SYMPOSIUM D

SEVENTH INTERNATIONAL SYMPOSIUM ON THE SCIENTIFIC BASIS FOR NUCLEAR WASTE MANAGEMENT

Chair
G. L. McVay, Pacific Northwest Laboratory
November 14-17, 1983

Session D1

Chairs: W. Lutze, Hahn-Meitner Institute
D. G. Brookins, University of New Mexico
Monday Morning, November 14, Imperial Ballroom

REPOSITORY RELEVANT RESEARCH: SALT (EUROPE, WIPP)

8:30 Introductory Remarks, G. L. McVay

8:35 D1.1 Investigations of a Hypothetical Inflow of Water or Brine into a Repository for HLW in Salt Formations: Results and Future Activities, H.-J. Herbert, Institut für Tieflagerung, FRG

8:50 D1.2 Numerical and Experimental Investigations on the Time Dependent Behavior of a Salt Dome with a HLW Repository, J. Prij and L.H. Vons, Netherlands Energy Research Foundation

9:05 D1.3 Chemical Durability of a Multicomponent Glass in a Simulated Carnallite/Rock Salt Environment, R. Conradt, H. Roggendorf and H. Scholze, Fraunhofer-Institut für Silicatforschung, FRG

9:20 D1.4 Laboratory Investigations on Radiolysis Effects on Rock Salt with Regard to the Disposal of High-Level Radioactive Wastes, N. Jockwer, Institut für Tieflagerung, FRG

9:35 D1.5 Corrosion Resistant Alloys Against Salt Brine Attack, V. Mirschinka and R. Odoj, Institute for Chemical Technology, KFA/FRG

9:50 D1.6 Colloid Generation and the Actinide Migration in Gorleben Groundwaters, J.I. Kim, G. Buckau, F. Baumgärtner and H.C. Moon, Institut für Radiochemie der Technischen, Universität München, FRG

10:10 BREAK

10:30 D1.7 Migration of Radionuclides: Experiments Within the Site Investigation at Gorleben, E. Warnecke, A. Hollmann, G. Stier-Friedland, Physikalisch-Technische Bundesanstalt, FRG

10:50 D1.8 The Geochemistry of the Castile Brines: Implications for Their Origin and Impact on the WIPP Site, W.E. Coons, D. Meyer, R.L. Olsen and J.K. Register, D'Appolonia Consulting Engineers

11.10 D1.9 Diffusion of Colloids and Other Waste Species in Brine-Saturated Backfill Materials, E.J. Nowak, Sandia National Laboratories

11:30 D1.10 The Waste Package Materials Field Test in S.E. New Mexico Salt, M.A. Molecke and T.M. Torres, Sandia National Laboratories

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fm. Chang + Jungling
12/29/83

Session D2

Chairs: M.J. Apted and C.C. Allen
Rockwell Hanford Operations
Monday Afternoon, November 14, Imperial Ballroom

REPOSITORY RELEVANT RESEARCH: BASALT

1:30 D2.1 Hydrothermal Testing of Barrier Materials: Data Needs for Site-Specific Waste Package Modeling, M.J. Apted, Rockwell Hanford Operations

1:40 D2.2 Status of Geologic and Hydrologic Characterization of a Potential Nuclear Waste Repository Site in Basalts, S.M. Price and R.E. Gephart, Rockwell Hanford Operations

2:00 D2.3 Hydrothermal Interactions Between Columbia River Basalt and Grande Ronde Groundwater: Considerations for a Nuclear Waste Repository, J. Myers, D.L. Lane, and M.J. Apted, Rockwell Hanford Operations

2:20 D2.4 Experimental Studies of Backfill Stability, C.C. Allen, D.L. Lane, R.A. Palmer and R.G. Johnston, Rockwell Hanford Operations

2:40 D2.5 Corrosion Behavior of Low-Carbon Steels in Grande Ronde Groundwater in the Presence of Basalt-Bentonite Backfill, R.P. Anantatmula, C.H. Delegard and R.L. Fish, Rockwell Hanford Operations

3:00 BREAK

3:15 D2.6 Irradiation-Corrosion Evaluation of Metals in Grande Ronde Groundwater for Nuclear Waste Package Applications, J.L. Nelson, R.E. Westerman and F.S. Gerber, Pacific Northwest Laboratory

3:35 D2.7 The Behavior of ^{99}Tc in Doped-Glass/Basalt Hydrothermal Interaction Tests, D.G. Coles, Pacific Northwest Laboratory

3:50 D2.8 Reactions in the System Basalt-Simulated Spent Fuel/Water, D.E. Grandstaff, G.L. McKeon, E.L. Moore and G.C. Ulmer, Temple University

4:05 D2.9 Gamma Radiolysis Effects on Basalt Groundwater, W.J. Gray, Pacific Northwest Laboratory

4:20 D2.10 Development of High Temperature and Pressure Eh and pH Sensing Instruments, M.J. Danielson and O.H. Koski, Pacific Northwest Laboratory, and J. Myers, Rockwell Hanford Operations

4:35 D2.11 Probabilistic Modeling in Post-Closure Performance Assessment, R.G. Baca, P.M. Clifton and R.C. Arnett, Rockwell Hanford Operations

Session D3

Chairs: L. H. Johnson, AECL, Whiteshell
L. Werme, Swedish Nuclear Fuel Supply Company
Tuesday Morning, November 15, Imperial Ballroom

REPOSITORY RELEVANT RESEARCH: GRANITE

8:00 D3.1 The Source Term for Release of Nuclides From a Radioactive Waste Repository 1. Vitrified Waste in Granite, F.T. Ewart, J.B. Morris, J. Severn, B.M. Sharpe, H.P. Thomason, UKAEA, Harwell, UK

Enclosure 1

8:20 D3.2 The Corrosion of Spent UO_2 Fuel in Synthetic Groundwater, R.S. Forsyth, K. Svanberg, Studsvik Energiteknik; and L. Werme, The Swedish Nuclear Fuel Supply Company, Sweden

8:40 D3.3 Multicomponent Leach Tests on Nuclear Waste Glass Precursor Frit in Standard Canadian Shield Saline Solution, R.B. Heimann, D.D. Wood and R.F. Hamon, AECL, Whiteshell, Canada

8:55 D3.4 Numerical Analysis of Radionuclide Migration Through Engineered Barriers, S.C.H. Cheung, T. Chan and R.S. Lopez, AECL, Whiteshell, Canada

9:10 D3.5 Investigation of Groundwater Composition in Relation to Spent Nuclear Fuel Disposal, F. Karlsson, The Swedish Nuclear Fuel Supply Company, Sweden

9:30 D3.6 Actinide and Technetium Solubility Limitations in Groundwaters of Crystalline Rocks, B. Allard, Chalmers University of Technology, Sweden

9:45 D3.7 Uranium Migration at Some Hydrothermal Veins Near Marysvale, Utah: A Natural Analog for Radwaste Isolation, M. Shea, Office of Crystalline Repository Development

10:05 BREAK

10:20 D3.8 Migration in a Single Fissure, H. Abelin, J. Gidlund and I. Neretnieks, Royal Institute of Technology, Sweden

10:40 D3.9 Diffusion in the Matrix of Granitic Rock Field Test in the Stripa Mine: Part 2, L. Birgersson and I. Neretnieks, Royal Institute of Technology, Sweden

