WEC PROPRIETARY CLASS 3

Westinghouse Energy Systems

WESTINGHOUSE PROPRIETARY CLASS 3

WCAP-13257

EVALUATION OF PRESSURIZED THERMAL SHOCK FOR 32 AND 34 EFPY FOR KEWAUNEE

M. A. Ramirez
J. M. Chicots

March 1992

Work Performed Under Shop Order KFIP-108

Prepared by Westinghouse Electric Corporation
for the Wisconsin Public Service Corporation

Approved by: T. A. Meyer
T. A. Meyer, Manager
Structural Reliability & Plant Life Optimization

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Division
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1. INTRODUCTION

A limiting condition on reactor vessel integrity known as Pressurized Thermal Shock (PTS) may occur during a severe system transient such as a loss-of-coolant-accident (LOCA) or a steam line break. Such transients may challenge the integrity of a reactor vessel under the following simultaneous conditions:

- severe overcooling of the inside surface of the vessel wall followed by high repressurization;
- significant degradation of vessel material toughness caused by radiation embrittlement; and
- the presence of a critical-size defect in the vessel wall.

In 1985 the Nuclear Regulatory Commission (NRC) issued a formal ruling on Pressurized Thermal Shock. It established the screening criteria for pressurized water reactor (PWR) vessel embrittlement as measured by the nil-ductility reference temperature, termed RT_{PTS}[1]. RT_{PTS} screening values were set for beltline axial welds, forgings and plates and for beltline circumferential weld seams for end-of-license plant operation. The screening criteria was determined using conservative fracture mechanics analysis techniques. All PWR vessels in the United States have been required to evaluate vessel embrittlement in accordance with the criteria through their end-of-license. The Nuclear Regulatory Commission has amended its regulations for light water nuclear power plants to change the procedure for calculating radiation embrittlement. The revised PTS Rule was published in the Federal Register, May 15, 1991 with an effective date of June 14, 1991[4]. This amendment makes the procedure for calculating RT_{PTS} values consistent with the methods given in Regulatory Guide 1.99, Revision 2[2].

The purpose of this report is to determine the Pressurized Thermal Shock reference temperature RT_{PTS} values for the Kewaunee reactor vessel to address the Pressurized Thermal Shock (PTS) Rule. Section 2 discusses the Rule and its requirements. Section 3 provides the methodology for calculating RT_{PTS}. Section 4 provides the reactor vessel beltline region material properties for the Kewaunee reactor vessel. The neutron fluence values used in this analysis are presented in Section 5. The results of the RT_{PTS} calculations are presented in Section 6. The conclusions and references for the PTS evaluation follow in Sections 7 and 8, respectively.

2. PRESSURIZED THERMAL SHOCK

The PTS Rule requires that the PTS submittal be updated whenever there are changes in core loadings, surveillance measurements or other information that indicates a significant change in projected RT_{PTS} values.

The Rule outlines regulations to address the potential for PTS events on pressurized water reactor (PWR) vessels in nuclear power plants that are operated with a license from the United States Nuclear Regulatory Commission (USNRC). PTS events have been shown from operating experience to be transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may result in the propagation of flaws postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The Rule establishes the following requirements for all domestic, operating PWRs:

* All plants must submit projected values of RT_{PTS} for reactor vessel beltline materials by giving values for time of submittal, the expiration date of the operating license, and the projected expiration date if a change in the operating license or renewal has been requested. This assessment must be submitted by six months after the effective date of this Rule if the value of RT_{PTS} for any material is projected to exceed the

screening criteria. Otherwise, it should be submitted with the next update of the pressure-temperature limits, or the next reactor vessel surveillance report, or 5 years from the effective date of this Rule, whichever comes first. These values must be calculated based on the methodology specified in this rule. The submittal must include the following:

1) the bases for the projection (including any assumptions regarding core loading patterns),
2) copper and nickel content and fluence values used in the calculations for each beltline material. (If these values differ from those previously submitted to the NRC, justification must be provided.)

* The RT_{PTS} (measure of fracture resistance) screening criteria for the reactor vessel beltline region is:

  270°F for plates, forgings, axial welds, and
  300°F for circumferential weld materials

* The following equations should be used to calculate the RT_{PTS} values for each weld, plate or forging in the reactor vessel beltline.

