70-687
50-54

CINTICHEM, INC.
P.O. BOX 816
TUXEDO, NEW YORK 10987 (914) 351-2131

August 6, 1991

Mr. Dominick A. Orlando, Project Manager
Decommissioning and Regulatory Issues Branch
Div. of Low Level Waste Management and
Decommissioning
Office of Nuclear Material Safety and
Safeguards
U. S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

Dear Mr. Orlando:

Reference: USNRC NMSS Letter Dated July 30, 1991

This letter contains the additional information to be included in our Decommissioning Plan as requested in the referenced letter.

NRC QUESTION #1

Please provide an estimate of the activity of the Sr-90 and transuranic radionuclides in the liquid effluents and include these in the calculation of dose from liquid effluents.

NRC QUESTION #3

Item 8 of this response also indicates a maximum potential individual dose of 0.9 mrem/yr from liquid effluents. This calculation does not include the effluent from the hot cell washdown and assumes the release of only 0.055 mCi (excluding K-40).

Cintichem has committed to keeping the concentration of radioactive material in liquid effluents below the limits specified in Appendix B, Table II of 10 CFR 20 and the total activity released in liquid effluents below 10 mCi/yr. Please provide a dose calculation using a conservative environmental pathway analysis and the assumption that the entire 10 mCi is released in one year.

RESPONSE TO QUESTIONS 1 AND 3

Attachment A partially answers questions 1 and 3, and also updates the January 11, 1991 NRC submittal on liquid effluents. It adds the liquid effluent from cell wash down after processing, Tritium from the pool, and transuranics. Quantities are in uCi or uCi/cc in the assumed 506,000 gallons (1.915E9cc).

JJM/150.91B

ADD: Nick Orlando Ltr. Encl.
, ,
NL16 1/1

RESPONSE TO QUESTIONS 1 AND 3 (continued)

Note that decommissioning liquid waste contains 150,000 uCi of Tritium plus 514 uCi of the other isotopes. Our outfall (001) also includes contribution from the radiopharmaceutical production facility of approximately 101 uCi of Mo-99 and 537 uCi of Tc-99m in process water effluent each year.

The decommissioning liquid effluent mixture yields a potential dose of 5.9 mRem (as calculated in Attachment A) assuming no dilution before drinking, and that the maximally exposed individual's entire water consumption originates at the 001 outfall pipe. However, it is unrealistic to assume that there is no dilution water. The effluent is diluted by site process water and storm drain runoff, both of which are discharged through 001. The annual estimate for the former is 1.0×10^7 gallons/yr and that for the latter 5.0×10^7 gallons/yr.

Therefore, the potential dose due to decommissioning liquid effluent is actually 0.05 mRem.

$$\frac{5.06 \times 10^5 \text{ gallons}}{6.05 \times 10^7 \text{ gallons}} \times 5.9 \text{ mRem} = \underline{0.05 \text{ mRem}}$$

If our non-tritium effluent were to increase to near 9.4 mCi (0.6 mCi due to existing Tc-99m generator production) the maximum potential offsite dose due to decommissioning would be 0.92 mRem.

$$\frac{9.4 \text{ mCi}}{0.51 \text{ mCi}} \times 0.05 \text{ mRem} = \underline{0.92 \text{ mRem}}$$

Although not accounted for in this calculation, we can be reasonably assured that any actual doses to an offsite individual would be due to drinking water from the Indian Kill stream. The stream flow is estimated at 7×10^8 gallons/yr and would reduce the calculated doses shown here by approximately a factor of ten.

NRC QUESTION #2

The information presented in Item E of Cintichem's response to NRC's request for additional information (RAI) dated January 11, 1991 suggests that the hot cell walls are contaminated with significant quantities of Ce-144 as well as Cs-137 and Sr-90. Please calculate, or measure, the Ce-144 activity on the hot cell walls and recalculate the projected maximum individual dose from airborne effluents, including Ce-144, resulting from the scabbling of the hot cell walls.

RESPONSE:

In the Decommissioning Plan's Emission Report, Attachment A, Case II, an estimate of the "maximum dose to the closest residential individual offsite due to environmental release caused by scabbling of the hot cell walls" was performed. This was done by using a dose rate measurement of 55 mRem/hr that was taken after a cleaning operation in cell 1 and calculating a uCi/cm^2 activity assuming all the dose rate was from Cs-137.

In addition, the same activity (uCi/cm^2) was assumed for Sr-90.

