**Department of Energy**
Washington, DC 20585


QA: N/A

November 6, 2007

B. John Garrick, Ph.D.
Chairman
Nuclear Waste Technical Review Board
2300 Clarendon Boulevard, Suite 1300
Arlington, VA 22201-3367

Dear Dr. Garrick:

Thank you for your April 19, 2007, letter providing the Nuclear Waste Technical Review Board's (Board) views on the Office of Civilian Radioactive Waste Management (OCRWM) Program, as presented to the Board at its January 24, 2007, meeting in Las Vegas, Nevada. As always, I appreciate the opportunity to interact with the Board.

The Program remains on track to complete the key milestones and meet its strategic objectives, as I outlined in my presentation.

In your letter, the Board raised some additional questions and asked for clarification of some of our plans. The enclosure to this letter provides detailed responses to the Board's inquiries.

If you have any questions concerning this letter, please contact Claudia M. Newbury at (702) 794-1361.

Sincerely,

Edward F. Sproat, III, Director
Office of Civilian Radioactive
Waste Management

Enclosure

**Response to Nuclear Waste Technical Review Board Comments from January 24, 2007, Board Meeting**

1) **The Nuclear Waste Technical Review Board (Board) noted that it was "interested in obtaining information on how the design will conform to preclosure safety requirements (i.e., the event sequences that require analysis and the implications for dose from those events)."** The following discussion provides information on level of design detail and implementation of the Preclosure Safety Analysis (PCSA).

The U.S. Department of Energy (Department) is developing the design for its License Application (LA) to the level of detail necessary to assure the availability of structures, systems and components (SSCs) as modeled in the PCSA. The level of design information will conform to the U.S. Nuclear Regulatory Commission (NRC) staff guidance including HLWRS-ISG-02 PCSA – Level of Information and Reliability Estimation. This approach will include a greater level of design detail for Important to Safety (ITS)/Important to Waste Isolation (ITWI) components than there will be for Non-ITS/Non-ITWI components. For example, Piping and Instrumentation Diagrams, Ventilation and Instrumentation Diagrams, electrical single line diagrams, and logic diagrams for ITS/ITWI SSCs will include sufficient component information to allow modeling for reliability assessment. Another example is that structural design for the Canister Receipt and Closure Facility (CRCF), the Receipt Facility (RF), and Wet Handling Facility (WHF) will include design details such as lumped mass, multi-stick model with soil springs; peak accelerations at mass nodes; typical thicknesses and rebar patterns for shear walls, floor and roof slabs; typical details for penetrations; foundation (basemat) thickness and rebar patterns; assessment of building stability for sliding and overturning effects; and sizing of principal structural steel members. The results of the analyses will be included in the LA submittal scheduled for June 30, 2008. Schematics with sufficient mechanical handling equipment component detail to support reliability assessment of speed control, brakes, travel limits, and the ability to hold load on loss of power will be included. The PCSA will include reliability assessment, including human reliability, for such items as ITS Heating, Ventilation and Air Conditioning (HVAC), ITS electrical power, WHF pool and support systems, and movable shield doors in addition to the mechanical handling equipment. Design calculations and drawings will be sufficient to allow the NRC to verify that the PCSA is adequate.

10 CFR 63.111(c) requires performance of a PCSA of the geologic repository operations area. The PCSA calculations and analyses are developed, reviewed, and approved in accordance with the overall design control and configuration management procedures Coordination and integration between the PCSA analysts and design engineering is accomplished as an integral part of daily routine activities similar to the interface between the separate engineering disciplines within an engineering, project and construction organization.

The PCSA process is iterative and includes analysis of evolving design information, site characteristics, and operational features to evaluate the potential hazards, potential event sequences, and calculate the radiological consequences for operations of the geologic repository operations area. As the design and the PCSA progress, there is continuous feedback from PCSA analysts to designers regarding the safety functions of SSCs and target reliabilities being modeled in the PCSA. PCSA analyses are revised, as necessary, to maintain consistency with repository design. When the LA is submitted, the design and PCSA will be based on the same design information.

