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Spent-Fuel Test—Climax: An Evaluation of the Technical Feasibility of Geologic Storage of Spent Nuclear Fuel in Granite

Executive Summary of Final Results

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Executive Summary of Final Results

Abstract

This summary volume outlines results that are covered in more detail in the final report of the Spent-Fuel Test—Climax project. The project was conducted between 1978 and 1983 in the granitic Climax stock at the Nevada Test Site. Results indicate that spent-fuel can be safely stored for periods of years in this host medium and that nuclear waste so emplaced can be safely retrieved. We also evaluated the effects of heat and radiation (alone and in combination) on emplacement canisters and the surrounding rock mass. Storage of the spent-fuel affected the surrounding rock mass in measurable ways, but did not threaten the stability or safety of the facility at any time.

Introduction

The National Waste Terminal Storage (NWTS) Program was established in 1976 by the predecessor of the U.S. Department of Energy (DOE) to evaluate the feasibility of retrievable deep geologic storage of commercial nuclear reactor wastes. In September 1983 the NWTS became the Civilian Radioactive Waste Management program. This large, multidisciplinary program plans to create an operational repository in the 1990s.

Although it was clear that a large-scale field test would be ideal for demonstrating essential technologies and revealing unexpected effects of waste emplacement, it was also evident that there would be few opportunities for such testing early in the program. Therefore, the Spent-Fuel Test—Climax (SFT—C) was undertaken to demonstrate the feasibility of spent-fuel handling and to address technical concerns related to granitic rocks.

The final report (which will be published as Lawrence Livermore National Laboratory, Livermore, CA, Report UCRL-53702, 1987) covers all aspects of the design, development, conduct, and results of the SFT—C. This summary volume is intended to be sufficiently detailed to present key aspects of site characterization, test facility and equipment design, and test results. These topics vary in complexity, and the depth of discussion varies accordingly.

Each chapter of this volume is a summary of the corresponding chapter in UCRL-53702. Please refer to the full report for discussions of each subject and for specific references. The Bibliography at the end of this volume lists all reports published concerning the SFT—C.

Background

The SFT—C was conducted under the technical direction of the Lawrence Livermore National Laboratory (LLNL) for the DOE. As part of the Nevada Nuclear Waste Storage Investigations, it was managed by the Nevada Operations Office of the DOE.

The SFT—C facility (shown in Fig. 1) was located 420 m below surface in the Climax stock

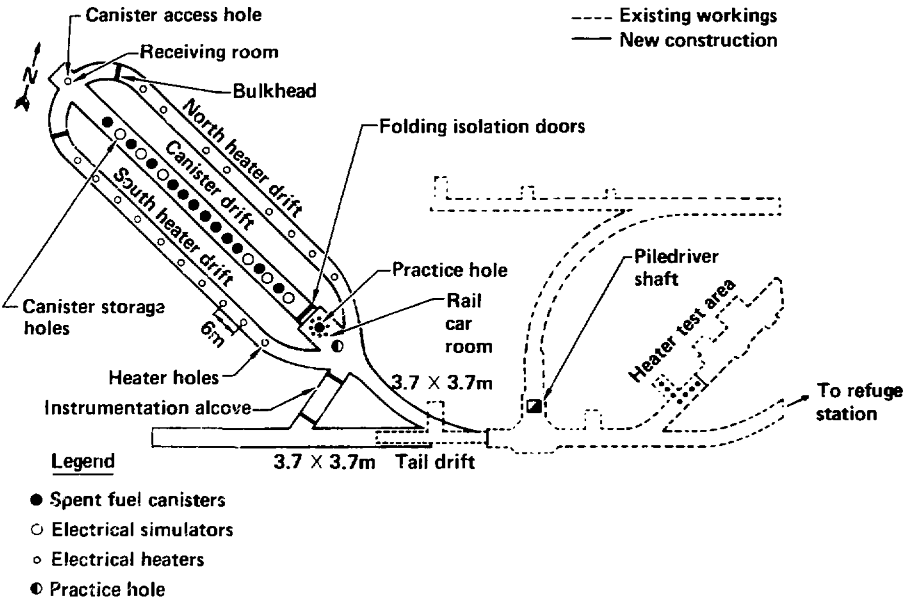

Figure 1. The SFT—C facility located 420 m below the surface in the Climax granite.

