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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(99)8
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JOINT IAEA/NEA WORKSHOP PROCEEDINGS ON REGULATORY REVIEW OF PLANT SAFETY ANALYSIS

Bratislava, Slovak Republic
July 6-10, 1998

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**ORGANISATION FOR ECONOMIC CO-OPERATION
AND DEVELOPMENT**

Pursuant to Article I 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 OECD Member countries except New Zealant and Poland. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of the 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, the 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.

Proceedings from the
Joint IAEA/NEA Regional Workshop on
Regulatory Review of Plant Safety Analysis
6-10 July, 1998, Bratislava, the Slovak Republic

Content of Proceedings:

Joint IAEA/NEA Regional Workshop on
Regulatory Review of Plant Safety Analysis
6-10 July, 1998, Bratislava, the Slovak Republic

FOREWORD

The Workshop was organised jointly by the OECD Nuclear Energy Agency (NEA), the International Atomic Energy Agency (IAEA) and the Nuclear Regulatory Authority (UJD) of the Slovak Republic.

The proposal for a joint IAEA/NEA workshop originated from discussions between Ms Carnino (IAEA), Mr Frescura (OECD/NEA) and Mr Misak (then Chairman of UJD the Slovak Republic) at the VVER regulators meeting (held 4 August 1997 in Helsinki). The proposal was subsequently approved by the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) and agreed to by the Technical Co-operation Department, IAEA. This proposal was implemented by adapting an IAEA workshop on "The Regulatory Review of Safety Analysis Reports" which was already planned under Regional programme RER/9/052. In preparing the workshop the aim was to meet the recipient countries requests on scope and content which were described at the RER/9/052 planning meeting (held 16-17 December 1997 in Vienna) and to concentrate more specifically on accident analysis. This led to a syllabus with a balance between technical and regulatory topics. The UJD of The Slovak Republic agreed to host the Workshop.

The main purpose was to assist regulatory authorities to meet their assigned tasks when reviewing the safety analysis of a nuclear power plant. In particular, the Workshop provided a forum for sharing experiences and opinions on performing and reviewing plant thermal-hydraulic safety analysis, including legal, institutional and technical aspects of the subject.

The workshop was designed for staff of regulatory authorities and other organisations who are responsible for or involved in the regulatory reviews of plant safety analysis. They came from the Europe Region Member States who are recipients of IAEA's Technical Co-operation programme. In addition, the NEA invited several internationally recognised experts to make presentations on selected topics. The NRA provided technical support and administrative assistance to the Workshop.

Responsible persons were Dr. Derek Lacey of IAEA, Mr. Andrzej Drozd of NEA and Mr Jan Husarcek of NRA. In addition to 6 participants from the Slovak Republic, there were also 16 participants from Armenia (2), Bulgaria (1), Czech Republic (4), Hungary (1), Lithuania (1), Romania (2), Russia (3) and Ukraine (2). The list of participants is attached.

The invited lecturers were from Belgium (Mr. Ray Ashley of AVN), Canada (Mr. Fabio De Pasquale of AECB), Finland (Dr. Juhani Hyvarinen of STUK), Hungary (Mr. Ivan Toth of KFKI) and the US (Mr. David Bessette of US NRC, and Mr. Michael Modro of INEEL). There was also a contribution from Mr. Eric Beckjord (consultant to the US NRC), who could not attend the workshop.

Derek Lacey IAEA	Andrzej Drozd NEA	Jan Husarcek NRA

Joint IAEA/NEA Regional Workshop on Regulatory Review of Plant Safety Analysis 6-10 July, 1998, Bratislava, the Slovak Republic

Summary and Conclusions

SUMMARY OF DISCUSSIONS

Based on the final discussion, the seminar was a valuable exchange of information and helpful in understanding the safety analysis and regulatory review issues. It gave a chance for some countries to see "where they are" regulatory-wise with respect to other countries. There was a strong support for a follow-up workshop within the next two years. A general opinion expressed by most of the participants was that the suggested next seminar should be based more on an "overall safety analysis" review rather than on very detailed presentations on specific issues. Also, there should be more discussions on Emergency Operating Procedures (EOPs) and emergency planning. In addition, to establish a "common language", the development of a glossary of terms would be most helpful.

In general, all of the participating countries are adapting regulatory and licensing practices previously developed by other countries. Such an approach allows to utilise and rely on a broad experience of countries with a well established nuclear power industry, strong regulatory organisation and licensing practices. Since there are obvious differences in engineering traditions and in countries' legal/regulatory frameworks, questions arise about how to reconcile these differences in adapting and developing practices suitable for a given country. It was advised as a good practice to identify from the very beginning what needs to be done and the goals to be accomplished not only in the perspective of safety and technical requirements but also in the context of the economical, political, legal and regulatory situation in a given country. There has to be a consistency in any approach, but one can never follow "blindly" the others. It became quite obvious that the needs of countries with a big nuclear program are very different from those of "small" countries. Adaptation of other countries' rules is a big effort in itself (e.g., Romania applying Canadian regulations). In addition, the French-German co-operation was given as an example that even a "common" approach may result in different "specific" requirements. However, the assistance of countries with a longer tradition of regulatory development appears to be very important and very much needed for the countries that are building and/or modifying their regulatory structure. The international organisations may serve as a helpful facilitator and/or moderator in this area.

