March 31, 2005

MEMORANDUM TO: Jack R. Strosnider, Director
Office of Nuclear Material Safety
and Safeguards

FROM: Robert C. Pierson, Director **/RA/**
Division of Fuel Cycle Safety
and Safeguards, NMSS

SUBJECT: TECHNICAL PAPER SUBMISSION TO AMERICAN NUCLEAR SOCIETY NUCLEAR CRITICALITY SAFETY DIVISION TOPICAL MEETING

BACKGROUND

The Commission has requested submission of public presentations, in order to assure that discussions of new policy and sensitive topics receive appropriate review before presentation in a public forum. Staff members in the Technical Support Group have prepared four papers which have been accepted at the 2005 American Nuclear Society Nuclear Criticality Safety Division Topical Meeting. This memorandum submits the proposed papers to the Office of Nuclear Material Safety and Safeguards, for review, and consideration of further submittal to the Executive Director for Operations.

DISCUSSION

The following papers are attached for review:

1. "The Criticality Safety Inspection Process at the U.S. Nuclear Regulatory Commission"

2. "Interim Staff Guidance 3, Nuclear Criticality Safety Performance Requirements and Double Contingency Principle"

3. "Justification of Minimum Margin of Subcriticality for Safety"

4. "Qualification of NRC Criticality Safety Reviewers in the Division of Fuel Cycle Safety and Safeguards"

The fore mentioned papers contain no new policy, and the second and third papers, above, have been publicly released. Accordingly, these papers should not require additional review.

Please refer any questions to Tamara Powell, of my staff, at 415-5095.

Attachments: As stated

March 31, 2005

MEMORANDUM TO: Jack R. Strosnider, Director
Office of Nuclear Material Safety
and Safeguards

FROM: Robert C. Pierson, Director **/RA/**
Division of Fuel Cycle Safety
and Safeguards, NMSS

SUBJECT: TECHNICAL PAPER SUBMISSION TO AMERICAN NUCLEAR SOCIETY NUCLEAR CRITICALITY SAFETY DIVISION TOPICAL MEETING

BACKGROUND

The Commission has requested submission of public presentations, in order to assure that discussions of new policy and sensitive topics receive appropriate review before presentation in a public forum. Staff members in the Technical Support Group have prepared four papers which have been accepted at the 2005 American Nuclear Society Nuclear Criticality Safety Division Topical Meeting. This memorandum submits the proposed papers to the Office of Nuclear Material Safety and Safeguards, for review, and consideration of further submittal to the Executive Director for Operations.

DISCUSSION

The following papers are attached for review:

1. "The Criticality Safety Inspection Process at the U.S. Nuclear Regulatory Commission"

2. "Interim Staff Guidance 3, Nuclear Criticality Safety Performance Requirements and Double Contingency Principle"

3. "Justification of Minimum Margin of Subcriticality for Safety"

4. "Qualification of NRC Criticality Safety Reviewers in the Division of Fuel Cycle Safety and Safeguards"

The fore mentioned papers contain no new policy, and the second and third papers, above, have been publicly released. Accordingly, these papers should not require additional review.

Please refer any questions to Tamara Powell of my staff at 415-5095.

Attachments: As stated

DISTRIBUTION:
NMSS r/f FCSS r/f JStrosnider MFederline

ML050740053

OFFICE	TSG	Tech Editor	FCSS	FCSS
NAME	TPowell:dw	Ekraus: By Fax	MGalloway	RPierson
DATE	03/ 16 /05	03/ 23 /05	03/ 18 /05	03/ 31 /05

OFFICIAL AGENCY RECORD

The Criticality Safety
Inspection Process at the
U.S. Nuclear Regulatory Commission

Dennis C. Morey and Melanie A. Galloway
U.S. Nuclear Regulatory Commission
Mailstop T8-F43
Washington, D.C. 20555
dcm@nrc.gov or mam@nrc.gov

Introduction

The Office of Nuclear Material Safety and Safeguards (NMSS) at the U.S. Nuclear Regulatory Commission (NRC), licenses, through the Division of Fuel Cycle Safety and Safeguards, seven fuel fabrication facilities and five critical mass facilities, under 10 CFR Part 70, and two uranium enrichment facilities, under 10 CFR Part 76. Because these facilities contain critical masses of fissile material, NRC conducts criticality safety inspections at them, to assure that adequate measures are in place, as committed to in the licenses, to prevent inadvertent criticality.

This paper discusses: (1) the conduct of Headquarters based criticality safety inspection activities: (2) the difference between NRC Headquarters and region-based criticality safety inspection activities: and (3) other inspection-related functions, such as enforcement.

Programmatic Basis

A license application under either Part 70 or Part 76 requires a commitment to implement a criticality safety program that is fully described in the license or certificate application. In addition, these regulations contain requirements for criticality accident alarm systems. A license application generally requires performance of a criticality safety analysis, resulting in criticality safety limits and controls commensurate with the risk of inadvertent criticality at the facility.

