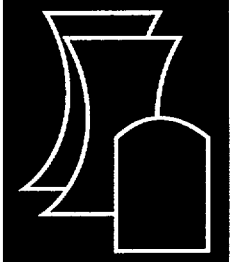
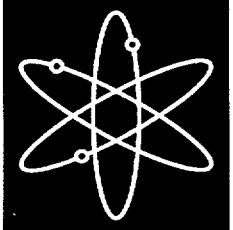
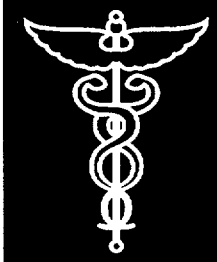
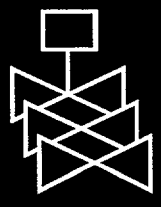
NUREG-1718









Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility

Final Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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NUREG-1718

Standard Review Plan
for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility

Final Report

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Enrichment Section
Special Projects Branch

Division of Fuel Cycle Safety and Safeguards
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This Standard Review Plan (SRP) (NUREG–1718) provides guidance to the NRC staff reviewers in the Office of Nuclear Material Safety and Safeguards who will perform safety, safeguards, and environmental reviews of the anticipated application for a license to possess and use special nuclear material for a mixed oxide fuel fabrication facility under 10 CFR Part 70. This guidance includes the construction approval review specifically related to plutonium processing and fuel fabrication. The SRP ensures the quality, uniformity, stability, and predictability of the staff reviews. It presents a defined basis from which to evaluate proposed changes in the scope and requirements of the staff reviews. The SRP makes information about NRC acceptance criteria widely available to interested members of the public and the regulated industry. Each SRP section addresses the responsibilities of persons performing the review, the review areas, the Commission's regulations pertinent to specific technical matters, the acceptance criteria used by the staff, how the review is accomplished, and the conclusions that are appropriate for the Safety Evaluation Report for both the construction approval review and the license review.

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15.0 MANAGEMENT MEASURES

EXECUTIVE SUMMARY

The NRC expects to receive a license application from Duke Cogema Stone and Webster to license a mixed oxide (MOX) fuel fabrication facility under 10 CFR Part 70. (Throughout this document, Duke Cogema Stone and Webster is referred to as "the applicant.") Under Part 70, the MOX facility is classified as a plutonium processing and fuel fabrication plant. An applicant for a license to possess and use special nuclear material at a plutonium processing and fuel fabrication facility must obtain the NRC's approval prior to starting facility construction. This means that the NRC will conduct two reviews. The first review will determine if the NRC can grant the applicant a construction approval. The NRC makes this determination based on contents of the license application that are specifically required by Part 70 for construction approval. The required material is described in detail in 10 CFR 70.22(f).

The second review will determine if the NRC can grant the applicant a possession and use license for special nuclear material. The NRC makes this determination based on the full content of the license application as described in all of 10 CFR 70.22(f) and Subpart H to 10 CFR Part 70.

The NRC developed this Standard Review Plan (SRP) to provide guidance to the NRC staff reviewers in the Office of Nuclear Material Safety and Safeguards who will perform safety, safeguards, and environmental reviews of the anticipated application for a license to possess and use special nuclear material for the MOX facility—including the construction approval review. The NRC developed NUREG–1718 in parallel with NUREG–1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," which the NRC staff is currently developing to support a rulemaking for 10 CFR Part 70. The NRC staff has attempted to ensure that this SRP is consistent with the requirements of the ongoing rulemaking. The NRC staff has also attempted to ensure that, where applicable for a MOX facility, NUREG–1718 is consistent with the draft of NUREG–1520. However, reviewers and other readers should be aware that this document incorporates guidance that makes it specific for a MOX facility.

The SRP ensures the quality, uniformity, stability, and predictability of the staff reviews. It presents a defined basis from which to evaluate changes in the scope and requirements of the staff reviews. The SRP makes information about NRC acceptance criteria widely available to interested members of the public and the regulated industry. Each SRP section addresses the responsibilities of persons performing the review, the review areas, the Commission's regulations pertinent to specific technical matters, the acceptance criteria used by the staff, how the review is accomplished, and the conclusions that are appropriate for the Safety Evaluation Report for both the construction approval review and the license review. Subject areas for the NRC staff reviews include:

- General information about the applicant and the plant site;
- The applicant's financial qualifications to construct and operate the facility;
- The applicant's organization and administration;
- The analysis of potential accidents, including:
    - The potential hazards;

- The potential likelihoods and consequences; and
- How the applicant will prevent or mitigate potential accidents, where necessary.

- The applicant's provisions to:

  - Protect its employees from exposure to radiation;
  - Protect against a nuclear criticality;
  - Protect the public and environment from radioactive material;
  - Provide for chemical safety;
  - Provide for protection against fires; and
  - Protect the workers, public, and environment during emergencies.

- The applicant's plans to:

  - Protect against the theft or loss of radioactive material;
  - Physically protect the radioactive material, including transportation; and
  - Protect information that is classified in the interest of national security.

- The applicant's management measures, which include:

  - Quality assurance;
  - Configuration management;
  - Maintenance;
  - Training and qualifications;
  - Plant procedures;
  - Audits and assessments;
  - Incident investigation; and
  - Records management.

In each of the subject areas, the document describes:

- The purpose of the review;
- Who should perform what role in the review;
- The specific material a reviewer would expect to see in the application;
- The applicable regulations and guidance;
- The basis for determining if the material is acceptable;
- Instructions for the review (including the construction approval); and
- An example of how to summarize the review and findings.

The NRC staff will use this document as the basis for licensing the MOX fuel fabrication facility.

ACRONYMS AND ABBREVIATIONS

2SX	*2nd Pass Solvent Extraction*
ACI	*American Concrete Institute*
AEC	*Active Engineered Control*
AEGL	*Acute Exposure Guideline Level*
AISC	*American Institute of Steel Construction*
ALARA	*As Low As Reasonably Achievable*
ANS	*American Nuclear Society*
ANSI	*American National Standards Institute*
AOA	*Area(s) of Applicability*
AP	*Aqueous Polishing*
ASCE	*American Society of Civil Engineers*
ASME	*American Society of Mechanical Engineers*
ASTM	*American Society for Testing and Materials*
BDC	*Baseline Design Criteria*
BFP	*Back Flow Preventer*
BOCA	*Building Code by Building Officials and Code Administrators International*
BTP	*Branch Technical Position*
CAAS	*Criticality Accident Alarm System*
CAM	*Continuous Air Monitor*
CAMS	*Continuous Air Monitoring System*
CCTV	*Closed Circuit Television*
CFR	*Code of Federal Regulations*
CM	*Configuration Management*
CSE	*Criticality Safety Evaluation*
D	*Dose*
DAC	*Derived Air Concentration*

DBP	*Dibutyl Phosphate*
DIW	*Deionized Water*
DOE	*Department of Energy*
DWM	*Division of Waste Management*
EA	*Environmental Assessment*
EAL	*Emergency Action Level*
EIS	*Environmental Impact Statement*
ERDA	*Energy Research and Development Administration*
ERPG	*Emergency Response Planning Guidelines*
FCSS	*Fuel Cycle Safety and Safeguards*
FHA	*Fire Hazards Analysis*
FKG	*Formula Kilogram*
FM	*Factory Mutual Research Corporation*
FMEA	*Failure Modes and Effects Analysis*
FNMCP	*Fundamental Nuclear Material Control Plan*
FOCI	*Foreign Ownership, Control, or Influence*
FONSI	*Finding of No Significant Impact*
HEPA	*High Efficiency Particulate Air*
HFE	*Human Factors Engineering*
HS&E	*Health, Safety and the Environment*
HSI	*Human Systems Interface*
HTP	*Hydrogenated Tetrapropylene*
I&C	*Instrumentation and Control*
IBC	*International Building Code by International Code Council*
ICRP	*International Council on Radiation Protection*
ID	*Inventory Difference*
IEEE	*Institute of Electrical and Electronic Engineers*
IROFS	*Items Relied on For Safety*

ISA	*Integrated Safety Analysis*
LEU	*Low Enriched Uranium*
MC&A	*Material Control and Accounting*
MCNP	*Monte Carlo Neutron Proton Code*
MDC	*Minimum Detectable Concentration*
MFT	*Mass Flow Totalizer*
MOX	*Mixed Oxide*
MP	*MOX Process*
M/S	*Mixer/Settler*
NCRP	*National Council on Radiation Protection*
NCS	*Nuclear Criticality Safety*
NDA	*Non-Destructive Assay*
NEPA	*National Environmental Policy Act*
NFPA	*National Fire Protection Association*
NIOSH	*National Institute for Occupational Safety and Health*
NIST	*National Institute of Standards and Technology*
NMSS	*Office of Nuclear Material Safety and Safeguards*
NRC	*Nuclear Regulatory Commission*
NSI	*National Security Information*
NVLAP	*National Voluntary Laboratory Accreditation Program*
OER	*Operating Experience Review*
OSHA	*Occupational Safety and Health Administration*
P^3	*Plutonium Purification Process*
P&IDs	*Piping and Instrumentation Diagrams*
PCFD	*Process Criticality Flow Diagram*
PEC	*Passive Engineered Control*
PFD	*Process Flow Diagram*
PHA	*Process Hazard Analysis*

PM	*Preventive Maintenance*
PPE	*Personnel Protective Equipment*
PSI	*Process Safety Information*
QA	*Quality Assurance*
QC	*Quality Control*
RD	*Restricted Data*
RG	*Regulatory Guide*
RSO	*Radiation Safety Officer*
RWP	*Radiation Work Permits*
SBC	*Southern Building Code by Southern Building Code Congress International Inc.*
SEC	*Securities and Exchange Commission*
SEID	*Standard Errors of Inventory Difference*
SER	*Safety Evaluation Report*
SNM	*Special Nuclear Material*
SRD	*Shipper-Receiver Differences*
SRP	*Standard Review Plan*
SSC	*Structure, System, and Component*
SSNM	*Strategic Special Nuclear Material*
T	*Likelihood Index*
TBP	*Tributyl Phosphate*
TEDE	*Total Effective Dose Equivalent*
TRT	*Tactical Response Team*
UBC	*Uniform Building Code by International Conference of Building Officials*
UL	*Underwriters Laboratories Inc.*
V&V	*Verification and Validation*

GLOSSARY

The following terms are defined here by the staff for the purposes of this Standard Review Plan (SRP). Many terms are taken from 10 CFR 70.4 or other regulations. Terms listed in this glossary represent the definition of the word in any chapter of this SRP. Words for which the definitions change between chapters are listed in the individual chapters.

Accident sequence	An unintended sequence of events that, given the failure of certain items relied on for safety (IROFS) identified in the sequence, would result in environmental contamination, a radiation exposure, a release of radioactive material, an inadvertent nuclear criticality, or an exposure to hazardous chemicals, provided the chemicals are produced from licensed radioactive material. The term "accident" may be used interchangeably with accident sequence.
Active-engineered control	A physical device that uses active sensors, electrical components, or moving parts to maintain safe process conditions and requires no human action.
Acute	This term is defined in 10 CFR 70.4.
Administrative control	Either an augmented-administrative control or a simple-administrative control.
Augmented-administrative control	A required or prohibited human action, combined with a physical device that alerts the operator that the action is needed or prohibited to maintain safe process conditions or that otherwise adds substantial assurance to the required human performance.
Available and reliable to perform their function when needed	This term is defined in 10 CFR 70.4.
Baseline design criteria	A set of criteria specifying design features and management measures that are required and acceptable under certain conditions for new processes or facilities specified in 10 CFR 70.64. These criteria are, in general, the acceptance criteria applicable to safety design described in the chapters of this SRP.
Configuration management (CM)	This term is defined in 10 CFR 70.4.

Consequence	Any result of interest caused by an event or sequence of events. In this context, adverse consequences refers to the adverse health or safety effects on workers or the public, and to adverse environmental impacts of accidents.
Consequence of concern	Adverse radiological, chemical, or environmental effects exceeding any of the levels specified in 10 CFR 70.61.
Construction approval	An approval provided by the NRC to an applicant that allows construction of the principal structures, systems, and components of a plutonium processing and fuel fabrication plant.
Controlled area	This term is defined in 10 CFR 20.1003.
Controlled parameter	A measurable parameter that is maintained within a specified range by one or more specific controls to ensure the safety of an operation.
Critical mass of special nuclear material (SNM)	This term is defined in 10 CFR 70.4.
Design bases	For the purposes of this SRP, this term is defined as in 10 CFR 50.2.
Deviation from safe operating conditions	A parameter outside its established safety limits, or an item relied on for safety that cannot perform its intended function.
Double contingency	This term is defined in 10 CFR 70.4.
Engineered control	Either an active-engineered control or a passive-engineered control.
Event	An occurrence; a change of conditions from a prior state.
External event	An event for which the likelihood cannot be altered by changes to the regulated facility or its operation. This would include all natural phenomena events plus airplane crashes, explosions, toxic releases, fires, etc., occurring near or on the plant site that cannot be controlled by actions of plant personnel.
Hazardous chemicals produced from licensed materials	This term is defined in 10 CFR 70.4.

Integrated safety analysis (ISA)	This term is defined in 10 CFR 70.4.
Integrated safety analysis summary	This term is defined in 10 CFR 70.4.
Items relied on for safety (IROFS)	This term is defined in 10 CFR 70.4.
Management measures	This term is defined in 10 CFR 70.4.
Mitigative control	A control intended to reduce the consequences of an accident sequence, not to prevent it entirely. When a mitigative control works as intended, the results of the sequence are called the mitigated consequences.
Natural phenomena event	Earthquakes, floods, tornadoes, tsunamis, hurricanes, and other events that occur in the natural environment and could adversely affect safety. Natural phenomena events, depending on their likelihood of occurrence, may be credible or incredible.
New processes at existing facilities	Systems-level or facility-level design changes to process equipment, process technology, facility layout, or types of licensed material possessed or used. This definition does not, generally, include component-level design changes or equipment replacement.
Passive-engineered control	A device that uses only fixed physical design features to maintain safe process conditions, and requires no human action.
Preventive control	A control intended to prevent an accident entirely, that is, to prevent any of the types of radiological or chemical consequences in 10 CFR 70.61 of any magnitude.
Principal structures, systems, and components (SSCs)	Safety controls that are identified in the design bases as providing protection against the consequences of accidents or natural phenomena. Designating a control as a principal SSC is effectively synonymous with designating that control as an IROFS.

Process hazard analysis (PHA)	That activity, and its product, that evaluates the identified hazards of operating the plant processes; describes potential accident sequences, including the items relied on to prevent or mitigate the progress of such sequences; and evaluates the likelihood and consequences of the sequences. This activity of necessity involves the determination of the likelihood of the initiating event and the likelihood of failure of the individual items (controls) relied on for safety, and, where more than one item is relied on in a sequence, the likelihood of various combinations of failures that lead to the assessment of the overall likelihood of arriving at the accident consequence.
Process safety information	Information pertaining to (1) the hazards of the material used or produced in the process, (2) the technology of the process, and (3) the equipment in the process.
Safe process conditions	The defined ranges or sets of acceptable values of one or more controlled parameters.
Safety control	A system, device, or procedure intended to regulate a device, process, or human activity to maintain a safe state. Controls may be engineered controls or administrative (procedural) controls. Controls may be preventive or mitigative. Effectively synonymous with "item relied on for safety." In the context of this SRP, use of the unmodified term "control" normally means safety control.
Simple-administrative controls	A human action that is prohibited or required to maintain safe process conditions.
Unacceptable performance deficiencies	This term is defined in 10 CFR 70.4.
Uncontrolled outcome	The sequence of events and consequences that result if no controls or barriers are available to prevent or mitigate an accident sequence. Thus the consequences of an uncontrolled outcome are, by definition, unmitigated. These consequences may also be referred to as uncontrolled consequences.
Unmitigated consequences	The consequences that result from an accident sequence when mitigative control fails or does not exist.
Worker	This term is defined in 10 CFR 70.4.

INTRODUCTION

The "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility" provides the U.S. Nuclear Regulatory Commission (NRC) with guidance for the review and evaluation of the health, safety, and environmental protection for a license application to possess and use special nuclear material (SNM) to fabricate MOX fuel under 10 CFR Part 70. The NRC developed this Standard Review Plan (SRP) in parallel with NUREG–1520 ("Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility"). This SRP is consistent with the guidance in NUREG–1520, yet contains modifications to make this guidance facility specific. The NRC only intends to use this guidance to review an application from the consortium of Duke Cogema Stone and Webster, which is under contract with the U.S. Department of Energy to construct and operate a MOX fuel fabrication facility at the Savannah River Site in Aiken, SC.

The NRC considers the MOX fuel fabrication facility to be a plutonium processing and fuel fabrication plant as defined in 10 CFR 70.4. Since 10 CFR Part 70 requires that the NRC give the applicant construction approval as part of licensing plutonium processing facilities, this SRP provides guidance on the construction approval review in addition to the review for a license to possess and use SNM. This SRP is further applicable to the review and evaluation of proposed amendments and license renewal applications for a MOX facility. Specific filing requirements for the construction approval, the possession and use license, and the issuance of such approvals are in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

Purpose

The principal purpose of the SRP is to ensure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope, level of detail, and acceptance criteria of reviews. This SRP should be used as the basis for the license review for a MOX fuel fabrication facility, including both the construction approval review and the review for a license to possess and use SNM. Moreover, although the SRP uses the term "applicant," this SRP is also intended to apply to license renewals and amendments.

Another important purpose of the SRP is to make information about regulatory reviews related to the MOX fuel fabrication facility widely available to improve communication and understanding of the staff review process. Because the SRP describes the scope, level of detail, and acceptance criteria for reviewers, it can serve as regulatory guidance for applicants who need to determine what information should be presented in a license application for a MOX fuel fabrication facility, including the portion of the application that provides material for the NRC's construction approval review.

The staff's responsibility in the review of a new license application (including the construction approval), license renewal application, or license amendment for a MOX fuel fabrication facility is to determine that there is reasonable assurance that: the design bases of the principal structures, systems, and components (SSCs) and quality assurance program provide reasonable assurance of protection against the consequences of potential accidents and natural phenomena (construction approval); and the facility can be operated in a manner that will not be inimical to the common defense and security and will provide reasonable protection

of the health and safety of workers, the public, and the environment including that the facility was constructed consistent with the application (license to possess and use SNM). To carry out this responsibility, the staff evaluates information provided by the applicant and, through independent assessments, determines that the applicant has demonstrated a reasonable design bases (for construction approval) and a reasonable safety program (for issuing a license to possess and use SNM) that are in accordance with regulatory requirements. To facilitate carrying out this responsibility, the SRP clearly states and identifies those standards, criteria, and bases that the staff should use in reaching regulatory decisions.

This SRP provides information to assist the staff (and applicant) in understanding the underlying objective of the regulatory requirements, the relationships among NRC requirements, the licensing process, the major guidance documents that the NRC staff has prepared for licensing facilities under 10 CFR Part 70, and the details of the staff review process set out in individual SRP sections. Analyses by the staff are intended to provide regulatory confirmation of reasonable assurance of safe design and operation. A staff determination of reasonable assurance leads to a decision to provide a construction approval, issue or renew a license, or approve an amendment. In the case of a staff determination of inadequate description or commitments, the staff should inform the applicant of what is needed and the basis upon which the determination was made.

Construction Approval

Prior to constructing a plutonium processing facility such as the MOX fuel fabrication facility, the applicant must obtain the NRC's approval. The material the applicant submits to support the NRC's construction approval review is part of the license application. The NRC does not require the applicant to submit a full license application to make a determination regarding the construction approval. Applicants must submit a description of the facility site; a description and safety assessment of the design bases of the principal structures, systems, and components (SSCs) of the facility, including provisions for protection against natural phenomena; and a description of the quality assurance program to be applied to the design, fabrication, construction, testing, and operation of the facility's SSCs. For the purposes of this guidance, the NRC is defining "design bases" as the information that identifies the specific functions to be performed by an SSC of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

The safety assessment of the design bases should explain why the applicant selected particular functions or values and demonstrate how the applicant determined that the design bases will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. Accident consequences are defined in the performance requirements of Subpart H to 10 CFR Part 70. In addition, the safety assessment should demonstrate how the requirements for new facilities identified in 10 CFR 70.64 are satisfied by the design bases. In effect, the safety assessment of the design bases should show that the design bases bounds, or at least meets, the acceptance criteria outlined in this SRP.

Prior to applying for a construction approval, the applicant should have designed and analyzed the facility in sufficient detail to allow the NRC to make a determination in accordance with 10 CFR 70.23(b). To allow this determination, the material submitted to obtain the NRC's construction approval should contain the information described in 10 CFR 70.22(f) in sufficient detail for the staff to review the safety assessment of the design bases.

Approval for a License To Possess and Use SNM

Part 70.65 requires that the applicant submit a Safety Program Description with the license application to possess and use SNM. The Safety Program Description must be sufficiently detailed to permit the staff to conclude that the design was completed and the facility constructed in accordance with the approved design bases and to obtain reasonable assurance that the facility will be operated without undue risk to the health and safety of workers or the public, i.e., meet the performance requirements of 10 CFR 70.61. To be acceptable, the license application, and therefore the Safety Program Description, should meet the acceptance criteria of this SRP.

The Safety Program Description is the principal document through which the applicant provides the information needed by the NRC staff to make a determination on the license application. When reviewed and approved by the staff, and incorporated in the NRC license by reference, the Safety Program Description, in its entirety and in its parts, is considered a binding commitment of the applicant regarding the design and operation of the licensed facility. The Safety Program Description is the safety basis on which the license is issued and may not be changed except under circumstances defined in 10 CFR 70.72.

Using the SRP

The requirements in 10 CFR Part 70 specify, in general terms, the information to be supplied in the license application, including the construction approval request. The specific information that should be submitted by the applicant and evaluated by staff is identified in this SRP. Prospective applicants should study the topic areas treated in this document (generally, chapter headings) and the subsections within each topic area, specifically the subsections titled "Areas of Review," "Acceptance Criteria," and "Review Procedures." The license application should contain a Safety Program Description that addresses all topics in the Table of Contents in the SRP. Staff should refer to each SRP chapter for specific guidance on how that topic should be addressed for the construction approval. In each case, the material should be structured in the same order as presented in this document.

The major topics addressed within the design bases (construction approval) or the Safety Program Description of a facility (possession and use) of a license application are addressed in separate SRP sections; each of those sections, or chapters, includes subsections described below.

Section 1. PURPOSE OF REVIEW

This section is a brief statement of the purpose for and objectives of reviewing the subject areas. It emphasizes the staff's evaluation of the ways the applicant can achieve identified performance objectives and ensures through the review that the applicant has used a

multidisciplinary, risk-informed, systems-oriented approach to establishing designs, controls, and procedures within individual technical areas.

Section 2. RESPONSIBILITY FOR REVIEW

This section identifies the organization and individuals by function, within the NRC, responsible for evaluating the subject or functional area covered by the SRP. If reviewers with expertise in other areas are to participate in the evaluation, they are identified by function. In general, the Project Manager has responsibility for the review product, a Safety Evaluation Report including safeguards and supporting environmental evaluations for an application. However, an identified Technical Specialist should have primary responsibility for a particular review topic, usually an SRP chapter. One or more specialists may have supporting responsibility. In some areas, the review is performed by a team of specialist reviewers, including the lead reviewer for the Integrated Safety Analysis (ISA) and the Project Manager. Although they perform their review tasks individually, the reviews are coordinated and integrated to ensure consistency in approach and risk-informed reviews. The Project Manager oversees and directs the coordination of the reviewers. The reviewers' immediate line management has the responsibility to ensure that an adequate review is performed by qualified reviewers.

Section 3. AREAS OF REVIEW

This section describes the topics, functions, systems, structures, equipment, components, analyses, data, or other information that should be reviewed as part of that particular subject area of the license application. Because the section identifies information to be reviewed in evaluating the construction approval as well as the license to possess and use SNM, it identifies the acceptable content of the license application in the areas discussed. If there is a distinction between the areas of review for the construction approval or the license to possess and use SNM, it is explicitly noted in each subject area. The areas of review identified in this section obviate the need for a separate Standard Format and Content Guide.

Topics identified in this section also set the content of the next two sections of the SRP. Both Section 4, "Acceptance Criteria," and Section 5, "Review Procedures," should address, in the same order, the topics set forth in Section 3 as areas to be reviewed. Section 3 also identifies the information needed or the review expected from other NRC individuals to permit the individual charged with primary review responsibility to complete the review.

Section 4. ACCEPTANCE CRITERIA

This section contains a statement of the applicable NRC criteria based on regulatory requirements, and the bases for determining the acceptability of the applicant's commitments relative to the design, programs, or functions within the scope of the particular SRP section. Technical bases consist of specific criteria such as NRC regulations, Regulatory Guides, NUREG reports, industry codes and standards, and Branch Technical Positions. To the extent practicable, the acceptance criteria identify, as objectively or quantitatively as is feasible, that specific criteria, and other technical bases must be bounded by the design bases or met by either the design bases (construction approval) or the Safety Program Description (license to possess and use SNM). The acceptance criteria (including Branch Technical Positions or other information) present positions and approaches that are acceptable to the staff.

The NRC's intent is to have the SRP present acceptance criteria for each technical function area (e.g., nuclear criticality safety, fire safety, and radiation safety) and for the management measures (e.g., quality assurance, maintenance, audits, and assessments) that allow the applicant to provide a level of protection commensurate with the accident risk inherent in the process activities proposed. For example, at process stations (or for an entire process or subprocess) for which the inherent risk to workers, the public, or the environment is demonstrably small, the applicant needs to provide only those design and operating controls that assure that small risk. The key element in the regulatory transaction involving presentation by the applicant, and review and approval by the NRC, is an adequate demonstration of acceptable control of risk by the applicant, which then supports a competent and informed review by NRC staff. The starting point for the applicant's demonstration of acceptable control of risk is the safety assessment of the design bases for the construction approval as followed by the ISA for the license to possess and use SNM.

The applicant's safety assessment of the design bases and ISA Summary (described in and reviewed in Chapter 5.0 of this SRP) are the primary supporting rationale for the safety level of design and operational features. There are, however, design and operational features and management measures that may be required independent of the ISA results presented by the applicant. This is to meet the requirements of 10 CFR 70.64 for new facilities or new processes at existing facilities or, for all facilities, other NRC requirements such as 10 CFR Parts 20 and 51. The level of detail presented in the ISA Summary and in other parts of the application represents the safety basis committed to by the applicant. That basis is subject to the provisions of 10 CFR Part 70 regarding changes that a licensee may make to the facility without prior NRC approval.

If the MOX facility is licensed and the licensee renews or amends the license, in responding to the requirements of 10 CFR Part 70, the licensee may propose items relied on for safety (IROFS) or supporting management measures that meet less stringent acceptance criteria than described in the SRP based on supporting analyses from the ISA. The ISA may be used to justify a reduced level of assurance for particular IROFS that are associated with lesser risk accident sequences, as defined by the applicant's analysis of likelihood and consequences pursuant to 10 CFR Part 70. The SRP criteria shown in this SRP apply to those IROFS and associated management measures that are involved in the higher risk accident sequences as defined in 10 CFR 70.61.

For construction approval of the MOX fuel fabrication facility, the acceptance criteria described in the SRP should be bounded by the applicant's safety assessment design bases. There is an additional requirement to comply with the baseline design criteria (BDC) of 10 CFR 70.64. The BDC are consistent with risk-informed regulation, in that, for new processes or new facilities, the NRC recognizes that good engineering practice dictates that certain minimum requirements be applied as design and safety considerations, generally independent of the risk-based information ultimately obtained through the ISA. However, the applicant may later use the license application to justify reduced criteria for some IROFS consistent with the ISA Summary for the final facility design. Proposed reductions in the level of assurance should be considered by the NRC staff and, if accepted, should also constitute compliance with the BDC.

The "Acceptance Criteria" are intended to communicate the underlying objectives but not to represent the only means of satisfying that objective. The applicant should tailor its safety program to the features of its particular facility. If approaches different from the SRP are

chosen, the applicant should identify the portions of its application that differ from the design approaches and acceptance criteria of the SRP and evaluate how the proposed alternatives provide an acceptable method of complying with the Commission's regulations. The staff retains the responsibility to make an independent determination of the adequacy of what is proposed.

The applicant should recognize that substantial time and effort on the part of the staff have gone into the development of the acceptance criteria and may be required to review and accept proposals that depart from the standard application described in the SRP. Thus, applicants resolving safety issues or safety-related design areas in ways other than those described in the SRP should plan for longer review times and more extensive questioning in these areas.

Section 5. REVIEW PROCEDURES

This section describes how the review should be performed and delineates differences between the construction approval review and the review for the license to possess and use SNM. It describes procedures that the reviewer should follow to achieve an acceptable scope and depth of review and to obtain reasonable assurance that the applicant has provided appropriate commitments to ensure that it will construct or operate the facility safely and securely. This includes identifying commitments the reviewer should verify and could include directing the reviewer to coordinate with others having review responsibilities for other portions of the application than those assigned to the reviewer. This section should provide whatever procedural guidance is necessary to evaluate the applicant's level of achievement of the acceptance criteria for the construction approval, the license, and license amendments.

Section 6. EVALUATION FINDINGS

This section presents the type of positive conclusion that is sought for the particular review area to support a decision to grant the construction approval or license. The review must be adequate to permit the reviewer to support this conclusion. For each section, a conclusion of this type should be included in the staff's Safety Evaluation Report (SER) in which the staff publishes the results of its review. The SER should also contain a description of the review, including aspects of the review that received special emphasis; matters that were modified by the applicant during the review; matters that require additional information or will be resolved in the future; aspects where the facility's design or the applicant's proposals deviate from the criteria in the SRP; and the bases for any deviations from the SRP or proposed exemptions from the regulations. Staff reviews may be documented in the form of draft SERs that identify open issues requiring resolution before the staff can make a positive finding in favor of the license issuance or amendment.

Section 7. REFERENCES

This section lists references that should be consulted in the review process. However, the references may not always be relevant to the review, depending on the action and approaches proposed by the applicant.

1.0 GENERAL INFORMATION
1.1 FACILITY AND PROCESS OVERVIEW

1.1.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the applicant provides a facility and process overview that describes the purpose of the facility. The facility and process overview should also familiarize reviewers, NRC management, or the public with the facility and process. The overview should be abstracted from, and therefore consistent with, material presented in the applicant's design bases (for the construction approval) or Safety Program Description and Integrated Safety Analysis (ISA) Summary (for the license to possess and use special nuclear material [SNM]), the environmental report, and the emergency plan.

1.1.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: ISA Reviewer, Environmental Reviewer, Emergency Protection Reviewer

Supporting: None

1.1.3 AREAS OF REVIEW

The facility and process overview should be submitted for the NRC's construction approval review and updated in the material submitted for the license to possess and use SNM. The areas of review for the overview should include:

A. The overall facility layout on scaled drawings. The following types of features should be identified:

   i. The location of facility buildings such as plant structures, buildings, towers, and tanks and other major manmade or geographical features;

   ii. Transportation right of ways;

   iii. Major ingress and egress routes for the site, including public access, if applicable; and

   iv. The controlled area, restricted area, or other boundaries proposed by the applicant, as appropriate.

B. The movement of personnel, materials, and equipment during facility operations.

C. A description of the major chemical or mechanical processes involving SNM, including:

i. The chemical and physical forms of SNM in the processes;
ii. The maximum amounts of SNM in the processes;
iii. The building locations of major components in the processes;
iv. A description of the process steps; and
v. Types, amounts, and discharge points of wastes discharged to the environment.

D. A text index with titles that describes all features identified in the scaled drawings.

1.1.4 ACCEPTANCE CRITERIA

1.1.4.1 Regulatory Requirements

The regulatory requirements for facility and process overview are 10 CFR 70.22, "Contents of Applications," and 10 CFR 70.65, "Additional Contents of Applications."

1.1.4.2 Regulatory Guidance

None.

1.1.4.3 Regulatory Acceptance Criteria

The reviewers should find the overview acceptable if:

A. The level of detail in the overview is appropriate for general familiarization with the facility and process, is appropriate for the level of design, and conveys the purpose of the facility.

B. The overview appropriately cross-references the material provided in support of Chapters 5.0, 8.0, and 14.0 of this SRP.

C. The overview is consistent with, yet less detailed than, the information provided in the application in support of Chapters 5.0, 8.0, and 14.0 of this SRP.

D. The applicant commits to updating the overview to reflect the completed design in the license application.

1.1.5 REVIEW PROCEDURES

1.1.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the specific items in Section 1.1.3, "Areas of Review." If the primary reviewer verifies that the facility and process overview is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 1.1.5.2. If the

primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

1.1.5.2 Safety Evaluation

After determining that the facility and process overview is acceptable for review in accordance with Section 1.1.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 1.1.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

The primary reviewer should consider the facility and process overview as background for the detailed descriptions provided in support of the application. Therefore, the primary reviewer should not perform a detailed technical analysis. However, the primary reviewer should coordinate with the supporting reviewers to ensure that the material presented here is consistent with material presented in support of other chapters of this SRP.

When the applicant updates the facility and process overview for the license to possess and use SNM, the primary reviewer should focus the review on any new or changed material. The primary reviewer should also confirm that the material presented in the facility and process overview remains consistent with the material provided in the license application in support of other chapters of this SRP.

1.1.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the appropriate Safety Evaluation Report (SER). The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

> *The staff reviewed the facility and process overview for approval to construct [insert name of facility] according to Section 1.1 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and found that [state the findings].*
>
> *The staff concluded that the (1) the level of detail in the facility and process overview provided an adequate understanding of the facility and process and conveyed the purpose of the facility, (2) the facility and process overview appropriately cross-referenced material presented in later sections of the application, and (3) the facility and process overview is consistent with, yet less detailed than, material in later sections of the application. As a result, the staff finds that the applicant meets the regulatory requirements for the facility and process overview to allow construction approval for the [insert name of facility].*

The staff could document the safety evaluation for the review for a license to possess and use SNM review as follows:

The staff reviewed the facility and process overview for a license application to possess and use SNM at [insert name of facility] according to Section 1.1 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] focusing on new or changed material when compared to the safety evaluation for the construction approval for [insert name of facility]. The staff found that [state the findings].

The staff concluded that (1) the level of detail in the facility and process overview provided an adequate understanding of the facility and process and conveyed the purpose of the facility, (2) the facility and process overview appropriately cross-referenced material presented in later sections of the license application, and (3) the facility and process overview is consistent with, yet less detailed than, material in later sections of the application. As a result, the staff finds that the application meets the regulatory requirements for the facility and process overview for a license to possess and use SNM.

1.1.7 REFERENCES

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

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

1.0 GENERAL INFORMATION
1.2 INSTITUTIONAL INFORMATION

1.2.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the applicant meets the requirements of 10 CFR Part 70 related to the ownership, the planned activities, and the nuclear material to be handled in connection with the requested license. The applicant's financial qualifications and facility security clearance to possess classified material are addressed in Chapters 2.0 and 3.0 of this SRP, respectively.

1.2.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: Primary Reviewer of Chapter 2.0, "Financial Qualifications"

Supporting: None

1.2.3 AREAS OF REVIEW

The applicant should submit the institutional information with the material submitted to support the construction approval review. The areas of review for the applicant's institutional information should include:

A. The corporate identity, including:

i. The applicant's full name and address, the State in which the applicant is incorporated or organized, or the location of the principal address;

ii. The address of the fuel cycle facility, if different from the corporate address, including the full description of the location (State, county, and municipality) as documented in the legal records;

iii. The name, address, and citizenship of each of the principal corporate officers;

iv. Parent or other affiliated companies;

v. Any foreign ownership or control of activities by any alien, foreign cooperation, or foreign government; and

vi. The presence and operations of any other companies on the site.

B. The type of license, period of the license, and the type, quantity, and form of licensed material the applicant proposes to acquire, deliver, receive, possess, produce, use, transfer or store, including:

   i. The elemental name, maximum quantity, and specifications, including chemical and physical form(s), of the SNM and strategic SNM; and

   ii. The isotopic content and weight percent enrichment of the SNM and strategic SNM.

C. The proposed authorized uses for the SNM or strategic SNM including a description of each activity or process in which the SNM or strategic SNM is acquired, delivered, received, possessed, produced, used, processed, transferred, or stored.

D. Specific requests for special exemptions or special authorizations that are listed and cross-referenced to a justification in the appropriate technical section of the application.

1.2.4 ACCEPTANCE CRITERIA

1.2.4.1 Regulatory Requirements

The regulations applicable to institutional information are found in 10 CFR 70.22(a)(1), (2), (3), and (4).

1.2.4.2 Regulatory Guidance

None.

1.2.4.3 Regulatory Acceptance Criteria

The reviewers should find the institutional information acceptable if the following criteria are met:

A. The information provided by the applicant pertaining to the corporate identity is complete and accurate.

B. The information provided by the applicant pertaining to the type, quantity, and form of licensed material is complete and accurate. The type, quantity, and form are consistent with the proposed activities.

C. The applicant's proposed activities and processes are consistent with the Atomic Energy Act of 1954, et seq., and the more detailed material submitted in support of Chapter 5.0 of this SRP.

D. The lists of special exemptions and special authorizations are complete and accurate.

E. The applicant commits to updating the institutional information in the license application.

1.2.5 REVIEW PROCEDURES

1.2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the applicant's institutional information adequately addresses the specific items in Section 1.2.3, "Areas of Review." If the primary reviewer verifies that institutional information is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 1.2.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

1.2.5.2 Safety Evaluation

After determining that the institutional information is acceptable for review in accordance with Section 1.2.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 1.2.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

The primary reviewer should not perform a detailed technical analysis of the material unless the applicant identifies foreign ownership, control, or influence (FOCI). The NRC will accept FOCI determinations made by other Government agencies (e.g., the U.S. Department of Energy or the U.S. Department of Defense), as described in Memorandums of Understanding between both agencies, e.g., NRC and DOE. The primary reviewer should coordinate verification of an existing FOCI determination or other FOCI activities (as appropriate) with the Division of Facilities and Security.

The primary reviewer should coordinate with the secondary reviewer so that the institutional information provided for this section may support the financial qualifications review performed under Chapter 2.0 of this SRP.

When the applicant updates the institutional information for the license application, the primary reviewer should limit the review to any new or changed material.

1.2.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

> *The staff reviewed the institutional information for approval to construct [insert name of facility] according to Section 1.2 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed, including a tabulated listing of the proposed material, form, quantity, and authorized use] and found that [state the findings].*
>
> *Based on the review, the staff concluded that the applicant provided sufficient information to support the requirements in 10 CFR Part 70 for the construction approval for [insert name of facility].*

The staff could document the safety evaluation for the review for a license to possess and use SNM as follows:

> *The staff reviewed the institutional information for [insert name of facility] according to Section 1.2 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed, including a tabulated listing of the proposed material, form, quantity, and authorized use] focusing on the new or changed material when compared to the safety evaluation for the construction approval review for [insert name of facility]. The staff found that [state the findings].*
>
> *Based on the review, the staff concluded that the applicant meets the regulatory requirements in 10 CFR Part 70 for ownership, location, planned activities, and nuclear material to be handled in connection with the license application to possess and use SNM for [insert name of facility].*

1.2.7 REFERENCES

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

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

1.0 GENERAL INFORMATION
1.3 SITE DESCRIPTION

1.3.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the information provided by the applicant adequately describes the geographic, demographic, meteorologic, hydrologic, geologic, and seismologic characteristics of the site and surrounding area. The site description should be abstracted from, and therefore consistent with, material presented in the applicant's design bases (construction approval) or Safety Program Description and ISA Summary (license application), the environmental report, and the emergency plan.

1.3.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: ISA Reviewer, Emergency Protection Reviewer, Environmental Reviewer

Supporting: None

1.3.3 AREAS OF REVIEW

The site description should be submitted for the NRC's construction approval review and updated in the application for a license to possess and use SNM. The areas of review for the applicant's site description should include:

A. Site Geography

i. Site location: State, county, municipality, topographic quadrangle (7½-minute series), longitude, and latitude;

ii. Public roads;

iii. Nearby bodies of water; and

iv. Any other significant geographic feature that may impact an accident consequence within 2 km (1.24 miles).

B. Demographics (including socioeconomics) and Land Use

i. Latest census results for the area of concern, including minority and low-income populations;

ii. Description, distance, and direction to nearby population centers;

iii. Description, distance, and direction to nearby public facilities (e.g., schools, hospital, parks);

iv. Description, distance, and direction to nearby industrial areas or facilities that may present potential hazards (including other nearby nuclear facilities);

v. Land use within 2 km (1.24 miles) of the facility (i.e., residential, industrial, commercial, agricultural); and

vi. Uses of nearby bodies of water.

C. Meteorology

i. Local wind directions and average and maximum wind speeds;

ii. Annual amount and forms of precipitation;

iii. The design basis values for analyzing the maximum snow or ice load and probable maximum precipitation; and

iv. Type, frequency, and magnitude of severe weather (e.g., lightning, tornado, hurricane).

D. Hydrology

i. Characteristics of nearby rivers, streams, and other bodies of water, as appropriate;
ii. Depth to the water table;
iii. Potentiometric surface map;
iv. Groundwater flow direction and velocity for the site;
v. Characteristics of the uppermost aquifer; and
vi. Design basis flood events used for accident analysis.

E. Geology

i. Characteristics of soil types and bedrock;
ii. Design basis earthquake magnitudes used for accident analysis; and
iii. Description of other geologic hazards, such as mass wastings.

1.3.4 ACCEPTANCE CRITERIA

1.3.4.1 Regulatory Requirements

Regulations applicable to the site description are contained in 10 CFR 70.22(f), which requires a description of the plant site for applications for special nuclear material in a plutonium processing and fuel fabrication plant.

1.3.4.2 Regulatory Guidance

None.

1.3.4.3 Regulatory Acceptance Criteria

The reviewer should find the applicant's site description, including the site geography, demographics (including socioeconomic data), meteorology, hydrology, and geology, acceptable if the following regulatory acceptance criteria are met:

A. Information is current and accurate. To the extent possible, data reflect observations and measurements made over a period of years, especially for conditions that are expected to vary seasonally (e.g., precipitations, wind speed and direction, and groundwater levels).

B. Data sources are appropriately referenced and documented.

C. Information is consistent with the more detailed material submitted by the applicant in the design basis (construction approval) or the Safety Program Description and ISA Summary (license application), environmental report, and emergency plan.