Interface activities are coordinated to ensure the design of the repository is consistent with the PCSA. This includes inputs from designers that are necessary to perform the preclosure safety calculations and analyses. The products developed by design engineering (e.g., project design criteria, system description documents, and drawings) and by the PCSA analysts (e.g., radiological hazards analyses and event sequence categorization) are closely coordinated between the respective organizations, and are subjected to procedurally required interface and interdisciplinary review before their issue.

The technical interface requirements between PCSA and design engineering are formally documented in the Preclosure Nuclear Safety Design Bases. This quality-affecting document provides the classification of systems, structures, and components ITS or not important to safety along with the associated safety function based on the results of completed event sequence analysis for each nuclear structure, and for subsurface areas and intra-site operations.

<u>Overview of PCSA Process</u>

In the PCSA required by 10 CFR 63.21(c)(5) and 10 CFR 63.112, an assessment of the safety of the geologic repository operations area is made and the ITS SSCs that are required to ensure that the credited safety functions can meet the performance objectives of 10 CFR 63.111 are identified. The four major portions of the analysis are (1) initiating events identification and event sequence development, (2) event sequence analysis and categorization, (3) radiological consequence, and (4) identification of SSCs ITS and specification of the nuclear safety design bases and procedural safety controls. The nuclear safety design bases for ITS SSCs and the procedural safety controls provide means to (1) prevent or reduce the likelihood of event sequences and (2) mitigate or reduce the consequences of event sequences.

Initiating events are considered only if they are reasonable (i.e., based on the characteristics of the geologic setting and human environment, and consistent with precedents adopted for nuclear facilities with comparable or higher risks to workers and the public (10 CFR 63.102(f)).

Initiating Events Identification and Event Sequence Development

To assess potential external and internal hazards, PCSA evaluates the site and uses descriptions of the repository facilities (surface and subsurface), SSCs, operational process activities, and characteristics of the waste stream to identify applicable hazards that may result in reasonable, credible, initiating events to be considered in further analyses. Examples of the internal hazard categories analyzed include, but are not limited to, collisions, drops, system failures (e.g., HVAC), floods, and fires. Master logic diagrams and process flow diagrams are being used to identify internal hazards and initiating events. Examples of external hazard categories analyzed include, but are not limited to, natural phenomena such as tornadoes and seismic events, and human activity such as aircraft crashes that could impart sufficient energy to be hazardous to a waste form.

Event Sequence Identification and Categorization

Potential event sequences are developed by safety analysis and evaluated based on the identification of credible potential external and internal initiating events. The event sequence analyses process quantifies (determines the overall probability or frequency) the sequences of events that lead to a potential radiological release or criticality. Event sequences are categorized in accordance with definitions of Category 1 and Category 2 event sequences in 10 CFR 63.2. Event sequences that have less than one chance in 10,000 of occurring during the preclosure period are screened out and categorized as beyond Category 2 event sequences.

Radiological Consequence Analyses

Analyses of radiological consequences of potential radionuclide releases and direct exposures from normal operations of repository surface and subsurface facilities, Category 1 event sequences, and Category 2 event sequences are performed as required by 10 CFR 63.111(c). Radiological consequences are calculated for workers and members of the public during normal operations and are added to the radiological consequences from the Category 1 event sequences to demonstrate compliance with 10 CFR 63.111(a) and (b).

For Category 2 event sequences, offsite public radiological consequences are evaluated for each Category 2 event sequence, individually. No worker radiological consequences are required to be calculated for Category 2 event sequences to demonstrate compliance with 10 CFR 63.111(b)(2).

Identification of SSCs ITS and Specification of the Nuclear Safety Design Bases and Procedural Safety Controls

The SSCs that perform safety functions credited in event sequence analyses and radiological consequence analyses are classified as ITS. The credited safety functions are documented in preclosure nuclear safety design bases.

For certain ITS SSCs, the PCSA specifies required reliability values for equipment or operator performance (or both) to ensure that event sequences involving those SSCs are prevented, the likelihood of occurrence is reduced, or the consequences are mitigated. The reliability specified by PCSA analyses is an engineering design requirement that is included in the preclosure nuclear safety design bases.

