70-3098

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 14, 1999

MEMORANDUM TO: Melanie A. Galloway, Acting Chief
Special Projects Branch
Division of Fuel Cycle Safety
and Safeguards, NMSS

THRU: Charles Cox, Acting Chief
Enrichment Section
Special Projects Branch
Division of Fuel Cycle Safety
and Safeguards, NMSS

FROM: Drew Persinko, Sr. Nuclear Engineer
Enrichment Section
Special Projects Branch
Division of Fuel Cycle Safety
and Safeguards, NMSS

SUBJECT: SUMMARY OF MEETING WITH DUKE COGEMA STONE & WEBSTER TO DISCUSS TECHNICAL TOPICS ASSOCIATED WITH THE MIXED OXIDE FUEL FABRICATION FACILITY

On November 16-17, 1999, the Nuclear Regulatory Commission (NRC) staff met with representatives from Duke Cogema Stone & Webster (DCS) to discuss technical topics associated with the mixed oxide (MOX) fuel fabrication facility. Topics discussed included design bases, quality assurance program, quality assurance classification and quality levels, process description overview, integrated safety analysis (ISA), natural phenomena hazards, and nuclear criticality safety.

The attendance list, meeting agenda, and slides used in the presentation are attached (Attachments 1, 2, and 3, respectively).

The meeting began with a brief update of the MOX project schedule by DCS. DCS indicated that the start and end dates have not changed; intermediate dates have been revised to reflect information from the August 31, 1999, NRC/DCS meeting. DCS still intends to submit an application in September 2000 with sufficient information for NRC to authorize construction. The complete license application is scheduled to be submitted in March 2003.

During the presentations, the staff indicated that it would like to obtain a more in-depth understanding of: (1) the formal and working relationships between the Cogema and SGN quality assurance organizations and programs and the overall DCS MOX quality assurance program (SGN is a wholly owned subsidiary of Cogema and provides process design expertise to Cogema); (2) the type of information DCS considers to be design basis information; (3) quality classification of specific systems and components (e.g., criticality alarm systems); and (4) hazard analysis and initial ISA results as the analyses progress.

NFO5

PDR ADOCK 07003098

Future meetings will be scheduled to discuss worker dose, use of polycarbonate material in glovebox construction, definition of site boundary and collocated workers, physical security, material control and accounting, International Atomic Energy Agency requirements, radiation protection, confinement systems, and fire protection.

Docket: 70-3098

Attachments: 1. Attendance List
2. Meeting Agenda
3. Slides

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Future meetings will be scheduled to discuss worker dose, use of polycarbonate material in glovebox construction, definition of site boundary and collocated workers, physical security, material control and accounting, International Atomic Energy Agency requirements, radiation protection, confinement systems, and fire protection.

Docket: 70-3098

Attachments: 1. Attendance List
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ATTENDEES
(Attending all or part of the meetings on Nov 16 and 17, 1999)

NAME	AFFILIATION
Andrew Persinko	Nuclear Regulatory Commission (NRC)
Melanie Galloway	NRC
Robert Pierson	NRC
Amy Bryce (via phone)	NRC
Rex Wescott	NRC
Albert Wong	NRC
Charles Cox	NRC
Yen-Ju Chen	NRC
Rocio Castaneira	NRC
A. Lynn Silvious	NRC
Wilkens Smith	NRC
Richard Lee	NRC
Robert Shewmaker	NRC
Kathryn Winsberg	NRC
Jack Spraul	NRC
Joel Kramer	NRC
Alex Murray	NRC
Christopher Tripp	NRC
Peter Lee	NRC
Julie Olivier	NRC
Michael Adjodha	NRC
J. Keith Everly, Jr.	NRC
Yawar Faraz	NRC
Jennifer Davis	NRC
Richard Milstein	NRC
Tin Mo	NRC
Ed Brabazon	Duke Cogema Stone & Webster (DCS)
Ray Fortier	DCS
Toney Mathews	DCS
Peter Hastings	DCS
Laurence Cret	DCS
Bill Hennessy	DCS
Richard Berry	DCS
Jim Brackett	DCS
Robert Freeman	DCS
John Matheson	DCS
Bob Foster	DCS
James Thornton	DCS
David Noxon	DCS
Jamie Johnson	Department of Energy (DOE)
Patrick Rhoads	DOE

ATTACHMENT 1

ATTENDEES

NAME	AFFILIATION
Don Williams	Oak Ridge National Laboratory
Faris Badwan	Los Alamos National Laboratory
Sidney Crawford	Consultant (self)
Steven Dolley	Nuclear Control Institute
Kevin Kamps	Nuclear Information and Resource Service

AGENDA
MOX FUEL FABRICATION FACILITY (MFFF) MEETING
NOVEMBER 16 - 17, 1999

November 16, 1999 (Tuesday) / 1-4 pm / Room T8A1

Schedule/Strategy for Licensing Submittals
Brief overview of changes to DCS licensing schedule as a result of August 31, 1999 NRC/NMSS meeting
Definition of Design Basis
Discussion of the definition of "design basis" for support of construction authorization, and overview of engineering documents expected to be available in support of Construction Authorization and License Application
Quality Assurance (QA) Program Overview
QA Classification and Quality Levels
Overview of DCS process for determining safety classification/quality level for SSCs

November 17, 1999 (Wednesday) / 8:30-4 pm / Room T3B45

Criticality Design
Present the MFFF criticality design approach, the interface process between the Process Group (France) and the Facility Group (US) including roles and responsibilities, the approach to benchmarking, and brief discussion of the AVLIS SER
Integrated Safety Analysis
Present the DCS understanding and approach to performing and documenting ISA methodology
Natural Phenomena Hazards
Identify expected natural phenomena for which the MFFF is to be designed

Format: A brief presentation by DCS personnel of the issue(s) and a summary of DCS proposed approach (or options for resolution) as appropriate, followed by a discussion between DCS and NRC Staff.

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STONE & WEBSTER

MOX Fuel Fabrication Facility

NRC Technical Exchange

Design Basis
Quality Assurance Program
QA Classification & Quality Levels

Duke Cogema Stone & Webster

November 16, 1999

NRC Technical Exchanges
Objectives

- Exchange/discussion of technical issues
  - "Identification/resolution of technical issues" from 31 Aug 1999 meeting
- Initiate technical interactions in support of MOX-specific guidance
- Present proposed approach, solicit NRC feedback

NRC Technical Exchanges
Schedule & Topics

Session	Date	Topics
1	16 Nov 1999	Update status of licensing schedule/strategy Defining *design basis* for Construction Authorization & LA DCS Quality Assurance program SSC classification and quality levels
2	17 Nov 1999	Integrated Safety Analysis/Natural Phenomena Hazards Criticality Design
3	07 Dec 1999	Worker Dose Use of Polycarbonate Material in Glovebox Construction Definition of Site Boundary/Collocated Worker Implications
4	21 Dec 1999	Physical Security Material Control and Accountability/IAEA Requirements
5	11 Jan 1999	Radiation Protection HVAC and Confinement Fire Protection

Revised Licensing Schedule/Strategy

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- Background
  - Original planning based on previous 10 CFR 70 draft
  - Restoration of Pu-specific provisions, discussion in 31-Aug-99 meeting resulted in reassessment of DCS licensing schedule
- Revised schedule

– MFFF Request for Construction Authorization	September 2000
– MFFF Final Design	March 2002
– Construction Authorization	March 2002
– Final License Application Submittal	March 2003
– Complete Construction	March 2006
– Facility Startup	April 2006
– Commence batch irradiation at mission reactors	September 2007

Licensing Schedule/Strategy

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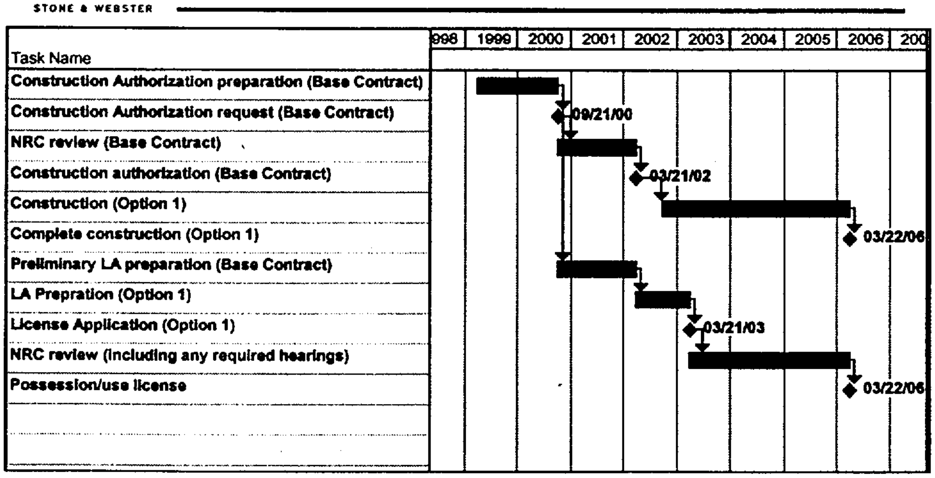

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Requirements for Construction Authorization (and Beyond)

1. Submittal and evaluation by NRC of environmental assessments [§70.23(a)(7) for CA]
2. Submittal and evaluation by NRC of design basis information described in §70.22(f) and QA program [§70.23(b) for CA]
3. Submittal and evaluation by NRC of the license application and related design information (described in remainder of §70.22, and incorporating new requirements of §70.61)
4. Confirmation by NRC of construction in accordance with LA [§70.23(a)(8)]

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NRC Technical Exchange

Design Basis

Richard Berry
November 16, 1999

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Design Basis

- Describe approach for establishing design basis
- Describe and identify supporting documents

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10 CFR 50.2

"Design bases means that 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 calculations and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals."

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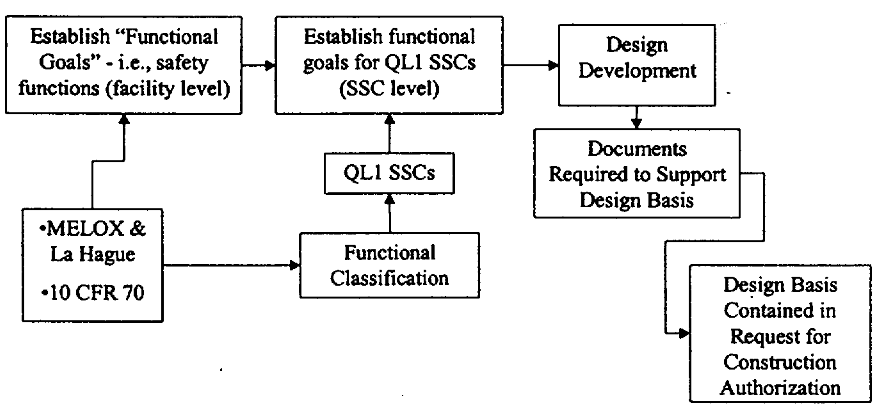
Design Basis Process



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Functional Goals
Quality Level 1 IROFS

- Provide Confinement System designs to prevent an unfiltered release of plutonium and associated chemical hazards.
- Provide system design features to prevent a criticality event.

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Confinement

- Static Confinement
  - Minimum of two boundaries for Pu.
- Dynamic Confinement
  - Required for gloveboxes and associated rooms containing Pu if not contained in qualified sealed containers.
  - Redundancy, separation and independence as required for dynamic confinement.

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Criticality

- Double Contingency
  - Requires at least two unlikely independent and concurrent changes in process conditions.
  - Mass and Moderation Control
  - Instrumentation and control systems with suitable redundancy and diversity to ensure high reliability.
- Geometrically Safe – preferred design approach

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Documents Supporting the Design Bases

Confinement

- Basis of Design Documents (the rules)
  - Site/Geotechnical
  - Structural
  - HVAC
  - Seismic
  - Electrical
  - Instrumentation and Controls
  - Integrated Safety Analysis
  - Equipment
  - MOX Process
  - Aqueous Polishing

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Documents Supporting the Design Bases

Confinement

- System Design Descriptions (the hows)
  - HVAC
  - Electrical
  - Instrumentation and Controls
  - Aqueous Polishing Descriptive Notes
  - MOX Process Descriptive Notes

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Documents Supporting the Design Bases

Confinement

- Reports and Plans
  - Functional Classification of Equipment (system level SSC summary)
  - Process Hazards Analysis (confirm safety classification of SSCs)
  - Seismology Report (basis for peak ground acceleration)
  - Electrical Independence Plan (plan for verifying that we meet separation and independence)
  - Preliminary Fire Hazards Analysis (includes definition of fire areas and ratings)
  - Radiation Zones

Documents Supporting the Design Bases

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Confinement

- Drawings and Diagrams
  - Site Plan
  - General Arrangement Drawings
  - Confinement Zone Drawings
  - HVAC Flow Diagram
  - Electrical One-Line
  - Instrumentation and Controls
- Calculations
  - HVAC Heating/Cooling Load
  - Preliminary Fire Loading
  - Structural Element Sizing
  - Preliminary Diesel (Standby and Emergency) Loads
  - Electrical UPS

Documents Supporting the Design Bases

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Criticality

- Basis of Design Documents
  - Nuclear Criticality Safety
  - Seismic
  - Instrumentation and Controls
  - Integrated Safety Analysis
  - Aqueous Polishing
  - MOX Process
- System Design Descriptions
  - Aqueous Polishing (Descriptive Notes)
  - MOX Process (Descriptive Notes)

Documents Supporting the Design Bases

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Criticality

- Reports and Plans
  - Process Hazards Analysis
  - Functional Classification of Equipment
  - Criticality Monitoring Plan
  - Criticality Evaluation Report
- Drawings and Diagrams
  - P&IDs (showing Quality Level 1 I&C systems)
  - Control Descriptions (related to P&IDs)
- Calculations
  - Criticality Calculations for all process steps

Values Chosen for Controlling Parameters

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- Confinement Zone Differential Pressure
- Peak Seismic Acceleration
- Other natural phenomena hazards
- Process Design Limits to Meet Mass and Moderation Control Requirements

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Typical Basis of Design Documents

Table of Contents

Note: Those Basis of Design Documents which support the Design Bases for the MFFF are expected to be completed prior to the submittal of the request for construction authorization in September, 2000.

