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HTGR ACCIDENT INITIATION AND PROGRESSION ANALYSIS STATUS REPORT

VOLUME VII. OCCUPATIONAL RADIATION EXPOSURES FROM GAS-BORNE AND PLATEOUT ACTIVITIES

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1. SUMMARY

As a part of the Accident Initiation and Progression Analysis (AIPA) program, calculations were performed of the occupational dose rates and man-rem exposures from gas-borne and plateout activities in a reference 3000-MW(t) HTGR plant (Ref. 1-1).

The median and 95%-confidence-level fission-product activities used in the calculations were determined in the study described in Volume V of this AIPA report. These activities differ from the design activity level criteria, which were typically used to calculate the shielding thicknesses for the equipment. The design activity level criteria are also described in Volume V.

The study included a preliminary survey to determine the most important contributors by operation or radiation source to the man-rem exposures. This survey was followed by detailed calculations for the most important cases. Median and 95%-confidence-level man-rem exposures per year were obtained for the gaseous activity in the containment building, moisture monitor system, analytic instrumentation, helium regeneration system, gas waste system, and reflector-block shipping. Median and 95%-confidence-level man-rem exposures per operation were obtained for the main-circulator removal, steam-generator tube plugging, and steam-generator removal and replacement. For each of these cases, the contributions to the man-rem exposures were calculated for the important isotopes.

The criteria described in the U.S. Code of Federal Regulations, 10CFR20, and in NRC Regulatory Guide 8.8 were used in determining the reasonableness of the results. The calculated median man-rem exposures are reasonable in all cases. Also, the calculated 95%-confidence-level man-rem exposures are reasonable in all cases except the values for reflector-block shipping and for steam-generator removal and replacement. The unreasonably

large man-rem exposures for these two cases are caused by the large uncertainties in the Cs-134 and Cs-137 plateout activities. To achieve reasonable 95%-confidence-level man-rem exposures for these cases, additional shielding or smaller 95%-confidence-level plateout activities for Cs-134 and Cs-137 will be needed.

REFERENCE

- 1-1. "General Atomic Standard Safety Analysis Report," Amendment 4, August 1, 1975 (NRC Docket STN 50-535).

2. CALCULATIONAL METHOD

2.1. MEDIAN MAN-REM EXPOSURES

The PATH code (Ref. 2-1) was used to calculate the dose rates from the median isotope activities. PATH is a highly flexible point-kernel code. The dose rates from the PATH calculations were multiplied by the exposure times per year or per operation to obtain the man-rem exposures per year or per operation.

In equation form, the median man-rem dose or exposure from a mono-energetic, point isotropic source may be stated as

$$D_{K,50\%} = C_K B_K t e^{-\mu_K R} A_{K,50\%}, \quad (2-1)$$

where K = subscript for each gamma ray, or for each isotope in the case of only one important gamma ray per isotope,

B_K = exposure buildup factor,

$A_{K,50\%}$ = median activity,

μ_K = macroscopic total cross section (cm^{-1}) evaluated at the initial gamma-ray energy,

$e^{-\mu_K R}$ = attenuation factor, which is the probability that the gamma ray travels a distance R (cm) without an interaction,

t = time of exposure,

C_K = all other factors, including $1/R^2$ effects and the conversion factor to obtain the dose rate.

The PATH code sums the dose rate results for each source location and for each gamma-ray energy; for simplicity, these summations are not included in the equations given here. The total median dose or exposure summed over the contributions from all isotopes is

$$D_{50\%} = \sum_K D_{K,50\%} \quad (2-2)$$

2.2. 95%-CONFIDENCE-LEVEL MAN-REM EXPOSURES

In this volume, the 95%-confidence level should always be taken to mean the upper one-sided 95%-confidence level.

As discussed in Volume V of this report, the circulating and plateout activities are assumed to have log normal frequency distributions. Assuming log normal distributions in the other factors given in Eq. 2-1, the dose is then also log normal, and

$$\sigma_{\ln D_K}^2 = \sigma_{\ln C_K}^2 + \sigma_{\ln B_K}^2 + \sigma_{\ln t}^2 + \sigma_{\mu_{KR}}^2 + \sigma_{\ln A_K}^2, \quad (2-3)$$

where each of the σ^2 in the equation is the variance of the expression in the subscript. The man-rem dose or exposure with a 95%-confidence level from isotope K is

$$D_{K,95\%} = D_{K,50\%} e^{1.65 \sigma_{\ln D_K}} \quad (2-4)$$

For the calculations discussed in this volume, the uncertainties from the source activities are usually much larger than those from the other effects. Consequently, a number of approximations for the values of $\sigma_{\ln B_K}^2$ and $\sigma_{\mu_{KR}}^2$ were made to simplify the calculations, with little effect on the results.

The ratio of the 95%-confidence-level buildup factor to the median buildup factor was taken to be 1.20 in the calculations. This value was chosen on the basis of typical scatter of data points about curves of the buildup factors. Variations as a function of material, gamma-ray energy, and the number of mean free paths were neglected. From the equation

$$e^{1.65 \sigma_{\ln B}} = 1.2, \quad (2-5)$$

$\sigma_{\ln B}^2$ was determined to be 0.0122.

The 2σ uncertainty in μ was taken to be 2%. This value was chosen on the basis of the uncertainties stated in Ref. 2-2 for shielding materials and gamma-ray energies of interest. (The uncertainties stated in Ref. 2-2 were determined on the basis of scatter of available data; the confidence level is not explicitly known and may differ from this assumption.) The value for $\sigma_{\mu_K R}$ then equals $(0.01)(\mu_K R)$, where $\mu_K R$ is determined from the important gamma-ray energy of an isotope and the important source location.

In the calculations, the uncertainties in the point-kernel method were neglected because of the difficulty in quantifying them. However, these uncertainties are small with respect to many of the uncertainties in the source activities, particularly those for the plateout activities. The exposure times for the various operations are typically not well known; in the calculations, the uncertainties in exposure time were neglected because of the difficulty in quantifying them. A change in the exposure time would, of course, result in a directly proportional change in the median value of the man-rem exposure. The conversion factor to obtain dose rate from the gamma-ray flux has a small uncertainty relative to many of the other uncertainties, and the uncertainties in the distance R were neglected because of the difficulty in quantifying them. Consequently, the uncertainties in C_K were assumed to be small. The neglected uncertainties would cause the 95%-confidence-level doses to be greater than those stated in this report.

For a number of cases, the sum of $\sigma_{\ln B}^2$ and $\sigma_{\mu_K R}^2$ was negligible with respect to $\sigma_{\ln A_K}^2$, so that $\sigma_{\ln D_K}^2 \approx \sigma_{\ln A_K}^2$. This was particularly true for the plateout activities.

The total 95%-confidence-level dose, $D_{95\%}$, was determined by summing the 95%-confidence-level dose for each isotope. In equation form, this is

$$D_{95\%} = \sum_K D_{K,95\%} \quad (2-6)$$

This method assumes that the $\ln D_K$ are completely correlated in a positive sense. Actually, the $\ln A_K$ are usually partially correlated and sometimes (as for different isotopes of the same element) are completely correlated. Similarly, there is a strong correlation in the $\ln$ of the other factors in Eq. 2-1 for different isotopes. Consequently, in practice the results are reasonably close to those obtained from a rigorous treatment of the correlation.

