July 10, 2014

MEMORANDUM TO: James Clifford, Director
Division of Nuclear Materials Safety
Region I

FROM: Gregory Suber, Deputy Director (Acting) **/RA/**
Environmental Protection and
Performance Assessment Directorate
Division of Waste Management
and Environmental Protection
Office of Federal and State Materials
and Environmental Management Programs

SUBJECT: RESPONSE TO TECHNICAL ASSISTANCE REQUEST, DATED MARCH 24, 2014, FOR THE REVIEW OF DERIVED CONCENTRATION GUIDELINE LEVELS FOR THE RELEASE OF THE PFIZER, INC., ECLIPSE CYCLOTRON VAULT, GROTON, CONNECTICUT

Region I submitted a Technical Assistance Request, dated March 24, 2014, requesting a review of site-specific derived concentration guideline levels (DCGLs) developed in support of license termination of the cyclotron facility operated within the Pfizer Inc. facility in Groton, Connecticut. Region I also requested a review of the licensee's method of determining compliance with the U. S. Nuclear Regulatory Commission (NRC) release limits for unrestricted use by using one set of DCGL values for surface activity and another for surface and volumetric activity combined. The licensee discusses the development of the proposed site-specific DCGL values and the use of the surface contamination screening values in "Pfizer, Inc. Eclipse Cyclotron Facility Decommissioning Final Status Report," which is included in the request for license termination package [ML14094A454].

The enclosed Technical Evaluation Report discusses the Performance Assessment Branch's review of the use of default screening values in areas without volumetric contamination and the development of site-specific DCGL values for areas with both surface and volumetric contamination. Based upon this review, the staff finds that the published default screening values listed in NUREG-5512, Vol. 1 are appropriate for surface contamination in areas without volumetric contamination at this site.

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The staff also found that the licensee-generated DCGLs for surface and volumetric contamination combined satisfy the NRC dose criteria for license termination under a building occupancy scenario at this site. However, additional information should be provided by the licensee about the volumetric contamination in steel remaining at the site to allow the NRC staff to determine if the residual contamination meets the NRC dose criteria in alternate scenarios.

If you have any questions regarding this review, please contact Christianne Ridge of my staff. She can be reached at 301-415-5673 or christianne.ridge@nrc.gov.

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Enclosure:
Technical Evaluation Report

cc w/enclosure:
Elizabeth Ullrich, Region 1

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Building 274 Cyclotron Vault Technical Evaluation Report
Prepared by: Christianne Ridge, Sr. Systems Performance Analyst

1. Background

Pfizer, Inc. has requested unrestricted release of the cyclotron vault in Building 274 of the Pfizer Inc., facility in Groton, Connecticut. Although the cyclotron has been removed, detectable concentrations of Co-60, Cs-134, and Eu-152 remain in the cyclotron vault. The remaining material is considered part of the Group 2 Decommissioning Group per NUREG-1757. Group 2 facilities are not required to submit a decommissioning plan. However they must demonstrate that the site meets U.S. Nuclear Regulatory Commission (NRC) screening criteria for residual radioactive material remaining at the site. To meet NRC requirements for the release of the cyclotron vault in Building 274 as acceptable for unrestricted use, all licensed material must be removed, areas decontaminated as necessary, and a close out survey must be performed.

The licensee used default screening values for surface contamination and developed site-specific DCGLs for areas that could contain both surface contamination and volumetric activation products. The NRC staff reviewed the licensee's use of the default screening values for surface contamination and verified that they are appropriate for areas of the site without volumetric activation products. For areas with volumetric activation products, the licensee developed site-specific DCGLs that accounted for both surface and volumetric contamination. The site-specific DCGLs were based on the building occupancy scenario described in NUREG-5512, Vol. 1. NRC staff reviewed the licensee's calculations for the building occupancy scenario and found the DCGLs to be appropriate for that scenario.

In addition, the licensee used NUREG-1640 to estimate the dose from the residual contamination in a variety of alternate scenarios, including renovation, demolition, recycling, and disposal. The licensee concluded that doses from residual contamination in the cyclotron vault would be much less than the 25 mrem/yr dose limit. The NRC staff reviewed the licensee's calculations and verified the conclusion for contaminated concrete at the site. However, the NRC staff cannot verify the licensee's conclusions about alternate scenarios for the steel remaining in the cyclotron vault without additional information about the activation products in the steel.