SSCs credited with preventing or ensuring that an event sequence is beyond a Category 2 event sequence are also identified as ITS with specific safety function design requirements.

**2) The Board stated that improvements should be made in the thermal management strategy that forms the basis for integrating waste management activities and requested clarification of how the Initial Handling Facility (IHF) fits into the Department's thermal-management strategy and the role of the IHF in general.** The following discussion provides additional information on the thermal management strategy and the role of the IHF.

With the change to the primarily canister-based approach relying on the use of Transport, Aging and Disposal (TAD) canisters, the Department plans on receiving up to 90% of the Commercial Spent Nuclear Fuel (CSNF) in TAD canisters loaded by the utilities. The Standard Contract (10 CFR Part 961) requires that the CSNF assemblies be a minimum of five years time out of reactor for classification as Standard Fuel; however, the Standard Contract does not impose any thermal limit on the CSNF to be accepted by Office of Civilian of Civilian Radioactive Waste Management (OCRWM). Selection of the CSNF assemblies to be delivered rests with the utilities.

Further, the Department's draft performance-based specification for the TAD canisters imposes temperature limits for protection of cladding at the utility sites, during transportation, and for the preclosure and postclosure periods at the repository. The performance-based specification imposes heat flux vs. canister-wall temperature limitations for the TAD canister at the time of emplacement. Other than these temperature limits, the thermal limits on CSNF that the Department must accept from the utilities are the NRC-approved individual assembly and total canister thermal limits from 10 CFR Part 71 Certificates of Compliance (CofC) for the TAD-based transportation systems (consisting of a TAD canister and its transportation overpack) that are determined by the TAD vendors.

Accordingly, with no set upper thermal basis and a lack of certainty of the specific thermal power of the TAD canisters, the Department is developing a thermal management strategy. It includes establishing thermal limits for handling of the TAD canisters and includes considerations for the design to allow for flexibility in the handling of the TAD waste stream to achieve thermal emplacement requirements.

There are several operational approaches, as part of the thermal management strategy, that are being planned for use at the repository. They include:

- Establishing a broad envelope for the emplacement process, that satisfies the TSPA constraints

- Allowing for the aging of TAD canisters to allow decay heat of the TAD canisters to achieve the thermal limits for emplacement

- Using low thermal power naval Spent Nuclear Fuel (SNF) and U.S. Department of Energy (DOE) High-Level Waste (HLW)/ SNF codisposal packages to blend the average thermal power in the emplacement drift to meet emplacement constraints

- Accounting for the decay of waste from its date of actual emplacement and the effects of ventilation during the preclosure period

As part of this strategy, the capability of the surface facilities is considered with respect to:

- Designing facilities that can meet potential thermal limits for receipt and handling of the TAD canister

- Accepting CSNF to meet DOE receipt rates

- Evaluating the capabilities of the facilities for the rates associated with closure of the waste package and subsequent emplacement in the proper thermal arrangement

- Evaluating the size of the aging facilities with respect to various waste streams

Each of the facilities has specific roles in the thermal strategy with respect to receipt of the TAD canisters, performing waste package closure, transporting TAD canisters to the aging facilities, and then returning them for handling and emplacement.

The IHF, in particular, receives and places the naval SNF canister into a waste package with subsequent closure, and has the capability to handle and close waste packages containing HLW, thus reducing the complexity of the Canister Receipt and Closure Facility. Waste packages are then placed into the transport and emplacement vehicle for emplacement in accordance with the thermal limits.

A thermal management study, using the above concepts to establish appropriate thermal emplacement limits, is currently underway to demonstrate the viability of a range of waste streams to meet the receipt and emplacement thermal limits for the repository.

A preliminary evaluation of proposed site operations, with these thermal constraints, has shown that there is considerable flexibility in the thermal limits for the waste packages and the thermal line load. Accordingly, there is considerable flexibility to receive waste streams of varying thermal characteristics while still meeting the preclosure and postclosure temperature and thermal limits used in the repository design and the 100-year preclosure operations period. Similarly, the Aging Facility has been shown to be of adequate size for a range of thermal powers associated with different waste streams. Since the thermal characteristics of the as-received waste stream is uncertain, the Department plans to perform a drift-by-drift analysis of the thermal loading to demonstrate preclosure and postclosure performance based on the as-received waste once the facility begins operations. This is similar to the nuclear industry's approach to conduct a core reload analysis of a reactor following refueling.