NRC inspection efforts focus primarily on licensee commitments that arise from their applications beginning with criticality safety analyses. The formal basis for the criticality safety inspection program is contained in Inspection Manual Chapter (IMC) 2600, the fuel cycle inspection program. This IMC contains the program direction for all fuel cycle inspection activities, including the resident inspectors at three facilities. Criticality safety inspection is done by Headquaters inspectors, under Inspection Procedure (IP) 88015, or by regional fuel cycle inspectors under IP 88020.

Inspection Procedures

The regional inspection procedure for criticality safety is IP 88020 and concerns primarily control implementation and maintenance/calibration. IP 88020 is performed by region-based fuel cycle inspectors in concert with other fuel cycle inspection activities. Headquarters-based criticality inspectors use IP 88015 to perform focused criticality safety inspections.

Attachment 1

Both inspection procedures contain 11 programmatic inspection areas:

1. Management and Administrative Practices
2. Nuclear Criticality Safety Function
3. Plant Activities
4. Configuration Control
5. Change Control
6. Operating Procedures
7. Maintenance
8. Training
9. Inspections, Audits, and Investigations
10. Criticality Alarm Systems
11. Emergency Response

These inspection areas may change as a result of fuel cycle inspection procedure revisions currently in progress. As the inspection of criticality safety has evolved, the areas of Operating Procedures, Maintenance, and Emergency Response have come to be inspected almost exclusively by regional staff, along with the maintenance and calibration of the criticality alarm system. Inspection of the Nuclear Criticality Safety Function, which includes review of criticality analyses, has come to be performed almost exclusively by Headquarters criticality inspectors.

Inspection Staff

The Division of Fuel Cycle Safety and Safeguards maintains three criticality safety inspectors in the Technical Support Group. These inspectors work solely in criticality safety and have been through a specialized qualification program.[1] The criticality inspectors are based at NRC Headquarters to allow association and interaction with criticality safety technical reviewers. Criticality safety inspectors also are expected to participate in criticality safety professional activities such as technical meetings and standards groups.

Inspection Focus

The NRC Headquarters criticality safety inspection focus is on program adequacy, specifically the completeness and adequacy of the criticality safety basis, beginning with validation. Inspectors select analysis for review, preferably new or changed analysis, and work through the analysis from assumptions to conclusions, which may result in a determination that criticality is not credible or may include requirements for limits on the operation. The inspectors then follow the limits to their resulting controls. Inspectors are responsible to assure not only that controls are implemented, but that they are adequate for the risk of the operation and existing safety margin.

Inspection Approach

Criticality safety inspectors go over the full safety basis to identify the basis for conclusions and the adequacy of calculations, validation, assumptions, reviews, approvals, and implementation.

---

[1] Discussed in Reference 4.

To determine the adequacy of analysis, the inspectors must establish that assumptions are bounding for the situation, that controls are appropriate and that conclusions about credibility make sense.

Criticality safety inspectors are also required to determine that a reasonable safety margin has been established and that controls actually assure the margin. In conjunction with safety margin review, the inspectors must also look at implementation of controls. This will put the inspectors on the operating floor of the facility where they can look for changes as they go over assumptions and look at in-place controls.

Analytical Review

Review of licensee criticality safety analysis is the foundation of the Headquarters criticality safety inspection effort. Having identified an appropriate slice of licensee analysis based on recent changes, non-compliances, events, or risk-significance, the inspectors follow the analysis through to implemented controls, beginning with the basic assumptions.

Criticality safety inspectors look at assumptions to assure that they reasonably bound the analyzed system or situation. Based on event experience, inspectors are particularly interested in assumptions that are based on parameters unrelated to neutron multiplication.[2] Another concern is reliance on an assumed limiting configuration that is not fixed or limited in any way[3]. Assumptions are of great interest when criticality is deemed not credible.

When criticality is deemed credible, the inspectors review the limits identified in the analysis so that the basis for them is clearly understood. The safety margin must be adequate in terms of primary parameters.[4]

When calculations are used to justify a limit, the inspectors will review the adequacy of models used and whether the licensee understands the bias relating to the system or situation (an upset condition, for example) analyzed. The inspectors are reviewing to determine that calculations reasonably and conservatively bound the system.

Criticality Alarm Systems

Regional fuel cycle inspectors, who may be at the facility more often than Headquarters criticality inspectors, review criticality alarm systems for maintenance, calibration, and changes to operations that may affect alarm system performance or coverage.
Criticality safety inspectors customarily review alarm placement and supporting calculations; overall system reliability (i.e., logic); and detector reliability. The basic regulatory requirement is

---

[2] For example, saturation level of an isotope that does not have a limiting effect on neutrons.

[3] For example, assuming a fissile content merely because it is substantially greater than normally seen, rather than actually limiting.

Reference 5.

to have dual detector coverage of fissile material operations, and within that framework, licensee commitments vary widely.

Event Review

NRC Headquarters criticality safety inspectors monitor event reporting and review criticality safety-related events to identify enforcement issues, licensee-specific trends, or generic trends. Event review typically involves interaction with Headquarters technical and licensing staff, along with regional fuel cycle inspection staff. Criticality safety inspectors may participate in activities ranging from response to events from reactive inspections, to drafting generic correspondence such as information notices.

