PNPF RETIREMENT SAFETY ANALYSIS
REEVALUATION OF RESIDUAL NUCLIDES

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I. SUMMARY

Subsequent to the completion of the Piqua Nuclear Power Facility Retirement Safety Analysis Report, analyses of samples of concrete from the reactor biological shield were completed. These analyses indicated that the neutron flux levels were significantly lower than those calculated during the design of the reactor and used for the determination of radioactivity content of the shield concrete. Also, the chemical and radiochemical analyses showed different, and in some cases higher, concentrations of some trace elements which are precursors of radioisotopes of potential concern.

Accordingly the neutron flux levels were recalculated using present day shielding analyses techniques and cross section data. The ANISN code was used to perform 20 group transport calculations with S8 quadrature coefficients and anisotropic scattering in the P-3 approximation. The neutron flux levels from these calculations were in good agreement with the results from the shield sample analyses. These results, adjusted to a slightly higher than calculated flux level to assure a conservative result, were used to re-analyze the radioisotopic sources to be left at the PNPf site.

From the re-analysis it was concluded that the source levels were either the same or lower than those predicted in the previous report, AI-AEC-Memo-12708.

II. INTRODUCTION

A 2-1/2-in. core, 94-1/2-in. long, was drilled from the PNPf concrete biological shield as a part of the PNPf retirement plan in order to assess the quantities of residual radioactivity. The concrete sample nearest the reactor core was analyzed for several elements and isotopes by the Battelle Memorial Institute (BMI). The published results⁽¹⁾ indicate that the thermal flux, element concentrations, and potential leaching rate of the concrete are less than what was conservatively assumed in the retirement safety analysis, AI-AEC-MEMO-12708.⁽²⁾

From the chemical/radiochemical analysis of the iron and calcium element/isotope concentrations, BMI calculated the average thermal neutron flux at two points in the sample using referenced cross sections, isotopic abundance, and half life data. The magnitude of this flux was found to be approximately 1/10 of the value used in the retirement safety analysis. Thus, radioactivity within the concrete may be 1/10 of the values calculated in the safety analysis. The chemical analysis also showed that the PNPf concrete aggregate is typical limestone and is not igneous rock as assumed in the safety analysis. Igneous rock contains greater concentrations of parent elements of the important radionuclides than limestone does and was chosen for the safety analysis as a degree of conservatism.

Although the reduced activities will not appreciably affect the recommended retirement period (~120 years) it may significantly influence the time for which the control of the concrete is necessary, or operations of any earlier removal of the major components from the site. In this light, additional analytical studies were made of the neutron flux distribution through the major PNPf shield components.

The flux distributions used in the safety analysis are from previous studies⁽³⁾ performed during the design of the PNPf in 1959. These calculations were based on 2-group diffusion theory which, by comparison with today's state of the art, is not very accurate for deep neutron penetrations. However, the calculations were not updated for use in the safety analysis since they were believed to be conservative (as confirmed by BMI) and because an order of magnitude increase in radioactivity would extend the retirement period by only 17 years; this is an increase of about 14 %, since ⁶⁰Co is the long-term dominating isotope.

The flux distributions have been recalculated using techniques which are considered in shielding analysis as the current state of the art. These included the use of the transport theory code ANISN with current cross section libraries having anisotropic scattering transfer matrices up through the P-3 approximation. The resulting fluxes are compared with BMI's measurements. The calculated element concentrations in the concrete have been revised as a result of the chemical analysis, and new radionuclide activities have been calculated.

III. NEUTRON FLUX DISTRIBUTION

From the results of the chemical/radiochemical concrete sample analysis, BMI⁽¹⁾ calculated the thermal flux magnitude at two points in the concrete shield. The locations are at a distance of 3-1/2 and 7-1/2 in. from the outer cavity liner wall and at an elevation of 7-3/8 in. below the reactor core midplane.⁽¹⁾ For each location, two sets of measurements (calcium and iron) were analyzed. The fluxes calculated from these measurements are compared in Figure III-1 with the diffusion theory results used in the retirement safety analysis.⁽²⁾ It can be seen that some discrepancy exists between the iron and calcium data. It was assumed that the experimental errors are equal for each of the data, and the average of the two points was taken as the experimentally determined values with error bands as denoted. These average values are about 1/10 of the magnitude used in the safety analysis.

The thermal flux in the concrete, as determined by BMI, was near the point of expected maximum, and is a valuable contribution for calculating maximum specific nuclear activities. However, the data provides limited information as to the slope of the flux distribution; although Figure III-1 does indicate that the flux slope between the two experimental points falls more rapidly with increased radii than that predicted by diffusion theory. In addition, no information has been gained as to the slope of the flux distribution between the core and the concrete shield.

In order to improve the understanding of the distribution, additional analytical studies were performed. The ANISN code⁽⁴⁾ was employed to perform a multigroup, anisotropic transport calculation. The mathematical models, procedures, and techniques of these calculations are described in the Appendix of this report.

The results of the revised flux calculations are presented in Tables III-1, -2, and -3, and are compared with the AI-AEC-MEMO-12708⁽²⁾ values. It can be seen that the revised fluxes are consistently lower than those previously used with the reduction in magnitude being more pronounced as distance is increased from the reactor core.

