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WISCONSIN PUBLIC SERVICE CORPORATION

600 North Adams • P.O. Box 19002 • Green Bay, WI 54307-9002

September 19, 1989

10 CFR 50.61

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

Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Pressurized Thermal Shock

- References:
- 1) Letter from D. C. Hintz (WPSC) to G. E. Lear (NRC) dated January 23, 1986
 - 2) Letter from D. C. Hintz (WPSC) to G. E. Lear (NRC) dated July 14, 1986
 - 3) Letter from D. L. Wiggington (NRC) to D. C. Hintz (WPSC) dated July 21, 1987
 - 4) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) dated February 13, 1989
 - 5) U. S. Nuclear Regulatory Commission Regulatory Guide 1.99 Revision 2 dated May 1988
 - 6) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) dated April 18, 1989
 - 7) Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC) dated May 12, 1989.
 - 8) Letter from J. G. Giitter (NRC) to C. R. Steinhardt (WPSC) dated May 26, 1989

By letters dated January 23, 1986 (reference 1) and July 14, 1986 (reference 2), Wisconsin Public Service Corporation (WPSC) provided the response required by the pressurized thermal shock (PTS) rule (10 CFR 50.61). The NRC responded in reference 3 stating that WPSC's reported values of copper and nickel content for the beltline circumferential weld were not acceptable, and that the RTPTS screening criterion would be reached prior to the end of licensed life based on the NRC chemistry values for the Kewaunee Nuclear Power Plant (KNPP).

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Reference 4 provided a summary of WPSC's current status and a list of long- and short-term options. WPSC also committed to performing a chemical analysis on the reactor vessel material specimen removed during the 1988 refueling outage at KNPP. The results of this chemical analysis reduced the best-estimate values for copper and nickel content of the circumferential weld to 0.28 wt.% and 0.74 wt.% respectively; this information was provided in reference 6. Reference 7 provided the NRC with the fluence projections based on a lower leakage core tentatively scheduled to be loaded next refueling. The NRC responded in reference 8 stating that the Kewaunee reactor pressure vessel meets the toughness requirements of 10 CFR 50.61 for at least 40 calendar years of operation.

The NRC evaluation in reference 8 was based on a more conservative estimate of fluence and not the estimate provided in reference 7. This was done because WPSC's fluence analysis had not been finalized and the report was still in draft form. This submittal transmits the final report (attachment 1) including the methodology used to calculate fluence. The projected fluence at 34 effective full power years (EFPY) for KNPP is 3.07×10^{19} n/cm². A value of 34 EFPY corresponds to the end of the KNPP licensed life, December 21, 2013. Using this fluence value, in combination with the material chemistry content provided in reference 6, results in a limiting RT_{P75} value of 265.8°F (using 10 CFR 50.61) for the beltline circumferential weld. This calculation, along with the calculation using the method described in NRC Regulatory Guide 1.99 Revision 2 (reference 5), is contained in attachment 2 to this letter.

This final report is being submitted in order to supply information necessary for the NRC to perform a thorough review of the updated fluence projections. As stated in reference 8, this report should provide the NRC with sufficient information to issue a supplemental safety evaluation on the increased RT_{P75} margin for KNPP. If you have any questions or need any additional information, please contact a member of my staff.

Sincerely,



K. H. Evers
Manager - Nuclear Power

PMF/jms

Attach.

cc - Mr. Robert Nelson, US NRC
US NRC, Region III

Attachment 1

To

Letter from K. H. Evers (WPSC) to Document Control Desk (NRC)

Dated

September 19, 1989

NEUTRON EXPOSURE EVALUATION FOR PTS EVALUATION

This section describes a discrete ordinates Sn transport analysis performed for the Kewaunee reactor to determine the neutron radiation environment within the reactor vessel on a fuel cycle specific basis for current and projected fuel management designs. Fast neutron exposure parameters in terms of fast neutron fluence ($E > 1.0$ MeV) and iron atom displacements (dpa) are established on a plant and fuel cycle basis for the first fifteen reactor operating fuel cycles as well as for the projected design of cycle 16; and, based on the results of these evaluations, projections of vessel exposure for future operating periods are estimated. Neutron dosimetry results from the first three surveillance capsules withdrawn from the Kewaunee reactor are integrated with the analytically derived exposure values to provide an overall "best estimate" of the current and future exposure of the pressure vessel.

