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YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

**BULK BIOLOGICAL SHIELDING
ASPECTS OF THE YANKEE CORE**

R & D SUBCONTRACT NO. 1 UNDER
USAEC - YAEC CONTRACT AT (30-3) - 222

OCTOBER 1957

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DEPARTMENT
PITTSBURGH, 30 P. O. BOX 355 PENNSYLVANIA

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Yankee Atomic Electric Company
Research and Development Program

BULK BIOLOGICAL SHIELDING ASPECTS OF
THE YANKEE CORE

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For The Yankee Atomic Electric Company
Under Research and Development Subcontract
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PREFACE

This report was prepared in the early stages of the Yankee Research and Development Program and was issued at that time as an internal working document. It is being reissued now in its original format as a topical report to summarize the work carried out in this area under Subcontract No. 1 to USAEC-YAEC Contract AT(30-3)-222.

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Bulk Biological Shielding Aspects of the Yankee Core

Abstract

The purpose of this report is to summarize the analysis of the radiation sources within the inner - or primary - concrete shielding of the Yankee Atomic Electric Plant. The dose due to these sources at a point on a transverse centerline and outside the plant container has been calculated. The dose at two points on the axial centerline has also been found as well as the extent to which the water is activated during its passage through the pressure vessel.

I. Introduction

The Yankee Atomic Electric Plant is powered by a reactor which is cooled and moderated by pressurized light water. The reactor core is of the heterogeneous type: UO_2 pellets are confined in stainless steel tubes; these are ordered in a square array having a generally cylindrical outline. This core and its auxiliary structure (most of which is steel), control mechanisms, etc. are contained in a steel pressure vessel. Between this vessel and the core are a steel core baffle and two steel thermal shields; the latter protect the vessel from excessive radiation damage and thermal stress.

Thermal neutrons absorbed in the various steel regions described above are the sources for most of the high energy gamma radiation. The dosages reported herein are due exclusively to rays originating in steel and zirconium since gamma radiations from fission and fission product decay in the core and neutron capture in the coolant water are much less energetic. The attenuation coefficients of concrete for these lower energy photons are, relative to the high energy coefficients, large enough to make negligible their contribution to the external dose.

Initially, this report describes the determination of the radiation dose at a distance of 60 feet from the core center along a transverse centerline as a function of total thickness of concrete shielding. The particular dose calculated is that due to the radiation arising from sources within the inner concrete annulus (see Section II); the minute contributions from sky shine and other large angle scattering and the significant contributions from the piping and other primary loop equipment are not included in this dosage. This first calculation is described in section III.

The section following describes the calculation of the dose both above and below the pressure vessel stemming from the high energy gamma sources located in this same region. Part A of section IV describes the calculation of the dosage below the plant container in the area containing the railroad spur. Part B describes the calculation of the dose at a point 85 feet above the pressure vessel head and just below the plant container. The latter calculation was made to facilitate the estimation of the hazard external to the plant due to scattering of such radiation by the plant container.

The coolant water is activated by the very fast neutrons prevalent in the core and reflector: as a result of this activation, both high energy gammas and fast neutrons are emitted by the coolant. The calculations of the N^{16} and N^{17} number densities in the primary loop are described in section V. Information on the distribution of these products of water activation was transmitted to the personnel of Large Plant Engineering. With these source strengths, they calculated the dosages outside the secondary shielding from the many different volume sources constituted by the primary loop.

FIGURE 1: (a)
Plan View of Primary and Secondary Shielding

FIGURE 1: (b)
Elevation View of Primary and Secondary Shielding

II. Description of the Shielding and the Sources

Figure 1 shows plan and elevation drawings of the Yankee Plant with particular emphasis on the placement of the concrete shielding. The following annular regions are encountered as one proceeds radially outward from the pressure vessel:

- (a) neutron water shield.
- (b) inner or primary concrete shield.
- (c) plant components' area.
- (d) outer or secondary concrete shield.

