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Nuclear energy — Determination of neutron fluence and displacement per atom (dpa) in reactor vessel and internals

Énergie nucléaire — Détermination de la fluence neutronique et du déplacement par atome (dpa) dans la cuve et les internes du réacteur

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Foreword

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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).

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For an explanation on the voluntary nature of standards, the meaning of ISO specific terms and expressions related to conformity assessment, as well as information about ISO's adherence to the World Trade Organization (WTO) principles in the Technical Barriers to Trade (TBT) see the following URL: www.iso.org/iso/foreword.html.

This document was prepared by Technical committee ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 6, *Reactor Technology*.

This document is based on the ANSI/ANS 19.10-2009 but extends to cover the evaluation of irradiation damage due to neutron fluence.

Introduction

This document is intended for use by

a) those involved in the determination of exposure parameters for the prediction of irradiation damage to the vessel and to the internals of a nuclear reactor, where the exposure parameters can be neutron fluence and/or displacements per atom (dpa),

b) those involved in the determination of material properties of irradiated reactor vessel and reactor internals,

c) regulatory agencies in licensing actions such as the writing of Regulatory Guides, analysis of reports concerning the integrity and material properties of irradiated pressure vessels and reactor internals.

Nuclear energy — Determination of neutron fluence and displacement per atom (dpa) in reactor vessel and internals

1 Scope

This document provides a procedure for the evaluation of irradiation data in the region between the reactor core and the inside surface of the containment vessel, through the pressure vessel and the reactor cavity, between the ends of active fuel assemblies, given the neutron source in the core.

NOTE These irradiation data could be neutron fluence or displacements per atom (dpa), and Helium production.

The evaluation employs both neutron flux computations and measurement data from in-vessel and cavity dosimetry, as appropriate. This document applies to pressurized water reactors (PWRs), boiling water reactors (BWRs), and pressurized heavy water reactors (PHWRs).

This document also provides a procedure for evaluating neutron damage properties at the reactor pressure vessel and internal components of PWRs, BWRs, and PHWRs. Damage properties are focused on atomic displacement damage caused by direct displacements of atoms due to collisions with neutrons and indirect damage caused by gas production, both of which are strongly dependent on the neutron energy spectrum. Therefore, for a given neutron fluence and neutron energy spectrum, calculations of the total accumulated number of atomic displacements are important data to be used for reactor life management.

2 Normative references

The following documents are referred to in the text in such a way that some or all of their content constitutes requirements of this document. For dated references, only the edition cited applies. For undated references, the latest edition of the referenced document (including any amendments) applies.

ANSI/ANS 19.10, *Methods for determining neutron fluence in BWR and PWR pressure vessel and reactor internals*

ASTM E170-16a, *Standard Terminology Relating to Radiation Measurements and Dosimetry*

3 Terms and definitions

For the purposes of this document, the terms and definitions given in ANSI/ANS 19.10, ASTM E170-16a and the following apply.

ISO and IEC maintain terminological databases for use in standardization at the following addresses:

— ISO Online browsing platform: available at https://www.iso.org/obp

— IEC Electropedia: available at http://www.electropedia.org/

3.1
accuracy of a measured/calculated value
difference between the "real" and the measured/calculated value, typically due to systematic errors in the measurement/calculation procedure

3.2
benchmark experiment
well-defined set of physical experiments with results judged to be sufficiently accurate for use as a calculational reference point

Note 1 to entry: The judgment is made by a group of experts in the subject area.

3.3
best-estimate fluence
most accurate value of the fluence based on all available measurements, calculated results, and adjustments based on bias estimates, least-squares analyses, and engineering judgment

3.4
calculational methodology
mathematical equations, approximations, assumptions, associated parameters, and calculational procedure that yield the calculated results

Note 1 to entry: When more than one step is involved in the calculation, the entire sequence of steps comprises the "calculational methodology."