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

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

* All values of RT_{PTS} must be verified to be bounding values for the specific reactor vessel. In doing this each plant should consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and surveillance results. Results from the plant-specific surveillance program shall be integrated into the embrittlement estimate if,

  (i) The plant-specific surveillance data has been deemed credible as defined in Regulatory Guide 1.99, Rev.2 [2], and
  (ii) The RT_{PTS} values change significantly. (Changes to RT_{PTS} values are considered significant if the value determined with RT_{PTS} equations (1) and (2), or that using capsule data, or both, exceed the screening criteria prior to the expiration of the operating license, including any renewed term, if applicable, for the plant.)

* Plant-specific PTS safety analyses are required before a plant is within 3 years of reaching the screening criteria, including analyses of alternatives to minimize the PTS concern.

* NRC approval for operation beyond the screening criteria is required.

3. METHOD FOR CALCULATION OF RT_{PTS}

In the PTS Rule, the NRC Staff has selected a conservative and uniform method for determining plant-specific values of RT_{PTS} at a given time.

For the purpose of comparison with the screening criteria, the value of RT_{PTS} for the reactor vessel must be calculated for each weld and plate or forging in the beltline region as shown below.

$$RT_{PTS} = I + M + \Delta RT_{PTS},$$

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

I = Initial reference temperature in °F (RT_{NDT}) of the unirradiated material. Measured values must be used if credible values are available.

M = Margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures.

M = 66°F for welds and 48°F for base metal if generic values of I are used.
M = 56°F for welds and 34°F for base metal if measured values of I are used.

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

CF = Chemistry factor in °F from Tables[4] for welds and for base metal (plates and forgings).
If plant-specific surveillance data has been deemed credible per Reg. Guide 1.99, Rev. 2, it may be considered in the calculation of the chemistry factor.

4. VERIFICATION OF PLANT-SPECIFIC MATERIAL PROPERTIES

Before performing the Pressurized Thermal Shock evaluation, a review of the latest plant-specific material properties was performed.

The beltline region is defined by the PTS Rule[4] to be "the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage." Figure 1, shown in the next page, identifies and indicates the location of all beltline region materials for the Kewaunee reactor vessel.

A summary of the pertinent chemical and mechanical properties of the beltline region plate and weld materials of the Kewaunee reactor vessel are given below in Table 1.

TABLE 1
KEWAUNEE REACTOR VESSEL
BELTLINE REGION MATERIAL PROPERTIES[3]

Material Description	CU (%)	NI (%)	I-RTNDT (°F)
Intermediate Shell, B6306	0.06	0.71	60
Lower Shell, B6307	0.06	0.75	20
Circumferential Weld *	0.283	0.745	-56

* % Weight Copper and Nickel values were determined per method described in Appendix A.

