MARTIN MARIETTA

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ENVIRONMENTAL RESTORATION PROGRAM

Annual Summary Report of the Decontamination and Decommissioning Surveillance and Maintenance Program at Oak Ridge National Laboratory for Period Ending September 30, 1994

ENERGY SYSTEMS

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MANAGED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

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Energy Systems Environmental Restoration Program
ORNL Decontamination and Decommissioning Program

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Waste Management and Remedial Action Division
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CONTENTS

ABBREVIATIONS

ACB	auxiliary charcoal bed
CSTL	Coolant Salt Technology Facility
D&D	decontamination and decommissioning
DF	Decontamination Facility
DMC	Documentation Management Center
DOE	U. S. Department of Energy
ER	Environmental Restoration Division
FPDL	Fission Product Development Laboratory
FPPP	Fission Product Pilot Plant
HEPA	high-efficiency particulate air
HLCDL	High-Level Chemical Development Laboratory
HRE	Homogeneous Reactor Experiment
I&C	Instrumentation and Control Division
IH	Industrial Hygiene Section
LITR	Low-Intensity Test Reactor
LLLW	low-level liquid waste
MRF	Metal Recovery Facility
MSCL	Molten Salt Corrosion Loop
MSRE	Molten Salt Reactor Experiment
NCS	nuclear criticality safety
NEPA	National Environmental Policy Act
OGR	Oak Ridge Graphite Reactor
OHF	Old Hydrofracture Facility
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
P&E	Plant and Equipment Division
PCB	polychlorinated biphenyl
PHS	preliminary hazard screening
QD	Quality Department
RP	Radiation Protection Section
RRD	Research Reactors Division
S&M	surveillance and maintenance
SAR	safety analysis report
SFMP	Surplus Facilities Management Program
SFS&M	surplus facilities surveillance and maintenance
STT	Shielded Transfer Tanks
TLD	thermoluminescent dosimeter
TRU	transuranic
USQ	unreviewed safety question
WBS	work breakdown structure
WEF	Waste Evaporator Facility
WMRAD	Waste Management and Remedial Action Division
WOCC	Waste Operations Control Center

PREFACE

This *Annual Summary Report of the Decontamination and Decommissioning Surveillance and Maintenance Program at Oak Ridge National Laboratory for Period Ending September 30, 1994*, was prepared to comply with the intent of DOE Order 5820.2A, "Radioactive Waste Management." This work was performed under Work Breakdown Structure 1.4.12.2.01 (Surveillance and Maintenance). The Annual Summary Report documents the accomplishments for FY 1994.

EXECUTIVE SUMMARY

The Surplus Facilities Management Program (SFMP) was established at Oak Ridge National Laboratory (ORNL) in 1976 to provide collective management of all surplus sites under ORNL's control on the Oak Ridge Reservation. Presently, over 50 facilities, grouped into projects, are currently managed by the Decontamination and Decommissioning Program, the successor program to the SFMP. Support includes (1) surveillance and maintenance planning; (2) routine surveillance and maintenance; and (3) special maintenance projects. This report documents routine surveillance and maintenance, special projects, and special maintenance performed on these facilities for the period of October 1993 through September 1994.

1. INTRODUCTION

The Oak Ridge National Laboratory (ORNL) Decontamination and Decommissioning (D&D) Program has continued to provide surveillance and maintenance (S&M) support for over 50 surplus facilities. The objectives are to (1) ensure adequate containment of residual radioactive materials remaining in the facilities, (2) provide safety and security controls to minimize the potential hazards to on-site personnel and the general public, and (3) manage the facilities in the most cost-effective manner while awaiting decommissioning. This support has included work in three principal areas: (1) S&M planning, (2) routine S&M, and (3) special projects designed to correct serious facility deficiencies beyond the scope of routine maintenance.

2. SURVEILLANCE AND MAINTENANCE PLANNING

2.1 DOCUMENTATION COMPLETED IN FY 1994

Staff members from Surplus Facilities Surveillance and Maintenance (SFS&M) Department, (representing ORNL's D&D Program), participated in a number of meetings and working sessions held by Martin Marietta Central Environmental Restoration Division. These activities led to the generation of the Environmental Restoration (ER) Section in the *Martin Marietta Energy Systems, Oak Ridge, Standard Requirements Identification Document*, that was published in April 1994.

SFS&M staff completed the compliance assessment for the U. S. Department of Energy (DOE) Order 5700.6c requirement units for the Price-Anderson Amendment Act implementation assessment for the Molten Salt Reactor Experiment (MSRE), which is the only ORNL facility, to date, that has been categorized as a Price-Anderson Amendment Act Nuclear Facility.

SFS&M staff generated a Conduct of Operations Manual this year to define the structure of the department for properly addressing the 18 Chapters from DOE 5480.19, *Conduct of Operations Requirements for DOE Facilities.*

ORNL's Waste Management and Remedial Action Division (WMRAD) determined the hazard classification for each of its facilities, in accordance with DOE Order 5481.1B, *Safety Analysis and Review System,* and the guidance provided in ES/CSET-2 R1, *Hazard Screening Application Guide.* These hazard classifications now require revision, in accordance with DOE orders and standards that have been issued since the original hazard classification.

A report has been issued regarding procedures for performing preliminary hazard screening (PHS) and nuclear hazard categorization in accordance with DOE Order 5480.23, DOE-STD-1027-92, and DOE-EM-STD-5502-94.

PHS documents have been drafted for all of ORNL SFS&M's facilities. These reports are scheduled to be finalized and issued early in FY 1995.

2.2 DOCUMENTATION RESCHEDULED FOR FY 1995

Updating and reissuance of the SFS&M's S&M Plan was scheduled for completion in FY 1994. This effort has been delayed because of lack of staff availability caused by the heavy manpower requirement generated by the MSRE investigation and subsequent activities. This update of the S&M Plan will be rescheduled for FY 1995.

Very little effort was expended toward the Aging Facilities determination for ORNL, as requested by the DOE Oak Ridge Operations Office. This, again, was due to the unavailability of D&D staff working on the MSRE and other high-priority activities. The Aging Facilities determination will be rescheduled for a later time.

2.3 PRELIMINARY FY 1995 WORK PLAN FOR MSRE

The occurrence of a gaseous compound of uranium in the MSRE fuel storage system indicates that long-term storage of the fuel salt in its present configuration may not be acceptable. The preliminary work plan[1] summarized below will provide the information and develop the long term storage or disposition approach for the MSRE fuel salt.

The Work Breakdown Structure (WBS) for the MSRE UF_6 Migration Remediation underway will be the basis for budgeting, tracking progress and costs, and reporting. The principal elements of the work plan were defined using the logic diagram shown in the referenced document, together with details of each task. The work plan is organized into 6 major packages with 26 tasks. The task titles are the following:

WBS 1.0	URANIUM DISTRIBUTION
WBS 1.1	Auxiliary Charcoal Bed
WBS 1.2	Drain Tanks
WBS 1.3	Piping and Vessels (Vent House, North Electrical Services Area, and Balance of Plant)
WBS 1.4	Containment Cells
WBS 2.0	FACILITY SAFETY EVALUATION
WBS 2.1	Nuclear Criticality Safety Evaluation
WBS 2.2	Corrosion Evaluation
WBS 2.3	Accidental Gas Release Risk Assessment
WBS 2.4	Safety Analysis and Documentation
WBS 2.5	Unreviewed Safety Question Determination
WBS 3.0	INTERIM CORRECTIVE MEASURES
WBS 3.1	Lower Water Level in the ACB Pit
WBS 3.2	Partition the Off-Gas System
WBS 3.3	Eliminate Water Sources
WBS 3.4	Revise and Implement S&M Procedures
WBS 3.5	Install Criticality Accident Alarm System
WBS 4.0	TESTS AND EVALUATION
WBS 4.1	Facility Configuration
WBS 4.2	Corrosion Tests
WBS 4.3	UF_6 Formation Mechanism
WBS 4.4	Evaluation of Annealing Data
WBS 4.5	Sample Drain Tank Cell Atmosphere
WBS 4.6	Auxiliary Charcoal Bed Chemical Reaction Analysis

[1] Refer to draft of *Molten Salt Reactor Experiment UF_6 Migration Remediation—Status Report,* August 17, 1994 and the *Molten Salt Reactor Experiment UF_6 Migration Remediation—Preliminary Work Plan,* dated September 30, 1994.

