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5. RADIATION SHIELDING

5.1 Design Basis

The NS Savannah shielding is designed in accordance with the criteria summarized in Table 5-1. These criteria are based on the April 1958 recommendations of the National Committee on Radiation Protection and Measurement and comply with the requirements of "Standards for Protection Against Radiation" (10CFR Part 20)[1].

5.2 Radiation Sources

5.2.1. Neutron Source

The volumetric neutron source strength is given by:

$$S_n = 3.1 \times 10^{10} \ \eta P \qquad (5\text{-}1)$$

where S_n = volumetric neutron source strength, neutron/cc-second

P = power density, watts/cc

η = neutrons/fission

and there are 3.1×10^{10} fissions per watt of power per second. The source strength as a function of position in the core is derived from the power density profile. The average value at a core power of 69 MW-t is 1.63×10^{12} neutron/cc-second and 1.9×10^{12} neutron/cc-second at 80 MW-t. The fission neutron energy spectrum can be represented by the equation:

$$N(E) = \left[\frac{2}{\pi e}\right]^{\frac{1}{2}} \left[\sinh (2\ E)^{\frac{1}{2}}\right] e^{-E} \qquad (5\text{-}2)$$

Where E = energy, mev

$N(E)$ = fraction of neutrons with energy E, normalized to one neutron

Table 5-1. Design Dose Rates

Location	Operating conditions	Type access	Design dose rate
Access spaces outside of secondary shield wall	Full power operation	Unrestricted	0.5 rem/yr
Access spaces outside of secondary shield wall (crew space)	Full power operation	Unrestricted (crew)	5.0 rem/yr
Cargo holds	Operation at 1/5 full power	Unrestricted (stevedores)	5.0 rem/yr
Inside containment	Reactor operating	None permitted	--
Inside containment	Reactor shut down ½ hour. No fission products in primary loop.	Limited	200 mr/hr
Locations occupied during fuel transfer operations	Reactor shut down 3 days. Refueling operations	Limited	200 mr/hr transient conditions. 10 mr/hr continuing conditions.

5.2.2. Primary Gamma Source

The sources of primary gamma rays are prompt fission gammas, fission product decay gammas, neutron capture (n,γ) reactions, and inelastic neutron scattering. The results are summarized in Table 5-2.

The prompt fission and fission product decay gamma source spectra are represented by the equation

$$\overline{N}(E) = 14.0\ e^{-1.10\ E} \tag{5-3}$$

where

$\overline{N}(E)$ = number of photons in the energy group per fission

E = photon group energy, mev

The neutron capture gamma source strength is determined for materials of significant cross sections by the equation

$$S_v(E) = \sum_{i=1}^{n} y_i(E)\ \Sigma_i\ \phi_{th} \tag{5-4}$$

where

$y_i(E)$ = yield, photons per absorption from i^{th} material in the energy interval E

Σ_i = capture cross section, i^{th} material, cm^{-1}

ϕ_{th} = thermal neutron flux, neutron/cm^2-second

The calculation of the gamma source from inelastic neutron scatter is based on the cross section and photon yield data available in the literature. The contribution to the total dose is relatively small, even though the calculation is made conservatively.

5.2.3. Secondary Gamma Source

Secondary gamma radiation results from the absorption of thermal neutrons in the shield outside the core. The intensity and energy distribution of the gamma source at any point is determined by the thermal flux profile through the shield slabs, the cross section for thermal neutron capture of the materials present, the yield and energy spectra of the resulting capture gammas, and the gamma attenuation properties of the materials involved. The source strength of the secondary gamma radiation is given by equation 5-4.

