ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON REGULATORY POLICIES AND PRACTICES
PROPOSED RULE ON RISK-INFORMED CHANGES TO
LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS

SUBCOMMITTEE: MAY 6, 2009
FULL COMMITTEE: MAY 7, 2009

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Cognizant ACRS Staff: David Bessette, Hossein Nourbakhsh

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS

ROCKVILLE, MARYLAND
MAY 6, 2009

1	Opening Remarks by the Chairman	Dr. William Shack	1:30 pm - 1:35 pm
2	Opening Remarks by NRR	William Ruland	1:35 pm - 1:40 pm
3	Overview of Staff Presentation	Tim Collins, NRR	1:40 pm - 1:45 pm
4	Background and Status of Rulemaking	Tim Collins, NRR	1:45 pm - 2:00 pm
5	Summary of 50.45a Rule Concept	Tim Collins, NRR	2:00 pm - 2:30 pm (20 min)
6	Rule Changes to Ensure Plant-Specific Applicability of Expert Elicitation and Impact of Seismic (ACRS Recommendation)	Rob Tregoning, RES	2:30 pm - 3:15 pm (45 min)
	Break		3:15 pm 3:30 pm
7	Rule Changes to Increase Defense-in-Depth for Beyond Transition Break Size Breaks (ACRS Recommendation)	Tim Collins, NRR	3:30 pm - 4:15 pm (45 min)
8	Rule Changes Related to Risk Assessment Process (ACRS Recommendation)	Steve Dinsmore, NRR	4:15 pm - 5:00 pm
8	Subcommittee Discussion and Plan for Full Committee		5:00 pm - 5:15 pm
9	Adjourn		5:00 pm

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STATUS SUMMARY

Purpose

To consider the draft revised proposed rule to amend 10 CFR 50.46 and 10 CFR 52 to allow alternate risk-informed break size evaluation of loss-of-coolant accidents that will be issued for public comment.

Background

The Definition of Large Break

10 CFR 50 Appendix A, Definitions and Explanations, defines LOCA as follows:

"*Loss of Coolant Accidents*. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system." [February 20, 1971]

The LOCA/ECCS Rule

10 CFR 50.46 and Appendix K were originally published on January 4, 1974. This LOCA/ECCS rule was the product of some 5 years of discussion, development, and contentious hearings. Before the adoption of the LOCA/ECCS rule, interim acceptance criteria for emergency core cooling systems for light water reactors were issued by the AEC on June 19, 1971 [Federal Register, Vol. 36, No. 125, pp 12,247 - 12,250]. By this point in time, the regulatory staff had already been using the double-ended guillotine break as the design basis large break LOCA for several years.

Key Historical Milestones Leading to the LOCA/ECCS Rule

During the review of several applications for construction in the 1965-66 time frame, it became apparent that because of increases in size and power, the role and performance of emergency core cooling systems needed more careful consideration. On October 27, 1966, the Director of Regulation of the Atomic Energy Commission appointed a Committee to conduct a review of power reactor emergency core cooling systems and core protection. The report of that Committee was published as "*Emergency Core Cooling: Report of Advisory Task Force on Power Reactor Emergency Core Cooling*," U.S, Atomic Energy Commission, 1967, generally referred to as the Ergen report. With respect to LOCA break size, the Ergen report states as follows (pp 41-43):

"Conclusion 6 - Break Size for Emergency Core-Cooling Design

a. We consider it unnecessary to assume that large and rapid failures will occur in any component or system which is designed, manufactured, inspected, protected against missiles, and operated in accordance with the requirements given in Conclusion 7 or their equivalent.

b. Because the record of conventional as well as nuclear plant performance to date clearly indicates that small leaks from a pressurized system can occur, we consider it necessary that back-up means be provided for introducing water into the primary system to assure continued core cooling.

c. In addition to a. and b., the emergency core-cooling system should also be capable of handling a large and rapid failure of those components and systems which are not designed, fabricated, inspected, protected against missiles, and operated in accordance with Conclusion 7 or its equivalent.

d. We expect that, as recommended herein, more and more elements of the primary system will be designed, manufactured, and inspected to the same degree of high standards as required by Section III of the ASME code, its revisions in process, and additional requirements such as those recommended in this report, to give the same reliability as reactor vessels. This evolution, which will further assure primary system integrity, should make it possible to design emergency core-cooling systems for reduced break sizes, because large and rapid failure of components meeting the recommended standards will not have to be considered. Eventually, a minimum in the reduced break size would still have to be specified as an acceptable basis for designing emergency core-cooling systems. In establishing such a minimum, a prudent safety factor based on engineering experience and judgement should be used. We consider that even with this safety factor the minimum acceptable break size eventually will be considerably smaller than the current design basis.

Discussion

The present concept of emergency core cooling will typically include several types of systems to inject water onto the core, i.e., high-pressure low flow injection systems, low-pressure high-flow injection systems or core spray systems. These emergency core-cooling systems are being designed to cover a range of breaks up to the double-ended break of the largest pipe connected to the reactor vessel. The use of the double-ended break of this largest pipe is an arbitrary criterion.

It can be argued that specifying the double ended break of the largest primary-coolant pipe connected to the reactor vessel for the loss-of-coolant accident is conservative and that by designing emergency core-cooling systems to protect against such eventuality, the safety of the reactor plant is increased. However, because of the severity of this maximum postulated accident, the number of emergency core-cooling systems installed in latest PWR and BWR reactor-plant designs has been increased to achieve the specified capacity or redundancy. Each one of these systems adds a significant amount of complexity to the primary coolant system in that they involve the addition of pipes, penetrations, valves, electrical controls, interlocks, test systems, sampling systems, and emergency-power sources.

The number of emergency core-cooling systems installed to meet the break size and redundancy requirements should in itself be a matter for careful consideration. Engineering experience over the years has, in general, shown that fewer pieces of equipment and a smaller number of systems are more desirable. Thus, in the final analysis, judgement must be applied in achieving a careful balance between the addition of systems to protect against postulated casualties and the resulting complications introduced by these added protective systems.

For the reasons given in Conclusion 7, we consider that by implementing the types of requirements suggested therein, it should not be necessary to assume that large and rapid failures will occur in those parts of the primary system that meet these requirements. Accordingly, for such high integrity primary systems it should not be necessary to design emergency core-cooling systems for very large primary system breaks. Such extreme protection would unnecessarily increase the system complexity wand could decrease the inherent reliability of the primary system as discussed previously. As a greater portion of the primary system is designed to the requirements suggested in Conclusion 7 it will lead to the design of emergency core-cooling systems for reduced break sizes, and these sizes will be considerably smaller that the present design basis. It is our opinion that a better balance will thus have been achieved in the design of the primary system and its protective emergency core-cooling system, resulting in increased plant safety."

Thus, while the vendors were designing plants to take into account large break LOCA, the regulatory staff evidently had not formally adopted this as a design basis accident. Howeverr, the staff had effectively adopted the vendors' design basis for large break LOCA.

In 1968, the Atomic Energy Commission appointed an internal study group headed by Harold Mangelsdorf to review its regulatory program to assure that licensing procedure were keeping pace with the then rapid expansion of the industry. The internal study group concluded [*Report to the AEC on the Reactor Licensing Program by the Internal Study Group*, June 1969]:

"1. It remains necessary to consider, in safety reviews, a wide variety of expected transients and postulated accidents.

2. The design basis accidents presently used in the safety evaluation of large water-cooled power reactors should not be changed until convincing technical evidence is available that the change is justified.

.....It has been suggested that the design-basis accidents presently used should be revised to reflect a more nearly consistent and mechanistic view of what would actually happen. The Group believes that sufficient knowledge in not available to justify such a revision."

The principal design basis accident that governs most aspects of core design and technical specifications is the large break loss-of-coolant accident (LOCA). The LOCA rule is defined in 10 CFR 50.46 and Appendix K.

The proposed rule revision to reduce the maximum design basis break size to the largest connecting pipe to the reactor coolant system may allow licensees to uprate power and/or change peaking factors. It could affect systems and components important to safety from the licensing basis. These include: containment spray system setpoint changes, fuel management improvements, optimization of plant modifications and operator actions to address postulated sump blockage issues, power uprates, and changes to the required number of accumulators, diesel start times, sequencing of equipment, and valve stroke times. modifying setpoints on accumulators or removing some from service, eliminating fast starting of one or more emergency diesel generators, etc

Risk-Informing 50.46

In 1995, the Commission published a Policy Statement on the Use of Probabilistic Risk Assessment (PRA). In its September 2, 1998, briefing to the Commission on the status of the PRA Implementation Plan, the staff proposed to assess various options for risk-informing 10 CFR 50. In a SRM dated September 14, 1998, the Commission directed the staff to go ahead and do so. The staff responded by preparing *"Options for Risk Informed Revisions to 10 CFR Part 50 - Domestic Licensing of Production and Utilization Facilities*," December 23, 1998 [SECY-98-300, ML982870048]. The staff proposed three options for risk informing 10 CFR 50. Option 3 involved focusing on the regulations that have the most significant potential for improving efficiency and reducing unnecessary burden.

On June 8, 1999, the Commission issued its SRM on SECY-98-300. The Commission approved a study by the staff to to risk-inform 10 CFR 50 and directed the staff to "*study on an aggressive timetable and provide, for Commission approval, a schedule for this activity*." (note: that was 10 years ago! Rome wasn't built in a day.) In response, the staff sent to the

Commission, *"Proposed Staff Plan for Risk Informing Technical Requirements in 10 CFR Part 50*," November 8. 1999 (SECY-99-264, ML9932301670).

In its review of SECY-99-264, the industry, in a letter from NEI to Chairman Meserve (ML0036962420), stated that the highest priority should be given to:

1. 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,* including Appendix K to Part 50, and

2. Rulemaking on 10 CFR 50.44, Standards *for Combustible Gas Control System in Light-Water-Cooled Power Reactors.*

This was based on a survey of all the plants for which 61 units responded.

On April 12, 2000, the Staff issued its 1st progress report to the Commission, *"Status Report on Risk Informing the Technical Requirements of 10 CFR Part 50 (Option 3),"* (SECY-00-0086, ML0036962580). This contained little of interest.

On September 14, 2000, the staff issued its 2nd progress report, *"Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control*)," (SECY-00-0198, ML003746990). It also contained little of interest.

On February 5, 2001, the staff issued its 3rd progress report, *"Third Status Report on Risk-Informing the Technical Requirements in 10 CFR Part 50 (Option 3),* (ML0102600324). In this SECY paper, the staff proposed a three part program.

"Phase 1: Assessment of the LBLOCA with Respect to ECCS Requirements. This assessment would consider whether or not operating experience, fracture mechanics and thermal-hydraulic analysis and leak before break considerations warrant use of a different LOCA as a design basis accident, rather than the currently defined LBLOCA. It should be recognized, however, that any redefinition of the LBLOCA will involve the consideration of many factors and could ultimately not be judged feasible. In addition, the staff assessment would consider whether or not the current assumptions and practices for analysis of the LBLOCA are reasonable in view of risk information and current understanding in the areas of thermal-hydraulics and fuel behavior. However, the focus would be on ECCS requirements only (i.e., 10 CFR 50.46).

Phase 2: Assessment of the LBLOCA with Respect to Other Plant Design Requirements. This phase would address whether or not changes to other plant design requirements (other than the ECCS) dependent upon the LBLOCA assumptions are warranted based upon risk insights and Phase 1 results (e.g., 10 CFR 50.49, containment).

Phase 3 Assessment of the Current ECCS Acceptance Criteria. This phase would focus on assessing the 2200F and 17% clad oxidation criteria, currently in 10 CFR 50.46. Current knowledge regarding cladding materials and burnup effects and risk insights will be used; however, if experimental work is needed this phase could result in an extended schedule."

On July 23, 2001, the staff issued its 4th progress report, "*Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)*" (SECY-01-033, ML0118004942). This paper contains a significant description of staff plans and recommendations. With respect to break size, the paper stated that:

- "ECCS spectrum of break sizes and locations. Given current estimates of the frequency of large-break LOCAs (NUREG/CR-5750 indicate 95th percentile values of 10^{-5} per critical year for pressurized water reactors and 10^{-4} per critical year for boiling water reactors), the reliability of the ECCS (and containment functions) is generally sufficient to assure that large-break LOCAs (> 6 inches in diameter) are not significant contributors to risk. However, the current estimates of large-break LOCA frequencies are uncertain and are not low enough to allow elimination of all large-break LOCA sizes from the design bases. In addition, plant equipment that is designed, at least in part, to the requirements of design-basis LOCAs also provides defense against a spectrum of beyond-design-basis accidents."

On March 29, 2002, the staff issued an update to its 4th progress report, "*Update to SECY-01-0133, Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Critereia)"* (SECY-02-0057, ML02206606071). The paper notes that several organizations stated that changing the spectrum of design basis break sizes is one of their highest priorities with respect to risk-informed regulation. Industry submitted a petition to allow the option of using an alternative to the currently required double-ended rupture of the largest pipe in the reactor coolant system. Commissioner Diaz commented extensively on this SECY paper, and these are provided as Attachment 10, page 56.

In its March 31, 2003, SRM on SECY-02-0057 (ML0309104760), the Commission provided extensive guidance with respect to design basis LOCA redefinition. The Commission agreed to consider redefining the design basis large-break loss-of-coolant accident (LOCA) in view of the apparent low risk associated with such events. The Commission directed the staff to complete the technical basis supporting the large break (LB) LOCA redefinition, supported by a 10-year estimation of LOCA frequencies. The SRM also directed the staff to prepare a proposed rule that would relax the current requirements for consideration of LBLOCA with coincident loss of offsite power. The Commission stated it would not support actual changes to ECCS coolant flow rates.

On March 3, 2004, the staff issued "*Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power* (SECY-04-0037, ML0404901331). In this paper, the staff sought Commission policy guidance on: retention of mitigation capability; use of best-estimate evaluation models; large break LOCA redefinition; and applicability to future plants.

On July 1, 2004, the Commission issued a SRM on SECY-04-0037 (ML04188304120) approving the development of a proposed rule to replace large break LOCA. The Commission stated that licensees should be required, by regulation, to retain the capability to successfully mitigate the full spectrum of LOCAs for break sizes between the new maximum break size and the double ended guillotine break of the largest pipe in the reactor coolant system. The mitigation capabilities for beyond design basis events, and any changes to these capabilities, should be controlled by NRC requirements commensurate with the safety significance of these capabilities and not by voluntary means. The Commission added that:

- "The low risk contribution of the large break LOCA, which allows removal of the large break LOCA from the design basis event category, should weigh heavily in the types of requirements that would be imposed in this area. Because of the low safety significance of the large-break LOCA, a high level criterion in the rule should include the requirement for the licensee to provide effective mitigation capabilities, including severe accident mitigation strategies directed at break sizes greater than the alternate maximum break size permitted by the rule, to maintain the core in a coolable geometry. Consistent with the approach taken in the 10 CFR 50.69 rulemaking on treatment and commensurate with the low safety significance of these capabilities, the staff should ensure that capabilities are provided in a performance-based manner and not in a prescriptive manner. Furthermore, to address the potential consequences from a beyond design basis LOCA, the staff should include a requirement for containment integrity."

Containment

The staff reviewed GDC 50, Containment Design Basis. GDC 50 specifies, in part, that the reactor containment structure shall be designed to accommodate, with sufficient margin, the calculated pressure and temperature from any LOCA. It also lists several factors that should be considered when determining the available margin. The staff determined that these factors should also be considered when determining the available margin for accommodating LOCAs larger than the transition break size. Under § 50.46a, LOCAs larger than the TBS are not design basis accidents since they are highly unlikely. Nevertheless, reactor containment designs should continue to consider beyond TBS LOCAs, but the methods used to calculate containment temperatures and pressures need not be as conservative as they are for design basis accidents. Thus, the staff proposes to modify GDC 50 to specify that under § 50.46a, leak tight containment capability should be maintained for ''realistically'' calculated temperatures and pressures for LOCAs larger than the TBS.

