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RISK-INFORMED REGULATION AND REACTOR OVERSIGHT

by

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Introduction

Good morning. I would like to express my thanks to Admiral Bowman for his invitation to present this talk, which continues the cordial relationship between my organization and yours. The NRC has worked with Naval Reactors for many years, reviewing the designs and safety evaluations of your new reactors. I believe that the interaction has benefitted us both.

As it happens, I have a particular fondness for naval reactors that stems from my childhood. When I was in 4th or 5th grade, I was obligated to generate a project for a science fair and, of course, left any thought about it until the night before it was due. I managed to parlay a plastic model of the Nautilus and a primitive understanding of Avogadro's principle into an undeserved first-place finish. With this as a backdrop, it was a particular pleasure about a year ago to spend a day on board an actual submarine, the U.S.S. Oklahoma City, seeing just how well the machines you design and the people who operate them perform. It was an unforgettable experience.

I should add that both the NRC and I personally have benefitted greatly from the expertise of many people who have been involved in the Navy nuclear program and subsequently joined the NRC staff. One of my reactor technical assistants was a graduate of the Naval Academy and is a submariner who is still in the Naval Reserves. I gave him some instructions as to how to handle a tough and urgent situation and he responded with an "Aye, Aye, Captain." I viewed his accidental utterance as a great compliment. It made my day.

I would like to focus this morning primarily on initiatives that the NRC is taking with regard to the way in which we regulate civilian power reactors -- specifically, the NRC's efforts to incorporate the consideration of risk insights into our regulations, our regulatory processes, and our oversight of power reactor operations. We refer to these initiatives collectively as "risk-informed regulation." When I speak of risk, I am referring in most cases to quantitative evaluation of risk, using probabilistic risk assessment, or PRA. As most of you are aware, the development of PRA techniques for use in evaluating the safety of nuclear power plants began about 30 years ago, with a program sponsored by the NRC's predecessor, the Atomic Energy Commission. The development and maturity of PRA since that time was a key consideration in the NRC's decision, in the mid-1990s, to move to a more risk-oriented regulatory approach.

The most substantive changes in the NRC's activities have occurred in our reactor oversight process, or ROP. A new, risk-informed process was tested for a selected number of pilot plants in 1999. Based on a highly favorable review of that effort by a panel of NRC staff and stakeholders – including organizations that are often critical of the NRC – the new process was implemented industry-wide in April 2000. Let me review briefly the way the system used to work, and then discuss how we have changed and improved it.

Reactor Oversight

The NRC's reactor oversight process as it existed until 1999 had its origins in the agency's response to the 1979 accident at Three Mile Island. Among the significant actions taken by the NRC were the stationing of resident inspectors at every operating power reactor site, and the establishment of an evaluation process, termed the Systematic Assessment of Licensee Performance, or "SALP." SALP was largely an inspection-based program, in which the NRC reviewed licensee performance on a 12- to 24-month cycle in four "functional areas": plant operations, maintenance, engineering, and plant support. A numerical rating for each area was determined, and a report was prepared discussing the licensee's performance. The period between SALP evaluations was based on the licensee's SALP score: poor performers were rated more frequently, while top plants were assessed less often. As time went along, two other oversight activities were incorporated into the process: a semiannual meeting of NRC senior managers focusing on plants with poor or declining performance, a product of which was the famous–or, perhaps infamous–"watch list"; and a semiannual plant performance review, the purpose of which was to assess overall plant performance and to plan future inspections.

SALP was developed when there was relatively little operational experience with nuclear power plants. A governing presumption was that plants were safe if they were in compliance with NRC regulations. As a result, the focus of the SALP process was often on compliance, regardless of the safety implications of a failure to comply. SALP was also the subject of considerable criticism over the years for a number of other reasons, including:

_ Claims that the SALP process was too subjective, too dependent on the judgment of the inspectors as to whether performance was acceptable;

S Claims that the bases for the numerical scores were, in some cases, obscure, and the meaning of a particular score was difficult to interpret for both the licensee and other stakeholders; and

- Claims that the process was largely retrospective, looking at past performance, and not reflective of the contemporaneous situation. It was asserted that problems might be cited that had long been corrected, while emergent issues could be overlooked.

