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

# Engineered Barrier System Design and Fabrication
# NRC Item 7(e.1), 7(e.2), and 7(e.4)

Presented to:
**NRC/DOE Preclosure Issues Technical Exchange**

Presented by:

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**Bechtel SAIC Company, LLC**


**July 24-26, 2001**

YUCCA MOUNTAIN PROJECT

# Outline

- Objective
- NRC comment: items from *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation*
- DOE response of individual items, as appropriate
- Conclusions

# Objectives

- Describe the basis for resolving NRC Items 7(e.1) A1 through C1 associated with waste package drop issues
- Describe the basis for resolving NRC Items 7(e.2) A1 through E1 associated with waste package fabrication and welding issues
- Describe the basis for resolving NRC Items 7(e.4) A1 and A2 associated with fire design criteria issues

# NRC Item 7(e.1) A1

## NRC Comment:

- The DOE needs to demonstrate that the mesh discretization of the finite element models used to simulate WP drop events are sufficient to provide the level of results resolution needed to assess potential failure

# DOE Response: NRC Item 7(e.1) A1

- **Benchmarking of the finite element analysis (FEA) code (LS-DYNA) against pour canister drop experiments has been performed and show acceptable fidelity with test results**
  - ***Drop Calculation of HLW Canister and Pu Can-in-Canister,*** **CAL-EBS-ME-000015 (work in progress)**
- **Mesh selection is an implicit part of testing performed for code qualification**
  - **Validation Test Report (VTR), SDN 10364-VTR-5.6.2-00 (LS-DYNA may be run either as a module of ANSYS or as a stand-alone code)**
- **Capability to select appropriate mesh density has been demonstrated through comparison to both test results and numerical evaluations**

# NRC Item 7(e.1) A2

## NRC Comment:

- **The DOE needs to provide documentation of all boundary conditions used in the WP drop finite element models. The documentation must also include the technical basis and/or rationale for the boundary conditions**

# DOE Response: NRC Item 7(e.1) A2

- **Many FEA codes require the use of specialized techniques (such as the use of arbitrary spring elements) to properly assess rigid body motion, such as that between the shells of the waste package**
- **LS-DYNA has the capability to handle unconstrained rigid body motions without modification of boundary conditions (i.e., addition of spring elements between surfaces to provide constraints)**
  - **Livermore Software Technology Corporation, *LS-DYNA Theoretical Manual, May 1998***

# NRC Item 7(e.1) B1

## Waste Package Drop Analysis Results - Applicable Failure Criteria

### NRC Comment:

- Has the applicable failure criterion been used? Has damage of the waste package contents been considered when assessing the results of the waste package drop analysis results?

# DOE Response: NRC Item 7(e.1) B1

- **The design objective for the waste package is to ensure no breach for pre-closure design-basis events**
- **While the outer corrosion-resistant barrier of the waste package is predicted to fail in the *Corner Drop of 21-PWR Waste Packages* calculation, the inner shell, which is the primary structural member, was computed to have large margin to failure**
  - ***Corner Drop of 21-PWR Waste Packages,* CAL-UDC-ME-000008**

# DOE Response: NRC Item 7(e.1) B1
(Continued)

- **In the event of a drop:**
  - An assessment would be made as to whether the waste form must be re-packaged
- **The waste form serves as a secondary barrier to release of radionuclides and will be evaluated as deemed appropriate in the future**

# NRC Item 7(e.1) B2

## Waste Package Drop Analysis Results - Failure Criteria Evaluation

### NRC Comment:

- **Has the failure criterion been evaluated properly? In *Corner Drop of 21-PWR Waste Packages,* no discussion was given as to whether waste package failed**

# DOE Response: NRC Item 7(e.1) B2

- In *Corner Drop of 21-PWR Waste Packages,* stresses in the waste package outer shell exceed the allowable, indicating that there may be a breach of the outer shell; however, stresses in the inner shell are below the breach criterion. As long as one of the shells remains intact, there is no release of radionuclides (also see discussion for NRC Item 7(e.1) B1)
  - *Corner Drop of 21-PWR Waste Packages,* CAL-UDC-ME-000008
- In the event of a drop, an assessment would be made as to whether the waste form must be re-packaged

# NRC Item 7(e.1) C1

## Design Basis Waste Package Drop Scenarios

## NRC Comment:

- **What impact orientations are considered? What is the technical basis for the impact orientations considered? Not all impact scenarios described in the waste package system description document appear to have been evaluated**

# DOE Response: NRC Item 7(e.1) C1

- **A wide variety of design-basis dynamic events are considered for the waste package in the *Preclosure Design Basis Events Related to Waste Packages* analysis, including vertical drop, horizontal drop, horizontal drop with emplacement pallet, tip over, and transporter runaway**
  - *Preclosure Design Basis Events Related to Waste Packages,* ANL-MGR-MD-000012
- **As a part of the normal design process, design-basis dynamic events will be re-evaluated as the design for both the surface facility and sub-surface facility mature**

# DOE Response: NRC Item 7(e.1) C1
(Continued)

- **Credible dynamic events will be identified and assessed. The design of the waste package and both the surface and sub-surface facilities will be adjusted to accommodate such challenges to waste package integrity**

# NRC Item 7(e.2) A1

## Chemical Composition and Variation Allowances

### NRC Comment:

- The DOE should provide the technical basis for compositional restrictions used for the procurement and verification of the materials used to construct the WP. If the compositional specification defined in ASTM B-575 is to be used, DOE should demonstrate that the compositional variations allowed for Alloy 22 will result in consistent WP performance

  The chemical composition specifications for Alloy 22 include variations for Cr, Mo, Fe, and W. Altering the compositions of these alloying elements within the range of the chemical composition tolerances may adversely affect the thermal stability and promote the precipitation of intermetallic phases that can decrease the corrosion resistance and impact strength of the alloy

# DOE Response: NRC Item 7(e.2) A1

- **The filler material and base material used for constructing the disposal container of Alloy 22 conforms to the ASME Code and is documented in the FY-00 and FY-01 *Waste Package Operations Fabrication Process Report.* This will also be addressed in the Closure Weld Report issued in September**
- **The samples being tested at LLNL are made from a number of heats of material. Therefore, any data on corrosion rates will take into consideration this variation**
  - ***Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003**

# NRC Item 7(e.2) A2

## Microstructure and Variation Allowances

## NRC Comment:

- **Alteration of the microstructure as a result of alloy processing and fabrication of the WP may adversely affect WP performance. Numerous heats of Alloy 22 (approximately 1 heat per waste package) will be required for the proposed HLW repository. Variations in the microstructure of the alloy cannot be determined from chemical analyses**

# DOE Response: NRC Item 7(e.2) A2

- **Waste Package Project weld flaw distribution study to be completed in this calendar year**
- **The filler material used for the welding of Alloy 22 and the base material conforms to the ASME Code and is documented in the FY-00 and FY-01 *Waste Package Operations Fabrication Process Report.* This will also be addressed in the Closure Weld Report to be issued in September 2001**
- **The samples being tested at LLNL are made from a number of heats of material. Therefore, any data on corrosion rates will take into consideration this variation**
  - ***Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003**

# NRC Item 7(e.2) A3

## Proposed Non-Destructive Evaluation Methods

## NRC Comment:

- **The size, distribution, and frequency of defects in the waste package are recognized as parameters that must be considered in the analyses of early waste package failures. These defects are also important to the mechanical integrity and long term performance of the WP. In the TSPA-SR the earliest waste package failure, which occurs approximately 12,000 years after repository closure, is attributed to the presence of initial defects. In the more recent SSPA, improper heat treatment is considered to lead to WP failure within the 10,000 year regulation period (i.e., one WP failure in less than 2,000 years). Because the size, distribution, and frequency of defects are principal characteristic of the waste package, the DOE should demonstrate the ability of the proposed inspection methods to adequately detect defects in the plate material. The proposed inspection method should be adequate to inspect the final fabricated container prior to waste loading, including postweld annealing**

# DOE Response: NRC Item 7(e.2) A3

- **The plate used to construct the waste packages will be ultrasonically inspected per the ASME Code prior to use in fabrication (§6.2.5, *Waste Package Operations Fabrication Process Report*)**
- **In addition, there will be visual and dimensional examination of the plate material per the ASME Code (§6.2.5, *Waste Package Operations Fabrication Process Report*)**
- **The FY-01 development program includes a study to identify the minimum flaw size that can be detected in Alloy 22 material of this design thickness**
- **Specifics regarding the testing of annealed cylinders are under development (§6.2.5, *Waste Package Operations Fabrication Process Report*)**
  - ***Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003 REV 02, (to be completed in September, 2001)**

# NRC Item 7(e.2) B1

## Contamination Controls

**NRC Comment:**

- **The process for inspection prior to welding to assure that the surfaces are free of potentially adverse contaminants should be provided. Improperly cleaned and contaminated waste package surfaces or filler metal could lead to higher distributions, sizes, and frequencies of weld defects. Because of the nature of the closure weld operation, inspection of the waste package surfaces may be limited by the remote inspection operation. In the Analysis of Mechanisms for Early Waste Package Failure (CRWMS M&O, 2000b) AMR it is assumed that an incorrect cleaning process cannot leave a residue that will adversely affect the performance characteristics of the weld**

# DOE Response: NRC Item 7(e.2) B1

- **Waste package weld area cleaning is addressed in §6.8.5 of the Fabrication Report for both FY-00 and the FY-01. The section states that "the surfaces or parts to be welded shall be visually clean and free of slag, scale, rust, oil, grease, and other deleterious foreign materials for a distance of at least one inch from the weld joint. Chemical cleaning agents for use on stainless steel or nickel alloy shall be approved by the purchaser before use" and will be chosen to leave no residue**
- **Similar words will address this issue in the Closure Weld document for FY-01 for the closure weld, except that the inspection of the closure weld will be remote with optics. This is an acceptable method of inspection in lieu of direct examination**
- **Adherence to these requirements will be provided by operational procedures**
  - ***Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003**

# NRC Item 7(e.2) B2

## Filler Metal Selection

### NRC Comment:

- Filler metal composition may also contribute to thermal instability of Alloy 22 in the weld regions. As previously stated [Section 7(e.2) A1] variations in the concentration may promote the stabilization of secondary phases, and decrease both the localized corrosion resistance that impact strength of the alloy in the weld region

# DOE Response: NRC Item 7(e.2) B2

- **The filler material used for the welding of Alloy 22 conforms to Section II, Part C of the ASME Code and is documented in the FY-00 and FY-01 *Waste Package Operations Fabrication Process Report,* §6.3. This will also be addressed in the Closure Weld Report to be issued in September 2001**
- **The samples being tested at LLNL are made from a number of heats of this wire. Therefore, any data on corrosion rates will take into consideration this variation**
- **The code does not require that this material be impact tested because it is not prone to brittleness**
  - ***Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003**
- **CLST 2.4 and 2.5**

# NRC Item 7(e.2) B3

## Welding Methods (speed, heat,etc)

### NRC Comment:

- **Demonstration that the parameters specified do not adversely affect the quality of the weld should be provided. Welding speed and specific heat input may affect the quality of the weld by increasing the frequency of defects and altering the thermal stability of the alloy. These parameters are expected to be specified during the weld procedure development**

# DOE Response: NRC Item 7(e.2) B3

- As a part of the standard fabrication develop process, the effect of process parameters on material performance will be developed
- Testing of the FY-00 mock up will be conducted after the induction annealing study is complete to ensure that performance is not adversely effected. Results will be documented in a future revision of the *Waste Package Operations Fabrication Process Report*
  - *Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003

# NRC Item 7(e.2) B4

## Environmental Restriction

## NRC Comment:

- **It is assumed that DOE will use an inert shielding gas during the welding of the WP. The complete range of environmental restrictions has not been identified**

# DOE Response: NRC Item 7(e.2) B4

- The purity of the argon and all other critical parameters will be provided in the welding specification
- The gas used for shielding is argon and is no different than normal manufacturing operations that are conducted daily in numerous manufacturing facilities
- The Surface Facility design group is addressing ventilation in the hot cell

# NRC Item 7(e.2) B5

## Weld qualification Tests

## NRC Comment:

- The ability of weld qualification tests to detect weld defects and poorly performing welds should be demonstrated. Mechanical tests may be used to demonstrate physical properties of the weld and the heat affected zone in the base alloy. These tests may not verify the integrity of the WP in terms of corrosion resistance. Specific tests to show the fabrication procedure does not alter the mechanical properties or corrosion resistance should be utilized in the weld procedure qualification

# DOE Response: NRC Item 7(e.2) B5

- **Weld qualification tests required to be conducted by the fabricator prior to any welding do not normally include corrosion testing**
- **However, corrosion tests are being conduct by LLNL at present with weld samples in the annealed and non-annealed condition to study the basic phenomenon. Further tests will be conducted on the FY-00 mock up after the annealing study that is currently underway and will examine the effect of welding and annealing on Alloy 22**
  - ***Waste Package Operations Fabrication Process Report,*** **TDR-EBS-ND-000003**

# NRC Item 7(e.2) C1

## Weld Flaws and Defects

## NRC Comment:

- **Proposed NDE methods can detect flaws and defects at the requisite level of resolution**

  **The assumption that ultrasonic testing (UT) inspections will be as reliable for Alloy 22 as it is for stainless steel welds will need to be verified. This verification should take into consideration the specific closure weld joint designs, weld dimension, and materials and the fact that the UT inspections will be accomplished remotely.**

# DOE Response: NRC Item 7(e.2) C1

- **Waste Package Fabrication has been performing Ultrasonic tests on Alloy 22 material since approximately 1997. These tests are documented in the annual fabrication reports. These tests have been performed on weld joints duplicating the final closure weld joint design**
- **In addition, tests are being conducted to determine the minimum flaw detection and will be reported on in the FY-01 Closure Weld document**
- **A flaw distribution study is under way. This will use numerous NDE techniques to detect the flaws, one of them being UT**

# NRC Item 7(e.2) C2

## Prescribed Surface Cleaning/Finish

**NRC Comment:**

- **Prescribed surface cleaning and finish are compatible with the proposed surface NDE method**

  **The surface finish of the waste package after all fabrication steps have been completed (e.g. welding, post weld treatments, and machining) must comply with the requirements necessary to perform NDE using the methods specified. Improper surface finish may mask some defects. For example, a rough surface finish may reduce the ability of surface sensitive NDE methods (ultrasonic testing or liquid penetrant testing) to detect surface breaking defects such as cracks formed during weld solidification. If undetected, these defects may act as initiation points for early postclosure failure mechanisms.**

# DOE Response: NRC Item 7(e.2) C2

- **The acceptable surface finishes for NDE are normally found in the NDE procedures which meet the ASME Code. Paragraph 1.2.1.10 of the *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* and §4.1 of the *Waste Package Operations Fabrication Process Report,* define the surface finish as 250 µin (6.35 µm)**
  - ***Uncanistered Spent Nuclear Fuel Disposal Container System Description Document,* SDD-UDC-SE-000001**
  - ***Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003**

# NRC Item 7(e.2) C3

## Weld Joint Design

### NRC Comment:

- Prediction of weld defects has been estimated for Alloy 22 using the RR-PRODIGAL weld simulation code and parameters used in the in-service inspection of stainless steel piping. As indicated in the *Analysis of Mechanisms for Early Waste Package Failure* (CRWMS M&O, 2000b) AMR, it is assumed that the information on the weld flaw density for gas tungsten arc welded (GTAW) stainless steels can be applied to GTAW Alloy 22 even though welding of Alloy 22 is recognized to be more difficult than stainless steel. Demonstration of the technical basis for the assumption, considering the geometry of the weld and the composition of the material, should be provided

# DOE Response: NRC Item 7(e.2) C3

- DOE does not find that Alloy 22 is more difficult than Stainless Steel to weld given the correct welding parameters, after welding three mock up joints of approximately sixty feet that duplicate the current design of the waste package
- The flaw distribution study is scheduled to weld another 200 feet of weld duplicating the weld joint design. This information will be used to substantiate the existing data on weld flaws
- The use of the Rolls-Royce Prodigal information will be phased out as applicable data becomes available
- CLST 2.6

# NRC Item 7(e.2) D1

## Post Weld Treatments

**NRC Comment:**

- **Proposed post-weld treatments must not degrade mechanical or corrosion characteristics of the base metal or the weld filler metal**

  **The present waste package fabrication method specifies that laser peening will be used for the inner Alloy 22 lid and induction annealing will be used for the outer Alloy 22 lid. Demonstration of the laser peening method as a means to mitigate tensile stresses in the weld regions without detrimental effects to either the mechanical properties or microstructures in the weld and adjacent base metal has not been provided**

# NRC Item 7(e.2) D1

(Continued)

Thermal gradients during local induction annealing, proposed for the outer closure lid of the waste package, may result in microstructural variations that reduce impact strength and corrosion resistance. In the Analyses of Early Failure Mechanisms (CRWMS M&O, 2000b) AMR it is suggested that independent tests will be conducted to verify that the thermal treatment was performed correctly. At present, the type of tests that will be used have not been identified and the ability of these test methods to detect improper thermal treatment of Alloy 22 has not been demonstrated.

# DOE Response: NRC Item 7(e.2) D1

- **Studies of the laser peening process are ongoing at LLNL**
- **The induction annealing tests are ongoing and tests will be also conducted on the FY-00 mock up**
  - **Results will be documented in a future revision of the *Waste Package Operations Fabrication Process Report* and the Waste Package Closure Weld Report which will be issued in September of FY-01**
- **CLST 2.4 and 2.5**

# NRC Item 7(e.2) E1

## Post Weld Repair

### NRC Comment:

- **Proposed remediation procedures must not degrade mechanical or corrosion characteristics of the base metal or the weld filler metal**

  **The details of waste package remediation are still under development. Repair of welding defects and the process of removing a closure weld and then rewelding the WP will result in increased thermal processing that may alter the mechanical properties and corrosion resistance of the WPS**

# DOE Response: NRC Item 7(e.2) E1

- Repair cycles at the fabricator will be limited and will be discussed in the FY-01 Fabrication document
- Repairs in the hot cell closure weld will be handled by feed back processes that identify the defect at the time of the initiation. This will make the repair minor and less intrusive. If there is an occasion where the defect is major, it would normally be handled by removing the lid and then the fuel and re-packaging
- Any repairs will be followed by appropriate stress mitigation

# NRC Item 7(e.4) A1

## Fire Design Criteria

### NRC Comment:

- **What is the technical basis for considering fire as a beyond-design basis event and an internal event with no release?**

  **The report on Repository Safety Strategy (CRWMS M&O, 2000c) bases classification of fire as a beyond-design-basis event on the information presented in Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation (CRWMS M&O, 2000d). Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation (CRWMS M&O, 2000d) specifies that the waste packages will be designed to withstand the fire environment defined in 10 CFR 71.73(c)(4).**

# NRC Item 7(e.4) A1

(Continued)

## NRC Comment: (Continued)

Criterion 1.2.2.1.11 of *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* (Pettit, 2000) also specifies the same fire design criteria. The fire design criterion in 10 CFR Part 71.73(c)(4) is a fire which is 1,475 °F for 30 minutes. However, Criterion 1.2.1.6 of *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* (Pettit, 2000) states that "WP shall maintain SNF zircaloy cladding temperature below 350°C (662 °F) under normal operations, and below 570 °C (1,058 °F) for short-term exposure to fire, as specified by Criterion 1.2.2.1.11." There is a clear inconsistency between the design criterion and cited reference (10 CFR Part 71 fire design criterion).

