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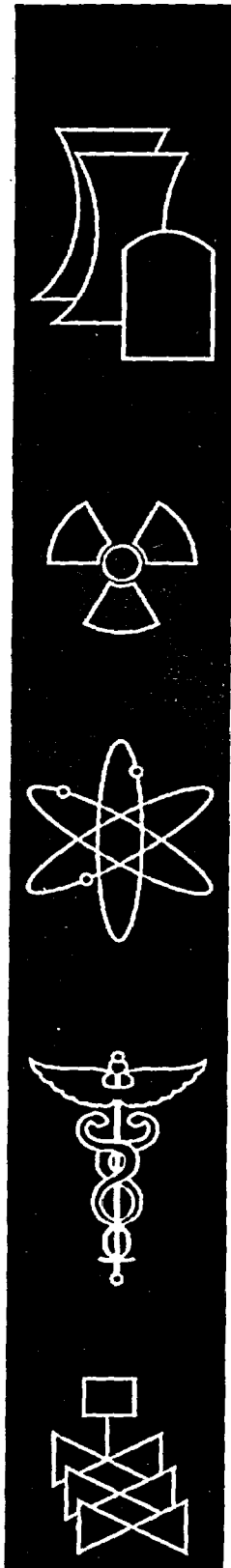

Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility

Draft Report for Comment

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

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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

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ABSTRACT

This Standard Review Plan (SRP) (Draft 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 impact reviews of the anticipated application for construction approval and the license application for operations for the Mixed Oxide (MOX) Fuel Fabrication Facility under the proposed 10 CFR Part 70 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 (SER).

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15.0 MANAGEMENT MEASURES

ACRONYMS AND ABBREVIATIONS

2SX	*2nd Pass Solvent Extraction*
AEC	*Active Engineered Control*
AEGL	*Acute Exposure Guideline Level*
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*
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 & 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*
NCS	*Nuclear Criticality Safety*
NCRP	*National Council on Radiation Protection*
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.*
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 SRP. Many of the terms are taken from 10 CFR 70.4. The definitions from this CFR section have not been changed in the list below, but are repeated for convenience. 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.

Active-engineered controls	*Controls that use active sensors to determine values of Controlled Parameters and automatically provide a response. Operation of these controls require no human intervention.*
Accident sequence	*In general, an unintended sequence of events or process failures that would result in adverse consequences. In the context of this SRP, an unintended sequence of events which results 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; or if the accident has the potential to jeopardize the safety of regulated activities. The term "accident" may be used interchangeably with accident sequence.*
Acute	*As used in 10 CFR 70.61, a single radiation dose or chemical exposure event or multiple radiation dose or chemical exposure events occurring within a short time (24 hours or less).*
Augmented-administrative controls	*Controls that use warning device(s) to notify humans that intervention is necessary to implement the controls. Operation of these controls require human intervention for implementation*
Available and reliable to perform their function when needed	*As used in Subpart H of 10 CFR 70 that, based upon the analyzed, credible conditions in the integrated safety analysis, items relied on for safety will perform their intended safety function when needed and management measures will be implemented that ensure continuous compliance with the performance requirements of 10 CFR 70.61, considering factors such as necessary maintenance, operating limits, common cause failures, and the likelihood and consequences of failure or degradation of the items and measures.*

Baseline Design Criteria	*A set of criteria specifying design features and assurance 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)	*Ensuring, as part of the safety program, oversight and control of design information, safety information, and modifications (both temporary and permanent) that might impact the ability of items relied on for safety to perform their function when needed.*
Control	*A system or device intended to regulate a device or process.*
Controlled Parameter	*A measurable parameter for which the value is maintained within a specified range by specific controls to ensure the safety of an operation.*
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.*
Credible event	*An initiating (or secondary) event that is not an incredible event (e.g., an event with a likelihood of occurrence greater than one in a million in any year). Any accident sequence identified in the ISA as initiated by a credible event must have its consequences assessed, and controls applied so as to comply with 10 CFR 70.61.*
Critical mass of special nuclear material (SNM)	*Special nuclear material in a quantity exceeding 700 grams of contained uranium-235; 520 grams of uranium-233; 450 grams of plutonium; 1500 grams of contained uranium-235, if no uranium enriched to more than 4 percent by weight of uranium-235; 450 grams of any combination thereof; or one-half such quantities if massive moderators or reflectors made of graphite, heavy water, or beryllium may be present.*
Deviation from safe operating conditions	*A parameter that is controlled to ensure adequate protection is outside its established safety limits, or that an item relied on for safety has been lost or has been degraded so that it cannot perform its intended function.*

Double contingency	*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 principle	*A licensed process should, in general, incorporate 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 protection	*A licensed process possesses double contingency protection if it has incorporated the double contingency principle. Double contingency protection is the standard; exceptions should be made only when it is not practicable and then redundancy and diversity of controls is expected to be present in the process.*
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	*Substances having licensed material as precursor compound(s) or substances that physically or chemically interact with licensed materials; that are toxic, explosive, flammable, corrosive, or reactive to the extent that they can endanger life or health if not adequately controlled. These include substances commingled with licensed material, and include substances such as hydrogen fluoride that is produced by the reaction of uranium hexafluoride and water, but do not include substances prior to process addition to licensed material or after process separation from licensed material.*

Integrated safety analysis (ISA)	*A systematic analysis to identify plant and external hazards and their potential for initiating accident sequences, the potential accident sequences, their likelihood and consequences, and the items relied on for safety. As used here, integrated means joint consideration of, and protection from, all relevant hazards, including radiological, nuclear criticality, fire, and chemical. However, with respect to compliance with the regulations of 10 CFR 70, the NRC requirement is limited to consideration of the effects of all relevant hazards on radiological safety, prevention of nuclear criticality accidents, or chemical hazards directly associated with NRC licensed radioactive material.*
Integrated safety analysis results	*The results of an ISA are all the ISA information that the applicant submits to the NRC. This includes the programmatic functions and commitments reviewed under SRP Section 5.3.1(A) plus any additional supporting information that the applicant keeps at the site.*
Integrated safety analysis summary	*The document submitted with the license application, license amendment application, or license renewal application that provides a synopsis of the results of the integrated safety analysis and contains the information specified in 10 CFR 70.65(b)*
Items relied on for safety	*Structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in 10 CFR 70.61 or to mitigate their potential consequences. This does not limit the licensee from identifying additional structures, systems, equipment, components, and activities of personnel (i.e, beyond those in the minimum set necessary for compliance with the performance requirements) as items relied on for safety.*
Management measures	*The functions performed by the licensee, generally on a continuing basis, that are applied to items relied upon for safety to ensure the items are available and reliable to perform their functions when needed. Management measures include configuration management, maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance elements.*

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 Controls	*Controls that use only fixed design features to control a Controlled Parameter. Operation of these controls require no human intervention.*
Preliminary process hazards analysis (PHA)	*An analysis undertaken during the early design or development phases of a process to identify the principal hazards and to enable them to be eliminated, minimized or controlled with minimal cost or disruption. The analysis also assists in identification and optimization of potential corrective, mitigative or preventive safety controls and management measures.*
Preventive control	*A control intended to prevent an accident entirely, i.e., to prevent any of the types of radiological or chemical consequences in 10 CFR 70.61 of any magnitude.*
Process safety information	*Information pertaining to the hazards of the material used or produced in the process, pertaining to the technology of the process, and pertaining to the equipment in the process.*

Safety control	*A system, device, or procedure intended to regulate a device, process, or human activity, so as to maintain a safe state. Effectively synonymous with "item relied on for safety". In the context of this SRP, use of the unmodified term "control" normally means safety control. Other controls will be referred to as "process controls". The function of safety controls is the avoidance of consequences of concern defined in 10 CFR 70.61. Controls may be active or passive engineered controls or administrative (procedural) controls. Controls may be preventive or mitigative. A process control may or may not be an item relied on for safety depending on whether the control of the process is required to assure safety.*
Simple-administrative controls	*Controls that requires only human intervention for implementation*
Unacceptable performance deficiencies	*Deficiencies in the items relied on for safety or the management measures that need to be corrected to ensure an adequate level of protection as defined in 10 CFR 70.61(b), (c), or (d).*
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	*An individual whose assigned duties in the course of employment involve exposure to radiation and/or radioactive material from licensed and unlicenced sources of radiation (i.e., an individual who is subject to an occupational dose as in 20 CFR 20.1003).*

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) facility specific guidance for the review and evaluation of health, safety, and environmental protection for applications to construct and operate a facility to fabricate MOX fuel. The MOX fuel fabrication facility is considered a plutonium processing and fuel fabrication plant as defined in 10 CFR 70.4. Since 10 CFR Part 70 requires construction approval for plutonium processing facilities, this SRP provides guidance to reviewers on construction approval in addition to the approval for a license to possess and use special nuclear material (SNM). This SRP is further applicable to the review and evaluation of proposed amendments and license renewal applications. Specific filing requirements for construction approval, the possession license, and for issuance of such approvals, are in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material." This SRP is guidance and, as such, is not binding on applicants for NRC licenses.

Purpose

The principal purpose of the Standard Review Plan (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. The SRP should be used as the basis for the review for the application for construction approval and the license application. 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 an application for construction approval or a license application for a MOX fuel fabrication facility.

The responsibility of the staff in the review of an application for construction approval, new or renewal license application, or license amendment for a MOX fuel fabrication facility is to determine that there is reasonable assurance that the facility can and will be constructed to operate 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 and the public and the environment. To carry out this responsibility, the staff evaluates information provided by an applicant and through independent assessments determines that the applicant has demonstrated a reasonable design basis (for construction approval) and a reasonable safety program (for license approval) that is 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 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 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.

Application for Construction Approval

To possess and use SNM in a plutonium processing and fuel fabrication facility, applicants must submit a description of the facility site; a description and safety assessment of the design bases of the principal structure, systems, and components 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 structures, systems, and components of the facility. For the purposes of this guidance, the NRC is defining the design basis as the information which identifies the specific functions to be performed by a structure, system, or component 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 basis should explain why the applicant selected particular functions or values and demonstrate how the applicant determined that the design basis 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 the proposed 10 CFR Part 70. In addition, the safety assessment should demonstrate how the requirements for new facilities identified in the proposed 10 CFR 70.64 are satisfied by the design basis. In effect, the safety assessment of the design basis should show that the design basis 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 in the application for construction approval should contain the information described in § 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 an 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 in accordance with the approved design basis and to obtain reasonable assurance that the facility will be operated without undue risk to the health and safety of workers or the public, e.g., meet the performance requirements of 10 CFR 70.61. To be acceptable, the license application, and therefore Safety Program Description, should meet the acceptance criteria of this SRP.

The Safety Program Description is the principal document with which the applicant provides the information needed by 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 Part 70.72.

Using the SRP

The requirements in 10 CFR Part 70 specify, in general terms, the information to be supplied in either the application for construction approval or the license application. The specific information that should be submitted by an 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 headed "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 in the application for 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 basis in the application for construction approval or the Safety Program Description of a facility 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 multi-disciplinary, 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 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 total 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 ISA and the project manager. Although they individually perform their review tasks, 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, and components, analyses, data, or other information that should be reviewed as part of that particular subject area of the application for construction approval or license application. Because the section identifies information to be reviewed in evaluating the adequacy of the application for construction approval or the license application, it identifies the acceptable content of the respective applications in the areas discussed. If there is a distinction between the areas of review for the application for construction approval or the license application, 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.

The 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, specific criteria, and other technical bases must be bounded by the design basis or met by the Safety Program Description. The acceptance criteria (including branch technical positions or other information) present positions and approaches that are acceptable to the staff.

It is NRC's intent that the SRP present acceptance criteria for each technical function area (e.g., nuclear criticality safety, fire safety, radiation safety), and for the management measures (e.g., quality assurance, maintenance, audits and assessments), that allow an 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 sub-process) 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 which assure that small risk. The key element in the regulatory transaction involving presentation by an 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 integrated safety analysis (ISA) for the license application and the safety assessment of the design basis for the application for construction approval.

The applicant's ISA Summary (described in and reviewed under Chapter 5 of this SRP) is 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 an 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, and it is that basis which is subject to the provisions of 10 CFR Part 70, as revised, regarding changes that a licensee may make to the facility without prior NRC approval.

An applicant for license renewal or an amendment for an existing facility responding to the requirements of 10 CFR Part 70 may propose items relied on for safety or supporting management measures that meet less stringent acceptance criteria than described in the SRP based on supporting analyses from the applicant's ISA. The ISA may be used to justify a reduced level of assurance for particular items relied on for safety, 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, as revised. The SRP criteria shown in this SRP apply to those items relied on for safety 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 basis. 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, NRC recognizes that good engineering practice dictates 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 items relied on for safety 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. An 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.

Applicants should recognize that substantial time and effort on the part of the staff have gone into the development of the acceptance criteria and that a significant amount of time and effort may be required to review and accept proposals that depart from the standard applications 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 review of the application for construction approval and the license application. 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 that 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 demonstrates the purpose of the facility. The facility and process overview should also familiarize reviewers, NRC management, or the general public with the facility and process. The facility and process overview should be abstracted from, and therefore consistent with, material presented in the applicant's design basis (for the application for construction approval) or Safety Program Description and Integrated Safety Analysis (ISA) Summary (for the license application), 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 as part of the application for construction approval and updated in the license application. The areas of review for the facility and process overview should include:

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

   i. The location of facility buildings such as plant structures, buildings, towers, and tanks and other major man-made 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 special nuclear material (SNM), including:

   i. The chemical and physical forms of SNM in the process;
   ii. The maximum amounts of SNM in 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 proposed 10 CFR 70.65, "Additional Contents of Applications."

1.1.4.2 Regulatory Guidance

There are no regulatory guides that apply to a general facility description for a mixed oxide (MOX) fuel fabrication facility.

1.1.4.3 Regulatory Acceptance Criteria

The reviewers should find the facility and process overview acceptable if:

A. The level of detail in the facility and process 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 facility and process overview appropriately cross-references the material provided in support of Chapters 5.0, 8.0, and 14.0 of this SRP.

C. The facility and process 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 update the facility and process 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 application for construction approval 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 the acceptance criteria in SRP Section 1.1.4.

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 application, 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 application for construction approval 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 for construction approval, 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 to allow construction approval for the [insert name of facility].

The staff could document the safety evaluation for the license application 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 the (1) 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 for construction approval, 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

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, 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 to understand ownership, the planned activities, and the nuclear material to be handled in connection with the requested license. The applicant's financial qualifications and 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: Office of the General Counsel
Office of Administration, Division of Facilities and Security

1.2.3 AREAS OF REVIEW

The applicant should submit the institutional information with its application for construction approval. 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 where 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, including:

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

   ii. For SNM and strategic SNM the specifications include the isotopic content and weight percent enrichment;

   iii. Identification of trace impurities or contaminants, such as fission products or transuranics characterized by identity and concentration;

   iv. The amounts of Agreement State licensed radioactive material for the proposed facility, if any; and

   v. Identification of moderator or reflector with special characteristics such as beryllium or graphite.

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

There is no regulatory guidance applicable to institutional information.

1.2.4.3 Regulatory Acceptance Criteria

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

A. Corporate Identity

The information provided by the applicant is complete and accurate. Any identified or proposed foreign ownership is explained.

B. Type, Quantity, and Form of Licensed Material

The information provided by the applicant 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. Special Exemptions or Special Authorizations

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

E. The applicant commits to update 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 application for construction approval 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 application for construction approval 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 the acceptance criteria in SRP Section 1.2.4.

The primary reviewer should not perform a detailed technical analysis of the material unless the applicant identifies foreign ownership or control. If the applicant identifies foreign ownership or

control, the primary reviewer should coordinate with the supporting reviewers so that the Division of Facilities and Security may make a determination of the acceptability of foreign ownership and control.

The primary reviewer should also 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 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 application for the construction approval 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 meets the regulatory requirements in 10 CFR 70.22 for ownership, location, planned activities, and nuclear material to be handled in connection with the construction approval for [insert name of facility].

The staff could document the safety evaluation for the license application 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 evalution for the construction approval 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 70.22 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

A. *Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.*

B. *Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, 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 basis (for the application for construction approval) or Safety Program Description and Integrated Safety Analysis (ISA) Summary (for the 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 with the application for construction approval and updated in the license application. The areas of review for the applicant's site description should include:

A. Site Geography

i. Site location: state, county, municipality, topographic quadrangle (7 ½ 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 socio-economics) 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, e.g., mass wastings.

1.3.4 ACCEPTANCE CRITERIA

1.3.4.1 Regulatory Requirements

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

1.3.4.2 Regulatory Guidance

There is no regulatory guidance applicable to the site description.

1.3.4.3 Regulatory Acceptance Criteria

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

A. The 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. The data sources are appropriately referenced and documented.

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

D. The applicant commits to update 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 application adequately 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 application for construction approval 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 the acceptance criteria in SRP Section 1.3.4.

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 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 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 of the application for construction approval 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 evalution for the license application 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 focussed on new or changed information when compared to the safety evaluation for the construction approval. 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 posses and use SNM.

1.3.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, 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 the proposed 10 CFR Part 70. The scope of this review does not include the applicant's financial qualifications for decommissioning if responsibility for eventual decommissioning will reside with the Department of Energy.

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 application for construction approval and updated in the license application. The areas of review for financial qualifications should include:

A. Project Costs

i. The 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, and construction costs.

B. Sources of Funds

i. Estimates of the total and incremental debt, equity, and revenues (if any) for each phase of the project, e.g., 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 applicant should provide the NRC with a copy of the most recent financial report and U.S. Securities and Exchange Commission 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. In the event that 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, earning before and after taxes, net earnings, and per-share earnings and dividends;

b. Consolidated statements of changes in share owners' equity;

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

d. Statements of cash flows to include 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 issuers of the on-site and off-site liability insurance to be provided for the proposed activities.

2.4 ACCEPTANCE CRITERIA

2.4.1 Regulatory Requirements

The 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

There are no regulatory guides that apply to the review of the financial qualifications for a MOX fuel fabrication facility under the proposed 10 CFR Part 70.

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)-(g).

E. Liability Insurance

On-site liability insurance is sufficient to cover reasonable expected onsite accidents and obligations. The applicant commits to maintaining off-site nuclear liability insurance in the maximum commercially available amount, unless the applicant shows that such liability will be borne by the Department of Energy.

2.5 REVIEW PROCEDURES

2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval adequately addresses 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 application is acceptable for review in accordance with Section 2.5.1, the primary reviewer should perform a safety evaluation for the application for construction approval 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 the acceptance criteria in Section 2.4. 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 license application, remain consistent with the material submitted with the application 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 application for construction approval 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 license application 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

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, 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 established and operates a security program to ensure that classified matter (i.e., National Security Information (NSI) and Restricted Data (RD)) is properly classified and protected in accordance with the requirements of 10 CFR Parts 25 and 95.

3.2 RESPONSIBILITY FOR REVIEW

Primary: Classified Matter Specialist

Secondary: Project Manager

Supporting: None

3.3 AREAS FOR REVIEW

The applicant's standard practice procedure plan for the protection of classified matter (Plan) should be submitted with the application for construction approval. The staff should review the applicant's Plan for an acceptable level of protection for using, processing, storing, reproducing, transmitting, transporting, classifying and safeguarding classified information.

3.4 ACCEPTANCE CRITERIA

3.4.1 Regulatory Requirements

The requirements applicable to protection of classified matter are contained in 10 CFR Parts 25 and 95 for the level of protection addressed in Section 3.3 as required by 10 CFR 70.22(m).

3.4.2 Regulatory Guidance

"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," dated 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 application for construction approval 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 application 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 the acceptance criteria in SRP Section 3.4.

The primary reviewer should verify that sufficient information has been provided in the application for construction approval 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 destroy 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, classifying, transporting, or destroying Top Secret information since no such information is authorized under Part 95.

If the primary reviewer concludes in the safety evaluation for the application for construction approval 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 application.

3.6 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 application for construction approval 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

A. Code of Federal Regulations, Title 10, Part 25, *Access Authorization for Licensee Personnel,* U.S. Government Printing Office, Washington, DC.

B. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

C. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

D. Code of Federal Regulations, Title 10, Part 95, *Security Facility Approval and Safeguarding of National Security Information and Restricted Data,* U.S. Government Printing Office, Washington, DC.

E. "*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,*" dated October 1999, as revised.

4.0 ORGANIZATION AND ADMINISTRATION

4.1 PURPOSE OF REVIEW

The purpose of the 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 SRP sections or chapters

Supporting: None

4.3 AREAS OF REVIEW

The applicant should submit organization and administration information with the application for construction approval and should resubmit updated information with the license application as described below. The areas of review for organization and administration should include:

A. Application for 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 health, safety, and the environment (HS&E) during design and construction.

b. Key management positions with HS&E responsibility during design and construction.

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 supervisory and management positions to perform their duties.

B. License Application

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. Key management 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 the proposed 10 CFR Part 70, concerning the applicant's corporate organization, qualifications of the staff, 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

There are no regulatory guides specific to the organization and administration for a mixed oxide (MOX) fuel fabrication facility.

4.4.3 Regulatory Acceptance Criteria

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

A. Application for 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 HS&E activities.

ii. Administration

a. The applicant commits to establish formal management measures as described in Chapter 15.0 of this SRP as necessary and appropriate to ensure 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 basis 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 with HS&E in position descriptions.

B. License Application

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 ensures 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 (construction or license) adequately addresses the specific items in Section 4.3, "Areas of Review." Specifically, the primary reviewer should compare the application for construction approval against Section 4.3(A) and the license application 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 (construction or license) 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, the primary reviewer should compare the application for construction approval against Section 4.4.3(A) and the license 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 application for construction approval or the license application, 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 (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 application for construction approval 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 provides reasonable assurance that the applicant has an acceptable organization, administrative policies, and qualified key management positions to satisfy the applicant's commitments for the design and construction of the facility.*

The staff could document the safety evaluation for the license application 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 provides reasonable assurance that the applicant has an acceptable organization, administrative policies, and qualified key management positions to satisfy the regulatory requirements for the operation of the facility.

4.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

C. NUREG-1324, "Proposed Method for Regulating Major Materials Licensees," Sections 3.1, Organization Plan, and 3.2, Managerial Controls and Oversight, U.S. Nuclear Regulatory Commission, 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 basis for the mixed oxide (MOX) fuel fabrication facility pursuant to 10 CFR 70.22(f) submitted as part of the application for construction approval; and

- The ISA for the license application, which includes:

  -- The ISA chapter of the license application, which 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 Basis for the Application for Construction Approval

The purpose of this review is to establish that the application for construction approval includes a description of the plantsite and a safety assessment of the design basis that demonstrates that the applicant's principle structures, systems, and components will provide protection against natural phenomena and the consequences of other accidents in accordance with the performance requirements of proposed 10 CFR 70.61. Pursuant to §70.22(f), the application for construction approval must be approved by the Commission prior to the beginning of construction.

The safety assessment of the design basis is neither an ISA nor a substitute for the ISA that is submitted with the license application (see Item B); instead, the safety assessment of the design basis allows the staff to determine if the applicant's design basis is adequate to meet 10 CFR 70.23(b) and to determine that the applicant, by using the safety assessment of the design basis, 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 basis 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 basis draw upon the acceptance criteria for the ISA for the license application.

B. The ISA for the License Application

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 ISA organization and procedures as may be explicitly required by the regulation, 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, worker, or the environment of the types specified in proposed 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 proposed 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 application for construction approval, which includes the applicant's design basis, safety assessment of the design bases, and principal structures, systems and components (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. The specific areas of review for the application for construction approval 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 maintain 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. The 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, and ISA results. The applicant submits the ISA Summary to the NRC with the license application 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, in particular, proposed 10 CFR 70.61. The specific areas of review are documented in Section 5.3.2(B).

5.3.1 Safety Assessment of the Design Basis for the Application for Construction Approval

For the application for construction approval for the MOX facility by 10 CFR 70.22(f), the areas of review should include those items of information relating to identification of hazards; identification of potential accident sequences, frequencies and severity of natural phenomena, and SSCs; and assessment of likelihoods and consequences of accidents. These areas of review are similar to those covered under Section 5.3.2 for the ISA 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 SRP. The areas of review should include:

A. The plantsite 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. 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.

C. The applicant's methods for conducting the safety assessment of the design basis, including the applicant's methods to evaluate chemical and radiological consequences and likelihood evaluation to show compliance with proposed §70.61.

D. The principal SSCs of the facility.

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

   i. A definition of the quantitative chemical consequence standards to be used in determining compliance with proposed §70.61;

   ii. Definition of the terms likely, unlikely, highly unlikely, and credible to be used in showing compliance with proposed §70.61;

   iii. Commitment to the methods of NUREG-1513 for hazard identification and process hazard analysis (PHA), or a description and validation of alternative methods;

   iv. A description of the design bases of the principal SSCs relied on for safety;

   v. A hazard identification for the principal SSCs relied on for safety;

   vi. A process hazard analysis identifying potential accidents; and

   vii. An assessment of the likelihoods and consequences of each general type of bounding case accident;

F. The provisions and design bases for protection against natural phenomena and the safety assessment of the design basis for natural phenomena events.

G. The design bases for protection against other accidents and the safety assessment of the design basis.

Review of the quality assurance program description required by §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 for the License Application

A. ISA Programmatic Commitments

The staff should review the license application to determine whether the applicant's commitments to perform and maintain an ISA are adequate. The areas of review should include:

   i. The applicant's commitment 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 requirements for ISA team training and qualifications (Section 15.4, "Training and Qualification of Plant Personnel").

iii. The applicant's ISA method for each individual process node, and the applicant's justification for that methods selection. 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. The applicant should follow the hazard identification with a systematic PHA of each facility process that 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 proposed 10 CFR 70.61; and
   e. Likelihood categorization for determining compliance with proposed 10 CFR 70.61.

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

   a. Performing and updating the ISA;

   b. Review responsibility;

   c. Documentation (including provisions for updating NRC on changes to IROFS or seeking NRC approval of changes per proposed §70.72); and

   d. Maintenance of ISA records per proposed §70.62(a)(2). The integrity of the ISA procedures should be controlled by the applicant's configuration management program.

B. ISA Results and Summary

The staff should review the ISA results (ISA summary and selected other ISA documentation) to find reasonable assurance that the applicant performed a systematic evaluation of the hazards and credible accident sequences and has determined that the performance objectives of proposed 10 CFR 70.61 have been satisfied. The review boundary should include those accidents that result in a release of licensed radioactive material or an inadvertent nuclear criticality event. In addition, the staff should review accidents involving hazardous chemicals when the chemicals are composed of or produced from the processing of licensed radioactive material; or if the accident has the potential to jeopardize the safety of regulated activities. The staff does not need to review event

sequences leading to consequences less than those identified in proposed 10 CFR 70.61(c) not requiring further consideration within the ISA. The areas of review should include:

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, and demography.

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. A description of each process analyzed as part of the ISA, including basic process function and theory, major components--their function and operation, process design and equipment, and process operating ranges and limits.

iv. The ISA results, primarily as documented in the ISA Summary, including:

   a. The list of hazardous materials and conditions resulting from the hazard identification task and a hazard interaction matrix table [see reference AIChE 1992, Section 3.3];

   b. Accident sequences identified by the ISA systematic PHA;

   c. Information demonstrating compliance with proposed 10 CFR 70.61, including:

      (1) Unmitigated and mitigated consequences of each postulated accident to facility workers or the public;

      (2) Comparisons of the consequences of each postulated accident to the consequence levels identified in proposed 10 CFR Part 70.61;

      (3) Assignment of accident sequences to likelihood categories and comparison to proposed 10 CFR 70.61 requirements.

v. The applicant's ISA team qualifications and ISA methods, including:

   a. ISA Team Qualifications: The ISA team leader(s) and team leader's(s') training and experience; team composition; and overall manager for the ISA process.

   b. ISA Methods: A descriptive summary of the methods used for each ISA task.

vi. The identification of, description of, and management measures applied to all IROFS that the applicant will use to ensure that, for each accident sequence, the performance requirements of proposed 10 CFR 70.61 are met, as interpreted in the acceptance criteria of Section 5.4. These criteria are risk informed in that IROFS applied to accident sequences having more severe consequences are to be correspondingly more reliable.

The applicant should also commit to maintain IROFS available and reliable for high and intermediate risk accident sequences.

Those management measures that are generically applied to all IROFS or to specified classes of IROFS may be described in Section 15.0, "Management Measures," and in Chapters 6.0 through 12.0, which cover specific safety disciplines. However, since the ISA identifies the IROFS as such and provides other information needed to apply management measures in a graded manner, the staff should review information from the ISA Summary and other ISA documentation needed to implement these measures.

vii. The applicant proposes quantitative chemical standards to assess the consequences from acute chemical exposure to licensed material or chemicals produced from licensed material.

viii. The applicant provides a list of IROFS that are the sole item for preventing or mitigating an accident sequence.

ix. For accident sequences evaluated as potentially having the consequences specified in proposed §70.61, but meeting the likelihood requirements of proposed 10 CFR 70.61 without IROFS, the basis for the applicant's evaluation of the sequence as being of acceptably low likelihood. Typically, these accident sequences are initiated by very low likelihood events e.g., natural phenomena.

5.4 ACCEPTANCE CRITERIA

5.4.1 Regulatory Requirements

10 CFR 70.23(b) requires that an applicant to construct and operate a plutonium processing and fuel fabrication facility such as a MOX facility, 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 proposed 10 CFR 70.62. Proposed 10 CFR 70.62(a)(2) requires that the applicant establish and maintain records of PSI, which is needed to perform and support the ISA. Proposed 10 CFR 70.62(c) specifies requirements for the tasks comprising the ISA, for the qualifications of ISA team personnel, and that the ISA must evaluate whether the applicant's facility, with its listed IROFS, meets the safety performance requirements of proposed §70.61. Proposed 10 CFR 70.64 specifies design criteria requirements for new facilities. Proposed 10 CFR 70.72 states requirements for keeping the ISA and its documentation current when changes are made to systems, structures, and components.

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 ISA Summary for 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 Basis for the Application for Construction Approval

The application for construction approval includes the safety assessment of the design basis and identifies the principal SSCs that will protect against natural phenomena and other accidents. The safety assessment of the design basis is not a substitute for the ISA that is submitted with the license application (see Section 5.4.3.2); instead, the safety assessment of the design basis allows the staff to determine if the applicant's design basis is adequate to meet 10 CFR 70.23(b). However, the processes the applicant uses to develop the safety assessment for the design basis should be analogous to the processes that the applicant will use to develop the ISA for the license application. Therefore, the acceptance criteria described in this section draw upon the acceptance criteria for the ISA for the license application, as described in Section 5.4.3.2.

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

A. The applicant's plantsite description includes sufficient information to permit a safety assessment of the design basis, including a site description as defined in Section 5.4.3.2(B)(i), a facility description as described in Section 5.4.3.2(B)(ii), and a process description as defined in Section 5.4.3.2(B)(iii). The level of detail the applicant provides to meet these acceptance criteria is consistent with the level of design.

B. The applicant commits to ISA programmatic commitments for completing the ISA license application (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.

C. The applicant's team qualifications for completing the safety assessment of the design basis are consistent with the acceptance criteria in Section 5.4.3.2(B)(v)(a).

D. The applicant's methods for conducting the safety assessment of the design basis are consistent with the methods the applicant will use to perform an ISA as described in Section 5.3.2(B)(v)(b). The applicant considers the level of design when it selects the methods for the safety assessment of the design basis. For example, the level of design in the application for construction may dictate that the applicant's methods for the consequence assessment are more approximate and less complete than expected for an

ISA, but should still provide reasonable estimates based on quantitative information and be consistent with valid methods.

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

E. The applicant describes the principal SSCs relied on for safety in sufficient detail to permit staff to evaluate the safety assessment of the design basis. In particular, the applicant describes the general features that indicate that the SSCs can be designed and constructed to meet the design basis. For natural phenomena hazards, the applicant provides the general aspects of the structures that make them resistant to failure. For internally initiated accidents, the applicant provides the general type of control(s) for parameters. 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. For administrative controls, the applicant identifies the types of human actions or prohibitions relied upon for safety.

   By definition (see the Glossary to this SRP or proposed 10 CFR 70.4) an SSC is an IROFS. Therefore, the applicant commits to evaluate any SSC identified in the design basis as part of the ISA. In addition, since the definition of IROFS includes equipment and personnel activities, where the applicant identifies administrative controls, it describes the type of human action or prohibition and flags the administrative controls for more detailed consideration in the ISA (see Section 5.4.3.2(B) and Chapter 12.0).

   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 basis must be described. However, approximate numbers of each general type of SSC or process is sufficient. It is the safety basis that is to be assessed.

F. The applicant's safety assessment of the design bases of the principal SSCs of the facility 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. The applicant shows that the design and design bases will result in a facility that will meet the performance requirements of proposed §70.61 and the defense-in-depth requirement of proposed §70.64(b). For processes vulnerable to criticality accidents, the applicant explains why it is expected that the given design and design bases will meet the double contingency requirement of proposed §70.64(a)(9).

   The applicant completes the safety assessment of the design bases by following steps analogous to the steps necessary to perform an ISA. However, the level of detail obtained at each step is correlated to the level of design. The applicant's safety assessment of the design basis addresses:

i. Hazard identification

The applicant identifies the approximate location and quantities of SNM and other hazardous materials (see Section 5.4.3.2(B)(ii)).

ii. Process hazard analysis and accident sequence identification

The applicant identifies the principal ways the hazards identified in Item i could impact the workers, public, or the environment, including:

a. Mechanisms to release hazardous materials;
b. Failures to control criticality parameters; and
c. Other potential initiating events for accidents;

As discussed in Item E, the accident severity will depend on the types of features, structures, control devices, or procedures the applicant will use to mitigate or prevent the accident consequences. The applicant compares the consequences of the accident with SSCs to the unmitigated accident consequences.

iii. Consequence Assessment

The applicant's consequence assessment is sufficiently quantitative to compare the consequence estimates against the performance requirements of proposed 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's safety assessment of the design basis with respect to likelihood provides reasonable assurance that the likelihood requirements of proposed §70.61 will be met by the final design. The applicant defines likely, unlikely, highly unlikely, and credible to evaluate the performance requirements for the safety assessment of the design basis and commits to use equivalent or refined definitions in the ISA for the license application. 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. The applicant's methods and definitions of likelihood terms are acceptable if they meet the same criteria as for ISA (see Section 5.4.3.2).

G. The applicant's safety assessment of the design basis for internal accidents provides reasonable assurance that the applicant will be able to meet the likelihood requirements of proposed §70.61. The applicant's safety assessment need not use the applicant's specified likelihood evaluation methods in detail, but it should be consistent with them. The applicant's safety assessment is consistent with the definitions of the likelihood terms. The applicant's safety assessment of the design basis includes:

i. The number and types of the principal SSCs;

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

a. For each SSC, the design basis parameters that will be specified or controlled for safety;

iii. The ranges and values of those parameters that constitute the design bases. The applicant demonstrates that these values are correct and incorporates sufficient safety margins to account for uncertainties. The applicant uses large safety margins when manual operations depend on operator actions as administrative controls.

H. The applicant's safety assessment of the design basis considers design basis events and shows that the likelihood for accidents resulting from natural phenomena will meet the performance requirements of proposed §70.61. The applicant's safety assessment of the design basis for natural phenomena:

i. Provides the frequency of occurrence of severity levels of the phenomena; and

ii. Demonstrates the ability of the SSC to withstand specified severity levels.

The applicant provides quantitative information that indicates that the frequencies of accidents are in accordance with the quantitative acceptance criteria for likelihood definitions given in Section 5.4.3.2(B). The acceptance criteria for assessment of the chemical and radiological consequences of accidents caused by natural phenomena are the same as described in Item G.

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.

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 proposed §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 proposed §70.61.

A. ISA Programmatic Commitments

For each required program function there may be several elements necessary to carry it out 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:

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

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

iii. There is reasonable assurance that the elements, as described, would be effective in accomplishing the safety program function.

Commitment statements in the application, to be acceptable, 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:

iv. 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 as is typically included on Material Safety Data Sheets (meeting the requirements of 10 CFR 1910.1200(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 (P&IDs); 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 basis.

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.

v. The applicant commits to engage personnel with appropriate experience and expertise in engineering and process operations to update and maintain current the ISA. 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 applicant commits to those aspects of the methods for each task of the ISA, as described in the ISA Summary, that are essential to assuring that Integrated Safety Analyses of particular processes will continue to correctly evaluate compliance with the performance requirements of proposed §70.61. The applicant's description of methods may be at a somewhat more generic level than in the ISA Summary in order to permit certain methodology changes without license amendment.

vii. The applicant includes procedures and criteria for changing the ISA either in the ISA commitments or in the commitment to design and implement a facility change mechanism that meets the requirements of proposed 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.

viii. 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 that used in conducting the original ISA for any modifications and revisions that the applicant deems necessary. The applicant commits to review of any facility changes that may increase the level of risk and, if dictated by revision of the ISA, to select and implement 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 proposed 10 CFR 70.72(d)(1).

ix. The applicant commits to promptly address 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 proposed 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.

x. The applicant commits to installation of IROFS (including administrative controls) and maintaining them in a functional state so that 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 is assured.

B. ISA Results and Summary

In principle, if the applicant performs an acceptable ISA, the applicant's results could show that the applicant's processes do not comply with the performance criteria of proposed §70.61. Thus it is necessary for staff to review the ISA to verify that the applicant's ISA results demonstrate compliance with proposed 10 CFR 70.61. The staff should use the ISA Summary as the primary source of information for making this compliance determination. However, it may be necessary for the staff to request additional information or make site visits in order to reach an adequate understanding of the characteristics of selected individual processes. The review is not merely an acceptance review of ISA Summary contents, but of whether those contents demonstrate that the applicant's processes and procedures comply with proposed §70.61. It is a review to determine that the process designs, IROFS, and specific management measures applied to each process are sufficient.

The following acceptance criteria address, in the order given in proposed §70.65(b), each of the required content elements of the ISA Summary; namely descriptions of: the site, the facility, each process, process hazards, types of accident sequences, information demonstrating compliance with performance requirements, team qualifications, ISA methods, list of IROFS, quantitative chemical consequence standards, list of IROFS that are the sole items preventing or mitigating an accident sequence, and definitions of likelihood terms. 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 proposed §70.61 and other ISA requirements of proposed 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 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 man-made features such as mountains, rivers, airports, population

centers, possibly 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 proposed 10 CFR 70.61.

c. Natural phenomena (e.g., tornados, 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 basis 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.

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 facility features that are relied on or required for safety and adequately supports an overall understanding of the facility structure and its general arrangement as it pertains to the ISA. As a minimum, the applicant 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 distance to boundaries in the prevailing wind directions.

b. Design information regarding the resistance of the facility to failures caused by credible external events, when those failures may produce consequences exceeding those identified in proposed §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 proposed 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(iv)(a)) and (2) an evaluation of the likelihood and consequences of the accidents identified (see Item B(iv)(c)). Where the applicant identified a need for IROFS in the ISA results (as identified in the ISA Summary, see Item B(iv)), 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.

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

b. Major components--their function and operation, 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. The applicant's discussion includes schematics indicating safety interrelationships of parts of the process. In particular, the applicant either provides schematics or descriptions that indicate the location and geometry of special nuclear material (SNM), moderators, and other materials in the process that are sufficient for the staff to understand the adequacy of controls on mass, geometry, moderation, reflection, and other criticality parameters affected by geometry (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, i.e., nuclear criticality, radiological hazards, chemical hazards, etc., as shown in Appendix A to this SRP.

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

iv. The ISA Results As Documented in the ISA Summary

The staff should not use the regulatory acceptance criteria for the applicant's ISA results merely to confirm that the applicant conducted an ISA; instead the staff should use the regulatory acceptance criteria to determine that the applicant will be in compliance with proposed §70.61. Proposed §70.61 effectively states that each of the applicant's

credible accident sequences must be correspondingly unlikely. High consequence events must be highly unlikely; and intermediate consequence events must be unlikely. The performance criteria of proposed §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 proposed §70.61 requires that the applicant must demonstrate that intermediate consequence events will be unlikely, and high consequence events will be highly unlikely.

The applicant provides two types of information for each of the three elements: (1) the methods used and (2) the results of applying these methods to each process. That is, the applicant's information demonstrates compliance if it describes methods and criteria that should, if properly applied, provide reasonable assurance that the applicant will meet the performance criteria. In each case, the applicant's resulting accident sequences, consequences, and likelihoods for each process demonstrate that the applicant properly applied the methods. The staff should refer to Section 5.4.3.2(B)(v) for the regulatory acceptance criteria for the applicant's ISA methods.

a. Hazards

The applicant's process hazards, as provided in the ISA Summary, identify hazards of all types specific to each process relevant to determining compliance with the performance criteria of proposed §70.61. The applicant should list hazards even if no accident exists that could exceed the minimum consequences of proposed §70.61. The applicant's hazard identification:

(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.

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

(3) Is complete. To satisfy the criteria of completeness the applicant:

(a) Uses a systematic method of hazard identification in accordance with the regulatory acceptance criteria for ISA methods (see Section 5.4.3.2(B)(v));

(b) Correctly applied the method (see Section 5.4.3.2(B)(v));

(c) Did not overlook a hazard. If the staff can identify a hazard not identified, then this criterion is not met.

b. Accident Sequences

The applicant's accident sequences, as determined by a systematic PHA, permit the staff to determine that the IROFS, as described in the ISA Summary, address each type of accident sufficiently to show that the applicant will meet the performance requirements of proposed §70.61. For this reason, the applicant should not, in general, merely state that a criticality, radiological accident, or chemical release is possible. These items are merely the hazards required by Section 5.4.3.2(B)(iv)(a). Nor should the applicant, in general, merely list controlled parameters without reference to the items relied on to control that parameter.

(1) The applicant's general description of accident sequences:

(a) Covers all types of sequences of failures of IROFS. The applicant's description permits the staff to determine that all accident sequences that could exceed the minimum consequence levels in proposed §70.61 are protected against by IROFS.

(b) Clearly shows the consequences and likelihood assigned to each type of accident sequence, and includes the results of any intermediate data or analysis that led to the consequence and likelihood assignments.

(c) Shows that each such type of accident is adequately addressed by IROFS and lists the specific IROFS that must fail for the type of accident to occur. The applicant's level of detail for each accident sequence is correlated to the number of combinations of failures of IROFS that lead to consequences referred to in proposed §70.61.

(2) The applicant's completeness for accident sequences

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 proposed §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.

In particular, the applicant's accident sequences are complete if the applicant:

(a) Uses a systematic method to identify accident sequences, e.g., a PHA, in accordance with the regulatory acceptance criteria for ISA methods (see Section 5.4.3.2(B)(v));

(b) Correctly applied the method (see Section 5.4.3.2(B)(v));

(c) Did not overlook an accident sequence. If the staff can identify an accident sequence not identified, then this criterion is not met.

(d) Indicates and explains why the applicant evaluated certain accidents as incredible events.

(e) Provides accident sequences. For processes having few IROFS, the applicant may use a numbered list of accident sequences. For processes with many accidents resulting from multiple combinations of IROFS failures, the applicant provides a logic diagram, such as a fault tree, that describes all accident sequences in a succinct explicit format. Appendix A to this SRP shows a third acceptable way of providing a general description of accident sequences; namely, in a tabular format. The applicant provides:

   (i) A tabular summary description of the accident sequences identified in the PHA. The tabular description consists of one row for each accident sequence. The applicant summarizes accident sequences initiated by the same type of event, consisting of the same sequence of control failures, and resulting in the same consequence category as a single row. This row lists the initiating event, the IROFS that must fail in order for the accident to occur, and the level of unmitigated consequences, if all IROFS fail. The tabular summary identifies the severity level of each type of consequence (radiological, criticality, chemical, environmental) according to the values defined in proposed 10 CFR 70.61. The applicant tabulates information sufficient for staff evaluation of compliance with the likelihood requirements of proposed 10 CFR 70.61, such as likelihood indices. Appendix A to this SRP provides an acceptable way of presenting this information; or

   (ii) A set of logic diagrams, such as fault trees or event trees for each process, presenting the same information as in Item (i); or

(iii) A numbered list of narrative summaries describing each type of accident sequence for the process and containing the same information as the tabular summary of Item (i).

c. Information Demonstrating Compliance with Proposed 10 CFR 70.61

The third required item in the ISA Summary is "information that demonstrates compliance with the performance criteria of §70.61" which addresses the consequences and likelihoods of the accident sequences.

(1) Consequences: The applicant's consequences demonstrate compliance with proposed 10 CFR 70.61 if:

(a) 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 proposed 10 CFR 70.61, or includes a reference to a value documented elsewhere in the ISA Summary that applies to or bounds that accident;

(b) The applicant calculated the consequences in accordance with the regulatory acceptance criteria for ISA methods (see Section 5.4.3.2(B(v));

(c) 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;

(d) The applicant's ISA Summary correctly assigns each type of accident to one of the consequence categories of proposed §70.61; and

(e) The applicant assigns criticality accidents as high consequence events. For processes with effective engineered shielding, criticalities may produce doses below the intermediate consequences of proposed §70.61. As stated in the regulation, notwithstanding shielding or other mitigative features, the applicant must place primary reliance on the prevention of criticalities. When the applicant uses shielding, the applicant may use preventive measures of lower reliability. That is, shielded criticality events need not be highly unlikely.

(2) Likelihood: The applicant's likelihoods demonstrate compliance with proposed 10 CFR 70.61 if:

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

(b) The applicant provides information that allows the staff to assess whether the applicant correctly assigned likelihoods as shown in the tabular method in Appendix A to this SRP. The applicant's information includes intermediate data such as whether an IROFS is active or passive, the degree of redundancy, information on independence, and methods and time intervals for surveillance of IROFS to limit the duration that they may be in a failed state. Much of the applicant's information relevant to likelihood of failure of individual IROFS is provided in the descriptive list of IROFS, a required item in the ISA Summary (see Section 5.4.3.2(B)(vi)). However, the applicant should show redundancy and independence among multiple IROFS used for a single type of accident through a method such as fault trees or the tabular format of Appendix A to this SRP.

(c) The applicant evaluated likelihoods in accordance with the regulatory acceptance criteria for ISA methods (see Section 5.4.3.2(B)(v));

(d) The applicant's evaluated likelihoods comply with acceptable definitions of the terms "unlikely" and "highly unlikely" for use with proposed §70.61 as evaluated in Section 5.4.3.2(B)(ix) of this SRP;

(e) The applicant evaluated unshielded nuclear criticality accident sequences with a likelihood of "highly unlikely" and, in general, possesses double contingency protection; and

(f) The applicant evaluated shielded nuclear criticality accident sequences, regardless of estimated radiation doses, as not substantially less unlikely than "highly unlikely," but may not possess double contingency protection.

v. ISA Team Qualifications and ISA Methods

a. ISA Team Qualifications: The applicant's ISA teams and team qualifications, as stated in the ISA Summary, include:

(1) 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, although the team leader need not be the cognizant engineer or expert for that process, the team leader can demonstrate an adequate understanding of all process operations and hazards under evaluation.

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

(3) The team represents a variety of process design and safety experience 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.

(4) A manager provides overall administrative and technical direction for the ISA.

b. ISA Methods: The applicant's descriptive summary of the ISA methods describes the methods used for each ISA task (e.g., see Section 5.3.2(A)(iii)), and the applicant's basis for selecting each method, so that the adequacy of the method is clear and appropriate according to the criteria described in this SRP and NUREG-1513. Or, if in any case, the applicant selects an alternative method, the applicant justifies the proper selection of that method in the ISA programmatic commitments (see Section 5.4.3.2(A)) and meets any additional regulatory acceptance criteria specified in Items (1)-(4). Specific acceptance criteria for the ISA methods for each task are:

(1) Hazard Identification Method: The applicant's hazard identification method leads to a hazard identification that satisfies the regulatory acceptance criteria specified in Section 5.4.3.2(B)(iv)(a).

(2) 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:

(a) The applicant uses criteria for an individual PHA process that are consistent with the principles of the PHA selection criteria in NUREG-1513;

(b) 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.

(c) 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 proposed §70.61[1].

(d) 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 proposed §70.61 and appropriately limited.

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

[1]The release of hazardous chemicals is of regulatory concern to NRC only to the extent that such hazardous releases result from the processing of licensed nuclear material or have the potential for adversely affecting radiological safety.

(f) 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 or 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.

(g) 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.

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

(i) The applicant provides justification, in the ISA Summary, that the individual method would effectively accomplish Items (a) through (h) above.

(3) Consequence Analysis Method. The applicant's method for ISA consequence evaluation, as described in the ISA Summary:

(a) Consists of or is consistent with, the approaches described in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," March 1998. (NUREG/CR-6410 also provides methods for estimating magnitudes of criticality events.) Or, if the applicant used an alternative method, the applicant described and justified in the alternative in the methods section of the ISA Summary (see Section 5.4.3.2(B)(vi)).

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

(c) Uses reasonably generic assumptions and data for the types of accidents analyzed.

(4) Likelihood Evaluation Method. The applicant's evaluation method for likelihood, as described in the ISA Summary, demonstrates compliance with the graded protection criteria of proposed 10 CFR 70.61 consistent with the guidance in the Appendix A to this SRP. Or, for individual accident sequences not conforming to the guidance in Appendix A, specific and adequate justification showing conformance to proposed 10 CFR 70.61 is provided.

vi. List of IROFS

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 Table A-7 in Appendix A to this SRP. The applicant's "list describing items relied on for safety" required by proposed 10 CFR 70.62(c)(vi):

a. Includes all IROFS in the identified accident sequences. 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 include both engineered controls and administrative controls. All such items must be listed, no matter how low their safety significance, if they are relied on to demonstrate compliance with the safety performance requirements of proposed §70.61. Such items may assure compliance by making the accident unlikely or by mitigating its consequences.

   For example, if a process upset is required before an accident may occur and in showing compliance with proposed §70.61, the applicant places reliance on the fact that this process upset is an unlikely event, then those features of the process that assure that the upset is of low frequency are IROFS. Similarly, if the dimension or the material composition of a piece of process equipment is essential to preventing an accident, then that dimension or material is an IROFS. In such cases, only those dimensions, features, or properties of the process that are essential to the safety function are IROFS. It is essential that the applicant identify such process features so that a description of their safety function is available to safety staff for change control.

b. 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 a.

c. Describes the IROFS, management measures, and the associated safety limits and margins to permit a determination of compliance with proposed 10 CFR 70.62(c)(vi). The applicant describes the essential features of each IROFS, including hardware controls, that are required to achieve adequate reliability. If the IROFS is an administrative control, the applicant describes the nature of the action or prohibition involved sufficiently to permit an understanding that, in principle, adherence to it should be reliable. The description of each IROFS contains any information needed to identify how the management measures, such as maintenance, training, configuration management, etc. of proposed 10 CFR 70.62(d) are applied to it.

d. Provides information concerning the assignment of management measures to engineered and administrative controls in accordance with proposed 10 CFR 70.62(d). If the applicant uses a system of graded management measures, the staff can determine the grade applied to each IROFS from information provided by the applicant. To show compliance with the performance requirements of proposed 10 CFR 70.61, the applicant's description of the IROFS and the

management measures applied to them must show how they meet all applicable provisions of the baseline design criteria (BDC) as described in Chapters 6.0 through 12.0 and Chapter 15.0, or a lesser set of measures if justified. If applicable, the applicant's primary justification for lesser management measures is lower risk significance.

The applicant's management measures include a description of the facility procedures for conducting and maintaining the ISA that includes: management policies; organizational responsibilities; administrative controls; and procedures governing the performance, review, and approval of the initial ISA and any revisions to the ISA. The applicant commits to evaluating the need for updating the ISA to reflect changes using a team with qualifications appropriate for the process in its changed configuration. In addition, the applicant commits to maintain the ISA under an adequate configuration management function. The applicant also identifies updates to the list describing the IROFS. The applicant describes facility procedures for reviewing process changes and new safety information to determine if prior NRC approval is required in accordance with proposed §70.72. Administrative controls ensure the independence of reviewing organizations and individual reviewers. The applicant establishes procedures to control records and supporting documentation concerning the ISA.

vii. 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 includes:

a. Three unambiguous quantitative standards for each of the applicable hazardous chemicals on site corresponding to each of proposed: (1) §70.61(b)(4)(i), (2) §70.61(b)(4)(ii) and §70.61(c)(4)(i), and (3) §70.61(c)(4)(ii).

b. The quantitative standard for proposed §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," which means death, although not the average result, could occur in a reasonable number of cases.

c. The quantitative standard for proposed §70.61(b)(4)(ii) and §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 proposed §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 Accute Exposure Guideline Level (AEGL) series of standards to be acceptable sets that meet the performance criteria of proposed §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-c.

viii. Definitions of Likelihood

Proposed §70.65 requires that the applicant's ISA Summary provide definitions of the terms unlikely, highly unlikely, and credible. The applicant's definitions of these terms is acceptable if, when taken together with the description in the ISA Summary of the applicant's method of assessing likelihoods, they provide reasonable assurance that the requirements of proposed §70.61 can be met. These likelihood, or frequency, definitions are needed because they are used in specifying the performance requirements of proposed §70.61 for each accident. Proposed §70.61 does not explicitly require the applicant to use quantitative definitions for these terms. However, in order to provide a basis for consistency, this section provides quantitative guidelines for the staff to interpret the applicant's definitions. An applicant may provided quantitative definitions, and these are acceptable if consistent with the quantitative guidelines in this section. If the applicant's definitions are qualitative, they are acceptable to the extent that they are (a) reasonably clear and objective and (b) reasonably consistent with the quantitative guidelines in this section.

Proposed §70.61 requires that accidents of a given level of consequences have a corresponding likelihood. Thus the meaning of the likelihood terms are on a "per accident" basis. To be acceptable, the applicant's definitions must be on a per accident basis.

The quantitative likelihoods are derived from Commission strategic safety performance goals. Hence, acceptable guidelines for quantitative frequencies for each level of likelihood depends on how many potential accidents there are in each of the two consequence categories. The number of accidents will not be known until ISA results are available for the industry. For this reason, the quantitative guidelines are expressed in terms of this, currently unknown, total number of accidents.

It should be noted that the quantitative likelihood definitions are maximum acceptable limits. That is, definitions based on lower limits are also acceptable.

The quantitative consequence categories defined in proposed §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 proposed

§70.61. In deriving the quantitative likelihood guidelines below, the typical high consequence accident is assumed to be equivalent to a nuclear criticality, in which a few workers would receive doses exceeding 100 rem, some of them possibly fatal. Thus for accidents producing "high consequences" similar to a typical criticality, the quantitative guideline for "highly unlikely" given below is appropriate. But, if an accident would produce much larger consequences, the quantitative definition of "highly unlikely" must be appropriately lower to be acceptable.

The term "credible" is used in proposed 10 CFR 70.61 in the following context: "The risk of each credible high consequence event shall be limited ... through the application of ...controls ...". Thus credible is a criterion for exemption from use of controls; controls mean IROFS. This implies that the reason that an event would not be credible must not depend on IROFS, but on external or natural phenomena or some feature of the facility that can be relied on without being in the facility change control system. In general, events which are not credible are either physically impossible, require very low likelihood external initiators, involve a long series of very unlikely events, or involve an extremely improbable series of human actions for which no motivation exists. Actions deliberately intended to cause accidents are also ignored; however, actions such as nuclear sabotage should be considered separately as part of the evaluation for physical protection (see Section 13.1).

The term credible is used in the rule in a way that implies that events that are not credible can be ignored. The guideline given is that a credible accident is one with a frequency greater than 10^{-6} per year. The rationale behind use of the term credible in the rule is that there may be events that have about the same maximal consequences as the typical high consequence event, but are of much lower likelihood. Such events can be ignored if their cumulative risk is negligible compared to the risk from the more typical events assessed. However, there is a potential for misinterpretation by the applicant. Such events must be incredible for reasons that are extremely unlikely to be changed. An accident cannot be incredible because of a feature of the plant that might be changed, because the feature could be changed so that the event is no longer unlikely. Thus any plant feature that makes events "incredible," or is otherwise needed to meet proposed §70.61, is an IROFS, and must be declared as such.

Subject to this guidance, the applicant's definitions of the terms likely, unlikely, highly unlikely, and credible as applied to each accident sequence in the ISA show compliance with proposed 10 CFR 70.61 if they are reasonably consistent with the following quantitative guidelines on a per accident basis:

(1) Unlikely: Less than 0.04/Ni per year;
(2) Highly unlikely: Less than 10^{-2}/Nh per year; and
(3) Credible: Greater than 10^{-6} per year.

where:

Ni = the total number of potential intermediate consequence accidents in regulated facilities. Staff currently expects that Ni will be less than 100.

Nh = the total number of potential high consequence accidents in regulated facilities. Staff currently expects that Nh will be on the order of 1000.

If the applicant provides qualitative definitions of the terms in Items (1)-(3), the definitions are acceptable if: (a) they are used within a consistent, systematic, and reasonably objective method for evaluating each accident sequence and (b) they are reasonably consistent with the above quantitative values.

5.5 REVIEW PROCEDURES

5.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval or the license application adequately addresses the items in Section 5.1.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the application for construction approval should addresses the items in Section 5.3.1.

B. License Application

Specifically, the license application should address Section 5.3.2.

If the primary reviewer verifies that the subject area material is adequately addressed in the application for construction approval or the license application, 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) (application for construction approval) or Section 5.5.1(B) (license application), 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 the acceptance criteria in SRP Section 5.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The primary reviewer should review the design basis and the safety assessment of the design basis. The primary reviewer should coordinate with the secondary reviewers to ensure consistency between the review conducted under this chapter and reviews of the design basis and safety assessment of the design basis conducted for other subject areas, e.g., Chapters 6.0-15.0.

B. License Application

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 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 in 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 that 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 application for construction approval as follows:

The staff reviewed the application for construction approval for [insert facility name] to possess and use SNM 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 basis demonstrates that the applicant's principle 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 basis show 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 the license application 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 Integrated Safety Analysis (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 applications 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

A. AIChE, *Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples*, American Institute of Chemical Engineers, New York, September 1992.

B. American National Standards Institute, ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors," American Nuclear Society, La Grange Park, IL, 1983.

C. American National Standards Institute, ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American Nuclear Society, La Grange Park, IL, 1983.

D. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, DC.

E. NUREG-1513, *Integrated Safety Analysis Guidance Document*, 1995.

F. U.S. Dept. of Commerce, Bureau of the Census, Statistical Abstract of the United States 1995, Table No. 688.

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 the proposed 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, including 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. 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 proposed 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 proposed 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 proposed 70.72(a) and proposed 70.72(b) and (2) facility changes requiring NRC approval in accordance with proposed 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 which 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 (ISA Summary) 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, 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. 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 an application for construction approval or the license application as required by 10 CFR 70.22 and proposed 70.65, respectively. In addition, the NCS review should be conducted to ensure compliance with 10 CFR 70.24 and proposed 10 CFR 70.61, 70.62, 70.64, 70.72, and Appendix A of 10 CFR Part 70.

6.4.2 Regulatory Guidance

The NRC Regulatory Guide (RG) 3.71, *"Nuclear Criticality Safety Standards for Fuels and Materials Facilities,"* August 1998, endorses the ANSI/ANS-8 national standards listed below in part or in full.

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

B. ANSI/ANS-8.3-1997, *"Criticality Accident Alarm System."*

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

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

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

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

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

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

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

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

K. ANSI/ANS-8.19-1996, *"Administrative Practices for Nuclear Criticality Safety."*

L. ANSI/ANS-8.20-1991, *"Nuclear Criticality Safety Training."*

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

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

O. 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 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

Throughout Section 6.4.3, are several examples of how the regulatory acceptance criteria may be met in practice. The examples are presented in italicized text.

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 creates 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 proposed 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 (Organization and Administration). 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 follow the more detailed requirements of ANSI/ANS-8.19-1996 (Sections 4 through 10).

As an example of how an applicant may meet the requirements of Item B, the reviewer may observe that: The criticality safety staff would typically be trained under a documented qualification program, which includes facility and process familiarization; criticality safety practices, procedures, and guides; use of criticality codes; and all other information needed to permit them to function as safety specialists. The NCS staff is involved in responding to

emergency and accident conditions as part of the emergency response organization. Independent of operations means they would be enabled to review and concur on procedures and facility operations and have shut-down authority over any operations the staff considers unsafe; they would report to the safety manager and would be 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.

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

D. 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."

E. 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 proposed 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 proposed 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.):

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 and 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 provide instruction in the training program regarding the use of process variables as NCS controls, if such controls are credited for NCS.

iii. The applicant commits to provide 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.):

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.):

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 weekly NCS walkthroughs (e.g., checklists) of all operating SNM process areas such that all operating SNM process areas should be reviewed at least every two weeks. Identified weaknesses should be incorporated into the facility corrective actions program and should be promptly and effectively resolved. An alternate plan for reduced frequency may be justified on the basis of risk.

iv. The applicant commits to conducting and documenting quarterly NCS audits such that all NCS aspects of management measures (see Sections 15.1 through 15.8) should be audited at least every 2 years. An alternate plan for reduced frequency may be justified on the basis of risk.

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 multi-parameter limits that may be referenced).

B. The applicant commits to the intent of the requirement in Regulatory Guide 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 making use of trends in the bias.

C. As part of providing reasonable assurance that an adequate margin of subcriticality has been provided, in accordance with proposed 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 which 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. The AOA is the range of material compositions and geometric arrangements within which the bias of a calculational method is established. 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, *"Nuclear Criticality Safety in Operations With Fissionable Material Outside Reactors"* (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.

viii. A description of the verification process and results.

D. The applicant commits to incorporate 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.

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.

As an example of how Items E and F may be met, the reviewer would observe that: The applicant establishes standard criteria for modeling arrays of fissile units under certain conditions. Sample criteria for certain criticality parameters follow. If reflection is not controlled, the array is 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) is assumed. If mass is not controlled, the units are assumed to be completely filled. If neutron poison is not controlled, no credit is taken for the material of construction of the containers and fissile material is modeled to the outer diameter. If interaction is not controlled, the units are 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. Less reactive values may be used if appropriate items relied on for safety are used to limit the parameters.

G. The applicant commits to consider the variability and uncertainty in a process and the NCS subcritical limit when setting NCS safety limits.

As an example of Item G, the reviewer should ensure that: If mass is controlled to a certain value on the basis of measuring concentration of material transferred to a waste tank, then the uncertainty in the volume of the tank, the identity of the solutions transferred, and the precision

of any in-line monitors or sampling methods should be taken into consideration in setting the mass limit.

6.4.3.3.2 Additional Technical Practices

CSEs should be considered the main source of information regarding the adequacy of criticality controls. The CSEs are the documents used to develop the safety basis of facility operations. The reviewer should confirm that the CSEs establish an adequate safety basis and then verify that all controls needed by sampling selected CSEs are as flowed into the ISA Summary as items relied on for safety (IROFS).

To provide for NCS, the applicant's commitment to NCS technical practices, in meeting the performance requirements of proposed 10 CFR 70.61(d) and Baseline Design Criteria of proposed 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 proposed 10 CFR 70.61, the applicant commits to the policy that: "No single credible event or failure could result in a criticality accident."

As an example of how the applicant may satisfy Item B, the applicant may observe that: A metering pump to a uranyl nitrate-plutonium nitrate blending tank may control both the concentration in the tank and the "enrichment" (plutonium isotopics) in the tank, but its failure would be considered a single event that defeats both parameters. Therefore, an additional control is needed to meet double contingency.

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. When choosing not to use a passive-engineered control, the applicant commits to provide justification in the CSE. This should also be documented in the ISA.

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

E. The applicant commits to perform an evaluation, for all controlled parameters, that shows that during both normal and credible abnormal conditions, the controlled parameter will be maintained.

F. The applicant commits to describe 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.

G. When controlled parameters are controlled for safety reasons by measurement, reliable methods and instruments should be 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.

As an example of Item G, the reviewer should ensure that: If dual independent sampling is the only control maintaining subcriticality upon transfer of dilute solution to an unfavorable geometry tank, the following conditions would be met: to qualify as independent, the two samples would be withdrawn by different individuals and at different points in the process or at different times with mixing to ensure a representative sample in between the measurements. They would be analyzed by different analysts using different methods in the lab, and a supervisor would be required to check results before authorizing the transfer. In addition, attention should be 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.

6.4.3.3.2.0 Methods of NCS Control

Several methods of NCS control (i.e., controlled parameters) are available. These are summarized below. Justification for not using geometry control as the preferred method should be fully documented in the NCS evaluations and ISA.

The controls used to establish limits on the following criticality parameters should be identified as IROFS in the CSEs and ISA. 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 which 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) 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, 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 shall be 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. The determination of the minimum critical mass should be based on spherical geometry or the actual fixed geometry of the system if it is controlled.

As an example of implementing mass control (especially Items A and C), the reviewer may observe that: If meeting the subcritical limits on mass for handling of filtercake depends on the relative percentage of plutonium in the filtercake, then reliable means are proposed to require sampling, NDA (non-destructive assay) scanning, or other direct measurement of the mass content. These controls are unnecessary if the filtercake is assumed to be 100% plutonium in the normal case calculations. The measurement equipment employed would be tested and calibrated regularly. Great caution is needed for NCS evaluation of mixtures of fissile isotopes of different elements.

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 which rely on geometry control are verified. 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) 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.

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.

As an example of implementing geometry control (especially Items A, C, and D), the reviewer would observe that: When taking credit for the slab thickness of a set of fuel rods, the diameter of the rods and the depth of the restraining device would be conservatively taken into account, by performing field measurements and adding the geometrical tolerance to the nominal dimensions. The safety limit would be no more than some specified percentage (say, 85%) of the minimum critical slab thickness based on calculations or handbook data. A higher subcritical limit may be proposed if the applicant can demonstrate that conservative assumptions–such as neglecting the neutron absorption effect of the cladding and assuming that all the space is taken up by fuel–would be sufficient to make up the difference in margin. This would also take into consideration mounding of fuel rods and variations in the ability of the operators to meet the depth requirement.

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, 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. 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).

As an example of implementing density control (especially Item A), the reviewer may observe that: Process variables that could be controlled, that may affect the density of pellets in trays removed from a sintering furnace, including: the length of residence time in the furnace, the temperature of the furnace, the additives added to the fuel, the force applied by the pellet press, the reduction in effective density due to geometrical packing, and so forth. These would only be significant if a lower density than theoretical was assumed.

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 non-fissile 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. 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).

As an example of implementing isotopics control (Items A and B), the reviewer may observe that: Plutonium and uranium oxides are stored in readily identifiable containers with distinctive colors, with storage arrays conspicuously posted in different areas of the facility. Training would be conducted and standards for handling uranium and plutonium oxides should be consistent throughout the facility to ensure uniform handling. Upon receipt, containers could be scanned and/or sampled and the results compared to shipping documents.