Enforcement

The consolidation of most fuel cycle inspection activities at Region II, in October 2003, reduced NMSS enforcement resources. As a result, enforcement actions not arising directly from stand-alone Headquarters criticality safety inspections will likely be pursued out of Region II. Headquarters criticality inspectors will still issue their own inspection reports, will include Level IV violations and related Notice of Violations (NOVs) in the reports and will still have the capability to pursue escalated enforcement out of NRC Headquarters.

Conclusion

NRC Headquarters, through the NMSS office, conducts criticality safety inspections at fuel cycle facilities and provides specialized support to the consolidated fuel cycle inspection program at Region II. Headquarters criticality safety inspectors are specifically qualified in criticality safety and spend 100 percent of their time on criticality safety inspection and related activities.

References

1. U.S. Nuclear Regulatory Commission, “Fuel Cycle Facility Operational Safety and Safeguards Inspection Program,” Inspection Manual Chapter 2600, October 2004. (Available on request)

2. U.S. Nuclear Regulatory Commission, “Headquarters Nuclear Criticality Safety Program,” Inspection Procedure 88015, December 1996. (Available on request)

3. U.S. Nuclear Regulatory Commission, “Regional Nuclear Criticality Safety Program,” Inspection Procedure 88020, December 1996. (Available on request)

4. D. Morey and F. Gee, “Qualification and Certification of NRC Criticality Safety Inspectors,” presented at the 2001 Topical Meeting on Practical Implementation of Nuclear Criticality Safety, Proceedings, November 2001.

5. U.S. Nuclear Regulatory Commission, “Justification of Minimum Margin of Subcriticality for Safety,” Interim Staff Guidance 10, draft October 2004.

Interim Staff Guidance 03
Nuclear Criticality Safety Performance Requirements and Double-Contingency Principle

Dr. Christopher S. Tripp
Lawrence J. Berg and Dennis C. Morey
U.S. Nuclear Regulatory Commission
Mailstop T8-F43
Washington, D.C. 20555
cst@nrc.gov or ljb@nrc.gov or dcm@nrc.gov

Introduction

Interim Staff Guidance (ISG)-03 describes the relationships between 10 CFR Part 70, Subpart H nuclear criticality safety requirements, specifically the performance requirements of 10 CFR 70.61, and the double-contingency principle (DCP) of 10 CFR 70.64. ISG-03 is intended for use by U.S. Nuclear Regulatory Commission (NRC) technical reviewers of license amendments and applications related to criticality safety of fuel cycle facilities. The purpose of ISG-03 is not to re-define terms such as double-contingency, but to provide NRC staff with a basis for technical review. This paper summarizes the discussion and comparison of these requirements.

Background

10 CFR Part 70, Subpart H, contains three requirements to ensure nuclear criticality safety:

1. 10 CFR 70.64(a)(9) requires that the design of new facilities and processes provide for criticality control, including adherence to the DCP.
2. 10 CFR 70.61(b) requires that high-consequence events (which typically will include criticality accidents) be highly unlikely.
3. 10 CFR 70.61(d) requires that nuclear criticality accidents be limited, by assuring that, under normal and abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality, and also requires that the primary means of criticality protection be prevention.

The first criterion is a baseline design criterion for new facilities and processes, similar to general design criteria in 10 CFR Part 50. Under second criterion, a criticality accident would be considered a high-consequence event that must be made highly unlikely or mitigated. The purpose of the third requirement is to preclude a situation where nuclear criticality would be permitted, based on not exceeding dose thresholds.

10 CFR 70.61(d)

This rule requires that, under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety, with preventive controls and measures the primary means of protection against criticality. To meet this performance requirement, all normal and credible abnormal conditions must be identified, using a systematic methodology.

Attachment 2

Normal conditions are those specifically allowed for in facility design as part of normal modes of operation [i.e., conditions that may occur without the failure of any items relied on for safety (IROFS)]. Abnormal conditions are events not planned for as a regular occurrence in the facility or operation design.

Judgment must be employed in determining what constitutes a credible abnormal condition. What is considered not credible is contained in NUREG-1520, Section 3.4.3.2:
- "...an external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years."
- "...a process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive…"
- "...process deviations for which there is a convincing argument, given physical laws, that they are not possible, or are unquestionably extremely unlikely…"

The requirement that nuclear processes be subcritical is satisfied if the licensee or applicant demonstrates that the most reactive credible conditions are subcritical. There are several different ways to demonstrate subcriticality:

- If subcriticality is demonstrated using an appropriately validated calculational method, then k_{eff} (including calculational uncertainties) must be less than the approved Upper Subcritical Limit, as specified in the license.
- Subcritical margin may also be expressed in terms of system parameters rather than system k_{eff}.
- Subcriticality may be demonstrated on the basis of subcritical limits included in the license, NRC-endorsed American National Standards Institute (ANSI) standards, or other documents that have been approved or endorsed by NRC.
- Widely accepted industry handbooks of criticality data may also be used and will be evaluated on a case-by-case basis.

10 CFR 70.61(b)

This rule requires that the risk of each credible high-consequence event be limited and controls applied to the extent needed to reduce the likelihood of occurrence of the event so that the event is highly unlikely.

