APPENDIX C

CHARPY V-NOTCH PLOTS FOR EACH CAPSULE USING SYMMETRIC HYPERBOLIC TANGENT CURVE-FITTING METHOD

 \mathcal{L}

Contained in Table C-1 are the upper shelf energy values used as input for the generation of the Charpy Vnotch plots using CVGRAPH, Version 4.1. The definition for Upper Shelf Energy (USE) is given in ASTM E185-82, Section 4.18, and reads as follows:

"upper shelf energy level - the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy."

If there are specimens tested in set of three at each temperature Westinghouse reports the set having the highest average energy as the USE (usually unirradiated material). If the specimens were not tested in sets of three at each temperature Westinghouse reports the average of all 100% shear Charpy data as the USE. Hence, the USE values reported in Table C-1 and used to generate the Charpy V-notch curves were determined utilizing this methodology.

The lower shelf energy values were fixed at 2.2 ft-lb for all cases.

UNIRR LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: LT

Capsule: UNIRR Total Fluence:

CAPSULE X LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material PLATE SA533B1 Heat Number: C4339-1 Orientation: LT

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature 300 345

Input CVN Energy Computed CVN Energy Differential 120 12114 -114 124 121.72 227

SUM of RESIDUALS = 15.53

CAPSULE V LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1

Orientation[.] LT

Capsule: V Total Fluence:

Charpy V-Notch Data (Continued)

Temperature 350 425

Input CVN Energy Computed CVN Energy 138 118.99 109 120.52

 $SUM of RESULTS = 13.74$ Differential 19 -11.52

C-8

CAPSULE Y LOWER SHELL PLATE C4339-1 (LONG)

Page 2

MateriaL PLATE SA533B1 Heat Number C4339-1 Orientation: LT

 \overline{a}

Capsule Y Total Fluence:

Charpy **V-Notch Data (Continued)**

Temperature Input CVN Energy Computed CVN Energy Differential 300 105 101.91 3.08 $\frac{111}{225}$ 10509 59 350 116 10721 8.78 SUS
 $\begin{array}{r} \text{Mferential} \ 3.08 \\ 59 \\ \text{SUM of RESIDUALS} = 25.33 \end{array}$

UNIRR LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number C4339-1 Orientation- LT

 \overline{a}

Capsule: UNIRR Total Fluence:

CAPSULE X LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1 Orientation: LT

 \sim

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature 300 345

 $\begin{array}{cccc}\text{Input lateral Expansion} & \text{Computed L.E.} & \text{Differential} \\ 87.5 & 90.5 & 91.3 & -302 \\ -3.02 & -8.0 & -1.0 & -1.0 \\ \end{array}$ 87.5 9052 *-3.02* $913 -8$ SUM of RESIDUALS = 2.29

CAPSULE Y LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1

Orientation LT

Capsule: Y Total Fluence:

 \pm $\bar{1}$

UNIRR LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1

 \overline{a}

Orientation[.] LT

Capsule: UNIRR Total Fluence

CAPSULE X LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature 300 345

Input Percent Shear Computed Percent Shear 100 99.54 100 9989

Orientation: LT

Differential .45 I $SUM of RESULTS = 13.42$

CAPSULE V LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1

Orientation: LT

Capsule: V Total Fluence:

Charpy V-Notch Data (Continued)

Temperature 350 425

Input Percent Shear Computed Percent Shear 100 97.36 100 99.22

SU **M** of RESIDUALS = 13.51 Differential 2.63 **.77**

CAPSULE Y LOWER SHELL PLATE C4339-1 (LONG)

Page 2

Material: PLATE SA533B1 Heat Number C4339-1 Orientation: LT

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

Temperature Input Percent Shear Computed Percent Shear Differentia 300 and 100 and 300 and 300 and 300 and 4.31 325 100 97.73 226 350 100 98.82 1.17 SUM of RESIDUALS $= 20.83$

UNIRR LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule UNIRR Total Fluence:

C-28

CAPSULE X LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperature Input CVN Energy Computed CVN Energy Differential 300 -3.48 -3.48 345 100 100 3339 9339 6.6

SUM of RESIDUAIS = 9.96

CAPSULE Y LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

e Input CVN Energy Computed CVN Energy $95 \t\t 04.42$ $95 \t 07.42$ 93 8952 $SUM of RESIDUALS = 16.16$ Temperatur 300 325 350 Differential 10.57 757 3.47

UNIRR LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1 Orientation: TL

Capsule: UNIRR Total Fluence

CAPSULE X LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperatur 300 <u>კ4ე</u>

73 7225 68.5 7292

eInput Lateral Expansion Computed LE.

Differential .74 -4.42 SUM of RESIDUALS $= 2.56$

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 $\frac{1}{3}$

CAPSULE Y LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule: Y Total Fluence:

UNIRR LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number- C4339-1 Orientation: TL

Capsule: UNIRR Total Fluence:

C-44

CAPSULE X LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule: X Total Fluence:

Charpy V-Notch Data (Continued)

Temperatur 300 345

Input Percent Shear Computed Percent Shear 100 9993 100 99.98 $SUM of RESIDUALS = 10.89$ Differential .06 .01

CAPSULE V LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number. C4339-1 Orientation: TL

Capsule: V Total Fluence:

Charpy V-Notch Data (Continued)

Temperatur 350 425

Input Percent Shear Computed Percent Shear 100 97.64 100 99.51 SUM of RESIDUALS

Differential 2.35 .48 = 28.54

 $\frac{1}{1}$

CAPSULE Y LOWER SHELL PLATE C4339-1 (TRANS)