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 MOX facility, where thick hydrogenous reflectors may provide shielding in several areas.)

B. At a minimum, reflection conditions equivalent to a one-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 one 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 twelve 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.

As an example of implementing reflection control (especially Items A, B, and C), the reviewer would observe that: Conservative assumptions about the thickness and composition of tangent concrete walls and floors for storage vaults, and firebrick around oxidation furnaces, are taken into account in the calculations. Beyond this, one inch nominal reflection is considered conservative to model water pipes, personnel, and other reflecting materials that may be nearby. If full water reflection is not adequately subcritical, postings and barriers could be erected to ensure that nominal reflection conditions are not exceeded.

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, 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, the design precludes 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.

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 moderating material on fissile material 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 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.

As an example of implementing moderation control (especially Items G, H, and I), the reviewer would observe that: If a fissile material system is not adequately subcritical with a few percent water density between array elements, water pipes and sprinklers would be excluded from that area. Also, the effects of fire suppression systems and activities would be evaluated. For areas under moderation control, overhead water pipes could be excluded or sleeved within secondary piping and a means provided to detect leakage from the inner pipe; drains could be provided to prevent water accumulation; and/or watertight cans and gloveboxes could be credited as a moderation barrier.

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, 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. High concentrations of SNM in a process are precluded.

C. When using a tank containing concentration-controlled solution, the tank is normally closed and locked.

D. When sampling of the concentration is needed, the sampling program uses dual independent sampling methods. The process should be designed such that a single operator acting alone cannot physically circumvent the sampling program.

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

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

G. 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).

As an example of implementing concentration control (especially Items A, B, C, D, and F), the reviewer may observe that: Plutonium nitrate tanks under concentration control should be locked and tagged, and dual independent sampling as well as an in-line monitor provided on the input line to prevent transfer of concentration solutions to the tank. (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. Specifying dual controls provides added reliability.) Process variables which maintain the plutonium nitrate in a solution form could be controlled--acid molarity, and possible precipitating agents should be excluded by removing hard-piped connections to the tanks and should be monitored by the sampling program.

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; or:

B. Unit spacing is controlled by rigorous procedures (if the spacing is identified in workstation procedures with visual indicators and postings). Justification for this method should be provided in the application and should demonstrate that multiple procedural violations will not by themselves lead to criticality.

As an example of implementing interaction control (Items A and B), the reviewer may observe that: 11-liter (2.906 gallons nominal) cylinders containing plutonium or uranyl nitrate are stored within birdcage drums, which are examined periodically for denting or other deformation. If it is important that they not be stored in a triangular pitch array, painted lines or circles on the floor and postings could be used to enforce administrative controls. In the absence of birdcage drums or other passive devices, calculations may explicitly demonstrate that placing together several cylinders in a tight configuration in the corner of the room will not result in a 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 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."*

As an example of implementing neutron absorber control (especially Item B), the reviewer would observe that: If credit is to be taken for the gadolinium percentage in MOX fuel, the concentration of this isotope is measured, and all items affecting it should be classified as IROFS. If fixed poison rods are credited, then the poison loading and dimensions would be declared as IROFS and monitored (as in pickling operations where it could be leached away).

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.

As an example of implementing volume control (especially Item A, the reviewer would observe that: The volume of a container of plutonium nitrate is controlled to some fixed percentage (say, 75%) of the minimum critical volume, assuming spherical geometry, 0.3 m (12 inches) of water reflection, and optimal concentration.

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, the process variables are identified as IROFS in the CSEs and ISA summary. Methods of causing the material to become inhomogeneous are evaluated. 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. Computer calculations use the appropriate cell-weighting to ensure that resonance self-shielding is taken into account.

C. A physical measurement of the scale of heterogeneity is obtained based on the observed physical characteristics of the material, and the calculations shown to be conservative with respect to these measurements.

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.

As an example of implementing herogeneity control (Items A, B, and C), the reviewer would observe that: A motorized stirrer and acid molarity are credited in maintaining a highly concentrated solution homogeneous. These could be identified as IROFS and the system evaluated over the entire range of heterogeneity if it is determined that heterogeneity is a significant effect, and if it is desirable to credit. If the solution is completely homogeneous, the system may be safe because of homogeneity, and if the material has precipitated out, it may be safe because it is a safe slab. The intermediate configuration may not be bounded and would typically be evaluated separately. The validation report contains benchmarks that contain heterogeneous effects, to ensure that the bias is known when using resonance self-shielding.

6.4.3.3.2.12 Process Variables

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

A. Process variables 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.

As an example of implementing process variables, the reviewer may observe that: Process variables identified as IROFS, including temperature in an oxidation furnace that is credited in excluding moderator, force of a pellet press credited in controlling density, and the presence of radionuclides that do not affect the reactivity but which can bias the measurement of monitoring equipment.

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 Regulatory Guide 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 proposed 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.

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

H. 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.

I. The applicant evaluates the effect of credible shielding in demonstrating the adequacy of the dual alarms to detect a nuclear criticality.

J. 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 either has an emergency plan or satisfies the alternate requirements found in 70.22(h)(1)(i). (See SRP Chapter 14.0)

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

As an example of adequate CAAS coverage (Items A through I), the reviewer may observe a commitment to place alarms such that two detectors cover all fissile material processing and transfer areas, with two detectors required to actuate the evacuation signal to minimize accidental actuation. In addition the applicant would perform shielding calculations (using, for instance, the Monte Carlo Neutron Proton (MCNP) Code) to demonstrate the alarm coverage radius, taking conservative estimates of the intervening shielding and using housekeeping practices that minimize the presence of this shielding material. If the CAAS alarm is in an area where process noise would render the alarm inaudible, alternate measures such as flashing strobe lights may be proposed.

K. 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.

As an example of possible justifications for the exclusion of CAAS coverage (Item K), the reviewer may observe: (1) the applicant has materials whose physical form and isotopic characterization would require masses vastly in excess of what the applicant is authorized to possess to cause criticality; (2) the facility has been demonstrated to have adequate shielding to prevent any operator from receiving a dose in excess of 20 rad. In this case, measures would still be provided to alert operations to the fact of a criticality; (3) operators are excluded from processing areas by hostile conditions to a distance adequate to ensure safety; or (4) 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 Proposed 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 proposed 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. 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. In addition, the increased risk associated with the less conservative margin should be commensurate with and offset by the unlikelihood of achieving the condition to begin with.

As an example of Item B(i), the reviewer would review the justification for the use of a minimum subcritical (administrative) margin. For example, justification for a margin of 0.02 for abnormal conditions may reference, among other things, the unlikelihood of achieving the abnormal condition. The applicant would then rigorously define what is meant by an abnormal condition. A spill of SNM when open containers of powder are manually handled would not be considered an abnormal occurrence; spills outside the tube of a favorable geometry sintering furnace would probably be an abnormal occurrence, unless operating history showed otherwise. An applicant wishing to use a minimum subcritical margin of 0.02 instead of a more conservative value (such as 0.05, which is typically considered acceptable for most cases where there are a statistically significant number of benchmarks) for instance, for abnormal events may demonstrate that the increase in risk by reducing the conservatism in k_{eff} by the proposed amount is offset by the low likelihood of occurrence of the abnormal condition. This should take into consideration the estimated uncertainty in the bias.

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 perform 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. (The AOA for the method should be defined in the validation report; see Section 6.4.3.3.1).

As an example of Item E, the reviewer should ensure that: The applicant's validation report states the range of material chemical and physical forms, isotopic concentrations, moderation range, other materials assumed validated, neutron energy ranges, any code options and/or

statistics used, and any other pertinent information that defines the AOA. The applicant would then commit to evaluating each application to determine whether it falls into the code's AOA and documenting this in the evaluation. If this information is not unambiguously spelled out in the validation report, then an equivalent analysis may be conducted in each evaluation. In addition, those cases which are most relevant would be evaluated when determining the bias and AOA. For instance, plutonium oxide lattice cases would not be in the same AOA as high-enriched uranium fast metal cases and the bias should be determined for each AOA.

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

6.4.3.3.5 Requirements of Proposed 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 proposed 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 should be 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.

As an example of meeting the double contingency principle (Items A and B), the reviewer should ensure that: If dual geometry control for a highly concentrated plutonium nitrate system is ensured by having an inner and an outer containment, means are provided for detecting whether the solution has intruded into the outer containment. Otherwise, the inner containment could have been breached and remained breached for an extended period of time. In this case, double contingency is not met even though two non-simultaneous failures are required. 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 non-diverse 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. If any such single event exists that could cause criticality, it by itself must qualify as highly unlikely.

As an example of common cause effects, as discussed in the preceding paragraph, a process may depend on geometry and moderation control, or on dual moderation controls, but a major fire in the facility may defeat both by causing sprinkler activation and by bringing material together into a more reactive configuration. Such externally-initiated common mode failure scenarios would be evaluated in the CSE and flowed 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. Even if the consequences of criticality are mitigated such that they do not rise to the threshold of proposed 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.

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.

As an example of possible exceptions to the double contingency principle (Item C), the reviewer may evaluate a process that requires the processing of large quantities of plutonium or uranium oxide to achieve an acceptable throughput. In the case of low-plutonium mixed oxide powder, the only practicable method of ensuring subcriticality may be moderation control. Multiple controls on moderation in a large geometry hopper or fluidized bed reactor may be sufficient to ensure double contingency. Such a case may or may not constitute single-parameter double contingency; single contingency may be authorized based on a demonstration that it would require a single almost incredible event–like a catastrophic breach in containment due to an explosion–before criticality is possible. Another example may be a system that relies only on an unusually rigorous passive geometry control.

In a shielded facility, single contingency may be authorized based on reduced risk to personnel. Controls would still be in place, however, to ensure against criticality, and will be evaluated on a case-by-case basis, taking mitigation into account as one of the factors.

6.4.3.3.6 Requirements of Proposed 10 CFR 70.65 (ISA Summary)

The applicant is required to meet the performance criteria in proposed 10 CFR 70.61(b) and (c) as well as the performance requirements in proposed 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 proposed 10 CFR 70.61(b) and (c); however proposed 10 CFR 70.61(d) requires the applicant to use prevention methods as the primary means to meet the performance requirements of proposed 10 CFR 70.61(b) and (c).

To provide for NCS, the applicant's implementation of the ISA requirements in proposed 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 (ISA Summary) related to accident sequences for NCS.

   ii. The applicant commits to evaluate 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 (ISA Summary) 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.

As an example of Item B(ii): A criticality in a shielded facility may not be a high consequence event, but still requires an explicit exemption from the double contingency requirement. Further justification may include areas where personnel are excluded because of hostile operating conditions.

C. Likelihoods:

i. The applicant meets the acceptance criteria in Section 5.4 (ISA Summary) 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 which 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 proposed 10 CFR 70.61(b) and 70.64(a)(9).

As an example of what is meant in Item C(ii), 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. For instance, a fissile material pump that functions as an IROFS would be considered to be sufficiently reliable if it was periodically functionally tested, if the oil were changed and the proper lubrication used in accordance with the manufacturer's specifications, if the environment in which it were used was within the pump's operating parameters (such as temperature), if gaskets and bearings were replaced on a frequency dictated by the manufacturer's wear data, and if its criticality significant characteristics were verified upon installation and configuration controlled.

For a simple administrative control such as spacing in an array, failure would not be considered unlikely if the operator were required to place units within half an inch center-to-center, if there were no passive spacing devices and no painted guides on the floor. A simple administrative control on spacing would be acceptable as a reliable control with the proper training, postings, floor markings, and supervisor attention, but its failure may still not be sufficiently unlikely to be credited as a single leg of double contingency. If two such spacing violations could lead to criticality, then a more rigorous interaction control would probably be required. 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 (ISA Summary) related to risks for NCS.

E. IROFS:

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

6.4.3.3.7 Requirements of Proposed 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 proposed 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 the lifetime of the facility. 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 effect 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 must 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 proposed 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.

An example of changes that could be made without prior NRC approval would include certain changes to a uranium solution pump. The attributes important to this pump are the plenum volume (pump is safe volume) and diameter, volume of the oil reservoir (safe volume and limited moderator into process), and pump capacity (required to prevent overflowing downstream equipment). A change that would not require prior NRC approval would be changing the manufacturer or model number of the pump, provided that this did not affect any of the above attributes. Changing from a centrifugal to a positive displacement pump may not require additional changes, unless this introduced additional accident sequences or reduced the reliability of the pump or any of the pump's characteristics important to NCS. Replacement of a backflow preventer with a check valve would require prior approval because although it did not

change any controlled parameters or introduce any new accident sequences, it reduced the margin of safety by replacing a robust control with one which was more likely to fail.

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

The applicant's description of measures to implement the reporting requirements in proposed 10 CFR 70 Appendix A 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.

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 issue the necessary report based on whether the IROFS credited for double contingency were lost, irrespective of whether the safety limits of the associated criticality parameters were actually exceeded.

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

As an example of the foregoing criteria (Items A through D), the reviewer should consider the hypothetical case of an upset in an unfavorable geometry waste water tank, which typically relies solely on mass control for criticality safety. An unauthorized transfer to this tank would be reportable as a one-hour report, even if it were unknown whether an unsafe mass had been transferred (since mass is what is maintaining subcriticality). Even if it were later determined that an unsafe mass did not accumulate in the tank, this report should not be retracted. In this case, the mass control was lost even if the mass did not physically exceed a safe value, because the multiplicity of positive controls needed for double contingency was not maintained - thus, it would be reportable under Items C and D above.

As a second example, consider the case of a slab of molybdenum boats containing green MOX pellets, which are heat-treated in a sintering furnace. Typical criticality controls for this example would include the mass in each boat, the depth of pellets in the boat, and moderation. The depth of pellets is controlled (in this hypothetical example) by the boat's dimensions, although the boats are demonstrated to be adequately subcritical when filled up to the top with pellets at theoretical density. An applicant controlling mass in these boats may establish an operating limit lower than the actual capacity of the boats in order to ensure that the material is processed uniformly and to give the operators a certain amount of margin in filling the boats. The formal

subcritical limit established in criticality safety evaluations typically exceeds this operating limit by a substantial amount. When controls are established with such conservatism, it may take several events before criticality is possible, including exceeding the analyzed safe slab depth of the boat and adding moderator. Because several events are needed for criticality even after an upset occurs, merely exceeding the operating limit would not be reportable as an immediate report under Appendix A Paragraph (a)(5). Double contingency would not be lost in this hypothetical case. The filling of a boat to two grams more than that allowed in the operating limit would not be considered significant (since the boats had been shown adequately subcritical even when overfilled well beyond this) and would not require reporting. This of course requires a significance determination as to what constitutes a significant loss of mass control in the CSE.

However, the filling of a boat until the material mounded over the top would violate both geometry and mass control and would be a significant loss of the control, since the mass would exceed the pre-analyzed condition. The resultant condition, and that of exceeding the subcritical limit, would constitute a twenty-four hour reportable events since the IROFS failed to meet the performance requirements of 10 CFR 70.61(e). Overflowing the material out of the boat and onto the floor, however, if it were not analyzed and shown to be subcritical as an upset condition in the criticality safety evaluation, would be reportable as an unanalyzed condition.

The information to be submitted in these reports should include, to the extent known at the time of the event, the quantities and isotopics of the materials involved, their moderation levels, and any other pertinent information needed to assess their k_{eff}, the particular procedural failure that led to the event, and the condition of the remaining geometry and moderation controls to allow NRC to determine the actual and potential significance of the event.

6.5 REVIEW PROCEDURES

6.5.1 Acceptance Review

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

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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.

The specific areas of interest during the design phase are described below:

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

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

iii. The commitment to design and operate 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 design and operate the facility in accordance with the BDC (that is, commits to the double contingency principle).

   b. The applicant commits to install and maintain a CAAS for applicable areas of the facility, or includes an exemption request with the construction 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 Baseline Design Criteria by providing validation reports to support calculations of subcritical limits, and proposed margins of subcriticality.

B. License Application

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 (application for construction approval or license application), 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) (application for construction approval) or 6.5.1(B) (license application), 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 the acceptance criteria in SRP Section 6.4.

The primary reviewer should consult with the supporting reviewers to identify and resolve any issues of concern related to the application for construction approval or the licensing application. 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:

i. 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.

ii. In support of the primary reviewer for Sections 15.1 through 15.8, the NCS reviewer should determine whether the acceptance criteria in Sections 15.1 through 15.8 have been met as they relate to NCS.

iii. In support of the primary reviewer for Chapter 5.0, the NCS reviewer should determine whether the acceptance criteria in Chapter 5.0 have been met as they relate to NCS.

iv. In support of the primary reviewer for Chapter 14.0, the NCS reviewer should determine whether the acceptance criteria in Chapter 14.0 have been met as they relate to NCS.

v. 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 application for construction approval) as follows:

Nuclear Criticality Safety (NCS)

The staff reviewed the Nuclear Criticality Safety (NCS) measures described in the application for construction approval according to Chapter 5.0 of the NUREG-1718. The staff is satisfied that: (1) The applicant's commitments to establish 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, staff concludes that there is reasonable assurance that a facility designed in compliance with the aforementioned application for construction approval will be found acceptable without major re-engineering or re-design. Therefore, the applicant's NCS design basis meets the requirements 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 Nuclear Criticality Safety (NCS) program of the license application for the [insert name of facility] according to Chapter 6.0 of the 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 which 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 of 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 workers and the environment.

Note: The NCS safety evaluation for the ISA Summary requirements for proposed 10 CFR 70.65 should be included in the safety evaluation that supports Chapter 5.0 of this SRP.

6.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

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

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

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

F. DOE/NCT-04, A Review of Criticality Accidents, W. R. Stratton, Revised by D. R. Smith, U.S. Dept. of Energy, March 1989.

G. *Nuclear Criticality Safety–Theory and Practice*, R. A. Knief, American Nuclear Society, La Grange Park, IL, 1985.

H. DOE Order 420.1 (Change 2), *Facility Safety*, October 24, 1996.

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 which 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 which 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 < 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 particular control failure is observed, it should no longer be credited as highly unlikely for double contingency.

incredible: Having a probability of occurrence < 10^{-6}/year/event. Demonstration of incredibility will be considered adequate if the resulting conditions are: (1) prohibited by physical laws or not

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[1]These definitions are predicated on the assumption that there are approximately 1000 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.

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 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 or 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 < 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.

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[2]See footnote on previous page.

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" (proposed §70.64(a)(3)) and that is based on defense-in-depth practices (proposed §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 the proposed §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, 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-basis documents; and inspection, maintenance, and testing of fire protection features and systems.

C. Manual Firefighting Capability: A baseline needs assessment should establish the minimum required capabilities of site firefighting forces. This assessment should include minimum staffing, organization and coordination of on-site and off-site firefighting resources, personal protective and firefighting equipment, training, and prefire emergency planning.

D. Fire Hazard 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 C of this SRP.

7.4 ACCEPTANCE CRITERIA

7.4.1 Regulatory Requirements

Proposed §70.(64)(a) has a Baseline Design Criterion for "fire protection" and requirements regarding defense-in-depth practices. In addition, proposed §70.61 contains performance requirements for the facility. The proposed 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

Much of the guidance that follows in this SRP is similar to the NRC fire protection guidance and regulations promulgated for light water nuclear power plants. Section II, General Requirements of 10 CFR Part 50 Appendix R, specifically establishes the basis for this guidance. Appropriate details from the organization and operations related guidance in NUREG-0800 Section 9.5.1, "Fire Protection Program," have been incorporated in this SRP. The guidance in this SRP also adheres to applicable fire protection policies developed by the US Department of Energy (DOE) for its plants processing uranium, plutonium and mixed oxides. 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 important to 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 National Fire Protection Association (NFPA) codes that can provide information on standard practices that may be applied for MOX facilities in other areas of fire safety[1]. Significant guidance from DOE STD-1066, "Fire Protection Design Criteria," has also been incorporated into this SRP. Guidance in regard to accident analysis may be found in NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," 1998.

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[1] NFPA Standard 801, Standards for Facilities Handling Radioactive Material, provides additional overall guidance on fire protection for fuel cycle facilities and NFPA Standard 803, Standard for Fire Protection for Light Water Nuclear Power Plants provides additional criteria that should be considered for MOX facilities.

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.

Where specific NFPA or other standards are referenced, it is the intent of the SRP 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. It is expected that the applicant will select and reference the most applicable standards for all known hazards and fire protection measures at its facility in its license application beyond those identified in this SRP Chapter.

7.4.3.1 Organization and Conduct of Operations

The following organizational and operational 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:

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 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 items relied on for safety, 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 items relied on for safety 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. Formulation and assurance of program implementation may be delegated to a staff composed of personnel prepared by training and experience in fire protection and personnel prepared by training and experience in 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 quality assurance of all fire protection features (e.g., detection systems, suppression systems, barriers, dampers, doors, penetrations 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 off-site or on-site management position that has management responsibility for the formulation, implementation, and assessment of the effectiveness of the MOX facility fire protection program;

b. The off-site or on-site 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 on-site management position responsible for the overall administration of the plant operations and emergency plans which include the fire protection and prevention program and which 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 safety-related areas; determine the effectiveness of housekeeping practices; assure the availability and acceptable condition of all fire protection systems/equipment, fire stops, penetration seals, and fire retardant coatings (if any); and assures the 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 evaluate test results and determine 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 which 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 on-site position responsible for fire protection quality assurance. 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 of audits are promptly reported to cognizant management personnel.

f. The positions which are part of the plant 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 and experience for such work.

(4) The personnel responsible for the training of 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 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. Prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance periods.

ii. Govern the handling and limitation of the use of ordinary combustible materials, combustible and flammable gases and liquids, high efficiency particulate air (HEPA) and charcoal filters, dry ion exchange resins, or other combustible supplies in safety-related areas.

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 items relied on for safety during all phases of operation, and especially during maintenance or modification operations. Use of wood products is permitted only when noncombustible products are not practicable 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 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 from the area of all waste, debris, scrap, oil spills, or other combustibles resulting from the work activity immediately following completion of the activity, or at the end of each work shift, whichever comes first.

vi. Govern leak testing (air movement in process lines and process buildings to control migration of radioactive materials) to use of commercially available techniques for procedures such as airflow determination. Open flames or combustion-generated smoke should not be permitted.

vii. Maintain periodic housekeeping inspections to ensure continued compliance with these administrative controls.

viii. Disarming of fire detection or fire suppression systems is controlled by a permit system. Fire watches should be established in areas where systems are so disarmed.

ix. Successful fire protection requires testing and maintenance of the fire protection equipment and the emergency lighting and communication. A test plan that lists the individuals and their responsibilities 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, e.g., 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 the control room 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 and the fire brigade leader of the type, size, and location of the fire.

xii. Define the strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. 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 zone covered by the specific fire fighting procedure that could increase the hazards in the area because of over pressurization 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 command control of the brigade, transporting fire suppression and support equipment to the fire scenes, applying the extinguishant to the fire, communication with the control room, and coordination with outside fire departments.

g. Potential radiological and toxic hazards in fire zones.

h. Operations requiring control room and shift supervisor coordination or authorization.

i. Instructions for plant operators and general plant personnel during fire.

xiii. Establish and implement a penetration seal tracking program to record pertinent information regarding the emplacement and modification of fire barrier penetration seals which are defined in the ISA Summary or FHA as items relied on for safety.

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 items relied on for safety are designed to qualify as Type I construction as defined by NFPA Standard 220. 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 practicable, 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 80A, "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 non-combustible 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 system.

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 (FM) -approved flame spread rating of 25 or less and a smoke developed rating of 50 or less, per ASTM E-84.

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 253 (ASTM-648) when applying the floor finish requirements of The Life Safety Code (NFPA 101) to MOX facilities. Carpets and rugs used in storage or industrial occupancies (no criteria in NFPA 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 occurs in the packaging materials.

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). Cable trays classified as relied on for safety or which may contribute a significant fire load are protected from fire in accordance with IEEE Standard 690.

H. Lightning protection for plant buildings determined to be items relied on for safety is designed in accordance with the applicable provisions of NFPA Standard 780.

I. The ventilation systems in areas containing items relied on for safety 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 Standards 69 and 90A. Where ventilation systems are designed to prevent the release of radioactive materials, all materials of construction including HEPA filters are of the fire-resistant type in accordance with the applicable provisions of UL Standard 586. Further fire protection guidance for nuclear filter plenums is contained in Appendix D 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) is 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 fire fighter 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 required to perform any safety related functions 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. Fire doors are designed and installed in accordance with the applicable provisions of NFPA Standard 80. Fire dampers are designed and installed in accordance with the applicable provisions of UL Standard 555.

M. Plant areas where a potential for large spills of flammable or combustible liquids exist are identified and means of containing, e.g., 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 30 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 50, 50A, 50B, 51, 55, 58, 69, 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 seismic Category I requirements, 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. Shut off valves are installed as close as possible to the reducing furnaces, or other using devices, but the shut off valves are so located 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 so located that the possibility and consequences of a catastrophic spill is 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 shut off 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 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 a 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. 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 off-site 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 seismic Category I Class 1E electric motor-driven fire pumps connected to Class 1E emergency power buses. 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 so arranged that a leak or other malfunction will not be able to drain off the water reserved for the fire service needs. The design of the distribution piping, valves, and fire hydrants are in accordance with the applicable provisions of NFPA Standard 24. 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 14.

R. Automatic fire suppression is incorporated in areas of significant fire loading or the potential for significant loading to protected areas determined to be items relied on for safety. 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 10, 11, 11A, 12, 13, 15, 16, 16A and 2001. 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, the fire suppression system is capable of delivering water to manual hose stations located within hose reach of areas containing items relied on for safety following the most severe earthquake expected in the geological area where the facility is located. In areas of high seismic activity, the staff will consider on a case-by-case basis the need to design the fire detection and suppression systems to be functional following such an expected most severe earthquake.

S. The applicant commits to provide 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.

T. Safety controls and interlocks for combustible liquids, flammable liquids and flammable gases and their associated delivery system 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-5XXX-99, "Stabilization, Packaging, and Storage of Plutonium-Bearing Materials."

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 non-combustible 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 in order to prevent gloves from falling or being blown off. If an inerting system is used, 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 shut down 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. The separation/isolation 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.

Z. Where flammable or combustible solvents are used, they are handled in a system that does not allow uncontrolled release of vapors. 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.

BB. Incinerators, boilers, and furnaces are located in separate fire areas with automatic suppression and installed and maintained in accordance with NFPA 54, 31, 8501 and/or other applicable industry standards.

CC. Facility laboratories using chemicals or nuclear materials are operated in accordance with the safety criteria in NFPA 45 and/or NFPA 801 as applicable.

DD. Provisions for the drainage and holdup of contaminated fire water following a fire is incorporated into the design.

7.4.3.3 Manual Firefighting 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 firefighting capability should be acceptable if:

A. The recommendations for organization, training, and equipment of Standard on Industrial Fire Brigades as specified in NFPA 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 items relied on for safety. 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 600 for the type of duties to be performed. The shift supervisor is not a member of the fire brigade. The brigade leader is competent to assess the potential safety consequences of a fire and 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 (NIOSH) 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 on-site 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. The 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. Training of the plant fire brigade should be coordinated with the local 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 local fire department staff as appropriate.

l. Local 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, is 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 is begun. 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. The 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 local 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 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 practicable.

(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 organize for the fire, assuming loss of automatic suppression capability.

(4) Assessment of 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 Hazard Analysis (FHA)

The FHA should be considered acceptable if it reflects current conditions throughout the facility and the applicant commits to review and update 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 C 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 (construction or license) adequately addresses the items in Section 8.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the application for construction approval 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 Application

Specifically, the license application 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 in the application for construction approval or the license application, 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) (application for construction approval) or Section 7.5.1(B) (license application), 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 the acceptance criteria in SRP Section 7.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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 system, 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 Application

The primary reviewer should focus on Section 7.4.3.1, "Organization and Conduct of Operations," Section 7.4.3.3, "Manual Firefighting Capability," and Section 7.4.3.4, "Fire Hazard Analysis (FHA)," 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 which address fire protection to insure 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 is 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 which 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 sections of the SRP 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 application for construction approval as follows:

The staff reviewed the application 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 application 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 firefighting capability.

The staff concluded that the applicant's proposed equipment, facilities, and commitments provide a reasonable level of assurance that applicant's design basis 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 license application as follows:

The staff reviewed the license application 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 which 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 firefighting 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 proposed 10 CFR Part 70.

7.7 REFERENCES

A. Factory Mutual Research Corporation, *Factory Mutual System Approval Guide-Equipment, Materials, Services, and Conservation of Property.*

B. IEEE Standard 690, *IEEE Standard for the Design and Installation of Cable Systems for Class 1E Circuits in Nuclear Power Generating Stations*, Institute of Electrical and Electronics Engineers, Inc.

C. NFPA Standard 10, *Standard for Portable Fire Extinguishers*, National Fire Protection Association, Inc.

D. NFPA Standard 11, *Standard for Low Expansion Foam*, National Fire Protection Association, Inc.

E. NFPA Standard 11A, *Standard for Medium- and High-Expansion Foam Systems*, National Fire Protection Association, Inc.

F. NFPA Standard 12, *Standard on Carbon Dioxide Extinguishing Systems*, National Fire Protection Association, Inc.

G. NFPA Standard 13, *Standard for the Installation of Sprinkler Systems*, National Fire Protection Association, Inc.

H. NFPA Standard 14, *Standard for the Installation of Standpipes and Hose Systems*, National Fire Protection Association, Inc.

I. NFPA Standard 15, *Standard for Water Spray Fixed Systems for Fire Protection*, National Fire Protection Association, Inc.

J. NFPA Standard 16, *Standard for the Installation of Deluge Foam-Water Sprinkler and Foam-Water Spray Systems*, National Fire Protection Association, Inc.

K. NFPA Standard 16A, *Standard for the Installation of Closed-Head Foam Water Sprinkler Systems*, National Fire Protection Association, Inc.

L. NFPA Standard 20, *Standard for the Installation of Centrifugal Fire Pumps*, National Fire Protection Association, Inc.

M. NFPA Standard 24, *Standard for the Installation of Private Service Mains and their Appurtenances*, National Fire Protection Association, Inc.

N. NFPA Standard 25, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*, National Fire Protection Association, Inc.

O. NFPA Standard 30, *Flammable and Combustible Liquids Code*, National Fire Protection Association, Inc.

P. NFPA Standard 31, *Standards for Installation of Oil Burning Equipment*, National Fire Protection Association, Inc.

Q. NFPA Standard 45, *Standard for Fire Protection for Laboratories Using Chemicals*, National Fire Protection Association, Inc.

R. NFPA Standard 50, *Standard for Bulk Oxygen Systems at Consumer Sites*, National Fire Protection Association, Inc.

S. NFPA Standard 50A, *Standard for Gaseous Hydrogen Systems at Consumer Sites*, National Fire Protection Association, Inc.

T. NFPA Standard 50B, *Standard for Liquified Hydrogen Systems at Consumer Sites*, National Fire Protection Association, Inc.

U. NFPA Standard 51, *Standard for Oxygen-Fuel Gas Systems for Welding, Cutting, and Allied Processes*, National Fire Protection Association, Inc.

V. NFPA Standard 54, *National Fuel Gas Code*, National Fire Protection Association, Inc.

W. NFPA Standard 55, *Standard for Compressed and Liquified Gases in Portable Cylinders*, National Fire Protection Association, Inc.

X. NFPA Standard 58, *Standard for Storage and Handling of Liquified Petroleum Gases*, National Fire Protection Association, Inc.

Y. NFPA Standard 69, *Standard on Explosion Prevention Systems*, National Fire Protection Association, Inc.

Z. NFPA Standard 70, *National Electric Code*, National Fire Protection Association, Inc.

AA. NFPA Standard 72, *National Fire Alarm Code*, National Fire Protection Association, Inc.

BB. NFPA Standard 80, *Standard for Fire Doors and Fire Windows*, National Fire Protection Association, Inc.

CC. NFPA Standard 80A, *Protection of Buildings from Exterior Fire Exposures*, National Fire Protection Association, Inc.

DD. NFPA Standard 90A, *Standard for the Installation of Air Conditioning and Ventilating Systems*, National Fire Protection Association, Inc.

EE. NFPA Standard 101, *Life Safety Code*, National Fire Protection Association, Inc.

FF. NFPA Standard 220, *Standard on Types of Building Construction*, National Fire Protection Association, Inc.

GG. NFPA Standard 600, *Standard on Industrial Fire Brigades*, National Fire Protection Association, Inc.

HH. NFPA Standard 780, *Lightning Protection Code*, National Fire Protection Association, Inc.

II. NFPA Standard 801, *Standards for Facilities Handling Radioactive Material*, National Fire Protection Association, Inc.

JJ. NFPA Standard 2001, *Standard on Clean Agent Extinguishing Systems*, National Fire Protection Association, Inc.

KK. NFPA Standard 8501, *Standard for Single Burner Oil Operation*, National Fire Protection Association, Inc.

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

MM. Underwriters Laboratories, Inc., *Underwriters Laboratories Building Materials Directory*.

NN. Underwriters Laboratories, Inc., *Underwriters Laboratories Fire Protection Equipment Directory*.

OO. Underwriters Laboratories Standard 555, *Standard for Fire Dampers and Ceiling Dampers*, Underwriters Laboratories, Inc.

PP. Underwriters Laboratories Standard 586, *High Efficiency Air Filtration Units*, Underwriters Laboratories, Inc.

QQ. U.S. Department of Energy, Fire Protection Design Criteria, DOE-STD-1066-97, March 1997.

RR. U.S. Department of Energy, Stabilization, Packaging, and Storage of Plutonium-Bearing Materials, Draft DOE-STD-5XXX-99, March 1999.

7.8 DEFINITIONS

combustible: A material, in the form and condition in which it is used, that will ignite and burn.

combustible liquid[2]: A liquid having a flash point at or above 100 °F (37.8 °C).

fire area: A location bounded by fire-rated construction, having a minimum fire resistance rating of 2 hours.

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 251, "Standard Methods of Fire Tests of Building Construction and Materials."

flammable liquid[3]: 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).

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[2] Definitions as used in NFPA Fire Protection Handbook and NFPA Standards

flammable gas[3]: A gas that will burn in the normal concentration of oxygen in the air.

gas[3]: Any substance that in a liquid state exerts a vapor pressure greater than 40 psia at 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 or 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 E136, "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 130 °F (54.4 °C) 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 the proposed 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 practicable, 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 at the facility (SRP Chapter 5.0). Chemical safety addresses the consequences of potential accidents involving licensed materials from hazardous chemicals and accidents due to chemicals that create potentially hazardous situations (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 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 proposed 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 53

(No. 210), 43950-43951, October 31,1998, the NRC oversees chemical safety issues related to (i) radiation risk produced by radioactive materials; (ii) chemical risk produced by radioactive materials; and (iii) plant conditions which affect the safety of radioactive materials and thus present an increased radiation risk to workers. The NRC does not oversee facility conditions which 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 of the 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 the proposed 10 CFR Part 70. The following sections are particularly relevant to chemical safety: safety performance requirements (proposed §70.61), safety program and ISA (proposed §70.62), and the baseline design criteria for new facilities or new processes at existing facilities (proposed §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

Listed in this section are the applicable portions of the NRC Inspection Manual and NUREGs that provide a basis that is generally acceptable to the NRC staff for satisfying the regulatory requirements listed in Section 8.4.1.

A. Nuclear Regulatory Commission (U.S.) (NRC). NRC Inspection Manual, Chapter 2603, "Inspection of the Nuclear Process Safety Program at Fuel Cycle Facilities." NRC: Washington, D.C.

B. U.S. NRC. NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook." NRC: Washington, D.C. 1998.

C. U.S. NRC. NUREG-1513, "Integrated Safety Analysis Document." NRC: Washington, D.C.

D. U.S. NRC. NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities." NRC: Washington, D.C. 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 (SRP Section 1.1) and the ISA Summary (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 MOX 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 be important to chemical process safety. 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 important to safety (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 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 potential initiators 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 that 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*, 1998. 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 (AEGL) or Emergency Response Planning Guide (ERPG) values or other cited values, such as those values developed by OSHA or NIOSH (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 proposed 10 CFR 70.61 are based upon acute chemical exposures, and, as such, 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 (SRP Chapter 5.0) and Environmental Protection (SRP Chapter 10.0) chapters and meet the requirements for proposed 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 focus 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 proposed 10 CFR 70.61.

8.4.3.5 Process Safety Information

Process safety information should be acceptable if it contains the following 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 as determined by the ISA.

D. Items relied on for safety are identified for those accident sequences that contain a chemical/process failure that may lead to radiological consequences that exceed the performance requirements of the proposed 10 CFR 70.61.

E. The applicant uses a graded approach to safety in accordance with proposed 10 CFR 70.62(a). The applicant ensures that the grading of items relied on for safety 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 such items as actuators, pumps, and eductors.

F. The application describes the management measures that assure the availability and reliability of items relied on for safety 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. 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 (construction or license) adequately addresses the items in Section 8.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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 license application in lieu of the actual material.

B. License Application for Operations

Specifically, the safety assessment includes 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 or license), 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) or 8.5.1.B (license), 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 the acceptance criteria in SRP Section 8.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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 chemical safety, meet or exceed the regulatory acceptance criteria in Section 8.4.

B. License Application for Operations

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 construction or operations), 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 application for construction approval as follows:

The staff reviewed the 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 for construction approval, 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 basis and safety assessment of the design basis 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 basis 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 proposed 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] 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 proposed 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 proposed 10 CFR Part 70.

8.7 REFERENCES

A. Chemical Manufacturers Association, *Responsible Care®, Process Safety Code of Management Practices.* Washington, D.C. 1990.

B. American Institute of Chemical Engineers (AIChE). Center for Chemical Process Safety, *Guidelines for the Technical Management of Chemical Process Safety.* AIChE: New York, 1989, Chapter 11, as revised.

C. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

D. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

E. Code of Federal Regulations, *Title 29 Labor*, Part 1900, §1910.119, "Process Safety Management of Highly Hazardous Chemicals."

F. U.S. NRC. NRC Inspection Manual, Chapter 2603, "Inspection of the Nuclear Process Safety Program at Fuel Cycle Facilities." NRC: Washington, D.C.

G. 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 53 (No. 210), 43950-43951, October 31, 1988.

H. U.S. NRC. NUREG/CR-6410, "Nuclear Fuel Cycle Accident Analysis Handbook." NRC: Washington, D.C. 1998.

I. U.S. NRC. NUREG-1601, "Chemical Process Safety at Fuel Cycle Facilities." NRC: Washington, D.C. 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 non-routine 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 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 for 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 is reasonably achievable (ALARA; see Items A-E). The applicant is also required to establish controls and management measures to meet the performance requirements established in proposed 10 CFR 70.61 (see Item F). The 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. The 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, in-service 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 non-routine operations (e.g., maintenance) including anticipated events; and

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 C);
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. Integrated Safety Analysis (ISA)

i. Postulated accident types of accident sequences in the ISA which have radiation safety consequences for workers, including all high 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. 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, the applicant describes the training program and postings for these individuals as required under proposed 10 CFR 70.61(f)(2) (Training and postings may be cross-referenced with Sections 9.2).

iii. The methodology in assessing the accident consequences. In particular, the primary reviewers of this SRP section should focus on the ISA source terms (see Item B), transport, and dosimetry analyses;

iv. The items relied on for safety, and associated management measures, to prevent or mitigate each accident sequence that results in radiological consequences in excess of the performance requirements of proposed 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 NRC through a regulation or RG. Further, inclusion in this section is not necessarily an endorsement of a particular standard by 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
Proposed 10 CFR 70.64	Requirements for New Facilities or New Process at Existing Facilities

9.1.4.1.2 Regulatory Guidance

RG 8.10, Rev. 1-R Sept 1975	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable

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 to ensure:

   i. Measures for reducing the need for time spent in radiation areas;

ii. Measures to improve the accessibility to components requiring periodic maintenance or inservice inspection;

iii. Measures to reduce the distribution and retention of radioactive materials throughout plant systems;

iv. Measures to control (reduce) contamination, facilitate decommissioning, and minimize secondary radioactive waste production in accordance with 10 CFR 20.1406;

v. Measures instructing designers and engineers in ALARA design objectives;

vi. Measures incorporating experience from operating plants and past designs; and

vii. A commitment to, and description of, 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 radiation protection related design improvements are sought, considered, and incorporated where practicable (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
Proposed 10 CFR 70.61(f)	Performance Requirements for New Facilities
Proposed 10 CFR 70.64(b)	Requirements for New Facilities or New Process at Existing Facilities

9.1.4.2.2 Regulatory Guidance

RG 3.29, May, 1975	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

9.1.4.2.3 Regulatory Acceptance Criteria

The 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 non-routine operations (including anticipated events); and

   ii. Items relied on for safety 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.)

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 non-routine 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 9.2.5.8, "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
Proposed 10 CFR 70.64	Requirements for New Facilities or New Process at Existing Facilities

9.1.4.3.2 Regulatory Guidance

RG 8.10, Rev. 1-R Sept 1975	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable

9.1.4.3.3 Regulatory Acceptance Criteria

The 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 non-routine 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 non-routine 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.

9.1.3.4 Ventilation Systems and Glovebox Design

9.1.4.9.1 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
Proposed 10 CFR 70.64	Requirements for New Facilities or New Process at Existing Facilities

9.1.4.4.2 Regulatory Guidance

ANSI/ASME N510-1980 (1989)	Testing of Nuclear Air Cleaning Systems
ERDA 76-21	Nuclear Air Cleaning Handbook, C. A. Burchsted, A. B. Fuller, J. E. Kahn

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 other functions 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.

The 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 applicant demonstrates that the design and operation of the ventilation system and/or gloveboxes protects workers and public from airborne radioactive material such that limits of 10 CFR Part 20 will not be exceeded during routine and non-routine operations and anticipated events. Recommendations for the design, construction, and testing of nuclear air cleaning systems (e.g., zoning, moisture separation, HEPA filtration, operational/maintenance considerations, etc.) that are generally acceptable to NRC staff are provided in Energy Research and Development Administration (ERDA) 76-21 (see also Section 11.4.5).

B. The applicant commits to design objectives for ventilation systems and gloveboxes that ensure that:

   i. During routine and non-routine 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, that are required to function during a loss of power (in addition to performing their specified functions) are provided with an uninterruptable power supply, unless they can tolerate a temporary loss of function without loss of data, and are provided with a stand-by 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 20.110(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
Proposed 10 CFR 70.64	Requirements for New Facilities or New Process at Existing Facilities

9.1.4.5.2 Regulatory Guidance

ANSI/ANS 6.4.2-1985	Specification of Radiation Shielding Materials

9.1.4.5.3 Regulatory Acceptance Criteria

The 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 non-routine operations and anticipated events. The applicant identifies and describes any areas that facilitate installation and removal of temporary shields for non-routine 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 non-occupational (general employee). An objective, for design purposes, of 20 percent 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 percent of the occupational dose limit of 5 rem in a 2000 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 non-penetrating 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 and cross sections of ANSI/ANS-6.1.2 (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 the decision between permanent or 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.

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

Proposed 10 CFR 70.61	Performance Requirements
Proposed 10 CFR 70.62	Safety Program and Integrated Safety Analysis
Proposed 10 CFR 70.64	Requirements for New Facilities or New Process at Existing Facilities

9.1.4.6.2 Regulatory Guidance

NUREG-1513 (DRAFT 1998) *Integrated Safety Analysis Guidance Document*

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 cause the consequences of concern described in proposed 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 resulting from licensed material or hazardous chemicals resulting from licensed material to the workers and 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 of concern for workers.

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 ensure the continued availability and reliability of safety controls to prevent and mitigate radiological consequences of concern 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 for construction approval or the license application adequately addresses the items in Section 9.1.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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 Application

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 in the application for construction approval or the license application, 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) (application for construction approval) or 9.1.5.1(B) (license application), the primary reviewer should perform a safety evaluation against the acceptance criteria described in

Section 9.4. 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 9.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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 reviewer of SRP Chapter 7.0, "Fire Protection," and the primary reviewer of 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 Application

While this section addresses the applicant's radiation safety *design*, the applicant's radiation protection *program* and management measures are reviewed under SRP Chapter 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, can not be cleanly categorized into either "design" or the "radiation protection program." Review of these areas should be coordinated with the reviewer of SRP Section 9.2, "Radiation Protection Program." 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 Analyses," (which requires additional process description information through proposed 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 the ventilation, gloveboxes, and air cleaning systems are not in

conflict with radiation protection and with the primary reviewer of SRP Chapter 11.0, "Plant Systems," for the non-radiation 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 application for construction approval as follows:

The staff reviewed the application for construction approval for [insert facility name] to 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 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 license application as follows:

The staff reviewed the license application for [insert facility name] to possess and use 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 applicants' 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 safety program of SRP Section 9.1, satisfy the requirements of 10 CFR Parts 20 and 70.

9.1.7 REFERENCES

All documents referenced in the acceptance criteria for this review area have been listed in Sections 9.1.4.1-9.1.4.6 and are not repeated here. However, in addition to those documents, the following references contain information that is specific to nuclear reactors (or other nuclear facilities), but which is also relevant to this review area. Applicants may choose to reference

portions of these documents in either the application for construction approval or the license application, with adequate justification.

A. Regulatory Guide 1.33, Rev. 2, *Quality Assurance Program Requirements (Operational)*, U.S. Nuclear Regulatory Commission, February 1978.

B. Regulatory Guide 8.8, Rev. 3, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable*, U.S. Nuclear Regulatory Commission, June 1978.

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 Integrated Safety Analysis (ISA) Summary are reviewed under SRP Chapter 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). The areas of review should include:

A. As Low As Reasonably Achievable (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 the 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 that are 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 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 NRC through a regulation or RG. Further, inclusion in this SRP is not necessarily an endorsement of a particular standard by 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 Is 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

RG 8.10, Rev.1-R, May 1977	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable

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 ensures:

i. That 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. That the management clearly defined the responsibilities of individuals to implement the ALARA policy.

iii. That the radiation safety manager will have the appropriate authority and independence to prevent unsafe practices.

iv. The qualification and appropriate staffing of the radiation safety organization, commensurate with size and complexity of the radiation protection program.

v. That workers and management will be held accountable for their radiological work performance through a review process or other similar method.

vi. That procedures and engineering controls will include formal plans and measures for applying the ALARA process to occupational exposures.

vii. That modifications to procedures, facilities, and equipment will be justified with respect to optimization of ALARA.

viii. That actions taken to maintain occupational exposures ALARA are documented as part of the radiation protection program.

ix. That performance reviews of ALARA actions are included as part of the radiation protection program review.

x. That individuals likely to receive an occupational dose in excess of 100 mrem (1 mSv) in a year (per 10 CFR 19.12) are 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 towards 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 pre-planning 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 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 effect 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 track 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

RG 8.10, Rev.1-R, May 1977	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable

9.2.4.2.3 Regulatory Acceptance Criteria

The 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, non-routine 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 off-site 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 experience in health physics with at least one 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 five years of experience may be substituted in lieu of a bachelor's degree.

9.2.4.3 Radiation Protection Procedures and Radiation Work Permits (RWPs)

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

RG 8.10, Rev.1-R, May 1977	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable

9.2.4.3.3 Regulatory Acceptance Criteria

The 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 perform 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., continuous air monitor [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., mock-up required, special shielding required);

vii. Hold points or monitoring points, if applicable;

viii. Personnel protective equipment (PPE) requirements;

ix. Authorization signature and date;

x. Actual doses, time, or other information resulting from the completed work authorization are recorded on the RWP [RG 8.10 C.2(a)];

xi. Expiration/termination date of the RWP; and

xii. The information on RWPs is sufficient 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 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 RWPs are not used past their termination dates. The system includes what types of records are 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. The 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 graded 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

RG 8.10, Rev.1-R, May 1977	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable
RG 8.29, Rev.1, Feb. 1996	Instructions Concerning the Risks from Occupational Radiation Exposure
NUREG-0041, Oct. 1976	Manual of Respiratory Protection Against Airborne Radioactive Materials
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

9.2.4.4.3 Regulatory Acceptance Criteria

The 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; or

B. 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).

C. 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.

D. 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.

E. To verify the radiation protection training received, the applicant commits to have 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

RG 8.25, Rev. 1, June 1992	Air Sampling in the Workplace
RG 8.36, July 1992	Radiation Doses to the Embryo/Fetus
NUREG-0041, Oct. 1976	Manual of Respiratory Protection Against Airborne Radioactive Materials
NUREG-1400, Sept. 1993	Air Sampling in the Workplace

ANSI N42.17B-1989	Performance Specifications for Health Physics Instrumentation--Occupational Airborne Radioactivity Monitoring Instrumentation

9.2.4.5.2 Regulatory Acceptance Criteria

The 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 an air sampling program that is consistent with the positions in RG 8.25, including evaluating the need for air sampling, locating samplers, sample representativeness, 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 demonstrates that its action levels for airborne activity use appropriate technical criteria to determine the necessary controls, where the demonstration includes 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

RG 8.24, Rev.1, Oct. 1979	Health Physics Surveys During Enriched Uranium 235 Processing and Fuel Fabrication
BTP, April 1993	License Condition for Leak Testing Sealed Byproduct Material Sources
BTP, April 1993	License Condition for Leak Testing Sealed Plutonium Sources
BTP, April 1993	License Condition for Plutonium Alpha Sources
BTP, April 1993	License Condition for Leak Testing a Sealed Source which Contains Alpha and/or Beta-Gamma Emitters
BTP, April 1993	License Condition for Leak Testing Sealed Uranium Sources
April 1993	Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, or Special Nuclear Material
ANSI N323-1978 r.1983	Radiation Protection Instrumentation Tests and Calibrations
ANSI N542-1977	Sealed Radioactive Sources Classification

9.2.4.6.3 Regulatory Acceptance Criteria

The 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 clearly stated. 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 practicable, 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 (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 is 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 5 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 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

RG 8.4, Feb. 1973	Direct and Indirect-Reading Pocket Dosimeters
RG 8.7, Rev. 1, June 1992	Instructions for Recording and Reporting Occupational Radiation Exposure Data
RG 8.28, Aug. 1981	Audible Alarm Dosimeters
RG 8.34, July 1992	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses
RG 8.35, June 1992	Planned Special Exposures
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

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 non-occupationally 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, 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 off-site 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

RG 8.7, Rev. 1, June 1992	Instructions for Recording and Reporting Occupational Radiation Exposure Data
RG 8.9, Rev. 1, July 1993	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program

RG 8.13	Instructions Concerning Prenatal Radiation Exposures (Draft DG-8014, Proposed Rev. 3, Oct. 1994)
RG 8.25, Rev. 1, June 1992	Air Sampling in the Workplace
RG 8.34, July 1992	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses
RG 8.35, June 1992	Planned Special Exposures
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

9.2.4.8.3 Regulatory Acceptance Criteria

The 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/HPS 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 use 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

RG 8.7, Rev. 1, June 1992	Instructions for Recording and Reporting Occupational Radiation Exposure Data
RG 8.34, July 1992	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses
RG 8.35, June 1992	Planned Special Exposures
ANSI N13.6-1966 r.1989	Practice for Occupational Radiation Exposure Records Systems

9.2.4.9.3 Regulatory Acceptance Criteria

The 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 sum internal and external dose 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

RG 8.15, Oct. 1976	Acceptable Programs for Respiratory Protection

NUREG-0041, Oct. 1976	Manual of Respiratory Protection Against Airborne Radioactive Materials
ANSI Z88.2-1992	Practices for Respiratory Protection
ANSI Z88.6-1984	Physical Qualifications for Respirator Use

9.2.4.10.3 Regulatory Acceptance Criteria

The 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, 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 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

RG 8.28, Aug. 1981	Audible Alarm Dosimeters
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

9.2.4.11.3 Regulatory Acceptance Criteria

The 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, ANSI N42.17B, and ANSI N323.

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 calibrate all instruments at least semi-annually and to recalibrate 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.

9.2.5 REVIEW PROCEDURES

9.2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application (construction or license) adequately addresses the items in Section 9.2.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

   The applicant is not expected to address the radiation protection program in detail with the application for 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 Application

   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 (construction or license), the primary reviewer should accept the application for the safety evaluation in Section 9.2.2. 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 the acceptance criteria in SRP Section 9.2.4.

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 application for construction approval and the license application is provided below.

A. Application for 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 Application

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 2.0) and Training and Qualifications (SRP Section 15.4). Applicants may choose to provide the information in this section explicitly or by providing a reference to those chapters. The primary reviewer of this section coordinates with the primary reviewers of the other chapters to verify the completeness and consistency of the information 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 Safety Evaluation Report (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 application for construction approval as follows:

The staff reviewed the 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 proposed 10 CFR Part 70.

The staff could document the safety evaluation for the license application 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 safety 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

All referenced documents in the acceptance criteria for this review area have been listed in Sections 9.2.4.1-9.2.4.12 and are not repeated here. However, in addition to those documents, the following documents contain information that is specific to nuclear reactors, but which is also relevant to this review area. Applicants may choose to reference portions of these documents in the application, with adequate justification.

A. Regulatory Guide 1.33, Rev. 2, *Quality Assurance Program Requirements (Operational)*, U.S. Nuclear Regulatory Commission, February 1978.

B. Regulatory Guide 8.8, Rev. 3, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable*, U.S. Nuclear Regulatory Commission, June 1978.

C. RG 1.97, Rev. 3, May 1983, *Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident*, U.S. Nuclear Regulatory Commission, May 1983.

10.0 ENVIRONMENTAL PROTECTION

10.1 PURPOSE OF REVIEW

The purpose of this review is to determine whether an applicant for the construction and operation of a mixed oxide fuel fabrication facility has established environmental protection measures that are adequate to protect public health and the environment and comply with the regulatory requirements imposed by the Commission in 10 CFR Parts 20, 51, and 70.

In addition, pursuant to 10 CFR Part 51, the NRC will determine if the applicant submits an environmental report that is adequate for NRC use in preparation of an Environmental Impact Statement (EIS) for construction approval for the mixed oxide (MOX) fuel fabrication facility and an EIS for licensing the MOX fuel fabrication facility. This determination will be coordinated through the Division of Waste Management (DWM) since on May 17, 1999, the Office of Nuclear Material Safety and Safeguards (NMSS) assigned DWM the responsibility to prepare each NMSS EIS. As a result, guidance for reviewing an environmental report used to prepare an EIS is not provided in this chapter.

The Division of Fuel Cycle Safety and Safeguards (FCSS) currently retains the responsibility for determining if, pursuant to 10 CFR Part 51, an environmental report is adequate to support a licensing action that will result in the preparation of an Environmental Assessment (EA) and a Finding of No Significant Impact (FONSI). However, this type of licensing action is not anticipated until after the MOX fuel fabrication facility is licensed to possess and use special nuclear material (SNM). Staff should contact DWM for coordination and guidance and refer to the supplementary guidance in Appendix E to this SRP.

10.2 RESPONSIBILITY FOR REVIEW

Primary: Environmental Engineer/Scientist

Secondary: Project Manager

Supporting: Health Physicist Reviewer
Chemical Safety Reviewer
Primary Reviewer of SRP Chapter 15
Environmental Protection Inspector

10.3 AREAS OF REVIEW

The regulatory requirements for environmental protection are contained in 10 CFR Parts 20, 51, and 70. The NRC staff environmental review under Parts 20 and 70 is focused on that part of the applicant's facility-wide safety program that is established to control and assess the level of radioactive releases (gaseous, liquid, and solid) to the environment during normal and anticipated operations. Therefore, the effluent control portion of the applicant's radiation protection program, as well as effluent and environmental monitoring practices, are reviewed.

Environmental Protection

This review complements the review conducted under Section 9.2 of this SRP, which addresses the radiation protection program as it applies to worker safety.

An applicant engaged in the fabrication of MOX fuel must perform an ISA in accordance with proposed Subpart H of 10 CFR Part 70 and submit an ISA Summary in accordance with proposed 10 CFR 70.65. Guidance on the ISA is covered in Chapter 5.0 of this SRP. The environmental review of the ISA Summary should include the identified potential accident sequences that result in radiological releases to the environment, the items relied on for safety that are specified by the applicant to reduce the risk of these accidents, and the associated management measures that provide reasonable assurance that the items relied on for safety will perform their designated safety functions as required by 10 CFR Part 70.

Thus, environmental protection includes three main components: (1) the radiation protection program, (2) effluent and environmental monitoring for normal and off-normal operations, and (3) the ISA Summary and other ISA documentation, as necessary.

The areas of review for each of these components should include:

A. Radiation Safety

i. ALARA goals for effluent control;
ii. Effluent controls to maintain public doses ALARA;
iii. ALARA reviews and reports to management; and
iv. Waste minimization practices and for new operations, design plans for waste minimization.

B. Effluent and Environmental Monitoring

i. In-place filter testing procedures for air cleaning systems;
ii. Known or expected concentrations of radionuclides in effluents;
iii. Physical and chemical characteristics of radionuclides in discharges;
iv. Discharge locations;
v. Environmental media to be monitored and the sample locations;
vi. Sampling collection and analysis procedures, including the minimum detectable concentrations of radionuclides, equipment used, and calibration information;
vii. Action levels and actions to be taken when the levels are exceeded;
viii. Permits, including air discharge and National Pollutant Discharge and Elimination System permits;
ix. Leak detection systems for ponds, lagoons, and tanks;
x. Pathways analysis methods to estimate public doses;
xi. Recording and reporting procedures, including event notification; and
xii. Solid waste handling and disposal programs.

C. Safety Assessment of the Design Basis or Safety Program Description and Integrated Safety Analysis Summary

The safety assessment of the design basis (application for construction approval) or the Safety Program Description and ISA Summary (license application) address similar material, as follows:

i. Accident sequences (and associated facility processes) which, if unmitigated, result in releases to the environment;

ii. Likelihood and consequences of these accident sequences;

iii. Controls relied on to reduce the unmitigated risk from "high" risk to an acceptable level; and

iv. Availability and reliability of items relied on for safety.

10.4 ACCEPTANCE CRITERIA

10.4.1 Regulatory Requirements

A. 10 CFR Part 20, specifically the effluent control and treatment measures necessary to meet the dose limits and dose constraints for members of the public specified in Subparts B, D and F, the requirements for minimization of contamination specified in 10 CFR 20.1406, the survey requirements specified in Subpart F, the waste disposal requirements of Subpart K, the records requirements of Subpart L, and the reporting requirements of Subpart M.

B. 10 CFR Part 51, specifically the applicant must establish effluent and environmental monitoring systems to provide the information required by 10 CFR 51.60(a).

C. 10 CFR Part 51, specifically the applicant must submit an environmental report as required by 10 CFR 51.60(b), or to support a categorical exclusion as described in 10 CFR 51.22(c).

D. 10 CFR Part 70, the applicant must demonstrate that proposed facilities and equipment, including measuring and monitoring instruments and devices for the disposal of radioactive effluents and wastes, are adequate to protect public health and the environment as specified 10 CFR 70.22(a)(7).

E. 10 CFR Part 70, the application for a plutonium processing facility as defined in 10 CFR 70.4 must submit a safety assessment of the design basis of the principal structure, systems, and components of the facility, including provisions for protection against natural phenomena, as specified in 10 CFR 70.22(f).

F. Proposed 10 CFR Part 70, an application for a facility to fabricate MOX fuel must contain an ISA summary that includes a list of the items relied on for safety established by the applicant and other elements as described in 10 CFR 70.65(b).

10.4.2 Regulatory Guidance

The regulatory guidance for environmental protection is contained in:

A. NRC Regulatory Guide 4.5, "Measurements of Radionuclides in the Environment Sampling and Analysis of Plutonium in Soil."

B. NRC Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations)--Effluent Streams and the Environment."

C. NRC Regulatory Guide 4.16, "Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants."

D. NRC Regulatory Guide 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors."

E. NRC Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities."

F. NRC Information Notice 94-07, "Solubility Criteria for Liquid Effluent Releases to Sanitary Sewerage Under the Revised 10 CFR Part 20," January 28, 1994.

G. NRC Information Notice 94-23: "Guidance to Hazardous, Radioactive and Mixed Waste Generators on the Elements of a Waste Minimization Program," March 1994.

H. ANSI N13.1-1982, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

I. ANSI N42.18-1980, "Specification and Performance of On-site Instrumentation for Continuously Monitoring Radioactive Effluents."

J. NCRP Report No. 123, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground," January 1996.

10.4.3 Regulatory Acceptance Criteria

A. Radiation Safety

In accordance with 10 CFR 20.1101, each licensee must implement a radiation protection program, which is discussed in detail in Chapter 9.0 of this SRP. The environmental review of the radiation protection program focuses on the applicant's methods to maintain public doses ALARA in accordance with 10 CFR 20.1101. NRC guidance on compliance with these regulations can be found in Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities," July 1993.

Specifically, 10 CFR 20.1101(d) requires the applicant to establish a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its decay products, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent (TEDE) in excess of 0.1 mSv (10 mrem) per year from these emissions. The applicant must have procedures to report to the NRC in accordance with 10 CFR 20.2203 when this dose constraint is exceeded and to take prompt appropriate corrective action to ensure against recurrence. NRC guidance on compliance with this regulation can be found in Regulatory Guide 4.20, "Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other Than Power Reactors," December 1996.

The environmental review of the radiation protection program also focuses on the applicant's waste minimization practices. Applicant's for new licenses are required to comply with 10 CFR 20.1406, which states that the applicant must describe how facility design procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste. Applicant's requesting amendment or renewal of existing licenses must minimize and control waste generation during operations as part of the radiation protection program in accordance with 10 CFR 20.1101 [62 FR 39082].

Guidance for waste minimization programs can be found in NRC Information Notice No. 94-23: "Guidance to Hazardous, Radioactive and Mixed Waste Generators on the Elements of a Waste Minimization Program," March 25, 1994.

The proposed radiation protection program is acceptable if, in addition to the acceptance criteria outlined in Section 9.2, it satisfies the following criteria: .

i. Radiological (ALARA) Goals for Effluent Control

ALARA goals for effluent control are set at a modest fraction (10% to 20%) of the values in Appendix B, Table 2, Columns 1 and 2 and Table 3 and the external exposure limit in 10 CFR 20.1302(b)(2)(ii), or the dose limit for members of the public, if the applicant proposes to demonstrate compliance with 10 CFR 20.1301 through a calculation of the TEDE to the individual likely to receive the highest dose.

An applicant's constraint approach is acceptable if it is consistent with guidance found in Regulatory Guide 4.20 and the applicant's description of the constraint approach provides sufficient detail to demonstrate specific application of the guidance to proposed routine operations and non-routine operations including anticipated events.

ii. Effluent Controls to Maintain Public Doses ALARA

The applicant describes and commits to using effluent controls (e.g., procedures, engineering controls, and process controls) to maintain public doses ALARA. Common control practices include filtration, encapsulation, adsorption, containment, recycling,

leakage reduction, and the storage of materials for radioactive decay. The applicant demonstrates a commitment to reducing unnecessary exposure to members of the public and releases to the environment.

Effluent controls during normal and likely facility conditions:

a. Are capable of handling the expected volume of potentially radioactive waste;

b. Are compartmentalized to reduce the potential for cross contamination. For example, storm water and sanitary sewage lines should be separate from lines carrying radioactive effluents. Laundry facilities and personnel decontamination facilities should send effluents to radioactive waste. There should be no means by which radioactive waste can bypass the effluent controls and be directly released to the environment.

c. Are capable of safe shut down, consistent with the operating status of the facility.

d. Are capable of safely handling the chemical characteristics of the effluent. For example, effluent controls in contact with strong acids or caustics should be corrosion resistant.

e. Achieve a decontamination factor for each radionuclide sufficient to reduce the total radioactivity to an acceptable release level on a "once through" treatment basis. Provisions are made to recirculate effluents for further decontamination when radioactivity is above an acceptable release level.

iii. ALARA Reviews and Reports to Management

As part of the annual review of the content and implementation of the radiation protection program as discussed in Section 9.2, the applicant commits to review the effluent controls to maintain public doses ALARA. This review includes analysis of trends in release concentrations, environmental monitoring data, and radionuclide usage; determines whether operational changes are needed to achieve the ALARA effluent goals; and evaluates all designs for system installations or modifications. The applicant also includes a commitment to report the results to senior management along with recommendations for changes in facilities or procedures that are necessary to achieve ALARA goals.

iv. Waste minimization

The application contains a description of how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment and minimize, to the extent practicable, the generation of radioactive waste. The waste minimization programs is acceptable if the programs include:

a. Top management support;

b. Identification of responsibilities for waste minimization activities and assessments;

c. Methods to characterize waste generation, including types and amounts, and waste management costs, including costs of regulatory compliance, paperwork, transportation, treatment, storage, disposal, etc.;

d. Periodic waste minimization assessments to identify waste minimization opportunities and solicit employee or external recommendations;

e. Provisions for technology transfer to seek and exchange technical information on waste minimization;

f. Provisions to incorporate operational experience; and

g. Methods for implementation and evaluation of waste minimization recommendations.

B. Effluent and Environmental Monitoring

The applicant is required to make, or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive material in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in 10 CFR 20.1301. Accordingly, the staff's acceptance criteria for the applicant's effluent and environmental monitoring for normal and off-normal operations are described in Sections 10.4.3(B)(i) and 10.4.3(B)(ii).

i. Effluent Monitoring

The reviewer should find that the applicant's effluent monitoring is acceptable if it meets the following criteria:

a. The known or expected concentrations of radioactive materials in airborne and liquid effluents are below the limits in 10 CFR Part 20, Appendix B, Table 2 or below site specific limits established in accordance with 20.1302(c) and are ALARA.

If, in accordance with 20.1302(c), the applicant proposes to adjust the effluent concentrations in Appendix B to 10 CFR Part 20 to take into account the actual physical and chemical characteristics of the effluents, the applicant provides information related to aerosol size distributions, solubility, density, radioactive decay equilibrium, and chemical form. This information is complete and accurate for the radioactive materials to justify the derivation and application of the alternative concentration limits.

b. If the licensee proposes to demonstrate compliance with 10 CFR 20.1301 through a calculation of the TEDE to the individual likely to receive the highest dose in accordance with 10 CFR 20.1302(b)(1), calculation of the TEDE by pathways analyses uses appropriate models and codes and assumptions that accurately represent the facility, the site, and the surrounding area; assumptions are reasonable; input data is accurate; all applicable pathways are considered; and the results are interpreted correctly.

