF-67 WF-70(N)	1.7E19 +A	2000
L2	WF-25 WF-67 WF-70(N)	1.7E19 +A +0.8E19	2005
W1	SA-1526 SA-1585 WF-70(N)	1.0E19	1997

Integration and Reporting

- "BAW-1543" format for entire MIRVP
  - Referenced in tech specs
  - Periodic updates
- Quarterly reporting at OG meetings
- Frequent NRC proactive reporting, as appropriate
- Special reports, as agreed with owners

Data Sharing For Plant Specific Application

Point Beach 1 Reactor Vessel

Sources of Weld Data

Beltline Weld	RVSP or Test Reactor Weld	Wire Heat	Data Source	Fracture Specimens	Fluence, E19
SA 1101	SA 1101	71249	TP3 Caps. T	WOL	.57
			TP3 Caps. V	WOL	1.2
			TP3 Caps. X	WOL	[2.8]
			HSST 2	0.5 - 4T CT	0.6 - 1.4
			HUPCAP-A2	0.936T CT	[3.0]
			HUPCAP-A5	0.936T CT	[3.0]
	SA 1769		HSST 2	0.5 - 4T CT	1.1 - 1.4
	SA 1094		TP4 Caps. T	WOL	0.75
SA 847	SA 1036	61782	Ginna Caps. V	WOL	0.6
			Ginna Caps. R	WOL	1.2
			Ginna Caps. T	WOL	1.8
			Ginna Caps. S	WOL	[4.1]
			Ginna Caps. P	WOL	[4.1]
			Ginna Caps. N	WOL	[6.1]
			HSST 2	0.5 - 4T CT	0.6 - 1.6
	SA 1135		SUPCAP-D1	0.4 - 0.936T CT	0.8
			SUPCAP-D2	0.4 - 0.936T CT	[1.7]
			HUPCAP-A2	0.936T CT	[3.0]
SA 775					
SA 812					

Point Beach 2 Reactor Vessel
Sources of Weld Data

Beltline Weld	RVSP or Test Reactor Weld	RVSP or Test Reactor Wire Heat	RVSP or Test Reactor Data Source	Fracture Specimens	Fluence, E19
SA 1484	SA 1484 WF 67	72442	HUPCAP-A3	0.936T CT	[1.7]
	↓	↓	SUPCAP-C1	0.4 - 0.936T CT	0.6
			SUPCAP-C2	0.4 - 0.936T CT	[1.7]
			HUPCAP-A4	0.936T CT	[3.0]
			HUPCAP-L1	0.936T CT	[1.7+Anneal]
			HUPCAP-L2	0.936T CT	[1.7+Anneal +Reirrad]

Summary

- MIRVP consolidates RVSPs for all B&W-Fabricated Operating PWRs
- MIRVP adds capsules to obtain higher fluence fracture toughness for Linde 80 welds
  - Life extension
  - Annealing response
- Appendix H requirements for IRVSPs believed to be satisfied

VI

Review of Materials Data Applicable to Point Beach Units 1 and 2

Topics to be Reviewed

Charpy Shift Data

Charpy Upper-Shelf Energy Data

Fracture Toughness Data

Fracture Toughness Correlation

Summary Evaluation of Reactor Vessel Weld End-Of-Life (32 EFPY) Fracture Toughness Point Beach Units 1 and 2

Weld Number	Weld Location	Chemical Composition (wt%)[b]		Neutron Fluence (n/cm^2)		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[e]		Initial USE (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)	Margin (°F)[e]	Inside Surface	T/4		RG 1.99/2	BAW-1803/1
Point Beach Unit 1												
SA-1426	NB/IS	0.29	0.55	3.18E18	2.15E18	-6[d]	68	188	170	70[d]	49	58
SA-812	IS-Longitudinal (Inside 27%)	0.17	0.52	1.79E19	1.21E19	-6[d]	68	222	208	70[d]	48	51
SA-775	IS-Longitudinal (Outside 73%)	0.19	0.63	--	--	--	--	---	---	--	--	--
SA-1101	IS/LS	0.26	0.60	2.28E19	1.54E19	10	28	258	240	65[d]	36	46
SA-847	LS-Longitudinal	0.25	0.54	1.61E19	1.09E19	-6[d]	68	252	234	70[d]	42	54
Point Beach Unit 2												
CE[a]	NB/IS	0.27	0.90	3.57E18	2.42E18	-56[c]	68	178	155	100	67	--
SA-1484	IS/LS	0.24	0.60	2.37E19	1.60E19	-6[d]	68	275	258	70[d]	40	52

