Savannah River Technology Center

Monthly Report

October 1992

MASTER

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CONTROL ROOM OPERATOR DOCUMENTATION

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Executive Summary

Reactor

- To provide analytical support for vertical tube storage basin cleanups, the Analytical Development Section (ADS) installed a particle size analyzer for radioactive samples. (page 6)
- SRS conducted a study to determine alternative modes of K-Reactor operation to provide a means for assuring the retention of SRS nuclear competence. (page 10)
- Data analysis of simulated test conditions in a MK22 assembly during the emergency cooling system (ECS) phase of a loss-of-pumping accident indicates that thermal excursions occur at lower powers than expected. (page 12)
- The changes in the potential for component corrosion in the cooling water system and the ECS, due to the cooling tower water system tie-in, are being evaluated. (page 13)
- Reactor Engineering requested that Equipment and Materials Technology perform a failure analysis on the rupture disks in the supplementary safety system/pump suction injection system. (page 14)
- Risk perspectives were performed for online and offline 484-D powerhouse operations. WSRCs results, corroborated by Los Alamos National Laboratory (LANL), show that safe operation of K Reactor has little sensitivity to the status of the 484-D powerhouse. (page 15)

Tritium

- Safety and Risk Analysis Management and Hydrogen Technology Section (HTS) are developing a tritium shipping container that uses uranium to transport tritium. The temperature profile of a prototype was measured to confirm that the design meets operational requirements. (page 16)
- The Replacement Tritium Facility (RTF) in-bed accountability technical support was supplied for Operational Readiness Review reports, mass flow controller testing, and definition of maximum accountable tritium concentrations on the Thermal Cycling Absorption Process deuterium beds. (page 16)
- The Hydrogen Technology Section (HTS) is tracking RTF requests for technical assistance. Benefits to RTF and HTS personnel include definition of the problem, scope, timing, resources required, and less duplication of efforts for problems previously addressed. (page 17)
- HTS and ADS personnel will review the RTF Transfer Lines' pre-operational process hazards. Several issues surfaced during the review of the transfer lines process drawings. (page 17)
- The Tritium Department used resistance upset welding to load twelve Materials Technology Section fabricated reservoirs. This accomplishment is a major milestone in the program to develop resistance welding as an improved method for res-

ervoir fabrication. To demonstrate the integrity of the upset girth welds for tritium service, long-term storage will follow. (page 18)

- A zirconium-iron-tin alloy (St198 from the SAES Getters/U.S.A.) is being evaluated for use in a non-oxidative tritium stripper system and was characterized by a variety of methods. St198 pellets consist of compacted granules of alloy containing Zr_2Fe primary phase and four secondary phases ($ZrFe_2$, Zr_5FeSn , α -zirconium, and η - $Zr_4Fe_2O_{0.6}$). (page 19)
- Activity Transfer Group (ATG) meetings for Gas Transfer Systems and Reservoir Surveillance Operations were held at the DOE (Albuquerque, New Mexico) on September 24 and 25, 1992. A draft of the Activity Transfer Plan for each activity was discussed. (page 20)
- Equipment and Materials Technology completed the functional and calibration tests for the Container Management Facility's post load leak test system. The system will be turned over to the Tritium Department, pending final review of system operating procedures. (page 20)
- The Non-Nuclear Consolidation Activity Transfer Plan (ATG No. 4) was revised after DOE and EG&G Mound Laboratories personnel resolved questions and concerns. The plan will be submitted to DOE (Albuquerque Operations), DOE (Savannah River Operations), and EG&G Mound Laboratories for approval and additional revisions by October 29, 1992. (page 21)

Separations

- The HB-Line Justification for Continued Operation was submitted for final WSRC approval on October 19, 1992. (page 23)
- The final version of the 5320 Safety Appraisal Report for Packaging was submitted to DOE for transmission to the Packaging Certification Office. (page 24)
- A specially-designed device was used to relocate a leaking EBR-II fuel storage canister in the storage basin. An encapsulating device was also delivered to the customer to overpack the canister for safe shipment to the dissolver. (page 24)
- DP-42 requested that SRTC evaluate the proposed technical developments in the Technical Assessment and Development Plans prepared by Los Alamos National Laboratory and Lawrence Livermore National Laboratory (LLNL), the lead Laboratories for Complex 21. (page 25)
- A computer-generated glovebox glove-change tracking system was developed and implemented in SRTC. It is patterned after a glove change system developed at Rocky Flats Plant and described at the 1992 American Glovebox Society Conference. (page 27)

Environmental

- Drilling of 18 closely spaced boreholes over the Pen Branch Fault for the Confirmatory Drilling Program was completed and cross-sections are under construction. (page 27)
- The dose received from a hypothetical atmospheric tritium release to people living near a DOE tritium processing facility (SRS or EG&G Mound Laboratories) was modeled. The dose to the maximum individual is higher if the release occurs at SRS, due to a shorter effluent stack. However, because of the proximity and size of

the population base near EG&G Mound Laboratories, the cancer risk is almost fourfold greater. (page 27)

- Screening level analyses for total residual chlorine, ammonia, and pH were performed on SRS sanitary outfalls. The results of the analyses were similar to earlier results and indicate that several sanitary plants discharge high concentrations of chlorine and/or ammonia. (page 27)
- The Radiological Assessment Program document on cesium in the SRS Environment (WSRC-RP-92-250) was issued. (page 27)
- Preliminary site scoring of the seven candidate sites for the new sanitary landfill was completed. Four sites were identified as candidates for subsurface characterization. (page 28)
- SRTC provided rapid support for pH measurements in the 484-D powerhouse large sump. Equipment was operating less than two days after the request was made. Additional equipment was ordered to provide continuous readouts, temperature-compensated measurements, and strip-chart recordings of pH. (page 29)

Waste Management

- Small scale tests demonstrated the ability of electron beam irradiation to destroy benzene in SRTC lab waste in the presence of nitrate and detergents. This method could give SRTC the ability to handle radioactive waste generated from in-tank precipitation (ITP) sample analysis operations. (page 33)
- Filter performance calculations show the smaller-pore size and thicker ITP filter walls will initially cause a 26% lower permeate flux, with 0.5 wt% slurry, compared to a standard 0.5 micron filter. The initial clean water flux rate should decrease by 61%. (page 36)
- The Late Washing-Nitric Acid flowsheet was demonstrated in the Integrated DWPF Melter System. The peak hydrogen generation rate decreased by 57%, compared to the design basis from the original Hydroxylamine Nitrate-Formic Acid flowsheet. The mercury concentration in the sludge was also reduced below the design basis of 0.45 wt%. (page 36)
- A bounding estimate of DWPF mercury emissions indicates high purge rates for H₂ dilution in the Chemical Process Cell. It also indicates that peaks in Formic Acid Vent Condenser temperature (due to oxidation of NO_x) will not cause emissions to exceed permitted quantities. (page 37)
- A development plan for a high-level liquid waste dynamic computer model was initiated and a graded implementation approach is being examined. (page 41)
- Proposed flowsheet modifications in the Defense Waste Processing Facility (DWPF) will subject alloy C-276 and Type 316L stainless steel to halide and mercury-containing solutions, which will be acidified with nitric acid. Electrochemical corrosion tests and a review of the corrosion database indicate that both alloys will perform satisfactorily under the new conditions. (page 43)

General

- Equipment and Materials Technology shipped the modular glovebox airlock docking port to LLNL for testing and demonstration. Drafting on a custom version is being finalized and the third concept is fully assembled. (page 48)

- The robotics team, funded by the Office of Technology Development, developed a FY93 program plan, "Waste Processing Operations", in which SRTC plays a prominent role. (page 48)
- "J-integral of Circumferential Crack in Large Diameter Pipes", is based on the results of finite element work. DOE is reviewing it for presentation at the Second International Conference on Nuclear Engineering in 1993. The report, which includes a crack stability analysis, is being prepared for SRTC. (page 48)
- A connection to the worldwide Internet was installed and limited access to the connection from the site network will be available in the near future. (page 48)
- The Technical Safety Appraisal that DOE has been conducting in SRTC and the Analytical Laboratories since October 20, 1992 was completed on October 29, 1992. Overall, SRTC and the Analytical Laboratories received a "good" rating on the appraisal. (page 49)

Progress and Accomplishments

Mark 53 JASON Library Documentation

K. A. Niemer

A technical report for the Mark 53 JASON library was written and submitted for review. The report, which is a DOE required deliverable of the closeout of the Pu-238 program, documents background, design, development, and user instructions for the Mark 53 JASON library.

The new Mark 53 JASON library is necessary for calculation of core reactivity and Pu-238 production in the mixed-lattice Mark 22/53 charge. The library contains information on:

- assembly names
- independent variables for correlation of data
- cell macroscopic cross sections vs. independent variables
- cell isotopic contents vs. independent variables, cell fission powers vs. independent variables
- default assembly staging instructions, editing instructions for isotopic contents
- data for converting fission power to computer power
- lithium-6 to total cell absorption ratios for tritium production calculations

New FaceMap Features

J. C. Roberts

FaceMap is a C program based on several software standards (e.g. XWindows) that allow it to run on diverse platforms, such as Macintosh, Sun, and IBM. The program displays "live" and historical data obtained from the K- Reactor core and outputs from reactor physics codes. The

primary display consists of a hexagonal grid, in which different colors represent variations in measured parameters; a second display shows color mapping to values.

Two features to aid in viewing historical data were added to the program. The first feature is a set of "VCR buttons" that allow users to pause the display or step forward or backward through the frames. Users can also locate individual frames by entering a date and time. The second feature allows users to print frames with the option of adding a centered label above the display. Both features proved extremely useful in meeting the Computing Technology Group's requests for a large number of FaceMap frames.

Plasma-Sprayed Heater Test Program

C. A. Baptiste and D. A. Coutts

The Plasma-Sprayed (PS) heater concept, developed in Nuclear Engineering Section (NES), can be used to make a more robust and less costly test rig than the offsite supplier's SPRIHTE wound-wire heater test rig. Production time of the PS heaters should be reduced and an azimuthal power profile can be produced. NES tested the heater and mechanical design features of a proposed PS heater Test Rig.

Installation of 28 thermocouples in the base aluminum tube was completed. Functional checks were performed and the thermocouples indicate normal responses. The tubes are being prepared for shipment to the thermal spraying facility.

Tank-Muff-Pump Tests—Update

J. S. Bollinger

Integral testing at the 1/4 scale separate effects test tank-muff-pump facility progressed to the point where additional single-phase pump testing can begin. A test matrix that includes additional single-phase tests was completed and transmitted to the Idaho National Engineering Laboratory. The purpose of the new test matrix is to fill in the single-phase homologous pump curves more completely for normal and dissipative pump operations. Tests also:

- examine the sensitivity of pump behavior to thermophysical property changes
- characterize the zero head and torque lines for quadrant one of the homologous curves
- determine the forward flow hydraulic resistance of the pump with the impeller locked
- determine the forward flow hydraulic resistance of the pump with the impeller free to rotate, but with the variable speed drive motor de-energized

A detailed description of the objectives of each set of tests will be provided in the final experimental operating specifications technical report.

TRAC and RELAP5 Conversion Update

D. P. Griggs, D. T. Herman, G. L. Gill, D. A. Burge, and P. K. Paul

The Nuclear Engineering Section (NES) completed the draft technical memorandum on the comparison of TRAC and RELAP5 loss-of-coolant accident-flow instability (LOCA-FI) analyses. NES also started work on the plenum pressure uncertainty analysis using RELAP5 benchmark results and corresponding plenum pressure data from the 1989 L-Reactor Test series, AC4M. This task is a duplication of the technique used for TRAC plenum pressure uncertainty described in WSRC-TR-90-263. Seven benchmark cases were used:

- one case with all AC pump motors operating normally
- three cases with five AC pump motors operating and one loop in backflow
- three cases with five AC pump motors operating and one loop with no flow (rotovalves closed)

For each case, the plenum pressures for assembly positions corresponding to K-14.1 flowzones one through four are compared with interpolations of the RELAP5 plenum pressures. Twenty-five data points are available for each case. The differences between the interpolated RELAP5 plenum pressures and the measured plenum pressures are calculated and the mean and standard deviation of this distribution of differences is calculated. The results of this analysis show a sample mean of +1.4 psi and a standard deviation of ± 2.0 psi. On the average, the interpolated RELAP5 plenum pressures are high by 1.4 psi and approximately 68% fall within -0.6 psi and +3.4 psi of the data. The positive mean of the sample indicates a "bias" in the interpolated RELAP5 results. Since the previous analysis with TRAC benchmarks against 1985 L-Reactor AC Flow Test data showed a negligible "bias" (approximately 0.1 psi), the uncertainty methodology for LOCA-FI limits treated plenum pressure uncertainty as a distribution with no bias. The appearance of a bias in the RELAP5 benchmark results suggest that an additional factor must be derived for the uncertainty analysis to account for the bias. Alternatively, additional optimization of the RELAP5 L-Reactor model could reduce or eliminate the bias.

Particle Size Analysis Support of Vertical Tube Storage Cleanup

S. F. McDaniel and L. L. Tovo

The Analytical Development Section (ADS) has the capability of performing particle size analysis on radioactive samples. The particle size instrumentation was placed in radioactive containment in response to a request from the Equipment Engineering Section for analytical support of the K-Reactor and P-Reactor Vertical Tube Storage cleanup. Particle size information about basin material is needed to determine a suitable filtering method and to verify that the method is working correctly. Approximately 40

samples are expected for the initial survey followed by periodic samples as the cleanup progresses.

The ADS particle size analyzer can measure particle sizes between 0.7 and 700 microns in powder or slurry samples. A diode laser sends light through a stream of particles to be analyzed. A photodiode array detector measures the amount and direction of light scattered from the particles. Data conversion into a particle size distribution is based on the diffraction theory.

Acceptance Criteria for Process Water System Expansion Joints

W. L. Daugherty, R. F. Miller, N. K. Gupta, and R. L. Sindelar

Acceptance criteria were developed for the Process Water System expansion joints. The criteria define acceptable levels of degradation of the expansion joint bellows (flow sleeve or tie rods) consistent with specified margins of safety. This effort demonstrates that some measure of degradation can be tolerated, provided that the recommended changes to the inservice inspection requirements are implemented.

The expansion joint bellows is subject to degradation through cracking or denting. In the case of dents, the resulting stress concentrations can lead to a reduced fatigue life. Possible degradation of the internal flow sleeve includes crevice corrosion, cracking, and distortion due to mechanical contact. Limits on degradation in each of these areas are defined. The acceptance criteria for the tie rods are based on American Society for the Mechanical Engineers Boiler and Pressure Vessel Code requirements.

Efforts are in progress to document these results and issue the final task report.

Acceptance Criteria for Process Water System Valve Casings

R. F. Miller

Three representative Process Water System valves were chosen and finite element models were constructed. Using the ABAQUS finite ele-

ment code, normal operating pressure loads and seismic loads were applied to the models. The maximum principle stress was determined for each case and conservative fracture assessment, based on linear elastic fracture mechanics that result in allowable flaw lengths in the valve casing is complete. The results of this analysis will lead to development of acceptance criteria for indications that may be found during in-service inspection. A final report summarizing the results of this analysis is being written.

Acceptance Criteria for Heat Exchanger and Head Shell

P. S. Lam and R. L. Sindelar

The final report, "Heat Exchanger Head and Shell Acceptance Criteria," (WSRC-TR-92-425) was reviewed by Materials Technology Section consultant, J. A. Begley of Packer Engineering, Inc. Non-mandatory recommendations to evaluate the degree of accuracy of the curvature correction approximating procedure for the plastic part of the J-integrals were made. Additional internal technical review is in progress.

Reference Guide to Reactor Process Water System Component Structural Integrity Documentation

G. S. Bumgarner

"Reference Guide to Reactor Process Water System (PWS) Component Structural Integrity Documentation" (WSRC-RP-92-1068) is complete. The report provides a single-source reference guide to other reports, technical memoranda, and questions, comments, and responses from reviewers addressing the structural integrity of SRS reactor PWS since 1984. The reference documentation serves as a resource for engineering experimentation and analytical activities.

Documentation on structural integrity and service life analyses is categorized by the following PWS components:

- main circulation and septifoil piping
- reactor tank
- expansion joints
- heat exchangers
- plenum inlet nozzles
- pumps
- valves
- flanges

The structural integrity of the pressure boundary is demonstrated through structural and fracture analyses, degradation and degradation rate evaluations, failure frequency evaluations, leak detection capability, and leak-before-break analyses.

Transite Wall Tests

T. J. Miller and R. C. Treanor

The Systems Structural Analysis group requested that Equipment and Materials Technology conduct tests on the structural integrity of the transite walls in the reactor areas. These walls primarily exist along the corridors and partitions in the Central Control Room of K Reactor and support safety related equipment, such as electrical panels, cabinets' and conduits. It is essential to maintain the structural integrity of the walls during a seismic event to ensure the functionality of the attached equipment. The objective is to establish the structural capacity of various wall features. This data will be used to perform seismic evaluations of equipment mounted on the walls.

The required tests were successfully completed in P Area and a test report is being written for the Systems Structural Analysis Group.

MELCOR/SR Verification and Validation Program from January through September 1992

S. Chow and T. E. Britt

MELCOR/SR is a computer code based on the Nuclear Regulatory Commission's Light Water Reactor severe accident code MELCOR 1.8.1, which was adapted to the SRS reactors by Science Applications International Corporation under WSRC sponsorship. Verification and Vali-

dation (V&V) will take place in accordance to, "V&V Plan for Safety and PRA Codes" (WSRC-RP-90-433). WSRC-RP-92-1097 describes the status of the MELCOR/SR V&V program from January to September 1992. Several milestones were completed during the reporting period. They include approval of the MELCOR V&V Task Plan, the MELCOR Integral Calculation Task Plan, the Software Development Plan, delivery and documentation of the code, and installation on the SRS Cray and one workstation. Several models are also being assessed, they include the fuel melting model, the material properties package, the high-efficiency particulate air filter model, and the carbon bed radiolytic and thermal desorption models. The results described in the status report are preliminary in nature because assessments have not undergone final reviews.

