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PALISADES NUCLEAR PLANT
ENGINEERING ANALYSIS COVER SHEET

EA-RDS-94-02

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Title Evaluation of Palisades Current PTS Screening Criteria Margin

INITIATION AND REVIEW

Calculation Status: Preliminary ○ Pending ○ Final ✗ Superseded ○

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0	Original Issue	Ross Snugg[?]	11-03-94	DCP		X		J.L. Biffer	11/03/94	DCP	
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Revision 1 discussion,

This revision incorporates administrative comments made as a result of the PRC meeting. None of the calculations or results change in this revision.

RDS 11-7-94

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Table of Contents

Tables

Attachments

1.0 Objective

This Engineering Analysis has been written to document calculations done to determine Palisades position with respect to the PTS screening criteria. These calculations incorporate the preliminary weld chemistry values obtained from the retired steam generators and the best available fluence data.

2.0 Summary

Calculations have been done to determine the Palisades reactor vessel material condition as it relates to the PTS screening criteria. Based upon the best available fluence values and axial weld chemistries which include the three preliminary copper and nickel weld values from the steam generators, the plant would exceed the 10 CFR 50.61 screening criteria after 210 EFPD's from 24:00 Hrs, October 31, 1994. This works out to a calendar date of May 29, 1995. If Palisades does not take credit for its inhouse fluence calculations, and instead uses cycle 9 fluence rates for cycle 11, the plant would exceed the 10 CFR 50.61 screening criteria after 115 EFPD's. This gives a calendar date of February 23, 1995, assuming continuous full power operation.

The other part of the data to be collected from the retired steam generator welds is the initial RT_{NDT}. This data is not yet available. If the initial RT_{NDT} results are equal to or less than the generic value for Palisades axial weld of -56°F, Palisades will recover a minimum of 10°F on its margin term. This gain would mean that the plant would exceed the 10 CFR 50.61 screening criteria in approximately 4.59 EFPY's.

Although the NRC rule on PTS is based on best estimate fluence and chemistry values, Palisades has not taken credit for the conservative bias of approximately 6% in its current Westinghouse calculational methodology. Recently Palisades received a Technical Evaluation of its fluence methodology from the NRC, reference 3.8. In this evaluation the NRC suggests that Palisades current fluence calculations are between 7% and 10% high. If Palisades is able to use the best estimate fluence values submitted in its 6-5-92 submittal, the plant could run for another 735 EFPD's.

3.0 Analysis Input

References given in section 3.1 cover the data used in this Engineering Analysis.

3.1 References

3.1 10 CFR 50, current issue.

3.2 6-5-92 NRC Fluence Submittal, Docket 50-255 - Lic. DPR-20, 10CFR50.61 Pressurized Thermal Shock, Revised Projected Values of RT_{PTS} for Reactor Beltline Materials.

3.3 6-10-93 NRC Fluence Submittal, Docket 50-255 - Lic. DPR-20, 10CFR50.61 Pressurized Thermal Shock, Reactor Vessel Neutron Fluence, Additional Information.

3.4 2-23-94 NRC Fluence Submittal, Docket 50-255 - Lic. DPR-20, 10CFR50.61 Pressurized Thermal Shock, Revised Information.

3.5 Preliminary Chemistry Data from AEA for Palisades Retired Steam Generators.

3.6 6-21-94 NRC Fluence Submittal, Docket 50-255 - Lic. DPR-20, Palisades Plant, Reactor Vessel Material Surveillance Capsule Test Report.

3.7 EA-P-PTS-93-03, NI Detector Adjustment Factors for Cycle 11 Operations, Rev. 1

3.8 NRC Fluence Evaluation, Docket 50-255, Palisades Plant, Transmittal of Technical Evaluation Report, 9-2-94.

All attachments relate directly to these references. The relevant pages from the separate references have been copied and included in the attachments so that all necessary information is readily available.

4.0 Assumptions

The calculations in this EA are based on the preliminary steam generator weld chemistry values provide by AEA, reference 3.5. All calculated values have been rounded off to three significant digits to be consistent with past submittals. Projections of EFPD's and EFPY's are based on inhouse fluence calculations for cycle 11 only. This inhouse model has been benchmarked against the Westinghouse fluence methodology and has been validated for use as a scoping tool. Westinghouse will be validating these calculations, however this data will not be available until the end of November. For dates that extend beyond cycle 11 it is important to note that the number of EFPD's or EFPY's may be changed by the fluence rates associated with the later cycles. The weld samples from the retired steam generator are only applicable to, and can only affect, Palisades axial weld chemistries. The 30° weld was and still is the limiting weld. This is the only weld addressed in this analysis. The welds removed from steam generator A contain W5214 weld material.

5.0 Analysis

10 CFR 50.61 provides the foundation of the PTS screening criteria. Calculations for the RT_{PTS} are done using equation 1 from the rule.

$$RT_{PTS} = I + M + \Delta RT_{PTS} \qquad Eq. 1$$

ΔRT_{PTS} = Irradiation adjustment of RT

$I = RT_{NDT}$ (Initial RT)

M = Margin term

Each of the items in Equation 1 will be discussed with respect to Palisades current situation.

5.1 Values of 'I' and 'M'

Palisades does not have an initial RT_{NDT} value for its reactor vessel welds. This forces the plant to use the generic value of -56°F for its axial welds, stated

in 10 CFR 50.61 for Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes, reference 3.1. The initial RT_{NDT} is one of the values that the plant intends to get from the retired steam generator welds, but has not yet received.

The value of M in Equation 1 is 66°F for welds when the generic value of I is used, and 56°F when a measured value of I is used. This is the 10°F margin term that the plant hoped to recover by measuring a value of initial RT_{NDT} from the retired steam generator welds.

5.2 Values for 'ΔRT_{PTS}'

The value of ΔRT_{PTS} is calculated from two factors, CF and f, as shown in Equation 2 from 10 CFR 50.61.

$$\Delta RT_{PTS} = (CF)\,f^{(0.28\,-\,0.10\log f)} \qquad Eq.\ 2$$

CF = *Chemistry Factor*
f = *Best estimate neutron fluence units of* $10^{19}\ n/cm^2$

5.2.1 Palisades 'CF' value.

The value of CF for Palisades comes from the table of generic weld CF's provided in a table in 10 CFR 50.61 for plants without credible surveillance data. This table relies on the copper and nickel content of the weld material to determine the CF. Attachment 4 gives the copper and nickel contents for comparable heat No. W5214 welds other than the steam generator welds which are shown in Attachment 5. Table 5.1 shows the chemistry values for the three 'A' steam generator welds from Attachment 5 and their averages. The samples taken from A steam generator were tandem heat No. W5214 welds, the B steam generator samples were from heat No. 34B009; only the heat No. W5214 values are of interest in this EA, since welds fabricated using weld wire from this heat are limiting. The new data taken for heat No. 34B009 does not change the limiting weld for the Palisades reactor vessel.

Sample	Weldment 'A' Copper	Weldment 'A' Nickel	'A/SG/A' Copper	'A/SG/A' Nickel	'A/SG/B' Copper	'A/SG/B' Nickel
1	0.341	1.093	0.367	1.154	0.353	1.203
2	0.310	1.003	0.291	1.156	0.233	1.149
3	0.266	1.090	0.278	1.059	0.237	1.024
Average	0.306	1.062	0.312	1.123	0.274	1.125

Table 5.1 Averages of Retired Steam Generator Weld Chemistries.

Table 5.2 uses the values from Table 5.1 and Attachment 4 to give all the weld sample values for copper and nickel. It also provides the averages of copper and nickel content for use in determining Palisades reactor vessel axial weld material CF from 10 CFR 50.61. Some of the copper values have been double counted because they were from tandem welds. This is the same averaging technique as used in Reference 3.4.

I.D.	Copper	I.D.	Nickel
D4463 IP2	0.20	D4494 IP2	0.94
" "	0.20	D4541	1.20
HBR2 Torus	0.159	D4577 & D4604	1.00
" "	0.159	D4673 Mill 1C	1.05
IP2 Sur	0.20	D4674 IP2	1.12
IP3 Sur	0.16	D4686 ML1	0.97
" "	0.16	D4687 IP21	0.92
IP3 Nozzle	0.15	D4688 Pal	0.99
HBR2 Sur	0.34	D4690	1.13
OC1 Sur	0.285	HBR2 Torus	0.99
Pal Weldment A	0.306	IP2 Sur	1.03
" "	0.306	IP3 Sur	1.12
PAL A/SG/A	0.312	IP3 Nozzle	1.09
" "	0.312	HBR2 Sur	0.66
PAL A/SG/B	0.274	Pal Weldment A	1.062
" "	0.274	Pal A/SG/A	1.123
Average	0.237	Pal A/SG/B	1.125
		Average	1.03

Table 5.2 Best Estimate Cu and Ni Values for Palisades Axial Welds.

The best estimate Cu value for Palisades axial welds is 0.237 and the Ni value is 1.03. These values can be used with Table 1 of 10 CFR 50.61, shown in Attachment 1, to determine a CF for use in calculating the Palisades PTS screening criteria fluence value. Using linear interpolation, as allowed by the rule, the CF = 242.36°F, which rounds to 242°F.

5.2.2 Palisades 'f' Values

To date Palisades has only officially submitted fluence values for cycles 1 through 10, Reference 3.3 and 3.6; these values are restated in a more convenient format in Attachment 6. In order to calculate Palisades current accumulated fluence it is necessary to use cycle 10 fluence values from Reference 3.6, and cycle 11 fluence values from Reference 3.7.

Westinghouse analysis shows that the calculational methodology used to create the data shown in the references above has a conservative bias of approximately 6%. Although the NRC rule on PTS is based on best estimate fluence and chemistry values, Palisades has not taken credit for the conservative bias in its current Westinghouse calculational methodology. Recently Palisades received a technical evaluation of its fluence methodology from the NRC, reference 3.8. In this evaluation the NRC evaluated the current Palisades fluence calculations as between 7% and 10% high. If necessary Palisades may choose in the future to recover this conservatism from its analysis. The best estimate fluence rates from Westinghouse for cycles 1 through 9 are shown in Attachment 3. For cycles 10 and 11 the best estimate fluence rates have been created by dividing the calculated fluence rates by 1.06.

Table 5.3 shows both the calculated and best estimate fluence rates, along with the cycle and cumulative fluence for both. The EFPD's for cycles 1 through 10 shown in Table 5.3 can be found in Attachment 6. For cycle 11 the EFPD's have been calculated as follows. The current burn-up, 7222.1 MWD/MTU, times the MTU of the core, 81.202 MTU, divided by the rated power, 2530 MW, gives 231.8 EFPD's. Since the limiting welds are the welds at the 30° positions, only fluence rates at these angles are used.

Cycle Number	Cycle EFPD's	Fluence Rate Calc's	Fluence Rate Best Est.	Cycle Fluerce Calc's	Cycle Fluerce Best Est.	Cumulative Fluence Calc's	Cumulative Fluence Best Est.
1	379.4	4.70E10	4.43E10	1.54E18	1.45E18	1.54E18	1.45E18
2	449.1	4.70E10	4.43E10	1.82E18	1.72E18	3.36E18	3.17E18
3	349.5	4.70E10	4.43E10	1.42E18	1.34E18	4.78E18	4.51E18
4	327.6	4.70E10	4.43E10	1.33E18	1.25E18	6.11E18	5.76E18
5	394.6	4.70E10	4.43E10	1.60E18	1.51E18	7.71E18	7.27E18
6	333.4	4.79E10	4.52E10	1.38E18	1.30E18	9.09E18	8.58E18
7	369.9	4.79E10	4.52E10	1.53E18	1.44E18	1.06E19	1.00E19
8	373.6	2.34E10	2.21E10	7.55E17	7.13E17	1.14E19	1.07E19
9	298.5	2.00E10	1.89E10	5.16E17	4.87E17	1.19E19	1.12E19
10	356.9	1.94E10	1.83E10	5.98E17	5.64E17	1.25E19	1.18E19
11	231.8	1.66E10	1.57E10	3.33E17	3.14E17	1.28E19	1.21E19

Table 5.3 Palisades Fluence Values.

Using Palisades most up to date fluence calculations, f = 1.28. A best estimate value of, f = 1.21, could be used if the plant can recover the conservative bias in its calculational methodology.