One of the results of the adoption of the TAD canister concept for simplifying repository waste handling operations was the segregation of functions to different waste handling facilities. The WHF is designed to receive CSNF and repackage it into TAD canisters. The CRCF are designed to receive disposable canisters (TAD, DOE SNF, and HLW) and transfer them into waste packages. The RF is designed to receive TAD canisters and dual-purpose canisters (DPC) and transfer them to aging overpacks to de-couple CSNF receipt from emplacement. The Initial Handling Facility is designed to receive disposable canisters (naval SNF and HLW) and transfer them into waste packages. The IHF reduces the operating load, complexity, and cost of the CRCF by processing all of the naval SNF. The IHF can process all 400 Naval Spent Nuclear Fuel Canisters in 17 years. The IHF also has the ability to process HLW canisters. There is a 300 ton crane in the IHF that is required to handle the transportation cask in which the naval SNF will be shipped. The CRCF design only requires a 200 ton crane with a lower maximum hook height than the IHF to handle the waste that it will receive, which has resulted in a less expensive and less complex design for the three CRCF. Also, since processing naval SNF in the CRCF would require removal of other waste forms from staging areas to ensure criticality safety, elimination of the naval SNF from the CRCF mitigates the resultant operational delays associated with clearing the CRCF of other waste forms prior to handling naval SNF, allowing increased throughput for the CRCF.

In the IHF, the radiation source terms from naval SNF and high-level radioactive waste are sufficiently low that mitigation is not required to meet site boundary dose limits. All other waste forms to be handled at the repository require mitigation to meet site boundary dose limits. Consequently, the IHF does not require the confinement function of the other waste handling facilities and can be constructed primarily from structural steel. This allows the IHF to be constructed considerably faster than the other waste handling facilities which are primarily built of reinforced concrete. The current schedule is for the IHF to be completed a year before CRCF 1. This period will be used to demonstrate equipment operations and refine operating procedures for cask handling, canister transfer, and waste package loading, closure and loadout. Lessons learned in the year will be applied to the other handling facilities. The IHF provides for an improved throughput of Naval SNF, while simplifying operations in the CRCF.

Therefore, throughput is improved for Naval Spent Nuclear Fuel and for waste going through the CRCF.

**3) The Board requested information on experience gained from safety and facility maintenance in the Exploratory Studies Facilities (ESF) could be applied to subsurface repository design and operations. The following information may be helpful in this regard.**

In the summer and fall of 2006 the Department conducted two workshops with outside experts in underground construction and environmental safety and health. A hazard analysis of current ESF operations and construction practices was also completed, and the result of these two efforts was the development of an Underground Safety and Health Requirements Document (DOE/RW-0586), issued in January 2007. This document was intended to be applied to continued site operations until construction authorization. Some specific experience gained from safety and facility maintenance in the ESF includes the following:

- Nominal excavation airflow design volumes are based on the 150 ft/min velocity established during ESF construction

- Drift orientation (azimuth 252) based on post excavation ESF information

- Measurements of steel set loads indicate no evidence of long-term time-dependent effects. The rock at the repository host horizon demonstrates a good self-supporting capacity, rock bolts with wire mesh are an adequate ground support system, and steel sets with lagging are a very conservative ground support system

- The two ground support systems, namely: the friction-type expandable rock bolts and cast-in-place concrete liner installed in the heated drift, performed very well while subjected to up to 200 degree C temperatures, supporting the use of that type of rock bolt in the ground support system proposed for emplacement drifts

- Lithophysal rock exposure in the ESF, particularly in the ECRB cross drift, revealed all the challenging rock mechanical aspects of testing the lithophysal rock, and the importance of integrating field activities such as mapping, in situ measurements, and field observations in the process of characterizing the lithophysal rock mass thermo-mechanical performance

- Use of a blowing system to deliver fresh air directly to the TBM face, so workers at the face will be in cleaner air. (An Exhaust system was used during ESF operation, intake air went to the working face through the TBM tunnel, where the airflow picked up a lot of dust in the tunnel)

- Use of 1,000-ft flexible tube segments for minimizing air leakage. (Compared with 20-ft steel duct segments used in ESF, this eliminates majority of the vent-line joints that are potential source of air leakage)

- Covered muck cars (instead of conveyer used in ESF, which was a major source of dust).

