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MASTER

OFFICE OF NUCLEAR WASTE ISOLATION

HEDL STABILIZER MATERIALS
PROGRAM

OCTOBER - DECEMBER QUARTERLY REPORT

Hanford Engineering Development Laboratory

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OFFICE OF NUCLEAR WASTE ISOLATION - HEDL STABILIZER MATERIALS PROGRAM
OCTOBER THROUGH DECEMBER QUARTERLY REPORT

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Enclosure: October through December Quarterly Report

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QUARTERLY REPORT
HEDL - SPENT FUEL STABILIZER MATERIALS PROGRAM
OCTOBER THROUGH DECEMBER 1980

I. Introduction

This report summarizes some of the activities of the following individuals employed at the Hanford Engineering Development Laboratory (HEDL) engaged in spent fuel stabilizer materials investigations.

V. J. Ferrell
R. L. Fish
N. Wynhoff

II. Program Objective

The objective of the Spent Fuel Stabilizer Materials Program is to identify, test, select, and qualify stabilizer materials for use in the design of spent fuel waste packages. Emphasis is placed on the selection of stabilizer materials with minimal interactions.

III. Progress Reported Previously

In support of the stabilizer material screening effort, all preliminary interaction, cost, and availability evaluations were completed. Criteria were established for the screening of stabilizer materials based on these preliminary analyses. Based on the criteria, the initial list of 67 potential stabilizer materials was reduced to 34 that were recommended to ONWI for further testing. The criteria used in screening included consideration of thermally induced stresses, criticality concerns, and material cost and availability.

Detailed planning for temperature limit, fill process, and prebreach disposal condition compatibility testing was completed. A test matrix was developed which emphasizes screening of materials rather than development of a data base for predicting long-term disposal effects. Procurement and preparation of test components for the temperature limit tests were ongoing last quarter.

IV. Activities During Reporting Period

The preliminary screening of candidate spent fuel stabilizer materials was completed and reported to the Office of Nuclear Waste Isolation in November 1980. Based on criteria set for thermally induced stresses, material cost, and material availability, an initial list of potential stabilizers was reduced to 34 that are recommended for additional screening tests.

The spent fuel stabilizer temperature limit test was completed during this report period. Tests were conducted to establish a conservative temperature limit for spent light water reactor (LWR) fuel during the stabilizer emplacement process. Melting of metals and metal alloys, fusing of glass stabilizers, or other possible emplacement processes may require elevation of the spent fuel temperature above levels desirable for maintaining fuel rod integrity.

(intact cladding) or reasonable inter-rod spacing for stabilizer flow. This test was conducted to establish the temperature below which these concerns would not exist. Stabilizers that require emplacement process temperatures above the limit will be eliminated from the screening program.

The temperature limit test specimen consisted of a helium-filled, Zircaloy-4 cladding section with welded end caps. A 416 stainless steel mandrel placed inside the cladding specimen occupied the same volume ratio and simulated the thermal expansion characteristics typical of fuel in a pressurized water reactor (PWR) fuel rod. The slightly higher 416 stainless steel thermal expansion coefficient and cladding specimen end restraints provided for a conservative simulation. Zircaloy-4 cladding specimens were backfilled with helium to 650 psi, sealed and leak tested. This pressure corresponds to a cladding stress in the specimen equivalent to the calculated end-of-life stress in 3-cycle PWR spent fuel cladding at temperatures experienced in a hot-cell. Calculations were based on an end-of-life pressure of 525 psi in Turkey Point PWR fuel rods.¹

The approach used to simulate the pouring or emplacement of the stabilizer around spent fuel assembly rods was as follows: The cladding specimens were placed in a test capsule, preheated in a separate furnace, and then placed in a test furnace at the selected test temperature for 16 hours. Calculations were performed to determine anticipated cool-down time for a full size waste canister. These calculations were used as a basis for using 16 hours as the test duration. While actual times above the melting point may be less than the 16 hours selected for this test, special processing techniques may require extended times above the melting point. Thus, 16 hours is considered to be reasonably conservative. After cooling to ambient temperature, the specimens were removed from the test capsule. The tests were conducted using different specimens at each of four temperatures ranging from approximately 500 to 650°C.

The diameter of each specimen was measured at three locations prior to each test run. The average diameter was calculated to provide a basis for determining the $\Delta D/D\%$ strain that occurred during simulated stabilizer emplacement. The maximum diameter of each specimen after test was measured and the $\Delta D/D\%$ strain calculated using the original average diameter as the reference.