2. Source Term

The cyclotron was licensed to produce short-lived radionuclides for Positron Emission Tomography (PET) imaging. These include C-11 and F-18, which are no longer a contamination concern because of their short half-lives. Cyclotron operation, however, also created activation products in concrete and steel at the facility. Radionuclides of concern were determined by gamma spectroscopy of concrete samples from the floor taken at the areas of highest activity as determined by surfaces scans. Gamma spectroscopy of the concrete floor samples detected Co-60, Cs-134, and Eu-152. Eu-134 was not detected but was included in the analysis.

The main source of surface contamination is expected to be concrete dust resulting from removing embedded fixtures and taking concrete core samples. Removal of embedded fixtures included removing the aluminum pan that lined the cyclotron pit and steel tracks for the movable

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shields. The aluminum pan and steel tracks were packaged for disposal. The licensee reported that samples of steel were not available for sampling. However, Section 14.1 of the licensee's submittal indicates that potentially activated steel remains in the cyclotron vault. The licensee did not characterize the mass of the remaining steel except to indicate it is "far less than 1 metric ton". Because the elemental composition of steel and concrete differ, they are expected to have different activation products. Thus the potential source term associated with the remaining steel at the site is unclear.

3. Review and Analysis of Proposed DCGLs for Areas with Surface Contamination Only

The licensee applied surface DCGLs in areas that do not have volumetric activation products. Because surface contamination is expected to result from concrete dust, the source term determined by the licensee based on concrete samples is appropriate for the development of surface DCGL values.

The licensee developed surface DCGLs based on two sources: (1) the default screening values provided in NUREG-1757 v. 1 Rev. 2 Table B.1 and (2) values calculated with the building occupancy scenario of DandD. NUREG-1757 Table B.1 values are provided for Co-60 and Cs-134. Because default screening values for Eu-152 and Eu-154 are not provided in that table, the licensee calculated default screening values for Eu-152 and Eu-154 with the building occupancy scenario of DandD using default parameters. The NRC staff duplicated these calculations with DandD Version 2.2.

Rather than individually identifying radionuclides and applying the sum of fractions rule, the licensee chose to apply the default screening value for Co-60 (i.e., 7100 dpm/100 cm^2) because it was the most limiting value (Table 1). The NRC staff agrees this approach is acceptable for this site. Because the default screening values for surface contamination were developed based on the assumption that the removal fraction of surface contamination is less than 10% of the total surface contamination, the licensee established the DCGL for removable surface contamination ($DCGL_{Removable}$) for areas without volumetric contamination as 10% of $DCGL_{Total}$. The NRC staff agrees this is an appropriate way to set the $DCGL_{Removable}$ for areas without volumetric activation products.

Table 1. Surface Activity Default Screening Values

Nuclide	mrem/yr per dpm/100 cm^2	Default Screening Value (dpm/100 cm^2)
Co-60	3.55×10^{-3}	7.1×10^{3}
Cs-134	1.96×10^{-3}	1.3×10^{4}
Eu-152	1.97×10^{-3}	1.3×10^{4}
Eu-154	2.18×10^{-3}	1.1×10^{4}

4. Review and Evaluation of the Proposed DCGLs for Areas with Surface and Volumetric Contamination

For areas where building structures could contain activation products, the licensee developed a $DCGL_{Total}$ that accounted for both surface contamination and volumetric activation products. The combined $DCGL_{Total}$ was developed to account for two components:

(1) direct exposure from both volumetric activation products and surface contamination combined resulting in a dose less than 24 mrem/yr; and

(2) inhalation and ingestion of removable surface contamination resulting in a dose less than 1 mrem/yr.

The licensee also considered alternate scenarios described in NUREG/CR-1640 to demonstrate that the residual contamination would not exceed the 25 mrem/yr dose limit in alternative scenarios. These alternate scenarios included renovation, demolition, recycling, and disposal.

4.1. Building Occupancy Scenario

4.1.1. External Dose

For areas with both volumetric and surface contamination, the licensee developed a dose rate limit for volumetric and surface contamination combined. The licensee based this combined $DCGL_{Total}$ on external dose alone. This approach was acceptable because the licensee based the $DCGL_{Total}$ on a dose of 24 mrem/yr and provided a separate demonstration that the internal dose from removable contamination would cause a committed effective dose equivalent of less than 1 mrem/yr.

The default assumption of the building occupancy scenario in NUREG-5512, Vol. 1, is that an individual could be exposed to contamination for 2340 hours per year. Based on an annual dose of 24 mrem/yr, the resulting dose rate limit is 24 mrem per year divided by 2340 hours per year, which results in a dose rate limit of 10.3 µrem per hour. The NRC staff agreed this was an appropriate external dose rate limit for areas with surface contamination and volumetric activation products.