3.5
code benchmark
comparison to the results of another code system that has been previously validated against experiment(s)

3.6
continuous-energy cross-section data
cross-section data that are specified in a dense point-wise manner that comprises the energy range

3.7
dosimeter reaction
neutron-induced nuclear reaction with a product nuclide having sufficient activity to be measured and related to the incident neutron fluence

3.8
displacements per atom (dpa)
mean number of times each atom of a solid is displaced from its lattice site during an exposure to displacing radiation, as calculated following standard procedures

3.9
least-squares adjustment procedure
method for combining the results of neutron transport calculations and the results of dosimetry measurements that provides an optimal estimate of the fluence by minimizing, in the least-squares sense, the calculation-to-measurement differences

3.10
multigroup cross-section data
cross-section data that have been determined by averaging the continuous-energy cross-section data over discrete energy intervals using specified weighting functions to preserve reaction rates

3.11
neutron fluence
time-integrated neutron fluence rate (i.e. the time-integrated neutron flux) as expressed in neutrons per square centimeter

3.12
precision of a measured/calculated value
standard deviation (if available from a set of repeated measurements/calculations) of the distribution of the measured or calculated physical value

3.13
reactor internals
reactor structure components that are within the pressure vessel such as the core baffle, core barrel, thermal shield, lower and upper core plates in PWRs and BWRs

3.14
solution variance
measure of the statistical variance of the Monte Carlo transport solution due to a finite number of particle histories

Note 1 to entry: Mathematically, it is the second central moment of the distribution about the mean value, which is used to measure the dispersion of the distribution about the mean.

4 Transport theory calculational models

4.1 General

4.1.1 Output requirements

The transport calculations need to be able to determine accurately the neutron flux or fluence distributions, and/or other response parameters such as reaction rates or dpa for the analysis of integral dosimetry measurements and for the prediction of irradiation damage to reactor pressure vessels and its internals.

Calculation methodologies described in this document focus on neutron fluence for determining radiation embrittlement of reactor vessel materials.

While neutron fluence (E > 1,0 MeV) (where neutron fluence (E > 1,0 MeV) represents the fluence of neutrons with energy above 1,0 MeV) has frequently been selected as the exposure parameter for determining radiation embrittlement of reactor vessel materials, the procedures in this document extend to include fluence spectrum above 0,1 MeV, in addition to thermal fluence below 0,625 eV.

Some parameters of the calculations would be determined based on

— direct use of the results: design or comparison to measurements (which imply envelope or best-estimate results, respectively),

— required response functions: (E > 1,0 MeV) neutron flux, (E > 0,1 MeV) neutron flux, thermal neutron flux (E < 0,625 eV), dpa/s, dosimeter reaction rates;

NOTE The figures for flux, given as examples of upper or lower limit, depend on the application.

— location(s) of interest: fineness of the spatial meshing.

4.1.2 Methodology: transport calculations with fixed sources

In the practice suggested in this document, a source distribution throughout the core is prepared using the results of core physics calculations; multidimensional transport theory calculations then are performed to propagate the neutrons to regions outside the core.

This document uses codes based on transport theory to determine multigroup three-dimensional flux distributions and to evaluate the reaction rates of dosimetry materials or dpa properties through proper use of response functions or cross sections.

Transport theory calculations should be performed using deterministic discrete ordinates (S_N)[2] or statistical Monte Carlo[3] approaches as discussed in 4.2.2 and 4.2.3, respectively. Other transport methods may be used if they are part of a benchmarked methodology.

4.2 Transport calculation

4.2.1 Input data

The four major types of input required are.

a) Material composition:

The material compositions should represent the physical configuration as closely as practical. Material compositions and densities (consistent with the geometric model), coolant and moderator density (consistent with operating conditions) are required.

b) Geometric model:

The geometric model should represent the physical configuration as closely as practical. "As-built" dimensions of the reactor configuration should be used when available.

c) Cross-section data:

Appropriate cross-section data should be used. Cross-section sets may be used if they are part of a benchmarked methodology. Major considerations include:

1) the accuracy of the data evaluation (ENDF/B, JEFF, JENDL...);

2) the energy group structure;

3) the order of the scattering anisotropy (i.e. P_n expansion);

4) the method used for group-collapsing.

d) Core neutron source:

The determination of the neutron source should include the temporal, spatial, and energy dependence together with the absolute source normalization. The spatial distribution(s) of sources shall be representative of the integrated or averaged distribution(s) during the considered irradiation duration(s). The neutron distribution should be accurate especially at the periphery of the core, in order to properly determine the fluence on the Reactor Pressure Vessel. Also, the neutron source spectrum (spectra) shall be determined and the average number(s) of neutrons produced per fission, ν, shall be selected. All these parameters are to be chosen with regards to the calculated data: representative of irradiation conditions (in case of comparisons to measurements), or envelope (in case of design phase for internals and/or vessel analyses).

