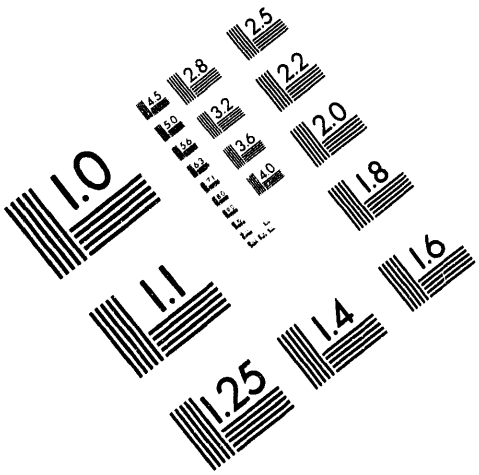

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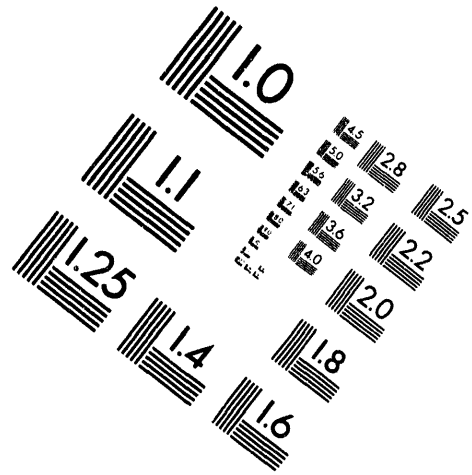

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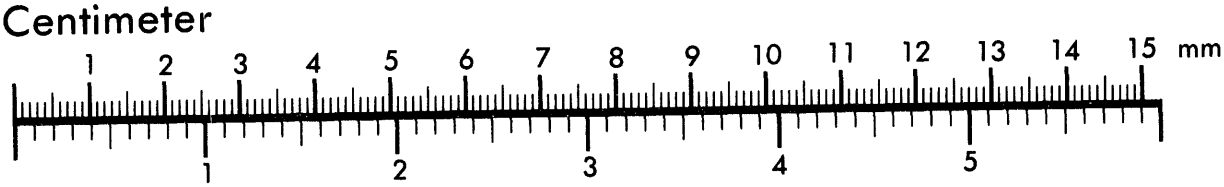

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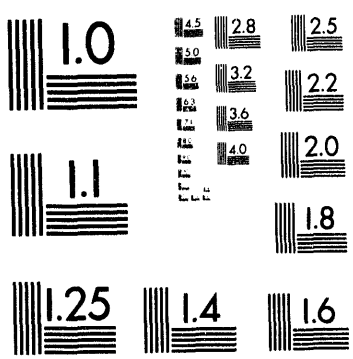

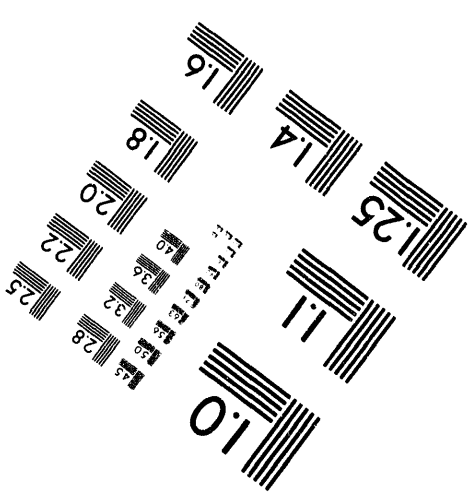

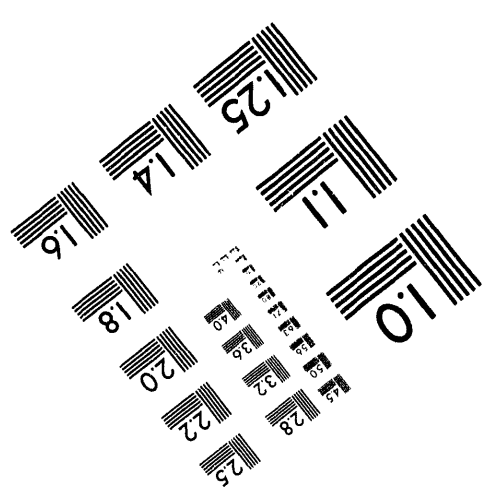

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APPLIED TECHNOLOGY SECTION MONTHLY REPORT MARCH 1994 (U)

by

Buckner, M. R.

Westinghouse Savannah River Company
Savannah River Site
Aiken, South Carolina 29808

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MASTER

SRT-ATS-940032

Westinghouse Savannah River Company
Inter-Office Memorandum

April 20, 1994

TO: FRED BERANEK, 773-A

FROM: M. R. BUCKNER, 773-42A
(5-3149)

MRB

Applied Technology Section

Monthly Report

MARCH 1994

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PACKAGING AND TRANSPORTATION
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Abstracts

ENGINEERING MODELING AND SIMULATION
(J. R. Pelfrey, Manager)

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Temperatures of Tank 17.1 Wasteforms - S. J. Hensel and J. W. Jerrell 5

Steady-state centerline temperatures were conservatively calculated for glass wasteforms produced in isotopic containment experiments with material from Tank 17.1 and documented in SRT-GTC-94-00027.

