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CSNI STATUS SUMMARY ON UTILIZATION OF BEST-ESTIMATE METHODOLOGY IN SAFETY ANALYSIS AND LICENSING

October 1996

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Paris

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

— to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
— to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
— to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium,Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973), Mexico (18th May 1994) the Czech Republic (21st December 1995), Hungary (7th May 1996), Poland (22nd November 1996) and the Republic of Korea (12th December 1996). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all European Member countries of OECD as well as Australia, Canada, Japan, Republic of Korea, Mexico and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

— *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
— *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
— *developing exchanges of scientific and technical information particularly through participation in common services;*
— *setting up international research and development programmes and joint undertakings.*

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

CSNI Status Summary on Utilization of Best-Estimate Methodology in Safety Analysis and Licensing

October, 1996

The Principal Working Group 2 Task Group on Thermal Hydraulic System Behavior has discussed the subject of the use of best-estimate codes in the licensing process. Codes that model thermal hydraulic processes important to assessing safety system performance have been in use for some time, at least since the 1970s.

Codes are generally referred to as "best-estimate" or "Appendix K." The former attempt to model processes and phenomena as accurately as possible within the constraints of inherent limitations of physical modeling, numerics, or other factors. The latter include intentional conservatisms in an attempt to bound problems in view of limitations in knowledge. RELAP4/MOD6 is the most prominent example of the Appendix K modeling approach, but several vendor codes have also been developed along the same lines. The reasons for the two categories are historical.

In 1974, the United States Nuclear Regulatory Commission (USNRC) promulgated rules governing analysis of loss-of-coolant accidents (10 CFR 50.46 and Appendix K). These rule codified a rather prescriptive procedure for performing LOCA analysis. In view of existing uncertainties in the data and limitations in modeling, artificial conservatisms were introduced to various parts of the analysis. In adopting Appendix K the Commission mandated that research be carried out to reduce the uncertainties associated with LOCA analysis and that the rule should be revisited when better information was available. Thus commenced an extensive research program that encompassed such major projects as LOFT and 2D/3D. It also led to the development of the first two-fluid codes such as TRAC.

After nearly 15 years, the mandate was considered to have been successfully completed. In 1988 a Compendium of ECCS Research for Realistic LOCA Analysis (NUREG-0800) was published by the USNRC and in 1989 the LOCA rule was revised to permit realistic analysis, with appropriate accounting for uncertainties (10 CFR 50.46 (a)(1)(i). Since to that time an accepted method of quantifying the uncertainty of large thermal hydraulic computer codes did not exist, the USNRC developed the Code Scalability, Applicability and Uncertainty (CSAU) methodology, published in the report, Quantifying Reactor Safety Margins (NUREG/CR-5249), in 1989. Other countries have established efforts to quantify code uncertainty as well. A discussion of some of these methods can be found in the proceedings of a CSNI workshop on the subject held in 1994 (NEA/CSNI/R(94)20)

The new rule and the methodologies to quantify code uncertainty established a basis for using best-estimate codes in the formal licensing process. Of course, such codes had come into widespread use prior to that time for safety studies.

The Task Group set out to determine the prevailing practices in member countries concerning safety assessment and safety review of transients affecting the reactor coolant system. The following information is from summary material provided by member countries in response to the eleven questions that were formulated. The information was assembled from presentations and from responses to a questionnaire that was distributed to the Task Group in 1994. It is for general information and status only and is not intended as comprehensive or official statements of position.

The available information is sufficient to constitute a superficial survey for the eleven questions that were formulated. The survey revealed a great deal of commonality of practices in the member countries, although the details may differ. The results of the survey are given in Appendix 1.

Responsibility for Safety Analysis

The utility is normally responsible for producing the safety analysis of the plant. This work is often contracted to the vendor, who generally prepares the initial safety analysis report. This reflects the philosophy that the owner and operator of the plant is the best entity to achieve safety in day to day operations.

Responsibility for Review and Evaluation of Safety Analysis

The review of the safety analysis submitted by the utility plant owner resides with a government authority. The practice of the utility owner submitting safety analysis to the government for review and approval is universally accepted amongst the member countries.