The spherical plant container holds this entire complex. The building housing the turbine, the generator and the control room is adjacent to this container. The primary loops extend throughout the annular plant components' area. There is virtually no shielding between the top of the pressure vessel and the plant container. There are six feet of concrete between the bottom of the vessel and the container. This was a structural requirement at the time of the calculation.

All calculations assume a thermal power level of 480 MW. The core characteristics have been described in detail elsewhere^{1/}. Two basic changes have been made which required the adjustment of some of the constants: the enrichment was raised to 2.66%^{2/}; the macroscopic thermal absorption cross section was increased to simulate the presence of control rods. This larger cross section was found by solving a modified one-group criticality equation.

Figure 2 depicts the locations and widths of the various regions which constitute sources and/or shields. The width of concrete is

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listed as 10 feet; calculations were also made for widths of 5 and 7.5 feet.

The thermal flux in the source regions was found by means of a digital computer code. Through this code, the distribution of fast and thermal neutrons in the regions from the core through the cold ($\rho \approx 1 \text{ gm/cm}^3$) water of the neutron tank was provided.

Table I provides the fast and thermal constants for the various regions which were required by the code as input data.

TABLE I
Two Group Constants

	Core	$\text{H}_2\text{O} \left(\begin{smallmatrix} \text{hot:} \\ \rho = 0.79 \end{smallmatrix} \right)$	$\text{H}_2\text{O} \left(\begin{smallmatrix} \text{cold:} \\ \rho \approx 1 \end{smallmatrix} \right)$	Fe	Void
fast: Σ_f	0.016867	0.03815	0.0491	0.006133	0
slow:	0	0	0	0	0
fast: Σ_a	0.0070001	0	0	0	0
slow:	0.1457	0.01107	0.01952	0.1521	0
fast: $\nu\Sigma_f$	0.0016284	0	0	0	0
slow:	0.1949	0	0	0	0
fast: D	1.2336	1.98	1.54	0.895	1.0×10^5
slow:	0.360	0.23861	0.14276	0.3987	1.0×10^5

The zirconium and steel in these regions are the sources for all of the capture gammas having significant energy. A mono-energetic spectrum was assumed, i.e., one 7.5 Mev gamma is considered to be emitted per thermal neutron capture. This is conservative, in that if lower energy γ 's are actually emitted, they would be less penetrating.

FIGURE 3: TWO GROUP FLUX DISTRIBUTION

III. Transverse Dosage

A. Specification of the Volume Sources Outside the Core

The two-group fluxes in the regions from the core through the water of the neutron tank were obtained by use of the WANDA code^{3/}. This one-dimensional code handles diffusion (in this instance) for cylindrical regions (infinite in the other direction). Figure 3 is a radial plot of the fast and slow fluxes; the values of the ordinate are normalized to the average power density; i.e., axial variation of the flux is ignored. Another part of the output, when properly normalized, provided the average flux in a given region. The actual total number of thermal neutron captures per region was found by multiplying this flux by the core height and the appropriate macroscopic cross section. This total will actually occur in a non-uniform distribution over a vertical range greater than the core height but the dose is a function of the grand total of captures and not their vertical distribution.

The dosage at a given radial distance will not, of course, stem from every capture over the entire circular arc of a regional annulus. In order to determine the volume that is effective in contributing to the dose 60 feet from the core center, an arbitrary but conservative specification was applied to the situation. Figure 4 depicts to scale a plan view of one-fourth of the reactor from the core center through five feet of concrete primary shielding. The core, baffle and reflector have been idealized to their equivalent cylindrical geometry; the figure of 5 feet is an arbitrary assumption for the inner concrete annulus.

