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Edwin A. Harvego
Richard R. Schultz
Ryan L. Crane

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Idaho National Laboratory

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DEVELOPMENT OF A STANDARD FOR VERIFICATION AND VALIDATION OF SOFTWARE USED TO CALCULATE NUCLEAR SYSTEM THERMAL FLUIDS BEHAVIOR

Edwin A. Harvego
Idaho National Laboratory
Idaho Falls, ID, USA

Richard R. Schultz
Idaho National Laboratory
Idaho Falls, ID, USA

Ryan L. Crane
American Society of Mechanical Engineers
New York, NY, USA

ABSTRACT

With the resurgence of nuclear power and increased interest in advanced nuclear reactors as an option to supply abundant energy without the associated greenhouse gas emissions of the more conventional fossil fuel energy sources, there is a need to establish internationally recognized standards for the verification and validation (V&V) of software used to calculate the thermal-hydraulic behavior of advanced reactor designs for both normal operation and hypothetical accident conditions. To address this need, ASME (American Society of Mechanical Engineers) Standards and Certification has established the V&V 30 Committee, under the jurisdiction of the V&V Standards Committee, to develop a consensus standard for verification and validation of software used for design and analysis of advanced reactor systems. The initial focus of this committee will be on the V&V of system analysis and computational fluid dynamics (CFD) software for nuclear applications. To limit the scope of the effort, the committee will further limit its focus to software to be used in the licensing of High-Temperature Gas-Cooled Reactors. In this framework, the Standard should conform to Nuclear Regulatory Commission (NRC) and other regulatory practices, procedures and methods for licensing of nuclear power plants as embodied in the United States (U.S.) Code of Federal Regulations and other pertinent documents such as Regulatory Guide 1.203, "Transient and Accident Analysis Methods" and NUREG-0800, "NRC Standard Review Plan". In addition, the Standard should be consistent with applicable sections of ASME NQA-1-2008 "Quality Assurance Requirements for Nuclear Facility Applications (QA)". This paper describes the general requirements for the proposed V&V 30 Standard, which includes; (a) applicable NRC and other regulatory requirements for defining the operational and accident domain of a nuclear system that must be considered if the system is to be licensed, (b) the corresponding calculation domain of the software that should encompass the nuclear operational and accident domain to be used to study the system behavior for licensing purposes, (c) the definition of the scaled experimental data set required to provide the basis for validating the software, (d) the ensemble of experimental data sets required to populate the validation matrix for the software in question, and (e) the practices and procedures to be used when applying a validation standard. Although this initial effort will focus on software for licensing of High-Temperature Gas-Cooled Reactors, it is anticipated that the practices and procedures developed for this Standard can eventually be extended to other nuclear and non-nuclear applications.

INTRODUCTION

To address the need for internationally recognized standards for verification and validation of software used in the thermal-hydraulic analyses of advanced nuclear power plants, the V&V 30 Committee has been established to develop an ASME standard for verification and validation of computational fluid dynamics and system analysis software that will be used in the design and analysis of advanced nuclear reactor systems, with an initial focus on High-Temperature Gas-Cooled Reactors. The V&V 30 Committee reports to the V&V Standards Committee, which is under the jurisdiction of the Board on Standardization and Testing, as depicted in the organizational structure shown in Figure 1. This structure is under the responsibility of the Standards and Certification Board of Directors. The title of the V&V 30 Committee is "Verification and Validation in Computational Nuclear System Thermal Fluids Behavior". As defined in the draft charter, the committee … "Provides the practices and procedures for verification and validation of software used to calculate nuclear system thermal fluids behavior. The software includes system