**4) The Board encouraged the DOE to evaluate surface-facility designs and operational concepts for opportunities to reduce the number of times waste is handled.** The Board suggested that DOE should, for example, assess the need for and, to the extent practicable, limit the size of large aging pads called for in the current surface facilities design. The current status of the repository design as modified to accommodate the TAD is described below.

The current design of the surface facilities has resulted in a significant reduction in the number of times the waste is required to be lifted and handled as compared to the previous repository design. As an example, in the former Dry Transfer Facility a loaded waste package was lifted by a crane a minimum of three times, and as many as six times, during handling. In the current design of the surface facilities, all crane lifts of a loaded waste package have been eliminated.

The current 21,000 MTHM capacity of the aging pads uses Total System Model delivery predictions that are based on a waste package thermal limit at emplacement of 11.8 kW. Evaluations are currently underway to determine the effect of increasing the thermal limit at emplacement on the postclosure analyses. If the Department chose to increase the waste package thermal limit at emplacement, more TAD canisters could be directly loaded into waste packages, thereby reducing the required capacity of the aging pads. Any such change would necessitate discussion with the NRC.

As discussed above, as part of the thermal strategy, the aging pads are a part of the overall program to handle the wide variability of the potential waste streams to be received. Evaluations of waste stream in the past with different waste package designs and thermal emplacement constraints identified that the 21,000 MTHM capacity (approximately 2500 "spots" for TAD canisters or dual-purpose canisters (should DOE accept them) may be needed to allow for thermal decay. Current evaluations suggest that the needed capacity of the aging facilities could possibly be reduced by as much as 50%, depending on the thermal characteristics of the waste stream and the emplacement strategy employed, even if emplacement of the lower thermal waste is deferred until the end of the emplacement period. Included in this consideration for this sizing is queuing of waste based on the throughput capability of the facilities. The uncertainty of the waste stream thermal characteristics and the thermal capability of the TAD canister causes the repository to retain the facilities' capacity of 21,000 MTHM as part of the current design. As the design matures, with respect to the throughput capability of the facilities, the TAD thermal capabilities as identified by the vendors, emplacement strategies during preclosure for postclosure acceptance are accepted by the NRC, and the characteristics of the waste stream become more certain, the Department will re-evaluate the need for the capacity of the aging facilities and adjust their capacity as necessary to support operations. Aging capacity will be developed in phases.

**5) While not directly discussed at the January meeting, the Board urged the DOE to evaluate the possible direct disposal of DPCs in Yucca Mountain (YM).** The Board suggested that the DOE should clarify its position regarding criticality and burn-up credit as part of an assessment of the feasibility of direct disposal of DPCs. DOE's plans with respect to DPCs are described below.

Should the Department accept DPCs, the direct disposal of existing DPCs is not planned and disposal of DPCs is not included in the LA. DOE does not currently plan that DPC disposal would be included in any amendments to the LA until the DPCs have been analyzed for postclosure criticality and other considerations. Several existing DPC designs rely on internal geometry and flux traps as well as neutron absorbers. During the postclosure period, internal geometry is lost due to material degradation, therefore credit is not taken for geometric controls. Also, any neutron absorber currently in DPCs may not have the same high level of corrosion resistance as the neutron absorber being specified for the TADs (borated stainless steel). If future analyses determine that direct disposal of DPCs is feasible, then the Department could propose an amendment to the license. However, currently the plan is to cut open DPCs in the WHF and transfer the fuel assemblies from DPCs to TADs. DOE intends to include burn-up credit in its evaluation of postclosure criticality and would expect burn-up credit to be considered in any direct disposal DPC analysis performed in the future.