The volume source per annular region which is effective in producing a dose at a larger distance was determined by means of a larger version of figure 4. This effective volume source per region was taken to be a slab having a height equal to the core's and a width determined in the following manner: the half-width is the lateral distance as measured from the figure between the transverse centerline and a line parallel to the latter starting from the curved region source which traverses an additional three attenuation lengths of shielding. For a ten foot thickness of concrete and assuming linear build-up, this method approximates the effective volume source by eliminating from consideration all of the differential volume sources which contribute less than 6% of the maximum differential dose, viz., that dose from elementary volumes located on the transverse centerline. Table II provides these slab widths. They were measured from each annulus outside the core representing a discrete source region.

TABLE II
Slab Widths

Source Region n	W_n (cm)
Baffle	172
Barrel	186
Thermal Shield	202
Pressure Vessel:	
Measured tangent to inside surface	202
" " " outside "	286
Neutron Tank Wall	328

The WANDA code can solve for the diffusion of neutrons through as many as 25 regions. Table III delineates the partition of the steel regions thicker than 0.61 attenuation lengths (for 7.5 Mev gammas) such that this maximum number of regions is employed.

TABLE III

Geometric WANDA Input Data

Discrete Region	Radial Width of Region (cm)	Number of Sub-regions	Radial Width of Sub-region (cm)
Core	94.49	1	94.49
Baffle	1.655	1	1.655
Reflector	21.97	1	21.97
Barrel	2.54	1	2.54
Water	5.08	1	5.08
Thermal Shield	7.62	4	1.905
Water	5.08	1	5.08
Pressure Vessel	20.96	10	2.096
Void	10.16	1	10.16
Tank Wall	5.398	3	1.799
Tank Water	76.2	1	76.2

The subdivision was performed to avoid calculations involving build-up and attenuation within the thicker steel annuli. The actual source distributions within each discrete region could have been represented by a sum of weighted exponential functions of the radius. Analytic methods for the calculation of the dosages from such sources are exceedingly clumsy^{4/}; furthermore they are less accurate than the method employed since small differences of large

quantities are involved.

The steel sources outside the core, at this stage of the presentation, are being approximated by stacked rectangular slabs. Concerning the slab widths listed in table II: for the 10 slabs comprising the effective volume source representing the pressure vessel, the width was assumed to vary linearly between the widths measured for the inside and the outside of the pressure vessel.

B. Determination of the Dose from the Sources Outside the Core

Prior to the actual dosage calculation, one more "operation" was performed on what once had been annular cylindrical sources varying from 1.6 to 21 cm in thickness: each rectangular slab was converted to a rectangular surface source via multiplication of the volumetric source density by the thickness of each slab. From this point, the dosage calculation was straightforward: at a distance of 60 feet from the core center, all these rectangular surface sources are indistinguishable from point sources of the equivalent magnitude. Therefore, the energy flux from the n^{th} surface source is

$$\frac{B_n S_{An} H W_n e^{-b_n}}{4\pi r_n^2}$$

where:-

- b_n = total number of attenuation lengths between n^{th} source and detector.
- W_n = width of n^{th} source (from table II).
- H = core height.
- S_{An} = n^{th} surface source strength (Mev/cm²-sec).
- r_n = distance between n^{th} source and detector ($\leq 60'$).
- B_n = the build-up factor corresponding to b_n .

Since the 2 or 3 attenuation lengths of shielding closest to the detector would be constituted by concrete (the shield wall of the control room; figure 2 does not show the concrete in its precise relationship to the steel wall of the plant container), concrete dose build-up factors^{5/} for a point isotropic source^{6/} were used. The dosages whose determination has been described in this section are summarized in part D.

C. Determination of the Core Dose

The zirconium and stainless steel contributions to the core's total macroscopic thermal neutron absorption cross section (listed in table I) are as follows:-

$$\Sigma^{Zr}: 1.24 \times 10^{-4} \text{ cm}^{-1}$$

$$\Sigma^{SS}: 123 \times 10^{-4} \text{ cm}^{-1}$$

From the WANDA output data, the average thermal flux in the core was calculated to be $2.77 \times 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$.