November 16, 1999 NRC Technical Exchange Page 14

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Typical System Design Description

Table of Contents

Note: Sections 1.0, 2.0 and 3.0 and limited portions of other sections,are expected to be complete for Design Basis Structures Systems, and Components prior to submittal of the request for construction authorization in September 2000.

November 16, 1999 NRC Technical Exchange Page 15

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NRC Technical Exchange

DCS MOX Quality Assurance Program Overview

R.J. Brackett
November 16, 1999

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MOX QA Program Overview
Agenda

- DCS QA Program Basis
- Base Contract Authorizations
- DCS Organization
- QA Program Approvals
- MFFF Process Design QA
- QA Program Attributes
- Questions

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Basis & Scope

- DCS QA Program Basis
  - 10CFR50, Appendix B
  - NQA-1-1989 through NQA-1b-1991 Addenda
- MOX Project Quality Assurance Plan (MPQAP) controls base contract QA activities
- Base Contract Authorizations
  - MOX Fuel Fabrication Facility Design and Licensing
  - Fuel Qualification Program
  - Identification of Utility Modifications

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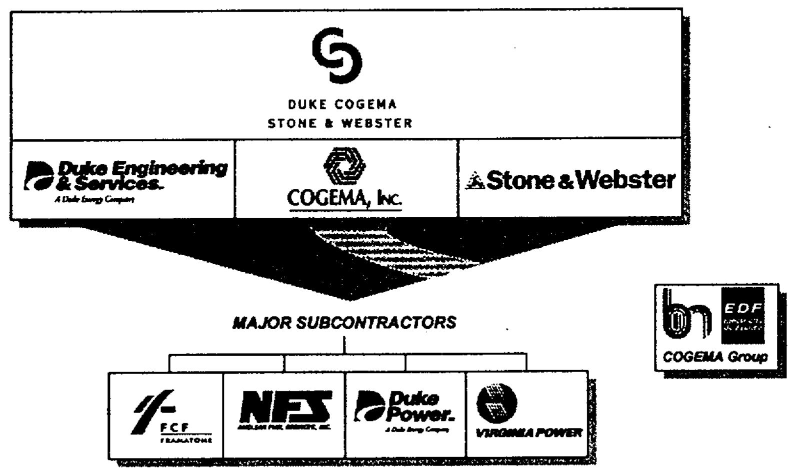
DCS Organization



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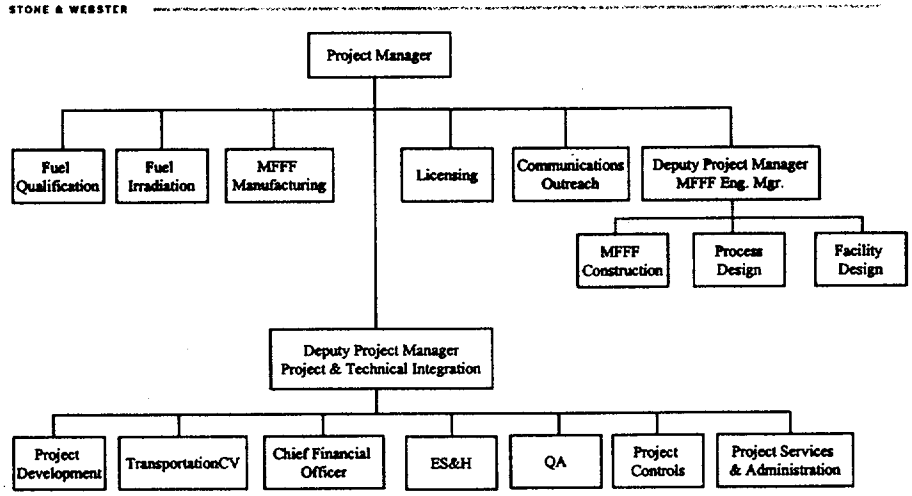
DCS Organization (continued)



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QA Program Approvals

DCS MPQAP	NRC
Duke Power/Virginia Power	NRC
Framatome COGEMA Fuels	NRC/DCS QA Manager
COGEMA/SGN	DCS QA Manager
Nuclear Fuel Services	DCS QA Manager

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MFFF Process Design QA

- Process Design output documents are produced using COGEMA design procedures & applicable portions of DCS project procedures
- COGEMA/SGN QA provides overview of activities
- Facilities Design performs design verification of QL-1 SSCs using DCS Project Procedures
- Final design deliverables reviewed by MFFF Engineering using DCS Project Procedures
- DCS QA performs overview (audits/surveillances) of Process Design Group activities

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QA Program Attributes

- Program development/implementation based on team experience
- Multiple work locations
- Although multiple QA Programs, final products are controlled by NRC approved QA Programs
- Four quality levels
- Heavy emphasis on self-assessment

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Questions

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NRC Technical Exchange

QA Classification and Quality Levels

Ray Fortier
November 16, 1999

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Technical Exchange Objective

- Describe Classification Approach
  - Nuclear Safety Philosophy
  - QA Program Basis
    - quality levels and criteria
  - System Engineering Approach
  - Integrated Safety Analysis Classification Process
  - SSC Classifications
- Solicit NRC Feedback
- For NRC Consideration in SRP Development

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Nuclear Safety Philosophy

- Safety Principles (Defense-in-Depth)
- Process Safety Information
- Americanization
- Integrated Safety Analysis
- Management Measures & Controls

Nuclear Safety Philosophy (Continued)

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- Safety Principles (Defense-in-Depth)
  - MELOX/La Hague Safety Principles & Classifications
  - Double Contingency (for criticality to occur)
    - requires at least 2 unlikely, independent, and concurrent changes in process conditions
  - Single Failure, Redundancy, Independence & Diversity
- Process Safety Information
  - Process Hazards
  - Process Technology
  - Process Equipment

Nuclear Safety Philosophy (Continued)

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- Americanization
  - Initially Classify SSCs with Deterministic Approach Based on MELOX/La Hague Designs
  - Apply U.S. Regulations, Codes & Standards
  - Conduct a Preliminary PHA Based on Nuclear Safety Philosophy
  - Confirm Preliminary PHA Results by Performing Risk Informed ISA
  - Maintain Updated ISA as a Living Document

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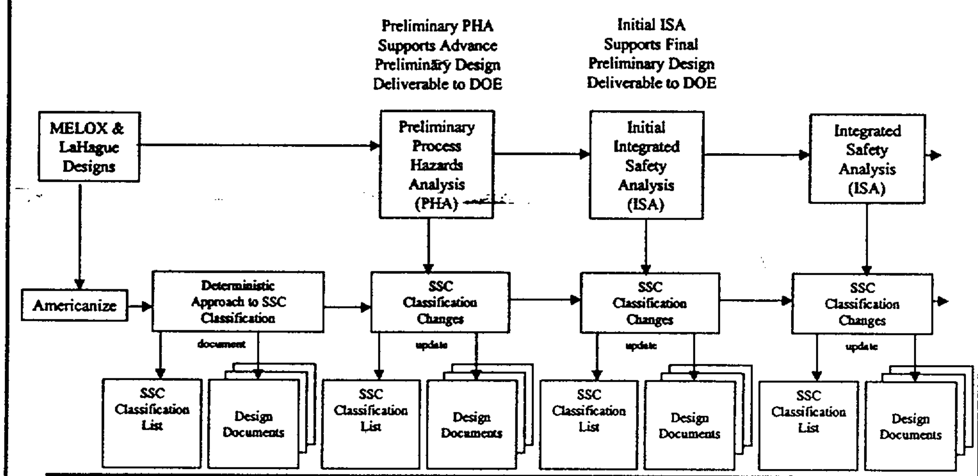
Nuclear Safety Philosophy (Continued)

Americanization & ISA Process



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Nuclear Safety Philosophy (Continued)

- Integrated Safety Analysis
  - Based on Risk Informed Logic
  - Identify Potential Hazards/Accidents
  - Analyze Hazards/Accidents
  - Evaluate Consequences and Likelihood of Hazards/Accidents
  - Identify SSCs Needed to Prevent/Mitigate Hazards/Accidents (design then administrative controls)
  - Identify Quality Level of SSCs
  - Identify SSC-Sensitive Operations/Maintenance (equivalent "technical specification" requirements)

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Nuclear Safety Philosophy (Continued)

- Management Measures & Controls
  - Configuration Management
  - SSC Maintenance
  - Training & Qualifications
  - Procedures
  - Audits & Assessments
  - Incident Investigations
  - Records Management
  - Other QA Elements

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QA Program Basis

- 10CFR50, Appendix B & ASME NQA-1
- Graded QA Approach with 4 SSC Quality Levels
  - Quality Level (QL) 1
    - Items (SSCs) Relied on for Safety (IROFS)
    - SSCs Relied on for Unlikely or Not Unlikely High Consequence Events (HCEs)
    - SSCs Relied on for Not Unlikely Intermediate Consequence Events (ICEs)
  - QL 2
    - SSCs Relied on for Unlikely ICEs
    - SSCs Relied on for Not Unlikely Low Consequence Events (LCEs)

QA Program Basis (Continued)

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- QL 3
  - SSCs Relied on for Unlikely LCEs
  - SSCs Relied on for Operational Performance (including maintenance & reliability)
- Conventional Quality (CQ)
  - SSCs that are not QL 1, 2 or 3

- SSC QL Classification Criteria
  - 10 CFR 70.61 Performance Requirements
  - Iterative Process
    - Deterministic
    - PHA
    - ISA

QA Program Basis (Continued)

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Quality Level	Consequence Category	Event Likelihood	Consequence Screening Criteria: Workers	Consequence Screening Criteria: Offsite Public	Consequence Screening Criteria: Environment	Consequence Screening Criteria: Plant Performance	Process Classification Level (Reference)
Quality Level 1 High Safety Significance. Relied on for high risk events.	High	Unlikely or Not Unlikely	D = 1 Sv (100 rem) = AEGL3, ERPG3	D = 0.25 Sv (25 rem) = 30 mg sol U intake = AEGL2, ERPG2	NA	NA	F1
	Intermediate	Not Unlikely	0.25 Sv = D < 1 Sv = AEGL2, ERPG2 but < AEGL3, ERPG3	0.05 Sv = D < 0.25 Sv = AEGL1, ERPG1 but < AEGL2, ERPG2	radioactive release >5000 x 10 CFR 20, Appendix B, Table 2	NA	F2
Quality Level 2 Low Safety Significance. Relied on for intermediate risk events.	Intermediate	Unlikely	0.25 Sv = D < 1 Sv = AEGL2, ERPG2 but < AEGL3, ERPG3	0.05 Sv = D < 0.25 Sv = AEGL1, ERPG1 but < AEGL2, ERPG2	radioactive release >5000 x 10 CFR 20, Appendix B, Table 2	NA	WSF*
	Low	Not Unlikely	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	NA	
Quality Level 3 Occupational Exposure Significance or Performance Significance. Relied on for low risk events.	Low	Unlikely	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	Cost > $XX Or Down Time > XX days	WSF
Conventional Quality	N/A	N/A	N/A	N/A	N/A	N/A	

ERPG: Emergency Response Planning Guidelines AEGL: Acute Exposure Guideline Levels

DEFINITIONS:

F1 - WHERE A FUNCTION IMPORTANT TO SAFETY (FIS) IS PERFORMED BY A SINGLE SSC, THIS SSC IS CLASSIFIED AS F1.

F2 - WHERE A FIS IS PERFORMED BY TWO REDUNDANT, INDEPENDENT AND SEPARATE SSCs, THIS IS CLASSIFIED AS F2.

WSF* - Where a failure of a SSC which does not contribute to a safety function (SSC classified as WSF) involves the loss of a SSC necessary to ensure an FIS, this WSF SSC has to satisfy a particular safety requirement subject to Quality Assurance: the SSC is classified WSF*.

WSF - Where a safety function is achieved by several identical redundant SSCs, at least one of them will be classified F2 while the others can be classified WSF (without safety function).

