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TDB-V.1.B	Estimated Critical Conditions Worksheet	R23 12-15-09
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TDB-V.9	Shutdown Margin Worksheet	R40 02-09-11
TDB-V.11	Instrument Bus Loads and Failure Modes	R44 12-29-10
TDB-V.12	Miscellaneous Formula Sheet	R10 05-02-08
TDB-V.13	Reactor Vessel Cooldown When Sweeping Steam Generators	R3 05-13-08
TDB-VI	Core Operating Limit Report	R39 11-20-09a
TDB-VII	Tank Curves	R15 08-06-09
TDB-VIII	Equipment Operability Guidance	R49 02-25-11
TDB-IX	RCS Pressure and Temperature Limits Report	R6 03-22-11

Fort Calhoun Station
Unit 1

TDB-IX

TECHNICAL DATA BOOK

RCS PRESSURE AND TEMPERATURE LIMITS REPORT

Change No.	EC 50073
Reason for Change	Update the minimum pressure values on page 8 as well as the revision numbers of FC06877 and EA-FC-01-022.
Requestor	T. McDonald
Preparer	K. Bessey
Issue Date	03-22-11 3:00 pm

Table of Contents

List of Figures

RCS PRESSURE AND TEMPERATURE LIMITS REPORT

INTRODUCTION

The purpose of this Technical Data Book (TDB) section is to provide Fort Calhoun Station (FCS) with an administrative document that defines updating the pressure and temperature (P-T) limit curves and low temperature overpressure protection (LTOP) setpoints and delineates Nuclear Regulatory Commission (NRC) review requirements as defined in the Technical Specifications (TSs) Definitions section. Reference 8.24 was created to document the development of this TDB section.

This Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) for FCS Unit No. 1 contains P-T limits corresponding to 40 Effective Full Power Years (EFPY) of operation. In addition, this report references the LTOP methodology and current analysis that contains the system limits and operating restrictions that protect the P-T limits from being exceeded during limiting LTOP events. Reference 8.1 allows the relocation of the P-T limit curves and LTOP system limits from the plants TSs and relocates them into a PTLR. Reference 8.2 is the topical PTLR that forms the basis for this document except as modified by the individual Sections.

This PTLR will be updated prior to exceeding the adjusted reference temperature (ART (RT_{NDT})) utilized to develop Figure 5-1. The PTLR, including any revisions or supplements thereto, shall be provided upon issuance of P-T limit curves to the NRC Document Control Desk with copies to the Regional Administrator and Senior Resident Inspector.

In addition, anytime it becomes necessary to change the methodology and/or any TSs that were used to develop data generated for this report, a license amendment will also be prepared describing the new methodology and/or TS change and will be submitted for NRC review and approval prior to implementation in this report.

1. NEUTRON FLUENCE VALUES

The most recent reactor vessel beltline neutron fluence has been calculated for the critical locations in Reference 8.3. (Note: The uncertainty associated with the fluence values stated in Reference 8.3 is ±15.5%.) This report/reference contains the following:

a) A description of the methodology used to perform the neutron fluence calculation.
b) A description of the computer codes used to calculate the neutron fluence values.
c) A description of how the computer codes for calculating the neutron fluence values were benchmarked.

The methodology stated in Reference 8.3 is consistent with the guidance of Draft Regulatory Guide DG-1053 (now Regulatory Guide RG 1.190), as stated by the NRC staff in the safety evaluations contained in References 8.4 and 8.5.

The values of fast neutron fluence (E >1 Mev) used in the ART calculations in Section 4 are located in Table 1-1 and are applicable for 40 EFPYs. (Note: The fluence associated with 40 EFPYs versus 48 EFPYs was used in the ART calculations for Figure 5-1 to prevent a reduction in the operating window between the P-T limit and the reactor coolant pump net positive suction head curves.) The 1/4 T and the 3/4 T neutron fluence values were calculated as follows:

a) The clad/base metal interface fluence values for the plates and circumferential weld use the peak neutron value listed in Table 6.2-1 of Reference 8.3 for 40 EFPY. This is due to these materials would be exposed to the highest fluence.
b) The clad/base metal interface fluence value used for the limiting axial welds was the value located at the 60° position for 40 EFPY. The axial welds for the 180° position is not limiting due to the fluence at this location is significantly less than at the 60° and 300° locations. The non-limiting 2-410 welds at the 0°, 120°, and 240° positions are located in geometrically symmetric locations as the 3-410 welds at 60°, 180°, and 300° positions. In Cycle 14, extreme low radial leakage fuel management was implemented to reduce the reactor vessel fast neutron flux. This management scheme and the incorporation of surveillance data from other nuclear power plants per Reference 8.14 ensures that FCS has the potential to operate to August 9, 2033 without exceeding the 10 CFR 50.61 pressurized thermal shock (PTS) screening criteria as approved by the NRC in Reference 8.5.
c) Equation 3 of Reference 8.22 was then used to calculate the 1/4 T and the 3/4 T fluence values as shown in Table 1-1.

(Note: The values in parentheses in Table 1-1 refers to weld wired heat numbers.)

Table 1-1, Neutron Fluence Values for 40 EFPY

Reactor Pressure Vessel Material	1/4 T	3/4 T
D 4802-1	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4802-2	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4802-3	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4812-1	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4812-2	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
D 4812-3	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2
2-410	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (12008/13253)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (12008/27204)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (13253)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
3-410 (27204)	1.4021×10^{19} n/cm^2	0.59629×10^{19} n/cm^2
9-410	1.9825×10^{19} n/cm^2	0.84312×10^{19} n/cm^2

2. REACTOR VESSEL SURVEILLANCE PROGRAM

The reactor vessel surveillance program is described in Section 2, Reference 8.2. The reactor vessel surveillance withdrawal schedule is located in Reference 8.6, Table 4.5-4. This schedule meets the requirements of ASTM-E-185-82 (Reference 8.25). The baseline report describing the pre-irradiation evaluation of the FCS reactor surveillance materials are presented in Reference 8.7. The reports describing the post-irradiation evaluation of the FCS surveillance capsules are contained in References 8.8 - 8.10. Each removed capsule has been evaluated in accordance with the testing requirements of the version of ASTM-E-185 in effect at the time of capsule removal.

3. LTOP SYSTEM LIMITS

The LTOP system setpoints have been developed by making a comparison between the peak transient pressure for each limiting LTOP event and the P-T limit curve of Figure 5-1 to ensure that the P-T limit curve is not exceeded.

These system setpoints and additional limitations for LTOP have been established based on NRC-accepted methodology and are described in References 8.15 and 8.16. (Note: The methodology described in Section 3 of Reference 8.2 was not used for the determination of the LTOP system setpoints.)

The LTOP analysis which contains the current system setpoints and operating restrictions to ensure the P-T limit curve is not exceeded during a limiting LTOP event is located in Reference 8.16. The applicable operating restrictions stated in Reference 8.16 will be maintained in the TSs. Reference 8.21 contains the methodology for incorporating the Reference 8.16 setpoints into the LTOP system actuation circuitry. These conservative values will then be used for incorporation into TDB Figures. The LTOP enable temperature is 350°F. (Reference 8.19)

4. BELTLINE MATERIAL ADJUSTED REFERENCE TEMPERATURE

The calculation of the ART for the reactor vessel beltline region has been performed using the NRC-accepted methodologies as described in Section 4, Reference 8.2. Application of surveillance data was used to refine the chemistry factor and the margin term in Reference 8.14. (See Section 7) The limiting weld for FCS is the 3-410 weld located at the 60°/300° position using weld wire heat 12008/13253. The RT_{PTS} value for the limiting weld is projected to be 268°F with a clad/base metal interface fluence of 2.43 x 10^{19} n/cm^2 at the end of license extension (August 9, 2033).

The ART values in the beltline region for FCS Unit 1 corresponding to 40 EFPY are listed in Table 4-1. (Note: The limiting ART value for the 1/4 T and 3/4 T (Weld 3-410, Weld Wire Heat 12008/13523) was incorporated into Figure 5-1 (References 8.19 and 8.23).)

Table 4-1, ART Values for Reactor Vessel Materials for 40 EFPY

Reactor Pressure Vessel Material	1/4 T (°F)	3/4 T (°F)
D 4802-1	131.56	112.27
D 4802-2	120.45	103.55
D 4802-3	120.76	103.60
D 4812-1	132.51	113.03
D 4812-2	111.14	95.89
D 4812-3	111.14	95.89
2-410	106.88	85.64
3-410 (12008/13253)	237.76	187.97
3-410 (12008/27204)	213.98	164.69
3-410 (13253)	196.26	150.84
3-410 (27204)	223.72	172.30
9-410	233.11	188.89

5. PRESSURE-TEMPERATURE LIMITS USING LIMITING ART IN THE P-T CURVE CALCULATION

The analytical methods used to develop the beltline RCS P-T limits are based on NRC reviewed methodologies as discussed in Section 5 of Reference 8.2. The NRC approved the use of ASME Code Case N-640 for FCS that allows the use of K_{IC} to calculate the reference stress intensity factor K_{IR} values for the reactor pressure vessel as a function of temperature in Reference 8.17. The limit for the maximum pressure in the vessel is 100 percent of the pressure satisfying Paragraph G-2215 of the 1996 Edition of Appendix G to the ASME Code for establishing LTOP limit setpoints. Additionally, an exemption was granted by the NRC to apply CE NSSS methods for determining P-T limit curves.

The ferritic reactor pressure vessel materials that have accumulated neutron fluences in excess of 1.0×10^{17} n/cm^2 (E >1 Mev) regardless of whether the materials are located within the region immediately surrounding the active core have been evaluated (Reference 8.18). This evaluation concluded that the limiting material remained the lower shell axial welds, 3-410 A/C.

Figure 5-1 was developed in Reference 8.19. Uncertainty was incorporated into Figure 5-1 as follows (Reference 8.19):

a) At 210°F and greater, pressure instrument uncertainty is incorporated into the P-T limit curve and below this temperature it is not. (Note: Normally pressure instrument uncertainty is not applied below the LTOP enable temperature (350°F) due to it being incorporated into the LTOP system setpoint curve. However, it was incorporated at 210°F and greater in Reference 8.19.) A pressure instrumentation uncertainty of 50 psi is being used, which bounds the wide and narrow range pressurizer pressure instruments that operators would use to determine RCS pressure.
b) The temperature uncertainty used is 14°F which bounds the instruments that operators would use to determine RCS temperature.

6. MINIMUM TEMPERATURE REQUIREMENTS IN THE P-T CURVES

The minimum temperature requirements specified in Reference 8.20 are applied to the P-T limit curves using the NRC-reviewed methodologies as described in Section 6 of Reference 8.2.

The minimum temperature values applied to the P-T limit curves for FCS Unit 1 corresponding to 40 EFPY are (Note: These limits were calculated in Reference 8.19 and incorporates instrument uncertainty):

a) Minimum Boltup Temperature: 64°F.
b) Minimum Hydrostatic Temperature Test Limits: See Figure 5-1. (Note: The in-service hydrostatic test curve is developed in the same manner as the heatup and cooldown curves with the exception that a safety factor of 1.5 is used in lieu of 2.)
c) Lowest Service Temperature: 164°F.
d) Flange Limit:
   1) Normal Operation: 144°F.
   2) Hydrostatic and Leak Testing: 114°F.
e) Core Critical Temperature Limit: 515°F per TS 2.10.1(1). (Note: This TS limit is more conservative than the core critical temperature limit required by Reference 8.20. Whenever the P-T limit curve of Figure 5-1 is modified, it must be verified that the new core critical peak temperature limit is less than 515°F, or else the core critical P-T limit curve must be included on Figure 5-1 and Section 6, item 'e' must be updated.)

In the development of P-T limits for CE NSSS's, the intent is to utilize the more conservative of either the lowest service temperature or the other minimum temperature requirements for the reactor vessel when the RCS is pressurized to greater than 20% of the pre-service hydrostatic test pressure (PHTP). The "minimum pressure criteria" specified in Reference 8.20 serves as a regulatory breakpoint in the development of P-T limits and is defined as 20% of PHTP. For CE NSSS plants, the PHTP is defined as 1.25 times the design pressure (Note: Design pressure = 2500 psia). The function of minimum pressure in the development of P-T limits is to provide a transition between the various temperature only based P-T limits, such as minimum bolt up and the lowest service temperature of flange limits.

For FCS Unit 1, the minimum pressure is calculated as follows:
Minimum Pressure = (1.25 x design pressure) x 0.20
= 1.25 x 2500 psia x 0.20
= 625 psia

Therefore, when the pressure correction factors (Reference 8.19) are applied to 625 psia, the minimum pressure(s) are as follows:

Actual RCS Temperature < 210°F = 562 psi
Actual RCS Temperature > 210°F = 555.6 psi

The pressure of 562 psi is the most significant value due to the RCS can not exceed this pressure until RCS temperature is greater than the lowest service temperature value stated in Section 6 item 'c' above. The lowest service temperature is the limiting minimum temperature value and is incorporated into Figure 5-1. The heatup and cooldown limit curve is more conservative than the minimum pressure value in the temperature range specified, but the in-service hydrostatic test curve is limited by the regulatory requirement (Reference 8.20).