granite at the Nevada Test Site (NTS). Facilities were constructed between June 1978 (when funding for the test was initiated) and April 18, 1980 (when spent-fuel emplacement began). Spent-fuel was emplaced between April 18 and May 28, 1980, and retrieved between March 3 and April 6, 1983. Individual spent-fuel canisters were exchanged in January and October 1981 and August 1982. Post-test characterization followed retrieval of the spent-fuel and was completed during 1985.

The operational objectives of the test were to demonstrate safe and reliable packaging, transport, short-term storage, and retrieval of spent nuclear reactor fuel. This storage period corresponds to spent-fuel ages of 2.5 to 5.5 years out of core (YOC).

A technical measurements program was implemented to acquire data for the ultimate qualification of granitic rock as a repository medium, to aid in the design of such a repository, and to predict its response in granitic rock. Numerous technical objectives were established at the beginning of the test, as presented in the technical concept report (see Ramspott et al., UCRL-52796, 1979). Our activities focused on these stated aims throughout the test.

We recorded data continuously on more than 900 computer channels during the 3-year fuel storage phase and for 6 months following retrieval to record thermal and thermomechanical responses as the simulated repository environment cooled. Although most data were acquired through a central data acquisition system (DAS), periodic displacement measurements and radiation dosimetry data were acquired manually and processed independently of the DAS. Acoustic emissions data were also acquired independently.

Chapter Summaries

Chapter 1 of UCRL-53702 includes the information found in this volume. Chapters 2 through 18 are summarized below.

Chapter 2. Test Objectives

Both operational and technical objectives guided the design, development, and execution of the SFT—C. The overall operational objective of the test was to evaluate the feasibility of safe and reliable short-term storage of spent reactor fuel assemblies at a plausible repository depth, and to retrieve the fuel assemblies afterward.

An underlying technical objective was to simulate the near-field thermal effects of a panel of a full-scale repository within a relatively small test volume. An additional technical objective was to evaluate the effects of heat alone and the combined effects of heat and radiation on the near-field canister environment and materials.

Although the highest priority of the SFT—C was a timely demonstration of spent-fuel handling, the test also addressed the ultimate qualification of granitic rocks for geologic disposal and the design of future repositories in granitic or other hard rocks. The focus of these objectives was to model and collect data on heat transfer, rock mechanical response, and radiation transport processes. Evaluating our ability to model such phenomena was also an important element of the SFT—C.

Chapter 3. Test Design

We designed the test by using a series of calculations of heat transfer, rock mechanical response, and radiation transport processes. To make the SFT—C demonstration meaningful and technically sound, we designed it to simulate the early, near-field history of a panel of a full-scale repository containing thousands of spent-fuel assemblies. A suite of increasingly sophisticated heat transfer calculations was the primary basis of the design of the SFT—C facility.

By emplacing a central linear array of spent-fuel assemblies and simulated assemblies and flanking that array with auxiliary heaters, we matched the thermal conditions of a conceptual repository within 4.5% in a central portion of the SFT—C. Thermal calculations also estimated the temperatures of fuel cladding, handling system hardware, and ventilation air.

We also calculated the response of the rock mass to excavating underground facilities, extensive heating, and cooling following retrieval of the spent-fuel assemblies. Most calculations were made with the assumption that the rock was a homogeneous, isotropic, linearly elastic material; however, we also incorporated dilatant and nondilatant joints in estimates of excavation response to mining. Calculated rock responses were used to site instrumentation and later provided a basis for comparison with field data. As field-properties data became available, the laboratory values used in initial design calculations were revised. *In situ* stress measurements also replaced early estimates of the stress state.

