A REGULATOR'S PERSPECTIVE ON THE STATE-OF-THE-ART IN NUCLEAR THERMAL-HYDRAULICS

Stephen M. Bajorek

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

Corresponding Author: Stephen.Bajorek@nrc.gov

ABSTRACT

An understanding of thermal-hydraulics, its basic phenomena, its application to nuclear power plants is vital to safe, as well as efficient design and operation. Thermal-hydraulics for two-phase flow and heat transfer with nuclear applications has been studied for roughly fifty years and the current body of knowledge is extensive. Yet, there remain safety issues and the licensing of new light-water reactor designs or new analysis methodologies is rarely simple. This is due to the state-of-the-art in reactor thermal-hydraulics as well as the perspective that the Regulatory, in this case the U.S. Nuclear Regulatory Commission, has towards that field of study.

This paper discusses the state-of-the-art in nuclear thermal-hydraulics, the Regulator's unique role and perspective, and a view of current challenges. The discussion is meant to point out how the Regulator's perspective has been shaped, and hopefully provide some guidance on fulfilling research needs for future applications that may need NRC review.

KEYWORDS

Thermal-hydraulics, Reactor Safety, Nuclear Regulation

1.0 INTRODUCTION

Knowledge of thermal-hydraulics has always been vital to the design, operation, and regulation of nuclear power plants. Since the opening of the Shippingport Atomic Power Station in 1957, numerous organizations from academia, private business, and government have worked to better understand the physical processes necessary for safe and economic nuclear power. The work has included a considerable amount experimentation, computational analysis and development of basic theory to explain results. Through these efforts, the nuclear industry now has a wealth of experimental data and sophisticated thermal-hydraulic systems analysis tools. Yet, certification of new and advanced light water reactors, and licensing new analysis methods remain a challenge. New safety issues as they are identified, are sometimes difficult to resolve. And, as the industry responds to economic pressures the necessary changes to plant operation must be understood before they are implemented.

This paper deals with the Regulator's role in thermal-hydraulic research and development, and in the Regulator's view of the state-of-the-art of thermal-hydraulic capabilities. To do this, first the "role of the Regulator" will be discussed and (hopefully) defined. Because the current state-of-the-art is partly a product of what has driven it in the past, some history is examined. Finally, current issues and programs will be discussed from the Regulator's perspective.

2.0 ROLE OF REGULATORY RESEARCH IN THE NUCLEAR INDUSTRY

The regulator has a unique role in technology development and research, and that role is unique by design. In 1946 the Atomic Energy Commission (AEC) was established in order to foster and control the peacetime development of atomic science. The congressional act that created the AEC transferred from military to civilian control of many nuclear laboratories that were instrumental in the initial understanding of nuclear science and safety. It also put the AEC in the politically difficult position of promoting what eventually grew into the nuclear power industry, but also of ensuring safety to the public. This dual role received increasing criticism during the 1960s and in 1974 the AEC was replaced by the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC). Several years later ERDA was combined with several other agencies associated with energy production into the present Department of Energy (DOE).

With the 1974 reorganization, role of the NRC was made clear. The NRC was to be independent and responsible for public health and safety. As part of its mission, the NRC is responsible for reactor licensing and renewal, licensing of radioactive materials, spent fuel management and safety and security. The DOE is responsible for a wide range of energy related areas, including nuclear energy related research, energy production, and waste disposal. While some areas appear to have overlap, the goal of the reorganization was to establish the DOE as a supporter of (nuclear) industry and the NRC as the independent regulator.

In reactor thermal-hydraulics, this division of responsibilities between the NRC and DOE (and other organizations) can become blurred. Both the NRC and DOE must understand the phenomena and basic physics of nuclear energy and its processes, and frequently must conduct or sponsor similar research to fulfill their respective missions. Both the NRC and DOE often rely on the same experts in academia and private industry for their skills, knowledge and unique capabilities. And, because both the NRC and DOE need talented engineers and scientists on their staffs, they are supportive of universities and educational programs in the nuclear sciences. However, in spite of much common ground, the NRC in order to maintain its independence as a regulator avoids thermal-hydraulic research unless it meets certain criteria.