Use of Best-Estimate Codes

Regulations normally permit use of best-estimate codes, but there may be added requirements for conservative assumptions, sensitivity studies or uncertainty studies. Examples of conservative assumptions may include equipment unavailability, operator actions or inactions, plant initial conditions, or code model features. As far as is known, no regulations rule out the use of Appendix K codes, that is codes using models that are "artificially" conservative.

Definition of Best-Estimate Code

Simply stated, a best estimate code should contain realistic models for relevant phenomena. A more precise definition of "best-estimate" code does not appear to be in general usage. The term is usually taken to mean that the code: (1) is free from deliberate pessimisms; and (2) contains sufficiently detailed models to describe the relevant processes in the transients that the code is designed to model. Codes that are designed to be unbiased include: ATHLET, CATHARE, CATHENA, RELAP5/MOD2 and MOD3, TRAC-PWR, TRAC-BWR, and COBRA-TRAC.

The USNRC's Regulatory Guide 1.157 provides an example of a definition of best estimate, provided here as an attachment.

Code Documentation

Member countries normally have requirements that are stated in general terms for documentation of code models and correlations. More precise requirements for code documentation may vary or may not have been generally promulgated. There is normally a requirement to review code models and correlations to ensure that the code has appropriate models for the important phenomena and that models are not applied outside the range of their validity. The specifics of what is required vary.

One example of an attempt to define documentation and review of models and correlations can be found in NUREG-1230 (p. 4-109), where it states:

> "For correlations, models, criteria or constants used in the code, the models and correlations document must:

1. Provide information on its original source, its data base, its accuracy, and its applicability to nuclear power plant conditions;

2. Provide an assessment of effects, if it is used outside its data base;

3. Describe how it is implemented in the code, that is, how it is coded;

4. Describe any modifications required to overcome computational difficulties; and

5. Provide an assessment of effects due to implementation (item 3) and/or due to modifications (item 4) on code overall applicability and uncertainty."

Note that this definition is not a prescribed regulatory requirement, an example of which is given in Appendix 2.

Code Assessment

There is normally a requirement that the code be assessed against relevant experimental data for the important phenomena expected to occur. The specifics of what is required will vary according to the particulars of the safety assessment under consideration. That is, the phenomena deemed important depend on the transient being evaluated.

Initial and Boundary Conditions

Initial and boundary conditions are either chosen to be conservative or are varied to evaluate the uncertainty. An example of a conservative boundary condition would be to use limit values from the plant's technical specifications as opposed to nominal operating conditions.

Operability of Active Equipment

Equipment availability may be treated as best estimate, with accounting for uncertainties, or single limiting failure may be assumed.

Operator Actions

Operators are generally modeled as responding to procedures. Hands-off operation is often assumed for some period of time. Uncertainties of operator behavior not routinely considered within the context using best-estimate codes for evaluating the performance of safety systems.

Appendix 1

Responses to Questionnaire

1. Who is Responsible for Safety Analysis?

Belgium:	Utility (contracted to the Vendor)
Canada:	Utility
Czech:	Vendor or Utility
Finland:	Utility (may be contracted to the Vendor or Consultant)
France:	Utility
Germany:	Utility (usually contracted to the Vendor)
Italy:	Utility
Japan:	Utility (usually contracted to the Vendor)
Netherlands:	Utility (usually contracted to the vendor or consultant)
Spain:	Utility
Sweden:	Utility
Switzerland:	Utility (usually contracted to the Vendor)
United Kingdom:	Utility
United States:	Vendor or Utility

2. Who is Responsible for Review and Evaluation of Safety Analysis?

Belgium:	The methodology and the results of the analysis has to be approved by AIB-Vincotte Nuclear (AVN). Some further review may be performed by the architect engineer.
Canada:	Atomic Energy Control Board
Czech:	State Office for Nuclear Safety (SUBJ)
Finland:	Finish Centre for Radiation and Nuclear Safety (STUK)
France:	Government (DSIN) with technical support from ISPN
Germany:	State Government (TUVs, usually with advice from GRS); Guidelines for performing reviews supplies by RSK.
Italy:	ANPA/DISP
Japan:	Ministry of International Trade and Industry (MITI), Nuclear Safety Commission (final approval)
Netherlands:	Ministry of Social Affairs (Nuclear Safety Department KFD) and Ministry of Environment
Spain:	Consejo de Seguridad Nuclear (CSN)
Sweden:	Utility, with final approval by SKI
Switzerland:	Swiss Federal Nuclear Safety Inspectorate (HSK)
United Kingdom:	Utility, Nuclear Installations Inspectorate
United States:	Nuclear Regulatory Commission

