KEY MESSAGES FOR THE NOVEMBER 7-9, 2006,
PRECLOSURE TECHNICAL EXCHANGE AND MANAGEMENT MEETING

1.0 AIRCRAFT HAZARDS

1.1 The U.S. Department of Energy (DOE) should be prepared to address U.S. Nuclear Regulatory Commission (NRC) comments (e.g., NRC letters dated August 2, 2005 and January 6, 2006) on aircraft crash hazard related issues.

1.2 If DOE intends to rely on flight restrictions in the license application (LA), DOE should describe how such plans would be enforced.

1.3 DOE should address uncertainties associated with assumptions, data and information used, methodologies selected, and analysis techniques. Greater attention to uncertainty analysis would be necessary, if the estimated annual crash frequency is close to the screening threshold frequency between a Category 2 event sequence and a beyond Category 2 event sequence. Uncertainties include, but are not limited to:

- Credit for any actions or inactions by pilots to reduce the probability of aircraft crashes;

- Classification of aircraft mishap types and further quantitative analyses using information derived from the mishap reports.

1.4 If structural robustness is credited in the preclosure safety analysis (PCSA) to eliminate an aircraft crash as a potential initiating event, DOE should provide appropriate analyses. The analyses should include secondary effects arising from an aircraft crash that have the potential to initiate an event sequence at the proposed facilities.

1.5 DOE should provide appropriate technical basis for eliminating from the PCSA, any flight related activities (e.g., cruise missile tests or ordnance jettisoned) in the vicinity of the proposed facilities.

1.6 DOE should provide regulatory and technical bases for the screening threshold frequency for aircraft crash event sequences that can be considered beyond Category 2 event sequences.

Enclosure

2.0 CONSEQUENCE AND SOURCE TERM

2.1 DOE should be prepared to discuss its plans for developing source term and consequence analyses in the PCSA and the radiation protection program.

2.2 The methods and parameters used to estimate direct radiation exposures and release consequences during preclosure should be based on accepted engineering practices and sound health physics principles.

2.3 Confinement and shielding design features should be described with appropriate technical bases to demonstrate their performance in mitigating radiation exposures.

2.4 Source terms assessment should consider the characteristics of the high-level waste processed at the facility, including the number of fuel assemblies, enrichment, burnup, decay time, and the types of failure phenomena during an event sequence that affect release fractions.

2.5 Dose calculations (direct radiation and release models) should consider the characteristics of the shielding and confinement design features, release fractions, deposition factors, weather parameters, and dosimetry for each event sequence.

2.6 The radiation protection program should be commensurate with the scope of activities at the facility, to assure compliance with the dose limits and that maintains exposures as low as reasonably achievable. Representative persons, locations, and occupancy times used in the dose estimates should be consistent with restricted areas, controls, and other protective features identified in the radiation protection program.

3.0 RELIABILITY METHODOLOGY

3.1 DOE's *Summary of Preclosure Safety Analysis Reliability Assessment Methodology*, submitted to the NRC on August 25, 2006, represents a step forward in terms of conducting effective reliability assessment and forms a basis for further discussion and clarification of the reliability assessment methods in performing a PCSA. Staff is currently reviewing the document and has identified the subsequent overarching topics for further discussion at the technical exchange.

3.2 Sufficient technical bases to justify reliability estimates of structures, systems and components (SSC) are necessary in the PCSA. Assumptions or limitations of the reliability estimate methodology, including treatment of uncertainty, should be justified. The DOE document provides several examples where further clarification in the technical bases would be expected due to the broad assumptions and generic input values used.

3.3 When estimating reliability, it is acceptable to use SSC analogues at the highest possible level (typically at the system level). If system-level data is unavailable, then consider analogues at the next level down. Sufficient technical justification is necessary to assess unique SSCs as an aggregate of individual sub-systems and components.

3.4 Uncertainty needs to be addressed in the PCSA. In particular, methods and criteria used to assess frequency and dose in an event sequence should be reasonable and clearly stated in the PCSA. This is especially true when the results are near the categorization limits.

3.5 NRC acknowledges that engineering judgment may play a significant role in estimating reliability and/or uncertainties in the PCSA. When engineering judgment is used, bases for the judgement should be provided.

4.0 HUMAN RELIABILITY ANALYSIS

4.1 The human reliability analysis (HRA) for the Yucca Mountain geologic repository operations area (GROA) should be fully integrated with the PCSA. The general guidance from NUREG-1792 and NUREG-1842 should be considered, along with operating experience from facilities and activities similar to those planned at the GROA, to develop and implement an HRA approach that is suitable for the GROA PCSA. Site- and facility-specific factors should be identified, and addressed appropriately, in the HRA for the Yucca Mountain PCSA.