Reactor Accident Program Verification and Modifications

A. A. Simpkins, R. P. Taylor, and D. P. Kearnaghan

Verification of the Source Term Setup Module of the Reactor Accident Program was completed. The verification report is under review and expected to be issued in early November 1992. Efforts are in progress to improve the capability of the Reactor Accident Program to handle accidents that go beyond those covered in the Safety Analysis Report. Conditional probability-weighted source terms for each internal event plant-damage state, identified by the level 2 Probabilistic Risk Assessment analysis, which are being calculated. Logic to select which source term to use during a release of radioactive material is also being developed. It will be based on observable plant parameters available to control room operators.

Level 1, Revision 1 Internal Events Probabilistic Risk Assessment Quantification

S. P. Tinnes, D. S. Cramer, H. A. Ford, M. D. Brandyberry, S. V. Topp, and R. P. Taylor

The K-Reactor restart configuration Level 1 Internal Events Probabilistic Risk Assessment

was requantified based on recent data revisions and modeling adjustments. It is specific to K Reactor and its physical state in June 1992 (operating at 30% power). The PRA includes the complete spectrum of potential "internal initiators". Primary and secondary loss-of-coolant accidents (LOCA) for different size leaks and leak locations, loss-of-heat-sink (LOHS) initiators, loss-of-river-water (LORW) initiators, and transients are included. The total severe core melt frequency point estimate is $1.5E-5$ per reactor-year. More than 300 different sequences from different initiating event classes contribute to the results. Primary and secondary LOCAs account for 64% of the core damage, with the remaining core damage split between transients. It is anticipated that an additional iteration will be required after the databases are finalized.

Verification and Validation of SARIS Supplementary Safety System Model in Savannah River Simulator

M. V. Gregory and H. W. Sundal

The subcontractor performing the Verification and Validation (V&V) of models used in the SARIS delivered the third (analysis of the supplementary safety system (SSS) model) of four report packages. The SSS model simulates the effect of gadolinium nitrate (ink) injection into the reactor Process Water System (PWS), which shuts down the reactor. Several minor discrepancies that have no immediate impact on the training value of the model and one discrepancy with an observable effect in the control room were identified.

The most significant discrepancy was the inability to correctly model the effect of the newly added pump suction injection (PSI) configuration, thus the V&V report serves mainly to quantify the discrepancy. Without accounting for the PSI in SARIS, the simulator predicts a sinusoidal response in reactor power after ink injection. Initially, the power drops as the ink is injected into the moderator space, rises as the ink is swept out of the reactor tank into the external loops, and drops again as the ink reenters the core on the next transit of the slug of fluid. The sinusoid damps out as the ink disperses in the process water on successive transits. Fortunately, this was the correct response before the PSI hardware

modification was made for the K-14 charge. With PSI injection, no sinusoidal effect exists due to ink being swept out of the core. The actual K-14 test data showed a monotonically decreasing reactor power after ink injection. Consequently, the SARIS model must be updated to show the new response caused by the modified hardware.

After incorporating comments, the SSS V&V report will be forwarded to SARIS custodians in Reactors Training and Procedures for disposition.

Development of Power-Dependent MARCO Libraries

G. R. Cefus

Two power-dependent data libraries are being created for the MARCO computer code, which calculates the margin of control (MOC) and the initial critical control rod position for each reactor charge. The libraries supplement the power-dependent libraries being developed for the JASON code.

For various reactor assemblies, MARCO obtains hot, dirty fewgroup parameters (corresponding to the "at power" condition) from a companion JASON calculation and modifies them to generate cold, clean fewgroup data (corresponding to the "initial critical" condition).

$$\Delta \text{MARCO} = \sum \text{JASON} + \Delta \sum \text{MARCO Library}$$

The $\Delta \sum$ from the MARCO library consists of three components that account for the changes in reactivity due to the decay of ^{135}Xe and temperature and the build-in of ^{149}Sm and ^{239}Pu .

Cross sections that reflect the reactivity change when transitioning from the "at power" condition to the "initial critical" condition are generated with the GLASS computer code. Initially, a GLASS depletion calculation is executed to provide exposure-dependent, hot, dirty cross sections. This information is used in executing three types of static GLASS calculations:

- hot ("at power" moderator and coolant temperatures) with no ^{135}Xe
- cold (moderator and coolant temperature at 20°C) with no ^{135}Xe
- cold with no ^{135}Xe , ^{149}Sm concentration = $^{149}\text{Sm} + ^{149}\text{Pm}$ concentrations and ^{239}Pu concentration = $^{239}\text{U} + ^{239}\text{Np} + ^{239}\text{Pu}$ concentrations at shutdown.

Information from the GLASS calculations are concatenated using the CHIP3 computer code. The MARCOLIB computer code is executed to correlate the data into the MARCO library form.

Study of Extended K-Reactor Operation

L. R. Chandler

DOE plans call for placing elements of K Reactor, M Area (reactor fuel and target fabrication), and F- and H Areas (Separations) in warm standby condition (beginning in mid-1993 for K and M Areas and 1998 for F and H Areas). DOE directed that nuclear competence be maintained at SRS, such that the areas can be restarted within five years from the time of a decision for renewed production.

WSRCs concern arose over the ability to maintain nuclear competence under the above stated conditions. P. Parks, of the Planning Support and Analysis Department, and L. Chandler, of the Applied Physics Group, coordinated a study to address this concern. Technical experts from various groups contributed to the study, "Extended K-Reactor Operation (U)" (WSRC-TR-92-381). A classified companion study, "Productivity Information for Extended K-Reactor Operation (U)," (WSRC-TR-92-382), provides tritium production information.

Four scenarios of continued K-Reactor operation are discussed in the main study. They feature:

- rapid production of tritium in the immediate future
- slow production of tritium to minimize the rate of usage of the existing reactor fuel
- resumption of the more rapid production at a later date

These continued operation scenarios are compared with two other scenarios that are included

in DOE plans. The study was made for Mark 22, Mark 16B, and Mark 62 charges with tritium as the main product of each charge. For each operating scenario a detailed K-Reactor operation schedule is provided.

The following topics were addressed in the study:

- total costs (capital and operating) and manpower needed to operate the necessary SRS facilities
- reactor fuel and component availability
- tritium extraction facility impacts
- reactor area impacts
- SRTC impacts
- H Canyon and waste handling impacts
- maintaining nuclear competence

The study concludes that WSRC considers K-Reactor restart within the stipulated five-year period a significant risk after a long standby. The risk is mainly due to SRSs limited ability to retain or attract highly capable and knowledgeable professionals in standby scenarios. K Reactor is a heavy-water reactor with safety and operating parameters and systems different from commercial light-water reactors. Without an operating mission, the technical competence required to assure successful restart is not likely to be retained or found elsewhere. Experience in restarting SRS reactors supports this judgment. The continued operation scenarios will maintain technical competence. An ongoing reactor operation mission will make the site an attractive career opportunity to many people and the required professional staff can be retained from the large pool that was assembled and trained from the recent successful K-Reactor restart.

SRTC Review of K-15 Fuel Nonconformance Reports

L. R. Chandler

The Reactor Division requested that the SRTC review and approve 27 Nonconformance Reports (NCR) concerning components to be irradiated in the K-15 reactor charge. Thirty-two related NCRs were also included for information purposes, not for review or approval. SRTC personnel reviewed the NCRs for the effects on reactor physics, reactor engineering (thermal-

hydraulics), reactor safety analyses, and reactor materials.

Approval was given for all NCRs, however, twelve will require additional technical work for use in K-Reactor operation. The K-15 (formerly P-11) and spare L-4 charges will contain components covered by the NCR. The major SRTC technical work for the NCRs will be to evaluate the effects of component dimensional variations larger than those included in the safety analyses and thermal-hydraulic limits. The expected effect is minimal, but has not been quantified.

SRTCs plan to include NCRs in the K-15 safety and operational analyses involves several steps:

- After issuance of the NCRs, personnel from the Safety Technology Section, the Materials Technology Section, the Nuclear Engineering Section, and the Scientific Computations Section will prepare a detailed technical work plan and schedule for the K-15.1 subcycle. This plan should be issued in October 1992.
- The work items and deliverables from the first step will be included in the K-Reactor Master Tracking System for the K-14.1 through 15.1 outage and other subcycles using components covered by NCRs. Much of the work for K-15.1 is expected to be generically applicable to other subcycles containing the components.
- After the work items and deliverables are completed, they will be closed out in the Master Tracking System and a final summary report will be issued.

The memorandum (SRT-APG-920121) documenting the SRTC review of K-15 Fuel NCRs provides more details concerning this issue.

K-Reactor Restart: Californium Source Rod & Partial Control Rod Modification

G. W. Richardson

Two californium (Cf) source rod assemblies are used in K Reactor during startup. Each source rod assembly occupies the axial channel (G) of a septifoil and its design is similar to that of a partial control rod. The Code Development Group's analysis results of the Cf source rod design indicate that guide tube overflow is highly probable when used in a type Q septifoil. Hydraulic test-

ing in the Heat Transfer Laboratory indicates a need for new design of a source rod. The Nuclear Engineering Section design is complete, the test assembly was fabricated, and hydraulic testing began.

The type Q septifoils operate in K Reactor at reduced flows to prevent possible guide tube overflow of reactor moderator via partial control rod extension tubes. Fabrication and use of partial rods in a full-rod configuration eliminates guide tube overflow at full-flow conditions and will be implemented during the scheduled cooling tower tie-in shutdown.

The newly designed full-length partial rods received full approval and were released for fabrication.

Cooling Tower Related Tasks

K. K. Reeves, K. L. Barbour, and D. A. Burge

The Nuclear Engineering Section completed four draft memos documenting three cooling tower related task items:

- reactor tank pressures due to the use of the river water system as an emergency cooling system (ECS) source during a loss-of-power accident
- the effect of a loss of 186 Basin inventory on ECS flow
- the effect of tripping the cooling water pumps on loss-of-coolant accident ECS limits

Code Development

R. A. Dimenna

The Nuclear Engineering Section completed a preliminary proposal, incorporating the views of the Scientific Computations Section and Idaho National Engineering Laboratory, for a K-Reactor simulator using RELAP5. The K-Reactor simulator idea appears to be a good method to tie code development, experimentation, and reactor analysis together in a meaningful way. It also supports code development and improves the ability to analyze and safely operate K Reactor.

Alternate Core Cooling Study

J. C. Whitehouse, G. L. Gill, J. R. Taylor, and
A. A. Zagrodnik

A task team composed of the Nuclear Engineering Section personnel was formed in August 1992 to study possible methods for cooling fuel and target assemblies that may remain in K-Reactor's moderator tank during future shutdowns. If suitable methods can be found, the life of a fuel charge can be extended up to five years. Five viable cooling options were identified, three of which involve leaving the moderator tank full.

The "wet" cooling options are:

- to continue limited moderator circulation with two DC powered Bingham pumps
- to install a new Auxiliary Heat Removal System that consists of four small pumps and heat exchangers.
- to use natural circulation in the fuel assemblies to transfer heat to the top shield cooling system

Two "dry" options were identified to allow maintenance of an empty moderator tank, they include:

- temporary installation of heat pipes in an outer ring of the reactor to remove decay heat from air that circulates between the fuel assemblies and tank by natural convection
- providing an external mechanism to circulate air through the fuel assemblies and remove heat

The "dry" cooling options can be used after the reactor has been shutdown for more than 10 months because the heat removal rates are low.

A report (WSRC-TR-92-443) was issued to document the task team's findings and recommendations for further work.

Loss-of-Pumping Accident Limit Criterion

A. J. Garrett

Recent analysis of test data that simulated conditions in a MK22 assembly during the Emergency Cooling System phase of a loss-of-pumping acci-

dent (LOPA) indicates that thermal excursions occur at lower powers than expected. These tests were performed with the SPRIHTE rig in the Heat Transfer Laboratory. SPRIHTE is a prototypical mockup of a MK22 assembly. The LOPA criterion was originally based on results from a single-channel test rig. Based on the current analysis, the LOPA limit may drop from 40% to 37% of historical power levels. This drop could be offset by a reduction in the number of inoperable thermocouples allowed in the reactor. The Nuclear Engineering Section is also investigating the potential for replacing the Stanton number criterion for prediction of the onset of significant voiding with another criterion that applies at the low flows characteristic of a LOPA.

Corrosion of Aluminum Clad Fuel and Target Alloys in the K-Reactor Disassembly Basin

J. P. Howell

Long-term corrosion tests are underway to evaluate the behavior of aluminum clad fuel and target alloys in the K-Reactor disassembly basin environment. The initial disassembly basin tests were started with coupons of 6063, 1100, and 8001 aluminum alloys immersed in the L-Reactor basin in May 1991. After 42 days of basin water exposure, results indicated pitting corrosion on the 8001 aluminum on the order of 30 mils deep, which is the cladding thickness of the fuel tube.

To better understand the alloy corrosion in the disassembly basin environment, components from actual fuel and target tubes were obtained from the Fuel Fabrication Facility in M Area. Nine-inch cylindrical "tube ends" were cut from the ends of actual extrusions. The ends did not contain the uranium or lithium core, but consisted of the 8001 and 1100 alloy cladding with a 5005/5052 alloy inert core. Similar to the actual Mark 22 assembly, the 8001 alloy tube was nested inside the slightly larger diameter 1100 alloy clad target tube. Six nested component tube ends were immersed from three to six feet beneath the surface of the water in the K-basin in December 1991.

Components were removed from the basin after exposure times varied from 45 to 182 days in a series of four withdrawals. Additional compo-

nents were immersed during this time period. Once removed from the basin, the tube ends were sent to SRTC where photography and metallurgical analysis were performed. The alloy corrosion was evaluated by serial metallography and calibrated optical microscopy. The depth of pits on the 8001 aluminum (the fuel tube cladding alloy) measured up to 53 mils and occurred during the 45-day exposure period. Pitting on the 1100 alloy, the target tube cladding, was not as rapid or severe for a given exposure up to about three months, but by the end of six months exposure, the maximum pit depth was similar to the 8001 at about 58 mils, or twice the thickness of the clad.

During the exposure period, the existing basin water chemistry was generally within the operational limits established during early site operations. Impurities such as Cl^- , NO_3^- , and SO_4^{2-} are controlled to the parts per million level and basin water conductivity is 170 to 190 mmho/cm. This test program demonstrated that the basin water is aggressive to the aluminum components at these levels. The Receiving Basin for Offsite Fuels (RBOF) at SRS and other basins around the country stored aluminum components for more than ten years without pitting corrosion. These basins maintain water chemistry in the parts per billion level (1000X less impurity concentrations) and water conductivity at less than 1.0 mmho/cm.

Recommendations were made to the Reactor Operations Department (ROD) to improve the basin chemistry. Efforts are underway to use the deionizers to lower the basin water conductivity and impurity levels. Inserts were proposed as insulators between the stainless-steel hangers and aluminum assemblies to eliminate potential galvanic corrosion. Plans are also being made to clean the basin floors to remove the sludge and residue collected over the years of operation. Studies indicate that extra deionizers used with the deionizers available to each basin could lower conductivity and impurity levels, which would enhance the storage performance of the aluminum components.

Structural Integrity Evaluation of New Partial Rods

N. K. Gupta

Partial control rods were modified to improve reactor operation by making them similar to the full control rods. The new rods have dummy aluminum slugs that were not present in the old partial rods. The new rods are identical to the full-control rods in length and other dimensions. Natural frequency calculations were performed for the new rods and the full-control rods to compare rigidity. The structural response ($\approx 1\%$) was close to the full-control rods. This similarity will ensure adequate rigidity of the new partial rods during reactor operation.

The control rods are fabricated by swaging an aluminum raincoat tube over aluminum and aluminum-lithium slugs. This operation leaves gaps between the tube and the slugs. Inspection of the gaps revealed larger gaps at certain locations specified (0.003"). Therefore, calculations were performed to ensure that larger gaps would not cause damage to the raincoat tubing. The basis for the specified 0.003" gap between the tube and aluminum-lithium slugs was also established and new gap inspection criteria were formulated.

Cooling Tower Water System

D. M. Barnes, E. W. Baumann, and G. R. Caskey

The changes in the potential for corrosion of components in the cooling water system (CWS) and the emergency cooling system (ECS), due to the cooling tower water (CTW) system tie-in, are being evaluated. Previously, cooling water was drawn from the Savannah River, passed through the system one time, and returned to the river. With the installation of the CTW, most of the water will re-circulate through the heat exchangers, cooling tower, and the 186-K basin. Consequently, the concentration of water impurities are expected to increase and their effect is being analyzed. In addition, the reactor power level is expected to be 30% of full power, which will lead to lower water temperatures.

An earlier assessment of corrosion in the CWS and ECS, in support of the Structural Integrity

Assessment Project, is being revisited taking into account the expected changes in impurity levels in the water and lower water temperatures. These changes are being evaluated to determine their effect on the active corrosion processes and the potential to activate the dormant corrosion process.

Safety Computer Upgrade Program

J. D. Bickley, J. K. Samborsky, D. M. Menci, D. M. Immel, and J. B. Jenkins

A comprehensive maintenance program to improve safety computer performance was initiated. Although it is not a part of the original proposal, the site added a task to suppress digital noise for conditions that caused spurious alarms. Temperature alarms occur when either computer is bypassed, false Automatic Backup Shutdown (ABS) alarm messages occur when bypass switches for scram instruments are operated, and multiple ABS messages occur when the lamp test is performed.

A walkdown and evaluation of noise on the process digital input and output subsystem of the safety computer was completed. The investigation found that the temperature alarms are caused by noise introduced in the thermocouple wiring by the operation of the Supplementary Safety System (SSS) bypass relays. The false ABS alarm messages were due to contact bounce on the buffer relay. The multiple ABS alarm messages were caused by the design of the input system. Information from this evaluation will be used to make design modifications to correct the problems.

