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Monthly Highlights Report

ORNL Projects for the NRC Office of Nuclear Regulatory Research

A. L. Lotts

October 1982

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-543-75,
40-550-75, 40-551-75, and 40-552-75

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Authors: A. L. Lotts, Program Director
V. B. Baylor, Technical Assistant

F. J. Homan, Manager, Engineering Technology Program
T. S. Kress, Manager, Accident Evaluation Program
G. F. Flanagan, Manager, Risk Analysis Program
D. N. Fry, Manager, Facility Operations Program
R. O. Chester, Manager, Health, Siting and Waste Management Program

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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ABSTRACT

Highlights of technical progress during September 1982 are presented for ORNL research projects for the Office of Nuclear Regulatory Research.

DIVISION OF ENGINEERING TECHNOLOGY

PROGRAM TITLE: Additional Requirements for Materials

PROJECT MANAGER: Randy K. Nanstad

ACTIVITY NUMBER: ORNL 41 11 54 30 3 (189 B0103)/NRC 10 19 01 01 2

TECHNICAL HIGHLIGHTS:

Ultrasonic Examination, Vessels (K. V. Cook, R. W. McClung)

The technical memorandum (TM) on scanning speed studies has been submitted to the Metals and Ceramics Division Reports Office for processing. The document identification numbers are ORNL/TM-8519 and NUREG/CR-2967. The title is *The Influence of Scanning Variables Upon Ultrasonic Response*.

R. W. McClung attended the ASME *Boiler and Pressure Vessel Code*, Section V, meetings held in New York (September 14–15). The item of most interest to this task is the rewrite of Articles 4 and 5 on ultrasonic examination. McClung (a working member of the Subgroup on Ultrasonic Testing) participated in the review of comments and the subsequent adjustments on the proposed reorganization of these articles (especially Article 5). Highlights of both the Subgroup meeting on Ultrasonic Testing and the Main Subcommittee on NDE of Section V were provided to A. Taboada, NRC Technical Monitor, and are available from our files.

Radiography (B. E. Foster)

On September 14, B. E. Foster participated in the meeting of the ASME *Boiler and Pressure Vessel Code*, Section V, Subgroup on Radiography (as a member). Highlights of the meeting were provided to A. Taboada, NRC Technical Monitor. Foster is a member of the task group reorganizing Article 2 of Section V with a target date for completion of January 1983. Other action was related to mottling of radiographs on steel castings and inquiries about (1) penetrameter and shim selection relative to weld and reinforcement thickness, and (2) required angles for exposures in double-wall film viewing. Additional discussion included circumferential weld radiography, penetrameter selection, and computerized imaging techniques, with no formal action.

Effect of Ferrite Content on Austenitic Welds (R. K. Nanstad, D. P. Edmonds, J. P. Strizak, T. L. Hebble)

A draft report, *Effects of Ferrite Content and Aging at 343°C on Fatigue and Impact Toughness of Type 308 Stainless Steel Weld Metals for LWR Applications*, by R. K. Nanstad, D. P. Edmonds, J. P. Strizak, and J. Fohl, has been prepared and is undergoing review. The report includes examination of as-welded and aged shielded metal-arc welds of type 308 stainless steel with nominal delta ferrite contents ranging from 1 to 15 FN.

A draft report, *An Analysis of Delta Ferrite Data from Production Stainless Steel Pipe Welds*, by T. L. Hebble, D. P. Edmonds, and D. A. Canonico, has been prepared and is undergoing review. The report describes analysis of data from a study to compare delta ferrite content as measured in the filler metal weld qualification pad (QW) with that in the resultant production weld (PW). A collection of 1449 paired ferrite measurements (QW and PW) with types 308, 308L, 316, and 316L stainless steel welds were analyzed to determine the necessity for ferrite measurements of the production welds required by Regulatory Guide 1.31 (Revision 1). This report describes analysis of the data and conclusions that were drawn.

Effect of Poor Practice During Half-Bead Weld Repair (D. O. Hobson, R. K. Nanstad)

A draft report, *Effects of Off-Specification Procedures on the Mechanical Properties of Half-Bead Weld Repairs*, by D. O. Hobson and R. K. Nanstad, has been prepared and is undergoing review. The report compares test results of specimens removed from the heat-affected zone of "poor practice" welds with companion specimens from a prototypical half-bead weld repair of an intermediate test vessel of the Heavy-Section Steel Technology Program.

Underclad Cracking (R. K. Nanstad, R. L. Swain)

The section of HSST Intermediate Test Vessel 4 (ITV-4), which was clad by a commercial fabricator using techniques with a high probability for producing underclad cracking, is being prepared for sectioning into Charpy V-notch (CVN) and precracked Charpy (PCVN) specimens. The specimens will be removed from various locations within the heat-affected zone (HAZ) as well as from the base metal (A508 class 2). The ITV-4 is known to be susceptible to the underclad cracking phenomenon and it provides a good source of test material for this project. The regions of overlapping beads in the HAZ will be emphasized because it is in those regions that cracking is most usually observed; although our examination of this particular clad piece did not reveal cracks. The objective of this program is to evaluate the fracture toughness of the HAZ beneath the clad in a material susceptible to underclad cracking to allow for a more quantitative evaluation of structural integrity in components fabricated with materials of similar susceptibility.

Irradiation Effects on Charpy Upper Shelf (R. G. Berggren)

Several meetings of the Metal Properties Council (MPC) Task Group on Charpy Upper-Shelf Behavior of Irradiated Materials have been held in the past few months and progress has been made in defining the directions of the project and selection of personnel to provide analytical support. A letter report is being prepared regarding the status of this task and will be sent to the NRC during October.

MEETINGS AND TRIPS:

R. G. Berggren attended the Metal Properties Council Task Group meetings on Charpy Upper-Shelf Behavior of Irradiated Materials on September 1 and also on September 29 in New York City.

R. K. Nanstad and W. R. Corwin attended a meeting of the Metal Properties Council Task Group on Nuclear Power Plant Materials on September 9 in Washington, D.C.

Personnel attended the ASME *Boiler and Pressure Vessel Code* meetings in New York on September 13–15. These included B. E. Foster in the Subgroup on Radiographic Testing (as a member), R. W. McClung in the Subgroup on Ultrasonic Testing (as a member) and the Subcommittee on NDE (as a visitor), and R. K. Nanstad in the Subgroup on Toughness and the Subcommittee on Properties.

R. K. Nanstad also attended a meeting of the Subcommittee on Thermal and Mechanical Effects of the Pressure Vessel Research Committee on September 29 in New York City.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: ASME Code Section III — Technical Suppo~t

PROGRAM MANAGER: G. T. Yahr

ACTIVITY NUMBER: ORNL 41 88 55 05 1 (189 B0474)/NRC 60 19 21 00

TECHNICAL HIGHLIGHTS:

Task 1: Dynamic Allowable Stresses — A test fixture for the study of allowable dynamic ,ad criteria as defined in Section III of the ASME Boiler and Pressure Vessel Code is being designed at the University of Akron under a PVRC subcontract that is funded by this project. The design is scheduled to be completed Oct. 22, 1982, with testing scheduled to begin in February 1983. Initially dynamic stresses will be superposed on pressure induced static stresses in a 3-in. stainless steel pipe by dropping the test device. A concentrated weight may be added to the center of the pipe in order to change the dynamic characteristics of the system.

PVRC also has a one year old project on the "Development of Dynamic Stress Criteria for Design of Piping Systems" under R. D. Campbell of Structural Mechanics Associates (SMA). This work includes a technical review of existing information on elastic-plastic response to dynamic loads on piping systems, the development of a potential design criteria based on the concept of dynamic-to-static safety margin, the analysis of several very simple analytical piping system models, and the development of a research program plan. We will continue to closely monitor this effort through our association with the PVRC Technical Committee on Piping Systems, and the PVRC Subcommittee on Dynamic Analysis of Pressure Components and the Subcommittee on Piping, Pumps, and Valves.

Task 2: Piping Support Reactions — Background material has been assembled at E. C. Rodabaugh Associates Inc. for a report that will identify the sources of uncertainties associated with piping support restraint loads and quantify them to the extent feasible.

Task 3: Fatigue Evaluation for Class 2 and 3 Piping Components — Preliminary comments were given to E. C. Rodabaugh on his draft report, *Comparisons of Code Fatigue Evaluation Methods for Class 1 Piping with Class 2 or 3 Piping*, and it was transmitted to the NRC project monitor for review. Approval was obtained from S. W. Taggart for inclusion of his previously unpublished comments on the basis for paragraph NB-3228.3 of ASME Section III simplified elastic-plastic analysis in Appendix A of this report.

Task 5: Preloading of Bolted Connections — The Bolted Joint Seminar presented by Raymond Bolting Services, Inc., was attended. A step-by-step design procedure was presented for determining the maximum and minimum preload required in a bolt. A method was also presented for estimating the maximum and minimum preload that actually exists in a tightened bolt. In a properly designed bolted joint, the actual preload range must fall within the required preload range.

Task 7: Evaluation of Section III Acceptance Standards and Fatigue Curves Using a Fracture Mechanics Approach — Comments were received from NRC on our draft proposal for evaluation of Section III acceptance standards and fatigue curves using a fracture mechanics approach. Final agreement on the work scope has not been reached.

MEETINGS AND TRIPS:

S. E. Moore attended the ASME Boiler and Pressure Vessel Code Working Group on Piping (SC D/SC III) meeting in New York City on September 13, 1982.

G. T. Yahr attended the ASME Boiler and Pressure Vessel Code Subgroup on Fatigue Strength (SC P) meeting in New York City on September 13, 1982.

G. T. Yahr attended the Raymond Bolting Seminar in Middletown, CT, on September 14-15, 1982.

G. T. Yahr attended the PVRC's Subcommittee on Dynamic Analysis of Pressure Component meeting in New York City on September 27, 1982.