For licensees voluntarily choosing to comply with § 50.46a, the structural and leak tight integrity of the reactor containment structure, including access openings, penetrations, and its internal compartments, shall be maintained for realistically calculated pressure and temperature conditions resulting from any loss of coolant accident larger than the transition break size.

Should licensees make plant modifications under § 50.46a resulting in containment pressures and temperatures that exceed the current design values by a small amount, the staff will evaluate the acceptability of revised containment structural integrity criteria. Criteria will be provided in a regulatory guide for containment structural integrity that could be used with § 50.46a. However, the acceptability of containment pressures and temperatures exceeding current values will also be evaluated for conformance with the LERF acceptance criteria specified in § 50.46a(f)(2) and the defense-in-depth acceptance criteria in § 50.46a(f)(3). The basis for allowing revision to containment structural integrity criteria is that LOCAs involving pipe breaks larger than the TBS are judged to be of very low probability and are no longer considered to be design basis accidents. However, a realistic assessment of containment structural capability for LOCAs involving pipe breaks larger than the TBS (without consideration of a loss-of-offsite-power and a single failure) is still required to provide defense-in-depth.

The inherent physical robustness of current reactor containment contributes significantly to the ‘‘built-in capability’’ of the plant to resist security threats. The Commission expects licensees not to make design modifications to the containment under § 50.46a that would reduce its structural capability (based on realistically calculated containment pressures and temperatures for breaks larger than the TBS) to a level that would compromise plant security.

EXPECTED COMMITTEE ACTION

The Subcommittee and the Full Committee are expected to take note of the NRR plans and information presented. A letter would be written only if the Committee has significant comments on the rulemaking at this time.

ATTACHMENTS:

ACRS letters on the subject of redefinition of the limiting design basis large break loss-of-coolant accident are provided as attachments.

Attachment 1: ACRS Letter of December 20, 2007

ACRSR-2276

December 20, 2007

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL NUREG-1829, "ESTIMATING LOSS-OF-COOLANT ACCIDENT (LOCA) FREQUENCIES THROUGH THE ELICITATION PROCESS," AND DRAFT NUREG-XXXX, "SEISMIC CONSIDERATIONS FOR THE TRANSITION BREAK SIZE"

Dear Chairman Klein:

During the 548th meeting of the Advisory Committee on Reactor Safeguards, December 6-8, 2007, we reviewed the draft final NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and draft NUREG-XXXX, "Seismic Considerations for the Transition Break Size." Our Reliability and Probabilistic Risk Assessment Subcommittee reviewed this matter during a meeting on November 27, 2007. During these reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. NUREG-1829 on estimating LOCA frequencies through the expert elicitation process, and the NUREG report on seismic considerations for the transition break size (TBS) should be published.

2. Regulatory decisions should be based on the totality of the results from the sensitivity studies rather than the results from individual methods of expert judgment aggregation.

3. A set of consistent guidelines should be established for the elicitation and aggregation of expert judgments including the performance of sensitivity studies. These guidelines should be used throughout the agency.

DISCUSSION

The Transition Break Size

An essential element of the proposed risk-informed alternative to the existing 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear reactors," is the concept of "transition break size." In a Staff Requirements Memorandum dated July 1, 2004, the Commission directed the staff to define the TBS as that break size that has a frequency of occurrence of about 10^{-5} per reactor year. Loss-of-coolant accidents due to breaks smaller than the TBS are expected to have frequencies of occurrence greater than 10^{-5} per reactor year and would remain design-basis accidents (DBAs). They would be analyzed using the methods, assumptions, and criteria currently prescribed in 10 CFR 50.46. Accidents due to breaks larger than the TBS are expected to have lower frequencies of occurrence and would become beyond design-basis accidents. Consequently, they would be analyzed without the additional conservatisms associated with DBAs.

The size of the transition break cannot be determined from operating experience or mechanistic calculations alone. We must rely on expert judgment supported by the available evidence and analyses. The resulting uncertainty is managed by selecting a conservative TBS and by ensuring that breaks greater than the TBS can be mitigated, i.e., by invoking a structuralist defense-in-depth principle for this range of break sizes.

The staff has produced two reports, NUREG-1829 and NUREG-XXXX, which help to provide the basis for selecting a conservative TBS. NUREG-1829 presents the results of a formal expert evaluation of the state of the art and NUREG-XXXX focuses on the impact of seismic events on TBS.

The authors of NUREG-1829 acknowledge the limitations of expert opinion elicitation processes as well as the fact that one could use several ways to aggregate these opinions. The study provides the results of a series of sensitivity studies that help decision makers understand the magnitude of the uncertainties in the TBS. As expected, many public comments addressed issues associated with individual aggregation methods. Although the authors of NUREG-1829 have provided reasonable answers to these comments, it is the totality of results from the sensitivity studies that shapes our state of knowledge rather than the results from individual methods.

NUREG-XXXX provides additional insights by investigating seismically induced failures in unflawed piping, flawed piping, and indirect piping failures caused by the failure of other components and supports. The results of the study indicate that, for Pressurized-Water Reactors (PWRs) east of the Rocky Mountains, the likelihood of seismically induced failures in unflawed piping of size greater than the TBS is very low for earthquakes with 10^{-5} and 10^{-6} annual probabilities of accedence. Even for pipes with long surface flaws, the depths of these flaws must be greater than 30-40% of the wall thickness for a high likelihood of failure during such earthquakes. Inspection programs, leak detection systems, and other measures taken to eliminate failure mechanisms such as stress corrosion cracking should make the likelihood of such cracks very low.

Both of these NUREG reports provide results and insights that can form the basis for the selection of the TBS. They should be published.

Expert Judgment

Using expert judgments to evaluate the state-of-the-art in issues that cannot be resolved by statistical or mechanistic methods is an approach that has been pioneered by the NRC. These issues usually involve rare events and divergence of opinions among knowledgeable investigators and practitioners.

The Senior Seismic Hazard Analysis Committee (SSHAC) investigated the paralyzing differences in probabilistic seismic hazards between the NRC and the Electric Power Research Institute (EPRI) (NUREG/CR-6372). SSHAC stated: "The Committee's most important conclusion is that differences in PSHA [Probabilistic Seismic Hazard Analysis] results are due to procedural rather than technical differences. Thus, in addition to providing a detailed documentation on state-of-the-art elements of a PSHA, this report provides a series of procedural recommendations." These recommendations dealt with the use of expert judgments.

It is worth pointing out that the SSHAC work was sponsored by the NRC, DOE, and EPRI. It was reviewed by a National Research Council Panel, which stated: "The panel believes that the SSHAC report makes a solid contribution to the methodology of hazard analysis, especially in the use of expert opinion."

The goal of the SSHAC guidance is to develop a probability distribution representing the state of knowledge of the informed technical community. To achieve this, the SSHAC guidance recommends that the appropriate method for aggregating expert estimates is one that encourages complete sharing of information and full consideration and discussion of the evidence supporting each expert's judgment. The approach asks the experts to state their own opinions first and then defend their positions, based on all the evidence at their disposal. This sharing of evidence puts the experts on equal footing and ensures that they understand the bases for the judgments of others. The approach then asks each expert to take on a new role, that of evaluator.

Under this reframing of the problem, the experts, acting as evaluators, propose probability distributions reflecting the state of knowledge of the informed technical community. This is done after significant interaction has taken place among them. Ideally, the experts agree upon a consensus distribution. The SSHAC report recommends that the results of any mechanistic aggregation of opinions be scrutinized and modified if they are inconsistent with the overall judgment of the experts and the study integrators. The National Research Council Panel agrees and states: "Do not accept the results of a mechanical combination rule unless they are consistent with judgment."

We note that this elicitation process gives considerable attention to the extreme values of the distribution, challenging each evaluator to consider all factors that could drive the results higher or lower. We acknowledge that this approach requires very effective control of bias and the interaction among experts, but that is true of all elicitation efforts.

For their baseline methodology, the authors of NUREG-1829 take the geometric average of each set (lower, median, and upper bound) of the expert supplied percentiles. This averaging is performed after the experts have exchanged views and their opinions have been adjusted for possible bias by the study integrators. The authors subscribe to the view that a group estimate should be defined as a value near the center of the group opinion; i.e., their approach focuses on getting the center value of the estimate right. In this study, the geometric mean does produce a value near the center of the group estimates[1].

The method called "Mixture Distribution Aggregation" in NUREG-1829 is the mechanistic aggregation approach recommended by SSHAC and was used by the team that developed NUREG-1150. In this method, the composite probability distribution of the frequency of a break of a certain size is the arithmetic average of the panelists' probability distributions (not of the percentiles).

In response to comments provided during the ACRS Subcommittee meeting, the authors of NUREG-1829 also produced results using the Mixture Distribution Aggregation method. The panelists went through a significant exchange of views. They were not asked, however, to act as evaluators, i.e., to produce distributions that reflect the views of the informed technical community; their distributions represented their own uncertainties. The authors of NUREG-1829 state: "The mixture distribution approach does not attempt to develop aggregated estimates that represent the central group opinion as does the baseline methodology, but rather attempts to exhibit the full range of variability among the panelist responses." We believe that employing a method that "exhibits the full range of variability among the panelist responses" is important and useful for a study whose results will form the basis of regulations. In these cases, understanding the breadth of informed opinion is more important than central estimates.

There is no compelling mathematical reason supporting a particular aggregation method[2]. Each requires assumptions that may or may not be justified. We find the attempt to develop a consensus distribution that represents the technical community's views intellectually appealing. To help the experts develop consensus, sensitivity studies need to be conducted including possible adjustment for bias and various aggregation schemes.

The elicitation of expert judgments is a process that the NRC will continue to use to inform regulatory decision making involving important matters. The method employed to process these judgments cannot be left up to the discretion of the team performing each new study. The Office of Nuclear Regulatory Research should investigate the existing methods and propose a set of consistent guidelines to be used throughout the agency.

Sincerely,
/RA/
William J. Shack
Chairman

1 It is important to recognize that the geometric average of percentiles can be controlled by a very low outlier. Similarly, the arithmetic average of percentiles can be controlled by a high outlier. In the current study, there are no extreme low outliers for the final evaluations; therefore, the geometric mean gives a fair estimate of the *center* of the distributions.

2. The theoretically correct method for combining expert judgments is to treat them as evidence in a Bayesian framework. To date, this approach is impractical. Development of a consensus distribution reflecting the breadth of concerns of the technical community is an excellent way to select an informed prior distribution for later Bayesian analysis.

REFERENCES

1. U.S. Nuclear Regulatory Commission, NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," and associated Appendixes A through M, 2005.

2. U.S. Nuclear Regulatory Commission, NUREG-XXXX, "Seismic Considerations for the Transition Break Size," 2005.

3. U.S. Nuclear Regulatory Commission, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," 1990.

4. U.S. Nuclear Regulatory Commission, NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," [Prepared by Senior Seismic Hazard Analysis Committee (SSHAC)], 1997.

5. Staff Requirements Memorandum from Annette L. Vietti-Cook, Secretary, U.S. Nuclear Regulatory Commission, to Luis A. Reyes, Executive Director for Operations, U.S. Nuclear Regulatory Commission, "Staff Requirements -SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power," dated July 1, 2004.

Attachment 2: ACRS Letter of November 16, 2006

ACRSR-2223

November 16, 2006

The Honorable Dale E. Klein
Chairman
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

SUBJECT: DRAFT FINAL RULE TO RISK-INFORM 10 CFR 50.46, "ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT WATER NUCLEAR POWER REACTORS"

Dear Chairman Klein:

During the 537th meeting of the Advisory Committee on Reactor Safeguards, November 1-3, 2006, we met with representatives of the NRC staff and the Boiling Water Reactor (BWR) Owners' Group to discuss the draft final rule to risk-inform 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," (the Rule). We also had the benefit of the documents referenced.

RECOMMENDATIONS

1. The Rule to risk-inform 10 CFR 50.46 should not be issued in its current form. It should be revised to strengthen the assurance of defense in depth for breaks beyond the transition break size (TBS). Such assurance would reduce concerns about uncertainties in determining the TBS.

2. The revision of draft NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," to include changes resulting from the resolution of public comments should be completed before the revised Rule is issued. This state-of-the-art review on the estimation of break size frequencies is an essential part of the technical basis for the Rule.

3. The interpretation that the Rule limits the total increase in core damage frequency (CDF) resulting from all changes in a plant that adopts the Rule to be "small" (i.e., $<10^{-5}$/yr) represents a significant departure from the current guidance for risk-informed regulation and should be reviewed for its implications.

DISCUSSION

In response to a Staff Requirements Memorandum (SRM) dated July 1, 2004, the staff has developed an alternative set of risk-informed requirements for emergency core cooling systems (ECCS). Licensees may voluntarily choose to comply with these requirements in lieu of meeting the existing requirements in 10 CFR 50.46. The Rule divides the spectrum of LOCA break sizes into two regions. The demarcation between the two regions is called a “transition break size.” The first region includes small breaks up to and including the TBS. The second region includes breaks larger than the TBS up to and including the double-ended guillotine break (DEGB) of the largest reactor coolant system pipe.

Because pipe breaks in the smaller break size region are considered more likely than pipe breaks in the larger break size region, each region would be subject to different ECCS requirements. Loss-of-coolant accidents in the smaller break size region would be analyzed using the methods, assumptions, and criteria currently used for LOCA analysis; accidents in the larger break size region would be analyzed using less stringent methods, assumptions, and criteria due to their lower likelihood of occurrence. Although LOCAs for break sizes larger than the TBS would become “beyond design basis accidents,” the Rule requires that licensees maintain the ability to mitigate all LOCAs up to and including the DEGB of the largest reactor coolant system pipe.

The fundamental principles of a risk-informed regulation should be to ensure that any increases in risk associated with a change are small, that changes are consistent with the defense-in-depth philosophy, and that adequate safety margins are maintained. Regulatory Guide 1.174 provides quantitative criteria for assessing changes in risk, but its guidance on ensuring consistency with the defense-in-depth philosophy and maintaining adequate safety margins is more subject to engineering judgment.

Probabilistic risk assessments of internal events typically show that large-break LOCAs (LBLOCAs) are relatively small contributors to CDF. The results in draft NUREG-1829 suggest that the contribution to CDF from breaks larger than the TBS proposed in the Rule is a small fraction of the already small contribution to CDF due to all LBLOCAs. Thus, the requirements for mitigation capabilities for breaks beyond the TBS should be based on defense-in-depth considerations to provide margin against unanticipated degradation phenomena, human errors, extremely large loads such as those associated with earthquakes beyond the safe shutdown earthquake, and other unanticipated events. The degree of defense in depth required can only be determined by judgment based on experience and best attempts to quantify uncertainties.

The Rule requires an analysis to demonstrate mitigation for breaks greater than the TBS, up to the DEGB of the largest pipe in the reactor coolant system. The requirements in the Rule provide a degree of assurance of this mitigation. It is our judgment, however, that the Rule should impose additional requirements to strengthen this assurance.

Because the Rule now defines pipe breaks greater than the TBS as “beyond design basis,” any equipment required solely to mitigate such breaks may no longer be safety-related and could be subject to less stringent maintenance and inspection requirements that could adversely affect its reliability. Such equipment could even be removed from technical specifications that control its availability. We agree that the low likelihood of breaks greater than the TBS justifies a relaxation in the requirements for mitigating such events, but this relaxation should instead result from the removal of additional requirements that make such events even more unlikely, such as the simultaneous loss-of-offsite-power (LOOP) and the assumption of the worst single failure. Confidence in the reliability and availability of the equipment needed to mitigate such breaks is important not only for defense in depth, but also for maintaining safety margins for breaks smaller than the TBS.

