Aircraft Crash Hazard

David Dancer

Nuclear Regulatory Commission

NRC/DOE Technical Exchange & Management Meeting
November 7-9, 2006

ENCLOSURE 4

Outline

- Key 10 CFR Part 63 Requirements
- Relevant Guidance and Precedent
- NRC/DOE Interactions
- NRC Key Messages
- Summary

Key 10 CFR Part 63 Requirements

- The Preclosure Safety Analysis must include:
  - an identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences – 63.112(b)
  - data pertaining to the Yucca Mountain site, and the surrounding region, to the extent necessary, used to identify naturally occurring and human-induced hazards at the geologic repository operations area – 63.112(c)
  - the technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards in the safety analysis – 63.112(d)

Technical Guidance and Precedent

- NUREG-0800 (NRC), *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (Section 3.5.1.6)

- License Application (Private Fuel Storage, LLC) for a Proposed Private Fuel Storage Facility in Utah, NRC No. Docket 72-22

Past NRC/DOE Interactions

- Technical Exchanges on Aircraft Hazards (most recently June 2005)
- August 2005 letter from NRC to DOE describing 13 NRC comments on the DOE methodology
- November 2005 letter from DOE to NRC responding to 6 of the 13 comments in NRC's August 2005 letter
- January 2006 letter from NRC to DOE providing high-level feedback to DOE's November 2005 Letter
- November 2006 letter from DOE to NRC transmitting revised frequency report

Key Messages

- Aircraft Crash Frequency
- Uncertainty
- Other Considerations

Aircraft Crash Frequency

- Regulatory and technical bases to support the selection of a threshold screening frequency
- DOE plans for enforcement of flight restrictions

Uncertainty

- Uncertainties in assumptions, data and information used, methodologies selected, and analysis techniques
- Greater attention to uncertainty as DOE gets closer to the threshold screening frequency

Uncertainty - Continued

- Examples of sources of uncertainties:
  - Credit for any actions or inactions by pilots to reduce the frequency of aircraft crashes
  - Classification of aircraft mishaps into various types, and further quantitative analyses using information from the mishap reports
  - Use of data sets for one week flight activities through Beatty Corridor to predict flight activities during the preclosure period
  - Use of Solomon's method versus NUREG-0800 to determine crash frequency outside the airway width

Other Considerations

- Technical basis for not considering other flight related activities in the PCSA
- If structural robustness is credited in the PCSA, analyses that will be included to demonstrate structural robustness

Summary

- Technical and regulatory bases for the screening criteria
- Enforcement of flight restrictions
- Consideration of uncertainty

Preclosure Source Terms and Consequences

Ali Simpkins and Tae Ahn

United States Nuclear Regulatory Commission

NRC/DOE Technical Exchange and Management Meeting
November 7 - 9, 2006

Outline

- Purpose
- Regulatory Requirements
- Key Messages
  - Communication of DOE Plans
  - Methods and Parameters
  - Source Terms
  - Confinement and Shielding
  - Radiation Exposure and Consequences
  - Radiation Protection Program
- Summary

Purpose

- Communicate NRC Key Messages to DOE
- Messages sent to DOE by letter dated November 2, 2006

Regulatory Requirements

- 10 CFR 63.111 Performance objectives of geologic repository operations area through closure
  - References 10 CFR 63.204 Preclosure Standard
  - References 10 CFR Part 20 Standards for Protection Against Radiation
- 10 CFR 63.112 Preclosure Safety Analysis of geologic repository operations area
- 10 CFR 20.1101 Radiation Protection Programs

Communication of DOE Plans

- Discuss plans for developing source terms and consequence analysis in the preclosure safety analysis and the radiation protection program in the license application

Adequacy of Methods and Parameters

- Analysis should be based on accepted engineering practices and sound health physics principles

- Consider normal operations exposures and Category 1 and 2 event sequences in consequence assessments for the preclosure safety analysis

Source Terms

- Characteristics of high-level waste to be processed
  - Number of fuel assemblies
  - Enrichment, burnup, and decay time
- Types of failure phenomena
  - Justification of release fractions: normal and accident conditions (e.g. drop or seismic)
- Release fraction
  - Effects of impact energy, oxidation, and high burnup
  - Building confinement

Confinement and Radiation Shielding

- Description and technical basis for design features should be provided in the preclosure safety analysis
  - Heating, ventilation, and air conditioning system
  - High efficiency particulate air (HEPA) filters
  - TAD canister confinement
  - TAD canister overpack

Direct Radiation Exposures and Airborne Release Consequences

- Direct radiation
  - Shielding design features
  - Dosimetry
- Airborne release models
  - Release fractions
  - Building confinement: building leakage, HEPA efficiency, and TAD canister protection
  - Dispersion factors
  - Meteorological parameters
  - Dosimetry

Radiation Protection Program

- Commensurate with the scope of activities at the facility to ensure compliance with the dose limits
- Maintains exposures As Low As is Reasonably Achievable (ALARA)
- Representative persons (locations, occupancy times) for estimating doses should be consistent with identified controls (e.g., restricted areas and protective features)

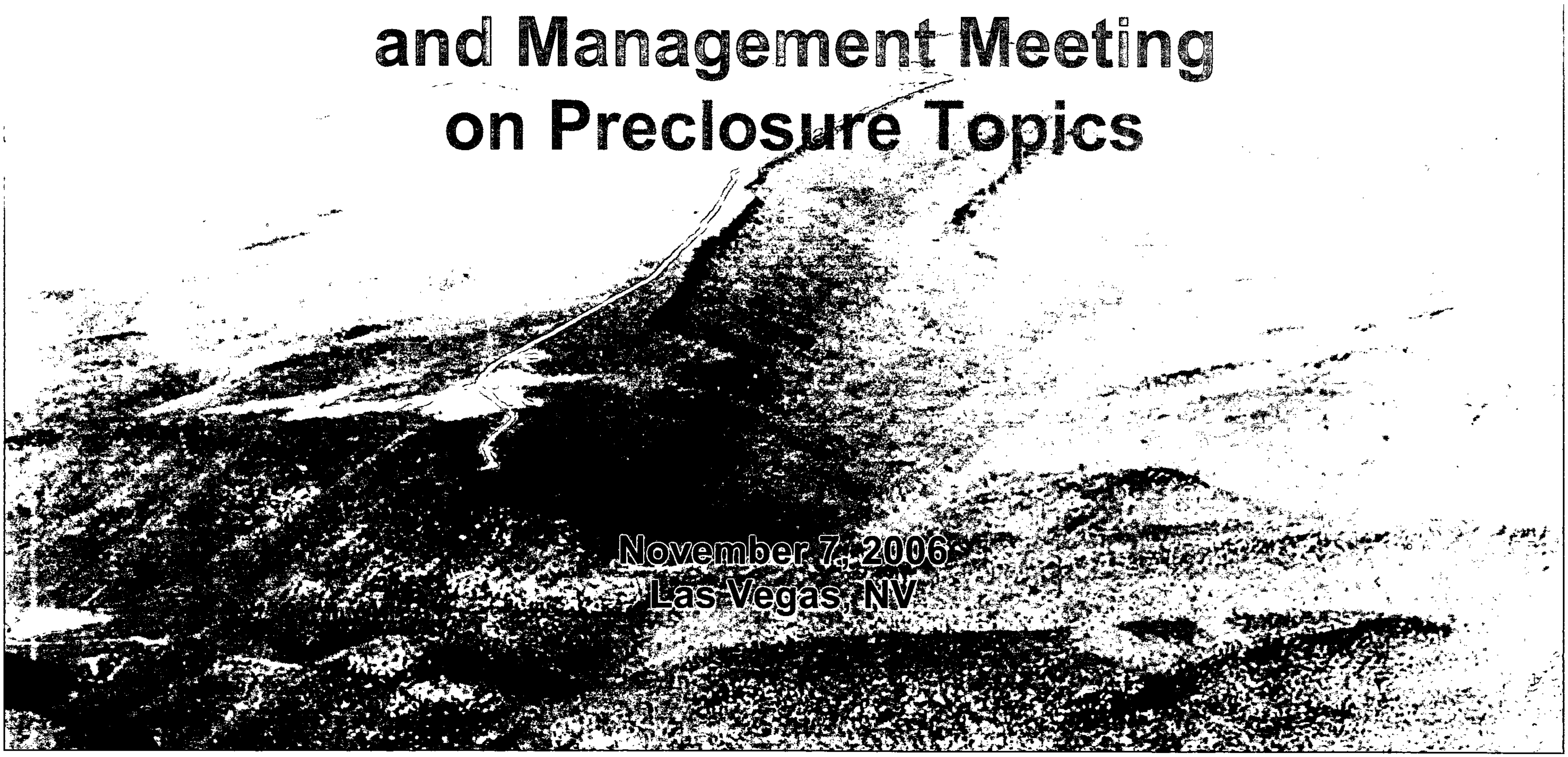
Summary

- Communication of DOE Plans
- Methods and Parameters
- Source Terms
- Confinement and Shielding
- Radiation Exposure and Consequences
- Radiation Protection Program

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics



November 7, 2006
Las Vegas, NV

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS: 1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY, 3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

<u>November 7 and 8, 2006</u>	<u>November 9, 2006</u>
8:00 AM – 5:00 PM (PT)	**8:00 AM – 12:00 PM (PT)**
11:00 AM – 8:00 PM (ET)	**11:00 AM – 3:00 PM (ET)**

U. S. Nuclear Regulatory Commission Hearing Center
Pacific Enterprise Plaza, Building 1
3250 Pepper Lane
Las Vegas, Nevada 89120

And via Teleconference to:

U. S. Nuclear Regulatory Commission	Center for Nuclear Waste Regulatory Analyses
Two White Flint North, Room T 7A-1	Conference Room A-237, Bldg. 189
11545 Rockville Pike	6220 Culebra Road
Rockville, MD	San Antonio, TX

INTERESTED PARTIES MAY PARTICIPATE VIA TELECON BY CALLING 1-800-638-8081, Passcode 8755# or 301-231-5539, Passcode 8755#

<u>Tuesday November 7, 2006 (Aircraft Hazards and Source Terms and Consequence Methodology)</u>

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:30 AM	NRC Key Messages on Aircraft Hazards Assessment	NRC
9:00 AM	Background and Overview of Updated Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
9:30 AM	Changes in Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
10:00 AM	Break	All
10:15 AM	Aircraft Hazards Sensitivity Analysis	DOE/BSC (K. Ashley)
11:00 AM	Response to 13 NRC Issues (NRC letter of Aug. 2, 2005)	DOE/BSC (K. Ashley)
11:30 AM	Lunch	All
1:00 PM	NRC Key Messages on Source Terms and Consequence Methodology	NRC
1:30 PM	Radioactive Source Terms and Release Methodology	DOE/BSC (D. Dexheimer)
2:30 PM	Break	All
2:45 PM	Consequence and Analysis Methodology	DOE/BSC (D. Dexheimer)
3:30 PM	Uncertainty and Sensitivity Analysis	DOE/BSC (D. Dexheimer)
3:50 PM	Documents to be Revised	DOE/BSC (D. Dexheimer)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

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NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS: 1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY, 3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

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Wednesday November 8, 2006 (Reliability Assessment, Technical Specifications, and Training)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:15 AM	NRC Key Messages: - Reliability Assessment	NRC
9:00 AM	Reliability Assessment Overview	DOE/BSC (M. Frank)
9:45 AM	Break	All
10:00 AM	Human Reliability Assessment	DOE/BSC (M. Frank)
11:30 AM	Lunch	All
1:00 PM	Reliability Assessment for Structures, Systems, and Components	DOE/BSC (M. Frank)
2:15 PM	Break	All
2:30 PM	NRC Key Messages: - Technical Specifications - Systematic Approach to Training	NRC
3:00 PM	DOE Plans for Development of Technical Specifications	DOE (W. Spezialetti)
3:30 PM	DOE Plans for Systematic Approach to Training	DOE/MTS (J. McMahon)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS:
1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY,
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8:00 AM – 5:00 PM (PT)
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Thursday November 9, 2006 (Preclosure Criticality and License Application Requirements Mapping)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:15 AM	NRC Key Messages on Preclosure Criticality	NRC
8:45 AM	Preclosure Criticality Discussion	DOE/BSC
9:45 AM	Break	All
10:00 AM	License Application Status and Requirements Mapping	DOE (R. Warther)
10:10 AM	License Application Requirements Mapping	DOE/BSC (G. Ashley)
11:00 AM	Public Comments	All
11:15 AM	Break/Caucus	All
11:30 AM	Summary Discussion/Closing Remarks	NRC/DOE
12:00 PM	Adjourn	All

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Aircraft Hazards

Presented to:
NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:
Pierre E. Macheret
Preclosure Safety Analyses Senior Engineer
Bechtel SAIC Company, LLC

Kathryn L. Ashley
Preclosure Safety Analyses Senior Engineer
Bechtel SAIC Company, LLC

November 7, 2006
Las Vegas, Nevada

Outline

- Background
- Overall analysis approach
- Changes in analysis since August 2005
- Design Basis, Assumptions, Results
- Conservatisms
- Sensitivity analysis
- 13 NRC items for DOE consideration

Background

- Last technical exchange (TE) with the NRC on Aircraft Hazards was June 1, 2005
- August 2, 2005, NRC closed KTI PRE.3.01 with 13 items for DOE consideration
- November 25, 2005, DOE responded to 6 of 13 items with the August 26, 2005 revision of the *Frequency Analysis of Aircraft Hazards for License Application*
- The October 2006 revision addresses 12 of the 13 items
- Last item concerns implementation of the Flight-restricted airspace, which will be discussed separately

Overview of Approach

- **10 CFR Part 63 Requirements:**
  - 63.2 Definition of Event Sequence
    - Event sequences that have at least one chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences
  - 63. 112 - Requirements for PCSA
    - Specifically 63.112 (d) specifies that the PCSA must include the technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards in the safety analysis
  - 63. 111(b) (2) - Performance objectives for the Geologic Repository Operations Area (GROA) through permanent closure
    - No member of the public at or beyond the site boundary will receive, as a result of a single Category 2 event sequence a TEDE of 5 rem

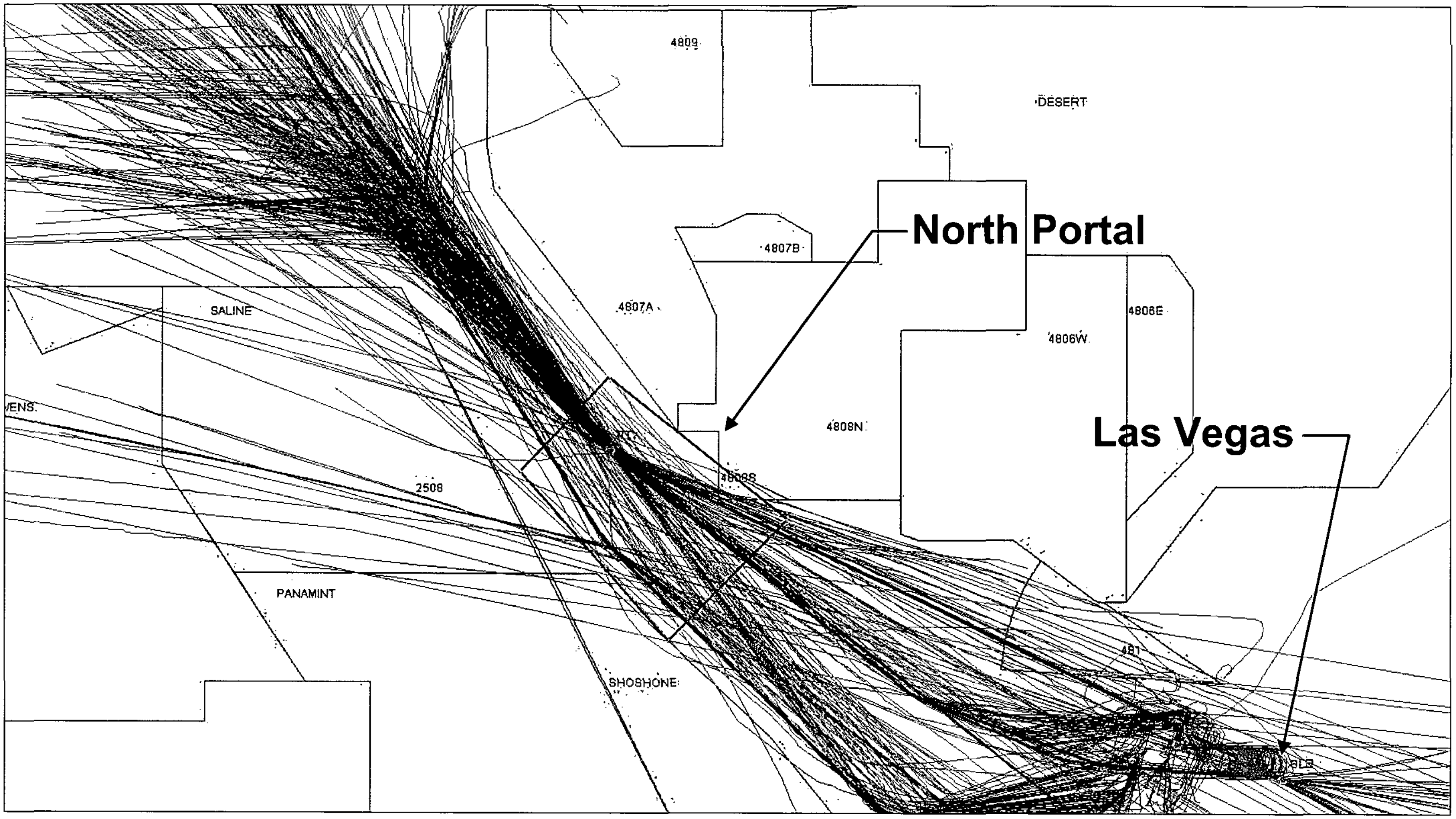
Overview of Approach (cont.)

- Beyond Category 2 event sequences can be screened out and not included in 10 CFR Part 63.111(b) (2) public dose calculations:
  - Probability < 1 in 10,000 before permanent closure
    - Time period is duration of emplacement operations
      - Not to exceed 50 years
  - Objective is met if aircraft crash frequency is $< 2 \times 10^{-6}$ per year
  - Analysis of dose consequences are not required for beyond Category 2 event sequences

Overview of Approach (cont.)

- **Air traffic on Beatty Corridor**
  - Airway heavily traveled by commercial, military and private aircraft
  - Corridor edge is greater than 5 miles from North Portal
- **Military aircraft operations**
  - Nevada Test and Training Range (NTTR)
  - Nevada Test Site (NTS)

Beatty Corridor Flights on 6/7/05

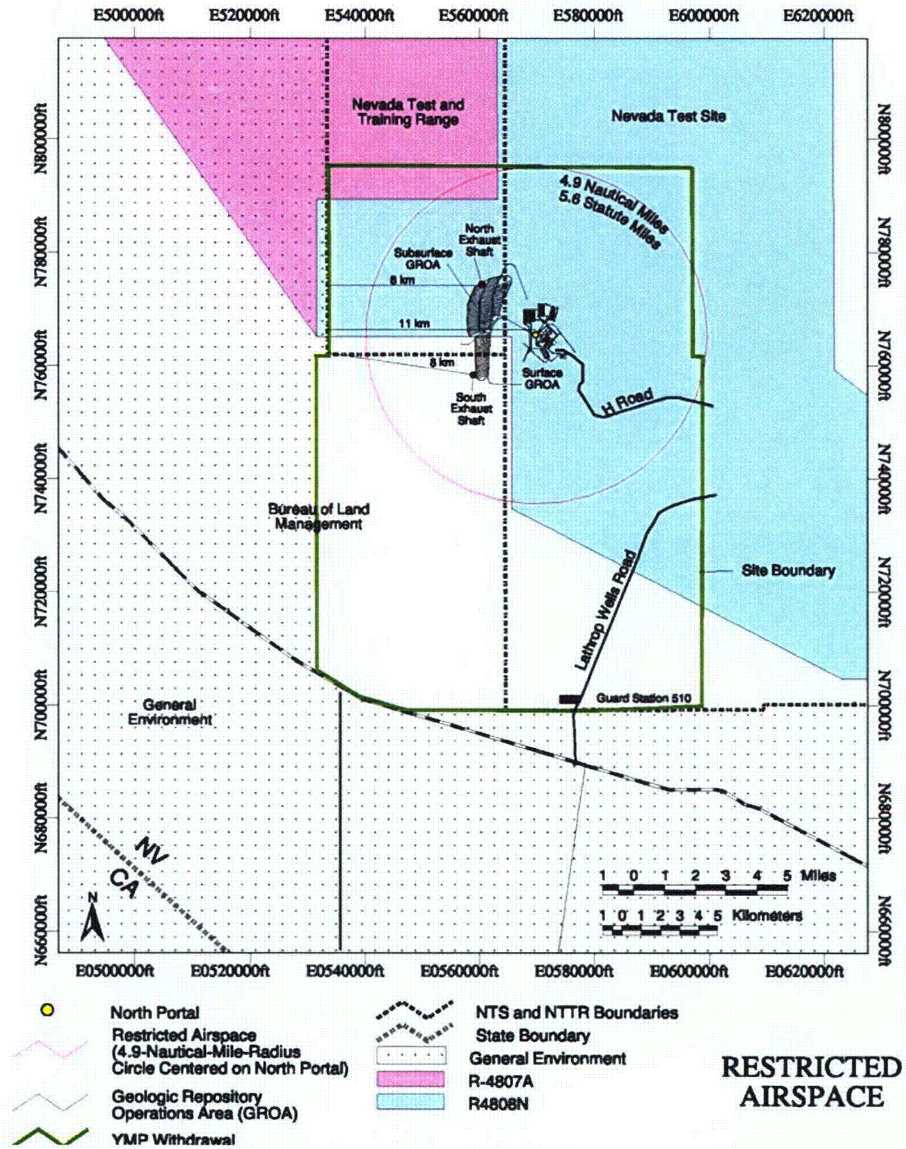


Graphic provided by FAA

Overview of Approach (cont.)

- **Analysis takes credit for a flight-restricted airspace over the repository:**
  - Flights by fixed-wing aircraft in Nevada Test Site (NTS) or Nevada Test and Training Range (NTTR) airspace within 4.9 Nautical Miles (5.6 statute mi) of the North Portal and below 14,000 ft mean sea level (MSL) are prohibited,
  - 1,000 overflights of the flight-restricted airspace per year are permitted above 14,000 ft MSL for fixed-wing aircraft,
  - Maneuvering over the flight-restricted airspace is prohibited; flight is straight and level,
  - Carrying ordnance over the flight-restricted airspace is prohibited,
  - Electronic jamming activity over the flight-restricted airspace is prohibited.

Overview of Approach (cont.)



RESTRICTED AIRSPACE

Overview of Approach (cont.)

- Analysis uses historical data provided by the Federal Aviation Administration (FAA) and the United States Air Force (USAF):
  - Flights in the Beatty Corridor
  - Military aircraft crashes on the Nevada Test and Training Range (NTTR) and the Military Operations Areas (MOA)
  - Aircraft crashes worldwide for the small military aircraft of concern (F-15, F-16, F-22, and A-10)

Changes in Analysis Since Aug. 2005

- **Updated FAA and USAF data through 2005**
- **Updated the surface facilities to reflect current design:**
  - Canister Receipt and Closure Facilities (3)
  - Receipt Facility
  - Wet Handling Facility
  - Initial Handling Facility
  - Aging Pads
  - Railcar and truck staging areas
  - Site transporters (2)

Changes in Analysis (cont.)

- Revised the flight-restricted airspace
- Included discussion on cruise missiles, jettisoned ordnance and electronic jamming
- Historic military crash events categorized into two types:
  - Type 0 – those events that are not applicable to overflights, for example; take-offs, landings, and aggressive maneuvering
  - Type 1 – those events that are applicable to overflights, which are all other events
- Performed sensitivity analysis

Design Basis

- Duration of Emplacement Operations is 50 years or less
- Flight-restricted airspace as described
- Helicopter flights within 0.5 mile of repository surface facilities are prohibited
  (no further consideration of potential helicopter crashes is required)

Assumptions

- Uniform distribution of overflights above the flight-restricted airspace and crash-impact points beneath the flight-restricted airspace
- Flight paths on the Beatty Corridor are approximately straight, parallel and uniformly distributed near Yucca Mountain
- Beatty Corridor flight counts increased 400% from the estimated annual counts based on actual flight counts from one week in June 2005 and one week in December 2005
- General aviation piston-engine aircraft counts further augmented with 10,000 additional counts to account for flights under 10,000 ft MSL.

Assumptions (cont.)

