# Regulatory Perspectives on Analytical

# Codes and Methods for Advanced Reactors

W. KLEIN-HEßLING

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH

Cologne, Germany

Email: walter.klein-hessling@grs.de

T. DRZEWIEKI

US Nuclear Regulatory Commission NRR/DANU/UART

Washington, D.C., USA

G. GRASSO

Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA)

Bologna, Italy

C. HERER

Institut de Radioprotection et de Sûreté Nucléaire (IRSN)

Fontenay-aux-Roses Cedex, France

N. KHRENNIKOV, V. PIVOVAROV

Scientific and Engineering Centre for Nuclear Radiation Safety (SEC NRS)

Moscow, Russian Federation

D. PLUMMER

Office for Nuclear Regulation

Bootle, United Kingdom

**Abstract**

Analytical codes and methods are used extensively in the design and safety analysis of nuclear reactors. Regulatory agencies establish requirements and/or expectations on the nuclear power plant designer or licensee for the development and use of analytical codes and methods in order to ensure the quality, credibility, and confidence in the analyses produced by the analytical codes and methods. In addition, regulatory agencies have used analytical codes and methods to perform confirmatory analyses as part of due diligence during a regulatory review. The Task Group on Analytical Codes and Methods (TGACM) of the OECD-NEA Working Group on the Safety of Advanced Reactors (WGSAR) has performed a review to (1) identify and clarify the requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plants, and (2) identify best practices for the use of confirmatory analyses by regulatory agencies. In the paper first results of this on-going work are presented, based on the responses to a survey from Canada, France, Germany, Italy, Russia, UK and USA. The differences in procedures and expectations on regulatory approval of codes and methods, quality assurance program and handling of possible bugs and errors are discussed. The second part of the survey is related to confirmatory analysis. Because the objective of these confirmatory analysis is mainly linked to support the regulator, there are differences in the required capabilities and expectations to codes used for the design and optimization of advanced reactors. Independent from the claimed inherent safety capabilities of the reactor concept, a simulation of severe accident phenomena is generally expected by the regulatory agencies. In conclusion, the regulatory expectations related to codes and methods used for advanced reactors should be considered in the development of these codes. Comparing code capabilities with safety relevant phenomena, the review provides information on further code development needs.

## INTRODUCTION

Analytical codes and methods are used extensively in the design and safety analysis of nuclear reactors. These codes and methods are commonly used to analyse the response of a complex engineering system to postulated events with potentially severe health, financial, and environmental implications. Guidance to perform deterministic analyses is documented in an IAEA report [1] where use of computer codes for deterministic safety analysis is presented. Regulatory agencies establish requirements and/or expectations on the nuclear power plant designer or licensee for the development and use of analytical codes and methods in order to ensure the quality, credibility, and confidence in the analyses produced by the analytical codes and methods. In addition, regulatory agencies have used analytical codes and methods to perform confirmatory analyses as part of due diligence during a regulatory review. These confirmatory analyses frequently require specialized expertise and can be a resource intensive aspect of a regulatory review.

The analytical codes and methods applied collectively have to account for all safety relevant phenomena, which may be particularly challenging for the advanced reactor concepts of Generation IV with new and only partially developed safety concepts. Independent from inherent safety capability, codes and methods are needed to simulate all types of transients and phenomena e.g., core degradation phenomena for these reactor concepts.

A technical report to further investigate regulatory expectations for analytical codes and methods has been developed by the OECD-NEA Working Group on the Safety of Advanced Reactors (WGSAR) [2]. This paper summarises some key findings of the detailed report. A questionnaire was created to gather information from the participants. The presented results are based on the answers to this questionnaire from Canada, France, Germany, Italy, Russian Federation, United Kingdom, and United States of America.

## Regulatory expectations

The scope of the WGSAR report covers the regulatory requirements and expectations set out by regulatory authorities for reactor designers and operators that use analytical codes and methods for design and analysis, the use of analytical codes and methods by regulatory authorities and/or their technical support organizations as a confirmatory analysis tool, and the existing analytical codes and methods. The regulatory requirements describe general requirements (e.g., law/regulations and enforcement), regulatory oversight and guidance, code assessment requirements and/or expectations, and user qualifications. The confirmatory analyses discussion describes the reasons for performing confirmatory analyses, expectations for the analytical tools, and quality assurance associated with those analytical tools.