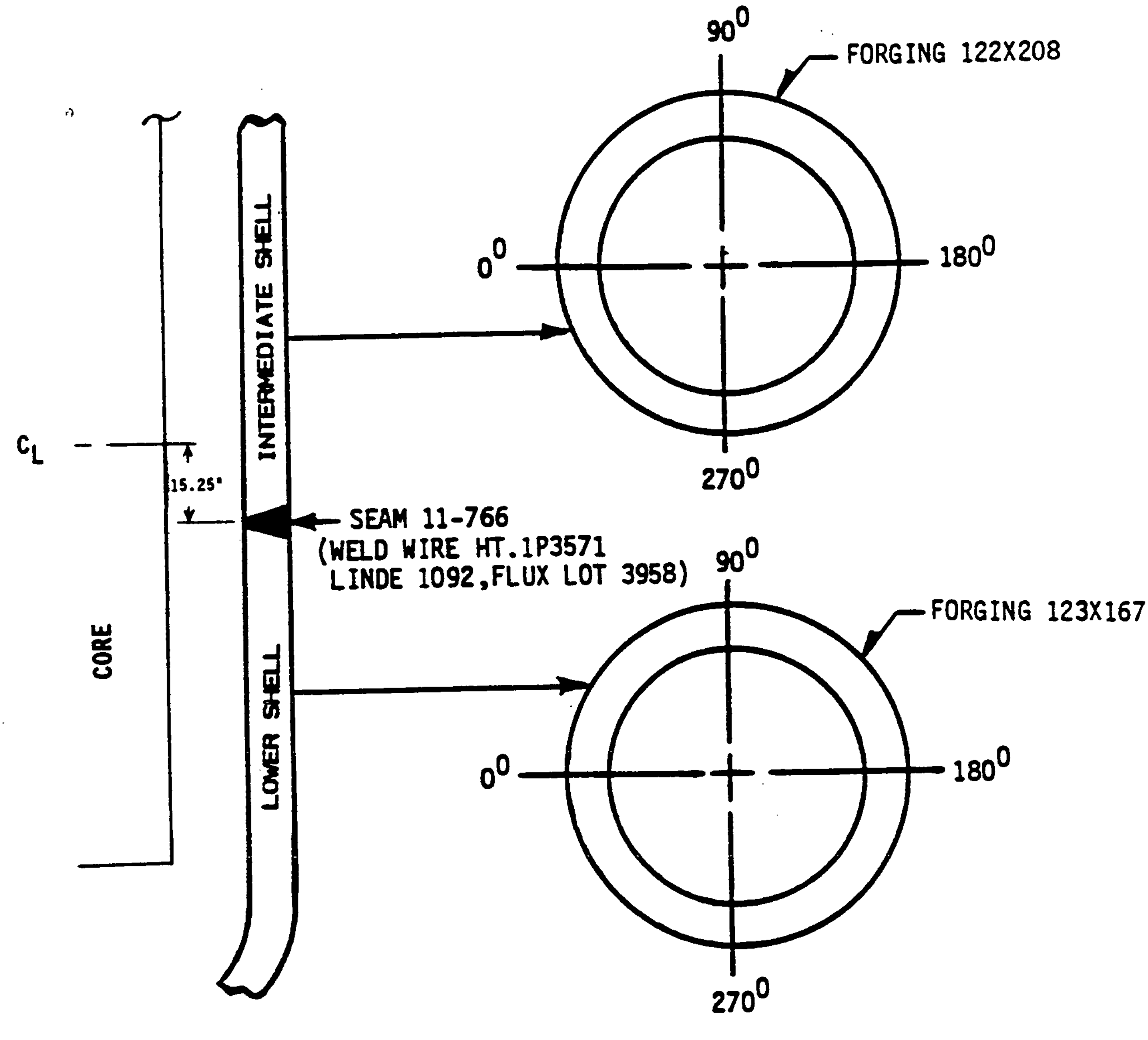

Figure 1. Identification and Location of Beltline Region Material for the Kewaunee Reactor Vessel

5. NEUTRON FLUENCE VALUES

The calculated fast neutron fluences (E>1 MeV) at the inner surface of the Kewaunee reactor vessel are shown below in Table 2 for 32 and 34 EFPY. These values were projected based on the recent implementation of the flux reduction core 1 design at fuel cycle 16 through cycle 22 and the future implementation of flux reduction core 2 design at cycle 23 and all subsequent fuel cycles. The neutron exposure projections presented in Table 2 were calculated based on the neutron dosimetry results from the first three surveillance capsules withdrawn from the Kewaunee reactor vessel integrated with the analytically derived exposure values obtained from a Sn transport analysis performed for the Kewaunee reactor vessel. The discrete ordinates calculations were performed in R,θ geometry using both the forward and adjoint methodologies with a P_3 cross section approximation and an S_8 order of angular quadrature. The power distributions used in the plant specific analyses represented cycle averaged relative assembly powers, burnups, and axial peaking factors. In constructing the cycle specific energy dependent source distributions account was taken of the burnup dependent inventory of fissionable isotopes, including U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242. It should be noted that the neutron exposure projections presented in Table 2 were multiplied by a factor of 1.167 to adjust for biases observed between the cycle specific calculations and the results of neutron dosimetry for the first three surveillance capsules removed from the Kewaunee reactor vessel. The factor of 1.167 was derived by taking the average of the measurement to calculation (M/C) flux ratios obtained from the dosimetry results of capsules V, R, and P removed from the Kewaunee reactor vessel.