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A summary of work supporting activities in IWT is given. One project (IWT) has been completed.

APPLIED STATISTICS
(R. C. Tuckfield, Manager)

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Radionuclide concentration measurements in ground water samples collected from six background ground water monitoring wells in the F and H area Seepage Basins were statistically analyzed to characterize and tabulate the average concentrations of a suite of radionuclides as specified in the RCRA post closure care permit for these waste sites. In addition, SCDHEC requested a statistical evaluation of the water quality data for two of the background wells to determine if there was an impact on these wells from either the burial ground complex or the abandoned process sewer lines. The final report was submitted to Environmental Restoration (ER) in a format that could be submitted to SCDHEC in support of the RCRA Part B post closure care permit.

Statistical Design for the FA-13.01 Slurry Sampling Program - C.P. Reeve 7

In the FA-13.01 test to be conducted at DWPF, multiple SME batches will be processed through the MFT, the melter, and into canisters. The material will be sampled at each stage of the process. A report giving the statistical design for the vial sampling plan and chemical preparation and analysis has been issued.

APPLIED PHYSICS
(C. E. Apperson, Manager)

Computer Code Certification

J80 Criticality Codes Certification - D. Biswas, S. L. Lee, R. E. Grove, R. L. Reed, and J. H. Hightower (Engineering & Scientific Computing Section) 8

Work is continuing on the J80 Criticality Code User's Manual and code testing work of J80 versions of KOKO and KENOCJ.

Criticality Methods and Analysis

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In support of the DOE mission to return and store foreign research reactor (FRR) spent nuclear fuel (SNF), preliminary criticality studies have been performed by APG for several dry storage casks and configurations. Analyzed designs and loading for a 16-inch canister and NUHOMS canister with 24 sleeves appear to be acceptable from a criticality perspective.

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Validation of the TWODANT and XSDRN codes with cross sections based on the ENDF/B-V evaluation is underway for uranium 235 solution experiments.

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E-Area Vault Equivalency Relations - T. G. Williamson and J. F. Mincey 10
Equivalency relations for fissile and fissionable nuclides relative to uranium-235 for criticality safety have been developed with cross section available in the JOSHUA system. A report on this work, WSRC-TR-94-071, Rev. A, has been issued.

Plutonium Disposition Study

Plutonium Burner Support - R. W. Rathbun, D. Biswas and S. Y. Lee 10
WSRC is supporting Westinghouse Electric Corporation (WEC) efforts in evaluating the "Spent Fuel Option" of the Plutonium Disposition Study. Present efforts, designated as Phase Id, will continue to the end of April 1994. All physics calculations are currently being performed on a rented HP-735 Workstation computing system. These calculations have concentrated on defining operating parameters for the PDR600 and PDR 1400 reactor designs. Also, physics calculations are ongoing for a test assembly in an existing reactor. By mid-April WSRC will provide WEC a draft report of all efforts during Phase Id. During mid-July all task participants will meet with DOE at Piney Point, Maryland for formal presentations.

Training

In-Field Criticality Safety Engineer Training - T. G. Williamson 11
The In-Field Criticality Safety Engineer Training continued with one module on discrete ordinates computations.

EXPERIMENTAL THERMAL HYDRAULICS
(D. R. Muhlbaier, Manager)

Spent Fuel Dry Storage Development-Separate Effects, Rig Design, Assembly, and Test - M. D. Fowley 12
A separate effects test is being fabricated with a Mockup MTR-type fuel installed in a close fitting can. The test will measure the performance of mockup heater under various storage conditions.

FRR Spent Fuel Dry Storage Development - H. N. Guerrero 12
The Department of Energy has a program to develop dry storage as a means of storing Foreign Research Reactor (FRR) spent fuel in the interim period prior to acceptance of the fuel in a final repository. The objective of the ETF portion of the program is to evaluate commercial dry fuel storage canister designs and obtain critical data to establish the technical basis for dry storage of FRR spent fuel.

Pilot-Scale Demonstration of HEME Dissolution Process - Zafar H. Qureshi 13
The High Efficiency Mist Eliminators (HEME) used in the melter feed preparation processes will be dissolved in a caustic solution for their ultimate disposal. A pilot-scale demonstration of this dissolution process will be conducted in the Thermal-Fluids Lab. to resolve some technical issues regarding dissolution and breakup of the fiberglass material. This task is scheduled to be complete by June 94.

Development of Liquid Level and Density Bubbler for DWPF Canyon Vessels - T. J. Steeper 14
An experimental rig has been designed to test several new canyon bubbler concepts for use in slurry/frit solutions.

Cross-Flow Filter Test - J. L. Steimke, C. M. Hart and S. E. Sponseller 14
A test is being run in the TFL to gain operating experience with cross-flow filter technology to support DWPF Late Wash activities.

Melter Feed Test - J. L. Steimke 15
Plans are being made for a test in the TFL of part of the Glass Melter Feed System.

