BS ISO 16966:2013

BSI Standards Publication

Nuclear energy — Nuclear fuel technology — Theoretical activation calculation method to evaluate the radioactivity of activated waste generated at nuclear reactors

bsi.

...making excellence a habit.™

National foreword

This British Standard is the UK implementation of ISO 16966:2013.

The UK participation in its preparation was entrusted to Technical Committee NCE/9, Nuclear fuel cycle technology.

A list of organizations represented on this committee can be obtained on request to its secretary.

This publication does not purport to include all the necessary provisions of a contract. Users are responsible for its correct application.

ISBN 978 0 580 75887 4

ICS 27.120.30

Compliance with a British Standard cannot confer immunity from legal obligations.

This British Standard was published under the authority of the Standards Policy and Strategy Committee on 31 December 2013.

Amendments issued since publication

Date	Text affected

BS ISO 16966:2013

INTERNATIONAL STANDARD

ISO 16966

First edition
2013-12-01

Nuclear energy — Nuclear fuel technology — Theoretical activation calculation method to evaluate the radioactivity of activated waste generated at nuclear reactors

Energie nucléaire — Technologie du combustible nucléaire — Méthode théorique de calcul de l'activation pour évaluer la radioactivité des déchets activés produits par les centrales nucléaires

Reference number
ISO 16966:2013(E)

COPYRIGHT PROTECTED DOCUMENT

ISO copyright office
Case postale 56 • CH-1211 Geneva 20
Tel. + 41 22 749 01 11
Fax + 41 22 749 09 47
E-mail copyright@iso.org
Web www.iso.org

Published in Switzerland

Contents

Page

Foreword

ISO (the International Organization for Standardization) is a worldwide federation of national standards bodies (ISO member bodies). The work of preparing International Standards is normally carried out through ISO technical committees. Each member body interested in a subject for which a technical committee has been established has the right to be represented on that committee. International organizations, governmental and non-governmental, in liaison with ISO, also take part in the work. ISO collaborates closely with the International Electrotechnical Commission (IEC) on all matters of electrotechnical standardization.

The procedures used to develop this document and those intended for its further maintenance are described in the ISO/IEC Directives, Part 1. In particular the different approval criteria needed for the different types of ISO documents should be noted. This document was drafted in accordance with the editorial rules of the ISO/IEC Directives, Part 2 (see www.iso.org/directives).

Attention is drawn to the possibility that some of the elements of this document may be the subject of patent rights. ISO shall not be held responsible for identifying any or all such patent rights. Details of any patent rights identified during the development of the document will be in the Introduction and/or on the ISO list of patent declarations received (see www.iso.org/patents).

Any trade name used in this document is information given for the convenience of users and does not constitute an endorsement.

For an explanation on the meaning of ISO specific terms and expressions related to conformity assessment, as well as information about ISO's adherence to the WTO principles in the Technical Barriers to Trade (TBT) see the following URL: Foreword - Supplementary information

The committee responsible for this document is ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 5, *Nuclear fuel cycle*.

Introduction

This International Standard presents guidelines on activation calculation methods for evaluating the radionuclide content of activated waste generated at nuclear reactors.

This International Standard addresses the basic process of planning, executing, and reporting of results for itemized component characterizations (point method) based on neutron source estimation and component elemental compositions and physical parameters and usage in the reactor. This International Standard also introduces the range method that extends the point method to define a radionuclide distribution applicable to a collection of components of similar types and exposure histories that take into account stochastic variations of the input parameters for material composition, neutron fluence rates, and exposure histories applicable to the ranges of these parameters found in the collection.

INTERNATIONAL STANDARD ISO 16966:2013(E)

Nuclear energy — Nuclear fuel technology — Theoretical activation calculation method to evaluate the radioactivity of activated waste generated at nuclear reactors

1 Scope

This International Standard gives guidelines for a common basic theoretical methodology to evaluate the activity of radionuclides in activated waste generated at nuclear reactors using neutron activation calculations.

The evaluation of any additional activity contributed by deposited contamination is not addressed in this International Standard.

2 Terms and definitions

For the purposes of this document, the following terms and definitions apply.

2.1
activation calculation
method of theoretical calculation for determining the radioactivity induced by neutron irradiation

2.2
activated waste
radioactive waste which contains radioactivity induced by irradiation

EXAMPLE Control rod, channel box, burnable poison, core support structures, reactor internal structures, and materials in close proximity to the reactor core, etc.

Note 1 to entry: It can also contain additional radioactivity in the form of surface contamination.

2.3
difficult-to-measure nuclide
nuclide whose radioactivity is difficult to measure directly from the outside of the waste packages by non-destructive assay means

[SOURCE: ISO 21238:2007, modified]

EXAMPLE Alpha-emitting nuclides, pure beta-emitting nuclides, and characteristic X-ray-emitting nuclides.

2.4
control index
index which has a constant relationship with the irradiation conditions affecting the activity concentration of the nuclide contained in the activated waste and enables the calculation of the activity concentration of the target radionuclide by the use of a conversion factor

EXAMPLE Fuel burnup.

2.5
key nuclide
gamma-emitting nuclide whose radioactivity is correlated with that of difficult-to-measure nuclides and can be readily measured directly by non-destructive assay means

[SOURCE: ISO 21238:2007, modified]

EXAMPLE ^{60}Co.

2.6
parent element
chemical element which produces a target radionuclide via neutron irradiation

2.7
target radionuclide
radionuclide of which the activity and/or concentration has to be declared for disposal or transportation of waste packages

2.8
neutron fluence rate
at a given point in space, the number of neutrons incident on a small sphere in a small time interval, divided by the cross-sectional area of that sphere and the time interval

[SOURCE: ISO 31-10:1992, modified]

3 Theoretical evaluation method

3.1 General methodology

The following two methodologies can be applied for estimating the radioactivity of the activated waste:

— Point method, a technique applicable to calculate the radioactivity concentration in a representative piece or specific point of an activated waste item;

— Range method, a technique that is an extension of the point method applicable to a class of activated component of similar property and exposure condition to calculate the average radioactivity of a set or collection of activated wastes by evaluating the range of its radioactivity concentrations typical of that set.

3.2 Point method

Point method provides the basic structure for performing activation calculations. It is generally performed on an item by item basis using directly applicable or best estimate values for key parameters, including neutron flux and material specification along with specific information on history of usage.

This method can be used for the evaluation of all kinds of activated wastes, including in-core hardware and reactor internals. This method, in general, offers more precision on specific items and can be necessary in the situation where activated wastes are close to the disposal limits. This method is most often applied with corroborative dose rate survey and normalization with dose-based key nuclide estimates.

NOTE See Annex B.

3.3 Range method

3.3.1 General

The neutron and irradiation conditions of the target activated waste vary depending on the neutron fluence rate at its physical position in the reactor. The total radioactivity of the whole activated item(s) can be estimated by repeating the activation calculation to cover all the necessary conditions of the neutron irradiation among the whole activated item of the specified type. This provides typical or average values and distribution.

The radioactivity concentration of some types of activated waste has a close relationship with the fuel burnup (e.g. components associated with fuel elements). Once equilibrium has been reached, the radionuclides produced in the same part of the same activated waste have constant composition ratios because those parts have the same elemental composition and irradiation conditions.

Where reactor components are installed at a fixed location in the reactor (e.g. reactor pressure vessel) with a specified elemental composition, only the neutron fluence rates differ depending on the axial and radial position of this location.

The following three theoretical evaluation methods are applicable as typical range methods:[1]

— Conversion method;

— Correlation method;

— Distribution evaluation method.

NOTE See Annex C.

3.3.2 Conversion method

In activated waste such as channel boxes and burnable poison which are used as fuel assembly parts, the induced radioactivity can be closely correlated with some common reactor control parameters such as fuel burnup, since the cases where the items are of similar design and material, remain in the reactor for the same length of time, and are subject to the same neutron fluence as the fuel.

In this method, the relationship (conversion factor) between the control index (such as fuel burnup) and the radioactivity concentration of such fuel assembly parts can be calculated by a series of activation calculations which cover the possible range of burnup variations. The radioactivity concentration of the activated waste can then be estimated by multiplying the control index such as burnup and the conversion factor within its valid range.

NOTE See Annex C.

3.3.3 Correlation method

The correlation method can be used if a particular part or individual item of an activated group is observed to have composition ratios of radionuclides produced simultaneously by irradiation that are constant once equilibrium has been reached within a specified range because the elemental composition, neutron, and irradiation conditions (times) of the particular part are very similar.

Therefore, the correlation between the difficult-to-measure (DTM) nuclides in the whole of the activated waste and the key nuclides produced simultaneously is evaluated by a series of activation calculations which cover the elemental composition, neutron, and irradiation conditions of several specific parts of several activated wastes of the same kind. The ratios between the key nuclides and the DTM nuclides are calculated by evaluating this correlation, and the radioactivity concentration of each nuclide in the activated waste can be calculated by multiplying the ratio between the key nuclides and the DTM nuclides by the radioactivity concentration of the key nuclide. Note that the post irradiation decay time should be considered in this method since the ratio between the key nuclides and the DTM nuclides changes over time due to the difference in decay half-life.

NOTE See Annex C.

3.3.4 Distribution evaluation method

Fixed reactor components are activated by direct neutron irradiation in the reactor. In the case of such activated wastes, the elemental composition and irradiation conditions (time) can be considered to be similar for all items of the group, and only the neutron fluence rate differs depending on their installation locations in the reactor.

In this method, the radioactivity concentration distribution and range of each radionuclide in all activated wastes can be calculated by a series of activation calculations which completely cover the neutron fluence rate at the installation location of the activated waste. The average radioactivity

concentration of each nuclide in the waste can then be calculated based on the resulting radioactivity concentration distribution.

NOTE See Annex C.

4 Calculations

4.1 General

The basic process for performing activation calculations for the purpose of estimating radionuclide concentrations in activated wastes involves several steps.

a) Establish the context. This defines the purpose of the calculation, the radionuclides of interest, the accuracy and precision required, the basic geometry, and the overall scope of the required calculations. This is an important first step for selecting an appropriate calculation methodology and input parameters (e.g. involved materials, quantity of items, similarity of items, target waste form, accessibility of sampling, etc.).

b) Select the calculation methodology, e.g. point or range method. The selection depends on the context of the calculation and the availability of input data.

c) Select and determine the input parameters. The input parameters and calculation boundary conditions depend on the chosen calculation methodology. Conversely, the availability of data can influence the selection of calculation methodology.

d) Perform the calculation(s) using the selected methodology.

e) Process the raw results of the calculation(s) to determine correlation factors, conversion factors, etc., depending on the chosen methodology.

4.2 Selection and determination of input parameters and conditions

4.2.1 Input parameters

A general procedure for setting input parameters and conditions can be established to allow point-based or range-based calculation methodologies. Activation calculations require a number of basic input parameters and conditions as listed below:

— elemental composition;

— neutron fluence rates;

— irradiation history (integrated exposure/decay time).

4.2.2 Elemental composition

Parent chemical elements are selected based on the context and scope of the calculation and the component materials involved (e.g. steel alloys). Elemental composition data on those parent elements can then be collected from a variety of sources for use in the subsequent activation calculations.

a) Selection of parent elements

The list of parent elements is selected for each kind of activated waste using one or more of the following concepts.

— The initial list of parent elements is assembled from the known basic chemical composition of the material being evaluated (e.g. steel alloys), along with elements which are known or suspected to be present as contaminants or in trace amounts.

— The elements which produce radionuclides that are significant in the safety assessment (including those used as key nuclides for scaling other nuclides) should be specifically included in the list of parent elements. Consideration should be given to key nuclides with a sufficiently long half-life, such that the half-life does not significantly affect the ratio between the nuclides over the evaluation timeframe.

The parent elements can be screened out using one or more of the following steps.

— Radioisotopes can be screened out from the original list of parent elements to be used in the activation calculations.

EXAMPLE Radioisotopes which are produced after activation of material such as Pu, etc.

— Elements which do not produce radionuclides of interest to the context of the calculation can be screened out from the list of parent elements.

— Elements which are demonstrated to be removed during the material refining process can be screened out from the list of parent elements. However, often the element cannot be completely removed and the residual trace amount can still be significant for the activation calculation, depending on the scope and context of the calculation.

— Elements which result in only a very small contribution to the short-term or long-term risk or result in radionuclides of short half-life can normally be screened out from the list of parent elements, depending on the scope and context of the calculation.

b) Collection of elemental composition data

Elemental composition data on the waste can be collected using one (or more) of the following methods, which is selected in consideration of the kind and material of the waste and of the context and scope of the calculation:

— direct elemental analysis of an actual sample of the original material (e.g. one preserved for quality management purposes) or a sample of the same kind of material;

— collection of literature data and mill sheets which show the results of element analysis of the same kind or similar materials;

— collection of elemental composition data given in the material specifications for the original material. (If original material specifications are not available, it would be acceptable to use the national standard material specification or other recognized standard applicable to the material being evaluated.)

c) Selection of input elemental composition

The elemental composition for the material to be evaluated can be set using one of the following methods.

— Set representative values. The representative (or best estimate) values of the chemical element concentrations are set based on the chemical element data collected on the parent elements of the radionuclides to be evaluated. This is normally used for the point method.