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QA Program Basis (Continued)

Quality Level	Consequence Category	Event Likelihood	Consequence Screening Criteria				Process Classification Level (Reference)
			Workers	Offsite Public	Environment	Plant Performance	
Quality Level 1 High Safety Significance. Relied on for high risk events.	High	Unlikely or Not Unlikely	D =1 Sv (100 rem) = AEGL3, ERPG3	D = 0.25 Sv (25 rem) = 30 mg sol U intake = AEGL2, ERPG2	NA	NA	F1
	Intermediate	Not Unlikely	0.25 Sv = D < 1 Sv = AEGL2, ERPG2 but < AEGL3, ERPG3	0.05 Sv = D < 0.25 Sv = AEGL1, ERPG1 but < AEGL2, ERPG2	radioactive release >5000 x 10 CFR 20, Appendix B, Table 2	NA	F2
Quality Level 2 Low Safety Significance. Relied on for intermediate risk events.	Intermediate	Unlikely	0.25 Sv = D < 1 Sv = AEGL2, ERPG2 but < AEGL3, ERPG3	0.05 Sv = D < 0.25 Sv = AEGL1, ERPG1 but < AEGL2, ERPG2	radioactive release >5000 x 10 CFR 20, Appendix B, Table 2	NA	WSF*
	Low	Not Unlikely	accidents of lesser radio-logical and chemical exposures to workers than those above in this column	accidents of lesser radio-logical and chemical exposures to the public than those above in this column	radioactive releases producing effects less than those specified above in this column	NA	
Quality Level 3 Occupational Exposure Signifi-cance or Perform-ance Significance. Relied on for low risk events.	Low	Unlikely	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	Cost > $XX Or Down Time > XX days	WSF
Conventional Quality	N/A	N/A	N/A	N/A	N/A	N/A	

ERPG: Emergency Response Planning Guidelines AEGL: Acute Exposure Guideline Levels

DEFINITIONS:

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WSF – Where a safety function is achieved by several identical redundant SSCs, at least one of them will be classified F2 while the others can be classified WSF (without safety function).

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System Engineering Approach

- Hierarchy of Design Documents (Design Process)
  - DOE Contract/SOW
  - Design Requirements Document
    - plant level analysis (quality level determination)
  - Basis of Design Documents
  - System Description Documents
    - system level analysis (quality level determination)
  - Other Design Documents
    - calculations
    - drawings
    - specifications & other technical documents

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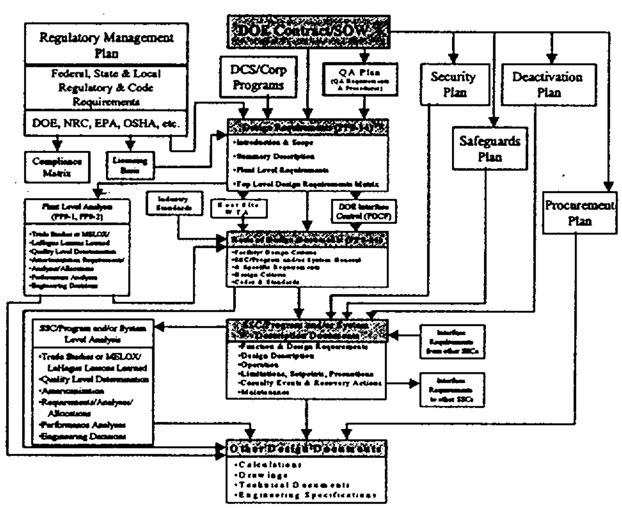
System Engineering Approach (Continued)



System Engineering Approach (Continued)

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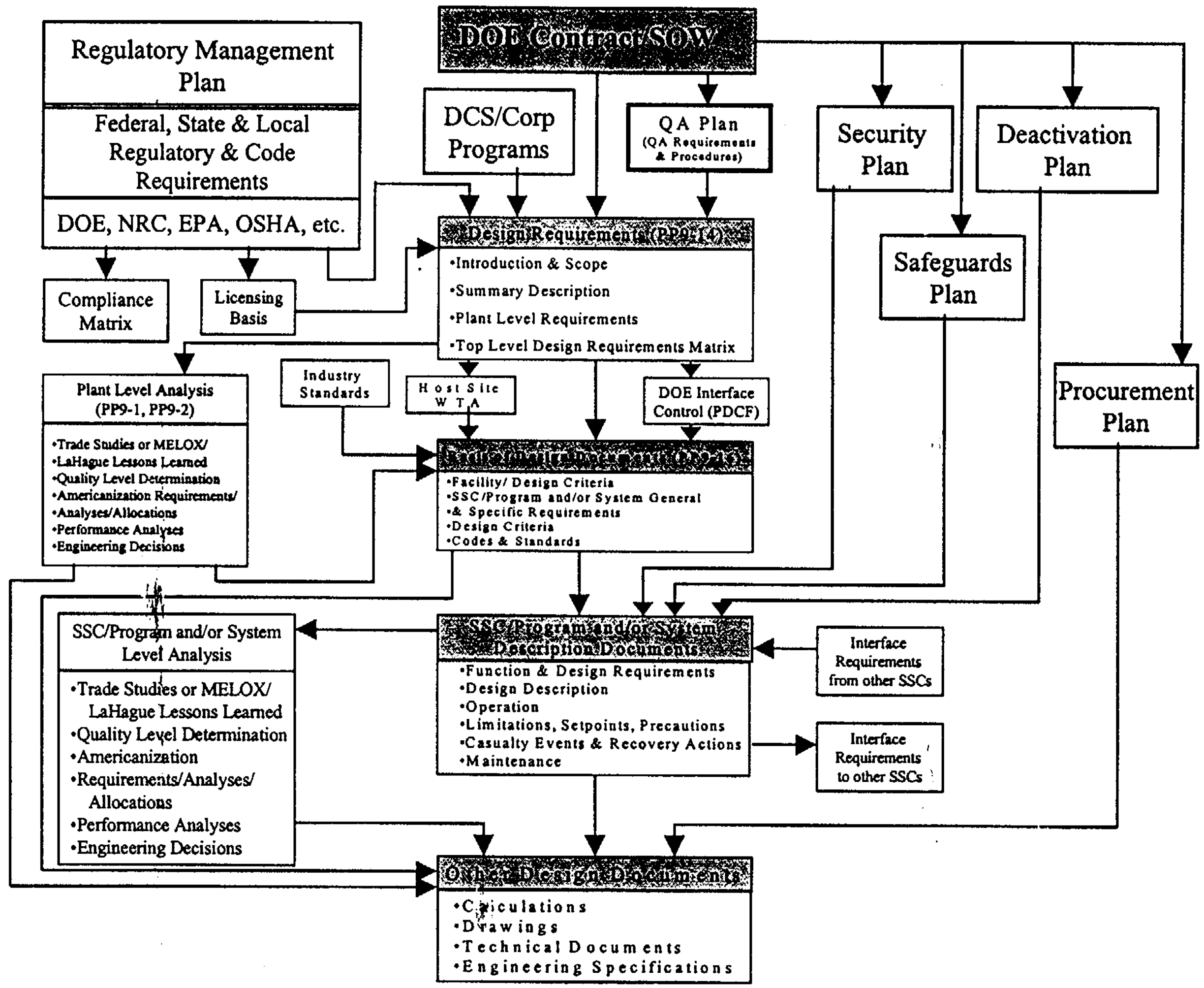
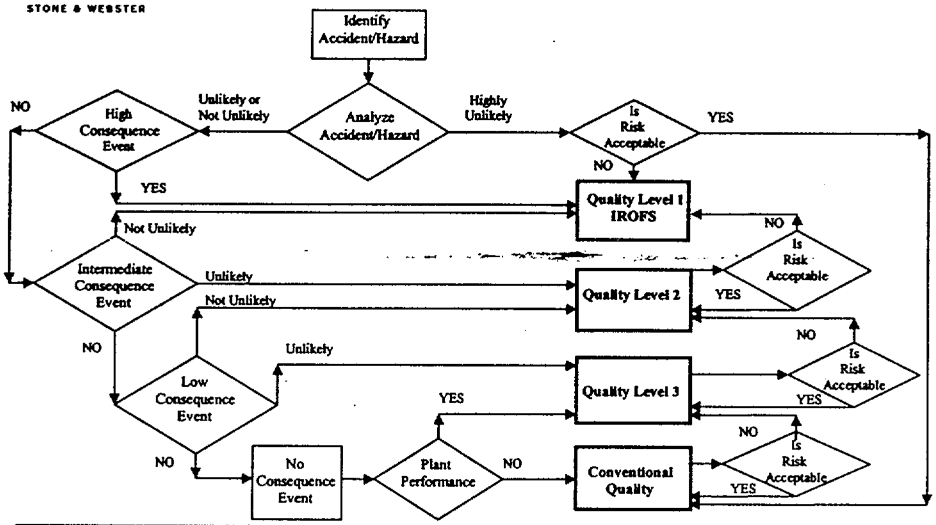


Integrated Safety Analysis Classification Process

DUKE COGEMA STONE & WEBSTER



SSC Classifications

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- Deterministic Functional Classification of SSCs
  - Typical Quality Level 1 (IROFS) SSCs
    - MOX fuel fabrication building
    - glovebox static confinement boundary
    - glovebox dynamic confinement filtered exhaust vent. system
    - emergency power system
    - glovebox low differential pressure control system
  - Typical Quality Level 2 SSCs
    - 2 over 1 SSCs
    - fire protection system

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SSC Classifications (Continued)

- Typical Quality Level 3 SSCs
  - area radiation monitors
  - key process equipment
- Typical Conventional Quality SSCs
  - administration building
  - domestic water system

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MOX Fuel Fabrication Facility

NRC Technical Exchange

Process Overview
Integrated Safety Analysis
Criticality

Duke Cogema Stone & Webster

November 17, 1999

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NRC Technical Exchanges
Objectives

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  - "Identification/resolution of technical issues" from 31 Aug 1999 meeting
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NRC Technical Exchanges
Schedule & Topics

Session	Date	Topics
1	16 Nov 1999	Update status of licensing schedule/strategy Defining *design basis* for Construction Authorization & LA DCS Quality Assurance program SSC classification and quality levels
2	17 Nov 1999	Integrated Safety Analysis/Natural Phenomena Hazards Criticality Design
3	07 Dec 1999	Worker Dose Use of Polycarbonate Material in Glovebox Construction Definition of Site Boundary/Collocated Worker Implications
4	21 Dec 1999	Physical Security Material Control and Accountability/IAEA Requirements
5	11 Jan 1999	Radiation Protection HVAC and Confinement Fire Protection

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NRC Technical Exchange

Integrated Safety Analysis

Bill Hennessy
Dave Noxon
November 17, 1999

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ISA BASIS OF DESIGN

- Proposed 10 CFR 70, NUREG-1520, NUREG-1513
- §70.64 baseline rqmts, §70.61 performance rqmts, §70.62 safety pgm/ISA
- IROFS definition, likelihood & consequence criteria
- OSHA 29 CFR 1910, EPA 40 CFR 68
- chemical process safety

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ISA Phases

- Preliminary PHA
- ISA completion during design phase
- License application / ISA Summary
- ISA maintenance during construction & is a living document to be used throughout the life of the facility

Nuclear Safety Philosophy

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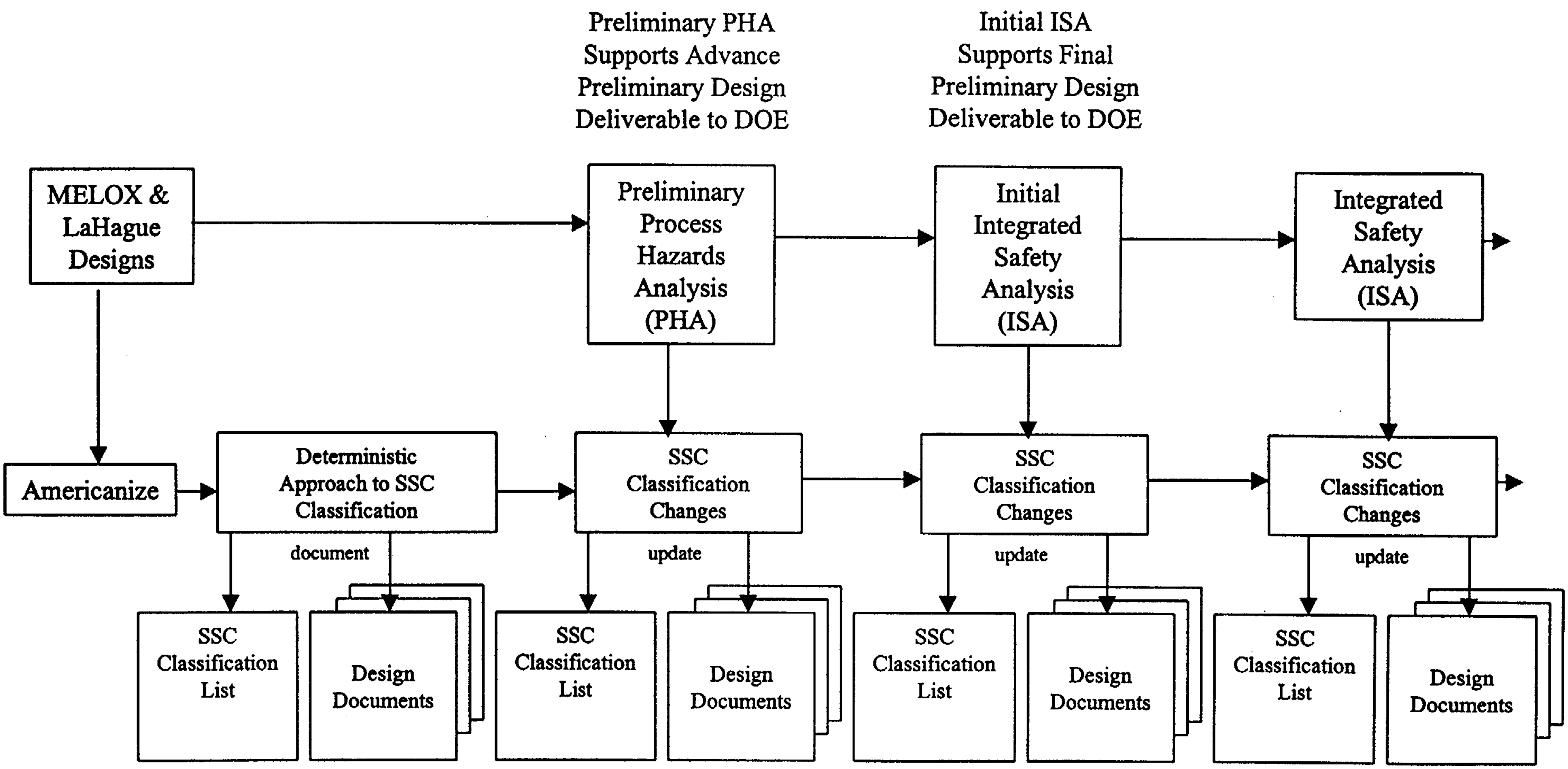
Americanization & ISA Process