   National Council on Radiation Protection (NCRP) Report No. 123, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground," January 1996, provides acceptable methods for calculating the dose from radioactive effluents. Computer codes are acceptable tools for pathways analysis if the applicant is able to demonstrate that the code has undergone validation and verification to demonstrate the validity of estimates developed using the code for established input sets. Dose conversion factors used in the pathways analyses are acceptable if they are based on the methodology described in International Council on Radiation Protection (ICRP) 30, "Limits for Intakes of Radionuclides by Workers," as reflected in Federal Guidance Report 11. Such methods are acceptable for determining the dose to the maximally exposed individual during normal facility operations and anticipated events.

c. All liquid and airborne effluent discharge locations are identified and monitored. Monitoring locations are identified, and for those effluent discharge points that have input from two or more contributing sources within the facility, sampling each contributing source is evaluated for effective effluent control.

d. Airborne effluents from all routine operations and non-routine operations, as well as anticipated events associated with the facility, including effluents from areas not used for processing special nuclear material such as laboratories, experimental areas, storage areas, and fuel element assembly areas, are continuously sampled.

   Effluents are sampled unless the applicant has established, by periodic sampling or other means, that radioactivity in the effluent is insignificant and will remain so. In such cases, the effluent is sampled at least quarterly to confirm that effluents are not significant. For the purposes of this SRP, an effluent is significant if the concentration averaged over a calendar quarter is equal to 10% or more of the appropriate concentration listed in Table 2 of Appendix B to 10 CFR Part 20.

e. The sample collection and analysis methods and frequencies are appropriate for the effluent medium and the radionuclide(s) being sampled. Sampling methods ensure that representative samples are obtained by use of appropriate sampling equipment and sample collection and storage procedures. For liquid effluents, representative samples are taken at each release point for the determination of concentrations and quantities of radionuclides released to an unrestricted area, including discharges to sewage systems. For continuous releases, samples are continuously collected at each release point. For batch releases, a representative sample of each batch is

collected. If periodic sampling is used in lieu of continual sampling, the applicant shows that the samples are representative of actual releases. Monitoring instruments are calibrated at least annually, or more frequently if suggested by the manufacturer.

f. Radionuclide specific analyses are performed on selected composited samples unless either:

- The gross alpha and gross beta activities are so low that individual radionuclides could not be present in concentrations greater than 10 percent of the concentrations specified in Table 2 or 3 of Appendix B to 10 CFR Part 20; or

- The radionuclide composition of the sample is known through operational data, such as the composition of the feed material.

Monitoring reports in which estimates of quantities of individual radionuclides are based on methods other than direct measurement include an explanation and justification of how the results were obtained.

Examples of cases in which operational data may not be adequate for the determination of radionuclide concentration are (1) facilities processing uranium in which extraction, ammonium diuranate precipitation, ion exchange, or other separation processes could result in concentration of thorium isotopes (principally Th-234); (2) facilities in which uranium of varying enrichments is processed; and (3) facilities processing plutonium in which significant variation in the Pu-238/Pu-239 ratio among batches and the continuous in-growth of Am-241 would preclude the use of feed material data to determine the radionuclide composition of effluents.

Radionuclide analyses are performed more frequently than usual under three circumstances: (1) at the beginning of the monitoring program until a predictable and consistent radionuclide composition in effluents is established; (2) whenever there is a significant unexplained increase in gross radioactivity in effluents; or (3) whenever a process change or other circumstance might cause a significant variation in the radionuclide composition.

g. The minimum detectable concentration (MDC) for sample analyses is not more than 5 percent of the concentration limits listed in Table 2 of Appendix B to 10 CFR Part 20. If the actual concentrations of radionuclides in samples are known to be higher than 5 percent of the 10 CFR Part 20 limits, the analysis methods need only be adequate to measure the actual concentration. However, in such cases, the MDC is low enough to accommodate fluctuations in the concentrations of the effluent and the uncertainty of the MDC.

h. The laboratory quality control (QC) procedures are adequate to support the validity of the analytical results. These QC procedures include the use of established standards such as those provided by the National Institute of Standards and

Technology (NIST), as well as standard analytical procedures, such as those established by the National Environmental Laboratory Accreditation Conference.

i. The proposed action levels and actions to be taken if the action levels are exceeded are appropriate. The action levels are incremental, such that each increasing action level results in a more aggressive action to assure and control effluents. A slightly higher than normal concentration of a radionuclide in effluent triggers an investigation into the cause of the increase. An action level is specified that will result in the shutdown of an operation if this level is exceeded. These action levels are selected based on the likelihood that a measured increase in concentration could indicate potential violation of the effluent limits.

j. The descriptions of applicable Federal and/or State standards for discharges and any permits issued by local, State, or Federal governments for gaseous and liquid effluents are complete and accurate.

k. The systems for the detection of leakage from ponds, lagoons, and tanks are adequate to detect and assure against any unplanned releases to groundwater, surface water, or soil.

l. Releases to sewer systems are controlled and maintained to meet the requirements of 10 CFR 20.2003, including (i) the material is water soluble; (ii) known or expected discharges meet the effluent limits of 10 CFR 20 Appendix B, Table 3; and (iii) the known or expected total quantity of radioactive material released into the sewer system in a year does not exceed 5 Ci (185 GBq) of ^{3}H, 1 Ci (37 GBq) of ^{14}C, and 1 Ci (37 GBq) of all other radioactive materials combined. Solubility is determined in accordance with the procedure described in NRC Information Notice 94-07.

m. Reporting procedures comply with the requirements of 10 CFR 70.59 and the guidance specified in Regulatory Guide 4.16. Reports of the concentrations of principal radionuclides released to unrestricted areas in liquid and gaseous effluents are provided and include the MDC for the analysis and the error for each data point.

n. The applicant's procedures and facilities for solid and liquid waste handling, storage, and monitoring result in safe storage of the material and timely disposition.

ii. Environmental Monitoring

The applicant's environmental monitoring is acceptable if it is commensurate with the scope of activities at the facility and the expected impacts of operations as identified in the environmental report and meets the following criteria:

a. Background and baseline concentrations of radionuclides in environmental media have been established through sampling and analysis.

b. A preoperational monitoring program is initiated prior to operation. The preoperational program should be of sufficient length to allow a sufficient data base for comparison with operational data.

c. Monitoring includes sampling and analyses for important pathways for the anticipated types of radionuclides released from the facility into the environment from routine operations and non-routine operations including anticipated events. The pathways include air, surface water, groundwater, soil, sediments, and vegetation, as appropriate. Important environmental media are sampled to estimate radionuclide concentrations in important biota.

d. The description of monitoring identifies adequate and appropriate sampling locations and frequencies for each environmental medium, the frequency of sampling, and the analyses to be performed on each medium. Sampling methods ensure that representative samples are obtained by use of appropriate sampling equipment, sample collection, and sample storage procedures.

e. Monitoring procedures employ acceptable analytical methods and instrumentation to be used, and monitoring procedures and analytical methods are subject to quality controls. The applicant commits to a program of instrument maintenance and calibration appropriate to the instrumentation, as well as participation in round-robin measurement comparisons if the applicant proposes use of its own analytical laboratory for analysis of environmental samples.

f. Appropriate action levels and actions to be taken if the levels are exceeded are specified for each environmental medium and radionuclide.

   Action levels are selected based upon a pathways analysis that demonstrates that below those concentrations, doses to the public will be below the limits in 10 CFR Part 20, Subpart B, and are ALARA. The action levels specify the concentrations at which an investigation would be performed and levels at which process operations would be shut down.

g. MDCs are specified for sample analyses and are at least as low as those selected for effluent monitoring in air and water. MDCs for sediment, soil, and vegetation are selected based upon the action levels to ensure sampling and analytical methods are sensitive and reliable enough to support application of the action levels.

h. Data analysis methods and criteria to be used for evaluating and reporting the environmental sampling results are appropriate and will indicate when an action level is being approached in time to take corrective actions.

i. The description of the status of all licenses, permits, and other approvals of facility operations required by Federal, State, and local authorities is complete and accurate.

j. Environmental monitoring is adequate to assess impacts to the environment from potential radioactive and nonradioactive releases as identified in high and medium risk accident sequences in the ISA.

C. Safety Assessment of the Design Basis and the Safety Program Description and Integrated Safety Analysis Summary

In accordance with 10 CFR 70.22(f), an applicant for a MOX fuel fabrication facility is required to submit a safety assessment of the design basis as part of its application for construction approval. Prior to submitting the license application for possession and use of SNM, pursuant to Subpart H to the proposed 10 CFR Part 70, an applicant for a MOX fuel fabrication facility is required to submit a Safety Program Description and an ISA Summary. Both the design basis and the ISA Summary include items relied on for safety identified to prevent or mitigate against accidents. The applicant's treatment of environmental protection in the safety assessment of the design basis or the Safety Program Description and ISA Summary is acceptable if the applicant:

i. Provides a complete list of accident sequences that result in radiological releases to the unrestricted area.

ii. Provides a reasonable estimate for the likelihood of each accident sequence identified.

iii. Uses acceptable methods for estimating consequences from accident sequences which result in radiological releases to the environment. For the purposes of this review, consequences include dose to the public and the 24-hour averaged release of radioactive material outside the restricted area as defined in proposed 10 CFR 70.61. Acceptable methods are described in NUREG/CR-6410, "Nuclear Fuel Cycle Facility-Accident Analyses Handbook." Models used for consequence analysis are verified and validated.

iv. Items relied on for safety (or systems and components) are identified for each accident sequence that results in consequences greater than the limits defined in proposed 10 CFR 70.61. The items relied on for safety prevent or mitigate risk sequences to an acceptable level of protection.

v. Affords adequate levels of assurance to the items relied on for safety to ensure that they will be reliable and available to perform their safety functions. This may be accomplished through configuration management, training, maintenance activities, or other management measures as appropriate.

10.5 REVIEW PROCEDURES

10.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval or license application adequately addresses the items in Section 10.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the safety assessment of the design basis should address Section 10.3(C) consistent with the level of design. Sections 10.3(A) and (B) should be addressed to the extent that the material therein supports information provided in the environmental report, e.g, environmental monitoring as a mitigation measure. 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 Application for Operations

Specifically, the safety assessment of the license application should address Sections 10.3(A)-(C) in full.

If the primary reviewer verifies that environmental protection is adequately addressed (construction or operations), the primary reviewer should accept the application for the safety evaluation in Section 10.5.2. 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.

10.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 10.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 10.4. 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 10.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for 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 environmental protection, meet or exceed the regulatory acceptance criteria in Section 10.4.3(C).

B. License Application

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. The primary reviewer should identify the mechanisms that will allow the applicant to identify and correct potential problems.

In support of the primary reviewer for Chapter 9.0, the environmental protection reviewer should determine whether the acceptance criteria in Chapter 9.0 have been met as they relate to environmental review of the radiation protection program. The primary reviewer should also support the primary reviewer for Chapter 8.0 to ensure that the acceptance criteria for Chapter 8.0 have been met as they relate to effluent controls to maintain public doses ALARA.

In support of the primary reviewer for Chapter 5.0, the environmental reviewer should review the ISA Summary. All accident sequences identified in the ISA that can have significant consequences due to releases to the unrestricted area, should be reviewed to determine that the list of potential accidents is complete and properly identified. Detailed review should only be conducted of the accident sequences which, when left unmitigated, are rated as "high-consequence" events by the applicant, as well as approximately 10% of the "intermediate-consequence" events and a smaller number of the lower risk sequences. However, additional "high-consequence" and "intermediate-consequence" events may be evaluated based on the results of the initial review.

The primary reviewer should provide input on the ISA Summary to the primary reviewer of Chapter 5.0 and input on management measures (if any) to the primary reviewer of Chapter 15.0.

In addition, for renewal and amendment applications, review of environmental protection by the primary reviewer will include coordination with the inspector responsible for environmental protection (supporting reviewer). Any comments or concerns that the inspector identifies will be addressed and resolved, and the Safety Evaluation Report (SER) (described in Section 10.6.1) for the licensing action will contain a statement indicating if the inspection staff has any objections to approval of the proposed licensing action. In addition, if applicable, the primary reviewer will review inspection reports and semi-annual effluent reports submitted in accordance with 10 CFR 70.59 to assure licensee performance in environmental protection.

When the safety evaluation for the application for construction approval or license application is complete, the primary reviewer, with assistance from the other reviewers, should prepare the environmental protection input for the SER as described in Section 10.6 using the acceptance criteria from Section 10.4.

10.6 EVALUATION FINDINGS

Documentation of the evaluation findings for the environmental protection review is contained in two types of products. The SER documents the review of the environmental protection measures and the design basis (application for construction approval) or the Safety Program Description and ISA Summary (license application). The EA or EIS documents the staff's independent assessment of the environmental impacts of the proposed action.

Environmental protection measures may be summarized in the EA or EIS. However, the EA or EIS does not become part of the license. 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 (see the supplementary guidance in Appendix E to this SRP).

If an EA or EIS was prepared for the licensing action, the date the document was issued should be reported in the environmental protection section of the SER. If the EA resulted in a FONSI, the FONSI's publication date in the Federal Register should be included in the SER. If an EIS was prepared, the SER would include the Federal Register publication date for the Record of Decision. When applicable, the SER also documents the determination that an action meets a categorical exclusion.

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

> *The staff prepared an environmental impact statement (EIS) [publication date] for the construction approval for [insert name of facility]. Based on the EIS, the NRC stated in its Record of Decision [publication date in the Federal Register] that the preferred option was [state preferred option here].*
>
> *For the preferred option, the staff reviewed the environmental protection measures for construction approval for [insert facility name] according to Chapter 10.0 of NUREG-1718. The staff evaluated [state what was evaluated] and found [state what was found]. The staff concluded that the applicant's design basis has adequate environmental protection measures to protect the public and the environment against natural phenomena and the consequences of potential accidents in accordance with the regulatory requirements imposed by the Commission in 10 CFR Part 70.*

The staff could document the safety evaluation for the license application as follows:

The staff prepared an environmental impact statement (EIS) [publication date] for this licensing action as required by 10 CFR 51.20. Based on the EIS, the NRC stated in its Record of Decision [publication date in the Federal Register] that the preferred option was [state preferred option here].

For the preferred option, the staff reviewed the environmental protection measures for issuing a license to possess and use SNM for [insert facility name] according to Chapter 10.0 of NUREG-1718. The staff evaluated [state what was evaluated] and found [state what was found]. The staff concluded that the applicant has adequate environmental protection measures, including: (1) environmental and effluent monitoring and (2) effluent controls to maintain public doses ALARA as part of the radiation protection program, to protect public health and the environment and comply with the regulatory requirements imposed by the Commission in 10 CFR Parts 20, 51, and 70.

10.7 REFERENCES

A. American National Standards Institute, N13.1-1982, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

B. American National Standards Institute, N42.18-1980, "Specification and Performance of On-site Instrumentation for Continuously Monitoring Radioactive Effluents."

C. National Council on Radiation Protection and Measurements, NCRP Report No. 123 I & II, *"Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground,"* January 1996.

D. NRC Information Notice No. 94-23: "Guidance to Hazardous, Radioactive and Mixed Waste Generators on the Elements of a Waste Minimization Program," March 25, 1994.

E. NRC Information Notice 94-07, "Solubility Criteria for Liquid Effluent Releases to Sanitary Sewerage Under the Revised 10 CFR Part 20," January 28, 1994.

F. U.S. Nuclear Regulatory Commission, NMSS/FCSS/Fuel Cycle Licensing Branch, Rev. 6, *"Materials Licensing Procedures Manual,"* April 1998.

G. U.S. Nuclear Regulatory Commission, Regulatory Guide 4.15, Rev. 2, *"Quality Assurance for Radiological Monitoring Programs (Normal Operations)--Effluent Streams and the Environment,"* February 1979.

H. U.S. Nuclear Regulatory Commission, Regulatory Guide 4.16, Rev. 2, *"Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants,"* December 1985.

I. U.S. Nuclear Regulatory Commission, Regulatory Guide 4.20, *"Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other Than Power Reactors,"* December 1996.

J. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.37, *"ALARA Levels for Effluents from Materials Facilities,"* July 1993.

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

L. U.S. Nuclear Regulatory Commission, NUREG -1520, *"Draft Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,"* April 1998.

M. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.52, Revision 2, *"Standard Format and Content for the Health and Safety Sections of License Applications for Fuel Cycle Facilities,"* January 1995.

11.0 PLANT SYSTEMS

11.1 PURPOSE OF REVIEW

The purpose of this review is to determine that the plant systems--systems that are identified as items relied on for safety pursuant to the proposed 10 CFR Part 70 and encompassed by the hazard and accident analyses of the Integrated Safety Assessment (ISA)--will be available and reliable to perform their intended safety function when needed. Examples of plant systems are: (a) a ventilation system necessary to provide certain decontamination functions for normal, off-normal, and accident conditions; (b) a cooling system necessary to provide a heat sink to prevent certain process elements from exceeding temperature limits; and (c) an electrical distribution system necessary to support various items relied on for safety.

11.2 RESPONSIBILITY FOR REVIEW

Primary: Discipline specific engineers

Secondary: Chemical Process Engineer, Health Physicist, Fire Protection Specialist, Human Factors Engineer

Supporting: Primary Reviewers of SRP Sections 1.1 and 13.1, and Chapters 2, 3, 4 and 14
Primary Reviewers of Applicable Sections of SRP Chapter 15

11.3 AREAS OF REVIEW

The review for the construction approval should focus on the layout and design of the plant systems, their components, and any related information considering the present stage of the applicant's design process. The review for the license application should focus on design modifications and any other system features not adequately described during the construction approval review.

Also, the review for the licensee application for operations should encompass the adequacy of the design and operation of plant systems identified in the ISA Summary as items relied on for safety such as electrical and ventilation systems.

The license application for operations documentation, to be reviewed by the staff, should include specific items listed below for each system. The documentation for construction approval should address the following items to the extent possible considering the stage of design information available.

Plant Systems

A. Safety Function

i. Identification of safety function as related to the performance requirements of proposed §70.61 and the ISA;

ii. Functional requirements stemming from the baseline design criteria (BDC) and the ISA process including environmental design considerations (temperature, pressure, humidity, etc., resulting from normal, off-normal, and accident operating conditions) for items relied on for safety with site factors (including natural phenomena that occur infrequently and conditions that are continuously present), redundancy, independence, defense-in-depth, and reliability/availability goals (including continued operation of plant systems that perform essential utility services).

B. System Description

i. Purpose (safety and non-safety);

ii. System design, including performance features;

iii. Structures (including their materials, shielding, and physical protection) and components;

iv. Instrumentation and controls (manual and automatic);

v. System interfaces;

vi. Drawings (including arrangements, plans, elevations, and sections for structures), specifications, and procedures;

vii. Assurance measures including applicable industrial codes and standards, environmental qualification, quality assurance, inspection, testing, and maintenance.

C. Safety Analysis

i. How functional requirements are satisfied by system design;

ii. How non-safety features do not prevent the plant system from performing its intended safety function;

iii. How long-term performance, testing, and maintenance features are addressed;

iv. How potential failure modes are analyzed including consideration of communication failures, common-mode failures and human errors;

v. How material-related failure modes are analyzed to include the effects of corrosion, erosion, and fatigue under normal, off-normal, and accident conditions.

vi. How data, information, and evaluations are developed as a result of site-related investigations, studies of historical data, and any newly developed information addressing the geology, seismology, hydrology, meteorology, and geotechnical aspects of the site as well as site proximity events considered as natural phenomena events (such as earthquakes, high winds, tornadoes, tornado missiles and floods) and other external events (such as nearby transportation accidents, airplane crashes, fires external to the facility) that may produce conditions that could influence the performance of plant facilities that are necessary to protect health and minimize danger to life or property.

Because the results of the ISA identify the items relied on for safety that form the safety functions discussed above, the primary reviewer should also review the ISA Summary (see SRP Chapter 5) to determine which plant systems have been identified as items relied on for safety, their safety categories, their assumed operating modes and conditions, the impact of their inoperability, and any related limiting operations or plant mode restrictions. The review should also encompass any additional assumptions used in ISA qualitative/quantitative evaluations related to performance requirements for plant systems such as redundancy, independence, reliability, quality, etc.

11.4 ACCEPTANCE CRITERIA

As part of the application for construction approval, the applicant should commit to providing plant systems which meet or exceed the acceptance criteria in the following subsections of this SRP section.

11.4.1 Regulatory Requirements

The staff's requirements applicable to all plant systems are the following:

10 CFR Part 70.22, specifically relating to the requirement that the applicant is to provide a description of the equipment and facilities and propose procedures to protect health and minimize danger to life and property.

10 CFR Part 70.23, specifically relating to the requirement that the Commission determine that the proposed equipment, facilities, and procedures are adequate to protect health and minimize danger to life and property.

10 CFR Part 70.61(e), as proposed, specifically relating to the requirement that each engineered or administrative control or control system that is needed to meet the performance requirements be designated as an item relied on for safety.

10 CFR Part 70.62, as proposed, specifically relating to the establishment and maintenance of a safety program and to the performance of an ISA.

10 CFR Part 70.64, as proposed, specifically relating to the application of BDC and defense-in-depth practices to new facilities or new processes at existing facilities.

11.4.2 Electrical Systems

11.4.2.1 Regulatory Guidance

The regulatory guidance and associated industry standards that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria for electrical systems are:

NUREG-0800, Standard Review Plan, Chapter 8, "Electric Power," Table 8-1, Acceptance Criteria and Guidelines for Electric Power Systems, U. S. Nuclear Regulatory Commission.

NRC Regulatory Guides for Division 1, Power Reactors, and associated Institute of Electrical and Electronic Engineers (IEEE) Standards encompassing the design, installation, and testing of equipment and components such as emergency diesel generators, batteries and cables in electrical systems performing safety functions.

Although the above documents apply to nuclear power plants, they provide background information, provide guidance which the staff expects to be applied to the MOX fuel fabrication facilities, and address subjects that the reviewer must verify have been adequately considered by the applicant.

11.4.2.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's electrical systems' design and operation acceptable if they satisfy the requirements listed in Section 11.4.1 and follow the relevant guidelines mentioned under Section 11.4.2.1. The requirements and guidelines for electrical systems are those related to the BDC and defense-in-depth. Typically, these include specific design considerations such as two physically independent offsite power sources with redundant and independent onsite ac and dc power sources that are designed with:

- Test, calibration, and in-service surveillance capabilities;
- Electrical and physical separation;
- No single failure vulnerability;
- Sufficient capacity and capability;
- Adequate protective relaying and breaker control;
- Status monitoring;
- Proper equipment qualification, quality assurance, and reliability; and
- Adequate design for natural phenomena.

Additionally, the electrical systems' design and operation should fulfill the functional requirements determined from the ISA and the electrical systems should be available and reliable to perform their intended safety function when needed.

11.4.3 Instrumentation and Control Systems

11.4.3.1 Regulatory Guidance

The regulatory guidance and associated industry standards that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria for instrument and control systems are:

NUREG-0800, Standard Review Plan, Chapter 7, "Instrumentation and Controls," U. S. Nuclear Regulatory Commission.

NRC Regulatory Guides for Division 1, Power Reactors, and associated IEEE Standards encompassing the design, installation, and testing of equipment, components, and computer software in instrumentation and control systems performing safety functions.

Although the above documents apply to nuclear power plants, they provide background information, provide guidance which the staff expects to be applied to the MOX fuel fabrication facilities, and address subjects that the reviewer must verify have been adequately considered by the applicant.

11.4.3.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's instrumentation and control systems' design and operation acceptable if they satisfy the requirements listed in Section 11.4.1 and follow the relevant guidelines mentioned under Section 11.4.3.1. The requirements and guidelines for instrumentation and control systems are those related to the BDC and defense-in-depth. Typically, these include specific design considerations such as redundant and/or diverse instrument channels with coincident logic providing automatic actuation with additional manual operation capability. The instrument channels are designed with:

- Test, calibration, and in-service surveillance capabilities;
- Electrical, physical, and control/protection separation;
- No single failure vulnerability;
- Adequate instrument spans, setpoints, and control ranges;
- Fail safe failure mode;
- Status monitoring;
- Proper equipment qualification, quality assurance, and reliability; and
- Adequate design for natural phenomena.

Additionally, the instrumentation and control systems' design and operation should fulfill the functional requirements determined from the ISA and the instrumentation and control systems should be available and reliable to perform their intended safety function when needed.

11.4.4 Cooling Water System

11.4.4.1 Regulatory Guidance

None, at present, specific to MOX fuel fabrication facilities.

11.4.4.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's cooling water system's design and operation acceptable if they satisfy the requirements listed in Section 11.4.1. The requirements and guidelines for the cooling water system are those related to the BDC and defense-in-depth. Typically, the cooling water system is designed to demonstrate:

A. Transfer of heat loads from safety-related structures, systems, and components to an appropriate heat sink under normal, off-normal, and accident conditions;

B. Adequate water supply under normal, off-normal, and accident conditions;

C. Adequate component redundancy; the capability to isolate components, systems, or piping for maintaining system safety function under varying system configuration; and the capability of integrated system control;

D. Supporting management measures (including tests and other verification methods) ensure the structural integrity and system leak tightness (including the prevention of cross-contamination (radioactive and chemical)), the operability and adequate performance of active system components, and the capability of the system to perform required functions during normal and accident situations;

E. Capability for withstanding environmental hazards resulting from pipe line breaks and dynamic effects associated with flow instability and attendant loads such as water hammer or cavitation and measures to prevent such dynamic conditions from occurring;

F. Capacity and capability for detecting leaks and cross-contamination (radioactive and chemical); for inservice component inspection and system maintenance; and for operational functional testing of the system and its components.

G. A quality assurance program is established for the design, construction, testing, operation, and maintenance of all structures and components of the cooling water system that are identified as items relied on for safety in accordance with the criteria in Chapter 15.1, "Quality Assurance," and Appendix F of this SRP.

H. The cooling water system and its components which are identified as items relied on for safety are designed to withstand the effects of tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena in accordance with criteria established in the ISA.

Additionally, the cooling water system's design and operation should fulfill the functional requirements determined from the ISA, and the cooling water system should be available and reliable to perform its intended safety function when needed.

11.4.5 Ventilation Systems

11.4.5.1 Regulatory Guidance

The regulatory guidance and associated industry standards that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria for ventilation systems are:

NUREG-0800, Standard Review Plan, Chapters 9.4.1, "Control Room Area Ventilation System," 9.4.3, "Auxiliary and Radwaste Area Ventilation System," and 9.4.5, "Engineered Safety Feature Ventilation System," and Regulatory Guides as cited in the acceptance criteria of these chapters.

Although the above documents apply to nuclear power plants, they provide background information, provide guidance which the staff expects to be applied to the MOX fuel fabrication facilities, and address subjects that the reviewer must verify have been adequately considered by the applicant.

Applicable DOE standards and industry codes and standards issued by the American National Standards Institute, American Society of Mechanical Engineers, Air Movement and Control Association, American Society of Heating, Refrigerating and Air-Conditioning Engineers, National Fire Protection Association, and Underwriters Laboratories.

11.4.5.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's ventilation systems' design and operation acceptable if they satisfy the requirements listed in Section 11.4.1 and follow the relevant guidelines mentioned under Section 11.4.5.1. The requirements and guidelines for ventilation systems are those related to the BDC and defense-in-depth.

Typically, specific design considerations for ventilation systems are:

A. Confinement of radioactive contamination by zones and pressure differentials

   i. Confinement of radioactive material is provided by multiple zones with each zone bounded by barriers such as vessel, glovebox, building, and internal room walls.

   ii. The systems have the capability to direct ventilation air from areas of low radioactivity to areas of progressively higher radioactivity. Devices are provided to control and indicate pressure differentials between confinement zones. Alarms are provided to indicate when pressure differentials are not maintained in a prescribed range.

   iii. The systems have the capability to detect the need for isolation and to isolate portions of the systems relied on for safety in the event of failures or malfunctions elsewhere in the systems. The isolated systems have the capability to function under such conditions.

   iv. Supply air fans are interlocked with an exhaust air plenum pressure sensor to prevent supply fan operation unless the exhaust fans are running. This will prevent pressurization of any process room or area should exhaust ventilation fail.

   v. Additional design guidance may be found in Regulatory Guides 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

B. Test, calibration, and in-service surveillance capabilities

   i. Provisions are made so that components of ventilation systems can be tested periodically for operability and required functional performance. The provisions include capability for periodic measurement of air flows in exhaust ducts and or at equipment, hoods, and exhaust ducts.

   ii. The capability is also provided to test, under conditions as close to design as practicable, the operating sequence that would bring ventilation systems into action, including the transfer to alternate power sources and the design airflow delivery capability.

   iii. Additional test guidance may be found in Regulatory Guides 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants."

C. Redundancy of fans, dampers, and power supplies and no single failure vulnerability

i. There are two automatically operated isolation dampers in series to separate nonessential portions of the system from essential portions.

ii. Essential components and subsystems are able to function in the event of loss of offsite power. In the event of the failure of a single active component (equipment or control device) or loss of offsite power, the resulting systems flow capacity will not cause the loss of preferred direction of air flow from areas of low potential radioactivity to areas of higher potential radioactivity.

iii. The systems are capable of automatically actuating components not operating under normal conditions or actuating standby components (redundant equipment) in the event of failure or malfunction, as needed.

D. Sufficient capacity and capability

i. The heating and cooling functions of the ventilation systems are sufficient to maintain a suitable temperature range in the areas serviced, assuming proper performance of equipment contained in those areas.

ii. Equipment identified as items relied on for safety are capable of functioning under the worst anticipated ventilation systems' performance.

iii. The systems are capable of preventing the accumulation of flammable or explosive gases from processes within the facility.

iv. The systems are capable of controlling airborne particulate material (dust) accumulation.

v. Ventilation systems are capable of operating during a normal power outage at capacities required to maintain confinement of contaminants.

E. Monitoring and alarms

i. All exhausting ducts and stacks which may contain plutonium contaminants are provided with two monitoring systems: a continuous air monitoring system (CAMS) and a fixed sampler. The probes for sampling purposes are designed for isokinetic sampling and located to obtain representative samples. Each system is connected to an emergency power supply. The continuous stack sampler alerts cognizant personnel through an audible and visual annunciator if the airborne radioactive effluents reach prescribed limits.

ii. Air monitoring and warning systems (including CAMS) are installed in areas where radioactive material is handled. Air sampling heads provide a representative of the potential airborne radioactivity being breathed.

iii. Duct runs and flow distributers assure uniform representative air flow past monitoring and sampling stations as well as through filter installations.

iv. Acceptance criteria for air monitoring and warning systems specific to radiation safety for design features, the radiation protection program, and effluent monitoring can be found in Sections 9.1.4.4.3(C), 9.2.4.5, and 10.4.3.B of this SRP.

F. Environmental qualification and quality assurance

i. The ventilation systems, including detectors, monitoring systems, and controls, are qualified for all expected and credible severe environments in which the systems are expected to function.

ii. A quality assurance program is established for the design, construction, testing, operation, and maintenance of all structures, systems, and components of the ventilation systems that are identified as items relied on for safety in accordance with the criteria in Chapter 15.1, "Quality Assurance," and Appendix F of this SRP.

G. Adequate design for natural phenomena

i. The ventilation systems and their components which are identified as items relied on for safety are designed to withstand the effects of tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena in accordance with criteria established in the ISA.

ii. Design considerations are also made for protection from offsite releases of toxic chemicals as a result of natural phenomena, if appropriate.

H. Appropriate fire protection and smoke control

i. The ventilation systems are designed to withstand any credible fire and explosion and continue to act as confinement barriers.

ii. Ventilation systems are capable of operating during a fire in the areas they ventilate and safely handle products of combustion through appropriate ventilation channels. A supply air system remains operational; however, the option to discontinue air supply to the involved spaces is maintained.

iii. The materials of construction for the ventilation systems are fire resistant to protect against fires occurring within or without the systems. Approved smoke and heat detectors are provided in the system.

iv. Detailed fire protection guidance for filter plenums is found in Appendix D of this SRP.

I. Assuring a safe air supply to the control room and other occupied areas

i. The ventilation systems confine and prevent uncontrolled release of radioactive aerosols, noxious fumes, and vapors into rooms and areas normally occupied by personnel.

ii. There is continuous monitoring of recirculated air to occupied areas and diversion of contaminated air to a once-through exhaust system if allowable radiation standards are exceeded.

iii. The control room heating and cooling subsystems are capable of maintaining a suitable ambient temperature for control room personnel and equipment.

iv. There are provisions to isolate portions of the system in the event of fires, failures, and malfunctions.

v. The ventilation systems are capable of keeping essential equipment in the control room operational under the worst anticipated degraded conditions of the ventilation system.

vi. The control room ventilation has provisions for an internal recirculation filtering mode or can discharge airborne contaminants from the control room area using a once-through ventilation mode, as applicable.

vii. Applicable guidance may also be found in Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

J. Removal and replacement of filters and other expected maintenance is designed to permit only minimum exposure of personnel to radioactivity.

i. Items relied on for safety allow for routine in-place testing of high efficiency particulate air (HEPA) filtration systems as outlined in ASME N510.

ii. Potential doses from expected maintenance of ventilation systems can be minimized by providing ready access to the systems, by providing space to permit the activities to be accomplished expeditiously, by separating filter banks and components to reduce exposures to radiation from adjacent banks and components, and by providing sufficient space to accommodate auxiliary ventilation of shielding of components.

iii. Additional guidance may also be found in Chapter 9.0, "Radiation Safety," of this SRP and Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable."

K. Gloveboxes and process enclosures

i. Gloveboxes are constructed using the highest quality of materials and workmanship to assure total containment and minimize leakage. Gloveboxes are constructed of non-combustible materials. (see Chapter 7.0 of this SRP).

ii. The design of enclosures is based on downdraft ventilation flow to minimize the spread of fire. Heat detectors and combustible gas and vapor detection meters are provided on gloveboxes or enclosures where fire or explosion hazards exist. An inerting environment or automatic suppression are provided in these boxes or enclosures. Where automatic suppression is not provided, fire detectors are installed and provisions made for manual fire suppression. (See Chapter 7.0 of this SRP.)

iii. Small gloveboxes or enclosure systems supplied with gases under positive pressure have positive-acting pressure-relief devices (discharging into an exhaust system) to prevent overpressurization. Further, should these systems be recirculating, all necessary cleanup and detection equipment for noxious, corrosive, or explosive vapors or gases are considered.

iv. The minimum instrumentation for a glovebox or enclosure ventilation system includes devices to indicate the pressure differential between the box or enclosure and the surrounding work area, the filter resistance, and the exhaust flow rate from the box or enclosure. (The applicant should specify the maximum operable pressure differential.) When box operations are not in full-time attendance for a continuous process, a sensor is provided to monitor abnormal pressure or temperature and alarm at a point where cognizant personnel are stationed.

Additionally, the ventilation systems' design and operation should fulfill all of the functional requirements determined from the ISA, and the systems should be available with adequate reliability to perform all of their intended safety functions when needed.

11.4.6 Civil-Structural Systems

The civil-structural systems include the buildings and support structures of the facilities that are to house, support, confine, or contain the various other plant systems, components, and equipment associated with licensed nuclear materials or hazardous chemicals associated with licensed nuclear materials that may adversely affect items relied on for safety.

11.4.6.1 Regulatory Guidance

The regulatory guidance and associated industry standards that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria for civil-structural systems are:

RG 1.132, "Site Investigations for Foundations of Nuclear Power Plants"

RG 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants"

In addition to regulatory guides, other sources of guidance are industry consensus standards such as:

ANSI/ANS-2.11-1978(R1989), "Guidelines for Evaluating Site-Related Geotechnical Parameters at Nuclear Power Plant Sites"

ANSI/ANS-2.19-1981(R1990), "Guidelines for Establishing Site-Related Parameters for Site Selection and Design of an Independent Spent Fuel Storage Installation (Water Pool Type)"

These documents provide guidance on the considerations that influence the array of natural phenomena that must be considered in the group of external events as well as the determination of site related technical information. Additionally these documents identify the subjects that the reviewer should verify that the applicant has appropriately integrated into the design of the facility. Other regulatory guides and industry consensus documents are listed below:

RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"

RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants"

RG 1.76, "Design Basis Tornado for Nuclear Power Plants"

RG 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants"

RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis"

RG 1.117, "Tornado Design Classification"

RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"

RG 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion"

ANSI/ANS-2.8-1992, "Determining Design Basis Flooding at Power Reactor Sites"

ASCE 1-82, "Guidelines for Design and Analysis of Nuclear Safety-Related Earth Structures"

ASCE 4-86, "Seismic Analysis of Safety-Related Nuclear Structures"

ASCE 7-95, "Minimum Design Loads for Buildings and Other Structures"

For most of the structural materials that will be utilized in the civil-structural systems, there are existing design codes or standards that are based on using allowable stresses or on using a strength approach with load or resistance factors. The list of regulatory guides and industry codes and standards that may have been used by the applicant is provided below:

RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)"

ACI 349-97, "Code Requirement for Nuclear Safety-Related Concrete Structures"

AISC N690-84, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities"

Although some of the above documents apply to nuclear power plants, they provide background information, provide guidance which the staff expects to be applied to the MOX fuel fabrication facilities, and address subjects that the reviewer must verify have been adequately considered by the applicant.

The governing building code should be considered as a guidance document in that it will prescribe absolute minimum requirements for the civil-structural systems independent of the facility requirements resulting from the ISA. Embedded within the building codes or other standards and documents that are incorporated by reference, there can be guidance regarding the design, analysis, construction, and testing portions of all of the elements for consideration described above. Listed below are the major national building codes which may become a single building code in 2000. One of these documents, or a local building code, will govern as the minimum requirement for all civil-structural systems at the facility. These building codes are listed below:

BOCA, Building Officials and Code Administrators International, Inc.

SBC, Southern Building Code Congress International, Inc.

UBC, International Conference of Building Officials

IBC, International Code Council, Inc. (to release the International Building Code 2000 that will replace the three major U.S. building codes)

11.4.6.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's civil-structural systems' design and operation acceptable if it satisfies the requirements listed in Section 11.4.1 and follows the relevant guidelines mentioned under Section 11.4.6.1. The requirements and guidelines for the civil-structural systems are those related to the BDC and defense-in-depth.

If civil-structural systems are involved in the prevention or mitigation of the consequences of any of the events or accidents, there should be a clear linkage identified between the structural performance requirements and the condition or event and its consequence. The magnitude of the parameters defining the site factors and all the accidents, events, and conditions of operation may vary for the different safety categories. It is expected that these values will be related to the results of the ISA. These parameters must then be translated into structural loads, structural movements, or other structural parameters that are capable of providing input to the definition of a physical model of the environment in which the civil-structural system must perform. Some of the parameters are considered as being deterministic while others are considered to be risk-based as a result of the ISA. For example, the return period or frequency of a specific design event should be reflected in, or derived from, the ISA (see Appendix B).

Since many of the input parameters may occur simultaneously, it is necessary that there are identifiable combinations of these parameters in the form of loading combinations defined for the facility. These loading combinations should be linked to the results of the ISA. The resulting load combinations should be clearly identified as representing the unique set of loading functions for the facility at the site that form part of the design bases of the facility. In addition, the reviewer should verify that the minimum requirements of the governing building code for the facility have been incorporated into the design bases. The reviewer should verify the acceptability of the loads, loading conditions, and the analysis models used in the design phase.

The reviewer should verify that the application provides the bases for sizing the various structural elements and members of the civil-structural systems. This aspect of the design basis can be used to quantify the safety margins that may be provided based on the loads and load combinations identified as a result of the ISA for the various operational and accident event scenarios. The reviewer should ascertain how the portion of these safety margins that arise from the sizing of individual structural members have been addressed in Chapter 5.0 of this SRP when the ISA was performed.

The reviewer should verify that a quality assurance program is established for the design, construction, testing, operation, and maintenance of all components of the civil-structural systems that are identified as items relied on for safety in accordance with the criteria in

Chapter 15.1, "Quality Assurance," and Appendix F of this SRP. The reviewer should also ensure that the civil-structural systems and their components, which are identified as items relied on for safety, are adequately designed for natural phenomena including the capability to withstand the effects of tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena in accordance with criteria established in the ISA.

Overall, the reviewer must ensure that all the relevant parameters have been incorporated into the design and the design reflects the supplemental design bases of the other plant systems as well as the requirements of facility operations and the baseline design criteria. Additionally, the civil-structural systems' design and operation should fulfill all of the functional requirements determined from the ISA and the systems should be available with adequate reliability to perform all of their intended safety functions when needed.

11.4.7 Material Transport System (Pumps and Valves)

11.4.7.1 Regulatory Guidance

The regulatory guidance and associated industry standards that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria for the material transport system are:

NRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants." This regulatory guide provides a quality group classification scheme to classify, design, fabricate, and test the material transport system components in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

Although the above document applies to nuclear power plants, it provides background information, provides guidance which the staff expects to be applied to the MOX fuel fabrication facilities, and addresses subjects that the reviewer must verify have been adequately considered by the applicant.

Additional guidance is provided in Department of Energy Standard 1128-98, "Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities," Section C.4.3.1, "Piping and Valves."

11.4.7.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's material transport system's design and operation acceptable if it satisfies the requirements listed in Section 11.4.1 and follows the relevant guidelines mentioned under Section 11.4.7.1. The requirements and guidelines for the material transport system are those related to the BDC and defense-in-depth. Typically, the material transport system is designed to demonstrate:

A. Adequate capacity to handle the expected volume of radioactive material during normal operating and accident conditions;

B. Redundancy or diversity of components required to prevent the release of radioactive materials to the environment or needed for the safe operation of the material transport system;

C. The material transport system can be safely shutdown during normal operations and accident conditions. Provisions for emergency power are included for critical process components.

D. Tank and piping systems are of welded construction to the fullest extent possible.

E. Tank and piping systems are designed to take advantage of gravity flow to reduce the potential for contamination associated with pumping and pressurization.

F. The design of the material transport system assures that accidental criticality will not occur under normal operating conditions or under credible accident conditions.

G. All components of the system expected to be in contact with strong acids or caustics are corrosion resistant.

H. Use of traps is avoided, and the piping is designed to minimize entrapment and buildup of solids in the system.

I. Systems and devices are evaluated to determine the need for hoods, gloveboxes, and shielding for personnel protection. Generally, wet processing operations involving gram quantities of plutonium and operations involving 50 micrograms or more of plutonium in respirable form are conducted in a glovebox. (See Chapter 9.0 of this SRP)

J. Surface finishes in the work area are of materials which have satisfactory decontamination characteristics for their particular application.

K. A quality assurance program is established for the design, construction, testing, operation, and maintenance of all components of the material transport system, that are identified as items relied on for safety, in accordance with the criteria in Chapter 15.1, "Quality Assurance," and Appendix F of this SRP.

L. Material transport system components, which are identified as items relied on for safety, are adequately designed for natural phenomena including the capability to withstand the effects of tornados, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena in accordance with criteria established in the ISA.

Additionally, the material transport system's design and operation should fulfill all of the functional requirements determined from the ISA and the system should be available with adequate reliability to perform all of its intended safety functions when needed.

11.4.8 Heavy Lift Cranes

11.4.8.1 Regulatory Guidance

The regulatory guidance and associated industry standards that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria for heavy lift cranes are:

NRC Regulatory Guide 1.13, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants." This regulatory guide provides design criteria for cranes used in handling heavy loads, specifically refueling casks.

Although the above document applies to nuclear power plants, it provides background information, provides guidance which the staff expects to be applied to the MOX fuel fabrication facilities, and addresses subjects that the reviewer must verify have been adequately considered by the applicant.

ANSI/ANS 57.7-1988, "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)," provides further design criteria:

11.4.8.2 Regulatory Acceptance Criteria

The reviewer should find the applicant's design and operation of cranes for lifting heavy loads acceptable if they satisfy the requirements listed in Section 11.4.1 and follow the relevant guidelines mentioned under Section 11.4.8.1. The requirements and guidelines for heavy lift cranes are those related to the BDC and defense-in-depth. Typically, heavy lift cranes are designed to demonstrate:

A. The handling equipment is designed in accordance with the American National Standard for Overhead Hoists, ANSI/AMSE B30.16-1987.

B. The purchase of equipment and materials is based on the codes and standards which represent a level of capability to meet the design requirements specified in American National Standard Lightning Protection Code, ANSI NFPA 78-1986, and the Specifications for Overhead Traveling Cranes, CMAA 70.

C. Cranes capable of carrying heavy loads are prevented, preferably by design rather than by interlocks, from moving over safety and containment systems.

D. Cranes are designed to provide single failure-proof handling of heavy loads, so that a single failure will not result in loss of capability of the crane-handling system to perform its safety function.

E. The crane structures and their support equipment are designed to withstand all design loads while remaining in place.

F. The crane system design is based on an analysis that considers personnel safety and the confinement of radioactive material under conditions of system failure and misoperation.

G. A quality assurance program is established for the design, construction, testing, operation, and maintenance of all structures and components of the heavy lift cranes, that are identified as items relied on for safety, in accordance with the criteria in Chapter 15.1, "Quality Assurance," and Appendix F of this SRP.

H. Heavy lift crane structures and components, which are identified as items relied on for safety, are adequately designed for natural phenomena including the capability to withstand the effects of tornadoes, tornado missiles, earthquakes, floods, and any other appropriate severe natural phenomena in accordance with criteria established in the ISA.

Additionally, the design and operation of heavy lift cranes should fulfill all of the functional requirements determined from the ISA and the heavy lift cranes should be available with adequate reliability to perform all of their intended safety functions when needed.

11.5 REVIEW PROCEDURES

11.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application (construction or license) adequately addresses the specific items in Section 11.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application for operations is provided below:

A. Application for Construction Approval

Specifically, the application for construction approval should contain the applicant's commitments to provide plant systems which meet or exceed the acceptance criteria in Section 11.4 and should also address the layout and design of the plant systems, their components, and any related information considering the present stage of the applicant's design process.

B. License Application for Operations

Specifically, the license application for operations should address the items described in Section 11.3 in full and update the information provide with the application for construction approval to encompass design modifications and any other system features not adequately described during the construction approval review.

If the primary reviewer verifies that plant systems are adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 11.5.2. 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.

The secondary and supporting reviewers should confirm that the described plant systems are consistent with descriptions in other sections of the application. Information provided for plant systems should be of comparable quality and detail and should not contradict or adversely impact information contained in other sections of the application.

11.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 11.5.1(A) (application for construction approval) or Section 11.5.1(B) (license application for operations), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 11.4. 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 11.4.

Guidance specific to the application for construction approval and the license application for operations is provided below:

A. Application for Construction Approval

The primary reviewer should verify the applicant's commitment to provide plant systems that meet or exceed the acceptance criteria in Section 11.4. The primary reviewer should focus on the layout and design of the plant systems, their components, and any related information considering the present stage of the applicant's design process.

B. License Application for Operations

The primary reviewer should establish that the applicant's plant systems' designs and operations provide reasonable assurance that the plant systems satisfy the acceptance criteria in Section 11.4 and will be available and reliable to perform their intended safety functions when needed. Also the primary reviewer should ensure that adequate documentation is provided in the ISA Summary for all plant systems that are identified as items relied on for safety.

Secondary and supporting reviewers should confirm that the provisions made in the application for plant systems are in accordance and consistent with their specified sections of the SRP. For example, the review performed by the primary reviewer of Chapter 15 of this SRP--as a supporting reviewer--should encompass the adequacy of management measures applied to plant systems. The reviewer of radiation safety under Chapter 9 should evaluate the design and operation of plant systems, such as the ventilation systems and certain instrumentation and controls, with regards to adequate radiation protection. The reviewer of human factors under Chapter 12 should confirm that the principles of human factors engineering are applied to the instrumentation and control systems' design. Also, the primary reviewer of Chapter 5 should determine the adequacy of items relied on for safety (including plant systems) to assure that the likelihood and consequences of identified accidents meet the performance requirements of the proposed 10 CFR 70.61.

For an existing facility being reviewed for a license amendment or renewal, the NRC reviewers may wish to visit the site and facility personnel in order to gain a better understanding of the represented plant systems and their intended safety functions. For a planned facility, the NRC reviewers may wish to meet with the design team in order to gain a better understanding of the process, its potential hazards, and safety approaches.

When the safety evaluation is complete, the primary reviewer--with assistance from the other reviewers--should prepare the plant systems' input for the Safety Evaluation Report (SER) as described in Section 11.6 using the acceptance criteria from Section 11.4. The secondary reviewers should coordinate the plant systems input with the balance of the reviews and the SER.

11.6 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 construction approval by stating that the applicant has committed to providing plant systems which meet or exceed the acceptance criteria in Section 11.4.

The staff could document a safety evaluation for the license application for operations as follows:

The staff evaluated ..... [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Based on the review of the license application, the NRC staff concluded that the applicant's plant systems' designs and operations satisfy the staff's acceptance criteria and are adequately available and reliable to perform their intended

safety functions when needed. In doing so the applicant has satisfactorily addressed the applicable regulatory requirements including the performance requirements, the baseline design criteria, and the defense-in-depth practices contained in the proposed 10 CFR Part 70.

11.7 REFERENCES

A. NUREG-0800, Standard Review Plan, Chapter 8, "Electric Power," Table 8-1, Acceptance Criterial and Guidelines for Electric Power Systems, U. S. Nuclear Regulatory Commission.

B. IEEE Nuclear Power Standards Collection, published by the Institute of Electrical and Electronics Engineers, Inc.

C. NUREG-0800, Standard Review Plan, Chapter 7, "Instrumentation and Controls," U. S. Nuclear Regulatory Commission.

D. NUREG-0800, Standard Review Plan, Chapter 9, "Auxiliary Systems," U. S. Nuclear Regulatory Commission.

12.0 HUMAN FACTORS ENGINEERING FOR PERSONNEL ACTIVITIES

12.1 PURPOSE OF REVIEW

The purpose of this review is to establish that human factors engineering (HFE) is applied to personnel activities identified as items relied on for safety and personnel activities that support the management measures for the items relied on for safety. The application of HFE to personnel activities ensures that the potential for human error in the facility operations was addressed during the design of the facility by facilitating correct, and inhibiting wrong, decisions by personnel and by providing means for detecting and correcting or compensating for error.

For the purposes of this chapter, the phrase "personnel activities" represents personnel activities identified as items relied on for safety and personnel activities that support the management measures for the items relied on for safety, e.g., maintenance.

12.2 RESPONSIBILITY FOR REVIEW

Primary: Human Factors Specialist

Secondary: ISA Reviewer
Primary Reviewer of SRP Section 15.4, "Training"
Primary Reviewer of SRP Section 15.5, "Procedures"
Instrumentation and Control (I&C) Reviewer

Supporting: Fuel Cycle Facility Inspector

12.3 AREAS OF REVIEW

The areas of review for the HFE for personnel activities should include:

A. A description of the personnel actions, the associated human systems interfaces (HSIs), and the consequences of incorrectly performing the action for each personnel activity.

B. The applicant's plans for HFE design review, including the:

   i. Goals and scope;
   ii. Team composition, organizational authority, and responsibilities;
   iii. Process and procedures;
   iv. Issues tracking; and
   v. Functional description.

C. Operating experience review;

D. Function and task analysis;

E. HSI design, inventory and characterization;

F. Staffing;

G. Procedure development;

H. Training program development; and

I. Human factors verification and validation (V&V).

All nine areas of review (A-I) may not be necessary for a specific application. The areas of review should be based on the applicant's provisions to address personnel activities consistent with the findings of the ISA; the similarity of the associated HFE issues for similar type plants; and the determination of whether an item relied on for safety has special or unique safety significance.

12.4 ACCEPTANCE CRITERIA

12.4.1 Regulatory Requirements

The regulatory requirements for HFE for personnel activities are:

A. Proposed 10 CFR 70.61(e), which requires a safety program to ensure that each item relied on for safety will be available and reliable to perform its intended function when needed.

B. Proposed 10 CFR 70.62(d), which requires the establishment of management measures.

C. Proposed 10 CFR 70.64(b)(2), which requires features that enhance safety by reducing challenges to items relied on for safety.

12.4.2 Regulatory Guidance

There are no regulatory guides that apply to human factors engineering for personnel activities for a mixed oxide (MOX) fuel fabrication facility.

12.4.3 Regulatory Acceptance Criteria

The HFE for personnel actions should be acceptable if:

A. Identification of Personnel Activities

The applicant appropriately identified the personnel activities such that the reviewer can understand the actions, the HSIs involved, and the consequences.

B. HFE Design Review Planning

The applicant's approach for planning HFE design review, includes:

i. Identification of appropriate goals and scope to ensure that HFE practices and guidelines are implemented during design, construction, and operation of the facility.

ii. Implementation by an HFE team that has the appropriate composition, experience, and organizational authority to ensure that HFE is considered in the design of HSI for personnel activities. The HFE team's responsibilities include ensuring the proper development, execution, oversight, and documentation of the HFE function. Depending on the identification of personnel activities, it may be appropriate for the HFE team to be comprised of a single individual.

iii. An HFE team that attains the HFE goals and scope through established processes and procedures and tracks HFE issues.

iv. An HFE function that ensures that all aspects of the personnel activities including the HSI are developed, designed, and evaluated on the basis of a structured approach using HFE.

C. Operating Experience Review (OER)

The applicant identified HFE-related events or potential events that have occurred in existing facilities that are similar to the proposed facility. The applicant:

i. Reviewed the HFE-related events or potential events for relevance;

ii. Analyzed the HSI technology employed for the relevant HFE events or potential events; and

iii. Conducted (or reviewed existing) operator interviews and surveys on the HSI technology for the relevant HFE events or potential events.

D. Functional Allocation Analysis and Task Analysis

i. Functional allocation analysis: The function allocation analysis is based on the OER. Personnel activities are functionally allocated to take advantage of human strengths and avoid demands that are not compatible with human capabilities.

ii. Task analysis: The task analysis includes the task analysis scope, identification and analysis of critical tasks; detailed description of personnel demands (e.g., input, processing, and output); iterative nature of the analysis; and incorporation of job design issues. The task analysis addresses each operating mode for each personnel activity

(e.g., start-up, normal operations, emergency operations, shutdown). The task analysis results support the functional allocation.

E. HSI Design, Inventory, and Characterization

The HSI design incorporates the functional allocation analysis and task analysis into the detailed design of HSI components (e.g., alarms, displays, controls, and operator aids) through the systematic application of HFE. The HSI design includes the overall work environment, the work space layout (e.g., control room and remote shutdown facility layouts), the control panel and console design, the control and display device layout, and information and control interface design details. The HSI design process ensures the application of HFE to the HSI required to perform personnel activities. The HSI design process excludes the development of extraneous controls and displays. The HSI design documentation includes a complete HSI inventory and the basis for the HSI characterization.

F. Staffing

Staffing is based on a review of the number and qualifications of personnel for each personnel activity during all plant operating conditions. The applicant conducts this review in a systematic manner that incorporates the functional allocation and task analysis results. The categories of personnel are based on the types of personnel activities. Staffing considerations include issues identified in the OER, functional allocation, HSI design, procedure development, and V&V.

G. Procedure Development

The applicant's procedure development for personnel activities incorporates HFE principles and criteria, along with all other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, and validated consistent with the acceptance criteria in Section 15.5 of this SRP. Because procedures are considered an essential component of the HSI design, they are derived from the same design process and analyses as the other components of the HSI (for example, displays, controls, operator aids) and subject to the same evaluation processes. Procedures include, as needed to support the personnel activity: generic technical guidance, plant and system operations, abnormal and emergency operations, tests (for example, preoperational, startup, and surveillance), and alarm response.

H. Training Program Development

The applicant's training program development addresses all personnel activities. The training program development indicates how the elements of a systems approach to training will be incorporated into the training program, how the knowledge and skill requirements of personnel will be evaluated, how the training program development is coordinated with the other activities of the HFE design process, and how the training program will be implemented in an effective manner consistent with human factors principles and practices.

The training program development results in a training program that provides personnel with the qualifications commensurate with the personnel activities. The training program development addresses the applicable acceptance criteria provided in Section 15.4.

I. Verification&Validation

V&V confirms that the design incorporates HFE to HSI that enables the successful completion of personnel activities. The V&V should be applied to personnel activities (see Item A) and HSI design (see Item E). The V&V process should consist of the following:

i. HSI task support verification: HSI components are appropriately provided for personnel activities through HSI task support verification. The verification shows that each HSI identified the task analysis (see Item D(ii)) and the HSI design (see Item E) is appropriately provided, yet minimizes the incorporation of information, displays, controls, and decorative features that unnecessarily complicate personnel activities.

ii. HFE design verification: The HFE design verification shows that each HSI identified for a personnel activity incorporated HFE into the design. Deviations from accepted HFE principals and guidelines should be justified or documented for resolution/correction. If all HSI components are not addressed by HFE design verification, then an alternative multidimensional sampling methodology should be used to assure comprehensive consideration of the safety significance of HSI components. The sample size should be sufficient to identify a range of significant safety issues.

iii. Integrated system validation: The applicant commits to a performance-based evaluation of the integrated design to ensure that the HFE/HSI supports safe operation of the plant. Integrated system validation is performed after HFE problems identified in HFE design activities are resolved or corrected because these may negatively affect performance and, therefore, validation results. Validation is performed by evaluating personnel activities using appropriate measurement tools. All personnel activities should be tested and found to be adequately supported in the design, including personnel activities outside the control room.

iv. Human factors issue resolution verification: The applicant verifies that HFE issues identified during the design process were addressed and resolved. Issue resolution verification should be documented in the HFE issue tracking system established by the HFE team (see Item B). Issues that can not be resolved until the HSI design is constructed, installed, and tested should be identified and incorporated into the final HFE/HSI design verification.

v. Final HFE/HSI design verification: The applicant should commit to perform a final HFE/HSI design verification if the applicant can not demonstrate that it has fully evaluated the actual installation of the final HSI design in the plant through the V&V activities described above. Final HFE/HSI design verification should demonstrate that in-plant HFE design implementation conforms to the HFE design (see Item E) as modified V&V activities.

V&V activities should be performed in the order listed above, as necessary. However, the applicant may find that it is necessary to iterate in order to address design corrections and modifications that occur during V&V.

12.5 REVIEW PROCEDURES

12.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval or license application adequately addresses the items in Section 8.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the safety assessment of the design basis should address Section 12.3(A)-(E) consistent with the level of design and the consequences of incorrectly performing the personnel activity consistent with the safety assessment of the design basis. 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 Application

Specifically, the safety assessment of the license application should fully address Section 12.3(A)-(I) consistent with the consequences of incorrectly performing the personnel activity.

If the primary reviewer verifies that the HFE for personnel activities is adequately addressed (construction or license), the primary reviewer should accept the application for the safety evaluation in Section 12.5.2. 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.

12.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 12.5.1.A (construction) or 12.5.1.B (operations), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 12.4. 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 12.4.

The primary reviewer should use a tiered approach for evaluating HFE for personnel activities. The upper tier is the program description level, e.g., missions or goals. The middle tier is when functions are allocated to tasks (personnel activities) for the purposes of specifying the alarms, information, and controls. The tasks are arranged into meaningful jobs and the HSI should be designed to best support job task performance. The lower tier is the detailed design (of the HSI, procedures, and training) and how they are incorporated into the facility design. Evaluation of the HFE design should be broad-based and include aspects of normal and emergency operations, testing, maintenance, etc., consistent with findings in the safety assessment of the design basis (application for construction approval) or in the ISA Summary (license application for operations).

Guidance specific to the application for construction approval and the license application for operations is provided below.

A. Application for Construction Approval

In general, the primary reviewer should perform an upper tier review for the safety assessment of the design basis. As the level of design permits, the primary reviewer should perform a middle tier review on those personnel activities that are identified as preventing or mitigating accident consequences.

B. License Application for Operations

In general, the primary reviewer should perform a lower tier review for personnel activities that prevent or mitigate "high-consequence" events, a middle tier review for personnel activities that prevent or mitigate "intermediate-consequence" events, and a high-level review for any remaining HFE activities.

The primary reviewer should review the ISA Summary to ensure personnel activities have been suitably characterized as items relied on for safety. The extent that HFE elements are applied should be based on the number, type, complexity, and potential consequences of the personnel activities.

The secondary reviewer should ensure that the types of personnel activities relied on for safety are appropriate. The primary reviewer should coordinate with the I&C reviewer for Chapter 11.0, "Plant Systems," to confirm that HFE principles are appropriately addressed in the I&C approach.

The supporting reviewers should assist in the tiered approach of the review in that they may look at more specific examples of human factors engineering application.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the HFE input for the Safety Evaluation Report (SER), as described in Section 12.6 using the acceptance criteria from Section 12.4. The secondary reviewer should coordinate the chemical safety input with the balance of the reviews and the SER.

12.6 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 application for construction approval as follows

The staff reviewed the application of human factors engineering to personnel activities for the application for construction approval for [insert facility name] according to Chapter 12.0 of NUREG-1718. The staff evaluated [insert a summary statement of the evaluation] and found [insert a summary of the findings].

The staff concluded that the applicant has established an adequate design basis, as it relates to HFE, that meets the requirements for construction approval in 10 CFR Part 70.

The staff could document the safety evaluation for the license application for operations as follows:

The staff reviewed the application of human factors engineering (HFE) to personnel activities for the license application to possess and use SNM at [insert facility name] according to Chapter 12.0 of NUREG-1718. The staff evaluated [insert a summary statement of the evaluation] and found [insert a summary of the findings].

The staff concluded that the applicant applied HFE to personnel activities identified as items relied on for safety and personnel activities that support the management measures for the items relied on for safety and that its personnel activities meet the requirements associated with human factors given in 10 CFR Part 70.

12.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

C. NUREG-0700, Rev.1, Vol.1-3, *Human-System Interface Design Review Guideline*, U.S. Nuclear Regulatory Commission, June 1996.

D. NUREG-0711, *Human Factors Engineering Program Review Model*, U.S. Nuclear Regulatory Commission, July 1994.

E. MIL-STD-1472D, *Human Engineering Design Criteria for Military Systems, Equipment and Facilities*, March 1989.

13.0 SAFEGUARDS
13.1 PHYSICAL PROTECTION

13.1.1 PURPOSE OF REVIEW

The purpose of this review is to determine with reasonable assurance that the applicant has committed to having a physical protection system that provides high assurance that activities involving special nuclear material (SNM) are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety. The physical protection system should be designed to protect against the design basis threats of theft or diversion of formula quantities of strategic special nuclear material (SSNM) and radiological sabotage as stated in 10 CFR 73.1(a). Physical protection requirements for applicants possessing formula quantities of SSNM are found in 10 CFR 73.20, 73.45, and 73.46.

13.1.2 RESPONSIBILITY FOR REVIEW

Primary: Physical Protection Specialist

Secondary: None

Supporting: Regional Physical Protection Inspector

13.1.3 AREAS OF REVIEW

The reviewer should review the applicant's submittal for an acceptable physical protection system that protects against the design basis threats of both theft or diversion of formula quantities of SSNM and radiological sabotage. The reviewer should ensure that the applicant has described how the general performance requirements of 10 CFR 73.20, the performance capabilities outlined in 10 CFR 73.45, and the specific measures included in 10 CFR 73.46 will be met through development, implementation, and maintenance of a physical protection system.

13.1.4 ACCEPTANCE CRITERIA

13.1.4.1 Regulatory Requirements

Specific references are as follows:

A. In 10 CFR 73.20, the general performance objective and requirements for a fixed site physical protection system are defined.

B. In 10 CFR 73.45, the performance capabilities for fixed site physical protection systems are defined.

C. In 10 CFR 73.46, specific measures for fixed site physical protection systems, subsystems, components, and procedures are detailed.

D. Appendices B, C, G, and H to 10 CFR 73.46 provide additional requirements applicable to the MOX facility.

13.1.4.2 Regulatory Guidance

The regulatory guidance for physical protection includes:

A. Regulatory Guide 5.52, Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants), Rev. 3, December 1994.

B. Regulatory Guide 5.55, Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities (for comment), March 1978.

C. Regulatory Guide 5.44, Perimeter Intrusion Alarm Systems, Rev. 3, October 1997.

D. INFCIRC/225/ Rev. 4 (Corrected), The Physical Protection of Nuclear Material and Nuclear Facilities, June 1999.

E. NUREG-BR0252, User's Guide to Physical Protection Documents Published by the NRC, November, 1998.

13.1.4.3 Regulatory Acceptance Criteria

The reviewer will find the applicant's physical protection system acceptable if the physical protection plan commitments are consistent with 10 CFR 73.20, 73.45 and 73.46. The physical protection plan for the mixed oxide (MOX) facility shall contain inspectable commitments which shall be the basis for the NRC physical protection inspection program. Therefore, it is imperative that commitments be expressed in unambiguous terms. NRC has determined that public disclosure of the details of the physical protection system for a MOX facility could impact on common defense and security and should be classified as Confidential, National Security Information.

13.1.4.3.1 Introduction and Schedule for Implementation

The applicant should state its corporate name, the facility name, and the location of the facility. The applicant should describe the MOX facility and the type of SNM that will be utilized, its general layout, its surrounding area and the surrounding terrain. The reviewer should ensure that the applicant has included a map of the entire facility and other maps and illustrations, as appropriate. The applicant should indicate on these maps the owner controlled area; the location of all buildings; the locations of physical protection systems, subsystems, and major components; the protected area and all entry/exit points; vehicle barriers; all material access

areas; vital (if applicable) areas; controlled access areas; vaults; entry/exit control points; alarm stations; security posts; and response force staging areas.

The applicant should describe the schedule for implementing the physical protection plan. SSNM may not be stored or used at the MOX facility until the physical protection system is fully implemented and operational.

13.1.4.3.2 General Performance Objectives

The reviewer will determine that the applicant's commitments in this section are consistent with §73.46. In addition, the reviewer should verify the following:

The applicant has described, in general terms, how the physical protection system will have as its objective to provide high assurance that activities involving SNM are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

The applicant has described how, through the development, implementation, and maintenance of a physical protection system, the general performance objective and requirements outlined in 10 CFR 73.20 and the performance capability requirements of 10 CFR 73.45 will be met.

Further, the reviewer should ensure that the applicant has identified and described those portions of the physical protection system for which redundant and diverse components, as well as redundant and diverse subsystems and components, are necessary in order to ensure adequate performance, as required by 10 CFR 73.20(b)(2). In general terms, the applicant should describe the subsystems and components to be used to provide this redundancy and diversity and the ways in which these subsystems and components are redundant and diverse.

Finally, the reviewer should verify that the applicant has described how the physical protection system is designed, tested, and maintained to ensure its continual effectiveness, reliability, and availability. This verification should be conducted onsite by the reviewer prior to plan approval.