- Only repository surface facilities where spent fuel is present are considered in determining the effective target area
- Relevant surface facilities are within a 1-mile circle centered on the North Portal
- Facilities are assumed to be at 100% capacity and in continuous use for the full emplacement period

Assumptions (cont.)

- Military Aircraft of Concern for flights over and outside the flight-restricted airspace are the F-15, F-16, F-22 and A-10
- Small Military Aircraft Crash Rate is the F-16 crash rate updated to reflect contemporary flight operations experience
- Crash frequency density outside the flight-restricted airspace is based on the ratio of the number of crashes that have occurred on the NTTR and Military Operations Areas, over the total surface encompassed by these areas

Assumptions (cont.)

- No credit for pilot action
- No credit for the robustness of transportation casks, aging casks or buildings to withstand an impact by an aircraft

Analytical Results

Three contributors to the crash frequency:

- **Flights in the Beatty Corridor**

  uses FAA data for military, commercial, and private aircraft (lowest contribution to the total crash frequency)

- **Flights outside of the Flight-Restricted Airspace**

  uses USAF data on the crashes that occurred on the NTTR and MOAs (largest contribution to the total crash frequency)

- **Overflights of the Flight-Restricted Airspace**

  uses USAF data on F-15, F-16, F-22 and A-10 crashes worldwide (medium contribution to the total crash frequency)

Analytical Results (cont.)

Preliminary indications are that event sequences that result from aircraft hazards are beyond Category 2

Conservatisms in Analysis

- Uncertainties in analysis are compensated by conservative assumptions:
  - No credit for pilot action
  - No credit for aging casks, transportation casks, or buildings to withstand crash without breach
  - All overflights of Flight Restricted Airspace are assumed at 14,000 ft MSL (mean sea level)

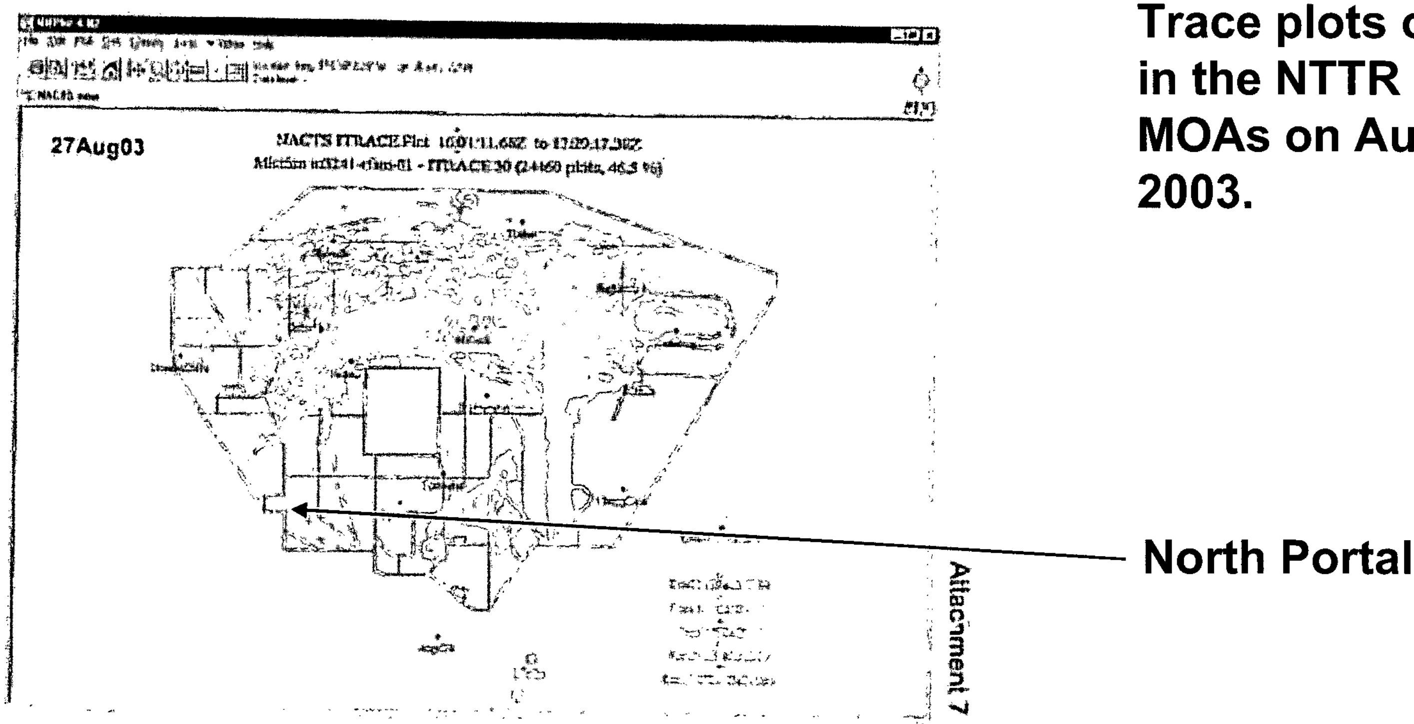
Conservatisms in Analysis (cont.)

- No credit for phased construction
- Distance to Beatty Corridor is chosen to be conservatively short
- Flight counts on the Beatty Corridor were increased by 400%
- F-16 crash rate used for military aircraft overflights of Flight Restricted Airspace and small military aircraft on the Beatty Corridor, which is higher than other small military aircraft

Conservatisms in Analysis (cont.)

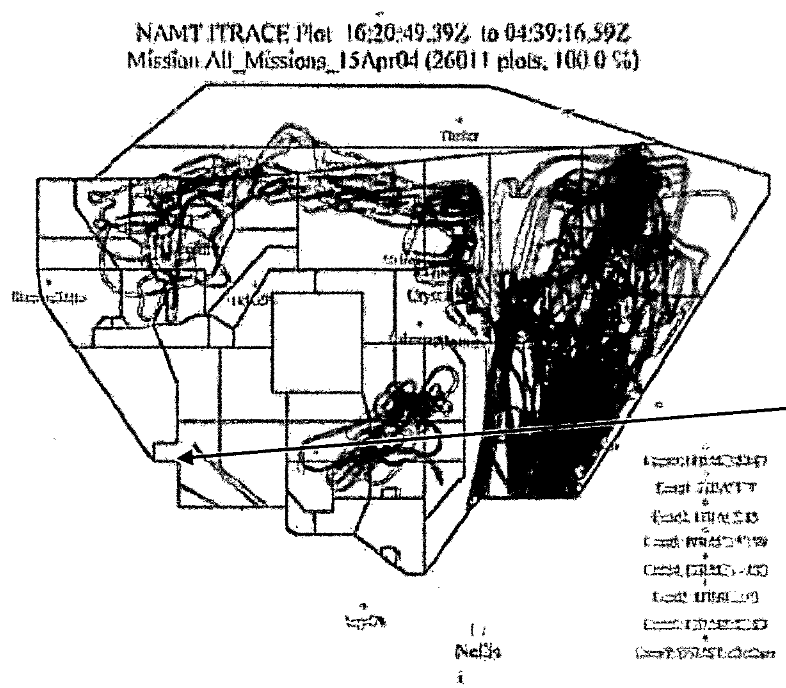
- Effective target areas overestimated by not taking credit for terrain, landscaping, proximity to other buildings and using the concrete pad dimensions for the aging pads instead of just the area occupied by the casks
- Applying the crash frequency density determined by the crashes that occurred on the NTTR and MOAs to flights outside of the Flight-Restricted Airspace even though aggressive maneuvering occurs well within the NTTR

Conservatisms in Analysis (cont.)



Trace plots of flights in the NTTR and MOAs on August 27, 2003.

Conservatisms in Analysis (cont.)



Trace plots of flights in NTTR and MOAs on April 15, 2004.

North Portal

Attachment 8

Sensitivity Analysis

Pilot Action:

- Overall crash frequency would be reduced significantly if pilot action were credited in the analysis for:
  - Flights outside of the Flight-Restricted Airspace
  - Overflights of the Flight-Restricted Airspace

Sensitivity Analysis (cont.)

Altitude of Overflights:

- Using a higher altitude than the minimum allowable of 14,000 ft MSL would reduce the overall crash frequency (low impact)

F-16 Crash Rate:

- Using a weighted averaged crash rate for F-16s and F-15s would reduce the overall crash frequency (low impact)

Sensitivity Analysis (cont.)

- **Distance to the Edge of the Beatty Corridor:**
  - Using a more realistic distance (8 miles) from the facilities to the edge of the Beatty Corridor would reduce the overall crash frequency (low impact)

- **Phased Construction:**
  - Assuming that the aging pads have an 80% capacity factor and that several of the facilities do not become operational until ten years after initial operations would reduce the overall crash frequency (medium impact)

Sensitivity Analysis (cont.)

Counts in the Beatty Corridor:

- Annual estimates from observed counts in the Beatty Corridor were increased by 400% for use in the analysis:
  - Further increasing the Beatty Corridor flight counts used in the analysis by a factor of two results in a small increase to the overall crash frequency
  - Further increasing the Beatty Corridor flight counts used in the analysis by a factor of ten results in a medium increase to the overall crash frequency, which still remains below the screening threshold

Sensitivity Analysis (cont.)

Military Crash Density:

- The crash frequency density is based on the number of crashes that occur in the NTTR and the military operational areas (MOAs) over the time period of interest
- Starting in 2006, it is assumed that zero, one, and two crashes per year as well as the average crash rate of 1.16 crashes per year occurs every year for 10 years
- After 10 years:
  - For 0 crashes/yr – decrease in overall crash frequency (medium impact)
  - For 1 crash/yr – decrease in overall crash frequency (low impact)
  - For 1.16 crashes/yr – decrease in overall crash frequency (low impact)
  - For 2 crashes/yr – increase in overall crash frequency (medium impact, screening threshold not reached)

Sensitivity Analysis (cont.)

Categorizing Event Types:

- 10% of Type 1 changed to Type 0
  - No change in the overall crash frequency
- 10% of Type 0 changed to Type 1
  - No change in the overall crash frequency
- All Type 0 changed to Type 1
  - Overall crash frequency increased (low impact)

Note: Type 1 events are applicable to overflight
Type 0 events are not applicable to overflight

Sensitivity Analysis (cont.)

Glide Ratio / Distance to Crash:

- **10% increase in the glide ratio by increasing the distance to crash**
  - Overall crash frequency increases
    (medium impact, screening threshold not reached)
- **10% decrease in the glide ratio by decreasing the distance to crash**
  - Overall crash frequency increases
    (low impact)

Sensitivity Analysis (cont.)

Solomon Model Gamma Factor:

- To show the relationship between the gamma factor (γ) and the crash frequency, the crash frequency was determined with all aircraft types using the same value for γ:
  - $\gamma = 2$ for all aircraft
    - Overall crash frequency decreases (low impact)
  - $\gamma = 1.6$ for all aircraft
    - Overall crash frequency remains the same
  - $\gamma = 1$ for all aircraft
    - Overall crash frequency increases (large impact but overall frequency remains below screening threshold)

Sensitivity Analysis (cont.)

Honoring the Flight-restricted Airspace:

- To exceed the frequency threshold, more than 1,500 violations per year of the flight-restricted airspace would have to occur, assuming the following adverse flight conditions:
  - Additional flights are at 6,500 ft MSL (instead of 14,000 MSL)
  - Flights perform combat training (instead of overflights in a straight and level manner)

13 NRC ITEMS

- **Implementation and Monitoring of Flight Restricted Airspace:**
  - DOE has decreased Flight Restricted Airspace radius to 4.9 NM, which will be just within the northern border of the land withdrawal area. This will place the Flight Restricted Airspace completely within DOE/NNSA airspace control.
- **Pilot Actions Outside of Flight Restricted Airspace:**
  - No credit is being taken for pilot actions
  - The estimated frequency for crashes that originate outside of the Flight Restricted Airspace is the dominant contributor to the total crash frequency.
  - This was addressed in August 2005 revision to the analysis and augmented in October 2006 revision.

13 NRC ITEMS (cont.)

- **Crash Frequency Analysis Methodology Using Solomon's Model:**
  - This concerns altitude variations of flights
  - Analysis uses gamma factors inherent to the Solomon Model
  - Gamma factors are based on aircraft type, which includes the altitude of the aircraft
  - Addressed in the current revision of the analysis with a more detailed discussion and a sensitivity analysis.
    - The analysis shows that the data collected on military aircraft crashes compares well with Solomon's model.
    - About 5% of the crashes from the data travel approximately 3 mi or more following pilot ejection.
    - The Solomon Model shows that at 3 mi from the edge of the airway, about 3% of the planes travel further

13 NRC ITEMS (cont.)

- **Use of United States Air Force (USAF) Mishap Reports:**
  - A discrepancy existed in the number of USAF crash reports used in the May 2005 analysis and the data used by Private Fuel Storage
  - The October 2006 analysis adds information on about 160 additional mishaps
- **Categorizing USAF Mishap Reports:**
  - The frequency analysis relies on the list of aircraft mishaps and a categorization of the mishaps as applicable or not applicable to the repository. The categorization is based on the identified cause of the event.
  - A revised discussion clarifying the process of categorizing the events has been included in the August 2005 revision of the analysis and augmented in the October 2006 revision.
  - A sensitivity to the categorization has been included in the October 2006 analysis

13 NRC ITEMS (cont.)

- **USAF Mishap Reports (unknown causes of mishaps):**
  - Similar to previous concern
  - The number of "unknown crashes" has been reduced to one based on the additional information collected at the Air Force Safety Center. The event has been conservatively categorized as a Type 1 event, meaning that it is applicable to overflights.
- **USAF Mishap Reports (mismatch of data):**
  - Similar to previous two concerns
  - Data has been verified and compared to PFS licensing exhibits

13 NRC ITEMS (cont.)

- **Jettisoned Ordnance:**
  - A discussion of jettisoned ordnance has been included in the October 2006 analysis
  - DOE is proposing that no ordnance be carried over the Flight Restricted Airspace
  - Ordnance originating from outside the Flight Restricted Airspace has been screened on probability
- **Cruise Missile Testing at Nevada Test Site:**
  - Discussion has been included in the October 2006 revision of the *Identification of Aircraft Hazards* report
  - Cruise missiles have been treated as an ordnance and screened on probability

13 NRC ITEMS (cont.)

- Bird Impact:
  - A reference supporting the assumption on bird impacts has been included in the August 2005 revision to the analysis. In addition, the additional military crash data collected supports the assumption.
- Utilization Factor – Aging Pads:
  - Eliminated by the revised analysis inputs and assumptions
- Structural Credit – Analysis Methodology:
  - Eliminated by the revised analysis inputs and assumptions
- Structural Credit – Transportation Casks:
  - Eliminated by the revised analysis inputs and assumptions

Summary

- Preliminary indications are that event sequences as a result from aircraft hazards are beyond Category 2

- We believe that we have provided the methodology and basis, both today and in the Frequency Analysis of Aircraft Hazard report, such that it will lead to the resolution of your 13 items

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

Preclosure Source Terms and Consequence Methodology

Presented to:

NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:
Dale Dexheimer - Senior Engineering Specialist
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Acronyms

ALARA	As Low as Reasonably Achievable
ARF	Airborne Release Fraction
CD-1	Critical Decision - 1
CSNF	Commercial Spent Nuclear Fuel
DPC	Dual Purpose Canister
GROA	Geologic Repository Operations Area
HEPA	High Efficiency Particulate Air (filter)
HLW	High Level Waste
HVAC	Heating, Ventilation, and Air Conditioning
PIDAS	Perimeter Intrusion Detection and Alarm System
SNF	Spent Nuclear Fuel
TAD	Transport, Aging, and Disposal Canister
WHF	Wet Handling Facility

OUTLINE

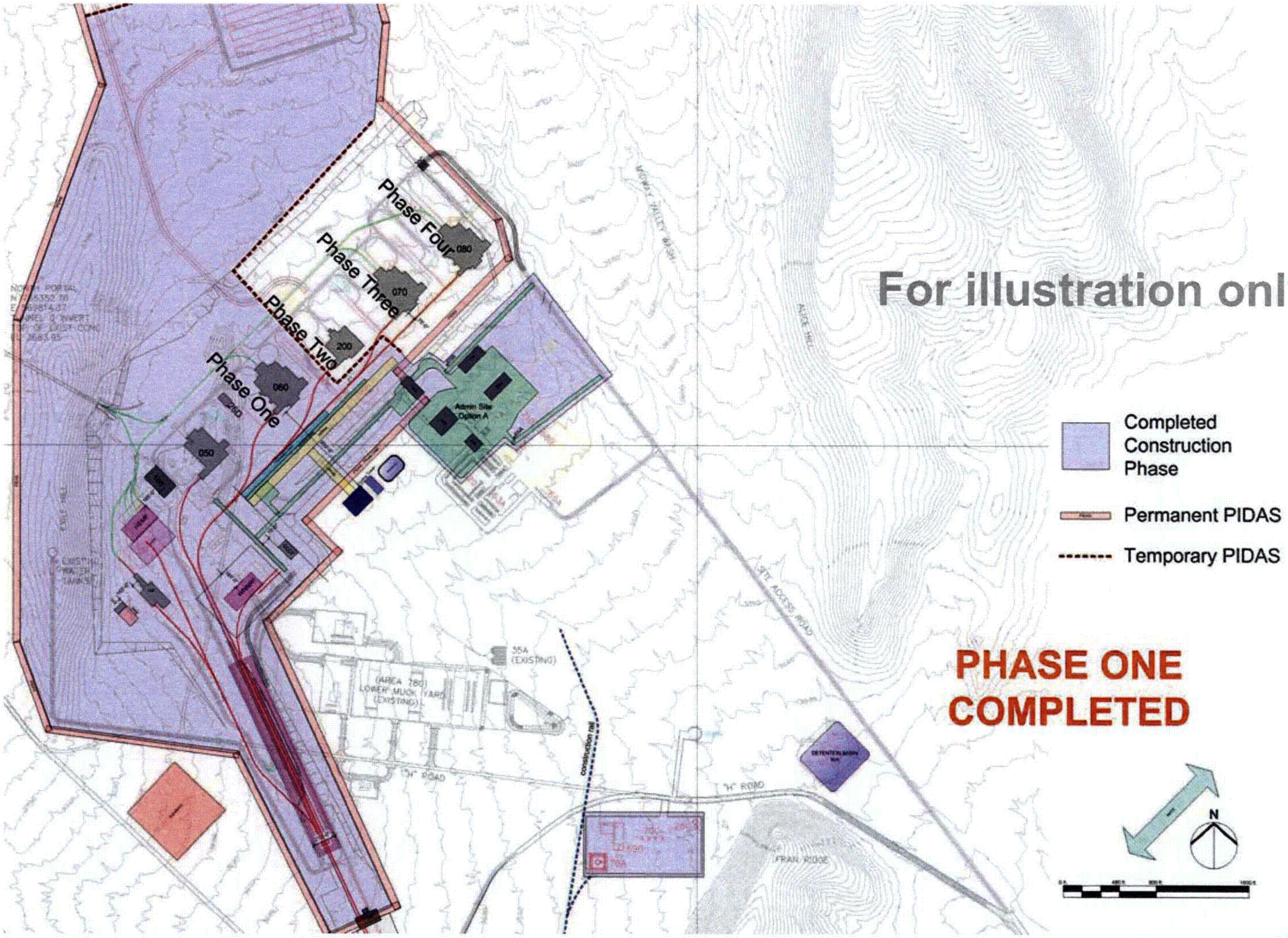
- Radiation Protection Program and ALARA Overview
- Source Terms
- Consequence Methodology
- Uncertainty and Sensitivity Analysis
- License Application Supporting Documents

Radiation Protection Program Overview

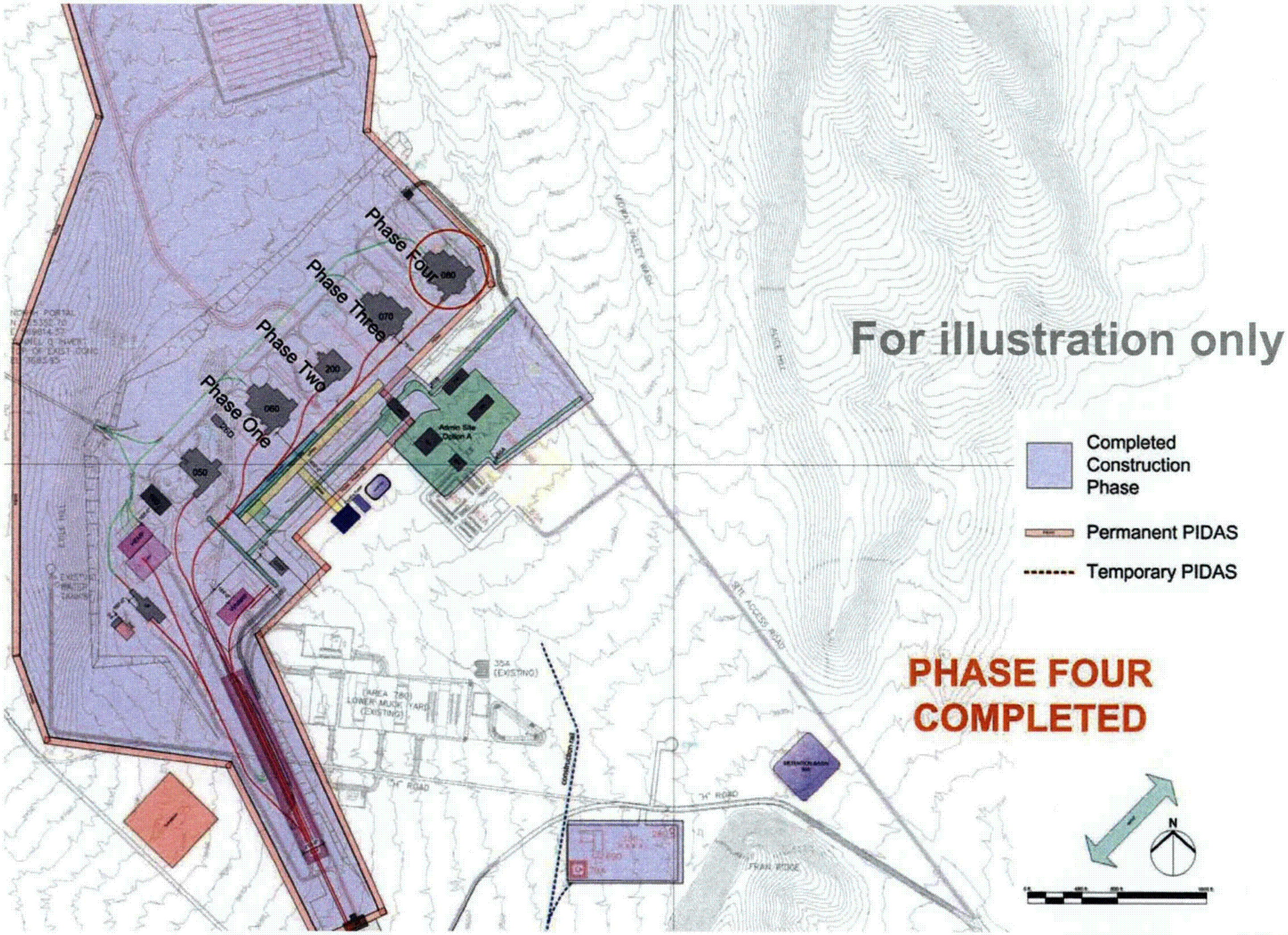
- Management policy and commitment to the goal of maintaining occupational worker and public doses below regulatory limits and consistent with ALARA principles
- Site and Operations Area Layout:
  - Offsite exposures - site location distant from population centers
  - Onsite exposures
    - GROA restricted area to protect individuals from radiation exposure is contiguous with security boundary
    - Additional radiological access controls are provided at individual facilities to further segregate areas
    - GROA restricted area expands during phased construction process

Note: On the following drawings, the security boundary is shown by the extents of the permanent and temporary Perimeter Intrusion Detection and Alarm System (PIDAS)

Radiation Protection Program Overview
Layout



Radiation Protection Program Overview Layout (cont.)



Radiation Protection Program Overview
Shielding

- **Radiation Shielding**
  - Contained radiation sources:
    - Processed waste forms (SNF and HLW)
    - Process wastes (HEPA, ion exchanger, etc.)
    - Staged and aged waste forms
  - Shielding Types:
    - Permanent bulk shielding
    - Transportation cask, aging overpacks, and transfer casks
    - Shield doors and labyrinths
    - Pool water

Radiation Protection Program Overview Shielding (cont.)