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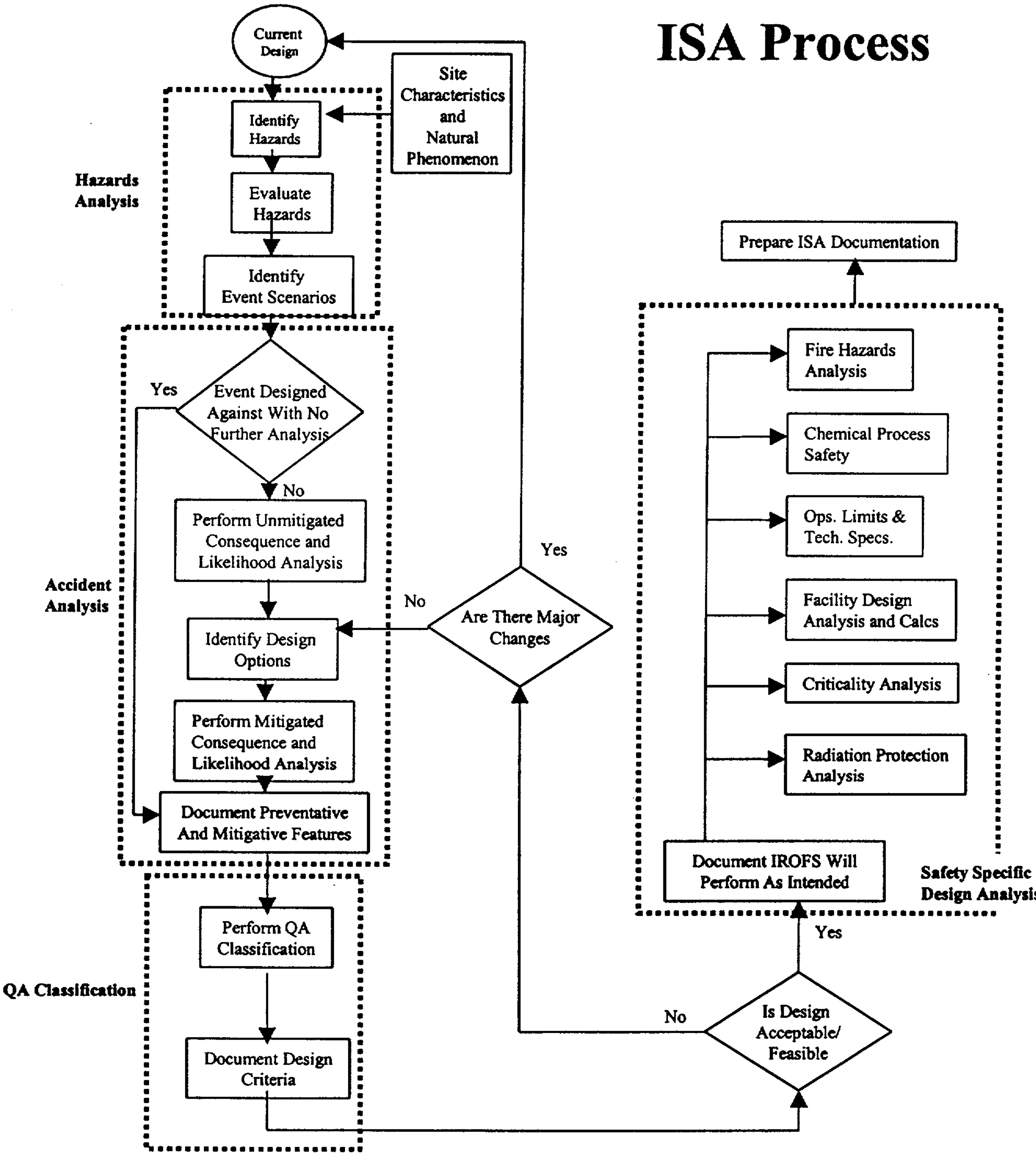
ISA Process



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ISA USAGE

- Risk Informed Decision Process
- Plant Performance Acceptability
- IROFS confirmation
- QA Classification confirmation
- Living plant evaluation tool / configuration management

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ISA Team Experience

- Extensive NRC and DOE experience
  - Licensing
  - Safety Analysis
  - Design Basis
  - Chemical Process Safety
- Process Group Safety Analysis
  - La Hague (Aqueous Polishing)
  - MELOX (MOX)

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ISA Support Team

- Process Group
  - Chemical Process
  - MOX Process
  - Fire Protection
  - Radiation Protection
  - Criticality
  - Confinement

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ISA Support Team

- Facility Group
  - Mechanical
  - Electrical
  - Instrumentation & Controls
  - Civil Structural/ Geotechnical
  - Fire Protection
  - Chemical Process
  - Radiation Protection
  - Criticality Engineering

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ISA /PHA Process

- Hazards Analysis
- Accident Analysis
- QA Classification
- Design Verification
- Safety Documentation

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Preliminary PHA
Hazards Identification Process

- MOX Preliminary PHA
  - Based on MELOX Safety Analysis
  - Focus on MOX USA differences
  - High level PHA approach
  - Review of industry data for completeness
- Aqueous Polishing Preliminary PHA
  - Based on La Hague Safety Analysis
  - More differences between AP & reference facility
  - More traditional HAZOP/ What-If analysis
  - Review of industry data for completeness

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Hazards

- **MELOX / La HAGUE HAZARDS**
  - **Nuclear risks :**
    - dispersal of nuclear materials
    - external exposure
    - criticality
    - thermal release
    - radiolysis

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Hazards

- **MELOX / La HAGUE HAZARDS**
  - **Internal Non-Nuclear Risks :**
    - fire
    - internal flooding
    - explosion
    - power or fluid supply failure
    - pressure vessels
    - load handling
    - chemical products
    - electrical equipment
    - heating and cooling fluids

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Hazards

- **MELOX / La HAGUE HAZARDS**
  - **External Non-Nuclear Risks :**
    - earthquake
    - flooding
    - aircraft crash
    - transportation and nearby facility accidents
    - extreme weather

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Hazard Sample

- Event Type
  - **Dispersal of Nuclear Materials**
- Location/SSC
  - **Glovebox**
- Cause
  - **<u>Pressure Higher/Lower than Design</u>**
    - Nitrogen Pressure Regulation Failure
    - Maintenance Error
    - HVAC Flow Perturbation

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Consequence

	Workers	Offsite Public	Environment
Consequence Category 3: High	D>1 Sv (100 rem) >ABGL3, ERPG3**	D>.25 Sv (25 rem) 30 mg sol U intake >ABGL2, ERPG2**	
Consequence Category 2: Intermediate	.25 Sv<D≤1 Sv >ABGL2, ERPG2** but <ABGL3, ERPG3**	.05 Sv<D≤.25 Sv >ABGL1, ERPG1** but <ABGL2, ERPG2**	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

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Likelihood

- Highly Unlikely
  - One or more highly reliable passive engineered features
  - Two redundant and independent active engineered features
  - Three or more active similar engineered features
- Unlikely
  - Redundant engineered features
  - Enhanced administrative controls
- Not Unlikely
  - Can be expected to occur in the plant life
  - Simple administrative Controls
  - Active equipment failures
- Credible
  - Based on highly unlikely

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Risk Matrix

	Likelihood Category 1: highly unlikely	Likelihood Category 2: unlikely	Likelihood Category 3: not unlikely
Consequence Cat. 3 High	3 acceptable	[redacted]	[redacted]
Consequence Cat. 2 Intermediate	2 acceptable	4 acceptable	[redacted]
Consequence Cat. 1 Low	1 acceptable	2 acceptable	3 acceptable

QA Classification

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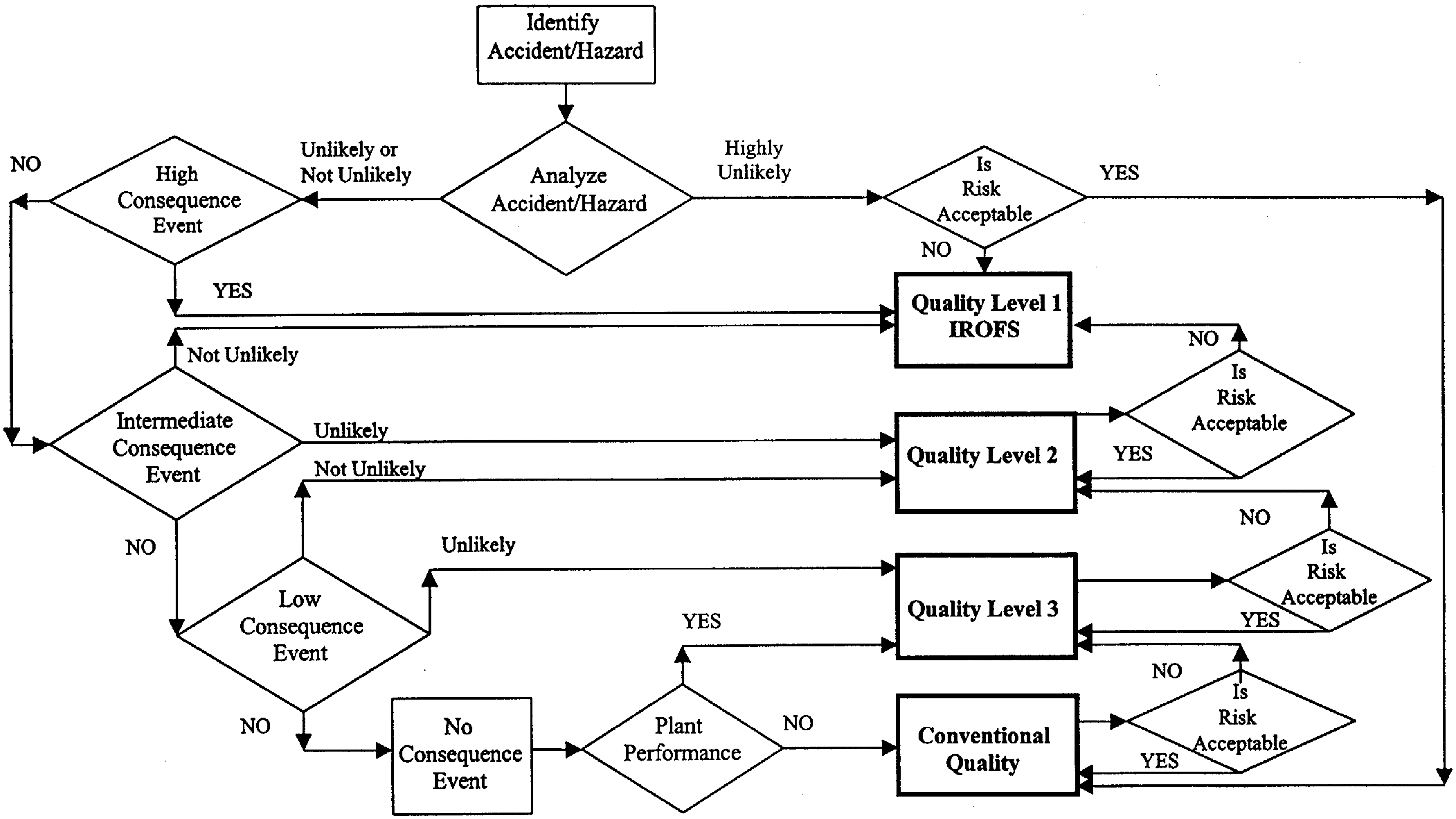

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QA Classification

- Functional/Deterministic Classification of SSCs
  - Typical Quality Level 1 (IROFS) SSCs
    - MOX fuel fabrication building
    - glovebox static confinement boundary
    - glovebox dynamic confinement filtered exhaust vent. system
    - emergency power system
    - glovebox low differential pressure control system
  - Typical Quality Level 2 SSCs
    - 2 over 1 SSCs
    - fire protection system

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QA Classification (Continued)

- Typical Quality Level 3 SSCs
  - area radiation monitors
  - key process equipment
- Typical Conventional Quality SSCs
  - administration building
  - domestic water system

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Design Verification/ Acceptability

- Engineering/Feasibility
- Cost
- Overall Plant Risk

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PHA Documentation

- Hazard Description
- Specific causes of concern for the hazard
- Unmitigated/unprevented risk
- Prevention features
- Mitigation features
- Specific Plant impacts of concern for the hazard
- Risk after controls have been applied
- Initial confirmation and justification of IROFS