WBS 5.0	REACTIVE GAS AND URANIUM DEPOSIT REMOVAL
WBS 5.1	Purge and Trap
WBS 5.2	Removal of Uranium Deposits
WBS 5.3	UF_6 - UO_3 Conversion
WBS 5.4	Uranium Transport and Storage
WBS 5.5	Safeguards Management
WBS 5.6	Waste Disposal
WBS 6.0	LONG-TERM FUEL DISPOSITION

For details of the tasks in this remediation activity, refer to Section 4.1.

3. ROUTINE SURVEILLANCE AND MAINTENANCE

3.1 MOLTEN SALT REACTOR EXPERIMENT

History and Background

The MSRE was a single-region, unclad-graphite moderated. homogeneous-fueled reactor built to investigate the practicality of the molten salt reactor concept for commercial power applications. It was operated from June 1965 to December 1969. After operations terminated, the facility (Building 7503) was placed in a shutdown status with the fuel in place in the fuel drain tanks, and an S&M program was initiated.

1994 Routine Surveillance Activities

- Continuous surveillance of elements and components of the MSRE have been conducted by WMRAD and documented in the Waste Operations Control Center (WOCC) log and the Documentation Management Center (DMC). Daily routine visual inspection of the building and annual preparation/and upgrading of inspection procedures have been conducted , with data entered into the facility log and the DMC.

- Annual surveillances of the following elements have been documented in annual checksheets and in DMC including (a) sump pump operability, (b) ventilation system check, and (c) verification check of switches and valves. This was accompanied by annual review of routine inspection records and preparation/maintenance of surveillance procedures. A separate semi-annual safety inspection was conducted and documented in a WMRAD memo.

- The Laboratory Protection Division conducted daily routine security patrols, monthly fire safety inspections, and an annual test of the fire sprinkler system.

- Radiation Protection (RP) personnel performed weekly, routine, radiological surveillances. "As Required" surveillances of maintenance activities and material transfers were also performed, with data for both activities entered into the RP database.

- Quality Department (QD) supportive activities were directed toward annual (and after replacement) DOP testing of high-efficiency particulate air (HEPA) filter system, and overhead crane inspections prior to crane operation. Documentation in WMRAD printout (HEPA) and by QD memo (crane service) was accomplished.

Routine Maintenance Activities

- Maintenance activities by WMRAD, Plant & Equipment (P&E), and Instrumentation and Control (I&C) Divisions provided support for exhaust filter changeouts, maintenance of heating/cooling systems, utilities, and overhead bridge crane, calibration, maintenance, and repair of instrumentation, maintenance materials and general maintenance and repair.

- All activities were documented in records (WMRAD and RP), reports (P&E), and in I&C preventative maintenance cards and QD printouts.

Other MSRE FY 1994 Activities

- The Nuclear Safety Review update for fuel and flush salt storage at MSRE was completed. The update was approved for a 2-year time period.

- An Energy Systems readiness review board gave approval to initiate sampling activities for the MSRE Vent House Sampling Project. Gas/aerosol sampling of the gas environment in the piping in the Vent House was conducted and completed. Sampling results and subsequent investigations and gas analyses have revealed the presence of a significant partial pressures of F_2 (350 mm Hg) and UF_6 (70 mm Hg) and a small amount of gaseous MoF_6 and HF in the piping. About 2.5 kg of uranium were located in the Auxiliary Charcoal Bed (ACB). A smaller quantity of uranium was found in the helium purge gas lines near the outer side of the concrete shield wall for the Drain Tank Cell.

- The second sample taken from the MSRE Vent House was sent to another laboratory because the dose rate was too high for the laboratory that originally received it. A glove box was fabricated at the new laboratory inside the existing hood that will aid in manipulating the sample bottle free from contamination.

- The second shield plug on the ACB was rotated to allow insertion of thermoluminescent dosimeters (TLDs) and neutron equipment to determine the radiation field within the bed. The charcoal bed water level was lowered 7 in. below the charcoal bed inlet to prevent inleakage of water should that inlet fail.

- The ACB, which is a part of the off-gas treatment for the fuel drain tanks, is located in a 10 ft diameter, 24 ft deep underground it which was filled with water to cool the charcoal traps and provide shielding during reactor operation. The ACB pit had remained full of water since reactor shutdown such that the charcoal traps were fully submerged when uranium was discovered to be located in approximately the first foot of charcoal at the inlet of the first U-tube.

- A remote video inspection was made of the entire length of the ACB which indicated no visible defects or signs of inleakage. Thermocouples in the vicinity of the uranium deposit were also read and indicated a modest temperature gradient (approximately 9°F) between the ambient water temperature and the centerline of the charcoal. The water level was lowered below the welds on the inlet piping (approximately 2 ft) on July 19, 1994. Preparations are underway to lower the water below the uranium deposit (an additional 2 ft) after additional temperature measurements and heat transfer calculations are completed and reviewed, if the projected temperature increase is acceptable.

- The valve which isolates the ACB from the drain tanks is inoperable (in the open position) for an unknown reason. In order to preclude additional gas transport to the ACB, priority is being given to isolating the ACB by other means. The source of the fluorine and UF_6 is in the fuel drain tanks which can be isolated from the ACB by remote actuation of three valves in the drain tank cell. The air system for these valves had been deactivated and control instrumentation had been removed after shutdown. New valve controls and an air supply have been installed to operate these valves.

- Candidate neutron absorbers to add to the charcoal pit water are being considered and test planning is underway to determine if they can be put in solution without adverse effects on the water chemistry.

- A portable criticality detection unit was installed and tied to local annunciator horns in Buildings 7503 and 7509, and was linked to the ORNL WOCC for 24-h monitoring.

For additional information related to the ongoing MSRE UF_6 Migration Remediation effort, refer to Sect. 4.1.

3.2 OAK RIDGE RESEARCH REACTOR

History and Background

The Oak Ridge Reservation (ORR) was a light-water-moderated and cooled, beryllium and water reflected research reactor designed and built for use as a general purpose research tool. The reactor core was a heterogeneous type that used enriched uranium fuel in the form of aluminum-clad, aluminum-uranium-alloy fuel plates. Reactor heat was transferred from the fuel by light water and dissipated to the atmosphere. The ORR was placed in service in March 1958 until the summer of 1960 at 20 MW. Improvements to the cooling system allowed an increase in power level to 30 MW from 1960 through shutdown in 1987.

1994 Activities

- Routine S&M were conducted as scheduled. This included routine site inspections, completion of facility checksheets and logbooks, radiological and industrial hygiene (IH) surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and ground maintenance have been included.

- In the July 1994 time frame, a tour of the ORR facility was conducted for D&D staff to determine the acceptance criteria adherence by Research Reactors Division (RRD) for facility turnover. Also, maintenance files for the ORR were transferred to the D&D office and S&M checks were started by WMRAD. The piping modification status and D&D acceptance issues were addressed between RRD and WMRAD staffs.

