Southern California Edison Company

P.O. BOX 800
2244 WALNUT GROVE AVENUE
ROSEMEAD, CALIFORNIA 91770

September 3, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Albert Schwencer
Acting Branch Chief
Licensing Projects Branch 3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

Enclosed are sixty-three copies of responses to NRC questions asked in NRC letters dated May 23, 1980, and June 16, 1980. Direct distribution of these responses will be made as part of the Amendment 21 distribution and will be in accordance with the service list provided by SCE's letter of October 29, 1979. An affidavit attesting to the fact that distribution has been completed will be provided within ten days of docketing of Amendment 21.

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Also enclosed are, 1) three (3) copies of the proprietary Combustion Engineering document CEN-135(S)-P referenced in the response to question 221.18 (Copy Nos. 000001, 000002, 000003), 2) three (3) copies of the non-proprietary version of CEN-135(S)-P, and 3) an affidavit setting forth the basis on which the information may be held from public disclosure by the Commission and addressing specifically the considerations listed in paragraph (b) (4) of Section 2.790 of the Commission's regulations.

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Accordingly, it is respectfully requested that the information which is proprietary to Combustion Engineering, Inc. be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. If you should have any questions concerning the proprietary nature of material transmitted herewith, please address these questions directly to:

8009050302

Mr. A. E. Scherer
Licensing Manager (9438-401)
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

We also request that you provide a copy of any questions concerning the proprietary nature of this submittal to SCE.

In addition, you recently requested that we advise you of the present cost of capital for the Applicants. The present cost of capital for the Applicants is 15% per annum, which is the appropriate value to be used in any cost-benefit studies concerning San Onofre Units 2 and 3. The Environmental Report will be amended to include this present cost of capital.

Please let me know if you have any questions.

Very truly yours,

K P Baskin

K. P. Baskin
Manager of Nuclear Engineering,
Safety, and Licensing

Enclosure

AFFIDAVIT PURSUANT

TO 10 CFR 2.790

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, A. E. Scherer depose and say that I am the Director, Nuclear Licensing of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations and in conjunction with the application of Southern California Edison Company and San Diego Gas and Electric Corporation, for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

CEN-135(S)-P, Response to NRC Question 221.18 on SONGS Units 2 and 3 FSAR.

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure is a description of functional changes made to the Core Protection Calculator System design, which is owned and has been held in confidence by Combustion Engineering.

2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a substantial competitive advantage to Combustion Engineering.

3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F.M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject documents herein are proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:

a. A similar product is manufactured and sold by major pressurized water reactors competitors of Combustion Engineering.

b. Development of this information by C-E required thousands of man-hours of effort and hundreds of thousands of dollars. To the best of my knowledge and belief a competitor would have to undergo similar expense in generating equivalent information.

c. In order to acquire such information, a competitor would also require considerable time and inconvenience related to development of analytical methods and computer models.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of changes based on operational experience with the Core Protection Calculator system at Arkansas Nuclear One Unit-2 Cycle 1 and general algorithm improvements, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

A. E. Scherer

Director

Nuclear Licensing

Sworn to before me

this 8th day of August, 1980

Lisa G. Waicunas

Notary Public

LISA G. WAICUNAS, NOTARY PUBLIC
State of Connecticut No. 54492
Commission Expires March 31, 1983

RESPONSES TO NRC QUESTIONS

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 & 3

SEPTEMBER 3, 1980

QUESTION	SUBJECT
121.18 - 121.24	Reactor Vessel Materials Appendix "G" Compliance
121.26	Reactor Coolant Pump Flywheel Integrity
221.18 & 221.20	Core Protection Calculator
212.159	Loss of Component Cooling Water to Reactor Coolant Pumps
331.19	Spent Fuel Transfer Tube Shielding

QUESTION 121.18

In Tables 5.3-16 through 5.3-20, it is stated that SA 540 Grade B24 steel is used for reactor vessel bolting material. Paragraph I.C, Appendix G, 10 CFR Part 50, requires that bolting and other types of fasteners not have a specified minimum yield strength greater than 130 ksi. Therefore, to establish the actual yield strength, indicate the class(es) of SA 540 Grade B24 material used.

RESPONSE

Fastener material for the reactor vessel is SA 540 Grade B24 or B23 Class 3. Nuts and washers are Grade B23, stud material is Grade B24. Minimum specified yield strength for Class 3 material is 130,000 psi.

Section I.C "Introduction and Scope" of 10 CFR Appendix G states that the requirements of the Appendix applies for bolting materials with specified minimum yield strengths not over 130,000 psi; it does not limit the materials used to actual yield strengths of 130,000 psi (i.e., bolting materials having specified minimum yield strength to 130,000 must meet the impact criteria of 10CFR50 Appendix G).

REFERENCES: No FSAR change was made.

QUESTION 121.19

Paragraph III.B.3, Appendix G, 10 CFR Part 50, specifies that calibration of temperature instruments and Charpy V-notch impact test machines comply with Paragraph NB-2360 of the ASME Code. Calibration of test equipment for San Onofre Unit Nos. 2 and 3 was conducted in accordance with Paragraph NA-4600 of the 1971 ASME Code through 1971 Summer Addenda. Paragraph NA-4600 requires that a procedure be in effect to ensure that measuring and testing equipment is calibrated and properly adjusted at specific periods, and that calibration is against certified measurement standards. Provide details of this required procedure and measurement standards used.

RESPONSE

Procedures and records of equipment calibration used for qualification of San Onofre Units 2 and 3 material are maintained in C-E's Chattanooga facility in accordance with subarticle NA-4900, "Records and Data Reports." These records are available for review.

C-E calibrated Charpy V-notch test machines in accordance with Watertown Arsenal standards every six months. Temperature instruments were purchased to the accuracy requirements of ASTM E-23 and were calibrated to NBS traceable standards every three months.

REFERENCES: No FSAR change was made.

QUESTION 121.20

Provide the qualifications of individuals who performed fracture toughness tests as required by Paragraph III.B.4, Appendix G, 10 CFR Part 50. Include training and experience.

RESPONSE

10CFR50, Appendix G, was not published when testing of material for San Onofre Units 2 and 3 was specified. Therefore, only Code compliance was required.

The personnel performing the Charpy and drop weight impact testing were qualified by schooling, training, and many years of experience. Their qualifications to perform this work was certified by qualified supervisory personnel. Records of the certification of personnel are maintained and available for review at C-E's Chattanooga facility.

REFERENCES: No FSAR change was made.

QUESTION 121.21

Data presented in Tables 5.2-5, 5.2-5A, 121.11-1 through 121.11-27, 121.12-1, and 121.12-2, either do not meet, or are not adequate to determine if the requirements of Appendix G, 10 CFR Part 50 are met. Therefore, to help demonstrate compliance with Appendix G, 10 CFR Part 50, supply the following:

(1) for the reactor vessel beltline materials, provide full Charpy V-notch curves, including data points, reported in impact energy and lateral expansion, both as a function of temperature;

(2) for welds and weld heat-affected-zones in the beltline region, provide fracture toughness data from either available data or additional tests. Include transition temperature data, upper shelf energy data, and the significant variables that affect fracture toughness properties, e.g., weld wire, flux, base metal combinations, and heat treatment. Correlate this information with data already presented in Tables 121.11-1 through 121.11-27, and provide analyses of the additional data to demonstrate compliance with all the fracture toughness requirements of Appendix G;

(3) for all reactor vessel beltline materials, define an initial reference temperature, RT_{NDT}, and the most limiting RT_{NDT}. Provide details of the method used to establish both values.

RESPONSE

(1) Beltline materials Charpy test results for Units 2 and 3 are shown in Figures 121.21-1 through 121.21-12, respectively. The data points have been included. Tables 121.12-1 through 121.21-12 provide the data in tabular form corresponding to the Figures. The revised FSAR figures will be provided by 9/15/80 showing the Charpy test results.

For those materials which were not tested in accordance with 10CFR50, Appendix G, a conservative estimate of the RT_{NDT} temperature has been derived using the procedures outlined in MTEB Position 5-2, "Fracture Toughness Requirements". For three plates in the San Onofre Unit 2 reactor vessel beltline, the difference between the requirements of 10 CFR 50, Appendix G, and what testing was performed was solely in the orientation of the Charpy specimens. Data was generated from 0 to 100% shear fracture and the method of MTEB 5.2, Paragraph B1.1(3) (b), was used to determine RT_{NDT}s. That is, for the temperature at which 50 ft-lbs and 35 mils lateral expansion was obtained (with longitudinally oriented specimens), $20^{o}F$ was added to this value to provide a conservative estimate of the temperature that would have been acquired if transversely oriented specimens were tested.

(2) The prescribed testing for weld materials in the San Onofre Units 2 and 3 included drop weight testing and three Charpy tests at $10^{o}F$. Additional testing on some of the heats of weld wire and lots of flux used in San Onofre Units 2 and 3 has been done to qualify the material to later Code editions for use on other vessels. The best available information was previously provided in Tables 121.11-1 through 27. Revised information, showing actual data for upper shelf energy is given

in attached Tables 121.21-17 through 25. Where not otherwise noted, the RT_{NDT} has been determined in accordance with 10CFR50, Appendix G, and the upper shelf energy reported is as a minimum the average of a group of tests where all specimens demonstrate 100% shear fracture.

Attached Figures 121.21-13 and 14, and 121.21-15 and 16 provide additional Charpy V-notch test results for the weld and HAZ materials of San Onofre Units 2 and 3, respectively.

The surveillance weldment for Unit 2 was fabricated using 3/16" diameter bare wire of Type Mil B-4, heat number of 90130 with Linde Type 0091 flux, lot number 0842, the same heat of weld wire and lot of flux as used in the reactor pressure vessel. Heat treatment of the surveillance material is equivalent to the heat treatment accorded the reactor vessel.

The surveillance weldment for Unit 3 was fabricated using 3/16" diameter bare wire of Type Mil B-4, heat number 90130 with Linde flux type 124, lot number 0951, the same heat of weld wire and lot of flux as used in the reactor pressure vessel. Heat treatment of the surveillance material is equivalent to the heat treatment accorded the reactor vessel.

(3) Initial RT_{NDT}s for all beltline materials exclusive of HAZ were reported in FSAR Tables 5.2-5, 5.2-5A, and additionally provided Tables 121.11-1 through 27, 121.12-1 and 121.12-2. Unless otherwise specified by footnote, all values were established consistent with 10CFR50, Appendix G. RT_{NDT}s for the surveillance HAZ and weld material are presented in Table 121.21-22. These values were determined in accordance with 10CFR50, Appendix G.

The most limiting beltline region initial value of RT_{NDT} for San Onofre Unit 2 is from Table 5.2-5 of the FSAR for plate C-6404-3 where RT_{NDT} = 18 F. The most limiting beltline region initial value of RT_{NDT} for Unit 3 is that of the HAZ material tested in the baseline surveillance program where RT_{NDT} = 74 F (Table 121.21).

REFERENCES: FSAR Section 5.2.3; Question/Response for 121.11 and 121.12. No FSAR change was made.

Table 121.21-1

San Onofre Unit 2

Charpy V-Notch Impact Data

Plate: C-6404-1
Orientation: Transverse
D_WNDT: -30°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	10	0	5
-40	9	0	4
-40	7	0	3
+10	23	10	19
+10	16	5	14
+10	20	10	15
+40	30	15	21
+40	35	15	25
+40	43	20	30
+70	63	35	48
+70	60	35	46
+70	44	20	31
+80	77	50	55
+80	64	40	44
+80	57	35	42
+100	81	50	58
+100	72	50	56
+100	83	50	62
+160	119	95	74
+160	113	95	76
+160	117	90	78
+212	121	100	75
+212	120	100	75
+212	124	100	79

Table 121.21-2

San Onofre Unit 2

Charpy V-Notch Impact Data

Plate: C-6404-2
Orientation: Transverse
DwNDT: -20°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	6	0	3
-40	7	0	3
-40	11	0	6
+10	18	10	13
+10	16	5	11
+10	13	5	8
+40	37	15	25
+40	23	10	17
+40	35	15	24
+70	52	25	38
+70	47	25	34
+70	51	25	40
+80	65	40	47
+80	69	40	48
+80	53	30	38
+100	75	50	55
+100	68	50	52
+100	73	50	53
+160	101	99	71
+160	95	95	66
+160	94	95	67
+212	105	100	74
+212	123	100	80
+212	117	100	76

Table 121.21-3

San Onofre Unit 2
Charpy V-Notch Impact Data

Plate: C-6404-3
Orientation: Parallel
DWNDT: 0°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	13	0	14
-40	15	0	10
-40	44	20	31
+10	45	25	31
+10	30	15	20
+10	41	25	26
+40	63	40	43
+40	45	30	32
+40	72	45	49
+110	115	80	79
+110	100	60	73
+110	116	80	80
+160	131	100	83
+160	136	100	82
+160	148	100	83

Table 121.21-4

San Onofre Unit 2
Charpy V-Notch Impact Data

Plate: C-6404-4
Orientation: Parallel
D_{WNDT}: 0° F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	11	0	6
-40	17	0	9
-40	10	0	5
+10	46	25	32
+10	42	20	29
+10	50	25	35
+40	67	40	45
+40	58	30	41
+40	63	35	42
+110	101	70	72
+110	105	60	74
+110	102	70	70
+160	124	100	80
+160	139	100	88
+160	133	100	84

Table 121.21-5

San Onofre Unit 2

Charpy V-Notch Impact Data

Plate: C-6404-5
Orientation: Parallel
DWNDT: -10°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	9	0	6
-40	15	0	10
-40	11	0	7
+10	26	15	18
+10	35	20	24
+10	24	15	19
+40	62	35	44
+40	83	40	58
+40	84	40	60
+110	126	80	75
+110	116	70	74
+110	112	70	72
+160	141	100	84
+160	134	100	82
+160	127	95	80

Table 121.21-6

San Onofre Unit 2

Charpy V-Notch Impact Data

Plate: C-6404-6
Orientation: Transverse
DW_{NDT}: -10°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	4	0	4
-40	9	0	6
-40	6	0	5
+10	14	5	13
+10	16	5	14
+10	13	5	12
+40	28	15	23
+40	27	15	22
+40	27	15	24
+50	61	30	46
+50	54	25	40
+50	65	30	47
+110	85	60	61
+110	73	50	52
+110	94	70	59
+160	116	90	78
+160	115	90	75
+160	118	90	78
+212	128	100	80
+212	124	100	76
+212	121	100	77

Table 121.21-7

San Onofre Unit 3

Charpy V-Notch Impact Data

Plate: C-6802-1
Orientation: Transverse
D_{WNDT}: 40°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	7	0	5
-40	6	0	4
-40	5	0	4
+10	29	10	20
+10	23	10	17
+10	18	5	14
+40	34	15	21
+40	27	10	19
+40	24	5	17
+80	43	25	34
+80	55	30	45
+80	56	30	44
+100	50	25	37
+100	52	25	40
+100	59	30	45

Table 121.21-8

San Onofre Unit 3

Charpy V-Notch Impact Data

Plate: C-6802-2
Orientation: Transverse
DW_{NDT}: +10 °F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	9	0	4
-40	6	0	3
-40	8	0	4
+10	26	10	19
+10	14	5	11
+10	28	10	24
+40	32	15	26
+40	40	20	30
+40	52	25	37
+60	44	25	34
+60	52	30	36
+60	54	30	37
+70	55	30	38
+70	57	35	39
+70	65	40	45
+100	66	40	50
+100	77	50	59
+100	69	40	54
+160	114	90	77
+160	110	90	75
+160	106	90	75
+212	117	100	80
+212	116	100	82
+212	113	100	79

Table 121.21-9

San Onofre Unit 3

Charpy V-Notch Impact Data

Plate: C-6802-3
Orientation: Transverse
DW_{NDT}: -10 °F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	8	0	4
-40	7	0	3
-40	9	0	5
+10	26	10	20
+10	26	10	20
+10	15	5	11
+40	37	15	27
+40	28	10	22
+40	35	15	26
+60	35	15	26
+60	49	25	34
+60	35	20	30
+80	57	30	47
+80	52	25	40
+80	60	40	46
+100	79	60	60
+100	70	50	54
+100	63	40	46
+160	100	90	71
+160	95	90	69
+160	94	90	67
+212	106	100	76
+212	109	100	78
+212	101	100	72

Table 121.21-10

San Onofre Unit 3

Charpy V-Notch Impact Data

Plate: C-6802-4
Orientation: Transverse
DW_{NDT}: -30 °F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	10	0	6
-40	14	0	8
-40	10	0	6
+10	25	10	18
+10	28	10	20
+10	19	5	13
+40	35	20	30
+40	39	20	32
+40	51	25	37
+60	48	20	35
+60	54	25	39
+60	49	20	34
+70	76	50	53
+70	63	40	47
+70	59	35	44
+100	104	60	65
+100	69	40	52
+100	70	40	54
+160	124	95	80
+160	112	90	71
+160	104	90	70
+212	118	100	78
+212	117	100	81
+212	115	100	77

Table 121-21.11

San Onofre Unit 3

Charpy V-Notch Impact Data

Plate: C-6802-5
Orientation: Transverse
DWNDT: 0°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	9	0	4
-40	10	0	5
-40	9	0	3
+10	18	5	12
+10	27	10	20
+10	15	5	11
+40	34	15	23
+40	34	15	24
+40	35	15	25
+60	39	15	26
+60	39	15	28
+60	38	15	27
+70	65	30	43
+70	57	30	40
+70	51	25	37
+100	66	30	47
+100	70	35	49
+100	79	40	56
+160	118	100	78
+160	116	100	75
+160	109	90	65
+212	119	100	78
+212	110	100	72
+212	117	100	74

Table 121.21-12

San Onofre Unit 3
Charpy V-Notch Impact Data

Plate: C-6802-6
Orientation: Transverse
DW_{NDT}: -40°F

Temperature (°F)	Energy Absorbed (ft-lbs)	Shear (%)	Lateral Expansion (mils)
-40	10	0	5
-40	10	0	4
-40	8	0	4
+10	23	10	17
+10	28	10	20
+10	25	10	18
+40	34	15	26
+40	43	20	34
+40	33	15	26
+70	50	25	39
+70	46	20	35
+70	58	30	45
+80	52	25	40
+80	57	30	45
+80	53	25	41
+100	50	30	42
+100	65	50	51
+100	59	40	46
+160	96	95	72
+160	90	90	69
+160	83	90	65
+212	90	100	70
+212	92	100	71
+212	92	100	70

