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NUCLEAR ROCKET OPERATIONS

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REPORT NO. RN-S-0423

TO

AEC-NASA SPACE NUCLEAR PROPULSION OFFICE

NUCLEAR ANALYSES IN SUPPORT OF
E/STS 2-3 PRELIMINARY DESIGN

NERVA PROGRAM

CONTRACT SNP-1

AUGUST 1967

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REPORT NO. RN-S-0423

NUCLEAR ANALYSES IN SUPPORT OF
E/STS 2-3 PRELIMINARY DESIGN

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ABSTRACT

Results are presented for nuclear analyses in support of the preliminary design of Engine Stage Test Stands 2 and 3 (E/STS 2-3) from December 1966 through August 1967. The results reported in two interim reports (RN-TM-0415 and RN-S-0390) are updated and incorporated in part herein.

This report documents the results of nuclear analyses of: (1) site layout, (2) facility shielding, (3) test stand design and (4) engine envelope and nuclear ground test module environment (with and without a facility intermediate shield). Generalized data related to materials activation are also included to facilitate analysis in support of further design efforts.

Conclusions are reached on the basis of the data presented. Some observations are made relative to the impact of possible design changes. The reported nuclear heating information is applicable to the mechanical design of the facility.

Due to the preliminary nature of the facility design, continuing study is required during the iterative process of design evolution. The shielding requirements and nuclear analyses in support of the facility design are reported herein. Mechanical and thermal analyses and detail engineering, which are the prerogative of the architect-engineer, are beyond the scope of this report.

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The Nuclear Analysis Department of Aerojet's Nuclear Rocket Operations wishes to express appreciation for the contributions to these analyses which were made by personnel from Kaiser Engineers. In particular, the technical and design liaison of Dr. J. Busch and the engineering support of Messrs. H. Isakari and R. Francis require specific acknowledgement.

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I. INTRODUCTION

This report presents the results of nuclear analyses performed in support of the preliminary design of the nuclear rocket Engine/Stage Test Stands 2 and 3 (E/STS 2-3). Nuclear radiation environments are described for operating and shutdown conditions; and nuclear requirements for radiation shielding are delineated. Mechanical and thermal aspects of shield design are specifically excluded since these areas are the prerogative of the Architect-Engineer. However, nuclear heat generation rates are presented for use in thermal analysis.

Figure 1 shows an artist's conception of the relative arrangement of the Nuclear Ground Test Module as it is shielded by the facility side shield. The service tower and service arms are also shown. The E/STS 2-3 complex consists of two such test stands with a common underground control center.

The information contained herein updates and supersedes some of the results and conclusions presented in two earlier interim reports^{1,2*}. These analyses are more extensive, computerized solutions than those of the Budgetary Study³, which covered the conceptual design of E/STS 2-3 for budgetary planning purposes. The first interim report dealt with nuclear analyses in support of site layout activities and employed the Budgetary Study shield configurations. The second interim report dealt with nuclear analyses in support of test stand design, and employed a later version of side-shield configuration, designated the March 1967 configuration. Continued study has resulted in the specification of preliminary design configurations for all test stand shields as described herein.

In accordance with the Criteria for Design of E/STS 2-3⁴, all calculations are based on a 5000 Mw nominal power level and a reference NERVA engine configuration^{4,5}, with a reference Nuclear Ground Test Module (NGTM)^{6,7}.

*Superscript numbers refer to the Reference List presented in Section XI.

Figure 1 - Relative Orientation of E/STS 2-3 Stage, Facility Side Shield, and Service Tower

Corresponding operating levels for 10,000 Mw operation can be approximated by multiplying these results by the ratio of the powers (i.e. 2.0). Approximate shutdown radiation levels following 45 minutes of operations at 10,000 Mw can be obtained by multiplication of the 30 minute 5000 Mw values by the ratio of the integrated powers (i.e. $\frac{45 \text{ minutes @ } 10,000 \text{ Mw}}{30 \text{ minutes @ } 5,000 \text{ Mw}} = 3.0$).

In interpreting these results it should be emphasized that all radiation levels are nominal values and that the criteria call for application of a factor of three for all operating radiation levels, and a factor of five for all radiation levels after shutdown, to compensate for uncertainties in the calculations and to allow for engine design evolution. In some instances, different uncertainty factors are appropriate.

For the most part, this report provides working information for evaluation of specific test stand design considerations. Analytical methods, models and sources of radiation are discussed briefly. Normal operational and shutdown radiation levels are reported for both unshielded and shielded conditions. Radiation levels resulting from nuclear accidents are also briefly reviewed. All of the analyses are based on the assumption that the LH₂ propellant tank is empty.

II. SCOPE OF ANALYSES

Emphasis is placed on analyses in support of radiation shield design. Shield configurations are presented, based on nuclear considerations. Radiation heating is quantitatively discussed and the factors governing the design of each shield are stated. In order to evaluate properly the necessity for and effectiveness of the radiation shields, unshielded and shielded radiation environments were analyzed. These analyses included evaluations of the radiation levels at the engine envelope and ground test module, test stand and duct vault, service tower, control center and access tunnels, and adjacent test stand and other NRDS facilities. Isodose contours are superimposed on the plot plan for the complex of two test stands at E/STS 2-3.

Neutron and gamma radiation levels and nuclear heating are analyzed where necessary. Residual activation of structural elements and the NERVA exhaust duct are included. Fission product contamination of the duct is also treated.

Discussion of radiation analyses of accident conditions is brief due to the lack of information available on this subject. This area requires further study when more information becomes available.

III. SUMMARY AND CONCLUSIONS

Preliminary shield designs are presented with the nuclear analysis which supports these designs. An attempt has been made to simplify shielding wherever possible by using plain and borated water. This includes a plain water movable side shield portion compared with the earlier borated water design. Alternative use of boral has been avoided.

Boron could be eliminated from the stationary portion of the side shield if shutdown radiation levels from the facility components are allowed to approach those emanating from the NGTM and associated piping. This is not considered to be desirable for the following reasons:

(a) Facility components are permanent or semipermanent, whereas the Nuclear Ground Test Module (NGTM) will be periodically replaced, thus allowing for built-in control of stage radiation level contributions.

(b) Ground test perturbations are minimized in this approach in a region where neutron population is significant, with copious amounts of moderator material present (LH_2 in NGTM and H_2O in movable side shield). The ground test perturbation to the flight environment levels may become unacceptable.

(c) Detailed calculations of residual radioactivant concentrations in this system are extremely difficult and complex and, therefore, are subject to a fairly large degree of uncertainty. A factor of five is to be applied to the nominal values to cover this effect. Use of a neutron poison such as boron gives additional assurance that unacceptable residual radiation levels will not be encountered.

Elimination of boron in the water of the bottom shield is a more likely candidate. The same arguments apply here as those indicated above for the stationary portion of the side shield, except that: (a) the flight environment at the engine components and tank bottom would be virtually unaffected by this change; and (b) analysis of this shield is much more straightforward due to the fact that straight-line-of-sight reactor leakage is predominant and easiest to analyze, and therefore carries with it a lesser degree of uncertainty.

The ultimate effects of the service arm penetration in the side shield have not been fully evaluated. Only preliminary analyses are available at this time. Redesign in this area is probable.

Preliminary design of the duct plenum shield may be changed after design evolution of the NERVA exhaust duct. This may affect facility shielding in the below-grade region.

The movable side shield must be filled with water when in the lowered position to provide necessary shielding for activation sources below grade.

Because all analyses are based on an empty LH_2 propellant tank, estimates of operating and shutdown leakage levels outside the facility side shield are conservative. With the present height of the side shield, only the last few seconds of expected power operation would be performed with essentially an open-ended shield due to the low level of LH_2 in the tank.

Control Center and access tunnel shield cover requirements are predicated on an accidental deposition of 10% of the fueled portion of the reactor above these areas, immediately following a 30 minute firing at 5000 Mw. The access tunnels can be used as fallout shelters in the event that occupancy of the control center is precluded by airborne or other contamination subsequent to a nuclear accident.

The perturbation of the flight environment, above the engine internal shield, is maintained within 25% by the incorporation of a facility intermediate shield, which is used in combination with an engine intermediate shield to intercept side leakage radiation.

This report summarizes analyses made to date and transmits preliminary shield design requirements. Design of the facility is an iterative process and it is expected that further iterations must be completed before design freezes are instituted. The current report is essentially a first look at many of the problem areas involved and is not expected to result in final design information.

IV. ANALYTICAL METHODS, MODELS, AND SOURCES OF RADIATION

A. ANALYTICAL METHODS

Direct gamma and fast-neutron radiation levels are calculated with the QAD-P5 point kernel integration program⁸, using the IBM-360 digital computer system. This program, developed by the Los Alamos Scientific Laboratory (LASL), was modified by Aerojet's Nuclear Rocket Operations (NRO) relative to the computation involving the neutron moments method. Neutron spectra are computed by selected S_n , discrete ordinates method, neutron transport codes⁹. These codes were also modified by NRO. Computer calculations employing these codes are augmented by hand calculations, such as surface and volume integrations, to arrive at source terms of secondary gamma and residual activation gamma radiation.

Scattered gamma radiation is computed by the GGG and GAMSCAT digital computer programs^{10,11}. NRO has modified the LASL-developed GGG program by incorporation of equations for polynomial fits to build up factors based on the data compiled by Capo¹². An extrapolation of these data to lower energies, which are important in scattering calculations, has also been included. An evaluation of the agreement between results obtained with this method and a more sophisticated Monte Carlo method was recently presented by Aerojet to the American Nuclear Society¹³.

Gamma ray energy flux and particle flux to dose conversion factors and attenuation factors are also based on compilation by Capo¹⁴. Coefficients for neutron flux and heating rate calculations are integral parts of the programs mentioned earlier.

Fast-neutron scattering is calculated by the NEUSCAT¹⁵ computer program on either the IBM-1130 or IBM-360 digital computer system.

Neutron-induced residual activation is calculated with the aid of the ACT-II digital computer program¹⁶. A combination of QAD and S_n codes is used to compute absolute source magnitudes based on the ACT-II results.

B. ANALYTICAL MODELS

Figures 2 and 3 show the basic analytical models used for the portions of the test stand above and below the pad level, respectively. Figures 4 and 5 show plan and elevation views of the side shield in which the service arm penetration can be seen. This penetration accommodates the retractable service arm which enters at an angle of 17° from the normal to the service tower front face.

Details of each of the shields are discussed in a later section. Some refinements in the basic models are necessary for certain calculations,

Figure 2 - Model for Radiation Analysis of E/STS No. 3 Test Stand Preliminary Design Configuration

Figure 3 - Location of Detectors on Engine Envelope and Propellant Tank for Design of Facility Intermediate Shield

Figure 4 - Idealized Plan View Showing Service Arm Penetration in Side Shield

Figure 5 - Idealized Side View Showing Side Shield and Module Support Structure

especially where geometric constraints in laminated shields, such as the intermediate shield, are involved. A discussion of details of this type is not deemed necessary in the context of this report.

The basic models shown here include the definition of an engine envelope. Details of the engine model are not described due to classification limitations in description of the nuclear subsystem. The classified supplement to the E/STS 2-3 Criteria⁴ provides information relative to reactor dimensions, compositions, etc.

C. SOURCES OF RADIATION

Detail information affecting the engine as the prime radiation source is available in the criteria document and its classified supplement⁴. Equivalent point sources can be used to represent the engine for much of the large separation distances involved in E/STS 2-3. Table 1 gives the equivalent point sources of gamma radiation for side leakage and end leakage (toward the propellant tank). These data are given for the operational case and for 1 hour and 24 hours after shutdown following 30 minutes of operation at 5000 Mw. Table 2 gives similar information for operational equivalent fast-neutron point sources.

The total secondary gamma source strengths for the side shield, facility intermediate shield, and bottom shield are given in Table 3. Figures 6 and 7 give the radial and axial secondary gamma source distributions in the side shield. Figures 8 and 9 give secondary source distributions in the facility intermediate shield. Figures 10 and 11 give secondary-source distributions in the bottom shield.

Total volume-integrated source strengths are shown in Tables 4 and 5 for sources of radioactivants resulting from residual neutron-induced

TABLE 1

EQUIVALENT POINT SOURCES OF GAMMA RADIATION
FOR ENGINE LEAKAGE

E (Mev)	5000 Mw Operation Equivalent S_o (Photons/sec)		1 Hour After 30 Minutes at 5000 Mw Equivalent S_o (Photons/sec)		24 Hours After 30 Minutes at 5000 Mw Equivalent S_o (Photons/sec)	
	Side Leakage $\alpha = 90^\circ$	End Leakage $\alpha = 0^\circ$	Side Leakage $\alpha = 90^\circ$	End Leakage $\alpha = 0^\circ$	Side Leakage $\alpha = 90^\circ$	End Leakage $\alpha = 0^\circ$
7.5	2.09×10^{17}	8.59×10^{16}				
6.5	4.14×10^{17}	1.67×10^{17}				
5.5	1.15×10^{18}	4.45×10^{17}				
4.5	3.66×10^{18}	1.30×10^{18}				
3.5	7.66×10^{18}	2.52×10^{18}				
2.8	4.79×10^{18}	1.42×10^{18}	3.30×10^{16}	1.01×10^{16}	1.27×10^{13}	3.85×10^{12}
2.5	6.97×10^{18}	2.03×10^{18}	6.53×10^{16}	1.95×10^{16}	2.14×10^{14}	6.40×10^{13}
2.0	1.05×10^{19}	2.78×10^{18}	7.12×10^{16}	1.93×10^{16}	2.88×10^{15}	7.84×10^{14}
1.5	1.42×10^{19}	3.16×10^{18}	2.08×10^{17}	4.70×10^{16}	1.65×10^{15}	3.74×10^{14}
1.0	2.14×10^{19}	3.59×10^{18}	1.79×10^{17}	3.04×10^{16}	3.16×10^{15}	5.36×10^{14}
0.7	3.48×10^{19}	4.62×10^{18}	3.62×10^{17}	4.84×10^{16}	1.75×10^{16}	2.33×10^{15}
0.3	2.38×10^{19}	1.54×10^{18}	1.12×10^{17}	7.05×10^{15}	5.25×10^{15}	3.29×10^{14}
TOTAL	1.295×10^{20}	2.37×10^{19}	1.03×10^{18}	1.82×10^{17}	3.07×10^{16}	4.42×10^{15}

NOTE: α = Meridian angle

TABLE 2

EQUIVALENT POINT SOURCES OF FAST NEUTRONS
FOR ENGINE LEAKAGE

E(Mev)	5000 Mw Operation Equivalent Source (Neutrons/sec)	
	Side Leakage $\alpha=90^\circ$	End Leakage $\alpha=0^\circ$
10.0	1.13×10^{16}	1.16×10^{16}
8.0	3.91×10^{16}	4.03×10^{16}
6.0	1.11×10^{17}	1.14×10^{17}
5.0	1.42×10^{17}	1.46×10^{17}
4.0	2.57×10^{17}	2.64×10^{17}
3.0	5.16×10^{17}	5.29×10^{17}
2.0	2.98×10^{18}	3.07×10^{18}
1.0	4.71×10^{18}	4.84×10^{18}
Total	8.77×10^{18}	9.01×10^{18}

TABLE 3

SECONDARY GAMMA SOURCE STRENGTHS IN SIDE SHIELD,
BOTTOM SHIELD, AND FACILITY INTERMEDIATE SHIELD

<u>REGION</u>		<u>SOURCE STRENGTH</u> <u>(Mev/Sec)</u>
<u>Side Shield</u>		
(a)	Movable Portion	3.00×10^{20}
(b)	Module Support Ring	8.83×10^{18}
(c)	Upper Portion	4.30×10^{18}
<u>Bottom Shield</u>		
(a)	1" Upper Aluminum Skin	8.17×10^{18}
(b)	Borated Water	2.44×10^{19}
<u>Intermediate Shield</u>		
(a)	Lower Aluminum Skin (1/2")	3.35×10^{18}
(b)	6" Borated Water & Lower Aluminum Flow Baffle	1.78×10^{19}
(c)	1" Aluminum Structural Plate	1.90×10^{17}
(d)	2" Borated Aluminum	1.13×10^{17}
(e)	2" Borated Water	1.37×10^{17}
(f)	5" Borated Aluminum	3.96×10^{16}
(g)	2" Borated Water and Upper Aluminum Flow Baffle	2.32×10^{16}
(h)	Upper Aluminum Skin (1")	1.47×10^{15}

Figure 6 - Side Shield Radial Source Distribution

Figure 7 - Side Shield Axial Source Distribution

Figure 8 - Facility Intermediate Shield Relative Radial Secondary Source Distribution

Figure 9 - Facility Intermediate Shield Relative Axial Secondary Source Distribution

Figure 10 - Bottom Shield Relative Radial Secondary Source Distribution

Figure 11 - Bottom Shield Relative Axial Secondary Source Distribution

TABLE 4

RESIDUAL ACTIVATION GAMMA SOURCE STRENGTHS IN SIDE SHIELD, BOTTOM SHIELD,
FACILITY INTERMEDIATE SHIELD AND SERVICE TOWER,
TWO WEEKS AFTER SHUTDOWN FOLLOWING 30 MIN - 5000 Mw RUN

Identification	Source Strength (Mev/sec)
Service Tower Aluminum Piping	9.22×10^7
Service Tower Steel Piping at Upper Service Arm	4.37×10^{10}
Service Tower Steel Platform at Upper Service Arm	3.90×10^{10}
Service Tower Concrete Activation	6.61×10^9
Service Tower Rebar Activation	2.31×10^{10}
Activation of Top of Side Shield	5.34×10^9
Activation of 6061 Aluminum Liner of Support Ring	4.56×10^{10}
Stainless Steel Activation in Top of Plenum	2.40×10^{12}
Bottom Shield Aluminum Activation	9.74×10^{10}
Activation of Lower Aluminum Skin of Intermediate Shield	1.45×10^{11}
Activation of Aluminum Side Skin of Intermediate Shield	1.33×10^9
Activation of 2-Inch Borated Aluminum Plate of Intermediate Shield	1.33×10^{10}
Activation of 5-Inch Borated Aluminum Plate of Intermediate Shield	5.15×10^9

TABLE 5

RESIDUAL GAMMA SOURCE STRENGTH IN NERVA EXHAUST DUCT FOLLOWING
5000 Mw - 30 MIN RUN

TIME AFTER SHUTDOWN	SOURCE STRENGTH (Mev/Sec)	
	ACTIVATION (347SS)	FISSION PRODUCTS
24 Hours	1.47×10^{14}	2.86×10^{14}
2 Weeks	2.13×10^{13}	1.48×10^{13}

activation. These data apply to the condition following 14-days decay subsequent to a 30-minute run at 5000 Mw. The comparable source strengths for other decay times are discussed later under specific shielding discussions.

V. UNSHIELDED ENGINE RADIATION ENVIRONMENT

The operational and shutdown radiation environments which result when no facility shielding considerations are included are described in this section. Shielded environments are described in a later section.

A. OPERATIONAL ENVIRONMENT

All operational values are reported for a 5000-Mw power level. Extrapolation to 10,000 Mw or any other power level can be approximated by multiplying by a direct ratio of the powers.

1. Distribution of Engine Leakage Radiation

Figure 12 shows how the fast-neutron flux and gamma ray dose rate resulting from engine leakage varies as a function of meridian angle measured from the engine axis. Zero degrees is in the direction of the NGTM tank and 180° is in the direction of the NERVA exhaust duct. These data are for an engine with no internal engine shield in the pressure vessel. Isoflux and isodose contours of engine leakage radiation are given in the criteria document⁴.

Table 6 gives similar data at 24 locations on the engine envelope and propellant tank bottom. Again, these are for an engine with no internal or external shielding.

Figure 12 - Fast Neutron Flux and Gamma Dose Rate Leakage at 5000 Mw on 7.5-Foot Meridian Ring

TABLE 6

ENGINE ENVELOPE AND GROUND-TEST-MODULE RADIATION LEVELS DUE TO ENGINE LEAKAGE @ 5000 Mw

Detector Identification	Detector Location		Gamma Ray Dose Rate, (Rad(c)/hr)	Fast-Neutron Flux, (n/cm ² -sec) E _n > 0.9 Mev
	Radius, ft	Distance Above or Below Core Center, ft*		
1	0	4.4	2.61×10^8	3.35×10^{13}
2	0	9.8	5.51×10^7	7.52×10^{12}
3	0	15.0	2.22×10^7	3.02×10^{12}
4	0	18.2	1.34×10^7	1.80×10^{12}
5	3.2	0	2.33×10^9	5.79×10^{13}
6	3.2	4.4	3.74×10^8	1.78×10^{13}
7	6.9	21.3	1.08×10^7	8.60×10^{11}
8	12.8	26.2	2.07×10^6	4.23×10^{11}
9	16.2	33.3	1.57×10^6	—
10	16.2	39.4	3.18×10^5	—
11	7.2	0	4.40×10^9	1.41×10^{13}
12	7.2	3.1	3.43×10^8	1.86×10^{13}
13	7.2	5.0	2.47×10^8	1.66×10^{13}
14	8.0	9.8	7.62×10^7	3.30×10^{12}
15	9.7	15.0	3.17×10^7	1.45×10^{12}
16	4.5	15.0	2.13×10^7	2.35×10^{12}
17	10.8	21.3	1.46×10^7	8.37×10^{11}
18	16.5	27.0	9.70×10^6	4.56×10^{11}
19	3.2	9.8	4.22×10^7	5.21×10^{12}
20	7.2	-13.7	1.48×10^8	2.26×10^{13}
21	3.2	-2.4	6.39×10^8	1.23×10^{13}
22	0	-2.4	8.77×10^9	1.36×10^{15}
23	0	-6.6	1.23×10^9	2.06×10^{14}
24	0	-13.7	2.39×10^8	4.07×10^{13}

*Distance designated with a (-) are below core center

2. Radiation Levels in Adjacent Vicinity

Figures 13 and 14 give the gamma dose rates and fast-neutron fluxes, respectively, to a distance of 1200 ft from the test stand centerline. Figure 15 extends these data to 9000 ft. Figure 16 shows the results of the superposition of an unshielded engine operating gamma isodose map on the current plot plan¹⁷ of the E/STS 2-3 complex. Subsequently, essential test complex component locations can be related readily to the dose rates that would exist at these locations with no facility shielding. The combination of Figures 13, 14, 15, and 16 could give similar isodose and isoflux data to 9000 ft if this were desired. From the preceding figures, the gamma dose rate and fast neutron flux at the adjacent stand are seen to be 8.0×10^2 rad/hr and 1.5×10^8 n/cm²-sec for unshielded operation. Similarly, Figure 15 shows that the gamma and neutron dose rates at ETS-1, approximately 10,000 ft away, are 2.0×10^{-2} and 1.0×10^{-4} rad/hr, respectively.

3. Control Center and Access Tunnel Radiation Levels

The gamma radiation levels were analyzed to determine dose rates in the control center and access tunnels which would result from unshielded engine operation. These data are presented in Table 7 as a function of concrete shielding thickness and distance from the test stand centerline. It should be emphasized that these data in no way affect the design of the shielding in these areas and are provided for information purposes only.

4. Service Tower and Test Pad Radiation Levels

Tables 8 and 9 give the radiation levels at the face of the service tower and at the surface of the test pad for operation with no facility shielding. For comparative purposes, the unshielded measured radiation levels on the shadow shield wall and pad at Test Cell A are given for NRX-A3 operation in Table 10 and for NRX-A5 operation in Table 11.