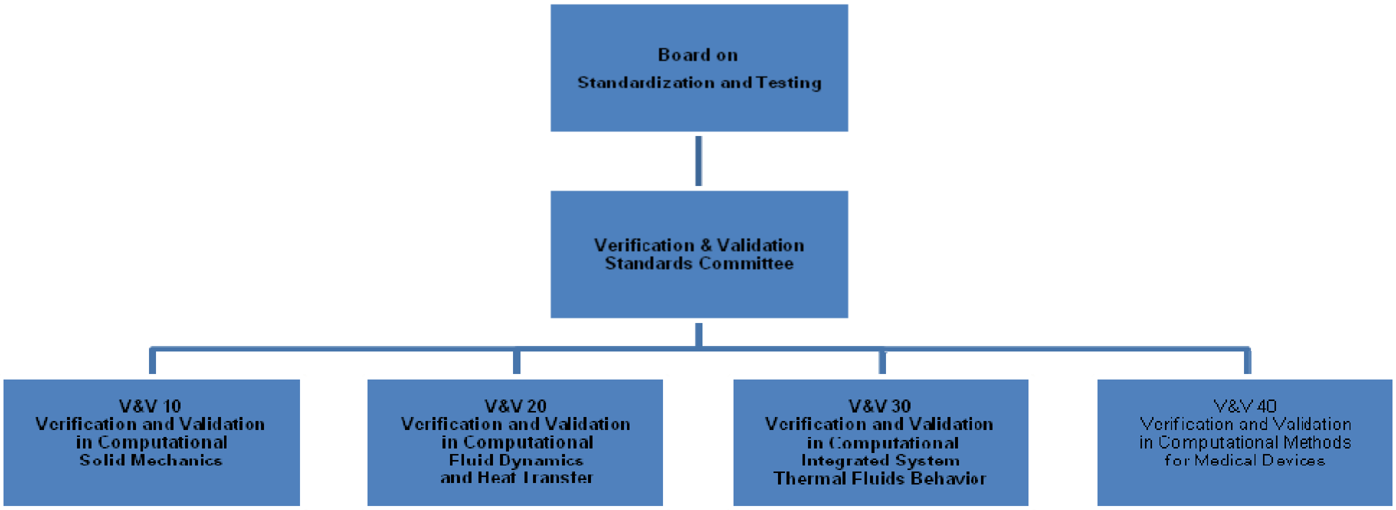

Figure 1. Relationship of Verification and Validation Standards Committee and Subcommittees to Board on Standards and Testing.

analysis and computational fluid dynamics, including the coupling of this software."

The processes and procedures that will be addressed in the new Standard will be used in the design and analysis for licensing of advanced reactor systems. As such, the Standard should conform to Nuclear Regulatory Commission (NRC) practices, procedures and methods for the licensing of nuclear power plants as embodied in the Code of Federal Regulations and other pertinent documents (such as Regulatory Guide 1.203, "Transient and Accident Analysis Methods"[1] and NUREG-0800, "NRC Standard Review Plan"[2]). The V&V 30 Committee will also consider other applicable regulatory requirements for licensing outside the U.S. In addition, the Standard should be consistent with applicable sections of ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" [3]. To ensure consistency between the V&V 30 Standard and applicable NRC and other regulatory practices, procedures and methods, an initial focus of the V&V 30 Committee will be the development of a mapping process that will allow direct comparison of specific requirements of the Standard with corresponding regulatory requirements. This mapping process will provide the basis for defining the scope and detailed content of the Standard, and also aid in defining the relationship between this Standard and other related standards.

ASME V&V 20-2009 "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer" [4] has been recently published. As a result of similarities in the requirements of this Standard, developed and approved by the V&V 20 Committee, and those of the Standard to be developed by the V&V 30 Committee, it is important to define the relationship between the work embodied in the V&V 20 Committee activities versus the work that will be forthcoming in the V&V 30 Standard. As noted in the V&V 20 Standard: "The scope of this Standard is the quantification of the degree of accuracy of simulation of specified validation variables at a specified validation point for cases in which the conditions of the actual experiment are simulated. Consideration of solution accuracy at points within a domain other than the validation points, i.e., a domain of validation, is a matter of engineering judgment specific to each family of problems and is beyond the scope of this Standard." This statement clearly limits the applicability of the V&V 20 Standard to the domain defined by the validation points.

In contrast, the aim of the proposed ASME V&V 30 Standard is to expand the domain of validation to encompass points beyond the range defined by the V&V 20 Standard. In other words, the V&V 30 Standard will complement and make reference to the V&V 20 Standard by defining a methodology for experimental validation of an expanded calculation envelope that encompasses the operational and accident domain of the nuclear system. Therefore, the V&V 30 Standard is expected to address: (a) applicable NRC and other regulatory requirements for defining the operational and accident domain of a nuclear system that must be considered if the system is to be licensed, (b) the corresponding calculation domain of the software that should encompass the nuclear operational and accident domain to be used to study the system behavior for licensing purposes, (c) the definition of the scaled experimental data set required to provide the basis for validating the software, (d) the ensemble of experimental data sets required to populate the validation matrix for the software in question, and (e) the practices and procedures to be used when applying a validation standard to demonstrate that the validated software is capable of performing the needed licensing calculations. Each of the above topics is being considered in the proposed V&V 30 Standard and is discussed in the subsequent sections of this paper.