Rupture Disk Failures

K. J. Imrich, P. R. Vormelker, K. M. Rehberger, M. R. Louthan, P. A. Kestin, and C. Neill

On May 17, 1992, Equipment and Materials Technology (E&MT) assisted in the removal and penetrant testing of leaking rupture discs in 105-K to ensure that damage was minimized during removal. This was the second rupture disk incident in less than one year. Site Services Quality/Quality Control dye penetrant examination (PT) revealed indications of ruptures in the disks.

Visual examination showed no apparent damage, but the tag and hook were bent. Failure analysis revealed a small crack in each tantalum disk, confirming the PT indication. The scanning electron microscope examination of the fracture surfaces showed post fracture damage (metal smearing) that masked the actual failure mechanism. The manufacturer and E&MT performed pressure testing to duplicate the failures. The results indicate that ductile overload or fatigue were the most probable causes and environmental effects were not. Three rupture disk failures occurred since December 1991, which indicates a reliability problem with the disk design. A new design, and possibly a new material, should be considered. The final report on this issue was released.

Cooling Tower—SRTC Task Team

T. E. Britt, K. J. Lansaw, and J. Buczek

In December 1991, the Safety Analysis and Engineering Services (SAES) Accident Management group identified potential problems associated with operating K Reactor with the cooling tower. An SRTC task team was formed in March 1992 to assess the impact of the cooling tower on the Safety Analysis Report Chapter 15 accident analyses. As part of the cooling tower safety analysis, a list of critical parameters was compiled and documented in SRT-SAE-920586 to ensure that a consistent set of data was used by organizations performing analyses with respect to the cooling tower. SAES and Reactor Engineering reviewed and approved the data, which meets NQA-1 requirements, and allowed certification for use in critical analyses.

Analysis of a High-Level Flux Monitor— Initiated Control Rod Reversal during a Single Rod Withdrawal Accident

D. A. Kalinich and R. S. Wittman

As part of the Revision 1 Level 2 PRA, calculations were performed to determine whether a control rod reversal, initiated by the high-level flux monitor (HLFM) in the absence of scram or supplemental safety system actuation, would prevent a single rod withdrawal accident from leading to fuel melt (the MARY code was used to

perform this analysis). The Applied Physics Group used the GRISSET and GRIMHX codes to calculate the worth of control rod reversal.

The input decks for these calculations are based on those used for the transient analyses in the Safety Analysis Report. However, they were modified to reflect reactor operation at 720 megawatts. Best-estimate parameters replaced more conservative parameters when possible. Calculations at HLFM setpoints of 130% and 140% show that control rod reversal will shut down the reactor before flow instability (a precursor to fuel melting under flow conditions) can occur.

Removal of the 484-D Powerhouse from the 115 kV Grid

D. S. Cramer

The initiator frequencies were reevaluated for loss of the SRS 115 kV Grid with 484-D powerhouse online versus offline. Loss of the SRS 115 kV Grid can initiate the loss-of-river water accident scenario. The core melt frequency, resulting from the initiator frequencies, were analyzed by using Probabilistic Risk Assessment techniques. Differences in the severe core melt frequencies are less than $1.0E-7$ per year or one percent of the total core melt frequency. Therefore, the status of 484-D online and offline does not have significant impact on Unresolved Safety Question 92-0084, which refers to the increase in the initiator when the powerhouse is taken offline. These results are lower than those previously obtained because 1987 procedure changes, human failure probabilities based on recent field studies, system upgrades, and the addition of new systems. The results were reviewed in a joint meeting with the DOE, Los Alamos National Laboratory, and WSRC.

Tritium

Tritium Shipping Container Development

L. K. Heung

A uranium hydride transport vessel (HTV) is being developed to ship tritium offsite. It ships tritium in a solid form and it is safer than the gas container it will replace. The HTV is heated to 450°C to recover the tritium. For safety purposes, the vessel is designed for a maximum temperature of 600°C, but the valves at the inlet and outlet lines are rated for 315°C. To confirm that the valves can not reach 315°C, a prototype was thermally tested to produce a temperature profile under maximum heating conditions.

The prototype, with nine thermocouples mounted at strategic points, was heated to 600°C in a crucible furnace. The results showed that the temperature at the valves was 122°C, far below the rated temperature.

Replacement Tritium Facility In-Bed Accountability Technical Support

J. E. Klein

Technical support of the Replacement Tritium Facility (RTF) in-bed accountability system was supplied in a number of areas described below.

- The operational readiness review checklist, "Materials Control and Accountability" (RTF-ORR-59) was revised and issued for field inspection.
- Hydride-bed-temperature requirements for integrated testing of the accountability system was determined in SRT-HTS-92-0155.
- Pressure transducer requirements for the in-bed accountability system leak-check and back-pressure measurements was determined (SRT-HTS-92-0160).
- Review of the process requirements (PR) criteria indicated that no PRs are needed for RTF

nuclear material accountability (SRT-HTS-92-0167).

- RTF in-bed accountability flow calibration specifications (SRT-HTS-92-0168) were supplied to the Savannah River Standards Lab for development of onsite flow calibration systems.
- The maximum tritium concentration allowed on the RTF Thermal Cycling Absorption Process deuterium beds, without requiring in-bed accountability, is 1.6 ± 0.1 mole percent tritium (SRT-HTS-92-0173).
- The Hydrogen Technology Section's (HTS) testing of the RTF in-bed accountability mass flow controllers (MFC) indicates that four of the eight units tested are functional units. HTS is testing the available MFCs for use in the RTF (SRT-HTS-92-0178).

Replacement Tritium Facility Process Requirements

A. S. Horen

Process Requirements (PR) documents are required for design areas in the RTF where the PR Criteria Guidelines (WSRC-RP-92-149) are met. PRs and their implementation satisfy many action items specified by Pre-Operational Process Hazards Reviews for the individual design areas. PR documents must be written and approved before tritium is introduced. The RTF PR responsibilities and schedule is documented in SRT-HTS-92-0147 and NMP-RTE-92-0365 and will meet tritium introduction requirements.

A PR draft, review, and approval schedule was developed in September 1992. PRs for tritium testing and operation will be reviewed and approved before tritium introduction (the projected approval date is November 16, 1992) and drafts were redistributed to team members in September 1992 for review and revision. The official reviewers are reviewing them for final approval.

PR implementation requirements and responsibilities were also identified. They were separated for actions that are necessary before tritium introduction and War Reserve production. This information is also documented in SRT-HTS-92-0147 and NMP-RTE-92-0365.

Replacement Tritium Facility Technical Issues Tracking

A. S. Horen, T. Motyka, L. K. Heung, S. Holder, I. A. Fisher, J. E. Kein, R. A. Malstrom, G. S. Nichols, F. B. Ramirez, J. H. Scogin, and J. R. Wermer

The Hydrogen Technology Section (HTS) is tracking Replacement Tritium Facility (RTF) requests for technical assistance. Benefits to RTF and HTS personnel include definition of the problem, scope, timing, resources required, and less duplication of efforts for problems previously addressed. The status of generated and completed RTF issues, and total man-hours spent addressing RTF issues is reported in the RTF Monthly Project Status Report.

HTS provides technical consultation and assistance for RTF Startup activities. RTF Technical Issues are tracked by the number of Technical Issues generated and resolved and the man-hours involved. Specific RTF issues, typically requested by RTF personnel, are tracked in this manner (general RTF issues and technology topics are not included).

Key RTF Technical Issues for September 1992 include:

- Process Requirements (PR)
- hot and cold nitrogen insulation issues and test evaluations
- Thermal Cycling Absorption Process (tubing) modification proposal
- in-bed accountability issues
- Operational Readiness Review items (catalytic purifier verification)

Forecasted activities in support of RTF will increase through tritium introduction and War Reserve production, specifically PRs and system-specific Technical and Operating guides.

Applicability of Hydride Vessel Technical Standard for Replacement Tritium Facility

A. S. Horen and L. K. Heung

The applicability of the Tritium Technical Standard, "Hydride Vessels (U)" (WSRC-TN-25) to

the RTF hydride vessels is documented in SRT-HTS-92-0148.

The Tritium Technical Standard is not applicable for the Replacement Tritium Facility (RTF) hydride vessels. The nitrogen heating and cooling system will govern the RTF vessel wall temperature and vessel pressure, which will be substantially lower than the limits stated in the technical standard. The issue of limiting the number of vessel cycles (heating and cooling) for RTF operation was investigated and concluded that no limit exists. Memos documenting this issue include:

- L. K. Heung. "Thermal Stress and Fatigue Life of RTF Nitrogen Lines (U)". SRL-HTS-89-0094 (August 15, 1989).
- M. S. Buckner. "Systems Engineering Position on Thermal Stress and Fatigue Analysis of Thermal Cycling Absorption Process Nitrogen Inlet Piping (U)". OPS-RTF-89-0265 (October 15, 1989).

Replacement Tritium Facility Transfer Line Process Hazards Review

A. S. Horen and R. A. Malstrom

Hydrogen Technology Section and the Analytical Development Section personnel formed a team for the Replacement Tritium Facility (RTF) Transfer Lines preoperational process hazards review. Several issues surfaced during transfer lines process drawings review. The transfer line components include:

- zeolite bed
- pressure transducer
- cold gold trap
- moisture meter
- automatic valve isolation (232/236)

The transfer line automatic isolation valves close when high gold trap temperatures or high moisture contents are detected (these parameters are hard wired interlocks). The automatic valve in the RTF is not interlocked to the process conditions, therefore, telephone communication will be required.

Liquid nitrogen chilled gold traps are used in the transfer lines from the tritium facilities to pre-

vent mercury contamination of the RTF as the primary defense. Because of process components in the RTF, which are incompatible with mercury, it is essential that mercury contamination be prevented.

The mercury analyzers were moved from the transfer line hoods to the mix tank glovebox (as a cost savings). The mercury analyzers vent directly to the mix tank glovebox, therefore, online analysis of transferred process gas is not possible. All gas transfers will be performed at subatmospheric pressure. It is recommended that a nitrogen flush of the transfer line header, located in the mix tank glovebox, be performed before and after each transfer and analyzed for mercury.

It is also noted that the moisture meters in the transfer hoods are located further downstream than the cold gold traps (operated at liquid nitrogen temperatures). Therefore, the measured moisture content of the gas transferred indicates a faulty moisture meter.

The transfer line operating information is being developed for RTF procedure generation and will be documented in a future memorandum.

Replacement Tritium Facility Loading Fixture Specification Development

D. D. Alexander and R. B. Looney

Replacement Tritium Facility (RTF) personnel requested that Equipment and Materials Technology develop a technical specification for procuring loading fixtures and delivering ten sets of ten loading fixture assemblies.

Fabrication and inspection for 1M/2M loading fixture bodies were completed. Construction and Central Services Works Engineering (CSWE) are fabricating Auxiliary 1M/2M fixture parts. Two fixture assemblies and five additional orders will be submitted. CSWE is also fabricating auxiliary parts for an additional fixture. The technical specification for procuring the remaining fixture is in the development cycle.

Storage Test Parts

W. R. Kanne, Jr.

Twelve reservoirs, fabricated using resistance upset welding, were loaded with tritium for the program to develop resistance welding technology for fabrication of reservoirs. Six reservoirs are of a spherical design and six are of a current production design. The reservoirs were fabricated from forged 21-6-9 stainless steel using the large resistance welder in to join the top and bottom hemishells at the girth. They will be stored as part of the program to demonstrate long-term integrity of upset girth welds for tritium service. Reservoir storage concludes a program to design, fabricate, qualify, and load forged 21-6-9 reservoirs for Los Alamos National Laboratory. The reservoirs will be removed for testing to determine any detrimental effects of tritium exposure from storage at intervals of several years. Four other reservoirs fabricated from bar stock 304L stainless steel remained in storage eight and a half years without incident. Planned destructive evaluation of two others in this reservoir set, after shorter storage periods, show no adverse effects from tritium service.

JBF Test Program

W. J. Rogier

Sandia National Laboratory-Livermore (SNLL) initiated the JBF program to examine long-term tritium storage effects on variously configured prototype units. Assembly, loading, storage, and function testing were to be performed at EG&G Mound Laboratories. However, due to consolidation uncertainties, it was halted pending the final outcome. In the interim, SNLL requested that SRS develop plans for taking over the JBF program. The following topics were discussed during the meeting:

- Loading the units in the Replacement Tritium Facility (RTF) may be necessary because of the possibility of program schedule uncertainties related to the date when the Record of Decision is made on reconfiguration. Therefore, an

introduction of the project was made to the RTF personnel in attendance.

- Possible loading schemes for 234-H and RTF were discussed and acceptable scenarios for both facilities were agreed upon.
- An older prototype unit was in the Hydrogen Technology Section's custody. The orientation of the unit fill valve is opposite of the orientation of drawings of the unit. A confirmation must be made on whether the valve is oriented with the gas to the valve seat (acceptable, as seen in the older unit) or if the valve is oriented with the gas to the valve stem (contrary to current procedure, as seen on unit drawings).
- Brad Meyer, of SNLL, verified a maximum loading pressure.
- Potential secondary designs and shelf testing were discussed. Secondary development will not be pursued until the Record of Decision is made.
- The JBF project plans to cancel formal meetings until the Record of Decision is made. Meanwhile, communication between personnel will be aimed at the resolution of unanswered questions. A schedule to transfer the units from EG&G Mound Laboratories and commencement of unit processing will be enacted when a positive response from the Record of Decision is received.

Evaluation of High Frequency Ultrasonics for Pinch Weld Inspection

E. A. Clark

Sandia National Laboratory-Livermore (SNLL) is evaluating modern ultrasonic nondestructive evaluation technology that employs high frequencies and modern data discrimination techniques for determining the quality of pinch welds. The pinch weld is the resistance forge weld that seals tritium reservoirs after loading. The Equipment Engineering Section St198 fabricated a series of test pinch welds at a variety of weld voltages, resulting in many bond lengths and types. Welds were made in air, nitrogen and hydrogen. For each weld voltage and atmosphere, four identical welds were made. Two of each set were destructively examined by metallography or hydrostatic testing. Metallography revealed various bond microstructures. The remaining two pinch welds at each condition are

available for evaluating modern nondestructive evaluation methods, including ultrasonic techniques.

In SNLLs technique, a transducer creates 25 MHz ultrasonic pulse that travels through water and enters the outside of the pinch weld. The wave travels through the tube side, the bond, the other tube side, and is then reflected from the back side. The reflected wave travels back through the bond and tube sides, emerges from the pinch weld and travels back to the transducer, which now serves as a detector. While traveling through the pinch weld, partial reflections from the front of the tube, the bond, and the rear surface occur. The echoes were detected and fourteen features of the echoes (e.g. rise time, pulse time, and energy content) were evaluated for their ability to distinguish between bond types. Modern multi-discriminant classification software identified four of fourteen features from that correctly predicted weld atmosphere. Data evaluation continues and additional tests are planned.

Characterization of SAES St198 Zirconium-Iron-Tin Getter

W. C. Mosley

SRS, EG&G Mound Laboratories and LANL are conducting a waste minimization program to develop a nonoxidative tritium stripper system based on metallic getters. Initial development is being performed using a Zirconium-Iron-Tin (Zr-Fe-Sn) intermetallic alloy getter procured as St198 from SAES Getters/U.S.A., Inc. SAES St198 was characterized by mercury porosimetry, inductively coupled plasma-mass spectrometry, scanning electron microscopy, energy dispersive x-ray analysis, electron probe microanalysis, and x-ray diffractometry.

The SAES St198 getter was supplied as durable, cylindrical pellets made by compaction of granules of Zr-Fe-Sn alloy up to 150 mm in size. The pellet density is 5.2 g/cc corresponding to 24.8% open porosity and little closed porosity. Bulk composition of St198 is 73.6 wt% (w/o) Zr, 23.3 w/o Fe, and 1.2 w/o Sn. St198 alloy consists of Zr_2Fe primary phase and four secondary phases ($ZrFe_2$, Zr_5FeSn , α -zirconium, and η - $Zr_4Fe_2O_{0.6}$).

The hydride-forming properties of the primary Zr_2Fe phase influence the hydrogen absorption characteristics of SAES St198 getter. Zr_2Fe hydrogen absorption produces ZrH_2 and $ZrFe_2$. Gettering behavior of St198 differs from that of Zr_2Fe because of α -zirconium, and possibly Zr_5FeSn , that react with hydrogen. The secondary phase $ZrFe_2$ does not form a stable hydride and reduces the hydrogen capacity of St198. The influence of η - $Zr_4Fe_2O_{0.6}$ on hydrogen absorption characteristics is uncertain.

SAES St198 characterizations are also being performed from hydrogen absorption tests at 200°C, 300°C, 350°C, 400°C and 500°C, SAES St198 from bench-scale getter performance tests at Los Alamos National Laboratory, and two getters made by UltraPure Systems, Inc. (UP1202 and UP1302) with compositions similar to SAES St198.

Life Storage Program

H. D. Brown

The Life Storage Program is an ongoing stockpile surveillance program designed to ensure reservoir reliability. This involves loading, unloading, reclamation, destructive testing, and metallurgical analysis of reservoirs.

Thirteen reservoirs were transferred to Tritium Operations for unloading and eight special type reservoirs were unloaded. Six special refills were filled at the request of W. R. Kanne and K. Dunn and will be stored in secondary containment in the Materials Testing Facility (MTF).

Permeation tests by ion-current measurement in the MTF run on a monthly basis. Approximately 85% of the scheduled measurements were made. The Pressure Burst Test Facility is preparing for restart and the tritium technical engineer in charge of the facility was temporarily assigned to the Replacement Tritium Facility.