2.3. SOURCE TERMS FOR EXPOSURE CALCULATIONS

The median and 95%-confidence-level activities for this study were obtained from the AIPA study of fission-product source terms discussed in Volume V. Because of the limited time frame of the AIPA studies, the calculations reported in this volume were sometimes performed with 95%-confidence-level activities or variances which differ from the final values given in Volume V, because of some subsequent improvements in the calculations presented in Volume V. For some of the circulating and total plateout activities, the final variances given in Volume V were higher than those used in this volume; the maximum effect on the 95%-confidence-level man-rem exposures from these differences is 11%. For the distributed plateout activities (in Ci/cm^2 at particular locations), the final 95%-confidence-level activities in Volume V for the cesium isotopes, Ag-110m, and Te-131m are lower than those used in this volume. The cesium isotopes are the most important of these. For the cesium 95%-confidence-level distributed activities, the ratio of the plateout value used in this study to the final value in Volume V is approximately 1.3 at the replaceable lower reflector, 2.4 at the steam-generator reheater, 1.5 at the steam-generator evaporator and economizer, 1.4 at the circulator, and 1.8 at the upper reflector. For Ag-110m, the corresponding distributed plateout ratios are approximately 1.1, 1.0, 1.6, 2.1, and 2.3. For the Te-131m 95%-confidence-level distributed activities, the ratio of the plateout value used in this study to the final value in Volume V is approximately 1.2 at the reheater. The other locations are not important for Te-131m.

The overall effect of the above differences is that the calculated median man-rem exposures are not affected, the calculated 95%-confidence-level man-rem exposures from circulating activities are low by 11% or less, and the calculated 95%-confidence-level man-rem exposures from distributed plateout activities may be high by a factor of 2.4 or less, depending upon the location and the important isotopes. However, the conclusions presented in this volume concerning operations that need more shielding or that require smaller activity levels are not affected by the above differences.

According to Volume V, indeterminate uncertainties induced by various design assumptions were not included in the analysis of activity uncertainties. The inclusion of these uncertainties would increase the 95%-confidence-level doses.

2.4. SHIELDING THICKNESSES

The shielding thicknesses used in this study were typically based on design activities, as described in Volume V. In the case of steam-generator removal and replacement, the shielding thicknesses were based on expected values, which are not as conservative as the design values; expected values were used in this case because of the extremely rare occurrence of this operation. The expected activities are also described in Volume V.

2.5. SELECTION OF OPERATIONS OR RADIATION SOURCES

A list was made of the operations or radiation sources that might produce significant occupational man-rem exposures from gas-borne and plateout activities in a 3000-MW(t) HTGR plant. Normal operating and maintenance conditions plus significant operational occurrences were considered. Rough estimates were made of the median and 95%-confidence-level man-rem exposures per year or per operation. These estimates were based on previous dose-rate calculations with design activities, on the estimated exposure times, and on preliminary ratios of median to design activities or 95%-confidence-level to design activities for the important

isotopes. In the case of the transfer of the steam generator out of the containment building, the rough estimate was made on the basis of expected activities rather than design activities.

The operations or radiation sources which had significant median or 95%-confidence-level man-rem exposures were then analyzed in detail, as described in Section 3. These operations or sources included the gaseous activity in the containment building, the moisture monitor system, the analytic instrumentation, the helium regeneration system, the gas waste system, the reflector-block shipping, the main-circulator removal, steam-generator tube plugging, and steam-generator removal and replacement. The operations or sources with relatively small estimated man-rem exposures were not analyzed in more detail; these included the auxiliary service cask with the high-temperature filter/adsorber (HTF/A), the control-rod-drive storage wells with high-temperature filter/adsorbers, the reflector-block storage wells, the steam-generator lower penetrations during operation, and the helium purification system. For these preliminary estimates, the location of the helium purification system was assumed to be in the top head of the prestressed concrete reactor vessel (PCRVR). Maintenance operations that are not anticipated could, of course, cause significant man-rem exposures, including internal exposures.

REFERENCES

- 2-1. Clark, S. S., and B. A. Engholm, "PATH - A Highly Flexible, General Purpose Gamma Shielding Program," Gulf General Atomic Report GA-9908, December 10, 1969.
- 2-2. Hubbell, J. H., "Photon Cross Sections, Attenuation Coefficients and Energy Absorption Coefficients from 10 keV to 100 GeV," National Bureau of Standards Report NSRDS-NBS 2, August 1969.

3. MAN-REM EXPOSURES FOR IMPORTANT OPERATIONS OR RADIATION SOURCES

3.1. GASEOUS ACTIVITY IN CONTAINMENT BUILDING

3.1.1. Configuration and Source Terms

A study was done of the occupational man-rem exposure per year inside the containment building above the refueling floor due to leakage of the primary coolant through the PCRV at the rate of 1% per year. No other sources of radiation were considered, and the containment building was conservatively assumed to have no ventilation. The geometry used in the calculation is shown in Fig. 3-1. Although this configuration does not exactly reproduce the containment geometry, it will provide more than adequate results.

The activity levels at equilibrium were derived from the following equation, assuming no purification or leakage of containment atmosphere:

$$(\text{leak rate into containment}) = (\text{decay rate}) \quad (3-1)$$

or

$$\ell A_p = \lambda A_c ,$$

where A_p = equilibrium activity in primary coolant, Ci,

A_c = equilibrium activity in containment, Ci,

ℓ = fractional leak rate from PCRV into containment,

λ = decay constant for isotope of interest.

To determine the number of curies per cm^3 in the containment building for a 3000-MW(t) plant (Ref. 3-1), the value of A_c was divided by a typical

Fig. 3-1. PATH code geometry for dose rates inside containment building

containment volume. Table 3-1 gives the calculated results for the median activities. It was assumed that only noble gases would be present in the containment atmosphere. (The halogens were assumed to plate out in the PCR.V.) Daughter products of noble gases were neglected, since a calculation showed that they had only a small effect.

3.1.2. Median and 95%-Confidence-Level Dose Rates

The PATH code was used to calculate the dose rate at 6 ft above the center of the refueling floor. The dose rate received from beta-ray inhalation was not considered in this calculation. In the PATH calculation, no credit was taken for air attenuation, and no buildup was assumed. The dose rate was found to be 0.035 mrem/hr for median activities and 0.054 mrem/hr for the activities corresponding to the 95%-confidence level. The uncertainty in the leak rate through the PCR.V would increase the latter dose rate. The calculated dose rates would not vary significantly for heights less than 6 ft above the floor. As a function of position on the refueling floor, the dose rates would not change greatly near the central region of the floor, but would decrease significantly near the containment.

3.1.3. Median and 95%-Confidence-Level Man-Rem Exposures

To calculate the man-rem exposures, the above dose rates were multiplied by the time corresponding to two workers in the containment building for 10 hr per week for each worker (Ref. 3-1). Consequently, the result for the median case is 0.036 man-rem per year; for the 95%-confidence-level case, the result is 0.056 man-rem per year.

The isotopes which are the principal contributors to the man-rem exposure for both the median and 95%-confidence-level values are given in Table 3-2. As shown by the table, Kr-88 is by far the largest contributor to the man-rem exposures.

TABLE 3-1
TOTAL MEDIAN ACTIVITY IN PRIMARY COOLANT AND ACTIVITY PER
CM³ IN CONTAINMENT BUILDING

Isotope	Total Activity in Primary Coolant ^(a) (Ci)	Ci/cm ³ Inside Containment Building
Kr-85m	776	7.94 - 14 ^(b)
Kr-87	878	2.59 - 14
Kr-88	1750	1.14 - 13
Kr-89	397	4.89 - 16
Kr-90	151	3.15 - 17
Kr-91	39.8	2.21 - 18
Xe-133m	28.9	3.64 - 14
Xe-133	596	1.75 - 12
Xe-135m	409	2.49 - 15
Xe-135	1010	2.15 - 13
Xe-137	247	3.66 - 16
Xe-138	445	2.45 - 15
Xe-140	24.2	2.19 - 18

(a) From Volume V.

(b) Read 7.94 - 14 as 7.94×10^{-14} . This notation also applies to other tables in this volume.