11:00 D3.10 Investigation of Hydraulic Properties in Crystalline Rock, L. Carlsson, A. Winberg, Swedish Geological, and B. Rosander, Chalmers University of Technology, Sweden

11:20 D3.11 An Integrated Approach to the Description of Radionuclide Release and Transport in the Geosphere, I. Neretnieks, Royal Institute of Technology, Sweden

Session D4

Chairs: V. M. Oversby and L. D. Tyler
Lawrence Livermore National Laboratory
Tuesday Afternoon, November 15, Imperial Ballroom

REPOSITORY RELEVANT RESEARCH: TUFF

1:15 D4.1 NNWSI Site Description and Site Selection Process, D.L. Vieth, Department of Energy, Nevada

1:35 D4.2 Geohydrologic Setting of Yucca Mountain, Nevada, W.W. Dudley, Jr., U.S. Geological Survey

2:05 D4.3 Mineralogy-Petrology and Geochemistry of Yucca Mountain Tuffs, D.L. Bish, A.E. Ogard and D.T. Vaniman, Los Alamos National Laboratory

2:25 D4.4 Design Considerations to Minimize the Impact of a Repository on a Host Rock, L.W. Scully, Sandia National Laboratories

2:45 D4.5 Post-Emplacement Environment of Waste Packages, K.G. Knauss, V.M. Oversby and T.J. Wolery, Lawrence Livermore National Laboratory

3:05 BREAK

3:20 D4.6 Corrosion Test Plan to Guide Canister Material Selection and Design for a Tuff Repository, R.D. McCright, R.A. Van Konynenburg and L.B. Ballou, Lawrence Livermore National Laboratory

3:40 D4.7 The NNWSI Waste Form Testing Program, V.M. Oversby, Lawrence Livermore National Laboratory

4:00 D4.8 Retardation of Radionuclides by Rock Units Along the Path to the Accessible Environment, A.E. Ogard, K. Wolfsberg, W.R. Daniels, R.S. Rundberg and B.J. Travis, Los Alamos National Laboratory

4:30 D4.9 NNWSI Performance Assessment Considerations, L.D. Tyler, Sandia National Laboratories

Session D5

Chairs: J. F. Kircher and S. J. Basham
Office of Nuclear Waste Isolation
Wednesday Morning, November 16, Imperial Ballroom

REPOSITORY RELEVANT RESEARCH: SALT (TEXAS, UTAH, MISSISSIPPI)

8:00 D5.1 Characterization of Earth Materials Properties for Conceptual Design of an Exploratory Shaft, Richton Dome, Mississippi, R. Haag and O. Swanson, Battelle Memorial Institute

8:20 D5.2 Geohydrology Surrounding a Potential High-Level Waste Repository in the Richton Salt Dome, Mississippi, O. Swanson, ONWI, and J. Tracy, Ertec

8:40 D5.3 Relationship of Engineering Geology to Conceptual Repository Design in the Gibson Dome Area, Utah, R. Helgerson and N. Henderson, Battelle Memorial Institute

9:00 D5.4 Geohydrology Surrounding a Potential High-Level Waste Repository in the Paradox Basin, Utah, A. Brandstetter and L. Kroitoru, Battelle Memorial Institute, R.W. Andrews, INTERA Environmental Consultants, Inc., and J.W. Thackston, Woodward-Clyde Consultants

9:20 D5.5 Relationship of Engineering Geology to Conceptual Repository Design in the Palo Duro Basin, Texas, M.A. Balderman, J.E. Hanley and S. Versluis, Office of Nuclear Waste Isolation

9:40 D5.6 Geohydrology Surrounding a Potential HLW Repository, Palo Duro Basin, Texas, M. Deyling and L. Kroitoru, Battelle Memorial Institute, and L. Picking, Stone and Webster Engineering Corporation

10:00 BREAK

10:15 D5.7 Far-Field Flow Uncertainty Analysis for the Palo Duro Basin, J.L. Devary, Pacific Northwest Laboratory; W.V. Harper, Office of Nuclear Waste Isolation; J.F. Sykes, University of Waterloo; and J.L. Wilson, INTERA Environmental Consultants

10:35 D5.8 Composition and Stratigraphic Distribution of Materials in the Lower San Andres, Units 4 and 5 Salt, Palo Duro Basin, Texas, N. Hubbard, Office of Nuclear Waste Isolation; D. Livingston and L. Fukui, Bendix

10:55 D5.9 Expected Environment for Waste Packages in a Salt Repository, L.R. Pederson, W.J. Gray, F.N. Hodges and G.L. McVay, Pacific Northwest Laboratory; D.E. Clark, Office of Nuclear Waste Isolation

11:10 D5.10 Evaluation of Iron Base Materials for Waste Package Containers in a Salt Repository, R.E. Westerman, J.L. Nelson, S.G. Pitman and W.L. Kuhn, Pacific Northwest Laboratory; S.J. Basham and D.P. Moak, Office of Nuclear Waste Isolation

11:25 D5.11 Evaluation of Spent Fuel as a Waste Form in a Salt Repository, G.L. McVay, W.J. Gray, D.J. Bradley and F.N. Hodges, Pacific Northwest Laboratory; R.W. Cote, Office of Nuclear Waste Isolation

11:40 D5.12 Performance Analysis of Conceptual Waste Package Designs in Salt Repositories, G. Jansen, G.E. Raines and J.F. Kircher, Office of Nuclear Waste Isolation

Session D6
Chairs: G. L. McVay, Pacific Northwest Laboratory
L. Werme, Swedish Nuclear Fuel Supply Company
Wednesday Afternoon, November 16, Imperial Ballroom

SUMMARY OF NATIONAL ACADEMY OF SCIENCES' REPORT ON RADWASTE DISPOSAL

1:30 Opening Remarks, G. L. McVay

1:35 6.1 *The National Research Council Study of the Isolation System for Geologic Disposal of Radioactive Wastes, T.H. Pigford, University of California, Berkeley

COMMENTS BY REVIEW PANEL

2:05 D6.2 *Review Panel Comments and Discussion on Geomedia Specific Repository Relevant Research: Salt, Basalt, Granite and Tuff; Panel Members: G.A. Cowan, Los Alamos Scientific Laboratory; C.A. Heath, NUS Corporation; and D.D. Runnells, University of Colorado

3:15 BREAK

3:30 D6.3 *Prediction of Waste Performance in a Geologic Repository, T.H. Pigford and P.L. Chambré, University of California, Berkeley

4:00 D6.4 *The Geomicrobiology of Nuclear Waste Disposal, J.M. West, AERE Harwell, UK, and I.G. McKinley, EIR, Switzerland

4:30 D6.5 *The Role of Colloids in Nuclear Waste Disposal, A. Avogadro, G. Bidoglio, A. Saltelli and C.N. Murray, Commission of European Communities, Ispra Establishment, Italy; and G. DeMarsily, Ecole des Mines de Paris, France

*Invited Talk

Session D7 (Poster)
Wednesday Evening, November 16, 7:00-10:00, Castle

WASTE FORM DEVELOPMENT AND CHARACTERIZATION

D7.1 Devitrification Behavior of SRL Defense Waste Glass, D.F. Bickford and C.M. Jantzen, Savannah River Laboratory