### Regulatory requirements

This part describes the requirements and expectations on the nuclear power plant designer or licensee for the development and use of analytical codes and methods in order to ensure the quality, credibility, and confidence in the analyses produced by the analytical codes and methods.

In general, requirements and expectations on the development and use of analytical codes and methods are derived from high level principles associated with the applications of the codes to support the safety case and safety analysis. Most of these requirements are based on light water and heavy water reactors (LWRs, HWRs). These are for example REGDOC 2.4.1 “Deterministic Safety Analysis” and REGDOC 2.5.2 “Design of Reactor Facilities: Nuclear Power Plants” in Canada, the German SiAnf, the regulatory documents of the top level NP-001-15 and NP-082-07 in Russia, the ONR Safety Assessment Principles for Nuclear Facilities - AV.1 to AV.8 in the UK and several requirements from title 10 of the Code of Federal Regulations (CFR) in the USA (e.g., 10 CFR 50.34, 10 CFR 50.43(e), 10 CFR 50.46, Appendix B to 10 CFR 50, Appendix K to 10 CFR 50, 10 CFR 52.47, 10 CFR 52.79). In all these documents there is a general requirement that the analytical codes and methods used in the safety analysis and in systems important to safety must be verified and validated. In certain countries more detailed requirements are provided for analysis of the reactor core. Examples are the German KTA 3101.1 and 3101.2.

In addition, guidance (typically non-binding) may be provided, for example in France the “ASN Guide No. 28 Qualification of scientific computing tools used in the nuclear safety case - 1st barrier”, and in the UK NS-TAST-GD-042 “Validation of Computer Codes and Calculation Methods”.

In the USA requirements on QA procedures are described in 10 CFR 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Pre-processing Plants”. These quality assurance programs include, among other items, independent verification of the analyses, planned periodic audits of the quality assurance program, and a corrective action program to ensure that any deficiencies are promptly identified and corrected.

In general, the participating countries are not developing specific guidance for modelling and simulation of advanced reactors, although some reviews of the applicability of the existing guidance have been completed or are planned. Where such reviews have been completed, it is considered that existing guidance is sufficiently generic to be suitable for application to advanced reactors.

The regulators expect the licensee to provide evidence of sufficient validation and verification for codes and methods used to support a safety submission. This demonstration is greatly simplified for codes and methods already approved (e.g., in USA and France).

For those countries that approve analytical codes and methods, this involves the submission of a report to the regulator, with the approval being linked to a particular usage case or range of applications. The approval processes can involve some degree of iteration, either by a staged process or regulatory feedback and requests for additional justification. The process also typically results in specifications (limited range) on the use of the code or method. One of the key strengths of this approach is the efficiency, as the code can be approved in advance of a licensing application and approved codes and methods can be re-used in the future and can potentially be applied across a fleet of plants. One of the key weaknesses of this approach is the difficulty in updating or modifying approved methodologies. Licensees may be reluctant to implement updates and improvements to approved codes due to the perceived risks associated with re-opening a regulatory review.

There is general consensus that bugs and errors in analytical codes and methods are expected to be managed using a combination of code quality assurance procedures, validation and verification, and in some cases, sensitivity analyses and/or confirmatory analyses. Comparison of code results against experimental data is seen as an important tool in code validation and elimination of defects in the code. The USA has mandatory reporting requirements for code defects that could affect safety.

In the case that analytical codes and methods are used in the design and safety analysis of advanced reactors, which have not been developed/maintained in accordance with a quality assurance program appropriate for the intended application, regulators expect that equivalence with current good practice is demonstrated to support the validation, verification and quality assurance of legacy analytical codes and methods. Only the USA has a specific process for licensees wishing to adopt a legacy code into their QA arrangements and reconciling quality assurance gaps, known as ‘commercial-grade dedication’.