S. E. Moore attended the PVRC meeting in New York City on September 27-30, 1982. Meetings attended included the Subcommittee on Reinforced Openings and External Loadings, the Subcommittee on Piping, Pumps, and Valves, the Technical Committee on Piping Systems, the Design Division, and the PVRC Main Committee.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Containment Leak Rate Testing

PROGRAM MANAGER: D. J. Naus

ACTIVITY NUMBER: ORNL #41 89 55 13 9 (189 #B0489)/NRC #40 10 01 06

TECHNICAL HIGHLIGHTS:

After discussions with the NRC technical monitor, a revised list of leak rate test reports to be reviewed has been assembled. The listing, which represents 27 containments, is attached. Also contained in the listing is the type of data contained in each of the reports. The letters "A," "B" and "C" in the report column designate that the test was either Type A, Type B or Type C. Designations "2nd" and "SUPPL" in this column indicate that the report contains information for either a secondary containment test, or supplementary information or corrections to previously published test results, respectively.

Of the 65 documents reviewed, there are data from 49 Type A tests, 46 Type B and Type C tests, 5 secondary containment test reports, and 7 supplementary test reports. Of the 49 Type A tests, 4 were conducted using the reference vessel test method, 43 used the absolute test method, and 2 used both. (One of the reference vessel tests was as late as 1980, though it was conducted in conjunction with an absolute test method for comparison.) The leak rates were determined using the total time method approximately 80% of the time, with the point to point method being used 35% of the time and the mass plot technique being used 48% of the time. These (and subsequent) percentages do not total 100% due to the practice of using more than one data analysis technique for each set of data. Full pressure tests are represented 44 times, and reduced pressure tests are represented 21 times. Of the 21 reduced pressure tests, 16 were conducted along with the full pressure tests in order to determine the correlation between the two types of tests. Therefore, reduced pressure tests were used only five times to determine the leak rate. Several utilities were unable to determine a reasonable correlation between the two tests and consequently used full pressure tests. Regarding test duration, 73% of the tests lasted 24 hours or longer, 15% lasted between 12 and 24 hours, and 12% were completed in under 12 hours.

The test data included in the reports are being input into a computer program and analyzed by the three data analysis techniques to allow a comparison of the techniques. The results, along with the input data, are plotted to determine any trends in the data and to view the sensitivity of the data measurements.

Arrangements are being made to visit three plant sites to observe integrated leak rate tests and to interview personnel. Tentative plans are to visit St. Lucie in October, Sequoyah in November, and Browns Ferry in December. Close contact is being maintained with the NRC Regional Office in Atlanta so as to keep abreast of the changes in scheduled test dates.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

CONTAINMENT LEAK RATE TEST REPORTS

DOCKET	CONTAINMENT	NUMBER	DATE	REPORT
50010	DRESDEN UNIT 1	241	3/19/74	A
50237	DRESDEN UNIT 2	115	12/9/70	A B C
		139	1/25/71	2ND
		173	5/28/71	2ND
		1092	8/16/76	A B C
50245	MILLSTONE UNIT 1	212	4/27/73	2ND
		232	6/12/73	A B C
		934	11/76	A B C
50247	INDIAN POINT UNIT 2	811	12/1/76	A
		8002050528	1/29/80	A B C
50249	DRESDEN UNIT 3	59	2/11/71	A B C
		491	9/3/74	A B C
50255	PALISADES	327	7/16/74	A B C
50259	BROWNS FERRY UNIT 1	102	8/13/73	A B C
		928	9/15/76	A B C
		1100	12/13/77	2ND
		1105	7/25/77	SUPPL
50280	SURRY UNIT 1	517	1/15/76	A B C
		623	7/1/76	SUPPL
		8111040104	6/81	A B C
50281	SURRY UNIT 2	126	7/3/73	A
		8009170321	9/16/80	A B C
		8203240369	12/81	A B C
50282	PRAIRIE ISLAND UNIT 1	166	10/4/73	A B C
		249	6/6/74	SUPPL
		749	5/15/77	A B C
		8101050262	12/22/80	A B C
50285	FORT CALHOUN UNIT 1	105	6/73	SUPPL
		733	75	B C
		734	1[illegible]/76	A
		952	[illegible]6/78	B C
50286	INDIAN POINT UNIT 3	241	[illegible]/19/75	A
		252	5/12/75	SUPPL
		7811030111	6/19/78	A
50298	COOPER	134	3/4/74	A
		767	2/10/77	A B C
		8009190478	5/23/80	A B C
50305	KEWAUNEE	?	8/24/73	A B C
		180	4/18/74	2ND
		427	5/20/76	B C
		633	6/29/77	A B C
		?	8/10/78	B C
50306	PRAIRIE ISLAND UNIT 2	232	1/31/75	A B C
		?	12/77	A B C
50313	ARKANSAS UNIT 1	147	3/8/74	A B C
		176	5/10/74	SUPPL
50315	D. C. COOK UNIT 1	?	2/26/75	A B C
		7811210260	11/14/78	A B C
50316	D. C. COOK UNIT 2	?	6/26/78	A B C
		8109150432	9/10/81	A B C
50317	CALVERT CLIFFS UNIT 1	169	3/8/74	A B C
		222	6/27/74	SUPPL
		?	4/24/78	A B C
50324	BRUNSWICK UNIT 2	?	12/18/74	A B C
		1268	3/15/78	A B C
50325	BRUNSWICK UNIT 1	405	11/29/76	A B C
		8111040306	10/29/81	A B C
50328	SEQUOYAH UNIT 2	8105270060	5/19/81	A B C
50335	ST. LUCIE	216	10/20/75	A B C
		7908100466	8/7/79	A B C
50336	MILLSTONE UNIT 2	237	7/16/75	A B C
		7908200369	8/10/79	A B C
50368	ARKANSAS UNIT 2	551	12/15/77	A B C
50369	MCGUIRE UNIT 1	8001090377	1/3/80	A B C
50416	GRAND GULF	8204160572	4/14/82	A B C

PROGRAM TITLE: Containment Penetration Integrity

PROGRAM MANAGER: D. J. Naus

ACTIVITY NUMBER: ORNL #41 88 55 06 2 (189 #B0814)/NRC #40 10 01 06

TECHNICAL HIGHLIGHTS:

A literature survey was completed to obtain analytical and experimental studies of elastomerically sealed penetrations. A very limited number of papers were found which had relevance to containment penetration seals, particularly with respect to leakage of seals due to extreme temperature, pressures and radiation effects. Only one analytical study used to evaluate leakage through elastomeric materials was discovered. This work was based on experimental test results but did not include effects of radiation dosage. Studies of radiation effects on elastomeric materials revealed that no experimental investigations have been conducted to determine the effects of radiation on seal leakage. The reports essentially conclude that unless a seal has been tested under operating conditions, that is, pressure, temperature, and radiation, the seal may or may not perform as intended.

Discussions with a utility, an architect-engineer and two penetration vendors were held to obtain an understanding of the design and qualification testing procedures for containment penetrations. Qualification tests are performed according to the A-E's or utilities specifications. This work is done by the penetration vendor or a subcontractor. A report of the testing is provided ultimately to the utility for the plant record system. Several of these reports were obtained but the testing criteria included only conditions associated with design basis accidents and did not include measurements of leakage rates of the penetration seals under pressure, temperature, and radiation.

Work has been inidiated on a draft of the report for this study. The draft will include a proposed testing program based on the results of the state-of-the-art knowledge of seal design procedures.

MEETINGS AND TRIPS:

On August 31, 1982, C. B. Oland and G. C. Robinson visted Sargent and Lundy and Commonwealth Edison in Chicago to discuss analytical studies of containment penetrations, qualification testing of electrical penetration assemblies, qualification testing of elastomeric seal materials, existing design details and accident sequences.

On September 1, 1982, C. B. Oland and G. C. Robinson visited Chicago Bridge and Iron Company and W. J. Woolley Company in Oak

Brook, Illinois. Discussions with these penetration vendors revealed that any qualification testing on elastomers is performed by their subcontractors according to the purchase specifications. Design details were discussed as well as problems associated with handling, fabricating, machining and installing various seals. Potential seal design configurations were also addressed.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Evaluation of Performance of Greased Prestressing Tendons in Nuclear Power Plant Structures

PROGRAM MANAGER: D. J. Naus

ACTIVITY NUMBER: ORNL #41 88 54 32 3 (189 #A9044)/NRC #10 19 01 01 2

TECHNICAL HIGHLIGHTS:

The final report for this activity has been completed, reproduced and distributed.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

J. R. Dougan, "Evaluation of Inservice Inspections of Greased Prestressing Tendons," NUREG/CR-2719, ORNL/TM-8278, September 1982.

PROBLEM AREAS:

None.

PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: ORNL #41 89 55 10 1 (189 #B0119)/NRC #60 19 01 30

TECHNICAL HIGHLIGHTS:

Task 1: Program Management, Fracture Mechanics and Analysis — Additional comparisons have been made between 3-D (ORVIRT) and 2-D (OCA) K_I calculations for inside surface flaws in PWRs subjected to combined pressure-thermal loading. At $t = 40$ min into a "typical" OCA, the 2-D calculated K-value for a surface crack having $a/w = 0.75$ and a length $\sim$6 ft (one shell course width) is $\sim$2.5 times as large as the 3-D value. These studies are continuing, but the preliminary results indicate that the 3-D calculations show a much greater tendency for crack arrest than do the 2-D computations for the deeper flaws.

Examination of existing small specimen fracture toughness data obtained in the transition range of temperature shows that the fracture toughness of forgings tends to be more variable than that of plate. In addition, when testing is done in displacement control, specimens in which visible stable crack growth precedes cleavage also tend to produce a load-displacement curve with a descending branch; and the two phenomena appear to correspond. Thus the maximum load point on a load-displacement curve is not necessarily the correct measurement point for calculating a cleavage fracture toughness.

Task 2: Irradiation Effects — Nuclear grade steel plate with dimensions of 4.62 by 2.39 by 0.22 m thick has been purchased for the 4T-K_{Ic} Program and will be shipped to the weldment fabricator when identified. Meetings have been held with Babcock and Wilcox Company and Combustion Engineering, Inc. to discuss weldment fabrication. Weld wire with two different copper compositions is currently being fabricated. Charpy impact control specimens [22 each of HSST Plate 02 (series 02G)] from capsule A of the Fourth HSST Irradiation Program were tested in the hot cell and remaining Charpy V-notch specimens will be tested during October, and results will be presented.

Task 3: Thermal Shock — Modifications were made to OCA-II to include radial variations in material properties in the calculation of longitudinal stresses. The ability to consider radial variations in material properties in the calculation of the tangential and radial stresses were included previously.