3. Do the Regulations Permit the use of Best-Estimate Codes?

Belgium:	Yes
Canada:	Yes
Czech:	Yes
Finland:	Yes
France:	No for DBA, Yes for beyond DBA
Germany:	Yes
Italy:	DBA analysis must be done based on 10 CFR 50 Appendix K requirements. May be supplemented by best-estimate analysis for specific safety issues.
Japan:	Yes
Netherlands:	Yes
Spain:	Yes
Sweden:	Yes
Switzerland:	Yes
United Kingdom:	Yes, not best estimate analysis
United States:	Yes

4. What are the Requirements for What Constitutes a Best-Estimate Code?

Belgium:	The models must realistically describe the physical processes encountered, based on comparison to relevant experiments.
Canada:	Realistic models of the physical system being modeled, validated against experiments.
Czech:	The computer codes and models which are used must be realistic. Computer codes and all correlations must be validated against experiments. In the case that the computer codes are standardized the validation is not required.
Finland:	Formal requirements do not exist. In practice, a code should not contain built-in conservative assumptions.
France:	Employ modeling that realistically describes the physical processed that occur in the reactor system.
Germany:	No formal requirements specified.
Italy:	Use to the greatest extent possible mechanistic models. The code should be well-assessed against separate effects and integral experimental data. Counterpart test data are essential in the validation process.
Japan:	The models and correlations must model the relevant phenomena, e.g. two-phase flow, heat transfer, multi-dimensional effects, frictional and form losses, reactor coolant pump, emergency core cooling systems, noncondensables, etc..
Netherlands:	Employ modelling that realistically describes the physical processes that occur in the reactor system (example USNRC Reg Guide 1.157).
Spain:	Definition of Regulatory Guide 1.157
Sweden:	The models must realistically describe the physical processes that occur.
Switzerland:	No formal requirements specified. Physical processes should be realistically modeled.
United Kingdom:	Employ modeling that realistically describes the significant physical processes that occur in the reactor systems.
United States:	Employ modeling that realistically describes the physical processes that occur in the reactor systems (defined in Regulatory Guide 1.157).

5. What Codes are used in Practice?

Belgium:	WCOBRA/TRAC, TRAC-PF1/MOD1, RELAP5/MOD2, NOTRUMP, COBRA3CP
Canada:	CATHENA, TUF
Czech:	RELAP5/MOD3, DYNAMIKA, ATHLET, CATHARE, MELCOR, DYJE, D4-11
Finland:	RELAP5, SMABRE, TRAB
France:	CATHARE
Germany:	RELAP5, ATHLET
Italy:	RELAP5
Japan:	Vendor codes, RELAP5 (not for licensing)
Netherlands:	RELAP5, TRAC-PF1, TRACG, RAMONA
Spain:	RELAP5/MOD2, TRAC-BF1
Sweden:	GOBLIN (BWR LOCA), BISON and RAMONA (BWR transients), COPTA (BWR containment), RELAP5 (PWR), WCOBRA-TRAC (PWR)
Switzerland:	RELAP5/MOD2, TRAC-BF1, RAMONA, Vendor codes (WCOBRA-TRAC, NOTRUMP)
United Kingdom:	NOTRUMP, WCOBRA/TRAC, TRAC-PF1/MOD1, BART
United States:	RELAP5/MOD3, TRAC-PF1/MOD2, TRAC-BF1, RAMONA, WCOBRA-TRAC, NOTRUMP, TRAC-G