PACKAGING AND TRANSPORTATION
(E. K. Opperman, Manager)

Packaging Impact Limiter Design - M. N. Van Alstine 16
Packaging and Transportation (P&T) has obtained development funding for an impact limiter system. DOE Transportation Management Division (EM-561) allocated Sandia National Laboratory, Albuquerque (SNLA) $200,000 to develop an impact limiter system under P&T task leadership.

Technical Review of CD-1 Cask for First Time Use - T. Kurt Houghtaling 16
First application of the WSRC 19Q manual to a First Time Use condition is nearly complete. An existing SRS shipping cask and proposed payload have been reviewed technically regarding the requirements of 10CFR71. Procedures for the technical review process will be developed from this prototype report.

Tritium Crucible Package Transport Design & Safety Assessment - P. S. Blanton and A. G. Eggers 17
Following Tritium's Line 3 Operational Readiness Review (ORR), hot targets that are presently held in various Reactor basins will be processed. Tentatively, it is believed that some forty four (44) spent melt crucibles will be moved over the next 5-6 years once processing begins. SRTC's Packaging & Transportation is providing Tritium with necessary design criteria for a package to transport processed spent melts to the ILT vault, and with documentation in the form of a Safety Assessment to support such a move.

Administrative

Awards and Honors

M. D. Wise received the Rothamsted Award for Excellence in Statistical Research from the Brigham Young University Department of Statistics. The Rothamsted Award is given "to graduate students who exemplify the work of [R. A.] Fisher at Rothamsted [Experimental Station]".

Personnel

John B. Justice, a Co-op Student who worked with Fitz Trumble for two consecutive sessions, terminated on March 25 to return to his studies at the University of Cincinnati.

Travel

Feb. 27-Mar.11	T. Kurt Houghtaling and P. S. Blanton attended a DOE-sponsored course on reviewing Safety Analysis Reports for Packagings in Berkeley, Calif. The two-week workshop covered all engineering disciplines that support radioactive material packaging and included both lecture and laboratory exercises. Credit for the workshop is based on achieving a passing grade on the eight-hour final examination.
March 8-11	M. R. Buckner attended the International Policy Forum: Management & Disposition of Nuclear Weapons Materials Conference in Leesburg, Virginia.
March 14-18	Poh-Sang Lam visited Intera Information Technologies, Inc. in Denver, Colo. with IWT personnel. The purpose was to discuss the application of their software ECLIPSE and its pre- and post-processors to the IWT radiological performance assessments.
March 17	WSRC personnel from the Packaging & Transportation Group (P&TG), Transportation, and Health Protection met with H&R Associates, Westinghouse Hanford, and our packaging consultant, Bob Jackson, for an independent review of the current WSRC 19Q Transportation Safety Manual approach to onsite radioactive materials packaging.
March 24	M. R. Buckner led the EMAC SPC/SPM User's Group meeting in Atlanta, Georgia. Buckner was recently appointed Chairman of the User's Group which includes members from each Westinghouse Business Unit.
March 24	R. E. Pevey. participated in Ph.D. orals for James Malo at the Georgia Institute of Technology, Atlanta, Georgia.
March 27-30	R. L. Reed participated in the Plutonium Vulnerability Assessment Working Group Kick-Off-Meeting in Gaithersburg, Md., as part of the Nuclear Material Disposition Project.

Presentations

March 3	"Criticality Assessment of Initial Operations at the Defense Waste Processing Facility" by B. C. Ha and T. G. Williamson, was presented at "WM '94" Waste Management Symposia, Tucson, Ariz.

APPLIED PHYSICS
(C. E. Apperson, Manager)

Computer Code Certification

J80 Criticality Codes Certification - D. Biswas, S. L. Lee, R. E. Grove, R. L. Reed, and J. H. Hightower (Engineering & Scientific Computing Section)

Customer: C. E. Apperson, Jr., APG

Abstract: Work is continuing on the J80 Criticality Code User's Manual and code testing work of J80 versions of KOKO and KENOCJ.

a) J80 Criticality Code User's Manual

The final version of the TGAL and TGAN sections in the J80 Code User's Manual has been completed. Review of the criticality code, KEFF, by the independent reviewer has been completed. The User's Manual update of the KEFF and TWOTRAN documents is in progress. Review of the ARRAY code has also been initiated.

b) KENO and KENOCJ

Results from formal testing of the J80 version of the KENO Monte Carlo neutron transport code are being documented in a draft test report.

Preliminary testing of the J80 version of the KENOCJ code has revealed some significant discrepancies in results between the J70 and J80 versions of the code for some computational options. Work is continuing on checking the generalized geometry routines of KENOCJ to resolve the discrepancies.

c) KENO Random Number Generator Routine

An independent review of the random number generator routine, RANMER, has been completed by Steve Harris. All complementary data testing results are satisfactory. Based on this, it has been concluded that RANMER is a good replacement for the current assembly language random number routine and no further testing is necessary.

d) TWOTRAN

The Software Test Report of the TWOTRAN criticality code has been completed, reviewed and signed off.

e) KOKO/KATISHA/POOHBAH

The testing of POOHBAH of the KOKO/KATISHA/POOBAH package has been initiated. Based on prior tests done by C. Henkel, it was discovered that slight differences occurred in some cross section values that were not within the acceptable tolerance level. There is some problem in accessing the JOSHUA dataset CRIT5202 in module GLASS, whereas. there is no problem when the dataset is accessed in module POOHBAH. The problem could be connected to the new system upgrade. Six KOKO test problems exercising the POOHBAH module have been run for both the J70 and J80 systems. Results are currently being compared.