Adding the two cross sections and multiplying their sum by the thermal flux and 7.5 results in the volumetric source strength - $2.58 \times 10^{12} \text{ Mev/cm}^3\text{-sec}$. The core was treated as a cylindrical source (of finite length)^{7/} by making use of the equivalent radius; again, concrete build-up factors were used.

D. Dosage Summary

Table IV summarizes the energy fluxes for the three different thicknesses of concrete; the total flux is presented as the sum of the fluxes from the 6 distinct regions which comprise the significant sources of radiation. The contribution of 2.5 Mev gammas originating from thermal capture in the water of the neutron tank is also

FIGURE 5: PRIMARY DOSAGE AT TRANSVERSE DISTANCE OF 60 FEET

provided in table IV. This value is a severe overestimate relative to the 7.5 Mev energy fluxes. The reason for this is that a very "safe" but unrealistic model was adopted for the 2.5 Mev source region; and, since the resultant flux did not significantly increase the flux due to 7.5 Mev gammas, the 2.5 Mev calculation was not refined. Accordingly, figure 5, which presents a plot of dosage versus total thickness of concrete, represents only dose due to 7.5 Mev gammas.

TABLE IV
High Energy Flux in Mev/cm²-sec at Detector Due to
Discrete Region Sources

Concrete Thickness:	10'	7.5'	5'
Tank Wall	0.1340	11.2	638
Pressure Vessel	0.0483	2.97	172
Thermal Shield	0.0075	0.785	47.1
Barrel	0.0191	1.10	69.3
Baffle	0.0331	1.99	121
Core	0.0099	0.592	35.5
Totals	0.252	18.7	1,080
<u>2.5 Mev γ Flux in Mev/cm²-sec</u>			
Tank Water			105

IV. Dose Above and Below the Pressure Vessel

A. Dose Below the Pressure Vessel and the Plant Container

The dose at grade level directly beneath the lowest point of the plant container was calculated. As before, this dose is a sum of contributions from the core and capture gamma sources outside the core. But these sources external to the core are now the bottom end plate and other steel hardware between the active part of the core and the bottom of the pressure vessel.

Particular designs for an end plate arrangement and fuel assembly extension were chosen. Since the first requirement of this type of shielding analysis is a flux plot, the parameters of this design had to be adjusted to the proper form for WANDA input data. This involved a somewhat arbitrary division of the volume below the core into 8 regions: steel, water and steel-water regions. The last type of region was treated as a homogeneous mixture for the purpose of the digital computer analysis. Table V presents data on these regions - volume fractions, widths, etc. Variations from one mechanical design to another would probably change the intrinsic source of radiation only slightly; certainly there would be no appreciable difference if the volume fractions and their locations relative to the core remain essentially the same.

TABLE V

Data on Regions Below Active Part of Core

Region	Width (cm)	Number of Sub-regions	Volume Fraction	
			H ₂ O	SS
Active Core to End Plate	12.70	6	0.8775	0.1225
Top Plate of Assembly	2.20	1	0	1
Between Top and Bottom Plates	15.04	7	0.8159	0.1841
Bottom Plate	3.08	2	0.0	1.0
Fuel Assembly Extension and Shroud Region	5.08	2	0.9587	0.0413
Remainder-of-Shroud Region	248.9	4	0.9430	0.0570
H ₂ O ($\rho \approx 0.79$)	29.2	1	1	0
Pressure Vessel	14.6	1	0	1

The WANDA problem was solved using spherical geometry. As previously, the average thermal flux in each sub-region was easily computed from the output. The source regions outside the core are discs or cylinders. The number of sub-regions into which each was divided is listed in table V.