13.1.4.3.3. Design Basis Threat (10 CFR 73.1(a))

The applicant has affirmed the intent to protect against the design basis threats of both theft or diversion of formula quantities of SSNM and radiological sabotage, as described in 10 CFR 73.1(a). For a MOX fuel fabrication facility, it is important that the physical protection system be designed both to protect against radiological sabotage, as well as to prevent theft of formula quantities of SSNM. With respect to radiological sabotage, the applicant is expected to establish a defensive strategy which would deny unauthorized access to areas of the plant which contain plutonium. The reviewer should ensure that the applicant has committed to maintain and update the physical protection plan to reflect any changes that are necessary to ensure the continual ability to protect against the design basis threats.

13.1.4.3.4 Security Organization (10 CFR 73.46(b))

The performance objective of the security organization is to manage, control, and implement the physical protection system in a manner that is consistent with the physical protection plan and continually maintains its effectiveness. The reviewer should ensure that the applicant has clearly described the security organization that will be used at the facility. The security organization should be acceptable if the applicant's commitments are consistent with the requirements in §73.46(b), and associated Appendices B, C, G, and H of 10 CFR 73, and the following criteria:

A. The applicant has stated whether the security organization is employed directly by the applicant or is a contractor to the applicant. If a contractor, the reviewer should ensure that the applicant described the written agreements between the applicant and contract guard force management that pertain to how the guard force will meet NRC's requirements in §73.46(b)(1) and in Appendix B, "General Criteria for Security Personnel," and Appendix H, "Weapons Qualification Criteria," to 10 CFR Part 73.

B. The applicant has described the structure and management of the security organization, including both uniformed security personnel and other persons responsible for security-related functions, consistent with §73.46(b)(1). This discussion should include a description of each supervisory and management position, including responsibilities and lines of authority to facility and corporate management.

C. The applicant has affirmed that at least one full-time member of the security organization will be onsite at all times with the authority to direct the physical protection activities of the security organization, consistent with §73.46(b)(2). The plan should also affirm that written security procedures will be used and provisions for written approval of such procedures, and any revision thereto, are developed and used, consistent with §73.46(b)(3).

D. The applicant has affirmed that an approved Guard Force Training Plan, in accordance with Appendix B to Part 73 will be in effect. The physical protection plan should commit to having all members of the security organization trained, equipped, and qualified to perform each assigned security duty per 10 CFR Part 73 Appendix B and Appendix H as appropriate, consistent with §73.46(b)(4).

E. The applicant has described how the security personnel, licensee employees or contractor employees will carry out their assigned duties or responsibilities upon the request of the NRC. The applicant should also affirm that, within any given period of time (e.g., at least one work shift or 8 hours), a member of the security organization will not be assigned to or have direct operational control over more than one of the redundant elements of a physical protection subsystem, if such assignment or control could result in the loss of effectiveness of the subsystem, consistent with §73.46(b)(5).

F. The applicant has affirmed that every guard, armed response person, and Tactical Response Team (TRT) member will be armed and should describe the armament assigned

to members of the security force by position title, consistent with §73.46(b)(6). The applicant should include a description of the qualification and requalification program for guard and TRT members in night firing with assigned weapons, and, for TRT members only, a description of the training program in response tactics, consistent with §73.46(b)(7) and (8). In addition, equipment to be used by members of the security force in providing effective response capabilities should also be described.

G. The applicant has described how scenarios for force-on-force exercises are developed, the design goals for conducting such exercises, and the frequency of exercises. The applicant should affirm that as a licensee it will permit NRC to observe one force-on-force exercise each year and that the NRC will receive a 60-day notice of the planned exercise, consistent with §73.46(b)(9).

H. The applicant has affirmed that the records required by §73.46(b)(3)(i), (4), (7), (8) and (9) will be maintained/retained and has described how they will be maintained/retained.

I. The applicant has described the physical fitness training program and medical examination for each guard, armed response person or TRT member consistent with §73.46(b)(10)-(12) to ensure that these personnel are able to perform their assigned duties under conditions of strenuous tactical engagements.

13.1.4.3.5 Physical Barrier Subsystems (10 CFR 73.46(c))

A performance objective of physical barriers is to define areas within which authorized activities and conditions are permitted. Other performance objectives of barriers are to channel persons, vehicles and material to or from entry/exit control points; to delay or deny unauthorized penetration attempts by persons, vehicles or material; and to delay any unauthorized SSNM removal attempts sufficient to assist detection and assessment and permit a timely response by the security force to prevent the intended act. The reviewer should ensure that the applicant has clearly described the physical barrier subsystems that will be used at the facility. This section should be acceptable if the applicant's commitments are consistent with the requirements in §73.46(c) and the following criteria:

A. The applicant has described the facility's protected, controlled access, material access, and vital (if applicable) area barriers, discussed the purpose of each barrier, and described the spatial relationship between the protected area and material access or vital areas, consistent with §73.46(c)(2).

B. The applicant has affirmed that the perimeter of the protected area will be provided with two physical barriers, as defined in §73.2. The inner barrier must be positioned, constructed and maintained to enhance assessment of penetration attempts and to delay attempts at unauthorized exit from the protected area, consistent with §73.46(c)(1). The applicant should commit to installing the protected area barrier fence so that it cannot be lifted to allow an individual to crawl under. The applicant should describe any access points in the protected area barrier, their use, and how they are controlled and protected to ensure the integrity of the barrier.

C. The applicant has described the location and size of all isolation zones at the facility. Affirmation should be given that the isolation zones adjacent to the physical barriers at the perimeter of the protected area should be at least 6.1 m (20 ft) wide and be maintained clear of obstacles or structures on either side of the barriers to permit assessment, consistent with §73.46(c)(3).

D. The applicant has affirmed that the location and placement of vehicle barriers will provide protection against radiological sabotage by the design basis explosive (classified) or the use of a vehicle for transporting personnel and their equipment into the protected area to aid in the theft of SSNM. The physical description of the barrier system should be included, along with a commitment that the barrier can adequately counter the design basis vehicle (classified), consistent with §73.46(c)(1). If other than a commercially available barrier is used, any testing conducted to validate the penetration resistance of the barrier should be discussed.

E. The applicant has described the lighting system provided to ensure illumination for all required monitoring, observation, and assessment activities for all exterior areas within the protected area. The commitment for illumination should be not less than 2.15 lumen per meter^2 (0.2 footcandle) measured horizontally at ground level, consistent with §73.46(c)(4). The applicant should discuss emergency backup power for protected area lighting and assessment capability if normal power is lost.

F. The applicant has described the purpose of each process material access area at the facility and the protection afforded SSNM (other than alloys, fuel elements, or fuel assemblies) while in these material access areas. Both physical and procedural protective measures should be described, consistent with §73.46(c)(5).

G. The applicant has affirmed that physical barrier subsystems will be in place to assure that SSNM is stored or processed only in a material access area, vital equipment is located only within a vital area, and both vital and material access areas are located within a protected area. Physical barriers will be maintained for a vital or material access area which are separated from any physical barrier at the perimeter of the protected area. The applicant should describe the level of physical hardening for the wall, floors, and ceilings of these areas. The number, location and types of entry/exit portals should be described. Methods used to provide hardening of the portals (during opened and closed conditions) should be described. Hardening for ventilation and other openings greater than 619.4 cm^2 (96 square inches), with the smaller dimension of 15.2 cm (6 inches) or greater, should be described. Access to vital equipment or SSNM will require passage through at least three physical barriers.

H. The applicant has affirmed that SSNM, other than alloys, fuel elements, or fuel assemblies, shall be stored in a vault when not undergoing processing if the material can be used directly in the manufacture of a nuclear explosive device, consistent with §73.46(c)(5). The applicant should describe the purpose; the construction of the walls, ceiling and floor; and the location and type of entry portal to each vault. The penetration delay time for the vault

should be estimated by the applicant based on the vault construction method and construction materials and considering penetration by tools and by explosive techniques. Affirmation should be given that the penetration delay time will be greater than the time required for the TRT to respond.

I. The applicant has described the construction and use of tamper-indicating containers for the storage of SSNM (other than alloys, fuel elements, or fuel assemblies), consistent with §73.46(c)(5).

J. The applicant has described how fuel elements and fuel assemblies will be stored and protected.

13.1.4.3.6 Access Control Subsystems and Procedures (10 CFR 73.46(d))

The performance objective of access authorization controls and procedures is to provide current authorization lists and entry criteria. The performance objectives of entry controls and procedures are to verify the identity of persons, vehicles, and materials and to assess such identity against current authorization lists and entry criteria before permitting entry, and to initiate timely response measures to deny unauthorized entries. The reviewer should ensure that the applicant has clearly described the access control subsystems that will be used at the facility. This section should be acceptable if the applicant's commitments are consistent with the requirements in §73.46(d) and the following criteria:

A. The applicant has described the numbered picture badge identification system used at the facility, consistent with §73.46(d)(1). This description should include a discussion of procedures used for badging individuals authorized access to the protected area and for individuals not employed by the applicant, but who require frequent and extended access to the protected area. Instructions that badged individuals receive in proper badge procedures should also be discussed, along with procedures for accommodating non-badged emergency response individuals during emergency situations. Verification of authorization can be accomplished by use of systems such as bio-metrics, personal identification numbers, card readers, or combinations thereof. Badges should not be taken off-site unless the applicant commits to using a highly reliable method of verifying personal identity such as bio-metrics. The applicant should affirm that blank badge material will be controlled. The applicant should affirm that the badge of an employee terminated for cause should be immediately retrieved or deleted from the computerized access system

B. The applicant has affirmed that badges will be required to be displayed by all individuals while inside the protected area, consistent with §73.46(d)(1).

C. The applicant has committed to procedures for determining an individual's need for access to a vital area, material access area, or controlled access area; procedures for the distribution and maintenance of lists of authorized individuals; procedures for ensuring the maintenance of the two-man rule within material access areas and vaults; procedures for ensuring that no activities other than those that require access to SSNM or necessary maintenance are permitted within material access areas; and methods used to visually

identify individuals authorized unescorted access to vital areas, material access areas, or controlled access areas, consistent with §73.46(d)(2). This discussion should note differences in procedures, if any, between working and non-working hours (nights, weekends, and holidays) and normal versus emergency conditions. The applicant should commit that access to material access areas and vaults requires a minimum of two individuals to be present. The applicant should commit to only allowing unescorted access to vital, material access or controlled access areas to individuals with a government security clearance and a need-to-know.

D. The applicant has described how it will control all points of personnel access into the protected area, under both normal and emergency conditions, consistent with §73.46(d)(4). This description should include a discussion of methods used to identify individuals and to verify individuals' authorization; methods used to verify emergency conditions; and procedures for conducting searches of individuals for firearms, explosives, and incendiary devices. The search function for detecting both firearms and explosives must use detector equipment. The equipment used should represent the current state-of-the-art equipment that is commercially available. The capabilities of the search equipment should be described. The applicant should also describe what actions it takes, including the use of pat-down searches, if it suspects an individual of trying to introduce contraband into the protected area or if the search equipment is not operating satisfactorily. The applicant should describe how it will determine that the equipment is operating properly. The applicant should commit to having the individual responsible for the last access control point prior to entering the protected area to be protected by a bullet-resisting structure hardened to at least the Underwriters' Laboratories, Inc. (UL)752/Class IV level and preferably to the 7.62 mm level of protection.

E. Individuals exempted from any of the aforementioned access controls should be identified. The distribution and maintenance of authorization lists should also be described.

F. The applicant has affirmed that it will establish and follow written procedures that will permit access-control personnel to identify materials in hand-carried packages that are not authorized entry to the protected area, during both normal and emergency conditions, consistent with §73.46(d)(5). Further, the applicant should describe procedures for searching hand-carried packages at personnel and vehicle access points for firearms, explosives, and incendiary devices.

G. The applicant has affirmed that it will establish and follow written procedures that will permit access-control personnel to identify materials in delivered packages that are not authorized entry to the protected area during both normal and emergency conditions, consistent with §73.46(d)(6). Further, methods used to check for proper identification and authorization should be described along with search procedures for firearms, explosives, and incendiary devices. Any activities exempted from the above procedures should be described. The development, distribution, and maintenance of authorized (or unauthorized) materials lists should be described.

H. The applicant has described procedures used for controlling all points of vehicle access (non-emergency and emergency) into the protected area and how written procedures are established and followed that will permit access-control personnel to identify vehicles that are authorized entry to the protected area, consistent with §73.46(d)(3). The distribution and maintenance of these procedures should be described. Search procedures of all vehicles requiring entry to the protected area for firearms, explosives, and incendiary devices should also be described. Any vehicles exempt from the aforementioned procedures should be described, consistent with §73.46(d)(7). Procedures used in escorting vehicles within the protected area, and areas where vehicles may have access, along with the purpose for the access, should be discussed.

I. The applicant has described the control and use of designated licensee vehicles within the protected area, consistent with §73.46(d)(8).

J. The applicant has described the methods it proposes to use to control all points of personnel access to material access areas, vital areas, and controlled access areas, including methods used to verify identification and authorization, consistent with §73.46(d)(9). The applicant shall affirm that at least two armed and appropriately trained guards shall be posted at each material access area control point whenever in use. Personnel exit searches from material access areas should also be discussed, and the applicant should affirm that at least two individuals, who are not authorized access to that material access area, will conduct separate, independent searches for concealed SSNM. The applicant should affirm that material access area exit searches for SNM and metal can detect standards, consistent with NRC classified criteria.

K. The applicant has described procedures for verifying material entry authorizations and procedures for verifying quantity and type of material, consistent with §73.46(d)(9). These descriptions should include the components to be used in the detection of unauthorized materials that are hand-carried by authorized individuals, or mailed or otherwise shipped, as part of an authorized shipment. The applicant has described how normal conditions differ between regular working hours and non-working hours (nights, weekends, and holidays).

L. The applicant has described methods used to control all points of vehicle access (non-emergency and emergency) to material access areas, vital areas, and controlled access areas, including the establishment and maintenance of written procedures that will permit access control personnel to identify those vehicles that are authorized entry to material access and vital areas, consistent with §73.46(d)(9). Vehicle exit searches should also be described, and the applicant should affirm that searches will be conducted by a team of at least two individuals.

M. The applicant has described procedures and areas used for searching contaminated wastes coming from a material access area, consistent with §73.46(d)(10).

N. The applicant has described containers, procedures, and areas used for shipping SSNM offsite, consistent with §73.46(d)(11) and (12). The applicant should affirm that the packaging and shipping process will be conducted by a team of at least two individuals.

O. The applicant has described individuals, by job function, who may be designated as escorts, and procedures used for escorting individuals during both routine and emergency situations. Such procedures should describe individuals requiring escort, escort/visitor ratios, badging procedures, and escort training and record-keeping, consistent with §73.46(d)(13). The applicant should commit to a maximum escort/visitor ratio of at least one escort to five visitors.

P. The applicant has described procedures for controlling all keys, locks, combinations, and related equipment used to control access to protected, material access, vital, and controlled access areas. The discussion should describe the circumstances under which such keys, locks, etc., are changed and procedures followed when the employment of an employee with access to such keys, locks, etc., is terminated, consistent with §73.46(d)(14). The applicant should commit to changing keys, locks, combinations, and related equipment at least when there is evidence of compromise to any of the items that a terminated employee had access.

Q. The applicant has committed to controlling information regarding the presence of NRC safeguards inspectors, consistent with §73.46(d)(15).

R. The applicant should describe record-keeping procedures for: (1) current written procedures that permit access-control personnel to identify vehicles that are authorized and those materials that are not authorized entry to protected, material access and vital areas; (2) findings of drum-scanning and tamper-sealing of containers of contaminated wastes exiting from material access areas; and (3) the required log of escorted individuals, consistent with §73.46(d)(3), (10) and (13).

13.1.4.3.7 Detection, Surveillance and Alarm Subsystems and Procedures (10 CFR 73.46(e))

The performance objectives of detection, surveillance, and alarm subsystems and procedures are to detect, assess, and communicate any unauthorized access or penetrations or such attempts by persons, vehicles, or materials at the time of the act or the attempt so that the response can be such as to prevent the unauthorized access or penetration. The reviewer should ensure that the applicant has clearly described the detection, surveillance, and alarm subsystems that will be used at the facility. This section should be acceptable if the applicant's commitments are consistent with the requirements in §73.46(e) and the following criteria:

A. The applicant has described the intrusion detection system that will be installed in the isolation zone between the two barriers at the protected area perimeter, consistent with §73.46(e)(1). The applicant should commit to providing a volumetric intrusion detection system, which is capable of detecting an individual weighing a minimum of 35 kg (77 lbs), whether the individual is running, walking, crawling, jumping, or rolling through the isolation zone of the protected area. The capabilities, installation, and testing of the intrusion detection equipment should be consistent with Regulatory Guide 5.44, Revision 3.

B. The applicant has described the location of all emergency exits and described the protection afforded them, consistent with §73.46(e)(2). The applicant should commit to maintaining all emergency exits in the protected, material access, vital and controlled access areas locked to prevent entry from outside and equipping them with local audible and visible alarms.

C. The applicant has described the protection and surveillance afforded: (1) unoccupied material access and vital areas; (2) the location of SSNM within process material access areas; (3) vaults and process areas that contain SSNM that has not been alloyed or encapsulated, including a description of procedures for access to these particular vaults and process areas, consistent with §73.46(e)(3). Equipment that is used to provide this protection, along with associated detection capabilities, should be described. The applicant should commit to having all unoccupied material access areas where plutonium is located equipped with volumetric intrusion detection equipment and closed circuit television (CCTV) for remote assessment. The applicant should affirm that access to unoccupied vaults and process areas requires that an individual other than the alarm station operator be present or have knowledge of access.

D. The applicant has described how all security stations and individuals (by job position), consistent with §73.46(e)(4), will be provided with duress alarms. The type of duress alarms used, where they are monitored, and emergency backup power should be described.

E. The applicant has described the location, construction, and characteristics of the central and secondary alarm stations, consistent with §73.46(e)(5). The applicant should commit to having all required alarms annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned independent onsite station. Continuous manning of alarm stations and methods used for annunciation of required alarms should be described, along with protection afforded the stations (both procedural and physical), so that a single act cannot remove the capability of calling for assistance or responding to an alarm. Affirmation also should be provided that the alarm stations are bullet-resisting to at least the UL 752/Class IV level, and preferably to the 7.62 mm level of protection. If other than commercially available armoring material is utilized, any testing or engineering studies conducted to validate the penetration resistance of the barrier should be described. Affirmation should be given that access to the alarm stations is controlled on a strict need-to-know basis and the central alarm station not contain any operational activities that would interfere with the execution of alarm response functions. The applicant should describe the annunciation systems at the alarm stations and commit to indicating the status of all alarms and alarm zones in both alarm stations.

F. The applicant has described (i) how detection equipment and alarm annunciation shall remain operable from independent emergency power sources, (ii) duration of operation in the event of loss of normal power, and (iii) indications given upon loss of normal power and transfer to standby power, consistent with §73.46(e)(6). The applicant should also affirm that switch over to standby power will be automatic and not cause false alarms.

G. The applicant has described the physical protection afforded alarm systems, including transmission media, to ensure that the system is not being tampered with, compromised, or on standby power without the knowledge of the licensee, consistent with §73.46(e)(7). The applicant should affirm that all tamper alarms will annunciate in either the access or secure mode.

H. The applicant has described methods used to monitor all exterior areas within the protected area for unauthorized persons, vehicles, materials, and activities and the duration or periodicity of such monitoring, consistent with §73.46(e)(8). Criteria used in defining authorized and unauthorized activities and conditions within the protected area should be described, along with methods for developing, maintaining, and distributing lists of authorized activities and conditions. The applicant should commit to monitoring or conducting random patrols within the protected area at least several times each shift.

I. The applicant has described methods used to observe individuals within material access areas, to assure that SSNM is not moved to unauthorized locations or moved in an unauthorized manner, consistent with §73.46(e)(9). The duration or periodicity of such monitoring should be described along with criteria used in defining authorized and unauthorized activities and conditions within the material access area. Methods for developing, maintaining, and distributing lists of authorized activities and conditions should be described. The applicant should commit to using CCTV to observe these areas periodically during working hours and for remote access during non-working hours.

13.1.4.3.8 Communication Subsystems (10 CFR 73.46(f))

The performance objective of communication subsystems is to provide for notification of an attempted unauthorized or unconfirmed removal of SSNM or attempted act of radiological sabotage so that response can be such as to prevent the unauthorized act. The reviewer should ensure that the applicant has clearly described the communication subsystems that will be used at the facility. The communication subsystems should be acceptable if the applicant's commitments are consistent with the requirements in §73.46(f) and the following criteria:

A. The applicant has described how each guard, watchman, armed response person, or TRT member on duty will be capable of maintaining continuous communications with the individual in each continuously manned alarm station, consistent with §73.46(f)(1). The applicant should also describe how the individual in each continuously manned alarm station will be capable of calling for assistance from other guards, watchmen, armed response personnel, or TRT members and from local law enforcement authorities.

B. The applicant has described the redundant and diverse systems used to ensure the capability of communications with the local law enforcement authority, consistent with §73.46(f)(2). Cellular phone service may be an acceptable alternative method of communications if the service is reliable and provides complete coverage of the area of concern.

C. The applicant has described methods used to keep the non-portable communications equipment it uses operable in the event of loss of normal power, consistent with §73.46(f)(3). The applicant should discuss the length of time the equipment will operate on the emergency power source. All sources of emergency power should be protected and located within the protected area.

13.1.4.3.9 Test and Maintenance Programs (10 CFR 73.46(g))

The performance objective of test and maintenance is to provide confidence that security equipment will be available and reliable to perform its function when needed. The review should ensure that the applicant has clearly described the test and maintenance programs that will be used at the facility. The test and maintenance programs should be acceptable if the applicant's commitments are consistent with the requirements in §73.46(g) and the following criteria:

A. The applicant has described the testing and maintenance programs for: (1) intrusion alarms; (2) emergency exit alarms; (3) communications equipment; (4) physical barriers; and (5) other physical protection-related devices and equipment such as CCTV, locks, emergency power sources, alarm annunciators, duress alarms, search equipment, etc. used pursuant to 10 CFR 73.46 during the installation and construction, pre-operational and operational tests of the physical protection subsystems and components, consistent with §73.46(g)(1)-(3). This discussion should also include the purpose for and intended level of testing and maintenance programs. In addition, specific methods for testing each type of equipment should be discussed, along with periodicity of testing, consistent with §73.46(g)(3). The applicant should commit to having a testing program for the perimeter intrusion detection system consistent with Regulatory Guide 5.44, Revision 3. The applicant should describe the sensitivity of the SNM, metal, explosive, and x-ray search equipment and the device used for calibration. The applicant should commit to using a device comparable to one which meets the American Society for Testing and Materials (ASTM) F792 standard, "Standard Practice for Design and Use of Ionizing Radiation Equipment for the Detection of Items Prohibited in Controlled Access Areas," consistent with NRC classified criteria.

B. The applicant has described the preventive maintenance program established to ensure the maintenance of all physical protection-related subsystems and components in operable and reliable condition, consistent with §73.46(g)(4) and (5). The applicant should describe corrective actions or compensatory measures used in the event of component failure within physical protection systems.

C. The applicant has described procedures used in performing repairs and maintenance of physical protection systems, consistent with §73.46(g)(5). The applicant should commit that all repairs and maintenance will be performed by two individuals working as a team and that performance verification tests will be conducted after maintenance has been completed.

D. The applicant has described how it will review and audit its security program, consistent with §73.46(g)(6). This discussion should include the periodicity of the review and audit, a

description of who will conduct the review and audit, items to be covered by the review and audit, how the review and audit will be documented, to whom the review and audit findings will be provided for review, and the record-keeping associated with the review and audit program.

E. The commitment for the frequency of the annual audit should not vary by more than plus or minus 1 month.

13.1.4.3.10 Contingency and Response Plans and Procedures (10 CFR 73.46(h))

The performance objective of contingency and response plans and procedures is to provide for predetermined response to safeguards contingency events so that the adversary will be engaged and impeded until offsite assistance arrives. The reviewer should ensure that the applicant has clearly thought out potential contingencies and has clearly described contingency and response plans that will be used by the facility. The contingency and response plans should be acceptable if the applicant's plans are consistent with the requirements in §73.46(h), developed in accordance with the criteria in Appendix C to Part 73, and the following criteria:

A. The applicant has established a safeguards contingency plan for dealing with threats, thefts, and radiological sabotage related to SSNM and its facility and commits to maintain and follow the plan, consistent with §73.46(h)(1).

B. The applicant has described the documented response arrangements it has made with the local law enforcement agencies, consistent with §73.46(h)(2). This should include estimated number of response individuals with specific response times of arrival that are consistent with NRC classified criteria.

C. The applicant has described the number of TRT members immediately available for response and the duties they will be assigned. TRT members may be physically located at the facility or at a nearby facility such that their response is timely, effective, and is not easily interdicted to ensure protection against the design basis threats defined in §73.1(a). In addition, the required force of guards or armed responders available on-site to assist the TRT should be described, along with a discussion of the rationale for determining the number of individuals in this force of guards or armed responders and the availability of this force, consistent with §73.46(h)(3).

D. The applicant has described its planned response procedures for dealing with detection of abnormal presence or activity of persons or vehicles within an isolation zone, the protected area, a material access area, or a vital area or evidence, or indication of intrusion into the protected area, material access area, or a vital area should be described, as well as the methods for assessing the threat and responding to the threat, consistent with §73.46(h)(4). The applicant should establish a defensive strategy which would deny unauthorized access to areas of the facility which contain plutonium. The applicant should commit to requiring guards to interpose themselves between vital areas and material access areas and any adversary attempting entry for purposes of radiological sabotage or theft of SSNM, to

intercept any persons exiting with SSNM, and to inform local law enforcement of the threat and request assistance.

E. The applicant has described the instructions that guards and armed responders will receive in the use of force, including the use of deadly force, in preventing or impeding theft of SSNM, consistent with §73.46(h)(5).

F. The applicant has described the methods that will be used for providing assessment of all protected areas alarms. The applicant should commit to using CCTV or other suitable means which limits exposure of responding personnel to possible attack to assess the protected area barrier and associated isolation zones, consistent with §73.46(h)(6). The applicant should commit to the CCTV providing unobstructed view of the protected area barrier and isolation zones with no blind spots.

G. The applicant has described methods that will be used for assessing alarms occurring within unoccupied vaults and unoccupied material access areas containing plutonium and the timeliness of assessment. The applicant should commit to using at least two security personnel to assess alarms by CCTV or other remote means that occur within unoccupied vaults and unoccupied material access areas, consistent with §73.46(h)(7).

H. The applicant has described methods that will be used for assessing alarms occurring within unoccupied vaults and unoccupied material access areas containing alloyed or encapsulated SSNM and the timeliness of assessment. The applicant should commit to using at least two security personnel to remotely assess alarms by CCTV, or by at least two security personnel who are searched before exiting the material access areas, consistent with §73.46(h)(8).

I. The applicant has described how it will establish, maintain, and retain as a record the current safeguards contingency plan, consistent with §73.46(h)(1) and (2).

13.1.4.3.11 Reporting of Safeguards Events (10 CFR 73.71)

Acceptance should be based on the fact that the applicant adequately addresses how and when it will report safeguards events to the NRC and follows the criteria in 10 CFR Part 73, Appendix G, "Reportable Safeguards Events."

13.1.5 REVIEW PROCEDURES

13.1.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application (construction or license) adequately addresses the applicable items in Section 13.1.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

J. Application for Construction Approval

Although the applicant is not expected to submit a physical protection plan with the application for construction approval, the applicant should commit to developing and implementing a physical protection system that meets or exceeds the acceptance criteria in Section 13.1.4. If provided by the applicant, the primary reviewer should evaluate the proposed location and construction technique and materials of the buildings; protected, vital, material access, and controlled access area barriers; vehicles barriers; alarm stations; security search or control points; and vaults to ensure that the commitments and program goals, as described in Section 13.1.3, are appropriate for physical protection at the design stage.

K. License Application

Specifically, the license application should address Section 13.1.3 in full. The applicant is expected to provide a physical protection plan with the license application.

If the primary reviewer verifies that physical protection is adequately addressed (construction or license), the primary reviewer should accept the application for the safety evaluation in Section 13.1.5.2. 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.

13.1.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 13.1.5.1(A) (construction) or 13.1.5.1(B) (license), the primary reviewer should perform an evaluation against the acceptance criteria described in Section 13.1.4. On the basis of its review, the reviewer may request that the applicant provide additional information or modify the application to meet the acceptance criteria in SRP Section 13.1.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The primary reviewer should establish that the applicant's proposed design, location, construction technique, and material for elements of the physical protection system and related commitments will lead to a physical protection plan that will meet or exceed the regulatory acceptance criteria in Section 13.1.4.

B. License Application

The primary reviewer should verify that sufficient information has been provided under Section 13.1.4.3, with respect to the physical protection plan, and that the information provided is consistent with the guidance in this SRP chapter.

When the evaluation is complete, the primary reviewer should prepare the physical protection input for the Safety Evaluation Report (SER), as described in Section 13.1.6 using the acceptance criteria from Section 13.1.4.

13.1.6 EVALUATION FINDINGS

The primary reviewer should document the physical protection evaluation by preparing material suitable for inclusion in the SER. The primary reviewer should describe the review, identify any alternative measures that will be used, explain the basis for the findings, and state the conclusions.

The reviewer could document the safety evaluation for the application for construction approval as follows:

The reviewer reviewed the application for construction approval for [insert name of facility] according to Section 13.1 of NUREG-1718. The reviewer evaluated [Insert a summary statement of what was evaluated] and found that [summarize the findings].

The reviewer concluded that the applicant provided adequate commitments and goals for the design of a physical protection system and that these commitments and goals should result in a physical protection plan that will meet or exceed the requirements in 10 CFR 73.20, 73.45 and 73.46 and guidance outlined in NUREG-1718. As a result, the applicant meets the requirements under proposed 10 CFR Part 70 for construction approval of the facility in the area of physical protection.

The reviewer could document the safety evaluation for the license application as follows:

The reviewer reviewed the license application for [insert facility name] according to Section 13.1 of NUREG-1718. The reviewer 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 reviewer concluded that the applicant adequately described and documented physical protection system and provided a plan to address the regulations in 10 CFR 73.20, 10 CFR 73.45 and 10 CFR 73.46. Meeting the requirements given above provides an acceptable basis for the finding that, insofar as physical protection is concerned, the applicant meets the associated requirements in 10 CFR Parts 73.20, 73.45 and 73.46 and therefore the physical protection plan is acceptable to support licensed operation under 10 CFR Part 70.

13.1.7 REFERENCES

A. Regulatory Guide 5.52, Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants), Rev. 3, December 1994.

B. Regulatory Guide 5.44, Perimeter Intrusion Alarm Systems, Rev. 3, October 1997.

C. Regulatory Guide 5.55, Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities, March 1978.

D. Code of Federal Regulations, Title 10, Part 73.20, General Performance Objectives and Requirements.

E. Code of Federal Regulations, Title 10, Part 73.45, Performance Capabilities for Fixed Site Physical Protection Systems.

F. Code of Federal Regulations, Title 10, Part 73.46, Fixed Site Physical Protection Systems, Subsystems, Components, and Procedures.

G. Code of Federal Regulations, Title 10, Part 73.71, Reporting of Safeguards Events.

H. Code of Federal Regulations, Title 10, Part 11, Criteria and Procedures for Determining Eligibility for Access to Or Control Over Special Nuclear Material.

I. Code of Federal Regulations, Title 10, Part 25, Access Authorization for Licensee Personnel.

J. Code of Federal Regulations, Title 10, Part 95, Security Facility Approval and Safeguarding of National Security Information and Restricted Data.

K. Regulatory Guide 5.7, Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas, Rev. 1, May 1980.

L. Battelle Columbus Division, NUREG-CR 5172, Tactical Training Reference Manual, April 1989.

M. Battelle Columbus Division, NUREG-CR 5081, Tactical Exercise Planning Handbook, April 1989.

13.0 SAFEGUARDS
13.2 MATERIAL CONTROL AND ACCOUNTING (MC&A)

13.2.1 PURPOSE OF REVIEW

The purpose of this review is to ensure that the Fundamental Nuclear Material Control Plan (FNMCP) submitted by the applicant describes how an MC&A system will be established, implemented, and maintained, and to ensure that the FNMCP is adequate to protect against, detect, and respond to the loss or theft of strategic special nuclear material (SSNM) by achieving the following five performance objectives stated in the Code of Federal Regulations (CFR), Title 10, Part 74.51(a):

A. Prompt investigation of anomalies potentially indicative of SSNM losses;

B. Timely detection of the possible abrupt loss of five or more formula kilograms of SSNM from an individual unit process;

C. Rapid determination of whether an actual loss of five or more formula kilograms occurred;

D. Ongoing confirmation of the presence of SSNM in assigned locations; and

E. Timely generation of information to aid in the recovery of SSNM in the event of an actual loss.

These objectives will be achieved by meeting the system capabilities requirements stated in 10 CFR 74.51(b).

13.2.2 RESPONSIBILITY FOR REVIEW

Primary: Safeguards Technical Analyst (MC&A Specialist)

Secondary: Project Manager

Supporting: MC&A Physical Scientist (MC&A Inspector)
Physical Protection Reviewer

13.2.3 AREAS OF REVIEW

The staff should review the applicant's FNMCP to ensure that the plan, in meeting the five performance objectives stated in Section 13.2.1, addresses:

A. Process Monitoring Program: For each unit process, the applicant's establishment of a production quality control program capable of monitoring the status of material in process;

B. Item Monitoring Program: The applicant's establishment of a process to verify the presence and integrity of SSNM items on a statistical sampling basis;

C. Alarm Resolution Program: The applicant's establishment of an alarm resolution program that is capable of:

   i. Resolving the nature and causes of any MC&A alarm within approved time periods;

   ii. Notifying NRC of any MC&A alarms that remain unresolved beyond the time periods;

   iii. Determining the amount of actual SSNM lost and taking corrective actions;

   iv. Providing an ability to rapidly assess the validity of alleged thefts; and

   v. Taking appropriate actions when the abrupt loss detection estimate exceeds two kilograms of plutonium.

D. Quality Assurance and Accounting Programs: The applicant's establishment of a quality assurance and accounting capability to address the following 11 elements: management structure, personnel qualification and training, measurements, measurement control, physical inventory, accounting, shipping and receiving, scrap control, human errors, independent assessments, and custodial responsibility.

13.2.4 ACCEPTANCE CRITERIA

13.2.4.1 Regulatory Requirements

Regulatory requirements applicable to the MC&A program and the FNMCP are specified in 10 CFR Part 74, "Material Control and Accounting of Special Nuclear Material." Subpart E, Formula Quantities of Strategic Special Nuclear Material, particularly applies to mixed oxide (MOX) fuel fabrication facilities.

13.2.4.2 Regulatory Guidance

NUREG-1280, "Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment," Rev. 1, April 1995.

13.2.4.3 Regulatory Acceptance Criteria

The performance objectives and acceptance criteria discussed below pertain to plutonium, both before and after processing into MOX. MOX contains uranium in the form of either depleted, natural, or low enriched uranium (LEU). The reviewer must be aware of which type of uranium will be processed into MOX and verify that this is stated in the process description section of the FNMCP. If the applicant uses LEU to produce MOX, the reviewer must verify that, up until the

time of processing, the LEU feed material is adequately controlled to enable the MC&A system to meet the objectives and requirements of 10 CFR Part 74.31.

It is important that the applicant establishes the basis for determining the formula quantity of SSNM for a facility processing MOX. Formula kilogram (FKG) means SSNM in any combination in a quantity of 1,000 grams computed by the formula:

$$\text{grams} = (\text{grams contained U-235}) + 2.5\ (\text{grams U-233} + \text{grams plutonium}).$$

Formula quantity means SSNM in any combination in a quantity of 5,000 grams or more computed by the above formula. Where the uranium oxide used in the process has an enrichment level lower than 20%, the determination of FKG is based on the amount of plutonium only. Therefore, two kilograms of plutonium yields five FKG or a formula quantity of SSNM.

13.2.4.3.1 Performance Objectives

Reviewers should use a risk-informed, performance-based approach to review the applicant's program and capability in meeting the performance objectives in 10 CFR 74.51(a). The reviewers should give high priority to the overall timely detection and resolution program. The reviewers should evaluate if the applicant appropriately considered and incorporated a collusion protection program in the MC&A system (i.e., threats from an insider; and potential diversion strategies during fuel processing, in material storage, or from recovery/recycling products). The primary reviewer of this section should coordinate with the primary reviewer of Section 13.1 where the applicant designed the detection program to be complimentary to the physical protection requirements in 10 CFR Part 73 to minimize redundant systems while maintaining adequate safeguards assurance.

13.2.4.3.2 Process Monitoring

Part 74.53 requires that licensees monitor internal transfers, storage, and processing of SSNM. The applicant's process monitoring program should be capable of: (1) promptly detecting a significant abrupt loss, diversion, or theft of two kilograms of plutonium with 95% power of detection, and (2) monitoring the status of material in process. The "prompt" detection is dependent upon the classification of the materials, i.e., Category IA or IB, as specified in §74.53.

The applicant's process monitoring program should at least consist of:

A. Clearly defined process subdivisions and measurement points to satisfy unit detection criteria and the category of material being processed;

B. Adequate material control tests for each unit process for detecting abrupt losses with at least 95% power of detection, evaluation and update of the action threshold on semi-annual basis, and ability to detect losses involving material substitution;

C. Basis for material classification, i.e., Category IA and IB materials;

D. Clear classification of inaccessible locations;

E. Identification of all credible substitute materials and the methods of preventing substitution;

F. A listing of material types exempted from the abrupt loss detection tests with their locations and basis for exemption;

G. Adequate trend analysis techniques and decision criteria, especially for the indication of trickling diversions; and

H. Adequate material balance tests and evaluation for research and development operations.

It is necessary for the applicant to submit a study of potential diversion scenarios as supporting information. Such study should include, but not be limited to, abrupt losses, trickle diversion, insider and/or outsider diversion, unauthorized production, and material substitution.

The applicant's process monitoring program should be found acceptable if it meets the criteria specified in Chapter 1 of NUREG-1280, Rev. 1, April 1995. In addition, NUREG/CR-4604, *Statistical Methods for Nuclear Material Management*, provides guidance on statistical tests in providing 95% power of detection.

13.2.4.3.3 Item Monitoring

Part 74.55 requires that licensees establish an item monitoring program capable of providing timely plant-wide detection of the loss of items that total two kilograms of plutonium with 99% power of detection. The "timely" detection is dependent upon the classification of the materials, i.e., Category IA or IB, and the degree of tamper-safing that is employed, as specified in §74.55. The applicant's item monitoring program should at least consist of:

A. A clear item identification system;
B. A basis for item classification, i.e., Category IA and IB materials;
C. A tamper-safing procedure and system;
D. Accessibility control;
E. Accounting and control procedures;
F. Item measurement systems;
G. Item verification procedures; and
H. Item sampling techniques.

The applicant's item monitoring program should be found acceptable if it meets the criteria specified in Chapter 2 of NUREG-1280, Rev. 1, April 1995. In addition, NUREG/CR-4604, *Statistical Methods for Nuclear Material Management*, provides guidance on statistical tests in providing 99% power of detection.

13.2.4.3.4 Alarm Resolution

Part 74.57 requires that the licensees' alarm resolution and reporting programs assure:

A. Resolution of the nature and cause of any MC&A alarm within approved time periods;

B. Reporting to NRC within 24 hours of any unresolved MC&A alarm beyond the specified time period;

C. Determining the amount of SSNM lost and taking corrective actions when a material loss has occurred;

D. The ability to rapidly assess the validity of alleged thefts; and

E. Taking appropriate actions when an abrupt loss detection estimate exceeds two kilograms of plutonium.

Specifically, the programs should address alarm resolution procedures, decision rules and their basis, and response time.

The applicant's programs for resolving and reporting indications of missing SSNM should be found acceptable if they meet the criteria specified in Chapter 3 of NUREG-1280, Rev. 1, April 1995. In addition, the applicant should establish the capability to respond rapidly to alarms occurring externally to the MC&A system, as stipulated in Chapter 3.3 of NUREG-1280, Rev. 1, April 1995.

13.2.4.3.5 Quality Assurance and Accounting Programs

Part 74.59 requires that licensees establish a quality assurance and accounting capability to address the 11 areas discussed in Sections 13.2.4.3.5 (A) through (K).

A. Management Structure

Part 74.59(b) establishes requirements for the licensees' MC&A management structure, organization, responsibilities, procedures, etc. The applicant's MC&A program's management structure should demonstrate the checks and balances of the program to ensure effective functioning of the MC&A program by providing:

i. Clear overall responsibility for MC&A responsibilities;

ii. Independence of MC&A functions from production responsibilities;

iii. Separation of key MC&A responsibilities from each other to provide controls and checks; and

iv. Adequate review, approval, and use of approved written MC&A procedures.

The applicant's organization for developing and implementing the MC&A program and procedures should be found acceptable if it meets the criteria specified in Chapter 4.1 of NUREG-1280, Rev. 1, April 1995. (SRP Chapter 2.0 provides additional guidance on organization and administration, and SRP Chapter 11.5 provides additional guidance on procedures.)

B. Personnel Qualification and Training

Part 74.59(c) establishes qualifications and training requirements for key MC&A personnel. The applicant's personnel qualification and training programs should ensure that qualified and adequately trained personnel are implementing and maintaining an effective MC&A program by ensuring that:

i. Personnel who work in key positions where mistakes could degrade the effectiveness of the MC&A program are trained to maintain a high level of safeguards awareness and are qualified to perform their duties and/or responsibilities;

ii. Continuing qualification of key personnel will be verified on an ongoing basis or at least every 2 years; and

iii. The training program emphasizes the job purposes and scope and provides a balance between theory and practice.

The applicant's personnel qualification and training programs should be found acceptable if they meet the criteria specified in Chapter 4.2 of NUREG-1280, Rev. 1, April 1995. (SRP Chapter 11.4 provides additional guidance on training and qualification.)

C. Measurement

Part 74.59(d) requires that licensees establish and maintain a system of measurements. The applicant's measurement program should ensure that:

i. All source material, SNM, and SSNM information in accounting records are based on measured values;

ii. Key measurement systems and measurement points are identified;

iii. At each measurement point, the appropriate measurement method and system is used for the accurate and precise determination of the material type;

iv. The MC&A system enables the estimation of the standard deviation associated with each measured quantity; and

v. Necessary data are provided for performing material control tests.

The applicant's measurement program should be found acceptable if it meets the criteria specified in Chapter 4.3 of NUREG-1280, Rev. 1, April 1995. The following documents also provide additional guidance/information on measurement methods: NUREG-0228, *Calorimetric Assay of Plutonium*, NUREG-0256, *Methods for the Accountability of Mixed Oxide*, NUREG/CR-0602, *Active Nondestructive Assay of Nuclear Materials*, NUREG/CR-2078, *Handbook of Nuclear Safeguards Measurement Methods, September 1983*, and NUREG/CR-5550, *Passive Nondestructive Assay of Nuclear Materials*.

D. Measurement Control

Part 74.59(e) requires that licensees ensure the quality of measurement systems and material processing practices. The applicant's measurement control program should include:

i. Performing engineering analyses and evaluations on all MC&A measurement systems;

ii. Establishing and verifying procedures for mixing and sampling source material, SNM, and SSNM and maintaining sample integrity during transport and storage;

iii. Generating current data on the performance of measurement processes;

iv. Utilizing the measurement control data for the estimation of standard errors of inventory difference (SEID) and the standard deviation associated with the process differences;

v. Ensuring SEID is less than 0.1% of the active inventory;

vi. Applying bias corrections in accordance with approved written procedures;

vii. Investigating and taking corrective actions when the associated measurement biases exceed limits; and

viii. Establishing and maintaining a statistical control system to monitor the quality of each type of program measurement.

The measurement control program applies to measurement systems utilized for inventory, shipper-receiver measurement, monitoring cumulative shipper-receiver differences, and detection and response purposes. In addition, the applicant should ensure the traceability of calibration and control standard measurements to a national standard or nationally accepted measurement system.

The applicant's measurement control program should be found acceptable if it meets the criteria specified in Chapter 4.4 of NUREG-1280, Rev. 1, April 1995. NUREG/CR-4604 and TID-26298, *Statistical Methods in Nuclear Material Control*, 1973, provide additional guidance on measurement error standard deviation.

E. Physical Inventory

Part 74.59(f) contains the basic requirements for scheduling, performing, and evaluating physical inventories. The applicant's physical inventory program should ensure that it provides for:

i. Performing a physical inventory at least every 6 calendar months (unless otherwise required to satisfy 10 CFR Part 75);

ii. Within 45 days after the start of the ending inventory:

   a. Calculating inventory difference (ID) and estimating SEID;
   b. Investigating, resolving, and reporting excessive ID and SEID;
   c. Reconciling and adjusting the book inventory; and
   d. Performing reinventory as necessary.

iii. Implementing policies, practices, and procedures designed to ensure the quality of physical inventories; and

iv. Control and maintenance of records and documentation associated with the physical inventories.

The applicant should appropriately describe the procedures and/or processes for verifying the location and identity of all quantities of SSNM and for verifying that all quantities are based on measurements, inventory cutoff and cutoff verification, and reconciliation. It is critical that the applicant demonstrates its ability to eliminate holdup before physical inventory and to measure holdup if it cannot be eliminated.

The applicant's physical inventory program should be found acceptable if it meets the criteria specified in Chapter 4.5 of NUREG-1280, Rev. 1, April 1995. NUREG/BR-0096, *Instructions and Guidance for Completing Physical Inventory Summary Reports*, provides additional guidance on completing NRC Form 327, *SNM and SM Physical Inventory Summary Report.*

F. Accounting

Part 74.59(g) requires that licensees establish auditable records sufficient to demonstrate that the requirements of 10 CFR 74.51, 74.53, 74.55, 74.57, and 74.59 have been met. The applicant's accounting programs should establish and maintain records in an auditable form, available for inspection, for at least 3 years, unless a longer retention time is required by 10 CFR Part 75. The programs should specify in what form those records will be kept. The programs should provide adequate safeguards against tampering with and loss of records. (SRP Chapter 11.8 provides additional guidance on Records Management.)

The applicant's programs for record keeping should be found acceptable if they meet the criteria specified in Chapter 4.6 of NUREG-1280, Rev. 1, April 1995. NUREG/BR-0006,

Instructions for Completing Nuclear Material Transaction Reports and Concise Note Forms (Form DOE/NRC 741, 741A, and 740M), and NUREG/BR-0007, *Instructions for Completing Nuclear Balance Report and Physical Inventory Listing (Forms DOE/NRC 742 and 742C)*, provide additional guidance on the use of NRC-required forms for reporting transactions involving nuclear materials.

G. Shipments and Receipts

Part 74.59(h)(1) requires that licensees establish procedures for the measurement of shipments and receipts and for the review, evaluation, and investigation of shipper-receiver differences (SRD). The applicant should establish a program to timely and accurately quantify the content of SSNM and other nuclear materials in shipments and receipts. The program should provide:

i. Accurate identification and measurements of the quantity shipped and received;
ii. Clear definition of statistically significant SRDs;
iii. Review and evaluation of SRD;
iv. Investigation and corrective actions when SRD exceed the specified limit; and
v. Documentation of SRD evaluations, investigations, and corrective actions.

The program should identify a reasonable time frame for completing the verification measurements of receipts. The documentation of shipments and receipts should be completed and transmitted within the time frame specified in NUREG/BR-0006. The applicant's program for shipper-receiver comparisons should be found acceptable if it meets the criteria specified in Chapter 4.7 of NUREG-1280, Rev. 1, April 1995.

H. Scrap Control

Part 74.59(h)(2) establishes requirements regarding the segregation of internally generated scrap from scrap received from other nuclear facilities and regarding the prompt recovery of scrap which cannot be measured to within ±5%. The applicant's scrap control program should ensure that:

i. Internally generated scrap and scrap from other licensees or contractors are segregated until accountability is established; and

ii. Any scrap measured with a standard deviation greater than 5% of the measured amount is recovered, so that the results are segregated by inventory period and received within 6 months of the end of the inventory period in which the scrap was generated, except where it can be demonstrated that the scrap measurement uncertainty will not cause noncompliance with 10 CFR 74.59(e)(5).

In addition, the applicant's scrap control program should address that:

iii. Scrap and waste will only be stored in approved locations and disposed by approved methods;

iv. The facility's recovery capability is adequate to preclude buildup of excess amounts of scrap;

v. Special handling procedures for waste or independent measurement verification are described;

vi. Scrap generated onsite and offsite are adequately separately, and the individuals performing measurements on scrap materials have the authority to reject containers that demonstrably violate segregation practices; and

vii. Procedures and processes for offsite scrape recovery are discussed.

The applicant's scrap control program should be found acceptable if it meets the criteria specified in Chapter 4.8 of NUREG-1280, Rev. 1, April 1995.

I. Human Errors

Part 74.59(h)(3) requires that the licensees incorporate checks and balances in MC&A system to control the rate of human errors in MC&A information. The applicant's program should minimize human errors in the following areas:

i. The development and management of MC&A procedures, especially procedures for processing MC&A data;

ii. The use of job performance aids, such as illustrations and graphs;

iii. The methods and technologies used to automate MC&A functions; and

iv. The quality control system used to monitor the frequency and types of human errors.

The applicant's human error controls should be found acceptable if they meet the format and criteria specified in Chapter 4.9 of NUREG-1280, Rev. 1, April 1995. (SRP Chapter 16 provides additional guidance on human factors.)

J. Independent Assessment

Part 74.59(h)(4) requires that the licensees independently assess the past performance of MC&A program. The applicant's audit and assessment program should be acceptable if it:

i. Independently assesses the effectiveness of MC&A system at least every 12 months;

ii. Documents the results of the assessment;

iii. Documents management's findings on whether the MC&A system is effective;

iv. Documents any actions taken on recommendations from prior assessments; and

v. Assesses the measurement control program of any outside contractor laboratory performing MC&A measurements for the applicant.

The selection of assessment team members should assure and balance independence and knowledge in the MC&A area. An assessment by a third party organization is not required, but is often an effective way to bring both knowledge and independence to the assessment effort. (SRP Section 11.6 provides additional guidance on Audits & Assessments.)

The applicant's program for assessing and reviewing the MC&A program should be found acceptable if it meets the criteria in Chapter 4.10 of NUREG-1280, Rev. 1, April 1995.

K. SSNM Custodianship

Part 74.59(h)(5) establishes requirements for assigning custodial responsibility for SSNM. The applicant's assignment of custodial responsibility should ensure that such responsibility is clearly defined and can be effectively executed. The applicant's SSNM custodial assignments should be found acceptable if they meet the criteria specified in Chapter 4.11 of NUREG-1280, Rev. 1, April 1995.

13.2.5 REVIEW PROCEDURES

13.2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval or license application adequately addresses the items in Section 13.2.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the safeguards assessment of the design basis should address Section 13.2.3 at the level of commitments and program goals.

B. License Application

Specifically, the license application should address Section 13.2.3 in full and should include the FNMCP. The secondary and supporting reviewers should confirm that the FNMCP is consistent with descriptions in other sections of the application. Information provided in the FNMCP should be of comparable quality and detail, and should not contradict or adversely impact information contained in other sections of this application.

If the primary reviewer verifies that MC&A is adequately addressed (application for construction approval or license application), the primary reviewer should accept the application for the safety evaluation in Section 13.2.5.2. 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.

13.2.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 13.2.5.1(A) (application for construction approval) or 13.2.5.1(B) (license application for operations), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 13.2.4. 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 13.2.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

   The primary reviewer should establish that the applicant's design basis for MC&A and related commitments will lead to an FNMCP that will meet or exceed the regulatory acceptance criteria in Section 13.2.4.

B. License Application

   The primary reviewer should establish that the applicant's FNMCP provides reasonable assurance in satisfying the acceptance criteria in Section 13.2.4 of this SRP. Also the primary reviewer should ensure that adequate documentation is provided.

For an existing facility, the NRC reviewers may wish to visit the site and hold discussions with facility personnel in order to gain a better understanding of the safeguards systems. For a planned facility, the NRC reviewers may wish to meet with the design team in order to gain a better understanding of the process, its potential safeguards concerns, and safeguards system/design approaches.

When the evaluation is complete, the primary reviewer, with assistance from other reviewers, should prepare input for the Safety Evaluation Report (SER) as described in Section 13.2.6 using the acceptance criteria from Section 13.2.4.

13.2.6 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 application for construction approval as follows:

The staff reviewed the application for construction approval for [insert name of facility] according to Section 13.2 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 material control and accounting and that these commitments and goals should result in a MC&A program and FNMCP that will meet or exceed the regulatory acceptance criteria outlined in NUREG-1718. As a result, the applicant meets the requirements in the area of MC&A to approve construction of the facility under proposed 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] according to Section 13.2 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 NRC staff concluded that the applicant's FNMCP satisfies the staff's acceptance criteria. Specifically, the applicant has satisfactorily addressed the applicable regulatory requirements in 10 CFR Parts 74.51, 74.53, 74.55, 74.57, and 74.59.

13.2.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 74, Subpart E, *Formula Quantities of Strategic Special Nuclear Material.*

B. NUREG-1280, *Standard Format and Content Acceptance Criteria for the Material Control and Accounting (MC&A) Reform Amendment,* Rev. 1, April 1995.

C. NUREG/CR-4604, *Statistical Methods for Nuclear Material Management,* December 1988.

D. NUREG-0228, *Calorimetric Assay of Plutonium,* May 1977.

E. NUREG-0256, *Methods for the Accountability of Mixed Oxide,* April 1977.

F. NUREG/CR-0602, *Active Nondestructive Assay of Nuclear Materials,* January 1981.

G. NUREG/CR-2078, *Handbook of Nuclear Safeguards Measurement Methods,* September 1983.

H. NUREG/CR-5550, *Passive Nondestructive Assay of Nuclear Materials,* March 1991.

I. TID-26298, *Statistical Methods in Nuclear Material Control*, 1973.

J. NUREG/BR-0096, *Instructions and Guidance for Completing Physical Inventory Summary Reports (NRC Form 327)*, October 1992.

K. NUREG/BR-0006, *Instructions for Completing Nuclear Material Transaction Reports and Concise Note Forms (Forms DOE/NRC 741, 741A, and 740M)*, Revision 3, January 1989.

L. NUREG/BR-0007, *Instructions for Completing Nuclear Balance Report and Physical Inventory Listing (Forms DOE/NRC 742, and 742C)*, Revision 2, July 1989.

13.2.8 DEFINITIONS

formula kilogram (FKG): SSNM in any combination in a quantity of 1,000 grams computed by the formula: grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium).

formula quantity: SSNM in any combination in a quantity of 5,000 grams or more computed by the formula: grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium).

14.0 EMERGENCY MANAGEMENT

14.1 PURPOSE OF REVIEW

The purpose of this review is to determine if the applicant established, before the start of operations, adequate emergency management facilities and procedures to protect the public, the workers, and the environment. The applicant should also show how the emergency management facilities and procedures comply with NRC regulations while coexisting with the Department of Energy's (DOE) emergency planning requirements and that DOE's requirements do not contradict any NRC requirements.

An emergency plan is required when an evaluation shows that the maximum dose to a member of the public offsite due to a release of radioactive materials would exceed 0.01 Sv (1 rem) effective dose equivalent. This section applies to facilities authorized to possess enriched uranium (U) or plutonium (Pu) for which a criticality accident alarm system is required, uranium hexafluoride (UF_6) in excess of 50 kg (110 lb) in a single container or 1000 kg (2200 lb) total, or Pu in excess of 2 Ci in unsealed form or on foils or plated sources.

Emergency capability is incorporated into the baseline design criteria of 10 CFR Part 70, as revised, and is intended to ensure control of licensed material, evacuation of personnel, and availability of emergency facilities.

14.2 RESPONSIBILITY FOR REVIEW

Primary: Emergency Preparedness Specialist

Secondary: Project Manager

Supporting: Regional Emergency Preparedness Inspector
Fuel Facility Inspection staff

14.3 AREAS OF REVIEW

The NRC staff should review the applicant's submittal for an acceptable level of evidence of planning for emergency preparedness directed at situations involving real or potential radiological hazards. The review should address those design features, facilities, functions, and equipment that may affect some aspect of emergency planning or the capability of an applicant to cope with plant emergencies. In addition, the review should address coordination with offsite organizations. The staff should either review the emergency plan made in accordance with 10 CFR 70.22(i)(1)(ii) and with the guidance contained in the acceptance criteria below, or should review the applicant's evaluation that demonstrates that the maximum dose to a member of the public would not exceed 0.01 Sv (1 rem) effective dose equivalent in accordance with 10 CFR 70.22(i)(1)(i).

The NRC staff reviewer should review the material presented, as described below.

14.3.1 Specific Items to be Reviewed When the Applicant Submits an Evaluation

If the applicant submits an evaluation to demonstrate that the maximum dose to a member of a public would not exceed 0.01 Sv (1 rem) effective dose equivalent, the staff should review the evaluation against 10 CFR 70.22(i)(1)(i), and NUREG-1140, "A Regulatory Analysis of Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees." NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," also contains useful information. Areas to be evaluated should include the following:

A. A description of the facility and proposed licensed activities;
B. Types of materials used, including both radioactive material and hazardous chemicals;
C. Types of accidents;
D. Detection of accidents;
E. Site specific information used to support the evaluation; and
F. An evaluation of the consequences, both onsite and offsite.
G. The evaluation should address one or more of the factors provided in 10 CFR 70.22(i)(2).

14.3.2 Specific Items to be Reviewed When the Applicant Submits an Emergency Plan

If the applicant submits an emergency plan, the staff should evaluate the emergency plan against 10 CFR 70.22(i)(1)(ii) and Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," which provides a standard format and content for an emergency plan. Elements in the emergency plan should include:

A. Facility description (including both onsite and offsite emergency facilities);
B. Types of accidents;
C. Classification of accidents;
D. Detection of accidents;
E. Mitigation of consequences (and safe shutdown);
F. Assessment of releases (both radioactive materials and hazards chemicals);
G. Responsibilities of applicant;
H. Notification and coordination;
I. Information to be communicated and parties to be contacted;
J. Training;
K. Safe shutdown (recovery and plant restoration);
L. Exercises (and drills);
M. Hazardous chemicals inventories and locations; and
N. Responsibilities for developing and maintaining the emergency program and its procedures.

14.4 ACCEPTANCE CRITERIA

14.4.1 Regulatory Requirements

10 CFR Part 70.22(i)(1)(i) specifies when an emergency plan does not have to be submitted to the NRC and, if an emergency plan is required to be submitted, 10 CFR Part 70.22(i)(3), contains the information that must be included in the emergency plan.

10 CFR Part 70.64(a)(6) requires that applicants address the control of licensed material, evacuation of personnel, and availability of onsite emergency facilities that facilitate the use of available offsite services.

14.4.2 Regulatory Guidance

Regulatory guidance for preparing an emergency plan includes:

A. Regulatory Guide 3.67, "*Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities,*" January 1992.

B. NUREG-1140, "A Regulatory Analysis of Emergency Preparedness for Fuel Cycle and Other Radioactive Materials," January 1988.

C. NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," 1998.

14.4.3 Regulatory Acceptance Criteria

If the applicant's proposed total possession limit for radioactive material exceeds the emergency plan threshold in 10 CFR 70.22(i)(1), the applicant may either submit a site specific evaluation that demonstrates maximum public exposure is less that the limits in 70.22(i)(1)(i), or an emergency plan. If the applicant submits an evaluation, the regulatory acceptance criteria in Section 14.4.3.1 apply. If the applicant submits an emergency plan, the regulatory acceptance criteria in Section 14.4.3.2 apply.

14.4.3.1 Evaluation

The adequacy of the applicant's evaluation that the maximum dose to a member of the public offsite due to a release of radioactive materials would not exceed 0.01 Sv (1 rem) or an intake of 2 mg (7.04×10^{-5} ounces) of soluble uranium should be evaluated by the reviewer against the requirements in 10 CFR Part 70.22(i)(2) and the specific criteria given in this section of the SRP. The applicant's evaluation should be acceptable if the regulatory requirements and the following criteria are met:

14.4.3.1.1 Facility Description

The applicant's evaluation includes a description of the facility and site, the area near the site, and the licensed activities conducted at the facility sufficient to support the evaluation. The facility description should be acceptable if it includes:

A. A detailed drawing of the site showing (1) onsite and near offsite (within 1.6 km [1 mile]) structures with building numbers and labels, (2) roads and parking lots onsite and main roads near the site, (3) site boundaries, showing fences and gates, (4) major site features, (5) water bodies within approximately 1.6 km (1mile), and (6) the location(s) of nearest residence(s);

B. The stack heights, typical stack flow rates, and the efficiencies of any emission control devices; and

C. A general description of the proposed licensed and other major activities conducted at the facility, and the type, form, solubility and maximum quantities of radioactive and other hazardous material normally onsite.

14.4.3.1.2 Types of Accidents

The applicant's evaluation describes each type of accident identified by the Integrated Safety Analysis (ISA) Summary that has the maximum offsite consequences exceeding the limit of 10 CFR 70.22(i)(1)(i). The types of accidents should be acceptable if they include:

A. The process and physical location where each accident could occur;

B. Complicating factors and possible onsite and offsite consequences, including non-radioactive hazardous material released;

C. The accident sequence that has the potential for the greatest radiological and toxic chemical impact.

14.4.3.1.3 Detection of Accidents

The applicant's evaluation should be acceptable if, for each type of accident identified, the applicant identifies:

A. The means of detecting the accident;
B. The means of detecting any release of radioactive or other hazardous material;
C. The means of alerting the operating staff; and
D. The anticipated response of the operating staff.

14.4.3.1.4 Maximum Public Exposure

In addition to the acceptance criteria in Sections 14.4.3.1 - 14.4.3.3, the applicant's evaluation should be acceptable if it includes a description of the following information sufficient to allow the primary reviewer to independently verify the calculations:

A. Type of accident (e.g., fire, exposure, chemical release, nuclear criticality);

B. Location of accident;

C. Maximum source term;

D. Solubility of material;

E. Facility design or engineered safety features in the facility and the proposed release fraction;

F. Location and distance of nearest member of the public to the facility;

G. Dose model used and the process used to verify the reliability of the model and validity of the assumptions;

H. Assumed worst case weather condition; and

I. Maximum calculated dose to a member of the public at the facility boundary.

The applicant's site specific evaluation should include a list and a description of the factors in 10 CFR 70.22(i)(2) that the applicant considered in evaluating maximum dose to members of the public. The applicant should demonstrate why the factors used in the evaluation are appropriate when compared to the factors in NUREG-1140. If the factors and evaluation show that the maximum dose to a member of the public offsite due to a release of radioactive materials could not exceed 0.01 Sv (1 rem) effective dose equivalent or the intake of soluble uranium of 2 mg (7.04×10^{-5} ounces), no emergency plan is required in accordance with 10 CFR 70.22(i)(1)(i). If the primary reviewer finds that the maximum dose to a member of the public could exceed 0.01 Sv (1 rem), the applicant must either submit an emergency plan consistent with the requirements in Section 14.4.3.2, or decrease the total possession limit for radioactive material below the emergency plan threshold in 10 CFR 70.22(i)(1).

14.4.3.2 Criteria When an Emergency Plan is Required

The adequacy of the applicant's proposed emergency plan should be evaluated by the reviewer against the requirements in 10 CFR Part 70.22(i)(3), and the specific criteria given in Section 14.4.3.2 of the SRP. The applicant's emergency plan should be acceptable if the regulatory requirements and the following criteria are met:

14.4.3.2.1 Facility Description

14.4.3.2.1.1 Operational Facilities

The applicant's emergency plan includes a description of the facility and site, the area near the site, and the licensed activities conducted at the facility sufficient to support emergency management activities. The description should be acceptable if it includes:

A. A detailed drawing of the site showing:

   i. Onsite and near offsite (within 1.6 km [1 mile]) structures with building numbers and labels;

   ii. Roads and parking lots onsite and main roads near the site;

   iii. Site boundaries, showing fences and gates;

   iv. Major site features; and

   v. Water bodies within approximately 1.6 km (1 mile).

B. A general area map (approximately 16 km [10 mile] radius), a United States Geological Survey topographical quadrangle (7 ½ minute series; including the adjacent quadrangle(s) if site is located less than 1.6 km [1 mile] from the edge of the quadrangle), and a map or aerial photograph indicating onsite structures and near-site structures (about 1.6 km [1 mile] radius). The general area map indicates the location of sensitive facilities near the site such as hospitals, schools, nursing homes, nearest residence(s), fire departments, prisons, environmental sampling locations, and other structures and facilities important to emergency management.

C. The stack heights, typical stack flow rates, and the efficiencies of any emission control devices;

D. A general description of licensed and other major activities conducted at the facility and the type, form, and quantities of radioactive and other hazardous materials normally onsite by location (use and storage) and building, including the hazardous characteristics (exposure rates, pH, temperature, and other characteristics) important to emergency management.

E. Certification that the applicant has met responsibilities under Emergency Planning and Community Right To Know Act of 1986, Title III, Public Law 99-499, in accordance with 10 CFR 70.22(i)(3)(xiii).

14.4.3.2.2 Onsite and Offsite Emergency Facilities

The applicant's emergency plan includes a list and description of onsite and offsite facilities that could be relied upon in the event of an emergency. The onsite and offsite emergency facilities should be acceptable if they include:

A. A list and description of both onsite and offsite emergency facilities by location and purpose of the facility.

B. A description of emergency monitoring equipment which is available for personnel and area monitoring, as well as that for assessing the release of radioactive or hazardous materials to the environment.

C. A description of the onsite and offsite services which support emergency response operations, including:

   i. Decontamination facilities;
   ii. Medical treatment facilities;
   iii. First aid personnel;
   iv. Fire fighters;
   v. Law enforcement assistance; and
   vi. Ambulance services.

D. In addition, the applicant's emergency facilities, equipment, and resources are ready to support emergency response operations, including:

   i. Facilities of adequate size and appropriate location that are designated, equipped, and ready for emergency use;

   ii. Adequate backup facilities required by the emergency plan and supporting documents that are available and ready for use;

   iii. Appropriate equipment and supplies necessary to support emergency response activities that are accessible during accident conditions;

   iv. Emergency equipment that is inventoried, tested, and serviced on a periodic basis to ensure accountability and reliability;

   v. Sufficient reliable primary and backup communications channels that are available to accommodate emergency needs;

   vi. Offsite emergency resources and services that are identified, and are ready to ensure their timely mobilization and use;

vii. Operational engineering information, such as current as-built drawings and procedures, that are readily available in the emergency facilities;

viii. Sufficient equipment for personnel protection and monitoring; and

ix. Systems in place to alert onsite and offsite personnel in the event of an emergency.