7. APPLICATION OF SURVEILLANCE DATA TO ART CALCULATIONS

Post-irradiation surveillance capsule test results for FCS Unit 1 are given in References 8.8 - 8.10. Additional reports containing surveillance capsule data from other nuclear power plants are located in References 8.11 - 8.13. These additional surveillance reports, along with others that are contained in Reference 8.14 (Attachment 1), were deemed credible and approved for use in the FCS surveillance program as stated by the NRC staff in Reference 8.5. Additionally, Reference 8.5 requires the following:

a) Future core loadings are limited to the core neutron leakage to values similar to those for Cycles 15 and 16 which will satisfy the requirement of end of license (August 9, 2033) fluence accumulation of 2.43×10^{19} neutrons/cm^2 to the limiting welds.
b) Caution is exercised to preclude misloading any of the peripheral assemblies which would invalidate the loading requirements.
c) New data from the Mihama Unit 1, Diablo Canyon Unit 1 and Palisades plants is assessed by the FCS staff as it becomes available, since the data from these plants were used in the FCS PTS analysis.

The use of surveillance data from these "Sister" reactor vessels (as stated in Section 7 item 'c' above) is required to ensure that FCS does not exceed PTS screening criteria during its extended lifetime (August 9, 2033).

A review of the surveillance programs of Mihama Unit 1 (12008/27204), Diablo Canyon Unit 1 (27204), Palisades Supplemental Capsules (27204), and the FCS W-275S Capsule (27204 and 12008/13253) concluded further data should be available for use in the FCS reactor vessel surveillance program as follows: (Note: The values in parentheses correspond to weld wire heat numbers.)

a) Mihama Unit 1 (Weld Wire Heat 12008/27204)

The data from Capsules 1-3 were used in Reference 8.14. The removal schedule for the remaining Mihama Unit 1 capsules as of 2000 was:

1) Capsule 4 was scheduled for removal in 2001; results are expected in 2002.
2) Capsule 5 is scheduled for removal in 2010; results are expected in 2011.
3) Capsule 6 is currently considered in standby with no scheduled removal date. Attempts to obtain additional information from KANSAI Electric Company by OPPD, MHI, and AREVA NP have not yielded any response or additional data.

b) Palisades (Weld Wire Heat 27204/27204)

The removal schedule for the Palisades capsules are:

1) Capsule SA-60-1 was pulled and evaluation data are found in internal report ATI-99-006-002 (8/4/99). The capsule report should be submitted to the NRC in 2003 or 2004. The data was used in Reference 8.14.
2) Capsule SA-240-1 was pulled and was evaluated by Framatome. A summary of the data was provided to OPPD by Palisades Staff and evaluated by Westinghouse for continued validity.

c) Diablo Canyon Unit 1 (Weld Wire Heat 27204)

The removal schedule for the Diablo Canyon Unit 1 capsules and the status of the results that are reported to the NRC are:

1) Capsule DC1-S data are contained in Reference 8.11 and was used in Reference 8.14.
2) Capsule DC1-Y data are contained in Reference 8.12 and was used in Reference 8.14.
3) Capsule DC1-V was removed in 2002 and submitted to the NRC (ML031400347). This is the last of the three original capsules containing 27204 weld material.
4) Capsule DC1-C (supplemental) and DC1-D (supplemental) were removed, but were stored in the spent fuel pool. Due to planned changes to 10CFR50.61, there are presently no plans for re-insertion or evaluation. (Note: DC1-D was fabricated using the FCS 1-410B (27204) nozzle dropout.)

R6

8. REFERENCES

8.1 NRC GL 96-03, "Relocation of Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

8.2 CE NPSD-683-A, Rev 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP Requirements from the Technical Specifications," April 2001.

8.3 WCAP-15443, Revision 0, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel," July 2000 [Contained in Letter LIC-00-0064 from OPPD (W. G. Gates) to NRC (Document Control Desk), dated August 3, 2000].

8.4 Safety Evaluation by the Office of NRR Related to Amendment Number 197 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated March 27, 2001.

8.5 Safety Evaluation by the Office of NRR Related to Amendment Number 199 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated June 6, 2001.

8.6 USAR Section 4.5.3, Revision 3, dated May 29, 2002.

8.7 TR-O-MCD-001, "Evaluation of Baseline Specimens Reactor Vessel Materials Irradiation Surveillance Program," dated March 22, 1977.

8.8 TR-O-MCM-001, Revision 1, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-225," dated August 28, 1980 [Contained in Letter LIC-81-0011 from OPPD (W.C. Jones) to NRC (H.R. Denton), dated January 23, 1981].

8.9 TR-O-MCM-002, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-265," dated March 7, 1984 [Contained in Letter LIC-84-124 from OPPD (W.C. Jones) to NRC (D.G. Eisenhut), dated April 25, 1984].

8.10 BAW-2226, "Fort Calhoun Station Unit No. 1 Evaluation of Irradiated Capsule W-275," dated July 21, 1994 [Contained in Letter LIC-94-0250 from OPPD (T.L. Patterson) to NRC (Document Control Desk), dated December 9, 1994].

8.11 WCAP-11567, "Analysis of Capsule S from the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," December 1987.

8.12 WCAP-13750, "Analysis of Capsule Y from the Pacific Gas and Electric Company Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," July 1993.

8.13 WCAP-13440, "Supplemental Reactor Vessel Radiation Surveillance Program for the Pacific Gas and Electric Company Diablo Canyon Unit No. 1," December 1992.

8.14 CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," dated July 2000 [This document is located in the Attachment 1.].

8.15 FC06876, Rev. 0, "Performance of Low Temperature Overpressure Protection System Analyses Using RELAP5: Methodology Paper."

8.16 FC06877, Rev. 3, "Low Temperature Overpressure Protection (LTOP) Analysis, Revision 1."

8.17 Safety Evaluation by the Office of NRR Related to Amendment Number 207 to Facility Operating License Number DPR-40 Omaha Public Power District Fort Calhoun Station, Unit Number 1, dated April 22, 2002.

8.18 Letter LTR-CI-01-25, Rev. 0 from WEC (S. T. Byrne) to OPPD (J. Jensen), "Assessment of Extended Beltline Limit for Fort Calhoun Station Reactor Pressure Vessel," dated December 18, 2001.

8.19 EA-FC-01-022, Rev. 2, "Pressure and Temperature Limit Curve for 40 EFPY."

8.20 10 CFR 50 Appendix G, "Fracture Toughness Requirements."

8.21 FC06863, Rev. 3, "LTOP Setpoint Instrument Loop Uncertainty and LTOP Trip Curve Development."

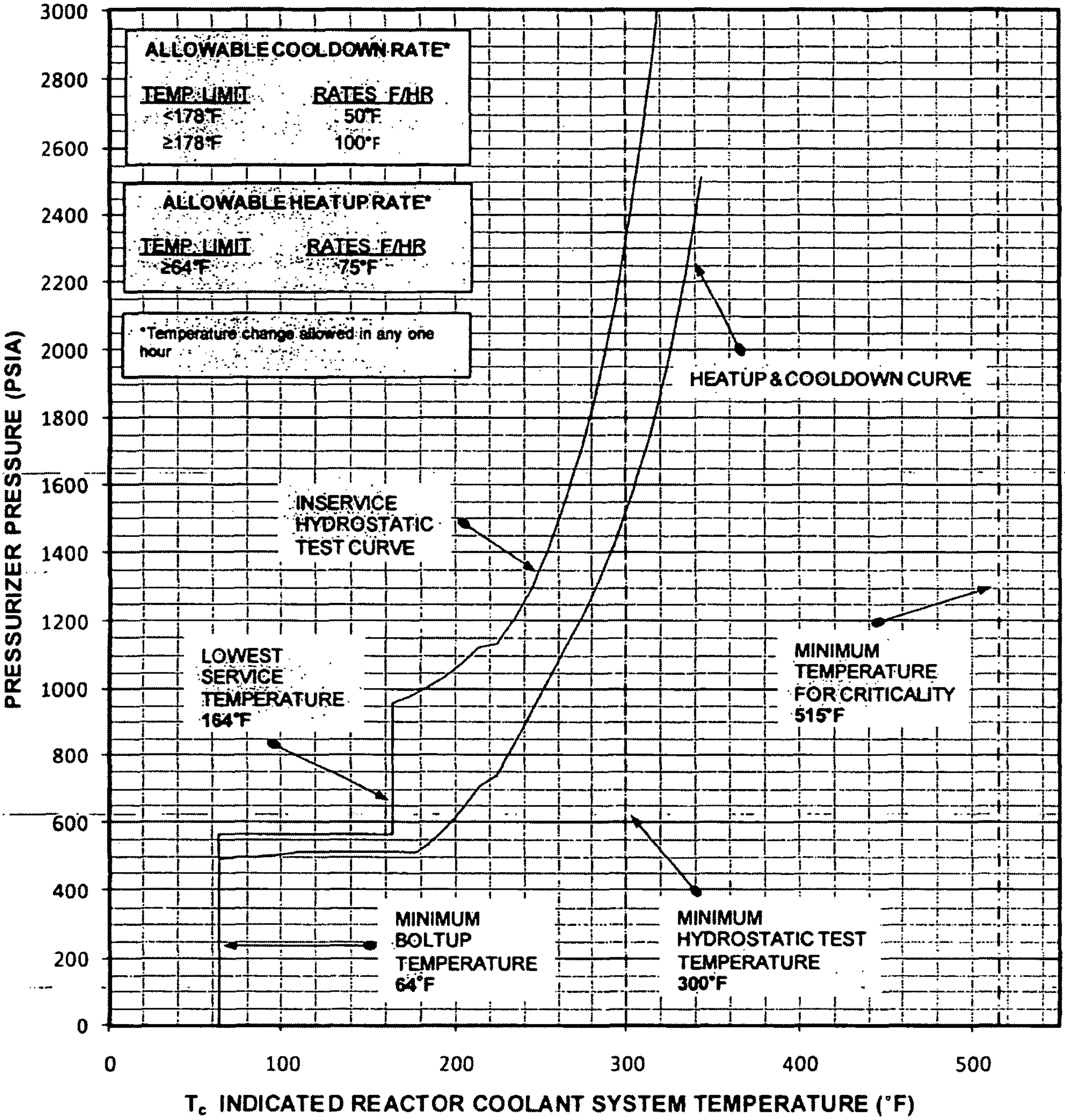
8.22 Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials."

8.23 FC06799, Rev. 0, "40 EFPY Pressure and Temperature Limit Curve Inputs."

8.24 EA-FC-02-025, Rev. 0, "Development of the RCS PTLR."

8.25 WCAP-15741, Rev. 0, "Reactor Vessel Surveillance Program Withdrawal Schedule Modifications," dated September 2001 [Contained in Letter LIC-01-0107 from OPPD (R. L. Phelps) to NRC (Document Control Desk), dated November 8, 2001].

Figure 5-1 – Fort Calhoun Station Composite P-T Limits 40 EFPY



Attachment 1 – CEN-636, Revision 2, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials," dated July 2000

Report Prepared for the
Omaha Public Power District,
Fort Calhoun Station

Final Report dated
July 19, 2000
CEN-636, Revision 02
Verification Status: Complete

Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials

Basis for Prediction of RT_{PTS}
for the Fort Calhoun RPV

Westinghouse Electric
CE Nuclear Power
Windsor, Connecticut

Evaluation of Reactor Vessel Surveillance

Data Pertinent to the Fort Calhoun

Reactor Vessel Beltline Materials

Basis for Prediction of RT_{PTS}

CEN-636, Revision 02

VERIFICATION STATUS: COMPLETE

Prepared by: Chris Hoffmann Date: 7-19-00

C. L. Hoffmann

Reviewed by: C J Gimbrone Date: 7/19/2000

C. J. Gimbrone

Approved by: Richard W Bradshaw Date: 7/19/2000

R. W. Bradshaw

Record of Revision		
No.	Date	Pages Involved
Original Issue	10/22/99	all
1	07/5/00	all
2	07/19/00	34, 35

Table of Contents

List of Tables

List of Tables (cont'd)

List of Figures

1.0 Objective

This report evaluates surveillance data to demonstrate that the Fort Calhoun reactor pressure vessel will not exceed the Pressurized Thermal Shock (PTS) screening criteria (Reference 1) through the end of the current and renewal license terms (August 9, 2013 and August 9, 2033, respectively). This evaluation is based on the use of Position 2.1 of Regulatory Guide 1.99 (Reference 2) to calculate chemistry factors for the limiting weld wire heat combinations and justify reduction of the standard deviation for shift by one-half based on credible surveillance data. The PTS screening criteria projections are based on conservative values of neutron fluence that were calculated using the methods of the U.S. Nuclear Regulatory Commission's Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". The approach used for calculating RT_{PTS} complies with 10CFR50.61(b)(3). The objective of this report is to support NRC approval of the report's conclusions.