**6) The Board also requested an explanation of the technical basis for the selection of borated stainless steel as a neutron absorber in TAD canisters.** The technical basis is described below.

The Department completed a comprehensive sensitivity study as documented in the calculation, "Evaluation of Neutron Absorber Materials Used for Criticality Control in Waste Packages" (CAL-DS)-NU-000007). This calculation looked at a range of absorber specifications, concentrations and geometric arrangements. The final recommended neutron absorber material for the TAD was borated stainless steel with a boron loading of 1.16 wt % at a minimum thickness over 10,000 years of 0.6 cm. The basis for the recommendation, as taken directly from the calculation, is as follows:

- Commercial experience with fabricability, commercial availability, and neutronics experience of absorber materials containing boron is much broader than with the Ni-Gd alloy. Also, ceramic based materials (B4C) would need special cladding and welding to ensure that they remain in place over long time periods of corrosion

- There are a relatively large number of criticality benchmark experiments with boron absorber in geometries representative of the TAD than with Gd absorber

- Expected corrosion rates for the Ni-Gd alloy and the borated stainless steel using powder metallurgy are expected to be relatively similar for the in-package pH ranges expected in the repository provided with boron loading is kept a relatively low levels

- A minimum absorber plate thickness of 0.6 cm with a credited boron loading of 0.87 wt% with natural boron provides a loading curve that is nearly identical to the proxy TAD configuration loading curve. This is the minimum thickness required after being subjected to 10,000 years of corrosion

- Further, additional corrosion testing of borated stainless steel have corroborated the expected corrosion rates.

**7) The Board expressed concern that, while technical interaction between DOE and the nuclear utilities is ongoing, it is not apparent to the Board that this dialogue includes all key issues warranting coordination within a successful waste management system.**

The Department believes that its current level of dialogue with nuclear utilities has been both appropriate and constructive. For example, the Department's discussions with both utilities and cask vendors has led to the successful development of the Preliminary Performance Specification for the canister. The Department also has continuing interactions with utilities on numerous topics including of nuclear operations, licensing, emergency preparedness, training, and configuration management. Additionally, the Department, with the assistance of the Electric Power Research Institute and the Nuclear Energy Institute, is working with a group of utilities to obtain additional data on spent nuclear fuel characteristics that it believes will be helpful in efforts to obtain an NRC license for the construction and operation of repository at YM.

The Department intends to expand the ongoing dialogue with nuclear utilities on additional issues as the program progresses into the licensing phase of the repository and beyond.

**8) The Board expressed concern that DOE has assigned postclosure planning responsibility to the Office of the Chief Scientist (OCS), while preclosure planning responsibility has been assigned to the Office of the Chief Engineer (OCE).** The Board indicates that it has not observed a systematic or comprehensive linking of these two components or recognition by DOE of the interdependencies of important repository design and operating elements (e.g., thermal management).

The Environmental Protection Agency, in 40 CFR 197, and the NRC, in 10 CFR 63, provide different standards and expectations with regard to pre- and post-closure safety.

The Department's organizational structure is reflective of these differences in requirements and associated areas of expertise. However, the Department has long recognized that these topics are not totally divorced from each other and require close coordination of activities and clear definition of interfaces. The OCE has been given responsibility for the development and control of top-level requirements documents including management of the technical change control process. This ensures consistent assignment and integration of requirements throughout the program, establish single point accountability for managing changes within the program, and develop a clearinghouse for integration at the management level.

Currently, the interface between postclosure activities performed under the direction of the OCS by the Lead Laboratory (LL), and preclosure activities performed under direction of the OCE by Bechtel SAIC Company, LLC (BSC), is managed through several processes and management actions, including the following:

- The LL and BSC have established a formal process for information exchange. Interface Exchange Drawings (IEDs) have been issued to document and control the exchange of information across the organizational boundary between preclosure functions (e.g., repository engineering, design, operations, and preclosure safety and criticality analyses) and post-closure and scientific investigation functions (e.g., post-closure performance modeling and assessment, post-closure criticality analyses, and site-specific geotechnical, environmental, meteorological, and seismic investigations). Control of the exchange of information across this boundary is necessary to ensure compatibility between the design of systems, structures and components and interfacing processes and scientific analyses.

