An Alternate Approach to Justifying Iodine-129 Concentration Values during Assessments for Decommissioning Evaluations and for Waste Profiles - 11216

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ABSTRACT

Demonstration that a site meets the established release criterion requires that an assessment be made of the contribution to dose from all potential radionuclides of concern. This is usually straightforward when measured concentrations are available for the radionuclides of concern. However, when a potential radionuclide has not been "detected" in the available samples, an evaluation must be made to determine the reasonable contribution that the "undetected" radionuclides of concern might be expected to contribute. A typical approach to evaluating the potential dose associated with a radionuclide is to either 1) assume that an "undetected" radionuclide is present at the Minimum Detectable Concentration (MDC) associated with the measurement technique being used; or 2) establish a relationship between the undetected radionuclide and another radionuclide designated as the surrogate. The first approach is conservative since the MDC is considered to be the highest possible concentration for that radionuclide, but nevertheless, this approach can lead to an undesirable consequence when the MDC value is larger than the Derived Concentration Guideline Level (DCGL) for that specific radionuclide because it would be virtually impossible to demonstrate compliance with the site release criterion. Therefore; an alternate approach must be developed to establish a justifiable basis to define a reasonable maximum concentration for the "undetected" radionuclide that is not based on the MDC for the radionuclide. The second approach can be used if there are measured values for both the undetected and surrogate radionuclides. This paper discusses a third approach of using fission yields as the technical basis for estimating the maximum potential concentration of I-129 in a media in contrast to using analytical measurement information.

INTRODUCTION

When decommissioning a site it is necessary to consider the contribution of all potential radionuclides of concern when demonstrating that the release criteria have been met. This is usually straightforward when measured concentrations are available for the radionuclides of concern, but when a potential radionuclide has not been "detected" in the available samples, an evaluation must be made to determine the reasonable contribution that the "undetected" radionuclides might be expected to contribute.

The "undetected" radionuclide considered in this paper is Iodine-129 (I-129) which has been identified to be one of the potential radionuclides of concern in contaminated soils at a site. Experience has indicated that a typical MDC for I-129 is approximately 15 pCi/g using reasonable analytical laboratory procedures. If special steps are taken, it is possible to reduce the MDC to approximately 0.5 pCi/g; however, even this lower MDC can be too large for the intended purpose. The NRC screening value for I-129 in soil is 0.5 pCi/g (see Appendix H, Table H.2, "Screening Values (pCi/g) of Common Radionuclides for Soil Surface Contamination Levels" of Reference [1]).

In this paper, a new approach is presented that uses fission yields as the technical basis for establishing a logical ratio of I-129 to another radionuclide that can then serve as the surrogate. This new approach was used because of significant problems encountered while attempting to use the two industry standard approaches to evaluate the potential dose associated with a radionuclide.

The Fission Yield approach uses two other potential radionuclides of concern as surrogates, Cesium-137 (Cs-137) and Strontium-90 (Sr-90). This is a reasonable scenario since all three of the radionuclides are fission products and would be expected to be present at locations where the contamination is associated with fission products. Since all three radionuclides are fission products, an alternate approach to establishing the maximum possible concentration for I-129 would be to calculate the expected ratio of I-129 to either Cs-137 or Sr-90 based on fission yield information from a published source such as the "Chart of the Nuclides [2].

DISCUSSION OF THE PROBLEM FOR DECOMMISSIONING EVALUATIONS

One typical approach to evaluating the potential dose associated with a radionuclide is to assume that an "undetected" radionuclide is present at the MDC associated with the measurement technique being used. This approach is conservative since the MDC could be considered to be the highest possible concentration for an "undetected" radionuclide. This approach, however, will lead to an unreasonable conclusion when the MDC value is larger than the DCGL for that specific radionuclide. Using this assumption when the MDC is greater than the DCGL, it is impossible to demonstrate compliance with the site release criterion. Therefore, an alternate approach must be developed to establish a justifiable basis to define a reasonable maximum concentration for the "undetected" radionuclide that is not based on the MDC for the radionuclide.

To illustrate the problem that can occur if the usual approach of assuming that the I-129 concentration is the MDC value when evaluating the potential contribution of I-129 to potential dose at the site, consider the assumptions of Table I.

Table I. Assumptions for Illustration Example

Parameter	Assumed Value
DCGL for Sr-90	1.7 pCi/g (USNRC Screening Value[1])
DCGL for Cs-137	11 pCi/g (USNRC Screening Value)
DCGL for I-129	0.5 pCi/g (USNRC Screening Value)
Soil Concentrations for Sr-90	Vary from 0.5 to 5,000,000 pCi/g
Soil Concentrations for Cs-137	Vary from 0.5 to 5,000,000 pCi/g
Soil Concentrations for I-129	Equal to typical MDC value of 0.5 pCi/g

Based on the assumptions of Table I, and setting the Sr-90 and Cs-137 concentrations equal (ratio of Sr-90 to Cs-137 equals 1), Figure 1 shows the percent contribution to the total dose for each of the three radionuclides. Using the MDC for the maximum I-129 concentration would result in making the determination that I-129 is a major contributor to the total dose especially at low concentrations of Sr-90 and Cs-137. This conclusion would be unrealistic and would result in the conclusion that the soil could never be released as meeting the decommissioning criteria.