D7.2 Preparation of High-Level Radioactive Insoluble Waste for Vitrification, R.E. Eibling and B.A. Hamm, Savannah River Laboratory

D7.3 Gel Adsorption Processing for Waste Solidification in "NZP" Ceramics, L.J. Yang, S. Komarneni and R. Roy, The Pennsylvania State University

D7.4 Properties of Saltstone for Disposal of Savannah River Plant Low-Level Radioactive Salt Waste, C.A. Langton and M.D. Dukes, Savannah River Laboratory

D7.5 Metal Matrices and Related Technology Development in the Canadian Nuclear Fuel Waste Management Program, P.M. Mathew, AECL, Whiteshell, Canada

D7.6 A Demonstration Plant for Nonradioactive SYNROC Fabrication: Design Status and Process Development, E.J. Ramm and K.D. Reeve, Australian Atomic Energy Commission; and A.E. Ringwood, Australian National University

D7.7 In-Can Hot Pressing of Borosilicate Glass for the Immobilization of High-Level Waste, G. Ondracek and E.H. Toscano, Institut für Material- und Festkörperforschung I

LEACHING AND WASTE PACKAGE INVESTIGATIONS

D7.8 Radionuclide Transport Modeling of Diffusion Cell Experiments Specific to a Backfill Barrier in a Salt Repository, H.M. Anderson, J.M. Pietz and D.M. Smith, University of New Mexico

D7.9 Effects of Metals and Metal Oxides on the Leaching of Nuclear Waste Glasses, A. Barkatt, W. Sousanpour, A. Barkatt and M.A. Boroomand, The Catholic University of America

D7.10 Activity Diagrams for Calcium/Hydrogen and H_4SiO_4 and Their Relation to Reactions in Systems Containing Radioactive Waste Forms, Cement and Rock in the Presence of Water, M.W. Barnes and D.M. Roy, The Pennsylvania State University

D7.11 Factors Influencing Mass Diffusion in Mixtures of Bentonite and Sand, S.C.H. Cheung, D.W. Oscarson and R.S. Lopez, AECL, Whiteshell, Canada

D7.12 Long-Term Radioactivity Release from High-Level Waste. Part IV: The Effect of Leaching Mechanism and Surface-Area-to-Solution Volume, F.K. Altenhein, E. Freude, B. Grambow and W. Lutze, Hahn-Meitner Institut; R.C. Ewing, The University of New Mexico

D7.13 The Role of Eh in Nuclear Waste Form Dissolution, W.P. Freeborn and W.B. White, The Pennsylvania State University

D7.14 Rough Preliminary Estimate of the Colloidal Sodium Induced in Rock Salt by Radioactive Waste Canister Radiation, P.W. Levy and J.A. Kierstead, Brookhaven National Laboratory

D7.15 Influence of Redox Condition on Iron Content of Leached Glass Surface, A. Manara, F. Lanza, G.D. Mea and C. Rossi, Commission of the European Communities, Ispra, Italy

D7.16 Time Dependent Leaching in Two-Phase Composite Glasses, C.J. Montrose, A. Barkatt and P.B. Macedo, Catholic University of America

D7.17 New Data on the Ion-Induced Modifications of Aqueous Dissolution of Silicates, G.D. Mea, G. Bezzon and C. Rossi-Alvarez, University di Padova, Italy; J.C. Dran and J.C. Petit, Laboratoire Rene Bernas, France

D7.18 Stability of Radioactive Waste Glasses Assessed from Hydration Thermodynamics, M.J. Plodinec, C.M. Jantzen, and G.G. Wicks, Savannah River Laboratory

D7.19 Selection of Barrier Metals for a Waste Package in Tuff, E.W. Russell and W.C. O'Neal, Lawrence Livermore National Laboratory

D7.20 Leaching Behavior of Simulated Radioactive Waste Glass by Neutron Activation Method, S. Sato, H. Furuya, K. Ohta, Kyushu University; and T. Tamai, Kyoto University, Japan

D7.21 Predicting Amounts of Radiolytically Produced Species in Brine Solutions, S.A. Simonson and W.L. Kuhn, Pacific Northwest Laboratory

D7.22 The Retention of Redox Sensitive Waste Elements in Compacted Bentonite, B. Torstenfelt and B. Allard, Chalmers University of Technology, Sweden

D7.23 Radiation Damage Growth Models for the Interpretation of Radiation Effects on Nuclear Waste Form Leaching, Abder M. Ougouag, University of Illinois, and Albert J. Machiels, Electric Power Research Institute

REPOSITORY AND FAR-FIELD RELATED INVESTIGATIONS

D7.24 Capillary Barrier Field Testing, W.V. Abeele and G.L. DePoorter, Los Alamos National Laboratory

D7.25 Mechanisms for the Interaction of Americium (III) and Neptunium (V) with Geologic Media, B. Allard, Chalmers University of Technology, and K. Esbensen, University of Umeå, Sweden

D7.26 Rock Properties Input to the Site Screening Process, Arthur A. Bauer, Battelle Project Management Div., and Joe L. Ratigan, RE/SPEC Inc.

D7.27 Geochemical Studies of Columbia River Basalts, D.G. Brookins, M.T. Murphy, University of New Mexico; H.A. Wollenberg, S. Flexser, Lawrence Berkeley Laboratories

D7.28 On the Environmental Impact of a Leaky Repository for Spent Nuclear Fuel, O. Brotzen, FBAB, Sweden

D7.29 Verification/Validation of the SWIFT Models, D.S. Ward and M. Reeves, GeoTrans, Inc., and L.E. Duda, Sandia National Laboratories

D7.30 A Natural Analogue Approach for Estimating the Health Risks from Release and Migration of Radionuclides from Radioactive Waste, T.L. Gilbert, Argonne National Laboratory

D7.31 Radiometric Dating of Wolfcamp Groundwaters Using ^{4}He and ^{40}Ar, A. Zaikowski and B. Kosanke, Bendix; N. Hubbard, Office of Nuclear Waste Isolation

D7.32 Chemical Migration by Contact Metamorphism Between Pegmatite/Country Rocks: Natural Analogs for Radionuclide Migration, J.C. Laul, Pacific Northwest Laboratory, and J.J. Papike, South Dakota School of Mines

D7.33 Studies of Altered Vitrophyre for the Prediction of Nuclear Waste Repository-Induced Thermal Alteration at Yucca Mountain., Nevada, S.S. Levy, Los Alamos National Laboratory

D7.34 A Proposed Model for the Thermal Conductivity of Dry and Water-Saturated Tuff, M. Moss and G. Haseman, Sandia National Laboratories

D7.35 Description of the SCOPE (Simplified Codes for Performance Evaluation) System for Use in Exploratory Analysis of Radionuclide Transport in a Geologic Medium, G.M. Petrie, S.C. Sneider and R.A. Craig, Pacific Northwest Laboratory

D7.36 Comparison of Batch and Autoradiographic Methods in Sorption Studies of Radionuclides in Rock and Mineral Samples, S. Pinnioja, T. Jaakkola and J.K. Miettinen, University of Helsinki, Finland

Session D8
Chairs: W. L. Kuhn, Pacific Northwest Laboratory
J. F. Kircher, Office of Nuclear Waste Isolation
Thursday Morning, November 17, Imperial Ballroom