TABLE 121.21-13

954/cjm

San Onofre II Beltline Region Material

Weld Seam No. 2-203A

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.065	.32	.017	.009	.84	<.01	.90	.23	.0005	<.01	
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.007	.014	.017	.03	.001	.01	.01	.012	.005	.002	<.001

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -60

RT_{NDT}(°F) -60

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	ft-lbs	Mils L.E.
+10	118	70
+10	104	64
+10	158	40

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19}

Anticipated ΔRT_{NDT} (°F) 86

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 26

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

TABLE 121.21-14

San Onofre II Beltline Region Material

954/cjm

Weld Seam No. 2-203B

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.077	.36	.016	.009	.91	<.01	.91	.24	.0005	.01	
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.008	.017	.010	.03	.003	.02	.01	.012	.005	.003	<.001

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -60

RT_{NDT}(°F) -60

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	ft-lbs	Mils L.E.
+10	106	66
+10	108	72
+10	105	71

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19}

Anticipated ΔRT_{NDT} (°F) 86

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 26

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

TABLE 121.21-15

San Onofre II Beltline Region Material

Weld Seam No. 2-203C

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.074	.36	.016	.010	.88	<.01	.95	.25	.0005	<.01	
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.008	.015	.008	.03	.002	.01	.01	.011	.005	.003	<.001

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -60

RT_{NDT}(°F) -60

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	ft-lbs	Mils L.E.
+10	106	66
+10	108	72
+10	105	71

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm²) 3.68×10^{19}

Anticipated ΔRT_{NDT} (°F) 96

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 36

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

954/cjm

TABLE 121.21-16

San Onofre II Beltline Region Material

954/cjm

Weld Seam No. 3-203A

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.14	.13	.011	.011	1.27	.05	.12	.52	.0004	<.01	
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.006	.009	.008	.05	.004	<.01	.01	.013	.007	.002	<.001

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -50

RT_{NDT}(°F) -50

Upper Shelf Energy (ft-lb) NA

Temp. (°F)	CVN Data for Use ft-lbs.	Mils L.E.
+10	153	85
+10	131	81
+10	125	77

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19}

Anticipated ΔRT_{NDT} (°F) 106

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 56

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

TABLE 121.21-17

San Onofre II Beltline Region Material

Weld Seam No. 3-203B

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.017	.10	.011	.010	1.17	.05	.06	.39	.0004	<.01	
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.004	.007	.006	.04	.002	<.01	.01	.011	.005	.002	<.001

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -50

RT_{NDT}(°F) -50

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	Ft-lbs.	Mils L.E.
+10	153	85
+10	131	81
+10	125	77

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19}

Anticipated ΔRT_{NDT} (°F) 96

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 46

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

954/cjm

TABLE 121.21- 18

954/cjm

San Onofre II Beltline Region Material

Weld Seam No. 3-203C

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.18	.12	.011	.010	1.33	.06	.11	.54	.0004	<.01	
V	**Co**	**N**	**Cu**	**Al**	**Ti**	**W**	**As**	**Sn**	**Zr**	**Pb**
.006	.009	.005	.06	.004	<.01	.01	.014	.006	.002	<.001

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -50

RT_{NDT}(°F) -50

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	Ft-lb.	Mils L.E.
+10	153	85
+10	131	81
+10	125	77

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19} n/cm^2

Anticipated ΔRT_{NDT} (°F) 96

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 46

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

TABLE 121.21-19

San Onofre III Beltline Region Material

954/cjm

Weld Seam No. 2-203A

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.15	.15	.010	.006	1.24			.61			
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.006			.05							

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -40

RT_{NDT}(°F) -40

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	Ft-lb.	Mils L.E.
+20	125	77
+20	138	81
+20	145	83

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19} n/cm^2

Anticipated ΔRT_{NDT} (°F) 78

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 38

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

TABLE 121.21-20

San Onofre III Beltline Region Material

954/cjm

Weld Seam No. 2-203B

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.15	.15	.010	.006	1.24			.61			
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.006			.05							

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -40

RT_{NDT}(°F) -40

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp. (°F)	Ft-lb.	Mils L.E.
+20	125	77
+20	138	81
+20	145	83

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19} n/cm^2

Anticipated ΔRT_{NDT} (°F) 78

Anticipated ΔUSE (%) 25

Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 38

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

954/cjm

TABLE 121.21-21

San Onofre III Beltline Region Material

Weld Seam No. 2-203C

Filler Metal Type Mil B-4 mod.

Process Submerged Arc

Composition (w/o)

C	Si	S	P	Mn	Cr	Ni	Mo	B	Cb	
.15	.15	.010	.006	1.24			.61			
V	Co	N	Cu	Al	Ti	W	As	Sn	Zr	Pb
.006			.05							

Unirradiated Fracture Toughness Properties:

Drop Weight T_{NDT}(°F) -40

RT_{NDT}(°F) -40

Upper Shelf Energy (ft-lb) NA

CVN Data for Use

Temp (°F)	Ft-lb.	Mils L.E.
+20	125	77
+20	138	81
+20	145	83

Estimated Irradiated Fracture Toughness Properties:

Maximum EOL fluence (n/cm^2) 3.68×10^{19} n/cm^2

Anticipated ΔRT_{NDT} (°F) 78

Anticipated ΔUSE (%) 25

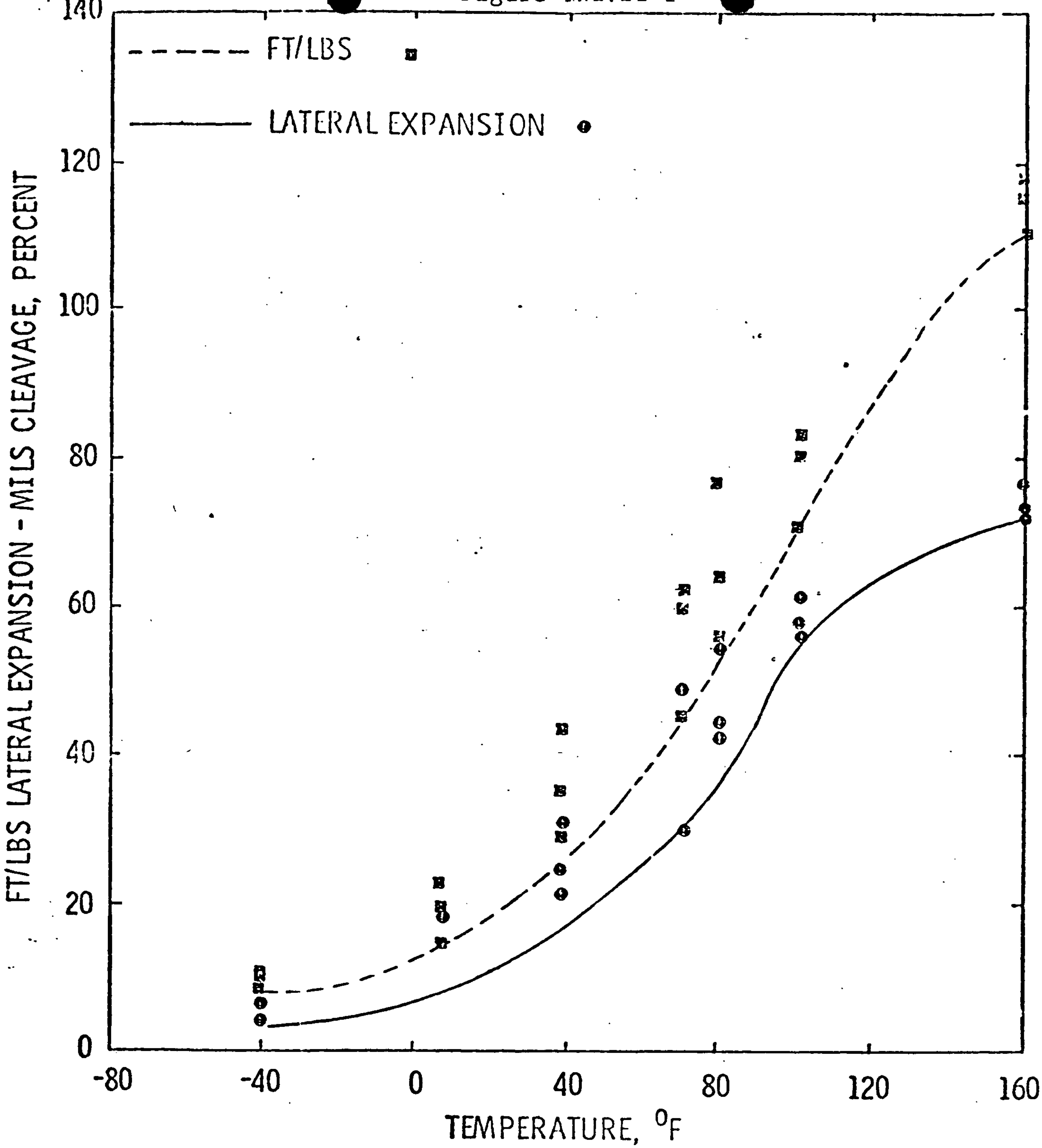
Adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) 38

Adjusted USE ($USE_{initial}$ + ΔUSE) >75

Table 121.21 - 22

Impact Properties for San Onofre Units 2&3 Surveillance Material

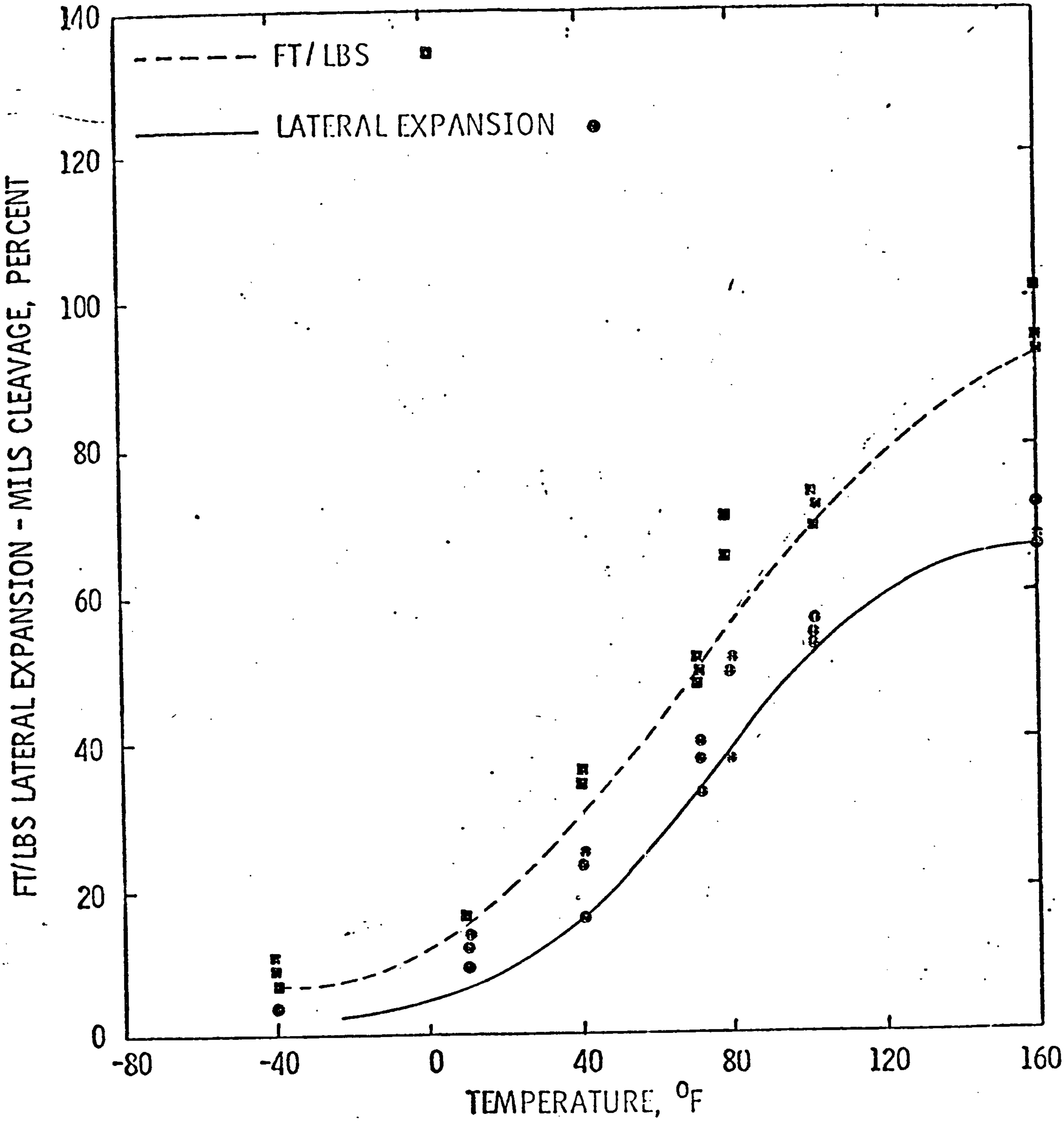
Unit	Material	DWTT NDT (^{o}F)	RT_{NDT} (^{o}F)	Cv Use Ft - lb
2	Weld Metal	-50	-50	135.5
	HAZ	-10	-10	139
3	Weld Metal	-60	-34	78
	HAZ	-40	74	79



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
CHARPY TEST RESULTS INTERMEDIATE SHELL - PLATE (C-6404-1)

10079-11D-679
31AUG

Figure 121.21-2

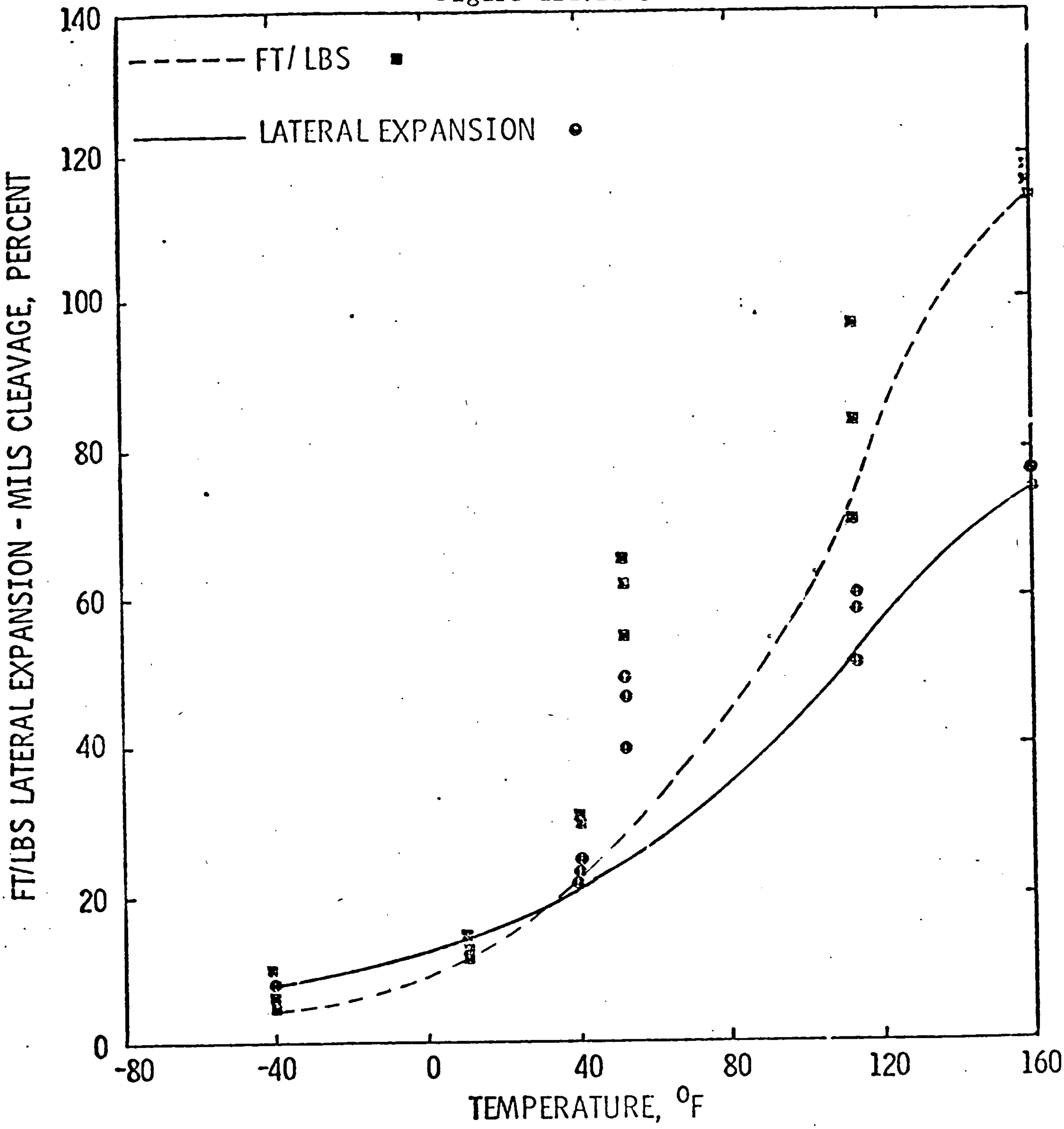


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CHARPY TEST RESULTS
INTERMEDIATE SHELL - PLATE
(C-6404-2)