The Rule also provides restrictions on the unavailability of the non-safety-related equipment needed to mitigate breaks beyond the TBS, but it imposes no other requirements. We believe that the equipment needed to mitigate these breaks deserves some special treatment and control. The staff has dealt with the regulatory treatment of non-safety systems in other contexts, and similar approaches would be appropriate here.

The Rule should also increase confidence in the ability to mitigate breaks greater than the TBS by requiring licensees to submit the codes used for the analyses of breaks beyond the TBS to the NRC for review and approval.

The Rule is an enabling rule that will permit licensees to make changes that increase operational flexibility and reduce regulatory burden, which could result in increases or decreases in risk. The Rule contains a risk-informed change process that will control all changes in risk that occur after a licensee adopts the Rule. The risk-informed change process in the Rule uses the current 10 CFR 50.59 change process and the 10 CFR 50.65 maintenance rule categorization to screen changes that can impact risk.

However, as currently envisioned by the staff, it allows the licensee in some cases to implement changes that have a ΔCDF greater than 10^{-6}/yr but less than 10^{-5}/yr without prior review by the staff. Regulatory Guide 1.174 would typically allow such changes only if the total CDF, including external events and low-power/shutdown events, is less than 10^{-4}/yr. Licensees should submit such changes to the staff for prior review and approval. Licensees could still implement changes that result in a ΔCDF < 10^{-6}/yr without prior review and should track the quantified changes in CDF in the 24 month report.

The Rule requires that the total increase in CDF resulting from all changes in a plant that adopts the Rule be "small" (i.e., < 10^{-5}/yr). This "cap" on the increase in risk applies regardless of whether the changes in CDF result from changes related to 10 CFR 50.46. This represents a significant departure from the current guidance for risk-informed regulation and should be reviewed for its implications.

Maintaining sufficient safety margin is another important element of risk-informed regulation that is not treated quantitatively in Regulatory Guide 1.174. It is likely that, with this Rule, the NRC will find requests for additional power uprates at pressurized water reactors (PWRs) acceptable. However, the uprates will clearly decrease safety margins, even for breaks below the TBS. The Rule currently contains acceptance criteria for fuel cladding performance under LOCA conditions based on the current 10 CFR 50.46. The Office of Nuclear Regulatory Research is now completing an examination of the adequacy of these criteria for high-burnup fuel. The adequacy of the acceptance criteria for cladding performance is important to maintain adequate safety margins. The Rule should not be finalized until the fuel cladding acceptance criteria for LOCAs involving breaks at or below the TBS are reviewed and/or revised to assure their adequacy for the higher burnup fuel and more demanding conditions of current reactor operating conditions. Alternatively, the acceptance criteria in the Rule could be expressed in terms of general requirements, such as a high degree of confidence in maintaining a coolable geometry and retaining some ductility in the cladding. Specific cladding and core criteria could be placed in the associated regulatory guide.

An important element in the selection of the TBS is the state-of-the-art review of break size frequencies conducted by the Office of Nuclear Regulatory Research, documented in draft NUREG-1829. There is substantial uncertainty in the determination of these frequencies. If there is a high degree of assurance that breaks greater than the TBS can be mitigated, the impact of this uncertainty on the selection of the TBS is substantially reduced. The selection of the TBS could then include consideration of the benefits of small changes in the break size. For example, the current TBS for BWRs inhibits implementation of longer diesel start-up times, which are almost universally agreed to lead to improved emergency diesel generator operability. If the staff strengthens the defense in depth for breaks greater than the TBS, the TBS proposed by the BWR Owners' Group could be acceptable and would not be inconsistent with the results in draft NUREG-1829.

Although the Rule defines TBSs for BWRs and PWRs, licensees should not presume that these automatically apply to all plants. As part of the adoption of the Rule, licensees should have to demonstrate that the results in draft NUREG-1829 are applicable to their plants. The staff should provide guidance for this demonstration in the associated regulatory guide. As part of this demonstration, licensees should demonstrate that the reactor coolant system piping of diameter corresponding to the TBS or larger meets the deterministic requirements currently used to credit leak-before break for dynamic analysis of reactor coolant piping. Such demonstrations will provide additional assurance of the very low likelihood of failures greater than the TBS. Many plants should have already performed such analyses.

The staff is revising draft NUREG-1829 to incorporate, as appropriate, the changes resulting from the resolution of public comments. This revision should be completed prior to issuing the revised Rule.

For internal events, the occurrence of a LBLOCA and a LOOP can generally be considered as independent events, and thus the simultaneous occurrence of a break greater than the TBS and a LOOP is a very unlikely event. However, a LOOP is very likely for any seismic event that is large enough to induce failures in reactor piping systems. As part of its effort to establish the TBS, the staff performed a study of the likelihood of seismically induced failures in unflawed piping, flawed piping, and indirect failures of other components and component supports that could lead to piping failure.

The study focused on piping systems in PWRs east of the Rocky Mountains. We have not yet completed our review of the staff's study in this area. However, the results of the study indicate that for these plants the likelihood of seismically induced failures in unflawed piping of size greater than the TBS is very low for earthquakes with 10^{-5} and 10^{-6} annual probabilities of accedence. Even for pipes with long surface flaws, the depths of these flaws must be greater than 30-40% of the wall thickness for a high likelihood of failure during such earthquakes. Inspection programs, leak detection systems, and other measures taken to eliminate failure mechanisms such as stress corrosion cracking should make the likelihood of such cracks very low. Because seismic hazards are very plant specific, licensees adopting the Rule will have to demonstrate that the results developed by the staff bound the likelihood of seismically induced failure in their plants. For unflawed piping, the results of the individual plant examination of external events (IPEEE) program may provide the needed information. Licensees may have to perform additional calculations to demonstrate a comparable robustness of flawed piping.

Although substantial progress has been made in the development of a risk-informed 10 CFR 50.46, the Rule should not be issued in its current form. It would be significantly strengthened by addressing the issues raised in this report.

Additional comments by ACRS Member Graham B. Wallis and ACRS Member Sanjoy Banerjee are presented below.

Sincerely,

/RA/
Graham B. Wallis
Chairman

Additional Comments from ACRS Member Graham B. Wallis

My colleagues have suggested some significant improvements to the draft final rule, which I support, if it should be issued as final.

However, I am not persuaded that an adequate case has been made for this rule or that its consequences have been sufficiently explored.

The probabilities for breaks of various sizes, as assessed in draft NUREG-1829, can be accommodated within the framework of the existing rule's "realistic (best estimate)" alternative without any new rulemaking. This can be done in numerous ways while preserving suitable caution and defense in depth. The details can be worked out between the staff and licensees through an evolutionary process that includes thorough consideration of practicality, enforcement, technical uncertainties, benefits, and risks.

Additional Comments from ACRS Member Sanjoy Banerjee

I support the Recommendations in the ACRS letter regarding the draft final rule to risk inform 10 CFR 50.46, but would add the further Recommendation that the draft NUREG-1829 be externally peer reviewed before being issued.

I have arrived at this Recommendation after reviewing NUREG-1829 and transcripts of 5 meetings regarding the work contained in it, held by the ACRS Regulatory Policies and Practices Subcommittee from 11/21/03 to 11/16/04. Based on this, it is my opinion that the quality of the NUREG and the credibility of its conclusions, would be substantially enhanced by eliciting, and responding to, comments from external and independent peer reviewers. This point was also raised at several of the ACRS Subcommittee meetings, but no substantive external peer review appears to have been conducted.

Amongst the several issues which, in my opinion, may be elucidated by such a review are the wide divergence in the initial estimates for various LOCA frequencies, and the methods used to narrow the range of uncertainty in the final results from which the conclusions are drawn.

References:

1. Memorandum from Michael Marshall Jr., Acting Branch Chief, Financial, Policy, and Rulemaking Branch, Division of Policy and Rulemaking, Office of Nuclear Reactor Regulation, to Dr. Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, “Advisory Committee on Reactor Safeguards Review of the Draft Final Rule to Amend 10 CFR 50.46, ‘Risk-informed changes to loss-of coolant accident technical requirements’,” dated October 26, 2006.

2. Report from Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, to Nils. J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, “Proposed Rulemaking to Modify 10 CFR 50.46, ‘Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements’,” dated March 14, 2005.

3. Report from Mario V. Bonaca, Chairman, Advisory Committee on Reactor Safeguards, to Nils. J. Diaz, Chairman, U.S. Nuclear Regulatory Commission, “SECY-04-0037, ‘Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power’,” dated April 27, 2004.

4. Staff Requirement Memorandum from Annette L. Vietti-Cook, Secretary, U.S. Nuclear Regulatory Commission, to Luis A. Reyes, Executive Director for Operations, U.S. Nuclear Regulatory Commission, “Staff Requirements - SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss-of-Offsite Power,” dated July 1, 2004.

5. U.S. Nuclear Regulatory Commission, NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” Draft Report for Comment, June 2005.

6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” November 2002.

7. U.S. Nuclear Regulatory Commission, “Seismic Considerations for the Transition Break Size,” December 2005, ADAMS ML053470439.

8. Letter from Randy C. Bunt, Chair, BWR Owners' Group, to Graham B. Wallis, Chairman, Advisory Committee on Reactor Safeguards, “Draft Final Rule Language, Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements, ADAMS Accession NO. ML062760146, dated October 3, 2006,” dated October 13, 2006.

Attachment 3: ACRS Letter of March 14, 2005

March 14, 2005

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PROPOSED RULEMAKING TO MODIFY 10 CFR 50.46, "RISK-INFORMED CHANGES TO LOSS-OF-COOLANT ACCIDENT TECHNICAL REQUIREMENTS"

Dear Chairman Diaz:

During the 520th meeting of the Advisory Committee on Reactor Safeguards on March 3-5, 2005, we reviewed the proposed rule for a voluntary alternative to 10 CFR 50.46, "Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements," (Reference 1). We also reviewed a draft version of a proposed rule (Reference 2) during the 518th meeting on December 2-4, 2004 and issued a letter on December 17, 2004 (Reference 3). During these reviews, we had the benefit of discussions with the NRC staff, the Nuclear Energy Institute, Westinghouse Owners Group and members of the public. We also had the benefit of the documents referenced.

RECOMMENDATION

The proposed rule for risk-informing 10 CFR 50.46 should be released for public comment.

DISCUSSION

The current proposed rule is consistent with the first two recommendations of our December 17, 2004 letter (Reference 3). It contains requirements intended to provide reasonable assurance of a coolable core geometry for breaks up to the double-ended guillotine break of the largest pipe in the reactor coolant system and permits operation only in configurations for which such capability has been demonstrated. The transition break size in the current version of the rule is equivalent to a single-ended rupture of the largest pipe attached to the reactor coolant system rather than the double-ended rupture in the earlier version.

The staff agrees with our recommendation that a better quantitative understanding of the possible risk benefits of a smaller transition break size is needed before finalizing the selection of the transition break size. The staff is attempting to identify areas where quantification of potential benefits might be meaningful. We have also heard a presentation from the industry on efforts to develop quantified estimates of the safety benefits associated with a smaller transition break size. These estimates are expected to be available during the rule comment period.

One of the changes in the proposed rule from the one that we reviewed in December is the omission of a quantitative criterion for the likelihood of late containment failure. We continue to believe that this should be considered in determining changes in risk due to changes in the licensing basis. We accept, however, that this is not an issue unique to changes in the licensing basis made possible by a risk-informed 10 CFR 50.46, and should be dealt with in the more general context of a revision to Regulatory Guide (RG) 1.174.

The proposed rule is an enabling rule. A licensee who wishes to make changes to its facility, technical specifications, or procedures based on the new rule will need to submit an application for a license amendment to allow such changes. The process of evaluating the risk due to such changes is critical to risk-informing 10 CFR 50.46.

Since 1998, the NRC has been evaluating the acceptability of risk-informed changes to the licensing basis using RG 1.174. The guidance and acceptance criteria in RG 1.174 are intended to ensure that any increases in risk associated with changes to the licensing basis are small and that sufficient defense in depth and safety margins are maintained to address uncertainties.

The staff argues that it is necessary to include some of the high-level guidance of RG 1.174 in the proposed rule, and a new regulatory guide would be developed to provide additional guidance. The language in the draft proposed rule and in the statement of considerations is consistent with RG 1.174 (including the bundling of changes in risk due to unrelated changes in the licensing basis). It is not clear why the process of accepting the changes to the licensing basis that will be possible due to changes in 10 CFR 50.46 should be specified in the rule itself when it is already in RG 1.174, which is currently in use for evaluating risk-informed changes to the licensing basis. As part of the public comment process, input should be sought on the need to incorporate in the rule requirements for the acceptability of changes to the licensing basis and to develop a new regulatory guide for evaluating such changes.

The proposed rule contains provisions intended to ensure that plants that adopt a risk informed 10 CFR 50.46 will still have a capability to mitigate loss-of-coolant accidents beyond the transition break size and permits operation only in configurations for which such capability has been demonstrated. However, the rule provides only high-level requirements for the analytical methods needed to demonstrate such capability and the statement of considerations just outlines a possible approach. The staff is developing a regulatory guide to provide more detailed guidance on acceptable methods for such analyses. The development of this regulatory guide is critical to the success of a risk informed 10 CFR 50.46. We look forward to interacting with the staff on the development of this guide and discussing the draft final rule after resolution of public comments.

Sincerely,

/RA/
Graham B. Wallis
Chairman

References:

1. Memorandum dated December 2, 2004, from Catherine Handy, Program Director, Policy and Rulemaking Program, NRR, to various members NRR, Subject: Office Concurrence

on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (Pre-Decisional).

2. Letter dated February 14, 2005, from Catherine Handy, Program Director, NRR, to multiple addresses, NRR, Subject: Office Concurrence on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (TAC # MB8397) 2004 (Pre-Decisional).

3. Letter dated December 17, 2004, from, Mario V. Bonaca, Chairman, ACRS, to Luis A. Reyes, EDO, NRC, Subject: Risk-Informing 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors

Attachment 4: ACRS Letter of December 17, 2004

December 17, 2004

Mr. Luis A. Reyes
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: RISK-INFORMING 10 CFR 50.46, "ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS FOR LIGHT-WATER NUCLEAR POWER REACTORS"

Dear Mr. Reyes:

During the 518th meeting of the Advisory Committee on Reactor Safeguards on December 2-4, 2004, we reviewed a draft version of a proposed rule for a voluntary alternative to 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water Nuclear Power Reactors" (Reference 1). We also reviewed draft proposed rule language (Reference 2) during the 517th meeting on November 4-6, 2004. Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on October 28-29, 2004.

During these reviews, we had the benefit of discussions with the NRC staff, the Nuclear Energy Institute, Westinghouse Owners Group, and members of the public. We also had the benefit of the documents referenced. Although the proposed rule language has not been finalized, we present our views on some of the basic elements of a risk-informed 10 CFR 50.46.

CONCLUSIONS AND RECOMMENDATIONS

1. A risk-informed 10 CFR 50.46 should maintain defense in depth by including requirements intended to provide reasonable assurance of a coolable core geometry for breaks up to the double-ended guillotine break (DEGB) of the largest pipe in the reactor coolant system.

2. The results of the expert opinion elicitation need to be further reviewed and assessed by the staff before finalizing the selection of the transition break size. Nevertheless, it appears that a transition break size corresponding to the single-ended rupture of the largest pipe attached to the reactor coolant system bounds the range of break sizes corresponding to a frequency of 1×10^{-5}/year.

3. A better quantitative understanding of the possible risk benefits of a smaller transition break size is needed to arrive at a final choice of the transition break size. If the defense-in-depth capability to mitigate breaks greater than the transition break size is maintained, a smaller choice of transition break size may be supportable.

DISCUSSION

Loss-of-coolant accidents (LOCAs) have been the focus of nuclear plant safety since the first commercial reactor designs. LOCAs can arise from many causes, and the current design basis requires the demonstration of the capability to mitigate a spectrum of break sizes up to the DEGB of the largest pipe in the reactor coolant system. Since the Three-Mile Island accident and the earliest probabilistic risk assessments, it has been recognized that small-break LOCAs are more risk significant than large-break LOCAs (LBLOCAs). This has been reflected in operator training, procedures, etc., but it has not been fully reflected in the regulations.