MODELING

8:00 D8.1 *The Impact of Alfa-Radiolysis on the Release of Radionuclides from Spent Fuel in a Geologic Repository, I. Neretnieks, Royal Institute of Technology, Sweden

8:30 D8.2 Thermodynamic Coupling of Heat and Matter Flows in Near-Field Regions of Nuclear Waste Repositories, C.L. Carnahan, Lawrence Berkeley Laboratory

8:50 D8.3 TOUGH: A Numerical Model for Nonisothermal Unsaturated Flow in Fractured Porous Media, K. Pruess and J.S.Y. Wang, Lawrence Berkeley Laboratory

9:10 D8.4 Numerical Simulation of Flow and Transport in Partially Saturated, Fractured Tuff, B.J. Travis, T.L. Cook, and R.S. Rundberg, Los Alamos National Laboratory

9:30 D8.5 Rock Joint Description and Modelling for Prediction of Repository Performance, N. Barton, Norwegian Geotechnical Institute; K. Bakhtar, Terra Tek Engineering; S. Bandis, Thessaloniki, Greece

*Invited Talk

9:50 D8.6 Post-Emplacement Safety Analysis of the Subseabed Disposal of High-Level Radioactive Waste, M.F. Kaplan and C.M. Koplik, The Analytic Sciences Corporation

10:10 BREAK

10:30 D8.7 Probabilistic Reliability Analysis of High-Level Waste Packages, C. Pescatore and C. Sastre, Brookhaven National Laboratory

10:50 D8.8 Analytic Models for Assessing the Performance of Engineered Barriers in a Basalt Repository, J.I. Scott and C.M. Koplik, The Analytic Sciences Corp.

11:10 D8.9 Impurities and Inclusion Size Effects on the Migration Velocity of Liquid Inclusions in Salt, A.J. Machiels, Electric Power Research Institute

11:30 D8.10 Transport of ^{14}C and Uranium in the Carrizo Aquifer of South Texas: A Natural Analog of Radionuclide Migration, R.W. Andrews and F.J. Pearson, Jr., INTERA Environmental Consultants, Inc.

Session D9
Chairs: D. J. Bradley, Pacific Northwest Laboratory
L. H. Johnson, AECL, Whiteshell
Thursday Morning, November 17, Berkeley Room

LEACHING/SOURCE TERM INVESTIGATIONS

8:40 D9.1 Modeling Chemical Interactions in the Hydrated Layers of Nuclear Waste Glasses, T.M. Sullivan, University of Illinois, and A.J. Machiels, Electric Power Research Institute

9:00 D9.2 An Experimental and Modelling Approach of the Near-Field Release and Transport Processes, F. Lanza, A. Avogadro, and A. Saltelli, Commission of the European Communities, Ispra, Italy

9:20 D9.3 Methods of Simulating Low Redox Potential (Eh) for a Basalt Repository, C.M. Jantzen, Savannah River Laboratory

9:40 D9.4 Leach Testing of Waste Glasses Under Near-Saturation Conditions, B. Grambow, Hahn-Meitner Institute, and D.M. Strachan, Pacific Northwest Laboratory

10:00 BREAK

10:20 D9.5 The Relationship Between Reaction Layer Thickness and Leach Rate for Six Nuclear Waste Glasses, L.A. Chick and L.R. Pederson, Pacific Northwest Laboratory

10:40 D9.6 Leach Behavior of SRL TDS-131 Defense Waste Glass in Water at High/Low Flow Rates, A. Barkatt, W. Sousanpour, A. Barkatt, M. Boroomand and P.B. Macedo, The Catholic University of America

11:00 D9.7 Dissolution of Technetium from Nuclear Waste Forms, M.Y. Khalil and W.B. White, The Pennsylvania State University

CANCELED

11:20 ~~D9.8~~ Effects of Water Chemistry and Irradiation Damage on the Leach Resistance of SYNROC, J.L. Woolfrey and D.M. Levins, Australian Atomic Energy Commission

11:40 D9.9 Static Leaching of Radioactive Glass Under Conditions Simulating a Granitic Repository for High-Level Waste: Phase I, H-P. Hermansson, H. Christensen, I-K. Björner, Studsvik Energiteknik, Sweden; D.E. Clark, University of Florida; H. Yokoyama, Central Research Institute of Electrical Power Industry, Japan; L.O. Werme, SKBF/KBS, Sweden

Session D10
Chairs: V. M. Oversby, Lawrence Livermore National Laboratory
M. J. Apted, Rockwell Hanford Operations
Thursday Afternoon, November 17, Imperial Ballroom

FAR FIELD, SORPTION AND MIGRATION

1:30 D10.1 *Solubility Constraint: An Important Consideration in Safety Assessment of Nuclear Waste Disposal, D. Rai, Pacific Northwest Laboratory

2:00 D10.2 Conclusions From an NEA Workshop: The Role of Phenomenological Sorption Modeling in Performance Assessment of Radioactive Waste Disposal Systems, A.B. Muller, AECD Nuclear Energy Agency, France; I. Neretnieks, Royal Institute of Technology, Sweden; and D. Langmuir, Colorado School of Mines

2:20 D10.3 Kinetics of the Adsorption of Radionuclides on Tuff from Yucca Mountain, R.S. Rundberg, Los Alamos National Laboratory

2:35 10.4 Porosities of and Diffusivities in Crystalline Rock and Fissure Coating Materials, K. Skagius and I. Neretnieks, Royal Institute of Technology, Sweden

2:50 D10.5 Strontium Isotopic Study of Fracture Filling Minerals in the Grande Ronde Basalt, Washington, D.G. Brookins and M.T. Murphy, University of New Mexico; H.A. Wollenberg, Lawrence Berkeley Laboratory

3:05 D10.6 A Natural Analogue Study of Radionuclide Migration in Clays, I. McKinley, EIR, Switzerland; A.B. McKenzie, R.S. Scott, SURRC, Scotland; J.M. West, Harwell, UK

3:20 BREAK

3:35 D10.7 Generation and Transport Properties of Colloidal Tri- and Tetravalent Actinide Species in Geologic Environments, U. Olofsson, M. Bengtsson and B. Allard, Chalmers University of Technology, Sweden

3:50 D10.8 The Pu (IV) Polymer: Its Formation and Chemistry in Dilute Aqueous Solutions, T.W. Newton and V.L. Rundberg, Los Alamos National Laboratory

4:05 D10.9 The Behavior of Americium in Aqueous Carbonate Systems, R.J. Silva, Lawrence Berkeley Laboratory

*Invited Talk

4:20 D10.10 Permeability and Fluid Chemistry of Some Nevada Test Site Tuffs, C.A. Morrow, D.E. Moore and J.D. Byerlee, U. S. Geological Survey

4:35 D10.11 Use of Natural Radionuclides to Predict the Behavior of Radwaste Radionuclides in Far-Field Aquifers, N. Hubbard, Office of Nuclear Waste Isolation; J.C. Laul and R. Perkins, Pacific Northwest Laboratory

Session D11
Chairs: L. A. Boatner, Oak Ridge National Laboratory
D. G. Brookins, University of New Mexico
Thursday Afternoon, November 17, Berkeley Room

