THE U.S. NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS REVIEW OF THE U.S. DEPARTMENT OF ENERGY'S KEY TECHNICAL ISSUE AGREEMENT RESPONSES RELATED TO THE POTENTIAL GEOLOGIC REPOSITORY AT YUCCA MOUNTAIN, NEVADA: CONTAINER LIFE AND SOURCE TERM 1.12, 1.13, 6.02 ADDITIONAL INFORMATION NEED–1; AND 6.03 ADDITIONAL INFORMATION NEED–1

1.0 INTRODUCTION

By letter dated December 9, 2003, the U.S. Department of Energy (DOE) submitted a report titled "Technical Basis Document No. 6: Waste Package and Drip Shield Corrosion" (Bechtel SAIC Company, LLC, 2003) to satisfy the informational needs of numerous key technical issue (KTI) agreement items pertaining to the waste package and drip shield materials at the potential repository at Yucca Mountain, Nevada. The information was requested by the U.S. Nuclear Regulatory Commission (NRC) during technical exchanges in September 2000, February 2001, July 2001, August 2001, and September 2001. Specific agreements addressed in this NRC review of the information provided by DOE in the Technical Basis Document include Container Life and Source Term (CLST) Agreement CLST.1.12, 1.13, 6.02 Additional Information Need (AIN)–1, and 6.03 AIN–1 (Schlueter, 2000). CLST.1.12 was addressed in Appendix G of Technical Basis Document Number 6 (TBD-6). CLST.1.13 was addressed in Appendix B of TBD-6. CLST.6.02 AIN-1 and CLST.6.03 AIN-1 were addressed in Appendix H of TBD-6

2.0 AGREEMENTS

Wording of these agreements is provided in the subsequent paragraphs.

CLST.1.12

"Provide the documentation for Alloy 22 and titanium for the path forward items listed on slides 34 and 35: (1) Qualify and optimize mitigation processes, (2) Generate stress corrosion cracking (SCC) data for mitigated material over full range of metallurgical conditions (base metal, as-welded, welded and aged, cold worked), (3) New vessels for Long Term Corrosion Test Facility (LTCTF) will house many of the SCC specimens (4) Continue slow strain rate testing in same environments as above, specimens in the same range of metallurgical conditions (5) Determine repassivation constants needed for film rupture SCC model to obtain value for the model parameter '*n*', (6) Continue reversing direct current potential drop crack propagation rate determinations in same types of environments and same metallurgical conditions as for slow strain rate testing and LTCTF tests, (7) Evaluate SCC resistance of welded and laser peened material versus nonwelded unpeened material, (8) Evaluate SCC resistance of induction annealed material, (9) Evaluate SCC resistance of full thickness material (with welds) obtained from demonstration prototype cylinder of Alloy 22 (mock-up site recommendation design). DOE will provide the documentation in a revision to AMRs (ANL–EBS–MD–000005 and ANL–EBS–MD–000006) prior to LA."

CLST.1.13

"Provide the data that characterizes the distribution of stresses due to laser peening and induction annealing of Alloy 22. DOE will provide the documentation in a revision to AMR (ANL–EBS–MD–000005) prior to LA."

CLST.6.02

"Provide additional justification for the use of a 400-ppm hydrogen criterion or perform a sensitivity analysis using a lower value. DOE stated that additional justification will be found in the report, 'Review of Expected Behavior of Alpha Titanium Alloys under Yucca Mountain Condition' TDR–EBS–MD–000015, which is in preparation and will be available in January 2001."

CLST.6.02 AIN–1

"1. Provide better justification to verify the critical hydrogen concentration chosen is a realistic and representative value for the onset of cracking in Titanium Grade 7. An evaluation of the critical hydrogen concentration for Titanium Grade 24 should also be given.

2. Provide an evaluation of possible detrimental effects of the fluoride on hydrogen uptake and its effects on the critical hydrogen concentration and subsequent cracking.

3. Provide results for the Titanium Grade 24 structural drip shield components."

CLST.6.03

"Provide the technical basis for the assumed fraction of hydrogen absorbed into titanium as a result of corrosion. DOE stated that additional justification will be found in the report 'Review of Expected Behavior of Alpha Titanium Alloys under Yucca Mountain Condition' TDR–EBS–MD–000015, which is in preparation and will be available in January 2001."

CLST.6.03 AIN–1

"Provide an evaluation of the possible detrimental effects of fluoride on possible hydrogen uptake rates, as well as enhanced corrosion resulting in higher than currently estimated hydrogen generation rates."