- **Design bases:**
  - Maximum source terms for waste forms
  - Barrier shielding thicknesses determined at location of maximum dose rate
  - Building areas classified by required access and occupancy and potential radiation level
  - Exterior building surfaces at grade: $\leq$ 0.05 mrem/hr
  - Interior barrier thicknesses based on expected occupancy: $\leq$ 0.25 mrem/hr in normally occupied spaces

Radiation Protection Program Overview Confinement and Ventilation

- **Confinement and Ventilation:**
  - Waste Form Confinement:
    - Waste form matrix – SNF fuel pellet and HLW glass matrix
    - Cladding - SNF
    - Sealed Containers – TAD, DPC, HLW and DOE SNF Canisters, and Waste Package
    - Sealed Casks – transportation
  - Facility Confinement and Ventilation:
    - Building areas classified by level of, or potential for airborne contamination
    - HVAC air flow design limits spread of radioactive contamination between confinement zones
    - Exhaust HEPA filtration mitigates airborne releases from waste processing areas
    - Subsurface air pressure gradients and ventilation barriers control flows

Radiation Protection Program Overview
Radiation Monitoring

- **Area and Airborne Radiation Monitors:**
  - Monitor radiological conditions and provide local display and alarm functions:
    - Area radiation monitors
    - Continuous air monitors
  - At surface and subsurface exhausts:
    - Airborne effluent monitors
  - Criticality detection and alarm monitors at locations consistent with Reg. Guide 3.71 and ANS standards

Radiation Protection Program Overview
ALARA

- **Goal to minimize the number of workers receiving doses > 500 mrem/yr**
- **ALARA is an integral part of the design process with strong management support and commitment**
- **Design principles considered for ALARA:**
  - Eliminate or reduce radiation sources prior to occupancy
  - Contain or confine radioactivity to reduce releases
  - Minimize time in radiation or radioactive airborne areas
  - Maximize distance from radiation sources
  - Use radiation shielding, including temporary shields

Radiation Protection Program Overview
ALARA (cont.)

- **ALARA in the Design Process:**
  - Baseline design developed with consideration of general ALARA design criteria and features
  - Perform Multi-discipline ALARA design reviews:
    - Involves designers and operational health physics personnel
    - Considers worker activities in each facility/area:
      - Normal operations, maintenance, surveillance, testing, etc.
      - Category 1 event sequences
    - Perform worker dose assessments as necessary
    - Identify design options to reduce potential exposures

Source Terms
Changed by All Canister Approach

Source Terms Changed by CD-1 Process:

- All canister approach (CSNF, Naval SNF, DOE SNF and HLW)
- Naval SNF, DOE SNF and HLW canisters placed directly into Waste Packages
- CSNF arrives mainly in TADs with some in DPCs or uncanistered in a transportation cask
- No handling of bare CSNF in air
- Features of TAD handling process:
  - CSNF waste to be placed into TADs at originating facility
  - Welded shut, not opened at YMP handling facilities
  - Provides confinement of the spent fuel sources, no contamination spread to handling areas
  - Placed directly into waste packages

Source Terms
Changed by All Canister Approach (cont.)

- Features of DPC or uncanistered fuel in cask handling process:
  - CSNF arriving in DPCs or cask handled under water in the WHF
  - Remotely controlled equipment to remove and insert into Waste Packages
  - Cask opened, DPCs cut open, and CSNF assemblies removed and staged under water

Source Terms
Changed by All Canister Approach (cont.)

- Features of DPC or uncanistered fuel in cask handling process:
  - Assemblies placed into TAD within overpack under water
  - TAD lid set in place and overpack lid secured in place under water
  - TAD and overpack raised and moved into welding station
  - Water level lowered to just below TAD lid
  - TAD welded shut, drained, vacuum dried, and inerted

Source Terms
Normal Operation

- **Minimal surface facility airborne releases:**
  - From processing DPCs or uncanistered CSNF
  - DPCs cut open, and CSNF assemblies removed under water
  - Releases from 1% CSNF with clad defects, under water in the WHF
  - 100% gaseous release (e.g., ^{85}Kr, ^{3}H) with credit for water decontamination factor for iodine:
    - Follow Regulatory Guide 1.183 guidance
- **Minor subsurface facilities releases:**
  - Resuspension of emplaced waste package surface contamination
  - Neutron activation of air and silica dust

Source Terms
Normal Operation (cont.)

- Direct radiation from contained sources:
  - In-process waste forms within shielded transfer equipment, shield walls, and/or pool water
  - Staged or aged waste forms outside facilities within shielded casks or overpacks
  - Accumulated waste (HEPA, dry active waste, pool ion exchanger, filters, etc.) within shield walls
  - Low-level liquid waste tank

Source Terms
Normal Operation (cont.)

- **Average PWR or BWR assemblies based on:**
  - Base case annual spent fuel arrival scenario for entire emplacement period:
    - Historical and projected fuel discharge characteristics (burnup, enrichment and decay times) for each year
    - Determine annual average fuel characteristics for each year
  - Select the maximum of the annual averages
  - Inventories calculated with SCALE v4.4
- **CSNF crud consisting of ^{60}Co and ^{55}Fe:**
  - Based on bounding measured PWR and BWR data:
    - PWR ^{60}Co - 140 $\mu Ci/cm^2$ and ^{55}Fe – 5,902 $\mu Ci/cm^2$
    - BWR ^{60}Co – 1,254 $\mu Ci/cm^2$ and ^{55}Fe – 7,415 $\mu Ci/cm^2$
  - To be decayed to the average age of spent fuel

Source Terms
Event Sequences

- Design goal to prevent or mitigate Category 1 and Category 2 event sequences through design features:
  - Category 1 event sequences ($\geq$1 during preclosure period)
  - Category 2 event sequences ($\geq 1 \times 10^{-4}$ during preclosure period)
- Hazards analyses underway:
  - Internal hazards analysis
  - External hazards analysis
- Event sequence analyses and categorization
  - Screening process to determine frequency of event sequences that could potentially result in a release of radioactivity or direct exposure

Source Terms
Event Sequences (cont.)

- Maximum PWR assemblies:
  - 5% initial enrichment, 80 GWd/MTHM burnup, 5 year cooling time
  - Maximum PWR fuel characteristics from fuel discharge data (477 kg initial uranium loading, 69 GWd/MTHM, 5.0% enrichment, and 5 years decay) conservatively increased to 80 GWd/MTHM to provide adequate design margin
- Maximum BWR assemblies:
  - 5% initial enrichment , 75 GWd/MTHM burnup, 5 year cooling time
  - Maximum BWR fuel characteristics from fuel discharge data (197 kg initial uranium loading, 65.55 GWd/MTHM, 4.28% enrichment, and 5 years decay) conservatively increased to 75 GWd/MTHM to provide adequate design margin
- CSNF crud with 5 year decay

Source Terms
Event Sequences (cont.)

- **Vitrified HLW Canisters:**
  - Per canister source terms for vitrified HLW provided by four sites: Savannah River Site, Hanford, West Valley, and Idaho National Laboratory
  - Basis: Projected maximum or bounding inventory per canister from each site
- **Naval SNF Canisters:**
  - Source A - Crud release for canister breach without fuel damage
  - Source B - Fuel and crud release for canister breach with fuel damage
  - Basis: Bounding releases provided by the Naval Nuclear Propulsion Program (NNPP)

Source Terms
Event Sequences (cont.)

- **DOE SNF Canisters:**
  - Over 250 DOE fuel types
  - Calculated for specific reactor types, fuel types, burnups and decay times
  - Basis: Bounding and average inventories estimated by National Spent Nuclear Fuel Program

Consequence Methodology
Normal Operation

- **Key Assumptions:**
  - Maximum annual throughput for each year of operation
  - Credit for HEPA filter efficiency of 99% per stage:
    - Two stage HEPA filter efficiency credit 99.99%
    - ASME N509-1989 defines a HEPA filter as having an efficiency of $\geq$ 99.97% for 0.3 µm particles
    - Testing per Regulatory Guide 1.52 requires a leakage and penetration of less than 0.05% per stage
  - Aging pads and staging areas at full capacity
  - Annual average meteorology

Consequence Methodology Category 1 & 2 Event Sequences

- Key Assumptions:
  - 100% of the fuel assemblies are damaged following a drop event
  - HEPA filter efficiency of 99% per stage:
    - Two stage HEPA filter efficiency of 99.99%
    - Credit only during HVAC operating period supported by reliability analyses
  - 95th percentile meteorology

Consequence Methodology
Onsite Meteorology

- ARCON96 methodology
- Based on 5 year (1998-2002) hourly meteorological data from Site 1 located 1 km SSW of north portal
- Dispersion factors determined:
  - Between all surface and subsurface facility exhausts and intakes
  - At restricted area boundary distance from surface facilities exhausts
  - Building wake included
  - Elevation differences and wind direction
  - Exhaust velocity and vent diameter considered

Consequence Methodology Site Meteorology

- Based on 5 year (1998-2002) hourly meteorological data from Site 1
- Annual average χ/Q determined using Regulatory Guide 1.111 for a ground level release
- 95th percentile χ/Q determined using Regulatory Guide 1.145 for a ground level release
- χ/Qs calculated at:
  - Site boundary from surface facilities and subsurface exhaust shafts (sector dependent)
  - Nearest resident as identified in Biosphere Model Report
  - Restricted area boundaries surrounding each subsurface exhaust shaft

Consequence Methodology Dose Receptor

- **Facility Radiation Worker:**
  - Occupational workers within the restricted area (GROA)
  - Direct radiation from contained sources
  - Airborne doses from surface and subsurface facility releases
  - 2000 hrs/yr for normal operations
  - Considered for Category 1 Event Sequences

Consequence Methodology Dose Receptor (cont.)

- **Offsite public:**
  - Located at the site boundary
  - Consumes locally produced food grown at location of closest resident
  - Characteristics of receptor based on Biosphere Model site-specific survey data:
    - Occupancy
    - Fraction consumption of locally grown food
  - Considered for normal operations, Category 1 and Category 2 Event Sequences
- **No credit taken for any post accident protective measures for offsite members of the public**

Consequence Methodology Dose Receptor (cont.)

- **Onsite public:**
  - During phased construction:
    - On site outside of the restricted area
    - construction workers adjacent to operating facilities
  - Following completion of construction:
    - On site outside of the restricted area
    - Located at the restricted area boundary closest to the facility
  - 2000 hrs/yr for normal operations
  - Considered for normal operations and Category 1 Event Sequences
- **No credit taken for any post-accident protective measures for onsite members of the public**

Consequence Methodology Confinement Barriers

- **SNF Cladding:**
  - Normal operations - cladding confinement intact:
    - 1% fuel assemblies have cladding defect
  - Drop event:
    - Largely intact but damaged following a drop event
    - Release fractions based on single tear
- **TAD, DCP, or Canister:**
  - Normal operations - canister confinement intact
  - HLW canister drop event:
    - Particulate leak path factor (LPF) of 0.1
      - » NUREG/CR-6672 indicates a particulate LPF of 0.02 for impact speed of 60 mph pressurized to 5 atmospheres

Consequence Methodology Confinement Barriers (cont.)

- **Transportation Cask:**
  - Cask drop event with CSNF or HLW:
    - Particulate Leak Path Factor of 0.1
    - Based on NUREG/CR-6672 – 60 mph impact
- **Transfer Cask and Aging Overpack:**
  - No retention
- **Handling Facility Structure:**
  - Provides ventilation confinement for operational HEPA filtration
  - No credit for building retention of released particulates through plateout or gravitational settling:
    - Future MELCOR study to evaluate

Consequence Methodology
TAD Drop and Breach

- **Key Assumptions:**
  - TAD contains 21 PWR or 44 BWR assemblies
  - Initial burst and subsequent oxidation of CSNF
  - Burst isotopic fractions available for release based on Interim Staff Guidance – 5 (ISG-5) and other literature
  - CSNF clad unzipping due to oxidation begins about two hours after TAD is postulated to breach
  - CSNF fully oxidized to U_3O_8 powder in about 30 days
  - Oxidized CSNF release fractions based on *Commercial Spent Nuclear Fuel Handling in Air Study* (March 2005)

Consequence Methodology Initial Burst Release

- **Airborne Release Fractions:**
  - 0.3 for gases (Tritium, Krypton, Iodine)
  - 2×10^{-4} for volatiles (Cesium, Rubidium)
  - 3×10^{-5} for particulates, including Strontium
  - 1.5×10^{-2} for crud (Cobalt, Iron)
- **Respirable Fractions:**
  - 1.0 for gases, volatiles, and crud
  - 5×10^{-3} for particulates

Consequence Methodology Oxidation Release

- Releases occur continuously during oxidation/unzipping process
- Airborne Release Fractions:
  - 1.0 for Tritium
  - 0.3 for other gases (Krypton and Iodine)
  - 1.2×10^{-3} for fuel fines and particulates
- Respirable Fractions:
  - 1.0 for all oxidation releases
- To be re-evaluated pending completion of fuel testing at Pacific Northwest National Laboratory

Consequence Methodology
GENII Version 2

- GENII developed by Pacific Northwest National Laboratory
- Customized by Pacific Northwest National Laboratory to meet specific YMP needs:
  - Input site specific χ/Q values
  - Sensitivity analyses using Federal Guidance Report 13 dose conversion factors
- Key features include:
  - Models for radionuclide transport in air
  - Decay and progeny generation
  - Plume dispersion and depletion
  - Building wake
  - Plume rise
  - Wet and dry deposition
  - Intake by plants and animals
  - Consumption of contaminated plants and animal products
  - Capability to perform sensitivity/uncertainty analyses

Consequence Methodology
Dose Conversion Factors

- The exposure pathways modeled in GENII Version 2 include inhalation, air submersion, groundshine, resuspension, and ingestion
- ICRP 60 based weighting factors applied to each of the 23 organs to obtain the effective dose equivalent
- Federal Guidance Report 13 dose conversion factors for inhalation and ingestion:
  - Inhalation lung class based on ICRP-72 recommendations
  - Selection based on fission product release or oxide form
  - Inhalation dose conversion factors based on 1 μm particle size
- Shielded direct doses calculated using ANSI 1977 flux to dose conversion factors
  - Evaluation to compare with ICRP-74 dose conversion factors based on ICRP-60 weighting factors

Uncertainty and Sensitivity Analysis

- Relative importance of the input parameters
- Establish individual parameter contribution to total uncertainty
- Use SUM^3 statistical analysis module of GENII Version 2
- Perform regression analysis to determine the sensitivity of parameters

Dose Criteria
Normal Operations and Cat 1 Events

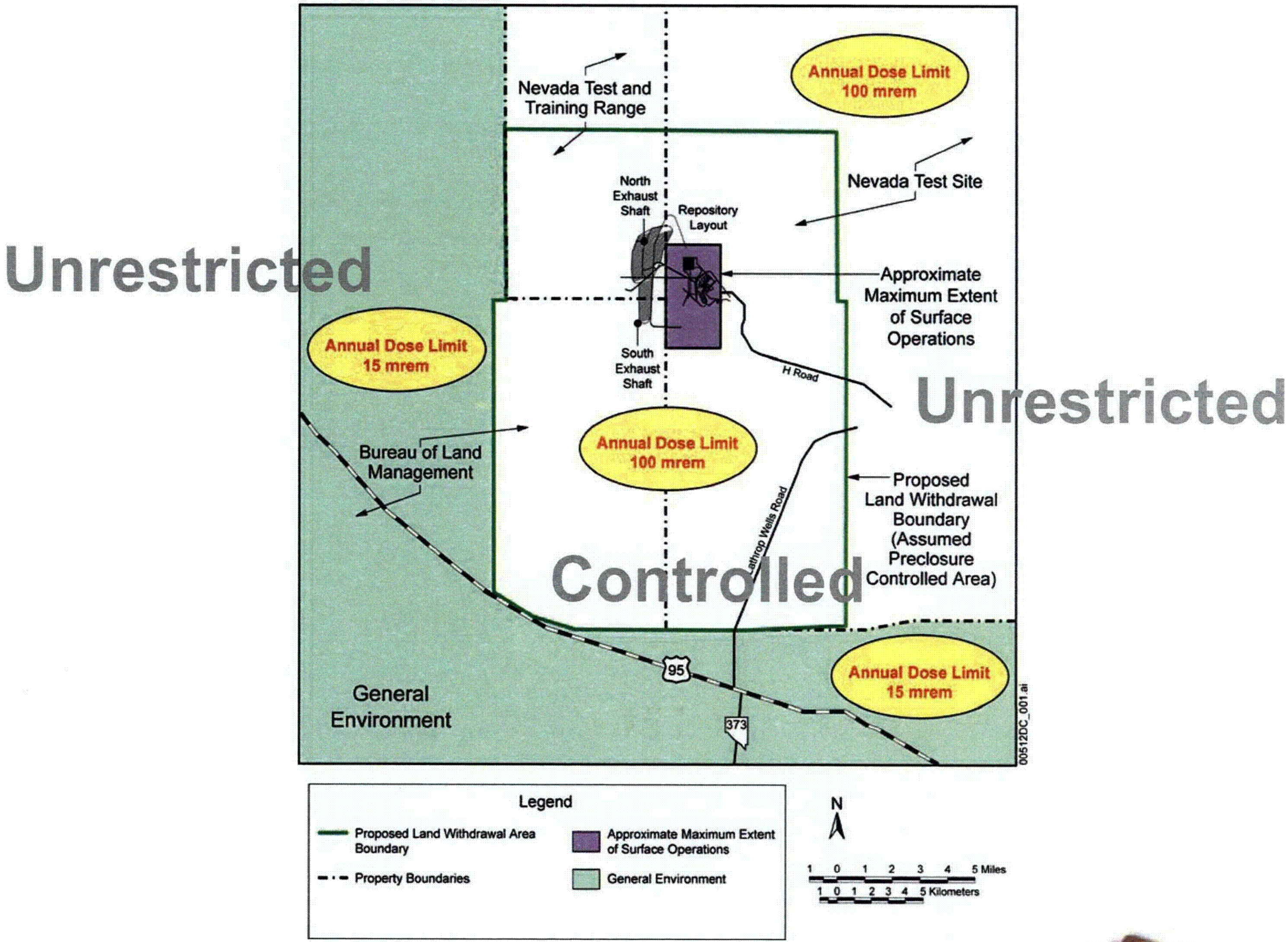
Event Sequence Type	Dose Type	Performance Objectives		
		Worker	Onsite Member of the Public	Offsite Member of the Public
Normal operations and Category 1	TEDE	5 rem/year	100 mrem/year	15 mrem/year 100 mrem/year
Normal operations and Category 1	Highest TODE	50 rem/year	NA	NA
Normal operations and Category 1	LDE	15 rem/year	NA	NA
Normal operations and Category 1	SDE	50 rem/year	NA	NA
Normal operations and Category 1	External dose: Highest of DDE, LDE, or SDE for the unrestricted area	NA	NA	2 mrem in any 1 hour

Notes: DDE – deep dose equivalent; LDE – lens dose equivalent; SDE – shallow dose equivalent; TEDE – total effective dose equivalent; TODE – total organ dose equivalent; NA – not applicable.

Dose Criteria
Category 1 Dose Evaluation Methods

- Category 1 event doses are aggregated with normal operation doses
  - Weighted by the event frequency
  - Added to the normal operation doses
  - Demonstrate compliance with 10 CFR Part 63.111
- Individual Category 1 event doses are compared with 10 CFR Part 63.111
- Any combination of Category 1 events that can occur in any one year are compared with 10 CFR Part 63.111

Radiological Control Boundaries



Dose Criteria
Category 2 Event Sequences

Event Sequence Type	Dose Type	Performance Objectives		
		Worker	Onsite Member of the Public	Offsite Member of the Public
Category 2	TEDE	NA	NA	5 rem/event
Category 2	Highest TODE	NA	NA	50 rem/event
Category 2	LDE	NA	NA	15 rem/event
Category 2	SDE	NA	NA	50 rem/event

Notes: LDE – lens dose equivalent; SDE – shallow dose equivalent; TEDE – total effective dose equivalent; TODE – total organ dose equivalent; NA – not applicable.

LA Supporting Documents

- GROA Airborne Release Dispersion Factor Calculation
- Normal Operation Airborne Release Worker Dose Calculation
- Category 1 Event Sequence Airborne Release Worker Dose Calculation
- Commercial SNF Accident Release Fractions
- GROA Worker Dose Assessment
- Preclosure Consequence Analysis

Reliability Assessment

Albert Wong

United States Nuclear Regulatory Commission

NRC/DOE Technical Exchange and Management Meeting
November 7 - 9, 2006

Outline

- Background
- Discuss Key Messages
  - Technical Basis
  - Reliability Estimation
  - Uncertainty
- Summary

Background

- Tech Exchange (TE) on 10 CFR Part 63 Preclosure Safety Analysis (PCSA), May, 16 – 17, 2006
  - NRC communicated key messages
- DOE *Summary of PCSA Reliability Assessment Methodology*, 08/25/06
- HLWRS-ISG-02, *Preclosure Safety Analysis - Level of Information and Reliability Estimation* (out for public comment)
- NRC key messages from the May TE remain unchanged.
  - Need for reliability estimate technical bases
  - Need to address uncertainty in reliability estimates

Technical Basis

- Technical bases needed to support reliability estimates for structures, systems and components (SSCs) in the PCSA
  - Applicability (e.g., operating experiences, environment) of failure data from other facilities and industries
  - Assumptions/limitations
  - Uncertainty

- Technical bases should be clearly articulated and documented in the PCSA.

Reliability Estimation

- Approaches to estimate reliability:
  - Accepted engineering practice
  - Empirical data
  - Modeling
- Determine the reliability of SSCs at the highest level (typically at the system level)
- If necessary, then determine the reliability at the next level down
  - Estimate system reliability based on sub-system or component level reliability
  - Justifications needed to assess unique SSCs as a aggregate of individual sub-systems or components

Uncertainty - General

- PCSA: A systematic examination of the site; the design; and the potential hazards, initiating events, and their resulting event sequences and potential radiological exposures to worker/public (10 CFR Part 63.102(f))

- Statement of Considerations, 66 FR 55742, Nov 2, 2001
  - DOE has flexibility to select the type of analysis
  - Need to support the approach
  - Need to take into account uncertainties

Uncertainty - Issues

- Particular attention should be given when:
  - An event sequence ***frequency*** is near a category boundary, or
  - An event sequence ***dose*** is close to a dose limit
- Consideration of uncertainty should include sensitivities to parameter distributions and assumptions used in a specific scenario
- Uncertainty in the quantified inputs to an event sequence should be taken into account
- Engineering judgments, if used, should have defensible technical bases

Summary

- Reliability estimates for SSCs in the PCSA should be based on defensible technical bases
- Reliability Estimation should start with SSC analogues at the highest level, then go to the next level down, if necessary
- Uncertainty should be taken into account in estimating reliability of SSCs in the PCSA

Human Reliability Analysis

Tina Ghosh

United States Nuclear Regulatory Commission

NRC/DOE Technical Exchange and Management Meeting
November 7 - 9, 2006

Outline

- Key regulatory requirements
- Key messages
- Summary

Key 10 CFR Part 63 Requirements

- The PCSA of the GROA must include:
  - An identification and systematic analysis of naturally occurring and human-induced hazards at the GROA, including a comprehensive identification of potential event sequences [63.112(b)]
  - An analysis of the performance of the SSCs to identify those that are important to safety. This analysis identifies and describes the controls that are relied on to limit or prevent potential event sequences or mitigate their consequences [63.112(e)]

HRA Integration with PCSA

- The human reliability analysis (HRA) for the Yucca Mountain GROA should be fully integrated with the overall preclosure safety analysis (PCSA).

General Guidance for HRA

- Recent NUREGs developed for nuclear power plants, can provide *general* guidance
  - NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA)”
  - NUREG-1842, “Evaluation of Human Reliability Analysis Methods Against Good Practices”

- Analysis elements that need to be addressed for the GROA PCSA include:
  - a technically appropriate HRA process
  - treatment of errors of commission (EOCs)
  - integration of the HRA into the overall PCSA.

Site-Specific and Facility-Specific HRA

- In addition to guidance from these NUREGs, qualitative insights from operating experience from facilities and activities similar to those planned at the GROA should be considered to develop and implement the HRA approach.

- HRA methods, data, and assumptions need to be justified for application to the GROA.

- Such justification should be made through the results of a qualitative HRA analysis.

Qualitative HRA Analysis

- Conceptual Modeling of Human Performance
  - Includes, for example, the identification of human failure events, and the identification of important factors influencing human performance (both traditional "performance shaping" and contextual factors)
  - Should be a large focus of the overall HRA effort
  - Should be the basis for all other HRA process steps
  - Should guide selection of appropriate HRA quantification methods

Ties to Training and Programmatic Controls

- Training and programmatic controls should be tied to the HRA.
- Assumptions made in the HRA should be supported by an appropriate personnel training program and other administrative controls for safety.
- Insights from HRA should be reflected in the development and implementation of training and administrative programs for safety.