The average thermal flux in all the source regions was increased by a factor of one-third to take into account the concentration of power in the lower part of the core during initial operation due to blanket control rod insertion at the top. The dosage at the point of interest was then determined by treating the sources

outside the core as disc sources^{8/} and the core itself as a (finite) cylinder^{9/}. Despite the fact that some of the sources external to the core are rather "tall" cylinders, the same procedure as that outlined in part III was used to convert these volume sources to surface (disc) sources. A brief examination of the steel volume fractions for the various regions (refer to table V) indicates the method to be a conservative device since there is so little source material in the "taller" regions.

Figure 6 shows the locations and widths of the various source and/or shield regions as well as the voids in the volume of space between the detector and the core. Once again, the diameters of these disc sources relative to the distances from each to the detector are such that the expression for the dose stemming from such sources reduces to the expression for the dosage from a point source. The dose build-up factors used in this part were those for a point isotropic source in an iron medium^{10/}. Table VI lists the total dose calculated as well as a breakdown of the dose according to its origin.

B. Dose Above the Pressure Vessel

Radiation which leaves the top of the pressure vessel and travels vertically upward encounters the vapor container after passing through an 85 foot air gap (discounting the presence of the bridge crane). The extent or the existence of hazard stemming from the scattering of this radiation by the container had to be evaluated. As far as the sources of this radiation are concerned, their geometry and distribution are essentially a "mirror image"

FIGURE 7: REGIONS BETWEEN CORE AND VAPOR CONTAINER

of the situation below the core - refer to the schematic of figure 6. Figure 7, similar in the source regions to figure 6, depicts the differences in the geometry and shielding above, as compared with below, the core.

Aside from not increasing the source density by one-third, the analysis was carried out as described in part A. Table VI provides the results of the analysis. The contribution of high energy gammas from N^{16} in the water of the upper plenum of the pressure vessel is also negligible because of the short half-life of N^{16} and the comparative stagnancy of the water in this region - see section V.

To indicate their unimportance as a source of dose either above or below the pressure vessel, the 2.5 Mev gamma contribution (due to thermal neutron captures in water) was calculated in one instance - see table VI below.

TABLE VI

Dosage Above and Below the Pressure Vessel (mr/hr)

γ Energy (Mev):	7.5	2.5
<u>Below the P.V.:</u>		
Core Contribution:	0.0013	-
Contribution of the Other Sources:	<u>0.0068</u>	-
Total Dose	8.1×10^{-3}	
<u>Above the P.V.:</u>		
Core:	0.17	-
Other Sources:	<u>1.85</u>	0.008
Total Dose	2.0

FIGURE 8: PRIMARY LOOP FLOW SCHEME

V. Activation of O^{16} and O^{17} in the Primary Coolant

This section describes the calculation of the equilibrium number densities of N^{16} and N^{17} in the water of the primary loop.

The equations used to determine the densities of these radioisotopes have been derived previously^{11/12/} for slightly different flow circuits. Figure 8 schematically depicts the primary loop flow scheme; the sub-division of the loop flow volume is determined by which portions of the latter produce the radioisotopes.

Five first order linear differential equations - one for each volume -- were set up to describe the variation with time of this number density. Only in the core and reflector regions is there substantial activation; therefore only two of the five equations contained production terms. Premised on the fact that equilibrium has been attained, the parameter of time was referred to the beginning of each cycle as the origin. The five boundary conditions were based on the stipulation that the densities are equal at the exit of one flow volume and the entrance of the next - see figure 8. Following algebraic manipulation, the equations "boiled down" to one significant equality:

$$n_e = A I_R + B I_C,$$

where n_e equals the number density of N^{16} or N^{17} at the inlet to the external flow volume (i.e., the core outlet); A and B are functions of the decay constant and the residence times in the various parts of the circuit; and I_R and I_C are the volume rates of production of the activated nuclei in the reflector and core, respectively. Table VII provides the "non-stagnant" flow volumes of the loop^{13/} and the flow

areas of the water between core and pressure vessel $\frac{14}{1}$. These data, together with the values for the volumetric flow rate ($204 \text{ ft}^3/\text{sec}$) and the entrance area to the reflector volume (1.5 ft^2) allowed the determination of the various residence times. The times are also listed in table VII.