TABLE 2

NEUTRON EXPOSURE PROJECTIONS* AT KEY LOCATIONS ON THE KEWAUNEE PRESSURE VESSEL CLAD/BASE METAL INTERFACE FOR 32 AND 34 EFPY

EFPY	0°	15°	30°	45°
32	3.17	2.22	1.89	1.70
34	3.31	2.33	2.00	1.81

*Fluence x 10^{19} n/cm^2 (E>1.0 MeV)

6. DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Kewaunee reactor vessel as a function of 32 and 34 EFPY fluence values.

In determining the chemistry factor for Kewaunee, both, the Tables in 10CFR Part 50.61 [4] and the surveillance capsule data were used. The use of plant specific surveillance capsule data arises because of the following reason:

> There have been three capsules removed from the Kewaunee reactor vessel (capsules V, R and P), hence the data is credible per Regulatory Guide I.99, Revision 2, and the surveillance capsule materials are representative of the Kewaunee reactor vessel materials.

Surveillance capsule data was available for the weld metal only, therefore, only the chemistry factor for the circumferential weld was calculated using the surveillance capsule data as shown in Table 3.

Comparing the chemistry factor value of 191.27°F obtained using the surveillance capsule data to the chemistry factor value of 210.675°F obtained using the Tables in Reference 4, it can be seen the latter value is higher. This chemistry factor value was used in determining the RT_{PTS} value for the weld metal, since it was more conservative. Note, the RT_{PTS} values for the weld metal determined using the surveillance capsule data are shown with a ** in Tables 4 and 5, shown in page 10.

As stated previously, the fluence data were projected based on the neutron dosimetry results from the first three surveillance capsules integrated with the analytical predictions performed for the Kewaunee reactor vessel, taking into account any measurement to calculation biases. The fluence projection was based on the actual implementation of flux reduction core design 1 at cycle 16 through cycle 22 and on the assumption that flux reduction core design 2 will be implemented at cycle 23 and all subsequent fuel cycles. Tables 4 and 5 provide a summary of the RT_{PTS} values for all beltline region materials for 32 and 34 EFPY, respectively, using the PTS Rule.

TABLE 3

CALCULATION OF CHEMISTRY FACTOR USING KEWAUNEE SURVEILLANCE CAPSULE DATA[3]

BELTLINE MATERIAL	CAPSULE	CAPSULE FLUENCE (n/cm^2) *	FF	DRTNDT AT 30 FT-LB (°F)	FF*DRTNDT	(FF)^2
Circ. Weld	V R P	0.559 2.070 2.890	0.837 1.198 1.282	175 235 230	146.53 281.55 294.81 -------- 722.89	0.70 1.44 1.64 ------ 3.78

* Capsule fluence x 10^{19} n/cm^2 (E>1.0 MeV)

CHEMISTRY FACTOR FOR CIRC. SEAM WELD = 722.89 / 3.78 = $\boxed{191.27}$

TABLE 4
RT_{PTS} VALUES FOR KEWAUNEE FOR 32 EFPY

Material	ΔRT_{NDT}(°F) (CF x FF*)	+	Initial RT_{NDT} (°F)	+	Margin (°F)	=	RT_{PTS} (°F)
Intermediate Shell	48.24		60		34		142.24
Lower Shell	48.24		20		34		102.24
Circ. Weld **	274.67		-56		66		284.67

TABLE 5
RT_{PTS} VALUES FOR KEWAUNEE FOR 34 EFPY

Material	ΔRT_{NDT}(°F) (CF x FF*)	+	Initial RT_{NDT} (°F)	+	Margin (°F)	=	RT_{PTS} (°F)
Intermediate Shell	48.61		60		34		142.61
Lower Shell	48.61		20		34		102.61
Circ. Weld **	276.79		-56		66		286.79

* Fluence factor based upon peak inner surface neutron fluence of 3.17×10^{19} and 3.31×10^{19} , respectively for 32 and 34 EFPY, with M/C bias.