All Part 70 provisions must also be met regardless of whether the licensee attempts to meet the performance requirements starting from 10 CFR 70.61(b) or 10 CFR 70.61(d). To meet the regulations and prevent criticalities, an applicant/licensee may use one of the three following approaches:

- Demonstrate compliance with 10 CFR 70.61(d);
- Demonstrate compliance with 10 CFR 70.61(b), considering only preventive controls and including an approved margin of subcriticality;
- Separately demonstrate compliance with both 10 CFR 70.61(d) and 10 CFR 70.61(b).

10 CFR 70.64(a)(9)

The design of new facilities and processes must provide for criticality control, including adherence to the DCP. American National Standards Institute/American Nuclear Society-8.1 recognizes that adherence to the DCP can be one means, but is not necessarily the only means, of meeting the underlying subcriticality requirement.

Strict adherence to the DPC may not be practicable and should be accompanied by a convincing demonstration that adherence is actually not practicable. The DCP does not necessarily require two controls; it requires "...at least two…changes in process conditions" be needed before criticality is possible. Meeting this may necessitate one, two, or more than two controls, depending on the possible conditions that can lead to criticality. Passive engineered controls are generally preferable to active engineered controls, and engineered to administrative controls. Also, process design should rely on geometry control as opposed to control of other parameters whenever practicable, and on diverse means of control, to minimize the potential for common-mode failure.

10 CFR 70.61 and Double Contingency

A number of different approaches to double contingency have been used and some cases used in the past may not be sufficiently robust to satisfy the performance requirements of 10 CFR 70.61. To clarify when adherence to the DCP will establish a sufficient basis for meeting performance requirements, the following terms are defined:

Unlikely changes in process conditions should be expected to occur rarely, or not at all, during the lifetime of the facility.
Independent changes in process conditions are such that one contingency neither causes another contingency nor increases its likelihood of occurrence.
Concurrent does not mean that the two changes in process conditions must occur simultaneously, but that the effect of the first contingency persists until the second contingency occurs.
Changes in process conditions does not imply that reliance on two different parameters is mandatory to meet the DCP.

The following illustrates the conditions under which adherence to the DCP is sufficient to meet the performance requirement of 10 CFR 70.61:

- Controls are established on system parameters to preclude changes in process conditions, and these controls are designated as IROFS, in accordance with 10 CFR 70.61(e);
- The condition resulting from the failure of a leg of double-contingency has been shown to be subcritical with an acceptable margin;
- Controls are sufficiently reliable to ensure that each change in process conditions necessary for criticality is "unlikely."

Because the DCP is only one means of meeting the performance requirements, it is possible to meet the DCP without meeting the conditions above.

Examples

Some specific examples of control systems that meet 10 CFR 70.61(d) through use of the DCP are:

- A passive geometry control where no credible failure mode (e.g., bulging, corrosion, or leakage) exists, and that has been placed under configuration management.
- Two passive controls where there is a credible failure mode, and there are sufficient management measures to ensure that the controls continue to perform their safety functions (e.g., periodic surveillance to detect corrosion/bulging):
- One passive control under configuration management and one active engineered control whose reliability is ensured by periodic functional testing, maintenance, and an alarm to automatically indicate its failure.
- One engineered and one enhanced administrative control, in which the instrumentation and devices included in the administrative control are subject to periodic functional testing and maintenance, and the operator action is performed routinely or reinforced by periodic drills and training:
- One engineered control and one simple administrative control, in which the reliability of the administrative control is subject to a high degree of redundancy[5].
- Two administrative controls that are independent (e.g., performed by different
- individuals or verified by a supervisor), for which human factors have been considered in the design of the process, such that the operation is not prone to error, and there is sufficient margin to require multiple failures before the criticality control limit can be exceeded.
- Other considerations ensuring that there is no credible event leading to criticality.

Some specific examples of control systems that would not meet 10 CFR 70.61(d) through use of the DCP are:

- Double contingency consisting of two single operator actions without any supervisor verification or redundancy.
- A leg of double contingency consisting of an administrative control for which correct performance of the action cannot be readily confirmed or is subjective.
- A leg of double contingency consisting of complex administrative tasks composed of multiple steps that are susceptible to error.
- A leg of double contingency consisting of an administrative control with insufficient margin to ensure that the safety limit will not be exceeded.
- A leg of double contingency consisting of an engineered control in which there is no reasonable means to detect and correct the failure within a given time.
- A leg of double contingency consisting of a control in an environment where its safety function is degraded.
- A leg of double contingency consisting of a control where its behavior under adverse conditions is uncertain.
- A leg of double contingency consisting of undeclared design features or process conditions that are not precluded by being explicitly controlled.

---

[1] Use of two independent samples is generally not considered adequate for both legs of double contingency because of difficulty ensuring complete independence between samples.

Conclusion

To meet the regulations and prevent criticalities, an applicant/licensee may use one of the three approaches below in conjunction with other Part 70 requirements:

- Demonstrate compliance with 10 CFR 70.61(d);
- Demonstrate compliance with 10 CFR 70.61(b), considering only preventive controls and including an approved margin of subcriticality;
- Separately demonstrate compliance with both 10 CFR 70.61(d) and 10 CFR 70.61(b).