Although the design-basis LBLOCA requirements have led to the development of robust safety systems, the burdens imposed by the design-basis requirement to deal with the DEGB of the largest pipe in the reactor coolant system are not commensurate with its risk importance, and the resulting requirements may have inhibited opportunities to optimize the system response for the entire range of challenges that must be met including those more likely to occur. For example, the current LBLOCA requirements result in rapid diesel start times. The testing necessary to demonstrate that these start times can be achieved increases wear on the diesel and reduces the reliability of the diesel in the case of more risk-important sequences that do not require such rapid start times.

A risk-informed 50.46 rule will be an enabling rule. It will not impose any specific changes that would be made in the design or operation of nuclear power plants. It will permit licensees to make changes that may decrease risk by optimizing system responses to accidents that are more likely to occur, and changes such as power uprates that will result in risk increases.

In a Staff Requirements Memorandum (SRM) dated July 1, 2004, the Commission approved the development of a proposed rule to risk-inform the requirements addressing LBLOCAs. The proposed rule was to use the initiating event frequencies from the expert elicitation process and other relevant information to guide the determination of an appropriate alternative break size.

The staff was also to ensure that any changes to the plant or operating procedures would follow a change process consistent with Regulatory Guide (RG) 1.174. RG 1.174 permits only small increases in risk as long as it is reasonably assured that sufficient defense in depth and margins are maintained.

In our report, dated April 27, 2004, we concluded that the process and criteria in RG 1.174 are appropriate for evaluating the acceptability of changes proposed under a revised rule, but recommended explicit consideration of late release frequency (LRF) in addition to core damage frequency (CDF) and large early release frequency (LERF) to ensure that all of the safety objectives are addressed. The SRM and the proposed rule language posit a process, akin to the current 10 CFR 50.59 process, to permit licensees to make changes that result in "inconsequential" changes in risk without prior NRC review and approval. We agree that a process for making such changes is needed. The staff argues that the existing 10 CFR 50.59 process is not suitable, since it addresses design-basis issues, while the new process must address the acceptability of changes with respect to risk. Additional input on the need for a new change process can be obtained when a draft rule is issued for public comment.

In the proposed rule language, the staff introduces a transition break size (TBS). The TBS is chosen to ensure that the frequency of LOCAs corresponding to breaks larger than the TBS is less than 1×10^{-5}/reactor-year. This frequency is consistent with the goal set in SECY-00-198, Attachment 1, “Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50" for rare initiators and the criterion proposed for a vessel failure frequency due to pressurized thermal shock, when it is recognized that those are unmitigated events, and that a substantial mitigative capability will be maintained for LOCAs beyond the TBS.

For LOCAs corresponding to break sizes smaller than the TBS, the requirements are equivalent to those in the current 10 CFR 50.46. We agree that defense in depth should be maintained through the requirement that sufficient mitigating capability be available to prevent severe core damage (i.e., loss of coolable geometry) for breaks greater than the TBS up to the DEGB of largest pipe in the reactor coolant system. Because of the low frequency of such breaks, it should not be necessary to postulate a simultaneous loss of offsite power and single failure of the most critical component. Credit may be taken for operation of any equipment supported by appropriate availability data. Nominal operating conditions rather than technical specification limits, actual fuel burnup in decay heat predictions, and actual operating peaking factors can be used. Some increase in the degree of core damage beyond that implied in the current 10 CFR 50.46 should also be considered acceptable. The integrity of the reactor containment structure should be maintained using realistically calculated pressure, temperature, and containment capacity.

Because breaks with sizes greater than the TBS are not risk significant, and hence equipment needed to mitigate such breaks might be considered unimportant in 10 CFR 50.65(a)(4) assessments of acceptable configurations, the staff has included additional configuration control requirements to ensure the capability to mitigate such large breaks during all modes of operation when the reactor is critical. We agree that such configuration control requirements are appropriate.

The draft version of a proposed rule discussed with us proposes a TBS that is reactor specific and equivalent in area to a double-ended rupture of the largest pipe attached to the reactor coolant system. For a pressurized water reactor (PWR) this would correspond to surge, shutdown cooling, or safety injection lines that are typically 12-14 inches in diameter Schedule 160 pipe. For a boiling water reactor (BWR) these would be residual heat removal or feedwater lines, which are typically 20 inches in diameter Schedule 80 pipe.

The selection of the TBS requires estimates of LOCA frequencies as a function of break size. The most comprehensive assessment of this information is the expert opinion elicitation conducted by the Office of Nuclear Regulatory Research (RES). We believe that additional work needs to be done to complete the expert opinion elicitation and have issued a separate report on this matter, dated December 10, 2004. Hence, some of our judgments below on the implications of the elicitation must be considered preliminary.

The elicitation sought to develop LOCA frequency estimates for PWR and BW R piping and non-piping passive components. It focused on developing average values for the fleet of operating plants, and thus the uncertainty bounds represent bounds on these average values and not on LOCA frequency estimates for individual plants. Thus they are only applicable to plants that can demonstrate that they have no additional degradation mechanisms, no significant differences in the conditions that produce degradation, and no significant differences in their capability to detect degradation than is typical of most plants in the fleet.

The elicitation also did not consider the impact on the frequency of LBLOCAs of the power uprates that could likely result from a risk-informed 10 CFR 50.46. Such uprates could have substantial impacts on flow-assisted corrosion rates in secondary systems in PWRs. PWR power uprates are not likely to have a significant impact on primary system piping. The BWR feedwater piping is susceptible to flow-assisted corrosion. The potential impact of power uprates on LOCA frequency will have to be addressed as part of the licensing reviews of the uprates.

In its efforts to develop a new rule, the staff has considered other potential mechanisms that could cause pipe failure that were not explicitly considered in the expert elicitation process such as active system LOCAs, seismic loading, heavy load drops, and LOCA–induced water hammer loading. No active system LOCAs were identified that would result in break sizes greater than about 4 inches. The staff concluded that heavy load drops would have little effect on the choice of the TBS. For seismic loads with magnitudes of occurrence of 1 x10^{-5}/yr, the staff has found that undegraded piping or piping with minor degradation has little likelihood of failure. More severely degraded piping could fail under such seismic loads, but the relatively low frequency of degradation in primary piping and the low frequency of the expected loading suggest that these will not have a significant impact on the choice of the TBS. RES is still performing some confirmatory research in this area.

Thus it appears that the expert elicitation has addressed the important potential contributors to the LBLOCA frequency. However, the choice of a TBS is strongly dependent on how the uncertainties in the elicitation are addressed.

For PWRs the break size (i.e., the equivalent diameter of the flow area) corresponding to a frequency of 1 x 10^{-5}/reactor-year from the expert opinion elicitation reported in Reference 3 ranges from 4-11 inches depending on the approaches used to aggregate and assess the expert opinions, whether the mean or 95th percentiles of the resulting distributions are used, and how the results are interpolated between the discrete break sizes in the elicitation.

The staff's choice of a break size corresponding to a double-ended break of the largest piping attached to the reactor coolant system appears to conservatively bound the range of values determined through the elicitation. The large disparity in size between the main reactor coolant system piping and the largest attached piping also provides an argument for the selection of the failure of the attached piping as the TBS.

Although uncertainties in the elicitation could affect the choice of the TBS in the range of sizes up to the diameter of the attached piping, the physics of the failure processes give a very-high confidence in the low-failure probability of the main coolant piping. The staff notes that this choice for the TBS makes it very unlikely that any future reevaluations of the break frequency versus break size will result in the need for licensees to make any plant modifications as a result of implementing the revised 10 CFR 50.46 thus helping to ensure a more stable regulatory environment. It also bounds the flow areas associated with breaks of components such as bolted connections. Although these connections were considered in the elicitation, they are more likely to be affected by human errors and are thus perhaps subject to even greater uncertainty than the piping failure.

Based on our current understanding of the results of the expert opinion elicitation, it appears that the choice of the double ended rupture is overly conservative. Choosing the TBS as the diameter of the largest attached pipe (i.e., a single-ended rupture) would still bound the elicitation results and would be consistent with the argument that the failure of the main coolant piping is much more unlikely than the failure of the smaller attached piping. If the defense-in depth capability to mitigate breaks greater than the TBS is maintained, a less conservative choice of TBS (e.g., one based on the mean value of the final "best estimate" distribution from the elicitation) may also be supportable.

A better quantitative understanding of the impact of the TBS on parameters, such as required diesel start time, is needed to help optimize the choice of a TBS to balance the defense in depth provided by the larger TBS in any new draft rule with the possible risk benefits of smaller break sizes. Since much of this may be plant specific and will require detailed plant information, it may have to be sought when a draft rule is issued for public comment. Any discussion of risk benefits should also include consideration of the impact of power uprates, which are the likely consequence of a risk-informed 10 CFR 50.46, on such risk benefits.

We would like to review any new draft rule before it is issued for public comment.

Sincerely,

/RA/
Mario V. Bonaca
Chairman

References:

1. Memorandum dated December 2, 2004, from Catherine Handy, Program Director, Policy and Rule making Program, NRR, to various members NRR, Subject: Office Concurrence on Proposed Rule - Risk Informed Changes to Loss-of-Coolant Accident Technical Requirements (Pre-Decisional For Internal ACRS Use Only).

2. Memorandum dated October 14, 2004, from Catherine Handy, Program Director, Policy and Rulemaking Program, NRR, to John T. Larkins, Executive Director, ACRS, Subject: Review of Risk-Informed 10 CFR 50.46 Proposed Rule Executive Summary and Draft Proposed Rule Language (Pre-Decisional For Internal ACRS Use Only).

3. Staff Requirement Memorandum dated July 1, 2004, from Annette L. Vietti-Cook, SECY, NRC to Luis A. Reyes, EDO, NRC Subject: SECY-04-0037 - Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss of-Offsite Power.

4. Regulatory Guide 1.174, entitled, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Rev. 1, Office of Nuclear Regulatory Research, November 2002.

5. ACRS Report dated April 27, 2004, from Mario V. Bonaca, Chairman, ACRS to Nils J. Diaz, Chairman, NRC, Subject: SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA with Coincident Loss of-Offsite Power."

6. 10 CFR 50.59, "Changes, tests, and experiments."

7. 10 CFR 50.65(a)(4), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."

8. ACRS Report dated December 10, 2004, from Mario V. Bonaca, Chairman, ACRS to Nils J. Diaz, Chairman, NRC, Subject: Estimating Loss-of-Coolant Accident Frequencies through the Elicitation Process.

9. Memorandum dated November 4, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive Director, ACRS, Subject: Transmittal of Draft NUREG on Passive System LOCA Frequency Development for use in Risk-Informed Revision of 10 CFR 50.46, Appendix K to Part 50, and GDC and Appendices (Pre-Decisional For Internal ACRS Use Only

Attachment 5: ACRS Letter of December 10, 2004

December 10, 2004

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: ESTIMATING LOSS-OF-COOLANT ACCIDENT FREQUENCIES THROUGH THE ELICITATION PROCESS

Dear Chairman Diaz:

During the 518th meeting of the Advisory Committee on Reactor Safeguards on December 2-4, 2004, we reviewed the draft NUREG Report, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies through the Elicitation Process," (Reference 1). Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on November 16, 2004. During these reviews, we had the benefit of discussions with the NRC staff and of the documents referenced.

RECOMMENDATION

- The draft NUREG Report should be revised prior to being issued for public comment.

DISCUSSION

In a staff requirements memorandum (SRM) dated March 31, 2003 (Reference 2), the Commission directed the staff to develop a risk-informed alternative to the current requirements in 10 CFR 50.46 related to the analysis of the performance of emergency core cooling systems (ECCS) during LOCAs. The focus of this effort is the selection of a risk-informed transition break size (TBS) for the alternative design-basis LOCA. In an SRM dated July 1, 2004 (Reference 3), the Commission directed the staff to use LOCA frequencies derived from an expert-opinion elicitation process, supported by historical data and fracture mechanics and other relevant information to determine an appropriate alternative break size. This alternative break size could be the break size that has a mean frequency of occurrence of 10^{-5} per reactor year.

Expert-opinion-based probability distributions of uncertain quantities have been used extensively in probabilistic risk assessments (PRAs) starting with WASH-1400 (Reference 4). The NUREG-1150 studies (Reference 5) formalized the process of elicitation and utilization of expert judgments. Later, major studies sponsored by both government and industry refined the process and applied it to seismic risk assessments (References 6 and 7).

Generating distributions from expert opinions involves the selection of the experts, elicitation of their judgments, and the processing of the individual judgments to produce a composite distribution.

An important question is what kinds of sources of uncertainties the expert-opinion produced distribution of LOCA frequencies should represent. Ideally, this distribution should reflect the uncertainties due to all scenarios and mechanisms with the potential of causing or contributing to a LOCA. Plant-to-plant variability is an important source of uncertainty. In addition, these uncertainties should reflect the opinions of the expert community at large (i.e., the composite distribution should represent the uncertainties in the state of the art).

Of course, this ideal situation is very difficult to achieve. It is impossible to elicit the opinions of the whole community of experts and the analysts have to rely on a group of experts that is representative of the range of the community's views. The expert panel in this study was selected carefully to represent a broad range of expertise.

In the elicitation process, it is very important that the analysts ask the experts questions that will lead to the development of a composite distribution useful to the decision makers. The experts must fully understand the questions and the underlying assumptions. In this context, we have identified several issues that must be addressed.

The Report does a good job describing the limitations of the results with respect to the scenarios and mechanisms considered. The elicitation process assumed normal plant operational cycles and did not consider the effects of operating profile changes (e.g., due to power uprates). The effects of "rarer" transients, such as seismic events, were also not considered.

It is unclear to what extent the experts considered plant-to-plant variability. The Report states that the elicitation was focused on developing generic, or average, values for the fleet of plants. The panelists were instructed to account for broad plant specific factors. It states further that "the uncertainty bounds do not represent LOCA frequency estimates for individual plants that deviate from the generic values." We conclude that plant-to-plant variability may not be fully reflected in the composite distribution. This conclusion is consistent with the statement in Section H-1 that "Several panelists expressed that safety culture deficiencies at a single plant could increase the LOCA frequencies at that plant by a factor of 10 or more."

The decision makers will have to compensate for the uncertainties created by these limitations by evaluating their impact and resorting to structuralist defense-in-depth measures (e.g., by adding conservatism to the ultimate results of the study). The Report should include a better explanation of what a generic frequency value for the fleet of plants means and to what extent plant-to-plant variability affected the results.

The LOCA size categories are defined by determining an effective break size using correlations that relate break size to the flow rate associated with the break. We were told that some experts assumed that the calculated break size corresponded to double the flow rate while others did not. The question is whether one uses the flow rate from one end of the severed pipe or this flow rate is doubled to include coolant loss from both ends. The analysts should correct the results to make them consistent. The Report should state clearly what the understanding of the experts was when they answered questions about the LOCA size categories.
The Report acknowledges that possible ways for correcting the individual expert opinions to compensate for potential biases and the method of aggregation of these opinions can have a significant impact on the results. Sensitivity analyses are presented to show the impact of a number of approaches.

The aggregation method chosen is what the Report calls "geometric" averaging, e.g., the group's 95^{th} percentile is taken to be the nth root of the product of the 95^{th} percentiles provided by n experts. The results from "arithmetic" averaging are also presented as a sensitivity analysis. This means that the group's estimate is taken to be the sum of the individual estimates divided by n. We note that these averaging methods deal with the characteristic values of the individual distributions directly [i.e., the group median and the group 95th percent are the geometric (or arithmetic) average of the individual medians and 95th percentiles, respectively]. This practice is at variance with the methods employed in References 5-7, in which the arithmetic averaging method is applied to the probability distributions of the experts.