Regarding code validation, there is a general preference for comparisons with experimental data and operational measurements. Regarding code-to-code comparisons, there is some disparity in the answers regarding the acceptability for assessing uncertainty. For example, Canada and the USA stated that code-to-code comparisons are not used to assess uncertainty. France and Russia stated that expert judgment is used in some instances. Germany and the UK responded that specific methods are not provided and that they are addressed on a case-by-case basis.

For the characterisation of the estimated uncertainty of the analytical code or method, there is general consensus, that statistical analyses to obtain predictions at a high confidence levels are acceptable (e.g., Wilks, 2-sigma) and preferred when sufficient data is available. There is also general recognition that sufficient data may not exist to obtain statistical values with high confidence levels and a conservative approach (e.g., maximum deviation) can be used. There is consensus that the experimental uncertainty needs to be addressed when assessing the uncertainty in the code being tested. However, there is some disparity regarding the method used to incorporate this uncertainty.

To consider the estimated code uncertainty in the safety analysis along with other sources of error, there is general consensus, that uncertainty needs to be accounted for with a conservative bias used in support of design basis analysis and that this is handled on a case specific basis. France and the USA discussed uncertainty propagation approaches using sampling techniques to determine a suitably conservative value (e.g., 95 percent confidence at the 95th percentile). The responses also indicated that conservative biasing of the inputs is also an acceptable practice.

The uncertainty assessment for passive systems is not conceptually different than for active systems although the range of parameters can be very different. However, in France the application of the uncertainty methodology needs to be appropriately justified for passive systems. Furthermore, there are ongoing activities within OECD and IAEA to address this topic. The responses of the UK and the USA indicate that differences in the phenomena or scale of the phenomena for passive systems may lead to different treatment of those systems in the safety analyses (e.g., chaotic behaviour and single-failure criteria).

In case a code certification is required, there is general consensus that a code approval/certificate needs to specify purpose and scope of the approval, range of applicability, and any applicable restrictions on the use of the code.

No general consensus was identified in the responses regarding methods used to identify knowledge gaps and phenomenological modelling requirements. Canada, the UK, and the USA indicated the use of expert judgement discussing a phenomenon identification and ranking process to identify potential gaps. Russia clarified that the main evidence for determining the adequacy of the code is contained in the code certification, and France clarified that the methods to determine knowledge gaps and modelling requirements are handled on a case-by-case basis.

Regarding user qualification, it is expected that code users are qualified. However, in Germany there is no formal request on the user itself. In Canada the personnel must be qualified and in France there are requirements inside the QA procedure. In Russia the user qualification is more on the organisation level. The qualification is part of the certification process. In the USA personnel qualification records are required to be maintained as part of the quality assurance program. The NRC does not impose specific criteria regarding the necessary credentials/training for the user of an analytical code/method, but reasonable expectations include demonstrated competence with the analytical code or method through experience or training.

The solutions concerning the consideration of uncertainties caused by user choices (e.g., selection of input deck parameters, models) is quite country specific. In Canada, France and USA it is expected, that the user choices are covered by the verification and validation process. Furthermore, certain sensitivity analyses are expected. Russia takes only the final result of safety case, with the selected choices. In case of doubts, the regulator can initiate confirmatory calculations or require additional calculations from the licensee. In the UK the development and qualification of reference input decks is expected. Variance from these should be controlled.

Another aspect that was not specifically covered in the questionnaire is the importance of qualification of coupled codes where indeed each code must be separately qualified but also specific challenges related to coupled codes must be considered, including user’s effects.

### Confirmatory analyses

Confirmatory analyses are typically studies performed by the reviewer repeating the calculations presented by the applicant. Regulatory authorities or their technical support organizations may also use confirmatory analyses to investigate sensitivities to analyse inputs or to examine accident sequences not presented in a license application. This is somewhat different to independent verification of deterministic safety analysis by the licensee (cf. Requirement 21 of IAEA General Safety Requirements (GSR) Part 4) that is provided by the applicant. The independent verification should consist of two main parts: an overall (qualitative) review focused on the quality and comprehensiveness of the safety analysis; and specific detailed reviews of important aspects of the analysis.

