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Nuclear Power Plant Analysis at the U.S. Nuclear Regulatory Commission

Remarks by The Honorable Peter B. Lyons
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I'm pleased to be here today to give you a brief overview of how the U.S. Nuclear Regulatory Commission (NRC) is using computer modeling and analysis techniques to meet our challenges in licensing, in beyond design-basis assessments, and in evaluating potential safety issues arising from operating experience. I must note that my remarks and views are my own and might not necessarily be those of the Commission.

Our regulatory responsibilities cover a very wide range. In licensing and oversight of these activities, we use a number of analytical tools to derive information that is independent from the calculations submitted by a licensee or an applicant. My remarks today will touch on how the NRC uses these tools to perform our independent regulatory mission, both now and for the foreseeable future.

Importance of Experimental Validation

I was very interested to note that tomorrow's plenary session is focused on experimental facilities. This leads me directly to my fundamental point today, which is the importance of experimental validation. I emphasize this point by way of a personal story from my own past. I started my professional work career at Los Alamos National Laboratory trying to better understand our lack of predictive success in the field of laser fusion. I participated in many of the early laser fusion experiments at a time when there was immense optimism that, based on the best calculations available at the time, modestly sized and fairly inexpensive lasers would provide enough energy to ignite fuel and enable efficient production of fusion energy.

Today there is very little discussion about laser fusion supplying grid power in the near future. The early predictions for success with small lasers have been replaced by construction of the multi-billion dollar and two million joule National Ignition Facility, where ignition and energy gain might be demonstrated, with attempts starting around 2010. This is a far cry from the early predictions. It seems that careful experiments, some done by my group at Los Alamos, simply did not support the optimism of the early calculations. Clearly, computational models are as good, or as bad, as the depth of the physics and engineering underpinning them.

When I came to the NRC about 3½ years ago, I was pleased to find that experimental validation is an important aspect of our regulatory licensing activities. NRC code development utilizes experimental data from a combination of bench-marking tests and experiments. These include tests to model specific physical phenomena, integral tests to assess predictive capability of the dynamic system, and scaling studies to ensure validity when using scale-model experiments as benchmarks.

The performance of a reactor plant under both the design-basis accident conditions for licensing, and in some cases for beyond-design-basis severe accidents, is predicted using analytical codes that have each been benchmarked against experimental data for specific phenomena. In all cases, our uses of computer codes, models, and simulation for licensing are bounded by the extent to which they have been verified experimentally.

As I have noted, I am a firm believer in making the necessary effort to validate predictive computer codes and models. I also believe that pooling international research capabilities to accomplish such validation can be immensely beneficial from both an efficiency and effectiveness perspective.

Licensing Codes and Analyses

The NRC has always employed prudent conservatism to account for uncertainty in our state of knowledge. The earliest thermal-hydraulic codes of the 1960s predicted plant performance under the most limiting design-basis accidents using a collection of discrete evaluation models that each addressed one aspect or physical phenomenon. This isn't too surprising for a number of reasons, not the least of which was the state of computer technology and the computational speeds that were achievable then. Our modeling capability has always been limited by our computational capability. However, even as our computational capabilities increased by orders of magnitude, research continued to be necessary to benchmark our models. A combined U.S. and international research effort in the 1970s and 1980s provided much better understanding of the physical phenomena related to reactor responses under upset and accident conditions. This enabled a revision of NRC 10 CFR Part 50 requirements in 1988 that allowed the use of realistic calculations and models to evaluate the performance of emergency core cooling systems. The use of this rule has permitted higher fuel burnup, increases in operational flexibility, and power uprates, all while maintaining adequate and well-understood safety margins.

Since we began developing thermal-hydraulic accident analysis codes in the 1960s, they have evolved into complex, sophisticated codes that combine calculations that couple two-phase flow, multi-mode heat transfer to and from surrounding structures, fuel cladding oxidation chemistry, cladding stress and strain, multi-dimensional reactor kinetics, and external control system effects. However, in general these codes are not based on "first principles." They use averaged equations that contain many idealizations and semi-empirical models bounded within specific flow regimes and geometries. For example, two-phase flow and heat transfer are much different with horizontal fuel bundles, such as the natural uranium-fueled, heavy water moderated CANDU reactor, than in vertical fuel bundles common to plants in the United States. In addition, the NRC has had to develop film condensation models to evaluate passive heat removal systems for the licensing review of General Electric's Economic Simplified Boiling Water Reactor - the ESBWR.

The U.S. Department of Energy and the NRC have also co-sponsored the development of nuclear analysis codes, such as the SCALE code, for reactor physics, criticality safety, radiation shielding, and waste characterization. The NRC's future need for such capability will be driven by the safety analyses of high burnup and mixed-oxide fuels, spent fuel storage, and criticality controls in new fuel cycle facilities, as well as new reactor designs.

We continue to advance and refine the codes we use in support of the NRC's safety mission. For example, we have consolidated most of our reactor thermal-hydraulics codes into the TRAC/RELAP5 Advanced Computational Engine (TRACE) code that handles two-phase compressible flow in up to three dimensions. Analysts can also couple this code to three-dimensional neutron kinetics models to study feedback effects. Plant-specific TRACE code runs can handle up to tens of thousands of discrete fluid volume nodes. Our steady progress in this area is driven by our safety mission and the care we are taking to ensure a very high confidence in the adequacy of these tools. Continuing development of our codes is focused on improving the physical models, such as situations where liquid films and droplets in the same computational volume can move at different speeds and in different directions, and better numerical methods.