The use of fast neutron fluence ($E > 1.0$ MeV) to correlate measured material properties changes to the neutron exposure of the material for light water reactor applications has traditionally been accepted for development of damage trend curves as well as for the implementation of trend curve data to assess vessel condition. In recent years, however, it has been suggested that an exposure model that accounts for differences in neutron energy spectra among surveillance capsule locations and positions within the pressure vessel wall could lead to an improvement in the uncertainties associated with damage trend curves as well as to a more accurate evaluation of damage gradients through the pressure vessel wall.

Because of this potential shift away from a threshold fluence toward an energy dependent damage function for data correlation, ASTM Standard Practice E853, "Analysis and Interpretation of Light Water Reactor Surveillance Results", recommends reporting iron atom displacements (dpa) along with fluence ($E > 1.0$ MeV) to provide a data base for future reference. The energy dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693, "Characterizing Neutron Exposures in Terms of Displacements per

Atom". The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the pressure vessel wall has already been promulgated in Revision 2 to Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials". Therefore, in keeping with the philosophy espoused in the current ASTM standards governing pressure vessel exposure evaluations, dpa data is also included in this report.

METHOD OF ANALYSIS

In performing the fast neutron evaluations for the Kewaunee reactor pressure vessel, two distinct sets of transport calculations were carried out. The first, a single computation in the conventional forward mode, was used primarily to obtain relative neutron energy distributions throughout the reactor vessel as well as to establish relative radial distributions of exposure parameters [fluence($E > 1.0$ MeV) and dpa] through the vessel wall. The neutron spectral information is required to determine exposure parameter ratios; i.e., dpa/fluence($E > 1.0$ MeV), within the pressure vessel geometry; while, the relative radial gradient information is required to permit the projection of cycle specific exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consisted of a series of adjoint analyses relating the fast neutron flux ($E > 1.0$ MeV) at several locations on the reactor vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses, when combined with cycle specific neutron source distributions, yielded absolute predictions of neutron exposure at the locations of interest for each of the fuel cycles designed for use in the Kewaunee reactor. It is important to note that the cycle specific neutron source distributions utilized in conjunction with the adjoint importance functions included not only spatial variations of fission rates within the reactor core; but, also accounted for the effects of varying neutron yield per fission and fission spectrum introduced by the build-in of plutonium as the burnup of individual fuel assemblies increased.

A plan view of the Kewaunee reactor geometry at the core midplane is shown in figure 1. Since the reactor exhibits 1/8 core symmetry only a 0-45 degree sector is depicted. In addition to the core, reactor internals, pressure vessel, and primary biological shield, the model also included explicit representations of the surveillance capsules attached to the thermal shield.

The forward transport calculation for the reactor model shown in figure 1 was carried out in R,Theta geometry using the Dot two-dimensional discrete ordinates code [1] and the SAILOR cross-section library [2]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In the current analysis anisotropic scattering was treated with a P3 expansion of the scattering cross-sections and the angular discretization was modeled with an S8 order of angular quadrature. This reference forward calculation was normalized to a core midplane power density characteristic of operation at a thermal power level of 1650 MWt.