D. The applicant commits to updating the site description in the license application.

1.3.5 REVIEW PROCEDURES

1.3.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the site description addresses the specific items in Section 1.3.3, "Areas of Review." If the primary reviewer verifies that the site description is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 1.3.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

1.3.5.2 Safety Evaluation

After determining that the site description is acceptable for review in accordance with Section 1.3.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 1.3.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

The primary reviewer should not perform a detailed technical analysis of the material since this material is considered background for the more detailed material submitted elsewhere in the application. However, the primary reviewer should coordinate with the secondary reviewers to ensure that the site description adequately summarizes material presented in support of the ISA Summary, the emergency plan, and the environmental report.

When the applicant updates the site description in the license application, the primary reviewer should review the new or changed information. The primary reviewer should also verify with the secondary reviewers that the updated site description in the license application remains consistent with material that supports other chapters of this SRP.

1.3.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation of the construction approval review as follows:

> *The staff reviewed the site description for approval to construct [insert name of facility] according to Section 1.3 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and found that [state the findings].*
>
> *Based on the review, the staff concluded that the applicant's site description meets the regulatory requirements in 10 CFR 70.22(f) for construction approval.*

The staff could document the safety evaluation for the review for the license to possess and use SNM as follows:

> *The staff reviewed the site description for [insert name of facility] according to Section 1.3 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and focused on new or changed information when compared to the safety evaluation for the construction approval review. The staff found that [state the findings].*
>
> *Based on the review, the staff concluded that the applicant's site description meets the regulatory requirements in 10 CFR 70.22 for a license to possess and use SNM.*

1.3.7 REFERENCES

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

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

2.0 FINANCIAL QUALIFICATIONS

2.1 PURPOSE OF REVIEW

The purpose of this review is to determine that the applicant appears to be financially qualified to engage in the proposed activities in accordance with 10 CFR Part 70. The scope of this review does not include the applicant's financial qualifications for decommissioning if responsibility for eventual decommissioning resides with the U.S. Department of Energy (DOE).

2.2 RESPONSIBILITY FOR REVIEW

Primary: Financial Specialist

Secondary: Project Manager

Supporting: None

2.3 AREAS OF REVIEW

The financial qualifications should be submitted with the material for the construction approval review and updated in the material submitted for the license to possess and use special nuclear material (SNM). Although construction funding is expected to be provided by DOE, financial qualifications information is needed for the construction approval to ensure that the applicant will construct the facility properly with adequate funding provided for engineering, design, materials, and quality assurance. To support the construction approval review, the applicant needs to adequately demonstrate that it has planned for the necessary funds, considers alternative sources of funding, and considers any contingencies such as delays in government funding, other shortfalls, and cost overruns. To support the review for a license to possess and use SNM, the applicant needs to provide information on projected costs, revenue sources, and contingencies. The applicant needs to demonstrate that it has sufficient financial strength and revenue sources to properly operate the facility. The areas of review for financial qualifications should include:

A. Project Costs

   i. Engineering, design, and construction costs for the full planned capacity of the facility.

   ii. If construction will be staged, incremental estimates for each stage of facility construction.

   iii. The total project cost, including interest, escalation, and financing in addition to the engineering, design, quality assurance, and construction costs.

iv. Projected operating costs.

B. Sources of Funds

i. Estimates of the total and incremental debt, equity, and revenues (if any) for each phase of the project, such as construction and operation.

ii. Funding plans for the proposed action, including, but not limited to, the debt equity and revenues.

iii. The source(s) and planned or existing funding commitments or contracts upon which the applicant relies, including Government contracts.

C. Contingency Funds

The contingencies for cost overruns and revenue shortfalls during construction and operation.

D. Financial Qualifications

i. The financial description of the applicant, of any partnership established to finance the proposed action, and of any parent or other affiliated companies upon whom the applicant is relying for sources of construction or operating funds.

ii. The most recent financial report and U.S. Securities and Exchange Commission (SEC) Report 10–K, for itself, any planned or existing partners, and any parent or other affiliated companies upon whom the applicant is relying for the sources of construction funds. If an annual financial report and U.S. Securities and Exchange Commission Report 10–K is not available for the applicant, a partner, or other affiliated company, the applicant should provide audited financial statements that include:

a. Statements of earning to include revenues, costs and expenses, earnings before and after taxes, net earnings, and per-share earnings and dividends;

b. Consolidated statements of changes in shareowners' equity;

c. Statements of financial position to include assets, liabilities, and equity;

d. Statements of cash flows, including cash flows from operating, investing, and financing activities;

e. Management's discussion of financial operations, resources, liquidity, and significant selected financial data;

f. Any notes applicable to the financial statements needed to clarify or explain significant items, assumptions, potential risks and liabilities, or limitations; and

g. An independent auditor's report describing the accounting principles used and any opinions or qualifications applicable to the financial statements.

E. Liability Insurance

The applicant should provide a description, the amounts, and the issuers of the public liability insurance to be provided for the proposed activities as required for Price-Anderson coverage under 10 CFR 140.13a. Under this provision, an applicant for a license to possess and use plutonium would need to provide $200,000,000 in primary coverage. The coverage would not be needed for a MOX facility construction approval. The NRC would indemnify the licensee above $200,000,000 to a maximum of $560,000,000. An indemnification fee of $5000/year would be required. The applicant should provide an Indemnification Agreement in the form of Part 140, Appendix H.

2.4 ACCEPTANCE CRITERIA

2.4.1 Regulatory Requirements

Regulatory requirements for financial qualifications are found in 10 CFR 70.23(a)(5) and the note in 10 CFR 70.22(a), which reads, "NOTE: Where the nature of the proposed activities is such as to require consideration of the applicant's financial qualifications to engage in the proposed activities in accordance with the regulations in this chapter, the Commission may request the applicant to submit information with respect to his financial qualifications."

2.4.2 Regulatory Guidance

None.

2.4.3 Regulatory Acceptance Criteria

The reviewer should find the applicant's financial qualifications acceptable if the following acceptance criteria are met:

A. Project Costs

The applicant's engineering, design, and construction costs, staged project costs, and total project costs are appropriate for the size and scope of the proposed actions.

B. Sources of Funds

The applicant's sources of funds (including the applicant's funding plan(s) and debt, equity and revenue levels (if any) for each stage of the project) and planned or existing

source(s) of funding commitments are consistent with the estimated construction costs of the proposed action.

C. Contingency Funds

The applicant's contingency funds are appropriate for unforeseen construction and operating contingencies. The applicant indicates its plans for the case where cost overruns are much higher than anticipated, e.g., in excess by 30%.

D. Financial Qualifications

The financial data for the applicant, planned or existing partners, or other affiliated companies support the financial commitments of each; are consistent with generally accepted accounting practices; and represent a reasonable financial basis for constructing and operating the facility.

The applicant commits to providing its annual report to the NRC. If the applicant does not issue an annual report, the applicant commits to annually provide the NRC with the information described in Section 2.3(D)(ii)(a) through (g).

E. Liability Insurance

Public liability insurance is sufficient to cover reasonable expected onsite accidents and obligations as required under 10 CFR 140.13a. The applicant commits to maintaining public nuclear liability insurance in the maximum commercially available amount, unless the applicant shows that such liability will be borne by the DOE. If the applicant intends to use liability provisions in place by the DOE, the applicant should describe the intended arrangements.

2.5 REVIEW PROCEDURES

2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the financial qualifications adequately address the specific items in Section 2.3, "Areas of Review." If the primary reviewer verifies that financial qualifications are adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 2.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

2.5.2 Safety Evaluation

After determining that the financial qualifications are acceptable for review in accordance with Section 2.5.1, the primary reviewer should perform a safety evaluation for the construction approval review against the acceptance criteria described in Section 2.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria. The primary reviewer should coordinate with the secondary reviewer to ensure consistency between this chapter and the applicant's material supporting Section 1.2, "Institutional Information."

The primary reviewer should verify that the applicant's updated financial qualifications, when submitted with the full license application, remain consistent with the material submitted for construction approval and continue to meet the acceptance criteria in Section 2.4.

2.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the Safety Evaluation Report (SER). The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

> *The staff reviewed the financial qualifications for construction approval for [insert name of facility] according to Chapter 2.0 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and found that [state the findings].*
>
> *The staff concluded that the applicant appears financially qualified to engage in the proposed activities in accordance with 10 CFR Part 70. As a result, the staff finds that the applicant's financial qualifications support the staff's approval of construction.*

The staff could document the safety evaluation for the review for the license to possess and use SNM as follows:

> *The staff reviewed the financial qualifications for [insert name of facility] according to Chapter 2.0 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and focused on new or updated material when compared to the safety evaluation for construction approval. The staff found that [state the findings].*
>
> *The staff concluded that the applicant appears financially qualified to engage in the proposed activities in accordance with 10 CFR Part 70. As a result, the staff finds that the applicant's financial qualifications meet the regulatory requirements for issuing a license to possess and use SNM.*

2.7 REFERENCES

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

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

3.0 PROTECTION OF CLASSIFIED MATTER

3.1 PURPOSE OF REVIEW

The purpose of this review is to confirm that the applicant has established procedures for safeguarding SECRET and CONFIDENTIAL National Security Information (NSI) and Restricted Data (RD) received or developed in conjunction with activities licensed, certified, or regulated by the Commission.

3.2 RESPONSIBILITY FOR REVIEW

Primary: Classified Matter Specialist

Secondary: Project Manager

Supporting: None

3.3 AREAS OF REVIEW

The applicant's standard practice procedures plan for the protection of classified matter (Plan) should be submitted before the applicant begins safeguarding SECRET and CONFIDENTIAL NSI and RD received or developed in conjunction with activities licensed, certified, or regulated by the Commission. The staff should review the applicant's Plan to ensure that it outlines the facility's proposed security procedures and controls for the protection of classified information; contains a floor plan of the area in which the matter is to be used, processed, stored, reproduced, transmitted, transported, or handled; and contains foreign ownership, Control, or influence (FOCI) information. If the facility already has an approved FOCI determination with another Government agency (e.g., the U.S. Department of Energy or U.S. Department of Defense), the NRC can accept the agency's FOCI as long as it meets the criteria set forth in the Memorandums of Understanding that the NRC has with the respective agencies.

3.4 ACCEPTANCE CRITERIA

3.4.1 Regulatory Requirements

Requirements applicable for the protection of classified matter are contained in 10 CFR Parts 25 and 95 for the level of protection addressed in Section 3.3.

3.4.2 Regulatory Guidance

The regulatory guidance applicable to protection of classified matter is contained in:

Nuclear Regulatory Commission (U.S.) (NRC). "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified Matter for NRC Licensee,

Certificate Holder and Others Regulated by the Commission." NRC: Washington, D.C. October 1999, as revised.

3.4.3 Regulatory Acceptance Criteria

The adequacy of the applicant's Plan is based on compliance with 10 CFR Parts 25 and 95. The information provided by the applicant should be of sufficient depth to allow the staff to evaluate the adequacy and appropriateness of the applicant's Plan. Acceptance is based on the verification that the applicant has committed to provide, in the Plan, a detailed description of the proposed security procedures and controls for the protection of classified matter and to follow such procedures. These security procedures and controls are based on the requirements of 10 CFR Parts 25 and 95.

3.5 REVIEW PROCEDURES

3.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the protection of classified matter adequately addresses the specific items in Section 3.3, "Areas of Review." If the primary reviewer verifies that the protection of classified matter is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 3.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

3.5.2 Safety Evaluation

After determining that the protection of classified matter is acceptable for review in accordance with Section 3.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 3.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

The primary reviewer should verify that sufficient information has been provided in the license application to satisfy the intent of 10 CFR Parts 25 and 95 requirements with respect to the Plan and that the information provided is consistent with the guidance in this SRP chapter. The primary reviewer should determine if the applicant has provided sufficient information to assess whether the applicant can use, process, store, reproduce, transmit, transport, or handle NSI and/or RD in connection with NRC activities, in a manner that will provide adequate protection and prevent unauthorized access. The primary reviewer should verify that the applicant will not be using, processing, storing, reproducing, transmitting, transporting, or handling Top Secret information since no such information is authorized under Part 95.

If the applicant submits material to support the protection of classified matter with the material it submits for the construction approval (or otherwise before applying for a license to possess and

use special nuclear material [SNM]), and the primary reviewer concludes in the corresponding safety evaluation that the applicant has met Parts 25 and 95 relating to classified matter protection, the primary reviewer does not need to repeat the review as part of the safety evaluation for the license to possess and use SNM to the extent that it remains the same.

3.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the Safety Evaluation Report. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

> *The staff reviewed the standard practice procedures plan for the protection of classified matter (Plan) for [name of facility] according to Chapter 3.0 of NUREG–1718. On the basis of the following finding, the staff concludes that the Plan is acceptable for implementation.*
>
> *[State what was reviewed and why it was acceptable.]*
>
> *The applicant adequately described and documented the protection of classified matter and has provided a plan to address those parts of 10 CFR Parts 25 and 95 relating to classified matter protection. Meeting the staff's requirements as given above provides an acceptable basis for the finding that, insofar as classified matter protection is concerned, the applicant meets the applicable requirements within Parts 25 and 95.*

3.7 REFERENCES

Code of Federal Regulations, *Title 10, Energy,* Part 25, "Access Authorization for Licensee Personnel."

———. *Title 10, Energy,* Part 70, "Domestic Licensing of Special Nuclear Material."

———. *Title 10, Energy,* Title 10, Part 95, "Security Facility Approval and Safeguarding of National Security Information and Restricted Data."

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Nuclear Regulatory Commission (U.S.) (NRC). "Standard Practice Procedures Plan Standard Format and Content for the Protection of Classified Matter for NRC Licensee, Certificate Holder and Others Regulated by the Commission." NRC: Washington, D.C. October 1999, as revised.

4.0 ORGANIZATION AND ADMINISTRATION

4.1 PURPOSE OF REVIEW

The purpose of this review is to ensure that the applicant's organizational structure and administrative policies and procedures provide reasonable assurance that the applicant will plan, implement, and control site activities in a manner that ensures the safety of the workers, the public, and the environment. The review also ensures that the qualifications for key management positions are adequate.

4.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: Primary reviewers for all other Standard Review Plan (SRP) sections or chapters

Supporting: None

4.3 AREAS OF REVIEW

The applicant should submit organization and administration information with the material submitted for the construction approval and should resubmit updated information with the license application to possess and use special nuclear material (SNM) as described below. The areas of review for organization and administration should include:

A. Construction Approval

i. Organization

a. The identification and functional description of the specific organizational groups responsible for designing and constructing the facility. Organizational groups should include contractors, consultants, and other outside service organizations in addition to the applicant.

b. Authorities and responsibilities among the organizational groups and the means of communication. This should include, but not be limited to, the process designers, architect engineering firm, and the construction contractor.

c. Organizational charts that depict the lines of responsibility and authority and the key management positions.

ii. Administration

a. Plans to transition the organization from the design and construction phase to the operations phase.

iii. Key Management Positions

a. The individual responsible for the principal structures, systems, and components (SSCs), during design and construction.

b. Key management positions with responsibility for the principal SSCs during design and construction.

c. The qualification criteria for each key management position with responsibility for the principal SSCs, including:

(1) Academic credentials;
(2) Continuing education or training; and
(3) Work experience.

d. The availability of supervisory and management positions to perform their duties.

B. License To Possess and Use SNM

i. Organization

a. The identification and functional description of the specific organizational groups responsible for operating the facility. Organizational groups should include contractors, consultants, and other outside service organizations in addition to the applicant.

b. Authorities and responsibilities among the organizational groups and the means of communication.

c. Organizational charts that depict the lines of responsibility and authority and the key management positions.

ii. Administration

a. Administrative policies and procedures that describe the implementation and relationships among the design basis, integrated safety analysis (ISA), the resulting safety program, and supporting management measures.

iii. Key Management Positions

a. The individual responsible for health, safety, and the environment (HS&E) during operations.

b. Key management positions with HS&E responsibility during operations. These positions should include the Plant Manager, Operations Manager, Shift Supervisor, and HS&E Managers or equivalent.

c. The qualification criteria for each key management position with HS&E responsibility, including:

(1) Academic credentials;
(2) Continuing education or training; and
(3) Work experience.

d. The availability of supervisors and managers to perform their duties.

4.4 ACCEPTANCE CRITERIA

4.4.1 Regulatory Requirements

The regulatory requirements for organization and administration are found in 10 CFR 70.22, 70.23, and other sections of 10 CFR Part 70, concerning the applicant's corporate organization, staff qualifications, and the adequacy of the proposed equipment, facilities, and procedures to provide adequate safety for workers, the public, and the environment.

4.4.2 Regulatory Guidance

None.

4.4.3 Regulatory Acceptance Criteria

The applicant's organization and administration should be acceptable if:

A. Construction Approval

i. Organizational Structure

a. Clear and unambiguous controls and communications exist between the organizational groups for designing and constructing the facility.

b. Lines of communication, responsibility, and authority are clearly delineated between the organizational groups.

c. A corporate officer is responsible for the activities that are related to the design and construction of the principal SSCs.

ii. Administration

a. The applicant commits to establishing formal management measures as described in Chapter 15.0 of this SRP as necessary and appropriate to provide reasonable assurance of the availability and reliability of the items relied on for safety.

b. The organization in conjunction with the administration, and specifically the plans to transition from design and construction to operation, are adequate to maintain the design bases of the facility at all times.

iii. Key Management Positions

a. The personnel to design and construct the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities, as indicated in the organizational structure.

b. The key management will be appropriately available during design and operation. Additionally, the number of key management as indicated on the organizational charts is appropriately defined for the scope of each organizational function.

c. The applicant documents the qualifications, responsibilities, and authorities for key management positions related to the design and construction of the principal SSCs in position descriptions.

B. License To Possess and Use SNM

i. Organizational Structure

a. Clear and unambiguous controls and communications exist between the organizational groups for operating the facility.

b. Lines of communication, responsibility, and authority are clearly delineated between the organizational groups.

c. The HS&E organization(s) is independent of the operations organization(s), allowing it to provide objective HS&E audits, reviews, or control activities. "Independent" means that neither organization reports to the other in an administrative sense. Both may report to a common manager.

d. A corporate officer is responsible for HS&E activities.

e. The individual with overall responsibility (or delegated responsibility) for HS&E functions has the authority to shut down operations if they appear unsafe. If this individual shuts down operations, the applicant requires that the same individual approve the restart of operations. Typically, this individual should

have the same authority as the Production or Operations Manager and have direct line responsibility to the Plant Manager.

ii. Administration

a. The activities essential for effective implementation of the management measures or any other identified HS&E functions are documented in formally approved, written procedures prepared in compliance with a formal document control program. This documentation provides reasonable assurance that management measures are appropriately implemented for all items relied on for safety.

b. The applicant commits to a simple mechanism for reporting potentially unsafe conditions or activities to the HS&E organization and/or to upper management that is available for use by any person in the plant. Reported concerns are investigated, addressed, and resolved promptly.

iii. Key Management Positions

a. The personnel to operate the facility have the appropriate breadth and level of experience for their respective authorities and responsibilities, as indicated in the organizational structure.

b. The key management will be appropriately available during operation. Additionally, the number of key management as indicated on the organizational charts is appropriately defined for the scope of each organizational function.

c. The applicant documents the qualifications, responsibilities, and authorities for key management positions with HS&E in position descriptions.

4.5 REVIEW PROCEDURES

4.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 4.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use SNM.

Specifically, the primary reviewer should compare the application for the construction approval review against Section 4.3(A) and the application for the license to possess and use SNM against Section 4.3(B). If the primary reviewer verifies that the organization and administration is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 4.5.2. If the primary reviewer identifies significant deficiencies in the

material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

4.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 4.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 4.4. Specifically, for the construction approval review, the primary reviewer should compare the application against Section 4.4.3(A). For the review for the license to possess and use SNM, the primary reviewer should compare the application against Section 4.4.3(B). On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet the acceptance criteria in SRP Section 4.4.

To facilitate the safety evaluation for either the construction approval review or the review for the license to possess and use SNM, each reviewer should examine the material provided in Section 4.3, "Areas of Review." In addition, the primary reviewer should verify with the secondary reviewers that the planned implementation of the organization and administration is consistent with other parts of the application, including any additional acceptance criteria in their respective review areas.

4.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the Safety Evaluation Report. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

> *The staff reviewed the organization and administration for construction approval for [insert name of facility] according to Chapter 4.0 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and found that [state the findings].*
>
> *The staff concluded that the applicant's organization and administration provide reasonable assurance that the applicant has an acceptable organization, appropriate administrative policies, and qualified key management positions to ensure the principal structures, systems, and components will protect against the consequences of accidents and natural phenomena.*

The staff could document the safety evaluation for the license to possess and use SNM review as follows:

> *The staff reviewed the organization and administration for the license to possess and use SNM for [insert name of facility] according to Chapter 4.0 of NUREG–1718. The staff evaluated [insert a summary of the material reviewed] and found that [state the findings].*

The staff concluded that the applicant's organization and administration provide reasonable assurance that the applicant has an acceptable organization, appropriate administrative policies, and qualified key management positions to satisfy the regulatory requirements for a license to possess and use SNM.

4.7 REFERENCES

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

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Nuclear Regulatory Commission (U.S.) (NRC). NUREG–1324, "Proposed Method for Regulating Major Materials Licensees." (Sections 3.1, "Organization Plan," and 3.2, "Managerial Controls and Oversight.") NRC: Washington, D.C. 1992.

5.0 INTEGRATED SAFETY ANALYSIS (ISA)

5.1 PURPOSE OF REVIEW

The types of submittals from the applicant that are addressed by this chapter are:

- The applicant's safety assessment of the design bases for the mixed oxide (MOX) fuel fabrication facility, which is submitted pursuant to 10 CFR 70.22(f) for construction approval; and

- The ISA for a license to possess and use special nuclear material (SNM), which includes:

  -- The ISA chapter of the license application that contains the applicant's ISA programmatic commitments; and

  -- The applicant's declaration that it completed an ISA in accordance with the regulations and the ISA Summary of the processes, methods, personnel, and results of the ISA.

A. Safety Assessment of the Design Bases

The purpose of this review is to establish that the material the applicant submits to obtain a construction approval includes a description of the plant site and a safety assessment of the design bases that demonstrates that the applicant's principle structures, systems, and components (SSCs) will provide protection against natural phenomena and the consequences of other accidents in accordance with the performance requirements of 10 CFR 70.61. Pursuant to 10 CFR 70.23(b), the Commission must grant approval before the applicant may begin constructing the facility.

The safety assessment of the design bases is neither an ISA nor a substitute for the ISA that is submitted with the application for a license to possess and use SNM (see Item B); instead, the safety assessment of the design bases allows the staff to determine if the applicant's design bases is adequate to meet 10 CFR 70.23(b) and to determine that the applicant, by using the safety assessment of the design bases, is building a foundation for the ISA to support the license application. Moreover, the processes the applicant uses to develop the safety assessment for the design bases should be analogous to the processes that the applicant will use to develop the ISA for the license application. Therefore, the areas of review and acceptance criteria described for the safety assessment of the design bases draw upon the acceptance criteria for the ISA for the license application.

The relationship between the safety assessment of the design bases and the ISA is shown in Figure 5-1.

B. The ISA

i. ISA Programmatic Commitments

The purpose of the review of the ISA chapter of a license application is to determine that the applicant established and commits to organization and procedures as may be explicitly required by the regulation for the ISA, or sufficient to accomplish an ISA function required by the regulation, and provides a formal system to manage changes to the ISA.

ii. ISA Results and Summary

The purpose of the review of the ISA results, primarily as described in the ISA Summary, is to establish reasonable assurance that the applicant:

a. Performed a comprehensive ISA of the MOX fuel fabrication facility and its processes using effective systematic methods and competent staff.

b. Identified and evaluated all hazards and credible accident sequences in the ISA that involve process deviations or other events internal to the facility (e.g., explosions and fires) and credible external events (e.g., floods, high winds, and earthquakes) that could result in consequences to the public, workers, or the environment of the types specified in 10 CFR 70.61.

c. Designated engineered and administrative items relied on for safety (IROFS) and evaluated the set of items for each accident sequence to provide reasonable assurance, through preventive or mitigative measures, that the safety performance requirements of 10 CFR 70.61 are met.

5.2 RESPONSIBILITY FOR REVIEW

Primary: ISA reviewer

Secondary: Reviewers in specific technical areas, including nuclear criticality safety, fire protection, chemical safety, radiation safety, and environmental protection

Supporting: Fuel Facility Inspection Staff

5.3 AREAS OF REVIEW

The staff should review the material submitted for construction approval, which includes the applicant's design bases, safety assessment of the design bases, and principal SSCs of the facility. The safety assessment of the design bases is expected to consist of tasks analogous

to the initial tasks in an ISA as described in Section 5.3.2. Specific areas of review for the construction approval review are documented in Section 5.3.1.

The applicant's ISA programmatic functions and commitments should be documented in the license application. The ISA is part of the safety program and consists of the process safety information (PSI), the methods used by the licensee to perform the ISA, the qualifications of the team performing the ISA, the method of documenting and implementing the results of the ISA, and the process used to keep the ISA current when changes are made to the facility. When the applicant submits the license application, the staff should review the applicant's ISA programmatic functions and commitments, primarily as documented in the application. Specific areas of review are documented in Section 5.3.2(A).

The applicant's ISA Summary, and other ISA documentation, should document the methods, personnel used, and ISA results. The applicant submits the ISA Summary to the NRC with the application for a license to possess and use SNM, with additional ISA documentation available for NRC review at the facility site. The term "results of the ISA" includes all the ISA information that the applicant submits to the NRC (including the programmatic functions and commitments reviewed under Section 5.3.2(A)) plus any additional supporting information that the applicant keeps at the site. The staff should also evaluate the results of the ISA, primarily as described in the ISA Summary. Review of selected additional information or review of information at the applicant's site will, in general, be necessary to attain reasonable assurance of acceptability of the results for compliance with the regulations, particularly, 10 CFR 70.61. Specific areas of review are documented in Section 5.3.2(B).

5.3.1 Safety Assessment of the Design Bases

To determine if the NRC can grant construction approval for a MOX facility in accordance with 10 CFR 70.22(f), areas of review should include:

A. The applicant's plant site description related to the safety assessment of the design basis, including information needed for quantification of the likelihood and severity of the natural phenomena such as earthquakes, tornadoes, floods, natural fires, hurricanes, and other wind storms.

B. A description of the team that performed the safety assessment of the design bases.

C. A definition of the quantitative chemical consequence standards to be used in determining compliance with 10 CFR 70.61;

D. The applicant's methods for conducting the safety assessment of the design basis, including:

   i. The method for hazard identification;

   ii. The method for analyzing select accidents;

iii. Definition of the terms likely, unlikely, highly unlikely, and credible that the applicant will use to show compliance with 10 CFR 70.61 as well as the method used to develop the definitions; and

iv. The applicant's methods to evaluate chemical and radiological consequences and likelihood evaluation to show compliance with 10 CFR 70.61.

E. The safety assessment of the design bases of the principal SSCs of the facility, including:

i. A hazard identification;

ii. A process hazard analysis (PHA) that examines selected accidents; and

iii. An assessment of the likelihoods and consequences of the accidents examined in the PHA (see Item ii).

F. A description of the design bases of the principal SSCs, including:

i. The provisions and design bases for protection against natural phenomena and

ii. The design bases for protection against other potential accidents.

G. The applicant's ISA elements and commitments (see Section 5.3.2(A)), including a description of how the applicant plans to incorporate the safety assessment of the design basis in the ISA performed for the license application.

Evaluation of the adequacy of methods, safety margins, and other discipline-specific safety design bases are contained in the appropriate chapters of this standard review plan (SRP). Review of the quality assurance program description required by 10 CFR 70.22(f) and of other non-ISA elements of the submittal are addressed by the other chapters of this SRP. In particular, the adequacy of safety management measures and generic technical aspects of methods used to analyze design bases for fire and chemical safety, radiological protection, and natural phenomena hazard estimation and evaluation of facility response may be addressed in other chapters.

5.3.2 The ISA

A. ISA Programmatic Commitments

The staff should review the application for a license to possess and use SNM to determine whether the applicant's commitments to perform and maintain an ISA are adequate. In the following, the phrases "process node" or "process" are used to refer to a single reasonably compact piece of equipment or workstation where a single unit process or processing step is conducted. The MOX fuel fabrication facility is expected to be divided into several major process lines or areas, each consisting of many process nodes. Areas of review should include:

i. The applicant's description of and commitments to compile and maintain a current and accurate set of PSI, including information on the hazardous materials, equipment, and technology used in each process. The applicant should explain this activity in detail in the description of its configuration management program (Section 15.2, Configuration Management).

ii. The applicant's description of and commitments to requirements for ISA team training and qualifications (Section 15.4, Training and Qualification of Plant Personnel).

iii. The applicant's description of and commitments to ISA methods, method selection criteria or specific methods to be used for particular classes of process nodes (usually process workstations). For the purposes of this review, the applicant should begin the ISA with an identification of hazards (chemicals, radiological materials, fissile materials, etc.) that may present a potential threat to the public, facility workers, or the environment. Based on a systematic analysis of each plant process, the ISA PHA identifies a set of individual accident sequences or process upsets that could result from the hazards. The applicant's ISA methods address:

   a. Hazard identification;
   b. PHA (accident identification);
   c. Accident sequence construction and evaluation;
   d. Consequence determination and comparability to 10 CFR 70.61; and
   e. Likelihood categorization for determining compliance with 10 CFR 70.61.

iv. The applicant's description of and commitments to management procedures for conducting and maintaining the ISA. The object of this review is to ensure that the overall integrity of the ISA is maintained as a current and accurate safety basis for the facility. The applicant's ISA management procedures include procedures for:

   a. Performing and updating the ISA;
   b. Review responsibility;
   c. ISA documentation;
   d. Reporting ISA Summary changes per 10 CFR 70.72(d)(1) and (3); and
   e. Maintenance of ISA records per 10 CFR 70.62(a)(2).

   The integrity of the applicant's ISA management procedures should be controlled by the applicant's configuration management program (see Section 15.2).

B. ISA Results and Summary

The staff reviews the ISA results (primarily the ISA Summary, but may include other ISA documentation) to find reasonable assurance that the applicant has performed a systematic evaluation of the hazards and credible accident sequences and has identified IROFS and management measures that satisfy the performance requirements of 10 CFR 70.61. The review includes those accidents that result in a release of radioactive material, a nuclear

criticality event, or any other exposure to radiation resulting from use of licensed material. In addition, the staff reviews accidents involving hazardous chemicals produced from licensed materials, that is, chemicals that are licensed materials, or have licensed materials as precursor compounds, or substances that physically or chemically interact with licensed materials and that are toxic, explosive, flammable, corrosive, or reactive to the extent that they endanger life or health. These include substances that are commingled with licensed material or are produced by a reaction with licensed material. If a chemical accident has the potential to cause, or reduce protection from, a radiation exposure accident, then it also must be addressed. On the other hand, event sequences having unmitigated consequences less than those identified in 10 CFR 70.61(c), once identified as such, do not require further consideration within the ISA.

i. The site description (see Section 1.3, Site Description) concerning those factors that could affect safety such as geography, meteorology (e.g., high winds and flood potential), seismology, demography, and nearby industrial facilities and transportation routes.

ii. The facility description concerning features that could affect potential accidents and their consequences. Examples of these features are facility location, facility design information, and the location and arrangement of buildings on the facility site.

iii. The description in the ISA Summary of each process analyzed as part of the ISA. Specific areas reviewed include basic process function and theory, functions of major components and their operation, process design and equipment, and process operating ranges and limits. It is expected that, for certain processes, additional information or a visit to the facility will be necessary to permit staff to understand the process adequately. Reviewer visits to the facility do not obviate the need for accurate, current drawings, process descriptions, and related information that are needed to evaluate facility safety.

iv. The applicant's ISA team qualifications as described in the ISA Summary.

v. The applicant's ISA methods as described in the ISA Summary. Additional information concerning methods provided in the application. Documentation of specific examples of the application of methods may be requested or reviewed onsite to confirm understanding of specific methods.

vi. The applicant's quantitative standards for the chemical consequences levels specified in 10 CFR 70.61, as described in the ISA Summary.

vii. The applicant's definitions of unlikely, highly unlikely, and credible used in 10 CFR 70.61, as described in the ISA Summary.

viii. The information resulting from the ISA that demonstrates compliance with the performance criteria of 10 CFR 70.61. In addition to the information specifically required as noted in Items ix through xi below, this information includes for each applicable process:

a. The consequences evaluated for each postulated accident sequence and comparison to the consequence levels identified in 10 CFR 70.61. Information such as inventory and release path factors that support the results of the consequence evaluation.

b. Information showing how each accident sequence has been assessed to have the likelihood required by 10 CFR 70.61.

c. Information describing how each accident sequence, for each process, is protected sufficiently by the IROFS listed in the ISA Summary to comply with 10 CFR 70.61.

ix. Information, in the ISA Summary, listing hazards and interactions for each process.

x. Information provided in the ISA Summary that describes all accident sequences.

xi. The list, in the ISA Summary, describing the IROFS for all accidents in each process sufficiently to understand their safety function in meeting the appropriate consequence and likelihood requirements of 10 CFR 70.61.

xii. The list, in the ISA Summary, identifying those IROFS which are the sole item relied on in an accident sequence to assure compliance with 10 CFR 70.61.

xiii. The information, in the ISA Summary, demonstrating compliance with the criticality monitoring requirements of 10 CFR 70.24.

xiv. The information, in the ISA Summary, demonstrating compliance with baseline design criteria required by 10 CFR 70.64(a)(1) through (5) and (7) through (10) for new facilities, or new processes at existing facilities, and required to be submitted in accordance with 10 CFR 70.65(b)(4). Since these elements all bear on the adequacy of IROFS, it is efficient to include their review in the ISA Summary review.

It is expected that, in addition to reviewing the application and ISA Summary, the NRC staff will select subsets of certain areas for which additional information will be reviewed, in some cases at the site. The method for selecting specific processes or accidents for additional review is described in Section 5.5 of this chapter, "Review Procedures."

5.4 ACCEPTANCE CRITERIA

5.4.1 Regulatory Requirements

The regulations in 10 CFR 70.23(b) require an applicant who is planning to construct and operate a plutonium processing and fuel fabrication facility, such as a MOX facility to obtain NRC approval prior to initiating construction. The NRC's approval is based on information the

applicant submits pursuant to 10 CFR 70.22(f), which includes the safety assessment of the design bases.

The requirement to perform an ISA is specified in 10 CFR 70.62. The regulations in 10 CFR 70.62(a)(2) require that the applicant establish and maintain records of PSI, which are needed to perform and support the ISA. Also, 10 CFR 70.62(c) specifies requirements for the tasks comprising the ISA and for the qualifications of ISA team personnel, and requires that the ISA must evaluate whether the applicant's facility, with its listed IROFS, meets the safety performance requirements of 10 CFR 70.61. The regulations of 10 CFR 70.64 specify design criteria requirements for new facilities. Additionally, 10 CFR 70.72 states requirements for keeping the ISA and its documentation current when changes are made to SSCs.

5.4.2 Regulatory Guidance

Guidance applicable to performing an ISA and documenting the results is contained in NUREG–1513, "Integrated Safety Analysis Guidance Document." A sample approach for a risk evaluation of one process is provided in Appendix A to this SRP to illustrate an acceptable form and content.

5.4.3 Regulatory Acceptance Criteria

5.4.3.1 Safety Assessment of the Design Bases

The construction approval review includes the safety assessment of the design bases of the principal SSCs that will protect against natural phenomena and other accidents. The safety assessment of the design bases is not a substitute for the ISA that is submitted with an application for a license to possess and use SNM (see Section 5.4.3.2); instead, the safety assessment of the design bases allows the staff to determine if the applicant's design bases are adequate to meet 10 CFR 70.23(b).

The steps the applicant follows to develop the safety assessment for the design bases should be analogous to the steps that the applicant will use to develop the ISA; however, the reviewer should expect the application of these steps to be adjusted according to the level of design when the applicant applies for construction approval. To avoid repetition, the acceptance criteria in this section are cross-referenced with the acceptance criteria for the ISA.

The staff should find the applicant's safety assessment of the design bases of the principal SSCs acceptable if the following criteria are met:

A. The applicant's plant site description includes sufficient information to support the safety assessment of the design bases, including:

   i. A site description. The site description should contain similar information as defined in Section 5.4.3.2(B)(i). The level of detail should be sufficient to allow an evaluation of natural phenomena and other external accidents.

ii. A facility description. The facility description should contain similar information as described in Section 5.4.3.2(B)(ii). The level of detail should allow the reviewer to understand the relationship between the design bases of the principal SSCs and the facility.

iii. A process description. The process description should contain similar information as defined in Sections 5.4.3.2(B)(iii)(a) and (b) consistent with the level of design. In addition, the applicant should provide additional detail (see Sections 5.4.3.2(B)(iii)(c) and (d)) where the process design is established through the design bases (e.g., operating ranges and limits).

The level of detail the applicant provides to meet the above acceptance criteria is consistent with the level of design.

B. The team that performed the safety assessment of the design bases should meet the acceptance criteria of Section 5.4.3.2(B)(iv).

C. The applicant proposes quantitative standards for chemical consequences to assess the consequences from acute chemical exposures. The applicant's quantitative standards should be consistent with Section 5.4.3.2(B)(vi).

D. The applicant's methods for conducting the safety assessment of the design bases are a logical precursor to the methods the applicant will use to perform an ISA as described in Section 5.4.3.2(B)(v). The applicant considers the level of design when it selects the methods for the safety assessment of the design bases. For example, the level of design when the applicant requests construction approval may dictate that the applicant's methods are more approximate and less complete than would be expected for an ISA. However, the applicant should still provide reasonable estimates based on quantitative information that is consistent with valid methods.

i. The applicant uses the methods of NUREG–1513 for hazard identification or shows that the method the applicant selected will result in a hazard identification consistent with Section 5.4.3.2(B)(v)(a).

ii. The applicant's methodology for the PHA should allow the applicant to examine selected accidents. At a minimum, this should include:

a. Natural phenomena and other types of bounding accidents;

b. Other events such as fires, explosions, criticalities, radiological (or hazardous chemical as applicable under 10 CFR Part 70) exposures, and loss of containment; and

c. Potential accidents from other classes of hazards indicated by the hazard identification.

iii. The applicant's definitions for likelihood should be consistent with the guidance in Section 5.4.3.2(B)(vii). The applicant's methodology for applying likelihoods to the accidents examined in Item ii above should:

a. Show how each principal SSC acts to prevent or mitigate the accident;

b. Address any design bases that ensure that the principal SSC will function as intended (e.g., safety margins for criticality).

c. Assign the accident sequences as unlikely, highly unlikely, or neither.

iv. The applicant's methodology for assessing the consequences of accidents is consistent with the acceptance criteria in Section 5.4.3.2(B)(v)(c).

v. The applicant describes how the methods used for the safety assessment of the design bases differ from the applicant's methods for the ISA (see Item G) and provides plans to transition from the design bases to the ISA.

E. The applicant's safety assessment of the design bases shows that the design and design bases will result in a facility that will meet the performance requirements of 10 CFR 70.61 and the defense-in-depth requirement of 10 CFR 70.64(b).

The applicant's safety assessment of the design bases includes:

i. A hazards analysis based on the methods described in Item D(i) that identifies the approximate location and quantities of SNM and other hazardous materials.

ii. A process hazard analysis and accident sequence identification based on the methods described in Item D(ii). In particular, the applicant:

a. Provides the frequency of occurrence of severity levels of the phenomena and demonstrates the ability of the SSC to withstand specified severity levels. The applicant may demonstrate the frequencies of natural phenomena and assess the likelihood that the safety functions of the SSCs will not fail when subject to natural phenomena by reference to accepted standards rather than by individual analyses.

A discussion of what tasks constitute a safety assessment of design bases for protection against natural phenomena is found in Appendix B of this SRP. Accepted standards for natural phenomena assessment are referenced therein.

b. Indicates the controlled parameters for safe operation, provides the limiting values of any controlled parameter, and explains and assesses the means of controlling those parameters to within those limiting values.

c. Explains for processes vulnerable to criticality accidents, why it is expected that the given design and design bases will meet the double contingency requirement of 10 CFR 70.64(a)(9).

As discussed in Item iii below, the accident consequences will depend on the design bases of the principal SSCs. When analyzing accident sequences, the applicant should examine the failure of ALL features, structures, control devices, equipment, or procedures to ensure that all principal SSCs are appropriately identified.

iii. Consequence assessment

The applicant's consequence assessment is sufficiently quantitative to compare the consequence estimates against the performance requirements of 10 CFR 70.61. The applicant does not determine the consequences for all accidents and all SSCs individually; however, the applicant demonstrates that the consequence assessment is bounding through the applicant's analysis of representative processes sufficient to cover all principal types of hazardous materials.

iv. Likelihood Assessment

The applicant provides information that indicates that the frequencies of accidents are in accordance with the acceptance criteria for the applicant's likelihood definitions. The applicant's safety assessment of the design bases with respect to likelihood provides reasonable assurance that the likelihood requirements of 10 CFR 70.61 will be met by the final design. The applicant commits to using equivalent or refined definitions of likely, unlikely, highly unlikely, and credible in the ISA. In addition, the applicant describes the likelihood evaluation method to be used in the ISA. The applicant makes these methods and definitions part of the design bases.

F. The applicant describes the principal SSCs. This description should include:

i. The number, types, and description of the principal SSCs. In particular, the applicant describes the general features that indicate that the principal SSCs can be designed and constructed to meet the design bases.

The description of the principal SSCs need not be at the level of detailed engineering drawings. However, principal safety function features, devices, amounts of hazardous materials, and the principal dimensions, layout, and location relevant to safety must be given. Each general type of principal SSC or process using the same design bases must be described. However, approximate numbers of each general type of principal SSC or process is sufficient. It is the safety basis that is to be assessed.

ii. For each principal SSC, the parameters that will be specified or controlled for safety and the ranges and values of those parameters that constitute the design bases. For active engineered controls, the applicant states the type of sensing and the type of control device. For passive engineered controls, the applicant states the general geometry,

materials, and how they prevent the accident. This applicant should also address administrative controls (if any). The applicant demonstrates that these parameters are consistent with the process description (see Item A(iii)) and incorporates sufficient safety margins to account for uncertainties.

iii. For natural phenomena hazards, the general aspects of the principal SSCs that make them resistant to failure.

iv. The functional relationship of each principle SSC to the top-level safety function for a process, for example, by a fault tree.

By definition (see the Glossary to this SRP), all principal SSCs are IROFS. Therefore, the applicant either commits to assigning all principal SSCs as IROFS when the applicant performs an ISA, or commits to providing justification as to why an IROFS designation is not necessary for a principal SSC.

G. The applicant commits to ISA programmatic commitments for completing the ISA for a license to possess and use SNM (see Section 5.4.3.2(A)). The commitments are consistent with the regulatory acceptance criteria in Section 5.4.3.2(A) considering the level of design.