Figure III-1. Radial Neutron Flux Distribution – AI-AEC-MEMO-12708
With BMI Measurements

TABLE III-1
AVERAGE AND VOLUME INTEGRATED THERMAL FLUX IN PNPf
MAIN SOURCE COMPONENTS

Component	AI-AEC-MEMO-12708 (Calculated)		AI Revised (Calculated)			
	$\bar{\phi}_{TH}$ (n/cm ² -sec)	$\int \bar{\phi}_{TH} dv$ (n-cm/sec)	$\bar{\phi}_{TH}$ (n/cm ² -sec)	Relative Fraction	$\int \bar{\phi}_{TH} dv$ (n-cm/sec)	Relative Fraction
Upper Core Grid	5.8×10^{11}	9.9×10^{16}	No change			
Lower Core Grid	2.9×10^{12}	5.5×10^{17}	No change			
Sampling Tube Support Plate	9.8×10^{10}	1.3×10^{16}	No change			
Lower Support Barrel	1.0×10^{12}	5.4×10^{17}	9.5×10^{11}	0.95	5.2×10^{17}	0.96
Thermal Shield	5.0×10^{10}	7.0×10^{16}	3.1×10^{10}	0.62	4.4×10^{16}	0.63
Reactor Vessel	1.0×10^{10}	6.3×10^{15}	4.8×10^9	0.48	3.0×10^{15}	0.48
Cavity Liner	4.0×10^9	1.7×10^{15}	3.5×10^8	0.09	1.5×10^{14}	0.09

$$\text{Relative Fraction} = \frac{\text{Current Estimate}}{\text{AI-AEC-MEMO-12708 Estimate}}$$

TABLE III-2
MAXIMUM THERMAL AND FAST FLUX IN PNPf SECONDARY
SOURCE COMPONENTS (n/cm²-sec)

Component	AI-AEC-MEMO-12708 (Calculated)		AI Revised (Calculated)			
	$\max \phi_{TH}$ (n/cm ² -sec)	$\max \phi_F$ (n/cm ² -sec)	$\max \phi_{TH}$ (n/cm ² -sec)	Relative Fraction	$\max \phi_F$ (n/cm ² -sec)	Relative Fraction
Pressure Vessel Insulation	4.5×10^9	1.7×10^9	1.0×10^9	0.22	1.2×10^9	0.70
Pressure Vessel Insulation Banding	4.5×10^9	1.7×10^9	1.0×10^9	0.22	1.2×10^9	0.70
Fuel Channel TC Assembly	6.6×10^{11}	2.7×10^{10}	No change			
Fuel Channel TC Assembly Brazing Material	6.6×10^{11}	2.7×10^{10}	No change			
Embedded TC's	1.1×10^{10}	1.0×10^9	1.4×10^9	0.13	8.5×10^8	0.85
Mixed Mean TC's	7.0×10^{10}	8.2×10^9	2.8×10^{10}	0.40	5.7×10^9	0.70
Mixed Mean TC End Spacer Springs	7.0×10^{10}	8.2×10^9	2.8×10^{10}	0.40	5.7×10^9	0.70
Thermal Shield and Lower Support Barrel TC's	1.8×10^{12}	4.8×10^{11}	2.4×10^{12}	1.33	2.3×10^{11}	0.50
Moderator Flow TC's	6.6×10^{11}	2.7×10^{10}	No change			
All TC Conductors	6.6×10^{11}	2.7×10^{10}	No change			
Serpentine Shield	$<5 \times 10^8$	$<5 \times 10^8$	No change			

$$\text{Relative Fraction} = \frac{\text{Current Estimate}}{\text{AI-AEC-MEMO-12708 Estimate}}$$

TABLE III-3
MAXIMUM AND INTEGRATED FLUX
IN PNPF CONCRETE SHIELD

	AI-AEC-MEMO-12708 (Calculated)	AI Revised (Calculated)	Relative* Fraction
Maximum Thermal Flux (n/cm ² -sec)	1.0 x 10 ¹⁰	1.4 x 10 ⁹	0.14
Maximum Fast Flux (n/cm ² -sec)	1.0 x 10 ¹⁰	1.0 x 10 ⁹	0.10
Volume Integrated Thermal Flux (n-cm/sec)	3.5 x 10 ¹⁶	6.8 x 10 ¹⁵	0.19
Volume Integrated Fast Flux (n-cm/sec)	3.5 x 10 ¹⁶	1.5 x 10 ¹⁵	0.04

*Relative Fraction = $\frac{\text{Current Estimate}}{\text{AI-AEC-MEMO-12708 Estimate}}$

No change was made to the axial distribution (upper and lower grid plates, etc.). The estimated uncertainty in these calculations is approximately 20%.

The revised fluxes are also displayed in Figures III-2 through -4. Figure III-4 compares the calculated results with BMI's results. It can be seen that averaged BMI results are higher by a factor of 1.9 to 3.6. Although the difference between the lower error bands and the calculated values is small, the curves were normalized to the experimental values, as a degree of conservatism, by averaging the ratio of experimental to calculated for all points. This normalization is included in the Revised Data of Table III-3.

