ENCLOSURE

SUMMARY HIGHLIGHTS OF THE
U.S. DEPARTMENT OF ENERGY/U.S. NUCLEAR REGULATORY COMMISSION
TECHNICAL EXCHANGE ON WASTE PACKAGE DESIGN
June 4-5, 2003
U.S. NUCLEAR REGULATORY COMMISSION
ROCKVILLE, MD

INTRODUCTION

On Wednesday, June 4, 2003, the U.S. Department of Energy (DOE) and U.S. Nuclear Regulatory Commission (NRC) staff began a two day Technical Exchange in Las Vegas, Nevada, in which the DOE presented 1) the current scope of waste package design information, 2) application of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) code to the waste package, 3) status of ongoing work to resolve key technical issues (KTI), and 4) an update on postclosure seismic evaluations as it applies to the proposed geological repository at Yucca Mountain. The KTI agreements cover information that NRC staff expects would be needed during the review of a license application (if submitted) to dispose of high-level radioactive waste at Yucca Mountain, Nevada, in accordance with the requirements of 10 CFR Part 63. The NRC goal of issue resolution during the pre-licensing period is to assure that DOE has assembled enough information on a given issue for NRC to accept a license application for review.

The detailed agenda for this meeting can be found in Attachment 1. The Technical Exchange included a audio connection between NRC in Rockville, Maryland, and the Center for Nuclear Waste Regulatory Analyses (CNWRA) in San Antonio, Texas. In addition to staff from DOE, NRC, the CNWRA and DOE's contractors, the meeting was attended by representatives from the State of Nevada; Clark County, Nevada; the Nevada Nuclear Waste Task Force; the Electric Power Research Institute (EPRI); the Nuclear Waste Technical Review Board (NWTRB); and the public. Attachment 2 contains the list of attendees who were present at the conference locations.

OPENING REMARKS

The meeting commenced with opening remarks by DOE and NRC. DOE indicated that the purpose of this technical exchange was to present: 1) the current scope of waste package design information in license application; 2) application of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) code to the waste package; 3) the status of ongoing work to resolve KTI's; and 4) update on postclosure seismic evaluations. The DOE has been encouraged by recent communications with the NRC in this area and looked forward to a productive interaction. The NRC stated that they were interested in seeking necessary and sufficient information to understand the DOE approach for a potential license application and if such information would be sufficient to make risk-informed regulatory decisions. Consistent with the detailed agenda found in Attachment 1, DOE presented their approach to the waste package design to address KTI agreements and related project specific requirements. In addition, the NRC provided DOE with feedback concerning their design approach for the proposed repository system components. The DOE presentations can be found in Attachment 3.

PRESENTATIONS AND DISCUSSION

On June 4, 2003 DOE presented its waste package design approach which included discussions on (1) preclosure design, (2) postclosure analyses, and (3) fabrication and closure methods. DOE led four presentations on day one of the technical exchange. The four presentations covered the following:

- Background - Waste Package Design
- Design for License Application
- Application of the ASME Code for YMP Waste Package
- Key Technical Issue (KTI) status

Background - Waste Package Design

A summary of the current status of the waste package and drip shield designs was presented. In addition, the functional and operational requirements for the waste package and drip shield were discussed. The level of design detail to be delineated in the License Application and the extent of applying the ASME B&PV Code to the waste package fabrication and design activities was clarified. A brief overview of the prototype objectives and activities was provided as well. A brief overview of the methodology being used to assess the waste package response to seismic excitation was also presented.

During the DOE background presentation, the NRC asked the DOE to explain the "no breach criteria" applied to the waste package. NRC stated that DOE needs an appropriate (probabilistic or qualitative) definition and basis for the "no breach" criteria. DOE, during the discussion, indicated that "no breach" really referred to their design goal that event sequences leading to the breach of a waste package will be Category 2 and beyond type event sequences which make the likelihood of waste package breach incredible. The DOE stated that the "no breach" criteria applied to the waste package during the preclosure period. Further, in response to the NRC staff questions regarding the role of the inner and outer barriers for the waste package, DOE indicated that the inner barrier's function is to provide the structural integrity required for the waste package and that the outer barrier's primary function is corrosion protection.

The NRC stated during the DOE presentation that there are two separate states involving drop accidents with the waste package and that these potential drops deal with the "no breach" criteria during the preclosure period. In addition to the waste package no breach criteria, which result in the removal of the waste package, certain drop sequences evaluated by DOE may result in unacceptable damage to the outer barrier (corrosion protection) requiring removal of the waste package because it may not be capable of meeting its long term performance objectives. DOE indicated that the waste package (WP) damage caused by preclosure handling required to take the WP out-of-service has yet to be established. In addition, WP damage that occurs after emplacement during the preclosure period will be assessed for retrieval prior to permanent closure and inspections will be used as a primary means to conduct such assessments (the type and frequency of inspections has not been established).