- In August 1994, polychlorinated biphenyls (PCBs) were discovered in basement room 11 in light fixtures and ballast. Clean up of PCBs was completed on August 29.

3.3 HOMOGENEOUS REACTOR EXPERIMENT

History and Background

This facility was originally constructed (1951) to house the Homogeneous Reactor Experiment-1 (HRE-1), the first of two experimental aqueous homogeneous reactors to be developed for nuclear power application analysis. In 1953, a decision was made to replace HRE-1 with a new experiment (HRE-2), and the second reactor was constructed between 1953 and 1956. The reactor was shut down in April 1961.

1994 Activities

- Routine S&M was conducted. This included routine site inspections, completion of facility checksheets and logbooks, radiological and industrial hygiene surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and ground maintenance have been included.

- On October 20, 1993, a safety inspection of HRE was conducted. Findings included mercury (approximately 100 mL) and oil (approximately 200 mL). A satellite area was set up in the HRE Laboratory Room to remove the mercury and oil.

- The condensate and turbine pits at the HRE were sampled for radiological and metal contents; approximately 10,000 gal of water were pumped to the building sump for discharge. Because of inleakage, several assays have been conducted on the inleaking water and a determination made that the potable water outside and west of Building 7500 is leaking. The point of discharge is being pursued.

3.4 OAK RIDGE GRAPHITE REACTOR

History and Background

The Oak Ridge Graphite Reactor (OGR) was the first reactor constructed at ORNL, being placed in service in 1943 at a 1-MW power level. In 1944, improvements in the cooling system and fuel cladding allowed the power level to be increased to an average level of 3.6 MW. The reactor was successfully operated for 20 years and was shut down in November 1963. In September 1966, the OGR was designated as a National Historical Landmark. Located in Building 3001, the OGR was an air-cooled, graphite moderated and reflected, heterogeneous, natural-uranium-fueled reactor. The moderator assembly is a 24-ft cube of graphite blocks, with spaces allowed for experimental access, thermocouples, and fuel slugs. The fuel channels extend through the block for fuel loading and unloading as well as providing for coolant air flow. The assembly is surrounded by a 7-ft thick reinforced concrete shield.

1994 Activities

- Routine S&M was conducted as scheduled. This included routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and ground maintenance have been included.

- An inspection of the 3018 stack that supports the OGR was completed. The inspection revealed that no water is present between the cell ventilation fans and the stack. This inspection was conducted to supply information on the source of legacy contamination for future remedial efforts at Building 3003.

- In October 1993, maintenance activities included removal of lead bricks on third floor; replacement of a plug in oil reservoir tank on west end of 3001; and removal of asbestos around the ladder on the fifth floor leading to the catwalk.

- In November, samples were taken from transformers and motors on the mezzanine to check for PCBs before removal.

3.5 LOW-INTENSITY TEST REACTOR

History and Background

In 1951, the Low-Intensity Test Reactor (LITR), in Building 3005, was converted from a hydraulic mockup of the materials testing reactor to an operating reactor for the purpose of supplying a variety of irradiation facilities for ORNL and other research groups. The LITR was a water-moderated- and cooled-reactor using enriched uranium as fuel and beryllium as a reflector. The reactor was designed for 500-KW power level and later converted to a 3-MW testing reactor prior to permanent shutdown in 1968.

The LITR tank is made up of five cylindrical steel and aluminum sections connected by gasketed flanges; it contains the reactor controls, coolant pipes, and the reactor internals. All but the lowest tank section is above ground. Not an integral building, the enclosure for the reactor is a composite of essentially independent rooms built on an as-required basis. The facility is primarily steel of a 70 x 62 x 57 ft. size.

1994 Activities

- Routine S&M was conducted as scheduled. This included routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and ground maintenance have been accomplished.

- In October 1993, a DOP test was completed on HEPA filters in LITR.

- In November, approximately 400 lbs of scrap metal were removed from the mid-riff level and top floor of 3005; broken boards on the wooden walkway on the mid-riff level were replaced.

- In January 1994, the flexible hose in the LITR mid-riff level was replaced with 3-in. galvanized tubing and the old filter was replaced with new filter housing for inlet air. These changes enhanced the off-gas system's capabilities through use of better construction materials.

3.6 HIGH LEVEL CHEMICAL DEVELOPMENT LABORATORY

History and Background

The High Level Chemical Development Laboratory (HLCDL) in Building 4507 was designed and operated as a laboratory and small-scale pilot plant for development studies of reactor fuel processing, separation and recovery of transuranic (TRU) materials, and separation of fission products from aqueous wastes. The building consists of four shielded hot cells equipped with master-slave manipulators and associated support equipment in the cell operating area. Chemical makeup and cell charging area above the cells contain a shielded manipulator cave, maintenance glovebox, and a 10-ton gantry crane for handling shielded casks. An underground tank pit south of the building contains two tanks formerly used for storage of radioactive solutions used in the facility operations. The facility is currently inactive, but has been maintained in good condition.

1994 Activities

- Routine S&M was conducted as scheduled. This included routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and ground maintenance have been accomplished.

- Other activities included completion of relamping of the facility which included operating areas and the vent house. Increased lighting enhanced accessibility during surveillance and maintenance tasks and addressed employee safety.

- A DOE/Martin Marietta Energy Systems inspection was performed to assess RP instrumentation. The team reviewed operational status, placement, calibrations, and proper use of equipment. All RP noncompliances identified have been corrected.

- A radiological survey was performed of ventilation pit 4556. As a result of this survey, ventilation balance/adjustment followed.

3.7 OLD HYDROFRACTURE FACILITY (BUILDING 7852)

History and Background

The Old Hydrofracture Facility (OHF) at Site 7852 in Melton Valley was an experimental and operational plant for the injection of waste grout into a fractured shale formation. The experimental design was tested in 1964–65 using dilute and concentrated waste solutions. Beginning in 1966, operational injections of concentrated liquid waste from the LLLW system were routinely made until facility shutdown in 1980. The facility was shutdown when the New Hydrofracture Facility, just south of this site, was constructed.

The facility consists primarily of 4 bulk storage tanks for cement and other solid constituents of the grout mix, waste and injection pumps, a waste/grout mixer, and assorted piping and other equipment. The wellhead, injection pumps, and mixer are enclosed in concrete cells.

1994 Activities

- Routine S&M was conducted as scheduled. Routine S&M includes routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and ground maintenance have been accomplished.

3.8 SHIELDED TRANSFER TANKS

History and Background

These shielded transfer tanks (STTs) were used for shipment of ^{137}Cs-loaded ion exchange resins from Richland, Washington, to ORNL for processing. The resins were removed from the tanks and the ^{137}Cs converted to usable forms at the Fission Product Development Laboratory (FPDL). The casks were reused several times over their service lifetime.

There are five STTs in the D&D Program (four STT Model No. II and one Model III). Model II tanks consist of a 500-gal, 3/8-in. thick stainless steel liner surrounded by a 3-1/2-in. lead shield, all encased in a 3/4-in. mild steel outer shell. The Model II tank consists of a 200-gal stainless steel liner encased in 9 in. of steel. Both types of tanks have provisions for lifting. Model II tanks are approximately 6 ft in diameter by 7 ft tall, with a loaded weight of 38,000 lbs. Model III tank is 8 ft tall and 4 ft in diameter. All tanks are located at the Decontamination Facility 7819.

1994 Activities

- Routine S&M was conducted at the STT facility. Routine S&M includes routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections.

- Surveys revealed no increase in surface contamination, as indicated by the quarterly radiological inspections. No areas of concern were observed.