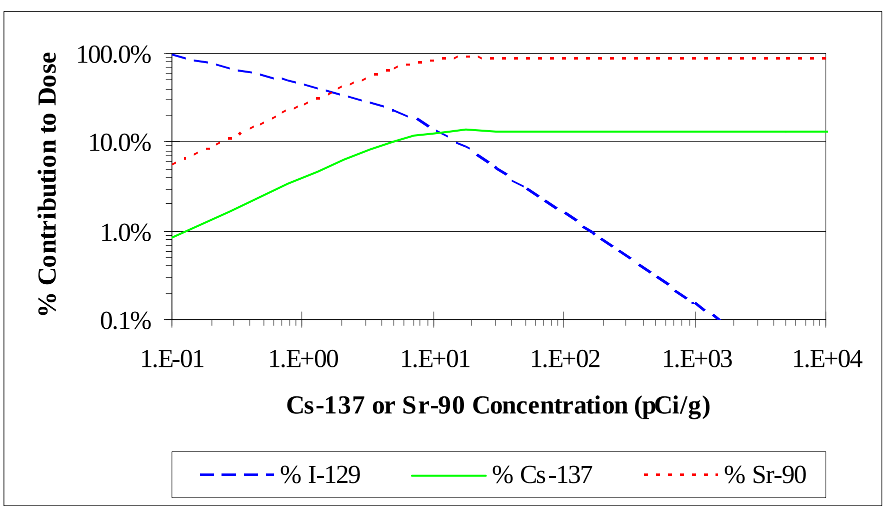

Figure 1. Percent contribution to total dose based on assumptions in Table I

The I-129 contribution does not drop below a contribution of 10% until the Sr-90 and Cs-137 contributions are both about 20 pCi/g. In this situation it is impossible to ever demonstrate compliance with the release criteria based on DCGLs because the fraction of DCGL for I-129 is equal to 1 without adding the contribution for Sr-90 and Cs-137 as required by the sum-of-fractions rule.

[1] The USNRC soil screening values are given in Appendix H, Table H.2 of Reference [1].

This example illustrates the problem that will be encountered if one makes the assumption the potential contribution from I-129 should be estimated by setting the concentration of I-129 to be equal to the MDC value obtained from the analytical measurements.

The second typical approach is to develop a relationship between the "undetected" radionuclide and another radionuclide designated as the surrogate as discussed in Section 4.3.2 of the MARSSIM [3]. The usual problem for I-129 in this case is that all of the measurement values are reported as MDC values and it is not possible to develop a consistent ratio to any other radionuclide.

CONSIDERATION OF FISSION YIELD INFORMATION

The use of fission yields as the technical basis for establishing a realistic ratio to another radionuclide permits the estimation of the maximum potential concentration of I-129 in a media in contrast to using analytical measurement information. These ratios can be considered to be reasonable if the source of the contamination is related to contamination from reactor operations or fuel handling. Example calculations are provided to demonstrate both the issues and the resolution that results when the ratio from fission yields is used rather than using the MDC for I-129 in the final dose assessment evaluation. The approach described is also valuable in: 1) determining that I-129 can be defined as an insignificant contributor to dose and can be removed from further detailed consideration during the conduct of a final status survey, and 2) development of a waste profile for disposal of the waste material.

If the contamination can be associated with sources associated with fission products then it is reasonable to assume that the radionuclides will be present in the ratios based on the fission yields in the fuel. The three radionuclides considered in this paper, Sr-90, I-129 and Cs-137, are fission products. When this assumption is valid then an alternate approach to develop more realistic potential concentration values for I-129, can be based on fission yield information. As an example the following information is based on using Cs-137 as the surrogate radionuclide and treating I-129 as the related radionuclide. Normally the ratio between the surrogate and the related radionuclides would be based of analytical measurement data.

In this approach, an estimated concentration ratio is developed between I-129 and Cs-137 based on fission yield using the following equation:

$$\text{Activity Ratio for I} - 129 = \frac{F_I}{F_C} \times \frac{S_C}{S_I} \times \frac{A_I}{A_C}$$

Where:

F_C = fission yield for Cs-137 = 6.19% [2]
F_I = fission yield for I-129 = 0.54% [2]
S_C = specific activity (g/mCi) for Cs-137 = 1.15 x 10-5 [4]
S_I = specific activity (g/mCi) for I-129 = 5.67 [4]
A_C = atom mass (g/g-atom) for Cs-137 = 137
A_I = atom mass (g/g-atom) for I-129 = 129

Table II shows the expected ratio of I-129 to Cs-137 based on the published fission yields for both radionuclides.