NOMENCLATURE

CFD:	Computational Fluid Dynamics
EMDAP:	Evaluation Model Development and Application Process
NRC:	Nuclear Regulatory Commission
PIRT:	Phenomena Identification and Ranking Table
V&V:	Verification and Validation

OPERATIONAL AND ACCIDENT DOMAIN

The verification and validation requirements for software intended for the design and analysis of advanced nuclear power systems are determined by the operational and accident envelopes of the reactor plant being considered. Specifically, as depicted in Figure 2, the V&V requirements can only be satisfied if the calculational envelope of the thermal-hydraulic software is demonstrated to either match, or encompass, the system operational and accident envelopes.

To ensure that vendor analyses adequately address the operational and accident domain of concern, the NRC has issued Regulatory Guide 1.203 to describe processes considered acceptable for development and assessment of evaluation models used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. In this context, the evaluation model includes all the numerical models used to calculate the nuclear system behavior, including system analysis and CFD software. The Evaluation Model Development and Assessment Process (EMDAP), as defined in Regulatory Guide 1.203, is shown schematically in Figure 3.

Element 1 in EMDAP (Establish Requirements for Evaluation Model Capability) basically establishes the system envelope as depicted in Figure 2. This step in the process requires the clear definition of all systems, components, phases, geometries, fields, and processes that need to be considered. It includes the complete operational and accident domain and the identification and ranking of phenomena and processes that must be addressed in the analysis process. Establishment of the system operational and accident domain or envelope provides the basis for defining the calculation envelope to be considered in the V&V process as discussed in the next section.

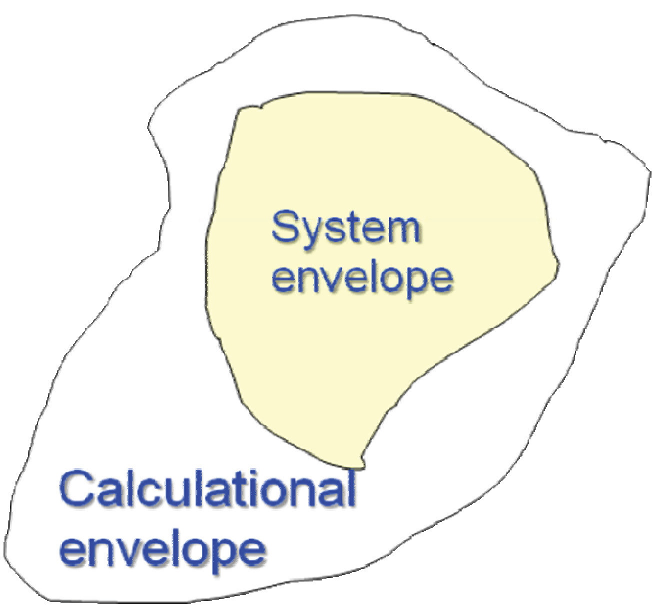

Figure 2. Venn Diagram of System and Calculation Envelope.

SOFTWARE CALCULATIONAL DOMAIN

The range of applicability for a given software package is in part determined by the physics and the models contained within the software. Therefore, the V&V process must confirm the software physics models properly calculate the key phenomena over the entire range of conditions encompassed by the calculation envelope. Successful V&V can only be achieved if an adequate, high-fidelity data matrix and/or exact analytical solution set are available to benchmark the calculation results over the range of conditions that encompass the entire system envelope. Figure 4 shows the process used to ensure that the calculation domain and analysis tools are adequate for their intended purpose.