Page 2

Material: PLATE SA533B1 Heat Number: C4339-1 Orientation: TL

Capsule Y Total Fluence:

Charpy V-Notch Data (Continued)

Temperature Input Percent Shear Computed Percent Shear Differential 300 5.76 325 330.98 100 96.98 301 350 1.55 100 1.55 98.44 1.55 1.55 SUM of RESIDUALS $= 28.42$

 $\overline{1}$

* UNIRRADIATED WELD

Page 2

Material: WELD

Heat Number: WIRE HEAT 0227

Orientation:

Capsule: UNIRR Total Fluence:

 $\overline{}$ $\frac{1}{4}$

CAPSULE Y WELD

Page 2

Material: WELD Heat Number: WIRE HEAT 0227 Orientation:

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

 $\fbox{Temperature} \quad 375$

 $\begin{array}{c} \text{Input CVN Energy} \\ \text{66} \end{array}$

 $\begin{array}{c} \text{Computed CVN Energy} \\ 5729 \end{array}$ ³⁷⁵ Energy Differential 8.7

SUM of RESIDUALS = 7.73

UNIRRADIATED WELD

Page 2

Material: WELD Heat Number. WIRE HEAT 0227 Orientation:

Capsule: UNIRR Total Fluence:

C-58

CAPSULE Y WELD

Page 2

Material: WELD Heat Number. WIRE HEAT 0227 Orientation:

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

Temperature აი

Input Lateral Expansion Computed LE 50 45.87

Differential 412 SUM of RESIDUALS $= 13$

. UNIRRADIATED WELD

Page 2

Material WELD Heat Number. WIRE HEAT 0227

Orientation:

Capsule: UNIRR Total Fluence

CAPSULE Y WELD

Page 2

Material: WELD Heat Number: WIRE HEAT 0227 Orientation

Capsule: Y Total Fluence:

Charpy V-Notch Data **(Continued)**

Temperature 375

Input Percent Shear $100\,$

 $\begin{tabular}{ll} Computed Percent Shear & Differential \\ 99.69 & \hspace*{2.5cm} 3 \\ \end{tabular}$ 100 .3

SUM of RESIDUALS $= 6.44$

UNIRRADIATED HEAT AFFECTED ZONE

Page 2

Material: HEAT AFFD ZONE Heat Number: C4339-1 SIDE OF WELD Orientation:

Capsule: UNIRR Total Fluence:

 $\bar{1}$

CAPSULE Y HEAT AFFECTED ZONE

Page 2

Material: HEAT AFFD ZONE Heat Number: C4339-1 SIDE OF WELD Orientation:

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

 $\frac{1}{2}$ Temperature Input CVN Energy Computed CVN Energy Differential 325 33 325 33 52 SUM **of** RESIDUALS =23.75

C-73

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UNIRRADIATED HEAT AFFECTED ZONE

Page 2

Material: HEAT AFFD ZONE Heat Number: C4339-1 SIDE OF WELD Orientation:

Capsule: UNIRR Total Fluence:

C-77

 $\frac{1}{2}$

CAPSULE Y HEAT AFFECTED ZONE

Page 2

Material: HEAT AFFD ZONE Heat Number: C4339-1 SIDE OF WELD Orientation:

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

Temperature 325

Input Lateral Expansion Computed LE Differential 66 66 67.32 5UM of RESIDUALS = $.53$

UNIRRADIATED HEAT AFFECTED ZONE

Page 2

Material: HEAT AFFD ZONE Heat Number: C4339-1 SIDE OF WELD Orientation:

Capsule UNIRR Total Fluence:

C-82

CAPSULE Y HEAT AFFECTED ZONE

Page 2

Material: HEAT AFFD ZONE Heat Number: C4339-1 SIDE OF WELD Orientation:

Capsule: Y Total Fluence:

Charpy V-Notch Data **(Continued)**

Temperature Input Percent Shear Computed Percent Shear Differential
100 945 200 2011 325 100 100 945 5.49 $SUM of RESIDUALS = 7.53$

UNIRRADIATED CORRELATION MONITOR MATERIAL

Page 2

Material: SRM SA533B1 Heat Number: HSST PLATE 02 Orientation: LT

Capsule: UNIRR Total Fluence:

C-90

CAPSULE Y CORRELATION MONITOR MATERIAL

Page 2

Material: SRM SA533B1 Heat Number: HSST PLATE 02 Orientation: LT

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

Temperature **375**

Input CVN Energy Computed CVN Energy Differential 96.36 PHS 10.63 107
SUM of RESIDUALS = 25.26

UNIRRADIATED CORRELATION MONITOR MATERIAL

Page 2

Material: SRM SA533B1 Heat Number: HSST PLATE 02 Orientation: LT

Capsule: UNIRR Total Fluence:

C-95

CAPSULE Y CORRELATION MONITOR MATERIAL

Page 2

Material: SRM SA533B1 Heat Number. HSST PLATE 02 Orientation: LT

 $\ddot{}$

Capsule: Y Total Fluence:

Charpy V-Notch Data (Continued)

Temperature **373**

Input Lateral Expansion Computed **LE** Differential 69 725 -3.35 SUM of RESIDUALS $= 1.73$

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 $\begin{array}{c} \begin{array}{c} \hline \end{array} \end{array}$

UNIRRADIATED CORRELATION MONITOR MATERIAL

Page 2

Material: SRM SA533B1 Heat Number: HSST PLATE 02 Orientation⁻ LT

Capsule: UNIRR Total Fluence:

C-102

 $\frac{1}{4}$

CAPSULE Y CORRELATION MONITOR MATERIAL

Page 2

Material: SRM SA533B1 Heat Number. HSST PLATE 02

Orientation: LT

Capsule: Y Total Fluence:

Charpy V-Notch Data **(Continued)**

Temperature 375

100 9723

Input Percent Shear Computed Percent Shear Sul

Differential M of RESIDUALS = 29.02

APPENDIX D

VALIDATION OF **THE RADIATION** TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

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D 1 Neutron Dosimetry

Comparisons of-measured dosimetry results to both the calculated and least squares adjusted values for all surveillance capsules withdrawn from service to date at Surry Unit 2 Unit 2 are described herein. The sensor sets from these capsules have been analyzed in accordance with the current dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."^[D-1] One of the main purposes for presenting this material is to demonstrate that the overall measurements agree with the calculated and least squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 6.2 of this report. This information may also be useful in the future, in particular, as least squares adjustment techniques become accepted in the regulatory environment.

D. 1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of the five neutron sensor sets withdrawn to date as a part of the Surry Unit 2 Reactor Vessel Materials Surveillance Program are presented. The capsule designation, location within the reactor, and time of withdrawal of each of these dosimetry sets were as follows:

The azimuthal locations included in the above tabulation represent the first octant equivalent azimuthal angle of the geometric center of the respective surveillance capsules.

The passive neutron sensors included in the evaluations of Surveillance Capsules X, W, V, S, and Y are summarized as follows:

Pertinent physical and nuclear characteristics of the passive neutron sensors are listed in Table D-1. The use of passive monitors such as those listed above does not yield a direct measure of the energy dependent neutron flux at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average neutron flux level incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- the measured specific activity of each monitor,
- the physical characteristics of each monitor,
- the operating history of the reactor,
- the energy response of each monitor, and
- the neutron energy spectrum at the monitor location.

The radiometric-counting of the neutron sensors from Capsules X and W was carried out by the Battelle Memorial Institute. The radiometric counting of the sensors from Capsules V, S, and Y was completed at the Pace Analytical Services Laboratory located at the Westinghouse Waltz Mill Site. In all cases, the radiometric counting followed established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor was determined by means of a high-resolution gamma spectrometer. For

the copper, iron, nickel, and cobalt-aluminum sensors, these analyses were performed by direct counting of each of the individual samples In the case of the uranium fission sensors, the analyses were carried out by direct counting preceded by dissolution and chemical separation of cesium from the sensor material.

The irradiation history of the reactor over the irradiation periods experienced by Capsules X, W, V, S, and Y was based on the reported monthly power generation of Surry Unit 2 from initial reactor criticality through the end of the dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$
R = \frac{A}{N_0 F Y \sum \frac{P_j}{P_{ref}} C_j [I - e^{-\lambda t_j}] [e^{-\lambda t_d}]}
$$

where.

- R *=* Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
- A = Measured specific activity (dps/gm).
- *No =* Number of target element atoms per gram of sensor.
- F *=* Weight fraction of the target isotope in the sensor material.
- Y *=* Number of product atoms produced per reaction.
- P, *=* Average core power level during irradiation period j (MW).
- P_{ref} = Maximum or reference power level of the reactor (MW)
- **C,** *=* Calculated ratio of sensor reaction rate during irradiation period j to the time weighted average sensor reaction rate over the entire irradiation period.
- *X=* Decay constant of the product isotope (1/sec). λ
- t_1 = Length of irradiation period j (sec).
- t_{d} = Decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the irradiation period.

For capsules remaining in a single location for the entire irradiation period, the spectrum averaged reaction cross-section is essentially constant and, therefore, the cycle dependent neutron flux ($E > 1.0$ MeV) can be substituted for individual reaction rates in the computation of the **C,** term. However, for cases such as
Capsule Y where relocation of the capsule resulted in significant changes in the relative neutron energy spectrum, the explicit sensor reaction rates must be used to compute the time history corrections.

In the equation describing the reaction rate calculation, the ratio $[P_1]/[P_{ref}]$ accounts for month-by-month variation of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio **C,,** which was calculated for each fuel cycle using the transport methodology discussed in Section 6 2, accounts for the change in sensor reaction rates caused by variations in flux level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, **C,** is normally taken to be 1.0. However, for multiple-cycle irradiations, particularly those employing low leakage fuel management, the additional **C,** term should be employed. The impact of changing flux levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low leakage to low leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel cycle specific neutron flux values used in the time history corrections for Capsules X, W, V, and S as well as the individual sensor reaction rates used in the time history corrections for Capsule Y are listed in Table D-2. These values represent the cycle dependent results at the radial and azimuthal center of the respective capsules at the axial elevation of the active fuel midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, corrections were made to the 238 U measurements to account for the presence of 235 U impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation Corrections were also made to the both the 238 U and 237 Np sensor reaction rates to account for gamma ray induced fission reactions that occurred over the course of the capsule irradiations. The correction factors applied to the Surry Unit 2 fission sensor reaction rates are summarized as follows:

These factors were applied in a multiplicative fashion to the decay corrected uranium fission sensor reaction rates.

Results of the sensor reaction rate determinations for Capsules X, W, V, S, and Y are given in Table D-3. In Table D-3, the computed reaction rates for each sensor indexed to the radial center of the capsule are listed The fission sensor reaction rates as listed include the applied corrections for 238 U impurities, plutonium build-in, and gamma ray induced fission effects.