In response to these criticisms and others and in concert with the decision to move toward a more risk-informed regulatory philosophy, the agency sought to provide a more objective, timely, and safety-focused process for accomplishing oversight responsibilities. The result of this effort was our new ROP.

The Reactor Oversight Process

(Second slide) The basic framework of the ROP reflects the NRC's overall safety mission, which is to protect public health and safety. You see the mission represented at the top of the framework. The next level shows the three strategic performance areas that support the accomplishment of our mission. Reactor safety refers to protection against the impacts of reactor accidents. Radiation safety refers primarily to releases as a result of normal operation, as opposed to accident-related impacts. And you also see a third area, safeguards, which relates to efforts to ensure that special nuclear materials are properly protected from accidental or deliberate misuse. This third element is not limited to nuclear power plant sites, but it is an important aspect of our licensees' responsibilities.

The next level of the framework comprises what we call the seven "cornerstones" for achieving acceptable safety performance. The four reactor safety cornerstones reflect the NRC's defense-in-depth philosophy: accident prevention and the mitigation of accident consequences, with an appropriate balance between them. That is, our licensees should strive to see that accidents do not happen. But we also require the capability to deal with accidents if they should occur, and to minimize their consequences. The cornerstones follow logically from the accident mitigation and prevention functions. Accidents begin with initiating events, which should be minimized. They are kept from progressing by the action of mitigating systems. If those systems are unavailable or ineffective, there are engineered barriers that prevent or hinder the release of radioactive material. Should that material escape into the environment, emergency preparedness provides the means by which action is taken to protect members of the public from health impacts of radiation exposure.

The two cornerstones under radiation safety reflect the NRC's regulatory limits on both worker exposure and routine releases to the environment. The last cornerstone, related to safeguards, indicates the need to provide protection against misuse of nuclear materials.

The last row of the framework is also extremely important. These are called "cross-cutting areas," and reflect aspects of plant operation that are common to all of the strategic performance areas and cornerstones. These are human performance, the establishment and maintenance of a safety-conscious work environment, and problem identification and resolution. These are elements of what is broadly referred to as "safety culture." I will come back to that topic, but let me proceed right now to explain how the ROP framework is actually implemented.

(Third slide) This slide is very complicated, but for now, I shall focus on the bottom half, which shows the two means of assessing licensee performance: performance indicators and inspections. Recall that the goals in developing this new process were to provide a more objective, timely, and scrutable means for assessing licensee performance, as well as to improve the focus on issues of true risk-significance. The issue of objectivity has been addressed by establishing performance indicators for each of the seven cornerstones. These indicators are quantitative measures of system performance, such as safety system functional failures, or, in some cases, programmatic performance, such as emergency preparedness drill participation. To augment the performance indicators and to assess performance and programmatic areas for which a quantitative assessment is not practical, we still conduct inspections. However, the inspection program has been revised to focus on risk-significant issues.

Once the performance indicators and inspection findings have been compiled, their risk-significance must be assessed. For performance indicators, the numerical values are compared to established thresholds. Inspection findings are evaluated by means of a significance determination process (or SDP), in which simplified risk models are used to assess the safety-significance of each finding. The simplified risk models are, in essence, very generalized PRAs.

The goals of timeliness and scrutability are served by the reporting process. Inspection and performance indicator assessments are reported quarterly, and the results in each area are color-coded, corresponding to the safety-significance determined in the evaluation process. The next slide illustrates how the information is displayed, with respect to the performance indicators for each cornerstone. (Fourth slide.) This is taken from our website. A “green” finding or performance indicator indicates very low safety significance. White is the first threshold, and that color indicates low-to-moderate safety significance. Yellow is the next threshold, representing substantial safety significance. High safety significance is indicated by a red performance indicator or inspection finding.

