CNWRA

Date: November 29, 2000

To: T. Ahn

From: E. Whitt

Subject: **TRIP REPORT** - Attendance at the Atalante 2000 - International Conference on Scientific Research on the Back-End of the Fuel Cycle for the 21st Century

Remarks: Please review and approve this paper also please indicate whether or not report should go into PDR/ADAMS. **This is a non-ticketed item.**

From: Tae Ahn
To: Vijay Jain
Date: Tue, Dec 5, 2000 12:27 PM
Subject: Fwd: PROGRAMMATIC ACCEPTANCE OF TRIP REPORT

CC: Emarsha Whitt

From: Tae Ahn
To: Deborah DeMarco, Emarsha Whitt
Date: Tue, Dec 5, 2000 10:13 AM
Subject: PROGRAMMATIC ACCEPTANCE OF TRIP REPORT

December 5, 2000

SUBJECT: PROGRAMMATIC ACCEPTANCE OF "TRIP REPORT - ATTENDANCE AT THE ATLANTE 2000 - INTERNATIONAL CONFERENCE ON SCIENTIFIC RESEARCH ON THE BACK-END OF THE FUEL CYCLE FOR THE 21ST CENTURY," OCTOBER 24-26, 2000, AVINGTON, FRANCE

I have done a programmatic review of the trip report. I consider it fully acceptable to go into PDR/ADAMS.

Tae M. Ahn
Project Manager (Materials Engineer)
High-Level Waste Branch
Division of Waste Management
Office of Nuclear Material Safety
And Safeguards

Ref.: a non-ticketed item for review

CC: N. King Stablein

CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES

TRIP REPORT

SUBJECT: Attendance at the Atalante 2000—International Conference on Scientific Research on the Back-End of the Fuel Cycle for the 21st Century (Charge Number 20.01402.571)

DATE/PLACE: October 24–26, 2000, Avignon, France

AUTHOR: V. Jain

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AUTHORS: V. Jain

PERSONS PRESENT: V. Jain, CNWRA

BACKGROUND AND PURPOSE OF TRIP:

Atalante 2000—International Conference on Scientific Research on the Back-End of the Fuel Cycle for the 21st Century, was held October 24–26, 2000, at the Palais des Papes in Avignon, France. Conference attracted about 450 delegates from all over the world. Conference included a visit to Atalante facility in Marcoule, France.

The purpose of the trip was to present a paper titled "High-Level Waste Glass Dissolution in Simulated Internal Waste Package Environments" at the conference and to gain useful information from papers presented at the meeting to update our background information on spent fuel and HLW glass waste form. The conference proceedings are available on the Internet (http://www.cea.fr/html/Atalante2000.html).

The summary provided in this report is based on the authors' attendance at selected sessions and brief notes taken during presentations on topics relevant to treatment and disposal of radioactive wastes.

SUMMARY OF PERTINENT POINTS:

Visit to Atalante Facility, Marcoule, France

Atalante is a centralized research facility for the investigation of the back-end of the nuclear fuel cycle. This includes spent fuel reprocessing, and high-level waste treatment and conditioning. The facility is designed in a modular form and has seven modules. Atalante facility is used for

- Development of high-temperature processes (vitrification and glass-ceramic) for high-level waste conditioning and determination of long-term behavior of HLW packages, glass, and irradiated spent fuel

- Design, synthesis and testing of new molecules for separation of minor actinides

- Process development and investigation of PUREX process, dissolution of irradiated targets, and separation of actinides

- Characterization of spent fuel

- Development of pyrochemical extraction, selective precipitation, and electrodeposition techniques or separation of long-lived radionuclides

- Fabrication of targets containing minor actinides (Np, Am), radiation sources such as gamma photon emission and neutron emission sources

Atalante facility has

- Specially designed cells capable of handling large quantity of radioactive materials. Some modules are licensed for handling up to 20 kg of fissile material

- Cells fitted with scaled melter system consisting of melter, calciner, and off-gas system

- Sample preparation equipment such as cutting and polishing tools, and physical characterization equipment such as viscometer, calorimeter, and dilatometer

- Cells that contain scanning electron microscope, X-ray diffraction, microprobe, and inductively coupled plasma mass-spectroscope

- Full range of analytical services in the hot-cells such as spectrophotometry, chromatograpghy, X-ray flourescence, infrared spectroscopy, electrochemical measurements, continuous scanning collimated gamma spectrometry

Overview Presentations

R. Schenkel, TUI, Germany, provided an overview of various R&D facilities. He indicated that facilities at British Nuclear Fuels Limited (BNFL) and Atalante are the major centers for backend of the fuel-cycle research. He stressed the need for reducing radiotoxicity. Sue Ion, BNFL, UK, stressed the need to use nuclear energy in order to meet CO_2 standard proposed at the Kyoto convention. She stated that current nuclear power plants avoid 2.3 billion tonnes of CO_2 emissions which is 10 percent of the total CO_2 emission. Her talk walked through various activities currently conducted at the BNFL research center. B. Tissot, CNE, Commissariat a l'energie (CEA), France and J. L. Nigon, COGEMA, France discussed the R&D needs for reducing radiotoxicity and french proposal for conducting R&D work in the area of back-end of the fuel-cycle.