2.0 Introduction and Background

The Fort Calhoun reactor vessel was fabricated by Combustion Engineering in Chattanooga, Tennessee during the time period 1966 to 1969. The vessel shell was fabricated using steel plates purchased to SA-533 Grade B, Class 1 requirements. The plates were joined together using automatic submerged arc welding using copper-coated electrodes. The primary coolant nozzles and the vessel flange were fabricated using forgings purchased to SA-508 Class 2 requirements. The forgings were joined to the vessel shell using automatic and manual submerged arc welding.

The reactor vessel shell, primary coolant nozzles and the vessel flange were designed to operate at high temperatures and pressures. The reactor vessel beltline materials were also designed for exposure to the fast neutrons generated in the reactor core. The material purchase specifications together with the forming, welding, and post-weld heat treatment processes were intended to provide for a high level of fracture toughness. The pre-service inspection and hydrostatic testing processes were intended to minimize the presence of fabrication-induced defects that could grow during the service lifetime. During the lifetime of the reactor vessel, periodic in-service inspections are conducted to look for defect indications in the vessel welds. In addition, a reactor vessel surveillance program is maintained throughout the life of the vessel to monitor the effect of neutron irradiation on the beltline materials.

Given the fact that the beltline welds in the Fort Calhoun vessel were fabricated using copper coated electrodes, the copper content in those welds is high (relative to vessel welds fabricated using non-copper coated electrodes). Such high copper welds have been shown to be more sensitive to the hardening effects of fast neutron irradiation than vessels fabricated during the mid- and late-1970s using non-copper coated welding electrodes. Neutron irradiation causes a reduction of the fracture toughness in the reactor vessel beltline materials. This toughness reduction is manifested as a shift in the reference temperature, RT_{NDT}, to a higher value. The shift increases as a function of the fast neutron fluence and chemical content (specifically the copper and nickel content as used in Reference 2). The magnitude of the shift is sensitive to the product form (e.g., plate or weld material).

The methodology for predicting shift that is currently acceptable to the NRC is provided in References 1 and 2. These two documents plus a handout entitled "Evaluation and Use of Surveillance Data" (Reference 3) from a November 12, 1997 NRC-Industry Meeting provide a set of NRC requirements and guidelines for using relevant and credible surveillance data to refine predictions of the shift in RT_{NDT} and calculation of the adjusted reference temperature, ART. (Values of ART, or RT_{PTS} in Reference 1, are obtained using the sum of the initial RT_{NDT}, the shift of RT_{NDT} with irradiation, and a margin term.) In the longer term, work is proceeding on the development of an improved methodology for predicting values of ART. This longer term work entails an ASTM effort to revise ASTM Standard E900 and an NRC effort to revise Regulatory Guide 1.99. A recent report on that program is NUREG/CR-6551 (Reference 4).

The approach being taken in this document is to apply Position 2.1 of Regulatory Guide 1.99 (Reference 2) using surveillance data applicable to the limiting Fort Calhoun beltline welds. (Position 2.1 provides a procedure for adjusting the chemistry factor used to predict shift and for reducing the standard deviation for shift in the margin term.) Several weld wire heats in various combinations were used in the beltline welds for the Fort Calhoun vessel. Therefore, numerous sources of surveillance data are being evaluated to give the broadest possible picture of the irradiation performance for the Fort Calhoun beltline welds. Data reviewed for applicability to Ft. Calhoun are Mihama Unit 1, Diablo Canyon Unit 1, D.C.

Cook Unit 1, Salem Unit 2, and a supplemental surveillance capsule from Palisades. Other welds that used one of the electrode heats in combination with another to produce the surveillance weld were also reviewed. These are labeled in Table 2 as "not fully applicable" to the Fort Calhoun vessel limiting beltline welds. The applicable data were then analyzed in accordance with Position 2.1, chemistry factors were calculated, and data predictability assessed. The results of this Position 2.1 analysis were then used to calculate the adjusted reference temperature, RT_{PTS}, applying the adjusted chemistry factor and the reduced standard deviation for shift from the analysis. The revised values of RT_{PTS} are being reported to the NRC in accordance with the requirements of 10CFR50.61 (b)(3).

3.0 <u>Description of Fort Calhoun Reactor Vessel Beltline Materials</u>

The Fort Calhoun reactor vessel beltline materials and surveillance materials are described in Table 1. The first column gives the plate code or the weld seam identification. The second column gives the heat number for the plate or welding electrode. The third column gives the flux type and lot number for the welds. The fourth column gives the chemistry factor based on the best estimate copper and nickel content. (The material identification and the weld chemistry factor values are from Reference 5.)

The Fort Calhoun beltline consists of the intermediate and lower shell courses of the reactor vessel. Plates D-4802-1, D-4802-2, and D-4802-3 comprise the intermediate shell course. Plates D-4812-1, D-4812-2, and D-4812-3 comprise the lower shell course. The plates and shell courses were joined together using automatic submerged arc welding using Mil B4 copper coated electrodes and Linde 1092 or Linde 124 flux. Weld seams 2-410 A/C (where "A/C" means seams A, B, and C) are the axial welds between the plates to form the intermediate shell. Weld seams 3-410 A/C are the axial welds between the plates to form the lower shell. Weld seam 9-410 is the circumferential weld between the intermediate and lower shell course. Weld seams 2-410 A/C and 9-410 were deposited using the single arc process. Weld seams 3-410 A/C were deposited using the tandem arc process.

Table 1 also provides a description of the Fort Calhoun surveillance program plate and weld material. The surveillance plate was obtained from plate D-4802-2. The surveillance weld was fabricated using the same welding process as was used for weld seam 9-410 but with a different heat of wire.

The beltline materials are evaluated using Reference 2 to identify the limiting material at end of the license period. The limiting material is the beltline plate or weld with the highest RT_{PTS} value. The limiting materials in the Fort Calhoun vessel beltline are from the lower shell course welds. As stated in the Introduction, the objective of this evaluation is to apply Position 2.1 of Reference 2 to surveillance data that are applicable to the limiting material, the lower shell course welds. The results of this Position 2.1 analysis can then be used to calculate the adjusted reference temperature, RT_{PTS}, at the end of the license period applying the adjusted chemistry factor and the reduced standard deviation for shift from the analysis.

4.0 Description of Surveillance Data Relevant to Fort Calhoun

In Table 1, the weld wires used to fabricate the lower shell course welds (3-410 A/C) in the Fort Calhoun vessel were identified as heat numbers 12008, 13253, and 27204. The approach taken was to match up those heats or combination of heats with those used to fabricate the surveillance welds in other reactor vessels manufactured by Combustion Engineering during a similar period of time.

The surveillance weld matches are identified in Table 2. A match is defined as having the same heat number in the surveillance weld as is in one of the welds in Table 1. In the case of a mixture of heats in the surveillance weld or Fort Calhoun beltline weld, at least one of the two heats in the mixture had to match. The matches are based on CEOG Report CE NPSD-1119 (Reference 6) and similarly developed sources. (In all the matches cited, the traceability of the surveillance weld wire heat was established based on fabrication records as stated in Reference 6.) Data from five PWR surveillance programs (References 7 through 18) were identified as likely sources of information relative to the three heats from the Fort Calhoun weld seam 3-410 A/C. Data determined to be applicable to Fort Calhoun are Mihama Unit 1, Diablo Canyon Unit 1, the weld from the Palisades supplemental surveillance program, the supplemental surveillance capsule for Fort Calhoun, Salem Unit 2, and D.C. Cook Unit 1. Data from three BWR surveillance programs were also identified using Reference 6. Only the Fitzpatrick weld was fully representative of the weld wire heats used in weld seam 3-410 A/C. The remaining two BWR welds were either a mixture or were representative of another weld (9-410). Analysis of the Fitzpatrick surveillance weld was not done given the limited number of measurements and the uncertainty regarding the effects of differences in irradiation environment between a BWR and the Fort Calhoun PWR vessel.

The data from four of the five PWR surveillance programs and from the Fort Calhoun surveillance program were compiled from the database assembled for the previously cited ASTM E900 effort (Reference 4). That database had been reviewed, updated and augmented by knowledgeable individuals from the Industry and, therefore, provides a credible source of information for each surveillance program. In addition the individual post-irradiation test reports were reviewed to the extent possible to assess the reasonableness of the data updates. The data from the Mihama Unit 1 surveillance program were obtained through a proprietary agreement between Kansai Electric Power Company and the Omaha Public Power District. [Note: Only the non-proprietary data are presented in this report.]

The surveillance program data sets are provided in Tables 3 through 6. The Fort Calhoun surveillance data (References 19 through 21) are provided in Tables 8A, 8B and 8C. Each table contains the surveillance capsule identity, the measured shift, the reported neutron fluence, and the irradiation temperature. [Note: The irradiation temperature for the surveillance specimens was taken as that of the reactor coolant cold leg. The temperatures were obtained from the E900 database and from Kansai for Mihama Unit 1.]

5.0 Regulatory Position 2.1 Analysis of Relevant Surveillance Data

The objective of this section is to analyze the surveillance data in accordance with Position 2.1 of Reference 2. The Position 2.1 analysis will be augmented using the guidance provided by the NRC (Reference 3). The guidance provides a set of NRC review requirements and guidelines for using relevant and credible surveillance data from other reactor vessels to refine predictions of the shift in RT_{NDT} and calculation of the adjusted reference temperature, RT_{PTS}. Position 2.1 of Regulatory Guide 1.99 is applied to available surveillance data that were identified in the preceding section as relevant to the beltline welds in the Fort Calhoun vessel.

5.1 Credibility of Surveillance Data:

Regulatory Guide 1.99 presents five credibility criteria by which surveillance data from a given reactor are judged before the surveillance data can be used in place of Regulatory Position 1. The five criteria are discussed in turn below:

Criterion 1: "Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement according to the recommendations of this guide."

The chemistry factors for each of the three beltline welds (determined using Table 1 of Reference 2) range from 89 °F to 231 °F. [Note: The highest chemistry factor for the beltline plates is less than the lowest beltline weld, 89 °F. Therefore, the beltline plates will not limit vessel operation and are excluded from the subsequent discussion.] The surveillance weld was fabricated using weld wire heat 305414 with Linde 1092 flux lots #3947 and #3951. It was made from different welding consumables than those used for the Fort Calhoun beltline welds. The surveillance weld is representative of but not identical to the beltline welds, so it does not meet Criterion 1. Therefore, it can not be used in a Position 2.1 analysis of the Fort Calhoun beltline welds. The focus of this report is on the use of data from surveillance welds that were fabricated using the same weld wire heats as were used in the Fort Calhoun vessel limiting beltline weld; i.e., surveillance weld data that meet Criterion 1 for the Fort Calhoun beltline welds. The surveillance program welds listed in Table 2 include most of the weld heats listed in Table 1. The one not represented at all, weld wire heat #51989, has a chemistry factor of 89 °F and thus is not a controlling beltline weld. The surveillance welds in Table 2 include the individual heats of controlling beltline weld materials and, therefore, satisfy the first criterion for the most limiting combinations of weld wire heats.

Criterion 2: "Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30-foot-pound temperature and the upper-shelf energy unambiguously."

As part of the effort to review the surveillance data for the ASTM E900 effort, all of the data were computer curve fit by Modeling and Computing Services as part of an effort sponsored by the U.S. Nuclear Regulatory Commission (Reference 4). The computer curve fit results (index temperature and transition temperature shift) were used for the E900 effort and reported in that database. Therefore, the individual test results for the materials data applied from Table 2 exhibited behavior consistent with pressure vessel materials, scatter was well within expected ranges, and there were no difficulties experienced in deriving the 30 foot-pound temperature. The second criterion is satisfied.

Criterion 3: "When there are two or more sets of surveillance data from one reactor, the scatter of RT_{NDT} shift values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28 °F for welds and 17 °F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter shall not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82."

The weld metal shift measurements for the materials were evaluated individually against this criterion in Tables 3 through 6 and in Table 8. The results of that evaluation are provided in Section 5.4. In all but one case (Cook Unit 1), the data scatter criterion was satisfied. [The November 1997 Guidelines (Reference 3) expanded on the use of this criterion. Those guidelines were taken into consideration in this report.]

Criterion 4: "The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within ± 25°F."

This criterion could not be addressed using temperature monitor data because there was an inconsistent use of monitors among the various surveillance programs. However, both NRC guidance (Reference 3) and the NRC sponsored work (Reference 4) used the reactor coolant inlet temperatures as a best estimate for the irradiation temperature of the Charpy specimens in the capsule. Implicit in the NRC sponsored approach is the assumption that Criterion 4 will be met. It is based on the premise that the reactor coolant will cool the vessel wall and the adjacent surveillance specimens the same. In the data analysis that follows, the reactor coolant inlet temperatures from the ASTM E900 database (Reference 4) were used to provide an estimate of the temperature of the Charpy specimens, and the differences in irradiation temperature were treated explicitly. Thus Criterion 4 is satisfied.

Criterion 5: "The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material."

There are limited sets of correlation monitor material (termed standard reference material in the Fort Calhoun vessel) data from the various surveillance capsules. For Fort Calhoun, the correlation monitor material measurements were addressed in Reference 20. For the other surveillance data, no such analysis could be performed. Therefore, the Fort Calhoun correlation monitor material measurements satisfy Criterion 5.

