U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER^{TM} iPWR DESIGN

15.0.2 REVIEW OF TRANSIENT AND ACCIDENT ANALYSIS METHODS

REVIEW RESPONSIBILITIES

Primary - Organization with Responsibility for Review of Transient and Accident Analyses

Secondary - None

I. AREAS OF REVIEW

In order to establish a licensing basis, licensees must analyze transients and accidents in accordance with the requirements of Title 10, *Code of Federal Regulations* (10 CFR)10 CFR 50.34, 10 CFR 50.46, 10 CFR 52.47 and 10 CFR 52.79 and where applicable, per NUREG-0737, "Clarification of TMI Action Plan Requirements." These accidents and transients are described in the Design-Specific Review Standard (DSRS) for mPower^{TM} iPWR, Section 15.0. This section of the DSRS describes the review process and acceptance criteria for analytical models and computer codes used by vendors, combined license (COL) applicants, and licensees to analyze accident and transient behavior. The purpose of the review is to verify that the evaluation model is adequate to simulate the accident under consideration. This includes methods to estimate the uncertainty in the calculation, as in the case of a best estimate loss of coolant accident (LOCA); or to ensure that the results of the analysis are demonstrably conservative, as in the case of a LOCA in Appendix K to 10 CFR Part 50. This review is independent of the type of application submitted (e.g., license amendment, topical report, design certification, or combined license application).

The guidance in this section should be applicable to the transients and accidents described in the DSRS. Guidance to the industry is set forth in Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods." The following sections discuss the areas to be reviewed.

1. Documentation. The development of an evaluation model for use in reactor safety licensing calculations requires a substantial amount of documentation. This documentation includes/covers (a) the evaluation model, (b) the accident scenario identification process, (c) the code assessment, (d) the uncertainty analysis, (e) a theory manual, (f) a user manual, and (g) the quality assurance program.

2. Evaluation Model. An evaluation model is the calculation framework for evaluating the behavior of the reactor coolant system during a postulated accident or transient. It includes one or more computer programs and other information necessary for application of the calculation framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters, and other

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information necessary to specify the calculation procedure. Evaluation models are sometimes referred to as a licensing methodology.

3. Accident Scenario Identification Process. The accident scenario identification process is a structured process used to identify and rank the reactor component and physical phenomena modeling requirements based on (a) their importance to acceptable modeling of the scenario and (b) their impact on the figures of merit for the calculation (e.g., peak cladding temperature and maximum and average cladding oxidation thicknesses). It is also used to identify the key figures of merit or acceptance criteria for the accident.

4. Code Assessment. The code assessment provides a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario.

5. Uncertainty Analysis. Uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. Examples of safety parameters are peak cladding temperature (PCT), cladding oxidation thickness, and departure from nucleate boiling ratio (DNBR).

6. Quality Assurance Plan. The quality assurance plan covers the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the evaluation model. The program also ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

7. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50, Appendix B, which requires the applicant to describe the quality assurance plan used in the design, fabrication, construction, and testing of the structures, systems, and components of the facility.

2. 10 CFR 50, Appendix K, Part II, which requires the applicant to describe the emergency core cooling system (ECCS) evaluation model and computer codes used in ECCS performance analysis in sufficient detail to permit technical review of the analytical approach.

3. 10 CFR 50.34(b) (4), which requires a final analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility.

4. 10 CFR 50.46, which required that an ECCS be provided that must be designed so that its calculated cooling performance following postulated LOCA conforms to the criteria set forth in the regulation. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident.

5. 10 CFR 52.47(a)(2), which requires a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, including the evaluations required to show that safety functions will be accomplished.

6. 10 CFR 52.79(a)(5), which requires a final safety analysis (FSAR) report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components of the facility as a whole. The report must include an analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility.

<u>Documentation</u>

The submittal must identify the specific accident scenarios and plant configurations for which the codes will be used. The evaluation model documentation must be scrutable, complete, unambiguous, accurate, and reasonably self-contained. Consistent nomenclature must be used throughout the entire model documentation. Any referenced material must be readily available from a technical library. Copies of any referenced documents that are not readily obtainable from a technical library or the U.S. Nuclear Regulatory Commision (NRC) Public Document Room, including proprietary reports, must be included with the documentation or provided upon request. The code documentation must be sufficiently detailed that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of Appendix B to 10 CFR Part 50, and the documentation requirement in Appendix K to 10 CFR Part 50. It is desirable that the documentation include the responses to requests for additional information, sorted according to the review issue so that it is easy to follow the entire review history for a single issue. The reviewer can help obtain this goal by issuing requests for additional information organized by review issue. The documentation must include the following components:

A. An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.

B. A complete description of the accident scenario including plant initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident.

C. A complete description of the code assessment comprising a description of each assessment test, why it was chosen, success criteria, diagrams of the test facility that show the location of instrumentation that is used in the assessment, a code model nodalization diagram, and all code options used in the calculation.