We calculated radiation transport for two distinct purposes: To design acceptable radiation shielding and to determine the radiation dose to the granite rock near the spent-fuel emplacement boreholes. Hand calculations of isotropic abundances, gamma energy spectra, and neutron attenuation test data established the shielding necessary to limit dose rates to 100 mrem/h at the surface of the shielding casks and 0.5 mrem/h at the storage drift floor. Monte Carlo simulation of the complex spent-fuel storage configuration was used to estimate dose and dose rate at various positions outward from the spent-fuel assemblies. In addition, these calculations estimated the thermal contribution of gamma attenuation to be 40 W per canister.

Chapter 4. Site Characterization and Geologic Investigations

We studied the local structural geology and hydrology, measured the *in situ* state of stress, measured physical properties (*in situ* and in the laboratory), studied mineralogy and petrology of pre- and post-test cores obtained near the canister emplacement holes, and analyzed (by microfracture counts) potential drilling damage and radiation effects. We summarize below the more important observations.

Structural Geology

The Climax stock quartz monzonite (CSQM) in which the SFT—C was located is part of a two-component stock of Cretaceous Age that intrudes

Paleozoic carbonates and is partially overlain by Tertiary tuffs. We identified eight joint sets during mapping and core logging; three sets account for 80% of all joints mapped. An average frequency of three joints per metre produces a moderately fractured, blocky structure.

Hydrology

Relatively low annual rainfall (150 to 200 mm), high evapotranspiration, and sparse overlying sediment contribute to low infiltration of water into the stock exposed at the surface above the test facility. The zone of total saturation appears to occur at about 975 m above mean sea level, about 145 m below the test horizon.

In Situ Stress

In the north and south heater drift pillars, the maximum principal stress (sigma 1) is essentially vertical, the intermediate principal stress (sigma 2) is horizontal and aligned parallel to the long axes of the pillars, and the least principal stress (sigma 3) is parallel to the pillar width. "Free field" state-of-stress measurements indicate that the maximum principal stress is oriented toward the east-northeast and is nearly horizontal, the intermediate principal stress is nearly vertical, and the least principal stress is nearly horizontal and oriented north-northwest. These results are consistent with both the conclusions of previous investigators and the motions of active faults at NTS. The maximum secondary principal stress levels measured in boreholes ISS-9 and ISS-10 (located at and above the facility horizon) apparently differ systematically. The observed difference does not appear to be related to any geologic structural anomaly.

Physical Properties

Laboratory measurements indicate that Young's modulus increases with confinement, decreases with increasing temperature, and is relatively unaffected by prior treatments of heat and stress. *In situ* deformation modulus apparently depends on prior heating and stress conditions. The average field modulus is about half that measured in the laboratory. The coefficient of thermal expansion increases substantially with temperature and decreases somewhat as confinement increases. Thermal conductivity and diffusivity decrease with increasing temperature and with decreasing confinement. *In situ* testing produced values of 3.1 W/m·K and 1.2 mm^2/s, respectively. *In situ* permeability was about 1 nD at ambient temperature, decreasing to 0.2 nD at 50°C. No permanent change in gas permeability occurred as a result of heating and subsequent cooling.

Mineralogy and Petrology

The quartz monzonite is a porphyritic rock composed of a ground mass that is predominantly equant, subhedral grains of plagioclase, K-feldspar, quartz, and biotite. Grains range from 0.5 to 2.0 mm in diameter. Modal percentages for the primary phases of quartz, plagioclase, and K-feldspar are 17.3 ± 2.1%, 32.8 ± 5.6%, and 31.2 ± 5.3%, respectively. Igneous accessory phases make up less than 3 vol%. The Climax record cores show significant chemical, petrographic, and modal variations. These variations result from both igneous and hydrothermal processes.

Radiation and Thermal Effects

No statistically significant changes in strength or Young's modulus were observed after intense laboratory irradiation of CSQM. Mineralogical and petrographical characteristics did not change significantly in samples obtained from near spent-fuel storage boreholes or auxiliary heaters. The microfracture structure of Climax granite is highly heterogeneous on the scale of 0.1 to 10 mm, making it difficult—if not impossible—to discern damage produced by elevated stress, temperature, gamma irradiation, or combinations of the three. No statistically significant evidence of changes in microfracturing were observed to be produced by laboratory gamma irradiation.

