

## 5. NUCLEAR CRITICALITY SAFETY

### 5.1 Purpose of Review

The primary purpose of the review is to determine, with reasonable assurance, that the applicant has designed a facility that will provide adequate protection against criticality hazards related to the storage, handling, and processing of licensed materials, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 70, "Domestic Licensing of Special Nuclear Material." The facility design must adequately protect the health and safety of workers and the public during normal operations and credible accident conditions (§70.23 (a)(3)) from the accidental criticality risks in the facility. It should also protect against facility conditions that could affect the safety of licensed materials and thus present an increased risk of criticality or radiation release.

Another purpose of this review is to determine, with reasonable assurance, whether the licensee's or applicant's nuclear criticality safety (NCS) program, as described in the license application and integrated safety analysis (ISA) summary, is adequate to meet the regulatory requirements in 10 CFR Part 70 and will support safe possession and use of nuclear material at the facility. The review should examine the parts of the license application and ISA Summary that describe the NCS program. The review should ensure that either the license application for a new facility or the license amendment to an existing facility meets the regulatory requirements in 10 CFR Part 70. The review should also ensure that, if applicable, the criteria specified in 10 CFR Part 70 for meeting 10 CFR 70.61, "Performance Requirements," are satisfied and that the contents of the ISA Summary required by 10 CFR 70.65, "Additional Content of Applications," meet the regulatory requirements for the NCS-related areas of the ISA Summary.

### 5.2 Responsibility for Review

Primary: Nuclear Process Engineer (NCS Reviewer)

Supporting: Licensing Project Manager  
Fuel Cycle Inspection staff  
Primary Reviewers for Chapters 1, 3, 8, and 11 of this Standard Review Plan (SRP)

### 5.3 Areas of Review

#### 5.3.1 License Application

The staff should review the license application and ISA Summary, if applicable, to determine whether the application meets the 10 CFR Part 70 requirements for the NCS-related areas. The regulatory requirements for the license application review should comply with the general and additional content of an application, as required by 10 CFR 70.22, "Content of Applications," and 10 CFR 70.65, respectively. The NCS reviewer should evaluate the application or amendment to determine whether the applicant has met the requirements of 10 CFR 70.23, "Requirements for the Approval of Applications," to ensure that the applicant has proposed equipment, facilities, and procedures to protect health and minimize danger to life or property, and 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," as applicable, to ensure that the design provides for criticality control, including adherence to the double contingency principle.

The NCS reviewer should review the ISA-related requirements in 10 CFR 70.62, "Safety Program and Integrated Safety Analysis," and 10 CFR 70.65, including the requirement for criticality monitoring and alarms. The regulation established in 10 CFR 70.62(a) requires an applicant to develop, implement, and maintain a safety program that will reasonably protect the health and safety of the public and the environment from criticality hazards associated with processing, handling, and storing licensed materials during normal operations, anticipated operational occurrences, and credible accidents. The NCS program addresses process-specific risks, as well as other criticality-related areas, such as performing calculations, making criticality evaluations, or demonstrating the subcritical margin. In addition, the NCS review should verify compliance with 10 CFR 70.61 for meeting the performance requirements and ensuring that, under normal and credible abnormal conditions, all nuclear processes are subcritical.

### **5.3.2 Nuclear Criticality Safety Program**

The NCS reviewer should ensure that the applicant has committed to and implemented effective management of the NCS program in the license application and has provided enough qualified resources for an effective NCS program. The primary objective of an effective NCS program is to prevent an inadvertent nuclear criticality. Although 10 CFR Part 70 does not require a nuclear safety program directly, an applicant should provide commitments pertaining to NCS in the following areas:

- establishing and maintaining NCS safety practices and procedures
- establishing and maintaining NCS safety limits and procedures for determining operating limits
- conducting NCS evaluations to ensure that, under normal and credible abnormal conditions, all nuclear processes remain subcritical, with an approved margin of subcriticality for safety
- providing training in procedures for criticality-related possession and use of nuclear material and for response to an inadvertent nuclear criticality
- complying with NCS baseline design criteria (BDC) requirements in 10 CFR 70.64(a), if the application is for a new facility or for a new process at an existing facility
- complying with the NCS ISA Summary change process requirements of 10 CFR 70.72, "Facility Changes and Change Process"
- protecting against the occurrence of any identified accident sequence in the ISA Summary that could lead to an inadvertent nuclear criticality
- complying with the NCS performance requirements of 10 CFR 70.61

The reviewer should determine whether the applicant has identified responsibilities and authorities of individuals to implement and administer the NCS program. The reviewer should evaluate the following matters related to the applicant's organization and administration:

- the general organization and administration methods used by the applicant (see SRP Chapter 2)
- the areas of review listed in SRP Section 2.3 as they relate to NCS, including the experience, educational requirements, responsibilities, and authorities of NCS management and staff

### **5.3.3 Integrated Safety Analysis Summary**

The reviewer should determine whether the applicant has committed to the facility safety program, including the process safety information, ISA, and management measures in 10 CFR 70.62, and whether the commitments demonstrate the applicant's ability to implement and maintain the NCS controls. The NCS reviewer should evaluate the following areas in the ISA Summary:

- Process descriptions are narrative descriptions of the site, facility, and processes with respect to criticality safety for normal operations. The criticality process description can include flow diagrams, major process steps, and major pieces of equipment, with emphasis on the criticality safety controls. The ISA Summary should include a reasonably simple description of each process (unit operations).
- Criticality accident sequences should include accident sequences involving licensed materials and an interpretation of the sequence of events, as described in the ISA Summary. It is assumed that all criticality accident sequences have high consequences; therefore, the applicant should include every credible event that could result in an uncontrolled criticality.
- Criticality accident consequences should be identified in the ISA Summary, including the assumption that all criticality accidents are high-consequence events and that the bases and methods the applicant used are based on using preventative controls.
- Criticality process items relied on for safety (IROFS) should include a list of items relied on for criticality safety and a description of their safety function, as described in the ISA Summary. The applicant should use enough IROFS to demonstrate that, under normal and abnormal credible conditions, all nuclear processes are subcritical.
- Criticality IROFS management measures should include management measures to ensure the reliability and availability of the IROFS described in the ISA Summary.

#### Review Interfaces

The criticality safety reviewer should examine information in the following other areas to ensure that it is consistent with Chapter 5 of the application:

- facility and process description applied to criticality safety, as described in Chapter 1 of this SRP

- administration and organization of the criticality safety functions, as described in Chapter 2 of this SRP
- safety program, ISA commitments, and ISA documentation applied to criticality safety under SRP Chapter 3
- emergency plan applied to criticality safety under SRP Chapter 8
- configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, record management, and other quality assurance elements, as described in SRP Chapter 11

## **5.4 Acceptance Criteria**

The applicant should provide NCS commitments and describe how the commitments will be met. Commitments and descriptions are expected when the acceptance criteria are relevant to the possession and use of nuclear materials and the materials to be licensed.

### **5.4.1 Regulatory Requirements**

Acceptance criteria are based on meeting the relevant requirements of the following regulations:

- The general and additional contents of an application for criticality safety are in 10 CFR 70.22 and 10 CFR 70.65, respectively. General information that must be included in the license application appears in 10 CFR 70.22. Information that must be included in the ISA Summary, including the requirements for criticality monitoring and alarms, appears in 10 CFR 70.65.
- The requirements for the approval of the application are in 10 CFR 70.23.
- Requirements for new facilities or new processes at existing facilities that require a license amendment under 10 CFR 70.72 appear in 10 CFR 70.64, including the requirement to adhere to the double contingency principle.
- Requirements to maintain and establish a safety program are in 10 CFR 70.62.
- The criticality safety review should be conducted to provide reasonable assurance of compliance with the performance requirements in 10 CFR 70.61.

### **5.4.2 Regulatory Guidance**

The following additional guidance may be used to supplement the review of the NCS program:

- NUREG-1513, "Integrated Safety Analysis Guidance Document," May 2001
- NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," March 1998

### **5.4.3 Regulatory Acceptance Criteria**

The reviewer should find the applicant's criticality safety program information acceptable if it provides reasonable assurance that the acceptance criteria discussed below are adequately addressed and satisfied. The applicant may elect to incorporate some or all of the requested criticality safety information in the facility and process description (SRP Section 1.1) or in the ISA Summary, rather than in this section. Either approach is acceptable, as long as the information is adequately cross-referenced.

#### 5.4.3.1 License Application

The reviewer should consider that the applicant's commitment to the criticality accident alarm system (CAAS) requirements in 10 CFR 70.24, "Criticality Accident Requirements," is acceptable if the applicant or licensee has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant describes a facility CAAS that meets the requirements of 10 CFR 70.24.
- The applicant commits to American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.3-1997, "Criticality Accident Alarm System," as modified by Regulatory Guide 3.71, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," issued October 2005 by the U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 3.71 lists the following exceptions to the standard:
  - At or above the 10 CFR 70.24 mass limits, the applicant should require CAAS coverage in each area where special nuclear material (SNM) is handled, stored, or used.
  - A requirement of 10 CFR 70.24 is that two detectors cover each area needing CAAS coverage.
  - A requirement of 10 CFR 70.24 is that a CAAS be capable of detecting a nuclear criticality that produces an absorbed dose in soft tissue of 20 rads (0.2 Gy) of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within 1 minute.
- The applicant commits to having a CAAS that is appropriate for the facility for the type of radiation detected, intervening shielding, and the magnitude of the minimum accident of concern.
- The applicant commits to having a CAAS that is designed to remain operational during credible events, such as a seismic shock equivalent to the site-specific, design-basis earthquake or the equivalent value specified by the Uniform Building Code.
- The applicant commits to having a CAAS that is designed to remain operational during credible events, such as a fire, an explosion, a corrosive atmosphere, or other credible conditions.
- The applicant commits to having a criticality accident alarm that is clearly audible in areas that must be evacuated or provides alternative notification methods that are documented to be effective in notifying personnel that evacuation is necessary.