In summary, the surveillance data are shown to satisfy the criteria above. The data are assessed individually for Criteria 3 and 4 in Section 5.4, Analysis of Surveillance Data. The plant specific Fort Calhoun surveillance data are assessed for Criterion 5 also in Section 5.4. Therefore, the surveillance data are acceptable for use with Position 2.1 of Regulatory Guide 1.99, Revision 2.

5.2 Traceability of Mihama 1 Surveillance Data

In the specific case of the Mihama Unit 1 surveillance program, foreign data from a Westinghouse designed Pressurized Water Reactor (PWR) are being applied to a domestic Combustion Engineering designed PWR. In order to establish that the weld surveillance data from the Mihama Unit 1 reactor vessel are applicable to the Fort Calhoun vessel, the following information was evaluated: a. Unirradiated and irradiated Charpy data for tandem weld wire heat 12008/27204; b. Irradiation temperature of the capsule based on PWR cold leg; c. Neutron flux of capsules; d. Gamma heating of capsules; e. Neutron spectrum of capsules; and f. Chemistry of surveillance data.

Each of these items is addressed below:

a. Unirradiated and irradiated Charpy data for tandem weld wire heat 12008/27204

The individual Charpy specimen data for the unirradiated tandem weld wire heat 12008/27204 are provided in Table 2 of Reference 15. Those data were used to establish the unirradiated Charpy curve. The individual Charpy specimen data for the irradiated tandem weld wire heat 12008/27204 were obtained from Kansai (Reference 17) and were used to establish the irradiated Charpy curve. Those data were checked against the Charpy index temperatures cited by Kansai in Reference 16 for the Charpy shift values from each of the three surveillance capsules (V, R and S per Reference 15) and shown to be consistent.

b. Irradiation temperature of the capsule based on PWR cold leg-

Kansai reported a value of 289 °C (552 °F) for the Mihama Unit 1 cold leg temperature (Reference 16). In an evaluation of the capsule configuration (Reference 22), it has been confirmed that that temperature is reasonable for similarly configured reactor vessels designed by Westinghouse.

c. Neutron flux of capsules-

The neutron flux corresponding to each irradiated and tested capsule from Mihama Unit 1 was reported by Kansai in Reference 17 together with their source reference and a description of the methodology used to calculate the neutron flux. In Reference 22, it has been confirmed that the reported flux is reasonable for similarly configured reactor vessels designed by Westinghouse.

d. Gamma heating of capsules-

In Reference 22, Westinghouse has confirmed that the design and construction of the Mihama Unit 1 surveillance capsules are the same as that for other surveillance capsules that they fabricated during this timeframe. Therefore, it is reasonable to conclude that the gamma heating in the Mihama Unit 1 surveillance capsules is the same as that in similar domestic Westinghouse capsules.

e. Neutron spectrum of capsules-

In a CEOG sponsored program (Reference 23) it was demonstrated that surveillance data applicable to Combustion Engineering fabricated reactor vessel materials were equally predictable using Regulatory Guide 1.99, Revision 2 for plants designed by both Westinghouse and Combustion Engineering. It was concluded from this that the irradiation environment was similar for the surveillance capsules from Westinghouse and Combustion Engineering plants. There was no definitive difference between the spectra such that one needs only to consider differences in the irradiation temperature and the neutron flux. Neutron spectrum was considered to be no more than a second order variable for embrittlement. (For example, embrittlement correlation development work reported in Reference 4 did not identify neutron spectrum as an independent or dependent variable.)

In Reference 24 no discernible differences were found between the neutron spectra for the surveillance capsules from Westinghouse and Combustion Engineering plants. Reference 22 confirmed that the Mihama Unit 1 neutron spectrum is comparable to domestic Westinghouse PWRs. Therefore, the neutron spectra in the Mihama Unit 1 surveillance capsules is not expected to adversely affect the application of those surveillance data to the Fort Calhoun vessel.

f. Chemistry of surveillance data-

Kansai reported copper and nickel contents of 0.19 and 1.08 w/o for the Mihama Unit 1 surveillance weld (Reference 16). Weld analyses by Combustion Engineering and the best estimate for the weld (Reference 6) for heat 12008 and 27204 yielded copper and nickel contents as follows:

WDC-351	(n/a) Cu	0.98 Ni
WDC-1817	0.19 Cu	0.98 Ni
Best estimate	0.219 Cu	0.996 Ni

The Kansai values are fully consistent with a weld deposit made using heats 12008 and 27204. Traceability of the Mihama Unit 1 surveillance weld has been established based on fabrication records from CE-Chattanooga.

5.3 Analysis Approach

The analysis in the following section utilizes the ratio method of Reference 2. The ratio method is based on the relative chemistry factors. Regulatory Guide 1.99 (Reference 2) states that, "if there is clear evidence" of a difference in copper and nickel content, the measured shift should be adjusted by multiplying by the ratio of the chemistry factors for the vessel weld to that of the surveillance weld (i.e., the ratio method). For this evaluation, the ratio method was used to adjust the surveillance data from other programs to the best estimate chemistry for the Fort Calhoun reactor vessel. (This was done whether or not the copper and nickel contents were significantly different.) References 5 and 6 were used to obtain best estimate copper and nickel contents for the weld wire heats so that chemistry factors could be computed for the Fort Calhoun welds.

The effect of differences in the neutron irradiation environment is considered when applying surveillance data from another reactor pressure vessel. These differences have been addressed by the Combustion Engineering Owners Group, BGE, and Duke Power (see References 23, 24, and 25, respectively). The effect of neutron irradiation environment is taken to mean changes in measured transition temperature shift caused by differences in irradiation temperature, neutron flux and neutron energy spectrum. For the BGE and Duke evaluations (References 24 and 25), there was no expected influence of neutron flux or neutron energy spectrum given the use of only PWR surveillance data. The actual values of neutron flux and neutron energy spectrum were compared for the various plants being considered, and the values were within expected ranges for which no difference in irradiation behavior would be expected. The Duke evaluation entailed the comparison of data from two Westinghouse designed reactor vessels. The BGE evaluation entailed comparisons of data from a Combustion Engineering and a Westinghouse designed reactor vessel. For the CEOG evaluation (Reference 23), a statistical analysis of surveillance data from both Combustion Engineering and Westinghouse designed reactor vessels demonstrated that there was no significant effect of differences in the irradiation environment for vessel materials fabricated by Combustion Engineering. In this report, data from the Combustion Engineering and Westinghouse vessel designs were considered in the analysis. Therefore, prior work suggests that there is no significant effect of neutron flux and neutron energy spectrum expected relative to the results in Table 7.

The effect of irradiation temperature was explicitly considered in the BGE evaluation (Reference 24) using the rationale stated in Reference 3. That rationale assumes there is a 1.0 °F effect on the chemistry factor for each 1.0 °F difference in irradiation temperature. (The higher the irradiation temperature, the lower the chemistry factor would be, and vice versa, per Reference 3. Irradiation temperature is taken as the reactor coolant inlet temperature.) The analysis in the following sections utilizes a modified approach from that given in Reference 3 for adjusting surveillance data for differences in irradiation temperature. A description of the rationale and benefits for the ratio and T_{cold} adjustments for analysis of surveillance data follows.

The rationale and benefits of this approach were described at a March 13, 2000 meeting between the NRC and the Omaha Public Power District in regard to the application of Position 2.1 of Regulatory Guide 1.99, Revision 2 to two heats of surveillance welds applicable to the Fort Calhoun vessel. The chemistry factor calculation has traditionally been done by the NRC as described in Reference 3. However, in order to analyze surveillance data from two separate programs it was necessary to first adjust for both CF differences and T_{cold} differences. Two issues were considered. The first is the viability of the T_{cold} adjustment method. The second is the appropriateness of adjusting the data prior to performing the data scatter analysis.

a) Viability of the T_{cold} Adjustment Method - In November 1997, the NRC presented a set of guidelines (Reference 3) to the industry that supplemented the guidelines contained in Regulatory Guide 1.99, Revision 02. The activities surrounding Generic Letter 92-01 and its antecedents prompted the need for the supplemental guidelines. That Generic Letter had addressed some of the material variability issues including copper and initial RT_{NDT} and the effect of irradiation temperature on the degree of embrittlement. In the November 1997 NRC-Industry meeting, the NRC presented ways they considered acceptable to treat each aspect:

The "ratio method" was the prescribed way to treat differences in the copper and nickel content between the surveillance program weld being analyzed and the best estimate for the vessel weld.

The use of the σ_i term was the prescribed way to treat variability in initial RT_{NDT}. A value of σ_i = 17 °F was assigned for use with the generic initial RT_{NDT} = -56 °F for welds fabricated by Combustion Engineering. A value of σ_i = 0 °F was assigned for use with a measured initial RT_{NDT} (just as is the case for plates and consistent with the practice for welds).

Position 2.1 of Reference 2 was the prescribed way to analyze surveillance data to derive a chemistry factor (CF) using two or more sets of credible data. The data are to be adjusted for chemistry differences using the ratio method. If the difference between the adjusted measured shift and the predicted shift using the derived CF is less than or equal to σ_Δ = 28 °F, data scatter is deemed acceptable and the derived CF as well as a reduced σ_Δ (28/2 = 14 °F) could be used for predicting future embrittlement of the vessel beltline weld.

The effect of irradiation temperature on the degree of embrittlement was considered initially in the credibility criteria for use of surveillance data (the capsule temperature was to be within 25 °F of the vessel wall) and in November 1997 in a post-CF derivation adjustment to the CF. The initial accounting was done to satisfy the applicability issue; i.e., for irradiation temperatures between 525 °F and 590 °F, the Regulatory Guide 1.99, Revision 02 embrittlement correlation was applicable without adjustment. The adjustment suggested in November 1997 was done to satisfy the NRC concern that the irradiation temperature of the surveillance capsule in plant "X" was at a higher temperatures than that of vessel "Y" to which the data were to be applied. It was widely believed that higher irradiation temperatures would result in less shift than at lower irradiation temperatures. The "rule-of-thumb" was that the effect was on the order of 1.0 °F increase/decrease in shift for each 1.0 °F difference in irradiation temperature.

At the March 13, 2000 meeting a method was presented for making the T_{cold} adjustment at the same time as was done for the ratio method. The approach followed was to use the recommended equation from NUREG/CR-6551 (Reference 4) to adjust the data for the effect of irradiation temperature differences. The method used was to compute the predicted shift at both temperatures of interest. The temperature effect is then the difference in the two shifts that is added to or subtracted from the measured shift, whichever is appropriate.

The equation in Reference 4 takes into consideration both time and temperature in the computation, thus providing a more rigorous treatment than that afforded by the rule-of-thumb given in Reference 3. It also offers the benefit of the numerical analysis of 609 data points for defining the apparent effect of irradiation temperature differences. (That is, the coefficients for temperature, copper, etc., were developed from the data and refined by statistical analysis.) Finally, use of the recommended equation from Reference 4 to adjust the data <u>before</u> the sum-of-the-squares analysis is mathematically more desirable than making the rule-of-thumb adjustment after the sum-of-the-squares analysis. (The Position 2.1 analysis approach was specifically designed to give more weight to the surveillance data at the higher fluences in recognition of the fact that the higher fluence data were more indicative of the expected behavior than were the low fluence data. Adjusting the data for temperature differences after the sum-of-the-squares analysis would not provide the same significance weighting. The Reference 3 guidelines approach, therefore, diminishes the significance of the effect of temperature on the high fluence data which is in conflict with the intent of the Position 2.1 analysis approach.)

The approach described above fully adjusts the data for both of the Reference 3 issues. Those are the chemistry differences (i.e., using the ratio method) and the T_{cold} differences. The shift measurements are adjusted prior to deriving the chemistry factor and prior to analyzing the scatter in the data.

b) Appropriateness of Data Adjustment Prior to Data Scatter Analysis - The third credibility criterion of Regulatory Guide 1.99, Revision 02 is to ascertain that the scatter of the surveillance measurements about a best-fit line derived using Position 2.1 is no more than 28 °F for welds. If this can be shown, then the derived chemistry factor can be used together with a reduced value for prediction uncertainty ($\sigma_\Delta/2$ =

14 °F). The concept is that the availability of credible measurements from the surveillance program greatly reduces the uncertainty of the prediction, and the lack of significant data scatter demonstrates that the material itself is not anomalous. In other words, the weld material is adequately represented by the embrittlement correlation contained in Regulatory Guide 1.99, Revision 02.

The applicability of the irradiation temperature adjustment depends on the source of the data. In using Position 2.1 to evaluate plant-specific surveillance data, the only data adjustment necessary is for the chemistry difference using the ratio method (if there is a significant difference between the surveillance weld and the vessel weld). There is no need to adjust for irradiation temperature because the capsule temperature and the cold leg temperature are essentially the same (i.e., it is the same vessel).