- An additional document that ensures consistency and integration between the LL and BSC design is the Postclosure Modeling and Analysis Design Parameter Report, which augments the IEDs by documenting a review of Analysis and Model Reports to identify parameters and constraints to design (e.g., design bases that must be met by the design). These constraints to design are included in the design requirements documents, thus assuring that postclosure modeling and performance analyses bases are being met.

- The contractors exchange review copies of in-process technical documents for inter-contractor review if there are impacts on either the content of an IED or the Post Closure Modeling and Analysis Design Parameter Report.

- A joint management review in the Technical Review and Management Board is performed by the LL and BSC on any proposed changes to the IEDs or the Post Closure Modeling and Analysis Design Parameters Report.

- A regularly scheduled Subsurface Integration Meeting is hosted by BSC engineering with Department and LL attendees. The purpose of the meeting is to provide a means to discuss specific issues that affect both preclosure and postclosure work.

The need for integration between offices is not limited to just the OCS and the OCE, particularly with regard to the Board's example of thermal management. The OCS, OCE, and Office of Waste Acceptance and Management are jointly developing the Thermal Management Strategy discussed above. An integrated team evaluated potential waste streams and associated parameters, and set bounds for the thermal envelope in the facility preclosure operations while meeting the initial conditions for the TSPA for postclosure. This was a significant integration effort that is now being implemented. Those parameters, defined in the study are being included into the control documents described above, for implementation into the ongoing design and TSPA analyses.

**9) The Board suggested that DOE monitor the upcoming rulemakings by the Department of Homeland Security and Pipeline and Hazardous Materials Safety Administration and the Federal Motor Carrier Safety Administration to ensure that DOE's approach is consistent with new regulations.**

Current and proposed rulemakings and legislation related to hazardous materials transportation security may impact the Department's system planning, and will be closely monitored by DOE. Accordingly, the Department will continue to closely follow developments in this area.

**10) The Board discussed the importance of developing more-realistic estimates of seismic ground motion for both preclosure and postclosure periods and noted its support for scientific and engineering activities aimed at developing such realistic estimates.**

During the last year work has been ongoing to refine seismic analyses. To address the evolution of the area where surface facilities will be sited, ground motions for design and preclosure safety analyses have been updated. In updating these ground motions, an alternate approach to incorporating site response has been implemented that results directly in a site-specific seismic hazard curve. In addition, reasonable limits to extreme (very low probability) ground motions at YM are directly incorporated. Limits are assessed both on the basis of geologic evidence that indicates a level of ground motion that has not been experienced at the site and on an evaluation of earthquake source parameters that are consistent with the geologic setting of the site.

Analyses and modeling of seismic consequences during the postclosure period are being updated to take into account the transportation, aging, and disposal canister concept and to evaluate performance for the period after 10,000 years. As part of this work, response to seismic loading is being assessed for additional states of degradation and failure of the engineered barrier system and for the effects of multiple seismic events.

**11) The Board considers the question of $^{36}Cl$ measurements an outstanding issue whose resolution could greatly enhance confidence in understanding fluid flow within YM.**

The Cl-36 studies can be viewed as consistent in one important aspect which is that the studies conducted to date consistently indicate that fast pathways, as indicated by bomb-pulse Cl-36 are either rare or non-existent. This is consistent with the way the unsaturated zone is modeled in process models and the TSPA, in which a small percentage of fast pathways are included in the models for unsaturated zone flow. Links to the completed reports on the work conducted by DOE investigators, including conflicting results and interpretations, were provided in a presentation at the January 24, 2007 Nuclear Waste Technical Review Board meeting.

**12) The Board expressed concern that budget constraints in fiscal year (FY) 2007 and the elimination of funding for this purpose in OCRWM's budget request for FY 2008 will negatively affect the continuation of the Science and Technology (S&T) program.**

Funding constraints will cause the Department to reduce or eliminate funding for the independent S&T program. The Department is investigating other avenues, such as the DOE Office of Science and cooperative research programs, to maintain the capability to investigate new and unproven techniques and technologies.