One generic "good practice" recommendation made was not to rely on a single analysis. Various approaches should be applied and various possible solutions should always be discussed. When an issue cannot be solved analytically, e.g., BWR strainers clogging, then it has to be "treated" differently.

SPECIFIC ISSUES RAISED

Frequency of periodic licensing reviews. There were no clear indications which of the existing models should be recommended, e.g., in the US a plant gets a license for life with required updates of the FSAR to reflect every modification made. In Belgium there is a 10-year review schedule (license renewal), during which all the main points must be checked; and in Canada there is a 2-year licence renewal cycle with periodic safety reviews for which a 10-year cycle is being considered.

The presented approaches to safety evaluations are based on (a) analyses review by various experts (Belgium); (b) deterministic approach used to evaluate "safety margins" (Canada); (c) independent experts' judgement (e.g., Bulgaria); (d) deterministic analyses supported by a probabilistic evaluation (Finland), and (e) non-prescriptive guidelines for safety review (e.g., Ukraine).

The relationship between the regulators and the industry is an important element of the safety culture. As there are many cultural differences, a "proper" regulator/industry interface has to reflect the legal tradition of a given country.

Validations of codes. The codes are validated through "various" means and international projects have often been extremely useful in this regard. The final result has to be a building of confidence in the use of codes. In Russia, for example, every code that has been evaluated has a "passport" issued were the limits of its applicability are specified. In general, an important part of the review is having an "integrity" and "consistency" in the final approval of analysis.

The of scaling and extrapolation (applicability) to "real plants" always needs to be carefully considered. This complex issue is usually "solved" (although sometimes without a rigorous verification) through discussions with the industry.

Regarding applicability of the Western codes, in general, the codes were found to be rather flexible except for Severe Accident applications.

Presentations on prescriptive and non-prescriptive regulations indicate that, practically, there is not much difference for a reviewer, because typically the same things need to be done regardless of the approach. A good example may be Canada where, although a relatively non-prescriptive regulatory approach is used, some rather detailed guidelines on how to perform safety analysis reviews are currently being produced. An extreme case may be the Czech approach, who are in a process of developing a fairly rigorous "algorithm" of plant safety review based on the NRC Standard Review Plan.

FOLLOW-UP ACTIVITIES

A follow-up questionnaire has been distributed and a summary of the results is included in the proceedings.

A follow-up workshop within the next two years is strongly recommended by the participants. Prior to the next Workshop a survey should be carried out to determine different countries' priorities of topics covered in Bratislava and those suggested during discussion (like EOPs and emergency planning). Based on the results of the survey issues of common interest should be established.

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Abstracts of selected presentations

Initiating events and acceptance criteria (Juhani Hyvärinen)

Safety practices vary among nations due to many factors (cultural differences) such as legal traditions, administrative structures (federal vs. central government), and attitude towards regulatory evolution. Additional significant factors include possible vendor presence in the country, size of the nuclear industry and regulator's resources. All these elements have directly affected the evolution of rules and practices pertaining to nuclear energy utilisation, and explain why practical approaches to nuclear safety differ in different countries also among the so-called Western nations.

Safety analysis can be defined as a process of familiarisation with the plant in question, its specific strengths and weaknesses, and can make use of any engineering tool available, up to the point of advanced safety analysis computer codes and large-scale experiments. The primary purpose is to check compliance with safety requirements and verify adequacy of safety functions: subcriticality, core cooling, containment/confinement performance. Ideally, this results in physical understanding of plant behaviour under all conditions.

The Regulator needs to define the basic assumptions such as single failure criterion, allowance of operator actions, use of non-safety grade systems, selection between control and protection system signals, treatment of boundary conditions, e.g. minimum vs. nominal capacities, etc. Also there is a need to address **uncertainties.**

The problem is that several kinds of uncertainties exist. Usually quantifiable are stochastic (e.g. constitutive laws, some parameters), boundary condition (valve setpoints etc.) related uncertainties; incompletely understood physics (too complex processes) may be quantifiable in a bounding sense.
Poorly or not at all quantifiable are uncertainties related to numerical accuracy, (e.g. violation of conservation laws, diffusion, dispersion), bifurcating physical processes (trip signals, critical phenomena), physical model simplifications (e.g. 1D flow), code model structure (nodalisation), and honest mistakes in data, coding, model, structure, user.

Introducing safety margins (i.e., "conservatism") is currently the only known solution to address uncertainty. These "conservatisms" can be incorporated either within the criteria or imposed on the analyses, but then the question arises, how large is large enough? Experience seems to indicate that it is wise to avoid excess conservatism - safety goals may be conflicting (e.g. core cooling vs. avoidance of pressure vessel thermal shock).

"Conservative" assumptions aim to maximise the physical threat but cannot be made unless a good (at least) qualitative physical picture of relevant physics is at hand. Stacking conservative assumptions may lead to misleading or physically meaningless results. As a rule of thumb it appears advisable to put conservatisms in rate processes (e.g. heat transfer coefficients) but never violate conservation laws (some existing regulatory guidelines do not comply with this and have hence caused much trouble in practice).