In using Position 2.1 to evaluate surveillance data from another plant, both the ratio method and irradiation temperature adjustments must be considered. The Reference 3 guidance is to adjust the shift measurements by the ratio method, calculate the CF, and then adjust the derived CF for temperature differences. The analysis of data scatter is done on the ratio adjusted data, so it is not examining the scatter of the original measurements. The Reference 3 approach provides a temperature adjustment but is done without regard to the time dependence of the presumed temperature effect. In using Position 2.1 to evaluate surveillance data from two other plants, both the ratio method and irradiation temperature adjustments must be considered, and they need to be done prior to the sum-of-the-squares analysis. Doing the analysis on data adjusted for both the ratio method and irradiation temperature accounts for the time dependence of the presumed temperature effect and permits the sum-of-the-squares analysis emphasis on the high fluence data. Doing the analysis without both initial adjustments coupled with the subsequent correction for a temperature effect is inconsistent with the intent of Position 2.1 and places an unrealistic burden on the user to demonstrate the data scatter criterion is met.

c) Illustration of the T_{cold} Adjustment Method - The Position 2.1 analyses were run two ways as shown in Tables 4A, 4B, 6A and 6B. Tables 4A and 6A give the derivation for each surveillance set of CF based on the fully adjusted numbers (i.e., for both CF and T_{cold} differences). Tables 4B and 6B give the derivation for each surveillance set of CF based on the numbers adjusted for CF, followed by the Reference 3 suggested approach to address T_{cold} differences.

For the Mihama 1 surveillance data analysis, Tables 6A and 6B, the derived CFs for weld wire heats 12008 with 27204 were as follows:

> $CF_{TCOLD+CF}$ = 206.6 °F based on shifts adjusted for FCS T_{cold} (543 °F) and best estimate chemistry (Table 6A)
>
> CF= 200.9 °F based on shifts adjusted for best estimate chemistry, and CF_{TCOLD}= 209.9 °F after adjustment for FCS T_{cold} (i.e., 552 °F – 543 °F= 9°F adjustment) (Table 6B)

Therefore, in the case of the Mihama 1 surveillance data, the difference in the derived CFs is small (3.3 °F), but the CF is larger using the rule-of-thumb approach of temperature adjustment. The data scatter is identical for each because the adjustments used were the same in each case.

For the Diablo Canyon 1 surveillance plus the Palisades supplemental capsule data analysis, Tables 4A and 4B, the derived CFs for weld wire heat 27204 (tandem) were as follows:

> $CF_{TCOLD+CF}$ = 215.5 °F based on shifts adjusted for FCS T_{cold} (543 °F) and best estimate chemistry (Table 4A)
>
> CF= 220.2 °F based on shifts adjusted for best estimate chemistry, and CF_{TCOLD}= 210.2°F after adjustment for FCS T_{cold} (i.e., 543 °F – 533 °F= 10 °F adjustment) (Table 4B)

The 10°F temperature difference corresponds to the data with the highest fluence exposure because that data has the greatest significance to the CF derivation. For the weld wire heat 27204 surveillance data, the difference in the two derived CFs is small (5.3 °F), but the CF obtained using the rule-of-thumb approach of temperature adjustment is smaller than the CF derived from the fully adjusted data.

The data scatter criterion is met in the case of the CF derived using the fully adjusted data. This is justified because the analysis entails the use of data from two different vessels and three unique T_{cold} values. It would be unreasonable to expect test results that are presumed sensitive to irradiation temperature to be predictable without first removing the bias due to irradiation temperature. As was expected, the data scatter criterion was not met with the data that were corrected only for CF differences.

This method of analyzing surveillance data using both a chemistry factor and irradiation temperature adjustment is seen to result in comparable values to those obtained using the NRC guidelines in Reference 3. Use of the NRC guidelines resulted in a larger adjustment (positive or negative) in the two cases considered because that approach does not take into account time-at-temperature. The approach using the fully adjusted data provides the capability to analyze data irradiated at multiple temperatures.

5.4 Surveillance Data Analysis

D.C. Cook Unit 1- The Cook surveillance weld was fabricated using weld wire heat 13253 (Reference 6). The chemistry factors for the Cook surveillance weld and the Fort Calhoun vessel weld are 206.4 °F and 189.05 °F, respectively. The Cook shift measurements in Table 3 (References 7 through 9) were adjusted for chemistry factor differences using the ratio 189.1 °F /206.4 °F= 0.916. The shifts were adjusted to the Fort Calhoun irradiation temperature, 543 °F, using the approach outlined in the preceding section. The computed adjustments were –3.2 °F, -5.1 °F, -6.1 °F, and –7.2 °F for capsule T, X, Y and U, respectively. The fully adjusted shift measurements are shown in Table 3.

The chemistry factor derived based on the four capsule results is 116.9 °F. The predicted shifts based on this chemistry factor were compared to the adjusted Charpy shifts. The adjusted minus predicted shifts for capsules Y and U are well in excess of σ_{Δ} for welds (28 °F). The chemistry factor was re-derived based on three capsule results, where capsule U was excluded because it was the most overpredicted value. The resultant chemistry factor value based on capsules T, X and Y is 137.4 °F, which is higher than the chemistry factor value based on all four capsules. The adjusted minus predicted shifts for those three capsules are within σ_Δ for welds (28 °F). The adjusted minus predicted shift for capsule U is greater than σ_Δ but is negative (i.e., conservative). Therefore, the Cook Unit 1 surveillance data are predictable when the capsule U results are excluded. The derived chemistry factor of 137.4 °F is much lower than the values for the surveillance weld (206.4 °F) from Table 1 and for the Fort Calhoun vessel weld (189.05 °F).

Diablo Canyon Unit 1- The Diablo Canyon surveillance weld was fabricated using weld wire heat 27204 (Reference 6). The chemistry factors for the Diablo Canyon surveillance weld and the Fort Calhoun vessel weld are 221.8 °F and 226.81 °F, respectively. The analysis included the use of data for weld heat 27204 irradiated in the Palisades reactor vessel in a supplemental capsule. The chemistry factor for the Palisades supplemental surveillance weld is 229.04 °F. The Diablo Canyon (References 10 and 11) and Palisades (Reference 18) shift measurements in Table 4 were adjusted for chemistry factor differences using the ratio 226.81°F /221.8°F= 1.022 for the Diablo Canyon data and 226.81 °F/229.04 °F = 0.990 for the Palisades data. The shifts were adjusted to the Fort Calhoun irradiation temperature, 543 °F, using the approach outlined in the preceding section. The computed adjustments were –1.6 °F, -2.0 °F, and –9.0 °F for capsules S and Y from Diablo Canyon and for capsule SA-60-1 for Palisades, respectively. The fully adjusted shift measurements are shown in Table 4A. A comparative analysis is provided in Table 4B in which the shift measurements were adjusted only for the chemistry factor differences.

The chemistry factor derived in Table 4A based on the three capsule results is 215.5 °F. The predicted shifts based on this chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shifts for the three capsules are all less than σ_Δ. The chemistry factor derived in Table 4B based on the three capsule results is 220.2 °F before adjusting for irradiation temperature differences. The adjusted chemistry factor is 210.2 °F using the guidelines of Reference 3. The predicted shifts based on the Table 4B chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shift for capsule S (fluence of 2.84E18 n/cm^2) is in excess of σ_Δ for welds (28 °F), but the difference is negative (i.e., conservative). The derived chemistry factors of 215.5 and 220.2 °F are slightly lower than the values for the surveillance welds (221.8°F and 229.04 °F) from Table 1 and for the Fort Calhoun vessel weld (226.81 °F). The weld heat 27204 surveillance data are predictable when the data are fully adjusted to account for the differences in both chemical content and irradiation temperature.

Salem Unit 2- The Salem surveillance weld was fabricated using weld wire heat 13253 (Reference 6). The chemistry factors for the Salem surveillance weld and the Fort Calhoun vessel weld are 198.1 °F and 189.05 °F, respectively. The Salem shift measurements in Table 5 (References 12 through 14) were adjusted for chemistry factor differences using the ratio 189.1 °F /198 °F= 0.955. The shifts were adjusted to the Fort Calhoun irradiation temperature, 543 °F, using the approach outlined previously. The computed adjustments were –1.7 °F, -2.2 °F, and –3.0 °F for capsules T, U, and X, respectively. The fully adjusted shift measurements are shown in Table 5.

The chemistry factor derived in Table 5 based on the three capsule results is 190.4°F. The predicted shifts based on this chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shifts for the three capsules are all less than σ_Δ. The derived chemistry factor of 190.4 °F is very similar to the values for the surveillance weld (198.1 °F) from Table 1 and for the Fort Calhoun vessel weld (189.05 °F). Therefore, the Salem Unit 2 surveillance data are predictable.

Mihama Unit 1- The Mihama Unit 1 surveillance weld was fabricated using weld wire heats 12008 and 27204. The chemistry factors for the Mihama surveillance weld and the Fort Calhoun vessel weld are 227.2 °F and 231.06 °F, respectively. The Mihama shift measurements in Table 6 (Reference 16) were adjusted for chemistry factor differences using the ratio 231.06 °F /227.2 °F= 1.017. The shifts were adjusted to the Fort Calhoun irradiation temperature, 543 °F, using the approach outlined in the preceding section. The computed adjustments were +4.3 °F, +5.3 °F, and +7.4 °F for capsules 1, 2 and 3, respectively. The fully adjusted shift measurements are shown in Table 6A. A comparative analysis is provided in Table 6B in which the shift measurements were adjusted only for the chemistry factor differences.

The chemistry factor derived in Table 6A based on the three capsule results is 206.6 °F. The predicted shifts based on this chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shifts for the three capsules are all less than σ_Δ. The chemistry factor derived in Table 6B based on the three capsule results is 200.9 °F before adjusting for irradiation temperature differences. The adjusted chemistry factor is 209.9 °F using the guidelines of Reference 3. The predicted shifts based on the Table 6B chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shifts for the three capsules are all less than σ_Δ. The derived chemistry factors of 206.6 and 209.9 °F are lower than the values for the surveillance weld (227.2 °F) from Table 1 and for the Fort Calhoun vessel weld (231.06 °F). The Mihama surveillance data are predictable when the data are fully adjusted or partially adjusted to account for the differences in both chemical content and irradiation temperature.

Fort Calhoun - The Fort Calhoun surveillance weld was fabricated using weld wire heat 305414 (Reference 6). The chemistry factor for the Fort Calhoun surveillance weld is 212 °F. The shift measurements in Tables 8A, 8B and 8C are from References 19 through 21). No chemistry factor adjustment was made because the data are not being related to any vessel weld. The data are being used only to assess predictability of the Fort Calhoun surveillance weld data.

The chemistry factor derived in Table 8A based on the three capsule results is 229.0 °F. The predicted shifts based on this chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shifts for the three capsules are all less than σ_Δ. Therefore, the Fort Calhoun weld surveillance data are predictable. The derived chemistry factor of 229.0 °F is higher than the value for the surveillance weld (212 °F) in Table 1.

The Fort Calhoun surveillance plate was fabricated using heat A1768-1. The chemistry factor for the Fort Calhoun plate is 65 °F based on Table 2 of Reference 2). No chemistry factor adjustment was made because there is no difference between the surveillance plate and the vessel plate chemistry. The data are being used to assess the predictability of the Fort Calhoun surveillance plate data.

The chemistry factor derived in Table 8B for the surveillance plate based on the three capsule results (where the longitudinal and transverse measurements were combined) is 72.0 °F. The predicted shifts based on this chemistry factor were compared to the measured Charpy shifts. The measured minus predicted shifts for the five measurements are all less than σ_Δ. Therefore, the Fort Calhoun plate surveillance data are predictable. The derived chemistry factor of 72.0 °F is similar to the Table 2 value (65 °F).

The standard reference material in the Fort Calhoun surveillance program was from HSST Plate 01. The chemistry factor for the plate is 131.7 °F using the reported chemical content from the E900 database with Table 2 of Reference 2. No chemistry factor adjustment was made because there is no corresponding vessel plate chemistry. The data are being used to assess the predictability of the Fort Calhoun standard reference material data.

The chemistry factor derived in Table 8C for the standard reference material based on the two capsule results is 138.3 °F. The predicted shifts based on this chemistry factor were compared to the measured Charpy shifts. [Note: This exceeds the requirements of Regulatory Guide 1.99, Revision 2, Criterion 5 in which it is necessary only to show the data are within the scatterband of available measurements.] The measured minus predicted shifts for the two measurements are both less than σ_Δ. The derived chemistry factor of 138.3 °F is similar to the Table 2 value (131.7 °F). Therefore, the Fort Calhoun standard reference material data are predictable.

6.0 Evaluation of Surveillance Data Credibility and Applicability to Fort Calhoun

The results of the preceding analysis are summarized in Tables 7 and 9. The derived chemistry factors are provided in Table 7 for each of the surveillance program welds that are applicable to the Fort Calhoun beltline welds. The derived values correspond to the best estimate chemistry for the weld wire heat(s) used to fabricate the surveillance program welds. The ratio method was applied to adjust the chemistry of the specific surveillance program weld to the best estimate chemistry for the vessel weld. Also shown in Table 7 are the chemistry factors obtained using Table 1 of Reference 2 for the surveillance weld and the best estimate chemistry for the weld wire heat.