3.9 FISSION PRODUCT PILOT PLANT

History and Background

The Fission Product Pilot Plant (FPPP) in Building 3515 was used in the fission product recovery development program for the separation of curie quantities of various radionuclides from low-level liquid waste (LLLW) waste streams. It was abandoned in 1958 when it was replaced by the FPDL. The facility consisted primarily of an unlined concrete-shielded cell, approximately 20 x 10 x 8 ft high, with an adjacent operating area. The process contained several small (few gallon capacity) stainless-steel vessels and columns, with associated piping, valving, and controls. The concrete-block and reinforced-concrete building is located on the east side of the South Tank Farm (site 3507).

1994 Activities

- Routine S&M was conducted as scheduled with no problems encountered. Routine S&M includes routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. General facility upkeep and grounds maintenance have been accomplished.

3.10 WASTE EVAPORATOR FACILITY

History and Background

The Waste Evaporator Facility received the LLLW waste stream from ORNL laboratories and other processing areas during the 1950s for concentration prior to final disposition by shale fracture techniques. This activity was suspended when the presently active evaporator facility (Building 2531) was brought on line. Subsequent installations of experimental equipment were used to develop fission-product purification processes and demonstrate contaminated waste incineration. The facility consists of a stainless steel-lined floor, reinforced concrete cell with underground piping, valve pit, and an attached wood-framed operating area. The building dimensions are 22 x 28 x 8 ft high. The evaporator facility is located on the west side of the South Tank Farm (site 3507).

1994 Activities

- Routine S&M was conducted as scheduled with no problems encountered. Routine S&M includes routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. General facility upkeep and ground maintenance were accomplished.

3.11 FISSION PRODUCT DEVELOPMENT LABORATORY INACTIVE CELLS

History and Background

The FPDL was originally designed and constructed to separate kilocurie quantities of ^{137}Cs, ^{90}Sr, ^{144}Ce, and ^{147}Pm from Redox- and Purex-type waste streams. It was modified in 1963 to allow production of megacurie amounts of ^{137}Cs, ^{90}Sr, and ^{144}Ce, primarily for use in the Atomic Energy Commission's SNAP Program. When this program ended in 1975, the facility was placed in standby, and initial decontamination efforts were undertaken. From then until 1989, a significant portion of the facility was reactivated for chemical separation and purification of fission products and manufacture/fabrication of radioactive sources. In 1989, the building was again placed in standby for use as interim storage of beta-gamma sources from other shutdown facilities.

The FPDL consists of 24 large-volume shielded hot cells with associated manipulator galleries and service areas. It is enclosed in a reinforced concrete, steel, and block structure approximately 125 ft long by 62 ft wide and 44 ft high. Associated tank farm cells are located adjacent to the building, extending about 14 ft below grade.

1994 Activities

- Routine S&M was conducted as scheduled, with no problems encountered. Routine S&M includes routine site inspections, completion of facility checksheets and logbooks, radiological and IH surveillance, and routine maintenance activities required as a result of site inspections. Maintenance of containment systems and equipment, general facility upkeep, and grounds maintenance have been accomplished.

3.12 METAL RECOVERY FACILITY

History and Background

The Metal Recovery Facility (MRF) was a pilot and small-scale production nuclear fuel reprocessing plant used for the processing of various waste solutions, scrap, and miscellaneous fuel elements for the recovery of uranium, plutonium, neptunium, and americium. Shutdown occurred in 1960 after some 25 different processing campaigns because of lack of secondary containment. The MRF is a one-story metal-sided building about 90 ft long by 70 ft wide by 24 ft high. It consists of seven concrete or concrete-block cells, makeup area, offices, storage area, and control room, with a below-grade concrete dissolver pit located inside and a fuel-handling canal located adjacent to the building, both with controlled access.

1994 Activities

- Routine S&M was performed, with no problems encountered. Substantial rainwater inleakage due to roof leaks continued to be a problem at the MRF. Floor drain dye testing was conducted to comply with an Environmental Compliance request. Several structural deficiencies in the floor drain system were noted.

3.13 MOLTEN SALT CORROSION LOOP

History and Background

This Molten Salt Corrosion Loop (MSCL) was used for preparation and handling of fuel for molten salt reactor technology development and the MSRE, which operated between 1965 and 1969. In 1974 and 1975, the facility was used to produce fuel salts for the MSBR development program. The facility consists of a large cell with separate subcells of irregular dimensions, but generally covering an area approximately 16 ft by 16 ft and 3 stories high. Constructed of metal, the cell has seven entrances on three levels. Within the cell are contaminated equipment, ductwork, friable insulation, and remnants of chemical processing equipment and material.

1994 Activities

- Routine S&M was performed as required, with no problems encountered.
- Relamping and clean up of old lamps has been performed.
- A monthly walk-through was performed for environmental, Occupational Safety and Health Administration (OSHA), IH, and HP compliance.

3.14 COOLANT SALT TECHNOLOGY FACILITY

History and Background

The Coolant Salt Technology Facility (CSTF) facility was used to support development of molten salt reactor technology, and specifically, for study of the effects of injecting tritium into a flow of boron trifluoride. The facility consists of a loop flow system surrounded by a steel enclosure approximately 24 x 8 x 10 ft tall. The system is comprised of pumps, tankage, valves, piping, and the remnants of a control system. The top of the steel enclosure is open to the outside, but enclosure panels are locked to prevent access. The facility is inactive, but is located in an active building within a central area of ORNL facilities at the Y-12 Plant.

1994 Activities

- Routine surveys were conducted of the outside of the facility. The CSTL is locked and posted, denoting radiation/contaminated areas.
- All utility/process lines have been labeled properly.
- A monthly walk-through was performed for environmental, OSHA, IH, and HP compliance.

3.15 DECONTAMINATION FACILITY (DF)

History and Background

This facility, located in Building 9419-1 at the Y-12 Plant, is an ORNL facility that was used to decontaminate equipment and materials associated with the development of the molten salt reactor technology. It consists of a roofed building with semi-deteriorated transite siding, located on a concrete pad. The structure is about 25 x 50 x 18 ft high.

1994 Activities

- Cleanup was performed including removal of accessible waste from the facility. The remaining items will be included when D&D is performed.
- An ER Maintenance Job Request for roof repair was submitted to maintenance personnel at the Y-12 Plant.
- The building was surveyed monthly to assure that it is secure and that no issues have arisen.

3.16 OIL STORAGE TANK

History and Background

The Oil Storage Tank is located on the mezzanine above Room 126 in active building 9201-3 in this ORNL facility at the Y-12 Plant. The tank was used as a reservoir for machine shop cutting oil during the development of molten salt reactor technology. The machine shop was located on the 1st floor just below this mezzanine. This facility consists of a glass-lined steel tank of 3000-gal capacity. It has a top opening currently covered by a trash can lid. The tank is empty with only a contaminated oily residue on the exterior. Beside the tank is a smaller, 100 gal, accumulation tank. The two tanks are surrounded by a spill-containment dike. There is an open drain directly under the large tank within the dike. The two tanks are located in a room about 20 x 40 ft, with the tanks occupying half this space.

1994 Activities

- Routine S&M was performed this year, as required, with no problems encountered.
- A monthly walk-through was being performed for environmental, OSHA, IH, and HP compliance.

4. SPECIAL MAINTENANCE PROJECTS

4.1 MOLTEN SALT REACTOR EXPERIMENT UF_6 MIGRATION REMEDIATION

Higher than normal radiation was detected in a remote section of the facility in 1988 when planning was underway for upgrades to enhance S&M of the facility. Annual annealing of the salt mixture was suspended after 1989 when annealing was suspected of contributing to the radiation increase. The radiation level has been monitored since that time while DOE performed an assessment of the facility to determine areas where the facility might be out of compliance. That assessment was completed in 1993. An evaluation was also initiated in 1993, and has continued in 1994, to thoroughly investigate the mechanism for migration of contamination associated with fuel stored in the MSRE.