The available safety margin can be estimated by doing sensitivity analyses: understand physically what goes on, identify key phenomena & parameters, vary them over sufficient range. Formal uncertainty quantification methods often have difficulty dealing with other than quantifiable uncertainties, yet non-quantifiable uncertainties may dominate. Hence it is not advisable to believe a single result, be it "best-estimate" or "conservative".

The selection of initiating events to be analysed should reflect plant specific features and cover all relevant transient and accident scenarios. One may use original design basis, operational experience, and peer experience as a guide; again, it is essential to know your plants. The Internal event spectrum should exercise all safety functions, external events should expose vulnerabilities (interact with PSA to identify system interdependencies). The usual practice is to classify events according to their expected frequency of occurrence, and use more relaxed physical acceptance criteria for more rare events.

Regarding VVER specificity in terms of initiating events, relatively large primary to secondary leaks need to be considered in depth because steam generator primary collector lid & bolt integrity has caused problems. Reactor pressure vessel embrittlement also requires attention.

The acceptance criteria are the allowable bounding values for key process parameters, e.g. energy deposition in fuel (RIA), peak temperatures (cladding during LOCA, containment), peak pressures (primary & secondary system), or radiological limits. Their definition may or may not include safety margin. Ideally, they should, for their part, work towards balanced risk: stringent criteria for frequent events, less stringent for rare events. It is advisable to try and avoid too abrupt transitions between categories.

Examples of criteria on criticality are: limits on reactivity feedback coefficients, shutdown margin, e.g. minimum 1%, maximum control rod worth, demonstrate redundant shutdown capability (rods and boron). As another example, consider core cooling: DNB criterion for transients, overpressure and thermal shocks for integrity study, LOCA & RIA criteria for accidents, and adequacy of ECCS inventory, recirculation reliability for long-term cooling. Typical containment/confinement performance criteria relate to leak rate (normal operation), isolation valve leak tightness testing limits, overpressure (structural limit; transients, accidents) and temperature (equipment qualification).

In conclusion, safety analysis equals safety function verification, by whatever engineering means are available, and should produce understanding of plant behaviour. Analysis always involves uncertainties whence safety margins are needed; initiating event selection should reflect plant specific features. Criteria and event selection also direct design efforts and should promote balance between various events and between different systems.

<u>Approaches to accident analysis and computer code use</u> (Juhani Hyvärinen)

The basic goal of accident analysis is to understand plant behaviour under accident conditions and to verify the adequacy of safety functions. Anticipated transient, Design Basis Accident (DBA) and Beyond DBA (BDBA) analyses are almost always done using ”system codes”, and coupled thermal hydraulic - reactor dynamic code systems are also in use or being developed. The division between transient and accident codes which, was clear in the past, is presently rapidly vanishing.

Basically there are two approaches to computer code usage, Conservative and Best-Estimate. Traditional conservatism (e.g. 10CFR50.59 Appendix K) attempted to maximise the physical threat by deliberately distorting calculation (typically making quite extreme assumptions on physical processes, sometimes even violating conservation laws). This approach results in fair certainty in the main variable (if overestimated, all others are respectively underestimated), even with ”one shot”, but has **no use** outside main variable - criterion comparison and is potentially dangerous if other variables used. An advantage is that well established acceptance criteria exist.

Best-estimate approach attempts to generate a “realistic” estimate of the physically most likely behaviour - there are no deliberate distortions. If successful, it is useful for many purposes (design, EOP development, PSA,...) but necessitates careful uncertainty analysis. Recall that some uncertainties are difficult if not impossible to quantify; likewise well established acceptance criteria do not exist to same extent as with the conservative approach.

Finally it is possible to introduce moderate conservatisms into best-estimate framework (to generate "intelligent conservatism"). This requires first to understand the physics (at least qualitatively), then introduce one or more conservatisms, and do sensitivity studies (with and without conservatisms). When making use of such results one should retain adequate safety margins, recognising the broad context and constraints from other than the ”main” safety issue. This approach requires competence and is difficult to formalise in detail, but pays off it requires a lot less effort than Best-Estimate approach.

Some key features of ”Western” system codes, both thermal hydraulic and reactor-dynamic, were also discussed. These codes were mostly developed for ”general purpose” use and incorporate multitude of interacting phenomena such two-phase flow, heat transfer, reactor dynamics, and specific components (valves, pumps, turbines,...). However, developers have had primarily their own reactors in mind, which affects component models & sometimes general modelling facilities **but not** essential physics. Specific details of physical models vary, as do numerical schemes, but none has proven superior to others (nor will). There are no major differences in predictive capabilities.

Generally, a computer code needs to be validated before application. This is a demonstration that the code can be used for some practical purpose, and is usually done by comparing code predictions (blind or open) to experimental data from both separate effects and integral tests (regrettably, most test facilities tend to simulate 1-D rather than 3-D reality). A general conclusion from ~30 International Standard Problems is that general purpose codes usually achieve reasonable qualitative **pre**dictions, followed by various degrees of quantitative agreements between final calculation and the test data.