Again, the slope of the flux between the two experimentally determined points in the concrete is steeper than the normalized calculated values (designated AI Revised in the tables). As a further check on this slope, thermal flux values were calculated from the BMI measurements of ⁶⁰Co and ¹⁵²Eu activities at four different points in the concrete. The results are presented in Table III-4. These fluxes were plotted in Figure III-4 and they agree well with the slope of the normalized calculated distribution and also substantiate the magnitude of normalization.

Figure III-2. Calculated Radial Neutron Flux Distributions
(Core Region)

Figure III-3. Calculated Radial Neutron Flux Distribution
(Main Source Region)

Figure III-4. Radial Neutron Flux Distribution
(Concrete Shield Region)

TABLE III-4
COMPARISON OF THERMAL NEUTRON FLUX DISTRIBUTION IN PNPFC CONCRETE SHIELD

Concrete Core Sample Section	Distance From Core G _L (cm)	BMI Measured ⁶⁰ Co Activity* (μ Ci/cm ³)	⁶⁰ Co Inferred Thermal Flux (n/cm ² -sec)	BMI Measured ¹⁵² Eu Activity* (μ Ci/cm ³)	¹⁵² Eu Inferred Thermal Flux (n/cm ² -sec)	Average Inferred Thermal Flux (n/cm ² -sec)	AI Revised Thermal Flux (n/cm ² -sec)	R ₂ [†]
94.5	147.3	2.1×10^{-2}	1.5×10^9	2.1×10^{-2}	1.1×10^9	1.3×10^9	1.1×10^9	1.18
92.5	152.4	1.5×10^{-2}	1.0×10^9	1.4×10^{-2}	8.0×10^8	9.0×10^8	8.6×10^8	1.04
90.5	157.5	1.2×10^{-2}	8.4×10^8	9.7×10^{-3}	5.6×10^8	7.0×10^8	6.2×10^8	1.13
88.5	162.6	7.9×10^{-3}	5.5×10^8	7.1×10^{-3}	4.1×10^8	4.8×10^8	4.1×10^8	1.17

*Concentration after 2-1/2 years decay time

$$^{\dagger}R_2 = \frac{\text{BMI Measured}}{\text{AI-AEC-MEMO-12708, Sup A Estimate}}$$

IV. RADIOACTIVITIES

A. MAIN SOURCE

The ratio of revised thermal flux to the value used in the safety analysis are presented in Table III-1 for the major components within the PNPf cavity liner. It can be seen that the revised integrated fluxes, with the exception of one, are at least 50% of the original predictions. Since this factor is within the uncertainty range of the concrete normalization magnitude (2.82), no change will be made to the estimate of activity levels for these components. The one exception is the cavity liner. The revised flux level is only 9% of the original value. Since the cavity liner is in contact with the concrete shield, it is reasonable to expect that the flux to which it was exposed would be increased by the same normalizing factor as for the concrete. With this assumption, the revised activity in the cavity liner would be 35% of the original amount which is within the range of calculational uncertainty. Therefore, no changes will be made in the predicted activity level for any of the main source components.

B. SECONDARY SOURCES

Table III-2 presents comparisons of the revised and safety analysis fluxes. All fast flux values in the secondary source components are at least 50% of those previously used. The components having significant thermal flux changes are only small contributors to the secondary sources (see Table VI-4 of Reference 2). These include: Pressure Vessel Insulation, Pressure Vessel Insulation Banding, Embedded Thermocouples, Mixed Mean Thermocouples, and Mixed Mean Thermocouple End Spacers, and Springs. The major secondary source contributors depend upon the axial source distribution which remains unchanged from the safety analysis. Thus, radioactivities of the secondary sources remain effectively unchanged from those reported in Reference 2.

C. CONCRETE SHIELD

1. Chemical Composition

The BMI chemical and spectrographic analyses indicate that some of the parent elements of important radionuclides exist in the concrete shield at lower concentrations than the levels assumed for the safety analysis. BMI found that

the concrete aggregate origin was typical limestone and not igneous rock as was conservatively assumed in the safety analysis. Actually, for additional conservatism, the content of each major element was maximized in the safety analysis across the granite-rhyolite to gabbro-basalt series which classified all igneous rocks.

The element concentrations determined by the BMI chemical and spectrographic measurements are presented in Tables IV-1 and IV-2 respectively. Both sets are compared with the concentrations assumed for the safety analysis. It is interesting to note that considerably more boron exists in the PNPf shield (300 ppm) than what is normally expected in typical concrete (8 to 10 ppm). Thus, the PNPf shield contains a ^{10}B concentration of about 60 ppm. This probably accounts for the absence of peaking of the revised thermal flux in the concrete. Significant peaking was predicted in the diffusion theory results (See Figures III-1 and III-4) which assumed no boron content in the concrete.