Confirmatory analyses also support understanding the results of the applicant’s analyses and provide additional confidence. Confirmatory analysis helps understanding the proposed design and assessing its compliance with regulations. Additionally, confirmatory calculations can also provide evaluations of the impact of uncertainties.

Confirmatory analyses are not always performed during the regulatory assessment; this is decided on a case-by-case basis. Potential reasons for undertaking confirmatory analysis include:

* To confirm key design values;
* To check the available margins when they are limited;
* To check the results when safety issues are important;
* To maintain competences in capabilities to perform calculations;
* To follow a graded approach and focus on important issues;
* To help understanding the design, integrated system performance, and analytical methods;
* To perform additional sensitivity studies;
* To assess approach when first-of-a kind design is proposed (or new for the reviewing staff).

Regulatory findings are based on the analyses and data submitted by the applicant and not upon confirmatory analyses. The process of request for additional information and answers from applicant generally provides supplementary information. It is also suggested to compare the results with similar cases previously reviewed and check for consistency using engineering judgment.

The use of different analytical codes and methods is sometimes required (depending on the country regulation) or necessary because the tools used by the applicant are not available for the reviewer. Although in the latter case, in some countries’ regulations it is required that the analytical methods be made available by the applicant to the reviewer. There are drawbacks and benefits to both approaches. Using the same tool provides confidence to the correct use of the tool (when results from applicant and confirmatory analysis are similar) and therefore can help to detect any flaws in the preparation or execution of the calculation (when discrepancies exist between applicant and reviewer). A side-by-side comparison is also possible when the same tools are used. However, this implies that the reviewer has a sufficient knowledge of the applicant’s tool. On the contrary, when the tools are different, similar results can provide a higher degree of confidence. However, when results are different, it may be challenging to identify the sources of discrepancies.

The approach to quality assurance (and by extension qualification of analytical codes and methods) is also different depending on a country’s regulations. On the one hand, for both the applicant and reviewer, the qualification of analytical tools is required. On the other hand, some countries required only qualification of the applicant’s tools, because they are the bases for safety demonstration, and there are no specific requirements for the confirmatory analyses’ tools. However, in the latter case, if discrepancies are detected with applicant’s results, this does not conclusively indicate an issue in the applicant’s analysis.

## Common positions

This section presents the common positions on the requirements and best practices applicable to nuclear power plant designers or licensee for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plants. These common positions were established by participating WGSAR members based on the questionnaire answers.

1. There is general consensus that a code approval/certificate shall specify purpose and scope of the approval, range of applicability, and any applicable restrictions on the use of the code.
2. Analytical codes and methods used in the design and safety analysis for a nuclear facility shall go through a process of verification and validation to establish the necessary confidence in the codes ability to determine safety margins.
3. Analytical codes and methods should be developed and maintained under a quality assurance program. This quality assurance program should include a comparison of code predictions against experimental data, verification of the implementation of numerical methods, and code lifecycle management.
4. Bugs and errors in analytical codes and methods should be managed using a combination of code quality assurance procedures, verification and validation, and in some cases, sensitivity analyses and/or confirmatory analyses. Comparison of code results against experimental data is seen as an important component in code validation and elimination of defects in the code.
5. For legacy codes that were not originally developed under a quality assurance program, the user of the code should demonstrate that the quality of code is commensurate with current good practice (i.e., verification and validation).
6. Code calculational uncertainty should be determined using statistical methods to determine the calculation error at high confidence levels (either determined by parametric (e.g., 2 sigma) or non-parametric (e.g., Wilks) methods) when sufficient data are available. There is general recognition that sufficient data may not exist to obtain statistical values with confidence levels and a conservative (e.g., maximum deviation) can be used.
7. Application of a code to a design calculation or safety analysis shall include appropriate conservatisms to accommodate uncertainties and unknown or non-modelled phenomena, and, when applicable, insufficient qualification of the code.
8. Uncertainty assessment for passive systems is conceptually similar to active systems although the range of parameters can be very different.
9. Code users or the organisations to which the users belong to shall be qualified. It is recognised that the used procedures are country specific.