Figure 13 - Unshielded Gamma Ray Dose Rate vs Distance from
Operating 5000 Mw Engine (No Propellant Tank)

Figure 14 - Unshielded Fast Neutron Flux vs Distance from
Operating 5000 Mw Engine (Direct with No Scattering)

Figure 15 - Fast Neutron and Gamma Ray Dose Rate from Operating Engine vs Horizontal Distance from Core Center, 5000 Mw

Figure 16 - Gamma Doses (rad (c)/hr) from E/STS No. 3 Dummy Operation at 5000 Mw with No Side Shield

TABLE 7

OPERATING GAMMA DOSE RATES UNDER VARIOUS THICKNESSES OF CONCRETE
VS DISTANCE FROM TEST STAND WITHOUT FACILITY SIDE SHIELD

<u>Distance,</u> <u>ft</u>	<u>Concrete</u> <u>Thickness,</u> <u>ft</u>	<u>Gamma Ray</u> <u>Dose Rate,</u> <u>Rad(c)/hr</u>
30	0	1.26×10^8
50	0	8.5×10^6
	1	4.8×10^4
	2	7.1×10^2
	4	5.7×10^{-1}
75	0	3.2×10^6
100	0	1.9×10^6
	1	1.9×10^2
	2	1.2×10^{-1}
150	0	9.0×10^5
200	0	5.8×10^5
	1	2.5×10^{-2}
300	0	3.0×10^5
500	0	9.9×10^4
750	0	2.9×10^4
1000	0	1.0×10^4

TABLE 8

OPERATING RADIATION LEVELS AT SERVICE TOWER - NO FACILITY SHIELDING

Dose Point	Dose Point Location		100% LH ₂ in			
	Vertical Distance Above or Below Core Center, ft	Horizontal Distance From C, ft	No LH ₂ in Ground Test Module		Ground Test Module	
			Gamma Dose Rate, (Rad(c)/hr)	Fast Neutron Flux, (n/cm ² -sec) E _n > 0.9 Mev	Gamma Dose Rate, (Rad(c)/hr)	Fast Neutron Flux, (n/cm ² -sec) E _n > 0.9 Mev
1	-13.1	0	1.1 x 10 ⁷	5.59 x 10 ¹¹	1.1 x 10 ⁷	5.59 x 10 ¹¹
2	- 6.6	0	1.1 x 10 ⁷	3.73 x 10 ¹¹	1.1 x 10 ⁷	3.73 x 10 ¹¹
3	0	0	1.2 x 10 ⁷	3.87 x 10 ¹¹	1.2 x 10 ⁷	3.87 x 10 ¹¹
4	6.6	0	1.2 x 10 ⁷	5.66 x 10 ¹¹	1.2 x 10 ⁷	5.66 x 10 ¹¹
5	13.1	0	1.2 x 10 ⁷	5.88 x 10 ¹¹	1.2 x 10 ⁷	5.88 x 10 ¹¹
6	19.7	0	9.7 x 10 ⁶	5.04 x 10 ¹¹	9.7 x 10 ⁶	5.04 x 10 ¹¹
7	26.2	0	7.6 x 10 ⁶	4.13 x 10 ¹¹	7.6 x 10 ⁶	4.13 x 10 ¹¹
8	32.8	0	5.7 x 10 ⁶	2.66 x 10 ¹¹	5.7 x 10 ⁶	2.66 x 10 ¹¹
9	39.4	0	4.4 x 10 ⁶	1.92 x 10 ¹¹	4.4 x 10 ⁶	1.92 x 10 ¹¹
10	45.9	0	3.3 x 10 ⁶	1.40 x 10 ¹¹	3.3 x 10 ⁶	1.40 x 10 ¹¹
11	52.5	0	2.6 x 10 ⁶	1.08 x 10 ¹¹	2.6 x 10 ⁶	1.08 x 10 ¹¹
12	55.8	0	2.2 x 10 ⁶	9.53 x 10 ¹⁰	2.2 x 10 ⁶	9.53 x 10 ¹⁰
13	59.0	0	2.0 x 10 ⁶	8.63 x 10 ¹⁰	2.0 x 10 ⁶	8.63 x 10 ¹⁰
14	62.3	0	1.8 x 10 ⁶	7.86 x 10 ¹⁰	1.8 x 10 ⁶	7.86 x 10 ¹⁰
15	65.6	0	1.6 x 10 ⁶	7.23 x 10 ¹⁰	1.6 x 10 ⁶	7.23 x 10 ¹⁰
16	68.9	0	1.4 x 10 ⁶	6.73 x 10 ¹⁰	1.4 x 10 ⁶	--
17	72.2	0	1.3 x 10 ⁶	6.25 x 10 ¹⁰	1.3 x 10 ⁶	--
18	75.4	0	1.2 x 10 ⁶	5.84 x 10 ¹⁰	9.8 x 10 ⁵	--
19	78.7	0	1.0 x 10 ⁶	5.51 x 10 ¹⁰	6.5 x 10 ⁵	--
20	82.0	0	9.6 x 10 ⁵	5.18 x 10 ¹⁰	3.0 x 10 ⁵	--
21	85.3	0	8.8 x 10 ⁵	4.91 x 10 ¹⁰	1.9 x 10 ⁵	--
22	88.6	0	8.0 x 10 ⁵	4.67 x 10 ¹⁰	1.3 x 10 ⁵	--
23	91.8	0	7.4 x 10 ⁵	4.44 x 10 ¹⁰	8.9 x 10 ⁴	--
24	0	5	1.17 x 10 ⁷	3.76 x 10 ¹¹	1.17 x 10 ⁷	5.7 x 10 ¹¹
25	0	10	1.12 x 10 ⁷	3.59 x 10 ¹¹	1.12 x 10 ⁷	3.59 x 10 ¹¹
26	0	15	1.03 x 10 ⁷	3.42 x 10 ¹¹	1.03 x 10 ⁷	3.42 x 10 ¹¹
27	0	20	9.43 x 10 ⁶	2.92 x 10 ¹¹	9.43 x 10 ⁶	2.92 x 10 ¹¹
28	0	25	8.44 x 10 ⁶	2.69 x 10 ¹¹	8.44 x 10 ⁶	2.69 x 10 ¹¹
29	0	27	8.04 x 10 ⁶	2.57 x 10 ¹¹	8.04 x 10 ⁶	2.57 x 10 ¹¹

TABLE 9

RADIATION LEVELS AT TEST PAD DURING 5000 Mw OPERATION
WITH NO FACILITY SHIELDS

Radial Distance From Axis, ft	Gamma Ray Dose Rate Due to Engine Leakage, (Rad(c)/hr)	Fast Neutron Flux Due to Engine Leakage, (n/cm ² -sec) E _n > 0.9 Mev
23	3.03×10^7	2.67×10^{12}
28	2.21×10^7	1.61×10^{12}
33.3	1.60×10^7	9.83×10^{11}
40	1.13×10^7	5.96×10^{11}
50	7.22×10^6	3.95×10^{11}
75	3.11×10^6	1.16×10^{11}

TABLE 10

MEASURED RADIATION LEVELS AT TEST CELL A PAD AND WALL DURING NRX-A3 - POWER @ 1100 Mw

Location	Distance Above or Below Core Center, ft	Distance North or West of Core Center, ft	Distance East or West of Core Center, ft	Gamma Ray Dose Rate, (Rad(c)/hr)	Fast-Neutron Flux (n/cm ² -sec)	
					E _n >0.9 Mev Fission Foil Interpolation	E _n >2.9 Mev (Nickel)
TCW-1	13+	5N	13W	1.11×10^7	1.40×10^{12}	--
2	8+	5N	13W	1.89×10^7	1.69×10^{12}	--
3	3+	5N	13W	--	2.96×10^{12}	--
4	0	5N	13W	4.62×10^7	2.60×10^{12}	--
5	5-	0	13W	3.30×10^7	2.62×10^{12}	--
6	7.5-	0	13W	2.47×10^7	2.00×10^{12}	--
7	10.0-	0	13W	1.85×10^7	9.00×10^{11}	--
8	0	0	13W	4.53×10^7	2.58×10^{12}	--
9	0	0	13W	3.21×10^7	1.14×10^{12}	--
10	Neutronics Tank C		13W	1.72×10^7	--	--
11	0		13W	1.02×10^7	2.01×10^{12}	--
TP-1	15-	5N	10E	5.77×10^6	2.77×10^{11}	1.22×10^9
2	15-	5N	0	3.95×10^6	--	1.19×10^9
3	15-	5N	10W	3.87×10^6	--	1.04×10^9
4	15-	5N	20W	7.35×10^6	5.91×10^{11}	5.99×10^9
5	15-	10N	10E	7.25×10^6	--	--
6	15-	10N	0	4.53×10^6	--	--
7	15-	10N	10W	8.73×10^6	--	--
8	15-	15N	10E	1.01×10^7	--	--
9	15-	15N	0	7.50×10^6	3.16×10^{11}	7.22×10^9
10	15-	15N	10W	8.24×10^6	--	--
11	15-	20N	10E	8.40×10^6	4.80×10^{11}	--
12	15-	20N	0	6.68×10^6	5.00×10^{11}	--
13	15-	20N	10W	7.09×10^6	5.40×10^{11}	--
14	15-	28.5N	10E	6.01×10^6	--	--
15	15-	28.5N	0	4.94×10^6	1.90×10^{11}	--
16	15-	28.5N	10W	5.44×10^6	--	--

TABLE 11

MEASURED GAMMA DOSE RATES ON TEST CELL A PAD AND WALL
DURING OPERATION OF NRX-A5

Location	Distance Above or Below Core Center, ft	Distance North of Core Center, ft	Distance East or West of Core Center, ft	Gamma Ray Dose Rate @1120 Mw, (Rad(c)/hr)
TCW-1	0	65	13W	1.05×10^7
TCW-2	0	40	13W	2.32×10^7
TCW-3	0	5	13W	1.83×10^8
TCW-4	3	5	13W	1.47×10^8
TCW-5	8	5	13W	1.06×10^8
TCW-6	13	5	13W	5.72×10^7
TP-1	17.5	40	10W	5.32×10^6
TP-2	17.5	60	10W	2.46×10^6
TP-3	17.5	5	30E	8.77×10^6
TP-4	17.5	5	50E	6.67×10^6

5. Service Tower Concrete Heating

Figure 17 shows the radiation heating as a function of depth into the concrete face of the service tower for unshielded operation at 5000 Mw. Figure 18 gives the variation in total concrete heating rate versus elevation on the service tower for unshielded operation.

6. Duct Vault Radiation Levels and Nuclear Heating

Table 12 gives the gamma dose rate and fast-neutron flux along the NERVA Exhaust Duct, with no facility shielding. Table 13 gives the duct vault concrete radiation heating with no facility shielding. These values do not include the effect of duct collar shielding.

B. SHUTDOWN ENVIRONMENT

The shutdown environments associated with the condition where no facility shielding is used are described in this section. All data are based on a single 30-minute run at 5000 Mw. An empty propellant tank is assumed in all calculations. This assumption particularly affects the air-scatter calculations.

1. Distribution of Shutdown Engine Leakage Radiation

The angular distribution of engine leakage can be roughly approximated by multiplying the meridian ring distribution of Figure 12 by the ratio of the equivalent point source for shutdown and operating conditions given in Table 1. For side leakage ($\alpha = 90^\circ$) at 24 hours this ratio is 2.4×10^{-4} . That is the ratio by which the full power data must be reduced to arrive at the dose rates 24 hours after shutdown following 30 minutes of operation at 5000 Mw. Similarly the end leakage ratio is 1.9×10^{-4} . Essentially then, 24 hours after shutdown the dose rates are 2×10^{-4} times the 5000 Mw operational numbers.

Figure 17 - Radiation Heating Rates in E/STS 2-3 Service Tower Opposite Core Center with No Facility Shields

Figure 18 - Total Concrete Heating vs Elevation on Face of Service Tower

TABLE 12

UNSHIELDED GAMMA DOSE RATE AND FAST-NEUTRON FLUX
ON SURFACE OF NERVA EXHAUST DUCT

Detector Location, ft		Dose Rate, (Rad(c)/hr)	Fast-Neutron Flux, (n/cm ² -sec)
Radius	Distance Below Top of Duct		
5' 10"	0	8.8×10^7	1.5×10^{13}
5' 4"	6' 10"	5.1×10^7	8.8×10^{12}
4' 11"	13' 9"	3.3×10^7	5.7×10^{12}
4' 5"	20' 7"	2.3×10^7	4.0×10^{12}

TABLE 13

CONCRETE RADIATION HEATING IN DUCT VAULT DURING
OPERATION WITH NO FACILITY SHIELD

<u>Detector Location, ft</u>		<u>Heating Rates, Btu/in³-sec</u>
<u>Side Surface of Vault</u>	<u>Z-Distance Below Top Surface</u>	
	0.0	8.83×10^{-3}
	10.0	5.33×10^{-3}
	20.0	3.17×10^{-4}
	30.0	1.58×10^{-4}
	40.0	1.26×10^{-4}
<u>Bottom of Vault</u>	<u>X-Distance From Core Centerline</u>	
	0.0	2.01×10^{-3}
	6.5	1.11×10^{-13}
	13.0	6.23×10^{-6}
<u>Top Surface of Vault</u>	<u>X-Distance Along Top Surface</u>	
	0.5	1.02×10^{-2}
	1.0	1.12×10^{-2}
	2.0	2.82×10^{-3}

Figure 19 gives the necessary information to relate the 24-hour decay data, reported herein, to data at any other times after shutdown. These data are based on Perkins and King's evaluation of fission product decay¹⁸.

Tables 14, 15, and 16 give the 1- and 24-hour shutdown engine leakage radiation levels at the engine envelope and tank bottom, the service tower, and the pad, respectively. Dose rates for other decay periods can be obtained by applying the relative decay curve given in Figure 19.

2. Radiation Levels in Adjacent Vicinity

Figures 20, 21, 22, and 23 give the unshielded gamma dose rates as a function of distance, to 1200 ft from the test stand, for 1- and 24-hour decay times. Figure 24 gives an extension of these data to 9000 ft. Air scattering is the principal contributor at these greater distances. Ground roughness effects due to changes in terrain are not considered in these calculations.

3. Control Center and Access Tunnel Radiation Levels

Table 17 gives shutdown gamma dose rates as a function of depth in concrete for various distances from the test stand. These data are for information purposes only and do not affect the design of shielding cover for these areas.

4. Engine Leakage Radiation Levels at Service Tower and Test Pad

Tables 15 and 16 present the engine leakage radiation levels at the service tower and pad for 1- and 24-hour decay after 30 minutes of operation at 5000 Mw. These data apply for shutdown after normal operation, with the side shield in its lowered position.

Figure 19 - Total Gamma Ray Energy Released After Operation at 1 Watt for 1 Hour

TABLE 14

GAMMA RADIATION LEVELS AT ENGINE ENVELOPE AND GROUND-TEST MODULE AFTER SHUTDOWN
WITH INTERMEDIATE SHIELD

Identification	Detector Location		Engine Leakage Gamma Ray Dose Rate (Rad(c)/hr)**	
	Radius, ft	Distance Above or Below Core Center ft*	1 Hour After Shutdown	24 Hours After Shutdown
1	0	4.4	1.39×10^6	2.57×10^4
2	0	9.8	3.02×10^5	5.82×10^3
3	0	15.0	1.22×10^5	2.34×10^3
4	0	18.2	7.21×10^4	1.37×10^3
5	3.2	0	1.47×10^7	3.33×10^5
6	3.2	4.4	2.36×10^6	5.48×10^4
7	6.9	21.3	6.15×10^4	1.27×10^3
8	12.8	26.2	4.37×10^4	9.37×10^2
9	16.2	33.3	2.70×10^4	5.78×10^2
10	16.2	39.4	1.96×10^4	4.15×10^2
11	7.2	0	2.82×10^6	6.59×10^4
12	7.2	3.1	2.26×10^6	5.17×10^4
13	7.2	5.0	1.58×10^6	3.72×10^4
14	8.0	9.8	4.78×10^5	1.18×10^3
15	9.7	15.0	1.95×10^5	4.43×10^3
16	4.5	15.0	1.19×10^5	2.36×10^3
17	10.8	21.3	8.79×10^5	1.94×10^3
18	16.5	27.0	5.94×10^4	1.33×10^3
19	3.2	9.8	2.27×10^5	4.28×10^3
20	7.2	-13.7	1.00×10^6	2.53×10^4
21	3.2	-2.4	3.77×10^6	8.04×10^4
22	0	-2.4	5.97×10^7	1.52×10^6
23	0	-6.6	8.48×10^6	2.18×10^5
24	0	-13.7	1.63×10^6	4.17×10^4

* Distances designated with (-) are below core center

** Based on 30-minutes operation at 5000 Mw

TABLE 15

GAMMA RAY RADIATION LEVELS ON FACE OF SERVICE TOWER AFTER 30 MINUTES
AT 5000 Mw, WITHOUT FACILITY SHIELD, NO LH_2 IN GTM

Dose Point	Dose Point Location		Gamma Ray Dose Rate, (Rad(c)/hr)	
	Vertical Distance Above or Below Core Center, ft	Horizontal Distance From C, ft	1 Hour After Shutdown	24 Hours After Shutdown
1	-13.1	0	7.25×10^4	1.70×10^3
2	- 6.6	0	7.21×10^4	1.66×10^3
3	0	0	7.57×10^4	1.75×10^3
4	6.6	0	7.95×10^4	1.85×10^3
5	13.1	0	7.44×10^4	1.75×10^3
6	17.7	0	6.14×10^4	1.44×10^3
7	26.2	0	4.76×10^4	1.11×10^3
8	32.8	0	3.62×10^4	8.35×10^2
9	39.4	0	2.76×10^4	6.32×10^2
10	45.9	0	2.08×10^4	4.74×10^2
11	52.5	0	1.52×10^4	3.55×10^2
12	55.8	0		
13	59.0	0	1.22×10^4	2.75×10^2
14	62.3	0		
15	65.6	0	9.72×10^3	2.16×10^2
16	68.9	0		
17	72.2	0	7.37×10^3	1.64×10^2
18	75.4	0		
19	78.7	0	4.46×10^3	9.55×10^1
20	82.0	0		
21	55.3	0	3.02×10^3	6.20×10^1
22	88.6	0		
23	91.8	0	2.66×10^3	5.43×10^{-1}
24	0	5	7.44×10^4	1.72×10^3
25	0	10	7.11×10^4	1.64×10^3
26	0	15	6.59×10^4	1.52×10^3
27	0	20	5.99×10^4	1.38×10^3
28	0	25	5.36×10^4	1.23×10^3
29	0	27	5.11×10^4	1.18×10^3

TABLE 16

RADIATION LEVELS AT TEST PAD AFTER 30 MINUTES AT 5000 Mw OPERATION
WITHOUT FACILITY SHIELD
(ENGINE IN PLACE)

Radial Distance From Axis, ft	Gamma Ray Dose Rate - Rad(c)/hr	
	1 Hour After Shutdown	24 Hours After Shutdown
23	1.97×10^5	4.74×10^3
28	1.43×10^5	3.45×10^3
33.3	1.03×10^5	2.44×10^3
40	7.24×10^4	1.70×10^3
50	4.61×10^4	1.07×10^3
75	3.70×10^4	4.51×10^2

Figure 20 - Unshielded Gamma Ray Dose Rate vs Distance 1 Hour
After 30 Minutes at 5000 Mw (No Propellant Tank)

Figure 22 - Unshielded Gamma Ray Dose Rate vs Distance 24 Hours
After 30 Minutes at 5000 Mw (No Propellant Tank)

Figure 23 - Gamma Doses (rad (c)/hr) from E/STS No. 3, 24 Hours After Shutdown Following 30 Minutes at 5000 Mw with No Side Shield

Figure 24 - Unshielded Gamma Ray Dose Rate vs Distance 1 and 24 Hours After 30 Minutes at 5000 Mw (No Propellant Tank)

TABLE 17

GAMMA DOSE RATES UNDER VARIOUS THICKNESSES OF CONCRETE
VS DISTANCE FROM TEST STAND 1 HOUR AFTER SHUTDOWN
WITHOUT FACILITY SIDE SHIELD

<u>Distance,</u> <u>ft</u>	<u>Thickness,</u> <u>ft</u>	<u>Gamma Ray</u> <u>Dose Rate,</u> <u>Rad(c)/hr</u>
30	0	1.20×10^5
50	0	4.8×10^4
	1	4.88×10^1
	2	1.22×10^{-1}
	4	*
	6	
75	0	2.12×10^4
100	0	1.15×10^4
	1	4.49×10^{-2}
	2	
150	0	4.74×10^3
200	0	2.50×10^3
	1	
300	0	9.77×10^2
500	0	2.70×10^2
	1	
750	0	8.58×10^1
1000	0	3.44×10^1
	1	

*Indicates no increase over background levels.

5. Duct, Vault Radiation Levels

Table 18 gives the gamma radiation levels in the duct vault due to engine leakage with no facility shielding, following 24 hours decay after a 30-minute firing at 5000 Mw. Table 19 gives activation and fission product dose rate contributions above pad, two weeks after shutdown following a 30 minute - 5000 Mw run, with no duct cover shield in place.

TABLE 18

UNSHIELDED GAMMA RADIATION LEVELS IN DUCT VAULT
24 HOURS AFTER SHUTDOWN OF 30 MIN 5000 MW RUN
(ENGINE IN PLACE)

<u>Side Surface of Vault</u>	<u>Z-Distance Below Top Surface, ft</u>	<u>Dose Rate, (Rad(c)/hr)</u>
	0	4.49×10^3
	10	1.23×10^2
	20	5.25×10^1
	30	2.37×10^1
	40	1.03×10^1
<u>Top Surface Of Vault</u>	<u>X-Distance Along Top Surface , ft</u>	
	0.5	5.36×10^3
	1.0	5.91×10^3
	2.0	6.87×10^3
<u>Bottom of Vault</u>	<u>X-Distance from Core Centerline, ft</u>	
	0.0	5.68×10^2
	6.5	*
	13.0	2.02×10^0

*Indicates no increase over background levels

TABLE 19

ACTIVATION AND FISSION PRODUCT DOSE-RATE CONTRIBUTIONS ABOVE PAD
TWO WEEKS AFTER SHUTDOWN FOLLOWING 30 MIN - 5000 MW RUN
WITH NO DUCT COVER SHIELD
(ENGINE REMOVED)

Detector Location		Dose Rates (Rad(c)/hr)						Totals
X-Distance From Core Centerline, ft	Z-Distance Above Pad, ft	Fiss. Prod. in Diff. Mouth	Activation in Top of Plenum	Activation in Diff. Mouth	Activation in Bottom Shield	Activation in Lower Al Skin of Int. Shield	Activation in Al Side Skin of Int. Shield	
0	0	3.286(+1)	3.705(0)	4.975(1)	1.942(-2)	5.227(-3)	8.284(-4)	8.634(1)
0	1	2.828(+1)	5.040(0)	4.283(1)	1.344(-2)	4.312(-3)	7.511(-4)	7.617(1)
3	1	3.077(+1)	6.298(0)	4.660(1)	1.095(-2)	3.553(-3)	7.360(-4)	8.368(1)
6	1	2.725(+1)	1.475(+1)	4.131(1)	4.710(-3)	2.293(-3)	2.015(-3)	8.331(1)
7.15	1	1.366(+1)	1.257(+1)	2.107(1)	5.976(-3)	2.688(-3)	2.957(-3)	4.731(1)
7.38	1	1.176(+1)	1.126(+1)	1.817(1)	5.171(-3)	2.387(-3)	2.015(-3)	4.120(1)
9	1	3.99(0)	5.105(0)	5.924(0)	2.430(-3)	1.344(-3)	9.265(-4)	1.502(1)
12	1	9.13(-1)	2.031(0)	1.283(0)	7.554(-4)	7.289(-4)	9.568(-5)	4.229(0)
15.25	1	3.22(-1)	1.074(0)	4.657(-1)	2.532(-4)	2.027(-4)	5.192(-5)	1.862(0)
0	10	6.66(0)	1.531(0)	1.010(1)	9.264(-4)	1.076(-3)	5.662(-4)	1.829(1)
0	20	2.44(0)	4.922(-1)	3.685(0)	5.187(-4)	6.607(-4)	2.350(-4)	6.619(0)
0	30	1.24(0)	2.311(-1)	1.874(0)	3.365(-4)	3.673(-4)	1.239(-4)	3.346(0)

VI. SHIELD DESIGN

The nuclear radiation shielding requirements are described in this section, with a brief description of the various test-complex shields. The nuclear aspects of shield design are discussed, including radiation heating. Mechanical and thermal aspects of shield design are excluded, since these are the prerogative of the Architect-Engineer.