2. Activities

Maximum specific activities were calculated for each of the significant radionuclides in the concrete shield. This was accomplished by combining the maximum increase in parent element concentrations, from either Table IV-1 or

TABLE IV-1
ELEMENT CONCENTRATIONS IN PNPf CONCRETE SHIELD
(Chemical Analysis)
(wt %)

Element	BMI (Measured)	AI-AEC-MEMO-12708 (assumed)			R_1^*
		Cement	Gravel	Concrete	
Ca	18.2-19.9	45.6	7.9	12.0	1.58
Fe	0.8-1.0	2.1	8.7	7.4	0.12
Co	$<1.0 \times 10^{-3}$	-	5.0×10^{-3}	4.1×10^{-3}	0.24
Li	5.0×10^{-3}	-	5.0×10^{-3}	4.1×10^{-3}	1.21
U	$<2.0 \times 10^{-4}$	-	-	1.0×10^{-4}	2.0

$$*R_1 = \frac{\text{BMI Measured}}{\text{AI-AEC-MEMO-12708 Estimate}}$$

TABLE IV-2
ELEMENT CONCENTRATIONS IN PNPf CONCRETE SHIELD
(Spectrographic Analysis)
(wt %)

Element	BMI (Measured)	AI-AEC-MEMO-12708 (Assumed)			R ₁
		Cement	Gravel	Concrete	
Al	1.0	2.4	9.1	7.8 wt %	0.13
Ba	0.01	-	-	1.0×10^{-4}	100.
B	0.03	-	1.0×10^{-3}	8.2×10^{-4}	36.6
Ca	20-30	45.6	7.9	12.0	2.1
Cu	<0.001	-	-	1.0×10^{-4}	10.
Fe	0.7	2.1	8.7	7.4	0.09
Mg	5-10	0.5	4.9	4.0	1.87
Mn	0.03	-	0.2	0.2	0.15
K	0.3	0.5	4.9	4.1	0.07
Si	3-7	10.7	34.0	29.2	0.17
Na	0.2	0.4	4.0	3.3	0.06
Sr	0.005	-	-	-	-
Ti	0.03	0.1	0.9	0.8	0.04
Be	$<1.0 \times 10^{-3}$	-	-	1.0×10^{-4}	10.
Bi	$<1.0 \times 10^{-3}$	-	-	-	-
Cr	$<1.0 \times 10^{-3}$	-	7.0×10^{-3}	5.7×10^{-3}	0.18
Co	$<1.0 \times 10^{-3}$	-	5.0×10^{-3}	4.1×10^{-3}	0.24
Pb	$<1.0 \times 10^{-3}$	-	-	-	-
Mo	$<1.0 \times 10^{-3}$	-	-	1.0×10^{-4}	10.
Ni	$<1.0 \times 10^{-3}$	-	-	1.0×10^{-4}	10.
Sm	$<1.0 \times 10^{-3}$	-	-	1.0×10^{-4}	10.
V	$<1.0 \times 10^{-3}$	-	2.0×10^{-2}	1.6×10^{-2}	0.06
Zr	$<1.0 \times 10^{-3}$	-	-	-	-
S	-	-	1.0×10^{-1}	8.2×10^{-2}	1.0
Cl	-	-	4.0×10^{-2}	3.3×10^{-2}	1.0
P	-	-	0.2	0.16	1.0
O	-	35.5	26.0	31.2	1.0
Sm	-	-	1.0×10^{-3}	8.2×10^{-4}	1.0
Eu	$3.0 \times 10^{-5}\dagger$	-	1.0×10^{-4}	8.2×10^{-5}	0.36
Dy	-	-	4.0×10^{-4}	3.3×10^{-4}	1.0
Gd	-	-	7.0×10^{-4}	5.7×10^{-4}	1.0

$$R_1 = \frac{\text{BMI Measured}}{\text{AI-AEC-MEMO-12708 Estimate}}$$

†Inferred from ^{152}Eu measurement and ^{151}Eu (n, γ) ^{152}Eu reaction.

IV-2, with the appropriate change in maximum flux (fast or thermal according to the type of reaction) from Table III-3. This results in a total change fraction for each of the reactions and is used to modify the safety analysis activities presented in Tables VII-5 and VII-6 of Reference 2. The revised activities and the intermediate calculational steps are presented in Table IV-3.

Specific and total activities of the significant radionuclides in the concrete have been plotted in Figure IV-1 and IV-2. Total activity equals the ratio of (integrated flux)/(maximum flux) times the specific activity. The significant radionuclides which have the largest reductions in activity from those of the safety analysis are ^{152}Eu , ^{154}Eu , ^{60}Co , and ^{55}Fe .

Specific activities reported from the BMI radiochemical analysis⁽¹⁾ are compared in Table IV-4 with the calculated safety analysis activities and also the revised activities. It can be seen that the revised activities are in good agreement with those of the measurements.

Element concentrations which were shown by the BMI measurements to be greater than assumed in the safety analysis were also investigated (see Table IV-5). All the resulting specific activities at decay times of 25 years were found to be no greater than $\sim 10^{-5} \mu\text{Ci}/\text{cm}^3$. Thus, no new troublesome radionuclide was identified as a result of the element concentration measurements.

3. External and Internal Radiation

The external radiation hazard constituted by the activated concrete is presented in Figure IV-3. The required retirement period is seen to be about 80 years for the concrete inside surface dose rate to decay to the safe radiation level, 0.2 mrem/hr.* This compares with a 130-year retirement period reported in AI-AEC-MEMO-12708.