4.1.2. Internal Dose

For areas with both surface contamination and volumetric activation products, the licensee developed a $DCGL_{Removable}$ for removable contamination using the building occupancy scenario of DandD with default parameters. The licensee used DandD to generate inhalation and ingestion doses for unit concentrations for Co-60, Cs-134, Eu-152, and Eu-154 (Table 2). NRC staff verified these values with DandD Version 2.2.

Because DandD assumes 10 percent of the surface contamination is removable, using unit concentrations is equivalent to calculating doses resulting from a removable concentration of 0.1 decays per minute (dpm) per 100 cm^2. Dividing 0.1 dpm per 100 cm^2 by the projected total internal dose (i.e., inhalation plus ingestion), the licensee calculated the removable activity equivalent to 1 mrem/yr internal dose. NRC staff verified the resulting DCGL values calculated by the Licensee.

Rather than individually identifying radionuclides and applying the sum of fractions rule, the licensee chose to apply the $DCGL_{Removable}$ for Eu-154 (i.e., 150 dpm / 100 cm^2) because it was the most limiting value for internal dose. The NRC staff agrees this approach is acceptable.

Table 2 Removable Activity Contribution to Total Dose

Nuclide	Annual Dose (mrem per removable 0.1 dpm/100 cm^2)			Removable Activity Equivalent to 1 mrem/yr Internal Dose (dpm/100 cm^2)
	Inhalation	Ingestion	Total Internal	
Co-60	4.87×10^{-4}	2.93×10^{-5}	5.16×10^{-4}	1.94×10^{2}
Cs-134	9/34 x 10^{-5}	7.21×10^{-5}	1.66×10^{-4}	6.04×10^{2}
Eu-152	5.12×10^{-4}	7.32×10^{-6}	5.19×10^{-4}	1.93×10^{2}
Eu-154	6.54×10^{-4}	1.06×10^{-5}	6.65×10^{-4}	1.50×10^{2}

4.2 Alternate Scenarios

The licensee evaluated alternate scenarios to verify that residual contamination would not exceed the dose limit in scenarios other than the building occupancy scenario. NUREG-1640 considers a wide range of scenarios, including renovation, demolition, recycling, and disposal. The licensee considered steel and concrete separately.

For each material (i.e., steel or concrete), the licensee chose the scenario from NUREG-1640 that yielded the highest dose for each radionuclide (Table 3). The licensee used the scenario dose conversion factor provided in the NUREG to calculate a dose based on the highest measured concentration of each radionuclide in samples of concrete.

The scenario dose conversion factors calculated in NUREG-1640 were based on the volumes of concrete and steel resulting from decommissioning a nuclear power plant. These volumes were much larger than the volumes of contaminated material present in the cyclotron vault. Therefore, the licensee adjusted the estimated doses by the ratios of the masses of contaminated concrete and steel in the cyclotron vault to the concrete and steel masses used in NUREG-1640. In each calculation, the licensee conservatively chose the lowest volume used in the range of volumes considered in NUREG-1640 and an upper estimate of the volume of contaminated structural material in the cyclotron vault.

Table 3 Scenario Dose Conversion Factors (DCFs) and Critical Groups for Alternate Scenarios

Nuclide	Concrete		Steel	
	Mass-Based DCF (mrem/yr per pCi/g)	Critical Group	Mass-Based DCF (mrem/yr per pCi/g)	Critical Group
Co-60	1.073	Road Building	0.1924	Scrap Yard
Cs-134	0.592	Road Building	0.1628	Electric Arc Furnace Dust – Dump Trailer
Eu-152	0.444	Road Building	0.0814	Scrap Yard

4.2.1. Concrete

The lower end of the range of volumes considered in NUREG-1640 was 143,000 metric tons of concrete cleared in 1.7 years. Rounding 1.7 years to two years, the lower bound of the annual contaminated concrete removal rate used in the NUREG-1640 is 7.15 x 10^3 metric tons in one year. The licensee estimated that less than 41 metric tons of potential activated concrete remained at the site, and adjusted the dose by a factor of 41 metric tons divided by 7.15 x 10^3 metric tons, or 0.06% (Table 4). The NRC staff duplicated these calculations.

To calculate doses in alternate scenarios, the licensee used the highest mass-based scenario dose conversion factor from NUREG-1640 adjusted for an applicable volume, with the highest concentrations of Co-60, Cs-134, and Eu-152 in concrete samples from the site. Because the licensee based the radionuclide concentrations on samples of concrete, the source term used by the licensee is appropriate for this calculation.