D. A determination of the code uncertainty for a sample plant accident calculation. (Appendix K models do not require a determination of the code uncertainty.

E. A theory manual that is a self-contained document and that describes (a) field equations, (b) closure relationships, (c) numerical solution techniques, (d) simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods, (e) pedigree or origin of closure relationships used in the code, and (f) limits of applicability for all models in the code.

F. A user manual that provides (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitations and options that should be avoided for particular accidents, components, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model.

G. A quality assurance plan that describes the procedures and controls under which the code was developed and assessed, and the corrective action procedures that are followed when an error is discovered.

It is not important that the documentation be provided in exactly the format stated above but the information in the review package must be clearly organized in a reasonable manner.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information." The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

1. Evaluation Model. Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from

both qualitative and quantitative points of view. The degree of imprecision that is allowed in the models will ultimately be determined by the amount of uncertainty that can be tolerated in the calculation. Models that cause non-physical predictions to the extent that misinterpretation of the calculated results or trends in the results may occur are not acceptable. For Appendix K LOCA analyses, ECCS evaluation models must meet the specific requirements contained in Appendix K to 10 CFR Part 50.

2. Accident Scenario Identification Process. The purpose of the accident scenario identification process is to identify and rank the reactor component and physical phenomena modeling requirements based on (a) their importance to the modeling of the scenario and (b) their impact on the figures of merit for the calculation. The accident scenario identification process must be a structured process. It must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario. The models that are present in the code and their degree of fidelity in predicting physical phenomena must be consistent with the results of this process. For example, if the accident scenario identification process determines that a certain physical phenomenon is important to the scenario under consideration, the code must have a relatively accurate model for that phenomenon and a detailed assessment of that model must be provided. Phenomena that have lower ranking may be represented by models with larger inherent uncertainty. The formality and complexity of this process should be commensurate with the complexity and importance of the event under consideration.

3. Code Assessment. Assessments of all code models intended to be used in the evaluation model must be provided. All assessments must be performed with the frozen version of the evaluation model that has been submitted for review. Assessments performed with other versions of the evaluation model should be justified on a case by case basis because even “small” changes to the evaluation model can have unintended consequences on calculation results that were thought to not be impacted by the changes.

   Separate effects testing must be performed, if the state of knowledge and existing test data are insufficient, to demonstrate the adequacy of the physical models to predict physical phenomena that were determined to be important by the accident scenario identification process. Separate effects testing must also be used to determine the uncertainty bounds of individual physical models.

   Integral effects testing must be performed, if the state of knowledge and existing test data are insufficient, to demonstrate that the interactions between different physical phenomena and reactor coolant system components and subsystems are identified and predicted correctly.

   Assessments against both separate effects tests and integral effects tests must be performed with the code. All models need to be assessed over the entire range of conditions encountered in the transient or accident scenario. Assessments should also compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models. Code options used in the assessment calculations must be the same as those used in plant accident calculations.

A scaling analysis must be performed that identifies important non-dimensional parameters related to geometry and key phenomena. Scaling distortions and their impact on the code assessment must be identified and evaluated in the assessment. Calculations of actual plant transients or accidents can be considered, but only as confirmatory supporting assessments for the evaluation model. This is because the data available from plant instrumentation is usually not detailed enough to support code assessment of specific models. Plant data can be used for code assessment if it can be demonstrated that the available instrumentation provides measurements of adequate resolution to assess the code. The assessment cases must compare code predictions to all important measured variables in order to show that good predictions of one test variable do not result from compensating errors. Assessments must include a description of all assessment cases, specific models that are being assessed in each case, and acceptance criteria used. Acceptance criteria must be supported by quantitative analysis whenever possible.

ECCS evaluation models must include a specific assessment to meet the criteria in Appendix K to 10 CFR Part 50, if the Appendix K evaluation model is used. Small-break ECCS evaluation models must also meet the assessment requirements of Three Mile Island (TMI) Action Item II.K.3.30, where applicable.

4. Uncertainty Analysis. The uncertainty analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistent with the results of the accident sequence identification process. When the code is used in a licensing calculation, the combined code and application uncertainty must be less than the design margin for the safety parameter of interest. The analysis must include a sample uncertainty evaluation for a typical plant application. In some cases, bounding values are used for input parameters as described in DSRS sections or Regulatory Guides and are used for plant operating conditions such as accident initial conditions, set points, and boundary conditions.

5. Quality Assurance Plan. The code must be maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50.

III. REVIEW PROCEDURES

The review procedures described below are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Programmatic Requirements - In accordance with the guidance in NUREG-0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements. If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant structures, systems, and components (SSCs), but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in

NUREG-0800 “Introduction,” Part 2. Commission regulations and policy mandate programs applicable to SSCs that include:

A. Maintenance Rule Standard Review Plan (SRP) Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guides 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” and RG 1.182; “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants”.