Adherence to the DCP will satisfy the performance requirement of 10 CFR 70.61(d) [and therefore also 10 CFR 70.61(b)], provided the following conditions are met:

- Controls are established on system parameters to preclude changes in process conditions, and these controls are designated as IROFS, in accordance with 10 CFR 70.61(e).
- The condition resulting from the failure of a leg of double contingency has been shown to be subcritical, with an acceptable margin.
- Controls are sufficiently reliable to ensure that each change in process conditions necessary for criticality is "unlikely."
- Management measures are established to ensure that controls are available and reliable to perform their safety function.

References

1. Code of Federal Regulations, "Domestic Licensing of Special Nuclear Material," Part 70, Title 10, "Energy."

2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520, March 2002.

3. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of a License Application for a Mixed Oxide (MOX) Fuel Facility," NUREG-1718, August 2000.

Justification of Minimum Margin of Subcriticality for Safety

Dr. Christopher S. Tripp and Dennis C. Morey
U.S. Nuclear Regulatory Commission
Mailstop T8-F43
Washington, D.C. 20555
cst@nrc.gov or dcm@ nrc.gov

Introduction

10 CFR 70.61(d) requires that licensees demonstrate that "...under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety." A subcritical limit includes allowances for bias and bias uncertainty as well as additional margin that is the minimum margin of subcriticality (MoS). This paper provides guidance for U.S. Nuclear Regulatory Commission (NRC) reviewers to use in determining what constitutes an acceptable MoS and how the chosen MoS can be adequately documented. It is intended to provide a basis for technical review for NRC reviewers of license amendments and applications related to criticality safety at fuel cycle facilities. This paper summarizes the discussion of adequate MoS.

Background

NRC observed increasing challenges to MoS at licensed facilities, including increased reliance on computer calculations of k_{eff}, reduced conservatism in explicit modeling; and increased interest in reducing MoS. Difficulties and inefficiencies were experienced in recent licensing actions that affected acceptability of MoS, including characterization of unknown uncertainties in calculations, lack of benchmarks, and adequacy of methodology. Therefore, NRC determined that there was a need for a consistent regulatory approach to approval of MoS. Industry representatives agreed on the need for guidance at the July 2004 Integrated Safety Analysis (ISA) workshop.

This paper provides a method for evaluating MoS that is systematic and consistently applied, is risk-informed; takes facility/process-specific considerations into account; and provides guidance on some acceptable ways to provide adequate assurance of subcriticality. The guidance presents five criteria that *may* be used to justify MoS (may use any or all) including: (1) benchmark similarity: (2) system sensitivity: (3) knowledge of neutron physics: (4) rigor of validation methodology;and (5) margin in system parameters. It also provides guidance on several possible ways to meet criteria.

A MoS is needed when subcritical limits are based on calculational methods. Because the bias for a system other than a critical experiment is not known with a high degree of confidence, the MoS allows for unknown errors that may affect a calculated value of k_{eff}. There are significant differences in the MoS that have been approved for various licensed fuel cycle facilities, as shown in Table 1.

Attachment 3

Table 1

Licensed FuelCycle Facility	k_{eff} **Normal**	k_{eff} **Upset**
A	$k_{eff} \leq 0.94$ for LEU $k_{eff} \leq 0.92$ for HEU	$k_{eff} \leq 0.97$ for LEU $k_{eff} \leq 0.95$ for HEU
B	$k_{eff} + 2\sigma + \text{bias} \leq 0.90$	$k_{eff} + 2\sigma + \text{bias} \leq 0.95$
C	$k_{eff} \leq 0.9634$ USL	$k_{eff} \leq 0.9634$ USL
D	$k_{eff} \leq 0.9605$ USL	$k_{eff} \leq 0.9605$ USL
E	$k_{eff} + 2\sigma + \text{bias} \leq 0.87$	$k_{eff} + 2\sigma + \text{bias} \leq 0.95$
F	$k_{eff} \leq k_{calc} - 2\sigma - \text{bias} - 0.05$	$k_{eff} \leq k_{calc} - 2\sigma - \text{bias} - 0.03$
G	$k_{eff} + 3\sigma + \text{bias} \leq 0.97$	$k_{eff} + 3\sigma + \text{bias} \leq 0.97$
H	$k_{eff} + 2\sigma + \text{bias} + \text{uncertainty} \leq 0.95$	$k_{eff} + 2\sigma + \text{bias} + \text{uncertainty} \leq 0.98$

Approved Subcritical Limits

Note: LEU - Low Enriched Uraniun; HEU - High Enriched Uranium; USL - Upper Subcritical Limit

Terminology

For ease of relating specific concepts, this paper defines some terms as follows:

Bias: The difference between the calculated and true values of k_{eff} for a fissile system or set of systems.

Bias Uncertainty: The calculated uncertainty in the bias as determined by a statistical method.
Margin of subcriticality (MoS): Margin in k_{eff} applied in addition to bias and bias uncertainty to ensure subcriticality (also known as subcritical, arbitrary, or administrative margin). This term is shorthand for "minimum margin of subcriticality."
Margin of safety: Margin in one or more system parameters that represents the difference between the value of the parameter at which it is controlled and the value at which the system becomes critical.[1]
Upper Subcritical Limit: The maximum allowable k_{eff} value for a system. Generally, the USL is defined by the equation USL = 1 - bias - bias uncertainty - MoS.[2]
Subcritical Limit: The value of a system parameter at which it is controlled to ensure criticality

[1] This represents an additional margin beyond the MoS.