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Hazard Table - Sample

Event Type	*Location/SSC*	*Cause*	*Risk* *No Controls*	*Prevention*	*Mitigation*	*Risk* *With Controls*
Dispersal of nuclear materials	Glovebox	<u>Pressure higher/lower than design pressure</u> 1. Nitrogen pressure regulation failure 2. Maintenance error 3. HVAC flow perturbation	Not Unlikely *Worker* H *Facility* L *Public* L *Risk Level* 9	1. Safety valves for overpressure and underpressure 2. Minimum and maximum design pressure 3. Elimination of high pressure/ high volume lines from inside gloveboxes.	1. Pressure sensor generating alarm 2. C3b static and dynamic confinement 3. Air monitoring 4. Facility evacuation procedure	Unlikely *Worker* I *Facility* L *Public* L *Risk Level* 4
External Exposure	source	<u>Increment of PuO2 Powder dust</u>	Not Unlikely *Worker* L *Facility* L *Public* L *Risk Level* 3	1. Dedusting systems fixed or mobile in glove box 2. Leaktight design for the main equipment of powder transfer 3. Powder dust capture near production 4. Regular cleaning of glove boxes	1. Radioprotection shields 2. Health physics monitoring 3. Facility evacuation procedure	Unlikely *Worker* L *Facility* L *Public* L *Risk Level* 2
Criticality	Units with mass control	<u>Critical mass reached</u> 1. Fail to control fissile material mass balance (input vs. output) 2. Slow, undetected fissile material accumulation (i.e., contamination) outside of jar, hopper, dosing equipment 3. Improperly characterized fissile material	Unlikely *Worker* H *Facility* I *Public* L *Risk Level* 6	1. Allowable mass less than critical mass 2. Total mass weighing 3. Mass balance 4. Bar code traceability	1. C3b static and dynamic confinement (filtration (two filters) system) 2. Criticality monitoring 3. Facility evacuation procedure 4. Wall thickness 5. Safe haven	Highly Unlikely *Worker* H *Facility* I *Public* L *Risk Level* 3

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Natural Phenomenon Recurrence Frequency

- Historic Precedents
  - NRC license facilities
  - DOE facilities
- 'Highly Unlikely' frequency factors
  - margin of safety in SSCs
  - initiating event

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Nuclear Criticality Safety
for the MOX Fuel Fabrication Facility

Laurence Cret, Process Group

Jim Thornton, Facility Group

Bob Foster, Facility Group

November 17, 1999

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Nuclear Criticality Safety-Agenda

- Criticality safety evaluation & analysis design approach
- Interface between the Process Group and the Facility Group
- Overview of MOX Fuel Fabrication Facility (MFFF)
- Design principles regarding the criticality risk including an overview of the MFFF areas where there are criticality evaluations/analyses planned and the expected control modes.
- Preparation of the criticality safety evaluations
- Approach to benchmarking
- AVLIS SER lessons learned
- Japanese criticality accident lessons learned
- Summary

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NCS-Design Analysis Approach

- Criticality Safety Evaluation Methodology
- Criticality Safety Criteria
- Criticality Control Modes
- Criticality Safety Programs
- Benchmark Determination

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Criticality Safety Evaluation Methodology

- NCSEs prepared according to standard US procedures and criticality methodologies (based on ANSI/ANS-8.1as invoked by RG 3.71)
- U.S. standard criticality code (KENO) and neutron cross-sections (238 group) included in SCALE 4.4 applied
- NCSEs originated by the Process Group
- NCSEs independently reviewed by the Facility Group

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Criticality Safety Criteria

- Double contingency principle compliance
- Criticality Analysis
  - Upper Safety Limit (USL)
    - Administrative safety margin, Δk_m Justification, typically 0.05
    - Account for method bias and uncertainty based on statistical analysis of applicable benchmark experiment results
  - Credible worst-case treatment and/or statistical accounting for design mechanical, material, and fabrication uncertainties
- Single parameter limits of ANSI/ANS-8.1

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Criticality Control Modes

- Geometry control whenever possible
- Mass and moderation control when required for process and operability reasons

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Benchmark Determination-Process

- Selection
  - Cover range of diversity in MFFF applications:
    - High moderated Pu oxide
    - Pu nitrate
    - Pu oxalate
    - Low moderated oxide powders
    - Arrays of pellets or rods
- Data Analysis
  - Confirmation of areas of applicability
  - Determination of method bias and uncertainty

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Benchmark Determination-Selection

- Available Benchmark Experiments
  - OECD International Criticality Benchmark Handbook (e.g.):
    - Aqueous Solutions - PU-SOL-THERM-XXX
    - Plutonium-Metal - PU-MET-FAST-XXX
    - PuO_2/Polystyrene Slabs - PU-COMP-MIXED-001, -002
    - MOX Pins - MIX-COMP-THERM-005, -009
    - Intermediate Energy Pu Experiments - MIX-MET-INTER-001
  - EPRI Clean Critical Experiments (UO2 and MOX pins in water)
  - SAXTON Partial Plutonium Core (UO2 and MOX pins in water)

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Benchmark Determination-Analysis

- Validation will apply trending analysis of applicable parameters to assure conservative treatment of method bias and uncertainty (criticality benchmark guidance presented in NUREG/CR-6361 and NUREG/CR-6102)
- Sensitivity and uncertainty techniques applied as necessary (Draft NUREG/CR-5593)
  - Demonstrate benchmark experiment similarity to design applications
  - Justify safety margin adequacy where data scarce or significant extrapolation necessary

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Criticality Safety Programs

- Administrative programs in accordance with ANS-8.19-1996 – Administrative Practices for Nuclear Criticality Safety.
- QA program in accordance with ANS-8.19-1996 – Administrative Practices for Nuclear Criticality Safety.
- Training program in accordance with ANS-8.20-1991 –Nuclear Criticality Safety Training.
- Operational inspections, audits, assessments, and investigations function to be regularly performed in accordance with standard NCS principles

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DCS Criticality Safety

Roles and Responsibilities

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DCS Roles and Responsibilities

- Process Design Team
  - Establish basic process flows and material throughputs
  - Develop equipment concepts and facility layouts
  - Establish preliminary functional requirements
- Facilities Design Team
  - Develop design criteria based on US requirements
  - Establish functional classifications and quality req'mnts
  - Develop site specific facilities requirements
  - Prepare License Application

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NCS Process Group

- Safety Analysis Group
  - Determine assumptions for normal/off-normal events
  - Provide/confirm assumptions used in NCSE
  - Provide input to ISA
- Criticality Safety Evaluation Group
  - Performs studies to evaluate MFFF design
  - Originate NCSE using US standard criticality methods
  - Perform validation using appropriate benchmarks

NCS Facility Group

- Provide guidance on US standard methods
- Provide guidance on criticality benchmarks
- Review/confirm NCSEs using independent analyses and methods
- Review validation of NCSEs to benchmarks
- Provide input to ISA
- Prepare criticality License Application information

MOX Fuel Fabrication Facility (MFFF) Criticality control design principles

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Criticality control design principles (1/2)

- Split the facility in criticality control units
- For each unit:
  - Define the reference fissile medium (e.g. PuO_2+H_2O, Pu nitrate ...)
  - Define the criticality control mode (e.g. geometry, mass, moderation ...)

  ➔Calculate the allowed range for the parameters of the control mode (e.g. dimensions, mass, %H_2O)

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Criticality control design principles (2/2)

- For each unit (cont'd):
  - Guarantee that the parameters of the control mode (+parameters defining reference fissile medium) remain in the allowed range by:
    - Design
    - Operation control

  ➔**Double contingency principle**

Definition of the reference fissile media (1/4)

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- A fissile medium is defined by:
  - Chemical form
  - Pu and/or U isotopics
  - Maximum density (for powders)
  - %Pu (MOX Process)

Definition of the reference fissile media (2/4)

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The reference fissile medium for each unit is defined as follows:

- Chemical form:
  - Aqueous Polishing:

    A safe side assumption is made taking into account the nominal conditions, but also possible process upsets (e.g. unwanted Soda introduction that may cause precipitates …)
  - MOX Process:

    No chemical transformations → oxide form is always assumed

Definition of the reference fissile media (3/4)

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The reference fissile medium for each unit is defined as follows (cont'd):

- Pu and U isotopics:

  A safe-side assumption is made knowing the range of isotopics that will be handled by the facility:

	Nominal range	Used in criticality calculations
Pu236 / Pu total	< 1 ppb	0
Pu238 / Pu total	< 0.05%	0
Pu239 / Pu total	90 - 95%	96%
Pu240 / Pu total	5 - 9%	4%
Pu241 / Pu total	< 1%	0
U235 / U total	0.25%	0.3 %
U238 / U total	99.75%	99.7%

Definition of the reference fissile media (4/4)

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The reference fissile medium for each unit is defined as follows (cont'd):

- Powder maximum densities:

  Safe-side assumptions are made for the different types of products ("fresh" powders, ball-milled master mix, final mix, recylced scraps ...) based on MELOX experience feedback

- %Pu:
  - Safe side assumptions made based on process values:

	Nominal range	Used in criticality calculations
%Pu in Master Mix	20%	22%
%Pu in Final Mix	2.3% - 4.8 %	6.3%
	(design for up to 6%)	

  - Parameter to be guaranteed during operation

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Choice of the criticality control mode (1/9)

- Possible control modes are:
  - Geometry (shape and size)
  - Mass
  - Moderation
  - Concentration
  - Supplemental neutron absorber

  or a combination of these modes

November 17, 1999 NRC Technical Exchange Page 21

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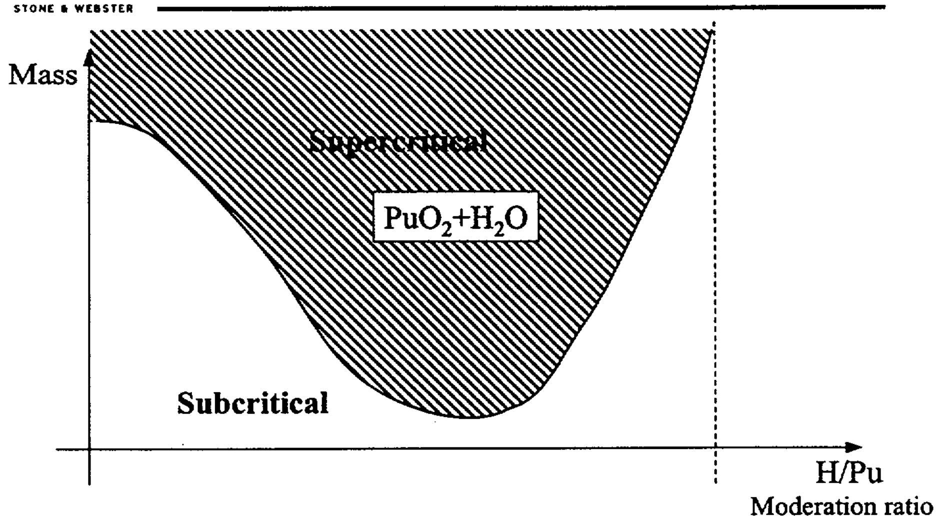
Choice of the criticality control modes (2/9)



November 17, 1999 NRC Technical Exchange Page 22

Choice of the criticality control modes (3/9)

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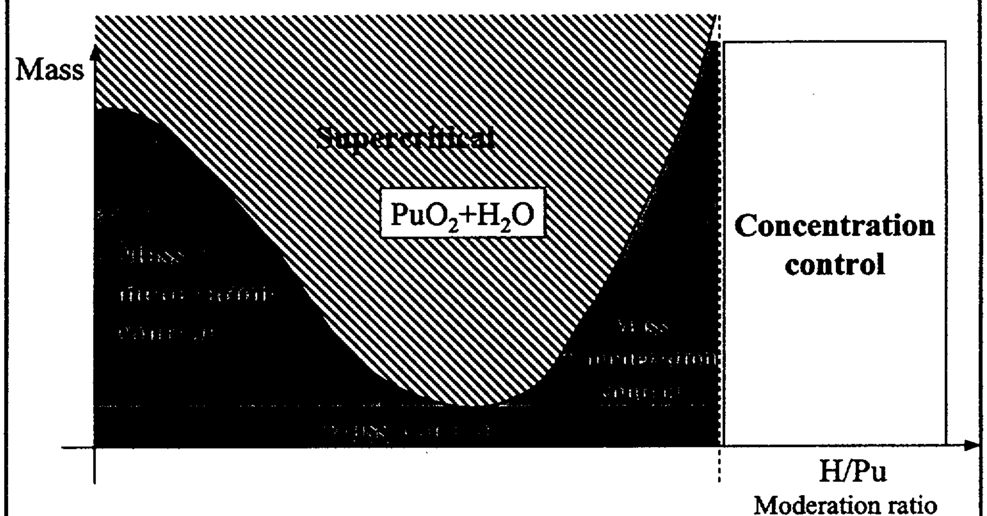

Choice of the criticality control modes (4/9)