The reference core power distribution utilized in the forward analysis was derived from statistical studies of long term operation of Westinghouse 2-loop plants. Inherent in the development of this reference core power distribution is the use of an out-in fuel management strategy; i.e., fresh fuel on the core periphery. Furthermore, for the peripheral fuel assemblies, a 2 sigma uncertainty derived from the statistical evaluation of plant to plant and cycle to cycle variations in peripheral power was added to the nominal assembly power level for all fuel assemblies adjacent to the core baffle plates. An axial peaking factor of 1.20 was also employed to scale the axially averaged power distribution to the midplane value. Since it is unlikely that a single reactor would have a power distribution at the nominal + 2 sigma level and would maintain an axial peaking factor of 1.20 for a large number of fuel cycles, the use of this reference case is expected to yield somewhat conservative results. This is especially true in cases where low leakage fuel management has been employed.

All adjoint analyses were also carried out using a P3 cross-section approximation from the SAILOR library and an S8 order of angular quadrature. Adjoint source locations were taken at the 0, 15, 30, and 45 degree azimuthal locations at the pressure vessel inner diameter. Here the angular orientation is relative to the core cardinal axes as shown in figure 1. Again these calculations were run in R,Theta geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case flux ($E > 1.0$ MeV). Having the importance functions and appropriate core power distributions, the response of interest could be calculated as:

$$\bar{\phi}(R_0, \theta_0) = \int_R \int_{\theta} \int_E I(R, \theta, E) S(R, \theta, E) R dR d\theta dE$$

where: $\bar{\phi}(R_0, \theta_0)$ = Neutron flux ($E > 1.0$ MeV) at the adjoint source location of radius R_0 and azimuthal angle θ_0

$I(R, \theta, E)$ = Adjoint importance function at radius R , azimuthal angle θ , and neutron source energy E

$S(R, \theta, E)$ = Neutron source strength at core location R, θ and energy E

Although the adjoint importance functions used in the Kewaunee analyses were based on a response function defined by the threshold neutron flux ($E > 1.0$ MeV), prior calculations have shown that, while the implementation of low leakage loading patterns significantly impact the magnitude and spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, for a given location the ratio of dpa/fluence ($E > 1.0$ MeV) is insensitive to changing core source distributions. In the application of these importance functions to the Kewaunee reactor, therefore, iron atom displacements were computed on a cycle specific basis by using dpa/fluence ($E > 1.0$ MeV) ratios from the forward analysis in conjunction with the cycle specific fluence ($E > 1.0$ MeV) solutions from the individual adjoint evaluations.

The power distributions used in the adjoint analyses represented cycle averaged relative assembly powers, burnups, and axial peaking factors. Therefore, the adjoint results provided data in terms of fuel cycle averaged neutron flux which, when multiplied by the appropriate fuel cycle length, in turn yielded the incremental fast neutron fluence for the fuel cycle. In constructing the cycle specific energy dependent source distributions account was taken of the burnup dependent inventory of fissioning isotopes, including U-235, U-238, Pu-239, Pu-240, Pu-241, and Pu-242.

The transport methodology, both forward and adjoint, using the SAILOR cross-section library, has been benchmarked against neutron dosimetry data obtained at the ORNL PCA facility [3]. Extensive comparisons of analytical predictions with measurements from power reactor surveillance capsules and reactor cavity dosimetry programs have also been made. The benchmarking studies indicate that the use of SAILOR cross-sections and the reference core power distribution produces flux levels that tend to be conservative by from 7-22%. When plant specific power distributions are used with the adjoint importance functions, the benchmarking studies show a tendency to underpredict fluence levels at surveillance capsule positions by from 5-10%; while calculations applicable to reactor cavity locations tend to be biased low by approximately 10-20% depending on the thickness of the pressure vessel. In performing the exposure evaluations for the Kewaunee reactor the comparisons of predictions with measurements obtained from previously withdrawn surveillance capsules (V, R, and P) were factored into the overall analysis to provide best estimate exposure levels with a minimum uncertainty.