[2] If the USL is defined as described above, then the MoS represents the difference between the average calculated k_{eff} (including uncertainties) and the USL, thus:

$$MoS = (1 - bias - bias\ uncertainty) - \text{USL}$$

safety, and at which k_{eff} does not exceed the USL (also known as safety limit).
<u>Operating Limit</u>: The value of a system parameter at which it is administratively controlled to ensure that the system will not exceed the subcritical limit.[3]

Benchmark Similarity

The bias of calculations is estimated based on critical benchmarks with similar geometric form, material composition, and neutronic behavior to the system being evaluated. Therefore, the degree of similarity between the benchmarks and the actual system is a key consideration for determining an appropriate MoS.

Several methods to determine similarity are discussed herein, including: parameter comparison; use of screening criteria;use of an analytical method such as Oak Ridge National Laboratory's TSUNAMI code; sensitivity studies; or physical arguments.

Evaluation of benchmarks should include:

- Reliability of critical experiments
- Independence of benchmarks
- Benchmark parameters cover needed range
- Statistically significant number of benchmarks
- Parameters affecting bias represented

System Sensitivity

The system's k_{eff} sensitivity i.e., the sensitivity of the system k_{eff} to changes in key parameters should be well understood, to have full confidence in the selected MoS.

Evaluating the system k_{eff} sensitivity should include:

- Sensitivity of k_{eff} to changes in underlying nuclear data.
- Sensitivity of k_{eff} to changes in geometric form and material composition.
- Magnitude of MoS compared to expected magnitude of changes in k_{eff} resulting from errors in the underlying system parameters.

Neutron Physics of the System

Fissile systems that are known to be subcritical with a high degree of confidence do not require as much MoS as systems where subcriticality is less certain.

Evaluation of the neutron physics of the system should include:

- If geometric form and material composition are rigid and unchanging.
- If geometric form and material composition are subject to strict quality assurance.
- Reasons other than criticality calculations such as handbooks, standards, and reactor fuel studies, to conclude system will be subcritical.

[3] Not all licensees have a separate subcritical and operating limit. Use of administrative operating limits is optional, because the subcritical limit should conservatively take parametric tolerances into account.

- If cross-sections are well-known in the energy range of interest.

Rigor of Validation

The validation methodology must also be suitable for the data analyzed. If either the data or the methodology is not adequate, a high degree of confidence in the results cannot be attained.

Evaluating the data and methodology should include:

- Methodology consistent with the distribution of data (e.g., normal).
- Sufficient benchmarks to determine behavior of bias across the entire area of applicability.
- Functional form of bias represents a good fit to the benchmark data.
- Identify any benchmark clusters for which the overall bias appears to be non-conservative.
- Apply additional margin to account for extrapolation or wide interpolation.
- Account for all apparent bias trends.
- Excessive discarding of benchmarks as statistical outliers.

Performing an adequate code validation is not alone sufficient demonstration that an appropriate MoS has been chosen.

Parameter Margin

Other types of margin that can provide additional assurance of subcriticality are frequently expressed in terms of system parameters rather than k_{eff}. Generally, the margin to criticality in system parameters (termed the *margin of safety*) is a better indication of the inherent safety of the system than margin in k_{eff}.

In addition to establishing subcritical limits on controlled system parameters, licensees frequently establish operating limits to ensure that subcritical limits are not exceeded. The difference between the subcritical limit and the operating limit of a system parameter is a type of margin that may be credited in justifying a lower MoS than might otherwise be acceptable.

The difference between the subcritical limit and the operating limit can be confused with the MoS because systems in which k_{eff} is highly sensitive to changes in process parameters may require both: (1) a large margin between subcritical and operating limits; and (2) a large MoS. This is because systems in which k_{eff} is highly sensitive to changes in process parameters are highly sensitive to normal process variations and potential errors. MoS must be shown to be consistently and dependably present.

Evaluating the margin in system parameters should include:

- How much margin in k_{eff} is present because of conservatism in the modeling practices?
- Will this margin be present for all normal and credible abnormal condition calculations?

Normal vs. Abnormal Conditions

Some licensees distinguish between normal and abnormal condition k_{eff} limits by having a higher k_{eff} limit for abnormal conditions, which is permissible but not required. There is a certain likelihood associated with the MoS, that processes calculated to be subcritical will, in fact, be critical. A somewhat higher likelihood is permissible for abnormal than for normal condition calculations because the abnormal condition should be at least unlikely to occur, in accordance with the DCP. Also, there is often additional conservatism present in the abnormal condition because uncontrolled parameters are analyzed at their worst-case credible conditions.

Statistical Arguments

The argument has been used that the MoS can be estimated based on comparing the results of two statistical methods. In the USLSTATS code issued with the SCALE code package, there are two methods for calculating the USL: (1) the Confidence Band with Administrative Margin Approach, which calculates USL-1; and (2) the Lower Tolerance Band Approach, which calculates USL-2. Justification that the MoS chosen in the Confidence Band Approach is adequate has been based on a comparison of USL-1 and USL-2.