Criticality Methods and Analysis

Criticality Support for FRR Spent Nuclear Fuel - E. F. Trumble

Customer: Economic Development Department

Abstract: In support of the DOE mission to return and store foreign research reactor (FRR) spent nuclear fuel (SNF), preliminary criticality studies have been performed by APG for several dry storage casks and configurations. Analyzed designs and loading for a 16-inch canister and NUHOMS canister with 24 sleeves appear to be acceptable from a criticality perspective.

Foreign research reactor fuel is typically highly enriched uranium in an uranium-aluminum alloy. Because of proliferation concerns, DOE plans to bring "home" all of the FRR SNF and store it in the US. The aluminum cladding on these fuels, however, make them less than suitable for long term (40 years) storage in wet basins (such as RBOF). Therefore, dry storage concepts are being put forward for the storage of this fuel until a repository can be completed.

To support this effort, several canisters and configurations for storing this FRR SNF have been analyzed for criticality safety. These configurations stem from designs that meet the heat transfer requirements for the project, and thus are subsequently undergoing a preliminary criticality safety analysis. The two designs analyzed consist of a 16 inch diameter canister which will contain 20 MTR (Material Test Reactor) elements and a 5 foot diameter cask which will hold 120 elements. These two designs, containing elements of up to 400 g U-235/assembly, were evaluated for criticality safety under varying conditions. The results of these preliminary analysis show that these canisters and their proposed spacings are acceptable as long as the fuel integrity and spacings are maintained.

Work is ongoing in support of refinements of these canisters and configurations, analyzing heavier loaded fuels (more U-235/assembly) and analyzing more extreme accident conditions.

Validation Studies

Code Validation for Uranium Solution Experiments - T. G. Williamson

Customer: SRTC Applied Physics

Abstract: Validation of the TWODANT and XSDRN codes with cross sections based on the ENDF/B-V evaluation is underway for uranium 235 solution experiments.

Validation of the TWODANT and XSDRN codes with the LAW44 cross sections from Oak Ridge National Laboratory which are based on the ENDF/B-V evaluation is underway for uranium 235 solution experiments. The experiment descriptions have been written and confirmed by repeating much of Hugh Clark's earlier work with HRXN-ANISN. The validation of the XSDRN and TWODANT codes for these experiments is almost complete.

ENGINEERING MODELING AND SIMULATION
(J. R. Pelfrey, Manager)

Engineering Modeling

Temperatures of Tank 17.1 Wasteforms - S. J. Hensel and J. W. Jerrell

Customer: DWPT

Abstract: Steady-state centerline temperatures were conservatively calculated for glass wasteforms produced in isotopic containment experiments with material from Tank 17.1 and documented in SRT-GTC-94-00027.

The wasteforms are cylinders of length 14.76 in. and 2.95 in. diameter. They are composed of glass within a 0.125 in. thick stainless steel sheath. The thermal conductivity of the glass was taken to be that of Pyrex glass. Natural convection to stagnant air from a vertical free standing cylinder was assumed as a boundary condition for the wasteform, and radiative heat transfer was neglected. The heat source in the wasteform was distributed uniformly, and the nominal decay energy was calculated from the specific activity, the decay energy and the concentration of the isotopes in the wasteform.

Analysis Support For IWT

Customer: M. G. Looper (LIWCIA009)

Abstract: A summary of work supporting activities in IWT is given. One project (IWT) has been completed.

1. Evaporation Rates from CIF Ash Tank

An analysis was performed to estimate evaporation rates from the CIF Ash Tank during operation. Using a control volume approach, mass and energy transfer were balanced in the control volume. These balance equations were then used to estimate the evaporation rates from the Ash Tank for various Ash Tank and Kiln temperatures. A supplemental calculation was performed by solving the one-dimensional heat conduction equation with radiant heat flux on the water surface exposed to the Kiln. From this, an expression for the evaporation rate was derived which agreed exactly with that derived by the material and energy balances.

APPLIED STATISTICS
(R. C. Tuckfield, Manager)

Analysis of Background Radionuclide Concentrations for the F and H Area Seepage Basins (U) - E. W. Jenkins, R. C. Tuckfield, J. H. Weber, and M. D. Wise

Customer: Environmental Restoration

Abstract: Radionuclide concentration measurements in ground water samples collected from six background ground water monitoring wells in the F and H area Seepage Basins were statistically analyzed to characterize and tabulate the average concentrations of a suite of radionuclides as specified in the RCRA post closure care permit for these waste sites. In addition, SCDHEC requested a statistical evaluation of the water quality data for two of the background wells to determine if there was an impact on these wells from either the burial ground complex or the abandoned process sewer lines. The final report was submitted to Environmental Restoration (ER) in a format that could be submitted to SCDHEC in support of the RCRA Part B post closure care permit.