The internal radiation hazard for the important radionuclides in the concrete is presented in Figure IV-4. The only four isotopes that are not shown to be harmless by the ratio of (specific activity)/(safe specific activity) are ^3H , ^{14}C , ^{152}Eu , and ^{154}Eu . These concentrations reported in the safety analysis were 8 and 10 times larger for ^3H and ^{14}C and 20 times larger for the europium

*See Chapter V Reference 2 for a discussion of the significance of this level.

TABLE IV-3

SPECIFIC ACTIVITY OF SIGNIFICANT ISOTOPES IN PNPf CONCRETE SHIELD

Product Isotope	Parent	Change in Parent Composition (fraction)	Reaction	Change in Flux Magnitude (fraction)	Total Change (fraction)	Activity at 25 yr ($\mu\text{Ci}/\text{cm}^3$)	Activity at 50 yr ($\mu\text{Ci}/\text{cm}^3$)	Activity at 100 yr ($\mu\text{Ci}/\text{cm}^3$)
^3H	^6Li	1.21	n, α	0.10	0.12	3.7×10^{-1}	9.0×10^{-2}	5.4×10^{-3}
^{152}Eu	^{151}Eu	0.36	n, γ	0.14	0.050	7.0×10^{-3}	1.6×10^{-3}	9.0×10^{-5}
	^{153}Eu	0.36	n,2n	0.10	0.036	2.8×10^{-8}	6.8×10^{-9}	3.6×10^{-10}
	^{152}Gd	1.0	n,p	0.10	0.10	4.5×10^{-13}	1.2×10^{-13}	8.4×10^{-15}
^{39}Ar	^{42}Ca	2.1	n, α	0.10	0.21	1.2×10^{-5}	1.1×10^{-5}	9.7×10^{-6}
	^{39}K	0.07	n,p	0.10	0.007	1.8×10^{-4}	1.7×10^{-4}	1.5×10^{-4}
^{154}Eu	^{153}Eu	0.36	n, γ	0.14	0.050	3.1×10^{-4}	1.1×10^{-4}	1.4×10^{-5}
	^{154}Gd	1.0	n,p	0.10	0.10	8.7×10^{-12}	2.3×10^{-12}	3.7×10^{-13}
^{60}Co	^{59}Co	0.24	n, γ	0.14	0.034	1.9×10^{-3}	6.8×10^{-5}	9.5×10^{-8}
^{41}Ca	^{40}Ca	2.1	n, γ	0.14	0.30	7.2×10^{-4}	7.2×10^{-4}	7.2×10^{-4}
	^{42}Ca	2.1	n,2n	0.10	0.21	6.5×10^{-10}	6.5×10^{-10}	6.5×10^{-10}
^{151}Sm	^{150}Sm	1.0	n, γ	0.14	0.14	1.7×10^{-4}	1.2×10^{-4}	9.5×10^{-5}
	^{152}Sm	1.0	n,2n	0.10	0.10	2.4×10^{-8}	2.0×10^{-8}	1.4×10^{-8}
	^{151}Eu	0.36	n,p	0.10	0.036	2.8×10^{-11}	2.3×10^{-11}	1.6×10^{-10}
	^{154}Gd	1.0	n, α	0.10	0.10	4.5×10^{-13}	3.7×10^{-13}	2.6×10^{-13}
^{36}Cl	^{39}K	0.07	n, α	0.10	0.007	4.7×10^{-8}	4.7×10^{-8}	4.7×10^{-8}
	^{35}Cl	1.0	n, γ	0.14	0.14	4.5×10^{-5}	4.5×10^{-5}	4.5×10^{-5}
	^{37}Cl	1.0	n,2n	0.10	0.10	2.6×10^{-11}	2.6×10^{-11}	2.6×10^{-11}
^{14}C	^{17}O	1.0	n, α	0.10	0.10	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-5}
^{55}Fe	^{54}Fe	0.12	n, γ	0.14	0.017	6.3×10^{-4}	1.1×10^{-6}	2.7×10^{-12}
	^{56}Fe	0.12	n,2n	0.10	0.012	1.7×10^{-7}	2.9×10^{-10}	7.8×10^{-16}

Figure IV-1. Activation Products in PNPf Concrete

Figure IV-2. Activation Products in PNPf Concrete

TABLE IV-4
ISOTOPE CONCENTRATIONS IN PNPFC CONCRETE SHIELD -
CORE SAMPLE SECTION 94.5
(2.5 year decay time)

Isotope	Specific Activity ($\mu\text{Ci}/\text{cm}^3$)				
	BMI (Measured)	AI-AEC-MEMO-12708 (Calculated)	R_1^*	Sup A (Calculated)	R_2^*
^{55}Fe	2.6×10^{-1}	1.3×10^1	0.20	2.1×10^{-1}	1.24
$^{45}\text{Ca}^\dagger$	4.3×10^{-2}	-	-	-	-
^{152}Eu	2.1×10^{-2}	5.3×10^{-1}	0.04	2.1×10^{-2}	1.0
^{60}Co	2.1×10^{-2}	1.1	0.02	3.4×10^{-2}	0.62
^{36}Cl	4.0×10^{-5}	3.3×10^{-4}	0.12	4.5×10^{-5}	0.89

$$*R_1 = \frac{\text{BMI Measured}}{\text{AI-AEC-MEMO-12708 Estimate}}$$

$$R_2 = \frac{\text{BMI Measured}}{\text{Sup A Estimate}}$$

$^\dagger^{45}\text{Ca}$ was not considered in AI's analysis because of the relatively short half-life (163 days).