Disposal of Spent Fuel and Vitrified Waste Forms

B. Grambow, Subatech, France, presented a paper on the performance of spent fuel in European geological disposal environments. The performance of spent fuel was studied in three different geological environments -clay, granite, and salt. Results showed that general reaction mechanisms were similar in various

environments. However, differences in corrosion rates were attributed to groundwater chemistry, surface complexation, and electrochemical corrosion potential. Radionuclide retention was highest in presence of iron.

A. Sasahara, Central Research Institute of Electric Power Industry, presented a paper on the post storage examination of the irradiated SF stored for twenty years in wet and dry conditions. Three fuel rods of BWR-MOX fuel were stored under wet conditions and two fuel rods were stored in dry conditions. In addition, a PWR-UO_2 fuel rod was also stored under dry conditions. The BWR-MOX wet stored fuel was visually examined for cladding condition and tested using puncture test. The BWR-MOX dry stored fuel was examined for gas in the storage capsule. The PWR-UO_2 fuel was additionally characterized using electron micro probe analysis for hydrogen content in cladding. The data was compared with the examination conducted prior to storage. The examination of PRW-MOX fuel indicated no significant changes in the fuel integrity after 20 years of storage. The microstructures were similar and a little change was observed in the fission gas release (almost all He accounted to alpha-decay during irradiation). The surface of UO2 fuel cladding showed a 10 μm thick oxide film but the cladding was intact and there was no significant migration and fission gas release during 20 years of dry storage.

J. F. Lucchini, CEA, France, presented a paper on the effects of alpha radiolysis on spent fuel (SF) matrix corrosion. The presenter investigated the effect of external alpha radiation at the UO_2/H_2O interface and the effect of the radiolytic species H_2O_2 on UO_2 corrosion using UO_2 pellets containing alpha emitters. The corrosion rate was determined by electrochemical methods. The results showed that the amount of UO_2 released was 2-3 orders of magnitude higher for alpha doped UO_2 than for undoped UO_2. Results also showed that the U and H_2O_2 concentration increases and the pH decreases in the solution, as the alpha flux increases. The U release rate was lower for leaching tests with H_2O_2 solution than for those performed under irradiation indicating that radiolytic species other than H_2O_2 are involved in U dissolution.

V. Jain, CNWRA, presented a paper on the effect of waste package corrosion products on HLW glass dissolution. The test conditions were selected to simulate an internal WP environment containing steel corrosion products and oxidized by radiolysis. The modified product consistency test (PCT), with regular solution exchanges, was used to determine the leaching rates of simulated HLW glasses (WVDP Ref. 6 and DWPF Blend 1) in the presence of aqueous solutions of $FeCl_2$ and $FeCl_3$ at 90°C. Substantially high initial boron and alkali release rates, approximately a factor of 50 to 70 times greater than those in deionized water, were measured in 0.25 M $FeCl_3$ solutions. The preliminary leaching results suggest that both solution pH and the presence of corrosion species such as iron chloride are significant contributing factors for enhanced glass dissolution. The pH dependence data demonstrated that the ions are incongruently released from the glass at rates determined by the pH of the leaching solution. Based on the preliminary normalized boron release results, the presence of iron containing corrosion products enhance the contribution of the vitrified HLW to the source term. However, the magnitude of this effect on the dose for the Yucca Mountain repository needs further evaluation.

X. Orlhac, Universite Montepellier, France, presented a poster on the long-term thermal stability of the French nuclear waste glass. The glasses with and without platinoid elements were subjected to heat treatment at various temperatures and were analyzed for the amount and type of crystals in the glass. The results showed that a) a crystallization equilibrium line can be used to define the maximum possible crystallization fraction at a given temperature and b) platinoid elements serve as catalyst for crystallization, but do not affect the maximum crystallized fraction.