— Set a concentration range. The minimum and maximum chemical element concentrations are set in consideration of the concentration distribution based on the chemical element data collected on the parent elements of the radionuclides to be evaluated. This is normally used for the range method.

— Set a concentration distribution. The representative chemical element concentration distribution is set in consideration of the concentration distribution based on the chemical element data collected on the parent elements of the radionuclides to be evaluated. This is normally used for the range method.

NOTE See C.5.

4.2.3 Neutron fluence rates

The neutron flux and energy spectrum shall be determined using a suitable neutron transport computer code for the activated material accounting for its position relative to the reactor core. The code should be suitably matched to the requirements of the calculation in context of modelling detail and overall accuracy. Alternative methods including Monte Carlo simulation of the transport process can be justified in instances where additional detail or corroboration is required.

NOTE See C.6.

4.2.4 Irradiation history

The irradiation time and non-irradiation time during the calculation timeframe under consideration (e.g. reactor operating time and post-irradiation decay time) can be set in either of the following two methods for the activated waste to be evaluated. The overall calculation timeframe could involve several cycles of irradiation and non-irradiation time.

a) Setting a detailed irradiation history individually for each waste

   The irradiation history is set in detail for each activated hardware item, based on its individual history. This is normally used for the point method.

b) Setting a representative irradiation history

   Irradiation history, which is considered to be adequate or conservative for a group of activated materials, are set based on the irradiation history of a representative or limiting item of the group.

When the irradiation history is used as input to further calculations by the Conversion method, Correlation method, or Distribution evaluation method, a range of the irradiation conditions can be set which are representative of several activated materials, instead of setting detailed conditions for individual materials.

This is normally used for range method.

NOTE See C.7.

4.3 Activation calculations

4.3.1 Calculation code

The activation calculation code should be verified in accordance with the user's quality management system. Users should understand and be familiar with the methodology and appropriately trained in the use of the program and its limitations.

EXAMPLE ORIGEN code, etc.

4.3.2 Setup input data

The input data for activation calculation listed below (for point or range method) are required for each theoretical calculation method to be applied. In accordance with 4.2, these data are set for each kind of activated waste to be evaluated:

— elemental composition;

— neutron fluence rates;

— irradiation history (integrated exposure/decay time).

When the range method is applied, input data for each individual calculation and condition can be selected at random within the range of the input data distribution in 4.2 and can be set as the input

conditions for the activation calculation. Alternatively, input data representative of each condition can be set as the input conditions for the activation calculation.

4.3.3 Determining the number of calculations

The number of calculations can be determined by one of two methods:

a) Point method

The required calculation number should be determined case by case.

(e.g. for a large or complex object, it is necessary to divide it into several segments for calculations or use an average or representative value, if the neutron fluence or materials can vary in different parts of the object.)

b) Range method

When the range method is applied, the number of activation calculation results obtained should be adequate for their use as evaluation data for determining the radioactivity concentrations. Whether their number is adequate or not can be judged in consideration of the number of the activation calculations made and the changes in the stability of the statistical values obtained from these activation calculation results.

4.4 Validation and uncertainties

4.4.1 Validation

Validation involves determining the accuracy and applicability of the software results with respect to its intended application. Only validation can evaluate such things as the level of accuracy of physical approximations, the applicability of physical correlations, the appropriateness of numerical method approximations, etc.

Validation should be performed using software that is under configuration management and has a clear version number. It is normally performed by the user of the software for a specific application.

Validation confirms that a calculation can be consistently and correctly conducted by demonstrating (with objective and documented evidence) that the calculation code and calculation routine of the theoretical calculation method produce the expected results for a given input. This can be done, for example, by applying the calculation code to known test cases, or by comparing the results with those of an alternate calculation method that has been previously validated or has a known analytical solution.

NOTE 1 It is important to make a distinction between software “verification” and “validation”. Verification is the step where the software is checked against its specifications. This is normally done by the developer of the software prior to delivery to the user.

NOTE 2 See A.4.

4.4.2 Dealing with uncertainties

The allowable level of uncertainty depends on the particular radionuclides and how the waste is ultimately treated in disposal. An estimation of accuracy and uncertainty of the calculation result should be performed in order to quantify the representativeness of the theoretical calculation carried out. Accuracy represents how close the theoretical calculated value is to the real value. As such, it is a measure of the bias, and many times a degree of conservativeness is established to safely guarantee of the limits involved (Waste Acceptance Criteria, radiological protection, transport processes, etc.). Uncertainty represents the variability around the consigned value. A highly precise value is characterized by a low uncertainty and vice versa.

Representativeness can be considered to be a parameter which accounts for both the conservativeness/bias and the uncertainty around a value. It gives an image of distance and variability of the theoretical calculated value to the actual value.

NOTE See Annex D.

4.5 Records

Reports should be presented in such a way as to provide a traceable account of the work performed with data sources and assumptions clearly identified. This allows the calculation to be easily reproduced by others for validation purposes.

Results shall be reported in a concise, easy to interpret, manner that allows the result to be understood without additional manipulation. Reporting units should be specified in the record and in any event should be in International System of Units unless specifically directed to use alternative units.

NOTE See Annex E.

Annex A
(informative)

Application and example of the theoretical activation calculation method

A.1 General

Annex A provides explanatory notes and gives examples and practices to assist the understanding or use of this International Standard.

A.2 Basic application of the theoretical activation calculation method

A.2.1 General

Activated radioactive waste from reactor core areas generally has significant levels of radioactivity. The empirical evaluation methods, relying on waste sampling and radiochemical analysis, are often limited for such waste by potentially high radiation exposures to involved labour.

The theoretical evaluation method using activation calculations provides a reasonable alternative method for determining the concentrations of radionuclides produced as a result of activation by neutron irradiation inside and near the reactor core. In this method, the types and amounts of the radionuclides are theoretically determined by the use of activation calculations.

Performance of theoretical evaluation depends on the neutron fluence rates local to the subject material and irradiation history of exposure. The radioactivity concentrations of the materials are determined from the calculated results or the evaluation factors determined from those results.

Figure A.1 shows a basic step by step procedure for applying the theoretical evaluation method. These steps are discussed in more detail below.

A.2.2 STEP 1: Setting the basis for the calculation

The context, scope, and purpose of the calculation need to be established prior to starting the calculations. This assists in determining the radionuclides of interest, the irradiation geometry, parent materials, accuracy and precision required, etc.

The theoretical evaluation of the radioactivity concentrations of activated waste is dependent on knowledge of the reactor operating conditions (irradiation conditions) and the characteristic properties (physical and chemical) of the activated materials being evaluated. This includes the following principal parameters:

— elemental composition of the exposed material;

— neutron fluence rates (including flux and spectrum);

— irradiation history (including irradiation and decay times).

Once the context, scope, and purpose have been determined, the relevant data should be collected. This process is further described in Annex C.

A.2.3 STEP 2: Selecting the evaluation method

In the theoretical evaluation method, the radioactivity concentrations of the activated waste are determined by performing one or more activation calculations based on the characteristic properties, irradiation history, and other conditions of the waste to be evaluated. It is desirable that the most appropriate evaluation method should be selected from among the point method and the range methods, according to the context of the calculation determined in STEP 1.

In STEP 2, a validated evaluation method is selected by technical judgment based on the property evaluation methods that are described in 3.2 and 3.3 for the conditions of the activated waste being evaluated and the properties of the irradiation source:

— installation conditions of the base material being irradiated (e.g. radial and axial positions, located inside or outside the reactor);

— irradiation history data of the component [including both irradiation time and non-irradiation, (decay) time];

— operation history data of the reactor;

— information on a non-destructive evaluation plan for gamma-emitting nuclides in the activated waste and other conditions.

The selection could be based on availability and completeness of data. For example, the range method would be generally preferable when condition information cannot be specifically determined for the material or location of interest.

A.2.4 STEP 3: Implementing the calculation process

Setting of input data for activation calculations can be grouped into two types:

— input data for point method;

— input data for range methods.

In the case of the point method, input data are defined to reflect the representative input conditions at the specified point by the use of the database of calculation input data collected in STEP 1.

When setting input data for the range methods, it is necessary to prepare multiple sets of input data which cover the whole range of the input conditions regarding the activated waste being evaluated, by means of random sampling from the database of calculation input data collected in STEP 1.

A.2.5 STEP 4: Presentation of results

Using the input conditions set in STEP 3, the radioactivity concentrations of the target materials are directly calculated by a verified and validated calculation code or evaluation factors such as composition ratios of correlation and conversion factors are calculated.

If evaluation factors are calculated, the radioactivity concentrations of the activated waste contained in waste packages are determined by the control index data stored as waste management operation data or the measured or calculated concentrations of key nuclides such as ^{60}Co being multiplied by the evaluation factors calculated.

NOTE For an exhaustive declaration of activity of the wastes, surface contamination of components in contact with radioactive fluids has to be added if it is non-negligible in comparison with activation.

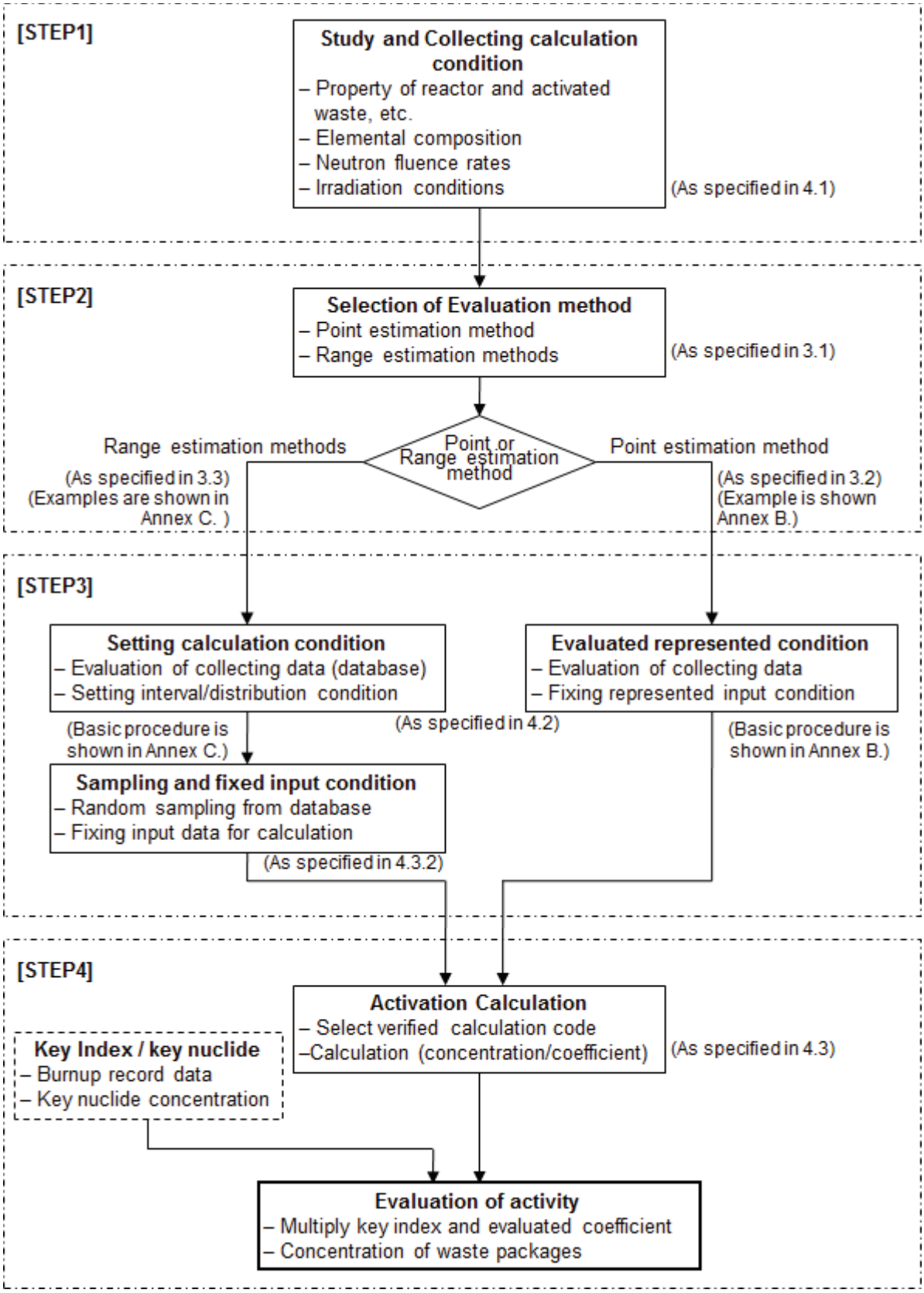


Figure A.1 — Basic flow of application for theoretical activation calculation method

A.3 Example of the theoretical calculation method

A.3.1 General

The following two theoretical methods are available for calculating the radioactivity concentrations of the activated waste:

— Point method, a method for evaluating the radioactivity concentrations of a specific part of a specific kind of activated waste;

— Range method, a method for determining the average or other concentrations of the activated waste by evaluating the concentration ranges of the waste as a whole.