FAST NEUTRON EXPOSURE RESULTS

Best estimate fast neutron ($E > 1.0$ Mev) exposure results for the Kewaunee reactor are presented in tables 1 through 5. In these tabulations data is presented at several azimuthal locations around the circumference of the reactor vessel for the axial elevation of the vessel girth weld.

In tables 1 and 2, cycle specific maximum neutron flux levels at 0, 15, 30, and 45 degrees on the reactor vessel inner radius are presented for

cycles 1-13 (taken as a base case for comparison), for cycles 14 and 15 which have been implemented in the Kewaunee reactor, and for the projected design of cycle 16.

In regard to the data presented in tables 1 and 2, it should be noted that the former set was taken directly from the cycle specific adjoint calculations; while the latter set was multiplied by a factor of 1.167 to adjust for biases observed between cycle specific calculations and the results of neutron dosimetry for the first three surveillance capsules removed from the Kewaunee reactor. The factor of 1.167 was derived by taking the average of the measurement to calculation ratios (M/C) as follows:

	CALCULATED FLUX (n/cm ² -sec)	MEASURED FLUX (n/cm ² -sec)	<u>M/C</u>
CAPSULE V	1.31E+11	1.61E+11	1.229
CAPSULE R	1.19E+11	1.42E+11	1.193
CAPSULE P	7.66E+10	8.27E+10	1.080
AVERAGE			1.167

In developing this average M/C ratio, dosimetry from capsules V and R was reevaluated using procedures consistent with those used in the recently completed capsule P analysis [4].

An examination of the data provided in tables 1 and 2 indicates that relative to the cycle 1-13 base case the following flux reduction factors were achieved for each of the subsequent fuel cycle designs:

	<u>FLUX REDUCTION FACTOR</u>
CYCLE 14	1.18
CYCLE 15	1.17
CYCLE 16	1.59

These listed flux reduction factors apply to the maximum flux location at the 0 degree azimuth. Flux reduction factors at other azimuthal locations would be somewhat different and may be computed from the data provided. It should also be noted that the calculated flux reduction factors are independent of the measurement to calculation bias derived from the surveillance dosimetry comparisons.

In table 3 the fast neutron exposure history for the Kewaunee pressure vessel is given. This exposure projection was based on the assumption that the irradiation times for cycles 1-13, cycle 14, and cycle 15 were 11.08, 0.86, and 0.92 EFPY, respectively; and that the cycle 16 design was implemented for all subsequent fuel cycles. Data is provided in table 3 both with and without the measurement to calculation bias.

In table 4, the relative radial distribution of fast neutron flux and fluence within the reactor vessel wall is listed for the four azimuthal locations for which cycle specific data was computed. A two-dimensional description of the maximum exposure of the reactor vessel wall can be constructed using the data given in tables 1 through 4 along with the relation

$$\phi(R,\theta) = \phi(\theta) F(R)$$

where: $\phi(R,\theta)$ = Fast neutron fluence at location R, θ within the reactor vessel wall

$\phi(\theta)$ = Fast neutron fluence at azimuthal location θ on the reactor vessel inner radius from tables 1 through 3

$F(R)$ = Relative fast neutron fluence at radius R into the vessel wall from table 4

Analysis has shown that the radial variations within the vessel wall are relatively insensitive to the implementation of low leakage fuel management schemes. Thus, the above relationship provides a vehicle for a reasonable evaluation of fluence gradients within the vessel wall.

All of the best estimate fast neutron ($E > 1.0$ MeV) data can be converted to iron atom displacements (dpa) by making use of the following set of dpa/fluence ratios applicable to the vessel inner radius:

	<u>dpa/fluence($E > 1.0$ MeV)</u>
0 deg Vessel IR	1.655E-21
15 deg Vessel IR	1.657E-21
30 deg Vessel IR	1.652E-21
45 deg Vessel IR	1.656E-21

Distributional information within the pressure vessel wall may then be calculated by normalizing the inner radius values to the relative dpa gradient data provided in table 5.