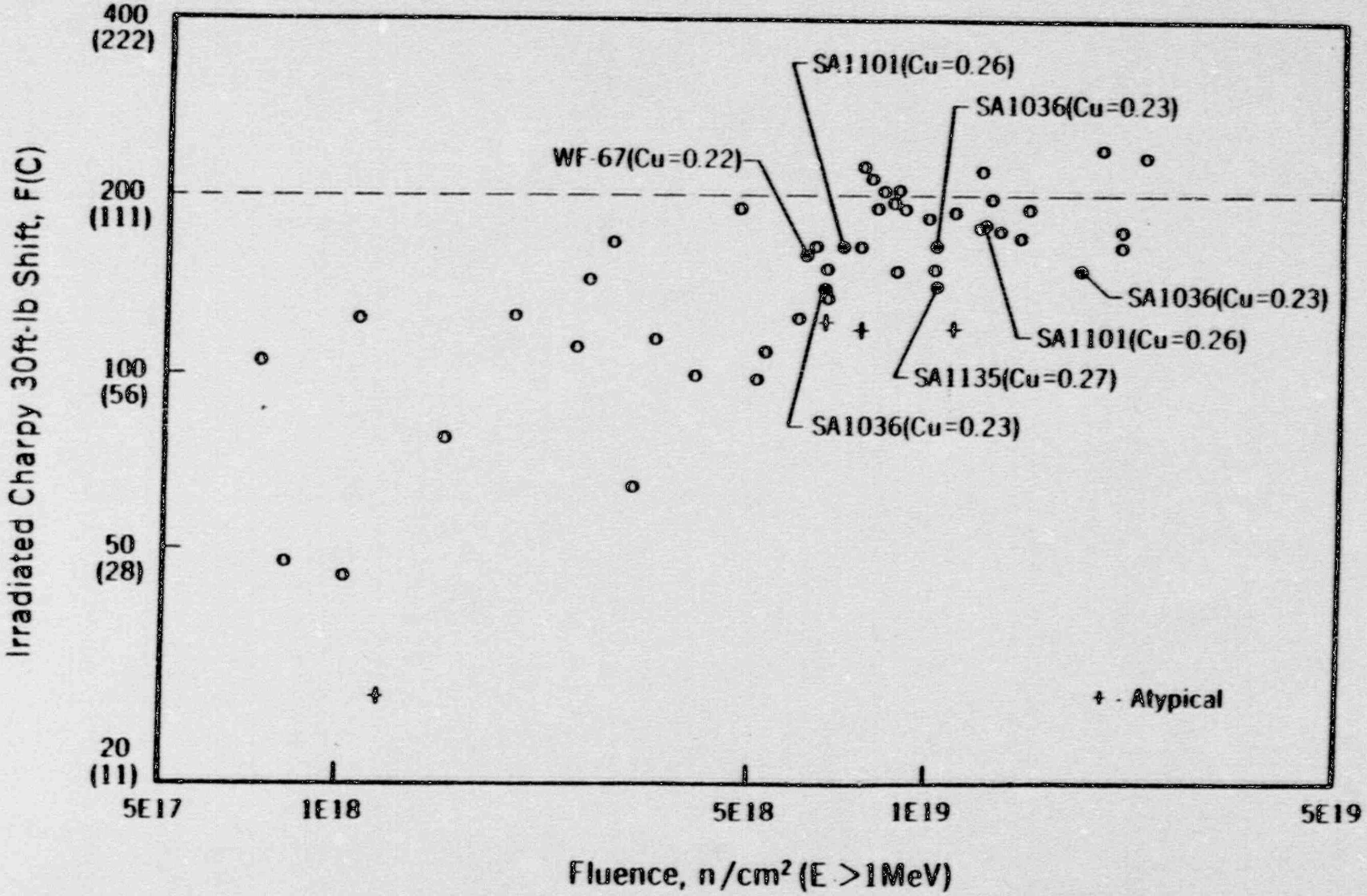
[a] Weld fabricated by Combustion Engineering, Chattanooga, TN