4.1.1 WBS 1.0 Uranium Distribution

4.1.1.1 WBS 1.1 auxiliary charcoal bed

A full mapping of the ACB to determine the quantity and location of uranium will be performed. Portable gamma spectrometry has measured the ^{208}Tl through water and concrete shields which introduces some uncertainty in the measurement. This measurement is the basis for the uranium estimate of the deposit in the ACB. Gamma and neutron TLD technology has also been used on the ACB to establish the spatial distribution of the uranium and to confirm the quantitative uranium estimates from spectrometry. The gamma spectrometry measurements will be repeated after the water is removed from the ACB to establish a baseline for further periodic monitoring.

Progress Made

The amount of uranium deposited in the ACB has been estimated using a portable gamma spectrometer and gamma dose rate measurements. Raw data have been acquired and computer curve-fitted to estimate the quantity and length of deposit. First data indicated the deposit is in the range of 2 to 2.2 kg and deposited within the first 10 in. of charcoal. Later data indicates the deposit (approximately 2.6 kg) is located within the first 12 in. of the bed in the upper end of the concrete pit. The main concern associated with this deposit is a potential for criticality if the water were able to enter the ACB due to a failure in the welded ACB assembly.

Neutron dosimetry and spectrometry are also underway to quantify the amount of neutron activity. Bubble dosimetry has indicated a neutron dose rate of 2 mrem/h. The first two attempts at neutron scanning of the ACB have encountered difficulties due to water inleakage to the detector and effects from saturation of the neutron detector with high gamma activity. A third attempt was successful using a different sensor configuration and has indicated the neutron activity to be in the same region as the gamma activity. However, differentiation between spontaneous fission neutrons and alpha-n reaction neutrons has not been completed.

4.1.1.2 WBS 1.2 drain tanks

To the extent possible, the amount of uranium in the fuel drain tanks will be determined. Remote gamma scanning and other detection options will be employed.

Progress Made

An estimate of the amount of uranium remaining in the drain tanks is desirable to fully understand and map the distribution throughout the facility. An assessment is planned to reactivate the drain tank weighing system to estimate the amount of fuel salt remaining in the tanks. This activity is lower priority than other more pressing tasks and has not been initiated yet.

4.1.1.3 WBS 1.3 piping and vessels (Vent House, north electric service area, and balance of plant)

All gas phase piping and auxiliary vessels connected to the drain tanks will be identified. Of particular interest will be vessels with geometries that are not inherently criticality safe. Attempts will be made to locate and quantify any uranium deposits in piping and small vessels. An assessment of records from past annealing operations and valve alignments will be made to identify potential locations for uranium deposit.

Progress Made

As with the drain tanks, knowledge of uranium in other parts of the facility is important to fully mapping its distribution. A small quantity is known to exist in piping in the North Electric Service Area, but has not been quantified. It is also possible that some uranium may exist in the particle trap (beneath the floor of the Vent House) and the four main charcoal beds. Efforts to estimate the quantity in the particle trap will be challenged by several feet of concrete shielding and its inaccessibility. Some uranium could conceivably exist in the other four charcoal beds. It is expected, however, that any uranium present would be of smaller quantities since the beds are isolated from the drain tanks and only one bed has been used for one annual reheat since reactor shutdown. Further, the diameter of piping for the four main charcoal beds is as follows: first 80 ft is 0.5 in. pipe, the second 80 ft is 3 in. pipe, and the last 80 ft is made from 6 in. pipe. Therefore any uranium present would most probably reside in the first section of piping (0.5 in.) and would therefore exist in more favorable criticality geometry. Efforts to estimate quantities of uranium in these other areas are also lower in priority than other activities.

4.1.1.4 WBS 1.4 containment cells

Tests will be devised to determine if any uranium deposits exist in the contained and shielded cells due to gaseous transport through process pipes or historical residues from fuel processing operations. Currently there is no indication of gaseous residues from the process piping. The drain tank cell, fuel reprocessing cell, and reactor cell, as well as auxiliary enclosures, will be included in the assessment, to the extent feasible, by remote methods.

Progress Made

Tests to determine if residual uranium exists in the fuel processing cell have not yet been initiated.

4.1.2 WBS 2.0 Facility Safety Evaluation

4.1.2.1 WBS 2.1 nuclear criticality safety evaluation

This task is to perform nuclear criticality safety (NCS) evaluations on the ACB, the fuel drain tanks, and any other system where uranium deposits are discovered or could be accumulated. These evaluations will be summarized in NCS approvals and NCS evaluation forms. For certain as-found conditions (e.g., the water-filled charcoal bed cell) an NCS approval of that condition as an acceptable configuration may not be possible. This task will also perform criticality safety evaluations for tests, facility changes (such as removing water from the charcoal bed pit), and mitigating actions. Hypothetical criticality accident studies will also be conducted as appropriate and results factored into the accidental gas release assessment (WBS 2.3).

Progress Made

The discovery of uranium in the off-gas system, and especially in the ACB (Sect. 1.1), has invalidated significant portions of the previous NCS review and approval (NSR) that was completed in 1993. That review was based on the information available at the time, which assumed that all the uranium was in the drain tanks, and that at least two good physical barriers against possible hydrogenous moderation of the uranium existed throughout the system. In fact, upon the initial discovery of uranium in the off-gas system, the NCS analysis was still based on the lack of available moderation, so the additional new geometry appeared to have at least the safety of the old. However, upon the discovery of uranium in the ACB, which resides in a large pit filled with water, it was obvious that there was only one good physical barrier, the stainless steel piping of the ACB, against introduction of moderator into the off-gas system. Furthermore, criticality calculations show that, if water from the pit were to leak into the ACB and diffuse to the deposit, based on our somewhat uncertain knowledge of the actual size of the deposit and its axial extent, it would not be possible to guarantee that the system would remain subcritical. If a significant amount of water were to be able to make its way to one or both of the two fuel drain tanks, the system would almost assuredly become critical.

As additional data are acquired to refine our knowledge of the amounts of uranium deposited (Chap. 1), additional NCS evaluations will be performed as appropriate. After the distribution of uranium throughout the facility has been satisfactorily determined, additions to the existing NSR will be developed to document the analyses that have already been done and any new analyses required. Eventually, a new request for NCSA will be developed to replace the existing NSR and its additions.

Other activities at the facility may require future NCS evaluations such as the purging and trapping of UF_6 from the piping system. These activities will be carried out as they are identified.

4.1.2.2 WBS 2.2 corrosion evaluation

Since corrosive gases were measured in the gas sample from the fuel drain tanks vent piping, this task will conduct an evaluation of corrosion of piping, valves, traps, etc., that have been exposed to the reactive gases.

Progress Made

The task to assess the corrosion potential for various materials in the primary containment system has not been initiated yet.

4.1.2.3 WBS 2.3 accidental gas release risk assessment

This task will perform a risk assessment to human health and the environment in the unlikely event of an accidental gas release from the reactor piping systems. Results from this assessment will be used to support an update of the safety analysis and documentation (WBS 2.4).

Progress Made

A risk assessment is currently underway that evaluates an incredible worst case as well as credible cases for an accidental release. The assessment is still in progress and results are not yet available.

4.1.2.4 WBS 2.4 safety analysis and documentation

This task will generate safety analyses and documentation incorporating the current facility configuration, hazards, risk assessments, and controls for assuring safety. A hazards screening revision is underway for the current facility condition. Interim safety bases for surveillance, maintenance, and near-term uranium migration investigations and remedial actions will be developed and documented in a Basis of Interim Operation. A safety analysis report (SAR) will be generated for the MSRE facility based on the results of the tests and assessments. This task will also conduct safety evaluations for the tests or mitigating actions at the MSRE.