# DOE Response: NRC Item 7(e.4) A1

- The technical basis for classifying fire as a beyond-design-basis event is that significant fire hazards will be intentionally precluded at the repository through the design of the systems, structures, and components
- Future analysis of any off-normal waste package events will be based on the Category 1 and 2 credibility criteria defined in 10 CFR 63
- The *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* (Criterion 1.2.1.6), lists the waste form surface temperature not to be exceeded during an off-normal event (570 °C [1058 °F] for short-term exposure to fire). This limit is compared to the peak calculated waste form surface temperature from the off-normal event analysis

# DOE Response: NRC Item 7(e.4) A1
(Continued)

- **Once sufficient information is available on the design of the repository systems, structures and components that interface with the waste package, the technical basis for off-normal waste package events will be documented in the *Preclosure Design Basis Events Related to Waste Packages* analysis. The results from the analysis of off-normal events will be documented in the appropriate design analysis reports (e.g., Design Analysis for UCF Waste Packages)**
  - ***Uncanistered Spent Nuclear Fuel Disposal Container System Description Document,* SDD-UDC-SE-000001**
  - ***Preclosure Design Basis Events Related to Waste Packages,* ANL-MGR-ME-000012**
  - ***Design Analysis for UCF Waste Packages,* ANL-UDC-MD-000001**

# NRC Item 7(e.4) A2

## Fire Degraded Waste Package

## NRC Comment:

- **It does not appear that DOE has considered the degradation of the waste package materials when assessing the potential consequences of a design-basis fire**

# DOE Response: NRC Item 7(e.4) A2

- **Any waste package involved in an off-normal event would be evaluated to ensure that its post-closure performance requirements will not be compromised**
- **For any waste package whose post-closure performance cannot be ensured, it will be necessary to repackage its contents into a fresh waste package**

# Cited Documents

The following documents were cited as sources of information in this presentation:

- *Design Analysis for UCF Waste Packages,* ANL-UDC-MD-000001
- *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document,* SDD-UDC-SE-000001
- *Analysis of Mechanisms for Early Waste Package Failure,* ANL-EBS-MD-000023
- *Preclosure Design Basis Events Related to Waste Packages,* ANL-MGR-MD-000012
- *Waste Package Operations Fabrication Process Report,* TDR-EBS-ND-000003
- *Corner Drop of 21-PWR Waste Packages,* CAL-UDC-ME-000008

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

# Engineered Barrier System Design and Fabrication
# NRC Item 7(e.3)
## Differential Thermal Expansion

Presented to:
**NRC/DOE Preclosure Issues Technical Exchange**

Presented by:

**Bruce Stanley**
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**July 24-26, 2001**

YUCCA MOUNTAIN PROJECT

# Agenda

- **Objective**
- **NRC Item 7.e.3 – Differential Thermal Expansion Issues**
- **DOE Response**
- **Discussion of Path Forward**

# Objective

- **Discuss NRC staff issue 7.e.3 on Differential Thermal Expansion Issues**

# NRC 7.e.3 - Differential Thermal Expansion Issues

## NRC Comment:

- **What provisions have been made for thermal expansion in the design of the gantry crane rails?**

  **Any thermal expansion joints used to prevent the gantry crane rails from deforming beyond allowable tolerances or buckling under thermal load must be capable of supporting the gantry crane without causing derailment**

- **What provisions have been made for thermal expansion of the invert structural frame beams attached to the drift wall?**

# NRC 7.e.3 - Differential Thermal Expansion Issues
(Continued)

**DOE Response: (Crane rails)**

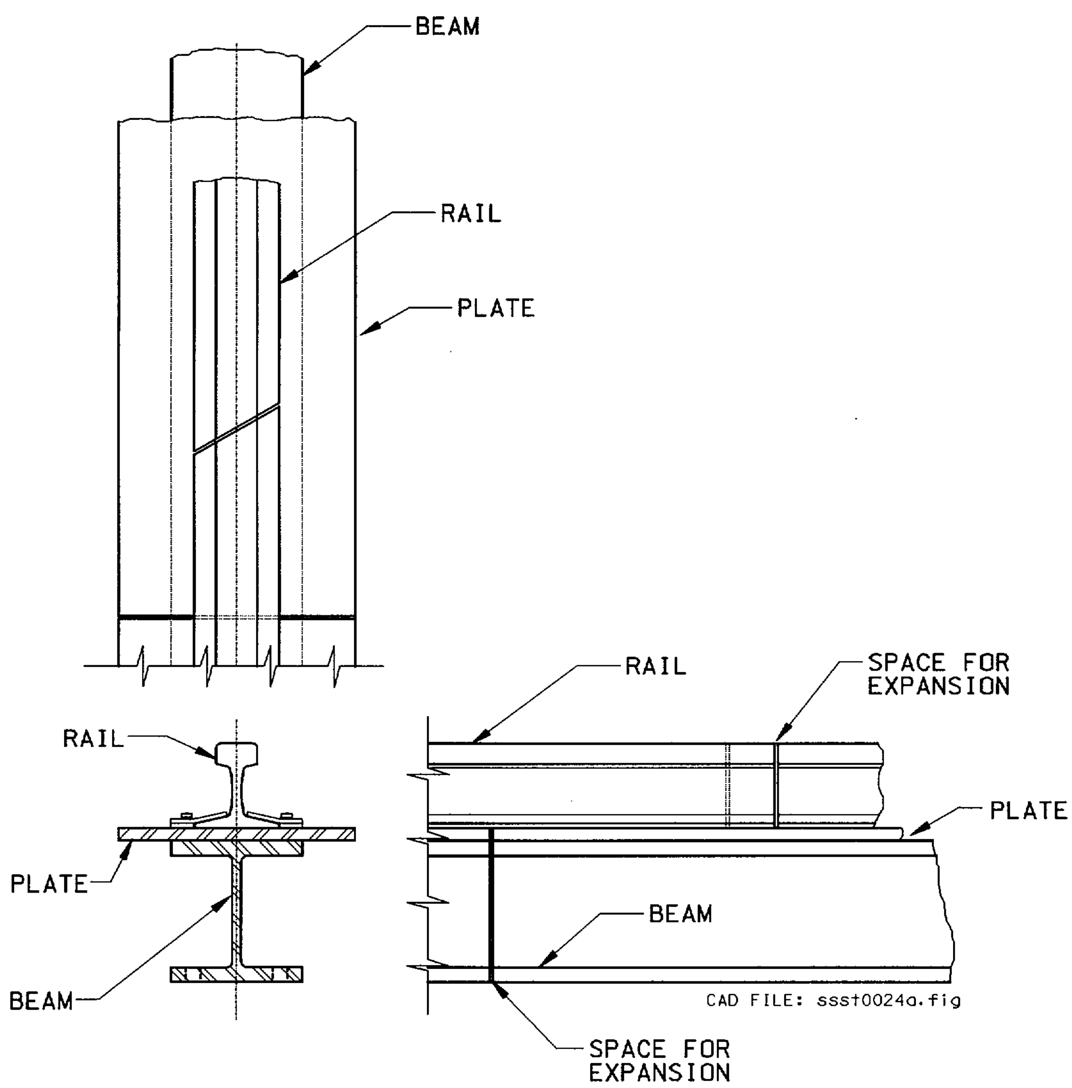
- Preliminary calculations performed to establish viability of concept
- 40 foot rail section at 200°C will expand about 0.53 inches
- Although not detailed at this time, a combination of fixed and slotted anchors will accommodate expansion
- Configuration of expansion gaps are adequate to support transporter weight at various temperatures
- Rail system will be designed to be maintainable for the required service life

# NRC 7.e.3 - Differential Thermal Expansion Issues
(Continued)

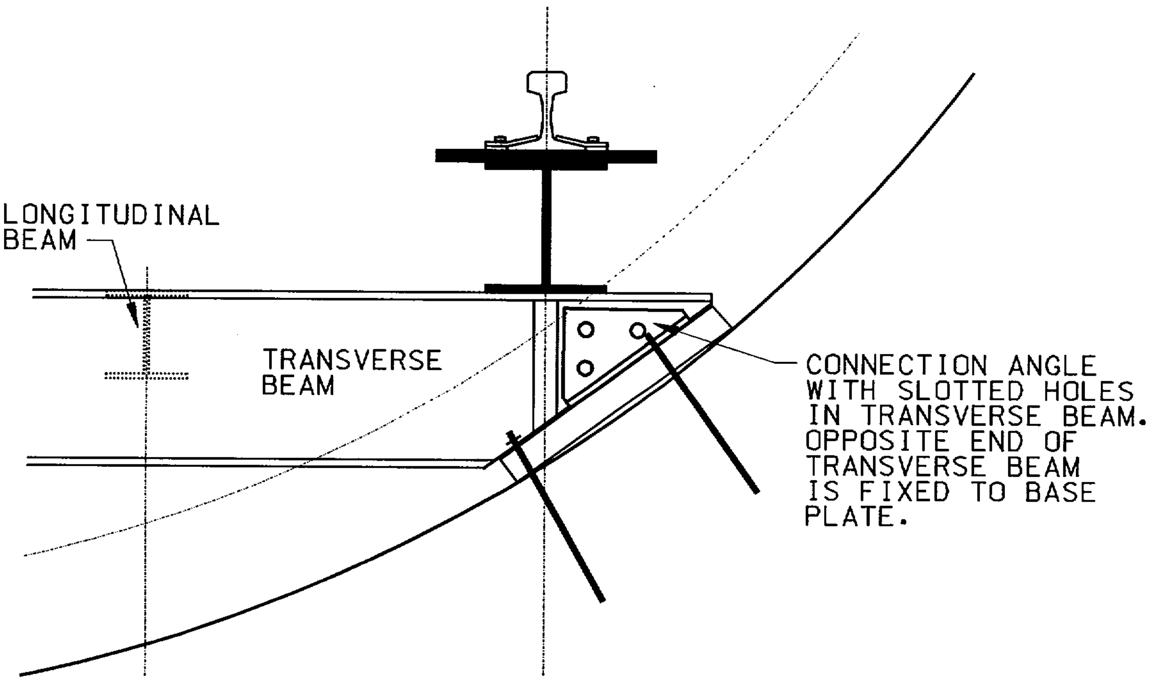
## DOE Response: (Invert transfer beams)

- Invert transfer beams are anchored on one end, and feature a slotted connection on the other end, allowing for expansion
- Preliminary estimate of a typical transfer beam expansion is about 0.13 inches at 200°C
- Design not yet detailed, and the invert configuration may change for LA

# View of Emplacement Drift Rail



# Transverse Beam Connections




NOTE:
LONGITUDINAL BEAMS WILL BE FIXED AT ONE END AND ALLOWED TO EXPAND AT OPPOSITE END.

CAD FILE: ssst0024a.fig

## July 24-26, 2001 NRC/DOE Preclosure Issues Technical Exchange Meeting
## DELTA Analysis

| NRC Item 2(a)-High Level Waste Characterization/burn-up Credit | | |
|---|---|---|
| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| DOE is planning to use burn-up credit in the design of the criticality control system of the waste packages for commercial Spent Nuclear Fuel (SNF). NRC would require the verification of the SNF assembly burn-up using physical measurements. | Waste Package<br>Thomas Doering | Disposal Criticality Analysis Methodology Topical Report, Rev. 1 cover letter noted that the process for verification of fuel assembly burnup would be addressed in the Preclosure Criticality Analysis Process Report.<br><br>Revision 1 of the Preclosure Criticality Analysis Process Report is currently scheduled for FY-02, and will include the approach for verification of fuel assembly burnup for burnup credit.<br><br>Burnup Credit - General Approach<br><br>• DOE acknowledges that burnups used in design evaluations and performance assessment must be demonstrated to be adequate and/or conservative.<br>• Burnup credit, requiring assembly burnup values, is only being sought for commercial spent nuclear fuel.<br>• DOE believes burnup information for the majority of the fuel developed and available through reactor records is the best source of assembly burnup (not specifically the RW-859 form in its current state).<br>• Assembly burnup information from Reactor Records are from NRC recognized and continually monitored quality programs (is the basis for nuclear power operations and cycle reload analysis safety licensing).<br>• DOE believes the uncertainty in physical measurement (without the use of supporting operational history inputs) is higher than reactor records.<br>• DOE acknowledges the need to have some burnup measurement capability at the surface facility. |

| **NRC Item 2(a)-High Level Waste Characterization/burn-up Credit** | | |
|---|---|---|
| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| | | • DOE believes only a very small percentage of commercial spent nuclear fuel will potentially have inadequate reactor records such that measurement will be required or no burnup credit will be applied for that assembly.<br>• DOE sees the need for regulatory consistency among 10 CFR Parts 50, 63, 71 and 72 in the sufficiency criteria for verifying fuel assembly burnup values. |

| **NRC Item (2.b)-Preclosure Criticality Issues** | | |
|---|---|---|
| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| A. Flooding, caused by either internal or external events, of the Waste Handling Building functional areas (e.g., waste receipt, carrier/cask transport, carrier/cask preparation, waste handling-carrier bay, or waste-handling-canister transfer areas) does not appear to have been considered when evaluating potential criticality hazard analyses. DOE should consider the potential for criticality as a result of external and internal flooding events. | Waste Package<br>Thomas Doering | A.<br><br>DOE understands that the potential for flooding and subsequent criticality needs to be evaluated.<br><br>Flooding is an item considered in the Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation, Section 5.1.1.3.1, Item B.5.c.<br><br>DOE has established design requirements that precludes preclosure criticality unless two unlikely independent, sequential or concurrent, events occur (e.g., Disposal Container Handling System Description Document, SDD-DCH-SE-000001).<br><br>DOE will be evaluating the potential for flooding and subsequent criticality as part of the normal criticality safety evaluations, which will be available for NRC review in the licensing documentation. |
| B. DOE should calculate the probability of criticality for the category 1 and 2 events in Tables 5-5 and 5-6 of the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O, 2000d). If criticality is determined to have a probability of occurrence greater than $10^{-6}$/yr, DOE would need to calculate the consequences of the criticality event. Criticality is most likely to result from those events involving fuel assembly drops and drops of assembly baskets, Navy canisters, DOE canisters, or commercial canisters in which fuel assemblies spill out of the basket or canister, separating the fissile | | B.<br><br>DOE understands that the probability of criticality should be considered for the Category 1 and 2 events in Tables 5-5 and 5-6 of the Preliminary Preclosure Safety Assessment.<br><br>The initial assessment is that the probability of criticality will be less than 10-6/yr because of the preclosure criticality safety strategy (Section 6 of the Preliminary Preclosure Safety Assessment) and the design requirements to implement the strategy. |

| NRC Item (2.b)-Preclosure Criticality Issues | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| material from the neutron absorbing additives. | | DOE has established design requirements that preclude preclosure criticality unless two unlikely independent events occur (e.g., Disposal Container Handling System Description Document, SDD-DCH-SE-000001). The probability of two unlikely independent events occurring will be less than 10-6/yr.<br><br>DOE will be documenting these evaluations as part of the normal criticality safety evaluations, which will be available for NRC review in the licensing documentation. |
| C. DOE should provide the technical basis for making "Criticality Event in Pool" and "Criticality Associated with Small Canister Staging Rack" Beyond Design Basis Events (BDBEs) (see Table 5-12 of the PPSA). It is not clear how DOE can reach this conclusion without analyzing the criticality potential from fuel assembly, assembly basket, or canister drop events. | | C.<br><br>DOE understands that the technical basis for designating events BDBEs should be given.<br><br>The initial technical basis used to designate "Criticality Event in Pool" and "Criticality Associated with Small Canister Staging Rack" as BDBEs was the preclosure criticality safety strategy (Section 6 of the Preliminary Preclosure Safety Assessment) and the design requirements to implement the strategy.<br><br>DOE has established design requirements that preclude preclosure criticality unless two unlikely independent events occur (e.g., Disposal Container Handling System Description Document, SDD-DCH-SE-000001).<br><br>DOE will be documenting these evaluations as part of the normal criticality safety evaluations, which will be available |

| NRC Item (2.b)-Preclosure Criticality Issues | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| | | for NRC review in the licensing documentation. |
| D. DOE should analyze the probability of criticality resulting from misload events listed in Table 5-7 of the PPSA under the category entitled "Internal Event Sequences with No Release." If criticality is determined to have a probability of occurrence greater than $10^{-6}$/yr, DOE would need to calculate the consequences of the criticality event. | | D.<br><br>DOE understands that the probability of potentially critical events from misloads, as identified in Table 5-7 of the PPSA, should be determined.<br><br>Based on the preclosure criticality safety strategy (Section 6 of the Preliminary Preclosure Safety Assessment) and the design requirements to implement the strategy (i.e., System Description Documents), the probability will be less than 10-6/yr.<br><br>DOE's design requirements preclude preclosure criticality unless two unlikely independent events occur (e.g., Disposal Container Handling System Description Document, SDD-DCH-SE-000001). The probability of two unlikely independent events occurring will be less than 10-6/yr.<br><br>DOE will be documenting these evaluations as part of the normal criticality safety evaluations, which will be available for NRC review in the licensing documentation. |

| **NRC Item 3(a)-Aircraft Hazards** | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| The DOE excluded aircraft crash hazard from further consideration for the design of the proposed repository based on calculations documented in **"MGR Aircraft Crash Frequency Analysis (ANL-WHS-000001, Revision 00, R. Morissette, 1999)."** The NRC Staff concludes that the exclusion of aircraft crash from the list of potential human-induced hazards that may affect the proposed repository is premature. | ISA<br>Dennis Richardson/Dealis Gwyn | DOE agrees with the comments and a more extensive evaluation is planned for LA. The analysis referenced (*MGR Aircraft Crash Frequency Analysis,* 9/29/99) is a preliminary evaluation of aircraft hazards at Yucca Mountain. It was performed to provide an understanding of the potential risks and the work that is needed to be performed to ensure the hazard is adequately addressed for LA. It also provided assurance that aircraft hazards do present site suitability issues that cannot be resolved through more detailed analyses, engineering solutions, and/or administrative controls.<br><br>DOE agrees with comment and will develop a vicinity map with aircraft types and activities identified. An evaluation of the aircraft activities within this vicinity will determine which activities will require quantitative crash frequency analysis. DOE is defining "vicinity" as the area where the flight activity will have an impact on the evaluation of aircraft hazards. DOE will obtain available information from Nellis AFB documents and staff regarding future flight activities, aircraft types, and changes in military exercises. |
| The DOE should provide a detailed analysis of the aircraft crash hazards by taking into consideration all types of aircraft flying in the vicinity of the proposed site with a reasonable projection into future flight activities including introduction of new type(s) of aircraft and change in military exercises. | | DOE will obtain information form DOE/Nevada Operations regarding potential changes to flight activities in DOE Controlled airspace over the Nevada Test Site. Using available information, types of aircraft (e.g., large twin engine, small engine) and projected flight activities for the preclosure period will be estimated for the evaluation. |

| NRC Item 3(a)-Aircraft Hazards | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| The annual aircraft crash probability will be the summation of probabilities from all types of aircraft from different operations. | | DOE agrees with comment and will sum the annual frequencies from all operations that required quantitative crash frequency analysis within the vicinity. |

**NRC Item 3(e)-Tornado Missile Hazards**

| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
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| The DOE has not assumed the characteristics of the missile in the Uncanistered Spent Nuclear Fuel Disposal Container System Description Document (Pettit, 2000) commensurate with the bounding characteristics of the tornado missiles for the region. No basis has been provided for the assumed alternate characteristics. | ISA<br>Dennis Richardson/Dealis Gwyn | Tornado missiles are not a hazard for Disposal Container/Waste Packages while they are inside the Waste Handling Building or the subsurface facility.<br><br>Necessary portions of the Waste Handling Building will be designed to withstand credible tornado missile hazards.<br><br>Tornado missiles are not a hazard for the subsurface facility.<br><br>During the brief exposure time when a transporter carrying a Waste Package travels between the surface and subsurface facilities, preliminary screening analysis indicates that none of the disposal containers, including the Uncanistered Spent Nuclear Fuel Disposal Container, will be required to withstand the characteristics of a design-basis tornado missile because it is an incredible event scenario (i.e., frequency < 1E-06/yr).<br><br>The missile criteria in the *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* is not related to tornado missiles and therefore does not represent "alternative characteristics" for design-basis tornado missiles.<br><br>SDD criteria requires a disposal container to withstand the impact of a 0.5 kg missile (modeled as 1 cm diameter, 5 cm long valve stem) with a velocity of 5.7 meters per second without breaching. Criteria addresses a potential hazard identified in a hazards analysis of the Waste Handling Building. |

| NRC Item 4(a)-Events Screened Out By Design | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| DOE, in their current ISA, identified about 35 event sequences from the GROA operations, that has been classified as "Internal Events Sequence with No Release. The events in this category are credible; however, these events are eliminated by design. SSCs credited to prevent these events by design are waste packages, bridge cranes, lifting fixtures, shipping casks, canisters, etc. DOE should adopt the NRC position on this issue enumerated in the FEP Screening Methodology: NRC Staff Views and Comments, May 14, 2001 which has been agreed upon by DOE and will be incorporated in agreement between NRC and DOE in the upcoming TSPA Technical Exchange. The staff position is paraphrased below: | ISA<br>Dennis Richardson/Dealis Gwyn | |
| ▯ DOE can screen preclosure design basis events based on a proposed design concept<br>▯ Consistent with overall risk-informed performance-based philosophy in proposed Part 63<br>▯ Screening can be based on either:<br>1. Probability, or<br>2. Consequences<br><br>▯ DOE will need to demonstrate that the particular design feature can perform its intended mitigation function over the time period of regulatory interest<br><br>▯ For supporting screening arguments, probability values for component failure or events potentially leading to the failure of the design feature, range, and distributions or relevant variables and/or | | DOE agrees that the screening of design basis events must be defensible. One of the factors to consider is how well the screening basis is understood (e.g., failure probabilities, event sequence probabilities, consequences). Uncertainties must be addressed to the extent they may impact either the categorization or the consequences of a potential design basis event. DOE agrees that all design basis event categorizations, component failure probabilities, consequence analyses, etc will have to be technically defensible to support their use. This defense may be in terms of quantified uncertainties, "stacking of conservatism's," or a qualitative argument as to the appropriateness of the information to support the preclosure safety analysis process. |

| NRC Item 4(a)-Events Screened Out By Design | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| boundary assumptions should be: technically defensible, and account for uncertainty and variability. Similarly, screening by consequence should be technically defensible and account for uncertainty and variability in the parameters. | | |

| NRC Item 4(b)-Justification of Probability Estimates | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| 1) DOE should justify the estimated probability of failure for the equipment and components used in surface and subsurface operations event sequence analysis. For example, the data used by DOE to determine probability of drop events for assemblies and shipping casks are based on analysis of the drop events of the cranes obtained from the industry. DOE should provide justification that:<br>a) the data used from the industry to estimate failure probability has been adequately analyzed<br>b) the data used are appropriate for use in repository operations. | ISA<br>Dennis Richardson/Dealis Gwyn | 1) Similar to the discussion in 4a), DOE agrees that failure probabilities must be justified sufficient to support the design basis event categorization process.<br><br>Appropriate attention will be given to event scenarios that are near thresholds (i.e., Category 1/Category 2, Category 2/BDBE) to either ensure that technical basis supports the event categorization or that the categorization is conservative (e.g., an event that is borderline Category 2/BDBE may be conservatively categorized as Category 2).<br><br>The basis for the categorization will demonstrate that the inputs used (e.g., failure rates) are correct and appropriate for its use at a potential repository. |
| 2) DOE has presented ISA analyses with only point estimates of frequency of failure of different components. However, it is not clear whether the probability estimates used in these analyses represent mean, median, or some other point estimates. Frequency of component failure is highly uncertain. Consequently, the analyses presented by the DOE do not consider the uncertainty and variability associated with each frequency or probability estimate. By ignoring the uncertainty and variability associated with the event sequences using only one point estimate, there is a distinct possibility of incorrectly classifying an event or an event sequence with associated consequences. DOE should conduct sensitivity and uncertainty analyses to estimate the probability of failure during the preclosure period. Frequencies of component failures should be assigned probability distributions and mean probability of failure | | 2) Categorization of design basis events will be defensible, which includes the inputs used. DOE will justify the correctness and appropriateness of failure rates used in preclosure safety analyses. This would include discussions on the uncertainties and sensitivities associated with any failure rates (or other inputs used in the analyses). Where applicable mean values will be used to categorize events. |

| NRC Item 4(b)-Justification of Probability Estimates | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| should be estimated.<br><br>TALKING POINTS ON FOR JUNE 28 TELECON ON THE USE OF DISTRIBUTIONS FOR ASSESSMENTS OF UNCERTAINTY AND VARIABILITY<br><br>FREQUENCY<br><br>1. PROBABILITY ESTIMATE OF COMPONENT FAILURE: DOE is encouraged to consider uncertainty and variability in their probability estimate of component failure. To account for uncertainty and variability, DOE may assign distributions to component failures.<br><br>2. EVENTS SEQUENCE CATEGORIZATION: If DOE obtains a probability distribution for the frequency of a preclosure event sequence, the mean value of that distribution can be used to categorize the event sequence, provided that the probability distributions of the component failures are valid and account appropriately for uncertainty and variability. | | 1. DOE will, as appropriate, assign uncertainty distributions to failure rate estimates of component failure. These distributions will be used to estimate the mean component failure rate and the variability in the estimated failure rate.<br><br>2. Probability distribution functions will be used for estimating the uncertainty in an event sequence frequency and the mean frequency, rather than a point estimate, will be used to categorize the event sequence as a Category 1, Category 2 or Beyond Design Basis Event. |

