Standard for the Determination of Steady-State Neutron Reaction-Rate Distributions and Reactivity of Nuclear Power Reactors

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Abstract

American National Standard ANSI/ANS*-19.3-2005 [1] covers "The Determination of Steady-State Neutron Reaction-Rate Distributions and Reactivity of Nuclear Power Reactors". The 2005 version is a new revision of this Standard, which had previously been issued in 1995.

In this revision, the sections on the various types of power reactors have been updated to cover the latest methodologies of calculation in current use, and a section on HWR [CANDU®[1]] reactors has been added. Also, the sections on verification and validation were revised to more fully define, discuss, and distinguish between these topics, and describe actions related to them.

KEYWORDS: Standards, Reaction Rates, Reactivity, Power Reactors

1. Introduction

American National Standard ANSI/ANS-19.3-2005 [1] covers "The Determination of Steady-State Neutron Reaction-Rate Distributions and Reactivity of Nuclear Power Reactors". The 2005 version is a new revision of this Standard, which had previously been issued in 1995. The 2005 revision was produced by a subcommittee of the ANS Reactor Physics Standards Committee (ANS-19). The subcommittee responsible for ANS-19.3 was 10-member strong (see Acknowledgements) and had broad industry representation. The main objective of the 2005 revision was to ensure that the Standard would reflect advances in the technology and current practice, and that it would cover all the power-reactor types currently operating.

2. Scope

The design and operation of nuclear reactors require knowledge of the conditions under which a reactor will be critical, as well as the degree of subcriticality or supercriticality when these conditions change. In addition, knowledge is required of the spatial distribution of neutron reaction rates in reactor components as a prerequisite, for example, for inferring temperature distributions. Reaction-rate spatial distributions and reactivity can be calculated by various techniques. As nuclear cross sections have become more accurate and as computational methods have been refined, the tendency has been to rely more heavily on calculations. Available reactor experimental data have been used to validate the calculations.

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* ANSI = American National Standards Institute; ANS = American Nuclear Society
[1] CANDU® is a registered trademark of AECL.

American National Standard ANSI/ANS-19.3 is meant to provide guidance for performing and validating the sequence of calculations for nuclear-power reactors leading to the prediction of steady-state neutron reaction-rate spatial distributions, reactivity, and long-term changes in isotopic composition due to fuel depletion. The standard provides guidance for the selection of computational methods, criteria for verification and validation of computational methods used by reactor core analysts, criteria for evaluation of accuracy and range of applicability of data and methods, and requirements for documentation.

Since the Standard covers power reactors of types PWR, BWR, LMR, HTGR, and HWR[CANDU], it is, of necessity, of a general nature, but at the same time it reflects the diverse physics computational procedures employed in the various types of reactor. The areas of major emphasis in the Standard are the computational methods, the verification of calculations, the validation of methods, and the documentation of computations performed. It is not the Standard's purpose to require or even recommend any particular methodology or computer program. However, the Standard does require that any method used be adequately verified, validated, and tested, and that it be demonstrated to yield results within acceptable errors.

3. Relation to Other Standards

This Standard is related to other ANS reactor-physics Standards, e.g.:

- "Nuclear Data Sets for Reactor Design Calculations", ANSI/ANS-19.1-1983 (R2002) [2], which defines the criteria to be employed in the preparation of application independent cross-section data files from experimental data and theoretical models.
- "Guide for Acquisition and Documentation of Reference Power Reactor Physics Measurements for Nuclear Analysis Verification", ANSI/ANS-19.4-1983 (R1989) [3]; and Requirements for Reference Reactor Physics Measurements, ANSI/ANS-19.5-1995 [4], which contain criteria for performing and documenting experiments for the validation of computational systems by comparison with the experimental results.
- "Determination of Thermal Energy Deposition Rates in Nuclear Reactors", ANSI/ANS-19.3.4-1983 (R2002) [5], which provides criteria for the establishment of the thermal-energy deposition-rate distribution within a nuclear-reactor core. Since the accuracy with which this can be done is dominated by the accuracy with which neutron reaction rates can be calculated, ANSI/ANS-19.3.4 is closely related to ANSI-ANS-19.3.
- "Quality Assurance Program Requirements for Nuclear Facility Applications", ANSI/ASME-NQA-2-1986 (R1989) [6], which includes additional documentation requirements.
- "Guidelines for the Documentation of Computer Software", ANSI/ANS-10.3-1995 [7].