Progress Made

In light of the additional data acquired recently concerning migration of radioactive and reactive gases, the previous hazard screening completed as part of the Safety Analysis Update Program will be repeated. Subsequent to a new hazard screening, it is anticipated that a Basis for Interim Operations will be developed, to be followed by a SAR. A cost estimate and schedule for these activities have been developed; however, they cannot fully begin until the full extent of the facility hazards has been identified.

4.1.2.5 WBS 2.5 unreviewed safety question determination

This task will be to provide unreviewed safety question (USQ) determinations for the current conditions and proposed mitigating actions at the MSRE. An initial USQ for the MSRE has been determined because the presence of fissile uranium in the charcoal trap is outside of the previous safety basis for the facility.

Progress Made

The procedure for determining if an USQ exists has been completed with the determination that one does indeed exist. The previous facility authorization basis consisted of the hazard screening report (discussed in Sect. 2.4) and the NSR (discussed in Sect. 2.1). The determination of having an USQ is based on the fact that new information concerning migration of radioactive and reactive gases increases the probability and consequences of occurrence of an accident previously evaluated in the authorizing documents.

This determination was included a 10-day update to the original occurrence report. Additional reporting that may be required for an USQ will be made as appropriate.

4.1.3 WBS 3.0 Interim Corrective Measures

4.1.3.1 WBS 3.1 lower water level in the ACB pit

Concerns over water leaking into the ACB and eventually into the fuel drain tanks that could lead to a criticality accident will be eliminated if the ACB pit is drained of water and is maintained in a dry state. The water level in the ACB pit will be lowered in two stages. Thermal analysis and heat transfer calculations will be performed to assure that the heat now being generated by decay of radionuclides and by the reaction of UF_6 and F_2 with the charcoal will not result in a significant rise in ACB temperature that could be detrimental to the ACB vessels or lead to undesired conditions within the ACB. Structural analysis of the ACB has determined that no detrimental effects to the charcoal bed pit will occur if the water is drained from the pit. Questions concerning the effect of ground water causing the pit to "float" out of the ground and the structural integrity of the ACB support saddle have been resolved.

Progress Made

The level was lowered the first increment below the inlet piping on July 19. Preparations are underway to allow the water to be safety lowered the second increment below the area where the uranium deposit exists. The preparatory steps include (1) performing peer reviewed heat transfer calculations to predict possible temperature increases from chemical reaction and/or decay heat after removal of water; (2) procuring and fabricating a movable thermocouple that will monitor the outer wall temperature of the ACB during the lowering; and (3) developing a procedure for the lowering process.

The heat transfer calculations were completed predicting what temperatures may be expected without the cooling effect of the water. With the water at design capacity, a 9°F temperature gradient was measured from the water to the centerline of the charcoal at the top of the bed. A temperature rise is anticipated as the water is removed due to some combination of decay heat and heat of chemical reaction of F_2 and UF_6 with the charcoal. An estimated decay heat source of 2.5 W has been used based on available measurements, and a range of chemical reaction heat source values has been modeled since this value is not presently known. Also, three different heat transfer coefficients for the charcoal have been modeled, since the actual value and effects on the coefficient from reaction with halogens is also not known. Using these parameters, the calculations have yielded a range of upper temperatures that could be experienced ranging from 97°F to 265°F at the centerline of the charcoal after water is removed.

Because of the possible temperature increase that may be experienced and its effect on accelerating a chemical reaction, accurate temperature monitoring is essential during the water lowering process. Existing thermocouples only measure water temperature and the centerline of the charcoal near the top of the column. As water is lowered, distance is increased from the single thermocouple in the charcoal, and therefore, accuracy with which an increase could be detected is significantly diminished. Therefore, plans are underway to install a remotely manipulated contact thermocouple which will track down the outside wall of the ACB as the water is lowered. Both the thermocouple carrier and the thermocouple assembly are on order. Fabrication of the tool to contact the ACB wall and manipulate the whole assembly is in progress. A small video device is also planned to be installed to permit visual confirmation of proper operation and location of the temperature sensor. Based on current information, it is expected complete fabrication and checkout of this device could require 3 to 4 weeks. However, other means to expedite this process are being explored through making innovative use of in-house equipment. In addition, drilling of an additional access hole through the concrete shield plug not currently installed is being planned, which will simplify tooling and also facilitate other data acquisition activities.

A procedure has been drafted for the actual process of lowering the water. Due to the potential temperature increase, a slow reduction of water level is planned. The draft procedure outlines a process for lowering the water in 1-in. increments, with holding periods on the order of 30 min to allow the temperature to stabilize. This process will be used for the second phase of water lowering which will drop the water an additional 2 ft to a depth below the uranium deposit. Prior to actual implementation, this procedure will be finalized and reviewed to ensure adequate contingency planning is incorporated.

In the interim, the temperature profile of the water itself has been mapped down to bottom of the pit. These temperatures will provide a reference for temperatures to be measured during the water lowering process. The temperature of the water ranges from a high of 73°F at the top of the pit down to 61°F at the bottom.

4.1.3.2 WBS 3.2 partition the off-gas system

In order to prevent further transfer of uranium (as UF_6) to the ACB and to other potential locations in the off-gas system, the piping will be partitioned by closing valves in the off-gas system. The two fuel drain tanks and the flush drain tank can be isolated from the ACB by closing valves HCV-573, HCV-575, and HCV-577; and closure of valve HCV-533 will isolate the particle traps (located below the Vent House) and holdup volume tank HV-2 (located in the charcoal bed cell). The line number 561 to the ACB will also be blocked. Prior plans to repair and close valve HV-561 have been deferred because of its remote location and high radiation levels. Other techniques to seal off the line 561 will be explored.

Closure of these valves will be preceded by installation and activation of pressure indicators to provide continuous monitoring of tanks or piping that is isolated from current monitoring or off-gas systems.

Changes in cell ventilation may also be required to ensure that air operated valves do not result in the pressurization of sections of the confinement system. Configuration changes will be evaluated for safety, environmental, operational, and maintenance effects prior to implementation. The facility configuration management, quality assurance, and readiness review processes will support the implementation of these changes.

Progress Made

Since uranium is known to have been deposited over time through the migration of UF_6 to the ACB, high priority is also placed on isolating the ACB from the drain tanks. The most direct way to accomplish this isolation would be to close valve 561 on the inlet to the ACB. This valve is presently in the open position and is inoperable. The valve itself is located in a heavily shielded valve box below the floor of the Vent House, and was operated by an extension rod with a tee-handle above the floor. It is believed the extension rod is uncoupled from the valve stem rendering the valve inoperable. Engineering drawings of the valve box have been reviewed to determine an approach to accessing and effecting repair of the valve. The process would require removal of a portion of the Vent House roof to permit crane access for removal of three 7-in. thick sections of steel armor plate to get to the top the valve box. Then some additional disassembly of the box itself would be required. The radiological conditions inside the box are not known, but a significant radiation field is expected to be encountered. Subsequent repair of the valve operator would then be a shielded remote maintenance operation. Due to extensive work involved and time required to plan and carry out this process, a more immediate plan to partition the piping system has been developed as described below.

The alternate plan will involve isolating portions of the piping system with the objectives of isolating the supply of migrating gases (the drain tanks), and limiting the volume of gas open to and available for reaction in the ACB. The process as currently planned will isolate the drain tanks by closing valves 573, 575, and 577. This will isolate the source of additional gas from the ACB, isolate the largest volume component (1177 L) from the ACB, and preclude any possibility of water inleakage to the drain tanks. A second partition will be effected by the closure of valve 533 which will similarly isolate an additional 233 L from the ACB. This two-step isolation would then leave only 19 L of gas volume open to the ACB. However, before these isolation can be implemented, additional pressure instrumentation must be brought on-line to permit monitoring of pressures in isolated areas of the system. Specifically, the original pressure transducers must be reactivated on the three drain tanks. Instrument drawings are currently being reviewed in conjunction with activities to prepare valves 573, 575, and 577 for reactivation. A valve control panel is currently being designed from which the valves will be activated and indicators from the valve position switches will be mounted.