| NRC Item 5(a/b)-Dose Calculations for Design Basis Events | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| DOE has proposed methodologies for calculating the anticipated annual average doses for Category-1 Design Basis Events (DBEs) and per-event-sequence dose for Category-2 DBEs in its November 2000 "Repository Safety Strategy" (RSS) and Design Basis Event Frequency and Dose Calculation for Site Recommendation (June 2000). The Category-1 dose is based on the annual exposure to a hypothetical subsistence farmer living at the site boundary while the Category-2 dose is based on a short-term (8-hour) acute exposure to an individual at the site boundary.<br><br>The total Category-1 annual dose estimate is based on contributions from three sources: (1) Category-1 DBEs; (2) routine releases from normal operations at the surface waste handling facility; and (3) normal operational releases from the subsurface facility. The annual dose resulting from Category-1 DBEs is calculated using the equation:<br><br>$D = F_i D_i$<br><br>where:<br><br>total annual dose<br>$F_i$ frequency of the $i^{th}$ Category-1 event sequence<br>dose resulting from the $i^{th}$ Category-1event sequence | ISA<br>Dennis Richardson/Dealis Gwyn | |

| NRC Item 5(a/b)-Dose Calculations for Design Basis Events | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| **Discussion**<br><br>Based on the review of DOE's "Repository Safety Strategy and Design Basis Events Dose Calculations for Site Recommendation," and subsequent discussions with DOE and its contractors, the NRC staff understands that the above equation will be used for demonstrating compliance with preclosure performance objectives (such as 25 mrem/yr and 100 mrem/yr under Category-1 DBEs) in the license application for construction authorization. However, it is assumed here that, after any license to receive and possess waste is issued, and waste handling operations start, actual (measured) doses would be used to monitor and report ongoing compliance with regulatory dose limits. Calculation of annual doses as a part of preclosure safety assessments (PCSA/ISA) for demonstrating compliance during design stage, two classes of Category-1 events would be identified: (1) events that occur one or more times a year; and (2) events that occur less than once a year but at least once during the operational period. Calculation of the "annual" dose is complicated by consideration of the above two classes of events in a given year. Specifically, a method is needed to appropriately combine events expected to occur one or more times a year and those expected to occur less than once a year in order to calculate the doses in a given year of operation. DOE has provided the above approach, which estimates annual dose based on aggregation of <u>all</u> Category-1 events in an "annualized" manner (i.e., consequence weighted by the frequency).<br><br>During recent discussion among NRC, DOE and their contractors, DOE made further clarifications. The following | | |

| NRC Item 5(a/b)-Dose Calculations for Design Basis Events | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| summarizes the NRC staff understanding of DOE approach:<br><br>1. A frequency weighted sum of <u>all</u> Category-1 DBE doses (an in the equation above) will be added to the routine operational releases to demonstrate compliance with the regulatory dose limits.<br>2. In addition, the dose estimated to result from any single Category-1 event sequence would not be allowed to exceed the regulatory dose limits.<br>3. For the License Application Design of structures, systems and components important to safety under Category-2 DBEs, doses are calculated on a per-event-sequence basis and compared with the regulatory limit (5rem/event sequence). | | DOE agrees with the NRC staffs understanding of the DOE proposed approach to show compliance for Category 1 and Category 2 Design Basis Events and considers this issue resolved |
| <u>Staff Position</u><br><br>**The staff believes this approach (i.e., sum of the annualized/frequency weighted doses for Category-1 DBEs and per-event-sequence-doses for category-2 DBEs) is acceptable because it is reasonable and technically defensible. In addition, it simplifies DOE's demonstration of compliance in the license application PCSA/ISA and NRC's review and compliance determination. It should be noted that this staff position is limited to the use of the approach discussed here and does not express any regulatory position regarding the dose estimates presented in the various DOE documents.**<br><br>TALKING POINTS FOR JUNE 28 TELECON ON THE USE OF DISTRIBUTIONS FOR ASSESSMENTS OF | | DOE agrees with the NRC staffs understanding of the DOE proposed approach to show compliance for Category 1 and Category 2 Design Basis Events and considers this issue resolved |

| NRC Item 5(a/b)-Dose Calculations for Design Basis Events | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| UNCERTAINTY AND VARIABILITY<br><br>CONSEQUENCE<br><br>1. COMPARING A DOSE DISTRIBUTION TO THE DOSE LIMIT. If DOE performs a dose calculation that results in a final dose distribution, the mean value of that distribution can be compared to the dose limit for demonstrating compliance. In addition to the mean value being below the dose limit, the assessment needs to be valid using appropriate scenarios, models, and parameters for the specific situation.<br><br>2. USE OF DISTRIBUTIONS FOR PARAMETERS. DOE can use a distribution for an individual parameter as long as<br>▯ the distribution is valid for the case under assessment,<br>▯ the values selected for distribution provide reasonable assurance that doses will not be underestimated, and<br>▯ the distribution accounts appropriately for uncertainty and variability.<br>If a point estimate is used for individual parameters, reasonable assurance should be provided that the value selected will not result in a dose underestimation considering uncertainty and variability.<br><br>**Points needing clarification**<br><br>(1) Future revisions of the RSS and other reports must document that no single Category-1event sequence will result in a dose that exceeds the regulatory limits. | | (1) DOE will ensure that the appropriate project documents that will be used to support LA will be consistent in terminology, definitions, and equations. The process for |

| NRC Item 5(a/b)-Dose Calculations for Design Basis Events | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| | | demonstrating compliance with Category 1 limits will be clarified. DOE will demonstrate that the annual exposure to the public due to Category 1 events (frequency weighted), including normal operations is less that the regulatory limit. Also, DOE will demonstrate that no single Category 1 event (which is evaluated on a per event basis) will exceed the regulatory limit. |
| (2) In order to facilitate the staff review and help focus the design review on the particular event sequences that might contribute higher shares of doses to the total calculated annual dose, it will be necessary for the DOE to provide a table of dose contributions from individual Category-1 event sequences in addition to the sum. | | (2) DOE in future preclosure safety documents will provide a table of dose contributions from individual Category 1 event sequences in addition to the sum. . |
| (3) The approach used by DOE for demonstrating compliance with the regulatory limits for combination of Category-1 event sequences that could occur in given year should be made transparent in the RSS. | | (3) The DOE approach for Category 1 compliance will be described in appropriate design documents in a clear and technically defensible manner. |
| (4) The RSS should also clarify how the dose calculation approach will be used in developing the list of structures, systems and components important to safety (Q-list). | | (4) DOE agrees to clarify the approach that will be used to develop the list of SSCs important to safety (Q-List) in the appropriate project documents. |
| (5) The RSS should explain in clear terms how the bounding dose term (referred to in DOE's Quality Level (QL) categorization process will be used in binning the | | (5) DOE will clarify this point in the appropriate design documents that support LA. |

| NRC Item 5(a/b)-Dose Calculations for Design Basis Events | | |
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| items on the Q-list. | | |

## NRC Item 6(a)-Q-List Methodology

| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
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| 1. DOE should provide adequate justification for including and excluding SSCs important to safety. For example, shield doors and isolation doors, used in surface and subsurface, are currently excluded from the Q-List. DOE should provide acceptable justification for not identifying and classifying these SSCs which perform radiation protection function during surface and subsurface operations. Similarly, the rail system has not been classified. | ISA<br>Dennis Richardson/Dealis Gwyn | 1. DOE agrees with the comment and will provide adequate justification for the classification of all SSCs. The examples cited are not excluded from the Q-List; they have not been specifically classified. The classification levels reflect the level of development of the MGR architecture, which at this point in time does not reflect all the major components. The items classified and the quality classification of the items will evolve consistent with the design and the integrated safety analysis. At the time of LA, the Q-List will reflect the classifications of major components. |
| 2. DOE identification of SSCs important to safety is based on the QL classification process, which is in accordance with the guidance in the DOE procedure QAP-2-3. DOE classifies SSCs into four categorizes as QL-1, 2 and 3, and conventional quality. SSCs binned in QL 1,2, and 3 are considered to be important to safety. Proposed Part 63.112 requires that the SSCs important to safety be identified by ISA. The classification analysis in QAP 2-3 is based on answering a set of checklist questions in each class. The QL process appears to be qualitative and from the classification analyses presented in several classifications documents do not appear to use the ISA results. DOE should provide a "walk through" example from identification of a SSC important to safety to its QL classification and provide adequate justification that exclusion of SSCs as important to safety is based on ISA process. | | 2. DOE agrees that the classifications need to be based on the ISA results. The preliminary classification work is based on engineering judgment, project strategies, and preliminary calculations. The classification analyses, hazard analyses, categorization and consequence evaluations, and the preclosure safety analyses are based on a preliminary evaluation of a conceptual design. The classification analyses that support LA will be more directly linked to ISA analyses. Adequate justification will be included for SSC classification.<br><br>DOE has reviewed the qualitative criteria and will discuss proposed criteria that include more quantitative criteria, where appropriate, as well as clarifying other criteria Quality Level 1 preclosure Category 1 and 2 criteria will be modified to more clearly link with the "takeaway" process (i.e., if the items fails, will the event sequence result in a dose in excess of 100 mrem or 5 rem, as appropriate, similar questions will be included related to changes in frequency as a result of the item failure). Quality Level 2 and 3 will have similar criteria, |

| NRC Item 6(a)-Q-List Methodology | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| | | but with lower consequence impacts for Category 1 and 2 design basis events. Criteria that address changes in frequency for Category 1 and 2 design basis events will also be included. |
| 3. DOE should provide detailed explanation and examples showing how they propose to use the aggregated annualized dose expression $F_iD_i + D_e$, (where De is the maximum dose, Figure 8-1 of Repository Safety Strategy document) along with importance (one-off or take-away) analysis in the classification process of the SSCs involved in Category 1 event sequence. DOE should justify exclusion of dose from surface and subsurface normal operational releases in the above annualized dose expression. DOE should justify why the above expression was not used in the classification analysis of SSCs particularly in the Assembly Transfer Pool area where Category 1 DBEs were identified in the current ISA analysis. | | 3. DOE agrees that the equations that reflect Category 1 compliance should be clarified and that project documents should be updated to reflect that clarification. As a point of further clarification, contributions from surface and subsurface normal releases are included in the annualized dose (assumed annual event probability for normal operations dose of 1.0). To clarify, the equations will be modified as follows: $Dn + \Sigma FiDi + De$ (where Dn represents the dose from surface and subsurface normal releases). This will be clarified in future document revisions. DOE also recognizes that the basis for classifications must be technically defensible and link to the integrated safety analysis.<br><br>However, much of the basis at this point in design development is linked to implementing a desired safety strategy and engineering judgment. As the design and the ISA develop, stronger links between classification analyses and the ISA will be included. As ISA analyses are performed, classification analyses will be updated (or impact reviews performed) to ensure consistency and establish appropriate links. With respect to the Assembly Transfer Pool area SSCs, due to the preliminary nature of the design and supporting analyses, the logic was not captured in the classification analysis. As discussed above, these analyses, as well as others, will be updated to capture ISA inputs. |

**NRC Item 6(a)-Q-List Methodology**

| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
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| 4. DOE should consider multiple Category 1 design basis events occurring in a single year. When determining QL classification for Category 1 event sequences, DOE should consider combinations of Category 1 DBEs that could occur in the same year with a probability of at least 0.01. For such combinations, the doses from those particular event sequences can be summed together with the anticipated releases from normal operations to yield a total annual dose from Category 1 DBEs. | | 4. DOE will consider Category 1 combinations of design basis events occurring in a single year when performing SSC classifications. See response to NRC Item 6(a), Comment 3 and presentation for the steps to classify SSCs to address Category 1 Design Basis Events. |
| 5. The preclosure screening criteria, in the procedure QAP-2-3, requiring the SSCs to limit the onsite worker dose during normal operations and Category 1 DBEs from exceeding the occupational limits of 10 CFR Part 20 has been assigned to QL-3. DOE should provide rationale for classifying the SSCs required to limit onsite worker doses as QL 3 item. | | 5. DOE believes that classifying items that limit onsite worker dose as QL-3 will ensure that worker radiological risks are appropriately addressed. Reliance on activity controls (e.g., worker training, radiation protection program, procedures) has been demonstrated to be successful in the nuclear industry. It is DOE's position that these activity controls, in combination with QL-3 SSC controls, are more than adequate to address worker radiological safety. |

| NRC Concerns Involving the Use of NUREG/CR-6407 Classification Categories for Risk Significance Categorization of SSCIS | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| DOE's proposed risk categorization methodology is based on the quality levels defined in procedure QAP-2-3[20] and its associated screening criteria, as discussed earlier in this paper. DOE has stated that the quality level or "important to safety classification" is "consistent" [19] with, the three tier approach and classification categories described in NUREG/CR-6407. It is important to note that the approach identified in NUREG/CR-6407 (and its predecessor RG 7.10) predates all of the risk-informed policy and guidance developed by the NRC since the Commission's Final Policy Statement on the Use of PRA[4] issued in 1995. Further, the approach to classification identified in NUREG/CR-6407 does not require the consideration of risk insights or significance. It does not consider probability. It only assesses consequences in terms of the maximum amount of radioactive material permitted in the transportation package. It assigns classification categories using a strictly deterministic approach. and as such is clearly not considered risk-informed. The staff has several concerns regarding DOE's use of the classification categories described in NUREG/CR-6407 for the risk significance categorization of SSCIS of a potential GROA. The following discussion outlines these staff concerns , several of which involve the use of QL-2 screening criteria (as identified in procedure QAP-2-3).<br><br>**Concern 1:** Consistency with Regulation<br>Two of DOE's QL-2 screening criteria which are not consistent with the definition of event sequences provided in the proposed § 63.2 (QAP-2-3 Appendix II, Checklist Items | ISA<br>Dennis Richardson/Dealis Gwyn | <br><br><br><br><br>**Concern 1:** DOE agrees that the classification procedure be clarified to better link with the ISA approach and processes to be used in LA. In addition the ISA guide (which is currently under development) will better clarify the thought process that will be used to address the criteria. The ISA approach will |

| NRC Concerns Involving the Use of NUREG/CR-6407 Classification Categories for Risk Significance Categorization of SSCIS | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| 8.2.5 and 8.2.6). These screening criteria only consider the failure of one item in conjunction with *an additional item* or *administrative control* (i.e., indirect impact). Whereas the definition of event sequences (presented in the proposed final 63.2) states: An event sequence includes one or more initiating events and associated combinations of repository system component failures,... and does not place a limit on the number of component failures.<br><br>**Proposed Path to Resolution:** Revise the screening criteria in QAP-2-3 Appendix II, Checklist Items **8.2.5** and **8.2.6** to be consistent with the definition of event sequences (presented in the proposed final 63.2), as described in Concern 1. | | make extensive use of event trees, or alternative definitions of event sequences, that will clearly reveal any combination of events that lead to a release of, or exposure to, radioactivity. Events considered in potential event sequences will include potential failures or unavailability of SSCs as well as potential human errors (e.g., failure to comply/perform an administrative control). Potential common-cause or dependent failures will be identified.<br><br>Classification as QL-1, -2, or -3 will be assigned to SSCs important to safety as appropriate to their significance in preventing or mitigating event sequences. Consideration of multiple failures in credible scenarios will be included when determining items important to safety.<br><br>DOE will update the classification procedure to clarify the process and to better tie it to the ISA. Also, the ISA guide will better clarify how multiple failures will be considered when determining items important to safety. |
| **Concern 2:** Justification for Screening Criteria<br>The screening criterion identified in QAP-2-3 Appendix II, Checklist Items 8.2.3 may result in mis-categorization. This criterion states: As a result of DBE, could consequential failure of the item, which is not intended to perform a QL-1 radiological safety function, prevent QL-1 SSCIS from performing their intended radiological safety function? The purpose and justification for this screening criterion are unclear. According to DOEs definition of QL-1, this screening criterion appears to identify SSCIS whose failure could *directly* result in a condition adversely affecting public | | **Concern 2:** SSCs classified due only to interaction issues (i.e., seismic 2/1) have been traditionally classified as nonnuclear safety related in the commercial nuclear power industry and placed in augmented QA programs. Criteria 8.2.3 recognizes that the SSC itself does not have to function to meet regulatory requirements but its failure might potentially impact a QL-1 SSC function. This criteria is included in QL-2 to identify the item's potential safety significance, however, following NRC licensing precedent, full application of the QA program is not required. Inclusion of this criteria in QL-2 will require that the item be |

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| safety□ or risk, and as such should be not be categorized as QL-2 but QL-1 SSCIS. Additional clarification is required.<br>**Proposed Path to Resolution:** Provide additional justification for the use of the QL-2 screening criterion found in QAP-2-3 Appendix II, Checklist Item **8.2.3** <u>or</u> revise it to agree with existing DOE terminology <u>and</u> ensure that it is risk-informed and consistent with the proposed final rule and existing regulatory framework, as described in Concern 2. | | appropriately restrained (to prevent interaction); however, QA controls are not required related to the item's safety function).<br><br>NRC has agreed to revisit Concerns 2 and 3 in terms of existing NRC licensing precedent. (See also NRC Item 6(b), Talking Point 4). However, DOE agrees that the classification procedure can be clarified to highlight the item's role in the ISA process. The ISA guide will also provide additional guidance to the analyst on approaches to adequately address the criteria. |
| **Concern 3:** Justification for Screening Criteria<br>The screening criterion identified in QAP-2-3 Appendix II, Checklist Items 8.2.2 may result in mis-categorization. This criterion states: □Does the item provide fire protection, fire suppression, or otherwise protect important to radiological safety or waste isolation functions of QL-1 SSCIS from the hazards of a fire?□ Again the purpose and justification for this screening criterion are not clear. If the failure of this item has the potential to adversely affect the ability or function of a QL-1 SSCIS then according to DOE□s definition of QL-1, this screening criterion appears to identify SSCIS □whose failure could *directly* result in a condition adversely affecting public safety,□ or risk and as such would be not be categorized as QL-2 but QL-1 SSCIS. Additional clarification is required.<br><br>**Proposed Path to Resolution:** Provide additional justification for the use of the QL-2 screening criterion found in QAP-2-3 Appendix II, Checklist Items **8.2.2** <u>or</u> revise it to agree with existing DOE terminology <u>and</u> ensure that this | | **Concern 3:** See concern 2. This approach follows NRC licensing precedent with respect to fire protection systems. |

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| criterion is risk-informed and consistent with the proposed final rule and existing regulatory framework, as described in Concern 3.<br><br>**Concern 4:** Clarification of Terminology<br>The terms in conjunction with and indirect impact as described in QAP-2-3 Appendix II, Checklist Items 8.2.5 and 8.2.6. These screening criteria are not well-defined. As described in QAP-2-3, it appears that DOE could have a situation in which the failure of two QL-2 SSCIS could potentially have the same risk as the failure of a single QL-1 SSCIS. The purpose and justification for this screening criterion are unclear. Again, this screening criterion is more consistent with DOEs definition of QL-1. Further, it would appear that either one or both of these SSCIS would be categorized as QL-1.<br>The use of the three tier approach described in NUREG/CR-6407 and particularly the use of the term indirectlyas the basis for the risk significance categorization of SSCIS appears to have several limitations, as described above. The resulting QL-2 screening criteria seem to be ambiguous in some instances. DOE may want to reconsider the use and application of this approach or provide additional justification to address the stated concerns.<br>**Proposed Path to Resolution:** Provide additional justification for the use of the QL-2 screening criteria found in QAP-2-3 Appendix II, Checklist Item **8.2.5** and **8.2.6** <u>or</u> revise them to agree with existing DOE terminology <u>and</u> ensure that it is risk-informed, as described in Concern 4. | | **Concern 4:** See responses to NRC Items 6(a), Comment 2 and 6(b), Concern 1. Regarding references to NUREG-CR-6407, DOE will clarify that 1) it is cited as an example of a graded approach to items important to safety, and 2) the DOE approach to classification is risk-informed (see response to NRC Item 6(b), Concern 6), which includes some application of deterministic and engineering judgment as well as risk analysis. |

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| **Concern 5:** Uncertainty and sensitivity Analyses<br>DOE uses the ISA process previously discussed in this paper to characterize risk. This ISA process identifies the individual hazards and their associated credible event sequences (frequency greater than $10^{-6}$). The NRC has developed extensive policy and guidance [5, 6, and 13] that identifies one acceptable approach to risk categorization and risk-informed decision making using risk insights from a robust PRA and importance measures. There is no use of importance type measures to characterize an SSCIS contribution to risk, as described in the referenced NRC policy and guidance. It is not clear how DOEs categorization methodology systematically relates individual SSCIS to their contribution to the aggregate risk. Further, it is not clear how or if DOE plans to use sensitivity studies or uncertainty analyses to assess the impact of risk categorization decisions on the aggregate risk. Because DOEs risk thresholds are the same as the performance objective in 63.111, it is necessary to have a clear understanding of the uncertainties associated with the calculation of the likelihood of each of the credible event sequences. Additionally, uncertainty and sensitivity analyses are also important to address some of the potential complexities and the identification and quantification of potential sources of variation that may impact the summation or calculation of the risks associated with each of these event sequences. It does not appear that DOE is performing an a comprehensive or integrated analysis that weighs the SSCIS for one event sequence against those SSCIS in another event sequence or sequences. Without an integrated consequence/frequency assessment, the overall risk | | **Concern 5:** DOE concurs that uncertainty and sensitivity issues must be dealt with appropriately to support a LA. See response to NRC Items 4(a) and 4(b). |

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| implications of alternative design configurations cannot be compared. Additional justification may be required to demonstrate that the chosen analytical approach constitutes a comprehensive identification of hazards as required in proposed  63.112(b).<br>DOE has not performed any uncertainty or sensitivity analyses of the quantification of event sequence frequencies. Uncertainty analyses are important in that they can be used to identify and quantify sources of uncertainty and variability associated with the quantification of event sequence frequencies. It is important to understand the uncertainty and variability associated with the quantification of event sequence frequencies because DOEs risk thresholds are the same as the performance objective in 63.111. It is also necessary to have a clear understanding of the uncertainty and variability associated with DOEs frequency calculations because these frequency calculations are used to determine which frequency category each of the respective event sequences are binned into and accordingly which of the performance objectives apply to that particular event sequence. Uncertainty and sensitivity analyses will also be important in addressing some of the potential complexities associated with DOEs risk calculations for the event sequences and the aggregate or some measure of the overall or aggregate risk. DOE needs to consider the use of uncertainty and sensitivity analyses where applicable or provide justification that explains why these analyses are not necessary.<br><br>**Proposed Path to Resolution:** DOE needs to consider the use of uncertainty and sensitivity analyses where applicable to assess the impact of risk categorization | | |