The final step of the assessment process is to evaluate the results to determine necessary NRC follow-up activities. This is done by means of our “action matrix” (fifth slide). From the left to the right across the top are the results, increasing in the level of safety significance. The rows correspond to agency and licensee actions and communications. This matrix guides the disposition of performance indicator findings and the results of the NRC’s inspection activities. An “all green” report means that findings are referred back to the licensee for corrective action, and the subsequent inspection effort will be at the baseline level. Degradation in safety performance, as indicated by white, yellow, or red findings, results in increasing levels of NRC oversight in the disposition of findings and increased inspection effort. The action matrix also indicates how the agency is to communicate its findings to the licensee and to the public.

As I indicated, the results of the ROP performance assessment determine how the NRC will conduct inspections at a plant. All plants get at least the baseline inspection effort, while supplemental inspections may be included to respond to degradations in safety performance. This permits us to schedule our inspection activities in advance, and to inform licensees about those activities. Inspections are planned 12 months ahead for all plants, and are adjusted every 6 months as determined by the results of quarterly assessments. Once a year, NRC senior managers meet to discuss the results of plant assessments, in what is called the Agency Action Review. In addition, the NRC holds public meetings at plant sites to discuss licensee performance. SDP results are also used as an input to the NRC’s enforcement process, to ensure that enforcement actions are consistent with the safety significance of regulatory non-compliance.

As I noted earlier, the new ROP has been in operation for all power plant licensees for about than 22 months, and the initial indications are that it has been extremely successful in accomplishing most of its goals. This is not only the NRC’s conclusion; feedback from our licensees and stakeholders has been largely positive, as well. Under the new process, our assessments are more timely and the color-coded results are much easier to understand than was the case with the old SALP numerical scores. Performance indicators increase the objectivity of the agency's findings and there is a clear connection between the overall performance assessment and the commitment of NRC inspection resources and the enforcement process.

There are still some improvements to be made. For example, we are studying other performance indicators to see if we can establish an even better connection to risk. We also seek performance indicators that will help predict emergent problems, and thereby permit their avoidance, rather than to apply performance indicators that merely confirm existing problems. We are also working to improve the risk assessment tools that are used. It seems clear at this juncture, however, that the ROP has been a change for the better from nearly every perspective.

Other Elements of Risk-Informed Regulation

As I indicated in my introductory remarks, oversight of reactor operations is only one element of the NRC's reactor regulatory program – and the one that is most directly involved with the day-to-day activities of our licensees. The other elements of reactor regulation involve the development and establishment of the rules that define the NRC's requirements and the establishment of processes for the conduct of NRC business. We are also incorporating risk insights into these rules and processes.

For example, we are revising the so-called "special treatment" requirements -- the rules that define the special requirements for systems, structures and components (or "SSCs") that are important to safety. We have found through both operational experience and PRA analyses that some SSCs denoted as "safety-related," and thus subject to such requirements, are in fact not risk-significant, while other SSCs that are not formally safety-related are, in fact, risk-significant. As a result, we are undertaking efforts to define treatment according to risk impacts and are in the process of developing a new rule, 10 CFR 50.69, that will address this issue.

We are also looking at how risk insights might affect the technical bases for some regulations. For instance, 10 CFR 50.44 defines requirements for combustible gas control, principally hydrogen, during certain postulated accidents. As you may know, during the Three Mile Island accident, hydrogen built up in containment as a result of the chemical reaction between steam and the fuel cladding, and is believed to have burned, causing a pressure spike in the containment building. After TMI, requirements were instituted for licensees to install hydrogen recombiners that would operate after such an event to keep hydrogen from reaching concentrations that could burn or explode. Those requirements were based upon the best technical information available at the time. However, since then, we have learned much more about the progression of these types of accidents, and it turns out that because of their inherent design and operating characteristics, recombiners do not help much in these scenarios. Thus, our improved technical and risk insights indicate that allowing licensees to remove recombiners would not affect risk, and thus we are in the process of developing revisions to 10 CFR 50.44 for that purpose.