Drilling Damage Effects

The damage induced by hammer-drilling the 0.61-m-diam spent-fuel emplacement boreholes is limited to an annular ring less than 20 mm thick around the borehole.

Chapter 5. Site Development and Facility Construction

The Piledriver shaft, headframe, and hoist system provided access to the 420-m test level. Developed in the 1960s for testing nuclear weapons effects, this access was used to support a heater test known as Heater Test No. 1 before the SFT—C was authorized. After the shaft was internally refurbished, we drilled an additional 0.76-m-diam shaft and lined it with a 0.51-m-o.d. steel pipe section, which was cemented in place. This canister access shaft allowed encapsulated spent-fuel assemblies to be moved between the surface and the underground. Two 3.4-m × 3.4-m cross-section parallel drifts were driven about 70 m to connect drifts from the main shaft with the base of the canister access shaft. The floors of these drifts housed the auxiliary heaters during the heated phase of the SFT—C. After the heater drifts were built, the central canister storage drift was driven in two passes: 4.0-m-high × 4.6-m-wide top heading and a 2.1-m × 4.6-m bench. Controlled blasting techniques were used to excavate some portions of this drift.

Subsurface outfitting included installing ground control measures, placing a reinforced concrete floor with embedded shielding pits and rails, drilling emplacement boreholes for spent-fuel and heater installations, and installing nearly 1000 instruments.

On the surface, construction included the ventilation and filtration system, the DAS trailer, power and water supplies, and the spent-fuel canister hoisting system.

Chapter 6. Spent-Fuel Characterization

The 13 spent-fuel assemblies used in the SFT—C came from the Florida Power and Light Co. Turkey Point Unit 3, a Westinghouse-designed commercial pressurized water reactor. Hanford Engineering Development Laboratory and the Battelle Columbus Laboratory characterized the intact fuel assemblies and individual fuel rods. Their evaluation was as follows: The average length and cross-section width of the assemblies were 3.903 m (153.65 in.) and 211.1 mm (8.312 in.), respectively. The average weight of the assemblies was 663 kg (1459 lb). A combination of gas and wet sip testing indicated that no assemblies contained leaking rods. Visual inspections revealed no major anomalies (such as bulges, scars, or blisters) on the rods, although profilometry detected circumferential ridges as high as 0.076 mm (0.003 in.) at pellet interfaces.

In September 1979, a total neutron flux of 1.06×10^4 n/cm^2/s with an average energy of 1.4 MeV was measured and found to be distributed fairly symmetrically around the canister axial midplane. An average gamma flux of 9.11×10^4 R/h was determined. In March 1984, one year after fuel retrieval, the gamma dose rate was measured to be 2.33×10^4 R/h. ORIGEN2 calculations and a series of boiling-water calorimetries established the decay-power-generation curve for the spent-fuel assemblies. During the test, the heat-generation rate decreased from about 1500 to 600 W per canister.

Chapter 7. Spent-Fuel Handling System

Spent-fuel assemblies were shipped from the Turkey Point reactor to the engine maintenance, assembly, and disassembly (EMAD) facility in southeastern Nevada by truck-mounted, licensed, commercial shipping casks. Once the casks were received and encapsulated, a three-component handling system was used for all operations at NTS. Each assembly was encapsulated in a 356-mm- (14-in.-) o.d. stainless steel canister that was sealed, evacuated, and backfilled with helium gas. An integral shield plug and grappling knob provided the necessary interface with the handling system.