D 1.2 Least Squares Evaluation of Sensor Sets

Least squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as ϕ (E > 1.0 MeV) or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum In general, the least squares methods, as applied to surveillance capsule dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$
R_{i} \pm \delta_{R_{i}} = \sum_{g} (\sigma_{ig} \pm \delta_{\sigma_{ig}}) (\phi_{g} \pm \delta_{\phi_{g}})
$$

relates a set of measured reaction rates, $R₁$, to a single neutron spectrum, ϕ_g , through the multigroup dosimeter reaction cross-section, σ_{ig} , each with an uncertainty δ . The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least squares evaluation of the Surry Unit 2 surveillance capsule dosimetry, the FERRET code^[D-2] was employed to combine the results of the plant specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters (ϕ (E > 1.0 MeV) and dpa) along with associated uncertainties for the two in-vessel capsules withdrawn to date.

The application of the least squares methodology requires the following input:

- 1 The calculated neutron energy spectrum and associated uncertainties at the measurement location
- 2 The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.
- 3 The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the Surry Unit 2 application, the calculated neutron spectrum was obtained from the results of plant specific neutron transport calculations described in Section 6.2 of this report. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section D. 1.1. The dosimetry reaction cross-sections and uncertainties were obtained from the Sandia National Laboratory Radiation Metrology Laboratory (SNLRML) dosimeter cross-section library^[D-3]. The SNLRML library is an evaluated dosimetry reaction cross-section compilation recommended for use in LWR evaluations by ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)".

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum were input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties followed the guidance provided inASTM Standard E 944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance."

The following provides a summary of the uncertainties associated with the least squares evaluation of the Surry Unit 2 surveillance capsule sensor sets.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least squares evaluation:

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross-sections used in the least squares evaluations were taken from the SNLRML library. This data library provides reaction cross-sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross-sections and uncertainties are provided in a fine multigroup structure for use in least squares adjustment applications. These cross-sections were compiled from the most recent cross-section evaluations and they have been tested with respect to their accuracy and consistency for least squares evaluations. Further, the library has been empirically tested for use-in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the Surry Unit 2 surveillance program, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrun

The neutron spectra input to the least squares adjustment procedure were obtained directly from the results of plant specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$
M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg}
$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and Rg. specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$
P_{gg'} = [I - \theta J \delta_{gg'} + \theta e^{H}
$$

where

$$
H=\frac{(g-g')^2}{2\gamma^2}
$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when $g = g'$, and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the Surry Unit 2 calculated spectra was as follows:

D.1.3 Comparisons of Measurements and Calculations

Results of the least squares evaluations of the dosimetry from the Surry Unit 2 surveillance capsules withdrawn to date are provided in Tables D-4 and D-5. In Table D-4, measured, calculated, and best-estimate values for sensor reaction rates are given for each capsule. Also provided in this tabulation are ratios of the measured reaction rates to both the calculated and least squares adjusted reaction rates. These ratios of M/C and M/BE illustrate the consistency of the fit of the calculated neutron energy spectra to the measured reaction rates both before and after adjustment. Also included in the tabulation are the results of the X²/Degree of freedom statistical test associated with each of the least squares evaluations. In Table D-5, comparison of the calculated and best estimate values of neutron flux $(E > 1.0 \text{ MeV})$ and iron atom displacement rate are tabulated along with the BE/C ratios observed for each of the capsules.

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 $\frac{1}{2}$

The data comparisons provided in Tables D-4 and D-5 show that the adjustments to the calculated spectra are relatively small and well within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections. Further, these results indicate that the use of the least squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. From Section 6.4 of this report, it may be noted that the uncertainty associated with the unadjusted calculation of neutron fluence $(E > 1.0 \text{ MeV})$ and iron atom displacements at the surveillance capsule locations is specified as 12% at the 1σ level. From Table D-5, it is noted that the corresponding uncertainties associated with the least squares adjusted exposure parameters have been reduced to 6-7% for neutron flux ($E > 1.0$ MeV) and 6-8% for iron atom displacement rate. Again, the uncertainties from the least squares evaluation are at the 1σ level.

Further comparisons of the measurement results with calculations are given in Tables D-6 and D-7. These comparisons are given on two levels. In Table D-6, calculations of individual threshold sensor reaction rates are compared directly with the corresponding measurements. These threshold reaction rate comparisons provide a good evaluation of the accuracy of the fast neutron portion of the calculated energy spectra. In Table D-7, calculations of fast neutron exposure rates in terms of $\phi(E > 1.0 \text{ MeV})$ and dpa/s are compared with the best estimate results obtained from the least squares evaluation of the capsule dosimetry results. These two levels of comparison yield consistent and similar results with all measurement-to-calculation comparisons falling well within the 20% limits specified as the acceptance criteria in Regulatory Guide 1.190.

In the case of the direct comparison of measured and calculated sensor reaction rates, the M/C comparisons for fast neutron reactions range from 0.80-1.16 for the 23 samples included in the data set. The overall average M/C ratio for the entire set of Surry Unit 2 data is 0.97 with an associated sample standard deviation of 10.5%.

In the comparisons of best estimate and calculated fast neutron exposure parameters, the corresponding BE/C comparisons for the capsule data sets range from $0.84-1.01$ for neutron flux (E > 1.0 MeV) and from 0.85-0.98 for iron atom displacement rate. The overall average BE/C ratios for neutron flux (E > 1.0 MeV) and iron atom displacement rate are 0.95 with a sample standard deviation of 6.9% and 0.94 with a samole standard deviation of 6.0%, respectively.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 6.2 of this report are validated for use in the assessment of the condition of the materials comprising the beltline region of the Surry Unit 2 reactor pressure vessel.