4. Computational Methods

Calculations of system reactivity and reaction-rate distributions within nuclear-power reactors typically consist of a sequence of steps. The following is a typical sequence. However, it must be noted that not all sequences need to include all the steps.

1. Calculation of an application-dependent neutron spectrum, using averaged-data-set cross sections, isotopic number densities, and geometrical information (an averaged data set is prepared by averaging a basic nuclear data set with a specified weighting function over a specified energy-group structure)
2. Collapsing of cross sections to few-group form, using this application-dependent spectrum
3. Homogenization of cross sections and number densities over lattice cells, pin cells or assemblies

4. Calculation of reactivity and of the few-group flux spatial distribution in the reactor
5. Computation of reaction rates in physical reactor components
6. Calculation of changes in the isotopic composition of the fuel, and possibly of other reactor components, with increasing exposure.

It is not a requirement of the standard that a particular sequence of calculations be used, and the sequence does not in itself demonstrate compliance with the standard. The use of a specific computational procedure shall be justified by a procedure of verification and validation (see further below). However, the preceding sequence does provide an adequate framework within which most of the problems in steady-state reactor-physics calculations can be discussed.

Careful consideration must be given to all conditions that may significantly affect results. Such conditions are, for instance, the presence of control elements, burnable or soluble absorbers, dissimilarity in fuel assemblies, composition and geometric layout of fuel in an assembly, dependence of coolant or moderator density upon conditions, and structural material. Also, the non-uniform spatial distribution of saturating fission products (xenon, samarium, etc.), actinides, or other nuclides, depletion-dependent conditions (such as previous power history, coolant history, control history, history of fuel assemblies), fuel temperature (Doppler feedback), and other important spatial distributions, should be considered.

The Standard covers and discusses each of the computational steps listed. It also goes into further detail to describe practices that are in common use in applications for the different reactor types.

4.1 Basic Data

The primary sources of basic nuclear data that are used for the generation of multigroup constants are evaluated data sets. The properties and criteria for selecting these sources of basic nuclear data are specified in [2].

4.2 Preparation of Multigroup Constants

When preparing multigroup constants directly from evaluated data sets, the procedures for the preparation of averaged data sets described in [2] shall be followed. The multigroup constants can be sensitive to the selection of an energy-dependent weighting spectrum, and to the choice of group structure. An estimate of the reactor spectrum is needed and should be obtained from measurements in identical or similar reactors, or from analytical models of neutron slowing down or source spectra. Results may be sensitive to the modeling of the spectra. The preparation of application-dependent multigroup constants shall entail use of an application-dependent energy-spectrum estimate. A weighting spectrum is selected to preserve important system-dependent characteristics during the averaging process. These characteristics usually include reaction rates, and may include other quantities.

4.3 System-Dependent-Spectrum Calculations

The multigroup cross-section set should be used in the calculation of the neutron energy spectra in the system under investigation. The energy spectra are established by the geometry, material composition, and the operating conditions of the reactor and are a function of the interplay between neutron leakage, absorption and scattering. The neutron energy spectrum may vary from one region of the core to another and it may be necessary to compute the spectra for several representative regions of the reactor core.

Many reactor cores can be thought of as composed of repeating units called cells, such as a single fuel pin cell or a fuel assembly cell (e.g., sometimes this formalism is extended to absorber pins or water holes), with its associated structures, coolant, and moderator (where this is distinct from the coolant). It is necessary to inspect the cell and its surroundings to determine if the spectrum in the cell is generated by the cell and its

similar surroundings alone, or if the spectrum in the cell is influenced by parts of the reactor not made of similar cells. When the spectrum is influenced by regions of the core external to the cell, a supercell may be defined, and the spectrum characteristic of the supercell is computed. The supercell may be a repeating unit of the reactor containing non-cell materials such as water channels, control rods, and structural materials. Other non-cell regions such as absorber pins, when present, should be included in the supercell if they significantly influence the spectrum. For either a cell or a supercell, outer boundary conditions are specified consistent with symmetry assumptions.