Ground water samples collected from six background wells in the F and H area Seepage Basin from 1988 to 1993 were statistically analyzed to determine the average concentration for a required list of radionuclides in each of three aquifers in the two areas. Plots of the results over time, by laboratory, and method were done to identify trends, outlier values, large detection limits, and other problems with the data. After qualification of the data, the average concentration was estimated in each of the six wells and over all three aquifers in both F and H area for 25 radionuclides. The average concentration was compared with the 90th percentile of the radionuclide detection limit. All constituents were less than the comparison detection limits except for Americium-243 in the Barnwell/McBean aquifer in H-area, Plutonium-239/240 for the Barnwell/McBean aquifer in H area, and Radium-226 in the Congaree aquifer in F area and the Water Table in H area. The Americium-243 detection limit was 0.25 pCi/L with the average concentration in the Barnwell/McBean aquifer in H area of 0.61 pCi/L. However, few Americium-243 analyses were done for any of the wells. The detection limit for Plutonium-239/240 was 1.0 pCi/L with the H area Barnwell/McBean average of 1.05 pCi/L. The radium-226 detection limit was 1.0 pCi/L with the F area Congaree average of 1.02 pCi/L and the H area Water Table average of 1.16 pCi/L. In addition, SCDHEC requested a statistical evaluation of the water quality data from two of the background wells in F area to ensure these wells did not indicate an impact from either the burial ground complex or the abandoned process sewer line.

The evaluation included comparing the historical concentrations of these radionuclides with those reported in 1993 by means of statistical process control charts. The charts showed that nearly all quarterly concentration measurements were within the usual +3 s control limits for all constituents in both wells. In addition, between well comparisons were made for these two wells with corresponding background wells in H-area in the same aquifer by statistically comparing the proportion of the results above detection limits. There were no statistically significant differences between the two F-area background wells and the H-area background wells in the proportion of results reported above detection. Since there was no significant increase in proportion of results above detection and the 1993 results were not significantly greater than would be expected, there is no evidence of past or present contamination to the two F-area background wells in question.

Statistical Design for the FA-13.01 Slurry Sampling Program - C.P. Reeve

Customer: DWPT/DWPF

Abstract: In the FA-13.01 test to be conducted at DWPF, multiple SME batches will be processed through the MFT, the melter, and into canisters. The material will be sampled at each stage of the process. A report giving the statistical design for the vial sampling plan and chemical preparation and analysis has been issued.

Prior to the start of the FA-13.01 test the slurry in the Melter Feed Tank (MFT) at DWPF, which has been sampled as part of the WP-19 and FA-10.02 tests, will be pumped back into the Slurry Mix Evaporator (SME). The FA-13.01 test begins as this initial batch is pumped through the system (SME --> MFT --> melter --> canister. Three more SME batches (designated 3, 4, and 5) will follow sequentially and be processed in the same way, resulting in a total of about 20 canisters of glass.

As each batch passes through the system it will be sampled at each stage. Samples will be taken via the Hydragard sampler, and Low and High grab samplers. A report (SRT-ASG-940028) has been issued which gives the statistical design for the slurry sampling portion of the program, i.e., sampling from the SME and MFT. The design includes schemes for the collection and identification of the sample vials, and the sequencing of their vitrification, preparation, and analysis. A total of 240 vials will be taken for elemental analysis. Solids analyses will not be done. Elemental weight percents of Al, B, Fe, K, Li, and Si will be measured by the ICP instrument.

E-Area Vaults

E-Area Vault Equivalency Relations - T. G. Williamson and J. F. Mincey

Customer: A. Q. Goslen (SWE)

Abstract: Equivalency relations for fissile and fissionable nuclides relative to uranium-235 for criticality safety have been developed with cross section available in the JOSHUA system. A report on this work, WSRC-TR-94-071, Rev. A, has been issued.

Equivalency relations for fissile and fissionable nuclides relative to uranium-235 for criticality safety have been developed with Hansen Roach cross section available in the JOSHUA system. These equivalency relations are based on accepted safe mass values from the ANS/ANSI Standards. The study indicated a modification of equivalency values for some of the nuclides and the need for further study with other cross section sets. A report on this work, WSRC-TR-94-071, Rev. A, has been issued.

Plutonium Disposition Study

Plutonium Burner Support - R. W. Rathbun, D. Biswas and S. Y. Lee

Customer: Westinghouse Electric Corporation (WEC), NTD

Abstract: WSRC is supporting Westinghouse Electric Corporation (WEC) efforts in evaluating the "Spent Fuel Option" of the Plutonium Disposition Study. Present efforts, designated as Phase Id, will continue to the end of April 1994. All physics calculations are currently being performed on a rented HP-735 Workstation computing system. These calculations have concentrated on defining operating parameters for the PDR600 and PDR 1400 reactor designs. Also, physics calculations are ongoing for a test assembly in an existing reactor. By mid-April WSRC will provide WEC a draft report of all efforts during Phase Id. During mid-July all task participants will meet with DOE at Piney Point, Maryland for formal presentations.

All future core physics calculations will be performed with a rented HP-735 computer workstation at SRS. This platform installation was necessary since the WEC/Cray computer is no longer being supported. This HP system has been installed and been through a full operational check.