Table 5-2. Core Volumetric Gamma Source Strengths Average Levels - 69 MW-t Operation

Photon energy mev	Prompt and fission product decay, photons/cc-sec	Thermal and resonance capture photons/cc-sec	Inelastic scatter photons/cc-sec	Total photons/cc-sec	Total mev/cc-sec
1	3.15×10^{12}	1.99×10^{11}	4.6×10^{11}	3.81×10^{12}	3.81×10^{12}
2	1.03×10^{12}	2.67×10^{11}	2.5×10^{11}	1.55×10^{12}	3.10×10^{12}
3	3.91×10^{11}	8.49×10^{10}	1.25×10^{11}	6.01×10^{11}	1.80×10^{12}
4	1.18×10^{11}	1.04×10^{11}	5.9×10^{10}	2.81×10^{11}	1.12×10^{12}
5	3.94×10^{10}	2.28×10^{11}	2.8×10^{10}	2.95×10^{11}	1.48×10^{12}
6	1.31×10^{10}	1.23×10^{11}	1.14×10^{10}	1.57×10^{11}	9.42×10^{11}
7	4.28×10^{9}	4.99×10^{10}	3.7×10^{9}	5.79×10^{10}	4.05×10^{11}
8	1.44×10^{9}	4.48×10^{10}	7.1×10^{8}	4.69×10^{10}	3.75×10^{11}
9	4.83×10^{8}	1.87×10^{10}	1.07×10^{8}	1.93×10^{10}	1.74×10^{11}

Total energy: 1.32×10^{13} mev/cc-sec

Secondary gammas constitute a major portion of the gamma ray dose rate at the exterior of the primary shield; consequently, secondary gammas are very important in determining the primary shield thickness.

Steel and water are the source of secondary gammas in the shield materials surrounding the core. Neutron capture by hydrogen yields a single photon of 2.23 mev for each thermal neutron absorbed. Gamma yields for type-304 stainless steel are listed in Table 5-3.

Table 5-3. Gamma Yields for Type-304 Stainless Steel

Energy interval (mev)	N(E) photons produced per thermal neutron absorbed
8.51 - 9.50	0.12
7.51 - 8.50	0.28
6.51 - 7.50	0.21
5.51 - 6.50	0.12
4.51 - 5.50	0.10
3.51 - 4.50	0.13
2.51 - 3.50	0.14
1.51 - 2.50	0.25
0.51 - 1.50	0.37

5.2.4. Fission Product Source

During reactor operation, fission product activity approaches an equilibrium level within a few hours after startup and contributes about 25% of the total gamma ray source strength. Fission products are the principal source of radiation during refueling operations after shutdown.

Before and after shutdown, radiation levels from fission products and neutron-activated materials near the core are functions of time, position in relation to the core, and reactor operating conditions. To simplify the analysis, a burnup history is assumed. The history assumed consists of 18 years of steady state operation at an average power level of 50 MW-t, followed by 600 days operation at 69 MW-t. Assuming no burnup of parent material, this history permits mathematical analysis based on the effects of two steady state activity buildup components:

1. Steady state operation at the average power level for 20 years.
2. Steady state operation at the difference between full power and average power for the final 600 days.

The cumulative gamma ray source strength from fission products, for a given operating and shutdown time and a given discrete photon energy interval, is obtained by summing the contributions from all individual radioactive isotopes produced by fission.

An indication of the dominant fission product sources after attenuation through thick shields is obtained from the tabulation at the outer surface of the primary shield tank, as shown in Table 5-4.

5.2.5. Neutron Activation Products Source

During operation, coolant activation produces the main source of gamma radiation outside the secondary shield. After shutdown, steel activation and corrosion products that have been activated in the core are the chief gamma source other than the fission products in the fuel.

Oxygen isotopes react with neutrons to provide the significant coolant activities shown in Table 5-5.

Source strengths from steel activation are predominantly determined by the thermal neutron flux levels. However, (n,p) reactions with fast neutrons may yield significant sources either in steel which contains a large amount of nickel and a small amount of cobalt or in regions that possess high fast-to-thermal neutron flux ratios.

Table 5-4. Significant Fission Product Radiation on Radial Centerline at Outside Surface of Primary Shield Tank Lead After Shutdown from 600 Days Operation

Time after shutdown	Activity isotope	Activity half-life	Energy groups	Precursor effecting decay rate isotope	Precursor effecting decay rate half-life	% total dose rate
$\frac{1}{2}$-h	I-132	2.4-h	V	Te-132	77-h	5
	I-134	52-m	IV	Te-134	44-m	5
	I-135	9.7-h	IV	none		3.3
	Kr-87	78-m	VI	none		24.5
	Kr-88	2.8-h	V	none		11.7
	La-140	40-h	IV	Ba-140	12.8-d	11
	Rb-88	18-m	V	Kr-88	2.8-h	5
	Rb-89	15.4-m	VI	none		11
	Sr-92	2.7-h	III	none		0.8
	Others					22.7
	Total					100.0
10-h	I-132	2.4-h	V	Te-132	77-h	7.0
	I-135	6.7-h	III, IV	none		3.4
	Kr-88	2.8-h	V	none		5.0
	La-140	40-h	IV, VI	Ba-140	12.8-d	78.0
	Sr-91	9.7-h	III	none		0.5
	Other					6.1
	Total					100.0