To define the scope of the required validation matrix,

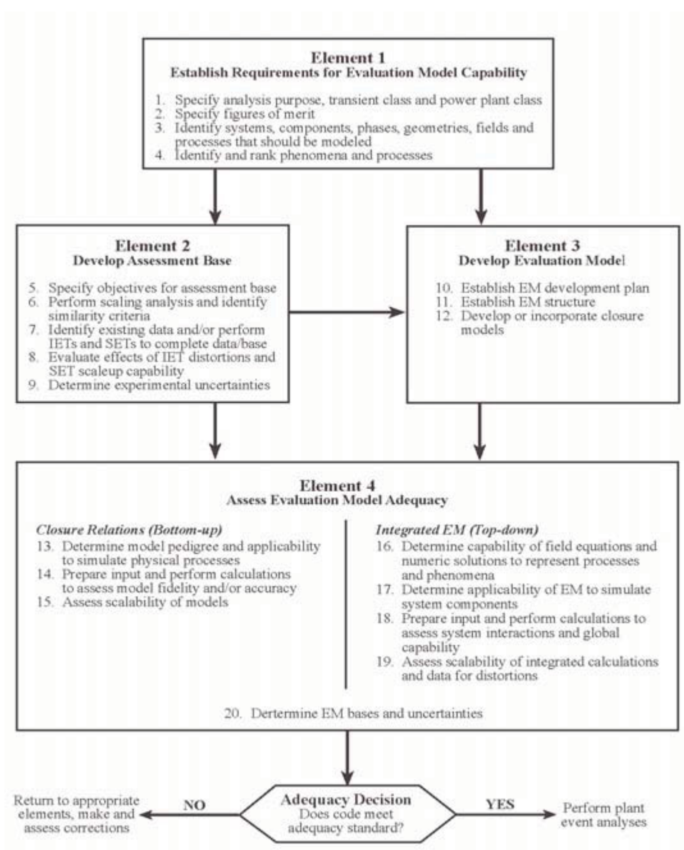

Figure 3. Elements of Evaluation Model Development and Applications Process (from NRC Regulatory Guide 1.203).

operational and accident scenarios that require analysis are first identified. Each transient scenario is then evaluated by a group of experts using the Phenomena Identification and Ranking Table (PIRT) process [5] to identify and rank important phenomena associated with each scenario. Existing analysis tools are then evaluated against validated experimental data to determine whether important phenomena can be calculated. If not, the development of new analysis tools and/or the further development or modification of existing software must then be undertaken. The process is completed when it can be demonstrated that the resulting software predicts the important phenomena over the entire calculation envelope that encompasses all scenarios in the operational and accident domains of interest.

A key component in this process is the validation of experimental data used for the assessment of the software. This component is an element of the validation activities identified in Figure 4. The validation of experimental data will be a major focus of the proposed V&V 30 Standard, and will include processes and procedures for the evaluation of existing data, the design of new experiments, scaling of experimental data to the prototype plant, decomposition of processes to the lowest level that can be modeled and assessed with experimental data, and determination of acceptable experiment uncertainties.

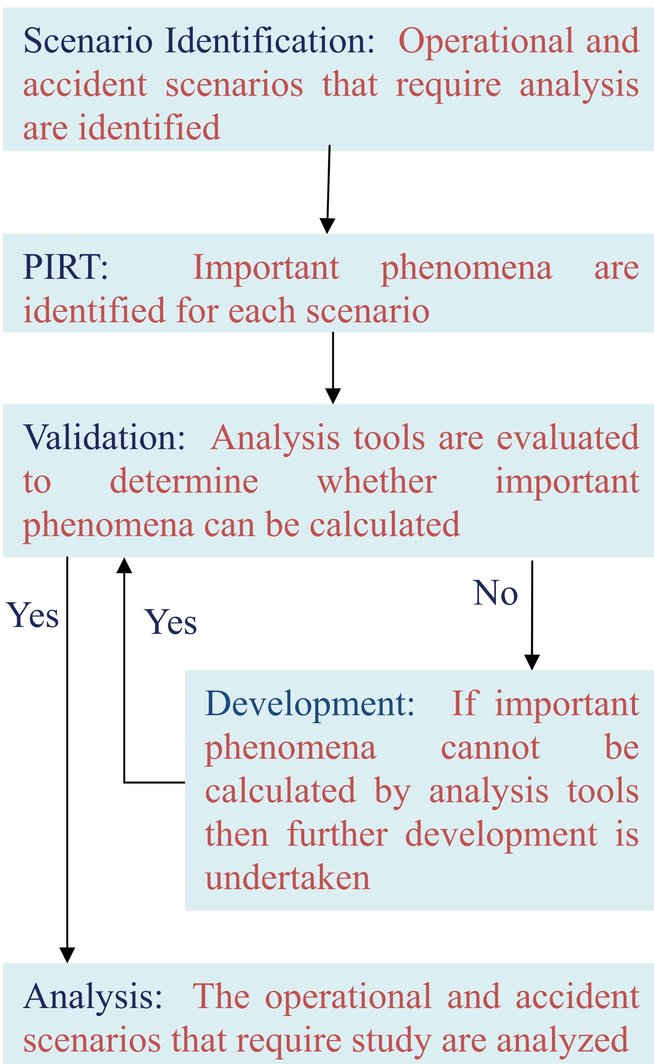

Figure 4. Process for evaluating calculation domain and analysis tools.