Calculations pertaining to TSE-7 were made which indicate that the closed-form equation that we have been using for semielliptical flaws substantially overestimates K_I values for large aspect ratios (this was determined by comparing the results with 2-D K_I values). The proposed test conditions for TSE-7 are now being reviewed with the aid of results from 3-D analyses.

The test cylinders for TSE-7 and 8 and PTSE were received, and preparations were commenced for obtaining material properties using the prolongation of the cylinder to be used for TSE-7 (test cylinder TSC-4). This source of material will also be used for fabricating a new set of thermocouple thimbles for TSE-7.

Task 4: Intermediate Vessel Test — The special weld seam, containing the flaw, was flame-cut from vessel V-8A. The block of material removed was examined ultrasonically to determine the extent of tearing of the flaw. It was then cut so as to recover as much uncracked material as practicable without disecting the flaw. The flaw was split open at liquid nitrogen temperature. Detailed fractographic examinations and flaw measurements are under way. The fracture surfaces from vessel V-8A have been photographed and are being prepared for dimensional mapping. Following those measurements, one fracture surface will be sectioned and examined by scanning electron microscopy.

At present it is evident that the flaw was well sharpened by fatigue and grew during the test about 13 mm in depth and 130 mm in length. The flaw profile appears as about 3/4 of an ellipse: that is, it tunneled at both ends.

A series of ORVIRT-3-D post-test calculations of the tunneled flaw geometry was planned. It has been observed, before the V-8A test, that semielliptical flaws would have G (or J_I) distributions highly peaked along the segment of the flaw profile where preferential growth would produce tunneling. The first post-test analyses of the tunneled flaw indicate a more uniform distribution, which is consistent with test results.

The blocks of unfractured material from the V-8A seam weld have been cut up to provide enough material for J-R tests by Babcock & Wilcox Company and the Metals and Ceramics (M&C) Division. Most of the material will be used by the M&C Division for K_{Ic} determinations of importance to the pressurized-thermal-shock test PTS-2.

Task 5: Pressurized Thermal Shock — Pressurized-Thermal-Shock Test Facility (PTSTF) design and construction activities continue to remain on schedule and within projected costs. A scope of work for the PTSTF data acquisition system has been reviewed and program approval to proceed with procurement and software design has been given. Methods to provide programmable, controlled pressurization during pressurized-thermal-shock tests using the existing intensifier, are being evaluated.

Discussions are continuing on a contract to place a patch of TSE-6 material in ITV-7 and a patch of degraded weld metal in ITV-8. A specification to assure that desired material properties will be attained in the preparation of both of these vessels is nearing completion.

Material produced by the Babcock & Wilcox Company in the preliminary trials of V-8A welding procedures was returned to ORNL for testing. Selection of a welding process for preparation of vessel V-8 for PTS-2 will be based on these tests. Specimen fabrication drawings have been completed for Charpy-V and miniature tensile specimens to be machined from the weld metal of three of the nine trial submerged-arc weldments for vessel V-8A. These weldments were used to evaluate the effects of flux and subsequent post-weld heat treatment (PWHT) on the mechanical properties. The weldment designations, flux combinations, and PWHT temperature are as follows:

Weld	Linde 60 (%)	Linde 80 (%)	PWHT (°C)
V822	75	25	579
V813	60	40	552
V823	75	25	552

Parts of the cylinders procured originally for TSE-9 will be used in the refabrication of V-7 for PTS-1. TSE material has been cut for fracture tests by the M&C Division to provide the data needed for proceeding with the work on V-7 and for pretest analysis of PTS-1.

Task 6: Cladding Evaluations — A new plate of A533, grade B, class 1 steel has been received. This plate will be used for a new series of clad plate tests utilizing a three-wire series arc-clad application technique typical of actual vessel fabrication practice in the late 1960s. A meeting was held with personnel from Combustion Engineering in Chattanooga to discuss our specific requirements for cladding test plates as well as our need for cladding material for properties characterization.

In regard to the continuation of the existing test series, it was concluded that repair (to eliminate unwanted geometric effects of the deep-groove flawing technique) of the remaining plates clad with T308/309 weld metal would cause unacceptable fracture properties of the plate. Hence, these plates will be stored until a suitable use is available. Preparations are being made to test the one remaining unclad plate under loading conditions similar to those leading to arrest in the plates with T308/309 cladding (CP-3 and CP-8). Testing is scheduled for October.

MEETINGS AND TRIPS:

Dr. Karl Kussmaul from MPA - Stuttgart, FRG, toured the vessel test site at ORGDP on September 2.

R. K. Nanstad, W. R. Corwin and J. G. Merkle attended a cleavage-fibrous transition and crack arrest progress review meeting between the University of Maryland and Battelle Columbus Laboratories at Columbus, Ohio, on September 10, 1982.

G. D. Whitman participated in the Symposium on Neutron Embrittlement and Reactor Pressure Vessel Integrity at North Carolina State University, Raleigh, NC, on September 14 and 15.

R. H. Bryan, R. W. McCulloch and K. R. Thoms on September 15 visited Babcock & Wilcox Company, a potential vendor for the repair of ITV-7 and -8, to discuss the requirements of the initial pressurized-thermal-shock tests and possible methods of preparing vessels ITV-7 and -8.

J. G. Merkle attended the ASME Section XI Task Group on Piping and Working Group on Flaw Evaluation meetings in Mystic, CT, on September 20-21, 1982. At the latter meeting, Merkle presented calculations explaining safety criteria proposed by W. E. Cooper for the low-shelf toughness (A-11) vessel overpressure condition.

Dr. Manuel Fuentes Perez and Professor J. M. Bastero de Eleizalde of Centro de Investigaciones de Guipuzcoa and the University of Navarra, respectively, visited ORNL on September 28 to discuss loss-of-coolant accident analysis and consequences.

G. C. Robinson, W. R. Corwin, R. K. Nanstad and R. G. Berggren visited Combustion Engineering, Chattanooga, TN, on September 28 to discuss stainless steel cladding fabrication and thick-section submerged-arc welding for program use.

REPORTS AND PUBLICATIONS:

None

PROBLEM AREAS:

None

PROGRAM TITLE: Improved Eddy-Current In-Service Inspection for Steam Generator Tubing

PROJECT MANAGER: Robert W. McClung

ACTIVITY NUMBER: ORNL 41 89 55 12 1 (189 B0417-8)/NRC 60 19 11 05

TECHNICAL HIGHLIGHTS:

The measurements of magnitude and phase for defects in large-scale tube models have been completed. We are now in the process of improving our computer codes and theoretical estimations to take advantage of the new measurements.

We have tested a new pancake probe using our present standards, which do not contain a method of varying lift-off. The probe "trained" very well, but was sensitive to lift-off variations when we tried to repeat the remeasurements. We have machined a standard that can provide lift-off when the standard is rotated. We are now installing lift-off shims in the standard.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

ACTIVITY NUMBER: ORNL 41 89 55 12 (189 #B0415)/NRC #60 19 10 05

TECHNICAL HIGHLIGHTS:

Task 1: Benchmark Fields -

A. PCA - Transport Calculations and Dosimetry - No Activity.

B. ORR-PSF - The simulated pressure vessel capsule (SPVC) has been disassembled in the ORNL hot cells and the metallurgical specimens and dosimetry sensors separated for shipment. Additional instructions were received to remove all streaming plugs from the compact tension specimens and the thermal sleeves from tensile specimen. This task required five (5) cell days to complete. An additional 1½ cell days were required to separate out the dosimetry sensors into separate containers. However, the metallurgical specimens were shipped in time to meet the October 1, 1982 target date for arrival at Buffalo.

The HEDL dosimetry will be shipped after ENSA returns one of the three (3) casks used in shipping the metallurgical specimens.

ORNL has not received DOE Form EV-391, or purchase orders from CEN-SCK, and Rolls Royce Associates, Ltd. to ship their dosimetry.

C. BSR - HSST - The dosimetry analysis for capsules A, B, and C has been completed and is being readied for publication.

Task 2: ASTM Recommended Procedures for LWR-PV Irradiation Surveillance Program

The draft standards E706(I-I) and E706(IIA) are currently being balloted on the E10.05 and E10 levels respectively.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Technology and Costs of Termination Surveys Associated with Decommissioning of Nuclear Facilities

PROGRAM MANAGER: J. P. Witherspoon

ACTIVITY NUMBER: ORNL #41 88 54 32 1 (189 # A9042)/NRC # 10 19 02 05 3

TECHNICAL HIGHLIGHTS:

Preparation of a draft addendum to NUREG/CR-2241 continued and sections on surveys for storage or entombment and abbreviated termination surveys were completed. The draft is expected to be completed by the end of October.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

DIVISION OF ACCIDENT EVALUATION

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: M. B. Herskovitz

ACTIVITY NUMBER: ORNL # 41 89 55 11 8 (189 #B0413)/NRC #60 19 10 01

TECHNICAL HIGHLIGHTS:

Analysis for Test Run 053 from CCTF-II was essentially completed. Efforts are continuing on Runs 054 and 055.

Attempt are underway to obtain the test data from SCTF-I Run 514 and CCTF-II Run 056. The AIRS sensors that were functioning during these tests will be analyzed.

Discussion of the results from two shakedown tests at PKL-II have been postponed until both sides have completed their analyses.

Continuing testing of the AIRS 20 strain gage amplifier unit which is used for the UPTF drag body and breakthrough detector. Investigating the possibility of using a data logger for system checkout and calibration.

Laboratory testing of the breakthrough detector is continuing.

dP Measurement System - A computer model of the purge control system was developed. The operation of the purge controls under the UPTF pressure transient was calculated with the model. A static heat transfer analysis was conducted using the latest sense line assumptions. The results are being forwarded to FRG as a commitment from the July interface meeting. The latest version of the logic diagram showed some negative slack for the dP system for UPTF. The elimination of this slack will be accomplished by expediting the design phase and purchasing of components.

MEETINGS AND TRIPS:

M. B. Herskovitz attended a short course sponsored jointly by NRC and ANS on basic thermal hydraulic mechanisms in LWR analysis at Bethesda, MD, on September 14 and 15

R. A. Hess attended a Logic Diagram Review Meeting concerning UPTF at Sandia Labs in Albuquerque, NM, on September 28 through October 1.