5.4.3.2 The ISA for the License Application

The acceptance criteria for an ISA are based on meeting the relevant requirements in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." The ISA will form the basis for the safety program by identifying potential accidents, designating IROFS and management measures, and evaluating the likelihood of each accident sequence for compliance with 10 CFR 70.61. The acceptance criteria in Section 5.4.3.2(A) address the programmatic commitments made by the applicant to perform and maintain an ISA. The acceptance criteria in Section 5.4.3.2(B) address the ISA results and whether those results demonstrate the ability of the applicant to meet the performance requirements of 10 CFR 70.61.

A. ISA Programmatic Commitments

Several elements may be necessary to carry out each required program function effectively. These elements may include: organization, assignment of responsibilities, management policies, required activities, documented procedures for activities, use of industry consensus standards, and technical safety practices. The applicant's commitment to each ISA requirement of the rule should be acceptable if it:

- Describes each necessary safety program element sufficiently to understand how well it supports the safety program function;

- Commits to each safety program element, as described, and to maintaining onsite written procedures for carrying out that function, if necessary; and

- Provides reasonable assurance that the elements, as described, would be effective in accomplishing the safety program function.

To be acceptable, commitment statements in the application should be declarative sentences with main verbs such as: shall, will, is, or must. Sentences with phrases expressing optional alternatives or recommendations—such as: should, may, will be considered, or as appropriate—may be acceptable if there are supporting statements giving the criteria for selecting the option. If no selection criteria are given, then phrases stating recommendations or options are not commitments. However, it may be acceptable for some safety elements of lesser importance not to be stated as commitments.

The staff should find the applicant's ISA programmatic commitments acceptable if the following criteria are met:

i. The applicant commits to compiling and maintaining current a database of PSI. As part of this commitment, the applicant will use the written PSI to update the ISA and to identify and understand the hazards associated with the processes. The applicant's compilation of written PSI includes:

   a. The hazards of all materials used or produced in the process, including information on chemical and physical properties such as toxicity, acute exposure limits, reactivity, chemical and thermal stability or other applicable information that would typically be included on Material Safety Data Sheets (see 10 CFR 1910.122(g)).

   b. Equipment used in the process, including information of a general nature on topics such as the materials of construction; piping and instrumentation diagrams; ventilation; design codes and standards employed; material and energy balances; safety systems (e.g., interlocks, detection or suppression systems); electrical classification and relief system design; and the design bases.

   c. Technology of the process, including block flow diagrams or simplified process flow diagrams, a brief outline of the process chemistry, safe upper and lower limits for controlled parameters (e.g., temperature, pressure, flow, concentration) and an evaluation of the health and safety consequences of process deviations.

ii. The applicant commits to keeping the ISA and ISA Summary accurate and up-to-date by means of a suitable configuration management system. The applicant's ISA accounts for any changes made to the facility or its processes (e.g., changes to the site, operating procedures, control systems). The applicant succinctly outlines its management policies, organizational responsibilities, revision time frame, and procedures to perform and approve revisions to the ISA. The applicant commits to evaluating any facility changes or changes in the process safety information that may alter the parameters of an accident sequence by means of the facility's ISA methodology. The applicant commits to using an ISA team with similar qualifications to those used in conducting the original ISA for any modifications and revisions that the applicant deems necessary. The applicant commits to reviewing any facility changes that may increase the level of

risk and, if dictated by revision of the ISA, to selecting and implementing new or additional IROFS and appropriate management measures. The applicant commits to submitting to the NRC revisions of the ISA Summary within the time frame specified in 10 CFR 70.72(d)(1).

iii. The applicant commits to promptly addressing any safety-significant vulnerabilities or unacceptable performance deficiencies identified in the ISA. Whenever an update of the ISA is conducted, the applicant commits to taking prompt and appropriate actions to address any vulnerabilities that may have been identified. If a proposed change results in a new type of accident sequence (e.g., different initiating event, changes in the consequences as defined in 10 CFR 70.61) or increases the risk of a previously analyzed accident sequence to an unacceptable level, the applicant commits to promptly evaluating the adequacy of existing IROFS and associated management measures and to making necessary changes, if required.

iv. The applicant includes procedures and criteria for changing the ISA, along with its commitment to design and implement a facility change mechanism that meets the requirements of 10 CFR 70.72. The applicant should discuss the evaluation of the change within the ISA framework and procedures and responsibilities for updating the facility ISA.

v. The applicant commits to engaging personnel with appropriate experience and expertise in engineering and process operations to update the ISA and keep it current. The ISA team for a process shall consist of individuals knowledgeable in the facility's ISA methodology and in the operation and hazards of the particular process.

vi. The regulations in 10 CFR 70.62(c) require that an ISA of appropriate complexity be conducted for each process and that it accomplish six results. The application is acceptable if it describes sufficiently specific methods and criteria that would be effective in accomplishing each of these tasks. Such effective methods and criteria are described in NUREG–1513, NUREG/CR–6410, Item v of Section 5.4.3.2(B), and Appendix A to this SRP. The applicant must provide sufficient features, criteria, equations, and data so that the staff can evaluate how the ISA for a particular process shows that the applicant will meet the performance requirements of 10 CFR 70.61.

vii. The applicant commits to implementing all IROFS and maintaining them in a functional state so they are available and reliable when needed. Management measures (which are evaluated in Chapter 15.0) comprise the principal mechanism by which the reliability and availability of IROFS are assured.

B. ISA Results and Summary

The preceding section addressed commitments to ISA requirements of the safety program. This section addresses whether the results of carrying out that program, i.e., the ISA methods and results, demonstrate compliance with the performance criteria of 10 CFR 70.61. Information in the ISA Summary should provide the primary basis for

drawing a conclusion that staff is reasonably assured that the identified IROFS will satisfy the performance requirements of the rule. However, the basis for the staff conclusion would not be limited to a determination that the applicant's ISA program has the capability only to identify the appropriate IROFS. Rather, the focus of the staff review would be on the sufficiency of the IROFS identified in the ISA Summary. This requires a determination of whether the identified IROFS are adequate to control the potential accidents of concern at the facility. The accidents of concern are those whose consequences would be at the high and intermediate consequence levels absent any preventive or mitigative controls. In this context, adequacy means the capability of the IROFS to prevent the related accidents with sufficient reliability, or to sufficiently mitigate their consequences. This, in turn, requires staff to make a determination concerning the completeness of the accident sequences identified in the ISA Summary. To support such a review, the information in the ISA Summary needs to provide enough information concerning the accidents to which the IROFS relate to be able to assess their contributions to prevention or mitigation. The ISA Summary must contain enough information concerning the ISA procedures, methods, and human resources employed to have confidence that the potential accidents identified are reasonably complete.

The completeness and adequacy of the IROFS is not the only consideration for satisfying the performance requirements of 10 CFR 70.61. In addition, staff needs to determine that appropriate management measures will be in place that will ensure the availability and reliability of the identified IROFS, to the degree needed to satisfy the likelihood element of the performance requirement.

The following acceptance criteria address, in the order given in 10 CFR 70.65(b), each of the required content elements of the ISA Summary. The acceptance criteria are not simply that the ISA Summary elements are described in the document submitted, but rather that the information submitted is sufficient to demonstrate that the applicant's process safety design and safety procedures meet the performance requirements of 10 CFR 70.61 and other ISA requirements of 10 CFR Part 70. Thus the staff will accept the applicant's ISA results if the staff finds that the following criteria are met:

i. Site Description

The applicant's site description in the ISA Summary includes or references the following safety-related information with emphasis on those factors that could affect safety:

a. The site geography, including the site location and the location of other prominent natural and manmade features, such as mountains, rivers, airports, population centers, possible hazardous commercial and manufacturing facilities, etc., adequate to permit evaluation of:

(1) The likelihoods of accidents caused by external factors and

(2) The consequences of potential accidents.

b. Population information, based on recent census data, that shows population distribution as a function of distance from the facility adequate to permit evaluation of regulatory requirements, including the public consequences listed in 10 CFR 70.61.

c. Natural phenomena (e.g., tornadoes, hurricanes, and earthquakes) and other external events, characterized sufficiently to assess their impact on facility safety and to assess their likelihood of occurrence. The applicant identifies the design bases events for the facility and indicates which events are considered incredible and the basis for that determination. The assessment also indicates which events could occur without adversely impacting safety. Natural phenomena are addressed in more detail in Appendix B to this SRP.

The level of detail for this material is greater than that which would be acceptable in the general information contained in Chapter 1.0 because the information is needed to evaluate the ISA.

ii. Facility Description

The applicant's facility description identifies and describes the general features that affect the reliability or availability of IROFS. The information provided should adequately support an overall understanding of the facility structure and its general arrangement as it pertains to the ISA. As a minimum, the applicant adequately identifies and describes:

a. The facility location and the distance from the site boundary in all directions, including the distance to the nearest resident and the distance to boundaries in the prevailing wind directions.

b. Design information regarding the facility's resistance to failures caused by credible external events, when those failures may produce consequences exceeding those identified in 10 CFR 70.61.

c. The location and arrangement of buildings on the facility site.

If the applicant provides facility description information in the license application, the applicant may provide a reference to the appropriate section.

iii. Processes

The applicant's description of the processes analyzed as part of the ISA provides sufficient detail to provide staff with an understanding of the theory of operation and to allow the staff to determine compliance with the performance requirements of 10 CFR 70.61. The applicant may provide a description at the systems level if it permits the staff to conduct: (1) an evaluation of the completeness of the hazard and accident identification tasks (see Item B(viii), Information Demonstrating Compliance with the Performance Requirements) and (2) an evaluation of the likelihood and consequences

of the accidents identified (see Item B(viii)). Where the applicant identified a need for IROFS in the ISA results (as identified in the ISA Summary), the applicant provides an adequate explanation of how the IROFS reliably prevent the process from exceeding safety limits for each case identified in the ISA results. The process description includes:

a. Basic process function and theory, including a general discussion of the basic theory of the process;

b. Function and operation of major components, including the general arrangement, function, and operation of major components in the process; process schematics showing the major components and instrumentation; and, if appropriate, chemical flow sheets showing the compositions of the various process streams.

c. Process design and equipment, including a discussion of the process design, equipment, and instrumentation that is sufficiently detailed to permit an adequate understanding of the results of the ISA. In particular, it is usually necessary for criticality safety to diagram the location and geometry of the fissile and other materials in the process, for both normal and bounding abnormal conditions. This can be done using either schematic drawings or textual descriptions indicating the location and geometry of fissile materials, moderators, etc., sufficient to permit an understanding of how the IROFS limit the mass, geometry, moderation, reflection, etc. (see Chapter 6.0 for more information on nuclear criticality safety).

d. Process operating ranges and limits, including the operating ranges and limits for measured process variables (e.g., temperatures, pressures, flows, and compositions) that are controlled by IROFS to ensure safe operation of the process. The process operating limits and ranges are consistent with those the applicant evaluated as adequate for safety in the ISA. The applicant may elect to present this information as a tabular summary of all IROFS grouped according to hazard type, that is, nuclear criticality, radiological hazards, chemical hazards, etc., as shown in Appendix A to this SRP.

iv. ISA Team Qualifications

The applicant's ISA teams and team qualifications, as stated in the ISA Summary, meet the following acceptance criteria:

a. The ISA team has a team leader who is formally trained and knowledgeable in the ISA methodology chosen for the hazard and accident evaluations. In addition, the team leader should have an adequate understanding of all process operations and hazards under evaluation, but should not be the cognizant engineer or expert for that process.

b. At least one member of the ISA team has thorough, specific, and detailed experience in the process under evaluation.

c. The team represents a variety of process design and safety experiences in those particular safety disciplines relevant to hazards that could credibly be present in the process, including, if applicable, radiation safety, nuclear criticality safety, fire protection, and chemical safety disciplines.

d. A manager provides overall administrative and technical direction for the ISA.

v. ISA Methods:

It is important that the reviewer determine what the methods and criteria used in the ISA are and whether they are adequate in principle before evaluating results for individual processes. The summary of ISA methods is considered acceptable if it describes the methods used for each ISA task. In accordance with NUREG–1513, it is expected that different specific analytical techniques will be used in different processes depending on their nature and complexity. Specific acceptance criteria for methods used in each ISA task are as follows:

a. Hazard Identification Method: The applicant's hazard identification method leads to a hazard identification that:

(1) Provides a list of materials (radioactive, fissile, flammable, and toxic) or conditions that could result in hazardous situations. The list includes maximum intended inventory amounts and the location of the hazardous materials at the site[1].

(2) Provides a hazards interaction table showing potential interactions between materials, including conditions that could result in hazardous situations.

b. PHA Method: To perform the PHA, the applicant selects one of the individual methods described in NUREG–1513 in accordance with the selection criteria of that document. The applicant may use individual PHA methods not described in NUREG–1513, provided that:

(1) The applicant uses criteria for an individual PHA process that are consistent with the principles of the PHA selection criteria in NUREG–1513.

(2) The applicant's PHA method adequately addresses all the hazards identified in the hazard identification task. The method justifies any hazards eliminated from further consideration.

[1] At least the following hazardous materials should be included in the inventory list if present onsite: ammonia, fines (e.g., UO_2 or PuO_2 dust, beryllium), flammable liquids and gases, fluorine, hydrofluoric acid, hydrogen, nitric acid, organic solvents, propane, and Zircalloy.

(3) The applicant's PHA method provides reasonable assurance that the applicant identifies all significant accident sequences (including the IROFS used to prevent or mitigate the accidents) that could result in the consequences identified in 10 CFR 70.61.[2]

(4) The applicant's PHA method accounts for the interactions of identified hazards and proposed IROFS, including system interactions, to ensure that the overall level of risk at the facility is consistent with the requirements of 10 CFR 70.61 and appropriately limited.

(5) The applicant's PHA method addresses all modes of operation, including startup, normal operation, shutdown, and maintenance.

(6) The applicant's PHA method addresses hazards resulting from process deviations (e.g., high temperature, high pressure), initiating events internal to the facility (e.g., fires, explosions), and credible hazardous external events (e.g., floods, high winds, and earthquakes, airplane crashes). The applicant provides justification for the determination that certain events are incredible and, therefore, not subject to analysis in the ISA.

(7) The applicant's PHA method considers initiation of, or contribution to, accident sequences by human error through the use of human-systems interface analysis or other appropriate methods.

(8) The applicant's PHA method considers common-mode failures and system interactions in evaluating systems that are to be protected by double contingency.

(9) The applicant provides justification, in the ISA Summary, that the individual method would effectively accomplish Items (1) through (8) above.

c. Consequence Evaluation Method: The applicant's method for ISA consequence evaluation, as described in the ISA Summary:

(1) Consists of or is consistent with the approaches described in NUREG/CR–6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," March 1998;

(2) Provides a scientifically correct and reasonable estimate of the consequences; and

[2] Including accident sequences that involve the releases of hazardous chemicals, as defined in 10 CFR 70.4.

(3) Uses reasonably generic assumptions and data that are reasonably conservative for the types of accidents analyzed.

d. Likelihood Evaluation Method: The method for evaluation of the likelihood of accident sequences, as described in the ISA Summary, is considered acceptable if it provides reasonable assurance that the IROFS and management measures described comply with the graded performance criteria of 10 CFR 70.61; and the method is consistent with acceptable definitions of the likelihood terms in accordance with Item vii below. Specific criteria are:

(1) The method includes clearly showing how each IROFS involved acts to prevent or mitigate the accident sequence being evaluated.

(2) When multiple IROFS are involved in an accident sequence, the method considers the interaction of all the IROFS involved, as in a logic diagram or tabulation, that accounts for the impact of redundancy, independence, and surveillance to correct failures on the likelihood of occurrence of the accident.

(3) The method has objective criteria for evaluating, at least qualitatively, the likelihood of failure of individual IROFS. Such likelihood criteria should include the following when applicable: means to limit potential failure modes, the magnitude of safety margins, the type of engineered equipment (active or passive) or human action that constitutes the IROFS, and the types and grading, if any, of the management measures applied to the IROFS.

(4) Finally, the method evaluates each accident sequence as unlikely, highly unlikely, or neither, as defined by the applicant in accordance with Item vii below.

(5) For nuclear criticality accident sequences, the method evaluates compliance with 10 CFR 70.61(d). That is, even in a facility with engineered measures to limit the consequences of nuclear criticalities, preventive control(s) must be in place sufficient to assure subcriticality for credible abnormal events. A moderately higher standard of likelihood may be permitted in preventing such events consistent with ANSI/ANS–8.10, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors." In particular, criticality cannot result from any single administrative error. In addition, criticality accidents must meet an approved margin of subcriticality for safety. Acceptance criteria for such margins are reviewed as programmatic commitments, but the ISA methods and Summary must consider and document the magnitude of those margins when they are part of the reason why exceedance of safety limits is unlikely.

One acceptable method of likelihood evaluation is described in Appendix A.

vi. Quantitative Standards for Chemical Consequences

The applicant's proposed quantitative standards to assess consequences from acute chemical exposure to licensed material or chemicals produced from licensed material include:

a. Three unambiguous quantitative standards for each of the applicable hazardous chemicals onsite corresponding to each of: [1] 10 CFR 70.61(b)(4)(i); [2] 10 CFR70.61(b)(4)(ii) and 10 CFR 70.61(c)(4)(i); and [3] 10 CFR 70.61(c)(4)(ii).

b. The quantitative standard for 10 CFR 70.61(b)(4)(i) correctly categorizes as such, all exposures that could endanger the life of a worker. The applicant is appropriately conservative in applying the language "could endanger," so as to include exposures that would result in death for average and susceptible persons, but not for hyper-susceptible persons, consistent with the methods used for the Environmental Protection Agency Acute Exposure Guidelines.

c. The quantitative standard for 10 CFR 70.61(b)(4)(ii) and 10 CFR 70.61(c)(4)(i) correctly categorizes as such all exposures that could lead to irreversible or other serious, long-lasting health effects to individuals. Similar to Item (b), the standard should have appropriate conservatism.

d. The quantitative standard for 10 CFR 70.61(c)(4)(ii) correctly categorizes as such all exposures that could cause mild transient health effects to an individual.

The staff finds the use of the Emergency Response Planning Guidelines (ERPG) and Acute Exposure Guideline Level (AEGL) series of standards to be acceptable sets that meet the performance criteria of 10 CFR 70.61. When the applicant chooses to select ERPG or AEGL values, a reference to this fact is sufficient. However, if such standards are not available for all of the applicant's chemicals or if the applicant opts to select another standard, the ISA Summary lists the actual values the applicant selected for each chemical and provides information or a reference justifying that the selected standards meet Items (a) through (c).

vii. Definitions of Likelihood

The regulations of 10 CFR 70.65 require the applicant's ISA Summary to provide definitions of the terms unlikely, highly unlikely, and credible. The applicant's definitions of these terms is acceptable if, when used with the applicant's method of assessing likelihoods, they provide reasonable assurance that the performance requirements of 10 CFR 70.61 can be met. The applicant's method of likelihood evaluation and the definitions of the likelihood terms are closely related. Qualitative methods require qualitative definitions. Such a qualitative definition would identify the qualities of IROFS controlling an accident sequence that would qualify that sequence as "unlikely" or "highly unlikely."

An applicant may use quantitative methods and definitions for evaluating compliance with 10 CFR 70.61. It is recommended that, in any case, the reviewer focus on objective qualities and information provided concerning accident likelihoods.

Also, 10 CFR 70.61 requires that an applicant's credible high-consequence events be highly unlikely. Thus the meaning of the phrase "highly unlikely" is on a per event basis. The same is true for the terms "unlikely" and "credible." Hence, applicant definitions should be on a per event basis. The events referred to are occurrences of consequences, which is herein synonymous with the phrase "accident sequence." This is important to recognize since there may be hundreds of potential accident sequences identified in an ISA. Thus, the likelihood of each individual sequence must be quite low.

a. Acceptance Criteria for the Definition of "Credible"

The regulations in 10 CFR 70.65 require that the applicant define the term "credible." This term "credible" is used in 10 CFR 70.61 to state the performance requirements that all credible events be controlled to be unlikely or highly unlikely, as appropriate. Thus, to be "not credible" could be used as a criterion for exemption from use of controls. There is a danger of circular reasoning here. In the safety program embodied in the rule, the fact that an event is "not credible" must not depend on any plant feature that could credibly fail to function, or be rendered ineffective as a result of a change to the system. Each plant feature that is needed to assure that accident events are sufficiently unlikely is an "item relied on for safety" (IROFS). There must be high assurance, provided by management measures, that such features are not removed or rendered ineffective during system changes. One cannot claim that a process does not need IROFS because it is "not credible" due to characteristics provided by IROFS.

Nevertheless, there are events, including external events and some types of plant upsets, that have inherent qualities that clearly make them not credible, even in the absence of management measures. The applicant may define such events by describing what qualities they must possess to be not credible.

Three acceptable sets of qualities that define an event as not credible are:

(1) An external event whose frequency of occurrence can conservatively be estimated as less than once in a million years;

(2) A process deviation that consists of a sequence of many unlikely human actions or errors for which there is no reason or motive. In determining that there is no reason for such actions, consideration must have been given to a wide range of possible motives, short of intent to cause harm. Necessarily, no such sequence of events can ever have actually happened in any fuel cycle facility.

(3) Process upsets for which there is a convincing argument, based on physical laws, that are not possible, or are unquestionably extremely unlikely. The validity

of the argument must not be dependent on any feature of the design or materials which is not controlled by the plant's system of IROFS.

The following discusses a further consideration for evaluating the acceptability of the applicant's definition of "credible." The implication of the use of "credible" in 10 CFR 70.61 is that events which are not "credible" may be neglected. For this to be acceptable on a risk basis, unless the event is impossible, it must be of negligible likelihood, that is, "not credible" must mean impossible, practically impossible, or of negligible likelihood. Negligible likelihood means sufficiently low that, considering the consequences, the addition to total risk is small. Note that consideration must thus be given to how many such events have, in fact, been neglected.

An applicant may demonstrate by quantitative reasoning, that a particular event is of negligible frequency. Such a demonstration must be convincing despite the absence of designated IROFS. Typically, this can only be achieved for external events known to be extremely unlikely.

b. Acceptance Criteria for Qualitative Definitions of Likelihood

If the applicant's definitions are qualitative, they are acceptable to the extent that they are: (1) reasonably clear and based on objective criteria; and (2) can reasonably be expected to consistently distinguish accidents that are highly unlikely from those that are merely unlikely.

The phrase "objective criteria" means the extent to which the method relies on specific identifiable characteristics of a process design, rather than subjective judgements of adequacy. Objective criteria are needed to achieve consistency. Consistency means the degree to which the same results are obtained when the method is applied by different teams of analysts. This is important in order to maintain an adequate standard of safety because ISAs of future plant modifications may be performed by individuals not involved in the initial ISA.

(1) Reliability and Availability Qualities

Qualitative methods of evaluating the likelihood of an accident sequence involve identifying the reliability and availability qualities of each of the events that constitute the sequence. The following lists of qualities is not necessarily complete, but contains many of the factors most commonly encountered. Some of these qualities relate to the characteristics of individual IROFS, such as:

(a) Safety margin in the controlled parameter compared to process variation and uncertainty;

(b) Whether the IROFS is an active engineered control, a passive engineered control, an administrative control, or an enhanced administrative control;

(c) The type and grade of management measures applied to the control;

(d) Fail-safe, self-announcing, or surveillance measures to limit down time;

(e) Failure modes;

(f) Demand rate; and

(g) Failure rate.

Other reliability qualities relate characteristics of the system of IROFS protecting against the accident sequence as a whole, such as:

(a) Defense-in-depth;
(b) Degree of redundancy;
(c) Degree of independence;
(d) Diversity; and
(e) Vulnerability to common cause failure.

(2) Qualitative Methods

Methods of likelihood evaluation, and the definitions of the rule's likelihood terms, may mix qualitative and quantitative information. Certain types of objective quantitative information may be available concerning specific processes in a plant. Some examples of such objective quantitative information are:

(a) Reports of failure modes of equipment or violations of procedures recorded in maintenance records or corrective actions programs;

(b) The time intervals at which surveillance is conducted to detect failed conditions;

(c) The time intervals at which functional tests or configuration audits are held;

(d) For a fail-safe, monitored, or self-announcing IROFS, the time it takes to render the system safe;

(e) Demand rates, that is, how frequently process operations are conducted which place a demand on an IROFS. Some situations amount to effectively continuous demand.

Such items of quantitative information should be considered in evaluating the likelihood of accident sequences, even in purely qualitative evaluations. For example, knowing the value to which down time is limited by surveillance can indicate that a system's availability is extremely high. For redundant systems,

such high availability can virtually preclude concurrent independent failures of the multiple controls.

(3) Likelihood Indexing Methods

One acceptable type of definition for the likelihood terms "unlikely" and "highly unlikely" could be based on a risk indexing method. Such a method is described in the example in Appendix A. The example described in Appendix A is intended to rely primarily on a qualitative evaluation of reliability/availability factors. In such methods, qualitative characteristics of the system of IROFS, such as those listed above, are used to set a quantitative likelihood index for each accident sequence. The definition of "unlikely" then is an acceptable limit on this likelihood index.

(4) Purely Qualitative Methods

A purely qualitative method of defining "unlikely" and "highly unlikely" is acceptable if it incorporates all of the applicable reliability and availability qualities to an appropriate degree. For example, one statement of applicable qualities is double contingency protection:

Double Contingency Protection: The quality of a process design that incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Double contingency addresses explicitly several reliability/availability qualities; namely:

(a) Factors of safety—safety margins;
(b) At least two—redundancy;
(c) Unlikely—low failure rate, low down time;
(d) Concurrent—low down time;
(e) Independent—independence; and
(f) Process conditions—physical events, not virtual human errors.

One acceptable definition of highly unlikely is a system of IROFS that possesses double contingency protection with each of the applicable qualities to an appropriate degree. For example, as implied by the modifier, "at least," sometimes more than just two-fold redundancy may be appropriate.

A qualitative method may also be proposed for defining "unlikely." Such a qualitative method might simply list various combinations of reliability qualities for a system of IROFS that would qualify as "unlikely." For example, single high reliability IROFS, such as engineered hardware controls with high grades of applicable management measures might qualify as an acceptable definition of

unlikely. Systems relying on administrative controls would normally have to make use of enhancing qualities such as large safety margins and redundancy in order to qualify as unlikely. A single simple administrative control, regularly challenged, without any special safety margin or enhancement, where a single simple error would lead to the accident, would not qualify as "unlikely" to fail.

c. Acceptance Criteria for Quantitative Definitions of Likelihood

An applicant, although not required to do so, may choose to provide quantitative definitions of the terms unlikely and highly unlikely. Quantitative guidelines are developed below. These guidelines serve two purposes: (1) they can be used as acceptance criteria for quantitative definitions, if provided; and (2) they provide guidance to the reviewer when objective quantitative reliability/availability information exists. The reviewer is cautioned not to interpret these guidelines as requiring that quantitative definitions or evaluations are required.

The goals from which these quantitative guidelines were derived are for specific types of accidents. Therefore the guidelines should not be used for accidents that differ significantly from these specific types. The high consequence guideline, for example, is based on a goal of no inadvertent criticalities. Thus it is only appropriate to use this guideline for accidents whose consequences are similar to a nuclear criticality accident, that is, one where a few fatal or near fatal worker doses may occur. For substantially more severe high consequence accidents, more stringent likelihood criteria would be acceptable. For less severe high consequence accidents, less stringent criteria may be applied. It should also be noted that the quantitative guidelines are derived from goals, not limits, and have been judged to be the highest values consistent with those goals.

(1) Quantitative Guidelines

The development of quantitative guidelines here does not imply that quantitative demonstration of compliance with 10 CFR 70.61 is required. As stated above, the phrase "highly unlikely" applies on a "per accident" basis. Hence, quantitative frequency guidelines for the likelihood definitions depend on how many potential accidents there are in each of the two categories. The quantitative guidelines stated below are derived from safety performance goals for the whole industry. The number of potential accidents for the whole industry will not be known until ISA results are available. For this reason, the quantitative guidelines provided below are expressed in terms of two variables, Nh and Ni. Nh is the total number of potential high-consequence accidents for the industry; and Ni is the number of intermediate-consequence accidents, as identified in the ISAs.

Since the numbers of potential accidents will not initially be known, for review of early applications, the staff should use values of Nh and Ni that are sufficiently high to allow for the contribution not just of the application being reviewed but of

the entire group of potential applicants under Part 70. If the number of accidents actually identified in all the industry ISAs exceeds these values significantly, adjustments may be needed.

(a) Highly Unlikely: The guideline for acceptance of the definition of "highly unlikely" has been derived as the highest acceptable frequency that is consistent with a goal of having no criticality accidents, and no accidents of similar consequences, in the industry. To within an order of magnitude, this is taken to mean a frequency limit of less than one such accident in the industry every 100 years. This has been translated below into a guideline limiting the frequency of individual accidents. The goal is to have no such accidents, thus it is reasonable to reduce accident frequencies substantially below these guidelines when feasible.

(b) Unlikely: Intermediate consequence events include significant radiation exposures of workers, those exceeding 0.25 Sieverts (25 rem). It is taken as a goal that there be no increase in the rate of such significant exposures. This rate is currently about one exposure per 2.5 years. Since the uranium fuel cycle industry has not contributed to such exposures, an allocation of one tenth of this value, or 0.04 per year has been used as appropriate for this industry. Once adjusted to a per accident basis, this value of 0.04 per year for the industry can then be used as an appropriate guideline limiting all types of accidents with intermediate consequences, because their health consequences are all comparable. The definition and use of the term "unlikely" should be consistent with this frequency guideline.

(2) Quantitative Guidelines for Use with Acceptance Criteria

Subject to the guidance above, the applicant's quantitative definitions of the terms unlikely and highly unlikely, as applied to individual accident sequences identified in the ISA, are acceptable for showing compliance with 10 CFR 70.61 if they are reasonably consistent with the following quantitative guidelines:

Likelihood Term of § 70.61	Guideline
unlikely	less than 0.04/Ni per year
highly unlikely	less than 10^{-2}/Nh per year

where:

Ni = the total number of potential intermediate-consequence accidents in regulated facilities. Although it is currently expected that Ni may be very low for the uranium fuel cycle industry, a value of at least 10 should initially be assumed.

Nh = the total number of potential high-consequence accidents in regulated facilities. Although currently not known, the value of Nh should initially be assumed to be at least 1000.

It should be noted that the stated quantitative guidelines are used to define the largest likelihood values that would be acceptable limits. Definitions based on lower limits are also acceptable. The performance requirements of 10 CFR 70.61 are limits, not goals, thus staff should use these guidelines in that sense.

The quantitative consequence categories defined in 10 CFR 70.61 are broad, especially the "high-consequence" category, which is open ended. For this reason, the meaning of "highly unlikely" for an individual accident should be graded in inverse proportion to the magnitude of consequences when these consequences are significantly greater than the lower limits defining high consequences in 10 CFR 70.61.

viii. Information Demonstrating Compliance with the Performance Requirements

Items 3, 4, 6, and 8 of 10 CFR 70.65(b) require the applicant to describe certain information in the ISA Summary that results from the ISAs the applicant performed on individual processes. Item 4 of 10 CFR 70.65(b) requires that the applicant's ISA Summary contain: "information that demonstrates compliance with the performance criteria of 10 CFR 70.61." Since the requirements of 10 CFR 70.61 are expressed in terms of consequences and likelihoods of events, the information needed is that which shows that all events are of appropriate consequences and likelihood. The requirements of 10 CFR 70.61 effectively states that each credible accident sequence must have a likelihood corresponding to its consequences. Thus the information submitted is acceptable if it provides consequence and likelihood information for each accident showing that: high consequence events are highly unlikely and intermediate consequence events are unlikely.

The performance criteria of 10 CFR 70.61 have three elements: (1) completeness, (2) consequences, and (3) likelihood. Completeness refers to the fact that the applicant must address each credible event. Consequences refers to the magnitude of the chemical and radiological doses used by the applicant to categorize accidents as being of high or intermediate consequences. Likelihood refers to the fact that 10 CFR 70.61 requires that the applicant must demonstrate that intermediate consequence events will be unlikely, and high consequence events will be highly unlikely.

To be acceptable, the information the applicant provides must correspond to the applicant's ISA methods, consequence, and likelihood definitions, which are also described in the submittal. The applicant's information must show the basis and the results of applying these methods to each process. In addition, the applicant's information must show that the methods have been properly applied in each case.

The applicant may present information showing completeness, consequences, and likelihood for accident sequences in various formats, including logic diagrams or tabular summaries.

a. Completeness: The applicant demonstrates completeness by correctly applying an appropriate method of accident identification, as described in NUREG–1513, "Integrated Safety Analysis Guidance Document." Completeness can be effectively displayed by using an appropriate diagram or description of the accidents identified.

b. Consequences: The applicant's consequences demonstrate compliance with 10 CFR 70.61 if:

   (1) The applicant's ISA Summary includes, for each accident, an estimate of its quantitative consequences (doses, chemical exposures, criticality) in a form that can be directly compared to the consequence levels in 10 CFR 70.61, or includes a reference to a value documented elsewhere in the ISA Summary that applies to or bounds that accident;

   (2) The applicant calculated the consequences using an acceptable method described in the ISA Summary methods section (see Section 5.4.3.2(B)(v));

   (3) The applicant used reasonably conservative estimates for source terms and other process specific data used for the type of accident and provided intermediate data. For example, for consequence analysis the applicant would provide intermediate data such as the inventory of hazardous material and the facts about the accident that result in release path reduction factors;

   (4) The applicant's ISA Summary correctly assigns each type of accident to one of the consequence categories of 10 CFR 70.61–namely, high, intermediate, or low (less than intermediate); and

   (5) The applicant assigns unshielded criticality accidents as high consequence events. For processes with effective engineered shielding, criticalities may produce doses below the intermediate consequences of 10 CFR 70.61. As stated in the regulation, the applicant must place primary reliance on the prevention of criticalities. This applies notwithstanding shielding or other mitigative features. Therefore, regardless of the actual consequences, shielded criticalities must meet likelihood criteria, as described in the following section of this SRP. If needed, the *Nuclear Fuel Cycle Facility Accident Analysis Handbook* (NUREG/CR–6410) provides methods for estimating magnitudes of criticality events that can be applied for workers or members of the public at varying distances from the event.

c. Likelihood: The applicant's likelihoods demonstrate compliance with 10 CFR 70.61 if:

(1) The applicant provides an evaluation of the likelihood of each type of accident sequence in the ISA Summary;

(2) The applicant evaluated likelihoods in accordance with an acceptable method–see the regulatory acceptance criteria for ISA methods (see Section 5.4.3.2(B)(v));

(3) The applicant's evaluated likelihoods comply with acceptable definitions of the terms "unlikely" and "highly unlikely" from the applicant's ISA Summary as evaluated in Section 5.4.3.2(B)(vii) of this SRP. Note that, when interpreted as required accident frequencies, these terms refer to long-run average frequencies, not instantaneous values, that is, a system complies with the performance requirements of 10 CFR 70.61 as a long-run average. Otherwise failure of any IROFS, even for a very short period, would be a violation of the requirement, which is not the intent;

(4) The applicant evaluates all nuclear criticality accident sequences with a likelihood of "highly unlikely," e.g., with double contingency protection as required by the baseline design criterion for nuclear criticality safety in new facilities (see 10 CFR 70.64(a)(9)). If the applicant proposes an exception to double contingency, the reviewer should refer to Section 6.4.3.3.5(C) of this SRP.

(5) The applicant must have preventive control(s) in place sufficient to ensure subcriticality for credible abnormal events. A moderately higher standard of likelihood may be permitted in preventing such events consistent with ANSI/ANS Standard 8.10. In addition, 10 CFR 70.61(d) requires that the risk of criticality must be limited by an approved margin of subcriticality for safety. Validation methods to establish margins to assure that a particular parameter value is actually subcritical are reviewed as programmatic commitments, not as part of the ISA. However, when a safety margin is part of the reason why exceedance of safety limits is unlikely, the margin should be listed in the ISA Summary description of that accident. For example, if the process is safe against double batching, the number of batches, and other conditions, required for actual criticality should be described in the ISA Summary. The likelihood of erroneously accumulating the critical number of batches should then be reflected in the evaluation of the likelihood of the accident sequence. (See Chapter 6.0 of this SRP for additional information on nuclear criticality safety.)

ix. Process Hazards

The applicant's description of the process hazards, as provided in the ISA Summary, identify for each process, all the types of hazards relevant to determining compliance with the performance criteria of 10 CFR 70.61, that is, the acceptance criterion is completeness. The applicant should list the identified hazards that could credibly result in the minimum consequences on 10 CFR 70.61 even if later analyses of a particular

hazard show that no accident sequence exists that could exceed these minima. The applicant may generally exclude certain hazards from further consideration for the entire facility if the applicant can justify by bounding case analyses that, for the conditions or credible inventories onsite, the minimum consequence levels of 10 CFR 70.61 cannot be exceeded. If the applicant uses such a justification, the applicant's bounding inventories or conditions, if under the control of the applicant, must be identified as IROFS. The applicant's hazard identification is consistent with the provisions outlined in Section 5.4.3.2(B)(v)(a).

x. Types of Accident Sequences

The applicant's description of the accident sequences permit the staff to determine that (1) the applicant identified all the accidents that could exceed the consequence criteria of 10 CFR 70.61 and (2) the applicant identified how the IROFS listed in the ISA Summary protect against each type of accident.

Types of accidents differ if they consist of a different set of failures of IROFS. Thus the applicant may summarize several processes using a set of IROFS that are functionally of the same type (same mechanical, physical and/or electrical principle of operation) as a single type of accident. However, the applicant should individually identify the individual processes covered by this system in a way that the reviewer can determine completeness in addressing all processes. For this reason, the applicant should not, in general, merely list the type of hazard, or just the controlled parameters, without reference to the IROFS controlling that parameter or hazard. The applicant's general description of accident sequences should cover all types of sequences of initiating events. Initiating events may be either failure of an IROFS or an external event. Human errors can be initiating events or failures of IROFS. The applicant's accident description is acceptable if it permits the staff to determine how each accident sequence that could exceed the minimum consequence levels in 10 CFR 70.61 is protected against by IROFS.

The applicant may do this by: showing a fault tree where the basic events are failures of the IROFS; providing a table where each row displays the events in an accident sequence, where, in general, each event is failure of an IROFS; or a narrative summary for each process describing the sequence of events in each type of accident. Refer to Appendix A to this SRP for examples.

The general description of types of accident sequences, to show completeness, must use systematic methods and consistent references. Therefore, each description is acceptable if:

a. A method of hazard identification and process hazard analysis was used in accordance with the criteria of NUREG–1513;

b. The method selected was correctly applied;

c. No hazard or accident sequence that could cause a failure to meet 10 CFR 70.61 was overlooked; and

d. A method of identifying plant processes was used, so that the completeness of the analysis in covering all processes can be evaluated.

When the applicant identifies accident sequences through the PHA, the applicant may identify accidents whose consequences may initially be unknown, then later are analyzed and shown to be below the consequence levels identified in 10 CFR 70.61. The applicant's ISA Summary must either list all the accidents identified or state that certain accidents are possible, but were not listed due to insufficient consequences. However, the applicant need not list every conceivable permutation of accidents as a separate accident sequence. The applicant may group accidents having characteristics that all fall in the same category as a single type of accident, if: (a) the initiating events have the same type of effect on the system, (b) they all consist of failure of the same IROFS, (c) they all result in violation of the safety limit on the same parameter, and (d) they all result in the same type and severity categories of consequences. A primary purpose of showing completeness is to assure that existing IROFS are adequate. Once the applicant demonstrates that a type of accident has the same characteristics, it is not necessary for the applicant to distinguish among the different events within the type. On the other hand, if a different initiating event poses a different type of challenge to a control, then the applicant should address that initiating event separately, because it may reveal a weakness of the control.

xi. List of IROFS

The applicant's "list describing items relied on for safety" required by 10 CFR 70.62(c)(vi):

a. Includes all IROFS in the identified accident sequences.

The primary function of the "list describing all items relied on for safety" is to document the safety basis of all processes in the facility to assist in assuring that these items are not degraded or removed without a justifying safety review. One example of a tabular description of IROFS meeting these criteria is in Appendix A to this SRP. No item, aspect, feature, or property of the processes that is needed to show compliance with the safety performance requirements of the regulation may be left off this list. IROFS may be hardware with a dedicated safety function or hardware with a property that is relied on for safety. Thus IROFS may be the dimension, shape, capacity, or composition of hardware. In some processes, the frequency of demands made on IROFS must be controlled or limited to comply with 10 CFR 70.61. In such processes, whatever features are needed to limit the frequency of demands are themselves IROFS.

b. The description of the IROFS.

The description of the IROFS, the identification of the grade of management measures applied to them, and the associated safety limits and margins is adequate to permit a determination of compliance with 10 CFR 70.61, that is, it includes the characteristics of its preventive, mitigative, or other safety function, and the assumptions and conditions under which the item is relied upon to support compliance with the performance requirements of 10 CFR 70.61.

The applicant describes the essential features of each IROFS that are required to achieve adequate reliability. The applicant should provide sufficient information about engineered hardware controls to permit an evaluation that, in principle, controls of this type will have adequate reliability. Because the likelihood of failure of IROFS often depends on safety margins, the applicant should, in general, describe the safety parameter controlled by the item, the safety limit on the parameter, and the margin to true failure. For IROFS that are administrative controls, the applicant should sufficiently describe the nature of the action or prohibition involved to permit an understanding that, in principle, adherence to it should be reliable. The applicant should indicate features of the IROFS that affect its independence from other IROFS, such as reliance on the same power supplies.

The description of each item must contain any information needed to identify how the management measures, such as maintenance, training, configuration management, etc., are applied to it. If the applicant uses a system of graded management measures, the grade applied to each control should be determinable from information provided. The regulations in 10 CFR 70.62(d) require that applicants "...establish management measures to provide continuing assurance of compliance with the performance requirements of 10 CFR 70.61." The reliability required for an IROFS is proportionate to the amount of risk reduction relied on. Thus the quality of the management measures applied to an IROFS may be graded commensurate with the reliability required. The management measures shall assure that IROFS are designed, implemented, and maintained, as necessary, to be available and reliable to perform their function when needed. The degree of reliability and availability of IROFS assured by these measures should be consistent with the evaluations of accident likelihoods. In particular, for redundant IROFS, all information necessary to establish the average vulnerable outage time is required in order to maintain acceptable availability. Otherwise failures must be assumed to persist for the life of the plant. In particular, the time interval between surveillance observations or tests of the item should be stated since restoration of a safe state cannot occur until the failure is discovered.

