Southern California Edison Company

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September 9, 1992

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U. S. Nuclear Regulatory Commis n
Attention: Document Control
Washington, D.C. 20555

Gentlemen:

Subject: **Docket Nos. 50-361 and 50-362**
Reference Temperatures for Pressurized Thermal Shock
(10 CFR 50.61) - Reactor Vessel Beltline Materials
San Onofre Nuclear Generating Station
Units 2 and 3

Reference: September 9, 1992, letter from Harold B. Ray (SCE) to Document Control Desk (NRC), Subject: Docket No. 50-362, Amendment Application No. 102, Changes to Technical Specifications 3/4.4.8.1, 3.4.8.3.1 and 3.4.8.3.2, San Onofre Nuclear Generating Station, Unit 3

ransmittal provides the information required by paragraph (b)(1) of 10 CFR , "Fracture Toughness Requirements for Protection Against Pressurized hermal Shock Events," for San Onofre Units 2 and 3. The enclosed information is being submitted in conjunction with the above reference which updates the San Onofre Unit 3 reactor vessel pressure-temperature limits. Included are the current and projected values of the reference temperatures (RT_{PTS}) for the reactor vessel beltline materials at the inner vessel surface. Also included as required, are 1) the bases for the current and projected values and the assumptions regarding core loading patterns, 2) the copper and nickel contents, and 3) the fluence values used in the calculation for each beltline material.

The reference temperatures have been calculated by the method given in paragraph (b)(2) of 10 CFR 50.61. The results are that no reference temperature for any beltline material is projected to exceed the Pressurized Thermal Shock (PTS) screening criteria prior to February 16, 2022 for Unit 2 or November 15, 2022 for Unit 3. These dates are 40 years after the issuance of the respective operating licenses, which are well beyond October 18, 2013, the current expiration date of both the Units 2 and 3 operating licenses.

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Because the reference temperatures for the beltline materials are projected to remain below the PTS screening criteria, 10 CFR 50.61 requirements other than those in paragraph (b)(1) are not applicable to San Onofre Units 2 and 3. As required by paragraph (b)(1), the assessment provided by this letter will be updated by Southern California Edison if there is a significant change in the projected values of RT_{PTS} or upon request for a change to the expiration date for operation of the facilities.

The assessment provided by this submittal is applicable to both Units 2 and 3. Therefore, this assessment will not be resubmitted in conjunction with the submittal of the update to the Unit 2 pressure-temperature limits by October 30, 1992.

If you have any questions regarding the results of this PTS assessment, please let me know.

Very truly yours,

R M Rosenblum

Enclosure

cc: J. B. Martin, Regional Administrator, NRC Region V
C. W. Caldwell, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3
M. B. Fields, NRR Project Manager, San Onofre Units 2 and 3

RESPONSE TO 10 CFR 50.61
PRESSURIZED THERMAL SHOCK (PTS) RULE REQUIREMENTS
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

10 CFR 50.61 describes the Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events. Specifically, paragraph (b)(1) requires the licensee of each pressurized water reactor (PWR) for which an operating license has been issued, to submit projected values of reference temperatures for pressurized thermal shock (RT_{PTS}) at the inner vessel surface of reactor vessel materials from the time of submittal to the expiration of the operating license. Paragraph (b)(2) provides the PTS screening criteria and prescribes the method by which the values of RT_{PTS} must be calculated. The PTS screening criteria are 270°F for plates, forgings, and axial weld materials, and 300°F for circumferential weld materials. For the purpose of comparison with these criteria, the value of RT_{PTS} for the reactor vessel must be calculated as follows. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline.

Equation 1: $RT_{PTS} = I + M + \Delta RT_{PTS}$

where:

I = initial reference temperature (RT_{NDT}) of the unirradiated material as defined in the ASME Code, Paragraph NB-2331

M = the margin to be added to account for uncertainties in the initial RT_{NDT}, copper and nickel contents, fluence, and the calculational procedures

ΔRT_{PTS} = the mean value of the shift in reference temperature caused by irradiation, calculated as follows:

Equation 2: $\Delta RT_{PTS} = (CF)\ f^{(0.28-(.10 \log f))}$

where:

CF(°F) = the chemistry factor, a function of nickel and copper content. CF is given in 10 CFR 50.61 Table 1 for welds and Table 2 for base metal (plates and forgings).

f = the best estimate neutron fluence, in units of $10^{19} n/cm^2$ (E greater than 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question.