Table 5-4. (Cont'd)

Time after shutdown	Activity			Precursor effecting decay rate		% total dose rate
	isotope	half-life	Energy groups	isotope	half-life	
3-d	I-132	2.4-h	III, V	Te-132	77-h	8.8
	La-140	40-h	IV, VI	Ba-140	12.8-d	86.6
	Pr-144	17-m	V	Ce-144	285-d	4.6
	Total					100.0
100-d	La-140	40.2-h	IV, VI	Ba-140	12.8-d	16
	Pr-144	17-m	III, V	Ce-144	285-d	80
	Rh-106	30-s	IV	Ru-106	1.01-y	4
	Total					100
1000-d	Pr-144	17-m	III, V	Ce-144	285-d	93
	Rh-106	30-s	VI	Ru-106	1.01-y	6
	Total					99

Table 5-5. Neutron Induced Activity in Coolant

Reaction	Type	Yield	Energy (mev)	Half-life
0-16 (n,p) N-16	Fast	0.75	6.1 γ	7.35 sec
0-17 (n,p) N-17	Fast	0.07	7.1 γ	4.14 sec
0-18 (n,γ) 0-19	Thermal	1.0	~1.0 n	29.4 sec
		0.7	1.6 γ	

Application to the NS Savannah reactor and primary shield requires two steps:

1. The dominant volumetric source activities in each material are identified, based on a reference neutron flux level of 10^{10} n/cm^2-second as a function of time before and after shutdown.

2. The variation of the volumetric gamma source strengths are obtained as a function of position using the ratio of the actual neutron flux to the reference level of 10^{10} n/cm^2-second.

The equilibrium-induced activities in type-304 stainless steel and SAE-212 carbon steel by (n,γ) and (n,p) reactions are summarized in Table 5-6 for an infinite operating time, assuming neutron flux levels of 10^{10} n/cm^2-second for both thermal and fast neutrons.

During operation and shutdown the dominant activities are 2.59-hour and 5.28-year CO-60 in the stainless steel and 2.59-hour Mn-56 and 46-day Fe-59 in the carbon steel. The 2.59-hour Mn-56 activity is unimportant 20 hours after shutdown.

5.3. Radiation Shields

5.3.1. Primary Shield Tank

The primary shield attenuates the core neutron and gamma ray sources to such an extent that they do not unduly influence the secondary shield thicknesses. Material

