February 23, 2001

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research /RA/

SUBJECT: SMIRT 16 CONFERENCE PAPER

Attached for your information is a paper we have prepared for presentation at the SMIRT 16 Conference, to be held in Washington, DC, in August 2001. I am one of the co-authors, and may present the paper at the conference.

The paper, entitled "Using PRA Information to Identify Changes in Nuclear Reactor Safety Regulations," includes no discussion of policy issues. Its focus is recent staff work to identify and assess possible risk-informed changes to 10 CFR 50. More specifically, it discusses the staff's recommendations on changes to 10 CFR 50.44, our ongoing work to assess possible changes in 10 CFR 50.46, Part 50 special treatment requirements, and 10 CFR 50.61. The content of the paper is drawn from existing Commission papers and memoranda, including SECY-00-0140, SECY-00-0198, and Mr. Travers' February 5, 2001, memorandum to the Commissioners ("Third Status Report on Risk-Informing the Technical Requirements in 10 CFR Part 50 (Option 3)"). Except for Mr. Travers' February 5 memorandum, all material used and referenced is publically available. The February 5 memorandum should also be made publically available in the next few weeks.

If you have any questions or comments, please feel free to discuss them with me.

Attachment: As stated

cc: S. Collins

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Using PRA Information To Identify Changes in Nuclear Reactor Safety Regulations

Ashok C. Thadani, Thomas L. King, Mark A. Cunningham, and Mary T. Drouin
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC USA

ABSTRACT

The United States Nuclear Regulatory Commission's 1995 Probabilistic Risk Assessment Policy Statement indicates that "the use of PRA technology should be increased in all regulatory matters . . . in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." Using this guidance, the NRC staff has undertaken several projects to make the regulations contained in 10 CFR Part 50 impose regulatory burdens on licensees that are commensurate with their safety importance. For instance, the NRC staff is studying its reactor safety requirements in 10 CFR Part 50 to identify areas of unnecessary conservatism and potential additional safety requirements and to assess the feasibility of alternative approaches to changing the requirements. This study has led to recommendations on specific changes to requirements on combustible gas control during accidents. Additional recommendations are anticipated in 2001 with respect to the technical feasibility of changes to emergency core cooling and "special treatment" requirements.

INTRODUCTION

The United States Nuclear Regulatory Commission (NRC) has made use of probabilistic risk analysis (PRA) information for many years. A key milestone in this use was the issuance of the Commission's 1995 PRA Policy Statement [1] that indicated that: "the use of PRA technology should be increased in all regulatory matters . . . in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." Using this guidance, the NRC staff has undertaken several projects to make the regulations contained in 10 CFR Part 50 [2] impose regulatory burdens on licensees that are commensurate with their safety importance.

In one of the projects, the staff is studying the 10 CFR Part 50 technical requirements to identify areas of unnecessary conservatism and potential additional safety requirements. The staff has developed and is now using a general framework for identifying and prioritizing potential changes [3]. The framework has been used to identify and assess the technical feasibility of making changes to requirements contained in 10 CFR 50.44 [4], which defines the Commission's requirements on control of combustible gases during reactor accidents.

The staff is evaluating the potential value and technical feasibility of changing the requirements contained in 10 CFR 50.46 [5], which defines the Commission's requirements for reactor emergency core cooling systems, and modifying and consolidating the 10 CFR Part 50 requirements for "special treatment" of important systems, structures, and components. In a parallel but related activity, the staff is now reevaluating the technical basis of 10 CFR 50.61 [6], which defines the Commission's requirements regarding pressurized thermal shock accidents.

FRAMEWORK FOR EVALUATING POTENTIAL CHANGES

The staff has developed a framework that describes the approach, process, and guidelines the staff will apply in reviewing, formulating, and recommending risk-informed alternatives to 10 CFR Part 50 technical requirements. The framework is provided in [3]; some of its key features are as follows:

- The framework utilizes a risk-informed, defense-in-depth approach to accomplish the goal of protecting public health and safety. This defense-in-depth approach builds on previously established guidance in: (a) Regulatory Guide 1.174 [7], (b) the Commission's White Paper on risk-informed and performance-based regulation [8], (c) the reactor oversight program [9], and (d) the Advisory Committee on Reactor Safeguards (ACRS) recommendations on defense-in-depth [10].
- The defense-in-depth approach includes elements that are dependent upon risk insights and elements that are employed independent of risk insights. Risk insights are used to set guidelines that limit the frequency of accident initiating events; limit the probability of core damage, given accident initiation; limit releases of radioactive material during core damage accidents; and limit public health effects caused by releases of radioactive material.