B. Quality Assurance Program SRP Sections 17.3 and 17.5 (DSRS Section 13.4, Table 13.4, Item 16).

C. Technical specifications (DSRS Section 16.0 and SRP Section 16.1) – including brackets value for DC and COL. Brackets are used to identify information or characteristics that are plant specific or are based on preliminary design information.

D. Reliability Assurance Program (SRP Section 17.4).

E. Initial Plant Test Program (Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants,”DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

F. ITAAC (DSRS Chapter 14).

2. For new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) that are identified in the version of NUREG-0933 current on the date six months before application and that are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the TMI requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

3. Acceptance Review. An acceptance review is a preliminary process of assessment of the completeness and apparent acceptability of quality of the documents submitted in order to commit NRC to a detailed review of the application. The reviewers of the code or codes comprising the evaluation model should perform an acceptance review consistent with the guidance provided in Office of Nuclear Regulatory Research (NRR) Office Instruction (OI) No. LIC-101, Revision 1, “License Amendment Review Procedures,” dated March 27, 2002, and LIC-500, Revision 4, “Topical Report Process,” dated December 21, 2009. The reviewers should ensure that all areas discussed in the section titled Documentation are addressed in the topical report submitted by a licensee or vendor. The documentation is not required to have the same section titles or organization as described above, but all of the relevant subject matter must be included. The material must be sufficiently detailed to permit the reviewers to begin a review

without immediately requesting additional documentation in order to understand the licensing submittal.

The results of the acceptance review should be documented in a letter to the organization submitting the evaluation model for review. Submittals that do not contain the required material should also be processed in accordance with NRR OI No. LIC-101, Revision 1, and LIC-500, Revision 4. The reviewers should document what material is missing from the submittal. This information should then be provided in the letter to the submitting organization so that the material can be revised to meet acceptance review requirements for any future submittals. After a successful completion of the acceptance review, a detailed review of the code documentation may proceed.

4. Detailed Review

A. Documentation. The reviewers should review the documentation to determine if (i) all documentation listed in Section II, "Documentation," above has been provided, (ii) the evaluation model overview provides an accurate roadmap of the evaluation model documentation, (iii) all documentation is accurate, complete, and consistent and, (iv) all symbols and nomenclature have been defined and consistently used.

The reviewers should confirm that the theory manual is a self-contained document and that it describes the field equations, closure relationships, numerical solution techniques, and simplifications and approximations (including limitations) inherent in the chosen field equations and numerical methods. The reviewers should also confirm that the theory manual identifies the pedigree or origin of closure relationships used in the code and the limits of applicability for all models in the code.

The reviewers should confirm that the user manual provides guidance for selecting or calculating all input parameters and code options. The guidance must be clear and unambiguous. As a result of the code development or review process many code options may be determined to be inappropriate for specific licensing calculations. The reviewers should confirm that the guidance in the manual specifies the required and acceptable code options for the specific licensing calculations. The reviewers should also confirm that required input settings are hardwired into the input processor so that the code stops with an error message if the required input is not provided or if the input is not within an acceptable range of values or that administrative controls (an independent reviewer QA check) are in place that accomplish the same purpose. The reviewers should confirm that computer codes that are used for multiple accidents and transients include guidelines that are specific to each transient or accident. Code assessment cases can be examined to ensure that the modeling used in the assessment cases is consistent with the user guidelines.

B. Evaluation Model. The reviewers should determine whether the mathematical modeling and computer codes used to analyze the transient or accident have been previously reviewed and accepted. For changes to previously approved models, the reviewers can limit their review to the new material if they determine that there is nothing new that will invalidate the previous approval, including the range of applicability for the analysis method. Otherwise the entire model must be reviewed. For a new model that has not been previously reviewed, the reviewers initiate an evaluation of the entire analytical model.

The reviewers should determine if the physical modeling described in the theory manual and contained in the mathematical models is adequate to calculate the physical phenomena influencing the accident scenario for which the code is used. A scenario will have a set of governing physical phenomena that drive the results of the calculation. The key physical phenomena, including constitutive equations needed for model closure, must be defined for the calculation being performed by the code. Physical phenomena that are important for one accident scenario may not be important for a different accident scenario. The key physical phenomena can also be specific to a particular plant design.