All of the surveillance materials analyzed in Tables 3 through 6 are credible with respect to being applicable to the limiting materials in the Fort Calhoun reactor vessel beltline. This applicability is with respect to weld wire heat number, welding flux type, and welding process. Any differences in copper and nickel content between a surveillance weld and the Fort Calhoun reactor vessel beltline weld with the same weld wire heat(s) were addressed through use of the ratio method in accordance with Reference 2. Any difference in irradiation temperature between the surveillance weld and the Fort Calhoun reactor vessel beltline weld was addressed through use of the T_{cold} adjustment method described in Section 5.3. The data were evaluated for scatter using the criterion that the surveillance measurements were to be predictable within one σ_Δ of the predicted shift using the derived chemistry factor in accordance with Reference 2.

In the case of heat 13253 from D.C. Cook Unit 1, Table 3, there are measurements from four surveillance capsules. The high fluence measurement, capsule U, is significantly overpredicted. The derived chemistry factor based on capsules T, X, and Y from D.C. Cook Unit 1 is 137.4 °F. In the case of heat 13253 from Salem Unit 2, Table 6, all three measurements are predictable within one σ_Δ but the derived chemistry factor (190.4 °F) is higher than obtained from the D.C. Cook Unit 1 data (137.4 °F). Therefore, a conservative chemistry factor adjusted for the Fort Calhoun weld irradiation temperature and chemical content and made with heat 13253 is 190.4 °F. It is based on the fully credible surveillance data from Salem Unit 2. The derived chemistry factor and the vessel weld best-estimate chemistry factor from Table 1 of Regulatory Guide 1.99, Revision 2 are very similar (190.4 °F and 189.1 °F, respectively).

In the case of heat 12008 and 27204 from Mihama Unit 1 (Table 6A), all three surveillance measurements are predictable within one σ_Δ. The derived chemistry factor is 206.6 °F and includes adjustments for differences in irradiation temperature and chemical content between the Mihama Unit 1 surveillance weld and the Fort Calhoun beltline weld. It is based on the fully credible data from Mihama Unit 1. The derived chemistry factor, 206.6 °F is less than the vessel weld best-estimate chemistry factor, 231.06 °F from Table 1 of Reference 2.

In the case of heat 27204 (tandem) from Diablo Canyon Unit 1 and the Palisades supplemental capsule (Table 4A), all three surveillance measurements are predictable within one σ_Δ. The derived chemistry factor is 215.5 °F and includes adjustments to the irradiation temperature and chemical content of the Fort Calhoun beltline welds. It is based on the fully credible data from Diablo Canyon Unit 1 and Palisades. The derived chemistry factor, 215.5 °F is less than the vessel weld best-estimate chemistry factor, 226.8 °F from Table 1 of Reference 2.

In Table 9, the Fort Calhoun surveillance program results are summarized. These data are credible and predictable. The data scatter based on the derived chemistry factors in Tables 8A, 8B, and 8C are within one σ_Δ for all of the Fort Calhoun surveillance materials, and the scatter is especially small for the surveillance plate and the standard reference material (SRM). The Fort Calhoun surveillance program results were further evaluated as follows:

1. One of the criteria of Regulatory Guide 1.99, Revision 2 is to ascertain that the SRM (correlation monitor) data are consistent with the trend of the database for that material. This is addressed in part in Figures 1 and 2 where it can be seen that the two Fort Calhoun results (at 527 °F and 538 °F) are as predictable as the other HSST Plate 01 data. It is further addressed in Table A2. The twelve sets of data from Combustion Engineering plants were evaluated following Position 2.1 of Reference 2. Those data provide a derived chemistry factor of 130.3 °F. That value is to be compared with the predicted chemistry factor of 131.7 °F based on the best estimate copper and nickel for HSST Plate 01 and the derived chemistry factor of 138.3 °F from the Fort Calhoun measurements alone. The preceding results demonstrate that the Fort Calhoun SRM data are consistent with the trend of the database for that material. The similarity between the derived chemistry factors and the predicted value indicate that the Fort Calhoun vessel irradiation environment is comparable to that of the other Combustion Engineering designed plants.

2. A comparison was made between the Fort Calhoun surveillance weld and the Fort Calhoun beltline welds. The surveillance weld for Fort Calhoun was fabricated using a heat of wire that is not found in any of the beltline welds. It is unique in that it was purchased to a 0.60% nickel specification rather than the 0.0%, 0.75% and 1.00% nickel specifications used to purchase welding electrode heats for the Fort Calhoun beltline welds. The derived chemistry factor for the Fort Calhoun surveillance program weld data is higher than that predicted using Table 1 of Reference 2. That is in contrast to the derived chemistry factors for the surveillance welds from other plants shown in Table 7. The chemistry factors for those welds are consistently equal to or lower than

the predicted chemistry factors. In other words, the surveillance weld data that correspond to the weld wire heats used in the Fort Calhoun beltline welds are conservatively predicted. There is no immediate explanation available for the observation that the Fort Calhoun surveillance weld material (i.e., heat #305414) data were underpredicted by Reference 2, whereas the 0.75% and 1.00% nickel specification heats were conservatively predicted. There are no Fort Calhoun beltline welds with a 0.60% nickel content. Therefore, this issue is not applicable.

The data in Table 7 encompass three of the five most limiting weld wire heat combinations used in the Fort Calhoun reactor vessel beltline. The surveillance data coverage by weld seam is as follows:

Welds 3-410 A/C: D.C. Cook 1 heat 13253, Diablo Canyon 1 heat 27204, Palisades supplemental capsule heat 27204, and Salem 1 heat 13253.

Weld 9-410: No applicable data. [Note: The chemistry factor associated with the best estimate copper and nickel content for heat 20291 is 188.41 °F. This weld is unlikely to be limiting because it is a circumferential weld for which the PTS screening criterion is 300 °F.]

Welds 2-410 A/C: No applicable data. [Note: The chemistry factor associated with the best estimate copper and nickel content for heat 51989 is 89.03 °F. These welds will not become limiting for the Fort Calhoun vessel.]

Position 2.1 of Reference 2 allows one to use credible surveillance data to determine the adjusted reference temperature. This is done by deriving a value for the chemistry factor (CF). If the data scatter is within prescribed limits, then the derived CF may be used with half the normal value for σ_Δ to calculate the adjusted reference temperature. Based on the preceding, there are credible surveillance data for three of the limiting heats used in the Fort Calhoun reactor vessel beltline. For each surveillance weld, a chemistry factor was derived using the ratio method together with an adjustment for irradiation temperature. As shown in Table 7, the derived chemistry factors obtained were less than or equal to the value obtainable from Table 1 of Reference 2. Position 2.1 states that "if this procedure gives a higher value of adjusted reference temperature than that given by using the procedures of Regulatory Position 1.1 (i.e., Table 1 of Reference 2), the surveillance data should be used. If this procedure gives a lower value, either may be used." Given the availability of credible surveillance data that show the Regulatory Position 1.1 chemistry factors to be conservative, those chemistry factors may be used. In the calculation of the margin, If the data scatter is within prescribed limits one may use half the normal value for σ_Δ when determining the adjusted reference temperature.

7.0 <u>Calculation of RT_{PTS}</u>

The limiting beltline material for the Fort Calhoun vessel is that from the lower shell axial welds, 3-410 A/C. The preceding analysis has demonstrated that there are credible surveillance data available for three of the four most limiting weld wire heat combinations used to fabricate those axial welds. These three sets of credible data pertain to each of the heats used for the lower shell axial welds, although not for each possible combination of heats. Given the availability of credible and predictable surveillance data for the three weld wire heat combinations, it is justified to use the derived CF and to use half the normal value for σ_Δ to calculate the margin when determining the adjusted reference temperature. For the one weld wire heat combination for which surveillance data are not yet available, the CF from Table 1 of Reference 2 and the normal value for σ_Δ will be used to calculate the adjusted reference temperature, RT_{PTS}.

Provided below is the determination of the RT_{PTS} for the limiting beltline materials predicted for the end of the current license for Fort Calhoun (August 9, 2013). The neutron fluence was conservatively determined to be 1.728 $x10^{19}$ n/cm^2 (E>1Mev) for that date using an unbiased estimate (see Reference 26). This was projected out to the end of a renewed license period, August 9, 2033, using the same unbiased estimate. (The projected value actually corresponds to the end of that fuel cycle, March 2034 and, therefore, contains an added conservatism.) The projected neutron fluence value is 2.431 $x10^{19}$ n/cm^2 (E>1Mev) (Reference 26). The fluence was calculated in a manner consistent with the methods of the U.S. Nuclear Regulatory Commission's Draft Regulatory Guide DG-1053 (Reference 27). The RT_{PTS} calculation was performed as follows:

$$RT_{PTS} = \text{Initial } RT_{NDT} + \text{Shift} + \text{Margin}$$

Following are the calculations for each of the three heats combinations for which credible and predictable surveillance data are available and for the fourth limiting heat combination for which surveillance data are not yet available.

a. <u>Heat 13253</u>

Initial RT_{NDT} = - 56 °F (generic value for CE welds)

Shift = Chemistry Factor X Fluence Factor

- Chemistry Factor (CF) = 190.4 °F (based on Salem 2 surveillance data)
- Fluence factor (FF) is a function of neutron fluence, f, in units of $1x10^{19}$ n/cm^2
- $FF= f^{(.28 - 0.1 \times \log f)}$

Margin = $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

- σ_Δ = 28 °F/2 = 14 °F (half the value for welds)
- σ_i = 17 °F (for generic CE welds)
- $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2} = 2(17\ °F^2 + 14\ °F^2)^{1/2} = 44.0$ °F

RT_{PTS} = - 56 °F + 190.4 °F X $f^{(.28 - 0.1 \times \log f)}$ + 44.0 °F

For the end of the current license for Fort Calhoun (August 9, 2013), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 219.0 °F + 44.0 °F = 207 °F

For the end of the renewed license period for Fort Calhoun (August 9, 2033), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 235.9 °F + 44.0 °F = 224 °F

These projected values are less than the PTS screening criterion value of 270 °F for axial welds. Thus the vessel weld will remain below the PTS screening criterion for a period exceeding 20 years beyond the current 40 year license term.

b. Heat 12008 and 27204

Initial RT_{NDT} = - 56 °F (generic value for CE welds) [Note: A measured value of initial RT_{NDT} = -58 °F is available for this weld. For purposes of this calculation the more conservative generic value and its associated margin was used.]

Shift = Chemistry Factor X Fluence Factor

- Chemistry Factor (CF) = 206.6 °F (based on Mihama 1 surveillance data)
- Fluence factor (FF) is a function of neutron fluence, f, in units of 1×10^{19} n/cm^2
- $FF = f^{(.28 - 0.1 \times \log f)}$

Margin = $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

- σ_Δ = 28 °F/2 = 14 °F (half the value for welds)
- σ_i = 17 °F (for generic CE welds)
- $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2} = 2(17\ °F^2 + 14\ °F^2)^{1/2} = 44.0\ °F$

RT_{PTS} = - 56 °F + 206.6 °F X $f^{(.28 - 0.1 \times \log f)}$ + 44.0 °F

For the end of the current license for Fort Calhoun (August 9, 2013), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 237.7 °F + 44.0 °F = 226 °F

For the end of the renewed license period for Fort Calhoun (August 9, 2033), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 256.0 °F + 44.0 °F = 244 °F

These projected values are less than the PTS screening criterion value of 270 °F for axial welds. Thus the vessel weld will remain below the PTS screening criterion for a period exceeding 20 years beyond the current 40 year license term.

c. Heat 27204

Initial RT_{NDT} = - 56 °F (generic value for CE welds)

Shift = Chemistry Factor X Fluence Factor

- Chemistry Factor (CF) = 215.5 °F (based on Diablo Canyon 1 and Palisades surveillance data)
- Fluence factor (FF) is a function of neutron fluence, f, in units of $1x10^{19}$ n/cm^2
- FF= $f^{(.28 - 0.1 \times \log f)}$

Margin = $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

- σ_Δ = 28 °F/2 = 14 °F (half the value for welds)
- σ_i = 17 °F (for generic CE welds)
- $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2} = 2(17\ °F^2 + 14\ °F^2)^{1/2} = 44.0\ °F$

RT_{PTS} = - 56 °F + 215.5 °F X $f^{(.28 - 0.1 \times \log f)}$ + 44.0 °F

For the end of the current license for Fort Calhoun (August 9, 2013), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 247.9 °F + 44.0 °F = 236 °F

For the end of the renewed license period for Fort Calhoun (August 9, 2033), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 267.0 °F + 44.0 °F = 255 °F

These projected values are less than the PTS screening criterion value of 270 °F for axial welds. Thus the vessel weld will remain below the PTS screening criterion for a period exceeding 20 years beyond the current 40 year license term.

d. <u>Heat 12008 and 13253</u>

Initial RT_{NDT} = - 56 °F (generic value for CE welds)

Shift = Chemistry Factor X Fluence Factor

- Chemistry Factor (CF) = 208.68 °F (from Table 1, Reference 2 for weld heats 12008 and 13253)
- Fluence factor (FF) is a function of neutron fluence, f, in units of $1x10^{19}$ n/cm^2
- FF= $f^{(.28 - 0.1 \times \log f)}$

Margin = $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

- σ_Δ = 28 °F (value for welds)
- σ_i = 17 °F (for generic CE welds)
- $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2} = 2(17\ °F^2 + 28\ °F^2)^{1/2} = 65.5\ °F$

RT_{PTS} = - 56 °F + 208.68 °F X $f^{(.28 - 0.1 \times \log f)}$ + 65.5 °F

For the end of the current license for Fort Calhoun (August 9, 2013), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 240.1 °F + 65.5 °F = 250 °F

For the end of the renewed license period for Fort Calhoun (August 9, 2033), the RT_{PTS} is:

RT_{PTS} = - 56 °F + 258.6 °F + 65.5 °F = 268 °F

These projected values are less than the PTS screening criterion value of 270 °F for axial welds. Thus the vessel weld will remain below the PTS screening criterion for a period exceeding 20 years beyond the current 40 year license term.

e. Plate Code D4802-2 (Heat A1768-1)

Initial RT_{NDT} = 18 °F (measured value)

Shift = Chemistry Factor X Fluence Factor

- Chemistry Factor (CF) = 72.0 °F (based on Fort Calhoun surveillance data)
- Fluence factor (FF) is a function of neutron fluence, f, in units of 1×10^{19} n/cm^2
- FF= $f^{(.28 - 0.1 \times \log f)}$ where f= 2.45×10^{19} n/cm^2 and 3.45×10^{19} n/cm^2 for the current and renewed license period, respectively (Reference 26).