TABLE 3-2
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FROM GASEOUS ACTIVITY IN
CONTAINMENT BUILDING

Isotope	Percent Contribution to Man-Rem Exposure	
	Median Value	95%-Confidence- Level Value
Kr-88	57	56
Xe-135	17	18
Xe-133	15	15
Kr-87	5.2	5.0
Kr-85m	3.6	3.6
Xe-138	0.8	0.8
Xe-135m	0.3	0.3
Kr-89	0.3	0.3

3.2. MOISTURE MONITOR SYSTEM

3.2.1. Configuration and Source Terms

Calculations were performed to determine the man-rem exposure received per year during the maintenance and operation of the moisture monitor system. This system is composed of three major components plus associated piping. The largest component is the hygrometer module, of which there are six. The only other component requiring regular maintenance is the compressor module. Figures 3-2, 3-3, and 3-4 show diagrams of the components, indicating sources of interest. The lines in the modules are not shielded, and the lines outside the modules are typically shielded.

The gas-borne activities in the lines of the hygrometer module and compressor module were determined from the inert gases in the medium circulating activities. The results presented in Table 3-3 are based on 13,500 lbm of circulating helium inventory in the primary circuit at 700 psia and 600°F; the latter conditions approximately correspond to the conditions at the reactor inlet.

For the line sources used in the moisture-monitor-system calculations, plateout source strengths were assumed to be in equilibrium with some of the inert gases. The data given in Table 3-3 for gas-borne activities include these sources.

Because of the similarity of the filters in the hygrometer module to those in the HTF/A, it was assumed that the filters in the hygrometer module contain the same activity concentration as the HTF/A. The median values of the activity concentration in the filters were obtained using the HTF/A design activity data and the ratios of the median circulating activity data to the corresponding design activity data. The isotopes used and the median source strengths are given in Table 3-3 under the heading of filters.

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Fig. 3-2. Hygrometer module

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Fig. 3-3. Compressor module

Fig. 3-4. Schematic of moisture monitor system

TABLE 3-3
SPECIFIC MEDIAN ACTIVITY (Ci/cm³) FOR MOISTURE MONITOR SYSTEM

Isotope	Gas Borne ^(a)	Filters
Rb-88	*1.12 - 6	5.12 - 7
Rb-89	--	1.18 - 7
Rb-90	*9.65 - 8	
Rb-91	*2.54 - 8	
Kr-85m	4.96 - 7	
Kr-85	--	
Kr-87	5.61 - 7	
Kr-88	1.12 - 6	
Kr-89	2.54 - 7	
Kr-90	9.65 - 8	
Kr-91	2.54 - 8	
Ag-110m	--	8.61 - 8
Te-129m	--	2.97 - 7
Te-131m	--	1.72 - 7
Te-132	--	1.78 - 6
I-131	--	1.22 - 5
I-132	--	3.51 - 6
I-133	--	8.53 - 6
I-134	--	2.69 - 6
I-135	--	3.54 - 6
Xe-133m	1.85 - 8	
Xe-133	3.81 - 7	
Xe-135m	2.61 - 7	
Xe-135	6.45 - 7	
Xe-137	1.59 - 7	
Xe-138	2.84 - 7	
Cs-134	--	4.04 - 6
Cs-138	*2.84 - 7	1.34 - 7
Cs-140	*1.55 - 8	
Ba-137m	--	6.67 - 6

(a) Daughter products of noble gases are identified by an asterisk.

3.2.2. Median and 95%-Confidence-Level Dose Rates

All the dose-rate determinations were made using the PATH code. The resulting median dose rates were 0.65 mrem/hr for the hygrometer module and 0.201 mrem/hr for the compressor module.

Input data for the calculation of the 95%-confidence-level dose rates are given in Tables 3-4 and 3-5. The values of $\sigma_{\mu_K R}^2$ were taken for the most important source points and for the strongest sources. The value of $\sigma_{\ln B}^2$ was taken to be 0.0122, as discussed in Section 2.2. The calculated 95%-confidence-level dose rates were 1.24 mrem/hr for the hygrometer module and 0.46 mrem/hr for the compressor module. The uncertainties in some of the assumptions in calculating activities for daughter products of noble gases cannot be quantified and were not included in this analysis.

3.2.3. Median and 95%-Confidence-Level Man-Rem Exposures

For each of the six hygrometer modules, approximately 8 man-hours per month are required for maintenance. The compressor module also requires 8 man-hours per month. The man-rem exposures for the hygrometer module and the compressor module were calculated on the basis of these times and the above dose rates. The results were 0.39 man-rem per year for the median exposure and 0.76 man-rem per year for the 95%-confidence-level exposure.

Table 3-6 gives the important isotopes, including the percentage of the total man-rem exposure contributed by each isotope.

3.3. ANALYTIC INSTRUMENTATION

3.3.1. Configuration and Source Terms

The analytic instrument room is made up of several components. The components analyzed were instrument boards C, D, and E and filters F36101/2, F36103/4, and F36401. Most of the other components in the room

TABLE 3-4
INPUT DATA FOR 95%-CONFIDENCE-LEVEL DOSE RATES FOR
HYGROMETER MODULE

Isotope	$\sigma_{\mu_K R}^2$	$\sigma_{\ln A_K}^2$
Kr-88	4.0 - 4	0.065
Rb-88	4.0 - 4	0.44
Cs-138	5.3 - 4	0.39
Kr-89	5.3 - 4	0.065
Kr-87	2.3 - 3	0.065
Xe-138	4.0 - 4	0.079
Rb-90	2.3 - 4	1.06
Xe-135	2.3 - 3	0.079
Kr-90	7.8 - 4	0.065
Xe-135m	1.3 - 3	0.079

TABLE 3-5
INPUT DATA FOR 95%-CONFIDENCE-LEVEL DOSE RATES FOR
COMPRESSOR MODULE

Isotope	$\sigma_{\mu_K R}$	$\sigma_{\ln A_K}^2$
Kr-88	1.44 - 3	0.065
Rb-88	1.52 - 3	0.44
Cs-138	1.44 - 3	0.39
Kr-89	1.94 - 3	0.065
Kr-87	1.44 - 3	0.065
Xe-138	1.52 - 3	0.079
Rb-90	1.23 - 3	1.06
Kr-90	3.48 - 3	0.065
Rb-91	1.23 - 3	1.85
Cs-140	1.44 - 3	2.23

TABLE 3-6
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FOR
MOISTURE MONITOR SYSTEM

Isotope	Percent Contribution to Man-Rem Exposure	
	Median Value	95%-Confidence- Level Value
Kr-88	36	29
Rb-88	13	20
Cs-138	11	16
Kr-89	8.4	6.9
Kr-87	6.8	5.7
Xe-138	5.4	4.7
Rb-90	3.7	11
Xe-135	2.7	2.4
Kr-90	2.7	2.2
Xe-135m	1.8	1.5

do not contain radioactive materials and therefore were not considered. The geometry for the filters and the instrument boards is given in Figs. 3-5, 3-6, and 3-7. The filters are on lines going to the instrument boards. Lead shielding around the F36101/2 and F36103/4 filters is not shown in Fig. 3-5.

The gas-borne activities and the activities for the F36101/2 and F36103/4 filters were obtained from Table 3-3. The activities for the F36401 filter were calculated from those for the low-temperature adsorber (LTA) using ratios with the design activities, as described in Section 3.2.1 for the HTF/A. The same procedure was also used to obtain the source data that pertain to the early regeneration of the radioactive-gas waste system (system 63) for a 3000-MW(t) plant. These data are given in Table 3-7.

3.3.2. Median and 95%-Confidence-Level Dose Rates

Operating plant personnel spend about 8 man-hours per day in the analytic-instrument room; approximately half of this time is spent taking readings from radioactive components. For the purpose of this analysis, the nonradioactive instrument boards are assumed to be 4.0 ft from the radioactive ones; this results in personnel spending approximately 4 hr at a distance of 18 in. from board D and 4 hr at a distance of 30 in. from board D. The contribution from the three filters to the yearly dose is small owing to small access times. Also, the contribution to the yearly dose from the early regeneration of the radioactive-gas waste system is small.