REPORTS, PUBLICATIONS, AND PAPERS:

Hardy, J. E., 'Mass Flow Measurements under PWR Reflood Conditions in a Downcomer and at a Core Barrel Vent Valve Location,' NUREG/CR-2710 (ORNL/TM-8331), August 1982.

PROBLEM AREAS:

None.

PROGRAM TITLE: Advanced Two-Phase Instrumentation

PROGRAM MANAGER: G. N. Miller

ACTIVITY NUMBER: 41 89 55 11 5 (189 #B0401-2)/NRC 60 19 11

TECHNICAL HIGHLIGHTS:

This month's effort in ultrasonics for the two-phase level instrumentation consisted of development of the pulse measurement for both torsional and extensional stress waves in the stainless steel probe, packaging the prototype electronics for the pulse timing, and a high-temperature test to determine the effects of temperature on the magnetostrictive transducer material. This completed work will allow us to proceed to the next stages of testing a working probe at high temperature and pressure, and obtaining a continuous output of level information in engineering units.

A method of magnetic biasing fields to enhance the torsional and then extensional signals was tested and found to be very successful. A set of three coils arranged alternately with the transducer excitation and pickup coils was used for the extensional bias which resulted in a larger pulse than that previously obtained with permanent magnets. These same coils were then used to provide an ac field of slowly diminishing magnitude to demagnetize the magnetostrictive rod. The subsequent application of a large (10 amp) current along the rod resulted in a clean and strong torsional signal.

The signal-conditioning electronics have been redesigned for passive differentiation and the addition of a signal-strength level control. These features allow the selection of the most intense pulses to be processed in a low-noise circuit for accurate peak detection. The information derived from the peak of the pulse echo is used to start or stop a fast binary counter driven by an ultra-stable, fast oscillator. Thus an accurate measurement of the time lapse between any two stress pulses can be determined.

One of the more important discoveries, from a practical viewpoint, was that the signals from torsional stress pulses do not significantly degrade at higher temperatures as long as the torsional bias field (mentioned above) is maintained over the length of rod. Specifically, a remendur magnetostrictive rod was heated to 700°F while observing the extensional and torsional signals. At around 350°F, the torsional pulse completely disappears; however, the application of a torsional bias field restores the pulse to at least 80% of its original amplitude even up to 700°F. The amplitude of the torsional pulses was seen to be approximately proportional to the applied torsional biasing current and to decrease about 0.05% per degree F. Similarly, an extensional bias must be maintained at elevated temperatures in order to detect the extensional pulses.

MEETINGS AND TRIPS:

None.

REPORTS, PUBLICATIONS AND PAPERS:

Award: IR-100 Award, September 23, 1982.

PROBLEM AREAS:

There is a potential problem of locating the magnetostrictive material, Remendur. Vendors that have been identified require a $40,000 minimum order. We cannot afford this expenditure now. We have had some material fabricated here at ORNL; however, the performance is not as expected yet. We are working on this problem.

PROJECT TITLE: Evaluation of Bundle Heat Transfer Models and Correlations

PROJECT MANAGER: C. B. Mullins

ACTIVITY NUMBER: ORNL #41 89 55 13 6 (189 #B0463)/NRC 60 19 01 10

TECHNICAL HIGHLIGHTS:

Analysis of the intermediate flow heat transfer test series is continuing. Verification of a radiation model used in the analysis has begun.

Screening of film boiling, void fraction, and steam cooling data recently acquired is continuing and 60% complete.

MEETINGS AND TRIPS:

C. B. Mullins attended the joint NRC/ANS meeting on Thermal Hydraulics in Bethesda during mid September.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Fission Product Release from Fuel

PROGRAM MANAGER: M. F. Osborne, R. A. Lorenz, and R. P. Wichner

ACTIVITY NUMBER: ORNL #41 89 55 10 8 (189 #B0127)/NRC #60 10 01 40

TECHNICAL HIGHLIGHTS:

1. Fuel Procurement and Characaterization

The shipment of irradiated LWR fuel from Savannah River Laboratory was promised before the end of FY 1982, but, as yet, has not been received.

2. Fission Product Release Tests and Results

Based on current results, several additional samples from test HI-2 (20 min at 1700°C in steam) were selected for analysis during the month. Consequently, the results from a few samples are still outstanding. Currently available release data for test HI-2 are compared with the test HI-1 data in the following table; ^{129}I data were obtained by activation analysis, whereas the other nuclides can be measured directly by gamma spectrometry.

Test No.	Temperature (°C)	Time (min)	Fission product release (%)[a]				
			^{85}Kr	^{137}Cs	^{129}I	^{125}Sb	^{110m}Ag
HI-1	1400	30	2.83	1.75	2.04	>0.017[b]	0[c]
HI-2	1700	20	50	49	>36[d]	>1.5[b]	2.12

[a]Percent of total inventory in the fuel specimen based on ORIGEN calculations and radiochemical analyses, decay corrected to July 15, 1981.

[b]Not measurable on some components because of interference from ^{134}Cs and ^{137}Cs; an additional 0.08% might have been present in HI-1 and up to ~0.2% in HI-2.

[c]Not detected; very small amounts might have been present on some components.

[d]Does not include results from thermal gradient tube.

Significant differences in the results of the two experiments are: the total release of Kr, Cs, and I increased from ~2% at 1400°C to ~50% at 1700°C, and the release of Ag increased from undetected to >2%. Most of the Ag and the Sb detected was located on the Pt/Au thermal gradient tube in both tests. Moreover, the small fraction of these two elements that could be removed by chemical treatment indicated that they had either alloyed or reacted with the Pt/Au at 500 to 1000°C.

Spark source mass spectrometry (SSMS) was used to obtain elemental analysis of smear samples from two areas of the thermal gradient tube and the glass wool prefilter, and also of solution samples from the same components. Either the amount of ^{137}Cs, as measured by gamma spectrometry, or a known quantity of Er added as a standard was used to determine mass values for the fission product, structural, and impurity elements detected. The results are incomplete at this time; some individual numbers are being rechecked, and some additional basic and acid leach solutions have been submitted for SSMS analysis.

Most of the aerosol mass as collected by the glass wool was composed of released fission products. The average aerosol concentration assuming 250 mg total aerosol solids and 20-min vaporization time would be 1.5 g/m^3 at 1700°C and 7.4 g/m^3 at 130°C, the filter temperature. At 1700°C the "aerosol" components were mainly in the vapor form but condensed as the exiting gas stream cooled.

The test apparatus is currently being prepared for the next test of irradiated fuel. Following a temperature calibration run with an unirradiated specimen, test HI-3 will be conducted at 2000°C; the steam flow rate will be reduced to ~0.1 L/min (compared to 1.0 L/min in earlier tests) which should result in some cladding melting prior to oxidation.

3. Species Identification by Laser-Induced Fluorescence (LIF)

Our current interest is species identification by the LIF technique. LIF provides species — specific signals with essentially the same apparatus as laser-Raman, but the LIF signal strength is orders of magnitude greater for those species that fluoresce. Both molecular iodine and atomic iodine are known to give strong fluorescence signals. We have recently demonstrated that CsI fluoresces and that the I_2 fluorescence signal in the presence of several atmospheres of water vapor at 700°C is quenched by only a factor of 10. A calibration for I_2 vapor at low concentration in 700°C steam is in progress.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

ACTIVITY NUMBER: ORNL #41 89 55 11 2 (189 #B0122)/NRC #60 19 13 02

TECHNICAL HIGHLIGHTS:

ORECA-FSV Code Development: Development work on the 3-D core code for Fort St. Vrain, ORECA-FSV, continued with the modeling of long-term uncontrolled heatup scenarios. A model for calculation of coolant temperatures around the entire coolant loop was formulated. These temperatures must be known in order to calculate primary coolant pressure. Presently, coolant temperature at the entrance to either the core inlet plenum or the core outlet plenum is a code input. When the new model is implemented, the code input will be steam generator tube temperature (as determined by secondary coolant conditions).

BLAST Code Development: A recent and very helpful review by Brookhaven National Laboratory of the BLAST steam generator code's numberical integration technique concluded that "the current [first-order] BLAST solution algorithm is correct and sufficiently accurate." Furthermore the review noted "it appears that with the computational effort already expanded [for the first-order implicit solution] only minor modifications would be required to obtain the benefits of a full second-order method."

Using the equation formulations suggested by BNL, the second order implicit integration method was incorporated into the Kernforschungsanlage (KFA) BLAST version as a user option. A null transient and several rapid transients were successfully executed using calculational time steps of ten times the minumum water-side nodal transport time. Since this version also includes a quasi-static helium side representation, problem execution is very rapid. Results compared favorably with the original first-order integration technique.

After additional confirmatory testing of the new method, the second order implicit method will be sent to KFA for their consideration and will be used in ORNL code benchmarking efforts.

ORTURB Code Development: The input subroutine for the turbine-generator plant simulator code ORTURB has been modified to improve user ease of specifing turbine and feedwater heater design geometry. Work is continuing to improve other aspects of the physical and mathematical modeling.

2240 MWT HTGR Siting Study: Development continued on the 2240 MWT HTGR version of the ORECA (3-D core model) code in support of the source term and siting evaluation. Models for the core auxiliary cooling system (CACS) were developed and incorporated into ORECA. The number of CACS loops in operation (0 to 3) and their operating periods can be specified by the user. The helium flow rates through each loop are also user-specified, with the exception that the core auxiliary heat exchanger (CAHE) temperature control system may reduce the helium flows as required if the CAHE coolant outlet temperatures exceed a specified limit. The CACS ducting temperature is also modeled, both for the active and inactive loops, because of the effects it can have on gas temperatures and hence the overall system pressure. The duct model for a loop consists of a single node for a cover plate, separated from the liner cooling system (LCS) sink temperature by Kaowool insulation.

A simplified GA fuel failure model, based on a temperature-only failure criterion, was incorporated into ORECA and gives the total core fuel failure fraction as the postulated uncontrolled core heating accident (UCHA) prgresses. Several runs using the new detailed LCS, PCRV, and CACS models were made to note the sensitivity of the results to various accident sequence and model parameter assumptions. Runs can be made at a cost of 30¢/hr.