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| decisions or provide justification that explains why these analyses are not necessary.<br><br>**Concern 6:** Relative Importance<br>DOE is not using estimates of the aggregate risk to determine the contribution of individual event sequences or their associated SSCIS to an overall measure of risk. DOEs approach to risk categorization individually identifies a measure of risk associated with each of the credible event sequences and their associated SSCIS. Additionally, DOE has provided a cursory indication of the estimated aggregate risk for frequency category 1 and frequency category 2 event sequences. A comparison of the individual risk to the overall risk is necessary to ensure that the SSCIS are being categorized consistent with their relative contribution to overall importance to safety or risk significance. The staff is concerned that there is no comprehensive analysis or comparison tying the event sequences together to determine their contribution to the overall risk. The importance of comparing the risks associated with individual or grouped event sequences or their associated SSCIS to the overall risk is discussed in several of NRC policy and guidance documents, including: RG 1.174, RG 1.176, the NRC White Paper,(8) and NUREG-0800 Chapter 19 (even going as far as suggesting the use of importance measures). DOE needs to consider some type of comparison of the individual risk to the overall risk as described above, or provide justification for why they are not doing so.<br><br>It does not appear that DOE is performing an a | | **Concern 6:** DOE believes that it is not necessary to define or apply a measure of aggregate risk for the preclosure operations.<br><br>Proposed 10 CFR 63.111 or 112, or classification of items important to safety does not require the use of an aggregate risk parameter.<br><br>The event sequence frequency Categories 1 and 2 and their respective performance criteria (prop. 10 CFR 63.111) were developed by the NRC as a risk-informed rule. These definitions define regions of compliance and non-compliance on a two dimensional graph of frequency vs. consequence (dose to receptor). The boundary between the two regions is considered to provide a risk metric for preclosure safety analysis.<br><br>Each event sequence end-state (frequency, dose) is represented by a point in the frequency-dose (risk) domain. DOE will demonstrate regulatory compliance by justifying that all credible event sequences are within the frequency-consequence boundaries defined by proposed 10 CFR 63.<br><br>DOE considers the insights gained from event-sequence frequency-dose calculations and sensitivity analyses (e.g., the "take-away" process) coupled with engineering judgment provide a robust risk-informed bases for determining the appropriate classification of SSCs. The "takeaway" process is |

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| comprehensive or integrated analysis that weighs the SSCIS for one event sequence against those SSCIS in another event sequence or sequences. Without an integrated consequence/frequency assessment, the overall risk implications of alternative design configurations cannot be compared.<br>**Proposed Path to Resolution:** Consider performing some type of comprehensive analysis identifying the aggregate risk, relative importance of each of the event sequences, and the relative importance of the SSCIS, as described in Concern 6. | | a sensitivity analysis that provides risk insights similar to the objectives of using the risk importance measures considered in RG 1.174, and which are sufficient for classifying SSCs important to safety.<br><br>The "take-away" process is applied to single SSCs on a sequence-by-sequence basis to assign the appropriate QL-1, QL-2, or QL-3 classification. If a given SSC appears in more than one sequence, its safety classification will be determined on its most limiting case.<br><br>DOE recognizes that RG 1.174 and 1.176 provide a well-crafted philosophy for applying risk-informed decisions in nuclear regulation. To the extent applicable to the preclosure operations, DOE will apply those philosophies. However, the technical approach for applying risk-informed analyses was developed specifically for nuclear power plants.<br><br>The DOE position is that the specific technical approaches presented in RG. 1.174 and 1.176 are not directly applicable for important to safety SSC classification for the following reasons:<br>• The risk-informed approach is used to applied to justify a change in the licensing basis for SSCs that have already been classified as Safety-Related vs Non-Safety-Related..<br>• The two "risk measures" (core-damage frequency and large early release frequency) have no relevance nor counterparts to the repository preclosure operations.<br>• The two "risk measures" (core-damage frequency and large early release frequency) address only one leg of the (frequency X consequence) risk paradigm. |

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| | | • The risk measures ignore other potential sources of radiological events that are not related to protecting the reactor core.<br>• No guidance is provided on the process for evaluating, nor interpreting, the relative risk importance measures from diverse radiological hazards in a radiation facility.<br>• The MGR comprises several discrete operations with specific hazards requiring important to safety SSCs for prevention or mitigation.<br>• The DOE classification process provides insights on the relative risk importance of each prevention or mitigation SSC in each operation (hazard).<br>• Ranking the relative risk importance of all SSCs across operations does not appear to be relevant for the classification<br>• Their "risk importance" relative to an important to safety SSC associated with a different hazard or operation is irrelevant. |
| **Concern 7:** Expert Panel<br>DOEs classification analyses and subsequent risk categorization may benefit from the use of a multi-disciplinary review group similar to the expert panel described in RG 1.176. DOEs proposed approach to risk categorization relies heavily on the screening criteria identified in procedure QAP-2-3 and the associated classification analyses. Specifically, DOE is relying heavily on those individuals performing and reviewing these classification analyses. NRC guidance recommends the use | | **Concern 7:** DOE notes that the ISA preparation, SSC classification, and specification of QA controls will involve a multidisciplinary team from safety analysis, licensing, design, criticality, fire safety, quality assurance, etc. Further, all documents will be subjected to multidisciplinary review by others. |

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| of a multi-disciplinary review group of technical and professional personnel, referred to as the "expert panel," to support risk-informed decision-making process. This expert panel performs an integrated assessment of quantitative and qualitative risk insights to determine the safety significance ranking of SSCIS.<br>**Proposed Path to Resolution:** Consider the use of an expert panel (multi-disciplinary) to support the safety significance ranking of SSCIS, as described in Issue 7. | | |
| **Points Requiring Additional Clarification**<br><br>In addition to the concerns identified above the staff have identified several point that require further clarification: | | |
| 1. Additional explanation and examples are required to show how DOE proposes to integrate in a transparent manner, the use of the equation $\sum F_i D_i + D_e$ (RSS Figure 8-1), the "take away" analyses (RSS Section 8), the screening criteria in procedure QAP-2-3, and the classification analyses. | | 1. See presentation on Item 6a, Comment 2. Project documentation will be updated to provide consistency, identify relationships, and clarify DOE's approach. |
| 2. Clarification is required as to how DOE is proposing to include multiple category 1 event sequences in the proposed categorization process. | | 2. See response to NRC Item 6(a), Comment 3. |
| 3. Clarification is required as to whether the routine releases from surface and subsurface facilities during normal operational are factored into the equation $\sum F_i D_i + D_e$ (RSS Figure 8-1). | | 3. Normal releases are included in classification of SSCs for Category 1 design basis event sequences. See response to NRC Item 6(a), Comment 3. |

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| 4. Additional explanation is required regarding the establishment and use of the bounding dose term ($D_e$) in the Q-list and categorization process. | | 4. See response to NRC Item 6(a), Comment 3. |
| TALKING POINTS ON QL CATEGORIZATION ISSUE FOR NRC/DOE JULY TE (TELECON JUNE 28, 2001)<br><br>1. DOE has reviewed the Acceptance Criteria (presented in the draft staff position) that the NRC staff intends to use in the review of DOE's proposed approach to categorization. DOE has not identified any concerns with the acceptance criteria. | | 1. DOE has not identified any concerns with the acceptance criteria. |
| 2. DOE agrees that several of the QL-2 screening criteria were vague and that the basis for, and application of, the associated screening criteria were not transparent. To address this DOE agrees to provide a detailed presentation during the July TE. | | 2. See responses to NRC Items 6(a), Comment 2 as well as responses to NRC Items 6 (a) and 6(b) related to screening criteria. |
| 3. DOE agrees to present additional information and examples (at the July TE) clarifying how it proposes to integrate in a transparent manner, the use of the equation □$F_iD_i + D_e$ (RSS Figure 8-1), the □take away□ analyses (RSS Section 8), the screening criteria in procedure QAP-2-3, and the classification analyses for category 1 event sequences. | | 3. See responses to NRC Items 6(a), Comment 2 as well as responses to NRC Items 6 (a) and 6(b) related to screening criteria.<br><br>4. See responses to NRC Item 6(b), Concerns 1, 4, 5, and 7. |

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| 4. DOE agrees to address concerns 1, 4, 5, and 7 (draft staff position) during the scheduled presentations at the July TE.<br><br>NRC agrees to remove concerns 2 and 3 based on discussions with DOE, discussions with NRR, and further review of related guidance. Concern 2 is being removed on the basis that DOE is following guidance contained primarily in RG 1.89, *Fire Protection for Operating Nuclear Power* Plants (April 2001). Concern 3 is being removed on the basis that DOE is following guidance contained primarily in RG 1.29, *Seismic Design Classification* (Revision 3, September 1978). DOE agrees to provide detailed examples outlining how screening criteria 8.2.2 and 8.2.3 are being implemented.<br><br>NRC agrees to consider removing concern 6 (draft staff position) based on the outcome of presentations during the TE.<br><br>The draft staff position will be revised to reflect the above changes as appropriate. | | Examples are provided in the presentation associated with NRC Item 6(a), Comment 2. |
| 5. NRC agrees to consider DOE's criteria of using 100 mrem and 25 mrem as basis for categorizing QL-1 and QL-2 respectively. DOE agrees to provide the technical basis during the presentations at the July TE. | | 5. See presentation associated with NRC Item 6(a), Comment 2. DOE is using 100 mrem for Quality Level 1 for Category 1 events. DOE is using 25 mrem based on proposed 10 CFR 63 performance objective (which is based on 10 CFR 20 and is conservative as industry precedent suggests that 500 mrem is an appropriate discriminator for safety related and nonnuclear safety related).<br><br>6. The DOE classification procedure includes criteria for |

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| 6. It was not clear during the last two telecons if DOE intends to categorize SSCs important to waste isolation (ITWI). DOE agrees to provide additional information clarifying if and how ITWI SSCs will be categorized; including, detailed examples outlining how the screening criteria will be implemented. | | classification of SSCs important to waste isolation as QL-1, QL-2, QL-3, or conventional quality.<br><br>Criteria, steps, and conceptual design examples are summarized below.<br><br>• TSPA is key to identifying Quality Level 1, Important to Waste Isolation structures, systems, and components<br>• If item is credited in TSPA to meet performance objectives, item is classified as Quality Level 1<br>• Preserving initial conditions will drive Quality Level 2<br>• Monitoring used to demonstrate site is performing within licensing specifications will drive Quality Level 3<br>• Classification procedure does not differentiate engineered items as related to principal factors or defense in depth (both classified as Quality Level 1)<br><br>Conceptual Design Examples<br><br><u>Emplacement Drifts (Subsurface Facility System)</u><br>• Quality Level 1<br>• Constructed within natural barrier, but do not form part of the natural barrier<br>• Sizing and placement of emplacement drifts invokes the waste isolation requirements of proposed 10CFR63.113(c) and 10CFR63.113(d)<br>• Directly credited in performance assessment to demonstrate ability of geologic repository to meet the proposed 10CFR63.113 dose requirements<br><u>Uncanistered SNF Disposal Container</u><br>• Quality Level 1 |

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| | | • Uncanistered SNF disposal container is part of the engineered barrier<br>• Performs waste isolation function<br>• Credited in performance assessments to demonstrate the ability of repository to limit annual public dose to regulatory limits in first 10,000 years<br><u>Drip Shields (Emplacement Drift System)</u><br>• Quality Level 1<br>• Part of engineered barriers<br>• Credited in performance assessments to demonstrate the ability of repository to limit annual public dose to within regulatory limits in first 10,000 years<br>• Protect waste package from rockfall<br>• Divert water, if present, around the waste package<br><u>Emplacement Drift Monitoring System (Performance Confirmation Emplacement Drift Monitoring System)</u><br>• Quality Level 3<br>• Provides information for analysis and information to conduct field and laboratory experiments<br>• May be used to assess off normal events that occur within the emplacement drift<br>• Monitors variables to verify that operating conditions are within licensing specifications<br><u>Subsurface Closure and Seal System</u><br>• Conventional Quality<br>• System is a barrier to limit air flow, water flow and human intrusion<br>• System is not relied upon for meeting postclosure performance objectives<br>• Amount of water entering boreholes small compared to fracture pathways in Yucca Mountain |

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| | | • Boreholes do not intersect drifts |

| NRC Item 7(a)-Level of Design Details | | |
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| 1. Differentiated Approach to Providing Information in the License Application-<br><br>Proposed 10 CFR 63.21 identifies the required contents of the LA. NRC regulations require the applicant to provide information as complete as possible in light of information available at time of docketing the license application for construction authorization. The applicant must update its application in a timely manner to allow commission review before issuance of a license. Proposed 10 CFR 63.24(b) permits DOE to update the LA with additional information required for submitting the LA to receive and possess HLW. The information DOE is required to provide in the LA (§63.21) for the construction authorization decision must be sufficient to demonstrate compliance with the requirements of proposed §63.31 Construction Authorization. DOE is required to perform preclosure safety assessment (PCSA similar to Integrated Safety Assessment) at the appropriate level of rigor to identify the structures, systems, and components important to safety and waste isolation (§63.112). In the absence of final design, DOE's PCSA needs to be conservative in identifying the SSCs important to safety and waste isolation. In general, the level of detail of the information in the LA should enable the NRC staff to determine if there is reasonable assurance that DOE has demonstrated compliance with the applicable regulations including receiving and possessing and disposing HLW. Demonstrating compliance requires providing sufficient technical basis to allow NRC to make a finding of reasonable assurance that the types and amounts of radioactive materials | Licensing<br>Marty Bryan/Jerry Self | DOE understands the requirement to submit information for CA that is sufficient for the NRC to make a safety determination. Based on the NRC's concurrence the project will move forward in providing information that is differentiated relative to information availability and graded according to is quality level. A differentiated approach refers to the fact that some information (primarily operational related activities) will not be available at the time of the License Application for Construction Authorization (LA (CA)) and additional details will be provided at the time of the License to Receive and Possess Waste. The design related information uses a graded approach that recognizes that the type and amount of information based provided is based on its safety significance. At the time of the License Application to Receive and Possess Waste the information will be updated, as appropriate, to reflect the most current information available.<br><br>DOE is developing an LA Guidance Database (based on guidance in the Technical Guidance Document (TGD), Rev. 1). The guidance in the TGD is based on the requirements of the proposed 10 CFR 63, nuclear industry experience and NRC licensing precedent. This guidance is presently being used to identify the LA Products needed to support LA development. This database will include the guidance and associated project products and becomes a transparent identification of what is planned for the LA. This is a living database and will be updated when the final 10 CFR 63 and the Yucca Mountain Review Plan become available. |

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| described in the application can be received, possessed, and disposed of in the geologic repository operations area of the proposed design without unreasonable risk to the health and safety of the public.<br><br>The differentiated approach proposed by DOE may result in the LA presenting all the information required for the NRC staff to make a determination on compliance with Proposed **10 CFR 63.31** for construction authorization if DOE provides all reasonably available information and updates as necessary, and the updated LA should contain the information needed to make a determination on compliance with standards for issuance of a license to receive and possess HLW as per proposed **10 CFR 63.41. The differentiated approach is acceptable provided DOE submits information at the CA and LRPW that is sufficient for NRC to make a 'safety determination'.**<br><br>2. Differentiated Approach in the Level of Detail of Design Information in License Application-<br>NRC policy permits the quality assurance program to control activities affecting the quality of the identified structures, systems, and components, to an extent commensurate with their importance to safety. Provision of controlling QA activities commensurate with their importance to safety permits graded QA approach. A properly conducted PCSA at the appropriate level of rigor identifies the SSCIS, and DOE proposes to implement graded QA to SSCIS. Similarly, a properly conducted post closure performance assessment (PA) identifies SSCIS and their performance requirements in the post closure time. DOE proposes to | | Based on the LA Guidance Database and clarification provided by the NRC, DOE plans to provide the following graded design information, as appropriate, for the different Quality Level SSCs.<br><br>For Quality Level 1 SSCs, the following information is what the NRC is expecting to see in the LA on an SSC basis:<br><br>• Applicable codes and standards<br>• Design criteria<br>• Regulatory design bases<br>  – Combination of system functions and performance parameters<br>• General system description<br>• Information on dimensions<br>• Material properties<br>• Specifications<br>• Analytical and design methods used in design<br>• Piping and instrumentation diagrams<br>• Electrical one-line diagrams<br>• General arrangement drawings<br>• Handling diagrams<br><br>For Quality Level 1 SSCs, the following information is planned to be included in the LA on an SSC bases. This information is based on the requirements already captured in the LA guidance and additional information as requested by the NRC:<br><br>• Applicable codes and standards<br>• Design criteria |

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<tr><th colspan="3">NRC Item 7(a)-Level of Design Details</th></tr>
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<tr><td>categorize, commensurate with their importance to safety, SSCIS into three categories (QL-I, QL-II, and QL-III) for QA implementation. DOE proposal on the criteria for categorizing SSCIS is under consideration by the staff and requires further discussions with DOE. QA categorization is not the subject of this paper, and therefore, the criteria/process for categorization of SSCIS is not addressed here. Acceptable list of SSCIS and QA categorization methodology are assumed in this document, and further discussion only relates to DOE's proposal for "Differentiated Approach in the Level of Detail of Design Information in LA."<br><br>DOE has extended its QA categorization of SSCIS into a corresponding classification for the purpose of defining the level of detail of design information to be provided in the LA. In addition, because the Repository Surface Facility design has licensing precedent, the level of detail of design DOE plans to provide for these SSCs in the LA is less than that required for the SSCs that do not have licensing precedent.<br><br>Proposed 10 CFR 63.21 (C) identifies the kinds of design information to be provided in the LA for SSCIS. NRC regulations don't specifically address the concept of level of detail of design information in the LA for an item to be commensurate with the safe significance of the particular item. The information in the LA for all SSCIS should be sufficient for NRC to make a finding of reasonable assurance on DOE's demonstration of compliance with regulations.</td><td></td><td>• Design bases<br>  – Combination of system functions and performance parameters<br>• General system description<br>• Materials of construction<br>• Analytical and design methods used in design<br>• Provide necessary drawings based on SSC matrix<br>• Discussion on system function to prevent, limit, or mitigate a DBE<br><br>For Quality Level 2 SSCs, the following information is what the NRC is expecting to see in the LA on an SSC basis:<br><br>• Applicable codes and standards<br>• Design criteria<br>• Regulatory design bases<br>• Combination of system functions and performance parameters<br>• General system description<br>• General arrangement drawings<br><br>For Quality Level 2 SSCs, the following information is planned to be included in the LA on an SSC bases. This information is based on the requirements already captured in the LA guidance and additional information as requested by the NRC:<br><br>• Applicable codes and standards<br>• Design criteria<br>• Design bases<br>  – Combination of system functions and performance</td></tr>
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| In Nuclear Power Plant licensing, NRC has accepted lesser level of detail of design information in the preliminary safety analysis report (PSAR) because the information was not available at that stage. However, the final safety analysis report (FSAR) provided all the information required for the staff to determine compliance with the regulations. Both PSAR and FSAR LAs contained sufficient information to enable the staff to make determination on compliance with the regulations. Also, NRC was able to closely monitor the design and construction activities of the licensee between PSAR and FSAR stages.<br><br>**Although there is no regulatory precedent to grade the level of detail of design information in LA, the staff agrees in principle with DOE that level of detail of design information for SSCIS in the La can be tailored commensurate with their importance to safety as long as the information is sufficient for the staff to make a finding on DOE's demonstration of compliance with the regulations.** A properly conducted PCSA and PA, and a transparent process for categorizing the SSCIS that takes into consideration the uncertainties in the underlying information are essential to this process and is a key assumption in this approach. **In the absence of specific technical/design criteria in the performance-based proposed 10 CFR Part 63 regulations, the staff needs the following information in the LA in addition to those proposed by DOE to reach a conclusion of reasonable assurance on DOE's demonstration of safety:**<br>Additional Information | | parameters<br>• General system description<br>• Materials of construction<br>• Provide necessary drawings based on SSC matrix<br><br>For Quality Level 3 SSCs, the following information is what the NRC is expecting to see in the LA on an SSC basis:<br><br>• Applicable codes and standards<br>• Design criteria<br>• Regulatory design bases<br>– Combination of system functions and performance parameters<br>• General system description<br><br>For Quality Level 3 SSCs, the following information is planned to be included in the LA on an SSC bases. This information is based on the requirements already captured in the LA guidance and additional information as requested by the NRC:<br><br>• Applicable codes and standards<br>• Design criteria<br>• Design bases<br>– Combination of system functions and performance parameters<br>• General system description<br>• Materials of construction<br>• Provide necessary drawings based on SSC matrix<br><br>For Conventional Quality Level SSCs, the following |

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| For QL-1 SSCIS-Information on dimensions, material properties, specification, and analytical and design methods used in the design.<br>For QL-2 SSCIS-Regulatory Design Bases, General Arrangement Drawings.<br>For QL-3 SSCIS-Applicable Codes and Standards, Regulatory Design Bases.<br>**The level of design information proposed by DOE together with the above identified additional information are expected to be sufficient, for the initial or first stage of LA review for CA, for the staff to make finding of reasonable assurance that DOE has demonstrated compliance with the applicable proposed 10 CFR 63 regulations.**<br>**Depending on the complexity of the SSCIS, on an as-needed basis, the staff may request additional information to enable a review of the LA.** | | information is what the NRC is expecting to see in the LA on an SSC basis:<br><br>General description that is sufficient to demonstrate the conventional quality classification<br><br>For Conventional Quality Level SSCs, the following information is planned to be included in the LA on an SSC bases. This information is based on the requirements already captured in the LA guidance:<br><br>• General description that is sufficient to demonstrate the conventional quality classification<br>• Provide necessary drawings based on SSC matrix |

| NRC Item 7(e.1)-Waste Package Drop Analysis | | |
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| A. WP Drop Finite Element Models<br><br>1. The DOE needs to demonstrate that the mesh discretization of the finite element models used to simulate WP drop events are sufficient to provide the level of results resolution needed to assess potential failure.<br><br>It is well known that the results of numerical models based on the finite element method are dependent on the mesh discretization. Because the results obtained from finite element models oftentimes represents the entirety of DOE's safety case, which appears to be the situation for the structural response of the WP to unintentional drops during handling operations, DOE needs to demonstrate that the mesh discretizations are adequate. For instance, based on a review of the *Corner Drop of 21-PWR Waste Packages* (CRWMS M&O, 2000a), it is not clear how many elements are used through the thickness of the inner and outer barrier components of the model near the impact zone. As a result, it is difficult to determine whether these models have been constructed satisfactorily.<br><br>2. DOE needs to provide documentation of all boundary conditions used in the WP drop finite element models. The documentation must also include the technical basis and/or rationale for the boundary conditions.<br><br>Based on WP drop analysis reports reviewed by the | Waste Package<br>Thomas Doering | A. WP Drop Finite Element Models<br><br>1. Benchmarking of the finite element analysis (FEA) code (LS-DYNA) against pour canister drop experiments have been performed and show acceptable fidelity with test results. (*Drop Calculation of HLW Canister and Pu Can-in-Canister*, CAL-EBS-ME-000015 (work in progress))<br><br>Mesh selection is an implicit part of testing performed for code qualification. (*Validation Test Report* (VTR), SDN 10364-VTR-5.6.2-00 (LS-DYNA may be run either as a module of ANSYS or as a stand-alone code.))<br><br>Capability to select appropriate mesh density has been demonstrated through comparison to both test results and numerical evaluations.<br><br>2. Many FEA codes require the use of specialized techniques (such as the use of arbitrary spring elements) to properly assess rigid body motion, such as that between the shells of the waste package.<br><br>LS-DYNA has the capability to handle unconstrained rigid body motions without modification of boundary conditions (i.e., addition of spring elements between surfaces to provide |