5.3 Palisades PTS Screening Criteria Limits

Equations 1 and 2 from 10 CFR 50.61 can be solved for f, as shown in Attachment 2, giving Equation 3 shown below.

$$f = 10^{\frac{0.28 - \sqrt{0.0784 - 0.4 \log \frac{(RT_{PTS} - I - M)}{CF}}}{0.2}} \qquad \text{Eq. (3)}$$

The maximum RT_{PTS} allowed for Palisades axial welds is 270°F, reference 3.1. Using this 270°F value for RT_{PTS}, -56°F for I, 66°F for M, and 242°F for CF, in Equation 3, gives a screening criteria fluence value of $1.31*10^{19}$ n/cm^2. This value and Palisades current fluence accumulation can be used to determine the number of EFPD's remaining before the plant reaches the PTS screening criteria. This is shown on the following page.

$$Margin = 1.31*10^{19}\ n/cm^2 - 1.28*10^{19}\ n/cm^2 = 3.0*10^{17}$$

$$Fluence/EFPD = 1.66*10^{10}\ n/(cm^2\text{-}sec) * 3600\ sec/Hr*24\ Hr/Day$$

$$Fluence/EFPD = 1.43*10^{15}\ n/cm^2$$

$$EFPD's = \frac{3.0*10^{17}}{1.43*10^{15}} = 210\ EFPD's$$

Table 5.4 shows the number of EFPD's/EFPY's left before the plant reaches the screening criteria using different values of (RT_{PTS} - I - M). This table includes values of EFPD's/EFPY's for both calculated and best estimate fluence data. The best estimate fluence rate for cycle 11 is $1.36*10^{15}$.

Value of RT_{PTS} - I - M	Screening Criteria Fluence Limit	Margin Using Calculated Fluence	Margin Using Best Est. Fluence
260	1.31E19	210 EFPD's	735 EFPD's
262	1.35E19	490 EFPD's	2.82 EFPY's
264	1.39E19	769 EFPD's	3.62 EFPY's
266	1.43E19	2.87 EFPY's	4.43 EFPY's
268	1.47E19	3.64 EFPY's	5.23 EFPY's
270	1.52E19	4.59 EFPY's	6.24 EFPY's
272	1.57E19	5.55 EFPY's	7.25 EFPY's
274	1.61E19	6.32 EFPY's	8.05 EFPY's
276	1.67E19	7.47 EFPY's	9.26 EFPY's
278	1.72E19	8.42 EFPY's	10.3 EFPY's
280	1.77E19	9.38 EFPY's	11.3 EFPY's

Table 5.4 Possible Margin Gains.

It is important to note two things about Table 5.4. First, the times stated do not take into account any capacity factor deviation from 100%; outages would add to the number of days or years calculated. Second, the data assumes that all subsequent fluence will be accumulated at cycle 11 fluence rates.

If cycle 9 fluence rates are used to estimate the cycle 11 fluence, rather than using the inhouse calculations, it can be shown that Palisades has 115 EFPD's left in cycle 11 before reaching the screening criteria.

$$\textit{PTS screening criteria fluence} = 1.31*10^{19}$$

$$\textit{End of cycle } 10 \textit{ fluence} = 1.25*10^{19}$$

$$\textit{Cycle } 9 \textit{ fluence rate} = 2.00*10^{10}*3600*24 = 1.73*10^{15}$$

$$EFPD's = \frac{1.31*10^{19} - 1.25*10^{19}}{1.73*10^{15}} = 347\ EFPD's$$

$$\textit{Margin} = 347\ EFPD's - 232\ EFPD's\ (\textit{thru } 10\text{-}31\text{-}94) = 115\ EFPD's$$

6.0 Conclusion

The objective of this EA has been met. Palisades PTS screening criteria margin has been calculated using the preliminary and partial chemistry data received from testing done on the retired steam generator welds. The data provided shows that the Palisades reactor vessel weld material would not have reached its PTS screening criteria fluence value. A longer summary is available in section 2.0.

Attachment 1

(c) The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to § 50.90.

39 FR 10558

§ 50.60 Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12

48 FR 24009

§ 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

(a) *Definitions.* For the purposes of this section:

(1) "ASME Code" means the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Rules for the Construction of Nuclear Power Plant Components," edition and addenda as specified by § 50.55a, Codes and Standards.

(2) "Pressurized Thermal Shock Event" means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

(3) "Reactor Vessel Beltline" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

(4) "Initial RT_{NDT}" means the reference temperature for a reactor vessel material as defined in the ASME Code, Paragraph NB-2331. RT_{NDT} means the reference temperature as adjusted for the effects of neutron radiation for the period of service in question.

(5) "RT_{PTS}" means the reference temperature calculated by the method given in paragraph (b)(2) of this section for use as a screening criterion.

50 FR 29937

(b) *Requirements.*

(1) For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RT_{PTS} for reactor vessel beltline materials by giving values for the time of submittal, the expiration date of the operating license, the projected expiration date if a change in the operating license has been requested, and the projected expiration date of a renewal term if a request for license renewal has been submitted. The assessment must use the calculative procedures given in paragraph (b)(2) of this section. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. The submittal must list the copper and nickel contents, and the fluence values used in the calculation for each beltline material. If these quantities differ from those submitted in response to the original PTS rule and accepted by the NRC, justification must be provided. If the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license or the proposed expiration date if a change in the license has been requested, or the end of a renewal term if a request for license renewal has been submitted, this assessment must be submitted by December 16, 1991. Otherwise, this assessment must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel material surveillance report, or 5 years from the effective date of this rule whichever comes first. These submittals must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon a request for a change in the expiration date for operation of the facility.

56 FR 22300

(2) The pressurized thermal shock (PTS) screening criterion is 270°F for plates, forgings, and axial weld materials, or 300°F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be calculated as follows, except as provided in paragraph (b)(3) of this section. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline.

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

(i) "I" means the initial reference temperature (RT_{NDT}) of the unirradiated material measured as defined in the ASME Code, Paragraph NB-2331. Measured values must be used if credible values are available; if not, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and −56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

(ii) "M" means the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and the calculational procedures. In Equation 1, M is 66°F for welds and 48°F for base metal if generic values of I are used, and M is 56°F for welds and 34°F for base metal if measured values of I are used.

(iii) ΔRT_{PTS} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

Equation 2: $\Delta RT_{PTS} = (CF) f^{(0.28 - 0.10 \log f)}$

(iv) CF (°F) is the chemistry factor, a function of copper and nickel content. CF is given in table 1 for welds and in table 2 for base metal (plates and

56 FR 22300

forgings). Linear interpolation is permitted. In Tables 1 and 2 "Wt-% copper" and "Wt-% nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. If these values are not available, the upper limiting values given in the material specifications to which the vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data [1] may be used if justification is provided. If none of these alternatives are available, 0.35% copper and 1.0% nickel must be assumed.

(v) "f" means the best estimate neutron fluence, in units of 10^{19} n/cm² (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.

TABLE 1.—CHEMISTRY FACTOR FOR WELDS, °F

56 FR 22300

Copper, Wt-%	Nickel, Wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	126	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

[1] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time period is an example of "generic data."

TABLE 2.—CHEMISTRY FACTOR FOR METAL, °F

56 FR 22300

Copper, Wt-%	Nickel, Wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

(3) To verify that the values of RT_{PTS} calculated as required by paragraph (b)(2) of this section are bounding values for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and surveillance results. Results from the plant-specific surveillance program shall be integrated into the embrittlement estimate if,

(i) The plant-specific surveillance data has been deemed credible as defined in Regulatory Guide 1.99 Revision 2, and

(ii) The RT_{PTS} value changes significantly.[2]

Any information that is believed to improve the accuracy of the RT_{PTS} value significantly shall be reported to the Director, Office of Nuclear Reactor Regulation. Values of RT_{PTS} that have been modified using the procedures of this paragraph are subject to the approval of the Director, Office of Nuclear Reactor Regulation when used as provided in this section.

(4) For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, or the projected expiration date if a change in the license has been requested, or the end of a renewal term if a request for license renewal has been submitted, the licensee shall submit by March 16, 1992, an analysis and schedule for implementation of such flux reduction programs as are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2) of this section. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated Commission approval of detailed plant-specific analyses, submitted to demonstrate acceptable risk at values of RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

(5) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(4) of this section indicates that no reasonably practicable flux reduction program will prevent the value of RT_{PTS} from exceeding the PTS screening criterion before the expiration date of the operating license, or the projected expiration date if a change in the operating license has been requested, or the end of a renewal term if a request for license renewal has been submitted, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine reactor vessel materials properties based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least 3 years before the value of RT_{PTS} is projected to exceed the PTS screening criterion or by one year after the effective date of this amendment, whichever is later.

(6) After consideration of the licensee's analyses (including effects of proposed corrective actions, if any) submitted in accordance with paragraphs (b)(4) and (b)(5) of this section, the Commission may, on a case-by-case basis, approve operation of the facility at values of RT_{PTS} in excess of the PTS screening criterion. The

[2] Changes to RT_{PTS} values are considered significant if either the value determined in paragraph (b)(2) of this section or the alternate value determined in paragraph (b)(3) of this section, or both values, exceed the screening criterion, prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

Commission will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

56 FR 22300

(7) If the Commission concludes, pursuant to paragraph (b)(6) of this section, that operation of the facility at values of RT_{PTS} in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(4) and (b)(5) of this section, the licensee shall request and receive Commission approval prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

49 FR 26036

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all commercial light-water-cooled nuclear power plants.

(b) *Definition.* For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.

(c) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

54 FR 13361

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

49 FR 26036

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

51 FR 40303

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4.

(d) *Implementation.* By 180 days after the issuance of the QA guidance for non-safety related components, each licensee shall develop and submit to the Commission, as specified in § 50.4, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

53 FR 23203

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.

(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

(b) *Limitation of scope.* Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988,* if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

53 FR 23203

(c) *Implementation.*—(1) *Information Submittal.* For each light-water-cooled nuclear power plant licensed to operate on or before *July 21, 1988,* the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by *April 17, 1989.* For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;

(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.

(2) *Alternate ac source:* The alternate ac power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the

Attachment 2

4.0 PROJECTED RT_{PTS}

The following describes how the PTS reference temperatures are determined for each of the Palisades reactor vessel beltline materials and includes projections for when each material will exceed the applicable screening criterion. The results are dependent on the best-estimate values for chemistry and fluence that have been addressed earlier in this report. Additionally this section provides response to NRC concerns as to how surveillance results from Palisades and other reactor vessels could affect the projected RT_{PTS} values.

4.1 Determination and Projection of the PTS Reference Temperatures

The base equation for the PTS reference temperature from 10CFR50.61 is:

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

"I" is defined as the initial reference temperature (RT_{NDT}) of the unirradiated material. "I" values for the Palisades reactor vessel beltline materials are:

Axial Weld	I_a	= -56°F	Generic Value 10CFR50.61 (b)(2)(i) for Welds made with Linde 1092 and 124 Fluxes
Circ Weld	I_c	= -56°F	
Plate	I_p	= 0°F	Value* reported in Reference 6. This represents the limiting plate.

* A less conservative value of I_p = -10°F was measured by Battelle Columbus Laboratories in 1977 (Reference 39). A value of -5°F was used in CPCo's 1986 (Reference 16) and 1991 (Reference 1) PTS submittals. Confirmation of -5°F could not be found by measurement or calculation.

"M" is defined as the margin term added to cover uncertainties as in the values of initial RT_{PTS} (Cu and Ni content, fluence and the calculational procedures). Values of "M" for the Palisades vessel beltline material are:

Axial Weld	$M_a = 66°F$	Value specified in 10CFR50.61(b)(2)(ii) for welds if generic values of "I" are used.
Circ Weld	$M_c = 66°F$	
Plate	$M_p = 34°F$	Value specified for base metal in 10CFR50.61 if measured value of "I" is used

"ΔRT_{PTS}" is defined as:

$$\Delta RT_{PTS} = (CF)\, f^{(0.28 - 0.10 \log f)} \tag{2}$$

"CF", the chemistry factor, a function of Cu and Ni content, is derived from Tables 1 and 2 of 10CFR50.61.

In Section 2, the chemistry factors were determined to be:

$CF_a = 217°F$ for the axial welds.

$CF_c = 228°F$ for the circumferential weld.

$CF_p = 165°F$ for the vessel plate material.

"f" is the best-estimate neutron fluence in units of 10^{19} n/cm^2 (E > 1 MeV) at the clad-base metal interface of the vessel.