The current values of RT_{PTS} as of September 11, 1992, are provided in the attached Table 1 for Unit 2 and Table 3 for Unit 3. The projected values of RT_{PTS} at end of life (EOL) are provided in the attached Tables 2 and 4 for Units 2 and 3, respectively. The Heat Affected Zone (HAZ) material is no longer included in the surveillance program for beltline materials pursuant to ASTM Standard E185-92 (Reference 1). Therefore, the attached Tables 1 through 4 do not contain HAZ data. In addition, there are two sets of Weld 9-203 data for Unit 3 because two values of initial reference temperatures were calculated due to the two sets of weld wire heat/flux lot number combinations used. The bases for the current and projected RT_{PTS} values in Tables 1 through 4 are discussed below.

Initial Reference Temperature and Margin

The initial reference temperature (I) or RT_{NDT} values for the unirradiated plate and weld materials, which were determined from Charpy V-notch (C_{vn}) absorbed energy tests and drop weight tests in accordance with ASME Section III, NB-2331, are documented in Reference 1 and in Appendix A to Reference 2. The initial reference temperatures are indicated in the column labeled "Initial RT_{NDT} °F" of the attached Tables 1 through 4. Because there was no measured RT_{NDT} value available for Weld 8-203, a generic RT_{NDT} value of -56°F was used as required by Reference 3.

The margin (M) that is added to cover uncertainties in the values of RT_{NDT}, copper and nickel content, fluence, and the calculational procedures is 56°F for welds and 34°F for base metals when measured values of initial reference temperatures are used as prescribed by Reference 3, paragraph (b)(2)(ii). Because there was no measured value available for Weld 8-203, a margin of 66°F was used as required by Reference 3.

Copper and Nickel Content

The copper (Cu) and nickel (Ni) contents of vessel beltline materials are based on updated material properties evaluated in response to Generic Letter 9[illegible], "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," Revision 1. The updated material properties are documented in Reference 1 and attached as Appendix A to Reference 2. Because there was no measured value available for the Cu and Ni contents of Weld 8-203, upper bound values were used as required by Reference 3.

Chemistry Factor

Chemistry Factor (CF) is a function of the copper and nickel content of the vessel beltline materials. CF is obtained from Reference 3, Tables 1 and 2 for welds and base metal (plates and forgings), respectively. Linear interpolation is permitted to determine chemistry factors.

Fluence

The fluence (f) for all plates and welds (except Weld 8-203) is assumed to be the peak vessel fluence at the core centerline. The peak fluence is based on core power distributions typical of low leakage checkerboard loading patterns and extended cycle lengths (i.e., 24-month nominal fuel cycle) similar to Unit [illegible] Cycle 5. The San Onofre Units 2 and 3 surveillance program described in Reference 4 obtains fast neutron flux measurements from threshold detectors inserted into each of the six irradiation capsules. The location and identification of the Units 2 and 3 beltline materials are shown in Reference 5. Figures 5.2-8 and 5.2-9 are attached for your information.

Except for Weld 8-203, updated fluence projections in Tables 1 through 4 were obtained from the test results and analysis of the surveillance capsule removed from Unit 3 at 4.33 EFPY (Reference 6). A reduced fluence has been credited for Weld 8-203 because the weld is not actually located in the beltline region (References 1 and 4).

The current fluence is calculated to be the fluence on September 11, 1992 scheduled submittal of this report. On this date, Units 2 and 3 will have experienced a vessel exposure of approximately 6.45 EFFY and 6.10 EFPY, respectively. The fluence for all plates and welds (except Weld 8-203) with a Unit 2 exposure of 6.45 EFPY has been computed to be 0.933 X 10^{19} n/cm^2. The equivalent fluence for a Unit 3 exposure of 6.10 EFPY has been computed to be 0.889 X 10^{19} n/cm^2. The fluence for Weld 8-203 with a Unit 2 exposure of 6.45 EFPY has been computed to be 0.0254 X 10^{19} n/cm^2. The equivalent fluence for Weld 8-203 with a Unit 3 exposure of 6.10 EFPY has been computed to be 0.0242 X 10^{19} n/cm^2.