14.4.3.2.3 Types of Accidents

The applicant's emergency plan includes a description for each accident identified by the ISA for which protective actions may be needed. The types of accidents should be acceptable if they include:

A. The process and physical location(s) where accidents could occur;

B. Complicating factors and possible onsite and offsite consequences, including nonradioactive hazardous material releases that could impact emergency response efforts;

C. The accident sequence that has the potential for the greatest radiological and toxic chemical impact; and

D. Figure(s) projecting dose and toxic substance concentration as a function of distance and time for various meteorological stability classes.

14.4.3.2.4 Classification of Accidents

The applicant's emergency classification system for classifying events at the facility should be acceptable if it includes:

A. The following two event classifications:

i. "Alert:" Events that may occur, are in progress, or have occurred that could lead to a release of radioactive material or hazardous chemicals incident to the process, but the release is not expected to require a response by an offsite response organization to protect persons offsite; and

ii. "Site area emergency:" Events that may occur, are in progress, or have occurred that could lead to a significant release of radioactive material or hazardous chemicals incident to the process that could require a response by offsite emergency response organizations to protect persons offsite.

B. For each accident in the emergency plan, the classification (alert or site area emergency) that is expected for each accident is identified.

C. The emergency plan specifies emergency action levels (EALs) at which an alert or site area emergency will be declared. EALs are specific conditions that require emergency response measures to be performed. The applicant's EALs are consistent with Appendix A of Regulatory Guide 3.67 and are compared with the Environmental Protection Agency's Protective Action Guides (EPA 400-R-92-001, May 1992 Revision). Transportation accidents more than 1.6 km (1 mile) from the facility are not classified.

D. The emergency plan designates the personnel positions and alternates with the responsibility for accident classification during normal and back shift hours.

14.4.3.2.5 Detection of Accidents

The emergency plan should be acceptable if it describes, for each type of accident identified:

A. The means of detecting the accident;
B. The means of detecting any release of radioactive or other hazardous material;
C. The means of alerting the operating staff; and
D. The anticipated response of the operating staff.

14.4.3.2.6 Mitigation of Consequences

The applicant's emergency plan should be acceptable if it adequately describes mitigation of consequences, including:

A. The emergency plan describes for each accident identified, adequate measures and equipment for safe shutdown and for mitigating the consequences to workers onsite and offsite as well as to the public offsite.

B. For impending danger from an accident initiator, the emergency plan describes the following:

   i. The criteria that will be used to determine whether a single process or the entire facility will be shut down;

   ii. The steps that will be taken to ensure a safe orderly shutdown of a single process or the entire facility;

   iii. The approximate time required to accomplish a safe shutdown of processes; and

   iv. The compensatory measures required for safety during the shutdown period following an accident.

14.4.3.2.7 Assessment of Releases

The applicant's emergency plan should be acceptable if it describes how the applicant assesses any radioactive material or hazardous chemical releases, including:

A. The applicant's procedures to promptly and effectively assess the release of radioactive material or hazardous chemicals associated with the processing of radioactive material, including:

   i. The procedures for estimating or measuring the release rate or source term;

   ii. Valid computer codes used to project doses or concentrations to the public or environment and associated assumptions, along with adequate justifications to show the validity of the assumptions;

   iii. The types, methods, frequencies, implementation time, and other details of onsite and offsite sampling and monitoring that will be performed to assess a release of radioactive material or hazardous chemicals; and

   iv. Method for assessing collateral damage to the facility, especially safety controls.

B. The applicant's procedure for validating any code used to assess releases of radioactive material or hazardous chemicals.

14.4.3.2.8 Responsibilities

The applicant's emergency plan should be acceptable if it describes the emergency response organization and administration which ensures effective planning, implementation, and control of emergency preparedness activities and meets the following criteria:

A. The organizational structure and chain of command are clearly defined;

B. Staffing and resources are sufficient to accomplish assigned tasks;

C. Responsibilities and authority for each management, supervisory, and professional position are clearly defined. Responsibility is assigned for the coordination of onsite and offsite radiation/hazardous material emergency response preparedness;

D. Interfaces with supporting groups, both onsite and offsite, are clearly defined;

E. Mutual cooperation agreements exist with local agencies such as fire, police, ambulance/rescue, and medical units;

F. Plant management measures include audit and assessment (SRP Section 15.6) of emergency preparedness to ensure site readiness to handle emergencies and to identify and correct problems;

G. The onsite emergency response organization as described provides reasonable assurance of effective command and control of the site during the assessment, mitigation, and recovery phase of an accident;

H. The emergency public information staff provides advance and ongoing information to the media and public on subjects that would be discussed during an emergency, such as radiation hazards, chemical hazards, site operation, and site emergency plans; and

I. The schedule of emergency preparedness procedure development provides for availability of procedures to support start-up and operation of new processes/facilities onsite.

14.4.3.2.9 Notification and Coordination

The applicant's emergency plan should be acceptable if it adequately describes the applicant's notification and coordination procedures, including:

A. Reasonable assurance that emergency notification procedures will enable the emergency organization to correctly classify emergencies, notify emergency response personnel, and initiate or recommend appropriate actions in a timely manner, based on the following:

   i. Classification of emergency events are based on the current emergency plan;

   ii. Notification procedures minimize distractions of shift operating personnel and include concise, preformatted messages. Appropriate follow-up messages to offsite authorities are issued in a timely manner;

   iii. Information on the nature and magnitude of the hazards are made available to appropriate emergency response personnel;

   iv. Radiological and chemical source term data are available to the command post, technical support center, emergency operations center, and appropriate State personnel, in cooperation with NRC;

   v. When available, offsite field monitoring data are logged, compared with source term data, and used in the protective action recommendation process;

   vi. Protective Action Guides are available and used by appropriate personnel in a timely manner;

vii. The emergency public information program ensures timely dissemination of accurate, reliable, and understandable information;

viii. Systems are in place, if required, to alert, notify, and mobilize onsite and offsite response personnel in the event of an emergency;

ix. Notification and coordination with responsible parties when some personnel, equipment, and facility components are not available.

B. How and by whom the following actions will promptly and effectively be taken:

i. Decision to declare an alert or site area emergency;

ii. Activation of onsite emergency response organization during all shifts;

iii. Prompt notification of offsite response authorities that an alert or site area emergency has been declared, including the licensee's initial recommendation for offsite protective actions (normally within 15 minutes);

iv. Notification to the NRC Operations Center (as soon as possible and, in any case, no later than one hour after a declared emergency);

v. Decision on what onsite protective actions to initiate;

vi. Decision on what offsite protective actions to recommend;

vii. Decision to request support from offsite organizations; and

viii. Decision to terminate the emergency or enter recovery mode.

14.4.3.2.10 Information To Be Communicated

The applicant's emergency plan should be acceptable if it describes the information to be communicated during an emergency and includes:

A. A standard reporting checklist to facilitate timely notification;

B. The types of information to be provided concerning facility status, radioactive or hazardous chemical releases, and protective action recommendations;

C. A description of preplanned protective action recommendations to be made to each appropriate offsite organization;

D. The offsite officials to be notified, as a function of the classification of the event;

E. The recommended actions to be implemented by offsite organizations for each accident treated in the emergency plan.

14.4.3.2.11 Training

The applicant's emergency plan includes an adequate training program for onsite and offsite emergency response personnel to ensure knowledge of the emergency plan, assigned duties, and effectively respond to an actual emergency. The training program should be acceptable if it includes:

A. The topics and general content of training programs used for training the onsite and offsite emergency response personnel to satisfy the objectives described above;

B. The administration of the training program, including responsibility for training, the positions to be trained, the schedules for training, the frequency of retraining, use of team training and the estimated number of hours of initial training and retraining;

C. The training to be provided on the use of protective equipment such as respirators, protective clothing, monitoring devices, and other equipment used in emergency response;

D. The training program for onsite personnel who are not members of the emergency response staff; and

E. The instructions and tours that will be offered to fire, police, medical, and other emergency personnel to the extent necessary commensurate with the results of the ISA.

14.4.3.2.12 Safe Shutdown (recovery and plant restoration)

The applicant's emergency plan describes the plans for adequately restoring the facility to a safe status after an accident and recovery after an emergency. The safe shutdown should be acceptable if it includes:

A. Appropriate methods and responsibilities for assessing the damage to and the status of the facility's capabilities to safely control radioactive material or hazardous chemicals associated with the process;

B. Procedures for promptly determining the actions necessary to reduce any ongoing releases of radioactive or other hazardous chemicals and to prevent further incidents;

C. Provisions for promptly and effectively accomplishing required restoration action; and

D. Describing the key positions in the recovery organization.

14.4.3.2.13 Exercises and Drills

The applicant's emergency plan commits to conducting exercises and drills in a manner that demonstrates the capability of the organization to plan and perform an effective response to an emergency. The commitment should be acceptable if it demonstrates:

A. Task-related knowledge is demonstrated through periodic participation by all qualified individuals for each position in the emergency response organization;

B. Drill performance is assessed against specific scenario objectives, using postulated accidents, that adequately test personnel, equipment, and resources, including previously identified weaknesses;

C. Effective player, controller, evaluator, and observer pre-drill briefings are conducted;

D. Scenario data and exercise messages provided by the controllers effectively maintain the time line and do not interfere with the emergency organization's response to exercise scenario events, except where safety considerations are involved;

E. Trained evaluators are used to identify and record participant performance, scenario strengths and deficiencies, and equipment problems;

F. Prestaging of equipment and personnel is minimized to realistically test the activation and staffing of emergency facilities;

G. Critiques are conducted in a timely manner and include a follow-up plan for correcting identified weaknesses and improving training effectiveness;

H. Emergency drills demonstrate that resources are effectively used to control the site, to mitigate further damage, to control radiological/chemical releases, to perform required onsite activities under simulated radiation/airborne and other emergency conditions, to provide accurate assessments and status during an accident, and to initiate recovery;

I. Emergency drills demonstrate personnel protection measures, including controlling and minimizing hazards to individuals during events such as fires, medical emergencies, mitigation activities, search and rescue, and other similar events;

J. The emergency drill demonstrates that onsite communications effectively support emergency response activities;

K. The emergency drill demonstrates that the emergency public information organization disseminates accurate, reliable, timely, and understandable information;

L. Provisions are made for conducting quarterly communications checks with offsite response organizations;

M. Offsite organizations are invited to participate in the biennial onsite exercise that tests the major elements of the emergency plan and response organizations.

14.4.3.2.14 Responsibilities for Developing and Maintaining Current the Emergency Program and Its Procedures

The applicant's emergency plan describes the responsibilities for developing and maintaining the emergency program and its procedures. The responsibilities should be acceptable if they include:

A. The means for ensuring that the revisions to the emergency plan and the procedures which implement the emergency plan are adequately prepared, kept up to date normally (within 30 days of any changes), and distributed to all affected parties including the NRC.

B. The provisions for approval of the implementing emergency procedures, making and distributing changes to the procedures, and ensuring that each person responsible for an emergency response function has immediate access to a current copy of emergency procedures.

C. The provisions for approval of changes to the emergency plan and the procedures and those individuals authorized to make these changes;

D. Procedures for allowing offsite response organizations 60 days to comment on the emergency plan before submitting it to the NRC, and to provide NRC any comments received within 60 days along with the plan; and

E. Procedures for modifying the emergency plan in accordance with 10 CFR 70.32(i).

14.5 REVIEW PROCEDURES

14.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application (construction or license) adequately addresses the items in Section 14.1.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The applicant is not expected to submit either an emergency plan as described in Section 14.3.2 or an evaluation as described in Section 14.3.1 with the application for construction approval. However, the primary reviewer should evaluate the safety assessment of the design basis to ensure that the commitments and program goals are appropriate for emergency protection at the design stage.

B. License Application

Specifically, the license application should either contain an evaluation described in Section 14.3.1 or an emergency plan as described in Section 14.3.2.

If the primary reviewer verifies that emergency protection is adequately addressed in the application for construction approval or the license application, the primary reviewer should accept the application for the safety evaluation in Section 13.1.5.2. 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.

14.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 14.5.1(A) (application for construction approval) or 14.5.1(B) (license application), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 14.4. 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 14.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The primary reviewer should ensure that the design basis includes appropriate commitments for emergency protection at the design stage. For example, if the safety assessment of the design basis shows a dose to a member of the public that exceeds the limits in 10 CFR 70.22(i)(1)(i), the applicant should commit to providing an emergency plan with the license application.

B. License Application

i. No Emergency Plan

The primary reviewer should verify that the applicant's evaluation is consistent with the potential accident sequences described in the ISA. The ISA reviewer and the primary reviewer should coordinate to assure the resolution of any issues concerning the evaluation relative to ISA information. The final step for the primary reviewer should be to prepare a Safety Evaluation Report (SER) in accordance with Section 14.6 which either agrees with the applicant's conclusion that no emergency plan is required or indicates that the staff does not accept the applicant's evaluation and recommends that an emergency plan be required by the applicant.

ii. Emergency Plan

After it is determined that an acceptable application containing an emergency plan has been received from the applicant, the primary reviewer should conduct a complete review of the emergency plan and determine its acceptability in accordance with Section 14.4.3.2. The reviewer should verify that emergency planning is consistent with the potential accident sequences described in the ISA. The ISA reviewer and emergency plan reviewer should coordinate to assure the resolution of any issues concerning the emergency plan relative to ISA information. This information may be supplemented by a personal visit to the site by the primary reviewer and meetings with the applicant. The final step for the primary reviewer should be to prepare an SER in accordance with Section 14.6, "Evaluation Findings."

14.6 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 application for construction approval as follows:

The staff evaluated the application for construction approval for [insert facility name] in accordance with Chapter 14.0 of NUREG-1718. The staff evaluated [insert a summary statement of what was evaluated] and found that [insert a summary statement of the findings]. The NRC staff determined that the applicant's commitments, including the commitment to provide an emergency plan with the license application [if the applicant's design basis safety assessment shows it is required], are adequate to meet the requirements for a construction approval in accordance with 10 CFR Part 70.

The staff could document the safety evaluation for the license application, where the applicant submits an emergency plan, as follows:

The staff evaluated the emergency plan submitted as part of the license application for [insert facility name] to possess and use SNM in accordance with Chapter 14.0 of NUREG-1718. The staff evaluated [insert a summary statement of what was evaluated] and found that [insert a summary statement of the findings]. In accordance with 10 CFR 70.22(i), the licensee commits to maintaining and executing an emergency plan for responding to the radiological hazards resulting from a release of radioactive material and to any associated chemical process hazards. NRC staff determined that the applicant's emergency plan is adequate to demonstrate compliance with 10 CFR 70.22(i), including: (1) the plant is properly configured to limit releases of radioactive materials in the event of an accident, (2) a capability exists for measuring and assessing the significance of accidental releases of radioactive materials, (3) appropriate emergency equipment and procedures are provided onsite to protect workers against radiation and other chemical hazards that might be encountered following an accident, (4) a notification system has been established for notifying Federal, State, and local government agencies and recommending appropriate protective actions to protect members of the public, and (5) necessary recovery actions are established for returning the plant to a safe condition following an accident. The requirements of the emergency plan are implemented through approved written procedures. Changes which decrease the effectiveness of the emergency plan may not be made without NRC approval. The NRC will be notified of other changes which do not decrease the effectiveness of the emergency plan within six months of the changes.

The NRC staff concluded that the applicant's emergency plan meets the requirements of 10 CFR 70.22(i).

14.7 REFERENCES

A. U.S. Nuclear Regulatory Commission, Part 30 Statements of Consideration and Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees, Federal Register 54, 14051, 1989.

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

C. NUREG/BR-0150, Vol. 1, Rev. 4, RTM-96 Response Technical Manual, U.S. Nuclear Regulatory Commission, 1996.

D. EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, Environmental Protection Agency, May 1992.

15.0 MANAGEMENT MEASURES
15.1 QUALITY ASSURANCE

15.1.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the applicant has a quality assurance (QA) program that will provide reasonable assurance against natural phenomena and the consequences of potential accidents through the QA program's application to the design, fabrication, construction, testing and operation of the applicant's structures, systems, and components[1] (SSCs); the applicant is required to describe the QA program as part of the application for construction approval under 10 CFR 70.22(f). This review also establishes that the applicant has a QA program that will provide reasonable assurance that all items relied on for safety[2] (IROFS) will be available and reliable to perform their designated safety functions when needed, which the applicant is required to describe as part of license application under proposed 10 CFR Part 70.

15.1.2 RESPONSIBILITY FOR REVIEW

Primary: QA Engineer/Specialist

Secondary: Project Manager

Supporting: Fuel Cycle Facility Inspector
Primary Reviewers of applicable SRP Chapters 5.0 through 15.0

15.1.3 AREAS OF REVIEW

The applicant is required to submit a description of the QA program with the application for construction approval and should update the QA program in the license application. The areas of review should include:

A. Organization

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[1] "Structures, systems, and components" are, by definition, items relied on for safety (see Footnote 2). For the purposes of the review guidance provided under this section, references to items relied on for safety are intended to include the SSCs identified in the application for construction approval.

[2] "Items relied on for safety" is defined in the proposed 10 CFR 70, as revised, as "structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at the facility that could exceed the performance requirements specified in §70.61 or to mitigate their potential consequences."

Management Measures

B. QA Function[3]
C. Design Control
D. Procurement Document Control
E. Instructions, Procedures,[4] and Drawings
F. Document Control
G. Control of Purchased Items
H. Identification and Control of Items
I. Control of Special Processes
J. Inspection
K. Test Control
L. Control of Measuring and Test Equipment
M. Handling, Storage, and Shipping
N. Inspection, Test, and Operating Status
O. Nonconformances
P. Corrective Action
Q. QA Records
R. Audits and Assessments[5]
S. Applicant's Provisions for Continuing QA

15.1.4 ACCEPTANCE CRITERIA

15.1.4.1 Regulatory Requirements

The regulation, 10 CFR Part 70, as proposed, requires that the applicant establish an appropriate QA program to ensure that all items relied on for safety perform their designated safety functions and are continually available and reliable. The regulatory requirements for QA are addressed in the following:

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.

In addition, an applicant to possess and use special nuclear material in a plutonium processing and fuel fabrication facility such as the mixed oxide (MOX) fuel fabrication facility is required,

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[3] SRP Section 15.4 addresses training and qualification of plant personnel. Section F2 of SRP Appendix F on QA addresses training and qualification of other personnel.

[4] SRP Section 15.5 addresses plant procedures. Section F5 of SRP Appendix F on QA addresses other procedures.

[5] Guidance for audits and assessments is given in SRP Section 15.6 as referenced in SRP Appendix F on QA.

pursuant to §70.22(f), to describe the QA program to be applied to the design, fabrication, construction, testing, and operation of the structures, systems, and components of the facility. The footnote of §70.22(f) states that the description of the QA program should include a discussion of how the criteria in Appendix B of 10 CFR Part 50 will be met.

15.1.4.2 Regulatory Guidance

Guidance for QA is addressed in the following:

American Society of Mechanical Engineers, "Quality Assurance Requirements for Nuclear Facility Applications." (An American National Standard), NQA-1-1994, New York. 1994.

Note that while this standard has separate sections for "requirements" and "guidance," NRC's regulatory QA requirements exist only in the applicable Commission regulations.

15.1.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find that the applicant's QA program adequately addresses and satisfies the regulatory acceptance criteria below. The applicant may reference material in other sections of application for construction approval or the license application, or incorporate material by reference, provided that these references are clear and specific.

The applicant should identify the SSCs (application for construction approval) or items relied on for safety (license application) and the degree of their importance. The graded approach for the application of QA should be described unless the applicant chooses to apply the highest level of QA and quality control to all SSCs or items relied on for safety.

For SSCs (application for construction approval) or items relied on for safety (license application), the applicant should apply either Option A or Option B (whichever the applicant chooses with the application for construction approval) as described below.

> Option A. Address the regulatory acceptance criteria given in this section and provide a commitment to implement and maintain the QA program in conformance with the applicable "requirements" of Parts I and II of NQA-1-1994 or equivalent.
>
> **OR**
>
> Option B. Address the checklist provided in SRP Appendix F on QA.

Depending on the option chosen, the applicant should address the criteria specified below. That is, if Option A is used, the applicant should (a) include a commitment that it will implement and

maintain its QA program to comply with the applicable "requirements" of NQA-1-1994[6] (that is, the basic and supplemental "requirements" of Parts I and II) or equivalent and should (b) be responsive to the three regulatory acceptance criteria given below. Note that, if Option A is used, only a verification of that commitment and of the response to the regulatory acceptance criteria given below should be performed.

A. Organization

The applicant should describe the organizational structure, functional responsibilities, charts of the lines of responsibilities, interrelationships, and areas of responsibility and authority for all organizations performing activities relied on for safety, including the applicant's organization and, if applicable, the organization of the applicant's principal contractors (architect/engineer, constructor, construction manager, and/or operator). Persons or organizations responsible for ensuring that appropriate QA has been established and verifying that activities affecting quality/safety have been correctly performed should have sufficient authority, access to work areas, and organizational independence to carry out their responsibilities.

B. QA Function

QA should be well-documented, planned, implemented, and maintained to ensure the availability and reliability of controls relied on for safety. It should be implemented during all phases of the facility's life. It should be functional prior to performing the Integrated Safety Analysis required by Part 70, as revised.

C. Applicant's Provisions for Continuing QA

The applicant's provisions for continuing QA should address review and updates based on reorganizations, revised activities, lessons learned, changes to applicable regulations, and other QA changes.

If Option B is used, the application should address the checklist items in SRP Appendix F on QA.

Also, the applicant should commit to update the QA program to reflect any changes between the application for construction approval and the application for a license.

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[6] This SRP section refers to regulatory QA requirements and NQA-1 "requirements." Regulatory QA requirements are given in the Part 70, as revised. NQA-1 "requirements" are the Basic and Supplementary Requirements given in Parts I and II of ASME NQA-1-1994.

15.1.5 REVIEW PROCEDURES

15.1.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval or license application adequately addresses the items in Section 15.1.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the application for construction approval should address Section 15.1.3 in full and should identify whether Option A or Option B of Section 15.1.4.3 has been chosen.

B. License Application

The areas of review for the updated material in the license application should include Items A through S identified in Section 15.1.3.1, with special attention on the identification of any new or changed aspects of the QA program.

Note that the applicant's commitment to implement and maintain its QA in conformance with the applicable basic and supplemental "requirements" of Parts I and II of ASME NQA-1-1994 or equivalent should satisfy the acceptance review criteria in Item A or B of this section.

If the primary reviewer verifies that QA is adequately addressed in either the application for construction approval or the license application, the primary reviewer should accept the application for the safety evaluation in Section 15.1.5.2. 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.

15.1.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 15.1.5.1(A) (application for construction approval) or 15.1.5.1(B) (license application), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.1.4. 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 15.1.4.

Guidance specific to the application for construction approval and the license application is provided below.

Management Measures

A. Application for Construction Approval

The primary reviewer should review the application for construction approval to determine whether the applicant, for SSCs, has met either Option A or Option B as defined in Section 15.1.4.3.

In either case, the applicant should also (a) describe how the QA will be graded for items of lesser or no effect on consequences of concern (unless the applicant chooses to apply the highest level of QA and quality control to all SSCs) and (b) list the SSCs as determined in the safety assessment of the design basis. The primary reviewer should determine whether the applicant and its principal contractors have adequately planned for QA to be accomplished. Some of the information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear and specific.

The secondary reviewer should confirm that the applicant's and the applicant's principal contractors' QA commitments are consistent with other sections of the application.

The other supporting reviewers should determine, within their areas of review, whether SSCs have been specified with the appropriate level of QA.

The review should result in a determination that there is reasonable assurance that the applicant's and the applicant's principal contractors' QA programs will provide reasonable assurance against natural phenomena and the consequences of potential accidents through the QA program's application to the design, fabrication, construction, testing and operation of the applicant's SSCs.

B. License Application

When the applicant updates the QA program for the license application, new or changed material should include any items relied on for safety identified since the applicant provided the SSCs in the application for construction approval. The primary reviewer should focus the review on any new or changed material and determine whether the necessary QA policies, procedures, and instructions will be in place and will be applied to IROFS before personnel begin activities relied on for safety. The primary reviewer should also confirm that the material presented remains consistent with the material provided in the license application in support of other chapters of this SRP.

The supporting reviewer (Fuel Cycle Facility Inspector) should become familiar with the applicant's and principal contractors' QA commitments and determine whether ongoing activities are in agreement with them.

The review should result in a determination that there is reasonable assurance that the applicant's and the applicant's principal contractors' QA will provide reasonable assurance

that items relied on for safety will be available and reliable to perform their safety functions in a satisfactory manner when needed.

When the safety evaluation is complete, the primary reviewer, with assistance from the other reviewers, should prepare the QA input for the Safety Evaluation Report (SER), as described in Section 15.1.6 using the acceptance criteria from Section 15.1.4. The secondary reviewer should coordinate the QA input with the balance of the reviews and the SER.

15.1.6 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 a safety evaluation for the application for construction approval as follows:

The staff reviewed the quality assurance (QA) program for the application for construction approval for [insert facility name] according to Chapter 15.1 of NUREG-1718. [Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the application acceptable.] Based on its review of the application for construction approval, the NRC staff concluded that (A) the applicant has adequately described its QA program and (B) the applicant's QA program meets the regulatory requirements of 10 CFR Part 70 and thus the applicant's QA program, as applied to SSCs, will provide reasonable assurance of protection against natural phenomena and the consequences of potential accidents.

The staff could document a safety evaluation for the license application as follows:

The staff reviewed the quality assurance (QA) program for the application for construction approval for [insert facility name] according to Chapter 15.1 of NUREG-1718. [Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the application acceptable.] Based on its review of the license application, focusing on new or updated material when compared to the safety evaluation for the construction approval, the NRC staff concludes that (A) the applicant has adequately described its updated QA program and (B) the applicant's updated QA program meets the regulatory requirements of 10 CFR Part 70 and thus provides reasonable assurance that all items relied on for safety will be available and reliable to perform their designated safety functions when needed.

15.1.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. 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.

C. American Society of Mechanical Engineers (ASME), "Quality Assurance Requirements for Nuclear Facility Applications," (An American National Standard). ASME NQA-1-1994, New \York. 1994.

15.0 MANAGEMENT MEASURES
15.2 CONFIGURATION MANAGEMENT

15.2.1 PURPOSE OF REVIEW

The purpose of this review is to establish with reasonable assurance that the applicant has a plan for or has implemented an acceptable configuration management (CM) system. The review should result in a determination that the applicant has described and committed to a CM system during design and construction (as described in the application for construction approval) and operations (as updated and described in the license application) that provides reasonable assurance that the applicant will maintain design information, safety information, and modifications (both temporary and permanent for design and operations) that might impact the ability of structures, systems, or components[1] (SSCs) (application for construction approval) or items relied on for safety (IROFS) (license application) to perform their function when needed in a consistent and up-to-date manner. The review should also result in a determination that the applicant's CM system captures formal documentation governing the design and continued maintenance of the SSCs (application for construction approval) or items relied on for safety (license application) and supporting management measures, as identified and described in the integrated safety analysis (ISA) programmatic commitments and ISA Summary (see Chapter 5.0). The review should ensure that the CM system is adequately coordinated and integrated with the other management measures such as maintenance, quality assurance, training and qualifications, procedures, and audits and assessments.

15.2.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: Primary ISA Reviewer, Quality Assurance Reviewer, Records Management Reviewer, Organization and Administration Reviewer

Supporting: Fuel Cycle Facility Inspector

15.2.3 AREAS OF REVIEW

The applicant should submit a description of the CM system with the application for construction approval and should submit updated information with the license application. The applicant's descriptions and commitments for CM should be reviewed with an emphasis on the processes for documenting an established baseline configuration and controlling changes to it to preclude inadvertent degradation of safety. An examination should be conducted of the descriptions of the organizational structure responsible for CM activities and the process, procedures, and

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[1] "Structures, systems, and components" are, by definition, items relied on for safety (see proposed 10 CFR 70.4 or the glossary to this SRP).

documentation required by the applicant for modifying the SSCs (application for construction approval) or items relied on for safety (license application) and the supporting management measures. The review should focus on the applicant's management level controls that ensure (a) the disciplined documentation of engineering, installation, and operation of modifications; (b) the training and qualification of affected staff; (c) revision and distribution of operating, test, calibration, surveillance, and maintenance procedures and drawings; (d) post-modification testing; and (e) operational readiness review.

The following topics should be reviewed:

A. CM Policy

The review should cover the applicant's description of overall CM systems, including at least the following topics: (a) the scope of the SSCs (application for construction approval) or items relied on for safety (license application) to be included in the CM system, (b) objectives of each CM activity, (c) a description of each CM activity, and (d) the organizational structure and staffing interfaces.

The review should examine the applicant's establishment of a baseline CM policy applicable to all design and construction (application for construction approval) and operations (license application), initially independent of the safety assessment of the design basis (application for construction approval) or the ISA results (license application), respectively. The review should also examine any reduced level of CM that the applicant may propose for certain SSCs (application for construction approval) or items relied on for safety (license application) based on the safety assessment of the design basis or ISA results, respectively.

Specifically, the primary reviewer should review the CM plan that provides management commitments and policy directives and defines key responsibilities, terminology, and equipment scope. The method for initiating immediate corrective actions should be examined. The secondary reviewers should examine the safety assessment of the design basis (application for construction approval) or the ISA Summary (license application) for the identification of dependence on CM of SSCs or items relied on for safety. Appropriate interfaces both within the CM system and with other facility organizations and functions should be examined. In particular, the quality assurance reviewer should assist in examining the functional interfaces with quality assurance (QA), maintenance, and training (including qualification). The reviewers should look for the applicant's identification of required databases and the rules for their maintenance. The reviewers should examine implementing procedures for the CM system.

B. Design Requirements

The review should cover the applicant's demonstration that design requirements and associated design bases have been established and are maintained by an appropriate organizational unit. The applicant's CM controls on the design requirements and the safety

assessment of the design basis (application for construction approval) or the ISA (license application) should be evaluated. The review should be coordinated with the primary reviewer of Chapter 5.0.

C. Document Control

The review should include the applicant's methods used to establish and control documents within the CM system.

D. Change Control

The review should examine the applicant's commitments to ensure that the CM system maintains strict consistency among the design requirements, the construction or physical configuration, and the facility documentation. An important component of this review is the applicant's process, within the CM system, for ensuring that the safety assessment of the design basis (application for construction approval) or the ISA (license application) will be systematically reviewed and modified to reflect design or operational changes from an established safety basis, and that all other documents that are affected by safety basis changes will be properly modified, authoritatively approved, and made available to personnel.

E. Assessments

The review should examine the applicant's commitments to conduct initial and periodic assessments of the CM system to determine the system's effectiveness and to correct deficiencies, consistent with the acceptance criteria in SRP Section 15.6, "Audits and Assessments."

15.2.4 ACCEPTANCE CRITERIA

15.2.4.1 Regulatory Requirements

The staff's requirements applicable to CM are the following:

10 CFR 70.62(d), relating to the requirement that the applicant or licensee is to establish management measures to provide continuing assurance of compliance with the performance requirements.

10 CFR 70.64(a)(1), relating to the requirement that the design of new facilities or the design of new processes at existing facilities be developed and implemented in accordance with management measures.

10 CFR 70.65(a), relating to the requirement that the application is to include a description of the management measures.

10 CFR 70.72(a), relating to the requirement the licensee is required to establish a CM system.

15.2.4.2 Regulatory Guidance

There are no regulatory guides that apply to CM for a new facility licensed under 10 CFR Part 70.

15.2.4.3 Regulatory Acceptance Criteria

The reviewers should determine that an applicant's CM system is acceptable if it satisfies the following criteria:

A. CM Policy

The applicant's description of its overall CM system describes at least the following topics: (a) the scope of the SSCs (application for construction approval) or items relied on for safety (license application) and supporting management measures to be included in the CM system (coordinate with reviewer of Chapter 5.0), (b) a description of each CM system activity, (c) the objectives of each CM system activity, (d) any reduced level of CM that the applicant may propose for certain SSCs (application for construction approval) or items relied on for safety (license application) based on the safety assessment of the design basis or the ISA results, respectively, and (e) the organizational structure and staffing interfaces.

The scope of SSCs (application for construction approval) or items relied on for safety (license application) includes all those SSCs (application for construction approval) or items relied on for safety (license application) as defined by the safety assessment of the design basis or the ISA, respectively; furthermore, those items are included in the QA, maintenance, and training and qualifications programs. The functional interfaces with QA, maintenance, and training and qualification are of particular importance and should be addressed individually.

B. Design Requirements

The applicant demonstrates that design requirements and associated design bases have been established and are maintained by an appropriate organizational unit. The applicant demonstrates that the CM system provides for keeping design requirements and the safety assessment of the design basis (application for construction approval) or the ISA (license application) current and that suitable hazard/accident analysis methods, including controlled computer codes, if applicable, are available to evaluate safety margins of proposed changes. Technical management review and approval procedures are described.

The design process leading to drawings and other statements of requirements proceeds logically from the design basis. Specific personnel are assigned the responsibility for maintaining the design bases and requirements. These may be the same personnel that maintain the safety assessment of the design basis (application for construction approval) or the ISA (license application) and controlled computer codes. SSCs (application for construction approval) or items relied on for safety (license application) to be listed under CM are clearly defined in the requirements documents, along with the assignment of any grades or quality levels. The grades or quality levels, if specified, are based on the qualitative risk associated with postulated accident sequences in which the SSCs (application for construction approval) or items relied on for safety (license application) are required to function. The applicant should have indicated in the safety assessment of the design basis (application for construction approval) or the ISA (license application) what level of CM attributes are applied to a particular item. However, in the safety assessment of the design basis or ISA this indication may only consist of an index or category designation. The definition of the multiple CM levels, if used, should be in the CM chapter of the application.

C. Document Control

The applicant describes an acceptable method to establish and control documents within the CM system, including cataloging the document data base, the information content of the document data base, maintenance and distribution of documents, document retention policies, and document retrieval policies. A list of the types of documents controlled is established and includes key documents, such as drawings, procurement specifications, engineering analyses, operating procedures, training/qualification records, and maintenance procedures.

The applicant's material shows that the CM system will capture documents that are relevant and important to safety. This includes design requirements; the safety assessment of the design basis (application for construction) or the ISA (license application); as-built drawings; specifications; all safety-important operating procedures; procedures involving training, QA, maintenance, audits and assessments; emergency operating procedures; emergency response plans; system modification documents; assessment reports; and others, as necessary, that the applicant may deem part of the CM system. A controlled document database is used to control documents and track document change status. Rules of storage for originals or master copies of documents within the CM system should follow the guidance of "Records Management" discussed in SRP Section 15.8.

D. Change Control

The applicant demonstrates that the CM system will maintain strict consistency among the design requirements, the physical configuration, and the facility documentation. The applicant commits to an acceptable process for identifying and authorizing proposed

changes; performing appropriate technical, management, and safety reviews of proposed changes in configurations of SSCs (application for construction approval) or items relied on for safety (license application); approving changes; tracking and implementing changes; and documenting changes (including placement of documentation in a document control center and dissemination to affected functions such as training, engineering, operations, maintenance, and QA). The applicant describes an acceptable process, within the CM system, for ensuring that the safety assessment of the design basis (application for construction approval) or the ISA (license application) is systematically reviewed and modified to reflect design or operational changes from an established safety basis, and that all documents outside the safety assessment of the design basis (application for construction approval) or the ISA (license application) that are affected by safety basis changes are properly modified, authoritatively approved, and made available to personnel.

Post-modification testing of items (or procedure drills or walk-throughs) may be performed in conjunction with periodic item performance monitoring and normal maintenance functions.

E. Assessments

The applicant confirms that assessments, including initial and periodic examinations of the CM system, will be conducted to determine the system's effectiveness and to correct deficiencies. The applicant indicates that such assessments will be systematically planned and conducted in accordance with an overall facility audit and assessment program as described by the applicant and reviewed by the NRC in accordance with Section 15.6 of this SRP.

Both document assessments and physical assessments (system walkdowns) will be conducted periodically to check the adequacy of the CM system. All assessments and follow-ups are documented. These reports can provide a supporting basis for future changes. Assessments will include reviews of safety systems from design requirements through implementation.

Also, the applicant should commit to update the CM system to reflect any changes between the application for construction approval and for a license.

15.2.5 REVIEW PROCEDURES

15.2.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for construction approval or license application adequately addresses the items in Section 15.2.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The application for construction approval should address each item in Section 15.2.3 with an emphasis on the CM for managing the design basis during design and construction. This should include a reviewer determination that the applicant committed to a formal CM system for establishing the design basis and reviewing proposed changes to SSCs.

B. License Application

The license application should address each item in Section 15.2.3 with an emphasis on the CM for operation (e.g., procedures, maintenance, and training) and any new or changed material in the CM program that will arise as a part of the transition from design and construction (design basis) to operations (integrated safety analysis). This should include a reviewer determination that the applicant committed to a formal CM system for establishing and managing the ISA and reviewing proposed changes to items relied on for safety or items, procedures, and processes that may impact items relied on for safety.

If the primary reviewer verifies that CM is adequately addressed in the application for construction approval or the license application, the primary reviewer should accept the application for the safety evaluation in Section 15.2.5.2. 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.

15.2.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with either Section 15.2.5.1(A) (application for construction approval) or 15.2.5.1(B) (license application), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.2.4. 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 15.2.4.

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The primary reviewer should determine whether the applicant has adequately planned for CM to be accomplished during design and construction and whether necessary policies, personnel, procedures, and instructions will be in place to begin CM early, that is, during the safety assessment of the design basis and the design and construction of the SSCs. The

secondary reviewers should confirm that the applicant's CM commitments are consistent with other sections of the application.

B. License Application

When the applicant updates the CM system for the license application, the primary reviewer should focus the review on any new or changed material. Particularly, the primary reviewer should ensure that the applicant has adequately planned for CM to be accomplished during operations and whether necessary policies, personnel, procedures, and instructions will be in place to transition from CM during design and construction to CM during operations, that is, from the safety assessment of the design basis and the design and construction of the SSCs to the ISA and the items relied on for safety.

The primary reviewer should also confirm that the material presented remains consistent with the material provided in the license application in support of other chapters of this SRP.

The supporting reviewer (Fuel Cycle Facility Inspector) should become familiar with the applicant's CM commitments and determine whether ongoing activities are in agreement with them.

The license application review should result in a determination that there is reasonable assurance that the CM system will provide additional assurance that items relied on for safety will perform satisfactorily in service and that activities relied on for safety will be performed satisfactorily.

When the safety evaluation is complete, the primary staff reviewer, with assistance from the other reviewers, should prepare the CM input for the safety evaluation report (SER) as described in Section 15.2.6 using the acceptance criteria from Section 15.2.4.

15.2.6 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 a safety evaluation for the application for construction approval as follows:

The staff reviewed the Configuration Management (CM) system for (name of facility) according to Section 15.2 of NUREG-1718. [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.]

Based on its review of the application for construction approval, the NRC staff concluded that the applicant suitably and acceptably described its commitment to a proposed CM system, including the method for managing changes in procedures, facilities, activities, and equipment for SSCs identified in the safety assessment for the design basis. Management level policies and procedures, including an analysis and independent safety review of any proposed activity involving SSCs, are described that will ensure that the relationship between design requirements, construction, and facility documentation is maintained as part of a new design or change in an existing design. The administrative control will ensure that the organizational structure, procedures, and responsibilities necessary to implement CM are in place or committed to; that the design requirements and bases are documented and supported by analyses and the documentation is maintained current; that documents, including drawings, are appropriately stored and accessible; that drawings and related documents adequately describe SSCs; that procedures adequately describe how the applicant will achieve and maintain strict consistency among the design requirements, facility construction, and the facility documentation; that methods are in place for suitable analysis, review, approval, and implementation of identified changes to SSCs.

In situations where the applicant proposes a graded CM system based on risk significance the following can be added:

The applicant described its approach to applying at least two levels of CM attributes to SSCs and identified which SSCs involve lower risk and may receive the reduced level of CM requirements. The applicant's proposed reduced CM features are found adequate to contribute to the reliability and availability of the lesser risk items relied on for safety identified in the application.

The staff could document a safety evaluation for the license application using similar paragraphs as those used for the construction approval, but encompassing the new or updated material when compared to the safety evaluation for the construction approval, and addressing CM as applied to IROFS during operations, including controls to assure configuration verification, correct functional tests, accurate documentation for equipment and procedures, adequate methods or plans for initial and periodic examination of the CM system's effectiveness, and thorough assessments and follow-up reports of corrective actions.

15.2.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. 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.

C. Department of Energy (U.S.) (DOE). DOE-STD-1073-93, "DOE Standard: Guide for Operational Configuration Management Function." Parts I and II, DOE: 1993.

15.0 MANAGEMENT MEASURES
15.3 MAINTENANCE

15.3.1 PURPOSE OF REVIEW

The purpose of this review is to establish reasonable assurance that the facility will have an adequate maintenance program for items relied on for safety--with the exception of personnel activities--to ensure their availability and reliability to perform their intended safety functions when needed. The maintenance performed to meet the availability and reliability requirements for the items relied on for safety should be commensurate with risk levels identified in the ISA Summary.

15.3.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: Quality Assurance, Criticality, Chemical, Fire, Radiation Protection and Environmental Reviewers

Supporting: Fuel Cycle Facility Inspector

15.3.3 AREAS OF REVIEW

The applicant's description of its maintenance program should be reviewed during the license application with emphasis on demonstrating that items relied on for safety with the exception of personnel activities (safety controls) are inspected, calibrated, tested and maintained so as to ensure their ability to perform their safety functions when needed. The safety controls should be identified by the ISA Summary (discussed in Chapter 5.0 of this SRP). Individual components and support systems for the safety controls may have to be individually maintained to ensure the availability and reliability of the control function. The reviewers should review the applicant's description of how each of the following essential components is implemented within the site organization:

A. Surveillance/monitoring;
B. Corrective maintenance;
C. Preventive maintenance; and
D. Functional testing.

15.3.4 ACCEPTANCE CRITERIA

15.3.4.1 Regulatory Requirements

The requirement for maintenance is addressed in the following:

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.

Specific references are as follows:

A. In § 70.4, "Definitions," the term management measures is defined. Maintenance is included as a management measure.

B. In § 70.62(d), the applicant is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

C. In § 70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

D. In § 70.64(a)(8), inspection, testing, and maintenance are required to be addressed as one of the Baseline Design Criteria to provide reasonable assurance that items relied on for safety will be designed to allow them to be adequately inspected, tested and maintained to ensure their availability and reliability to perform their function when needed.

E. In § 70.65(a), the application is required to include a description of the management measures.

15.3.4.2 Regulatory Guidance

There are no regulatory guides that apply to maintenance for a new facility licensed under 10 CFR Part 70.

15.3.4.3 Regulatory Acceptance Criteria

As part of the application for construction approval, the applicant should commit to establishing a maintenance program which meets or exceeds the acceptance criteria in Section 15.3.4.

The applicant's maintenance program should be considered acceptable (for the license approval) if it adequately addresses the following:

A. Safety Controls Identified in the ISA

An assessment of whether components and support systems need to be individually maintained to ensure the availability and reliability of specific safety controls. The reliability and availability of a particular item should be commensurate with the risk levels identified in the ISA.

B. Essential Components

i. Surveillance/monitoring: The surveillance/monitoring function, its responsible organization, and the conduct of surveillance/monitoring at specified frequencies to measure the degree to which safety functions or safety controls meet performance specifications. This activity is used in setting preventive maintenance frequencies for safety controls and the determination of performance trends for safety controls. How results from incident investigations (described in Section 15.7 of this SRP) and identified root causes are used to modify the affected maintenance function and eliminate or minimize the root cause from recurring should be addressed. For surveillance tests that can only be done while equipment is out of service, proper compensatory measures should be prescribed for the continued normal operation of a process.

ii. Corrective maintenance: The documented approach used to perform corrective actions or repairs on safety controls. The maintenance function should provide a planned, systematic, integrated and controlled approach for the repair and replacement activities associated with identified failures of safety controls.

iii. Preventive maintenance: A description of the preventive maintenance (PM) function that demonstrates a commitment to conduct preplanned and scheduled periodic refurbishing or partial or complete overhaul for the purpose of ensuring that unplanned outages of selected safety controls do not occur. This activity includes using the results of the surveillance/monitoring component of maintenance. Instrumentation calibration and testing should be addressed as part of this component.

iv. Functional testing: A description of the functional testing function that demonstrates a commitment to the functional testing of safety controls after corrective or preventive maintenance or calibration. Functional testing should be conducted using approved procedures that include compensatory measures while the test is being conducted.

C. Work Control Methods

A list of maintenance-related work control methods.

D. Relationship of the Maintenance Elements to Other Management Control Sections Discussed in SRP Chapter 15.0

A discussion of how the maintenance function utilizes, interfaces with, or is linked to these elements.

15.3.5 REVIEW PROCEDURES

15.3.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application for a license adequately addresses the specific items in Section 15.3.3, "Areas of Review." If the primary reviewer verifies that maintenance is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 15.3.5.2. 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.

15.3.5.2 Safety Evaluation

For construction approval, the reviewer should determine that the applicant has committed to a maintenance program that will meet or exceed the acceptance criteria in Section 15.3.4.

For a license application for operations approval and after determining that the application is acceptable for review in accordance with Section 15.3.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.3.4. 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 15.3.4.

The primary reviewer should establish that the applicant's maintenance program meets or exceeds the acceptance criteria. The primary reviewer should determine if the applicant has adequately planned the work to be accomplished and whether necessary policies, procedures, and instructions either are in place or will be in place before work starts. The primary reviewer should also determine that there is reasonable assurance that the applicant's quality assurance, configuration management, and maintenance programs, as described in SRP Sections 15.1 through 15.3, are coordinated.

When an applicant's maintenance program references other sections of the application, the primary reviewer should confirm that these sections of the application are consistent with the applicant's selection of acceptance criteria and the proposed method for implementation.

The primary reviewer should coordinate with secondary staff reviewers to ensure there is no contradiction between maintenance and other areas of the application. The secondary staff reviewers should ensure that the scope of the applicant's maintenance program includes the items relied on for safety that are in their primary review areas of the application. The supporting staff reviewer (Fuel Cycle Facility Inspector) should become familiar with the applicant's maintenance program and determine whether ongoing activities are in agreement with it.

15.3.6 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 construction approval by stating that the applicant has committed to establishing a maintenance program that meets or exceeds the acceptance criteria contained in Section 15.3.4 of NUREG-1718.

The staff could document the safety evaluation for the license application for operations as follows:

The staff reviewed the license application for [insert facility name] according to Section 15.3 of NUREG-1718. Based on the review of the license application, the staff concluded that the applicant committed to maintenance of items relied on for safety with the exception of personnel activities (safety controls). [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] The applicant's maintenance commitments contain the basic elements to ensure availability and reliability: surveillance/monitoring, corrective maintenance, preventive maintenance, and functional testing. The applicant's maintenance function is proactive, using surveillance/monitoring and maintenance records to analyze equipment performance and identify the root causes of repetitive failures.

In addition, the surveillance/monitoring activities described in this section of the application provide assurance of the validity of the ISA by examination and calibration and testing of equipment that monitors process safety parameters and acts to prevent or mitigate accident consequences.

The maintenance function: (1) is based on approved procedures; (2) employs work control methods that properly consider personnel safety, awareness of facility operating groups, quality assurance, and the rules of configuration management; (3) links items relied on for safety requiring maintenance to the ISA; (4) justifies the preventive maintenance intervals in the terms of equipment reliability goals; (5) provides for training that emphasizes importance of ISA identified controls, regulations, codes, and personal safety; and (6) creates documentation that includes detailed records of all surveillances, inspections, equipment failures, repairs, and replacements.

The staff concludes that the applicant's maintenance function meets the requirements of 10 CFR Part 70 and provides reasonable assurance that the environment and the health and safety of the public are protected.

15.3.7 REFERENCES

A. Code of Federal Regulations, *Title 10, Energy*, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. 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.

C. Code of Federal Regulations, *Title 10, Energy*, Subpart 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

D. Code of Federal Regulations, *Title 29, Labor*, Subpart 1910.119, "Process Safety Management of Highly Hazardous Chemicals."

E. Code of Federal Regulations, *Title 40, Protection of Environment*, Part 68, "Risk Management Program for Chemical Accidental Release Prevention."

F. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Guidance on Management Controls/Quality Assurance, Requirements for Operation, Chemical Safety, and Fire Protection for Fuel Cycle Facilities." *Federal Register:* Vol. 54, No. 53. pp. 11590–11598. March 21, 1989.

G. Nuclear Regulatory Commission, (U.S.) (NRC). Regulatory Guide 1.160, Rev. 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." NRC: Washington, D.C. March 1997.

H. Nuclear Regulatory Commission, (U.S.) (NRC). Inspection Procedure 88025, "Maintenance and Surveillance Testing." NRC: Washington, D.C. May 23, 1984.

I. Nuclear Regulatory Commission, (U.S.) (NRC). Inspection Procedure 88062, "Maintenance and Inspection." NRC: Washington, D.C. January 1996.

15.0 MANAGEMENT MEASURES
15.4 TRAINING AND QUALIFICATION OF PLANT PERSONNEL

15.4.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that personnel who perform activities relied on for safety at the plant[1] will understand, recognize the importance of, and be qualified to perform these activities as required by the proposed 10 CFR Part 70 in a manner that adequately protects the public and worker health and safety and the environment.

15.4.2 RESPONSIBILITY FOR REVIEW

Primary: Training, Quality Assurance or Human Factors Engineer/Specialist

Secondary: Licensing Project Manager

Supporting: Fuel Cycle Facility Inspector

15.4.3 AREAS OF REVIEW

The regulation, 10 CFR Part 70, as proposed, requires that personnel who perform activities relied on for safety be trained, tested, and retested as necessary to ensure that they understand, recognize the importance of, and are qualified to perform these activities in a manner that adequately protects the public and worker health and safety and the environment. Personnel at the facility should have the knowledge and skills necessary to start-up, operate, maintain, modify, and decommission the facility in a safe manner. The applicant should address the training and qualification of plant personnel with the application for a construction approval and should submit updated information with the license application.

The training, testing, retesting, and qualification of these personnel as described in the application for construction approval should be reviewed. This should include the training, testing, retesting, and qualification of managers, supervisors, designers, technical staff, plant operators, technicians, maintenance personnel and other personnel whose level of knowledge is relied on for safety.

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[1]This SRP section provides guidance for the review of information on the training and qualification of plant personnel who perform activities relied on for safety. Section F2 of SRP Appendix F on quality assurance and Supplement 2S-4 of ASME NQA-1-1994 provide review guidance on the subject of training and qualification of other personnel (for example, construction personnel) who perform activities relied on for safety.

Management Measures

The following areas should be reviewed:

A. Organization and management of training;
B. Trainee selection;
C. Conduct of needs/job analysis and identification of tasks for training;
D. Development of learning objectives as the basis for training;
E. Organization of instruction using lesson plans and other training guides;
F. Evaluation of trainee mastery of learning objectives;
G. Conduct of on-the-job training;
H. Systematic evaluation of training effectiveness;
I. Personnel qualification; and
J. Applicant's provisions for continuing assurance.

15.4.4 ACCEPTANCE CRITERIA

15.4.4.1 Regulatory Requirements

The requirement for training and qualification is addressed in the following:

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.

Specific references are as follows:

A. In §70.4, "Definitions," the term management measures is defined. Training and qualification are included as a management measure.

B. In §70.62(d), the applicant is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

C. In §70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

D. In §70.65(a), the application is required to include a description of the management measures.

An additional requirement for training and qualification is addressed in the following:

U.S. Code of Federal Regulations, "Notices, Instructions and Reports to Workers: Inspection and Investigations," Part 19, Title 10, "Energy." The specific reference is to §19.12, "Instructions to Workers."

15.4.4.2 Regulatory Guidance

NRC guidance applicable to training and qualification of personnel that provide guidance for implementing and satisfying the regulatory requirements and acceptance criteria is:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Training Review Criteria and Procedures," NUREG-1220, Rev.1, January 1993.

15.4.4.3 Regulatory Acceptance Criteria

With the application for construction approval, the applicant should commit to meet or exceed the acceptance criteria in Section 15.4.4 and to update the training and qualification of plant personnel description to reflect any changes between the applications for construction approval and for license.

The NRC reviewers should find the applicant's submittal regarding training and qualification of plant personnel provides reasonable assurance that the regulatory acceptance criteria below are adequately addressed and satisfied.

In addition to the regulatory acceptance criteria given below, SRP Sections 9.1.5.4 and 9.1.5.6 provide criteria for training and qualification of plant personnel for radiation safety functions.

A. Organization and Management of Training

The organization and management of training of plant personnel should be acceptable if the start-up, operation, maintenance, and modification of the facility are organized, staffed, and managed to facilitate planning, directing, evaluating, and controlling a systematic training process that fulfills job-related training needs. Formal training should be provided for each position or activity for which the required performance is relied on for safety. The application should state what training will be conducted and which personnel will be provided this training. Training should include retraining of previously trained and qualified personnel based on specified criteria.

The following commitments should be in the application regarding organization and management of training:

i. Line management should be responsible for the content and effective conduct of the training.

ii. The job function, responsibility, authority, and accountability of personnel involved in managing, supervising, and implementing training should be clearly defined.

iii. Performance-based training should be used as the primary management tool for analyzing, designing, developing, conducting, and evaluating training.

iv. Training procedures should be documented and implemented to ensure that all phases of training are conducted reliably and consistently.

v. Training documents should be linked to the configuration management system to ensure that design changes and plant modifications are accounted for in the training.

vi. Exceptions from training may be granted to trainees and incumbents when justified, documented, and approved by management.

vii. Auditable training records should be maintained. Training records, both programmatic and individual, should support management information needs and provide required data on each individual's training, job performance, and fitness for intended duty. (Refer to SRP Section 15.8 and Appendix H for detailed guidance on records management.)

B. Trainee Selection

Selection of trainees who will perform activities relied on for safety should be acceptable if minimum requirements for selection of trainees are specified. Trainees should meet entry-level criteria defined for the position including minimum educational, technical, experience, and (if necessary) physical fitness requirements.

C. Conduct of Needs/Job Analysis and Identification of Tasks for Training

The conduct of needs/job analysis and identification of tasks for training should be acceptable if the tasks required for competent and safe job performance are identified, documented, and included in the training.

Operations personnel, training staff, and other subject matter experts, as appropriate, should have conducted or should conduct a needs/job analysis to develop a valid task list for specific jobs. The jobs treated in this manner should include, as a minimum, those responsible for managing, supervising, performing, and verifying the activities specified in the Integrated Safety Analysis Summary (ISA - see SRP Chapter 5) that prevent or mitigate accidents. Each task selected for training (initial or continuing) from the facility-specific task list should be matrixed to supporting procedures and training materials. The facility-specific list of tasks selected for training and the comparison to training materials should be reviewed on an established schedule and updated as necessitated by changes in procedures, facility systems/equipment, or job scope.

D. Development of Learning Objectives as the Basis for Training

The development of learning objectives as the basis for training should be acceptable if learning objectives that identify training content and define satisfactory trainee performance are derived from job performance requirements and the needs/job analysis. Learning

objectives should state the knowledge, skills, and abilities the trainee should demonstrate; the conditions under which required actions will take place; and the standards of performance the trainee should achieve upon completion of the training activity. Learning objectives should be sequenced based on their relationship to each other.

E. Organization of Instruction Using Lesson Plans and Other Training Guides

The organization of instruction using lesson plans and other training guides should be acceptable if the plans/guides are based on the required learning objectives derived from specific job performance requirements and the needs/job analysis. Plans/guides should be used for in-class training and on-the-job training and should include standards for evaluating proper trainee performance. Review and approval requirements should be established for all plans/guides and other training materials before their issue and use.

F. Evaluation of Trainee Mastery of Learning Objectives

The evaluation of trainee mastery of learning objectives should be acceptable if trainees are evaluated periodically during training to determine their progress toward mastery of job performance requirements and at the completion of training to determine their mastery of job performance requirements.

G. Conduct of On-the-Job Training

The conduct of on-the-job training should be acceptable if on-the-job training used for activities identified in the ISA Summary is fully described. On-the-job training should be conducted using well-organized and current performance-based training materials. On-the-job training should be conducted by designated personnel who are competent in the program standards and methods of conducting the training. Completion of on-the-job training should be by actual task performance. When the actual task cannot be performed by the trainee and is therefore "walked-down," the conditions of task performance, references, tools, and equipment should reflect the actual task to the extent possible.

H. Systematic Evaluation of Training Effectiveness

A systematic evaluation of training effectiveness and its relation to on-the-job performance should be acceptable if it ensures that the training program conveys all required skills and knowledge and is used to revise the training, where necessary, based on the performance of trained personnel in the job setting. A comprehensive evaluation of individual training programs should be conducted periodically by qualified individuals to identify program strengths and weaknesses. Feedback from trainee performance during training and from former trainees and their supervisors should be used to evaluate and refine the training. Change actions (for example, procedure changes, equipment changes, facility modifications) should be monitored and evaluated for their impact on the development or modification of initial and continuing training and should be incorporated in a timely manner. Change

actions should be accomplished through the configuration management system (see SRP Section 15.2). Improvements and changes to initial and continuing training should be systematically initiated, evaluated, tracked, and incorporated to correct training deficiencies and performance problems.

I. Personnel Qualification

The following commitments should be in the application regarding personnel qualification for managers, supervisors, designers, technical staff, plant operators, technicians, maintenance personnel, and other plant staff required to meet NRC regulations:

i. Managers should have a minimum of a B.S./B.A. or equivalent. Each manager should have either management experience or technical experience in facilities similar to the MOX facility.

ii. Supervisors should have at least the qualifications required of personnel being supervised with either one additional year experience supervising the technical area at a similar facility or should have completed the supervisor training.

iii. Technical staff identified in the ISA Summary whose activities are relied on for safety to satisfy the performance requirements identified in 10 CFR Part 70, as proposed, should have a B.S. in the appropriate technical field and three years experience.

iv. Facility operators, technicians, maintenance personnel, and other staff whose actions are required to comply with NRC regulations should have completed the applicant's training process or have equivalent experience or training.

v. Candidates for process operators should be required to meet minimum qualifications described in the application. Candidates for job functions other than process operators should also be required to meet minimum qualifications, but these minimum qualifications need not be described in the application.

J. Applicant's Provisions for Continuing Assurance

The applicant's provisions for continuing assurance of training and qualification of plant personnel should be acceptable if the applicant's submittal addresses periodic retesting of personnel as necessary to ensure that the personnel continue to understand, recognize the importance of, and are qualified to perform their activities that are relied on for safety.

15.4.5 REVIEW PROCEDURES

15.4.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application (construction or license) adequately addresses the specific items in Section 15.4.3, "Areas of Review."

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

Specifically, the safety assessment of the design basis should address Section 15.4.3 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. The primary reviewer should also verify that the applicant has committed to meeting or exceeding the acceptance criteria of Section 15.4.4.

B. License Application for Operations

Specifically, the license application should address Section 15.4.3 in full. The applicant is expected to have developed a program for the training and qualification of plant personnel prior to facility licensing for operations.

If the primary reviewer verifies that the training and qualification of plant personnel is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 15.4.5.2. 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.

15.4.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 15.4.5.1(A) (construction) or 15.4.5.1(B) (license), the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.4.4. 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 15.4.4.

Management Measures

Guidance specific to the application for construction approval and the license application is provided below.

A. Application for Construction Approval

The primary reviewer should verify that the applicant's commitments and goals as they relate to the training and qualification of plant personnel are adequate to meet or exceed the acceptance criteria in Section 15.4.3.

B. License Application for Operations

The primary reviewer should focus the review on any new or changed material covering the training and qualification of plant personnel which the applicant updated with the license application. The primary reviewer should also confirm that the material remains consistent with the material provided in the license application in support of other chapters of this SRP.

The primary reviewer should recognize that the rigor and formality of a systematic approach to training and the required qualification of plant personnel may be graded to correspond to the hazard potential of the facility and to the complexity of the training needed. The primary reviewer should determine whether the applicant has adequately planned for the training and qualification of plant personnel to be accomplished and whether necessary policies, procedures, and instructions will be in place and appropriate training and qualification will be accomplished before these personnel begin activities relied on for safety. Some of the information may be referenced to other sections of the application, or incorporated by reference, provided that these references are clear and specific.

The secondary reviewer should confirm that the applicant's commitments regarding the training and qualification of plant personnel are consistent with other sections of the applicant's submittal.

The supporting reviewer (Fuel Cycle Facility Inspector) should become familiar with the applicant's commitments for the training and qualification of plant personnel and determine whether ongoing activities are in agreement with them.

The review should result in a determination that there is reasonable assurance that the applicant's training and qualification of plant personnel will ensure that only properly trained and qualified personnel will perform activities relied on for safety.

15.4.6 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 a safety evaluation for the application for construction approval as follows:

> *[Here the primary reviewer provides a summary statement of what was evaluated (including a applicant commitments) and why the reviewer finds the applicant's submittal acceptable.] Continued with: Based on its review of the license application, the NRC staff concludes that the applicant adequately described its training and qualification of plant personnel (or made commitments to meet the acceptance criteria of Section 15.4.4 of NUREG-1718) and that the applicant's training and qualification of plant personnel (will, based on commitments) meet the requirements of 10 CFR Part 70 and provide reasonable assurance of protection of public health and safety and of the environment.*

The staff could document a safety evaluation for the license application using a similar paragraph as that used for the construction approval, but encompassing any new or updated material (and possible fulfilled commitments) when compared to the safety evaluation for the construction approval.

15.4.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. 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.

C. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Training Review Criteria and Procedures," NUREG-1220, Rev. 1, January 1993.

D. American Society of Mechanical Engineers (ASME), "Quality Assurance Requirements for Nuclear Facility Applications," (An American National Standard). ASME NQA-1-1994, New York. 1994.

E. U.S. Code of Federal Regulations, "Notices, Instructions and Reports to Workers: Inspection and Investigations," Part 19, Title 10, "Energy."

15.0 MANAGEMENT MEASURES
15.5 PLANT PROCEDURES

15.5.1 PURPOSE OF REVIEW

The purpose of this review is to establish that there is reasonable assurance that the applicant is capable and committed to providing management control of facility operations identified as items relied on for safety through the development, review, approval, control, and implementation of written plant procedures[1] that will protect the workers, the public, and the environment during testing, startup, and operation of the facility.

15.5.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: Primary staff reviewers of other Management Measures
Human Factors Engineer

Supporting: Fuel Cycle Facility Inspector

15.5.3 AREAS OF REVIEW

The staff's review of the license application should address the process the applicant has developed for the production, use, and management control of written plant procedures. This should include the basic elements of identification, development, verification, initial review, comment resolution, approval, validation, issuance, change control, and periodic review. There should be two general types of plant procedures:

A. Plant procedures used to directly control process operations, commonly called "operating procedures." These are procedures for workstation operators, and they should include directions for normal operations as well as off-normal incidents caused by human error or failure of equipment. Procedures of this type should include required actions to ensure nuclear criticality safety, chemical safety, fire protection, emergency planning, and environmental protection.

B. Plant procedures used to perform activities that support the process operations, commonly referred to as "management control procedures." These are procedures used to manage the conduct of activities such as configuration management, radiation safety, maintenance,

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[1]This SRP section provides guidance for the review of information on plant procedures identified as items relied on for safety. Section F5 of SRP Appendix F on quality assurance and Basic Requirement 5 of ASME NQA-1-1994 provide review guidance for other procedures (for example, construction procedures) relied on for safety.

human-systems interface, quality assurance, design control, test control, startup, plant personnel training and qualification, audits and assessments, incident investigations, record-keeping, and reporting.

15.5.4 ACCEPTANCE CRITERIA

15.5.4.1 Regulatory Requirements

The requirement for plant procedures is addressed in the following:

Nuclear Regulatory Commission (U.S.), Washington, D.C. "Domestic Licensing of Special Nuclear Material (10 CFR Part 70)." *Federal Register:* Vol. 64, No. 145. pp. 41338-41357. July 30, 1999.

Specific references are as follows:

A. In §70.4, "Definitions," the term *management measures* is defined. Procedures are included as a management measure.

B. In §70.22(a)(8), the application is required to include proposed procedures to protect health and minimize danger to life or property.

C. In §70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

D. In §70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

E. In §70.65(a), the application is required to include a description of the management measures.

15.5.4.2 Regulatory Guidance

Appendix A of Reference 2 contains a list of typical procedures for the operation of nuclear power plants. Similar procedures should be developed and implemented for this facility.

15.5.4.3 Regulatory Acceptance Criteria

As part of the application for construction approval, the applicant should commit to establish a process for the production, use, and management control of written plant procedures which meets or exceeds the acceptance criteria in Section 15.5.4.

The reviewers should determine that the applicant's process for developing and implementing plant procedures is acceptable (for license approval) if the process satisfies the following:

A. Plant procedures should be written or planned for the conduct of all operations involving controls identified in the ISA as activities relied on for safety and for all management control systems supporting those controls.

B. Operating procedures contain the following elements:

i. Purpose of the activity;
ii. Regulations, polices, and guidelines governing the procedure;
iii. Type of procedure;
iv. Steps for each operating process phase;
v. Initial startup;
vi. Normal operations;
vii. Temporary operations;
viii. Emergency shutdown;
ix. Emergency operations;
x. Normal shutdown;
xi. Startup following an emergency or extended downtime;
xii. Hazards and safety considerations;
xiii. Operating limits;
xiv. Precautions necessary to prevent exposure of hazardous chemicals or licensed special nuclear material;
xv. Measures to be taken if contact or exposure occurs;
xvi. Safety controls associated with the process and their functions; and
xvii. Time period for which the procedure is valid.

C. Management control procedures contain elements reflecting the important elements of the functions described in the applicable chapters of this SRP. Management control procedures should exist for the following activities:

i. Configuration management;
ii. Radiation safety;
iii. Maintenance;
iv. Human-systems interface;
v. Quality assurance;
vi. Training and qualification;
vii. Audits and assessments;
viii. Incident investigations;
ix. Records management;
x. Nuclear criticality safety;
xi. Fire safety;
xii. Chemical process safety;

xiii. Design control;
xiv. Test control;
xv. Startup; and
xvi. Reporting requirements.

D. The applicant's method for identifying plant procedures includes using ISA results to identify needed procedures. Operating procedures should provide specific direction regarding administrative controls to ensure operational safety.

E. The applicant's method for identifying, developing, approving, implementing, and controlling plant procedures should include, as a minimum:

i. Operating limits and controls are specified in the procedure;

ii. Procedures include required actions for off-normal conditions of operation as well as normal operations;

iii. If needed, safety checkpoints are identified at appropriate steps in the procedure;

iv. Procedures are validated through field tests;

v. Procedures are approved by management personnel responsible and accountable for the operation;

vi. A mechanism is specified for revising and reissuing procedures in a controlled manner;

vii. The quality assurance and configuration management programs at the facility ensure that current procedures are available and used at all work locations; and

viii. The facility training program ensures that the required persons are trained in the use of the latest procedures.