In addition to presenting shield configuration and radiation heating data, this section also includes a brief discussion on the factors affecting shield design and materials selection for each facility shield.

The facility shields are designed to meet the Criteria Document Requirements 4 and 5. Review of these criteria indicated the need for an additional basis for design for the Test Stand with the engine removed, to augment the "extension of pad" access requirement 4. This latter criteria requirement is interpreted to mean beyond the side shield outside radius (i.e., 21 feet). Based on experience to date at NRDS, a reasonable criterion of 25 mr/hr 14 days after shutdown was chosen for the facility contribution to the dose rate on the test stand within the radius of the side shield, with the side shield in its lowered position and the duct cover shield in place. This allows for additional contributions to total shutdown radiation levels from the NGTM aft bulkhead, service arm, and associated lines and valves that are used for several tests.

An uncertainty factor of two should be applied to all radiation heating rates reported herein, since these values are nominal computer results. This factor is believed adequate to cover the uncertainties in bulk heating rates; however, an uncertainty factor of three should be applied when working with spatially-dependent, pointwise heat-generation rates. That is, a factor of two should be applied in determining total temperature rises in each shield, but a factor of three should be applied for such calculations as would affect surface boiling of water or thermal stress in structural components.

The preceding factors are based on evaluation of a neutron and gamma energy balance for the entire system. This adequacy is further assured by the following factors: (a) unshielded engine configuration, (b) minimum reflector thickness and (c) past experience in experimental verification of similar calculations in the NERVA development engine program.

The E/STS 2-3 shielding consists of a side shield, facility intermediate shield, bottom shield, duct cover shield, and shielding over the underground control center and access tunnels from the common control center to E/STS 2 and E/STS 3. Each of these shields is discussed briefly below. In addition to these facility shields, there is a NERVA exhaust-duct plenum shield which is a part of the exhaust duct design.

A. SIDE SHIELD

1. Shield Configuration

The side shield is a cylindrical, water-annulus type shield which is built in two portions. The upper portion is stationary. As shown in Figure 25, it is made up of the 9-ft 11-in.-high, 57-in.-thick support ring shield tank, which acts as a combined shield and support structure for the NGTM and engine, and an 18-ft-high, 30-in.-thick annular tank, which rests on the top of the support ring shield. Both parts of this stationary upper portion of the side shield are borated water tanks, containing 1% by weight of natural boron. The tanks and structural stiffeners are nuclear-grade aluminum alloy. Nuclear grade aluminum is available in such alloys as 2219 and 6061, or their equivalents, which have a minimum of cobalt and zinc concentration, consistent with cost constraints. Table 20 gives the elemental composition of Nuclear Grade Aluminum Alloys as specified by procurement practices at the Oak Ridge National Laboratory.

Figure 25 - E/STS 2-3 Side Shield Nuclear Design Requirements,
Upper Portion

TABLE 20

ELEMENTAL CONCENTRATIONS IN NUCLEAR PURITY
ALUMINUM ALLOYS

Element	Weight Fraction	
	Maximum Allowable 1100NP, 6061NP, and 2219NP	6061 Com'l. Max. For Comparison
Cd	3.0(-5)	
Cr		3.5(-3)
Ni	2.0(-4)	5.0(-4)
Co	1.0(-5)	5.0(-4)
Zn	4.0(-4)	2.5(-3)
Fe	4.0(-3)	7.0(-3)
Ti	4.0(-4)	1.5(-3)
Ga	4.0(-4)	(Total Ni + Co + Others = 1.5(-3))
Cd	3.0(-5)	
Li	8.0(-5)	
B	1.0(-5)	
Other	2.0(-4)	

The lower portion of the side shield is shown in Figure 24. This portion is movable. It is a 6-ft-thick plain (unborated) water tank with nuclear grade aluminum structural elements and is shown in the raised operating position in Figure 26. When in the lowered position, it is recessed into a water moat, which also serves as a flotation device for raising the shield. It is in the lowered position during installation and retrieval of the engine and/or stage from the test stand. It is filled with plain water when it is in the moat during work on the stand following a hot firing. The inside radius of the lower portion of the side shield is 15 ft 3 in.

The mating plane between the movable lower portion and stationary upper portion of the side shield is 3 ft 3 in. below the engine/stage interface station "O". The lower unborated water portion is 30 ft 6 in. above the pad and 38 ft 2 in. in overall height. The borated water upper portion extends from 30 ft 6 in. to 58 ft 5 in. above the pad level. The top of the side shield occurs at an angle of 24° -46' from the engine/stage vertical centerline. This point is 37 ft 11 in. above the horizontal centerline of the active core.

2. Shield Heating

Table 21 gives the breakdown of bulk radiation heating in the various segments of the side shield. Table 22 gives the same information for the condition where the facility intermediate shield is included in the analysis. This shield is described in the next section of this report. Inclusion of the facility intermediate shield results in less heating in the side shield.

Figure 27 shows the radial distribution of radiation heating in the plain water side shield, opposite the core center. Figure 28 gives the axial distribution of the radiation heating as a function of elevation on the surface of the side shield lower (movable) portion. Figure 29 gives the heating rate in aluminum as a function of depth in the plain water side shield.

Figure 26 - E/STS 2-3 Side Shield Nuclear Design Requirements, Lower Portion

TABLE 21

NUCLEAR RADIATION HEATING IN SIDE SHIELD
WITHOUT FACILITY INTERMEDIATE SHIELD

SHIELD REGION	NUCLEAR HEAT GENERATION RATE, Btu/sec			
	NEUTRON CAPTURE HEATING	NEUTRON KINETIC HEATING	ENGINE GAMMA HEATING	TOTAL
Movable Portion	4.42×10^4	4.02×10^3	2.15×10^4	6.97×10^4
Module Support Ring	3.75×10^3	3.39×10^2	2.39×10^3	6.48×10^3
Upper Portion	1.40×10^3	1.18×10^2	8.89×10^2	2.41×10^3
Total Side Shield	4.93×10^4	4.48×10^3	2.48×10^4	7.86×10^4

TABLE 22

NUCLEAR RADIATION HEATING IN SIDE SHIELD
WITH FACILITY INTERMEDIATE SHIELD

SHIELD REGION	NUCLEAR HEAT GENERATION RATE, Btu/sec			
	NEUTRON CAPTURE HEATING	NEUTRON KINETIC HEATING	ENGINE GAMMA HEATING	TOTAL
Moveable Portion	3.69×10^4	3.56×10^3	1.69×10^4	5.74×10^4
Module Support Ring	3.42×10^3	3.11×10^2	1.95×10^3	5.68×10^3
Upper Portion	1.40×10^3	1.18×10^2	8.89×10^2	2.41×10^3
Total Side Shield	4.17×10^4	3.99×10^3	1.97×10^4	6.55×10^4

Figure 27 - Radial Distribution of Radiation Heating in Plain Water of Side Shield at Reactor Midplane

Figure 28 - Axial Distribution of Radiation Heating in Plain Water at Inner Surface of Movable Side Shield

Figure 29 - Radial Distribution of Radiation Heating in Aluminum
as a Function of Depth in Plain Water Side Shield

This figure should not be interpreted as indicating depth in aluminum. Rather, it represents radiation heating per unit volume of aluminum at various depths in the plain water side shield.

Similarly, the radial and axial distributions of radiation heating in the upper (borated water) portion of the side shield are given in Figures 30 and 31.

3. Factors Affecting Shield Design

The side shield design is principally dictated by shutdown-access requirements with a hot engine on the stand. Secondly, the side shield design is affected by shutdown access limitations with the engine removed, and by control of the ground test radiation environment within 125% of the expected flight radiation levels above the facility intermediate shield.

The lower movable portion thickness of 6 ft of plain water is predicated on the heat capacity needed for design of an uncooled, movable-side-shield segment. The lower portion is unborated to reduce costs and simplify operations. Since this shield segment is lowered into the moat, and therefore can be shielded after shutdown, neutron-induced activation is a more manageable problem. Earlier studies^{1,2} were based on a borated water side shield pending assurance from the nuclear subsystem contractor that reactivity perturbations would not be excessive with a plain water side shield.

The stationary upper portion of the side shield is designed to allow only a reasonable minimum solid angle for air scatter over the open-ended side shield with a hot shutdown engine on the stand and an empty NGTM propellant tank. This configuration also minimizes neutron escape during operation, which would result in neutron-induced residual activation of test stand components; therefore, activation of major components such as the support columns is substantially reduced. More than 90% of the anticipated full-power firing duration will be accomplished with either the side shield water or LH_2

Figure 30 - Radial Distribution of Radiation Heating in B.H₂O
Stationary Side Shield at Station 0

Figure 31 - Axial Distribution of Radiation Heating in $B \cdot H_2O$
at Inner Surface of Stationary Side Shield

propellant acting as an interceptor of leakage neutrons. Only during the last few seconds of operation, with a 300 lb/sec LH_2 flow rate, will the test stand configuration essentially be an open-ended shield arrangement. Activation of components that must be physically handled later is minimized by this approach.

Since the upper portion of the side shield is a fixed stationary part of the test stand, it is desirable to minimize activation of the tankage involved. Hence, borated water is used to suppress this activation by roughly a factor of ten below what use of plain water would entail.

An additional consideration, although not a controlling one, is the fact that selection of a cylindrical, annular side shield provides a means of controlling ground test perturbations of the flight environment by the incorporation of a facility intermediate shield of the type described herein. It should be noted in this connection, that a complete borated water side shield would be required if this consideration dictated the major design constraints.

As already indicated shutdown shield requirements and available heat load without cooling were the governing constraints, combined with a desire to eliminate boron wherever possible to minimize cost and operational complexity.

B. FACILITY INTERMEDIATE SHIELD

1. Shield Configuration

As shown in Figures 32 and 33, the facility intermediate shield is a 22-in.-high cylindrical annulus-type shield with a 7 ft 3 in. inside radius and a 15 ft 3 in. outside radius. The bottom of the intermediate shield is 3 ft 8 in. above core center. This is based on the reference engine design with the internal engine shield below the core support plate. Provisions

(Figure 31 shows Detailed Cross-Sectional View)

Figure 32 - E/STS 2-3 Engine Preliminary Design Facility
Intermediate Shield Cutaway View

- Notes: 1. B·H₂O is 1% by weight natural boron
 2. B·Al is 0.35% by weight natural boron

Figure 33 - Facility Intermediate Shield Cross Sectional View

should be made for variations of up to 4 ft 6 in. in the elevation of this shield, to allow accommodation of varying-length engines and engines with various internal shields.

The shield has a laminar construction. Nuclear-grade structural aluminum is used throughout as both tankage and stiffening members, some of which are not shown in either Figure 32 or Figure 33. Borated water is the combined neutron shield material and coolant, containing 1% by weight natural boron. Gamma radiation shielding is provided by the borated aluminum. This material has the same atom density as the borated water, which results in a 0.3% by weight boron concentration in the aluminum.

2. Shield Heating

Figure 34 shows the location of point-by-point heat generation calculations in the facility intermediate shield. Table 23 gives the radiation heating rates at each of these locations. The total radiation heating is broken down into the contributing radiation type, including primary and secondary sources of radiation and neutron kinetic heating. Table 24 gives the bulk radiation-heating rates by region in the shield. The data in both tables are based on operation with no engine intermediate shield relative to heating in the water annulus and inner aluminum plate.

3. Factors Affecting Shield Design

The facility intermediate shield is designed to limit contributions due to ground testing to no more than 25% of the predicted flight environment. An engine intermediate shield is assumed to be included to augment the facility intermediate shield in order to preclude exceeding this value, since it affects the engine envelope above the shield internal to the pressure vessel. To obtain data relative to propellant tank bottom heating and radiation levels, operation without an engine intermediate shield is contemplated. The facility intermediate shield is expected to be compatible

Figure 34 - E/STS 2-3 Facility Intermediate Shield Heating Rate Locations

TABLE 23

LOCAL RADIATION HEATING RATES IN E/STS 2-3 FACILITY INTERMEDIATE SHIELD

Locations	Material	Radius in	Distance Above Shield Bottom in	Engine Leakage Gamma	Int. Shield Captures	Side Shield Captures	Bottom Shield Captures	Side Shield Scatter	Bottom Shield & Duct Jacket Scatter	Neutron Kinetic	Total Btu/in ³ -sec
A1	A1	87	0.0	3.48(-2)*	3.50(-2)	1.75(-3)	2.41(-4)	3.86(-4)	2.88(-5)	9.31(-4)	7.31(-2)
A2	A1	91	0.0	3.28(-2)	3.27(-2)	1.77(-2)	2.41(-4)	3.90(-4)	2.86(-5)	8.43(-4)	6.88(-2)
A3	A1	94	0.0	3.13(-2)	3.04(-2)	1.78(-3)	2.40(-4)	3.93(-4)	2.84(-5)	7.87(-4)	6.49(-2)
A4	A1	99	0.0	2.91(-2)	2.76(-2)	1.80(-3)	2.39(-4)	3.99(-4)	2.80(-5)	7.09(-4)	5.99(-2)
A5	A1	111	0.0	2.44(-2)	2.26(-2)	1.85(-3)	2.35(-4)	4.12(-4)	2.71(-5)	5.80(-4)	5.01(-2)
A6	A1	135	0.0	1.76(-2)	1.56(-2)	1.97(-3)	2.26(-4)	4.54(-4)	2.53(-5)	3.92(-4)	3.63(-2)
A7	A1	159	0.0	1.31(-2)	1.10(-2)	2.12(-3)	2.16(-4)	5.31(-4)	2.37(-5)	2.71(-4)	2.73(-2)
A8	A1	183	0.0	1.01(-2)	7.81(-3)	1.65(-3)	2.05(-4)	4.90(-4)	2.23(-5)	1.94(-4)	2.05(-2)
B1	A1	87	1.0	3.43(-2)	3.50(-2)	1.17(-3)	2.02(-4)	3.70(-4)	2.77(-5)	9.17(-4)	7.20(-2)
B2	B.H ₂ O	88	1.0	1.07(-2)	2.58(-1)	4.98(-4)	8.58(-5)	1.58(-4)	1.18(-5)	8.77(-3)	2.78(-1)
B2	A1	88	1.0	2.58(-2)	3.52(-2)	1.17(-3)	2.01(-4)	3.71(-4)	2.76(-5)	7.38(-4)	6.35(-2)
B3	B.H ₂ O	91	1.0	7.39(-3)	2.46(-1)	5.03(-4)	8.57(-5)	1.59(-4)	1.18(-5)	5.17(-3)	2.59(-1)
B3	A1	91	1.0	1.79(-2)	3.35(-2)	1.19(-3)	2.01(-4)	3.74(-4)	2.75(-5)	4.38(-4)	5.36(-2)
B4	B.H ₂ O	94	1.0	6.94(-3)	2.32(-1)	5.07(-4)	8.56(-5)	1.60(-4)	1.17(-5)	4.62(-3)	2.44(-1)
B4	A1	94	1.0	1.68(-2)	3.16(-2)	1.19(-3)	2.01(-4)	3.77(-4)	2.73(-5)	3.92(-4)	5.06(-2)
B5	B.H ₂ O	99	1.0	6.29(-3)	2.11(-1)	5.15(-4)	8.54(-5)	1.62(-4)	1.15(-5)	3.96(-3)	2.20(-1)
B5	A1	99	1.0	1.53(-2)	2.87(-2)	1.21(-3)	2.00(-4)	3.82(-4)	2.69(-5)	3.36(-4)	4.62(-2)
B6	B.H ₂ O	111	1.0	5.00(-3)	1.72(-1)	5.28(-4)	8.37(-5)	1.68(-4)	1.11(-5)	2.85(-3)	1.81(-1)
B6	A1	111	1.0	1.21(-2)	2.35(-2)	1.24(-3)	1.96(-4)	3.95(-4)	2.60(-5)	2.42(-4)	3.77(-2)
B7	B.H ₂ O	135	1.0	3.21(-3)	1.19(-1)	5.62(-4)	8.07(-5)	1.85(-4)	1.04(-5)	1.63(-3)	1.23(-1)
B7	A1	135	1.0	7.81(-3)	1.62(-2)	1.32(-3)	1.89(-4)	4.35(-4)	2.43(-5)	1.39(-4)	2.60(-2)
B8	B.H ₂ O	159	1.0	2.13(-3)	8.37(-2)	6.05(-4)	7.73(-5)	2.16(-4)	9.76(-6)	1.01(-3)	8.77(-2)

(-2) = x 10⁻²

TABLE 23 (cont.)

LOCAL RADIATION HEATING RATES IN E/STS 2-3 FACILITY INTERMEDIATE SHIELD												
Heating Rate Locations	Material	Radius in	Distance Above Shield Bottom in	Engine Leakage Gamma	Int. Shield Captures	Side Shield Captures	Bottom Shield Captures	Side Shield Scatter	Bottom Shield & Duct Jacket Scatter	Neutron Kinetic	Total Btu/in ³ -sec	
75B	B8	A1	159	1.0	5.18(-3)	1.14(-2)	1.42(-3)	1.81(-4)	5.09(-4)	2.28(-5)	8.63(-5)	1.88(-2)
	B9	B.H ₂ O	1.83	1.0	1.45(-3)	5.96(-2)	4.73(-4)	7.30(-5)	2.00(-4)	9.16(-6)	6.67(-4)	6.25(-2)
	B9	A1	1.83	1.0	3.53(-3)	8.12(-3)	1.11(-3)	1.71(-4)	4.70(-4)	2.14(-5)	5.69(-5)	1.35(-2)
	C1	A1	87	4.0	3.25(-2)	3.50(-2)	4.53(-4)	1.51(-4)	3.56(-4)	2.72(-5)	9.37(-4)	6.94(-2)
	C2	B.H ₂ O	88	4.0	1.01(-2)	2.58(-1)	3.10(-4)	6.42(-5)	1.52(-4)	1.16(-5)	8.40(-3)	2.77(-1)
	C2	A1	88	4.0	2.44(-2)	3.52(-2)	4.53(-4)	1.50(-4)	3.57(-4)	2.71(-5)	7.10(-4)	6.13(-2)
	C3	B.H ₂ O	91	4.0	6.92(-3)	4.54(-2)	3.13(-4)	6.41(-5)	1.53(-4)	1.16(-5)	2.92(-3)	5.58(-2)
	C4	B.H ₂ O	94	4.0	4.90(-3)	2.14(-2)	3.16(-4)	6.41(-5)	1.54(-4)	1.15(-5)	1.10(-3)	2.80(-2)
	C5	B.H ₂ O	99	4.0	3.43(-3)	1.94(-2)	3.21(-4)	6.39(-5)	1.56(-4)	1.13(-5)	4.35(-4)	2.38(-2)
	C6	B.H ₂ O	111	4.0	2.41(-3)	1.59(-2)	3.29(-4)	6.26(-5)	1.62(-4)	1.09(-5)	2.59(-4)	1.91(-2)
	C7	B.H ₂ O	135	4.0	1.41(-3)	1.10(-2)	3.50(-4)	6.04(-5)	1.78(-4)	1.02(-5)	1.39(-4)	1.31(-2)
	C8	B.H ₂ O	159	4.0	8.50(-4)	7.72(-3)	3.77(-4)	5.78(-5)	2.08(-4)	9.57(-6)	6.59(-5)	9.29(-3)
	C9	B.H ₂ O	183	4.0	5.26(-4)	5.50(-3)	2.95(-4)	5.46(-5)	1.92(-4)	8.98(-6)	3.44(-5)	6.61(-3)
	D1	A1	87	7.0	3.07(-2)	3.50(-2)	2.75(-4)	1.08(-4)	3.49(-4)	3.94(-5)	9.39(-4)	6.74(-2)
	D2	A1	88	7.0	2.30(-2)	3.52(-2)	2.75(-4)	1.07(-4)	3.50(-4)	3.93(-5)	7.11(-4)	5.97(-2)
	D2	B.H ₂ O	88	7.0	9.53(-3)	2.58(-1)	1.93(-4)	4.57(-5)	1.49(-4)	1.68(-5)	8.42(-3)	2.76(-1)
	D3	B.H ₂ O	91	7.0	6.52(-3)	2.14(-2)	1.95(-4)	4.57(-5)	1.50(-4)	1.68(-5)	2.88(-3)	3.12(-2)
	D4	B.A1	94	7.0	1.09(-2)	3.30(-3)	4.63(-4)	1.07(-4)	3.51(-4)	3.90(-5)	7.54(-5)	1.52(-2)
	D4	B.H ₂ O	94	7.0	4.49(-3)	3.28(-3)	1.97(-4)	4.56(-5)	1.50(-5)	1.66(-5)	9.88(-4)	9.58(-3)
	D5	B.A1	99	7.0	6.40(-3)	1.52(-3)	4.70(-4)	1.07(-4)	3.56(-4)	3.86(-5)	1.63(-5)	8.91(-3)
	D5	B.H ₂ O	99	7.0	2.61(-3)	1.51(-3)	2.00(-4)	4.55(-5)	1.52(-4)	1.64(-5)	2.11(-4)	4.74(-3)
	D6	B.A1	111	7.0	3.31(-3)	1.22(-3)	4.82(-4)	1.05(-4)	3.70(-4)	3.72(-5)	3.66(-6)	5.53(-3)
	D6	B.H ₂ O	111	7.0	1.34(-3)	1.22(-3)	2.05(-4)	4.46(-5)	1.58(-4)	1.58(-5)	4.69(-5)	3.03(-3)
	D7	B.A1	135	7.0	1.76(-3)	8.45(-4)	5.12(-4)	1.01(-4)	4.07(-4)	3.48(-5)	1.51(-6)	3.66(-3)
	D7	B.H ₂ O	135	7.0	7.14(-4)	8.40(-4)	2.18(-4)	4.30(-5)	1.74(-4)	1.48(-5)	1.93(-5)	2.02(-3)
	D8	B.A1	159	7.0	9.90(-4)	5.95(-4)	5.52(-4)	9.68(-5)	4.75(-4)	3.27(-5)	6.34(-7)	2.74(-3)

TABLE 23 (cont.)