Documentation

- In a license application, documentation of the HRA for DOE's PCSA should be sufficiently detailed such that both the qualitative and quantitative analysis, as well as the ties to training and programmatic controls, can be reviewed.

Summary

- HRA should be fully integrated with the PCSA.
- General guidance on HRA should be considered along with site- and facility-specific factors.
- Qualitative analysis for HRA is important.
- HRA assumptions should be reflected in training and administrative programs.

Preclosure Licensing Specifications and Training

Rosemary Reeves

United States Nuclear Regulatory Commission

NRC/DOE Technical Exchange and Management Meeting
November 7 - 9, 2006

Outline

Preclosure Licensing Specifications:

- License Specifications - Overview
- Content of LA
- Probable Subjects of License Specifications
- Revisions to License Specifications
- Summary

License Specifications

- License specifications (LS) define the conditions for safe operation of the GROA and assure key safety controls are maintained
- NRC will impose LS based on important design assumptions and considerations in the PCSA and items identified by DOE

License Specifications

- Consist of license conditions and technical specifications
- May include:
  - functional and operating limits
  - monitoring instrument and control settings
  - limiting conditions
  - surveillance requirements
  - design features
  - administrative controls
- Should be specific, observable, measurable, or verifiable

Content of LA

10 CFR Part 63.21(c)(18) requires:

- The Safety Analysis Report must include identification and justification for selection of variables, conditions, or other items that are probable subjects for license specifications
- Special attention given to items that significantly influence final design

Probable Subjects for License Specifications

- Amounts, physical form, and radionuclide content of the high-level waste being disposed
- Key design features of ITS SSCs
- Key administrative controls and programs
- Key parameters and limiting conditions of operation

Revisions

- Revision of LS would require amendment to construction authorization or a license amendment, requiring NRC approval
- **However,** 10 CFR 63.44 provides criteria for changes to the SAR without a license amendment

Summary

- NRC would like to understand DOE's plans for developing preclosure licensing specifications and their bases
- License specifications assure key safety controls will be maintained per LA
- DOE will identify probable subjects for license specifications, but NRC will impose them and approve changes to them

Training

Outline

- Training Overview
- Regulatory Bases
- Training Requirements
- Systems Approach to Training
- Guidance on Training
- Summary

Training - Overview

- 10 CFR Part 63 includes requirements for personnel training, indoctrination, and qualification
- Comprehensive and integrated training & qualification program
- Consider training in the design and engineering of the facilities, components, and operations
- Important to safe operation and human reliability

Regulatory Bases

- Part 63.21(b)(3): The LA shall contain a description of the security organization personnel training and qualification plan
- Part 63.21(c)(22)(iii): The Safety Analysis Report shall contain personnel qualifications and training requirements
- Part 63.31(a)(3)(iv): Personnel training program for operations must comply with 10 CFR 63, Subpart H

Regulatory Bases (continued)

- Subpart H:
  - Operation of ITS systems & components only by trained and certified personnel
  - Training, proficiency testing, certification, and requalification
  - Physical condition and general health of operators
- Subpart G: QA program indoctrination and training of personnel performing activities affecting quality, special processes, and auditors

Training Requirements

- Specific training requirements for:
  - Operators/Supervisors of ITS Systems & Components
  - Specialized Trades (e.g. welders, crane operators)
  - Material Control & Accountability
  - Quality Assurance Personnel & Auditors
  - Security Force
  - Radiation Protection Personnel
  - Maintenance Personnel

Systems Approach to Training

- A systems approach to training (SAT) as defined in 10 CFR 55.4 may be considered, using risk-informed and performance-based criteria
- SAT elements include:
  - Analysis
  - Learning objectives
  - Design and implementation
  - Trainee evaluation
  - Program evaluation and revision

Guidance on Training

- NUREG 1804, Yucca Mountain Review Plan
- NUREG 1220, Training Review Criteria and Procedures
- ANSI/ANS 3.1-1993
- NRC Regulatory Guide 1.8
- INPO-managed accreditation program
- Other Standards, NRC Regulatory Guides and NUREGs

Summary

- 10 CFR Part 63 identifies various requirements for training, indoctrination and qualification of personnel
- Elements of training program should be factored into design and engineering of facilities, components, and operations
- Systems Approach to Training has been successful in other applications and may be considered for YMP

Preclosure Criticality

Sheena Whaley

United States Nuclear Regulatory Commission

NRC/DOE Technical Exchange and Management Meeting
November 7 - 9, 2006

Outline

- Key 10 CFR 63 Requirements
- Relevant Guidance
- Criticality Event Sequences
- Criticality Safety Analysis Administrative Margin
- Neutron Absorbing Materials
- Preclosure Burnup Credit
- Summary
- Questions and Answers

Key 10 CFR 63 Requirements

- A description and discussion of the design, including the relationship of the design bases to the design criteria (63.21(c)(3), 63.112(f))
- The identification and systematic analysis of potential event sequences (63.112(b))
- An analysis of means to prevent and control criticality to identify potential items important to safety (63.112(e)(6))

Existing Guidance

- Primary Guidance
  - Regulatory Guide 3.71: Nuclear Criticality Safety Standards for Fuels and Material Facilities
- Useful technical practices
  - NUREG -1520: Standard Review Plan (SRP) for the Review of a License Application for a Fuel Cycle Facility
  - NUREG -1536: SRP for Dry Cask Storage Systems
  - NUREG-1718: SRP for the Review of an Application for a Mixed Oxide Fuel Fabrication Facility
  - NRC/NMSS Interim Staff Guidance documents
- May need adaptation for Part 63 PCSA requirements

Criticality Event Sequences

- Criticality analyses should include credible events
  - Depends on fuel handling processes and type and condition of fuel
- Potential credible events:
  - Fuel misloads,
  - Damaged fuel, and
  - Optimum moderation
- Analyze most reactive credible fuel configuration

Criticality Safety Analysis Administrative Margin

- Technical basis needed for the administrative margin
- Acceptable upper subcritical limit administrative margin, evaluation of biases and uncertainties
- For commercial spent nuclear fuel, typical administrative margin of 0.05 and evaluating biases and uncertainties at a 95 percent confidence level
  - Criteria might also be applied to Category 1 and 2 event sequences with a sufficient technical basis

Criticality Safety Analysis Administrative Margin (continued)

- Smaller margins generally require more technical justification
- Draft NRC Fuel Cycle Safety and Safeguards Interim Staff Guidance-10, Revision 2, "Justification for Minimum Margin of Subcriticality for Safety," provides guidance

Neutron Absorbing Materials

- Reliability and performance credited in PCSA should determine testing requirements
- If neutron absorber is important to safety:
  - May need requirements on the absorber quality and the handling, fabrication, and test activities
- Qualification plan for neutron absorbers not currently approved
- Demonstrate acceptability and durability of the neutron absorber over the licensed service life

Preclosure Burnup Credit

- NRC-DOE interaction on DOE burnup credit strategy
- Existing NRC guidance only provides for partial burnup credit - lack of relevant experimental data
- Analyses should address uncertainty in data used to justify burnup credit

Summary

- Key 10 CFR 63 Requirements
- Relevant Guidance
- Criticality Event Sequences
- Criticality Safety Analyses Administrative Margin
- Neutron Absorbing Materials
- Preclosure Burnup Credit

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

November 8, 2006
Las Vegas, NV

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS:
1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY,
3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

November 7 and 8, 2006
8:00 AM – 5:00 PM (PT)
11:00 AM – 8:00 PM (ET)

November 9, 2006
8:00 AM – 12:00 PM (PT)
11:00 AM – 3:00 PM (ET)

U. S. Nuclear Regulatory Commission Hearing Center
Pacific Enterprise Plaza, Building 1
3250 Pepper Lane
Las Vegas, Nevada 89120

And via Teleconference to:

U. S. Nuclear Regulatory Commission
Two White Flint North, Room T 7A-1
11545 Rockville Pike
Rockville, MD

Center for Nuclear Waste Regulatory Analyses
Conference Room A-237, Bldg. 189
6220 Culebra Road
San Antonio, TX

INTERESTED PARTIES MAY PARTICIPATE VIA TELECON BY CALLING
1-800-638-8081, Passcode 8755# or 301-231-5539, Passcode 8755#

Tuesday November 7, 2006 (Aircraft Hazards and Source Terms and Consequence Methodology)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:30 AM	NRC Key Messages on Aircraft Hazards Assessment	NRC
9:00 AM	Background and Overview of Updated Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
9:30 AM	Changes in Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
10:00 AM	Break	All
10:15 AM	Aircraft Hazards Sensitivity Analysis	DOE/BSC (K. Ashley)
11:00 AM	Response to 13 NRC Issues (NRC letter of Aug. 2, 2005)	DOE/BSC (K. Ashley)
11:30 AM	Lunch	All
1:00 PM	NRC Key Messages on Source Terms and Consequence Methodology	NRC
1:30 PM	Radioactive Source Terms and Release Methodology	DOE/BSC (D. Dexheimer)
2:30 PM	Break	All
2:45 PM	Consequence and Analysis Methodology	DOE/BSC (D. Dexheimer)
3:30 PM	Uncertainty and Sensitivity Analysis	DOE/BSC (D. Dexheimer)
3:50 PM	Documents to be Revised	DOE/BSC (D. Dexheimer)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS:
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Wednesday November 8, 2006 (Reliability Assessment, Technical Specifications, and Training)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:15 AM	NRC Key Messages: - Reliability Assessment	NRC
9:00 AM	Reliability Assessment Overview	DOE/BSC (M. Frank)
9:45 AM	Break	All
10:00 AM	Human Reliability Assessment	DOE/BSC (M. Frank)
11:30 AM	Lunch	All
1:00 PM	Reliability Assessment for Structures, Systems, and Components	DOE/BSC (M. Frank)
2:15 PM	Break	All
2:30 PM	NRC Key Messages: - Technical Specifications - Systematic Approach to Training	NRC
3:00 PM	DOE Plans for Development of Technical Specifications	DOE (W. Spezialetti)
3:30 PM	DOE Plans for Systematic Approach to Training	DOE/MTS (J. McMahon)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS:
1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY,
3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

November 7 and 8, 2006	November 9, 2006
8:00 AM – 5:00 PM (PT)	8:00 AM – 12:00 PM (PT)
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INTERESTED PARTIES MAY PARTICIPATE VIA TELECON BY CALLING
1-800-638-8081, Passcode 8755# or 301-231-5539, Passcode 8755#

Thursday November 9, 2006 (Preclosure Criticality and License Application Requirements Mapping)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:15 AM	NRC Key Messages on Preclosure Criticality	NRC
8:45 AM	Preclosure Criticality Discussion	DOE/BSC
9:45 AM	Break	All
10:00 AM	License Application Status and Requirements Mapping	DOE (R. Warther)
10:10 AM	License Application Requirements Mapping	DOE/BSC (G. Ashley)
11:00 AM	Public Comments	All
11:15 AM	Break/Caucus	All
11:30 AM	Summary Discussion/Closing Remarks	NRC/DOE
12:00 PM	Adjourn	All

Overview of Preclosure Safety Analysis Reliability and Event Sequence Methodology

Presented to:
NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:
Michael V. Frank
Supervisor - Hazards, Event Categorization and Reliability Analysis
Bechtel SAIC Company, LLC

November 8, 2006
Las Vegas, NV

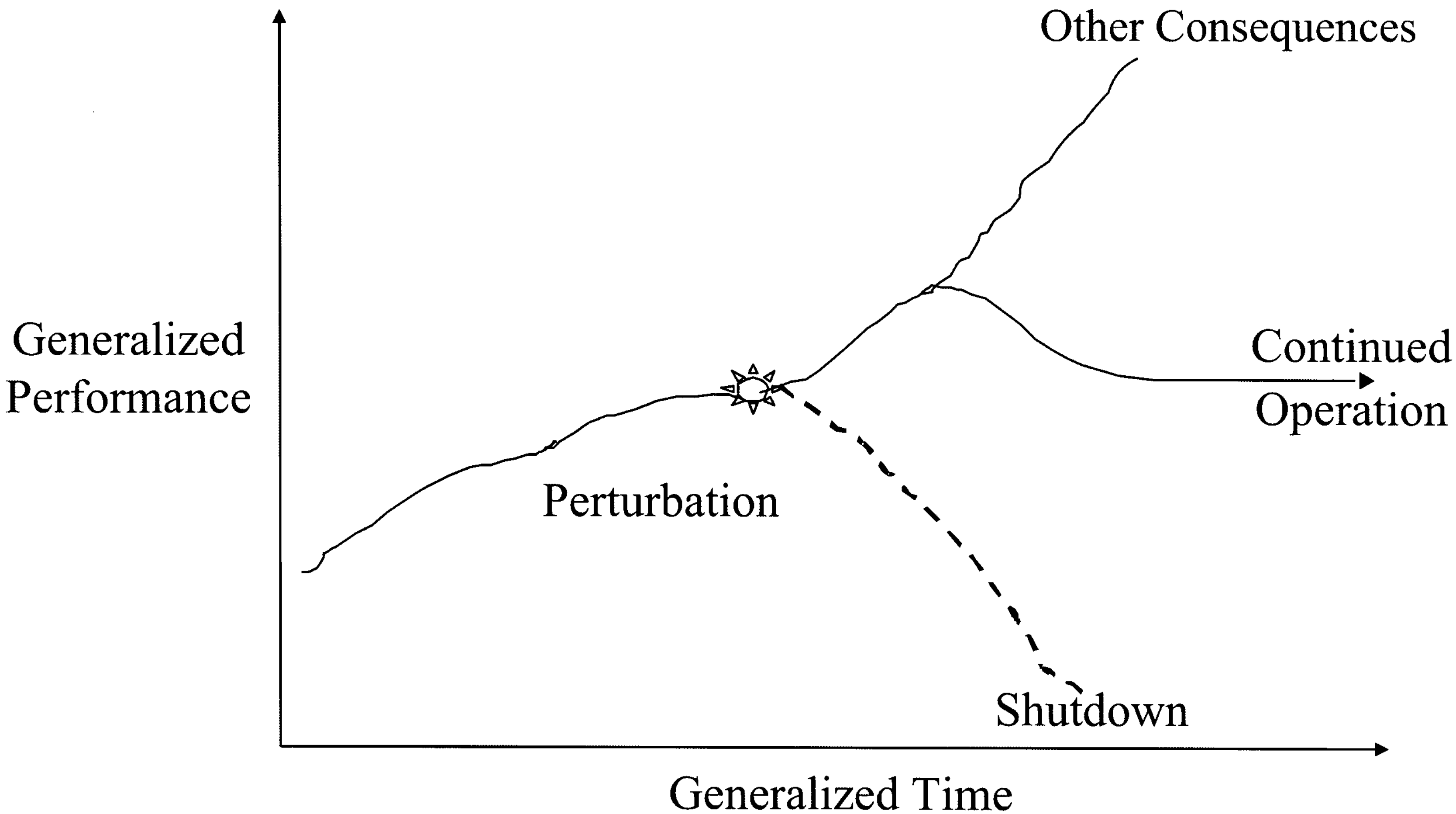
Outline

- Engineering and PCSA Coordination
- Concepts of Event Sequence Development
- Six Steps for Event Sequence Development and Reliability Analysis
- Summary

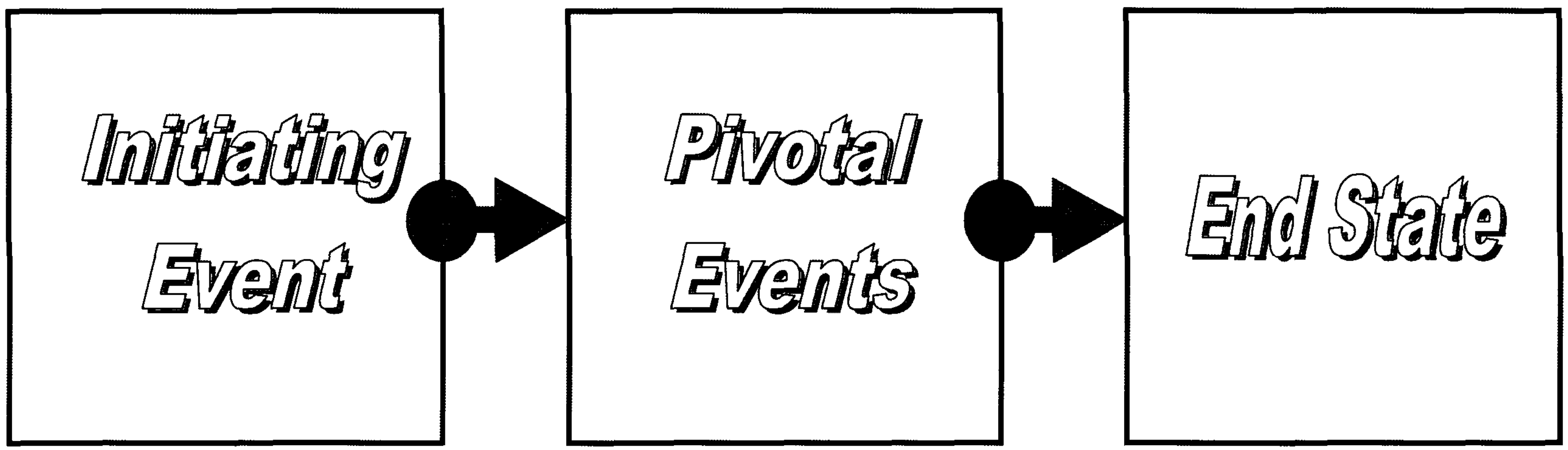
Engineering and PCSA Coordination

- Design, engineering and PCSA evolve together:
  - Engineering provides information for PCSA as it is developed
  - PCSA provides ongoing feedback to ensure that requirements are met
  - The PCSA is updated as the design changes to provide a check on compliance and feedback to design and engineering effort
- Important outputs:
  - List of SSCs that are important-to-safety (ITS)
  - Quantitative reliability allocation
  - Inputs to technical specifications and limiting conditions of operation to assure that operations remain in compliance

System Response to Perturbation



Principle of Event Sequences

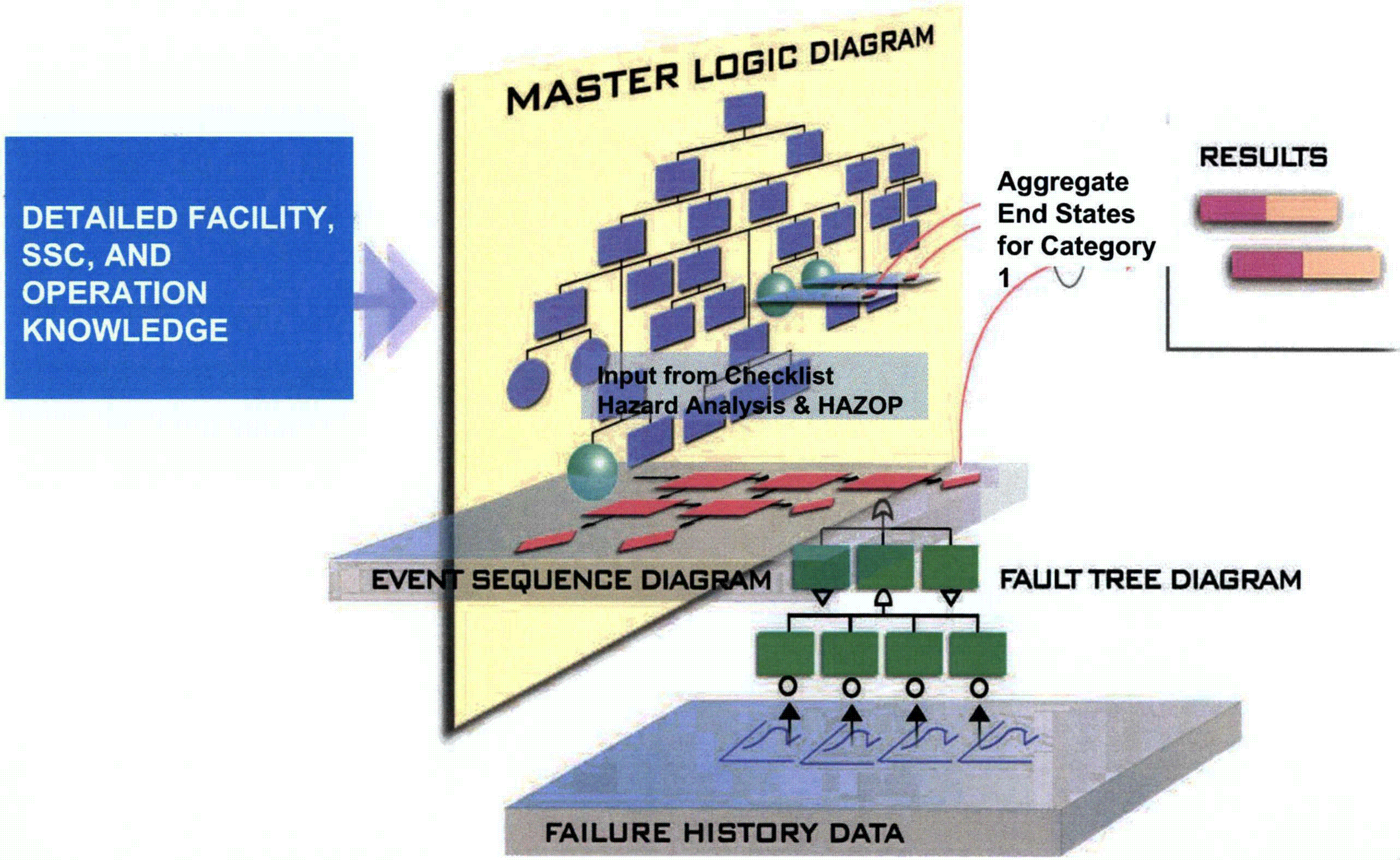


- The Perturbation

- Aggregative
- Mitigative
- Protective/preventive

- Condition for which consequences such as worker dose, public dose, and reactivity will be evaluated.

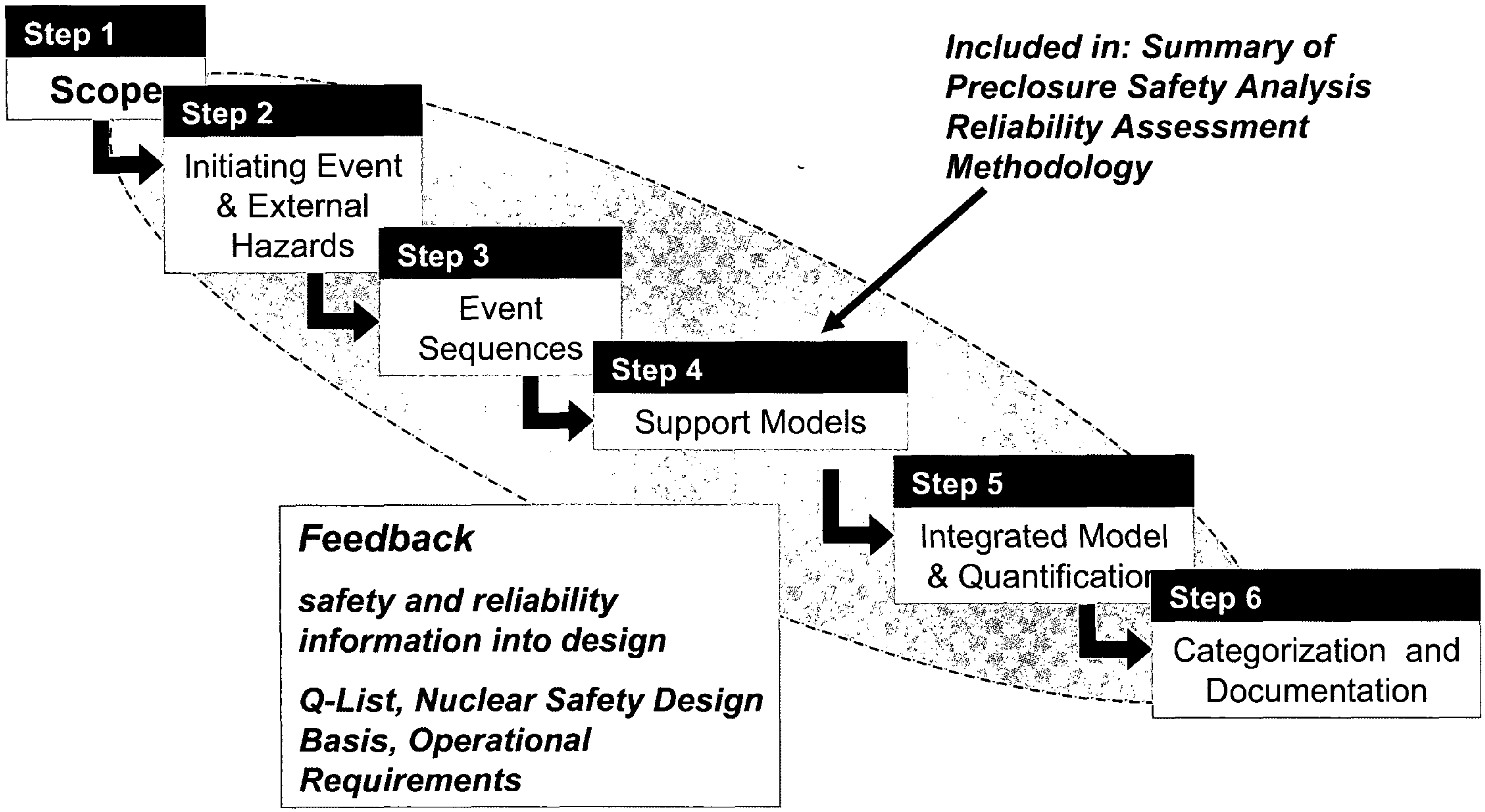
Event Sequence Development and Reliability Analysis



Event Sequence Development and Reliability Analysis

- The PCSA is using standard, accepted methods and techniques of risk assessment:
  - ASME and ANS standards
  - NUREG/CR-2300
  - Fault Tree Handbook
- Widely used in nuclear, chemical, and aerospace industries
  - Used in EPRI Bolted Storage Cask analysis

Steps of Event Sequence Categorization



Step 1: Scope

- Identify internal and external events:
  - Include events that could affect public and/or worker populations
  - Internal events consider:
    - Equipment failures
    - Human errors
    - ITS to non-ITS interactions
  - External events consider:
    - Seismic events
    - Internal floods and fires
    - External floods, fires/explosions, windstorms, construction hazards, aircraft crashes, and other

Step 1: Scope (cont.)