Leak Test Systems for the Container Management Facility

J. V. Cordaro, W. D. Thompson, J. W. Anderson, G. E. Reeves, D. D. Rucker, J. S. Martin and H. B. Hoover

In late September 1991, Equipment and Materials Technology (E&MT) received authorization from the Tritium Projects Management Department to begin work on leak test systems for the Container Management Facility (CMF) S-4655 project. E&MT is responsible for designing, installing, and checking the systems. The CMF must be built so that Tritium Production can handle the new H1616-1 and H1616-2 shipping containers. DOE mandated that all loaded reservoirs be shipped in the Sandia National Laboratory - Albuquerque designed shipping containers.

The leak test system for post load leak testing passed functional and calibration tests and is ready to be turned over to Tritium. Tritium agreed to delay the final turnover (originally scheduled for October 1, 1992) until November 1, 1992. The delay was caused by:

- Tritium's delay in reviewing the operating procedures issued by E&MT
- Bechtel Construction's delay in completing the B punch list items

Bechtel Construction completed B punch list items on October 16, 1992. Post installation testing of the six-belljar station assembly for annual recertification was completed. Functional testing is underway to determine if helium contamination of valve O-rings in the belljar assembly will cause unacceptable delays in leak testing.

Activity Transfer Group Meetings

K. A. Dunn and K. M. Keeler

As a result of the Non-nuclear Consolidation Plan to downsize the DOE Nuclear Weapons Complex, Activity Transfer Groups (ATG) were formed to provide the structure, objectives, and deliverables for the transfer of operations from donor sites to receiver sites. Reservoir Surveillance Operations (RSO) and Gas Transfer Systems (GTS) at EG&G Mound Laboratories are scheduled for transfer to SRS as part of the Non-

conformance Plan. Meetings of the RSO and GTS ATG outlined schedules for transfer of equipment and technology and provided an Activity Transfer Plan (ATP) for the activities. The ATP addresses items such as assumptions, issues and concerns, scheduling, qualification plans from the Design Agencies, Environmental, Safety, and Health concerns with the workload, and critical skills for the transfer from EG&G Mound Laboratories to SRS. Input from the ATG members is addressed in the ATP. Specific items discussed at the ATG meetings are shown below.

Gas Transfer Systems

The main ATP text has not changed since the August 1992 ATG meeting. A schedule for shipping units from EG&G Mound Laboratories to SRS was reviewed. According to the schedule (H1616-2 and UC-609), enough shipping containers exist to adequately ship the units within the time frame.

Reservoir Surveillance Operations

Few changes were made to the RSO ATP. Burst testing costs were updated to extend testing to Kansas City. "Pneumatic Burst Testing vs. Hydrostatic Burst Testing (U)" (SRT-HTS-92-0149), was also revised to include the most recent cost estimates for burst testing at the three sites.

Units required for GTS and RSO startup were discussed for incorporation into the Reconfiguration Pilot Production Program Definition (R-PPPD) plan. The R-PPPD includes units for qualification of the pneumatic and/or hydrostatic burst tester at SRS, in relation to those at EG&G Mound Laboratories and Kansas City. It also includes the reservoirs and hardware necessary to test the GTS and RSO function testers. In addition, two systems are required to qualify the environmental tests (drop, centrifuge, vibration, and function). The units and hardware are included in the R-PPPD transmitted on October 5, 1992.

DOE-Albuquerque agreed on the acceptance statement for the proposed moratorium on surveillance testing after the shutdown of EG&G Mound Laboratories and before the startup of RSO at SRS (also incorporated into the ATP).

Commercial Sales and Inertial Containment Facility Loadings Activity Transfer Plan Activity Transfer Group No. 4

W. N. Posey

The NCP Activity Transfer Plan-Activity Transfer Group No. 4, was revised after DOE (Albuquerque Operations (ALO)) and EG&G Mound Laboratories personnel resolved questions and concerns. The revised plan will be submitted as the final Draft (Revision 4) to DOE-ALO, (Savannah River Operations) and EG&G Mound Laboratories by October 29, 1992 for approval and additional revisions upon their review.

On November 20, 1992, WSRC will issue the plan as approved for transferring Commercial Sales and Inertial Containment Facility (ICF) loadings to SRS after EG&G Mound Laboratories' tentative March 1995 production cessation date.

The project's mission is to establish facilities and technologies at SRS that accurately load, package, and ship small tritium-loaded SS cylinders, U-beds, ICF microspheres, and high purity ^3He loaded cylinders to commercial customers, federal agencies, and other DOE facilities.

Tritium Commercial Sales/ICF units and ^3He customer loading schedule requirements will be met as follows:

- Small SS cylinders will begin loading in the RTFs mass spectrometer glovebox, with less than 1000 curies of T_2 at less than atmospheric pressure, on January 15, 1995.
- U-beds will begin loading, with less than 28,000 curies T_2 at less than atmospheric pressure, in the Product Container Loading Facility on March 1, 1995. In the long term, the U-beds will be loaded in one RTF Loading Line upon completion in June 1996.
- ICF Microspheres will begin loading in RTFs Loading Lines, at less than 400°C and less than 10000 psi DT and/or other gas mixtures, beginning June 1995 and will continue loading in one line for the long term.
- ^3He "pre-purified" to contain less than $5 \times 10^{-7}\text{ccT}_2/\text{cc}^3\text{He}$, will be "final-purified" to contain less than $10\text{-}11\text{ccT}_2/\text{cc}^3\text{He}$. The purified ^3He will be loaded into customer supplied

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containers, ranging in internal volume from 10cc to ~44 liters at pressures from 15 psia to 1800 psia, according to customer requirements.

The SRS loading schedule will satisfy commercial sales customer requirements.

Bechtel Savannah River, Inc. is revising the Tritium Consolidation Conceptual Design Report, which will be reissued by November 1992 to reflect the changes made since April 1992.

Separations

Storage of Americium-Curium Solutions

J. H. Gray and R. W. Wainwright

Laboratory studies show that uranium should not be used to poison americium-curium solutions neutralized with excess caustic because carbon dioxide absorbed from air, forms soluble uranyl carbonate complexes.

Uranium dissolution was observed in the laboratory using synthetic americium-curium solutions that initially contained excess caustic. For two months, the uranium concentration remained between 2 to 4 ppm. Over a three week period, the soluble uranium increased from 4 ppm to more than 3000 ppm. As the uranium redissolved, it separated from the americium-curium in the solids, therefore, it could not provide criticality control.

EBR-II Container Materials

J. H. Gray and R. W. Wainwright

The dissolution rates for a new type of aluminum and iron set-screws and bolts were measured for encapsulating the leaking EBR-II fuel container. The aluminum completely dissolved in a 6 molar nitric acid - 0.05 molar fluoride solution after boiling for six hours. However, the iron set-screws completely dissolved after 15 minutes of boiling. One hour of boiling was required to completely dissolve the iron bolts in the same solution.

Elbow Cutting Pipe Crawler

W. T. Zollinger, R. C. Treanor, and C. W. Robinson

Separations requested that Equipment and Materials Technology (E&MT) supply a pipe crawler to remove an elbow in the FB-Line ventilation system to alter air flow paths and prevent the high-particulate air filter from clogging with soot

and cutting off air flow to an entire floor of the building during a fire.

The task required the development of a specialized pipe crawler capable of crawling through 90° elbows and up a vertical pipe in the 36" process air duct. The pipe crawler contains two main segments— a tractor for locomotion, and a cutting attachment to hold and articulate a plasma arc torch. The control panel uses a 230 ft. tether for teleoperation in the duct, which reduces the radiation exposure to field personnel (it is the same panel used for other E&MT pipe crawlers).

Testing in the full scale mockup in F Area began. It includes debugging the crawling geometry and cutting process. A formal demonstration of the equipment will be scheduled after the testing is successfully completed. Elbow removal was shifted to the FY94, therefore, the equipment will be completed and used for demonstrations until that time.

H-Canyon Safety Analysis Report Addendum

J. M. Low, P. L. Fisk, and C. R. Lux

As a result of the examination of the nominal source term issue, DOE indicated a desire for an immediate safety analysis report (SAR) revision for H Canyon. This revision addresses the Frame Waste Recovery Facility that supports HB-Line restart.

Nuclear Processes Safety Research issued a SAR addendum for review and comment. Using revised nominal and maximum source terms, the addendum addresses all accidents for Frame Waste Recovery Facility. The addendum is presented in the form of a complete Chapter 5, pre-Nuclear Regulatory Commission format SAR and includes recalculated doses and risks using ICRP 30 dose values for earthquake, external impact, fire, uncontrolled reactions, transfer error, overflow, leak, and coil and tube failure. Facility Safety and other Environmental, Safety, Health and Quality Assurance sections provided comments in a timely manner. These, and anticipated comments from Separations, will be addressed (WSRCs approval goal date is October 30, 1992).

Tank Solids

J. H. Gray and R. W. Wainwright

The latest in a series of solution samples from two tanks was received and the solid composition was determined. Concern arose that the solids in the tanks contain plutonium that precipitated during storage. The solids from one tank were filtered and the composition was determined by the scanning electron microscopy (SEM). Results from the SEM scans confirmed that the solids are similar to previous samples and that plutonium is not present.

HB-Line Restart

D. P. Eisele

The HB-Line Justification for Continued Operation was submitted for final WSRC approval on October 19, 1991.

PuO₂ Facility Fault Tree

L. W. Christiansen

A fault tree that calculates the frequency of criticality in the HB-Line PuO₂ Facility was completed in support of HB-Line restart. It shows a frequency of 3.5E-10 per year of inadvertent criticality in the facility. This frequency is primarily due to the accidental transfer of fissile material from the H Canyon. There is also a contribution due to accidental transfer of ²³⁹Pu from the HB-Line Vault, however, since this transfer would have to remain undetected in HB Line and H Canyon on transfer and receipt, its contribution is small.

Hydrogen Gas Generation in HB Line

M. E. DelGenio

The alpha radiolysis of nitric acid generates hydrogen gas in the process vessels in HB Line. This hydrogen gas evolution in HB Line was analyzed for the Pu-238 campaign in August 1991 (SRL-OAG-910103). The analysis evaluated the hydrogen generation rate, the percent of hydrogen in the vapor space of the vessels, and

the time to reach the Technical Standard limit of 3.9% hydrogen in the vapor space under static conditions. This information was provided in response to issues raised by the Office of Nuclear Safety on their evaluation of HB Line. A reevaluation (SRT-SEP-92-0074) of the values was performed and issued in October 1992 in support of the revision of Limiting Condition of Operation 3.5.1 in the HB-Line Operational Safety Requirements.

HB-Line Direct Support

M. A. Whitney and A. G. Eggers

The Packaging and Transportation (P&T) group provides support to HB Line in procedures and test equipment. Particular attention focused on the revised HB Line and P&T Safety Analysis Report Packages in areas that control HB-Line acceptance and periodic testing.

Onsite Packaging Inventory

R. S. Maurer, A. G. Eggers, and B. Jackson

Drafts of the following documents are complete:

- Criteria Basis for Onsite Transfers of Radioactive Material
- Criteria for Onsite Transfers of Radioactive Material
- Safety Assessment Requirements for Onsite Transfers of Radioactive Material
- Contents Characterization and Package Usage Data Sheets

These draft documents form a set of interrelated documents that describe a system for performing onsite transfers of radioactive material. A peer review meeting was held on October 13, 1992 for the DOE and potential customers to become current with the proposed technical approach and to provide feedback to P&T. Another meeting is scheduled for early November 1992 to complete the presentation of this material. The documents will be updated as necessary and a formal review will be scheduled.

Compilation of 5320 Safety Analysis Report Package

M. A. Whitney, K.M. Stawney, and A. G. Eggers

The final version of the 5320 Safety Analysis Report Package, Revision 3, was submitted to DOE for transmittal to the Packaging Certification Office.

HB-Line Operational Safety Requirement Revision Review

E. A. Kyser

The proposed revision to HB-Line LCO 3.5.1 regarding Air Purge to vessels was reviewed. Reference to hydrazoic acid was removed from the basis of the LCO. A literature review of this issue resolved the concerns of a possible explosion resulting from the formation of hydrazoic acid.

Hydrazine is added as a nitrous acid scavenger before the precipitation process to keep plutonium in the +3 valence state. Upon reaction with nitrous acid, the plutonium forms hydrazoic acid, which is somewhat volatile and explosively unstable in high concentrations in liquid and vapor forms. Hydrazoic acid vapor is not flammable in low concentrations, but it can undergo an explosive decomposition (without oxygen) at moderately low concentrations. However, explosive vapor concentrations in HB Line are prevented by the small concentrations used in the process and the vapor-liquid behavior of hydrazoic acid in the process solutions. The Operational Safety Requirement LCO controls the amount of hydrazine added to the HB Line (limiting the concentration of hydrazoic acid that can form). The amount of hydrazine normally used in HB-Line processing cannot form explosive vapor or liquid mixtures. Purge gas calculations should not be affected by the use of hydrazine in HB-Line vessels.

Decontamination of Rocky Flats Plant Residues

R. A. Pierce

In response to a request made by Ed Moore, of Separations, a preliminary report was written on decontaminating Rocky Flats Plant (RFP) incinerator ash and ash heels in HB Line. The report examines the basic flow scheme if a common-electrolyte electrochemical dissolver were installed in HB Line. As expected, modifications would be minor, compared to building new facilities. A comparable report with a higher level of detail was written for New Special Recovery.

Receiving Basin for Offsite Fuel—Fuel Slug Retrieval

M. Hapstack, R. L. Minichan, and D. C. Patterson

A storage canister that houses EBR-II fuel assemblies in the basin of 244-H Receiving Basin for the Offsite fuel facility developed a leak. The combination of water and sodium inside the fuel canister generated hydrogen that combined with uranium to form expanding oxides and hydrides in the top portion. As a result, the upper 1/3 of the canister split. Equipment and Materials Technology was requested to develop a set of tools to safely relocate the fuel canister for shipment to the 221-F dissolver without jeopardizing its integrity.

Separations developed a canister relocation tool that successfully moved the canister to a tilt table. In mid-October 1992, an encapsulating device was delivered to the customer so the canister can be overpackaged for safe shipment to the dissolver.

Residue Elimination Project—Modeling Review

S. D. Fink

The Residue Elimination Project (REP), of the Los Alamos Technical Office, requested that a team comprised of members from the Technology Assessment Selection Panel (TASP) Collaborative Modeling Effort initiate an independent

review of modeling efforts associated with the REP. The review is justified as part of TASP, in that a scenario for the proposed Complex 21 facility includes the processing of the residue materials stored at RFP. The team members were J. Ostic and K. Gruetzmacher (Los Alamos National Laboratory), E. Merewether (Fluor Daniels Inc.), and the author. WSRCs effort is funded by the Albuquerque Field Office.

An initial Verification and Validation REP modeling effort was conducted mid-October 1992 at Rocky Flats Plant, to evaluate three options:

- shipment-as-waste to the Waste Isolation Pilot Plant or another repository
- shipment-as-residue to another DOE complex processing facility
- performing actinide separation at RFP before disposal as waste

DOE personnel requested that the effort be continued with an objective of issuing a final report by late November or early December 1992.

Special Nuclear Material Casting Developments

M. A. Atristain, V. L. Skidmore, H. B. Haynes, E. Proctor, L. B. Romanowski, M. A. Killian, M. L. Rogers, and D. D. Rucker

Development of a melt injection system and a squeeze casting system as potential candidates for future Special Nuclear Material (SNM) casting technology continues. The first version of a melt injection system is being used to cast aluminum. The second version, which will be used at Lawrence Livermore National Laboratory for testing with plutonium, is up for bids. A squeeze casting system is near the test stage with surrogate materials. Equipment and Materials Technology also investigated various metal alloys, coatings, and metal surface treatments, for SNM casting equipment components. Radiography of several bismuth-tin castings was completed and metallurgical examination of the castings is in progress.

Los Alamos National Laboratory Oxide/ Skull Roast Furnace

W. J. Randall

The clamshell furnace design oxidizes plutonium metal machining scrap and skull in an open crucible. This operation results in oxide dust contaminating the inside of the glovebox, requiring extensive decontamination to minimize exposure to technicians. Equipment and Engineering Technology (E&MT) was asked to design a new furnace system to reduce the contamination and improve the collection of plutonium oxide dust.

E&MT wrote specifications for a new furnace and the associated components, based on a design in New Special Recovery, and routed them to Los Alamos National Laboratory (LANL) for comment. D. Olivas, LANL Section Leader for component fabrication at the Nuclear Materials Technology Division, will return comments by the end of October 1992. LANL will then purchase the components and E&MT will assist in assembly and testing.

Technology Assessment Selection Panel Plutonium Simulation Effort

S. D. Fink

Los Alamos National Laboratory (lead laboratory) revised the Programmatic Environmental Impact Statement data package for the Complex 21 plutonium processing facility for transmission to Fluor Daniels, Inc. The information reflects a conservative envelope of operation sufficient for several identified processing options, including the BASELINE flow sheet, the Aqueous Nitrate Alternative, two chloride processing alternatives, and alternatives that include Mediated Electrochemical Oxidation and advanced disassembly and metal purification processes.

Los Alamos National Laboratory FY93 Technical Assessment Development Plans Review

F. R. Graham and S. D. Fink

At the direction of DP-42, the lead laboratories for Complex 21 developed Technical Assessment Development Plans (TADP) for FY93. Frank Graham, in his support role for DP-40, was asked to evaluate the Los Alamos National Laboratory TADP for the plutonium processing facility.

Initial comments, with emphasis on the Nitrate Recovery (Fink) and Disassembly and Metal Purification (Graham) areas of the BASELINE flowsheet, were provided. Additional information for these and other areas of the flowsheet were requested.

