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SUBJECT: Forwards response to request for addl info dtd 960614 re reactor pressure vessel structural integrity.

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August 12, 1996

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Reactor Pressure Vessel Structural Integrity

References:
1) Letter from CR Steinhardt (WPSC) to Document Control Desk (US NRC) dated April 28, 1995
2) Letter form CR Steinhardt (WPSC) to Document Control Desk (US NRC) dated April 30, 1996
3) Letter from RJ Laufer (US NRC) to ML Marchi (WPSC) dated June 14, 1996
4) Letter from RJ Laufer (US NRC) to ML Marchi (WPSC) dated June 18, 1996
5) Letter from Maine Yankee Atomic Power Company [License No. DPR-36, Docket No. 50-309] to US NRC, titled, "Unirradiated Properties of Maine Yankee Vessel Materials and Analysis and Evaluation of First Surveillance Capsule," dated October 22, 1975

By letters dated April 28, 1995, and April 30, 1996 Wisconsin Public Service Corporation (WPSC) submitted proposed amendments to modify Kewaunee Nuclear Power Plant (KNPP) Technical Specification limits relating to heatup, cooldown, and low temperature overpressure protection (LTOP). In references 3 and 4, the Nuclear Regulatory Commission (NRC) informed WPSC that the NRC staff has reviewed the subject information and finds that additional information is required to complete their reviews. The June 14, 1996 letter requested additional information regarding reactor vessel and surveillance capsule fluence. The June 18, 1996 letter requested material property data for IP 3571 Linde 1092 welds. This letter transmits WPSC's response to your RAI's.

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WPSC contracted Westinghouse Electric Corporation, (the vendor that developed the fluence model/methodology used to analyze the KNPP reactor vessel surveillance capsules and determine the reactor pressure vessel fluence) to develop a response to the RAI dealing with pressure vessel fluence. Attachment 1 provides Westinghouse's response.

WPSC's response to the RAI relating to Charpy impact and drop weight test data from IP 3571 Linde 1092 welds is provided in Attachment 2.

If you have any questions or require additional information, please contact a member of my staff.

Sincerely,

Clark R. Steinhardt
Senior Vice President - Nuclear Power

CAT/jmf

Attach.

cc - US NRC Region III
US NRC Senior Resident Inspector

Subscribed and Sworn to
Before Me This 12th Day
of August 1996

Notary Public, State of Wisconsin

My Commission Expires:
June 13, 1999

ATTACHMENT 1

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

August 12, 1996

Response to NRC Request For Additional Information
Dated June 14, 1996
Regarding Pressure Vessel Fluence

Request 1.0

In Ref. 1 (page 6-7) it is stated that; "The FERRET approach used the reaction rate and the calculated neutron energy spectrum at the geometric center of the specimens as input and proceeded to adjust apriori calculated group fluxes to produce a best fit (in a least squares sense) to the reaction rate data. The exposure parameters along with the associated uncertainties were then obtained from the adjusted spectra."

Request 1.1

Would the group flux adjustments be required if plant specific sources were utilized in the forward calculation?

WPSC Response

Yes. Regardless of the input used in the transport calculation there are uncertainties associated with the calculated results. In the case of a plant specific neutron transport calculation, the overall uncertainties are due to the individual uncertainty components inherent in the transport cross-sections, core source definition, reactor geometry, and methods approximations. Likewise, uncertainties exist in measured reaction rates and neutron dosimetry cross-sections. In performing the adjustment with the FERRET code, the input variables (measured reaction rates, calculated energy dependent neutron flux, energy dependent dosimeter reaction cross-sections) are adjusted within the context of their respective uncertainties to provide a "best estimate" of the neutron damage parameters ($\Phi(E > 1.0$ MeV), $\Phi(E > 0.1$ MeV), dpa) at the measurement location.

As noted in ASTM Standard E 944 - 89, "Adjustment methods provide a means for combining the results of neutron transport calculations with neutron dosimetry measurements in order to obtain optimal estimates for neutron damage exposure parameters with assigned uncertainties. The inclusion of measurements reduces the uncertainties for these parameter values and provides a test for consistency between measurements and calculations and between different measurements."