These methods are two different statistical treatments of the data, and a comparison between them can only demonstrate whether the MoS is sufficient to bound statistical uncertainties included in the Lower Tolerance Band Approach but not included in the Confidence Band Approach. NRC does not consider this an acceptable justification for selection of the MoS.

Summary of Comments

Industry comments received can be grouped into the following areas; (1) the paper presents specific MoS values; (2) the paper imposes new regulatory requirements; (3) the paper emphasizes effective neutron multiplication (k_{eff}) as an indicator of safety; (4) the paper emphasizes the use of a specific code, TSUNAMI, to evaluate benchmark applicability; and (5) the paper seeks to replace or supplant existing American National Standards Institute/ American Nuclear Society (ANSI/ANS)-8 guidance on criticality safety.

The paper provides technical reviewers information in areas where regulations and other guidance, such as a Standard Review Plan is not specific. This guidance is not mandatory for licensees. It is also worth noting that ANSI/ANS-8 Standards do not provide any guidance on evaluation of subcritical margin. ANSI/ANS-8 Standards evolved under a paradigm of an expert-based program employing substantial conservatism through double contingency.

Conclusion

Determination of an adequate MoS is strongly dependent on the specific processes and conditions at the facility being licensed. For facility processes involving unusual materials or new process conditions, the validation should be reviewed to ensure that there are no anomalies associated with unique system characteristics. In any case, the MoS should not be reduced below a minimum of 0.02.

Reducing the MoS below 0.05 for low-enriched processes or 0.1 for high-enriched or plutonium processes requires substantial additional justification, which may include:

- A high degree of similarity between chosen benchmarks and anticipated normal and credible abnormal conditions.
- Demonstration that the system k_{eff} is highly insensitive to changes in underlying system parameters.
- Demonstration that the system is known to be subcritical with a high degree of confidence.
- Demonstration that the validation methodology is exceptionally rigorous and accounts for all potential sources of error.
- Demonstration that there is dependable and consistent conservatism in k_{eff} because of conservatism in modeling practices.
- Demonstration for abnormal conditions may include offsetting an increased likelihood of being critical by the unlikelihood of achieving the abnormal condition.

Other technical justification demonstrating that there is a high degree of confidence in the calculation of k_{eff} may be used.

References

1. U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,” NUREG-1520, March 2002.

2. U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of a License Application for a Mixed Oxide (MOX) Fuel Facility,” NUREG-1718, August 2000.

3. U.S. Nuclear Regulatory Commission, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” NUREG/CR-6698.

4. U.S. Nuclear Regulatory Commission, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” NUREG/CR-6361.

Qualification of NRC Nuclear Criticality Safety Technical Reviewers in the Division of Fuel Cycle Safety and Safeguards

Tamara D. Powell and Melanie A. Galloway
U.S. Nuclear Regulatory Commission
Mailstop T8-F42
Washington, D.C. 20555
tdp@nrc.gov or mam@nrc.gov

Abstract

To ensure public health and safety, the U.S. Nuclear Regulatory Commission (NRC) performs nuclear criticality safety reviews for licenses, amendments, and renewals associated with fuel cycle facilities licensed under Title 10 of the Code of Federal Regulations, Parts 70 and 76, using technical staff from the NRC Headquarters office. Employees assigned as nuclear criticality safety technical reviewers must complete a rigorous program involving self-study, formal training, and oral review before final NRC certification. The qualification process provides assurance that each individual has attained the skills, competencies, and knowledge needed to adequately perform his or her assigned tasks. The high standard associated with certification as an NRC nuclear criticality safety technical reviewer is commensurate with the NRC responsibility to review the safety of licensed facilities against inadvertent criticality.

Introduction

The mission of the U.S. Nuclear Regulatory Commission (NRC) is to regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. The Office of Nuclear Material Safety and Safeguards (NMSS) has overall responsibility for ensuring the public health and safety through licensing, inspection, and environmental reviews for the regulation of activities involving the use and handling of radioactive materials. Within NMSS, the Division of Fuel Cycle Safety and Safeguards (FCSS) develops, implements, and evaluates overall Agency safety policy for fuel cycle, special nuclear material, uranium recovery, and associated waste processing facilities licensed under the Atomic Energy Act of 1954, as amended, or certified in accordance with the Energy Policy Act of 1992.

Technical staff in FCSS' Technical Support Group (TSG) perform criticality safety reviews for licenses, amendments, and renewals associated with fuel cycle facilities licensed under Title 10 of the Code of Federal Regulations, 10 CFR Parts 70 and 76. Nuclear criticality safety technical reviewers (criticality safety reviewers) must understand the facilities, equipment, processes, and activities of the programs they license. The qualification process is intended to provide criticality safety reviewers with sufficient information to conduct license reviews that are technically correct and in accordance with NRC regulations, policies, and procedures. Within 2 years, newly appointed or assigned reviewers must complete a qualification program leading to certification that includes formal training and an oral review board.