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| NRC to date, it is not clear if the inner and outer barriers of the WP are □tied□ or allowed to □slide□ at their interfaces. Moreover, springs are often used to provide numerical stability by preventing unconstrained rigid body motion in finite element models where interactions between unconnected structural components are being simulated, e.g., the inner and outer barriers of the WP. In general, there is very little discussion provided in the DOE reports pertaining to the details of the finite element model construction. For the sake of traceability and transparency, DOE needs to sufficiently document the technical basis and/or rationale for the finite element model boundary conditions and other relevant aspects of its construction.<br><br>B. WP Drop Analysis Results<br><br>1. Has the Applicable Failure Criterion Been Used?<br><br>It is not clear at this time as to whether the integrity of the WP or SNF will be the limiting or controlling factor in establishing allowable fall heights. Has damage of the WP contents been considered when assessing the results of the WP drop analysis results and establishing allowable drop heights? | | constraints)(Livermore software Technology Corporation, *LS-DYNA Theoretical Manual*, May 1998).<br><br>B. WP Drop Analysis Results<br><br>1. The design objective for the waste package is to ensure no breach for pre-closure design-basis events.<br><br>While the outer corrosion-resistant barrier of the waste package is predicted to fail in the Corner Drop of 21-PWR Waste Packages calculation, the inner shell, which is the primary structural member, was computed to have large margin to failure (*Corner Drop of 21-PWR Waste Packages*, CAL-UDC-ME-000008).<br><br>In the event of a drop, an assessment would be made as to whether the waste form must be re-packaged.<br><br>The waste form serves as a secondary barrier to release of |

**NRC Item 7(e.1)-Waste Package Drop Analysis**

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| 2. Has the Failure Criterion Been Evaluated Properly?<br><br>From the perspective of failure of the WP controlling allowable drop heights, DOE is using the failure criterion advocated by the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code (American Society of Mechanical Engineers, 1998), i.e., the calculated stress intensity (as defined by the ASME B&PV Code) must be greater than 90 percent of the ultimate tensile strength of the material for failure to occur. The ASME B&PV Code stress intensity is defined as the difference between the first and third principal stresses, i.e., the diameter of the largest Mohr Circle for a given state of stress. Therefore, a calculated stress state must have a Mohr□s circle diameter greater than 90 percent of the ultimate tensile strength of the material for failure to occur.<br><br>In the *Corner Drop of 21-PWR Waste Packages* calculation report (CRWMS M&O, 2000a) the finite element analysis results were given in terms of the maximum shear stress (no discussion was given as to whether the WP failed or not). For example, the maximum shear stresses for the inner and outer barriers of the 21-PWR WP when dropped from 2.4 m with a material temperature of 400 □F was given as | | radionuclides and will be evaluated as deemed appropriate in the future.<br><br>2. In *Corner Drop of 21-PWR Waste Packages*, stresses in the waste package outer shell exceed the allowable, indicating that there may be a breach of the outer shell; however, stresses in the inner shell are below the breach criterion. As long as one of the shells remains intact, there is no release of radionuclides, thus, the results are acceptable (also see discussion for NRC Item 7(e.1) B1). (*Corner Drop of 21-PWR Waste Packages*, CAL-UDC-ME-000008)<br><br>In the event of a drop, an assessment would be made as to whether the waste form must be re-packaged. |

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| 215 and 452 MPa, respectively. Because the maximum shear stress is equal to the <u>radius</u> of the largest Mohr circle, it must be multiplied by a factor of two before it can be compared to the allowable ultimate tensile strength. For the inner barrier, the maximum calculated stress intensity is approximately 430 MPa (2 □ 215 MPa). For the outer barrier, the maximum calculated stress intensity is approximately 904 MPa (2 □ 452 MPa). In the case of the outer barrier, the calculated stress intensity is clearly above the allowable, i.e., 846 MPa (0.9 □ 940 MPa). Note that the allowable stress intensity for the inner barrier is 567 MPa (0.9 □ 630 MPa).<br><br>It is not clear if the conclusion presented in the *Yucca Mountain Science and Engineering Report* (U.S. DOE, 2001) indicating that the Naval SNF Long waste package can survive a vertical drop design basis event is affected by the observations made above. | | |
| C. Design Basis WP Drop Scenarios<br><br>1. What impact orientations are considered? What is the technical basis for the impact orientations considered? Not all impact scenarios described in the WP system description documents appear to have been evaluated.<br><br>Impact orientations that may be the worst case for the WP may not represent the worst case if the structural integrity of the MPC or SNF is considered. | | C. Design Basis WP Drop Scenarios<br><br>1. A wide variety of design-basis dynamic events are considered for the waste package in the *Preclosure Design Basis Events Related to Waste Packages* analysis, including vertical drop, horizontal drop, horizontal drop with emplacement pallet, tip over, and transporter runaway (*Preclosure Design Basis events Related to Waste Packages,* ANL-MGR-MD-000012).<br><br>As a part of the normal design process, design-basis dynamic |

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| | | events will be re-evaluated as the design for both the surface facility and sub-surface facility mature.<br><br>Credible dynamic events will be identified and assessed. The design of the waste package and both the surface and sub-surface facilities will be adjusted to accommodate such challenges to waste package integrity. |

| NRC Item 7(e.2)-Waste Package Welding/Fabrication Issues | | |
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| Welding and fabrication processes are considered to be important for both preclosure and postclosure performance. The effects of welding and fabrication processes on the long-term mechanical integrity and corrosion resistance of the emplaced waste packages have been previously identified as postclosure issues that may affect both the waste package and overall repository performance. The principal characteristics of the waste package, which include fabrication and welding, is identified as a preclosure issue. Principal characteristics will be influenced by the waste package design, welding parameters, repair methods, post weld processing methods, and non destructive evaluation (NDE) techniques. Fabrication qualification tests as well as pre- and post-fabrication inspection methods will be critical to the evaluation of the waste package principal characteristics.<br><br>1. The DOE should provide the technical basis for compositional restrictions used for the procurement and verification of the materials used to construct the WP. If the compositional specification defined in ASTM B-575 is to be used, DOE should demonstrate that the compositional variations allowed for Alloy 22 will result in consistent WP performance.<br><br>The chemical composition specifications for Alloy 22 include variations for Cr, Mo, Fe, and W. Altering the compositions of these alloying elements within the range of the chemical composition tolerances may adversely | Waste Package<br>Thomas Doering | A.1 The filler material and base material used for constructing the disposal container of Alloy 22 conforms to the ASME Code and is documented in the FY-00 and FY-01 Waste Package Operations Fabrication Process Report. This will also be addressed in the Closure Weld Report issued in September.<br><br>The samples being tested at LLNL are made from a number of heats of material. Therefore, any data on corrosion rates will take into consideration this variation. (*Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003) |

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| affect the thermal stability and promote the precipitation of intermetallic phases that can decrease the corrosion resistance and impact strength of the alloy. | | |
| 2. Alteration of the microstructure as a result of alloy processing and fabrication of the WP may adversely affect WP performance. Numerous heats of Alloy 22 (approximately 1 heat per waste package) will be required for the proposed HLW repository. Variations in the microstructure of the alloy cannot be determined from chemical analyses. | | A.2 Waste Package Project weld flaw distribution study to be completed in this calendar year.<br><br>The filler material used for the welding of Alloy 22 and the base material conforms to the ASME Code and is documented in the FY-00 and FY-01 Waste Package Operations Fabrication Process Report. This will also be addressed in the Closure Weld Report issued in September 2001.<br><br>The samples being tested at LLNL are made from a number of heats of material. Therefore, any data on corrosion rates will take into consideration this variation. (*Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003) |
| 3. The size, distribution, and frequency of defects in the waste package are recognized as parameters that must be considered in the analyses of early waste package failures. These defects are also important to the mechanical integrity and long term performance of the WP. In the TSPA-SR the earliest waste package failure, which occurs approximately 12,000 years after repository closure, is attributed to the presence of initial defects. In the more recent SSPA, improper heat treatment is considered to lead to WP failure within the 10,000 year regulation period (i.e., one WP failure in less than 2,000 years). Because the size, distribution, and frequency of defects are principal characteristic of the waste package, the DOE should demonstrate the ability of the proposed inspection methods to adequately detect defects in the plate | | A.3 The plate used to construct the waste packages will be ultrasonically inspected per the ASME Code prior to use in fabrication (§6.2.5, *Waste Package Operations Fabrication Process Report*).<br><br>In addition, there will be visual and dimensional examination of the plate material per the ASME Code (§6.2.5, *Waste Package Operations Fabrication Process Report)*.<br><br>The FY-01 development program includes a study to identify the minimum flaw size that can be detected in Alloy 22 material of this design thickness.<br><br>Specifics regarding the testing of annealed cylinders are under development (§6.2.5, *Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003 REV 02, (to be |

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| material. The proposed inspection method should be adequate to inspect the final fabricated container prior to waste loading, including postweld annealing. | | completed in September, 2001). |
| B. Welding Procedures<br><br>1. Contamination Controls<br><br>The process for inspection prior to welding to assure that the surfaces are free of potentially adverse contaminants should be provided. Improperly cleaned and contaminated waste package surfaces or filler metal could lead to higher distributions, sizes, and frequencies of weld defects. Because of the nature of the closure weld operation, inspection of the waste package surfaces may be limited by the remote inspection operation. In the *Analysis of Mechanisms for Early Waste Package Failure* (CRWMS M&O, 2000b) AMR it is assumed that an incorrect cleaning process cannot leave a residue that will adversely affect the performance characteristics of the weld. | | B.1 Waste package weld area cleaning is addressed in §6.8.5 of the *Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003 for both FY-00 and the FY-01. The section states that "the surfaces or parts to be welded shall be visually clean and free of slag, scale, rust, oil, grease, and other deleterious foreign materials for a distance of at least one inch from the weld joint. Chemical cleaning agents for use on stainless steel or nickel alloy shall be approved by the purchaser before use" and will be chosen to leave no residue.<br><br>Similar words will address this issue in the Closure Weld document for FY-01 for the closure weld, except that the inspection will be remote with optics. This is an acceptable method of inspection in lieu of direct examination.<br><br>Adherence to these requirements will be provided by operational procedures. |
| 2. Filler Metal Selection<br><br>Filler metal composition may also contribute to thermal instability of Alloy 22 in the weld regions. As previously indicated (Section 7.e.2. A.1), variations in the concentration may promote the stabilization of secondary phases, and decrease both the localized corrosion resistance and impact strength of the alloy in the weld region. | | B.2 The filler material used for the welding of Alloy 22 conforms to Section II, Part C of the ASME Code and is documented in the FY-00 and FY-01 *Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003, §6.3. This will also be addressed in the Closure Weld Report to be issued in September 2001.<br><br>The samples being tested at LLNL are made from a number of heats of this wire. Therefore, any data on corrosion rates will take into consideration this variation.<br><br>The code does not require that this material be impact tested |

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| | | because it is not prone to brittleness.<br><br>Completion of Container Life and Source Term Agreements 2.4 and 2.5 regarding waste package fabrication and welding. |
| 3. Welding Method (speed, heat, etc)<br><br>Demonstration that the parameters specified do not adversely affect the quality of the weld should be provided. Welding speed and specific heat input may affect the quality of the weld by increasing the frequency of defects and altering the thermal stability of the alloy. These parameters are expected to be specified during the weld procedure development. | | B.3 Item should be considered as resolved for the following reasons:<br><br>As a part of the standard fabrication develop process, the effect of process parameters on material performance will be developed.<br><br>Testing of the FY-00 mock up will be conducted after the induction annealing study is complete to ensure that performance is not adversely effected. Results will be documented in a future revision of the *Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003. |
| 4. Environmental Restrictions<br><br>It is assumed that DOE will use an inert shielding gas during the welding of the WP. The complete range of environmental restrictions has not been identified. | | B.4 The purity of the argon and all other critical parameters will be provided in the welding specification.<br><br>The gas used for shielding is argon and is no different than normal manufacturing operations that are conducted daily in numerous manufacturing facilities.<br><br>The Surface Facility design group is addressing ventilation in the hot cell. |
| 5. Weld Qualification Tests<br><br>The ability of weld qualification tests to detect weld defects and poorly performing welds should be demonstrated. Mechanical tests may be used to | | B.5 Weld qualification tests required to be conducted by the fabricator prior to any welding do not normally include corrosion testing.<br><br>However, corrosion tests are being conduct by LLNL at |

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| demonstrate the physical properties of the weld and the heat affected zone in the base alloy. These tests may not verify the integrity of the WP in terms of corrosion resistance. Specific tests to show the fabrication procedure does not alter the mechanical properties or corrosion resistance should be utilized in the weld procedure qualification. | | present with weld samples in the annealed and non-annealed condition to study the basic phenomenon. Further tests will be conducted on the FY-00 mock up after the annealing study that is currently underway and will examine the effect of welding and annealing on Alloy 22 (Waste package Operations Fabrication Process Report, TDR-EBS-ND-000003). |
| C. Weld Flaws and Defects.<br>1. Proposed NDE methods can detect flaws and defects at the requisite level of resolution<br><br>The assumption that ultrasonic testing (UT) inspections will be as reliable for Alloy 22 as it is for stainless steel welds will need to be verified. This verification should take into consideration the specific closure weld joint designs, weld dimensions, and materials and the fact that the UT inspections will be accomplished remotely. | | C.1 Waste Package Fabrication has been performing Ultrasonic tests on Alloy 22 material since approximately 1997. These tests are documented in the annual fabrication reports. These tests have been on weld joints duplicating the final closure weld joint design.<br><br>In addition, tests are being conducted to determine the minimum flaw detection and will be reported on in the FY-01 Closure Weld document.<br><br>A flaw distribution study is under way. This will use numerous NDE techniques to detect the flaws, one of them being UT. |
| 2. Prescribed surface cleaning and finish are compatible with the proposed surface NDE method.<br><br>The surface finish of the waste package after all fabrication steps have been completed (e.g. welding, post weld treatments, and machining) must comply with the requirements necessary to perform NDE using the methods specified. Improper surface finish may mask some defects. For example, a rough surface finish may reduce the ability of surface sensitive NDE methods (ultrasonic testing or liquid penetrant testing) to detect surface breaking defects such as cracks formed during weld solidification. If undetected, these | | C.2 The acceptable surface finishes for NDE are normally found in the NDE procedures that meet the ASME Code. Paragraph 1.2.1.10 of the *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document*, SDD-UDC-SE-000001and §4.1 of the *Waste Package Operations Fabrication Process Report*, TDR-EBS-ND-000003 define the surface finish as 250 $\mu$in (6.35 $\mu$m). |

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| defects may act as initiation points for early postclosure failure mechanisms.<br><br>3. Weld joint design does not detrimentally contribute to the weld flaws and defects rate.<br><br>Prediction of weld defects has been estimated for Alloy 22 using the RR-PRODIGAL weld simulation code and parameters used in the in-service inspection of stainless steel piping. As indicated in the *Analysis of Mechanisms for Early Waste Package Failure* (CRWMS M&O, 2000b) AMR, it is assumed that the information on the weld flaw density for gas tungsten arc welded (GTAW) stainless steels can be applied to GTAW Alloy 22 even though welding of Alloy 22 is recognized to be more difficult than stainless steel. Demonstration of the technical basis for the assumption, considering the geometry of the weld and the composition of the material, should be provided.<br><br>D. Post Weld Treatments<br><br>1. Proposed postweld treatments must not degrade mechanical or corrosion characteristics of the base metal or the weld filler metal.<br><br>The present waste package fabrication method specifies that laser peening will be used for the inner Alloy 22 lid and induction annealing will be used for the outer Alloy 22 lid. Demonstration of the laser peening method as a means to mitigate tensile stresses in the weld regions without detrimental effects to either the mechanical properties or microstructure in the weld and adjacent base metal has not been provided. Thermal gradients during local induction | | C.3 After welding three mock up joints of approximately sixty feet that duplicate the current design of the waste package, DOE does not find that Alloy 22 is more difficult than Stainless Steel to weld given the correct welding parameters.<br><br>The flaw distribution study is scheduled to weld another 200 feet of weld duplicating the weld joint design. This information will be used to substantiate the existing data on weld flaws.<br><br>The use of the Rolls-Royce Prodigal information will be phased out as applicable data becomes available.<br><br>Completion of Container Life and Source Term Agreement 2.6<br><br>D.1 Studies of the laser peening process are ongoing at LLNL.<br><br>The induction annealing tests are ongoing and tests will be also conducted on the FY-00 mock up. Results will be documented in a future revision of the Waste Package Operations Fabrication Process Report and the Waste Package Closure Weld Report that will be issued in September of FY-01.<br><br>Completion of Container Life and Source Term Agreements 2.4 and 2.5 regarding waste package fabrication and welding. |

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| annealing, proposed for the outer closure lid of the waste package, may result in microstructural variations that reduce impact strength and corrosion resistance. In the *Analyses of Early Failure Mechanisms* (CRWMS M&O, 2000b) AMR it is suggested that independent tests will be conducted to verify that the thermal treatment was performed correctly. At present, the type of tests that will be used have not been identified and the ability of these test methods to detect improper thermal treatment of Alloy 22 has not been demonstrated. | | |
| E. Postweld Repair<br>1. Proposed remediation procedures must not degrade mechanical or corrosion characteristics of the base metal or the weld filler metal.<br><br>The details of waste package remediation are still under development. Repair of welding defects and the process of removing a closure weld and then rewelding the WP will result in increased thermal processing that may alter the mechanical properties and corrosion resistance of the WPs. | | E.1 Repair cycles at the fabricator will be limited and will be discussed in the FY-01 Fabrication document.<br><br>Repairs in the hot cell closure weld will be handled by feed back processes that identify the defect at the time of initiation. This will make the repair minor and less intrusive. If there is an occasion where the defect is major, it would normally be handled by removing the lid and then the fuel and re-packaging.<br><br>Any repairs will be followed by appropriate stress mitigation . |

| NRC Item 7(e.3)-Differential Thermal Expansion Issues | | |
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| A. What provisions have been made for thermal expansion in the design of the gantry crane rails?<br><br>Any thermal expansion joints used to prevent the gantry crane rails from deforming beyond allowable tolerances or buckling under thermal load must be capable of supporting the gantry crane without causing derailment.<br><br>B. What provisions have been made for thermal expansion of the invert structural frame beams attached to the drift wall?<br><br>Excessive differential thermal expansion between the invert structural frame beams and the drift wall may cause damage to the beams and/or drift wall. For example, damage to the drift wall may affect the stability of the invert structural framework itself, i.e., it is no longer adequately anchored to the drift wall. Or, unwanted drift side wall instabilities may arise from coalescing fractures originating from the invert beam anchor points due to localized differential thermal stresses between the invert beams and drift wall. | Subsurface Design<br>Bruce Stanley | A. Gantry Crane Rails<br>• Preliminary calculations have been performed to establish viability of concept<br>• 40 foot rail section at 200° C will expand about 0.53 inches<br>• Although not detailed at this time, a combination of fixed and slotted anchors will accommodate expansion<br>• Configuration of expansion gaps are adequate to support transporter weight at various temperatures<br>• Rail system will be designed to be maintainable for the required service life<br><br>B. Invert Structural Frame Beams<br>• Invert transfer beams are anchored on one end, and feature a slotted connection on the other end, allowing for expansion<br>• Preliminary estimate of a typical transfer beam expansion is about 0.13 inches at 200° C<br>• Design not yet detailed, and the invert configuration may change for LA |

| NRC Item 7(e.4)-Fire Design Criteria | | |
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| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| A. What is the technical basis for considering fire as a beyond-design-basis event and an internal event with no release?<br><br>1. The report on *Repository Safety Strategy* (CRWMS M&O, 2000c) bases classification of fire as a beyond-design-basis event on the information presented in *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O, 2000d). *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O, 2000d) specifies that the waste packages will be designed to withstand the fire environment defined in 10 CFR 71.73(c)(4). Criterion 1.2.2.1.11 of *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* (Pettit, 2000) also specifies the same fire design criteria. The fire design criterion in 10 CFR Part 71.73(c)(4) is a fire which is 1,475° F for 30 minutes. However, Criterion 1.2.1.6 of *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* (Pettit, 2000) states that "WP shall maintain SNF zircaloy cladding temperature below 350° °C (662° F) under normal operations, and below 570 □C (1,058° F) for short-term exposure to fire, as specified by Criterion 1.2.2.1.11." There is a clear inconsistency between the design criterion and cited reference (10 CFR Part 71 fire design | Waste Package<br>Thomas Doering | A.1.<br>The technical basis for classifying fire as a beyond-design-basis event is that significant fire hazards will be intentionally precluded at the repository through the design of the systems, structures, and components.<br><br>Future analysis of any off-normal waste package events will be based on the Category 1 and 2 credibility criteria defined in the final 10 CFR 63.<br><br>The *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document*, SDD-UDC-SE-000001 (Criterion 1.2.1.6), lists the waste form surface temperature not to be exceeded during an off-normal event (570 °C [1058 °F] for short-term exposure to fire). This limit is compared to the peak calculated waste form surface temperature from the off-normal event analysis.<br><br>Once sufficient information is available on the design of the repository systems, structures and components that interface with the waste package, the technical basis for off-normal waste package events will be documented in *Preclosure Design Basis Events Related to Waste Packages*, ANL-MGR-ME-000012. The results from the analysis of off-normal events will be documented in the appropriate design analysis reports (e.g., *Design Analysis for UCF Waste Packages*, ANL-UDC-MD-000001). |

| NRC Item 7(e.4)-Fire Design Criteria | | |
|---|---|---|
| NRC Comment | BSC Responsible Group/Individual | DOE Proposed Resolution |
| criterion).<br><br>2. It does not appear that DOE has considered the degradation of the WP materials when assessing the potential consequences of a design-basis fire. In *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (CRWMS M&O, 2000d) and Criterion 1.2.2.1.11 of *Uncanistered Spent Nuclear Fuel Disposal Container System Description Document* (Pettit, 2000), the DOE specifies that the waste packages will be designed to withstand the fire environment specified in 10 CFR 71.73(c)(4). However, when subjected to a temperature of 1475° F (800° C) for 30 minutes, Alloy 22 will have a significantly lower repassivation potential and, as a result, will be more susceptible to corrosion. In addition, the impact strength and ductility of Alloy 22 may be adversely affected by a design basis fire as well. Consequently, a waste package that has been subjected to a design-basis fire may exhibit faster corrosion rates and be more susceptible to failure by mechanically disruptive events if emplaced in the repository. The DOE has not provided a plan for dealing with a waste package after it has been subjected to this design-basis fire load.<br><br>DOE should analyze the potential consequences of waste packages after being subjected to a design-basis fire and propose any necessary preventive or corrective actions. | | A.2.<br>Any waste package involved in an off-normal event would be evaluated to ensure that its post-closure performance requirements will not be compromised. For any waste package whose post-closure performance cannot be ensured, it will be necessary to discard that waste package and repackage its contents in a fresh waste package. |

# NRC Staff Review of DOE's Dose Calculation Methodology for Category-1 and Category-2 Design Basis Events

DOE has proposed methodologies for calculating the anticipated annual average doses for Category-1 Design Basis Events (DBEs) and per-event-sequence dose for Category-2 DBEs in its November 2000 "Repository Safety Strategy" (RSS) and "Design Basis Event Frequency and Dose Calculation for Site Recommendation" (June 2000). The Category-1 dose is based on the annual exposure to a hypothetical subsistence farmer living at the site boundary while the Category-2 dose is based on a short-term (8-hour) acute exposure to an individual at the site boundary.