F. The application should include the following statement regarding adherence to plant procedures: "Activities involving special licensed nuclear material will be conducted in accordance with approved procedures."

G. The applicant should discuss plant procedure categories used at the facility. An acceptable discussion should clearly state areas for which a plant procedure is required. The applicant should provide a list of the types of activities that are covered by the plant procedures. This should include the topics of administrative plant procedures; system plant procedures that address startup, operation, and shutdown; abnormal operation/alarm response; maintenance activities that address system repair, calibration, inspection and

testing; and emergency procedures. Appendix G to this SRP provides an acceptable list of the items to be included under each topic.

H. The applicant should indicate that following an incident–such as an accident, unexpected transient, significant operator error, or equipment malfunction–or following any modification to a system, a review of written plant procedures will take place.

I. The applicant should indicate how technical accuracy of plant procedures will be ensured as written. The discussion should identify who is responsible for verification.

J. The applicant should indicate how documents will be distributed in accordance with current distribution lists. A process limiting the use of outdated plant procedures should be addressed.

K. The applicant should describe how formal requirements governing temporary changes to plant procedures will be developed and implemented.

L. Formal requirements for design control of items relied on for safety should be provided and should identify who is responsible for design inputs, processes, outputs, changes, interfaces, and records.

M. A description of the test control program should be provided and should indicate that an effective program has been established for tests, including commissioning and preoperational tests. Acceptable plant procedures for test control should provide criteria for determining when a test is required or how and when testing activities are performed.

   i. Tests should be performed under conditions that simulate the most adverse design conditions, as determined by analysis.

   ii. Test results should be documented, evaluated, and their acceptability determined by a responsible individual or group.

N. Plant procedures for maintenance involving safety controls should commit to the topics listed below for corrective and preventive maintenance, functional testing after maintenance, and surveillance/monitoring of maintenance activities:

   i. Pre-maintenance activity involving reviews of the work to be performed, including reviews of facility procedures for maintenance for accuracy and completeness.

   ii. Steps that require notification of all affected parties (operators and supervisors) prior to performing work and upon completion of maintenance work.

   iii. Control of work by comprehensive facility procedures to be followed by maintenance technicians.

O. The applicant should commit to conduct periodic reviews of plant procedures to ensure their continued accuracy and usefulness. The applicant should establish the time frame for these reviews. At a minimum all procedures should be reviewed every 5 years and emergency procedures should be reviewed every year.

P. The applicant should describe the use and control of procedures.

Q. A pre-operational testing (startup) program should be described. Information pertaining to how, and to what extent, the facility operating, emergency, and surveillance procedures will be user-tested during this test program should be provided.

15.5.5 REVIEW PROCEDURES

15.5.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application a license adequately addresses the specific items in Section 15.5.3, "Areas of Review." If the primary reviewer verifies that procedures are adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 15.5.5.2. 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.

15.5.5.2 Safety Evaluation

For construction approval, the reviewer should determine that the applicant has committed to a maintenance program that will meet or exceed the acceptance criteria in Section 15.5.4.

After determining that the application for license approval is acceptable for review in accordance with Section 15.5.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.5.4. 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 15.5.4.

The safety evaluation forms the basis for staff findings and supports the reviewers' conclusions that the applicant has committed to:

A. Controls that are identified in the ISA for facility safety procedures (i.e., procedures that constitute administrative controls for safety).

B. The independent verification and validation of procedures before use.

C. The review and approval by an independent multi-disciplinary safety review team and control by the configuration management function of any change to facility procedures.

D. Follow approved procedures while processing licensed special nuclear material.

E. Having procedures for the notification of operations personnel before and after maintenance is performed on safety controls.

Secondary staff reviewers should ensure that the applicant's facility procedures do not conflict with their primary review areas.

The supporting staff reviewer (Fuel Cycle Facility Inspector) should become familiar with the applicant's written plant procedures and determine whether ongoing activities are in agreement with them.

15.5.6 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 construction approval by stating that the applicant has committed to establish a process for the production, use, and management control of written plant procedures with meets or exceeds the acceptance criteria in Section 15.5.4.

The staff could document a the safety evaluation for the license application as follows:

> *The applicant has described suitably detailed processes for the development, review, approval, control, and implementation of procedures. [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Special attention has been paid to items relied on for safety, as well as to systems important to the health of workers and the public and to the protection of the environment during testing, startup, and operation of the facility.*

15.5.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

C. U.S. Nuclear Regulatory Commission, (U.S.), Washington, D.C. "Guidance on Management Controls/Quality Assurance, Requirements for Operation, Chemical Safety, and Fire Protection for Fuel Cycle Facilities." *Federal Register:* Vol. 54, No. 53. pp. 11590–11598. March 21, 1989.

D. Nuclear Regulatory Commission, (U.S.) (NRC). Regulatory Guide 1.33, Rev. 2, "Quality Assurance Program Requirements (Operation)." NRC: Washington, D.C. February 1978.

E. American Society of Mechanical Engineers (ASME), "Quality Assurance Requirements for Nuclear Facility Applications," (An American National Standard). ASME NQA-1-1994, New York. 1994.

15.0 MANAGEMENT MEASURES
15.6 AUDITS AND ASSESSMENTS

15.6.1 PURPOSE OF REVIEW

The purpose of this review is to establish that the applicant developed and adequately described a system of audits and assessments[1] that provides reasonable assurance that an adequate level of protection will be maintained at the facility and to ensure that items relied on for safety will be available and reliable to perform their safety function when needed, as required by the proposed 10 CFR Part 70.

15.6.2 RESPONSIBILITY FOR REVIEW

Primary: Quality Assurance (QA) Engineer/Specialist

Secondary: Project Manager

Supporting: Fuel Cycle Facility Inspector

15.6.3 AREAS OF REVIEW

The applicant should submit a description of the system of audits and assessments with the application for construction approval and should submit updated information with the license application for operations. The applicant's system of audits and assessments should consist of two distinct levels of activities:

A. An independent internal and external audit activity to evaluate the scope, status, adequacy, programmatic compliance and implementation effectiveness of QA and other management measures that ensure continued availability and reliability of items relied on for safety.

B. An internal assessment activity to evaluate the scope, status, adequacy, programmatic compliance, and implementation effectiveness of QA and other management measures that ensure continued availability and reliability of items relied on for safety.

The following areas should be reviewed (construction approval):

A. Audits and assessments-general;
B. Audits;

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[1]Audits and assessments are evaluations of the scope, status, adequacy, programmatic compliance, and implementation effectiveness of QA and other management measures. Audits are conducted or led by "independent" personnel from the QA organization. Assessments are conducted by or for management above or outside the QA organization.

C. Assessments; and
D. Applicant's provisions for continuing assurance.

15.6.4 ACCEPTANCE CRITERIA

15.6.4.1 Regulatory Requirements

The requirement for audits and assessments is addressed in the following:

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.

Specific references are as follows:

A. In §70.4, "Definitions," the term management measures is defined. Audits and assessments are included as a management measure.

B. In §70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

C. In §70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

D. In §70.65(a), each application is required to include a description of the management measures.

15.6.4.2 Regulatory Guidance

There is no regulatory guidance applicable to this area of the SRP.

15.6.4.3 Regulatory Acceptance Criteria

The NRC reviewers should find that the applicant's audits and assessments provide reasonable assurance that the regulatory acceptance criteria below are adequately addressed and satisfied.

A. Audits and Assessments

General: Audits and assessments should be acceptable if:

i. Internal audits, external audits, and assessments are to be conducted with a graded approach based on the results of the integrated safety analysis (ISA - see SRP

Chapter 5). The stated objective of the audits and assessments should be to objectively evaluate the effectiveness and proper implementation of QA and other management measures for items relied on for safety and to address the technical adequacy of the items being audited/assessed.

ii. The applicant describes, provides a commitment to, and provides justification for a frequency and scope of audits and assessments of items relied on for safety. A commitment to perform audits and assessments in all areas where the requirements for QA and other management measures are applicable should be provided. Audits and assessments will be regularly scheduled on the basis of the status and the safety significance of the items being audited/assessed and will be initiated early enough to ensure the implementation of effective QA and other management measures.

iii. Policy directives are established for audits and assessments. Policy directives cover schedules, guidance for conducting the audits and assessments, assigned responsibilities, and procedures for recording the audit/assessment results and ensuring that identified deficiencies are corrected in a timely and effective manner for each activity audited/assessed.

iv. The applicant identifies the position title, qualifications, and responsibilities of the manager responsible for the overall success of the audits and assessments. Other organizational responsibilities for audits and assessments should be identified in the application.

v. The applicant describes the training and qualification requirements for audit and assessment personnel. (SRP Section 15.4 addresses training and qualification requirements in detail.)

vi. The applicant describes the authority each audit and assessment team has to investigate any aspect of the audited/assessed items with access to all relevant information.

vii. Performance indicators are established so that audit and assessment personnel can determine the degree to which items relied on for safety are meeting performance requirements.

viii. Audits and assessments are conducted according to written procedures/checklists. (SRP Section 15.5 provides procedure guidance.)

ix. Audits and assessments include detailed walk-downs of plant areas, including out-of-the-way and limited-access areas, with provisions for accurate, documented descriptions of any deficiencies.

x. The applicant describes provisions for on-the-spot corrective actions with appropriate documentation.

xi. Audit and assessment results are reviewed with management having responsibility in the area audited/assessed.

xii. Audit and assessment findings and recommendations are documented and distributed to appropriate management for review and response. As described in SRP Section 15.1, a corrective action program is administered to ensure timely and effective corrective action.

xiii. Audit and assessment deficiency data are analyzed and trended and resultant reports, which indicate quality trends and the effectiveness of management measures, are given to appropriate management for review, response, corrective action, and follow-up.

B. Audits

Audits should be acceptable if, in addition to addressing the acceptance criteria in Section 15.6.4.3.1 above,

i. Audit personnel have no direct responsibility for the items they audit.

ii. Audits are led by appropriately qualified and certified audit personnel from the QA organization.

iii. Audit team membership may include personnel (not necessarily from the QA organization) having technical expertise in the areas being audited.

iv. Technical and programmatic audits are performed internally (that is, within the applicant's organization) and externally (that is, within the organization of suppliers, contractors, and subcontractors) and these audits provide a comprehensive independent verification and evaluation of procedures and activities affecting the quality of items relied on for safety.

v. Auditing organizations schedule and conduct appropriate follow-up to ensure timely and effective corrective action.

vi. Audit reports are issued to appropriate management on a timely basis.

vii. Reports on the status of corrective actions for audit findings are issued periodically to appropriate management.

viii. Internal audits address compliance with selected operating limits during facility operation.

C. Assessments

Assessments should be acceptable if, in addition to addressing the acceptance criteria in Section 15.6.4.3.1 above, the application indicates that responsible management personnel (or that qualified, but not necessarily certified, personnel with no direct responsibility for the items being assessed who are designated by the responsible management) perform the assessments.

D. Applicant's Provisions for Continuing Assurance

The applicant's provisions for continuing audits and assessments should be acceptable if changes to the program of audits and assessments due to reorganizations, revised activities, lessons learned, changes to applicable regulations, and other changes are reviewed and reflected in the program description.

Also, the applicant should commit to update the system of audits and assessments to reflect any changes between the application for construction approval and the license application for operations.

15.6.5 REVIEW PROCEDURES

15.6.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the specific items in Section 15.6.3, "Areas of Review." If the primary reviewer verifies that audits and assessments are adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 15.6.5.2. 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.

15.6.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 16.6.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.6.4. 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 15.6.4.

The primary reviewer should determine whether the applicant has adequately planned for audits and assessments to be accomplished and whether necessary policies, personnel, procedures, and instructions will be in place to begin audits and assessments early, that is, during the ISA process and the design of items relied on for safety.

Management Measures

The secondary reviewer should confirm that the applicant's audit and assessment commitments are consistent with other sections of the submittal.

The supporting reviewer should become familiar with the applicant's audit and assessment commitments and determine whether the ongoing audits and assessments of the applicant and the applicant's suppliers, contractors, and subcontractors are in agreement with them.

The review should result in a determination that there is reasonable assurance that the audits and assessments will provide additional assurance that items relied on for safety will perform satisfactorily in service and that activities relied on for safety will be performed satisfactorily.

When the applicant updates the system of audits and assessments, the primary reviewer should focus the review on any new or changed material. The primary reviewer should also confirm that the material presented remains consistent with the material provided in the license application for operations in support of other chapters of this SRP.

15.6.6 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 a safety evaluation for the application for construction approval as follows:

> *[Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Based on its review of the application for construction approval, the NRC staff concludes that the applicant has adequately described its system of audits and assessments, and the applicant's system of audits and assessments meet the requirements of 10 CFR Part 70 and provides reasonable assurance of protection of public health and safety and of the environment.*

The staff could document a safety evaluation for the license application for operations as follows:

> *[Here the primary reviewer provides a summary statement of what was evaluated and why the reviewer finds the submittal acceptable.] Based on its review of the license application for operations, focusing on new or updated material when compared to the safety evaluation for construction approval, the NRC staff concludes that the applicant has adequately described its updated system of audits and assessments, and the applicant's updated system of audits and assessments meet the requirements of 10 CFR Part 70 and thus provides reasonable assurance of protection of public health and safety and of the environment.*

15.6.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

15.0 MANAGEMENT MEASURES
15.7 INCIDENT INVESTIGATIONS

15.7.1 PURPOSE OF REVIEW

The purpose of this review is to establish, with reasonable assurance, that the applicant will have a system in place for the systematic investigation of incidents,[1] assignment and acceptance of corrective actions, and follow-up to ensure completion of the actions. The review should confirm that incidents will be investigated and corrective action taken to prevent (or minimize) their recurrence or their leading to more serious consequences. Furthermore, the review should find that the results of incident investigations will be compared against the Integrated Safety Analysis (ISA) Summary (see SRP Chapter 5.0) to provide assurance that there is continued compliance with the performance requirements contained in 10 CFR Part 70, as proposed.

15.7.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: Quality Assurance Engineer/Specialist and ISA Reviewers

Supporting: Fuel Cycle Facility Inspector

15.7.3 AREAS OF REVIEW

The staff's review of the license application should encompass the following areas:

A. The description of the functions, qualifications, and responsibilities of the management person who would lead the investigation team and those of the other team members, the scope of the team's authority and responsibilities, and assurance of cooperation of management.

B. The team's ability to obtain all the information considered necessary and independence from responsibility for or to the functional area involved in the incident under investigation.

C. The maintenance of documentation consistent with SRP Section 15.8, "Records Management."

D. Guidance for the team conducting the investigation on how to apply a reasonable, systematic, structured approach to determine the root cause(s) of the problem.

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[1]Incidents are unplanned events such as accidents, unexpected transients, equipment malfunctions, and operator error.

E. The system for comparing the results of the investigation against the ISA.

F. The system for monitoring to ensure completion of any corrective measures specified, including revisions to the ISA.

15.7.4 ACCEPTANCE CRITERIA

15.7.4.1 Regulatory Requirements

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.

Specific references are as follows:

A. In §70.4, "Definitions," the term management measures is defined. Incident investigations are included as a management measure.

B. In §70.62(d), the applicant or licensee is required to establish management measures to provide continuing assurance of compliance with the performance requirements.

C. In §70.64(a)(1), the design of new facilities or the design of new processes at existing facilities is required to be developed and implemented in accordance with management measures.

D. In §70.65(a), the application is required to include a description of the management measures.

15.7.4.2 Regulatory Guidance

There is no specific regulatory guidance for the overall conduct of incident investigations. See the references at the end of this section for guidance on specific aspects of incident management such as corrective action and root cause analysis.

15.7.4.3 Regulatory Acceptance Criteria

As part of the application for construction approval, the applicant should commit to establishing a system for the systematic investigation of incidents, assignment and acceptance of corrective actions, and follow-up to ensure completion of the actions which meets or exceeds the acceptance criteria in Section 15.7.4.

The NRC reviewers should find the license application for operations acceptable if the applicant's system of incident investigations 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 that these references are clear and specific.

A. Acceptability should be based on commitments for the prompt investigation of incidents that include the following elements:

   i. The establishment of teams to investigate incidents that may occur during operation of the facility, to determine the root cause(s) of the incident, and to recommend corrective actions.

   ii. The monitoring and documenting of corrective actions (including effectiveness) through completion.

   iii. The maintenance of documentation so that "lessons learned" may be applied to future operations of the facility. Details of the incident sequence should be compared to incident sequences already considered in the ISA, and actions should be taken to ensure that the ISA includes the evaluation of the risk associated with incidents of the type actually experienced.

B. Acceptability should be based on the adequacy of the applicant's commitments to establish and use a plan for the investigation of incidents. Acceptability should also be based upon the following acceptance criteria:

   i. The licensee has described the overall plan and method for investigating incidents. The plan is separate from any required emergency plan.

   ii. The functions, responsibilities, and scope of authority of investigation teams are documented in the plan.

   iii. Qualified internal or external investigators are appointed to serve on investigation teams. Each team should include at least one process expert and one team member trained in root cause analysis.

   iv. The investigation process and investigation team are independent of the line function(s) involved with the incident under investigation, and participants are assured of no retribution from participating in investigations.

   v. A reasonable, systematic, structured approach is used to determine the root cause(s) of incidents. The level of investigation should be based on a graded approach relative to the severity of the incident.

vi. Auditable records and documentation related to incidents, investigations, and root cause analysis are maintained.

vii. For each incident, an incident report is prepared that includes a description of the incident, contributing factors, root cause analysis, findings, and recommendations. Relevant findings should be reviewed with all affected personnel, and the reports should be made available to the NRC on request.

viii. Documented corrective actions are taken within a reasonable period to resolve findings from incident investigations.

15.7.5 REVIEW PROCEDURES

15.7.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the license application for operations adequately addresses the specific items in Section 15.7.3, "Areas of Review." If the primary reviewer verifies that incident investigations are adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 15.7.5.2. 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.

15.7.5.2 Safety Evaluation

For construction approval, the reviewer should determine that the applicant has committed to a system for the systematic investigation of incidents, assignment and acceptance of corrective actions, and follow-up to ensure completion of the actions that will meet or exceed the acceptance criteria in Section 15.7.4.

For a license application for operations and after determining that the application is acceptable for review in accordance with Section 15.7.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.7.4. 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 15.7.4.

The review should determine if the applicant has adequately planned for incident investigations to be conducted with resulting corrective actions to be appropriately implemented.

The primary reviewer should confirm that the organizational structure for incident investigations is consistent with SRP Chapter 4, "Organization and Administration."

The quality assurance secondary reviewer should verify that methods used for determining root causes, the procedures for tracking and implementing the corrective actions, and the process of applying the "lessons learned" to the other operations are appropriate for incident investigations.

The ISA reviewers should verify that the applicant ensures the results of the investigation are compared against the ISA and the necessary follow-up actions occur.

The supporting reviewer(s) should become familiar with pertinent procedures and determine whether planned future and ongoing activities are consistent with them.

15.7.6 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 construction approval by stating that the applicant has committed to establishing a system for the systematic investigation of incidents, assignment and acceptance of corrective actions, and follow-up to ensure completion of the actions that meets or exceeds the acceptance criteria in Section 15.7.4.

The staff could document a safety evaluation for the license application for operations as follows:

> *Based on its review of the license application for operations, [Insert a summary statement of what was evaluated and why the reviewer finds the submittal acceptable,] the NRC staff concluded that the applicant has committed to and established an organization responsible for investigating incidents that occur during operation of the facility, determining the root cause(s) of each incident and taking corrective actions for ensuring a safe facility and safe facility operations in accordance with the acceptance criteria of Section 15.7.4 of the SRP; committed to review the results of the investigation against the ISA; committed to monitoring and documenting corrective actions through completion; and committed to the maintenance of related documentation and apply "lessons learned" to future operations of the facility.*
>
> *Accordingly, the staff concludes that the applicant's description of the incident investigation process complies with applicable NRC regulations and is adequate.*

15.7.7 REFERENCES

A. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

B. Proposed 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 64 FRN 41338, July 30, 1999.

C. Department of Energy (U.S.) (DOE). DOE-STD-1010-92, "Guide to Good Practices for Incorporating Operating Experiences." DOE: Washington, D.C. July 1992.

D. Department of Energy (U.S.) (DOE). DOE-NE-STD-1004-92, "Root Cause Analysis Guidance Document." DOE: Washington, D.C. February 1992.

E. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-4616, "Root Causes of Component Failures Program: Methods and Applications." NRC: Washington, D.C. December 1986.

F. Nuclear Regulatory Commission (U.S.) (NRC). NUREG/CR-5665, "A Systematic Approach to Repetitive Failures." NRC: Washington, D.C. February 1991.

G. Nuclear Regulatory Commission (U.S.) (NRC), NRC Information Notice 96-28, "Suggested Guidance Relating to Development and Implementation of Corrective Action." NRC: Washington, D.C. May 1996.

15.0 MANAGEMENT MEASURES
15.8 RECORDS MANAGEMENT

15.8.1 PURPOSE OF REVIEW

The purpose of this review is to verify that the applicant has established a facility records management system that complies with NRC requirements.

15.8.2 RESPONSIBILITY FOR REVIEW

Primary: Project Manager

Secondary: None

Supporting: Primary reviewers of SRP Sections 15.1, "Quality Assurance," and 15.2, "Configuration Management"

15.8.3 AREAS OF REVIEW

The applicant should submit a description of the facility records management system with the application for construction approval and should submit updated information with the license application for operations.

Areas related to the handling and storing of records generated or needed in the design, construction, and operation phases of the facility, including the following, should be reviewed with the application for construction approval.

A. The process whereby records–such as training records, dosimetry records, effluent records, and records regarding the facility structures, systems, or components that are items relied on for safety–are specified, created, verified, categorized, indexed, inventoried, protected, stored, maintained, distributed, and deleted or preserved. The process may be linked with or be a part of the facility quality assurance and configuration management systems.

B. The handling and control of various kinds of records and the methods of recording media that comprise the records including contaminated and classified records.

C. The physical characteristics of the record storage facilities with respect to the preservation and protection of the records for their designated lifetimes.

15.8.4 ACCEPTANCE CRITERIA

15.8.4.1 Regulatory Requirements

The requirements for records management are addressed in the following:

A. Code of Federal Regulations, *Title 10, Energy*, Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

B. Code of Federal Regulations, *Title 10, Energy*, Part 20, "Standards for Protection Against Radiation."

C. Code of Federal Regulations, *Title 10, Energy*, Part 21, "Reporting of Defects and Noncompliance."

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

E. 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.

15.8.4.2 Regulatory Guidance[1]

Regulatory guidance applicable to the area of records management is as follows:

U.S. Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1460, Rev. 1, "Guide to NRC Reporting and Recordkeeping Requirements." NRC: Washington, D.C. July 1994.

15.8.4.3 Regulatory Acceptance Criteria

The reviewer should find the applicant's records management system acceptable if it satisfies the following criteria:

A. Records are specified, prepared, verified, characterized, and maintained.

B. Records are legible, identifiable, and retrievable for their designated lifetimes.

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[1] Additional guidance for records is given in SRP Appendix F on quality assurance (Section F17) and in ASME NQA-1-1994 (Basic Requirement 17 and Supplement 17S-1) as referenced in SRP Section 15.1, "Quality Assurance."

C. Records are protected against tampering, theft, loss, unauthorized access, damage, or deterioration for the time they are in storage.

D. Procedures are established and documented specifying the requirements and responsibilities for record selection, verification, protection, transmittal, distribution, retention, maintenance, and disposition.

E. The organization and procedures are in place to promptly detect and correct any deficiencies in the records management system or its implementation.

Examples of the types of records that could be included in the system and that contribute to providing reasonable assurance of protection of worker and public health and safety and of the environment are listed in Appendix E to this SRP. Records should be categorized by relative safety importance to identify record protection and storage needs and to designate the retention period for individual kinds of records. The procedures should assign responsibilities for records management; specify the authority needed for records retention or disposal; specify which records must have controlled access and provide the controls needed; provide for the protection of records from loss, damage, tampering, or theft during an emergency; and specify procedures for ensuring that the records management system remains effective.

For records consisting of computer codes/computerized data relied on for safety, the application should establish and describe procedure(s) for maintaining readability and usability of older codes/data as computing technology changes.

Also, the applicant should commit to update the facility records management system to reflect any changes between the application for construction approval and a license application for operations.

15.8.5 REVIEW PROCEDURES

15.8.5.1 Acceptance Review

The primary reviewer should perform an acceptance review to determine if the application adequately addresses the specific items in Section 15.8.3, "Areas of Review." If the primary reviewer verifies that the records management system is adequately addressed, the primary reviewer should accept the application for the safety evaluation in Section 15.8.5.2. 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.

15.8.5.2 Safety Evaluation

After determining that the application is acceptable for review in accordance with Section 15.8.5.1, the primary reviewer should perform a safety evaluation against the acceptance criteria described in Section 15.8.4. 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 15.8.4. The primary reviewer should coordinate this review with the primary reviewers of SRP Sections 15.1, "Quality Assurance," and 15.2, "Configuration Management."

When the applicant updates the facility records management system for the license application for operations, the primary reviewer should focus the review on any new or changed material. The primary reviewer should also confirm that the material presented remains consistent with the material provided in the license application for operations in support of other chapters of this SRP.

15.8.6 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 a safety evaluation for construction approval as follows:

> *The staff reviewed the applicant's records management system [Insert a summary statement of what was evaluated] and concluded that there is reasonable assurance that the system will (1) be effective in collecting, verifying, protecting, and storing information about the health and safety aspects of the facility and its operations and will be able to retrieve the information in readable form for the designated lifetimes of the records; (2) provide record storage facilities with the capability to protect and preserve records that are stored there during the mandated periods, including protection of the stored records against loss, theft, tampering, or damage during and after emergencies; and (3) ensure that any deficiencies in the records management system or its implementation will be detected and corrected in a timely manner. The staff concludes that the applicant's facility records management system meets the requirements of 10 CFR Part 70 and is acceptable.*

The staff could document a safety evaluation for the license application for operations using a paragraph similar to the one use for the construction approval, but encompassing the new or changed material when compared to the safety evaluation for the construction approval.

15.8.7 REFERENCES

A. Code of Federal Regulations, *Title 10, Energy*, Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

B. Code of Federal Regulations, *Title 10, Energy*, Part 20, "Standards for Protection Against Radiation."

C. Code of Federal Regulations, *Title 10, Energy*, Part 21, "Reporting of Defects and Noncompliance."

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

E. Code of Federal Regulations, Title 10, Part 70, Domestic Licensing of Special Nuclear Material, U.S. Government Printing Office, Washington, D.C., 1999.

F. 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. 41338-31357. July 30, 1999.

G. American Society of Mechanical Engineers (ASME), "Quality Assurance Requirements for Nuclear Facility Applications," (An American National Standard). ASME NQA-1-1994, New York. 1994.

H. U.S. Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1460, Rev. 1, "Guide to NRC Reporting and Recordkeeping Requirements." NRC: Washington, D.C. July 1994.

APPENDIX A
EXAMPLE PROCEDURE FOR RISK EVALUATION

NRC requirements in proposed 10 CFR 70.61 require that the occurrence of consequences of concern, defined in proposed §70.61 be sufficiently unlikely. In addition, proposed 10 CFR 70.62(c) requires that the applicant perform an Integrated Safety Analysis (ISA) to identify all potential accident sequences and to assess their consequences. These two requirements are related. The consequences of concern result from accident sequences identified in the ISA. Thus, to show that the likelihood of occurrence of the consequences is sufficiently low, the applicant must show that for each of the accident sequences identified in the ISA, the resulting consequences are sufficiently unlikely.

As defined in proposed 10 CFR 70.61, the required likelihood is graded according to the severity of the consequences of the accident. Accidents in the intermediate consequence category of proposed §70.61(c) must be "unlikely," while those in the high consequence category of proposed §70.61(b) must be "highly unlikely." The procedure described in this appendix is one way by which the applicant may use the ISA results to demonstrate that the requirements of proposed 10 CFR 70.61 have been met. If the applicant evaluates accidents using a different method, the method should produce similar results in terms of how accidents are categorized. This method should be regarded as a screening method, not as a definitive method of proving the adequacy or inadequacy of the controls for any particular accident. The method requires the applicant to identify and evaluate the characteristics of controls used to limit accident sequences in a consistent manner. This will permit identification of accident sequences with defects in the combination of controls used. Such controls can then be further evaluated or improved to establish adequacy. The procedure also ensures the consistent evaluation of similar controls by different ISA teams. Sequences or controls that have risk significance, and are evaluated as marginally acceptable, are good candidates for more detailed evaluation by the applicant and the reviewer.

The tabular accident summary resulting from the ISA should identify, for each sequence, what safety controls must fail for consequences of concern in proposed 10 CFR 70.61 to occur. Chapter 5.0 specifies acceptance criteria for these safety controls, such that the performance requirements of proposed §70.61 are met. These criteria require that safety controls be sufficiently unlikely to fail. However, the criteria of Chapter 5.0 do not provide for a method for assessing likelihood. This appendix describes an acceptable procedure for this required assessment of likelihood.

A1. DETERMINING COMPLIANCE WITH GRADED PROTECTION REQUIREMENTS

Proposed 10 CFR Part 70.61 describes requirements for a graded system of protection sufficient to bound the risk of identified accidents by making accidents of higher potential consequences have a proportionately lower likelihood of occurrence. The regulation specifies two categories of consequences of concern into which an accident may fall. The first category is referred to in proposed §70.61 as "high consequences," the second as "intermediate consequences." Implicitly there is a third category, namely, those accidents

that produce consequences less than "intermediate." These will be referred to as "low consequence" accidents. Since the primary purpose of process hazard analysis is to identify all accidents having consequences of concern, it will, in some cases, be necessary to identify accidents that produce radioactive or chemical exposures, then subsequently determine that some of these exceed the threshold values of the regulation. For this reason, the list of accidents resulting from such analysis will include such low consequence accidents in order to show that they have been considered. Otherwise, the analysis will not have demonstrated its completeness.

The limits defining the three accident consequence categories are given in Table A-1. Note that the categories are numbered in ascending order of the magnitude of their consequences. The usefulness of this numbering will be evident later. The symbols CHEM3, CHEM2, and CHEM1 refer to quantitative standards selected by the applicant in accordance with proposed 10 CFR 70.61(b)(4)(ii), 70.61(c)(4)(ii) –*e.g.*, AEGL or ERPG, as appropriate.

Consequence Category 3--High Consequences: An accident resulting in any consequence specified in proposed §70.61(b); that is: an acute worker exposure of 1 Sv (100 rem)[1] or greater TEDE[2], or a chemical exposure that could endanger the life of a worker (as defined by the applicant); or acute exposure of a member of the public outside the controlled area to a radiation dose (D) of 0.25 Sv (25 rem) or greater TEDE, a 30 mg soluble uranium intake, or a chemical exposure that could lead to irreversible or other serious long-lasting health effects, as defined by the applicant (represented herein as CHEM3).

Consequence Category 2--Intermediate Consequences: An accident resulting in any consequence specified in proposed §70.61(c). That is, acute exposure of a worker to a radiation dose of 0.25 Sv (25 rem) or greater but less than 1 Sv (100 rem) TEDE, or chemical exposure that could lead to irreversible or other serious long-lasting health effects, as defined by the applicant (represented herein as CHEM2); or acute exposure of a member of the public outside the controlled area to a radiation dose 0.05 (5 rem) or greater but less than 0.25 Sv (25 rem) TEDE, or a chemical exposure that could cause mild transient health effects, as defined by the applicant (represented herein as CHEM1); or prompt release of radiation outside the restricted area that would, if averaged over a 24-hour period, exceed 5000 times the values specified in Table 2 of Appendix B to 10 CFR Part 20.

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[1]A nuclear criticality would normally be considered a high consequence event because of the potential for producing a high radiation dose to a worker.

[2]TEDE is Total Effective Dose Equivalent (see 10 CFR Part 20), represented by 'D'.

Consequence Category 1--Low Consequences: Any accident with potential adverse radiological or chemical consequences but at exposures less than consequence Categories 3 and 2 above.

TABLE A-1: Consequence Severity Categories Based on Proposed 10 CFR 70.61

	Workers	Offsite Public	Environment
Consequence Category 3: high	$D^2 \geq 1$ Sv (100 rem) ≥ CHEM3	D ≥ 0.25 Sv (25 rem) 30 mg sol U intake ≥ CHEM2	
Consequence Category 2: intermediate	0.25 Sv ≤ D < 1 Sv ≥ CHEM2 but < CHEM3	0.05 Sv≤D < 0.25 Sv ≥CHEM1 but < CHEM2	radioactive release > 5000 x Table 2 App B 10 CFR 20
Consequence Category 1: low	Accidents of lesser radiological and chemical exposures to workers than those above in this column	Accidents of lesser radiological and chemical exposures to the public than those above in this column	Radioactive releases producing effects less than those specified above in this column

Corresponding to the two consequence categories of the rule (Categories 2 and 3 above), proposed §70.61 requires corresponding levels of graded protection, that is, engineered or administrative controls (or a combination thereof), sufficient to ensure that the likelihood of these adverse events is correspondingly low. The two categories of likelihood thus prescribed are:

Likelihood Category 1: Consequence Category 3 accidents must be "highly unlikely;" and

Likelihood Category 2: Consequence Category 2 accidents must be "unlikely."

Implicitly there is a third category into which an accident could fall, that is it could fail to be "unlikely." This category will be referred to in this document as:

Likelihood Category 3: "Not unlikely."

Although this likelihood category includes unintended events that might actually be expected to happen, others might be less frequent. For this reason, the term "likely" was not used for these events.

Per proposed §10 CFR 70.61, the applicant must use the ISA is to document its compliance with the performance requirements. This evaluation should be done using a tabular summary of identified accident sequences. One acceptable way of doing so is for the applicant to assign two category numbers to each accident sequence, one based on its consequences and one for likelihood. The product of these two category numbers is then used as a risk index. Listing this calculated risk index in the tabular summary provides a simple method for showing that the graded protection requirements have been met for each accident sequence. A risk index value less than or equal to "4" means the sequence is acceptable. If the applicant provides this risk index in one column of the tabular summary, the reviewer can quickly scan this column to confirm that each accident conforms to the safety performance requirements of proposed 10 CFR 70.61. This system is equivalent to assigning each accident to a cell in a 3 by 3 matrix. This conceptual matrix is shown in Table A-2. The values in the risk matrix cells are the risk index numbers.

TABLE A-2: Risk Matrix

	Likelihood Category 1: highly unlikely	Likelihood Category 2: unlikely	Likelihood Category 3: not unlikely
Consequence Cat. 3 High	3 acceptable	6 unacceptable	9 unacceptable
Consequence Cat. 2 Intermediate	2 acceptable	4 acceptable	6 unacceptable
Consequence Cat. 1 Low	1 acceptable	2 acceptable	3 acceptable

To demonstrate compliance with the system described above, the applicant needs to assign consequence categories to each identified accident in order to determine which likelihood requirement applies. Then those accident sequences identified as high or intermediate consequences must be assigned to a likelihood category. To be acceptable, these assigned consequences and likelihoods must have a valid basis, and the applicant must demonstrate this basis in the documentation submitted in the application. The following sections describe an acceptable method for making these assignments.

A2. CONSEQUENCE CATEGORY ASSIGNMENT

The assignment of consequence categories is based on estimated consequences of prototype accidents. Criteria for the presentation of these estimates by the applicant is described in Section 5.4.3.2(B)(iv). Although consequences of accidents can be determined by actual calculations, it is not necessary that such a calculation be performed for each individual accident sequence listed. Accident consequences may be estimated by comparison to similar events for which reasonably bounding conservative calculations have been made. The applicant should document the bases for bounding calculations of the consequence assignment in the submittal. NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," describes valid methods and data to be used by the applicant and may be used for confirmatory evaluations by the reviewer.

A3. LIKELIHOOD CATEGORY ASSIGNMENT

An assignment of an accident sequence to a likelihood category is acceptable if it is based on the record of failures at the facility or other methods that have objective validity. Failure data from other facilities may also be used, but care should be taken to ensure its applicability. Because the sequences leading to accidents often involve multiple failures, a combination of failure frequency and probability values determines the likelihood of the whole accident sequence. These values include the frequencies of initiating events and failure likelihoods of safety controls. As described below, the applicant may estimate an approximate likelihood category for an accident sequence by considering all the events involved. This method uses the number, type, independence, and observed failure history of safety controls. However, correctly evaluating the appropriate likelihood of accidents using such a qualitative approach depends on the informed judgement of the analyst. Safety controls, even those of the same types, have a wide range of reliability. The ultimate criterion for acceptability, is that the frequencies of initiating events and the likelihood of failure of safety controls involved is sufficiently low so that the entire accident sequence is "highly unlikely" or "unlikely" as required by proposed 10 CFR 70.61. The virtue of the approach is that it requires explicit consideration of some of the underlying events and factors that affect the likelihood of the accident. Another virtue is that the more explicit the criteria for assignment are, the more consistent are the results.

Underlying any evaluation of an accident sequence as "unlikely" or "highly unlikely" is an implied assessment of its "likelihood" or frequency of occurrence. The structured procedure described below will indicate which likelihood category may be appropriate for an event. In order to maintain internal consistency in evaluating different control systems and accidents, it was necessary to derive this structured procedure based on the underlying frequencies of events. The following numerical guidelines were thus used to obtain consistency:

Likelihood Category 1: Highly unlikely, a frequency of less than 10^{-5} per year per accident;

Likelihood Category 2: Unlikely[3], a frequency of less than $4x10^{-4}$ per year per accident (but more frequent than 10^{-5}); and

Likelihood Category 3: Not unlikely, more frequent than $4x10^{-4}$ per year per accident

In assigning specific numerical values to these likelihood categories, we are making definitive assumptions about the number of accident sequences. The Commission's strategic goals are stated in terms of total industry risk, so that the per accident probabilities must be expressed as the cumulative likelihood divided by the total number of accident sequences. For the purposes of this example, it will be assumed throughout the remainder of this appendix that there are 100 intermediate consequence accidents and 1000 high consequence accidents across the industry (this is consistent with SRP Section 5.4.3.2).

With this assumption, each individual accident sequence in this likelihood category should have a frequency no greater than 10^{-5} per year (i.e., one accident of this type every 100,000 years). This number can be multiplied by the total number of accident sequences to give the cumulative likelihood of all accident sequences in a given category at the facility, in units of yr^{-1}.

In assessing the adequacy of safety controls, individual accident frequencies greater than 10^{-5} per year may not be assigned a likelihood Category 1, that is, "highly unlikely." The NRC has a strategic safety performance measure of no inadvertent nuclear criticalities. For this reason, the acceptability of any given frequency depends on the total number of accidents that may be identified. Since the total number and consequences of all potential accidents at a facility is not accurately known until its ISA is completed, it is difficult to establish a definitive acceptable frequency. Individual accidents may need to be limited to lower frequencies to meet the performance requirements. On the other hand, the fact that a particular accident sequence is below this value does not automatically mean that it is

[3]A distinction must be drawn between the concept of "unlikely" in regard to intermediate consequence events and "unlikely" in regard to the double contingency principle. The above definition of unlikely does not apply to a nuclear criticality (which should be regarded as a high consequence event in unshielded facilities in most instances). In meeting double contingency, unlikely typically means $\leq 10^{-2}$.

clearly acceptable. The frequencies should be used as a guideline in developing more consistent and objective standards for safety goals. These likelihoods may be derived by considering the Commission goal that there should be no accidental criticalities at any regulated facility.

As an example, the value of 10^{-5} per year per accident in a facility with 100 potential accident sequences (Consequence Category 3) would yield a cumulative frequency for Consequence Category 3 accidents of:

$$100 \text{ accidents} \times 10^{-5} \text{ per year per accident} = 10^{-3} \text{ per year.} \qquad \text{(Eq. A-0)}$$

These Category 3 accidents generally result in fatalities. The average statistic for all manufacturing industries is that a facility with 250 manufacturing workers would expect 10^{-2} on-the-job deaths per year (see References, Statistical Abstract of the U.S.). The number of 10^{-3} per year is consistent with the Commission goal that there should be no accidental criticalities at regulated facilities. With approximately 10 regulated facilities in the United States, this should ensure that the likelihood of an accidental criticality anywhere in the country is no greater than 10^{-2}. A recurrence period of 100 years is sufficient to provide reasonable assurance that a criticality accident will not occur during the lifetime of any regulated facility.

Similarly, accident sequences having frequencies more than 4×10^{-4} per year per accident are considered "not unlikely" (assuming on the order of 100 accident sequences of this type in the industry). Again this value should not be taken as a definitive criterion for acceptability. It is a guideline value to assure consistency. It may need to be adjusted based on the numbers and severity of accidents. The rationale for the value 4×10^{-4} is that accidents of the corresponding severity, Consequence Category 2, are not common and should remain so. This is based on a Commission strategic goal, that there should be no increase in reportable radiation releases, as discussed in SRP Section 5.4.3.2(B)(ix). To achieve this, the product of this frequency per accident per year with the assessed number of potential accidents should provide adequate confidence that such accidents will not occur. Note again that these values of 10^{-5} and 4×10^{-4} are per year per accident.

The accident evaluation method described below does not preclude the need to comply with the double contingency principle for sequences leading to criticality. Although exceptions are permitted with compensatory measures, double contingency, should be applied. The reason double contingency is needed is the fact that there is usually insufficient firm data as to the reliability of the control equipment and administrative control procedures used in criticality safety. If only one item were relied on to prevent a criticality, and it proved to be less reliable than expected, then the first time it failed, a criticality accident could result. For this reason, it is prudent to require two independent controls. Inadequate controls can then be determined by observing their failure, without also suffering the consequence of a criticality. Even with double contingency, it is essential that each of the items relied on for

safety (IROFS) be sufficiently unlikely to fail. This is so that, if one of the two items that establish double contingency is actually ineffective, criticality will still not be likely.

A4. RISK INDEX EVALUATION SUMMARY

As mentioned in Section A3, an acceptable way for the applicant to present the results of the ISA is a tabular summary of the identified accident sequences. Table A-9 is an acceptable format for such a table. This table lists several example accident sequences for a powder blender at a MOX facility. Table A-9 summarizes two sets of information: (1) the accident sequences identified in the ISA and (2) a risk index calculated for each sequence to show compliance with the regulation.

A fault tree is another acceptable method of presenting the results. As shown by the example, for the purposes of documenting compliance with the double contingency principle, a fault tree provides a fuller description of the control systems, and the logical progression of the accident, than a tabular format can, and is thus considered the preferred method. Both of these methods will be presented in the tables which follow.

Accident sequences result from initiating events, followed by failure of one or more controls. Thus, there are columns in Table A-9 for the initiating event and for controls which may be mitigative or preventive. In most cases, the initiating event will be the failure of one of the preventive controls. There may also be accident sequences resulting from external events such as fires or earthquakes.

With redundant safety controls, and in certain other cases, there are sequences where an initiating event occurs that places the system in a vulnerable state. While the system is in this vulnerable state, a safety control must fail in order for the accident to result. Thus, the frequency of the accident depends on the frequency of the first event, the duration of vulnerability, and the frequency of the (second) control failure. For this reason, it is necessary to consider the duration of the vulnerable state and to assign it a duration index. The values of all index numbers for a sequence are added to obtain a total likelihood index, T. Sequences are then assigned to one of the three likelihood categories of the Risk Matrix depending on the value of this index in accordance with Table A-3.

Table A-3: Determination of Likelihood Category

LIKELIHOOD CATEGORY	LIKELIHOOD INDEX T (= sum of index numbers)
1	$T \leq -5$
2	$-5 < T \leq -4$
3	$-4 < T$

The likelihood category in Table A-3 applies to the accident sequence of a whole and is used to assess the overall likelihood of the sequence, not the likelihood of individual controls used in meeting double contingency.

The values of index numbers in sequences are assigned considering the criteria in Tables A-4 through A-6. Each table applies to a different type of event. Table A-4 applies to events which have frequencies of occurrence, such as initiating events and certain control failures. When failure probabilities are required for the event, Table A-5 provides the index values. Table A-6 provides index numbers for durations of failure. These are used in certain accident sequences where two controls must simultaneously be in a failed state. In this case, one of the two controlled parameters will fail first. It is then necessary to consider the duration that the system remains susceptible to failure of the second. The reverse sequence, where the second control fails first, should also be considered as a separate accident sequence. (Since the example chosen concerns mainly criticality safety, the failure of each control relied on to meet the double contingency principle must be considered as the initiating event of an accident sequence.) This is necessary because the duration of failure of the second control will usually differ from that of the first. The values of these duration indices are not merely judgmental. They are directly related to the time interval of surveillance monitoring for failures. That is, the duration of a failure is the time until it is detected plus the time to restore the system to a state where it is not vulnerable to the second failure.

If the probability of failure for the first preventive control is P_1 (in units of events per yr), its duration of failure is d_1 (in years), and the probability and duration of failure of the second control is P_2 and d_2, then P_1P_2 is the probability that both controls will fail within the year. The probability that both controls will be in a failed state simultaneously is $P_1P_2(d_1+d_2)$. The two terms $P_1P_2d_1$ and $P_1P_2d_2$ correspond to the direct and reverse accident sequences (that is, where Control 1 fails first followed by Control 2, and *vice versa*). Thus, we see that taking the duration index into account can produce a substantial reduction in the overall likelihood of the accident sequence.

For all these index numbers, the more negative the number is, the less likely is the failure. Accident sequences may consist of varying numbers of events, starting with an initiating event. The total likelihood index is the sum of the indices for all the events in the sequence, including those for duration.

Consequences are assigned to one of the three consequence categories of the Risk Matrix based on calculations or estimates of the actual consequences of the accident sequence (see Table A-1). Multiple types of consequences can result from the same event. The consequence category for an event is chosen for the most severe consequence.

As shown in the first row of Table A-9, the failure duration index can make a large contribution to the total likelihood index. Therefore, the reviewer should verify that there is

adequate justification that the failure will be corrected in the time ascribed to the duration index. In general, duration indices with values less than minus one (-1), corresponding to 36 days (about one month), to be acceptable, should be based on the intentional monitoring frequency of the process. The failure duration for an unmonitored process should be conservatively estimated.

Table A-7 provides a more detailed description of the accident sequences used in the example of Table A-9. The reviewer needs the information in Table A-7 to understand the nature of the accident sequences listed in Table A-9. Table A-9 lacks sufficient room to explain any but the simplest failure events.

Table A-8 is used to explain the safety controls and external initiating events that appear in the accident sequences in Table A-9. The reviewer needs the information in Table A-8 to understand why the initiating events and safety controls listed in Table A-9 have the low likelihood indices assigned. Thus, Table A-8 needs to address such information as: the margins to safety limits, the redundancy of a control, and the measures taken to assure adequate reliability of a control. Table A-8 must also justify why those external events, which are not obviously extremely unlikely, have the low likelihoods which are being relied on for safety. The applicant should provide separate tables to list the controls for criticality, chemical, fire, radiological, and environmental accidents.

Table A-4: Failure Frequency Index Numbers

FREQUENCY INDEX NUMBER	BASED ON EVIDENCE	BASED ON TYPE OF CONTROL**	COMMENTS
-6 *	External event with freq. < 10^{-6} per yr		If initiating event, no controls needed
-4 *	No failures in 30 yr for hundreds of similar controls in industry	Exceptionally robust passive engineered control (PEC), or an inherently safe process, or 2 independent active engineered controls (AECs), PEC, or enhanced admin. controls.	Rarely justified by evidence, since few systems are found in such large numbers. Further, most types of single control have been observed to fail.
-3 *	No failures in 30 yr for tens of similar controls in industry	A single control with redundant parts, each a PEC or AEC	
-2 *	No failure of this type in this facility in 30 yr	A single PEC	
-1	A few failures may occur during facility lifetime	A single AEC, an enhanced administrative control, an admin. control with large margin, or a redundant admin. control	
0	Failures occur every 1 - 3 yr	A single administrative control	
1	Several occurrences per yr	A frequent event	Not for safety controls, just initiating events
2	Occurs every week or more often	Frequent event, an inadequate control	Not for safety controls, just initiating events

* Numbers less than (more negative than) "-1" should not be assigned to controls unless the configuration management, auditing, and other management measures are of high quality, because without these measures, the controls may be changed or not maintained.

** The failure frequency index assigned to a control of a given type in column 3 may be one value higher or lower than the value given in column 1, since the reliability of different types of controls can vary widely. Criteria justifying assignment of the lower (more negative) failure frequency index should be given in the narrative describing ISA methods. Exceptions should be individually justified.

Table A-5: Failure Probability Index Numbers

PROBABILITY INDEX NUMBER	PROBABILITY OF FAILURE ON DEMAND	BASED ON TYPE OF CONTROL	COMMENTS
-6 *	10^{-6}		If initiating event, no controls needed
-4 or -5 *	10^{-4} - 10^{-5}	Exceptionally robust passive engineered control (PEC), or an inherently safe process, or 2 redundant controls better than simple admin controls (active engineered control (AEC), PEC, or enhanced admin.)	Rarely can be justified by evidence, since few systems are found in such large numbers. Further, most types of single controls have been observed to fail.
-3 or -4 *	10^{-3} - 10^{-4}	A single PEC or an AEC with high availability	
-2 or -3 *	10^{-2} - 10^{-3}	A single AEC, or an enhanced admin control, or an admin control for routine planned operations	
-1 or -2	10^{-1} - 10^{-2}	An admin control that must be performed in response to a rare unplanned demand	

* Probability index numbers less than (more negative than) "-1" should not be assigned to controls unless the configuration management, auditing, and other management measures are of high quality, because, without these measures, the controls may be changed or not maintained.

Figure A-2 presents the same information as a set of fault tree diagrams. A discussion and comparison of the two methods follows the example.

Definitions and explanations of the terms used in the following tables and figures will follow the example.

As an understanding of the example systems is important, process descriptions for hypothetical MOX processes follow. These hypothetical systems were chosen because of their relatively high degree of importance for nuclear criticality and because they represent the extremes in terms of operational and control complexity. The first example, the solvent extraction system, is a complex chemical operation that is most amenable to a fault tree presentation of the results of the ISA summary (though to compare the strengths and weaknesses of the two methods, both fault trees and a tabular format are presented). The second example, mixed oxide blending, is much more straightforward in terms of controls and the results of the ISA can be summarized more effectively in terms of a table of accident sequences.

These examples should only be considered typical of the degree of information required and the ways in which it may be displayed. It is anticipated that the applicant's ISA Summary and process description may differ markedly from the example. These examples should not be construed to preclude other methods of presenting the ISA summary results.

A5. OVERALL PROCESS DESCRIPTION

The purpose of the front-end Plutonium Purification Process (P^3) is to remove impurities such as gallium and americium from the plutonium oxide feed, producing a more suitable plutonium feed stream for the oxide blending process. This process description is for illustrative purposes only and should not be expected to conform to any particular applicant's process. The actual license application would require a more detailed process description than that presented below, but the following brief summary is presented to aid in understanding the example:

Raw plutonium oxide (PuO_2) powder is received from the shipper and batched into a glovebox at the front end of the Aqueous Polishing (AP) processing line. The containers of PuO_2 are fed into an electrically-heated dissolver unit in the glovebox, consisting of a favorable geometry recirculation loop with electrodes at either end. The PuO_2 is digested by the addition of nitric acid in the presence of Ag^{++} ions, resulting in the formation of an impure plutonium nitrate ($Pu(NO_3)_4$) solution at a concentration of ~250 gPu/l. Plutonium can exist in several oxidation states in nitric solutions simultaneously, which complicates the process chemistry considerably. Although the plutonium in PuO_2 is tetravalent (Pu(IV)), it undergoes disproportionation, or self-oxidation and reduction, to both Pu(III) and Pu(VI) through the reaction

$$3Pu^{4+}\ (IV) + 2H_2O \rightleftharpoons 2Pu^{3+}\ (III) + PuO_2^{2+}\ (VI) + 4H^+ \qquad \text{(Eq. A-1)}$$

Table A-6: Failure Duration Index Numbers

DURATION INDEX NUMBER	AVG. FAILURE DURATION	DURATION IN YEARS	COMMENTS
-5	5 minutes	10^{-5}	
-4	1 hour	10^{-4}	
-3	8 hours	0.001	
-2	A few days	0.01	
-1	One month	0.1	Formal monitoring to justify indices less than "-1"
0	One year	1	
1	More than 3 years	10	

The plutonium must be adjusted to the tetravalent state to ensure effective extraction. This is done as a two-step process. First, Ag^{++} is generated at the cathode and acts as an oxidation agent to drive both Pu(III) and Pu(IV) to Pu(VI). Tetravalent plutonium is oxidized through the reaction $Pu(IV) + Ag^{++} = Pu(VI) + Ag$.

After the operators determine that complete PuO_2 dissolution is achieved by means of independent dual sampling, the $Pu(NO_3)_4$ is fed through a favorable geometry in-line filter into the solvent extraction feed preparation slab tank. (*N.B.* Plutonium in $Pu(NO_3)_4$ is actually in the tetravalent state; the chemical form after oxidation is more accurately characterized as a mixture of Pu(VI) cations in a NO_3^--rich solution.) The free Pu(VI), or PuO_2^{+2} plutonyl ions, must be adjusted from the hexavalent to the tetravalent state Pu(IV) by the addition of excess HNO_3 and H_2O_2 (a reducing agent) at a low plutonium concentration. This is done in the favorable geometry preparation tank. The entire aqueous polishing process is conducted on a batch basis, with approximately 14 kg (30.8 lb) Pu processed through dissolution, solvent extraction, precipitation, and calcination in each batch. The powder is then mixed together with natural uranium oxide to form the master blend.

A6. SOLVENT EXTRACTION PROCESS (PLUTONIUM PURIFICATION)

The Solvent Extraction, Scrub, and Strip columns consist of identical long (~20 feet [6.1 m]), 5-inch (12.7 cm) diameter Pyrex columns containing a series of stationary perforated plates. For solvent extraction, the aqueous $Pu(NO_3)_4$ solution is added at the top of the column and a mixture of TBP, or tributyl phosphate (chemical formula $(C_4H_9)_3PO_4$), and a diluent (30% hydrogenated tetrapropylene, or HTP) is added at the bottom of the column. The difference between the relative specific gravities of the two streams causes the aqueous solution to sink to the bottom and the organic mixture to rise to the top of the column. The immiscible fluids are pulsed in the columns by means of positive-displacement pumps. This pulsing breaks up the interface between the fluids and increases the surface area, resulting in intimate mixing to increase the efficiency of extraction. The tetravalent plutonium ion Pu^{4+} becomes complexed to the organic through the reaction:

$$Pu^{4+}(aq) + 4NO_3^-(aq) + 2TBP(o) = Pu(NO_3)_4 \cdot 2TBP(o) \qquad \text{(Eq. A-2)}$$

The existence of a salting agent such as HNO_3 or $Al(NO_3)_3$ increases the acid molarity of the excess nitric ion, and causes the above reaction to be shifted to the right.

In the scrub column, the fissile-bearing organic stream from the top of the solvent extraction column is fed into the bottom of the scrub column. Additional nitric acid is added to the top of the scrub column and the same countercurrent operation repeated, to remove additional impurities from the organic into the aqueous phase. The plutonium remains in the organic phase at the end of the scrub operation. The aqueous *raffinate* stream–which should now contain low levels of plutonium but concentrated fission products–is transferred to raffinate storage while the fissile-bearing organic stream fed into the bottom of the strip column.

In the strip column, deionized water is added to reduce the acid molarity, causing the left hand side of Equation A-2 to be favored. The aqueous product stream containing purified plutonium nitrate is then transferred to the first pass evaporator; the spent organic phase must be reconditioned for reuse in the first pass solvent extraction. The evaporator consists of a tube-and-shell heat exchanger in which the concentration of the $Pu(NO_3)_4$ increases from 40 gPu/l to around 250 gPu/l. The second and third pass solvent extraction lines are nearly identical to the first, except that the $Pu(NO_3)_4$ is at a higher concentration.

Raffinate and solvent conditioning streams attach to the process at various points. The first, second, and third pass raffinate streams are transferred to a bank of favorable geometry columns (from the extraction and scrub columns, solvent regeneration, and evaporator condensate), where they are sampled (by dual independent sampling) for Pu content. Only after they meet the release criteria of 0.015 gPu/l are the contents discharged (through an in-line monitor that is interlocked to the waste tank isolation valves) to a set unfavorable geometry waste water tanks, for waste water treatment and eventual discharge from the site. In addition, organic solvent from the strip columns must be regenerated because it contains a build-up of metallic impurities (primarily gallium and americium), nitric acid (acquired through the reaction $H^+(aq) + NO_3^-(aq) + 2TBP(o) = HNO_3 \cdot TBP(o)$), and various radiolytic decomposition products of TBP and kerosene, such as dibutyl phosphate (DBP). The solvent is washed with Na_2CO_3, NaOH, and HNO_3 in a series of favorable geometry Mixer/Settlers (M/Ss) to remove impurities, filtered, and recycled to the solvent extraction columns. Gallium and americium is further removed by electrolytic deposition on charged plates in the M/S units. Makeup solvent from bulk chemical tanks is added as needed to maintain the solvent inventory. The M/Ss consist of a safe geometry box partitioned by a short wall into a mixing chamber and a settling chamber. The mixing chamber contains a rotary impeller which draws the heavier liquid (aqueous wash solution) from the bottom of the mixing chamber and emulsifies it into the lighter liquid (organic solvent) in the top of the mixing chamber. Following this intimate mixing (which operates under the same principle as the pulsed extraction columns), the solution gravity drains into the settling chamber, where it separates into two distinct layers. The organic is drawn off to the next wash stage or to the fresh solvent column, while the aqueous is discharged to the raffinate storage columns.

Following third pass solvent extraction, the purified $Pu(NO_3)_4$ must be re-converted to PuO_2 for blending with UO_2. This is accomplished by the addition of oxalic acid ($H_2C_2O_4$) to cause the precipitation of plutonium as plutonium oxalate ($Pu(C_2O_4)_2$). Hydrogen peroxide (H_2O_2) is added to the plutonium nitrate solution to ensure that it is in the proper oxidation state. After sampling, the solution is transferred to the precipitation column, a short 4-inch (10.2 cm) diameter glass column contained within a glovebox, in which the plutonium oxalate is prepared. Precipitation proceeds through the reaction:

$$Pu(IV)(NO_3)_4 + 2H_2C_2O_4 = Pu(C_2O_4)_2 + 4HNO_3 \qquad \text{(Eq. A-3)}$$

The resulting precipitate is prepared through the slow addition of oxalic acid to the column, and is thixotropic in nature. The nitric acid content must also be adjusted to obtain the

desired level of consistency. The resultant plutonium oxalate slurry collects at the bottom of the column. The residual nitric-water solution contains only low levels of plutonium nitrate and is sampled for discharge. Solutions which contain greater than the release criteria of 0.015 gPu/l are recycled to solvent extraction for re-extraction. This dilute nitric solution is decanted and filtered before transfer to acid recovery, and the material at the bottom of the bowl drained out before being air-dried in the glovebox. When the material is dried, it forms a cake containing plutonium oxalate hydrates (such as $Pu(C_2O_4)_2 \cdot 6H_2O$). The material is gravity drained from the bottom of the precipitator, where it is automatically dropped through a chute into an inclined, rotary-kiln calciner in a continuous process. The slurry is then calcined in an electrically heated oxidation furnace at 300 °C and then converted to PuO_2 at 900 °C in an oxygen-rich atmosphere in the same furnace. The PuO_2 is collected into a moderation-controlled hopper, which is connected to a favorable geometry tumbling mixer to achieve proper homogenization. The mixer consists of two rotating drums with a spiral blade in the intervening space with a cadmium shaft for neutron poison. After homogenization, the material is transferred to a glovebox where it is sampled, bottled, and transferred to the Mixed Oxide Blending Operation of the MOX Process (MP) Line.

The overall process flow is shown in Figure A.1.

Controlled parameters in the solvent extraction process are geometry, concentration, spacing, interstitial moderation, and process variables (material form). The solvent extraction columns were modeled using an optimal plutonium nitrate concentration of--140 GPU/l, without taking credit for the presence of gallium –a mild neutron poison--or excess nitric acid. The solution was modeled to the outer diameter of the columns, and thus took credit for the diameter but not the column thickness. Credit was not taken for the plutonium isotonic (~4wt% ^{240}Pu), as the models assume the feed consists solely of ^{239}Pu. Concentration was not controlled for the extraction columns, but was credited for keeping the waste tanks subclinical upon solution transfer from the refined storage columns.

Because the design relies primarily on passive engineered features (*i.e.*, fixed geometry and spacing), the potential for nuclear criticality in the solvent extraction system itself is extremely unlikely. The main accidents of concern are transfer of concentrated solution to unfavorable geometry process equipment. As shown in Figure A-1, the unfavorable geometry systems that are connected to the process consist of (i) steam supply for the evaporators, (ii) demonize water, nitric acid, and solvent regeneration bulk chemical supply tanks, (iii) waste water system tanks, and (iv) the floor.

The example shown in the following tables is for the second pass solvent extraction (2SX) in the P^3 Process Node. The list of accident sequences and controls is for illustrative purposes only and is not meant to be exhaustive.

1. PROCESS CRITICALITY FLOW DIAGRAM

   Figure A-1 is an example of one method of describing the process flow. A good understanding of the process flow and the criticality control systems that exist at each

node in the process is essential to evaluating the results contained in the ISA Summary. The information contained in this Process Criticality Flow Diagram (PCFD) is a more condensed form of the information that would be expected in the process description, process flow diagrams (PFDs), and criticality safety evaluations. Presenting the information in this way is advantageous to the applicant, as it is a more efficient means of providing needed process knowledge to the ISA or safety discipline reviewer. Basically, the PFD. is a PFD modified to contain the features relied on for criticality safety. Note several useful features of this diagram:

The different process steps are divided into two categories by shape, those relying on favorable geometry and those which are unfavorable geometry. Distinguishing between these two types of systems may be done by several other means. Geometry control is typically ranked as the most preferable control due to its inherent stability and robustness, and is the primary control relied on in most of this particular system. In certain other systems, it may be somewhat more advantageous to draw a distinction between process steps that are moderation controlled and uncontrolled areas, or between concentration controlled and uncontrolled areas. By reviewing this diagram, it is immediately apparent where the transition from favorable to unfavorable geometry takes place.

Another feature of this diagram is that the engineered features relied on for criticality safety are clearly identified by shading. There is a simple graphic representation of the barriers that exist between favorable and unfavorable geometry equipment, which are drawn as bars across the flow path between these systems. This makes it readily apparent what features prevent the backflow of concentrated fissile solution to unfavorable geometry equipment, among other scenarios. Adding the labels that correspond to each of the IROFS (as in Table A-8) would provide a ready cross-reference, but may result in too much added complexity for such a system.

The use of different line patterns to distinguish between the various streams–particularly with respect to different fissile compositions–facilitates understanding of the process flow. Another useful feature is the division of the entire diagram into different zones corresponding to various process nodes. This provides a clearer boundary definition and allows the review to see how the system functions together as an integrated whole, including how perturbations in criticality controls in one process node or piece of equipment flows down into the next. The engineered controls tabulated in the ISA Summary (such as Table A-8) should include all features relied on for safety within the boundary of that process node. Finally, this diagram displays the actual controlled parameters at each process step; to the degree possible, this should be extended to display the actual controlled values of those parameters.

This diagram should be consulted in reviewing the sample tables.