Core physics data concerning fuel performance and transient performance are being generated and sent to the appropriate WEC personnel. All depletion calculations are concentrating only on burning the fuel to 40,000 megawatt-days per metric tonne metal (MWD/MTM), hence achieving the defined "Spent Fuel Standard" for all discharged fuel assemblies. Calculations were performed for increasing integral fuel burnable absorber (IFBA) along with switching to zirconium cladding, in place of SS304 cladding, with results indicating that increasing the IFBA density from 3 to 4 mg/inch provides nearly equivalent fuel assembly reactivity. Because of this conclusion, along with recent decisions concerning the test assembly, all future calculations for the large, 1400 MWe plant will utilize zirconium based cladding. Calculations for the PDR600, for this phase, are maintaining the stainless steel cladding. Plutonium enrichment for either design is 6.6% plutonium.

Calculations have proceeded for the test assembly core physics predictions. A generic 2-Loop reactor has been modeled and fueled with the test assembly configuration. Preliminary results indicate that power peaking is within design limits for the proposed assembly. Each of the proposed test assemblies will contain 31 test rods at the 6.6% plutonium enrichment employed in the PDR600 and PDR1400 designs.

Results of this phase are being drafted and sent to WEC. All draft input will be provided by mid-April. Final input will be provided by April 22nd.

Phase Id will end the last week of April 1994, at which time a formal document is due by WEC to DOE. DOE is requesting that all participants meet in mid-July for formal presentations at the Piney Point, Maryland facility.

Training

In-Field Criticality Safety Engineer Training - T. G. Williamson

Customer: WSRC Criticality Community

Abstract: The In-Field Criticality Safety Engineer Training continued with one module on discrete ordinates computations.

The In-Field Criticality Safety Engineer Training continued with one meeting in March with a final module on discrete ordinates computations. Because of the pressure of other work and because of the impending reorganization of the site criticality functions, the beginning of the series of sessions on the Monte Carlo method is being delayed.

EXPERIMENTAL THERMAL HYDRAULICS
(D. R. Muhlbaier, Manager)

Spent Fuel Dry Storage Development-Separate Effects, Rig Design, Assembly, and Test - M. D. Fowley

Customer: Economic Development Division, Spent Fuels Programs.

Abstract: A separate effects test is being fabricated with a Mockup MTR-type fuel installed in a close fitting can. The test will measure the performance of mockup heater under various storage conditions.

The fabrication of the fuel sleeve has been completed. Assembly is approximately 80% complete. The Data Acquisition System (DAS) is approximately 80% complete. Testing will begin shortly.

The shop completed the last stages of the fabrication process 3/21/94. All parts were trial fitted and found to fit together nicely. Rework was required on one part to replace an incorrect service tube size. The duct heater was successfully tested for proper operation when it was installed.

The lower section of the flow loop (two blowers, duct heater, transition and offset ducts) has been assembled and insulated. The upper portion (test section, elbows, and straight ducts) will be completed once the test section has been assembled. The DAS has been assembled and calibrated. Some instrumentation has not yet been terminated due to the unassembled, upper portion of the flow loop. Testing should start in the middle of April.

FRR Spent Fuel Dry Storage Development - H. N. Guerrero

Customer: Economic Development Division, Spent Fuels Programs.

Abstract: The Department of Energy has a program to develop dry storage as a means of storing Foreign Research Reactor (FRR) spent fuel in the interim period prior to acceptance of the fuel in a final repository. The objective of the ETF portion of the program is to evaluate commercial dry fuel storage canister designs and obtain critical data to establish the technical basis for dry storage of FRR spent fuel.

Current Milestone Schedule:

Canister Test: A Program Review meeting was held to review plans for the integral canister test. The proposed test was to simulate a 3 ft. high section of a 16-inch diameter canister for the dry vault design. It would contain four MTR mockups inside a can. It was decided to test the canister in a once-through wind tunnel where the test canister is surrounded by unheated dummy tubes. This results in a test boundary condition where the external heat flux would be the natural one arising from the cross-flow of air around the canister. The heat flux distribution around the canister circumference would be measured by heat flux sensors and then the local heat transfer coefficient (calculated from the heat flux and the measured canister wall to bulk air temperature difference) is provided as an input to the computer models. This testing concept was adopted as a result of FIDAP modeling of the canister which included the effect of the double walls of fuel can and canister, a 1/8-inch thick helium gap, and a typical heat transfer coefficient distribution taken from the literature. The results showed that the locally varying heat flux coefficient has a significant effect on the individual fuel temperatures and that the internal heat transfer cannot be decoupled from the external heat transfer. Since the objective of the test is to benchmark computer models against a set of prototypic data, this test design would achieve this objective of closely simulating the actual temperature distribution inside the canister. The design inputs document for the canister test is presently in preparation.

Conceptual Design Studies: The FIDAP model for the NUHOMS canister design continues to be refined. It was observed from a local model of the MTR fuel and sleeve that radiation heat transfer inside the sleeve is a significant mechanism , much more so than convection. This means that the assumption of homogenizing the fuel sleeve/MTR fuel region with an effective thermal conductivity to simplify the global analysis of the multi-sleeve canister cannot be valid. This is because in conduction, heat from one side of the sleeve to the other side can then take place. In reality, the MTR fuel is hotter than the sleeve and heat flows only from the MTR fuel to the sleeve and then to the gas flowing between the sleeves. Future FIDAP modeling will require that radiation inside the fuel sleeve be modeled.