The mathematical equations that comprise an evaluation model can be characterized as being either a field equation or as a constitutive or closure relationship. Field equations are a set of rigorously derived equations that contain no approximations other than the initial assumptions used in deriving the equations. The range of applicability of the field equations is limited only by the validity of the assumptions used in their derivation. An example of a set of field equations is the set of fluid transport equations for mass, momentum, and energy that are derived from macroscopic balances of these quantities. Although these equations are mathematically exact, the equations contain more unknown quantities than there are equations. Some of these unknown quantities are the equation of state, the stress tensor, and the heat flux. In order to be able to solve the field equations the unknown quantities must be expressed in terms of the known quantities from the field equations. The equations that relate the unknown quantities to the known quantities are constitutive or closure relationships. These equations are often models or approximations that are much more restrictive in their range of validity than the set of field equations that they are used with. For example, using Newton's law of cooling as a closure relationship for heat flux from a heat structure to a fluid will limit the application to model situations where radiation heat transfer is not significant even though the field equations are valid. The reviewer must therefore ensure that the field equations of the evaluation model are adequate to describe the set of physical phenomena that occur in the accident and ensure that the closure relationships are valid over the full range of conditions encountered during the accident.

The modeling must be consistent with the results of the accident scenario identification process in that there must be models for all important phenomena in the accident scenario. Components and physical phenomena that are identified as being important in the accident scenario identification process must be modeled with a high degree of fidelity. Phenomena of lower importance may be represented by less accurate models.

The reviewers should determine if the simplifying assumptions and assumptions used in the averaging procedure are valid for the accident scenario under consideration. Simplifying assumptions and averaging are often applied to detailed physical and mathematical modeling to obtain simplified mathematical models that can be solved more readily and with less computational effort. Examples of common simplifications are incompressible flow models, one-dimensional flow models, common two-phase flow models such as the homogeneous equilibrium model (HEM), drift flux, and the two-fluid model, and simplified reactor kinetics models such as point kinetics or one-dimensional kinetics. Even models commonly thought of as detailed models usually contain simplifying assumptions and averaging procedures applied to first-principles models. Reviewers should confirm that justifications are provided for all simplifications, assumptions, and averaging.

The reviewers should confirm that the level of detail in the model is equivalent to or greater than the level of detail required to specify the answer to the problem of interest. For example, a one-dimensional flow model cannot provide information about the velocity profile in the vicinity of a pipe bend or the degree of thermal stratification in a horizontal pipe. A detailed, three-dimensional flow model would be needed to provide this type of information.

The reviewers should confirm that the equations and derivations are correct. There must be sufficient text to adequately describe the derivation, including all assumptions and equations. The derivations must be sufficiently detailed to allow the reviewers to understand the logical progression of steps involved in the derivation. Simplifying assumptions must have a technical justification and a range of validity associated with them.

Models that are typically used in nuclear reactor analysis are highly phenomenological and/or empirical in nature. They are either proposed using physical or engineering judgment based on observations of experimental data or derived using averaging procedures applied to detailed first-principle models. These models often represent processes that occur on length and time scales that are too small to be resolved in the computation or processes that we do not have sufficient understanding to model from first principles. These models require closure relationships based on information from experimental measurements or detailed first-principle calculations. The reviewers should ensure that the range of validity of the closure relationships is specified and is adequate to cover the range of conditions encountered in the accident scenario. This is especially true of empirical correlations that are derived directly from experimental data without recourse to any physical modeling.

In most applications, especially those with a large number of processes and parameters, it is difficult, if not impossible, to design test facilities that preserve total similitude between the experiment and the nuclear power plant. Therefore, optimum similarity criteria are identified and scaling rationales developed for selecting existing data or designing and operating experimental facilities. The reviewers should confirm that the similarity criteria and scaling rationales are based on the important phenomena and processes identified by the accident scenario identification process and appropriate scaling analyses.

The reviewers should confirm that scaling analyses were conducted to ensure that the data and the models will be applicable to the full scale analysis of the plant transient. Scaling compromises that are identified must be addressed in the bias and uncertainty evaluation. The experimental data base must be demonstrated to be sufficiently diverse, so that the expected plant specific response is bounded and that the evaluation model calculations are comparable to the corresponding tests in non-dimensional space. This demonstration allows extending the conclusions relating to code capabilities, drawn from assessments comparing calculated and measured test data to the prediction of plant specific transient behavior.

C. Accident Scenario Identification Process. The accident scenario identification process is required in order to determine the needed modeling and assessment requirements for the code. The accident scenario identification process is also needed to identify and rank the reactor component and physical phenomena modeling requirements based on their importance to acceptable modeling of the scenario and their impact on the figures of merit for the calculation. This process is highly dependent on the type of reactor and the accident scenario of interest.

Often a single computer code is used to analyze multiple accident or transient classes. A separate accident scenario identification description is needed for each accident or transient class for which the code is to be used in order to describe the accident progression and dominant physical phenomena for that particular accident. This description must explicitly reference code models and assessment cases that are specifically applicable to each scenario to avoid confusion when the same code is used for multiple accident scenarios.

The processes and phenomena that evaluation models should simulate are found by examining experimental data and experience, as well as code simulations related to the specific scenario. Independent techniques to accomplish the ranking include expert opinion, selected calculations, and decision-making methods.