6. What are the Requirements Concerning Code Documentation?

Belgium:	Review for completeness, clarity, and consistency. Note and resolve shortcomings.
Canada:	Under discussion.
Czech:	Codes must be fully documented including description of models, validation of correlations, and validation of the complete code.
Finland:	A description of the general principles, physical models and numerical methods must be given. If correlations are used, then the experimental data used to derive the correlation must be presented or referenced.
France:	Formal requirements under discussion.
Germany:	No formal requirements specified.
Italy:	Completeness in terms of model description and implementation, user guidelines, model validation and limits of applicability.
Japan:	Must meet guidelines.
Netherlands:	Describe models, range of applicability, validation references in accordance with IAEA Safety Guide QA6.
Spain:	10 CFR 50 Appendix B
Sweden:	Describe basic principles, physical models, numerical methods, and extent of assessment.
Switzerland:	Describe models and validation, including range of applicability.
United Kingdom:	All codes used in licensing submissions must be fully documented (to include user document, verification statement, validation evidence, etc.)
United States:	Must meet 10 CFR 50 Appendix B.

7. What are the Requirements for Review of Code Models and Correlations?

Belgium:	Review models and correlations for range of applicability and comparison with data; determine bias and uncertainty.
Canada:	Suitability of the model to represent physical phenomena, validation against suitable experimental data, and use of the model within validated range.
Czech:	Must be validated against experimental data. Models and correlations must be applied only in the range of their validity.
Finland:	The reliability of the calculational method shall be justified. Correlations must be supported by comparison with experimental data.
France:	Formal requirements under discussion. For a particular application, the models and correlations are examined with respect to their data base.
Germany:	Validity of correlations shall be demonstrated on the basis of experimental results.
Italy:	Relevant physical phenomena must be modeled and assessed against qualified test data.
Japan:	List of acceptable models and correlations maintained. Other models and correlations must validated by comparison with experimental data. Code must be shown to be applicable to new applications.
Netherlands:	Models and correlations must be verified against experimental data. Uncertainties of models and correlations must be quantified.
Spain:	Verify models and correlations against relevant data and quantify uncertainties and biases.
Sweden:	Models and correlations are reviewed for range of applicability, uncertainty, and bias.
Switzerland:	Models and correlations must be verified against experimental data. Uncertainties of models and correlations must be quantified.
United Kingdom:	Models and correlations must be developed and verified against relevant data. Ranges of validity must be established. Models and correlations should not be applied outside of the range of their validity.

United States: Models and correlations must be developed and verified against relevant data. Models and correlations should not be applied outside the range of their validity. Uncertainties and biases of models and correlations should be stated.

8. What are the Requirements Concerning Code Assessment?

Belgium:	Perform sufficient assessment to assure the results are reasonable, self consistent, and do not violate physical laws.
Canada:	State-of-the-art theoretical basis and validation against separate effect and integral tests.
Czech:	Code must be validated against separate effect and integral test and plant transients.
Finland:	Calculational methods and physical models are assessed for relevant phenomena using separate effects tests, integral tests, plant transients and by comparison with already verified models. Numerical methods are verified by means of adequate reference calculations.
France:	Code should be assessed against relevant integral and separate effects experiments including beyond DBA experiments.
Germany:	Relevant assessment should be performed. Submitted analyses should be experimentally confirmed.
Italy:	Relevant physical phenomena should be assessed against qualified test data. Counterpart tests data should be used as much as possible. Code uncertainty should be quantified.
Japan:	Models and correlations must be shown to be valid through comparison with experimental data.
Netherlands:	Code should be assessed against separate effect tests, integral tests, and plant transients (scaling should be treated appropriately).
Spain:	Code should be assessed against relevant integral and separate effects experiments of different scales. Comparisons against transient plant data can also be taken into account.
Sweden:	The code should be assessed against relevant physical phenomena using separate effect and integral data. For transient codes used for older plants it has been the practice to compare with plant transients that have occurred.
Switzerland:	Code should be assessed against separate effects tests, integral tests, and plant transients.
United Kingdom:	Codes must be adequately verified and validated for their intended applications.

United States: Code should be assessed against relevant integral and separate effects experiments of different scales.