The median dose rates obtained from the PATH code were 0.15 mrem/hr from board D and 0.06 mrem/hr from the contribution of boards C and E to the board D value.

In the calculation of the 95%-confidence-level dose rates, the values of $\sigma_{\mu_K}^2$ were very small with respect to the values of $\sigma_{\ln B}^2 + \sigma_{\ln A_K}^2$ and

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F-36103/4

Fig. 3-5. Filters F36101/2 and F36103/4 for analytic instrumentation

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Fig. 3-6. Filter F36401 for analytic instrumentation

Fig. 3-7. Source boards C, D, and E for analytic instrumentation

TABLE 3-7
SPECIFIC MEDIAN ACTIVITY (Ci/cm³) FOR ANALYTIC INSTRUMENTATION

Isotope	F36401 Filter	System 63 ^(a)
Rb-88	1.17 - 3	
Kr-85m	8.50 - 4	1.33 - 8
Kr-85	8.11 - 4	0.68 - 3
Kr-87	2.70 - 4	--
Kr-88	1.20 - 3	2.41 - 10
Xe-131m	1.12 - 4	3.03 - 6
Xe-133m	3.86 - 4	0.64 - 5
Xe-133	1.90 - 2	4.43 - 4
Xe-135m	2.60 - 5	
Xe-135	2.31 - 3	1.85 - 6
Xe-138	2.50 - 5	--
Cs-138	2.16 - 5	--

(a) Based on a volume of 1050 ft³.

were therefore neglected. The resulting 95%-confidence-level dose rates were 0.24 mrem/hr from board D and 0.096 mrem/hr for the contributions of boards C and E to the board D value.

3.3.3. Median and 95%-Confidence-Level Man-Rem Exposures

From the median dose rate and exposure time given above, a median value of 0.61 man-rem per year was obtained. The 95%-confidence-level result was 0.98 man-rem per year.

Table 3-8 gives the contribution by isotope to the exposure. The most important isotope is Kr-88, followed by xenon and other krypton isotopes.

3.4. HELIUM REGENERATION SYSTEM AND RADIOACTIVE-GAS WASTE SYSTEM

For the balance-of-plant shielding, the helium regeneration system and the radioactive-gas waste system are the most important systems with gas-borne activities. The plateout activities in these systems are negligible, because these activities are removed by the HTF/A. The major source input for these two systems comes from regeneration of the LTA in the helium purification system.

3.4.1. Source Terms

To calculate the activity in either the helium regeneration system or the radioactive-gas waste system, the following assumptions were made:

1. Regeneration of the LTA after a minimum of 48 hr of isolated decay.
2. Holdup capacity of the system equal to the maximum activity, which is accumulated over 40 yr.

TABLE 3-8
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FOR
ANALYTIC INSTRUMENTATION

Isotope	Percent Contribution to Man-Rem Exposure	
	Median Value	95%-Confidence- Level Value
Kr-88	47	47
Xe-133	10.1	10.5
Kr-89	9.9	9.9
Kr-87	8.4	8.3
Xe-138	7.1	7.3
Kr-85	6.8	6.7
Xe-135	3.2	3.3
Kr-90	3.0	3.0

With 48-hr decay, all short-lived nuclides have negligible activity. Only Kr-85 and Xe-133 remain important. By using the equation

$$A = \frac{PC}{\lambda M} (1 - e^{-\lambda t}) \quad , \quad (3-2)$$

where A = total activity in the gas waste system and helium regeneration system, Ci,

P = helium purification rate = 2070 lb/hr,

C = circulating gas-borne activity in the primary circuit, Ci,

M = total circulating helium inventory in the primary circuit = 13,500 lb,

λ = decay constant (in units of hr^{-1} in this equation),

t = accumulation time (in units of hr in this equation),

the accumulated activity with no decay for a 3000-MW(t) plant was calculated. The results are given in Table 3-9A. With 48-hr decay preceding the LTA regeneration, the activity reduces to the values given in Table 3-9B.

3.4.2. Median and 95%-Confidence-Level Dose Rates

The present shielding design for the systems calls for 2.5 ft of ordinary concrete; this is required to limit the dose rate from the design gaseous activity to 2.5 mrem/hr or less at the outside surface of the concrete shield. Krypton-85 is the only contributor to the dose rate with this shield thickness, because Xe-133 emits very-low-energy gamma rays which are easily attenuated.

The dose rate from the median Kr-85 activity was calculated from the ratio of the median to design activities and from the dose rate for the design activity. The result at the outside surface of the shield was 0.083 mrem/hr.

TABLE 3-9
TOTAL ACCUMULATED ACTIVITY IN RADIOACTIVE-GAS WASTE SYSTEM AND
HELIUM REGENERATION SYSTEM

A. No Decay

Isotope	Activity (Ci)		
	Median	95%-Confidence-Level	Design
Kr-85	21,200	32,400	642,000
Xe-133	16,800	27,000	232,000

B. 48-Hr Decay Preceding LTA Regeneration

Isotope	Activity (Ci)		
	Median	95%-Confidence-Level	Design
Kr-85	21,200	32,400	642,000
Xe-133	12,900	20,700	178,000

In the calculation of the 95%-confidence-level dose rates, $\sigma_{\ln A}^2$ for Kr-85 was taken to be 0.065 (Vol. V). The value of $\sigma_{\mu_{KR}}^2$ was calculated from $[(0.01)(\mu_{KR})]^2$, as described in Section 2.2. With $R = 2.5$ ft or 76.2 cm for the concrete thickness and $\mu = 0.1889 \text{ cm}^{-1}$; $\sigma_{\mu_{KR}}^2 = 0.0208$. With $\sigma_{\ln B}^2 = 0.0122$ included, $\sigma_{\ln D}^2 = 0.098$. From Eq. 2-4, the 95%-confidence-level dose rate equals 0.139 mrem/hr.

3.4.3. Median and 95%-Confidence-Level Man-Rem Exposures

The personnel access time is about 20 man-hours per week for the helium regeneration system and about 10 man-hours per week for the gas waste system. The resulting median values for the man-rem exposure per year are 0.086 for the helium regeneration system and 0.043 for the gas waste system. The 95%-confidence levels for the man-rem exposure per year are 0.145 for the helium regeneration system and 0.073 for the gas waste system.

3.5. REFLECTOR-BLOCK SHIPPING

3.5.1. Configuration and Shielding Thicknesses

A study was performed of the man-rem exposure per year from shipping replaceable reflector blocks. The calculations were based on an 8-yr accumulation of activity and a 100-day decay after shutdown.

It was assumed that the reflector blocks will be shipped in 55-gal drums, with each drum containing one full-size or two half-size similar blocks. Also, it was assumed that lead shielding will be wrapped around the outside of the drums and that the voids in the drums will be filled with magnetite composed of 79% concrete with a density of 2.35 g/cm^3 and 21% iron with a density of 7.89 g/cm^3 .

The amount of lead shielding required to achieve a dose rate of 200 mrem/hr on the surface was determined for the top replaceable reflector blocks with the design activities given in Table 3-10. This dose-rate

criterion corresponds to the value given in the U.S. Code of Federal Regulations, Title 49. The activities shown in the table were calculated from the Ci/cm^2 plateout data for the blocks and from the surface area and volume of the blocks. The activities in the table are assumed to be typical of the activities for the blocks. The results showed that 0.65 in. of lead shielding is required. A similar calculation with design activities for the bottom replaceable reflector blocks showed that 1.2 in. of lead shielding is needed. The activity level for the side blocks was taken the same as for the top blocks. The assumption is believed to be conservative, but cross contamination from adjacent blocks was not investigated in this study.