The reviewers should confirm that the process used for accident scenario identification is a structured process. The reviewers should confirm that the description of each accident scenario provides a complete and accurate description of the plant initial and boundary conditions and the accident progression. The specified initial and boundary conditions must comply with values required by regulations or specified as acceptable in a regulatory guide or DSRS section that covers the accident. The reviewers should confirm that the dominant physical phenomena influencing the outcome of the accident are correctly identified and ranked. The reviewers should keep in mind that the initial phases of the accident scenario identification process can rely heavily on expert opinion and can be subjective. Therefore, iteration of the process, based on experimentation and analysis, is important.

One example of an acceptable structured process is the phenomena identification and ranking table (PIRT) process, which is described in NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of-Coolant Accident." The process is also described in a series of papers in Nuclear Engineering Design.

D. Code Assessment. The reviewers should confirm that the code assessment adequately covers all of the important code models and the full range of conditions encountered in the accident scenarios. The assessment must be consistent with the accident scenario identification process in that all models must have assessment commensurate with their importance and required fidelity. The reviewers should also verify that all assessment cases were performed with a single version of the evaluation model.

The reviewers should confirm that the numerical solution conserves all important quantities. Even when the mathematical equations conserve mass, momentum, and energy, the numerical method used to solve the equations may not conserve any of these quantities. The reviewers should also confirm that all code options that are to be used in the accident simulation are appropriate and are not used merely for code tuning.

The reviewers should ensure that all code closure relationships based in part on experimental data or more detailed calculations have been assessed over the full range of conditions encountered in the accident scenario by means of comparison to separate effects test data. Scaling analyses may be needed to demonstrate that the assessment results apply to the full-scale plant accident conditions. Even closure relationships such as equations of state and material properties, which are based on interpolating functions, need to be assessed against standards for the properties. For example, a relatively small error in thermodynamic properties such as phasic densities as a function of temperature and pressure may cause a larger error when propagated to a neutron kinetics model in a main steamline break or an anticipated transient without scram (ATWS) calculation.

The reviewers should ensure that integral test assessments properly demonstrate physical and code model interactions that are determined to be important for the full size plant accident scenarios. The integral test will usually not be full-scale, and therefore will contain scaling distortions. These distortions can affect both local and overall elements of the analysis, such as two-phase flow regime transitions and global dynamic response of the test facility, when compared to the full scale plant. The reviewers should ensure that the documentation contains comparisons of all important experimental measurements with the code predictions in order to expose possible cases of compensating errors. Such errors may result in good predictions of key parameters, derived from poor predictions of contributing parameters. These cases are often indicative of tuning the code to an integral experiment. One example of this is described in NUREG/CR-5249, in which key parameters, namely large-break LOCA (LBLOCA) reflood peak cladding temperatures were predicted well, even though an important contributing factor, the core void fraction during reflood, was not predicted accurately.

In the case of LOCA evaluation models, specific assessment test cases are required in order to meet the requirements of Appendix K to 10 CFR Part 50. Specific test cases are also specified in the TMI action items for pressurized-water reactor (PWR) small- break LOCA (SBLOCA) evaluation models. The reviewers should confirm that these assessments have been performed where applicable.

The reviewers should refer to published literature for sources of assessment data for specific phenomena, accident scenarios, and plant types. These include the ECCS Compendium, Nuclear Energy Agency (NEA) validation data documents, and International Standard problems.

E. Uncertainty Analysis. The reviewers should confirm that the method for calculating uncertainty contains all important sources of uncertainty and that a sample uncertainty calculation for a prototypical plant gives a reasonable estimate of the calculation uncertainty. The reviewers should confirm that the accident scenario identification process was used in identifying the important sources of uncertainty.

The reviewers should confirm that sources of code uncertainty have been addressed. These include uncertainties in theoretical models or closure relationships determined from comparison to separate effects tests, uncertainties due to scaling of the basic models and closure relationships, and uncertainties due to plant nodalization and solution techniques. The reviewers can sometimes determine whether the code uncertainty is reasonable by applying a simple analytical model. This can be done if the dominant contribution to uncertainty is confined to a small number of models. In other cases, detailed audit calculations may be needed to confirm the estimate of uncertainty in a calculation.

The reviewers should confirm that sources of calculation uncertainties have been addressed, including uncertainties in plant model input parameters for plant operating conditions (e.g., accident initial conditions, set points, and boundary conditions). Calculation uncertainties are specific to a given licensing calculation. Therefore, it is sometimes acceptable to use bounding values in the licensing calculation for these types of uncertainties.

The reviewers should confirm that the uncertainties in the experimental data base have been addressed. These uncertainties arise from such items as measurement errors and experimental distortions. For separate effects tests and integral effects tests, the reviewers should confirm that the differences between calculated results and experimental data for important phenomena have been quantified for bias and deviation. The reviewers should confirm that data sets and correlations with experimental uncertainties that are too large when compared to the requirements for evaluation model assessment are not used.

When the code is used in a licensing calculation, the reviewers should confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest in the calculation.