The NRC conducts research primarily for the following reasons:

(a) To become better informed on an applicant's analysis and the acceptability of assumptions and methods applied.

(b) To confirm the safety margin or "robustness" of a new design or of a conventional design to a new scenario.
(c) To resolve emerging safety issues.
(d) To provide a basis for establishing a new regulation or changing and existing rule.

The obvious theme contained in each of these reasons is that the thermal-hydraulic research conducted by the NRC is safety-related. Economics, or improving the efficiency of plant operation is important and of interest to the NRC but is rarely the motivation to perform research unless it has a potential impact on safety.

It is not important to the NRC that research advances the state-of-the-art. Clearly, improved understanding of complex physical processes such as turbulence or development of more refined numerical methods benefit science. But, unless safety margins are involved this type of research is addressed by the National Science Foundation (NSF).

Therefore, the Regulator's view of thermal-hydraulics and on-going research is filtered by its influence on safety and the perception of safety margins. Support of the nuclear industry as it pertains to new and improved designs, and more economic operation is the responsibility of DOE, and advancement of science for the benefit of society is the responsibility of the NSF.

3.0 HISTORICAL PHASES OF REGULATORY THERMAL-HYDRAULICS

The current view on the state-of-the-art in thermal-hydraulics is partly the result of events that dictated the course of thermal-hydraulic research over the past 50 years. The motivation for the work can be summarized into four periods, during which the major objectives varied. These were the "pre-NRC" period where the nuclear industry was expanding rapidly, but still quite young. The "Regulatory era" where the rules were first imposed on industry and then revised, the "Advanced LWR period" where growth in conventional reactors stalled, but was accompanied by development of passively cooled LWRs, and finally with the present era in which interest in new reactors has lessened but there are significant advancements in thermal-hydraulic analysis because of high performance computing. This section considers each period.

<u>The "pre-NRC Period", < 1974</u>

During this early period, most of the effort was directed towards basic design and development of reactor systems. There were few experimental studies in large scale facilities, with the Semiscale tests [1,2] being the first that were relevant to safety. Considerable progress was made in understanding the basic principles of two-phase flow however, with the pioneering work of Zuber [3] and Wallis [4], Rohsenow [5] (and co-workers) being primary examples. Studies such as these identified many of the key features of two-phase flow and boiling heat transfer. They were instrumental in enabling analysts to account for non-equilibrium between gas and liquid fields, and highlighted the need for additional research.

Thermal-hydraulic systems codes at this time were in their infancy and limited by computers of the day. Development of systems codes began in the mid-60s with the FLASH [6] and RELAP [7] codes which assumed homogeneous equilibrium in the flow. System nodalizations were not very detailed.

The "Regulatory Era", 1974 - 1988

After extensive rulemaking hearings, 10CFR50.46 and Appendix K were formally published in January 1974. Because technology was limited and there few experimental studies available the rule and Appendix K restrictions were conservative. Clearly, from the Regulator's perspective the state-of-the-art was poor and the Appendix K restrictions on analysis methods were a reflection of that perception. A close examination of the Commissioner's Opinions [8] however, and a later study by the American Physical Society (APS) [9] on the recently published rule left no doubt the rule was probably conservative. However, the complexity of the Evaluation Models and uncertainty in various elements made it difficult to demonstrate this conservatism. The APS study group proposed an approach to establish the degree of conservatism associated with system designs that were similar to the proposed approach in the 1988 revision to 50.46:

> "... If the degree of conservatism of the ECCS codes is to be established (assuming they are indeed conservative) then codes which predict the LOCA phenomena realistically must also be available for comparison."

The recommendations of the APS study group helped establish advanced code development programs and the experimental programs that provided data for model development and assessment.