- The applicant commits to rendering operations safe, by shutdown and quarantine if necessary, in any area where CAAS coverage has been lost and not restored within a specified number of hours. The number of hours should be determined on a process-by-process basis, because shutting down certain processes, even to make them safe, may carry a larger risk than being without a CAAS for a short time. The applicant should commit to compensatory measures (e.g., limiting access, halting SNM movement) when the CAAS system is not functional.
- The applicant commits to the following emergency management provisions (see SRP Chapter 8):
  - The applicant commits to the requirements in ANSI/ANS-8.23-1997, “Nuclear Criticality Accident Emergency Planning and Response,” as they relate to NCS.
  - The applicant either has an emergency plan or satisfies the alternative requirements in 10 CFR 70.22(h)(1)(i).
  - The applicant commits to the provision of fixed and personnel accident dosimeters in areas that require a CAAS. These dosimeters should be readily available to personnel responding to an emergency, and there should be a method for prompt onsite dosimeter readouts.
  - The applicant commits to providing emergency power for the CAAS or provides justification for the use of continuous monitoring with portable instruments.

Using the reasonable assurance of safety standard as described in the introduction to this SRP, the reviewer should determine whether the applicant has met the requirements of 10 CFR 70.61. The introduction, as well as Section 3.1 of the SRP, describing the review of the ISA and ISA Summary, includes guidance on the level of detail needed to achieve this standard. The reviewer should consider the applicant’s commitments to demonstrating that all nuclear processes will be subcritical under normal and credible abnormal conditions to be acceptable if the application includes the following acceptance criteria or identifies and justifies an alternative:

- As one approach, the applicant commits to the following national standards, as they relate to these requirements: ANSI/ANS-8.7-1975, “Guide for Nuclear Criticality Safety in the Storage of Fissile Materials”; ANSI/ANS-8.9-1987, “Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials”; ANSI/ANS-8.10-1983, “Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement”; ANSI/ANS-8.12-1987, “Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors”; ANSI/ANS-8.15-1981, “Nuclear Criticality Control of Special Actinide Elements”; and ANSI/ANS-8.17-1984, “Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.” Alternatively, the applicant commits to base the safety limits on validated calculational methods.
- The applicant describes a program that ensures compliance with the double-contingency principle, where practicable (see Appendix 5-A for detailed guidance regarding the double-contingency principle). Processes in which there are no credible accident sequences that lead to criticality meet the double-contingency principle by definition.

This principle, as given in ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," states that at least two changes in process conditions must occur before criticality is possible. If there are no process changes leading to criticality, then the principle is satisfied. Each process that has accident sequences leading to criticality should have sufficient controls in place to ensure double-contingency protection. This may be provided by either (1) control of two independent process parameters, or (2) control of a single process parameter, such that at least two independent failures would have to occur before criticality is possible. The first method is preferable, because of the inherent difficulty in preventing common-mode failure when controlling only one parameter.

The reviewer should note that the double-contingency principle requires two unlikely, independent, and concurrent changes in process conditions before criticality is possible. This does not necessarily mean that two controls are required. In some cases, it may be appropriate to credit the natural and credible course of events (e.g., unsintered powder cannot exceed a maximum density, there is no means of enriching beyond 5 weight percent uranium-235, the low historical likelihood of flooding) without establishing explicit controls. The reviewer should exercise judgment in determining whether the applicant has established sufficient means to ensure that occurrence of the contingencies is "unlikely." In addition, the term "concurrent" means that the effect of the first process change persists until a second change occurs. It does not mean that the two events must occur simultaneously. The possibility of an inadvertent criticality can be markedly reduced if failures of NCS controls are rapidly detected and the process rendered safe. If not, processes can remain vulnerable to a second failure for extended periods of time.

In a very few processes, double-contingency protection is not practicable. In those rare instances, the applicant should provide adequate justification for why such cases are acceptable. The justification should demonstrate that there is sufficiently low risk that an exception is warranted. The reviewer should note that the double-contingency principle, as stated in ANSI/ANS-8.1-1998, is a recommendation ("Process designs should, in general, incorporate sufficient factors of safety..."). The more important requirement is the one incorporated in 10 CFR 70.61(d) ("it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions"). Thus, as long as the applicant can meet the underlying requirement to be subcritical under normal and credible abnormal conditions through other means, an exception may be justifiable.

- The applicant meets the acceptance criteria in SRP Chapter 3 as they relate to the subcriticality of operations and the margin of subcriticality for safety.

The ISA and supporting ISA documentation (such as piping and instrumentation diagrams, criticality safety analyses, dose calculations, process safety information, and ISA worksheets) would be maintained on site at an existing facility. For an applicant seeking a license before completion of a facility, a full level of detail concerning hardware, procedures, and programs usually would not exist. However, at the time of the preoperational readiness review for a new facility, or a new process at an existing facility, such details must be available to demonstrate compliance with the safety program requirements of Subpart H, "Additional Requirements for Certain Licensees Authorized To Possess a Critical Mass of Special Nuclear Material," to 10 CFR Part 70.

The reviewer should consider the applicant's commitment to the BDC requirements in 10 CFR 70.64(a) acceptable if the applicant has committed to the double-contingency principle

in determining NCS controls and IROFS in the design of new facilities or new processes at existing facilities that require a license amendment under 10 CFR 70.72. The applicant could also identify and justify an alternative in the application.

The applicant must meet the performance requirements in 10 CFR 70.61(b) and (c), as well as in 10 CFR 70.61(d), which include the requirement to limit the risk of inadvertent nuclear criticality by ensuring that all nuclear processes remain subcritical. The applicant's evaluation of NCS accident sequences may be performed in a manner consistent with the applicant's evaluation of non-NCS accident sequences used to meet 10 CFR 70.61(b) and (c); however, 10 CFR 70.61(d) requires the applicant to use prevention methods as the primary means to meet the performance requirements of 10 CFR 70.61(b) and (c). In addition, for new facilities and new processes at existing facilities, 10 CFR 70.64(a)(9) requires compliance with the double-contingency principle. This requires considerations in addition to those necessary to meet 10 CFR 70.61 for the noncriticality hazards.

The reviewer should consider the applicant's commitment to the requirements in 10 CFR 70.65(b) acceptable if it has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant meets the acceptance criteria in SRP Section 3, as they relate to the identification of NCS accident sequences, consequences of NCS accident sequences, likelihoods of NCS accident sequences, and descriptions of IROFS for NCS accident sequences.
- The applicant should consider the upsets listed in Appendix A to ANSI/ANS-8.1-1983 in identifying NCS accident sequences.

The applicant may use the guidance in ANSI/ANS-8.10-1983, as modified by Regulatory Guide 3.71, in determining the consequences of criticality accident sequences. In general, such events should be considered "high-consequence" events unless controls are in place to provide shielding or other isolation between the source of radiation and facility personnel. Consideration of events as other than those of high consequence should be justified in the ISA Summary. The reviewer should note that the requirements of 10 CFR 70.61(d) are still applicable (i.e., criticality is to be prevented).

The application should also address the BDC for new facilities or new processes at existing facilities that require a license amendment under 10 CFR 70.72. These baseline criteria must (§70.64 (a)(9)) be applied to the design of new processes but do not require retrofits to existing facilities or existing processes; however, all facilities and processes must comply with the performance requirements in 10 CFR 70.61. Section 2.4 of NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities," dated September 3, 1997, contains a list of items that should be considered in an adequate facility design. For new facilities and new processes in existing facilities, the design should provide for adequate protection against criticality accidents.

The application should do the following:

- The applicant describes how it performed the ISA for the new process and how the ISA satisfies the principles of the BDC and the performance requirements in 10 CFR 70.61. The applicant also explains how it applies defense in depth to higher risk accident sequences. Acceptable defense-in-depth principles for the criticality safety design are

those that support a hierarchy of controls: prevention, mitigation, and operator intervention, in order of preference.

- The applicant describes proposed facility-specific or process-specific relaxations or additions to BDC, along with justifications for relaxations.
- The ISA Summary describes how the criticality safety BDC were applied in establishing the design principles, features, and control systems of the new process.

#### 5.4.3.2 Nuclear Criticality Safety Program

The reviewer should consider the applicant's management of the NCS program acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant describes and commits to implementing and maintaining an NCS program to meet the regulatory requirements of 10 CFR Part 70.
- The application states the NCS program objectives, which should include those listed in this chapter.
- The application outlines an NCS program structure that is consistent with current industry practices (e.g., ANSI/ANS-8.1-1998 and ANSI/ANS-8.19-1996, "Administrative Practices for Nuclear Criticality Safety") and current industry practice that defines the responsibilities and authorities of key program personnel.
- The applicant commits to using the NCS program to establish and maintain NCS safety limits and NCS operating limits for fissile material use and possession and commits to maintaining management measures to ensure the availability and reliability of the controls.
- The applicant commits to preparing NCS postings, to NCS training, and to NCS procedure training.
- The applicant commits to evaluating modifications to the facility or safety program for their impact on criticality safety.