Some reflector blocks have activities that are significantly different from those in Table 3-10. For this calculation, the shielding for these blocks was also assumed to be sufficient to give a dose rate of 200 mrem/hr for design activities.

3.5.2. Median and 95%-Confidence-Level Dose Rates

The median activities used in this calculation are given in Table 3-10. For the top replaceable reflector blocks, the calculated median surface dose rate with 0.65 in. of lead is 2.7 mrem/hr; for the bottom replaceable reflector blocks, this dose rate with 1.2 in. of lead is 54 mrem/hr. These results, which are given in Table 3-11, are well under the 200 mrem/hr limit.

Dose rates with a 95%-confidence level were calculated using Eqs. 2-4 and 2-6. For the reflector blocks, the value of $\sigma_{\ln A_K}^2$ for each important isotope is very large, so that $\sigma_{\ln D_K}^2$ essentially equals $\sigma_{\ln A_K}^2$. For the top replaceable reflector blocks, the 95%-confidence-level surface dose rate with 0.65 in. of lead is 110 rem/hr. For the bottom replaceable reflector blocks, this dose rate with 1.2 in. of lead is 1.7 rem/hr.

To achieve reasonable dose rates for the given 95%-confidence-level activities, additional shielding is required. Calculations show that 3.35 in. of lead will reduce the 95%-confidence-level dose rate from the top

TABLE 3-10
ACTIVITY LEVELS FOR REPLACEABLE REFLECTOR BLOCKS
(100-DAY DECAY, 8-YR ACCUMULATION)

Isotope	Activity (Ci/cm ³)	
	Design	Median
Top blocks		
Ag-110m	2.34 - 6	8.11 - 10
Cs-134	4.38 - 5	7.81 - 7
Ba-137m	3.96 - 5	7.09 - 7
Bottom blocks		
Ag-110m	1.41 - 5	1.32 - 6
Cs-134	1.60 - 4	6.08 - 5
Ba-137m	1.44 - 4	5.49 - 5

TABLE 3-11
DOSE RATES ON SURFACE OF 55-GAL DRUM CONTAINING REPLACEABLE
REFLECTOR BLOCKS

Lead Thickness (in.)	Dose Rate from Top Replaceable Reflector Blocks		Dose Rate from Bottom Replaceable Reflector Blocks	
	Median Activities	95%-Confidence- Level Activities	Median Activities	95%-Confidence- Level Activities
0.65	2.7 mrem/hr	110 rem/hr		
1.2			54 mrem/hr	1.7 rem/hr
2.1			6.6 mrem/hr	190 mrem/hr
3.35	0.005 mrem/hr	190 mrem/hr		

blocks to 190 mrem/hr and 2.1 in. of lead will reduce the corresponding dose rate from the bottom blocks to 190 mrem/hr. The increase in the lead thickness is significantly larger for the top blocks than for the bottom blocks because the ratio of the 95%-confidence-level activity to the design activity is much larger for the top reflector blocks than for the bottom reflector blocks.

3.5.3. Median and 95%-Confidence-Level Man-Rem Exposures

Table 3-12 gives the number of reflector blocks to be shipped each year for a 3000-MW(t) plant. Based on this information, the dose rates in Table 3-11, and an estimated exposure time of 3 min per drum, the man-rem exposure per year was determined for each year after startup. In this calculation, it was assumed that the dose rate received by the workers is the dose rate on the shield surface and that physical shielding or access control is provided during loading operations to ensure the same radiation protection as provided by the shielded drums. The results of the calculations are given in Table 3-13.

Table 3-14 gives the contribution by isotope to the man-rem exposures for each year of the cycle. Cesium-134 is the main contributor in all cases; the other significant contributors are Ag-110m and Ba-137m (from Cs-137).

3.6. MAIN-CIRCULATOR REMOVAL

3.6.1. Source Terms and Configuration

The source terms for the main-circulator removal are the 5-day-decay plateout activities on the helium-wetted surface. The 5-day decay time was chosen to account for the time required for depressurization of the PCRV and preparation for the circulator removal, and to reduce shielding requirements.

TABLE 3-12
REPLACEABLE-REFLECTOR-BLOCK SHIPPING SCHEDULE FOR
3000-MW(t) HTGR

Type of Block	Year	Number of Half-Size Blocks	Number of Full-Size Blocks
Top	0-4	0	0
	5	114	19
	6	96	18
	7	108	18
	8	102	18
Side	0-4	0	0
	5	0	0
	6	78	342
	7	12	48
	8	66	294
Bottom	0-4	0	
	5	133	
	6	114	
	7	126	
	8	120	

TABLE 3-13
MAN-REM EXPOSURES FOR SHIPPING REPLACEABLE REFLECTOR BLOCKS

Year	Man-Rem Per Year With Shielding Designed for Design Activities		Man-Rem Per Year With Shielding Designed for 95%-Confidence-Level Activities	
	Median Activities	95%-Confidence- Level Activities	Median Activities	95%-Confidence- Level Activities
5	0.19	420	0.022	1.4
6	0.22	2500	0.019	4.9
7	0.19	690	0.021	1.8
8	0.22	2200	0.020	4.4
Unweighted average	0.21	1400	0.021	3.1

TABLE 3-14
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FOR REFLECTOR-BLOCK SHIPPING

Year	Isotope	Percent Contribution with Shielding Designed for Design Activities		Percent Contribution with Shielding Designed for 95%-Confidence-Level Activities	
		Median Activities	95%-Confidence- Level Activities	Median Activities	95%-Confidence- Level Activities
5	Ag-110m	9.5	0.2	16	2.3
	Cs-134	85	89	82	97
	Ba-137m	5.7	11	1.7	1.1
6	Ag-110m	7.3	0.2	16	1.1
	Cs-134	86	89	82	98
	Ba-137m	6.9	11	1.7	0.5
7	Ag-110m	9.1	0.2	16	1.8
	Cs-134	85	89	82	97
	Ba-137m	5.9	11	1.7	0.8
8	Ag-110m	7.5	0.2	16	1.1
	Cs-134	86	89	82	98
	Ba-137m	6.8	11	1.7	0.5
Unweighted average	Ag-110m	8.3	0.2	16	1.6
	Cs-134	85	89	82	98
	Ba-137m	6.3	11	1.7	0.2

The plateout on the circulator in terms of Ci/cm^2 for each nuclide of importance was calculated using the equation

$$A_i = \frac{T_i}{S} f_i, \quad (3-3)$$

where A_i = surface activity on the circulator for nuclide i , Ci/cm^2 ,
 T_i = total plateout inventory in the primary circuit for nuclide i , Ci ,
 S = total plateout surface area in the primary circuit,
 $= 4.7 \times 10^8 \text{ cm}^2$,
 f_i = nonuniform plateout distribution factor for nuclide i , which equals the ratio of the actual surface activity on the circulator to the uniform surface activity throughout the primary circuit.

The total plateout inventory in the primary circuit for each nuclide was adjusted to account for the 5-day decay. The actual surface activity required to calculate the nonuniform distribution factor was taken from the data for the midpoint of the main circulator. Table 3-15 presents the resulting activity values for the main circulator at 5-day decay, together with the variance of the natural logarithm of the activity, $\sigma_{\ln A}^2$, for each nuclide.

The geometry of the main circulator and its handling cask was modeled as shown in Fig. 3-8. The steel shield thickness for the lower portion of the circulator handling cask was taken as 10 in. to be consistent with the design described in Ref. 3-1. A dose point was placed on the outside surface of the shield directly opposite the bottom part of the circulator, where the plateout surface area is relatively large.