A draft report by D. T. Goodin of GA on an improved fuel failure model was received and reviewed. This model includes time-at-temperature effects and is considered a significant improvement over the temperature-only-effects model. A general-purpose subroutine for implementing the new model equations for arbitrary fuel temperature histories was developed and tested, and will be incorporated into ORECA as an alternative fuel failure fraction calculation.

Fission Product Release from HTGRs: T. B. Lindemer, A. D. Kelmers, and R. P. Wichner developed a task proposal outlining the activities that they believed to be of importance in improving fission product and actinide release calculations for hypothetical severe accidents. The proposed work focuses on the chemical behavior and transport of fission products and actinides from the failed fuel particle, in the graphite core, and in the PCRV. The proposal was submitted to NRC and discussed at length with them at a meeting.

MEETINGS AND TRIPS:

S. J. Ball, A. D. Kelmers, T. B. Lindemer, and R. P. Wichner met with RES and NRR representatives in Silver Spring, MD on September 29 to discuss proposed work on fission product chemistry and transport modeling.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Iodine and Tellurium Chemistry

PROGRAM MANAGER: J. T. Bell/R. P. Wichner

ACTIVITY NUMBER: ORNL #41 89 55 13 5 (198 #B0453-1)/NRC #60 19 01 0

TECHNICAL HIGHLIGHTS: (L. M. Toth, E. C. Beahm)

Attempts to prepare "HOI" have continued. Silver oxide has been reacted with iodine in the most recent attempts to duplicate previously referenced preparations. ($AgNO_3$ was used last month.) This method proved to be less satisfactory than the preparations using the $AgNO_3$ because the insoluble Ag_2O reacted much slower with the iodine, thus allowing most of the HOI species to dissociate before all the Ag_2O had reacted.

Preparations of the HOI species with $AgNO_3$ at pH = 7 and 9, followed by titrations to determine its lifetime, showed that the higher pH increases the reaction rate by a factor of 3. These results are qualitatively in agreement with trends recognized in previous kinetics studies.

An "HOI" species has been prepared using $AgNO_3$ at 0°C where the disproportionation is slow enough to measure the species spectrophotometrically. The reaction rate is a factor of seven slower at 0°C than at room temperature. An unusual spectrum was obtained that is similar in many respects to I_3^- and suggests a species, if it is due to a +1 valence species of iodine, that is structurally like that of the triiodide, namely HOI_2^-. (It is similar to I_3^- if one considers the OH^- and I^- are interchangeable.) It is interesting to note that HOI_2^- has already been reported in a thesis by Y. T. Chia in 1958 and has a spectrum that is in some respects similar to the one seen here. Further work is in progress to clarify these findings.

In our attempts to obtain a partition coefficient for HOBr, we have obtained aqueous solutions of 0.1 M HOBr and have identified a strong band at 265 nm. However, we have not been able to find any evidence of HOBr in the vapor under conditions similar to those used successfully for HOCl. The current results suggest that the HOBr partition coefficient is considerably greater than that of HOCl. Higher concentrations will be sought in an effort to find a band in the vapor and thereby quantify this observation.

An iodine volatility test was run at 323 K, pH = 7.0 (buffered) and 9.5×10^{-5} g-atom/L. As shown in the table, the partition coefficient values increased rapidly in the first 6 h of equilibration. Following this initial increase, the partition coefficient remained essentially the same for ~14 h, then again a large increase was observed. We plan a further test at the same conditions to determine the verity of the plateau of 14 h in the iodine partition coefficient value.

Iodine partition coefficients: sample 9.5×10^{-5} g-atom/L, pH 7.0 (buffered), 323 K

Iodine partition coefficient	Time after initial mixing
205 ± 5	9.0×10^{2} s (15 min)
412 ± 13	7.7×10^{3} s (2 h, 8 min)
568 ± 22	1.5×10^{4} s (4 h, 10 min)
697 ± 30	2.2×10^{4} s (6 h)
729 ± 30	8.0×10^{4} s (22 h, 15 min)
1324 ± 82	9.0×10^{4} s (24 h, 53 min)
1619 ± 110	1.0×10^{5} s (27 h, 42 min)
2152 ± 188	3.4×10^{5} s (94 h, 27 min)
2272 ± 214	4.3×10^{5} s (119 h, 10 min)

We are now in the process of recalibrating the counting equipment using the same techniques described in the monthly highlights report for June and July 1982. Two identical experimental systems are in use so that we may have a continuous operation, one system counting while the other system is equilibrating. The dual system operation also reduces by one-half the number of iodide/iodine conversions.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: LWR Aerosol Release and Transport

PROGRAM MANAGER: R. E. Adams

ACTIVITY NUMBER: ORNL # 41 89 55 11 1 (189 #B0121)/NRC # 60 A 13 03

TECHNICAL HIGHLIGHTS

NUCLEAR SAFETY PILOT PLANT (NSPP): R. E. Adams, R. F. Benson, M. T. Hurst

Analytical procedures are being finalized for the chemical analysis of the collected aerosol samples taken during the recent mixed concrete/iron oxide experiment (No. 601). Delays have been encountered in the analysis of these samples because of budget-induced reductions in the staff of the analytical laboratory. Results are now anticipated in October.

The next aerosol experiment planned will involve study of the behavior of a limestone-aggregate concrete aerosol in a steam environment; conduct of this test is scheduled for late October.

LWR CORE-MELT STUDIES: G. W. Parker, G. E. Creek (consultant), A. L. Sutton, Jr.

Core-melt experiments, to date, have utilized a close-coupled induction coil in the induction furnace. This heating system normally produces a rapid rise in temperature of the test fuel bundle to 1400°C, or above, in 1-2 minutes. This heating arrangement has been adequate to demonstrate the interaction between molten silver from the control rod alloy and the Zircaloy fuel rod cladding producing a low-melting intermetallic phase. It is now of interest to investigate the influence of extensive surface oxidation of the Zircaloy cladding on this interaction. A loose-coupled induction coil has been fabricated and installed in the induction furnace. This arrangement should allow production of a slower rate of temperature rise in the fuel bundle allowing a longer period of time in the 1200-1350°C temperature range for the metal-water reaction to form a thicker oxide layer on the Zircaloy cladding prior to rupture of the stainless steel sheath of the silver alloy control rod.

Design of the 10 kg core-melt furnace has been completed and construction of the housing and the copper crucible has been initiated in the ORNL shops. The quartz containment housing has been ordered from a commercial fabricator. The 250 kW rf induction generator, with a 250-350 kHz output specification, is out for bid.

ANALYTICAL: M. L. Tobias and J. C. Petrykowski

The results of the pre-test code predictions for the special sodium spray fire test (AB-5) of the CSTF ABCOVE project have been received from the other participants and are being compared to the QUICK and HAARM-3 calculations made at ORNL. These two codes were also utilized by some of the other participants as well as the aerosol codes, MAEROS, HAA-3C, HAA-4A, CONTAIN and MSPEC.

Initial attempts to measure the magnitude of the air currents within the NSPP vessel produced by the small fan-mixer were relatively unsuccessful. Air currents are present in the vessel, but the velocity is below the 5 cm/s minimum velocity rating of the hot-film anemometer used. At very low gas velocities, free heat convection from the hot-film is dominant and masks the heat loss to the local gas flow field. An attempt will be made to make these measurements by operating the hot-film sensor at a lower temperature.

MEETINGS AND TRIPS:

T. S. Kress attended the meeting of CSNI Group of Experts on Nuclear Aerosols in Reactor Safety, Paris, France, September 9-10, 1982.

G. W. Parker attended a NUS/NRC meeting to discuss plans for the LOFT Project, Gaithersburg, MD, September 26-27, 1982.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL #41 89 55 10 6 (189 #B0120)/NRC 69 19 13 01

TECHNICAL HIGHLIGHTS:

Techniques similar to the ones we use to generate strain data from photographs of bundle cross sections are being developed at the UKAEA Springfields Laboratory for use in the MERLIN program and at INEL for use on the LOFT L2-6 test bundle. To aid their development activities and to check on the accuracy of our techniques, both organizations were supplied with photographs of two sections of the B-5 bundle that were particularly difficult to process. The Springfields technique results in strain based on tube outside diameter (the same as we and virtually all other investigators use), while the INEL technique results in strains based on tube inside diameter. Although we collect the data for determining the latter strains, we have not reduced the data for comparison. (The two strains are uniquely related for uniform strain distributions in round tubes with constant wall area.)

We recently received preliminary results from each group for comparisons with our results. Additional analysis was required to make the comparison with the INEL results. The analysis provided further insight to our understanding of the results and showed that all three sets of measurements are in agreement.

MEETINGS AND TRIPS:

Dr. R. Del Negro and Dr. P. Berna, of the CEA-Cadarache Laboratories visited ORNL September 13 for discussions on cladding oxidation and ballooning research.

REPORTS, PAPERS, AND PUBLICATIONS:

A report, entitled "Variations in Zircaloy-4 Cladding Deformation in Replicate LOCA Simulation Tests", was submitted to the reproduction department for printing and distribution. The report presents results of five single rod heated shroud replicate tests that were conducted to study statistical variations in cladding deformation parameters.

PROBLEM AREAS:

None.

PROGRAM TITLE: Near-Term TRAP-MELT Verification

PROGRAM MANAGER: A. L. Wright/R. P. Wichner

ACTIVITY NUMBER: ORNL #41 89 44 13 8 (189 #B0488)/NRC #60 19 13

TECHNICAL HIGHLIGHTS:

The objective of the TRAP-MELT Verification Test Program is to conduct the most immediately useful tests related to the deposition and transport of aerosols and fission products under conditions simulating those possible in severe accidents.

Two preliminary aerosol transport experiments were performed in August; initial attempts to make a mass balance in these tests indicated that only ~40% of the expected aerosol mass could be accounted for. However, more detailed evaluations for the first experiment indicate that roughly 32 g of iron oxide material was deposited on the plasma torch quartz liner. This brought the mass balance for the first test up to 76%. A similar evaluation of the deposits on the quartz liner in the second test is being performed.