The total Category-1 annual dose estimate is based on contributions from three sources: (1) Category-1 DBEs; (2) routine releases from normal operations at the surface waste handling facility; and (3) normal operational releases from the subsurface facility. The annual dose resulting from Category-1 DBEs is calculated using the equation:

$$D = \sum F_i D_i$$

where:

| | |
|---|---|
| $D$ | total annual dose |
| $F_i$ | frequency of the $i^{th}$ Category-1 event sequence |
| $D_i$ | dose resulting from the $i^{th}$ Category-1 event sequence |

**<u>Discussion</u>**

Based on the review of DOE's "Repository Safety Strategy and Design Basis Events," "Dose Calculations for Site Recommendation," and subsequent discussions with DOE and its contractors, the NRC staff understands that the above equation will be used for demonstrating compliance with preclosure performance objectives (such as 25 mrem/yr and 100 mrem/yr under Category-1 DBEs) in the license application for construction authorization. However, it is assumed here that, after any license to receive and possess waste is issued, and waste handling operations start, actual (measured) doses would be used to monitor and report ongoing compliance with regulatory dose limits.

Calculation of annual doses as a part of preclosure safety assessments (PCSA/ISA) for demonstrating compliance during design stage, two classes of Category-1 events would be identified: (1) events that occur one or more times a year; and (2) events that occur less than once a year but at least once during the operational period. Calculation of the "annual" dose is complicated by consideration of the above two classes of events in a given year. Specifically, a method is needed to appropriately combine events expected to occur one or more times a year and those expected to occur less than once a year in order to calculate the doses in a given year of operation. DOE has provided the above approach, which estimates annual dose based on aggregation of <u>all</u> Category-1 events in an "annualized" manner (i.e., consequence weighted by the frequency).

During recent discussion among NRC, DOE and their contractors, further clarifications were made by DOE. The following summarizes the NRC staff understanding of DOE approach:

1) A frequency weighted sum of <u>all</u> Category-1 DBE doses (as in the equation above) will be added to the routine operational releases to demonstrate compliance with the regulatory dose limits.

2) In addition, the dose estimated to result from any single Category-1 event sequence will not be allowed to exceed the regulatory dose limits.

3) For the License Application Design of structures, systems and components important to safety under Category-2 DBEs, doses are calculated on a per-event-sequence basis and compared with the regulatory limit (5rem/event sequence).

**<u>Preliminary Staff Position</u>**

The staff believes this approach (i.e., sum of the annualized/frequency weighted doses for Category-1 DBEs and per-event-sequence-doses for category-2 DBEs) is acceptable because it is reasonable and technically defensible. In addition, it simplifies DOE's demonstration of compliance in the license application PCSA/ISA and NRC's review and compliance determination. It should be noted that this staff position is limited to the use of the approach discussed here and does not express any regulatory position regarding the dose estimates presented in the various DOE documents.

**<u>Points needing clarification</u>**

(1) Future revisions of the RSS and other reports must document that no single Category-1event sequence will result in a dose that exceeds the regulatory limits.

(2) In order to facilitate the staff review and help focus the design review on the particular event sequences that might contribute higher shares of doses to the total calculated annual dose, it will be necessary for the DOE to provide a table of dose contributions from individual Category-1 event sequences in addition to the sum.

(3) The approach used by DOE for demonstrating compliance with the regulatory limits for combination of Category-1 event sequences that could occur in a given year should be made transparent in the RSS.

(4) The RSS should also clarify how the dose calculation approach will be used in developing the list of structures, systems and components important to safety (Q-list).

(5) The RSS should explain in clear terms how the "bounding" dose term (referred to in DOE's Quality Level (QL) categorization process will be used in binning the items on the Q-list.

# DIFFERENTIATED APPROACH TO PROVIDING INFORMATION IN THE LICENSE APPLICATION

## 1.0 DOE PROPOSAL

Department of Energy (DOE) guidance document, "Technical Guidance Document for License Application Preparation, YMP/97-03, Rev 1, September 1999," (TGD) provides direction to Management and Operating Contractor (M&O) on preparation of License Application for Construction Authorization (CA) and License Application to Receive and Possess HLW (LRPW). Civilian Radioactive Waste Management System M&O, June 1999, "Level of Design Detail Necessary for the License Application for Construction Authorization," document presents proposal for grading the level of design details to be presented in the License Application (LA). These documents present DOE's proposal on (1) the kinds of information to be provided in the initial license application for construction authorization and subsequent update of the LA to receive and possess HLW, and (2) the level of detail of design information to be presented in the LAs to be commensurate with their importance to safety.

### 1.1 Differentiated Approach for Information in the License Application for Construction Authorization and License Application to Receive and Possess HLW

There are two situations that could occur relative to the information required at the time of docketing the LA for CA and updating the LA to receive and possess HLW.

- In the first situation, all of the information needed at the time to receive and possess HLW is available at the time of docketing the LA. In this case, the information will be provided in the LA at the time of docketing for CA (i.e., no differentiation in information to be provided in the LA). However, information such as site characteristics and radiation safety assessment etc. provided at the time of docketing for CA will be brought up to date to reflect the changes during construction, as part of the LA update to receive and possess HLW.

- In the second situation, some of the information needed at the time to receive and possess HLW will not be available at the time of docketing the LA for CA, nor will it be needed for the Nuclear Regulatory Commission (NRC) to issue a CA. In this case, the LA will differentiate between the information required at the time of docketing LA for CA and the information required at the time of LA update to receive and possess HLW.

DOE is proposing to differentiate between the information to be provided in the LA for CA and the information required in the LA update for LRPW. A differentiated approach means providing information required for CA at the time of docketing the LA and updating the LA with additional information for LRPW. The differentiated approach is proposed by DOE because some of the information needed for LA to receive and possess the HLW will not be available at the time of docketing the LA for CA, and because some information needed to support the LA update is not needed to support the CA.

DOE plans to present description of systems and summary of design in the LA for CA and place the supporting documents in the record center. LA will include - description of systems required to protect the health and safety of the public; description of engineered structures, systems, and components (SSCs) that are required to meet the post closure performance objectives; description of systems that process radioactive waste; information on fire protection and protecting required safety SSCs from interactions from nonsafety SSCs; and information on SSCs that are important to waste isolation. The updated LA submittal for license to receive and possess waste will incorporate, as appropriate, the updated designs and results of analyses for all safety systems. Some of this information may be totally new information.

## 1.2 Differentiated Approach in the Level of Design Detail for Structures Systems and Components

Preclosure Safety Assessment (PCSA), conducted at the appropriate level of rigor, will establish structures, systems, and components important to safety (SSCIS). DOE will categorize the SSCIS commensurate with their importance to safety/risk and apply graded Quality Assurance (QA) in implementing the QA program. DOE proposed three Quality Level (QL) categories (QL-I, QL-II, and QL-III) based on consequence (dose) criteria which is an indication of importance to safety/risk. Using the same categorization criteria proposed for the implementation of graded QA, DOE is proposing to grade the level of design detail to be provided for the SSCIS in the LA. QL-I items will have detailed design information and QL-III items will have minimal design information in the LA. In addition, the level of design detail proposed to be provided by DOE for the repository surface facility design, which has licensing precedent, is less than that for those SSCIS that do not have licensing precedent. Providing unnecessary details will be avoided, although references will be used to point to additional details in a given area. For structures, systems, and components that require research and development to confirm the adequacy of design, a plan for obtaining the needed information will be presented, and schedules for obtaining the information will be provided in the LA.

Using the proposed categorization, DOE proposes (based on draft Part 63) to provide the following level of detail of design information for the SSCIS. The LA will include representative discussions of the following:

For QL-I SSCIS the following information shall be included in the LA:
- Applicable Codes and Standards
- Design Criteria and Regulatory Design Bases
- General System Description
- Piping and Instrumentation Diagrams
- Electrical one-line Diagrams
- General Arrangement Drawings
- Handling Diagrams

<u>For QL-II</u> SSCIS the following information shall be included in the LA:

- Applicable Codes and Standards
- Design Criteria
- General System Description

For QL-III SSCIS, the following information shall be included in the LA:

- Design Criteria
- General System Description

Non-Safety SSCs
General description on non-safety SSCs will be included in the LA that is sufficient to demonstrate the non-safety classification.

## 2.0 APPLICABLE REGULATORY REQUIREMENTS (from proposed Part 63 published for public comment)

LICENSE APPLICATION

§63.21 Content of application.
§ 63.24 Updating of application and environmental impact statement
§ 63.31 Construction authorization.
§ 63.32 Conditions of construction authorization.
§ 63.41 Standards for issuance of a license.

## 3.0 STAFF POSITION

### 3.1 Differentiated Approach in Providing Information

Proposed 10 CFR 63.21 identifies the required contents of the LA. NRC regulations require the applicant to provide information as complete as possible in light of information available at time of docketing the license application for construction authorization. The applicant must update its application in a timely manner to allow commission review before issuance of a license. Proposed 10 CFR 63.24(b) permits DOE to update the LA with additional information required for submitting the LA to receive and possess HLW. The information DOE is required to provide in the LA (§63.21) for the construction authorization decision must be sufficient to demonstrate compliance with the requirements of proposed §63.31 Construction Authorization. DOE is required to perform preclosure safety assessment (PCSA similar to Integrated Safety Assessment) at the appropriate level of rigor to identify the structures, systems, and components important to safety and waste isolation (§63.112). In the absence of final design, DOE's PCSA needs to be conservative in identifying the SSCs important to safety and waste isolation. In general, the level of detail of the information in the LA should enable the NRC staff to determine if there is reasonable assurance that DOE has demonstrated compliance with the applicable regulations including receiving and possessing and disposing HLW. Demonstrating compliance requires providing sufficient technical basis to allow NRC to make a finding of reasonable assurance that the types and amounts of radioactive materials described in the application can be received, possessed, and disposed of in the geologic repository operations area of the proposed design without unreasonable risk to the health and safety of the public.

Differentiated approach proposed by DOE may result in the LA presenting all the information required for the staff to make a determination on compliance with Proposed 10 CFR 63.31for construction authorization if DOE provides all reasonably available information and updates as necessary, and the updated LA should contain the information needed to make a determination on compliance with standards for issuance of a license to receive and possess HLW as per proposed 10 CFR 63.41. The differentiated approach is acceptable provided DOE submits information at the CA and LRPW that is sufficient for NRC to make a 'safety determination'.

### 3.2 Differentiated Approach in the Level of Detail of Design Information in LA

NRC policy permits the quality assurance program to control activities affecting the quality of the identified structures, systems, and components, to an extent commensurate with their importance to safety. Provision of controlling QA activities commensurate with their importance to safety permits graded QA approach. A properly conducted PCSA at the appropriate level of rigor identifies the SSCIS, and DOE proposes to implement graded QA to SSCIS. Similarly, a properly conducted post closure performance assessment (PA) identifies SSCIS and their performance requirements in the post closure time. DOE proposes to categorize, commensurate with their importance to safety, SSCIS into three categories (QL-I, QL-II, and QL-III) for QA implementation. DOE proposal on the criteria for categorizing SSCIS is under consideration by the staff and requires further discussions with DOE. QA categorization is not the subject of this paper, and therefore, the criteria/process for categorization of SSCIS is not addressed here. Acceptable list of SSCIS and QA categorization methodology are assumed in this document, and further discussion only relates to DOE's proposal for "Differentiated Approach in the Level of Detail of Design Information in LA."

DOE has extended its QA categorization of SSCIS into a corresponding classification for the purpose of defining the level of detail of design information to be provided in the LA. In addition, because the Repository Surface Facility design has licensing precedent, the level of detail of design DOE plans to provide for these SSCs in the LA is less than that required for the SSCs that do not have licensing precedent.

Proposed 10 CFR 63.21(c) identifies the kinds of design information to be provided in the LA for SSCIS. NRC regulations don't specifically address the concept of level of detail of design information in the LA for an item to be commensurate with the safety significance of the particular item. The information in the LA for all SSCIS should be sufficient for NRC to make a finding of reasonable assurance on DOE's demonstration of compliance with regulations.

In Nuclear Power Plant licensing, NRC has accepted lesser level of detail of design information in the preliminary safety analysis report (PSAR) because the information was not available at that stage. However, the final safety analysis report (FSAR) provided all the information required for the staff to determine compliance with the regulations. Both PSAR and FSAR LAs contained sufficient information to enable the staff to make determination on compliance with the regulations. Also, NRC was able to closely monitor the design and construction activities of the licensee between PSAR and FSAR stages.

Although there is no regulatory precedent to grade the level of detail of design information in LA, the staff agrees in principle with DOE that level of detail of design information for SSCIS in the LA can be tailored commensurate with their importance to safety as long as the information is sufficient for the staff to make a finding on DOE's demonstration of compliance with the regulations. A properly conducted PCSA and PA, and a transparent process for categorizing the SSCIS that takes into consideration the uncertainties in the underlying information are essential to this process and is a key assumption in this approach. In the absence of specific technical/design criteria in the performance-based proposed 10 CFR Part 63 regulations, the staff needs the following information in the LA in addition to those proposed by DOE to reach a conclusion of reasonable assurance on DOE's demonstration of safety.

<u>Additional Information</u>

For QL-1 SSCIS:
Information on dimensions, material properties, specification, and analytical and design methods used in the design.

For QL-2 SSCIS:
Regulatory Design Bases, General Arrangement Drawings

For QL-3 SSCIS
Applicable Codes and Standards, Regulatory Design Bases,

The level of detail of design information proposed by DOE together with the above identified additional information are expected to be sufficient, for the initial or first stage of LA review for CA, for the staff to make finding of reasonable assurance that DOE has demonstrated compliance with the applicable proposed 10CFR 63 regulations. Depending on the complexity of the SSCIS, on a as-needed basis, the staff may request additional information to enable a review of the LA.

## 4. REFERENCES


1. DOE (Department of Energy) September 1999. Technical Guidance Document for License Application Preparation, YMP/97-03, Revision 1

2. CRWMS (Civilian Radioactive Waste Management System) M&O (Management and Operating Contractor), June 1999, Level of Design Detail Necessary for the License Application for Construction Authorization.

DRAFT - June 13, 2001

# STAFF POSITION ON RISK SIGNIFICANCE CATEGORIZATION OF STRUCTURES SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY

## Purpose

The purposes of this paper are to: (1) identify the attributes of an acceptable approach to risk significance categorization of structures, systems and components (SSCs) consistent with their importance to safety; (2) evaluate the U.S. Department of Energy's (DOE's) proposed approach for the risk significance categorization of Structures, Systems and Components Important to Safety (SSCIS) for the proposed geologic repository operations area (GROA) at the Yucca Mountain site; (3) identify staff concerns; and (4) suggest a path to resolution for concerns identified by the staff evaluation.

To this end, this paper discusses the governing regulation and applicable policy and guidance and develops generic acceptance criteria based on this background information. Further, it discusses DOE's proposed approach to risk categorization and evaluates its approach against the generic acceptance criteria and necessary background information. It identifies staff comments and concerns and provides a possible path to resolution for these comments and concerns.

## Applicable NRC Regulation

The proposed 10 CFR Part 63 allows the risk significance categorization of SSCs to an extent consistent with their importance important to safety (§63.143) and identifies the performance objectives governing preclosure operations (§63.111). However, the proposed 10 CFR Part 63 does not identify or designate any specific process or methodology for risk categorization. So it is essential for the reader to have a clear understanding of the regulations governing the design, construction and operations of a potential GROA at the Yucca Mountain site and other similar NRC-regulated facilities. With this in mind, relevant portions of applicable regulatory requirements and preclosure performance objectives for the GROA are provided in Appendix-A.

## Applicable NRC Policy and Guidance

There are no specific regulatory guidance documents or policy specifically relating to the risk significance categorization of SSCs consistent with their importance to safety for a potential GROA. However, the NRC has developed extensive direction (in the form of regulatory policy and guidance) on risk-informed decision-making that is directly related to risk categorization and the issues being considered by this paper. A summary of relevant information from such documents is provided in Appendix-B.

## Attributes of an Acceptable Risk Significance Categorization Process for the GROA SSCIS

The following discussion identifies the attributes of an acceptable approach to risk significance categorization of SSCIS consistent with their relative importance to safety. These attributes are based on the governing regulation and applicable policy and guidance discussed in the appendices. The following attributes represent the minimum characteristics necessary for an acceptable approach to risk significance categorization of SSCIS.

**The risk significance categorization of SSCIS shall be consistent with the existing and proposed regulatory framework:**

- The identification of SSCIS shall be consistent with the governing regulation and applicable policy and guidance;
- The identification of SSCIS (Q-List generation) shall be done using an Integrated Safety Assessment (ISA) methodology that is consistent with and fulfills the requirements in proposed § 63.112;
- The categorization methodology shall consider the frequency of Design Basis Events (DBE Categories 1 & 2) in the proposed § 63.2;
- The categorization methodology shall consider the dose limits in the proposed § 63.111 (including Part 20); and
- The categorization methodology shall provide due consideration of uncertainties and sensitivity analyses for DBE frequencies in a manner that is consistent with the applicable portions of existing NRC policy and guidance, including: *NRC's final policy statement on PRA*; RG 1.174; RG 1.176; SECY-98-144, *The White Paper on Risk-Informed and Performance-Based Regulation*; SECY-99-100; and NUREG-0800 Chapter 19, Use of PRA in Plant-Specific Risk-Informed Decision-Making: General Guidance.

**The risk significance categorization of SSCIS shall be consistent with their relative importance to safety:**

- The categorization methodology shall ensure that SSCIS are categorized consistent with their risk significance and relative importance to safety (§63.143);
- The distinctions between quality levels should have a well defined and well documented technical basis;
- The probabilities and consequences of failures of SSCIS at the various quality levels shall be well defined and consistent with applicable portions of existing NRC policy and guidance; and
- The categorization methodology for SSCIS shall be supported by appropriate qualitative descriptions and quantitative or semi-quantitative methods.

**The risk significance categorization of SSCIS should demonstrate flexibility:**

- The categorization methodology should demonstrate flexibility to accommodate the iterative nature of the design process;
- The categorization methodology should permit the revision of the categorization level of individual and groups of SSCIS as a result of the introduction of new data and/or design changes; and
- The categorization methodology should be flexible enough to accommodate multiple iterations of the ISA and subsequent evaluation of risk significance.

**The documentation and analysis for the risk significance categorization of SSCIS shall be transparent and traceable:**

- The risk significance categorization methodology should be developed and presented in such a manner that the reviewer can gain a clear understanding of every step of what has been done, what the results are, and the technical bases for the results; and
- The categorization methodology should include an unambiguous and complete record of the decisions and assumptions made, and the process used in arriving at a given conclusion or result.

The previous discussion outlines the staff's expectations for an acceptable categorization methodology and is based on an extensive review of existing and proposed regulatory requirements, policy, and guidance. These attributes are expected to be developed into acceptance criteria and introduced into the appropriate section of the Yucca Mountain Review Plan (YMRP) that is currently under development.

## DOE's Proposed Approach to the Risk Significance Categorization of SSCIS

DOE has developed a process for the categorization of SSCIS [9, 10, 21]. DOE has presented its approach to risk categorization to the staff during several technical interactions [21]. A summary of DOE's approach is presented here.

The first step in the process involves performing an ISA to identify those SSCs that will be relied upon to protect the health and safety of the public and onsite workers. These SSCIS are then categorized using a classification procedure and placed into one of three categories. The following discussion outlines the elements of DOE's approach to risk categorization of SSCIS.

The Preliminary Preclosure Safety Assessment (PPSA)[9] and the Repository Safety Strategy (RSS)[10] provide a detailed descriptions of the individual elements of the ISA process, as well as a graphical representation of this process (Figures in Appendix-C and Appendix-D, respectively). DOE's ISA is comprised of the following elements:

- Hazard Identification
- Event Identification
- Event Sequence Identification
- Quantitative Frequency Assessment
- Beyond Design Basis Event Determination
- Assignment of Frequency Categories 1 or 2
- Consequence Analysis
- Selection of Categories 1 and 2 DBEs
- Determination if DBE is within Assigned Performance Objectives for the DBE
- Determination of the Need for Preventive or Mitigative Features
- Assessment of Impact on the Design.