The organization and administration part of the application (see SRP Chapter 2) contains information related to NCS organization and administration acceptance criteria. The reviewer should find the applicant's NCS organization and administration acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant meets the acceptance criteria in SRP Section 2.4 as they relate to NCS, including organizational positions, functional responsibilities, experience, and qualifications of personnel responsible for NCS.
- The applicant meets the intent of ANSI/ANS-8.1 and ANSI/ANS-8.19 (see Regulatory Guide 3.71), as they relate to organization and administration.

- The NCS organization should be independent of operations to the extent practical.
- The applicant commits to providing distinctive NCS postings in areas, operations, work stations, and storage locations relying on administrative controls for NCS.
- The applicant commits to requiring its personnel to perform activities in accordance with written, approved procedures when the activity may affect NCS. Unless a specific procedure deals with the situation, personnel shall take no action until the NCS staff has evaluated the situation and provided recovery procedures.
- The applicant commits to requiring its personnel to report defective NCS conditions to the NCS program management.
- The applicant describes organizational positions, experience of personnel, qualifications of personnel, and functional responsibilities.
- The applicant commits to designating an NCS program director who will be responsible for implementing the NCS program.

Information related to acceptance criteria for the NCS safety program may appear in the ISA or management measures part of the application. The applicant's NCS management measures (required by 10 CFR 70.62) should be considered acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- training (see SRP Chapter 11)
  - The applicant meets the intent of ANSI/ANS-8.19 and ANSI/ANS-8.20, "Nuclear Criticality Safety Training," as they relate to training.
  - The applicant commits to training all personnel to recognize the CAAS signal and to evacuate promptly to a safe area.
  - The applicant commits to providing instruction and training regarding the policy in the SRP guidance for NCS organization procedures (see SRP Chapter 11).
- procedures (see SRP Chapter 11)
  - The applicant commits to ANSI/ANS-8.19-1996 as it relates to procedures.
- audits and assessments (see SRP Chapter 11)
  - The applicant commits to ANSI/ANS-8.19-1996 as it relates to audits and assessments.
  - The applicant commits to conducting and documenting walkthroughs (i.e., observation of operations to ensure compliance with criticality limits) of all operating SNM process areas, such that all operating SNM process areas will be reviewed at some specified frequency. The reviewer should consider the complexity of the process, the degree of process monitoring, and the degree of reliance on administrative controls in assessing the acceptability of the specified

frequency. Identified weaknesses should be referred to those responsible for facility corrective actions and should be promptly and effectively resolved. A graded approach may be used to justify an alternative NCS walkthrough schedule.

- The applicant commits to conducting and documenting periodic NCS audits (such that all NCS aspects of management measures (see SRP Chapter 11) will be audited at least every 2 years). A graded approach may be used to justify an alternative NCS audit schedule.

The reviewer should consider the applicant's NCS technical practices acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- NCS evaluations will be performed using industry-accepted and peer-reviewed methods.
- NCS limits on controlled parameters will be established to ensure that all nuclear processes are subcritical, including an adequate margin of subcriticality for safety.
- Methods used to develop NCS limits will be validated to ensure that they are used within acceptable ranges and that the applicant used both appropriate assumptions and acceptable computer codes.
- The applicant commits to demonstrating (1) the adequacy of the margin of subcriticality for safety by ensuring that the margin is large compared to the uncertainty in the calculated value of  $K_{eff}$  (effective multiplication factor), (2) that the calculation of  $K_{eff}$  is based on a set of variables within the method's validated area of applicability, and (3) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.

The margin of subcriticality for safety is an allowance for any unknown uncertainties that have not been accounted for in validation and a measure of the degree of confidence that systems calculated to be subcritical are actually subcritical. The margin is used to define an upper subcritical limit, as follows:

$$k\text{-subcritical} = 1.0 - \text{bias} - \text{bias uncertainty} - \text{margin of subcriticality for safety}$$

The reviewer must use judgment in assessing whether the margin of subcriticality for safety is sufficient to provide reasonable assurance of subcriticality (in accordance with 10 CFR 70.61(d)). The reviewer should consider the following factors, as applicable, in making this judgment, as well as any other available information that provides the needed confidence:

- conservatism in the calculations, beyond that needed to accommodate uncertainties in the modeled parameters (e.g., geometric tolerances)
- confidence in subcriticality generated by the applicant's validation process, including the following:

- similarity between the benchmark experiments and calculations to be performed
  - sufficiency of the benchmark data (both quality and quantity)
  - rigor of the validation methodology (e.g., trending, statistical testing)
  - conservatism in the statistical parameters (e.g., 95/95 lower tolerance limit)
- sensitivity of the system to changes in modeled parameters (and therefore to errors)
  - corroborating evidence of subcriticality from other sources (e.g., knowledge of neutron physics for well-characterized systems, such as finished fuel)
  - risk considerations, including the likelihood of actually attaining an abnormal condition

In general, a margin of subcriticality for safety of 0.05 has been found acceptable for typical nuclear processes involving low-enriched uranium, without a detailed justification. The use of increasingly smaller margins should require increasingly more rigorous justification, and the reviewer should evaluate other physical systems on a case-by-case basis.

- The applicant includes a summary description of a documented, reviewed, and approved validation report (by NCS function and management) for each methodology that will be used to perform an NCS analysis (e.g., experimental data, reference books, hand calculations, deterministic computer codes, probabilistic computer codes). The summary description of a reference manual or validation report should include the following:
  - a summary of the theory of the methodology that is sufficiently detailed and clear to be understood, including the method used to select the benchmark experiments, determine the bias and uncertainty in the bias, and determine the upper subcritical limit
  - a summary of the physical systems and area(s) of applicability covered by the validation report, noting that it is not necessary to include the full range of numerical parameters that defines the area of applicability. since a general description (e.g., low-enriched homogeneous uranium fluoride solutions, low-enriched fuel pellets, and rods containing gadolinia) is sufficient
  - a description of the methods used to justify applying the methodology outside the area or areas of applicability
  - a summary of the plant-specific benchmark experiments used to validate the methodology, noting that it is not necessary to include all benchmark experiments used, since a brief description of the individual benchmark data sets will suffice

- a description of the margin of subcriticality for safety and its justification
- a description of the controlled software and hardware used
- a description of the verification process used, including verification upon changes to the calculational system and upon some specified period
- The applicant's validation methodology, as described above, should be found acceptable if either (1) the applicant commits to following ANSI/ANS-8.24-2006, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations," as endorsed by Regulatory Guide 3.71, or (2) the methodology follows current industry practices in terms of selecting the benchmark experiments, assessing their applicability, determining the area(s) of applicability, extending the area(s) of applicability beyond the range of benchmark data, and statistically analyzing the data. This requires that the NCS reviewer remain aware of current practices in the area of criticality code validation.

The reviewer may examine the applicant's validation report to ensure that the methodology is sufficiently rigorous and is being applied in a manner consistent with its assumptions (e.g., normal distribution of benchmarks).

- The applicant commits to incorporating each validation report into the facility configuration management program.
- The applicant commits to performing NCS analyses in accordance with documented and approved procedures, which incorporate the following principles:
  - NCS safety limits and NCS operating limits will be established, assuming optimum credible conditions (i.e., the most reactive conditions physically possible or limited by written commitments to regulatory agencies) unless specified controls are implemented to control the limit to a certain range of values.
  - NCS safety limits, NCS operating limits, and limits on NCS-controlled parameters will be derived from the NCS analyses.
  - NCS operating limits will be derived from NCS safety limits by considering the uncertainty and variability in operating parameters to ensure that processes will remain subcritical under both normal and credible abnormal conditions.
  - The margin of subcriticality for safety for a process should be large, relative to the uncertainty in the calculated value of  $K_{eff}$ .

Controlled parameters available for NCS control include the following: mass, geometry, density, enrichment, reflection, moderation, concentration, interaction, neutron absorption, and volume. The reviewer should consider the applicant's commitment to NCS technical practices acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant's use of a single NCS control to maintain the values of two or more controlled parameters constitutes only one component necessary to meet double-contingency protection.
- The applicant commits to the preferred use of passive-engineered controls to ensure NCS. In general, the applicant should commit to the following order of preference for NCS controls: (1) passive engineered, (2) active engineered, (3) enhanced administrative, and (4) simple administrative. When using other than a passive-engineered control, the applicant should justify the choice of the type and manner.
- When they are relevant, the applicant should consider heterogeneous effects. Heterogeneous effects are particularly relevant for low-enriched uranium processes, where, all other parameters being equal, heterogeneous systems are more reactive than homogeneous systems.

The use of mass as a controlled parameter should be considered acceptable in the following circumstances:

- When mass limits are derived for a material that is assumed to have a given weight percent of SNM, determinations of mass are based on either (1) weighing the material and assuming that the entire mass is SNM or (2) conducting physical measurements to establish the actual weight percent of SNM in the material.
- When fixed geometric devices are used to limit the mass of SNM, a conservative process density is assumed in calculating the resulting mass.
- When the mass is measured, instrumentation subject to facility management measures is used.

The use of geometry as a controlled parameter should be considered acceptable if the following applies:

- Before beginning operations, all dimensions and nuclear properties that use geometry control are verified. The facility configuration management program should be used to maintain these dimensions and nuclear properties.

The use of density as a controlled parameter should be considered acceptable in the following circumstances:

- When process variables can affect the density, the ISA Summary shows the process variables to be controlled by IROFS.
- Density is measured by the use of instrumentation subject to facility management measures.

The use of enrichment as a controlled parameter should be considered acceptable if the following apply:

- Either a method of segregating enrichments is used to ensure that differing enrichments will not be interchanged or the most limiting enrichment is applied to all material.
- Measurements of enrichment are obtained by using instrumentation subject to facility management measures.