The NRC, now assigned to be the Regulator, initiated a broad series of research. This included several large scale systems tests as part of the 2D-3D Program [10] and support for the development of advanced systems codes such as TRAC-P [11], COBRA/TRAC [12] and RELAP5 [13], each of which modeled non-equilibrium between the phases. This intense period of research enabled a rule change in 1988 which allowed "Best Estimate" LOCA calculations as an alternative to the models stipulated by Appendix K. This was a period of significant growth in our understanding of thermal-hydraulics, system behavior, code development and the physical processes in two-phase flow. That 10CFR50.46 could be revised demonstrated the maturing of the field of nuclear thermal-hydraulics and that the Regulator's perspective had changed such that realistic analysis methods became acceptable.

The "Advanced LWR Period", 1988 – 2009

The following period might be referred to as the "Advanced LWR" period (if one is an optimist), or the "post-Chernobyl" period (if one is a pessimist). There was little growth in the nuclear power industry early in this period, and from the Regulator's perspective this was a period of determining how to license reload cores under the new 1988 rule change, and respond to changes in plant operating conditions such as a power uprate or fuel upgrade.

The first applications using the 1988 rule revision were approved for Westinghouse [14] and later for AREVA [15]. In the mid-90s Westinghouse submitted its application for design certification for the AP600, and General Electric submitted its application for the SBWR thus beginning a roughly 15 year period in which an emphasis was placed on advanced plants generally with passive cooling systems. The NRC responded with independent experimental programs using ROSA [16] for AP600 and the PUMA facility [17] for the SBWR. Confirmation that the new safety features would function as intended and that there was a large margin of safety was the goal of these studies, which continued using the APEX facility [18] for the AP1000 design certification.

The "Current Era", > 2011

The most recent period began on March 11, 2011 with the events at Fukushima. Clearly, since then there has been diminished interest in advanced plants although this also has much to do with the economics of the domestic energy industry. Much of the interest in thermal-hydraulics from the Regulator's perspective however transitioned from advanced plants to containment phenomena and severe accident analysis. The NRC has participated in several international programs investigating spent fuel pool thermal-hydraulics under LOCA conditions, and filtered containment venting. Work on in-vessel thermal-hydraulics has been limited primarily to Small Modular Reactors (SMRs).

4.0 CURRENT ISSUES AND DEVELOPMENT

The Regulator's perspective has been shaped over the past forty years by the need to support rulemaking, revisions to 10CFR50.46, and advanced LWR design certification. A large body of research is now available to develop and assess codes for Loss-of-Coolant-Accident (LOCA) analysis, in large part due to the many experimental programs conducted to better understand large and small break LOCA phenomena. A number of tests (mainly Proprietary) have also been conducted to understand passive safety system performance and natural circulation in the new LWRs.

In spite of the extensive amount of thermal-hydraulic research, questions arise as industry needs and requirements evolve and the NRC must consider new safety issues. The following discussion covers several topics that are of interest to regulatory thermal-hydraulics or may be in the near future.

Fuel and Cladding Research

An important area not to be overlooked in thermal-hydraulics related work is on-going research into reactor fuel and improved cladding materials. Research results have shown that hydrogen uptake into the cladding can affect integrity at high burnup [19], and it may be necessary to revise existing LOCA Evaluation Models to account for new limits that might be imposed on high burnup fuel. The proposed new limits would restrict the maximum equivalent cladding reacted (ECR) to lower values for high burnup fuel [20]. While thermal-hydraulic systems codes, such as the TRACE code under development by the NRC [21], are sometimes considered codes that only solve two-phase flow and heat transfer they must also simulate fuel performance during a transient. To accommodate these fuel research findings, Evaluation Models will need to be revised. Rather than isolating only the hot assembly and fuel at the maximum expected peaking factor, the Evaluation Models will need to simulate second- and third cycle fuel which may be non-limiting for peak cladding temperature (PCT) but may reach the revised limit on ECR for high burnup fuel. This will likely require changes to core and vessel nodalizations in previously licensed models and additional detail in fuel modeling, such as the initial distributions of oxide and crud.