Table D-1

Nuclear Parameters Used In The Evaluation Of Neutron Sensors

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Notes: The 90% response range is defined such that, in the neutron spectrum characteristic of the Surry Unit 2 surveillance capsules, approximately 90% of the sensor response is due to neutrons in the energy range specified with approximately 5% of the total response due to neutrons with energies below the lower limit and 5% of the total response due to neutrons with energies above the upper limit.

Table D-2

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ϕ (E > 1.0 MeV) [n/cm²-s] at the Surveillance Capsule Center Core Midplane Elevation

Table D-2 (Continued)

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Sensor Reaction Rates [rps/a] at the Surveillance Capsule Center Core Midplane Elevation

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Table D-3

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Measured Sensor Activities And Reaction Rates

Table D-4

Comparison of Measured, Calculated, and Best Estimate Reaction Rates At The Surveillance Capsule Center

		Reaction Rate [rps/atom]			
Reaction	Measured		²⁴²⁰ Best Estimate	$\bf MC$.	
⁶³ Cu(n, α) ⁶⁰ Co	5.87E-17	5.23E-17	5.84E-17	1.12	1.01
${}^{54}Fe(n,p)$ ⁵⁴ Mn	6.56E-15	5.79E-15	6.29E-15	1.13	1.04
58 Ni(n,p) ⁵⁸ Co	8.26E-15	7.96E-15	8.42E-15	1.04	0.98
$^{238}U(n,f)^{137}Cs$ (Cd)	2.75E-14	$2.82E-14$	2.90E-14	0.98	0.95
$^{237}Np(n,f)^{137}Cs$ (Cd)	1.89E-13	2.14E-13	2.01E-13	0.88	0.94
${}^{59}Co(n,\gamma){}^{60}Co$	3.94E-12	3.53E-12	3.93E-12	1.12	1.00

Capsule X (χ^2 /DOF = 0 25)

Capsule W (χ^2 /DOF = 0.27)

-3.26		Reaction Rate [rps/atom]	- 9		æ, 233
	leasured	Calculated	Best Estimate	M/C.	$2x + 2$
${}^{63}Cu(n,\alpha){}^{60}Co$	4.70E-17	4.05E-17	4.56E-17	1.16	1.03
${}^{54}Fe(n,p){}^{54}Mn$	4.39E-15	4.21E-15	4.40E-15	1.04	1.00
58 Ni(n,p) ⁵⁸ Co	5.92E-15	5.74E-15	5.95E-15	1.03	0.99
$^{238}U(n,f)^{137}$ Cs(Gd)	1.79E-14	1.95E-14	1.95E-14	0.92	0.92

Capsule V (χ^2 /DOF = 0.16)

Table D-4 cont'd

Comparison of Measured, Calculated, and Best Estimate Reaction Rates At The Surveillance Capsule Center

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Capsule S (χ^2 /DOF = 0.31)

Capsule Y (γ^2 /DOF = 0.36)

		Reaction Rate [rps/atom]		134	
Reaction	Jeasured		Best Estimate	M/C-	
⁶³ Cu(n, α) ⁶⁰ Co	3.21E-17	3.27E-17	3.15E-17	0.98	1.02
54 Fe(n,p) ⁵⁴ Mn	3.04E-15	3.35E-15	3.12E-15	0.91	0.97
²³⁸ U(n,f) ¹³⁷ Cs (Cd)	1.36E-14	1.54E-14	1.46E-14	0.88	0.93
$^{237}Np(n,f)^{137}Cs$ (Cd)	1.28E-13	1.11E-13	1.17E-13	1.15	1.09
${}^{59}Co(n,\gamma)$ ⁶⁰ Co	1.53E-12	1.68E-12	1.57E-12	091	0.97
${}^{59}Co(n,\gamma)$ ⁶⁰ Co (Cd)	1.06E-12	8.19E-13	1.03E-12	1.29	1.03

Table D-5

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Comparison of Calculated and Best Estimate Exposure Rates At The Surveillance Capsule Center

Table D-6

Comparison of Measured/Calculated (M/C) Sensor Reaction Rate Ratios Including all Fast Neutron Threshold Reactions

Notes⁻

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1. The overall average M/C ratio for the set of 23 sensor measurements is 0.97 with an associated sample standard deviation of 10.5%.

Table D-7

Comparison of Best Estimate/Calculated (BE/C) Exposure Rate Ratios

Appendix D References

- D-1. Regulatory Guide RG-1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U. S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.
- D-2. A Schmittroth, *FERRET Data Analysis Core*, *HEDL-TME 79-40*, *Hanford Engineering* Development Laboratory, Richland, WA, September 1979.
- D-3. RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium", July 1994.

ATTACHMENT 2

EVALUATION OF APPLICATION OF SURRY UNIT 2 CAPSULE Y SURVEILLANCE DATA TO SURRY UNIT 2 LOWER SHELL PLATE MATERIAL C4339-1 AND INTERMEDIATE TO LOWER SHELL CIRCUMFERENTIAL WELD MATERIAL R3008

ATTACHMENT 2

EVALUATION OF APPLICATION OF SURRY UNIT 2 CAPSULE Y SURVEILLANCE DATA TO SURRY UNIT 2 LOWER SHELL PLATE MATERIAL C4339-1 AND INTERMEDIATE-TO-LOWER SHELL CIRCUMFERENTIAL WELD MATERIAL R3008

BACKGROUND

Surry Unit 2 surveillance Capsule Y was withdrawn from the Surry Unit 2 reactor vessel on March 31, 2002. Capsule Y contains Surry Unit 2 lower shell plate material C4339-1 and intermediate-to-lower shell circumferential weld material R3008 (weld wire heat 0227).