Seven sources with appropriate locations as indicated in Fig. 3-8 were considered. Source 1 is a volumetric source in the region of compressor and stator blades with a homogenized density of $1.94 \text{ g}/\text{cm}^3$ of steel. The volumetric source strengths were obtained by dividing the product of the plateout surface activity and total plateout surface area in the region

TABLE 3-15
PLATEOUT ON MAIN CIRCULATOR WITH 5-DAY DECAY

Isotope	Surface Activity (Ci/cm ²)		$\sigma_{\ln A}$	$\sigma_{\ln A}^2$
	Median	95%-Confidence Level		
Zr-95	3.66 - 10	2.04 - 6	5.23	27.4
Nb-95	5.62 - 10	3.13 - 6	5.23	27.4
Ag-110m	3.47 - 9	3.16 - 5	5.53	30.6
Te-129m	1.68 - 8	1.18 - 4	5.37	28.8
Te-129	1.07 - 8	7.65 - 5	5.37	28.8
Te-131m	6.67 - 10	4.71 - 6	5.37	28.8
Te-131	1.23 - 10	8.68 - 7	5.37	28.8
Te-132	3.88 - 8	2.73 - 4	5.37	28.8
I-131	6.57 - 6	4.61 - 4	2.58	6.66
I-132	2.75 - 6	1.93 - 4	2.58	6.66
I-133	1.24 - 7	8.72 - 6	2.58	6.66
Cs-134	2.85 - 6	3.30 - 2	5.67	32.2
Ba-137m	7.53 - 6	9.40 - 2	5.72	32.7
Ba-140	1.18 - 8	3.02 - 8	0.57	0.325
La-140	1.38 - 8	3.54 - 8	0.57	0.325
Ce-144	1.49 - 10	1.68 - 6	5.65	31.9
Pr-144	1.49 - 10	1.68 - 6	5.65	31.9

ALL DIMENSIONS ARE IN INCHES.
CIRCLED NUMBERS STAND FOR SOURCES.
Z STANDS FOR HEIGHT.
R STANDS FOR RADIUS

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Fig. 3-8. Geometry of main circulator for PATH runs

under consideration by the volume of that region. Sources 2 through 5 were all regarded as cylindrical shell sources. Note that the cylindrical shells were used to approximate any source geometry of conical shape so that the source could be directly input to the PATH code. Sources 6 and 7 are disk sources on the bottom and top surfaces, respectively, of the compressor.

3.6.2. Median and 95%-Confidence-Level Dose Rates

The calculated median dose rate at the point of interest was 7.6×10^{-3} mrem/hr.

In the calculation of the 95%-confidence-level dose rates, the values of $\sigma_{\ln A_K}^2$ for the important isotopes are much larger than the sum of $\sigma_{\ln B}^2$ and $\sigma_{\mu_{KR}}^2$. Therefore, the uncertainty in the dose rate from a particular nuclide is essentially identical to the uncertainty in the corresponding plateout activity. The calculated 95%-confidence-level dose rate from all the isotopes was 15.5 mrem/hr.

3.6.3. Median and 95%-Confidence-Level Man-Rem Exposures

Once the dose rates are known, the total man-rem exposure during the main-circulator removal can be calculated with the following assumptions:

1. Fifty man-hours are required to complete the removal procedures, which include bolting, unbolting, connecting, and disconnecting of piping and wires; equipment removal, transport, and storage; and other necessary peripheral work.
2. Physical shielding and/or access control is provided to ensure the same degree of radiation protection as provided by the main-circulator handling cask.

3. The dose rate received by workers is that on the surface of the shield, which means that the decrease of the dose rate with distance from the shield is not considered.
4. There is no credit for radioactive decay in the course of the removal operations.

The accumulated man-rem exposure for completing the removal operation is then simply the dose rate received times the total exposure time in man-hours. With a median dose rate of 7.6×10^{-3} mrem/hr and a 95%-confidence-level dose rate of 15.5 mrem/hr, the corresponding man-rem exposures per operation are 0.00038 and 0.78, respectively.

The five isotopes which make the greatest contributions to the man-rem exposures are given in Table 3-16. Iodine-132 is by far the most important isotope for the median value. Because of the very large uncertainty in the cesium plateout activities, Cs-134 is by far the most important isotope for the 95%-confidence-level value.

3.7. STEAM-GENERATOR TUBE PLUGGING

3.7.1. Configuration and Shielding Thicknesses

A study was performed of the man-rem exposure received during steam-generator tube plugging at 24 hr after shutdown. A cross section of the lower penetrations is shown in Fig. 3-9.

The depleted uranium thicknesses required for shielding in the tube-plugging operations were calculated on the basis of design plateout activities. The material compositions used in the calculation are given in Table 3-17.

The design plateout source spectrum at the midpoint of the reheater at 24 hr after shutdown is given in Table 3-18. The various sources are described in Table 3-19. The relative strengths for these plateout sources

TABLE 3-16
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FOR
MAIN-CIRCULATOR REMOVAL

Isotope	Percent Contribution to Man-Rem Exposure	
	Median Value	95%-Confidence- Level Value
I-132	80	2.8
Cs-134	14.5	82.0
Ba-137m	2.4	14.4
La-140	2.2	
I-131	0.4	
Ag-110m		0.7
Te-131m		0.05

Fig. 3-9. Dose rates in Mark II steam-generator lower penetrations due to design plateout values

TABLE 3-17
MATERIAL COMPOSITIONS USED IN PATH MODEL FOR TUBE-PLUGGING STUDY

Region	Material	Density (g/cm ³)	Volume Fraction
PCRV	Concrete	2.35	1.0
Superheat bundle	Iron	7.89	0.172
Superheat tubesheet	Iron	7.89	0.9313
Reheat bundle	Iron	7.89	0.136
Reheat tubesheet	Iron	7.89	0.8494
Feedwater bundle	Iron	7.89	0.199
Feedwater tubesheet	Iron	7.89	0.883
Piping and liner	Iron	7.89	1.0
Shielding	Depleted uranium	18.7	1.0

TABLE 3-18
DESIGN PLATEOUT SOURCE SPECTRUM FOR TUBE-PLUGGING STUDY
(1-DAY DECAY)

Gamma-Ray Energy (MeV)	Strength (MeV/cm ² -sec)
0.3	1.28 + 7
0.6	2.23 + 7
0.8	1.76 + 7
1.0	2.70 + 6
1.25	3.90 + 6
1.6	4.59 + 6
2.0	6.17 + 5
2.4	3.43 + 5

TABLE 3-19
SOURCES USED IN PATH MODEL FOR TUBE-PLUGGING STUDY

Source	Source Description	Relative Strength
1	Plateout on steam generator cavity wall	1.0
2	Plateout on lower portion of superheater	1.0
3	Plateout on feedwater shroud (above Kaowool)	1.0
4	Plateout on lower Kaowool coverplate	1.0
5	Plateout on outside of superheat support	1.0
6	Plateout on superheat tubesheet bundle	0.357
7	Plateout on reheat bundles (two sources)	0.518
8	Plateout on reheat piping (two sources)	0.0006
9	Plateout on reheat piping (two sources)	0.0006
10	Plateout on feedwater shroud (below Kaowool) ^(a)	0.2
11	Plateout on feedwater support ^(a)	0.2

(a) Assuming no purge flow or flow restriction.

were adjusted to accommodate the conversion from surface to equivalent volume sources. This conversion simplified the modeling of the tubes for sources 6 and 7. Shell sources 10 and 11 were determined on the basis of expected activities, because of the lower flow rates in these regions. Table 3-19 also gives the relative magnitude of each source with respect to the source strength given in Table 3-18. Shell sources 8 and 9 are plateout sources between the Kaowool thermal barrier and reheat penetration structure and piping. Source 8 is on the outside of the penetration support structure between the steam generator floor cavity liner¹ and level with the reheat tubesheet. Source 9 is on the outside of the reheat piping and covers the entire area between the bottom of the penetration (below the PCR²) and the tubesheet. Both sources 8 and 9 were assumed to have plateout activities limited by the fraction of total flow rate that passes through the reheater annulus.