Glovebox Glove Tracking System

Doug Walpole, Gardner Whittle, James Clark
and Dan Walker

A computer-generated glovebox glove-change tracking system was developed and is being implemented in the Chemical Technology Section. It is patterned after a glove change system developed at the Rocky Flats Plant and described at the 1992 American Glovebox Society Conference in Albuquerque, New Mexico. The system will track port numbers, last change dates, and expiration dates for each glove port. It will also instruct users to begin procuring replacements 60 days before glove changes are due. When the change date passes, the system supplies a "Do Not Use" warning. The database includes: laboratory room number, glovebox orientation diagram, glove type, thickness, and change frequency. Workplace safety was enhanced and developing a more realistic change frequency based on actual glove use could result in a cost savings.

Environmental

Pen Branch Fault Confirmatory Drilling Completed

A. L. Stieve

Based on earlier work, Defense Programs and New Production Reactor initiated funding for the Confirmatory Drilling Program to confirm conclusions on the nature of the Pen Branch Fault (PBF). Eighteen closely spaced boreholes were drilled over the fault. Sixteen holes were drilled to 300 ft and two holes were drilled to basement (~1000 ft). The boreholes were continuously cored and geophysically logged. Core descriptions were also completed and five cross-sections are under construction.

Site Specific Response Spectra

R. C. Lee

October 1992 activities in support of the Site Specific Response Spectra program included:

- the kick-off meeting at the University of Texas (Austin) to discuss the subcontract for dynamic tests of SRS soil samples in a hole adjacent to PBF confirmatory hole. A demonstration of the university's resonant column and cyclic torsional testing system was provided and testing procedures were reviewed.
- development of a system to deconvolve instrument response from the vertical array records. When this process is completed, the August 1992 Charleston, and other records, can be examined for site response and information.
- a presentation to the Seismic Advisory Committee on the status of the program

Environmental Effects of Potential Tritium Releases at the Replacement Tritium Facility and EG&G Mound Laboratories

D. M. Hamby

DOE hired SCIENTECH to determine the potential health effects on people living near a DOE tritium processing facility (e.g. SRS or EG&G Mound Laboratories). SCIENTECH requested that the dose resulting from a hypothetical atmospheric tritium release be modeled. At any given downwind distance, the maximum dose to the individual is higher if the release occurs at SRS, due to a shorter effluent stack (15 meters at SRS, as opposed to 60 meters at EG&G Mound Laboratories). However, because of the proximity and size of the population base residing within 50 miles of EG&G Mound Laboratories, the cancer risk is almost fourfold greater.

National Pollutant Discharge Elimination System Outfall Screening Analyses

W. L. Specht

Screening level analyses for total residual chlorine, ammonia, and pH were performed on SRS sanitary outfalls. The results were similar to earlier results and indicate that several sanitary plants discharge high concentrations of chlorine and/or ammonia. A sample will be collected for metal analyses with the second round of toxicity testing that will be conducted before the end of the year.

Radiological Assessment Program Cesium Issued

Evans, Carlton, Murphy, Strom, Geary, and Pinder

The Radiological Assessment Program document on cesium in the SRS Environment (WSRC-RP-92-250) was issued. It is the second in a series of eight documents on individual radioisotopes released from SRS. A document on tritium was previously issued and documents on iodine, uranium, plutonium, strontium, carbon, and technetium are being prepared.

New Sanitary Landfill Siting Studies

L.D. Wike, R.K. Aadland, J.S. Haselow, and J.B. Gladden

Field reconnaissance of the seven candidate sites for the new sanitary landfill was conducted and the preliminary site scoring was completed. The initial scoring was based on site knowledge and features that could be determined from the surface. Of the seven sites, four had similar scores and are candidates for subsurface characterization. Subsurface characterization is required before final site acceptability can be determined. The results of the initial evaluation was presented to DOE and WSRC Waste Management staffs.

Initial Cataloging of SRS Historical Photography was Completed

H. E. Mackey

Initial cataloging of the 80,000 frames of SRS historical photography was completed. It provides documentation on the existing collection of imagery by acquisition date, type of photography, and location. The ultimate objective of this effort is to develop a computerized database of SRS photography to improve access to this resource.

Study of Mercury and Lead in SRS Streams and the Savannah River

J. E. Halverson, D. M. Beals, and G. Hall

A study investigating the presence and transport of four Environmental Protection Agency listed elements in SRS streams and rivers was written to complete a DOE milestone. The elemental concentrations (lead, mercury, barium, and cadmium) are approximately 100 times lower than drinking water requirements. Barium is the only element that consistently has higher concentrations than blank sample results, however, the concentration at all locations is below regulatory requirements. The measured concentrations of mercury, lead, and barium are higher in the Savannah River than in onsite streams. Mercury, barium, and cadmium exist in higher concentrations in Pen Branch (at Road B) than any other

onsite stream (the source of this increased level is undetermined at this time). Four Mile Creek samples show undetectable levels of mercury, but it is more elevated than other onsite streams. Follow-up studies are planned for Pen Branch and Par Pond.

Wetland Evaluations Conducted

V. A. Rogers

Delineations of wetland boundaries were performed for a TNX seepage basin, the proposed Central Sanitary Treatment Plant, a waste site near Central Shops, an area near the Par Pond dam, the SRS Savannah River boat dock area (Millstone steam generator transport), and the seven candidate sites for the New Sanitary Landfill.

Par Pond Aerial Gamma Survey Results Reviewed

H. E. Mackey

The results of three Par Pond aerial gamma surveys were reviewed with EG&G Mound Laboratories (Nevada) staff. Preliminary reports were reviewed and the draft reports will be delivered to WSRC in early 1993. Preliminary plots of the distribution of gamma emitting radionuclides are available for review.

Dosimetry Technical Support

D. M. Hamby

Technical support of the Engineering Development Group's dosimetry function for October 1992 include:

- providing response to the Environmental Protection Agency comments on the F and H

**Baseline Risk Assessment to Environmental
Restoration Department**

- providing input for self-assessment of technical work control on Tank 48 heat transfer experiments
- reviewing dose calculations for the K-Reactor Safety Analysis Report
- supplying the site As Low As Reasonably Achievable Release Committee with dose-release factors specific to A Area
- calculating drinking water doses assuming a hypothetical spill of waste tank contents in the Savannah River for Environmental Sciences Section.

**Predictive Vegetation Modeling for Par
Pond**

H. E. Mackey

A South Carolina Universities Research and Education Foundation (SCUREF) task order was awarded to implement predictive Geographic Information System modeling of the distribution of aquatic vegetation in Par Pond. The task takes advantage of earlier work conducted in L Lake. The distribution of aquatic vegetation is important in determining probable distributions of fish and other aquatic organisms in the reservoir and will be important for optimizing future sampling programs.

Seismic Monitoring Network

D. A. Stevenson

All Seismic Monitoring Network stations are operating and records are reviewed daily for evidence of localized earthquake activity. Seismic monitoring at SRS in October 1992 continues with four onsite remote recording stations. Background noise surveys for the six offsite seismic network upgrade stations were completed. DOE, through the U.S. Army Corps of Engineers, is negotiating long-term leases for four of the proposed sites. Two of the original sites are unacceptable because of high levels of background noise. Alternatives are being sought.

SRS Geologic Map

C. A. Eddy and M. Denham

The South Carolina Geological Survey was selected to prepare a surface geologic map of SRS. The survey has long been performing this type of work across the state and recently mapped a number of quadrangles adjacent to the site. The information from this program will be used in environmental information documents and safety analysis reports.

Groundwater Modeling

J. S. Haselow

Status reports were issued for two SCUREF projects for development of a multiphase groundwater flow and transport code for development of stochastic codes that simulate the effects of heterogeneity and diffusion on the performance of recovery wells.

Power House Sump pH Monitoring

R. Livingston and B. Mackey

SRTC provided rapid support for pH measurements in the 484-D (Powerhouse) large sump. Equipment was installed and is operating less than two days after the request was made. The sump receives boiler wash effluent and treatment plant water for mixing before discharge to the ash basin. A battery operated pH instrument and special mounting hardware were installed to allow the use of a standard lab pH electrode. Additional equipment was ordered to provide continuous readouts, temperature compensated measurements, and a strip chart recording of pH.

TNX Groundwater Remediation

R. L. Nichols, J.S. Haselow, and J. Bollinger

Characterization of the TNX groundwater contaminant plume will be documented in a report. Revision of the Interim Action Proposed Plan for remediation of groundwater in the contaminant plume will begin after receipt of the Environmental Protection Agency's comments. It pro-

poses installation of an interim groundwater corrective action system. To support this proposal, a preliminary two-dimensional groundwater contaminant zone of capture study was completed for the TNX contaminant plume. The results of the modeling study indicates that three recovery wells, pumping 55 gallons per minute, will be required to contain the contaminated water at TNX. A three-dimensional simulation will be completed to confirm the results of the simplified two-dimensional model.

Soil vapor samples were collected from monitoring well TBG5. The vapor samples were analyzed for chlorinated volatile organics using gas chromatography-mass spectrometry methods. Trichloroethylene (TCE) was detected at 0.13 ppm in the vapor. Groundwater in this monitoring well contains approximately 5000 ppb TCE. Atmospheric air in equilibrium with water containing 5000 ppb would contain approximately 350 ppm TCE. The low test results in the vapor sample indicates that the vapor is diluted in the subsurface by air that migrated from the surface. The large amount of dilution suggests that the unsaturated zone is conducive to vapor flow. Additional tests are being planned to gain a better understanding of the dynamics of subsurface vapor transport to support future corrective action plans that address remediation of the vadose zone.

Technical Support

J. S. Haselow, B. B. Looney, and R. N. Strom

Technical reviews of documents completed during the month included:

- Environment Protection Agency comments on the F- and H-Area integrated document
- South Carolina Department of Health and Environmental Control comments on the F- and H-Area Part B permit
- Mixed Waste Management Facility-Part B permit
- Geotrans report on the F- and H-Area flow and transport modeling
- the Interra report on the sanitary landfill flow and transport modeling
- the Energy Research Foundation's groundwater report

Meetings with Environmental Restoration personnel were held to discuss potential sites for innovative remediation technology demonstrations, such as barriers and bioremediation.

A- and M-Area Groundwater Remediation

B. B. Looney and J. Rossabi

As a result of the identification of a separate phase of solvent in a monitoring well adjacent to the M-Area Settling Basin last year, an investigation to determine the extent of the Dense Non-Aqueous Phase Liquid (DNAPL) separate phase material was initiated. Preparation of a report documenting the initial results of the investigation is underway with close cooperation from the Environmental Restoration Department (ERD). The report will document the results of a cased hole logging to determine if DNAPL can be detected in existing monitoring wells where the casing/screen is affected by the presence of separate phase solvents. The cone penetrometer detects DNAPLs using microresistivity sensors and maps the structure of clay layers that have potential for directing the flow of DNAPLs in the subsurface.

Integrated Demonstration for Cleanup of Organics in Soils and Groundwater at Non Arid Sites

Looney, Eddy, Haselow, Hazen, Rossabi, Jarosch, Berry, Kaback, Walton, Bergren, Kirr, and Barnard

Remediation

Operations at the in-situ bioremediation demonstration are continuing smoothly. The test, designed to demonstrate in-situ remediation of volatile organic compounds in soils and groundwater through microbial degradation by methanotrophic bacteria, has been ongoing for eight of the twelve months scheduled. The injection of air and methane into a horizontal well below the water table is encouraging the microbial activity, while a vacuum draws a second horizontal well above the water table in the unsaturated zone. Methane injection resulted in increases in the

methanotrophic bacteria densities known to degrade volatile organic compounds (VOC) by as much as four to five orders of magnitude. A campaign involving injection of one percent methane continued for several months. During that time most of the methane was utilized by the subsurface microorganisms to assist with degradation of the VOCs. At the end of July 1992 the methane injection increased to four percent. Since that time, the amount of methane in the extraction well slowly increased. The increased methane discouraged the activity of certain microorganisms, such as nitrogen fixers, and made the system more nutrient limiting. Thus, a new operating campaign was initiated to reduce the likelihood of nutrient limitations. Air pulsing versus air and methane will continue for the next few months. In December 1992, a decision will be made as to whether addition of nitrogen as a nutrient will be requested of the SCDHEC. This campaign will continue for the last few months of the year-long demonstration. Several lines of evidence of VOC microbial degradation are documented.

Characterization and Monitoring

A report documenting the contaminant plume at the Integrated Demonstration site after the completion of the in-situ air stripping demonstration is in preparation. Data collected after the demonstration will be compared to data collected before testing was initiated. Three-dimensional volumetric images were created on the geoscience workstation to assist in visualization of the changes in the character of the contaminant plume as a result of the in-situ remediation demonstration. Significant decreases in the size and concentration of the contaminant plume were measured.

The Environmental Engineering Section's (EES) personnel reviewed a draft document, written at Los Alamos National Laboratory, comparing the cost savings associated with field screening using the Oak Ridge National Laboratory ion trap mass spectrometer, as opposed to the contract laboratory procedures. Field screening technologies have tremendous potential to save substantial costs during site characterization and monitoring. This information should be used in discussions with South Carolina Department of Health and Environmental Control or the Envi-

ronmental Protection Agency when WSRC proposes reductions in sampling interval requirements.

Directional Drilling

Draft reports were received from subcontractors documenting the results of two horizontal drilling technology demonstrations. SEC Donahue prepared a report documenting the results of the Eastman Christensen Environmental Systems technology demonstration. CDM Federal Programs completed a report documenting the results of the utility industry demonstrated technology (Charles Machine Works/Ditch Witch). ESS personnel are reviewing these reports.

Two additional horizontal wells are planned for installation under the M-Area Settling Basin. This demonstration will target the river crossing horizontal drilling technology. Two blind hole wells will be drilled at a depth of 115 feet and extend for a length of 500 feet in the horizontal. These wells will be turned over to the Environmental Restoration Department for incorporation into the Vadose Zone Remediation Program. The Cherrington Environmental Company will drill the wells with Revert drilling fluid (expected to begin in November 1992).

Ion Trap Mass Spectrometer

B. R. Buchanan and M. Keenan

In an effort to improve the Ion Trap Mass Spectrometer method, software was coded to read the data and perform chemometric analysis. Finnigan Mat supplied the structure to the data file, which allowed the use of partial least squares (PLS) to quantify tetrachloroethene (PCE) and trichloroethane. An internal standard, bromofluorobenzene, verified instrument performance and normalized the data. The PLS model for PCE uses 4 vectors and has a standard error of prediction (SEP) of 4.2 (2.6%). The model also uses 4 vectors and has an SEP of 3.3 (8%). The models predict within ten percent, except at the extreme low end. Thus, concentrations predicted at lower than 40 ppb will be reported as more than 40 ppb. By writing the calculated data directly to a Lotus 1-2-3 worksheet, transcription

errors are avoided and researchers have convenient access to data.

Algal Strains Exhibit Differing Metal Binding Affinities

E. W. Wilde

Two thermophilic algae, *Cyanidium* and *Mastigocladus*, were screened for short-term sorption of seven metals (aluminum, cadmium, chromium, copper, mercury, lead, and zinc) that are present in SRS effluents or liquid wastes. *Cyanidium* is more effective in removing cadmium, copper, lead, and zinc and *Mastigocladus* is more effective in removing aluminum. Although, both algae are effective in removing mercury, neither removed chromium effectively. With the exception of chromium and lead, one or both algae removed more than 90% of the metals from dilute solutions after a 30 minute contact time.

Algae Binding of Mercury pH Sensitive

E. W. Wilde

Model systems for the removal of mercury, by processed algal biomass, are modified by the acidity of test waters with a drop in pH from 7 to 5, increased the removal of mercury in *Spirulina* by two orders of magnitude. At near neutral pH, *Nostoc* reduced mercury concentrations in test solutions from 1000 ug/l to 300 ug/l. At pH 5 mercury, concentrations were reduced from 1000 ug/l to 2.5 ug/l with a two-minute contact time.

Electrolytic Migration on Old TNX Basin

J. P. Bibler

Oral reports on progress in mercury removal from soil taken from the old TNX Basin using electrokinetic migration technology, were presented to Comprehensive Environmental Response, Compensation, and Liability Act representatives at the Environmental Protection Agency (EPA) (Atlanta, Georgia) and South Carolina Department of Health and Environmental Control representatives. These reports were requested when the EPA approved Phase 1

laboratory experiments. Both groups are interested in the effort and expressed appreciation for the information. M. Wilson arranged and attended the meetings.

Electrolytic Migration of the 904-A Trench Soil

J. P. Bibler

A report on T. F. Meaker's study on one-dimensional electrokinetic migration on a sample of 904-A trench soil is nearly complete. Meaker's work shows that electrokinetic migration technology can remove chromium ion from soil. Removals appear to be accomplished at the cathode for the chromium (III) ion. Any chrome (VI) initially present was apparently reduced to chrome (III), which was transported back through the soil and trapped in a polymer matrix situated around the cathode. This effort also shows that pH control at the electrodes can be accomplished by incorporating an appropriate chemical in the polymer matrix to react with H⁺ and OH⁻ generated from the electrolysis of water at the anode and cathode, respectively. This report represents an Office of Technology Development milestone.

Waste Management

Waste Management Waste Sample Tool

D. C. Patterson

Most waste tanks have a primary vessel for waste storage and a secondary containment (not intended to house waste) to protect the environment. In some tanks, waste seeped into the secondary containment, thus, causing concern for corrosion potential. This is important because the annulus pan is the final barrier between the leaking waste and the environment. Waste Management Engineering's primary concern focuses on waste in Tanks 9 and 10. To solve this problem, EMT was asked to provide a tool to retrieve a sample 40 feet below a 5" diameter access hole to the annular space.

Several tools were used in the past, but proved ineffective on the thin layer of dry waste. This difficulty prompted Waste Management Technology's decision to retrieve a liquid sample. A concept that operates on the syringe principal was developed. If the syringe hole is plugged and the plunger is withdrawn, a negative pressure is created on the inside of the cylinder. When the syringe is unplugged, while in the liquid, the liquid rushes into the chamber to equalize the pressure. Another possible design option uses an evacuated cylinder to perform the same function.