WASTE FORM AND RELATED MATERIALS EVALUATION AND CHARACTERIZATION

2:10 D11.1 Immobilization of High-Level Waste in SYNROC-E, S.E. Kesson and A.E. Ringwood, Australian National University, Australia

2:25 D11.2 A Polyphase Ceramic for Immobilizing ICPP High-Level Nuclear Waste, A.B. Harker and J.F. Flintoff, Rockwell International

2:45 D11.3 Characterization of Hydrofracture Grouts for Radionuclide Partitioning, D.P. Stinton, E.W. McDaniel and H.O. Weeren, Oak Ridge National Laboratory

3:05 BREAK

3:20 D11.4 Corrosion Behavior of TRUW Base and Reference Glasses, P.P. VanIseghem, W. Timmermans, S.C.K./C.E.N.; and R. DeBatist, Rijksuniversitair Centrum Antwerpen, Belgium

3:35 D11.5 Surface Layer Crystallization of Simulated Waste Glass at Elevated Temperatures, T. Inoue, H. Yokoyama, T. Onchi and H. Koyama, Central Research Institute of Electric Power Industry, Japan

3:50 D11.6 Stress Corrosion Cracking in Borosilicate Glass from the Savannah River Plant, A.E. Ringwood and P.E. Willis, Australian National University, Australia

4:10 D11.7 Longevity of Borehole and Shaft Sealing Materials: Characterization of Ancient Cement-Based Building Materials, C.A. Langton, Savannah River Laboratory, and D.M. Roy, Pennsylvania State University

4:25 D11.8 Partitioning of High-Level Waste as Pretreatment in Waste Management, M. Kubota, I. Yamaguchi, K. Okada, Y. Morita, K. Nakano and H. Nakamura, Japan Atomic Energy Research Institute, Japan

Enclosure 2

Some Notes on Symposium D "Seventh International Symposium on th Scientific Basis for Nuclear Waste Management"

Repository Relevant Research: Salt (Europe, WIPP)

1. D1.7 Migration of Radionuclides: Experiments Within the Site Investigation Program at Gorleben, E. Warnecke, A. Hollmann, G. Stier-Friedland, Physikalish-Technische Bundsanstalt, FRG

   This series of experiments were started in 1981 and would end in 1984. It provided parametric values for many chemical processes in salt (e.g., sorption, desorption, temperature, Eh, pH, radionuclide concentration and oxic and anocix conditions) site specific in the FRG.

2. D1.8 The Geochemistry of the Castile Brines: Implications for Their Origin and Impact on the WIPP Site, W.E. Coons, D. Meyer, R.L. Olsen and J.K. Register, D'Appolonia Consulting Engineers

   This project evaluated the major and minor component chemistry of the brines and concluded that Castle Brines appeared to be in equilibrium with all major minerals of the geologic environment and therefore did not have the capacity for substantially degrading the integrity of the WIPP site. If the conclusions were correct, results of the evaluation could be used to predict the equilibrium conditions of the specific WIPP site after waste emplacement.

3. D1.9 Diffusion of Colloids and Other Waste Species in Brine Saturated Backfill Materials, E.J. Nowak, Sandia National Laboratories

The results of one dimensional diffusion experiments of uncompacted and compacted (up to 2000 psi) brine saturated bentonite were presented and compared with those obtained from theoretical diffusion calculations. The quantitative results were given in terms of distribution coefficient Kd.

4. D1.10 The Waste Package Materials Field Test in S.E. New Mexico Salt, M.S. Molecke and T.M. Torres, Sandia National Laboratories

This talk presented results of backfill material and tests for low-density bentonite clay, low density bentonite (70% Wt%) silica sand (30% Wt%) mixtures, both clay and brine injected, high-density bentonite-sand annular compacts, air and finely-crushed salt. The tests were done at 350 m below surface with radiation for time periods of one to five months. Since the tests served as a precursor to the forthcoming WIPP in-situ simulated high-level waste package performance experiments, they could provide NRC with information on the instrumentations (thermocouples, pressure gages, electrical sensitivity, moisture sensors and vertical displacement gages) of the WIPP in situ experiments.

Repository Relevant Research: Basalt

5. D2.1 Hydrothermal Testing of Barrier Materials: Data Needs for Site-Specific Waste Package Modeling, M.J. Apted, Rockwell Hanford Operations

   This presentation outlined BWIP's interpretation of data requirements for site specific waste package to satisfy Federal Regulatory criteria. The data requirements were organized in accordance to whether they were criteria specific, site specific or design specific and to the time periods of interest (pre-emplacement, containment, isolation/slow release).

6. D2.2 Status of Geologic and Hydrologic Characterization of a Potential Nuclear Waste Repository Site in Basalts, S.M. Price and R.E. Gephart, Rockwell Hanford Operations

   This project provided some parametric values for BWIP's Hanford site including typical rock fracture sizes (.1 to .2mm), groundwater head and hydraulic conductivity values. These parameters may be required for WMEG's waste package performance analysis.

7. D2.3 Hydrothermal Interactions Between Columbia River Basalt and Grande Ronde Groundwater: Considerations for a Nuclear Waste Repository J. Myers, D.L. Lane, and M.J. Apted, Rockwell Hanford Operations

   The experiments discussed in this talk were done in Dickson-type autoclaves using Grande Ronde basalt and synthetic groundwater at temperatures of 100°, 150°, 200° and 300°C at a pressure of 3000 bars and varying initial water:rock mass ratios of from 2:1 to 50:1.

It was found in the experiments that solution pH remained alkaline with rapid adjustment of solution Eh to reducing values at all temperatures. At the end of the experiments, alteration minerals were characterized. It was somewhat unfortunate that none of the tests included radiation effects. The data from these experiments could still provide a basis for interpreting the results of more complex waste/barrier/rock interaction experiments to determine the suitability of Grande Ronde basalts for a waste repository.

8. D2.4 Experimental Studies of Backfill Stability, C.C. Allen, D.L. Lane, R.A. Palmer and R.G. Johnston, Rockwell Hanford Operations

   The series of experiments reported in these studies were designed to identify the range of conditions under which a backfill of Na - bentonite and crushed basalt could retain its desirable physical and chemical properties. It appears that irreversible dehydration could occur and cause loss of swelling ability at temperature of 400°C in one year and 300°C in 379 days. At other less severe test conditions (e.g., 320 days at 370°C, 60 days at 200°C) little irreversible alteration occurred.

9. D2.5 Corrosion Behavior of Low-Carbon Steels in Grande Ronde Groundwater in the Presence of Basalt-Bentonite Backfill, R.P. Anantatmula, D.H. Delegard and R.L. Fish, Rockwell Hanford Operations

   Three low carbon steels (AISI 1006, 1020 and 1025) were tested at 150°C and 250°C in Hanford Grande Ronde groundwater (9.75 pH under anoxic conditions (≤0.1 mg/L oxygen in water) in backfill (75 Wt% basalt + 25 Wt% bentonite) for 1 and 2 weeks. The following conclusion (by the authors) can be of value to NRC's waste package analyses:

a. The average corrosion of carbon steels under anoxic conditions with the backfill was at least a factor of 7 lower than that for oxic conditions at 150°C.

b. The dissolved oxygen in groundwater was reduced to very low levels within a week.

c. Under anoxic conditions with backfill, the corrosion of carbon steels was independent of the carbon composition of the steel.

d. The weight loss at 250°C was about the same as that at 150°C due to the formation of a very adherent layer of iron-rich clay (saponite) on the surface of the steels at 250°C.

e. Pitting was not detected on any of the specimens.

f. Assuming linear corrosion kinetics, extrapolation of the two-week data to 1000 year results in ~ 10 mm penetration of the low carbon steel in Grande Ronde groundwater under anoxic conditions with backfill.