Single Fuel Element Test: All parts of the flow loop have been received and the components are now being assembled. The test mockup was installed inside its aluminum can or housing and initial vacuum testing is currently being done. Except for some minor fitup problems, no significant problems are anticipated.

Pilot-Scale Demonstration of HEME Dissolution Process - Zafar H. Qureshi

Customer: Defense Waste Processing Technology

Abstract: The High Efficiency Mist Eliminators (HEME) used in the melter feed preparation processes will be dissolved in a caustic solution for their ultimate disposal. A pilot-scale demonstration of this dissolution process will be conducted in the Thermal-Fluids Lab. to resolve some technical issues regarding dissolution and breakup of the fiberglass material. This task is scheduled to be complete by June 94.

During this month four dissolution tests were completed successfully. The first three tests were done using mockup # 3 (2" thick fiberglass) and the fourth test was conducted with mockup # 2 (1.25" thick fiberglass). All the tests took about 30 hours to complete. The preliminary results indicated that the filter dissolution is essentially a two step process. The first step is chemical one, that takes about 16-24 hours for the digestion of the fiberglass where the fibrous matrix is converted into a "packed snow" type lump of mass. The second step is mechanical one where the packed mass is slowly eroded by the air agitated solution. The air sparging method of agitating is marginally successful. The last test required very high sparging levels (double the plant design sparge rates) to mechanically breakdown the filter.

The facility will undergo some repairs/upgrades before resuming the next test series. These include new metal cover for the tank, relocation of the sight tube connections, a new sample port and replacement of the pump seal.

The next test series will start with a fresh batch of NaOH solution. Mockup #3 will be dissolved under continuous heating and sparging conditions. All the previous tests were done during the normal day time shift and the filter was left overnight in the warm solution without any heating or sparging. It is postulated that the overnight soaking of filter may have aided the gelling of the digested filter. Thus a continuous operation is needed to reduce the effect of overnight gelling of the digested filter.

Development of Liquid Level and Density Bubbler for DWPF Canyon Vessels - T. J. Steeper

Customer: Defense Waste Processing Technology (DWPT), SRTC

Abstract: An experimental rig has been designed to test several new canyon bubbler concepts for use in slurry/frit solutions.

Bubblers have been used for years in the canyons for liquid level measurements because of their simplicity and low cost. However, in tanks containing slurry and glass frit they plug so rapidly that a standard bubbler is not practical. A Holledge gauge liquid level measurement system was installed but is complex, easily damaged, and expensive. This study will evaluate some new concepts and materials which may solve the bubbler plugging problem and allow them to be used instead of the Holledge probes. The design of the test equipment is complete, and component fabrication is in progress in the 717-A machine shop.

Cross-Flow Filter Test - J. L. Steimke, C. M. Hart and S. E. Sponseller

Customer: DWPT

Abstract: A test is being run in the TFL to gain operating experience with cross-flow filter technology to support DWPF Late Wash activities.

Background: In preparation for delivery to the Defense Waste Processing Facility, slurry from Tank 49 will be washed in a process that uses a cross-flow filter in a vertical orientation. Most of the flow passes through the porous tubes of the filter in the axial direction. A small part of the flow is driven through the porous membrane of the tubes by a pressure difference. Filtrate collects in the shell around the porous tubes and is collected. To provide technical support the Thermal Fluids Laboratory (TFL) is testing a cross-flow filter in a flow loop that allows flow in either the forward direction (tube side to shell side) or reverse direction. Brief flow in the reverse direction, back-pulsing, is useful for dislodging filter cake, which increases production of filtrate.

Summary: A Mott cross-flow filter is now being tested in the test loop at the TFL. The first tests are shakedown tests using aqueous solutions in the loop rather than slurry. First the flow loop was filled with clean water with a trace of caustic to allow the magnetic flowmeters to read. The hydraulic resistance of the filter was 2.5 psid ft^2/gpm, about what Mott claims for the 0.5 micrometer pore filter. Next the flow loop was filled with a cleaning solution, one percent oxalic acid. Some two-month-old acid solution was reused because the laboratory has been unable to get rid of any waste and is rapidly filling with jugs of liquid. With the oxalic acid the filter resistance immediately increased to 25 psid ft^2/gpm, then decreased to 20. Finally, the filter was washed with one percent caustic solution. The resistance decreased back to the original 2.5. All resistances were adjusted to 37°C. The solution temperature was allowed to heat up from 22°C to 34° during the cleaning cycle. One possible cause for the increase in resistance with the oxalic acid was that the acid solution had grown bacteria that partially plugged the pores of the filter. D. McCabe suggested another explanation, that the oxalic acid reacted with traces of tetraphenylborate from a previous wash cycle, producing a tar which plugged pores.

Melter Feed Test - J. L. Steimke

Customer: DWPT

Abstract: Plans are being made for a test in the TFL of part of the Glass Melter Feed System.