Code validation (of general-purpose codes) can eliminate gross violations of Laws of Nature and simple coding errors. It does **not** prove that the code can reproduce Nature exactly; often some small detail in systems or during the course of events has a major influence which is difficult to capture, yet overall qualitative agreement is reachable. Code user experience & competence is a major factor influencing the credibility of predictions. One should **not** expect too much quantitative accuracy from general purpose codes. The required accuracy is case dependent; and insights from ISPs along with other experience are necessary to determine its acceptable level. Likewise, physically detailed codes are not always “better” than codes with more simple physical models. A plant analysis may be sensitive to, e.g., trip values, set points or bifurcating physical phenomena, thus rendering non-unique results. Real plants may exhibit a complex behaviour and this must be respected: the proper way is always performing sensitivity analyses.

Special purpose codes, e.g. 3D computational fluid dynamics (CFD) code, are potentially better suited to employ “correct” physics. Their validation, however, can be (and often has been) done in much more stringent fashion than it is possible for general purpose accident analysis tools. You have to know your tools!

The application of present state-of-the-art general-purpose thermal-hydraulics codes to VVER transients and accidents is not a major challenge, since all the essential physics is usually there. There are some special items, such as DNB correlations for the triangular fuel rod lattice, or capability to treat the hexagonal core structure in reactor dynamics code, that may be missing from off-the-shelf codes. In addition, there are some features such as hot leg loop seals and the horizontal steam generator that may require substantial modelling effort (in terms of nodalisation structure) or modifications of specific models. Severe accident analysis is usually comparable to accident analyses only before the beginning of core degradation. Analyses of degrade core phase and beyond encounter numerous difficulties which are basically independent from code or reactor type.

In conclusion, codes are valuable analytical tools; however, proper understanding of the results depends on an analyst. Independent of the application, code user should be competent, cognisant and experienced. The ”western codes” (with at most minor modifications) are very useful for safety analysis (transients, DBA and BDBA); however their application to severe accidents can be more troublesome. System code results are never “exact”, but more likely qualitatively correct. Moderate conservatism may be an efficient approach in terms of required analysis effort and uncertainty treatment.

Conservative Versus Best-Estimate Safety Analysis (David Bessette)

A widespread practice in nuclear safety analysis concerns the use of conservative assumptions in analysing transients and accidents. The conservatisms consist of assumptions in initial and boundary conditions, equipment availability, limiting single failure, human interventions, and so on. This approach was taken in many cases such as siting criteria and loss of coolant accidents. The results of the analysis are compared against a limiting value. If it falls within then the plant is considered safe. This forms the design basis envelope. Appendix K to 10 CFR 50 is an example. It specifies a list of conservatisms that must be used. Conservative prescriptive rules were developed during times when there were large uncertainties in phenomena and processes expected to occur during accidents.

In the intervening years since these rules were formulated, a considerable amount of safety research has been performed and a great deal has been learned. This allows more realistic, or best-estimate analyses to be conducted and uncertainties to be quantified. In doing so, much is learned about plant design margin, the relative importance of various phenomena, and the expected plant behaviour. This can lead to improved plant operation and more efficient allocation of resources on the part of the utility and the regulator. From the operator’s perspective, knowing realistically how the plant behaves is better than a view shaped by conservative analysis.

The large break loss-of-coolant accident is an example. From the perspective of peak cladding temperature, a flat power profile is desirable since this will reduce the peak initial stored energy and decay heat. From the perspective of vessel embrittlement and pressurised thermal shock, however, a peaked profile is preferred. A conservative analysis approach does not yield the optimum solution to such a “conflicting” problem.

Prescriptive vs. Non-Prescriptive Regulations (Ray Ashley)

Prescriptive regulations do not leave the licensee with flexibility but gives a rather clear “recipe” that allows to evaluate beforehand submitted analyses. The consistency of the regulator's decisions are ensured by his own “prescriptiveness”.

For the non-prescriptive (or less prescriptive) regulators, there is a need for well structured internal rules or assessment guidelines to ensure consistency. As this gives large flexibility to the licensee, discussion between regulators and licensees are needed to avoid unacceptable divergence. Such discussions are facilitated when the credibility of the regulator is well established, for instance, through international recognition.

Periodic Reassessment (Ray Ashley)

The need for regular reassessments seems to be commonly agreed. Although differences appear as for the periodicity to adopt, 10 years seems appropriate. To be noted is the expectation of Bulgaria to have it each 5 years and the license renewal of Canada each 2 years at most.

(Note: the list of subjects for reassessments could be further developed in a future workshop when more countries will have to face the problem)

Streamlining the user approach (Ray Ashley)

It was acknowledged that a good knowledge of the specific plat designs was needed to perform and to assess safety analyses. Recommendations were presented to avoid input deck preparation or assessment purely based on generic features, and in a more general way, to reduce the user effect in licensing calculations results, based on a recent international survey.

Safety analysis review guides (Fabio De Pasquale)

Based on the requirements published in AECB document C-6, and the experience gained in the review of accident analysis submissions, a set of guidelines was developed to facilitate the review of Large- and Small-Break Loss of Coolant Accidents (LOCA). The purpose of the guides was to assist in and standardise the review of analyses that have been submitted in support of an application to commence or continue operation of a CANDU nuclear power plant. An important objective of this project was also to make the guides sufficiently general that they would be useful regardless of the specific methodology or approach taken by the licensees in meeting the basic safety objectives and fundamental requirements that are stated in the AECB regulatory documents.