The ISA provides input to the quality assurance classification process. Repository SSCs credited with event prevention or mitigation in the ISA fall within the definition of important to safety as described in the proposed §63.2. The ISA is the tool used to determine an SSC's functional role as part of the repository preclosure safety case. SSCs are categorized in a graded fashion to assure quality assurance controls are commensurate with the item's importance to safety [10].

The categorization process considers the configuration and function of SSCs and their effects on repository radiological safety. Classification analyses are performed based on the system design and the System Description Documents (SDDs). These analyses use the DBEs from the ISA to evaluate GROA preclosure operations facility SSCIS against the categorization screening criteria in QAP-2-3 to determine the QL of the respective SSCIS. This categorization process screens the SSCIS into one of the following category or quality levels:

- Quality Level 1 (QL-1): Permanent items (SSCIS) whose failure could *directly* result in a condition adversely affecting public safety. These permanent items are determined to have a high safety significance.

- Quality Level 2 (QL-2): Permanent items whose failure or malfunction could *indirectly* result in a condition adversely affecting public safety, or whose direct failure would result in consequences in excess of normal operational limits. These permanent items are determined to have low public safety significance.

- Quality Level 3 (QL-3): Permanent items whose failure or malfunction would not significantly impact public or worker safety, including those defense-in-depth design features intended to keep doses As Low As Reasonable Achievable (ALARA). These item are determined to have minor impact on public or worker safety.

- Conventional Quality: Permanent item not meeting any of the criteria for Quality Levels 1, 2, or 3.

DOE's classification process is based on, and is considered by DOE to be consistent with, the classification process outlined in NUREG/CR-6407. A detailed summary of the quality level screening criteria is included in Appendix E . Additionally, figures presented at the March 8, 2000, technical exchange[19] may serve to summarize DOE's risk categorization process and are included as Appendices C and D of this paper. Finally, this iteration of the categorization process is completed when the SSCs are added to the Q-List as described in the procedure YAP-2.7Q [22].

**Evaluation of DOE's Approach and Discussion**

The staff recognizes the inherent challenge of trying to develop an acceptable approach to risk categorization of SSCIS for the GROA, and acknowledges the considerable level of effort that must have gone into the development of DOE's proposed approach. Based on DOE presentations and discussions during NRC-DOE technical interactions, and a review of DOE's RSS [10] and DOE's PPSA [9]), the staff has the following observations on the proposed DOE approach to risk categorization.

The proposed 10 CFR Part 63 (and 10 CFR Parts 20, 50, and 70) do not identify or require any specific process or methodology for the risk significance categorization of SSCIS. Further, there is no regulatory guidance or policy specifically addressing risk categorization of SSCIS for a potential GROA. However, the NRC has developed extensive direction (in the form of regulatory policy and guidance) on risk-informed decision-making that is directly related to risk categorization and the issues being considered by this paper. In order to adequately review DOE's proposed risk categorization methodology, it is necessary to consider the applicable policy and guidance governing the design, construction and operations of a potential GROA at the Yucca Mountain site and other similar NRC-regulated facilities.

DOE's proposed risk categorization methodology is based on the quality levels defined in procedure QAP-2-3[20] and its associated screening criteria, as discussed earlier in this paper. DOE has stated that the quality level or "important to safety classification" is "consistent" [19] with, the three tier approach and classification categories described in NUREG/CR-6407. It is important to note that the approach identified in NUREG/CR-6407 (and its predecessor RG 7.10) predates all of the risk-informed policy and guidance developed by the NRC since the Commission's Final Policy Statement on the Use of PRA[4] issued in 1995. Further, the approach to classification identified in NUREG/CR-6407 does not require the consideration of risk insights or significance. It does not consider probability. It only assesses consequences in terms of the maximum amount of radioactive material permitted in the transportation package. It assigns classification categories using a strictly deterministic approach. The staff has several concerns regarding DOE's use of the classification categories described in NUREG/CR-6407 for the risk significance categorization of SSCIS of a potential GROA. The following discussion outlines these staff concerns , several of which involve the use of QL-2 screening criteria (as identified in procedure QAP-2-3).

**Concern 1:** Two of DOE's QL-2 screening criteria which are not consistent with the definition of event sequences provided in the proposed § 63.2 (QAP-2-3 Appendix II, Checklist Items 8.2.5 and 8.2.6). These screening criteria only consider the failure of one item in conjunction with *"an additional item* or *administrative control* (i.e., indirect impact)." Whereas the definition of event sequences (presented in the proposed final § 63.2) states: "An event sequence includes one or more initiating events and *associated combinations of repository system component failures*,..." and does not place a limit on the number of component failures.

**Concern 2:** The screening criterion identified in QAP-2-3 Appendix II, Checklist Items 8.2.3 may result in mis-categorization. This criterion states: "As a result of DBE, could consequential failure of the item, which is not intended to perform a QL-1 radiological safety function, prevent QL-1 SSCIS from performing their intended radiological safety function?" The purpose and justification for this screening criterion are unclear. According to DOE's definition of QL-1, this screening criterion appears to identify SSCIS "whose failure could *directly* result in a condition adversely affecting public safety" or risk, and as such should be not be categorized as QL-2 but QL-1 SSCIS. Additional clarification is required.

**Concern 3:** The screening criterion identified in QAP-2-3 Appendix II, Checklist Items 8.2.2 may result in mis-categorization. This criterion states: "Does the item provide fire protection, fire suppression, or otherwise protect important to radiological safety or waste isolation functions of QL-1 SSCIS from the hazards of a fire?" Again the purpose and justification for this screening criterion are not clear. If the failure of this item has the potential to adversely affect the ability or function of a QL-1 SSCIS then according to DOE's definition of QL-1, this screening criterion appears to identify SSCIS "whose failure could *directly* result in a condition adversely affecting public safety," or risk and as such would be not be categorized as QL-2 but QL-1 SSCIS. Additional clarification is required.

**Concern 4:** The terms "in conjunction with" and "indirect impact" as described in QAP-2-3 Appendix II, Checklist Items 8.2.5 and 8.2.6. These screening criteria are not well-defined. As described in QAP-2-3, it appears that DOE could have a situation in which the failure of two QL-2 SSCIS could potentially have the same risk as the failure of a single QL-1 SSCIS. The purpose and justification for this screening criterion are unclear. Again, this screening criterion is more consistent with DOE's definition of QL-1. Further, it would appear that either one or both of these SSCIS would be categorized as QL-1.

The use of the three tier approach described in NUREG/CR-6407 and particularly the use of the term "indirectly"as the basis for the risk significance categorization of SSCIS appears to have several limitations, as described above. The resulting QL-2 screening criteria seem to be ambiguous in some instances. DOE may want to reconsider the use and application of this approach or provide additional justification to address the stated concerns.

**Concern 5:** DOE has not performed any uncertainty or sensitivity analyses of the quantification of event sequence frequencies. Uncertainty analyses are important in that they can be used to identify and quantify sources of uncertainty and variability associated with the quantification of event sequence frequencies. It is important to understand the uncertainty and variability associated with the quantification of event sequence frequencies because DOE's "risk thresholds" are the same as the performance objective in §63.111. It is also necessary to have a clear understanding of the uncertainty and variability associated with DOE's frequency calculations because these frequency calculations are used to determine which frequency category each of the respective event sequences are binned into and accordingly which of the performance objectives apply to that particular event sequence. Uncertainty and sensitivity analyses will also be important in addressing some of the potential complexities associated with DOE's risk calculations for the event sequences and the aggregate or some measure of the overall or aggregate risk. DOE needs to consider the use of uncertainty and sensitivity analyses where applicable or provide justification that explains why these analyses are not necessary.

**Concern 6:** DOE is not using estimates of the aggregate risk to determine the contribution of individual event sequences or their associated SSCIS to an overall measure of risk. DOE's approach to risk categorization individually identifies a measure of "risk" associated with each of the credible event sequences and their associated SSCIS. Additionally, DOE has provided a cursory indication of the estimated aggregate risk for frequency category 1 and frequency category 2[9] event sequences. A comparison of the individual risk to the overall risk is necessary to ensure that the SSCIS are being categorized consistent with their relative contribution to overall importance to safety or risk significance. The staff is concerned that there is no comprehensive analysis or comparison tying the event sequences together to determine their contribution to the overall risk. The importance of comparing the risks associated with individual or grouped event sequences or their associated SSCIS to the overall risk is discussed in several of NRC policy and guidance documents, including: RG 1.174, RG 1.176, the NRC "White Paper,"[8] and NUREG-0800 Chapter 19 (even going as far as suggesting the use of importance measures). DOE needs to consider some type of comparison of the individual risk to the overall risk as described above, or provide justification for why they are not doing so.

**Concern 7:** DOE's classification analyses and subsequent risk categorization may benefit from the use of a multi-disciplinary review group similar to the "expert panel" described in RG 1.176. DOE's proposed approach to risk categorization relies heavily on the screening criteria identified in procedure QAP-2-3 and the associated classification analyses. Specifically, DOE is relying heavily on those individuals performing and reviewing these classification analyses. NRC guidance [6] recommends the use of a multi-disciplinary review group of technical and professional personnel, referred to as the "expert panel," to support risk-informed decision-making process. This expert panel performs an integrated assessment of quantitative and qualitative risk insights to determine the safety significance ranking of SSCIS.

## Points Requiring Additional Clarification

In addition to the concerns identified above the staff have identified several point that require further clarification:

1. Additional explanation and examples are required to show how DOE proposes to integrate in a transparent manner, the use of the equation $\sum F_i D_i + D_e$ (RSS Figure 8-1), the "take away" analyses (RSS Section 8), the screening criteria in procedure QAP-2-3, and the classification analyses.
2. Clarification is required as to how DOE is proposing to include multiple category 1 event sequences in the proposed categorization process.

3. Clarification is required as to whether the routine releases from surface and subsurface facilities during normal operational are factored into the equation $\sum F_i D_i + D_e$ (RSS Figure 8-1).
4. Additional explanation is required regarding the establishment and use of the bounding dose term ($D_e$) in the Q-list and categorization process.

## Summary and Path to Resolution

Based on this review the staff concludes that DOE's proposed risk categorization methodology has merits; however, the staff has identified several concerns. The staff is aware that procedure QAP-2-3 is in the process of being revised (incorporated into a new procedure) and based on informal discussion with DOE this may serve to address some of the concerns identified in the above discussion section of this paper. The contents of this paper is expected to serve as discussion points for preclosure technical exchange tentatively scheduled for July 2001. Staff concerns have been summarized in the following table.

| **Staff Concerns on DOE Approach and Proposed Path To Resolution** | | |
|---|---|---|
| **Concern** | **Proposed Path to Resolution** | **Acceptance Criteria** |
| 1<br>Consistency with Regulation | Revise the screening criteria in QAP-2-3 Appendix II, Checklist Items **8.2.5** and **8.2.6** to be consistent with the definition of event sequences (presented in the proposed final § 63.2), as described in Concern 1. | The identification of SSCIS shall be consistent with the governing regulation and applicable policy and guidance |
| 2<br>Justification for Screening Criteria | Provide additional justification for the use of the QL-2 screening criterion found in QAP-2-3 Appendix II, Checklist Item 8.2.3 or revise it to agree with existing DOE terminology <u>and</u> ensure that it is risk-informed and consistent with the proposed final rule and existing regulatory framework, as described in Concern 2. | The identification of SSCIS shall be consistent with the governing regulation and applicable policy and guidance.<br><br>The categorization methodology shall ensure that SSCIS are categorized consistent with their risk significance and relative importance to safety (§63.143).<br><br>The documentation and analysis for the risk significance categorization of SSCIS shall be transparent and traceable. |

| Staff Concerns and Proposed Path To Resolution (continued) | | |
|---|---|---|
| **Concern** | **Proposed Path to Resolution** | **Acceptance Criteria** |
| 3<br>Justification for Screening Criteria | Provide additional justification for the use of the QL-2 screening criterion found in QAP-2-3 Appendix II, Checklist Items **8.2.2** or revise it to agree with existing DOE terminology and ensure that this criterion is risk-informed and consistent with the proposed final rule and existing regulatory framework, as described in Concern 3. | The identification of SSCIS shall be consistent with the governing regulation and applicable policy and guidance.<br><br>The categorization methodology shall ensure that SSCIS are categorized consistent with their risk significance and relative importance to safety (§63.143).<br><br>The documentation and analysis for the risk significance categorization of SSCIS shall be transparent and traceable. |
| 4<br>Clarification of Terminology | Provide additional justification for the use of the QL-2 screening criteria found in QAP-2-3 Appendix II, Checklist Item **8.2.5** and **8.2.6** or revise them to agree with existing DOE terminology and ensure that it is risk-informed, as described in Concern 4. | The categorization methodology shall ensure that SSCIS are categorized consistent with their risk significance and relative importance to safety (§63.143).<br><br>The documentation and analysis for the risk significance categorization of SSCIS is consistent with their relative importance to safety and shall be transparent and traceable. |
| 5<br>Uncertainty and sensitivity Analyses | DOE needs to consider the use of uncertainty and sensitivity analyses where applicable to assess the impact of risk categorization decisions or provide justification that explains why these analyses are not necessary. | The risk categorization methodology should provide due consideration of uncertainties in DBE frequencies consistent with discussion provided in existing NRC policy and guidance,<br><br>The risk categorization methodology for SSCIS shall be transparent and supported by appropriate qualitative descriptions and quantitative or semi-quantitative methods,<br><br>The risk categorization methodology to derive the relative importance to safety (i.e., High, Medium, or Low) shall be risk-informed (considering both frequency and consequence). |

| Staff Concerns and Proposed Path To Resolution (continued) | | |
|---|---|---|
| **Concern** | **Proposed Path to Resolution** | **Acceptance Criteria** |
| 6<br>Relative Importance | Consider performing some type of comprehensive analysis identifying the aggregate risk, relative importance of each of the event sequences, and the relative importance of the SSCIS, as described in Concern 6. | The risk categorization methodology should provide due consideration of uncertainties in DBE frequencies consistent with discussion provided in existing NRC policy and guidance.<br><br>The risk categorization methodology for SSCIS shall be transparent and supported by appropriate qualitative descriptions and quantitative or semi-quantitative methods. |
| 7<br>Expert Panel | Consider the use of an expert panel (multi-disciplinary) to support the safety significance ranking of SSCIS, as described in Issue 7. | The categorization methodology shall ensure that SSCIS are categorized consistent with their risk significance and relative importance to safety (§63.143).<br><br>The identification of SSCs important to safety (SSCIS) shall be consistent with the governing regulation and applicable policy and guidance.<br><br>The documentation and analysis for the risk significance categorization of SSCs consistent with their importance to safety shall be transparent and traceable. |

## APPENDIX-A
## Applicable NRC Regulation

Proposed 10 CFR 63 (ref a) provides regulations governing the licensing and operation of the U.S. Department of Energy to receive and possess source, special nuclear, and byproduct material at a GROA sited, constructed, or operated at Yucca Mountain, Nevada, in accordance with the Nuclear Waste Policy Act of 1982, as amended, and the Energy Policy Act of 1992.

To have a clear understanding of the requirements governing the categorization of SSCs it is first necessary to review several key sections of the proposed rule, including: § 63.2,Definitions; § 63.111, Performance Objective for the Geologic Repository Operations Area through Permanent Closure; § 63.112, Requirements for Preclosure Safety Analysis of the Geologic Repository Operations Area; and § 63.142, Quality Assurance Criteria.

The proposed rule provides definitions of the terms design basis event (DBE), important to safety, and Integrated Safety Analysis (ISA). Specifically, proposed § 63.2 defines design basis events (DBEs) as:

> "Design basis events means: (1) Those natural and human-induced events that are expected to occur one or more times before permanent closure of the geologic repository operations area (referred to as Category 1 events). (2) Other natural and man-induced events that have at least one chance in 10,000 of occurring before permanent closure of the geologic repository (referred to as Category 2 events)."

Proposed § 63.2 also defines important to safety (ITS) as:

> "... those engineered features of the geologic repository operations area whose function is: (1) To provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of § 63.111(b)(1) for Category 1 design basis events; or (2) To prevent or mitigate Category 2 design basis events that could result in doses equal to or greater than the values specified in § 63.111 (b)(2) to any individual located on or beyond any point on the boundary of the site."

Proposed § 63.2 defines the Integrated Safety Analysis (ISA) as:

> "... An analysis to identify hazards and their potential for initiating events sequences, the potential event sequences and their consequences, and site, structures, systems, components, equipment, and activities of personnel, that are relied upon for safety.

Proposed § 63.111 specifies performance objectives governing each of the following areas: protection against radiation exposures and releases of radioactive material, numerical guides for design objectives, preclosure safety analysis, performance confirmation, and retrievability of waste. A summary of these performance objectives has been included in the following discussion:

§ 63.111 Performance Objectives for the Geologic Repository Operations Area through Permanent Closure.

(e) *Protection against radiation exposures and releases of radioactive material.*
(1) The geologic repository operations area must meet the requirements of part 20 of this chapter.
(2) During normal operations, and for Category 1 event sequences, the annual dose to any real member of the public, located beyond the boundary of the site may not exceed a TEDE of 0.25 mSv (25 mrem).

(b) *Numerical guides for design objectives.*

(1) The geologic repository operations area must be designed so that taking into consideration Category 1 event sequences and until permanent closure has been completed, the aggregate radiation exposures and the aggregate radiation levels in both restricted and unrestricted areas, and the aggregate releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in paragraph (a) of this section.

(2) The geologic repository operations area must be designed so that taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on, or beyond, any point on the boundary of the site, will receive, as a result of a single category 2 event sequence, the more limiting of a TEDE of 0.05 Sv (5 rem), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem), and the shallow dose equivalent to the skin may not exceed 0.5 Sv (50 rem).

(c) *Preclosure safety analysis.* A preclosure safety analysis of the geologic repository operations area that meets the requirements specified in § 63.112 must be performed. This analysis must demonstrate that: (1) The requirements of § 63.111(a) will be met; and (2) The design meets requirements of § 63.111(b).

(d) *Performance confirmation.* The geologic repository operations area must be designed so as to permit implementation of a performance confirmation program that meets the requirement of subpart F.

(e) *Retrievability of waste.*

(1) The geologic repository operations area must be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and thereafter, until the completion of a performance confirmation program and Commission review of the information obtained from such a program. To satisfy this objective, the geologic repository operations area must be designed so that any or all emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 years after waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission. This different time period may be established on a case-by-case basis consistent with the emplacement schedule and the planned performance confirmation program.

(2) This requirement may not preclude decisions by the Commission to allow backfilling part, or all of, or permanent closure of the geologic repository operations area, before the end of the period of design for retrievability.

(3) For purposes of paragraph (e) of this section, a reasonable schedule for retrieval is one that would permit retrieval in about the same time as that required to construct the geologic repository operations area and emplace waste.

The proposed rule specifies the use of an ISA of the Geologic Repository Operations Area to, in part, to provide a comprehensive identification of hazards. Specifically, and proposed § 63.112(b) requires:

> "An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential accident/event sequences that would result in unacceptable consequences (i.e., design basis events)."

The proposed rule specifies the use of a ISA of the Geologic Repository Operations Area to, in part, identify those SSCs that are important to safety. Specifically, and proposed § 63.112(e) requires:

> "... An analysis of the performance of the major design structures, systems, and components, both surface and subsurface, to identify those that are important to safety, including identification and description of controls relied on to limit or prevent potential accidents or mitigate their consequences, and including measures taken to ensure the availability of identified safety systems ..."

Subpart G of the proposed rule outlines the scope, applicability, and implementation of the Quality Assurance Program. Specifically, proposed § 63.142 states:

> "The quality assurance program applies to all systems, structures, and components important to safety, to design and characterization of barriers important to waste isolation, and thereto."

Proposed section §63.143 states:

> "DOE shall implement a quality assurance program based on the criteria of Appendix B of 10 CFR 50, as applicable, and appropriately supplemented by additional criteria, as required by §63.142."

It is important to note that criterion II to Appendix B of 10 CFR 50 contains language requiring the categorization of SSCs in a manner that is commensurate with their safety significance:

> "The quality assurance program shall provide control activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety."

These sections of the proposed rule form the regulatory basis for the risk-informed categorization of ITS SSCs for the GROA.

The use of ISA and risk categorization as required in the revised 10 CFR 70, *Domestic Licensing of Special Nuclear Material: Possession of a Critical Mass of Special Nuclear Material*, provides another NRC staff approved approach to risk-informed decision-making. The new §70.61(a) requires an applicant or licensee to perform and ISA to demonstrate compliance with the performance requirements stated in §70.61(b), (c), and (d). These performance requirements outline the interaction between consequences, likelihood, and ultimately risk as it is defined in this rule and are summarized in the following discussion:

- §70.61(b) requires the risk of each credible high-consequence event must be limited. It further states that engineered and/or administrative controls shall be applied to the extent needed to reduce the likelihood of occurrence of the event such that, upon implementation of the controls, that the event is highly-unlikely or its consequences are less severe than those identified in §70.61(b)(1)-(4).

- §70.61(c) requires the risk of each credible intermediate-consequence event must be limited. It further states that engineered and/or administrative controls shall be applied to the extent needed to reduce the likelihood of occurrence of the event such that, upon implementation of the controls, that the event is unlikely or its consequences are less severe than those identified in §70.61(c)(1)-(4).

- §70.61(d) requires that the risk of nuclear criticality must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical including use of an approved margin of subcriticality

The rule prescribes consequence categories and acceptable levels of risk while allowing the applicant or licensee to determine and justify the gradation of likelihood categories.

Revised §70.4 defines items relied on for safety (IROFS) as SSCs and activities of personnel that are relied on to prevent potential accidents as a facility that exceed the performance requirements in §70.61 above or to mitigate their potential consequences.

The new §70.62(d) requires that each applicant or licensee establish management measures to ensure compliance with the performance requirements of Sec. 70.61. It states that the measures applied to a particular engineered or administrative control or control system may be graded commensurate with the reduction of the risk attributable to that control or control system. These management measures shall ensure that engineered and/or administrative controls and control systems that are identified as IROFS, are implemented and maintained, as necessary, to ensure they are available and reliable to perform their function when needed, in compliance with the performance requirements §70.61.

**APPENDIX-B**
**Applicable NRC Policy and Guidance**

The NRC has also developed extensive direction (in the form of policy and guidance) on the use and application of risk insights in the regulatory decision making process. The following discussion captures the portions of the policy and guidance that provide insight into the risk-informed categorization process identified in the proposed 10 CFR 63, as discussed above.