It is further stated in ASTM Standard E 944 - 89, "The algorithms of the adjustment codes tend to decrease the variances of the adjusted data compared to the corresponding input values. The least squares adjustment codes yield estimates for the output data with minimum variances that is, the "best estimates." This is the primary reason for using these adjustment procedures."

Request 1.2

Having done the group flux adjustments was the forward calculation repeated to obtain the proper spectra?

WPSC Response

No. As noted in the reply to 1.1, the adjustments performed by FERRET are designed to obtain a "best estimate" neutron exposure at the measurement point within the combined uncertainties of the calculated spectrum, reaction rate measurements, and dosimetry cross-sections. Unlike a code system such as LEPRICON, the FERRET procedure does not perform adjustments at reactor locations remote from the measurement point, such as within the pressure vessel wall. Likewise, the cross-sections used in the plant specific transport calculation itself are not adjusted. Therefore, other than an indication of the bias that may exist between the "best estimate" neutron exposure and the transport codes purely analytical prediction of the neutron exposure at the measurement location, the FERRET evaluation provides no information that could be used to change the neutron energy spectrum calculated with the transport code.

Request 2.0

In Ref. 1 (page 6-8) it is stated that: "In the FERRET evaluations, a log-normal least squares algorithm weights both the a priori values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values f are related to the flux by some response matrix A...."

Request 2.1

Are the uncertainties pre-assigned? How are the elements of matrix A determined?

WPSC Response

For each neutron sensor contained in a surveillance capsule dosimetry set, the relationship among the measured reaction rate, R, the neutron energy spectrum, $\phi(E)$, and the sensor reaction cross-section, $\sigma(E)$, may be expressed as follows:

$$R \pm \Delta_R = \int (\Phi(E) \pm \Delta_\Phi(E))(\sigma(E) \pm \Delta_\sigma(E))dE$$

where: R = The measured reaction rate for an individual neutron sensor.

Δ_R = The uncertainty associated with the measured reaction rate.

$\Phi(E)$ = The calculated energy dependent neutron flux at the location of the neutron sensor.

$\Delta_\Phi(E)$ = The energy dependent uncertainty associated with the calculated neutron spectrum.

$\sigma(E)$ = The energy dependent reaction cross-section for an individual neutron sensor.

$\Delta_\sigma(E)$ = The energy dependent uncertainty associated with the individual neutron sensor reaction cross-sections.

The neutron dosimetry sets irradiated in the Kewaunee surveillance capsules include sensors incorporating the following reactions:

Cu-63 (n,α) Co-60
Fe-54 (n,p) Mn-54
Ni-58 (n,p) Co-58
U-238 (n,f) Cs-137 Cd Covered
Np-237 (n,f) Cs-137 Cd Covered
Co-59 (n,γ) Co-60
Co-59 (n,γ) Co-60 Cd Covered

Therefore, the response matrix considered in the adjustment procedure includes the seven reaction rate equations corresponding to the individual sensors included in the dosimetry set.

In establishing the input to the response matrix, the reaction rate data are obtained from counting laboratory measurements for each of the sensors irradiated in the surveillance capsule. Prior to input to the adjustment procedure, the measured data are corrected to account for the plant specific flux/power history applicable to the capsule location, for radioactive decay following the end of irradiation of the capsule, and for the presence of competing reactions in individual sensors.

The energy dependent neutron flux at the dosimetry location is obtained directly from the results of the plant specific neutron transport calculations applicable to the irradiation of each individual capsule. The relative energy distribution at the location of the dosimetry set is obtained from the reference forward transport calculation. The magnitude of the neutron flux is obtained from the results of fuel cycle specific adjoint transport analyses. In performing the adjoint calculations, the effects of plutonium fissioning are included for individual fuel assemblies within each fuel cycle comprising the total irradiation period. In this respect, each fuel cycle is treated independently and no attempt is made to establish an average core source over multiple fuel cycles. The intent of this combined forward/adjoint calculation is to provide an analytical prediction for direct comparison with measured reaction rates.

The energy dependent dosimetry cross-sections input to the adjustment procedure are obtained directly from the evaluated nuclear cross-section data files. In this case, the ENDF/B-V dosimetry cross-sections were employed. This cross-section data base was the latest dosimetry cross-section library available at the time of the Kewaunee evaluation.