The approach taken was to develop a set of higher level criteria that the reviewer must confirm are met by the analysis. The criteria were developed in a hierarchical, top-down manner. For each criterion, the most relevant information was provided. This included:

- A concise statement of the objective,
- A list of key parameters,
- A statement of the success criterion (stated mathematically where possible and appropriate),
- A list of the key analysis tasks,
- An explanation of the variables that significantly affect the key parameters,
- A set of guidelines to assist the reviewer in confirming that the success criterion is met.

The guidelines are aimed at reviewers who have limited experience in the review of CANDU safety analyses, but who have a reasonably good understanding of the fundamentals of reactor safety. The guidelines are also aimed at reviewers who may have significant experience in the review of CANDU reactors, but who may not be familiar with all of the issues related to Large- and Small-Break LOCAs.

Approval for the use of codes (Ray Ashley)

Although the assessment processes of the codes themselves appear similar amongst the countries, namely based on the validation matrices, the kinds of approval are quite different. They range from the certificate (or "passport") delivered by Gosatomnadzor in Russia to the restricted approval for a specific application (which can be repeated case by case) by AVN in Belgium. In depth analyses of a code are sometimes needed to assess the transposition of its validity - established on scaled facilities - to the reactor case. The importance of a reliable maintenance team was stressed and recommendations were presented to reach the objective.

Applicability of results and basic requirements (Mike Modro)

Deterministic safety analyses form the basis for a licensing of nuclear power plants but are also applied to validation of emergency operating instructions (EOI), validation of plant simulators, support of accident management and emergency planning, in support of PSA analyses, and in support for variety of engineering analyses needed for plant operation.

The issue of licensing analyses was discussed extensively in the other sections. However, it is worth to mention that there are two basic approaches in the licensing analyses: the evaluation method (conservative) and the best-estimate method. The evaluation method employs conservative assumptions are embedded into the analytical models. Also, conservative assumptions are used for initial and boundary conditions in the analyses. Alternative best-estimate approaches are based on models of phenomena and systems that are as realistic as possible but for the licensing approval uncertainties of the analyses need to be quantified. In the non-licensing applications mostly the best-estimate methods are used, sometimes supplemented with conservative assumptions where large uncertainties are expected such as some of the PSA analyses. For evaluation of emergency procedures or instructions best-estimate methods are also used but mainly in the context of bounding analyses. However, for design or engineering related analyses, usually a best-estimate approach is applied.

Independent of what application or what method is used, a confidence in the results of the analysis must be generated. The regulator, through review of the analysis and their bases, verifies and assures the confidence in plant safety performance as required by the law. Essentially, in this process three basic questions must be addressed: (1) is the code/method used an acceptable analysis tool for the application?; (2) are the calculated results reasonable and logical?; and (3), are the documents supporting the validity of the analysis in order? The review focuses on the verification that all processes and phenomena important to safety are identified, that the models and numerics are adequate, that the experimental database exists and that validation of codes, methods and input models was performed.

Application of Western Codes to VVER Designs (Iván Tóth, Cs. Györi and I. Trosztel)

Construction of the first VVER units began in Hungary at the end of the 70-ies. The safety analysis report of the Paks NPP was supplied by the Soviet designers and contained mostly generic VVER-440/213 analyses. With our today's knowledge it can be criticised that, e.g., the list of initiating events was not complete, documentation of the results was unsatisfactory, the length of the transient times did not assure verification of safe end-conditions, the conservatisms applied could not be verified.

The computer codes available at that time in Hungary did not allow for an independent assessment of plant transients. It was only in the early 80-ies that the first thermal-hydraulic system codes, e.g. RELAP4/mod6 became available which, along with the construction of the first integral-type test facility (PMK) for VVER-440 reactors in Hungary, gave an important impetus to assessment capabilities. Code validation work based on PMK test results initiated an increase of confidence in calculations performed for the Paks NPP. The aim of these calculations was first of all to acquire deeper knowledge of plant behaviour and, consequently, the best estimate approach was applied.

An important amount of licensing-type calculations was performed in the early 90-ies in the framework of the AGNES project, the aim being the complete reassessment of the safety of the Paks NPP. Best estimate codes were applied with a methodology that was state-of-the-art at that time in European countries. With the lack of a proper legislation (no established criteria or guidelines) all these elements had to be defined for the project. Later on, the requirements applied in the AGNES project served as a basis for the authority, when stating the licensing requirements for the Periodical Safety Review of the Paks units and also the new legislation, came into force last year, drew much on its basic ideas.

While most transients in VVER plants can be handled without restrictions by applying the bounding approach, some of them (e.g. LBLOCA) might need uncertainty methods to satisfy acceptance criteria. As an example the approach applied for Paks LBLOCA analyses is presented. The first step was to apply the bounding approach, where a large number of conservative assumptions were taken in order to cover largely unknown code and model uncertainties. Obviously, this method calls for an adequate margin to be maintained with respect to acceptance criteria. In a second step a systematic revision and reduction of pessimistic assumptions was carried out, but the final solution is the application of uncertainty methods.

Present efforts concentrate on the application of the uncertainty analysis for selected accidents. In this context the proposed OECD/CSNI activity to revisit an earlier LBLOCA ISP including uncertainty analysis is of great interest. For VVER-type reactors identification and ranking of the important phenomena should be carried out, followed by a quantitative assessment of uncertainties. A major problem is the lack of VVER-specific test data for LBLOCA. At present, the GRS uncertainty methodology is applied in Hungary to a SBLOCA test performed in the PMK facility.