- Place event sequences into one of three categories:
  - Category 1 (expected during the preclosure period)
  - Category 2 (not expected during the preclosure period)
  - Beyond Category 2 (less than 1E-04 over the preclosure period)
- All waste handling buildings, BOP, subsurface, aging and intra-site operations within the GROA

Step 2: Initiating Event Identification & Screening

- Identification of a comprehensive list of initiating events is an important step in event sequence development:
  - This is different from Nuclear Power Plants which follow years of established precedence dating from 1974 (WASH-1400)
- PCSA is using Master Logic Diagrams and Process Flow Diagrams:
  - Widely used method for nuclear, chemical process, and aerospace risk assessments
  - Top event of each Master Logic Diagram (MLD) relates to exposure of individuals to radiation
  - Performed for each of the four waste handling buildings, subsurface areas, aging and intra-site transportation, and BOP
  - MLDs use two different types of hazard analyses: checklist hazard analysis and HAZOP

Step 2: Initiating Event Identification & Screening (cont.)

- **Initiating Event Screening:**
  - Screen if occurrence probability is less than 1 in 10,000 over pre-closure period
- **External hazard screening:**
  - Start with comprehensive list of facility hazards
  - Screen hazard event if it cannot cause damage to waste
  - Screen based on probability as above

Step 3: Develop Event Sequences

- PCSA is using event sequence diagrams and event trees for both internal events and seismic events
- Models the system, equipment and facility response to an initiating event
- Event sequence modeling includes:
  - Functional Dependence: One component or system depends on another to supply vital functions
  - Environmental Dependence: System functionality depends on maintaining environment within designed or qualified limits
  - Spatial Dependence: One system or component fails by virtue of close proximity to another
  - Human Dependence: A system, component or function fails because of human activity involved with the process
  - Common Cause Failures: parametric representation of failure dependence of redundancy that is not explicitly modeled
  - Failures owing to: normal random events, seismic, fire, flood, and other external hazards

Step 4: Develop Models to Support Sequences

- Models are developed to calculate initiating event frequencies and pivotal event probabilities used in event sequences
- Methods used to calculate initiating event frequencies and pivotal event probabilities include:
  - Fault trees:
    - Systems and equipment
    - Operation
  - Human Reliability Analysis (HRA)
  - Data Analysis:
    - Equipment and component
    - Seismic fragilities

Step 4 (cont.): Data Development

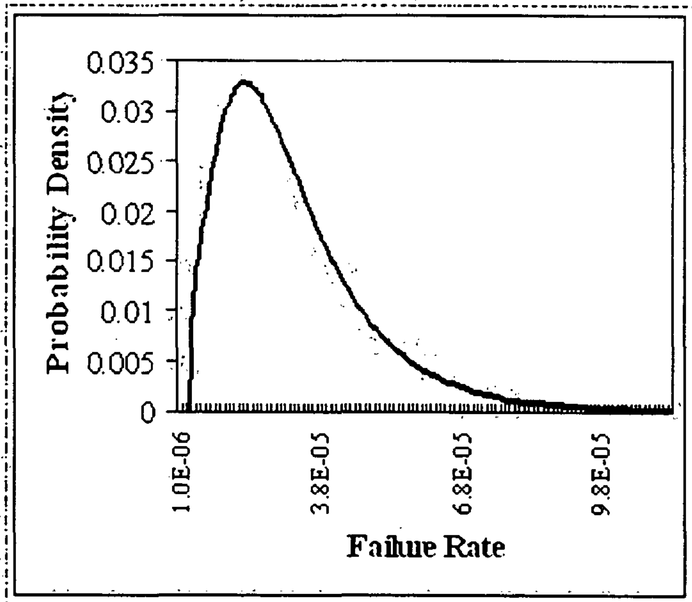
- **Data Development for Frequency and Probability Calculations:**
  - The YMP is a first of a kind:
    - Compilations of historical records
    - Specific facility records
    - Structural analyses
  - Bayesian framework is applicable to YMP and will allow easy updating of information when test and operational data become available:
    - The utilization of Bayes' Theorem in nuclear reactor PRAs emerged to deal with high reliability equipment that has few failures within an industry that was developing its operating experience
    - Bayes analysis provides for appropriate aggregating of different data sources (e.g. NPRD, EPRD, NUCLARR, IEEE-STD-500, NSWC 98-LE1)

Step 5: Integrate Model and Quantification with Uncertainties

- **PCSA uses SAPHIRE ver. 7.27 as the integration and quantification tool:**
  - **Internal and seismic events**
  - **Internal fire event sequences**
  - **Internal flood event sequences**
- **The location dependence feature of SAPHIRE will be useful for fire, seismic and flood events, integration and quantification**

Step 5 (cont.): Uncertainty Analysis

Component	Failure Mode	Prior Lower Estimate	Prior Upper Estimate	Qualification Test Data	Updated Mean	Sources for Prior
		Failures/hour		Failure in x hours		
Isolation Valve, solenoid	Fails open	1E-06	6E-05	0 in 500	1E-05	AIAA, NPRD, FMD
Control Valve, motor	Fails open	1E-05	2E-04	0 in 500	6E-05	AIAA, NPRD, FMD
Controller	Spurious signal	4E-07	5E-06	0 in 500	2E-06	AIAA, NPRD, IEEE-Std-500, FMD
Controller	Fails to respond	4E-06	2E-05	0 in 500	1E-05	AIAA, NPRD, IEEE-Std-500, FMD
RPM transducer/transmitter	Fails low	5E-05	5E-03	1 in 500	1E-03	APU Proof of Concept Study
Wire	Open circuit	1E-07	1E-06	0 in 5000	4E-07	NPRD (based on connector failures)



Probability of each basic event x becomes uncertain

$$P(x) = 1 - e^{-\lambda(x)t}$$

Step 6: Categorization & Documentation

- **Interpretation of Results:**
  - If the mean value (including uncertainties) of an event sequence < 10^{-4} over the preclosure period then no further analysis is performed
  - If the mean value (including uncertainties) of an event sequence is in Category 1 or 2, then dose is analyzed to evaluate compliance
  - Category 1 event sequences are aggregated such that total weighted dose to workers and public is obtained
  - If an event sequence in Category 1 or 2 indicates a potential fuel reconfiguration or presence of moderator, then reactivity calculation is performed
- **Documentation:**
  - Master logic diagrams with supporting hazard analyses, event sequence and reliability analysis, accident sequence categorization, Q-list, Nuclear Safety Design Bases, and Operational Requirements

PCSA Team Organization

- Each task performed by a two or more person team:
  - Teams for WHF, IHF, CRCF, RF, subsurface structures and equipment, intra-site transport and balance of plant, data development, and internal/external hazards
  - Team members check each other
  - Teams review each other's work
- Each document undergoes an interdisciplinary review from engineering and operations organizations

Summary

- Standard methods supported by well-known references and widely used:
  - Includes comprehensive identification of event sequences
  - Reliability analysis includes passive and active components and uncertainties
  - Quantification with uncertainties
  - Categorization using mean values of event sequence probability distributions
- Internal and External Events Included
- Multiple levels of reviews:
  - Assures accurate and traceable documentation with sufficient justification for analysis inputs

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Human Reliability Analysis (HRA) for the Preclosure Safety Analysis (PCSA)

Presented to:
NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:
Michael V. Frank
Supervisor - Hazards, Event Categorization and Reliability Analysis
Bechtel SAIC Company, LLC

November 8, 2006
Las Vegas, NV

Outline

- What is Human Reliability Analysis?
- Contrast between YMP and Nuclear Power Plants with respect to HRA
- Example Identification of Human Failure Events (HFEs)
- Approach to the HRA for YMP PCSA
- Evaluation of HRA Methods for YMP PCSA
- Summary

What HRA is and is Not

- **HRA is:**
  - a field of study that attempts to characterize and quantify human reliability, often within the context of a probabilistic risk assessment
  - a means for representing systematic effects
- **HRA is not:**
  - an approach to predict actions or error rates of an individual or one control room crew during a specific accident scenario
  - geared toward day-to-day human variability
  - a subset of psychology (although psychological theories may be used to derive models)
  - the same as human factors (HF), however, HF is part of the context within which human errors may be identified and quantified.

Contrast of Nuclear Power Plant (NPP) and YMP

NPP	YMP
• Central control in control room	• Decentralized (local), hands on control
• Most important human actions are in response to accidents	• Most important human actions are initiating events
• Post-accident response is important and occurs in minutes to hours. Short time response important to model in HRA.	• Post-accident response evolves more slowly (hours to days). Short time response not important to model.
• Multiple standby systems are susceptible to pre-initiator failures.	• Standby systems do not play major role in YMP safeguards, therefore few opportunities for pre-initiator failures.

Contrast of NPP and YMP (cont.)

NPP

- Auxiliary operators sent by central control room operators to where needed in the plant
- Most actions are remote
- Reliance on instrumentation /gauges as operators' "eyes"
- High complexity of systems, interactions, and phenomena. Actions may be skill, rule or knowledge based
- Many in operation for decades; HRA may include walk-downs and consultation with operators

YMP

- Local control reduces time to respond
- Most actions are local
- Most actions are local, either hands on or televised. Less reliance on man-machine interface.
- Relatively simple process with simple actions. Actions are largely skill based.
- First of a kind; HRA performed for license application, therefore walk-downs and consultation with operators are limited

Examples of Potential HFEs Associated with the Wet Handling Facility

Entry & Receipt Area → X-fer → **Preparation Area** → X-fer → **In-Pool Operations** → X-fer → **Decontamination & Preparation Area** → X-fer → **Exit to Other Facilities**

Entry & Receipt Area

- **HFE** (*initiator*) - Failure to properly set parking brake and chock the truck trailer may lead to cask drop
- **HFE** (*initiator*) - Drop of cask from a crane

Preparation Area

- **HFE** (pre-*initiator, latent failure*) - Failure to maintain steam trap or improper steam trap installation may cause water condensation in vent line that would prevent steam release during water addition to the cask prior to transfer to pool. This may cause cask over-pressurization and lead to radiological release into the building
- **HFE** (*initiator*) - Too much cold water quenching may lead to fuel assembly damage

In-Pool Operations

- **HFE** (*initiator*) - Drop of canister or fuel assembly owing to crane or transfer machine operator error may lead to fuel geometry reconfiguration

Decontamination & Preparation Area

- **HFE** (*initiator*) - Failure to lower water level inside the TAD coupled with failure to catch that mistake may result in poor welding of TAD lid
- **HFE** (*initiator*) - Failure to dry TAD interior prior to inerting may lead to water hydrolysis

Exit to Other Facilities

- **HFE** (*initiator*) - Failure to properly transfer loaded STC/TAD onto siteTransporter (ST) may result in STC fall over

STC: Site Transfer Canister

TAD: Transport, Aging, and Disposal Canister

HRA Approach

- HRA is integrated within the overall PCSA
- Comprised of both qualitative and quantitative aspects
- Specific methods selected for human error probability (HEP) quantification should be adaptable to the YMP
- The bases for the identification of human actions, human failure events, and human error probabilities include engineering design, operation, maintenance philosophy, training, procedures and management philosophy. In turn, HRA feedback influence the evolution of these bases

HRA Approach

- **Develop event sequences and fault trees:**
  - Identification of initiating events includes human errors (Master Logic Diagrams supported by Hazard and Operability studies)
  - Includes review of layout drawings, system and operation documents, equipment design documents
  - Development of event sequence diagrams and fault trees includes a systematic Identification of HFEs
- **Quantitative screening using screening value:**
  - e.g., 0.1 for HEPs
  - Screen out event sequences below the Category 2 lower threshold

HRA Approach

- **Quantify human reliability for Category 1 or 2 event sequences that involve human actions:**
  - The main objective is to identify areas of design or operation for which safeguards should be added
- **Thorough documentation:**
  - Traceable from HFE identification through quantification noting assumptions and judgments

Evaluation

of

HRA Methods

HRA Method Types

- **Procedure focused:**
  - For example, Technique for Human Error Rate Prediction (THERP)
- **Time-response focused:**
  - For example, Time Reliability Curve (TRC) , Human Cognitive Reliability (HCR), Success Likelihood Index Method (SLIM)
- **Context and/or cognition driven:**
  - For example, Cognitive Reliability Error Analysis Method (CREAM), A Technique for Human Event Analysis (ATHEANA), The Human Error Assessment and Reduction Technique (HEART)
- **Simplified:**
  - For example, Accident Sequence Evaluation Program (ASEP), Standard Plant Analysis Risk-HRA (SPAR-H)

Procedure Focused

- Example: THERP
- Concentrates on failures that occur during step-by-step tasks
- Of limited use for YMP because important actions are not procedure-driven:
  - Many operations are skill-based and/or semi-automated (e.g., crane operation, trolley operation, canister transfer machine operation, emplacement vehicle (TEV) operation)

Time - Response

- Examples: HCR, TRC, SLIM
- Based on NPP control room experience and studies
- Most YMP actions do not occur in control rooms and the time to respond is at least hours and may be days:
  - Time response models are correlated with much shorter duration simulator exercises. These models do not apply to YMP.

Context - Driven

- Examples: ATHEANA, CREAM, HEART
- Independent of facility and process
- More broadly applicable to various industries, tasks, situations (including YMP) because they allow context-specific Performance Shaping Factors (PSFs) to be considered
- Can support the variety of contexts, individual performance factors (e.g., via PSFs) and human factor approaches

Simplified

- Examples: ASEP, SPAR-H
- Methods pre-suppose NPP actions and define PSFs based on past NPP PRAs
- Too limited for application beyond the NPP environment:
  - No ability to investigate context, individual and human factors that are beyond NPP experience
  - HEPs calibrated to other NPP methods

A Perspective

- **Much of HRA thought and method development has concentrated on representation of NPP control room and associated Auxiliary Operator activities at full power operation**
- **Nearly all methods have been developed to be used within a PRA**
- **Exceptions:**
  - **CREAM, ATHEANA: useful in discovering human failure modes without being associated with PRA event sequences or fault trees**
  - **HEART: created as tool to assess human performance in facilities owing to a variety of factors: contextual, individual performance, cognitive, human factors**

Three Mile Island Insights

- Subsequent to accident, TMI sequence of events was analyzed in detail with respect to human performance
- Multiple insights into procedures and operator behavior emerged:
  - Operators do not act like pieces of equipment who might take a wrong action
  - Operator actions far more complex and occur within context of:
    - their previous background and expectations
    - current influences
    - perceived plant conditions

A Technique for Human Event Analysis (ATHEANA)

- First major attempt at introducing contextual elements that could trigger cognitive errors
- Attempted to reconcile observed human performance with existing theories of human cognition and reliability models
- May be applied with an understanding of system and facility design, plant conditions and PSFs
- Identifies important human-system interactions, human failure modes and causes
- Results in recommendations for improving human performance based on these causes
- Current version of ATHEANA uses HEART as HEP quantification tool

Cognitive Reliability and Error Analysis Method (CREAM)

- Based on Hollnagel's Context and Control mode of human cognition
- Retrospective and Prospective search to/from human failure modes:
  - Interactive tabulations permit trace-back to basic human error modes (retrospective) and trace forward from error modes to higher level problems (prospective)
- HEPs developed using CREAM are considered probability of a cognitive error rather than probability of a human action error:
  - Two quantification techniques: basic and extended

CREAM - BASIC METHOD

- Basic Process:
  - Identify scenario to be analyzed (e.g., from event sequence)
  - Perform task analysis of human actions and interactions with the hardware
  - Rate common performance conditions (using explicit guidance) to obtain a Cognitive Performance Condition (CPC) score
  - Determine probable cognitive control mode for CPC score: scrambled, opportunistic, tactical, strategic
  - Cognitive failure frequency ranges provided for each cognitive control mode

CREAM - Extended Method

- Task analysis
- Assign cognitive activity to task
- Cognitive demands and functional failures are associated with cognitive activity
- Cognitive failure probability uses information from HEART

Human Error Assessment and Reduction Technique (HEART) (Williams, 1988)

- Provides procedure using error producing conditions to adjust nominal HEPs, all of which are based on the author's extensive database
- Generic tasks define basic HEP, together with a range:
  - For example, task 'D' (defined as a "fairly simple task performed rapidly or given scant attention") is associated with basic HEP of 0.09 (0.06 to 0.13)
- Application requires considerable thought to define error producing conditions that apply to human actions during an event sequence
- Pre-application of qualitative aspect of ATHEANA or CREAM is helpful to sort out context and identify appropriate HFEs and PSFs

Emphasis of YMP HRA

- YMP PCSA emphasizes creating a safe design over performing detailed human reliability analyses:
  - Analyses detailed enough to identify areas to introduce safeguards
- Different methods may give widely varying results; Different practitioners using the same method may yield widely varying results; Large uncertainties in HEPs of each individual method due to sparse data, approximate cognition theories, dependent PSFs not accurately modeled, etc.
- Provide technical justification to selection of methods

Selection of Quantification Method

- **Method depends on the human failure event:**
  - **For example:**
    - **HEART may be sufficient for skill based HFEs that rely on man-machine interface**
    - **CREAM may be sufficient for highly cognitive (knowledge-based) tasks such as decision-making**
- **New quantification method for ATHEANA is under development**
- **The criteria to select a quantification method for YMP is still evolving**

Summary

- **YMP is now being designed:**
  - Risk-informed regulation has changed usual design process toward close concurrent work by safety analysts, design engineers, analysis engineers, and operations engineers
- **Emphasis is on safe design and operation:**
  - Relative HEP values are therefore more important than absolute values
  - High relative values indicate areas of design improvement, operational constraints, and/or need for procedural safety controls
  - HRA insights will be fed back to the design and operation processes
- **Operational constraints and procedural safety controls may become technical specifications**

Summary (cont.)

- Differences between YMP and conventional NPPs constrain the selection and application of HRA methods
- PCSA team includes human reliability analysts with background in various HRA approaches, event sequence development and hazard analyses to assure integration of HRA within the overall PCSA
- Quantitative method applied will depend on the HFE being analyzed
- Document analysis to level that permits traceability and facilitates review

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

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Reliability Methodology

Presented to:
NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:
Michael V. Frank
Supervisor - Hazards, Event Categorization and Reliability Analysis
Bechtel SAIC Company, LLC

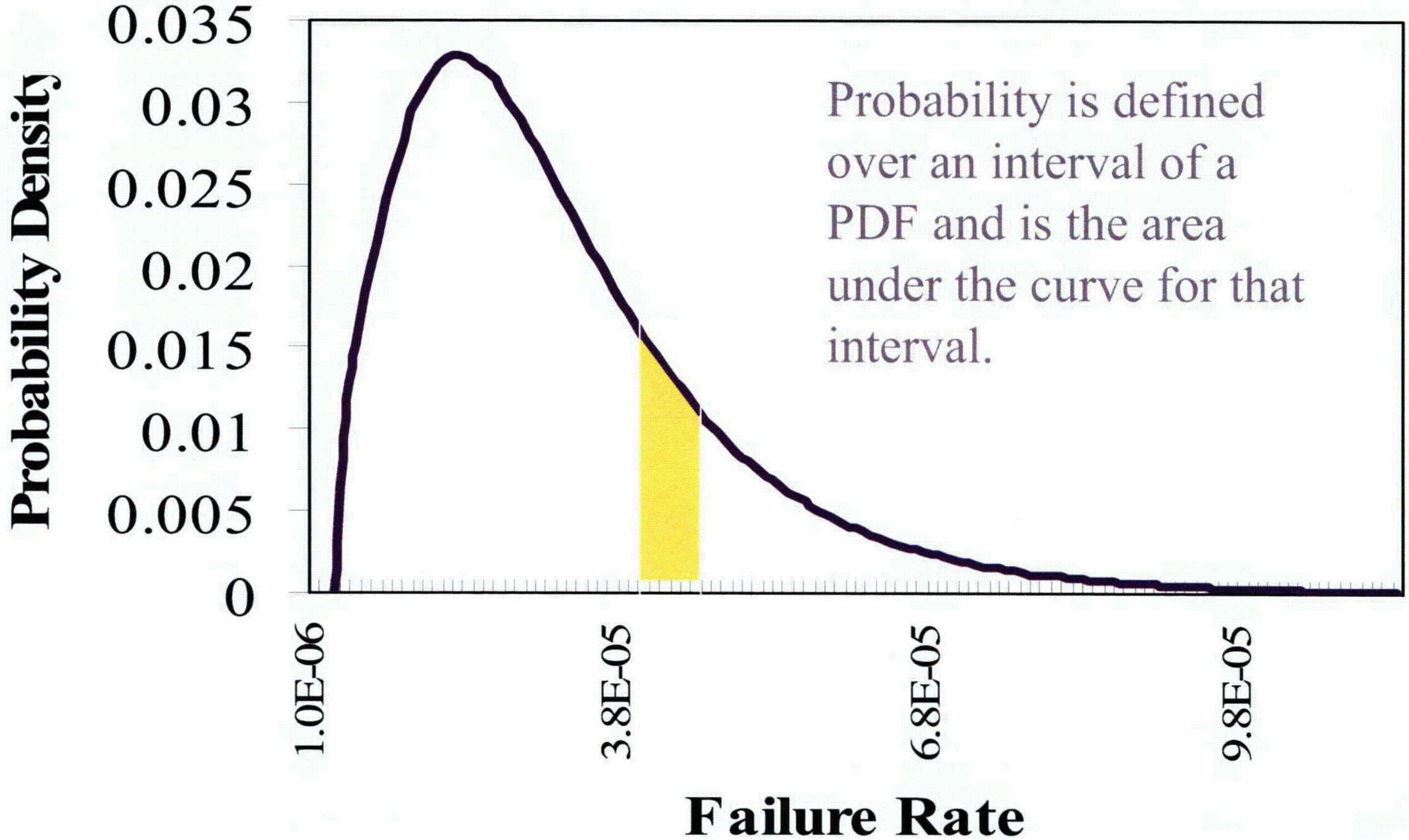
November 8, 2006
Las Vegas, NV

Outline

- Fundamental concepts
  - Uncertainties
  - Mean values and comparison to a requirement
- Reliability Methods and Data
  - Active equipment reliability & data sources
    - Type of data and Data sources
    - Dependent and common cause failures
  - Bayesian analysis
  - Passive equipment reliability
  - Process of expert judgment
  - Software reliability
  - Seismic Fragility Analysis
- Summary

Fundamental Concepts

Continuous Probability Distribution



Mean Value

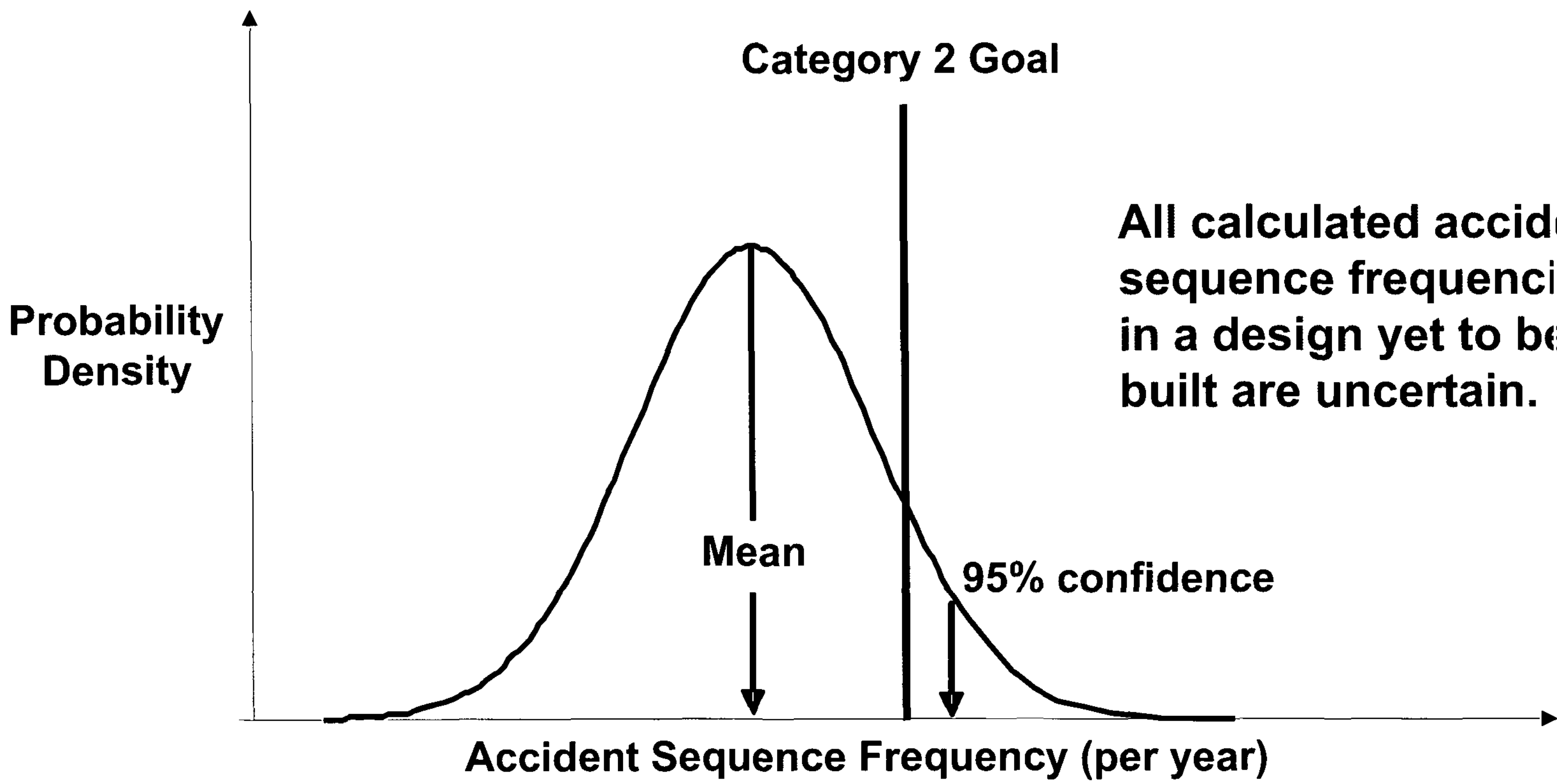
- A mean value is the first moment of a probability distribution
- In a reliability analysis, it is usually the result of the uncertainty analysis, rarely an input

$$\overline{\lambda} = \int \lambda p(\lambda) d\lambda$$

- By the rules of probability, the mean of a PDF of an uncertain event is the probability that must be assigned to the occurrence of that event. (Howard, 1988*)

* R. Howard, "Uncertainty about Probability: A Decision Analysis Perspective", *Risk Analysis*, Vol. 8, No.1, 1988.