Similarly, an additional pressure transducer must be installed in the system to permit monitoring of pressure in the 19-L volume isolated by closure of valve 533. The most desirable location for this instrument is the sampler enricher station located in the high bay. This is another area that is currently shielded and will require careful radiological monitoring before and during this operation. A fluorine-compatible transducer is currently on order for this application.

Closure of valve 561 will continue to be pursued, but as a lower priority activity in favor of the plan described above. The timing for its closure will also be factored into plans for purging and trapping of gases in the system (Chap. 5.0).

4.1.3.3 WBS 3.3 eliminate water sources

A study of possible water sources and scenarios of how water could be accidentally introduced into the MSRE piping and/or vessels will be made. Appropriate steps will be taken to eliminate any sources of water that have a possibility of entering into the MSRE drain tank piping system. Cooling-tower and treated water systems were drained when the surveillance and maintenance program was first established. The water in the charcoal bed cell is addressed in WBS 3.1. The other primary sources of water are (1) reactor cell annulus, (2) nuclear instrument penetration, (3) vapor condensing tanks, and (4) fire water system (only exterior to cells). Each of these will be analyzed for potential pathways to fuel storage or uranium deposits. For water sources requiring removal, sampling of each water source will be completed to define appropriate disposal approaches. Initial assessments and sampling have indicated that the water in the vapor condensing tanks is at level that prevents it from being forced into the reactor cell. The volume of water in the nuclear instrument penetration is less than one-half of the volume required to cause an overflow into the drain tank cell. Sumps in the drain tank cell and reactor cell have been monitored since facility shutdown and have not shown accumulations. The results of these early assessments indicate that water intrusion from these sources is highly improbable. As a result, the removal of these sources of water will proceed at a lower priority consistent with the availability of environmentally compliant treatment or disposal options.

Progress Made

None reportable at the present time.

4.1.3.4 WBS 3.4 revise and implement S&M procedures

The S&M procedures will be reviewed and revised to reflect the current conditions of the MSRE. As physical conditions change during the course of remediation, S&M procedures will be modified accordingly. Safety documentation generated in WBS 2.1 and 2.3 will be a principal requirements basis for modified procedures. The implementation of the revised procedures, additional monitoring requirements, and facility management activities associated with this work plan are included in this task.

Progress Made

As facility conditions are assessed or changed through remediation actions, S&M activities will be revised as necessary. To date, a number of revisions have already been made. A portable criticality detection device has been borrowed from the Y-12 Plant and installed at the facility along with annunciator horns. It has also been linked via phone line with the ORNL WOCC which provides 24-h monitoring. In addition, general specifications and cost estimate for a permanent criticality detection system have been developed. Additional parameters are also being monitored in conjunction with ongoing tests and evaluations. Thermocouples have been identified and reactivated to measure temperature on the ACB.

After relocation of personnel from offices in the facility, a guard has been stationed continuously at the site to control access. Plans have been made to add additional fencing to fully encircle the site, and thereby eliminate the need for the guard's continuous presence. Completion of the fence installation is underway.

4.1.3.5 WBS 3.5 install criticality accident alarm system

The potential consequences of a criticality accident require that the MSRE be equipped with a criticality accident alarm system. A permanent criticality accident alarm system, based on a neutron system, will be installed.

Progress Made

As an interim measure, a portable criticality accident alarm, based on a gamma-ray detector system, was installed in MSRE on July 22, 1994.

4.1.4 WBS 4.0 Tests and Evaluations

Numerous tests and evaluations will be conducted to support WBS 1.0, WBS 2.0, WBS 3.0, and WBS 5.0. These tests and evaluations will provide technical data and engineering details to support other actions.

4.1.4.1 WBS 4.1 facility configuration

This task will compile drawings and other technical information in both tabular and drawing form. The result of this effort will provide a detailed description (pipe routing, holdup volume, materials of construction, confinement penetrations. etc.) of the primary and secondary confinement systems for the MSRE fuel and gas space. This information will provide a baseline to be used to evaluate corrosion of the system, areas for uranium accumulation, potential release points, and other safety issues.

Progress Made

The objectives of the subtask include ensuring that all potential water sources are identified and properly managed, and determining materials of construction in the piping system that are exposed to the reactive gas environment (pipe, valves, fittings, gages, etc.) and therefore have the potential for corrosion.

The particle traps in the off-gas piping system were designed to trap particulate matter during reactor operation. Due to high temperatures experienced in the traps during operation, they were contained in a water and jacket enclosure which also contained a water cooling coil. Water lines to both the jacket and the cooling coil have been identified in the Vent House and have been verified as disconnected from any water supply. A probe has been inserted to the bottom of the water jacket that indicated no water is present. The probe was also checked for radioactivity and none was found. There remains a possibility that residual water could still exist in the cooling coil. Near term plans have been developed to force any residual water out of the cooling coil. Six thermocouples located in the particle traps have been identified and reactivated. Current temperatures are in the ambient range (69 to 74°F).

A preliminary list of materials of construction of the piping system has been compiled. The list includes Hastelloy-N (drain tanks and primary piping), and several alloys of stainless steel throughout the off-gas system.

4.1.4.2 WBS 4.2 corrosion tests

WBS 2.2 will evaluate corrosion of the MSRE primary confinement system. As part of WBS 2.2, existing corrosion data will be compiled. Where there is no data existing for the condition under consideration, additional data will be developed under this task. This task will execute simple corrosion tests to evaluate corrosion rates and determine corrosion products. Data are needed on gas phase corrosion of stainless steels by UF_6 and F_2 and verification of MoF_6 as a corrosion product.

Progress Made

At this time, plans for these tests have not yet been developed.

4.1.4.3 WBS 4.3 UF_6 formation mechanism

The mechanism postulated for UF_6 formation at the temperatures in the drain tanks has not been previously documented. To prevent reoccurrence of the collection of reactive gases and to aid in mitigating the current situation, the mechanism of UF_6 generation must be understood. This task will conduct tests with fuel salt containing depleted uranium under the radiation condition in the MSRE to verify UF_6 generation and to determine rates of formation. A gamma irradiation facility in the High Flux Isotope Reactor spent fuel pool will be used to simulate the MSRE radiation. Earlier tests of the MSRE salt have shown that the primary mechanism for F_2 generation is via gamma radiation damage of the solid fluoride salt. Alpha radiation was not a significant contributor.

Progress Made

Tests are being conceptualized which will expose simulated salt with depleted uranium to radiation conditions similar to the MSRE. More detailed plans in this area have not yet been developed.

4.1.4.4 WBS 4.4 evaluation of annealing data

Annealing of fuel is a part of the MSRE S&M Program. During the MSRE program at ORNL, data were collected and reports prepared on the annealing process that recombines the free fluorine that may have formed from radiolysis of the fuel salt. This task will evaluate existing annealing data and develop additional data through testing, as needed, to provide a technical base for annealing to be conducted during future S&M.

Progress Made

This subtask has not yet been initiated.

4.1.4.5 WBS 4.5 sample drain tank cell atmosphere

The drain tank cell provides secondary confinement for the fuel in storage in the drain tanks. Any leaks in the drain tanks and associated piping must first be detectable in the drain tank cell atmosphere. This task would sample the atmosphere in the drain tank cell and provide analyses for UF_6, F_2, and HF that may indicate leakage in the primary confinement system.