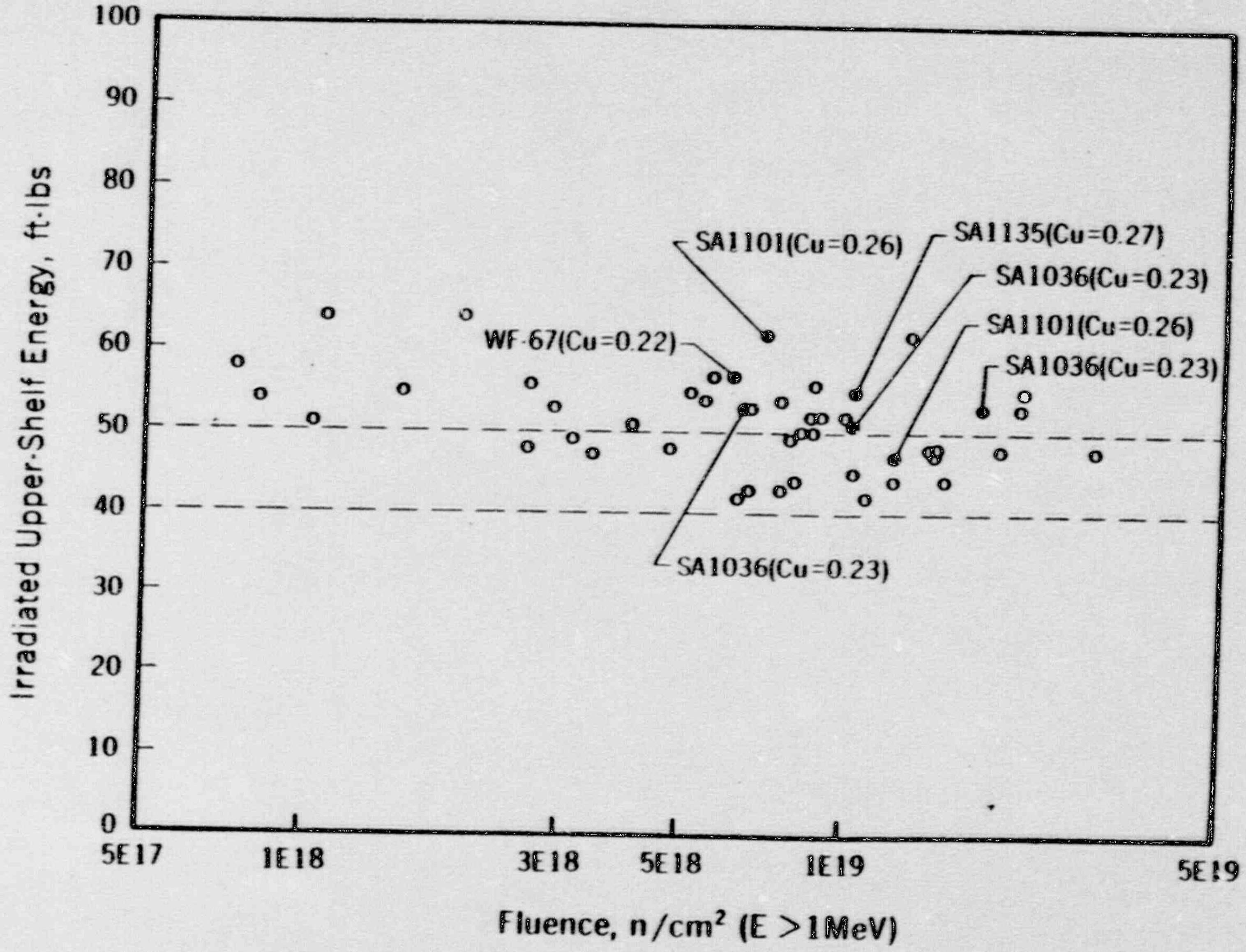
[b] BAW-1500, September 1978

[c] Estimated on basis of reactor vessels fabricated by Combustion Engineering at about the same time