In each of the first two tests, the aerosol "plateout" near the bottom of the pipe was very nonuniform; this occurred because the aerosol source was directed toward one of the side-walls. The third preliminary test was performed this month; in it the plasma torch was moved 1 ft further away from the vertical pipe in an effort to eliminate nonuniform deposition. We attempted to burn iron oxide powder for 7 min, but a powder feed—oxygen feed mismatch lead to an aerosol production time of roughly 5 min. Preliminary analysis of the deposition pattern at the bottom of the pipe indicates that it is at least an order of magnitude more uniform than in the first two experiments.

Fission Product Interaction with Aerosols (R. D. Spence)

The induction plasma torch is an attractive alternative to the dc plasma torch as an aerosol generator. The advantages are free access (no internals) to the plasma zone, no contamination from electrodes, longer residence time in the flame, the possibility of using oxygen in the plasma gas, and lower velocities. Unfortunately, the commercially available induction plasma torch requires higher flowrates than the smallest dc plasma torch. This disadvantage outweighs all of the advantages mentioned above. In addition, the induction plasma torch is much more expensive than a dc plasma torch.

Direct current plasma torches are available that require less than 20 standard liters per minute (slpm) (<40 standard cubic feet per hour or SCFH) of plasma gas. The possibility of using laminar flow may lower the plasma gas flowrate requirement to less than 10 slpm (<20 SCFH). The size of our experimental apparatus is a function of the plasma gas flowrate. Even the lowest flowrates discussed above will still require venting of the aerosol.

A rough draft of the work plan is being written and should be finished by the end of the month, satisfying the major milestone for FY 1982.

MEETINGS AND TRIPS:

A meeting was held at ORNL on September 8, 1982, with R. R. Sherry and L. Chan of the Fuel Behavior Branch. This meeting was held to discuss the present status of the TRAP-MELT project at ORNL.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Severe Accident Sequence Analysis (SASA)

PROGRAM MANAGER: S. A. Hodge

ACTIVITY NUMBER: ORNL #41 89 55 13 4 (189 #B0452)/NRC 60 19 01 3 0

TECHNICAL HIGHLIGHTS

Major ongoing work concerns the MARCH upgrade project, the analysis of fission product transport for the scram discharge volume break accident sequence, and the accident sequence analysis for the loss of decay heat removal (DHR) events at Browns Ferry Unit One.

Final preparations to commence accident analyses for the Limerick plant (MK-II containment) have been completed. Up-to-date versions of the Limerick FSAR and PRA are on hand. The plant has been visited and a working session at the simulator has provided familiarization with the control room equipment. A set of Limerick training manuals was obtained at the simulator.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress during September are presented below with a brief initial statement of the purpose of each group.

Group I: (R. M. Harrington) Determines and analyzes the events of the accident sequence which would occur prior to core uncovery, using the ORNL-developed simulation program BWR-LACP to provide plotted studies of the plant response to operator actions.

Preliminary results have been obtained for several loss of decay heat removal (DHR) sequences using the plant response code BWR-LACP. The calculations were performed for a 22-hr period following the reactor trip; containment failure pressure was not reached during this period. The IREP results for this same sequence predict failure of injection well before 22 hr; the ORNL results do not agree with this prediction because of the realistic, best-estimate assumption of a full condensate storage tank (CST) at the beginning of the sequence (instead of the 35% full CST which the IREP study assumed).

A summary presentation of the preliminary loss of DHR results and analytical assumptions was prepared for presentation to TVA engineers for their review and comment.

The BWR-LACP loss of DHR calculations to date have assumed a homogeneous suppression pool water temperature. It is known that the pool will become thermally stratified during the loss of DHR sequences. This effect is expected to be important in the determination of the containment failure time. It is planned to estimate thermal stratification and local temperature effects using a model being developed by D. H. Cook.

Group II: Determines and analyzes the events of the accident sequence that would occur following core uncovery, including core melt and containment failure.

March 2.1 Development Plan (S. R. Greene) The final March 2.1 development plan coordinated with Battelle Columbus Laboratories has been forwarded to the NRC SASA technical monitor for comment and approval. The plan calls for delivery of this code, which will significantly improve BWR analysis capability, by February 28, 1983.

An important part of the ORNL MARCH upgrade project concerns identification of all BWR-4 and later boiling water reactor design systems whose operation might significantly affect the course of a severe accident so that the modeling requirements for an adequate analysis code can be identified. In addition to the information already available from previous SASA work, this has required the acquisition of material concerning BWR-5/MK-II and BWR-6/MK-III designs. To this end, copies of the LaSalle and GE Standard Plant (GESSAR) FSARs have been assembled from the material available at the ORNL Nuclear Safety Information Center (NSIC).

FORTRAN-77 Compiler at ORNL (S. R. Greene, L. J. Ott) Battelle Columbus Laboratories will release a FORTRAN-77 version of MARCH 2.0 in late October. The ORNL Computer Sciences Division has recently installed a FORTRAN-77 compiler on the Laboratory computer system for user testing and checkout. We have tested this compiler with the existing MARCH version 1.1 with satisfactory results, and anticipate no difficulties in running the future versions of MARCH.

Modifications to MARCH for Version 2.1 (L. J. Ott) Head curve modifications for BWR low-pressure ECC systems have been transmitted to Battelle Columbus Labs for incorporation into MARCH. This completes Task 2.6 in the MARCH 2.1 development plan.

Development of simplified models to analyze the heatup, oxidation, and melting of BWR channel boxes and control blades (Task 2.4 of the MARCH 2.1 development plan) was begun in September. These models will be used in conjunction with the existing fuel rod heatup models in MARCH subroutine BOIL.

Activities at RPI under the subcontracted effort there, which is under the direction of Dr. R. T. Lahey and Dr. M. Podowski, continues in several areas:

Heat transfer model for a fuel pin. The case of heat transfer across a fuel pin after cladding failure was developed. The model proposed assumes that molten zirc/UO_2 escapes from all the nodes above the failure zone, leaving a gap between the fuel pellets and the remaining solid zirc. The decay heat generated in the fuel and the energy released due to the oxidation of zirc result in further melting of the metallic clad. It is assumed that the new molten material is also relocated outside the pin.

Derivation of an improved model for falling zirc/UO_2 mixtures. It has been found that modeling the dynamics of molten zirc/UO_2 by using a continuous film concept is neither physically appropriate nor numerically efficient. Consequently, a slug model will be applied. The model uses both the momentum and thermal energy conservation equations for a lumped mass of molten zirc to calculate such parameters as the slug temperature and velocity, and also the zirc freezing rate. It also allows for evaluating the initial slug geometry (i.e., thickness and length) as a function of the cladding failure size.

Whole core model. The development of a program for the entire core is based on dividing the core into radial and axial nodes. It is assumed that each radial zone consists of an average fuel pin and two sections of channel boxes operating at different temperatures. Also, there is a control blade between neighboring zones. The model accounts for heat convection from the fuel pins, channel boxes, and control blades to the coolant, as well as for radiative heat transfer between the fuel pins, channel boxes, and control rod blades. At present, work is underway to test this code, and to incorporate the detailed fuel pin model.

Pressure Suppression Pool (PSP) Modeling (D. H. Cook) An informal agreement has been discussed with the General Electric Company for the study of localized pool heating under accident conditions.

A small computer program written to predict time dependent local temperatures in the PSP during safety relief valve discharge is currently being improved to include: multi-layer capability in the vertical direction, whole pool modeling (for predicting asymmetric transients), inter-cell calculation of a plume rise event in the bay of discharge, horizontal entrainment by a hot layer across the surface, and pressure/temperature coupling to the wetwell airspace. This code will be used to estimate the PSP response in the loss of DHR study.

Group III: Determines the magnitude and timing of fission product release from the fuel, establishes the various pathways for fission product release to the atmosphere, and performs the fission product transport calculation for each Severe Accident sequence analyzed.

Structural Material Aerosol Formation in the Reactor Vessel (R. A. Lorenz) Aerosol from structural material components will form during the 58-min period while the core is undergoing melting at 2400°C in the SDV break accident sequence. According to the ORNL model, the steam and hydrogen flowing through the heated core will become saturated with structural material component vapors. The amount of steam and hydrogen flow has been calculated and the associated mass of structural material carried out of the reactor vessel during core melting is approximately 170 kg or 3000 mols.

Aerosol Transport Analysis (A. L. Wright, R. P. Wichner) The HAARM-3 aerosol code is being used to calculate aerosol behavior in the drywell and in the reactor building/refueling floor. The first estimate of the core-concrete aerosol source for this sequence was roughly 5000 kg. However, comparison to estimates made by others (in particular NUREG-0772) indicates that this source is probably too large. The aerosol source calculation was redone with improvements, particularly to include the influence of the hydrostatic pressure of the core-concrete melt layer on aerosol production. The revised source is now roughly 1620 kg of core-concrete aerosol, which seems more reasonable.

Using the revised aerosol source, all calculations for the draft of the fission product transport report were completed. The aerosol leaking from the drywell is the source to the reactor building. Assuming that 162 kg of aerosol collects on the SGTS filters before they rupture, only about 17% of the 1620 kg of aerosol produced in the drywell is released to the environment.

During the period from reactor vessel melt-through until drywell failure, flow from the drywell to the reactor building is through the CRD WITHDRAW lines, which vary from 20 to 150 ft in length. A scoping calculation has been performed to determine the approximate degree of aerosol deposition in the lines, assuming

1. Steady flow, evenly distributed between the 185 WITHDRAW lines.
2. Constraint temperature (380 K).
3. Carrier gas is nitrogen.
4. Partial size of 1 μm and density of 5 g/cm^3.

The velocity in the lines is quite high, ensuring turbulent flow. The resulting aerosol deposition in the lines is quite small, on the order of 0.3% in the 150-ft lines and much smaller in the shorter lines.

Organic Iodide Production Rate (J. W. Nehls) A formation rate equation for organic iodides has been developed from the experimental work of Postma and Zavadoski (WASH-1233), Parker et al (ORNL-4502), and Lorenz et al (ORNL/NUREG/TM-25). The equation represents the formation of organic iodides by nonradiolytic means and their destruction by high temperature.

The section of the fission product transport report pertaining to organic iodide production is essentially complete and work has been started on a more comprehensive report describing the development of the rate equation in much more detail.

MEETINGS AND TRIPS: S. R. Greene, R. M. Harrington, S. A. Hodge, and R. P. Wichner attended the NRC/IDCORE information exchange meeting held at TEC Headquarters in Knoxville on August 24 and 25. Each gave a short presentation summarizing past and planned future SASA activities in the area of accident sequence analysis or fission product transport analysis.