The limiting fluence is determined by setting RT_{PTS} equal to the screening criteria and solving for f. First, rearranging equations (1) and (2):

$$RT_{PTS} = I + M + (CF)\, f^{(0.28 - 0.10 \log f)}$$

$$(0.28 - 0.10 \log f) \log f = \log \left(\frac{RT_{PTS} - I - M}{CF} \right)$$

$$0.10 (\log f)^2 - 0.28 \log f + \log \left(\frac{RT_{PTS} - I - M}{CF} \right) = 0$$

Using the quadratic equation to solve for log f:

$$\log f = \frac{0.28 \pm \sqrt{(0.28)^2 - 4\,(0.10) \log \left(\frac{RT_{pts} - I - M}{CF} \right)}}{2\,(0.10)}$$

Because the positive root of the equation provides meaningless results, the equation may be simplified to:

$$f = 10 \exp \left[\frac{0.28 - \sqrt{0.0784 - 0.4 \log \left(\frac{RT_{pts} - I - M}{CF} \right)}}{0.2} \right]$$

The maximum allowed values of RT_{PTS} is defined in 10CFR50.61(b)(2) for each of the Palisades beltline is:

Axial Weld	$RT_{PTSa} = 270°F$
Circumferential Weld	$RT_{PTSc} = 300°F$
Plate Material	$RT_{PTSp} = 270°F$

Attachment 3

Table 8-4 (Continued)

Palisades Fast Neutron Fluence (E > 1.0 MeV) Through Cycle 9
At the Reactor Vessel Clad-Base Metal Interface

Cycle	Cycle Length (EFPD)	Cycle Flux (n/cm^2-s)	Cycle Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
		30 Degrees		
1	379.4	4.43E+10	1.45E+18	1.45E+18
2	449.1	4.43E+10	1.72E+18	3.17E+18
3	349.5	4.43E+10	1.34E+18	4.51E+18
4	327.6	4.43E+10	1.26E+18	5.77E+18
5	394.6	4.43E+10	1.51E+18	7.28E+18
6	333.4	4.52E+10	1.30E+18	8.58E+18
7	369.9	4.52E+10	1.44E+18	1.00E+19
8	373.6	2.21E+10	7.13E+17	1.07E+19
9	298.5	1.89E+10	4.87E+17	1.12E+19
		45 Degrees		
1	379.4	2.81E+10	9.22E+17	9.22E+17
2	449.1	2.81E+10	1.09E+18	2.01E+18
3	349.5	2.81E+10	8.49E+17	2.86E+18
4	327.6	2.81E+10	7.96E+17	3.66E+18
5	394.6	2.81E+10	9.58E+17	4.62E+18
6	333.4	2.86E+10	8.23E+17	5.44E+18
7	369.9	2.86E+10	9.14E+17	6.35E+18
8	373.6	1.67E+10	5.39E+17	6.89E+18
9	298.5	1.09E+10	2.80E+17	7.17E+18

Attachment 4

CALCULATION OF THE MEAN COPPER AND NICKEL CONTENT
OF WELDS FABRICATED USING WELD WIRE FROM HEAT No. W5214

The following identifications and copper content values are from Table I.1.

1. COPPER CONTENT

Sample Identification	Weight % Copper
D4463 - IP2-flange 1-042B	0.20
" " " "	0.20
HBR2 - Torus Flange	0.159
" " "	0.159
IP2 - Surveillance	0.20
IP3 - Surveillance	0.16
" "	0.16
IP3 - Nozzle Cutout	0.15
HBR2 - Surveillance	0.34
OC1 - Surveillance	0.285
Total	2.013

2.013 ÷ 10 = 0.201 = Mean Copper Content

2. NICKEL CONTENT

Sample Identification	Weight % Content
D4494 - IP2 1-042	0.94
D4541	1.20
Average of D4577 & D4604	1.00
D4673 Millstone 1C	1.05
D4674 IP2 3-042B	1.12
D4686 ML1 2-072A	0.97
D4687 IP21-042A	0.92
D4688 PAL S/G 5-943	0.99
D4690	1.13
HBR2 Torus Flange	0.99
IP2 - Surveillance	1.03
IP3 - Surveillance	1.12
IP3 - Nozzle Cutout	1.09
HBR2 - Surveillance	0.66
Total	14.21

14.21 ÷ 14 = 1.015 = Mean Nickel Content

Attachment 5

TESTING OF WELDMETALS FOR CPCO

PRELIMINARY ANALYSIS RESULTS ON SECTIONS TAKEN FROM LARGE WELDMENTS ('A' AND 'B') AND TREPANS

Results of the chemical anlysis of weldmetal samples taken from sections through the two large weldments and four Trepans are given in the table below. Although these data are considered to be true and accurate, final checks have still to be performed and as such the data should be treated as being of a preliminary nature until all the checks have been made.

	Mn	Cr	Cu	Mo	Ni	P	Si	S	V
Section through Large Weldment 'A'									
A1/1/X	1.157	0.0371	0.341	0.502	1.093	0.010	0.264	0.0146	0.0023
A1/1/Y	1.249	0.0339	0.310	0.507	1.003	0.010	0.265	0.0174	0.0022
A1/1/Z	1.176	0.0317	0.266	0.487	1.090	0.011	0.288	0.0181	0.0022
Section through Large Weldment 'B'									
B1/2/X	1.249	0.0400	0.235	0.546	1.215	0.012	0.173	0.0159	0.0028
B1/2/Y	1.304	0.0383	0.159	0.537	1.010	0.012	0.166	0.0182	0.0028
B1/2/Z	1.265	0.0389	0.196	0.540	1.098	0.012	0.182	0.0177	0.0028
Section through Trepan 'A/SG/A'									
A/SG/A/2/X	1.137	0.0401	0.367	0.508	1.154	0.011	0.247	0.0115	0.0021
A/SG/A/2/Y	1.120	0.0328	0.291	0.498	1.156	0.013	0.284	0.0178	0.0020
A/SG/A/2/Z	1.114	0.0333	0.278	0.498	1.059	0.012	0.284	0.0182	0.0021
Section through Trepant 'A/SG/B'									
A/SG/B/3/X	1.173	0.0407	0.353	0.515	1.203	0.011	0.243	0.0110	0.0021
A/SG/B/3/Y	1.102	0.0389	0.233	0.523	1.149	0.012	0.291	0.0158	0.0024
A/SG/B/3/Z	1.105	0.0411	0.237	0.519	1.024	0.011	0.302	0.0141	0.0025
Section through Trepan 'B/SG/A'									
B/SG/A/2/X	1.292	0.0412	0.195	0.551	1.272	0.017	0.186	0.0181	0.0031
B/SG/A/2/Y	1.234	0.0379	0.195	0.550	1.331	0.016	0.202	0.0170	0.0027
B/SG/A/2/Z	1.246	0.0388	0.206	0.544	1.338	0.016	0.211	0.0165	0.0027
Section through Trepan 'B/SG/B'									
B/SG/B/2/X	1.273	0.0397	0.162	0.541	1.126	0.016	0.191	0.0177	0.0029
B/SG/B/2/Y	1.237	0.0402	0.208	0.536	1.136	0.016	0.204	0.0165	0.0027
B/SG/B/2/Z	1.188	0.0391	0.209	0.532	1.307	0.016	0.212	0.0156	0.0025

G. Gage
31st October 1994

Attachment 6

TABLE 6-13 (Continued)

CALCULATED FLUENCE (E>1.0 MeV) THROUGH CYCLE 10
AT THE PRESSURE VESSEL CLAD-BASE METAL INTERFACE

Cycle	Cycle Length (EFPD)	Cycle Flux (n/cm2-sec)	Cycle Fluence (n/cm2)	Cumulative Fluence (n/cm2)
		30 Degree		
1	379.4	4.70E+10	1.54E+18	1.54E+18
2	449.1	4.70E+10	1.82E+18	3.36E+18
3	349.5	4.70E+10	1.42E+18	4.78E+18
4	327.6	4.70E+10	1.33E+18	6.11E+18
5	394.6	4.70E+10	1.60E+18	7.71E+18
6	333.4	4.79E+10	1.38E+18	9.09E+18
7	369.9	4.79E+10	1.53E+18	1.06E+19
8	373.6	2.34E+10	7.55E+17	1.14E+19
9	298.5	2.00E+10	5.16E+17	1.19E+19
10	356.9	1.94E+10	5.98E+17	1.25E+19
		45 Degree		
1	379.4	2.98E+10	9.78E+17	9.78E+17
2	449.1	2.98E+10	1.16E+18	2.13E+18
3	349.5	2.98E+10	9.00E+17	3.04E+18
4	327.6	2.98E+10	8.44E+17	3.88E+18
5	394.6	2.98E+10	1.02E+18	4.90E+18
6	333.4	3.03E+10	8.73E+17	5.77E+18
7	369.9	3.03E+10	9.68E+17	6.74E+18
8	373.6	1.77E+10	5.71E+17	7.31E+18
9	298.5	1.15E+10	2.97E+17	7.61E+18
10	356.9	1.32E+10	4.07E+17	8.02E+18

Attachment 7

Palisades Cycle Flux Values at Critical Locations

Cycle	EFPD	Cycle Flux 0°	E + 10 16°	30°	45°
1	379.4	4.59	[illegible].03	4.70	2.98
2	449.1	4.59	6.03	4.70	2.98
3	349.5	4.59	6.03	4.70	2.98
4	327.6	4.59	6.03	4.70	2.98
5	394.6	4.59	6.03	4.70	2.98
6	333.4	4.87	6.25	4.79	3.03
7	369.9	4.87	6.25	4.79	3.03
8	373.6	2.16	4.89	2.34	1.77
9	298.5	2.08	3.06	2.00	1.15
10	356.9	1.51	2.40	1.94	1.32
11	422.0	1.42	2.21	1.66	1.09

Values for cycles 1 through 10 are from WCAP14014.
Values for cycle 11 are from Palisades in-house calculations.

ENCLOSURE 3
TO
10CFR50.61 PRESSURIZED THERMAL SHOCK - REVISED INFORMATION

Consumers Power Company
Palisades Plant
Docket 50-255

A NON-PROPRIETARY VERSION
OF THE
CPC NOVEMBER 18, 1994 SUBMITTAL

January 23, 1995

A NON-PROPRIETARY VERSION
OF THE
CPC NOVEMBER 18, 1994 SUBMITTAL

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
10CFR50.61 - PRESSURIZED THERMAL SHOCK - ADDITIONAL INFORMATION

Consumers Power Company (CPC) submittals dated February 23, 1994, November 8, 1994, and November 10, 1994 described our plan to more accurately determine the chemical and physical properties of the weld materials in the Palisades reactor vessel and the progress we have made. We have implemented a plan, the Palisades Reactor Vessel Integrity Project Plan (PRVIPP), to a point where we have performed chemistry and physical testing on weld material from our retired steam generators.

Our November 8, 1994 letter provided preliminary chemistry results from the testing of steam generator weld material. It also provided Revision 1 of Palisades engineering analysis EA-RDS-94-02 which postulated when the Palisades reactor vessel material would exceed the screening criterion if the preliminary chemistry results were representative of three welds fabricated with weld wire from Heat No. W5214. Our November 10, 1994 letter informed the staff that we: (1) had received preliminary low temperature toughness data from the physical testing material and were suspicious of its credibility for use in determining the initial RT_{NDT} of the weld material in the Palisades reactor vessel, and (2) were aware of preliminary information, in regard to the steam generator weld fabrication methodology, that indicated the data from the three welds from each steam generator should be treated as being representative of one weld. Our November 10, 1994 letter also stated that, on or before November 18, 1994, we would make a submittal containing: (1) our analysis of the steam generator weld test data and its effect on the operability of the Palisades reactor vessel, and (2) a description of the actions we plan to take in the near future as we continue to implement the PRVIPP.

This is that submittal.This letter transmits Revision 2 of Palisades engineering analysis EA-RDS-94-02 (Enclosure 1). Revision 2 incorporates steam generator weld material chemistry data into the industry database to estimate that the limiting Palisades reactor vessel material (welds fabricated using wire from Heat No. W5214) will not exceed the screening criterion until January 1999. In reaching this conclusion, the analysis continues to use the generic value of initial RT_{NDT} prescribed in 10CFR50.61 for the flux type used in the Palisades reactor vessel. While fracture toughness data obtained using ASTM test standard E 208 showed a higher than originally anticipated NDTT for

the steam generator material, subsequent testing and evaluation has lead to the conclusion that the steam generator material test results cannot be considered credible for use in establishing an initial RT_{NDT} for as fabricated Palisades reactor vessel material. This is because of differences such as material thickness, number of weld passes, post-weld heat treatment, and the effects of thermal aging from having been exposed to a medium high temperature environment for a long period of time. These effects are addressed further in Attachment 9 to Enclosure 1.

The analysis presented in Enclosure 1 shows that the Plant may be operated for an additional 3.16 effective full power years (EFPY) or approximately four calendar years before the limiting Palisades reactor vessel material (Heat No. W5214) exceeds the screening criterion (January 1999). It will, therefore, be necessary for the Company to submit, within approximately one calendar year, our plan to allow for operation through the end of licensed life. Our short-term actions, to be completed in 1995, will support the development of that plan. These short-term actions will include:

1. Independently analyze steam generator weld samples.

2. Evaluate performing microstructure analyses of the broken steam generator impact test samples.

3. Evaluate performing additional fracture toughness analyses using alternate methodology.

4. Before March 1, 1995, submit a request to use a site-specific surveillance plan using the available representative industry data on Heat No. W5214 welds. Preliminary analysis using this data, which is subject to staff approval, projects the time before the Palisades reactor vessel material will exceed the criterion to approximately seven EFPY. Additionally, we will evaluate heat treating the steam generator weld material samples and incorporating them in this plan.