The uncertainty associated with the measured reaction rates include components due to the basic measurement process, the irradiation history/radioactive decay corrections, and the corrections for competing reactions within the U-238 and Np-237 fissile sensors. In developing the overall reaction rate uncertainties, the component due to sensor counting is derived from the expected accuracy using the appropriate ASTM counting standard. In general, counting times can be chosen so as to make the random statistical variation in the counting rate negligible compared to other counting uncertainties such as counting geometry and detector calibration, etc. The uncertainty component due to irradiation history/radioactive decay includes the effects of product half-lives, target abundance in the sensor material, and the product yield in the fission monitors.

The uncertainties in the calculated neutron energy spectrum at the measurement location are input to the least squares adjustment procedure in the form of a covariance matrix generated from the relationship discussed in the reply to item 4.0 of this RAI. As noted in the reply to item 4.0, the uncertainty matrix includes an overall normalization uncertainty; i.e, potential bias, as well as individual correlated group uncertainties propagated over a range of energies.

The uncertainties in the dosimetry reaction rate cross-sections, in the form of variances and covariances are obtained from the master dosimetry cross-section library along with the energy dependent cross-sections themselves. Data are provided for all of the reactions included in the Kewaunee surveillance capsule dosimetry sets.

Request 2.2

Which cross-sections does FERRET adjust, how and why?

WPSC Response

As noted in the reply to 2.1, the only cross-sections adjusted in the FERRET procedure are the reaction cross-sections for the individual sensors contained in the dosimetry package. The adjustment algorithm modifies the entire set of input data (dosimetry cross-sections, calculated group fluxes, measured reaction rates) within the constraints of the uncertainties associated with each of the input variables to provide a "best estimate" neutron spectrum consistent with the set of seven reaction rate equations defined by the sensor set package. The mathematical algorithm in current adjustment codes, including FERRET, tries to make the adjustments as small as possible relative to the uncertainties in the corresponding input data. Again as noted in ASTM Standard E 944 - 89, "The more recent codes like STAYS'L, FERRET, LEPRICON, and LSL-M2 are based explicitly on statistical principles such as "Maximum Likelihood Principle" or "Bayes Theorem" which are generalizations of the well-known least squares principle."

Request 3.1

How have the plant specific uncertainties related to the radial and azimuthal location of the dosimeters been evaluated? Where are they reported?;

and,

Request 3.2

How have the uncertainties related to the core and vessel geometry been evaluated? Where are they reported?

WPSC Response

The use of the bias factors derived from the plant specific measurement data base acts to remove plant specific biases associated with the definition of the core source, actual vs. assumed reactor dimensions, and operational variations in water density within the reactor. As a result, the

overall uncertainty in the best estimate exposure projections within the vessel wall depends on the individual uncertainties in the measurement process, the uncertainty in the dosimetry location, and, in the uncertainty in the calculated ratio of the neutron exposure at the point of interest to that at the measurement location.

The uncertainty in the derived neutron flux for an individual measurement is obtained directly from the results of a least squares evaluation of dosimetry data. The least squares approach combines individual uncertainty in the calculated neutron energy spectrum, the uncertainties in dosimetry cross-sections, and the uncertainties in measured foil specific activities to produce a net uncertainty in the derived neutron flux at the measurement point. The associated uncertainty in the plant specific bias factor, K, derived from the C/M data base, in turn, depends on the total number of available measurements as well as on the uncertainty of each measurement.

In the case of the Kewaunee dosimetry evaluations the following is a summary of the bias factor development for $\Phi(E > 1.0 \text{ MeV})$ from the results of the four surveillance capsule sensor sets withdrawn to date.

	C/M Ratio	% Std Deviation
Capsule V	0.862	8
Capsule R	0.871	10
Capsule P	0.872	8
Capsule S	1.012	8
Average	0.904	10

Thus, at the measurement locations, the uncertainty in the projected best estimate fast neutron fluence is estimated to be $\pm$ 10% at the 1σ level.