10. D2.6 Irradiation-Corrosion Evaluation of Metals in Grande Ronde Grounwater for Nuclear Waste Package Applications, J.L. Nelson, R.E. Westerman and F.S. Gerber, Pacific Northwest Laboratory

The corrosion behavior of several iron-base and titanium-base alloys was determined in synthetic Grande Ronde groundwater at temperatures of 150°C to 250°C and under irradiation dose rates to 2×10^6 rad/hr. At 250°C corrosion rate for radiated conditions (of 3×10^5

rads/hr gamma field) was 2 to 3 times higher than non-radiated conditions. For titanium alloys, corrosion rate was low but significant amount of hydrogen was absorbed during a 10-month exposure to groundwater at 250°C and irradiated at a dose rate of 1 x 10^4 to 2 x 10^6 rad/hr.

11. D2.7 The Behavior of ^{99}Tc in Doped-Glass/Basalt Hydrothermal Interaction Tests, D.G. Coles, Pacific Northwest Laboratory

    This talk reported a series of autoclave tests evaluating the effect of basalt on the concentration of ^{99}Tc released into solution from PNL 76-68 glass waste form.

    The results indicated that the presence of basalt under hydrothermal conditions (200°C and 30 MPa) reduced the concentration of ^{99}Tc in solution by nearly four orders of magnitude (~0.01 mg/l after 1400 hours of run time). Therefore for ^{99}Tc under those hydrothermal conditions, solubility is the dominant constraint rather than leach rate.

12. D2.8 Reactions in the System Basalt-Simulated Spent Fuel/Water, D.E. Grandstaff G.L. McKeon, E.L. Moore and G.C. Ulmer, Temple University

    Experiments in which Grande Ronde Basalt and basalt with simulated spent fuel were reacted with synthetic Hanford groundwater were conducted to determine steady-state concentrations which could be used in radionuclide release rate models. These tests were performed at temperatures of 100°, 200° and 300°C; 30 megapascals pressure, and a solution: solid mass ratio of 10:1 for durations of up to 5000 hours. Some of the conclusions made so far were that pH was not greatly influenced by spent fuel (≤7.5) but redox conditions were greatly altered. Composition of products of glass dissolution and precipitation were discussed also.

13. D2.9 Gamma Radioloysis Effects on Basalt Groundwater, W.J. Gray, Pacific Northwest Laboratory.

    Reference Grande Ronde Basalt groundwater, containing up to 700 mg/l methane and lesser amounts of nitrogen and argon, was irradiated in cobalt-60 source to doses of about 5 MGY. Hydrogen gas and polymer hydrocarbons similar to polyethylene were formed. The pH of the solution remained at the initial value of 9.5 during the test and the absolute amount of dissolved nitrogen did not change. This suggests that gamma radiolysis did not lead to formation of nitric acid for site-specific Grande Ronde Basalt groundwater as has previously postulated.

14. D2.10 Development of High Temperature and Pressure Eh and pH Sensing Instruments, M. J. Danielson and O. H. Koski, Pacific Northwest Laboratory, and J. Myers, Rockwell Hanford Operations.

    Progress in the development of Eh and pH sensing electrochemical probes were reported. Sensing metals under consideration for Eh measurement are titanium, gold, and platinum. The pH electrodes under consideration was zirconia based electrodes.

15. D2.11 Probabilistic Modeling in Post-Closure Performance Assessment, R. G. Baca, P. M. Clifton and R. C. Arnett, Rockwell Hanford Operations.

    This model stochastically analyzed groundwater travel time. It uses a Monte Carol technique to generate inputs such as (1) transmissivity (or hydraulic conductivity), (2) effective thickness (or effective porosity), or (3) boundary conditions. A log-normal distribution is assumed for transmissivity and normal distributions for the effective thickness and boundary conditions. A demonstration of this method was presented using Hanford site data.

Repository Relevant Research: Salt (Texas, Utah, Mississippi)

16. D5.1 Characterization of Earth Materials Properties for Conceptual Design of an Exploratory Shaft, Richton Dome, Mississippi, R. Haag and O. Swanson, Battelle Memorial Institute.

D5.2 Geohydrology Surrounding a Potential High-Level Waste Repository in the Richton Salt Dome, Mississippi, O. Swanson, ONWI, and J. Tracy, Ertec.

D5.3 Relationship of Engineering Geology to Conceptual Repository Design in the Gibson Dome Area, Utah, R. Helgerson and N. Henderson, Battelle Memorial Institute.

D5.4 Geohydrology Surrounding a Potential High-Level Waste Repository in the Paradox Basin, Utah, A. Brandstetter and L. Kroitoru, Battelle Memorial Institute, R. W. Andrews, INTERA Environmental Consultants, Inc., and J. W. Thackston, Woodward-Clyde Consultants.

D5.5 Relationship of Engineering Geology to Conceptual Repository Design in the Palo Duro Basin, Texas, M. A. Balderman, J. E. Hanley, and S. Versluis, Office of Nuclear Waste Isolation.

D5.6 Geohydrology Surrounding a Potential HLW Repository, Palo Duro Basin, Texas, M. Deyling and L. Kdritoru, Battelle Memorial Institute, and L. Picking, Stone and Webster Engineering Corporation.

D5.7 Far-Field Flow Uncertainty Analysis for the Palo Duro Basin, J. L. Devary, Pacific Northwest Laboratory; W. V. Harper, Office of

Nuclear Waste Isolation; J. F. Sykes, University of Waterloo; and J. L. Wilson, INTERA ENvironmental Consultants.

D5.8 Composition and Stratigraphic Distribution of Materials in the Lower San Andres, Units 4 and 5 Salt, Palo Duro Basin, Texas, N. Hubbard, Office of Nuclear Waste Isolation; D. Livingston and L. Fukui, Bendix.

D5.9 Expected Environment for Waste Packages in a Salt Repository, L. R. Pederson, W. J. Greay, F. N. Hodges and G. L. McVay, Pacific Northwest Laboratory; D. E. Clark, Office of Nuclear Waste Isolation.

Some of the geomechanical and geohydrology properties assessed in these studies (D5.1 to D5.9) can be of value in defining repository environment in salt for waste package performance evaluations, e.g., plasticity, unit weight, moisture content, bulk density, porosity, shear strength, permeability, saturation and hydrologic flow properties.

17. D5.10 Evaluation of Iron Base Materials for Waste Package Containers in a Salt Repository, R. E. Westerman, J. L. Nelson, S. G. Pitman and W. L. Kuhn, Pacific Northwest Laboratory; S. J. Basham and D. P. Moak, Office of Nuclear Waste Isolation.

This project is developing corrosion models for low carbon steel containers in salt repositories. It has plans to consider the following: effects of temperature level, radiation level, oxygen level, material parameters and failure mode analysis. Constitutive type relations will be developed for predicting performance of iron-based waste package containers in salt repositories.