Long term actions being considered are:

1. Using a plant-specific surveillance program.

2. Performing reactor vessel weld sampling.

3. Establishing methods to better define fluence.

4. Utilizing a lower leakage core.

5. Installing reactor vessel prestressed bands.

6. Performing a Regulatory Guide 1.154 analysis.

7. Performing a reactor vessel anneal.

CONCLUSION

The latest calculations, using conservative fluence values, show that the Palisades reactor vessel can operate for 3.16 EFPY before exceeding the 10CFR50.61 screening criterion. This will allow operation at a 75 percent capacity factor until January 1999. When a site-specific integrated surveillance plan is approved, these values are expected to be increased by approximately four EFPY. Planned short-term actions, short term-actions being evaluated, and long term actions being considered may further increase the time before exceeding the criterion.

The version of Engineering Analysis EA-RDS-94-02 which is included as Enclosure 1 to this letter contains the same information as that submitted November 18, 1994 and now requested to be withdrawn except that the information on Sheet 8 and in Attachment 4 is no longer considered proprietary.

SUMMARY OF COMMITMENTS

1. Before March 1, 1995, we will submit a site-specific integrated surveillance plan for staff approval.

2. Before March 1, 1995 we will submit a plan to further evaluate the weld material from the retired steam generators.

Enclosure

ENCLOSURE 1
TO
THE NONPROPRIETARY VERSION
OF
THE CPC NOVEMBER 18, 1994 SUBMITTAL

Consumers Power Company
Palisades Plant
Docket 50-255

ENGINEERING ANALYSIS
EA-RDS-94-02, REVISION 2

Consumers Power
POWERING MICHIGAN'S PROGRESS

PALISADES NUCLEAR PLANT
ENGINEERING ANALYSIS COVER SHEET

EA-RDS-94-02

Total Number of Sheets 10

Title Evaluation of Palisades Current PTS Screening Criteria Margin

INITIATION AND REVIEW

Calculation Status: Preliminary ○ Pending ○ Final ○ Superseded ○

Rev #	Description	Initiated By	Initiated Date	Init Appd By	Review Method: Alt Calc	Review Method: Detail Review	Review Method: Qual Test	Technically Reviewed By	Technically Reviewed Date	Revr Appd By	CPCo Appd
0	Original Issue	Ross Snuggerud	11-03-94	GCP		✓		Jim L Biffer	11-03-94	GCP	
0	Original Issue 2nd Review				✓	✓		George H Goralski	11-03-94	RJS	
1	Admin Revision	Ross Snuggerud	11-07-94	GCP		✓		Jim L Biffer	11-07-94	GCP	
2	Updated Revision	Ross Snuggerud	11-17-94	[illegible]		✓		GHGoralski [illegible]	11-17-94	[illegible]	

Revision 1 Discussion

This revision incorporates administrative comments made as a result of the PRC meeting. None of the calculations or results change in this revision.

Revision 2 Discussion

This revision incorporates the final chemistry data and discussion of the results of the initial RT_{NDT} test data. This has resulted in several changes from the two previous versions.

Table of Contents

Tables

Attachments

1.0 Objective

This Engineering Analysis is written to document calculations which determine the compliance status of the Palisades reactor vessel weld material in respect to the PTS screening criteria. They incorporate the final weld chemistry values obtained from the retired steam generators and the best available fluence data.

2.0 Summary

Calculations have been performed to determine the Palisades reactor vessel material condition as it relates to the PTS screening criteria. Based upon the best available fluence values and axial weld chemistries which include the averages of the eighteen copper and nickel weld samples from the steam generators, the plant would exceed the 10 CFR 50.61 screening criteria after 3.16 EFPY's from 24:00 Hrs, October 31, 1994. Assuming a 75% capacity factor this works out to a calendar date of mid January 1999.

The other part of the data to be collected from the retired steam generator welds was the initial RT_{NDT}. The data collected from these measurements suggests that the material has been affected by its use in the steam generators and cannot be used to provide the initial RT_{NDT} for this weld material.

3.0 Analysis Input

References given in section 3.1 cover the data used in this Engineering Analysis.

3.1 References

3.1 10 CFR 50, current issue.

3.2 6-5-92 CPCo Submittal, Docket 50-255 - Lic. DPR-20, 10CFR50.61 Pressurized Thermal Shock, Revised Projected Values of RT_{PTS} for Reactor Beltline Materials.

3.3 6-10-93 CPCo Submittal, Docket 50-255 - Lic. DPR-20, 10CFR50.61 Pressurized Thermal Shock, Reactor Vessel Neutron Fluence, Additional Information.

3.4 2-23-94 CPCo Submittal, Docket 50-255 - Lic. DPR-20, 10CFR50.61 Pressurized Thermal Shock, Revised Information.

3.5 Testing of Weldmetals for CPCo Additional Chemical Analysis, Letter from Dr. G. Gage, AEA, to John Kneeland, CPCo, November 14, 1994.

3.6 6-21-94 CPCo Submittal, Docket 50-255 - Lic. DPR-20, Palisades Plant, Reactor Vessel Material Surveillance Capsule Test Report.

3.7 EA-P-PTS-93-03, NI Detector Adjustment Factors for Cycle 11 Operations, Rev. 1

3.8 Palisades SG Upper Shell Long Seam Fabrication Technique, Letter from Carl J. Gimbrone, ABB, to John Kneeland, CPCo, November 15, 1994.

3.9 Server, W.L., Credibility of Using Steam Generator Welds as Surrogates for the Palisades Reactor Pressure Vessel Welds.

3.10 NRC Fluence Evaluation, Docket 50-255, Palisades Plant, Transmittal of Technical Evaluation Report, 9-2-94.

All attachments relate directly to these references. The relevant pages from the separate references have been copied and included in the attachments so that all necessary information is readily available.

4.0 Assumptions

4.1 The weld samples from the retired steam generator are only applicable to, and can only affect, Palisades axial weld chemistries, because only heat No. W5214 and 34B009 weld materials were removed from the steam generators.

4.2 The 30° weld was and still is the limiting weld. This is the only weld addressed in this analysis.

4.3 The welds removed from steam generator A contain the heat No. W5214 weld material. The welds removed from steam generator B contain the heat No. 34B009

weld material. The chemistry factor for heat No. 34B009 weld material is still lower than the chemistry factor for heat No. W5214 weld material.

4.4 The calculations in this EA are based on integrating the averages of the eighteen steam generator weld chemistry values provide by AEA, reference 3.5, with the previously available industry data. The samples taken from the steam generator constitute one weld and should be averaged into the industry data as one weld, as supported in reference 3.8.

4.5 The retired steam generator weld material is not capable of providing credible RT_{NDT} values for the Palisades axial welds, as supported in reference 3.9.

4.6 All calculated values have been rounded off to three significant digits to be consistent with past submittals. This is consistent with the accuracy of measured values and those values reported in the regulatory guidance.

5.0 Analysis

10 CFR 50.61 provides the foundation of the PTS screening criteria. Calculations for the RT_{PTS} are done using equation 1 from the rule.

$$RT_{PTS} = I + M + \Delta RT_{PTS} \qquad Eq.\ 1$$

$$\Delta RT_{PTS} = \text{Irradiation adjustment of } RT$$
$$I = RT_{NDT} \text{ (Initial } RT)$$
$$M = \text{Margin term}$$

Each of the items in Equation 1 will be discussed with respect to Palisades current situation.

5.1 Values of 'I' and 'M'

Palisades does not have an initial RT_{NDT} value for its reactor vessel welds. This means the plant must use the generic value of -56°F for its axial welds, stated in 10 CFR 50.61 for Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes, reference

3.1. The initial RT_{NDT} was one of the values that the plant intended to get from the retired steam generator welds, but analysis of these welds showed that the material had been affected by its use in the steam generators and could no longer be used to provide initial RT_{NDT}, reference 3.9.

The value of M in Equation 1 is specified in 10 CFR 50.61 as 66°F for welds, when the generic value of I is used.

5.2 Values for 'ΔRT_{PTS}'

The value of ΔRT_{PTS} is calculated from two factors, CF and f, as shown in Equation 2 from 10 CFR 50.61.

$$\Delta RT_{PTS} = (CF) f^{(0.28 - 0.10 \log f)} \qquad Eq.\ 2$$

CF = Chemistry Factor

f = Best estimate neutron fluence units of 10^{19} n/cm^2

5.2.1 Palisades 'CF' value.

The value of CF for Palisades comes from the table of generic weld CF's provided in a table in 10 CFR 50.61 for plants without credible surveillance data. This table relies on the copper and nickel content of the weld material to determine the CF. Attachment 4 gives the copper and nickel contents for comparable heat No. W5214 welds other than the steam generator welds which are shown in Attachment 5. Explanations of the weld designations are provided in attachment 10. Table 5.1 shows the chemistry values for the three 'A' steam generator welds segments from Attachment 5 and their copper and nickel averages. The samples taken from A steam generator were tandem heat No. W5214 welds, the B steam generator samples were from heat No. 34B009: only the heat No. W5214 values are of interest in this EA, since welds fabricated using weld wire from this heat are limiting. The new data taken for heat No. 34B009 does not change the limiting weld for the Palisades reactor vessel.

Sample	Weldment 'A' Copper	Nickel	'A/SG/A' Copper	Nickel	'A/SG/B' Copper	Nickel
1	0.341	1.093	0.367	1.154	0.353	1.203
2	0.310	1.003	0.291	1.156	0.233	1.149
3	0.266	1.090	0.278	1.059	0.237	1.024
4	0.328	1.116	0.365	1.193	0.359	1.204
5	0.310	1.006	0.292	1.127	0.239	0.960
6	0.266	1.104	0.281	1.066	0.228	1.107
	Average Cu	0.297		Average Ni	1.101	

Table 5.1 Averages of Retired Steam Generator Weld Chemistries.

Table 5.2 uses the values from Table 5.1 and Attachment 4 to give all the weld sample values for copper and nickel. It also provides the averages of copper and nickel content for use in determining Palisades reactor vessel axial weld material CF from 10 CFR 50.61. Some of the copper values have been double counted because they were from tandem welds. This is the same averaging technique as used in Reference 3.4.

I.D.	Copper	I.D.	Nickel
D4463 IP2	0.20	D4494 IP2	0.94
" "	0.20	D4541	1.20
HBR2 Torus	0.159	D4577 & D4604	1.00
" "	0.159	D4673 Mill 1C	1.05
IP2 Sur	0.20	D4674 IP2	1.12
IP3 Sur	0.16	D4686 ML1	0.97
" "	0.16	D4687 IP21	0.92
IP3 Nozzle	0.15	D4688 Pal	0.99
HBR2 Sur	0.34	D4690	1.13
OC1 Sur	0.285	HBR2 Torus	0.99
Palisades SG	0.297	IP2 Sur	1.03
" "	0.297	IP3 Sur	1.12
Average	0.217	IP3 Nozzle	1.09
		HBR2 Sur	0.66
		Palisades SG	1.101
		Average	1.02

Table 5.2 Best Estimate Cu and Ni Values for Palisades Axial Welds.

The best estimate Cu value for Palisades axial welds is 0.217 and the Ni value is 1.02. These values can be used with Table 1 of 10 CFR 50.61, shown in Attachment 1, to determine a CF for use in calculating the Palisades PTS screening criteria fluence value. Using linear interpolation, as allowed by the rule, the CF = 233.54°F, which rounds to 234°F.

5.2.2 Palisades 'f' Values

To date, Palisades has only officially submitted fluence values for cycles 1 through 10, Reference 3.3 and 3.6; these values are restated in a more convenient format in Attachment 6. In order to calculate Palisades current accumulated fluence it is necessary to use cycle 10 fluence values from Reference 3.6, and apply cycle 9 fluence rates, reference 3.3, to cycle 11. Based on current core design the use of cycle 9 fluence rates for cycle 11 at the 30° weld is approximately 17% conservative.

Attachment 6 shows that the EOC 10 accumulated fluence at the 30° weld location is $1.25*10^{19}$ n/cm^2.

5.3 Palisades PTS Screening Criteria Limits

Equations 1 and 2 from 10 CFR 50.61 can be solved for f, as shown in Attachment 2, giving Equation 3 shown below.

$$f = 10^{\frac{0.28 - \sqrt{0.0784 - 0.4 \log \frac{(RT_{PTS} - I - M)}{CF}}}{0.2}} \qquad \text{Eq. (3)}$$

The maximum RT_{PTS} allowed for Palisades axial welds is 270°F, reference 3.1. Using this 270°F value for RT_{PTS}, -56°F for I, 66°F for M, and 234°F for CF, in Equation 3, gives a screening criteria fluence value of $1.49*10^{19}$ n/cm^2. This value and Palisades current fluence accumulation can be used to determine the number of EFPD's remaining before the plant reaches the PTS screening criteria. This is shown below.

$$\textit{PTS screening criteria fluence} = 1.49*10^{19}$$

$$\textit{End of cycle } 10 \textit{ fluence} = 1.25*10^{19}$$

$$\textit{Cycle } 9 \textit{ fluence rate} = 2.00*10^{10}*3600*24 = 1.73*10^{15}$$

$$EFPD's = \frac{1.49*10^{19} - 1.25*10^{19}}{1.73*10^{15}} = 1387\ EFPD's$$

$$\textit{Margin} = 1387\ EFPD's - 232\ EFPD's\ (\textit{thru } 10\text{-}31\text{-}94) = 1155\ EFPD's$$

Using the 75% capacity factor and the 1155 EFPD's gives Palisades 4.21 years before reaching the PTS screening criteria. This works out to a date sometime in mid January, 1999.