The NRC staff asked the DOE if they had looked at how invert degradation (i.e., failure of the invert to perform its intended function) would have on its current analysis. DOE stated that they have not evaluated a degraded/failed invert, but DOE said that they will be evaluating the potential effects of a failed invert in the system model. In addition to the potential affects a degraded invert could have on the current analysis assumption and results, the NRC staff reminded DOE that the 150 degree C temperature could be exceeded during the natural backfill condition. DOE stated that they were aware of that possibility.

In the presentation of the revised drip shield design the staff asked for the driver for the redesign of the drip shield. The DOE responded that revisions to the drip shield design were related to dynamic rockfall events.

Design for License Application

DOE provided an overview of the waste package fabrication and design information they plan to provide in the License Application. Specifically, DOE conveyed their intent to provide the details of the waste package design methodology and design bases for the purposes of (i) controlling and facilitating the evolution of the design, (ii) managing design changes, and (iii) facilitating project integration.

At the present time there is 1 (one) waste package design with 10 (ten) configurations that are being designed to accommodate the various waste forms that will be disposed of at the proposed repository. These different waste package configurations all share the same basic design (i.e., inner and outer barrier materials and lid designs); but exhibit variations in overall dimensions, weights, and basket design. Outline drawings for all 10 waste package configurations, including (i) components, materials, and nominal dimensions; (ii) lid and closure details; and (iii) internal configurations without assembly details, will be provided in the License Application. Detailed analyses assessing the performance of the waste package subjected to the requisite design bases loading conditions will be provided for 4 representative configurations [i.e., the 21 Pressurized Water Reactor (PWR) with absorber plate, 44 Boiling Water Reactor (BWR), 5 Defense High-Level Waste and DOE (DHLW/DOE) Spent Nuclear Fuel (SNF) Co-Disposal Short, and Navy Canistered SNF Long waste packages]. The assembly drawings for these representative waste package designs will be provided in the License Application. NRC staff indicated that the approach presented by DOE seemed reasonable.

Regarding the design of the WPs, NRC staff was concerned with aspects of the twist lock and welded design of the upper and lower trunnion sleeves onto the Alloy 22 outer barrier. Specifically, NRC staff asked DOE for their structural analysis regarding the ability of the trunnion welds to withstand the stresses associated with lifting the entire WP from these trunnions. Furthermore, NRC staff asked for future DOE analyses on residual stress measurements associated with solutionizing, quenching, and lifting a fully loaded WP prototype by the trunnion sleeves.

An overview of the preclosure event sequences that are currently being used to assess the performance of the proposed waste package design was provided. During the presentation, the NRC asked the DOE if they had evaluated nuclear power plant and the waste isolation pilot project overhead handling system operating experience for handling of the WP. DOE stated that they had considered industry operating experience and that all overhead handling systems would be designed as single-failure-proof systems. Further, DOE indicated that they have not established a credible design basis fire, however, they are evaluating a range of times and temperatures to determine which will be the most limiting for their WP "no breach" criteria.

NRC staff questioned whether DOE would conduct a re-evaluation of the WP and what analysis would be conducted to describe cladding performance during an event involving a fire. Further, the NRC staff questioned DOE concerning waste form alteration during a fire event sequence. DOE stated that they would replace any waste package damaged as a result of a fire. However, the NRC reiterated the need to evaluate any changes to the waste form as it would be repackaged and changes to the waste form may challenge the basic assumption about the waste form within a WP. DOE agreed that they would evaluate any potential change to the waste form as part of its analysis of the various times and temperatures of fires involving waste packaging.

Application of the ASME Code

An overview of the applicability of, and exceptions to, the ASME B&PV Code for the fabrication and design of the waste package was provided. DOE indicated their intent to design, fabricate and inspect the waste package inner barrier according to the requirements of the ASME B&PV Code, Section III, Division I, Subsection NC (for Class 2 Components). One notable exception to the use of the code, however, was that accident load failure will be evaluated to project specific criteria rather than the criteria as defined by the Code. The inner barrier will be fabricated to the requirements of ASME B&PV Code, Section III, Division 1, Subsection NC (for Class 2 Components) and is expected to be N-stamped. Waste package welds will be nondestructively examined by radiographic (RT) and liquid penetrant (PT) tests. The inner barrier will be pressure and helium leak tested at the fabricator facilities. Fabrication of the outer barrier will be consistent with the requirements of ASME B&PV Code, Section III, Division 1, Subsection NC (for Class 2 Components). However, the outer barrier will not be ASME Code stamped. The outer barrier base metal will be examined using ultrasonic testing (UT) in accordance with ASME Code, Section III, NB-2532.2. The outer barrier fabrication welds will be examined by RT, PT, and UT. The Alloy 22 outer barrier will be solution annealed and quenched during fabrication. The outer barrier surface finish will be controlled to as yet to be determined specifications. After the outer barrier is solution annealed, the fabrication weld will be examined using volumetric methods.