Attachment 4

Qualification Requirements

The criticality safety reviewer qualification plan is designed to assure that the employee receives the necessary training required to effectively provide criticality safety support in FCSS, and to ensure the transmission of institutional knowledge of criticality safety regulation to new staff. On appointment or assignment as an FCSS criticality safety reviewer, an NRC employee is required to prepare a criticality safety reviewer qualification journal to guide and document his or her qualification process. The qualification journal also provides assurance that the employee has attained the minimum knowledge and skills necessary to function as a criticality safety reviewer for fuel cycle facilities. The journal implements NRC Inspection Manual Chapter 1246 (Reference 1), Appendix A, Section VII, and the FCSS Division Procedure for Qualification Programs (Reference 2). Initial qualification is achieved through self-study, formal classroom courses and on-the-job training.

Specific skills required are:

- Understanding of the regulatory requirements for FCSS-regulated fuel cycle facilities.
- Knowledge of the fuel cycle facilities and processes.
- Knowledge of criticality safety principles and skills to perform criticality safety reviews of FCSS-regulated fuel cycle facilities.

These specific skills are incorporated into nine training areas that must be completed:

- NRC Orientation
- Regulatory Framework Orientation
- Office Instructions/Procedures
- Regulatory Guidance
- Industry Codes and Standards
- NRC Management Directives
- Review of Significant Fuel Cycle Events
- Directed Review of Selected Licensing Casework
- Formal Training

The prospective reviewer will meet with the supervisor and determine the individual qualification methods and training requirements to be met in each area, based on the employee's background, experience level, or prior NRC certification. Technical staff with significant education, training, and at least 5 years of applicable experience may be grandfathered, and thus documented as fully qualified. Qualification requirements can only be waived on the basis of well-supported, documented, justification and approval by the Chief, TSG.

To help new staff complete their qualification journals, each incoming nuclear criticality safety staff member who is not a qualified reviewer will be assigned a senior staff member as a mentor for various training areas. Each training area has specific self-study activities that are to be completed, and the mentor for each area is responsible for "signing off" that the individual has gained general knowledge and understanding of each activity. To further focus the qualification of the employee, a reference facility is assigned to complete a directed review of selected licensing casework.

The qualification process will be evaluated by supervisors and candidates for qualification will be monitored to assess progress toward completion of the qualification. Regular progress reviews and "sign-offs" should be scheduled. The target for completion of qualification is 12 months for most personnel, particularly those with extensive NRC or applicable industry experience. Technical staff who are recent college graduates may need additional time for full qualification.

Formal Training

All criticality safety reviewers complete formal training that includes, but is not limited to, the following courses. Some of these courses are 3-to 5-day classroom courses with written final examinations:

- NRC: What It Is and What It Does
- The Regulatory Process
- Root-Cause and Incident Investigation
- Site Access Training
- Fuel Cycle Processes
- Introduction to Risk Assessment in NMSS
- NMSS Environmental Review Overview
- Nuclear Criticality Safety Self-Study Course

NRC conducts all of the above training at the Headquarters complex in Rockville but some courses may not always be available to the particular employee during a qualification period, because of scheduling problems. An individual qualification plan may include additional formal training requirements, based on an employee's education, training, and experience, relative to the skill set necessary for successful performance as a criticality safety reviewer. For example, criticality safety reviewers normally take a SCALE course offered by Oak Ridge National Laboratory, but for a more experienced employee, that may not be necessary.

Oral Qualification Board

On completion of all qualification requirements contained in the approved qualification plan, the training area mentors will recommend to the supervisor that the employee is ready for certification. The employee will demonstrate the minimum knowledge and skills by successfully completing an oral review held by a qualification board consisting of criticality staff and management. First, the supervisor will interview the employee and review the employee's qualification journal, which will, by then, contain the completed qualification plan, with evidence that all requirements have been fulfilled. When the supervisor is satisfied that the employee is ready for oral review, the supervisor will convene and chair an oral board consisting of at least three, but no more than five members. Senior FCSS managers are also invited and may choose to participate. Reasonable questions concerning the NRC licensing program, enforcement, allegations, criticality safety, licensed facilities, and risk may be asked. The questions asked should allow and encourage the candidate to answer in such a way as to demonstrate depth of knowledge and understanding (i.e., not just a yes or no). An outline of typical questions is provided to the employee and panel members before the review, however, the review is not limited to these questions. At the conclusion of the review, the panel meets in private and discusses the employee's responses. If all panel members are satisfied that the employee is fully capable of performing independently as a certified NRC reviewer, the panel will recommend certification. If the panel identifies weaknesses, the employee must develop a training plan to satisfy the deficiencies before certification. Occasionally, subsequent to additional training, a second panel is convened, and an additional review is conducted to complete certification.

Summary

Employees appointed or assigned as criticality safety reviewers must complete a rigorous program involving self-study, formal training, and oral review before final NRC certification. The high standard associated with certification as an NRC criticality safety reviewer is commensurate with the NRC responsibility to ensure the safety of licensed facilities against inadvertent criticality.

References

1. NRC Inspection Manual Chapter 1246, *Formal Qualification Programs in the Nuclear Material Safety and Safeguards Program Area*, U.S. Nuclear Regulatory Commission (June 1996).

2. Memorandum to FCSS staff from Robert C. Pierson, Director of Fuel Cycle Safety and Safeguards, *Division Procedure for Qualification Programs*, July 17, 2002.