6.0 Conclusion

The objective of this EA has been met. Palisades PTS screening criteria margin has been calculated using the chemistry data received from testing performed on the retired steam generator welds. The data provided shows that the Palisades reactor

vessel weld material has not exceed the PTS screening criteria.

Attachment 1

(c) The holder of a license authorizing operation of a production or utilization facility who desires (1) a change in technical specifications or (2) to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to § 50.90.

§ 50.60 Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation.

(a) Except as provided in paragraph (b) of this section, all lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.

(b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.

§ 50.61 Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

(a) *Definitions.* For the purposes of this section:

(1) "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, "Rules for the Construction of Nuclear Power Plant Components," edition and addenda as specified by § 50.55a, Codes and Standards.

(2) "Pressurized Thermal Shock Event" means an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel.

(3) "Reactor Vessel Beltline" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

(4) "Initial RT_{NDT}" means the reference temperature for a reactor vessel material as defined in the ASME Code, Paragraph NB-2331. RT_{NDT} means the reference temperature as adjusted for the effects of neutron radiation for the period of service in question.

(5) "RT_{PTS}" means the reference temperature calculated by the method given in paragraph (b)(2) of this section for use as a screening criterion.

(b) *Requirements.*

(1) For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall submit projected values of RT_{PTS} for reactor vessel beltline materials by giving values for the time of submittal, the expiration date of the operating license, the projected expiration date if a change in the operating license has been requested, and the projected expiration date of a renewal term if a request for license renewal has been submitted. The assessment must use the calculative procedures given in paragraph (b)(2) of this section. The assessment must specify the bases for the projection, including the assumptions regarding core loading patterns. The submittal must list the copper and nickel contents and the fluence values used in the calculation for each beltline material. If these quantities differ from those submitted in response to the original PTS rule and accepted by the NRC, justification must be provided. If the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license or the proposed expiration date if a change in the license has been requested, or the end of a renewal term if a request for license renewal has been submitted, this assessment must be submitted by December 16, 1991. Otherwise, this assessment must be submitted with the next update of the pressure-temperature limits, or the next reactor vessel material surveillance report, or 5 years from the effective date of this rule, whichever comes first. These submittals must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon a request for a change in the expiration date for operation of the facility.

(2) The pressurized thermal shock (PTS) screening criterion is 270°F for plates, forgings, and axial weld materials, or 300°F for circumferential weld materials. For the purpose of comparison with this criterion, the value of RT_{PTS} for the reactor vessel must be calculated as follows, except as provided in paragraph (b)(3) of this section. The calculation must be made for each weld and plate, or forging, in the reactor vessel beltline.

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

(i) "I" means the initial reference temperature (RT_{NDT}) of the unirradiated material measured as defined in the ASME Code, Paragraph NB-2331. Measured values must be used if credible values are available; if not, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and −56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

(ii) "M" means the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and the calculational procedures. In Equation 1, M is 66°F for welds and 48°F for base metal if generic values of I are used, and M is 56°F for welds and 34°F for base metal if measured values of I are used.

(iii) ΔRT_{PTS} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

Equation 2: $\Delta RT_{PTS} = (CF) f^{(0.28 - 0.10 \log f)}$

(iv) CF (°F) is the chemistry factor, a function of copper and nickel content. CF is given in table 1 for welds and table 2 for base metal (plates and

forgings). Linear interpolation is permitted. In Tables 1 and 2 "Wt-% copper" and "Wt-% nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld. If these values are not available, the upper limiting values given in the material specifications to which the vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data [1] may be used if justification is provided. If none of these alternatives are available, 0.35% copper and 1.0% nickel must be assumed.

(v) "f" means 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.

TABLE 1.—CHEMISTRY FACTOR FOR WELDS, °F

Copper, Wt-%	Nickel, Wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	228	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

[1] Data from reactor vessels fabricated to the same material specification in the same shop as the vessel in question and in the same time period is an example of "generic data."

56 FR 22300

TABLE 2.—CHEMISTRY FACTOR FOR METAL, °F

Copper, Wt-%	Nickel, Wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	196	208	214
0.26	109	130	151	180	206	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	256	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	206	234	264	282
0.34	149	164	184	209	238	[illegible]	290
0.35	153	168	187	212	241	272	288
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	249	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

56 FR 22300

(3) To verify that the values of RT_{PTS} calculated as required by paragraph (b)(2) of this section are bounding values for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and surveillance results. Results from the plant-specific surveillance program shall be integrated into the embrittlement estimate if,

(i) The plant-specific surveillance data has been deemed credible as defined in Regulatory Guide 1.99 Revision 2, and

(ii) The RT_{PTS} value changes significantly.[9]

Any information that is believed to improve the accuracy of the RT_{PTS} value significantly shall be reported to the Director, Office of Nuclear Reactor Regulation. Values of RT_{PTS} that have been modified using the procedures of this paragraph are subject to the approval of the Director, Office of Nuclear Reactor Regulation when used as provided in this section.

(4) For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion before the expiration date of the operating license, or the projected expiration date if a change in the license has been requested, or the end of a renewal term if a request for license renewal has been submitted, the licensee shall submit by March 16, 1992, an analysis and schedule for implementation of such flux reduction programs as are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2) of this section. The schedule for implementation of flux reduction measures may take into account the schedule for submittal and anticipated Commission approval of detailed plant-specific analyses, submitted to demonstrate acceptable risk at values of RT_{PTS} above the screening limit due to plant modifications, new information or new analysis techniques.

(5) For each pressurized water nuclear power reactor for which the analysis required by paragraph (b)(4) of this section indicates that no reasonably practicable flux reduction program will prevent the value of RT_{PTS} from exceeding the PTS screening criterion before the expiration date of the operating license, or the projected expiration date if a change in the operating license has been requested, or the end of a renewal term if a request for license renewal has been submitted, the licensee shall submit a safety analysis to determine what, if any, modifications to equipment, systems, and operation are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events if continued operation beyond the screening criterion is allowed. In the analysis, the licensee may determine reactor vessel materials properties based on available information, research results, and plant surveillance data, and may use probabilistic fracture mechanics techniques. This analysis must be submitted at least 3 years before the value of RT_{PTS} is projected to exceed the PTS screening criterion or by one year after the effective date of this amendment, whichever is later.

(6) After consideration of the licensee's analyses (including effects of proposed corrective actions, if any) submitted in accordance with paragraphs (b)(4) and (b)(5) of this section, the Commission may, on a case-by-case basis, approve operation of the facility at values of RT_{PTS} in excess of the PTS screening criterion. The

[9] Changes to RT_{PTS} values are considered significant if either the value determined in paragraph (b)(2) of this section or the alternate value determined in paragraph (b)(3) of this section, or both values, exceed the screening criterion, prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.

Commission will consider factors significantly affecting the potential for failure of the reactor vessel in reaching a decision.

(7) If the Commission concludes, pursuant to paragraph (b)(6) of this section, that operation of the facility at values of RT_{PTS} in excess of the PTS screening criterion cannot be approved on the basis of the licensee's analyses submitted in accordance with paragraphs (b)(4) and (b)(5) of this section, the licensee shall request and receive Commission approval prior to any operation beyond the criterion. The request must be based upon modifications to equipment, systems, and operation of the facility in addition to those previously proposed in the submitted analyses that would reduce the potential for failure of the reactor vessel due to PTS events, or upon further analyses based upon new information or improved methodology.

56 FR 22300

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all commercial light-water-cooled nuclear power plants.

(b) *Definition.* For purposes of this section, "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.

(c) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

49 FR 26036

54 FR 13361

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

49 FR 26036

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Commission as specified in § 50.4.

51 FR 40303

(d) *Implementation.* By 180 days after the issuance of the QA guidance for non-safety related components, each licensee shall develop and submit to the Commission, as specified in § 50.4, a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

53 FR 23203

§ 50.63 Loss of all alternating current power.

(a) *Requirements.* (1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:

(i) The redundancy of the onsite emergency ac power sources;

(ii) The reliability of the onsite emergency ac power sources;

(iii) The expected frequency of loss of offsite power; and

(iv) The probable time needed to restore offsite power.

(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

(b) *Limitation of scope.* Paragraph (c) of this section does not apply to those plants licensed to operate prior to *July 21, 1988*, if the capability to withstand station blackout was specifically addressed in the operating license proceeding and was explicitly approved by the NRC.

53 FR 23203

(c) *Implementation.*—(1) *Information Submittal.* For each light-water-cooled nuclear power plant licensed to operate on or before *July 21, 1988*, the licensee shall submit the information defined below to the Director of the Office of Nuclear Reactor Regulation by *April 17, 1989*. For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance.

(i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section;

(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom; and

(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications.

(2) *Alternate ac source:* The alternate ac power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the

Attachment 2

4.0 PROJECTED RT_{PTS}

The following describes how the PTS reference temperatures are determined for each of the Palisades reactor vessel beltline materials and includes projections for when each material will exceed the applicable screening criterion. The results are dependent on the best-estimate values for chemistry and fluence that have been addressed earlier in this report. Additionally this section provides response to NRC concerns as to how surveillance results from Palisades and other reactor vessels could affect the projected RT_{PTS} values.

4.1 Determination and Projection of the PTS Reference Temperatures

The base equation for the PTS reference temperature from 10CFR50.61 is:

$$RT_{PTS} = I + M + \Delta RT_{PTS} \qquad (1)$$

"I" is defined as the initial reference temperature (RT_{NDT}) of the unirradiated material. "I" values for the Palisades reactor vessel beltline materials are:

Axial Weld	I_a	= -56°F	Generic Value 10CFR50.61 (b)(2)(1) for Welds made with Linde 1092 and 124 Fluxes
Circ Weld	I_c	= -56°F	
Plate	I_p	= 0°F	Value* reported in Reference 6. This represents the limiting plate.

* A less conservative value of I_p = -10°F was measured by Battelle Columbus Laboratories in 1977 (Reference 39). A value of -5°F was used in CPCo's 1986 (Reference 16) and 1991 (Reference 1) PTS submittals. Confirmation of -5°F could not be found by measurement or calculation.

"M" is defined as the margin term added to cover uncertainties as in the values of initial RT_{PTS} (Cu and Ni content, fluence and the calculational procedures). Values of "M" for the Palisades vessel beltline material are:

Axial Weld	$M_a = 66°F$	Value specified in 10CFR50.61(b)(2)(ii) for welds if generic values of "I" are used.
Circ Weld	$M_c = 66°F$	
Plate	$M_p = 34°F$	Value specified for base metal in 10CFR50.61 if measured value of "I" is used

"ΔRT_{PTS}" is defined as:

$$\Delta RT_{PTS} = (CF)\, f^{(0.28 - 0.10 \log f)} \tag{2}$$

"CF", the chemistry factor, a function of Cu and Ni content, is derived from Tables 1 and 2 of 10CFR50.61.

In Section 2, the chemistry factors were determined to be:

CF_a = 217°F for the axial welds.

CF_c = 228°F for the circumferential weld.

CF_p = 165°F for the vessel plate material.

"f" is the best-estimate neutron fluence in units of 10^{19} n/cm^2 (E > 1 MeV) at the clad-base metal interface of the vessel.