R. A. Lorenz attended the International Meeting on Thermal Nuclear Reactor Safety at Chicago, Illinois, August 30—September 2, 1982.

L. J. Ott visited RPI and met with Drs. Podowski and Taleyarkhan on September 9 to discuss the subcontracted modeling effort there. Dr. Manahan of BCL was also present. It was determined that the channel box/control rod blade heatup models developed by ORNL should be made completely compatible with and no more complex than the current MARCH fuel rod heatup model.

S. R. Green, R. M. Harrington, S. A. Hodge, and L. J. Ott visited the Limerick plant site on September 17 for an excellent plant tour hosted by the Philadelphia Electric Company. R. M. Harrington and S. A. Hodge stayed over the following day for a working session at the Limerick simulator.

REPORTS, PAPERS, AND PUBLICATIONS: R. A. Lorenz presented the paper "The Vaporization of Structural Materials in Severe Accidents" at the International Meeting on Thermal Nuclear Reactor Safety.

S. R. Greene, R. M. Harrington, and R. P. Wichner have submitted papers for presentation at the Tenth Water Reactor Safety Meeting.

Volume 1 of the report NUREG/CR-2672, *SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis*, has been submitted for reproduction and issuance.

PROBLEM AREAS: None

DIVISION OF RISK ANALYSIS

PROGRAM TITLE: Acceptable Level of Risk Criteria for Nuclear Power Plants

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #4 88 55 02 2 (189 #B0424) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

Reviews have been received for our paper, "Health and Safety Standards: Theoretical Rationale and Application to Safety Goals for Nuclear Power." Revision of this paper is now in progress. Our paper titled, "Modeling the Societal Impacts of Multiple-Fatality Accidents," was completed and sent out for review. Ralph Keeney and Robert Winkler completed a paper titled, "Von-Neumann-Morgenstern Utility and Equity of Public Risks."

MEETINGS AND TRIPS:

Paul Slovic traveled to Rockville on September 9th to meet with Pat Rathbun, George Flanagan, Ken Solomon and several researchers from ORNL to discuss plans for a program of research on psychological, social, and decision science contributions to risk analysis. A number of research directions were discussed, as summarized in George Flanagan's memorandum of September 10th.

REPORTS, PAPERS, AND PUBLICATIONS:

See above.

PROBLEM AREAS: None

PROGRAM TITLE: Analysis of Proposed New IAEA Basis for Transportation Regulatory System

PROGRAM MANAGER: K. F. Eckerman

ACTIVITY NUMBER: ORNL 40 10 01 06 (189 B0810-2)/NRC 60-82-283

TECHNICAL HIGHLIGHTS:

A listing of the revised numerical guidance incorporating changes introduced as a result of the Special Working Group on the "Q-System" was received from MacDonald. A spot check of the data will be performed. All other tasks under the contract have been completed.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Analysis of Reliability Data from Nuclear Power Plants

PROGRAM MANAGER: Raymond J. Borkowski

ACTIVITY NUMBER: ORNL 41 88 55 03 01 (189 #B0445) NRC #60 19 40 01

TECHNICAL HIGHLIGHTS:

Task 1: Reports. Comments on the pump report are being incorporated.

Task 2: Data Collection and Encoding. Population data encoding of the valves in Plant 3 are approximately three-quarters complete. Visits to four nuclear stations (6 units) were made during September for the purpose of lining up additional plants for the IPRD project.

Task 3: Generic System Definition. This task is complete.

Task 4: Data Analysis. In progress.

Task 5: Comparison with Other Data Bases. Complete.

Task 6: Human Error Data. Letter report from SAI "Development of Criteria for Pilot Plant Data Records Selection for Extraction of Potential Human-Related Maintenance Error from In-Plant Maintenance Records," was submitted to ORNL for review.

MEETINGS AND TRIPS:

R. J. Borkowski and J. P Drago met with personnel from the following plants to discuss and invite them to participate in the In-Plant Reliability Data project:

D. C. Cook Nuclear Plant in Bridgman, Michigan (2 PWRs)
Prairie Island Nuclear Plant in Welch, Minnesota (2 PWRs)
Monticello Nuclear Plant in Monticello, Minnesota (1 BWR)
Duane Arnold Energy Center in Palo, Iowa (1 BWR)

All plants were interested in cooperating with the IPRD program. Additional funding in FY-1983 would be needed to encode the data from these plants.

REPORTS, PUBLICATIONS, AND PAPERS: None.

PROBLEM AREAS: None.

PROGRAM TITLE: Common Cause Failure Analysis Procedure

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #41 88 55 03 0 (189 #B0456) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

Testing of the common cause screening program is continuing. Modifications to program output formats are being made to allow more efficient user access to the information provided by the computer program. Task 1 is on schedule.

We have received permission from Arkansas Power and Light Company to access plant information used in support of the ANO-1 IREP analysis and to visit the plant. A visit to the Sandia National Laboratories has allowed us to obtain plant drawings, operating and maintenance procedures, and complete IREP documentation for the ANO-1 power plant. The IREP fault trees for plant systems are currently being modified as necessary for the common cause failure analysis.

The IREP Procedures Guide has been received by the project staff. This documentation supports the review of the NREP Procedures Guide for defining common cause failure analysis procedures.

Fault trees and additional information for the ANO-1 scram system common cause failure analysis were obtained during the Sandia visit. Further development of the scram system fault trees is underway to extend the detail of the analysis. Additional information requirements have been identified and requested from Arkansas Power and Light Company.

MEETINGS AND TRIPS:

D. J. Campbell visited the Sandia National Laboratories to obtain information for the ANO-1 common cause failure analysis.

REPORTS, PAPERS AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Definition of Scenarios and Evaluation of Methods for Analyzing Source Terms of Major Accidents Involving UF_6 at NRC-Licensed Fuel Cycle Facilities

PROJECT MANAGER: M. Siman-Tov

ACTIVITY NUMBER: ORNL #41 88 55 05 6 (189 B0495-2) NRC 60 19 21

TECHNICAL HIGHLIGHTS:

Task 1. Literature Review and Scenario Identification

Task 1A. A draft report documenting a study of postulated UF_6 releases was recently received from Exxon Nuclear, Inc. This paper provides some useful information applicable to NRC's Accident Analysis Handbook (AAH). Unless additional NRC documents are discovered, this activity is complete and is being documented in the interim progress report now in preparation.

Task 1B-1E. These activities are complete and are being documented in the interim progress report.

Task 1G. A preliminary review of available methods for developing UF_6 source terms is being summarized in the interim progress report. Additional UF_6 source term methods presented at a 1978 conference entitled " A Specialists' Meeting on Safety Problems Associated with the Handling and Storage of UF_6" will be discussed in the report due to NRC in December.

Task 1H. As of September 30, 1982, the interim progress report was being reviewed by UCC-ND and DOE management. The report will be forwarded to NRC as soon as all comments have been incorporated into a revised report.

Task 3. Review of Analytical Models

Task 3A-3C. Preliminary results of these activities will be summarized in the interim progress report. Additional UF_6 source term models will be discussed in the December report.

Task 4. Preparation of Final Draft Documentation for Inclusion in the NRC Accident Analysis Handbook

Task 4A. Flow diagrams of UF_6 handling processes at NRC-licensed UF_6 production and at fuel fabrication facilities are being provided in the interim progress report.

MEETINGS AND TRIPS:

M. Siman-Tov attended the Fuel Cycle Risk Assessment Program meeting September 15-16, 1982, at Rockville, Maryland. This meeting provided us an overview of ongoing NRC fuel cycle risk assessment activities.

REPORTS, PAPERS AND PUBLICATIONS:

The September interim progress report is being reviewed by management and should be forwarded to NRC early in October.

PROBLEM AREAS:

None.

PROGRAM TITLE: Evaluation of Pressurized Thermal Shock

PROGRAM MANAGER: J. D. White

ACTIVITY NUMBER: ORNL #41 88 55 04 1 (189 #B0468)/NRC #60 19 51

Oconee - Unit 1

TECHNICAL HIGHLIGHTS:

Probabilistic Risk Analyses

Probability estimates for the branches on the Oconee PTS event trees have been completed with the following exceptions.

Branch	Status
PORV lift/RT	Information requested from G. Hollohan, ORAB, NRC
SG level controlled	Information requested from Duke by ORNL in August.
RCPs tripped/HPI	Human factors component--typical human error rates will be assumed initially
HPI throttled/HPI initiation	Human factors component--typical human error rates will be assumed initially

In addition, information concerning the impact of instrument air failures and vital bus failures is required to conclude work on these unique initiators. This information was requested from Duke by ORNL in August and has not been received.

The methodology for combining the probabilistic and mechanistic results of the program has been defined. This methodology will be presented to the NRC for review in a meeting on October 13, 1982.

Thermal-Hydraulic Modeling

Calculations were done by INEL (September 30) using RELAP-5 for ORNL-identified sequences involving main steamline break, steam generator overfeed, and small break LOCA (non-stagnant) transients. Results are presently under examination.

A TRAC-compatible version of the Babcock and Wilcox integrated control system was constructed and implemented by LANL. Some further work on the implementation is needed before transients can be calculated.

Calvert Cliffs - Unit 1

Task 1 - Modeling the plant - The total plant is being modeled by LANL. ORNL and SAI are providing assistance, principally in the modeling of the feedwater and control systems. This work is underway. LANL anticipates a one month delay (until April 1) due to difficulties in control system modeling.

Task 2 - Identify event sequences - ORNL is studying plant and instrument drawings, safety analysis reports, and operating procedures for Calvert Cliffs - Unit 1. This work is on schedule.

Task 3-8 - Not scheduled to begin until later dates.

Task 9 - Review models and results with plant owner - Modeling techniques and program strategy were presented in a meeting with representatives from Baltimore Gas & Electric, Combustion Engineering, LANL, INEL, NRC, and ORNL on September 21.

MEETINGS AND TRIPS:

September 21 (ORNL): reviewed data concerning Calvert Cliffs-Unit 1; identified additional data needs; reviewed proposed plant modeling and fracture mechanics techniques; reviewed program strategy.