Although the regulations do not explicitly list the content and grading of management measures as a separate element of an ISA Summary, such information is required to "demonstrate compliance with the performance requirements" by the IROFS. Normally this information would be available in the license application. If sufficiently detailed information is not provided with the application, submittal of additional information may be required.

xii. List of Sole Items Relied on for Safety (IROFS)

A subset of the complete list of all IROFS that identifies the IROFS that are the sole item for preventing or mitigating an accident sequence. The subset includes a descriptive title of the item, provides an unambiguous and clear reference to the process to which the item applies, and provides a clear and traceable reference to the description of the item as it appears in the list of all IROFS described in Item xi.

xiii. Information Demonstrating Compliance with the Requirements of 10 CFR 70.24 for Criticality Monitoring

The regulations in 10 CFR 70.24 have specific sensitivity requirements for criticality monitors. To demonstrate compliance, the applicant should describe its method for evaluating an acceptable response of at least two detectors to a criticality at any location where SNM may be handled, used, or stored. The applicant should provide a diagram of the locations of all detectors relative to the potential locations of SNM. Information supporting determination of the gamma and neutron emission characteristics of the minimum credible accident of concern capable of producing the effects specified in 10 CFR 70.24 should be provided. The applicant should provide the actual neutron and gamma doses and dose rates at the detector locations as well as information showing the response characteristics of the detectors to neutron and gamma doses and rates characteristic of credible accidents.

The regulations in 10 CFR 70.24 also require specific emergency preparations. The applicant should provide information demonstrating that its equipment and procedures are adequate to assure that these requirements are met.

xiv. Information Demonstrating Compliance with Requirements of 10 CFR 70.64 for New Facilities

The regulations in 10 CFR 70.64 specify baseline design criteria that an applicant must use, as applicable, for new facilities. If the application involves such new facilities or process, then an acceptable set of information would address each baseline design criterion listed in 10 CFR 70.64 and would show how the criterion is met. For criteria such as double contingency to which each individual process must comply, the process-specific information may be provided along with the other process information in the ISA Summary. Design bases events and safety parameter limits should be given. Methods, data, and results of analysis showing compliance with these design bases should be given for individual processes and structures. Specific acceptance criteria for the baseline design criteria are given in other chapters of this SRP.

5.5 REVIEW PROCEDURES

5.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 5.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use SNM.

Guidance specific to the application for construction approval review and the review for the license to possess and use SNM is provided below.

A. Construction Approval

Specifically, the application for construction approval should address Section 5.3.1.

B. License To Possess and Use SNM

Specifically, the license application should address Section 5.3.2.

If the primary reviewer verifies that the subject area material is adequately addressed for the construction approval review or the review for the license to possess and use SNM, the primary reviewer should accept the application for the safety evaluation in Section 5.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

5.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 5.5.1(A) (construction approval review) or Section 5.5.1(B) (review for a license to possess and use SNM), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 5.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

Guidance specific to construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

The primary reviewer should review the design bases of the principal SSCs and the safety assessment of the design bases. The primary reviewer should coordinate with the secondary reviewers to ensure consistency between the review conducted under this chapter and reviews of the design bases and safety assessment of the design bases conducted for other subject areas, that is, Chapters 6.0 through 15.0.

B. License To Possess and Use SNM

i. The primary reviewer should review the ISA programmatic commitments, as described in the license application, and the ISA results, as described in the ISA Summary. The primary reviewer should coordinate with the secondary reviewers to ensure consistency between the review conducted under this chapter and the review conducted under other chapters. For example, the primary reviewer of the ISA Summary should coordinate with the primary reviewer of nuclear criticality safety (NCS) to ensure that NCS is consistent throughout the license application.

ii. The primary reviewer should evaluate the risk significance of the accident sequences using the risk indices from Appendix A, which provides an example for evaluating risk significance. For accident sequences categorized as lower risk significance, the primary reviewer selects a representative sample of sequences for specific evaluation, while the remainder receive a less detailed review.

iii. The primary reviewer should coordinate with the secondary reviewer who is reviewing Chapter 15.0, "Management Measures," to ensure that the management practices proposed by the applicant are consistent with the material submitted in support of Chapter 15.0.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the input for the Safety Evaluation Report (SER), as described in Section 5.6 using the acceptance criteria from Section 5.4. The secondary reviewers should coordinate the input with the balance of the reviews and the SER.

5.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

The staff reviewed the material submitted for construction approval for [insert facility name] according to Chapter 5.0 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found [insert a summary statement of the findings]. The staff found that the applicant's safety assessment of the design bases demonstrates that the applicant's principal structures, systems, and components will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents. The staff concluded that the applicant's safety assessment of the design bases shows that it meets the requirements for issuing a construction approval in accordance with 10 CFR Part 70.

The staff could document the safety evaluation for license to possess and use SNM as follows:

The staff reviewed the ISA programmatic commitments in the license application and ISA Summary for [insert facility name] to possess and use SNM according to Section 5.0 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found [insert a summary statement of the findings]. The staff verified that the applicant performed an ISA to identify and evaluate the hazards and potential accidents associated with the facility, and to establish engineered and administrative controls to ensure facility operation will be within the bounds of the ISA.

The staff confirmed that the applicant's license application contains appropriate commitments, including commitments to: (1) compile and maintain process safety information; (2) engage personnel with appropriate training to conduct the ISA; (3) use appropriate methods to conduct the ISA; and (4) implement appropriate measures and procedures to ensure that the ISA stays accurate and up-to-date.

The staff confirmed that the applicant's ISA Summary (1) identified all hazards at the facility; (2) analyzed for accident sequences through the use of process hazards analysis; (3) evaluated and assigned consequences to the accident sequences; and (4) evaluated the likelihood of each accident consistent with the guidance in NUREG–1718. Moreover, the applicant identified all items relied on for safety, including administrative and engineered controls. As a result, the NRC staff concluded that the applicant's postulated accidents resulting from the facility hazards that may be anticipated to occur (or are considered unlikely or highly unlikely) should be in compliance with the performance requirements of 10 CFR Part 70.

The staff concludes that (1) the identification and evaluation of the hazards and accidents as part of the ISA and (2) the establishment of controls to maintain safe facility operation from their consequences meet the requirements for a license to possess and use SNM under 10 CFR Part 70, and provide reasonable assurance that the health and safety of the public, the workers, and the environment will be adequately protected.

5.7 REFERENCES

American Institute of Chemical Engineers (AIChE). "Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples." AIChE: New York, NY. 1992.

American National Standards Institute/American Nuclear Society (ANSI/ANS). ANSI/ANS–8.1–1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." ANSI: La Grange Park, Illinois. September. 1983.

———. ANSI/ANS–51.1–1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." ANSI: La Grange Park, Illinois. 1983.

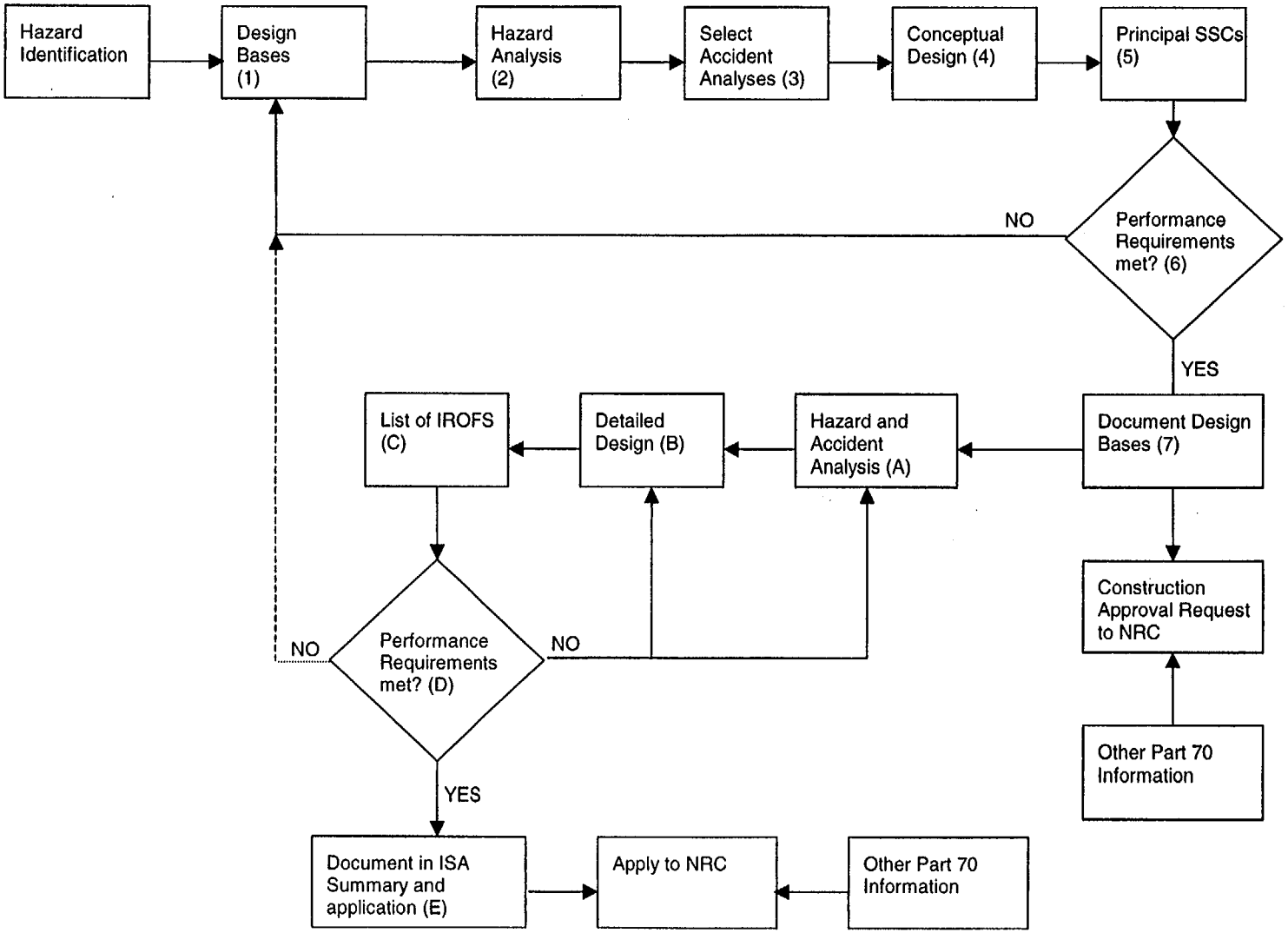
Census Bureau (U.S.) (CB). Table No. 688.1995, "Statistical Abstract of the United States (115th Edition)." U.S. Government Printing Office, Washington, D.C. 1995

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

Nuclear Regulatory Commission (U.S.), Washington, DC. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338-41357. July 30, 1999.

Nuclear Regulatory Commission (U.S.) (NRC). NUREG–1513, "Integrated Safety Analysis Guidance Document." NRC: Washington, D.C. 1995

Figure 5-1: Relationship between the safety assessment of the design bases (Steps 1-7) and the ISA (Steps A-D)



6.0 NUCLEAR CRITICALITY SAFETY (NCS)

6.1 PURPOSE OF REVIEW

The purpose of this review is to determine whether the applicant, in the license application and supported by materials on the docket, has (1) established an adequate organization with which to implement the NCS program; (2) established an adequate NCS program to ensure safe operation of the facility; (3) implemented adequate controls and limits on parameters relied on to prevent nuclear criticality; and (4) assessed accident sequences identified in the Criticality Safety Evaluations (CSEs) and documented in the integrated safety analysis (ISA) leading to a nuclear criticality, as required by 10 CFR Part 70.

6.2 RESPONSIBILITY FOR REVIEW

Primary: Nuclear Process Engineer (NCS Reviewer)

Secondary: Chemical Safety Reviewer

Supporting: Project Manager and Fuel Cycle Inspector (as needed)

6.3 AREAS OF REVIEW

The staff should review the application to determine whether the applicant has (1) described an adequate NCS program; (2) implemented the facility management measures; (3) identified and committed to the responsibilities and authorities for individuals implementing the NCS program; and (4) established an adequate criticality accident alarm system (CAAS).

6.3.1 Organization and Administration

The primary reviewer should review the applicant's organization and administration to determine whether the applicant has identified the responsibilities and authorities for organizations and individuals implementing the NCS program. This review should include:

A. For familiarity, the general administrative organization methods used by the applicant.

B. The administrative organization of the NCS program, including authority and responsibilities of each position identified, and organizations and individuals with responsibility for NCS.

C. Experience and education requirements of management and staff positions with NCS responsibility.

6.3.2 Management Measures

The primary reviewer should review the applicant's management measures in support of the applicant's ability to implement and maintain the NCS program, and to ensure the continued availability and reliability of items relied on for safety (IROFS). The following areas of the application related to the applicant's management measures should be reviewed:

A. Management functions discussed in SRP Sections 15.1 through 15.8, specifically as they relate to NCS.

B. The commitment to measures implementing the requirements of 10 CFR 70.64 (Baseline Design Criteria) to ensure that the initial facility design meets these baseline design criteria (BDC) for NCS.

C. The implementation of the requirements of 10 CFR 70.72 (Facility Change and Change Process) to ensure that: (1) facility changes are managed to maintain the integrity of the facility's safety basis and to ensure they receive the appropriate level of NCS review in accordance with § 70.72(a) and § 70.72(b) and (2) facility changes requiring NRC approval in accordance with § 70.72(c) are appropriately identified and treated.

6.3.3 Technical Practices

The primary reviewer should review the applicant's implementation of NCS technical practices to ensure the safe operation of the facility. This review should include:

A. The commitment to derive and implement NCS controls and limits in accordance with technical practices as described in the application, by incorporating them into the applicant's NCS program.

B. Technical practices, including a description of the management measures that ensure operability of the CAAS and emergency response procedures.

C. The technical practices to ensure that limits on controlled parameters have an adequate safety margin. These practices should include those to ensure that the methods used to develop NCS limits are properly validated.

D. The technical practices to ensure that sufficient NCS controls, developed in the CSEs and flowed into the ISA, are identified for each process.

E. The areas of review listed in Section 5.3 as they relate to NCS, specifically: (1) potential accident sequences that could result in nuclear criticality; (2) specific controls relied on to provide reasonable assurance that an inadvertent criticality will not occur; and (3) a demonstration that the likelihood of failure is sufficiently low so as to demonstrate compliance with the double contingency principle.

F. The commitment to prepare and maintain applicable safety basis documentation in enough detail so that criticality controls and double contingency analysis can be reviewed and inspected by NRC and licensee staff.

6.4 ACCEPTANCE CRITERIA

To provide for NCS, the applicant's use of standards should be considered acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application.

If an applicant intends to conduct activities where a standard applies and the standard has been endorsed by an NRC Regulatory Guide (RG), then a commitment to comply with all of the requirements (i.e., "shalls") and the appropriate recommendations (i.e., "shoulds") of the standard should constitute an acceptable program under the NRC regulations with respect to the safety aspects addressed by the standard. If the applicant does not intend to comply with all recommendations in the standard, alternative methods of meeting the intent of the standard should be proposed. Notwithstanding such a general commitment to a standard, the applicant should clarify broad requirements in the standard by more specific commitments in the application. These commitments should be in sufficient detail to show how the applicant's program will meet the standard. Any variations from the requirements of the standard should be identified and justified in the application.

Throughout this chapter, reference is made to specific portions of the standards. This is not meant to imply that they are more important than other portions of the standards, but only that further elaboration is needed.

Individual commitments to the acceptance criteria are expected only when the acceptance criteria are relevant to the operations and materials to be licensed.

6.4.1 Regulatory Requirements

The regulatory basis for the review should be the general and additional contents of a license application (including construction approval) as required by 10 CFR 70.22 and § 70.65, respectively. In addition, the NCS review should be conducted to ensure compliance with 10 CFR 70.24 and 10 CFR 70.61, 70.62, 70.64, 70.72, and Appendix A of 10 CFR Part 70.

6.4.2 Regulatory Guidance

The Nuclear Regulatory Commission (U.S.) (NRC), Regulatory Guide (RG) 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," dated August 1998, endorses the American National Standards Institute's (ANSI's) and American Nuclear Society's (ANS's) ANSI/ANS 8 national standards listed below in part or in full:

ANSI/ANS–8.1–1983 (Reaffirmed in 1988), "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

ANSI/ANS–8.3–1997, "Criticality Accident Alarm System."

ANSI/ANS–8.5–1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."

ANSI/ANS–8.6–1983 (Reaffirmed in 1995), "Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ."

ANSI/ANS–8.7–1975 (Reaffirmed in 1987), "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."

ANSI/ANS–8.9–1987 (Reaffirmed in 1995), "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials."

ANSI/ANS–8.10–1983 (Reaffirmed in 1988), "Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement."

ANSI/ANS–8.12–1987 (Reaffirmed in 1993), "Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors."

ANSI/ANS–8.15–1981 (Reaffirmed in 1995), "Nuclear Criticality Control of Special Actinide Elements."

ANSI/ANS–8.17–1984 (Reaffirmed in 1997), "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors."

ANSI/ANS–8.19–1996, "Administrative Practices for Nuclear Criticality Safety."

ANSI/ANS–8.20–1991, "Nuclear Criticality Safety Training."

ANSI/ANS–8.21–1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors."

ANSI/ANS–8.22–1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators."

ANSI/ANS–8.23–1997, "Nuclear Criticality Accident Emergency Planning and Response."

These ANSI standards are not requirements, but represent practices that have been found generally acceptable to the NRC staff. The reviewer should check the current version of RG 3.71 to determine the currently endorsed versions of these standards. Reference in this chapter to a specific version should not be construed as discouraging the applicant from using the most recent version of a standard. However, if the applicant commits to an unendorsed standard, responsibility for demonstrating that this constitutes an acceptable methodology rests with the applicant.

6.4.3 Regulatory Acceptance Criteria

6.4.3.1 Organization and Administration

The importance of management measures and the corporate safety culture in preventing accidental criticality cannot be overstated. Programmatic failure has been a major contributor to most of the historic accidents, much more than failures of a technical or analytical nature. The most theoretically robust control systems will not work if the facility management does not make safety a top priority and create an atmosphere of safety consciousness and accountability. Although the majority of this chapter is devoted to the technical aspects of the NCS program, the primacy of administration, organization, and management measures is stressed by placing it first in this chapter.

To provide for NCS, the applicant's organization and administration implementing the safety program in 10 CFR 70.62(a) should be considered acceptable if the applicant has met the following acceptance criteria. (Information related to these acceptance criteria may be consolidated with other organization and administration descriptions elsewhere in the application in response to Chapter 4.0.):

A. The applicant meets the acceptance criteria related to NCS in SRP Section 4.4.3. Further, the applicant has described organizational positions, functional responsibilities, experience, and adequate qualifications of persons responsible for NCS.

B. The applicant commits to the endorsed requirements related to organization and administration in ANSI/ANS–8.1–1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," (applicable section is Section 4.1). Where similar requirements also exist in ANSI/ANS–8.19–1996, "Administrative Practices for Nuclear Criticality Safety," the applicant commits to following the more detailed requirements of ANSI/ANS–8.19–1996 (Sections 4 through 10).

C. The NCS organization is independent of operations to the extent practical. The staff is enabled to review and concur on procedures and facility operations and have shut-down authority over any operations it considers unsafe; the staff reports to the safety manager and is independent of operations at the highest practical level, preferably to an official at a sufficiently high level to have the authority to make commitments to the NRC and have accountability for the overall safety of the facility.

D. The applicant commits to providing NCS postings for administrative controls in areas, operations, work stations, and storage locations that provide operators with a reference for ensuring conformance and safe operation.

E. The applicant commits to the policy that: "All personnel shall report defective NCS conditions to the NCS function, directly or through a designated supervisor, and take no further action not specified by approved written procedures until NCS has analyzed the situation." The NCS staff is involved in responding to emergency and accident conditions as part of the emergency response organization.

F. The applicant's administration of the facility should include commitments that foster ownership of safety by organizations at all levels, including operations, maintenance, engineering, and management (not just the NCS organization). The applicant commits to a corporate policy of instilling a safety ethic in the workforce and making safety a top priority.

6.4.3.2 Management Measures

To provide for NCS, the applicant's management measures required by 10 CFR 70.62(d) should be considered acceptable if the applicant has met the following acceptance criteria. Management measures may be graded in accordance with 10 CFR 70.62(d), with appropriate justification provided if the highest level of assurances is not used.

A. Training (these acceptance criteria are in addition to those specified in SRP Section 15.4.4):

   i. The applicant commits to the endorsed training requirements in both ANSI/ANS–8.19–1996, "Administrative Practices for Nuclear Criticality Safety" (applicable sections are Sections 4.2, 4.4, 5.2, 5.3, 6.2 through 6.5, 10.2, and 10.5) and ANSI/ANS–8.20–1991, "Nuclear Criticality Safety Training" (Sections 5 through 8).

   ii. The applicant commits to providing instruction in the training program regarding the use of process variables for NCS control, if controls on such parameters are credited for NCS (e.g., IROFS).

   iii. The applicant commits to providing instruction in the training program regarding the policy discussed in Item D of Section 6.4.3.1.

B. Procedures (these acceptance criteria are in addition to those specified in Section 15.5.4):

   i. The applicant commits to the endorsed procedural requirements in ANSI/ANS–8.19–1996, "Administrative Practices for Nuclear Criticality Safety" (applicable section is Section 7).

   ii. Administrative controls that are incorporated into procedures are reiterated in distinctive and readable criticality safety postings. Postings and procedures should be controlled to ensure that they reflect the current administrative controls and limits.

C. Audits and Assessments (these acceptance criteria are in addition to those specified in Section 15.6.4):

   i. The applicant commits to the endorsed audit and assessment requirements in ANSI/ANS–8.19–1996, "Administrative Practices for Nuclear Criticality Safety" (applicable sections are Sections 4.6, 6.6, 7.8, and 8.4).

ii. Operations are reviewed at least annually to ascertain that procedures are being followed and that process conditions have not been altered to adversely affect NCS. These reviews are conducted, in consultation with operating personnel, by applicant staff who are knowledgeable in NCS and who (to the extent practicable) are not immediately responsible for the operations.

iii. The applicant commits to conducting and documenting periodic NCS walkthroughs (e.g., checklists) of all operating special nuclear material (SNM) process areas. Identified weaknesses should be incorporated into the facility corrective actions program and should be promptly and effectively resolved. A less than weekly frequency may be justified on the basis of risk, such as based on the results of the ISA. Without any such risk-informed determination of the walkthrough frequency, a commitment to conducting weekly walkthroughs should be considered acceptable, such that all operating SNM process areas should be reviewed at least every 2 weeks.

iv. The applicant commits to conducting and documenting periodic NCS audits. A less than quarterly frequency may be justified on the basis of risk, such as based on the results of the ISA. Without any such risk-informed determination of the audit frequency, a commitment to conducting quarterly audits should be considered acceptable, such that all NCS aspects of management measures (see Sections 15.1 through 15.8) should be audited at least every 2 years.

6.4.3.3 Technical Practices

6.4.3.3.1 Analytical Methodology

To provide for NCS, the applicant's NCS methodologies should be considered acceptable if the applicant has met the following acceptance criteria:

A. The applicant commits to the endorsed technical requirements in ANSI/ANS–8.1–1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" (Sections 4.2 and 4.3; Sections 5 and 6 contain single-parameter and multiparameter limits that may be referenced).

B. The applicant commits to the intent of the requirement in RG 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," related to validation reports, that is, the applicant should demonstrate: (1) the adequacy of the margin of subcriticality for safety by assuring that the margin is large compared to the uncertainty in the calculated value of k_{eff} and (2) determination of the area(s) of applicability (AOA) and use of the code within the AOA, including justification for extending the AOA by using trends in the bias.

C. As part of providing reasonable assurance that an adequate margin of subcriticality has been provided, in accordance with 10 CFR 70.61(d), the applicant has, at the facility, a documented, reviewed, and approved validation report (by NCS and management) for each methodology that will be used to make an NCS determination (e.g., experimental data, reference books, hand calculations, deterministic computer codes, probabilistic computer

codes). These methodologies may include Monte Carlo or deterministic computer codes, hand calculations, handbooks, experiments, or other applicable methods. The validation report should contain the following, in sufficient detail to permit an independent reconstruction of results by the NCS reviewer:

i. A description of the theory of the methodology in sufficient detail, clarity, and lack of ambiguity that allows understanding of the methodology, including validity of assumptions and independent duplication of results.

ii. A description of the AOA that identifies the range of values for which valid results have been obtained for the parameters used in the methodology. As defined in ANSI/ANS 8.1-1983, the AOA is the range of material compositions and geometric arrangements within which the bias of a calculational method is established. Other variables that may affect the neutronic behavior of the calculational method should also be specified in the definition of the AOA. Particular attention should be given to validating the code for calculations involving mixed oxides of differing isotopics and defining the isotopic ranges covered by the available benchmark experiments. In accordance with the provisions in ANSI/ANS–8.1–1983 (applicable section is Section 4.3.2), any extrapolation of the AOA beyond the physical range of the data should be supported by an established mathematical methodology.

iii. A description of the use of pertinent computer codes, assumptions, and techniques in the methodology.

iv. A description of the verification of the proper functioning of the mathematical operations in the methodology (e.g., mathematical testing).

v. A description of the benchmark experiments and data derived therefrom that were used for validating the methodology.

vi. A description of the bias, uncertainty in the bias, uncertainty in the methodology (e.g., from statistics, computational convergence, and nuclear cross section data), uncertainty in the data, uncertainty in the benchmark experiments, and Margin of Subcriticality for Safety, as well as the basis for these items, as used in the methodology. If the bias is determined to be advantageous to the applicant, the applicant shall use a bias of 0.0 (e.g., in a critical experiment where the k_{eff} is known to be 1.0 and the code calculates 1.02, the applicant cannot use a bias of 0.02 to allow calculations to be made above the value of 1.0).

vii. A description of the software and hardware that will use the methodology.

D. The applicant commits to incorporating each documented, reviewed, and approved validation report (by NCS and management) into the configuration management program.

E. The applicant commits to performing NCS evaluations using specific standardized methods, including the use of only validated calculational methods. The applicant should commit to incorporating these methods into the facility safety program. This should include standard specifications for how single units and arrays should be modeled for different physical configurations (such as a minimum reflection condition to account for incidental reflectors in the room, minimum statistics, and so forth).

F. The applicant commits to assuming credible optimum conditions (i.e., most reactive conditions physically possible) for each controlled parameter unless specified controls are implemented to limit the controlled parameter to a certain range of values.

   Specifically, if reflection is not controlled, the array should be assumed to be fully flooded, unless a more reactive water density or reflector material exists. If moderation is not controlled, the optimal weight percent of water (or more reactive credible moderator) should be assumed. If mass is not controlled, the units should be assumed to be completely filled. If neutron poison is not controlled, no credit should be taken for the material of construction of the containers and the fissile material is modeled to the outer diameter. If interaction is not controlled, the units should be stacked together in the most reactive configuration in the corner of the room (reflected by concrete walls and the floor). Optimal values of these parameters may be determined using sensitivity studies.

G. The applicant commits to considering the variability and uncertainty in a process and the NCS subcritical limit when setting NCS safety limits. If a controlled parameter is dependent on other physical parameters (such as mass controlled indirectly by concentration and volume), then the uncertainties in each of the independent parameters should be folded into the estimated uncertainty of the dependent parameter.

6.4.3.3.2 Additional Technical Practices

CSEs should be considered the main source of information regarding the adequacy of criticality controls. CSEs are the documents used to develop the safety basis of facility operations. By sampling selected CSEs, the reviewer should confirm that the CSEs establish an adequate safety basis. The reviewer should then verify that all controls (from the sampled CSEs) are included in the ISA Summary as IROFS.

To provide for NCS, the applicant's commitment to NCS technical practices, in meeting the performance requirements of 10 CFR 70.61(d) and BDC of § 70.64(a)(9), should be considered acceptable if the applicant has met the following acceptance criteria:

A. Although the applicant may use a single NCS control to maintain the values of two or more controlled parameters, this use constitutes only one component necessary for double contingency protection.

B. Based on the performance requirements in 10 CFR 70.61, the applicant commits to the policy that: "No single credible event or failure could result in a criticality accident." This

commitment should involve an evaluation to identify common-mode failures that may simultaneously defeat two or more controls.

C. The applicant commits to the preferred use of passive-engineered controls to ensure NCS. The applicant should commit to the following preference, in general, for controls to ensure NCS: (1) passive-engineered, (2) active-engineered, (3) augmented-administrative, and (4) simple-administrative. The applicant should demonstrate how it is meeting this commitment to the preferred design approach, such as by providing justification when using other than passive-engineered control. This demonstration should also be documented in the ISA.

D. The applicant commits to incorporating controls on controlled parameters into the facility management measures of 10 CFR 70.62(d).

E. The applicant commits to describing controlled parameters for each process used as NCS control. Examples of controlled parameters available for NCS control are: mass, geometry, density, enrichment, reflection, moderation, concentration, interaction, neutron absorber, and volume.

F. When controlled parameters are controlled for safety reasons by measurement, reliable methods and instruments are used. It is acceptable if the applicant commits to representative sampling, reliable measurement instruments and methods, and dual independent measurements where there is significant susceptibility to human error.

6.4.3.3.2.0 Methods of NCS Control

Several methods of NCS control are available. These are summarized below. Passive geometry control is the preferred method of preventing criticality, and the applicant should commit to using this method whenever practical. The applicant should demonstrate how it is meeting this commitment, such as by providing justification when using other than passive geometry for criticality control.

The controls used to establish limits on the following criticality parameters should be identified as IROFS in the CSEs and ISA Summary. Tolerances on the controlled parameters should be conservatively taken into account in setting operating limits and controls established to prevent exceeding subcritical values of parameters.

The use of single-parameter limits (favorable geometry, safe volume or mass, etc.) may be invalid when interactions with other units are taken into account. Interaction should be fully evaluated, and spacing controls should be used in conjunction with those other controls as needed to ensure subcriticality.

6.4.3.3.2.1 Mass Control

The use of mass as a controlled parameter should be considered acceptable if:

A. 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 the entire mass is SNM or (2) conducting physical measurements to establish the actual weight percent of SNM in the material. When process variables can affect the bounding weight percent of SNM in the mixture, controls to maintain the process variables are identified as IROFS in the CSEs and ISA Summary. The acceptance criteria in Section 6.4.3.3.2.12 are met.

B. Theoretical densities for fissile mixtures are used unless lower densities are ensured by the establishment of NCS controls.

C. When physical measurement of the mass is needed, the measurement is obtained by using instrumentation subject to facility quality assurance measures as specified in 10 CFR 70.22(f).

D. When overbatching of SNM is possible, the mass of SNM in a single batch is limited so that the mass of the largest credible overbatch resulting from a single failure is safely subcritical, taking system sensitivities into account. Overbatching beyond double batching should be considered in establishing the margin of safety.

E. When overbatching of SNM is not possible, the mass of SNM in the batch is limited to be safely subcritical, taking system sensitivities into account.

F. In setting mass limits, tolerances in determining the mass should be taken into account. Determination of the minimum critical mass should be based on spherical geometry or the actual fixed geometry of the system if it is controlled.

6.4.3.3.2.2 Geometry Control

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

A. Before beginning operations, all dimensions and nuclear properties that rely on geometry control are verified. Dimensional tolerances should be conservatively taken into account when setting geometrical limits. The facility configuration management program should be used to maintain these dimensions and nuclear properties.

B. All credible means of transferring fissile materials to unfavorable geometry are evaluated and controls (IROFS) are established against this contingency.

C. When using large single units, conservative margins of safety (such as 90% of the minimum critical cylinder diameter, 85% of the minimum critical slab thickness, and 75% of the minimum critical sphere volume) are used. Justification should be provided for proposed alternatives to these limits, taking system sensitivities into account. Such justification may include a demonstration that other conservative assumptions–such as neglecting neutron absorption effects and making conservative density assumptions–would be sufficient to make up the difference in margin. Reliance on engineering judgement does not substitute

for this justification. The ability of the control scheme to maintain process parameters within the prescribed limits should be considered when setting less conservative margins of safety.

D. Possible mechanisms for changes to the fixed geometry should be evaluated and controls established as needed. Where such credible mechanisms exist (such as deformation by static loads or pressure, corrosion, etc.), the applicant should describe the design and surveillance program for these units.

6.4.3.3.2.3 Density Control

The use of density as a controlled parameter should be considered acceptable if:

A. When process variables can affect the density, controls to maintain the process variables are identified as IROFS in the CSEs and ISA Summary. Process characteristics that can affect the density non-conservatively are controlled. The acceptance criteria in Section 6.4.3.3.2.12 are met.

B. When physical measurement of the density is needed, the measurement is obtained by using instrumentation subject to facility quality assurance measures as specified in 10 CFR 70.22(f).

6.4.3.3.2.4 Isotopics Control

Isotopic abundance (isotopics) is taken to include both the $^{235}U/U$ concentration (enrichment) and the concentration of fissile and nonfissile plutonium isotopes (such as ^{239}Pu, ^{240}Pu, ^{241}Pu) as well as the relative abundance of plutonium to uranium.

The use of isotopics as a controlled parameter should be considered acceptable if:

A. When taking credit for isotopic mixtures, where different isotopic mixtures could co-exist, controls are established to clearly label and segregate the SNM of different isotopic mixtures. Labels and postings for different isotopic mixtures should be distinctive and readily apparent to operators, and should be uniform to the extent practicable throughout the facility to minimize the potential for error. Moreover, determinations of isotopic content shall be based on dual independent sampling and analysis of each lot of fissile material.

B. When physical measurement of the isotopics is needed, the measurement is obtained by using instrumentation subject to facility quality assurance measures as specified in 10 CFR 70.22(f).

6.4.3.3.2.5 Reflection Control

The use of reflection as a controlled parameter should be considered acceptable if:

A. When determining subcritical limits for an individual unit, the wall thickness of the unit and all reflecting adjacent materials of the unit are conservatively bounded by the assumed

reflection conditions, leaving allowances for transient reflectors as discussed in the next item. (This effect may be significant for a mixed oxide MOX facility, where thick hydrogenous reflectors may provide shielding in several areas.)

B. At a minimum, reflection conditions equivalent to a 1-inch tight-fitting water jacket are assumed to account for personnel and other transient incidental reflectors not evaluated in the unreflected unit models. This will be considered bounding for all hydrogenous reflectors further than 1 foot away from the surface of the unit. Justification for less conservative reflection conditions should be included in the application.

C. When loss of reflection control can lead to criticality, by itself or in conjunction with any other single failure, rigid and testable personnel barriers are established and maintained through the configuration management and maintenance programs.

D. Full water reflection of units may be assumed to be represented by 12 inches of close-fitting water. Under certain conditions, however, materials such as concrete, beryllium, carbon, and polyethylene may be more effective than water.

E. Conservative reflection conditions are established when evaluating the criticality safety of arrays.

6.4.3.3.2.6 Moderation Control

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

A. When using moderation, the applicant commits to the requirements in ANSI/ANS–8.22–1997, "Nuclear Criticality Safety Based on Limiting and Controlling Moderators."

B. When process variables can affect the moderation, controls to maintain the process variables are identified as IROFS in the CSEs and ISA Summary. The acceptance criteria in Section 6.4.3.3.2.12 are met.

C. When physical measurement of the moderator is needed, the measurement is obtained by using instrumentation subject to facility quality assurance measures as specified in 10 CFR 70.22(f).

D. When designing physical structures that are credited for moderation control, the design is sufficient to preclude the ingress of moderation.

E. When sampling of the moderator is needed, the sampling program uses dual independent sampling and analysis methods. The process should be designed such that a single operator acting alone cannot physically circumvent the sampling and analysis program. (More detailed guidance is provided in Section 6.4.3.3.2.7 on concentration control).

F. When developing firefighting procedures for use in a moderation-controlled area, restrictions are placed on the use of moderator material. Moderation-controlled areas should be physically segregated from potential ignition sources. The effects of the fire and the activation and/or failure of the fire suppression should be evaluated as applicable.

G. Limits on moderators as firefighting agents are established in the CSE and flow into the ISA. The ISA may weigh the competing risks from criticality accidents and fires and determine that the overall risk to the worker and the public is minimized by allowing the use of water. The CSE is the primary document used to develop the safety basis, and thus should govern the safety of operations; the CSE should be revised so that the safety conclusions harmonize between the two documents.

H. After evaluating all credible sources of moderator for the potential for intrusion into a moderation-controlled area, the ingress of moderator is precluded or controlled.

I. The effects of varying levels of interstitial moderation are evaluated when the calculational methods consider interacting arrays of fissile units. If a fissile material system is not adequately subcritical with a few percent water density between array elements, water pipes and sprinklers should be excluded from the affected area. Overhead water pipes may alternately be sleeved within secondary piping, provided a means is provided to detect leakage from the inner pipe.

J. Favorable geometry drains are provided to prevent water accumulation if that accumulation can lead to unsafe configurations of fissile material.

6.4.3.3.2.7 Concentration Control

The use of concentration as a controlled parameter should be considered acceptable if:

A. When process variables can affect the concentration, controls to maintain the process variables are identified as IROFS in the CSEs and ISA Summary, including assumptions relied on to determine solubility limits. The acceptance criteria in Section 6.4.3.3.2.12 are met.

B. Sufficient controls are established to preclude the introduction of materials concentrated to a level higher than assumed for normal conditions.

C. When using a tank containing concentration-controlled solution, the tank is normally closed and locked. Access should be controlled to ensure that a single operator cannot defeat the control mechanism.

D. When sampling of the concentration is needed, the sampling program uses dual independent sampling methods. The two samples should be withdrawn by different individuals and at different points in the process or at different times with mixing to ensure a representative sample between measurements. They should be analyzed by different analysts using different methods in the lab, and a supervisor should be required to check

results before authorizing the transfer. In addition, attention is paid to common-mode failures that can defeat both samples, such as circumventing this robust system by having a single isolation valve leak through or by not having the transfer valve locked or tagged so that an operator can effect the transfer by himself. The process is designed such that a single operator acting alone cannot physically circumvent the sampling program.

(Note: It may be difficult to determine that a single sampling and analysis constitutes a robust control, due to the number and complexity of steps involved. In this case, dual sampling may need to be credited as a single leg of double contingency.)

E. When concentration control is the only means of ensuring subcriticality in unfavorable geometry equipment (such as waste water storage tanks), particular attention should be given to the robustness of concentration controls where transfer to unfavorable geometry occurs. In such cases, due to the difficulties involved with dual sampling, another means (such as an in-line monitor) should be used in conjunction with dual sampling to provide reasonable assurance of safety.

F. After identifying possible precipitating agents, precautions are taken to ensure that such agents will not be introduced inadvertently.

G. All other concentrating mechanisms are identified and controls established to prevent overconcentration. Surveillance is provided to ensure the effectiveness of these controls.

H. When physical measurement of the concentration is needed, the measurement is obtained by using instrumentation subject to facility quality assurance measures as specified in 10 CFR 70.22(f).

6.4.3.3.2.8 Interaction Control

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

A. When maintaining a physical separation between units, engineered devices (i.e., spacers) with a minimum spacing are used. The structural integrity of the spacers should be sufficient for normal and credible abnormal conditions. Moreover, if the engineered devices are part of the structure of the unit (such as moveable birdcage drums) or subjected to significant mechanical stresses, they should be periodically inspected for deformation; or

B. Unit spacing is controlled by rigorous procedures (if the spacing is identified in workstation procedures with visual indicators and postings). This should include visible guides (such as painted lines and postings) to ensure that spacing limits are not violated. Justification for this method should be provided in the application and should demonstrate that multiple procedural violations will not by themselves lead to criticality.

C. When evaluating the criticality safety of units in an array or pairs of arrays, the spacing limits in ANSI/ANS–8.7–1975, "Nuclear Criticality Safety in the Storage of Fissile Materials,"

are followed (Sections 5 and 6), or else spacing is based on validated calculational methods.

6.4.3.3.2.9 Neutron Absorber Control

The use of neutron absorber as a controlled parameter should be considered acceptable if:

A. When using borosilicate-glass Raschig rings, the applicant commits to the endorsed requirements in ANSI/ANS–8.5–1996, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material."

B. When using fixed neutron absorbers, the applicant commits to the endorsed requirements in ANSI/ANS–8.21–1995, "Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors."

6.4.3.3.2.10 Volume Control

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

A. When using volume control, geometrical devices are used to restrict the volume of SNM and engineered devices limit the accumulation of SNM.

B. When physical measurement of the volume is needed, the measurement is obtained by using instrumentation that is subjected to quality assurance.

C. Volume is limited to some percentage of the minimum critical volume, assuming spherical geometry, optimal concentration, and full water reflection.

6.4.3.3.2.11 Heterogeneity Control

The use of heterogeneity as a controlled parameter should be considered acceptable if:

A. When process variables can affect the heterogeneity, controls to maintain the process variables are identified as IROFS in the CSEs and ISA Summary. Methods of causing the material to become inhomogeneous are evaluated and appropriate controls established. If such mechanism for causing inhomogeneity are credible, such as precipitation, the most reactive configuration of the material should be considered in computer calculations (*i.e.*, assuming homogeneous solution before precipitation and a uniform slab after precipitation may not be sufficient to bound all credible system configurations) if heterogeneity is not explicitly controlled. The acceptance criteria in Section 6.4.3.3.2.12 are met.

B. Computer calculations that take heterogeneity into account are appropriately validated with benchmark experiments that display effects of heterogeneity, to ensure that the bias is known when using resonance self-shielding. Computer calculational methods that require self-shielding corrections to correctly calculate k_{eff} use the appropriate cell weighting to ensure that this effect is taken into account.

C. Assumptions about the physical scale of heterogeneity (as used in computer calculations) are based on the observed physical characteristics of the material, and appropriately controlled. Calculations are shown to be conservative with respect to these measurements (i.e., model heterogeneous conditions that are at least as reactive as suggested by physical data).

Heterogeneous effects are particularly relevant to deriving NCS limits for low-enriched uranium processes, where heterogeneous systems are typically more reactive than homogeneous systems for all other parameters being equal.

6.4.3.3.2.12 Process Variables

The use of process variables as a controlled parameter should be considered acceptable if:

A. Controls needed to maintain process variables that are relied on for criticality safety are identified as IROFS in the CSEs and ISA Summary and are subject to quality assurance sufficient to ensure that the associated controlled parameter safety limit is not exceeded. These may include furnace temperature credited in excluding moderation, mechanical forces credited in limiting density, and the effect of background radiation on mass measurement (NDA) instrumentation.