The handling system had three major components: The surface transport vehicle (STV)—a special trailer with a rotatable shielding cask and a commercial tractor—was designed and developed to move the encapsulated spent-fuel from EMAD to the SFT—C. Top and bottom gates permitted loading through the top at EMAD and unloading and loading through the bottom at the SFT—C. All functions were remotely controlled. The underground transfer vehicle (UTV)—a rail-mounted, remotely controlled shielding cask with an onboard jib crane—received the spent-fuel canisters underground and emplaced or retrieved them from the individual storage boreholes. A canister-handling system—a specially designed wire-line hoist with control and braking subsystems—lowered and raised the spent-fuel between the STV and the UTV. An automatically actuated brake travelled with the spent-fuel canister for protection in the event of a hoist or cable failure. Control systems and closed-circuit television permitted remote operations with no risk of excess radiation exposure to personnel.

Chapter 8. Safety Assessment

The conduct of the SFT—C required that a safety assessment document (SAD) be prepared. A similar document was also developed to cover supporting operations at the EMAD facility. The SAD showed that no unacceptable radiological or nonradiological consequences to site personnel, the public, nor the environment would result from normal operations, abnormal operations, or postulated accidents. Calculations showed that nuclear criticality could not occur for any postulated fuel configuration, even with flooding.

Chapter 9. Selection, Deployment, and Performance of Instruments

Nearly 1000 instruments were deployed and operated for 3.5 years to monitor the temperatures of rock, air, and metallic components of the test; displacements and stress changes in the rock mass; radiation dose to personnel and to the rock; thermal energy input; characteristics of the ventilation airstream; and the operational status of the test. Careful selection, installation, calibration, and maintenance of these instruments allowed acquisition of about 15.3×10^6 high-quality data points.

Our studies show that currently available continuous air monitor (CAM) and remote area monitor (RAM) systems adequately monitor radiation safety in an underground environment. However, long-term gamma radiation dosimetry techniques were augmented by short-term thermoluminescence dosimetry (TLD) measurements.

Properly sheathed, commercially available RTDs, thermocouples, and thermistors provide reliable, accurate, long-term temperature data. Moisture-balance and energy-removal calculations were found to require better calibrations of flowmeters than are commonly available. In addition, systematic changes in Watt transducer calibrations suggest a need for periodic calibrations if accuracies better than 5% are needed.

Available vibrating-wire stressmeters have been failure-prone in the past and do not provide data with an accuracy better than about $\pm 50\%$. The several types of rock displacement gauges functioned reliably and accurately with one exception: Linear potentiometers failed early in the test, resulting in loss of near-field rock displacement data.

Chapter 10. Data Acquisition and Management Systems

A dual HP1000 disk-based DAS was developed to control instrument scanning, to provide preliminary data conversion, and to generate remote alarms if the data were outside of anticipated limits. The DAS performed all functions from acquiring analog signals through digitizing and archiving the raw data records on magnetic tape for subsequent detailed conversion by the data management system (DMS).

The DAS functioned accurately and reliably throughout the 3.5-year storage and cool-down phases. System statistics show that system availability averaged about 96% (functionally disabled index was 4%), providing about 15.3×10^6 data points during the test. The accuracy of dc voltage measurements was maintained within ± 4 μV. The accuracy of four-wire resistance measurements was occasionally outside the anticipated ± 0.0092-Ω envelope because of periodic digital voltmeter failures.

In addition to several utility functions, the DMS included a binary reading and screening code, a conversion code with algorithms for individual calibrations and temperature compensation, and a set of file manipulation codes for organizing the raw and converted data files into individual files that contain all data for each instrument. This latter feature allowed easy access to the data for plotting. The DMS statistics indicate that of the 15.3×10^6 points recorded by the DAS, 8.7×10^6 were retained, processed, and archived for analysis. An additional 6.3×10^6 data points from radiation monitors and test status monitors were valid but not retained. Only 2.5×10^5 points (1.6% of the total) were discarded as invalid.

Chapter 11. Spent-Fuel Handling Experience

Spent-fuel handling was at the heart of the SFT—C, which had as its main goal the demonstration of the feasibility of safe and reliable transport, storage, and retrieval of spent nuclear fuel. Operating procedures, administrative controls, and personnel training contributed to attaining this objective.