This evaluation provides revised Surry Unit 2 data tables for the NRC's Reactor Vessel Integrity Database (RVID) and an evaluation of changes relative to the previous RVID update for Surry Unit 2 (1). The evaluation considers the impact of Surry Unit 2 Capsule Y
surveillance data on (a) licensing basis reactor coolant system (RCS) surveillance data on (a) licensing basis reactor coolant system (RCS) pressure/temperature (P/T) limit curves, (b) the associated Low Temperature Overpressure Protection System (LTOPS) setpoints and enabling temperature, (c) 10 CFR 50.61 Pressurized Thermal Shock (PTS) screening calculations, and (d) Charpy Upper Shelf Energy (CvUSE) values. The evaluation was performed in a manner consistent with applicable regulatory guidance. Reference Temperature for the Nil Ductility Transition (RTNDT) is performed in accordance with Regulatory Guide 1.99 Revision 2 (3), and the regulatory guidance provided in the meeting minutes from the November 12, 1997 NRC/Industry meeting on reactor vessel integrity (5). PTS screening calculations were performed in accordance with 10 CFR 50.61 (2). CvUSE values were developed in accordance with Regulatory Guide 1.99 Revision 2 (3). Evaluation results are presented in a format consistent with the data requirements of the NRC's Reactor Vessel Integrity Database (RVID).

DISCUSSION OF CHANGES TO PREVIOUSLY REPORTED INFORMATION

Surry Unit 2 revised RVID data tables are presented in Appendix A. Shaded cells in Appendix A indicate a changed value relative to those currently presented in RVID (Version 2.0.1, dated 7/6/00). The following changes have been incorporated into the revised tables:

Surry Unit 2 Lower Shell Plate material C4339-1 and Intermediate-to-Lower Shell Circumferential Weld Material R3008

- The RG 1.99 Revision 2 Position 2.1 chemistry factor (CF) calculation includes consideration of the capsule Y analysis results. The Capsule Y data are documented in Table 5-12 of Reference (4).

- Because the surveillance capsules were irradiated in a single reactor and the surveillance material was derived from a single source, irradiation temperature and chemistry corrections are not applied in the credibility determination.
- The surveillance data applicable to C4339-1were determined to be non-credible. However, the data were within 2σ of the RG 1.99 Rev. 2 Position 1.1 curve based on a CF for the average surveillance material chemical composition. Therefore, the Position 1.1 CF value was applied to the C4339-1 beltline material with a full margin term. The surveillance data for the R3008 material were determined to be credible, so the RG 1.99 Rev. 2 Position 2.1 CF values were applied for this material.

EVALUATION OF EXISTING P/T LIMITS AND LTOPS SETPOINTS

The existing Surry Units 1 and 2 P/T limits and LTOPS setpoints (7)(8) are based on a limiting 1⁄4-thickness (¼-T) RTNDT of 228.4°F. When the P/T limits and LTOPS setpoints were developed, this value of RTNDT was determined to bound all Surry Units 1 and 2 reactor vessel beltline materials at fluences corresponding to 28.8 EFPY and 29.4 EFPY for Surry Units 1 and 2, respectively (7)(8). RTNDT calculations have been performed for all Surry Unit 2 reactor vessel beltline materials at a neutron fluence value corresponding to 30.1 EFPY (1). The results are presented in Appendix A. After consideration of the aforementioned changes to previously reported information, the most limiting **1/-T** RTNDT value for Surry Unit 2 is 208.8°F at the fluence value corresponding to a cumulative core burnup of 30.1 EFPY (1). However, the P/T Limits and LTOPS Setpoints are based on a limiting $4-$ T RTNDT value of 228.4 \degree F. Therefore, the existing P/T Limits and LTOPS Setpoints remain valid and conservative.

EVALUATION OF PTS SCREENING CALCULATIONS

PTS screening calculations have been performed for all Surry Unit 2 reactor vessel beltline materials at a neutron fluence value corresponding to 30.1 EFPY (1). The results of these calculations are presented in Appendix A. After consideration of the aforementioned changes to previously reported information, it is concluded that all Surry Unit 2 beltline materials continue to meet the 10 CFR 50.61 screening criteria.

REPORT OF CvUSE VALUES

CvUSE data and calculations are presented in Appendix A. Although surveillance Capsule Y only contained Surry Unit 2 lower shell plate material C4339-1 and intermediate-to-lower shell circumferential weld material R3008 (weld wire heat 0227), the CvUSE table reflects the following changes:

- The current listing in the RVID for weld material L737/4275 (nozzle-to-intermediate shell circumferential weld) identifies the material as Linde 80 material when in fact it consists of SAF 89 material.

- The current listing in the RVID for weld material R3008/0227 (intermediate-to-lower shell circumferential weld) identifies the material as Linde 80 material when in fact it consists of Grau Lo material.
- 14-T USE fluence values are calculated using the RG 1.99 Rev. 2 Position 1.1 attenuation equation. The RG 1.99 Rev. 2 Position 1.1 methodology includes the vessel clad material thickness in the fluence calculation (i.e., "...depth into the vessel wall measured from the inner (wetted) surface."):

 $f/f_{surf} = exp (-0.24x), x = [0.25[*]vessel thickness] + [clad thickness]$

For Surry Unit 2, the maximum allowable f/f_{surf} would be:

 $x = [0.25*8.079 \text{ in}] + [0.157 \text{ in}] = 2.177 \text{ in}, \frac{f}{f_{\text{surf}}} = 0.593$

However, Surry Unit 2 attenuation was calculated by the more conservative approach of calculating wall depth by taking 25% of the total including the clad thickness:

 $x = [0.25*(8.079 \text{ in} + 0.157 \text{ in})] = 2.059 \text{ in}$, $\frac{f}{f_{\text{surf}}} = 0.610$

The higher f/f_{suff} produces a higher \triangle RTNDT, so the 1/4-T fluences calculated for Surry Unit 2 are conservative with respect to the RG 1.99 Rev. 2 Position 1.1 methodology.