In the neutronics and reactor physics arena, in the near term, we are looking ahead to define our needs for the possible licensing of the Next Generation Nuclear Plant, which is expected to use a high-temperature, gas-cooled reactor. There is a clear possibility for other advanced reactor designs and technologies, and our evaluations may lead us to conclude that we need to improve the computational efficiency of our SCALE code. In the long term, there may be a need for higher resolution and three-dimensional visualization tools. As applicants for new plants move toward the use of massively parallel, high-performance computers to justify adequate safety with less reliance on integral experimental data, the NRC will need equivalent tools to carefully evaluate if this approach is acceptable, and the extent to which experimental data must remain an essential element of code validation.

Another use of simulation codes is in the spent fuel storage and transportation area. The NRC certifies a wide variety of spent fuel and radioactive material storage and transportation casks and systems. We perform technical reviews and analyses as needed to assure criticality safety, proper containment, and heat removal. In some of these cases, we take a slightly different regulatory approach. Rather than developing codes that are themselves independent from the codes used by applicants, we instead obtain and learn to run the codes used by applicants. We achieve regulatory independence by running these codes to explore the sensitivity of the results to various assumptions and to verify the appropriateness of assumptions such as materials properties. We also take advantage of opportunities to independently validate some of the materials properties assumed in very severe environments simulated by these codes. For example, we recently obtained pieces of structural steel damaged during an actual severe highway fire in California in April 2007 caused by an accident involving a gasoline tanker truck. We analyze such specimens to assess how a transportation cask might withstand such severe fire conditions. As another example, to provide assurance that spent fuel in dry transportation casks will not become critical under full water immersion, which is a design-basis event, the NRC obtained French burnup data to use in our criticality analysis code. This is another example of using data to ensure a sound analytical and technical basis for our regulatory decisions.

Analysis of Beyond Design-basis Severe Accident Phenomena

There are a number of important regulatory needs for computational tools other than licensing. For example, following the core melt accident at Three Mile Island Unit 2 in 1979, the NRC undertook a very substantial research program to develop insights into severe accident scenarios involving significant core melting. Such severe accidents were those that could progress well beyond the calculated design-basis accident results that were verified by the NRC's licensing computer codes. The NRC sponsored development of computer-based models to predict the phenomenology of severe accident progression and benchmarked them, to the extent possible, with applicable severe accident experimental research results. These models are not used to demonstrate or confirm regulatory licensing or compliance, but rather are used to inform the Commission's safety policies and highlight areas for regulatory focus.

Another example is the use of fault-tree logic and statistical approaches to generate probabilistic risk models of nuclear power plants. These have grown in complexity along with steadily increasing computational power and speed. Today we continue to add new and more computationally challenging elements to our risk models, such as the risk from internal fires. While this increase in complexity is aimed at providing more complete and detailed risk insights, it also creates significant challenges in making such insights understandable to the NRC, licensee decision-makers, and others who use them.

The future of the NRC's computer simulation capabilities may include the integrated coupling of probabilistic plant risk models with severe accident (core-melt) progression models. Both types of models have ongoing programs to develop a greater degree of detail or accuracy. For example, the NRC has initiated work with Sandia National Laboratories to improve the realism of our probabilistic models using the most current accident research data and analysis. Its objective is to update a 25-year-old, unrealistically conservative analysis by using the latest severe accident research and best available estimation techniques. Such large-scale-model integration has not been previously accomplished, but today we have the computational capability to do it. One application being considered for such a large integrated model would be in faster-than-real-time simulation for use during training exercises or actual events.

Another beyond-design-basis analysis capability that the NRC has developed since 2001 is that of assessing the effects of an impact from a commercial aircraft to a nuclear power plant. We clearly do not have full, integrated testing to validate these computer simulations and analyses. However, we use state-of-the-art structural and fire analyses and computer modeling to realistically predict the consequences. These studies have confirmed that given robust plant designs, and the additional enhancements we have required to safety, security, and emergency preparedness and response, it is unlikely that significant radiological consequences would result.

Analyses addressing operating experience and safety issue resolution

The NRC has also been able to respond rapidly to produce its own independent computer simulations when a potential safety issue is identified from operating experience. One such example was the 2006 discovery at Wolf Creek of five circumferential crack indications in primary coolant system dissimilar metal welds that were significantly larger than previously seen in the industry. The issue was generic, and the NRC was initially concerned enough to contemplate the need for regulatory action to cause near-term plant shutdowns and inspections. In response, U.S. nuclear industry developed an advanced finite element analysis to better understand the relative safety significance of the problem. To review and evaluate the adequacy of its analysis, the NRC established a parallel confirmatory study to benchmark the industry's results and evaluate its quality. We did this by independently developing our own advanced computer model and using it to evaluate the sensitivity of industry's results to a variety of assumptions, including specific geometries, operating loads, stress intensity factors, initial flaw assumptions, crack growth, and leak rate models. This significant effort, by both industry and the NRC, gave us the regulatory confidence we needed to establish an acceptable inspection schedule for all affected operating plants. The area of materials degradation is one of ongoing research to develop the experimental data needed to validate our increasingly complex models.

Concluding Remarks

In conclusion, there may come a time when our computational capabilities and understanding of the relevant physical phenomena are good enough to rely fully on a "first principles" approach to simulation and modeling. Such calculations would need to demonstrate an improved fidelity to more accurately predict performance than do the current models. In any case, I do not see the need for experimental validation ever being eliminated. The NRC will utilize validated computational tools when they are adequate for the task, but we must always act conservatively in consideration of uncertainty, and we expect our licensees to do the same when such validation does not exist.

Thank you for your attention, and I hope you have a very successful and productive conference.