Electron Beam Irradiation Destruction of Benzene in the SRTC Waste Simulant

J. P. Bibler

SRTC liquid laboratory waste is collected in waste tanks before shipment to an F-Area evaporator for volume reduction. To be received at the evaporator, however, the waste must be analyzed to determine its radionuclide content and to ascertain that it is not hazardous due to pH, heavy metal concentrations, or organic constituents. Fortunately, SRTC has the capability to treat waste that exceeds the established limits so it can be shipped to the evaporator. However, no technology exists to address removal of benzene and

benzene derivatives expected to enter the SRTC waste stream from the Defense Waste Processing Facility analytical samples. The concentrations of the compounds could exceed Resource Conservation and Recovery Act limits and require treatment for removal before the evaporator accepts waste.

High energy electron beam technology is being studied as a method for destroying benzene compounds in a SRTC low activity waste matrix. Researchers at Florida International University, the University of South Carolina, and the University of Miami completed Phase 1 of a study designed to evaluate the importance of major waste components (particularly nitrate ion and laboratory detergents) that define energy needs for beam experiments. Phase 1 studies were performed using a small Co-60 energy source. An extensive analytical effort to identify benzene degradation products and follow the concentrations at different doses accompanied Phase 1. The report concluded that benzene in a simulant of SRTC low activity liquid waste can be taken to CO₂ and water using energy from an electron beam source. Phase 2 work, which uses the actual beam on a 4,000 gallon batch of simulant containing about 100 mg/L benzene (initial concentration), will soon begin. Studies on the effect of the beam on microbes living in the waste also began. This effort is being conducted under a SCUREF contract.

Removal of Mercury from SRTC Radioactive Waste

J. P. Bibler and J. J. DeGange

A report describing the construction, operation, and performance of the in-tank probes in Tanks L and K for removal of mercury from SRTC radioactive waste was issued. The in-tank ion exchange column probes are secured in a vacant flange in the tanks. J. J. DeGange, of SRTC Laboratory Services, designed and fabricated the probes, which are operating on two high-activity and two low-activity laboratory waste batches for the removal of mercury. Waste with elevated mercury concentrations (more than 0.200 mg/L) is circulated through a small column of GT-73 cation exchange resin in the probe. During treatment, samples were taken from the waste tank and analyzed for their mercury content. Twenty

four hours of probe treatment were required to lower the mercury concentration to a value less than 0.200 mg/L. To date, four ~4,000 gallon batches of high-mercury waste were processed, enabling shipment from SRTC to an F-Area evaporator for volume reduction.

Variable Depth Sampler

M. J. Dalmaso

Final design of the Variable Depth Sampler (VDS) modification is complete. The modifications, identified during testing at TNX, include guides to keep the hoist cable aligned with the take-up drum, an improved sample transfer system, and an improved shaft sealing mechanism. The next step includes preparation of a scope of work detailing the required work for a machine/fabrication shop. This will be completed by October 24, 1992. The VDS will be ready for shipment to the in-tank precipitation facility by January 30, 1993.

Decontamination Solution Tested

W. N. Rankin

Nova Clean, a detergent manufactured for use in cleanrooms, was identified as effective, non-hazardous, and environmentally acceptable as a decontamination cleaning agent.

Wiping tests were carried out to compare its effectiveness with the effectiveness of Momar 800 and Iradecon 210 (the most effective cleaner found so far). Results indicate that the Nova Clean performed about as well as the Iradecon 210. The Mrad/hour level of the specimens cleaned with Nova Clean was reduced 40% (vs. 44% for Iradecon 210). The d/m level of the specimen cleaned with Nova Clean was reduced 67% (vs. 70% for Iradecon 210).

A larger quantity of Nova Clean will be ordered for large-scale evaluations. It is almost as effective as Iradecon 210, but it may be a safer alternative.

Pallets Evaluated for Decontaminability

W.N. Rankin and D.F. Steedly

An investigation on decontaminating pallets was completed. Presently, wooden pallets are used in radiologically controlled areas (RCA). However, decontamination of wooden pallets for reuse in another RCA is impossible. An alternative pallet, which is easier to decontaminate, would minimize the amount of waste generated from disposal.

Plastic pallets were evaluated as a possible alternative. The surface of three types of plastic pallets were characterized. One manufacturer claimed that the plastic surface was "impermeable for easy cleaning or sterilization". Although plastic pallets do not absorb water like wood, the surface contains many irregularities that entrap radioactive material. Therefore, making them impossible to decontaminate.

Plans are to obtain Type 304L stainless steel pallets with an electropolished surface finish, which has few surface irregularities where radioactive material can accumulate. This option may be more viable because the pallets pick up little contamination and are easy to clean. These pallets will be evaluated in contaminated areas.

Extended Sludge Processing Temperature Concerns

M. S. Hay and N. E. Bibler

A task team was formed to resolve concerns over temperature increases that may occur in Extended Sludge Processing (ESP) tanks due to heat generated by slurry pumps. The concerns arose after Tank 42H and 51H temperature data from sludge washing in 1987 showed temperatures rising by ~1.5°C for each day of slurry pump operation. It is unknown whether cooling coils were in used.

To determine the effect of tank temperature on ESP operations (e.g. number of washes required, total waste water generated, etc.) a spreadsheet model of the sludge washing process was developed in Microsoft Excel. The spreadsheet model is easy to use and a given scenario can be setup and run in minutes. Various scenarios were cal-

culated to illustrate the change in the number of washes and water usage at temperatures ranging from 40°C to 60°C and wash water batch sizes from 200 to 600 Kgal.

A SRTC computer model generated for determining temperatures in Tank 48 is being modified to model the ESP tanks. It predicts the expected temperatures during slurry pump operation. The spreadsheet model results indicate a need to provide data for validation of the ESP temperature model.

Ammonium Ion in the Feed to Late Washing

D. D. Walker

The ammonium ion concentration in the precipitate slurry to be sent to the Late Washing facility was estimated at 500 mg/L. The radiolysis of nitrate and nitrite ions in the presence of tetraphenylborate solids produces the ammonium ion. Radiolysis experiments were completed to show that the increase in ammonium ion is proportional to the radiation dose for about 150 Mrads (one year of tank farm storage). The formation rate decreases at higher doses and the ammonium ion concentration reaches a peak of 460 mg/L at about 250 Mrads. Agitation during storage in Tank 49 should help purge ammonia from the slurry and Late Washing will further reduce the concentration. This information will be used in the design of the ammonia scrubbers that will be added to the Defense Waste Processing Facility.

Cooling Coil Leak Detection and Repair for the Type III Waste Tanks

D. K. Appel, L. R. Park, W. T. Zollinger, G. W. Lewis, A. E. Smoak, I. D. Hill, and V. Cech

Equipment & Materials Technology proposed a concept to make use of several existing technologies to tackle the tough restraints in inspecting and repairing waste tank cooling coils. A development and testing program was initiated to determine the limitations of using a technique to deploy a "rabbit" in the 2" cooling coil and to transport eddy current (ET) coils to detect leaks in the pipe.

The rabbit and ET exam would be performed in a way to avoid the limitations of numerous bends in a cooling coil loop. The rabbit will maneuver through the pipe by water pressures generated by a positive displacement pump installed between the inlet and outlet of the loop. The ET signals from the coils on the rabbit would be converted into pressure pulses and transmitted through the water back to a receiver in the valve house outside the tank. When a leak is detected, a Nitinol "Memory Metal", sleeve will be delivered to the leak sight on the rabbit and allowed to expand to form a press fit patch inside the pipe.

A commercial ET instrument (MIZ 17) was imitated in miniature form. The circuits that generate and detect magnetic fields were miniaturized to fit in the rabbit, fortunately they produce the correct field patterns. However, optimization and circuit tuning is continuing. Martin Marietta performed an acoustic telemetry test in the test facility to determine the feasibility of the acoustic data transmission technique. Test results are not encouraging because of multiple harmonic interferences. The acoustic signal was easily transmitted through the pipe, but significant data processing will be required to filter the harmonics. Therefore, pursuit of the technique will be delayed until other options are considered.

The full scale mockup, the clear PVC pipe mockup, the Rabbit Launching System, the ET circuitry, and the rabbit positioning controls were fabricated and integration of these systems are in progress. However, preliminary tests in the clear PVC mockup are delayed because of vendor delays in delivering the positive displacement pump, but the time can be regained if it is delivered by October 20, 1992 (the new scheduled delivery date). The prototype rabbits, computerized pump controls, and PVC mockup are ready for testing when the pump arrives.

The task is approximately 70% complete and the final recommendation is due by December 15, 1992.

In-Tank Precipitation Filter Support

D. J. McCabe

In-Tank Precipitation (ITP) replaced a Mott sintered metal filter with a spare filter. The original filter was partially fouled with process water and exhibited a low cleanwater flux rate. Preparations are underway to test the gas flow rate to verify the extent of fouling.

The vendor determined the flow rate before shipment, therefore, filters can be used to compare the current state with pristine conditions. They were procured in 1989 before the manufacturer changed manufacturing techniques, therefore, the pore sizes are smaller and the walls are thicker than expected.

M. R. Poirier's calculations suggest that the smaller pore size and thicker walls will initially cause a 26% lower permeate flux (with 0.5 wt% slurry) than a new Mott filter. The clean water permeate flux rate was 61% lower for the ITP filters versus a new filter. Whether a new filter will retain the higher slurry flux for an extended duration is not known. The filter design was based on flux measurements with a pre-1989 filter with characteristics similar to the ITP filters. The filter design had a gas flow rate of 18 scfm/sq.ft. and the ITP filter gas flow rate is 16 scfm/sq.ft. (at 5 psi). A new Mott filter would be approximately 35 scfm/sq.ft. and would have a thinner wall.

Slurry Mix Evaporator Condensate Tank Waste Water Filtration

M. R. Poirier

Laboratory tests with Slurry Mix Evaporator Condensate Tank (SMECT) wastewater from the Integrated Defense Waste Processing Facility (DWPF) Melter System shows that waste stream will rapidly plug the cartridge filters DWPF intends to use.

Waste generated from the DWPF Cold Chemical Runs must be treated and disposed of in compliance with regulations and permits. Plans are to prefilter the waste water using cartridge filters before sending the water to one of several alter-

native treatment facilities (possibly the F- and H-Effluent Tritium Facility (ETF)).

Previous testing demonstrated waste water from the Off-Gas Condensate Tank (OGCT) would not adversely affect the ETF if effectively prefiltered. The testing also demonstrated that OGCT waste water would rapidly plug the DWPF cartridge filters.

The current test examines the effect of waste water from the SMECT. The sample was adjusted to 7.0 pH and filtered with a cartridge filter. The feed turbidity was more than 200 NTU. A 1.0 m Pall Profile and a 1.2 m CUNO polypropylene filter were used. Both filtered and plugged rapidly, demonstrating that cartridge filters are not an effective solution to prefilter the waste. OGCT would not adversely affect the ETF if prefiltered. The testing also demonstrated that OGCT waste water would rapidly plug the DWPF cartridge filters.

Demonstration of the Late Washing-Nitric Acid Flowsheet in the Integrated Defense Waste Processing Facility Melter System

J. A. Ritter

The Late Washing-Nitric Acid (LW-NA) flowsheet was demonstrated in the one-fifth scale Integrated Defense Waste Processing Facility Melter System (IDMS). It was developed to replace the Hydroxylamine Nitrate (HAN) flowsheet that mitigates the formation of ammonium nitrate in the Defense Waste Processing Facility (DWPF) offgas system. Subsequently, the Nitric Acid (NA) flowsheet was developed to replace the Formic Acid flowsheet to balance the redox of the melter feed (controlled by the amounts of nitrate and formate). The elimination of the nitrate in HAN and the increase in formate in the Late Wash (LW) flowsheet upset the redox of the melter feed. The NA flowsheet also exhibited a potential for reducing the total acid in the system, thereby, reducing the hydrogen generation. However, the mercury stripping capability of the LW-NA flowsheet should be proven on a large scale.

One objective of IDMS Run PX5 was to obtain a peak hydrogen generation rate, based on credible deviations from the nominal and maximum

operating conditions of the DWPF. These conditions included 21% excess nitric acid, 31% excess precipitate hydrolysis aqueous (PHA), 56% greater PHA addition/evaporation rate, and maximum levels of formic acid (0.3 M) and, copper (1,050 mg/L) in the PHA. The goal was not to exceed the peak hydrogen generation rate design basis, based on the HAN-Formic Acid flowsheet. Another objective was to demonstrate mercury stripping using the LW-NA flowsheet. Concern arose as to whether the design basis for mercury stripping could be achieved during the PHA addition/evaporation cycle.

To achieve the objectives, the IDMS facility was modified before PX5. The DWPF flowsheet requires the same PHA addition and evaporation rates. Previously, this was not possible because of a flow rate limitation in the condensate treatment facility. An intermittent addition rate of 6 gal/min was used, along with a continuous evaporation rate of 0.7 gal/min (neither of which was prototypic of DWPF (1.28 gal/min)). Moreover, bench scale experiments showed that the peak hydrogen generation rate increased with an increase in the PHA addition and evaporation rate. As a result, a 2,500 gal SMECT, which allowed the PHA addition and the evaporation rates to be continuous in PX5 (about 2 gal/min), was installed in the IDMS.

The peak hydrogen generation rate in PX5 was 0.64 lb per hour. This level corresponded to a 57% decrease compared to the design basis obtained from the HAN-Formic Acid flowsheet (1.5 lb per hour). The preliminary data from PX5 indicates that the residual mercury concentration in the sludge was 0.03 wt% (dry basis), far below the design basis of 0.45 wt%. These results show that the objectives for the LW-NA flowsheet were accomplished in PX5. Other objectives are being investigated.

Bounding Estimate of Defense Waste Processing Facility Mercury Emissions

R. A. Jacobs

Purges required for H₂ flammability control and verification of elevated Formic Acid Vent Condenser (FAVC) exit temperatures, due to NO_x reactions, lead to significant changes in Chemical Process Cell operating conditions. Mercury emis-

sions were updated based on the new operating requirements, Integrated Defense Waste Processing Facility (DWPF) Melter system (IDMS) experience, and development of an NO_x/FAVC model that predicts FAVC exit temperature. Using conservative assumptions and maximum purge rates, the maximum calculated mercury emission is approximately 130 lbs per year. A range of 100 to 120 lbs per year is conservatively predicted for other operating conditions. DWPF permitted mercury emissions are 175 lbs per year (0.02 lbs per hour annual average).

Material Balance Tables for the Defense Waste Processing Facility Cold Chemical Run—Test 1

A. S. Choi

Several major revisions were made to the Defense Waste Processing Facility (DWPF) flowsheet model to calculate the material balance for Test 1 of the DWPF Cold Chemical Runs (CCR).

First, the nitric acid flowsheet, which substitutes nitric acid for a portion of the formic acid required to treat the sludge feed was added to the model. Its main purpose is to alleviate the glass redox problem that arises from the elimination of hydroxylamine nitrate (HAN).

Second, the chemistry of precipitate hydrolysis in the Salt Processing Cell was revised to reflect laboratory- and pilot-scale test data, which was one of the main unresolved issues in the DWPF flowsheet modeling. The calculated distribution of organic products between the organic and aqueous streams closely conforms to the actual data.

Third, a new basis for the material balance was developed to ensure that the strategy of blending sludge, precipitate hydrolysis aqueous (PHA), and frit produces waste glass of an acceptable quality at the design production rate and the maximum waste oxide loading. Under the new basis, several important process parameters such as PHA-to-sludge mass ratio, weight percent frit in glass, and percent calcination are calculated iteratively between the model and the Product Composition Control System. For the CCR Test 1, the total sludge waste oxides content is 30%.

Additional revisions and the full material balance tables are described in WSRC-TR-90-408 (Rev. 1).

Statistical Analyses of the At-Line Benzene Monitor Study

T. B. Edwards

Statistical analyses in support of experiments on the calibration and use of the At-Line Benzene Monitor (ALBM) for the Defense Waste Processing Facility was completed. The Applied Statistics Group and Defense Waste Processing Technology personnel assisted in monitor evaluations and methods for calibration and use.

Many statistical analyses to support this evaluation effort were conducted. Factors affecting the calibration process, such as solution matrix and sample temperature, were determined. The process of preparing benzene standards was also investigated. Appropriate calibration curves that provided lower prediction limits for ALBM measurements corresponding to standard benzene concentrations were estimated. Results of the studies are documented in SCS-ASG-92066.

Recommendations for Paring WP-19 Sampler Test

C. P. Reeve

The original design for the WP-19 sampler test is prohibitively large. By eliminating duplications in the laboratory analyses, a design less than half the size of the original was developed.

The number of sample vials and subsequent analyses in the original WP-19 sampler test design on four Defense Waste Processing Facility tanks were large. A request was made to pare down the number of samples and/or analyses and detect a 3.3% bias between samplers. Single aliquots and preps, rather than double aliquots and preps, reduced the amount of work to less than half. More detailed recommendations are documented in SCS-ASG-92054.

The Defense Waste Processing Facility Remote Drying Oven

J. M. DeWent

The current remote drying oven units are too large to be transferred into the analytical cells through the cell's access port. Defense Waste Processing Facility-Technical (DWPF-T) requested that Defense Waste Processing Technology design a remote drying oven for the analytical cells.

Equipment & Materials Technology (E&MT) evaluated the laboratory oven market for a suitable replacement for the Precision Scientific STM40 oven used in the analytical cells. A Humboldt Manufacturing Company (Model H-30125) laboratory oven was procured and tested, but deemed insufficient. E&MT was unsuccessful in identifying a laboratory oven (including the STM40) that meets the dimension limits and heating capacity specified by DWPF-T. As a result, E&MT is evaluating the possibility of modifying a Fisher Scientific oven (Model 718F).

The Fisher Scientific oven controls are side mounted, which may facilitate easier modifications. If modification of the Fisher Scientific oven cannot be completed, DWPF-T and E&MT will discuss laboratory oven design options.