4.2.2 Discrete ordinates (SN) method

In order to ensure an accurate representation of three-dimensional effects, three-dimensional discrete ordinates transport calculations should be used when practical. When three-dimensional calculations are not practical, a synthesis method may be used to determine the three-dimensional flux or fluence distribution. In this approach, the fluence distribution is determined by synthesizing the results of one- and two-dimensional discrete ordinates solutions (see References [4] and [29]). The results depend on the specific locations where the neutron flux/fluence has to be determined (location of interest), i.e., not only at the core mid-plane, in general. Note that the use of synthesis technique may lead into inaccurate results if the material and/or source distributions are highly three dimensional.

4.2.3 Monte Carlo transport method

In addition to the considerations a) to d) in 4.2.1, the Monte Carlo model construction could require a technique to reduce the solution variance. The geometric model used in the Monte Carlo analyses should reflect the actual physical configuration. The great flexibility in typical Monte Carlo codes allows a very detailed representation, and this should be used to represent all the important features of the geometry under consideration. Typically, Monte Carlo codes allow use of either multigroup or continuous-energy cross sections. The continuous-energy cross sections are recommended. Variance-reduction techniques that have been validated for these applications may be used to reduce the variance in the Monte Carlo calculation (some of them are presented in the References [3] and [5]). Techniques that may be used to improve the statistics at locations far from the core include the following, provided that preliminary checking has been done:

a) source biasing;

b) geometry splitting with Russian roulette;

c) increasing importances;

d) surface restarts;

e) weight windows.

4.2.4 Adjoint fluence calculations

Adjoint calculations may be performed:

— as upstream calculations, to estimate the space- and energy- dependent importance of the core neutrons to a specific location (on the vessel or on the considered internal), in order to determine the source biasing in the direct mode calculation;

— or else, to replace multiple transport calculations in direct mode:

Because the reactor conditions are generally dependent on the fuel cycle, multiple transport calculations are required to track the fluence during plant operation. However, when the operating conditions that affect the transport calculation (e.g. downcomer and core bypass coolant densities, core mechanical design) remain the same, the multiple transport calculations may be replaced by a single adjoint calculation[6].

The adjoint is calculated for an adjoint source located at the vessel or other location of interest that is taken to be proportional to the energy-dependent response cross section. Typically, in the case of flux and/or fluence (E > 1 MeV), the source is taken to be unity above 1,0 MeV and zero below 1,0 MeV. When a dosimeter reaction rate is required, the source typically is taken to be equal to an energy-dependent dosimeter cross section. The fluence (or reaction rate response) at the location of interest is then determined for each cycle by integrating the cycle-specific core neutron source over the calculated adjoint function.

If Monte-Carlo method is used, and if adjoint mode is not available in the code, there may exist options in direct mode that identify the originated sources (spatially and in energy).

4.3 Validation of neutron fluence calculational values

Prior to performing transport calculations for a particular facility, the calculational methodology shall be validated by

a) comparing results with benchmarked calculations and measurements, and

b) demonstrating that it accurately determines appropriate benchmark results.

4.4 Determination of calculational uncertainties

Calculational uncertainties associated with the methodology for predicting neutron fluence typically include the following:

a) nuclear data (e.g. transport cross sections, dosimeter reaction cross sections, and fission spectra);

b) geometry (e.g. locations of internals and deviations from the nominal dimensions);

c) isotopic composition of material (e.g. density and composition of coolant water, vessel internals, the core barrel, thermal shielding, the pressure vessel with cladding, and concrete shielding);

d) neutron sources (e.g. space and energy distribution depending on fuel burnup);

e) methods error (e.g. mesh density, angular expansion, convergence criteria, macroscopic group cross sections, fluence perturbation by surveillance capsules, spatial synthesis, and cavity streaming).

These uncertainties should be evaluated before and/or when performing transport calculations for a particular facility.