L. Bouchet, CEA, presented a poster on the phenomenological study of the reactions between glass frit and simulated fission products calcine. The dissolution of calcine in glass frit was studied using micro probe analysis, and samples were characterized using SEM/EDS and XRD. The study indicates impregnation of calcine by molten glass frit which leads to the formation of different crystals in the melt.

A poster by C. Lopez, CEA, examined the solubility of Pu surrogates in nuclear waste glasses. Ce is widely used as non-radioactive surrogate for Pu. In addition, Hf and Nd are used for Pu^{4+} and Pu^{3+} respectively. The study showed that solubility of CeO_2 increase from 0.25 wt % at 1,100 C to more than 15 w t % at 1,400 °C. Also, $Ce^{3+}/\ Ce_{Total}$ increased from 0.5 to 0.9. Hf and Nd also showed an increase in solubility with temperature. The results showed that Hf and Nd have a stable valance state that does not change with temperature.

I. T. Kim, Korea Atomic Energy Research Institute, Korea, presented a poster on the assessment and prediction of the long-term leaching behavior of glassy waste forms for stabilizing radioactive incineration ash. The waste form was developed using French R7T7 HLW glass. Tests were conducted on non-radioactive glasses containing Sr and Cs are radionuclide surrogates. Leaching tests were conducted at 70 °C for 820 days. The amount of leached material was calculated by a) solving transport equations for a semi-infinite medium with uniform initial concentration and b) assuming leach rate as inversely proportional to the square root of time, and assuming dissolution rate to control the release of radionuclides. Both models provided an estimation of leach rates with an accuracy of over 90 percent.

F. Bart, CEA, presented results of his studies on glasses containing high-concentrations of Cs. The glasses, high in alumina, were melted between 1,500 and 1,600 °C. The glasses showed low Cs volatility and much higher durability compared to borosilicate glasses.

Conversion of Weapon-grade Pu to MOX Fuel and Ceramic Waste Form

A. Michel, Pu 2000 Organizing committee, S. L. Zygmunt, Los Alamos National Laboratory, USA, and P. Brossard, CEA, France, gave presentations on the current status of conversion of weapon-grade Pu (w-Pu). Dr. Zygmunt indicated that initial engineering and chemical research and development has been completed for the Russian program for converting w-Pu to MOX fuel. The schedule requires disposal of 34 metric tonnes of w-Pu by 2012. A similar program is followed in United States. Mr. Brossard provided a summary of the joint efforts by French and Russian teams to evaluate a suitable process for MOX fuel processing. Based on this evaluation the CHEMOX process (an oxalic acid based process currently used in France) was selected.

In addition, several papers focused on development of ceramic waste forms that will be used for w-Pu disposition as a waste form. G. R. Lumpkin, Australian Nuclear Science and Technology Organization (ANSTO), presented a review paper on the radiation damage in pyrochlore and zirconolite ceramics used for immobilization of actinide-rich wastes. These waste forms are being evaluated for disposal of surplus weapons Pu. A. Jostsons, ANSTO, reviewed titanate ceramics and glass/ceramic composites for the immobilization of HLW and actinide-rich wastes. Dr. Jostsons reviewed the development history of the titanate ceramics and compared it with the existing waste forms such as glass.

Partitioning and Transmutation of Minor Actinides

Several oral presentations and posters focused on partitioning and transmutation of minor actinides. This area has been a subject of significant research in Europe, Japan and Russia. The main objective is to reduce radiotoxicity by separating minor actinides such as Am , Cm and Np, and subjecting them to further transmutation into short lived nuclides. Readers are referred to the website for further details on this topic.

SUMMARY OF ACTIVITIES:

The author, V. Jain, presented a paper at the meeting and attended oral presentations and posters relevant to waste forms. Europeans along with Russians and Japanese are carrying out significant research on partitioning and transmutation of minor actinides. CEA is conducting both radioactive and nonradioactive studies on development and characterization of glass and glass ceramic wasteforms, and spent nuclear fuels.

CONCLUSIONS:

The meeting was very useful in keeping current with the ongoing worldwide advancements in the back end of the nuclear fuel cycle. The participation at the meeting was a good opportunity to gather information and generate discussion on the nuclear waste management technologies.

PROBLEMS ENCOUNTERED:

None.

PENDING ACTIONS:

None.

RECOMMENDATIONS:

Participation at such meeting should be encouraged because these meeting provide an excellent avenue to discuss and present scientific findings to the international community.

SIGNATURES:

Vijay Jain, Manager
Corrosion Science & Process Engineering Element

11/21/00
Date

CONCURRENCE:

Budhi Sagar
Technical Director

11/21/2000
Date

VJ:BS:jg