While detailed discussion of the validation of experimental data is beyond the scope of this paper, general discussions of the experiment scaling requirements, which are requirements for the experiments used to generate the ensemble of experimental data sets necessary to populate the validation matrix for the software in question, and the practices and procedures to be used when applying the proposed V&V 30 validation Standard are discussed in the following sections.

SCALED EXPERIMENTAL DATA SETS

While in general it is preferred that experiments used for validation of software be performed at full scale to eliminate potential scaling distortions, the time and cost of conducting such experiments is often prohibitive due to the size and complexity of the nuclear power plants. For this reason, methods have been developed for evaluating the existing scaled experimental facilities, or developing new experimental facilities, to produce the experimental data for validating software and models used in the analyses of full-scale prototype plants.

These scaling methods typically include a combination of top-down and bottom-up scaling analyses. The top-down scaling approach is most applicable to the validation of system analysis codes and is usually performed for scaled integral experiments designed to simulate the integral behavior of the complete prototype plant[1]. These scaled integral experimental facilities include simulation of most of the major components in the complete prototype plant, and are usually scaled on a volume basis. On the other hand, the bottom-up scaling approach is most applicable to the validation of CFD software used to evaluate localized phenomena, and typically evaluates phenomena occurring in individual components (e.g. local heat transfer, countercurrent flow in reactor core) to confirm that scaling distortion do not occur at the local level.

In the top-down scaling approach, the system to be addressed is defined, and the transient scenarios within the operational and accident domain are divided into several phases, and sub-phases if necessary, where each phase is defined by the phenomena groups which dominate. The system responses of interest in each phase are represented by the governing conservation equations that determine the behavior of the systems and/or components. These equations are then

[1] Scaled integral behavior is the thermal-fluid behavior of a system that includes all of the key phenomena and physics that are present and that interact with one another.

nondimensionalized and the nondimensional groups are evaluated to determine the relative importance of each group. A set of nondimensionalized coefficients characterizing the system response are then defined. The nondimensional groups from the calculated response of the prototype plant can then be compared with the measured data from the experimental facilities to quantitatively evaluate the scalability of the test facilities to the prototype plant for the parameters and scenarios of interest.

As indicated earlier, the bottom-up scaling approach is primarily concerned with localized phenomena. This approach addresses the details lost in the averaging at the component level in the top-down scaling, thereby providing insights into qualitatively different responses between the test facility and plant. The bottom-up scaling approach is also useful in understanding apparent distortions between the test facility and plant that are observed but not easily explained by the nondimensional groups obtained from the top-down scaling approach.

DATA NEEDS FOR SOFTWARE VALIDATION MATRIX

The data needs for software validation depend on the particular software being considered. For advanced reactors, such as the High-Temperature Gas Reactors (HTGRs), both systems analysis and CFD software will be required. The systems analyses software will be used to analyze the overall behavior of the prototype reactor or experimental facility for selected scenarios in the operational or accident domain, while the CFD software will be used to study more detailed flow behavior and the potential for hot spots and/or unacceptably large thermal gradients in localized areas of the reactor system. Depending on the particular application, the systems analysis and CFD software may be used separately or coupled together to perform the analyses.

The proposed V&V 30 Standard will provide the processes and procedures for determining the data needed to populate the software validation matrices for both systems analysis and CFD software. The software validation processes included in the Standard will address both the evaluation of existing experimental data and the procedures for defining new data needs. The software validation matrices should include both separate effects experiments for evaluating localized phenomena and integral effects experiments for evaluating global system responses. Ideally, the experiment validation matrices will include data from experimental facilities at different scales, so that scaling effects can be evaluated to identify any potential scaling distortions and provide confidence in scaling assessments performed as part of the software validation process.

When new experimental facilities are required, the design of these new facilities must be rigorous to ensure compliance with NRC and other regulatory requirements in the licensing of a new reactor plant, particularly an advanced reactor design with a limited history of regulatory review. Important features to be considered in the experiment design include: (1) assurance that the proposed experiment facility captures key phenomena being investigated; (2) the experiment is scaled to provide a direct link between the scaled facility and prototype plant; (3) adequate high-quality measurements are available to ensure that experimental data uncertainties are quantifiable and acceptably low; and (4) experiment results can be decomposed to the lowest level modeled by the software to ensure that system behavior at the component level is properly being calculated by the governing software physics.