As we stated above, the analysts performed numerous sensitivity analyses. Yet, the Executive Summary lists only the "baseline" results and states: "This study does not recommend whether the LOCA frequency estimates corresponding to the baseline or a particular sensitivity analysis should be used in applications." By not stating what, in their judgment, the most appropriate distribution is, the analysts place an extraordinary burden on the users of the results who are generally not familiar with the intricacies of expert opinion elicitation and aggregation. The final distribution reported in the Executive Summary should be the composite distribution that the analysts, based on the sensitivity analyses, believe represents the expert community's current state of knowledge regarding LOCA frequencies.[1] Providing such a distribution would also be consistent with PRA practice, which utilizes epistemic distributions for the frequencies of initiating events (in this case, LOCA frequencies) and not confidence intervals for individual percentiles. Thus, the results would be useful to a broader class of applications than just the selection of the TBS.

During our December, 2004 meeting, the analysts presented to us results from the aggregation method that averages probability distributions (what they called a "mixture distribution"). They also provided us with a revised chapter of the Report. It is evident that this work is still in progress and is not ready for public comment.

We look forward to reviewing the Report after the staff responds to our comments.

Sincerely,

/RA/
Mario V. Bonaca
Chairman

1. This means that the analysts should act as a Technical Facilitator/Integrator, a concept described in detail in NUREG/CR-6372 (Reference 7).

References:

1. Memorandum dated November 4, 2004, from Michael E. Mayfield, Director, Division of Engineering Technology, RES, to John T. Larkins, Executive D irector, ACRS, Subject: Transmittal of Draft NUREG on Passive System LOCA Frequency Development for use in Risk-Informed Revision of 10 CFR 50.46, Appendix K to Part 50, and GDC and Appendices (Pre-Decisional For Internal ACRS Use Only).

2. Staff Requirements Memorandum dated March 31, 2003, from Annette L. Vietti-Cook, SECY, NRC, to William D. T ravers, EDO, NRC, Subject: SECY-02-0057 - Update to SECY-01-0133, "Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)."

3. Staff Requirements Memorandum dated July 1, 2004, from Annette L. Vietti-Cook, SECY, NRC, to Luis A. Reyes, EDO, NRC, Subject: Staff Requirements - SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant-Accident (LOCA) Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power".

4. U.S. Nuclear Regulatory Commission, Reactor Safety Study, an Assessment of Accident Risks in U.S. Nuclear Power Plants, Report NUREG-75/014 (WASH-1400), 1975.

5. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Report NUREG-1150, 1990.

6. Electric Power Research Institute, *Seismic Hazard Methodology for the Central and Eastern United States*, EPRI Report NP-4726, 1988.

7. R.J. Budnitz, G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris, *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance and Use of Experts*, Report NUREG/CR-6372, 1997

Attachment 6: ACRS Letter of April 27, 2004

ACRSR-2076

April 27, 2004

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SECY-04-0037, "ISSUES RELATED TO PROPOSED RULEMAKING TO RISK INFORM REQUIREMENTS RELATED TO LARGE BREAK LOSS-OF-COOLANT ACCIDENT (LOCA) BREAK SIZE AND PLANS FOR RULEMAKING ON LOCA WITH COINCIDENT LOSS-OF-OFFSITE POWER"

Dear Chairman Diaz:

During the 511th meeting of the Advisory Committee on Reactor Safeguards on April 15-17, 2004, we reviewed SECY-04-0037, "Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to Large Break Loss-of-Coolant Accident (LOCA) Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power." Our Subcommittee on Regulatory Policies and Practices reviewed this matter during a meeting on April 1, 2004. During these reviews, we had the benefit of discussions with the NRC staff and the Nuclear Energy Institute. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The risk-informed revision to 10 CFR 50.46 should permit a wide range of applications of the new break size as long as it can be demonstrated that the resulting changes in risk are small and adequate defense-in-depth is maintained.

2. The process and criteria in Regulatory Guide (RG) 1.174 are appropriate for evaluating the acceptability of changes proposed under a revised rule. However, explicit criteria to ensure mitigative capability for breaks beyond the new maximum break size and to limit the risk associated with late containment failure should be developed as part of the revised rule to ensure that sufficient defense-in-depth is maintained as plant changes are made.

3. We concur with the recommendation of the staff that the appropriate metric for the design basis maximum break size is the direct LOCA initiating event frequency.

4. Additional criteria and guidance are not needed for tracking cumulative risk due to the changes resulting from a risk informed 10 CFR 50.46.

5. The results of the expert elicitation for the frequency of LOCA events are yet to be finalized and peer-reviewed, but the process employed and the qualifications of the

panel members appear to be well suited to the problem. The results should help provide a technical basis for the selection of the new maximum break size.

DISCUSSION

The double-ended guillotine break (DEGB) of the largest pipe in the system has always been recognized as an unlikely event. It was intended to be a surrogate accident that bounded the consequences of a wide spectrum of reactor accidents. Probabilistic risk assessments (PRAs) for existing plants show that the defense-in-depth provided by the emergency core cooling system (ECCS) capability and robust containment designs developed to deal with this accident have resulted in plants with low core damage frequency (CDF) and low risk to the public.

However, experience and PRA have also shown that the focus on such large, highly unlikely breaks can have detrimental effects on safety. The demands on equipment resulting from the need to demonstrate the equipment's capability to deal with the DEGB can reduce the equipment's reliability and capability to function during the much more likely small and medium LOCAs. Improved understanding of the likelihood of various initiating events and the responses of reactor systems to those events suggests that a risk-informed approach to dealing with large break LOCAs could result in greater operational flexibility with little increase or even decreases in risk.

In a Staff Requirements Memorandum (SRM), dated March 31, 2003 (Reference 2), the Commission directed the staff to complete the technical basis supporting the redefinition of the maximum design-basis break size and to provide proposed rule changes to the Commission. In its evaluation of the SRM, the staff identified a number of policy and technical issues that it felt needed to be resolved to ensure that the new rulemaking for maximum break size redefinition does not result in any unintended consequences. The staff discusses these issues in SECY-04-0037.

Because the consequences of 10 CFR 50.46 in the regulatory system are pervasive, the staff believes the Commission needs to provide additional guidance on the scope of changes to be permitted under a new rule. The staff distinguishes between a "narrow" scope and a "broad" scope rule change. In a "narrow" scope rule change, specific areas of application would be identified, similar to the current use of leak-before-break to restrict the sizes of breaks considered in determining dynamic effects. An example would be to permit the use of the redefined maximum break size in determining the start times for emergency diesels. A "broad" scope rule would permit a wide range of applications of the new break size as long as it could be demonstrated that the resulting changes in risk are small and adequate defense-in-depth is maintained. We believe that the revised rule should support a broad scope of applications.

It may be possible to deal with some applications generically in the revised rule, but in most cases applications of the new rule will be developed by licensees and will require plant-specific demonstrations that the resulting changes in risk are acceptable. RG 1.174 provides a process for determining the acceptability of changes in risk associated with changes in the licensing basis. In SECY-04-0037, the staff's preliminary conclusion is that the numerical criteria listed in RG 1.174 for defining acceptable changes to a plant's licensing basis are not stringent enough to use for modifying the fundamental building blocks and protections provided in the current regulations. We disagree. The uncertainties may be different in different situations, but if a certain change in risk is acceptable in terms of a change to a licensing basis, we see no reason why there should be a different level of acceptable risk for a modification of a rule, even one as fundamental as 10 CFR 50.46.

The number and kind of changes that will be possible for a licensee to make under the new rule will depend strongly on the scope and technical detail of the licensee's PRA. What is important is a convincing demonstration that the resulting changes are indeed small enough to meet the RG 1.174 criteria. If a limited scope PRA is used, contributions to the CDF and the total CDF and corresponding large, early release frequency (LERF) quantities from the omitted portions of the PRA and the associated uncertainties must still be conservatively estimated and demonstrated to be consistent with the RG 1.174 criteria.

The expert elicitation and other evaluations of the likelihood of large pipe breaks demonstrate that the frequency of such failures due to normal loads and conventional modes of degradation is quite low. It is much more difficult to quantify the potential for such failures due to abnormal loads, security issues, and human errors. Thus a capability to mitigate breaks beyond the new maximum break up to the DEGB of the largest pipe needs to be maintained.

In the March 31, 2003 SRM, the Commission directed the staff not to permit changes in ECCS coolant flow rates or reduce containment capabilities. However, the degree of defense-in-depth provided by these systems may change as plants make changes in response to the new rule. We believe that the staff should be directed to develop criteria and guidance to quantify the capability to mitigate DEGB beyond the new maximum break size and thus ensure that sufficient defense-in-depth is maintained. One possibility is a criterion for the conditional probability of core damage given a DEGB beyond the new maximum break size, but other approaches are possible.

Calculations of the conditional probability could be performed using the realistic approaches taken in PRAs to assess core damage rather than the conservative approach taken in Appendix K to assess core damage. Some degree of core damage could be permitted to occur, but coolability would be maintained and rapid failure of the vessel precluded. It may also be necessary to develop guidance to ensure the functionality of equipment that may no longer be required under design basis conditions, but would be needed to mitigate a beyond design basis break.

RG 1.174 includes consideration of the risks associated with late containment failures, but it does not provide any explicit criteria for evaluating such risks. Such a criterion was developed in the Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50 document (Reference 3) where it was proposed that the conditional probability of a large late release (i.e., one that does not contribute to LERF, but occurs within approximately 24 hours of the onset of core damage) be limited to 10^{-1} or less. This criterion or a suitable alternative should also be considered when considering changes associated with a revised rule.

One of the important technical issues raised by the staff in SECY-04-0037 is the choice of the appropriate metric to determine the design basis LOCA maximum break size. The staff argues that a metric based on the expected frequency of pipe breaks is more direct than one based on the impact of LOCAs on CDF and LERF. Also, the staff argues that most licensees will be following a phased approach in upgrading their PRAs and any definitions based on CDF and LERF could result in maximum break sizes that vary simply because of changes in PRA methods. We concur with the staff's conclusion that break frequency is the best metric. The rationalist approach to defense-in-depth considers the frequency of an initiating event as a basic criterion in assessing the confidence that must be provided for the response to the initiating event.

The staff proposes to identify large break LOCA sizes applicable to various categories of plants if possible. If not, the staff would specify a plant-specific implementation process necessary to determine the appropriate plant-specific break size. We believe that it is possible and desirable to make generic definitions applicable to categories of plants.

As a consequence of the redefinition of the maximum break size, licensees may propose plant changes that will result in increases in risk. The RG 1.174 process will ensure that the change in risk associated with any specific change in the licensing basis will be small, but there is certainly a possibility that a significant number of changes will be proposed because of the change to 10 CFR 50.46. The staff proposes to determine the information that needs to be tracked for individual changes authorized by the rule over the life of the plant and to develop analysis guidelines for cumulative risk estimates that can be compared to applicable risk criteria. We believe that the limitations implied by the RG 1.174 criteria, the inclusion of specific defense-in-depth criteria for mitigation of beyond design basis breaks, and an explicit criterion for late containment failure will limit changes in cumulative risk to acceptable levels. RG 1.174 provides sufficient guidance and criteria to track and control cumulative risk, and additional requirements are not necessary.

The elicitation process to determine degradation related LOCA frequencies was well structured and the expert panel has an appropriate range of expertise (Reference 4). Although the results are still under final review, we expect that they will be confirmed by the planned peer review and will provide a technical basis for the selection of a maximum break size in terms of the frequency of the initiating event.

There are important policy and technical issues to be considered in the development of a risk informed 10 CFR 50.46. We look forward to interacting with the staff as it pursues this effort after receiving further guidance from the Commission.

Sincerely,

/RA/
Mario V. Bonaca
Chairman

References:

1. Memorandum from William D. Travers, EDO, to the Commissioners, SECY-04-0037, Subject: Issues Related to Proposed Rulemaking to Risk-Inform Requirements Related to LBLOCA Break Size and Plans for Rulemaking on LOCA With Coincident Loss-of-Offsite Power, March 3, 2004.

2. Staff Requirements Memorandum (SRM) from Annette L. Vietti-Cook, Secretary, to William D. Travers, EDO, Subject: SECY-02-057 - Update to SECY-01-0133, "Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)," dated March 31, 2003.

3. Draft, Revision 2, “Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50,” August 2000.

4. Memorandum from Michael E. Mayfield, RES, to John T. Larkins, Executive Director, ACRS, Subject: Forwarding of Commission Paper on “Loss-of-Coolant Accident (LOCA) Break Frequencies for the Option III Risk-Informed Reevaluation of 10 CFR 50.46, Appendix K to 10 CFR Part 50, and General Design Criteria (GDC) 35," and its Corresponding Attachment (Pre-Decisional For Internal ACRS Use Only), March 29, 2004.

Attachment 7: ACRS Letter of July 25, 2001

July 25, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: FEASIBILITY STUDY ON RISK-INFORMING THE TECHNICAL REQUIREMENTS OF 10 CFR 50.46 FOR EMERGENCY CORE COOLING SYSTEMS

During the 484th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 2001, we met with representatives of the NRC staff and the industry to discuss the status of staff and industry initiatives to risk inform the technical requirements of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." Our Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment discussed this matter with representatives of the NRC staff, the Nuclear Energy Institute, the Westinghouse Owners Group, and the Boiling Water Reactor Owners Group on July 9, 2001. We also had the benefit of the documents referenced.

Recommendations

1. We recommend that the Commission approve the staff's request to proceed with rulemaking to modify the existing 10 CFR 50.46 to replace the prescriptive emergency core cooling system (ECCS) acceptance criteria with a performance-based requirement and to modify the 10 CFR Part 50, Appendix K evaluation model.

2. We recommend that the Commission approve the staff's request to proceed with the development of a voluntary risk-informed alternative to 10 CFR 50.46, Appendix K, and General Design Criterion (GDC) 35 of 10 CFR Part 50, Appendix A.

3. The staff should continue to develop the technical bases and requirements for redefining the large-break loss-of-coolant accident (LBLOCA).

Discussion

The ECCS requirements codified in 10 CFR 50.46, Appendix K, and GDC 35 are intended to ensure that plants can safely cope with a LBLOCA. The ECCS has been designed to accommodate pipe breaks up to and including a double-ended guillotine break of the largest pipe in the reactor coolant system. GDC 35 requires that the ECCS be capable of providing sufficient core cooling for a full spectrum of postulated LOCAs using either offsite power or onsite power. To comply with this requirement, ECCS evaluations generally assume that pipe breaks are coincident with a loss of offsite power (LOOP). In addition, the system must have sufficient diversity and redundancy to accomplish its safety function assuming a single failure.

Because LBLOCAs are rare, the current requirements for ECCS performance may have a detrimental effect on safety. These requirements focus attention and resources on events that are extremely unlikely to happen rather than on events which can have a larger contribution to risk. For example, the postulated occurrence of a LOOP coincident with a LBLOCA leads to requirements for rapid emergency diesel generator (EDG) start times and load sequencing. Such requirements could reduce the reliability of the EDGs and diminish the capability of the system to deal with the more likely small and medium break LOCAs.

The industry has proposed a revision of 10 CFR 50.46 that is based on a redefinition of the LBLOCA. Instead of dealing with a full spectrum of break sizes up to and including the double-ended guillotine break of the largest pipe in the reactor coolant system, the industry proposes to define a new maximum LBLOCA size based on leak-before-break (LBB) methodology and probabilistic assessments of the frequency and consequences of the new LBLOCA size.

The staff has accepted LBB methodology for the analysis of dynamic effects of pipe failure for pipe sizes down to 8-inches in some cases. The NRC pioneered the application of probabilistic fracture mechanics to piping through the development of the PRAISE code. The staff argues, however, that the prediction of leak rates for all sizes of cracks in all locations in piping systems is technically much more demanding than predicting whether a detectable leak will occur before failure. The staff also argues that a more rigorous assessment of uncertainties is needed to justify the redefinition of the LBLOCA for ECCS requirements. Thus, the staff believes this is a longer-term activity that will require a substantial technical effort.