TABLE IV-5
SPECIFIC ACTIVITY EFFECTS OF GREATER THAN ASSUMED CONCENTRATIONS
OF ELEMENTS IN PNP CONCRETE SHIELD

Product Isotope	Parent	Change in Parent Concentration (fraction)	Reaction	Change in Flux Magnitude (fraction)	Total Change (fraction)	Activity at 25 yr ($\mu\text{Ci/cm}^3$)	Activity at 50 yr ($\mu\text{Ci/cm}^3$)	Activity at 100 yr ($\mu\text{Ci/cm}^3$)
^{133}Ba	^{132}Ba	100.	n, γ	0.14	14.0	2.7×10^{-6}	2.1×10^{-7}	1.6×10^{-9}
^{10}Be	^{10}B	36.6	n,p	0.10	3.66	1.4×10^{-8}	1.4×10^{-8}	1.4×10^{-8}
^{63}Ni	^{63}Cu	10.0	n,p	0.10	1.0	2.8×10^{-8}	2.2×10^{-8}	1.6×10^{-8}
^{60}Co	^{63}Cu	10.0	n, α	0.10	1.0	1.9×10^{-8}	7.4×10^{-10}	9.8×10^{-13}
^{10}Be	^9Be	10.0	n, γ	0.14	1.4	8.0×10^{-6}	1.4×10^{-6}	4.5×10^{-8}
^{93}Mo	^{92}Mo	10.0	n, γ	0.14	1.4	2.4×10^{-8}	2.4×10^{-8}	2.4×10^{-8}
^{59}Ni	^{58}Ni	10.0	n, γ	0.14	1.4	2.9×10^{-7}	2.9×10^{-7}	2.9×10^{-7}
^{63}Ni	^{62}Ni	10.0	n, γ	0.14	1.4	3.6×10^{-5}	2.9×10^{-5}	2.1×10^{-5}
^{121}Sn	^{120}Sn	10.0	n, γ	0.14	1.4	2.5×10^{-8}	1.2×10^{-8}	3.2×10^{-9}
^{121}Sn	^{122}Sn	10.0	n,2n	0.10	1.0	4.8×10^{-9}	2.4×10^{-9}	5.0×10^{-10}

Figure IV-3. Concrete Surface Dose Rate

Figure IV-4. Concrete Biological Shield (Specific Activity/
Safe Specific Activity) vs Time for Non-Occupational
Internal Exposure

isotopes. However, they were shown in Section VII-C⁽²⁾ to be safe from an internal radiation hazard on other bases. Thus, the revised quantities insure a greater degree of conservatism.

The results of BMI's concrete leaching rate measurements showed that, after 21 days, about 3% of the activity can be leached by water from a 30 gm concrete sample. If 3% of all the activity in the concrete shield were leached out during the first 80 years of retirement, and if all the leached activity were deposited on the floor of the reactor building, safe surface radiation levels could possibly be exceeded. Of course, transporting a quantity of this magnitude is highly unlikely since the leaching tests were made with small samples having relatively large surface to volume ratios. The major portion of concrete in the shield would be expected to have much less surface/volume access by water.

V. CONCLUSIONS

Radioactivity predictions for the Main and Secondary Sources remain unchanged from those of the retirement safety analysis. Thus, by today's standards, all restrictions on entering the reactor complex could be removed in 120 years if the small quantities of silver in the thermocouples were removed. Uncertainties in the analysis indicate that reevaluation would be appropriate at that time.

Radioactivity predictions for the concrete have been reduced from those of the safety analysis, decreasing the required control period from 130 to 80 years, for the inside surface dose rate to decay to the safe radiation level of 0.2 mrem/hr. Analysis of the composition measurements of the concrete identified no new limiting radionuclides.

Although water leaching measurements of the concrete were found to be small, water control must still be maintained for 75 years. This is to prevent corrosion and erosion of the cavity liner and reactor vessel materials to preclude the possibility of radiation levels in the occupied portions of the reactor building from exceeding 0.2 mrem/hr.

APPENDIX

MULTIGROUP FLUX CALCULATIONS

A. GENERAL DISCUSSION

The neutron flux estimates presented in Figures III-1, -2, and -3 are the results of a series of multigroup, anisotropic scattering, transport calculations. These were performed using the ANISN Code⁽⁴⁾ with S_8 quadrature coefficients, a cylindrical model of the PNPf geometry, and a current, 20-group cross-section library⁽⁵⁾ expressly generated for the estimated PNPf energy spectrum.

B. MULTIGROUP LIBRARY

The nuclides comprising the library are H, ^{10}B , C, O, Al, Cr, Fe, Ni, Mo, ^{235}U , and ^{238}U . For hydrogen and iron, transfer matrices up through the P-3 approximation are included. All other nuclides are for the diffusion theory approximation, i.e., transport corrected P-0. The microscopic data are from two sources: the cross sections for boron 10 and molybdenum are based on data contained in the July 1968 ENDF/B data files;^(6,7) all other cross-section information comes from the Atomic International Evaluated Nuclear Data Files dated May 1967.