Table II. Fission Yield Calculations for I-129 to Cs-137

Radionuclide	Fission Yield (% atoms)	Half Life [3] (years)	Specific Activity (g/mCi)	Atom Mass (g/g-atom)	Activity Ratio of I-129 to Cs-137
Cs-137	6.19	30.1	1.15x10-5	137	
I-129	0.54	1.60 x 10+7	5.67	129	1.67 x 10-7

The expected ratio based on the fission yields is then I-129/Cs-137 = 1.67 x 10-7. If information is available on the time since irradiation ceased for the contamination present, it would be possible to correct the above fission yields by applying a decay factor for the elapsed time. This correction would increase the value of the ratio. For example if it known that the appropriate decay time is approximately 30 years then the corrected ratio would be approximately 3.4 x 10-7.

There is an uncertainty associated with the source document used to obtain fission yield information. For example if the data from Table 21.3 of [5] is used then the calculated ratio of I-129 to Cs-137 is 3.9 x 10-7. As a conservative measure for this paper a value of 1 x 10-6 will be assumed for the ratio of I-129 to Cs-137.

RECALULATION OF I-129 CONTRIBUTION USING FISSION YIELD RATIO

Table III provides the results when one starts with the proposition that the ratio of I-129 to Cs-137 should based on the fission yield ratio. Using this approach demonstrates that the I-129 dose not contribute a significant fraction to the total dose estimate. Therefore it would be possible to make the determination that I-129 can be defined as an insignificant contributor to dose and can be removed from further detailed consideration during the conduct of a final status survey.

Table III. Calculation of I-129 contribution to dose based on fission yield ratio

Radionuclide	Relationship	Ratio to Surrogate	DCGL (pCi/g)	Fraction of DCGL	Percentage of Contribution to Dose
Cs-137	Surrogate	1.00E+00	11	9.09E-02	99.9978%
I-129	Related	1.00E-06	0.5	2.00E-06	0.0022%
			Totals	9.09E-02	100%

DISCUSSION OF WASTE PROFILES

The approach of using fission yield ratios to determine an expected I-129 concentration could also be used in the determination of waste classifications. Waste Classifications are regulated under the provisions of Title 10 of the Code of Federal Regulations, Part 61, Section 55, “Waste classification” (10CFR61.55). The provisions in 10CFR61.55 establish the concentration limits for certain radionuclides for waste to be considered suitable for near surface disposal.

In order to make a shipment of a waste material to a disposal site it is necessary to establish that the concentrations of radionuclides in the waste material, i.e. a waste profile, do not exceed the concentration limits of Table 1 of 10CFR61.55. This table includes I-129 as a long-lived radionuclide with a limiting value of 0.08 Ci/m^3. The normal practice to determine if a waste is suitable for disposal at a specific location is to develop a waste material profile by submitting sample(s) to a radioanalytical laboratory with I-129 included in the suite of analyses.

The problem that may arise when developing a waste profile is different than that discussed above because in this situation it is not often a problem if the estimate is unrealistically conservative. It is only necessary to demonstrate that the concentration limit is not exceeded. The result of the typical approaches would be the overstatement of the amount of I-129 in the waste shipment. The fission yield approach described above could be useful in the development of a technical basis document to justify a more realistic estimate of the actual concentration of I-129 in a waste shipment.

CONCLUSIONS

This paper demonstrates a new Fission Yield approach to calculating the contribution of I-129 to dose estimates for the decommissioning process. The new approach was used because of unrealistic high estimates of percent contribution problems that arose from using the typical calculation methods. The problem originates in the difficulties in making analytical measurements of sufficient accuracy for I-129 especially when the sampled media has low concentrations of total radioactivity normally associated with soils that are expected to meet release criteria. Development of a ratio to a surrogate radionuclide using the fission yields for the radionuclides can establish a valid technical basis for estimating the concentration of I-129.

In addition, it is often necessary to estimate the maximum concentration of I-129 in waste shipments to demonstrate compliance with the requirements of Table 1 of 10CFR61.55. In this situation, the typical approaches for assuming concentrations of "undetectable" radionuclides may result in overestimating the presence of I-129 in the waste shipment. Even though this is not often an issue with waste disposal, the Fission Yield approach presented in this paper could provide an alternate resolution if an issue developed.

The Fission Yield approach described in this paper has been demonstrated to be valuable in: 1) determining that I-129 can be defined as an insignificant contributor to dose and can be removed from further detailed consideration during the conduct of a final status survey, and 2) the development of a waste profile for disposal of the waste material.

REFERENCES

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