The strain was plotted as a function of test temperature to facilitate the temperature limit determination as shown in Figure 1. The temperature limit was based on a 7.5% $\Delta D/D$ cladding strain criterion or failure, whichever occurred first. Since failure was not detected in any of the cladding specimens, the 7.5% strain criterion was used for establishing 590°C as the stabilizer emplacement process temperature limit. This was justified on the basis of stabilizer flow restriction. In PWR fuel assemblies, a reduction in rod-to-rod spacing in excess of 25% would be expected to significantly reduce flow of stabilizer material into the interstices between rods. A

¹R. B. Davis, R. L. Fish, V. Pasupathi, and D. E. Stellrecht, "Remote Characterization of Spent LWR Fuel for Geologic Disposal Demonstrations", to be published in Proceedings of the 28th Conference on Remote Systems Technology, February 1981.

FIGURE 1. EFFECT OF TEMPERATURE ON STRAIN
OF ZIRCALOY 4 CLADDING SPECIMENS

(TIME AT TEMPERATURE = 16 HOURS)

strain of 7.5% in the cladding specimens is equivalent to the expansion of fuel rods resulting in this 25% reduction in rod spacing. Unless we have assurance of continuous filling of the fuel assembly interstices, lithostatic loads may not be uniformly distributed across the spent fuel assembly, thus compromising one of the incentives for using solid stabilizer materials.

For purposes of screening, the emplacement process temperature limit criterion of 590°C provides a basis for acceptance or rejection of stabilizer materials. Immediate impact of this screening criterion on the stabilizer program is the rejection of copper and all its alloys (casting temperature $\geq 954^\circ\text{C}$) from further consideration as stabilizers for intact spent fuel rod waste packages.

Contacts with Nevada Nuclear Waste Storage Investigations (NNWSI), Basalt Waste Isolation Project (BWIP), and Westinghouse/Advanced Energy Systems Division (W/AESD) personnel were made to establish stabilizer performance functions and requirements. Preliminary discussions indicate that the primary function of the stabilizer should be to resist collapse of the canister due to geologic lithostatic and hydrostatic pressures. Agreement was voiced that either a particulate or a solid stabilizer could perform this function. This functional requirement would eliminate gases used alone as stabilizers. Although other functions would be served as a natural consequence of using either a particulate or a solid, these functions are secondary in importance. Therefore, testing efforts should probably focus on measuring properties that are related to maintaining the stabilizer's ability to perform its primary function as well as identifying materials that exhibit minimal interactions with the cladding and candidate canister materials. Some persons contacted expressed concern about the use of lead and its alloys because of its toxicity and the use of graphite and bentonite because of potential airflow problems with fine particles in hot cell filling operations. Concern was also expressed about the stability of Bondate in a radiation field. These concerns will be addressed and evaluated during the final screening process.

Concrete "Formed Under Elevated Temperature and Pressure" (FUETAP) has been added to the list of potential stabilizer materials as a result of discussions with John G. Moore of Oak Ridge National Laboratory (ORNL). The strength properties and moisture content of concrete emplaced using the FUETAP process pass the screening criteria established previously. This material will be included in the interaction screening tests.

A design review for the prebreach compatibility tests was held in October. Recommendations from that review incorporated into the test program included revision of the engineering test plan, completion of capsule stress and pressure overchecks, and establishing Quality Assurance requirements for retention of test program records.

Preparations were initiated for the prebreach fill process and disposal condition material interaction screening tests. Approximately 90% of the component machining was completed for these tests. All stabilizer materials have been received and test unit fabrication is 50% complete. Loss-on-ignition (LOI) tests were conducted on each of the particulate stabilizers.

As shown in the attached Table 1, only basalt and the bentonite exceeded the LOI water loss criterion of 1% established for use of particulate stabilizers. These particulates were dried for an additional 2-1/2 hours at 300°C; subsequent water content analyses indicated 0.6% in the bentonite and 0.5% in the basalt.

V. Problems

None.

VI. Activities Planned for Next Quarter

Stabilizer fill process screening tests will be conducted and prebreach disposal condition screening tests will be initiated. Stabilizer performance functions will be established.

TABLE 1

LOI TESTS

<u>Ident. No.</u>	<u>Particulate</u>	<u>% Wt. Loss</u>	
F-24	Graphite	.62%	} Weight Loss Over 2 Hrs. @ 550°C in Helium
F-23	Mullite	.06%	
F-25	Basalt	2.26%	
F-26	Zircon	.22%	
F-27	Zirconia	.08%	
F-29	Silica Amorphous	.25%	
F-30	Silica Quartz	.27%	
F-6	Na Bentonite	7.6% - H ₂ O Analysis @ 300°C	

H₂O Analysis

F-6	bentonite	} dried additional 2-1/2 hours @ 300°C	} 0.6% 0.5%	after 2	} 0.6% 0.5%
F-25	basalt			days in air	