Margin = $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2}$

- σ_Δ = 17 °F/2 = 8.5 °F (half the value for plates)
- σ_i = 0 °F (for measured value)
- $2(\sigma_i^2 + \sigma_\Delta^2)^{1/2} = 2(0\ °F^2 + 8.5\ °F^2)^{1/2} = 17.0\ °F$

RT_{PTS} = 18 °F + 72.0 °F X $f^{(.28 - 0.1 \times \log f)}$ + 17.0 °F

For the end of the current license for Fort Calhoun (August 9, 2013), the RT_{PTS} is:

RT_{PTS} = 18 °F + 89.4 °F + 17.0 °F = 124 °F

For the end of the renewed license period for Fort Calhoun (August 9, 2033), the RT_{PTS} is:

RT_{PTS} = 18 °F + 95.3 °F + 17.0 °F = 130 °F

These projected values are less than the PTS screening criterion value of 270 °F for plates. Thus the vessel plate will remain below the PTS screening criterion for a period exceeding 20 years beyond the current 40 year license term.

8.0 Conclusions

1) The Fort Calhoun surveillance program data are credible and predictable as summarized in Table 9.

2) There are four sets of credible surveillance weld data available from other plants that are applicable to the Fort Calhoun reactor vessel beltline welds. The derived chemistry factor given in Table 7 for each set was less than or equal to the value obtainable from Table 1 of Regulatory Guide 1.99.

3) Given the availability of credible and predictable surveillance weld data, it is justified to use half the normal value for σ_Δ to calculate the margin when determining the adjusted reference temperature for the Fort Calhoun vessel beltline materials.

4) The highest projected value of RT_{PTS} is 250 °F at the end of the current license. This was determined using the normal value for σ_Δ (28 °F) and the limiting material chemistry factor of 208.68 °F from Table 1 of Regulatory Guide 1.99, Revision 02. It corresponds to weld wire heats 12008 and 13253 for Fort Calhoun weld 3-410 A/C. The highest projected value of RT_{PTS} at the end of the renewed license term is 268 °F for that same weld material as shown in Table 10. These projected values are less than the PTS screening criterion value of 270 °F for plates and axial welds and

Thus the vessel plates and welds will remain below the PTS screening criterion for a period exceeding 20 years beyond the current 40 year license term.

5) In the analysis of the surveillance data, the data were adjusted for both differences in copper and nickel content and for differences in irradiation temperature. It was necessitated by the fact that the data available for one of the heats was from two different reactor vessel surveillance programs that in turn had to be adjusted for the Fort Calhoun vessel. The irradiation temperature adjustment method was based on the use of NUREG/CR-6551 (Reference 4). In the two cases evaluated, the adjustment method resulted in a derived chemistry factor that was comparable to that obtained using guidelines (Reference 3) developed previously. The proposed method with its dual adjustments was successfully used to reconcile surveillance data from two different plants.

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21. "OPPD Fort Calhoun Station, Evaluation of Irradiated Capsule W-275", November 1994, BAW-2226.

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23. "Application of Reactor Vessel Surveillance Data for Embrittlement Management", Combustion Engineering Owners Group Report CEN-405-P, Revision 3, September 1996.

24. Robert E. Denton, Baltimore Gas and Electric Company, "Request for Approval of Updated Values of Pressurized Thermal Shock (PTS) Reference Temperatures (RT_{PTS}) Values (10CFR50.61)", letter dated July 21, 1995.

25. "Duke Power Company, Evaluation of McGuire Unit 1, Surveillance Weld Data Credibility", Technical Report No. ATI-98-012-T005, revision 1, November 1998, transmitted by Duke Energy Corporation letter, H.B. Barron to U.S. Nuclear Regulatory Commission, "Reactor Vessel Radiation Surveillance Program", dated January 7, 1999.

26. S.L. Anderson, "Fast Neutron Fluence Evaluations for the Fort Calhoun Unit 1 Reactor Pressure Vessel", Westinghouse Report WCAP-15443, July 2000.

27. U.S. Nuclear Regulatory Commission's Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence".

Table 1
Identification of Reactor Vessel Plates and Welds in the Fort Calhoun Reactor Vessel Beltline

Plate or Weld Identification	Plate or Weld Electrode Heat No.	Weld Flux Type and Lot No.	Chemistry Factor (°F)[a]
Plate D4802-1	C2585-3	N/A	82.2
Plate D4802-2	A1768-1	N/A	65
Plate D4802-3	A1768-2	N/A	73.1
Plate D4812-1	C3213-2	N/A	83
Plate D4812-2	C3143-2	N/A	65
Plate D4812-3	C3143-3	N/A	65
Surveillance Plate D4802-2	A1768-1	N/A	72.0[c]
2-410 A/C	51989	Linde 124, #3687	89.03
3-410 A/C	12008 & 13253 (T)[b]	Linde 1092, #3774	208.68
3-410 A/C	13253 (T)[b]	Linde 1092, #3774	189.05
3-410 A/C	12008 & 27204 (T)[b]	Linde 1092, #3774	231.06
3-410 A/C	27204 (T)[b]	Linde 1092, #3774	226.81
9-410	20291	Linde 1092, #3833	188.41
Surveillance Weld	305414	Linde 1092, #3947 and #3951	212

Notes:
a) Chemistry Factor from Table 1 or 2 of Reference 2.
b) "T" denotes a tandem arc weld; other welds are single arc.
c) Chemistry Factor as derived based using surveillance measurements in Table 8B of this report.

Table 2

Identification of Reactor Vessel Surveillance Program
Welds Applicable to the Fort Calhoun Vessel Beltline Welds

Reactor Vessel	Weld Electrode Heat No.	Flux Type and Lot No.	Copper Content (%)	Nickel Content (%)
DC Cook 1	13253	Linde 1092, #3791	.27	.74
Salem 2	13253	Linde 1092, #3774,3833	.254	.726
Diablo Canyon 1	27204	Linde 1092, #3714	.20	1.00
Mihama 1	12008 & 27204	Linde 1092, #3724	.19	1.08
Fort Calhoun Suppl.	27204	Linde 1092, #3714	.19	1.07
Palisades Suppl.	27204	Linde 1092, #3714	.19	1.07
Diablo Canyon 2*	12008 & 21935	Linde 1092, #3869	.219	.871
Fort Calhoun*	305414	Linde 1092, #3947,3951	.35	.60
McGuire 1*	12008 & 20291	Linde 1092, #3854	.198	.874
Fitzpatrick (BWR)	12008 & 13253	Linde 1092, #3774	n/a	n/a
Cooper (BWR)*	20291	Linde 1092, #3833	n/a	n/a
Pilgrim (BWR)*	12008 & 20291	Linde 1092, #3833	.161	.794

* These are not fully applicable to the Fort Calhoun vessel limiting beltline welds.

REFERENCE USE

Table 3

**Test Results from the D.C. Cook Unit 1
Reactor Vessel Surveillance Program
(Surveillance Weld Wire Heat No. 13253)**

Capsule Identity	Charpy Shift, °F	Adjusted[(a)] Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
T	70	60.9	2.69E18	537
X	146	128.7	8.13E18	537*
Y	184	162.5	1.23E19	537
U	109	92.6	1.77E19	537

* not reported; assumed to be same as other reported values

Capsule Identity	Adjusted[(a)] Charpy Shift, °F	(FF) x Adjusted Shift	Fluence Factor (FF)	(FF)^2	Adjusted – Predicted[b] Shift, °F
T	60.9	39.1	.6424	.4127	60.9-88.3=-27.4
X	128.7	121.2	.9419	.8872	128.7-129.4=-0.7
Y	162.5	171.9	1.0577	1.1187	162.5-145.3=17.2
U	92.6	107.1	1.1569	1.3383	92.6-159=-66.4

$CF_{(ALL)}$=439.3/3.7569= 116.9 °F   Σ=439.3   Σ =3.7569
$CF_{(w/o\,U)}$=332.2/2.4186= 137.4 °F   Σ =332.2   Σ =2.4186

(a) Shift adjusted for FCS T_{cold} (543 °F) and best estimate chemistry
(b) Predicted using $CF_{(w/o\,U)}$= 137.4 °F

Table 4A
Test Results from Diablo Canyon Unit 1 and Supplemental Capsule with T_{cold} and CF Pre-Adjustment for Weld Heat 27204

Capsule Identity	Charpy Shift, °F	Adjusted[(a)] Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
DC1-S	113	114	2.84E18	539
DC1-Y	233	236	9.41E18	540
SA-60-1	250	239	1.62E19	533

Capsule Identity	Adjusted[(a)] Charpy Shift, °F	(FF) x Adjusted Shift	Fluence Factor (FF)	$(FF)^2$	Adjusted – Predicted** Shift, °F
DC1-S	114	74.8	.6562	.4306	114-141=-27
DC1-Y	236	232.0	.9830	.9662	236-212= 24
SA-60-1	239	270.8	1.1331	1.2840	239-244= -5
		Σ =577.6		Σ =2.6808	

CF=577.6/2.6808= 215.5 °F

(a) Shift adjusted for FCS T_{cold} (543 °F) and best estimate chemistry

REFERENCE USE

Table 4B
Test Results from Diablo Canyon Unit 1 and Supplemental Capsule with Separate Adjustment for T_{cold} and CF for Weld Heat 27204

Capsule Identity	Charpy Shift, °F	Adjusted[a] Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
DC1-S	113	115.5	2.84E18	539
DC1-Y	233	238.1	9.41E18	540
SA-60-1	250	247.5	1.62E19	533

Capsule Identity	Adjusted[a] Charpy Shift, °F	(FF) x Adjusted Shift	Fluence Factor (FF)	(FF)^2	Adjusted – Predicted** Shift, °F
DC1-S	115.5	75.8	.6562	.4306	115-144=-29
DC1-Y	238.1	234.0	.9830	.9662	238-216= 22
SA-60-1	247.5	280.4	1.1331	1.2840	247-249= -2
		Σ =590.2		Σ =2.6808	

CF=590.2/2.6808= 220.2 °F
CF_{Tcold}= 220.2 °F + (533 °F – 543 °F) = 210.2 °F

(a) Shift adjusted for best estimate chemistry

Table 5

**Test Results from the Salem Unit 2
Reactor Vessel Surveillance Program
(Surveillance Weld Wire Heat No. 13253)**

Capsule Identity	Charpy Shift, °F	Adjusted[a] Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
T	145	136.8	2.75E18	539
U	180	169.7	5.50E18	539
X	188	176.6	1.07E19	539

Capsule Identity	Adjusted[a] Charpy Shift, °F	(FF) x Shift	Fluence Factor (FF)	(Fluence Factor)^2	Measured minus Predicted Shift, °F
T	136.8	88.6	.6480	.4199	136.8-123.4=13.4
U	169.7	141.3	.8328	.6936	169.7-158.6= 11.1
X	176.6	179.9	1.0189	1.0382	176.6-194= -17.4
		Σ = 409.8		Σ = 2.1517	

CF=409.8 /2.1517= 190.4 °F

(a) Shift adjusted for FCS T_{cold} (543 °F) and best estimate chemistry

REFERENCE USE

Table 6A
Test Results from Mihama Unit 1 Surveillance Capsules with T_{cold} and CF Pre-Adjustment for Weld Heats 12008 and 27204

Capsule Identity	Charpy Shift, °F	Adjusted[(a)] Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
1	187.2	194.8	6.0 E18	552
2	205.2	214.1	1.2 E19	552
3	226.8	238.2	2.1 E19	552

Capsule Identity	Adjusted[(a)] Charpy Shift, °F	(FF) x Adjusted Shift	Fluence Factor (FF)	(FF)^2	Adjusted – Predicted** Shift, °F
1	194.8	166.9	.85696	.7344	195-177= 18
2	214.1	225.0	1.05086	1.1043	214-217= -3
3	238.2	286.3	1.20182	1.4444	238-248= -10
		Σ =678.2		Σ =3.2831	