Confirmatory analyses consist of cross calculations carried out by the reviewer on the same case, using the same or different tools. The common positions below were established by participating WGSAR members based on the questionnaire answers.

1. Use of analytical codes and methods by a regulatory authority or its technical support organization enhances the safety review by (1) increasing the regulatory authorities understanding of the plant design, integrated system performance, safety margins, sensitivities, and analytical methods, (2) allowing for an efficient assessment of accident scenarios not presented in the licensee safety case, and (3) supporting the analysis of the safety case.
2. Regulatory decisions are based upon analyses and data submitted by the applicant and not confirmatory analyses. Issues identified through confirmatory analyses should be brought to the attention of the design authority (i.e., plant designer or facility owner) and resolved through the safety review process.

## Evaluation of codes

In addition to regulatory expectations and methods related to confirmatory analyses, the survey also considered available simulation codes and methods. To evaluate the status of these codes, their capabilities are reflected against specific safety relevant phenomena of Gen-IV reactor concepts. Exemplary phenomena different to light-water reactors are listed below for each of the six Gen-IV reactor concepts proposed by Gen-IV International Forum (GIF):

One of the main specifics of the SFR which pose a threat to the safety of the reactor are associated with the properties of the sodium coolant – burning in air and violent interaction with water with the release of a large amount of hydrogen. Additionally, the coupling between reactor kinetics and thermal hydraulics has to be considered in detail, including changes of reactor core geometry. The safety-related specifics of a SFR are increased radiation swelling of the fuel and claddings, radiation embrittlement of steels, as well as a high specific power, which, in combination with a relatively low boiling point of sodium creates a threat to introduce large positive reactivity and in the case of deterioration of heat removal, it can lead to rapid fuel melting.

Examples related to a pebble type VHTR are the simulation of dust generation and abrasion core wall interface effects on bypass flow.

Regarding MSRs, salt performance as described by thermophysical properties has been identified as an important phenomenon to capture during normal operation and accident scenarios. Specifically, radiative heat transfer is significant at the elevated temperatures which leads to greater importance of salt optical properties. Consideration of phase change in the form of freezing is an issue that could occur due to localized cool spots (e.g., in a heat exchanger tube) or during over-cooling events.

The overall system thermal hydraulics of an LFR is very similar to that of an SFR, so that codes and methods developed for the latter can easily be accommodated for application to the former. However, in case of a postulated core degradation scenario, the relocation of the fuel, in case of its dispersal from the core, is not straightforward and depends on the specific thermal and hydraulic conditions, due to the similar densities of oxide fuels and coolant.

Stability calculations are essential to identify the operating and start-up conditions for SCWRs in view of the significant variation in density in the core, which could lead to dynamics instability. Another interesting SCWR characteristic is the strong coupling between thermal hydraulics and neutronics. Instability could lead to power fluctuations within the core.

The codes can be categorised into the type of code, the field of application regarding the plant state and scenario, and regarding the type of reactor concept. Regarding neutron kinetics the methods are point kinetics, 2D/3D modelling using Monte-Carlo-Methods or neutron diffusion methods. The thermal-hydraulics topic is distinguished between Lumped-Parameter (LP), sub-channel methods and CFD. Further criteria are related to mechanical codes and structure analysis and fuel performance. Additional categories are the capability to simulate fission product behaviour and transport and to simulate severe accident phenomena.

 Regarding safety relevant phenomena for the different Gen-IV concepts, specific needs on analytical codes and methods can be preliminarily derived. Partly these are applicable to the different Gen-IV concepts.

One of the complicating factors for advanced non-LWRs is that it may be necessary to have a tight coupling between several analysis codes because of the feedback between physical phenomena. Fast reactors for example, require a tight coupling between neutronics, thermal-hydraulics, and in some cases thermal-mechanics to account for rapid changes in power due to reactivity feedback that can occur with changes in temperature. Thermomechanical expansion of the core is an important phenomenon that must be accounted for because of its impact on neutron leakage. As mentioned earlier, coupled methodologies require specific qualification.