Joint IAEA/NEA Regional Workshop on
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PROSPECTUS

TITLE	Regional Workshop on Regulatory Review of Plant Safety Analysis (Under IAEA TC project RER/9/052 and OECD NEA sponsorship)
PLACE:	Bratislava, the Slovak Republic
DATE:	6 to 10 July, 1998
DEADLINE FOR NOMINATIONS:	To be supplied by TC
ORGANISERS:	International Atomic Energy Agency, OECD Nuclear Energy Agency and Slovakian Nuclear Regulatory Authority
TARGET	This workshop is designed for the Regulatory Bodies and other organisations COUNTRIES: of CEEC and FSU countries (10 in all) involved in the regulatory review of NPP safety analysis.
PARTICIPATION:	The workshop is open for (to be agreed, possibly 30) participants
LANGUAGE:	English
PURPOSE OF THE WORKSHOP: assist	It is necessary world-wide to increase the level of safety culture and strengthen regulatory authorities in their duties and responsibilities. In particular, it is necessary to regulatory authorities to meet their assigned tasks when reviewing the safety analysis of a nuclear power plant.
regulatory	This involves sharing experiences and opinions on performing and reviewing deterministic accident analysis for nuclear power plants. The workshop will cover selected legal, framework and technical aspects of the subject. At the beginning of the workshop, the participants will be informed about approaches and practices in their respective countries. The rest of the workshop will be subdivided into several modules covering the most important aspects of the accident analysis. Each module will consist of presentations by invited expert or by workshop participants and discussion of all participants. At the end, the participants will agree upon main conclusion of the workshop and suggestions of further activities.
PARTICIPANTS' QUALIFICATIONS:	The participants should be professional staff members of the nuclear safety regulatory body or of organisations performing work on their behalf. Workshop participants should have responsibility for reviewing safety analyses and be involved in making regulatory decisions.
NATURE OF THE WORKSHOP:	The workshop will consist of presentations and discussions in working groups. All participating countries should be prepared to present their national experience/approach during the workshop. Specific issues will be discussed and experts' opinion will be formulated in working groups.

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SYLLABUS

DAY 1 (Monday afternoon)

MODULE 1 INTRODUCTION TO WORKSHOP

1. Introduction/background (1 hr)
    - Overview (outline) of the workshop
    - NEA/ IAEA remarks on safety analysis

MODULE 2 LEGAL AND REGULATORY REQUIREMENTS AND PRACTICES

2.1 National Presentations on Regulatory Requirements and Approaches in Reviewing and Assessing NPP Safety Analysis

Each country represented {invited lecturers and workshop participants} to present national practice and experience (<u>not longer than 25 minutes</u>) covering major elements of the following topics:
- legal requirements
- availability of methodological guidance
- institutional arrangements
- required QA including independent review or confirmatory analysis
- use of conservative versus best-estimate approach
- major safety related applications of accident analysis at present, including the relationship with plant modernisation programmes

The time assigned for this module is Monday afternoon and Tuesday morning. The presentations will be made by the invited countries from outside the region as well as by the participants. After every three or four presentations there will be a short moderated overview/wrap-up session. Time permitting, after the presentations there may be Q&A session and/or panel discussion.

DAY 2 (Tuesday morning)

2.2 Continuation of National Presentations on Regulatory Requirements and Approaches in Reviewing and Assessing NPP Safety Analysis.

MODULE 3: TECHNICAL ASPECTS OF REVIEW AND ASSESSMENT

DAY 2 (Tuesday afternoon)

3.1 Initiating events and acceptance criteria (1 hr)
- selection of internal initiating events
- considerations of external hazards
- selection of events beyond the original plant design
- interrelation with the PSA for selection of events
- high level acceptance criteria
- detailed acceptance criteria

3.2 Basic approaches and computer codes used in accident analysis; use of Western codes for other applications (1 hr)
- DBA, BDBA, SA etc. applications
- applicability of codes for other modes of plant operations
- elements of conservative and best-estimate approaches
- requirements for code validation, determination of applicability
- evaluation of code uncertainties
- code certification

DAY 3 (Wednesday, July 8)

3.3 Examples of accident analysis (by invited experts)
- Case study of a conservative approach for an internal event (2 hr)
- Case study of a best-estimate approach for an internal event (2 hr)
- Case study of application of Western codes to VVER designs (2 hr)

3.4 Discussion on case studies (1 hr)

MODULE 4: REGULATORY ASPECTS OF REVIEW AND ASSESSMENT

DAY 4 (Thursday, July 9)

4.1 Basic elements of QA in accident analysis and user effect (1.5 hr)
- requirements on QA of the organisation performing analysis
- adequate representation of the plant status in the input data
- adequate recording of analysis, possibility for inspection
- requirements for Additional Information
- users effects issues

Applicability of results and basic requirements for each area of application (1 hr)
- design analysis
- licensing analysis
- validation of EOP and of plant simulators
- support for accident management and emergency planning
- PSA supporting analysis