6.4.3.3.3 Requirements of 10 CFR 70.24 (Criticality Accident Requirements)

To provide for NCS, the applicant's description of measures to meet the requirements in 10 CFR 70.24 should be considered acceptable if the applicant has met the following acceptance criteria:

A. The applicant has fully demonstrated that the facility CAAS meets the requirements of 10 CFR 70.24.

B. The applicant has fully demonstrated that the facility meets the remaining criticality accident requirements of 10 CFR 70.24.

C. The applicant commits to the endorsed requirements in ANSI/ANS 8.3–1997, "Criticality Accident Alarm System."

D. Beyond these requirements, the applicant commits to any additional requirements in RG 3.71, "Nuclear Criticality Safety Standards for Fuels and Materials Facilities," (Section C) which modify requirements in the ANSI/ANS–8.3 standard.

E. In accordance with the performance requirements of 10 CFR 70.61(a) to limit the risk of high-consequence events, including externally initiated events:

i. 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.

ii. The applicant commits to having a CAAS that is designed to remain operational during normal operating conditions and should be resistant to damage during other credible events, to the extent practical (up to design basis events). These events would include fires, explosions, corrosive atmospheres, etc.

F. The applicant commits to having a CAAS alarm that is clearly audible in areas that must be evacuated or that provides alternate notification methods that are documented to be effective in notifying personnel when evacuation is necessary.

G. 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., limit access, halt SNM movement) when the CAAS system is not functioning due to maintenance.

H. The applicant evaluates the effect of credible shielding in demonstrating the adequacy of the dual alarms to detect a nuclear criticality. The applicant verifies alarm coverage radius (such as through the use of shielding codes) and controls through good housekeeping practices the presence of shielding material.

I. In accordance with the provisions of 10 CFR 70.24(b)(1) and (b)(2):

i. The applicant commits to the requirements in ANSI/ANS–8.23–1997, "Nuclear Criticality Accident Emergency Planning and Response."

ii. The applicant has either an emergency plan or satisfies the alternate requirements found in § 70.22(h)(1)(i). (See Chapter 14.0 of this Standard Review Plan (SRP).)

iii. The applicant commits to providing emergency power for the CAAS.

J. Exceptions to the CAAS requirements of 10 CFR 70.24 will be considered when the risk of nuclear criticality is sufficiently low that the exposure of facility personnel is not a regulatory or safety concern. The applicant should provide justification by demonstrating that the risk to facility personnel is significantly less than that afforded under the double contingency principle. To support this justification, the applicant may take credit for shielding or other dose mitigation or demonstrate that a criticality is incredible due to amounts and forms of SNM that are or may be present.

Factors that should be considered when evaluating CAAS requirement exemptions include: (1) whether the applicant is authorized to possess less than a minimum critical mass; (2)

whether the facility has been demonstrated to have adequate shielding to prevent any operator from receiving a dose in excess of 20 rad (measures should still be provided to alert operations to the fact of a criticality); (3) whether operators are excluded from processing areas by hostile conditions to a distance adequate to ensure safety; and (4) whether other process conditions exist such that there are no identifiable accident sequences that could credibly lead to a criticality.

6.4.3.3.4 Requirements of 10 CFR 70.61 (Subcriticality of Operations and Margin of Subcriticality for Safety)

To provide for NCS, the applicant's description of measures to implement the subcriticality of operations and margin of safety for subcriticality requirements in 10 CFR 70.61 should be considered acceptable if the applicant has met the following acceptance criteria:

A. The applicant commits to technical practices as applicable in the endorsed versions of ANSI/ANS–8.1, 8.5, 8.7, 8.9, 8.10, 8.12, 8.15, 8.21, and 8.22.

B. The applicant submits justification for the minimum subcritical margin (frequently referred to as the administrative or arbitrary margin) for normal and credible abnormal conditions. (Note: A minimum subcritical margin of 0.05 is generally considered to be acceptable without additional justification when both the bias and its uncertainty are determined to be negligible.) Abnormal conditions should meet the following criterion to provide reasonable assurance of adequate protection:

   i. If the fact that a condition is abnormal is credited for double contingency (that is, the abnormality is taken as justification for having a lower margin of subcriticality than would be permissible for normal conditions), then the abnormal conditions should meet the standard of being at least "unlikely" from the standpoint of the double contingency principle. A condition that occurs on a regular basis during facility operations would not be considered abnormal. The applicant should rigorously define what is meant by an abnormal condition for this purpose. In addition, the increased risk associated with the less conservative margin should be commensurate with and offset by the unlikelihood of achieving the condition in any case.

C. The applicant commits to determining subcritical limits for k_{eff} calculations such that:

$$\text{k-subcritical} = 1.0 - \text{bias} - \text{margin}$$

   where margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality.

D. The applicant commits to determining operation limits for controlled parameters, such that there is an adequate margin of safety to ensure the subcritical limit will not be exceeded. The applicant should commit to performing studies of the sensitivity of k_{eff} to variations in the parameters. The margin of safety should be based on these sensitivity studies and the ability of the control to maintain the operating limits.

E. The applicant commits to determining whether each calculation to establish subcritical limits for facility processes lies within the AOA of the calculational method employed, and documenting the determination that it is within the AOA in the CSEs. (The AOA for the method should be defined in the validation report but may also be done on a case-by-case basis in the CSEs; see Section 6.4.3.3.1). To the extent possible, the applicant should use benchmark cases that are similar in physical and neutronic characteristics to the specific application to determine the bias. Separate AOAs should be defined wherever there are significant differences in system characteristics (such as for plutonium and uranium oxides).

F. The applicant meets the acceptance criteria in Section 5.4 as they relate to subcriticality of operations and margin of subcriticality for safety.

6.4.3.3.5 Requirements of 10 CFR 70.64 (BDC) [for new facilities and processes only]

To provide for NCS, the applicant's description of measures to implement the BDC requirements in 10 CFR 70.64 should be considered acceptable if the applicant has met the following acceptance criteria:

A. The applicant commits to the double contingency principle in determining NCS controls in the design of new facilities or new processes at existing facilities. When evaluating double contingency protection, the term "unlikely" should be used in a manner consistent with Section 4.2.2 of ANSI/ANS–8.1–1983.

B. Protection is provided by either the control of two (or more, as needed) independent process parameters or a system of multiple independent controls on a single process parameter.

The former method, two-parameter control, is the preferred approach due to the difficulty of preventing common-mode failure when controlling only one parameter. In all cases, no single credible event or failure shall result in a criticality accident.

The term "concurrent" as used in double contingency means, for the purpose of this review, that the effect of the first process change persists until the second change occurs, at which point the system is potentially at or above critical. It does not mean that the two events initiating the change must occur simultaneously.

Means should be provided for detecting and correcting failures that may not be readily apparent (such as failure of containment in inaccessible equipment) to ensure that single failures do not persist for indefinite periods of time. The time interval needed to detect and correct the failure should be considered and may be credited in the determination that the two failures in combination are highly unlikely.

Dependence between the two or more events in the accident sequence should be taken into account in assessing the likelihood, so that the occurrence of both events taken together is highly unlikely. This dependence can occur because one event causes the other to become

more likely, or because occurrence of some other event increases the likelihood of both events. This latter type can be the occurrence of a fire or other environmental degradation, the use of nondiverse equipment, or the same operator performing two actions. Another type of dependence that must be considered is common-cause failure, that is, a single-event failure. Such a common-mode failure is most likely when reliance is placed on a single controlled parameter. If any such single event exists that could cause criticality, it by itself must qualify as highly unlikely. Such common-mode failure scenarios should be evaluated in the CSEs and flow into the ISA.

C. Adequate justification for allowing an exception to the double contingency principle includes:

i. The impracticality of implementing the double contingency principle is thoroughly documented by showing excessive costs and severe operational burdens that would be imposed on the facility compared to the risk reduction gained by implementing the principle; and

ii. Enough redundancy and diversity exists to ensure that the probability of criticality remains highly unlikely, or the nature of the process is such that no credible accident scenarios are identified. Even if the consequences of criticality are mitigated such that they do not rise to the threshold of 10 CFR 70.61(b)(1), criticality shall still be highly unlikely. However, the mitigation may constitute grounds, along with other considerations, for granting exemption from the requirements to establish double contingency. A criticality in a shielded facility may not be a high-consequence event, but still requires an explicit exemption from the double contingency requirement.

Care should be taken to use a definition of "unlikely" that is consistent with the definition in SRP Section 6.8, rather than the definition in Chapter 5 (Integrated Safety Analysis) for intermediate-consequence events. Although the terminology used is the same, the context differs. As Section 6.8 states, the scope of the definitions there are confined to this chapter.

6.4.3.3.6 Requirements of 10 CFR 70.65 (ISA Summary)

The applicant is required to meet the performance criteria in 10 CFR 70.61(b) and (c) as well as the performance requirements in § 70.61(d), which include the requirement to limit the risk of an inadvertent nuclear criticality by assuring that all nuclear processes remain subcritical. The applicant's evaluation of NCS accident sequences should 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).

To provide for NCS, the applicant's implementation of the ISA requirements in 10 CFR 70.65 should be considered acceptable if the applicant has met the following acceptance criteria or has identified and justified an alternative in the application:

A. Accident Sequences:

i. The applicant meets the acceptance criteria in Section 5.4 related to accident sequences for NCS.

ii. The applicant commits to evaluating the loss of each criticality control as a separate accident sequence. (Appendix A of ANSI/ANS–8.1–1983 provides guidance on the types of accident sequences that should be considered.)

B. Consequences:

i. The applicant meets the acceptance criteria in Section 5.4 related to consequences for NCS.

ii. In determining the consequences of a criticality, the applicant may commit to the requirements in ANSI/ANS–8.10–1983, "Criteria for Nuclear Criticality Safety Controls in Operations With Shielding and Confinement." The justification for considering a criticality accident as other than a high-consequence event should be fully documented and provided as part of the application. (See guidance for shielded facilities in Section 6.4.3.3.5.)

C. Likelihoods:

i. The applicant meets the acceptance criteria in Section 5.4 related to likelihoods for NCS.

ii. In demonstrating compliance with the double contingency principle, the term "unlikely" is taken to mean that an event—or a set of events credited as one leg of double contingency—is not anticipated to occur during the lifetime of the facility at any particular point in the process or in any particular accident sequence. In demonstrating unlikelihood, the applicant may credit credible process conditions, previous facility history (for existing facilities), and management measures that ensure the availability and reliability of controls when needed. The applicant may choose to define the terms "unlikely" and "highly unlikely" differently, but must still demonstrate compliance with the performance requirements of 10 CFR 70.61(b) and 70.64(a)(9).

For passive and active engineered controls, management measures such as maintenance, configuration management, surveillance, and so on may be considered adequate to ensure that failure of the control is unlikely. Consideration should be given to the environment in which the device is to be used and whether the expected operating conditions are within the manufacturer's specifications. Surveillance and periodic maintenance frequencies may be determined to be adequate if based on the manufacturer's recommendations when operated within the applicable specifications.

For a simple administrative control such as spacing in an array, a determination that failure is unlikely may be based on operator training, supervisor oversight, operator aids, and other human factors considerations. Processes relying on administrative requirements should be designed to minimize the opportunity for human error. Multiple failures of the same control

may not be considered distinct contingencies, if they can result from the same operator's error. More than two controls may be needed as one leg of double contingency, if they are not individually unlikely to fail.

D. Risk:

   i. The applicant meets the acceptance criteria in Section 5.4 related to risks for NCS.

E. IROFS:

   i. The applicant meets the acceptance criteria in Section 5.4 related to IROFS for NCS.

   ii. The IROFS should be described in sufficient detail to permit the reviewer to make a determination of adequacy in verifying consequences and likelihoods. Those attributes of the controls that are deemed important for safety should be fully described in the ISA Summary, rather than general statements that controls will be established on specific controlled parameters.

6.4.3.3.7 Requirements of 10 CFR 70.72 (Facility Change Process)

To provide for NCS, the applicant's description of measures to implement the facility change process requirements in 10 CFR 70.72 should be considered acceptable if the applicant has met the following acceptance criteria:

A. The applicant commits to a change control process that is sufficient to ensure that the safety basis of the facility will be maintained during its lifetime. This change process must be documented in written procedures and must ensure that:

   i. All potentially affected SNM processes are evaluated to determine the effect of the change on the safety basis of the process, including the effects on bounding process assumptions, on the reliability and availability of nuclear criticality controls, and on the criticality safety of connected processes. The change control process should have procedures for the review and approval of facility changes by the criticality safety organization to determine the potential effects on nuclear criticality safety.

B. The change control process must be connected to the facility's configuration management system to ensure that changes to the criticality safety basis are incorporated into procedures, evaluations, criticality postings, drawings, any other safety basis documentation, and the ISA.

C. The applicant commits to a program to determine whether facility changes require prior NRC approval in accordance with the criteria of 10 CFR 70.72(c). This program must be documented in written procedures and must involve individuals qualified to determine the incremental effect of changes to the safety basis as documented in the ISA and established in the CSEs; the change shall be compared to the baseline (latest NRC-approved) version of the ISA.

D. In evaluating whether the applicant's change control process is adequate, the reviewer should examine the level of detail with which the IROFS are specified in the ISA Summary (see Section 6.4.3.3.6). Controls should be specified with enough specificity to ensure that all attributes important to safety are captured. Having a level of detail that exceeds this will result in an unduly burdensome change process.

E. Those attributes of IROFS that are important to criticality safety are identified in the CSEs and flowed into the ISA. Changes to attributes not previously identified as being important to criticality safety do not typically require change review, but all changes should be evaluated to ensure that new failure modes are not introduced and that equipment reliability is not degraded.

6.4.3.3.8 Requirements of 10 CFR 70 Appendix A (Reportable Safety Events)

The applicant's description of measures to implement the reporting requirements in Appendix A of 10 CFR 70 should be considered acceptable if the applicant has met the following acceptance criteria:

A. The applicant has a program for evaluating the criticality significance of criticality safety events and an apparatus in place for making the required notification to the NRC Operations Center. The determination of significance should be made by qualified individuals (such as facility NCS staff). The determination of loss or degradation of double contingency protection should be made against the current version of the facility safety basis documents. Any degradation of double contingency should be considered as reportable unless a significance threshold has been pre-defined.

B. The applicant incorporates the reporting criteria of Appendix A and the report content requirements of 10 CFR 70.50 into the facility emergency procedures.

C. The applicant commits to issuing the necessary report based on whether the IROFS credited for double contingency were lost, regardless of whether the safety limits of the associated criticality parameters were actually exceeded. The applicant may define normal operating limits more conservative than the subcritical limits derived in the CSEs. In this case, a loss of controls instituting these normal operating limits would not rise to the level of a reportable event. The applicant therefore should commit to pre-defining which control failures rise to the level of reportable events.

D. The applicant makes the following commitment: If it cannot be determined within 1 hour of whether the criteria of 10 CFR Appendix A Paragraph (a) or (b) apply, the event shall be treated as a 1-hour report.

E. The applicant commits to include sufficient information in the event report to permit the reviewer to make a determination as to the safety significance. The information to be submitted in these reports should include, to the extent known at the time of the event, the physical and chemical characteristics of the as-found condition, the specific controls relied

on for criticality safety, the specific failure that occurred, and the remaining controls that are in place to prevent an accidental criticality.

6.5 REVIEW PROCEDURES

6.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 6.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use SNM.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

Specifically, the safety assessment of the design basis should address Sections 6.3.1 to 6.3.3 consistent with the level of design. Where information is under development or not yet available, the applicant may use a commitment to provide the material with the license application in lieu of the actual material.

Specific areas of interest during the design phase are described below:

i. The commitment to establishing an NCS organization and administration in accordance with the acceptance criteria of Section 6.4.3.1.

ii. The commitment to establishing management measures for NCS in accordance with the acceptance criteria of Section 6.4.3.2.

iii. The commitment to designing and operating the facility using technical practices that are in accordance with the acceptance criteria of Section 6.4.3.3. In particular:

   a. The applicant commits to designing and operating the facility in accordance with the BDC (that is, commits to the double contingency principle).

   b. The applicant commits to installing and maintaining a CAAS for applicable areas of the facility, or includes an exemption request with the application.

   c. The applicant commits to the following design criteria: Geometry control shall be the preferred mode of control for criticality safety and shall be designed into the facility to the greatest extent practical. Where geometry control is not practical, reliance shall be based on other passive-engineered controls to the greatest practical extent.

d. The applicant provides a description of the overall process and, for each major process step, identifies which criticality safety parameters will be relied on to satisfy the BDC.

e. The applicant demonstrates an ability to design the facility in accordance with the BDC by providing validation reports to support calculations of subcritical limits, and proposed margins of subcriticality.

B. License To Possess and Use SNM

Specifically, the safety assessment of the license application should address Section 6.3 in full.

If the primary reviewer verifies that NCS is adequately addressed (construction or license), the primary reviewer should accept the application for the safety evaluation in Section 6.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

6.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 6.5.1(A) (construction approval) or 6.5.1(B) (license to possess and use SNM), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 6.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

The primary reviewer should consult with the supporting reviewers to identify and resolve any issues of concern related to the construction approval review or the review for the license to possess and use SNM. For the license application, the primary reviewer (acting as a secondary or supporting reviewer) should also coordinate with other reviewers concerning NCS regarding the following:

A. In support of the primary reviewer for Chapter 9.0, the NCS reviewer should determine whether the acceptance criteria in Chapter 9.0 have been met as they relate to NCS.

B. In support of the primary reviewer for Sections 15.1 through 15.8, the NCS reviewer should determine whether the acceptance criteria in those sections have been met as they relate to NCS.

C. In support of the primary reviewer for Chapter 5.0, the NCS reviewer should determine whether the acceptance criteria in that chapter have been met as they relate to NCS.

D. In support of the primary reviewer for Chapter 14.0, the NCS reviewer should determine whether the acceptance criteria in that chapter have been met as they relate to NCS.

E. In determining whether the acceptance criteria have been met, the reviewer should become familiarized with the proposed operation and the dominant criticality safety risks. The reviewer should select a risk-informed sample of accident scenarios from the applicant's ISA Summary to review in evaluating the applicant's technical practices, in conjunction with the applicant's CSEs.

6.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the Safety Evaluation Report (SER). The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

The staff reviewed the Nuclear Criticality Safety (NCS) measures described in the application material for construction approval according to Chapter 5.0 of NUREG–1718. The staff is satisfied that: (1) The applicant's commitments to establishing an NCS organization and administration, management measures, and technical practices for NCS are in broad agreement with regulatory acceptance criteria; (2) the adequate implementation of these commitments is likely to generate an acceptable license application; and (3) the applicant has established design criteria that in broad agreement with the Baseline Design Criteria of 10 CFR 70.64. Based on these findings, the staff concludes that there is reasonable assurance that a facility designed in compliance with the aforementioned application material for construction approval will be found acceptable without major reengineering or redesign. Therefore, the applicant's NCS design bases meets the requirements to approve construction of the facility under 10 CFR Part 70.

The staff could document the safety evaluation for the review for the license to possess and use SNM as follows:

The staff reviewed the Nuclear Criticality Safety (NCS) program of the license application for the [insert name of facility] according to Chapter 6.0 of NUREG–1718. The staff evaluated [state what was evaluated] and found that [state the findings]. The staff has reasonable assurance that: (1) 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, administration, and management measures; (2) the applicant's conduct of operations will be based on NCS technical practices that will ensure that the fissile material will be possessed, stored, and used safely according to the requirements in 10 CFR Part 70; (3) the applicant will develop, implement, and maintain a criticality accident alarm system in accordance with the requirements in 10 CFR 70.24 and in accordance with its emergency management program; and (4) the applicant will have in place an NCS program in accordance with the subcriticality of operations and margin of subcriticality for safety requirements in 10 CFR 70.61 and Baseline Design Criteria in 10 CFR 70.64.

Based on this review, the staff concludes that the applicant's NCS program meets the requirements for a license to possess and use SNM under 10 CFR Part 70 and provides reasonable assurance for the protection of public health and safety, including that of workers and the environment.

Note: The NCS safety evaluation for the ISA Summary requirements for 10 CFR 70.65 should be included in the safety evaluation that supports Chapter 5.0 of this SRP.

6.7 REFERENCES

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6.8 NCS DEFINITIONS

The terms defined below are in addition to the definitions that apply to the entire SRP. Where the definition below disagrees with the global usage, the term below governs. These are terms with a specific meaning to nuclear criticality safety, and the scope of these definitions is confined to SRP Chapter 6.

abnormal condition: Any event that is not planned for as a regular occurrence in the facility or operation design. Any event whose occurrence would result in suspension of fissile material operations and movement and require specific recovery actions to restore adequate protection. A condition that can only be reached by exceeding the safety limits of a controlled parameter but that is planned for in CSEs.

adequate protection: A condition that exists when the risk of criticality is sufficiently low. Adequate protection is presumed to exist when double contingency is maintained, for example.

administrative margin: Margin in k_{eff} in addition to the bias and uncertainties in the bias, to allow for unquantified uncertainties in calculating k_{eff}.

area(s) of applicability: The range of physical parameters (e.g., isotopic abundance, moderation, neutron energy, etc.) characterizing a fissile material system over which the code is validated. That is, the range of parameters covered by the benchmark experiments and for which the bias has been determined. The AOA may be extended by extrapolating the bias using conservative assumptions and methods.

bias: The numerical difference between the calculated and experimental values of k_{eff}. For a group of experiments over a particular AOA, the bias is established as a function of the trending parameter(s).

concurrent: In the context of double contingency, the effect of the first process change persists until the second change occurs. It does not mean simultaneous, but rather that both controls are in a failed state at the same time.

contingency: A loss of criticality control that results in one or more controlled parameters exceeding their safety limits.

control: A system, device, or personnel action intended to regulate a device or process. For criticality safety, any item relied on to prevent or mitigate a criticality accident; synonymous with item relied on for safety or barrier.

degradation: Degradation of a control or controlled parameter occurs when an IROFS identified in the ISA, which maintains the controlled parameter within its safety limits, continues to perform its function but with reduced reliability and availability such that the likelihood of its failure is no longer unlikely.

equivalent replacement: In the context of 10 CFR 70.72(c)(2), any item substituted for an IROFS that does not differ in any attribute(s) identified as important for NCS in the ISA or otherwise relied on for NCS. Substitution of an IROFS should not cause the bounding values of any controlled parameters to be exceeded, should not introduce new accident sequences or failure modes, and should not decrease the reliability and availability of the IROFS for which it is being substituted. If substitution causes at least one of the above, NRC prior approval is required.

highly unlikely:[1] Having a probability of occurrence of $< 10^{-5}$/year/event. Such events should not be expected to occur during the lifetime of the facility. As facility- and process-specific failure data are generated, the definition of highly unlikely should be refined; that is, if a

[1] These definitions are predicated on the assumption that there are approximately 1,000 high consequence accident sequences in the industry (see SRP Section 5.4.3.2). These numbers would need to be adjusted if this assumption is invalid.

particular control failure is observed, it should no longer be credited as highly unlikely for double contingency.

incredible: Having a probability of occurrence of < 10^{-6}/year/event. Demonstration of incredibility will be considered adequate if the resulting conditions are: (1) prohibited by physical laws or not achievable with quantities and materials allowed at the facility, (2) having no identifiable accident sequence that could lead to upset conditions, or (3) requiring a combination of several events such that the probability of occurrence is significantly less than that required to meet the double contingency principle.

independent: In the context of double contingency, two control failures are independent if the occurrence of one does not cause or increase the probability of occurrence of the other; if the probability of both occurring is independent of the order in which they occur; and if there are no identifiable common-mode failures. In the context of dual independent sampling, this implies that no single procedural error by an operator or laboratory analyst can lead to incorrect sample results. In the context of independent reviews, this means that a qualified criticality analyst, employed by the applicant or the NRC, should be capable of verifying the criticality safety basis of the covered operation without resorting to additional sources of information beyond those included with the criticality safety evaluation.

loss of control: Loss of a control of a controlled parameter occurs when an IROFS identified in the ISA, which maintains the controlled parameter within its safety limits, ceases to function as designed, or cannot be verified to function as designed, whether or not the controlled parameter actually exceeds its safety limits.

margin of subcriticality: The difference between the bias (or calculated value at which k_{eff} is expected to be critical) and the calculated value of k_{eff}, including allowances for uncertainty in the bias.

normal condition: A condition specifically allowed for as part of one of the normal modes of operation in the facility design, in which all controlled parameters are within their safety limits.

operating limit: A value of a controlled parameter to which actual operations are restricted, with sufficient margin to ensure that exceeding the safety limit is an unlikely event.

process variable: Any physical characteristic of a fissile material operation that is controlled within certain limits to maintain subcriticality (e.g., temperature or pressure) by indirectly limiting the value of a controlled parameter (mass, geometry, concentration, etc.).

redundancy and diversity: Having multiple controls sufficient to ensure that criticality is highly unlikely, but not meeting the full requirements of the double contingency principle.

safety limit: A value of a controlled parameter established by criticality safety evaluation. This typically would be equal to the subcritical limit, but could conceivably be less.

safety margin: The difference between the value of a controlled parameter at which a system is critical and the subcritical limit of that parameter.

subcritical: Demonstrated to not be critical. Having a value of k_{eff} less than the bias minus the uncertainty in the bias and minus the administrative margin.

subcritical limit: The bounding value of a controlled parameter in the normal case conditions. The actual operating limit is the value at which the parameter is controlled to ensure that the subcritical limit is not exceeded.

type of accident: In the context of 10 CFR Part 70, two accident sequences constitute different types of accidents if they differ in regard to the initiating event, the consequences (k_{eff} or the values of the criticality parameters of the resulting condition), or the physical mechanism by which the system reaches the ultimate state.

unlikely:[2] Having a probability of occurrence of $< 10^{-2}$/year/event. Such events should only be expected rarely during the lifetime of the facility, if at all. Demonstration of unlikelihood will be considered adequate if appropriate assurance measures are applied. As facility- and process-specific failure data are generated, the definition of unlikely should be refined; that is, if it is found that a control fails on a regular basis, it should no longer be credited as unlikely for double contingency.

validation: The process of demonstrating with reasonable assurance that a calculational method can accurately compute the value of k_{eff} for a certain AOA, by comparing calculations to accepted benchmark experiments similar in composition to the desired applications.

verification: The process of demonstrating with reasonable assurance that a calculational method performs mathematical functions correctly and consistently over a period of time.

[2] See footnote referenced in "highly unlikely" definition.

7.0 FIRE PROTECTION

7.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the applicant designed a facility that provides for "adequate protection against fires and explosions" (§ 70.64(a)(3)) and that is based on defense-in-depth practices (§ 70.64(b)). This review should also establish that radiological consequences from fires are considered in determining how the facility will meet the performance requirements of § 70.61.

7.2 RESPONSIBILITY FOR REVIEW

Primary: Fire Protection Engineer

Secondary: Project Manager

Supporting: Chemical Safety Reviewer
Nuclear Criticality Safety Reviewer
Quality Assurance Reviewer
Physical Security Reviewer

7.3 AREAS OF REVIEW

The review should address the adequacy of the following areas of fire protection:

A. Organization and Conduct of Operations: Organization and conduct of operations includes organization and management, training and qualifications, fire prevention, engineering review of design changes, quality assurance (QA), and documentation and recordkeeping.

B. Fire Protection Features and Systems: Plant fire protection features and systems include construction features; passive fire-rated barriers; process and operational features; fire detection and alarm systems; fire suppression systems and equipment; design bases documents; and inspection, maintenance, and testing of fire protection features and systems.

C. Manual Fire Fighting Capability: A baseline needs assessment should establish the minimum required capabilities of site fire fighting forces. This assessment should include minimum staffing, organization and coordination of onsite and offsite fire fighting resources, personal protective and fire fighting equipment, training, and prefire emergency planning.

D. Fire Hazards Analysis (FHA): The FHA consists of a systematic analysis of the fire hazards, an identification of specific areas and systems important to plant fire safety, the development of design basis fire scenarios, an evaluation of anticipated consequences,

and a determination of the adequacy of plant fire safety. FHA requirements are listed separately in Appendix D of this Standard Review Plan (SRP).

7.4 ACCEPTANCE CRITERIA

7.4.1 Regulatory Requirements

The regulations in 10 CFR 70.64(a) has a baseline design criterion for "fire protection" and requirements regarding defense-in-depth practices. In addition, § 70.61 contains performance requirements for the facility. The sections of 10 CFR Part 70 require that there be reasonable assurance of public health and safety and of the environment from the fire and explosion hazards of processing licensed material during normal operations, anticipated operational occurrences, and accidents.

7.4.2 Regulatory Guidance

The guidance in this SRP establishes the criteria for the staff in its review of the fire protection program provided by an applicant for authorization to construct and license to possess and use special nuclear material (SNM) at a mixed oxide (MOX) facility. The program must establish the fire protection policy for the protection of structures, systems, and components relied on for safety at the plant and the procedures, equipment, and personnel required to implement the program at the plant site.

While providing specific guidance in selected areas of fire safety, the staff's position as presented in this SRP also references a National Fire Protection Association, Inc. (NFPA) code that can provide information on standard practices that may be applied for MOX facilities in other areas of fire safety.[1] Significant guidance from Department of Energy DOE–STD–1066–97, "Fire Protection Design Criteria," has also been incorporated into this SRP. Guidance in regard to accident analysis may be found in Nuclear Regulatory Commission (NRC) NUREG/CR–6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook."

Additional industry documents that may provide useful background information for consideration in the design of MOX fuel fabrication facilities are listed in Section 7.7.

7.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find that the applicant's fire protection is acceptable if it provides reasonable assurance that the regulatory acceptance criteria below are adequately addressed and satisfied. Some of the information may be referenced to other sections of the SRP, or incorporated by reference, provided an adequate summary is provided and a single reference essentially contains all the information.

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[1] National Fire Protection Association, Inc. (NFPA) Standard 801, "Standards for Facilities Handling Radioactive Material," provides additional overall guidance on fire protection for fuel cycle facilities.

Where specific NFPA or other standards are referenced, the intent of the SRP is to refer the user to the latest standard. Because these standards may have been retitled or renumbered since the publication of this SRP, specific dates are not listed in the reference list. If the applicant references an NFPA or other industry standard, it should be dated (as the code of record) so that its criteria can be applied in the review of the applicant's submittal. Specified standards will normally be considered as acceptable means of meeting the review criteria. Alternative means, as well as deviations from specific sections of the standards, will also be considered but may require justification through analysis. Also, depending on the application, standards other than those referenced may be more appropriate for the fire protection required. In addition, hazards may exist or occur at the facility that are not specifically addressed in this SRP chapter. In its license application, the applicant is expected to select and reference the most applicable standards for all known hazards and fire protection measures at its facility beyond those identified in this SRP Chapter.

7.4.3.1 Organization and Conduct of Operations

The following organizational and operational guidance is appropriate for the MOX facility because of the significantly increased potential for fire-induced high radiological consequences over that for other types of fuel cycle facilities:[2]

A. Fire Protection Program

A fire protection program should be established at each MOX facility. The program should establish the fire protection policy for the protection of items relied on for safety (IROFS) at the plant and the procedures, equipment, and personnel required to implement the program at the plant site. The fire protection program should be acceptable if:

i. The fire protection program extends the concept of defense-in-depth to fire protection in fire areas that may affect IROFS, with the following objectives:

a. To prevent fires from starting;

b. To detect rapidly, control, and extinguish promptly those fires that do occur; and

c. To provide protection for IROFS so that a fire that is not promptly extinguished by the fire suppression activities will not result in uncontrolled release of radioactive materials.

ii. Responsibility for the overall fire protection program is assigned to a person who has management control over all organizations involved in fire protection activities.

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[2] Memorandum to the Commission on the Office of Nuclear Material Safety and Safeguards Fire Protection Plan, August 26, 1977, Attachment A.

Formulation and assurance of program implementation may be delegated to a staff composed of personnel prepared by training and experience in fire protection and MOX process safety to provide a balanced approach in directing the fire protection program for the MOX plant. The staff is responsible for:

a. Fire protection program requirements, including consideration of potential hazards associated with postulated fires, with knowledge of building layout and systems design;

b. Post-fire safety considerations;

c. Design, maintenance, surveillance, and QA of all fire protection features (e.g., detection systems, suppression systems, barriers, dampers, doors, penetration seals, and fire brigade equipment);

d. Fire prevention activities (administrative controls and training);

e. Fire brigade organization and training; and

f. Prefire planning.

iii. The organizational responsibilities and lines of communication pertaining to fire protection is defined between the various positions through the use of organizational charts and functional descriptions of the responsibilities of each position. The positions/organizations listed below are specifically designated; however, positions and responsibilities may be combined as appropriate depending on the scope of the responsibilities:

a. The upper level offsite or onsite management position that has management responsibility for the formulation, implementation, and assessment of the effectiveness of the MOX facility fire protection program;

b. The offsite or onsite management position(s) directly responsible for formulating, implementing, and periodically assessing the effectiveness of the fire protection program for the applicant's MOX plant, including fire drills and training conducted by the fire brigade and plant personnel and reporting the results of these assessments to the upper level manager responsible for fire protection, with recommendations for improvements or corrective actions as deemed necessary; and

c. The onsite management position responsible for the overall administration of the plant operations and emergency plans that include the fire protection and prevention program and that provide a single point of control and contact for all contingencies.

d. The onsite position(s) that:

(1) Implements periodic inspections to: minimize the amount of combustibles in areas with IROFS; determine the effectiveness of housekeeping practices; assure the availability and acceptable condition of all fire protection systems and equipment, fire stops, penetration seals, and fire-retardant coatings (if any); and assure that prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence;

(2) Is responsible for the fire fighting training for production plant personnel and the plant's fire brigade; design and selection of equipment; periodic inspection and testing of fire protection systems and equipment in accordance with established procedures; and evaluation of test results and determination of the acceptability of the systems under test;

(3) Assists in the critique of all fire drills to determine how well the training objectives have been met;

(4) Reviews and evaluates proposed work activities to identify potential transient fire loads;

(5) Implements a program for indoctrination of all plant contractor personnel in appropriate administrative procedures that implement the fire protection program and the emergency procedures relative to fire protection; and

(6) Implements a program for instruction of personnel on the proper handling of accidental events, such as leaks or spills of flammable materials, that are related to fire protection.

e. The onsite position responsible for fire protection QA. This position is responsible for assuring the effective implementation of the fire protection program by planned inspections, scheduled audits, and verification that the results of these inspections or audits identifying significant adverse conditions are promptly reported to cognizant management personnel.

f. The positions that are part of the plant fire brigade:[3]

[3] If a capable fire department exists onsite (if site is a DOE reservation) or nearby, an agreement with such a department may be made for facility use of that department's services. The facility's baseline needs assessment (7.3.C) will establish the minimum required capabilities for onsite fire fighting, which may or may not include a facility fire brigade.

(1) The plant fire brigade positions are responsible for fighting fires. The authority and responsibility of each fire brigade position relative to fire protection is clearly defined.

(2) The responsibilities of each fire brigade position corresponds with the actions required by the fire fighting procedures.

(3) The responsibilities of the fire brigade members under normal plant conditions do not conflict with their responsibilities during a fire emergency.

g. Personnel qualifications

(1) The position responsible for formulation and implementation of the fire protection program has within its organization (or as a consultant) a manager selected on the basis of education, experience, and advancement as an industrial fire protection engineer.

(2) The qualifications for members of the fire brigade include satisfactory completion of a physical examination for performing strenuous activity.

(3) The personnel responsible for the maintenance and testing of the fire protection systems are qualified by training or experience for such work.

(4) The personnel responsible for training the fire brigade are qualified by training and experience for such work.

(5) During operation and construction or major modification of the MOX facility, the superintendent (or equivalent position) of the MOX facility has the lead responsibility for all site fire protection.

B. Administrative Controls

Administrative controls should be used to maintain the performance of the fire protection system and personnel. These controls should establish procedures to:

i. Govern bulk storage of combustible materials inside or adjacent to buildings or systems having IROFS during operation or maintenance periods.

ii. Govern the handling and limitation of the use of ordinary combustible materials, combustible and flammable gases and liquids, combustible high efficiency particulate air (HEPA) and charcoal filters, dry ion exchange resins, or other combustible supplies in areas containing IROFS.

iii. Govern the handling of and limit transient fire loads such as combustible and flammable liquids, wood and plastic products, or other combustible materials in

buildings containing IROFS during all phases of operation, and especially during maintenance or modification operations. Use of wood products is permitted only when noncombustible products are not practical from a process consideration. If wood or wood products are required, the wood is pressure treated with a flame retardant. Equipment or supplies shipped in untreated combustible packing or containers may be unpacked inside the plant production areas if required for valid operating reasons. However, all combustible materials are to be removed from the area immediately following unpacking. Such transient combustible material, unless stored in approved containers, is not to be left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material such as wood or paper excelsior or polyethylene sheeting is placed in metal containers with tight-fitting, self-closing metal covers.

iv. Govern the use of ignition sources by use of a hot work permit system to control welding, flame cutting, brazing, or soldering operations. A separate permit is issued for each area where work is to be done. If work continues over more than one shift, the permit is valid for not more than 24 hours when the facility is operating or for the duration of a particular job during plant shutdown.

v. Control the removal of work-generated combustible waste from the work activity immediately following completion of the activity, or at the end of each work shift, whichever comes first.

vi. Prohibit the use of open flames or combustion-generated smoke for leak testing.

vii. Maintain periodic housekeeping inspections to ensure continued compliance with these administrative controls.

viii. Control disarming of fire detection or fire suppression systems by a permit system. Fire watches should be established in areas where systems are so disarmed.

ix. Test and maintain the fire protection equipment and the emergency lighting and communication. A test plan that lists the responsible personnel positions in connection with routine tests and inspections of the fire detection and protection systems is developed. The test plan contains the types, frequency, and detailed procedures for testing. Procedures also contain instructions on maintaining fire protection during those periods when the fire protection system is impaired or during periods of plant maintenance, such as, fire watches or temporary hose connections to water systems.

x. Control actions to be taken by an individual discovering a fire, for example, notification of control room, attempt to extinguish fire, and actuation of the local fire suppression system(s).

xi. Control actions to be taken by a designated operator to determine the need for fire brigade assistance upon report of a fire or receipt of alarm on control room

annunciator panel; for example, announcing the location of fire over public address system, sounding fire alarms, and notifying the shift supervisor (or equivalent position) and the fire brigade leader of the type, size, and location of the fire.

xii. Define the strategies for fighting fires in all areas containing IROFS and areas presenting a hazard to IROFS. These strategies, which are reflected in the prefire plans, designate:

a. Fire hazards in each area covered by the specific prefire plans.

b. Fire extinguishants best suited for controlling the fires associated with the fire hazards in that area and the nearest location of these extinguishants.

c. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and best station or elevation for fighting the fire. All access and egress routes that involve locked doors are specifically identified in the procedure, with the appropriate precautions and methods for access specified.

d. Management of plant systems to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical systems in the area or zone covered by the specific fire fighting procedure that could increase the hazards in the area because of overpressurization and/or electrical hazards).

e. Vital heat-sensitive system components that need to be kept cool while fighting a local fire, particularly hazardous combustibles that need cooling.

f. Organization of fire fighting brigades and the assignment of special duties according to job title so that all fire fighting functions are covered by any complete shift personnel complement. These duties include having command control of the brigade, transporting fire suppression and support equipment to the fire scenes, applying the extinguishant to the fire, communicating with the control room, and coordinating with outside fire departments.

g. Potential radiological and toxic hazards in fire areas or zones.

h. Operations requiring control room and designated management coordination or authorization.

i. Instructions for plant operators and general plant personnel during fires.

xiii. Establish and implement a penetration seal tracking program to record pertinent information regarding the emplacement and modification of fire barrier penetration

seals that are defined in the Integrated Safety Analysis (ISA) Summary or FHA as IROFS.

7.4.3.2 Fire Protection Features and Systems

The facility fire protection features and systems should be considered acceptable if the following conditions are met:

A. Buildings containing IROFS are designed to qualify as Type I construction as defined by NFPA Standard 220, "Standard on Types of Building Construction." This includes structural building components such as walls, floors, roofs, columns, and beams as well as interior building features. The process layout separates and isolates, as much as practical, operations presenting fire hazards. This can be accomplished by distance, or compartmentalizing using fire barriers, or both. In addition, adequate fire safety criteria for adjoining process facilities, or facilities close to each other, or near bulk hazardous material storage is defined in NFPA Standard 80A, "Recommended Practice for Protection of Buildings from Exterior Fire Exposures."

B. The structural shell (and its supporting members) surrounding any area handling plutonium, where the plutonium could be accidently dispersed and cause exposure to either operating personnel or the public, is designed with sufficient fire resistance that it will remain standing and continue to act as a confinement structure during any credible accident conditions resulting from fires. The fire resistance rating of this shell is at least 2 hours and is attained by integral parts of this structure (concrete slabs, walls beams, columns and ceilings/roofs). Penetrations in the shell incorporate equivalent protection.

C. Special facilities such as SNM storage, radioactive waste, or other facilities with a potential for significant releases of radioactivity are designed and constructed using building components of fire-resistant and noncombustible material, particularly in locations vital to the functioning of confinement systems. The fire resistance rating of SNM storage facilities is at least 2 hours and is attained by integral parts of this structure (concrete slabs, walls, beams, columns, and ceiling/roofs). Combustible materials are not used in the construction of confinement systems.

D. Exposed interior walls or ceilings (including ceilings formed by the underside of roofs) and any factory-installed facing material have an Underwriters Laboratories Inc. (UL) listed/Factory Mutual Research Corporation approved flame spread rating of 25 or less and a smoke developed rating of 50 or less, per the American Society for Testing and Materials (ASTM) ASTM–E–84, "Standard Test Method for Surface Burning Characteristics of Building Materials."

E. The use of carpets and rugs is minimized to the extent practicable in buildings containing SNM. If determined to be necessary, carpets and rugs are tested in accordance with NFPA Standard 253 (ASTM–E–648, "Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source") when applying the floor finish requirements of The Life Safety Code (NFPA Standard 101) to MOX facilities.

Carpets and rugs used in storage or industrial occupancies (no criteria in NFPA Standard 101) have a critical radiant flux not less than 0.45 watts per square cm (0.40 BTU per second per square ft) in areas unprotected by an automatic fire suppression system and 0.22 watts per square cm (0.20 BTU per second per square ft) in protected areas.

F. Storage racks in SNM (oxides, pellets, or fuel rods) storage facilities are noncombustible and designed to securely hold storage containers in place, ensure proper separation of storage containers, and maintain structural integrity during a fire. No combustible material is stored in the SNM storage facilities in a location that would endanger the storage facility or stored material if a fire should occur.

G. Electrical wiring for MOX facilities is designed and provisions exist to maintain such wiring in accordance with the applicable provisions of the National Electric Code (NFPA Standard 70).