18. D5.11 Evaluation of Spent Fuel as a Waste Form in a Salt Repository, G. L. McVay, D. J. Bradley and F. N. Hodges, Pacific Northwest Laboratory; R. W. Cote, Office of Nuclear Waste Isolation.

The evaluation of spent fuel as a waste form in a salt repository was discussed in terms of the experimental program being carried out and planned by ONWI. It also discussed the use of test data in obtaining numerical estimates of parameters for mathematical modeling of spent fuel leaching mechanism. Some initial findings included (a) leaching of unclad spent fuel did not seem to be temperature dependent in brine and (b) oxidized zircaloy did not have much effect on UO_2 leach rate (this does not mean cladding has no effect on spent fuel leaching because the cladding can absorb other radionuclides).

19. D5.12 Performance Analysis of Conceptual Waste Package Designs in Salt Repositories, G. Jansen, G. E. Raines and J. F. Kircher, Office of Nuclear Waste Isolation.

This evaluation demonstrates how WAPPA waste package and TEMP thermal codes can be used for sensitivity studies to address criteria imposed by 10 CFR 60.113. Using what the authors claimed to be conservative preliminary design values for salt thermal conductivity, brine wetting time, overpack corrosion rate, leach rate and radiation field, the package life was evaluated and found to be most sensitive to overpack corrosion rate which is in turn highly dependent on temperature. It concluded with the following finding for waste package design:

(a) "Avoidance of brine wetting until a time well past the temperature peak or achievement of lower maximum temperatures through aging of the waste could dramatically increase package life for a given design."

(b) "For the Sr-90 and Cs-137 leach rates to drop below the 10 CFR 60.113 limit, a package lifetime of about 300 year would be required. For <u>all</u> other nuclides, shorter package life times are achieved."

20. 6.1 The National Research Council Study of the Isolation System for Geologic Disposal of Radioactive Wastes, T. H. Pigford, University of California, Berkeley.

Waste Form

Borosilicate glass is the appropriate choice of a waste form for further testing and for use in current repository designs. Back-up research and development on alternative waste forms should be continued.

Testing

Testing in a simulated repository environment is necessary to develop an adequate prediction of the long term performance of waste package in a geologic repository.

Long-Term Isolation and Environmental Releases

The major contributors to geologic isolation are:

(a) The slow dissolution of key radioelements are limited by solubility and by diffusion and convection in groundwater.

(b) Long water travel times from the waste to the environment.

(c) Sorption retardation in the media surrounding the repository.

Other Contributors

(a) Dilution of the radionuclides at the surface can reduce the individual radiation exposures that can result from those waste radioactivity that may ultimately reach the environment. Some conceptual repositories were found to not meet the individual dose criterion, although they could still meet the release limits proposed by the EPA standard.

<u>Other Conclusions</u>

(a) The most important issue in terms of long term release is release rates at the unheated conditions. This also implies that at thermal conditions, stresses on waste package are severe.

(b) The present large backlog of cooled waste implies that there will be a lot of access time for high temperature waste form.

(c) Waste form leach rate comparison at present is not a good way of justifying waste form selection because they are compared without full consideration of the environment and the system constraints.

21 D6.2 Review Panel Comments and Discussion on Geomedia Specific Repository Relevant Research: Salt, Basalt, Granite and Tuff; Panel Members: G. A. Cowan, Los Alamos Scientific Laboratory; C. A. Heath, NUS Corporation; and D. D. Runnels, University of Colorado.

The following is an abstract of some of the comments from the review panel.

C. A. Heath

(a) In general, there is still a lack of focus in some programs as regard to where and how their results are used to solve the waste problem.

(b) Researchers should challenge the (EPA and NRC) requirements and criteria if they feel these are unreaslistic (e.g., the assumed age of the waste at the time of emplacement and their radiation level).

(c) There is inadequate cross-talk among workers on different programs (e.g., one paper on a specific repository site quoted 30 references with none from similar work for the other two sites).

(d) There should be more care given to the validity of some of the results of some studies (e.g., a prediction carried to 10^9 years is clearly out of focus. According to the USGS, there will be an ice age before reaching such time period. Similarly some data (on permeability) presented have discrepancies of 10^{13}). Careless use of such results can only undermine public confidence on the overall program.

G. A. Cowan

(a) Our level of technology indicates that we can design a waste package that lasts (for 300 - 1000 year life).

(b) There is a general lack of discussion on quality control on all programs (QC of data, test results, and analysis).

(c) The experiments (tests) seem to lag analyses by big gaps. More experimental results should be obtained to verify the results of the analyses.

(d) Cowan agrees with Heath that there is a lack of focus in DOE's systems analyses.

D. D. Runnels

(a) It is good to see an increase of reported studies in this symposium on colloids.

(b) Runnels was bothered by some "unrealistic" use of parameters (e.g., to address redox conditions at places other than the borehole) the use of Eh measurement at the borehole is meaningless because of Eh variations away from the borehole.

(c) There was inadequate number of papers on effects of radiation.

(d) There is an astonishing lack of error analysis in almost all studies presented.

(e) There is a lack of focus on DOE's work - too many programs on the same topic.

21. D6.3 Prediction of Waste Package in a Geologic Repository, T. H. Pigford and P. L. Chambre, University of California, Berkeley.

This model is a theoretical analysis of the time-dependent rate of radioelement mass transfer by diffusion and convection into the groundwater surrounding the waste. In developing this model, the analysis assumes a constant concentration at the waste form surface equal to the solubility of the radioelement and some specified ranges of repository conditions.

Some of the results of this analysis include:

(a) The analytical solutions for time-dependent mass transfer are reduced to asymptotic steady-state approximation.

(b) The predicted steady-state dissolution rates are considerably below those observed in laboratory leaching experiments with borosilicate glass and other waste forms.

(c) The time to reach steady-state dissolution can be as short as a few years, when convective mass transfer in the concentration boundary is important to many thousands of years, when mass transfer is mainly by diffusion in the groundwater.

(d) The solid-liquid chemical reaction rates measured in the laboratory experiments are greater than the rates of diffusion-convective mass transfer in the concentration boundary layer surrounding the waste form in a geologic repository.

(e) Retardation by backfill is not important for steady-state.

22. D6.4 The Geomicrobiology of Nuclear Waste Disposal, J. M. West, AERE Harwell, UK, and I. G. McKinley, EIR, Switzerland.

Current British and European research in this area was reviewed. Little data was reported excepting the confirmation that biochemistry must be considered in nuclear waste disposal because of confirmation that microbes and bacteria could thrive at extreme environments ($-20°C \geq T \geq 250°C$ at pressures of up to 265 atmospheres).

Enclosure 3

Repository Relevant Research: Granite

The majority of the presentations on granite research were made by the Swedish contingent however papers were presented by the United Kingdom, Canada and the United States.

Simulated vitrified radioactive waste glasses containing neptunium, plutonium or technetium were tested in the presence of concrete, granite and granitic groundwater in order to determine a source term for nuclide release, at UKAEA Harwell. The results indicated that technetium was weakly sorbed while neptunium was strongly sorbed, in fact sorption of Np was complete onto concrete in just 5 days in solution. Sorption coefficients for concrete were an order of magnitude greater than those for granite. Plutonium was not detectable in the solution indicating slower leach kinetics.