10079-11D-480

Figure 121.21-3

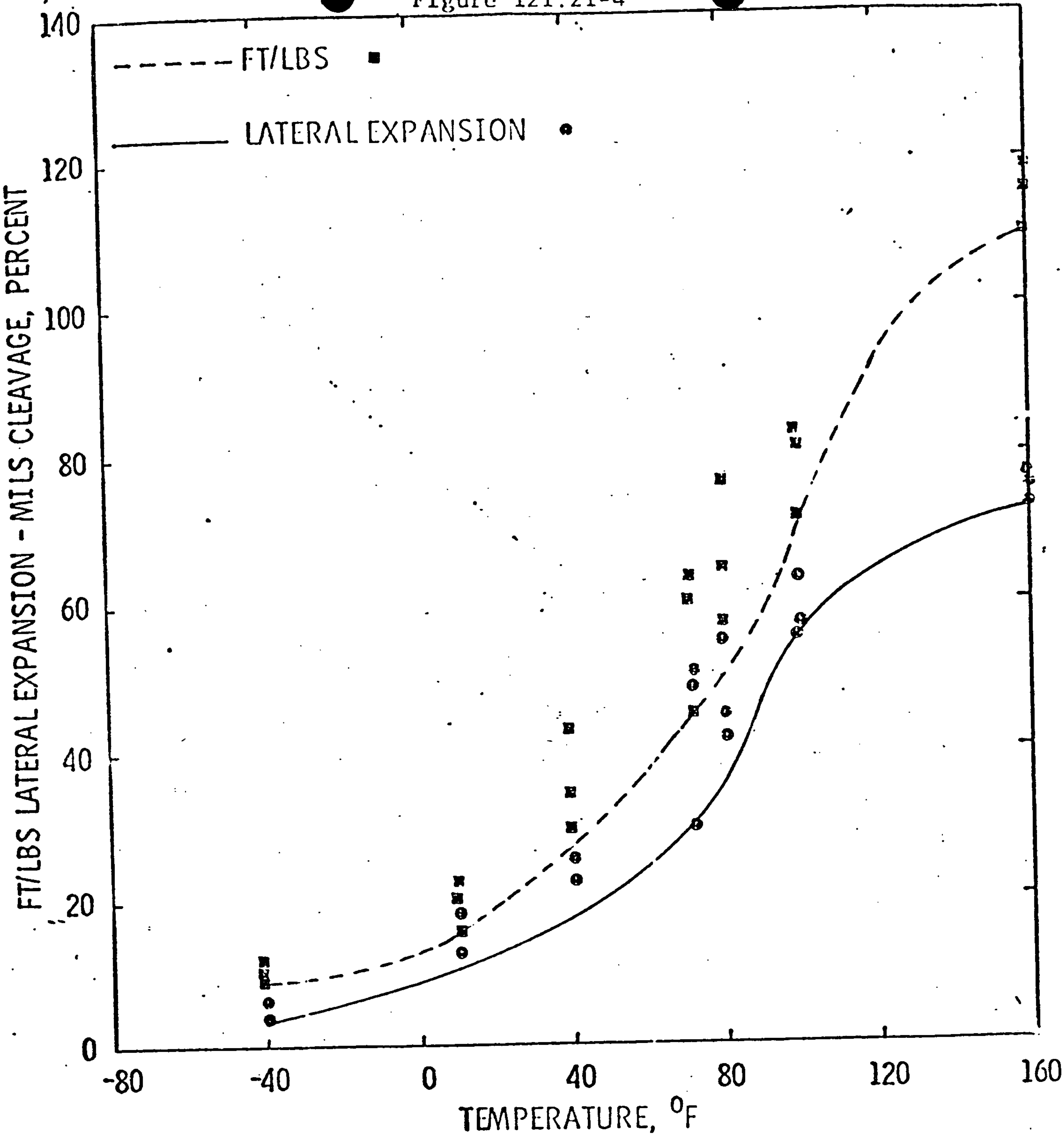


SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CHARPY TEST RESULTS
INTERMEDIATE SHELL - PLATE
(C-6404-3)

10079-11D-681

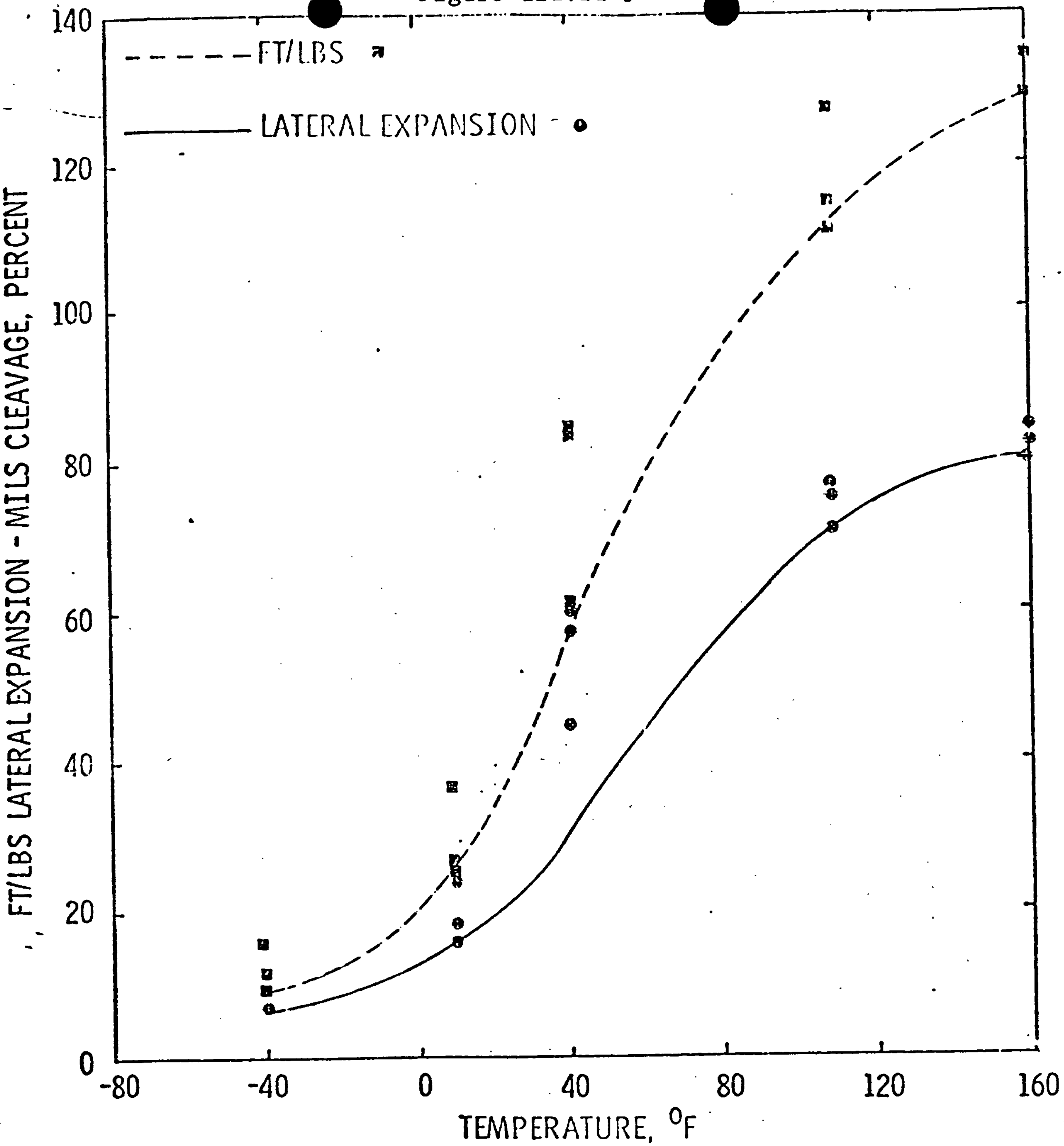
Figure 121.21-4



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

CHARPY TEST RESULTS
LOWER SHELL - PLATE (C-6404-4)

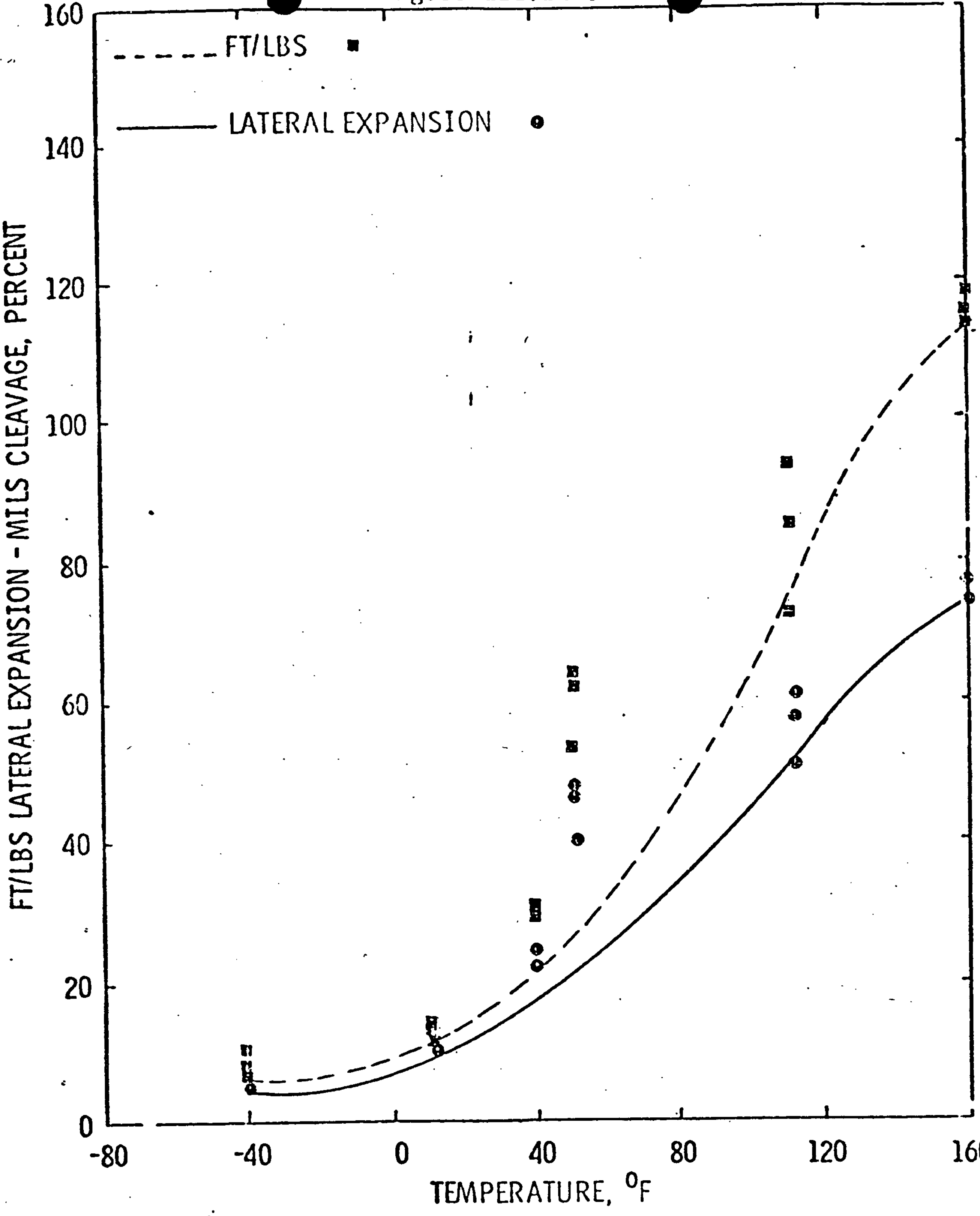
Figure 121.21-5



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
CHARPY TEST RESULTS LOWER SHELL - PLATE (C-6404-5)

SU075-11D-683
31AUG

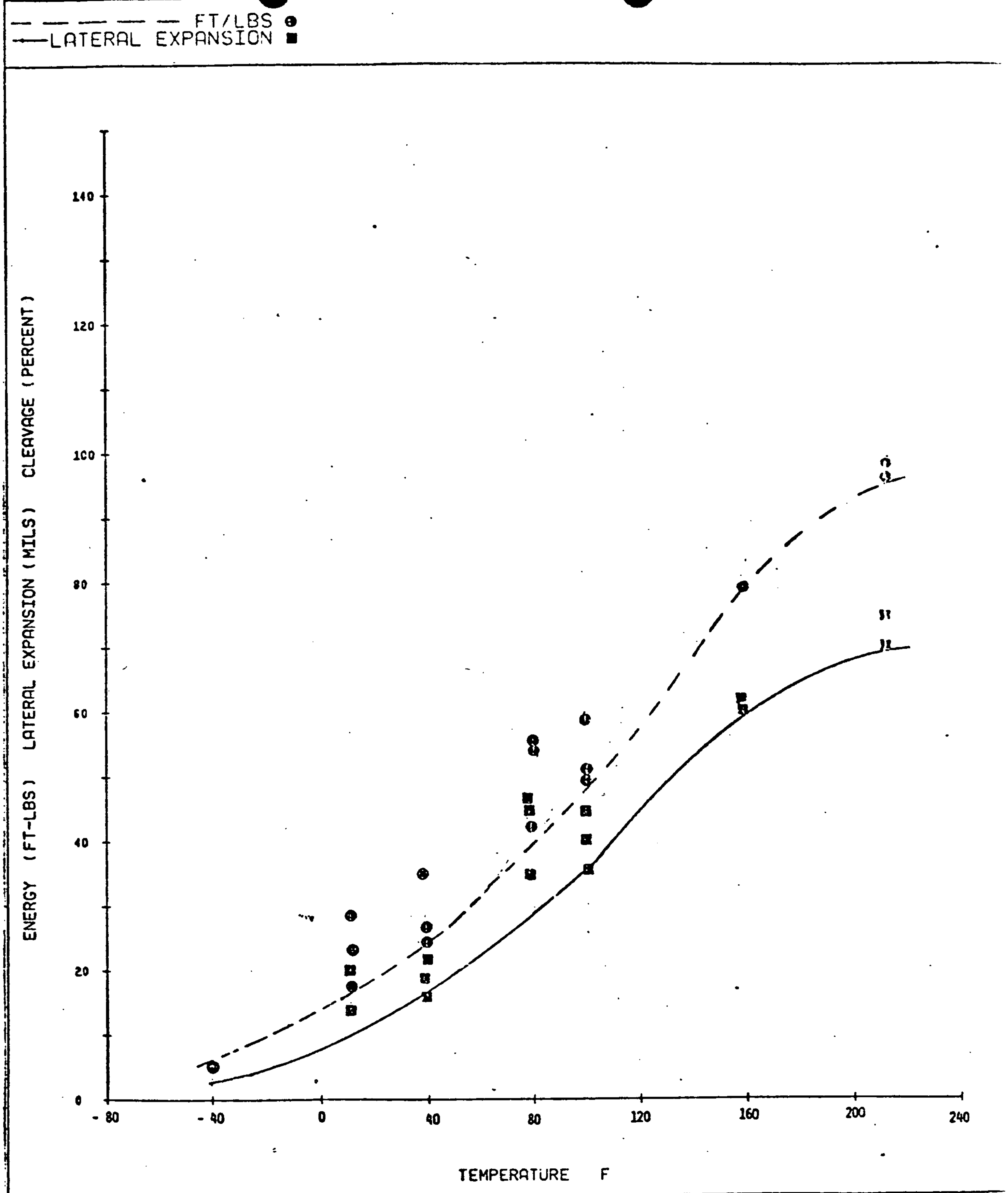
Figure 121.21-6



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
CHARPY TEST RESULTS LOWER SHELL - PLATE (C-6404-6)