3.0 RELEVANCE TO OVERALL PERFORMANCE

CLST.1.12 and 1.13 Agreements are related to the stress corrosion cracking of the proposed waste package outer containers. The waste package, composed of the containers and the waste forms, is the primary engineered barrier controlling the release of radionuclides from spent nuclear fuel and high-level waste glass. Because corrosion processes, promoted by the presence of an aqueous environment contacting the surface of the containers, will be the primary cause of container failure under undisturbed conditions, the mode and rate of corrosion needs to be evaluated to determine container lifetimes. Corrosion processes potentially important in the degradation of the engineered barriers include humid-air and uniform aqueous corrosion, localized (pitting, crevice, and intergranular) corrosion, microbially influenced corrosion, stress corrosion cracking, and hydrogen embrittlement. Fabrication processes, such as cold working, welding, and postweld heat treatments, may alter the corrosion resistance of the waste package materials.

CLST.6.02 and 6.03 Agreements are related to the hydrogen embrittlement of the proposed drip shields that will be emplaced prior to closure of the potential repository. Drip shield performance is an important factor regarding safety because the drip shields are incorporated into the design of the engineered barrier system to limit the amount of water contacting the waste package as a result of dripping and damage to the waste package from rockfall. Initiation of aqueous corrosion of the waste packages depends on the deliquescence of dust or the contact with seepage water. The presence of drip shields will delay the contact of seepage water with the waste package surface, resulting in a significantly longer container lifetime. In addition, once the containers are breached, the amount of water available for the dissolution of spent nuclear fuel and high-level waste glass and advective transport of the released radionuclides could be limited, even by the presence of a partially damaged drip shield.

The NRC performed a risk insights analysis that indicates stress corrosion cracking of the waste package closure welds has a medium significance to waste isolation (NRC, 2004). Stress corrosion cracks that penetrate through the waste package outer containers could allow water to contact the waste forms and lead to the release of radionuclides; however, the transport of water and the release rate of the radionuclides may be restricted by the small apertures of the cracks. The integrity of the drip shield also has a medium significance to waste isolation because, while intact, the drip shield will limit the quantity of water contacting the waste packages and waste forms and limit the formation of aggressive environments on the waste package surfaces. The geometry of cracks in the waste package and the drip shield may be altered by applied loads from external events such as rockfall, seismic events, and drift degradation.

4.0 RESULTS OF THE NRC REVIEW

CLST.1.12, 1.13, 6.02, and 6.03 Agreements are included in the integrated subissue for degradation of engineered barriers. These agreements resulted from a staff review of DOE documentation that is consistent with Review Method 2 in Section 2.2.1.3.1.2 of NRC (2003). The NRC review of the response for these agreements also was conducted in accordance with the aforementioned review method. This review method includes evaluation of the sufficiency of the experimental data used to support parameters used in conceptual models and process-level models.

4.1 CLST.1.12

The focus of CLST.1.12 was to ensure that assessment of stress corrosion cracking as a potential degradation mode for the drip shields and the waste packages considered the range of materials, metallurgical conditions, fabrication processes and the variations in chemistry of the solutions contacting the waste package and drip shield surfaces. The DOE response (Bechtel SAIC Company, LLC, 2003) identifies that postweld heat treatment will be used to mitigate the residual tensile stresses at the surface of the Alloy 22 disposal container. In addition, mechanical processes (i.e., laser peening or controlled plasticity burnishing) will be used to mitigate residual tensile stresses in the waste package closure welds. The technical basis document indicates the effect of residual stress mitigation using laser peening is demonstrated for welded Type 316 stainless steel. When tested in a boiling magnesium chloride solution, stress corrosion cracks of the stainless steel did not propagate into the laser peened area of the test sample. Although the occurrence of stress corrosion cracking may be precluded by the formation of compressive surface stresses through solution annealing and stress mitigation processes, DOE residual stress calculations on the 21 pressured water reactor mockup waste package container outer cylinder indicate tensile residual stresses may be present on some locations on the inside and outside surfaces (Structural Integrity Associates, Inc., 2002).