In developing the overall uncertainty associated with the exposure of the pressure vessel wall, additional uncertainty is included to account for the fact that the vessel wall is removed from the measurement points and its exposure is impacted by factors including downcomer water density variations and the actual and vs as-built vessel inner radius that do not directly impact the measurements. In evaluating these additional components of the overall uncertainty at the pressure vessel wall, the impact of positioning uncertainties for dosimetry within the surveillance capsules is taken from parametric studies of sensor position performed as part a series of analytical sensitivity studies included in the qualification of the methodology. The uncertainties

in the exposure ratios relating dosimetry results to positions within the vessel wall are again based on the analytical sensitivity studies of the, downcomer water density variations and vessel inner radius tolerance. Thus, this portion of the overall uncertainty is controlled entirely by dimensional tolerances associated with the reactor design and by the operational characteristics of the reactor.

For the exposure evaluation for the pressure vessel wall, the net uncertainty in the bias factor, K, is combined with the uncertainty from the analytical sensitivity study to define the overall fluence uncertainty in the best estimate fluence projections.

The individual components of that overall uncertainty are summarized as follows:

C/M Bias Factor	10%
Dosimetry Position	4%
Downcomer Water Temp.	4%
Vessel Inner Radius	5%

Combining these uncertainty components via the square root of the sum of the squares results in a net uncertainty of $\pm$ 13% (1σ) to be applied to the best estimate exposure projections for the Kewaunee reactor pressure vessel. This level of uncertainty is well within the requirement of $\pm$ 20% (1σ) associated with the PTS rule.

Request 4.0

Why is the expression for $M_{gg'}$ (Ref. 1, top of page 6-9) applicable to the Kewaunee measurements and how was the Mearker development justified for Kewaunee?

WPSC Response

The expression for $M_{gg'}$ does not apply to the surveillance capsule measurements. Rather, the $M_{gg'}$ relationship is used to assign a covariance matrix to the calculated neutron spectrum obtained from the plant specific transport analysis.

The covariance matrix for the input trial spectrum is constructed from the following relation:

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

where R_n specifies an overall fractional normalization uncertainty for the input neutron flux solution. The fractional uncertainties R_g and $R_{g'}$ specify additional random uncertainties on a groupwise basis for the input spectrum. These groupwise uncertainties are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\ \delta_{gg'} + \theta\ e^{-H}$$

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 short range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1 when g = g' and 0 otherwise. For the trial spectrum used in the current evaluations, a short range correlation of $\gamma = 6$ groups was used. This choice implies that neighboring groups are strongly correlated when θ is close to 1.

The reference to the publication by R. E. Maerker was intended to show that this approach to the construction of a covariance matrix applicable to the calculated neutron spectrum is consistent with the approach used by Maerker in his evaluations of the PCA, ANO-1, and H. B. Robinson benchmark analyses. In these analyses, Maerker provided information regarding the covariances associated with the calculated spectra in each of these environments and suggested that the conclusions were generally applicable to LWR analysis. The approach used in the Kewaunee evaluations is consistent with these views.

Request 5.0

Please provide the results of the measurements before the FERRET adjustments for capsules V, R, P, and S. Also provide the values of the vessel fluence based on the V, R, and P capsules before the update that is shown in this submittal is performed.

WPSC Response

The results of the reaction rate measurements prior to the application of the FERRET adjustment procedure are provided in Tables 6-18 through 6-21 of WCAP-14279, "Analysis of Capsule S from the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program," March 1995 for Capsules V, R, P, and S, respectively. These measured reaction rates are repeated here as follows:

	Measured Reaction Rate [rps/atom]			
	Capsule V	Capsule R	Capsule P	Capsule S
Cu-63(n,α)Co-60	7.21e-17	5.84e-17	4.74e-17	4.39e-17
Fe-54(n,p)Mn-54	8.58e-15	7.33e-15	5.24e-15	4.61e-15
Ni-58(n,p)Co-58	1.07e-14	9.74e-15	6.38e-15	6.55e-15
U-238(n,f)Cs-137	Cd 5.18e-14	4.64e-14	2.82e-14	2.34e-14
Np-237(n,f)Cs-137	Cd 5.02e-13		2.50e-13	1.76e-13
Co-59(n,γ)Co-60	1.03e-11	8.94e-12	4.49e-12	4.01e-12
Co-59(n,γ)Co-60	Cd 4.23e-12	2.16e-12	1.71e-12	1.76e-12

As noted in the reply to Item 3.2 of this RAI, the current evaluation, including all four surveillance capsules withdrawn to data, resulted in an average C/M ratio of 0.904 $\pm$ 10%. The use of this ratio, in turn, resulted in a bias factor [1/(C/M)] of 1.11 $\pm$ 10% being applied to the calculated fluence to produce "best estimate" fluence values at the pressure vessel surface.