LOCAL RADIATION HEATING RATES IN E/STS 2-3 FACILITY INTERMEDIATE SHIELD

Distance Above Shield Bottom in	Engine Leakage Gamma	Int. Shield Captures	Side Shield Captures	Bottom Shield Captures	Side Shield Scatter	Bottom Shield & Duct Jacket Scatter	Neutron Kinetic	Total Btu/in ³ -sec
7.0	4.00(-4)	5.91(-4)	2.35(-4)	4.12(-5)	2.03(-4)	1.39(-5)	8.09(-6)	1.49(-3)
7.0	5.71(-4)	4.24(-4)	4.32(-4)	9.14(-5)	4.40(-4)	3.06(-5)	2.72(-7)	1.99(-3)
7.0	2.30(-4)	4.21(-4)	1.84(-4)	3.89(-5)	1.88(-4)	1.30(-5)	3.42(-6)	1.08(-3)
8.0	1.07(-2)	2.41(-3)	3.16(-4)	8.03(-5)	3.04(-4)	2.57(-5)	7.46(-5)	1.39(-2)
8.0	4.39(-3)	2.32(-3)	1.34(-4)	3.42(-5)	1.30(-4)	1.09(-5)	9.80(-4)	8.00(-3)
8.0	4.44(-3)	9.41(-4)	3.21(-4)	8.00(-5)	3.08(-4)	2.54(-5)	1.87(-5)	6.13(-3)
8.0	2.11(-3)	5.68(-4)	3.30(-4)	7.88(-5)	3.20(-4)	2.45(-5)	2.68(-6)	3.43(-3)
8.0	1.04(-3)	3.92(-4)	3.50(-4)	7.58(-5)	3.52(-4)	2.29(-5)	1.01(-6)	2.23(-3)
8.0	5.49(-4)	2.76(-4)	3.77(-4)	7.26(-5)	4.11(-4)	2.15(-5)	4.07(-7)	1.71(-3)
8.0	2.98(-4)	1.97(-4)	2.95(-4)	6.76(-5)	3.80(-4)	2.01(-5)	1.65(-7)	1.26(-3)
9.0	1.05(-2)	2.70(-3)	2.15(-4)	5.93(-5)	2.45(-4)	2.40(-5)	7.41(-5)	1.38(-2)
9.0	4.30(-3)	2.63(-3)	9.13(-5)	2.52(-5)	1.04(-4)	1.02(-5)	9.73(-4)	8.13(-3)
9.0	3.25(-3)	1.30(-3)	2.18(-4)	5.93(-5)	2.48(-4)	2.37(-5)	2.09(-5)	5.12(-3)
9.0	1.32(-3)	1.12(-3)	9.27(-5)	2.52(-5)	1.06(-4)	1.01(-5)	2.72(-4)	2.95(-3)
9.0	1.38(-3)	7.80(-4)	2.24(-4)	5.81(-5)	2.58(-4)	2.29(-5)	2.07(-6)	2.72(-3)
9.0	5.55(-4)	7.36(-4)	9.50(-5)	2.47(-5)	1.10(-4)	9.75(-6)	2.65(-5)	1.56(-3)
9.0	6.32(-4)	5.39(-4)	2.38(-4)	5.60(-5)	2.84(-4)	2.14(-5)	6.82(-7)	1.77(-3)
9.0	2.53(-4)	5.08(-4)	1.01(-4)	2.38(-5)	1.21(-4)	9.11(-6)	8.70(-6)	1.02(-3)
9.0	3.17(-4)	3.79(-4)	2.57(-4)	5.36(-5)	3.32(-4)	2.01(-5)	2.65(-7)	1.36(-3)
9.0	1.26(-4)	3.57(-4)	1.09(-4)	2.28(-5)	1.41(-4)	8.56(-6)	3.37(-6)	7.68(-3)
9.0	1.62(-4)	2.70(-4)	2.01(-4)	5.06(-5)	3.07(-4)	1.88(-5)	1.03(-7)	1.01(-3)
9.0	6.46(-5)	2.54(-4)	8.52(-5)	2.15(-5)	1.31(-4)	8.00(-6)	1.32(-6)	5.66(-4)
11	1.00(-2)	2.09(-3)	1.57(-4)	4.53(-5)	1.52(-4)	1.95(-5)	7.27(-5)	1.25(-2)
11	4.12(-3)	2.06(-3)	6.66(-5)	1.94(-5)	6.44(-5)	8.26(-6)	9.55(-4)	7.29(-3)
11	4.67(-3)	6.49(-4)	1.58(-4)	4.52(-5)	1.54(-5)	1.91(-5)	1.41(-5)	5.57(-3)
11	1.90(-3)	5.99(-4)	6.70(-5)	1.93(-5)	6.52(-5)	8.12(-6)	1.83(-4)	2.84(-3)

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TABLE 23 (cont.)

LOCAL RADIATION HEATING RATES IN E/STS 2-3 FACILITY INTERMEDIATE SHIELD

Heating Rate Locations	Material	Radius in	Distance Above Shield Bottom	Engine Leakage Gamma	Int. Shield Captures	Side Shield Captures	Bottom Shield Captures	Side Shield Scatter	Bottom Shield & Duct Jacket Scatter	Neutron Kinetic	Total
			in								Btu/in ³ -sec
G3	B.A1	111	11	1.11(-3)	3.29(-4)	1.62(-4)	4.44(-5)	1.60(-4)	1.84(-5)	1.06(-6)	1.82(-3)
G3	B.H ₂ O	111	11	4.45(-4)	3.11(-4)	6.90(-5)	1.89(-5)	6.76(-5)	7.83(-6)	1.36(-5)	9.33(-3)
G4	B.A1	135	11	4.51(-4)	2.27(-4)	1.73(-4)	4.28(-5)	1.76(-4)	1.73(-5)	2.40(-7)	1.09(-3)
G4	B.H ₂ O	135	11	1.80(-4)	2.15(-4)	7.33(-5)	1.82(-5)	7.45(-5)	7.34(-6)	3.05(-6)	5.71(-4)
G5	B.A1	159	11	2.20(-4)	1.60(-4)	1.86(-4)	4.08(-5)	2.06(-4)	1.62(-5)	8.36(-8)	8.29(-4)
G5	B.H ₂ O	159	11	8.73(-5)	1.51(-4)	7.91(-5)	1.74(-5)	8.70(-5)	6.89(-6)	1.06(-6)	4.30(-4)
G6	B.A1	183	11	1.09(-4)	1.14(-4)	1.45(-4)	3.87(-5)	1.90(-4)	1.52(-5)	2.90(-8)	6.12(-4)
G6	B.H ₂ O	183	11	4.33(-5)	1.08(-4)	6.14(-5)	1.05(-5)	8.05(-5)	6.46(-6)	3.66(-7)	3.16(-4)
H1	A1	87	13	2.73(-2)	3.50(-2)	7.19(-5)	2.39(-5)	4.22(-5)	8.52(-6)	9.58(-4)	6.34(-2)
H2	A1	88	13	2.04(-2)	3.52(-2)	7.23(-5)	2.39(-5)	4.23(-5)	8.52(-6)	7.37(-4)	5.65(-2)
H2	B.H ₂ O	88	13	8.43(-3)	2.58(-1)	3.05(-5)	1.01(-5)	1.80(-5)	3.64(-6)	8.73(-3)	2.67(-1)
H3	B.H ₂ O	91	13	5.75(-3)	4.54(-2)	3.07(-5)	1.01(-5)	1.81(-5)	3.64(-6)	2.79(-3)	5.40(-2)
H4	B.A1	94	13	9.63(-3)	1.75(-3)	7.36(-5)	2.38(-5)	4.29(-5)	8.53(-6)	7.15(-5)	1.16(-2)
H4	B.H ₂ O	94	13	3.95(-3)	1.73(-3)	3.11(-5)	1.01(-5)	1.82(-5)	3.61(-6)	9.38(-4)	6.68(-3)
H5	B.A1	99	13	2.94(-3)	3.40(-4)	7.40(-5)	2.37(-5)	4.35(-5)	8.36(-6)	1.93(-5)	3.45(-3)
H6	B.A1	111	13	5.51(-4)	7.65(-5)	7.60(-5)	2.33(-5)	4.52(-5)	8.05(-6)	9.51(-7)	7.81(-4)
H7	B.A1	135	13	1.90(-4)	5.28(-5)	8.09(-5)	2.24(-5)	4.97(-5)	7.57(-6)	1.20(-7)	4.04(-4)
H8	B.A1	159	13	8.40(-5)	3.71(-5)	8.71(-5)	2.14(-5)	5.82(-5)	7.09(-6)	3.93(-8)	2.95(-4)
H9	B.A1	183	13	3.80(-5)	2.64(-5)	6.78(-5)	2.03(-5)	5.36(-5)	6.65(-6)	1.27(-8)	2.13(-4)
I1	B.A1	94	16	9.00(-3)	1.81(-3)	2.38(-5)	8.83(-6)	5.71(-6)	1.59(-6)	6.86(-5)	1.09(-2)
I1	B.H ₂ O	94	16	3.69(-3)	1.79(-3)	1.02(-5)	3.73(-6)	2.41(-6)	6.66(-7)	9.00(-4)	6.40(-3)
I2	B.A1	99	16	2.24(-3)	3.96(-4)	2.45(-5)	8.80(-6)	5.79(-6)	1.56(-6)	1.92(-5)	2.70(-3)
I2	B.H ₂ O	99	16	9.04(-4)	3.61(-4)	1.03(-5)	3.71(-6)	2.44(-6)	6.54(-7)	2.30(-4)	1.54(-3)
I3	B.A1	111	16	2.60(-4)	1.22(-4)	2.42(-5)	8.65(-6)	6.02(-6)	1.50(-6)	9.20(-7)	4.24(-4)
I3	B.H ₂ O	111	16	1.03(-4)	1.17(-4)	1.06(-5)	3.64(-6)	2.54(-6)	6.29(-7)	1.17(-5)	2.49(-4)

75D

TABLE 23 (cont.)

LOCAL RADIATION HEATING RATES IN E/STS 2-3 FACILITY INTERMEDIATE SHIELD

Heating Rate Locations	Material	Radius in	Distance Above Shield	Engine Leakage Gamma	Int. Shield Captures	Side Shield Captures	Bottom Shield Captures	Side Shield Scatter	Bottom Shield & Duct Jacket	Neutron Kinetic	Total Btu/in ³ -sec
			Bottom in						Scatter		
I4	B.A1	135	16	5.88(-5)	8.45(-5)	2.68(-5)	8.32(-6)	6.62(-6)	1.41(-6)	4.65(-8)	1.86(-4)
I4	B.H ₂ O	135	16	2.31(-5)	8.04(-5)	1.13(-5)	3.50(-6)	2.79(-6)	5.91(-7)	5.75(-7)	1.22(-4)
I4	B.H ₂ O	135	16	2.31(-5)	8.04(-5)	1.13(-5)	3.50(-6)	2.79(-6)	5.91(-7)	5.75(-7)	1.22(-4)
I5	B.A1	159	16	2.29(-5)	5.95(-5)	2.88(-5)	7.58(-6)	7.75(-6)	1.32(-6)	1.36(-8)	1.28(-4)
I5	B.H ₂ O	159	16	8.95(-6)	5.66(-5)	1.21(-5)	3.35(-6)	3.27(-6)	5.53(-7)	1.71(-7)	8.50(-5)
I6	B.A1	183	16	9.17(-6)	4.24(-5)	2.24(-5)	7.54(-6)	7.14(-6)	1.24(-6)	4.00(-7)	8.99(-5)
I6	B.H ₂ O	183	16	3.56(-6)	4.03(-5)	9.42(-6)	3.18(-6)	3.01(-6)	5.20(-7)	5.01(-8)	6.00(-5)
J1	A1	87	18	2.46(-2)	3.50(-2)	1.79(-5)	6.72(-6)	3.20(-6)	9.53(-7)	9.47(-4)	6.06(-2)
J2	A1	88	18	1.82(-2)	3.52(-2)	1.80(-5)	6.72(-6)	3.22(-6)	9.50(-7)	7.15(-4)	5.41(-2)
J2	B.H ₂ O	88	18	7.54(-3)	2.58(-1)	7.48(-6)	2.84(-6)	1.35(-6)	4.00(-7)	8.47(-3)	2.74(-1)
J3	A1	91	18	1.25(-2)	9.58(-2)	1.81(-5)	6.70(-6)	3.24(-6)	9.48(-7)	2.00(-4)	1.08(-1)
J3	B.H ₂ O	91	18	5.14(-3)	2.14(-2)	7.51(-6)	2.83(-6)	1.36(-6)	3.99(-7)	2.65(-3)	2.92(-2)
J4	A1	94	18	8.58(-3)	1.72(-3)	1.83(-5)	6.70(-6)	3.26(-6)	9.40(-7)	6.63(-5)	1.04(-2)
J4	B.H ₂ O	94	18	3.52(-3)	1.64(-3)	7.61(-6)	2.83(-6)	1.37(-6)	3.96(-7)	8.70(-4)	6.04(-3)
J5	A1	99	18	3.66(-3)	3.07(-4)	1.84(-5)	6.67(-6)	3.28(-6)	9.29(-7)	1.30(-5)	4.01(-3)
J5	B.H ₂ O	99	18	1.49(-3)	2.20(-4)	7.66(-6)	2.82(-6)	1.38(-6)	3.89(-7)	1.69(-4)	1.89(-3)
J6	A1	111	18	2.84(-4)	5.03(-5)	1.89(-5)	6.56(-6)	3.40(-6)	8.93(-7)	7.26(-7)	3.65(-4)
J6	B.H ₂ O	111	18	1.12(-4)	1.80(-6)	7.89(-6)	2.76(-6)	1.43(-6)	3.76(-7)	9.25(-6)	1.36(-4)
J7	A1	135	18	4.70(-5)	3.47(-5)	2.00(-5)	6.31(-6)	3.76(-6)	8.36(-7)	1.99(-8)	1.13(-4)
J7	B.H ₂ O	135	18	1.84(-5)	1.24(-6)	8.38(-6)	2.66(-6)	1.58(-6)	3.52(-7)	2.40(-7)	3.29(-5)
J8	A1	159	18	178(-5)	2.44(-5)	2.16(-5)	6.03(-6)	4.38(-6)	7.84(-7)	5.19(-9)	7.50(-5)
J8	B.H ₂ O	159	18	6.95(-6)	8.73(-7)	9.04(-6)	2.54(-6)	1.84(-6)	3.30(-7)	6.49(-8)	2.16(-5)
J9	A1	183	18	7.02(-6)	1.74(-5)	1.68(-5)	5.72(-6)	4.07(-6)	7.36(-7)	1.38(-9)	5.17(-5)
J9	B.H ₂ O	183	18	2.72(-6)	6.22(-7)	7.02(-6)	2.41(-6)	1.71(-6)	3.10(-7)	1.72(-8)	1.48(-5)

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TABLE 24

RADIATION HEATING RATES BY REGION IN E/STS 2-3
FACILITY INTERMEDIATE SHIELD

<u>Shield Region</u>	<u>Radiation Heating Rates, Btu/sec per region @ 5,000 Mw</u>
$\frac{1}{2}$ " Aluminum Bottom Plate	1.3×10^3
1" Borated Water	1.12×10^4
$\frac{1}{2}$ " Borated Aluminum	2.4×10^3
5" Borated Water	8.8×10^3
1" Aluminum Support Plate	2.1×10^2
2" Borated Aluminum	3.5×10^2
2" Borated Water	2×10^2
5" Borated Aluminum	$2 \times 10^{2*}$
2" Borated Water	$3 \times 10^{1*}$
$\frac{1}{2}$ " Borated Aluminum	$1 \times 10^{1*}$
$1\frac{1}{2}$ " Borated Water	$2 \times 10^{1*}$
1" Aluminum Top Plate	$3 \times 10^{1*}$
Sub Total	2.5×10^4
Inner Annulus	
$1\frac{1}{2}$ " Aluminum	7.3×10^2
$3/4$ " Borated Water	4.0×10^2
$1/2$ " Borated Aluminum	5.5×10^2
$5\frac{1}{4}$ " Borated Water	3.0×10^3
Sub Total	4.7×10^3
Grand Total	3.0×10^4

*These data are based on heating with an engine intermediate shield present and with a relatively thick internal engine shield. The shield and coolant flow passages are designed to accommodate operation with no internal engine or engine intermediate shields.

with relatively thick engine internal shielding requirements (e.g. $\sim 85 \text{ gm/cm}^2$). Analyses pertaining to the 25% requirement were performed with a nuclear opaque engine intermediate shield. This shield mates the internal engine shield with the facility intermediate shield.

The facility intermediate shield also serves as a shutdown shield when it is in its lowered position over the bottom shield, as shown in Figure 35. This is one method to control the radiation level on the test stand with the hot engine removed.

Selection of the specific laminated configuration resulted from an attempt to design an optimum shield for both operating and shutdown constraints. The choice of materials, particularly borated aluminum, is based on having a reasonably thin shield to mate with the weight-constrained engine and engine intermediate shields. Borated water is believed necessary to suppress reactivity perturbations in the reactor. It is also the most efficient way to remove neutrons and heat simultaneously during operation while reducing residual activation levels. Even with borated water and boron in the aluminum shield segments, residual activation and aluminum capture gamma radiation levels are significant as indicated in Section VII.

Elimination of boron in either the water or the aluminum of this shield is more difficult to achieve than anywhere else in the system. Having introduced an intermediate shield to the shield enclosure, control of capture and activation reactions becomes a prime consideration. No simple method of resolving this problem is currently available. However, continued analysis in this area may be warranted.

Figure 35 - Bottom Shield and Duct Cover Shield Configurations

C. BOTTOM SHIELD

1. Shield Configuration

Figure 35 shows the location and relative dimensions of the 5-ft-thick bottom shield. It is a 1% borated water cylindrical annulus enclosed in nuclear-grade structural aluminum and mates with the duct plenum shield on the inside at an inner radius of approximately 7 ft 2 in. The outside of the bottom shield is at a radius of roughly 15 ft 3 in. and mates with the movable side shield when it is in its lowered position in the moat.

2. Shield Heating

Figures 36 and 37 give the relative radial and axial radiation-heating rate distributions, respectively, in the bottom shield. The total bulk heating rate in the shield is 2.1×10^4 Btu/sec. The principal mechanism of heating is the energy deposition associated with (n, α) absorptions in the boron of the borated water.

3. Factors Affecting Shield Design

The principal objectives of the bottom shield are to reduce radiation heating and residual concrete activation in the duct vault. The thickness is governed by the concrete heating design limitation of approximately 1×10^{-4} Btu/in.³-sec. The use of water as a gamma shield material results in a shield which is considerably thicker than would be required for purely neutron-shielding reasons. Therefore, neutron kinetic heating, secondary gamma heating, and concrete activation levels are all quite low in the duct vault concrete, which is shielded by the bottom shield. Five feet of borated water is required to attenuate primary gamma radiation and bottom shield secondary gamma radiation sources.

Figure 36 - Radial Distribution of Heating Rates in Bottom Shield

Figure 37 - Axial Distribution of Heating Rates in Bottom Shield

The boron in the water serves three purposes. The first is the suppression of activation in the tankage and structural members. Without the boron, aluminum activation in this shield would increase by about a factor of ten. The second function of the boron is to suppress reactivity perturbations in the reactor. Finally, the use of boron results in a shield which is more efficient in the reduction of concrete heating.

Elimination of the boron would require a thicker shield of, perhaps, 7 ft total thickness as compared to the present 5 ft design. This alternative of eliminating the boron in the bottom shield appears attractive when one considers the fact that the subcontractor for the nuclear subsystems has pointed out that elimination of boron in all the shields would bring about a reactivity perturbation of only 1 cent. Since the criteria provide for a total reactivity perturbation from the facility components of 50 cents, elimination of boron in all the facility shields would amount to only a 2% change in this total. Furthermore, study of Table 32, presented in Section IX.A, shows that increasing the bottom shield activation by a factor of ten nominally would bring this activation contribution up to the level of contributions from other activation sources. As a consequence, increasing the bottom shield thickness to a total thickness which would still satisfy nuclear heating requirements in the duct vault concrete, appears to be an attractive alternative to the use of boron in the water of this shield.

D. DUCT COVER SHIELD

1. Shield Configuration

As shown in Figure 35, the duct cover shield is composed of two segments. The basic shield is a 6-in.-thick cylindrical lead slab 8 ft 9 in. in radius. It rests on a 2-in.-thick cylindrical steel slab which

extends to a radius of 12 ft. On centerline the total shield is, therefore, 2 in. of steel plus 6 in. of lead. Beyond a radius of 8 ft 9 in. only the 2 in. steel slab is required.

The duct cover shield is to be rolled into place over the duct after a hot firing. It should be stored sufficiently far from the reactor to prevent its activation during firings. An expedient to distance during operation, would be to locate some small amount of neutron shielding material over the duct cover shield if it is only a few feet from the side shield during operation.

2. Factors Affecting Shield Design

The basic shielding thicknesses for the duct cover shield are based on reducing the combined contributions from duct activation and fission product contamination of the duct. This results in 10 mrem/hr when a factor of five uncertainty is applied to the 2 mrem/hr nominal value 2 weeks after a firing. It will be remembered that an allowable of 25 mrem/hr was specified for the pad area. Calculations of contamination are based on distribution in the duct of 0.06% of the total fission product inventory from a 30-min, 5000 Mw firing. Residual radioactants in the steel of the duct are based on neutron leakage for the same run duration and power level. Radiation levels following a 45-min, 10,000-Mw firing would be roughly three times these levels. The duct cover thickness could readily be increased to compensate for such an eventuality. On the other hand, the levels of radiation reported in Section VII for this shield configuration may be thought to be sufficiently low so as not to require additional shielding for the 10,000-Mw engine case. The supporting calculations are based on allowing only a 2-in.-clearance between the steel of the duct cover shield and the top surface of the intermediate shield. Increasing this gap would increase radiation streaming and, hence, raise the radiation levels above the pad due to below-grade sources.

E. CONTROL CENTER AND ACCESS TUNNELS

1. Shield Configuration

The Control Center and access tunnel shielding is depicted in Figure 38. The equivalent of 5 ft 6 in. of concrete is deemed appropriate to insure personnel safety in these areas, following a nuclear accident. No distinction is made between occupancy requirements in the control center and the access tunnels. This equivalent concrete thickness can be achieved by a combination of concrete and earth fill based on the rough approximation that both materials are equivalent in shielding effectiveness.

2. Factors Affecting Shield Design

The control center and access tunnel shielding is predicated on an accident condition involving concentration of 10% of the fueled region of the core directly over these locations. The design is based on the requirement that the integrated dose in these regions be limited to 3 rem total exposure. Shielding of the access tunnels is based on use of the tunnels for shelter in the event that air-particulate contamination precludes occupancy of the Control Center.

Figure 38 - Sketch of Control Center and Access Tunnel Shielding

VII. SHIELDED RADIATION ENVIRONMENT DURING OPERATION

The operational radiation environment resulting from normal engine tests with facility shielding in place are described in this section.

A. ENGINE ENVELOPE AND NUCLEAR GROUND TEST MODULE

Figure 39 shows the location of engine envelope and tank bottom detector points selected for evaluation of the operational radiation environment in the engine test compartment and for evaluation of the effect of the facility on the simulated flight environment. Radiation environment analysis at these locations forms the basis for shield design optimization as it is influenced by requirements inside the test compartment.

The objective of the design optimization of the facility intermediate shield was to limit the operational neutron and gamma radiation environments to 25% of that specified as the flight environment^{4,5}. In addition, provisions are currently included in the facility intermediate shield design for testing with segments of the intermediate shield removed. Tables 25, 26, and 27 describe fast-neutron and gamma-radiation levels during engine operation at the engine envelope and tank bottom with and without the facility intermediate shield.