Probability Distribution Overlapping Category Goal



All calculated accident sequence frequencies in a design yet to be built are uncertain.

Reliability Methods and Data

Equipment Failure Types

- For equipment failures owing to motion or change of state (termed active for this presentation)
- For equipment failures owing to a demanded load exceeding a load capacity (termed passive for this presentation)

Active Equipment Reliability Data Sources

Types of Data

- Empirical data collected from industrial reliability/monitoring/testing studies for specific components and structures, or empirical modeling (i.e., computer-based design and simulation) conducted by equipment vendors or generally available in the industry. This data may be from nuclear and non-nuclear sources.
- Generic reliability databases
- Accepted engineering practices and expert judgments

Generic Field Data Sources

- **From Reliability Analysis Center:**
  - **Non-electronic Part Reliability Data:**
    - Mechanical and electrical components and equipment
    - No uncertainties but variability may be deduced
  - **Electronic Part Reliability Data:**
    - Electronic part
    - No uncertainties but variability may be deduced
  - **Failure Mode/Mechanism Distributions:**
    - Lists failure modes or mechanisms of mechanical, electrical, and some electronic parts

Generic Field Data Sources (Cont'd)

- OREDA:
  - Severe offshore environment, will evaluate for YMP application
- NPRDS:
  - Failure rates, MTBF, MTTR for nuclear power plant equipment
- CCPS:
  - Electrical, mechanical, and piping systems (included uncertainties) for chemical process industry equipment

Reliability Model/Testing Based Data Sources

- Naval Surface Weapons Laboratory LEI-98-LE1
  - Mechanical Parts
    - Mechanical equipment may be analytically constructed from parts
- MIL-HDBK-217F
  - Electronic Parts
- These are just some examples of generic data sources. Review and selection of the specific data sources to be used is currently underway.

Dependent Failures

- Functional
  - Shared system, structure, components
  - Treated in event trees and fault trees
- Environmental
  - Environmental stressors (e.g., temperature, pressure, etc.)
  - Treated at basic event level
  - Handbook data sources include environmental dependence
- Human
  - Man-machine interface (e.g., common training, common maintenance personnel)
  - Treated using human failure event identification and human reliability analysis
- Spatial
  - Common location
  - Treated in event trees and fault trees and in fire, flood and earthquake analyses

Common Cause Failures

- Multiple failures of similar equipment for which a specific mechanism cannot be determined a-priori
- Parametric methods such as Alpha Factor and Multiple Greek Letter:
  - The parameters in these models are derived from nuclear power plant incident records
  - Evidence that these factors are also applicable in space systems
- Alpha factor method used because it includes a mathematically consistent treatment of epistemic uncertainty

Bayes Methods to be used in the PCSA

Bayes' Theorem for YMP

- The YMP is a first of a kind. It will have no operating experience during the period of development of the LA:
  - Reliability data will have to be developed largely from sources other than permanent geologic repositories
- A Bayesian approach is applicable for YMP:
  - This will allow easy updating of information when test and operational data become available
  - Bayes analysis provides for appropriate aggregating of different data sources
  - Best method to quantify uncertainties

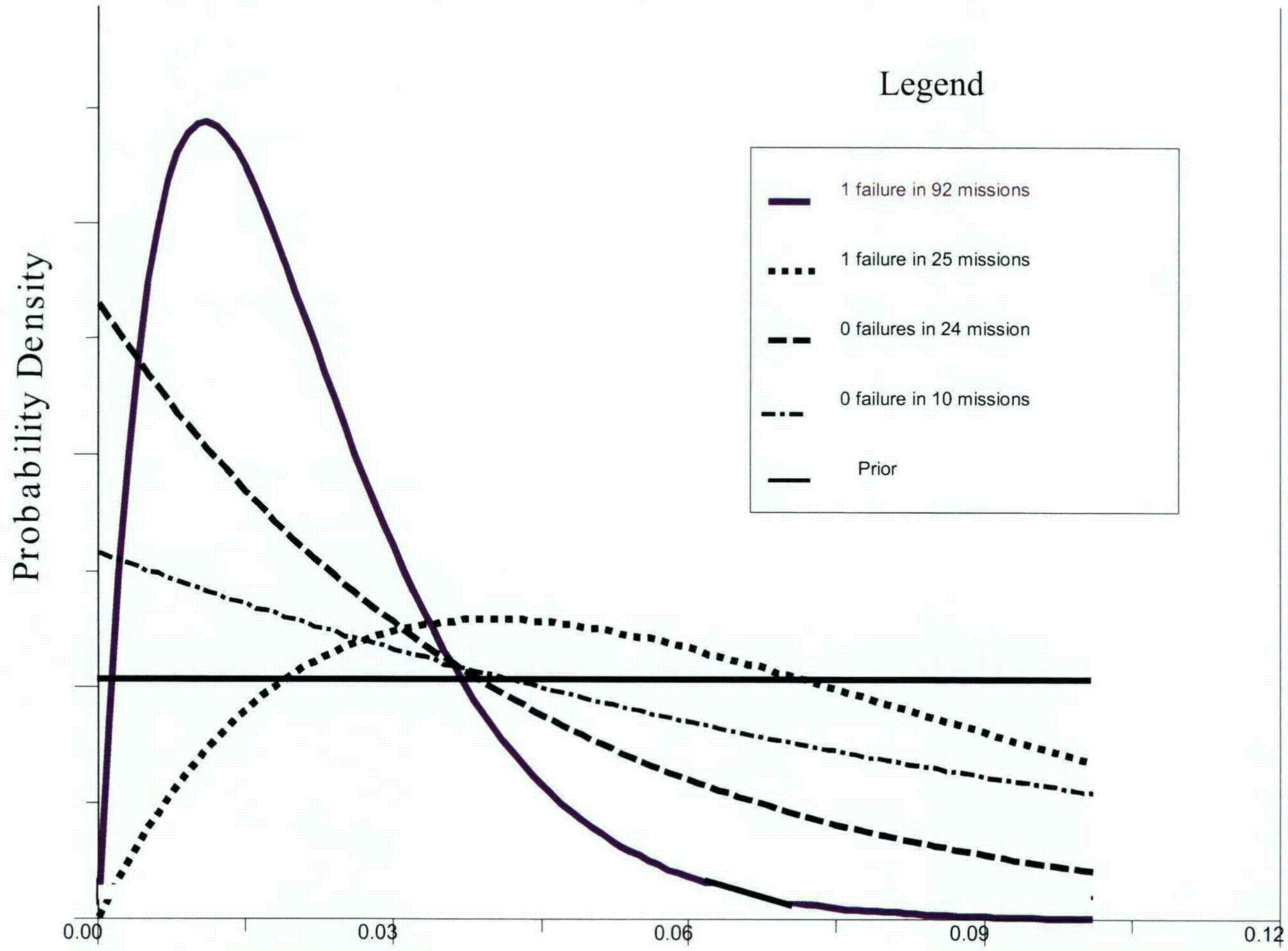
Bayes' Theorem is Fundamental to Estimating State-of-Knowledge Uncertainty

Bayes' Theorem has been proven to be a coherent method of mathematically expressing a decrease in uncertainty gained by an increase in knowledge (for example, knowledge about failure frequency). It has been particularly useful in estimating our knowledge about the frequency of rare events.

Example: Methods for "Prior" Distributions

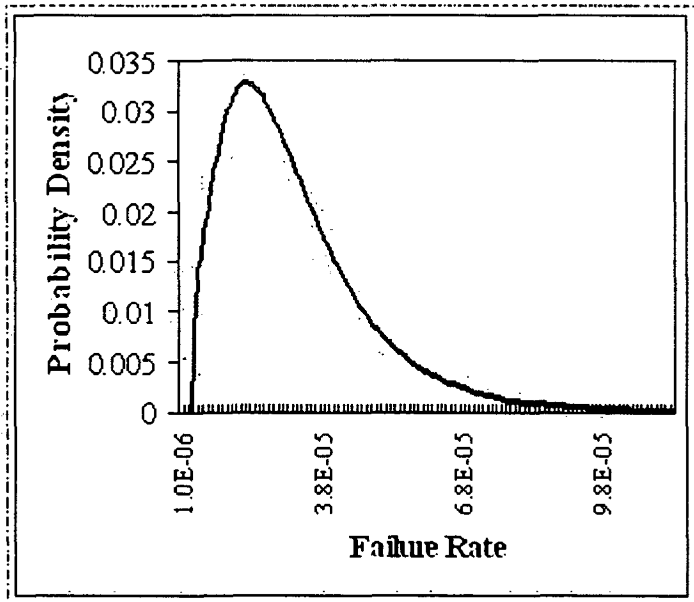
- Concentrate on Prior Distributions until test and operational information is available:
  - Conjugate
  - Maximum Entropy
  - Judgment
  - Empirical Bayes
  - Regression Analysis
  - Weighted Posterior
  - Two Step Bayes
- Specific method will depend on data available for each failure rate to be developed

Effect of Experience on Uncertainties Using Bayes' Theorem



Concept of Epistemic Uncertainty in Failure Rate

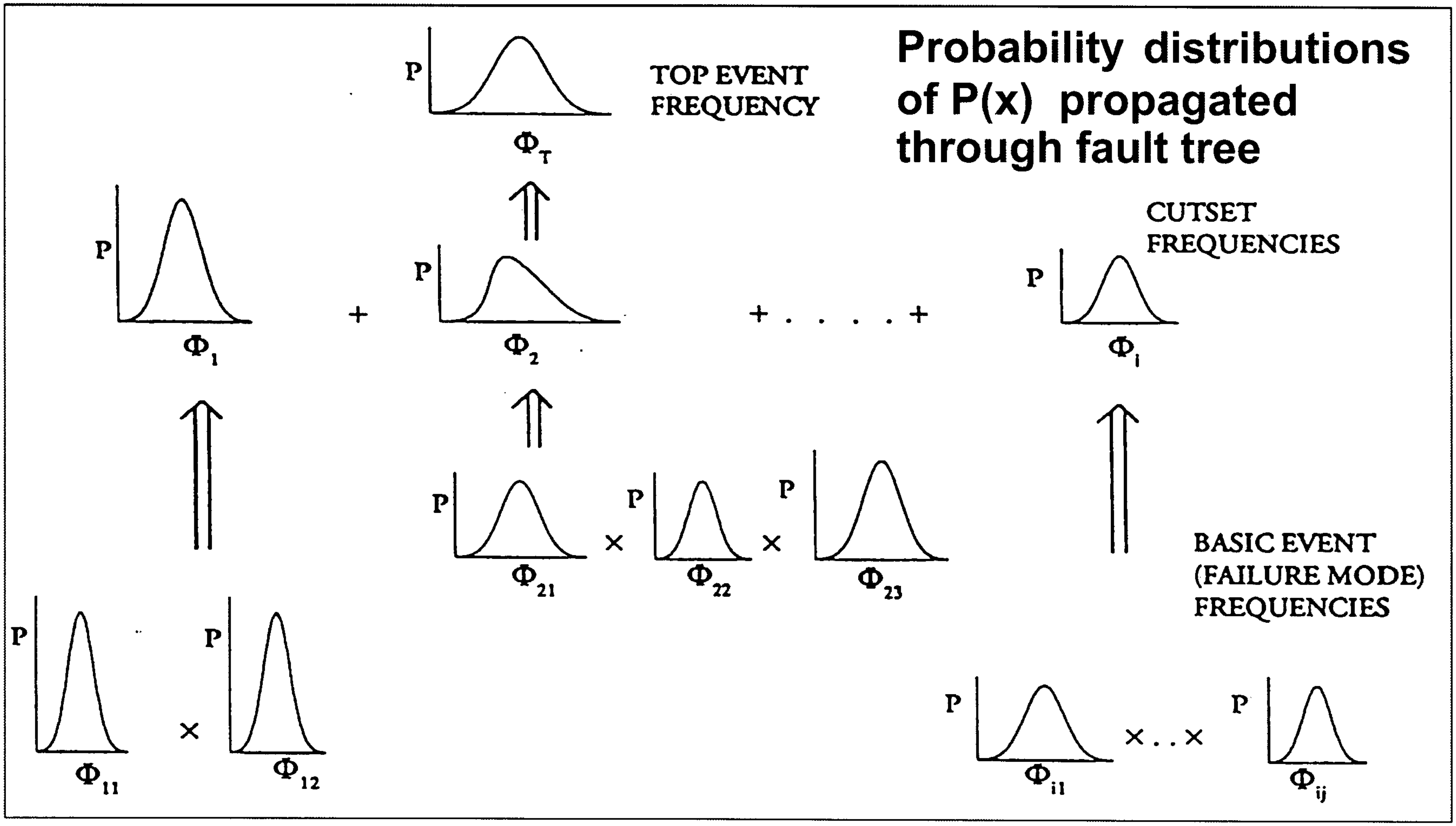
Component	Failure Mode	Prior Lower Estimate	Prior Upper Estimate	Qualification Test Data	Updated Mean	Sources for Prior
		Failures/hour		Failure in x hours		
Isolation Valve, solenoid	Fails open	1E-06	6E-05	0 in 500	1E-05	AIAA, NPRD, FMD
Control Valve, motor	Fails open	1E-05	2E-04	0 in 500	6E-05	AIAA, NPRD, FMD
Controller	Spurious signal	4E-07	5E-06	0 in 500	2E-06	AIAA, NPRD, IEEE-Std-500, FMD
Controller	Fails to respon	4E-06	2E-05	0 in 500	1E-05	AIAA, NPRD, IEEE-Std-500, FMD
RPM transducer/transmitter	Fails low	5E-05	5E-03	1 in 500	1E-03	APU Proof of Concept Study
Wire	Open circuit	1E-07	1E-06	0 in 5000	4E-07	NPRD (based on connector failures)



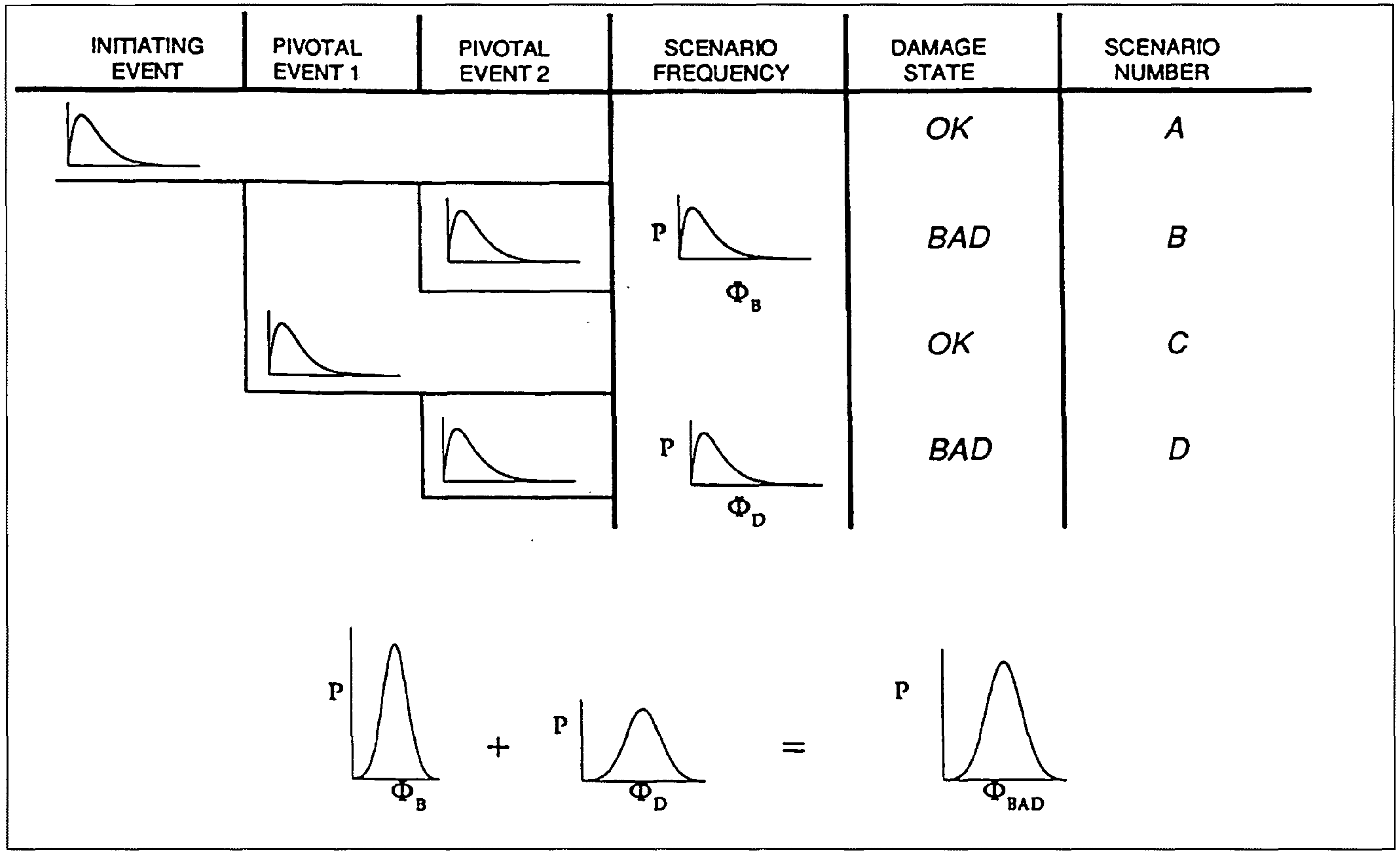
Probability of each basic event x becomes uncertain

$$P(x) = 1 - e^{-\lambda(x)t}$$

Uncertainty Propagation Fault Tree



Concept of Propagation of Uncertainties through Scenarios



Mean Value and Uncertainties

- **Propagation of uncertainties through fault trees and event sequences to obtain the mean value, which is based on the probability distribution function (PDF)**

Passive Equipment Reliability

Multiple Methods

- All are equivalent, depends on output of structural or thermal analysis:
  - Stress strength
  - Factor of Safety (FS)
  - Simplified FS based on code allowable
  - Implications from the use of codes and standards
  - Draft ISG-2 and variations
  - Monte Carlo
- Define load limit:
  - Yield
  - Code Allowable
  - Code Ultimate
  - Tested Ultimate
  - Material property (minimum, best estimate)

Large Number of Components Requires Screening

- **Screening is performed because it is impractical and unnecessary to use rigorous methods for each component:**
  - **Event sequences that are below Category 2 will be screened out if used with conservative estimates of passive failure probability**
  - **For example, if P = 0.01 for impacts within code allowable limits and P=1 for impacts beyond the limits**

Judgment with Codes and Standards

- **Design of passive equipment to established codes and standards implies a high degree of reliability (for stresses within code allowable limits)**
- **Engineers knowledgeable with the codes and standards, the function of the equipment and structural analyses of the equipment combine their knowledge to develop failure probabilities with uncertainties**
  - **Technical basis and justification of judgment is essential**

Software Reliability

Software Reliability

- **Use local hardware as safeguards against software driven erroneous commands:**
  - For example, Cranes and canister transfer machine have redundant motion limit switches; large shield doors are electrically interlocked with each other; trolleys and other package ground transfer devices have inherent electro-mechanical speed limiters.
- **The CCCF has monitoring, inhibit, and permissive functions:**
  - Permissives allow operation. They do not cause operation.
- **Controller failure rates will be developed for functional failure modes at the subsystem level. As such, both hardware and software failures are included.**

Seismic Fragility Analysis

Seismic Analysis Approach

- **YMP has elected to use Hybrid Method:**
  - Based less on judgment than traditional "factor of safety" method
  - The median seismic capacity ($C_{50\%}$, or the peak ground acceleration (PGA) at which there is a 50% probability of unacceptable performance) will be calculated directly from the HCLPF value (95% confidence of a 5% probability of failure) and the composite logarithmic standard deviation β_c:
    - HCLPF can be taken to be the seismic capacity at the 1% non-exceedance probability of unacceptable performance ($C_{1\%}$)
    - β_c considers both aleatory and epistemic uncertainty

Hybrid Method

- HCLPF will be directly calculated using the conservative deterministic failure margin (CDFM) approach ($C_{1\%} \approx C_{CDFM}$)
  - Structural design engineers can understand and perform CDFM from standard design code calculations
    - Simple parameter alterations to obtain the HCLPF
- Therefore, $C_{50\%}$ is calculated as follows:
  - $C_{50\%} = C_{CDFM} \times e^{2.326\,\beta_c}$
    - Small range of composite lognormal standard deviations (typical range of 0.3 to 0.5 for structures and for equipment mounted on the ground or at low elevations)

Seismic Screening Approach

- **Divide into component categories:**
  - For example, cranes, trolleys, pipes, electrical cabinets, switchgear, diesel generators, compressors, pumps, pipes, etc.
- **NPP generic fragilities:**
  - YMP to be built to the same civil/structural codes and standards
  - Screen out event sequences or component categories
- **Use CDFM to calculate HCLPF for "weakest" example in category**
- **Obtain composite fragility curve for categories in remaining event sequences**

Summary

- **Mean value used for event sequence categorization:**
  - Obtaining the mean values of an event sequence requires developing and propagating the probability distributions throughout the analysis
- **Active equipment reliability data & data sources**
- **Bayes' Theorem is the basis for development of active component probability distributions**
- **Passive failure methods compatible with draft ISG-2**
- **Seismic fragilities developed using 1% non-exceedance frequency, median, and composite uncertainty**
- **Software reliability is part of the process control equipment overall reliability**

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Technical Specifications

Presented to:

NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:

William Spezialetti
Office of the Chief Engineer – Engineering Design
Office of Civilian Radioactive Waste Management
U.S. Department of Energy

November 8, 2006
Las Vegas, NV

General Requirements for Yucca Mountain License Specifications

- **10 CFR Part 63.21(c)(18)**
  - The Safety Analysis Report must include probable subjects of license specifications
- **10 CFR Part 63.42(a)**
  - License specifications may be included as conditions of the license
- **10 CFR Part 63.43(a)**
  - License includes license conditions derived from the analyses and evaluations included in the application

Similar Requirements for Nuclear Power Plant Technical Specifications

- 10 CFR Part 50.34 (a)(ii)(E)(5)
  - Identification of probable subjects of technical specifications

Content of Technical Specifications Patterned after Nuclear Power Plants and Independent Spent Fuel Storage Installations

- **Content**
  - Definitions
  - Approved contents (adapted from NUREG-1745*)
  - Limiting conditions for operation
  - Surveillance requirements
  - Design features
  - Administrative controls

* Standard Format and Contents for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance

Technical Specifications Based on 10 CFR Part 63 and Yucca Mountain Review Plan

- 10 CFR Part 63.43 specifies six categories:

1. Restrictions as to the physical and chemical form and radioisotopic content of radioactive waste.
   [YMP proposed implementation: condition of license and administrative controls]
2. Restrictions as to size, shape, and materials and methods of construction of radioactive waste packaging.
   [YMP proposed implementation: design features and LCOs/surveillances]
3. Restrictions as to the amount of waste permitted per unit volume of storage space, considering the physical characteristics of both the waste and the host rock.
   [YMP proposed implementation: design features and administrative controls]

10 CFR Part 63.43 Categories (Cont.)

4. Requirements relating to test, calibration, or inspection, to assure that the foregoing restrictions are observed. [YMP proposed implementation: surveillance requirements and administrative controls]

5. Controls to be applied to restrict access and to avoid disturbance to the site and to areas outside the site where conditions may affect compliance with 10 CFR Parts 63.111 and 63.113
[YMP proposed implementation: physical security plan and administrative controls]

10 CFR Part 63.43 Categories (Cont.)

6. Administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure that activities at the facility are conducted in a safe manner and in conformity with the other license specifications.
   [YMP proposed implementation: administrative controls and quality assurance program]

Yucca Mountain Review Plan Adds a Seventh Category

- Section 2.5.10.1 (1)(e)
  - Characteristics of drifts, drip shields, backfill, ventilation systems, and other structures, systems, and components
    [YMP proposed implementation: design features, administrative controls, and LCOs/surveillances]

Format of Technical Specifications

1. Use and application:
   - Three proposed modes:
     - Operating (movement of waste forms allowed)
     - Standby (waste forms present, but no movement allowed)
     - Shutdown (no waste forms present)
2. Approved contents (replaces safety limits and limiting safety system settings)

Format of Technical Specifications (cont.)