If the Capsule S results were not included in the evaluation and the bias factor was determined solely on the results of the Capsules V, R, and P dosimetry, the resultant average C/M ratio would be 0.868 $\pm$ 11% corresponding to a bias factor of 1.15 $\pm$ 11%. This bias factor calculated without the Capsule S data agrees within 4% with the bias factor obtained from the entire four capsule data base. The agreement is well within the statistical uncertainty associated with the individual bias factors themselves.

The following tabulation provides a comparison of the 34 EFPY neutron fluence (E > 1.0 MeV) at the pressure vessel inner radius based on the application of bias factors of 1.11, 1.15, and 1.00. The value of 1.11 represents the application of the complete four capsule measurement data base, the value of 1.15 represents the application of a three capsule measurement data base excluding Capsule S, and the value of 1.00 represents a fluence evaluation based on the use of the ENDF/B-VI transport calculation with no impact of the dosimetry.

	$\Phi(E > 1.0\ MeV)\ [n/cm^2]$		
Azimuthal Angle	Bias = 1.11	Bias = 1.15	ENDF/B-VI Calculation
0.0°	3.49e+19	3.63e+19	3.15e+19
15.0°	2.32e+19	2.42e+19	2.10e+19
30.0°	1.94e+19	2.02e+19	1.75e+19
45.0°	1.69e+19	1.76e+19	1.53e+19

Considering that the total uncertainty in the fluence projections at the pressure vessel wall (bias uncertainty + reactor geometry uncertainty) for the two cases where the plant specific measurements were utilized is on the order of 13%-14% and that the uncertainty in the analytical result without benefit of plant specific measurement is on the order of 17%-18%, the above tabulation is statistically consistent; all of the projection values fall within roughly 1σ of each other. However, the first column of data (Bias = 1.11), based on all of the available measurement and calculational information, is considered to represent the "best estimate" exposure projection for the Kewaunee pressure vessel wall.

Request 6.0

It is reported in Ref. 1 that for capsules V, R, and P the C/M is estimated at 0.86, 0.87, and 0.87, respectively. For capsule S it is estimated at 1.01. It is reported in Ref. 2 that M/C = 1.167 (page 7) or C/M = 0.86, that is, about the same as in Ref. 1. Assuming that the difference in V, R, and P versus S (in Ref. 1) is due to the application of ENDF/B-VI cross-sections and the presence of the thermal shield, why did the recalculation of V, R, and P using ENDF/B-VI leave the C/M unchanged.

WPSC Response

As presented in WCAP-14279, all four capsules (V, R, P, and S) were evaluated with the same analytical techniques and the same transport and dosimetry cross-section libraries. Therefore, the observed differences in the C/M ratios is not impacted by the methodology itself. Rather, these are real observed differences in the measurements. As noted in the reply to Item 3.2, when the 1σ uncertainty of 8%-10% associated with each of the C/M ratios is considered, the four measurements form a statistically consistent data base with all of the data falling within $\pm$ 12% of the average of 0.904. This is well within the $\pm$ 20% range that is suggested as acceptable in DG-1053 (previously DG-1025).

The reason for no apparent change in the C/M ratios for Capsules V, R, and P between the two submittals is due primarily to a change in the treatment of axial peaking factors used in the neutron transport calculations performed for the two evaluations. In the prior analysis, the axial peaking factor was conservatively taken as 1.2 for all fuel cycles. In the current evaluation, cycle specific peaking factors ranging from 1.07 to 1.12 were utilized in the transport calculations. Therefore, from a purely analytical standpoint, when the same axial peaking factors are employed the current ENDF/B-VI calculations would exceed the prior ENDF/B-IV analysis by approximately 9%. An increase of this magnitude is consistent with the expected comparisons between ENDF/B-IV and ENDF/B-VI neutron transport calculations for reactors with a thermal shield internals design.