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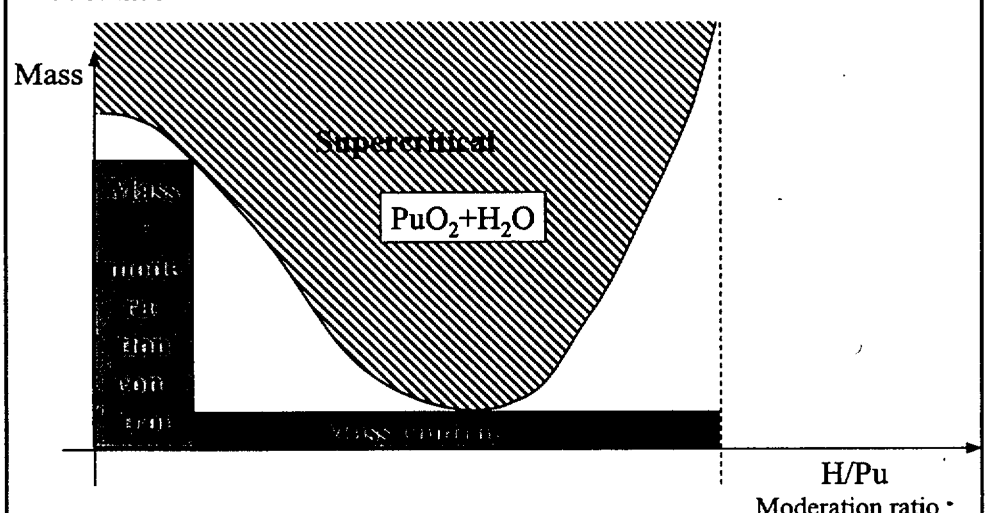

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Choice of the criticality control modes (5/9)

- Geometry control
  - Is used:
    - For storages (large quantities of fissile matter)
    - For process equipment if compatible with their process function (i.e. in Aqueous Polishing, in some pellet / rod handling equipment)
  - Implies:
    - Thorough control of equipment dimensions upon fabrication
    - Accidental situations taken into account:
      - Aseismic design of geometry
      - Criticality-safe design of drip trays in Aqueous Polishing

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Choice of the criticality control modes (6/9)

- Moderation control (MOX Process)
  - Is used:

    Combined with mass control
    - For some MOX Process equipment (when their needed capacity is not compatible with mass control alone: powder area, some units in the pellet and rod areas)
  - Implies:
    - Control of organic materials (pore-former, lubricant) added to the powder
    - No fluids admitted in process cells
    - If fluids are necessary for process :
      - Double barrier between fluids and fissile matter, or
      - Use of fluids with no hydrogen, or
      - Limited amount of fluid

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Choice of the criticality control modes (7/9)

- Mass control
  - Is used:

    Generally combined with moderation control*

    - For some MOX Process equipment (when their process function is not compatible with geometry control: powder area, some units in the pellet and rod areas)
  - Implies:
    - Limitation of the mass that can be handled in each unit
    - Control of the mass during operation:

      Weighing, mass balances

* Allowable mass with moderation control is higher than without moderation control: see figure on previous slide

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Choice of the criticality control modes (8/9)

- Concentration control (Aqueous Polishing)
  - Is used:
    - For equipment processing solutions with a very low concentration (liquid waste)
  - Implies:
    - Low nominal concentration
    - Control of the concentration during operation

Choice of the criticality control modes (9/9)

- Supplemental neutron absorber control
  - Is used:

    Combined with geometry control

    - For Aqueous Polishing vessels (reflection mitigation) to increase allowable dimensions (so that the process functions can be satisfactorily performed)
    - For storages (neutronic isolation) in order to allow for a more compact arrangement
  - Implies:
    - Thorough control of shields upon fabrication
    - Accidental situations taken into account:
      - Aseismic design
      - If needed, protection of shields against high temperatures (i.e. loss of H)

Control of criticality parameters through design and operation (1/3)

The most important practical implications of the **double contingency principle** are:

- Aqueous polishing:
  - Controls for a transfer from a safe geometry vessel to an ordinary geometry vessel:

    Double concentration control (e.g. follow up of process parameters + sampling before transfer)
  - Criticality-safety design of drip-trays to collect potential leaks
  - Controls to guarantee chemical form (i.e. fissile medium)

    e.g. after dissolution: double control of absence of PuO_2 in receiving tank + interlock

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Control of criticality parameters through design and operation (2)

The most important practical implications of the **double contingency principle** are (cont'd):

- MOX Process:
  - Design controls used whenever possible:
    - Geometry control mode
    - No fluid pipes in process rooms ; if fluid needed for process equipment, double wall or reduced quantity
    - Master Mix and Final Mix jar docking devices are different
  - Operation controls used for:
    - Pu content
    - Mass
    - Moderation (organic additives)

Replacements pages
For criticality safety
presentation.

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Control of criticality parameters through design and operation (3/3)

The most important practical implications of the **double contingency principle** are (cont'd):

- MOX Process (cont'd):
  - Operation controls distinguish 2 types of parameters:
    - Parameters with a double control (e.g. Pu contents, mass of Pu per jar) by the normal operating system* + a specific means (e.g. "criticality PLC (Programable Logic Controller)":
      - ↳ are considered as guaranteed in the safety analyses
    - Parameters with a single control (e.g. mass balances) by the normal operating system*:
      - ↳ allowed level is well below calculated "admissible" level (i.e. well below level corresponding to USL)
      - ↳ exceedance of allowed level is analyzed in the safety analyses

** Normal operating system = operator + normal PLC's + computerized production management system*

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Application to the MOX FFF: orders of magnitude

For Information Only

Acceptable dimensions for W-Pu (optimum moderation)

	Reflector (water)	PuO2 (d <= 7)	Pu(NO3)3
Sphere volume	20 cm	**1.5 lit.**	**7.0 lit.**
	2.5 cm	**2.5 lit.**	**10 lit.**
Cylinder diameter	20 cm	**8.5 cm**	**15 cm**
	2.5 cm	**11 cm**	**18 cm**
Slab thickness	20 cm	**2.4 cm**	**5.6 cm**
	2.5 cm	**5.1 cm**	**9.6 cm**

Corresponding to k_{eff} = 0.93, 96% ^{239}Pu, calculated with French codes

Acceptable mass of PuO_2 at optimum moderation: **390 gPu** (reflector: 20 cm water)

"Safe" masses of oxide (3% moderation)

	PuO2 (d = 3,5)	Master Mix (d = 5,5)	Final Mix (d = 3,5)	Pellets (d = 11)
R-Pu (71% Pu239)	**45 kg**	**160 kg** (30% Pu)	**1 900 kg** (12.5% Pu)	**400 kg** (12.5% Pu)
W-Pu (100% Pu239)	**30 kg**	**180 kg** (20% Pu)	**Not calculated**	**590 kg** (6% Pu)

Critical masses × 0.7, calculated with French codes

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Application to the MOX FFF: Aqueous Polishing - Dissolution

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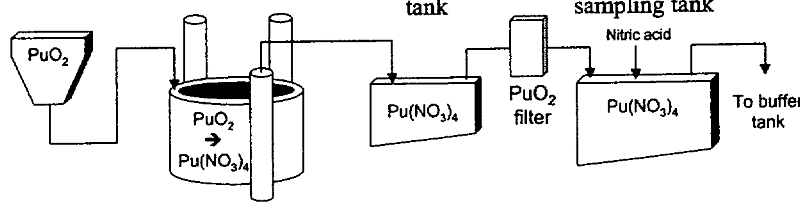
Reference fissile medium	$PuO_2 + H_2O$	$PuO_2 + H_2O$	$PuO_2 + H_2O$	$Pu(NO_3)_3 + H_2O$
Control mode	Geometry (+ Cadmium)	Geometry (+ Cadmium)	Geometry (+ Cadmium)	Geometry (+ Cadmium)
	Dissolution feed	Dissolver	Reception tank	Dilution and sampling tank



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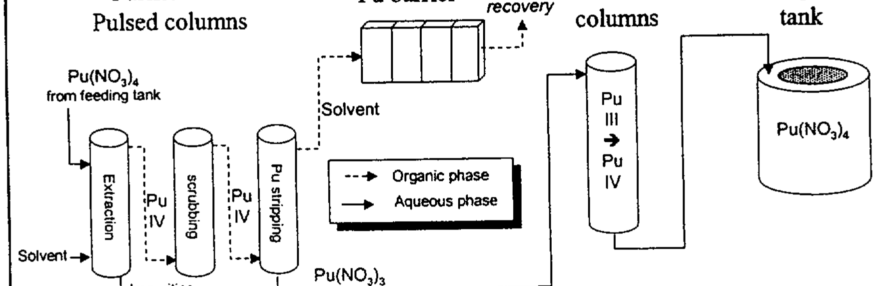
Application to the MOX FFF: Aqueous Polishing - Purification

For Information Only

Reference fissile medium	$Pu(NO_3)_3 + H_2O$	$Pu(NO_3)_3 + H_2O$	$Pu(NO_3)_3 + H_2O$	$Pu(NO_3)_3 + H_2O$
Control mode	Geometry	Geometry	Geometry	Geometry
	Purification Pulsed columns	Pu barrier	Oxidation columns	Reception tank



Application to the MOX FFF: Aqueous Polishing - Precipitation

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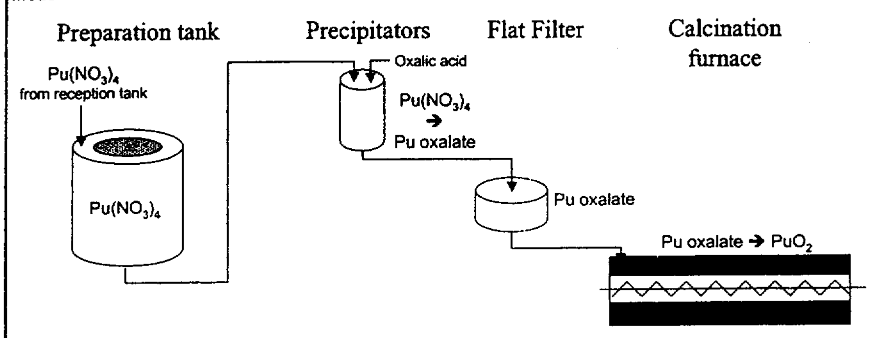
Reference fissile medium	$Pu(NO_3)_3$+ H_2O	PuO_2F_2 + H_2O	PuO_2F_2 + H_2O	PuO_2+ H_2O
Control mode	Geometry	Geometry	Geometry	Geometry



Application to the MOX FFF: Aqueous Polishing - Canning

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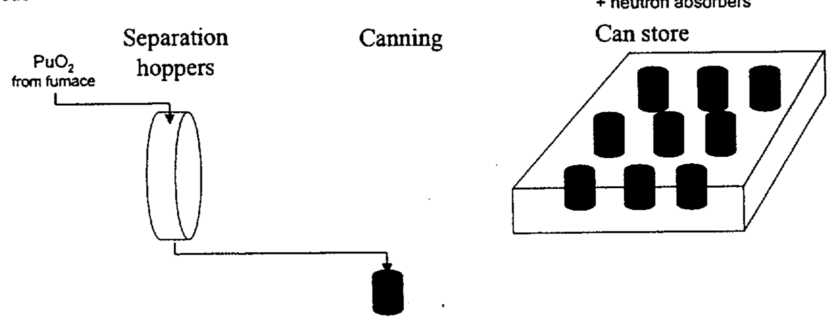
For Information Only

Reference fissile medium	PuO_2+ H_2O	PuO_2+ H_2O	PuO_2+ H_2O
Control mode	Geometry	Geometry	Geometry + neutron absorbers



Application to the MOX FFF: MOX - Powder area

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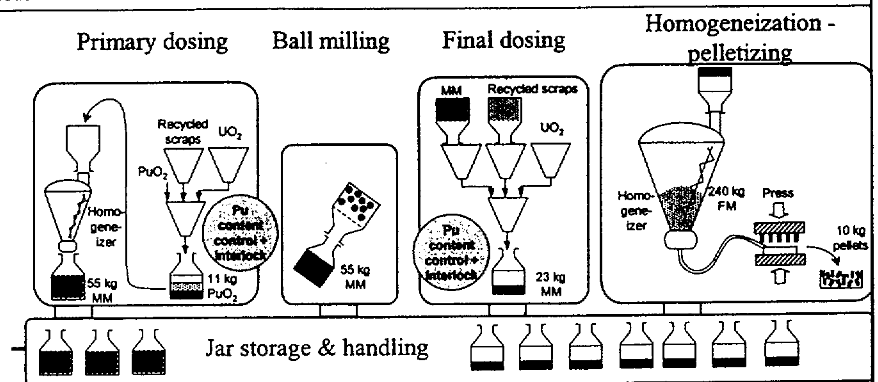
Reference fissile medium	PuO_2, MM (<22%Pu)	MM (<22%Pu)	MM (<22%Pu)	MM (<22%Pu) FM (< 6.3%Pu)
Control mode	Mass + moderation	Mass + moderation	Mass + moderation	Mass + moderation



Geometry + mass + moderation

Application to the MOX FFF: MOX - Pellet area

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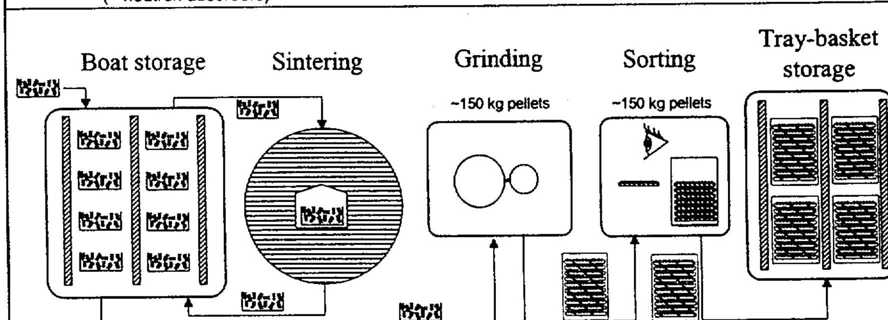
For Information Only

Reference fissile medium	Pellets (< 6.3%Pu)	Pellets (< 6.3%Pu)	Pellets (< 6.3%Pu)	Pellets (< 6.3%Pu)	Pellets (< 6.3%Pu)
Control mode	Geometry (+ neutron absorbers)	Geometry	Mass + moderation	Mass + moderation	Geometry (+ neutron abs.