[d] BAW-1803, January 1983

[e] Regulatory Guide 1.99, Revision 2, May 1988


400
(222)
200
(111)
100
(56)
50
(28)
20
(11)
Irradiated Charpy 30ft-lb Shift, F(C)
5E17
1E18
5E18
1E19
5E19
Fluence, n/cm² (E >1MeV)
SA1101(Cu=0.26)
SA1036(Cu=0.23)
WF-67(Cu=0.22)
SA1036(Cu=0.23)
SA1101(Cu=0.26)
SA1135(Cu=0.27)
SA1036(Cu=0.23)
+ - Atypical



Irradiated Upper-Shelf Energy, ft-lbs
100
90
80
70
60
50
40
30
20
10
0
5E17
1E18
3E18
5E18
1E19
5E19
Fluence, n/cm² (E >1MeV)
SA1101(Cu=0.26)
SA1135(Cu=0.27)
SA1036(Cu=0.23)
SA1101(Cu=0.26)
SA1036(Cu=0.23)
WF-67(Cu=0.22)
SA1036(Cu=0.23)

VII

APPENDIX G FRACTURE TOUGHNESS MARGINS

- o BACKGROUND - NRC REQUEST TO ASME FOR CRITERIA
- o CURRENT PROPOSED CRITERIA
- o OVERALL APPROACH TO ESTABLISH MARGINS
- o MARGINS FOR SERVICE LEVEL C AND D
- o MARGINS FOR SERVICE LEVEL A AND B, INCLUDING DISCUSSION OF RELATION TO ATWS EVENTS

CURRENT PROPOSED CRITERIA
(8-10-89)

SERVICE LEVEL A AND B

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

$$@\ J_{applied} = J_{material}$$

$$SF < 2.0 \text{ on pressure loads}$$

SERVICE LEVEL C AND D, INCLUDING PTS CONDITIONS

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

$$@\ J_{applied} = J_{material}$$

OR

Integrated Frequency of Through-Wall Penetration
Must Be Less Than 5×10^{-6} Per Reactor-Year

OVERALL APPROACH TO ESTABLISH MARGINS

The Probability of Vessel Failure is estimated as -

$$P_f = P1 \times P2 \times P3$$

where -

P1 = probability (per vessel year) that a given transient will occur

P2 = probability that the vessel has a flaw larger than the postulated size

P3 = probability that the fracture toughness is less than the assumed value

For Service Level A and B - P1 = f (SF x Pressure)

For Service Level C and D - P1 = Frequency of PTS Events

MARGINS FOR SERVICE LEVELS C AND D

Frequency of Event (P1): For PTS events leading to severe challenges, $P1 = 10^{-4}$ to 10^{-3} events per reactor year (Source: Westinghouse Owners Group Work)

Probability of a Flaw (P2): Most recent and most widely accepted work on this subject is the revised Octavia distribution

P2 (For a flaw > 1.0" deep) = 4.63×10^{-4} per 6 foot weld

P2 (For a flaw > 0.5" deep) = 4.68×10^{-3} per 6 foot weld

Using a $P_f = 5 \times 10^{-6}$ per reactor year (consistent with RG 1.154) -

$$5 \times 10^{-6} = 10^{-3} \times P2\ (\text{Flaw}) \times P3\ (\text{Low Toughness})$$

Thus, $$5 \times 10^{-3} = P2\ (\text{Flaw}) \times P3\ (\text{Low Toughness})$$

For Flaws > 1.0" $5 \times 10^{-3} = 2.4 \times 10^{-3} \times$ P3 (Low Toughness) → OK