TABLE 1

CALCULATED FAST NEUTRON FLUX ($E > 1.0$ MeV)
 AT THE KEWAUNEE REACTOR VESSEL INNER RADIUS
 (NO M/C BIAS)

	NEUTRON FLUX (n/cm^2 -sec)			
	<u>0 DEGREE</u>	<u>15 DEGREE</u>	<u>30 DEGREE</u>	<u>45 DEGREE</u>
CYCLES 1-13	3.76E+10	2.36E+10	1.75E+10	1.58E+10
CYCLE 14	3.19E+10	2.08E+10	1.63E+10	1.41E+10
CYCLE 15	3.22E+10	2.09E+10	1.62E+10	1.37E+10
CYCLE 16	2.37E+10	1.78E+10	1.52E+10	1.32E+10

TABLE 2
 CALCULATED FAST NEUTRON FLUX (E > 1.0 MeV)
 AT THE KEWAUNEE REACTOR VESSEL INNER RADIUS
 (WITH M/C BIAS)

	NEUTRON FLUX (n/cm ² -sec)			
	<u>0 DEGREE</u>	<u>15 DEGREE</u>	<u>30 DEGREE</u>	<u>45 DEGREE</u>
CYCLES 1-13	4.39E+10	2.75E+10	2.04E+10	1.84E+10
CYCLE 14	3.72E+10	2.43E+10	1.90E+10	1.65E+10
CYCLE 15	3.76E+10	2.44E+10	1.89E+10	1.60E+10
CYCLE 16	2.77E+10	2.08E+10	1.77E+10	1.54E+10

TABLE 3
 CALCULATED FAST NEUTRON FLUENCE (E > 1.0 MeV)
 AT THE KEWAUNEE REACTOR VESSEL INNER RADIUS

<u>EFPY</u>	<u>FLUENCE (n/cm²)</u>			
	<u>0 DEGREE</u>	<u>15 DEGREE</u>	<u>30 DEGREE</u>	<u>45 DEGREE</u>
<u>WITHOUT M/C BIAS</u>				
12.86 (EOC 15)	1.50E+19	9.43E+18	7.04E+18	6.31E+18
15	1.66E+19	1.06E+19	8.07E+18	7.20E+18
20	2.03E+19	1.34E+19	1.05E+19	9.28E+18
25	2.40E+19	1.62E+19	1.29E+19	1.14E+19
30	2.77E+19	1.90E+19	1.53E+19	1.35E+19
32	2.92E+19	2.01E+19	1.63E+19	1.43E+19
34	3.07E+19	2.12E+19	1.73E+19	1.51E+19
48	4.12E+19	2.91E+19	2.40E+19	2.09E+19
<u>WITH M/C BIAS</u>				
12.86 (EOC 15)	1.75E+19	1.10E+19	8.22E+18	7.36E+18
15	1.94E+19	1.24E+19	9.42E+18	8.40E+18
20	2.37E+19	1.56E+19	1.23E+19	1.08E+19
25	2.80E+19	1.89E+19	1.51E+19	1.33E+19
30	3.23E+19	2.22E+19	1.79E+19	1.58E+19
32	3.41E+19	2.35E+19	1.90E+19	1.67E+19
34	3.58E+19	2.47E+19	2.02E+19	1.76E+19
48	4.81E+19	3.40E+19	2.80E+19	2.44E+19

TABLE 4

RELATIVE RADIAL DISTRIBUTIONS OF NEUTRON FLUX ($E > 1.0$ MeV)
 WITHIN THE PRESSURE VESSEL WALL

<u>Radius</u> <u>(cm)</u>	<u>0°</u>	<u>15°</u>	<u>30°</u>	<u>45°</u>
168.04 (1)	1.00	1.00	1.00	1.00
168.71	0.935	0.938	0.936	0.937
170.12	0.816	0.817	0.814	0.818
171.53	0.680	0.689	0.683	0.691
172.94	0.563	0.573	0.566	0.574
174.35	0.462	0.473	0.465	0.473
175.75	0.376	0.388	0.380	0.388
177.16	0.305	0.316	0.309	0.316
178.57	0.246	0.256	0.250	0.256
179.98	0.196	0.206	0.201	0.206
181.39	0.155	0.164	0.160	0.164
182.80	0.118	0.128	0.125	0.129
183.83	0.0946	0.104	0.103	0.105
184.80 (2)	0.0857	0.0967	0.0956	0.0982