The dose-rate results of this analysis based on the model described and the design plateout source spectrum given in Table 3-18 are presented in Fig. 3-9.

3.7.2. Median and 95%-Confidence-Level Dose Rates

Tube-plugging dose rates with median plateout values were calculated with the shielding specified in Fig. 3-9 in position. The plateout source spectrum was calculated by taking the total median circuit activity for the fission products of interest and dividing by the total plateout surface area in the 3000-MW(t) HTGR. This uniform plateout distribution was then adjusted, so the results correspond to plateout at the midpoint of the reheater. Table 3-20 presents the median plateout source spectrum. The dose rates obtained by a PATH analysis using this spectrum and the relative source strengths given in Table 3-19 are presented in Fig. 3-10.

Dose rates due to 95%-confidence-level plateout values were calculated with the model previously described and the specified shielding in place. For the plateout activities of interest, the sum of $\sigma_{\ln B}^2$ and $\sigma_{\ln R}^2$ is small compared with $\sigma_{\ln A_K}^2$, so the uncertainty in the dose rate for a

TABLE 3-20
 MEDIAN PLATEOUT SOURCE SPECTRUM FOR TUBE-PLUGGING STUDY
 (1-DAY DECAY)

Gamma-Ray Energy (MeV)	Strength (MeV/cm ² -sec)
0.1	1.28 + 5
0.3	7.15 + 5
0.6	2.33 + 6
0.8	1.13 + 6
1.0	9.34 + 4
1.25	1.94 + 5
1.6	1.04 + 5
2.0	1.36 + 4
2.4	4.44 + 3
2.9	7.38 + 1
3.5	2.63

Fig. 3-10. Dose rates in Mark II steam-generator lower penetrations due to median plateout values

particular isotope is essentially determined by the uncertainty in the plateout activity. The plateout spectrum used in the calculation is given in Table 3-21. The resulting tube-plugging dose rates are presented in Fig. 3-11.

3.7.3. Median and 95%-Confidence-Level Man-Rem Exposures

The average tube-plugging operation is expected to take approximately 60 man-hours. For the median and 95%-confidence-level dose rates at the superheater tubesheet, for example, the man-rem exposures per operation are 0.078 and 0.60, respectively. These exposures will vary for other locations in the lower penetration.

The important isotopes that contribute to the man-rem exposure are given in Table 3-22. Silver-110m is the most important isotope for the median value, and Cs-134 is the most important isotope for the 95%-confidence-level value. At the reheater, the I-132 median and 95%-confidence-level activities are much lower than the design activity. Consequently, I-132 is not important for the median and 95%-confidence-level man-rem exposures for tube plugging, even though it is important in studies involving design activities.

3.8. STEAM-GENERATOR REMOVAL AND REPLACEMENT

Steam-generator removal and replacement involve a number of operations, which are listed in Table 3-23. The information given in this table concerning the operations, the number of men required, the number of shifts required, and the man-hours is taken from Ref. 3-1.

As shown in Table 3-23, the main contribution to the man-rem exposure results from transfer of the steam generator out of the containment building.

TABLE 3-21
95%-CONFIDENCE-LEVEL PLATEOUT SOURCE SPECTRUM FOR
TUBE-PLUGGING STUDY
(1-DAY DECAY)

Gamma-Ray Energy (MeV)	Strength (MeV/cm ² -sec)
0.1	3.86 + 5
0.3	1.21 + 6
0.6	6.56 + 7
0.8	2.33 + 7
1.0	7.11 + 5
1.25	2.36 + 6
1.6	1.25 + 6
2.0	2.65 + 4
2.4	7.10 + 4
2.9	1.73 + 3
3.5	6.18 + 1

Fig. 3-11. Dose rates in Mark II steam-generator lower penetrations due to 95%-confidence-level plateout values

TABLE 3-22
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FOR TUBE PLUGGING AT
SUPERHEATER TUBESHEET

Isotope	Percent Contribution to Man-Rem Exposure	
	Median Value	95%-Confidence- Level Value
Ag-110m	40	4
Te-131m	20	3
La-140	20	5
Cs-134	15	85

TABLE 3-23
MANPOWER REQUIREMENTS AND MAN-REM EXPOSURES FOR STEAM-GENERATOR REMOVAL AND REPLACEMENT

Operation	No. of Men Required	No. of Shifts Required (a)	Man-Hours	Dose Rate for Different Sources (mrem/hr)			Exposure for Different Sources (man-rem)		
				Design (b)	Median	95%-Confidence-Level	Design (b)	Median	95%-Confidence-Level
Removal of main circulator	3	2	48		1.19 - 3	13.9	0.48	5.9 - 5	0.7
Removal of concrete plug and diffuser	2	2	32	10	0.24	3 + 3	0.32	7.6 - 3	96
Removal of compressor inlet duct	3	3	72	10	0.24	3 + 3	0.72	0.017	216
Removal of steam generator									
Preparation	2	5	80	10	2.5	650	0.8	0.2	52
Lifting from PCRV	2	2	32	10	0.24	3 + 3	0.32	7.6 - 3	96
Transfer out of containment	2	7	112	(100) (c)	65	1.4 + 5	(11.20) (c)	7.3	1.6 + 4
Installation of new steam generator									
Preparation	2	5	80	10	0.24	3 + 3	0.80	0.019	240
Lowering into PCRV	2	2	32	10	0.24	3 + 3	0.32	7.6 - 3	96
Reinstallation of compressor inlet duct	3	3	72	10	0.24	3 + 3	0.72	0.017	216
Reinstallation of concrete plug and diffuser	2	2	32	10	0.24	3 + 3	0.32	7.6 - 3	96
Reinstallation of main circulator	3	2	48		1.19 - 03	13.9	0.48 (d)	5.9 - 5	0.7
Total							16.5	7.6	1.7 + 4

(a) Eight hours per shift.

(b) The values in parentheses are based on expected activities.

(c) These values assume that shielding is adjusted so that the dose rate is 100 mrem/hr along the entire length of the cask.

(d) This value is from Ref. 3-3. Actual values for design sources should be lower.

3.8.1. Transfer of Steam Generator Out of Containment Building

3.8.1.1. Configuration and Shielding Thicknesses. The essential configuration of the Mark II steam generator for this calculation is shown in Fig. 3-12. The shielding requirements for the transfer of the steam generator out of the containment building were calculated on the basis of expected source activities with a decay time of 100 days. For purposes of this study, the expected plateout activities were assumed to be small with respect to the Co-60 activation activities over the bottom part of the steam generator. In the calculation, the steel shielding thicknesses were determined so that the dose rates on the outside surface of the shield would be equal to or less than 100 mrem/hr. The ratios of the plateout area to volume for the calculation are given in Table 3-24.

The results show that a steel wall thickness of 4.9 in. is required near the top of the evaporators and a wall thickness of 4.0 in. is required opposite the midpoint of SH1 and SH2 of Fig. 3-12. Consequently, the steel cask was assumed to have a wall thickness of 4.9 in. over the length of the evaporators and economizers and a wall thickness decreasing to 3.8 in. at the top of SH2.

3.8.1.2. Median and 95%-Confidence-Level Dose Rates. The median source activities for this calculation are given in Table 3-25. The calculated median dose rate was 330 mrem/hr opposite the evaporator inlet and 180 mrem/hr opposite the midpoint of superheaters SH1 and SH2.