The limiting fluence is determined by setting RT_{PTS} equal to the screening criteria and solving for f. First, rearranging equations (1) and (2):

$$RT_{PTS} = I + M + (CF)\, f^{(0.28 - 0.10 \log f)}$$

$$(0.28 - 0.10 \log f) \log f = \log\left(\frac{RT_{PTS} - I - M}{CF}\right)$$

$$0.10 (\log f)^2 - 0.28 \log f + \log\left(\frac{RT_{PTS} - I - M}{CF}\right) = 0$$

Using the quadratic equation to solve for log f:

$$\log f = \frac{0.28 \pm \sqrt{(0.28)^2 - 4\,(0.10) \log\left(\frac{RT_{pts} - I - M}{CF}\right)}}{2\,(0.10)}$$

Because the positive root of the equation provides meaningless results, the equation may be simplified to:

$$f = 10 \exp\left[\frac{0.28 - \sqrt{0.0784 - 0.4 \log\left(\frac{RT_{pts} - I - M}{CF}\right)}}{0.2}\right]$$

The maximum allowed values of RT_{PTS} is defined in 10CFR50.61(b)(2) for each of the Palisades beltline is:

Axial Weld	RT_{PTSa} = 270°F
Circumferential Weld	RT_{PTSc} = 300°F
Plate Material	RT_{PTSp} = 270°F

Attachment 3

Table 8-4 (Continued)

Palisades Fast Neutron Fluence (E > 1.0 MeV) Through Cycle 9
At the Reactor Vessel Clad-Base Metal Interface

Cycle	Cycle Length (EFPD)	Cycle Flux (n/cm^2-s)	Cycle Fluence (n/cm^2)	Cumulative Fluence (n/cm^2)
		30 Degrees		
1	379.4	4.43E+10	1.45E+18	1.45E+18
2	449.1	4.43E+10	1.72E+18	3.17E+18
3	349.5	4.43E+10	1.34E+18	4.51E+18
4	327.6	4.43E+10	1.26E+18	5.77E+18
5	394.6	4.43E+10	1.51E+18	7.28E+18
6	333.4	4.52E+10	1.30E+18	8.58E+18
7	369.9	4.52E+10	1.44E+18	1.00E+19
8	373.6	2.21E+10	7.13E+17	1.07E+19
9	298.5	1.89E+10	4.87E+17	1.12E+19
		45 Degrees		
1	379.4	2.81E+10	9.22E+17	9.22E+17
2	449.1	2.81E+10	1.09E+18	2.01E+18
3	349.5	2.81E+10	8.49E+17	2.86E+18
4	327.6	2.81E+10	7.96E+17	3.66E+18
5	394.6	2.81E+10	9.58E+17	4.62E+18
6	333.4	2.86E+10	8.23E+17	5.44E+18
7	369.9	2.86E+10	9.14E+17	6.35E+18
8	373.6	1.67E+10	5.39E+17	6.89E+18
9	298.5	1.09E+10	2.80E+17	7.17E+18

Attachment 4

CALCULATION OF THE MEAN COPPER AND NICKEL CONTENT
OF WELDS FABRICATED USING WELD WIRE FROM HEAT No. W5214
The following identifications and copper content values are from Table 1.1.

1. COPPER CONTENT

Sample Identification	Weight % Copper
D4463 - IP2-flange 1-042B	0.20
" " " "	0.20
HBR2 - Torus Flange	0.159
" " "	0.159
IP2 - Surveillance	0.20
IP3 - Surveillance	0.16
" "	0.16
IP3 - Nozzle Cutout	0.15
HBR2 - Surveillance	0.34
OC1 - Surveillance	0.285
Total	2.013

2.013 ÷ 10 = 0.201 = Mean Copper Content

2. NICKEL CONTENT

Sample Identification	Weight % Content
D4494 - IP2 1-042	0.94
D4541	1.20
Average of D4577 & D4604	1.00
D4673 Millstone 1C	1.05
D4674 IP2 3-042B	1.12
D4686 ML1 2-072A	0.97
D4687 IP21-042A	0.92
D4688 PAL S/G 5-943	0.99
D4690	1.13
HBR2 Torus Flange	0.99
IP2 - Surveillance	1.03
IP3 - Surveillance	1.12
IP3 - Nozzle Cutout	1.09
HBR2 - Surveillance	0.66
Total	14.21

14.21 ÷ 14 = 1.015 = Mean Nickel Content

Attachment 5

AEA

AEA Technology

Facsimile

Date	14 November 1994
To	John Kneeland
Company	Consumers Power Company
Facsimile number	(9) 0101 616 764 8196
From	Dr. G. Gage
Address	Materials Performance Department: B388, Harwell Laboratory, Didcot, Oxfordshire, OX11 0RA, United Kingdom
Telephone	235 434466
Facsimile number	235 432337
Page	1 of 3
Copies:	Neil Irvine , AEA O'DONNELL: (9) 0101 412 655 2928

MPD/082
#2474

Message: **TESTING OF WELDMETALS FOR CPCO**

ADDITIONAL CHEMICAL ANALYSIS

Attached is a table giving the values for the copper and nickel contents of the repeat, second set of analyses. These values are the average values of the three determinations performed on each sample. They have also been normalised appropriately based on analysis results for the standards, however they have not been verified by the section manager.

To aid comparison I have also provided the data from the first set of samples alongside.

Also attached is a copy of the fax that I have received from TWI concerning visual examination of the fracture surfaces of drop weight specimens; AQ1, AQ2, AJ1 and BI2.

Unfortunately I will be out of the office tomorrow, Tuesday 15/11/94, and hence not contactable. As such I would propose making a start on testing of the remaining three batches of Charpy specimens (Weld B; transverse, Weld A: longitudinal and Weld B: longitudinal) on Wednesday. I will try to contact you before doing so in order to confirm that such testing is in keeping with your wishes; Richard Miller indicated that you would be reviewing your requirements today (Monday).

IF THERE ARE ANY PROBLEMS WITH TRANSMISSION OF THIS FACSIMILE PLEASE RING TEL No. 235 434332

COMPARISON OF COPPER AND NICKEL CONTENTS OF SAMPLES ANALYSED IN SECOND SET WITH THOSE OBTAINED ON THE ORIGINAL SAMPLES

SECOND SET	Ni	Cu	Cu	Ni	FIRST SET
Section through Large Weldment 'A'					Section through Large Weldment 'A'
AR1/X	1.116	0.328	0.341	1.093	A1/1/X
AR1/Y	1.006	0.310	0.310	1.003	A1/1/Y
AR1/Z	1.104	0.266	0.266	1.090	A1/1/Z
Section through Large Weldment 'B'					Section through Large Weldment 'B'
BR1/X	1.092	0.198	0.235	1.215	B1/2/X
BR1/Y	0.906	0.198	0.189	1.010	B1/2/Y
BR1/Z	1.057	0.196	0.196	1.098	B1/2/Z
Section through Trepan 'A/SG/A'					Section through Trepan 'A/SG/A'
A/SG/A/3/X	1.193	0.365	0.367	1.154	A/SG/A/2/X
A/SG/A/3/Y	1.127	0.292	0.291	1.156	A/SG/A/2/Y
A/SG/A/3/Z	1.066	0.281	0.278	1.059	A/SG/A/2/Z
Section through Trepant 'A/SG/B'					Section through Trepant 'A/SG/B'
A/SG/B/2/X	1.204	0.359	0.353	1.203	A/SG/B/3/X
A/SG/B/2/Y	0.960	0.239	0.233	1.149	A/SG/B/3/Y
A/SG/B/2/Z	1.107	0.228	0.237	1.024	A/SG/B/3/Z
Section through Trepan 'B/SG/A'					Section through Trepan 'B/SG/A'
B/SG/A/3/X	1.256	0.191	0.195	1.272	B/SG/A/2/X
B/SG/A/3/Y	1.292	0.192	0.195	1.331	B/SG/A/2/Y
B/SG/A/3/Z	0.998	0.204	0.206	1.138	B/SG/A/2/Z
Section through Trepan 'B/SG/B'					Section through Trepan 'B/SG/B'
B/SG/B/3/X	1.117	0.165	0.162	1.126	B/SG/B/2/X
B/SG/B/3/Y	1.088	0.203	0.208	1.136	B/SG/B/2/Y
B/SG/B/3/Z	1.292	0.209	0.209	1.307	B/SG/B/2/Z

The above data correspond to the average values of the three determinations performed on each sample. All results have been subjected to the normalisation procedure based on calibration with standard specimens, however data for the second set of samples have yet to be verified by the section manager.

G. Gage
14th November 1994

TWI

TELEFAX

TWI, Abington Hall, Abington, Cambridge CB1 6AL UK
Telephone: +44 (0)223 891162 Telex 81183 Telefax +44 (0)223 892588

To:	Dr Gareth Gage	Fax Room Ref:	TF/ 1
Company:	AEA Technology	From:	Mr K Bell
Dept:	Materials Performance	Dept:	Structural Integrity
Town:	Harwell, Didcot	Date:	14 November 1994
Country:	UK	Dept Ref:	KB/kb/60.94
Fax No:	0235 - 432337	No of pages:	1 of 1

Please telephone Fax Room on 0223 891162 Ext. 2220 if pages are not received or are unclear

MESSAGE

Pellini Testing of Steam Generator Welds
TWI Project 620751

Dear Gareth,

The four specimens (AEA refs AQ1, AQ2, AJ2, and BI2, TWI refs W01-06, W01-09, W01-04, and W02-04 respectively) have been heat tinted at 280°C for 2 hours, cooled using liquid N_2, broken open and examined under a low powered binocular microscope (~x10).

All four specimens had been extensively fractured during the Pellini test (more than 80% of the fracture surface was tinted).

None of the specimens show any gross welding defects which might have influenced the results of the tests. One specimen (BI2) has a small planar discontinuity (~ 3 x 0.5mm) on the fracture face, 5mm sub-surface.

Two specimens (AJ2 and BI2) showed some evidence of shear lip formation, up to 1mm wide, on the top (tension) surface of the specimen.

I am arranging for all of the specimens, broken and unbroken (excluding W02-01, which Richard Miller took with him after the day of testing), to be returned to you at AEA Harwell.

Best regards.

Yours sincerely,

Keith Bell

Keith Bell,
Senior Project Engineer,
STRUCTURAL INTEGRITY DEPARTMENT

Normal TWI conditions of contract apply as appropriate

Attachment 6

TABLE 6-13 (Continued)

CALCULATED FLUENCE (E>1.0 MeV) THROUGH CYCLE 10
AT THE PRESSURE VESSEL CLAD-BASE METAL INTERFACE

Cycle	Cycle Length (EFPD)	Cycle Flux (n/cm2-sec)	Cycle Fluence (n/cm2)	Cumulative Fluence (n/cm2)
		<u>30 Degree</u>		
1	379.4	4.70E+10	1.54E+18	1.54E+18
2	449.1	4.70E+10	1.82E+18	3.36E+18
3	349.5	4.70E+10	1.42E+18	4.78E+18
4	327.6	4.70E+10	1.33E+18	6.11E+18
5	394.6	4.70E+10	1.60E+18	7.71E+18
6	333.4	[illegible]	1.38E+18	9.09E+18
7	369.9	4.79E+10	1.53E+18	1.06E+19
8	373.6	2.34E+10	7.55E+17	1.14E+19
9	298.5	2.00E+10	5.16E+17	1.19E+19
10	356.9	1.94E+10	5.98E+17	1.25E+19
		<u>45 Degree</u>		
1	379.4	2.98E+10	9.78E+17	9.78E+17
2	449.1	2.98E+10	1.16E+18	2.13E+18
3	349.5	2.98E+10	9.00E+17	3.04E+18
4	327.6	2.98E+10	8.44E+17	3.88E+18
5	394.6	2.98E+10	1.02E+18	4.90E+18
6	333.4	3.03E+10	8.73E+17	5.77E+18
7	369.9	3.03E+10	9.68E+17	6.74E+18
8	373.6	1.77E+10	5.71E+17	7.31E+18
9	298.5	1.15E+10	2.97E+17	7.61E+18
10	356.9	1.32E+10	4.07E+17	8.02E+18

Attachment 7

Palisades Cycle Flux Values
at Critical Locations

Cycle	EFPD	Cycle Flux E + 10			
		0°	16°	30°	45°
1	379.4	4.59	6.03	4.70	2.98
2	449.1	4.59	6.03	4.70	2.98
3	349.5	4.59	6.03	4.70	2.98
4	327.6	4.59	6.03	4.70	2.98
5	394.6	4.59	6.03	4.70	2.98
6	333.4	4.87	6.25	4.79	3.03
7	369.9	4.87	6.25	4.79	3.03
8	373.6	2.16	4.89	2.34	1.77
9	298.5	2.08	3.06	2.00	1.15
10	356.9	1.51	2.40	1.94	1.32
11	422.0	1.42	2.21	1.66	1.09

Values for cycles 1 through 10 are from WCAP14014.
Values for cycle 11 are from Palisades in-house calculations.

Attachment 8

ABB

November 15, 1994
RVG-94-089

Mr. John Kneeland
Consumers Power Company
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

Subject: **Palisades SG Upper Shell Long Seam Fabrication Technique**

Reference: ABB Letter P-PENG-94-022, Chemical Analysis of SG Weld Seam Samples and Materials Consultation (ABB CENO Proposal No. 1017-840-019-A) dated November 7, 1994.

Dear Mr. Kneeland:

In support of your efforts to analyze and evaluate several steam generator (SG) long seam welds, ABB CE is providing material testing and consulting services on the Palisades RPV integrity. These services are described in the referenced letter. At a site meeting, ABB CE was requested to provide additional steam generator fabrication information.

As requested, ABB has reviewed the fabrication data for the original Palisades steam generators. This review was focused on the welding sequence of the long seam welds of the upper shell of the steam generators. From this effort we were able to determine that the original Palisades SGs' three long seam welds in the upper shell were made using a sequential weld method. Therefore these three seams should be considered as one weld with respect to chemistry. This method is discussed briefly below.