NOTES: (1) Base Metal Inner Radius
 (2) Base Metal Outer Radius

TABLE 5

RELATIVE RADIAL DISTRIBUTIONS OF IRON DISPLACEMENT RATE (dpa)
WITHIN THE PRESSURE VESSEL WALL

Radius (cm)	0°	15°	30°	45°
168.04 (1)	1.00	1.00	1.00	1.00
168.71	0.944	0.947	0.945	0.946
170.12	0.832	0.833	0.830	0.834
171.53	0.714	0.723	0.717	0.726
172.94	0.625	0.636	0.628	0.637
174.35	0.545	0.558	0.549	0.558
175.75	0.466	0.481	0.471	0.481
177.16	0.400	0.414	0.405	0.414
178.57	0.344	0.358	0.350	0.358
179.98	0.290	0.305	0.297	0.305
181.39	0.243	0.257	0.251	0.257
182.80	0.196	0.212	0.208	0.214
183.83	0.163	0.179	0.177	0.181
184.80 (2)	0.154	0.174	0.172	0.177

NOTES: (1) Base Metal Inner Radius
(2) Base Metal Outer Radius

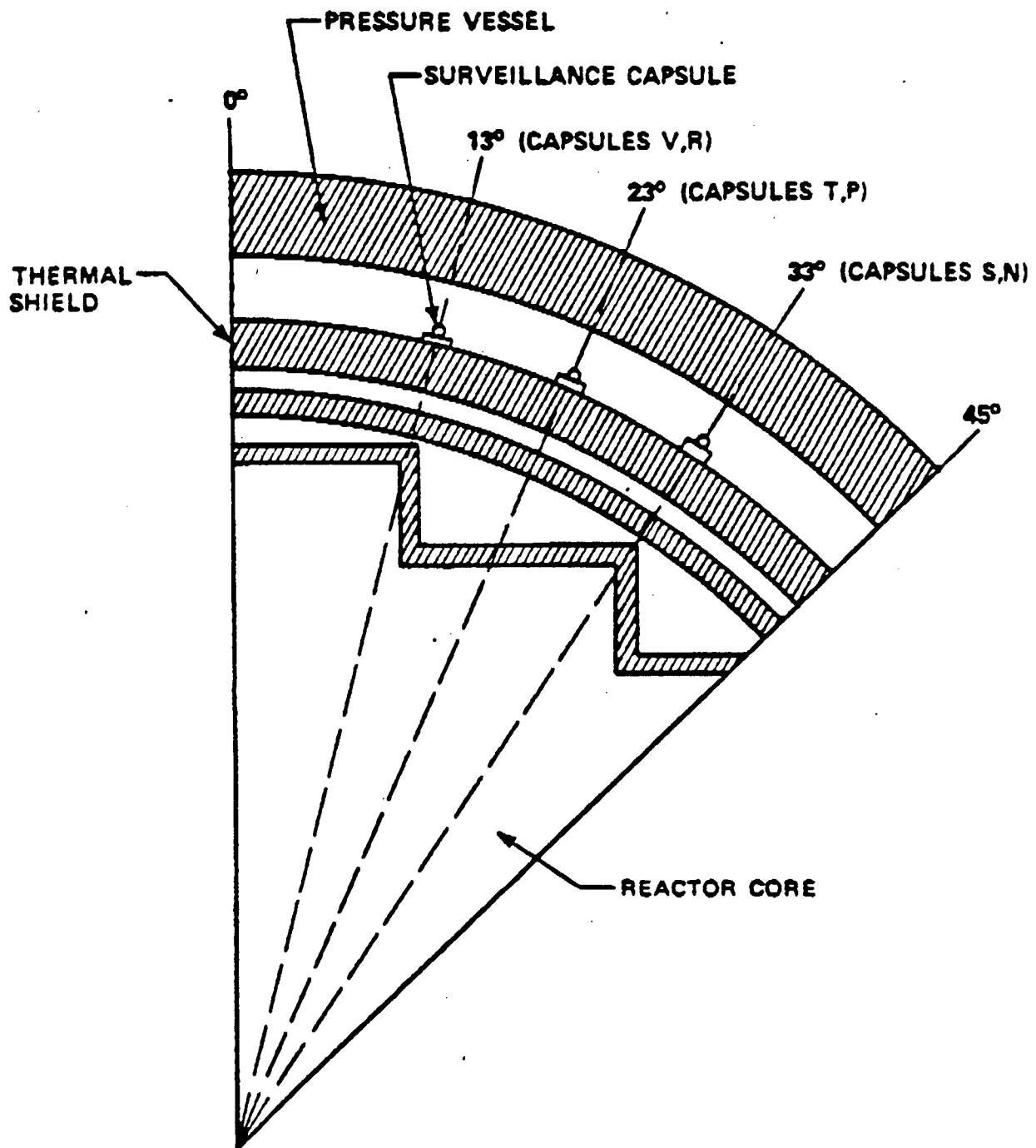


Figure 1. Kewaunee Reactor Geometry

REFERENCES

1. R. G. Soltesz, et al., "Nuclear Rocket Shielding Methods Modification, Updating, and Input Data Preparation. Vol. 5 - Two-Dimensional Discrete Ordinates Transport Technique", WANL-PR(LL)-034, Vol. 5, August 1970.
2. "ORNL RSIC Data Library Collection DLC-76, SAILOR Coupled Self-Shielded, 47 Neutron, 20 Gamma Ray, P3, Cross Section Library for Light Water Reactors".
3. W. N. McElroy, et al., "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments And Blind Test", NUREG/CR-1861, July 1981.
4. S. E. Yanichko, et al., "Analysis of Capsule P From The Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program", WCAP-12020, November 1988.