Interfacial Area Transport Equation

Development of an Interfacial Area Transport Equation (IATE) for use in systems codes is under investigation by several organizations, including the NRC. The objective is to derive a system of equations and closure relations that track the change in interfacial area concentration in a two-phase flow as is passes through a channel. Successful development of an IATE would allow for dynamic flow pattern transitions, and be a major improvement upon the current "static" flow pattern maps now in use in systems codes. This has the benefit of providing better accuracy in estimates of the local interfacial area, and possibly on avoiding numerical discontinuities in codes that sometimes occur with flow pattern transitions. An improved estimate of the interfacial area and its dynamic behavior enables improvements in interfacial drag and interfacial heat and mass transfer between the phases.

Successful application of an IATE however will require extensive research and development of models that account for all of the individual mechanisms for bubble interaction; random collision and wake entrainment (coalescence), and turbulent impact, shearing off, and surface instability (break up). Ishii and Kim [22] describe these mechanisms for one- and two-group formulations of the IATE. A transient balance of these mechanisms into an interfacial transport equation allows for a dynamic tracking of the flow structure.

Pursuit of an IATE is a noble goal. It will certainly improve our understanding of two-phase flow. But, from a regulatory perspective development of an IATE is far from complete, nor is it clear what safety improvements might occur. Considerable work is necessary to fully understand the interaction mechanisms and properly correlate them. While progress has been made for simple geometries, a nuclear plant contains many complex structures; rod bundles with spacer grids, fuel nozzles, support plates, plenums, guide tubes, steam generator tubes, etc. How the interaction mechanisms are affected by these structures must be understood before an IATE will be acceptable in analysis of a power plant transient. The IATE must also be shown valid for a broad range of pressure; atmospheric to at least 15.5 MPa and incorporate the correct source and sink terms for subcooled boiling, flashing, and condensation. Thus, from a Regulator's perspective development of an IATE is far from mature.

Uncertainty Quantification

In 1988 the NRC amended the requirements of 10CFR50.46 so that these regulations reflected the improved understanding of ECCS performance during reactor transients that was obtained through the extensive research performed since original rule was published in 1974. The amended rule allows for the use of "best estimate" or "realistic" Evaluation Models for analysis of ECCS performance. The amended rule also required that uncertainty in a realistic evaluation model be quantified, and that when applicable limits were calculated "there is a high probability that the criteria will not be exceeded." Regulatory Guide 1.157 [23] identified a 95% probability level as being acceptable for comparisons of best-estimate predictions to the applicable regulatory limits, but was vague with respect to acceptable methods in which to determine the code uncertainty. Nor, did it specify if a confidence level should be determined. As a result, vendors have developed Evaluation Models utilizing several different methods to combine uncertainty parameters and determine the PCT and other variables to a 95% probability.

In nearly all applications, the models and parameters that are ranged as part of an uncertainty methodology are selected from a Phenomena Identification and Ranking Table (PIRT) that is developed by a group of experts. The PIRT is one of the initial steps in the Code, Scaling, Applicability, and Uncertainty (CSAU) method that is summarized by Boyack et al. [24]. The CSAU process has worked well by providing a structure for code qualification and what are the acceptable features of a best estimate EM. However, it is recognized that the PIRT is subjective. One group of "experts" is likely to come up with a PIRT that is considerably different than that of another equally qualified group of "experts." For the Regulator, this creates an additional uncertainty (for lack of a better word) in the review process. The subjectivity of the PIRT makes it possible to omit an important parameter from consideration, or to place greater emphasis on parameters that may not deserve such scrutiny. The distribution and range of various parameters as applied in the statistical approach is another concern that must be addressed. Thus, from the Regulator's perspective, the treatment of uncertainties in a best estimate application where the state-or-the-art lacks maturity. Absent, is a general consensus on what represents the best statistical approach to address uncertainties in complex systems codes, and what standards should be followed to identify all important parameters, their ranges and distributions.