Although the calculations in NUREG-1640 involve many approximations, because the dose results were orders of magnitude below the 25 mrem/yr dose limit, adjustments of the approximations involved in the NUREG-1640 scenarios are unlikely to cause the estimated doses to exceed the dose limit. Therefore, the NRC staff determined this approach was adequate for demonstrating volumetrically contaminated concrete at the site would meet the 25 mrem/yr unrestricted release limit in these alternate scenarios.

Table 4. Alternate Scenario Dose Estimate for Volumetrically Contaminated Concrete

Nuclide	Average Activity Concentration (pCi/g)	Mass-Based Scenario DCF (mrem/yr per pCi/g)	Dose Using NUREG-1640 Volume Assumptions (mrem/yr)	Volume-Corrected Dose (mrem/yr)
Co-60	0.63	1.073	0.68	3.9×10^{-4}
Cs-134	0.12	0.592	0.07	4.1×10^{-5}
Eu-152	0.72	0.444	0.32	1.8×10^{-4}
		Total	1.07	6.1×10^{-4}

4.2.2. Steel

The lower end of the range of volumes considered in NUREG-1640 was 14.7×10^3 metric tons of concrete cleared in 1.7 years. Rounding 1.7 years to two years, the lower bound of the annual contaminated concrete removal rate used in the NUREG-1640 is 7.3×10^3 metric tons in one year. The licensee estimated that less than one metric ton of potential activated steel remained at the site, and adjusted the dose by a factor of 1 metric ton divided by 7.3×10^3 metric tons, or 0.02% (Table 5). The NRC staff duplicated these calculations.

Table 5. Alternate Scenario Dose Estimate for Volumetrically Contaminated Steel

Nuclide	Average Activity Concentration (pCi/g)	Mass-Based Scenario DCF (mrem/yr per pCi/g)	Dose Using NUREG-1640 Volume Assumptions (mrem/yr)	Volume-Corrected Dose (mrem/yr)
Co-60	0.63	0.192	0.12	6.8×10^{-4}
Cs-134	0.12	0.163	0.02	1.1×10^{-4}
Eu-152	0.72	0.081	0.06	3.3×10^{-4}
		Total	0.20	1.1×10^{-3}

As discussed in Section 2, because the elemental composition of steel and concrete differ, they are expected to have different activation products. Thus, the source term that the licensee used, which was based only on samples of concrete, may not represent the contaminant concentrations in activated steel in the cyclotron vault. Therefore, the concentrations of Co-60, Cs-134, and Eu-134 that the licensee measured in concrete could underrepresent the concentrations of those radionuclides in steel. In addition, it is possible that the estimated dose should include other radionuclides in addition to Co-60, Cs-134, and Eu-152.

The Licensee's surface scans performed with a sodium-iodide (NaI) detector and tissue equivalent gamma scintillation meter provide assurance that gamma radiation at the accessible steel surfaces was not significantly higher than gamma radiation on the concrete surfaces. However, insufficient information was presented about the volumetric concentration. In addition, it is unclear whether there could be significant concentrations of alpha or beta-emitting radionuclides that are common activation products in steel (e.g., Ni-63) in the steel, which could become inhalation or ingestion hazards in alternate scenarios such as recycling. For this reason, the NRC staff was unable to determine if volumetric activation products in steel could present a hazard in alternate scenarios based on the information provided by the licensee.

4.3. Independent Analyses

In addition to duplicating the licensee's calculations, the NRC staff performed independent comparisons to evaluate the hazard posed by potential contamination of steel with volumetric activation products in alternate scenarios.

Although the licensee has not presented information about the activity concentrations of activation products in volumetrically contaminated steel at the site, the licensee's surface gamma scans provide some assurance that activities of other gamma-emitting steel activation products (e.g., Mn-54, Co-58, Fe-59) are not significantly greater than the licensee's measured activity of Co-60. Based on examination of NUREG-1640 Table 2.1, the scenario dose conversion factors for Mn-54, Co-58, and Fe-59 are comparable to the scenario dose conversion factor for Co-60. Furthermore, no scenario dose conversion factor for any radionuclide exceeds the scenario dose conversion factor for Co-60 by more than a factor of 7. Therefore, because the estimated dose based on Co-60 is several orders of magnitude lower than the 25 mrem/yr dose limit, it appears unlikely that the dose limit would be exceeded by gamma-emitting radionuclides in volumetrically contaminated steel at the site.

The licensee provided less information to bound the concentrations of alpha- or beta- emitting activation products in the steel (e.g., Ni-63), which could present an inhalation or ingestion hazard in alternate scenarios. Therefore, the licensee should provide justification for assuming that the source term developed based on samples of concrete is representative of the volumetrically contaminated steel at the site or characterize the source term represented by the steel.