TABLE VII
Primary Loop Flow Parameters

Loop Element	Volume (ft^3)	Residence Time (sec)
Reflector	120	3.59
Lower Plenum	380	1.86
Core	130	0.637
Top Plenum	80	$t_e \left\{ \begin{array}{l} 7.73 \\ 0.556 \end{array} \right.$
Hot Leg	42.5	
Steam Generator	242	
Cold Leg	68.3	
Vessel Inlet Chamber	85	
Thermal Shield Region	95	0.556
Flow Areas (ft^2):- Reflector - 12		
Thermal Shield Region - 7.67		

Of the various quantities, the production rates in the core and reflector were the most difficult to determine. In the case of both isotopes, this production is due to the high energy neutrons which originate in fission; the production rate, I_C (or R), is equal to

$$C \bar{\sigma} \bar{\phi}_C \text{ (or } R \text{)}$$

where

C = number density in the loop water of the target isotope — O^{16} or O^{17} .

$\bar{\sigma}$ = the cross section for the reaction averaged over the fission spectrum:-

For O^{16} (n, p) N^{16} : $\bar{\sigma} = 2 \times 10^{-5}$ barns

For O^{17} (n, p) N^{17} : $\bar{\sigma} = 7 \times 10^{-6}$ barns

$\bar{\phi}_C$ (or R) = the high energy neutron flux in the core (or reflector) also averaged over the fission spectrum.

In the core, the flux of neutrons having energies greater than that of the threshold for the reaction was obtained by numerical integration of the right-hand side of the following equality:

$$\phi_C = n_b \frac{A}{2} \int_{E_T}^{\infty} \frac{e^{-(E - \sqrt{2E})}}{\Sigma_T E} dE$$

where:

n_b = average fission neutron source density.

$A = \sqrt{2/\pi} e$

E_T = the threshold energy for the reaction: 10 Mev for N^{16}
8.5 Mev for N^{17} .

$\Sigma_T(E)$ = the energy = dependent macroscopic removal cross section in the core

$$= 0.0765 + \left(\frac{0.301}{E + 4.2} \right) + \left(\frac{0.0652}{E + 0.134} \right)$$

After ϕ_C was calculated, $\bar{\phi}_C$ was obtained by multiplying ϕ_C by $S/S(E_T)$; S is equal to one - the area under the (normalized) fission

spectrum - and $S(E_T)$ is the area under the normalized spectrum above E_T ^{15/}.

The quantity ϕ_R was determined by a comparison with results of a previous calculation. I_R , an average value over the reflector, was calculated to be equal to $0.179 I_C$; I_C was $2.7 \times 10^7 \text{ cm}^{-3} \text{ sec}^{-1}$ for the production of N^{16} and $3.7 \times 10^3 \text{ cm}^{-3} \text{ sec}^{-1}$ for the production of N^{17} .

The resultant number densities in the external portion of the primary loop as a function of residence time therein are provided below:

$$\begin{aligned} N^{16} (\text{cm}^{-3}) &= 2.86 \times 10^7 \exp(-0.0936 t_e) \\ N^{17} (\text{cm}^{-3}) &= 2.89 \times 10^3 \exp(-0.1674 t_e). \end{aligned} \quad (t_e \text{ is in seconds})$$

VI. Conclusion

The calculations described in this report have provided a basis for the specification of the Yankee reactor's shielding.

In particular, the "primary" transverse contribution to the total dose has been reported as a function of gross thickness of the concrete shielding. The safety of the area below the vapor container has been determined. It has been established that there is not need of any special shielding above the pressure vessel, at least insofar as protection against "primary" radiation scattered by the vapor container is needed. The source strengths for the "secondary" radiation have been provided by means of the water activation calculation.

VII. / Acknowledgment

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