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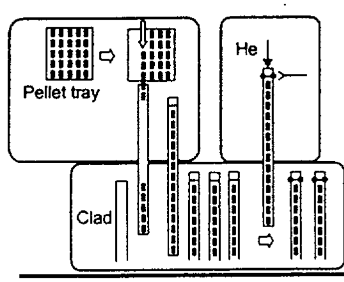
Application to the MOX FFF: MOX - Rod area

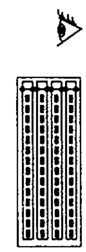
For Information Only

Reference fissile medium	Pellets (< 6.3%Pu)	Rods (< 6.3%Pu)	Rods (< 6.3%Pu)
Control mode	Geometry (Mass + moderation*)	Geometry (Mass + moderation*)	Geometry (+ neutron absorbers)

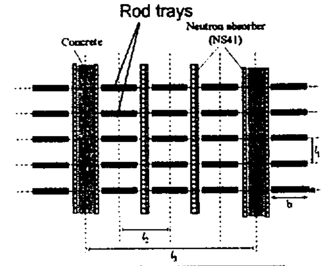
Rod filling & welding



Rod control



Rod storage



* *So called "secondary control mode": for some accidental situations (e.g. earthquake)*

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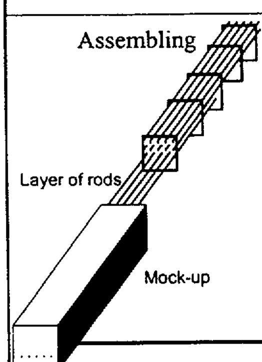
Application to the MOX FFF: MOX - Assembly area

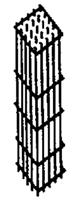
For Information Only

Reference fissile medium	Rods (< 6.3%Pu)	Rods (< 6.3%Pu)	Rods (< 6.3%Pu)
Control mode	Geometry + moderation	Geometry	Geometry

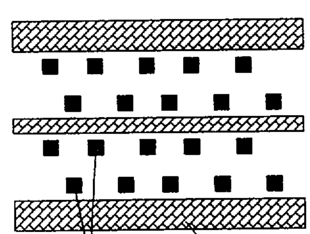
Assembling



Assembly control



Assembly storage



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Preparation of Criticality Safety Evaluations

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Nuclear Criticality Safety Evaluations

- Criticality design and verification process
- Design and verification analysis methods
- Nuclear Criticality Safety Evaluation approach
- Treatment of normal and upset conditions
- Related hazard assessments and operational programs

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Criticality Design Process

- Nuclear Criticality Safety Evaluations (NCSEs) prepared in accordance with US standards and criticality methodologies
  - ANSI/ANS-8.1 as invoked by RG 3.71
  - Selection of US developed computer codes and nuclear data
- NCSEs originated by the Process Group
- NCSEs independently reviewed by the Facility Group

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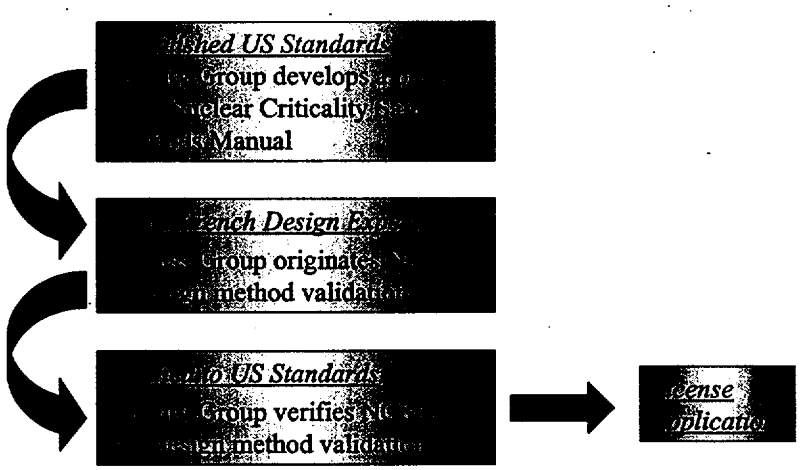
Criticality Design Process



Design and Verification/Analysis Methods

- KENO and SCALE-238 neutron cross-section library selected for use in originating NCSEs
  - Process Group familiarity with KENO
  - 238-group fine-structure best suited to intermediate neutron energy range system design applications
- KENO and MCNP 4B applied in verification
  - 238GROUUPNDF5 library used with KENO-IV
  - ENDF60 continuous energy library used with MCNP 4B
- Computer codes used in both origination and verification of NCSEs will be verified & validated

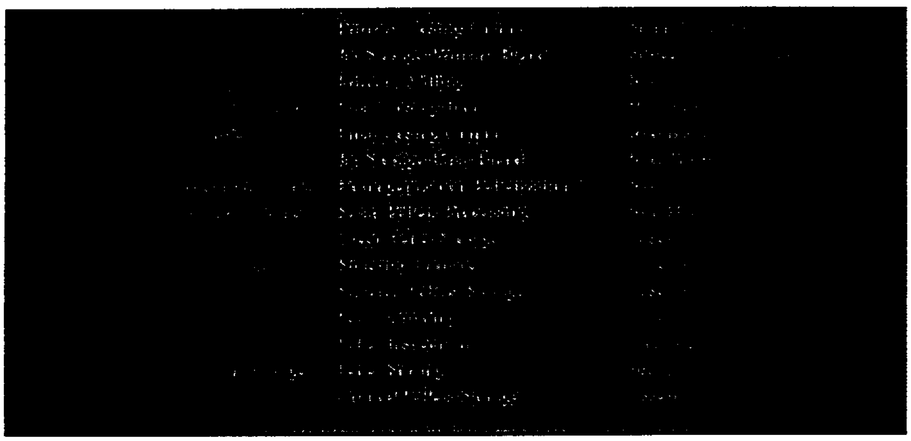
NCSE Approach

- NCSE performed supporting each process station
- k_{eff} calculated for each station using validated method
- Safety criterion for normal and upset conditions:

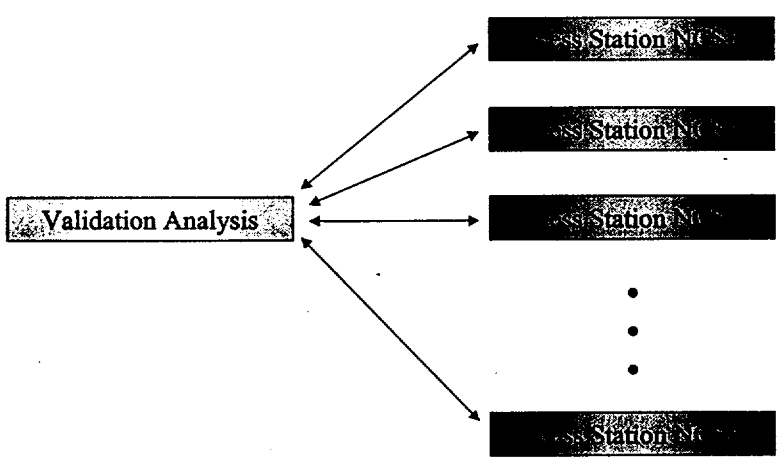
$$k_{eff} + \Delta k_{eff} \leq (1 + \beta) - \Delta\beta - \Delta k_{marg}$$

where: k_{eff} is the calculated result for a given case
Δk_{eff} is the total uncertainty in k_{eff} (95/95 tolerance)
β is the method bias established in validation
$\Delta\beta$ is the standard deviation σ in β
Δk_{marg} is administrative safety margin

Process Stations



Integrated Validation Approach



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Application of Safety Criterion

- Safety criterion applied as an Upper Safety Limit (USL) on the calculated k_{eff} of a design application system
  - Application of trending analysis techniques documented in NUREG/CR-6361 for LWR fuel transport and storage packaging
  - $USL = 1 - \Delta k_{marg} + \beta - \Delta \beta$, where $USL \geq k_{eff} + \Delta k_{eff}$
  - USL calculated based on a linear regression fit of benchmark critical experiment results analyzed as a function of important system parameters (e.g., average neutron energy causing fission)
- USLs justified on an design application specific basis in the NCSEs

November 17, 1999 NRC Technical Exchange Page 49

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Treatment of Normal and Upset Conditions

- NCSEs shall include consideration of the full range of potential normal and upset conditions
- Material composition (e.g., Pu density) and mechanical tolerance uncertainties applied as credible worst-case or incorporated statistically at a 95% probability/95% confidence level
- Upset conditions (e.g., presence of water) generally incorporated as a credible worst-case modeling assumption for normal operation to minimize impact on process and administrative controls

November 17, 1999 NRC Technical Exchange Page 50

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Hazard assessments/operational programs

- Systematic hazard assessment performed to identify all potential upset conditions requiring analysis and to demonstrate compliance with Double Contingency Principle
- NCSEs and hazard assessment provide input for establishment of MFFF administrative controls and process limits

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Summary

- NCSEs originated by Process Group and independently verified by Facilities Group
- Design and verification preformed using two significantly diverse computer code systems
- NCSE acceptance criterion based on USL Method 1 documented in NUREG/CR-6361consistent with ANSI/ANS-8.17 guidance
- NCSEs shall address full range of normal and upset conditions

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Criticality Validation

Approach to Benchmarking

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Criticality Methods Validation

- Selected Criticality Analysis Methods
- Method verification and validation process
- Benchmark Validation Data Analysis
  - Establishing area(s) of applicability
  - Determination of calculational bias
  - Justifying margin of subcriticality

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Criticality Analysis Methods

- KENO/238GROUPNDF5 and MCNP 4B/NDF60 applied in NCSE origination and verification
- Computer codes used in both origination and verification of NCSEs will be verified & validated
- Validation performed integral with NCSE origination to confirm experiment set applicability to design conditions

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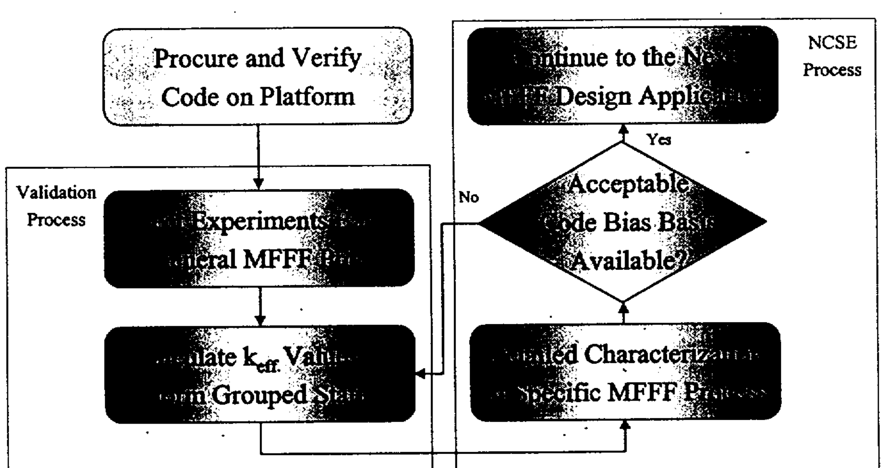
Verification & Validation Process Flow



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Method Verification and Validation Process

- General MFFF process characterization and benchmark experiment selection
  - Experiment configurations must cover a wide range of diversity in MFFF design applications and control methods (e.g., supplemental neutron absorber materials)
  - Experiments grouped based on similarity to design applications for statistical analysis
  - Grouped statistics and trending results provide basis for design application specific USLs

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Generalization of MFFF Processes

Form	Reference Density*	Reference Pu Content*
PuO2 Powder	7	100%
PuO2+H2O	7	100%
Pu Nitrate	Solution	100%
PuO2 Oxalate	Precipitate	100%
PuO2 Powder	3.5	100%
UO2 Powder	3.5	0%
MOX Powder	3.5	22%
MOX Powder	3.5	22%
MOX Powder	3.5	6%
MOX Powder	3.5	6%
MOX Pellets	11	6%
MOX Rods	11	6%
MOX Assemblies	11	6%

* Values are approximate intended for illustration only.