** RTpts values for circumferential weld using surveillance capsule data were calculated as 259.37 °F and 261.29 °F, for 32 and 34 EFPY respectively.

7. CONCLUSIONS

As shown in Tables 4 and 5 all the RT_{PTS} values for the intermediate and lower shells and the weld metal remain below the NRC screening criteria for PTS using the projected fluence values for 32 and 34 EFPY. A plot of the RT_{PTS} values versus the fluence is shown in Figure 2 for the most limiting material in the Kewaunee reactor vessel beltline region, the circumferential weld 11-766. This plot includes RT_{PTS} values determined using both, the Tables in Reference [4] and the surveillance capsule data.

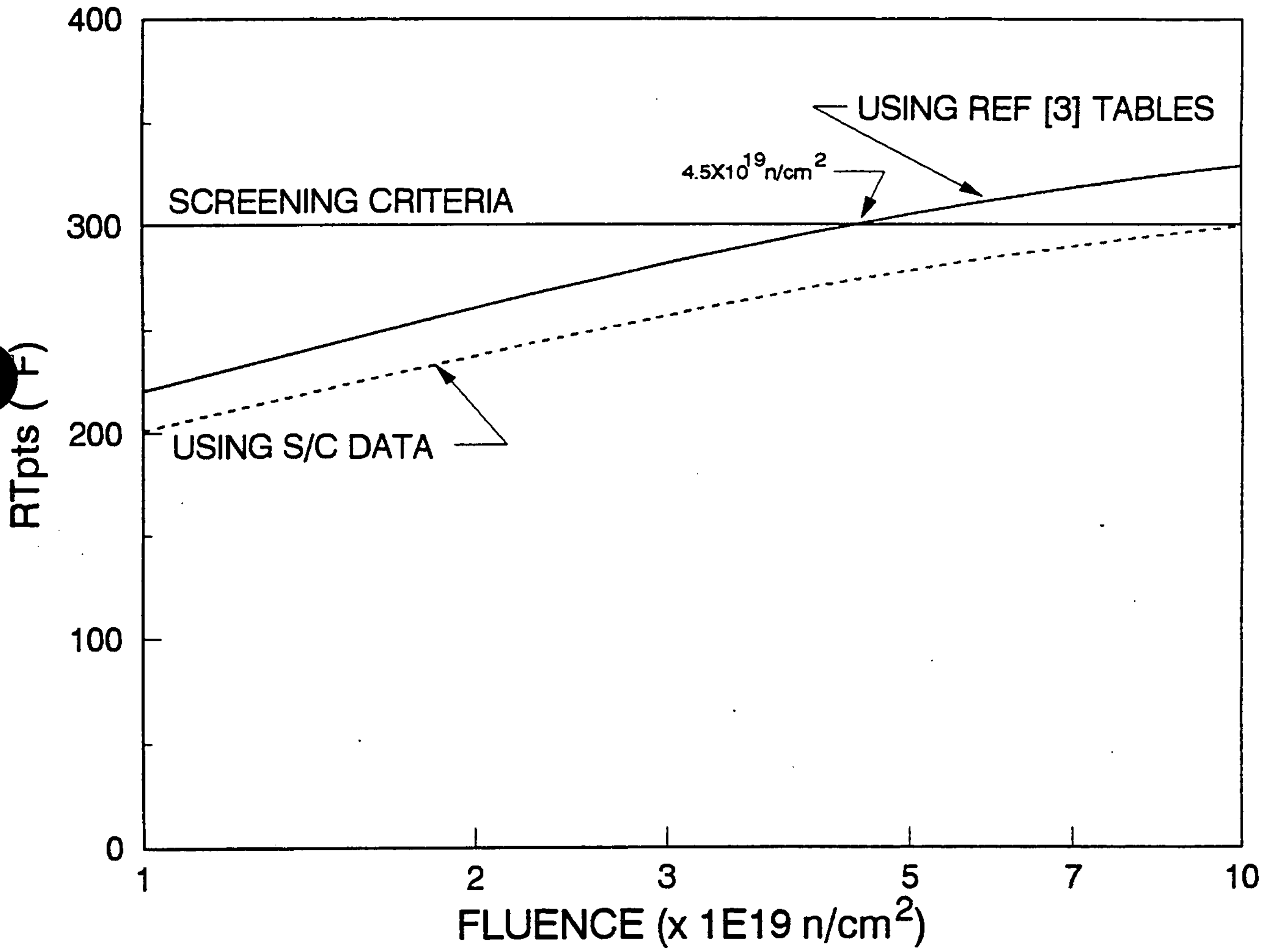

Figure 2. RT_{PTS} versus Fluence Curves for Kewaunee Limiting Material - Circumferential Weld, 11-766.