The use of reflection as a controlled parameter should be considered acceptable in the following circumstances:

- In the evaluation of an individual unit, the wall thickness of the unit and all reflecting adjacent materials of the unit are considered. The materials adjacent to the unit should be farther than 30 centimeters (12 inches).
- After all fixed reflectors are accounted for, the controls to prevent the presence of any transient reflectors (e.g., personnel) are identified as IROFS in the ISA Summary.

The use of moderation as a controlled parameter should be considered acceptable if the following apply:

- When using moderation, the applicant commits to ANSI/ANS-8.22-1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators."
- When process variables can affect the moderation, the ISA Summary shows the process variables to be controlled by IROFS.
- Moderation is measured by using instrumentation subject to facility management measures.
- The design of physical structures prevents the ingress of moderators.
- When moderation needs to be sampled, dual independent sampling methods are used.
- Firefighting procedures for use in a moderation-controlled area evaluate the use of moderator material.
- After evaluation of all credible sources of moderation for the potential for intrusion into a moderation-controlled area, the ingress of moderation is prevented or controlled.

The use of concentration as a controlled parameter should be considered acceptable in the following circumstances:

- When process variables can affect the concentration, the ISA Summary shows the process variables to be controlled by IROFS.
- Concentrations of SNM in a process are limited unless the process is analyzed to be safe at any credible concentration.
- When using a tank containing concentration-controlled solution, the tank is normally closed and locked to prevent unauthorized access.

- When concentration needs to be sampled, dual independent sampling methods are used.
- After identification of possible precipitating agents, precautions are taken to ensure that such agents will not be inadvertently introduced.

The use of interaction as a controlled parameter should be considered acceptable if the following applies:

- To maintain a physical separation between units, engineered controls are used to ensure a minimum spacing. If engineered controls are not feasible, augmented administrative controls are used.
- The structural integrity of the spacers or racks should be sufficient for normal and credible abnormal conditions.

The use of neutron absorption as a controlled parameter should be considered acceptable in the following circumstances:

- When using borosilicate-glass raschig rings, the applicant commits to ANSI/ANS-8.5-1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."
- When using fixed neutron absorbers, the applicant commits to ANSI/ANS-8.21-1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors."
- In the evaluation of absorber effectiveness, neutron spectra are considered (e.g., cadmium is an effective absorber for thermal neutrons but ineffective for fast neutrons).

The use of volume as a controlled parameter should be considered acceptable if the following apply:

- Fixed geometry is used to restrict the volume of SNM.
- When the volume is measured, the instrumentation used is subject to facility management measures.

The reviewer should consider the applicant's description of additional commitments for the NCS program acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

- The applicant commits to using the NCS program to promptly detect any NCS deficiencies by means of operational inspections, audits, or investigations and to refer to those responsible for the facility's corrective actions any unacceptable performance deficiencies in IROFS, NCS function, or management measures, so as to prevent recurrence.

- The applicant commits to supporting the facility change mechanism process by performing NCS evaluations to determine changes to processes, operating procedures, criticality controls, IROFS, and management measures.
- The applicant commits to retaining records of NCS deficiencies and to documenting any corrective actions taken.
- The reviewer should consider the applicant's description of measures to implement the facility change process requirements in 10 CFR 70.72 acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:
  - The applicant describes a change control process that is sufficient to ensure that the safety basis of the facility will be maintained during the lifetime of the facility. The change process should be documented in written procedures and should ensure that all changes to SNM processes are evaluated to determine the effect of the change on the safety basis of the process, including the effect on bounding process assumptions, the reliability and availability of NCS controls, and the NCS of connected processes. The change control process should include procedures for the review and approval of facility changes by the NCS function to determine the potential effects on NCS.
  - The change control process should be connected to the facility's configuration management system to ensure that changes to the NCS basis are incorporated into procedures, evaluations, postings, drawings, other safety-basis documentation, and the ISA Summary.
- The applicant's description of measures to implement the reporting requirements in Appendix A, "Reportable Safety Events," to 10 CFR Part 70 for criticality safety-related commitments should be considered acceptable if the commitments are consistent with the overall Appendix A program commitments and the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:
  - The applicant has a program for evaluating the criticality significance of NCS events and an apparatus in place for making the required notification to the NRC Operations Center. Qualified individuals should make the determination of significance of NCS events. The determination of loss or degradation of double-contingency protection should be made against the license and Appendix A to 10 CFR Part 70.
  - The applicant incorporates the reporting criteria of Appendix A to 10 CFR Part 70 and the report content requirements of 10 CFR 70.50, "Reporting Requirements," into the facility emergency procedures.
  - The applicant commits to issuing the necessary report, based on whether the criticality controls and IROFS credited were lost (i.e., they were unreliable or unavailable to perform their intended safety functions), irrespective of whether the safety limits of the associated parameters were actually exceeded.

- If the licensee cannot ascertain within 1 hour whether the criteria of 10 CFR Part 70, Appendix A, paragraph (a) or (b) apply, the applicant commits to treating the event as a 1-hour reportable event.

The applicant may use standards as a means to meet regulatory requirements. Regulatory Guide 3.71 endorses the ANSI/ANS-8 national standards, with some exceptions. The NRC endorsement of these standards means that they provide procedures and methodology generally acceptable to the NRC staff for the prevention and mitigation of nuclear criticality accidents. However, application of a standard is not a substitute for detailed NCS analyses for specific operations.

If the applicant intends to conduct activities to which an NRC-endorsed standard applies, the applicant should meet the intent of the standard by satisfying the following acceptance criteria:

- The license application contains a commitment to follow the requirements (i.e., “shall” statements) of the standard, subject to any exceptions taken by the NRC. The application clearly specifies the version of the standard and the specific provisions to which the applicant is committing.
- If there are requirements in a standard that the applicant does not commit to, the applicant provides sufficient information for the staff to determine if the requirements are not relevant to the applicant’s activities or the license application contains other commitments that are equivalent.

If the licensee commits to a standard that the NRC has not endorsed, is not the most current version endorsed by the NRC, or is an unendorsed version of a previously endorsed standard, the license application should include justification for this commitment.

Regulatory Guide 3.71 endorses, in part or in full, the following ANSI/ANS-8 national standards:

- ANSI/ANS-8.1-1998
- ANSI/ANS-8.3-1997
- ANSI/ANS-8.5-1996
- ANSI/ANS-8.6, “Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ,” 1983 (reaffirmed in 1995)
- ANSI/ANS-8.7-1975 (reaffirmed in 1987)
- ANSI/ANS-8.9-1987 (reaffirmed in 1995)
- ANSI/ANS-8.10-1983 (reaffirmed in 1988)
- ANSI/ANS-8.12-1987 (reaffirmed in 1993)
- ANSI/ANS-8.15-1981 (reaffirmed in 1995)
- ANSI/ANS-8.17-1984 (reaffirmed in 1997)

- ANSI/ANS-8.19-1996
- ANSI/ANS-8.20-1991
- ANSI/ANS-8.21-1995
- ANSI/ANS-8.22-1997
- ANSI/ANS-8.23-1997
- ANSI/ANS-8.24-2007
- ANSI/ANS-8.26, "Criticality Safety Engineer Training and Qualification Program," 2007

#### 5.4.3.3 *Integrated Safety Analysis Summary*

The reviewer should find the applicant's criticality safety information acceptable if it provides reasonable assurance that the acceptance criteria presented below are adequately addressed and satisfied. The applicant may elect to incorporate some or all of the requested process information in the facility and process description (SRP Section 1.1) or the ISA Summary, rather than in this section. Either approach is acceptable, as long as the information is adequately cross-referenced.

The regulation in 10 CFR 70.65(b)(3) requires, in the ISA Summary, a description of each process in the facility. The applicant's descriptions of the criticality processes are acceptable if they meet the conditions described below.

##### 5.4.3.3.1 Criticality Process Description

Process descriptions are sufficiently detailed to allow an understanding of the criticality to allow development of potential accident sequences.

Process descriptions are sufficiently detailed to allow an understanding of the theory of operation.

##### 5.4.3.3.2 Criticality Accident Sequences

The use of accident sequences to demonstrate compliance is acceptable in the following circumstances:

- The applicant provides a general description of the accident sequences identified in the ISA process for criticality hazards.
- The ISA Summary describes the hazards identified in the ISA. Each accident sequence identified by the applicant in the ISA should include a criticality hazard evaluation of potential interactions and key assumptions, vessels, process equipment, and facility personnel. The hazard evaluation should use appropriate accepted methods.

- The applicant provides reasonable assurance that measures to mitigate the consequences of accident sequences identified in the ISA Summary are consistent with actions described in SRP Chapter 8. (Note that some facilities are not required to have an emergency plan.) Preventive controls and measures should be the primary means of protection against nuclear criticality accidents (§70.61(d)).
- All the credible criticality accident sequences should be assumed to have high consequences.

#### 5.4.3.3.3 Criticality Items Relied on for Safety

The regulation in 10 CFR 70.65(b)(6) requires a list briefly describing all IROFS in sufficient detail to understand their functions in relation to the performance requirements.

The applicant provides, in the ISA Summary, a list of criticality safety controls (i.e., IROFS) suitable to prevent criticality accidents. This list should also briefly describe the IROFS, in sufficient detail to permit an understanding of their safety functions. The applicant should demonstrate that the likelihood of each credible high-consequence event will be highly unlikely.