4.2 Qualitative HRA analysis (the conceptual modeling of human performance) should be given appropriate focus, as it is the basis for all other HRA process steps. These steps should include, for example, the identification of human failure events, and the identification of important factors influencing human performance.

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

5.0 PRECLOSURE LICENSING SPECIFICATIONS

5.1 DOE should be prepared to discuss its plans for developing licensing specifications and their bases.

5.2 License Specifications are an important element of a NRC license because they help assure key safety controls will be maintained in accordance with the bases of the LA.

5.3 Licensing specifications consist of license conditions and technical specifications. Preclosure license specifications may control or limit: (1) the amounts, physical form, and radionuclide content of the high-level waste that is allowed for disposal; (2) key design features of structures, systems and components important to safety; (3) key administrative controls and programs; and (4) key parameters and limiting conditions of operation that may require testing, inspection, surveillance, reporting, and/or monitoring during the preclosure period.

5.4 DOE must identify and provide a technical basis for the variables, conditions, or other items that are determined to be probable subjects for license specifications. DOE must give special attention to those items that may significantly influence the final design [10 CFR 63.21(c)(18)].

5.5 NRC will impose/include any conditions, including licensing specifications, it considers necessary to protect the health and safety of the public based on important assumptions and considerations made in the preclosure safety analysis and other items identified by DOE.

5.6 Changes to license specifications will require either an amendment of the construction authorization or a license amendment.

5.7 The provisions of 10 CFR 63.44 provide criteria under which DOE may make changes to the geologic repository operations area or procedures as described in the safety analysis report (SAR), and conduct tests or experiments not described in the SAR, without obtaining either an amendment of the construction authorization or a license amendment.

6.0 TRAINING

6.1 10 CFR Part 63 includes a number of requirements for personnel training, indoctrination, qualification, and/or training of key personnel. An integrated training and qualification program addressing the regulatory requirements should be utilized and described in the license application.

6.2 Training should be considered in the design and engineering of the facilities, components, and processes, as well as the PCSA.

6.3 A systems approach to training (SAT) as defined in 10 CFR 55.4 has been successful in the industry and may be considered for the Yucca Mountain repository. SAT elements, include: systematic analysis of the jobs to be performed, learning objectives, training design and implementation, testing or evaluation, and performance feedback on training effectiveness. These elements, combined with human factors and reliability issues should be factored into facility engineering and design, safety analyses, and construction and operations planning.

6.4 Guidance to support the review of the training program is provided in NUREG 1804, *Yucca Mountain Review Plan*, and in NUREG-1220, *Training Review Criteria and Procedures*. The characteristics of the training program should be consistent with ANSI/ANS-3.1-1993, endorsed by the NRC in Regulatory Guide 1.8 and the INPO-managed accreditation program, also endorsed by the Commission.

7.0 PRECLOSURE CRITICALITY

7.1 10 CFR 63 identifies several requirements which are applicable to the analysis of preclosure criticality event sequences. Regulatory Guide 3.71, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, provides useful guidance in demonstrating compliance with these and other regulatory requirements with respect to criticality safety. Existing criticality safety guidance for fuel cycle facilities and transportation and storage systems is considered generally acceptable provided it is adapted to the unique PCSA requirements of 10 CFR 63.

7.2 The preclosure criticality safety analysis should include credible events. In evaluating potential event sequences, consider the fuel handling processes used and type and condition of fuel to be handled.

7.3 A technical basis for the administrative margin used in preclosure criticality analyses should be provided. The NRC has generally found using a 0.05 administrative margin and evaluating all biases and uncertainties at a 95 percent confidence level acceptable for commercial spent nuclear fuel. In general, a smaller administrative margin would require more substantial technical justification.

7.4 The reliability of neutron poisons will need to be addressed if relied upon in the PCSA. Qualification and acceptance testing of neutron poison materials should be commensurate with their reliability and performance credited in the PCSA.

7.5 DOE should inform NRC as early as possible of its plans to use burnup credit to demonstrate compliance with pre-closure criticality safety requirements. Note that NRC guidance and licensing precedence provide for only partial burnup credit due in part to the lack of relevant experimental data necessary to adequately validate calculated isotopic concentrations and cross sections. Preclosure criticality analyses should address uncertainty in data used to justify burnup credit.