Atomic Energy of Canada, Ltd., presented the results of a multicomponent leach test on a borosilicate glass precursor frit in Standard Canadian Shield Saline Solution. In addition to the frit and solution the tests included Grade-2 titanium container material, a buffer material (montmorillonite) and granitic rock. Leaching was carried out at 200°C for up to 28 days. Initially a protective Mg-silicate layer formed, however this began to flake off after about 12 days producing a rapid increase in the glass dissolution rate. The second Canadian paper was a finite-element analysis of radionuclide migration through the engineered barriers. One notable results was that use of a backfill possessing the same effective diffusion coefficient as the buffer (where the buffer and backfill are adjacent, concentric barriers) reduced the total flux of nuclides migrating out of the waste package by approximately 30%.

The Swedish presentations concentrated on the period beginning with the release of radionuclides from the waste form and continuing with the migration of the nuclides through the engineered barriers and into the host rock. Investigations of the groundwater composition showed that copper as a canister material is relatively stable with the exception of a small amount of sulphide corrosion and initial oxidation due to emplacement operations. The groundwater is predicted to be reducing as a result of the high concentrations of Fe^{2+} in the rock and in the water, under these conditions technetium and the actinides will exist in their lowest oxidation state thereby limiting their solubility to soluble oxide and carbonate phases which could exist under these conditions. Therefore, the three parameters emphasized in the studies were the redox potential, pH and total carbonate concentration.

Repository Relevant Research: Tuff

A great deal of the work presented at the Boston meeting was the same material presented to the NRC at a NNWSI Waste Package Workshop in October at Dublin, CA, which has been summarized in the NNWSI trip report. The highlights of these papers relevant to the waste package work will again be mentioned here.

The current reference repository is located in the Topopah Springs member of Yucca Mountain approximately 150 meters above the water table, therefore the tuff waste package will lie in the unsaturated zone. The maximum annual water recharge is expected to approach 8 mm and as a result of the unsaturated conditions in the porous rock, the in-situ waste package will be subjected only to atmospheric pressure. L. Scully indicated that emphasis remains on horizontal waste emplacement due to the economic savings to be achieved by removing a smaller quantity of rock to emplace the same quantity of waste.

K. Knauss discussed the early results of geochemical testing examining the effects of temperature and rock interaction on the chemistry of Yucca Mountain groundwater. Little significant change was noted except that the Si concentration tended to increase towards cristobalite (SiO_2) saturation. The pH remained near 7.0 throughout the 4 month experiments.

As indicated by R. Van Konynenburg, Type 304L stainless steel has been selected for the canister reference material with 316L and 321 stainless and Incoloy 825 as secondary materials. Typically 304L exhibits good resistance to several forms of corrosion (considered to be the major failure mechanism), although it is susceptible to stress corrosion cracking and localized corrosion under certain environmental conditions. The viewpoint held by the NNWSI group is that the in-situ environmental conditions of Yucca Mountain will not produce these forms of corrosion, however this viewpoint is not universal and further testing will be necessary before acceptance could be determined. As a result of the lack of groundwater contact as well as the ability of the reference canister to resist corrosion, the current waste package design does not employ either an overpack or packing material on the basis that the cost savings outweighs any engineering benefit. Although such options have not been ruled out altogether.

V. Oversby discussed several testing programs under way to establish the performance of the waste forms under saturated and unsaturated conditions. Reprocessed glass waste forms representing commercial and defense high-level waste will be tested in vapor and totally immersed to simulate repository temperatures greater than 95°C and saturated conditions less than 95°C, respectively. A second test will have a single drop of groundwater dripping onto the waste form at three day intervals to simulate the expected annual amount of recharge water. Both tests include canister material and Topopah Spring tuff to simulate in-situ interactions. A third program is examining the amount of protection provided by Zircaloy cladding on spent fuel rods by intentionally inducing defects and subsequently monitoring the release of radionuclides to the solution. Preliminary data indicates some protection may be provided by the cladding.

Leaching/Source Term Investigations

C. Jantzen, SRL, discussed a test method for reproducing measured Eh values from anoxic basaltic groundwater for the purpose of laboratory leach tests. Reproducible measurements were achieved under extremely sterile conditions (the platinum electrode was polished with diamond paste under an argon atmosphere and tests were performed in an argon glove box). Dr. Runnels again expressed his opinion that such Eh values would be meaningless in actual multi-component in-situ situations.

As described by T. Sullivan, the hydrated (gel) layers formed at the surface of nuclear waste glasses as a result of aqueous corrosion were examined to characterize their effect on the proceeding degradation phenomena. The gel layers act as saturated porous ionic solide therefore network dissolution, diffusion and precipitation (alteration product formation) can readily occur within these layers.

D. Strachan presented other leach tests performed to determine the overall effect of the gel layer on waste glass degradation under near-saturation conditions. Specimens were leached for two months, the gel layer was then removed and leaching continued. The specimens leached in deionized water continued to leach at the same rate while others leached in 0.001M $MgCl_2$ exhibited a large increase in weight loss after removal of the gel layer as a result of a chemical potential difference between the glass and the new gel layer. Other static leach experiments discussed by L. Chick, and performed at PNL to characterize the effect of reaction layer thickness, resulted in a linear leach rate with time therefore it was concluded that the leach rate is independent of reaction layer thickness. A linear leach rate with time was also observed in dynamic leach experiments by A. Barkatt while testing SRL TDS-131 Defense Waste Glass in high flow rates. At slow flow rates a surface layer formed which contained most of the Al and Si to diffuse out of the glass matrix. The concentrations of these elements increased in the surface layer producing an overall decrease in the concentration of leached species in solution to a value which remained constant with further contact time.

Waste Form and Related Materials Evaluation and Characterization

S. Kesson discussed the latest in a series of SYNROC products (hot-pressed SYNROC-E) which is composed of a matrix of rutile (TiO_2) along with the SYNROC phases which can immobilize up to 7 Wt% calcined HLW. The rutile matrix provides excellent leach-resistance such that preliminary tests have indicated that only 0.7 Wt% of Cs is leached during the first 50 days and the percentage decreases thereafter. It was noted that a protective rutile surface layer is expected to form with time, further decreasing the leach rate. Ms. Kesson later presented a paper for her collegues at the Australian National University which proposed a mechanism for stress-corrosion cracking in borosilicate glass. Samples of SRL glass were obtained which contained crystalline spinel phases surrounded by microcracks. A system of localized stress fields in conjunction with atmospheric water vapor was proposed as the mechanism for SCC. This sparked numerous comments expressing concern that the necessary conditions for SCC were not present, viz., an electrochemical cell and a constant source of stress. Also the supplier of the glass specimens from SRL mentioned that to obtain the specimens, the melt canisters were opened with an acetylene torch and the glass was then removed with a hammer and chisel probably inducing thermal and mechanical stresses which may have produced cracking.

A polyphase ceramic waste form has been developed specifically for high-level calcined wastes produced at the Exxon Idaho Chemical Processing Plant (ICPP). This ceramic waste form has a waste loading of over 70 Wt% for the ICPP calcine. The ICPP wastes are rich in CaF_2, ZrO_2 and Al but possess a low actinide content. By utilizing ceramic waste forms instead of borosilicate glass, a volume reduction of a factor of three has been achieved.