3. Limiting Conditions for Operation (LCO) and Surveillance Requirements (SR)
4. Design features
5. Administrative controls

Technical Specification Bases

- Licensee controlled document
- Format and content patterned after nuclear plant standard technical specifications
- Administrative controls in technical specifications
- Technical Specifications link to Preclosure Safety Analyses (PCSA)

PCSA Risk Insights Reflected in Technical Specifications

- Limiting Conditions for Operation:
  - Selection of PCSA significant SSCs (structures/systems/components)
  - Limiting Conditions for Operation would specify that Important-to-Safety (ITS) SSCs be operable
  - Completion times for specified actions
- Surveillance intervals / frequencies
- Reliability monitoring

Reliability Monitoring is a Part of Technical Specifications

- Proposed program to be included in the Administrative Controls section of Technical Specifications
- Reliability monitoring and control program would contain details on how reliability assumptions for the component/system would be verified to support operability determination
- Maintenance program and surveillances assure component/system are maintained at assumed reliability

Reliability Monitoring and Control Program Contains Details for Determining Operability

- Monitor component/system run times, demands, and failures to determine reliability
- If needed reliability is not met, corrective action would be taken
- Increased attention initiated as appropriate
- Reanalysis where appropriate

Example Technical Specifications

- 3.3.1 Canister Receipt and Closure Facility (CRCF) Cranes
- 3.3.8 CRCF Surface Nuclear HVAC

Example Technical Specification (cont.)

3.3 CANISTER RECEIPT and CLOSURE FACILITY (CRCF)

3.3.1 CRCF Cranes

LCO 3.3.1 Each Gantry and Crane in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: OPERATING

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Gantry and Crane.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Gantry or Crane inoperable.	A.1 Suspend waste handling operations with the affected unit.	Immediately.
	AND	
	A.2 Place the load on affected unit in a safe and stable condition.	[1] hour
B. Required Action and associated Completion Time not met.	B.1 Be in STANDBY for the affected unit.	[ ] days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform FUNCTIONAL TEST of the control system.	In accordance with the Reliability Monitoring Program
SR 3.3.1.2	Perform load test.	In accordance with the Reliability Monitoring Program

Example Technical Specification (cont.)

CRCF Cranes
3.3.1

Table 3.3.1-1 (page 1 of 1)
CRCF Gantry and Crane Ratings

Gantry or Crane	Rating
Cask Handling Crane	200 tons
Waste Package and Canister Handling Crane	100 tons

Example Technical Specification (cont.)

3.3 CANISTER RECEIPT and CLOSURE FACILITY (CRCF)

3.3.8 CRCF Surface Nuclear HVAC (SNHVAC)

LCO 3.3.8 Each SNHVAC Train shall be OPERABLE

APPLICABILITY: During movement of WASTE FORMS

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One SNHVAC Train inoperable.	A.1	Restore SNHVAC train to OPERABLE status.	[7] days
B	Two [or more] SNHVAC Trains inoperable.	B.1	Suspend movement of WASTE FORMS.	Immediately
		AND		
		B.2	Place the WASTE FORMS in a safe and stable condition.	[1] hour
		AND		
		B.3	Restore at least one train to OPERABLE status.	[2] days
C.	Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in STANDBY for CRCF	[1] day

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.8.1	Operate each SNHVAC train for ≥ 15 minutes	[ ] days
SR 3.3.8.2	Perform required SNHVAC testing in accordance with Ventilation Filter Testing Program (VFTP)	In accordance with the VFTP.
SR 3.3.8.3	Perform a FUNCTIONAL TEST of each SNHVAC train.	[ ] months
SR 3.3.8.4	Verify each SNHVAC train can maintain a negative pressure of > [0.125] inches water guage	[ ] months on a STAGGERED TEST BASIS

Submittal of Technical Specifications

- With LA submittal (Section 5.10):
  - List of probable subjects for Technical Specifications [10 CFR Part 63.21 (c)(18)]
  - Basis for selection
- Subsequent to LA submittal:
  - Representative Technical Specifications for common ITS SSCs:
    - Cranes, HVAC, emergency power, etc.

Tech Specs and TAD CSNF Loading

- Loading of TADs licensed under 10 CFR Part 63 will meet the bounding conditions specified in the Approved Contents section of the Technical Specifications
- TADs will be discussed at a subsequent NRC/DOE Technical Exchange, tentatively scheduled for early 2007

Summary

- Technical Specifications will meet 10 CFR Part 63 license specification requirements
- Technical Specification format and content patterned after nuclear power plant and independent spent fuel installation (ISFSI) standard technical specifications
- Risk insights and reliability assumptions reflected in Technical Specifications

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

www.ocrwm.doe.gov

Systematic Approach to Training

Presented to:
NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

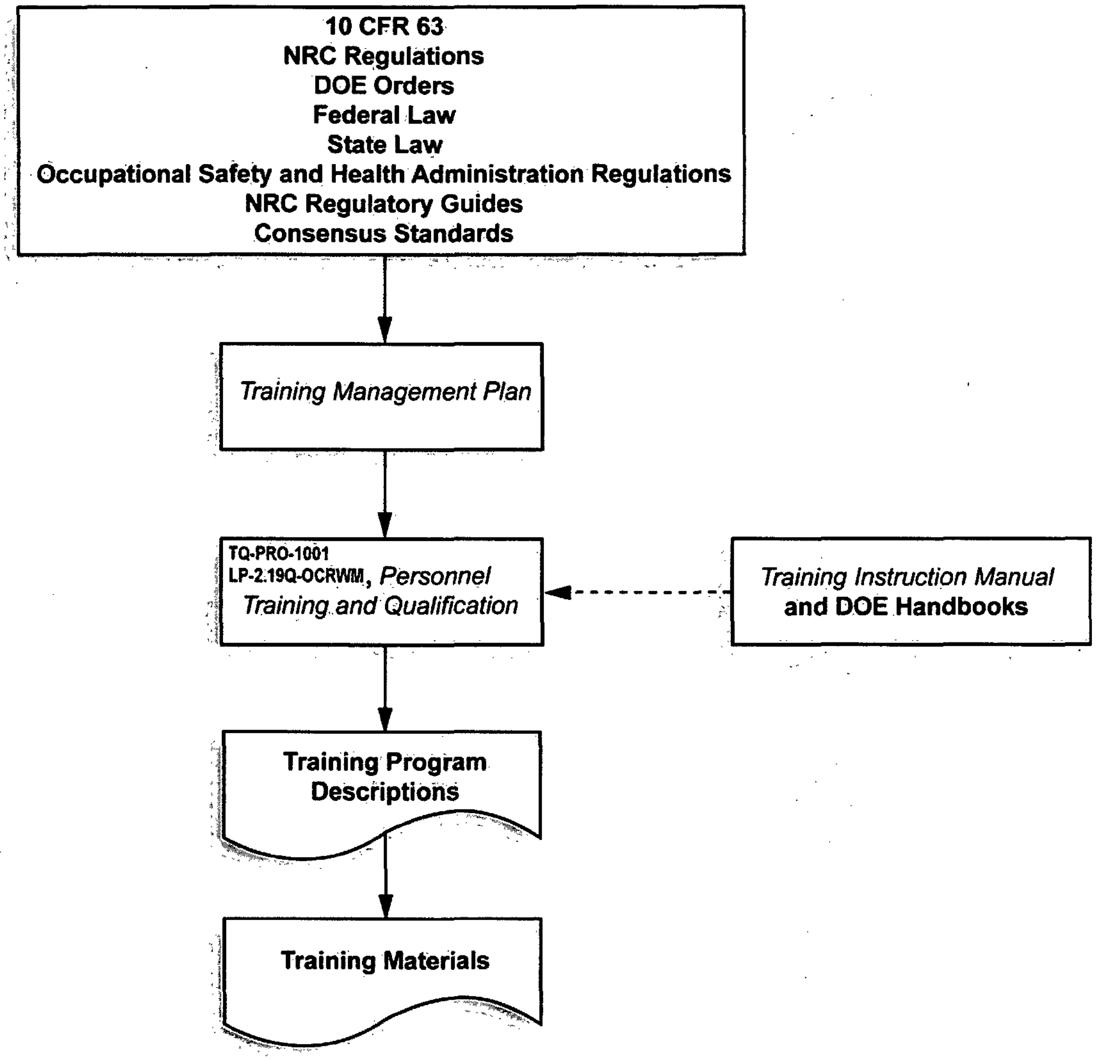
Presented by:
Jerry McMahon
OCRWM Training – Senior Policy Advisor
Office of Civilian Radioactive Waste Management
Management and Technical Support (MTS)

November 8, 2006
Las Vegas, NV

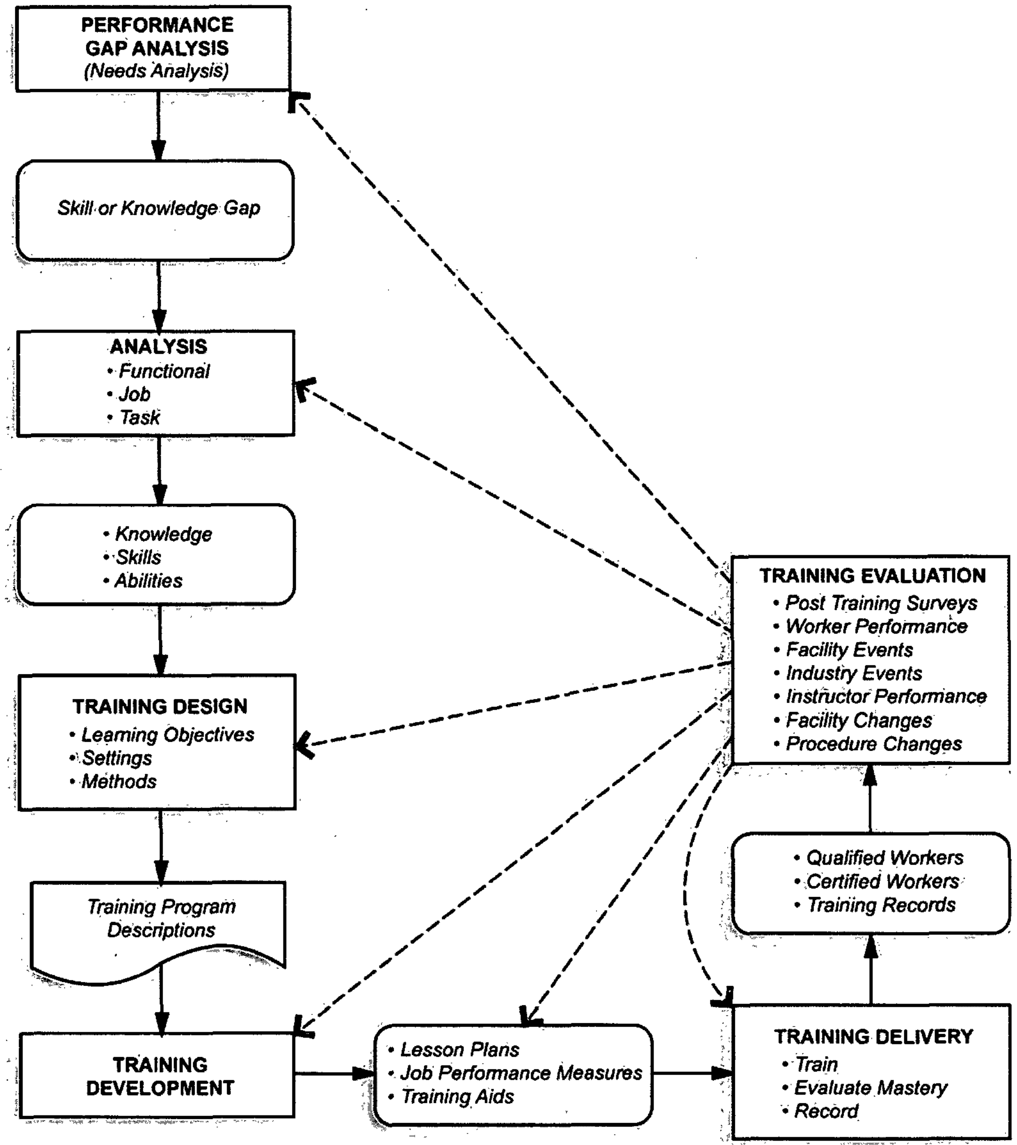
OCRWM Director's Four Objectives

- Submit a high-quality, docketable license application to the Nuclear Regulatory Commission by June 30, 2008
- Design, staff and train the OCRWM organization so that it has the skills and culture to design, license, and manage the construction and operation of the Yucca Mountain Project with safety, quality, and cost effectiveness
- Address the federal government's mounting liability associated with unmet contractual obligations to move spent fuel from nuclear plant sites
- Develop and implement a comprehensive national spent fuel transportation plan that accommodates state, local and tribal concerns to the extent possible

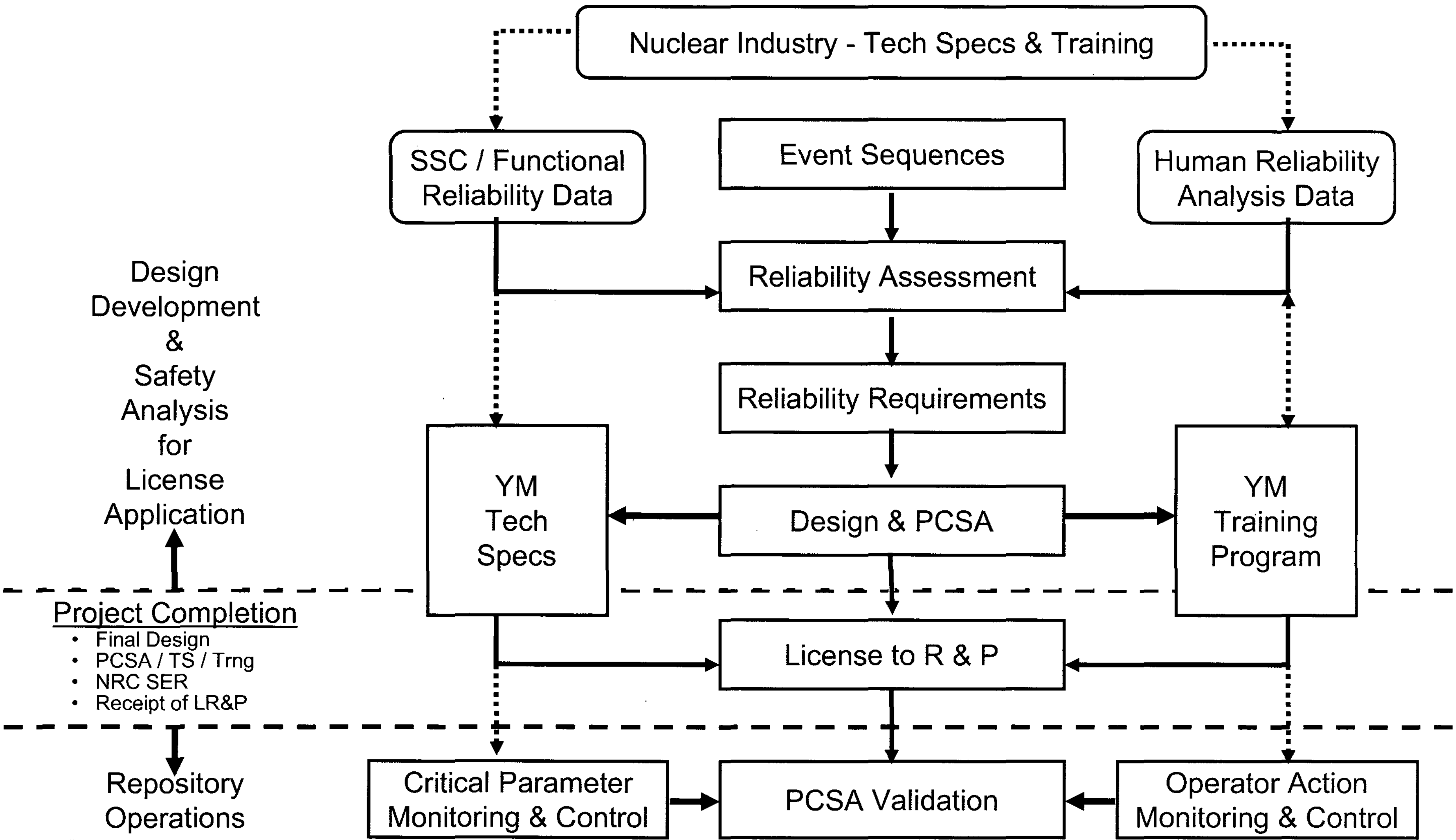
Training Management Plan Document Relationship



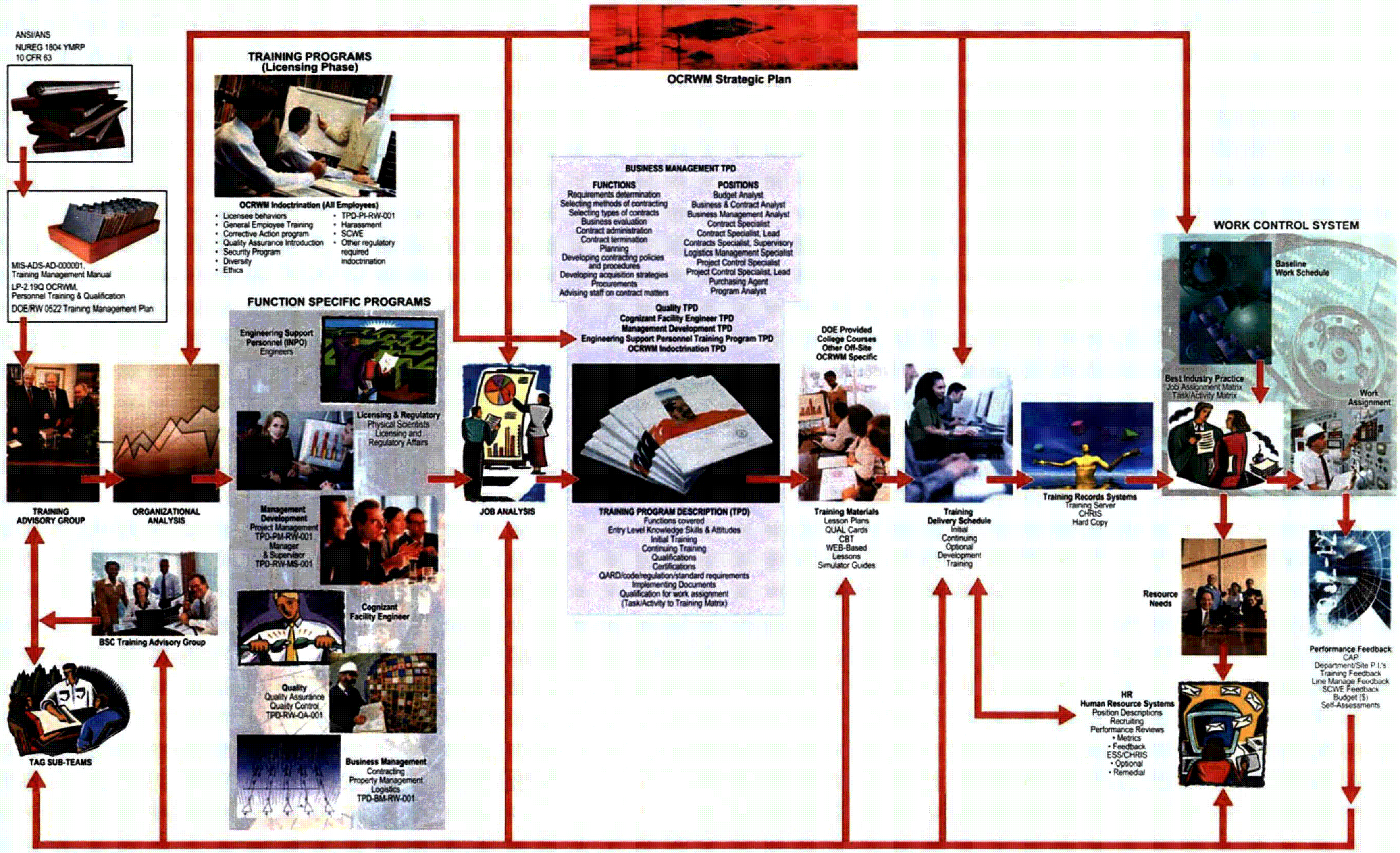
Systematic Training Process



Training Program Interface with PCSA



OCRWM Training and Qualification Process



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NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

November 9, 2006
Las Vegas, NV

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS: 1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY, 3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

November 7 and 8, 2006
8:00 AM – 5:00 PM (PT)
11:00 AM – 8:00 PM (ET)

November 9, 2006
8:00 AM – 12:00 PM (PT)
11:00 AM – 3:00 PM (ET)

U. S. Nuclear Regulatory Commission Hearing Center
Pacific Enterprise Plaza, Building 1
3250 Pepper Lane
Las Vegas, Nevada 89120

And via Teleconference to:

U. S. Nuclear Regulatory Commission
Two White Flint North, Room T 7A-1
11545 Rockville Pike
Rockville, MD

Center for Nuclear Waste Regulatory Analyses
Conference Room A-237, Bldg. 189
6220 Culebra Road
San Antonio, TX

INTERESTED PARTIES MAY PARTICIPATE VIA TELECON BY CALLING
1-800-638-8081, Passcode 8755# or 301-231-5539, Passcode 8755#