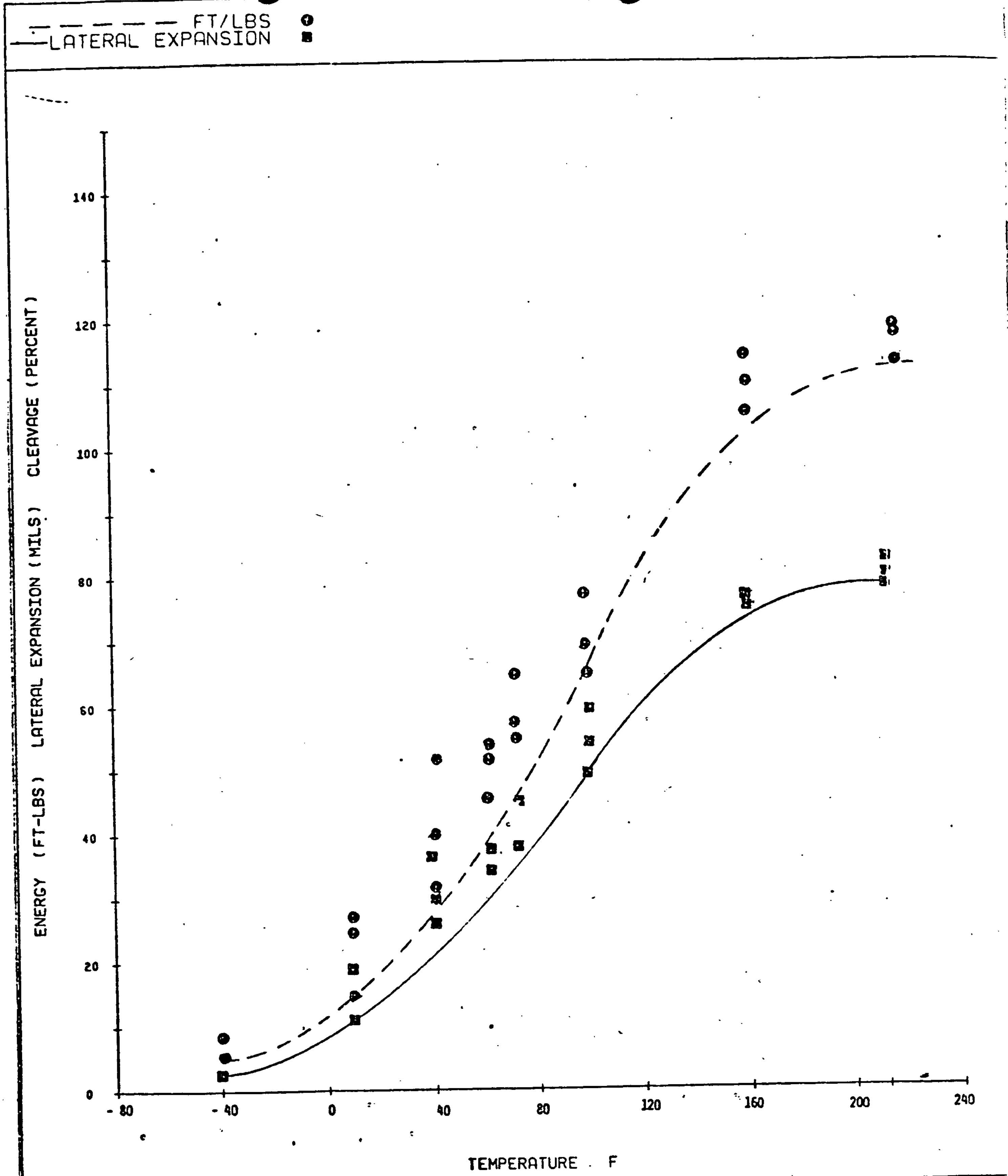
10079-11D-684
31AUG

Figure 121.21-7



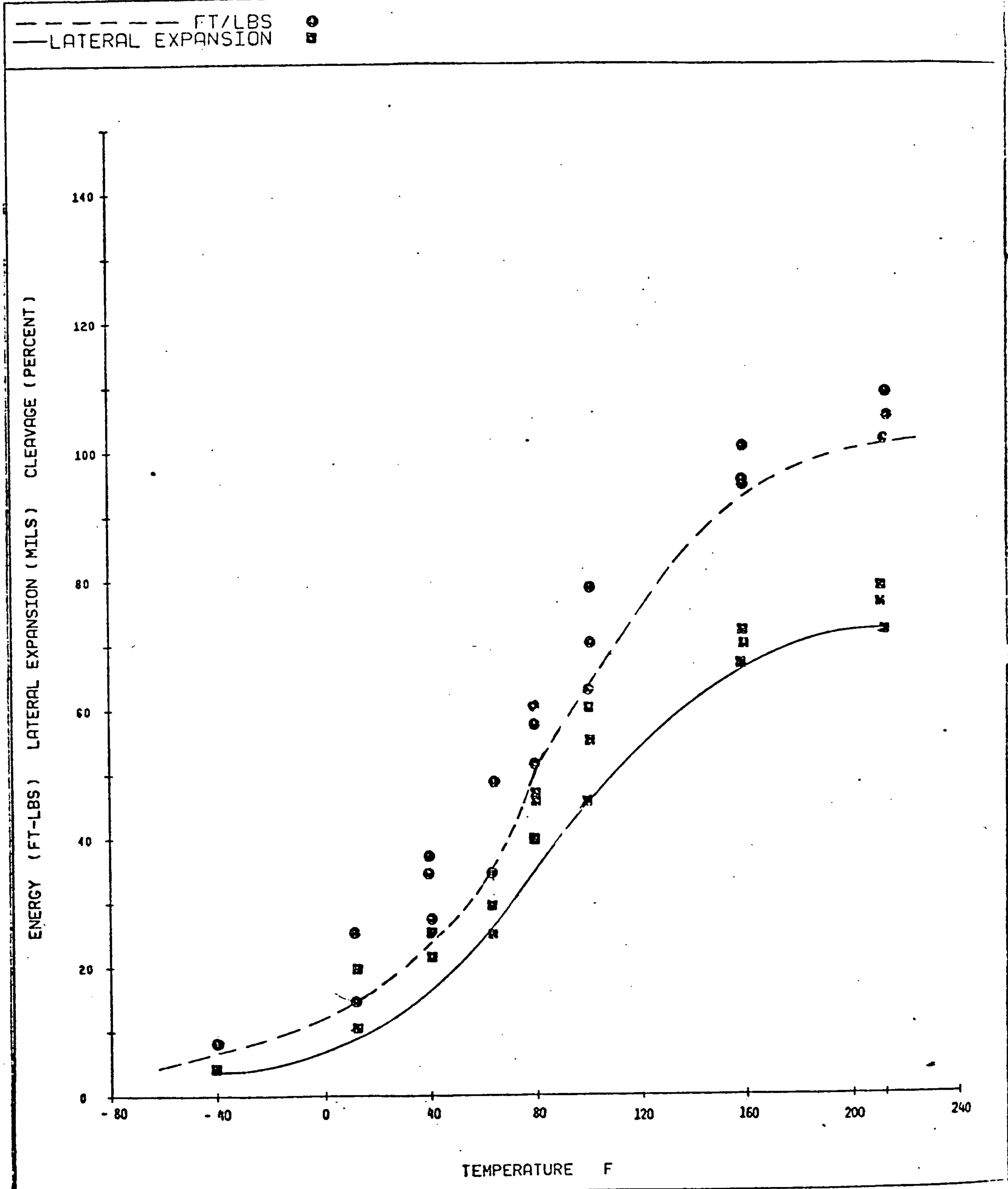
SAN ONOFRE
NUCLEAR GENERATING STATION
UNIT 3
CHARPY TEST RESULTS
INTERMEDIATE SHELL PLATE C-6802-1

Figure 121.21-8



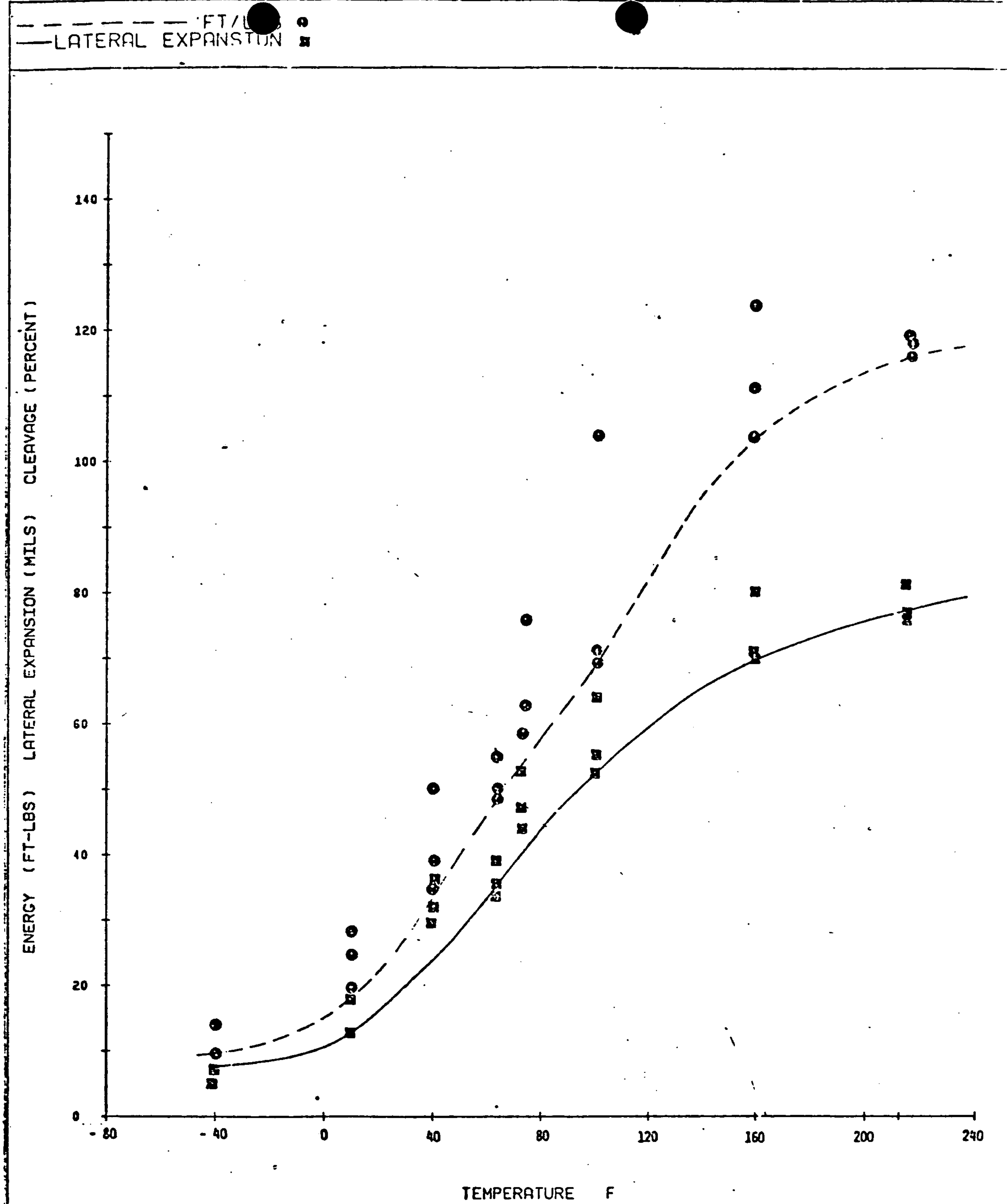
SAN ONOFRE
NUCLEAR GENERATING STATION
UNIT 3
CHARPY TEST RESULTS
INTERMEDIATE SHELL PLATE C-6802-2

Figure 121.21-9



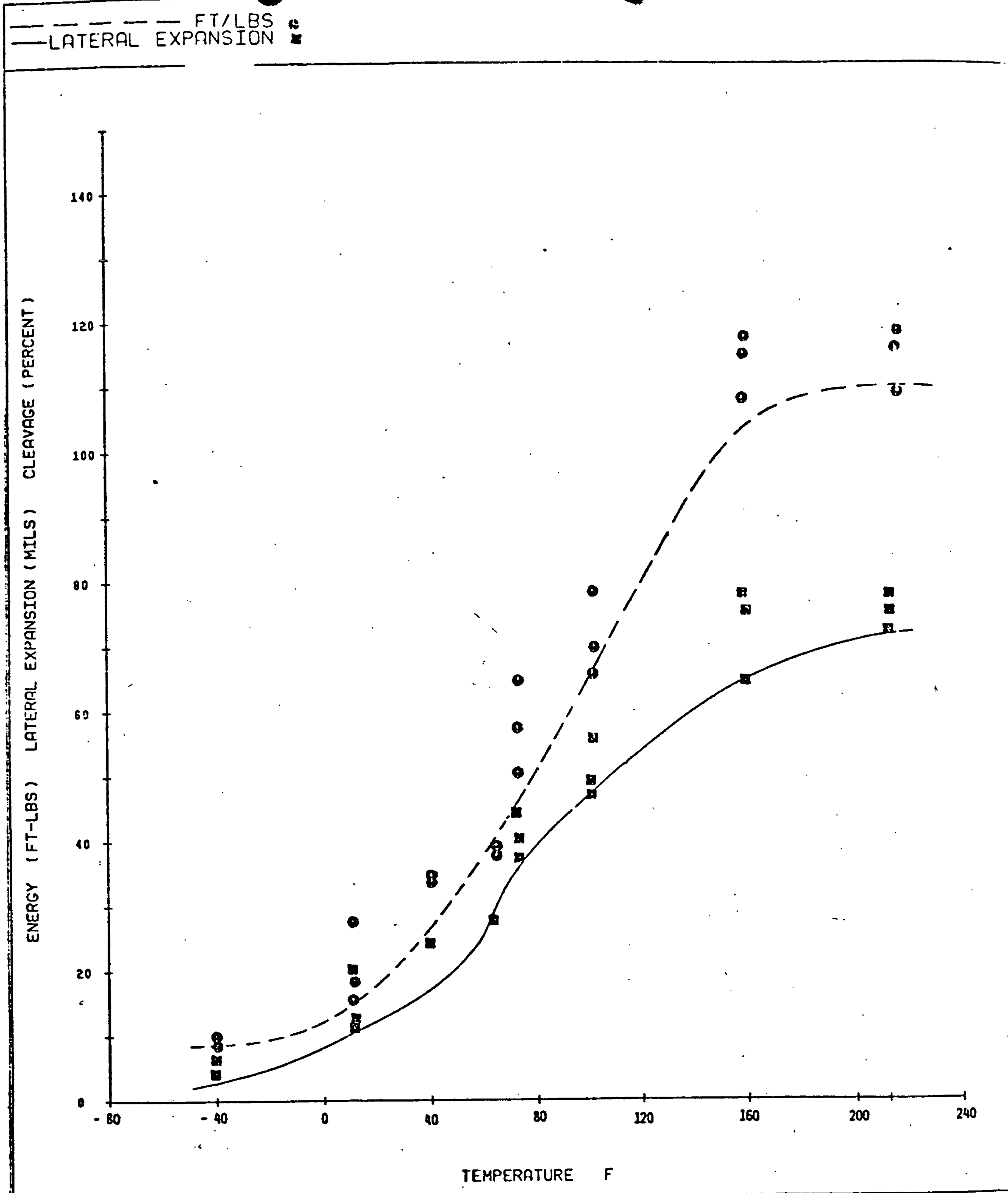
SAN ONOFRE
NUCLEAR GENERATING STATION
UNIT 3
CHARPY TEST RESULTS
INTERMEDIATE SHELL PLATE C-6802-3