5 Reactor pressure vessel neutron dosimetry measurements

5.1 Introduction

Accurate neutron dosimetry provides reasonable assurance that predictions of the reactor vessel neutron fluence at any critical location are accurate and reliable. In this regard, ratios of the calculated to the measured dosimeter response are determined for each dosimeter. The measured to calculated (M/C) ratios are then used to assess the existence of any techniques of convergence acceleration operative within the calculational process.

5.2 General requirements for reactor vessel neutron metrology

Specific procedures identified in applicable standards on neutron metrology published by ASTM International should be followed (see References [7] to [20]). The general requirements for neutron monitors used for reactor pressure vessel dosimetry are outlined below, as are several specific requirements unique to stable-product neutron dosimeters:

a) Types of activation dosimeters:

The recommended set of activation and fissile dosimeters covering the spectral energy range from ~0,08 MeV to 10,0 MeV includes ^{237}Np, ^{238}U, ^{58}Ni, ^{54}Fe, ^{46}Ti, ^{63}Cu, and possibly ^{93}Nb. Additional ^{59}Co dosimeters enable to determinate the thermal contribution of the response in fast dosimeters, especially fission due to ^{235}U present as traces in ^{238}U dosimeters. Cobalt is generally diluted with aluminium in order to reduce the overall activity of the dosimeter.

b) Nuclear and material properties of dosimeters:

The physicochemical properties shall be compatible with the prevailing service conditions; for example, the dosimeter should not melt and should be chemically stable and corrosion resistant. Basic nuclear properties to be considered when implementing fissionable dosimeters include activation product half-life, reaction cross-section, gamma-ray yield, and fission yield.

c) Dosimeter mass and isotopic composition:

Dosimeters shall be of high isotopic purity and sufficient mass for adequate activation. The impact of impurities should be evaluated.

d) Dosimeter geometry and configuration:

In general, dosimeters are in the form of thin activation foils, although other shapes are available. The foil thickness is an important consideration for self-shielding during irradiation and photon absorption or fission-product loss from recoil during counting.

e) Spectral coverage:

Neutron dosimeters should possess adequate spectral coverage. In particular, the dosimeter should enable separate benchmarking calculations of the neutron fluence in the relevant energy ranges: <0,625eV, >0,1 MeV, and >1,0 MeV.

f) Selection of alternative combinations of dosimeters:

ASTM E844-14[14] and ASTM E1005-16[15] provide guidance on composing an appropriate dosimetry package.

g) Irradiation geometry and dosimeter location:

Dosimeters should be placed in locations demonstrated to be representative of the location of interest. The dosimeter location should be determined accurately and recorded. Structures and materials surrounding a dosimeter that can influence its response should be avoided when possible. When these structures or materials are present, their effect should be assessed and included within the overall fluence determination.

h) Dosimeter encapsulation:

Neutron dosimeters are often placed within some form of encapsulating neutron filters or within the in-vessel surveillance capsule. The filter and capsule design should minimize perturbations to the neutron flux and spectrum. Such perturbations should be assessed and included within the overall fluence determination.

i) Irradiation parameters:

Exposure time, the associated power history, and the effects of dosimeter burnout should be accurately determined.

j) Dosimeter analysis:

Radio assay of active species is most commonly done by direct nuclear counting with a high-resolution gamma-ray spectrometer (usually Li-drifted Ge detectors). When conditions preclude direct counting, one can employ radiochemical dissolution (e.g. for Nb dosimeter). In either case, a complete description of the gamma-ray spectrometer and the counting techniques employed should be included as part of the dosimetry documentation.

k) Spectrum unfolding:

From measured reaction rates given by different foils, the more representative spectrum of the whole irradiation should be calculated using suitable unfolding codes.

5.3 Stable-product neutron dosimeters

In addition to radiometric dosimeters, stable product neutron dosimeters also are used for reactor fluence determinations. These devices include solid-state track recorders (SSTRs) and helium accumulation fluence monitors (HAFMs). These devices provide a permanent measurement record because of their stable responses. The provisions of ASTM standards ASTM E854-03[17] for SSTRs and ASTM E910-07[18] for HAFMs should be observed.

5.4 Dosimeter response parameters

As discussed in Clause 4, dosimeter response should be calculated and compared to the measured values described in this section. The M/C ratio can be used to validate the calculational methodology (see Clause 6).