  Safety function success probabilities (commensurate with accident frequencies, consequences, and uncertainties) are achieved via appropriate redundancy, independence, diversity, defenses against common-cause failure mechanisms, defenses against human errors, and safety margins.

Defense-in-depth elements are employed independent of risk insights: prevention and mitigation are maintained; reasonable balance is provided among prevention, containment, and consequence mitigation; over-reliance is avoided on programmatic activities to compensate for weaknesses in plant design; independence of barriers is not degraded; and the defense-in-depth objectives of the current General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 [11] are maintained.

As a working definition, for use in the study, defense-in-depth is assessed by the application of the following strategies to protect the public:

- limit the frequency of accident initiating events
- limit the probability of core damage given accident initiation
- limit radionuclide releases during core damage accidents
- limit public health effects caused by core damage accidents

- The framework considers both design-basis and beyond-design-basis core-melt accidents.
- The framework considers uncertainties.
- For risk significant accidents in which one or more of the high-level strategies are precluded (e.g., containment bypass accidents), the remaining strategies may be more tightly regulated; that is, regulations should provide a very high confidence in the remaining strategies. Similarly, more stringent requirements may be imposed in the presence of large uncertainties regarding the effectiveness of one of the strategies.
- Following Commission direction [12], the staff is using metrics consistent with the Commission's Safety Goals [13], and associated subsidiary objectives, to define how safe is safe enough. That is, the framework is constructed in such a way that risk-informed alternatives to 10 CFR Part 50 will be developed consistent with this direction. The framework uses quantitative guidelines, based on the Safety Goals and its subsidiary objectives, to assist the staff in determining the appropriate balance between prevention and mitigation and whether or not to recommend a risk-informed alternative to the current requirements.

The framework is used for identifying candidate regulations and DBAs that are candidates for a risk-informed evaluation, for performing the evaluations, and for identifying proposed changes. The staff has performed an initial screening of the regulations and DBAs of 10 CFR Part 50. As a result of this screening, the staff identified 23 regulations in and 9 appendices to 10 CFR Part 50 as potential candidates for change [14].

In the process of risk-informing existing regulations, it is also important to identify risk-significant events not explicitly addressed in current regulations. An initial attempt has been made to find "holes" in the current Part 50 regulations on issues that are important to accident risks. Some risk-significant accident types and related events do not find any mention in the current regulations. Examples include seal loss of coolant accidents (LOCAs), which are important contributors to pressurized water reactor (PWR) core damage frequencies, and liner melt-through, which can be a significant contributor to LERF for boiling water reactors (BWRs) with Mark I containments.

Application of the framework provides an assessment of the relative priority of changing Part 50 technical requirements and, for those requirements identified as higher priority, an evaluation of the technical feasibility of potential changes. The staff makes recommendations to the Commission for initiating rulemaking for those technical requirements appearing feasible to change. Upon Commission approval, the staff proceeds to perform the rulemaking, including development of the regulatory and detailed technical analyses. As discussed below, the staff has recently begun such a process for changing its requirements on combustible gas control during accidents.

ASSESSMENT OF COMBUSTIBLE GAS CONTROL REQUIREMENTS

In June 1999, the Commission approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50 [15]. The Commission specifically directed the staff that, "if the staff identifies a regulatory requirement which warrants prompt revision . . ., the Commission should be . . . provided with a recommended course of action." In April 2000, the staff notified the Commission [14] of its intention to develop recommendations for changing 10 CFR 50.44, which provides agency requirements with respect to the control of combustible gases such as hydrogen that could be generated during accidents, subsequently burn or detonate, and challenge the integrity of containment structures.

Based upon current risk information and research results, the staff concluded that little to no risk significance or benefit was associated with some of the combustible gas control requirements of this regulation, resulting in a potential for unnecessary burden. In addition, the staff concluded that the current requirements did not address some risk-significant concerns from accident scenarios. Therefore, the staff recommended changes to the requirements that include both safety enhancements (some of which may have an associated additional burden) and reductions in unnecessary burden.