8. REFERENCES

[1] 10CFR Part 50, "Analysis of Potential Pressurized Thermal Shock Events," July 23, 1985.

[2] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

[3] WCAP-12020, "Analysis of Capsule P from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program", S.E. Yanichko, et al., November 1988. (Westinghouse Proprietary Class III).

[4] 10CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", May 15, 1991. (PTS Rule)

APPENDIX A

Recently, Westinghouse has completed the chemical analysis on four irradiated Charpy V-notch specimens from capsules V and R from the Kewaunee plant [A-1] and two specimens from the capsule at the 253° location from the Maine Yankee plant [A-2]. These six specimens were analyzed by the Inductively Coupled Plasma Spectrometry (ICPS) method. A review of the data indicates that welds made with weld wire heat IP3571 can result in a variation of copper content. This heterogeneity in copper content is believed to have occurred because of a variation in weld wire copper coating thicknesses from the front end to back end of a welding wire reel and a variation in coating thickness between reels of wire. These additional data along with the chemical analysis measurements from Westinghouse Data Base for Kewaunee beltline region circumferential weld and the four previously tested irradiated Charpy V-notch specimens are tabulated in Table A-1. With the additional data, the mean as-deposited chemistry values for copper and nickel content, which are obtained by averaging the eighteen data points, are 0.283 Wt % and 0.745 Wt % respectively.

TABLE A-1

CHEMICAL ANALYSIS MEASUREMENTS

	Source	Cu (wt %)	Ni (wt %)
	Additional Data		
ICPS	Kewaunee Surveillance Capsule "V"	0.214	0.816
	Kewaunee Surveillance Capsule "V"	0.434	0.800
	Kewaunee Surveillance Capsule "R"	0.066	0.736
	Kewaunee Surveillance Capsule "R"	0.207	0.769
	Maine Yankee Surveillance Capsule 253°	0.432	0.745
	Maine Yankee Surveillance Capsule 253°	0.356	0.728
	Existing Data		
ICPS	Kewaunee Surveillance Capsule "P"	0.18	0.74
ICPS	Kewaunee Surveillance Capsule "P"	0.35	0.74
ICPS	Kewaunee Surveillance Capsule "P"	0.19	0.73
ICPS	Kewaunee Surveillance Capsule "P"	0.17	0.72
	Kewaunee Unirradiated Surveillance Weld	0.20	0.77
	CE Weld Qualification M 1.42 (Simple Wire)	0.40	0.82
	CE Weld Qualification M 1.43 (Tandem Wire)	0.37	0.75
	Maine Yankee Surveillance (ETI Report CR 75-269)	0.36	0.78
X-ray Fluorescence Spectrometry	Maine Yankee Surveillance Capsule 263° (BCL Rpt. 585-21)	0.25	0.66
	Maine Yankee Surveillance Capsule 263° (BCL Rpt. 585-21)	0.25	0.70
	Maine Yankee Surveillance Capsule 263° (BCL Rpt. 585-21)	0.33	0.71
	Maine Yankee Surveillance Capsule 263° (BCL Rpt. 585-21)	0.33	0.70
	SIMPLE AVERAGE	0.283	0.745

References

[A-1] Analytical Report 14570, "Metals Analysis of Low Alloy Steel", from Westinghouse Electric Corporation AES Analytical Laboratories, February 14, 1992.

[A-2] WCAP-12819, "Analysis of the Maine Yankee Reactor Vessel Second Wall Capsule Located at 253°", E. Terek, et al., March 1991. (Westinghouse Proprietary Class III).