The projected fluence is computed to be the fluence at 32 EFPY (EOL). The maximum design basis integrated fast neutron (E>1 Mev) fluence at the vessel inside surface, including tolerance, has been computed to be 4.20 X 10^{19} n/cm^2 for a vessel exposure of 32 EFPY. A reduced fluence of 0.114 X 10^{19} n/cm^2 at 32 EFPY has been credited for Weld 8-203.

Current and Projected Values of RT_{PTS}

Current values of RT_{PTS} for San Onofre Units 2 and 3 as of September 11, 1992, are given in Tables 1 and 3. By September 11, 1992, Units 2 and 3 are expected to have experienced approximately 6.45 EFPY and 6.10 EFPY of operation, respectively. For these current exposures the maximum values of RT_{PTS} for any beltline material are 117.8°F (Plates C-6404-1, 2, 3, and 4) for Unit 2 and 144.8°F (Plate C-6802-1) for Unit 3.

Projected values of Units 2 and 3 RT_{PTS} in Tables 2 and 4 are given for 32 EFPY (EOL). The EOL dates are February 16, 2022 and November 15, 2022 for Units 2 and 3, respectively. These dates are forty years after the issuance of the respective unit operating licenses and are well beyond October 18, 2013, the current expiration date of both operating licenses. The maximum projected RT_{PTS} values at 32 EFPY for any beltline material are 146.5°F (Plate C-6404-5) for Unit 2 and 159.6°F (Plate C-6802-1) for Unit 3.

The Units 2 and 3 projected RT_{PTS} values in this report and the projected RT_{PTS} values previously reported in Reference 7 both meet the Reference 3 PTS screening criteria. However, they are different, and as required by 10 CFR 50.61(b)(1), the difference must be justified. The difference in projected RT_{PTS} values at 32 EFPY is due to the following:

- The final 10 CFR 50.61 PTS rule (Reference 4) primarily changed the method for calculating the shift in RT_{PTS} and the "Margin" to be consistent with Regulatory Guide 1.99, Revision 2, May 1988. The previous method of calculating the shift in RT_{PTS} and "Margin" was based on the original 10 CFR 50.61 rule and was inconsistent with Regulatory Guide 1.99.

- Updated fluence projections from the analysis of the surveillance capsule removed from Unit 3 in May 1990 at 4.33 EFPY were used (Reference 6). Previously submitted projections of RT_{PTS} were based on the original calculated fluences.

- Updated material properties evaluated in response to Generic Letter 92-01, "Reactor Vessel Integrity, 10 CFR 50.54(f)" were used (Reference 1). The plate Cu and Ni content were obtained by averaging two measurements made by ABB-Combustion Engineering when the plates were received and when the surveillance program was defined. The weld Cu and Ni content were obtained from the UFSAR except for Weld 9-203 which was obtained from a welding material certification (WMC). Where chemistry data was not available, 0.35% Cu and 1.00% Ni were assumed. Initial fracture toughness data (initial RT_{NDT}) for plates were obtained from the material certification reports (MCRs) and the baseline surveillance in accordance with the most recent version of ASME Section III, NB-2331. Initial fracture toughness data for the beltline welds were obtained from the UFSAR and WMC. Previous values did not consider all of the above.

Conclusion

The projected values of RT_{PTS} for all beltline materials indicated in Tables 1 through 4 do not exceed the PTS screening criteria in 10 CFR 50.61(b)(2). The PTS screening criteria are 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials. In the year 2022 the limiting materials in the reactor vessel beltline region are projected to be intermediate shell plate C-6404-5 for Unit 2 with the RT_{PTS} equal to 146.5°F, and intermediate shell plate C-6802-1 for Unit 3 with the RT_{PTS} equal to 1[illegible].6°F. The results indicate that reactor vessel integrity will be maintained for the San Onofre Units 2 and 3 reactor vessels throughout their service lives. Hence, the actions required by Paragraphs (b)(3) and (b)(4) are not applicable.

References

1. July 6, 1992 letter from R. M. Rosenblum (SCE) to Document Control Desk (SCE), Subject: Docket Nos. 50-361 and 50-362, Generic Letter 92-01, Revision 1 "Reactor Vessel Structural Integrity, 10 CFR 50.54(f), San Onofre Nuclear Generating Station, Units 2 and 3.