Eleven encapsulated spent-fuel assemblies were stored at the SFT—C between April 18 and May 28, 1980, at a rate of about two per week. To maintain a state of readiness for both personnel and equipment, we exchanged single spent-fuel assemblies between the SFT—C and EMAD three times: January 12 to 14, 1981; October 26 to 28, 1981; and August 16 to 18, 1982. Between March 3 and April 6, 1983, all 11 assemblies were retrieved and returned to temporary storage at the EMAD. These operations showed that spent-fuel can be safely handled with available technologies.

Total radiation exposure to operating personnel was 0.4 person-rem. Although the highest exposure was 10 mrem per handling operation, the average was about 30% of this value. Neutron exposure was greater than gamma exposure. Expressed relative to the energy generated while the spent-fuel was in the reactor, the normalized dose commitment at the SFT—C was 0.002 person-rem/MW·y (electric), a small fraction of the 1 to 4 person-rem/MW·y (electric) received during power plant operations.

Chapter 12. Thermal Sources

Three principal sources deposited energy into the rock mass during the SFT—C: spent-fuel assemblies, electrical simulators, and auxiliary heaters. Facility lighting also contributed to the input energy.

These thermal sources were monitored during the test to ascertain their energy input for use in thermal and thermomechanical calculations of SFT—C response. Measurements show that total thermal energy input to the SFT—C was 1041 MW·h during the 3-year storage phase, with 19 MW·h added during post-retrieval activities. The input energy partition was 25.3% from the 11 spent-fuel assemblies, 14.2% from the 6 electrical simulators, 57.7% from the 20 auxiliary heaters, and 2.8% from the facility lights. Electrical sources of heat, associated controllers, and instrumentation were very reliable.

Chapter 13. Heat Transfer Analyses and Measurements

Measured temperatures were compared with the heat transfer calculations described briefly in Chapter 3, with the following observations and conclusions. The intended SFT—C simulation of emplacing thousands of spent-fuel assemblies in a hypothetical repository was successful. Throughout the test, measured temperatures in the vicinity of the spent-fuel storage boreholes were within 3°C of those calculated at the axial midplane of the heat sources. Measured temperatures were somewhat higher than calculated near the top and somewhat lower than calculated near the bottom of the heat sources. Comparison of measured and calculated temperatures throughout the approximately 10,000 m^3 instrumented volume indicates very good agreement. Pairwise plots of measured and calculated temperature increases may be fit by straight lines with near-unity slopes and near-zero intercepts. Associated mean-square errors are generally about 2°C, only slightly greater than the 1.1°C ISA special limits of error for the thermocouples. Analytical and finite-difference models of the finite-length geometry of the SFT—C produced marked improvements in the level of agreement between data and calculations near the ends of the test array, where the infinite-length assumption of the early models is no longer valid. Temperature-measuring instruments functioned accurately and reliably during the test, as confirmed by calibrations.

Chapter 14. Ventilation System Analyses and Measurements

We documented the energy removed from the SFT—C by the ventilation system by measuring inlet and outlet air temperatures, dewpoints, and air flowrates for the three drifts. These measurement systems functioned reliably. The ventilation system removed a total of 148 MW·h during the spent-fuel storage phase of the test. Of this, 76.7% was removed as sensible heat and 23.3% as latent heat of vaporization. About 20 tonnes of water were removed from the facility each year in the ventilation airstream.

Attempts to calculate energy removed by the ventilation system were marginally successful. After trying a variety of values for the pertinent parameters, we found that good agreement between measured and calculated rock temperatures was achieved only by calculating with an energy-removal rate much higher than that measured.

Chapter 15. Radiation Transport Calculations and Measurements

The radiation dose calculations described in Chapter 3 form the basis for comparison with data measured during the test. In addition to personnel dosimetry and areal radiation monitoring, we also instrumented the rock near selected spent-fuel storage boreholes. The latter measurements used optical-grade LiF dosimeters at the borehole wall and at positions 200 and 360 mm into the rock. Neutron dosimeter foils were also incorporated with these measurement packages. The RAM and CAM areal monitors performed well but exhibited drift that required periodic adjustment. The fade characteristics and temperature sensitivity of the LiF dosimeters required that they be augmented with short-term $MgBO_4$ and CaF_2 TLDs. The Monte Carlo radiation transport calculations were generally more accurate than the measurements. However, generally good agreement was observed, particularly with the short-term dosimeters.