Figure 121,21-10

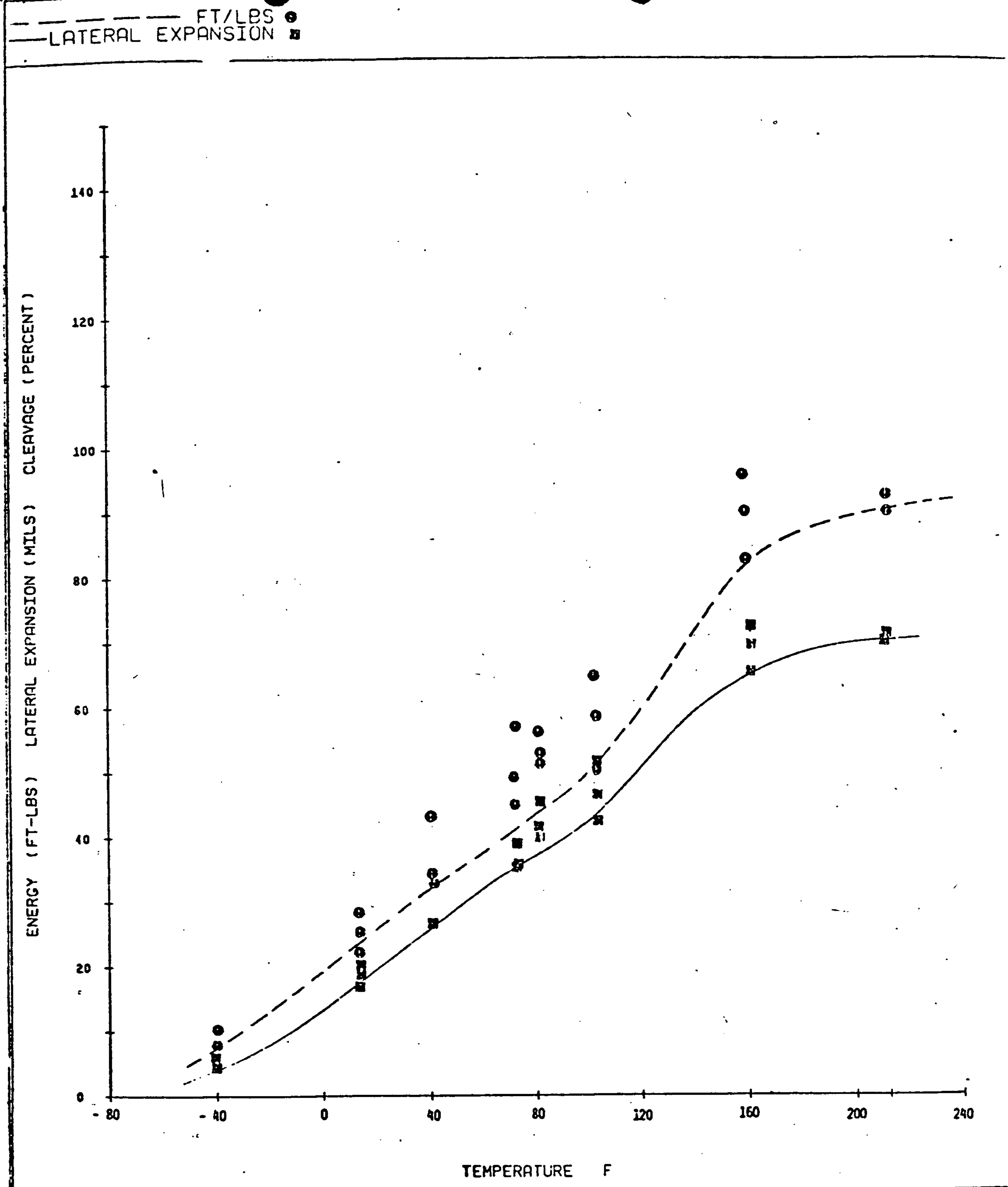


SAN ONOFRE
NUCLEAR GENERATING STATION
UNIT 3
CHARPY TEST RESULTS
LOWER SHELL PLATE C-6802-4

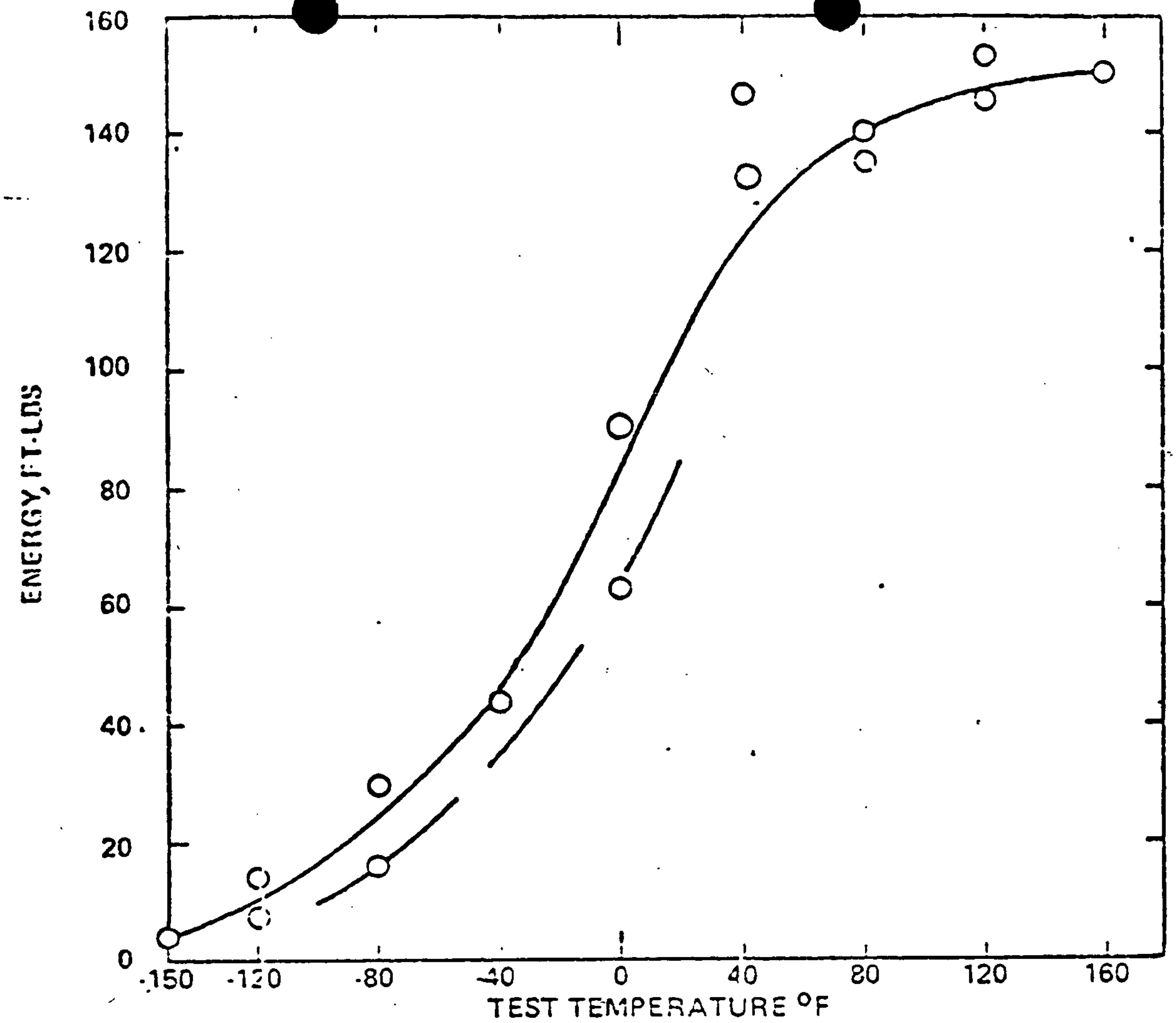
Figure 121.21-11



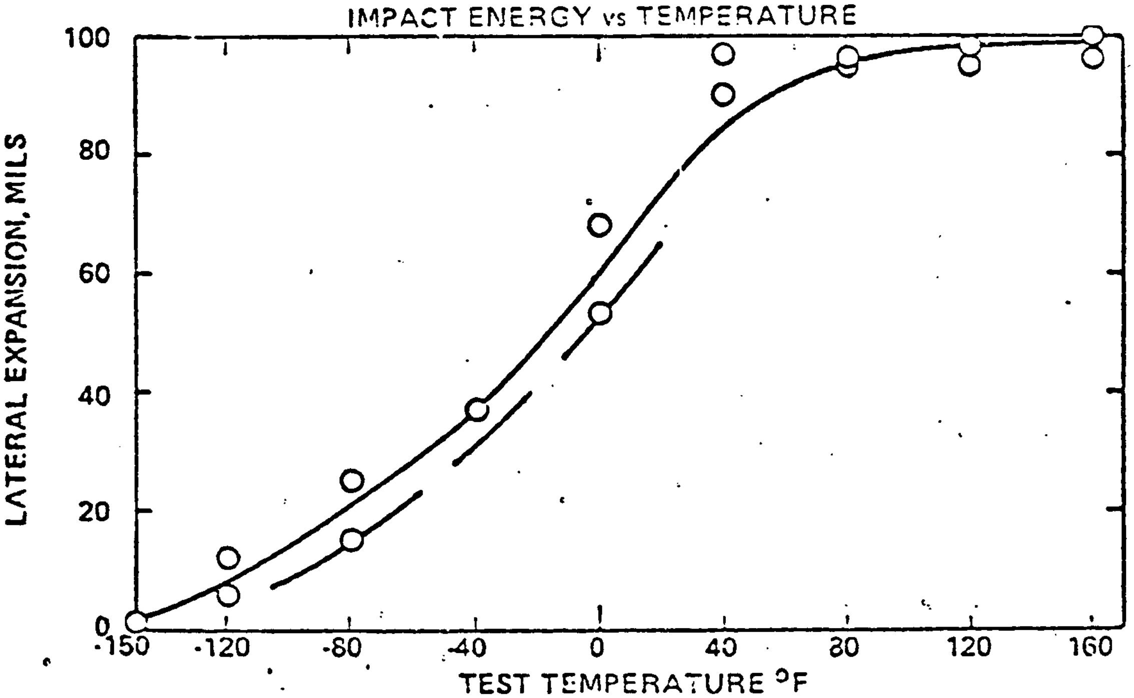
SAN ONOFRE
NUCLEAR GENERATING STATION
UNIT 3
CHARPY TEST RESULTS
LOWER SHELL PLATE C-6802-5



SAN ONOFRE
NUCLEAR GENERATING STATION
UNIT 3
CHARPY TEST RESULTS
LOWER SHELL PLATE C-6802-6

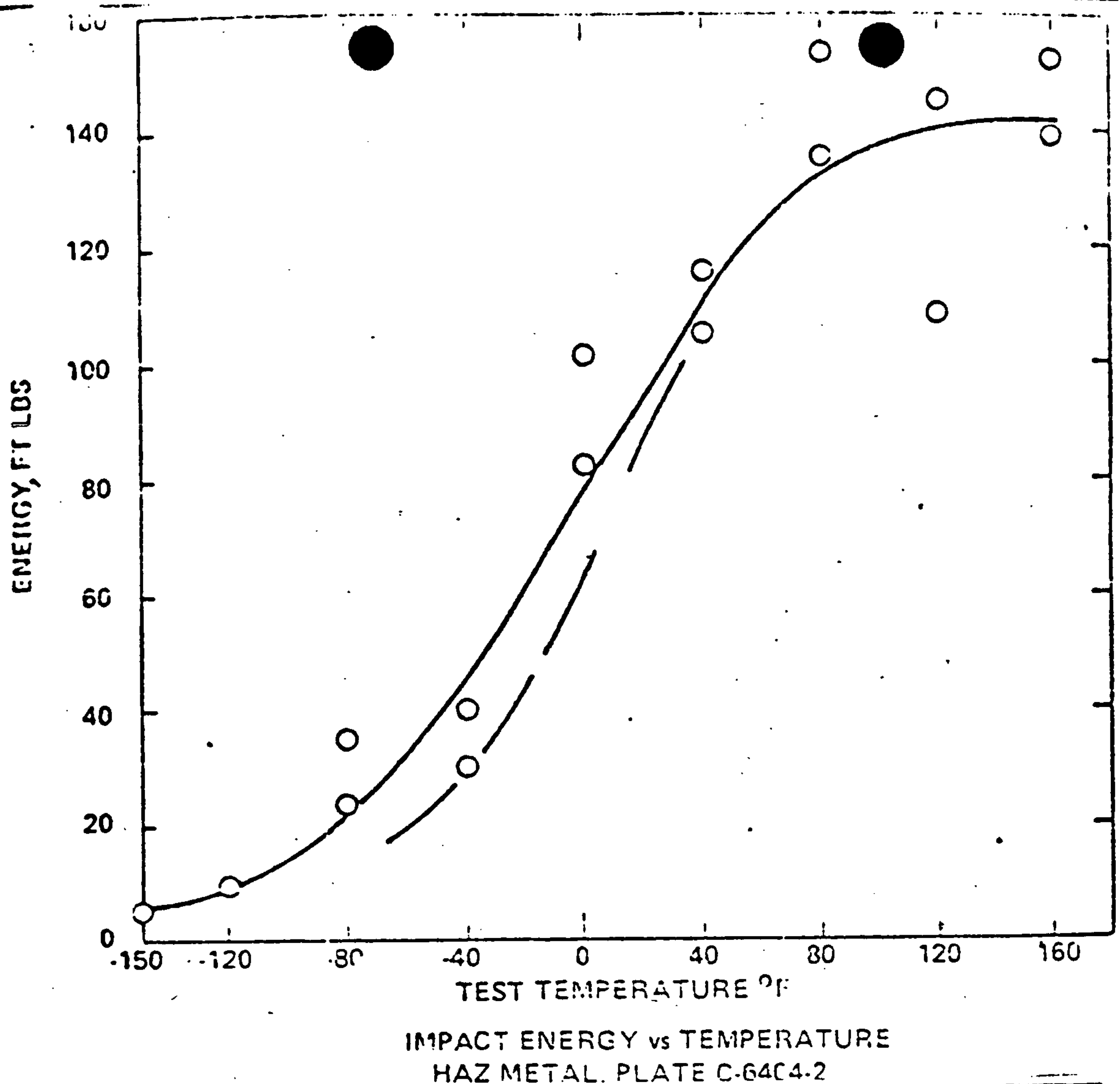


IMPACT ENERGY vs TEMPERATURE



LATERAL EXPANSION vs TEMPERATURE
WELD METAL, PLATE C-6404-1 & 3

FIG. 121.21-13 CHARPY V-NOTCH TEST RESULTS
FOR SAN ONOFRE UNIT 2 SURVEILLANCE WELD METAL



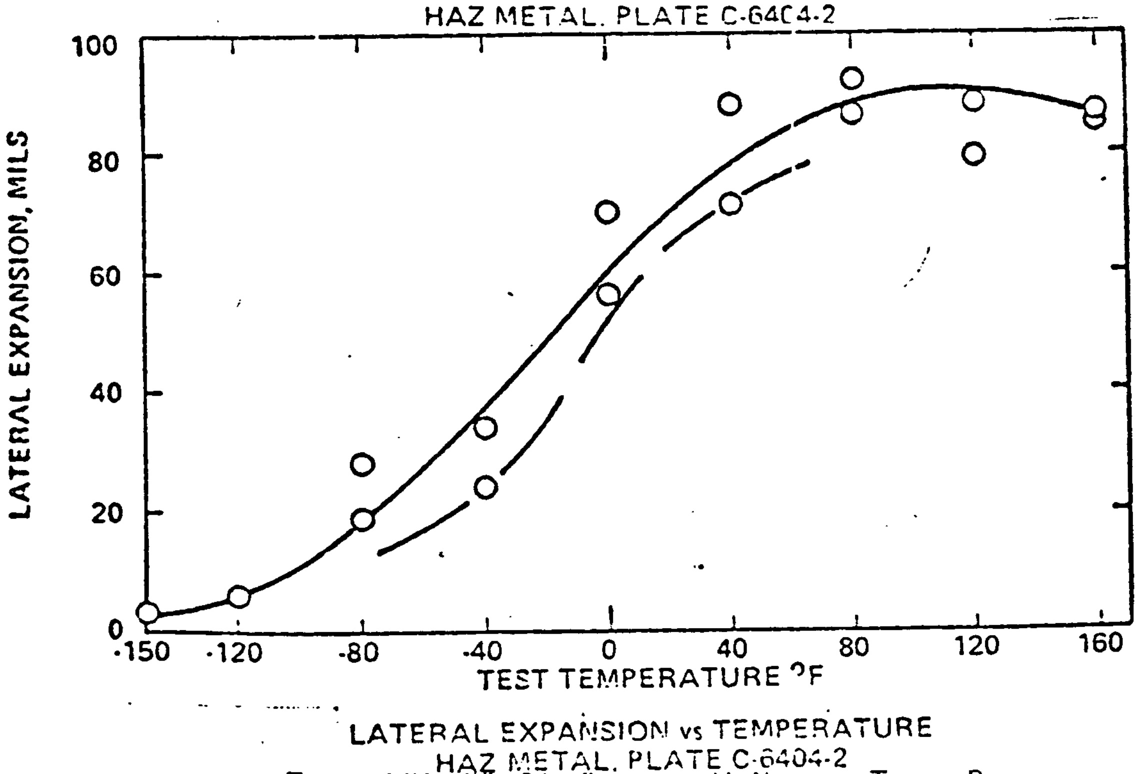
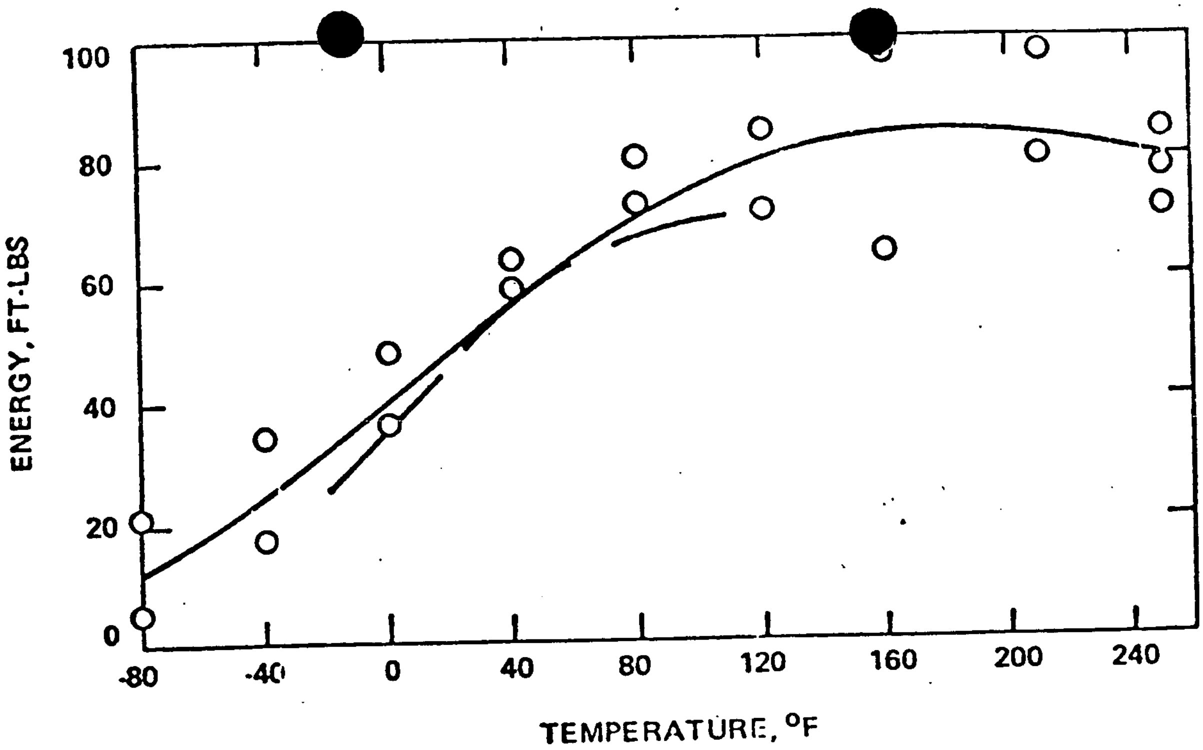
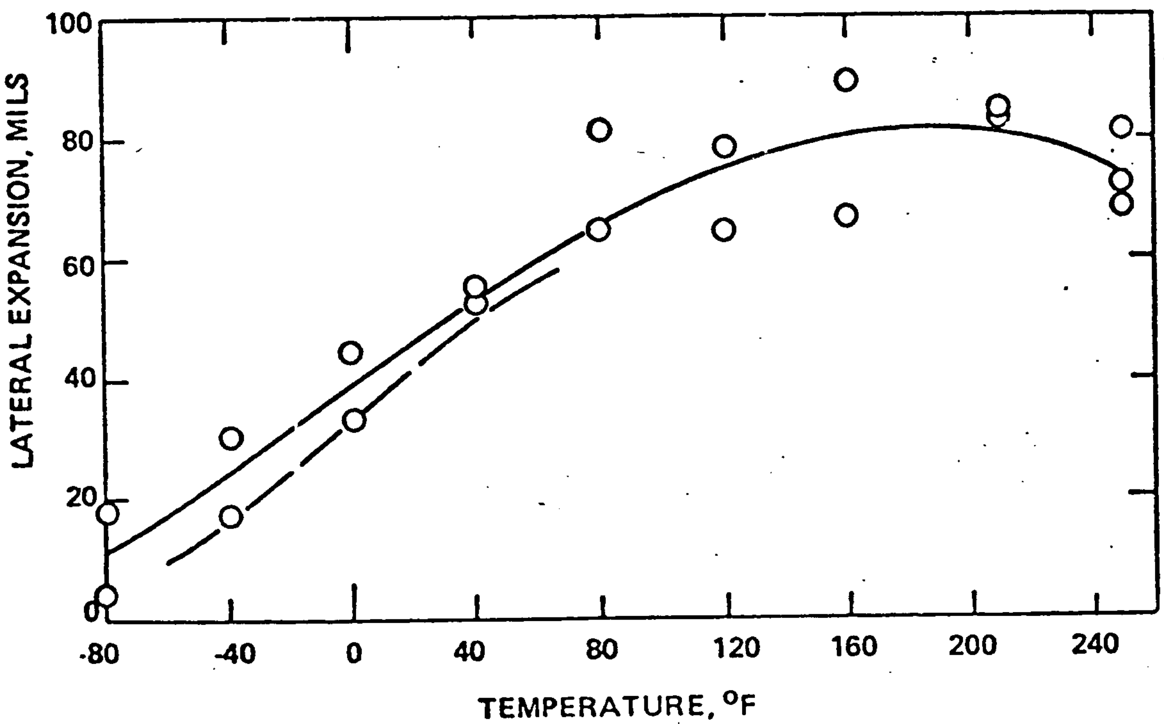