If the spectrum in the cell is influenced by external regions, these effects should be included in the spectrum calculations. These effects may be caused by leakage across the cell or supercell boundaries, and thus may be energy and direction dependent. Temperature effects in fuel, in moderator or coolant, or in both (e.g., Doppler broadening), and variations in density of coolant or moderator, or both, shall be included in the calculation. Corrections for a non-uniform temperature distribution should be made, or the temperature distribution should be included in the calculation.

Various numerical procedures may be utilized to describe neutron transport phenomena. Different degrees of approximation may be made depending on the nature of the problem and the objectives of the calculation. A very detailed type of calculation is continuous-energy Monte Carlo. This technique has served primarily as a guide to the validity of other procedures but is now in mainstream application as well. Other transport models generate numerical solutions of the integral transport equation (sometimes referred to as collision probability methods) or of the integrodifferential transport equation. Few-group diffusion-theory solution of the neutron diffusion equation is the next level of approximation. The analyst shall demonstrate that the model used is appropriate to the problem under consideration, for example that the spatial mesh, the order of scattering (P1, P3, etc.), and the order of quadrature (in S_n methods) are adequate to achieve stated accuracy levels for the calculated reactivity and reaction rates.

4.4 Collapsing to Few Groups

When performing full reactor calculations it is usually adequate and desirable to collapse the cross sections from the multigroup structure into a few groups. The actual group structure chosen should depend on the type of calculation and the sensitivity of that calculation to the group structure. When collapsing cross sections to a few groups, important system characteristics - such as reaction rates in a unit cell, reactivity of the cell and core, or reaction-rate ratios - should be preserved. This preservation is an attempt to maintain some of the detailed representation of the multigroup calculation in the coarser few-group calculation. The calculation used in the collapse shall include or account for all important effects of space and energy that cannot be adequately modeled in the calculations to follow, such as self-shielding and spectrum dependence on surrounding materials. The cross sections of each nuclide present to a significant degree shall be retained individually whenever calculations of individual reaction rates are to be carried out. These cross sections should also be the starting points for calculations of change in isotopic composition with time (depletion calculations).

4.5 Calculation of Reactivity and Neutron-Flux Distribution

A number of models are in common use for performing neutron flux calculations. Some of these are

(1) Solving the diffusion equations by finite-difference or synthesis methods,
(2) Solving the transport equations by discrete-ordinates or collision-probability methods or by the method of characteristics, and
(3) Solving the reactor neutron balance equations by nodal methods.

Usually the model used to describe neutron transport in the reactor calculation is an approximation to a more accurate model. For example, diffusion theory is an approximation to transport theory. Thus, there

will be some inherent uncertainty or error because of the model. In the implementation of a model via a computer program, it is common for additional approximations to be made. Thus, the solution produced by the computer program will be an approximation to the solution of the model equations.

A number of modeling assumptions or approximations commonly made may contribute to a computational bias and/or to uncertainties, for example only, assumptions of symmetry for configurations that are not precisely symmetric, the choice of a small number of energy groups to represent the neutron energy variation in the core, the use of bucklings to simulate leakage effects in the directions not explicitly represented, the assumption that dissimilar media may be homogenized, etc.

These assumptions or approximations are in principle amenable to numerical studies aimed at establishing the deviation of their results from more precise solutions. Numerical methods should normally be used only within the range of parameters for which the biases or uncertainties of the methods are known. When stepping outside this range, caution should be exercised to try to ensure that there are no fortuitously canceling errors, and to be in a better position to judge the reasonableness of the behavior of the method and of the results obtained.

4.6 Calculation of Reaction Rates

When a model that simplifies the physical description is used, means shall be provided to convert the results of the model calculation into reaction rates in the physical components, as required by the application. The procedure shall specify how local reaction rates are obtained, and the basis or justification for the technique employed shall be described.

The reaction rates calculated are used for a variety of purposes, for example,

(1) Computation of heat-generation rates for heat transfer and thermalhydraulic calculations,
(2) Computation of change in isotopic composition of fissile materials as a function of position in the core, in order to predict fuel inventory,
(3) Computation of shutdown margins, control-rod worths, and reactivity worths of other components, and
(4) Computation of the relationship between detector response and in-core reaction rates.