The inner barrier closure weld will be fabricated to the requirements of ASME B&PV Code, Section IX, and subsequently visually inspected (VT) and helium leak tested. Prior to closing the outer barrier, the inner barrier will be evacuated and backfilled with helium before the inerting plug is sealed. The inner, outer barrier closure weld will be evaluated using VT and eddy current (ET) tests. The outer barrier closure weld will be a full penetration structural weld that meets the requirements of ASME B&PV Code, Section IX, and subsequently VT, ET, and UT tested before and after stress mitigation.

The NRC staff questioned DOE regarding the specific stress mitigation method to be used during the outer lid fabrication process. DOE indicated that the primary stress mitigation methods to be used on the project were laser peening and low plasticity burnishing.

NRC staff questioned DOE regarding how selection and use of the Code would be used to meet the project performance objectives. DOE stated that the relevant project specific requirements will be used in design, fabrication, testing and inspection of waste packages. Portions of these project specific requirements can be met through the appropriate application of the ASME Code. NRC staff indicated that DOE will need to include any and all relevant ASME Code quality assurance (QA) requirements along with all relevant project specific QA requirements in its procurement of waste package prototypes and waste package for service at the repository. DOE recognizes this and will include quality assurance (QA) requirements for both the project specific requirements as well as the relevant requirements specified by the ASME Code. DOE recognizes and has acknowledged that waste package fabricators will have to meet the ASME Code QA requirements along with all relevant project specific QA requirements.

In addition, the NRC staff questioned DOE concerning how they envisioned selection and reconciliation of ASME Code versions would be completed. DOE indicated that they have plans to use the 2001 ASME Code with 2002 addenda. In response to NRC staff concerns on ASME Code selection, DOE plans to develop a road map for the selected Code as it relates to the version incorporated by reference in 10 CFR 50. Further, DOE, to the extent practicable, will list and evaluate any deviations/additions to the Code as necessary for its project specific application.

Key Technical Issue Status

An overview of the status of resolution for KTI Agreements CLST 2.01, 2.02, 2.03, 2.06, 2.08, 2.09, PRE 7.02, 7.03, 7.04, and 7.05 was presented. The status overview addressed the current progress of the work that has been completed to date, the work that has yet to be completed, and the anticipated completion date.

DOE discussed its path forward for CLST 2.03 which includes fracture toughness testing, literature surveys, and corresponding analyses to establish appropriate failure mechanisms for bTi-7, Ti-24, and Alloy 22. The NRC staff asked DOE how the path forward would establish the variation of Ti-7, Ti-24, and Alloy 22 mechanical properties due to welds, materials degradation processes, stress mitigation techniques on the lid closure weld, and other thermomechanical effects like the combination of cold work and heat. The NRC staff also questioned DOE regarding their use of the failure criteria with only their current mill annealed Ti-7 and Alloy 22 mechanical properties data. Furthermore, NRC questioned whether the Failure Assessment Diagram (FAD) demonstrated that the Tresca failure criterion bounds a fracture mechanics approach at specific areas on the WP and drip shield (DS) (e.g. WP longitudinal, circumferential, and closure welds; and the Ti-24 bulkhead weld to the DS side plate, etc.).

DOE presented the path forward for CLST 2.08 which indicated that additional work was needed to address the KTI by January 2004. The NRC staff questioned whether the additional work to be completed would evaluate the effects of rock fall and corrosion and if that work would be completed to meet the target schedule date. DOE's response to the NRC staff was that they are still exploring their options regarding any additional testing that would be conducted and that they were working diligently toward the target schedule for submission to the NRC.

The path forward proposed by DOE to address the adequacy of the finite element models used in PRE 7.02 to assess the responses of the drip shield and waste package to design basis accidents was presented by DOE. Although the proposed approach appears to adequately address concerns regarding mesh discretization, residual and differential thermal expansion stresses, strain rate effects, dimensional and material variability, and so on; the assumption that the waste package contents (i.e., the basket and waste form) mass can be lumped into the mass of the inner barrier may require further attention. Given that the basket and waste form represents approximately 35 to 40 percent of the overall mass of the waste package, the concern is that dynamic amplification effects may not be properly accounted for when employing the combined mass assumption.