Analytical methods are generally needed to assess fuel performance under conditions of normal operation, including anticipated operational occurrences, and accident conditions. The analytical methods need sufficient modelling capabilities to capture fuel geometry, material properties, and physics sufficient to capture degradation mechanisms and failure modes. Fuel swelling, fission gas release, creep, stress/strain, chemical interactions with the reactor environment (e.g., coolant) and neighbouring materials (e.g., fuel/clad chemical interaction), and changes in thermal-physical properties with exposure (e.g., thermal-conductivity degradation) are generally expected to be phenomena addressed by fuel performance codes for solid fuel designs. Higher burnups anticipated for the Gen-IV concepts are expected to introduce specific modelling needs that must be reflected in fuel performance codes. Metal fuel is susceptible to the formation of low melting-point eutectics which could contribute to cladding failure under accident conditions. Fuel performance modelling for TRISO fuel is a major consideration for determining fuel failure fraction and contribution to accident source term. Unknowns such as kernel migration and silicon carbide morphology modelling and the impact on radionuclide release have been identified. For MSRs, salt chemistry changes associated with fission product building and the resulting impact on thermophysical properties has been identified as an important phenomenon that is unique to reactors utilizing fuel salts.

Most modern liquid-metal and molten salt reactors are “pool” as opposed to “loop” designs. That is, flow circulation and heat transfer take place within the reactor vessel rather than through an external flow loop. The reactor vessels often contain large regions in the plena where thermal stratification and internal recirculation can be important. LP type codes generally have difficulty simulating these processes because they lack the models and experimental support, in addition to detailed nodalisation required for fluid mixing. Regarding simulations inside reactor vessel codes can be distinguished between LP or CFD type codes. The LP type codes (e.g., ASTEC, ATHLET, CATHARE2, MELCOR, RELAP5) use often point-kinetic models. Most of the used CFD codes do not consider neutron kinetics, one exception is the SIMMER code. Examples of neutron kinetic codes are DYN3D, MCNP6, PARCS, SERPENT, TORT-TD. Partly these are coupled to thermal-hydraulic codes.

Most of thermal-hydraulic codes can handle steady-state conditions, transients and accident conditions. However, regarding fast transients such as 3D modelling of SFR core mechanics in transient conditions including very fast ones (e.g., driven by the pressure pulse caused by sodium vapour bubble collapse) knowledge gaps remain. Another example is the 3D modelling of transition from forced convection to natural convection for the pool-type SFR including fluid dynamics and thermal stratification and the simulation of systems such as the dynamic modelling of thermal electromagnetic pumps.

As long as scenarios (e.g., leading to core degradation) cannot be practically eliminated, the regulators consider these kinds of scenarios have to be investigated. Regarding severe accident simulation capability, few codes are available. These are for example ASTEC (for SFRs) and MELCOR (for SFRs, VHTRs and MSRs).

## Conclusions

In this paper the on-going work of the OECD-NEA Working Group on the Safety of Advanced Reactors (WGSAR) related to analysis codes and methods used in the design and safety analysis of Gen-IV plant concepts has been presented. The objective is to identify and clarify the requirements and best practices for such codes and methods, and to identify best practices for the use of confirmatory analyses by regulatory agencies. Based on the survey, WGSAR has expressed common positions underlining the need of verification and validation, the estimation of uncertainties and configuration management. However, differences between countries exist. Examples are certification of codes and expectations on user qualification. No general consensus was identified in the responses regarding methods used to identify knowledge gaps and phenomenological modelling requirements.

Furthermore, common positions related to confirmatory analysis are drawn. In fact, expectations on codes and methods are independent from their application on Gen-III or Gen-IV type plants. However, within the licensing process of Gen-IV simulation codes and methods must be available and these must be able to simulate the regarded safety relevant phenomena, which are partly specific to Gen-IV concepts.

References

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