The microscopic data were processed with the GRISM⁽⁸⁾ code to produce the multigroup library. The energy group structure is shown in Table A-1. In addition, for hydrogen and iron, the higher order transfer matrices through P-3 were produced by the computer code PRISM-II.^(9,10) All data were spectrum averaged with the neutron spectrum shown in Figures A-1 and A-2, which is representative of the neutron distribution in the biological shield. This spectrum was arbitrarily used to weight all orders of transfer. Reference 10 tabulates the Legendre coefficients used for the hydrogen calculation. In the iron calculation, inelastic scattering was treated as isotropic in the laboratory frame.

Self-shielded resonance absorption integrals, for molybdenum, ^{235}U and ^{238}U , were computed and are compared with experimentally normalized* values

*Resonance integrals for heavy isotopes in PNPf fuel were calculated in Reference 11 and fitted to match nuclear measurements of the PNPf Critical Assembly.

TABLE A-1
MULTIGROUP ENERGY STRUCTURE

Group Number	E _{upper}	E _{lower}
1	10.0 Mev	4.5 Mev
2	4.5 Mev	2.25 Mev
3	2.25 Mev	1.125 Mev
4	1.125 Mev	563 kev
5	563 kev	282 kev
6	282 kev	141 kev
7	141 kev	70.5 kev
8	70.5 kev	35.2 kev
9	35.2 kev	17.6 kev
10	17.6 kev	8.8 kev
11	8.8 kev	4.4 kev
12	4.4 kev	2.2 kev
13	2.2 kev	1.1 kev
14	1.1 kev	550 ev
15	550 ev	275 ev
16	275 ev	100 ev
17	100 ev	25 ev
18	25 ev	3 ev
19	3 ev	0.414 ev
20	0.414 ev	0.001 ev

TABLE A-2
RESONANCE ABSORPTION INTEGRALS
AT T = 300°K

Nuclide	Energy Cutoff (ev)	Computed (barns)	Experimental* (barns)
Molybdenum	0.625	14.34	13.43
Uranium-235	0.625	232.6	191.0
Uranium-238	0.5	14.58	14.57

*Resonance integrals for heavy isotopes in PNPf fuel were calculated in Reference 11 and fitted to match nuclear measurements of the PNPf Critical Assembly.

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Figure A-1. Spectrum vs Energy for the PNPf Concrete Shield
(10^{-3} to 10^1 ev)

Figure A-2. Spectrum vs Energy for the PNPf Concrete Shield
(10^1 to 10^7 ev)

in Table A-2. The parameters adopted for the self-shielding calculations, per nuclide, are given in Table A-3.

TABLE A-3
SELF-SHIELDING PARAMETERS

Nuclide	Effective Moderator Scattering Cross Section per Atom Resonance Absorber (barns)	Potential Scattering Cross Section per Atom Resonance Absorber (barns)
Molybdenum	50.0	250.0
Uranium-235	40.0	560.0
Uranium-238	16.6	10.8

C. CALCULATIONAL PROCEDURE

A diagram of the calculational procedure is presented in Figure A-3 and consists of four major steps using the ANISN code. The steps are described below.

1. Unit Fuel Cell

Group constants were calculated for a single unit fuel cell with geometry from Reference 11 as shown in Figure A-4 and Table A-4. The atom number densities⁽¹¹⁾ for each of the 13 regions are shown in Table A-5. The calculation resulted in region-averaged angular flux and volume-weighted macroscopic cross sections.

2. Unit Control Cell

Group constants were also calculated for a unit control cell as in (1) above. The control cell geometry and densities are identical to the unit fuel cell with the exception that the organic in the center region is divided into three additional regions to approximate the control rod assembly of Reference 12 (see Figure A-5 and Table A-6). The control cell calculation resulted in region-averaged angular flux and volume-weighted macroscopic cross sections.

3. Reactor Core

Eigen values and flux profiles were calculated for the PNPf core region using the geometry model of Figure A-6.⁽¹³⁾ The outer region extends well

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Figure A-3. Calculation Procedure for PNPf Multigroup Flux

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Figure A-4. PNPF Unit Fuel Cell Cross Section

TABLE A-4
REGION DIMENSIONS FOR THE PNPf UNIT FUEL CELL

Outer Radius	Region Material	Outer Radius (in.)	Outer Radius (cm)	Volume (cm ³ /cm)	Volume Fraction
R ₁	Organic	1.531	3.8887	47.507	0.23622
R ₂	Steel	1.541	3.9141	0.622	0.00309
R ₃	Organic	1.625	4.1275	5.381	0.02675
R ₄	Aluminum	1.697	4.3104	4.861	0.02417
R ₅	Fuel	1.905	4.8387	15.186	0.07551
R ₆	Aluminum	1.977	5.0216	5.661	0.02815
R ₇	Organic	2.145	5.4483	14.037	0.06979
R ₈	Aluminum	2.211	5.6159	5.824	0.02895
R ₉	Fuel	2.419	6.1443	19.522	0.09707
R ₁₀	Aluminum	2.483	6.3068	6.359	0.03162
R ₁₁	Organic	2.595	6.5913	11.526	0.05731
R ₁₂	Steel	2.605	6.6167	1.056	0.00525
R ₁₃	Organic	3.150	8.0010	63.570	0.31609
				<hr/> 201.112	