Figure A-1: Criticality Flow Diagram for P [3] Operation

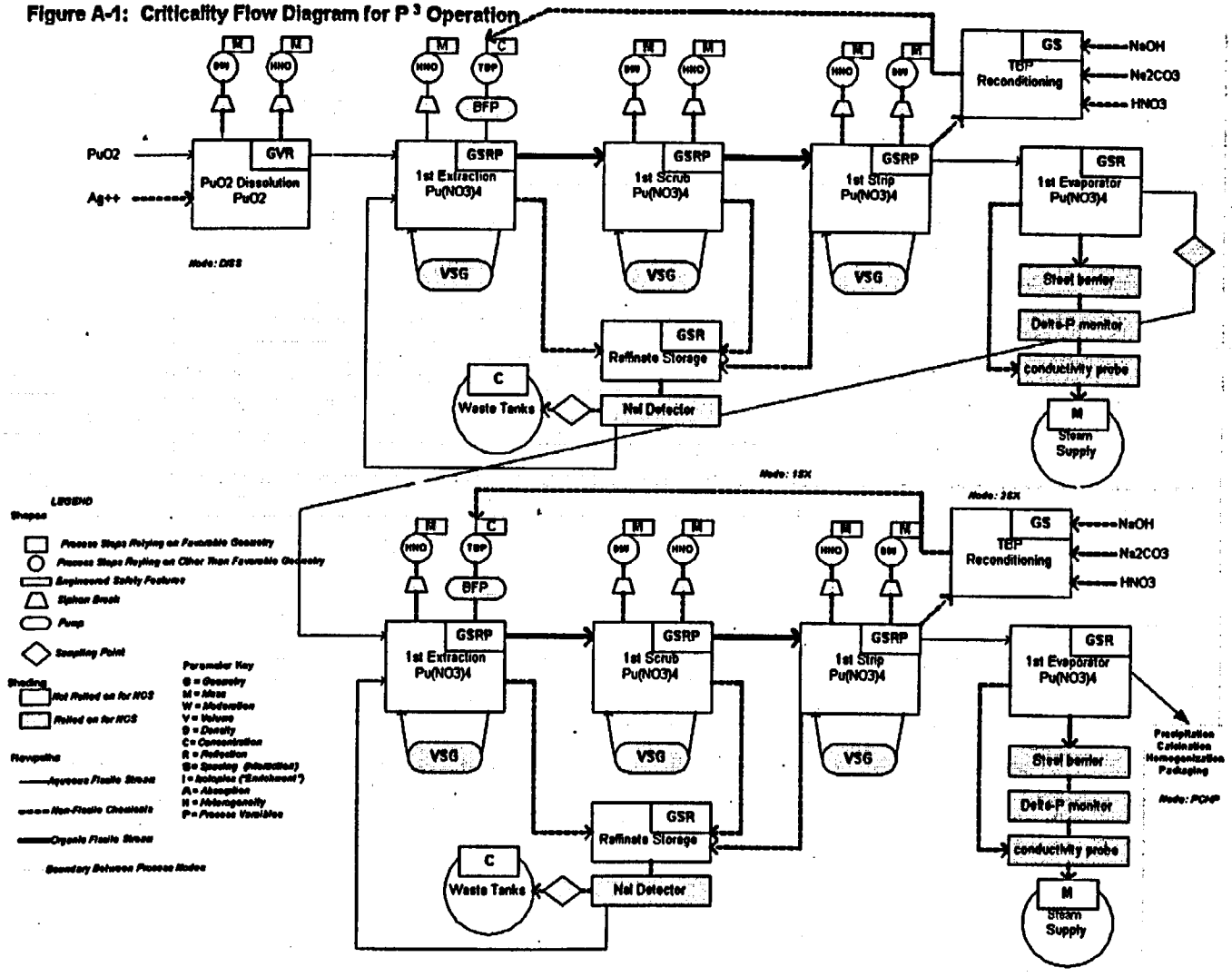


Table A-7: Accident Sequence Descriptions

Accident Sequence (see Table A-9)	DESCRIPTION
Loss of MASS	
MOB-001	Exceeding the mass limit of the blend tank, by adding too much UO_2 blendstock. This will have the effect of increasing the overall mass present, but will simultaneously decrease the plutonium "enrichment." The overall effect of this is to increase the distance from the subclinical curve of mass as a function of plutonium "enrichment" as more blendstock is added. The system is adequately subclinical under conditions of double batching uranium. To achieve criticality, this would have to be followed by a loss of moderation control.
MOB-002	Exceeding the mass limit of the blend tank, by adding too much PuO_2. This will have the effect of increasing both the overall mass and plutonium "enrichment." At ~33 kg PuO_2 (73 lb) (and 23 wt%) the subclinical mass limit will be exceeded. Therefore, this could lead to criticality without any additional upsets and therefore dual controls are established on the plutonium mass.
MOB-003	Exceeding the mass limit of the blend tank by performing the blending operation while there is still blended oxide present from the previous batch in the tank. Assuming the previous batch was properly mixed, it would require an additional 50 kg (110 lb) of PuO_2+UO_2 to exceed the subclinical mass limit. Therefore this could lead to criticality without any additional upsets and therefore dual controls are established to ensure the blend tank is empty of material before another batch is started.
Loss of MODERATION	
MOB-004	Exceeding the moderation limit (1wt% H_2O) by adding UO_2 which has not been properly sampled. This could lead to criticality without any additional failures. Dual independent sampling is required to ensure moisture limits are adhered to. Also, material will not freely flow through orifice if wet.
MOB-005	Exceeding the moderation limit (1wt% H_2O) by adding PuO_2 which has not been properly sampled. This could lead to criticality without any additional failures. Dual independent sampling is required to ensure moisture limits are adhered to. Also, material will not freely flow through orifice if wet. In addition, both the plutonium feed hopper and blend hopper are heated. Material is added at a sufficiently slow rate that contact with the heated blendstock will cause moisture in the plutonium to be driven off.
MOB-006	Exceeding the moderation limit (1wt% H_2O) by introduction of liquid water from overhead water lines or roof leaks. The blend tank is completely enclosed within an airtight and watertight enclosure. There are no overhead water lines allowed. The most likely cause of this scenario is backflow of condensate from the ventilation header, which serves to remove evolved water from the heated material. The ventilation header is sloped and equipped with drain lines to ensure against condensate backflow. Even in the event of water intrusion, the heating is sufficient to drive off any realistic accumulation of liquid water.
Loss of PLUTONIUM "ENRICHMENT"	
MOB-007	Exceeding the plutonium "enrichment" by adding too little blendstock to the blending hopper. This will have the effect of increasing plutonium "enrichment" while decreasing the overall mass. This will eventually reach criticality without any additional failures, by only when more than half the original UO_2 blendstock is omitted.
MOB-008	Exceeding the plutonium "enrichment" by adding too much PuO_2 feed to the blending hopper. This is identical to Scenario MOB-002 and will be discussed as a loss of mass control.
MOB-009	Exceeding the plutonium "enrichment" by adding PuO_2 to the blending hopper without first adding blendstock. This is the bounding case of Scenario MOB-007. Controls are established to ensure that blendstock is added and in the correct proportion before addition of PuO_2 feed is allowed.

Accident Sequence (see Table A-9)	DESCRIPTION
MOB-010	Exceeding the plutonium "enrichment" by the formation of clumps of higher enrichment PuO_2 in the blending hopper. Clumping can be caused by i) too high a plutonium feed rate, ii) failure of the magnetic stirrer, iii) failure of the deflection plate, or iv) failure of moderation control, resulting in a more cohesive mix. Calculations show there are sufficient controls such that homogeneity is not necessary to ensure subcriticality. However, criticality could occur if clumping were followed by a loss of moderation control.

2. DETERMINATION OF LIKELIHOOD CATEGORY IN Table A-3

The likelihood category is determined by calculating the likelihood index, T, then using this table. The term T is calculated as the sum of the indices for the events in the accident sequence.

3. DETERMINATION OF FAILURE FREQUENCY INDEX NUMBERS IN Table A-4

Table A-4 is used to assign frequency index numbers to facility initiating events and control system failures as found in the columns of Table A-9. The term failure must be understood to mean not merely failure of the control device or procedure, but also as violation of the safety limit by the process. In the example in Table A-9, accident sequence 2SX-001 involves loss of volume control due to pump failure. If criticality is the concern, failure does not occur unless an unsafe volume of uranium-oil mixture collects in the oil reservoir before the leak is stopped. For radiological consequences, any amount leaked may cause exposure. In assessing the frequency index, this factor should be considered because many control failures do not cause safety limits to be exceeded.

Table A-4 provides two columns with two sets of criteria for assigning an index value, one based on type of control, the other directly on observed failure frequencies. The types of controls are administrative, active engineered, passive engineered, etc. Since controls of a given type have a wide range of failure frequencies, assignment of index values based on this table should be done with caution. Due consideration should be given as to whether the control will actually achieve the corresponding failure frequency in the next column. Based on operational experience, more refined criteria for judging failure frequencies may be developed by an individual applicant. In the column labeled "Based on Type of Control," references to redundancy allow for controls that may themselves have internal redundancy to achieve a necessary level of reliability.

Another objective basis for assignment of an index value is actual observations of failure events. These actual events may have occurred in a comparable process elsewhere or in the licensed facility. Justification for specific assignments may be noted in the Comments column of Table A-9.

As previously noted, the definition of failure of a safety control to be used in assigning indices is, for non-redundant controls, a failure severe enough to cause an accident with consequences. For redundant controls, it is a failure such that, if no credit is taken for functionality of the other control, an accident with consequences would result. If most control malfunctions would qualify as such failures, then the index assignments of this table are appropriate. If true failure is substantially less frequent, then credit should be taken and adequate justification provided.

Note that indices less than (more negative than) "-1" should not be assigned to controls unless the configuration management, auditing, and other required management

measures are of high quality, because, without these measures, the controls may be changed or inadequately maintained. The reviewer should be able to determine this from a tabular summary of safety controls provided in the application. This summary should include identification of the process parameters to be controlled and their safety limits and a thorough description of the control and its applied management measures.

4. DETERMINATION OF FAILURE PROBABILITY INDEX NUMBERS IN Table A-5

Occasionally, information concerning the reliability of a safety control may be available as a probability on demand. That is, a history may exist of tests or incidents where the system in question is demanded to function. To quantify such accident sequences, the applicant must know the demand frequency, the initiating event, and the demand failure probability of the safety control. This table provides an assignment of index numbers for such controls in a way that is consistent with Table A-4. The probability of failure on demand may be the likelihood that it is in a failed state when demanded (availability), or that it fails to remain functional for a sufficient time to complete its mission.

5. DETERMINATION OF FAILURE DURATION INDEX NUMBERS IN Table A-6

The failure duration index is important because of reasons discussed above–it represents the window of opportunity after failure of the first preventive control during which the failure of the second could lead to adverse consequences. Cases in which the loss of the first control may remain undetected for long periods of time (such as leakage of hidden or baffled piping when credited as primary containment, or failure of items only tested when challenged) will typically not credit the failure duration in reducing the probability of the accident scenario. In this case, the duration index *D* should be taken as 0. Duration indices less than -1 should be based on periodic surveillance and/or maintenance periods, or the fact that failure would be readily apparent within a certain time frame. For example, a failure duration index of -2 would be based on the fact that a weekly surveillance requirement has been established for that item. A failure duration index of -3 may be based on a requirement to perform a certain measurement once per shift, or the fact that failure would immediately reveal itself to operators who are required to be continually present.

Table A-8: Criticality Safety Limits and Controls

IROFS for the Second Pass Solvent Extraction system

IROFS Identifier	Parameters and Limits	IROFS Description	Management Measures	QA Grade
2SX-PE1	VOLUME: <4.5 L (1.2 gal)	SX Pump PMPX-001 has a safe volume chamber.	Configuration control	B
2SX-AE1	GEOMETRY: < 7.6 cm (3") depth	SX Pump PMPX-001 has an active level switch on the oil reservoir, which automatically shuts the recirculation valve and sounds an audible alarm in the control room if the slab depth is exceeded.	1. Configuration control 2. Control room constantly monitored. 3. Biweekly functional check.	A
2SX-AE2	PROCESS VAR: ΔP < 0.34 atm (5 psi)	Pressure differential gauge on heat exchanger HX-001 is set to alarm if $P_{tube} - P_{shell} < 0.34$ atm (5 psi).	1. Configuration control. 2. Control room constantly monitored. 3. Functional check each shift.	A
2SX-AE3	PROCESS VAR: not applicable.	Conductivity probe on heat exchanger shell side to detect intrusion of plutonium. Set point will be sufficient to detect a concentration of 0.1 GPU/l.	1. Configuration control. 2. Functional test weekly.	A
2SX-ADM1	PROCESS VAR: ΔP < -0.34 atm (-5psi)	Procedures require operator response to differential pressure gauge alarm.	1. SOP 5349 2. Training/postings.	B
2SX-PE2	MASS: 0 mass in nitric acid supply	Siphon break installed in nitric acid supply line.	1. Configuration control.	C
2SX-ADM2	MASS: 0 mass in nitric acid supply	Utility (in this case, nitric acid) supply gauges are continually monitored in the control room whenever fissionable material is being processed. Facility procedures require shut down when utility pressure lost.	1. SOP 9483 2. Training/postings.	B
2SX-PE3	MASS: 0 mass in DIW supply	Siphon break installed on Deionize Water (DIW) line.	1. Configuration control.	C
2SX-ADM3	MASS: 0 mass in DIW supply	Utility (in this case, DIW) supply gauges are continually monitored in the control room whenever fissionable material is being processed. Facility procedures require shut down when utility pressure lost.	1. SOP 6879 2. Training/postings.	A
2SX-ADM4	PROCESS VAR: Acid molarity —.	DIW must be added to reduce acid molarity in the strip column to <—M. This ensures the plutonium will be stripped back into the aqueous phase.	1. SOP 0292 2. Training/postings.	A
2SX-ADM5	CONCENTR: < 0.1 GPU/L in the solvent regeneration columns	Procedures require weekly check of solvent regeneration columns by dual independent sampling. In addition, at the start of each batch, a checklist requires operators to visually check the columns for observed plutonium intrusion (greenish color).	1. SOP1929 2. Training/postings 3. QA Lab procedure ensures independ.	B
2SX-PE4	MASS: 0 mass in bulk chemical supply	Backflow preventer (BFP) installed on bulk chemical and DIW lines to prevent backlog to organic solvent supply tanks.	1. Configuration control. 2. Annual surveillance.	B
2SX-PE5	GEOMETRY: diam < 10.2 cm (4")	Columns must be composed on no greater than 10.2 cm (4") diameter glass (extraction, scrub, strip, and precipitation).	1. Configuration control.	C
2SX-PE6	GEOMETRY: depth < 5.2 cm (2") Area > 4.65 ^{M2} (50 ft^2)	Catch pans beneath columns must be no more than 5.2 cm (2") deep. In addition, they must have an area of 4.65 ^{M2} (50 ft^2) or more to ensure that they are capable of handling the largest spill from the columns.	1. Configuration control.	C

IROFS Identifier	Parameters and Limits	IROFS Description	Management Measures	QA Grade
2SX-PE7	SPACING: columns > 61 cm (24") center-to-center	Drawings require columns be installed no more than 61 cm (24") center-to-center (c-to-c).	1. Configuration control.	C
2SX-ADM6	MODERATION: water not allowed in fighting fires	Facility emergency response procedures prohibit the use of water in fighting fires in the solvent extraction area, when plutonium is being processes. There is no automatic sprinkler system in this area. Foams and fogging agents may be used.	1. Emergency Plan. 2. Training/postings. 3. Annual drill. 4. Configuration control.	C
2SX-PE8	MODERATION: no overhead lines in SX area	Overhead water lines are prohibited in the solvent extraction area.	1. Configuration control.	C
2SX-PE9	GEOMETRY: width < 7.6 cm (3")	Width of the solvent regeneration M/Ss must be less than 7.6 cm (3").	1. Configuration control.	C
2SX-PE10	GEOMETRY: diameter < 7.6 cm (3")	Diameter of the floor drains must be less than 7.6 cm (3").	1. Configuration control.	C
2SX-PE11	GEOMETRY: depth < 2.54 cm (1")	Floor must be sloped to drain into the favorable geometry floor drains; variation in floor level must not allow solution more than 2.54 cm (1") deep to accumulate.	1. Configuration control. 2. Annual audit.	C
2SX-AE4	CONCEPT: < 0.015 GPU/L	In-line monitor interlocked to isolation valve, to terminate feed if concentration > limit. Safety grade items are the monitor, the interlock electronics, and the isolation valve.	1. Weekly calibration and functional source check. 2. Configuration control.	A
2SX-ADM7	CONCEPT: < 0.015 GPU/L	Dual independent samples must be drawn and confirmed before transfer of refined to the waste water tanks is permitted. The results of sampling must be reviewed by the operator and a supervisor (who maintains control of the key to the valve lock).	1. SOP 9045 2. QA Lab procedure 3. Training/postings	A
2SX-PE12	SPACING: columns > 61 cm (24") c-to-c	Structural supports must be designed to withstand credible loads with a safety factor ≥ 2. Must be designed to withstand seismic loads > —g.	1. Pre-startup load testing. 2. Configuration control.	C
2SX-ADM8	CONCEPT: < 0.015 GPU/L	Excess nitric added in extraction and scrub columns sufficient to maintain a pH of –. Needed to keep refined concentration at a sufficiently low level.	1. SOP 3934 2. Lab QA procedure	B
2SX-ADM9	CONCEPT: < 0.015 GPU/L	Concentration in second pass extraction limited to —GPU/L. Along with 2SX-ADM8, needed to ensure extraction efficiency to keep refined concentration sufficiently low.	1. SOP 0945 2. Lab QA procedure	B
2SX-ADM10	SPACING: containers > 30.5 cm (12") from columns	Facility procedures require that fissile material contains and portable equipment be maintained at least 30.5 cm (12") from columns and pumps. Reinforced by postings and blue lines painted on floor (Limited Movement Areas).	1. Supervisor walk-through shiftly. 2. Facility directive 07. 3. Training/postings.	C
2SX-ADM11	MAT'L FORM: oil < 4L (1.1 gal)	The amount of oil in the oil reservoir of any pump shall be limited to 4L (1.1 gal). This limits the concentration of hydrogenous moderators other than water to ensure subclinical calculations are bounding.	1. Configuration control.	C
2SX-ADM12	MAT'L FORM: no precipitating agents	Lids to bulk chemical supply tanks must be locked and controlled by supervisors, to ensure against the inadvertent addition of precipitating agents. Addition of all reagents must be certified by a facility chemical engineer prior to fissionable material processing.	1. Facility directive 29. 2. Training/postings.	C

IROFS Identifier	Parameters and Limits	IROFS Description	Management Measures	QA Grade
2SX-PE13	MAT'L FORM: no precipitating agents	BFP installed on the line connecting the precipitation columns and the second pass evaporator. This prevents the backlog of oxalic acid into the SX operation.	1. Configuration control. 2. Annual surveill.	B

IROFS Identifier: cross-referenced with Preventive Controls in Table A-9. Parameters and Limits: describe actual parameter limits, and all attributes of the IROFS that are important to criticality safety. Management Measures: These are the measures needed to ensure IROFS availability and reliability. QA Grade: This is optional – all controls may be classified as Grade-A. If there is a graded QA Program, this signifies not the relative safety-significance of the control, but the degree of management attention needed once the item is installed to ensure its availability and reliability (*e.g.*, the siphon break is Grade-C, not because its failure is of minor NCS significance, but because once installed it requires essentially no maintenance.)

Note: Engineered features such as alarms and instrumentation needed to trigger an administrative response should be categorized as separate IROFS from the administrative controls; these design features are required to be maintained as IROFS.

6. DETERMINING MANAGEMENT MEASURES FOR SAFETY CONTROLS

Table A-8 is an acceptable way of listing those IROFS in all the accident sequences leading to consequences of concern. The IROFS listed should include all safety controls and all external events whose low likelihood is relied upon to meet the performance requirements of proposed 10 CFR 70.61. Staff reviews this list to determine whether measures have been applied to each safety control adequate to assure their continual availability and reliability in conformance to proposed 10 CFR 70.62(d). The types of management measures include maintenance, training, configuration management, audits and assessments, quality assurance, etc. These management measures are indicated in the Baseline Design Criteria (BDC) and described in greater detail in SRP Chapters 6.0 through 12.0 and 15.0. Safety controls meeting all the provisions of these chapters have acceptable management measures, that is, they comply with proposed §70.62(d). Safety controls may, with justification, have lesser management measures than those described. However, every item relied on for safety in accident sequences leading to consequence categories 2 or 3 should be assigned at least a minimal set of management measures. Specifically, in order to defend against common mode failure of all controls on a process, this minimal set of measures must include an adequate degree of: (a) configuration management, (b) regular auditing for the continued effectiveness of the control, (c) adequate labeling, training, or written procedures to assure the awareness of the operating staff of the safety function performed, (d) surveillance and corrective maintenance, and (e) preventive maintenance, if applicable.

If lesser or graded management measures are applied to some controls, Tables A-8 and A-9 and the narratives preceding them, in order to be acceptable, must identify to

which controls these lesser measures are applied. In addition, information indicating that acceptable reliability can be achieved with these lesser measures must be presented. It is not necessary that the specifics of these measures, such as the surveillance interval, type of maintenance, or type of testing, be described as applied to each control. It is recognized that such specific measures must be applied differently to each control to whatever degree is necessary to achieve adequate reliability. It is the formality, documentation, and QA requirements applied to these direct management measures that may be graded generically in a risk-informed manner.

The following describes the application of management measures to IROFS based on the risk importance of the item in an accident sequence, as defined by (1) the risk index, and (2) the failure likelihood index, "T." In summary, items relied on to prevent or mitigate accidents with consequences in the two highest categories identified in proposed §70.61 should satisfy the applicable B.C. of proposed §70.64.

For each of the accident sequences evaluated in Table A-9 as being in an acceptable risk category (a risk index of less than or equal to 4):

(1) If the initiating event is not a control failure, then management measures for that event are not necessary. For sequences claimed to be highly unlikely or unlikely, the assessment that the initiating event has such a low frequency must be adequately justified in the application.

(2) Regardless of the degree to which this initiating event is relied on in the accident sequence, for accident sequences resulting in nuclear criticality, double contingency should still be established. This requires at least one more IROFS in the accident sequence, in addition to the initiating event, that requires management measures to ensure compliance with the double contingency principle.

(3) If the initiating event is a control failure, management measures for that IROFS should be applied sufficient to maintain the claimed failure frequency. The selection of management measures should take into consideration the failure likelihood assumed in finding the accident sequence risk acceptable, as well as the inherent nature of the control.

[Basis: If the required failure frequency index for a control with management measures applied (assumed in the accident sequence) is comparable to the failure index without management measures, such as for rigid dimensions of equipment not susceptible to changing, a relatively low level of management measures may be warranted.]

(4) If the initiating event is a control failure, management measures may be graded less than the highest level depending on the importance of the control to the overall risk of the accident sequence.

[Basis: If the unavailability of the IROFS makes a negligible increase in the overall risk, then that IROFS has a relatively low importance in the accident sequence. Assigning weights to the various IROFS in terms of management measures may be done by comparing the overall risk with and without (mitigated vs. unmitigated) that particular IROFS.]

7. RISK-INFORMED REVIEW OF SAFETY CONTROLS

The staff reviews the safety controls and external events listed in Table A-8 in a risk-informed manner as described in Section 5.5. The procedure for identifying systems of safety controls having higher risk significance is described in Section 5.5. These controls will be subject to a more detailed review by staff to assure their adequacy.

Table A-9: Example Accident Sequence Summary and Risk Index Assignment

Process: P^3 (Plutonium Purification Process) Node: 2SX (Second Pass Solvent Extraction)

Accident Sequence	Initiating Event (a)	Preventive Control 1 (b)	Preventive Control 2 (c)	Likelihood* Index T and Category C (d)	Conse-quence Category (e)	Risk Indices (f=d x e)	Comments & Recommendations
2SX-001	Pump chamber leaks	2SX-PE1: Pump chamber is safe volume F1 = -1. Regular maintenance prevents frequent pump failure. D1 = -3. Pump failure would be detected by oil presence in the solution in clear glass columns. Process continually monitored by operators.	2SX-AE1: Level switch keeps oil level at safe slab depth. Automatically actuates isolation valve and alarms if level exceeded. F2 = -2. Regular maintenance ensures low failure rate. D2 = -2. Biweekly surveillance.	T = -5 C = 1	3	3	
2SX-002	Heat exchanger tube leaks	2SX-AE2: Differential gauge alarms if pressure differential across evaporator not maintained. F1 = -1. Regular maintenance ensures low failure rate. D1 = -3. Failure would be detected during one shift because concentration monitored frequently for QA.	2SX-ADM1: Operator response required to respond to alarm if pressure differential lost. F2 = -2. Failure to evaporate would be noticed by operators on floor, and control room operator required by training and procedure to respond to alarm. Control room manned by two operators at all times. D2 = 0. Failure of this control may not be readily noted. Credit not taken.	T = -4 C = 2	3	6	This scenario requires other controls to ensure adequate low likelihood. Recommend installation of a drain line on the steam supply to prevent liquid accumulation.
2SX-003	Motive force causes potential backlog to nitric acid	2SX-PE2: Siphon break installed on supply line. F1 = -4. The most likely scenario is that the siphon break was never installed in the first place. There is a rigorous configuration control program for safety grade items. D1 = 0. All safety grade items audited annually to confirm their continued presence.	2SX-ADM2: Utility supply pressure not maintained above atmospheric. F2 = -2. Utilities used throughout facility for many different purposes. They are used frequently and so are tested on a continuel basis. D2 = -2. Control room continually manned; these process variables are monitored constantly for QA purposes.	T = -5 C = 1	3	3	
2SX-004	Motive force causes potential backlog to DIW	2SX-PE3: Siphon break installed on supply line. see 2SX-003 for explanation.	2SX-ADM3: Utility supply pressure not maintained above atmospheric. see 2SX-003 for explanation.	T = -5 C = 1	3	3	

Appendix A

Accident Sequence	Initiating Event (a)	Preventive Control 1 (b)	Preventive Control 2 (c)	Likelihood* Index T and Category C (d)	Conse-quence Category (e)	Risk Indices (f=d x e)	Comments & Recommendations
2SX-005	Concentrated plutonium not stripped from organic	2SX-ADM4: Sufficient DIW added to ensure acid molarity low enough to guarantee stripping. Must be sampled and checked before stripping. 2SX-ADM5: Solvent regeneration columns M/Ss monitored weekly for uranium build-up. F1 = -2. Process variables (acid molarity and concentration) monitored shiftly and monitored continuously for QA purposes, ensuring their reliability D1 = -3. Major process upset would be noted by operators almost immediately.	2SX-PE4: BFP on organic buil chemical supply line. F2 = -2. Regular maintenance ensures low failure rate. D2 = -2. Failure would be detected during weekly surveillance.	T = -8 C = 1	3	3	
2SX-006	Solution spills from column	2SX-PE5: Columns are favorable geometry glass. F1 = -1. Columns have capacity to break even though they are sealed within a steel scaffold; operational history shows that this is an infrequent occurrence. D1 = -3. Breakage would be readily apparent. The process floor is continually manned and good housekeeping practices are instituted.	2SX-PE6: Catch pans are safe slab, and have sufficient area to hold the contents of more than two columns when filled to the maximum. F2 = -4. For this control to fail would either require improper installation, or the breakage of several columns. Configuration management reliability is judged to be -4. D2 = -3. See Control 1 explanation.	T = -7 C = 1	3	3	
			...additional accident sequences would follow this...				
2SX-008	Earthquake occurs of sufficient strength to cause structural failure F = -5.	2SX-PE7: Columns separated at sufficient distance to ensure subcriticality. 2SX-PE5: Columns are favorable geometry. F1 = -3. If columns are subjected to extreme stresses they will tend to break rather than displace. Probability of displacing so that the columns would come to rest with their axes parallel is very low. D1 = -2. Several days is the longest that the condition would be likely to persist before control of the site was reestablished.	2SX-PE8: There are no water lines or other sources of water installed to burst in the event of an earthquake. F2 = -4. This is the standard frequency used elsewhere where a passive design feature that relies only on configuration management is used, when there are no other failure mechanisms. D2 = -2. The presence of water would be readily detected by responders following the earthquake.	T = -12 C = 1	3	3	This scenario takes credit for an external event. Site characteristics provide the likelihood of seismic activity and flood levels quoted.

*Likelihood index T is a sum. Uncontrolled: T=frqi or frq1; Controlled: includes all indices T=a+b+c+d

Note 1: For these sequences the initiating event is failure of one of the controls, hence the frequency is assigned under that control.

The final results column of Table A-9 gives the risk index for each accident sequence that was identified in the ISA. The risk index will be used by staff to identify all risk significant sets of controls. These sets of controls will be reviewed with greater scrutiny than controls established to prevent or mitigate accident sequences of low risk.

8. ACCIDENT SUMMARY AND RISK INDEX ASSIGNMENT FOR TABLE A-9

The definitions for the contents of each column in the accident summary tabulation, Table A-9, are provided below.

(1) Accident Sequence

This column is provided to list the accident sequences identified by the applicant in the ISA. It is important to the proper documentation of the ISA that the applicant subdivides the facility into a set of uniquely identified units, referred to here as "nodes". The applicant should give symbols, names, or numbers to these nodes that permit them to be uniquely identified. For example, the Plutonium Purification process described in Section A6 has the unique identifying symbol P^3. The specific node corresponding to second pass solvent extraction has the unique identified 2SX. Additional identifier characters have been added to form the identifier, 2SX-001, to identify the first accident sequence identified in that node. Because the applicant should list all the facility safety controls of significance used elsewhere in the ISA, tabulations of the unique node (and accident) identifier can be used to find the accidents that these safety controls have been shown to prevent. By reviewing this table, the reviewer can then evaluate (1) the adequacy of the controls for preventing accidents and (2) the bases for making the consequence and likelihood assignments in the table.

(2) Initiating Event or Control Failure

This column is provided to list initiating events or control failures, typically identified in the process hazard analysis phase of the ISA, that may lead to consequences of concern. Initiating events are of several distinct types: (1) external events, such as hurricanes and earthquakes, (2) facility events external to the node being analyzed (e.g., fires, explosions, failures of other equipment, flooding from facility water sources), (3) deviations from normal of the process in the node (i.e., credible abnormal events), and (4) failures of safety controls of the node. The tabulated initiating events should only consist of those that involve an actual or threatened failure of safety controls, or that cause a demand requiring controls to function in order to prevent consequences of concern. The frequency index number for initiating events is referred to in the table using the symbol *"F."* Table A-4 provides criteria for assigning a value to *F*. Usually, insufficient room is present in a tabular presentation like Table A-9 to describe accurately the events indicated. Consequently, the applicant should provide supplementary narrative information to adequately describe each accident sequence of Table A-7. Cross referencing

between this information and the table should be adequate; for instance, the unique symbolic accident sequence identifiers can be used. Table A-7 is an example of a list of supplementary accident sequence descriptions corresponding to Table A-9.

(3) Preventive Control 1

This column is provided to list a control designed to prevent consequences of concern. If separate controls are used to prevent different consequences, separate rows in the table should be defined corresponding to each type of consequence. Sequences where two controls must simultaneously be in a failed state require assignment of three index numbers: the failure frequency of the first control, F_1, the duration of this failure, D_1, and the failure frequency of the second control, F_2. For such sequences, the initiating event is failure of the first control. In these cases, F_1 is assigned using Table A-4. The failure duration of the first control is assigned using Table A-6. Other sequences may be more easily described as a failure of the safety controls on demand after the occurrence of an initiating event. In these cases, the failure probability index number, *prf1*, is assigned using Table A-5. The symbol "b" is used in the column heading for the indices associated with this control.

(4) Preventive Control 2

This column is provided in case a second preventive control exists. The failure frequency or failure probability on demand is assigned as for Preventive Control 1. The symbol "c" is used in the column heading for the indices associated with this control.

In cases where no second preventive control exists (especially when the B.C. require double contingency), this column should contain a description of the consequences resulting from the first control failure. For example, there are generally two ways to demonstrate double contingency – either by I) specifying a second independent control that has to fail concurrently before criticality is credible, or ii) showing by calculation that the worst credible physical conditions resulting from the control failure remain subclinical. References identifying the consequence calculations that relate to the accident sequences should be included somewhere in the table, such as in column "c" or "e."

(5) Likelihood Category

This column is provided to list the likelihood category number for the risk matrix, which is based on the total likelihood index for a sequence. The total likelihood index, T, is the sum of the indices for those events that comprise a sequence. These events normally consist of the initiating event, and failure of one or more controls, including any failure duration indices. However, accident sequences may

consist of varying numbers and types of undesired events. Methods for deciding what frequencies and failure durations need to be considered will be described later in this appendix. Based on the sum of these indices, the likelihood category number for the risk matrix is assigned using Table A-3. The symbol "d" is used for this category number in the column heading.

(6) Consequence Category

This column is provided to assign the consequence category numbers based on estimating the consequences of all types (i.e., radiological, criticality, chemical, and environmental) that may occur. Based on this estimate, accidents can be assigned to the categories defined in proposed 10 CFR 70.61. The symbol "e" is used for this category number in the column heading.

(7) Risk Index

This column is provided to list the risk index, which is calculated as the product of the likelihood category and consequence category numbers. This is shown in the column heading by the formula "f = d x e." Sequences with values of "f" less than or equal to "4" are acceptable. The risk index can be calculated as the product of the consequence category with the failure index of the first preventive control, giving a measure of the "unmitigated" risk, in the case where the second control is not available to perform its function. This is a way to assess the risk significance of the second control.

For sequences in which there is no second control specified, the unmitigated risk may be used to demonstrate an acceptable risk category. There may, however, be cases in which this is not possible; where there is a continuum of possible consequences resulting from occurrence of the accident sequence up to that point, credit may be taken for the unlikelihood of achieving an unacceptable physical state (*e.g.*, probability that the upset exceeds a subclinical mass). This will require a thorough, documented justification for the reviewer to find this approach acceptable.

(8) Comments and Recommendations

This column is needed to record ISA team recommendations, especially when the existing system of controls is evaluated as being deficient. This may happen because a newly identified accident sequence is not addressed by existing controls, or because a deficiency has been found in the existing controls.

9. ALTERNATE METHODS OF PRESENTATION (FAULT TREES)

Table A-9 displays the results of the ISA Summary in a tabular format by accident sequence. This approach is commonly developed from a What-If hazard evaluation

technique, which is but one of several methods available. The methods that may be used include What-If, HazOp, Failure Modes and Effects Analysis (FMEA), Fault

Figure A-2: Fault Tree for SX Operation

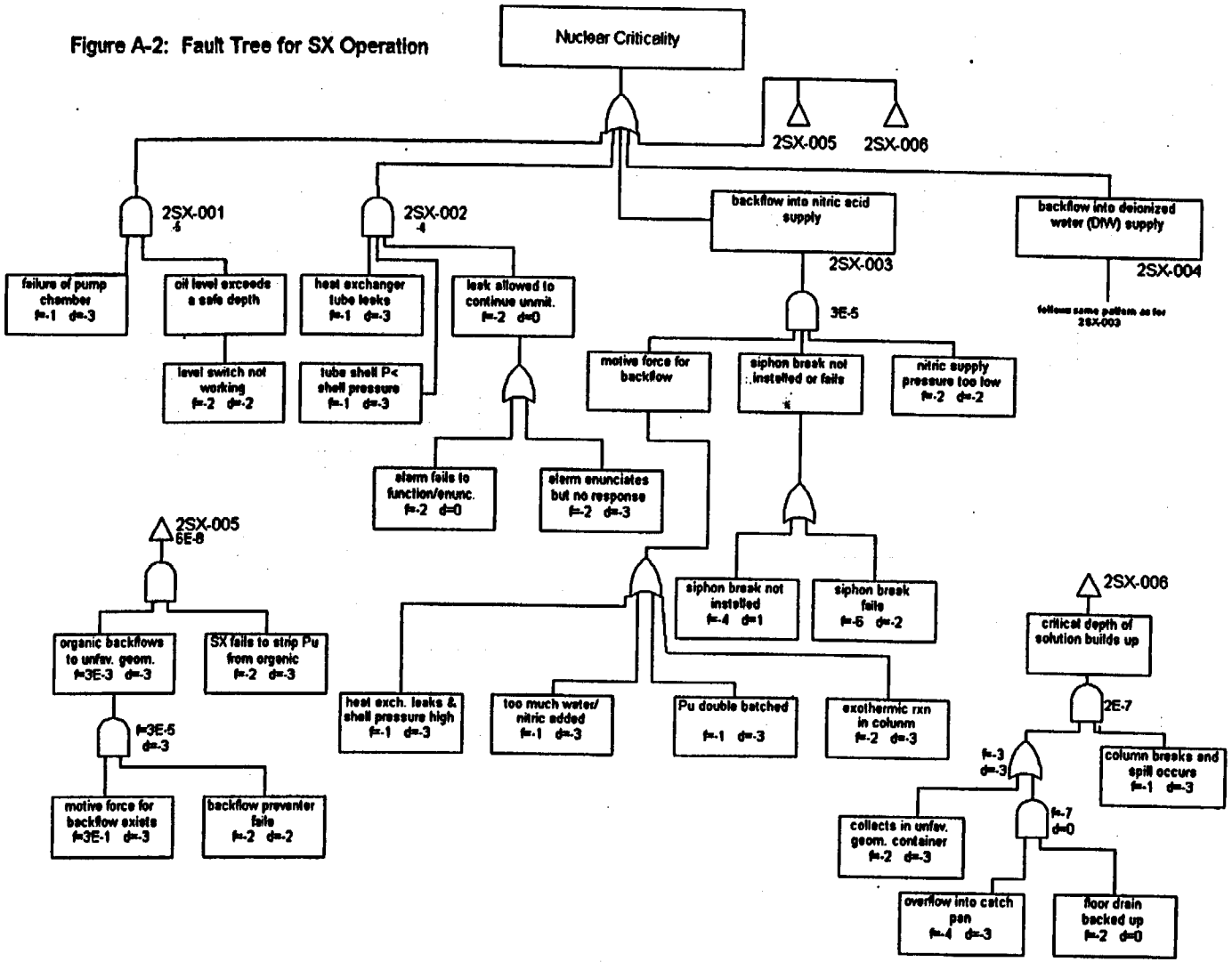
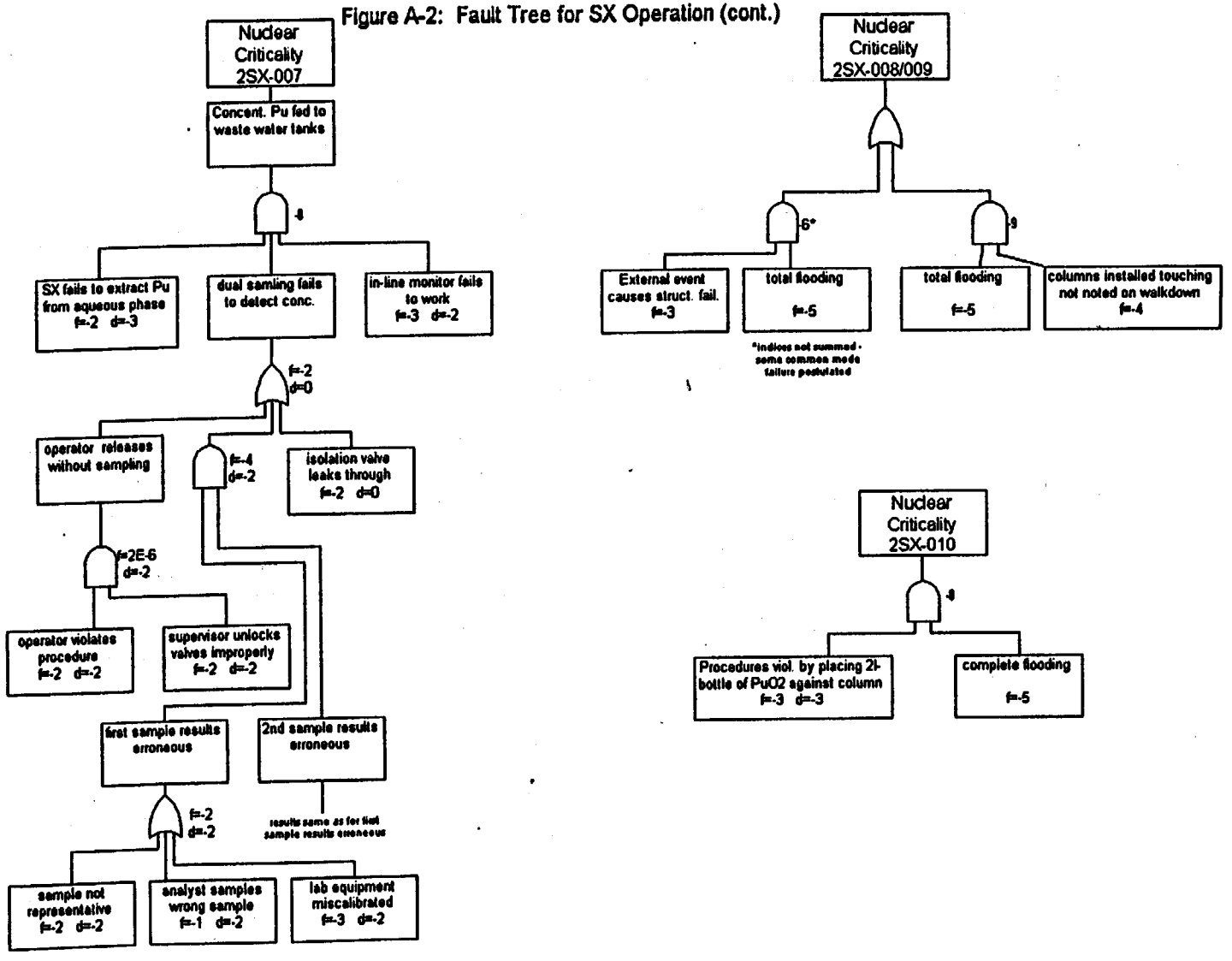


Figure A-2: Fault Tree for SX Operation (cont.)



Trees, and other methods. The What-If approach may be considered the least preferable of these approaches, particularly when there are a large number of accident sequences, as it is difficult to demonstrate completeness. That is, it will be difficult for the reviewer to verify that all credible accident sequences have been included in the hazard evaluation for a very complex process.

There are additionally reasons why a tabular format (Table A-9) may not be the best method of displaying the results of the ISA Summary for all processes. A variety of different techniques may be used rather than rigidly adhering to one format, if the multi-method approach enhances the clarity of the presented data. One of the main weaknesses of the tabular format is that it considers the accident sequence to consist of the failure of only two discrete controls. The establishment of double contingency may require more than two controls to ensure that at least two unlikely and concurrent upsets must occur before criticality is possible (and that the overall likelihood of criticality is highly unlikely). Several distinct controls may in general be combined into a single "control system." When grouped as shown in Figure A-2, there may be several distinct controls which must be grouped together to ensure each "leg" of double contingency is unlikely to fail. This definition of unlikely is in the context of the double contingency principle, which numerically is approximately $\leq 10^{-2}$, rather than the more restrictive value of $\leq 4 \times 10^{-4}$ as used in SRP Section 5.4.3.2(B)(ix). Use of this more conservative value would of course be acceptable, although it is highly doubtful whether many operations would be able to meet this without the virtual elimination of administrative criticality controls. Although this information may be presented in the table by listing multiple controls in each bin (*e.g.*, Scenarios 2SX-005 and -008), it would be more efficacious to use a fault tree (Figure A-2).

In addition, each accident sequence in the table considers the failure of the first control followed by failure of the second. Therefore there are actually two complementary accident sequences that must be considered in different rows of the table. This particular aspect of the logic – and the general logical flow of the accident as it unfolds–is masked by using an approach that follows a simple linear development of the sequence from initiating event to completion. In addition, the What-If approach often does not consider the control failure at a sufficiently high level. The answer to the question "*What if the pump chamber leaks into the oil reservoir?*" is often "*The pump cannot leak because....*" Considering the next to the top level event in the tree to be the loss of volume control ensures that the system will remain adequately subclinical even in the event that the pump failure occurs

The advantages of using a fault tree to present the ISA Summary data include: (1) the top-down approach of a fault tree (as opposed to the bottom-up approach of What-If) ensures that all credible changes in process conditions – or loss of controlled parameters–are considered; (2) robustness is ensured by considering the control failures at a sufficiently high level; and (3) the logical sequence of events that must occur to cause a criticality cannot be described thoroughly using the tabular approach. Figure A-2 shows a fault tree for the accident sequences that are described in

Table A-9; a cursory review demonstrates that these diagrams present a much higher level of information than is contained in Table A-9. For example, in order to have a criticality due to leakage through the heat exchanger into the steam supply, the following events would have occur: (1) the heat exchanger tubes would have to leak; and (2) the pressure on the tube side would have to drop below the pressure on the shell side; and (3) the leak would have to continue without being noticed, either by failure of the alarm to enunciate (mechanical) or failure of the operator to take appropriate actions (human error). This combination of events is then required to ensure that the overall consequence–getting concentrated uranium solution into the unfavorable geometry steam supply–is sufficiently unlikely (in fact, other controls are then recommended to reduce the likelihood index below -4). In addition, one can see that the loss of integrity of the evaporator tubes and loss of steam pressure are comparable events, and that reducing the frequency of mechanical failure of these items or the duration of alarm failure would result in the greatest drop in overall likelihood. In the event that other controls are credited in this scenario as a result of the recommendation, it would be difficult to convey the full amount of all the above information in the table.

A7. MIXED OXIDE BLENDING OPERATION

Oxide blending is a process whereby dry UO_2 and PuO_2 powder is combined to produce a homogeneous blend suitable for fabrication into mixed oxide fuel pellets and assemblies. The final mix consists of 20wt% PuO_2 (isotopically, ~96% ^{239}Pu and ~4wt% ^{240}Pu) and 80wt% U(0.7wt%)O_2. Process equipment downstream of the blending operation is designed with subclinical dimensions for 30wt% PuO_2 MOX. The main criticality controls in the blending operation are mass, moderation, and plutonium "enrichment" (defined for the purpose of this example as the weight percent of PuO_2 relative to the PuO_2-UO_2 blend).

A batch of UO_2 blendstock (~112 kg [246 lb]) is measured out into a favorable geometry feed hopper attached to a safety-grade scale. The material in the hopper is weighed and sampled for moisture, after which it is gravity fed into the favorable geometry blending hopper. This is a conical-bottom hopper which gravity drains into the cylindrical homogenizer. The low feed rate of the blendstock and plutonium oxide ensures that the powder attains a high degree of homogeneity as the two oxides are blended together. Homogeneity is not credited, however, for criticality safety until after the material passes through the homogenizer. In addition to ensuring criticality safety, moisture control is important to ensure that the powder will flow smoothly to ensure proper transfer and mixing.

PuO_2 powder is emptied from the 2-liter (0.45 gal) bottles into a plutonium oxide feed hopper through a hole in the bottom of a glovebox. This hopper is a 4-inch (10.2 cm) diameter cylindrical stainless steel vessel, which is heated to 150 °C (302 °F) to drive off residual moisture that may have accumulated. Several containers are emptied into the hopper until a mass of ~28 kg (61.6 lb) is reached. The powder is sampled and then gravity fed down a chute into the blending hopper. The flow rate of the plutonium oxide powder is controlled

using a mass flow totalizer (MFT), which is interlocked to the plutonium feed valve; the feed rate is maintained at a slow rate using a stopcock on the input line. If the preset mass of plutonium oxide is exceeded, the MFT shuts the valve and prevents the overall plutonium "enrichment" in the blender from exceeding the safety limit of 22wt% PuO_2. After blending, the material is agitated for 30 minutes before being sampled; only after two independent samples confirm the correct "enrichment" may the material be transferred to the cylindrical homogenizer for further processing. Following this, the *master mix* is ball-milled and sieved to ensure proper consistency before being combined with additional $U(0.7wt\%)O_2$, which results in a *final mix* of ~4wt% Pu.

Table A-10 presents the main accident sequences in the oxide blending operation. Table A-11 shows the IROFS credited for double contingency during oxide blending. Table A-12 shows the main accident sequences and the preventive controls used.

A table of accident sequences is used to communicate the ISA Summary information for the oxide blending operation; this operation is a simpler process from a criticality safety standpoint than the solvent extraction. Since the double contingency logic is based on a relatively simple set of controls on moderation, mass, and plutonium isotonic, this system is much more amenable to a tabular approach. Fault trees could be used profitably for this system, but there is a much lower level of complexity than in the first example, and tables may be adequately used. Several tables will in general be needed to summarize the information that must be presented; these should be cross-referenced to allow clear traceability of the control logic. The contents of the tables and figure for the oxide blending process are summarized below:

1. PROCESS CRITICALITY FLOW DIAGRAM IN FIGURE A-3

   This process is inherently much simpler than the solvent extraction example considered above, from the standpoint of criticality safety. Note that in this case, the entire operation is conducted in favorable geometry equipment, so that there is no attempt to distinguish between favorable and unfavorable process steps graphically. One should note that labels have been attached to each piece of equipment relied on for safety, so that this diagram may be cross-referenced with the tables. Each component relied on for safety must be identified for incorporation into the configuration management program. For example, not only the mass flow totalizer, but also the interlock back to the PuO_2 supply valve, and the valve itself, must be controlled to ensure that the active feature that prevents too high a plutonium "enrichment" in the blend hopper remains available and reliable to perform its function.

2. ACCIDENT SEQUENCES IN Table A-10

   By displaying the accident sequences in the manner shown, it is immediately apparent that the criticality controls on the process are mass, moderation, and plutonium isotonic. Each of the accident sequences describes the initiating event and presents such information as the controls that prevent the loss of that controlled parameter, the

safety significance of the initiating event, the probable cause, and so forth. The information should be succinctly provided in the ISA Summary to immediately put the accident sequences into the proper viewpoint.

3. ITEMS RELIED ON FOR DOUBLE CONTINGENCY IN TABLE A-11

Figure A-3 shows how a criticality flow diagram may be used effectively to summarize the contents of Table A-11. Please see Section A6 for a fuller discussion of this type of table.

4. SUMMARY OF ACCIDENT SCENARIOS AND RISK EVALUATION IN TABLE A-12

Note that Scenario MOB-001 has a consequence category of 0 (no consequences) instead of 3 (for criticality). This is not actually needed, because the likelihood index is sufficiently low based on the two preventive controls. However, this was done for illustrative purposes. As described in first entry in Table A-10, the loss of mass control due to the failure of both of these preventive controls cannot lead to criticality without a concurrent loss of moderation control. This should be documented in criticality calculations which would be referenced in the table. This is an acceptable way to treat accident scenarios where there is sufficient defense-in-depth that criticality cannot be achieved without the occurrence of additional events. In other words, the accident sequence defined by the failure of two preventive controls does not result in a criticality.

Scenarios MOB-006a and -006b (and MOB-010a and -010b) represent cases in which a single initiating event may occur due to two different causes. Generally deeper level events than the initiating event are not treated, but in this case it made sense to separate the sequences MOB-006 and -010 into more than one sub-sequence because different controls are needed for each pathway. Accident scenarios should be considered separate sequences if the controls relied on for safety are different, if the consequences are different (two scenarios leading to loss of mass control may result in different physical amounts and configurations), or likelihoods are different. Two accident sequences may have the same initiating events and the same consequences but different intermediate conditions or steps.

Figure A-3: Oxide Blending Operation

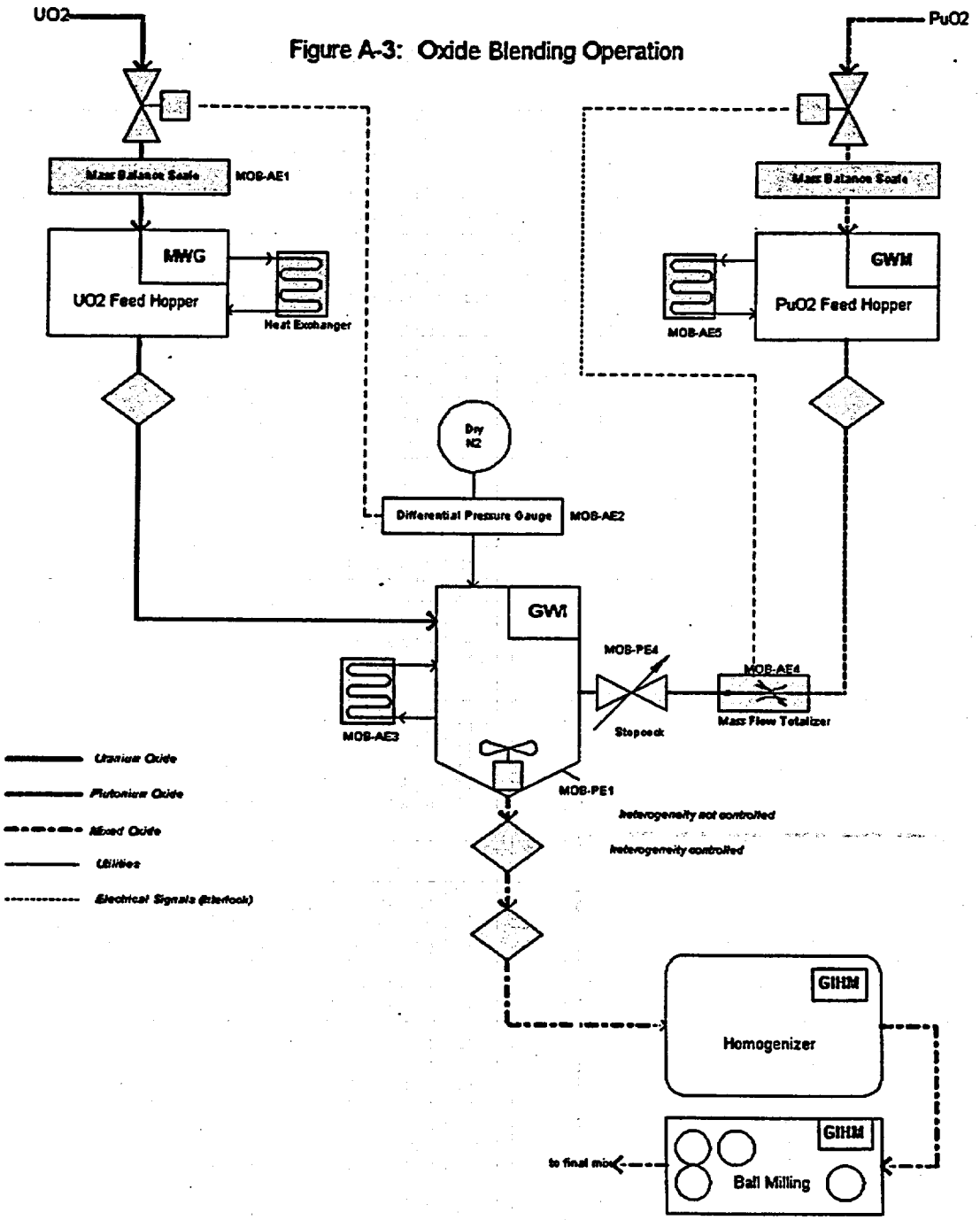


Table A-10: Accident Sequence Descriptions

Accident Sequence	DESCRIPTION
Loss of MASS	
MOB-001	The initiating event is exceeding the mass limit of the blend tank, by adding too much UO2 blendstock. This will have the effect of increasing the overall mass present, but will simultaneously decrease the plutonium "enrichment." The overall effect of this is to increase the distance from the subclinical curve of mass as a function of plutonium "enrichment" as more blendstock is added. The system is adequately subclinical under conditions of double batching uranium. To achieve criticality, this would have to be followed by a loss of moderation control.
MOB-002	The initiating event is exceeding the mass limit of the blend tank, by adding too much PuO2. This will have the effect of increasing both the overall mass and plutonium "enrichment." At ~33 kg PuO2 (72.6 lb) (and 23 wt%) the subclinical mass limit will be exceeded. Therefore, this could lead to criticality without any additional upsets and therefore dual controls are established on the plutonium mass.
MOB-003	The initiating event is exceeding the mass limit of the blend tank, by performing the blending operation while there is still blended oxide present from the previous batch in the tank. Assuming the previous batch was properly mixed, it would require an additional 50 kg (110 lb) of PuO2+UO2 to exceed the subclinical mass limit. Therefore this could lead to criticality without any additional upsets and therefore dual controls are established to ensure the blend tank is empty of material before another batch is started.
Loss of MODERATION	
MOB-004	The initiating event is exceeding the moderation limit (1 wt% H2O) by adding UO2 which has not been properly sampled. This could lead to criticality without any additional failures. Dual independent sampling is required to ensure moisture limits are adhered to. Also, material will not freely flow through orifice if wet.
MOB-005	The initiating event is exceeding the moderation limit (1 wt% H2O) by adding PuO2 which has not been properly sampled. This could lead to criticality without any additional failures. Dual independent sampling is required to ensure moisture limits are adhered to. Also, material will not freely flow through orifice if wet. In addition, both the plutonium feed hopper and blend hopper are heated. Material is added at a sufficiently slow rate that contact with the heated blendstock will cause moisture in the plutonium to be driven off.
MOB-006	The initiating event is exceeding the moderation limit (1 wt% H2O) by introduction of liquid water from overhead water lines or roof leaks. The blend tank is completely enclosed within an airtight and watertight enclosure. There are no overhead water lines allowed. The most likely cause of this scenario is backlog of condensate from the ventilation header, which serves to remove evolved water from the heated material. The ventilation header is sloped and equipped with drain lines to ensure against condensate backlog. Even in the event of water intrusion, the heating is sufficient to drive off any realistic accumulation of liquid water.
Loss of PLUTONIUM "ENRICHMENT"	
MOB-007	The initiating event is exceeding the plutonium "enrichment" by adding too little blendstock to the blending hopper. This will have the effect of increasing plutonium "enrichment" while decreasing the overall mass. This will eventually reach criticality without any additional failures, but only when more than half the original UO2 blendstock is omitted.
MOB-008	The initiating event is exceeding the plutonium "enrichment" by adding too much PuO2 feed to the blending hopper. This is identical to Scenario MOB-002 and will be discussed as a loss of mass control.
MOB-009	The initiating event is exceeding the plutonium "enrichment" by adding PuO2 to the blending hopper without first adding blendstock. This is the bounding case of Scenario MOB-007. Controls are established to ensure that blendstock is added and in the correct proportion before addition of PuO2 feed is allowed.
MOB-010	The initiating event is exceeding the plutonium "enrichment" by the formation of clumps of higher enrichment PuO2 in the blending hopper. This can be caused by i) too high a plutonium feed rate, ii) failure of the magnetic stirrer, iii) failure of the deflection plate, or iv) failure of moderation control, resulting in a more cohesive mix. Calculations show there are sufficient controls such that homogeneity is not necessary to ensure subcriticality. However, criticality could occur if this were followed by a loss of moderation control.

Table A-11: Criticality Safety Limits and Controls

IROFS Identifier	Parameters and Limits	IROFS Description	Management Measures	QA Grade
MOB-ADM1	MASS: UO_2 feed ~112 kg (246 lb)	Procedures, training, and postings require that the mass be checked and certified by an operator and supervisor prior to PuO_2 feed allowed.	1. Procedures and training. 2. Operator/supervisor must sign material balance sheets.	NA
MOB-AE1	MASS: UO_2 feed ~112 kg (246 lb)	Safety grade scale attached to feed hopper.	1. Weekly calibration using mass standards. 2. Tare weight re-certified whenever hopper is emptied.	A
MOB-PE1	MODERATION: Blend hopper is limited to 1wt% H_2O.	Blend tank comprises a welded stainless-steel barrier. Blending required to be under dry nitrogen atmosphere.		
MOB-AE2	MODERATION: Blend hopper is limited to 1wt% H_2O.	Blending required to be under a dry nitrogen atmosphere. IROFS is an differential pressure gauge interlocked to the feed supply valves and system alarm.	1. Monthly functional test. 2. Configuration control.	B
MOB-AE3	MODERATION: Blend hopper is limited to 1wt% H_2O.	Electric heater maintains powder at 150 °C (302 °F) in blend hopper. Low-T gauge and alarm interlocked to supply valves.	1. Weekly functional test. 2. Configuration control.	A
MOB-AE4	MASS: PuO_2 feed ~28 kg (61.6 lb)	Mass flow totalizer (MFT) interlocked to PuO_2 supply valve.	1. Configuration control. 2. Weekly function test.	A
MOB-PE2	GEOMETRY: diameter < 12.7 cm (5")	Diameter of oxide blender must be less than 12.7 cm (5").	Configuration control.	C
MOB-PE3	GEOMETRY: diameter < 10.2 cm (4")	Diameter of PuO_2 feed hopper must be less than10.2 cm (4").	Configuration control.	C
MOB-ADM2	MODERATION: Blend hopper is limited to 1 wt% H_2O	Procedures, postings, and training require the material in the UO_2 feed hopper to be sampled for moisture before it is released to the blending hopper. Supervisor concurrence required. Dual independent samples are required.	1. Procedures and training. 2. Lab QA procedures must be followed.	NA
MOB-AE5	MODERATION: Blend hopper is limited to 1 wt% H_2O	Electric heater maintains powder at 150 °C (302 °F) in PuO_2 feed hopper. Low-T gauge and alarm interlocked to supply valves.	1. Weekly functional test. 2. Configuration control.	A
MOB-ADM3	MODERATION: Blend hopper is limited to 1 wt% H_2O	Procedures, postings, and training require the material in the PuO_2 feed hopper to be sampled for moisture before it is released to the blending hopper. Supervisor concurrence required. Dual independent samples are required.	1. Procedures and training. 2. Lab QA procedures must be followed.	NA
MOB-ADM4	MASS: PuO_2 feed ~28 kg (61.6 lb)	Only a limited number of 2-liter (0.5 gal) bottles may be emptied into the PuO_2 feed hopper, such that the total mass does not exceed 28 kg (61.6 lb) as indicated on material balance sheets.	1. Procedures and training. 2. Material control program - the feed hopper is a process measurement node.	NA

IROFS Identifier	Parameters and Limits	IROFS Description	Management Measures	QA Grade
MOB-ADM5	MASS: Total blend hopper mass < 140 kg (308 lb)	Operators are required to check visually that the blend hopper is devoid of more than surface contamination after each campaign.	Procedures and training	NA
MOB-ADM6	MASS: Total blend hopper mass < 140 kg (308 lb)	Blend hopper must be NDA scanned after each campaign.	Procedures and training	NA
MOB-PE4	MODERATION: Blend hopper is limited to 1wt% H_2O.	Stopcock on PuO_2 feed line controls feed rate to 800 g/hr (1.8 lb/hr). This slow flow rate ensures that any moisture will be driven off on contact with the heated blendstock.	1. Flow rate checked during run by monitoring MFT. 2. Configuration control.	B
VEN-PE13	MODERATION: Blend hopper is limited to 1wt% H_2O.	Ventilation header must be sloped away from the blend hopper.	Configuration control.	C
VEN-PE15	MODERATION: Blend hopper is limited to 1wt% H_2O.	Ventilation header must be equipped with condensate drains at its lowest point, to prevent condensate backlog to the blend hopper.	1. Configuration control. 2. Periodic monitoring.	B
BLD16-65	MODERATION: Blend hopper is limited to 1wt% H_2O.	No overhead water lines are allowed in Building 16.	Configuration control.	C
MOB-ADM7	ISOTONIC: PuO_2 content 20wt%	Supervisor must check that UO_2 feed hopper is empty and that the appropriate mass has been added before PuO_2 transfer is authorized.	1. Procedures and training. 2. Material control program - the blendstock feed hopper is a process measurement node.	NA

Table A-12: Accident Sequence Summary and Risk Index Assignment

Accident Sequence	Initiating Event (a)	Preventive Control 1 (b)	Preventive Control 2 (c)	Likelihood* Index T and Category C (d)	Conse-quence Category (e)	Risk Indices (g=d x e)	Comments & Recommendations
MOB-001	Too much blendstock added.	MOB-ADM1: Blendstock mass certified before introduction of PuO_2. F1 = -3. Material control sensitivity ensures this receives appropriate attention and supervisor oversight. D1 = -2. Process is a batch process, campaign is running several days. Failure would be detected at start of subsequent campaign.	MOB-ADM7: Supervisor must ensure that the appropriate mass was emptied from blendstock feed hopper. F2 = -2. Required on checklist and reinforced by training and postings. D2 = -2. See MOB-001, Control 1.	T = -7 C = 1	0	0	Criticality not possible without an additional failure.
MOB-002	Too much PuO_2 added.	MOB-ADM4: No more than 26kg (61.6 lb) PuO_2 may be charged into the feed hopper. F1 = -3. Material control sensitivity ensures this receives appropriate attention and supervisor oversight. D1 = -2. See MOB-001, Control 1.	MOB-AE4: MFT limits total integrated PuO_2 which is transferred to blend hopper. F2 = -3. Reguter maintenance and testing ensures reliability. D2 = -2. Functionally tested weekly.	T = -8 C = 1	3	3	
MOB-003	Mixed oxide not cleaned out before next batch started.	MOB-ADM5: Visuel check that blend hopper empty before each campaign. F1 = -2. Required on checklist and reinforced by training and postings. D1 = -2. See MOB-001, Control 1.	MOB-ADM6: Blend hopper must be NDA scanned before each campaign. F2 = -2. Required on checklist and reinforced by training and postings. D2 = -2. See MOB-001, Control 1.	T = -5 C = 1	3	3	
MOB-004	Moderated blendstock added.	MOB-ADM2: Duel independent samples taken to confirm moisture level of blendstock. F1 = -3. Requires failure of two operators to follow procedures, and independence of sampling and lab analysis ensures reliability. D1 = -2. See MOB-001, Control 1.	MOB-AE3: Electric heater maintains temperature sufficient to drive off moisture in blend hopper. F2 =-4. Past history with this model of heater shows it to be very reliable. D2 = -2. Functionally tested weekly.	T = -8 C = 1	3	3	
MOB-005	Moderated PuO_2 added.	MOB-AE5: Electric heater maintains temperature sufficient to drive off moisture in feed hopper. F1 = -4. Past history with this model of heater shows it to be very reliable. D1 = -2. Functionally tested weekly.	MOB-AE3: Electric heater maintains temperature sufficient to drive of f moisture in blend hopper. F2 = -4. Past history with this model of heater shows it to be very reliable. D2 = -2. Functionally tested weekly.	T = -10 C = 1	3	3	

Accident Sequence	Initiating Event (a)	Preventive Control 1 (b)	Preventive Control 2 (c)	Likelihood* Index T and Category C (d)	Conse-quence Category (e)	Risk Indices (g=d x e)	Comments & Recommendations
MOB-006a	Water backlog from ventilation condensate.	VEN-PE13: Ventilation header sloped away from blend hopper. F1 = -3. The configuration control program requires installation according to design drawings and pre-startup verification. Several layers of management controls would have to fail to allow this to happen. D1 = 0. Would be checked during annual audit.	VEN-PE15: Ventilation header has condensate drains to prevent backlog. F2 = -3. The configuration control program requires installation according to design drawings and pre-startup verification. Several layers of management controls would have to fail to allow this to happen. D2 = 0. Would be checked during annual audit.	T = -6 C = 1	3	3	
MOB-006b	Water intrusion from external source.	MOB-PE1: Blend tank is watertight. F1 = -3. The ability of certified welders to ensure the integrity of welded vessels has been demonstrated. D1 = 0. Would be checked during annual audit.	MOB-AE2: Differential pressure gauge with interlock prevents introduction of feed if containment breached. F2 = -2. Based on past failure rate data when used in combination with HEPA filters. D2 = -1. Though sufficient to detect breach of the containment immediately (D=-5), its failure would be detected during monthly functional test. High demonstrated reliability means that D = -5 is actually more realistic.	T = -5 C = 1	3	3	
MOB-007	Too little blendstock added.	MOB-ADM1: Blendstock mass certified before introduction of PuO_2. F1 = -3. Material control sensitivity ensures this receives appropriate attention and supervisor oversight. D1 = -2. See MOB-001, Control 1.	MOB-ADM7: Supervisor must ensure that the appropriate mass was emptied from blendstock feed hopper. F2 = -2. Required on checklist and reinforced by training and postings. D2 = -2. See MOB-001, Control 1.	T = -7 C = 1	3	3	
MOB-008	same as MOB-002 (q.v.)						
MOB-009	PuO_2 added before blendstock.	MOB-ADM1: Blendstock mass certified before introduction of PuO_2. F1 = -3. Material control sensitivity ensures this receives appropriate attention and supervisor oversight. D1 = -2. See MOB-001, Control 1.	MOB-ADM7: Supervisor must ensure that the appropriate mass was emptied from blendstock feed hopper. F2 =-2. Required on checklist and reinforced by training and postings. D2 = -2. See MOB-001, Control 1.n	T = -7 C = 1	3	3	
MOB-010a	PuO_2 clump develops by: feed rate too high	MOB-PE4: Feed rate controlled by stopcock. F1 = -3. This is locked into place and tested before start-up. Has no moving or wear parts. D1 = -3. Failure would be detected during the course of one shift. Process is continually monitored by operators.		T = -6 C = 1	3	3	

Accident Sequence	Initiating Event (a)	Preventive Control 1 (b)	Preventive Control 2 (c)	Likelihood* Index T and Category C (d)	Conse-quence Category (e)	Risk Indices (g=d x e)	Comments & Recommendations
MOB-006a	Water backlog from ventilation condensate.	VEN-PE13: Ventilation header sloped away from blend hopper. F1 = -3. The configuration control program requires installation according to design drawings and pre-startup verification. Several layers of management controls would have to fail to allow this to happen. D1 = 0. Would be checked during annual audit.	VEN-PE15: Ventilation header has condensate drains to prevent backlog. F2 = -3. The configuration control program requires installation according to design drawings and pre-startup verification. Several layers of management controls would have to fail to allow this to happen. D2 = 0. Would be checked during annual audit.	T = -6 C = 1	3	3	
MOB-006b	Water intrusion from external source.	MOB-PE1: Blend tank is watertight. F1 = -3. The ability of certified welders to ensure the integrity of welded vessels has been demonstrated. D1 = 0. Would be checked during annual audit.	MOB-AE2: Differential pressure gauge with interlock prevents introduction of feed if containment breached. F2 = -2. Based on past failure rate data when used in combination with HEPA filters. D2 = -1. Though sufficient to detect breach of the containment immediately (D=-5), its failure would be detected during monthly functional test. High demonstrated reliability means that D = -5 is actually more realistic.	T = -5 C = 1	3	3	
MOB-007	Too little blendstock added.	MOB-ADM1: Blendstock mass certified before introduction of PuO_2. F1 = -3. Material control sensitivity ensures this receives appropriate attention and supervisor oversight. D1 = -2. See MOB-001, Control 1.	MOB-ADM7: Supervisor must ensure that the appropriate mass was emptied from blendstock feed hopper. F2 = -2. Required on checklist and reinforced by training and postings. D2 = -2. See MOB-001, Control 1.	T = -7 C = 1	3	3	
Accident Sequence	Initiating Event (a)	Preventive Control 1 (b)	Preventive Control 2 (c)	Likelihood* Index T and Category C (d)	Conse-quence Category (e)	Risk Indices (g=d x e)	Comments & Recommendations
MOB-010b	PuO_2 clump develops by: failure of moderation control	MOB-AE5: Electric heater maintains temperature sufficient to drive off moisture in feed hopper. F1 = -4. Past history with this model of heater shows it to be very reliable. D1 = -2. Functionally tested weekly.	MOB-PE4: Feed rate controlled by stopcock. F2 = -3. This is locked into place and tested before start-up. Has no moving or wear parts. D2 = -3. Failure would be detected during the course of one shift. Process is continually monitored by operators.	T = -9 C = 1	3	3	

*Likelihood index T is a sum. Uncontrolled: T=frqi or frq1; Controlled: includes all indices T=a+b+c+d

Note 1: For these sequences the initiating event is failure of one of the controls, hence the frequency is assigned under that control.