The path forward proposed by the DOE to assess the variations in the allowable compositions of base and weld filler materials in PRE 7.03 will include a study to determine mechanical properties. The NRC staff asked about the selection of the ERNiCrMo-14 filler material instead of ERNiCrMo-10 that has been used in previous studies and identified as the filler metal in previous DOE documents. The NRC staff concern is that during solidification segregation of alloying elements in the weld metal results in an enrichment of molybdenum in the interdendritic regions. Enrichment of molybdenum promotes the stability of secondary phases that can alter mechanical properties and corrosion resistance. The DOE indicated that base and filler metal compositions that have acceptable mechanical properties may be evaluated in corrosion tests.

Waste Package Seismic Response

On June 5, 2003, DOE presented its analytical approach for addressing the waste package seismic response. DOE, in its presentation, stated that the vibratory ground motion effects and seismically induced rock fall impacts in the postclosure period were (1) appropriately de-coupled in the analysis and (2) the impacts were properly treated in accordance with predominate features and corresponding damage to the system components. In addition, DOE stated that their preliminary results indicate that the waste package would not be breached and rupture of the drip shield would not occur.

An overview of the work being performed by the DOE to assess the response of the waste package to seismic shaking was presented. The overview addressed (i) the assumptions, analysis inputs, and the overall methodology being used to perform the numerical approximation; (ii) the problem domain division; (iii) the finite element model construction, and (iv) the results obtained to date. The results obtained from seismically induced dynamic rock block impact loads were also discussed.

During the presentation, the NRC staff questioned the DOE model which assumed the combination of the inner barrier and waste form to be a single rigid body without independently evaluating the potential alteration/degradation of the waste form. DOE indicated that future analyses would evaluate the potential damage to the waste form. The NRC staff also raised concerns about the size of the gap between the outer and inner barrier in the axial and radial direction. DOE stated that the gap was approximately 4 millimeters (mm) in the radial direction and 30 mm in the axial direction. The gap is based on the difference in the thermal expansion coefficients between the outer barrier and the inner barrier up to 239 degrees C. The axial gaps will vary, depending on the length of the waste package.

In the waste package seismic response, NRC asked whether DOE is addressing CLST 3.10 (analysis of the rockfall and vibratory loading effects on the mechanical failure of cladding, as appropriate). DOE stated that the acceleration and the accompanying inputs are being assessed, and the results will be translated to the performance assessment staff to implement CLST 3.10.

NRC commented that the validity of all rigid body assumptions used in the seismic analyses needs to be justified. Specific components that have been modeled as rigid bodies that were identified by the staff include the invert, drift wall, waste form, part of the outer barrier and waste package inner barrier. NRC observed that the seismic analyses performed to date does not consider the presence of accumulated rockfall within the drift. DOE acknowledged that the seismic assessment work is still in the preliminary stages and the models will continue to be refined. NRC also noted that potential material degradation mechanisms and strain rate effects were not considered in assessing their effect on failure criterion used to determine waste package breach in the seismic analyses. Further, NRC staff requested that the maximum value for the following parameters calculated for the waste package seismic analyses should be documented: (i) strain rate, (ii) ASME Code stress intensity, (iii) equivalent plastic strain, and (iv) residual ASME Code stress intensity. DOE indicated that some results are available and they can accommodate, as appropriate, the request in future documents.

CLOSING REMARKS

In closing, DOE stated that they would provide the additional information required to resolve KTI agreements and that they would provide more detail information concerning the selection and use of the ASME Code. The NRC, in its closing statements, indicated that they looked forward to reviewing information needed to address KTI agreements and expected DOE to provide the NRC staff with an adequate road map to how it was using the Code.

PUBLIC COMMENTS AND FINAL DISCUSSION

Public comments were provided by representatives of the State of Nevada, the Nevada Nuclear Waste Task Force, and NWTRB. The representative from the State of Nevada emphasized that DOE should have procured the prototype waste package and should have conducted its evaluation of the prototype prior to LA submission. Currently, the prototype waste package is scheduled to be completed and delivered in 2005. The NRC staff stated that while they would prefer to have the results from DOE's evaluation of the prototype waste package, the NRC's main concern is the evaluation of waste package design and design methodology. A representative from NWTRB stated that he didn't understand why DOE was going to submit detailed information on four waste packages and less information on the remaining six. In addition, the NWTRB representative questioned what DOE's expectation of NRC would be for reviewing the waste package design details within the LA. DOE stated that the details of the four waste package design configurations will serve as a surrogates for the other six waste package design configurations at LA.

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