Table 5-6. Equilibrium Neutron Induced Activity in Steels

Material	Composition element	Composition w/o	Parent isotope	Parent % abundance	Parent density atoms/cc	Daughter isotope	Daughter half-life[1]	Daughter photons	Average photon energy mev	Activation cross section[2] barns	Equilibrium activities[3] photons cc-sec
304 SS	W	0.205	W-180	0.14	7.33×10^{16}	W-181	140-d	1.0	1.83	6.58 ± 6.5	4.8×10^{3}
			W-186	28.4	1.49×10^{19}	W-187	24-h	1.1	0.5	22.4 ± 5	3.4×10^{6}
	Co	0.585[4]	Co-59	100	4.56×10^{20}	Co-60	5.28-y	2.00	1.25	23.7 ± 1	2.2×10^{8}
	Cr	19.0	Cr-50	4.31	6.86×10^{20}	Cr-51	27.8-d	0.08	0.32	7.24 ± 3.3	4.0×10^{6}
	Mn	1.5	Mn-55	100	1.26×10^{21}	Mn-56	2.58-h	1.00	0.84	8.83 ± 2	1.1×10^{8}
								0.40	2.00		4.4×10^{7}
	Fe	67.68	Fe-58	0.31	1.76×10^{20}	Fe-59	45.1-d	1.00	1.20	0.59 ± 0.12	1.0×10^{6}
			Fe-54	5.84	3.31×10^{21}	Fe-55	2.94-y	weak	1.40	1.5 ± 0.3	nil
	Ni[5]	9.5	Ni-64	1.16	9.24×10^{19}	Ni-65	2.56-h	0.14	1.01	1.71 ± 0.26	2.2×10^{5}
			Ni-58	67.76	5.13×10^{21}	Co-58	72-d	1	0.805	0.1 ± 0.09	5.1×10^{6}
			Ni-60	26.16	1.98×10^{21}	Co-60	5.28-y	2.00	1.25	0.1 ± 0.1	4.0×10^{6}
	Cu	0.29	Cu-63	69.1	1.50×10^{20}	Cu-64	12.8-h	0.005	1.35	2.57 ± 0.53	1.9×10^{4}
	Mo	0.165	Mo-98	23.75		Mo-99	67-h	0.20	0.78	0.086 ± 0.033	nil
	Sn	0.013	Sn-116	14.24		Sn-117M	14.0-d	2.0	0.162	0.004 ± 0.002	nil
			Sn-118	24.01		Sn-119M	275-d	1.0	0.089	0.007 ± 0.004	nil
SAE-212	Mn	0.9	Mn-55	100	7.55×10^{20}	Mn-56	2.58-h	1.00	0.84	8.83 ± 2	6.7×10^{7}
								0.40	2.00		2.7×10^{7}
Carbon Steel	C	0.35	C-13	1.11	3.26×10^{18}	C-14	5570-y	nil	--	0.00059	nil
	Si	0.30	Si-30	3.05	7.67×10^{18}	Si-31	2.65-h	nil	--	0.072	nil
	S	0.05	S-34	4.215	1.77×10^{19}	S-35	87.1-d	nil	--	0.17	nil
	P	0.04	P-31	100	3.35×10^{20}	P-32	14.30-d	none	--	0.15	none
	Fe	98.36	Fe-58	0.31	2.56×10^{20}	Fe-59	45.1-d	1.00	1.20	0.59 ± 0.12	1.5×10^{6}

1. Half-life abbreviations; y = years, d = days, h = hours.
2. Based on a neutron flux of 10^{10} n/cm^2-second.
3. Activation cross sections are corrected to 505 F and for Maxwellian distribution: σ_{th} (505 F) = 0.658 σ_{th} (72 F).
4. The cobalt content of type-304 stainless steel varies greatly from the maximum limit of approximately 0.75 weight percent. Generally, use of low cobalt steel reduces the concentration to about 0.05 weight percent. Thermal shields, core structural and grid plates are generally kept to 0.2 weight percent.
5. The cross sections for Ni-58 and Ni-60 are for (n,γ) reactions. The Ni-60 and Ni-58 cross sections for fast neutrons are best estimates.

outside the primary shield does not undergo sufficient neutron interaction to become an important source of gamma radiation. It also shields fission product gamma radiation emanating from the core and activation gamma radiation originating in the pressure vessel and insulation. The dose rate is reduced to levels which permit limited access to the equipment within the containment vessel soon after shutdown.

The primary shield consists of an annulus of light water, 33 inches thick, contained in the primary shield tank surrounding the reactor vessel. The water annulus is supplemented at the primary shield tank outer wall by lead, varying in thickness from 1 to 4 inches.

The water in the primary shield tank is cooled by intermediate cooling water through a coil within the shield tank. The water level in the primary shield tank is maintained by addition of water from the intermediate cooling system as necessary. The shield tank water level is indicated and annunciated on the main control console.

5.3.2. Secondary Shield

The secondary shield surrounding the containment vessel is constructed of concrete, lead and polyethylene. Below the containment vessel equator, the shield is a vertical wall of reinforced concrete with a maximum thickness of 4 feet and is extended at the forward end to form a compartment for the low pressure primary auxiliary systems located outside containment. Above the equator, the shielding consists of about 6 inches each of lead and polyethylene in that order placed on the containment vessel shell. Tanks within the inner bottom below the reactor compartment can be filled with water to provide additional lower shielding when the ship is in drydock (see Figure 4-3 and NYS Drawings 529-200-8 and 529-200-9).