In the calculation of the 95%-confidence-level dose rates, the sum of $\sigma_{\ln B}^2$ and $\sigma_{\ln \mu_{KR}}^2$ is negligible with respect to $\sigma_{\ln A_K}^2$ for the important isotopes. The calculated 95%-confidence-level dose rate was 660 rem/hr opposite the evaporator inlet and 44 rem/hr opposite the midpoint of superheaters SH1 and SH2. To achieve a dose rate of 100 mrem/hr for the given 95%-confidence-level activities, the thickness of steel opposite the evaporators must be increased from 4.9 to 11.0 in. and the thickness opposite the top of SH2 must be increased from 3.8 to 7.1 in. The increase

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KEY

EC1, EC2	ECONOMIZERS
EV1, EV2	EVAPORATORS
SH1, SH2, SH3	SUPERHEATERS

Fig. 3-12. Geometry for calculations of transfer of steam generator out of containment building

TABLE 3-24
RATIOS OF PLATEOUT AREA TO VOLUME FOR STEAM GENERATOR

Location (a)	cm^2/cm^3
EC1, EC2	0.425
EV1	0.401
EV2	0.371
SH1	0.298
SH2	0.730
SH3	0.151

(a) See Fig. 3-12.

TABLE 3-25
SOURCE STRENGTHS FOR STEAM-GENERATOR REMOVAL

Isotope	Source Strength at Various Locations(a) (Ci/cm ²)	
	Inlet of EV2	Middle of SH2, SH1
Zr-95	8.65 - 9	1.63 - 8
Nb-95	5.27 - 9	9.97 - 9
Ag-110m	2.22 - 7	3.86 - 7
Te-129m	1.08 - 7	2.04 - 7
Cs-134	2.38 - 4	2.96 - 5
Ba-137m	6.78 - 4	8.42 - 5
Pr-144	7.71 - 9	1.45 - 8

(a) See Fig. 3-12 for location.

in thickness is different at the two locations because the ratio of the 95%-confidence-level activity to the expected activity in the evaporator is much larger than that for SH1 and SH2.

Dose rates from plateout activities were calculated with the assumption of uniform occupancy times along the entire length of the steam generator. The variation in the plateout activities as a function of position was used in the calculation. For the shield wall thicknesses that are based on expected activities, the dose rate averaged over the length* was 65 mrem/hr for median activities and 140 rem/hr for the 95%-confidence-level activities.

3.8.1.3. Median and 95%-Confidence-Level Man-Rem Exposures. The man-rem exposures for the transfer of the steam generator out of the containment building were based on the dose rates averaged over the length of the cask and on the times given in Table 3-23. The resulting exposures are also given in Table 3-23. The important isotopes are listed in Table 3-26.

3.8.2. Other Operations for Steam-Generator Removal and Replacement

The other operations for steam-generator removal and replacement are given in Table 3-23. The calculations for the removal and replacement of the main circulator were done in a manner similar to those described in Section 3.6, except that 100-day decay after shutdown was used in this case to correspond to the decay time for steam-generator removal. The resulting dose rates and man-rem exposures are given in Table 3-23.

For the preparation of the steam generator for removal, the source activities were assumed to be similar to those at the superheater. Using the design dose rate shown in Table 3-23 and the ratios of the median and the 95%-confidence-level activities to the design activity for cesium, the

* The dose rate as a function of length was integrated with respect to length and divided by length to obtain an unweighted average value.

TABLE 3-26
CONTRIBUTION BY ISOTOPE TO MAN-REM EXPOSURE FOR TRANSFER OF
STEAM GENERATOR OUT OF CONTAINMENT BUILDING

Isotope	Percent Contribution to Man-Rem Exposure	
	Median Value	95%-Confidence- Level Value
Cs-134	70	70
Ba-137m	30	30

dose rates and exposures shown in Table 3-23 were obtained. The dose rates and man-rem exposures for the rest of the operations given in Table 3-23 were calculated in a similar manner, except that the activities for the circulator were used.

3.8.3. Median and 95%-Confidence-Level Man-Rem Exposures for Steam-Generator Removal and Replacement

The total man-rem exposures for these operations are given in Table 3-23. Since practically all the median and 95%-confidence-level man-rem exposures result from transferring the steam generator out of the containment, the percentage contribution by isotope to the total man-rem exposure will be essentially the same as given in Table 3-26. Cesium-134 and Ba-137m (from Cs-137) are the only important isotopes. In the case of the 95%-confidence-level dose rates and man-rem exposures, additional shielding or smaller 95%-confidence-level activities will be required to achieve reasonable values. Access control could also be used to reduce the man-rem exposure.

REFERENCE

- 3-1. "General Atomic Standard Safety Analysis Report," Amendment 4, August 1, 1975 (NRC Docket STN 50-535).

4. CONCLUSIONS

A study of the occupational dose rates and man-rem exposures from gas-borne and plateout activities in a reference 3000-MW(t) HTGR plant was performed. This study included calculations of the median and 95%-confidence-level man-rem exposures for the important operations or radiation sources and calculations of the most significant isotopic contributors for the important cases.

The man-rem exposures per year for the scheduled operations are given in the top part of Table 4-1. Both the median and 95%-confidence-level man-rem exposures for all the scheduled operations are reasonable with respect to the U.S. Code of Federal Regulations, 10CFR20, and NRC Regulatory Guide 8.8, except for the 95%-confidence-level man-rem exposure for reflector-block shipping. In the latter case, Cs-134 is the most important isotope, and the large 95%-confidence-level man-rem value is caused by the large uncertainties in the Cs-134 and Cs-137 plateout activities. For the gaseous activity in the containment building and for the analytic instrumentation, Kr-88 is the most important contributor, and all the other important contributors are also noble gases. For the moisture monitor system, Kr-88 is again the most important contributor, followed by Rb-88, Cs-138, and other isotopes. In the case of the helium regeneration system and the gas waste system, Kr-85 is the only isotope that makes a significant contribution to the man-rem exposures. The relatively small uncertainties in the noble-gas activities as well as the small uncertainties in the gamma-ray buildup factor and the macroscopic cross section result in relatively small ratios (between 1.6 and 1.9) of the 95%-confidence-level man-rem exposure to the median man-rem exposure for all the scheduled cases other than the reflector-block shipping. The inclusion of indeterminate uncertainties in the activities, in the exposure time, and in the point-kernel method would increase the 95%-confidence-level man-rem values significantly. Also, the inclusion of the

TABLE 4-1
MAN-REM EXPOSURES PER YEAR OR PER OPERATION FOR IMPORTANT CONTRIBUTORS

Scheduled Operation (or Radiation Source)	Man-Rem Per Year	
	Median	95%-Confidence Level
Gaseous activity in containment building	0.036	0.056
Moisture monitor system	0.39	0.76
Analytic instrumentation	0.61	0.98
Helium regeneration system	0.086	0.145
Gas waste system	0.043	0.073
Reflector-block shipping	0.21	1400

Unscheduled Operation	Man-Rem Per Operation	
	Median	95%-Confidence Level
Main-circulator removal	0.00038	0.78
Steam-generator tube plugging	0.078	0.60
Steam-generator removal and replacement	7.6	17,000

uncertainty in the leak rate through the PCRV would increase the uncertainty in the man-rem exposure for the containment building.

The three unscheduled operations considered in detail are given in the bottom part of Table 4-1. The man-rem exposures for the main-circulator removal and the steam-generator tube plugging are reasonable. For the main-circulator removal, the median man-rem exposure is unusually small because of a conservative cask design; I-132 is the most important isotope for the median case, followed by Cs-134. The conservative cask design also results in a reasonable value for the 95%-confidence-level exposure, even though the uncertainties in the Cs-134 and Cs-137 plateout activities are very large. For the tube-plugging operation, Ag-110m is the most important isotope for the median man-rem exposure, and Cs-134 is the most important isotope for the 95%-confidence-level value. The median man-rem exposure for the steam-generator removal and replacement is reasonable for this operation, but the 95%-confidence-level exposure is extremely large. Again, the very large uncertainties in the important Cs-134 and Cs-137 plateout activities are responsible. To achieve reasonable 95%-confidence-level man-rem exposures for these two cases, either the 95%-confidence-level activities should be reduced or the shielding thicknesses should be increased. In the future, additional experimental data and additional calculations should reduce the uncertainties in the cesium activities significantly.