CASL and NEAMS

Two DOE projects that may result in analysis tools submitted to the NRC for review are the Consortium for Advanced Simulation of Light Water Reactors (CASL) and Nuclear Energy Advanced Modeling and Simulation (NEAMS). One of CASL's objectives is to develop a highly integrated environment for engineering analysis by coupling a suite of advanced codes. NEAMS is also attempting to develop a simulation tool kit that makes use of several advanced codes. Both projects are impressive in scope and in recent accomplishments, especially in the coupling of complex codes. From the Regulator's position, the key concern is verification and validation (V&V), which is also referred to as assessment. As codes are coupled together and empirical correlations are replaced with models that are more fundamentally based, assessment must demonstrate not only that the codes can calculate the primary figures of merit (often peak cladding temperature and the equivalent cladding reacted) but that the codes are not subject to compensating errors. The difficulty is compounded when the physical modeling becomes highly detailed and the net result is due to competing mechanisms. The IATE may be a good example of this. Calculation of the net interfacial area concentration and void fraction depends on several mechanisms for bubble breakup and coalescence. Rather than demonstrating that a conventional flow pattern map is adequate, the applicant will likely need to show that each of the individual mechanisms has a sufficient database and are correctly modeled and can be applied in differing geometries. With some of these mechanisms almost microscopic in scale, highly detailed measurements are necessary. Thus, from the Regulator's perspective on the state-of-the-art for these advanced simulation tools is that they currently lack maturity from insufficient V&V.

Experimental Needs

Closely aligned with V&V is thermal-hydraulic testing and experimental data. A large amount of data for systems code assessment has alread been generated and is available [25]. Data for assessment of advanced LWRs with passive safety systems exists, although most information is Proprietary. This has been sufficient for code assessment to date, but as codes become more complex and allow for higher fidelity in coupled phenomena, new data may be necessary. Validation of these capabilities will require access to data that are representative of the depth of the modeling and of the coupling phenomena.

One possible problem is that the infrastructure for performing large scale thermal-hydraulic tests has decayed in the U.S. Except for some work by vendors for SMRs, there are no large scale test facilities in active use to generate the data that might be needed for advanced codes. From the Regulator's view, this is not necessarily a serious problem. These facilities can be constructed, or accessed through international collaborations, but if new codes and capabilities cannot be assessed with the existing database delays or conservatisms would result.

5.0 SUMMARY AND CONCLUSIONS

The state-of-the-art in nuclear thermal-hydraulics is mature, and the industry currently has a large body of experimental data and codes that are significantly advanced compared to those available in the early 70s. The state-of-the-art enabled a revision to 10CFR50.46 allowing "best estimate" LOCA Evaluation Models, and development of advanced systems codes has resulted in Evaluation Models capable of simulating a wide variety of plant transients. Testing and continued code development and assessment has made it possible to simulate advanced LWRs with passive cooling systems and obtain Design Certification.

Challenges exist in several areas, as the ability to perform complex calculations has outpaced efforts to obtain experimental data and validate advanced tools. Progress in gaining acceptance of highly detailed models and for codes involving complex coupling or in simulating transients with a quantified uncertainty would benefit from new experimental facilities and detailed measurements.

Disclaimer

The opinions expressed in this paper are those of the author and not necessarily those of the U.S. Nuclear Regulatory Commission. Any information presented here should not be interpreted as official NRC policy or guidance.

6.0 ACRONYMS

AEC	Atomic Energy Commission
APEX	Advanced Plant Experiment
AP1000	Advanced Passive 1000
CASL	Consortium for Advanced Simulation of Light Water Reactors
CFR	Code of Federal Regulations
CSAU	Code, Scaling, Applicability and Uncertainty
DOE	Department of Energy
ECCS	Emergency Core Cooling System
ERDA	Energy Research and Development Administration
IATE	Interfacial Area Transport Equation
LOCA	Loss of Coolant Accident
LOFT	Loss of Flow Test
LWR	Light Water Reactor
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NRC	Nuclear Regulatory Commission
NSF	National Science Foundation
OECD	Organization for Economic Cooperation and Development
PIRT	Phenomena Identification and Ranking Table
PUMA	Purdue University multi-dimensional integral test assembly
ROSA	Rig of Safety Assessment
RELAP	Reactor Excursion and Leak Analysis Program
SBWR	Simplified Boiling Water Reactor
SMR	Small Modular Reactor
TRAC	Transient Reactor Analysis Code
TRACE	TRAC/RELAP Computational Engine

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