- The values for unirradiated USE for materials C4331-2 and C4339-1 appeared to have been mistyped in the RVID. Corrected values have been provided.
- The percentage drops in CvUSE values were calculated using the RG 1.99 Rev. 2 Position 1.2 methodology. CvUSE data obtained from surveillance capsules compares favorably with predictions. For those Rotterdam and Linde 80 materials that are below 50 ft-lbs, equivalent margin analyses (EMAs) have been previously approved in References (9) and (10).

CONCLUSIONS

After consideration of the aforementioned changes to previously reported information, the most limiting 14-T RTNDT value for Surry Unit 2 is 208.8°F at a fluence value corresponding to a cumulative core burnup of 30.1 EFPY. The existing Surry Unit 2 Technical Specification RCS P/T limits, LTOPS setpoints, and LTOPS enabling temperature are based upon a $1/4$ -T RTNDT value of 228.4°F (7) (8). Therefore, the analyses supporting the Surry Unit 2 RCS P/T limits, LTOPS setpoints, and LTOPS enabling temperature remain valid and conservative (7) (8). In addition, after consideration of the aforementioned changes to previously reported information, all Surry Unit 2 reactor vessel beltline materials continue to meet the 10 CFR 50.61 PTS screening criteria for cumulative core burnups up to 30.1 EFPY. Finally, calculated Surry Unit 2 CvUSE values continue to be greater than the 50 ft-lb 1 OCFR50 Appendix G criterion.

NRC REACTOR VESSEL INTEGRITY DATABASE

Virginia Power requests that information presented in Appendix A be used to update the NRC Reactor Vessel Integrity Database (RVID).

FUTURE CAPSULE EXTRACTION PLANS

The currently docketed reactor vessel materials surveillance program includes withdrawal of the final Surry Unit 2 surveillance capsule at a fluence value corresponding to a cumulative core burnup of 30.1 EFPY. As a result of Dominion's license renewal efforts for Surry Unit 2, a submittal is planned to change the capsule withdrawal schedule to reflect the recommendations of the Generic Aging Lessons Learned (GALL) report (6).

REFERENCES

- (1) Letter from L. N. Hartz to USNRC, "Virginia Electric and Power Company, North Anna Power Station Units 1 and 2, Surry Power Station Units 1 and 2, Evaluation of Reactor Vessel Materials Surveillance Data," Serial Number 99-452A dated November 19, 1999.
- (2) Title 10, Code of Federal Regulations, Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."
- (3) Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May, 1988.
- (4) WCAP-16001, "Analysis of Capsule Y from Dominion Surry Unit 2 Reactor Vessel Radiation Surveillance Program," dated February 2003.
- (5) Memorandum from K. R. Wichman to E. J. Sullivan, "Meeting Summary for November 12, 1997 Meeting with Owners Group Representatives and NEI Regarding Review of Responses to Generic Letter 92-01, Revision 1, Supplement 1 Responses," dated November 19, 1997.
- (6) NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," dated July 2001.
- (7) Letter from R. F. Saunders to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Request for Exemption - ASME Code Case N-514, Proposed Technical Specifications Change, Revised Pressure/Temperature Limits and LTOPS Setpoint," Serial No. 95-197, June 8,1995.
- (8) Letter from B. C. Buckley to J. P. O'Hanlon, "Surry Units 1 and 2 Issuance of Amendments Re: Surry Units 1 and 2 Reactor Vessel Heatup and Cooldown

Curves (TAC Nos. M92537 and M92538)," Serial No. 96-020, dated December 28, 1995.

- (9) BAW-2178PA, "Low Upper Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads," dated April 1994.
- (10) BAW-2192PA, "Low Upper Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level A & B Conditions," dated April 1994.

APPENDIX A

REACTOR VESSEL MATERIALS DATA TABLES FOR SURRY UNIT 2

 $(11$ pages)

Facility: Surry Unit 1 Vessel Manufacturer: B&W and Rotterdam Dockyard

* 1/4T ART value of 228 4 F was used In the determination of PIT limits

Note Shaded cells indicate a changed value relative to the NRC's Reactor Vessel Integrity Database (RVID) Version 2 0 1 (Data Update on 7/6/00)

Facility: Surry Unit 2 **Vessel Manufacturer:** B&W and Rotterdam Dockyard

* 1/4-T ART value of 228 4 F was used In the determination of P/T limIts

Note Shaded cells indicate a changed value relative to the NRC's Reactor Vessel Integrity Database (RVID) Version 2 0 1 (Data Update on 716/00)

CvUSE Values Facility: Surry Unit 2 Vessel Manufacturer: B&W and Rotterdam Dockyard

Note Shaded cells indicate a changed value relative to the NRC's Reactor Vessel Integrity Database (RVID) Version 2 0 1 (Data Update on 7/6/00)

Table 2: Surry Unit 2 Plate Material C4339-1 (Combined Longitudinal and Transverse Data)

Table 3: Surry Unit 2 Plate Material C4339-1 (Combined Longitudinal and Transverse Data)

*For credibility check, measured shlft values are adjusted to average surveillance matenat chemistiy and irradiation temperature as required See Table 4

Table 4: Surry Unit 2 Plate Material C4339-1 (Combined Longitudinal and Transverse Data)

CF Determination

'Measured shift values are adjusted to the average surveillance matenal chemistry and irradiation temperature, and are venfied to be within 2 sigma of the trend curve based on RG 1 99 Rev 2 Position 11

** If surveillance data are non-credible but the Pos. 1 1 CF is shown to be conservative, the lower of the Pos 1.1 and Pos 2.1 chemistry factors is applied to the beitline material with a full margin term.