Tuesday November 7, 2006 (Aircraft Hazards and Source Terms and Consequence Methodology)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:30 AM	NRC Key Messages on Aircraft Hazards Assessment	NRC
9:00 AM	Background and Overview of Updated Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
9:30 AM	Changes in Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
10:00 AM	Break	All
10:15 AM	Aircraft Hazards Sensitivity Analysis	DOE/BSC (K. Ashley)
11:00 AM	Response to 13 NRC Issues (NRC letter of Aug. 2, 2005)	DOE/BSC (K. Ashley)
11:30 AM	Lunch	All
1:00 PM	NRC Key Messages on Source Terms and Consequence Methodology	NRC
1:30 PM	Radioactive Source Terms and Release Methodology	DOE/BSC (D. Dexheimer)
2:30 PM	Break	All
2:45 PM	Consequence and Analysis Methodology	DOE/BSC (D. Dexheimer)
3:30 PM	Uncertainty and Sensitivity Analysis	DOE/BSC (D. Dexheimer)
3:50 PM	Documents to be Revised	DOE/BSC (D. Dexheimer)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

AGENDA

**NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS:
1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY,
3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING**

November 7 and 8, 2006	**November 9, 2006**
8:00 AM – 5:00 PM (PT)	**8:00 AM – 12:00 PM (PT)**
11:00 AM – 8:00 PM (ET)	**11:00 AM – 3:00 PM (ET)**

**U. S. Nuclear Regulatory Commission Hearing Center
Pacific Enterprise Plaza, Building 1
3250 Pepper Lane
Las Vegas, Nevada 89120**

And via Teleconference to:

U. S. Nuclear Regulatory Commission	**Center for Nuclear Waste Regulatory Analyses**
Two White Flint North, Room T 7A-1	**Conference Room A-237, Bldg. 189**
11545 Rockville Pike	**6220 Culebra Road**
Rockville, MD	**San Antonio, TX**

**INTERESTED PARTIES MAY PARTICIPATE VIA TELECON BY CALLING
1-800-638-8081, Passcode 8755# or 301-231-5539, Passcode 8755#**

Wednesday November 8, 2006 (Reliability Assessment, Technical Specifications, and Training)

8:00 AM	**Introductions**	**NRC/DOE**
8:10 AM	**Opening Remarks**	**NRC/DOE (J. Williams)**
8:15 AM	**NRC Key Messages:** **- Reliability Assessment**	**NRC**
9:00 AM	**Reliability Assessment Overview**	**DOE/BSC (M. Frank)**
9:45 AM	**Break**	**All**
10:00 AM	**Human Reliability Assessment**	**DOE/BSC (M. Frank)**
11:30 AM	**Lunch**	**All**
1:00 PM	**Reliability Assessment for Structures, Systems, and Components**	**DOE/BSC (M. Frank)**
2:15 PM	**Break**	**All**
2:30 PM	**NRC Key Messages:** **- Technical Specifications** **- Systematic Approach to Training**	**NRC**
3:00 PM	**DOE Plans for Development of Technical Specifications**	**DOE (W. Spezialetti)**
3:30 PM	**DOE Plans for Systematic Approach to Training**	**DOE/MTS (J. McMahon)**
4:00 PM	**Public Comments**	**All**
4:15 PM	**Break/Caucus**	**All**
4:30 PM	**Summary Discussion/Closing Remarks**	**NRC/DOE**
5:00 PM	**Adjourn**	**All**

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS: 1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY, 3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

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9:45 AM	Break	All
10:00 AM	License Application Status and Requirements Mapping	DOE (R. Warther)
10:10 AM	License Application Requirements Mapping	DOE/BSC (G. Ashley)
11:00 AM	Public Comments	All
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11:30 AM	Summary Discussion/Closing Remarks	NRC/DOE
12:00 PM	Adjourn	All

License Application (LA) Status and Requirements Mapping

Presented to:
NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

Presented by:
Robert Warther
License Application Project Director
U.S. Department of Energy

November 9, 2006
Las Vegas, Nevada

LA Background

- Five principal organizations
- 71 sections
- Thousands of figures and tables
- Nearly 7,000 pages

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- Certify LSN by: Dec. 21, 2007
- LA Submittal to NRC by: June 30, 2008

LA Project Key Tools

- CD-1 Design
- Integrated schedule
- Prevent changes to schedule and design
- LA Project risk management and reduction:
  - Scope, cost, schedule
  - Technical risk
- LA Management Plan
- Monthly reports

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- Ensures completeness of LA
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- Provides an aid to reviewers:
  - BSC
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Office of Civilian Radioactive Waste Management

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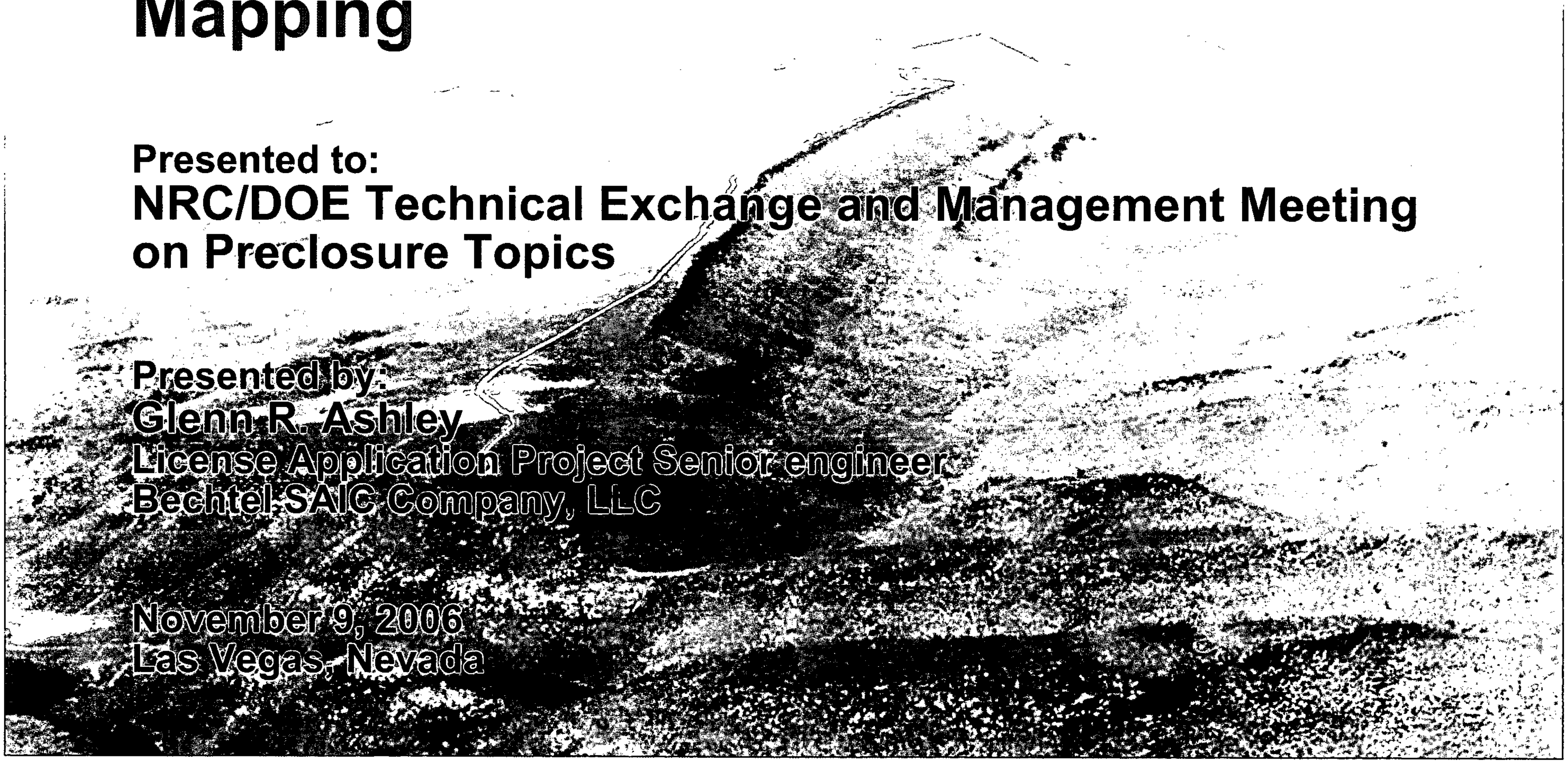
License Application (LA) Requirements Mapping



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Presented by:
Glenn R. Ashley
License Application Project Senior engineer
Bechtel SAIC Company, LLC

November 9, 2006
Las Vegas, Nevada

Requirements Mapping Ensures Completeness of LA

- Requirements and guidance mapped to LA sections and subsections:
  - 10 CFR 63.21
  - Yucca Mountain Review Plan (YMRP) Acceptance Criteria (NUREG-1804)
  - Other 10 CFR requirements as applicable
- Mapping verified as part of LA development

LA Requirements Mapping

10 CFR 63.21 and YMRP Mapping to LA Sections and CDR Groups

The relations illustrated here are summarized from our database mapping of requirements from 10 CFR 63.21 and the YMRP into the LA sections.

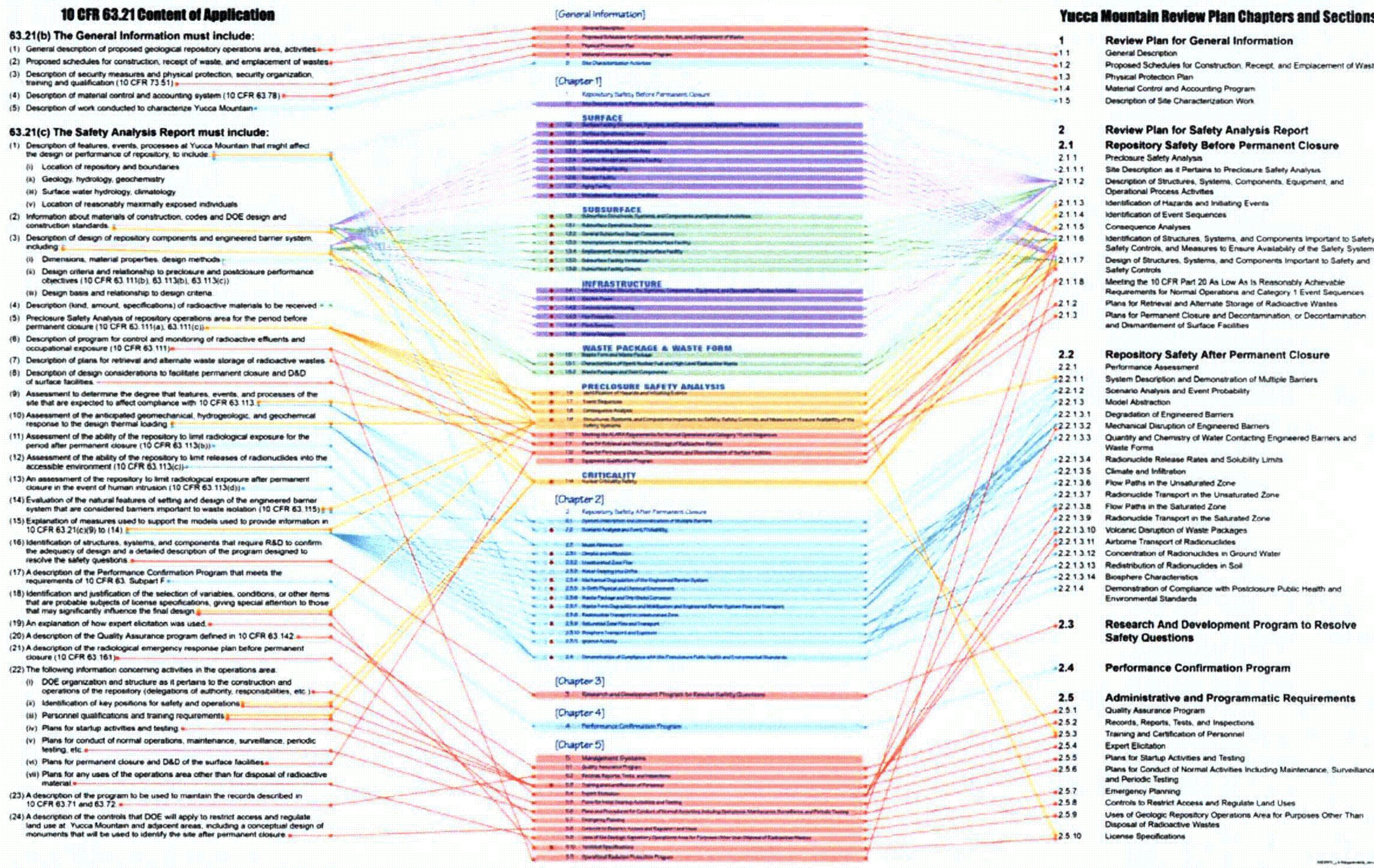
The ~110 relations shown between the YMRP outline of 50 topical areas of review and the LA sections actually represent ~3,000 discrete relationships between 503 YMRP acceptance criteria and subcriteria and the LA sections and subsections.

10 CFR 63 requirements are also traced at greater depth in our requirements traceability crosswalk database.

LEGEND: CDR GROUPS

- Surface Design Group
- Subsurface Design & Waste Package Group
- Preclosure Safety Analysis Group
- Programmatic Group
- Postclosure Group
- * Sections with subject matter to be addressed in planned NRC technical exchanges

Planning Date July 12, 2006



LA Design Sections Closely Aligned With YMRP

YMRP Section	Subject	SAR Section
2.1.1.1	Site Description as it Pertains to Preclosure Safety Analysis	1.1
2.1.1.2	Description of SSCs	1.2, 1.3, 1.4, 1.5
2.1.1.3	Identification of Hazards and Initiating Events	1.6
2.1.1.4	Identification of Event Sequences	1.7
2.1.1.5	Consequence Analyses	1.8
2.1.1.6	Identification of SSCs Important to Safety	1.9
2.1.1.7	Design of SSCs Important to Safety	1.2, 1.3, 1.4, 1.5
2.1.1.8	Meeting the 10 CFR 20 ALARA	1.10
2.1.2	Plans for Retrieval and Alternate Storage of Radioactive Wastes	1.11
2.1.3	Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities	1.12

LA Addresses Consistent Set of Requirements for ITS SSCs

- System description [2.1.1.2.3 - AC1, AC2]
- Operational processes and procedures [2.1.1.2.3 - AC6]
- Safety category classification [2.1.1.6.3 - AC1]
- Procedural safety controls to prevent event sequences or mitigate their effects [2.1.1.6.3 - AC2]

AC - Acceptance Criteria (from NUREG-1804)
ITS - Important to Safety
SSC - Structures, Systems and Components

LA Addresses Consistent Set of Requirements for ITS SSCs (cont.)

- Design bases and design criteria [2.1.1.7.3.1 - AC1]
- Design methodologies [2.1.1.7.3.2 - AC1]
- Consistency of materials with design methodologies [2.1.1.7.2.3 I - AC2]
- Design codes and standards [2.1.1.7.3.3 I - AC1]
- Design load combinations [2.1.1.7.3.3 I - AC3]

10 CFR 63.21 Mapping Typically at the LA Section Level

10 CFR 63.21(c) - Content of application:

(5) The Safety Analysis Report (SAR) must include:

A preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with § 63.111(a), as required by § 63.111(c). For the purposes of this analysis, it is assumed that operations at the geologic repository operations area will be carried out at the maximum capacity and rate of receipt of radioactive waste stated in the application.

LA Part-Sect.	Section Title
SAR-1.6	Identification of Hazards and Initiating Events
SAR-1.7	Event Sequences
SAR-1.8	Consequence Analyses
SAR-1.9	Structures, Systems, and Components Important-to-Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems
SAR-1.14	Nuclear Criticality Safety

YMRP Mapping Typically at the Individual Acceptance Criteria Level

- SAR-1.7 Event Sequences
- YMRP (NUREG-1804) Section 2.1.1.4.3:
  - AC 1
    "Adequate Technical Basis and Justification are Provided for the Methodology Used and Assumptions Made to Identify Preclosure Safety Analysis Event Sequences"
  - AC 2
    "Categories 1 and 2 Event Sequences are Adequately Identified"

Other 10 CFR Requirements Are Mapped As Applicable

SAR-1.7 Event Sequences:

10 CFR 63.112 Requirements for preclosure safety analysis of the geologic repository operations area:

(b) An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences;

(c) Data pertaining to the Yucca Mountain site, and the surrounding region to the extent necessary, used to identify naturally occurring and human-induced hazards at the geologic repository operations area;

(d) The technical basis for either inclusion or exclusion of specific, naturally occurring and human-induced hazards in the safety analysis

Requirements Mapping Information to Aid Reviewers and Validation

- Requirements cross-referenced in LA:
  - Table at front of each major section
  - References under subsection titles
- Other cross-reference reports are being considered to aid reviewers

Requirements Mapping Tables Provide Cross-Reference to 10 CFR 63 and YMRP

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
1.7.1	Technical Basis and Assumptions for Methods to Identify Event Sequences	**63.21(c)(5)** **63.112(b), (c), and (d)**	**Section 2.1.1.4.3: Acceptance Criterion 1**
1.7.2	Categorization of Internal Hazard Event Sequences	**63.21(c)(5)** **63.112(b), (c), and (d)**	**Section 2.1.1.4.3: Acceptance Criterion 1**
1.7.3	Categorization of External Hazard Event Sequences	**63.21(c)(5)** **63.112(b), (c), and (d)**	**Section 2.1.1.4.3: Acceptance Criterion 1**
1.7.4	Categorization Results	**63.21(c)(5)** **63.112(b), (c), and (d)**	**Section 2.1.1.4.3: Acceptance Criterion 2**

Summary

Requirements mapping:

- Ensures completeness of LA
- Aligns LA with requirements and guidance documents
- Provides an aid to reviewers
  - BSC
  - DOE
  - SNL
  - NRC

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NRC/DOE Technical Exchange and Management Meeting on Preclosure Topics

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS: 1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY, 3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

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Tuesday November 7, 2006 (Aircraft Hazards and Source Terms and Consequence Methodology)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:30 AM	NRC Key Messages on Aircraft Hazards Assessment	NRC
9:00 AM	Background and Overview of Updated Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
9:30 AM	Changes in Aircraft Hazards Analysis	DOE/BSC (P. Macheret)
10:00 AM	Break	All
10:15 AM	Aircraft Hazards Sensitivity Analysis	DOE/BSC (K. Ashley)
11:00 AM	Response to 13 NRC Issues (NRC letter of Aug. 2, 2005)	DOE/BSC (K. Ashley)
11:30 AM	Lunch	All
1:00 PM	NRC Key Messages on Source Terms and Consequence Methodology	NRC
1:30 PM	Radioactive Source Terms and Release Methodology	DOE/BSC (D. Dexheimer)
2:30 PM	Break	All
2:45 PM	Consequence and Analysis Methodology	DOE/BSC (D. Dexheimer)
3:30 PM	Uncertainty and Sensitivity Analysis	DOE/BSC (D. Dexheimer)
3:50 PM	Documents to be Revised	DOE/BSC (D. Dexheimer)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS: 1) AIRCRAFT HAZARDS, 2) SOURCE TERMS AND CONSEQUENCE METHODOLOGY, 3) RELIABILITY ASSESSMENT, 4) TECHNICAL SPECIFICATIONS, 5) SYSTEMATIC APPROACH TO TRAINING, 6) PRECLOSURE CRITICALITY, AND 7) LICENSE APPLICATION REQUIREMENTS MAPPING

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Wednesday November 8, 2006 (Reliability Assessment, Technical Specifications, and Training)

8:00 AM	Introductions	NRC/DOE
8:10 AM	Opening Remarks	NRC/DOE (J. Williams)
8:15 AM	NRC Key Messages: - Reliability Assessment	NRC
9:00 AM	Reliability Assessment Overview	DOE/BSC (M. Frank)
9:45 AM	Break	All
10:00 AM	Human Reliability Assessment	DOE/BSC (M. Frank)
11:30 AM	Lunch	All
1:00 PM	Reliability Assessment for Structures, Systems, and Components	DOE/BSC (M. Frank)
2:15 PM	Break	All
2:30 PM	NRC Key Messages: - Technical Specifications - Systematic Approach to Training	NRC
3:00 PM	DOE Plans for Development of Technical Specifications	DOE (W. Spezialetti)
3:30 PM	DOE Plans for Systematic Approach to Training	DOE/MTS (J. McMahon)
4:00 PM	Public Comments	All
4:15 PM	Break/Caucus	All
4:30 PM	Summary Discussion/Closing Remarks	NRC/DOE
5:00 PM	Adjourn	All

AGENDA

NRC/DOE TECHNICAL EXCHANGE ON PRECLOSURE TOPICS:
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License Application (LA) Status and Requirements Mapping

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Presented by:
Robert Warther
License Application Project Director
U.S. Department of Energy

November 9, 2006
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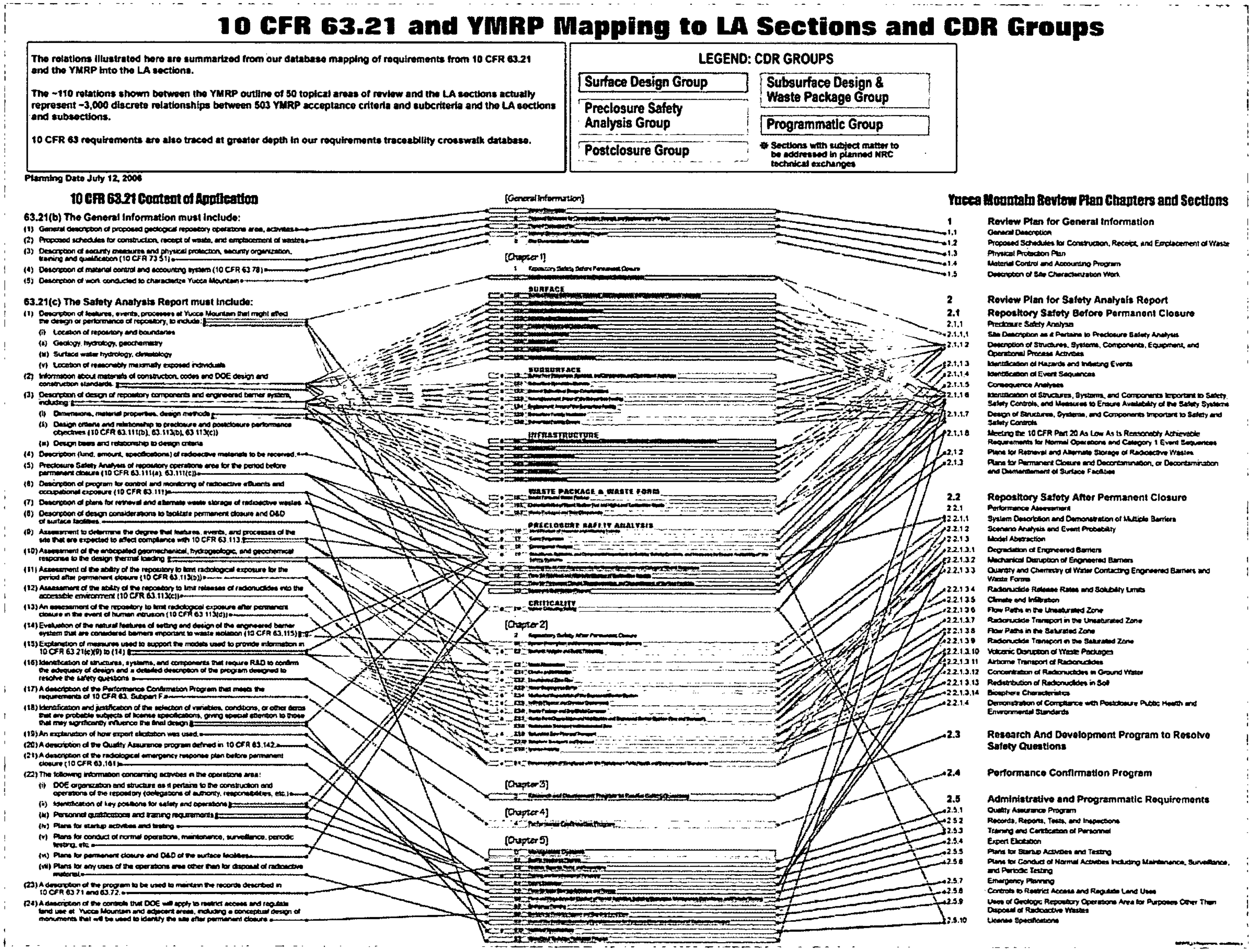

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10 CFR 63.21 Mapping Typically at the LA Section Level

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Requirements Mapping Tables Provide Cross-Reference to 10 CFR 63 and YMRP

SAR Section	Information Category	10 CFR Part 63 Reference	NUREG-1804 Reference
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1.7.4	Categorization Results	**63.21(c)(5)** **63.112(b), (c), and (d)**	**Section 2.1.1.4.3: Acceptance Criterion 2**

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