A detailed discussion of the staff's feasibility study and recommendations is provided in [3]. In summary, the staff considers the work described in [3] sufficient to establish the feasibility of risk-informed changes to the technical requirements of 10 CFR 50.44 and recommends the following for a risk-informed alternative to that regulation:

- Specify in the regulation a specific combustible gas source term using best available calculational methods for a severe accident that includes in-vessel (and ex-vessel) hydrogen and carbon monoxide generation in such a way that the alternative regulation addresses the likely sources of combustible gases. These sources would only address challenges to the containment that could result in a large release of radioactive material within 24 hours after the onset of core damage.
- Eliminate the requirement to measure hydrogen concentration in containment. Hydrogen monitoring is not needed to initiate or activate the combustible gas control systems for each type of containment, hence hydrogen monitors have a limited significance in mitigating the threat to containment in the early stages of a core-melt accident.
- Retain the requirement to ensure a mixed atmosphere. The intent of this requirement is to maintain those plant design features (e.g., open compartments) that promote atmospheric mixing, and this is considered an important defense-in-depth element.
- Eliminate the requirement to control combustible gas concentration resulting from a postulated LOCA. This type of accident is not risk significant and the means to control combustible gas concentration (e.g., recombiners) do not provide any benefit for the risk-significant accidents or, if a vent-purge method is used, can result in unnecessary releases of radioactive material to the atmosphere.
- Retain the requirement to inert BWR Mark I and Mark II containment structures. Removal of this requirement would result in the integrity of these structures being highly vulnerable to gas combustion.
- Retain the requirement for high point vents in PWR reactor coolant systems (RCS). Combustible gases in the RCS can inhibit flow of coolant to the core; therefore, the capability to vent the RCS provides a safety benefit in its ability to terminate core damage.
- Modify the requirement for the hydrogen control system for BWR Mark III and PWR ice condenser containment structures to control combustible gas during risk-significant core-melt accidents (e.g., station blackout). Since the control system uses igniters that are dependent on alternating current (ac), under station blackout conditions these containments may remain vulnerable to gas combustion. Alternatively, if station blackout could be shown by the licensee to be of low enough frequency, with due consideration of uncertainties and defense-in-depth, the sequence would not be risk significant and the licensee would comply with the requirement with the current igniter system. Such an approach represents a performance-based aspect of this recommendation.
- Include a performance-based second alternative within this regulation that would allow a licensee to use risk information and plant-specific analysis with respect to the generation and control of combustible gases. Licensees would be permitted to demonstrate that the plant would meet specified performance criteria (e.g., maintain containment integrity for at least 24 hours for all risk-significant events). This alternative may be especially attractive for future plants.
- Recommend that long-term (more than 24 hours) control of combustible gas be included as part of the licensee's Severe Accident Management Guidelines (SAMG) since combustible gases still pose a challenge to containment integrity in the long term with the possibility of a large, late radionuclide release.

It is recognized that, since this recommendation is based upon a feasibility study, additional work is required to support the actual rule change. In addition to the calculation of the combustible gas source term discussed earlier, such work would include performing detailed regulatory analyses on safety enhancements that have the potential to pass the NRC's backfit rule requirements [16], assessing the relation to and need for conforming changes in emergency operating procedures and SAMGs, assessing the implications of fire and seismic events on the combustible gas control system requirements in BWR Mark III and PWR ice condenser plants, and developing regulatory guides to implement the performance-based aspects of the recommended alternative rule.

In January 2001, the Commission approved the staff's recommendation [17], indicating that "the staff should proceed expeditiously with rulemaking on the risk-informed alternative version of 10 CFR 50.44, including completion of outstanding technical work (e.g., development of the combustible gas source terms) and necessary regulatory analyses." The staff is now proceeding to implement the changes to 10 CFR Part 50 via its rulemaking process.

ASSESSMENT OF EMERGENCY CORE COOLING REQUIREMENTS

The staff has initiated an assessment of potential changes to requirements for emergency core cooling systems contained in 10 CFR 50.46 [5]. Currently, the staff is evaluating two different approaches [18] - one proposed by the staff and a second proposed by the Westinghouse Owners Group (WOG). The staff approach under consideration consists of the following three phases:

- Phase 1: Assessment of the Large-Break Loss of Coolant Accident (LB LOCA) with Respect to Emergency Core Cooling System (ECCS) Requirements. This assessment would consider whether or not operating experience, fracture mechanics, thermal-hydraulic analysis, and previous staff decisions on the "leak before break" issue warrant defining a different LOCA as a design basis accident. Such a redefinition of the LOCA design basis accident would involve the consideration of many factors and could ultimately be judged not feasible. In addition, the staff assessment would consider whether or not the current assumptions and practices for analysis of the LB LOCA 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.
- Phase 2: Assessment of the LB LOCA with Respect to Other Plant Design Requirements. This phase would address whether or not changes to other plant design requirements (other than the ECCS requirements in 10 CFR 50.46) dependent upon the LB LOCA assumptions are warranted, based upon risk insights and Phase 1 results. This assessment could, for example, lead to changes to containment design and operational requirements.
- Phase 3: Assessment of the Current ECCS Acceptance Criteria. This phase would focus on assessing the acceptance criteria (2200°F and 17% clad oxidation) currently in 10 CFR 50.46. Current knowledge regarding cladding materials and burnup effects and risk insights will be used. If experimental work is needed, this phase may require considerable time to complete.

The WOG has taken the lead (with support from the other owner's groups) for industry to work with the staff on risk-informing 10 CFR 50.46. In public meetings, WOG has proposed an approach that is rather different than that of the staff, described above. The WOG approach would focus on redefining the LB LOCA design basis accident and would apply the redefined LB LOCA to all current requirements that are dependent upon or that utilize the LB LOCA assumption. More specifically, the WOG approach differs from the staff approach in two ways:

- Instead of a phased approach, all regulations affected by a redefinition of the LB LOCA would be addressed at one time.
- Only those technical requirements affected by a redefinition of the LB LOCA would be included. Reassessing analysis assumptions and acceptance criteria would then be of lower priority, if addressed at all.

The staff is currently assessing the costs and benefits of both proposals, including assessing the feasibility of redefining the LB LOCA. When this assessment is completed, the staff will establish a detailed schedule for completion of the feasibility study on risk-informing 10 CFR 50.46. The staff expects to provide this schedule in June 2001.

ASSESSMENT OF SPECIAL TREATMENT REQUIREMENTS

The staff is continuing to assess the technical aspects of current special treatment requirements contained in 10 CFR Part 50. In other work [19](the staff's "Option 2" work), the staff is addressing risk-informed changes to the regulatory scope for structures, systems, and components in need of special treatment (e.g., quality assurance, environmental qualification). The Option 2 work does not address changing the technical content of special treatment requirements, the design of the plant, or the design-basis accidents. In its "Option 3" work, the staff is assessing the risk-significance of technical requirements associated with the special treatment requirements in 10 CFR Part 50. The feasibility study is expected to be complete and recommendations provided to the Commission in the fall of 2001.

ASSESSMENT OF PRESSURIZED THERMAL SHOCK REQUIREMENTS

The Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61[6], was established in 1983 to resolve an issue concerning the integrity of embrittled pressurized water reactor (PWR) pressure vessels that involved a rapid cooldown of the inside wall of the vessel, accompanied by either sustained high reactor coolant system pressure or a subsequent repressurization of the system. The rule included a specified numerical value of a materials parameter (RT_{PTS}) that would be used as a screening criterion, above which licensees would be required to demonstrate that their pressure vessels could be operated safely. RT_{PTS} is a measure of the material toughness of the vessel at the end of its licensed life and the ability of the vessel materials to withstand a PTS event.

Since the rule was established, the staff has accumulated considerable experience with application of the rule and the associated Regulatory Guide, 1.154 [20], and they have performed extensive research on the key technical issues underlying the rule. With respect to the regulatory guide, experience has shown that it is difficult to use. Analyses performed as part of this research suggested that the rule could have conservatism could be reduced while still providing reasonable assurance of adequate protection to public health and safety.

The staff initiated a program in 1999 to revisit the technical bases for the PTS Rule, and if appropriate, to propose a revision to the rule and the regulatory guide. This revisitation and possible rule revision are intended to continue to provide reasonable assurance of adequate protection, improve the realism of the rule by incorporating these research results as well as current methods in thermal-hydraulics and probabilistic risk assessment, reflect current agency guidance on the use of risk information in regulatory activities, reduce licensee burden by eliminating unnecessary conservatism in the rule, and clarify the implications of the research results for PWRs that could approach the screening criterion, and provide for public participation during the revisitation of the technical basis and any subsequent rulemaking.