We agree that the effort to define a new LBLOCA size requires an extension of current LBB and probabilistic fracture mechanics methodology. We believe that it is technically feasible, but the justification of the new LBLOCA size will become increasingly difficult as the proposed maximum break size is decreased. The industry has stated that it is willing to invest substantial resources to accomplish this objective. The staff should continue to develop the technical basis and requirements for the redefinition of LBLOCA.

In its Feasibility Study, the staff has investigated a number of options for revising 10 CFR 50.46 that it believes can be implemented on a shorter time scale and will provide safety benefits and some reduction in unnecessary conservatism and associated regulatory burden.

One of these options would make changes in the Appendix K evaluation model and would replace the current prescriptive ECCS acceptance criteria with a performance-based requirement. This would permit licensees to use cladding materials other than zircaloy or ZIRLO without having to seek an exemption. The current criteria, such as the 2200F peak clad temperature and 17% oxidation limit, would be relegated to a regulatory guide as acceptance criteria for zircaloy and ZIRLO. We support the proposed development of the new performance-based acceptance requirement.

Possible changes in the evaluation models suggested in the staff Feasibility Study include replacing the current 1971 American Nuclear Society (ANS) decay heat curve with the 1994 ANS standard, replacing the current decay heat multiplier of 1.2 with an uncertainty estimate, and replacing the Baker-Just oxidation model with the Cathcart-Pawel oxidation model for heat generation. The intent of these changes is to use improved technical understanding to remove excessive conservatism from Appendix K models.

We are generally supportive of this effort, but note that in dealing with a mix of models in which some elements are conservative and some elements are nonconservative, removing "excessive" conservatism without a real understanding of the uncertainties in the overall model can lead to unsatisfactory results. For example, although the Cathcart-Pawel model gives a more accurate description of the oxidation behavior of unirradiated zircaloy tubing in laboratory studies, the more conservative Baker-Just model was deliberately chosen in an attempt to ensure that the effects of variables such as irradiation and behavior such as spalling of the oxide film that were not explicitly included in the models would not lead to nonconservative results. In addition, although the staff is developing performance-based acceptance criteria to permit use of other cladding materials, both the Baker-Just and Cathcart-Pawel models build "zircaloy behavior" into the evaluation model. The staff should consider a performance-based requirement for a heat generation model that includes the effects of cladding oxidation, irradiation, and the potential for cladding spallation rather than a prescriptive requirement.

Acceptable heat generation models for different cladding materials could then be discussed in a regulatory guide. If implementation of the Appendix K option proves to be more challenging than anticipated, then the staff should proceed with a rulemaking that includes only the update of the decay heat curve to the 1994 ANS standard.

The second shorter-term option recommended by the staff is a voluntary risk-informed alternative to 10 CFR 50.46 that would replace the current requirements intended to ensure ECCS reliability (i.e., the coincident LOOP and the single-failure criterion) with more risk-informed approaches that reflect the lower frequencies of LBLOCAs. Licensees could choose either generic deterministic reliability requirements developed by the NRC (e.g, a requirement that a coincident LOOP be postulated only for smaller, more frequent LOCAs) or show that they can meet an acceptable threshold value for the core damage frequency (CDF) and large, early release frequency (LERF) associated with the LOCA initiators with appropriate consideration of uncertainties. ECCS reliability evaluations could reflect plant-specific features and operational data. Frequencies of LOCAs with different break sizes could be determined using the analysis provided in NUREG/CR-5750, updated to reflect more recent operating experience.

Alternatively, probabilistic fracture mechanics together with a review of service history data could be used, but the technical work to support this would be similar in magnitude to that required to define the new LBLOCA size. We believe the approach outlined by the staff in this option would provide a much more realistic and risk-informed approach for ECCS requirements. The staff should proceed with the technical work and the rulemaking for this option.

We look forward to reviewing the technical work and regulatory guidance needed to support these rulemaking efforts as they evolve.
Sincerely,

/RA/
George E. Apostolakis
Chairman

References:

1. Draft memorandum received June 3, 2001, from William D. Travers, Executive Director for Operations, to The Commissioners, Subject: Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR part 50 (Option 3) and

Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria), and attached Feasibility Study report.

2. Memorandum dated January 19, 2001, from Annette L. Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-00-0198 - Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes 10 CFR 50.44 (Combustible Gas Control).

3. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.

4. Letter dated February 8, 2001, from Anthony R. Pietrangelo, Nuclear Energy Institute, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Preliminary Industry Response to NRC Questions on Redefinition of Large-Break Loss-of-Coolant Accident.

5. Letter dated October 17, 2000, from Robert H. Bryan, Westinghouse Owners Group to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: "WOG Large Break Loss of Coolant Accident (LBLOCA) Redefinition Discussion of Benefits."

6. Letter dated January 8, 2001, from Adrian Heymer, Nuclear Energy Institute, to Mary T. Drouin, Office of Nuclear Regulatory Research, Subject: "Draft Large Break LOCA Redefinition Program, Project Summary."

7. Letter dated January 19, 2000, from Joe F. Colvin, Nuclear Energy Institute, to Richard A. Meserve, Chairman, NRC, Subject: SECY-99-264, Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.

8. American Nuclear Society, ANSI/ANS-5.1-1994, American National Standard for Removing Decay Heat Power in Light Water Reactors, dated August 23, 1994.

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Attachment 8: ACRS Letter of November 20, 2000

November 20, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Travers:

SUBJECT:, PROPOSED FRAMEWORK FOR RISK-INFORMED CHANGES TO THE TECHNICAL REQUIREMENTS OF 10 CFR PART 50

During the 477th meeting of the Advisory Committee on Reactor Safeguards, November 2-4, 2000, we met with representatives of the NRC staff to discuss Attachment 1 to SECY-00-0198 entitled, "Framework for Risk-Informed Changes to the Technical Requirements of 10 CFR 50." Our Subcommittee on Reliability and Probabilistic Risk Assessment met on July 11, 2000, to discuss an earlier version of the proposed framework. We also had the benefit of the documents referenced.

The purpose of the framework is to provide guidance to the staff for the identification and development of risk-informed changes to the technical requirements of 10 CFR Part 50 (Option 3). The proposed framework is a work in progress. The staff has identified the elements that are important to the prioritization of candidate regulations to be risk informed. We agree with the staff that improvements will be made to the framework as experience is gained from evaluating its application to risk-informing candidate regulations such as 10 CFR 50.44 related to combustible gas control systems and 10 CFR 50.46 concerning emergency core cooling systems. We offer the following comments for consideration as the work progresses.

The structuralist approach to defense in depth has been applied to the top tiers of the framework by adopting the cornerstones of the revised reactor oversight process, i.e., limiting the frequency of accident initiating events and the conditional core damage probability (CCDP) given an initiating event, and limiting radionuclide releases and public health effects given core damage. The "tactics" for achieving these goals include safety margins and redundancy, diversity, and independence. We recommend that the tactics for implementing defense in depth be clarified. Will defense in depth be applied at all levels of the framework? Will it be invoked at lower tiers when it has already been applied to the top tiers?

In our May 19, 1999 report and the associated attachment, we offered a "preliminary proposal" to apply the structuralist approach at lower tiers only when there are significant uncertainties that have not been included in the probabilistic risk assessment (PRA) and could reduce confidence that the higher-level goals are met. For uncertainties that are included in the PRA, we recommended that the rationalist approach be followed, i.e., appropriate safety margins and redundancy, and diversity would be developed by quantitative analyses. Even though the framework is consistent with this approach, an expanded discussion of these issues would be beneficial.

We are pleased that the proposed framework recommends the quantification of safety margins in terms of probabilities. While present PRA methods can provide estimates of the contribution of multiple barriers (defense-in-depth measures) to the risk metrics, the contribution from safety margins is not normally quantified. We believe that the quantification of safety margins would be an important step toward the wider use of the rationalist approach. It would also make the integrated decision-making process of Regulatory Guide 1.174 easier to implement.

The framework proposes goals for the frequency of three groups of initiating events: anticipated, infrequent, and rare initiators. Even though this is reasonable for the standard initiating events for light water reactor PRAs, there is a potential pitfall. The concept of an initiating event is not defined rigorously. For an infrequent initiating event, the framework requires that the CCDP be less than or equal to 10^{-2} per reactor-year. One could envision partitioning this initiating event into a number of more specific initiating events, each with a frequency less than or equal to 10^{-5} per reactor-year. These new initiating events would then belong to the group of rare initiators, and there would be no constraints imposed on the CCDP. Thus, creative definitions of initiating events could be used to inappropriately relax the CCDP goal.

The external events in a PRA, such as earthquakes and fires, affect all of the cornerstones. The treatment of events that affect more than one cornerstone extensively should be discussed.

We look forward to reviewing additional refinements to the framework as progress is made in its application to developing risk-informed alternative regulations.

Sincerely,

/RA/
Dana A. Powers
Chairman

References:

1. Memorandum dated September 14, 2000, from William D. Travers, Executive Director for Operations, for the Commissioners, Subject: SECY-00-0198, Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).

2. Report dated September 13, 2000, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Richard A. Meserve, Chairman, NRC, Subject: Proposed Risk-Informed Revisions to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

3. Report dated May 19, 1999, from Dana A. Powers, Chairman, Advisory Committee on Reactor Safeguards, to Shirley Ann Jackson, Chairman, NRC, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System.

4. Paper by J. N. Sorensen, G. E. Apostolakis, T. S. Kress, D. A. Powers, "On the Role of Defense in Depth in Risk-Informed Regulation," presented at the American Nuclear Society, International Topical Meeting on Probabilistic Safety Assessment, PSA '99, Washington, DC, August 22-26, 1999.

5. Memorandum dated February 3, 2000, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-264 - Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50.

6. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

Attachment 9: ACRS Letter of May 19, 1999

May 19, 1999

The Honorable Shirley Ann Jackson
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Dear Chairman Jackson:

SUBJECT: THE ROLE OF DEFENSE IN DEPTH IN A RISK-INFORMED REGULATORY SYSTEM

During the 462nd and 461st meetings of the Advisory Committee on Reactor Safeguards, May 5-8 and April 7-10 1999, we discussed issues identified in the Staff Requirements Memorandum dated March 5, 1999, concerning the appropriate relationship and balance between probabilistic risk assessment (PRA) and defense in depth in the context of risk-informed regulation. We previously discussed this matter with the Commission during our meeting on February 3, 1999.

We are attempting to identify pitfalls that may exist along the path the Commission is taking toward risk-informed regulation so they may be addressed in a timely manner. We have communicated previously on the need for plant-specific safety goals that are practical for licensees to evaluate, the need for risk assessments for all modes of plant operation, and the need for research to support further use of risk information in regulatory activities. Several ACRS members, working with an ACRS Senior Fellow, have produced the attached paper in which two views of defense in depth are discussed along with a preliminary proposal regarding its role. Here, we further discuss the role that defense in depth should have in a risk-informed regulatory scheme.

Our motivation for this report has arisen because of instances in which seemingly arbitrary appeals to defense in depth have been used to avoid making changes in regulations or regulatory practices that seemed appropriate in the light of results of quantitative risk analyses. Certainly, we have seen defense in depth used as a basis for delaying changes in the existing regulatory practices:

- there has been reluctance to develop new, risk-informed limits on leakage from steam generator tubes because these are part of the defense-in-depth barriers,

- the development of extensions of the Regulatory Guide 1.174 process to define criteria for risk-informed revisions to 10 CFR 50.59 has been delayed because of defense in depth issues,

- the development of graded quality assurance measures has been overly conservative because of concerns about the imputed importance of quality assurance to defense in depth, and
- the development of regulatory requirements on software-based digital instrumentation and control systems was delayed because of concerns related to defense in depth.

We are concerned that arbitrary appeals to defense in depth could inhibit the effective use of risk information in the regulatory process. At the same time, we are mindful that risk analyses are not perfect. Defense in depth can be an effective means for compensating for any weaknesses in our ability to understand the risks posed by nuclear power plants.

As discussed in the attached paper, the defense-in-depth approach to safety arose in an earlier time when there was less capability to analyze a nuclear power plant as an integrated system. Subsystems were designed such that the necessity and sufficiency of defense in depth could be determined from experience and through exercising engineering judgment. Defense in depth was a design and operational philosophy that called for multiple layers of protection to prevent and mitigate accidents. Its practical implementation was most often associated with control of initiating event frequencies, redundancy and diversity in key safety functions, multiple physical barriers to fission-product release, and emergency response measures. This philosophy has been invoked primarily to compensate for uncertainty in our knowledge of the progression of accidents at nuclear power plants.

Improved capability to analyze nuclear power plants as integrated systems is leading us to reconsider the role of defense in depth. Defense in depth can still provide needed safety assurance in areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain. To avoid conflict between the useful elements of defense in depth and the benefits that can be derived from quantitative risk assessment methods, constraints of necessity and sufficiency must be imposed on the application of defense in depth and these must somehow be related to the uncertainties associated with our ability to assess the risk.

We believe that two different perceptions of defense in depth are prominent. In one view (the "structuralist" view as described in the attached paper), defense in depth is considered to be the application of multiple and redundant measures to identify, prevent, or mitigate accidents to such a degree that the design meets the safety objectives. This is the general view taken by the plant designers. The other view (the "rationalist"), sees the proper role of defense in depth in a risk- informed regulatory scheme as compensation for inadequacies, incompleteness, and omissions of risk analyses. We choose here to refer to the inadequacies, incompleteness, and omissions collectively as uncertainties. Defense-in-depth measures are those that are applied to the design or operation of a plant in order to reduce the uncertainties in the determination of the overall regulatory objectives to acceptable levels. Ideally then, there would be an inverse correlation between the uncertainty in the results of risk assessments and the extent to which defense in depth is applied. For those uncertainties that can be directly evaluated, this inverse correlation between defense in depth and the uncertainty should be manifest in a sophisticated PRA uncertainty analysis.

When defense in depth is applied, a justification is needed that is as quantitative as possible of both the necessity and sufficiency of the defense-in-depth measures. Unless defense-in-depth measures are justified in terms of necessity and sufficiency, the full benefits of risk-informed regulation cannot be realized.

The use of quantitative risk-assessment methods and the proper imposition of defense-in-depth measures would be facilitated considerably by the availability of risk-acceptance criteria applicable at a greater level of detail than those we now have. Development of the additional risk-acceptance criteria would have to take into consideration safety objectives embodied in the existing regulations. For example, risk-acceptance criteria are needed to meet the Commission's safety objectives with respect to worker health and environmental contamination and to meet additional public health and safety objectives [e.g., total fatalities, land interdiction]. All of these may not be currently reflected in conventional risk assessments.

We believe that a key missing ingredient needed to place quantitative limits on defense-in-depth measures is acceptance values on the level of uncertainty for each safety objective. Setting such acceptance values is a policy role, very much like setting safety goal values. The uncertainties that are intended to be compensated for by defense in depth include all uncertainties (epistemic and aleatory). Not all of these are directly assessed in a normal PRA uncertainty analysis. Therefore, when acceptance values are placed on uncertainty, these would have to appropriately incorporate consideration of the additional uncertainties not subject to direct quantification by the PRA. These considerations would have to be determined by judgment and expert opinion. As a practical matter, we suggest that the acceptance values be placed on only those epistemic uncertainties quantifiable by the PRA but that these be set sufficiently low to accommodate the unquantified aleatory uncertainties.

When acceptance values have been chosen as policy for the regulatory objectives and their associated uncertainties, it would be possible to develop objective limits on the amount of defense in depth required for those design and operational elements that are subject to evaluation by PRA. To do this, it is necessary to incorporate the effects of the defense-in-depth measures into the PRA uncertainty analysis and the designer or regulator must be able to adjust the defense in depth until the acceptance levels for the regulatory objectives and the acceptance values for the associated uncertainties have both been achieved.