A.3.2 Point method

This is a basic method employing activation calculations, which is intended to evaluate the radioactivity concentrations of a specific position of the activated waste. In this method, the concentrations of radionuclides produced by irradiation are determined by activation calculations based on the detailed neutron fluence rates and irradiation conditions of the waste.

As a feature of this method, it is suited for the evaluation of special kinds of activated wastes which are generated in small quantities, or which have very uniform properties and irradiation histories. However, it requires a very large number of activation calculations when applied to non-uniform activated wastes which are generated in large quantities and have various irradiation patterns in terms of neutron fluence, irradiation history, and other conditions. It can also be used where a single conservative or maximum value can be used for a family of wastes, as defined in the context and scope of the calculation. It is appropriate that the applicability should be judged through sufficient studies in this respect.

This method is also applied to each individual activation calculation in the range methods.

NOTE See Annex B.

A.3.3 Range methods

The radioactivity of some activated wastes is closely correlated with reactor parameters, such as fuel burnup. The radionuclides produced at the same position in the activated waste installed in the reactor have constant production ratios because they have the same elemental composition and irradiation conditions. The same kind of activated waste installed and irradiated in the reactor has the same elemental composition and irradiation conditions and is different only in the neutron fluence rates which depend on the position in the waste. The following three evaluation methods are applicable as the typical range method for evaluating the concentration distribution in the waste:

— Conversion method;

— Correlation method;

— Distribution evaluation method.

NOTE See Annex C.

A.4 Example of validation

A.4.1 General

It is important to make a distinction between software “verification” and “validation”. Verification is the step where the software is checked against its specifications. This is normally done by the developer of the software prior to delivery to the user.

Validation involves determining the accuracy and applicability of the software results with respect to its intended application. Only validation can evaluate such things as the level of accuracy of physical

approximations, the applicability of physical correlations, the appropriateness of numerical method approximations, etc.

Validation should be performed using software that is under configuration management and has a clear version number. It is normally performed by the user of the software for a specific application.

To confirm that a calculation can be conducted in such a way that it produces consistent and correct results, a process of validation of the computer code and calculation methods through benchmarking and comparisons with objective evidence is conducted and documented.

The calculation process can be divided into the stages of preparation, execution, and recording results. Appropriate confirmation and documentation thereof is made through each stage of the process. The results of the calculations are then validated against the expected results. Validation often requires the use of alternate calculations against a known benchmark to check the validity of the results or comparison with measured results. (For example, dose rate measurements can be used to cross-check or to validate calculated results and to reassess the validity of input data or be used as an alternative basis for activity estimation.)

A.4.2 Preparation for the calculation

This stage is to prepare appropriate documents such as procedures necessary to properly and consistently conduct radioactivity calculations using verified and validated radioactivity calculation methods and computer codes.

— Ensure applicability of principal equations and computer programs (verified calculation code: e.g. ORIGEN).

— Check suitability of the method employed (e.g. the method has been used previously and validated through comparisons with actual measurements).

— Provide or reference the documentation of the calculation procedure.

A.4.3 Execution of the calculation

This stage is to clarify necessary calculation conditions according to the documented radioactivity calculation method, to clarify and record the input data so that such data can be verified by third parties as required, and to clearly determine the method of consideration of conservativeness.

— Include a reference for the input data to allow traceability by third parties.

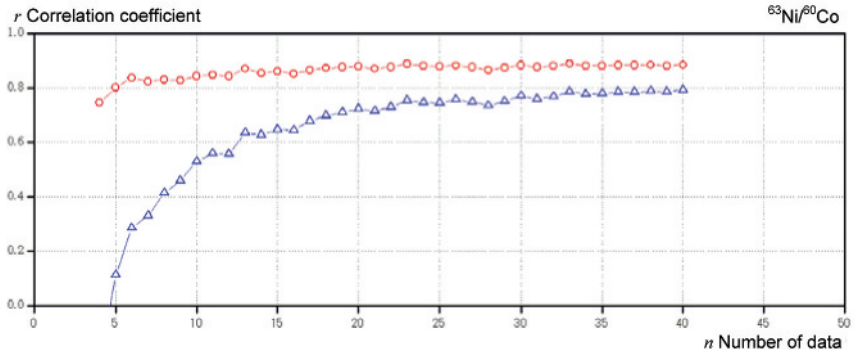
— List references and/or calculation practices for determining input data for calculation.

— Specify the processing method (such as how to set the applied concentration distribution or conduct random sampling from input database) when the input data processed based on the actual data are applied and consider condition of conservativeness.

— Validate the output record (e.g. number, format of result, etc.).

NOTE Methods can be applied that demonstrate appropriate maintainability depending on the amount of data.

The range method involves performing a series of calculations. The number of activation calculation results obtained should be adequate for their use as evaluation data for determining the radioactivity concentrations. The case of the Correlation method is taken as an example. Figure A.2 shows an example of evaluation of the stability of the correlation coefficient (average value, 95 % lower confidence limit value) between difficult-to-measure nuclide concentration and key nuclide concentration when the number of activation calculations increases.



Key

-○- average value of correlation coefficient

-Δ- 95 % lower confidence limit value of correlation coefficient

r correlation coefficient

n number of data

Figure A.2 — Image of changes in the stability of the correlation coefficient (average value, 95 % lower confidence limit value) with an increase in the number of activation calculations

A.4.4 Recording of the results

This stage is to record and evaluate the output of calculation (i.e. the results) based on the clarified calculation input conditions, according to the documented radioactivity calculation method.

— Provide a listing of the calculation results (output).

— Describe the method used and bases for validating the results.

— Assess the calculation results (e.g. sufficiency of the number of calculations, statistic stability of calculation results, etc.).

— Validate the record of output (e.g. compare calculation result and measurement).

— Clarify the factors that could have influenced the results, explicitly specify the purpose of calculation and expectations of the calculation results.

Annex B
(informative)

Suggested procedure for the point method for activation calculation

B.1 General

Annex B provides explanatory notes and gives examples and practices to assist the understanding or use of this International Standard.

The term point method is used in the context of this International Standard in the broad sense of performing typical activation calculations as it could be applied to a single object or assembly of known exposure history. For items used within the reactor core such as fuel assembly hardware, instrument strings, fuel channels, etc., it is usually sufficient to use average core neutron flux rates and energy spectra which are generally well known and managed. For these items, integrated exposures are typically tracked by nuclear power plant (NPP) engineering staff and can provide a sufficient basis when coupled with other information including dates of usage and reactor operating history.

For material outside of the reactor core, including structural components, a particle transport code is often required in order to calculate the neutron fluence rate (flux and spectrum) at the target location. The calculation needs to be based on all relevant information concerning the source, material types and elemental compositions, densities, geometry, and dimensions between the source and the target itself. At the target's position (ideally a representative point), the neutron fluence rate along with the irradiation condition is used by an ensuing activation code to determine the activation of the target material.

B.2 Sequence of performing an activation calculation

B.2.1 General

The aforementioned calculation sequence is described below in more detail by dividing it into three consecutive steps which constitute a general complete (neutron) activation calculation (see Figure B.1).

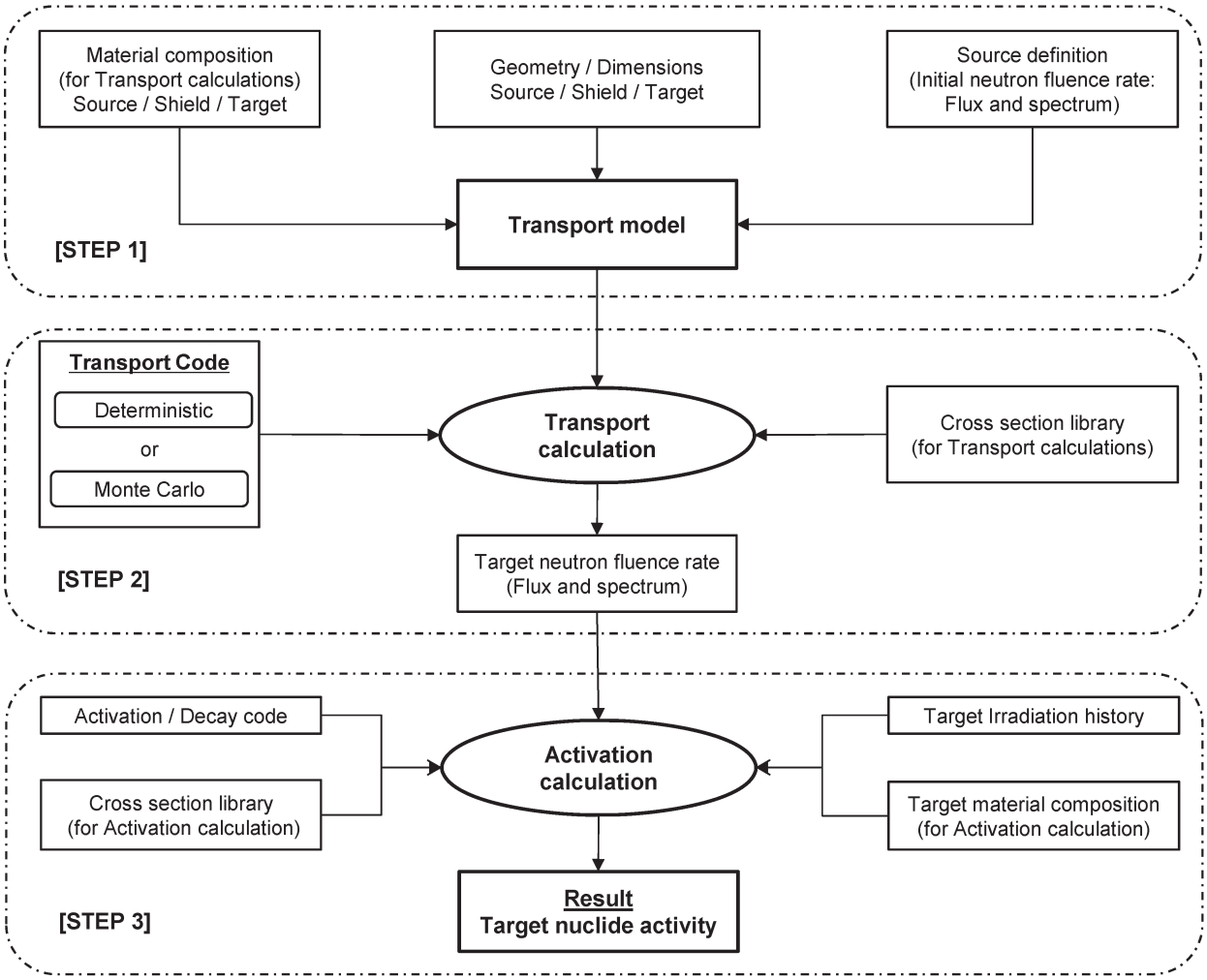


Figure B.1 — Three-step approach for performing an activation calculation

B.2.2 STEP 1: Determining the neutron fluence rate basis/setting up the transport model

B.2.2.1 Expendable in-core or near core assemblies

This includes reactor instrumentation, control assemblies, etc.

— Use recorded neutron fluence rate value used for tracking the service life of the component. Re-baseline neutron fluence rate from operating history (start and end of exposure) and fluence rates during exposure.

— The total activity distribution is most strongly impacted by the maximally exposed portion of the component. This distribution would conservatively serve as basis for the entire component.

B.2.2.2 Reactor internals, vessel, and adjacent structural material

A transport model has to be set up preparing all required information.

— Source definition

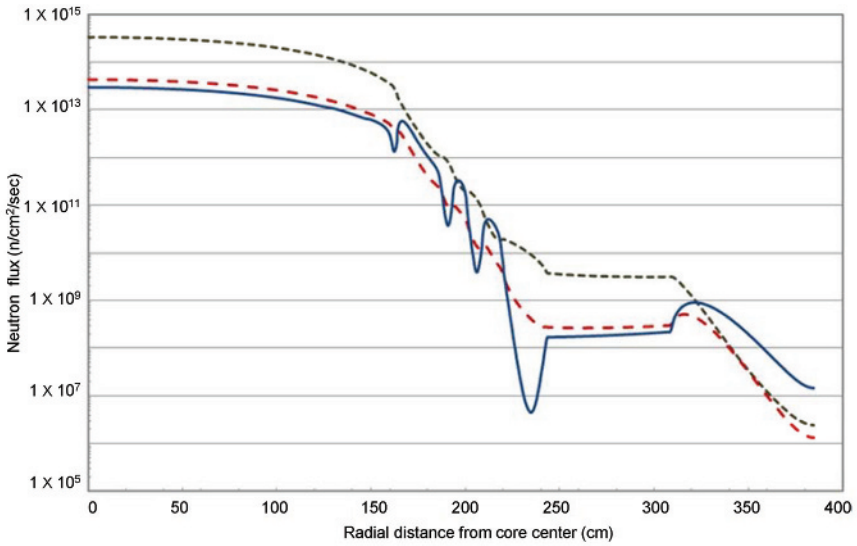
  — neutron fluence rate distribution, or

  - reactor core material compositions including distribution of fissionable isotopes.