Radon-thoron concentrations in the air gradually increased from 1×10^{-10} to about 6×10^{-10} Ci/m^3 as the rock mass was heated. The log of concentration was found to increase nearly linearly with a decrease in ventilation airflow rate. As noted in Chapter 11, measurements of radiation doses to man indicate that minor whole-body doses were received during spent-fuel handling operations and no whole-body dose above background was received during spent-fuel storage. Very low finger doses were recorded for technicians who installed thermocouples on the emplaced canisters.

Chapter 16. Rock Mechanical Response Calculations and Measurements

Calculations were made and data were obtained for each phase of the SFT—C: excavation, heating, and post-retrieval cooling. The calculations described briefly in Chapter 3 are the basis for comparisons with data. The response of rock near underground openings to the excavation process was dominated by the behavior of joints. This behavior has not been adequately modeled by either homogeneous, isotropic, elastic formulations or by two-dimensional models, which include joints with either dilatant or nondilatant behavior. The rock-mass response during extensive heating was calculated quite well by the linearly elastic formulations, provided that measured *in situ* stress values and proper field deformation properties were incorporated in the model. Accurate calculation of temperature changes is essential for good rock-response results. Since commonly available codes do not incorporate all heat-flow features of interest, we found it necessary to simulate such processes as thermal radiation and ventilation by means of "effective" conductive properties.

Most displacement instrumentation performed accurately and reliably during the test. Stress-change instrumentation failed, causing loss of all early data. In addition, near-field extensometer potentiometers also malfunctioned, producing loss of early data near the emplaced spent-fuel assemblies.

Rock response to post-retrieval cool-down was calculated to be very small. The heated phase instruments were augmented during cooling by instruments designed and fabricated to monitor displacements within selected canister emplacement holes after spent-fuel was retrieved.

Chapter 17. Acoustic Emission and Wave Propagation Monitoring

We studied acoustic emissions (AE) and wave propagation to improve our understanding of the rock mass response to heating and to determine if these techniques could reliably monitor changes in a full-scale repository. Continuous AE monitoring began about three months before spent-fuel emplacement and continued through the post-retrieval cooling phase. About six months after spent-fuel emplacement, we installed wave propagation instruments. Data were acquired and processed with an automated seismic processor (ASP).

The frequency of occurrence of AE is directly related to changes in the rate of energy deposition to the rock mass. Adjustments to heater power levels and emplacement or retrieval of heat sources (such as spent-fuel assemblies) produce rapid increases in AE that decrease to background levels within a few days. Analyses of AE data indicate small-scale shear displacement or fracturing on the order of 0.01 to 0.05 mm per event, with source dimensions of several centimetres. Changes in the ratio of S- to P-wave amplitudes recorded over path lengths of several metres qualitatively agree with temperature changes in the rock mass. These changes are hypothesized to result from fracture closure. No measurable variations in P- and S-wave velocities occurred during the monitoring period.

Chapter 18. Metallurgical Investigations

We conducted metallurgical analyses to examine failures in emplacement borehole liner welds and in Superinvar connecting rods from borehole extensometers, and to examine corrosion of a spent-fuel canister and thermocouple sheathing. Inadequate weld penetration caused at least one emplacement borehole liner to leak at the connection of the bottom plate and the pipe section. The stainless steel canister stored within the leaky liner showed no evidence of general or localized corrosion even though it withstood temperatures of about 95°C and radiation dose rates of 10^4 rad/h while immersed in Climax stock groundwater for at least eight months.

Superinvar connecting rods corrosion-cracked when in contact with Climax groundwater at temperatures near 55°C. Thermocouple sheathing corroded in two cases where the thermocouples were erroneously sheathed in stainless steel rather than Inconel 600, as specified.

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National Technical Information Service
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