H. Lightning protection for plant buildings determined to be IROFS is designed in accordance with the applicable provisions of NFPA Standard 780, "Lightning Protection Code."

I. Ventilation systems in areas containing IROFS are designed to minimize the spread of fire, smoke, hot gases, and products of combustion from the area of fire origin and prevent explosions in accordance with the applicable provisions of NFPA Standard 69, " Standard on Explosion Prevention Systems," and Standard 90A, "Standard for the Installation of Air Conditioning and Ventilating Systems." Where ventilation systems are designed to prevent the release of radioactive materials, HEPA filters of these systems will satisfy the requirements of UL Standard 586, "High Efficience Air Filtration Units." Further fire protection guidance for nuclear filter plenums is contained in Appendix E of this SRP.

J. Where fire barriers are penetrated by the confinement system's ventilation ducting, fire dampers are appropriately used to maintain the barrier integrity. However, the closure of such dampers does not compromise the functions of the confinement system where the loss of confinement might pose a greater threat than the spread of fire. In such cases, alternative fire protection means (e.g., duct wrapping, duct enclosure, or rerouting) are used as a substitute for fire barrier closure. Sprinkler systems, such as those designed as a "water curtain," are not considered a fire barrier substitute.

K. Building layout provides a safe means of egress for plant personnel in the event of fire in accordance with the applicable provisions of The Life Safety Code (NFPA Standard 101). Physical security of nuclear facilities, by design, may inadvertently institute controls that delay worker egress and firefighter access during fire events. Provisions are made to minimize these delays. Emergency lighting for the purpose of personnel egress is in accordance with NFPA Standard 101. The design basis for emergency lighting (location, intensity, and duration) required to perform any functions relied on for safety during a loss of power is determined from engineering evaluations and the ISA.

L. The design of openings in passive fire-rated barriers incorporates suitable automatic or fixed closure devices or components, such as fire doors, fire dampers, and fire-rated penetration seals. Manual activation of fire closure devices may be used where other safety considerations may preclude the use of automatic closure devices as determined by the ISA or FHA (see Appendix E, Section 2.4). Fire doors are designed and installed in accordance with the applicable provisions of NFPA Standard 80, "Standard for Fire Doors and Fire Windows." Fire dampers are designed and installed in accordance with the applicable provisions of UL Standard 555, "Standard for Fire Dampers and Ceiling Dampers."

M. Plant areas with the potential for large spills of flammable or combustible liquids are identified and means of containing, such as, dikes, and disposing of such spills are provided for in the facility design. The design of containment and drainage systems considers the rate of water discharge from fixed suppression systems and/or hose lines and is capable of preventing the spread of combustible liquids from pits or confining areas. Flammable and combustible liquids are stored, handled, and used in accordance with the applicable provisions in NFPA Standard 30, "Flammable and Combustible Liquids Code," and/or other industry standards.

N. Plant areas are identified where credible risk of creation of a flammable mixture with hydrogen or other flammable or oxidizing gases exists. Preventive measures in accordance with NFPA Standard 50, "Standard for Bulk Oxygen Systems at Consumer Sites;" Standard 50A, "Standard for Gaseous Hydrogen Systems at Consumer Sites;" Standard 50B, "Standard for Liquified Hydrogen at Consumer Sites;" Standard 51, "Standard for Oxygen-Fuel Gas Systems for Welding, Cutting, and Allied Processes;" Standard 55, "Standard for Compressed and Liquified Gases in Portable Cylinders;" Standard 58, "Standard for Storage and Handling of Liquified Petroleum Gases;" Standard 69, "Standard on Explosion Prevention Systems;" and/or other industry standards are provided.

O. Flammable gas is not introduced into SNM-processing buildings except when specifically required for process reasons. Where hydrogen is necessary for processes:

   i. Hydrogen lines introduced into plutonium processing buildings are either designed to maintain functionality when subjected to an earthquake, or sleeved such that the outer pipe is directly vented to the outside, or are equipped with excess flow valves so that the hydrogen concentration in the affected areas will not exceed 2% in case of a line break. Shutoff valves are installed as close as possible to the reducing furnaces, or other using devices, but the shutoff valves are located so that they are not likely to be involved in a fire involving the using device.

   ii. Bulk storage of hydrogen is outside of all process buildings. Cryogenic storage is located so that the possibility and consequences of a catastrophic spill are minimized. High pressure tube trailers are located so that the long axis of the tube cylinders are parallel and not perpendicular to the process buildings. Master shutoff valves are installed at the bulk storage tank or manifold.

iii. Entry of air into a furnace operating with reducing gas is precluded by the use of inert-gas-purged locks or other suitable means at the furnace entry and exit. Furnace gas is exhausted through an enclosed, noncombustible construction, filtered off-gas system.

iv. Process furnaces are provided with a system for automatically shutting off the furnace gas and purging with inert gas in the event of a power failure, loss of coolant water, loss of exhaust fan, overtemperature, low flow pressure and/or high flow in gas line, or detection of hydrogen in the vicinity of the furnace.

P. The facility design incorporates a fire alarm system, designed in accordance with the applicable provisions of NFPA Standard 72, "National Fire Alarm Code," provided throughout areas as determined to be relied on for safety by the ISA/FHA. The system incorporates features such as local and remote annunciation, primary and secondary power supplies, and audible and visual alarm devices. The alarm system also includes supervisory devices for all critical fire protection functions.

Q. The facility design incorporates an adequate and reliable water supply system, designed in accordance with NFPA standards for fire protection use. The system consists of the water source, dedicated storage facilities, fire pumps, a distribution-piping network, sectional isolation valves, and fire hydrants and standpipes, as applicable to the facility. The design of the fire pumps, where provided, is in accordance with the applicable provisions of NFPA Standard 20, "Standard for Installation of Centrifugal Fire Pumps." If pumps are required to meet system pressure or flow requirements, a sufficient number of pumps are provided to ensure that 100% capacity will be available assuming failure of the largest pump or loss of offsite power (e.g., three 50% pumps or two 100% pumps). This can be accomplished, for example, by providing either: electric motor-driven fire and diesel engine-driven pump(s); or two or more electric motor-driven fire pumps connected to emergency power buses and designed to maintain functionality when subjected to an earthquake. Common tanks are permitted for fire and sanitary or service water storage. When this is done, however, minimum fire water storage requirements are dedicated by passive means, for example, use of a vertical standpipe for other water services. Administrative controls, including locks for tank outlet valves, are unacceptable as the only means to ensure minimum water volume reserved for fire service needs. Note that if standpipes are used for other water services, they should be arranged so that a leak or other malfunction will not be able to drain off the water reserved for the fire service needs. Designs of the distribution piping, valves, and fire hydrants are in accordance with the applicable provisions of NFPA Standard 24, "Standard for the Installation of Private Service Mains and their Appurtenances." Water supply requirements in terms of stored volume and/or supply rates are determined in the FHA. Standpipe and hose systems are in accordance with the applicable provisions of NFPA Standard 14, "Standard for the Installation of Standpipes and Hose Systems."

R. Automatic fire suppression is incorporated in areas of significant, or potentially significant, fire loading to protect IROFS. Manual activation of fire suppression systems may be used

where other safety considerations may preclude the use of automatic suppression as determined by the ISA or FHA. The design and installation of fire suppression systems and equipment is in accordance with the applicable provisions of appropriate NFPA standards. Commonly applied NFPA Standards include NFPA Standard 10, "Standard for Portable Fire Extinguishers;" Standard 11, "Standard for Low Expansion Foam;" Standard 11A, "Standard for Medium- and High- Expansion Foam Systems;" Standard 12, "Standard on Carbon Dioxide Extinguishing Systems;" Standard 13, "Standard for the Installation of Sprinkler Systems;" Standard 15, "Standard for Water Spray Fixed Systems for Fire Protection;" Standard 16, "Standard for the Installation of Deluge Foam-Water Sprinkler and Foam-Water Spray Systems;" Standard 16A, "Standard for the Installation of Closed-Head Foam Water Sprinkler Systems;" and Standard 2001, "Standard on Clean Agent Extinguishing Systems." In addition, total reliance is not placed on a single fire suppression system. Appropriate backup fire suppression capability is provided. A single active failure or a crack in a moderate-energy line (pipe) in the fire suppression system does not impair both the primary and backup fire suppression capability. For example, neither the failure of a fire pump, its power supply or controls, nor a crack in a moderate-energy line in the fire suppression system should result in loss of function of both sprinkler and hose standpipe systems in an area protected by such primary and backup systems. Also, as a minimum, there should be capability for manual fire suppression in areas containing IROFS following the most severe earthquake expected in the geological area where the facility is located. The required water quantities, flow, and pressure would be determined from the FHA. The need for fire detection and/or suppression following an earthquake may be determined from the ISA/FHA.

S. The applicant commits to providing a program of regular inspection, testing, and maintenance of fire protection equipment in accordance with the provisions of appropriate NFPA or other industry standards. A commonly applied standard for water-based systems is NFPA Standard 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems."

T. Safety controls and interlocks for combustible liquids, flammable liquids, and flammable gases and their associated delivery systems are tested periodically and after maintenance operations.

U. Combustible and pyrophoric metals are stored and handled in accordance with the applicable codes and/or industry standards. Additional information on storage and handling of combustible and pyrophoric metals may be found in DOE Handbook–1081–9, "Primer on Spontaneous Heating and Pyrophorocity," December 1994, and DOE–STD–3013–99, "Stabilization, Packaging, and Storage of Plutonium-Bearing Materials," November 1999.

V. Operating controls and limits for the handling of pyrophoric materials are established. An adequate supply of the appropriate extinguishing agent should be available where combustible and pyrophoric metals are present.

W. Provisions are made to construct gloveboxes and windows of noncombustible materials. A means of fire detection is provided if pyrophoric materials, oxidizers, or organic liquids are handled. Fire suppression or a fixed inerting system is provided if combustible materials are present, or could be present, in quantities sufficient to cause a breach of integrity. If a fixed suppression system is utilized, the internal pressurization is calculated to prevent gloves from falling or being blown off. If an inerting system is used to reduce fire risk to IROFS, the oxygen concentration is continually monitored to assure that the oxygen concentration does not exceed 25% of the level required for combustion by means of an alarm and other measures (such as shutdown of operations and electric power to the glovebox) as warranted by the FHA/ISA.

X. Glovebox ventilation ducting is provided with separation/isolation dampers or doors to minimize fire propagation. Fire barriers are also provided between individual or groups of gloveboxes or within glove lines where warranted by the FHA. If closure of the separation/isolation dampers or doors does not compromise the functions of the confinement system, the separation/isolation mechanism is shut by a fusible device or upon activation of the glovebox automatic fire suppression or detection system. In the case of fire detection systems, precautions such as heat detectors or dual-zone smoke detectors should be used to avoid inadvertent damper operation and shutdown of the glovebox ventilation system.

Y. Glovebox primary exhaust openings are provided with prefilters and fire screens to reduce vapor mist and fire propagation. The fire screens are stainless steel screens (8–16 mesh) or a perforated stainless steel plate using the same opening sizes. Glovebox exhaust ventilation lines are also designed so that each box has its own exhaust port so that flame or hot fire gases will not travel from one glovebox to another through a common header or interconnection arrangement. Single exhaust manifolds that connect an entire glovebox line shall not be used. Exceptions may be made where necessary for confinement considerations if compensation for fire risk is provided, if necessary.

Z. Where flammable or combustible solvents are used, they are stored and handled in accordance with the guidance of NFPA Standard 30, "Flammable and Combustible Liquids Code." Approved operating controls and limits for the use of flammable or combustible solvents are established. An approved fixed fire suppression system is installed or the process carried out in an inert atmosphere such as nitrogen. The FHA should identify the specific hazards and the best fire protection method.

AA. Inert gas purge and vent systems are used for SNM-bearing solution tanks to minimize potential accumulation of a flammable mixture of hydrogen gas, including a means of venting hydrogen gas from process piping. If inert gas is not used to purge the system, the ventilation system must be capable of maintaining hydrogen concentrations below 25% of the lower flammable limit under all expected process conditions.

BB. Incinerators, boilers, and furnaces are located in separate fire areas with automatic suppression and installed and maintained in accordance with NFPA Standard 31, "Standard for Installation of Oil Burning Equipment;" Standard 54, "National Fuel Gas

Code;" and Standard 8501, "Standard for Single Burner Oil Operation;" and/or other applicable industry standards.

CC. Facility laboratories using chemicals or nuclear materials are operated in accordance with the safety criteria in NFPA Standard 45, "Standard for Fire Protection for Laboratories Using Chemicals," and/or NFPA Standard 801, "Standards for Facilities Handling Radioactive Material," as applicable.

DD. Provisions for the drainage and holdup of contaminated fire water following a fire are incorporated into the design.

7.4.3.3 Manual Fire Fighting Capability

The following manual fire fighting guidance for the MOX facility, because of the significantly increased potential for fire-induced high radiological consequences over that for other types of fuel cycle facilities, is closely related to the guidance provided for light water power reactors. The manual fire fighting capability should be acceptable if:

A. The recommendations for organization, training, and equipment specified in, "Standard on Industrial Fire Brigades," (NFPA Standard 600), are considered appropriate criteria for organizing, training, and operating a plant fire brigade.

B. A site fire brigade trained and equipped for fire fighting is established to ensure adequate manual fire fighting capability for all areas of the plant containing IROFS. The minimum fire brigade members to be available on each shift is determined from the baseline needs assessment (the minimum required for commercial reactor facilities is five). The brigade leader and at least two brigade members have sufficient training in or knowledge of plant safety and process systems to understand the effects of fire and fire suppression activities on the ability to control release of radioactive materials. The qualification of fire brigade members is in accordance with the guidance in NFPA Standard 600 for the type of duties to be performed. The shift supervisor or equivalent position is not a member of the fire brigade. The brigade leader is competent to assess the potential safety consequences of a fire and to advise control room personnel.

C. The minimum equipment provided for the brigade consists of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus using full-face positive-pressure masks approved by the National Institute for Occupational Safety and Health is provided for fire brigade, damage control, and control room personnel. An extra mask is available for each of the required fire brigade personnel. Control room personnel may be furnished breathing air by a manifold system piped from a storage reservoir if practical. Service or rated operating life is a minimum of one-half hour for the self-contained units.

D. At least two extra air bottles are located onsite for each self-contained breathing unit. In addition, an onsite 6-hour supply of reserve air is provided and arranged to permit quick

and complete replenishment of exhausted supply air bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air are used and compressors are operable assuming a loss of offsite power. Special care is taken to locate the compressor in areas free from dust and contaminants.

E. The fire brigade training program ensures that the capability to fight potential fires is established and maintained. The program consists of an initial classroom instruction program followed by periodic classroom instruction, fire fighting practice, and fire drills.

i. The initial classroom instruction includes:

a. Indoctrination of the plant fire fighting plan with specific identification of responsibilities for each individual;

b. Identification of the type and location of fire hazards and associated types of fires that could occur in the plant;

c. The toxic and corrosive characteristics of expected products of combustion;

d. Identification of the location of fire fighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes to each area;

e. The proper use of available fire fighting equipment and the correct method of fighting each type of fire; types of fires covered include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires involving uranium and/or plutonium metal, fires resulting from construction or maintenance activities, and record file fires;

f. The proper use of communication, lighting, ventilation, and emergency breathing equipment;

g. The proper method for fighting fires inside buildings and confined spaces;

h. The direction and coordination of the fire fighting activities (fire brigade leaders only);

i. Detailed review of fire fighting strategies and procedures;

j. Review of the latest plant modifications and corresponding changes in fire fighting plans;

k. The plant fire brigade training is coordinated with the responsible fire department so that responsibilities and duties are delineated in advance. This

coordination is part of the training course and is included in the training of the responsible fire department staff as appropriate.

l. The responsible fire departments are provided training in operational precautions when fighting fires on MOX facility sites and are made aware of the need for radiological protection of personnel and the special hazards associated with a MOX facility site.

Note: Items (i) and (j) may be deleted from the training of no more than two of the nonoperations personnel who may be assigned to the fire brigade.

ii. The instruction is provided by qualified individuals who are knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in a MOX plant.

iii. Instruction is provided to all fire brigade members and fire brigade leaders.

iv. Regularly planned meetings are held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary.

v. Periodic refresher training sessions are held to repeat the classroom instruction program for all brigade members over a 2-year period. These sessions may be concurrent with the regularly planned meetings.

vi. Practice

a. Practice sessions are held for each shift fire brigade on the proper method of fighting the various types of fires that could occur in a MOX facility. These sessions should provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting.

b. Practice sessions are provided at least once per year for each fire brigade member.

vii. Drills

a. Fire brigade drills are performed in the plant so that the fire brigade can practice as a team.

b. Drills are performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but as a minimum in at least two drills per year. A sufficient number of these drills, but not less than one for each shift fire brigade per year, are unannounced to determine the fire fighting readiness of the plant fire brigade, brigade leader,

and fire protection systems and equipment. Persons planning and authorizing an unannounced drill ensure that the responding shift fire brigade members are not aware that a drill is being planned until it starts. Unannounced drills are not scheduled closer than 4 weeks. At least one drill per year is performed on a "back shift" for each shift fire brigade.

c. Drills are preplanned to establish the training objectives of the drill and are critiqued to determine how well the training objectives have been met. Unannounced drills are planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or of individual fire brigade members are remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance is followed by a repeat drill within 30 days.

d. These drills provide for responsible fire department participation at least annually.

e. At 3-year intervals, a randomly selected unannounced drill is critiqued by qualified individuals independent of the MOX plant staff. A copy of the written report from such individuals is available for NRC review.

f. Drills include, at a minimum:

(1) Assessment of fire alarm effectiveness, time required to notify and assemble the fire brigade, and selection, placement, and use of equipment and fire fighting strategies.

(2) Assessment of the knowledge of each brigade member concerning his or her role in the fire fighting strategy for the area assumed to contain the fire. Assessment of the conformance of each brigade member with established plant fire fighting procedures and use of fire fighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practical.

(3) The simulated use of fire fighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drills so that brigade members are trained in fighting fires in various plant areas. The situation selected should simulate the size and arrangement of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, to obtain equipment, and to organize for the fire, assuming loss of automatic suppression capability.

(4) Assessment of the brigade leader's direction of the fire fighting effort as to thoroughness, accuracy, and effectiveness.

viii. Records

Individual records of training provided to each fire brigade member, including drill critiques, are maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These training records are available for NRC review. Retraining or broadened training for fire fighting within buildings is scheduled for all those brigade members whose performance records show deficiencies.

7.4.3.4 Fire Hazards Analysis

The FHA should be considered acceptable if it reflects current conditions throughout the facility and the applicant commits to reviewing and updating the FHA as necessary at defined, regular intervals to document that fire protection measures are adequate to ensure plant fire safety. In addition, the FHA should be revised to incorporate significant changes and modifications to the facility, processes, or inventories, as needed. (The level of detail provided in the FHA should reflect the complexity of the facility and the anticipated consequences from fire events. A more detailed description of the requirements for an FHA is provided in Appendix D of this SRP.)

7.5 REVIEW PROCEDURES

7.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 7.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use SNM.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

Specifically, the application material should adequately addresses commitments related to Sections 7.3(A), 7.3(C), and 7.3(D), and the fire protection features and systems identified in Section 7.3(B).

B. License To Possess and Use SNM

Specifically, the application material should address the areas described in Sections 7.3(A), 7.3(C), and 7.3(D) in full and update the information described in Section 7.3(B) to reflect any changes in fire protection features and design.

If the primary reviewer verifies that fire protection is adequately addressed for the construction approval review or the review for the license to possess and use SNM, the primary reviewer should accept the application for the safety evaluation in Section 7.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

7.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 7.5.1(A) (construction) or Section 7.5.1(B) (license), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 7.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

The primary reviewer should verify that the applicant's commitments and goals as they relate to fire protection are adequate to meet or exceed the regulatory acceptance criteria in Section 7.4.3. The primary reviewer should focus on Section 7.4.3.2, "Fire Protection Features and Systems," with emphasis on building construction, water supply and distribution systems, ventilation systems fire protection, major combustible liquid storage areas, and facility fire suppression and detection systems. Fire protection aspects of process areas and gloveboxes should be described to the extent possible, considering the present stage of the applicant's design process.

B. License To Possess and Use SNM

The primary reviewer should focus on Section 7.4.3.1, "Organization and Conduct of Operations," Section 7.4.3.3, "Manual Fire Fighting Capability," and Section 7.4.3.4, "Fire Hazards Analysis," with a re-review of Section 7.4.3.2 if any significant changes have been made or information added.

The primary reviewer should also review sections of the ISA Summary that address fire protection to ensure that those sections are consistent with the fire protection portion of the application. The primary reviewer should also assure that the requirements for placement and reliability of fire protection measures are consistent with the ISA Summary.

The secondary reviewer should confirm that descriptions in the fire protection section are consistent with descriptions in other sections of the application that may interface with fire safety. The secondary reviewer may also request support from other technical reviewers as required.

Supporting reviewers should confirm that provisions made in the applicant's fire protection section are in accordance with other SRP sections within their areas of responsibility. For example, the nuclear criticality safety reviewer, as a supporting reviewer of fire protection, should establish that the program described by the applicant provides reasonable assurance that a water-based suppression system will not adversely affect criticality safety. The physical security reviewer should assist in the review of access and egress requirements.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the fire protection input for the Safety Evaluation Report (SER), as described in Section 7.6 using the acceptance criteria from Section 7.4. The primary reviewer should coordinate the fire protection input with the balance of the reviews and the SER.

7.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

The staff reviewed the application material for construction approval for [insert facility name] according to Chapter 7.0 of NUREG–1718. The staff evaluated [state what was evaluated] and found [state what was found]. The applicant provided fire protection features and systems consistent with the level of design it provided in the license application's material for construction approval. In addition to the fire hazards analysis, the applicant also made commitments related to the fire safety organization and conduct of operation; fire protection features and systems; and manual fire fighting capability.

The staff concluded that the applicant's proposed equipment, facilities, and commitments provide a reasonable level of assurance that the applicant's design bases will provide adequate fire protection to meet the safety performance requirements and the baseline design criteria for construction approval in accordance with 10 CFR Part 70.

The staff could document the safety evaluation for the review for the license to possess and use SNM as follows:

The staff reviewed the license application for a license to possess and use special nuclear material for [insert facility name] according to Chapter 7.0 of NUREG–1718. The staff evaluated [state what was evaluated] and found [state what was found]. The applicant updated a fire hazards analysis that documents all significant facility fire hazards, fire protection features designed to control those hazards, and the overall adequacy of facility fire safety. In addition to the fire hazards analysis, the applicant also provided the following information in the license application on the fire safety organization and conduct of operation; the fire protection features and systems; and the manual fire fighting capability.

The staff concluded that the applicant's proposed equipment, facilities, and procedures provide a reasonable level of assurance that adequate fire protection will be provided and maintained for those items determined to be relied upon for safety to meet the safety performance requirements and the baseline design criteria of 10 CFR Part 70.

7.7 REFERENCES

American Society for Testing and Materials (ASTM). ASTM–E–84, "Standard Test Method for Surface Burning Characteristics of Building Materials."

Department of Energy (U.S.) (DOE). DOE–STD–1066–97, "Fire Protection Design Criteria." DOE: Washington, DC. March 1997.

Department of Energy (U.S.) (DOE). Draft DOE–STD–5XXX–99, "Stabilization, Packaging, and Storage of Plutonium-Bearing Materials." DOE: Washington, DC. March 1999.

Factory Mutual Research Corporation. "Factory Mutual System Approval Guide–Equipment, Materials, Services, and Conservation of Property."

Institute of Electrical and Electronics Engineers, Inc. (IEEE). Standard 690, "IEEE Standard for the Design and Installation of Cable Systems for Class 1E Circuits in Nuclear Power Generating Stations."

National Fire Protection Association, Inc. (NFPA). Standard 10, "Standard for Portable Fire Extinguishers."

———. Standard 11, "Standard for Low Expansion Foam."

———. Standard 11A, "Standard for Medium- and High-Expansion Foam Systems."

———. Standard 12, "Standard on Carbon Dioxide Extinguishing Systems."

———. Standard 13, "Standard for the Installation of Sprinkler Systems."

———. Standard 14, "Standard for the Installation of Standpipes and Hose Systems."

———. Standard 15, "Standard for Water Spray Fixed Systems for Fire Protection."

———. Standard 16, "Standard for the Installation of Deluge Foam-Water Sprinkler and Foam-Water Spray Systems."

———. Standard 16A, "Standard for the Installation of Closed-Head Foam Water Sprinkler Systems."

———. Standard 20, "Standard for the Installation of Centrifugal Fire Pumps."

———. Standard 24, "Standard for the Installation of Private Service Mains and their Appurtenances."

———. Standard 25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems."

———. Standard 30, "Flammable and Combustible Liquids Code."

———. Standard 31, "Standards for Installation of Oil Burning Equipment."

———. Standard 45, "Standard for Fire Protection for Laboratories Using Chemicals."

———. Standard 50, "Standard for Bulk Oxygen Systems at Consumer Sites."

———. Standard 50A, "Standard for Gaseous Hydrogen Systems at Consumer Sites."

———. Standard 50B, "Standard for Liquified Hydrogen Systems at Consumer Sites."

———. Standard 51, "Standard for Oxygen-Fuel Gas Systems for Welding, Cutting, and Allied Processes."

———. Standard 54, "National Fuel Gas Code."

———. Standard 55, "Standard for Compressed and Liquified Gases in Portable Cylinders."

———. Standard 58, "Standard for Storage and Handling of Liquified Petroleum Gases."

———. Standard 69, "Standard on Explosion Prevention Systems."

———. Standard 70, "National Electric Code."

———. Standard 72, "National Fire Alarm Code."

———. Standard 80, "Standard for Fire Doors and Fire Windows."

———. Standard 80A, "Recommended Practice for Protection of Buildings from Exterior Fire Exposures."

———. Standard 90A, "Standard for the Installation of Air Conditioning and Ventilating Systems."

———. Standard 101, "Life Safety Code."

———. Standard 220, "Standard on Types of Building Construction."

———. Standard 251, "Standard Methods of Tests of Fire Endurance of Building Construction and Materials."

———. Standard 253 (ASTM E–648, "Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source").

———. Standard 600, "Standard on Industrial Fire Brigades."

———. Standard 780, "Lightning Protection Code."

———. Standard 801, "Standards for Facilities Handling Radioactive Material."

———. Standard 803, "Standard for Fire Protection for Light Water Nuclear Power Plants."

———. Standard 2001, "Standard on Clean Agent Extinguishing Systems."

———. Standard 8501, "Standard for Single Burner Oil Operation."

Nuclear Regulatory Commission (U.S.) (NRC). "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Underwriters Laboratories, Inc. "Underwriters Laboratories Building Materials Directory."

———. "Underwriters Laboratories Fire Protection Equipment Directory."

———. Standard 555, "Standard for Fire Dampers and Ceiling Dampers."

———. Standard 586, "High Efficiency Air Filtration Units."

7.8 DEFINITIONS

combustible: A material, in the form and condition in which it is used, that will ignite and burn.

combustible liquid:[4] A liquid having a flash point at or above 37.8 °C (100 °F).

fire area: A location bounded by fire-rated construction, having a minimum fire resistance rating of 2 hours.

[4] Definitions as used in National Fire Protection Association, Inc. (NFPA) "Fire Protection Handbook" and NFPA Standards.

fire barrier: A continuous membrane such as a wall, floor, or roof that is constructed to limit fire spread and the movement of smoke. Fire barriers have fire resistance ratings and may have protected openings.

fire brigade: Facility personnel trained in plant fire fighting operations.

fire door: A fire-rated door assembly.

Fire Hazards Analysis (FHA): A comprehensive assessment of potential fires to ensure mitigative features are in place to limit damage from fires to an acceptable level.

fire prevention: Measures directed toward avoiding the inception of fires.

fire protection: Methods of providing for fire control or fire extinguishment.

fire resistance rating: Time, in minutes or hours, that a material or assembly withstood a fire exposure as specified in NFPA Standard 251, "Standard Methods of Tests of Fire Endurance of Building Construction and Materials."

flammable liquid:[4] Liquid with a flash point below 37.8 °C (100 °F) and a vapor pressure not exceeding 40 psia at 37.8 °C (100 °F).

flammable gas:[4] A gas that will burn in the normal concentration of oxygen in the air.

gas:[4] Any substance that in a liquid state exerts a vapor pressure greater than 40 psia at 37.8 °C (100 °F).

limited-combustible: A building construction material that, in the form in which it is used, has a potential heat value not exceeding 8,141 KJ/kg (3,500 BTU/lb) and has either a structural base of noncombustible material with a surfacing not to exceed 3.2 mm (1/8 in) that has a flame spread rating not greater than 50, or other material having neither a flame spread rating greater than 25 nor evidence of continual progressive combustion, even on surfaces exposed by cutting through the material on any plane.

noncombustible: A material that, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors, when subjected to fire or heat. Materials passing ASTM–E–136, "Standard Test Method for Behavior of Materials in Vertical Tube Furnace at 750 °F," should be considered noncombustible.

pyrophoric material: A material with an auto ignition temperature in air at or below 54.4 °C (130 °F) and 50% relative humidity.

oxidizing gases: Gases that support combustion.

reactive gases: Gases that will either react with other materials or within themselves by a chemical reaction other than combustion under reasonably anticipated initiating conditions.

8.0 CHEMICAL SAFETY

8.1 PURPOSE OF REVIEW

The purpose of this review is to establish reasonable assurance that the applicant has designed a facility that provides for adequate protection against chemical hazards related to the storage, handling, and processing of licensed material as required by 10 CFR Part 70. This review also establishes that the applicant's facility and system design and facility layout pertaining to chemical safety is based upon defense-in-depth practices and, where practical, favors passive control systems over active ones.

Safety issues are initially evaluated as part of the applicant's integrated safety analysis (ISA); the ISA Summary identifies potential accidents with the types of consequences specified in 10 CFR 70.61 (Standard Review Plan (SRP) Chapter 5.0). Chemical safety addresses chemical hazards of licensed material and hazardous chemicals produced from licensed material. In addition, it also address plant conditions that may affect the safety of licensed material (e.g., an inert gas incapacitating or suffocating operators or precluding entry to an area of the facility handling licensed materials), and the controls used to prevent the occurrence or mitigate the consequences of accidents. The review should determine that the applicant's facility design and items relied on for safety (IROFS) provide reasonable assurance of chemical safety at the facility for routine operations, off-normal conditions, and potential accidents.

8.2 RESPONSIBILITY FOR REVIEW

Primary: Chemical Process Specialist

Secondary: Project Manager

Supporting: Project Manager as the primary reviewer of Organization and Administration, ISA Reviewer, Health Physicist Reviewer, Environmental Protection Reviewer, Primary Reviewers of Applicable Sections of SRP Chapter 15.0, and Inspection Staff (as needed)

8.3 AREAS OF REVIEW

The regulations of 10 CFR Part 70 requires applicants to establish a safety program to demonstrate compliance with the performance requirements. This does not necessarily require that the applicant establish a separate chemical safety program, but does require that chemical hazards and accident sequences that affect radiological materials be considered and adequately prevented or mitigated.

At NRC-licensed facilities, as stated in U.S. Nuclear Regulatory Commission, "Memorandum of Understanding between the Nuclear Regulatory Commission and the Occupational Safety and Health Administration: Worker Protection at NRC-Licensed Facilities," *Federal Register*, Vol. 53, No. 210, October 31,1998, pp.43950–43951, the NRC oversees chemical safety issues

related to (1) radiation risk produced by radioactive materials; (2) chemical risk produced by radioactive materials; and (3) plant conditions that affect the safety of radioactive materials and thus present an increased radiation risk to workers. The NRC does not oversee facility conditions that result in an occupational risk but do not affect the safe use of licensed radioactive materials.

The following areas should be reviewed:

A. Chemical Process Description - including process chemistry, process flow diagrams, mass/energy balances, inventories, major/significant process steps, safe operating limits for key parameters (e.g., temperature and pressure), and major pieces of equipment.

B. List of Hazardous Chemicals Affecting Licensed Materials - including potential interactions between chemicals and other materials as described in the ISA Summary.

C. Chemical Accident Sequences - including unmitigated analyses involving the hazardous chemicals and licensed materials, as described in the ISA Summary.

D. Chemical Accident Consequences - including assumptions, bases, and methods used to estimate the consequences of accidents for the workers, co-located workers, and the public identified in the ISA that involve hazardous chemicals and licensed materials.

E. Chemical Safety Controls - including the quantity and quality of controls used to mitigate or protect against accidents involving the release of hazardous chemicals and/or licensed materials, as determined by the ISA.

F. Chemical Process Safety Interfaces - including a description of how chemical safety interfaces with and is affected by other areas of review, including quality assurance, training, configuration management, maintenance, etc. Because the results of the ISA form the basis for much of the chemical safety of the design and facility, the primary reviewer should also review the ISA (see SRP Chapter 5.0). Supporting reviewers should confirm that provisions made in the application for chemical safety are in accordance and consistent with specified sections of the SRP. For example, the health physicist that is a primary reviewer from SRP Chapter 9.0, "Radiation Safety," as a supporting reviewer for chemical safety, should establish that the chemical safety program will not have unacceptably adverse impacts on the radiological safety at the facility.

Information contained in the application should be of sufficient quality and detail to allow for an independent review, assessment, and verification by the reviewers. Some information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear, specific, and essentially complete. Trade secrets or proprietary information will be treated in accordance with 10 CFR 2.790.

8.4 ACCEPTANCE CRITERIA

8.4.1 Regulatory Requirements

Requirements for protection against the occurrence of adverse chemical process consequences that could result from the handling, storage, or processing of licensed material and hazardous chemicals are found in 10 CFR Part 70. The following sections are particularly relevant to chemical safety: safety performance requirements (§ 70.61), safety program and ISA (§ 70.62), and the baseline design criteria for new facilities or new processes at existing facilities (§ 70.64, specifically § 70.64(a)(5), chemical protection; and § 70.64(b), defense-in-depth practices), and where applicable, passive systems and features.

8.4.2 Regulatory Guidance

The regulatory guidance applicable to chemical safety is contained in:

A. Nuclear Regulatory Commission (U.S.) (NRC). Manual Chapter 2603, "NRC Inspection Manual: Inspection of the Nuclear Process Chemical Safety Program at Fuel Cycle Facilities." NRC: Washington, D.C. 1996

B. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR–6410, "Nuclear Fuel Cycle Accident Analysis Handbook." NRC: Washington, D.C. March 1998.

C. Nuclear Regulatory Commission (U.S.) (NRC). NUREG–1513, "Integrated Safety Analysis Document." NRC: Washington, D.C. April 2000.

D. Nuclear Regulatory Commission (U.S.) (NRC). NUREG–1601, "Chemical Process Safety at Fuel Cycle Facilities." NRC: Washington, D.C. August 1997.

8.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find the applicant's chemical process safety information acceptable if there is reasonable assurance that the regulatory acceptance criteria are adequately addressed and satisfied. The applicant may elect to incorporate some or all of the requested chemical process information in the facility and process overview (discussed in SRP Section 1.1) and the ISA Summary (discussed in SRP Chapter 5.0) rather than in this section. Either approach is acceptable as long as the information is adequately cross-referenced.

8.4.3.1 Chemical Process Description

The chemical process description should be acceptable if it addresses the baseline design criteria for chemical safety and contains the following information:

A. Chemical Process Summary: In the chemical process summary, the applicant includes the purpose or objective of the major chemical process steps (e.g., valence adjustment and oxidation), including the operations to be performed, overall mass, energy,

radioactivity (Bq or curie), and waste balances (including emission, effluents, the disposition of wastes, and chemical/radionuclide concentrations).

B. Chemical Process Details: In the chemical process description, the applicant identifies the names and formulae of chemical reactants and products (input and output) to process steps, rates of reactions, and the operating conditions (e.g., temperature, pressure, flow rate, and pH), and identifies which chemicals contact licensed materials or could significantly impact operations with licensed materials. The chemical process description includes sufficient information (e.g., mass/energy/radioactivity balances, process flow diagrams, and descriptive equations) to enable the reviewers to understand the hazards associated with the chemical processes.

C. Process Chemistry: The description of the process chemistry provides stoichiometric equations for the primary/side reactions and degradation phenomena of the chemical moieties. Generation of flammable gases (e.g., hydrogen from reactions unique to mixed oxide processes such as the degradation of organic solvents in the presence of higher alpha radiation from plutonium and americium) should be included. The process chemistry discussion addresses initial startup conditions, normal operations, shutdown, and process testing and qualification.

D. Chemical Process Equipment, Piping, and Instrumentation: The description of the chemical process equipment, piping, and instrumentation includes descriptions, diagrams, layouts, schematics, and process logic for the major equipment, piping, and controls that may contain IROFS. The applicant identifies the codes and standards used to construct the process equipment (e.g., American Society of Mechanical Engineers (ASME) B.31.3 Process Piping Code). In addition, the applicant describes specific areas of hazards, such as large inventories in vessels or columns. The applicant also includes the results of its evaluation of the potential deleterious effects of processes (e.g., pH, radiation, and upset conditions) on equipment.

E. Chemical Process Inventories: The chemical inventory information provides the complete chemical and radionuclide inventories within the facility for routine and credible off-normal conditions.

F. Chemical Process Ranges: The description of the range of chemicals includes the approximate input, in-process, and output ranges of chemical and radioisotope concentrations, mass flow rates, and other properties (e.g., significant enthalpy changes during an acid/base reaction).

G. Chemical Process Limits: The identification and description of chemical process limits identify and discuss the limits in terms of parameters that may be considered as IROFS (such as chemical concentrations, temperature, pressure) and address the consequences of exceeding these limits. The process description identifies those limits that conservatively bound potential off-normal and accident conditions and that would be suitable for subsequent consequence analyses.

8.4.3.2 List of Hazardous Chemicals and Potential Interactions

The list of hazardous chemicals and potential interactions should be acceptable if they contain the following information:

A. Chemicals: The list of hazardous chemicals includes the major chemicals used in the process. The list includes chemical form, concentration, maximum projected inventory and location, associated exposure limits (e.g., Occupational Safety and Health Administration (OSHA) Permissible Exposure Limit, Emergency Response Planning Guidelines, etc.), and safety precautions.

B. Chemical Interactions: The list of chemical interactions includes potential reactions and interactions between materials stored and used at the facility that have the potential to affect the safe handling of licensed radioactive materials, as determined by the ISA. The list includes a chemical interaction matrix (see NUREG–1513), or equivalent, for determining chemical incompatibilities and potentially unsafe interactions. The matrix summarizes the effects of intense radiolysis as a potential initiator of chemical reactions and interactions. The list uses standard groupings of chemicals (e.g., acids, bases, oxidizers, organics) and includes potential chemical/radiolytic interactions between chemicals and items not generally considered as reagents, such as ion exchange resins, sorbents, lead-lined gloves, glovebox covers, and sealing materials (e.g., mechanical pump seals and gaskets). The list includes potential deleterious effects of the degradation products of solvent/organic compounds (e.g., di-butyl phosphate generated by the degradation of tri-butyl phosphate) on licensed material. Additionally, the list includes possible adverse impacts to the pyrophoric licensed material resulting from the loss of the inert atmosphere, as appropriate.

C. Unusual and Unexpected: The list of hazardous chemicals and potential interactions addresses unusual and unexpected chemical interactions from the different facility conditions that may affect the safety of licensed materials, including those that impact controllability and habitability issues such as emission of inert gas, CO_2, or NOx. The applicant has addressed the potential accumulation of flammable/combustible gases in tank ullage spaces and vent lines, as appropriate.

8.4.3.3 Chemical Accident Sequences

The chemical accident sequences are acceptable if they contain the following information:

A. Chemical Accident Sequence Bases: The bases and references used in the chemical accident sequences are supported by applicable data and references. The applicant includes estimated annual frequencies and probabilities over the facilities' operational period. The accident sequences include the chemical hazard evaluation, which identifies the potential interactions between process chemicals, licensed materials, process conditions, facility personnel/operators, and structures, systems, and components.

B. Unmitigated Sequences: The applicant clearly delineates these chemical accident sequences as unmitigated for the purposes of analysis and item categorization.

C. Estimated Concentrations: The estimates of hazardous chemical concentrations include techniques, assumptions, and models that are consistent with industry practice, are verified and/or validated, and follow the guidance on atmospheric and consequence modeling found in NUREG/CR–6410, "Nuclear Fuel Cycle Accident Analysis Handbook." The applicant provides evidence that the techniques, assumptions, and models used are appropriate for the application and that they lead to a conservative estimate of potential consequences.

D. Concentration Limits: The chemical concentration limits have a supporting rationale or basis such as Acute Exposure Guideline Level or Emergency Response Planning Guide values or other cited values, such as those values developed by OSHA or the National Institute for Occupational Safety and Health. If the applicant does not use a published standard, or if a chemical has an unknown exposure standard, the applicant may propose an alternate standard accompanied by supporting documentation to justify the selection of such an alternative. The performance requirements of 10 CFR 70.61 are based upon acute chemical exposures; therefore, chemical concentration values such as OSHA permissible exposure limits or other time-weighted average values should not be used unless a rational basis is provided in the ISA.

8.4.3.4 Chemical Accident Consequences

The primary reviewer should coordinate the chemical accident consequence reviews with the primary reviewers of the ISA Summary (discussed in SRP Chapter 5.0) and Environmental Protection (SRP Chapter 10.0) chapters and meet the requirements for 10 CFR 70.61 and 70.62. The chemical accident consequences should be acceptable if they contain the following information:

A. Analysis: The accident consequence analysis is encompassed by the ISA, which identifies potential accident sequences with hazardous chemicals and licensed materials, and the consequences are estimated for both workers and members of the public. Dispersion models may be necessary for estimating the concentration and potential impacts of such chemicals at various distances from the point of release. In this case, the applicant provides information to support the conclusion that the models used are appropriate for the application and physical phenomena occurring, that the models have been validated and verified, and that the assumed data input leads to a conservative estimate of potential consequences. Consequence analysis follows the guidance found in NUREG/CR–6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook."

B. Latent Impacts: The applicant's accident consequence analysis considers if there are any residual, long-term impacts to worker and public health that could result from an acute chemical exposure to licensed material or hazardous chemicals produced from licensed material (i.e., as compared to the analysis in Item A, which focuses primarily on the prompt effects).

C. Uncertainty: The accident consequence analysis includes consideration of uncertainty and errors in comparing chemical hazards and radioactive material effects with the performance requirements of 10 CFR 70.61.