If the applicant takes a graded approach to safety, in accordance with 10 CFR 70.62(a), the reviewer should establish that the grading of IROFS is appropriate and sufficient to protect against criticality risks. For example, the applicant should consider reliance on passive controls of active systems and defense in depth, in accordance with 10 CFR 70.64(b). To reduce common-mode failures, the applicant should favor design features that use independent sources of motive force.

#### 5.4.3.3.4 Management Measures

The applicant should review management measures to ensure the availability and reliability of IROFS when they are required to perform their safety functions. Management measures may be graded commensurate with risk. The regulation in 10 CFR 70.65(b)(4) requires information that demonstrates the licensee's compliance with the performance requirements, including a description of the management measures.

The application should meet the following criteria:

- The application should describe the engineering approach, basis, or schemes employed for maintaining safety in normal operations.
- The ISA Summary must identify the administrative and engineered controls to prevent a criticality hazard. The applicant should also explain how any safety grading of IROFS and management measures has been made and how such grading is commensurate with the reduction in risk that the IROFS are designed to achieve.
- The application should demonstrate the management measures proposed to ensure that IROFS are available and reliable by briefly describing the following:
  - procedures to ensure the reliable operation of engineered controls (e.g., inspection and testing procedures and frequencies, calibration programs,

functional tests, corrective and preventive maintenance programs, criteria for acceptable test results)

- procedures to ensure that administrative controls will be correctly implemented, when required (e.g., employee training and qualification in operating procedures, refresher training, safe work practices, development of standard operating procedures, training program evaluation)
- the configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements used by the applicant (see SRP Sections 11.3.1 through 11.3.8)
- management provisions for the following:
  - training and qualifications of NCS management and staff
  - auditing, assessing, and upgrading the NCS program
  - maintaining current NCS safety-basis documentation
  - installing and maintaining a CAAS to detect and annunciate an inadvertent nuclear criticality
  - referring NCS deficiencies to the corrective action program
  - retaining records of the NCS program, including independent reviews, audits, and documentation of corrective actions taken

#### 5.4.3.3.5 Requirements for New Facilities or New Processes at Existing Facilities

The application should address the BDC for new facilities or new processes at existing facilities that require a license amendment under 10 CFR 70.72. The baseline criteria must be applied to the design of new processes but do not require retrofits to existing facilities or existing processes (§70.64(a)); however, all facilities and processes must comply with the performance requirements in 10 CFR 70.61.

The applicant should state clearly how the design of the new facility or process provides for criticality control, as required in 10 CFR 70.64(a)(9). The discussion should identify how the following were considered in the design:

- subcriticality under normal and abnormal conditions
- CAAS
- implementation of double contingency

The licensee could indicate its preference in the selection of controls, such as the following:

- engineered over administrative controls

- favorable geometry design
- two-parameter control

## **5.5 Review Procedures**

The reviewer should use the regulatory guidance of this chapter, references in this chapter, and the applicant's reports to the NRC (e.g., NRC Bulletin 91-01, "Reporting Loss of Criticality Safety Controls"; 10 CFR 70.50; and 10 CFR 70.74, "Additional Reporting Requirements").

### **5.5.1 Acceptance Review**

The primary reviewer should review the applicant's NCS information for completeness with respect to the requirements in 10 CFR 70.22, 10 CFR 70.24, 10 CFR 70.61, 10 CFR 70.62, 10 CFR 70.64, and 10 CFR 70.65, and the acceptance criteria in Section 5.4. If deficiencies are identified, then either the reviewer should ask the applicant to submit additional material before the start of the safety evaluation or the application should be denied.

### **5.5.2 Safety Evaluation**

After the application has been accepted, the primary reviewer should conduct a complete review of the application and determine if it meets the requirements for approval specified in Section 5.4. The primary reviewer should consult with the supporting reviewers, as appropriate, to identify and resolve any issues of concern related to the licensing review. The primary reviewer should also coordinate with other primary reviewers of SRP Chapters 2, 3, 8, and 11 to confirm that the application meets all acceptance criteria pertinent to NCS. The reviewer should also coordinate with other primary reviewers in radiation protection, chemical safety, and fire protection, as well as other disciplines as appropriate (e.g., seismic), to ensure appropriate consideration of any cross-cutting issues.

#### *5.5.2.1 License Application*

The primary reviewer should review the applicant's NCS information in the license application for completeness with respect to the requirements in 10 CFR 70.22, 10 CFR 70.23, 10 CFR 70.24, 10 CFR 70.61, 10 CFR 70.62, 10 CFR 70.64, and 10 CFR 70.65, and the acceptance criteria in Section 5.4.

During the license application review the reviewer should identify and note any items or issues that should be inspected during an operational readiness review, if such a review will be performed. These items could include confirming that the commitments made in the license application are implemented through procedures and training.

If, during the review, the primary reviewer determines a need for additional information, the reviewer coordinates a request for additional information with the licensing project manager. The reviewer should ascertain that the criticality safety approach is consistent with other sections of the application, including those addressed by SRP Chapters 2, 3, 4, 6, 8, and 11.

For an existing facility, the reviewer may consult NRC inspectors to identify and resolve any issues related to the licensing review. These interactions should be coordinated through the licensing project manager.

The primary reviewer will prepare safety evaluation report (SER) input for the licensing project manager in support of the licensing action.

#### *5.5.2.2 Nuclear Criticality Safety Program*

The reviewer should review all aspects of the applicant's NCS program, including management, organization, and technical practices. The reviewer should identify and note any items or issues relating to the NCS program and commitments that should be inspected during an operational readiness review, if such a review will be performed. These items could include confirming that the commitments made in the license application are implemented through procedures and training.

If, during the review, the primary reviewer determines a need for additional information regarding the NCS program, the reviewer coordinates a request for additional information with the licensing project manager.

For an existing facility, the reviewer may consult NRC inspectors to identify and resolve any issues related to the NCS program commitments. These interactions should be coordinated through the licensing project manager.

#### *5.5.2.3 Integrated Safety Analysis Summary*

The results of the ISA support the overall safety basis for the criticality safety evaluation. The reviewer should assess the criticality safety risks identified in the ISA Summary and ensure that the level of safety is reflected in the design and the operational plans for the facility. The reviewer should establish that the applicant's facility design, operations, and IROFS for criticality safety provide reasonable assurance that they will function as intended, be reliable and available to perform their safety function, and provide for the safe possession and use of licensed material at the facility.

### **5.6 Evaluation Findings**

SRP Chapter 3 contains the evaluation findings for the ISA Summary requirements for 10 CFR 70.65.

If the staff's review verifies that the safety program description presents sufficient information to satisfy the acceptance criteria in SRP Section 5.4, the staff may document its review as follows:

The staff has reviewed the Nuclear Criticality Safety (NCS) program and requirements for criticality safety for [name of facility] according to SRP Chapter 5. The staff has reasonable assurance of the following:

- The applicant will have in place a staff of managers, supervisors, engineers, process operators, and other support personnel who are qualified to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures.
- The applicant's conduct of operations will be based on NCS technical practices, which will ensure that the fissile material will be possessed, stored, and used safely, according to the requirements in 10 CFR Part 70.

- The applicant will develop, implement, and maintain a criticality accident alarm system in accordance with both the requirements in 10 CFR 70.24 and the facility emergency management program.
- The applicant will have in place an NCS program that meets the performance requirements in 10 CFR 70.61(b), the subcriticality requirement in 10 CFR 70.61(d), and the baseline design criteria requirements in 10 CFR 70.64(a).
- Based on this review, the staff concludes that the applicant's NCS program meets the requirements of 10 CFR Part 70 and provides reasonable assurance of the protection of public health and safety, including that of workers, and the environment.

## **5.7 References**

*U.S. Code of Federal Regulations*, Chapter I, Title 10, "Energy," Part 70, "Domestic Licensing of Special Nuclear Material."

U.S. Nuclear Regulatory Commission, "Integrated Safety Analysis Guidance Document," NUREG-1513, May 2001.

U.S. Nuclear Regulatory Commission, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410, March 1998.

American National Standards Institute/American Nuclear Society, "Criticality Accident Alarm System," ANSI/ANS-8.3-1997.

U.S. Nuclear Regulatory Commission, "Nuclear Criticality Safety Standards for Fuels and Material Facilities," Regulatory Guide 3.71, October 2005.

American National Standards Institute/American Nuclear Society, "Nuclear Criticality Accident Emergency Planning and Response," ANSI/ANS-8.23-1997.

American National Standards Institute/American Nuclear Society, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials," ANSI/ANS-8.7-975.

American National Standards Institute/American Nuclear Society, "Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate," ANSI/ANS-8.9-1987.

American National Standards Institute/American Nuclear Society, "Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement," ANSI/ANS-8.10-1983.

American National Standards Institute/American Nuclear Society, "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors," ANSI/ANS-8.12-1987.

American National Standards Institute/American Nuclear Society, "Nuclear Criticality Control of Special Actinide Elements," ANSI/ANS-8.15-1981.

American National Standards Institute/American Nuclear Society, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors," ANSI/ANS-8.17-1984.

American National Standards Institute/American Nuclear Society, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1998.

U.S. Nuclear Regulatory Commission, "Chemical Process Safety at Fuel Cycle Facilities," NUREG-1601, September 3, 1997.

American National Standards Institute/American Nuclear Society, "Administrative Practices for Nuclear Criticality Safety," ANSI/ANS-8.19.

H.K. Clark, "Maximum Safe Limits for Slightly Enriched Uranium and Uranium Oxide," DP-1014, Du Pont de Nemours and Co., Aiken, SC, 1966.