- Material and geometry setup for the transport problem, including intervening structures, coolant, etc.
  - material compositions and densities (including isotopic information being relevant for the transport calculation), and
  - geometry description for the transport problem.

Ideally the transport model is three-dimensional, although simpler cases for a given transport problem can be approximated with only two or even one-dimensional models, without additional shielding and/or with simplified source setup (up to a mono-energetic point source).

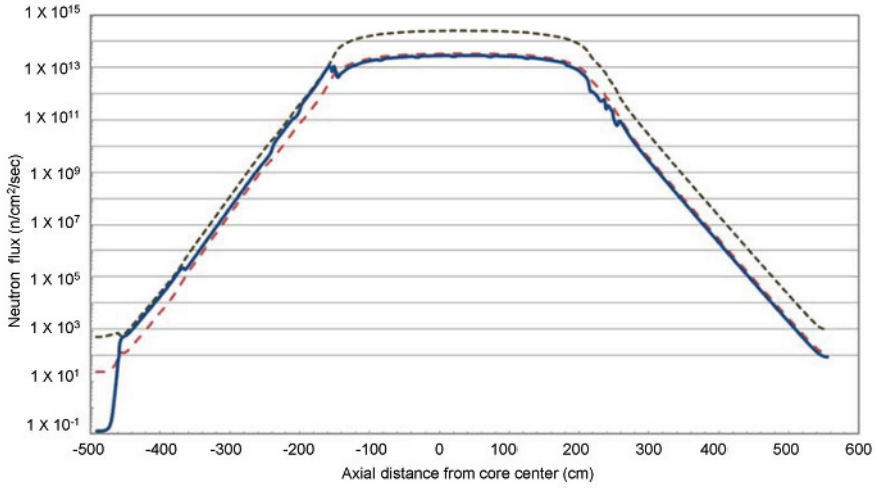
In the case of two- and three-dimensional transport models, there is often limited capability to perform detailed modelling of ex-core structures and simplifications can be required. A representation of the overall flux distribution can be created using a one-dimensional model with two cases representing radial and axial profiles. While not ideal, it should provide sufficient detail for activation estimates by weighting the flux in off-centre locations. Figures B.2 and B.3 show radial and axial neutron flux profiles for a typical PWR generated with the ANISN computer code. The radial trace extends from the centre of the core into the biological shield. The axial trace extends from the bottom to the top of the reactor vessel. Each zone is characterized on the basis of its material mix including fuel, fuel assembly materials, coolant (including poisons), control hardware, structural materials, etc.



Key

- fast neutron
- epithermal neutron
- thermal neutron

Figure B.2 — Example of PWR radial neutron flux trace



Key

- fast neutron
- epithermal neutron
- thermal neutron

Figure B.3 — Example of PWR axial neutron flux trace

The methodology for this analysis is generally described in two reports prepared by Battelle Pacific Northwest Laboratories.[2] [3] A third Battelle report focuses more directly on non-fuel assembly components.[4]

Much more detailed three-dimensional transport models (Monte Carlo or Deterministic) can be used to determine the required neutron spectrum information.[5] [6]

B.2.3 STEP 2: Performing the transport calculation

The transport calculation serves to determine the neutron fluence rate (flux and spectrum) at the location of the target (e.g. at a representative "point" within the target, therefore point method). This point information for the neutron flux and spectrum can also be applied through an averaging process over the volume of the target.

To perform the transport calculation the following components are required:

— the transport model setup (STEP 1) as an input data set;

— an adequate nuclear transport code, which can handle the transport model setup either

  — a Deterministic code (e.g. ANISN, etc.), or

  — a Monte Carlo code (e.g. MCNP, etc.);

— a cross-section library (e.g. ENDF/B, etc.), either in group or continuous energy structure, linked to the transport code.

The result of the transport calculation is the required neutron flux/spectrum at the reference point being used by the following activation calculation for the target.

B.2.4 STEP 3: Performing the activation calculation

The activation calculation serves to determine the nuclide vector for the reference point of the target, again either as a representative point value within the target or as an average value over the target volume.

To perform the activation calculation the following components are required:

— an inventory program to calculate activation and decay products (e.g. ORIGEN, etc.);

— a neutron reaction cross-section library linked to the activation code, either with pre-processed isotopic cross sections, or with the flexibility of the activation code to condense problem-dependent cross sections for the neutron spectrum at the reference point;

— a target material composition containing all primary constituents, along with important impurities and trace elements;

— the irradiation history of the target (e.g. information about changes of the neutron flux/spectrum over the exposure time including possible decay in between and at the end of the exposure).

The result of the activity calculation is the nuclide vector at the reference point of the target. Insignificant nuclide activities can be suppressed by choosing a corresponding cut-off value. The cut-off value(s) used should account for important long half-life radionuclides which typically have very low activity but are important for long-term disposal safety.

B.3 Dose rate survey

The result of an activation calculation is based on the knowledge of all relevant information describing the transport/activation problem, i.e. characteristics of source, material, densities, and geometry. Furthermore, the nuclear codes employed as well as the related cross-section libraries involved and any simplifications of the model have to be used with thorough understanding.

A dose rate survey of an activated target, if properly undertaken, can provide confirmation that the calculation matches the actual target activation condition. Therefore, it is preferable to additionally rely on a dose rate survey of the activated target to validate or even partly scale the nuclide vector from the activation calculation with the proper dose rate survey data.

B.4 Example for an activation calculation by the point method

One typical standard example for an activation calculation based on the point method is the determination of the nuclide inventory for an irradiated small component (e.g. jet pump support pin, bolt, etc.) outside the active zone of a reactor core (e.g. in the vicinity of the core barrel).

For the activation calculation, the neutron fluence rate (flux and spectrum), at least opposite to the location of the component, has to be defined. If there is no other previously established flux basis, a transport calculation has to be performed to develop this information (covering the relevant core area, the location of the component and any moderating water, and/or other shielding components in between). If the component is small relative to the height of the core, the gradient of the neutron fluence rate at its location is likely small enough to be represented as a point value.

Once the spectrum is defined, it can be used with an activation code and a linked cross-section library to determine the activity of the component material. The base component material specification includes trace constituents generally found in the material that can be important for disposal such as the chemical elements of cobalt and niobium. If the irradiation of the component has been more or less constant over its lifetime before it has been removed, only the decay during its storage time has to be additionally taken into account. Decay can be managed within the activation code in two ways:

— adjust the flux rate to account system availability and power levels;

— perform a multistep analysis to follow the operating and idle periods of the plant.

The dose rate survey of the component determines one or more dose rate values at defined distances from the component. Where the component is measured with shielding in place (e.g. under water or in a shielded cask), a transport simulation (basically a shielding-type calculation) can be set up to estimate the dose rate for comparison with the dose rate survey values. The difference between the calculated

and measured dose rates can serve to adjust the initial activity estimate to conform to the dose rate. A similar result can be achieved using direct gamma spectrum analysis and scaling to the measured gamma concentrations. For small items, a point kernel shielding program should provide an adequate simulation. Larger items could require multiple measurements and some additional calculations (sometimes with more complex transport codes) to estimate overall content of the leading gamma emitters. If the measured gamma radiation dose rate is mainly caused by the ^{60}Co inventory of the component (which is generally the case for activated steel components of reactors), the dose rate value can be related to the corresponding ^{60}Co concentration. In this case, the theoretical activation result is scaled to the estimated total cobalt determined from conversion of the dose rate.

Further, if the knowledge of the ^{59}Co impurity concentration for the component material is reliable, then the activation calculation can be (slightly) scaled to the ^{60}Co inventory (determined by the dose rate survey information) to derive the total nuclide inventory of the component (i.e. all relevant activated nuclides including those not contributing to the gamma dose rate of the dose rate survey scaled to ^{60}Co).

Annex C
(informative)

Suggested procedure for range method for setting input data for activation calculations

C.1 General

Annex C provides explanatory notes and gives examples and practices to assist the understanding or use of this International Standard.

C.2 Range method

C.2.1 Types of range methods

The radioactivity of some activated wastes is closely correlated with reactor parameters, such as fuel burnup. The radionuclides produced at the same position in the activated waste installed in the reactor have constant production ratios because they have the same elemental composition and irradiation conditions. The same kind of the activated waste installed and irradiated in the reactor has the same elemental composition and irradiation conditions and is different only in the neutron fluence rates which depend on the position in the waste. The following three evaluation methods shown in Table C.1, in which these features are considered, are applicable as the typical range method for evaluating the concentration distribution in the waste:

— Conversion method;

— Correlation method;

— Distribution evaluation method.

C.2.2 Conversion method

The concept of the Conversion method is based on the assumption that the production of radionuclides is directly proportional to the cumulative neutron exposure of the activated waste.

The radioactivity concentrations in the material can be determined by using a conversion factor proportional to the cumulative exposure (irradiance itself or fuel burnup, reactor operating period, etc.) as a control indices.

Through activation calculations using representative or conservative calculation conditions, the relationship between the control index value and the radioactivity concentrations is evaluated as a conversion factor, and the radioactivity concentrations can be determined for future members of the target waste family from the multiplication of the conversion factor by the applicable control index value.

This method is applied on the condition that the factors proportional to the cumulative exposure of the activated waste is managed by nuclear power plants and would be applicable as control indexes and that representative or conservative neutron fluence rates can be set for the target radioactive waste.

The following can be listed as examples of control indices:

— the cumulative exposure time for control rods, reactor vessels, or other core components which remain in place for more than one refuelling cycle;

— the fuel burnup for BWR channel boxes or other fuel assembly components which are replaced with each refuelling cycle.

When the half-life is short relative to the irradiation time, the radioactivity of a nuclide produced by activation at equilibrium can be expressed approximately (simplified with one condensed neutron energy group) by Formula (C.1):

$$a = f(\sigma, N, \Phi, t, \lambda) \tag{C.1}$$

where

a is the radioactivity concentration of the target radioactive waste (Bq/cm^3);

σ is the activation cross section of the parent nuclide (cm^2);

N is the atom number density of the parent nuclide before irradiation (cm^{-3});

Φ is the neutron fluence rate (cm^{-2}·sec^{-1});

λ is the decay constant of the product nuclide (s^{-1});

t is the irradiation time (s).

When the radioactivity concentration at a position in the target radioactive waste is considered, the activation cross section (σ) and the atom number density (N) can be regarded as constant and the decay constant (λ) is also constant. Therefore, the radioactivity concentration is in proportion to the irradiance as shown by Formula (C.2):

$$a = f(c, \Phi, t) \tag{C.2}$$

where

c is constant ($\sigma \times N \times \lambda$).

When the factor proportional to the irradiance ($\Phi \times t$) on the target radioactive waste is assumed as the control index (f_k), the radioactivity concentration can be expressed by Formula (C.3) as proportional to the control index.

$$a = f(C, i_k) \tag{C.3}$$

where

C is the conversion factor;

i_k is the control index (e.g. fuel burnup).

As described above, the radioactivity concentration can be expressed as the product of the conversion factor (C) and the control index value (i_k). Therefore, if the conversion factor (C) is evaluated in advance, the radioactivity concentration of the activated waste to be evaluated can be determined from the control index value.

C.2.3 Correlation method

The concept of the Correlation method is similar to that of the Scaling factor method which is applied to contaminated radioactive waste. The Scaling factor method is an empirical approach which is based on the results of radiochemical analysis of actually contaminated wastes. On the other hand, the Correlation method is a theoretical approach in which the composition ratios of difficult-to-measure nuclides and key nuclides are determined based on the results of activation calculations which cover the whole range of the elemental composition, the neutron fluence rate and the irradiation conditions of the activated waste being evaluated.

The Correlation method is based on the concept that, in the case of the activated waste, even if the radioactivity concentration changes with time during the period from the beginning to the end of irradiation, the chemical composition ratios of the produced nuclides are kept basically constant because each position of the waste has the same elemental composition, neutron fluence rates, and irradiation conditions. When the average composition ratios of the produced nuclides in the activated waste are evaluated by activation calculations, it is therefore necessary that the activation calculation conditions should be set in consideration of the ranges of the irradiation conditions of the material (irradiation position and reactor operating conditions).

In this method, the average nuclide composition ratio between difficult-to-measure nuclide and key nuclide is evaluated in consideration of the irradiation history (including post-irradiation decay) of the activated waste being evaluated. Even if detailed information is not available for the irradiation history of each waste, this method is applicable by measuring the concentrations of the key nuclides and applying the appropriate ratios for the difficult-to-measure radionuclides of interest.