While processes and procedures for qualifying experimental data and populating the software validation matrices will be part of the proposed V&V 30 Standard, it is not intended that the Standard will be prescriptive in nature. Rather, the processes and procedures will define acceptable practices for validating data and developing acceptable software validation matrices that will be consistent with licensing and regulatory requirements. Specific results from the implementation of this Standard will depend on the particular application and the software being used.

APPLICATION OF SOFTWARE VALIDATION STANDARDS

It is intended that the V&V 30 Standard will be used by vendors as the basis for verification and validation of software used for licensing advanced reactor designs. As such, this Standard will be developed to conform to current NRC and other regulatory requirements and guidelines and also provide additional details on acceptable processes that may be used to meet the current regulatory requirements. The initial focus of this Standard is on V&V of systems analysis and CFD software to be used in the design and analyses of HTGRs. However, it is anticipated that this Standard or additional standards will be developed in the future to include other reactor concepts, as well as potential non-nuclear applications.

This Standard is also being developed in conjunction with the V&V 10 Committee, which is responsible for ASME V&V 10-2006 "Guide for Verification and Validation in Computational Solid Mechanics" [6], V&V 20 Committee, which is responsible for ASME V&V 20-2009 "Verification and Validation in Computational Fluid Dynamics and Heat Transfer", and a newly proposed V&V 40 "Verification and Validation in Computational Methods for Medical Devices", which is currently under development. The ASME Standards and Guides are developed in accordance with the "Procedures for ASME Codes & Standards Development Committees", which are accredited by the American National Standards Institute (ANSI) and approved as American National Standards. ASME Standards are defined as technical definitions, instructions, rules, guidelines, or characteristics set forth to provide consistent and comparable results; insure that items are manufactured uniformly and provide for interchangeability; tests and analyses are conducted reliably and minimize the uncertainty of the results; and facilities are designed and constructed for safe operation. As indicated by the titles, each of the V&V Committees discussed in this paper will focus on the specific topic, or application, within its scope (i.e., computational solid mechanics, fluid dynamics and heat transfer, nuclear, and medical devices). In addition, the proposed V&V 30 Standard will be developed in close collaboration with the ASME Standards Committee on Nuclear Quality Assurance, and therefore, will be consistent with applicable sections of ASME NQA-1-2008.

SUMMARY

This paper describes the planned development of a new Standard for verification and validation of systems analysis and CFD software to be used in the design and analysis for the licensing of advanced nuclear power plants in the U.S. and abroad. The initial focus of the proposed Standard will be on advanced High-Temperature Gas Reactors, but it is anticipated that this Standard or additional standards will be developed in the future to include other reactor concepts as well as potential non-nuclear applications.

The organizational structure established by ASME for developing this Standard and other related V&V standards is described, along with the processes that will be employed to ensure consistency among the related standards and the nuclear regulatory environment.

General requirements for the V&V Standard are also described, which include; (a) applicable NRC and other regulatory requirements for defining the operational and accident domain of a nuclear system that must be considered if the system is to be licensed, (b) the corresponding calculation domain of the software that should encompass the nuclear operational and accident domain to be used to study the system behavior for licensing purposes, (c) the definition of the scaled experimental data set required to provide the basis for validating the software, (d) the ensemble of experimental data sets required to populate the validation matrix for the software in question, and (e) the practices and procedures to be used when applying a validation standard.

ACKNOWLEDGMENTS

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COPYRIGHT STATEMENT

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Transient and Accident Analysis Methods", Regulatory Guide 1.203, December 2005.
2. U.S. Nuclear Regulatory Commission, "NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition", NUREG-0800, Rev. 3, June 1996.
3. The American Society of Mechanical Engineers, ASME NQA-1-2008 – Quality Assurance Requirements for Nuclear Facility Applications, March 2008.
4. The American Society of Mechanical Engineers, ASME V&V 20-2009 – Standard for Verification and Validation in Computational Fluids and Heat Transfer", November 2009.
5. G. E. Wilson and B. E. Boyak, "The Role of the PIRT Process in Experiments, Code Development and Code Applications Associated with Reactor Safety Analysis", Nuclear Engineering and Design, 186 (pp. 23-37), 1998.
6. The American Society of Mechanical Engineers, ASME V&V 10-2006 "Guide for Verification and Validation in Computational Solid Mechanics", December 2006.