R.A. Knief, "Nuclear Criticality Safety—Theory and Practice," American Nuclear Society, La Grange Park, IL, 1985.

H.C. Paxton and N.L. Pruvost, "Critical Dimensions of Systems Containing  $^{235}\text{U}$ ,  $^{239}\text{Pu}$ , and  $^{233}\text{U}$ ," LA-10860-MS, Los Alamos National Laboratory, Los Alamos, NM, 1987.

N.L. Pruvost and H.C. Paxton, "Nuclear Criticality Safety Guide," LA-12808/UC-714, Los Alamos National Laboratory, Los Alamos, NM, 1996.

W.R. Stratton (D.R. Smith Revisor), "A Review of Criticality Accidents," DOE/NCT-04, U.S. Department of Energy, March 1989.

U.S. Department of Energy, "Facility Safety," DOE Order 420.1 (Change 2), October 24, 1996.

## APPENDIX A

### NUCLEAR CRITICALITY SAFETY PERFORMANCE REQUIREMENTS AND DOUBLE CONTINGENCY PRINCIPLE

Title 10 of the Code of Federal Regulations (10 CFR) Part 70, Subpart H contains three separate requirements to ensure nuclear criticality safety. One requirement, 10 CFR 70.64(a)(9), requires that the design of new facilities and processes provide for criticality control including adherence to the double contingency principle. A second requirement, 10 CFR 70.61(b), requires that high consequence events (which typically will include criticality accidents) be highly unlikely. A third requirement, 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 purpose of this appendix is to clarify the relationship between these three requirements.

#### Discussion

There are three separate requirements in 10 CFR Part 70 for ensuring nuclear criticality safety. The first requirement of 10 CFR 70.64(a)(9) is more prescriptive and deterministic than the performance requirements of 10 CFR 70.61. 10 CFR 70.64 establishes baseline design criteria for new facilities and processes, similar to general design criteria in 10 CFR Part 50. One of these baseline design criteria applies directly to criticality safety. Specifically, 10 CFR 70.64(a)(9) requires that the design “provide for criticality control including adherence to the double contingency principle.” Section 70.64(b) further specifies that new facilities or processes must incorporate defense-in-depth practices, which is defined as a “design philosophy, applied from the outset and through completion of the design, that is based on providing successive levels of protection such that health and safety will not be wholly dependent upon any single element of the design, construction, maintenance, or operation of the facility.” Section 70.64(b)(1) specifically mentions preference for the selection of engineered controls over administrative controls to increase overall system reliability.

Another more risk-informed and performance-based requirement is contained in 10 CFR 70.61. In short, this regulation stipulates that credible high consequence events shall be made “highly unlikely” or be mitigated (10 CFR 70.61(b)) and that intermediate consequences shall be made “unlikely” or be mitigated (10 CFR 70.61(c)). High and intermediate consequence thresholds for workers and members of the public are established for both chemical and radiological events. Under this risk-informed and performance-based regulation a criticality accident would typically be considered a high consequence event to the worker since the worker could receive a dose in excess of 100 rem TEDE (total effective dose equivalent).

In addition, there is a separate provision within 10 CFR 70.61 that specifically addresses criticality safety. Section 70.61(d) states that, in addition to meeting the requirements above for high and intermediate consequence events, the “risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes

are subcritical, including use of an approved margin of subcriticality for safety. Preventive controls and measures must be the primary means of protection against nuclear criticality accidents.” The purpose of this is to preclude a situation where nuclear criticality would be permitted as long as the dose thresholds of § 70.61(b) and § 70.61(c) are not exceeded.

Thus, 10 CFR Part 70 contains three separate and distinct requirements related to precluding nuclear criticality (10 CFR 70.64(a)(9), 10 CFR 70.61(b), and 10 CFR 70.61(d)), besides provisions in § 70.24 and § 70.52, which pertain to mitigating the consequences of a criticality accident and reporting its occurrence.

#### Section 70.61(d) of 10 CFR Part 70

Section 70.61(d) requires that under normal and credible abnormal conditions all nuclear processes are subcritical including use of an approved margin of subcriticality for safety. In addition, preventive controls and measures must be the primary means of protection against criticality. Meeting this performance requirement entails a number of factors. First, all normal and credible abnormal conditions must be identified. There are many different methods that may be employed to do this, but a systematic methodology should be used to provide reasonable assurance that the complete spectrum of credible conditions has been identified.

Normal conditions are those specifically allowed for as part of the normal modes of operation in the facility design (i.e., conditions that may occur without the failure of any items relied on for safety (IROFS)). Abnormal conditions are those events not planned for as a regular occurrence in the facility or operation design. They include those undesirable conditions that are the result of external events and process deviations, including those resulting from the failure of identified IROFS. Credible abnormal events include both credible single events (e.g., an external event or failure of a single IROFS) and credible sequences of events. Credible sequences of events include, but may not be limited to, chains of independent but not unlikely process deviations (i.e., not precluded by IROFS) and chains of related failures of IROFS (i.e., failures that are not independent). Some judgment must be employed in determining what constitutes a credible abnormal condition. It is not necessary to include multiple independent failures of IROFS within the spectrum of credible abnormal conditions. Additional guidance on what is considered not credible is contained in NUREG-1520, Section 3.4.3.2:

- a. “An external event for which the frequency of occurrence can conservatively be estimated as less than once in a million years.”
- b. “A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive....”
- c. “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. To provide adequate assurance of subcriticality, this must include margin. There are several different ways to demonstrate sub-criticality, as discussed below:

- If subcriticality is demonstrated using an appropriately validated calculation method, then  $k_{\text{eff}}$  (K effective) (including calculation's uncertainties) must be less than the approved upper subcritical limit (USL), as specified in the license. Meeting this requires that models bound actual anticipated conditions (e.g., tolerances and uncertainties appropriately taken into account, most reactive credible system parameters allowed are assumed), as specified in the license. Additional guidance is provided in the criticality chapter of NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," (Sections 5.4.3.4.1, 5.4.3.4.2, and 5.4.3.4.4).
- Subcritical margin may also be expressed in terms of system parameters rather than system  $k_{\text{eff}}$ . An example would be where the licensee or applicant has committed to use mass or dimensional limits that are some specified fraction of the critical values of those parameters. In such cases, the approach used must be approved by the NRC.
- Subcriticality may be demonstrated on the basis of subcritical limits included in the license, U.S. Nuclear Regulatory Commission (NRC) endorsed American National Standards Institute (ANSI) standards, or other documents that have been approved or endorsed by NRC. Approval or endorsement by the NRC implies that the Agency has found these references to include an acceptable margin of subcriticality for safety.
- Industry handbooks of criticality data may also be used if widely accepted in the nuclear industry and if used in accordance with any limitations of that data. The NRC, however, reserves the right to evaluate the use of such handbooks on a case-by-case basis.

The requirement that preventive controls and measures be the primary means of protection against criticality is satisfied if engineered or administrative controls relied on to meet § 70.61(d) are designed to prevent occurrence of the critical excursion rather than mitigate its consequences. By stating that prevention should be the *primary* means of protection, it is recognized that there may be extraordinarily rare occasions when prevention alone is not sufficient to meet § 70.61(d). Such cases require convincing demonstration that there is no practicable way to meet § 70.61(d) with solely preventive measures.

Some examples where the § 70.61(d) requirement has not been met:

- A process in which the most reactive credible conditions have not been modeled and have not been shown to have  $k_{\text{eff}}$  less than the approved USL.
- A process in which subcriticality is based on criticality calculations, but the model is outside the area of applicability of the calculation method.
- A process for which there is an unanalyzed or unanticipated credible abnormal condition (e.g., unanticipated failure of an IROFS or unanticipated external event).
- A process for which there is a credible common-mode event that can result in the failure of all criticality controls such that it can lead to a critical configuration.
- A process in which the designated IROFS are not sufficient to limit the system to a subcritical configuration.

## Relationship of 10 CFR 70.61(b) to 10 CFR 70.61(d)

Section 70.61(b) states “. . . the risk of each credible high consequence event must be limited. . . . Controls . . . shall be applied to the extent needed to reduce the likelihood of occurrence of the event so that . . . the event is highly unlikely . . . .”

Section 70.61(d) states “. . . the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical, including an approved margin of subcriticality . . . .”

As written, the rule language requires both provisions (i.e., § 70.61(b) and § 70.61(d)) be met, since § 70.61(d) states “In addition to complying with paragraphs (b) and (c) of this section . . . .” However, during the 10 CFR Part 70 rulemaking, regulated industry representatives met with NRC and submitted letters in which they expressed their desire that NRC not consider criticality accidents high consequence events and not associate quantitative likelihoods with double contingency. As discussed in the release notes issued with the 10 CFR Part 70 rulemaking, in response to industry arguments accidental criticality was explicitly removed from the high consequence (§ 70.61(b)) category and a separate performance requirement for criticality (§ 70.61(d)) was created. The staff felt that in so doing, both the industry’s desires as well as the staff’s needs would be met. Further, the staff felt that the § 70.61(d) requirement required the same information as that required by § 70.61(b). Saying all nuclear processes must be subcritical in § 70.61(d) implies that criticality events must be prevented. Moreover, since likelihood is never zero, some non-zero likelihood must be assumed; the highly unlikely requirement in § 70.61(b) is appropriate for this. Therefore, the staff felt that by removing criticality explicitly from § 70.61(b) and creating § 70.61(d) during the rulemaking, the staff still retained its desired outcome—to prevent criticality accidents from occurring. The final rule Statement of Considerations (SOC) stated that “. . .the NRC believes that a separate performance requirement for nuclear criticality prevention is appropriate. The staff recognizes that many (but not all) nuclear criticality accidents would reasonably be expected to result in worker doses that exceed the high- and intermediate-consequence standards in 10 CFR 70.61(b) or (c). However, regardless of the dose directly resulting from the accident, an inadvertent nuclear criticality should be avoided. This is consistent with the Commission’s goal to prevent inadvertent criticalities, as reflected in the NRC Strategic Plan (NUREG-1614) . . . .” However, there remained ambiguity regarding the relationship between § 70.61(b) and § 70.61(d). While the staff’s intent was to have a single performance requirement for criticality accidents, this cannot be substantiated by a literal examination of the final rule.