2. Calculation N-0220-026, Revision 1, "SONGS 2/3 Reference Temperature for Pressurized Thermal Shock," July 31, 1992.

3. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Final Rule Number 22300, Volume 56, No. 94, May 15, 1991.

4. Updated Final Safety Analysis Report, San Onofre Nuclear Generating Station, Units 2 and 3, Sections 5.2 and 5.3.

5. Updated Safety Analysis Report, San Onofre Nuclear Genarating Station, Units 2 and 3, Figures 5.2-8 and 5.2-9.

6. Westinghouse Report WCAP-12920, " Analysis of the Southern California Edison Company San Onofre Unit 3 Reactor Vessel Surveillance Capsule Removed from the 97° Location," March 1991.

7. January 22, 1986 letter from M. O. Medford (SCE) to G. Knighton (NRC), Subject: Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Units 2 and 3.

TABLE 1

CURRENT VALUES OF RT_{PTS} FOR REACTOR VESSEL BELTLINE MATERIALS - 6.45 EFPY
SAN ONOFRE UNIT 2

$$RT_{PTS} = I \text{ or } RT_{NDT} + M + \Delta RT_{PTS}$$

Component	Plate or Seam No.	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)	CF (°F)	M (°F)	Fluence (10^{19} n/cm^2)	6.45 EFPY	
								ΔRT_{PTS} (°F)	RT_{PTS} (°F)
Inter. Shell	C-6404-1	0.10	0.56	20	65	34	0.933	63.8	117.8
Inter. Shell	C-6404-2	0.10	0.59	20	65	34	0.933	63.8	117.8
Inter. Shell	C-6404-3	0.10	0.56	20	65	34	0.933	63.8	117.8
Lower Shell	C-6404-4	0.10	0.62	20	65	34	0.933	63.8	117.8
Lower Shell	C-6404-5	0.11	0.64	10	75	34	0.933	73.6	117.6
Lower Shell	C-6404-6	0.10	0.58	-10	65	34	0.933	63.8	87.8
Long. Weld	2-203A	0.03	0.90	-60	41	56	0.933	40.2	36.2
Long. Weld	2-203B	0.03	0.91	-60	41	56	0.933	40.2	36.2
Long. Weld	2-203C	0.03	0.95	-60	41	56	0.933	40.2	36.2
Long. Weld	3-203A	0.05	0.12	-50	40	56	0.933	39.2	45.2
Long. Weld	3-203B	0.04	0.06	-50	30	56	0.933	29.4	35.4
Long. Weld	3-203C	0.06	0.11	-50	42	56	0.933	41.2	47.2
Circum. Weld	8-203	0.31	1.00	-56	260	66	0.0254	51.7	61.7
Circum. Weld	9-203	0.07	0.29	-60	69	56	0.933	67.7	63.7

TABLE 2

PROJECTED VALUES OF RT_{PTS} FOR REACTOR VESSEL BELTLINE MATERIALS - 32 EFPY
SAN ONOFRE UNIT 2

$$RT_{PTS} = I \text{ or } RT_{NDT} + M + \Delta RT_{PTS}$$

Component	Plate or Seam No.	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)	CF (°F)	M (°F)	Fluence (10^{19} n/cm^2)	32 EFPY	
								ΔRT_{PTS} (°F)	RT_{PTS} (°F)
Inter. Shell	C-6404-1	0.10	0.56	20	65	34	4.20	88.9	142.9
Inter. Shell	C-6404-2	0.10	0.59	20	65	34	4.20	88.9	142.9
Inter. Shell	C-6404-3	0.10	0.56	20	65	34	4.20	88.9	142.9
Lower Shell	C-6404-4	0.10	0.62	20	65	34	4.20	88.9	142.9
Lower Shell	C-6404-5	0.11	0.64	10	75	34	4.20	102.5	146.5
Lower Shell	C-6404-6	0.10	0.58	-10	65	34	4.20	88.9	112.9
Long. Weld	2-203A	0.03	0.90	-60	41	34	4.20	56.0	52.0
Long. Weld	2-203B	0.03	0.91	-60	41	34	4.20	56.0	52.0
Long. Weld	2-203C	0.03	0.95	-60	41	34	4.20	56.0	52.0
Long. Weld	3-203A	0.05	0.12	-50	40	56	4.20	54.7	60.7
Long. Weld	3-203B	0.04	0.06	-50	30	56	4.20	41.0	47.0
Long. Weld	3-203C	0.06	0.11	-50	42	56	4.20	57.4	63.4
Circum. Weld	8-203	0.31	1.00	-56	260	66	0.114	115.4	125.4
Circum. Weld	9-203	0.07	0.29	-60	69	56	4.20	94.3	90.3