B. RADIATION LEVELS IN ADJACENT VICINITY

Figures 40 and 41 describe the direct and air-scattered contributions to the fast neutron and gamma dose rate to 9000 feet from engine centerline in a direction opposite the service tower during operation at 5000 Mw with a 6-ft water shield in place. Included are comparisons with the unshielded fast neutron and gamma dose rates. Figures 42 and 43 show the air-scattered gamma dose rate from behind the service tower to 9000 feet from the engine centerline in the direction of the service tower during operation at 5000 Mw. Data from Figures 40 through 43 are combined in Figure 44, which describes

Figure 39 - Location of Engine Envelope and Tank Bottom Dose Points

TABLE 25

GAMMA RADIATION LEVELS AT ENGINE ENVELOPE AND TANK BOTTOM
WITH NO FACILITY INTERMEDIATE SHIELD

Dose Point	Radius, ft	Axial Distance from Station 0, ft	GAMMA DOSE RATE, Rad(c)/hr			Criteria Shielded Engine Leakage ^(4,5)	Ratio, Total Facility to Criteria
			Facility Scatter	Side Shield Captures	Tot.Facility Contributions		
AXIS 1	0.0	-8.66	1.85(6)	4.64(7)	4.82(7)	1.3(7)	3.71
AXIS 2	0.0	-6.56	2.72(6)	5.96(7)	6.23(7)	5.1(6)	12.2
AXIS 3	0.0	-3.28	2.96(6)	5.55(7)	5.85(7)	2.0(6)	29.2
AXIS 4	0.0	0.0	2.73(6)	4.74(7)	5.01(7)	1.1(6)	45.5
AXIS 5	0.0	1.64	2.61(6)	4.34(7)	4.60(7)	4.41(6)	10.4
Tank Bot. 1	4.0	2.38	2.62(6)	4.19(7)	4.45(7)	9.79(6)	4.54
Tank Bot. 2	8.0	4.89	2.59(6)	3.55(7)	3.81(7)	9.79(6)	3.89
Tank Bot. 3	12.0	8.89	2.34(6)	2.52(7)	2.75(7)	9.28(6)	2.96
Tank Bot. 4	16.2	16.73	3.60(6)	1.19(7)	1.55(7)	2.61(6)	5.94
Eng.Env. 1	8.04	-8.66	3.84(6)	7.09(7)	7.47(7)		
Eng.Env. 2	8.04	-6.56	3.63(6)	6.59(7)	6.95(7)		
Eng.Env. 3	8.04	-3.67	3.82(6)	5.80(7)	6.18(7)		
Eng.Env. 4	9.68	0.0	3.28(6)	4.79(7)	5.12(7)		

TABLE 26

GAMMA RADIATION LEVELS AT ENGINE ENVELOPE AND TANK BOTTOM
WITH FACILITY INTERMEDIATE SHIELD

Dose Point	Radius, ft	Axial Distance from Station 0, ft	Facility Scatter	GAMMA DOSE RATE, Rad(c)/hr			Criteria Shielded Engine Leakage ^(4,5)	Ratio, Total Facility to Criteria
				Int. Shield Captures	Facility Captures	Totl.Facility Contributions		
AXIS 1	0.0	-8.66	-	-	-	-	1.3 (6)	-
AXIS 2	0.0	-6.56	9.90(-2)	2.30(3)	-	2.30(3)	5.1 (6)	4.50(-4)
AXIS 3	0.0	-3.28	2.94(2)	3.43(4)	1.53(4)	4.99(4)	2.0 (6)	2.50(-2)
AXIS 4	0.0	0.0	3.01(3)	1.18(5)	1.07(5)	2.28(5)	1.1 (6)	2.07(-1)
AXIS 5	0.0	1.64	6.79(3)	1.30(5)	2.45(5)	3.82(5)	4.41(6)	8.66(-2)
Tank Bot.1	4.0	2.38	1.07(4)	1.96(5)	4.44(5)	6.51(5)	9.79(6)	6.65(-2)
Tank Bot.2	8.0	4.89	1.51(4)	2.09(5)	4.70(5)	6.94(5)	9.79(6)	7.09(-2)
Tank Bot.3	12.0	8.89	1.26(4)	1.40(5)	3.51(5)	5.04(5)	9.28(6)	5.43(-2)
Tank Bot.4	16.2	16.73	9.32(2)	3.04(4)	6.66(4)	9.79(4)	2.61(6)	3.75(-2)
Eng.Env. 1	8.04	-8.66	1.35(4)	1.27(6)	1.10(6)	2.38(6)		
Eng.Env. 2	8.04	-6.56	1.38(4)	5.93(5)	8.83(5)	1.49(6)		
Eng.Env. 3	8.04	-3.67	1.49(4)	4.03(5)	6.98(5)	1.12(6)		
Eng.Env. 4	9.68	0.0	1.80(4)	2.83(5)	5.49(5)	8.50(5)		

TABLE 27

FAST NEUTRON RADIATION LEVELS AT ENGINE ENVELOPE AND TANK BOTTOM
WITH AND WITHOUT FACILITY INTERMEDIATE SHIELDFAST NEUTRON FLUX (n/cm^2 -sec, $E > .9\text{Mev}$)

Dose Point	Radius, ft	Axial Distance from Station 0, ft	Criteria, Shielded Engine Leakage (4,5)	Without Facility Int.Shield	Ratio, W/O Facility Int.Shield to Criteria	With Fac. Int.Shield	Ratio, With Int.Shield to Criteria
Axis 1	0.0	-8.66	4.5 (12)	5.72(11)	0.13	1.73(7)	3.80(-6)
Axis 2	0.0	-6.56	7.0 (11)	7.88(11)	1.13	1.79(7)	2.56(-5)
Axis 3	0.0	-3.28	3.5 (11)	9.65(11)	2.76	3.99(8)	1.14(-3)
Axis 4	0.0	0.0	1.9 (11)	8.88(11)	4.67	1.09(9)	5.74(-3)
Axis 5	0.0	1.64	2.11(11)*	7.32(11)		1.68(9)	
Tank Bot. 1	4.0	2.38	1.15(12)*	7.00(11)		3.46(9)	
Tank Bot. 2	8.0	4.89	1.15(12)*	6.92(11)		3.18(9)	
Tank Bot. 3	12.0	8.89	1.16(12)*	3.96(11)		2.28(9)	
Eng.Env. 1	8.04	-8.66		1.62(12)		4.00(9)	
Eng.Env. 2	8.04	-6.56		1.59(12)		3.80(9)	
Eng.Env. 3	8.04	-3.67		1.38(12)		3.40(9)	

* Fast Neutron Flux for $E > .1\text{ Mev}$ from Reference 5

Figure 40 - Gamma Dose Rate Opposite Service Tower with 6-Foot Water Shield During Operations at 5000 Mw

Figure 41 - Fast Neutron and Gamma Dose Rate Opposite Service Tower During Operation at 5000 Mw

Figure 42 - Gamma Dose Rate Behind Service Tower During Operation at 5000 Mw (to 1200 ft)

Figure 43 - Gamma Dose Rate Behind Service Tower During Operation at 5000 Mw (to 9000 ft)

Figure 44 - Gamma Doses (rad (c)/hr) from E/STS No. 3 During Operation at 5000 Mw with 6-Foot-Thick Water Shield

the gamma dose rates from E/STS No. 3 during operation at 5000 Mw with the 6-ft water shield in place.

C. CONTROL CENTER AND ACCESS TUNNEL RADIATION LEVELS

An evaluation has been made of the control center and access tunnel radiation levels during normal operation. Table 28 presents the gamma-ray dose rates resulting from engine leakage through the 6-ft water shield and contributions from air scatter during operation at 5000 Mw. These gamma dose rates are described in terms of equivalent concrete roof thickness up to 4 ft for distances up to 1000 ft from the engine centerline. From these data, concrete roof thicknesses can be determined as a function of distance from the engine centerline, which satisfy design requirements for normal operation. However, thickness requirements are dictated by conditions that may occur as a result of an accident.

D. SERVICE TOWER RADIATION LEVELS

An evaluation has been made of the radiation levels during 5000 Mw operation at the service tower face, behind service tower wing walls, and inside the service tower with the facility shields in place. Figure 45 gives gamma dose rates at the face of the service tower wall along the tower vertical centerline and five ft from centerline in the direction of the service arm. These data are given as a function of distance above or below Station "O". Figure 46 gives the total dose rate inside the service tower at the tower center as well as along the inner wall, for vertical distances from 80 ft below to 80 ft above Station "O" for 5000 Mw operation with facility shields in place. Figure 47 gives the total dose rate behind the wing walls (20 ft from centerline) as a function of vertical distance from Station "O" for shielded 5000 Mw operation. The face of the service tower is 40 ft from the engine centerline.

TABLE 28

OPERATING GAMMA DOSE RATE UNDER VARIOUS THICKNESSES OF CONCRETE
VS DISTANCE WITH SIDE SHIELD IN PLACE

Distance, ft	Concrete Thickness, ft	Gamma Ray Dose Rate, Rad(c)/hr		
		Engine Leakage	Air Scatter	Total
30	0	1.135×10^4	2.47×10^2	1.16×10^4
50	0	9.22×10^3	3.52×10^2	9.56×10^3
	1	7.03×10^1	1.43×10^0	7.18×10^1
	2	9.97×10^{-1}	2.52×10^{-3}	1.00×10^0
	4	6.45×10^{-4}		6.45×10^{-4}
75	0	5.34×10^3	3.77×10^2	5.72×10^3
100	0	3.24×10^3	2.75×10^2	3.61×10^3
	1	4.93×10^{-1}	6.90×10^{-1}	1.18×10^0
	2	2.93×10^{-4}	7.50×10^{-4}	1.04×10^{-3}
	4	--		*
150	0	1.47×10^3	3.12×10^2	1.78×10^3
200	0	8.08×10^2	2.34×10^2	1.04×10^3
	1	5.05×10^{-5}	2.53×10^{-1}	2.53×10^{-1}
	2		1.69×10^{-4}	1.69×10^{-4}
	4	--		
300	0	3.34×10^2	1.50×10^2	4.84×10^2
500	0	1.01×10^2	7.40×10^1	1.75×10^2
	1	--	2.21×10^{-2}	2.21×10^{-2}
	2	--	5.01×10^{-6}	
	4	--		
660	0	4.90×10^1	3.3×10^1	8.20×10^1
	1	--	6.94×10^{-3}	6.94×10^{-3}
	2	--	8.2×10^{-7}	
750	0	3.60×10^1	3.12×10^1	7.72×10^1
1000	0	1.62×10^1	1.33×10^1	1.95×10^1
	1	--	1.35×10^{-3}	1.35×10^{-3}
	2	--	1.00×10^{-7}	

*Indicates no increase over background

Figure 45 - Radiation Levels at Face of Service Tower

Figure 46 - Total Dose Rate Inside Service Tower

Figure 47 - Total Dose Rate of Points Behind Wing Walls,
Y = 20 Feet

E. SERVICE TOWER CONCRETE HEATING

Concrete heat generation rates during 5000 Mw operation have been determined for the service tower concrete, considering a 10-ft-wide cut-out in the top shield to accommodate the service arm (Figure 4). Figure 48 shows the heating rates at the face of the service tower, and Figure 49 describes heating rates as a function of depth into the concrete at locations of peak heating due to streaming through the 10-ft cut-out in the top shield.

F. DUCT VAULT RADIATION LEVELS

In the duct vault a limit of 10^{-4} Btu/in.³-sec in nuclear heat generation has been imposed upon the design for the duct vault concrete. To achieve this objective the bottom shield is augmented by a duct plenum shield as shown in Figure 35, in addition to the 5-ft-thick bottom shield. The radiation heating rates in the duct vault concrete for the preceding conditions are presented in Table 29 for the case of 5000 Mw engine operation.

Figure 48 - Heating Rates at Face of Tower

Figure 49 - Heating Rate vs Depth in Service Tower 45 Feet Above Station 0

TABLE 29

CONCRETE RADIATION HEATING RATES IN DUCT VAULT
WITH FACILITY SHIELDS IN PLACE

X-Distance From Side Surface of Vault, ft	Z-Distance Below Top Surface of Vault, ft	Concrete Heating Rates, Btu/in ³ -sec
0.0	0	3.78×10^{-5}
0.0	10	6.13×10^{-5}
0.0	20	1.68×10^{-4}
0.0	30	1.25×10^{-4}
0.0	40	6.44×10^{-5}
0.5	0	1.83×10^{-5}
0.5	10	2.19×10^{-5}
0.5	20	4.58×10^{-5}
0.5	30	3.25×10^{-5}
0.5	40	1.50×10^{-5}
1.0	0	1.03×10^{-5}
1.0	10	8.12×10^{-6}
1.0	20	1.63×10^{-5}
1.0	30	9.70×10^{-6}
1.0	40	4.82×10^{-6}
2.0	0	5.42×10^{-6}
2.0	10	1.26×10^{-6}
2.0	20	2.16×10^{-6}
2.0	30	1.21×10^{-6}
2.0	40	6.83×10^{-7}

VIII. RADIATION ENVIRONMENT AFTER NORMAL SHUTDOWN WITH SHIELDED ENGINE ON STAND

The radiation environment due to a facility shielded engine on the stand following a single normal engine operation for 30 min at 5000 Mw is presented in this section. Facility shields include the top shield, support ring shield, intermediate shield, movable side shield, and bottom shield.

A. ADJACENT VICINITY

Figures 50 and 51 give the direct and air-scattered contributions to the shutdown gamma dose rate out to 9000 feet from the engine centerline in the direction opposite the service tower. These data pertain to an engine shielded by the facility shields, at 1 hr after shutdown following 30 min operation at 5000 Mw. Figures 56 and 57 report the air-scatter contribution to the shutdown gamma dose rate behind the service tower for the same operating conditions and shield configurations as shown in Figures 50 and 51. A combination of the results, following 1 hr shutdown, shown in Figures 50, 51, 56, 57, is presented in the isodose map in Figure 52. Corresponding results 24 hr after shutdown following 30-min operation at 5000 Mw for the facility shielded engine are presented in Figure 55 derived from Figures 53, 54, 58, and 59 (24-hr decay), respectively.

B. CONTROL CENTER AND ACCESS TUNNELS

The radiation environment, 1 hr following a 30-min engine operation at 5000 Mw, at the access tunnel and control center, has been determined as a function of concrete thickness and distance from the engine centerline. Table 30 gives the gamma dose rate resulting from engine leakage through the 6-ft water shield and air-scatter contributions at 1 hr after shutdown. A comparison with the engine leakage that occurs when the movable side shield is lowered is also presented. Using this data and the corresponding tabulation (Table 28) for operating conditions, a specification of

Figure 50 - Shutdown Gamma Dose Rate to 1200 Feet Opposite Service Tower with 6-Foot Water Shield 1 Hour After 30 Minutes at 5000 Mw

Figure 51 - Shutdown Gamma Dose Rate to 9000 Feet Opposite Service Tower with 6 Foot Water Shield 1 Hour After 30 Minutes at 5000 Mw

Figure 52 - Gamma Doses (rad (c)/hr) from E/STS No. 3, 1 Hour After Shutdown Following 30 Minutes at 5000 Mw with 6-Foot-Thick Water Shield

Figure 53 - Shutdown Gamma Dose Rate to 1200 Feet Opposite Service Tower with 6-Foot Water Shield 24 Hours After 30 Minutes at 5000 Mw

Figure 54 - Shutdown Gamma Dose Rate to 9000 Feet Opposite Service Tower with 6-Foot Water Shield 24 Hours After 30 Minutes at 5000 Mw

Figure 56 -Shutdown Gamma Dose Rate to 1200 Feet Behind Service
Tower with 6-Foot Water Shield 1 Hour After 30 Minutes
at 5000 Mw

Figure 57 - Shutdown Gamma Dose Rate to 9000 Feet Behind Service Tower with 6-Foot Water Shield 1 Hour After 30 Minutes at 5000 Mw

Figure 58 - Shutdown Gamma Dose Rate to 1200 Feet Behind Service Tower with 6-Foot Water Shield 24 Hours After 30 Minutes at 5000 Mw

Figure 59 - Shutdown Gamma Dose Rate to 9000 Feet Behind Service Tower with 6-Foot Water Shield 24 Hours After 30 Minutes at 5000 Mw

TABLE 30

DOSE RATES UNDER VARIOUS THICKNESSES OF CONCRETE
VS DISTANCE 1 HOUR AFTER 30 MINUTES OPERATION
AT 5000 Mw WITH SIDE SHIELD IN PLACE

Distance, ft	Concrete Thickness, ft	Gamma Ray Dose Rate, Rad(c)/hr		Total
		Engine Leakage	Air Scatter	
30	0	1.08×10^1	1.9×10^0	1.27×10^1
50	0	1.03×10^1	2.72×10^0	1.30×10^1
	1	1.40×10^{-2}	1.1×10^{-2}	2.5×10^{-2}
	2	3.39×10^{-5}	1.95×10^{-5}	*
	4	--	--	
75	0	6.21×10^0	2.9×10^0	9.11×10^0
100	0	3.80×10^0	2.88×10^0	6.68×10^0
	1	2.24×10^{-5}	6.1×10^{-3}	6.12×10^{-3}
	2		5.77×10^{-6}	
	4	--		
150	0	1.72×10^0	2.4×10^0	4.12×10^0
200	0	9.37×10^{-1}	1.80×10^0	2.74×10^0
	1		1.95×10^{-3}	1.95×10^{-3}
	2	--	1.3×10^{-6}	
	4	--		
300	0	3.78×10^{-1}	1.17×10^0	1.55×10^0
500	0	1.08×10^{-1}	5.46×10^{-1}	6.54×10^{-1}
	1	--	1.7×10^{-4}	1.7×10^{-4}
	2	--	3.9×10^{-8}	
	4	--		
750	0	3.53×10^{-2}	2.4×10^{-1}	2.75×10^{-1}
1000	0	1.46×10^{-2}	1.03×10^{-1}	1.18×10^{-1}
	1	--	1.05×10^{-5}	
	2	--		

*Indicates no increase over background

the concrete roof thickness requirements may be determined for the access tunnel limited to normal operation. This exercise is academic since the control center and access tunnels must also meet accident condition requirements. The latter considerations dictate this shielding, as described in Section X.E.

C. SERVICE TOWER

During the first few hours following a 30-min run at 5000 Mw, access to the service tower is limited due to sources of radiation from the engine on the stand. Figure 60 gives the dose rate at the face of the service tower due to fission product decay 1 hr after the run with the side shield in place and with the side shield lowered. Figure 61 presents corresponding dose-rate data at 24 hr after operating the engine at 5000 Mw for 30 min. Figures 62 and 63 present 1- and 24-hr data, respectively, for detector points located at the interior wall of the service tower and data resulting from fission-product decay with the engine on the stand after 30 min of operation at 5000 Mw.

D. DUCT VAULT RADIATION LEVELS

While the engine is in place on the test stand, fission-product decay sources in the engine dominate the shut-down radiation environment in the duct vault. Table 31 presents the gamma radiation levels at the surface of the vertical concrete duct vault wall 24 hr after shutdown of a 30-min engine run at 5000 Mw. The data shows conditions with the facility shields in place around the engine. The detector points along the vertical wall are delineated, with the distance measured from the top surface of the concrete wall. The detector points along the top horizontal surface of the vertical concrete wall are indicated by $X=0$ at the vertical surface of the wall and extending along the top surface of the concrete wall directly below the test stand bottom shield.

Figure 60 - Dose Rates at Face of Service Tower Due to Fission Product Decay 1 Hour After 30 Minutes of Operation at 5000 Mw

Figure 61 - Dose Rates at Face of Service Tower Due to Fission Product Decay 24 Hours After 30 Minutes of Operation at 5000 Mw

Figure 62 - Dose Rates at Interior Wall of Service Tower Due to Fission Product Decay 1 Hour After 30 Minutes of Operation at 5000 Mw

Figure 63 - Dose Rates at Interior Wall of Service Tower Due to Fission Product Decay 24 Hours After 30 Minutes of Operation at 5000 Mw

TABLE 31

GAMMA RADIATION LEVELS IN DUCT VAULT
24 HOURS AFTER SHUTDOWN OF 30 MIN - 5000 Mw RUN
(ENGINE IN PLACE)

<u>Side Surface of Vault</u>	<u>Z-Distance Below Top Surface, ft</u>	<u>Dose Rate, Rad(c)/hr</u>
	0	9.22×10^1
	10	1.50×10^2
	20	4.10×10^2
	30	3.05×10^2
	40	1.57×10^2
<u>Top Surface of Vault</u>	<u>X-Distance Along Top Surface (Ft)</u>	
	0.5	4.46×10^1
	1.0	2.51×10^1
	2.0	1.32×10^1

IX. TEST STAND RADIATION ENVIRONMENT WITH THE ENGINE REMOVED

A major area of uncertainty in predicting radiation environments and shield design occurs in the neutron-activation analysis of the test stand components with the engine removed. This problem is attributed to the complexity of the analysis of neutron distributions in material whose capture yields residual activity of complex distributions and geometries, and to the uncertainty in composition of structural materials such as aluminum, steel, and concrete where trace elements often constitute the major radiation emitters during the decay period. To compensate for this condition an uncertainty factor of 5 is specified for the shutdown radiation environment. The shutdown radiation environment when the engine has been removed after 5000 Mw operation for 30 min is described in this section. These nominal values have not been increased by the preceding factor of 5.

A. CONTRIBUTIONS ABOVE PAD FROM SUB-GRADE SOURCES WITH THE DUCT COVER SHIELD IN PLACE

The design of the duct cover shield is discussed in Section VI. An analysis has been made to evaluate the shutdown radiation environment at detector locations above the pad with the duct cover shield in place. This analysis considers exhaust duct sources of fission product contamination and activation of the intermediate shield and the bottom shield. For this analysis the engine was assumed to have been removed from the test stand and the intermediate shield and movable side shield were assumed to be full of water in the lowered positions. Table 32 gives the activation and fission product plate-out dose rate contributions at selected detector locations above the pad at 310 hr after a 30 min, 5000 Mw run (with the duct cover in place). Sources include the bottom shield, intermediate shield, exhaust duct, and plenum activation as well as fission product plate-out in the duct. The fission product plate-out source is derived from the assumption that the fission product source is distributed in the exhaust duct inlet portion. The criteria stipulate that 0.06% of the total available fission

TABLE 32

ACTIVATION AND FISSION PRODUCT DOSE RATES ABOVE PAD
 310 HOURS AFTER SHUTDOWN OF 30 MIN - 5000 Mw RUN
 WITH DUCT COVER SHIELD IN PLACE

Detector Location		Dose Rates (Rad(c)/hr)					
X-Distance From Core Centerline, ft	Z-Distance Above Pad, ft	Fiss. Prod. in Diff. Mouth	Act. in Top of Plenum	Act. in Diff. Mouth	Act. in Bottom Shield	Act. in Int. Shield	Total
21	0	6.71(-6)	1.36(-5)	4.75(-6)	9.75(-7)	5.57(-8)	*
0	1	3.24(-4)		2.81(-4)		1.04(-4)	7.09(-4)
3	1	5.64(-4)		4.96(-4)		1.90(-4)	1.25(-3)
6	1	6.76(-4)	5.32(-5)	6.08(-4)	8.48(-9)	5.72(-4)	1.91(-3)
7.15	1	7.76(-6)	2.82(-5)	4.96(-6)	1.41(-7)	8.28(-4)	8.69(-4)
7.38	1	3.59(-6)	9.96(-6)	2.12(-6)	3.82(-8)	8.68(-4)	9.04(-4)
9	1	1.87(-7)		5.92(-8)	3.08(-4)	2.84(-4)	5.92(-4)
12	1	8.33(-5)		6.59(-5)	2.93(-4)	3.56(-4)	7.98(-4)
15.25	1	6.76(-6)		3.49(-6)	2.50(-4)	1.90(-4)	4.50(-4)
16	1	3.89(-6)		1.83(-6)	4.53(-4)	7.43(-5)	5.33(-4)
17	1	2.04(-6)		9.08(-7)	3.03(-4)	5.33(-5)	3.59(-4)
19	1	3.81(-6)		2.59(-6)	8.11(-5)	1.34(-5)	1.01(-4)
21	1	7.46(-6)		5.72(-6)	1.10(-5)	2.23(-6)	
22	1	4.94(-6)		3.59(-6)	3.61(-6)	6.82(-7)	
0	3	4.28(-4)		3.76(-4)		3.26(-4)	1.13(-3)
15	3	8.43(-5)		7.38(-5)	2.72(-4)	2.31(-4)	6.61(-4)
16	3	5.14(-5)		4.30(-5)	2.04(-4)	1.44(-4)	4.42(-4)
17	3	4.58(-5)		3.84(-5)	3.12(-4)	7.38(-5)	4.70(-4)
19	3	1.47(-5)		1.06(-5)	1.45(-4)	3.17(-5)	2.02(-4)
21	3	5.19(-6)		3.60(-6)	6.54(-5)	1.23(-5)	8.65(-5)
22	3	3.87(-6)		2.80(-6)	3.58(-5)	7.95(-6)	
0	6.53	6.12(-4)	8.66(-7)	5.48(-4)	4.76(-9)	3.35(-4)	1.50(-3)
0	10	6.36(-4)	5.84(-6)	5.80(-4)	3.40(-6)	2.24(-4)	1.45(-3)
0	20	3.76(-4)	9.68(-6)	3.51(-4)	4.00(-5)	1.23(-4)	9.00(-4)
0	30	2.16(-4)	6.36(-6)	2.02(-4)	6.11(-5)	8.33(-5)	5.69(-4)

*Indicates no increase over background

product inventory following a 30 min, 5000 Mw run be located in the duct. The assumed material compositions of each of the activation source components are as specified in Section VI. Figures 64 and 65 describe the resulting radial and axial distributions of radiation levels above the pad, with the duct cover shield in place due to subgrade sources, 2 weeks after shutdown following a 30 min, 5000 Mw run. The radiation environment above the pad due to sub-grade sources must be added to the contributions from above grade facility sources, discussed in Section IX.B below, to describe the complete radiation environment. Sub-grade source contributions are seen to be comparable to above-grade sources at 1 ft above the pad.