C.2.4 Distribution evaluation method

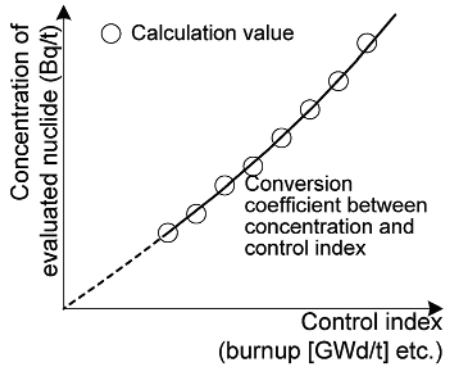
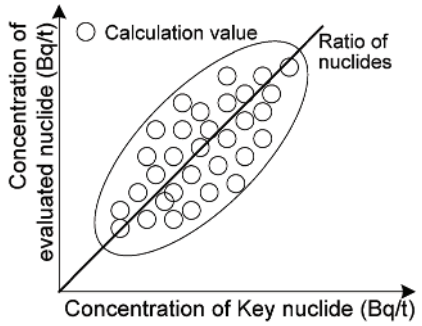
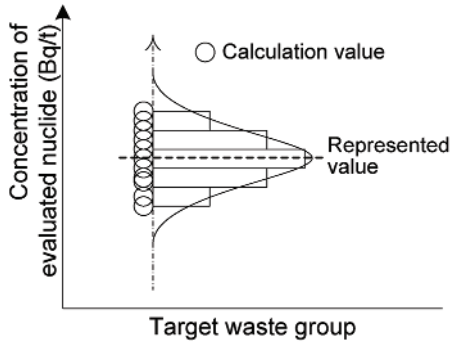
In the Distribution evaluation method, similar target activated waste is classified into one group (for instance, all graphite blocks from a given reactor are classified into one group), and the average radioactivity concentration is determined for each group, instead of determining the radioactivity concentrations of individual items.

The activation calculation in this method is basically equivalent to the case where radioactivity concentrations are calculated for individual items, but this method is applied on the premise that it is possible to ensure that the radioactivity concentration calculated is representative of the activated waste group being evaluated (for instance, average radioactivity concentration). It is therefore necessary that the calculation condition (neutron fluence rates) for the waste in the same group should be within a constant defined range of applicability for the calculation.

In this case representative or conservative values can be set for each condition without setting several different values.

Based on this concept, the activation calculations are made using different values in consideration of the variability of the calculation conditions for the activated waste groups, and the average radioactivity concentrations are determined based on the concentration distribution obtained through those calculations. In determining the average radioactivity concentrations of the waste group by the use of this method, it is appropriate to judge whether the results of activation calculations using several different values have a stable distribution within a constant range.

Table C.1 — Basic property of each range method and applicable activated wastes

Method	Conversion method	Correlation method	Distribution evaluation method
Image of evaluation			
Basic property	There is a strong correlation between a control index, such as burnup (normally nuclear fuel is controlled by burnup), and the concentration of generated nuclides in the activated waste. The concentration of the DTM nuclide in the activated waste is possible to be evaluated by multiplying the "conversion coefficient", calculated by activation calculations that are considered the range of actual reactor condition (e.g. elemental composition, neutron fluence rate, and irradiation condition), and the control index such as burnup.	There are specific relations among simultaneous generated nuclides in the same part of the activated core component because all conditions of irradiation are the same. The ratio of the DTM nuclide and key nuclide can be estimated by activation calculations that consider the range of the actual reactor operation conditions.	The elemental composition and irradiation condition are considered almost the same in the fixed reactor/core component. Only the neutron fluence rate is different in each part of the reactor/core component. The difference of the nuclide's concentrations is generated by the difference of the neutron fluence rate. The concentrations in the components or parts of the reactor can be estimated by activation calculations that consider the range of the actual reactor condition and the position of equipments.
Typical application	Channel box, Control rod, Burnable poison, Fuel assembly hardware, etc.	Channel box, Control rod, Burnable poison, Graphite block, Core shroud, Pressure vessel, etc.	Graphite block, Core shroud, Pressure vessel, etc.

C.3 Basic flow of input data setting in activation calculations

The following two methods are available for setting input conditions in activation calculations necessary to evaluate the radioactivity of target radioactive wastes:

— Point method, a method in which input conditions are set using the representative values (average values, conservative values, etc.) of the chemical element concentration, neutron fluence rate, and irradiation condition of the target radioactive waste;

— Range method, a method in which a sufficient number of input data sets are prepared so as to cover all the data as required on one or more of the chemical element concentration distributions, neutron fluence rate, and irradiation condition of the target radioactive waste.

A basic flow of the range method is shown in Figure C.1, which requires the preparation of a sufficient number of input data sets which cover all items of the chemical element concentration distribution, neutron fluence rate, and exposure time history of the target radioactive waste. Through the selection of conditions in the following steps, it is possible to set two or more sets of input conditions which can appropriately cover the irradiation condition of the target radioactive waste.

a) Select the target radioactive waste and identify its characteristic properties.

   As a first step, the characteristic properties (e.g. shape, material, etc.) of the radioactive waste selected are identified, and basic data necessary for input condition setting, such as in-core irradiation history and element analysis data, are collected.

b) Select the evaluation points in the target radioactive waste (irradiation points).

   The probability distribution of the irradiation point in the target radioactive waste being evaluated is set with consideration given to the shape of the target radioactive waste and its installation direction in the reactor. The irradiation point(s) to be set as an input condition is selected by random sampling from this probability distribution. (See C.4)

c) Set the elemental compositions.

   Necessary element analysis data (chemical element concentration data) on the target radioactive waste are collected in advance, and the distribution of chemical element concentrations is set through an assessment using this database. A chemical element concentration to be set as an input condition is selected by random sampling from the set distribution. (See C.5)

d) Set the installation positions in the reactor operation cycles.

   Typical rotation patterns are evaluated and set for the installation position of activated wastes in the reactor. A rotation pattern to be set as an input condition is selected by random sampling with consideration given to the probability of each typical rotation pattern. (See C.4.)

e) Set the irradiation times.

   Concerning the range of irradiation time of the target radioactive waste in the reactor, the distribution of irradiation time is evaluated considering the actual reactor operation data (operation and idle time) and other data. The irradiation time to be set as an input condition is selected by random sampling from the irradiation time distribution. (See C.7)

f) Set the neutron fluence rates.

   The neutron fluence rate distribution in the reactor is evaluated using a verified and validated calculation code. As a next step, the neutron fluence rate to be set as an input condition is selected from the neutron fluence rate distribution during the evaluated irradiation period with consideration given to the irradiation point selected in step b) and the rotation pattern selected in step d). (See C.6)

g) Set the activation cross sections.

The activation cross section of the target element is selected as an input condition with consideration given to the neutron fluence rate selected in step f). (See C.6) These are normally included as a standard library connected to the calculation code.

h) Prepare input data sets for calculation.

The input conditions evaluated and selected in the process from step b) to step g) are regarded as one set of input data for activation calculations. When the selected number of input data sets is insufficient, it is necessary to return to step b) and carry out additional input data evaluation and selection for subsequent activation calculations. However, when the number of input data sets is judged to be sufficient, activation calculations are carried out using these data sets as a database for input conditions.

NOTE The method for evaluating the sufficiency of the number of data sets is specified in 4.3.3.

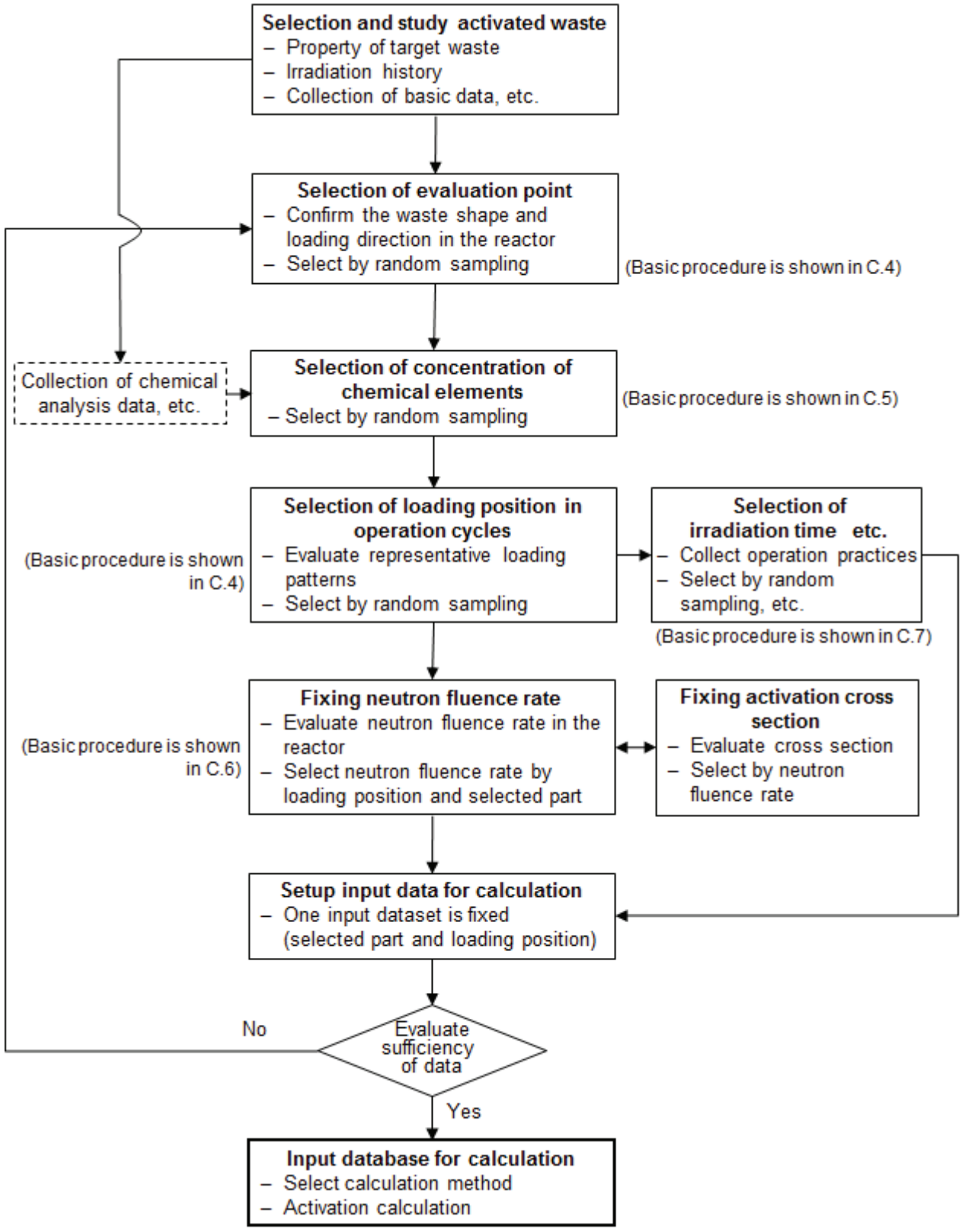


Figure C.1 — Basic setting flow for range method of input data for activation calculation

C.4 Selecting the evaluation point for the target radioactive waste

C.4.1 Setting of the irradiation position on the basis of the configuration and installation direction of the target radioactive waste

To set the neutron fluence rates such as the neutron flux and neutron spectrum, it is necessary to select the irradiation position for the target radioactive waste and identify the neutron fluence rates at the selected position in the reactor. As a first step, the irradiation position is selected with consideration given to the configuration of the target radioactive waste and its installation direction and location in the reactor. Next, appropriate neutron fluence rates are set for each irradiation position selected.

Table C.2 shows a basic philosophy for setting the irradiation position for use in activation calculations, considering the configuration and installation direction of the target radioactive waste.

Table C.2 — Basic philosophy in setting the irradiation position to be evaluated in the target radioactive waste

Shape of target radioactive waste and an installing direction in the reactor[a]	Conditions should be considered[b]	Example of activated waste[c]	Probability distribution of the irradiation position of the target radioactive waste
Axial direction of reactor	Axial installation position of target radioactive waste in the reactor	Channel box, control rod, core shroud, etc.	Uniform distribution
Radial direction of reactor	Radial installation position of the target radioactive waste in the reactor	Upper grid, etc.	Probability distribution according to the area ratio of the evaluated parts of target radioactive waste

[a] The shape of evaluated waste and its installation direction inside/outside of the reactor (installation in the axial direction of the reactor, installation in the radial direction of the reactor, etc.).

[b] Special consideration should be given to the shape of the target radioactive waste and its installation direction in the reactor from the viewpoint of the neutron fluence rate/neutron spectrum.

[c] Typical examples of representative radioactive wastes which have an installation position extending in the axial or radial direction of the reactor.

C.4.2 Setting of the irradiation position of the target radioactive waste in the reactor in consideration of the rotation of its installation position

The basic philosophy in setting the irradiation position of the target radioactive waste, depending on whether its installation position changes or rotates in each reactor operation cycle is shown in Table C.3. This influences the part of the target radioactive waste which is most exposed to the neutron fluence, and thus the uniformity of the radioactivity in the component.

Table C.3 — Basic philosophy in setting the installation position of the target radioactive waste inside/outside the reactor

Installation position of target radioactive waste in the reactor[a]	Conditions should be considered	Example of activated waste	Probability distribution to be considered regarding irradiation point in the reactor
Rotating position	Installation position rotating in each reactor operation cycle	Channel box[b], etc.	Actual distribution or representative patterns of the installation position in the reactor[f]
	Insert position changing in the reactor for fuel burnup control	Control rod for PWR[c], BWR[d], etc.	Actual distribution or representative patterns of the insert position in the reactor[f]
Fixed position	Loading position fixed during the reactor operation period	Core shroud, upper grid, etc.[e]	Fixed value

[a] Condition setting regarding the rotation of the installation position in the reactor.