5. Summary of DCGL Values

The licensee created two sets of DCGL values, one for areas that only have surface contamination and one for areas with both surface contamination and volumetric activation products (Table 6). The areas where each set of DCGLs was applied were not designated in the licensee's submittal. However, in practice, the licensee applied both $DCGL_{Total}$ limits (i.e., 7100 dpm per 100 cm^2 and 10 μrem/hr) to the entire survey unit. The licensee also applied the more restrictive of the two $DCGL_{Removable}$ value (i.e., 150 dpm per 100 cm^2) to the entire survey unit. The NRC staff determined this approach is acceptable.

Table 6 Summary of DCGL values

Area	$DCGL_{Total}$	$DCGL_{Removable}$
Areas without activated structures (surface contamination only)	7100 dpm per 100 cm^2	710 dpm per 100 cm^2
Areas with activated structures (surface and volumetric contamination)	10 μrem per hour	150 dpm per 100 cm^2

6. Conclusions

The NRC staff reviewed the licensee's approach of developing one set of DCGL values for surface contamination and another set of DCGL values that accounted for both surface and volumetric contamination and found the approach to be acceptable. The NRC staff reviewed the licensee's selection of DCGL values for total and removable contamination in areas without activated components and found the selection of $DCGL_{Total}$ of 7100 dpm per 100 cm^2 and $DCGL_{Removable}$ of 710 dpm per 100 cm^2 based on default screening values for Co-60 to be acceptable for this site.

The NRC staff reviewed the licensee's approach to develop one set of DGCL values that combines surface and volumetric contamination to apply to areas with both surface contamination and volumetric activation products. Although the areas where each set of DCGL values (i.e., surface contamination only as compared to surface contamination and volumetric contamination) were not delineated in the submittal, the licensee appears to have applied both $DCGL_{Total}$ values (i.e., 7100 dpm per 100 cm^2 and 10 μrem per hour) to the entire site, as well as the more restrictive of the two $DCGL_{Removable}$ values (i.e., 150 dpm per 100 cm^2). The NRC staff found this approach to be acceptable. The NRC staff found the $DCGL_{Total}$ value of 10 μrem/hr and $DCGL_{Removable}$ value of 150 dpm per 100 cm^2 would limit doses to below 25 mrem/yr in a building occupancy scenario.

The NRC staff reviewed the licensee's analysis of the potential dose that could be caused by residual contamination in concrete and steel remaining in the cyclotron vault under alternate scenarios. The NRC staff agreed with the licensee's conclusion that the residual contamination in concrete at the site is not expected to exceed the dose limit in alternate scenarios.

However, because the activation products in steel are expected to differ from the activation products in concrete, and because the licensee did not characterize the radionuclides present in the remaining steel at the site, the NRC staff was unable to determine if the licensee's analysis included all of the radionuclides that could be relevant to recycling or disposal of activated steel from the site. Therefore, the NRC staff was unable to verify the licensee's conclusion that the residual contamination in steel at the site would meet the dose limit in alternate scenarios. This conclusion could be verified if the licensee provides justification for the assumption that the source term developed based on samples of concrete represents or bounds the source term in the remaining steel at the site. Alternately, the licensee could perform additional calculations based on a revised source term for the steel remaining at the site.

7. References

Anigstein, R., H. J. Chmelynski, D. A. Loomis, S. F. Marschket, J. J. Mauro, R. H. Olsher, W. C. Thurber, and R. A. Meck, 2003. "Radiological Assessments for Clearance of Materials from Nuclear Facilities, Main Report," NUREG-1640, Vol. 1. Prepared by SC&A, Inc. and Gemini Consulting Company for the U.S. Nuclear Regulatory Commission, Washington, D.C.

Banovac, K.L., J.T. Buckley, R.L. Johnson, G.M. McCann, J.D. Parrott, D.W. Schmidt, J.C. Sheperd, T.B. Smith, P.A. Sobel, B.A. Watson, D.A. Widmayer, and T.H. Youngblood, 2006. "Consolidated Decommissioning Guidance: Decommissioning Process for Materials Licensees, Final Report," NUREG-1757, Vol. 1, Rev. 2. U.S. Nuclear Regulatory Commission, Washington, D.C.

Kennedy, W.E. and D.L. Strenge, 1992. "Residual Radioactive Contamination From Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent, Final Report," NUREG/CR-5512, Vol. 1. Prepared by PNL, Richland, WA for U.S. Nuclear Regulatory Commission, Washington, D.C.

Pfizer, Inc., 2014. "Eclipse Cyclotron Facility Decommissioning Final Status Report." Pfizer, Inc. March 2014. ML1409A454.