TABLE 3

CURRENT VALUES OF RT_{PTS} FOR REACTOR VESSEL BELTLINE MATERIALS - 6.10 EFPY
SAN ONOFRE UNIT 3

$$RT_{PTS} = I \text{ or } RT_{NDT} + M + \Delta RT_{PTS}$$

Component	Plate or Seam No.	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)	CF (°F)	M (°F)	Fluence (10^{19} n/cm^2)	6.10 EFPY	
								ΔRT_{PTS} (°F)	RT_{PTS} (°F)
Inter. Shell	C-6802-1	0.06	0.58	75	37	34	0.889	35.8	144.8
Inter. Shell	C-6802-2	0.04	0.58	10	26	34	0.889	25.1	69.1
Inter. Shell	C-6802-3	0.06	0.58	20	37	34	0.889	35.8	89.8
Lower Shell	C-6802-4	0.05	0.56	10	31	34	0.889	30.0	74.0
Lower Shell	C-6802-5	0.04	0.55	10	26	34	0.889	25.1	69.1
Lower Shell	C-6802-6	0.06	0.62	20	37	34	0.889	35.8	89.8
Long. Weld	2-203A	0.05	1.00	-40	68	56	0.889	65.8	81.8
Long. Weld	2-203B	0.05	1.00	-40	68	56	0.889	65.8	81.8
Long. Weld	2-203C	0.05	1.00	-40	68	56	0.889	65.8	81.8
Long. Weld	3-203A	0.04	0.16	-70	39	56	0.889	37.7	23.7
Long. Weld	3-203B	0.04	0.16	-70	39	56	0.889	37.7	23.7
Long. Weld	3-203C	0.04	0.16	-70	39	56	0.889	37.7	23.7
Circum. Weld	8-203	0.35	1.00	-56	272	66	0.0242	52.5	62.5
*Circum. Weld	9-203	0.06	0.04	-60	34	56	0.889	32.9	28.9
*Circum. Weld	9-203	0.05	0.04	-50	31	56	0.889	30.0	36.0

* For Unit 3 two sets of values were calculated for Weld 9-203 corresponding to the two sets of weld wire heat/flux lot number combinations used.