Background: The Melter Feed Tank contains a slurry consisting of glass frit, sludge and water which is agitated and pumped to the Glass Melter. Both of the melter feed loops contain recycle piping, a strainer to reject lumps and smaller diameter piping connecting the strainer to the melter. Two problems have been observed during the last four months of testing of the melter feed system. First, samples taken from the melter feed tubes, particularly feed tube #2, are enriched in sludge and depleted in glass frit compared to other samples taken from the Melter Feed Tank. Second, under some conditions the feed flows become very erratic. At present there is no consensus on the causes for the two problems.

Summary: DWPT has accepted a proposal from the Experimental Thermal-Fluids Group to build a test station at the Thermal Fluids Laboratory (TFL). The test section would accurately mock up part of the Melter Feed Loop. The test loop would serve three purposes. First, it would be used to determine the causes for the two problems. Second, the loop could be used to test possible hardware improvements. Third, if future problems arose with the Melter Feed Loops the TFL loop could be used to quickly provide technical support. Planning has begun for the test.

PACKAGING AND TRANSPORTATION
(E. K. Opperman, Manager)

Packaging Impact Limiter Design - M. N. Van Alstine

Customer D. L. Hayes (Tritium Department)

Abstract: Packaging and Transportation (P&T) has obtained development funding for an impact limiter system. DOE Transportation Management Division (EM-561) allocated Sandia National Laboratory, Albuquerque (SNLA) $200,000 to develop an impact limiter system under P&T task leadership.

Tritium Department has expressed need for a modern bulk tritium packaging. P&T sought and has obtained $200,000 in development funding from the DOE EM-561 Transportation Management Division. Obtaining the funding was conditional upon having a cooperative effort among DOE sites. EM-561 accepted P&T's proposal to have P&T lead SNLA development of a conceptual impact limiter design.

SNLA is then developing an impact limiter based on P&T design requirements and EM-561 funding. The objective is a small packaging impact limiter that can be assessed independently from thermal and containment systems. The design is to be validated analytically and by testing, and is to be readily adapted to the anticipated modern tritium packaging.

P&T has initially tasked SNLA to define allowable g-loads and to identify materials that would satisfy the g-load functions. Based on this, P&T will make a decision on the g-load and materials which SNLA is to pursue. P&T and SNLA personnel met and discussed these objectives in later March.

Technical Review of CD-1 Cask for First Time Use - T. Kurt Houghtaling

Customer Robert A. Adams, Technical Programs

Abstract: First application of the WSRC 19Q manual to a First Time Use condition is nearly complete. An existing SRS shipping cask and proposed payload have been reviewed technically regarding the requirements of 10CFR71. Procedures for the technical review process will be developed from this prototype report.

The CD-1 cask is one of the SRS fleet of 70 ton shipping casks (EP-85s) formerly used for moving highly radioactive reactor materials and components. Project 5550, Cadmium Control and Safety Rod Disposal, will use the CD-1 cask to collect cadmium rods from SRTC and the reactor areas. The mixed waste will then be stored on a TRU Pad at the SRS burial grounds until the mixed waste vaults are available. A technical review of the package is required by the WSRC 19Q, Transportation Safety Manual, because the intended payload is not one for which the cask is approved and qualifies as a First Time Use condition. The technical review is the first such activity since approval of the 19Q manual and follows the format and content guidelines provided by Regulatory Guide 7.9. The review is 95% complete and describes the cask and payload relative to the current requirements of 10CFR71, which address eight major areas of evaluation (structure, thermodynamics, containment, shielding, criticality, operation, maintenance and quality assurance). This prototype technical review is the basis from which one or more procedures will be written to guide this activity in the future.

Tritium Crucible Package Transport Design & Safety Assessment - P. S. Blanton and A. G. Eggers

Customer: Tritium

Abstract: Following Tritium's Line 3 Operational Readiness Review (ORR), hot targets that are presently held in various Reactor basins will be processed. Tentatively, it is believed that some forty four (44) spent melt crucibles will be moved over the next 5-6 years once processing begins. SRTC's Packaging & Transportation is providing Tritium with necessary design criteria for a package to transport processed spent melts to the ILT vault, and with documentation in the form of a Safety Assessment to support such a move.

The Tritium Extraction Facility (Line 3) can temporarily store 12 processed crucibles before having to move the spent melts from their facility to the Burial Grounds. Previous plans for movement of the spent melts involved having an external RAM packaging agency design and acquire a DOT Type B transportation cask that could transport the crucibles from the Tritium facility to the ILT vault.

The present path forward will have Packaging & Transportation act as the cognizant packaging design agency in determining necessary design criteria to transport the spent melt in an alternate Type B package, per the 19Q TSM to the ILT vault. P&T and Tritium are presently developing the necessary crucible transport design requirements. Upon completion of these requirements, a preliminary package and shield/transport design, schedule and cost estimate proposal will be presented to Tritium. This proposal is expected to be completed by the first of July.

The proposed alternate Type B package design will be based on the present crucible overpack to provide containment and a transport shield similar to the existing coffin design to meet necessary shielding requirements.

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Lam, Poh-Sang	773-42A 178	5-5306
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	Edwards, T. B.	773-42A	125	5-5148
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	Bridgmon, Gean	786-A	102	5-2727
	Harris, H. L., Jr.	773-42A	179	5-7386
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