In order to minimize distortion and weld shrinkage, a process was used that incrementally welded each of the upper shell long seams in sequence. These shell plates were machined with double U preps. The upper shell was fabricated in a horizontal position (i.e., lying on its side). After alignment of these plates, weld deposit was then performed using an automated welding process on the OD weld. The automatic welding machine was run the entire length of the seam. Slag was manually chipped from the upper surface of the entire length of the weld bead. The automatic welding machine was repositioned at the far end of the weld seam and a second pass of weld material was deposited along the entire weld seam. This process was continued until the initial OD weld deposit was approximately 1 1/2 inches thick. The shell was then rotated so that the second upper shell plate weld seam was in position. Weld deposit was then performed on the seam OD using the automatic welding process as described for the first seam. Upon completion of the second weld seam

procedure, the shell was rotated so that the third seam was in position. Weld deposit was performed on the third seam as described for seams one and two. The seam welding process then shifted to the ID of the weld joint. After backgrooving to sound weld metal and performing a magnetic particle examination, the first weld pass was deposited on the ID of the first weld joint using the automatic welding equipment. As with the OD welds, slag was removed and subsequent passes were performed until approximately one inch of weld was deposited on the ID of the first weld seam. The shell was then rotated so that the second weld seam was in position. This seam was also built up like the aforementioned ID weld. The process was repeated for the third weld seam. A second set of weld increments was performed sequentially on each of the three ID weld seams. The second set of weld passes completed each of the ID weld seams. Upon completion of the ID weld seams, the welding process was shifted to the OD of the weld seams. Again the second increment of the OD welding process was performed in a sequential fashion. The second increment of the OD weld completed the OD weld. Weld wire was fed into the automatic welding machine from 150 pound spools. Additional wire was added to the weld machine as necessary during the entire welding process.

Approximately six 150 pound spools of weld wire were required to fabricate the three upper shell long seams. The welding sequence described above results in mixing of a portion of each of the weld wire spools in each of the seams. Therefore effects of spool to spool variation in the weld deposits should be the same for each of the three seams. The chemical analysis results from the three welds should be averaged as a single datum point for a weld produced by this fabrication process.

This welding sequence has been described in this letter to aid Consumers Power Company in its evaluation of the weld deposit chemistry. This description of the welding procedure and the actual welding procedure are held by ABB Combustion Engineering as proprietary information. This document contains proprietary information and is not to be transmitted or reproduced without specific written approval from Combustion Engineering, Inc. Consistent with the requirements of 10 CFR 2.790, transmittal of any proprietary information provided herein to the Nuclear Regulatory Commission must be accompanied by an affidavit from Combustion Engineering, Inc.

If you have any questions regarding this letter, please call me at (203) 285-2567.

Sincerely,

COMBUSTION ENGINEERING, INC.

Carl J. Gimbrone

Carl J. Gimbrone
Supervisor, Reactor Vessel Integrity

Attachment 9

CREDIBILITY OF USING STEAM GENERATOR WELDS AS SURROGATES FOR THE PALISADES REACTOR PRESSURE VESSEL WELDS

An Independent Technical Opinion
by
W. L. Server, ATI Consulting

In developing the proper materials to use as surrogates for the Palisades reactor pressure vessel (RPV) beltline welds since no archive weld materials exist (except for the Palisades surveillance weld), the welds in the retired steam generators were selected as candidates. The pedigree of the welds in the steam generators was determined by ABB-CE, and some of the welds in the steam generator were found to be the same weld wire heats as in the RPV. Therefore, material was removed from the steam generators in order to further determine the adequacy of the materials as surrogates for measuring copper-nickel chemistry and use in a supplemental surveillance program. The two weld wires heats of concern were W5214 and 34B009. The following discussion provides details on the similarity and differences between the RPV welds and the steam generator welds as determined comparing fabrication information and measured chemistry and mechanical property data. Note that some of the comparison of chemistry and mechanical property data involves data from other sister RPVs that have the same weld wire heat in the beltline region and/or in their surveillance program: H. B. Robinson Unit 2 (HBR), Indian Point 2 (IP2), Indian Point 3 (IP3), Salem Unit 1 (S1), and two BWRs, Millstone Unit 1 (M1) and Oyster Creek (OC).

The fabrication information for the steam generator welds is very similar to that of the Palisades RPV in that the same weld wire heats from the same manufacturer were used with the same flux type (Linde 1092), welding procedure, and approximate time frame in the same CE shop. (These items of similarity are generally true for the sister RPV welds also.) The dissimilarity issues come in relative to differences in the flux lot numbers, post-weld heat treatment conditions, and number of weld arcs (related to vessel wall thickness). The effect of different flux lot numbers should be inconsequential, but the effects of post-weld heat treatment conditions and thickness can be important. Table 1 illustrates these differences as compiled by ABB-CE. The differences appear to be minor, but there are differences.

Related to post-weld heat treatment is the question of extended time at service temperature and pressure. The steam generators were in service for approximately 8 EFPY at a nominal temperature of 500°F under a pressure of about 700 psi. The effect of service aging on these materials needs careful consideration in determining equivalency with the RPV welds. It is known that thermal aging effects can arise at temperatures near or greater than 600°F for some ferritic steels, but extended time at slightly lower temperatures also could play a significant role. The results from mechanical testing should reveal any differences due to

inservice aging on the weld materials. It should be noted that Charpy V-notch testing of the Palisades surveillance weld after thermal aging revealed a fairly significant difference reflecting time at temperature (5 EFPY at 535°F) as shown in Figure 1. The Palisades surveillance weld was fabricated using a different weld wire heat (3277) but with equivalent flux type and welding procedures that closely matched or bounded the W5214 weld in the RPV; in fact, the surveillance weld baseline curve is very similar to that of the reported baseline curves for weld wire W5214 from IP2, IP3, and HBR as shown in Figure 2.

The pieces of steam generator welds were recently tested to determine copper and nickel chemistry, nil-ductility transition temperature (NDTT or NDT temperature), RT_{NDT}, and Charpy V-notch transition curves. There are no known NDTT measurements for the W5214 welds, and a value of -80°F for the 34B009 has been reported. The measured values for the steam generator materials are -20°F for weld wire heat W5214 and -50°F for weld wire heat 34B009. The 30°F difference between the reported value for 34B009 and the measured value here suggests a potential aging effect. Other issues are also important relative to the drop weight NDT temperature determination: the brittle weld starting bead was fabricated using the latest ASTM E208 specification which was different from that used in the late 1960s through mid 1980s; for some materials, this different starting weld bead can result in different measures of NDTT. Additionally, there appears to be a possible effect for the W5214 material relative to where the NDTT specimens were taken from the weld thickness which is somewhat supported by looking at some of the higher energy Charpy specimen results (i.e., one region of the weld with the highest copper level tends to provide higher levels of toughness possibly indicative of a lower NDTT). The RT_{NDT} value for the steam generator W5214 weld is equivalent to the NDTT since the Charpy V-notch data supports greater than 50 ft-lb and 35 mils lateral expansion at NDTT + 60°F.

The Charpy V-notch data also confirms a potential aging effect for the steam generator materials. The measured Charpy energy data for the steam generator W5214 weld metal is shown in Figure 3 where a comparison is made with the combined baseline data from IP2, IP3, and HBR in the same manner as shown in Figure 2. There is a definite shift in the 30 and 50 ft-lb transition temperatures and a drop in upper shelf energy. The shift difference is approximately 30°F and the decrease in upper shelf is about 10 ft-lb. These differences strongly suggest a thermal aging effect for the steam generator W5214 material. This difference would also suggest the inaccuracy of using the measured NDTT from the steam generator as applied to the Palisades RPV, even though the measured value of -20°F is in the upper range of known NDTT measurements for CE-fabricated welds. Charpy V-notch testing of the similar steam generator weld Heat No. 34B009 is now underway, and a comparison of these results with the original (i.e., unirradiated and unaged) material will provide further evidence of this aging effect.

The aging phenomena have been observed in other vessel weld materials as is described in a recent (unpublished) paper [Ref. 1] containing data from the Doel I and II pressure vessels. Charpy V-notch data from the Doel I reactor vessel weld in the unaged and aged condition is shown in Figure 4 which clearly indicates the effects of aging on shift in the Charpy curve.

The use of the steam generator material to add to the data base relative to copper-nickel chemistry should be adequate since the bulk measurements are unaffected by heat treatment or service aging considerations. The specific data from the chemistry measurements are discussed elsewhere, but the key results show good agreement between prior measurements for the 34B009 weld, but higher than expected copper values for the W5214 weld. These higher values for weld wire heat W5214 have been factor into an average for all of the measurements for W5214 and applied to the Palisades RP weld.

In summary, the Charpy V-notch and NDTT values from the steam generator welds appear to exhibit a service aging phenomenon which invalidate their use directly as measures of the virgin mechanical properties for the subject welds. It may be possible to thermally anneal these steam generator materials to restore the properties back to their equivalent virgin condition, and this approach should be pursued in combination with microstructural characterization work to better understand the embrittlement process. The success of achieving equivalent virgin mechanical properties will be used in further assessing the steam generator materials as appropriate surrogates for the supplemental surveillance program for the Palisades RPV.

Reference

1. Gerard, R., Fabry, A., Van de Velde, J., Puzzolante, J. L., Verstrepen, A., Van Ransbeeck, T., and Van Walle, E., **"In-Service Embrittlement of the Pressure Vessel Welds at the Doel I and II Nuclear Power Plants,"** Effects of Radiation on Materials: 17th International Symposium, ASTM STP 12XX, David S. Gelles, Randy K. Nanstad, Arvind S. Kumar, and Edward E. Little, Editors, American Society for Testing and Materials, Philadelphia, 1995 (currently under editorial review).

Table 1

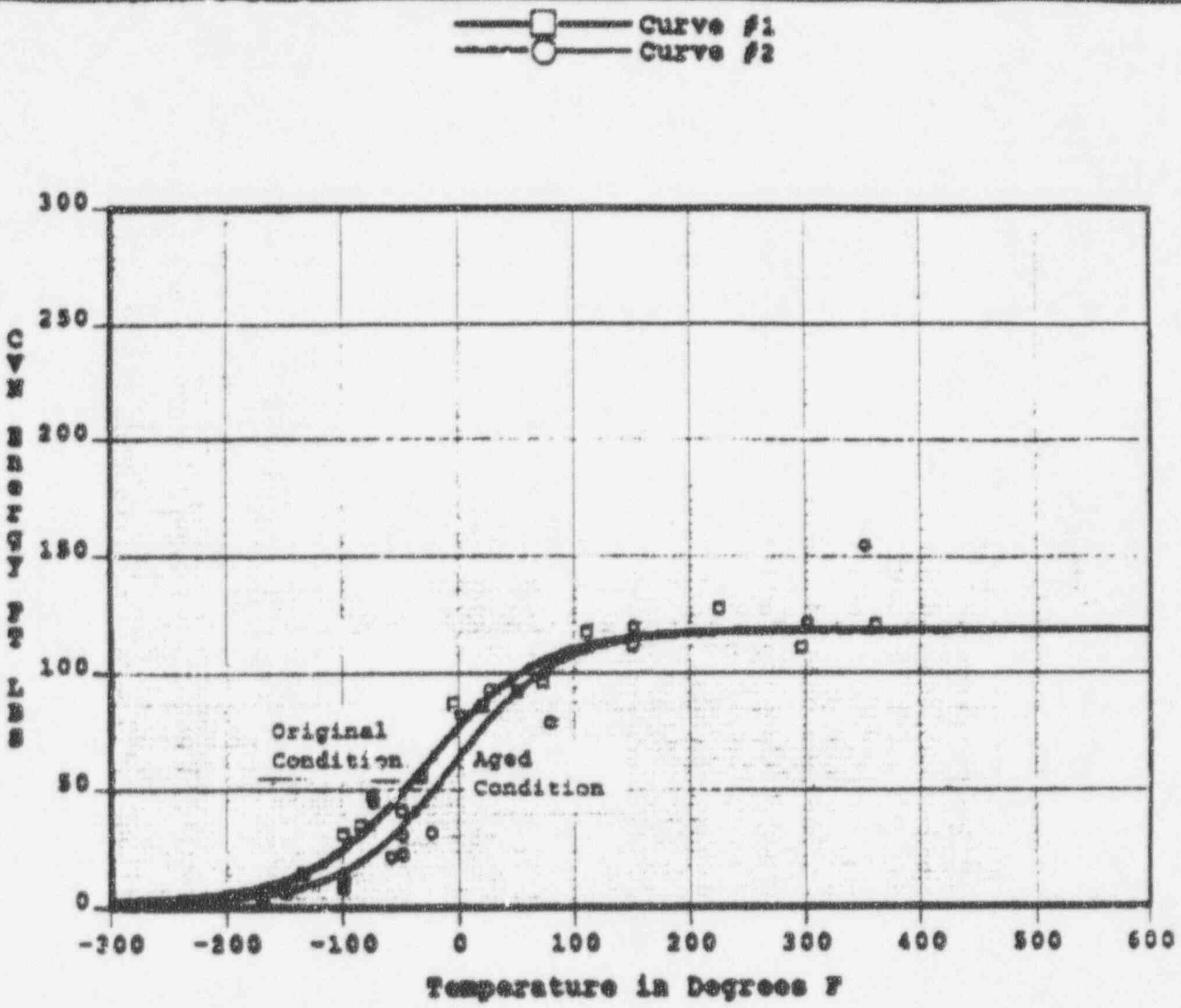
Palisades Vessel Beltline and Steam Generator
Weld Fabrication Details