5.5 Uncertainty estimates and measurement validation in standard neutron fields

In order to affect a meaningful comparison between measured results and the corresponding calculated quantities, the uncertainty and bias associated with the measurement process shall be carefully evaluated. Sources of uncertainty include the following: dosimeter physical parameters, irradiation characteristics (e.g. reactor power history and decay times), nuclear data (e.g. decay constants, fission yields, nuclear cross sections, and photon attenuation coefficients), and the nuclear counting process.

Additional uncertainty sources may be present, and their presence should be investigated on a case-by-case basis.

Because dosimetry measurements are used to validate the calculational methodology, the measurement process shall be validated by performing dosimetry measurements with dosimeters that are identical to those exposed to certified fluences in standard neutron fields. Aspects of measurement validation in standard neutron fields are discussed in ASTM E2006-16[19]. After validation in the standard neutron field, the measurement uncertainties and bias are typically insignificant compared to other uncertainties in the fluence determination.

6 Comparison of calculations with measurements

6.1 Introduction

If the measurement data are of sufficient quality and quantity to allow a reliable estimate of the calculational bias and the uncertainty is within the acceptable limits (i.e. they represent a statistically significant measurement database), the comparisons to measurements may be used to modify the calculation to account for bias by applying a correction, by adjusting the model, or both. Several methods of comparison may be used to validate the calculated results. When applying these methods, it is commonly assumed in these comparisons that the uncertainties associated with modelling, such as the spatial location of the detectors within the reactor vessel, are negligible. However, when this is not the case, the effect of these uncertainties on the comparisons should be addressed.

6.2 Direct comparison of calculated activities with measured sensor activities

One method of comparison is to directly compare the calculated dosimeter-specific activities at the end of irradiation with the corresponding measured dosimeter activities. This method enables various segments of the irradiation to be summed to get the total activity. The disadvantage is that experimental results from different irradiations cannot be directly compared without the introduction of transport theory calculations. An overall comparison of calculated and measured activities can be made by using a suitable weighted average of the M/C ratios. The weighting of individual sensor comparisons should include the uncertainties associated with measured activities as well as the energy spectrum coverage provided by each sensor.

6.3 Comparison of calculated rates with measured average full-power reaction rates

The second method of comparison is to derive the average full-power reaction rate for each sensor using the irradiation history of the dosimeter set. After a sufficient operating duration, these reaction rates are independent of both the length of the irradiation and the time at less than full-power operation. The advantage of this approach is that the reaction rate comparisons permit direct comparisons of measured results from different reactors and different cycles of irradiation within the same reactor. Further, comparisons of measured spectral indices (ratios of reaction rates from different sensors) provide comparisons of the energy spectra at different measurement locations. As discussed in 6.2, an overall M/C comparison can be made using a suitably weighted average of the reaction rate data.

6.4 Comparison of the calculations against measurements using least-squares methods

Another method of comparison to obtain a suitable weighting of the uncertainties in the measurements and calculations as well as the spectral coverage of the individual sensors is to apply least-squares adjustment procedures. Least-squares methods provide the capability of combining the measurement data with the neutron transport calculations resulting in an adjusted neutron energy spectrum with associated uncertainties.

7 Determination of the best-estimate fluence

The computed value of neutron exposure, produced with the guidance in Clause 4 to calculate the best-estimate values of fluence, is considered acceptable for safety analysis provided that both of the following are true:

a) The calculation has been validated as described in Clause 6;

b) The validation was based on a qualified database from measurements performed as described in Clause 5.

8 Calculational methods for dpa and gas production

Neutrons from fission events cover a wide range of energies up. These neutrons when interacting with elements along their transport paths could cause atom displacements directly or indirectly, creating displacement cascades, areas of low atomic density (high vacancy concentration), and of high atomic density (interstitial atoms), resulting in a change in the microstructure of the material which is commonly referred to as material damage.

8.1 Displacements per atom (dpa)

Although the notion of displacements per atom of material from their normal lattice sites (dpa) is not a physical quantity and cannot be measured (there is not a simple correspondence between dpa and a particular change in a material property), dpa may be one of the exposure parameters of interest for users to evaluate embrittlement (see 4.1.1).