TABLE A-5
PNPF UNIT FUEL CELL ATOM NUMBER DENSITIES

Region	Element or Isotope	Atom Density (10^{24} atoms/cm ³)
Fuel - 5, 9	²³⁵ U	0.0008592
	²³⁸ U	0.04288
	Mo	0.004327
	Al	0.0006837
Cladding - 4, 6, 8, 10	Fe	0.000132
	Al	0.05989
	Ni (plate)	0.00174
304 SS - 2, 12	Cr	0.01655
	Ni	0.006516
	Fe	0.06711*
Organic - 1, 3, 7, 11, 13	H	0.0325 ($\bar{T}$ = 560°F)
	C	0.0418

*Mn was lumped into Fe on an absorption equivalence ratio.

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Figure A-5. PNPf Unit Control Cell Central Regions

TABLE A-6
PNPF UNIT CONTROL CELL CENTRAL REGION
ATOM NUMBER DENSITIES

Region	Material	Volume Fraction	Element or Isotope	Atom Density (10^{24} atoms/cm ³)
Organic	Organic	1.0	H	0.0325
			C	0.0418
Poison	B ₄ C	0.5564	¹⁰ B	0.00669
			C	0.01026
	304 SS	0.1961	Cr	0.00324
			Ni	0.000128
			Fe	0.00132
	Organic	0.2283	H	0.00742
			C	0.00954
	Spacers	0.0192	Ni	0.000197

REGION RADII (cm)

r_1	8.001
r_2	21.169
r_3	25.301
r_4	62.490
r_5	83.820
r_6	87.630
r_7	97.155
r_8	107.315
r_s	72.185

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Figure A-6. Cylindrical Model of PNPf Core Geometry

beyond the reflector (outer edge of thermal shield) in order to ensure a good approximation of the angular flux within the reflector region. Subsequent calculations require the use of these fluxes as a shell source for the shield calculation. The atom densities of the terphenyl region are identical to the unit cell's organic region. The iron atom densities are 0.0839 and 0.0850 for the support barrel and thermal shield respectively.

Two core cases were investigated. The first represented an unpoisoned core with an organic outlet temperature of 575°F. The resulting effective neutron multiplication constant was 1.092, which agrees well with the measured value of 1.088.⁽¹¹⁾ The second case approximated the operating core with the center control rod fully inserted and three additional rods partially inserted such that $k_{\text{eff}} = 1.0$. This was accomplished by replacing the center fuel cell with the unit control cell constants and searching for the quantity of homogeneous ^{10}B in region III which results in k_{eff} equal to 1.0. This technique flattens the radial flux by depressing the flux near the center and increasing it near the edge. The resulting flux profiles for the core are shown in Figure III-1. Angular fluxes were obtained for each of the 20 groups for the subsequent shield calculation.

4. Shield

The shield geometry from Reference 13 is shown in Figure A-7, and the atom densities are presented in Table A-7. The resulting flux profiles are presented in Figure A-8 for each of the 20 groups. The fast group flux described in Chapter III represents the first three groups and includes neutrons energies between 1.125 and 10 Mev (see Table A-1).

REGION RADII (cm)

r_0	72.185
r_1	83.820
r_2	87.630
r_3	97.155
r_4	107.315
r_5	116.840
r_6	119.698
r_7	137.160
r_8	138.430
r_9	201.930

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Figure A-7. Cylindrical Model of the PNP Shield Geometry

TABLE A-7
PNPF SHIELD GEOMETRY ATOM NUMBER DENSITIES

Region	Element or Isotope	Atom Density (10^{24} atoms/cm ³)
Terphenyl - I, III, V	H	0.0325
	C	0.0418
Barrel and Vessel - II, VI	Fe	0.0839
Shield and Liner - IV, VIII	Fe	0.0850
Insulation and Bonding - VII	O	0.001129
	Al	0.000978
	Fe	2.47×10^{-5}
Concrete - IX	H	0.00934
	O	0.04103
	Al	0.02852
	^{10}B	$8.531 \times 10^{-6*}$
	Fe	0.000181

* ^{10}B was not included in the previous analysis
(AI-AEC-MEMO-12708).

Groups 1 Through 4

Figure A-8. Calculated Radial Neutron Flux Distributions
(Sheet 1 of 5)

Groups 5 Through 8

Figure A-8. Calculated Radial Neutron Flux Distributions
(Sheet 2 of 5)

Groups 9 Through 12

Figure A-8. Calculated Radial Neutron Flux Distributions
(Sheet 3 of 5)

Groups 13 Through 16

Figure A-8. Calculated Radial Neutron Flux Distributions
(Sheet 4 of 5)

Groups 17 Through 20

Figure A-8. Calculated Radial Neutron Flux Distributions
(Sheet 5 of 5)

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