B. CONTRIBUTIONS FROM ABOVE-GRADE SOURCES

Table 33 presents the dose-rate contributions at selected locations due to activation of above-grade facility components on the basis of a 2-week shutdown period after a 30 min 5000 Mw run. Detector locations include locations on the duct centerline above the pad, at the service arm, on the service tower face, inside the tower, and behind the wing wall.

A look at the access requirements at some of these locations is now in order. The total dose rate contribution in the area of the pad from above-grade and below-grade sources is about 8 mr/hr. Applying a factor of uncertainty of 5 to this value would bring the dose rate up to a level of 40 mrem/hr, which is nominally consistent with the objective of 25 mrem/hr. The total dose rate at the lower service arm is seen to be about 7 mr/hr. Using a factor of uncertainty of 5, one obtains a value of 35 mr/hr. The criteria document provides for an access requirement of less than 2-1/2 mr/hr at this location, which is a factor of ten lower than the indicated value. This access requirement, however, is not in accordance with past experience and a review for possible revision of this section of the criteria document is recommended before design changes are formulated.

Figure 64 - Radial Distribution of Radiation Level Above Pad Due to Subgrade Sources, 2 Weeks After Shutdown of 30 Minutes - 5000 Mw Run, with Duct Cover Shield in Place

Figure 65 - Axial Distribution of Radiation Levels Above Pad Due to Subgrade Sources, 2 Weeks After Shutdown of 30 Minutes - 5000 Mw Run, with Duct Cover Shield in Place

TABLE 33

DOSE RATE CONTRIBUTIONS, AT SELECTED LOCATIONS DUE TO RESIDUAL, ABOVE-GRADE SOURCES, TWO WEEKS AFTER SHUTDOWN

Detector Location Relative to Station 0.0				Dose Rate Contributions, Rad(c)/hr							Total
				Tower Concrete Activation	Side Shield Activation	Support Ring Activation	Piping Act. at Upper Umbilical Arm	Al. Tower Piping Activation	Steel Platform At Upper Umbilical Arm	Tower Rebar Activation	
X, ft	Y, ft	Z, ft	Comment								
22.0	-6.72	11.3	Service Arm	1.63(-4)	4.61(-4)	3.32(-3)	1.13(-3)	4.79(-6)	1.01(-3)	7.32(-4)	6.82(-3)
30.0	-9.17	11.3	Service Arm	1.32(-4)	1.41(-4)	1.06(-3)	1.17(-3)	5.38(-6)	1.05(-3)	5.36(-4)	4.11(-3)
40.0	-12.2	11.3	Service Arm	--	3.96(-5)	2.29(-4)	1.18(-3)	5.63(-6)	1.06(-3)	--	2.52(-3)
40.0	0.0	30.0	Service Tower	--	1.55(-4)	6.96(-4)	2.19(-3)	2.04(-5)	1.95(-3)	--	5.04(-3)
40.0	0.0	50.0	Service Tower	2.39(-7)	1.46(-4)	6.62(-4)	5.39(-3)	8.81(-5)	4.80(-3)	4.94(-7)	1.12(-2)
40.0	0.0	70.0	Service Tower	5.19(-6)	9.94(-5)	4.33(-4)	2.82(-2)	1.18(-4)	2.51(-2)	1.20(-5)	5.41(-2)
42.0	0.0	30.0	Inside Tower	--	1.63(-7)	3.46(-7)	--	--	--	--	*
42.0	0.0	50.0	Inside Tower	5.96(-8)	4.15(-8)	7.68(-8)	--	--	--	8.48(-8)	
42.0	0.0	70.0	Inside Tower	7.66(-7)	6.06(-9)	8.39(-9)	--	1.56(-9)	--	1.04(-6)	
0.0	0.0	-33.4	1 Ft Above Pad	3.55(-6)	1.44(-4)	4.99(-3)	1.66(-4)	1.30(-7)	1.48(-4)	1.50(-5)	5.47(-3)
0.0	0.0	-24.4	10 Ft Above Pad	1.89(-5)	3.40(-4)	8.38(-3)	3.85(-4)	5.47(-7)	3.43(-4)	7.24(-5)	9.54(-3)
40.0	-20.0	0.0	Behind Wing Wall	--	2.32(-9)	--	--	--	--	--	
40.0	-20.0	60.0	Behind Wing Wall	9.73(-7)	1.13(-8)	1.69(-8)	--	1.46(-9)	--	1.03(-6)	

*Indicates no increase over background

Principal above-grade activation sources include the support ring and service-arm platform and piping. Each of these sources is exposed to engine leakage radiation during a run. A reduction in the service-arm platform, support ring and columns, and piping sources could be accomplished by modifications in the design of the top section of the side shield relative to the service arm penetration. It should be pointed out that the data presented in Table 33 apply to a run with no facility intermediate shield in place. The latter condition will probably not have a significant effect but should not be discounted without further study.

Prior to the initiation of any design modifications which would reduce the contributions from these sources, these contributions should be compared with those from the Nuclear Ground Test Module. According to one study*, at comparable times after shutdown, dose rates of 7.0 mrem/hr and 150 mrem/hr rad (c)/hr exist at the pad and service arm, respectively, as a result of NGTM activation during a run. A factor of uncertainty of 5 should be applied to these values.

C. EXHAUST DUCT CONTRIBUTIONS TO THE VAULT RADIATION ENVIRONMENT

The dose rates at selected locations in the duct vault from residual activity in the exhaust duct 2 weeks after shutdown with the engine removed following a 30 min 5000 Mw run is given in Table 34. This data can be modified for decay times of up to 70 days using data shown in Figure 67 and Tables 39 through 43, describing the decay of various steels that may be selected for the duct design.

D. GENERAL DATA ON MATERIALS ACTIVATION

An effective method of reducing the shutdown radiation environment at critical access locations is to use fabrication materials having con-

*W.C. Loomis, NGTM Activation Dose Rates, R-AS-VP-57-67, prepared by Marshall Space Flight Center, Huntsville, Alabama, May 1, 1967.

TABLE 34

DOSE RATES AT SELECTED LOCATIONS IN DUCT VAULT FROM RESIDUAL
ACTIVITY IN EXHAUST DUCT TWO WEEKS AFTER SHUTDOWN

<u>Distance From Duct Centerline, ft</u>	<u>Distance Below Pad Level, ft</u>	<u>Rad(c)/hr</u>
12.5	9.0	8.0(-3)
7.0	23.1	3.0(0)
12.5	23.1	1.0(0)
7.0	45.8	1.9(0)
12.5	45.8	9.3(-1)
12.5	57.0	6.7(-1)
7.0	68.0	2.4(-1)
12.5	68.0	2.4(-1)

trolled quantities of isotopes responsible for residual activity. The effect of this approach is shown in Figures 66 through 69, which give relative gamma-source strengths after shutdown for principal structural materials such as steel, aluminum, and concrete. The composition of these materials is shown in Table 35. Similar materials often show differences in residual activity that may be decades apart in magnitude. For this reason critical facility components that contribute substantially to the radiation environment have been specified as nuclear grade materials. These materials can be obtained with no significant cost increase.

Neutron-induced activation of various structural materials has been calculated with the ACT-II computer program¹⁶. Three types of activation data have been reduced. One set is based on a uniform neutron spectrum and the other two sets are based on $\alpha = 0^\circ$ end-leakage, and $\alpha = 90^\circ$ side-leakage spectra.

1. Activation from Uniform Neutron Spectrum

The uniform neutron spectrum calculations are intended to be used to calculate activation of materials for spectra other than the engine leakage spectra by adjusting the reduced ACT-II output by the ratio of the four group neutron fluxes as they become available in further analyses (e.g. LH_2 scattered spectrum). These data are not included here, but are considered to be intermediate results which will be utilized in calculating activation levels for various neutron spectra during the next phase of this analytical effort. The uniform neutron spectrum employed to date is based on a neutron number flux of 1×10^{10} n/cm²-sec in each of the four neutron groups in the ACT-II calculations: 0 to 0.625 ev, 0.625 ev to 5 Kev, 5 Kev to 0.82 Mev, and neutrons with energies greater than 0.82 Mev.

Activation analyses were made of several steel, concrete, and aluminum compositions. Table 35 gives the elemental compositions of the steels and concretes.

Figure 66 - Decay of Neutron-Induced Activation in Concrete Following Exposure to End Leakage Spectrum for 30 Minutes at 5000 Mw

Normalized to $5.3 \times 10^5 \text{ Mev/cm}^3\text{-sec}$

Figure 67 - Decay of Neutron-Induced Activation of Steel Following Exposure to End Leakage Spectrum for 30 Minutes at 5000 Mw

Normalized to $1.33 \times 10^4 \text{ Mev/cm}^3\text{-sec}$

Figure 68 - Decay of Neutron Induced Activation in Mil Spec 6061 Aluminum and Nuclear Grade 1100 Aluminum Following Exposure to End Leakage Spectrum for 30 Minutes at 5000 Mw

Figure 69 - Decay of Neutron-Induced Activation of Nuclear Grade Aluminum Following Exposure to End Leakage Spectrum for 30 Minutes at 5000 Mw

TABLE 35
ELEMENTAL COMPOSITIONS FOR ACTIVATION ANALYSES

Elements	Materials, Weight Fraction							
	Concrete Special Composition	Concrete Bingham	Concrete 05	Carbon Steel (A-36)	Carbon Steel Special	SS-321	SS-304	SS-347
H	5.6(-2)	5.6(-3)	8.8(-3)					
O	5.0(-1)	5.0(-1)	4.9(-1)					
Al		4.6(-2)	5.2(-2)		3.8(-4)			
Cd	1.0(-5)	2.2(-5)						
Cr	3.0(-7)	9.0(-7)			9.0(-4)	1.8(-1)	2.0(-1)	1.9(-1)
Ni	5.0(-6)	1.7(-5)			1.4(-3)	1.1(-1)	1.0(-1)	1.3(-1)
Co	2.5(-8)	9.5(-8)				1.0(-3)	1.0(-3)	1.0(-3)
Cu	5.0(-5)	1.8(-4)			2.2(-3)	5.0(-3)		
Zn	2.0(-5)	5.6(-5)					5.0(-5)	
Fe	7.0(-2)	1.7(-2)	4.9(-2)	9.8(-1)	9.8(-1)	6.8(-1)	6.9(-1)	6.5(-1)
Mn	1.8(-2)		1.8(-2)	1.2(-2)	1.2(-2)	2.0(-2)	2.0(-2)	2.0(-2)
Sc	2.0(-6)	2.6(-6)						
Ti	6.8(-3)		6.8(-3)		1.4(-4)	7.0(-3)		
Na	1.2(-2)	1.7(-2)	1.2(-2)					
Ta	1.1(-6)	7.8(-6)					5.0(-4)	8.0(-3)
Sn	1.0(-6)	2.2(-6)						
Si	3.4(-1)	3.1(-1)	2.5(-1)	3.0(-5)	1.8(-3)	1.0(-2)		
Ca		8.3(-2)	9.7(-2)					
Mg		2.4(-3)	8.0(-4)					
K		1.9(-2)	1.0(-2)					
S		1.2(-3)	1.5(-3)		3.0(-4)			2.0(-4)
Mo					2.4(-4)	5.0(-3)		
P					1.8(-4)			4.5(-4)
V								
C			3.2(-3)					

TABLE 35 (cont.)

ELEMENTAL COMPOSITIONS FOR ACTIVATION ANALYSES

Elements	Materials, Weight Fraction						
	Nuclear Grade Al-1100	Al-2219	Nuclear Grade Al-2219	Nuclear Grade Al-5083	Nuclear Grade Al-5454	Nuclear Grade Al-6061	Al-6061
H							
O							
Al	9.9(-1)	9.3(-1)	9.2(-1)	9.3(-1)	9.5(-1)	9.7(-1)	9.6(-1)
Cd							
Cr				2.5(-3)	2.0(-3)	3.5(-3)	2.5(-3)
Ni	2.0(-4)	5.0(-4)	2.0(-4)	2.0(-4)	2.0(-4)	2.0(-4)	5.0(-4)
Co	1.0(-5)	5.0(-4)	1.0(-5)	1.0(-5)	1.0(-5)	1.0(-5)	5.0(-4)
Cu	2.0(-3)	6.0(-2)	6.8(-2)	1.0(-3)	1.0(-3)	4.0(-3)	3.0(-3)
Zn	4.0(-4)	1.0(-3)	4.0(-4)	4.0(-4)	4.0(-4)	4.0(-4)	2.5(-3)
Fe	1.0(-2)	3.0(-3)	3.0(-3)	4.0(-3)	4.0(-3)	7.0(-3)	7.0(-3)
Mn	5.0(-4)	3.0(-3)	4.0(-3)	1.0(-2)	1.0(-2)	1.5(-3)	1.5(-3)
Sc							
Ti		5.0(-4)	1.0(-3)	1.5(-3)	2.0(-3)	1.5(-3)	1.5(-3)
Na							
Ta							
Sn							
Si		2.0(-3)		4.0(-3)		4.0(-3)	6.0(-3)
Ca							
Mg		2.0(-4)	2.0(-4)	4.9(-2)	3.0(-2)	1.2(-2)	1.0(-2)
K							
S							
Mo							
P							
V		1.0(-3)	1.5(-3)				
C							
Zr			2.5(-3)				

The activations of 2219 and 6061 aluminum were calculated previously for Military Specification compositions. The presence of zinc, cobalt, and nickel caused the activity to remain at high levels for long decay times. Nuclear grade aluminum with the zinc, cobalt and nickel contents limited to .04, .001 and .02 wt%, respectively, can be obtained at no significant increase in cost. Table 35 repeats the compositions of 2219 and 6061 which were formerly analyzed and shows the compositions of the nuclear grade aluminums which were analyzed. In addition, the chromium content of the 6061 aluminums was varied and the resultant effect was shown to be an increase in the strength of the low energy gammas with increased chromium content.

Both single and cycled full-duration runs have been calculated. The activation due to 10 sec of full-power operation has also been calculated. These data will be used to evaluate activation as the LH_2 is consumed from the GTM. During a brief period, neutron streaming affects regions that would normally be shadow-shielded by the LH_2 during the major portion of the run.

2. Activation from $\alpha = 0^\circ$, End-Leakage Spectrum

Table 36 presents activation source strengths for end-leakage spectrum irradiation of a special concrete composition suggested by Kaiser Engineers. The end-leakage spectrum (in the $\alpha = 0^\circ$ direction) is indicated at the base of the table. As shown, the operating history is based on a single 30-min run at a power level of 5000 Mw. Data are given for four gamma energy groups and the summation of these groupwise contributions in units of $(\text{Mev}/\text{cm}^3\text{-sec})$.

Table 37 gives similar data for a material designated as conventional concrete by W. G. Bingham, of the Bechtel Corporation, in

TABLE 36

NEUTRON-INDUCED ACTIVATION IN SPECIAL COMPOSITION CONCRETE
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - $S_v(\text{mev}/\text{cm}^3\text{-sec})$				
	0.4 Mev	0.5 Mev	1.5 Mev	2.0 Mev	Total
1	4.12×10^1	1.53×10^7	3.37×10^5	1.44×10^7	3.00×10^7
3	2.20×10^1	8.91×10^6	3.07×10^5	8.64×10^6	1.79×10^7
5	1.61×10^1	5.21×10^6	2.79×10^5	5.25×10^6	1.07×10^7
7	1.21×10^1	3.04×10^6	2.55×10^5	3.24×10^6	6.53×10^6
10	8.32×10^0	1.36×10^6	2.22×10^5	1.67×10^6	3.25×10^6
30	2.64×10^0	6.32×10^3	8.82×10^4	1.83×10^5	2.77×10^5
50	1.55×10^0	3.76×10^1	3.51×10^4	7.02×10^4	1.05×10^5
70	1.13×10^0	8.33×10^0	1.40×10^4	2.79×10^4	4.19×10^4
100	8.93×10^{-1}	8.09×10^0	3.54×10^3	6.97×10^3	1.05×10^4
310	6.24×10^{-1}	7.62×10^0	6.09×10^1	4.26×10^{-1}	6.95×10^1
510	5.42×10^{-1}	7.07×10^0	5.10×10^1	1.69×10^{-4}	5.86×10^1
710	5.12×10^{-1}	6.86×10^0	4.80×10^1	1.52×10^{-4}	5.54×10^1
1010	4.51×10^{-1}	6.33×10^0	4.02×10^1	1.33×10^{-4}	4.70×10^1
3110	2.03×10^{-1}	3.80×10^0	1.24×10^1	5.66×10^{-5}	1.64×10^1

* $\alpha = 0^\circ$ End Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 37

NEUTRON-INDUCED ACTIVATION IN CONCRETE (BINGHAM)
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - $S_V(\text{mev}/\text{cm}^3\text{-sec})$				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	7.6×10^1	1.55×10^3	4.78×10^5	9.38×10^5	1.42×10^6
3	5.05×10^1	1.44×10^2	4.35×10^5	81.50×10^5	1.29×10^6
5	3.80×10^1	9.06×10^1	3.97×10^5	7.75×10^5	1.17×10^6
7	2.92×10^1	6.03×10^1	3.60×10^5	7.06×10^5	1.07×10^6
10	2.10×10^1	3.27×10^1	3.13×10^5	6.14×10^5	9.28×10^5
30	7.76×10^0	9.35×10^0	1.25×10^5	2.44×10^5	3.69×10^5
50	4.83×10^0	7.81×10^0	4.91×10^4	9.69×10^4	1.46×10^5
70	$<10^1$	$<10^1$	1.95×10^4	3.85×10^4	5.79×10^4
100			4.89×10^3	9.60×10^3	1.45×10^4
310			4.07×10^1	$<10^1$	5.08×10^1
510			3.69×10^1		4.56×10^1
710			3.40×10^1		4.21×10^1
1010			3.03×10^1		3.78×10^1
3110			1.47×10^1		1.87×10^1

* $\alpha = 0^0$ End Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

a paper delivered at a national meeting of the American Nuclear Society.* These calculations are based on a copy of the delivered paper, but differences exist between the delivered paper and the summary published in the ANS Transactions.** One of the major differences between the Bingham and the Type 05 concrete compositions is the inclusion of a number of elements in the parts per million category, which result in long-lived activation products. In addition, an independent investigation of concrete described in the delivered paper indicated that it is similar to existing NRDS concrete.

Table 38 presents the activation source strengths for end-leakage spectrum irradiation of Type 05 concrete, which has been found to parallel closely concrete mixtures used at the NRDS Test Cell A and ETS-1.

Tables 39 through 43 present volumetric gamma-ray source strengths resulting from activation of two carbon steel and three stainless steel compositions. The carbon steel designated as a special composition in Table 40 is based on materials data supplied by Kaiser Engineers. The other steel compositions are based on standard military specifications.

Tables 44 through 50 give similar end-leakage spectrum activation data for aluminum. Standard military specifications and nuclear-grade specifications were used for elemental compositions of Aluminum 2219 and 6061, and nuclear grade specifications only were used for the remainder.

Figure 66 presents the relative decay of the concrete activation total source strengths as a function of time after shutdown, normalized to Type 05 concrete activity 1 day after shutdown. The groupwise energy breakdown for any point on these curves can be obtained from the corresponding tables. The irradiation spectrum in these cases is based on $\alpha = 0^\circ$, end leakage.

*W.G. Bingham, Low Activation Shielding Materials for Nuclear Reactor Environmental Test Chambers, as presented at the Eleventh Annual Meeting of the American Nuclear Society, 21 June 1965.