The weight of the concrete shield is approximately 1187 tons and that of the lead and polyethylene is 616 tons. The containment vessel alone weighs 250 tons.

5.3.3. Local Shielding

Local shielding is provided on the demineralizers and filters of the primary loop purification system. The design of these shields is based on an assumed fuel pin failure which exposes 5% (363 kg) of the fuel inventory to the primary coolant.

The demineralizers are shielded with $4\frac{1}{2}$ inches of lead. This lead thickness reduces the dose rate adjacent to the demineralizers to 70 mr per hour 1 day after the release occurs.

The filters are shielded with 2 inches of lead. This shielding reduces the dose rate adjacent to the filters to below 200 mr per hour.

5.4. Shield Surveys

5.4.1. Sea Trial Surveys

In order to determine whether the shield design specifications in Table 5-1 had been met, a detailed survey of radiation from the NS Savannah's shield was made during a special sea trial (May 20 to May 25, 1962). The survey was accomplished with the reactor at a power of 69 megawatts during a cruise of 2821 miles in the Atlantic Ocean out of Yorktown, Virginia.

5.4.1.1. Equipment

A complete set of special equipment was developed by ORNL for measurement of the extremely low dose rates specified in the NS Savannah's design. The instrumentation was carefully calibrated against known radiation sources and neutron fluxes and compared to standard instrumentation in a series of experiments at the ORNL Bulk Shielding Facility (BSF). The BSF experiments employed an idealized mockup of the NS Savannah's shield, thus insuring that the comparisons would be valid for the energy spectrum encountered aboard the NS Savannah. A 41-man task force carried out the survey under the general supervision of E. P. Blizzard of ORNL.

5.4.1.2 Results

The results of the detailed gamma-ray and thermal neutron dose rate surveys show that the tolerance limits of 0.5 rem per year or 0.057 mrem per hour are not exceeded in any area where unlimited access is permitted. In the areas to which only the crew is permitted access, the tolerance limit of 5 rem per year or 0.57 mrem per hour, is exceeded only in a small region on D-deck in the water-sampling room against the forward bulkhead. Gamma-ray dose rates of the order of 1.2 to 1.9 mrem per hour were measured at this location. Entry to this area is controlled to insure that no hazard ensues.

The dose rates measured at approximately 80 special points were compared with values predicted by the shield design calculations. Generally, the measured dose rates are considerably lower than the predicted dose rates reflecting the intended conservatism in the design. It may be concluded from the results of the survey that the shield contains no significant voids or cracks and that the stringent dose-rate specifications have been met.

5.4.2 Routine Surveys

A sustaining program of health physics surveys is routinely conducted. Although these surveys are made with conventional portable health physics instruments which do not have the sensitivity of the instruments used for the shield survey, it is concluded that no significant change has occurred in the condition of the shielding.

5.5 Calculated Doses From Failed Fuel Rods

Past radiochemical data indicated that there had been no failed fuel rods in the reactor for approximately 15,000 effective full power hours (EFPH) of operation. During the last voyage before the Shuffle Outage, evidence of a minor fuel rod failure was detected. It appears that small amounts of fission products are released to the primary coolant whenever there is a significant change in reactor power level. Post shuffle operation indicates that the situation still exists; however, this has not limited operation nor access anywhere on the ship.

The shielding is designed to allow operation with failed fuel elements and fission products in the primary system. Continuous purification of the primary coolant restricts the fission product radioactivity levels in the primary coolant during operation and after shutdown. This sidestream purification consists of ion exchange and filtering. There is no known method of predicting the number of leaking pins which will result in a given dose for known operating and decay times; however, the distribution and relative amount of fission product activity in the various components are provided for any assumed amount of fuel exposed to the coolant water in the core. If the maximum desired dose rate is 200 mr per hour adjacent to the primary coolant piping one day after shutdown, it is calculated that a maximum of 363 kg of exposed fuel may be tolerated. Figure 5-1 shows the dose rate from a primary coolant pipe and the primary loop purification system demineralizers as a function of time after shutdown. During operation, the fission product activity is not significant compared to primary and secondary gamma radiation from the core.

Figure 5-1
Dose Rate From Failed Fuel