ATTACHMENT 2

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

August 12, 1996

Response to NRC Request For Additional Information
Dated June 18, 1996
Regarding Charpy Impact and Drop Weight Test Data
From IP 3571 Linde 1092 Welds

NRC Request 1

Provide all Charpy impact and drop weight test data from IP 3571 Linde 1092 welds that were fabricated by ABB/CE from sister plant surveillance welds, weld qualification, etc.

WPSC Response

Charpy impact and drop weight test data from IP 3571 Linde 1092 welds only exists for the Kewaunee Nuclear Power Plant and the Maine Yankee Nuclear Plant. Reference 1 transmitted to the NRC the results of both the drop weight and Charpy impact testing for Kewaunee. Yankee Atomic previously provided the NRC staff with drop weight and Charpy impact test data in report CR-75-269 (reference 5).

A tabulation of the unirradiated Charpy impact test data from heat number IP 3571 weld wire is provided on the following page:

TABLE 1
Unirradiated
Charpy Impact Test Data

Kewaunee		Maine Yankee	
Test Temp (°F)	Impact Energy (ft-lbs)	Test Temp (°F)	Impact Energy (ft-lbs)
-200	3.0	-320	2.2
-200	5.5	-150	3.4
-200	7.0	-100	7.3
-150	36.0	-75	14.9
-150	15.0	-50	28.0
-150	32.0	-35	32.9
-100	11.5	-20	45.5
-100	13.0	0	45.2
-100	20.0	10	50.4
-40	40.0	10	52.6
-40	41.0	10	42.7
-40	29.5	30	78.9
10	66.0	30	59.2
10	67.0	30	59.8
10	55.5	71	90.3
40	79.0	150	105.0
40	83.0	250	106.8
40	45.0	350	109.4
75	97.5		
75	92.0		
75	102.5		
210	125.0		
210	126.5		
210	126.0		

Research of available drop-weight measurements from Linde 1092 type weld heats revealed the following data:

Table 2
Measured Drop Weight Values for CE Fabricated Linde 1092 Type Welds

Heat Number	Plant	Drop Weight Values (°F)	RT_{NDTU} (°F)	Source
IP 3571	Kewaunee	-50	-50	WCAP-14042 Rev. 1
IP 3571	Maine Yankee	-30	-30	CR 75-269
13253	DC Cook 1	-70	-70	WCAP-8047
13253	Salem 2	-40	-40	WCAP-8824
21935/12008	Diablo Canyon 2	-60	-60	CE Weld Test Report Contract No. 6268 dated May 14, 1970
305424	Beaver Valley 1	-60	-60	WCAP-8457
34B009	Millstone 1	-80	-77	NEDC-30299
20291/12008	McGuire 1	-50	-50	WCAP-9195
34B009	Palisades	-50	-50	April 12, 1995 letter from NRC to Palisades
W5214	Palisades	-20	-20	

NRC Request 2

Does the unirradiated reference temperature (RT_{NDTU}) value of -50°F bound all the data in Item 1? If it does not, evaluate the effect of using either a bounding or generic (RT_{NDTU}) on the pressure-temperature limits, low temperature overpressure protection limits, and pressurized thermal shock evaluation.

WPSC Response

10 CFR 50.61 requires the use of best estimate data for assessment of the structural integrity of the reactor pressure vessel. Combining data in Item 1 results in a mean RT_{NDTU} value of -50.7°F and a standard deviation of 17.4°F. These results show that the mean RT_{NDTU} and standard deviation for all Linde 1092 type welds is in good agreement with the generic mean (-56°F) and standard deviation (17°F) values for CE fabricated vessels. The KNPP initial reference temperature of -50°F is bounding when compared to both the generic RT_{NDTU} and mean value for all Combustion Engineering 1092 welds. Based on this data, WPSC concludes that an initial reference temperature of -50°F and σ_I value of 0°F is both conservative and appropriate for KNPP. 10 CFR 50.61 requires that KNPP use a margin term of 28°F, since the KNPP surveillance program is credible and measured values are being used for initial RT_{NDTU}.