The DOE abstraction for stress corrosion cracking of the Alloy 22 waste package outer container is conservative. For example, the threshold stress intensity for stress corrosion cracking of Alloy 22 is 3 to 29 $MPa\cdot m^{1/2}$ [2.7 to 26.4 $ksi\cdot in^{1/2}$] with a mean value of 11 $MPa\cdot m^{1/2}$ [10 $ksi\cdot in^{1/2}$] and is based on crack tip blunting by passive corrosion. Experimental measurements of crack propagation suggest crack propagation rates decrease with time, and crack growth may be arrested at these stress intensities (Andresen, et al., 2004). Moreover, the initiation of stress corrosion cracking of Alloy 22 has been observed only in tests using either cyclic loading or constant straining with high applied potentials (Andresen, et al., 2004; King, et al., 2002). No stress corrosion cracking of Alloy 22 has been observed for constant deflection conditions in simulated groundwater under acidic and alkaline conditions (Fix, et al., 2004a).

The DOE indicated the drip shield will be emplaced in a stress-mitigated condition (Bechtel SAIC Company, LLC, 2003). The stress corrosion cracking initiation threshold stress criterion for the drip shield design is 50 percent of the at-temperature yield strength (Bechtel SAIC Company, LLC, 2003). Although DOE response to CLST.1.12 did not include additional information for the titanium alloys used to construct the drip shields, using 50 percent of the at-temperature yield strength for the initiation of stress corrosion cracking is likely to be a conservative design criterion. Stress corrosion cracking of Titanium Grade 7 has been observed under both cyclic and static loading conditions (Andresen, et al., 2001; Young, et al., 2003). Although stress corrosion cracking of Titanium Grade 12 was observed for constant deflection conditions, no cracking was observed for Titanium Grade 7 when tested using identical conditions (Fix, et al., 2004b).

Although the staff considers this agreement closed, DOE should consider the following comments:

- Although conservative, confirmatory tests would lend additional support to the models for stress corrosion cracking of the Alloy 22 waste package outer container and the titanium alloy drip shield. A range of microstructures could result from the fabrication and closure of waste packages. The stress corrosion cracking susceptibilities of stainless steels and nickel-base alloys are known to be dependent on environmental and metallurgical conditions. Fabrication processes that can result in metallurgical changes and the formation of residual stresses also can promote stress corrosion cracking. These confirmatory tests should provide assurance the threshold stresses, threshold stress intensities, and parameters used to model crack propagation are appropriate for the range of alloy compositions, metallurgical conditions, fabrication processes, and postweld stress mitigation methods used in the construction of the waste package and the drip shield.

- Data to confirm the validity of the stress mitigation processes for limiting stress corrosion crack initiation may support DOE's model abstractions. These data should confirm the calculated and measured stress profile and confirm the beneficial effects of the postclosure weld stress mitigation processes for controlling crack initiation for Alloy 22. In addition, the effects of machining the inside surface of the outer Alloy 22 cylinder, necessary to provide a radial gap 0–4 mm [0–0.16 in] between cylinders (Bechtel SAIC Company, LLC, 2001), on the resulting residual stress profile established by solution annealing should be evaluated.

Based on the NRC review of DOE response to CLST.1.12 Agreement in accordance with methods discussed in the appropriate section of NRC (2003, Section 2.2.1.3.1.2, Review Method 2), NRC found DOE response to the agreement to be satisfactory.

4.2 CLST.1.13

The focus of CLST.1.13 was the mitigation of residual stresses in the waste package closure welds. Mitigation of tensile residual stresses is intended to remove the mechanical driving force for the initiation and propagation of stress corrosion cracking. The DOE response (Bechtel SAIC Company, LLC, 2003) identifies that the originally proposed induction annealing processes will not be used, and closure weld residual stress mitigation will be accomplished using laser peening or controlled plasticity burnishing. Measurements using ring core and x-ray diffraction indicate that compressive stresses exist to depths of more than 6 mm [0.24 in] for low-plasticity, burnished Alloy 22. For the laser peened welds, compressive stresses exist to depths of at least 1.5 mm [0.06 in], based on x-ray diffraction measurements. Measurements using the ring core method suggest that laser peening can impart compressive stresses to depths of more than 5 mm [0.2 in] (Bechtel SAIC Company, LLC, 2003). Compressive stresses to depths of at least 6 mm [0.24 in] would mitigate stress corrosion cracking in the closure weld and are unlikely to be removed by corrosion as long as passivity is maintained.

Although the staff considers this agreement closed, DOE should consider the following comments:

- Residual stress measurements were conducted on a welded Alloy 22 test plate that may not be representative of the actual waste package closure weld. The information and conclusions presented in TBD-6 (Bechtel SAIC Company, LLC, 2003) should be supported by data from a waste package closure weld mockup or information supporting the representativeness of the Alloy 22 test plate.

