UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MEMORANDUM FOR: Robert E. Browning, Director
Division of Waste Management, NMSS

FROM: Karl R. Goller, Director
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and Earth Sciences, RES

SUBJECT: POTENTIAL PROBLEM AREAS FOR HLW REPOSITORY

I want to call your attention to two potential problem areas that I recently learned about, which we will need to consider and on which you may want to interact with DOE to resolve. The first concerns the potential secondary consequences of radiation damage to salt. The second concerns the amount of radionuclides that may be leached from spent fuel and how this is predicted from experimental measurements.

1. CONSEQUENCES OF RADIATION DAMAGE TO SALT

The gamma radiation from HLW emplaced in a salt repository will decompose the salt into colloidal sodium and chlorine gas. The sodium metal will combine with brine to produce hydrogen, which may cause hydrogen embrittlement or hydrogen damage of the overpack. The chlorine will combine with the brine and with brine radiolysis products to produce oxygenated chlorine ions. The quantity of such ions present is important because they are very aggressive with respect to attack of overpack materials and, further, their specific corrosion chemistry is poorly understood.

DOE has conducted at BNL a program of damaging salt with various total gamma doses representative of salt repository designs at different dose rates to project expected colloid and chlorine production. Of course, performing laboratory experiments at realistic total doses means performing them at far higher dose rates than are expected in the repositories. The experiments done at BNL show that the best model currently available for predicting sodium colloid formation (U. Jain and A. J. Lidiard, Phil. Mag. vol. 35pp. 245ff (1977)) is accurate at high dose rates but seriously underestimates colloid production (and thus chlorine production) at low dose rates. It appears that at dose rates expected in repositories, the use of this model could underestimate colloid production by an order of magnitude or more.

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The weakness of this model appears to be connected with a superficial treatment of the very early stages of colloid nucleation and growth. This is consistent with the observation that it is good for high dose rates but fails at low dose rates. In 1982 work was ongoing (under ONWI subcontract E511-01000) to develop analyses of defect transport in salt which would permit more accurate predictions of sodium colloid formation under expected repository dose rates, and the first major element (a calculation of transport along ⟨100⟩ dislocations in the absence of random walk diffusion) was approaching completion. No report of this work is available in either the refereed literature or publicly available DOE documents.

2. LEACHING OF RADIONUCLIDES FROM SPENT FUEL

The second potential problem concerns the effect of the mechanical integrity of spent fuel on long-term leach rates. LWR fuel is composed of sintered UO_2 pellets. In general, grain boundaries of any sintered material are more vulnerable to aqueous leaching than the grains themselves. This means that leach rates of spent LWR fuel could be controlled initially by the solubility of the material in the grain boundaries at the surface of the spent fuel and although the grains themselves may be significantly more leach resistant than the material in the boundaries, as the material in the boundaries is leached the fuel loses its integrity, crumbles, and exposes individual grains to groundwater attack. Because of the small size of the grains ($\sim 20\mu$) their surface to volume ratio is quite high. Under this situation, the long term release of radionuclides from spent fuel may be underestimated by short-term leaching tests of intact spent fuel pellets for high groundwater flow rates.

If you or your staff have any questions on these matters, please contact Dr. Michael McNeil on X74636 or Dr. Kyo Kim on X74637.

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