8.4.3.5 Process Safety Information

In addition to the information provided in the ISA Summary (see Chapter 5.0, Section 5.4.3.2(A)(iv), Items (a) through (c)), the chemical safety reviewer should use the following items as a checklist when reviewing the process safety information:

A. The applicant's identification of chemical process safety controls used to prevent or mitigate potential accidents are supported by appropriate safety analyses, and the applicant provides reasonable assurance that these safety controls will be available and reliable upon demand.

B. The application identifies the design basis that provides safety for normal operations. A description could include specified features such as materials of construction, sizing, system fabrication, and process control schemes.

C. The process safety control discussion includes a description of the process and engineering design features used to control each process step, including set point ranges and any special administrative or procedural controls. The discussion describes the process safety features that are relied upon for chemical process safety, including the number and quality of controls used to protect against (reducing frequency and probability of occurrence) or mitigate (reducing consequences) accidents involving the release of hazardous chemicals that are produced from the licensed material or that may impact the safety of licensed material, as determined by the ISA.

D. IROFS are identified for those accident sequences containing a chemical/process failure that may lead to radiological consequences that exceed the performance requirements of the 10 CFR 70.61.

E. The applicant uses a graded approach to safety in accordance with 10 CFR 70.62(a). The applicant ensures that the grading of IROFS is appropriate and sufficient to protect against chemical/process risk, including a consideration of relying upon passive over active systems, defense-in-depth, and fail-safe features. For common-mode failures, the applicant considers design features in the application that utilize independent sources of motive force and power for items such as actuators, pumps, and eductors.

F. The application describes the management measures that assure the availability and reliability of IROFS for chemical and process safety. Management measures may be graded commensurate with risk.

8.4.3.6 Chemical Process Safety Interfaces

The description of chemical process safety interfaces should be acceptable if the application addresses how the following areas of review interface with aspects of chemical safety at the facility (see the appropriate SRP sections and chapters as specified in parentheses):

A. Organizational Structure (SRP Chapter 4.0)
B. Human Factors (SRP Chapter 12.0)
C. Emergency Management (SRP Chapter 14.0)
D. Quality Assurance (SRP Section 15.1)
E. Configuration Management (SRP Section 15.2)
F. Maintenance (SRP Section 15.3)
G. Training and Qualification (SRP Section 15.4)
H. Plant Procedures (SRP Section 15.5)
I. Audits and Assessments (SRP Section 15.6)
J. Incident Investigations (SRP Section 15.7)
K. Records Management (SRP Section 15.8)

8.5 REVIEW PROCEDURES

8.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 8.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use special nuclear material (SNM).

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

Specifically, the safety assessment of the design basis should address Section 8.3(A)–(E) consistent with the level of design. Where information is under development or not yet available, the applicant may use a commitment to provide the material with the application for a license to possess and use SNM in lieu of the actual material.

B. License To Possess and Use SNM

Specifically, the safety assessment included as part of the license application should address Section 8.3(A)–(F) in full.

If the primary reviewer verifies that chemical safety is adequately addressed (construction approval or license to possess and use SNM), the primary reviewer should accept the application for the safety evaluation in Section 8.5.2. If the primary reviewer identifies

significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

8.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 8.5.1(A) (construction approval) or 8.5.1(B) (license to possess and use SNM), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 8.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

The primary reviewer should establish that the applicant's facility design, as described in the safety assessment of the design bases and other commitments as they relate to chemical safety, meets or exceeds the regulatory acceptance criteria in Section 8.4.

B. License To Possess and Use SNM

The primary reviewer should establish that the applicant's facility design, operations, and chemical safety items provide reasonable assurance that they will function as intended and provide for the safe handling of licensed materials at the facility.

When the safety evaluation is complete (either for construction approval or for the license to possess and use SNM), the primary reviewer, with assistance from the other reviewers, should prepare the chemical safety input for the Safety Evaluation Report (SER), as described in Section 8.6 using the acceptance criteria from Section 8.4. The secondary reviewer should coordinate the chemical safety input with the balance of the reviews and the SER.

8.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

The staff reviewed the license application for construction approval for [insert name of facility] according to Chapter 8.0 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found that [summarize the findings]. Based on the review of the application, the NRC staff concluded that the applicant adequately described and assessed accident consequences having potentially significant chemical consequences

and effects that could result from the handling, storage, or processing of licensed materials. The applicant's design bases and safety assessment of the design bases identified and evaluated those chemical process hazards and potential accidents. The staff reviewed these safety controls and finds them acceptable.

The staff concluded that the applicant's design bases for managing chemical process safety and the chemical process safety controls meet the requirements in the area of chemical safety to approve construction of the facility under 10 CFR Part 70.

The staff could document the safety evaluation for the license application as follows:

The staff reviewed the license application for [insert facility name] to possess and use SNM according to Chapter 8 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found [insert a description of the findings]. Based on the review of the license application, the staff concluded that the applicant adequately described and assessed accident consequences having potentially significant chemical consequences and effects that could result from the handling, storage, or processing of licensed materials. The ISA Summary identified those chemical process hazards and potential accidents, and established safety controls to ensure safe facility operation. To ensure that the performance requirements in 10 CFR Part 70 are met, the applicant will ensure that controls are maintained available and reliable. The staff reviewed these safety controls and the applicant's plan for managing chemical process safety and its potential effects upon licensed radioactive materials and finds them acceptable.

The staff concludes that the applicant's plan for managing chemical process safety and the chemical process safety controls meet the requirements to possess and use SNM according to the 10 CFR Part 70.

8.7 REFERENCES

Center for Chemical Process Safety (CCPS). Chapter 11, "Guidelines for the Technical Management of Chemical Process Safety." CCPS of the American Institute of Chemical Engineers: New York, New York. 1989.

Chemical Manufacturers Association (CMA). "Responsible Care®, Process Safety Code of Management Practices." CMA: Washington, D.C. 1990.

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

———. *Title 29, Labor,* Part 1910.119, "Process Safety Management of Highly Hazardous Chemicals."

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Memorandum of Understanding between the Nuclear Regulatory Commission and the Occupational Safety and Health

Administration: Worker Protection at NRC-Licensed Facilities," *Federal Register:* Vol. 53, No. 210, pp. 43950–43951. October 31, 1988.

———. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)," *Federal Register:* Vol. 64, No. 146. pp. 41338–41357. July 30, 1999.

Nuclear Regulatory Commission (U.S.) (NRC). Manual Chapter 2603, "Inspection Manual: Inspection of the Nuclear Process Chemical Safety Program at Fuel Cycle Facilities." NRC: Washington, D.C. 1996.

———. NUREG/CR–6410, "Nuclear Fuel Cycle Accident Analysis Handbook." NRC: Washington, D. C. March 1998.

———. NUREG–1513, "Integrated Safety Analysis Document." NRC: Washington, D.C. April 2000.

———. NUREG–1601, "Chemical Process Safety at Fuel Cycle Facilities." NRC: Washington, D.C. August 1997.

9.0 RADIATION SAFETY
9.1 RADIATION SAFETY DESIGN FEATURES

9.1.1 PURPOSE OF REVIEW

The purpose of this review is to determine with reasonable assurance that the applicant's design for construction and operation of the facility is adequate to protect the radiological health and safety of workers and to comply with the regulatory requirements of 10 CFR Parts 20 and 70 during routine and nonroutine operations, including anticipated events. This section also facilitates the review of the radiation safety aspects of accident sequences described in the Integrated Safety Analysis (ISA) Summary, through an interface with Standard Review Plan (SRP) Chapter 5.0.

The protection of members of the public and the control of effluent releases is not included in this section, but is covered in SRP Chapter 10.0, "Environmental Protection." While this chapter addresses the review of the applicant's radiation safety design as applied to construction and operation of the facility, the applicant's radiation protection program and management measures are reviewed under SRP Section 9.2, "Radiation Protection Program."

9.1.2 RESPONSIBILITY FOR REVIEW

Primary: Health Physicist

Secondary: Project Manager, Environmental Reviewer, ISA Reviewer, Fire Protection Engineer, Emergency Protection Specialist, and the Primary Reviewer of SRP Section 9.2 (if different from the Primary Reviewer of Section 9.1)

Supporting: None

9.1.3 AREAS OF REVIEW

As established in 10 CFR 20.1101, the applicant is required to use, to the extent practical, engineered controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA; see Items A through E). The applicant is also required to establish controls and management measures to meet the performance requirements established in 10 CFR 70.61 (see Item F). Areas of review include:

A. ALARA Design Considerations

i. Organizational relationships and responsibilities with respect to performing radiological design reviews;

ii. Application of ALARA into design-stage collective dose estimates;

iii. Descriptions and elements of the design review process for radiation protection; and

iv. How the applicant used experience from past designs and from operating plants to develop improved radiation protection design, when ALARA threshold values are exceeded.

B. Facility Design Features

i. Proposed equipment and facility design features and facility layout as they relate to occupational radiation protection and ALARA concepts;

ii. Design features incorporated to minimize contamination and waste production, and facilitate ease of operations, maintenance, replacement, and decommissioning consistent with maintaining doses at levels that are ALARA;

iii. Facility design goals as they relate to radiation safety; and

iv. A self-assessment of the individual and collective doses via a summary figure or table of predicted annual occupational doses for the types of work functions (e.g., operations, routine maintenance, special maintenance, inservice testing and surveillance, and waste management) provided at the facility.

C. Source Identification

i. The sources of radiation and contamination in the facility during routine and nonroutine operations (e.g., maintenance), including anticipated events.

ii. The sources of radiation that are used to evaluate consequences in the ISA Summary.

iii. Source identification describes the pertinent information needed for:

a. Input to shielding codes used in the design process (Item E);
b. Establishing related facility design features (Items A and B);
c. Plans and procedures development; and
d. Assessment of occupational dose (Item C).

iv. The methods for estimating source magnitudes and locations at the design stage and how this information is incorporated into the design.

D. Ventilation Systems and Glovebox Design

i. The design and operation of the ventilation systems and gloveboxes as described in support of Chapter 11.0, "Plant Systems," as related to radiological safety, including the:

a. Proposed design objectives;

b. Design and operation; and
c. Monitoring and alarms.

E. Shielding Evaluations

i. Shielding information for each of the radiation sources identified in Item C;

ii. The criteria for penetrations;

iii. Shielding materials;

iv. The methods (e.g., codes) by which the shield parameters (e.g., attenuation coefficients, buildup factors) were determined; and

v. Special protective features that use shielding, geometric arrangement, or remote handling to ensure that occupational radiation exposures will be ALARA in normally occupied areas.

F. ISA Summary

i. Postulated types of accident sequences in the ISA Summary that have radiation safety consequences for workers, including all high-risk and a sample of lower risk accident sequences that result in radiation doses of concern and accidents that result from operations and natural phenomena.

ii. The training program and postings for the individuals as required under 10 CFR 70.61(f)(2) (training and postings may be cross-referenced with Section 9.2) if the applicant's proposed controlled area (as identified under Item B) includes individuals who are not workers, as defined in 10 CFR 70.4.

iii. The methodology in assessing the accident consequences. In particular, the primary reviewers of this SRP section should focus on the source terms (see Item C), transport, and dosimetry analyses.

iv. The items relied on for safety (IROFS), and associated management measures, to prevent or mitigate each accident sequence that results in radiological consequences in excess of the performance requirements of 10 CFR 70.61.

9.1.4 ACCEPTANCE CRITERIA

Each subject area lists the applicable regulatory requirements and the NRC Regulatory Guides (RGs), NUREG reports, Branch Technical Positions (BTPs), and industry standards that provide a basis that is generally acceptable to the NRC staff for satisfying the applicable regulatory requirements. However, in some cases the use of industry standards has not been endorsed by the NRC through a regulation or RG. Further, inclusion in this section is not

necessarily an endorsement of a particular standard by the NRC. Therefore, their use is encouraged, but alternative, equivalent methods may be proposed in the application with adequate justification.

9.1.4.1 ALARA Design Considerations

9.1.4.1.1 Regulatory Requirements

10 CFR 20.1101(b)	"Radiation Protection Programs"
10 CFR 20.1406	"Minimization of Contamination"
10 CFR 20.1501(a)	"Surveys—General"
10 CFR 70.22(a)(4) and (7)	"Contents of Applications"
10 CFR 70.64	"Requirements for New Facilities or New Process at Existing Facilities"

9.1.4.1.2 Regulatory Guidance

The regulatory guidance applicable to ALARA design considerations is contained in:

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable." NRC: Washington, D.C. May 1977.

9.1.4.1.3 Regulatory Acceptance Criteria

The requirements related to ALARA design considerations are specified in Section 9.1.4.1.1. The applicant's ALARA design considerations should meet the regulatory requirements if the following regulatory acceptance criteria, or information describing acceptable alternatives, are met:

A. The applicant defines organizational functions that have the responsibility for performing radiological design and design reviews.

B. The applicant's design and design activities, with respect to radiation protection, incorporate provisions that provide reasonable assurance that the design will:

   i. Reduce the need for time spent in radiation areas;

   ii. Improve the accessibility to components requiring periodic maintenance or inservice inspection;

   iii. Reduce the distribution and retention of radioactive materials throughout plant systems;

iv. Control (reduce) contamination, facilitate decommissioning, and minimize secondary radioactive waste production in accordance with 10 CFR 20.1406;

v. Instruct designers and engineers in ALARA design objectives;

vi. Incorporate experience from operating plants and past designs; and

vii. Commit to, and describe, continuing radiation safety (ALARA) design reviews for facility or process modifications made during construction and operations.

C. The radiation protection (ALARA) design review process includes:

i. Design reviews and dose assessments performed by competent personnel including (or with the concurrence of) radiation safety staff and radiation safety management;

ii. Design reviews that include the review of previous jobs, designs, operating experience, and processes for applicability and improvements;

iii. Design reviews that include documentation (e.g., ALARA Design Review Checklists) and tracking of recommendations to completion; and

iv. Design reviews that are graded based on the hazard (e.g., are compared to defined ALARA trigger levels).

D. The applicant's process for seeking radiation protection-related design improvements includes a description of how those design improvements are sought, considered, and incorporated where practical (RG 8.10, C.1(f)).

9.1.4.2 Facility Design Features

9.1.4.2.1 Regulatory Requirements

10 CFR 20.1101(b)	"Radiation Protection Programs"
10 CFR 20.1201	"Occupational Dose Limits For Adults"
10 CFR 20.1301	"Dose Limits for Individual Members of the Public"
10 CFR 20.1406	"Minimization of Contamination"
10 CFR Part 20 Subpart H	"Control of Exposure from External Sources in Restricted Areas"
10 CFR 20.1701	"Use of Process or Other Engineering Controls"

10 CFR 70.22(a)(4) and (7)	"Contents of Applications"
10 CFR 70.23(a)(3)	"Requirements for Approval of Applications"
10 CFR 70.61(f)	"Performance Requirements for New Facilities"
10 CFR 70.64(b)	"Requirements for New Facilities or New Process at Existing Facilities"

9.1.4.2.2 Regulatory Guidance

Regulatory guidance facility design features is contained in:

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 3.29, "Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants." NRC: Washington, D.C. May 1975.

9.1.4.2.3 Regulatory Acceptance Criteria

Requirements related to facility design features are specified in Section 9.1.4.2.1. The applicant's facility design features should meet the regulatory requirements if the following regulatory acceptance criteria, or information describing acceptable alternatives, are met:

A. The facility and process drawings and descriptions identify clearly readable and scaled radiation safety design features that are:

   i. Relied on to reduce doses to meet 10 CFR Part 20 during routine and nonroutine operations (including anticipated events); and

   ii. IROFS to reduce accident doses.

B. The identification of the features in Item A include:

   i. Locations of detectors and alarm systems;

   ii. Locations of permanent shielding (including penetrations, labyrinths, shield doors, etc.);

   iii. Provisions for installation/removal of temporary shielding;

   iv. Locations and access control points for restricted areas;

   v. The controlled area, including the applicant's means to limit access to the controlled area for any reason;

   vi. The restricted area;

vii. Change rooms, showers, and locker rooms; and

viii. The contamination control, decommissioning facilitation, and waste minimization design features required by 10 CFR 20.1406. (The reviewer should also refer to SRP Chapter 10.0, "Environmental Protection.")

C. The applicant's self-assessment of the submitted facility design, shielding, layout, traffic patterns, expected maintenance, and sources shows that both collective and individual doses from significant activities are within the limits of 10 CFR Part 20, ALARA, and meet facility design goals for routine and nonroutine operations, including anticipated events. For purposes of design stage estimates, significant activities could be defined as dose-causing activities conservatively estimated to result in greater than 0.01 person-sievert (1.0 person-rem) per year.

D. Worker access controls for high and very high radiation areas meet 10 CFR 20.1601 and 20.1602, respectively. For general radiation areas, change rooms are provided for changing into personnel protective equipment (PPE). Change rooms are adjacent to shower and decontamination facilities and are provided with ventilation systems that filter dispersible radionuclides. Administrative (i.e., programmatic) aspects of access control and storage are reviewed under SRP Section 9.2.4.6, "Contamination Control."

9.1.4.3 Source Identification

9.1.4.3.1 Regulatory Requirements

10 CFR 70.22(a)(4) and (7)	"Contents of Applications"
10 CFR 70.64	"Requirements for New Facilities or New Process at Existing Facilities"

9.1.4.3.2 Regulatory Guidance

The regulatory guidance applicable to source identification is contained in:

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable." NRC: Washington, D.C. May 1977.

9.1.4.3.3 Regulatory Acceptance Criteria

Requirements related to source identification are specified in Section 9.1.4.3.1. The applicant's source identification should meet the regulatory requirements if the following regulatory acceptance criteria, or information describing acceptable alternatives, are met:

A. Internal and External Dose Considerations

The applicant provides quantitative descriptions and estimates of contained sources (RG 8.10, C.2(a)) and uses the quantitative descriptions as the basis for the radiation protection program, the internal radiation protection program, the ventilation system design, and the shield design calculations with consideration of routine and nonroutine operations, including anticipated events and accident conditions. The quantitative descriptions include:

i. Tabulations of the calculated concentrations of radioactive material, by isotopic composition, expected during routine and nonroutine operations (including anticipated events and accident conditions) for equipment cubicles, corridors, and operating areas normally occupied by operating personnel; and

ii. The models and parameters (e.g., source strength or geometry) for the calculations and the basis for the values used.

B. The contained and airborne radioactivity sources estimated at the design stage are based on an assumption of several years of facility operation. The applicant identifies specific assumptions, discusses uncertainties, and justifies the conservatism of each assumption.

9.1.4.4 Ventilation Systems and Glovebox Design

9.1.4.4.1 Regulatory Requirements

10 CFR 20.1101(b)	"Radiation Protection Programs"
10 CFR 20.1201	"Occupational Dose Limits for Adults"
10 CFR 20.1301	"Dose Limits for Individual Members of the Public"
10 CFR 20.1501(a)	"Surveys—General"
10 CFR 20.1701	"Use of Process or Other Engineering Controls"
10 CFR 70.22(a)(4) and (7)	"Contents of Applications"
10 CFR 70.64	"Requirements for New Facilities or New Process at Existing Facilities"

9.1.4.4.2 Regulatory Guidance

The regulatory guidance applicable to ventilation systems and glovebox design is contained in:

American Nuclear Standards Institute/American Society of Mechanical Engineers (ANSI/ASME). ANSI/ASME–N510–1980 (1989), "Testing of Nuclear Air Cleaning Systems."

Energy Research and Development Administration (ERDA). ERDA 76–21, "Nuclear Air Cleaning Handbook."

9.1.4.4.3 Regulatory Acceptance Criteria

A ventilation system is necessary to provide confinement integrity and to process off-gas before being exhausted to the environment. The review performed in this SRP section concerns those functions of the ventilation and air cleaning system that pertain to occupational radiation protection (specifically, controlling internal dose through limiting airborne radioactivity). Ventilation systems will have many functions other than controlling internal radiation exposure to workers through containment (e.g., off-gas management, heating and air conditioning, accident functions, controlling chemical exposures, reducing effluent releases, etc.). Explicit acceptance criteria related to the ventilation design, testing, redundancy, capacity and capability, monitoring, environmental qualifications, natural phenomena, fire protection, air supply, removal and replacement of filters, and gloveboxes can be found in Section 11.4.5.

Requirements related to radiation safety for ventilation and glovebox design are specified in Section 9.1.4.4.1. The applicant's ventilation and glovebox design, as related to radiation safety, should meet the regulatory requirements if the following regulatory acceptance criteria, or information describing acceptable alternatives, are met:

A. The design and operation of the ventilation system and/or gloveboxes protect workers and the public from airborne radioactive material such that limits of 10 CFR Part 20 will not be exceeded during routine and nonroutine operations and anticipated events. Recommendations for the design, construction, and testing of nuclear air cleaning systems (e.g., zoning, moisture separation, high efficiency particulate air filtration, operational/maintenance considerations, etc.) that are generally acceptable to NRC staff are provided in ERDA 76–21 (see also Section 11.4.5).

B. The applicant commits to designing objectives for ventilation systems and gloveboxes that ensure that:

   i. During routine and nonroutine operations and anticipated occurrences, airborne concentrations in occupied operating areas are well below the limits of 10 CFR Part 20, Appendix B; and

   ii. The use of engineering (i.e., design) controls shall be preferred over the use of respirators (10 CFR 20.1701).

C. Air monitoring and warning systems associated with the ventilation system and gloveboxes, which are required to function during a loss of power (in addition to performing their specified functions), are provided with an uninterruptible power supply, unless they can tolerate a temporary loss of function without loss of data, and are provided with a standby power supply. In addition to local alarms, the applicant provides readouts for air monitoring and alarm systems that are accessible during accidents. Certain programmatic aspects of air monitoring and warning systems are reviewed under SRP Section 9.2, "Radiation Protection Program."

9.1.4.5 Shielding

9.1.4.5.1 Regulatory Requirements

10 CFR 20.1101(b)	"Radiation Protection Programs"
10 CFR 20.1201	"Occupational Dose Limits for Adults"
10 CFR 20.1301	"Dose Limits for Individual Members of the Public"
10 CFR 20.1501(a)	"Surveys—General"
10 CFR 20.1701	"Use of Process or Other Engineering Controls"
10 CFR 70.22(a)(4) and (7)	"Contents of Applications"
10 CFR 70.64	"Requirements for New Facilities or New Process at Existing Facilities"

9.1.4.5.2 Regulatory Guidance

American National Standards Institute/ American Nuclear Society (ANSI/ANS). ANSI/ANS–6.4.2–1985, "Specification of Radiation Shielding Materials."

9.1.4.5.3 Regulatory Acceptance Criteria

Requirements related to shielding are specified in Section 9.1.4.5.1. The applicant's shielding design should meet the regulatory requirements if the following regulatory acceptance criteria, or information describing acceptable alternatives, are met:

A. The applicant's facility descriptions (e.g., facility layout diagrams submitted for SRP Section 1.1 or Chapter 5.0) detail the use of and locations where the applicant included permanent shielding into the design to lower dose rates to comply with 10 CFR Part 20 during routine and nonroutine operations and anticipated events. The applicant identifies and describes any areas that facilitate installation and removal of temporary shields for nonroutine operations. (Where the applicant identifies the use of temporary shielding, local audible and visible alarming radiation monitors are installed to alert personnel if shielding is not present, consistent with the external radiation hazard.) The use of permanent shielding is consistent with the external sources identified under Section 9.1.4.2.3(A).

B. Shielding design to minimize external and internal doses meets design goals and is described in sufficient detail to verify results.

C. The applicant derives permanent or temporary shielding requirements and specifications based on identified design objectives. The applicant's specified dose or dose-rate design objectives are based on fractions of 10 CFR Part 20 limits and personnel occupancy

predictions, for both continually and intermittently occupied areas of the facility. Occupancy accounts for duration and frequency of exposures and for the fact that doses in particular areas may either be occupational (radiation worker) or nonoccupational (general employee). An objective, for design purposes, of 20% of the applicable annual limits in 10 CFR Part 20 (e.g., 1.0 rem/yr for restricted areas), accounting for occupancy estimates, is acceptable to the staff. For continuously occupied areas, this translates to an average dose rate of 0.5 mrem/hr (20% of the occupational dose limit of 5 rem in a 2,000 hour work-year). (These objectives are comparable to the design limits of 10 CFR 835.1002.) Notwithstanding this design objective, management measures would need to supplement the design objective to further reduce doses consistent with ALARA. Another acceptable design objective is that the use of straight-line penetrations of shield walls should be minimized.

D. For each instance the applicant provides shielding associated with reducing doses from high or very high radiation areas, the shielding used and features such as penetrations, shield doors, and labyrinths meet design goals and are described in sufficient detail to verify results. The applicant demonstrates adequate attenuation through:

   i. Analyses (calculations); or

   ii. Reference to similar configurations that were previously analyzed or experimentally verified.

E. The applicant commits to and describes a radiation shielding test program that will verify the efficacy of installed shielding materials in meeting the radiation shielding design goals and the regulatory external dose requirements of 10 CFR Part 20. The applicant's objective for this commitment is to verify that the applicant provided sufficient shielding (particularly with regard to penetrations, labyrinths, shield doors, etc.) for the life of the facility, prior to initiation of operations; and to verify that design models and calculations are representative of actual operating conditions with respect to occupational radiation protection.

F. Shielding and features such as penetrations provided and/or installed to minimize nonpenetrating external radiation doses, including that to the skin, extremities, and lens of the eye, meet design goals and are described in sufficient detail to verify results.

G. Where used, the applicant's analyses for calculating shielding requirements are comparable to commonly acceptable shielding calculations and use realistic assumptions regarding source terms, cross-sections, shield and source geometries, and transport methods. The applicant uses codes that rely on the use of flux-to-dose conversion factors of ANSI/ANS–6.1.1–1991 and cross-sections of ANSI/ANS–6.1.2–1999 (recommends ENDF/B library). Generally, only Monte Carlo calculation methods would be acceptable to NRC staff for analyses of complex geometries (e.g., shield penetrations). The applicant's analyses descriptions are acceptable if provided in sufficient detail to allow independent confirmatory calculations.

H. The applicant considers facilitating waste minimization in accordance with § 20.1406 in its selection of shielding materials and of permanent versus temporary shielding, as one

design consideration. The applicant's descriptions of the physical and nuclear properties of shielding materials used for various functions in the facility are consistent with ANSI/ANS–6.4.2–1985.

I. In cases where the confinement barrier or process equipment provides the primary shielding and is relied on for safety as determined by the ISA, the quality assurance program is applied to all aspects of the shielding design, procurement, installation, maintenance, etc. For shielding that is relied on for safety, the design and analyses approaches used by the applicant should be described; for concrete, the methods in ANSI/ANS–6.4–1985 should be acceptable.

9.1.4.6 Integrated Safety Analyses (ISA) Summary

9.1.4.6.1 Regulatory Requirements

10 CFR 70.61	"Performance Requirements"
10 CFR 70.62	"Safety Program and Integrated Safety Analysis"
10 CFR 70.64	"Requirements for New Facilities or New Process at Existing Facilities"

9.1.4.6.2 Regulatory Guidance

Nuclear Regulatory Commission (U.S.) (NRC). NUREG–1513 (DRAFT April 2000), "Integrated Safety Analysis Guidance Document." NRC: Washington, D.C.

9.1.4.6.3 Regulatory Acceptance Criteria

The requirements related to the ISA Summary are specified in Section 9.1.4.6.1. The applicant's ISA Summary as it applies to design for radiation protection should meet the regulatory requirements if the following regulatory acceptance criteria, or information describing acceptable alternatives, are met:

A. The applicant uses appropriate and verified assessment methods, computer codes, and literature values.

B. The applicant considers a complete range of credible accident sequences that could adversely affect radiation protection and exceed the performance requirements described in 10 CFR 70.61.

C. The applicant makes reasonable estimates of the radiological consequences to workers (considering source term, transport, and dosimetry) of accident sequences. (Note that radiological consequences to the public and chemical consequences or hazardous chemicals resulting from licensed material to workers and the public are evaluated in Chapters 10.0 and 8.0, respectively.)

D. The applicant identifies effective controls and management measures to prevent and mitigate accident sequences and radiological consequences for workers that exceed the performance requirements of 10 CFR 70.61.

E. If the applicant's controlled area could be occupied by individuals who are not workers, as defined in 10 CFR 70.4, the applicant provides training and postings in accordance with 10 CFR 19.12(a)(1)–(5) and 10 CFR 19.11(a), respectively.

F. The applicant describes and commits to appropriate management measures to provide reasonable assurance of the continued availability and reliability of safety controls to prevent and mitigate radiological consequences for workers.

9.1.5 REVIEW PROCEDURES

9.1.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 9.1.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use SNM.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below. "

A. Construction Approval

Specifically, the safety assessment of the design basis should address Section 9.1.3(A)–(E) consistent with the level of design. Where information is under development or not yet available, the applicant may use a commitment to provide the material with the license application in lieu of the actual material.

B. License to Possess and Use SNM

Specifically, the safety assessment of the license application should update the material provided in the application for construction approval and address Section 9.1.3(A)–(F) in full.

If the primary reviewer verifies that radiation safety design features are adequately addressed for the construction approval review or the review for the license to possess and use SNM, the primary reviewer should accept the application for the safety evaluation in Section 9.1.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

9.1.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 9.1.5.1(A) (construction approval) or 9.1.5.1(B) (license to possess and use SNM), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 9.1.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

The primary reviewer should establish that the applicant's facility design as described in the safety assessment of the design basis and other commitments, as they relate to radiation safety, meet or exceed the regulatory acceptance criteria in Section 9.1.4.

The primary reviewer should coordinate the radiation safety design aspects of the ventilation, gloveboxes, and air cleaning systems of this SRP section with the primary reviewers of SRP Chapter 7.0, "Fire Protection," and SRP Chapter 11.0, "Plant Systems," to ensure that the application for construction approval contains adequate and consistent information and that conflicts do not exist between the various technical areas.

B. License To Possess and Use SNM

While this section addresses the applicant's radiation safety *design*, the applicant's radiation protection program and management measures are reviewed under SRP Section 9.2, "Radiation Protection Program," with the license application. Certain aspects of radiation safety, such as facility access controls, zoning, and security of stored material, cannot be cleanly categorized into either "design" or the "radiation protection program." Review of these areas should be coordinated with the reviewer of Section 9.2. The review should confirm that appropriate aspects of the radiation design, updated from the construction approval stage, are fed appropriately into the radiation protection program. Other considerations include:

i. The information in Section 9.1.4.2, regarding the facility and process design drawings and descriptions, could be included by a reference to SRP Chapter 1.1, "Facilities and Process Overview," or SRP Chapter 5.0, "Integrated Safety Analysis," which requires additional process description information through 10 CFR Part 70 Subpart H. The primary reviewer should perform the safety evaluation of this information as it pertains to radiation protection design, regardless of where it appears in the application. Particularly, the primary reviewer should confirm with the emergency protection specialist and the physical protection specialist that the applicant is able to limit access to the controlled area.

ii. The primary reviewer should coordinate the updated radiation safety design aspects of the ventilation, gloveboxes, and air cleaning systems of this SRP section with the primary reviewer of SRP Chapter 7.0, "Fire Protection," to ensure that the fire protection-related aspects of those systems are not in conflict with radiation protection, and with the primary reviewer of SRP Chapter 11.0, "Plant Systems," for the nonradiation protection-related aspects of the ventilation and air cleaning systems, to verify that the license application contains adequate and consistent information.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the radiation safety design input for the Safety Evaluation Report (SER), as described in Section 9.1.6 using the acceptance criteria from Section 9.1.4. The secondary reviewer should coordinate the radiation safety design input with the balance of the reviews and the SER.

9.1.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

The staff reviewed the license application construction approval for [insert facility name] according to Section 9.1 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found [insert a summary statement of the findings]. The applicant estimated the facility radiation sources capable of producing significant radiation levels and significant airborne radioactivity, based on [include the applicant's basis for radiation and airborne source terms]. These estimates demonstrate a conservative approach for the current level of design and are acceptable.

The applicant described organizational relationships and responsibilities with respect to performing radiological design reviews, which ensure the adequate application of ALARA in design-stage activities, including facility modifications made during construction.

The general shielding design and analysis methodology used by the applicant is acceptable. The applicant has provided an adequate treatment of features requiring special analyses, such as cell penetrations, and has shown by calculation that doses in work areas meet requirements. The basic radiation transport analysis used for the applicant's shield design is based on [list appropriate shielding computer codes used].

The ventilation system at [facility name] should ensure that worker exposures do not exceed the performance requirements of 10 CFR Part 70 under accident conditions.

The NRC staff concludes that there is reasonable assurance that the applicant's radiation safety design process and design features meet the requirements of 10 CFR Part 70.

The staff could document the safety evaluation for the review for the license to possess and use SNM as follows:

The staff reviewed the license application for [insert facility name] to possess and use special nuclear material (SNM) according to Section 9.1 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found [insert a summary statement of the findings]. The applicant supplied information on the radiation safety design features and design process that demonstrate, with reasonable assurance, that radiation doses will be within the limits of 10 CFR Part 20 and will be as low as is reasonably achievable (ALARA). The applicant considered contamination control, decommissioning facilitation, and waste minimization in developing the design features of the facility, as required by 10 CFR 20.1406. The applicant also incorporated radiation safety design features as a result of the applicant's radiation safety design review and from radiation dose experience gained during the operation of other facilities.

The applicant made estimates of facility radiation sources capable of producing significant radiation levels and significant airborne radioactivity, based on [include the applicant's basis for radiation and airborne source terms]. These estimates demonstrate a conservative approach and are acceptable.

The applicant described organizational relationships and responsibilities with respect to performing radiological design reviews, which ensure the adequate application of ALARA in design-stage activities, including future facility modifications.

The general shielding design and analysis methodology used by the applicant is acceptable. The applicant has provided an adequate treatment of features requiring special analyses, such as cell penetrations, and has shown by calculation that doses in work areas meet requirements. The basic radiation transport analysis used for the applicant's shield design is based on [list appropriate shielding computer codes used].

The ventilation system at [facility name] is designed to ensure that facility personnel are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR Part 20.

The NRC staff concludes that there is reasonable assurance that the applicant's radiation safety design process and design features are adequate and, in concert with an effective radiation protection program of SRP Section 9.1, satisfy the requirements of 10 CFR Parts 20 and 70.

9.1.7 REFERENCES

The staff and industry documents listed below include those referenced in the previous sections of this chapter and additional sources that may provide useful background information for consideration in the design of MOX fuel fabrication facilities.

American National Standards Institute/ American Nuclear Society (ANSI/ANS). ANSI/ANS–6.1.1–1991, "Neutron and Gamma-Ray Fluence-to-Dose Factors."

———. ANSI/ANS–6.1.2–1999, "Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations."

———. ANSI/ANS–6.4.2–1985, "Specification of Radiation Shielding Materials."

American Nuclear Standards Institute/American Society of Mechanical Engineers (ANSI/ASME). ANSI/ASME–N510–1980 (1989), "Testing of Nuclear Air Cleaning Systems."

Energy Research and Development Administration (ERDA). ERDA 76–21, "Nuclear Air Cleaning Handbook."

Nuclear Regulatory Commission (U.S.) (NRC). NUREG–1513 (DRAFT April 2000), "Integrated Safety Analysis Guidance Document," NRC: Washington, D.C.

———. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operational)." NRC: Washington, D.C. February 1978.

———. Regulatory Guide 3.29, "Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants." NRC: Washington, D.C. May 1975.

———. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable." NRC: Washington, D.C. June 1978.

———: Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable." NRC: Washington, D.C. May 1977.

9.0 RADIATION SAFETY
9.2 RADIATION PROTECTION PROGRAM

9.2.1 PURPOSE OF REVIEW

The purpose of this review is to determine whether the applicant's radiation protection program is adequate to protect the radiological health and safety of the workers and to comply with the regulatory requirements of 10 CFR Parts 19, 20, and 70.

The applicant's program for protection of members of the public and control of effluent releases is not included in this chapter but is in SRP Chapter 10.0, "Environmental Protection." While this chapter addresses the review of the applicant's radiation protection program, radiation safety design aspects of the facility and the radiation protection aspects of the ISA Summary are reviewed under SRP Section 9.1, "Radiation Safety Design Features."

9.2.2 RESPONSIBILITY FOR REVIEW

Primary: Health Physicist

Secondary: Project Manager, Environmental Reviewer, ISA Reviewer, and Quality Assurance Reviewer

Supporting: Fuel Cycle Facility Inspector

9.2.3 AREAS OF REVIEW

As specified in 10 CFR Part 20, the applicant is subject to very specific requirements for workers' protection against radiation. 10 CFR 20.1101 requires the applicant to develop, document, and implement a radiation protection commensurate with the scope and extent of licensed activities. The requirements for a radiation protection program are specified in 10 CFR 20.1101(a), (b), (c), and (d). Areas of review should include:

A. ALARA

i. The applicant's management policy and commitments for ALARA;

ii. ALARA considerations for design (see Section 9.1);

iii. ALARA considerations for operations, including:

a. The system for operational ALARA goals, along with their bases, and a qualitative description of how the applicant will achieve the goals (i.e., numerical goals are not expected, but the applicant should commit to achieving ALARA goals and describe a methodology for achieving them); and

b. Trend analysis.

iv. The planned organizational structure and how units of that structure interact to maintain occupational doses ALARA (e.g., the ALARA Committee);

v. The applicable activities and audits carried out by the individuals in management having responsibility for radiation safety and trend analyses.

B. Organizational Relationships and Personnel Qualifications

i. The applicant's organization of the radiological protection program and the organizational relationships between the positions identified as responsible for radiation protection functions and other line managers;

ii. The qualification requirements for the radiological protection personnel; and

iii. The assignment of specific responsibilities and authorities for key functions.

C. Radiation Safety Procedures and Radiation Work Permits (RWPs)

The applicant's commitments regarding the development, control, and use of approved written radiation safety procedures and RWPs for activities related to radiological safety.

D. Training

The applicant's radiological safety training for all personnel who have authorized access to a restricted area, including:

i. Training objectives;
ii. Management oversight;
iii. Methodology of training;
iv. Who receives the training;
v. A description and the frequency of the training and refresher training; and
vi. The effectiveness of the training.

E. Air Sampling

The applicant's radiological air sampling objectives, methods, and criteria in developing sampling procedures, including:

i. The frequency and methods of analysis of airborne concentrations;
ii. Sampling methods and frequency;
iii. Counting techniques;
iv. Lower limits of detection for specific radionuclides;
v. Specific calculations for concentrations;
vi. Establishment of action levels;

vii. Location of continuous air monitors (CAMs), if used; and
viii. Annunciators and alarms associated with CAMs.

F. Contamination Control

The applicant's control of radiological contamination within the facility, including:

i. The types and frequency of surveys;
ii. Administrative contamination threshold levels;
iii. The methods and choice of instruments used in the surveys;
iv. Establishment of action levels; and
v. The design features to control access, including:

a. Technical criteria and levels defining contamination and high-contamination areas;
b. The types and availability of contamination monitoring equipment;
c. Specific limits established for personnel decontamination;
d. Minimum provisions for personnel decontamination;
e. The minimum types of clothing needed to enter contaminated areas;
f. The release criteria for contaminated materials; and
g. The frequency of periodic review of all aspects of access control.

G. External Exposure

The applicant's program for monitoring personnel external radiation exposure, including:

i. The means to measure, assess, and record personnel exposure to radiation; and

ii. The method and criteria to select the type, range, sensitivity, and frequency for analyzing personnel dosimeters and the action levels.

H. Internal Exposure

The applicant's method and criteria to develop a program for monitoring personnel internal radiation exposure, including:

i. Criteria for determining when it is necessary to monitor an individual's internal exposure;
ii. Methods for determining the worker intake;
iii. Frequency of analysis;
iv. Minimum detection levels; and
v. Setting action levels.

I. Summing Internal and External Exposure

The applicant's program for summing internal and external exposure to demonstrate compliance with dose limits, including the method used to develop procedures for assessing worker's exposures in accordance with NRC regulatory requirements.

J. Respiratory Protection

The applicant's respiratory protection program, including:

i. The equipment to be used;

ii. The conditions under which respiratory protection will be required for routine and nonroutine operations;

iii. The protection factors that will be applied when respirators are being used; and

iv. The criteria for locating the respiratory equipment within the plant.

K. Instrumentation

The applicant's methods for selection of radiological measurement instrumentation, including:

i. The policy for the maintenance and use of operating instrumentation; and
ii. The types of instruments available, including their:

a. Ranges;
b. Counting mode;
c. Sensitivity;
d. Alarm setpoints;
e. Planned use; and
f. Frequency of calibration.

9.2.4 ACCEPTANCE CRITERIA

Each subject area lists the applicable regulatory requirements and the NRC RGs, NUREG reports, BTPs, and industry standards that provide a basis that is generally acceptable to the NRC staff for satisfying the applicable regulatory requirements. However, in some cases the use of industry standards has not been endorsed by the NRC through a regulation or RG. Further, inclusion in this SRP is not necessarily an endorsement of a particular standard by the NRC. Therefore, their use is encouraged, but the applicant may propose alternative, equivalent methods with adequate justification.

9.2.4.1 ALARA (As Low As Reasonably Achievable)

9.2.4.1.1 Regulatory Requirements

10 CFR 19.12	"Instruction to Workers"
10 CFR 20.1101(b)	"Radiation Protection Program"
10 CFR 20.2102	"Records of Radiation Protection Programs"
10 CFR 20.2110	"Forms of Records"

9.2.4.1.2 Regulatory Guidance

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.10, Rev. 1–R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." NRC: Washington, D.C. May 1977.

9.2.4.1.3 Regulatory Acceptance Criteria

The requirements related to ALARA in the applicant's radiation protection program are specified in Section 9.2.4.1.1. The applicant's program should meet the regulatory requirements if the following acceptance criteria are met:

A. Management's ALARA Policies and Commitments

The applicant provides a clear management commitment to policies and provisions for maintaining individual and collective doses at levels that are ALARA. The applicant's approach addresses the regulatory guidance of RG 8.10, and provides reasonable assurance that:

i. The management commitment will be communicated to all plant personnel through policy statements, instructions to personnel, and similar documents, as well as direct communication, training, and inspection of the workplace.

ii. The management will clearly define the responsibilities of individuals to implement the ALARA policy.

iii. The Radiation Safety Manager will have the appropriate authority and independence to prevent unsafe practices.

iv. The qualifications and staffing of the radiation safety organization will be commensurate with size and complexity of the radiation protection program.

v. Workers and management will be held accountable for their radiological work performance through a review process or other similar method.

vi. Procedures and engineering controls will include formal plans and measures for applying the ALARA process to occupational exposures.

vii. Modifications to procedures, facilities, and equipment will be justified with respect to optimization of ALARA.

viii. Actions taken to maintain occupational exposures ALARA will be documented as part of the radiation protection program.

ix. Performance reviews of ALARA actions will be included as part of the radiation protection program review.

x. Individuals likely to receive an occupational dose in excess of 100 mrem (1 mSv) in a year (per 10 CFR 19.12) will be instructed on procedures and equipment used to maintain doses ALARA.