9. What are the Requirements Concerning Initial and Boundary Conditions?

Belgium:	Perform sensitivity studies, retain worst case results and associated initial and boundary conditions within the expected ranges.
Canada:	Represent in a conservative manner.
Czech:	Use conservative initial and boundary conditions.
Finland:	Initial conditions are selected conservatively (e.g. at the 95% level). Limiting single failure and minimum safety system output assumed. Maintenance of redundant system also assumed. Excess safety system output also considered.
France:	Use most probable initial and boundary conditions.
Germany:	Use conservative initial and boundary conditions.
Italy:	Use conservative initial and boundary conditions.
Japan:	Uncertainties in initial and boundary conditions must be taken into account.
Netherlands:	Use conservative initial and boundary conditions or best-estimate conditions including uncertainty analysis.
Spain:	Uncertainties in initial and boundary conditions should be included in uncertainty analysis.
Sweden:	Initial and boundary conditions may be included in the uncertainty analysis or biased to worst case values.
Switzerland:	Use conservative initial and boundary conditions, otherwise account for their uncertainties as part of the uncertainty analysis.
United Kingdom:	Use conservative initial and boundary conditions.
United States:	Account for uncertainties in initial and boundary conditions in the uncertainty analysis.

10. What are the Requirements Concerning Operability of Active Equipment?

Belgium:	Use most limiting single failure. Remaining safeguards equipment is assumed to operate at minimum rated capacity plus or minus uncertainties whichever is conservative versus the applicable criterion.
Canada:	Dual failure assessment and only credit the second trip on the slower of the two shutdown systems.
Czech:	Use most limiting single failure. An additional repair unavailability may be required.
Finland:	Normally operating systems assumed to operate in most probable mode. Protection systems operate as designed. Safety systems assumed to operate at minimum output, with single failure (on demand) and an additional train assumed unavailable due to repair.
France:	Qualified equipment assumed to operate.
Germany:	Assume single failure and an additional repair unavailability.
Italy:	Use most limiting single failure in DBA analysis. Multiple failures are assumed in beyond DBA PSA studies.
Japan:	Uncertainties in equipment availability and performance must be taken into account. Limiting single failure may be assumed.
Netherlands:	Use most limiting single failure in the conservative analysis and use equipment availability and performance in the best-estimate analysis.
Spain:	Account for uncertainties in equipment availability and performance or consider limiting single failure.
Sweden:	Assume the most limiting single failure.
Switzerland:	Use most limiting single failure. An additional repair unavailability may be required.
United Kingdom:	The deterministic analysis must use the worst normally permitted configuration of equipment and make due allowance for common cause and single failures.
United States:	Account for uncertainties in equipment availability and performance in the uncertainty analysis. In practice, limiting single failure may be otherwise assumed.

11. What are the Requirements Concerning Operator Actions?

Belgium:	No operator action should be necessary for the first 30 minutes. Operator action is only assumed for long transients (e.g. SGTR, feedwater line break, LOCA during mode 3 or 4 operation).
Canada:	No credit for operator action for 15 minutes.
Czech:	No operator action within the first 30 minutes.
Finland:	Operator's act according to procedures. Credit for operator actions if the event is clearly identifiable, event-specific procedures exist and time is allowed for operator action.
France:	Assume operators perform according to procedure.
Germany:	Actions involving operational system covered by conservative initial and boundary conditions. Safety systems automated such that no operator action is necessary during the first half hour and operators cannot disable safety systems.
Italy:	Assume operator actions according to procedures. Assume worst single operator error. No operator action before 30 minutes (for future plant longer periods may be considered).
Japan:	Operator is assumed to perform according to procedures.
Netherlands:	Operators actions within the first 30 minutes are not credited in the Safety Analysis.
Spain:	Assume operator actions according to procedures.
Sweden:	No operator actions within the first 30 minutes. After that, operators assumed to perform according to procedures.
Switzerland:	No credit for operator actions in the first 30 minutes of a design basis accident.
United Kingdom:	No human action should be claimed in the analysis of a design basis fault during the first 30 minutes following its initiation.
United States:	Assume operator performs according to procedures.

Appendix 2

Example of Prescribed Regulatory Requirement for Code Documentation
(applicable to "best estimate" as well as "Appendix K" analysis)

Excerpt from 10 CFR 50

50.46 Acceptance criteria for emergency core cooling systems for light water nuclear power reactors.

(a)(1)(i) Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted....

Appendix K.II. Required Documentation

1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

   b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability - Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by 50.46(a)(1)(ii), compliance with required features of Section I of this Appendix K; and, for models covered by 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of 50.46(b) would not be exceeded.