Attachment 2

To

Letter from K. H. Evers (WPSC) to Document Control Desk (NRC)

Dated

September 19, 1989

PTS Evaluation

The currently accepted methodology used to calculate RT_{PTS} is provided in 10 CFR 50.61. The governing equation is:

$$RT_{PTS} = I + M + [-10 + 470 \text{ Cu} + 350 \text{ Cu Ni}]f^{0.27}$$

where:

$$I = \text{Initial } RT_{NDT} = -56^{\circ}\text{F}$$

$$M = \text{uncertainty margin} = 59^{\circ}\text{F}$$

$$\text{Cu} = \text{wt.\% copper in circumferential weld} = 0.28$$

$$\text{Ni} = \text{wt.\% nickel in circumferential weld} = 0.74$$

$$f = \text{Peak fluence on circumferential weld for 34 EFPY in units of } 10^{19} \text{ n/cm}^2 = 3.07$$

therefore:

$$RT_{PTS} = 265.8^{\circ}\text{F}$$

which is lower than the acceptable screening criterion of 300°F .

It is anticipated that the methodology described in Regulatory Guide 1.99 Revision 2 will replace that in 10 CFR 50.61. Therefore, the following calculation of the adjusted reference temperature is included. The applicable equation is:

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

where:

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

$$\text{CF} = 208.7$$

$$f = 3.07$$

therefore:

$$\Delta RT_{NDT} = 270.5$$

This corresponds to an $\text{ART} = 280^{\circ}\text{F}$ which again is lower than the screening criterion.