This overall program is described in SECY-00-0140 [21]. Key elements of the program include:

- Initiating Events and Their Frequency. This element provides information on the types of initiating events that could lead to PTS events and the frequencies of these events. The staff will review previous PTS studies, review more recent PRAs and operational events to identify new initiators, and estimate the frequencies of these initiators.
- Thermal Hydraulics. The thermal hydraulics element will provide the reactor vessel downcomer temperature and pressure boundary conditions for the fracture mechanics analysis, using state-of-technology computer models.
- Probabilistic Fracture Mechanics. The probabilistic fracture mechanics element of staff work will provide estimates of the probabilities of through-wall cracks for each of the sets of initiators and thermal hydraulic conditions identified in previous elements. This work will make use of the extensive research performed by the staff.
- Probabilistic Aspects of PTS Screening Criterion. In parallel with the development of revised technical information on PTS events and their frequencies and consequences, the staff is reassessing the basis for the "acceptable" frequency of such events. This reassessment will be performed to reflect guidance on the use of risk information established since the PTS rule was completed, including the Safety Goal Policy Statement [13], Regulatory Guide 1.174 [7], and the framework being used in the other Part 50 assessments [3].
- PTS Through-Wall Crack Frequency. The frequency of a through-wall crack, which is considered to be equivalent to vessel failure and core damage, will be estimated in this element. This frequency will consider all initiators identified in the first element and their frequencies, thermal hydraulic information, and probabilistic fracture mechanics information. A simple analysis (involving less than six staff-months of effort) of the impact of such vessel failures on containment performance will also be performed as part of this element. Uncertainties in these frequencies will be estimated.
- PTS Screening Criterion. The staff will develop recommendations for new values of RT_{PTS}, using the results of the PTS analyses and the reassessment of the probabilistic aspects of the screening criterion.
- Technical Basis for Revision of 10 CFR 50.61. The information created and assembled in previous tasks will be integrated into a form that will support a new version of the rule and regulatory guide. When completed, this material will be provided to the Commission with a recommendation on whether or not to proceed with rulemaking, as well as the priority of this rulemaking relative to other risk-informed Part 50 rulemakings.

By present schedules, this program will be completed in early FY2002.

ASSESSING OTHER POTENTIAL CHANGES

In parallel with the activities described above to evaluate specific regulations, the staff continues to assess and prioritize other potential changes to Part 50 regulations, including requirements as well as related design basis accidents. Factors that are being used in this prioritization include:

- potential for improving safety
- potential for reducing licensee and NRC burdens
- the anticipated complexity of changes
- NRC resources needed for putting changes in place (both short-term and long-term)
- licensee resources needed for putting changes in place
- calendar time for full implementation (NRC and licensee)
- application to current and/or future plants.

Additional factors that may be used in this prioritization include:

- the scope of the risk assessment that is required
- the extent to which risk information can be incorporated into risk-informing the requirement
- whether the risk-informing of a candidate requirement would require risk-informing other related requirements
- the number of regulations impacted by a particular design basis accident.

The results of this ongoing prioritization will be provided in the periodic staff status reports to the Commission on this program (i.e., reports such as [3], [15], and [17]).

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4. United States Code of Federal Regulations, Title 10, “Energy,” Chapter I, “Nuclear Regulatory Commission,” Part 50, “Domestic Licensing of Production and Utilization Facilities,” §50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors.”
5. United States Code of Federal Regulations, Title 10, “Energy,” Chapter I, “Nuclear Regulatory Commission,” Part 50, “Domestic Licensing of Production and Utilization Facilities,” §50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors.”
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9. USNRC, “Recommendations for Reactor Oversight Process Improvements,” SECY-99-007, January 8, 1999.
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14. USNRC, “Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3),” SECY-00-0086, April 12, 2000.
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17. Memorandum from Annette Vietti-Cook, USNRC, to William D. Travers, USNRC, “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 to 10 CFR 50.44 (Combustible Gas Control),” January 19, 2001.
18. Memorandum from William D. Travers, USNRC, to the Commissioners, “Third Status Report on Risk-Informing the Technical Requirements in 10 CFR Part 50 (Option 3),” February 5, 2001.
19. USNRC, “Risk-Informed Special Treatment Requirements,” SECY-00-0194, September 7, 2000.
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21. USNRC, “Reevaluation of the Pressurized Thermal Shock Rule (10 CFR 50.61) Screening Criterion,” SECY-00-0140, June 23, 2000.