Comparing the language in § 70.61(b) and (d), one concludes that § 70.61(d) is actually more restrictive than § 70.61(b). Section 70.61(d) essentially requires that there be no criticality accidents, with a high degree of assurance, whereas § 70.61(b) essentially requires that deaths and injuries (as implemented through a dose limit) be precluded (i.e., be made to be highly unlikely). If criticality accidents are prevented, then deaths and injuries are also prevented. However, the converse is not necessarily true; if deaths and injuries are prevented, criticality accidents are not necessarily prevented. Therefore, if one meets § 70.61(d), then one also automatically meets § 70.61(b); and if one meets § 70.61(b) through preventive means, and also meets the additional requirements specified in § 70.61(d), then one also meets § 70.61(d) in full. Thus, if a licensee chooses to address criticality event sequences under 10 CFR 70.61(b)

with a preventive strategy and has an approved margin of subcriticality for safety, then the licensee will have also met the requirements under 10 CFR 70.61(d). However, if the licensee chooses to address criticality event sequences under 10 CFR 70.61(b) with a mitigative strategy, then the licensee will not have met the requirements under 10 CFR 70.61(d) and additional controls will have to be identified to ensure subcriticality.

Another consideration is that both § 70.61(b) and § 70.61(d) set the standard that must be met (i.e., the performance requirements), but not the methodology. Methodology requirements are contained in § 70.62. One cannot look at § 70.61 in a vacuum. All other 10 CFR Part 70 provisions must also be met, including the § 70.62(c) provision that requires the integrated safety analysis (ISA) to include radiological hazards, facility hazards, potential accident sequences, and identification of IROFS as well as the assumptions and conditions under which the IROFS are relied upon to support compliance with § 70.61 performance requirements. It also requires that the ISA team include a person with experience in criticality safety. These requirements must be met regardless of whether the licensee attempts to meet the performance requirements starting from § 70.61(b) or § 70.61(d). The three options below can be seen to be equivalent when one considers that § 70.62 must also be met for all cases.

To meet the regulations and prevent criticalities, an applicant/licensee may use one of the three approaches below (in conjunction with other 10 CFR Part 70 requirements, including those in § 70.62):

1. Demonstrate compliance with § 70.61(d); or
2. Demonstrate compliance with § 70.61(b), considering only preventive controls and including an approved margin of subcriticality; or
3. Separately demonstrate compliance with both § 70.61(d) and § 70.61(b).

Use of any of the above three approaches will satisfy the regulations.

That both § 70.61(b) and § 70.61(d) apply to criticality is supported by this SRP. In addition, there are several references to the requirement to make criticality highly unlikely.

#### Double Contingency Principle § 70.64(a)(9)

In addition to complying with the performance requirement in § 70.61, new facilities and processes are required to comply with the baseline design criteria in § 70.64. Section 70.64(a)(9) requires that the design provide for criticality control, including adherence to the double contingency principle (DCP). In addition to this requirement for new facilities and processes, many existing facilities and processes have license commitments to meet the DCP for licensed activities. Although Subpart H of 10 CFR Part 70 is relatively new, this conceptual framework is not new. Licensees have historically committed to ANSI/American Nuclear Society -8.1 (ANSI/ANS-8.1). This standard also requires that nuclear processes be ensured to be subcritical under normal and credible abnormal conditions. By contrast, the DCP is stated as a recommendation of ANSI/ANS-8.1. Therefore, the standard recognizes that adherence to the DCP can be one means, but is not necessarily the only means of meeting the underlying subcriticality requirement. The conditions under which compliance with the DCP ensures that § 70.61(d) is met are discussed below.

The double contingency principle is a design principle intended to be used in designing a facility that meets the performance requirements of § 70.61. The definition in § 70.4 (“...process designs *should* incorporate sufficient factors of safety...”) implicitly recognizes that there may be some cases in which a strict adherence to the double contingency principle is not practicable. This should be an exceedingly rare situation and should be accompanied by a convincing demonstration that a strict adherence to the double contingency principle is not practicable. Section 70.64(a) allows for this in stating that licensees must maintain the application of this criterion unless the integrated safety analysis (ISA) demonstrates that it is not relied on for safety or otherwise does not require adherence.

The presence of two controls may not be necessary, or may not be sufficient, to meet the DCP. 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. In general, there will be many pathways to criticality and, therefore, more than two controls required to meet the DCP for an entire process.

In addition, § 70.64(b)(1) requires that the design must incorporate, whenever practicable, preference for the selection of engineered over administrative controls to increase overall system reliability. Passive engineered controls are generally preferable to active engineered controls, and engineered to administrative controls. In addition, process design should rely on geometry control as opposed to control of other parameters whenever practicable, and on diverse means of control (e.g., reliance on two different criticality parameters or different means of controlling one parameter) whenever practicable, to minimize the potential for common-mode failure. Cases in which these preferences cannot be complied with will generally require more justification to show adherence with the DCP. For example, one cannot claim that the double contingency principle is met with only two controls (regardless of type) if the resulting configuration fails to protect against all credible pathways to criticality or limit the risk of inadvertent criticality as required in 10 CFR 70.61(d).

#### Relationship between § 70.61 and § 70.64(a)(9)

As stated above, adherence to the DCP can be one means of meeting the performance requirements of § 70.61(d) (and, therefore, also § 70.61(b)). Historically, a number of different approaches to double contingency have been used. Some cases that have been used in the past may not be sufficiently robust to satisfy the performance requirements of § 70.61. Typically, this has been due to a reliance on controls that were not sufficiently robust (e.g., weak administrative controls). The purpose of this guidance is not to promote a new standard for all applications but rather to clarify when adherence to the DCP will establish a sufficient basis for meeting the performance requirements. To facilitate this, the following guidance is provided on the various terms in the definition of the DCP:

Unlikely changes in process conditions should be expected to occur rarely, or not at all, during the lifetime of the facility. Operational events that occur regularly should not be credited as a contingency relied on to meet the DCP (although they may constitute part of a contingency if a combination of events may be considered unlikely). Therefore, the occurrence of any such event generally reveals a deficiency in the design that should result in corrective action.

Determination that a contingency is unlikely should be based on objective attributes of the criticality controls, rather than on subjective judgment alone. Examples of such attributes are environmental factors that can degrade the reliability and availability of controls, margin, and redundancy and diversity of controls. (Guidance on some of the availability and reliability qualities that should be considered is provided in Section 3.4.3.2(9) of this SRP and NUREG-1718, "Standard Review Plan for the Review of a License Application for a Mixed Oxide (MOX) Fuel Fabrication Facility," Section 5.4.3.2(B)(vii).) Management measures should be provided, as needed, to ensure that the failure of the criticality controls is an unlikely contingency. (NOTE: Usage of the term "unlikely" in the DCP is not equivalent to the term as used in § 70.61(c) for intermediate consequence events.)

Independent changes in process conditions are such that one contingency neither causes another contingency nor increases its likelihood of occurrence. The existence of any credible common-mode failure of both contingencies means that it is not valid to consider them independent. For example, related actions performed by the same individual or using the same equipment will not generally be sufficiently independent to meet the DCP.

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. Prompt detection and correction of abnormal conditions should thus be provided to restore double contingency protection. The time required to detect and correct failures should be significantly shorter than the anticipated time between failures in order for there to be significant risk reduction provided from failure detection.

Changes in process conditions does not imply that reliance on two different parameters is mandatory to meet the DCP. Reliance on two different parameters is preferable to reliance on two controls on a single parameter, however, because of the difficulty in achieving complete independence when controlling one parameter. In those cases in which single parameter control is unavoidable, great care should be taken to ensure that no common-mode failures exist.

In addition to meeting the above, the following guidance is provided to illustrate the conditions under which adherence to the double contingency principle (in terms of the guidance above) 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 § 70.61(e). (Reliance should be based on items that are designated as IROFS in the ISA Summary and not on random factors that may or may not be maintained.)
- The condition resulting from the failure of a leg of double contingency has been shown to be subcritical with an acceptable margin (e.g.,  $k_{\text{eff}}$  is less than USL, parameters are within subcritical limits specified in the license or endorsed standards).
- Controls are sufficiently reliable to ensure that each change in process conditions necessary for criticality is "unlikely." Management measures are established to ensure that they are available and reliable to perform their safety function.

Because the DCP is only one means of meeting the performance requirements, it is possible to meet the DCP without meeting the conditions above (including designating criticality controls as IROFS in the ISA Summary). In this case, however, another method must be relied on to meet the § 70.61 performance requirements. However, in order to use compliance with the DCP as part of the demonstration of meeting the § 70.61 performance requirements, these conditions should be met.