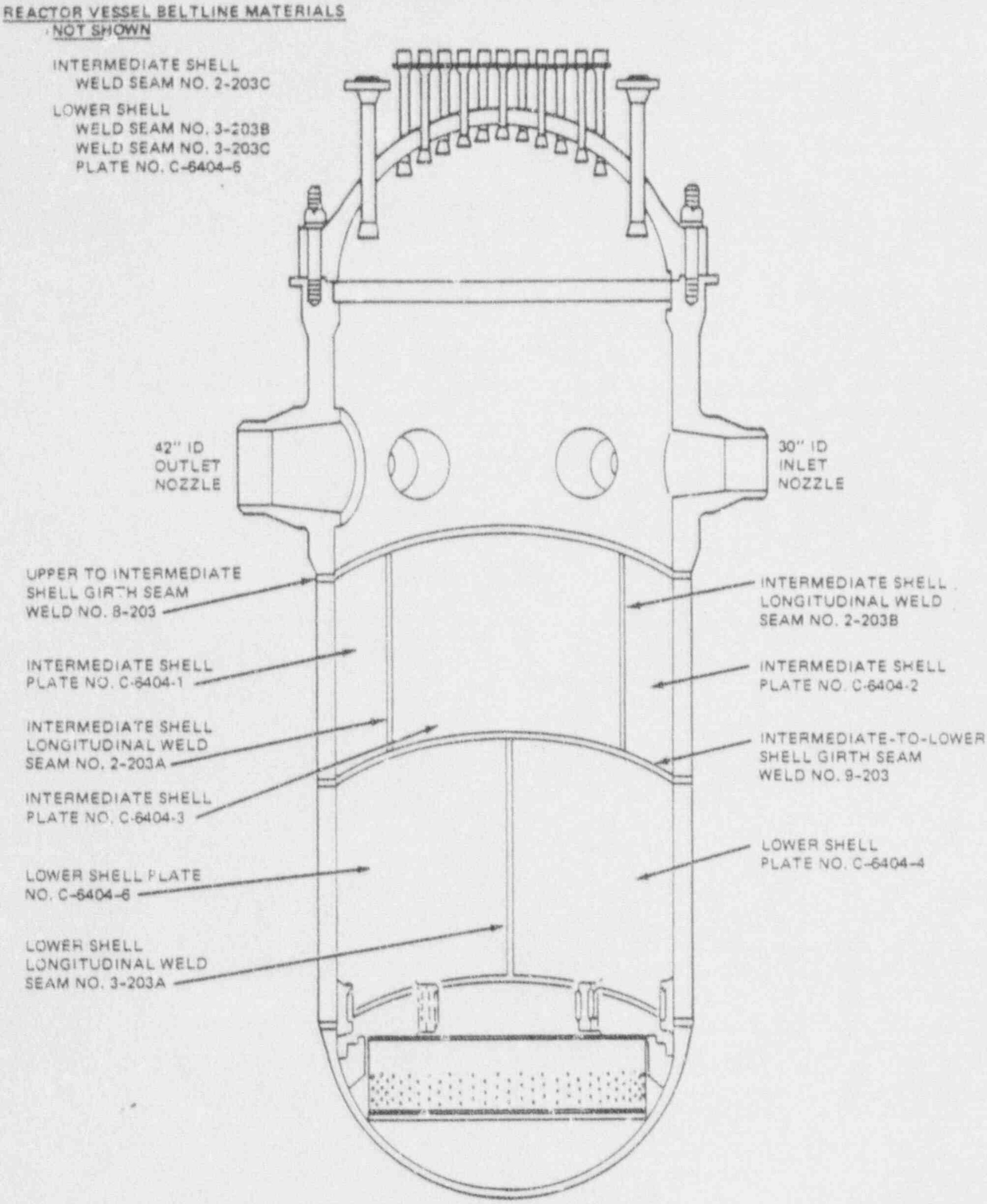
TABLE 4

PROJECTED VALUES OF RT_{PTS} FOR REACTOR VESSEL BELTLINE MATERIALS - 32 EFPY
SAN ONOFRE UNIT 3

$$RT_{PTS} = I \text{ or } RT_{NDT} + M + \Delta RT_{PTS}$$

Component	Plate or Seam No.	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)	CF (°F)	M (°F)	Fluence (10^{19} n/cm²)	32 EFPY	
								ΔRT_{PTS} (°F)	RT_{PTS} (°F)
Inter. Shell	C-6802-1	0.06	0.58	75	37	34	4.20	50.6	159.6
Inter. Shell	C-6802-2	0.04	0.58	10	26	34	4.20	35.5	79.5
Inter. Shell	C-6802-3	0.06	0.57	20	37	34	4.20	50.6	104.6
Lower Shell	C-6802-4	0.05	0.58	10	31	34	4.20	42.4	86.4
Lower Shell	C-6802-5	0.04	0.52	10	26	34	4.20	35.5	79.5
Lower Shell	C-6802-6	0.06	0.65	20	37	34	4.20	50.6	104.6
Long. Weld	2-203A	0.05	1.00	-40	68	56	4.20	93.0	109.0
Long. Weld	2-203B	0.05	1.00	-40	68	56	4.20	93.0	109.0
Long. Weld	2-203C	0.05	1.00	-40	68	56	4.20	93.0	109.0
Long. Weld	3-203A	0.04	0.16	-70	39	56	4.20	53.3	39.3
Long. Weld	3-203B	0.04	0.16	-70	39	56	4.20	53.3	39.3
Long. Weld	3-203C	0.04	0.16	-70	39	56	4.20	53.3	39.3
Circum. Weld	8-203	0.35	1.00	-56	272	66	0.114	120.8	130.8
*Circum. Weld	9-203	0.06	0.04	-60	34	56	4.20	46.5	42.5
*Circum. Weld	9-203	0.05	0.04	-50	31	56	4.20	42.4	48.4

* For Unit 3 two sets of values were calculated for Weld 9-203 corresponding to the two sets of weld wire heat/flux lot number combinations used.