CF=678.2/3.2831= 206.6 °F

(a) Shift adjusted for FCS T_{cold} (543 °F) and best estimate chemistry

Table 6B
Test Results from Mihama Unit 1 Surveillance Capsules with Separate Adjustment for T_{cold} and CF for Weld Heat 12008 and 27204

Capsule Identity	Charpy Shift, °F	Adjusted[(a)] Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
1	187.2	190.4	6.0 E18	552
2	205.2	208.6	1.2 E19	552
3	226.8	230.7	2.1 E19	552

Capsule Identity	Adjusted[(a)] Charpy Shift, °F	(FF) x Adjusted Shift	Fluence Factor (FF)	(FF)^2	Adjusted – Predicted** Shift, °F
1	190.4	163.2	.85696	.7344	190-172= 18
2	208.6	219.2	1.05086	1.1043	209-211= -2
3	230.7	277.3	1.20182	1.4444	231-241= -10
		Σ =659.7		Σ =3.2831	

CF=659.7/3.2831= 200.9 °F
CF_{Tcold}= 200.9 °F + (552 °F – 543 °F) = 209.9 °F

(a) Shift adjusted for best estimate chemistry

Table 7
Derived Chemistry Factors for Reactor Vessel Surveillance Program Welds Applicable to Fort Calhoun Vessel Weld 3-410

Reactor Vessel	Weld Electrode Heat No.	Flux Type and Lot No.	Derived Chemistry Factor[a], CF (°F)	RG 1.99 CF (°F) for Surveillance Weld Chemistry[b]	RG 1.99 CF (°F) for Best Estimate Weld Chemistry[c]
DC Cook 1	13253	Linde 1092 #3791	137.4	206.4	189.1
Diablo Canyon 1 and Supp. Capsule	27204	Linde 1092 #3714	215.5 (210.2)	221.8	226.8
Salem 2	13253	Linde 1092 #3774,3833	190.4	198	189.1
Mihama 1	12008 & 27204	Linde 1092 #3724	206.6 (209.9)	227.2	231.06

a) Adjusted to Best Estimate CF and T_{cold} for Fort Calhoun (543 °F); value in parentheses was determined by adjusting for T_{cold} after deriving chemistry factor.
b) Chemistry Factor (CF) from Table 1 of Reference 2 based on the copper and nickel content for the surveillance weld.
c) Chemistry Factor (CF) from Table 1 of Reference 2 based on the best estimate copper and nickel content for the weld wire heat or combination of heats.

Table 8A

**Test Results from the Fort Calhoun
Reactor Vessel Surveillance Program
(Surveillance Weld Wire Heat No. 305414)**

Capsule Identity	Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
W225	210	5.53E18	527
W265	225	7.71E18	534
W275	219	1.28E19	538

Capsule Identity	Charpy Shift, °F	(FF) x Shift	Fluence Factor (FF)	(FF)^2	Measured - Predicted Shift, °F
W225	210	175.2	.8343	.6961	210-191.1=18.9
W265	225	208.6	.9270	.8593	225-212.3=12.7
W275	219	234.0	1.0687	1.1421	219-244.7=-25.7
	CF=617.8/2.6975= 229.0 °F	Σ =617.8		Σ =2.6975	

Table 8B

**Test Results from the Fort Calhoun
Reactor Vessel Surveillance Program
(Surveillance Plate Heat No. A1768-1)**

Capsule Identity	Charpy Shift, °F (Lg,Tr)[a]	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
W225	60, N/A	5.53E18	527
W265	74,70	7.71E18	534
W275	73,72	1.28E19	538

a) "Lg" is longitudinal and "Tr" is for transverse orientation Charpy data

Capsule Identity	Charpy Shift, °F (Lg,Tr)	(FF) x Shift	Fluence Factor (FF)	$(FF)^2$	Measured - Predicted Shift, °F
W225	60	50.1	.8343	.6961	60-60.1=-0.1
W265	74,70	68.6,64.9	.9270	.8593	74-66.7=7.3 70-66.7=3.3
W275	73,72	78.0,76.9	1.0687	1.1421	73-76.9=-3.9 72-76.9=-4.9
		CF=338.5/4.6989= 72.0 °F	Σ =338.5	Σ =4.6989	

Table 8C

**Test Results from the Fort Calhoun
Reactor Vessel Surveillance Program
(Standard Reference Material)**

Capsule Identity	Charpy Shift, °F	Neutron Fluence, n/cm^2	Irradiation Temperature, °F
W225	124*	5.53E18	527
W265	N/A	7.71E18	534
W275	141*	1.28E19	538

* shift per Surveillance Program test report

Capsule Identity	Charpy Shift, °F	(FF) x Shift	Fluence Factor (FF)	(FF)^2	Measured - Predicted Shift, °F
W225	124	103.5	.8343	.6961	124-115.4=8.6
W275	141	150.7	1.0687	1.1421	141-147.8=-6.8
	CF=254.2/1.8382= 138.3 °F		Σ =254.2	Σ =1.8382	

Table 9

Derived Chemistry Factors for Fort Calhoun Reactor Vessel Surveillance Materials

Material Identity	Material Description	Derived Chemistry Factor (°F)	RG 1.99 Table 1 or 2 Chemistry Factor (°F)
Weld	Heat 305414, Linde 1092	229.0	212
Plate D4802-2	SA 533B Class 1	72.0	65
SRM	HSST Plate 01	138.3	131.7

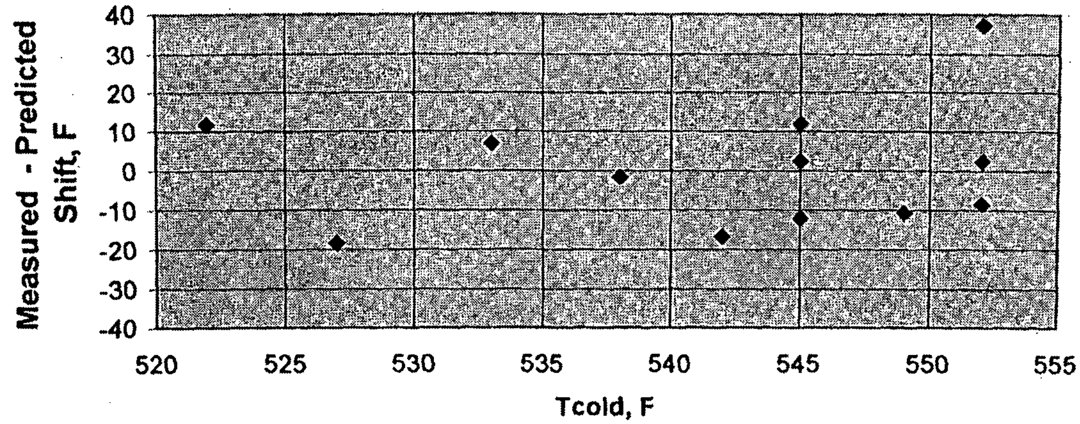
Table 10
Predicted RT_{PTS} for the Fort Calhoun Reactor Vessel Beltline Plates and Welds

Plate or Weld Identification	Plate or Weld Electrode Heat No.	Chemistry Factor (°F)	Predicted RT_{PTS} through 2033[c] (°F)
Plate D4802-1	C2585-3	82.2[a]	143
Plate D4802-2	A1768-1	72.0[b]	130
Plate D4802-3	A1768-2	73.1[a]	131
Plate D4812-1	C3213-2	83[a]	144
Plate D4812-2	C3143-2	65[a]	120
Plate D4812-3	C3143-3	65[a]	120
2-410 A/C	51989	89.03[a]	120
3-410 A/C	12008 & 13253 (T)	208.68[a]	268
3-410 A/C	13253 (T)	190.4[b]	224
3-410 A/C	12008 & 27204 (T)	206.6[b]	244
3-410 A/C	27204 (T)	215.5[b]	255
9-410	20291	188.41[a]	259

Notes:
a) Chemistry Factor from Table 1 or 2 of Reference 2 or derived using surveillance measurements in this report.
b) Chemistry Factor derived using surveillance measurements in this report.
c) Prediction based on fluence of 2.43×10^{19} n/cm^2 for axial welds and 3.45×10^{19} n/cm^2 for plates and weld 9-410.

REFERENCE USE

Figure 1
Effect of Tcold on SRM Data
HSST Plate 01 Results
Normalized to 1E19 n/cm2



REFERENCE USE

Figure 2
Effect of Tcold on SRM Data
HSST Plate 01 Results (CF=130.3 F)

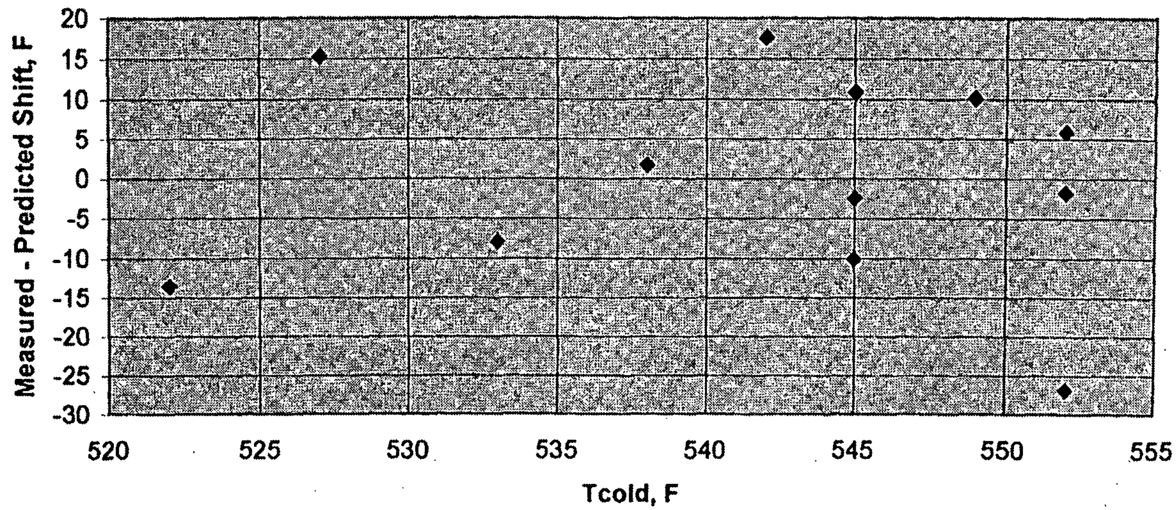


Table A1
Standard Reference Material Data from Combustion Engineering Designed Surveillance Capsules

Reactor Vessel	Surveillance Capsule	SRM Material Identification	Charpy Shift (°F)	Neutron Fluence (10^{19} n/cm^2)	Irradiation Temperature (°F)
Calvert Cliffs 1	W263	HSST 01	101	0.59	545
Calvert Cliffs 2	W263	HSST 01	120	0.806	545
Fort Calhoun	W225	HSST 01	124* (116)	0.553	527
Fort Calhoun	W275	HSST 01	141* (162)	1.28	538
Millstone 2	W104	HSST 01	136	0.884	549
Maine Yankee	A25	HSST 01	137	1.76	522
Maine Yankee	W253	HSST 01	156	1.25	542
Palisades	W110	HSST 01	143	1.78	533
Palo Verde 1	W137	HSST 01	98	0.345	552
Palo Verde 2	W137	HSST 01	96	0.407	552
Palo Verde 3	W137	HSST 01	67*	0.364	552
St. Lucie 1	W104	HSST 01	129	0.716	545

*Shift per surveillance report

Table A2
Analysis of Standard Reference Materials

Irradiation Temperature, (°F)	Shift (°F)	(FF) x Shift	$(FF)^2$	Fluence (10^{19} n/cm^2)	Fluence Factor (FF)	Measured-Predicted Shift, (°F)
545	101	86.08	0.7264	0.59	0.85229	101 - 111.1 = -10.1
545	120	112.74	0.8827	0.806	0.93950	120 – 122.4 = -2.4
527	124*	103.46	0.6961	0.553	0.83434	124 – 108.7 = 15.3
538	141*	150.69	1.1422	1.28	1.06873	141 – 139.3 = 1.7
549	136	131.30	0.9321	0.884	0.9654	136 – 125.8 = 10.2
522	137	157.28	1.3348	1.76	1.1554	137 – 150.5 = -13.5
542	156	165.70	1.1282	1.25	1.0622	156 – 138.4 = 17.6
533	143	165.65	1.3418	1.78	1.1584	143 – 150.9 = -7.9
552	98	69.26	0.4994	0.345	0.70669	98 – 92.1 = 5.9
552	96	72.06	0.5635	0.407	0.75066	96 – 97.8 = -1.8
552	67*	48.30	0.5196	0.364	0.72085	67 – 93.9 = -26.9
545	129	116.91	0.8214	0.716	0.90630	129 – 118.1 = 10.9

*Shift per surveillance report

<u>(FF) x Shift</u>	<u>$(FF)^2$</u>	
Σ=1379.43	Σ=10.5882	CF=(1379.43)/ (10.5882)=130.3 °F