4.7 Depletion Calculations

In a critical reactor, the rate of change in the concentration of a nuclide is the difference between the rate of production of that nuclide and the rate of its destruction (or loss). The most significant production mechanisms are neutron capture in the next lower isotope, the decay of another nuclide, and direct fission yield (in the case of fission products). The most significant destruction mechanisms are fission, neutron capture, and decay of the nuclide. Yields of fission products and decay constants of the nuclides of interest are basic nuclear data that must be available to any such calculation.

The product of the cross sections and fluxes give fission and capture reaction rates. These, together with the fission yields and decay constants, provide the production and loss terms for each nuclide. The result is a set of coupled differential equations for the concentration of all nuclides of interest. These equations are solved simultaneously by numerical methods, through the use of discrete time steps. The time steps shall be set sufficiently small to ensure numerical stability of the solution technique, and accuracy appropriate to the application. It is also necessary to ensure that the flux level in depleting regions does not rise so rapidly that the required time-step length becomes exceedingly short.

In order to lengthen the permissible time step, it is common practice to assume that some nuclides with very large decay constants are in equilibrium at all time steps. The choice of these nuclides may be under the control of the analyst. If so, the choice should be made consistent with the intended applicability of the results.

Fission and capture reaction rates near the beginning of exposure are normally obtained from the zero exposure cross sections of all nuclides of interest. As the depletion calculation proceeds from zero exposure, concentrations change and group-averaged cross sections may change as a result. Both spectrum changes and changes in resonance self-shielding factors contribute to these cross-section changes. Thus, it may be necessary to recalculate the supercell spectra with updated concentrations at intervals in order to obtain group-averaged cross sections and other parameters such as diffusion coefficients as a function of time, exposure, number density, etc. The procedure for selecting a satisfactory time interval is straightforward: the interval is reduced in successive calculations until the differences in concentrations are insignificant for the intended application.

The few-group cross sections obtained from the supercell calculation may be used to perform a spatial reactor depletion calculation. The cross sections are retained as functional fits to exposure, isotopic concentrations, and/or other variables (such as moderator density and temperature, fuel temperature, and void fraction) or as tabular data with interpolation and extrapolation. This is applicable if the supercell representation and spectrum have adequate accuracy to allow proceeding directly to depletion calculations through the use of macroscopic cross sections. This assumes that the number-density ratios will not change with time as the core depletes with locally and possibly globally varying power level, moderator temperature, void distribution, etc. If this cannot be justified, a more accurate approach shall be used. The supercell calculation is used to generate few-group cross sections as a function of exposure, isotopic concentrations, and/or other variables, which are retained for use in the few-group reactor core calculation. The depletion equations may then be solved as before, or individually for each explicit nuclide. The macroscopic cross sections must include dependence on separately defined history variables, such as power level, moderator temperature and density, fuel temperature, Xe-135 concentration, etc.

5. Verification, Validation

Verification and validation are complementary aspects, which influence the range of applicability of a computational system. Verification assesses the fidelity of the computational system to the theoretical models upon which it is based. For example, it is concerned with whether equations in the theoretical model are correctly coded in the computer program and are solved in a correct manner. On the other hand, validation assesses the accuracy of the computational system's outputs, as determined by comparison against measurements or other real-world data.

Verification may typically involve comparisons with closed-form analytic solutions, or with results from a previously verified method. Verification is typically performed within the organization developing the code, but independently of the code developer(s). The results are usually included within the documentation distributed with the code.

A systematic method of verification might include unit testing, then integral testing of lattice codes, linkage codes, and reactor-core codes. Analytic and/or simplified artificial benchmarks are the primary tool for unit testing within the context of verification. If the code does not replicate the reference result of the benchmark(s) to an acceptably fine level of precision, the coding shall be examined until the reasons for the discrepancies are identified and understood, after which the code shall be revised and the verification process repeated.

From the perspective of the ANS-19.3 Standard, validation usually involves comparisons with actual measurements, such as those from critical experiments or from operating reactors. Typically, the organization developing the code will perform much of the validation, and the results are included within the documentation distributed with the code. Further validation against results from an operating reactor of

interest or a very similar reactor also should be performed before the computational system is employed to model that reactor on a routine basis.