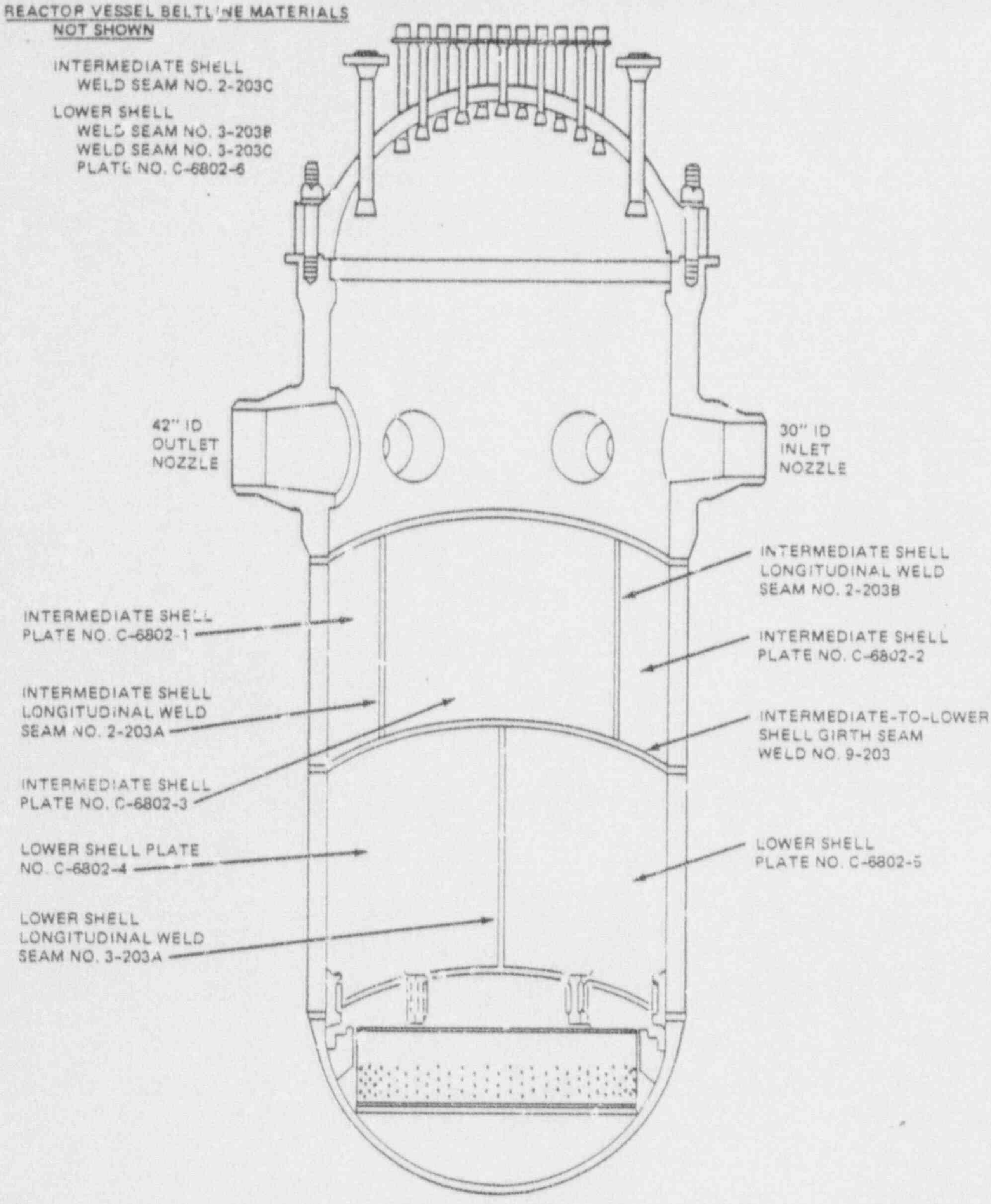
REACTOR VESSEL

Updated

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

UNIT 2
BELTLINE MATERIALS

Figure 5.2-8



REACTOR VESSEL

Updated

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UNIT 3 BELTLINE MATERIALS
Figure 5.2-9