Defense Waste Processing Facility Laboratory Instrument Installation Task

J. M. DeWent

Equipment & Materials Technology completed design and delivery of the buret control system, which is undergoing Defense Waste Processing Facility (DWPF) inservice testing in the analytical cells. DWPF personnel are being trained and they offered several suggestions to improve operation of the buret.

Project requirements and punch list items were completed and DWPF-T is reviewing check prints and draft copies of software documentation. The job will be closed after assessment of Quality Assessment measures outlined in a Quality Assessment Report.

Transuranic Waste Recovery—Local Ventilation

D. B. Stefanko

The test plan for the Idaho National Engineering Laboratory BWID project is essentially complete. The project involves benchscale tests of a local exhaust ventilation hood for mitigating contamination spread. Particles more than 20 microns will be spread inside an exhaust hood using a mechanical dust generator. The exhaust hood has three sides and a top, leaving an opening on one side for an excavator bucket. A ventilation system will be connected to the hood to provide an inward airflow through the hood entrance for contaminate containment. A method for gently blowing airborne particles toward the back of the hood will also be included.

Tests will be run through the hood entrance at many air velocities. After each test, particle filters will be collected from air samples strategically located around the hood entrance and compared to the baseline case without airflow. The test results will supply information about the feasibility of local exhaust ventilation for controlling contaminated dust at the dig site during transuranium waste retrieval. The test plan will be issued in early November 1992.

Transuranic Drum Vent and Purge Equipment

L. Williams

Los Alamos National Laboratory (LANL) and Westinghouse Hanford personnel visited SRS on October 14 and 15, 1992 to discuss transuranium (TRU) waste activities and issues. Discussions focused primarily on the development of drum vent and purge equipment. This equipment is needed to characterize one percent of its retrievably stored TRU waste. The characterization data will be used to develop a data baseline for inclusion in the Waste Isolation Pilot Plant No-Migration Variance Petition.

Vent and purge equipment is also needed during the recovery and processing of the less than 0.5 curie TRU drums in the mid 1990s. This equipment will eliminate the risk of drum deflagration

from flammable gases. Development of a functional prototype, in conjunction with LANL, is scheduled for February 1994. Other discussions included SRS drum corrosion studies, Pu-238 storage, safety documentation, and waste retrieval plans.

E-Area Vault Presentation

C. A. Langton

DOE-SR requested that Waste Management (WM) participate in a technical exchange with Chem Nuclear. The discussion topics were the E-Area vault design objectives, structural details, and construction techniques. A tour of the E-Area vaults was also scheduled. C.A. Langton, of Interim Waste Technology, participated in the technical exchange and made a presentation on the performance criteria and design objectives of the SRS waste disposal vaults and compared the Z-Area and E-Area disposal systems. In addition, a detailed description of the development and testing of the E-Area concrete mix was provided. Lessons learned were summarized and changes considered for future vaults were provided.

Progress in the Synthesis and Testing of a Replacement Reagent for Cesium Precipitation

E. G. Orebaugh and C. M. King

The process for decontaminating soluble waste salts accumulated in the waste tanks uses sodium tetraphenylborate to precipitate cesium. Unfortunately, this reagent can not be directly fed to the glass melter because of its high carbon content, therefore, it necessitates numerous chemical and engineering processes to remove it. In an attempt to identify more desirable alternative precipitating agents, Professor Fanning (Clemson University), working under a SCUREF contract with Interim Waste Technology, suggested the use of cobalt dicarbollide anine (DBCCo).

The DBCCo cesium salt ($\text{Cs}\{[\pi\text{-(3)-1,2-B}_9\text{C}_2\text{H}_{11}]\text{Co}\}$) is insoluble. It contains little carbon and no benzene and may be capable of feeding directly to the glass melter without further

treatment (sodium and potassium salts are also soluble).

The Soviets used this unique reagent as an extractant in solvent extraction systems for many years, but never as a precipitating agent in cesium decontamination of alkaline wastes. Fanning's contract was extended to allow investigation at the solubility of this reagent in simulated SRS waste and to evaluate its suitability as an alternative to tetraphenylborate reagent.

The Clemson University group synthesized a small quantity of the reagent and verified the literature solubility in water. A shift in the analytical band in the colorimetry hampered the solubility measurements in SRS waste media. Continuing work will quantify the solubility product in highly alkaline nitrate media and document the chemical stability of the salt, especially regarding hydrogen evolution.

Halogenated reagent derivatives to have lower aqueous solubility than their cesium salts. The Clemson University group needs a continuing contract to evaluate the derivatives. If results are needed more quickly, funding for full-time personnel will be required.

Impact of New Debris Rule on the Hazardous Waste/Mixed Waste Project

C.A. Langton

Personnel from Interim Waste Technology and the EPD set up a meeting with Ebasco and Systems Engineering to update design engineers about the newly promulgated Environmental Protection Agency (EPA)/Resource Conservation and Recovery Act (RCRA) Debris Rule. The regulations significantly impact the treatments required for much of the hazardous and mixed waste at SRS, which can be classified as debris, rather than process waste. Debris treatments, in many cases, require less processing or no processing before encapsulation or stabilization. Consequently, it impacts the processes being designed (Title I) for the Hazardous Waste / Mixed Waste (HW/MW) treatment building and should result in process changes included in the HW/MW project.

Ebasco and Systems Engineering design engineers were also updated on a Soils Rule that EPA plans to issue in draft and finalize in 1993. Since soil treatment is included in the HW/MW project, the outcome of this rule could also impact the facility design.

Inductively Coupled Plasma-Mass Spectrometry

S. Wyrick

The Inductively Coupled Plasma-Mass Spectrometry (ICP-MS) instrument has been running efficiently for the past month. During this time, two sets of waste tank samples were analyzed for characterization and trending studies. The analyses were used to define fission products and some deviations from the theoretical fission yield curves. Two types of Interim Waste Technology samples were analyzed.

The first type was measured to determine solubilities of uranium and plutonium in ITP tank solutions before sodium titanate absorption. The second type consisted of sodium titanate digested in sulfuric acid after exposure to uranium and plutonium to determine absorption characteristics of sodium titanate.

Initial analysis of the samples, using ten percent sulfuric acid, plugged the sample cones of the ICP-MS. However, the instrument could tolerate a one percent sulfuric acid matrix. Samples were also analyzed to support the Savannah River Ecology Laboratory's studies in "Colloid-Facilitated Transport of Contaminants at the Savannah River Site".

On October 6 to 8, 1992, the author attended the Gatlinburg Conference on Analytical Chemistry to present a poster session on ICP-MS analysis of gadolinium in reactor moderator water. The conference was beneficial to meet people involved in ICP-MS work from other sites. A more detailed trip report will be issued at a later date.

Statistical Treatment of Insoluble Solids Determinations for In-Tank Precipitation

T.B. Edwards, D. D. Walker, and C. J. Coleman

A statistical study was performed on the two methods to measure the weight percent insoluble solids in potassium tetraphenylborate (KTPB) slurries produced in the in-tank precipitation (ITP). Weight percent insoluble solid measurements are needed to assess the homogeneity of the tetraphenylborate (TPB) precipitate and the performance of the tank sampling system. Walker used the classical gravimetric method to analyze and prepare KTPB standards. Coleman also measured the total solids and soluble solids in the slurry, took the difference between the values, and analyzed the standards for weight percent insoluble solids. The determinations were performed according to Edwards' plan that is designed to randomize errors. The results, according to the statistical treatment of the data, are as follows:

- No evidence of bias in the preparation of the standards existed.
- The classical method for KTPB determinations produced excellent accuracy and precision.
- The method for measuring weight percent insoluble solids, by taking the difference between total solids and soluble solids, showed no evidence of absolute bias. However, the method is less precise than the classical method. A 95% prediction limit for the uncertainty of an individual analytical measurement is about 0.93 weight percent of samples containing 1 to 10 weight percent insoluble solids.

ITP Operations is preparing a report summarizing the results of this study.

The High Performance Liquid Chromatography Lab Expands Services

M. L. Hutchens

Last month, Analytical Development Section acquired the ability to perform organic analysis by High Performance Liquid Chromatography. The analyses was performed for three compounds. However, this month we expanded our capabilities to include routine analysis for 14

compounds. The instrument is capable of analyzing a wide variety of organic compounds.

The lab is also acquiring two new instruments that will allow overnight operation to help meet the sample load from Defense Waste Processing Technology.

Statistical Quality Control

K. L. Shanahan

The Defense Waste Processing Facility's Analytical Facility intends to use Statistical Quality Control technology to aid in lab quality efforts. A project that allows automatic quality control sample result transmittal from the Laboratory Information Management System (LIMS) to RS/1 for statistical analysis on a real-time basis began. RS/1 will send the analysis result back and a lab specialist will view it via LIMS.

High-Level Liquid Waste Computer Model Development

M. S. Hay, T. Hang, and K. L. Shanahan

A development plan was initiated for the creation of a dynamic High-Level Liquid Waste computer model that enables examination of future operating scenarios with a high credibility level in a short-time frame. The scope of the model evolved as other interested parties became involved in the development effort. A task team, lead by D. Harrel, focuses on defining DOE and SRS needs and expectations of the model. At the same time, an SRTC task team is reinvestigating development options in light of new model requirements. When the investigation is completed and documented, a new development plan will be drafted. The development plan will include a graded approach to the development effort to provide the customer with short term deliverables that will be immediately useful and serve as an additive to subsequent stages.

Procurement of the SPEEDUP modeling package (the dynamic modeling software for implementation of the model) is proceeding and SRTC and Waste Management upper management expressed interest in updating and integrating other site models into the SPEEDUP framework.

Eventually, this could lead to a sitewide waste management model with stand alone modules of each facility that could be accessed separately for more narrowly focused interests or in combinations for broader problems.

Materials Testing for Replacement High Level Waste Evaporator

G. T. Chandler

Successive evaporation runs of synthetic Purex and H-modified waste solutions were performed to determine the effect of increased temperatures on the operating conditions of the Replacement High Level Waste Evaporator (RHLWE). Condensed solutions under both operating conditions were cooled after each run to allow precipitation of solids. Under current operating conditions, two evaporation runs were performed on the Purex solution to obtain a solution with a specific gravity of 1.5. The solution hydroxide concentration increased from 1.3 to 11.5 M and the NaNO_3 concentration decreased from 2.9 to 1.1 M. The initial boiling point of the solution was 107°C and the boiling point after the second evaporation was 146°C. Under RHLWE operating conditions, three evaporation runs were performed on the Purex solution to obtain a solution with a specific gravity of 1.6. The solution hydroxide concentration increased from 1.3 to 15.1 M and the NaNO_3 concentration decreased from 2.9 to 1.1 M. The initial boiling point of the solution was 107°C and the boiling point after the second evaporation was 146°C. Solutions from the H-Modified runs were submitted to the ADS for analysis.

Constant extension rate tensile (CERT) tests in simulated waste solutions are underway to evaluate the susceptibility of Hastelloy G-3 to stress corrosion cracking (SCC) under RHLWE conditions. Baseline CERT tests with Type 304L stainless steel in 45 wt% NaOH at temperatures above and below the apparent SCC boundary are being performed to demonstrate the viability of the CERT method as an indicator of SCC. CERT tests were completed for Type 304L stainless steel in 45 wt% NaOH at 100°C (below the SCC boundary) and at boiling at 140°C (above the SCC boundary). The fracture surfaces of the test specimens are being examined for indications of SCC. Hastelloy G-3 and Type 304L stainless steel

simulated H-Modified waste immersion tests of are also underway. They are aimed at determining general corrosion rates under RHLWE conditions.

Failure Strain Criteria for the Type III In-Tank Precipitation Waste Tanks

J. K. Thomas and G. E. Mertz

Waste Management requested that an evaluation of the effects of multiaxial loading on failure strain be conducted. This evaluation supports development of a strain-based failure criteria for the Type IIIA waste tank primary and secondary liners; these criteria will be used with an ongoing structural analysis of the tank under hypothesized deflagration loadings. A technical memorandum (SRT-MTS-92-1181) documenting these results was issued.

The triaxiality factor (TF) approach was also employed in this evaluation. The TF is defined as three times the hydrostatic mean stress divided by the Mises equivalent stress. It was empirically determined that the failure strain under a multiaxial loading is the uniaxial failure strain divided by the TF. Thus, 2 TF indicates a failure strain one-half the value obtained from a uniaxial tensile test. Westinghouse Hanford employed similar approaches in the analysis of the response of their waste tanks to deflagration loadings. The Electric Power Research Institute also recommended the approaches for analysis of reactor containment structures to overpressure loadings. For a deflagration accident in a waste tank, the global failure mode would be tensile instability, therefore, the uniform strain is employed as the failure strain.

The Scientific Computational Section's (SCS) Computational Modeling Group performed a finite element analysis of a Type III waste tank for a deflagration loading and provided the primary and secondary liner stresses required to compute the TF with the corresponding strains. The current best-estimate and upper-bound pressures estimated for a deflagration event in the waste tanks are 45 and 74 psig, respectively. The evaluation results show that the strain at failure, for some liner locations, is reduced by almost a factor of 2, relative to that from a uniaxial tensile test. Based on the SCS liner strain cal-

culations and corresponding TF, the primary and secondary liners are predicted to survive the best-estimate deflagration loading (45 psig). At the upper-bound deflagration loading (74 psig), the primary liner is predicted to fail at the inner knuckle and the secondary liner is expected to survive. These conclusions do not consider failure strain reductions (knockdown factors) to account for sophistication or adequacy of the finite element structural model, design confidence, or materials considerations; these factors will be considered in a Systems Engineering report.

High-Level Waste Tank Inspection Guidelines

D. M. Barnes

On August 3 and 4, 1992, a meeting was held in Boston, Massachusetts, to discuss inspection guidelines for High-Level Waste (HLW) tanks as part of the DOE Tank Task Force. The meeting focused on the development of in-service inspection guidelines that HLW tank sites should use in the future. The goal of the panel is to establish guidelines (areas subject to inspection, responsibilities, provisions for accessibility and inspectability, examination methods, frequency of inspections, repair requirements, etc.) with active participation and support by various sites. The sites were requested to perform the following tasks:

- prepare a list of inspection programs at the tank farms and provide a brief description on each program
- identify the difficulties and challenges encountered in maintaining the tanks' structural integrity and performing inspections
- provide inputs that identify the sites' perception of what the guidelines should include

A response to panel recommendations was prepared by the SRTC, SS-Q/NDE and Waste Management. The next workshop is scheduled for October 29 and 30, 1992 in Richland, Washington, to discuss the input from various sites.

Failure Analysis of Effluent Treatment Facility Heat Exchanger Fans

R. L. Bickford and T. E. Skidmore

The Materials Technology Section (MTS) performed a failure analysis on two heat exchanger fans for the F- and H-Area Effluent Treatment Facility (ETF).

ETF requested that MTS perform a failure analysis on two fans used to cool heat exchangers. The fans failed within 36 hours of each other, for different reasons. Visual examination of the first fan led to the conclusion that at least one of the air lines powering the fan loosened, causing excess weight on one side of the rotary union at the top of the fan shaft. This uneven weight caused a corresponding uneven wear on the bottom of the union, eventually leading to catastrophic failure, causing it to drop into the fan blades below. Metallurgical analysis of the failed parts supported this conclusion, as a result, MTS recommended that support braces be used on both air lines and vibrational analyses be performed on the shafts.

The failure of the second fan appeared to be caused by a frozen bearing that rubbed against the shaft, causing a substantial decrease in the shaft diameter. This shaft was not available for metallurgical analysis because it was repaired for use as a spare part.

Impact of Defense Waste Processing Facility Flowsheet Modifications on Materials Performance

P. E. Zapp and B. J. Wiersma

Proposed flowsheet modifications for the Slurry Receipt and Adjustment Tank (SRAT) and the Slurry Mix Evaporator Condensate Tank (SMECT) that mitigate hydrogen and ammonium nitrate problems in the Defense Waste Processing Facility (DWPF) will expose the tanks to solutions acidified with nitric acid. The pH range in modified solutions will be 1 to 3. The impact of the modifications on alloy C-276 and Type 316L stainless steel and the SRAT construction materials and the SMECT was evaluated a survey of the DWPF corrosion database (for alloy C-276) and laboratory testing on simulated tank solutions.

Alloy C-276 limited corrosion resistance to high concentrations (more than 20%) of boiling nitric acid. The manufacturer's data shows that the corrosion rate may exceed 10 mils per year (mpy, where 1 mil = 0.001 in.) in ten percent boiling nitric acid. DuPont Company conducted studies of alloy C-276 at pHs in the range of interest in a variety of DWPF original flowsheet solutions. Coupon immersion tests to determine the uniform corrosion rate and reveal potential localized corrosion (crevice and pitting) were run. In the solution similar to the new SRAT solution, the corrosion rate of alloy C-276 at 90°C was 3.3 mpy and 4.0 mpy, measured on flat and flat welded coupons, respectively. Localized corrosion was not observed.

Cyclic potentiodynamic polarization (CPP) scans were performed on alloy C-276 to determine its susceptibility to localized corrosion in nonradioactive simulants of nitric acid flowsheet conditions of the SRAT. The Defense Waste Processing Technology (DWPT) section provided the test solutions and represents PUREX and HM waste. CPP scans were run at room temperature and at 95°C in each SRAT solution. All scans revealed immunity from localized pitting corrosion.

Two electrochemical corrosion tests were performed to assess the performance of Type 316L stainless steel under exposure to an acidified SMECT solution. The solution was simulated with a dilute, modified-flowsheet SRAT solution. The dilute solution was acidified to 1 pH with nitric acid with one drop of mercury metal. CPP scans indicated immunity from pitting. The Type 316L stainless steel uniform corrosion rate was determined through linear polarization and found to be 0.050 mpy at 70°C.

The combined results from the DuPont Company corrosion tests and the electrochemical tests indicate that C-276 and Type 316L stainless steel will give satisfactory service in the modified SRAT and SMECT conditions.