If surveillance data are non-credible and the Pos 11 CF is shown to be non-conservative, the greater of the Pos 1.1 and Pos 21 chemistry factors is applied to the beltline matenal with a full margin term

Credibility Assessment

(1) For the credibility determination, a temperature correction is not applied to measured values of transition temperature shift **if** applicable surveillance data were irradiated in a single reactor **(i** e , were irradiated at a similar temperature)

(2) For the credibility determination, a chemistry correction Is not applied to measured values of transition temperature shift **If** applicable surveillance data were obtained from a single source (I e, were machined from the same block of material)

(3) For determination of the beitline material chemistry factor, a temperature correction Is not applied to measured values of transition temperature shift **if** applicable surveillance data were irradiated in the reactor vessel which is being evaluated (i e, were irradiated at a similar temperature) A temperature correction is applied only in the conservative direction

(4) For determination of the beltline material chemistry factor, a chemistry correction (I e ,ratio procedure) Is not applied to measured values of transition temperature shift if the chemical composition of applicable surveillance data Is essentially Identical to the best-estimate chemical composition of the beltline material being evaluated

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Table 5: CvUSE Data Surry Unit 2 Plate Material C4339-1 (Combined Longitudinal and Transverse Data)

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Table 2: Surry Unit 2 Weld Material R3008

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Table 3: Surry Unit 2 Weld Material R3008

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*For credibility check, rneasured shift values are adjusted to average surveillance matenal chemstry and irradiation temperature as required See Table 4

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Table 4: Surry Unit 2 Weld Material R3008

CF Determination

*Measured shift values are adjusted to the average surveillance material chemistry and irradiation temperature, and are venfied to be within 2 sigma of the trend curve based on RG 1 99 Rev 2 Position 1 1

** If surveillance data are non-credible but the Pos 1.1 CF is shown to be conservative, the lower of the Pos 1.1 and Pos 21 chemistry factors is applied to the beltline matenal with a full margin term.

If surveillance data are non-credible and the Pos 11 CF is shown to be non-conservative, the greater of the Pos 11 and Pos 21 chemistry factors is applied to the beltline material with a full margin term

Credibility Assessment

(1) For the credibility determination, a temperature correction Is not applied to measured values of transition temperature shift If applicable surveillance data were Irradiated In a single reactor (I e, were irradiated at a similar temperature)

(2) For the credibility determination, a chemistry correction is not applied to measured values of transition temperature shift if applicable surveillance data were obtained from a single source (i e, were machined from the same block of material).

(3) For determination of the beltline material chemistry factor, a temperature correction is not applied to measured values of transition temperature shift if applicable surveillance data were Irradiated in the reactor vessel which is being evaluated (i e, were irradiated at a similar temperature) A temperature correction is applied only in the conservative direction

(4) For determination of the beitline matenal chemistry factor, a chemistry correction **(i** e ,ratio procedure) is not applied to measured values of transition temperature shift if the chemical composition of applicable surveillance data Is essentially Identical to the best-estimate chemical composition of the beltline material being evaluated

Table 5: CvUSE Data Surry Unit 2 Weld Material R3008

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Table 2: Surry Unit 1 and 2 Weld Material SA-1585 (Point Beach 1 Data Only)

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Table 3: Surry Unit 1 and 2 Weld Material SA-1585 (Point Beach I Data Only)

* For credibility check, measured shift values are adjusted to average surveillance material chemistry and irradiation temperature as required See Table 4.

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Table 4: Surry Unit 1 and 2 Weld Material SA-1585 (Point Beach 1 Data Only)

CF Determination

'Measured shift values are adjusted to the average surveillance materlal chemistry and Irradiation temperature, and are venfied to be within 2 sigma of the trend curve based on RG 1 99 Rev. 2 Position 1. 1.

** If surveillance data are non-credible but the Pos 11 CF is shown to be conservative, the lower of the Pos 1.1 and Pos 2.1 chemistry factors is applied to the beltline material with a full margin term.

If surveillance data are non-credible and the Pos 11 CF is shown to be non-conservative, the greater of the Pos 1.1 and Pos 2.1 chemistry factors is applied to the beltline material with a full margin term

Credibility Assessment

(1) For the credibility determination, a temperature correction is not applied to measured values of transition temperature shift if applicable surveillance data were irradiated In a single reactor (I e , were irradiated at a similar temperature)

(2) For the credibility determination, a chemistry correction is not applied to measured values of transition temperature shift if applicable surveillance data were obtained from a single source (i e, were machined from the same block of material)

(3) For determination of the beitline material chemistry factor, a temperature correction is not applied to measured values of transition temperature shift if applicable surveillance data were irradiated in the reactor vessel which is being evaluated (i e, were irradiated at a similar temperature) A temperature correction is applied only in the conservative direction

(4) For determination of the beitline material chemistry factor, a chemistry correction (i e, ratio procedure) is not applied to measured values of transition temperature shift if the chemical composition of applicable surveillance data is essentially identical to the best-estimate chemical composition of the beitline material being evaluated.

Table 5: CvUSE Data Surry Unit 1 and 2 Weld Material SA-1585 (Point Beach 1 Data Only)

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