I must also note, however, that certain types of containments are, in fact, susceptible to damage from hydrogen burns or explosions, and there are devices, called igniters, that are effective in controlling combustible gases in such situations. We are currently evaluating whether additional requirements may need to be established for the more vulnerable containment designs.

This points out an important aspect of risk-informed regulatory changes: the sword can cut both ways. There are cases in which we discover that we are imposing requirements that serve no meaningful purpose and we can appropriately relax these requirements. But there are also sure to be cases in which we discover that additional requirements are necessary to address previously unrecognized risk-significant issues.

Although we have accomplished a great deal in the last few years in incorporating risk considerations into our regulatory structure, we clearly have much to do before we are finished. Moreover, we are embarking on a parallel effort to risk-inform regulations that apply to our materials licensees – of which we have several thousand in many different categories, from medical and industrial uses of radioactive materials to waste disposal. Risk-informed regulation will be a major area of focus for the NRC for a number of years.

Advanced Reactors

Let me mention briefly one other area of significant current interest in reactor regulation: the possibility that new reactors of advanced designs could be built in the U.S. in the not-too-distant future.

I expect that most of you have seen articles and reports about all sorts of new reactor designs, such as the helium-cooled pebble-bed modular reactor. In the early 1990s, the NRC established a new licensing process for standardized reactor designs, in 10 CFR Part 52. The licensing process that was employed for all of our currently operating plants was a multi-step process, in which a construction permit was issued, the plant was built – and usually was being designed at the same time it was under construction. A separate operating license was required, which was not acted upon until construction was essentially completed. There were opportunities for hearings at each stage of the process, and the ultimate result was a long, costly licensing procedure, with no assurance that a completed plant could be operated. In fact, the Shoreham plant, on Long Island, is an example of a plant that was completely built but never operated.

The process established in Part 52 aimed to address the shortcomings of the old process by, among other changes, allowing plant designs to be reviewed and certified prior to their construction. The certified design can be approved through a rulemaking, and an applicant can then apply to build such a plant without any hearings on the technical issues resolved during certification. A combined construction permit and operating license can also be issued, and once the plant is built and been demonstrated to conform to the certified design, permission to operate is granted. The NRC has certified three reactor designs, and expects an application for a fourth to be submitted in the near future; several other reactor manufacturers are seriously considering certification, as well.

The regulations in Part 50 governing reactor design, analysis, and operation were written almost exclusively for water-cooled reactors. When we consider reactors with gas or other non-water coolants and core designs that are much different from conventional water reactors, we find that many of our current regulations do not apply, or must be reinterpreted to accommodate the novel technology. This has led us to think about the possibility of establishing a new licensing framework for advanced reactors. Rather than starting with the current body of regulations and trying to figure out which to keep, which to eliminate, and which to modify, a new framework would allow an integrated approach to regulation to be developed. This process would also allow a risk-informed approach to be taken at the start, rather than by the rule-by-rule procedure that is currently employed for revising our regulations. The NRC staff is currently discussing this issue with the industry and other stakeholders, and will be providing recommendations to the Commission later this year. This would clearly be a major undertaking for the NRC, but the ultimate outcome – a risk-informed approach to regulation that could be applied to a wide range of reactor technologies – would be of substantial benefit to the NRC in considering new reactor technologies.

Conclusion

I hope that I have been able to provide a glimpse at our processes for the regulation and oversight of civilian power reactors, and for the sorts of challenges we may face in the future as we consider new reactor designs. This is a very exciting time for the NRC as we move forward with many new initiatives.   We look forward to the challenges that await us.

Thank you.