Progress Made

Since the drain tank cell provides secondary containment for the stored fuel salt, information about gases present in the drain tank cell atmosphere would be of use. Methods to obtain and analyze a gas sample are currently under discussion.

4.1.4.6 WBS 4.6 ACB chemical reaction analysis

Removal of the uranium deposit from the ACB requires an understanding of the chemistry of the uranium, fluorine, and carbon compounds in the bed. Evaluation of methods for removal of the uranium will be completed. The mechanisms and potential for undesired conditions will be investigated. Remote sampling of the uranium deposit may be necessary to gain a sufficient understanding of the materials present.

Progress Made

This subtask has not yet been initiated.

4.1.5 WBS 5.0 Reactive Gas and Uranium Deposit Removal

Based on the information currently available, over 10% of the uranium originally stored as a solidified salt in the fuel drain tanks may have been converted to UF_6 and has migrated to other locations in the off-gas system. The gas phase samples taken to date also indicate some corrosion of the Hastelloy-N or stainless steel components may be occurring. Additional generation of F_2 due to radiolysis is expected and formation of more UF_6 is possible. It is expected that WBS 2.2 and WBS 4.2 will provide additional data to evaluate short-term corrosion concerns. However, because of the mobility of UF_6 and other existing information, it can be reasonably concluded that the MSRE primary confinement system is not adequate for long-term storage of the fuel salt.

Tasks under this WBS will remove the reactive gases, remove deposits of uranium as necessary, convert the recovered UF_6 to oxide, and package the oxide for long-term storage.

4.1.5.1 WBS 5.1 purge and trap

This task will include design, fabrication, installation, and operation of a purge and trap system to remove the reactive gases from the MSRE off-gas system. A shielded chemical trapping system with criticality-safe geometry will be designed to allow purged gases to be collected. The trapping system will include sodium fluoride (NaF) columns to collect UF_6 and soda lime columns to collect F_2 as calcium fluoride (CaF). The system will be capable of evacuating process lines to promote sublimation and transport of UF_6 deposits. The system will be designed to facilitate transport of the NaF loaded with UF_6 to other facilities for conversion to oxide. Soda lime columns will be designed to facilitate disposal as radioactive waste.

Progress Made

An important area of activity involves the removal of UF_6 and reactive gases (F_2 and HF) from the off-gas piping system. The task to design a system for removal by trapping on sodium fluoride and soda-lime has just recently been initiated. No reportable progress has yet been made; however, the task will entail the following sub-elements:

4.1.5.2 WBS 5.2 removal of uranium deposits

No reportable progress.

4.1.5.3 WBS 5.3 UF_6 to UO_3 conversion

No reportable progress.

4.1.5.4 WBS 5.4 uranium transport and storage

No reportable progress.

4.1.5.5 WBS 5.5 safeguards management

No reportable progress.

4.1.5.6 WBS 5.6 waste disposal

No reportable progress.

4.1.6 WBS 6.0 Long-Term Fuel Disposition

The task to explore long-term fuel disposition options will evaluate realistic alternatives to continued storage-in-place at the MSRE. Information acquired during the course of the present remediation task will have bearing on the feasibility and desirability of continuing on-site storage. If the present approach (with or without modifications) is deemed unacceptable, other alternatives will be evaluated with cost estimates and schedules developed. This task is important, but is secondary to all other areas of work focused on mitigation of the current situation, and as such, has not yet been initiated.

During this task, technical and regulatory plans for long-term storage and disposition will be developed. A project scoping document, with resource and schedule requirements, will be developed as the basis for multi-year funding. The end condition of this plan will be an MSRE facility in a safe condition for either long-term surveillance and maintenance or initiation of final decontamination and decommissioning. The fuel salt will have been removed and stored in a stable configuration or processed for ultimate disposition.

Progress Made

This subtask has not yet been initiated.

4.2 STATUS OF OTHER SPECIAL MAINTENANCE PROJECTS

The following special maintenance projects projected for FY 1994 experienced varying degrees of completeness because of the diversion of manpower required to address the MSRE remediation problem.

4.2.1 OGR Hydraulic Oil Removal

This work consists of the collection, sampling, and disposal of approximately 100 gal of potentially PCB-contaminated oil. The oil was used as hydraulic fluid to power horizontal rod drives that insert poison into the OGR core when necessary to shutdown the reactor. This oil is currently stored in metal reservoirs setting on concrete pedestals behind an access controlled fence at Building 3001.

Progress Made

- A project review checklist for NEPA determination was completed and submitted to the ORNL NEPA Coordination office. Current status of the determination is on hold pending resolution of raised questions.

4.2.2 HRE Storage Pool Segregation

This project consists of the sorting of items remaining in the HRE storage pool according to radiation level and material type. Material type determination may require sampling/analysis if a positive identification cannot be made by visual examination by past operational personnel. Storage options will be determined at a later time.

Progress Made

This subtask has not yet been initiated.

4.2.3 High Level Chemical Development Laboratory Containment Upgrade

This task includes upgrading the containment for the hot cells in Building 4507, specifically, by placing permanent covers over approximately 40 gloveports in plexiglass enclosures. The project involves potential breaching of the containment boundary to permanently fasten the connections.

Progress Made

- This subtask has not yet been initiated.

4.2.4 ORR Instrument Upgrade

This task involves the relocation of selected critical parameters from the ORR control room to an area specified on the second level close to the personnel entrance. Other non-essential parameters are to be deleted and the remainder of the control room inactivated as a surveillance control point. A control panel is to be added, providing clear identification of the instruments, and the parameters monitored (as operator aids) for routine surveillance and for emergency response.

Progress Made

- This project has all documentation completed from FY 1993 and is awaiting the GPP for LLLW Upgrade with the ORR (Feed & Bleed) to be completed. The GPP for LLLW upgrade includes fabrication of alarm panel for instrument upgrade in the control room. This must been done before the alarm lines can be connected to the WOCC.

4.2.5 OGR Containment Fans

This task involves the removal of aging ventilation fans, motors, and adaptive ductwork at Building 3003, replacing them with fans and motors of identical or similar specifications, according to specified functional requirements. In addition, some ductwork that attaches to the fans/motors may require replacement to provide acceptable hookup. Removed motors/fans and affected ductwork will require disposal following the new installation and will, most likely, be contaminated.

Progress Made

- The following tasks in support of the project included revised estimates received from Engineering for in-house versus subcontract work, revised draft document for task, Davis-Bacon determination made—subcontractor required, and Project Waste Management Plan initiated (in draft form, currently).

4.2.6 HRE Cells Investigation

This task includes investigation of all applicable building drawings for all underground cells at HRE; a possible dye test of several process cells associated with the HRE Steam Turbine used during reactor operation including three Turbine Pit sumps, repair of a manometer that gives indication of water in the Reactor Cell; assuring that all inputs to the Building 7562 12,000 gal LLLW Tank have been physically disconnected; and revising surveillances so as to check status of these underground cells on a routine basis.

Progress Made

This subtask has not yet been initiated.

4.2.7 Decontamination Facility Cleanup

This task includes surveying the facility for hazardous environments and radiological material; segregating surplus items within Building 9419-1 and disposing according to waste form, contamination status, and regulatory guidance; cleaning the floor, as appropriate, to remove residuals left behind from the larger items; repairing roof leaks; and removing damaged/deteriorated pipe insulation and packaging it for disposal. The South entrance door contains a lead weather strip, to be removed and replaced with a more suitable material to serve the same purpose.

Progress Made

- A subsequent meeting was held to determine resources to be used to complete the cleanup. A decision was made that ORNL would dispose of all wastes.

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