4.3 Practical guidance for regulators how to review accident analysis (1 hr)
- applicable guidelines (Standard Review Plan or similar)
- relation to Technical Specification
- elements needed for review and for accident analysis approval
- documents required for issuance of a corresponding license
- internal QA procedures for review of accident analysis
- applicability of “generic” plant analysis
- practicability of Safety Evaluation Reports

4.4 Regulatory approval of safety analysis (1 hr)

MODULE 5: WORKSHOP CONCLUSION

5.1 Formulation of conclusions and recommendations of the workshop (1.5 hr)

DAY 5 (Friday, July 10)

5.2 Workshop Closure (3 hr)

- discussion of miscellaneous topics:
  - periodic safety reviews
  - release of the information to the public
- discussion of conclusions and suggestions
- workshop evaluation
- potential follow-up activities (reports, training on selected issues, etc.)
- concluding remarks

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Summary of results of the follow up Questionnaire

Answers from 8 participants: Gabriel Petrescu Juraj Klepac, Inna Ougoleva, Nadejda Rijova, Evaldas Bubelis, Sandor Szirmai, Lubica Kubisova, Milos Kyncl. The scores represent either number of answers (questions 1-5) or an arithmetic average with the range of answers given in parenthesis.

1. very interesting / useful: **5**, somewhat interesting / useful: **3**
2. Would you recommend to include similar National Presentations in the future?
   6 [YES], **1** [NO], **1** [NO OPINION]
3. Based on the needs of your organisation, the importance of the topics was:
   Initiating events and acceptance criteria...............................8. [H]
   Basic approaches / computer codes.......................................8 [H]
   Examples of accident analysis...6 [H]; 2 [L]
   Basic elements of QA in accident analysis and user effect....7 [H]; 1 [L]
   Applicability of results and basic requirements.....................7 [H], 1 [L]
   Practical guidance for regulators...5 [H]; 3 [L]
   Regulatory approval of safety analysis.................................6 [H]; 2 [L]
4. The time assigned for discussions was: sufficient (5) somewhat sufficient (3)
5. The format/structure of the Workshop was adequate and they liked it (7), one would modify

6. Importance / usefulness of issues: [0 - 5], where “0” means “NO”, and “5” means “VERY”

(a) Frequency of periodic safety reviews,	2.86	[1-4]
(a1) How often should be a SAR (or FSAR) updated?	2.71	[1-4]
(a2) Are there objective criteria for frequency of these updates?	3	[1-4]
(a3) On what basis the frequency is, or should be determined?	3.43	[3-5]

(b) Approaches to safety evaluations:	4. 37	[4-5]
(b1) prescriptive / non-prescriptive regulations	4.62	[3-5]
(b2) deterministic / probabilistic evaluation	4.5	[4 -5]
(c) Relationship between the regulators and the industry	3.5	[3-5]
(d) Validations of codes	4.75	[4-5]
(e) Applicability of codes to “real plants”	4.87	[4-5]
(f) Applicability of the Western codes to Russian designs	4.37	[0,5,5,5,5,5,5,5]
(g) Elements of an “overall” safety analysis review process:	4	[2-5]
(g1) structure of / responsibility within a regulatory body	3	[0-5]
(g2) flow of documents during the review process,	3.37	[1-5]
(g3) requirements for / applicability of codes,	4.6	[4-5]
(g4) requirements for confirmatory analysis,	4.6	[4-5]
(g5) criteria used for approval of an analysis,	4.37	[3-5]
(g6) decision-making process leading to the final approval	4.12	[3-5]

(h) Case studies of:

(h1) details of analytical tools	3.7	[3-5]
(h2) details of a specific analysis	3.8	[3-5]
(h3) details of review of specific analysis	4.4	[3-5]

7. Just one comment regarding point 6:
   “all issues are important, but the usefulness depends on current issues being of high priority”

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Results of the Questionnaire

1. From your perspective, the National Presentations were:
   - very interesting / useful ❒ x,x,x,x,x
   - somewhat interesting / useful ❒ x,x,x
   - of little value / interest ❒
   - any comments?

2. Would you recommend to include similar National Presentations in the future?
   [YES] x,x,x,x,x [NO] x [NO OPINION] x

3. Based on the needs of your organisation, the importance of the topics was:
   Initiating events and acceptance criteria................ [H] x,x,x,x,x,x,x,x
   Basic approaches / computer codes......................... [H] x,x,x,x,x,x,x,x
   Examples of accident analysis................................. [H] x,x,x.x,x,x [L] x,x
   Basic elements of QA in accident analysis and user effect..[H] x,x,x,x,x,x,x [L] x
   Applicability of results and basic requirements........... [H] x,x,x,x,x,x,x [L] x
   Practical guidance for regulators................................ [H] x,x,x,x,x [L] x,x,x
   Regulatory approval of safety analysis........................ [H] x,x,x,x,x,x [L] x,x

4. The time assigned for discussions was:
   - sufficient ❒ x,x,x,x,x
   - somewhat sufficient ❒ x,x,x
   - not sufficient ❒

5. The format/structure of the Workshop was:
   - adequate and you liked it ❒ x,x,x,x,x,x,x
   - adequate, but you would modify it ❒ x
   - not adequate and should be changed ❒

6. Based on the regulatory issues you (your organisation) is dealing with, on a scale [0 - 5] how would you estimate the importance / usefulness of the following topics:
(NOTE: “0” means “NOT IMPORTANT”, and “5” means “VERY IMPORTANT”)