[b] Consideration is required for the installation position rotating in each reactor operation cycle (central position, outermost position, etc.).

[c] In the case of PWR control rods, consideration is required for each axial installation position during the rated output operation (the position of inserted control rods during the rated output operation and the position of extracted control rods during the rated output operation).

[d] In the case of BWR control rods, consideration is required for the installation position in the reactor and the time during which control rods are set at the insert position during the rated output operation.

[e] These radioactive wastes are irradiated at the fixed position in the reactor throughout the irradiation period.

[f] Irradiation position setting is based on actual distribution data such as the installation and inserting positions after rotation, or sometimes based on representative patterns (for instance, a pattern which gives higher values in activity concentration evaluation).

C.5 Setting the elemental composition

C.5.1 Selection of the method for setting the elemental composition

Concerning the method for setting the elemental composition, it is considered that either of the following two methods can be selected according to the elemental composition data collected and the type of the activity concentration determining method selected:

— a method in which the elemental composition is set based on representative element analysis data on the target radioactive waste (such as average value);

— a method in which the elemental composition is set by setting the concentration distribution or range of element analysis data on the target radioactive waste.

C.5.2 Setting of the basic profile of the concentration distribution shape for each chemical element

Concerning the concentration distribution of each chemical element contained in each kind of material, the basic profile of the concentration distribution shape can be set by considering the control specification and method for each chemical element in the material production process. A basic philosophy for setting the distribution shape profile is shown in Table C.4.

In general, chemical elements contained in materials are divided broadly into three groups: “main elements” whose contents are adjusted to the specified alloy composition control targets in the material production process, “impurity elements” whose contents are controlled by a specified upper limit for the alloy, and “trace elements” for which no control targets are specified. Therefore, either

normal distribution or log-normal distribution can be selected for the basic profile of the concentration distribution shape for each element, depending on whether control targets are specified for the elements which require content adjustment in the material production process.

For the main elements, uniform distribution can be conservatively applied, in which concentration data close to the upper limit of the distribution more frequently appear than in the case of normal distribution. Additionally, it is considered that the basic profile of the concentration distribution shape for each element can be set by considering the concentration distribution based on element analysis data collected.

Table C.4 — Basic philosophy in setting basic type of distribution of chemical element concentrations

Chemical element condition	Main elements	Impurity elements	Trace elements
	Controlled in a certain range of concentration	Controlled with an upper limit of concentration	Non-controlled
Basic philosophy	Main chemical elements of materials which are manufactured in specific factories under lot-based quality control. Their contents are controlled within the target ranges specified by national industrial standards of material, and their concentration ranges (concentration distribution) are comparatively narrow.	Chemical elements which are reduced or controlled in a certain manufacturing process as impurity elements contained in manufactured materials. Their contents are controlled below comparatively low control values, and the concentration distribution of each element is able to reflect its concentration distribution in the nature.	Chemical elements which are not controlled. The content of each element reflects its concentration distribution appearing in nature.
Reference concentration distribution of each chemical element	Normal distribution	Log-normal distribution	Log-normal distribution

C.5.3 Setting of the concentration distribution condition according to the number of element analysis data

In setting the distribution condition of chemical element concentrations, it is desirable that, with consideration given to representativeness and conservativeness, the methods listed in cases a) to d) below should be applied according to the lot of material of the target radioactive waste and the number of element analysis data values collected.

For standard deviations, appropriate values can be set with consideration given to conservativeness. In the case of “impurity elements” and “trace elements”, it is also acceptable to set appropriate values for standard deviations by reference to the distribution shape for the same element or an element similar in chemical properties.

However, when reference is made to the distribution shape for the same element or an element similar in chemical properties, it is necessary to collect data on the reference element and then verify that the standard deviation has no concentration dependence in the case of the same element, and that there is no difference in standard deviation between the target element and the reference element similar in chemical properties.

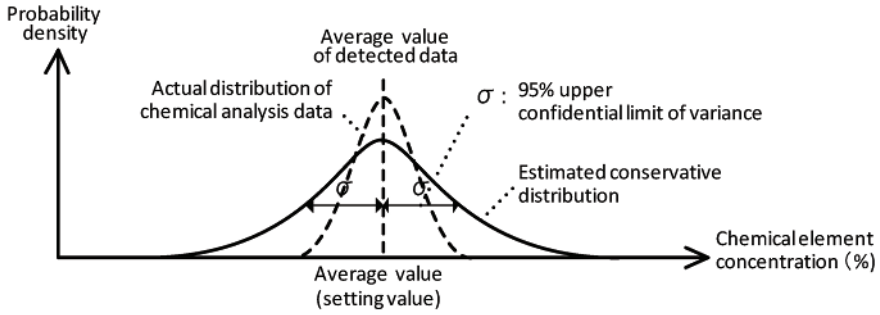
a) When element analysis data are sufficient

   When a sufficient number of element analysis data are collected for the target radioactive waste (material) and this database is representative, there is no particular need to consider conservativeness in setting the concentration distribution condition and it is possible to set it by using the average values, standard deviation, minimum/maximum values, etc. of element analysis data on the target radioactive waste.

b) When element analysis data are comparatively few

In some cases, the number of element analysis data is not sufficient for the target radioactive waste (material). In this case, the chemical element concentration distribution of the target radioactive waste can be set by applying average values and standard deviations in which conservativeness is considered by such a means as applying an upper limit to the confidence of element analysis data distribution in consideration of the quantitative insufficiency of analysis data.

EXAMPLE Image of setting of concentration distribution when number of data are comparatively few.



c) When element analysis data are very few

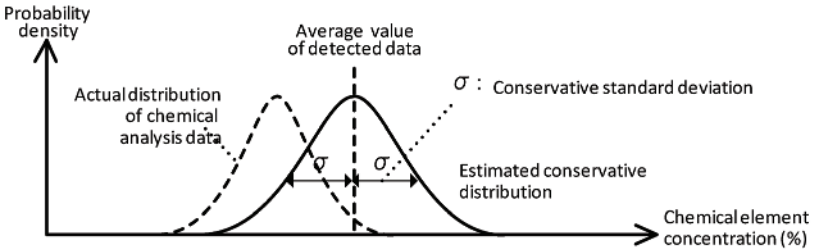
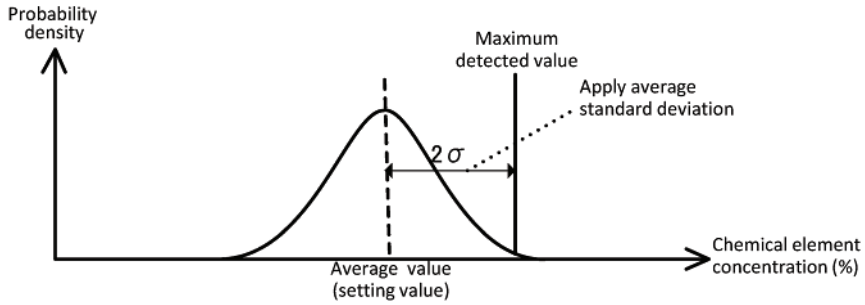
When most of the analysis data are below the detection limit (i.e. only one or two points are above the detection limit), it is impossible to calculate standard deviations and other values. Either of the following two methods is applicable to set average values and standard deviations.

— The concentration distribution can be estimated by using one or two detected values and/or applying a conservative value of standard deviation using same element data.

— The concentration distribution is assumed in a range of lower concentrations than the minimum detection values in element analysis data.

Table C.5 shows the basic philosophy in determining average values and standard deviations which are necessary to set a distribution of chemical element concentrations when the number of element analysis data is very few.

Table C.5 — Method for setting the chemical element concentration condition when chemical element analysis data are very few

Method	Outline of method	Consideration of conservativeness
Assuming the average concentration and standard deviation from detected values	The concentration distribution is set by applying the following values: — average value: apply detected value; — standard deviation: apply conservative standard deviation.	As shown in the following figure. Average concentration: Conservativeness is considered by applying the average of detected values. Standard deviation of concentration: Conservativeness is considered by applying a conservative standard deviation derived from detected values (e.g. Set the standard deviation such that it includes 90 % of the measured values above the detection limit).
	 EXAMPLE 1 Image of applying the average of detected data to the average of estimated distribution.	
Assuming the concentration distribution in a range below the detection limit	The concentration distribution is set in a range of lower concentrations than the minimum detected value in element analysis data.	As shown in the following figure. Average concentration: The average is determined by assuming maximum detected value located +2σ value of concentration distribution. Standard deviation of concentration: Assume average standard deviation of distribution from database (e.g. apply standard deviation evaluated from same element data in the nature or from element data which chemical property element).
	 EXAMPLE 2 Image of setting distribution in the case of minimum detected value.	

d) When element analysis data contain detection limit values only

When all element analysis data on the target radioactive waste (material) show detection limit values only, the average values and standard deviations can be set using any of the following three methods.

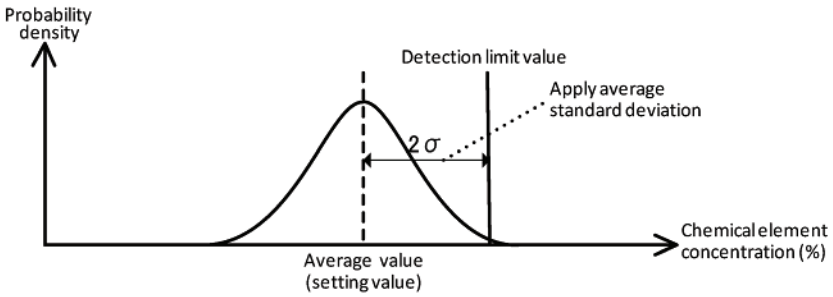
— Detection limit values in element analysis data are used as average values.

— Concentration distributions are set in a range of low concentrations which starts from detection limit values in element analysis data.

— The concentrations of parent elements are estimated from activity concentration data on radionuclides contained in a target radioactive waste of known irradiation history.

Table C.6 shows the philosophy in determining average values and standard deviations which are necessary to set the concentration distribution of chemical elements when element analysis data contain detection limit values only.

Table C.6 — Method for setting the concentration distributions of chemical elements when element analysis data contain detection limit values only (i.e. all data are not detectable)

Method	Outline of method	Consideration of conservativeness
Using detection limit values as they are	Detection limit values in element analysis data are used as average concentrations in activation calculations.	Sufficient conservativeness is considered by applying the minimum detected values in element analysis data to set the element concentrations.
Assuming a concentration distribution from detection limit values	Concentration distribution is set in a range of low concentrations below the detection limit values in element analysis data.	As shown in the following figure. Average concentration: The average is determined conservatively by assuming detection limit is located +2σ value of concentration distribution. Standard deviation of concentration: Assume average standard deviation of distribution from database (e.g. apply standard deviation evaluated from same chemical element data in the nature or from element data which chemical property element is similar with evaluated chemical).
	 EXAMPLE Image of setting distribution under non-detectable value.	
Estimating from radiochemical analysis data	The concentrations of parent elements are estimated from radiochemical analysis data on target radioactive waste whose irradiation condition is known definitely.	Appropriate conservativeness is considered, if necessary from the standpoint of radiochemical analysis data.

C.6 Setting of neutron fluence rate

C.6.1 Basic considerations in setting the neutron fluence rate

The neutron fluence rate is divided broadly into two factors: “the neutron flux/neutron spectrum” and “the activation cross section”. The neutron fluence rate varies depending on the reactor type/fuel condition, the kind of the target radioactive waste, the evaluated part of the target radioactive waste, etc. Therefore, in order to determine the neutron fluence rate by calculations, it is necessary to create a calculation model reflecting the actual condition of the reactor and make calculations using

computational codes and group constants suited for the purpose with consideration given to accuracy requirements, the condition of the neutron field, etc.

Table C.7 shows basic considerations in setting the neutron fluence rate.

Table C.7 — Basic considerations in setting the neutron fluence rate

Consideration item		Main items that need to be considered
Fuel condition		— Enrichment[a] — Burnup[a] — Fuel type[b]
Position in the reactor[c]	Axial direction	— Position of the target radioactive waste in axial direction of reactor[d] — Insertion ratio of the target radioactive waste in axial direction of reactor[e]
	Radial direction	— Position of the target radioactive waste in radial direction of the reactor[d] — Rotation of the target radioactive waste in the radial direction of the reactor[e]
Others		— Void distribution (BWR)[f] — Boron concentration (PWR) — Temperature distribution — Flux depression by self-shielding-effects[g]

[a] Enrichment of the fuel and burnup used during the irradiation period of the target radioactive waste.

[b] Fuel type i.e. UO2, MOX.

[c] Positions of the source of neutron generation, the target radioactive waste, and other materials which affect the behaviours of neutrons such as moderation, reflection, absorption, and leakage.

[d] This consideration applies not only to the case where the neutron fluence rate varies depending on parts of the target radioactive waste but also to the case where the target radioactive waste extends in the inside or to the outside of the reactor.

[e] This consideration applies to the case where the target radioactive waste travels to a range where the neutron fluence rate varies.

[f] This consideration applies to BWR plants. (In the case of BWR plants, the nuclear reaction is suppressed and the output falls with an increase in the amount of vapour (void) generated at the reactor core.