For best estimate LOCA analyses, uncertainty determination description and guidance are described in NUREG/CR-5249, RG 1.157, and Appendix A to RG 1.203. In these examples, the uncertainty analyses discussed have the ultimate objective of providing a singular statement of uncertainty with respect to the 10 CFR 50.46 acceptance criteria. This singular uncertainty statement is accomplished when the individual uncertainty contributions are determined (see RG 1.157).

For other events, a complete uncertainty analysis is not required. However, in most cases the DSRS guidance is to use “suitably conservative” input parameters. This suitability determination may involve a limited assessment of biases and uncertainties. The individual uncertainty (in terms of range and distribution) of each key contributor is determined from the experimental data, input to the nuclear power plant model, and the effect on appropriate figures of merit evaluated by performing separate calculations. The figures of merit and devices chosen must be consistent.

The NRC has developed the code scaling, applicability, and uncertainty (CSAU) methodology for code uncertainty evaluation. The CSAU process has been demonstrated for LBLOCA and boiling-water reactor ATWS. Other methods of uncertainty evaluation may be acceptable.

F. Quality Assurance Plan. The reviewers should confirm that the evaluation model is maintained under a quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50. As a minimum, the program must address design control, document control, software configuration control and testing, and corrective actions. The reviewers should confirm that the quality assurance program documentation includes procedures that address all of these areas. The reviewers may conduct an audit of the implementation of the code developer’s quality assurance program.

The reviewers should confirm that independent peer reviews were performed at key steps in the evaluation model development process. The peer review team should include programmers, developers, end users, and independent members with recognized expertise in relevant engineering and science disciplines, code numerics, and computer programming. Expert peer review team members, who were not directly involved in the evaluation model development and assessment, can enhance the robustness of the evaluation models. Further, they can be of value in identifying deficiencies that are common to large system analysis codes.

5. Independent Analysis. The reviewers may perform independent analyses in order to determine or confirm the importance of key phenomena and evaluate the impact of uncertainties in these phenomena on the key figures of merit in the plant calculation. The reviewers may want an independent determination of the importance of a high-ranking phenomenon, or to determine if a phenomenon not ranked as high by the licensee must be so ranked. Independent analysis may also be used to confirm uncertainty estimates. The NRR should be consulted, as needed, to accomplish these audits.

6. Review of General Purpose Computer Programs. Very often a general purpose transient analysis computer program, such as RELAP5, TRAC, or RETRAN, is developed to analyze a number of different events for a wide variety of plants. These codes can constitute the major portion of an evaluation model for a particular plant and event. Generic reviews are often performed for these codes to minimize the amount of work required for plant- and event-specific reviews. A certain amount of generic assessment may be performed for such a code as part of the generic code development. Applying portions of the review process to a general purpose transient analysis computer program is required to determine its suitability for use as the basis for an evaluation model.

The review process for the general purpose transient analysis computer program may indicate that the code models and generic assessment do not include all the appropriate geometry, phenomena, or the necessary range of variables to demonstrate code adequacy for some of the intended plant-specific event analyses. In order to avoid problems in the review, it is important that the submitted code documentation identify the intended range of applicability of the generic code, including its models and correlations. The "generic" assessment that accompanies the code must support the intended range of applicability of the code. The reviewer must ensure that the code models and assessment support the use of the code over its stated range of applicability. One goal of the review should be that evaluation models that use an approved general purpose transient analysis computer program can efficiently identify the models and assessment that support the analysis of the specific plant and accident types that the evaluation model will be used for in order to streamline future reviews of the evaluation models.

7. Review of Small Changes to Existing Evaluation Models. Application of the full review process described in this DSRS section may not be needed for all evaluation models submitted for review by the staff. Some evaluation models submitted for review are relatively minor modifications to existing evaluation models. The scope and depth of applying the review process to the evaluation model can be based on a graded approach. The following attributes of the evaluation model should be considered when determining the extent to which the full review process may be reduced for a specific application.

   A. Novelty of the revised evaluation model or changes to the model compared to the currently acceptable model.

      The level of effort involved in the review process should be commensurate with the extent of the changes made to an evaluation model. Small changes to a robust time tested evaluation model component such as a change to a simple heat transfer or drag correlation (possibly required by an error correction) may not require a full review of the entire evaluation model. In this case, scaling would only have to be considered within the context of how well the new model scales to full plant analysis if the model is developed from a reduced scale test program. Consideration would also have to be given to how well the assessment cases for the model represent full scale plant conditions. A small subset of the entire code assessment matrix may be adequate to test the phenomena that are affected by the model changes or the new model. Another subset of the code test cases may need to be performed to ensure that other parts of the model are not inadvertently impacted by the changes. The impact of any changes due to an error correction would have to be evaluated for the current license analysis of record. A large model change may require application of the review process on a much larger scale. Changing models from equilibrium, drift flux model to a two fluid, non-equilibrium model would be an example of a significant change that would require an extensive review for the new evaluation model.