Lexan Replacement Evaluation Program for the Defense Waste Processing Facility Electrical Connectors

T. E. Skidmore

Evaluation of candidate materials to replace Lexan polycarbonate, used as the insulating block material in the Defense Waste Processing Facility (DWPF) Westinghouse Hanford electrical connectors, is in progress.

DWPF observed cracking in the existing Lexan blocks, presumably because of chemical exposure during machining and/or operation. Although exposed to a wide variety of chemical fumes and radiation, the most severe conditions are assumed to be caused by exposure to aromatic hydrocarbons, particularly benzene. Using 40 wt% glass-filled Lexan polycarbonate as a reference composition, critical properties of several candidate materials were reviewed. These candidates include: Tefzel (ETFE), Thermocomp (polyetheretherketone), Ultrason E (polyethersulfone), Ultem (polyetherimide), Vespel (polyimide), Ryton (polyphenylene sulfide), and Torlon (polyamide-imide). The candidate materials have similar or superior mechanical, electrical, and radiation resistance properties, compared to the reference composition. However, the resistance to stress-cracking, as a result of benzene exposure, is not well defined for all materials and must be tested.

Test samples from each material and grade are being machined to contain the same hole pattern used in the existing 40-pin connector blocks. Each test sample will be stressed to the same level and exposed to benzene vapor for a specified period of time. Coupons of the same materials are to be immersed in benzene as well. Therefore, stress-cracking resistance and general chemical resistance will be fully evaluated. Based on these test results, an alternative material will be recommended. Completion of the test program is expected in December 1992.

Decomposition of Radioactive Tetraphenylborate Precipitates from In- Tank Precipitation

D. M. Ferrara, N. E. Bibler, and B. C. Ha

Radioactive tetraphenylborate (TPB) precipitate slurry from Tank 48 was successfully hydrolyzed in the SRTC Shielded Cells. This radioactive precipitate resulted from the successful demonstration of in-tank precipitation (ITP) in the SRS Tank Farm in April 1983. This process isolates cesium-137 from waste supernates to immobilize the cesium in borosilicate glass in Defense Waste Processing Facility (DWPF). The purpose of the hydrolysis is to separate the organics from the cesium-137 before sending it to the melter where it is immobilized in the glass. In 1991, a precipitate slurry sample was transferred from Tank 48 to SRTC. At SRTC, its composition was adjusted to the composition required from the proposed "Late Wash" process. The adjustments included reprecipitating the cesium-137 that was released to the supernate by radiolysis during the nine years of storage in Tank 48 and lowering the nitrite and hydroxide concentration in the slurry.

Two hydrolysis tests were performed. In each test, most of the cesium-137, potassium, and boron were solubilized during hydrolysis, indicating complete destruction of the TPB precipitate. The primary product of this hydrolysis are an aqueous solution of cesium-137, boron, potassium, and an organic solution, primarily benzene.

Most of the cesium remained in the precipitate hydrolysis aqueous (PHA) product (5.8 Ci/gal in the first test and 7.8 Ci/gal in the second). Benzene, which had less than 2×10^{-8} cesium-137 Ci/gal in the first test and 2.1×10^{-6} Ci/gal cesium-137 in the second, was distilled from the PHA. Mercury was also distilled from the PHA and was below our detection limits in the organic condensate.

High Performance Liquid Chromatography analyses of the PHA were performed to identify and measure organic compounds that may be present. Phenylboric acid is produced when the TPB is not completely hydrolyzed. Phenylboric acid could not be detected in the PHA, indicating that the TPB was hydrolyzed. Phenol and biphe-

nyl are radiolysis products of the TPB and are introduced into the reaction vessel with the precipitate feed. In addition, small quantities may be produced during hydrolysis. Low levels of phenol and biphenyl were present in the PHA.

No significant observable differences were observed between tests performed on the hydrolysis process with radioactive Tank 48 precipitate and studies with nonradioactive precipitates that were irradiated. In both cases, freshly precipitated TPB with small amounts of cesium-137 radiolysis floated and foamed when stirred. The irradiated precipitate settled quickly and did not foam. The precipitate hydrolysis aqueous (PHA) and organic solution produced from hydrolysis of Tank 48 radioactive precipitate slurry had the same overall appearance as the PHA and organic solution produced from nonradioactive precipitate.

Technical Bases for the Defense Waste Processing Facility's Glass Product Control Program

M. J. Plodinec

The SRTC Glass Technology Group and Defense Waste Processing Facility (DWPF) developed a Glass Product Control Program to ensure that the DWPF consistently produces an acceptable glass product. Before the DWPF can proceed to the Qualification Runs portion of the Startup Test Program, DOE and external reviewers must approve the program. SRTC completed a description of the technical bases that demonstrate that the program should be effective during production. DOE approved the document and released it for external review.

The DWPF Glass Product Control Program is primarily directed at controlling the chemical composition of the glass product. Several independent investigations concluded that glass composition is the only parameter affecting glass interactions with liquid water, which is under the direct control of the DWPF. The DWPF elected to control the composition of the glass through control of the composition of the melter feed, which will be exercised at the last feed preparation vessel before the melter, Slurry Mix Evaporator (SME).

The Glass Product Control Program consists of the following elements:

- qualification of waste, which required SRTC to develop frit formulations capable of immobilizing the SRS high-level waste. SRTC demonstrated that the waste types in the SRS Tank Farm can be made into glasses that will satisfy the product consistency specification.
- SME sampling and analysis, which relies on DWPFs capability to reliably take samples from process vessels for analysis. SRTC demonstrated that minimal errors are associated with sampling process vessels and SRTC analytical methods can be performed remotely.
- determination of acceptability of SME batches to produce glass, using the Product Composition Control System (PCCS)
- remediation of SME batches if it cannot be demonstrated that a SME batch will produce glass that satisfies the product consistency specification. The PCCS will be used to identify the strategy.
- The DWPF will verify that the glass product is acceptable through two forms of evidence. The predicted Product Consistency Test (PCT) results from the PCCS for each SME batch will be averaged and reported in the production records. The average chemical composition will also be used to develop a separate prediction of the PCT results
- confirm that the glass is acceptable through sampling. Samples taken, using the glass sampler, during production with the DWPF will be brought to SRTCs Shielded Cells Facility and subjected to the PCT. The experimental results will also be reported in the production records

Preliminary evaluations of the effectiveness of this program were carried out with actual SRS radioactive waste on a small-scale and simulated waste on an engineering-scale. Both types of tests demonstrated that the Glass Product Control Program will be effective in ensuring production of a glass that is consistently acceptable.

Development of Standard High-Level Waste Reference Glass

C. M. Jantzen

The DOE product consistency specification for Defense Waste Processing Facility (DWPF) glass requires comparison of the Product Consistency Test (PCT) results of production glass and benchmark glass in the DWPF Environmental Assessment (EA). SRTC developed and characterized an EA Glass Standard Reference Material for this purpose, the DWPF program, and other waste form producers. Samples were provided to West Valley, Westinghouse Hanford, and Nuclear Regulatory Commission support contractors.

The specified composition of the glass is based on the composition of the glass in the DWPF EA, and was designed to simulate this composition as closely as possible. For example, a range of Fe_{2+}/Fe was specified to match the reduced nature of the glass.

However, it was not always possible to exactly simulate the composition in the EA. The EA glass composition contained "other" constituents, such as NaNO_3 , zeolite, and coal, that break down during vitrification. The glass composition in the EA also included components, such as uranium, that hinder free distribution of the Standard Reference Material. The target composition was derived by converting the components that break down during vitrification to glass oxides (e.g. NaNO_3 to Na_2O), excluding uranium (SRTC work shows that uranium does not affect the durability of SRS waste glasses), and normalize the resulting all-oxide composition to 100%.

One thousand pounds of this material was procured from Corning, Inc. and are characterized as follows:

- SRTC and the manufacturer determined the chemical composition (including the iron redox ratio) of this material. SRTC results generally confirmed the manufacturer's results. The manufacturer's estimates of the uncertainties in the composition were also used.
- The homogeneity of the material was checked by analyses of samples from various locations in each drum sent to SRTC. Each sample was also analyzed by x-ray diffraction for crystalline phases. However, none were found. No

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discernible differences in the chemical composition or the iron redox ratio were found among the samples.

- PCT Standard value results were obtained from an internal "round robin" using this material. The purposes of this activity were to provide values of the mean PCT results for use in qualifying PCT data and to provide an initial set of data for control-charting the SRTC PCT results. Three researchers participated and two types of vessel materials were used.

Based on uniformity and conformance, the specified composition material was accepted for use as a Standard Reference Material. This glass will be used as an internal control in future testing involving the PCT, which will facilitate direct demonstrations of compliance with the DOE product consistency specification, especially, for testing conducted as part of the DWPF Startup Test Program.

General

Glovebox Docking Port

W. T. Zollinger, G. D. Teese, and J. A. Purcell

Equipment and Materials Technology developed prototype equipment for docking and undocking gloveboxes while maintaining containment for evaluation at Lawrence Livermore National Laboratory (LLNL) and Separations. Radioactive material will be passed between the boxes while docked and containment will be maintained during and after undocking the gloveboxes. This concept is also being pursued for Complex 21 work since it allows the processes to be modular, centralized, re-configurable, and easier to maintain.

Coordination with LLNL yielded three developmental paths. The first path facilitates port manufacturing using modified commercial equipment with doors at both ends of an airlock (supplied by LLNL). It is currently being tested and incorporated into a glovebox system at LLNL, using an IBM robot for demonstration. The second path continues development of a custom port with one door in the moveable module, as originally outlined. The drawings are still being finalized. A third concept was also developed. It is a 4" mockup constructed with available materials.

Waste Processing Operations Robotics Team

G. D. Teese, C. R. Ward, and L. Yarbrough

The Office of Technology Development (OTD) is funding a complex-wide Robotic Technology Development Program. Waste Processing Operations is one of several teams on which SRTC participates. The FY93 program for the team was developed during an intensive two-day meeting at Germantown, Maryland.

The program will concentrate on robotics for Mixed Waste processing. It culminates a large demonstration of Mixed Waste treatment pro-

cesses in TNX Area and it will accomplish several objectives:

- utilize waste handling installed hardware and minimize the costs associated with a large demonstration
- demonstrate the national laboratories' ability to transfer technology to production sites
- closely interface with the Waste Facility Operations efforts coordinated by SRTC.

The team viewed TNX as an ideal facility to demonstrate the broad range of automation and robotics that could be applied to waste processing.

SRTC will also participate in the liaison effort with the Mixed Waste Integrated Program and the Mixed Waste Treatment Project. Automation strategies will be developed and recommendations will be made. Automated material transport will also be investigated. OTD will provide \$280,000 for the demonstration, liaison effort, and material transport study.

SCUREF: Advanced Fracture Mechanics to Assess Complicated Piping Flaws

P. S. Lam, G. E. Mertz, and G. S. Bumgarner

The University of South Carolina completed the J-integral calculations for circumferential cracks under bending for SRS piping sizes and demonstrated crack stability analysis based on the extended Electric Power Research Institute (EPRI) estimation scheme. "J-integral of Circumferential Crack in Large Diameter Pipes," is based on the first part of the task. The J-integral and EPRI estimation scheme is under DOE review for presentation at the Second International Conference on Nuclear Engineering in 1993. The complete report, including the second part of the task (crack stability demonstration) is being prepared for SRTC.

Internet Connection

J. C. Jensen

A connection to the worldwide Internet was installed. A Cisco router is connected to the Southeastern Universities Research Association

network (SURAnet) via a T1 communication line between SRS and Greenville, South Carolina. Functional testing of the connection and critical services was completed. Several Los Alamos National Laboratory computers were connected, including the UNICOS Cray. File transfers that demonstrate capability and good performance were also performed.

Work is underway to connect the Internet router to the site network through a security controller, sometimes called a "firewall". The controller is part of a service purchased from the Advanced Network Systems Corporation. The controller will be initially configured to allow selected users to initiate connections and file transfers from the SRS site network. No incoming connections or file transfers are permitted. Electronic mail and network news services will be provided after gaining experience with the security controller.

Configuration Management Process Mapping Team

R. C. Edwards and R. A. Moyer

TMLA engineers participated in the sitewide process mapping team to collect and organize information on site configuration management practices. The mapping process and accompanying discussions took place in eight two-hour meetings. As a result, basic definitions were agreed upon and the overall task was reorganized into three subtasks, resulting in two additional task teams. The site Configuration Management (CM) group will incorporate the CM process map and definitions in a revised site CM 7E manual.

SRTC and Analytical Laboratories Technical Safety Appraisal

R. A. Moyer, P. B. Gerrard, and N. K. Savani

The Technical Safety Appraisal (TSA) that DOE conducted in SRTC and the Analytical Laboratories (AL) ended on October 29, 1992. Nuclear processes safety research (NPSR) was primarily responsible for the safety assessment (including safety documentation) function during this appraisal.

The evaluation grading system for each area was based on a scale from 1 to 3, with a "1" being superior performance (DOE will consider reduced level of oversight) and a "3" being acceptable performance (DOE will consider increased level of oversight). Overall, SRTC and the AL received a "good" rating on the appraisal.

Some comments regarding the Safety Assessment functional area at the closeout meeting were:

- The requirements of the Annual Safety Appraisal and Review System not being fulfilled in all cases.
- SRTC does not have DOE approved Safety Analysis Reports (SAR) and Operating Safety Requirements (OSR), otherwise, the program is in good shape. Excellent progress since Tiger Team with positive trend.
- The SAR upgrade program for SRTC and AL in good shape.

Eleven issues/concerns regarding the Safety Assessment function were identified. One issue/concern relates to safety documentation. It referred to the fact that "SRTC has no DOE-approved SAR/OSR". The remaining issues/concerns deal with: annual safety appraisal, SRTC Safety Review Committee, Operation Experience Review Program and SIRIM.

SRTC Safety Analysis Report

R. A. Moyer and N. K. Savani

Last month, the DOE and WSRC held two meetings to discuss what DOE-SR should do to expedite the approval of the Safety Analysis Report (SAR). DOE-SR requested information on WSRC Operations' review of the SAR and what assurances configuration envelopes SRTC operation. The requested information will be promptly provided to DOE.

Items of Interest

- The University of South Carolina and the Clemson Math Department invited J. Haselow to assist in developing a proposal for cooperative research between industry and universities. The proposal will be evaluated under a new DOE Nuclear Safety Facility program that aims to develop the competitiveness of under-funded research universities.
- B. Z. Cowart, V. C. Cobb, C. S. Teelon, and H. J. Hootman are inputting the disclosures files into the new computer system. Six hundred and twenty six of the 1332 disclosures were recorded.
- J. C. Corey attended the DOE Technology Integration Program Review and the WSRC Government Operations Contractor TT Meeting in Washington, D.C. from October 25 to 29, 1992.
- Solid Waste Management requested that E. Wilhite participate in a review of the SRS Solid Waste Management Department from September 29 to October 2, 1992. The review team consisted of Wilhite, two independent consultants, and two members from Westinghouse Hanford Company management. They interviewed Solid Waste Management personnel and reviewed pertinent documents. A report on the review is being prepared.
- L. Williams and M. Looper hosted a transuranic (TRU) waste technical exchange with Los Alamos National Laboratory and Hanford at SRS on October 14 and 15, 1992. The discussion included drum corrosion studies, Pu-238 storage, safety documentation, TRU retrieval plans, and drum vent and purge development.
- G. J. Hooker, P. E. O'Rourke, and R. Livingston visited the Westinghouse Diagnostics Business Unit in Orlando, Florida on October 22, 1992 to demonstrate the Fiber Optic Temperature Monitor, which may be useful for monitoring operating temperatures in electric generators. It was determined that placing optical fibers in the generator coils will be challenging. Westinghouse Technical Staff will determine whether to pursue this issue further.
- E. Blahut of the Great Lakes Industrial Technology Center visited University of South Carolina-Aiken on October 19, 1992 and the WSRC TT Office on October 20, 1992 to study the Summer Institute for TT. She is planning a similar program to evaluate NASA technologies at her facility.
- Heritage Environmental personnel met with C. S. Teelon. Subsequently, two licenses were filed.

Presentations

- A. Yu and B. Hiergesell attended a workshop on groundwater modeling for environmental applications and PORFLOW users group meeting in September 1992. Yu presented, "PORFLOW Modeling of Nitrate Release from a Saltstone Vault".
- Environmental Sciences Section managers presented information on the SRTC Geology, Hydrology, Seismology, and Ecology programs to the Energy Research and Development Administration Universities consortium to investigate the potential for collaborative work. Professors from the University of Georgia, Georgia Institute of Technology, Georgia State University, and Clark University were present.
- A poster describing the existing M-Area Groundwater Pump and Treat System with enhancements, such as the demonstration of innovative technologies at the Integrated Demonstration site, was prepared in with Environmental Restoration personnel for presentation at the National Groundwater Association workshop on Pump and Treat in Las Vegas, Nevada.
- C. Murphy presented, "Recent Tests of Tritium Conversion to Oxide", to the Tritium Focus Group of LANLs Los Alamos Study Center.
- B. Looney presented, "Innovative Remediation Technologies Demonstrated at SRS", at the Environmental Protection Agency/DOE National Technology Initiative Meeting in Raleigh, North Carolina.
- B. Looney and D. Kaback presented, "SRS Integrated Demonstration Program", at the South Carolina Hazardous Waste Conference in Columbia, South Carolina. The annual conference advertises the results of research funded by a governor implemented program that taxes hazardous and radioactive waste brought into the state of South Carolina.
- D. Stevenson presented, "SRS Seismic Network and Network Upgrade", to the Eastern Section of the Seismological Society of America in Richmond, Virginia.
- M. R. Buckner and members of the EMAC SPC/SPM User's Group led a breakout session at the Westinghouse Total Quality Symposium on October 5 and 6, 1992 in Baltimore, Maryland. The discussion topic was, "SPM-Do We Need a Mid-course Correction to Enhance Results?"

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