[g] When the target radioactive waste is a strong neutron absorber, consideration should be given to the neutron absorption effect on the neutron fluence rate because such waste has a depression effect (distortion of the neutron fluence rate distribution).

On the other hand, the activation cross section is normally automatically determined using an existing cross-section library attached to an applicable activation calculation code. However, depending on the result of evaluating the neutron fluence rate/neutron spectrum, it is sometimes possible to create a simplified cross section with consideration given to the characteristics of the neutron spectrum of the activated area. The use of a simplified cross section can reduce computational resource requirements, especially when large numbers of calculations are performed.

When existing cross-section libraries are applied, normally the difference between the libraries basically has only a slight effect on activation calculations as far as the target radioactive wastes are similar in terms of the type of reactor and the chemical composition of fuel. In this case, it is necessary to identify in advance the characteristics of the neutron spectrum which affect the creating of the one-group effective cross section.

C.6.2 Setting procedure for the neutron fluence rate

In setting the neutron fluence rate, either of the following two methods can be applied according to the method used for determining activity concentrations.

a) Setting for each individual item

   A method in which the neutron fluence rate is set in detail for each target radioactive waste on the basis of its irradiation history. The appropriate values are set for the activation cross section after confirming the representativeness of the neutron spectrum.

b) Setting using representative values

A method in which the neutron fluence rate is set so as to appropriately represent two or more target radioactive wastes or give a conservative evaluation result (i.e. higher activity concentrations) when their irradiation histories are considered.

When the neutron fluence rate is determined using activity concentration ratios or determined from two or more calculation results, the neutron fluence rate range can be set to appropriately represent two or more target radioactive wastes, instead of setting individual neutron fluence rates.

When the neutron fluence rate/neutron spectrum is evaluated for condition setting in activation calculations, it is necessary that reliable computational codes and validated group constants should be applied to the calculations with due consideration given to their intended uses. A main current approach to neutron calculations is to carry out nuclear reactor core performance calculations (the nuclear physics parameters calculation code for fuel assembly units and a three-dimensional nuclear thermal-hydraulic analysis) and solve the Boltzmann neutron transport equation. Applicable calculation techniques are the Sn method (the differential neutron transport equation), the Monte Carlo method, the direct integration method, etc. In addition to these calculation techniques, the extrapolation approach method and other methods are applied according to the characteristic features of each method.

EXAMPLE 1 JENDL (Japan Atomic Energy Agency), ENDF/B (Brookhaven National Laboratory), etc. are available as cross-section libraries (nuclear data).

EXAMPLE 2 ANISN, DOT, and TORT (Oak Ridge National Laboratory) are available as transport calculation codes using the Sn method, and MCNP (Los Alamos National Laboratory) is available as a transport calculation code using the Monte Carlo method.

C.7 Setting procedure for the irradiation condition

C.7.1 Basic philosophy in setting the irradiation condition

Irradiation time and non-irradiation time during irradiation periods are set as the input conditions of the target radioactive waste. An irradiation history can include multiple periods of irradiation (neutron flux on) and decay (neutron flux off). For comparatively short half-life nuclides such as ^{60}Co (nuclides with half-life shorter than about five years), it is necessary to consider the effect of decay during irradiation periods in the case of the target radioactive waste irradiated over a long period of time. Table C.8 shows a basic philosophy in setting irradiation time and non-irradiation time during irradiation periods.

Table C.8 — Basic philosophy in setting irradiation time and non-irradiation time during plant life

Condition item	Irradiation condition setting procedure		
	Setting method	**Basic philosophy in condition setting**	**Target radioactive waste**
Irradiation time (total)	Distribution-based setting	— When the irradiation time of the target radioactive waste is not necessarily uniform, an appropriate distribution pattern (e.g. normal distribution) is set based on actual data on irradiation time (e.g. total). — The frequency distribution is set using average values, standard deviations, etc. according to the distribution pattern of irradiation time.	Channel-box, control rod, etc.
	Case-by-case setting	— When irradiation time data are the same for each reactor and each target radioactive waste, uniform irradiation time is set considering the actual data.	Core shroud, upper grid, etc.
Non-irradiation time[a] (during the irradiation period)	Uniform setting	— The ratio calculated from the average frequency of non-irradiation periods and the average non-irradiation period (e.g. availability factor) is set uniformly based on the actual data on non-irradiation time (e.g. total) and frequency. — Non-irradiation time is set uniformly for each operation cycle, based on the calculated ratio of irradiation and non-irradiation time.	All kinds of target radioactive wastes
	Case-by-case setting	— Non-irradiation frequency and time are set case by case based on the actual irradiation data on the target radioactive waste.	

[a] Basically, the storage time elapsed after the irradiation period is not included in the irradiation condition, and it is desirable to make decay corrections on the evaluation.

C.7.2 Considerations in setting the irradiation history

In setting the irradiation history, i.e. irradiation time and non-irradiation time, it is necessary to set the irradiation history in detail for each target radioactive waste according to the type of the activity concentration determining the method selected or to set an irradiation history which appropriately represents two or more target radioactive wastes (or a conservative condition which gives a higher value in activity concentration evaluation).

When the concentration ratio or the conversion factor is used, or when the irradiation history is determined from two or more calculation results, it is also possible to set the range of the condition which appropriately represents two or more target radioactive wastes, instead of setting the irradiation history for each individual target radioactive waste. Table C.9 shows basic factors to be considered in setting the irradiation history.

Table C.9 — Basic factor needs to consider for setting irradiation condition

Condition items		Main consideration points
Irradiation time		— Total irradiation time[a]
Non-irradiation (nuclide decay) time	After the irradiation period	— Corrections for nuclide decay after the irradiation period[b] or — Appropriate corrections on the date of generation of target radionuclides[b c]
	During the irradiation period	— Time and frequency of plant shutdown[d]

[a] Of target radionuclides, those which have shorter half-lives than the plant irradiation period are only slightly affected by the initial part of the plant operation history but greatly affected by the neutron fluence rate on immediately before activity evaluation. On the other hand, those which have longer half-lives than the plant irradiation period are affected by the total irradiation time.

[b] This applies to comparatively short half-life nuclides.

[c] This applies to the case where concentration ratios are used and key nuclides are non-destructively measured from outside.

[d] For nuclides, such as ^{60}Co, which have almost the same half-lives as the irradiation time, it is necessary to consider nuclide decay during the irradiation period.

The basic considerations for setting irradiation history shown in Table C.9 can be considered under the philosophy described below.

a) Irradiation time

Irradiation time greatly affects activity concentrations. In determining the activity concentrations of the target radioactive waste, it is therefore necessary to set an appropriate irradiation time so as to represent the activity concentrations of the target radioactive waste, or so as to give higher values in activity concentration evaluation.

On the other hand, the effect of irradiation time length on activity concentration ratios is comparatively small as long as the irradiation time of the target radioactive waste is equal to that of wastes generated during irradiation periods (continuous irradiation of 10 years or so). However, concerning the irradiation time of decommissioning wastes (for instance, corresponding to long-term irradiation over a period of several decades), activity concentrations are affected by irradiation time in the case of comparatively short half-life nuclides such as ^{60}Co. For this reason, when concentration ratios to such comparatively short half-life nuclides are applied, it is necessary to give appropriate considerations, such as changing the condition setting in activity calculations according to the total irradiation time length of the target radioactive waste.

b) Non-irradiation time (after all irradiation terminates)

The activity concentrations of comparatively short half-life nuclides such as ^{60}Co are affected by decay which takes place after irradiation terminates completely. It is therefore necessary to make corrections on calculation results with appropriate consideration given to decay periods. (However, when making no decay corrections gives higher values in activity concentration evaluation, the results of after-irradiation activation calculations can be applied as they are.)

When activity concentration ratios combining comparatively short half-life and comparatively long half-life nuclides are used as combinations of key nuclides and difficult-to-measure nuclides, it is necessary to consider appropriate decay corrections for the results of measurements such as non-destructive external measurement of key nuclides.

EXAMPLE When ^{60}Co is applied as a key nuclide, decay correction is considered necessary. For instance, the activity concentration of ^{60}Co measured by non-destructive assay measurement is decay-corrected for the date of waste generation and multiplied by the "concentration ratio" calculated by activation calculations, whereby the concentration of a difficult-to-measure nuclide for the date of waste generation is calculated and then the activity concentration of the difficult-to-measure nuclide is decay-corrected to that for the date of activity evaluation.

c) Non-irradiation time (during irradiation periods)

When the total non-irradiation time is the same, the differences in length of each period of non-irradiation time during irradiation periods have a basically small influence on either of the activity concentration and concentration ratio even in the case of comparatively short half-life nuclides such as ^{60}Co. Therefore, in setting the conditions for activation calculations, it is appropriate to apply such a method as setting non-irradiation time (total) according to the actual reactor operation condition, and then equally distributing it over the irradiation period.

However, when a long period of non-irradiation time is included even if the total non-irradiation time is the same, the effect of decay on decreases in activity concentration is sometimes great during some period of time after a long period of non-irradiation terminates and the reactor operation restarts. It is therefore necessary to consider this when a long period of non-irradiation takes place.

Annex D
(informative)

Dealing with uncertainties

D.1 General

Annex D provides explanatory notes and gives examples and practices to assist the understanding or use of this International Standard.

D.2 Representativeness and conservativeness

One of the main objectives of theoretical calculations is to compare the output results with measured values, validating the model developed and providing a measure of its representativeness and/or conservativeness.

In the context of this International Standard, differences between the actual value and the calculated one have to be evaluated. The actual value is never known, unless an infinite number of calculations/measurements are performed which span the possible range of all of the variables involved (chemical composition, neutron fluence rates, irradiation condition, etc.). Thus, it is better to analyse the concept of representativeness by quantifying the following: "mean square difference between the actual value and the calculated or consigned value". By this, representativeness quantifies both accuracy (bias) and precision (uncertainty).

$$R^2 = b^2 + S^2 \tag{D.1}$$

where

R is representativeness;

b is the bias or skew, the mean difference between the consigned activity value calculated and the actual activity value (High accuracy means low bias, and low accuracy means high bias.);

S^2 is the variance of the consigned value (uncertainty). Low variance means high precision and high variance means low precision.

Thus, representativeness is characterized by the absence or the establishment of a bias, and by an acceptable variance. In the context of this International Standard, this is an objective and scientific definition of representativeness. Therefore, analysis of representativeness means that both uncertainties and accuracy need to be taken into account in this process.

Often, an envelope of the activity distribution, or conservativeness, can be assumed in order to safely guarantee the limits involved (Waste Acceptance Criteria, radiological protection requirements, shielding design, etc.). In these cases, uncertainty analyses have to be performed to check the confidence interval of the calculated conservative activity.

The determined activity value mainly should ensure accuracy or conservativeness, as established by the context and scope of the calculation, and the uncertainty should be properly evaluated for the stated purpose.

Uncertainty of the activity value can be evaluated using normal tools devised for that purpose. For example, the uncertainty propagation law methodology, described in ISO/IEC GUIDE 98-3:2008[7] is the most common method that addresses the solution of the analysis involved.

D.3 Point method

For a specific item being analysed, the established values for the main parameters involved (conservative or accurate ones), namely, chemical composition, neutron fluence rates, and irradiation condition, should have their uncertainty evaluated in advance in order to properly determine the final uncertainty in the calculated value.

Once the uncertainties of the involved parameters have been determined (one is based on measurements, or the other is based on specifications, judgments, etc.), a propagation of uncertainties is applied to the theoretical calculation.

Finally, this uncertainty should be linked to the consigned value for building its confidence interval (see Figure D.1).

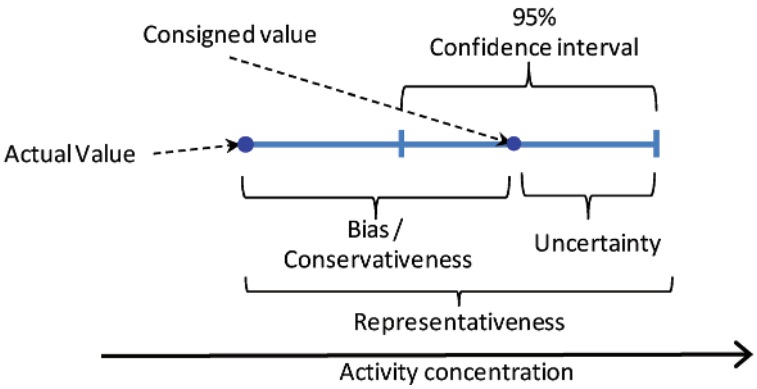


Figure D.1 — Image of representativeness analysis of the theoretical activity concentration

D.4 Range method

In this case, there are distribution functions for all or some of the main parameters involved (chemical composition, neutron fluence rates, and irradiation condition).

The distribution of the input data is managed using statistical schemes (e.g. Monte Carlo method, etc.), generating random data for each distribution function in order to produce the final input data that fit to their probability density function.

Due to the distribution of the input data, the final result also follows a distribution, and therefore the confidence interval of the calculated data can be built from such a final distribution (see Figure D.2).