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Benchmark Experiment Selection

- Available benchmark experiments likely to be considered
  - OECD International Criticality Benchmark Handbook
    - Aqueous Solutions - PU-SOL-THERM-XXX
    - Plutonium-Metal - PU-MET-FAST-XXX
    - PuO_2/Polystyrene Slabs - PU-COMP-MIXED-001, -002
    - MOX Pins - MIX-COMP-THERM-005, -009
    - Intermediate Energy Pu Experiments - MIX-MET-INTER-001
  - EPRI clean critical experiments (UO2 and MOX pins in water)
  - SAXTON partial plutonium core (UO2 and MOX pins in water)
- Experiments selected based on similarity to design applications and coverage of application attributes

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Benchmark Validation Data Analysis

- Establishing area(s) of applicability
- Determination of calculational bias
- Justifying margin of subcriticality

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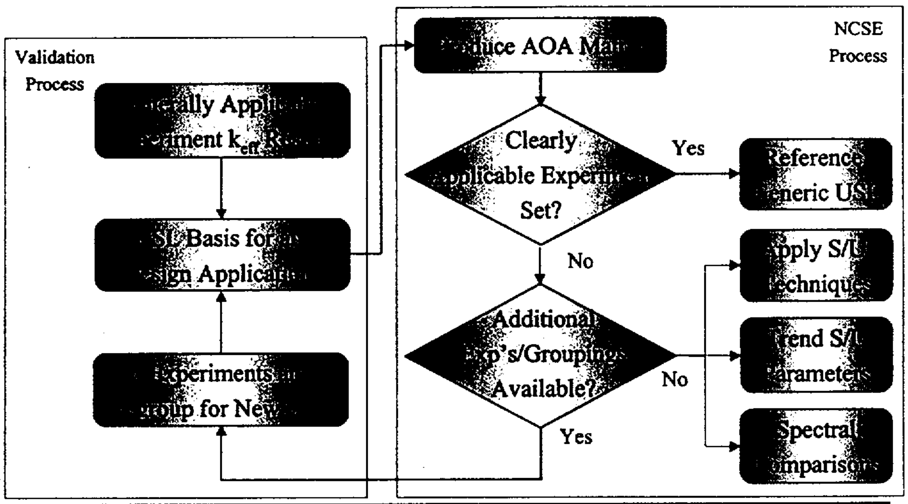
Area(s) of Applicability Design Integration Logic



November 17, 1999 NRC Technical Exchange Page 61

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Area(s) of Applicability Determination

Characteristic	Comment
Fissile Material	Specify the type of fuel and enrichment.
Moderation	Identify moderating materials and if possible, quantify a measure of moderation (e.g., H/X ratio). Interstitial moderation may be characterized by thickness of moderator.
Reflection	Identify the reflecting materials and associated thickness (if applicable).
Absorption	Identify the absorbing materials and associated thickness (if applicable).
Neutron Energy Spectrum	Identify the average energy group range or the neutron energy range.

November 17, 1999 NRC Technical Exchange Page 62

Area(s) of Applicability Determination

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- Two Examples
  - 1. Pellets Boats and Boxes Store
  - 2. Buffer Powder Store

Area(s) of Applicability Determination Example 1-Pellets Store

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Characteristics	Design Application: Pellet Arrays[a]	Benchmark Suite: MIX-COMP-THERM[b]	Comment
Fissionable	PuO_2-UO_2 SG = 11 $^{240}Pu/Pu_{total}$ = 0.04 $^{235}U/U_{total}$ = 0.003 0.02 < Pu/(Pu+U) < 0.065	6 < PuO_2-UO_2 SG < 10.2 0.08 < $^{240}Pu/Pu_{total}$ < 0.22 0.0016 < $^{235}U/U_{total}$ < 0.0072 0.015 < Pu/(Pu+U) < 0.066	In range except for $^{240}Pu/Pu_{total}$ and Specific Gravity
Absorber	None	None, 0-767 ppmb	In range
Moderator	Pure Water 40 < H/Pu < 340 (evaluated) Optimum H/Pu ≅ 170 Room Temperature	Pure water & borated water 75 < H/Pu < 1169 Room Temperature	In range
Scatterer	In fuel O Reflector: H_2O	In fuel O Reflector: H_2O	In range
Shape	Tri. Pitch Lattice Array Rectangular Core	Square and Tri. Pitch Arrays Cylindrical & Rectangular Cores	In range
Heterogeneity	Heterogeneous system: Triangular pitch pellets & rods	Heterogeneous system Square and triangular pitch rods	In range
Reflection	Water Regular concrete	Water	In range
Neutron Energy	Thermal system 0.17 < EALF < 0.26 eV (limiting cases)	Thermal systems 0.08 < EALF< 0.34 eV	In range

[a] Isolated water reflected boxes and boats of green & sintered pellets over range of pin pitches.
[b] Includes consideration of sets MIX-COMP-THERM-002, 003, 004, 005, and 009 in OECD Handbook.

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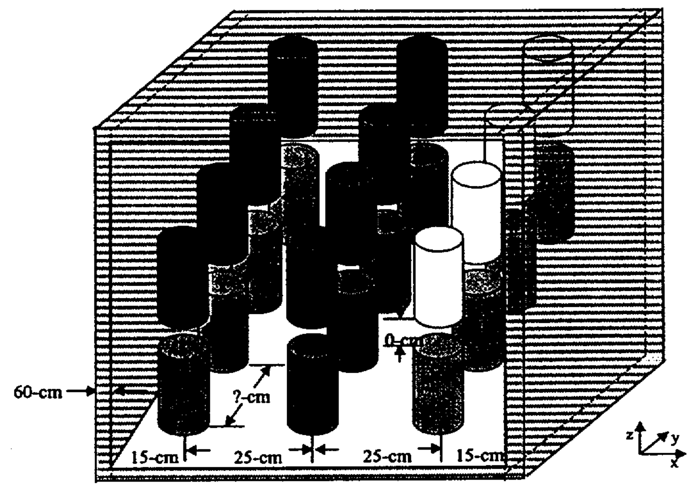
Area(s) of Applicability Determination
Example 2-PuO2 Buffer Store



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Area(s) of Applicability Determination
Example 2-PuO2 Buffer Store(Cont'd)

Characteristics	Design Application: PuO_2 Powder Storage Bin	Benchmark Suite: PU-COMP-MIXED-002	Comment
Fissionable	$2.8 < Pu\ SG < 3.1$ $^{239}Pu/Pu_{total} = 0.96$ $^{240}Pu/Pu_{total} = 0.04$	$1.1 < Pu\ SG < 2.3$ $0.75 < {}^{239}Pu/Pu_{total} < 0.98$ $0.02 < {}^{240}Pu/Pu_{total} < 0.18$	In range except for Pu Specific Gravity
Absorber	Interstitial borated concrete	None	Not in range
Moderator	Light Water $5 < H/X < 7$ Room Temperature	Polystyrene $0.04 < H/X < 49$ $0.0 < C/X < 49$ Room Temperature	In range
Scatterer	In core O: $4.5 < O/X < 5.5$ Reflector: H_2O	In core O: $2.0 < O/X < 2.3$ Reflector: H, C, O (Plexiglas)	Not in range
Shape	Cylinder array (3 x infinite); Single Unit Radius = 5 cm	Cuboid	Not in range
Heterogeneity	Heterogeneous system: PuO2 powder cylinders contained in borated concrete	Homogeneous system	Not in range
Reflection	Regular concrete	Plexiglas	Not in range
Neutron Energy	Mixed systems 1200 < EALF< 1500 eV	Thermal, mixed & fast systems 0.7 < EALF< 5000 eV	In Range

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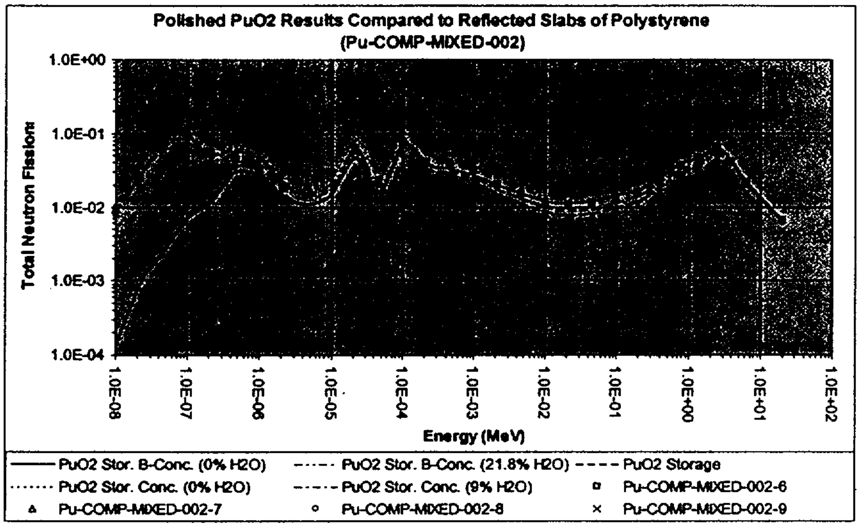
Area(s) of Applicability Determination Example 2-PuO2 Buffer Store (Cont'd)



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Area(s) of Applicability Determination Example 2-PuO2 Buffer Store (Cont'd)

- PU-COMP-MIXED-002 benchmark experiment set include important in range characteristics
  - PuO_2 composition data
  - Moderation (H/X)
  - Neutron energy
- Additional benchmarks required to address out-of-range areas of applicability
  - Interstitial borated concrete
  - Concrete reflector

Determination of Method Bias & Uncertainty

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- Validation will apply trending analysis of applicable parameters to assure conservative treatment of method bias and uncertainty (criticality benchmark guidance presented in NUREG/CR-6361 and NUREG/CR-6102)
- Sensitivity and uncertainty (S/U) techniques applied as necessary (Draft NUREG/CR-5593)
  - Demonstrate experiment similarity to design applications
  - Further justify subcritical margin in cases where data scarce or significant extrapolation necessary
  - Alternative spectral comparisons performed as alternative if S/U methods not available

Trending Analysis and Use of USLSTATS

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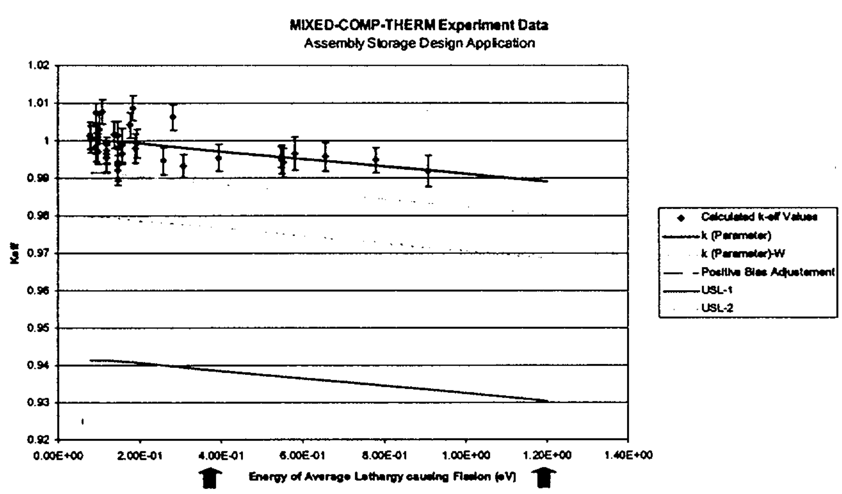

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Subcritical Margin

- Minimum subcritical margin of 0.05 Δk, typically, established as a design criterion for MFFF design applications
  - Identified as Δk_m in ANSI/ANS-8.17 subcriticality criteria
  - Consistent with design guides for commercial USNRC licensed power reactor applications where applicability of available experimental benchmark data is well established
- Subcritical margins (including minimum of 0.05 Δk, typically,) shall be justified on a design application specific basis
  - Area(s) of applicability analysis
  - USLSTATS trending
  - S/U and non-parametric techniques

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Subcritical Margin

- Situations where a higher subcritical margin may be required include:

  a) Significant extrapolation beyond benchmark area(s) of applicability required

  b) Data do not follow a normal distribution

  c) Insufficient applicable benchmark data

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Criticality Methods Validation Summary

- Nuclear Criticality Safety Evaluations will be originated by the Process Group and independently verified by the Facility Group
- Standard US criticality safety evaluation method and criteria applied
- Design and verification process integrated to ensure validation applicability to specific design applications
- Criticality calculations will be validated using the latest methods of benchmark validity determination including parameter trending spectral analysis, and ORNL S/U methods (to extent available)

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AVLIS Criticality SER

Lessons Learned

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AVLIS Criticality Lessons Learned (1)

- Provide information on criticality situations so that applicable benchmarks can be evaluated.
- Justify Areas of Applicability of benchmarks
- Address the full range of normal and upset situations.
- Use specific benchmarks for different MFFF situations
- Use standard statistical tools (USLSTATS) to analyze trends in benchmark data.

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AVLIS Criticality Lessons Learned (2)

- Typically use 0.05 administrative margin in addition to calculated situation specific benchmark bias. Rare cases of less administrative margin to be fully justified
- Use specific benchmarks and techniques to address intermediate energy range situations
- Extrapolations from the area of the benchmarks to be justified.

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Japanese Criticality Accident

Lessons Learned

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Japanese Accident Lessons Learned

- It is imperative to provide adequate training to workers about criticality safety
- Workers must realize the importance of following approved procedures
- Procedures must be under strict configuration management control to ensure approval by all appropriate entities
- Criticality safety must be designed into the facility

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Criticality Summary

- Criticality Safety Administrative Programs will be used on the MFFF
- Standard US Criticality Safety Evaluation Methodology
- Procedure has been prepared to ensure standard US methodology is used
- Standard US criticality code (KENO/Scale 4.4) will be used
- Criticality calculations will be validated using the latest methods of benchmark validity determination including parameter trending analysis and ORNL methods
- Standard administrative uncertainties will be used
- Nuclear Criticality Safety Evaluations will be originated by the Process Group and independently reviewed by the Facility Group