NRC's final policy statement on probabilistic risk assessment (PRA)(4) encourages greater use of PRA and risk insights to improve safety decision making and regulatory efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data. The probabilistic approach to regulation is considered an extension and enhancement of traditional (deterministic) regulation by considering risk in a more coherent and complete manner; ultimately focusing regulations on those items most important to safety.

Several Regulatory Guides (RGs) discuss the application of risk insights and risk importance measures to categorize SSCs with respect to safety significance, including: RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* (ref d), and RG 1.176, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance* (ref e).

Regulatory Guide 1.174 provides general guidance concerning an approach that the NRC has determined to be acceptable for analyzing issues associated with proposed changes to a plant's licensing basis (LB) and for assessing the impact of such proposed changes on the risk associated with plant design and operation. This RG forms the basis for the approach to graded quality assurance as discussed in RG 1.176 and referenced by DOE. One of the major considerations or decision criteria for incorporating risk insights into the risk-informed decision-making process is an estimate of the change in risk as a result of the proposed change. This approach supports NRC's desire to base its decisions on the results of traditional engineering evaluations, supported by insights (derived from the use of PRA methods) about the risk significance of the proposed changes. This regulatory guide is intended to improve the consistency in regulatory decisions in the areas in which the results of risk analyses are used to help justify regulatory action.

Regulatory Guide 1.174 establishes a set of key safety principles and expectations upon which the risk-informed decision-making approach is based and describes a four element process for evaluating risk-informed regulatory changes consistent with those safety principles. The key principles of integrated risk-informed regulatory decision-making are identified as: consistency with current regulation, consistency with defense-in-depth philosophy, maintaining sufficient safety margins, requiring increases in risk to be small and within the intent of the Commission's Safety Goal Policy Statement(4), and lastly monitoring the impact of the of the proposed changes using performance measurement strategies. The four element approach for evaluating risk-informed regulatory changes are identified as: defining the change, performing engineering analyses (traditional and PRA), defining the implementation and monitoring program, and submitting the proposed change.

RG 1.174 focuses on the use of PRA findings and risk insights as they relate to the regulatory decision-making process associated with proposed changes to a plant's LB. RG 1.174 indicates that some proposed licensing basis changes can be characterized as involving the categorization of SSCs according to their safety significance. An example is grading the application of quality assurance controls commensurate with the safety significance of equipment. Licensing Basis change requests for applications involving safety categorization will be evaluated according to the acceptance guidelines associated with each key principles and expectations presented in this regulatory guide, unless equivalent guidelines are proposed by the licensee. Since risk importance measures are often used in such categorizations, guidance on their use is provided in Appendix A of this regulatory guide. Other application-specific guidance documents address guidelines associated with the adequacy of programs (in this example, quality controls) implemented for different safety-significant categories (e.g., more safety significant and less safety significant).

Guidance on the use of risk importance measures, which are often used to support the categorization of SSCs, is provided in RG 1.174, Appendix A, *Use of Risk-Importance Measures to Categorize Structures, Systems, and Components with respect to Safety Significance.* Specific guidance on the categorization of SSCs according to safety significance is provided in RG 1.176. Of particular interest, it discusses grading the application of quality assurance controls commensurate with the safety significance of the equipment. Specific guidance on the categorization of SSCs according to safety significance is provided in RG 1.176. Of particular interest, it discusses grading the application of quality assurance controls commensurate with the safety significance of the equipment.

Regulatory Guide 1.176 describes an acceptable method for the categorization of SSCs at nuclear power plants in a manner commensurate with their safety significance (using an integration of insights from traditional engineering analyses, applicable qualitative considerations, and probabilistic analyses) and for applying appropriate quality assurance programs to each category of SSC. RG 1.176 focuses the risk-informed decision-making process discussed in RG 1.174 on proposed changes to the QA categorizations of certain SSCs. This RG provides considerable guidance relating to the categorization of safety significant SSCs, including:

- identification of system functions,
- system function safety-significant categorization,
- quantitative (importance measures) and qualitative safety categorization insights,
- identification and categorization of support systems,
- the use of an expert panel to perform an integrated assessment, and
- Performance monitoring, Operational feedback, and corrective actions.

This RG presents a categorization process that uses quantitative PRA results supplemented by traditional qualitative engineering evaluations to develop an initial categorization level based on the safety-significance of the respective SSC. Such a combined, integrated approach is necessary to utilize the strengths and avoid the inherent limitations in both probabilistic and traditional engineering analysis methodologies.

NUREG-0800, *Standard Review Plan*, Chapter 19, *Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance*, identifies the roles and responsibilities of organizations in the NRC that participate in risk-informed reviews of licensees' proposals for changes to the licensing basis of nuclear power plants. This chapter provides additional insights on the use and review of PRA in risk-informed regulatory decision-making and is directly applicable to the risk-informed categorization of SSCs consistent with their contribution to risk. Specifically, this chapter devotes substantial discussion to the adequacy and use of PRA, risk insights, and importance measures used in the risk-informed decision-making process. It also provides expanded discussion on the key principles, expectations, and elements discussed in RG 1.174. Further, Appendix C, *categorization of Plant Specific Elements with Respect to Safety Significance*, provides detailed guidance on the use, review, and expectations on the use of PRA and importance measures as they relate or contribute to the risk-informed decision-making. The guidance provided in this document is a logical extension of current NRC policy on the use of PRA in regulatory activities that are documented in the Commission's PRA policy Statement(4). It also notes that the decisionmaking process should use the results of the risk analyses in a manner that complements traditional engineering approaches, supports the defense-in-depth philosophy, and preserves safety margins; however, should not be the sole basis for regulatory decisions.

Draft NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*, Chapter 3, *Integrated Safety Analysis,* provides guidance on ISA methodology (including acceptance criteria for the quantitative and qualitative definitions of likelihood), hazard and accident analysis, IROFS, and an example procedure for risk evaluation.

The ISA is initially used to identify credible uncontrolled and unmitigated accidents that exceed intermediate- and high-consequence. Following this determination the ISA is also used to determine the IROFS that are needed to ensure that the probability of occurrence of those accidents that exceed intermediate- and high-consequence are unlikely and highly-unlikely, respectively. Draft NUREG-1520, Chapter 3, Appendix A, *Example Procedure for Risk Evaluation*, provides an approved methodology that could be used to categorize risk and demonstrate compliance with the performance requirements identified in §70.61. In this example a risk matrix is used to quantify risk in terms of risk index numbers (refer to Table A-3). These risk index numbers then provide a mechanism for the categorization of risk for the credible accident scenarios identified in the ISA. These risk index numbers are used to determine if the level of risk associated with an accident is acceptable or unacceptable (based on the performance requirements in §70.61). Further, these risk index numbers can be used to categorize IROFS commensurate with the reduction of risk attributable to the IROFS, as required in §70.62(d).

NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*[14], is included in this discussion because DOE has incorporated several aspects of the described classification methodology into DOE proposed approach to risk-informed categorization for the GROA. The methodology described in NUREG/CR-6407 presents an approach to the classification of components according to their importance to safety and was based on RG 7.10, *Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material*[16]. NUREG/CR-6407 and RG 7.10 present a method of classification of various components in transportation packaging. Each component of a transportation package is first identified as either ITS or not ITS. The components that are considered ITS are further categorized into one of the following three classification categories (depending on that components importance to safety):

- Critical to Safe Operation - These items include structures, components, and systems whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.

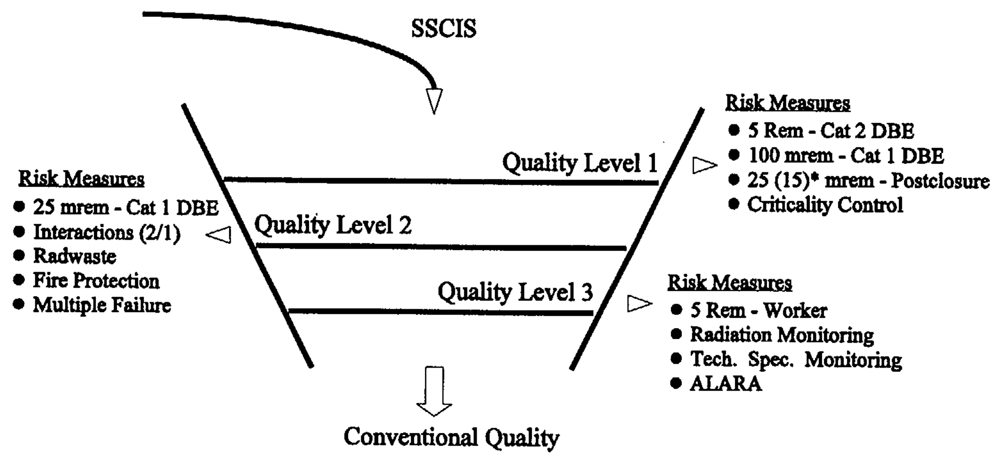
- Major Impact on Safety - These items include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of this type of item, in conjunction with the failure of an additional, could result in an unsafe condition.

- Minor Impact on Safety - These items include structures, components, and systems whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.

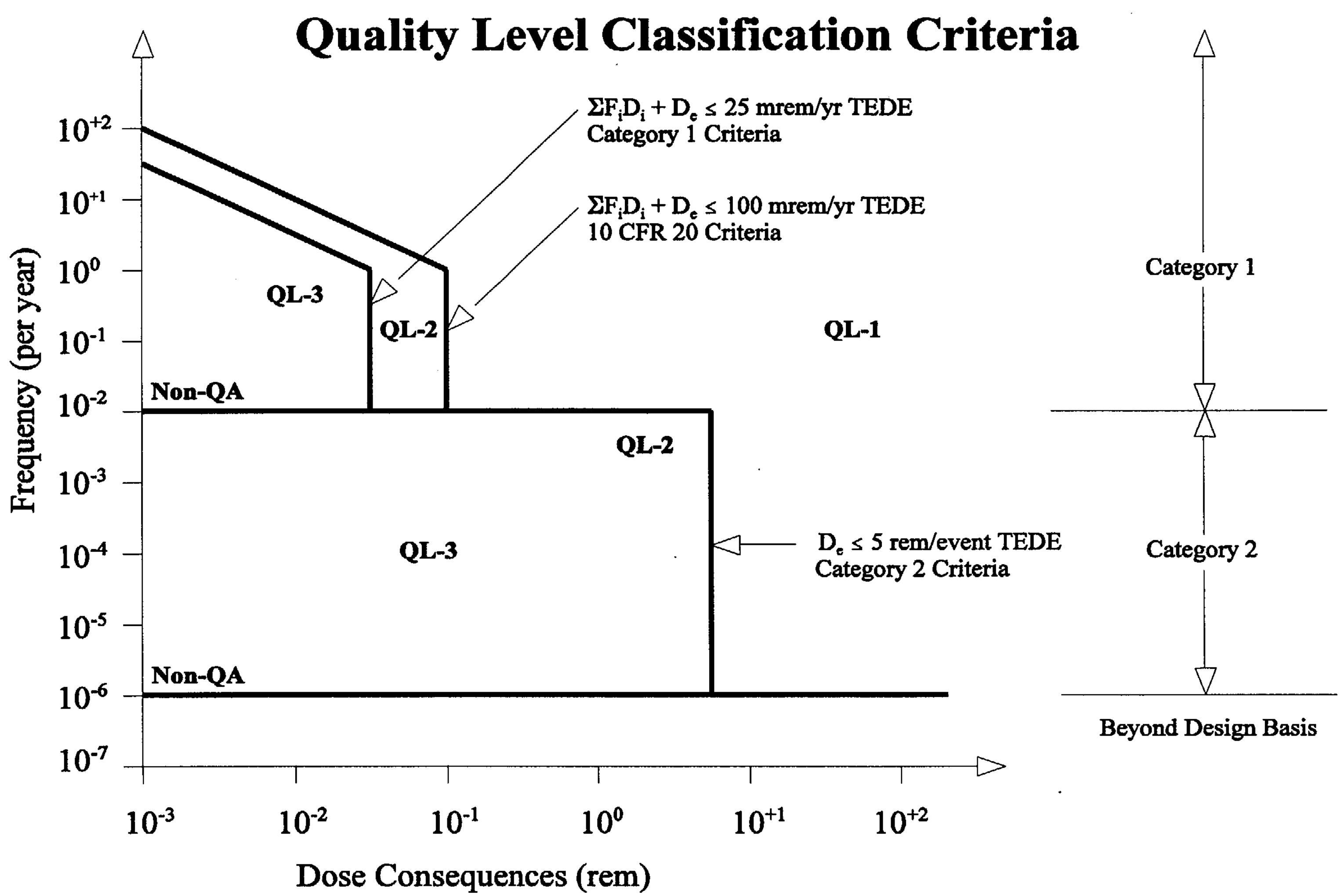
NUREG/CR-6407 provides a well defined list of typical components for each container type and assigns a primary safety function (containment, criticality control, shielding, heat transfer, structural integrity, and operations support) to each of these components. It then assigns an ITS classification category to each of the components based on the components safety function and container type.

**Appendix C**

# Classification Process




** EPA proposed limit is 15 mrem*




Where: $F_i$ = Annual Frequency (per year) of the ith event;
$D_i$ = Annual dose from ith event;
$D_e$ = Dose from bounding event (rem or rem per year); and
$\Sigma F_i D_i + D_e$ = The annual average radiation dose for the sum of anticipated releases and the bounding Category 1 DBE.

Figure 8-1. Quality Level Classification Criteria, RSS,
TDR-WIS-RL-000001 REV 04 ICN 01

**APPENDIX- E - Summary of Proposed Screening Criteria Identified in Procedure QAP-2-3, Rev. 10**

Each SSCs is pre-screened for importance to safety or waste isolation using the following criteria:

- Is the item directly or indirectly relied upon to provide an ITS function (confinement/containment, criticality control, shielding, heat transfer, structural integrity, or operations necessary for waste handling safety) for radioactive wastes received or handled?

Preclosure screening criteria for QL-1 SSCs are summarized as:

- Can failure of the item *directly result* in loss of waste package containment or criticality control for the spent nuclear fuel, high-level wastes, or other radioactive materials received for emplacement at the MGR?
- Is the item required to *prevent or mitigate* a Category 1 event sequence that could result in offsite doses of greater than or equal to performance objectives identified in 10 CFR 63.111(a)(1), 10 CFR 63(b)(1), and 10 CFR 20.1301(a)(1) [100 mrem]?
- Is the item required to *prevent or mitigate* a Category 2 event sequence that could result in offsite doses of greater than or equal to performance objective identified in 10 CFR 63.111(b)(2) [5 rem]?

Preclosure screening criteria for QL-2 SSCs are summarized as:

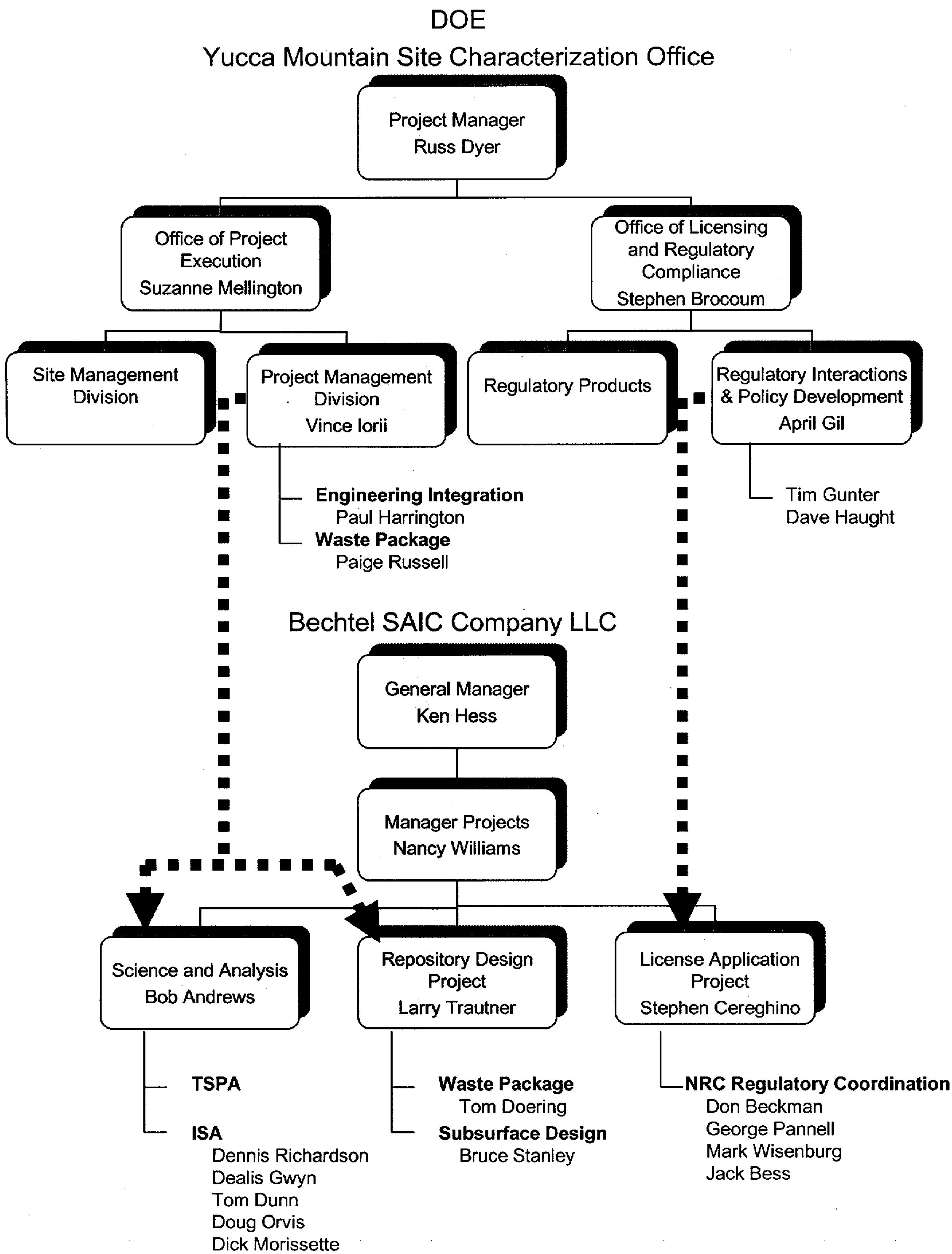
- Does the item function to provide control or management (i.e., collection and/or confinement) of site generated liquid, gaseous, or solid low-level or mixed waste?
- Does the item provide fire protection, fire suppression, or otherwise protect important to radiological safety or waste isolation functions of QL-1 SSCs from the hazards of a fire?
- As a result of DBE, could consequential failure of the item, which is not intended to perform a QL-1 radiological safety function, prevent QL-1 SSCs from performing their intended radiological safety function?
- 
- Is the item required to *prevent or mitigate* a Category 1 event sequence that could result in offsite doses of greater than or equal to performance objectives identified in 10 CFR 63.111(a)(2)?
- Is the item *in conjunction with an additional item or administrative control* (i.e., indirect impact), required to *prevent or mitigate* a Category 1 event sequence that could result in offsite doses of greater than or equal to performance objectives identified in 10 CFR 63.111(a)(1), 10 CFR 63(b)(1), and 10 CFR 20.1301(a)(1)?
- Is the item *in conjunction with an additional item or administrative control* (i.e., indirect impact), required to *prevent or mitigate* a Category 2 event sequence that could result in offsite doses of greater than or equal to performance objectives identified in 10 CFR 63.111(b)(2)?

Preclosure screening criteria for QL-3 SSCs are summarized as (occupational/monitoring/ALARA):

- Does the item function to provide an alarm to warn of significant increases in radiation levels or concentrations of radioactive materials?
- Does the item function to monitor variables to verify that operating conditions are within technical specification limits?
- Is the item used in MGR emergency response to provide prompt evacuation of personnel, or to monitor variables used in helping to determine the cause or consequence of DBEs (during post-accident investigations)?
- Does the item function as a part of the radiological, meteorological, or environmental monitoring systems required to assess radionuclide release or dispersion following a DBE?
- Is the item part of the design or design objectives for keeping levels of radioactive material in effluent to unrestricted areas as low as practicable during normal operations?
- Is the item required to limit onsite worker doses from normal operations and during Category 1 DBEs, including planned recovery operations, to less than 5 rem per year TEDE, 50 rem per year combined deep dose equivalent and committed dose equivalent to any individual organ or tissue (other than the eye), 15 mrem per year dose equivalent to the lens of the eye, or 50 rem per year shallow dose equivalent to the skin or any extremity?

**Appendix-F**
**References**


1. USNRC, 10 CFR 63, *Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada* (64 FR 8640);
2. USNRC, 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*;
3. USNRC, 10 CFR 70, *Domestic Licensing of Special Nuclear Material: Possession of a Critical Mass of Special Nuclear Material*;
4. USNRC, *The NRC's Final Policy Statement on Probabilistic Risk Assessment (PRA)*;
5. USNRC, Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*;
6. USNRC, Regulatory Guide 1.176, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance*;
7. USNRC, Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* (Revision 2);
8. USNRC, SECY-98-144, *White Paper on Risk-Informed and Performance-Based Regulation*;
9. CRWMS M&O, BC0000000-01717-0210-00001, Rev. 00, ICN 01, *Preliminary Preclosure Safety Assessment*;
10. CRWMS M&O, TDR-WIS-RL-000001 Rev. 04, ICN 01, *Repository Safety Strategy*;
11. USNRC, SECY-99-100, *Framework for Risk-Informed Regulation in the Office of NMSS*;
12. USNRC, SECY-95-265, *Response to August 9, 1995, Staff Requirements Memorandum Request to Analyze the Generic Applicability of the Risk Determination Process used in Implementing the Maintenance Rule*;
13. Not used.
14. USNRC, NUREG-0800, *United States Nuclear Regulatory Commission Standard Review Plan, Office of Nuclear Reactor Regulation*, Chapter 19, Use of PRA in Plant-Specific Risk-Informed Decision-Making: General Guidance;
15. USNRC, Draft NUREG-1520, *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility*
16. USNRC, NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*;
17. NUMARC 93-01, *Industry Guidance for Monitoring the effectiveness of Maintenance at Nuclear Power Plants* (Rev. 2);
18. USNRC, Regulatory Guide 7.10, *Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material*;
19. DOE/NRC Technical Exchange on Classification Analysis and Graded QA: *Risk-Informed Classification Process*; March 8, 2000;
20. CRWMS M&O, QAP-2-3, Rev. 10, *Classification of Permanent Items*;
21. Meeting minutes; and
22. CRWMS M&O, YAP-2.7Q, R01, ICN2, *Item classification an Maintenance of the Q-List* procedure.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