If we were to consider the 55 mRem/hr dose rate to be due to all the isotopes listed in the January 11, 1991 NRC submittal, the ratios to previously calculated doses would be as shown in Attachment B to this letter.

In summary, when the mix of isotopes listed in the January 11, 1991 NRC submittal is considered as the isotopic mix in the 55 mRem/hr dose rate due to contamination of hot cell walls during scabbling instead of only Cs-137, all corresponding dose rate or dose calculations can be corrected by a factor of 0.25.

We trust that this additional information satisfies those questions posed in the referenced letter regarding effluents and estimated exposures therefrom.

As a result of the supplemental information about liquid effluents contained in this and other correspondence since the Decommissioning Plan was initially submitted for review and approval, technical specification 7.1.3 shall now state:

7.1.3 Liquid waste released from the site shall comply with the limits specified in 10 CFR Part 20, Appendix B, Table II. Additionally the total annual release of H3 shall be less than 150 mCi and the total annual release of all other radionuclides shall be less than 10 mCi.

Very truly yours,

J. J. McGovern,
President/Plant Manager

JJMcG/bjc

Attachments

JJM/150.91B

cc: Dr. Paul J. Merges, Director
Bureau of Radiation, DHSR
New York State Department of
Environmental Conservation
50 Wolf Road
Albany, NY 12233-7255

Dr. Francis J. Bradley
Principle Radiophysicist
New York State Department of Labor
One Main Street, Room 813
Brooklyn, NY 12223

Ms. Annette Dorozynski
Supervisor
Town of Tuxedo
P. O. Box 725
Tuxedo, NY 10987

Director, Technical Development Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, NY 12223

Ms. Ava Gartner
Berle, Koss and Case
145 Rockefeller Plaza
New York, NY 10111

JJM/150.91B

ATTACHMENT /

	A	B	C	TOTAL	CONC.	uCi ingested**	uCi per 5 Rem***	Potential Internal Dose [Rem]
	uCi	uCi	uCi	uCi	uCi/cc			
Ce-134	1.6	.0015		1.6015	8.4E-10	6.1E-4	7E+1	4.4E-5
Cs-137	1.3	.0058	7.1	8.4058	4.4E-9	3.2E-3	1E+2	1.8E-4
Ce-144	189.6	.0210	25.2	214.8210	1.1E-7	8.0E-2	3E+2	1.3E-3
Nb-95	2.1	.0030	5.0	7.1030	3.7E-9	2.7E-3	2E+3	6.8E-6
Zr-95	100.0	.0250		100.0250	5.2E-8	3.8E-2	1E+3	1.9E-4
Sb-125	138.8	.0066	5.6	144.4066	7.5E-8	5.5E-2	2E+3	1.4E-4
Co-60			12.6	12.6000	6.6E-9	4.8E-3	5E+2	4.8E-5
K-40			16.2	16.2000				
Sr-90	1.3	.0058	7.1	8.4058	4.4E-9	3.2E-3	4E+1	4.0E-4
H-3			150000	150000 *	7.8E-5	5.7E+1	8E+4	3.6E-3
Pu-239	1.1E-4		6.0E-4	7.1E-4	3.7E-13	2.7E-7	1E+0	1.4E-6
Pu-240	4.9E-7		2.7E-6	3.2E-6	1.7E-15	1.2E-9	1E+0	6.0E-9
Pu-241	5.9E-14		3.2E-13	3.3E-13	2.0E-22	1.5E-16	7E+1	1.1E-17
Pu-242	9.9E-15		5.4E-14	6.4E-14	3.3E-23	2.4E-17	1E+0	1.2E-16
Am-241	1.1E-11		6.0E-11	7.1E-11	3.7E-20	2.7E-14	1E+0	1.4E-13
Cm-242	4.8E-12		2.6E-11	3.1E-11	1.6E-20	1.2E-14	5E+1	1.2E-15
U-235	1.9E-3		1.0E-2	1.2E-2	6.3E-12	4.6E-6	2E+1	1.2E-6
U-234	5.8E-2		3.2E-1	3.8E-1	2.0E-10	1.5E-4	2E+1	3.8E-5
U-236	1.7E-5		9.3E-5	1.1E-4	5.7E-14	4.2E-8	2E+1	1.1E-6
				150514				5.93E-3 Rem or 5.9 mRem

A. Cell wash down effluent after deionization <u>and</u> evaporation processes [uCi].

B. Effluent from floor drains, showers, sinks and other miscellaneous sources after evaporation. These data are directly from the January 11, 1991 NRC submittal plus Sr-90 [uCi].