For Flaws > 0.5" $5 \times 10^{-3} = 2.4 \times 10^{-2} \times$ P3 (Low Toughness) → OK if P3 < 0.2

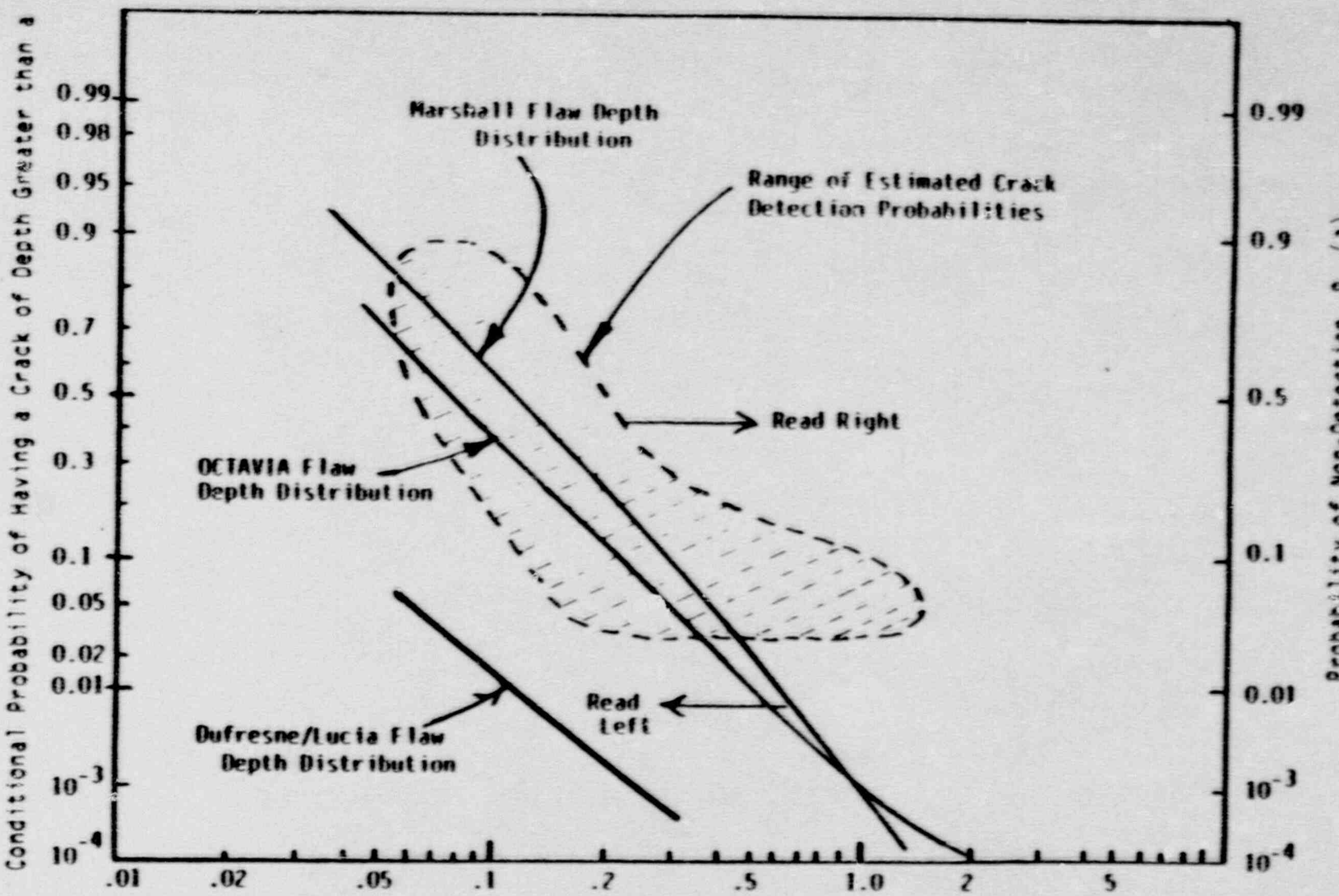


FIGURE 1. Flaw Size Probabilities Compared to Detection Probabilities