(a) Frequency of periodic safety reviews, 3,1,3,3,3,4,3,x
(a1) How often should be a SAR (or FSAR) updated? 2,1,3,2,4,4,3,x
(a2) Are there objective criteria for frequency of these updates? 4,1,3,3,3,4,3,x
(a3) On what basis the frequency is, or should be determined? 3,3,3,3,5,4,3,x

(b) Approaches to safety evaluations: 4,5,4,4,4,5,4,5
(b1) prescriptive / non-prescriptive regulations 4,x,4,3,4,4,4,5
(b2) deterministic / probabilistic evaluation 4,5,4,4,4,5,5,5

(c) Relationship between the regulators and the industry 3,3,4,3,5,4,3,3

(d) Validations of codes 4,5,5,5,5,5,4,5

(e) Applicability of codes to “real plants” 4,5,5,5,5,5,5,5

(f) Applicability of the Western codes to Russian designs 0,5,5,5,5,5,5,5

(g) Elements of an “overall” safety analysis review process: 4,2,5,4,4,5,4,4

(g1) structure of / responsibility within a regulatory body 3,0,5,4,3,4,2,3
(g2) flow of documents during the review process, 3,1,5,4,3,4,3,4
(g3) requirements for / applicability of codes, 4,5,5,4,5,5,4,5
(g4) requirements for confirmatory analysis, 4,5,5,4,5,5,4,5
(g5) criteria used for approval of an analysis, 4,3,5,4,4,5,5,5
(g6) decision-making process leading to the final approval 4,3,5,4,4,4,5,4

(h) Case studies of:

(h1) details of analytical tools 4,x,4,3,3,4,3,5
(h2) details of a specific analysis 4,x,4,3,3,5,3,5
(h3) details of review of specific analysis 4,x,5,5,3,5,4,5

7. Please, provide any comments you may have relevant to this questionnaire, e.g., confusing/wrong questions (which ones?), missing questions (suggest one), poor/wrong selection of given choices (examples/suggestions), etc.

No comments except for one regarding point 6:
“all issues are important, but the usefulness depends on current issues being of high priority”

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Follow up Questionnaire

1. From your perspective, the National Presentations were:
- very interesting / useful ❒
- somewhat interesting / useful ❒
- of little value / interest ❒
- any comments?

2. Would you recommend to include similar National Presentations in the future?
[YES] [NO] [NO OPINION]

3. Based on the needs of your organisation, the importance of the topics was:
Initiating events and acceptance criteria.................................. [HIGH] [LOW]
Basic approaches / computer codes... [HIGH] [LOW]
Examples of accident analysis... [HIGH] [LOW]
Basic elements of QA in accident analysis and user effect...... [HIGH] [LOW]
Applicability of results and basic requirements....................... [HIGH] [LOW]
Practical guidance for regulators... [HIGH] [LOW]
Regulatory approval of safety analysis.................................... [HIGH] [LOW]

4. The time assigned for discussions was:
- sufficient ❒
- somewhat sufficient ❒
- not sufficient ❒

5. The format/structure of the Workshop was:
- adequate and you liked it ❒
- adequate, but you would modify it ❒
- not adequate and should be changed ❒

6. Based on the regulatory issues you (your organisation) is dealing with, on a scale [0 - 5] how would you estimate the importance / usefulness of the following topics:
(NOTE: “0” means “NOT IMPORTANT”, and “5” means “VERY IMPORTANT”)

(a) Frequency of periodic safety reviews,	0	1	2	3	4	5
(a1) How often should be a SAR (or FSAR) updated?	0	1	2	3	4	5
(a2) Are there objective criteria for frequency of these updates?	0	1	2	3	4	5
(a3) On what basis the frequency is, or should be determined?	0	1	2	3	4	5
(b) Approaches to safety evaluations:	0	1	2	3	4	5
(b1) prescriptive / non-prescriptive regulations	0	1	2	3	4	5
(b2) deterministic / probabilistic evaluation	0	1	2	3	4	5
(c) Relationship between the regulators and the industry	0	1	2	3	4	5
(d) Validations of codes	0	1	2	3	4	5
(e) Applicability of codes to “real plants”	0	1	2	3	4	5
(f) Applicability of the Western codes to Russian designs	0	1	2	3	4	5

(g) Elements of an "overall" safety analysis review process:	0	1	2	3	4	5
(g1) structure of / responsibility within a regulatory body	0	1	2	3	4	5
(g2) flow of documents during the review process,	0	1	2	3	4	5
(g3) requirements for / applicability of codes,	0	1	2	3	4	5
(g4) requirements for confirmatory analysis,	0	1	2	3	4	5
(g5) criteria used for approval of an analysis,	0	1	2	3	4	5
(g6) decision-making process leading to the final approval	0	1	2	3	4	5

(h) Case studies of:

(h1) details of analytical tools	0	1	2	3	4	5
(h2) details of a specific analysis	0	1	2	3	4	5
(h3) details of review of specific analysis	0	1	2	3	4	5

7. Please, provide any comments you may have relevant to this questionnaire, e.g., confusing/wrong questions (which ones?), missing questions (suggest one), poor/wrong selection of given choices (examples/suggestions), etc.