   B. The complexity of the event being analyzed.

      The level of effort involved in applying the development process should be commensurate with the complexity of the evaluation model. At first glance the review process may seem too burdensome to apply to simple events. However, application of the review process to a simple evaluation models will automatically result in a simplified review process. In simple events the number of key physical phenomena should also be small and the code assessment only needs to cover the important phenomena even though the underlying general purpose transient analysis computer program may have models that cover a much wider range of conditions. An example of this is the system evaluation of a PWR pump trip analysis in which the important phenomena may be limited to a few quantities such as single-phase liquid wall drag and heat transfer, and pump inertia. In this case, very little assessment would need to be performed and there may be adequate full scale plant data for the code assessment so there would be no need for a scaling analysis. The other extreme is an evaluation model for a LBLOCA where the physical phenomena and the mathematical models are complex and cover a wide range of conditions. An extensive code development process and assessment and an extensive review process would be required in this case.

C. The degree of conservatism in the evaluation model.

The review process may be simplified if there is a large degree off conservatism in the evaluation model. The intended results of an analysis can be conservative due to a combination of code input and modeling assumptions. The amount of assessment required for a change to an evaluation model may be reduced significantly if the documented degree of conservatism is large or if the new model can be shown to give more conservative results than the previous model. However, conservatism in just one part of the evaluation model such as a heat transfer correlation cannot be used to justify conservatism in the evaluation model as a whole because other parts of the model may be non-conservative and cause the overall model to be non-conservative. The degree of conservatism in the overall evaluation model must be quantified and documented for the particular application in order to justify a reduction in assessment requirements using this argument.

D. The extent of any plant design or operational changes that would require a reanalysis.

The level of effort required to apply the process should be commensurate with the extent of changes made to the plant design or operation. Most of the changes to plant equipment or operations do not cause the plant to operate outside the range of validity of the evaluation model. In this case no additional development and assessment needs to be performed. This may not be the case for all changes. Examples of changes that may require changes to or additional assessment of the evaluation model are fuel bundle design changes (including grid spacer and intermediate flow mixer design changes), increases in the peak linear heat generation rate or operational changes that may cause reliance on a different safety grade trip which requires that accurate prediction of a quantity not required in the previous analysis. In these cases a limited application of the review process similar to that described in Section 6.1 should be sufficient.

8. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the licensee’s technical submittal meets the acceptance

criteria. DCs have referred to the licensee's technical submittal as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC submittal.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The reviewers should document all findings in an evaluation model SER that either accepts the evaluation model for the intended use or rejects the evaluation model. Acceptance may be subject to limitations determined during the review. If the evaluation model is to be accepted, it must be clearly demonstrated that the model is getting the right answer for the right reasons. There must be no evidence of compensating errors or arbitrary code tuning that produces the desired result for a single parameter. Any errors found in the documentation or issues related to the technical adequacy of the model must be addressed through the review process. The review process and the results of communications with the code submitter must also be documented as appropriate in the SER to provide a traceable history of the review process. Restrictions and limitations on use of the evaluation model must be explicitly documented in the SER. The SER must document areas that were reviewed, the method of review, and the findings in each area. The SER must also document areas that were not reviewed and provide the reasons for their omission.

The reviewers should determine if the evaluation model was previously reviewed and if the evaluation model previously had restrictions on its use or on the use of certain models in the codes. These previous restrictions must be explicitly addressed in the SER. The current review of the evaluation model may result in the imposition of new restrictions to be placed on its use. These restrictions must be explicitly identified. The applicant's documentation must also be revised to reflect these restrictions on use of the code. SER restrictions that have not been explicitly incorporated into evaluation model documentation have been determined to not be legally enforceable. Only statements in the approved description of the evaluation model are legally enforceable.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower^{TM}-specific DC, or COL, applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower^{TM} and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower^{TM} -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "an evaluation of the standard plant design against the SRP revision in effect six months before the docket date of the application." The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower^{TM} DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41), and COL applications.

VI. DEFINITIONS

All definitions are in the context of the objectives of this DSRS section and may not be generic to other uses.

Acceptance Review	An initial review of a submittal performed to ensure that sufficient information is included for the review to conduct the review.
Accidents	In this DSRS section, accidents and transients refer to events that are defined in NUREG-0800 to be analyzed to meet the requirements of the GDC, except for the fuel assembly misloading event and all radiological consequence analyses.
Codes	Calculational procedures that compose an evaluation model.
Code Options	User controlled options that control which procedures, models, correlations, etc., the code uses when performing calculations.
Code Tuning	The adjustment of parameters or options included in the code to achieve a predetermined result of code run.