FIG. 121.21-14 CHARPY V-NOTCH TEST RESULTS

SAN ONOFRE UNIT 2 SURVEILLANCE HAZ MATERIAL

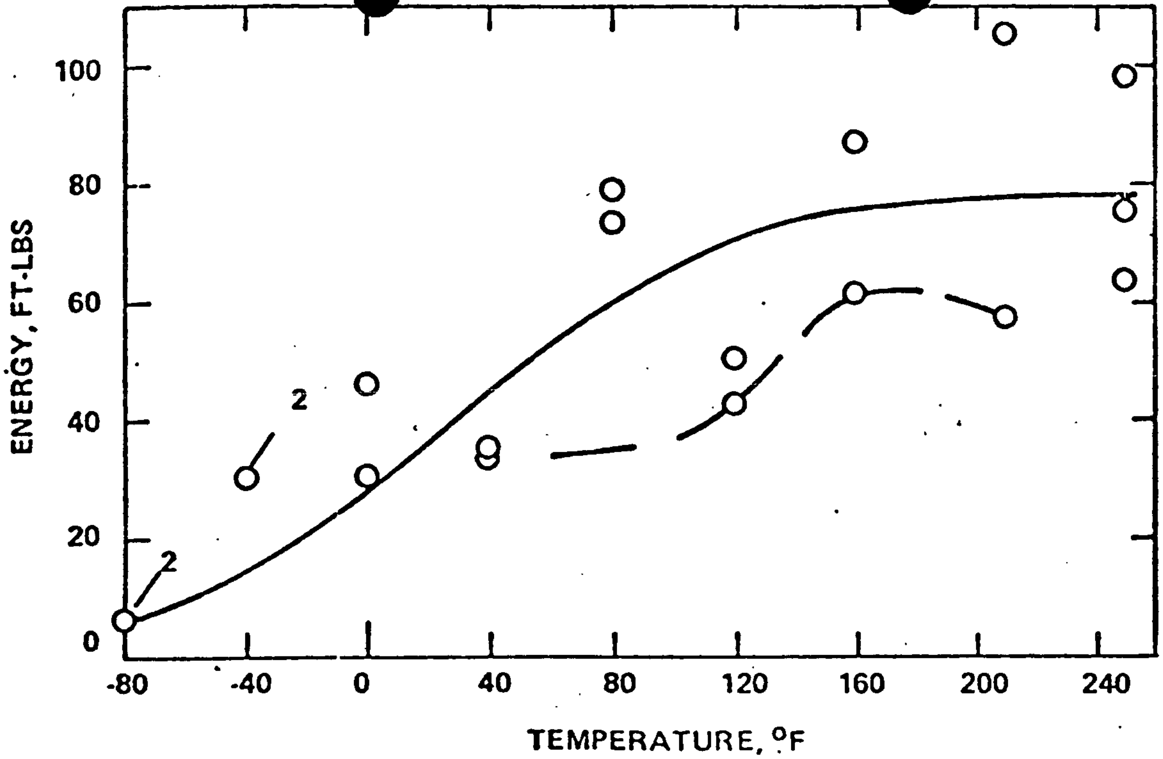


IMPACT ENERGY VS TEST TEMPERATURE WELD METAL, PLATES C-6802-2 & 3

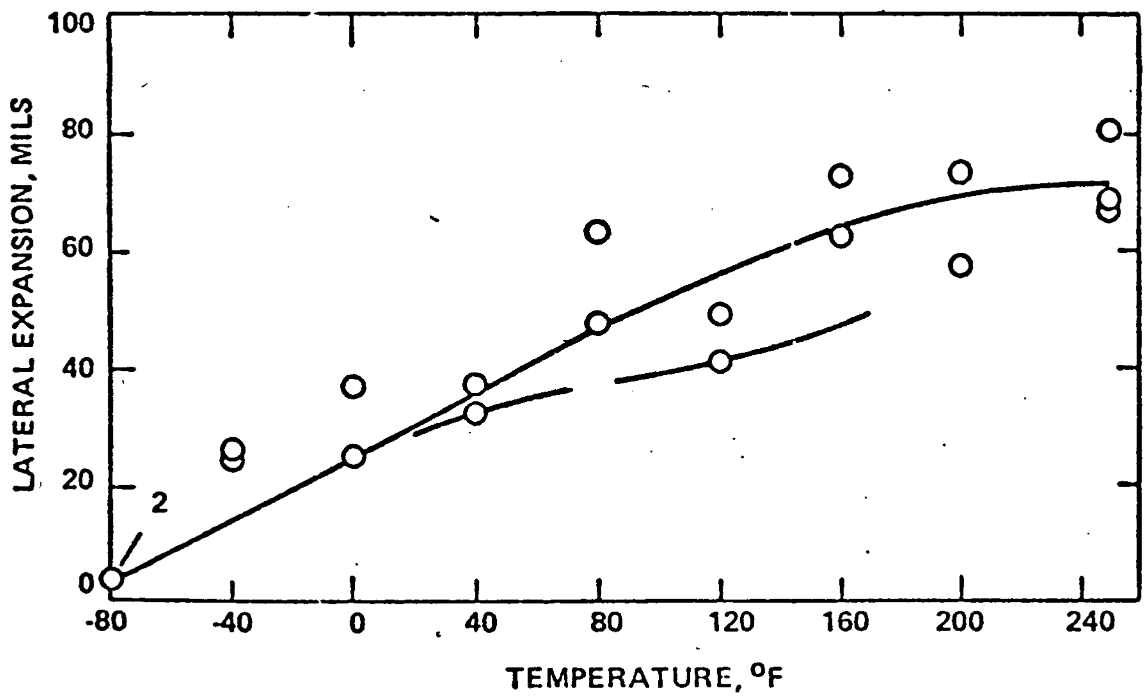


LATERAL EXPANSION VS TEST TEMPERATURE WELD METAL, PLATES C-6802-2 & 3

FIG. 121.21-15 CHARPY V-NOTCH TEST RESULTS SAN ONOFRE UNIT 3 SURVEILLANCE WELD METAL



IMPACT ENERGY VS TEST TEMPERATURE HAZ METAL, PLATE C-6802-1



LATERAL EXPANSION VS TEST TEMPERATURE HAZ METAL, PLATE C-6802-1

FIG. 121.21-16 CHARPY V-NOTCH TEST RESULTS

SAN ONOFRE UNIT 3 SURVEILLANCE HAZ MATERIAL

QUESTION 121.22

Paragraph III.C.2, Appendix G. 10 CFR Part 50, specified that every fracture toughness test specimen from the reactor vessel beltline must be subjected to a heat treatment that produces metallurgical effects equivalent to those produced in the vessel material throughout its fabrication process. Identify all specimens that do not meet this requirement and provide technical justification for use of such a specimen(s) in establishing fracture toughness properties of the reactor vessel beltline.

RESPONSE

All materials used in either the surveillance program or qualification testing have undergone heat treatment consistent with the requirements of Subarticle NB-2200, "Material Test Coupons and Specimens for Ferritic Steel Material." No material that had not undergone equivalent heat treatment was used to generate fracture toughness data.

REFERENCES: No FSAR change was made.

QUESTION 121.23

Revise the pressure-temperature limits, presented in Figures 16.3-7A and 16.3-7B of the Technical Specifications, to reflect data requested in Question 121.19.

RESPONSE

The pressure-temperature limit curves are presently being revised. All data including that requested in Question 121.21 will be evaluated for deriving these curves.

REFERENCES: No FSAR change was made.

QUESTION 121.24

The materials surveillance program uses six specimen capsules, containing reactor vessel steel specimens of the limiting base material, weld metal material, and heat-affected zone material. To help demonstrate compliance with Appendix H, 10 CFR Part 50, provide a table that includes the following information for each of the 342 specimens:

(1) actual surveillance material;
(2) beltline material from which the specimen was obtained;
(3) test specimen type and orientation;
(4) fabrication history of each test specimen;
(5) chemical composition of each test specimen; and
(6) heat of filler material, production welding conditions, and base metal combinations for weld specimens.

Provide the lead factor for each specimen capsule calculated with respect to the vessel inner wall.

RESPONSE

The requested information, in a format similar to that requested by the MTEB personnel, is presently attached in Tables 121.24-1 through 4.

Regarding Footnote A; Withdrawal Schedule: this is a recommended withdrawal sequence, since all capsules have the same lead factor, the actual capsule removed may be changed to accommodate operating exingencies during the refueling outage. The actual capsule removal will be reported in the report to the Commission required by 10CFR50, Appendix H.

Regarding Footnote B; Chemical Composition: complete chemical composition data for these materials is given in attached Tables 121.24-5 and 6.

Regarding Footnote C; Chemical Composition HSST Plate 01. ORNL Report 4377 on the HSST program contains chemical composition data for Plate 01.

REFERENCES: No FSAR change was made.

Table 121.24-1

San Onofre Unit 2 Surveillance Program

Capsule No.	Azimuthal Location	Withdrawal Schedule[A] Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. & Orientation	Chemical Composition[B]
1	83°	Standby	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal Linde 0091 Lot No. 0842 Mil B-4 Wire Heat No. 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ material Plate C-6404-2	12 CVN-T 3 Tensile	0.10 Cu .005 P
2	97°	4	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal Linde 0091 Flux Lot No. 0842 Mil B-4 Weld Wire Heat No. 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ material Place C-6404-2	12 CVT-T 3 Tensile	0.10 Cu .005 P
3	104°	17	1.15	1. Plate C-6404-2	12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal 0091 - Heat 0842 B-4 - Heat 90130	12 CVN 3 Tensile	0.30 Cu .003 P
				3. HAZ material Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P
				4. SRM Material HSST Plate 01	12 CVN-L	Ref. C

Table 121.24-2

San Onofre Unit 2 Surveillance Program

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. & Orientation	Chemical[B] Composition
4	284°	30	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal Linde 0091 Flux Lot No. 0842 Mil B-4 Wire Heat No. 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ Material Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P
5	263°	Standby	1.15	1. Plate C-6404-2	12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal Linde 0091 lot 0842 Mil B-4 Heat 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P
				4. SRM HSST Plate 01	12 CVN	Ref. C
6	277°	Standby	1.15	1. Plate C-6404-2	12 CVN-L 12 CVN-T 3 Tensile	0.10 Cu .005 P
				2. Weld Metal Linde 0091 Lot 0842 Mil B-4 Heat 90130	12 CVN 3 Tensile	0.03 Cu .003 P
				3. HAZ Material Plate C-6404-2	12 CVN 3 Tensile	0.10 Cu .005 P

Table 121.24-3

San Onofre Unit 3 Surveillance Program

Capsule No.	Azimuthal Location	Withdrawal Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No. & Orientation	Chemical Composition[A]
1	83°	Standby	1.15	1. Plate C-6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 2 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN-T 3 Tensile	0.05 Cu .008 P
2	97°	7	1.15	1. Plate C6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
3	104°	19	1.15	1. Plate C-6802-1	12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate 6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
				4. SRM Material [illegible] 01	12 CVN-L	Ref. C

Table 121.24-4

San Onofre Unit 3 Surveillance Program

Capsule No.	Azimuthal Location	Withdrawal[A] Schedule Calendar Year	Lead Factor	Surveillance Materials	Specimen Type No.&Orientation	Chemical[B] Composition
4	284°	30	1.15	1. Plate C-6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Flux Lot No. 0951 Mil B-4 Wire Heat No. 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
5	263°	Standby	1.15	1. Plate C-6802-1	12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Flux Lot No. 0951 Mil B-4 Heat 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P
				4. SRM Material HSST Plate 01	12 CVN	Ref C
6	277°	Standby	1.15	1. Plate C-6802-1	12 CVN-L 12 CVN-T 3 Tensile	0.05 Cu .008 P
				2. Weld Metal Linde 124 Lot 0951 Mil B-4 Heat 90069	12 CVN 3 Tensile	0.03 Cu .004 P
				3. HAZ Material Plate C-6802-1	12 CVN 3 Tensile	0.05 Cu .008 P

Table 121.24-5

Chemical Analyses Of
San Onofre Unit 2
Surveillance Test Materials

Elements (Weight %)	C-6404-2 BASE PLATE	C-6404-1/3 WELD	C-6404-2/3 H.A.Z.
C	0.23	0.17	0.17
Mn	1.43	1.34	1.38
P	0.005	0.003	0.004
S	0.009	0.009	0.009
Si	0.26	0.21	0.22
Ni	0.60	0.12	0.07
Cr	0.18	0.09	0.10
Mo	0.58	0.52	0.54
V	0.003	0.005	0.005
Cb	<.01	<.01	<.01
Ti	<.01	<.01	<.01
Co	0.012	0.012	0.019
Cu	0.10	0.03	0.04
Al Sol	0.033	0.010	0.009
Al Ins	0.001	0.002	0.009
Al Total	0.034	0.012	0.012
B	<.001	<.001	<.001
W	<.01	<.01	<.01
As	0.001	<.001	0.003
Sn	0.003	0.001	0.002
Zr	<.001	<.001	<.001
Pb	<.001	<.001	<.001
Sb	0.0025	0.0013	0.0014
N	0.005	0.004	0.005

Table 121.24-6

Chemical Analyses Of
San Onofre Unit 3
Surveillance Test Materials

Elements (Weight %)	C-6802-1 PLATE	C-6802-1/2 H.A.Z	C-6802-2/3 WELD
Si	0.23	0.40	0.39
Sul	0.014	0.009	0.009
Phos	0.008	0.003	0.004
Mn	1.38	1.55	1.54
Car	0.24	0.12	0.12
Cr	0.07	0.03	0.05
Ni	0.57	0.09	0.08
Mo	0.58	0.55	0.55
B	0.0005	0.001	0.001
Cb	<0.01	<0.01	<0.01
Ti	<0.01	<0.01	<0.01
Co	0.010	0.014	0.015
Cu	0.05	0.03	0.03
Al	0.033	0.005	0.006
N2	0.010	0.005	0.006
V	0.005	0.005	0.005
W	<0.01	0.01	0.01
As	0.009	<0.001	<0.001
Sn	0.005	0.001	0.002
Zr.	0.002	<0.001	<0.001

QUESTION 121.26

To help demonstrate the integrity of the reactor coolant pump flywheels, supply the Charpy V-notch impact and tensile data for each flysheel, explicitly stating the material used for each flywheel.

Also, confirm that welding, including repair welding, was not performed on any finished flywheel. If welding was performed, identify the flywheel(s) and location of the welds.

RESPONSE

Charpy V-notch impact and tensile data for the Reactor Coolant Pump flywheels is provided on Table 121.26-1. The material each of the flywheels were made from is: ASTM A543, Grade I, Type B.

Neither welding or repair welding was performed on any of the flywheels.

References: FSAR Section 5.4.1.2 and 5.4.1.4; No change was made to the FSAR.

TABLE 121.26-1

REACTOR COOLANT PUMP

FLYWHEEL MECHANICAL TESTS

Melt No.	Slab No.	Tensile PSI x 1000	Yield PSI x 1000	% Elong. in 2"	Bend Test	Charpy Impact Test Ft.Lbs. @ 212°F.			N.D.T. Temp. per ASTM E 208	Number of Flywheels from each Melt & Slab
B 8819	3	1106	974	17	OK	78	81	80	-130°F.	1
		1240	1046	17	OK					
A 4646	2	1230	1019	20	OK	73	74	74	-170°F. or Below	2
		1126	916	21	OK					
A 4546	5	1116	902	19	OK	90	88	91	-130°F.	1
		1136	969	22	OK					
B 8858	1	1070	958	20	OK	110	108	108	-100°F.	2
		1080	898	26	OK					
A 4546	4	1075	925	22	OK	81	86	85	-100°F.	1
		1136	947	20	OK					
B 8858	3	1196	1100	18	OK	65	68	66	-100°F.	1
		1056	891	24	OK					

QUESTION 221.18

With regard to the Core Protection Calculator system, we require that the following information be provided:

(1) Identification of the revisions to the Software Specifications CEN-57(A)-P and CEN-58(A)-P made for San Onofre Units 2 and 3;

(2) The test report for verification of the San Onofre 2 and 3 CPC software;

(3) The data base constants and changes to the CPC algorithms; and

(4) Modifications to the proposed technical specifications.

(5) Provide a commitment to (a) implement the final software change procedure approved for the ANO-2 facility in accordance with Appendix B provisions of 10 CFR Part 50, and (b) utilize the services of a qualified computer consultant to provide independent verification that approved changes in the software are properly made. Provide documentation or a reference to documentation describing the final version of the software to which change procedures are to be applied.

RESPONSE

In order to facilitate the NRC review of CPC system software modifications, the following software design and test information will be submitted:

(1) Table 221.18-1 is a summary of the software differences between San Onofre and ANO-2. The ANO-2 baseline software is defined in the Basis to the ANO-2 Technical Specification (Appendix A to Operating License NPF-6) Section 2.2.1 as modified by References 1 and 2.

CEN-135(S)-P is a detailed description of the software changes described in Table 221.18 which includes algorithm descriptions in symbolic algebra.

(2) A software test results document giving the results of Phase I and Phase II software testing will be submitted by December, 1980. Sufficient information will be provided to demonstrate the bases for the test cases, test procedures, test results, test conclusions and corrective actions.

Where appropriate, results of analyses to demonstrate the adequacy of major modifications will be included, such as uncertainty analyses of algorithms employing the CE-1 DNBR correlation. The safety related impact of the software changes relative to the previous software version will be discussed.

The document will also describe the number and types of software implementation test cases for the SONGS CPC software.

(3) The generation of detailed software design documentation and test documentation is included as part of the structured QA design documentation described in response to NRC position 16 on ANO-2. These types of design documents will be used in the design process on San Onofre 2 and 3 and include CPC and CEAC Functional Design Specifications and a Data Base Document.

(4) As noted in the response to San Onofre Question 032.30, the San Onofre 2 and 3 software changes are not expected to impact Technical Specifications. Should a change to Technical Specifications be found to be necessary, the recommended changes will be submitted on a schedule appropriate to the Technical Specification review.

(5) Subsequent to the completion of the San Onofre CPC software base design, any revisions to the San Onofre software base design and test documentation will be prepared in accordance with the CPC Protection Algorithm Software Change Procedure (CEN-39(A)-P, Revision 2, and its Supplement 1-P Revision 1) established in response to NRC Position 19 on ANO-2. SCE intends to utilize the services of a qualified consultant (may be an SCE employee) to independently review all software changes prior to implementation at San Onofre Units 2 and 3. Documentation to which modifications are made to produce a set of revised documents will be the previous revision level at that documentation for San Onofre 2 and 3.