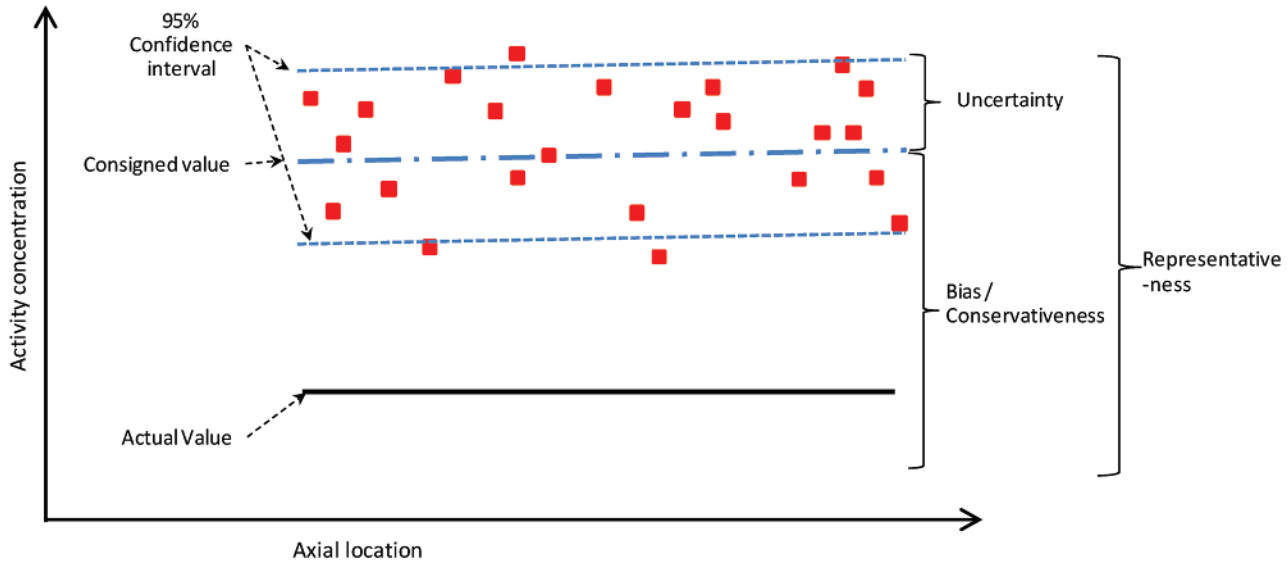


Figure D.2 — Image of representativeness analysis of activity concentration

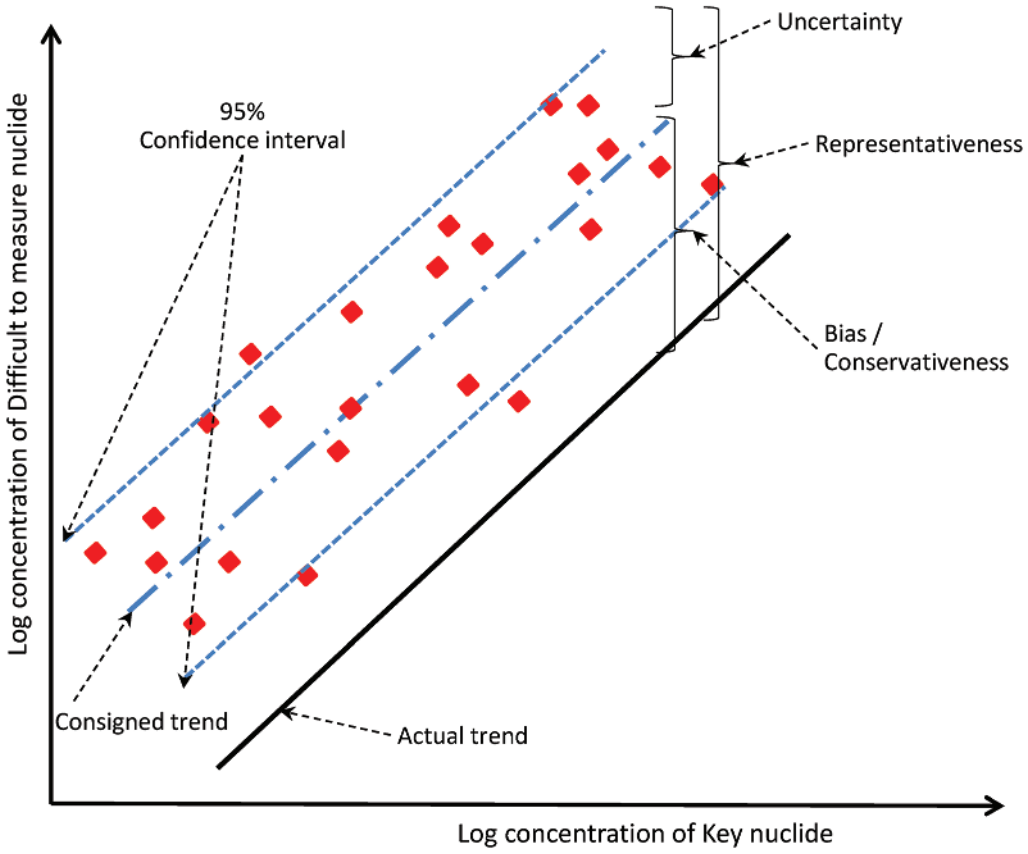


Figure D.3 — Image of representativeness analysis of activity concentration ratio

In the context of this International Standard, uncertainty only needs to include the variability that cannot be predicted by the theoretical model, the rest of the variability (as predicted by the model) can be discarded from the uncertainty. For instance, a variability of the neutron fluence rate in different parts of a large piece of material produces variability of the activity within that piece. However, this variability is not an uncertainty as a consequence of being explained by the known neutron fluence rate

range, and only the uncertainty of fluence rates in each part produces uncertainty in the activity at such part. In the same way, as can be seen in Figure D.3, the main data range of both key nuclide and difficult-to-measure nuclide concentrations are predicted by the trend, and only the distance of data from the trend is what the model cannot explain or predict (i.e. the uncertainty).

Annex E
(informative)

Reporting of results

E.1 General

Annex E provides explanatory notes and gives examples and practices to assist the understanding or use of this International Standard.

E.2 Results contents

The primary reporting of results should be presented in sufficient detail to enable a knowledgeable reviewer to trace the development of the argument and reconstruct the calculation. It should include the following information as applicable.

a) context and scope of calculation
    1) purpose of the calculation
    2) basic assumptions
    3) applicability and limitations

b) material description
    1) physical description
    2) weights
    3) material types by weight
    4) material test reports or other chemical composition reference or element analysis report
    5) neutron fluence rates by item
    6) neutron irradiation history by item

c) description of the radiation (dose rate) survey process
    1) summary of survey results (copies of original survey sheets should be retained as part of the calculation record)

d) general description of the method
    1) description of the general approach followed to perform the calculation directly or by reference (A detailed method verification and validation report for a computational code or model can be proprietary to the software vendor. However, it should be included at least as a reference to provide evidence that this step has been performed.)

e) summary of results
    1) final material basis
    2) discussion of use and application of the survey data
    3) radionuclide inventory list broken down for discrete objects included in calculation

4) other data summaries and analyses as can be required by regulatory authorities

f) discussion of uncertainty and/or validation of results

g) references (include as applicable)

1) computer programs

2) fabrication drawings

3) reactor operating data

4) component exposure histories

5) dose rate survey data (if collected)

6) methodology report (if not included)

h) evidence of reviews, verifications, etc

1) signature block on the report should include authors, reviewers, verifiers, and approvers as appropriate

E.3 Reusable results

Significant calculation segments and reports that can serve as references for other calculations should be broken out and separately reported, reviewed, verified, and approved.

E.4 Handling of proprietary information

Proprietary information should be addressed in separate documents and referenced as proprietary in the main calculation.

E.5 Reporting limits

Theoretical calculations, in contrast to measurements, don't have limits on minimum values. The values reported are limited only by the numerical processor. Cut-off levels should be established such as to limit the reporting of inconsequential and meaningless values. In general, values lower than 1 Bq/cm^3 averaged over the contents of the container would not factor significantly into the radiation risk arising from the waste nor would it add significantly to the disposal site inventory. The reporting limits should be clearly stated in the report.

E.6 Reporting precision

Achievable numerical precision is limited by the least number of significant digits in of any input. Sample results generally report three significant digits. Dose rate survey values and material test reports can be limited to one or two significant digits. Results should avoid reporting unrealistic precision. Numerical results should account for the precision of input parameters and correspondingly limit the number of significant digits reported. Rounding up of values is preferred to truncating digits.

Bibliography

[1] AESJ-SC-F015. *Basic procedure to determine the activity concentration of sub-surface disposal waste, Atomic Energy Society of Japan (AESJ) Standards*, 2010

[2] PNL-6096, "Spent fuel assembly hardware: Characterization and 10 CFR 61 Classification for Waste Disposal, Volume 2. Calculated Activity Profiles of Spent Nuclear Fuel Assembly Hardware for Pressurized Waste Reactors", Battelle Pacific Northwest Laboratory, Richland, Washington, June 1989

[3] PNL-6096, "Spent fuel assembly hardware: Characterization and 10 CFR 61 Classification for waste disposal, Volume 3. Calculated activity profiles of spent nuclear fuel assembly hardware for Boiling water reactors", Battelle Pacific Northwest Laboratory, Richland, Washington, June 1989

[4] PNL-10103, "Non fuel assembly components: 10CFR61.55 classification for waste disposal", Battelle Pacific Northwest Laboratory, September 1994

[5] Pantelias M. et al., "MCNP modelling of the Swiss LWRs for the calculation of the in- and ex-vessel neutron flux distribution", Proceedings of PHYSOR 2012-Advances in reactor physics, Knoxville TN, USA, April 15-20, 2012

[6] Brack D. et al., "Removal of pins and rollers, dose rate measurements, calculations of activity and the conception of control rod packaging at KKL", Proceedings of KONTEC 2009, 9th International Symposium "Conditioning of radioactive operational and decommissioning wastes", Dresden, Germany, April 15-17, 2009

[7] ISO/IEC GUIDE 98-3:2008, *Guide to Expression of Uncertainty in Measurement*

[8] ISO 80000-10; *Quantities and units — Part 10: Atomic and nuclear physics.* [1)]

[9] ISO 21238:2007, *Nuclear energy — Nuclear fuel technology — Scaling factor method to determine the radioactivity of low- and intermediate-level radioactive waste packages generated at nuclear power plants*

1) Revises ISO 31-10:1992.

ICS 27.120.30
Price based on 45 pages

This page deliberately left blank

British Standards Institution (BSI)

BSI is the national body responsible for preparing British Standards and other standards-related publications, information and services.

BSI is incorporated by Royal Charter. British Standards and other standardization products are published by BSI Standards Limited.

About us

We bring together business, industry, government, consumers, innovators and others to shape their combined experience and expertise into standards -based solutions.

The knowledge embodied in our standards has been carefully assembled in a dependable format and refined through our open consultation process. Organizations of all sizes and across all sectors choose standards to help them achieve their goals.

Information on standards

We can provide you with the knowledge that your organization needs to succeed. Find out more about British Standards by visiting our website at bsigroup.com/standards or contacting our Customer Services team or Knowledge Centre.

Buying standards

You can buy and download PDF versions of BSI publications, including British and adopted European and international standards, through our website at bsigroup.com/shop, where hard copies can also be purchased.

If you need international and foreign standards from other Standards Development Organizations, hard copies can be ordered from our Customer Services team.

Subscriptions

Our range of subscription services are designed to make using standards easier for you. For further information on our subscription products go to bsigroup.com/subscriptions.

With **British Standards Online (BSOL)** you'll have instant access to over 55,000 British and adopted European and international standards from your desktop. It's available 24/7 and is refreshed daily so you'll always be up to date.

You can keep in touch with standards developments and receive substantial discounts on the purchase price of standards, both in single copy and subscription format, by becoming a **BSI Subscribing Member**.

PLUS is an updating service exclusive to BSI Subscribing Members. You will automatically receive the latest hard copy of your standards when they're revised or replaced.

To find out more about becoming a BSI Subscribing Member and the benefits of membership, please visit bsigroup.com/shop.

With a **Multi-User Network Licence (MUNL)** you are able to host standards publications on your intranet. Licences can cover as few or as many users as you wish. With updates supplied as soon as they're available, you can be sure your documentation is current. For further information, email bsmusales@bsigroup.com.

BSI Group Headquarters

389 Chiswick High Road London W4 4AL UK

Revisions

Our British Standards and other publications are updated by amendment or revision.

We continually improve the quality of our products and services to benefit your business. If you find an inaccuracy or ambiguity within a British Standard or other BSI publication please inform the Knowledge Centre.

Copyright

Useful Contacts:

Customer Services
Tel: +44 845 086 9001
Email (orders): orders@bsigroup.com
Email (enquiries): cservices@bsigroup.com

Subscriptions
Tel: +44 845 086 9001
Email: subscriptions@bsigroup.com

Knowledge Centre
Tel: +44 20 8996 7004
Email: knowledgecentre@bsigroup.com

Copyright & Licensing
Tel: +44 20 8996 7070
Email: copyright@bsigroup.com

bsi.

...making excellence a habit.™