- Results of the residual stress profile measurements using ring core and x-ray diffraction appear to be dependent on the characterization method. Improved methods could reduce the uncertainty in the characterization of the residual stress profile of the closure weld.

Based on the NRC review of DOE response to CLST.1.13 Agreement in accordance with methods discussed in the appropriate section of NRC (2003, Section 2.2.1.3.1.2, Review Method 2), NRC found DOE's response to the agreement to be satisfactory.

4.3 CLST.6.02 AIN–1 and 6.03 AIN–1

The focus of CLST.6.02 and 6.03 was to ensure that the assessment of hydrogen embrittlement and hydrogen induced cracking as a potential degradation mode for the drip shields considered the range of materials, metallurgical conditions, fabrication processes, and the variations in chemistry of the solutions contacting the drip shield surfaces. The DOE response (Bechtel SAIC Company, LLC, 2003) provides estimations of the critical hydrogen concentrations for the titanium alloys proposed for the drip shield. The critical hydrogen concentration of Titanium Grade 7 was estimated to be 1,000 Fg/g and the critical hydrogen concentration of Titanium Grade 24 was estimated to be in the range 400–600 Fg/g. Estimates of critical hydrogen concentrations were supported by evaluating similar titanium alloys (Williams, et al., 1959; Jaffee and Williams, 1959; Moody and Gerberich, 1980; Shoesmith, et al., 1997; Hardie and Ouyang, 1999; Greene, et al., 2001; Qin and Shoesmith, 2004). Corrosion tests conducted by DOE suggest the oxide films on titanium alloys are resistant to fluoride, and DOE indicated no evidence has been reported for fluoride enhancing the hydrogen uptake for titanium alloys. The hydrogen uptake in 10,000 years is estimated to be 124 Fg/g assuming a passive corrosion rate measured in long-term corrosion facility test tanks and an uptake efficiency fraction of 0.015 (Bechtel SAIC Company, LLC, 2003).

Although the staff considers these agreements closed, DOE should consider the following comments:

- Although no evidence has been reported for fluoride enhancing the hydrogen uptake for titanium alloys, Brossia, et al. (2001) showed the corrosion rates of titanium alloys are strongly dependent on fluoride concentration. Subsequent analyses by Lin, et al. (2003) suggest the damage to the titanium drip shields by corrosion in fluoride-containing groundwater is limited by the low fluoride concentration in seepage water and consumption of fluoride in corrosion products. The conclusion that the oxide films on titanium alloys used for the drip shield are resistant to fluoride should be supported by data that bound the range of expected fluoride concentrations.

- According to test results reported by Okada (1983) and recent information reported by Qin and Shoesmith (2004), the hydrogen uptake efficiency is dependent on alloy composition, pH, and temperature. The dependence on pH appears to be the most significant, and much higher uptake efficiencies have been measured for titanium alloys in acidic solutions (pH < 3) (Okada, 1983). The hydrogen values for hydrogen uptake used in DOE's response are valid in alkaline solutions. The increased uptake efficiency for acidic conditions should be considered if acidic conditions are expected to occur within the emplacement drifts.

Based on the NRC review of DOE's response to CLST.6.02 Agreements AIN–1 and 6.03 AIN–1 in accordance with methods discussed in the appropriate section of NRC (2003, Section 2.2.1.3.1.2, Review Method 2), NRC found DOE's response to the agreements to be satisfactory.

5.0 SUMMARY

The NRC reviewed DOE's KTI agreement responses within the report to determine whether any important aspect of CLST.1.12, 1.13, 6.02 Agreements AIN–1, and 6.03 AIN–1 were excluded from the response. In addition, NRC performed an independent assessment to determine if the information provided would support submission of a potential license application for a geologic repository. Notwithstanding new information that could raise new questions or comments concerning these agreements, the information provided satisfies the intent of the agreements. On the basis of this review, NRC agrees with DOE that the information assembled in response to CLST.1.12, 1.13, 6.02 Agreements AIN–1, and 6.03 AIN–1 is adequate to support the submission of a license application for the potential repository at Yucca Mountain.

6.0 STATUS OF THE AGREEMENTS

Based on the preceding review, NRC agrees with DOE that the information provided with respect to CLST.1.12, 1.13, 6.02 Agreements AIN–1, and 6.03 AIN–1 is adequate to support the submission of the license application. Therefore, NRC considers CLST.1.12, 1.13, 6.02, and 6.03 Agreements to be closed.

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