REFERENCES: Response to Question 032.30. No FSAR change was made.

1. NRC Audit of the ANO-2 CPCS at the Arkansas Site, September 13-14, 1979.

2. DC Trimble (AP + L), "CPC Software Modifications," Letter #2-010-25 to NRC Director of Nuclear Reactor Regulation, dated January 31, 1980.

3. CEN-135(S)-P, "CPC/CEAC Software Modifications for San Onofre Unit 2 (Response to NRC Question 221.18)."

TABLE 221.18-1
CPC/CEAC SOFTWARE DIFFERENCES BETWEEN SONGS 2 AND 3 AND ANO-2

AREA OF CHANGE	DESCRIPTION	REASON
A. GENERAL CPC/CEAC MODIFICATIONS		
1. CEAC Subgroup Deviation Monitoring	The CEAC algorithm is modified to process a 2 CEA subgroup.	SONGS Units 2 and 3 have a 2-SFA subgroup.
2. Addressable Constants	Addressable constants are added for power distribution constants measured during start-up.	The accuracy of core power and power distribution calculations are improved.
3. Addressable Constants	LPD and DNBR pretrip alarm setpoints are made addressable constants.	Flexibility in setting pretrip alarm setpoints is added.
4. POWER Program	Fixed numbers currently in algorithms are changed to constants in the data base.	The flexibility to change power distribution data without algorithm design changes is enhanced.
5. FLOW and STATIC Programs	Improved curve fits for coolant properties (enthalpy, specific volume) are provided.	The accuracy of coolant property determination is increased.
6. Heat Flux and LPD Filtering	Heat flux and LPD filtering at low powers is provided.	The uncertainties below 20% power for increasing power events are reduced.
7. Minimum Compensated Cold Leg Temperature	The minimum compensated cold leg temperature calculation is modified.	The accuracy of neutron flux determination is increased.
8. Compensation of Planar Radials	Temperature compensation of planar radials is added.	The accuracy of the planar radial calculation is increased.
B. OPERATOR INTERFACE		
1. Point ID Table	The Point ID table is revised.	More useful CPC information is provided for startup testing and normal operation.
C. DNBR CALCULATION		
1. DNBR Correlation	The W-3 DNBR calculation is replaced by the CE-1 DNBR calculation.	The accuracy of the DNBR calculation is increased.

TMercurio:npm

QUESTION 221.20

Your response in Amendment 17 to question 221.18, item (3), is not clear. This part of the question requested that you provide the data base constants and changes to the CPC algorithms. Your response stated that "Generation of detailed software design documentation and test documentation is included as part of the structured QA design documentation described in response to NRC position 16 on ANO-2. These types of design documents will be used in the design process on San Onofre 2 and 3 and include CPC and CEAC Functional Design Specifications and a Data Base Document." We see no commitment to supply this documentation to the NRC in your response. We require that you submit the CPC and CEAC Functional Design Specifications and Data Base Document for our review. The San Onofre CPC design will not be approved prior to NRC review of these documents.

RESPONSE

In order to facilitate the NRC review of CPC system software modifications, the following software design and test information will be submitted:

(1) Table 221.18-1 is a summary of the software differences between San Onofre and ANO-2. The ANO-2 baseline software is defined in the Basis to the ANO-2 Technical Specification (Appendix A to Operating License NPF-6) Section 2.2.1 as modified by References 1 and 2.

CEN-135(S)-P is a detailed description of the software changes described in Table 221.18 which includes algorithm descriptions in symbolic algebra.

(2) A software test results document giving the results of Phase I and Phase II software testing will be submitted by December, 1980. Sufficient information will be provided to demonstrate the bases for the test cases, test procedures, test results, test conclusions and corrective actions.

Where appropriate, results of analyses to demonstrate the adequacy of major modifications will be included, such as uncertainty analyses of algorithms employing the CE-1 DNBR correlation. The safety related impact of the software changes relative to the previous software version will be discussed.

The document will also describe the number and types of software implementation test cases for the SONGS CPC software.

(3) The generation of detailed software design documentation and test documentation is included as part of the structured QA design documentation described in response to NRC position 16 on ANO-2. These types of design documents will be used in the design process on San Onofre 2 and 3 and include CPC and CEAC Functional Design Specifications and a Data Base Document.

(4) As noted in the response to San Onofre Question 032.30, the San Onofre 2 and 3 software changes are not expected to impact Technical Specifications. Should a change to Technical Specifications be found to be necessary, the recommended changes will be submitted on a schedule appropriate to the Technical Specification review.

(5) Subsequent to the completion of the San Onofre CPC software base design, any revisions to the San Onofre software base design and test documentation will be prepared in accordance with the CPC Protection Algorithm Software Change Procedure (CEN-39(A)-P, Revision 2, and its Supplement 1-P Revision 1) established in response to NRC Position 19 on ANO-2. SCE intends to utilize the services of a qualified consultant (may be an SCE employee) to independently review all software changes prior to implementation at San Onofre Units 2 and 3. Documentation to which modifications are made to produce a set of revised documents will be the previous revision level at that documentation for San Onofre 2 and 3.

REFERENCES: Response to Question 032.30. No FSAR change was made.

1. NRC Audit of the ANO-2 CPCS at the Arkansas Site, September 13-14, 1979.

2. DC Trimble (AP + L), "CPC Software Modifications," Letter #2-010-25 to NRC Director of Nuclear Reactor Regulation, dated January 31, 1980.

3. CEN-135(S)-P, "CPC/CEAC Software Modifications for San Onofre Unit 2 (Response to NRC Question 221.18)."

QUESTION 212.159

Loss of Component Cooling Water (CCW) to Reactor Coolant Pumps (RCPs)

In response to staff's request 010.58 concerning the possible complete loss of CCW supply to the RCPs due to a postulated single active failure in the CCW supply system, you stated that tests have been completed to demonstrate that the San Onofre Units No. 2 and 3 RCPs are capable of continued operation for a minimum of 30 minutes without shaft seizure or excessive seal leakage if CCW is lost. We have reviewed your test results presented in the FSAR, in ASME paper No. 80-C2/PVP-28 and in the Byron-Jackson test report GS-1520 and find that we require additional information for evaluation. They are as follows:

(1) The San Onofre 2 & 3 design incorporates a single CCR supply and return line to all four RCPs rotor bearing and pump seal heat exchangers in each unit. A single active failure of the isolation valves in either the CCW supply line or the return line will cause complete loss of CCW supply to both motor bearing and pump seal heat exchangers of all four RCPs. Your tests were performed separately for the motor bearing and pump seal on loss of CCW supply. Explain why your tests did not simulate a complete loss of CCW supply to all heat exchangers simultaneously. Justify that the results of the separate tests for the motor bearing and pump seal are applicable to the integral effects which would result from a postulated complete loss of CCW.

(2) The test report does not address the pump motor speed during the tests. State the motor test rpm and confirm that these motor speeds are compatible to the rpm associated with the design flow or otherwise justify why the lost results are applicable.

(3) Describe whether the motor was tested under "load" or "no load" conditions. Explain how the test results are applicable to when a loss of CCW under normal plant operating conditions.

(4) Describe the RCP inlet and discharge pressures during the tests and confirm that these pressures are compatible to normal plant operating conditions or otherwise justify why the test results are applicable.

(5) The test report indicates that pump shaft vibration was noted during the tests. Explain why prolonged vibration (30 minutes) is acceptable.

RESPONSE

(1) The test of loss of CCW to the RC Pump was performed separately for the motor and the pump in order to maximize the test time, i.e., if the test had to be terminated before 30 minutes duration because of failure of the pump shaft seals then it would not be known whether the motor could have continued to run for 30 minutes and vice versa.

The results of the separate tests for the motor bearings and pump shaft seal are valid to relate the integral effects which would result from a postulated complete loss of CCW simultaneously to both motor and pump for the following reasons: Unless there is a gross failure of some component such as pump shaft seals or motor bearings, the increasing temperature in the motor oil reservoirs does not affect the pump operation, i.e., it would neither accelerate nor delay the pump shaft seal failure. Also, as long as there is no gross failure of the pump shaft seals, the increasing instability of the shaft seal controlled bleedoff flow has no effect on the operation of the motor bearings or the operation of the motor itself. Thus, it is our judgment that the test as performed represents the integrated reactor coolant pump and motor performance during a loss of CCW.

(2) The RC pump is driven by an induction motor which operates at constant speed. The nominal rated speed is 1200 rpm. The actual rpm is about 1188 rpm. This is the rpm at which the pump operates, whether in a test loop for performance testing or loss of CCW tests or in the reactor coolant system in the nuclear power plant. The pump discharge flow in the pump manufacturer's test loop is comparable to that in the reactor coolant system, except for a possible minor variation due to differences in piping resistance, i.e., it is not feasible to duplicate the reactor coolant system (including the steam generator and the reactor vessel) in the pump manufacturer's shop. During the loss of CCW tests the rpm was monitored but since deviations were neither expected nor observed, no special mention was made of this fact.

(3) The motor was tested "under load", i.e., coupled to the pump, during the loss of CCW test. Every attempt was made to simulate actual operating conditions:

a. The motor was coupled to the pump.

b. The operation of the pump/motor assembly was stabilized in "hot" condition to simulate field conditions, and the test parameters were recorded in the steady state condition in order to define the baseline prior to shutting off the CCW flow. These parameters were continuously monitored during the loss of CCW test and also after the test.

(4) The RCP inlet and discharge pressures were stabilized in the hot operating condition prior to the initiation of the loss of CCW test. The RC pumps are tested at the pump manufacturer's shop under normal plant operating conditions of pressure, flow, head and temperature. The loss of CCW test was performed at hot operating conditions because this was a more severe test and this simulated best the conditions likely to be encountered during actual plant operation.

(5) The test report indicates that some increase (beyond that measured during normal steady state pump operation) in pump shaft vibration was noted during the test. This observation would normally be made in a test report since this parameter was measured and was one of the parameters used in the determination as to whether to continue the test or to abort for shop safety reasons. The values of shaft vibration which were observed during the loss of CCW test were not of significant concern and a post-test examination of the pump indicated no observable loosening of the bolts. It should also be noted that at the end of the 30 minute test the shaft vibration decreased again, i.e., after restoration of CCW flow.

QUESTION 331.19

It is our position that all accessible portions of the spent fuel transfer tube be shielded during fuel transfer. Use of removable shielding for this purpose is acceptable. This shielding shall be such that the resultant contact radiation levels shall be no greater than 100 rads per hour. All accessible portions of the spent fuel transfer tube shall be clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed to alert personnel if temporary fuel transfer tube shielding is removed during fuel transfer operations. Please provide a description of your modified design to comply with these positions.

RESPONSE

All normally accessible areas of the fuel transfer tube area are shielded to levels below 100 mR/hr or have access restricted by means of locked and/or alarmed barriers. All access barriers and concrete hatches are clearly identified as high radiation areas during fuel transfer operations.

Inside containment 100 lbs/ft^3 lead foam is used to fill the seismic gap between the containment wall and transfer tube concrete shield. Figures 12.3-30 and 12.3-31 show the fuel transfer tube shield design inside containment.

Outside containment a labyrinth shield design is used to reduce streaming through the seismic gap between the containment and fuel handling buildings. Figures 12.3-31 through 12.3-34 show the shield design outside containment.

REFERENCES: See revised FSAR subsections 12.3.2.2.2, 12.3.2.2.4 and new FSAR figures 12.3-30, 12.3-31, 12.3-32, 12.3-33 and 12.3-34.

Components of the letdown portion of the chemical and volume control system are located in shielded compartments that are normally Zone V, restricted access areas. Shielding is provided for each piece of equipment in the letdown system consistent with its postulated maximum activity (subsection 12.2.1) and with the access and zoning requirements of adjacent areas. This equipment includes the regenerative heat exchanger and letdown lines.

After shutdown, the containment is accessible for limited periods of time and all access is controlled. Areas are surveyed to establish allowable working periods. Dose rates are expected to range from 0.5 to 1000 mrem/h, depending on the location inside the containment (excluding reactor cavity). These dose rates result from residual fission products, neutron-activated materials, and corrosion products in the reactor coolant system.

Spent fuel is the primary source of radiation during refueling. Because of the extremely high activity of the fission products contained in the spent fuel elements and the proximity of Zone II areas, extensive shielding is provided for areas surrounding the spent fuel pool and the fuel transfer canal to ensure that radiation levels remain below zone levels specified for adjacent areas. Water provides the shielding over the spent fuel assemblies during fuel handling.

Six feet of concrete shielding is provided around the fuel transfer tube inside containment. To prevent radiation streaming, the six inch seismic gap between the containment wall and transfer tube concrete shield is filled with 100 lbs/ft^3 lead foam. With this shield design the maximum dose rate from a single spent fuel assembly passing through the fuel transfer tube, assuming three days decay, will be less than 5 mrem/hr. To provide access to the fuel transfer tube for local leakage rate tests, concrete hatches are provided above the transfer tube. Figures 12.3-30 and 12.3-31 show the fuel transfer tube shield design inside containment.

The secondary shield is a reinforced concrete structure surrounding the reactor coolant equipment, including piping, pumps, and steam generators.

This shield protects personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products carried away from the core by the reactor coolant. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield is sized to allow limited access to the containment during full power operation. The thickness of secondary shield walls is 4 feet 0 inches.

12.3.2.2.3 Auxiliary Building Shielding Design

During normal operation, the major components in the auxiliary building with potentially high radioactivity are those in the chemical and volume control system, the coolant radwaste and miscellaneous liquid radwaste systems, the waste gas system, and sold radwaste system.

Shielding is provided as necessary around the following equipment in the radwaste building to ensure that the radiation zone and access requirements are met for surrounding areas.

A. Letdown heat exchangers and piping

B. Purification and deborating ion exchangers

C. Chemical and volume control tank

12.3.2.2.4 Fuel Building Shielding Design

Concrete shield walls surrounding the spent fuel cask loading and storage area, fuel transfer and storage pools, and fuel transfer tube between the containment and fuel transfer pool are sufficiently thick to limit radiation levels outside the shield walls in accessible areas above elevation 45 feet to Zone II and below elevation 45 feet to Zone III. Access to the fuel transfer tube through the concrete radiation shield is provided by a labyrinth entrance and hatch through the floor of the shield.

Water in the spent fuel pool provides shielding above the spent fuel transfer and storage areas. Radiation levels at the fuel handling equipment are limited to 2.5 mrem/hr.

A labyrinth shield design is used to reduce streaming through the seismic gap between the containment and fuel handling building. No lead foam is used in this gap. This labyrinth reduces the dose rate to less than 100 mr/hr in most areas considering a single spent fuel assembly with three days decay. To preclude the possibility of personnel standing in a direct line with the labyrinth opening alarmed access control doors restrict access to this area. Access to the fuel transfer tube is gained by climbing a ladder at the 15 foot elevation into an access opening. Labyrinth shields reduce the radiation streaming through this access opening and seismic gaps to levels less than 100 mr/hr in normally accessible areas. Locked and alarmed gates along with permanent access barriers restrict access to the fuel transfer tube opening. Figures 12.3-32 through 12.3-34 show the shield design along with peak dose rates.

The spent fuel pool cooling and cleanup (SFPCC) system (section 9.1) shielding is based on the maximum activity discussed in subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the SFPCC system to be shielded includes the SFPCC heat exchangers, pumps and piping. (SFPCC filters and ion exchangers are located in the auxiliary building.)

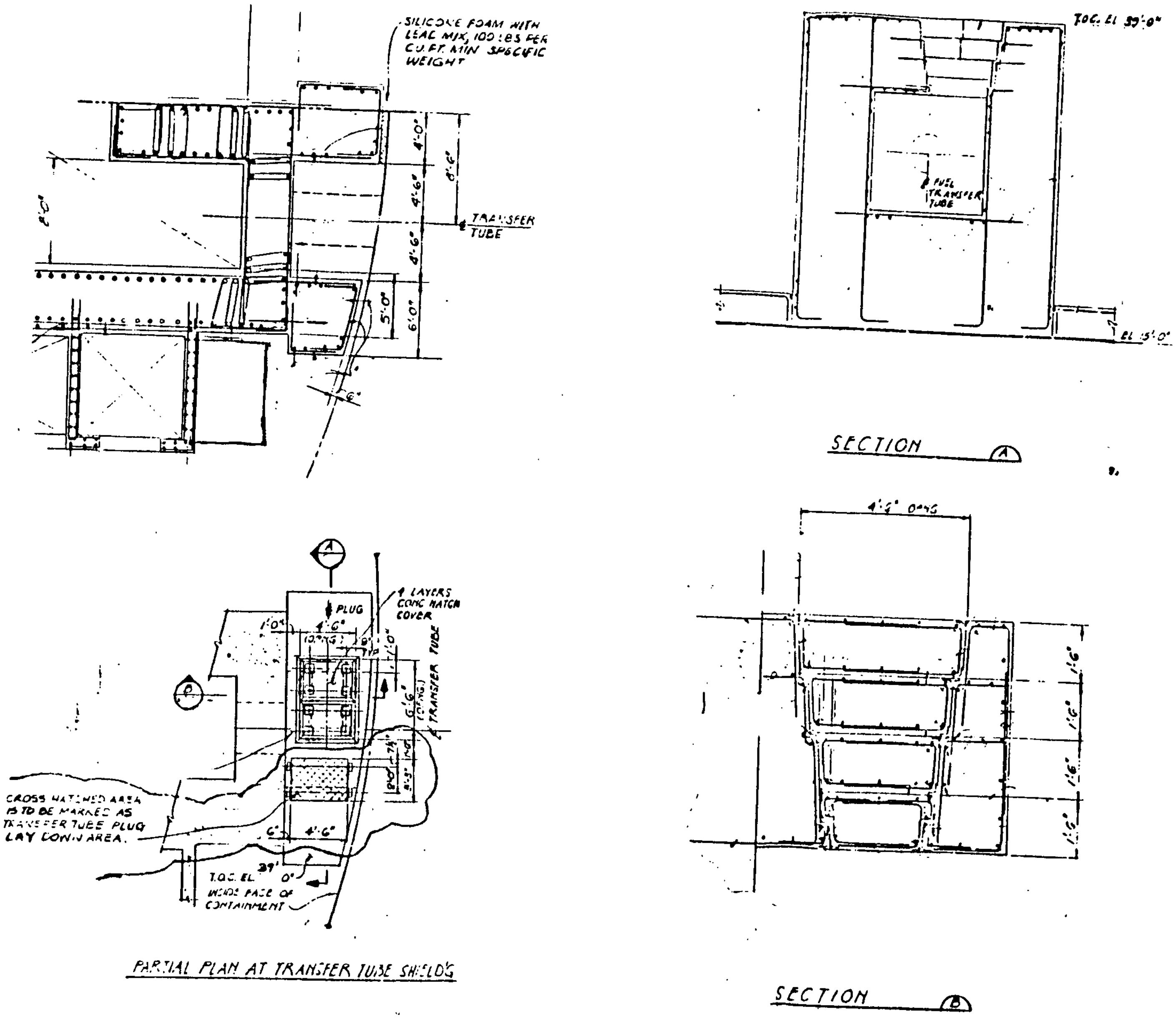
12.3.2.2.5 Safety Equipment Building Shielding Design

Following reactor shutdown, the low pressure safety injection (LPSI) system pumps and shutdown cooling (SDC) system heat exchangers are in operation to remove heat from the reactor coolant system. The radiation levels in the vicinity of this equipment will temporarily reach Zone V levels due to corrosion and fission products in the reactor water. Shielding is provided to attenuate radiation from SDC equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas.