The balance between core damage frequency (CDF) and conditional containment failure probability (CCFP) can serve as an example of this defense-in-depth concept. We have previously recommended that CDF be elevated to a fundamental safety goal. Let us suppose, for example sake, that our acceptance value on this is 10-4 per reactor year. If that is the value actually achieved by the design, then a CCFP of about 0.5 has been shown (NUREG-1150) to be generally sufficient to meet the safety goal regulatory objective of individual risk of prompt fatality [which can be adequately represented by an acceptance value of 10-5 per reactor year on large, early release frequency (LERF) as noted in Regulatory Guide 1.174]. Does this CCFP provide sufficient defense in depth?

In our view, three acceptance criteria must be satisfied -- one each on CDF, LERF, and the epistemic uncertainty associated with LERF. The Safety Goal Policy Statement suggests candidate acceptance values on CDF and LERF. In addition to these, we must establish the acceptance value on the uncertainty associated with LERF. For the particular value of LERF achieved, let's say that the acceptance value has been set by policy to be on the epistemic uncertainty that can be directly developed from the PRA [but which properly reflects the unquantified aleatory uncertainties]. Now suppose our PRA uncertainty analysis tells us that the quantified uncertainty for this design is greater than the acceptance value. Employing our concept, the design with the 0.5 CCFP does not have sufficient defense in depth. The design must, then, include provisions for more defense in depth [e.g., a better containment perhaps] or reduction of the LERF to values for which the achieved uncertainty is acceptable. The acceptance value on uncertainty for any given regulatory objective could be a function of the absolute value achieved for the regulatory objective. That is, as the achieved mean value for LERF gets further below the acceptance value, the acceptable level of uncertainty on its determination can be greater.

We believe this concept of defense in depth can provide a rational way to develop sufficiency limits wherever the defense-in-depth measures can be directly evaluated by PRA. We acknowledge however, that considerable judgment will have to be exercised to set limits on uncertainty, especially uncertainties not quantified by the PRA. Our preceding example suggests one approach to managing these uncertainties.

For those regulatory functions that are not well suited for PRA or where the current capabilities of PRAs are not sufficient, we suggest that the limits on application of defense in depth be placed at levels lower than the top-level safety objectives (see Figure 1 of attached paper). We emphasize that, even under these circumstances, the PRA can still dictate when defense in depth is needed. Let us illustrate how we envision defense in depth to be applied under these circumstances with an example. Fire is one of the initiating events of interest. PRAs quantify the occurrence of fires in nuclear power plants and, among other things, their impact on control and power cables. The plant response to the loss of the relevant systems (due to the loss of these cables) is also analyzed.

The frequency of fires in specific critical locations, that is, locations in which cables of redundant systems may be damaged, is estimated in the PRA using experience-based rates of occurrence of fires, multiplied by subjective estimates of the fraction of fires that are large enough to have the potential to cause damage and the fraction of those fires that occur in the specified critical locations. This is a highly subjective part of the risk assessment (therefore, highly uncertain). It is, therefore, a suitable area to invoke defense in depth and to impose prescriptive requirements regarding the prevention of fires in those critical locations [e.g., strict administrative controls and periodic inspections]. Thus, the relative inadequacy of the PRA model suggests how defense in depth should be applied at levels lower than the top-level safety objectives.

We further realize that the fire risk assessment does not include the damaging effects of the smoke generated by a fire. This is a case of omission of a potentially significant effect. Therefore, we would, again, resort to defense in depth and may demand barriers to limit the spread of smoke and to protect sensitive equipment.

Since the impact on the risk metrics of these lower-level defense-in-depth measures cannot be quantified, nor can the uncertainties, the necessity and sufficiency of the defense-in-depth measures will have to be simply prescribed and that prescription would constitute the acceptance criteria.

We note that our first example dealing with CDF and CCFP addresses the top level of Figure 1 of the attached paper. If one adopts the structuralist viewpoint at that level, as the paper's preliminary proposal suggests, then the tradeoffs of our example between CDF and CCFP will have to be performed under the assumption that at least some level of defense in depth will be required. If, on the other hand, one adopts the rationalist view even at that level, it is conceivable that the LERF objectives could be satisfied without a containment. Our second example dealing with fires exemplified the rationalist view at lower levels, as the preliminary proposal recommends.

We acknowledge that these preliminary thoughts on the role of defense in depth in a risk-informed regulatory system identify a direction but fall short of closing the issue. We recommend that the Commission give further consideration to this matter.

Sincerely,

/s/
Dana A. Powers
Chairman

References:

1. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

2. U. S. Nuclear Regulatory Commission, NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," December 1990.

3. Report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters.

4. Memorandum dated March 5, 1999, from Annette Vietti-Cook, Secretary of the NRC, to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting with the Advisory Committee on Reactor Safeguards, February 3, 1999.

Attachment:

U. S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, J. N. Sorensen, G. E. Apostolakis, T. S. Kress, D. A. Powers, "On the Role of Defense in Depth in Risk- Informed Regulation," to be presented at PSA 1999, August 22-25, 1999.
- -

ON THE ROLE OF DEFENSE IN DEPTH IN RISK-INFORMED REGULATION

To be presented at PSA '99
Washington, D.C.
August 22-25, 1999

J. N. Sorensen, Senior Fellow
G. E. Apostolakis, Member
T. S. Kress, Member
D. A. Powers, Member
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ABSTRACT

The nascent implementation of risk informed regulation in the United States suggests a need for reexamination of the Nuclear Regulatory Commission's (NRC) defense in depth philosophy and its impact on the design, operation, and regulation of nuclear power plants. This reexamination is motivated by two opposing concerns: (1) that the benefits of risk informed regulation might be diminished by arbitrary appeals to defense in depth, and (2) that the implementation of risk informed regulation could undermine the defense in depth philosophy. From either perspective, two questions are suggested: (1) How is defense in depth defined? (2) How should the implementation of risk informed regulation alter our view of defense in depth? A preliminary proposal for the role of defense in depth in a risk- informed regulatory system is presented.

HISTORICAL DEVELOPMENT

Defense in depth is a nuclear industry safety strategy that began to develop in the 1950s. A review of the history of the term indicates that there is no official or preferred definition. Where the term is used, if a definition is needed, one is created consistent with the intended use of the term. Such definitions are often made by example.

In a 1967 statement submitted to the Joint Committee on Atomic Energy by Clifford Beck, then Deputy Director of Regulation for the Atomic Energy Commission, three basic lines of defense for nuclear power reactor facilities were described. The first line was the prevention of accident initiators through superior quality of design, construction and operation. The second line was engineered safety systems designed to prevent mishaps from escalating into major accidents. The third line was consequence-limiting safety systems designed to confine or minimize the escape of fission products to the environment.

A 1969 paper by an internal study group of the Atomic Energy Commission identified the issue of balance among accident prevention, protection, and mitigation, with the conclusion that the greatest emphasis should be put on prevention, the first line of defense.
A 1994 NRC document identifies the elements of the defense in depth safety strategy as accident prevention, safety systems, containment, accident management, and siting and emergency plans. Other interpretations of defense in depth can be found in INSAG-3 and INSAG-10.

The historical record indicates an evolution of the term from a narrow application to the multiple barrier concept to an expansive application as an overall safety strategy. The term has increased in scope and gained stature over time. The history also indicates that defense in depth is considered to be a concept, an approach, a principle or a philosophy, as opposed to being a regulatory requirement per se.
Currently the term is commonly used in two different senses. The first is to denote the philosophy of high level lines of defense, such as prevent accident initiators from occurring, terminate accident sequences quickly, and mitigate accidents that are not successfully terminated. The second is to denote the multiple physical barrier approach, most often exemplified by the fuel cladding, primary system, and containment.

One of the essential properties of defense in depth is the concept of successive barriers or levels. This concept applies equally well to multiple physical barriers and to high level lines of defense. A closely related attribute would be requiring a reasonable balance among prevention, protection and mitigation.

EMERGING REGULATORY PRACTICE

The most recent NRC policy statement that deals with defense in depth is the Probabilistic Risk Assessment (PRA) Policy statement published in 1995, which states, in part:

"The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

The policy statement, thus, places PRA in a subsidiary role to defense in depth.

In 1998, the NRC published Regulatory Guide 1.174. This guide establishes an approach to risk-informed decision making, acceptable to the NRC staff, which includes the provision that proposed changes to the current licensing basis must be consistent with the defense in depth philosophy. The RG 1.174 discussion states that, "The defense in depth philosophy . . . has been and continues to be an effective way to account for uncertainties in equipment and human performance." The discussion goes on to say that PRA can be used to help determine the appropriate extent of defense in depth, which, by example, is equated to balance among core damage prevention, containment failure prevention and consequence mitigation. The regulatory guide thus addresses the concern of preventing risk-informed regulation from undermining defense in depth. Defense in depth is primary, with PRA available to measure how well it has been achieved.

STRUCTURALIST MODEL

We have identified two different schools of thought (models) on the scope and nature of defense in depth. These models came to be labeled "structuralist" and "rationalist."

The structuralist model asserts that defense in depth is embodied in the structure of the regulations and in the design of the facilities built to comply with those regulations. The requirements for defense in depth are derived by repeated application of the question, "What if this barrier or safety feature fails?" The results of that process are documented in the regulations themselves, specifically in Title 10, Code of Federal Regulations. In this model, the necessary and sufficient conditions are those that can be derived from Title 10. It is also a characteristic of this model that balance must be preserved among the high-level lines of defense, e.g., preventing accident initiators, terminating accident sequences quickly, and mitigating accidents that are not successfully terminated. One result is that certain provisions for safety, for example reactor containment and emergency planning, must be made regardless of our assessment of the probability that they may be required. Accident prevention alone is not relied upon to achieve an adequate level of protection.

There does not appear to be any question that the implementation of defense in depth up to the present time reflects the structuralist model. While this philosophy has served the industry well from the safety perspective, it is now realized that, in some instances, it has led to excessive regulatory burden. Furthermore, the lack of an integrated view of the reactor systems has resulted in some significant accident sequences not being identified until PRA was developed, e.g., the interfacing-systems LOCA sequence.

The next issue, then, becomes how should the insights from PRA be integrated into this structure to reduce unnecessary burden and make it more rational? In the structuralist model, defense in depth is primary, with PRA available to measure how well it has been achieved.

THE RATIONALIST MODEL

The rationalist model asserts that defense in depth is the aggregate of provisions made to compensate for uncertainty and incompleteness in our knowledge of accident initiation and progression. This model is made practical by the development of the ability to quantify risk and estimate uncertainty using probabilistic risk assessment techniques. The process envisioned by the rationalist is: (1) establish quantitative acceptance criteria, such as the quantitative health objectives, core damage frequency and large early release frequency, (2) analyze the system using PRA methods to establish that the acceptance criteria are met, and (3) evaluate the uncertainties in the analysis, especially those due to model incompleteness, and determine what steps should be taken to compensate for those uncertainties. In this model, the purpose of defense in depth is to increase the degree of confidence in the results of the PRA or other analyses supporting the conclusion that adequate safety has been achieved.

The underlying philosophy here is that the probability of accidents must be acceptably low. Provisions made to achieve sufficiently low accident probabilities are defense in depth. It should be noted that defense in depth may be manifested in safety goals and acceptance criteria which are input to the design process. In choosing goals for core damage frequency and conditional containment failure probability, for example, a judgement is made on the balance between prevention and mitigation.

What distinguishes the rationalist model from the structural model is the degree to which it depends on establishing quantitative acceptance criteria, and then carrying formal analyses, including analysis of uncertainties, as far as the analytical methodology permits. The exercise of engineering judgement, to determine the kind and extent of defense in depth measures, occurs after the capabilities of the analyses have been exhausted.

A PRELIMINARY PROPOSAL

The structuralist and rationalist models are not generally in conflict. Both can be construed as a means of dealing with uncertainty. Neither incorporates any reliable means of determining when the degree of defense in depth achieved is sufficient. In the final analysis, they both depend on knowledgeable people discussing the risks and uncertainties and ultimately agreeing on the provisions that must be made in the name of defense in depth. The fundamental difference is that the structural model accepts defense in depth as the fundamental value, while the rationalist model would place defense in depth in a subsidiary role.

The remaining question is which model provides the better basis for moving forward with risk-informed regulation. How can capricious imposition of defense- in-depth be prevented from undermining the focus that can be provided by risk- informed methods of regulation? PRA methods have identified gaps in the regulations and in the safety profiles of individual plants. They have also identified regulations and plant systems that do not make a significant contribution to safety. Typically, however, regulatory reactions to findings that regulations or plant systems are superfluous to safety have been less aggressive than reactions to apparent safety deficiencies.

Two options can be identified:

(1) Recommend defense in depth as a supplement to risk analysis (the rationalist view)

(2) Recommend a high-level structural view and a low-level rationalist view.

Option (1) requires a significant change in the regulatory structure. The place of defense in depth in the regulatory hierarchy would have to change. The PRA policy statement could no longer relegate PRA to a position of supporting defense in depth. Defense in depth would become an element of the overall safety analysis.

Option (2) is to a large degree compatible with the current regulatory structure. The structuralist model of defense in depth would be retained as the high-level safety philosophy, but the rationalist model would be used at lower levels in the safety hierarchy. An example is shown in Figure 1.

The PRA uncertainties increase as we move from the initiating events to risk (from left to right). The structuralist view dictates that intermediate goals be set, such as core damage frequency (CDF), large early release frequency (LERF) or conditional containment failure probability (CCFP),or frequency-consequence (F-C) curves. This would satisfy the requirement of balance between prevention and mitigation. We note that the actual numerical value chosen for core damage frequency can express a preference for prevention, and such a preference is unrelated to defense in depth. One could proceed and set goals at the "cornerstone" level, i.e., one level below. This could include goals on initiating- event frequencies, safety-function or safety-system unavailabilities, and so on. How far down one would go would be a policy issue. The structuralist view would not be applied at lower levels.

The rationalist model would be applied at levels lower than the cornerstones of Figure 1. Defense in depth would be used only to address uncertainties in PRA at the lower levels, thus becoming an element of the overall safety analysis. For events or processes that are not modeled in PRA, defense in depth would play its traditional role. Such is the case with the impact of smoke from fires on plant safety. Current fire risk assessments do not account for the effects of smoke, therefore, prescriptive defense-in-depth based measures would be taken to limit this impact.

We view Option (2) as a pragmatic approach to reconciling defense in depth with risk-informed regulation. There can be little doubt, however, that the rationalist model, Option (1), will ultimately provide the strongest theoretical foundation for risk-informed regulation. When more experience has been gained with the application of PRA in the design and regulation of nuclear power plants, when PRA models can adequately treat most of the phenomena of interest, the role of defense in depth can and should be changed to one of supporting the risk analyses. This transition will need to be supported by the development of subsidiary principles from which necessary and sufficient conditions could be derived.
Note

The views expressed in this paper are the authors' and do not necessarily represent the views of the Advisory Committee on Reactor Safeguards

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3. F. E. Haskin, and A. L. Camp,, "Perspectives on Reactor Safety," NUREG/CR-6042, Nuclear Regulatory Commission, Washington, DC, March 1994.

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5. International Nuclear Safety Advisory Group, "Defense in Depth in Nuclear Safety," INSAG- 10, International Atomic Energy Agency, Vienna, Austria, 1996

6. U. S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities; Final Policy Statement," Federal Register, 60 FR 42622

7. U. S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Regulatory Guide 1.174, June 1998

ATTACHMENT 10: COMMISSIONER DIAZ' COMMENTS ON SECY-02-0057 AND THE LOCA

The Commission's vote on SECY-02-0057 is a decision that transcends the regulation of the reliability and availability of core cooling for the management and mitigation of reactor transients and accidents. This decision will reflect on our capability and our commitment to be a risk-informed agency utilizing present regulatory tools, technology and operational safety experience.