**Ibid, Summary published in ANS Transactions, Vol. 8, No. 1, June 1965.

TABLE 38

NEUTRON-INDUCED ACTIVATION IN TYPE 05 CONCRETE
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - S_v (neut/cm ³ -sec)				
	0.4 Mev	0.5 Mev	1.5 Mev	2.0 Mev	Total
1	1.07×10^{-1}	1.59×10^7	3.48×10^5	1.50×10^7	3.12×10^7
3	5.22×10^{-1}	9.27×10^6	3.17×10^5	8.97×10^6	1.86×10^7
5	5.31×10^{-1}	5.42×10^6	2.89×10^5	5.46×10^6	1.12×10^7
7	5.45×10^{-1}	3.16×10^6	7.63×10^5	3.36×10^6	6.78×10^6
10	5.58×10^{-1}	1.41×10^6	2.38×10^5	1.72×10^6	3.37×10^6
30	6.40×10^{-1}	6.50×10^3	9.05×10^4	1.84×10^5	2.81×10^5
50	6.97×10^{-1}	3.28×10^1	3.58×10^4	7.09×10^4	1.07×10^5
70	7.27×10^{-1}	2.29×10^0	1.42×10^4	2.82×10^4	4.25×10^4
100	7.39×10^{-1}	2.09×10^0	3.58×10^3	7.03×10^3	1.06×10^4
310	5.01×10^{-1}	1.91×10^0	3.86×10^1	4.29×10^{-1}	4.14×10^1
510	3.00×10^{-1}	1.83×10^0	3.10×10^1	4.15×10^{-5}	3.37×10^1
710	2.4×10^{-1}	1.77×10^0	2.90×10^1	4.02×10^{-9}	3.10×10^1
1010	1.45×10^{-1}	1.71×10^0	2.30×10^1	3.76×10^{-15}	2.52×10^1
3110	3.59×10^{-2}	1.36×10^0	6.21×10^0	0.0	7.60×10^0

* $\alpha = 0^\circ$ End Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 39

NEUTRON-INDUCED ACTIVATION IN CARBON STEEL A-36
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - S_v (neut/cm ³ -sec)				
	0.4 Mev	0.6 Mev	1.5 Mev	2.0 Mev	Total
1	1.67×10^1	3.30×10^7	2.85×10^3	2.97×10^7	6.27×10^7
3	1.67×10^1	1.93×10^7	2.86×10^3	1.74×10^7	3.67×10^7
5	1.67×10^1	1.13×10^7	2.84×10^3	1.01×10^7	2.14×10^7
7	1.67×10^1	6.59×10^6	2.83×10^3	6.11×10^6	1.27×10^7
10	1.66×10^1	2.95×10^6	2.83×10^3	2.65×10^6	5.60×10^6
30	1.64×10^1	1.38×10^4	2.80×10^3	1.23×10^4	2.89×10^4
50	1.62×10^1	1.75×10^2	2.78×10^3	5.69×10^1	3.01×10^3
70	1.60×10^1	1.12×10^2	2.73×10^3	2.64×10^{-1}	2.86×10^3
100	1.57×10^1	1.12×10^2	2.68×10^3		2.80×10^3
310	1.37×10^1	1.10×10^2	2.34×10^3		2.46×10^3
510	1.20×10^1	1.07×10^2	2.06×10^3		2.17×10^3
710	1.06×10^1	1.05×10^2	1.81×10^3		1.93×10^3
1010	8.70×10^0	1.02×10^2	1.49×10^3		1.60×10^3
3110	2.25×10^0	8.30×10^1	3.88×10^2		4.73×10^2

* $\alpha = 0^\circ$ End Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 40

NEUTRON-INDUCED ACTIVATION IN CARBON STEEL (SPECIAL COMPOSITION)
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - S_v (rev/cm ³ -sec)				
	0.4 Mev	0.5 Mev	1.5 Mev	2.0 Mev	Total
1	1.11×10^3	3.43×10^7	8.57×10^3	3.10×10^7	6.53×10^7
3	2.04×10^2	2.01×10^7	6.69×10^3	1.81×10^7	3.82×10^7
5	1.54×10^2	1.17×10^7	5.92×10^3	1.05×10^7	2.23×10^7
7	1.39×10^2	6.87×10^6	5.40×10^3	6.18×10^6	1.30×10^7
10	1.27×10^2	3.07×10^6	4.84×10^3	2.76×10^6	5.84×10^6
30	1.03×10^2	1.44×10^4	5.27×10^3	1.28×10^4	3.25×10^4
50	8.99×10^1	1.94×10^2	2.96×10^3	6.20×10^1	3.31×10^3
70	7.92×10^1	1.28×10^2	2.78×10^3	1.42×10^0	2.99×10^3
100	2.49×10^1	1.21×10^2	2.67×10^3	1.35×10^{-1}	2.82×10^3
310	3.25×10^1	1.24×10^2	2.33×10^3	1.47×10^{-1}	2.49×10^3
510	2.48×10^1	1.21×10^2	2.05×10^3	1.35×10^{-1}	2.19×10^3
710	2.07×10^1	1.17×10^2	1.80×10^3	1.25×10^{-1}	1.94×10^3
1010	1.61×10^1	1.13×10^2	1.49×10^3	1.10×10^{-1}	1.62×10^3
3110	3.21×10^0	8.73×10^1	3.85×10^2	4.69×10^{-2}	4.76×10^2

* $\alpha = 0^\circ$ End Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 41

NEUTRON-INDUCED ACTIVATION IN 321 SS
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - S (mev/cm ³ - sec)				
	0.4	0.8	1.5	2.0	Total
0	2.29 x 10 ⁶	7.21 x 10 ⁷	7.73 x 10 ⁶	6.49 x 10 ⁷	1.47 x 10 ⁸
1	7.28 x 10 ⁴	5.50 x 10 ⁷	1.83 x 10 ⁵	4.95 x 10 ⁷	1.05 x 10 ⁸
3	1.03 x 10 ⁴	3.21 x 10 ⁷	1.00 x 10 ⁵	2.89 x 10 ⁷	6.11 x 10 ⁷
5	8.34 x 10 ³	1.88 x 10 ⁷	6.19 x 10 ⁴	1.69 x 10 ⁷	3.58 x 10 ⁷
7	7.30 x 10 ³	1.10 x 10 ⁷	3.93 x 10 ⁴	9.87 x 10 ⁶	2.09 x 10 ⁷
10	6.46 x 10 ³	4.90 x 10 ⁶	2.16 x 10 ⁴	4.41 x 10 ⁶	9.34 x 10 ⁶
24	5.55 x 10 ³	1.15 x 10 ⁵	6.01 x 10 ³	1.02 x 10 ⁵	2.29 x 10 ⁵
30	5.42 x 10 ³	2.42 x 10 ⁴	5.21 x 10 ³	2.05 x 10 ⁴	5.53 x 10 ⁴
50	5.08 x 10 ³	1.60 x 10 ³	4.19 x 10 ³	1.09 x 10 ²	1.10 x 10 ⁴
70	4.78 x 10 ³	1.48 x 10 ³	3.85 x 10 ³	1.44 x 10 ¹	1.01 x 10 ⁴
100	4.42 x 10 ³	1.46 x 10 ³	3.68 x 10 ³	1.38 x 10 ¹	9.59 x 10 ³
310	3.10 x 10 ³	1.35 x 10 ³	3.42 x 10 ³	1.26 x 10 ¹	7.88 x 10 ³
510	2.47 x 10 ³	1.25 x 10 ³	3.22 x 10 ³	1.16 x 10 ¹	6.95 x 10 ³
710	2.00 x 10 ³	1.16 x 10 ³	3.05 x 10 ³	1.07 x 10 ¹	6.22 x 10 ³
1010	1.47 x 10 ³	1.03 x 10 ³	2.83 x 10 ³	9.51 x 10 ⁰	5.34 x 10 ³
3110	1.77 x 10 ²	4.64 x 10 ²	2.03 x 10 ³	4.04 x 10 ⁰	2.67 x 10 ³
6000	1.71 x 10 ¹	1.68 x 10 ²	1.73 x 10 ³	1.25 x 10 ⁰	1.92 x 10 ³
8890	6.86 x 10 ⁰	7.04 x 10 ¹	1.63 x 10 ³	3.85 x 10 ⁻¹	1.70 x 10 ³

370 days

*α= 0° End-Leakage Spectrum:

<u>n/cm²-sec</u>	<u>Energy Range</u>
1.00 x 10 ¹⁰	E _n > .82 Mev
3.04 x 10 ¹⁰	5 kev < E _n < 0.82 Mev
3.24 x 10 ¹⁰	0.625 ev < E _n < 5 kev
2.61 x 10 ¹⁰	E _n < .625 ev

TABLE 42

NEUTRON-INDUCED ACTIVATION IN 304 STAINLESS STEEL
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - Sv (rev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	5.28×10^4	5.51×10^7	1.52×10^5	4.96×10^7	1.04×10^8
3	8.90×10^3	3.22×10^7	9.02×10^4	2.89×10^7	6.13×10^7
5	7.33×10^3	1.88×10^7	5.58×10^4	1.69×10^7	3.58×10^7
7	6.41×10^3	1.10×10^7	3.55×10^4	9.88×10^6	2.10×10^7
10	5.68×10^3	4.91×10^6	2.07×10^4	4.42×10^6	9.36×10^6
30	5.1×10^3	2.41×10^4	8.28×10^3	2.05×10^4	5.79×10^4
50	4.89×10^3	1.48×10^3	8.20×10^3	1.08×10^2	1.47×10^4
70	4.80×10^3	1.36×10^3	8.14×10^3	1.31×10^1	1.43×10^4
100	4.66×10^3	1.34×10^3	7.92×10^3	1.25×10^1	1.39×10^4
310	3.82×10^3	1.24×10^3	7.86×10^3	1.15×10^1	1.29×10^4
510	3.17×10^3	1.14×10^3	7.20×10^3	1.06×10^1	1.15×10^4
710	2.64×10^3	1.06×10^3	6.83×10^3	9.58×10^0	1.05×10^3
1010	2.02×10^3	9.44×10^2	5.63×10^3	8.64×10^0	3.60×10^3
3110	4.21×10^2	4.30×10^2	4.02×10^3	3.67×10^0	4.94×10^3

* $\alpha = 0^\circ$ End-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 43

NEUTRON-INDUCED ACTIVATION IN ^{347}SS
 FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - S (Mev/cm ³ - sec)				
	0.4	0.8	1.3	2.0	Total
0	2.19×10^6	7.19×10^7	7.40×10^6	6.47×10^7	1.46×10^8
1	6.18×10^4	5.50×10^7	2.62×10^5	4.95×10^7	1.05×10^8
3	1.72×10^4	3.21×10^7	1.82×10^5	2.89×10^7	6.12×10^7
5	1.51×10^4	1.88×10^7	1.37×10^5	1.69×10^7	3.59×10^7
7	1.39×10^4	1.10×10^7	1.11×10^5	9.87×10^6	2.10×10^7
10	1.30×10^4	4.90×10^6	9.09×10^4	4.41×10^6	9.41×10^6
24	1.22×10^4	1.16×10^4	7.50×10^4	1.02×10^5	2.01×10^5
30	1.21×10^4	2.45×10^4	7.46×10^4	2.05×10^4	1.32×10^5
50	1.20×10^4	1.85×10^3	7.41×10^4	1.12×10^2	8.81×10^4
70	1.18×10^4	1.74×10^3	7.37×10^4	1.69×10^1	8.73×10^4
100	1.17×10^4	1.72×10^3	7.32×10^4	1.63×10^1	8.66×10^4
310	1.05×10^4	1.58×10^3	6.94×10^4	1.49×10^1	8.15×10^4
510	9.54×10^3	1.46×10^3	6.59×10^4	1.38×10^1	7.69×10^4
710	8.71×10^3	1.35×10^3	6.27×10^4	1.27×10^1	7.28×10^4
1010	7.67×10^3	1.20×10^3	5.81×10^4	1.12×10^1	6.70×10^4
3110	3.80×10^3	5.38×10^2	3.47×10^4	4.78×10^0	3.90×10^4
6000	1.77×10^3	1.90×10^2	1.75×10^1	1.48×10^0	1.95×10^4
8890	8.54×10^2	7.69×10^1	9.27×10^3	4.55×10^{-1}	1.02×10^4
370 days					

* $\alpha = 0^\circ$ End-Leakage Spectrum: $\frac{n}{\text{cm}^2 \cdot \text{sec}}$ Energy Range

1.00×10^{10}	$E_n > 0.82$
3.04×10^{10}	$5 \text{ kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 44

NEUTRON-INDUCED ACTIVATION IN NUCLEAR GRADE 1100 ALUMINUM
FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - Sv(Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
0	7.58×10^3	1.65×10^6	6.36×10^5	7.03×10^8	7.05×10^8
1	2.61×10^2	4.93×10^5	2.10×10^4	4.49×10^5	9.63×10^5
3	1.28×10^2	2.78×10^5	1.06×10^4	2.70×10^5	5.59×10^5
5	1.20×10^2	1.62×10^5	9.68×10^3	1.64×10^5	3.36×10^5
7	1.09×10^2	9.49×10^4	8.80×10^3	1.02×10^5	2.06×10^5
10	9.37×10^1	4.24×10^4	7.64×10^3	5.23×10^4	1.02×10^5
24	4.62×10^1	9.87×10^2	3.97×10^3	8.28×10^3	1.33×10^4
30	3.42×10^1	1.98×10^2	3.01×10^3	5.78×10^3	9.02×10^3
50	1.26×10^1	2.20×10^0	1.19×10^3	2.22×10^3	3.42×10^3
70	4.65×10^0	1.28×10^0	4.80×10^2	8.83×10^2	1.37×10^3
100	1.08×10^0	1.27×10^0	1.32×10^2	2.21×10^2	3.55×10^2
310	5.83×10^{-2}	1.19×10^0	1.66×10^1	2.14×10^{-2}	1.79×10^1
510	5.22×10^{-2}	1.12×10^0	1.55×10^1	7.33×10^{-3}	1.67×10^1
710	4.68×10^{-2}	1.05×10^0	1.46×10^1	6.76×10^{-3}	1.57×10^1
1010	4.00×10^{-2}	9.64×10^{-1}	1.34×10^1	5.98×10^{-3}	1.44×10^1
3110	1.54×10^{-2}	5.49×10^{-1}	8.91×10^0	2.55×10^{-3}	9.48×10^0
6000	6.74×10^{-3}	2.98×10^{-1}	7.09×10^0	7.86×10^{-4}	7.40×10^0
8890	4.23×10^{-3}	1.89×10^{-1}	6.36×10^0	2.42×10^{-4}	6.55×10^0

370 days

* $\alpha = 0^\circ$ End-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > .82 \text{ Mev}$
3.04×10^{10}	$5 \text{ kev} < E_n < .82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
2.61×10^{10}	$E_n < .625 \text{ ev}$

TABLE 45

NEUTRON-INDUCED ACTIVATION IN ALUMINUM (2219)
 FOLLOWING 30 MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - $S_v(\text{Mev/cm}^3\text{-sec})$				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	7.36×10^3	2.88×10^6	4.91×10^4	2.59×10^6	5.53×10^6
3	3.24×10^2	1.67×10^6	3.51×10^4	1.52×10^6	3.23×10^6
5	3.00×10^2	9.75×10^5	3.15×10^4	8.96×10^5	1.90×10^6
7	2.73×10^2	3.86×10^5	2.85×10^4	5.28×10^5	9.43×10^5
10	2.34×10^2	2.55×10^5	2.44×10^4	2.43×10^5	5.23×10^5
30	8.56×10^1	1.18×10^3	8.76×10^3	6.46×10^3	1.65×10^4
50	3.13×10^1	7.84×10^0	3.39×10^3	2.15×10^3	5.58×10^3
70	1.15×10^1	2.37×10^0	1.42×10^3	8.52×10^2	2.29×10^3
100	2.49×10^0	2.39×10^0	5.67×10^2	2.12×10^2	7.84×10^2
310	4.14×10^{-2}	2.13×10^0	3.27×10^2	3.28×10^{-2}	3.29×10^2
510	3.07×10^{-2}	1.97×10^0	3.25×10^2	1.83×10^{-2}	3.27×10^2
710	3.65×10^{-2}	1.82×10^0	3.24×10^2	1.69×10^{-2}	3.26×10^2
1010	3.36×10^{-2}	1.62×10^0	3.22×10^2	1.50×10^{-2}	3.24×10^2
3110	2.14×10^{-2}	7.35×10^{-1}	3.10×10^2	6.36×10^{-3}	3.11×10^2

* $\alpha = 0^0$ End-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 46

NEUTRON-INDUCED ACTIVATION IN NUCLEAR GRADE 2219 ALUMINUM
FOLLOWING 30-MIN OPERATION AT 5000 MW WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, Hr	Volumetric Gamma Ray Source Strength - Sv(Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
0	8.96×10^3	6.24×10^6	1.25×10^7	6.87×10^8	7.06×10^8
1	2.73×10^2	3.98×10^6	5.37×10^4	3.59×10^6	7.62×10^6
3	1.34×10^2	2.32×10^6	3.94×10^4	2.10×10^6	4.46×10^6
5	1.25×10^2	1.36×10^6	3.55×10^4	1.24×10^6	2.64×10^6
7	1.14×10^2	7.92×10^5	3.20×10^4	7.28×10^5	1.55×10^6
10	9.76×10^1	3.54×10^5	2.73×10^4	3.32×10^5	7.13×10^5
24	4.82×10^1	8.26×10^3	1.32×10^4	1.46×10^4	3.61×10^4
30	3.56×10^1	1.67×10^3	9.65×10^3	6.92×10^3	1.83×10^4
50	1.31×10^1	2.38×10^1	3.44×10^3	2.17×10^3	5.65×10^3
70	4.80×10^0	1.03×10^1	1.23×10^3	8.58×10^2	2.10×10^3
100	1.09×10^0	6.90×10^0	2.75×10^2	2.14×10^2	4.97×10^2
310	2.57×10^{-2}	5.65×10^0	1.15×10^1	2.24×10^{-2}	1.72×10^1
510	2.36×10^{-2}	5.69×10^0	1.11×10^1	7.76×10^{-3}	1.68×10^1
710	2.17×10^{-2}	5.64×10^0	1.07×10^1	7.06×10^{-3}	1.64×10^1
1010	1.94×10^{-2}	5.46×10^0	1.03×10^1	6.24×10^{-3}	1.58×10^1
3110	1.03×10^{-2}	3.13×10^0	8.45×10^0	2.66×10^{-3}	1.16×10^1
6000	6.12×10^{-3}	1.05×10^0	7.38×10^0	8.19×10^{-4}	8.44×10^0
8890 (370 days)	4.27×10^{-3}	3.40×10^{-1}	6.76×10^0	2.53×10^{-4}	7.10×10^0

* $\alpha = 0^\circ$ End-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 47

NEUTRON-INDUCED ACTIVATION IN 5083 ALUMINUM
FOLLOWING 30-MIN OPERATION AT 5000 MW WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, Hr	Volumetric Gamma Ray Source Strength - Sv(Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	2.84×10^2	9.53×10^6	2.20×10^4	8.58×10^6	1.81×10^7
3	1.48×10^2	5.56×10^6	1.05×10^4	5.02×10^6	1.06×10^7
5	1.40×10^2	3.25×10^6	9.58×10^3	2.94×10^6	6.20×10^6
7	1.29×10^2	1.90×10^6	8.72×10^3	1.72×10^6	3.63×10^6
10	1.13×10^2	8.48×10^5	7.58×10^3	7.77×10^5	1.63×10^6
24	6.56×10^1	1.97×10^4	3.96×10^3	2.54×10^4	4.91×10^4
30	5.35×10^1	3.93×10^3	3.00×10^3	9.33×10^3	1.63×10^4
50	3.14×10^1	1.93×10^1	1.19×10^3	2.31×10^3	3.56×10^3
70	2.31×10^1	1.16×10^0	4.78×10^2	9.12×10^2	1.41×10^3
100	1.90×10^1	1.06×10^0	1.28×10^2	2.28×10^2	3.76×10^2
310	1.44×10^1	9.86×10^{-1}	1.17×10^1	2.19×10^{-2}	2.72×10^1
510	1.17×10^1	9.19×10^{-1}	1.12×10^1	7.33×10^{-3}	2.40×10^1
710	9.54×10^0	8.56×10^{-1}	1.08×10^1	6.76×10^{-3}	2.12×10^1
1010	6.99×10^0	7.71×10^{-1}	1.03×10^1	5.98×10^{-3}	1.80×10^1
3110	7.97×10^{-1}	3.87×10^{-1}	8.10×10^0	2.55×10^{-3}	9.29×10^0
6000	4.51×10^{-2}	1.74×10^{-1}	6.96×10^0	7.86×10^{-4}	7.18×10^0
8890 (370 days)	6.06×10^{-3}	9.40×10^{-2}	6.34×10^0	2.42×10^{-4}	6.44×10^0

* $\alpha = 0^\circ$ End-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 48

NEUTRON-INDUCED ACTIVATION IN NUCLEAR GRADE 5454 ALUMINUM
FOLLOWING 30-MIN OPERATION AT 5000 MW WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, Hr	Volumetric Gamma Ray Source Strength - Sv(Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
0	7.51×10^3	1.34×10^7	6.56×10^5	6.79×10^8	6.93×10^8
1	2.74×10^2	9.43×10^6	2.11×10^4	8.48×10^6	1.79×10^7
3	1.42×10^2	5.50×10^6	1.03×10^4	4.96×10^6	1.05×10^7
5	1.34×10^2	3.21×10^6	9.35×10^3	2.91×10^6	6.13×10^6
7	1.23×10^2	1.88×10^6	8.51×10^3	1.70×10^6	3.59×10^6
10	1.08×10^2	8.38×10^5	7.39×10^3	7.68×10^5	1.61×10^6
24	6.10×10^1	1.95×10^4	3.86×10^3	2.50×10^4	4.84×10^4
30	4.91×10^1	3.89×10^3	2.92×10^3	9.14×10^3	1.60×10^4
50	2.74×10^1	1.91×10^1	1.16×10^3	2.26×10^3	3.47×10^3
70	1.92×10^1	1.12×10^0	4.66×10^2	8.89×10^2	1.38×10^3
100	1.52×10^1	1.03×10^0	1.25×10^2	2.22×10^2	3.63×10^2
310	1.14×10^1	9.53×10^{-1}	1.15×10^1	2.14×10^{-2}	2.39×10^1
510	9.29×10^0	8.87×10^{-1}	1.11×10^1	7.25×10^{-3}	2.13×10^1
710	7.55×10^0	8.27×10^{-1}	1.07×10^1	6.68×10^{-3}	1.91×10^1
1010	5.53×10^0	7.45×10^{-1}	1.01×10^1	5.92×10^{-3}	1.64×10^1
3110	6.33×10^{-1}	3.74×10^{-1}	8.00×10^0	2.52×10^{-3}	9.01×10^0
6000	3.68×10^{-2}	1.68×10^{-1}	6.88×10^0	7.77×10^{-4}	7.09×10^0
8890	5.61×10^{-3}	9.18×10^{-2}	6.27×10^0	2.40×10^{-4}	6.37×10^0

(370 days)

* $\alpha = 0^\circ$ End-Leakage Spectrum:	$n/\text{cm}^2\text{-sec}$	Energy Range
	1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
	3.04×10^{10}	$5 \text{ kev} < E_n < 0.82 \text{ Mev}$
	3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
	2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 49

NEUTRON-INDUCED ACTIVATION IN NUCLEAR GRADE 6061 ALUMINUM
FOLLOWING 30-MIN OPERATION AT 5000 MW WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, Hr	Volumetric Gamma Ray Source Strength - Sv(Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
0	1.08×10^4	2.93×10^6	7.76×10^5	6.90×10^8	6.93×10^8
1	2.92×10^2	1.44×10^6	2.22×10^4	1.30×10^6	2.77×10^6
3	1.56×10^2	8.34×10^5	1.15×10^4	7.70×10^5	1.62×10^6
5	1.48×10^2	4.87×10^5	1.04×10^4	4.56×10^5	9.54×10^5
7	1.37×10^2	2.85×10^5	9.49×10^3	2.72×10^5	5.67×10^5
10	1.21×10^2	1.27×10^5	8.23×10^3	1.28×10^5	2.64×10^5
24	7.34×10^1	2.96×10^3	4.25×10^3	1.01×10^4	1.74×10^4
30	6.12×10^1	5.91×10^2	3.20×10^3	6.14×10^3	1.00×10^4
50	3.90×10^1	3.93×10^0	1.26×10^3	2.23×10^3	3.53×10^3
70	3.05×10^1	1.20×10^0	5.01×10^2	8.84×10^2	1.42×10^3
100	2.62×10^1	1.18×10^0	1.34×10^2	2.21×10^2	3.83×10^2
310	2.02×10^1	1.10×10^0	1.42×10^1	2.14×10^{-2}	3.55×10^1
510	1.64×10^1	1.03×10^0	1.34×10^1	7.33×10^{-3}	3.09×10^1
710	1.34×10^1	9.62×10^{-1}	1.27×10^1	6.76×10^{-3}	2.70×10^1
1010	9.79×10^0	8.74×10^{-1}	1.18×10^1	5.98×10^{-3}	2.25×10^1
3110	1.11×10^0	4.71×10^{-1}	8.52×10^0	2.55×10^{-3}	1.01×10^1
6000	6.10×10^{-2}	2.36×10^{-1}	7.03×10^0	7.86×10^{-4}	7.33×10^0
8890 (370 days)	6.89×10^{-3}	1.41×10^{-1}	6.36×10^0	2.42×10^{-4}	6.51×10^0

* $\alpha = 0^\circ$ End-Leakage Spectrum:

$n/cm^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 50

NEUTRON INDUCED ACTIVATION IN ALUMINUM 6061 FOLLOWING
30-MIN OPERATION AT 5000 Mw WITH END-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - S_v (Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	7.69×10^3	1.45×10^6	2.21×10^4	1.31×10^6	2.79×10^6
3	8.11×10^2	8.36×10^5	1.07×10^4	7.72×10^5	1.62×10^6
5	7.64×10^2	4.88×10^5	1.03×10^4	4.57×10^5	9.56×10^5
7	6.99×10^2	2.85×10^5	9.34×10^3	2.73×10^5	5.68×10^5
10	6.05×10^2	1.27×10^5	8.13×10^3	1.29×10^5	2.65×10^5
30	2.02×10^2	5.92×10^2	3.32×10^3	6.04×10^3	1.02×10^4
50	9.73×10^1	5.27×10^0	1.53×10^3	2.18×10^3	3.81×10^3
70	4.73×10^1	2.52×10^0	8.05×10^2	8.67×10^2	1.72×10^3
100	2.44×10^1	2.49×10^2	4.54×10^2	2.16×10^2	6.97×10^2
310	1.45×10^1	2.30×10^0	3.37×10^2	3.30×10^{-2}	3.54×10^2
510	1.18×10^1	2.14×10^0	3.36×10^2	1.83×10^{-2}	3.51×10^2
710	9.58×10^0	1.99×10^0	3.34×10^2	1.69×10^{-2}	3.45×10^2
1010	7.02×10^0	1.78×10^0	3.31×10^2	1.50×10^{-2}	3.40×10^2
3110	8.12×10^{-1}	8.56×10^{-1}	3.16×10^2	6.36×10^{-3}	3.17×10^2

* $\alpha = 0^\circ$ End-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
3.04×10^{10}	$5 \text{ Kev} < E_n < 0.82 \text{ Mev}$
3.24×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.61×10^{10}	$E_n < 0.625 \text{ ev}$

Figure 67 gives similar data for the steels investigated, normalized to the activity of A-36 steel 1 day after shutdown.