Hence, an appropriate damage exposure index is the number of times, on the average, that an atom has been displaced during an irradiation. This can be expressed as the total number of displaced atoms per atom of the material. The number of dpa associated with a particular irradiation depends on the primary knock-on atom (PKA) spectra in the material by the neutrons, and hence, depends on the material itself and on the neutron spectrum.

One can consider 3 major different metric for dpa estimation; the original Kinchin-Pease[21], the frequently used Norgett Robinson and Torrens[22] and the more recent and sophisticated Athermal Recombined Corrected[23]. All of them are the effective estimation of Frenkel pairs in materials by model equivalence (accurate molecular dynamics for the latter). Drawbacks of such metric is that they can only be used for crystalline materials with a nominal factor applied to address some degree of cascade efficiency. The calculations apply to the number of displaced atoms at the earliest stage of the cascade production rather than the number of displaced atoms that take part in radiation damage. The calculation of the number of displaced atoms that are effective in altering a material's properties involves further study using molecular and cluster dynamics. In some cases diffusional rate-theory can be used to calibrate the net damage based on some measured property changes (swelling or radiation-induced element segregation for instance). Users shall indicate which metric and associated parameters (displacement energy for instance) they account for dpa calculations.

Damage cross-sections to be folded by neutron spectrum are often performed by a processing code in which the electronic screening is described by a partition function (Lindhard for instance in the NJOY/HEATR processing code[24]). Users shall indicate which processing code (or partition function) and which nuclear data library is used for the calculation of such damage isotopic cross sections.

Authors of this document make users aware with the use of reduced dpa cross section in order to account for self-shielding corrections and that dpa response functions can have limitations/biases and associated uncertainties[25]. The calculation method of dpa from now shall be reviewed time by time at the academic progress.

8.2 Gas production

Lattice defects are generally dominated by the interaction of fast (high energy) neutrons because they create atoms with high recoil energy within the interacting material; this is referred as direct radiation damage. However indirect radiation damage due to recoil from (n,2n), (n,γ), (n,p), and (n,α) reactions, for example, can be a significant fraction of the total number of displaced atoms. Of these reactions the (n,α) is perhaps the most important in creating high energy recoils (due to the mass and energy of the emitted α-particle).

Whereas most common engineering alloys used in nuclear reactors (steels and Ni-alloys containing Fe, Cr and Ni) are prone to damage production from the various recoil reactions, especially at high neutron energies (>5 MeV), these recoil reactions can occur over a very wide range of neutron energies in Ni-containing alloys in particular. Ni is a special case, where thermal neutrons become very effective contributors to radiation damage production through a two-stage process involving transmutation of the most common Ni isotope. The main isotope in Ni, ^{58}Ni, has a high thermal neutron capture cross-section creating ^{59}Ni. The ^{59}Ni, in turn has very high (n,γ), (n,p), and (n,α) reaction cross-sections over a very large range of neutron energies (especially at thermal neutron energies). In addition to the enhanced displaced atom production, helium generated from thermal (n,α) reactions is an important contributor to materials degradation. These reactions need to be considered, when a component is subjected to high thermal neutron fluence with significant percentage of nickel in the alloy. To this end standardized codes and procedures have been developed that allow a user to calculate dpa due to direct atomic displacement by neutrons and from various reactions of the type described here (see for instance References [26],[27] and[28]).

Materials that are made of high nickel content under significant thermal neutron fluence can experience a considerable amount of atom displacements causing premature stress relaxation. Stress relaxation is undesirable for components such as bolts and tie-rods but can also be beneficial by reducing internal stresses, in welds for example, that could alleviate irradiation-assisted stress corrosion cracking. However, the biggest impact of the ^{59}Ni is swelling and embrittlement due to He production in areas of the reactor with high thermal neutron fluxes that can have serious consequences for components that operate for long periods in the reactor core region.

To have a proper assessment of the material damage properties inside the reactor vessel, an account of the helium production is required, which is strongly dependent on the thermal neutron fluence at the location of the component.

Other (n,α) reactions may also be considered, such as the (n,α) production in the boron of control rods. And other effects caused by the gas generation reaction may also be considered, such as steel swelling caused by Helium at high temperatures.