Weld Seam	2-112 A/C	3-112 A/C	1-951 A/C	1-951 A/C
Location of Weld	Int. Shell Long. Seams	Int. Shell Long. Seams	Steam Gen. No. 1	Steam Gen. No. 2
Weld Wire Heat Numbers	W5214 + Ni 200	W5214 + Ni200 34B009 + Ni200	W5214 + Ni200	34B009 + Ni200
Flux Type	Linde 1092	Linde 1092	Linde 1092	Linde 1092
Flux Lot No.(s)	3617	3692, 3617	3617	3708
Thickness (in.)	8.5	8.5	4.75	4.75
PWHT (Hours at Temp > 11.0 F)	14.75	14	17.4	9.5

Palisades Surveillance Weld Data

Hyperbolic Tangent Curve Fitting Routine Version 2.0 Printed at 11:12:12 on 11-15-1994

	Material: WELD SA302BM	
Capsule:	Heat No.:3277	Orientation:TL



Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0.00E+00	2.2	0.0	118.2	0.0	-86.7	0.0	-47.2	0.0
2	1.00E+13	2.2	-0.0	118.0	-0.2	-58.5	28.2	-22.1	25.0

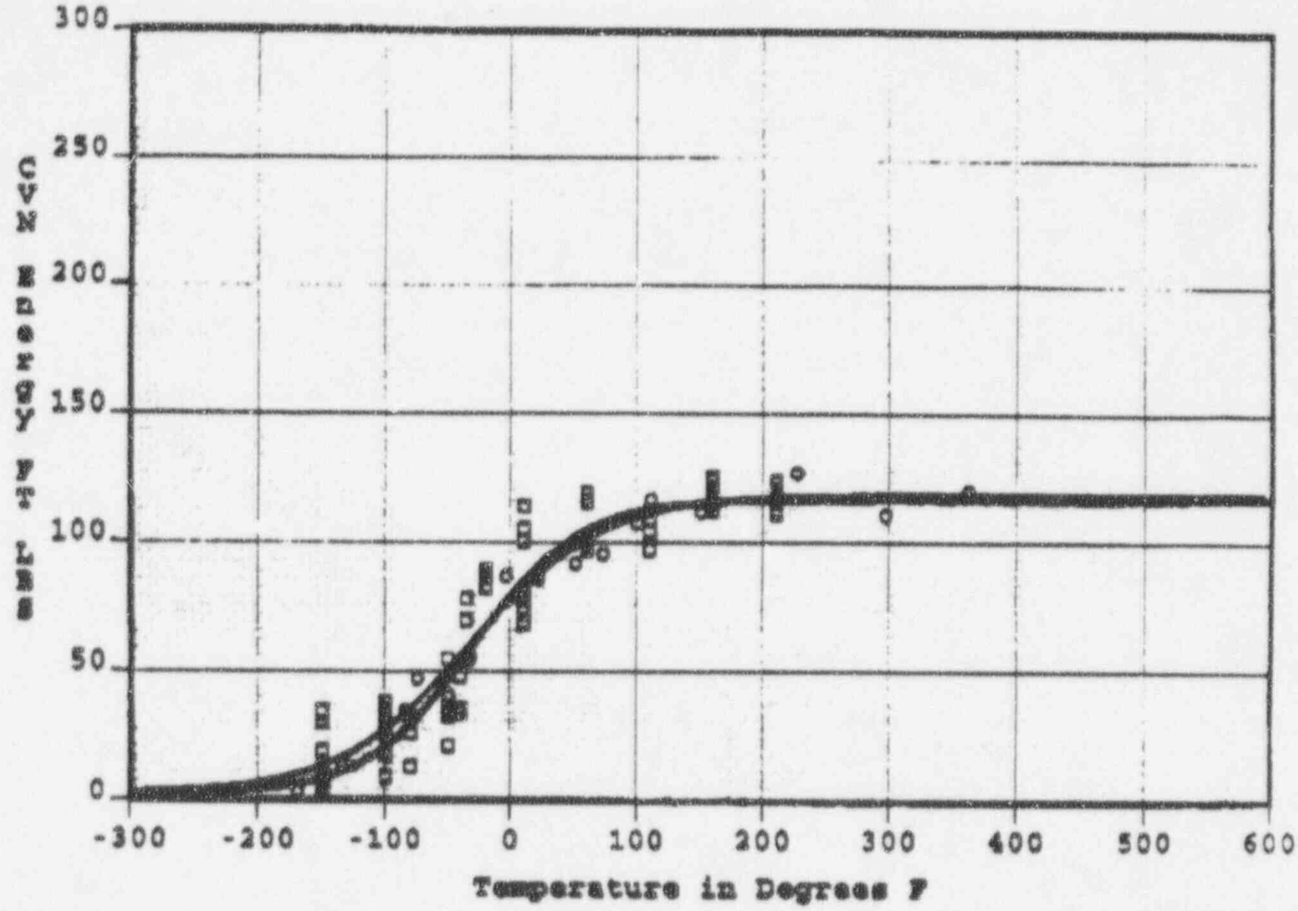
Figure 1

IP2, IP3 HBR2 and Palisades Surveillance Weld Data

Hyperbolic Tangent Curve Fitting Routine Version 3.0 Printed at 09:11:42 on 11-17-1994

	Material: Linde 1092 SAW	
Capsule:	Heat No.:W5214 & 3277	Orientation:

Curve #1 IP2, IP3 HBR2 Data
Curve #2 Palisades Data



Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 30	d-T @ 30	T @ 50	d-T @ 50
1	0.00E+00	2.2	0.0	116.8	0.0	-78.8	0.0	-44.6	0.0
2	0.00E+00	2.2	0.0	118.7	2.0	-93.5	-14.7	-51.1	-6.5

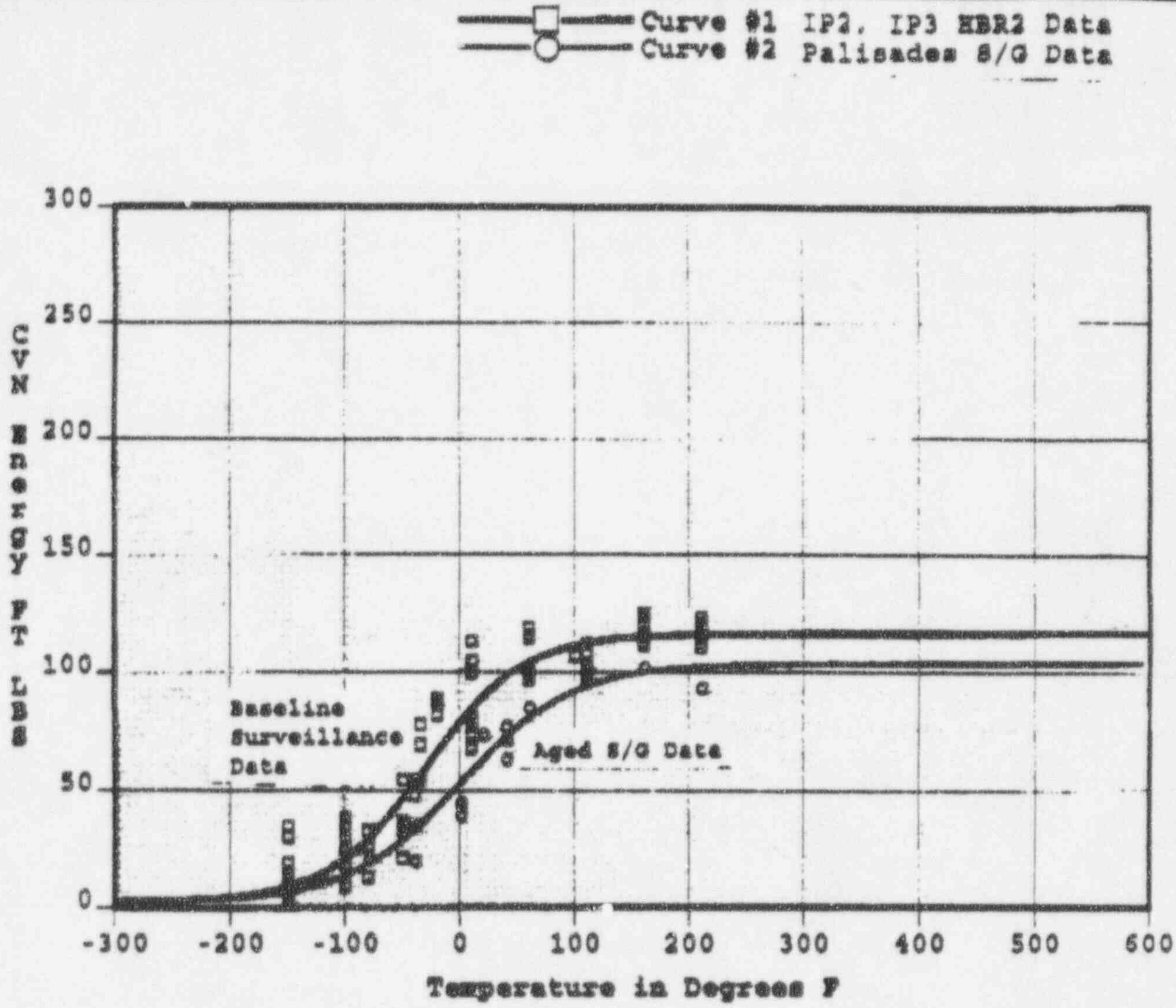
Figure 2

IP2, IP3 HBR2 Surveillance Data & Palisades S/G Data

Hyperbolic Tangent Curve Fitting Routine Version 2.0 Printed at 09:17:21 on 11-17-1994

Material: Linde 1092 SAW

Capsule: Heat No.:W5214 Orientation:

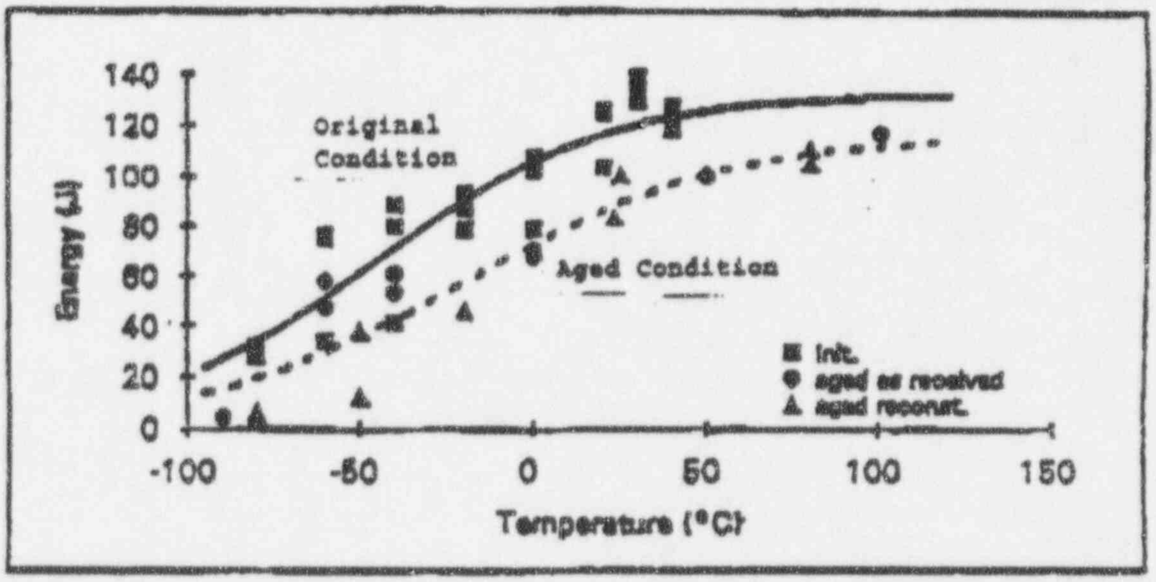


Curve	Fluence	LSE	d-LSE	USE	d-USE	T @ 10	d-T @ 10	T @ 50	d-T @ 50
1	0.00E+00	2.2	0.0	116.8	0.0	-78.8	0.0	-44.6	0.0
2	0.00E+00	2.2	0.0	104.0	-12.8	-67.5	11.2	-5.9	38.8

Figure 3

Doel I Unirradiated and Thermally Aged Weld Data

Aged for 63000 hours at 287°C (549°F)



Chemical Composition of Weld

C	Co	Cr	Mn	Ni	P	S	Si	Mo	Cu
0.066	-	0.08	1.31	0.125	0.016	0.013	0.33	0.484	0.12-0.35

Figure 4

Attachment 10

Sample ID Descriptions

Seams A and B

A Steam Generator
Seam
Slice
Sample Position

A/SG/A/2/X

Seam C

A Steam Generator
Sample Slice
Sample Position

A1/1/X

A Steam Generator
Sample Slice
Sample Position

AR/1/X

SAMPLE POSITION