Some specific examples of control systems that meet § 70.61(d) through use of the DCP follow:

*A passive geometry control in which no credible failure mode (e.g., bulging, corrosion, or leakage) exists and which has been placed under configuration management:*

- A favorable geometry vessel in a benign environment in which corrosion or other material degradation is not credible. In addition, the vessel is of such robust construction (e.g., thick stainless steel, steel surrounded by concrete) that it is unquestionably not going to leak, and there is no credible mechanism for the material to accumulate in an unfavorable configuration.
- A tank that is not authorized to contain fissile material is located far outside the fissile material handling areas and is physically isolated from fissile liquid processes by a blank flange or siphon break, such that backflow is not credible.

*Two passive controls in which there is a credible failure mode, and there are sufficient management measures to ensure the controls continue to perform their safety functions (e.g., periodic surveillance to detect corrosion/bulging):*

- A favorable geometry solution column, in which leakage of the tank is a credible upset. In addition, the column is in an area in which the solution would leak into a favorable geometry dike, and the leakage would be self-revealing (i.e., column is in a continually manned area) or the column and dike would be subject to periodic surveillance.
- A double-sleeved solution line in which leakage of the inner pipe would be quickly detected (e.g., by conductivity probe between the pipes or by transparent baffling).
- A storage array in which fissile material is stored in fixed geometry containers, and the spacing between containers is fixed by birdcages or other fixed devices, and geometry and spacing controls are ensured by configuration management and periodic walkthroughs.

*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:*

- A calciner relying on geometry and moderation control in which geometry control is provided by limiting the calciner interior to the height of a single layer of pellet boats, and moderation control is provided by monitoring of the calciner temperature. Temperature control is ensured by thermocouples that alarm if the temperature drops below a minimum set-point.

- A down-blending tank that is subcritical for uranium solutions with less than a limiting enrichment in which volume control is provided by the design of the tank and enrichment control is provided using mass flow totalizers and a mechanical stirrer. The failure of these active devices automatically stops the transfer of solution and actuates an alarm.
- A large geometry tank relying on Raschig rings for criticality control in which the Raschig rings are only approved up to a limiting concentration, and the concentration is controlled by an in-line sodium iodide detector that closes an isolation valve when actuated.

*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:*

- A powder handling glovebox relying on moderator and mass control in which moderator control is provided by the glovebox design (e.g., airtight, dry nitrogen atmosphere, sloped ventilation ductwork) and mass is procedurally controlled by limiting batch size. In addition, mass transfers must be logged into a computer tracking system that alarms if mass limits are exceeded.
- A vessel in which the volume of fissile solution is controlled by the diameter of the tank and by procedurally limiting the solution height. In addition, operator actions are backed up with a high-level switch equipped with an alarm.

*One engineered control and one simple administrative control in which the reliability of the administrative control is subject to a high degree of redundancy:*

- Solution transfer from favorable to unfavorable geometry relying on two controls on concentration. Two different operators are required to draw separate samples which are then analyzed in the laboratory by two different methods and shown to be within concentration limits before transfer is authorized. In addition, the area supervisor maintains control of a key to the transfer pump so that the procedure may not be inadvertently bypassed. This is backed up with an in-line sodium iodide detector that automatically closes an isolation valve if concentration limits are exceeded.

(NOTE: Use of two independent samples is generally not considered adequate for both legs of double contingency because of the difficulty in ensuring complete independence between the samples.)

*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:*

- A glovebox relying on dual mass control in which two operators or an operator and a supervisor must confirm that placing material into the glovebox will not result in the mass limit being exceeded. In addition, criticality would require the mass limit to be exceeded multiple times, which would be difficult to achieve and would be readily apparent.

- A drum storage array limited to a vertical stack of four drums in which there are no forklifts in the area capable of raising a drum above this height. In addition, the drums are very heavy and violating the stack height limit would require an immense physical effort.
- A planar storage array in which mass-controlled containers are procedurally limited to not less than 24 inches center-to-center, and in which criticality would require assembling a very large number of containers into a spherical heap and reflecting them intimately with water.

*Other considerations ensuring that there is no credible event leading to criticality:*

- A facility handling uranium enriched to no more than 1 weight percent (wt%) uranium-235 ( $^{235}\text{U}$ ).
- A facility in which the site-wide limit is less than a minimum critical mass.
- A facility storing contaminated soil or equipment with a very low uranium concentration in which there is no known concentration mechanism that can lead to a critical configuration.

Some examples of control systems that would not meet § 70.61(d) through use of the DCP:

*Double contingency consisting of two single operator actions without any supervisor verification or redundancy:*

- Solution transfers that rely only on two operators drawing separate samples or in which a single procedural deviation could cause an unauthorized transfer.
- A mass controlled system in which triple batching (i.e., two successive batching errors) could result in criticality when the mass transfers are done by a single operator.
- A storage array in which two violations on administrative spacing requirements could credibly lead to criticality.

*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 solution vessel in which the operator is required to confirm concentration or chemical form by visually observing a color change in the solution.
- A tank in which the operator is required to verify prior to operation that the tank is “essentially empty.”

*A leg of double contingency consisting of complex administrative tasks composed of multiple steps that are susceptible to error:*

- A glovebox in which the operator is required to calculate the mass of plastic, paper, and other miscellaneous materials in order to comply with moderator control.

- A solution transfer operation in which one leg consists solely on a single sample being correctly drawn, labeled, analyzed, recorded, and read.
- Maintenance on a dissolution process in which criticality safety relies on the correct performance of a procedure to replace an in-line filter. The procedure requires that the filter be removed, flushed, and re-installed in a multi-step process that has several opportunities for failure.

*A leg of double contingency consisting of an administrative control with insufficient margin to ensure that the safety limit will not be exceeded:*

- A glovebox in which mass is controlled administratively, and in which the normal mass limit is almost equal to the minimum critical mass.
- A planar storage array in which spacing between containers is administratively limited to be less than 24 inches center-to-center, and in which criticality will result if a few containers are placed 23 inches apart.

*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 solution process in which it is plausible for concentrated solution to be allowed to accumulate undetected over a long period of time in an unfavorable geometry.
- A vessel in which geometry control is provided by a double wall, but there is no means of detecting leakage between the walls. In addition, the vessel is of a type known to have a history of leakage (e.g., heat exchanger).

*A leg of double contingency consisting of a control in an environment where its safety function is degraded.*

- A solution vessel relied on for geometry control, but which is subject to pressure fluctuations that can cause the vessel to bulge beyond a favorable diameter.
- Instrumentation whose performance is degraded under conditions that can be reasonably expected during normal operations (e.g., temperature, pressure, presence of corrosive gases, or loss of essential utilities such as electricity, plant air, or water).

*A leg of double contingency consisting of a control where its behavior under adverse conditions is uncertain.*

- An unfavorable geometry pump in which mass control relies on the presumption that the pump will malfunction before an unsafe volume of uranium accumulates in the pump oil, and for which no failures of this type have been observed.

*A leg of double contingency consisting of undeclared design features or process conditions that are not precluded by being explicitly controlled.*

- A powder blending operation in which uranium oxide density that is less than the theoretical density is assumed, but the process variables affecting density (e.g., calcinations temperature, mechanical pressure of the pellet press) are not specifically controlled and there is no confirmatory sampling.
- A solvent extraction process in which nominal concentration of uranyl nitrate is assumed, but there is no in-line monitoring or confirmatory sampling.
- A vault in which the mass limit is not controlled by procedure or license limit, but is merely based on current inventory.
- A process relying on the favorable geometry of passive equipment, but for which the dimensions and/or material composition are not specifically identified as criticality controls.

This list is merely illustrative and not meant to be exhaustive. However, these examples demonstrate that double contingency that satisfies the performance requirements can be based on one, two, or more than two passive engineered, active engineered, or administrative controls, and that reliability and availability of those controls depends on management measures, safety margins, environmental conditions, human factors, and other process and control characteristics. Not every application similar to these examples will be found acceptable—determination must be made on the totality of the information available, and an analyst should consider all factors that may degrade the robustness of the controls.

## **Technical Review Guidance**

### Relationship of 10 CFR 70.61(b) and 10 CFR 70.61(d)

The reviewer needs to assure that all applicable 10 CFR Part 70 criticality provisions (including § 70.62(c)) are met. To meet the regulations and prevent criticalities an applicant/licensee may use one of the three approaches below (in conjunction with other 10 CFR Part 70 requirements, including those in § 70.62):

1. Demonstrate compliance with § 70.61(d); or
1. Demonstrate compliance with § 70.61(b), considering only preventive controls and including an approved margin of subcriticality; or
2. Separately demonstrate compliance with both § 70.61(d) and § 70.61(b).

Use of any of the above three approaches will satisfy the regulations.

Staff should not dictate which of the above three options must be met; rather, staff should assure that the applicant/licensee has met one of these options.

### Double Contingency Principle

One way, but not the only way, of meeting 10 CFR 70.61 is by applying the double contingency principle (defined in 10 CFR 70.4) to accident sequences leading to criticality that are required to be developed per § 70.62. Adherence to the DCP will satisfy the performance requirement of § 70.61(d) (and therefore also § 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 § 70.61(e). (Reliance should be based on items that are designated as IROFS in the ISA Summary and not on random factors that may or may not be maintained.)
- The condition resulting from the failure of a leg of double contingency has been shown to be subcritical with an acceptable margin (e.g.,  $k_{eff}$  is less than USL, parameters are within subcritical limits specified in the license or endorsed standards).
- Controls are sufficiently reliable to ensure that each change in process conditions necessary for criticality is “unlikely.” Management measures are established to ensure that they are available and reliable to perform their safety function.

In the absence of meeting these conditions, an alternate demonstration of compliance with the performance requirements should be provided.