APPENDIX B
NATURAL PHENOMENA/OTHER EXTERNAL EVENTS

Natural phenomena events (i.e., earthquakes, high winds, tornadoes, tornado missiles, floods) and other external events (e.g., transportation accidents, airplane crash, industrial accidents, and fires external to the facility) should be addressed in the integrated safety analysis (ISA) as initiating events.

Currently, there are no Regulatory Guides (RGs) in Division 3, Fuels and Materials Facilities, addressing natural phenomena events and other external events. Therefore, the following sections of NUREG-0800, Standard Review Plan (SRP), and Division 1 RGs for power reactors, which describe methods for performing evaluations for natural phenomena events and other external events, should be consulted:

B1. Floods

- SRP 2.4.1, "Hydrologic Description;"
- SRP 2.4.2, "Floods;"
- SRP 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers;"
- SRP 3.4.1, "Flood Protection;"
- RG 1.59, "Design Basis Floods for Nuclear Power Plants;" and
- RG 1.102, "Flood Protection Plan for Nuclear Power Plants."

B2. Wind and Tornadoes

- SRP 3.3.1, "Wind Loadings;"
- SRP 3.3.2, "Tornado Loadings;"
- RG 1.76, "Design Basis Tornado for Nuclear Power Plants;" and
- RG 1.117, "Tornado Design Classification."

B3. Earthquakes

- SRP 3.7.1, "Seismic Design Parameters;"
- SRP 3.7.2, "Seismic System Analysis;"
- SRP 3.7.3, "Seismic Subsystem Analysis;"
- RG 1.60, "Design Basis Response Spectra for Seismic Design of Nuclear Power Plants;"
- RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis;"

- RG 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment and Components;"
- RG 1.161, "Damping Values for Seismic Design of Nuclear Power Plants;" and
- RG 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion."

B4. Other External Events

- SRP 2.2.1 - 2.2.2, "Identification of Potential Hazards in Site Vicinity;"
- SRP 3.5.1.6, "Aircraft Hazards;" and
- RG 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants."

The applicant's approach for evaluating natural phenomena events and other external events should be in concert with the risk-informed approach described in proposed 10 CFR Part 70, Subpart H. Although the above references provide useful information for staff use in the review, some of the analysis methods described therein should be adapted to be risk-informed and to agree with the approach described in 10 CFR Part 70, Subpart H.

The applicant's risk from natural phenomena events and other external events shall meet the performance requirements described in 10 CFR Part 70, Subpart H. The applicant's evaluation to determine whether the performance requirements are met should be iterative. First, the applicant should perform evaluations to describe the likelihoods associated with a suite of magnitudes for each type of natural phenomena or other external events. For example, when assessing earthquakes, the applicant should describe likelihoods associated with a suite of maximum accelerations ("g" values); when assessing tornadoes or high winds, the applicant should describe likelihoods associated with a suite of maximum windspeeds; when assessing floods, the applicant should describe likelihoods associated with a suite of maximum water levels and velocities.

Next, the applicant should select a likelihood for each external event and identify the associated magnitude (e.g., water level, windspeed, acceleration level). For each external event, the applicant should identify failures of structures, systems, and components associated with the magnitude of the event, taking into consideration common-cause failures and the likelihoods of the failures, given the event. This step involves developing and applying intermediate assessment tools such as response spectra and floor response spectra for seismic analysis. The applicant should determine the consequences, in terms of radiation and chemical exposures to the public and workers and any nuclear criticalities, for each external event. The applicant should compare the consequences and the associated likelihoods to the performance requirements in proposed 10 CFR Part 70, Subpart H. If the likelihood and consequences of the

external event satisfies the performance requirements, the selection of the external event magnitude is acceptable. Otherwise, if the performance requirements are not satisfied, a less likely event should be selected, the magnitude identified, and the process repeated until the performance requirements are satisfied. This process should be performed for each natural phenomena event and for other external events.

APPENDIX C
FIRE HAZARDS ANALYSIS PROCEDURES

C1. The purpose of the fire hazards analysis (FHA) is to document specific fire hazards, fire protection features proposed to control those hazards, and the overall adequacy of facility fire safety. The FHA consists of a systematic analysis of the fire hazards, an identification of specific areas and systems important to facility fire safety, the development of design-basis fire scenarios, an evaluation of anticipated consequences, and a determination of the adequacy of facility fire safety.

C2. A preliminary FHA should be performed for the MOX facility early in the design phase to ensure incorporation of an acceptable level of protection in the evolving design.

C3. The FHA should be performed under the direction of a qualified fire protection engineer, with support from chemical, electrical, mechanical, and systems engineers, as well as operations staff as needed.

C4. The FHA should contain, but not be limited to, a conservative assessment of the following items and safety issues:

- Descriptions:
  - Construction (Type);
  - Fire hazards;
  - Fire protection features;
  - Critical process equipment; and
  - Operations.
- Potential for a toxic or radiation incident from a fire;
- Impact of natural hazards (earthquake, flood, or wind) on fire safety;
- Protection of items relied upon for safety;
- Life safety considerations;
- Emergency planning;
- Fire department/brigade response;
- Security and safeguards considerations related to fire protection; and
- Exposure fire potential and the potential for fire spread between two fire areas.

C5. The FHA should assume and evaluate the consequences of a single, worst-case automatic fire protection system malfunction during a fire. This could be a detection system that also functions to activate a pre-action type sprinkler system. The failures and/or events postulated in the analysis should be consistent with the probability criteria in the ISA.

C6. If redundant automatic fire protection systems are provided in the area, only the system that causes the most vulnerable condition is assumed to fail. Passive fire protection

features, such as blank fire-rated walls or continuous fire-rated cable wraps are assumed to remain viable in accordance with their fire endurance rating to the extent that they are properly constructed and maintained.

C7. The FHA is normally organized by the individual fire areas that comprise the facility. As defined in Section 7.7, a fire area is a location bounded by fire-rated construction, having a minimum fire resistance rating of 2 hours. The FHA through fire modeling (if necessary) and fire loading analysis should document that the fire ratings are appropriate for each fire area boundary. Where a facility is not subdivided by fire rated construction, the fire area should be defined by the exterior walls and roof of the facility.

C8. The FHA should contain an inventory of items relied on for safety that are susceptible to fire damage within each subarea. Loss of systems such as ventilation, cooling, or electrical power that could cause failures elsewhere in the facility should be evaluated. The FHA should also consider the improper operation of equipment due to spurious signals induced by fire damage. In addition, the effects of combustion products, manual firefighting efforts, and the activation of automatic fire suppression systems should be assessed.

C9. The FHA may need to produce fire related parameters (temperatures, pressures, and air velocities) for evaluating radioactive material dispersion through the facility air distribution system as a result of fire. The radiological consequences should then be determined as part of the integrated safety assessment.

C10. The quantity and associated hazards of flammable and combustible material that can be expected to be found within the fire area should be factored into the analyses. Consideration should also be given to the presence of transient combustibles associated with maintenance activities and storage. Average combustible loading, by itself, should not be used to estimate fire area fire severity. As a minimum, for each designated fire area, the following fire hazards should be evaluated for potential fire severity and consequent damage:

1. Fire load from solid combustible materials (both quantity and configuration) including those materials of construction, in-situ materials, and anticipated transient combustible materials. Combustibles are defined as materials which do not meet the definition of noncombustible material as presented in NFPA Standard 220. For the purposes of the fire load survey, combustibles which can be classified as limited-combustible (as per NFPA 220) may be so classified. In performing the fire loading survey, the end uses of the survey in the FHA and/or ISA should be kept in mind. These uses may include, but not be limited to: determining or verifying the proper design basis of the fire suppression system, determining the minimum required fire resistance for barriers, assuring adequate prefire planning, and input to fire propagation or radionuclide transport modeling. Each of these uses may require the data to be presented in different formats or level of detail.

2. Flammable and combustible liquids and gases used in the processes within the fire area (quantities or flow rates);

3. Process chemicals and materials (both quantity and location) that could present a toxic or radiological hazard, or that could significantly affect health or the quality of the environment through a release as a result of a fire emergency; and

4. Potential ignition sources.

C11. The FHA should contain an assessment of facility fire water requirements including capacity, pressure, and storage requirements. The assessment should include a list of water based automatic suppression systems and their maximum demands, interior hose stream requirements, and exterior hydrant requirements. With this assessment, the facility fire water system layout should also be provided, including the locations and characteristics of pumps, lines, tanks, towers, and sectionalizing valves.

C12. For each designated fire area determined to be important to facility fire safety, the FHA should provide input to the ISA regarding the postulated accident sequences caused or aggravated by fire. Either quantitative or qualitative methods may be used. Where quantitative analytical methods are used, all input data and assumptions are documented.

C13. The FHA should define those fire protection systems and procedures that provide reasonable assurance that the defined consequences of an accident sequence will not occur or will be mitigated. The coverage of fire detection and suppression systems should be shown within each fire area. For the identified fire protection measures, the applicant should specify compensatory measures to be implemented on a temporary basis in the event the identified systems are not operable. Both the compensatory measure(s) and the time schedule for implementation should be established.

Most of the guidance in this appendix originated from "The Implementation Guide for Use with DOE Orders 420.1 and 440.1--Fire Safety Program" (G-420.1/B-0, G-440.1/E-0, September 30, 1995). In some cases, the original guidance was modified to reflect specific needs for the mixed oxide (MOX) facility.

APPENDIX D
FIRE PROTECTION GUIDANCE FOR NUCLEAR FILTER PLENUMS

D1. Filter Plenum Construction

All high-efficiency particulate air (HEPA) filters should meet the requirements of ASME AG-1, Section FC and be listed as tested in accordance with Underwriters Laboratories, Inc. (UL) 586. Entrance filters and prefilters located upstream or made part of final HEPA filter exhaust plenums should be listed as Class 1 air filter units as tested in accordance with UL 900. Filter framing systems and filter plenum housing should be of noncombustible construction.

D2. Fire Rating Requirements for Plenum Housing, Openings, and Dampers

1. Filter plenum enclosures inside buildings or located less than 1.5 m (5 ft) from an adjacent building should be of 2-hour fire rated construction. For enclosures greater than 1.5 m (5 ft) from an existing building, the fire rating may be either one-hour or as determined by the FHA.

2. Door openings into a 2-hour rated filter plenum enclosure should be 1.5-hour minimum fire rated. Door openings into a 1-hour rated filter plenum enclosure should be 0.75-hour minimum fire rated.

3. For ducts not required to function as a nuclear confinement system:

   (1) A 1.5-hour damper should be used where the duct penetrates a 2-hour rated barrier.

   (2) A fire damper is not necessary where the duct penetrates a 1-hour barrier provided that automatic sprinkler protection is provided on both sides of the barrier and the duct passes through the wall and extends into the area outside the enclosure. Transfer grills and similar openings without ducting should be provided with an approved damper.

4. Fire dampers should not be utilized when penetrating fire rated construction where ducting is an integral part of the air filter system equipment that is required to continuously function as part of the confinement system. Such duct material may be made part of the fire rated construction by wrapping, spraying, or enclosing the duct with an approved material to provide a minimum 2-hour rating, or be qualified for a 2-hour fire rated exposure to the duct at the penetration location using the fire damper criteria as specified in UL 555.

5. All mechanical and electrical penetrations made into fire-rated plenum enclosures should be fire stopped by listed materials meeting the requirements of ASTM E-814.

D3. Materials and Hazards Inside Plenums

1. Filter plenum enclosures should only be used for ventilation control equipment. The storage and accumulation of combustible materials (including spare filters) as well as combustible and flammable liquids should not be permitted.

2. Electrical equipment should comply with NFPA 70, and all electrical wiring inside the enclosure should be in metal conduit.

3. The concentration of flammable vapors inside the final filter plenum should not exceed 25 percent of their lower flammable limit. If flammable and combustible gases are expected as a result of facility processes, fixed combustible gas analyzers should be provided with analyzer alarms set to sound at 25 percent of the lower flammable limit and transmitted to a continuously manned position.

D4. Fire Screens for Filter Plenums

1. Fire screens should be located upstream from the prefilters and final filter plenums.

2. Fire screens with metal meshes from 3 to 6 openings per cm (8 to 16 openings per inch) should be provided and located at least 1.2 m (4 ft) upstream from all prefilters and at least 6.1 m (20 ft) upstream from all final filter plenum enclosures.

3. Where prefilters are located in final filter enclosures, fire screens should be located at least 6.1 m (20 ft) upstream from the prefilters.

D5. Fire Detection Systems

1. Automatic fire detectors should be rate-compensated type heat detectors, approved for the specific use and conform to NFPA 72. The detectors should be of the 88 °C (190 °F) temperature range, unless operations require higher temperature air flows.

2. Heat detectors or pilot sprinkler heads should be provided in the final filter enclosure and in ducting prior to the final filter enclosure. Airflow should be considered when determining detector or pilot head location in ducting.

3. Detector installations should be engineered and installed for testing over the life of the detector. Where contamination levels permit, detectors can be removed and tested externally.

D6. Deluge Spray Suppression Systems

1. Automatic and manual water deluge spray systems should be provided inside all final filter plenums for protection of the filters where there is a leading filter surface area greater than 1.5 m^2 (16 ft^2).

2. Automatic deluge systems should be designed as per the applicable provisions of NFPA 13 and 15 and as follows:

   (1) Water spray density should be 10.2 lpm per m^2 (0.25 gpm per ft^2) over the entire filter area or 3.8 lpm per 14 m^3 per min (1.0 gpm per 500 ft^3 per min) air flow, whichever is greater.

   (2) Spray heads should be deluge type sprinkler heads.

   (3) The spray pattern of the deluge head should be in the form of a downward vertical water curtain approximately 15 cm (6 in) in front of the filter. Heads should be spaced so that each head does not exceed 1.2 linear m (4 linear ft) of curtain coverage.

3. Manual spray systems should be designed as per the applicable provisions of NFPA 15 and modified as follows:

   (1) Water spray density should be 10.2 lpm per m^2 (0.25 gpm per ft^2) over the entire filter area.

   (2) Nozzles should be deluge spray nozzles that form a full circle solid cone discharge.

   (3) Spray nozzles should be horizontally directed at the face of the HEPA filters so that all areas of the first stage filters and framing support system are wetted.

4. Automatic and manual water spray system water supplies should be hydraulically calculated and capable of supplying a simultaneous flow of the automatic and manual water spray systems as well as the overhead ceiling automatic fire sprinkler systems for the fire area providing air to the plenum for a minimum period of two hours.

5. Water for the deluge spray system should be provided by two separate water supply connections for reliability. One connection may be a fire department connection.

6. Demisters should be installed to protect the final stage of HEPA filters from being wetted by operation of the deluge water spray system.

Appendix D

Most of the guidance presented here is taken from U.S. Department of Energy (DOE) Standard, "Fire Protection Design Criteria" (DOE-STD-1066-97, March 1997). The items of guidance presented are considered to be pertinent to the filter systems likely to be used at the MOX facility. The items presented also represent the NRC responsibility for fire safety as related to facility nuclear safety rather than property protection. A more comprehensive discussion of Nuclear Filter Plenum Fire Protection can be found in Chapter 14 of the DOE Standard and the references cited in the standard.

APPENDIX E
THE NATIONAL ENVIRONMENTAL POLICY ACT AND ENVIRONMENTAL REPORTS

E1. Introduction

The Commission promulgated 10 CFR Part 51 to implement the National Environmental Policy Act (NEPA) of 1969, which requires an assessment of the environmental impacts for all major Federal actions. The NRC staff conducts an independent assessment for all licensing actions that may have a significant effect on the environment, based on the information provided by the applicant in an environmental report. An environmental report is required for actions listed in 10 CFR 51.60(b). This assessment is documented in an Environmental Assessment (EA) and Finding of No Significant Impact (FONSI) or Environmental Impact Statement (EIS).

The Commission determined that the actions listed in 10 CFR 51.22(c) have insignificant environmental impacts, and these actions are categorically excluded from the requirement for an EA and an environmental report. However, if pursuant to 10 CFR 51.23(c)(11), the action involves an amendment that involves changes in process operations or equipment, the applicant must justify that the action will not result in significant effects on the environment.

The Office of Nuclear Material Safety and Safeguards (NMSS) consolidated environmental review work into the Division of Waste Management (DWM) on May 17, 1999. DWM is responsible for preparing all NMSS EIS and reviewing each EA prepared in NMSS. The Division of Fuel Cycle Safety and Safeguards (FCSS) retains the responsibility to prepare each EA and FONSI and make determinations regarding the applicability of categorical exclusions. As a result, DWM is responsible for determining if the applicant's environmental report is adequate to allow the preparation of an EIS. FCSS is responsible for determining if the applicant's environmental report is adequate to support the preparation of an EA and FONSI or, as applicable, to make a determination regarding a categorical exclusion.

Staff coordination on the review of environmental reports used to prepare an EIS should be obtained through DWM. Supplementary guidance for FCSS staff use on determining the adequacy of environmental reports for an EA and FONSI or to justify the applicability of a categorical exclusion is provided in Section E2 for licensing actions after receipt of a license to possess and use SNM.

Information in Section E2 is presented in parallel with the content of an environmental report, as specified in 10 CFR 51.45. This includes:

- Date of application
  - Environmental considerations
  - Description of the proposed action
  - Purpose of the proposed action
  - Description of the affected environment
  - Discussion of considerations
- Analysis of environmental effects of the proposed action and alternatives
- Status of compliance
- Adverse information

The environmental report may include, reference, or supplement the information submitted to the NRC for prior licensing actions.

Section E3 discusses environmental reports for categorical exclusions and Section E4 reviews the NEPA documentation and coordination necessary for license amendments.

E2. Environmental Report Content

1. Date of Application

The date of an application for a license to possess and use special nuclear material for processing and fuel fabrication, scrap recovery, conversion of uranium hexafluoride, or for the conduct of any other activity, which the NRC has determined pursuant to 10 CFR 51 Subpart A will significantly affect the quality of the environment, is acceptable if the application is submitted at least 9 months before the commencement of construction, as required by 10 CFR Part 70.21(f).

2. Environmental Considerations

An adequate environmental report addresses the requirements of 10 CFR 51.45(b), as described below.

(1) Description of the Proposed Action

The description of the proposed action includes a brief summary of the significant characteristics of the proposed facility, including the major site features and the major plant design and operating parameters. The description includes a complete discussion about how special nuclear material will be processed at the facility. If future construction or expansion is proposed, the description includes a proposed project schedule showing the dates for initiation of site preparation, plant construction, and operation.

(2) Purpose of the Proposed Action

The statement of purpose demonstrates a need for the proposed project. This demonstration provides at least the following information: (1) the quantities of special nuclear material used for domestic benefit, (2) a projection of national and foreign requirements for the services, and (3) alternative sources of supply for the proposed facility's services. If delay of the proposed project would have effects on the nation's material disposition program or on the applicant's business (such as loss of contracts, jobs, or future business), the applicant should discuss these effects.

(3) Description of the Affected Environment

The description of the affected environment includes:

(a) Site location (including longitude and latitude) and facility layout;

(b) Regional demography and land use;

(c) Socioeconomic information, including that for low-income and minority populations within a 50-mile radius;

(d) Regional historic, archaeological, architectural, scenic, cultural, and natural landmarks;

(e) Local meteorology and air quality;

(f) Local surface water and groundwater hydrology;

(g) Regional geology and seismology; and

(h) Local terrestrial and aquatic ecology.

To the extent possible, this information is current and reflects observations and measurements made over a period of years, especially for conditions that are expected to vary seasonally (e.g., precipitation, wind speed and direction, and groundwater levels).

(4) Discussion of Considerations

The reviewer should find that the discussion of considerations is acceptable if it includes:

(a) Impacts of the proposed action on the environment, such as the:

- Effects of site preparation and construction on land use and water use;
- Effects of plant operation on the human population (including consideration of occupational and public radiation exposure) and important biota;
- Any irreversible commitments of resources because of site preparation and plant construction and operation, such as destruction of wildlife habitat, removal of land from agricultural use, and diversion of electrical power;
- Plans and policies regarding deactivation, decommissioning, and dismantling at the end of the plant's useful life;
- Environmental effects of the transportation of radioactive materials to and from the site;
- Environmental effects of accidents;
- Impacts on air and water quality; and
- Impacts on cultural and historic resources.

The environmental report discusses the impacts on the environment in proportion to their significance and considers the cumulative impacts of the proposed action. In addition, accident analyses provided in the environmental report are consistent with the accident scenarios and consequences described in the applicant's ISA Summary.

(b) Adverse environmental effects

The applicant describes any adverse environmental effects that cannot be avoided should the proposal be implemented. This description is presented in quantitative terms to the maximum extent possible. This discussion makes clear which of these effects are unavoidable and subject to later amelioration and which are unavoidable and irreversible. The description includes specific measures that the applicant could take or plan to take to mitigate adverse effects.

(c) Alternatives to the proposed action

The discussion of alternatives to the proposed action is sufficiently complete to aid NRC in developing and exploring, pursuant to Section 102(2)(E) of NEPA, "appropriate alternatives to recommended courses of action in any proposal which involves unresolved conflicts concerning alternative uses of available resources." To the extent practicable, the environmental impacts of the proposal and the alternatives are presented in comparative form.

The discussion of alternatives includes siting alternatives and design alternatives. Comparable levels of information on each site need not be presented as long as the applicant presents sufficient information to facilitate a fair and reasonable comparison. The following factors are considered when comparing alternative sites:

- Physical characteristics of the area, including demographic, geological, hydrological, meteorological, and seismological conditions of the site and surrounding area;
- Location of power sources and transmission lines;
- Location of the major product market;
- Location of raw materials, components, and sources of supply;
- Availability of air, rail, roads, and water for transport of raw materials and supplies, finished products, and solid wastes;
- Commitment of natural resources for site preparation and plant construction, including but not limited to the destruction or diminution of wildlife habitats, flora, woodlands, and marshlands;
- Commitment of capital for site preparation and plant construction;
- Cost of operation, including consideration of labor supply, prevailing wage rates, and other recurring or nonrecurring costs;
- Availability of municipal services or, conversely, the cost of providing services such as water and sewage treatment;
- Requirements for relocating homes and families; and
- Existing and projected land use and economic status of the community (e.g., urban, industrial, stable).

(d) Relationship between short-term uses and long-term productivity

The relationship between local short-term uses of the environment and the maintenance and enhancement of long-term productivity is discussed. Short-term uses are considered to be those that occur during the active life of the facility. Long-term productivity represents the use of the environment beyond deactivation and decommissioning of the facility.

(e) Irreversible or irretrievable commitments of resources

Any irreversible environmental commitments and irretrievable material resources that would be involved in the proposed action are discussed.

3. Analysis of Environmental Effects of the Proposed Action and Alternatives

An adequate environmental report analyzes the environmental effects of the proposed action and alternatives. In accordance with 10 CFR 51.45(c), the analysis considers and balances the environmental effects of the proposed action and the alternatives available for reducing or avoiding adverse environmental effects, as well as the environmental, economic, social, and other benefits of the proposed action.

This analysis quantifies, to the fullest extent practicable, the various factors considered. If the application involves renewal or amendment of a current license, environmental impacts are quantified using environmental monitoring data collected by the licensee. To the extent that there are important qualitative considerations or factors that cannot be quantified, the analysis discusses those considerations and factors in qualitative terms. The analysis contains sufficient data to aid the staff in its development of an independent analysis.

4. Status of Compliance

As required by 10 CFR 51.45(d), the applicant should list all Federal permits, licenses, approvals, and other entitlements, which must be obtained in connection with the proposed action. The list is acceptable if it is complete and current as of the application date.

In addition, 10 CFR 51.45(d) requires that the environmental report include a discussion of the status of compliance with applicable environmental quality standards and requirements including, but not limited to, applicable zoning and land-use regulations, and thermal and other water pollution limitations or requirements which have been imposed by Federal, State, regional, and local agencies having responsibility for environmental protection. The discussion is acceptable if it includes a discussion of whether each alternative will comply with such applicable environmental quality standards and requirements. The discussion includes, but is not limited to, the following Federal laws:

(1) The National Historic Preservation Act of 1966;
(2) The Fish and Wildlife Coordination Act of 1966;
(3) The Wild and Scenic Rivers Act of 1968;
(4) The Endangered Species Act Amendments of 1978; and
(5) The Coastal Zone Management and Improvement Act of 1990.

5. Adverse Information

In accordance with 10 CFR 51.45(e), the preceding discussions and analyses are acceptable if they include information that is adverse to the proposed actions as well as information supporting the proposed action.

E3. Categorical Exclusion

An environmental report is not required for actions identified in 10 CFR 51.60(b)(1) that involve an amendment to the MOX fuel fabrication facility that are not expected to result in significant environmental impacts. However, when these amendments involve changes in process operations or equipment, the applicant needs to justify that the changes will not result in significant environmental effects.

The information provided by the applicant to justify the categorical exclusion determination for changes in process operations or equipment is acceptable if it demonstrates the following as specified in 10 CFR 51.22(c)(11):

- There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite;
- There is no significant increase in individual or cumulative occupational radiation exposure;
- There is no significant construction impact; and
- There is no significant increase in the potential for or consequences from radiological incidents.

Review of the environmental report or information presented to support a categorical exclusion includes review of occupational exposure information. This review should be coordinated with the health physics reviewer to assess the adequacy of the information provided by the applicant.

E4. NEPA Documentation and Coordination

Before taking a licensing action such as a license amendment the NRC will determine whether the proposed action qualifies for a categorical exclusion under 10 CFR 51.22 or whether an EA or EIS should be prepared.

1. An EIS will be prepared if the action meets the criteria in 10 CFR Part 51.20. An EA is not necessary if an EIS will be prepared. Coordination with DWM must be initiated to prepare the EIS.

2. A categorical exclusion will suffice if the action meets the criteria for categorical exclusions as defined in 10 CFR Part 51.22(c). (An action that qualifies for a categorical exclusion is usually identified at the start of the licensing review, and an EA is not required.) No coordination with DWM is necessary.

3. An EA will be prepared if the action meets the criteria in 10 CFR Part 51.21. DWM will be informed that an EA will be prepared. DWM should review the completed EA. On completion, the NRC determines whether to prepare an EIS or a FONSI.

Requirements for the preparation of an EIS, EA, or FONSI are described in detail in 10 CFR Part 51. Documents prepared in accordance with NEPA will follow pertinent NMSS procedures.

E5. Environmental Assessment (EA)

The SPB staff will prepare an EA that identifies the proposed action and includes the following, in accordance with 10 CFR 51.30:

1. A brief discussion of:

   (1) The need for the proposed action;

   (2) Alternatives to the proposed action as required by Section 102(2)(E) of NEPA;

   (3) The environmental impacts of the proposed action and alternatives, as appropriate; and

   (4) As required for special case EAs, as defined by NMSS Policy and Procedures Letter 1-50, Revision 2, 1999, disproportionately high and adverse human health or environmental effects on low income and minority populations.

2. A list of agencies and persons consulted and identification of sources used. During preparation of an EA, the staff will consult with affected States on environmental issues and will document such contact in the EA. This documentation will include the following information identified in NMSS Policy and Procedures Letter 1-48, January 1995:

   (1) The name of each State, agency (including contacted individual's name), or person consulted;

   (2) Date of consultation(s);

   (3) Purpose for the consultation;

   (4) Brief summary of the views or comments expressed by the consulted party and the staff's resolution; and

(5) Reference to publicly available documents containing additional information, if applicable.

Much of the information used to prepare the EA is provided by the applicant in the environmental report. However, the staff will perform independent analyses of the environmental impacts of the proposed action and will discuss the conclusions of these analyses in the EA. The EA should focus on the impacts of the proposed action and should be no more than 15 pages, unless necessary to explain any complicated environmental issues associated with the proposed action.

On completion, the EA should be forwarded to DWM for review. DWM reviews the EA to ensure consistency among all EAs prepared by NMSS. When DWM completes its review, the staff will determine whether to prepare an EIS or a FONSI on the proposed action. As discussed in Appendix X(A6) and provided in 10 CFR 51.33, a determination to prepare a draft FONSI may be made. As provided in 10 CFR 51.25, an EA is not necessary if it is determined that an EIS will be prepared.

E6. Finding of No Significant Impact (FONSI)

When the staff makes a final finding that there are no significant environmental impacts for the proposed action, a final FONSI will be published in the Federal Register. The Commission will not take the proposed action until after the FONSI is published. Requirements for the preparation of a FONSI for materials licensing actions are contained in 10 CFR 51.32-51.35. A FONSI will include the following:

1. Identification of the proposed action;

2. Statement that the Commission has determined not to prepare an EIS for the proposed action;

3. Brief presentation of the reasons why the proposed action will not have a significant impact on the quality of the human environment;

4. The EA or a summary of the EA;

5. A note of any other related environmental documents; and

6. A statement that the finding and any related environmental documents are available for public inspection and where the documents may be inspected.

NRC may make a determination to prepare and issue a draft FONSI for public review and comment before making a final determination whether to prepare an EIS or a final FONSI on the proposed action. A draft FONSI may be prepared if a FONSI appears warranted, but the proposed action is similar to one that normally requires an EIS or is without precedent.

The draft FONSI will be identified as a "draft" and will contain the information specified above for a final FONSI. The draft FONSI will be accompanied by or will include a request for comments on the proposed action and the draft findings within 30 days, or a longer period as may be specified in the notice of the draft findings. This draft FONSI will be published in the Federal Register, distributed as provided in 10 CFR 51.74(a), and made available in accordance with 10 CFR 51.123.

When a draft FONSI is issued, a final determination to prepare an EIS or final FONSI will not be made until the last day of the public comment period has expired.

E7. Environmental Impact Statement (EIS)

When the NRC determines that an EIS will be prepared for the licensing action, coordination should be initiated with the Division of Waste Management. The Division of Waste Management will review the environmental report and prepare the EIS. The environmental reviewer should coordinate with the FCSS Project Manager and DWM to ensure consistency between the environmental review for licensing and the preparation of the EIS. This coordination minimizes potential issues between the safety evaluation and the NEPA analysis, and ensures the results of the NEPA analysis are appropriately incorporated into the Safety Evaluation Report (SER) for the application for construction approval and the SER for the license application.

APPENDIX F
CHECKLIST FOR EVALUATING ACCEPTANCE OF QUALITY ASSURANCE ELEMENTS

F1. Organization - The organizational elements responsible for Quality Assurance (QA) are acceptable if:

1. The responsibility for the overall QA is retained and exercised by the applicant.

2. The applicant identifies and describes the major delegation of work involved in establishing and implementing its QA program or any part thereof to other organizations.

3. When major portions of the applicant's QA program are delegated:

   (1) The applicant describes how responsibility is exercised for overall QA. The extent of management supervision should be given, including the position location, qualifications, and criteria for determining the number of personnel performing these functions.

   (2) The applicant evaluates the performance of work by the delegated organization (method and frequency-once per year, although a longer cycle is acceptable with other evaluations of individual elements-are stated).

   (3) Qualified individuals or organizational elements are identified by position title within the applicant's organization as responsible for the quality of the delegated work before activities are started.

4. Clear management controls and effective lines of communication exist for QA activities among the applicant, contractors, and suppliers to ensure direction of QA.

5. Organizational charts clearly identify all the onsite and offsite organizational elements that function under the purview of QA (such as design, engineering, procurement, manufacturing, construction, inspection, testing, instrumentation, control, operation, and maintenance), the lines of responsibility, and the criteria for determining the size of the QA organization, including the inspection staff.

6. The applicant describes the QA responsibilities of each of the organizational elements noted on the organization charts.

7. The applicant identifies a management position that retains overall authority and responsibility for QA. This position may be filled by a person having the title "QA Manager" or other individual performing that function. This position has the following characteristics:

(1) The position resides at least at the same organizational level as the position of the highest line manager directly responsible for performing activities that affect the quality/safety of facility operations (such as engineering, procurement, construction, and operation) and is independent of operational restraints.

(2) The person in the position has effective communication channels with other senior management personnel.

(3) The person in the position has responsibility for approval of QA manuals.

8. Conformance to established requirements (except for designs) is verified by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified or by individuals or groups trained and qualified in QA concepts and practices who are independent of the organization responsible for performing the task.

9. Persons and organizations performing QA functions have sufficient access to management at a level necessary to ensure the capability to:

(1) Identify quality/safety problems;

(2) Initiate, recommend, or provide solutions through designated channels; and

(3) Verify implementation of solutions.

Positions with the above authority are identified by position title and a description of how the above actions are carried out is provided.

10. When work contributes to a situation adverse to safety and has to be stopped, the following provisions apply:

(1) Designated QA personnel, sufficiently free from direct pressures resulting from operational concerns, have the responsibility, delineated in writing, to stop work in unsafe situations and to control further operations until the conditions that created the unsafe condition are corrected.

(2) The organizational positions with stop-work authority are identified.

11. Provisions are established for the resolution of disputes involving quality of items relied on for safety arising from a difference of opinion between QA personnel and personnel from other departments (engineering, procurement, manufacturing, etc.).

12. Designated QA individuals are involved in day-to-day activities relied on for safety of facility conditions and operations. QA staff members routinely attend and participate in status meetings to ensure that they are kept abreast of day-to-day activities and that there is adequate QA coverage of those activities.

13. Policies regarding the implementation of QA are documented and made mandatory. These policies are established at the facility management or corporate level.

14. The position description ensures that the individual directly responsible for the definition, direction, and effectiveness of overall QA has sufficient authority to effectively implement responsibilities. This position is to be sufficiently free from operational responsibilities to ensure independence of action. Qualification requirements for this individual are established in a position description that includes the following prerequisites:

    (1) Management experience through assignments to responsible positions;

    (2) Knowledge of QA regulations, policies, practices, and standards; and

    (3) Experience in performing QA or QA-related activities in design, construction, or operation in a fuel cycle plant, a power reactor, a low-level waste facility, or in a similar high-technology industry.

15. The person responsible for onsite QA is identified by position and has the appropriate organizational position, responsibilities, and authority to exercise proper control over QA. The duties of this individual are structured such that adequate attention can be given to ensuring that QA at the plant site is being effectively implemented.

Additional guidance for organization is given in SRP Section 4.0, "Organization and Administration."

F2. QA Function – The QA function for items relied on for safety is acceptable if:

1. The scope of QA includes:

    (1) A commitment that activities affecting the quality of design, construction, and operation will be subject to the applicable controls of QA. Activities covered by QA are identified on QA-defining documents.

    (2) A commitment that any test program for items relied on for safety will be conducted with QA controls and a description of how QA will be applied.

(3) A commitment that computer programs for functions related to safety will be procured/developed, modified, maintained, and used in accordance with QA controls and a description of how QA will be applied.

(4) A commitment that special items, environmental conditions, skills, or processes will be provided as necessary to ensure the quality of activities having an effect on safety.

2. A brief summary of the applicant's corporate QA policies is given.

3. The following provisions are established to ensure that quality-affecting procedures required to implement QA are consistent with QA commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official:

(1) The QA organization reviews and documents concurrence in the quality-affecting procedures.

(2) The organizational group or individual responsible for the policy statement is identified.

(3) The quality-affecting procedural controls of the principal contractors are provided for the applicant's review with documented agreement of acceptance before the initiation of activities relied on for safety.

4. Provisions are included for notifying the NRC of changes in the implementation of QA from that described in the application.

5. The QA organization and the necessary technical organizations participate early in the QA definition stage to determine and identify QA controls and the extent to which they are to be applied to items as they relate to safety. This effort may involve applying a defined, graded approach to the items in accordance with their importance to safety.

6. A description is provided that emphasizes how the detailed QA will be properly implemented and carried out.

7. A description is provided of how management (above or outside the QA organization) regularly assesses the scope, status, adequacy, and compliance of QA. These measures should include:

(1) Frequent appraisals of QA status through reports, meetings, audits and/or self assessments;

(2) Performance of an annual, preplanned, and documented assessment; and

(3) Identification and tracking of corrective actions based on assessment findings.

8. Activities which are items relied on for safety (such as design, procurement, and site investigation) initiated prior to formal NRC acceptance of the QA program are controlled by a QA program in accordance with this SRP section. Approved procedures and appropriately trained personnel should be available to implement the applicable portion of the QA program prior to the initiation of the activity.

9. A summary description is provided on how responsibilities and control of quality-affecting activities are transferred from the principal contractors to the applicant as the design and construction phase is completed.

10. Indoctrination, training, and qualification[1] are established so that:

(1) Personnel responsible for performing and verifying activities affecting quality are instructed as to the purpose, scope, and implementation of the applicable manuals, instructions, and procedures.

(2) Personnel performing and verifying activities affecting safety and/or quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

(3) For formal training and qualification, documentation includes a statement of the training objective and its content, the attendees, and the date of attendance.

(4) Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

(5) The certificate of qualifications clearly delineates the specific functions personnel are qualified to perform and the criteria used to qualify personnel in each function.

(6) Proficiency of personnel performing and verifying activities affecting safety/quality is maintained by retraining, reexamining, and/or recertifying, as determined by management or program commitment.

11. The applicant's ISA is developed and maintained under QA controls.

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[1] Guidance for training and qualification of plant personnel is given in SRP Section 15.4.

F3. Design Control[2] – Control of the design of items relied on for safety is acceptable if:

1. The scope of design control includes design activities associated with the preparation and review of design documents, including the correct translation of applicable regulatory safety requirements and associated design bases into design, procurement, and procedural documents.

2. Organizational responsibilities are described for preparing, reviewing, approving, and verifying design documents related to an item or its processes, such as system descriptions, design input and criteria, design drawings, design analyses, computer programs, specifications, and procedures.

3. Organizational responsibilities are described for planning and conducting site characterization, including reviewing, approving, and verifying analyses and conclusions.

4. Errors and deficiencies in approved design documents, including design methods (such as computer codes) that could adversely affect the performance of items and processes are documented, and action is taken to ensure that all errors and deficiencies are corrected.

5. Deviations from specified quality standards are identified, and procedures are established to ensure their control.

6. Internal and external design interface controls, procedures, and lines of communication among participating design organizations and across technical disciplines are established and described for the review, approval, release, distribution, and revision of documents involving design interfaces to ensure that items are compatible geometrically and functionally.

7. Procedures are established and described requiring documented verification of the dimensional accuracy and completeness of design drawings and specifications.

8. Procedures are established and described requiring that design drawings and specifications for items relied on for safety be reviewed by the QA organization to ensure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain necessary QA requirements, such as inspection and test requirements, acceptance requirements, and those pertaining to the extent of documenting inspection and test results. These reviews are documented.

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[2] Guidance for configuration management is given in SRP Section 15.2.

9. Guidelines or criteria are established and described for determining the method of design verification (design review, alternate calculations, or tests).

10. Procedures are established and described for design verification activities that ensure the following:

    (1) The verifier is qualified, and neither the verifier nor the verifier's immediate supervisor is directly responsible for the design. In exceptional circumstances, the designer's immediate supervisor may perform the verification provided:

        (a) The supervisor is the only technically qualified individual;

        (b) The need is individually documented and approved in advance by the supervisor's management; and

        (c) QA audits and self assessments cover frequency and effectiveness of the use of supervisors as design verifiers to guard against abuse.

    (2) Design verification is completed before release of procurement, manufacturing, or construction to another organization for use in other design activities. When this schedule cannot be met, the design verification may be deferred, provided the justification for this action is documented and the unverified portion of the design output document and all design output documents, based on the unverified data, are appropriately identified and controlled. Construction site activities associated with a design or design change should not proceed without verification past the point where the installation would become irreversible (i.e., require extensive demolition and rework).

    (3) Procedural control is established for design documents that reflect the commitments of the application for construction approval and license application for operations. Procedural control differentiates between documents that undergo formal design verification by interdisciplinary or multi-organizational teams and those that can be reviewed by a single individual (a signature and date are acceptable documentation for personnel certification). Design documents that pertain to plant safety and are subject to procedural control include, but are not limited to, specifications, calculations, computer programs, system descriptions, and drawings (including flow diagrams, piping and instrument diagrams, control logic diagrams, electrical single-line diagrams, diagrams of structural systems for major facilities, site arrangements, and equipment locations). Specialized reviews should be used when uniqueness or special design considerations warrant them.

(4) The responsibilities of the verifier, the areas and features to be verified, the pertinent considerations to be verified, and the extent of documentation are identified in procedures.

11. The following provisions are included if the verification method is only by test:

(1) Procedures provide criteria that specify when verification should be by test.

(2) Prototype, component, or feature testing is performed as early as possible before installation of plant items or before the installation would become irreversible.

(3) Verification by test is performed under conditions that simulate the most adverse design conditions as determined by analysis.

12. Procedures are established to ensure that verified computer codes are certified for use and that their use is specified.

13. Design and specification changes, including field changes, are subject to the same design controls that were applicable to the original design.

F4. Procurement Document Control – Control of procurement documents for the procurement of items relied on for safety is acceptable if:

1. Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, are inspectable, and are controllable; that there are adequate acceptance and rejection criteria; and that procurement documents have been prepared, reviewed, and approved in accordance with QA requirements. To the extent necessary, procurement documents should require that contractors and subcontractors provide acceptable QA. The review and documented concurrence of the adequacy of quality requirements stated in procurement documents are performed by independent personnel trained and qualified in QA practices and concepts.

2. Procedures are established to ensure that procurement documents identify applicable regulatory, technical, administrative, and reporting requirements; drawings; specifications; codes or industry standards; inspection and test requirements; and special process instructions that must be met by suppliers.

3. Organizational responsibilities are described for procurement planning; the preparation, review, approval, and control of procurement documents; supplier selection; bid evaluations; and the review of and concurrence with supplier QA before initiation of activities relied on for safety. The involvement of the QA organization is described.

F5. Instructions, Procedures,[3] and Drawings – Activities related to instructions, procedures, and drawings pertaining to items relied on for safety are acceptable if:

1. Organizational responsibilities are described for ensuring that activities affecting the quality of items relied on for safety are prescribed by documented instructions, procedures, and drawings and accomplished through implementation of these documents.

2. Procedures are established to ensure that instructions, procedures, and drawings that could affect safety include quantitative acceptance criteria (such as those pertaining to dimensions, tolerances, and operating limits) for determining that activities relied on for the safety of plant operations have been satisfactorily performed.

F6. Document Control – Control of documents related to items relied on for safety is acceptable if:

1. The scope of document control is described and the types of controlled documents are identified. As a minimum, controlled documents include:

   (1) Design documents (e.g., calculations, drawings, specifications, and analyses), including documents related to computer codes;

   (2) Procurement documents;

   (3) Instructions and procedures for such activities as fabrication, construction, modification, installation, maintenance, testing, and inspection;

   (4) Documents pertaining to as-built conditions;

   (5) QA and quality control manuals, procedures, and reports; and

   (6) Technical reports.

2. Procedures for the review, approval, and issuance of documents and changes thereto are established and described to ensure technical adequacy and inclusion of appropriate safety/quality requirements before implementation. The QA organization, or an individual other than the person who generated the document but who is qualified in QA, reviews and concurs with these documents in regard to QA-related aspects.

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[3] Guidance for plant procedures is given in SRP Section 15.5.

3. Procedures are established to ensure that changes to documents are reviewed and approved by the same organizations as those that performed the initial review and approval or by other qualified, responsible organizations delegated by the applicant.

4. Before commencing work, procedures are established to ensure that documents are available at the location where the activity will be performed.

5. Procedures are established and described to ensure that obsolete or superseded documents are removed and replaced by applicable revisions in work areas in a timely manner.

6. A master list or equivalent document control system is established to identify the current revision of instructions, procedures, specifications, drawings, and procurement documents. When such a list is used, it should be updated and distributed to predetermined responsible personnel.

7. Procedures are established and described to provide for the preparation of drawings pertaining to as-built conditions and related documentation in a timely manner to accurately reflect the actual design.

F7. Control of Purchased Items – Control of purchased items relied on for safety is acceptable if:

1. Organizational responsibilities are described for the control of purchased items including interactions between design, procurement, and QA organizations.

2. Verification of suppliers' activities during fabrication, inspection, testing, and shipment of items relied on for safety is planned and performed with QA organization participation in accordance with written procedures to ensure conformance to the purchase order requirements. The procedures, as applicable to the method of procurement, provide for:

   (1) The specification of the characteristics or processes to be witnessed, inspected, or otherwise verified; the method of verification and the required documentation; and the personnel responsible for implementing these procedures; and

   (2) Audits, surveillances, or inspections that ensure that the supplier complies with the quality requirements.

3. Procurement of spare or replacement parts for items relied on for safety is subject to QA controls, to codes or standards, and to technical requirements equal to or better

than the original technical requirements, or as required to prevent the procurement of defective items.

4. Selection of suppliers is documented and filed.

5. Items are inspected when received to ensure:

   (1) The item is properly identified and corresponds to the identification on the purchase document and the documentation when the item is received.

   (2) The item and acceptance records satisfy the inspection instructions before installation or use of the item.

   (3) Specified inspection, test, and other records (such as certificates of conformance attesting that the item conforms to specified requirements) are available at the facility before installation or use of the item.

6. Items accepted and released are identified as to their inspection status before they are forwarded to a controlled storage area or are released for installation or further work.

7. The supplier furnishes the following records to the purchaser:

   (1) Documentation that identifies the purchased item and the specific procurement requirements (e.g., codes, standards, and specifications) met by the item relied on for safety;

   (2) Documentation that identifies any procurement requirements that have not been met; and

   (3) A description of those items that do not conform to the procurement requirements and that are designated "accept as is" or "repair."

The procedure for review and acceptance of these documents is described.

8. For commercial "off-the-shelf" items where specific QA controls cannot be imposed in a practicable manner, special quality verification requirements are established and described to ensure that an acceptable item has been received by the purchaser.

9. Suppliers' certificates of conformance are periodically evaluated by audits, independent inspections, or tests to ensure that they are valid and that the results are documented.

F8. Identification and Control of Items – Identification and control of items relied on for safety are acceptable if:

1. Controls are established and described to identify and control items relied on for safety. The description should include organizational responsibilities.

2. Procedures are established that ensure that identification is maintained either on the item relied on for safety or on records traceable to the item, to preclude use of incorrect or defective items.

3. Identification of items relied on for safety can be traced to the appropriate documentation such as drawings, specifications, purchase orders, manufacturing and inspection documents, deviation reports, and physical and chemical test reports.

4. Correct identification of items is verified and documented before they are released for fabrication, assembling, shipping, and installation.

F9. Control of Special Processes – Control of special processes related to items relied on for safety is acceptable if:

1. Organizational responsibilities, including those for the QA organization, are described for the qualification of special processes, equipment, and personnel.

2. Procedures are established for recording evidence of an acceptable level of quality for special processes, using qualified procedures, equipment, and personnel.

3. Qualification records of procedures, equipment, and personnel associated with special processes are established, filed, and kept current.

F10. Inspection – Inspection of items relied on for plant or process safety is acceptable if:

1. The scope of inspection indicates that an effective inspection program has been established. Procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or for defining how and when inspections are performed. The QA organization participates in these functions.

2. Organizational responsibilities for inspection are described. Individuals performing inspections are other than those who performed or directly supervised the item/activity being inspected and do not report directly to the immediate supervisors who are responsible for the item/activity being inspected. If the individuals performing inspections are not part of the QA organization, the inspection procedures, personnel qualification criteria, and independence from undue pressure,

such as operational needs, should be reviewed and found acceptable by the QA organization before the initiation of the activity.

3. A qualification plan for inspectors is established and documented and the qualifications and certifications of inspectors are kept current.

4. Inspection procedures, instructions, or checklists provide for the following:

   (1) Identification of characteristics and activities to be inspected;

   (2) A description of the method of inspection;

   (3) Identification of the individuals or groups responsible for performing the inspection in accordance with the provisions of Item 10.b in this section;

   (4) Acceptance and rejection criteria;

   (5) Identification of required procedures, drawings, and specifications and revisions;

   (6) Identification of inspection personnel, measuring and test equipment used (including any data recorders), and the results of the inspection; and

   (7) Specification of the necessary measuring and test equipment, including accuracy requirements.

5. Inspection results are documented and evaluated and their acceptability is determined by a responsible individual or group.

F11. Test Control – Control of tests of items relied on for safety is acceptable if:

1. The description of the scope of test control indicates that an effective test program has been established for tests, including proof tests before installation and pre-operational tests. Procedures provide criteria for determining the accuracy requirements of test equipment and provide criteria for determining when a test is required or how and when testing activities should be performed.

2. Test procedures or instructions provide, as required, for the following:

   (1) The requirements and acceptance limits in applicable design and procurement documents;

   (2) Instructions for performing the test;

(3) Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation, including their accuracy requirements, completeness of items to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage;

(4) Test acceptance and rejection criteria,

(5) Mandatory inspection hold points for witness by owner, contractor, or inspector (as applicable);

(6) Methods of documenting or recording test data and results; and

(7) Provisions for ensuring that test prerequisites have been met.

3. Test results are documented and evaluated and their acceptability is determined by a responsible individual or group.

4. A qualification plan is established and documented for those individuals conducting the tests and certifications for those individuals performing the tests are kept current.

F12. Control of Measuring and Test Equipment – Control of measuring and test equipment (such as instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive test equipment) identified as items relied on for safety or used to measure or test other items relied on for safety is acceptable if:

1. The scope for the control of measuring and test equipment is described, along with the types of equipment to be controlled. This information indicates that effective calibrations and adjustments have been established.

2. QA and other organizations' responsibilities are described for establishing, implementing, and ensuring the effectiveness of the calibrations and adjustments.

3. Procedures are established and described for calibration (technique and frequency), maintenance, and control of measuring and test equipment. The review of and documented concurrence with these procedures are described and the organization responsible for these functions is identified.

4. Measuring and test equipment is identified and traceable to the calibration data.

5. Measuring and test equipment is labeled, tagged, or "otherwise controlled" to indicate the due date of the next calibration. The method to "otherwise control" measuring and test equipment should be described.

6. Measuring and test equipment is calibrated at specified intervals on the basis of the required accuracy, purpose, degree of usage, stability characteristics, and other conditions affecting the measurement. The test equipment should have sufficient accuracy to ensure that the equipment being calibrated is within required tolerance, and the basis of acceptance is documented and authorized by responsible management. The management authorized to perform this function is identified.

7. Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if they can be shown to be adequate to meet the requirements, and the basis of acceptance is documented and authorized by a responsible member of the management staff. The management staff member authorized to perform this function is documented.

8. Reference and transfer standards are traceable to nationally recognized standards; where national standards do not exist, provisions are established to document the basis for calibration.

9. Measurements are taken and documented to determine the validity of previous inspections and the acceptability of items inspected or tested since the last calibration when measuring and test equipment is found to be out of calibration. Inspections or tests are repeated on items determined to be suspect.

F13. Handling, Storage, and Shipping – Handling, storage, and shipping of items relied on for safety are acceptable if:

1. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and implemented by suitably trained individuals in accordance with predetermined work and inspection instructions.

2. Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of items in accordance with design and procedure requirements.

F14. Inspection, Test, and Operating Status – Inspection, test, and operating status of items relied on for safety are acceptable if:

1. Procedures are established to indicate the inspection, test, and operating status of items.

2. Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps.

3. Procedures are established and described to control the alteration of the sequence of required tests, inspections, and other operations relied on for safety. Such actions should be subject to the same controls as those for the original review and approval.

4. The status of nonconforming, inoperative, or malfunctioning items and processes is documented and identified to prevent inadvertent use. The organization responsible for this function is identified.

F15. Nonconforming Items – Control of nonconforming items relied on for safety is acceptable if:

1. Procedures are established and described for the identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming items (including computer codes) if disposition is other than to scrap. The procedures identify authorized individuals responsible for the independent review of nonconforming items, including their disposition and closeout.

2. QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconformance.

3. Documentation identifies the nonconforming item; describes the nonconformance, the disposition of the nonconforming item, and the inspection requirements; and includes signature approval of the disposition. Nonconformances are corrected or resolved before the initiation of preoperational testing of the item.

4. Reworked, repaired, and replacement items are inspected and tested in accordance with the original inspection and test requirements or acceptable alternatives.

5. Nonconformance reports are periodically analyzed by the QA organization to show quality trends, and the significant results are reported to upper management for review and assessment.

F16. Corrective Action – Corrective actions that affect or support items relied on for safety are acceptable if:

1. Procedures are established and described indicating that effective corrective actions have been established. The QA organization reviews and documents concurrence with the procedures.

2. Corrective action is documented and initiated after the determination of a condition adverse to safety/quality (e.g., nonconformance, failure, malfunction, deficiency,

deviation, defective item, a failure to follow operating procedures, or a human error) to preclude recurrence. The QA organization concurrence is required regarding the adequacy of the corrective action.

3. Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

4. Significant conditions adverse to safety, the root cause of the conditions, and the corrective action taken to preclude repetition are documented and reported to immediate management and upper levels of management for review and assessment.

F17. QA Records[4] – Control of QA records is acceptable if:

1. QA and other organizations are identified and their responsibilities are described for the definition and implementation of QA records.

2. Inspection and test records contain the following, where applicable:

   (1) A description of the type of observation;

   (2) The date and results of the inspection or test;

   (3) Information on conditions adverse to quality;

   (4) Identification of the inspector or data recorder;

   (5) Evidence as to the acceptability of the results; and

   (6) Action taken to resolve any discrepancies noted.

3. Suitable facilities for the storage of the records are described.

F18. Audits and Assessments – Guidance for audits and assessments is given in SRP Section 15.6.

F19. Applicant's Provisions for Continuing QA – The applicant's provisions for continuing QA are acceptable if the submittal addresses reviews and updates based on reorganizations, revised activities, lessons learned, changes to applicable regulations, and other QA changes that should be reflected in the license application's QA program description to keep it current.

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[4] Additional guidance for records management is given in SRP Section 15.8.

APPENDIX G
CHECKLIST FOR PLANT PROCEDURES

The list below shows activities that should be covered by written procedures. The list is not intended to be all inclusive nor is it intended to imply that procedures be developed with the same titles as those on the list.

G1. Operating Procedures

1. Procedures that address startup, operation, shutdown, control of process operations, and recovery after a process upset:

   (1) Ventilation;
   (2) Criticality alarms;
   (3) Shift routines, shift turnover and operating practices;
   (4) Decontamination operations;
   (5) Plant utilities (air, other gases, cooling water, fire water, steam);
   (6) Temporary changes in operating procedures; and
   (7) Abnormal operation/alarm response:

      (a) Loss of cooling water;
      (b) Loss of instrument air;
      (c) Loss of electrical power;
      (d) Loss of criticality alarm system;
      (e) Loss of containment;
      (f) Fires; and
      (g) Chemical process releases.

2. Maintenance activities that address repair, calibration, surveillance, and functional testing:

   (1) Repairs and preventive repairs of items relied on for safety;
   (2) Testing of criticality alarm units;
   (3) Calibration of items relied on for safety;
   (4) HEPA filter maintenance;
   (5) Functional testing of items relied on for safety;
   (6) Relief valve replacement/testing;
   (7) Surveillance/monitoring;
   (8) Pressure vessel testing;
   (9) Piping integrity testing; and
   (10) Containment device testing.

3. Emergency procedures:

   (1) Response to a criticality and

(2) Hazardous process chemical releases.

G2. Management Control Procedures

1. Training;
2. Audits and assessments;
3. Incident investigation;
4. Records management;
5. Configuration management;
6. Quality assurance;
7. Equipment control (lockout/tagout);
8. Shift turnover;
9. Work control;
10. Management control;
11. Procedure management;
12. Nuclear criticality safety;
13. Fire protection;
14. Radiation protection;
15. Radioactive waste management;
16. Maintenance;
17. Environmental protection;
18. Chemical process safety;
19. Operations;
20. Calibration control;
21. Preventive maintenance;
22. Design control; and
23. Test control.

APPENDIX H
HEALTH AND SAFETY RECORDS

The requirements for records management will be dependent upon the applicable hazards and risks determined for the facility. Examples of the types of records that should be included in the system required by 10 CFR Parts 19, 20, 21, 25, and 70 (as proposed) are listed in Section H1 below. Section H2 lists examples of the types of records that should be established and maintained to provide reasonable assurance that items relied on for safety will be available and reliable to perform their function when needed, as required by 10 CFR §70.64(1), as proposed. Section H2 is organized under the chapter headings of the SRP.

Although Sections H1 and H2 lists examples of records, the lists are not intended to be exhaustive or prescriptive in format. Furthermore, the applicant may choose to organize the records in ways other than shown here.

H1. Examples of Records Required by 10 CFR Parts 19, 20, 21, 25, and 70

1. Audits
2. Access authorization for personnel
3. Administrative procedures with safety implications
4. Air sample data
5. Bioassay data
6. Change control records for material control and accounting program
7. Radiation dose to individuals of the public
8. Radiation exposure history
9. Individual radiation monitoring data
10. Individual radiation monitoring results
11. Individual intakes of radioactive material
12. Radioactive material storage records
13. Planned special radiation exposures
14. Radiation protection (and contamination control) records
15. Radiation training records
16. Radiation work permits
17. Records of cumulative occupational radiation dose
18. Records of receipt, transfer, and disposal of radioactive material
19. Records of radioactive waste disposal
20. Reports of theft/loss of licensed material
21. Results of radiation surveys/calibrations
22. Results of measurements used to calculate radioactive effluents
23. Health and safety compliance records, medical records, personnel exposure records, etc.

H2. Examples of Records that Should Provide Reasonable Assurance that Items Relied on for Safety Will Be Available and Reliable to Perform their Function (Listed by SRP Section)

1. General Information:

   (1) Construction records;
   (2) Facility and equipment descriptions and drawings;
   (3) Design criteria, requirements, and bases for safety-related structures, systems, or components, as specified by the facility configuration management system;
   (4) Records of facility changes and associated integrated safety analyses, as specified by the facility configuration management system;
   (5) Safety analyses, reports, and assessments;
   (6) Records of site characterization measurements and data;
   (7) Records pertaining to onsite disposal of radioactive or mixed wastes in surface landfills; and
   (8) Specifications for items relied on for safety.

2. Financial Qualifications:

   None

3. Protection of Classified Matter:

   (1) Procedures to prevent tampering and loss of classified/sensitive records; and
   (2) Employee access authorization lists.

4. Organization and Administration:

   (1) Administrative procedures with safety implications;
   (2) Change control records for material control and accounting program;
   (3) Organization charts, position descriptions, and qualifications records;
   (4) Health and safety compliance records, medical records, personnel exposure records;
   (5) Quality assurance records (see Section H2.15(1) of this appendix);
   (6) Safety inspections, audits, assessments, and investigations; and
   (7) Safety statistics and trends.

5. Integrated Safety Analysis:

   (1) Integrated safety analysis and revisions and
   (2) Integrated safety analysis summary.

6. Nuclear Criticality Safety:

   (1) Nuclear criticality control written procedures and statistics;
   (2) Nuclear criticality safety analyses;
   (3) Records pertaining to nuclear criticality inspections, audits, investigations, and assessments;
   (4) Records pertaining to nuclear criticality incidents, unusual occurrences, or accidents; and
   (5) Records pertaining to nuclear criticality safety analyses.

7. Fire Protection:

   (1) Fire Hazard Analysis;
   (2) Fire prevention measures, including hot-work permits and fire-watch records;
   (3) Records pertaining to inspection, maintenance, and testing of fire protection equipment;
   (4) Records pertaining to fire protection training and retraining of response teams; and
   (5) Pre-fire emergency plans.

8. Chemical Safety:

   (1) Chemical process safety procedures and plans;
   (2) Records pertaining to chemical process inspections, audits, investigations, and assessments;
   (3) Diagrams, charts, and drawings;
   (4) Records pertaining to chemical process incidents, unusual occurrences, or accidents;
   (5) Chemical process safety reports and analyses; and
   (6) Chemical process safety training.

9. Radiation Safety:

   (1) Bioassay data;
   (2) Exposure records;
   (3) Radiation protection (and contamination control) records;
   (4) Radiation training records; and
   (5) Radiation work permits.

10. Environmental Protection:

   (1) Environmental release and monitoring records and

(2) Environmental Report and Supplements to the Environmental Report, as applicable.

11. Plant Systems:

(1) Written procedures and statistics for plant systems;
(2) Safety analyses and management measures for plant systems;
(3) Records pertaining to inspections, audits, investigations, and assessments of plant systems ; and
(4) Records pertaining to a description of equipment and facilities design (electrical systems, structures and components, cooling water systems, containment and confinement systems, ventilation systems, etc.).

12. Human Factors:

(1) Personnel performance trend analyses and
(2) Human factor improvements.

13. Security and Safeguards:

(1) Physical protection plans;
(2) Fundamental nuclear material control plans;
(3) Transportation plans;
(4) Records pertaining to granting unescorted access; and
(5) Records pertaining to material control and accounting of special nuclear material.

14. Emergency Protection:

(1) Emergency plan(s) and procedures;
(2) Comments on emergency plan from outside emergency response organizations;
(3) Emergency drill records;
(4) Memorandum of understanding with outside emergency response organizations
(5) Records of actual events;
(6) Records pertaining to the training and retraining of personnel involved in emergency preparedness functions; and
(7) Records pertaining to the inspection and maintenance of emergency response equipment and supplies.

15. Management Measures:

(1) Quality Assurance:

(a) Table 1 in Reference 2 contains a list of QA records generated during design and construction of a nuclear power plant that should be maintained as QA records. Although Reference 2 was developed for nuclear power plants, the QA record keeping requirements for the design and construction of this facility should be comparable; and
(b) Appendix A of Reference 3 contains a list of typical procedures for the operation of nuclear power plants. Although Reference 3 was developed for nuclear power plants, the QA record keeping requirements for the operation of this facility should be comparable.

(2) Configuration Management:

(a) Safety analyses, reports, and assessments that support the physical configuration of process designs and changes to those designs;
(b) Validation records for computer software used for safety analysis or MC&A;
(c) ISA documents including facility drawings, specifications, and purchase specifications for items relied on for safety; and
(d) Approved, current operating procedures and emergency operating procedures.

(3) Maintenance:

(a) Preventive maintenance records, including trending and root cause analysis;
(b) Calibration and testing data for items relied on for safety; and
(c) Corrective maintenance records.

(4) Training and Qualification of Plant Personnel:

(a) Personnel training and qualification record and
(b) Procedures.

(5) Plant Procedures:

(a) Standard operating procedures and
(b) Functional test procedures.

(6) Audits and Assessments:

(a) Audits of safety and environmental activities and
(b) Assessments of safety and environmental activities

(7) Incident Investigations:

(a) Investigation reports;
(b) How and when changes recommended by investigation reports are implemented;
(c) Summary of reportable events for the term of the license; and
(d) Incident investigation policy.

(8) Records Management:

(a) Policy;
(b) Material storage records; and
(c) Records of receipt, transfer, and disposal of radioactive material.

H3. References

1. 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. Nuclear Regulatory Commission, (U.S.) (NRC), Regulatory Guide 1.28, Rev. 3, "Quality Assurance Program Requirements (Design and Construction)." NRC: Washington, D.C. August 1985.

3. Nuclear Regulatory Commission, (U.S.) (NRC), Regulatory Guide 1.33, Rev. 23, "Quality Assurance Program Requirements (Operation)." NRC: Washington, D.C. February 1978.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This NUREG provides guidance to the NRC staff reviewers in the Office of Nuclear Material Safety and Safeguards who will perform safety and environmental impact reviews of the anticipated application for construction approval and license application for operations for the Mixed Oxide (MOX) Fuel Fabrication Facility under the proposed 10 CFR Part 70 specifically as related to plutonium processing and fuel fabrication. The standard review plan (SRP) presented in this NUREG 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 the 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 (SER).

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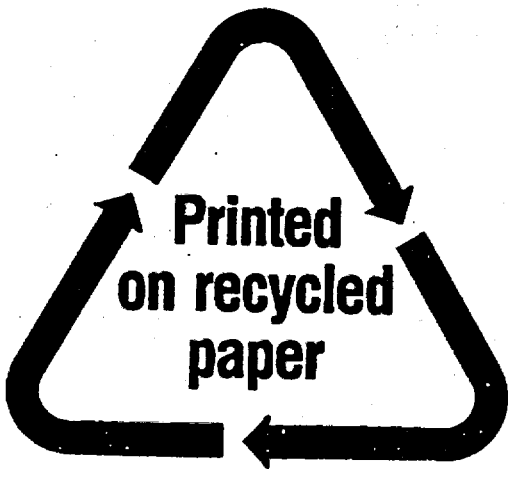

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