Integral testing for validation purposes usually involves comparisons with measurements from operating reactors. In these comparisons, due consideration must be given to uncertainties or tolerances in enrichment, composition, densities, as-built dimensions, and other engineering state variables. In addition, results from critical experiments or from more rigorous codes may be used to supplement these comparisons when appropriate.

If the computational system is to be used for licensing purposes or for in-core fuel management, comparisons should be made with measurements for the specific reactor of interest. If the reactor has not yet begun operation, such comparisons should be made with measurements from other reactors that are very similar to the one of interest. Furthermore, it is imperative that the model employed during integral testing be exactly the same as that which will be used for routine applications.

There are advantages to using results from critical experiments: the uncertainties associated with such measurements usually are smaller and more readily quantifiable than those from operating reactors, even when the reactors are at zero-power conditions. On the other hand, critical experiments typically are performed at room temperature and have little if any thermalhydraulic feedback. In addition, critical experiments are usually limited to zero-burnup fuel, and they often feature different geometries from those in power reators, for both the fuel design and the core arrangement.

The results obtained from validation, particularly integral validation, can be used to establish biases and uncertainties for the predictions from the computational system over its demonstrated range of applicability. In addition, confidence levels shall be specified for the established uncertainties. If the computational system is applied beyond its demonstrated range of applicability, the assigned uncertainties shall be increased by an amount that reflects the extent of the extrapolation and the possibility that other factors or phenomena can influence the results obtained.

The Standard requires that a written report or reports of the validation of the computational system shall be prepared, which shall

1. Describe the computational system - including methods, cross-section data, and validation procedure - with sufficient detail, clarity, and lack of ambiguity to demonstrate that the validation conforms to this standard.
2. Identify experimental data and results of more accurate calculation methods, if applicable, and state explicitly which parameters are compared.
3. State the area or areas of applicability.
4. State the biases and the uncertainties in the computational system over each area of applicability, together with the means for developing composite error estimates and confidence levels.

The Standard requires the documentation to be sufficiently complete to provide an auditable path. In those instances where the foregoing documentation is proprietary in nature, documentation edited through the exclusion of the proprietary information shall be prepared and areas omitted because of proprietary considerations noted.

6. Summary of Requirements

Compliance with the intent of the Standard for an intended area of applicability of the computational system used can be demonstrated by meeting the following requirements:

1. Selection of models and methods:
   a. Consideration of all conditions of reactor composition, temperature, and configuration that

significantly affect the calculated quantities and justification of the resulting model approximations
   b. Preparation of multigroup constants, if employed, in conformance with the ANSI/ANS-19.1 Standard [2], through the use of an application-dependent energy spectrum calculation
   c. Justification of geometric and neutronic transport approximations utilized in the spectrum calculation
   d. Inclusion of all important space and energy effects in the generation of few-group cross sections (if such cross sections are employed)
   e. Demonstration of capability, as required by the application, to retrieve required neutron reaction rates in the physical reactor components from the computations, and to justify any assumptions that need to be made for this retrieval
   f. Justification of the spectrum calculation interval used in depletion calculations, and justification that the numerical integration time step is sufficiently small to ensure numerical stability and accuracy appropriate to the application.
2. Component and integral verification of the computational system.
3. Validation of the computational system. Establishment of the degree of agreement over a limited area of applicability by correlating experimental results, exact analytical results, or results obtained with a previously validated model with results obtained from the system being validated.
4. Evaluation of accuracy and range of applicability of data and methods by establishment of biases and uncertainties, with confidence levels, for the calculations.
5. Documentation of details of all these procedures.

7. Conclusion

American National Standard ANSI/ANS-19.3 was revised and a new version has been issued in 2005. In this revision, the sections on the various types of power reactors have been updated to cover the latest methodologies of calculation in current use, and a section on HWR [CANDU] reactors has been added. Also, the sections on verification and validation were revised to more fully define, discuss, and distinguish between these topics, and describe actions related to them.

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7) American National Standard Guidelines for the Documentation of Computer Software, ANSI/ANS-10.3-1995 (Ibid).