C. Reactor pool/canal water after deionization. These data again are directly from the January 11, 1991 NRC submittal plus Sr-90 and H-3 [uCi].

* an exemption to the NYSDEC 10 mCi/yr license limit is being requested.

** based on drinking 7.3×10^5 ml annually [water intake of "Reference Man"]

*** ALI values from 10 CFR 20 Appendix B, Table 1, Col. 1

JJM/154.918

ATTACHMENT B

The inhalation dose from a mixture of fission isotopes which include Ce-144 is compared to that resulting from a mixture of equal amounts of Cs-137 and Sr-90. The former dose is 0.25 that of the latter. The approach uses the concept of "annual limit of intake (ALI)" in the May 21, 1991 Final Rule of 10 CFR 20.

The Ci basis for the CASE 1 Ce-144 mixture is shown in Column I. The quantity of each gamma emitting isotope relative to Cs-137 is listed in Column II. A gamma factor for each isotope is listed in Column III.

CASE 1 (Ce-144 mixture)

	I. Ci Basis	II. Quantity Relative to Cs-137	III. Gamma Factor
Cs-137	0.202	1.00	3.3
Cs-134	0.019	0.09	8.7
Ce-144	4.016	19.88	0.4
Nb-95	3.620	17.92	4.2
Zr-95	1.780	8.81	4.1
Sb-125	0.120	0.59	2.7
Sr-90	0.202	-	-

For a Case 2 mixture of equal amounts of Sr-90 and Cs-137 the table is as follows:

CASE 2 (Cs-137 and Sr-90)

	I. Quantity Relative to Cs-137	II. Gamma Factor
Cs-137	1.00	3.3
Sr-90	1.00	-

Let a reference dose rate for CASE 2 be 3.3(x) where x equals Cs-137 radioactivity and 3.3 is the gamma factor. Then an identical gamma dose rate could be expected due to CASE 1 isotopes as follows:

$3.3(y) + 8.7(0.09y) + 0.4(19.88y) + 4.2(17.92y) + 4.1(8.81y) + 2.7(0.59y)$

where y is the CASE 1 quantity of Cs-137. Set 3.3x equal to the above to obtain the relationship between the Cs-137 quantities for the two CASES.

$$3.3x = 125.0\ y \quad \text{or} \quad x = 37.9\ y$$

The CASE 2 Cs-137 radioactivity is 37.9 times greater than that for CASE 1.

Note that the number of "annual limits of intake (ALI)" is equivalent to dose for each radioisotope. The total relative dose is the sum of the ALIs for each CASE. A dimensionless quantity for radioactivity facilitates the comparison of dose between CASE 1 and CASE 2.

CASE 1

	I. Radioactivity Units	II. Radioactivity* Units per ALI	III. # ALIs
Cs-137	1.00	2×10^2	5.0×10^{-3}
Cs-134	0.09	1×10	9.0×10^{-4}
Ce-144	19.88	1×10^1	2.0×10^0
Nb-95	17.92	1×10^3	1.8×10^{-2}
Zr-95	8.81	1×10^2	8.8×10^{-2}
Sb-125	0.59	5×10^2	1.2×10^{-3}
Sr-90	1.00	4×10^0	2.5×10^{-1}
		Total #ALI (relative scale)	2.4×10^0

CASE 2

	I. Radioactivity Units	II. Radioactivity* Units per ALI	III. # ALIs
Cs-137	37.9	2×10^2	1.9×10^{-1}
Sr-90	37.9	4×10^0	9.5×10^{-0}
		Total #ALI (relative scale)	9.7×10^0

CONCLUSION

The internal dose from CASE 1 radioisotopes would be 0.25 that of CASE 2 radioisotopes on assumption of equalized external gamma radiation dose rates from both mixtures.

$$\frac{\text{CASE 1 \# ALI}}{\text{CASE 2 \# ALI}} = \frac{2.4}{9.7} = 0.25$$

Therefore, all doses calculated with a 50-50 mixture (Ce-137 and Sr-90) should be multiplied by 0.25 to obtain the corresponding dose from the Ce-144 (CASE 1) mixture.

* Based on 10 CFR 20 Final Rule (May 21, 1991) Inhalation ALI Appendix B, Table 1, Col. 2 Stochastic values.

JJM/150.91B