The Commission's decision to risk-inform our regulations was based on two simple principles: if it is not risk-significant, it is not important to safety, and, the focus of our regulations and resources will be on the issues more important to safety. In this regard, the low risk-significant Large Break Loss-of-Coolant-Accident (LBLOCA) is a true anachronism in today's safety construct, consuming resources that should be directed to the more risk-significant issues. The LBLOCA is a very small component of the Loss-of-Coolant-Accident (LOCA) contribution to risk, and the LOCAs are a small contributor to the total risk. Therefore, the LBLOCA is using resources disproportionate to its importance to safety, taking attention and resources from what we know is much more important. The LBLOCA was good in 1970, was so-so in 1978, but it is absurd now as a dominant Light-Water Reactor (LWR) safety criterion; it has been rendered obsolete by improvements in safety performance and analysis. Thus, I support a risk informed alternative within the definition of a LOCA as a voluntary option for licensees.

Before I vote on the specifics, I will present an overall justification for such a change and, on the way, some recommendations on "how-to". The NRC has been using four major performance goals to direct and measure the agency's achievements: maintain safety, increase effectiveness and efficiency, increase public confidence and reduce unnecessary burden. I am convinced that only three of these goals (safety, effectiveness and efficiency, and unnecessary burden reduction) are "vectors" amenable to regulatory control, and that increasing public confidence should be the result of the good performance of those three, as well as our ability to communicate the performance well. Safety is not determined by public confidence; public confidence should be established and grow from safety performance that is factually established and is well communicated. I also believe that major changes to our regulations need to improve safety, not just to maintain safety.

While the decision on this SECY impacts on increasing effectiveness and reducing unnecessary burden, my vote is based on improving the safety of the nuclear power plants. Yet, whatever direction is set by the Commission, it will be necessary, indeed indispensable, to properly communicate the supporting safety criteria and performance expectations. A very clear statement of the results of the Commission's decision regarding the ECCS and LOCAs will be needed, presenting the safety case and the benefits of implementation. Thus, I believe that providing rule language now, allowing for a risk-informed alternative to the present LBLOCA requirements that focuses on the appropriate LOCA events and other risk-significant issues will actually improve safety, demonstrate the Commission's commitment to risk-informed regulation and allow for early public participation. The path towards rulemaking will provide many opportunities for participation by all stakeholders and should be conducive to enhancing public confidence in our stewardship of nuclear safety issues.

There is a significant, although not always well utilized, body of knowledge regarding LOCAs and LBLOCAs. I will start with the regulatory definition of a LOCA. The term LOCA is often used quite loosely, but it is very clearly defined in our regulations as "those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant make up system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double ended rupture of the largest pipe of the reactor coolant system" (emphasis added).1 So a loss of reactor coolant is only a LOCA, under NRC regulation, if the cause is a break in the reactor coolant pressure boundary, with the maximum size limited to the largest pipe of the reactor coolant system. Thus, a LOCA is a subset of possible losses of reactor coolant. Furthermore, it is only a full-fledge regulatory LOCA if the loss of coolant exceeds the capability of the reactor coolant make up system. Again, the set of possible loss of coolant accidents that are dealt with in regulatory space are narrowed to those losses well beyond the ordinary. The reason for repeating the obvious is to emphasize that the regulatory definition of a LOCA does not include all possible leaks or breaks. Moreover, the original selection of the break size was not based on a well-established analysis, an analysis that is now possible.

I believe that, as a matter of improving safety, the consideration of very low probability Large Break LOCAs should be addressed as severe accident scenarios rather than as the design basis accident. Effectively, the current LBLOCA would not be a design basis accident when utilizing a risk-informed approach. With the alternative definition of the LOCA the really important, risk-significant, accident scenarios would remain within the design basis; in fact, their consideration would be enhanced by a new focus on their risk-importance.

My decision to support a risk-informed alternative definition of a LOCA in the regulatory context is based primarily on several important factors: the data available (or lack thereof), the ability to "learn" from failures or potential failures and take corrective action, the excellent state of current operational safety, and the existing capability of making sound risk-informed decisions that include state-of-the-art Probabilistic Risk Assessments (PRA). Let me address each in turn.

There are many very significant aspects of the data that can be singled out in the existing body of knowledge regarding the occurrence of coolant leaks and their association with LOCAs. The first is the scarcity of actual LOCA data. There are not enough LOCAs to estimate, with confidence, the frequencies of LOCA-type failures from historical data, particularly for medium to large breaks. It is very difficult to predict medium and large break LOCAs from zero occurences. Of course, the lack of data should really be construed as a success story, but it is a curse to analysts seeking to establish failure rates. Thus, all kinds of failure data (cracks, pinholes, leaks or ruptures) are brought in to substitute for actual LOCA data. Much of this failure data can be made useful to provide failure estimates. Piping failures are among the "easiest" -- but not easy -- to estimate since there is at least some reasonable body of knowledge that could be used to predict piping crack growth and potential ruptures. Failures due to human behavior (for example, failure to take corrective action) are much harder to predict. Yet, a Large Break LOCA has not occurred in any nuclear power plant in the world, good plants or bad plants, nada.

Due to the lack of actual data, medium and Large Break LOCA frequencies are very conservatively estimated by calculating the frequency of leaks or through-the-wall cracks that have challenged piping integrity. What about small-break LOCAs? They are about two orders of magnitude more probable. Of course, one very famous "small" LOCA occurred at TMI-2 from the failure of a valve to close, a failure augmented by human error. Indeed, it is in this area where data and PRAs demonstrate the need for regulatory concern. It has been more than 23 years since it has been well known that the LOCA risk is dominated by the small break LOCAs. Therefore, that is where our "attention and resources" should be focused.

Another reason for my decision to support a risk-informed alternative stems from the ability of nuclear regulators, the industry, and the technical infrastructure to learn from and correct actual or potential failures. This is especially true for significant safetyrelated failures or LOCA-type failures. It is not surprising that these groups have "learned to learn" from failure and lack of failure, and that everyday they should learn more efficiently from errors, because prevention and mitigation, followed by error minimization are fundamental nuclear regulatory and operational safety principles. No other TMI-type LOCA has occurred since the first occurrence, and that is probably to be expected because of the extensive actions taken to prevent another occurrence. Yet, the fact that the system "learns" and "corrects" is significant.

The capability of well-developed industrial systems to "learn" from errors is well documented. I would venture to add that error learning curves are most predictable in industrialized democracies, and that errors -especially those well publicized -are corrected rapidly in order to address the real or perceived risk that society associates with the industrial activity. Errors or failures in nuclear power plants are well publicized. Furthermore, the higher the perception of risk to society from an activity, the quicker and more successful should be the learning process.

The nuclear industry, after a somewhat shaky start --- due mostly to human errors in the design, construction and operation of a complex system new to the marketplace --learned well after the shock of TMI. When a rare and significant event like the DavisBesse hole-in-the-head occurs, the industry and the regulator are forced to learn and act quickly. It is highly improbable that another Davis-Besse type failure will occur in the U.S.A. because of the corrective actions that have been and are being taken based on what has been learned. One hole-in-the-head is bad enough. Other new and unknown occurrences will surely take place and, therefore, capabilities to mitigate the more probable and risk-significant spectrum of failures should be given more attention.

The learning has not been limited to major events -a la TMI -but also has included a significant part of operational safety issues. For example, once Intragranular Stress Corrosion Cracking (IGSCC) and Flow-Assisted Corrosion were identified as emergent failure mechanisms, the industry "learned" and the failure rates decreased almost exponentially as a function of accumulated experience. This is neither unique nor laudable: it is normal and expected.

Presently, the NRC and licensees are justifiably focused on the cracks found on PWR vessel head penetration nozzles and welds. I expect that this issue, due to the attention it is properly receiving, should not result in changes to the medium or Large Break LOCAs' frequencies. In the realm of reasonable assurance, it is reassuring to observe that in this country no error or failure from the operation of nuclear power plants has come close to breaching the very stringent safety standards established for the protection of public health and safety embodied in the NRC's strategic goals. We are committed to maintaining this record. The point is, the NRC now regulates in a "learned" and "learning" environment, a statement supported by the present operational safety performance of the plants in this mature industry. This fact allows us to conclude that significant new "errors" should be discovered and corrected before progressing to large failures, and more specifically, this environment should further decrease the probability of a LBLOCA.

When estimating failure rates, regulators today should focus not only on the existence of failures or errors --- many of which are due to human performance -- but also on the ability of the learned systems to cope with the failure, to detect deficiencies, to minimize consequences, to prevent --- or decrease significantly --- recurrence, and to properly value success. A truly effective regulatory system should balance the error data with the expected learned-system behavior to estimate future failure rates. This would be directly applicable to potential failures of the reactor coolant pressure boundary, and certainly applicable to LBLOCAs.

Another consideration in my decision to support a risk-informed alternative is the fact that the capability for making risk-informed decisions, based on relevant experience, deterministic models, defense in-depth and state-of-the-art PRA exists today. This capability is not equally utilized by everyone, but it is here. Selecting a risk-informed alternative to the LOCA rupture size will require this capability at the expert level, with an acceptable -in regulatory space -high quality PRA. It is important to point out that I believe that the precise size of the large break is not a risk determinant issue; there are many other more risk-significant issues.

I now offer the following specific proposal on how to better reduce to practice the "LOCA failure analysis and frequency estimation":

> By December 31, 2003, the staff shall present to the Commission a comprehensive "LOCA failure analysis and frequency estimation" that is realistically conservative and amenable to decision-making. Realistically conservative estimations, with appropriate margins for uncertainty, should be used. Unrealistic extrapolation of estimates to time periods beyond the knowledge base and those requisite time periods used by the industry to inspect, monitor, and correct should not be used. Full understanding of the LOCA frequencies has always been important, but it is time that it becomes a short-term high priority. The goal is to achieve a predictive and well managed safety envelope emboding the best data and the best methods.

To achieve the objective of the above proposal I believe the following must be done:

a. Use a 10-year period for the estimation of LOCA frequency distributions, with a rigorous re-estimation conducted every 10 years and a sanity check for new types of failures every 5 years. This periodicity is consistent with the In-Service Inspection (ISI) program required of all reactor licensees. Longer periods do not make sense, neither technically nor from a regulatory perspective.

b. Conduct a practical reconciliation of LOCA frequency distributions by the 1) expert use of service-data, 2) Probabilistic Fracture Mechanics (PFM) and 3) expert elicitation to converge the results. Limiting the interval to 10 years will benefit significantly all three methods, using realistic predictability and convergence of results as necessary criteria. I strongly recommend that both service-data and PFM estimates be "reduced" to an appropriate set by "expert discrimination" of what data should be treated. Not all data is "born" equally nor should it be treated equally. For the purpose of LOCA estimation, a better discrimination of failure data is needed before it is used as predictive data. This is an area that needs prompt and expert attention. Service-based LOCA estimates (a statistical analysis of service experience data) are more useful than PFM, especially if the projection is limited to 10 years. PFM (a phenomena-based method using fracture and failure analysis) can make a contribution, more so if it is used to selectively converge to service data predictions.

c. Finally, expert elicitation should use the converged (whenever possible) service-data and PFM results to provide the Commission a comprehensive "LOCA failure analysis and frequency estimation" predictive envelope that is realistically conservative. Expert elicitation is better when the data and analysis methods have first been screened for that purpose, and I believe that this has not yet been done.

In a related matter, in a briefing of Commission Technical Assistants on April 22, 2002, the staff stated that it is possible for some pipes to fail without a precursor leak (no leakbefore-break) and that this contribution to the pipe break probability should be included in the analysis. I believe that leak-before-break is an established technological fact for risk-significant failures and the Commission should be informed and kept up-to-date on the staffs efforts in this area. I prefer to deal with actual probabilities and not with all possibilities.

One final comment on the above recommendations. As a regulator, I want to know, with significant confidence what the failure rate estimates are for next year, and the year after. For both rulemaking and regulatory oversight, 10 year scenarios are very good; furthermore, I know we can do it even better the next time around. Also, for any safety reason, we can and should take any needed action, as the circumstances require. No service-data, no PFM and no expert elicitation can confidently predict beyond 10 years, nor do we need to using a risk-informed approach.

In summary, the re-consideration of the Large Break Loss-of-Coolant Accident has been a long time in the making. I am convinced that we now have the necessary justification to make this fundamental change to the Light-Water Reactor safety regulatory construct now. Therefore, I vote as follows:

1. With regard to the re-definition of the Large Break LOCA:

The staff should prepare a proposed rule change to 10 CFR Part 50 that allows for a risk-informed alternative to the present maximum LOCA break size. I believe the rule should be very specific and leave no doubt that the pertinent risk parameters are addressed and only the non-significant contributions to risk are handled through severe accident risk management. For example, the modified definition-of the LOCA, for use throughout Part 50 and wherever applicable, could read:

> Loss of coolant accidents (LOCA). Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system *or up to an alternate maximum break size determined by including at least XX% [e.g., 95%, 96%...] of the LOCA failure contributors to core damage frequency.*

Thereby, the most significant failures are included. The net effect of this change would not reduce protection or give up risk sequences; rather, the rule will establish a new risk-informed design basis accident. Only those failures smaller than the average by about two orders of magnitude would be removed for severe accident management; i.e., the capability to mitigate the double-ended rupture of the largest pipe in the reactor coolant system will be retained under severe accident management principles and activities.

While I would expect pertinent changes in the design basis and associated analysis to naturally occur, I concur with the staff that changes in hardware and operation "would require that it be demonstrated that the ECCS functional reliability is commensurate with the frequency of accidents in which ECCS success would prevent core damage or a large early release". In other words, I am not supporting changes to functional requirements unless they are fully risk-informed and protective of public health and safety. For example, I would not support actual changes to ECCS coolant flow rates or containment capabilities to mitigate accidents. I would support changes that provide for risk-informed sequencing of equipment with demonstrated functionality and reliability requirements that arise from the alternate criteria.

There is also no doubt that the redefinition of the LBLOCA would also require strict configuration controls, including during Low Power and Shutdown (LPSD) operations. Thus, I support requiring these strict configuration controls and believe that the ROP, the revised Maintenance Rule and Reg. Guide 1.174, are suitable for use in addressing such requirements.

One last point on the alternate break size. The conservative CDF and LERF safety criteria of Option 3, and particularly the capability of Reg. Guide 1.174 to deal both with absolute (CDF) and relative (delta-CDF) changes, are essential to effect an alternative break size with reasonable assurance of adequate protection.

Furthermore, as discussed above in the recommendation for determining LOCA frequency distribution amenable to decision-making, the rulemaking should be supported by a 10 year estimation of LOCA frequencies, to be delivered by December 31, 2003. This should be done in parallel with the rulemaking activities.

2. Regarding the recommendations in SECY-02-0057:

I approve the staff recommendations to proceed with rulemaking changes to 10 CFR 50.46, 10 CFR Part 50, Appendix K, and GDC 35, sooner rather than later, including an option to the Appendix K evaluation model requirement to permit use of a decay heat model based on the 1994 ANS standard. I support the unbundling and pursuing of separate rulemaking for each of the proposed changes. In order to improve the timeliness, I also approve not preparing a separate rulemaking plan for each rulemaking. However, I strongly believe we should seek early public and stakeholder comments on all of these proposals.

The staff proposed allowing the use of a decay heat model based on the 1994 ANS standard and stated that concerns with uncertainties and conservatism associated with the current standard would be addressed separately from any proposed rulemaking. This is a prudent approach. A similar approach could be used to handle issues separate from the rulemaking when pursuing rule changes associated with the redefinition of the Large Break LOCA.

Risk is measurable and manageable, and risk-informed decision-making is a very good tool to improve safety. It is available now, and I strongly recommend we use it for this particular significant issue in a manner protective of public health and safety