Compensating Errors	A set of errors that, combined, mask the effect of the individual errors.
Correlation Effort	The change in one parameter as a result of a change in another parameter.
Closure Relationships	Equations and correlations required to close the field equations so that they may be solved. They relate unknown quantities to the variables of the field equations. They may include physical correlations of transport phenomena such as equations that relate the shearing stresses to the rate of strain (the velocity field).
Constitutive Equations transport	Equations and correlations required to close the field equations so that they may be solved. They relate unknown quantities to the variables of the field equations. They may include physical correlations of phenomena such as equations that relate the shearing stresses to the rate of strain (the velocity field).
Code Scaling, Applicability, and Uncertainty	A process to determine the applicability, scalability, and uncertainty of a computer code in simulating an accident or other transient. A PIRT and Uncertainty process is normally embedded within a CSAU process. See Reference 4.
Design-specific Review Standard	Acceptable plan for NRC reviewers to evaluate mPower Design Certification application.
Empirical	Derived from experimental data without recourse to physical modeling.
Evaluation Model	Calculational framework for evaluating the behavior of the reactor system during a postulated Chapter 15 event, which includes one or more computer programs and all other information needed for use in the target application.
Field Equations	Equations that are solved to determine the transport of mass, energy, and momentum throughout the system.
Figures of Merit	Quantitative standards of acceptance that are used to define acceptable answers for a safety analysis (e.g., DNBR limits and fuel temperature limits).
Frozen	The condition whereby the analytical tool(s) and associated facility input decks remain unchanged (and under configuration control) throughout a safety analysis thereby ensuring traceability of, and consistency in, the final results.
General Design Criteria	Design criteria described in Appendix A to 10 CFR Part 50.
Homogeneous Equilibrium Model	An analytical model for two-phase flow that assumes (1) both phases move at the same velocity, (2) the fluid is in thermal equilibrium, and (3) the flow is isentropic and steady.

Integral Effects Test	An experiment in which the primary focus is on the interactions between parameters and processes.
Model	(without "evaluation" modifier) - Equation or set of equations that represents a particular physical phenomenon within a calculational device.
Peak Cladding Temperature	The maximum fuel element cladding temperature.
Phase	State of matter involved in transport process, usually liquid or gas.
Phenomena Identification and Ranking Table	May refer to a table, or to a process depending on context of use. The process relates to determining the relative importance of phenomena (and/or physical processes) to the behavior of a nuclear power plant following the initiation of an accident or other transient. A PIRT table is a listing of the results of application of the process.
Request for additional information	A documented request, usually in the form of questions, sent by the NRC to the submitter to obtain more information on areas under review.
Roadmap	A document used to facilitate navigation through a larger and complex set of documentation.
Scaling	The process in which the results from a subscale facility (relative to a nuclear power plant) and/or the modeling features of a calculational device are evaluated to determine the degree to which they represent a nuclear power plant.
Scaling Distortions	Errors introduced into the data as a result of scaling the experimental facility.
Scenario	Time sequence of events.
Sensitivity Studies	The term is generic to several types of analyses; however, the definition of most interest here relates to those studies associated with the PIRT process and used to determine the relative importance of phenomena or processes. This may also involve analysis of experimental data that are a source of information used in the PIRT process.
Separate Effects Test	An experiment in which the primary focus is on a single parameter or process.
Safety Evaluation Report	A report by the NRC that evaluates a submittal and either accepts or rejects the proposals in the submittal.
Standard Review Plan	Acceptable plan for NRC reviewers, as set forth in NUREG-0800.

Transients	In this DSRS, accidents and transients refer to those events that are defined in NUREG-0800 to be analyzed to meet the requirements of the GDC, except for the fuel assembly misloading event and all radiological consequence analyses.
Uncertainty	There are two separate but related definitions of primary interest: a. The inaccuracy in experimentally derived data typically generated by the inaccuracy of measurement systems. b. The inaccuracy of calculating primary safety criteria or related figures of merit typically originating in the experimental data and/or assumptions used to develop the analytical tools. The analytical inaccuracies are related to approximations in solving the equations and constitutive relations.
User Manual	A document that includes modeling guidelines for the accident under consideration, procedures for selecting code inputs, and procedures for transferring information between different pieces of the evaluation model.

VII. REFERENCES

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987. (Certain updated sections are available from the NRC.)

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3. B.E. Boyak, et al., "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of-Coolant Accident," NUREG/CR-5249, USNRC, December 1989.

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5. "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, December 1988.

6. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations, NEA/CSNI/R(93)14, September 1993.

7. "Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients," Committee on the Safety of Nuclear Installations, NEA/CSNI/R(96)17, July 1996.

8. "CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients," Committee on the Safety of Nuclear Installations, CSNI Report 132, March 1987.

9. RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” May 1989.

10. NUREG/CR-6200, W. Wulff, et al., “Uncertainty Analysis of Suppression Pool Heating During an ATWS in a BWR-5 Plant: An Application of the CSAU Methodology Using the BNL Engineering Plant Analyzer,” March 1994.