Figure 68 shows the decay for Military Specification Aluminums 2219 and 6061 following activation by the end-leakage spectrum. Figure 69 presents similar data for Nuclear Grade aluminums. The values are normalized to Aluminum 1100 activity 1 day after shutdown.

3. Activation from $\alpha = 90^\circ$ Side-Leakage Spectrum

Tables 51 and 52, respectively, give the resultant activation in Bingham and Type 05 concretes after exposure to the $\alpha = 90^\circ$ side-leakage spectrum. The side-leakage spectrum is shown at the bottom of each of these tables.

TABLE 51

NEUTRON-INDUCED ACTIVATION IN CONCRETE (BINGHAM)
FOLLOWING 30-MIN OPERATION AT 5000 MW WITH SIDE-LEAKAGE SPECTRUM*

Time After Shutdown, Hr	Volumetric Gamma Ray Source Strength - Sv(Mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	7.24×10^1	1.55×10^3	4.95×10^5	9.71×10^5	1.47×10^6
3	5.15×10^1	1.46×10^2	4.51×10^5	8.80×10^5	1.33×10^6
5	3.86×10^1	9.20×10^1	4.10×10^5	8.02×10^5	1.21×10^6
7	2.96×10^1	6.14×10^1	3.73×10^5	7.31×10^5	1.10×10^6
10	2.11×10^1	3.34×10^1	3.24×10^5	6.36×10^5	9.60×10^5
30	7.60×10^0	5.68×10^0	1.29×10^5	2.52×10^5	3.81×10^5
70	10^1	10^1	2.01×10^4	3.98×10^4	6.00×10^4
100			5.06×10^3	9.94×10^3	1.50×10^4
310			4.03×10^1	10^1	5.05×10^1
510			3.64×10^1		4.53×10^1
710			3.36×10^1		4.18×10^1
1010			2.99×10^1		3.74×10^1
3110			1.43×10^1		1.83×10^1

* $\alpha = 90^\circ$ Side-Leakage Spectrum:

$n/\text{cm}^2\text{-sec}$	Energy Range
1.00×10^{10}	$E_n > 0.82 \text{ Mev}$
2.64×10^{10}	$5 \text{ Kev} < E_n < 0.82$
2.96×10^{10}	$0.625 \text{ ev} < E_n < 5 \text{ Kev}$
2.72×10^{10}	$E_n < 0.625 \text{ ev}$

TABLE 52

NEUTRON-INDUCED ACTIVATION IN CONCRETE (05)
 FOLLOWING 30-MIN OPERATION AT 5000 Mw WITH SIDE-LEAKAGE SPECTRUM*

Time After Shutdown, hr	Volumetric Gamma Ray Source Strength - Sv(mev/cm ³ -sec)				
	0.4 Mev	0.8 Mev	1.5 Mev	2.0 Mev	Total
1	1.11 x 10 ¹	1.63 x 10 ⁷	3.60 x 10 ⁵	1.53 x 10 ⁷	3.20 x 10 ⁷
3	<10 ¹	9.50 x 10 ⁶	3.28 x 10 ⁵	9.20 x 10 ⁶	1.90 x 10 ⁷
5		5.55 x 10 ⁶	2.99 x 10 ⁵	5.60 x 10 ⁶	1.15 x 10 ⁷
7		3.24 x 10 ⁶	2.72 x 10 ⁵	3.45 x 10 ⁶	6.96 x 10 ⁶
10		1.45 x 10 ⁶	2.46 x 10 ⁵	1.77 x 10 ⁶	3.46 x 10 ⁶
30		6.73 x 10 ³	9.37 x 10 ⁴	1.91 x 10 ⁵	2.91 x 10 ⁵
50		3.35 x 10 ¹	3.71 x 10 ⁴	7.34 x 10 ⁴	1.10 x 10 ⁵
70		<10 ¹	1.47 x 10 ⁴	2.92 x 10 ⁴	4.39 x 10 ⁴
100			3.70 x 10 ³	7.28 x 10 ³	1.10 x 10 ⁴
310			3.98 x 10 ¹	<10 ¹	4.27 x 10 ¹
510			3.29 x 10 ¹		3.50 x 10 ¹
710			3.00 x 10 ¹		3.20 x 10 ¹
1010			2.47 x 10 ¹		2.65 x 10 ¹
3110			6.43 x 10 ⁰		7.82 x 10 ⁰

* α = 90° Side-Leakage Spectrum:

<u>n/cm²-sec</u>	<u>Energy Range</u>
1.00 x 10 ¹⁰	E _n < 0.82 Mev
2.64 x 10 ¹⁰	5 Kev < E _n < 0.82
2.96 x 10 ¹⁰	0.625 ev < E _n 5 Kev
2.72 x 10 ¹⁰	E _n < 0.625 ev

X. RADIATION LEVELS AFTER NUCLEAR ACCIDENTS

A. GENERAL CONSIDERATIONS

The Design Criteria document⁴ specifies the following accident conditions for a reactor power excursion or core meltdown:

1. A reactor power excursion resulting in reactor pressure vessel rupture, nozzle loss, and scattering of core pieces to the exhaust duct, tank shield, engine compartment, and surrounding area.
2. Core meltdown assuming 20% core vaporization and resulting damage to the test position, including rupture of the run tank.

In the earlier study in support of site layout activities¹, data for E/STS 2-3 were obtained by scaling the available data from the Kiwi-TNT programmed excursion^{19,20,21,22} on the Phoebus 1A-321 core-meltdown incident^{23,24}. This was recognized as only an interim solution and was so reported. No new data are currently available in this safety area; therefore, no additional nuclear analyses of this type are included in this report. Some of the earlier reported data¹ are included for convenience. Both the Kiwi TNT transient nuclear test and the Phoebus 1A-321 experimental results have been scaled up in direct proportion from the fission product source strengths of these tests to those 1 hr after a 5000 Mw reactor operation for 30 min and 20,000 Mw operation for 45 min. The differences between these values following the instantaneous fissions in Kiwi TNT are accounted for as shown in Figure 70. Ground depositions from the Kiwi TNT experiment in the downwind directions are evaluated in addition to the crosswind conditions discussed in the Budgetary Study³. The earlier work was based on rather limited preliminary data^{19,23}, but the present work is based on more recent information^{20,21,22,24} in addition to the initial information.

Figure 70 - Gamma Source Decay from Kiwi TNT, Phoebus 1A-321 and E/STS 2-3 10,000 Mw and 5000 Mw Inventory

As a means of obtaining an early assessment of control-center and access-tunnel shielding, some calculations have been made with the assumption that a small portion of the core will be accidentally deposited above these areas. Calculations related to this condition are discussed in this section.

B. EXTRAPOLATION OF KIWI TNT DATA TO E/STS 2-3

The Kiwi Transient Nuclear Test was a deliberate nuclear destruction experiment of a prototype nuclear rocket engine. A nuclear transient was induced in a virgin reactor by the rapid insertion of reactivity accomplished by modification of the control drums, which enabled rapid rotation not anticipated in E/STS 2-3. The transient produced a 3.1×10^{20} fission burst with two-thirds of the fission product activity being transported away by the effluent cloud. The data have been scaled up to simulate E/STS 2-3 conditions of 10,000 Mw operation for 45 min or 5000 Mw operation for 30 min.

The TNT dose rate survey isodose rate contours at 1 hr after the event are presented in Figures 19 and 20 of LA-3519-MS²². Based on these data, the 205° direction was designated as downwind and a 90° direction was selected as crosswind. Using these isodose rate contours, plots of the dose rate as a function of distance are obtained. If these data are then adjusted upward consistent with E/STS 2-3 fission-product inventories given in Figure 70, the dose-rate distribution about E/STS 2-3 following a 5000 Mw 30 min run at 1, 5, and 24 hr is shown in Figure 71. Comparable data for a 45 min 10,000 Mw run are shown in Figure 72. These Kiwi TNT gamma-source decay rates are based on adjustment of the scaled up sources by the Perkins and King¹⁸ gamma ray spectrum and decay rates following instantaneous fission.

Comparisons between the results of the Kiwi TNT-type accident conditions and the Phoebus 1A-321 conditions (both scaled to E/STS 2-3 integrated powers) together with comparisons of bare-core dose rate vs distance relationships are made in Section X.D of this report. Data relative to the TNT ground-deposition patterns, in addition to these comparisons, are included.

Figure 71 - Dose Rate vs Distance Scaled From a Kiwi TNT Type Transient for 1-Hour, 5-Hour, and 24-Hour Decay Following 5000 Mw E/STS 2-3 Operation for 30 Minutes

Figure 72 - Dose Rate vs Distance Scaled from a Kiwi TNT Type Transient for 1-Hour, 5-Hour, and 24-Hour Decay Following 10,000 Mw E/STS 2-3 Operation for 45 Minutes

C. EXTRAPOLATION OF PHOEBUS 1A-321 DATA TO E/STS 2-3

The Phoebus 1A-321, Experimental Plan 4, full-power reactor test was conducted at a power level of approximately 1050 Mw for 10.8 min. Near the termination of the run, a loss of coolant occurred that resulted in the rapid overheating of the core and expulsion of 5 to 8% of the core through the nozzle. This test represents a loss-of-coolant accident for which considerable experimental data were obtained and which is similar to an incident postulated in the E/STS 2-3 criteria⁴.

The scale-up of the Phoebus 1A-321 data to E/STS 2-3 conditions can be accomplished by adjusting the deposition dose rates reported for the estimated time of cloud passage to dose rates at 1 hr after the test, and scaling the data to E/STS 2-3 as was described for Kiwi TNT. Such scaling adjusts dose rates for differences in fission-product inventory between Phoebus 1A-321 and E/STS 2-3 conditions.

The deposition dose rates corrected to the estimated time of cloud passage are reported in Figure 26 of Reference 24. It was reported²⁴ that winds at run time were southerly at 6 to 15 knots through 10,000 ft MSL. On this basis, a cloud velocity of 10 mph was assumed to modify Phoebus 1A-321 deposition dose rates to those following 1 hr decay. Table 53 depicts the adjustment of Phoebus 1A-321 deposition data at the time of cloud passage to 1 hr after the test, using fission-product gamma decay data from Figure 28 in Reference 24.

The downwind direction was established at approximately the 310° direction and the crosswind direction was selected at the 270° direction. Due to the uncertainty associated with attempts to establish the time of cloud passage in the crosswind direction, since the cloud did not travel in that direction, the crosswind-deposition-dose rate was determined from the isodeposition contour shown in Figure 20 of Reference 24. It was assumed that the ratio of crosswind to downwind dose rates would be unaffected by the decay

TABLE 53

ADJUSTMENT OF PHOEBUS 1A-321 DOWNWIND
DEPOSITION DOSE RATED TO 1-HOUR DECAY

<u>Distance from Test Point, ft</u>	<u>Estimated Time of Cloud Passage, min</u>	<u>Deposition Dose Rate at Time of Cloud Passage, R/hr</u>	<u>Deposition Dose Rate After One Hour Decay R/hr</u>
2000	2.5	1.7×10^1	1.7×10^0
4000	5.0	5.5×10^1	7.62×10^0
8000	9.0	5.4×10^1	1.06×10^1
12000	13.0	2.4×10^0	5.73×10^{-1}
16000	18.0	9.2×10^{-2}	2.77×10^{-2}

from the time of cloud passage to the 1 hr decay time (i.e., same radioactive decay rate in the crosswind and downwind directions).

The Phoebus 1A-321 dose rates were then adjusted upward to represent the E/STS 2-3 fission product inventory based on variations in the intensity for appropriate operation and decay times and power levels. The resultant E/STS 2-3 dose rates, scaled up from Phoebus 1A-321 data, are presented in Figures 73 and 74, for 5000 Mw for 30 min and 10,000 Mw operation for 45 min, relative to 1, 5, and 24-hr decay for both downwind and crosswind conditions in each case.

D. COMPARISON OF RADIATION LEVELS FROM A BARE SHUTDOWN ENGINE WITH SCALED UP KIWI TNT AND PHOEBUS 1A-321 DATA

Figure 36 compares radiation levels, as a function of distance from the test stand, for a bare shutdown engine and for a scaleup of the Kiwi TNT and Phoebus 1A-321 downwind and crosswind measurements. These data are based on 1-hr decay following 30 min of operation at 5000 Mw. These conditions following 45 min of operation at 10,000 Mw can be determined by multiplying all of the curves in Figure 75 by a factor of three. As has been indicated, an additional factor of five is required⁴, in either case, to compare these data with the access criteria.

Use of the downwind data results in extremely high dose rates compared with the crosswind or bare engine values. A bare engine is roughly equivalent to the crosswind conditions at some locations.

The relative shapes of the dose vs distance curves are significantly different for the various cases. Since exposure values are predicated on 1-hr after an accident, in these cases wind direction cannot be predetermined with a great deal of accuracy. However, if the decay associated with times greater than 1 hr is considered, after an excursion (such as the Kiwi TNT) or loss of shield water accident (such as the bare engine), then the dose rates for these

Figure 73 - Dose Rate vs Distance Scaled from a Phoebus 1A-321 Type Accident for 1-Hour, 5-Hour, and 24-Hour Decay Following 5000 Mw E/STS 2-3 Operation for 30 Minutes

Figure 74 - Dose Rate vs Distance Scaled from a Phoebe 1A-321 Type Accident for 1-Hour, 5-Hour and 24-Hour Decay Following 10,000 Mw E/STS 2-3 Operation for 45 Minutes

Figure 75 - Gamma Ray Dose Rate Comparison of Bare Core with Scaled Up Kiwi TNT and Phoebus 1A-321 Data, 1 Hour After 30 Minutes Operation at 5000 Mw

cases shown in Figure 75 would be lower by an amount roughly equivalent to that shown in Figure 70. The Phoebus 1A-321 type of accident may have a controlled favorable wind direction because the loss-of-coolant is assumed to occur during a full-power run (which would not be conducted without favorable wind conditions). However, sudden wind reversals could occur which would increase the importance of the downwind data for the core-meltdown, scale-up case as in the Phoebus 1A-321 release.

Relative to the Kiwi TNT incident, approximately 66% of the fission products created were estimated to be transported away from the test point by the effluent cloud, and the remaining 33% were distributed in the form of core fragments dispersed in the test area. Most of this radioactive material was deposited within a few hundred feet²².

In the Phoebus 1A-321 loss-of-coolant accident, approximately 5 to 8% of the core was estimated to have rapidly exited through the nozzle, at the end of a 10.8 min power run²⁴. A preliminary evaluation indicated that 20 to 25% of the core was rapidly expelled²³. The 5 to 8% value is believed to be the best available estimate.

Isodose and isodeposition patterns were measured for both events¹⁹⁻²⁴ and these data are used extensively in this analysis.

The observed differences between the Kiwi TNT and Phoebus 1A-321 dose rate-vs-distance scale-up to E/STS 2-3 conditions can be explained by an investigation of the distribution of the ground depositions. In the case of the Phoebus (core meltdown) scale-up, about 80% of the ground deposition within 128,000 ft occurred within the first 10,000 ft of the test point. For the Kiwi (excursion) scale-up, less than 5% of the ground deposition within 128,000 ft occurred within the first 10,000 ft (Figure 39).

The preceding evaluation of the location of the ground-deposited fission product activity from Kiwi TNT data is based on the fraction of the

ground-deposited activity and its radial distance from the test point. This was determined using scale-up data based on the earlier referenced experimental fission product isodeposition contours.

When the two incidents are extrapolated to E/STS 2-3 conditions (e.g., 1 hr after 45 min at 10,000 Mw) the effects of the relatively increased close-in deposition for the Phoebus-type accident are evident, as shown in Figure 75. This demonstrates the importance of the deposition patterns, since the Phoebus 1A-321 incident resulted in only 5 to 8% total inventory release, compared with 100% for Kiwi TNT (66% long-range distribution and most of the remainder in large, close-in fragments). The release of a large inventory of fission products such as results from a scale-up to E/STS 2-3 conditions yields maximum dose rates, when the products are distributed locally as in the Phoebus 1A-321 case, where distances of between 5000 and 10,000 ft are involved and the core pieces deposited on the stand are diminished by air attenuation and distance. It can be concluded from Figure 75 that within 5000 ft maximum dose rates result from scaled-up Kiwi TNT data, and beyond 5000 ft Phoebus 1A-321 scale-up produces maximum dose rates.

E. CONTROL CENTER AND ACCESS TUNNEL CONDITIONS

Dose-rate calculations for the control center and access tunnel following a nuclear accident were performed on the basis that deposition of significant fragments of the core directly above occupied areas might occur. Specifically, for the purpose of these calculations, a source equivalent to 10% of the core fission product inventory at the end of 30 min of operation at 5000 Mw was utilized. The dose accumulated by an occupant of the control center in the time period from 0.01 to 1000 hr (essentially equivalent to infinite integral) is tabulated as a function of concrete shield thickness in Table 54. It is assumed that occupants will be about 15 ft below the source material. It appears that a total thickness equivalent to about 5-1/2 ft of concrete is required to limit this dose to 3 rem. Table 55 shows the cumulative dose as a function of time for a total exposure of 3 rem.

TABLE 54

DOSE IN CONTROL CENTER VS CONCRETE
SHIELD THICKNESS AT A POINT 15 FEET BELOW SHIELD
(With 10% of core fission products following 30-minutes operation
5000 Mw deposited above Control Center)

CONCRETE THICKNESS, <u>ft²</u>	DOSE <u>Rem</u>
0	5.6×10^6
1	6.0×10^5
2	3.1×10^4
3	1.8×10^3
4	1.2×10^2
5	7.7×10^0
6	5.2×10^{-1}
7	3.5×10^{-2}
8	2.4×10^{-3}
9	1.6×10^{-4}

TABLE 55

TIME INTEGRATED DOSE IN THE CONTROL CENTER
AS A FUNCTION OF TIME

(ASSUMING A 5 1/2-FOOT EQUIVALENT THICKNESS OF CONCRETE
AS THE ROOF AND A DEPOSITION OF ONE-TENTH OF THE CORE)

<u>TIME AFTER DEPOSITION, hr</u>	<u>CUMULATIVE TIME-INTEGRATED DOSE, Rem</u>	<u>TIME AFTER DEPOSITION, hr</u>	<u>CUMULATIVE TIME-INTEGRATED DOSE, Rem</u>
.01	0.0212	2.0	1.47
.02	0.0421	3.0	1.69
.03	0.0618	4.0	1.85
.04	0.0812	5.0	1.96
.05	0.0998	6.0	2.05
.06	0.118	7.0	2.12
.07	0.136	8.0	2.18
.08	0.154	9.0	2.23
.09	0.172	10.0	2.28
0.1	0.189	15.0	2.44
0.2	0.344	20.0	2.54
0.3	0.474	30.0	2.67
0.4	0.587	40.0	2.76
0.5	0.686	50.0	2.82
0.6	0.773	60.0	2.87
0.7	0.852	70.0	2.91
0.8	0.923	80.0	2.95
0.9	0.986	90.0	2.98
1.0	1.04	100.0	3.00

XI. REFERENCE LIST

1. AGC Report: RN-TM-0415, Interim Nuclear Analysis in Support of E/STS 2-3 Site Layout Activities, December 1966
2. AGC Report: RN-S-0390, Interim Nuclear Analysis in Support of E/STS 2-3 Test Stand Preliminary Design, April 1967
3. Final Project Report Budgetary Study for the Engine/Stage Test Stand (E/STS 2-3), Nuclear Rocket Development Station (NRDS), Jackass Flats, Nevada, Volume II of V, Kaiser Engineers Report No. 66-21 RE(B), May 1966
4. Criteria for Preliminary Design of E/STS 2-3, SNPO Oakland, 24 January 1967
5. W. C. Loomis, Nuclear Flight Propulsion Module (NFPM) Radiation Environment in Lower Bulkhead Region, R-AS-VP-125-67, prepared by Marshall Space Flight Center, Huntsville, Alabama, August 1967
6. NERVA II Reactor Conceptual Design Report, WANL-TME-1315, October 1965 (CR-D)
7. NASA/MSFC: Private Communication Transmitting Drawing Number SK 10-7295 Nuclear Ground Test Module
8. R. Malenfant, QAD: A Series of Point-Kernel General-Purpose Shielding Programs, LA-3573, October 1966
9. B. Carlson, C. Lee, and J. Worlton, The DSN and TDC Neutron Transport Codes, LAMS-2346, October 16, 1959
10. R. Malenfant, G. Graves, LASL Computer Program: GGG, Private Communication
11. The GAMSCAT Program, A Gamma Scattering Code Based on Differential Dose Albedos, Unpublished Aerojet-General Corporation Computer Code, 1964
12. M. A. Capo, Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source, APEX-510, General Electric Corporation, Cincinnati, Ohio, August 1959
13. E. A. Warman, D. R. Rogers, G. Garrington, and B. A. Lindsey, Applicability of Simplified Approximation Techniques for Air and Concrete Gamma Ray Scattering, AGC-NRO-1266, presented at the Thirteenth Annual Meeting, American Nuclear Society at San Diego, June 1967

14. M. A. Capo, Gamma Ray Absorption Coefficients for Elements and Mixtures, XCD-59, 10-19 GEANPD, 28 September 1959
15. NEUSCAT, Neutron Scattering Code: NRO Computer Program (to be published)
16. P. C. Heiser and L. O. Ricks, Activation Source Strength Program ACT-II, WANL-TNR-063
17. Kaiser Engineers Dwg. No. C1-1101, Revision R-D, Job No. 6628-12, Plot Plan
18. J. F. Perkins, R. W. King, Energy Release from the Decay of Fission Products, Nuclear Science and Engineering, 3,726-746, 1958
19. R. V. Fultyn, Airborne Radioactivity and Contamination Patterns from a Transient Test of a Nuclear Rocket Engine, presented at the International Symposium of Fission Product Release and Transport Under Accident Conditions held at Oak Ridge, Tennessee, April 5-7, CONF-650407, Volume I
20. On-Site Radiological Report of the Transient Nuclear Test, prepared by Radiation Service Department, Pan American World Airways, Inc., Nuclear Rocket Development Station Project, Las Vegas, Nevada, PAA 33-6, 13 July 1966
21. P. K. Lee, F. C. V. Worman, Los Alamos Scientific Laboratory, Integral Gamma and Neutron Measurements on the Kiwi TNT, LA-3304, 5 July 1965
22. P. K. Lee, R. V. Fultyn, Los Alamos Scientific Laboratory, Kiwi Transient Nuclear Test Dose Rate Survey, LA-3519-MS, 3 August 1966
23. Group H-8, Los Alamos Scientific Laboratory, Phoebus 1A-321 EP-IV Preliminary Evaluations of Environmental Effects, H8-MR-65-2, 2 June 1965
24. R. W. Henderson, O. W. Larson, and R. V. Fultyn, Radiation Measurements of the Effluent from the Phoebus 1A-321 Reactor, LA-3396-MW, 27 June 1966

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