WPSC has evaluated the effect of using a generic RT_{NDTU} on the pressure-temperature limits, low temperature overpressure protection limits, and pressurized thermal shock evaluation even though the unirradiated reference temperature of -50°F does bound both the generic RT_{NDTU} and mean value for all Linde 1092 welds. Two cases were evaluated: Case 1 utilized an initial RT_{NDTU} of -50°F and margin of 28°F, case 2 utilized an initial RT_{NDTU} of -56°F and margin of 44°F. The results of this evaluation are summarized below:

a) The LTOP enabling temperature is defined as RT_{NDT} plus 90°F. Code Case N514 relaxed this definition to a coolant temperature less than 200°F or a reactor vessel metal temperature less than RT_{NDT} plus 50°F, whichever is greater. Case 1 yields RT_{NDT} = 213°F and case 2 yields RT_{NDT} = 223. The enabling temperature could therefore be as high as 313°F. Comparison of the calculated allowable pressures listed in Table 3 from 70°F to 313°F show that the case 1 pressures are all within 110% of the case 2 pressures. If the case 2 data represents the valid Appendix G limit and Code Case N-514 is considered, the case 1 data is found to be acceptable since it limits the maximum

pressure in the vessel to 110% of the pressure determined to satisfy the Appendix G limit. The setpoint on the relief valve used for LTOP is currently set at 500 psig. Data for both case 1 and case 2 show that this setpoint is adequate. The most limiting situation is during an isothermal event at 70°F when the calculated allowable pressure is 589.5 psig. Under this situation, the calculated allowable pressure is 589.5 psig minus 87 psig [10 psig (piping losses, 33 psig (RXCP flow losses), 29 psig (instrument error), and 15 psig (valve over shoot)] which prevents violation of the Appendix G curve by a margin of 2.5 psig.

b) The RT_{PTS} screening criteria for a circumferential weld in the beltline region of the reactor vessel is 300°F. PTS calculations assuming end of life fluence were performed in accordance with 10 CFR 50.61 and found to be acceptable for both case 1 and case 2. The calculated RT_{PTS} values for case 1 and case 2 are 231°F and 241°F, respectively.

c) 10 CFR 50.61 requires that best estimate data be used for assessment of structural integrity of the reactor vessel. As explained above, an unirradiated reference temperature value of -50°F is bounding when compared to both the generic and best-estimate value for all 1092 welds. To evaluate the effect of using a bounding best-estimate RT_{NDTU} or generic RT_{NDTU} on the pressure-temperature limits, WPSC calculated allowable pressures for end of life fluence at discrete temperature intervals under steady state cooldown conditions. The results for case 1 and case 2 are shown in Table 3. These allowable combinations of pressure and temperature were independently verified by Westinghouse Electric Corporation. The values in Table 3 are slightly different from the equivalent values in the original submittal (WCAP-14278). This is because the values in Table 3 were obtained using the newly developed K solutions for Section XI of Appendix G. These solutions were approved by the ASME Code Main Committee in March 1996 and are being balloted by the Board on Nuclear Codes and Standards.

Table 3
Calculated Allowable Pressure(s)-Temperature(s) in Reactor Vessel Corresponding to Steady State Cooldown

Temperature °F	Pressure (psig)	
	Case 1 I = -50°F and M = 28°F	Case 2 I = -56°F and M = 44°F
70	593.9	589.5
95	608.3	601.9
100	611.8	605.0
112	621.4	613.3
120	621.4	619.7
122	621.4	621.4
140	621.4	621.4
180	621.4	621.4
>180	722.8	701.0
200	777.2	748.0
225	871.5	829.6
250	1007.1	946.9
263	1099.6	1027.0
273	1183.7	1100.0
300	1482.0	1357.7
313	1673.0	1522.9
325	1884.3	1705.7
341.2	2235.0 (operating pressure)	2008.8
350	2462.5	2205.8
350.8	2485.0 (design pressure)	2225.0
351.2	2496.1	2235.0 (operating pressure)
360.8	2785.1	2485.0 (design pressure)

The case 1 data allows operation of the reactor at higher pressures than the case 2 data. At 70°F, case 1 permits operation of the plant at 4 psig higher than case 2. At 341.2°F, the case 1 pressure is equal to the operating pressure at 100% power and is 226 psig higher than the corresponding case 2 pressure. In general, the calculated case 2 pressures vary from the case 1 pressures by less than 1% to about 10% with increasing temperatures.