Equipment in the component cooling water system is normally shielded to Zone II as shown in figures 12.3-1 through 12.3-25. Shielding is provided for this equipment to ensure Zone IV access after a design basis loss-of-coolant accident (LOCA).

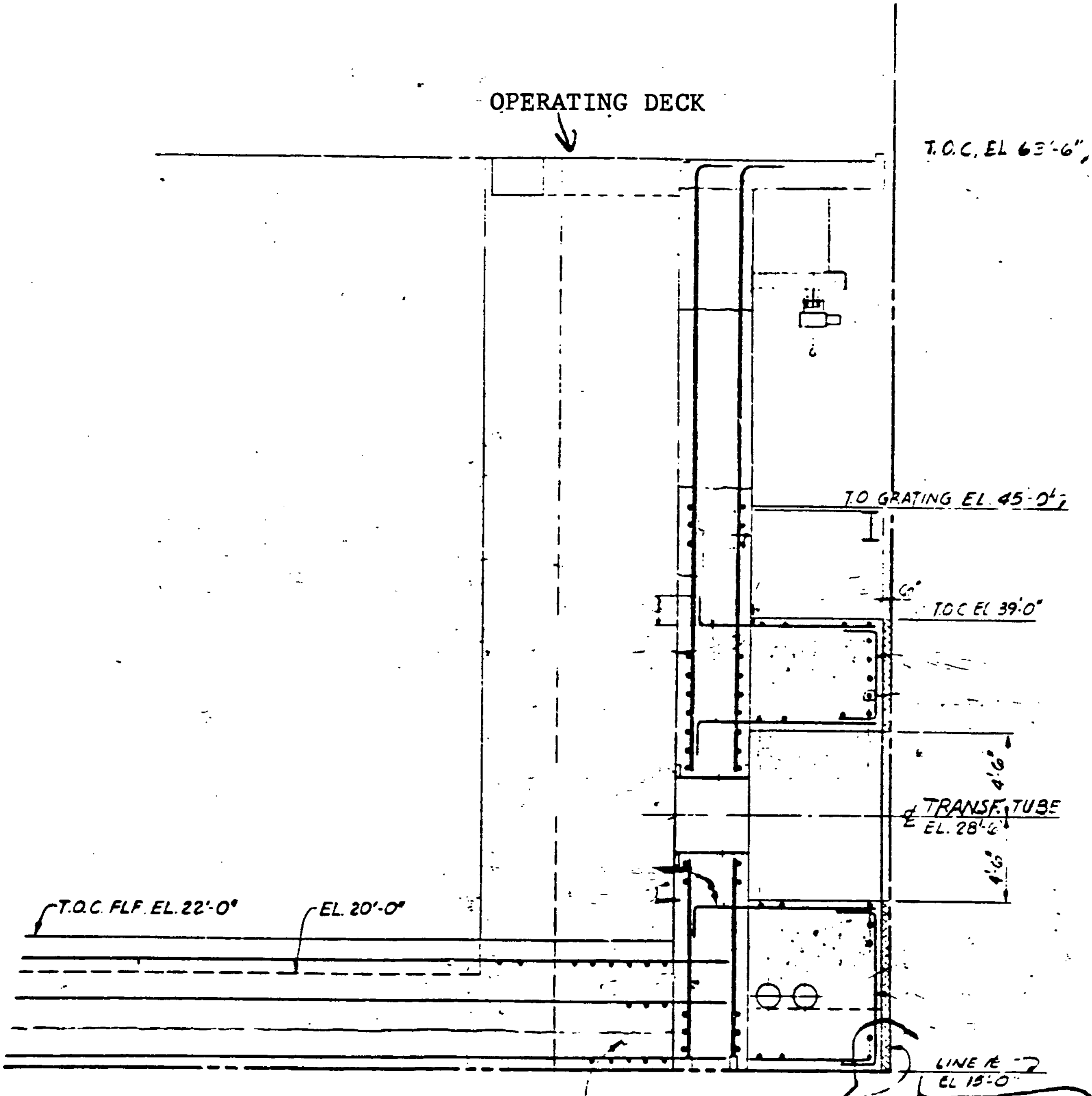
12.3.2.2.6 Turbine Building Shielding Design

Radiation shielding is not required for any equipment in the steam and power conversion system located in the turbine building. The steam generator blowdown system components located in the turbine building are provided with concrete shield walls. All areas in the turbine building are classified Zone I with the exception of areas housing the steam generator blowdown demineralizers which are classified Zone IV.

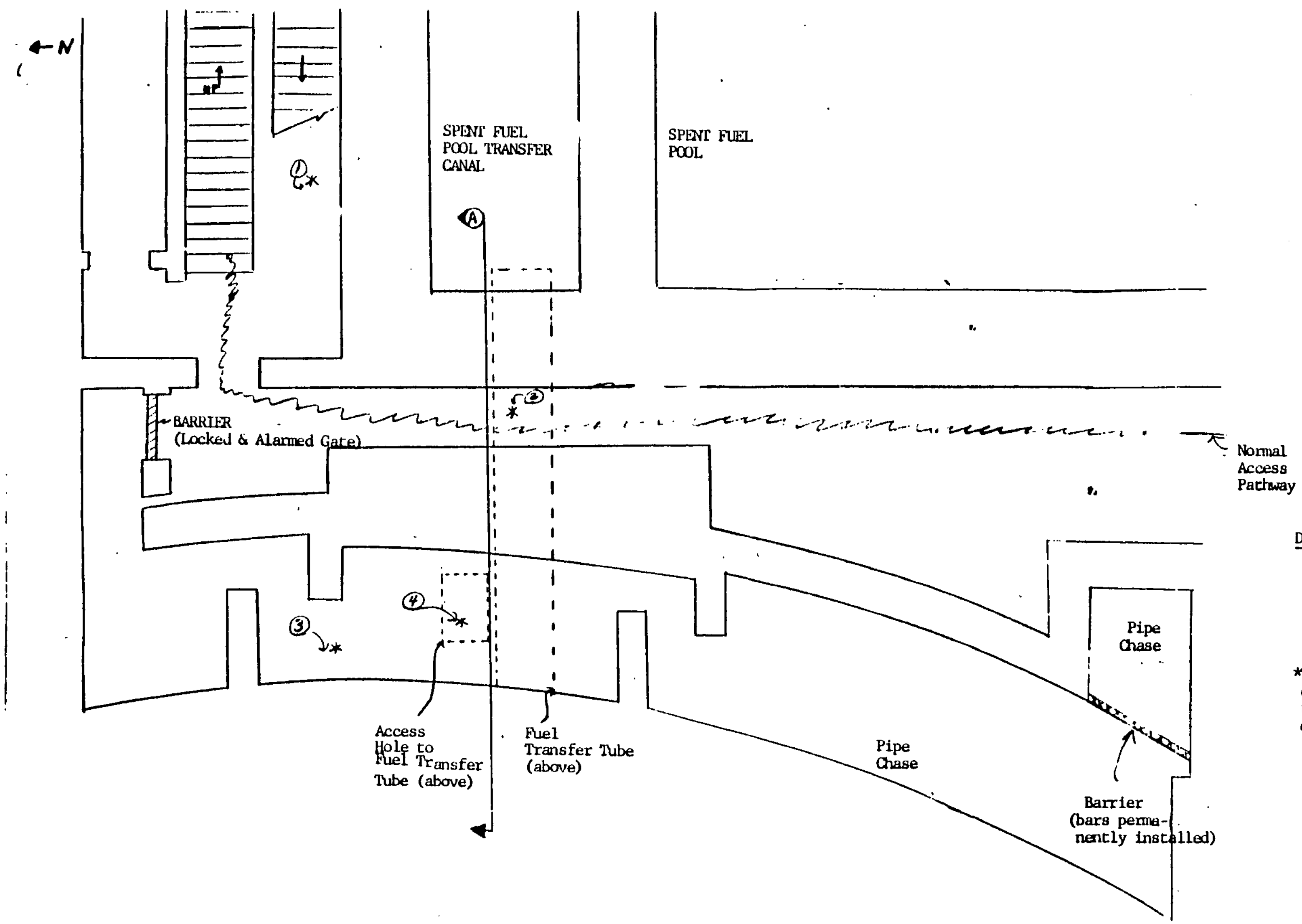


Fuel Transfer Tube
Inside Containment
Concrete Structure

Figure 12.3-30



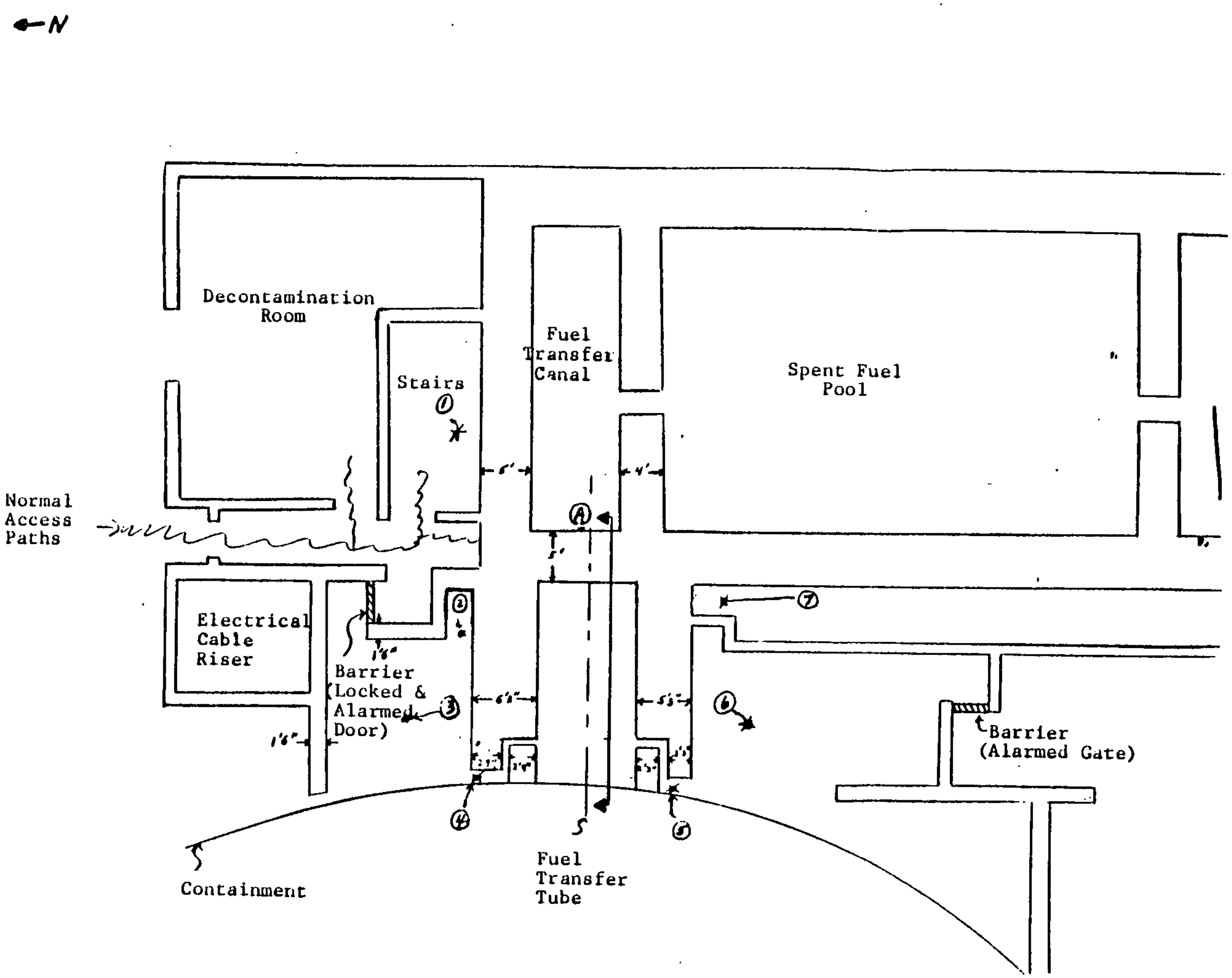
FUEL TRANSFER TUBE INSIDE CONTAINMENT
CONCRETE STRUCTURE SECTION VIEW
Figure 12.3-31



DOSE POINT	PEAK DOSE RATE*
1	.8 mr/hr
2	40. mr/hr
3	>100 mr/hr
4	> 10 R/hr

*These dose rates are based on single fuel assembly in transfer tube with three days decay.

FUEL TRANSFER TUBE OUTSIDE CONTAINMENT AT ELEV. 15'-0"
Figure 12.3-32
(Unit 2 shown)

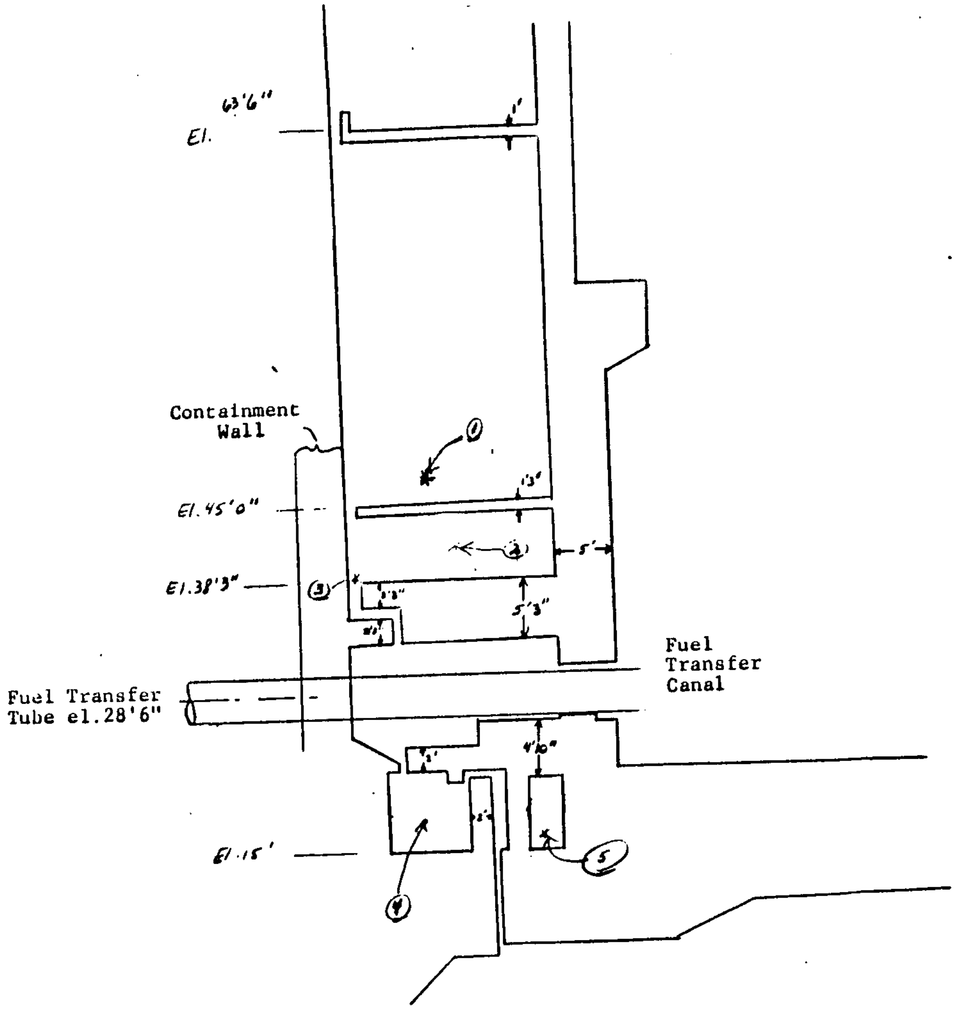


Dose Point	Peak Dose Rate *
1	.8 mr/hr
2	1.0 mr/hr
3	2.5 mr/hr
4	13 R/hr
5	60 R/hr
6	60 mr/hr
7	12 mr/hr

* These dose rates are based on single fuel assembly in transfer tube with three days decay.

Fuel Transfer Tube Outside Containment at elev. 30'

Figure 12.3-33
Unit 2 Shown



Dose Point	Peak Dose Rate*
1	15 mr/hr
2	60 mr/hr
3	60 R/hr
4	>10 R/hr
5	40 mr/hr

* These dose rates are based on single fuel assembly in transfer tube with three days decay.

Fuel Transfer Tube
Outside Containment
Sectional View A

Figure 12.3-34

(Unit 2 Shown)