B. Design Considerations

Facility and construction design aspects related to ALARA should be reviewed using SRP Section 9.1.

C. Operational Considerations

The applicant's operational considerations for ALARA are consistent with RG 8.10, particularly as it relates to the performance of the Radiation Safety Officer (RSO) and radiation protection staff.

i. The applicant establishes a system of operational radiological performance goals (also called ALARA goals). The applicant's bases for goals could be collective dose, contamination events, intakes of radioactive material, contamination areas, radioactive waste generation, and liquid and gaseous releases. The applicant's:

a. Goals are measurable, realistic, auditable, and challenging;

b. Senior management periodically reviews the goals and progress toward meeting them; and

c. Goals are evaluated and adjusted accordingly on at least an annual basis.

ii. RSO and radiation protection staff periodically review doses associated with procedures, RWPs, and ALARA goals to identify trends (with special audits for unusual exposures). The applicant commits to perform trending analyses of key performance indicators during facility operation. Examples of key performance indicators are:

a. Radiation exposures of plant workers through bioassay results, contamination surveys, and direct measurements;

b. Concentrations of airborne radioactivity in plant areas;

c. Radioactive contamination in plant areas and on equipment;

d. Operation/malfunctions of radiation measurement instrumentation and respiratory protection equipment;

e. Concentrations of radioactive material in gaseous and liquid effluents (see SRP Chapter 10.0); and

f. Operation of effluent treatment systems (see SRP Chapter 10.0).

iii. Adequate equipment and supplies are available to the radiation protection staff to perform all personnel dosimetry, environmental monitoring, and bioassay functions.

iv. The applicant establishes a system for receiving and reviewing radiation protection-related suggestions from employees, and workers are made knowledgeable of the process [RG 8.10 C.2(b)1].

v. A system of preplanning work exists such that progressively higher levels of approval are required for higher dose activities.

D. ALARA Committee

The applicant commits to an ALARA Committee that is based on the designation and assigned responsibility and authority for implementing the applicant's ALARA policy and commitments, including the following elements:

i. The ALARA committee is shown to be an organizational structure in which radiation protection personnel will interact, in a timely manner, with production personnel to ensure that the methods and techniques for reducing occupational dose are incorporated in facility operation.

ii. The ALARA committee membership includes a chairman and management or worker representatives from the radiation protection organization, environmental organization, engineering, safety, maintenance, and production.

iii. The ALARA committee performs or receives the results of audits of the radiation protection program at least annually and reviews the results of the radiation organization's internal audits.

iv. The ALARA committee evaluates all major design activities, experiments, or plant modifications that could affect radiation levels, doses, and radioactivity levels in liquid

and gaseous effluents. The ALARA committee considers the results of the ISA in determining whether further reduction in occupational radiation doses are reasonable.

v. The ALARA committee evaluates trend analyses and the adequacy and implementation of radiological performance (ALARA) goals.

vi. The applicant commits to tracking the reviews and recommendations of the ALARA committee to completion.

9.2.4.2 Organizational Relationships and Personnel Qualifications

9.2.4.2.1 Regulatory Requirements

10 CFR 70.22(a)(6) "Contents of Applications"

10 CFR 70.23(a)(2) "Requirements for Approval of Applications"

9.2.4.2.2 Regulatory Guidance

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.10, Rev. 1–R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." NRC: Washington, D.C. May 1977.

9.2.4.2.3 Regulatory Acceptance Criteria

Requirements for organizational relationships and personnel qualifications related to radiation protection are listed in Section 9.2.4.2.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The organizational relationships clearly identify radiation protection functions and responsibilities of the radiation protection staff, and the operations, support, and engineering organizations. Additionally, each position with radiation protection functions that include authorities and responsibilities (such as those identified in RG 8.10, C.1(c)) is defined and identified. Radiation protection functions include those of the RSO, radiation staff (specialists and technicians), radiation protection engineering, the radiation training, radiation monitoring and surveillance, dosimetry and counting services, and radiation protection auditing.

   i. The RSO (or equivalent) has direct responsibility for establishing and implementing the radiation protection program, has input to facility design and operational planning, has assigned organizational emergency duties through the site emergency plan, has stop-work authority, will be independent of operations, and has direct access to the Plant Manager [see RG 8.10, C.1(e)].

   ii. The functional organization of the radiation protection staff shows that radiation protection specialists have responsibility for specific activities assigned to the radiation

protection program (e.g., dosimetry, surveys, audits, bioassay, and calibration), with radiation protection technicians implementing these functions.

B. The Plant Manager, or equivalent, has overall responsibility and authority for safety.

C. The minimum staffing of the radiation protection organization ensures that, by shift, all routine radiation functions can be performed in a timely manner and that all radiation requirements can be met during routine operations, nonroutine operations such as anticipated events, and accidents. For periods of extended absence of the RSO (because of vacations, illness, etc.), a substitute with equivalent qualifications (see Item E) and training (e.g., emergency management duties) is available to act on his or her behalf.

D. If radiation technical support or audit activities (e.g., instrument calibration and dosimetry) are contracted to qualified offsite corporate or consultant organizations, the contractors are subject to the applicant's quality assurance (QA) program and QA controls.

E. The radiation protection personnel qualifications are based on the following education and experience criteria:

i. The RSO has a bachelor's degree in science or engineering and at least 5 years of experience in health physics, with at least 1 year at a uranium or plutonium processing facility.

ii. Radiation protection specialists have a bachelor's degree in science and engineering and at least 1 year of experience in applied radiological controls at an operating nuclear facility.

iii. Radiation protection technicians have a high school diploma or equivalent, technical training commensurate with their assigned duties (e.g., dosimetry, bioassay, etc.), and certification in a technician trainee program.

An additional 5 years of experience may be substituted in lieu of a bachelor's degree. Alternative qualifications with justification can be submitted by the applicant for the Radiation Safety Officer and the Radiation Safety Specialist.

9.2.4.3 Radiation Protection Procedures and Radiation Work Permits

9.2.4.3.1 Regulatory Requirements

10 CFR 20.1101	"Radiation Protection Program"
10 CFR 70.22	Contents of Applications"
10 CFR 70.23	"Requirements for Approval of Applications"

9.2.4.3.2 Regulatory Guidance

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.10, Rev. 1–R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." NRC: Washington, D.C. May 1977.

9.2.4.3.3 Regulatory Acceptance Criteria

Requirements for radiation protection procedures and engineering controls, such as RWPs, are listed in Section 9.2.4.3.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant commits to performing activities involving exposure to licensed material in accordance with written, approved radiation protection procedures and/or RWPs.

B. In support of the applicant's commitment in Item A, the applicant's system for implementing RWPs specifies:

   i. How a determination is made to use an RWP;

   ii. The approval levels and organizational positions authorized to approve and issue RWPs (see Item D);

   iii. The types of information included on an RWP (see Item C);

   iv. Provisions for updating/terminating RWPs, including a system to update RWPs when tasks or environmental changes affect worker safety (see Item F);

   v. A method for ensuring workers are aware of the requirements, controls, restrictions, and limits in an RWP;

   vi. Records to be kept for RWPs and retention times; and

   vii. Final disposition of RWPs.

C. The applicant commits to using RWPs for specific purposes only, and RWPs are reissued when the applicant makes significant changes in the task or changes that affect the safety of workers. The application states that the RWP will include a list of the safety requirements for work conducted under the authorization and include at least the following, as applicable:

   i. The number and identification of personnel working on the task;

   ii. Expected radiological conditions (radiation, contamination, and airborne levels);

   iii. Type and frequency of monitoring and dosimetry (e.g., CAM, self-alarming dosimetry);

iv. Estimated exposure times and doses for the authorization;

v. Limiting exposure times and doses for the authorization;

vi. Special instructions or equipment (e.g., mockup required, special shielding required);

vii. Hold points or monitoring points, if applicable;

viii. PPE requirements;

ix. Authorization signature and date;

x. Actual doses, time, or other information resulting from the completed work authorization recorded on the RWP (RG 8.10 C.2(a));

xi. Expiration/termination date of the RWP; and

xii. Sufficient information on the RWPs to allow independent inspection and reconstruction of the circumstances necessitating the RWP, the factors included, and the results.

D. The RSO (or an individual who has the qualifications of the RSO) reviews and approves radiation protection procedures and RWPs (RG 8.10, C.2(b)). The applicant requires approval from other organizational groups in the preparation and approval of the RWPs to ensure that provisions of the RWP address all potential hazards (not just radiological hazards) and that operations comply with all applicable regulations.

E. The applicant commits to a system that ensures that the RWPs are not used past their termination dates. The system includes the types of records to be kept, the retention times for these records, and the final disposition of the RWP. The record system allows independent auditors to reconstruct the circumstances necessitating the RWP, the factors included, and results.

F. The applicant periodically reviews, revises, and updates radiation protection procedures and/or RWPs to identify situations for reducing doses. Reviews occur at intervals not to exceed 2 years.

G. The applicant provides a mechanism to provide current copies of radiation protection procedures and RWPs to personnel and establishes a system for ensuring that RWPs are not used past their expiration date.

H. The applicant develops, maintains, and uses radiation protection procedures and RWPs under the appropriate QA program requirements in accordance with the applicant's QA program (SRP Section 15.1).

I. The applicant commits to the use of special reviews and approvals before conducting an activity involving licensed materials with an RWP that is not covered by a written radiation safety procedure.

9.2.4.4 Radiation Training

9.2.4.4.1 Regulatory Requirements

10 CFR 19.12	"Instruction to Workers"
10 CFR 70.22(a)(6)	"Contents of Applications"
10 CFR 70.23(a)(2)	"Requirements for the Approval of Applications"

9.2.4.4.2 Regulatory Guidance

American Society for Testing and Materials (ASTM). ASTM–C986–1989 r. 1995, "Developing Training Programs for the Nuclear Fuel Cycle."

———. ASTM–E1168–1995, "Standard Guide for Radiological Protection Training for Nuclear Facility Workers."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.10, Rev. 1–R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." NRC: Washington, D.C. May 1977.

———. Regulatory Guide 8.29, Rev. 1, "Instructions Concerning the Risks from Occupational Radiation Exposure." NRC: Washington, D.C. February 1996.

———. NUREG–0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials." NRC: Washington, D.C. October 1976.

9.2.4.4.3 Regulatory Acceptance Criteria

Requirements for radiation training are listed in Section 9.2.4.4.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. All personnel and visitors entering restricted areas either receive radiation protection training or are provided a general indoctrination in site-specific safe practices and escorted by an individual who has received radiation training. If this is not the case, than radiation protection training is given prior to occupational exposure and periodically thereafter (RG 8.29); refresher radiation protection training is completed not later than 2 years following the most recent radiation protection training. However, employees authorized to perform "higher risk" work should be requalified annually (ASTM–E1168–1995).

B. The applicant's process for developing a radiation protection training program follows the process outlined in ASTM–C986–89 (reapproved 1995). The radiation protection training program objectives, content, testing, requalifications, recordkeeping, and audits are consistent with the ASTM– E1168–1995 standard and Appendix A of RG 8.29. The applicant demonstrates equivalence if it elects not to use these standards.

C. The technical content and extent of radiation protection training is commensurate with the radiological risk present in the workplace (RG 8.29 and ASTM–C986–1995) and is accomplished by grading the training requirements for general employees, radiation workers, radiation technicians, and supervisors. In addition, training for all groups, except general employees, includes practical demonstrations by trainees of proper equipment use, dosimetry use, PPE use, and incident (e.g., spill) response.

D. To verify the radiation protection training received, the applicant commits to having each trainee acknowledge in writing that the training was received and understood (RG 8.29). The applicant maintains the records of most recent training and testing as specified in ASTM–E1168–1995.

9.2.4.5 Air Sampling

9.2.4.5.1 Regulatory Requirements

10 CFR 20.1204	"Determination of Internal Exposure"
10 CFR 20.1703	"Use of Individual Respiratory Protection Equipment"
10 CFR 20.1902	"Posting Requirements"
10 CFR 20.2103	"Records of Surveys"
10 CFR 20.2110	"Forms of Records"
10 CFR 20.2203(a)(3)(i)-(ii)	"Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Materials Exceeding the Limits"
10 CFR 70.22(a)(7)	"Contents of Applications"

9.2.4.5.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–N42.17B–1989, "Performance Specifications for Health Physics Instrumentation—Occupational Airborne Radioactivity Monitoring Instrumentation."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.25 Rev. 1, "Air Sampling in the Workplace." NRC: Washington, D.C. June 1992.

———. Regulatory Guide 8.36, "Radiation Doses to the Embryo/Fetus." NRC: Washington, D.C. July 1992.

———. NUREG–0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials." NRC: Washington, D.C. October 1976.

———. NUREG–1400, "Air Sampling in the Workplace." NRC: Washington, D.C. September 1993.

9.2.4.5.3 Regulatory Acceptance Criteria

Requirements for air sampling are listed in Section 9.2.4.5.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant commits to using an air sampling program that is consistent with the positions in RG 8.25, including evaluating the need for air sampling, locating samplers, determining sample representativeness and conditions for adjusting derived air concentrations (DACs), measuring sampled air volume, and evaluating results. NUREG–1400 complements RG 8.25 and presents examples, methods, and techniques for implementing the recommendations of RG 8.25.

B. The applicant's basis for the air sampling program includes:

   i. For each work area, a determination that the frequency for analyzing airborne levels of radioactivity, counting techniques, action levels and actions to be taken when action levels are exceeded, and alarm set points are adequate to meet Part 20; and

   ii. Calculations and verification of airborne concentrations in various areas are controlled under the applicant's QA program (SRP Section 15.1).

C. The applicant's use of and specifications for air sampling instrumentation are consistent with RG 8.25 and ANSI–N42.17B–1989. Calibration methods and frequencies for air sampling instruments ensure proper operation of the instrumentation, including the operation of flow rate meters. The applicant specifies the locations of detectors, readouts, annunciators, and alarms. (The applicant may provide this information in support of SRP Section 9.1.4.2; however, the applicant should provide a cross-reference to this material.)

D. The applicant's action levels for airborne activity use appropriate technical criteria to determine the necessary controls, including the minimum detectable concentrations (MDCs) for the radionuclides of interest.

9.2.4.6 Contamination Control

9.2.4.6.1 Regulatory Requirements

10 CFR 20.1406	"Minimization of Contamination"
10 CFR 20.1501	"Surveys and Monitoring—General"
10 CFR 20.1601	"Control of Access to High Radiation Areas"
10 CFR 20.1602	"Control of Access to Very High Radiation Areas"
10 CFR 20.1703	"Use of Individual Respiratory Protection Equipment"
10 CFR 20.1901	"Caution Signs"
10 CFR 20.1902	"Posting Requirements"
10 CFR 20.1904	"Labeling Containers"
10 CFR 20.1906	"Procedures for Receiving and Opening Packages"
10 CFR 20.2103	"Records of Surveys"
10 CFR 20.2110	"Forms of Records"
10 CFR 20.2203	"Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Materials Exceeding the Limits"

9.2.4.6.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–N323-1978 r.1983, "Radiation Protection Instrumentation Tests and Calibrations."

———. ANSI–N542-1977, "Sealed Radioactive Sources Classification."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.24 Rev. 1, "Health Physics Surveys During Enriched Uranium 235 Processing and Fuel Fabrication." NRC: Washington, D.C. October 1979.

———. BranchTechnical Position (BTP), "License Condition for Leak Testing Sealed Byproduct Material Sources." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing Sealed Plutonium Sources." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Plutonium Alpha Sources." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing a Sealed Source which Contains Alpha and/or Beta-Gamma Emitters." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing Sealed Uranium Sources." NRC: Washington, D.C. April 1993.

———. BTP, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." NRC: Washington, D.C. April 1993.

9.2.4.6.3 Regulatory Acceptance Criteria

Requirements for contamination control are listed in Section 9.2.4.6.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant's policy for controlling contamination is stated clearly. The policy mandates the use of personnel monitoring equipment, and that personnel perform a whole-body survey each time they leave a known contamination area, or a minimum hand and shoe survey each time they leave a potentially contaminated restricted area.

B. Features of the facility that help control contamination are consistent with RG 8.24 and included in the facility descriptions (e.g., fume hoods, step-off pads, personnel monitoring equipment at egress points).

C. The applicant's facility operating procedures include procedures that minimize, to the extent practical, contamination in the facility pursuant to 10 CFR 20.1406 and a commitment to a contamination survey program.

D. The contamination survey program is based on the information provided in RG 8.24 on contamination-level limits and types, methods, instruments, and frequencies of surveys. For each area, the applicant specifies the types of radiation, the criteria for contamination action levels (for both removable and fixed contamination), action levels, and actions to be taken if exceeded. Contamination surveys are conducted routinely for the accessible areas of the plant site where contamination is likely. The types of instruments and methods used in the surveys are adequate to allow assessment of working conditions. The instruments are sufficiently sensitive to measure contamination at or below the assigned action levels and tested and calibrated in accordance with ANSI–N323–1978 r. 1983 (or equivalent).

E. The applicant documents contamination surveys, investigations, corrective actions, and reviews, along with deficiencies. The RSO reviews this documentation for possible trends and needed corrective actions. The applicant tracks contamination levels and contaminated areas as part of the ALARA goals (see Section 9.2.4.1.3(C)).

F. The applicant's maximum personnel contamination levels for skin and clothing are established and specified consistent with RG 8.24. The applicant uses means to detect contamination in excess of these levels. If the applicant detects contamination in excess of these levels, the applicant then decontaminates; investigates; corrects; and documents the source, probable cause, and other pertinent information. The applicant states the minimum detectable levels.

G. The applicant's access control and security of stored radioactive material are in accordance with 10 CFR Part 20, and the applicant performs periodic reviews to verify:

   i. Proper posting, labeling, and operability of access controls;

   ii. Proper identification of restricted areas to prevent the spread of contamination; and

   iii. Sufficient numbers and appropriate locations of step-off pads, change facilities, PPE facilities, and personnel monitoring equipment.

H. The applicant establishes a system that ensures that equipment and materials removed from contaminated areas are not contaminated above specific release levels. The contamination levels of items (tools, equipment, etc.) given release clearance are in accordance with NRC's BTP, "Guidelines for Contamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material."

I. The applicant performs sealed source leak testing in accordance with written procedures and in accordance with the NRC BTPs listed in Section 9.2.4.6.2. The procedures include acceptable contamination levels, test frequencies, and actions if action limits are exceeded.

9.2.4.7 External Exposure

9.2.4.7.1 Regulatory Requirements

10 CFR 19.13	"Notifications and Reports to Individuals"
10 CFR 20.1201	"Occupational Dose Limits For Adults"
10 CFR 20.1203	"Determination of External Doses from Airborne Radioactive Material"
10 CFR 20.1206	"Planned Special Exposures"
10 CFR 20.1301	"Dose Limits for Individual Members of the Public"
10 CFR 20.1302	"Compliance with Dose Limits for Individual Members of the Public"

10 CFR 20.1501	"Surveys and Monitoring—General"
10 CFR 20.1502	"Conditions Requiring Individual Monitoring of External and Internal Occupational Doses"
10 CFR 20.1601	"Control of Access to High Radiation Areas"
10 CFR 20.1602	"Control of Access to Very High Radiation Areas"
10 CFR 20.1901	"Caution Signs"
10 CFR 20.1902	"Posting Requirements"
10 CFR 20.1906	"Procedures for Receiving and Opening Packages"
10 CFR 20.2101	"Records—General Provisions"
10 CFR 20.2103	"Records of Surveys"
10 CFR 20.2105	"Records of Planned Special Exposures"
10 CFR 20.2106	"Records of Individual Monitoring Results"
10 CFR 20.2110	"Forms of Records"
10 CFR 20.2202	"Notification of Incidents"
10 CFR 20.2203	"Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Materials Exceeding the Limits"
10 CFR 20.2206	"Reports of Individual Monitoring"

9.2.4.7.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–N13.11–1983, "Personnel Dosimetry Performance, Criteria for Testing."

———. ANSI–N13.15–1985, "Dosimetry Systems, Performance of Personnel Thermoluminescence."

———. ANSI–N13.27–1981 r. 1992, "Dosimeters and Alarm Ratemeters, Performance Requirements for Pocket-Sized Alarm."

———. ANSI–N322–1977, "Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters."

———. ANSI–N323–1978 r. 1983, "Radiation Protection Instrumentation Tests and Calibrations."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.4, "Direct and Indirect-Reading Pocket Dosimeters." NRC: Washington, D.C. February 1973.

———. RG 8.7 Rev. 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data." NRC: Washington, D.C. June 1992.

———. RG 8.28, "Audible Alarm Dosimeters." NRC: Washington, D.C. August 1981.

———. RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses." NRC: Washington, D.C. July 1992.

———. RG 8.35, "Planned Special Exposures." NRC: Washington, D.C. June 1992.

9.2.4.7.3 Regulatory Acceptance Criteria

The requirements for external exposure are listed in Section 9.2.4.7.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant determines who are and are not occupationally exposed individuals and who is to be monitored for exposure in accordance with RG 8.34. For nonoccupationally exposed workers, the limits for members of the public apply, and acceptability is based on compliance with the surveys required by 10 CFR 20.1302.

B. The applicant's type, range, sensitivity, accuracy, frequency for personnel dosimetry and area dosimetry, and methods for recording measured dose are justified for the types, energy, and amount of radiation and are consistent with ANSI N13.11-1983, ANSI N13.15–1985, ANSI N13.27–1981 r. 1992, ANSI N322–1977, and ANSI N323–1978 r. 1983.

C. The applicant may use administrative dose levels, below 10 CFR Part 20 limits, to demonstrate that doses are maintained ALARA. The applicant specifies administrative dose limits that are a fraction (e.g., 20 percent) of the 10 CFR Part 20 limits, and the applicant identifies the actions and approvals necessary to exceed the administrative dose limits.

D. A dosimetry processor, holding accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP), processes and evaluates personnel dosimetry (except those specified in 10 CFR 20.1501(c)).

E. The applicant's source identification and control program:

i. Identifies sources of external exposure throughout the facility along with controls and responsibilities for restricted, controlled, and unrestricted areas;

ii. Identifies methods for materials inventory, movement, and storage to prevent releases and limit external exposures; and

iii. Complies with 10 CFR 20.1906, 10 CFR Part 71, and U.S. Department of Transportation requirements (49 CFR 171–178) for the receipt and offsite transfer of radioactive materials.

9.2.4.8 Internal Exposure

9.2.4.8.1 Regulatory Requirements

10 CFR 19.13	"Notifications and Reports to Individuals"
10 CFR 20.1201	"Occupational Dose Limits for Adults"
10 CFR 20.1204	"Determination of Internal Exposure"
10 CFR 20.1206	"Planned Special Exposures"
10 CFR 20.1301	"Dose Limits for Individual Members of the Public"
10 CFR 20.1302	"Compliance with Dose Limits for Individual Members of the Public"
10 CFR 20.1502	"Conditions Requiring Individual Monitoring of External and Internal Occupational Doses"
10 CFR 20.1703	"Use of Individual Respiratory Protection Equipment"
10 CFR 20.1901	"Caution Signs"
10 CFR 20.1902	"Posting Requirements"
10 CFR 20.2101	"Records–General Provisions"
10 CFR 20.2103	"Records of Surveys"
10 CFR 20.2105	"Records of Planned Special Exposures"
10 CFR 20.2106	"Records of Individual Monitoring Results"
10 CFR 20.2110	"Forms of Records"

10 CFR 20.2202	"Notification of Incidents"
10 CFR 20.2203	"Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Materials Exceeding the Limits"
10 CFR 20.2206	"Reports of Individual Monitoring"

9.2.4.8.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–N13.22–1995, "Bioassay Program for Uranium."

———. ANSI–N13.30–1996, "Performance Criteria for Radiobioassay."

———. ANSI–N42.17B–1989, "Performance Specifications for Health Physics Instrumentation—Occupational Airborne Radioactivity Monitoring Instrumentation."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.7 Rev. 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data." NRC: Washington, D.C. June 1992.

———. Regulatory Guide 8.9 Rev. 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program." NRC: Washington, D.C. July 1993.

———. Regulatory Guide 8.13, "Instructions Concerning Prenatal Radiation Exposures (Draft DG–8014, Proposed Rev. 3, Oct. 1994)." NRC: Washington, D.C.

———. Regulatory Guide Regulatory Guide 8.25 Rev. 1, "Air Sampling in the Workplace." NRC: Washington, D.C. June 1992.

———. Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses." NRC: Washington, D.C. July 1992.

———. Regulatory Guide 8.35, "Planned Special Exposures." NRC: Washington, D.C. June 1992.

9.2.4.8.3 Regulatory Acceptance Criteria

Requirements for internal exposure are listed in Section 9.2.4.8.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant establishes and implements a program to monitor internal doses in accordance with the information, recommendations, and guidance in RG 8.9, RG 8.25, RG 8.34, and ANSI–N13.22–1995.

B. The applicant's internal dose monitoring program specifies:

   i. Criteria for participation;

   ii. Frequencies of routine measurements;

   iii. Use of confirmatory measurements;

   iv. Methods to be used;

   v. MDCs;

   vi. Action levels and actions to be taken when exceeded; and

   vii. Methods for determining worker doses from quantities of radionuclides in the body, in the work area air, and/or combinations of these.

C. When the applicant uses air sampling to determine worker intake, the applicant specifies the frequency of sampling and data analyses, the MDC, the action levels, and the actions taken when the levels are exceeded. The applicant uses bioassays to evaluate the effectiveness of using air sampling to determine worker intake.

D. When the applicant uses bioassay to determine worker intake, the applicant specifies the types of bioassay used, the frequency of data collection for each type, the MDCs, the action levels, and the actions taken when the levels are exceeded. The applicant commits to a continuing QA program on all phases of the bioassay program, including sample collection, qualifications of laboratory personnel, laboratory intercomparisons, computational checks, and use of appropriate blanks and standards.

E. The applicant commits to using engineering controls to limit the intake of radioactive material, including auxiliary ventilation systems (e.g., portable filtration systems) used to control airborne contaminants (e.g., when servicing primary ventilation or machining contaminated surfaces) and containment structures used to protect personnel working in adjacent areas, when feasible.

9.2.4.9 Summing Internal and External Exposure

9.2.4.9.1 Regulatory Requirements

10 CFR 20.1201	"Occupational Dose Limits for Adults
10 CFR 20.1202	"Compliance with Requirements for Summation of External and Internal Doses"
10 CFR 20.1206	"Planned Special Exposures"

10 CFR 20.1207	"Occupational Dose Limits for Minors"
10 CFR 20.1208	"Dose to Embryo/Fetus"
10 CFR 20.1301	"Dose Limits for Individual Members of the Public"
10 CFR 20.1302	"Compliance with Dose Limits for Individual Members of the Public"
10 CFR 20.2101	"Records—General Provisions"
10 CFR 20.2103	"Records of Surveys"
10 CFR 20.2104	"Determination of Prior Occupational Dose"
10 CFR 20.2105	"Records of Planned Special Exposures"
10 CFR 20.2106	"Records of Individual Monitoring Results"
10 CFR 20.2110	"Forms of Records"
10 CFR 20.2202	"Notification of Incidents"
10 CFR 20.2203	"Reports of Exposures, Radiation Levels, and Concentrations of Radioactive Materials Exceeding the Limits"
10 CFR 20.2206	"Reports of Individual Monitoring"

9.2.4.9.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–N13.6–1966 r. 1989, "Practice for Occupational Radiation Exposure Records Systems."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.7 Rev. 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data." NRC: Washington, D.C. June 1992.

———. Regulatory Guide RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses." NRC: Washington, D.C. July 1992.

———. Regulatory Guide RG 8.35, "Planned Special Exposures." NRC: Washington, D.C. June 1992.

9.2.4.9.3 Regulatory Acceptance Criteria

Requirements for summing internal and external exposure are listed in Section 9.2.4.9.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the applicant commits to summing internal and external doses in accordance with RGs 8.7, 8.34, and 8.36.

9.2.4.10 Respiratory Protection

9.2.4.10.1 Regulatory Requirements

10 CFR 20.1701	"Use of Process or Other Engineering Controls"
10 CFR 20.1702	"Use of Other Controls"
10 CFR 20.1703	"Use of Individual Respiratory Protection Equipment"
10 CFR 20.2110	"Forms of Records"

9.2.4.10.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–Z88.2–1992, "Practices for Respiratory Protection."

———. ANSI–Z88.6–1984, "Physical Qualifications for Respirator Use."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection." NRC: Washington, D.C. October 1976.

———. NUREG–0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials." NRC: Washington, D.C. October 1976.

9.2.4.10.3 Regulatory Acceptance Criteria

Requirements for respiratory protection are listed in Section 9.2.4.10.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant's respiratory protection program meets ANSI–Z88.2–1992, with defined responsibilities and requirements in the areas of training, control, and use of respiratory protection equipment, mask-fit testing, and breathing air purity. (ANSI–Z88.6-1984 provides additional guidance on medical qualifications and examinations for respiratory protection.)

B. The applicant maintains adequate numbers and locations of respiratory protection equipment and current training as needed to satisfy emergency response functions.

C. The applicant specifies methods to determine internal dose when respiratory protection equipment is used or when engineering and administrative controls for respiratory protection are used. The applicant's methods show a preference for engineered controls over respiratory protection equipment and the factors in the dose calculations include the time of exposure to airborne radioactive materials, the measurement and variability of airborne concentrations of radioactive material during the exposure, and for respirators, the respirator's protection factor and proper fitting.

9.2.4.11 Instrumentation

9.2.4.11.1 Regulatory Requirements

10 CFR 20.1501	"Surveys and Monitoring—General"
10 CFR 20.2103	"Records of Surveys"

9.2.4.11.2 Regulatory Guidance

American National Standards Institute (ANSI). ANSI–N13.4–1971, "Specification for Portable X- or Gamma-Radiation Survey Instruments."

———. ANSI–N42.12–1980, "Calibration and Usage of Sodium Iodide Detector Systems."

———. ANSI–N42.15–1980, "Performance Verification of Liquid Scintillation Counting Systems."

———. ANSI–N42.17A–1989, "Performance Specifications for Health Physics Instrumentation—Portable Instrumentation for Use in Normal Environmental Conditions."

———. ANSI–N42.17B–1989, "Performance Specifications for Health Physics Instrumentation—Occupational Airborne Radioactivity Monitoring Instrumentation."

Nuclear Regulatory Commission (U.S.) (NRC). Regulatory Guide 8.28, "Audible Alarm Dosimeters." NRC: Washington, D.C. August 1981.

9.2.4.11.3 Regulatory Acceptance Criteria

Requirements for instrumentation are listed in Section 9.2.4.11.1. The reviewer should find reasonable assurance that the applicant meets the regulatory requirements if the following acceptance criteria are met:

A. The applicant's policy for the maintenance and use of operating radiation instrumentation commits to continuing availability of sufficient numbers and types of instruments for all routine (Part 20) and emergency operations. The number and types of instruments available is consistent with the information on radiation measuring instruments and

instrument calibration in ANSI–N42.17A–1989, ANSI–N42.17B–1989, and ANSI–N323–1978 r. 1983.

B. The applicant's criteria for selecting radiation measuring instruments and equipment facilitates:

   i. Performing radiation and contamination surveys;

   ii. Sampling airborne radioactivity;

   iii. Monitoring area radiation;

   iv. Monitoring personnel;

   v. Performing radioactive analyses; and

   vi. Using high-range, portable instrumentation, with justified ranges, as necessary to monitor conditions during and after accidents.

C. The applicant commits to calibrating all instruments at least semiannually and to recalibrating if conditions occur that could otherwise affect the calibration, e.g., maintenance.

D. The applicant's radiation protection procedures (with respect to radiation protection instrument checks) establish daily operational checks of continuously operating radiation protection instruments.

E. The applicant identifies the locations of and describes the facilities related to radiation protection instrumentation, including:

   i. A radiochemistry laboratory equipped to perform the analyses required by 10 CFR 20.1501;

   ii. A low-background counting room equipped to perform routine counting of all plant samples (water, swipes, air); and

   iii. Instrument storage, calibration, decontamination, and maintenance facilities.

9.2.4.12 Additional Program Commitments

9.2.4.12.1 Regulatory Requirements

Regulations applicable to the additional program commitments are the following from Title 10, Part 20:

Subpart L "Records"

Subpart M	"Reports"
Section 70.61	"Performance Requirements"
Section 70.74	"Additional Reporting Requirements"

9.2.4.12.2 Regulatory Guidance

None.

9.2.4.12.3 Regulatory Acceptance Criteria

The applicant's commitment to implementing additional program commitments is acceptable if the applicant provides data and information, in the license application, that meet each of the following commitments:

A. To maintain records of the radiation protection program, including program provisions, audits, and reviews of the program content and implementation; radiation survey results (air sampling, bioassays, external exposure data from monitoring of individuals, internal intakes of radioactive material); results of corrective action program referrals; RWPs; and planned special exposures;

B. To establish a program to report to the NRC, within the time frames specified in 10 CFR 20.2202 and 10 CFR 70.74, any event that results in an occupational exposure to radiation exceeding the dose limits in Part 20;

C. To prepare and submit to the NRC an annual report of the results of individual monitoring, as required by 10 CFR 20.2206(b); and

D. To refer to the facility's corrective action program any incident that results in an occupational exposure to radiation that exceeds the dose limits in Part 20, Appendix B, or 10 CFR 70.74, and to report to the NRC both the corrective action taken (or planned) to ensure against a recurrence and the proposed schedule to achieve compliance with the applicable license condition(s).

9.2.5 REVIEW PROCEDURES

9.2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the items in Section 9.2.3, "Areas of Review," for either the construction approval review or the review for a license to possess and use SNM.

Guidance specific to the construction approval review and the review for a license to possess and use SNM is provided below.

A. Construction Approval

The applicant is not expected to address the radiation protection program in detail for the construction approval. However, the primary reviewer should evaluate the safety assessment of the design basis to ensure that the commitments and program goals, as related to the areas of review described in Section 9.2.3, are appropriate for radiation protection at the design stage.

B. License To Possess and Use SNM

Specifically, the license application should address Section 9.2.3 in full.

If the primary reviewer verifies that the radiation protection program is adequately addressed for either the construction approval review or the review for the license to possess and use SNM, the primary reviewer should accept the application for the safety evaluation in Section 9.2.5.2. If the primary reviewer identifies significant deficiencies in the material provided, the primary reviewer should request that the applicant submit additional information prior to the start of the safety evaluation.

9.2.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 9.2.5.1(A) (construction) or 9.2.5.1(B) (license), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 9.2.4. On the basis of its review, the staff may request that the applicant provide additional information or modify the application to meet those acceptance criteria.

The review performed in this section pertains to programmatic aspects of occupational doses during routine operations and anticipated events. The safety assessment of the design basis and doses from accidents are reviewed under the SRP chapter dealing with the ISA and ISA Summary (SRP Chapter 5.0) and the Radiation Safety Design Features Section (SRP Section 9.1). Doses to the public and the environment, including ALARA, are the subject of SRP Chapter 10.0, "Environmental Protection."

Guidance specific to the construction approval and the license application is provided below.

A. Construction Approval

The primary reviewer should establish that the applicant's design basis for radiation protection and related commitments will lead to a radiation protection program that will meet or exceed the regulatory acceptance criteria in Section 9.2.4.

B. License To Possess and Use SNM

The following items should be noted regarding the relationships between the primary reviewer and the secondary reviewers for this SRP section in performing the safety evaluation for the license application:

i. The plant organization, functional responsibilities, and qualifications of personnel are also reviewed as part of the SRP chapters on "Organization and Administration" (SRP Chapter 4.0) and "Training and Qualification of Plant Personnel" (SRP Section 15.4). Applicants may choose to provide the information in this section explicitly or provide a reference to those chapters. The primary reviewer of this section coordinates with the primary reviewers of the other chapters to verify that the information is complete and consistent and that the acceptance criteria are satisfied.

ii. The radiation protection training program and the respiratory protection training program could be described by the applicant in the SRP section on Training and Qualifications (SRP Section 15.4). Applicants may choose to provide the information in this section explicitly or by providing a reference to that section. The primary reviewer of this section uses the acceptance criteria in this section to evaluate these commitments, regardless of where they appear in the application.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the radiation protection program input for the SER, as described in Section 9.2.6 using the acceptance criteria from Section 9.2.4.

9.2.6 EVALUATION FINDINGS

The primary reviewer should document the safety evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, explain the basis for the findings, and state the conclusions.

The staff could document the safety evaluation for the construction approval review as follows:

The staff reviewed the license application for construction approval for [insert name of facility] according to Section 9.2 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found that [summarize the findings].

The staff concluded that the applicant provided adequate commitments and goals for the design basis as it applies to radiation protection and that these commitments and goals should result in a protection program that will meet or exceed the requirements and guidance outlined in NUREG–1718. As a result, in concert with the evaluation conducted under Section 9.1 of NUREG–1718, the applicant meets the requirements in the area of radiation protection to approve construction of the facility under 10 CFR Part 70.

The staff could document the safety evaluation for the review for the license to possess and use SNM as follows:

The staff reviewed the application for the license for [insert name of facility] to possess and use SNM according to Section 9.2 of NUREG–1718. The staff evaluated [insert a summary statement of what was evaluated] and found that [summarize the findings].

The applicant's radiation protection program includes: (1) an effective documented program to ensure that occupational radiological exposures are ALARA; (2) an organization with adequate qualification requirements for the radiation safety personnel; (3) approved written radiation protection procedures or RWPs for radiation protection activities; (4) radiation safety training for all personnel who have access to restricted areas; (5) requirements for an air sampling program; (6) control of radiological contamination within the facility; (7) a respiratory protection program; (8) requirements for radiological measurement instrumentation; and (9) a program for monitoring personnel external and internal radiation exposure. Conformance to this program should ensure safe operation and provide early detection of unfavorable trends to allow prompt corrective action.

The NRC staff concludes, with reasonable assurance, that the applicant's radiation protection program is adequate and that the applicant has the necessary technical staff to administer an effective radiation protection program that meets the requirements of 10 CFR Parts 19, 20, and 70 for a license to possess and use SNM.

9.2.7 REFERENCES

The staff and industry documents listed below include those referenced in the previous sections of this chapter and additional sources that may provide useful background information for consideration in the design of MOX fuel fabrication facilities.

American National Standards Institute (ANSI). ANSI–N13.4–1971, "Specification for Portable X- or Gamma-Radiation Survey Instruments."

———. ANSI–N13.6–1966 r. 1989, "Practice for Occupational Radiation Exposure Records Systems."

———. ANSI–N13.11–1983, "Personnel Dosimetry Performance, Criteria for Testing."

———. ANSI–N13.15–1985, "Dosimetry Systems, Performance of Personnel Thermoluminescence."

———. ANSI–N13.22–1995, "Bioassay Program for Uranium."

———. ANSI–N13.27–1981 r. 1992, "Dosimeters and Alarm Ratemeters, Performance Requirements for Pocket-Sized Alarm."

———. ANSI–N13.30–1996, "Performance Criteria for Radiobioassay."

———. ANSI–N42.12–1980, "Calibration and Usage of Sodium Iodide Detector Systems."

———. ANSI–N42.15–1980, "Performance Verification of Liquid Scintillation Counting Systems."

———. ANSI–N42.17A–1989, "Performance Specifications for Health Physics Instrumentation—Portable Instrumentation for Use in Normal Environmental Conditions."

———. ANSI–N42.17B–1989, "Performance Specifications for Health Physics Instrumentation—Occupational Airborne Radioactivity Monitoring Instrumentation."

———. ANSI–N322–1977, "Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters."

———. ANSI–N323-1978 r.1983, "Radiation Protection Instrumentation Tests and Calibrations."

———. ANSI–N542-1977, "Sealed Radioactive Sources Classification."

———. ANSI–Z88.2–1992, "Practices for Respiratory Protection."

———. ANSI–Z88.6–1984, "Physical Qualifications for Respirator Use."

American Society for Testing and Materials (ASTM). ASTM–C986–1989 r. 1995, "Developing Training Programs for the Nuclear Fuel Cycle."

———. ASTM–E1168–1995, "Standard Guide for Radiological Protection Training for Nuclear Facility Workers."

Nuclear Regulatory Commission (U.S.) (NRC). BTP, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing a Sealed Source which Contains Alpha and/or Beta-Gamma Emitters." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing Sealed Byproduct Material Sources." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing Sealed Plutonium Sources." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Leak Testing Sealed Uranium Sources." NRC: Washington, D.C. April 1993.

———. BTP, "License Condition for Plutonium Alpha Sources." NRC: Washington, D.C. April 1993.

———. NUREG–0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials." NRC: Washington, D.C. October 1976.

———. NUREG–1400, "Air Sampling in the Workplace." NRC: Washington, D.C. September 1993.

———. RG 1.33 Rev. 2, "Quality Assurance Program Requirements (Operational)." NRC: Washington, DC. February 1978.

———. RG 1.97 Rev. 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident." NRC: Washington, DC. May 1983.

———. RG 8.4, "Direct and Indirect-Reading Pocket Dosimeters." NRC: Washington, D.C. February 1973.

———. RG 8.7 Rev. 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data." NRC: Washington, D.C. June 1992.

———. RG 8.8 Rev. 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable." NRC: Washington, DC. June 1978.

———. RG 8.9 Rev. 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program." NRC: Washington, D.C. July 1993.

———. RG 8.10, Rev. 1–R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." NRC: Washington, D.C. May 1977.

———. RG 8.13, "Instructions Concerning Prenatal Radiation Exposures (Draft DG–8014, Proposed Rev. 3, Oct. 1994)." NRC: Washington, D.C.

———. RG 8.15, "Acceptable Programs for Respiratory Protection." NRC: Washington, D.C. October 1976.

———. RG 8.24 Rev. 1, "Health Physics Surveys During Enriched Uranium 235 Processing and Fuel Fabrication." NRC: Washington, D.C. October 1979.

———. RG 8.25 Rev. 1, "Air Sampling in the Workplace." NRC: Washington, D.C. June 1992.

———. RG 8.28, "Audible Alarm Dosimeters." NRC: Washington, D.C. August 1981.

———. RG 8.29, Rev. 1, "Instructions Concerning the Risks from Occupational Radiation Exposure." NRC: Washington, D.C. February 1996.

———. RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses." NRC: Washington, D.C. July 1992.

———. RG 8.35, "Planned Special Exposures." NRC: Washington, D.C. June 1992.

———. RG 8.36, "Radiation Doses to the Embryo/Fetus." July 1992.