September 22 (Oak Ridge): met with LANL, INEL, and NRC staff to discuss progress on Oconee PTS study; identified needs for additional information transfers; identified all 12 transients to be calculated.

REPORTS, PAPERS, AND PUBLICATIONS

None

PROBLEM AREAS

None

PROGRAM TITLE: LWR Accident Sequence Precursor Study

PROGRAM MANAGER: Wm. B. Cottrell

ACTIVITY NUMBER: ORNL #41 88 55 02 6 (189 #0435)/NRC 60 19 03

TECHNICAL HIGHLIGHTS

This program involves the review of licensee event reports (LERs) which have occurred starting in 1969, in order to identify potential accident precursor sequences. In order to identify precursor sequences of interest from the total LERs, it was necessary to develop appropriate criteria both for precursor sequences and for screening the large number of LERs to eliminate those of lesser significance. The precursor sequences thus selected require in-depth evaluation — including, for example, event tree analysis. The assessment was undertaken in two phases: the first phase included those LERs which occurred in 1969 1979; the second phase is for the LERs occurring in 1980 and 1981.

The work is currently organized into 7 tasks. During September, activities were concentrated primarily as follows:

Task 3: In-depth review of 1980 and 1981 LERs has been completed. The results of these reviews are being checked and write-ups on LERs selected as precursors are in typing.

A detailed response to comments received on the ASP report from Dairyland Power Cooperative (LaCrosse) was prepared.

MEETINGS AND TRIPS

W. B. Cottrell and J. W. Minarick met with F. M. Manning (NRC) on September 23rd to discuss expected work under the ASP program during the next fiscal year.

REPORTS, PAPERS, AND PUBLICATIONS

None

PROBLEMS

None

PROGRAM TITLE: LWR System Survey for PRA

PROGRAM MANAGER: Wm. B. Cottrell

ACTIVITY NUMBER: ORNL #41 88 55 02 4 (189 #B0431)/NRC #60 19 03

TECHNICAL HIGHLIGHTS

This program provides a simple and concise file of specific plant descriptive data on existing LWRs for easy information retrieval by NRC personnel. Descriptive information such as amount of redundancy, diversity, type, location and arrangement of key components and systems, which are especially important to safety reliability and risk analysis, are compiled on key safety systems.

Task A of this program includes the collection of information identified in the information list and assistance to SRR in the review and revision of this list and computer format. Task B of this program includes various evaluations of the data base.

NRC forwarded $15,000 to permit the publication of an updated report, which will include the six Task B studies as appendixes. This work in updating the report was essentially completed in August and the revised draft submitted for final editing prior to publication.

MEETINGS AND TRIPS

None

REPORTS, PAPERS, AND PUBLICATIONS

None

PROBLEM AREAS

None

PROGRAM TITLE: Mathematical and Statistical Problems in Risk Analysis

PROGRAM MANAGER: R. C. Ward/V. R. R. Uppuluri

ACTIVITY NUMBER: ORNL #41 88 55 03 0 (189 #B0444)/NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

Task 3: At the request of NRC, a computer program developed at Oak Ridge was mailed. This program assigns weights to k characteristics based on paired comparisons, and can be used either in an interactive mode or by the input of data in a block.

Task 4: In connection with the workshop on the Propagation of Uncertainties (POU), to be held at Oak Ridge during October 7-8, 1982 the final program was set.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS AND PUBLICATIONS:

None.

PROBLEM AREAS:

With the exception of funds to cover the POU workshop and a subcontract, all NRC funding has run out. At this time, there is no clear evidence that any additional funding will be forthcoming.

PROGRAM TITLE: The Impact of Truncation on IREP sequences.

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #42 88 55 05 9 (189 #B0496) NRC #6C 19 5 1

TECHNICAL HIGHLIGHTS:

The fault tree for the auxiliary feedwater system (AFWS) of the Calvert Cliffs that is developed in the IREP study is chosen for truncation tests. The truncation cut-off value is varied from 10^{-2} to 10^{-10}. The result showed that the probability of AFWS failure can be calculated with high accuracy by using a 10^{-7} cut-off. However, the time needed to calculate the top event probability with, say 10^{-9}, is more than 1000 times for that of 10^{-7} cut-off. With 10^{-9} cut-off the value of top event probability increased by only 5%. Clearly with uncertainties in the data base, 5% improvement in the top event calculation is very insignificant. Therefore, this example shows that one can establish limits for cut-off based on cost and accuracy of results needed.

Currently, the AFWS is being modularized in order to find the exact probability of top event. In addition, efforts are under way to study the impact of truncation on cut set size only. In our communications with Dale Rasmuson of NRC, we agreed that the cut set size cut-off can be a valid truncation method for small fault trees. Efforts are under way to use this method for large fault trees as well.

MEETINGS AND TRIPS:

A meeting on September 13 was held at the University of Maryland between D. M. Rasmuson of NRC and M. Modarres and H. Dezfuli of the University of Maryland. The status of this project was briefly reviewed. A trip was made by M. Modarres to participate in the International Meeting on Thermal Nuclear Reactor Safety. The meeting was held in Chicago from August 30 to September 1.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Utilization of Risk Analysis and Risk Criteria

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #40 10 01 06 5 (189 #B0458) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

A. K. A. Solomon (Rand) provided a "Work in Progress" Briefing to NRC on September 7 and 8. The briefing addressed two subjects:

1. The development of a decision rule for treating uncertainty in PRA's.

2. The development and application of a value impact methodology for addressing generic safety fixes.

Regarding the decision rule for dealing with uncertainty in PRA's, Rand has the following preliminary observations:

- Require that PRA's specify uncertainty distributions.

- Require that these distributions consider confidence intervals.

- Mandate that PRA's should demonstrate a high (98 percent or better) degree of confidence that the risk under assessment is smaller than the risk goal.

- Account for overestimates of certainty perception in PRA's.

Regarding the value impact work, Rand presented an abbreviated method for assessing value impact of generic safety issues. The Rand method considers direct exposure and dollar costs for the fix and the reduction in accident risk. Rand also factors in secondary costs, sensitivities, and discounting. Rand estimated the expected cost of about a dozen accident scenarios.

B. Rand provided to ORNL (letter of September 17, 1982, Solomon to Flanagan) FY 1982 Funds-NRC Budget Preparation. Rand identified four areas of continued research interest:

First, Rand proposes to continue their value impact assessment work--both further developing the data base and also assigning quantitative value impacts for a number of generic safety fixes. Rand initiated the value impact work in August of 1982.

Second, Rand proposes to continue its development of a decision tool for treating uncertainty in PRA assessments. Specifically, how should uncertainty be specified? What level of certainty is required for a PRA based decision? Rand initiated the uncertainty work in June of 1982.

Third, working closely with Decision Research and Oak Ridge National Laboratory, Rand will address a series of questions that constrain the implementation of PRA in decision making. These questions deal primarily with how risk and uncertainty is perceived and acted upon.

Fourth, Rand will continue to be available to review draft manuscripts and address relevant problems as they arise and by mutual agreement with ORNL and NRC.

MEETINGS AND TRIPS

On September 7 and 8 K. A. Solomon provided three briefings to approximately 25 members of NRC and ORNL at NRC. The substantive content of the briefing is reported above.

On September 9 K. A. Solomon also met with P. Slovic (Decision Research); G. Flanagan (ORNL); P. Rathburn (NRC) and others to develop a "Use of Risk Perception in the PRA Program." G. Flanagan summarized the findings of this meeting in a letter to P. Rathburn.

REPORTS, PAPERS, PUBLICATIONS

None

PROBLEM AREAS

None

DIVISION OF FACILITY OPERATIONS

PROGRAM TITLE: Bioassay Methods for Estimation of Internal Dose

PROGRAM MANAGER: K. F. Eckerman

ACTIVITY NUMBER: ORNL 60 19 31 (189 B0480-2)/NRC 60-82-103

TECHNICAL HIGHLIGHTS:

A draft version of the radiocesium report is now in typing. In comparison with most radioelements, a great deal is known about metabolic properties and retention times of the intracellular alkali metals, cesium, rubidium, and potassium. These elements are of particular interest not only because of the abundance of radiocesium as a long-lived fission product, but also because there is a great deal to be learned about retention models in general from intercomparisons of the wealth of tracer studies, bioassay results, and basic physiological studies available concerning intracellular alkali metals. This report is intended as a case study of the relationship of the ICRP 30 retention models and their underlying assumptions to the actual metabolism, retention, and excretion of the radionuclides. An attempt has been made to identify the factors leading to the variability in retention times that have been observed for cesium. Ways of incorporating consideration of biological variability into estimates of intake based on bioassay procedures have been suggested.

Work has been started on the development of mathematical formulations for translating bioassay-measurement results on plutonium and americium into estimates of intake. The physiological behaviour of these radioelements is much more complicated than that of cesium, and considerably less information is available on the metabolism of the actinides than on the metabolism of alkali metals. As with cesium, there is extensive recycling of plutonium and americium, but analysis of this recycling is complicated for the actinides because they may form complex chemical compounds in the body, thereby altering their metabolism. However, we feel that sufficient information is available to develop biologically based mathematical formulations of the retention of these elements.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Continuous On-Line Reactor Surveillance System Evaluations

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 12 8 (189 #B0442)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Task 1: In-Plant Demonstration of On-Line Noise Surveillance System. Magnetic tape recordings of 14 Sequoyah noise signals were obtained on August 18 and September 8. A recording was also made of the source range neutron detectors shortly after Sequoyah-1 was shut down for refueling on September 11, 1982.

A letter was sent to H. G. Parris, TVA Manager of Power by T. M. Novack, Assistant Director of Licensing, on August 4, 1982 requesting that automated noise surveillance be continued throughout the second fuel cycle. The current agreement between ORNL and TVA calls for the system to be removed after the first fuel cycle. TVA has not yet responded to the request by NRC.

In accordance with Milestone 1.C, a draft report, A Description of the Hardware and Software of the PSDREC Continuous, On-Line Reactor Surveillance System, was transmitted to NRC for review and comment.

Task 2: Procurement of NRC Surveillance System. No activity during this reporting period.

Task 3: Experiments in Test Facilities. Preparation of a test plan has been initiated for acquisition of additional data at LOFT in February 1983 in which fuel damage is expected.

Task 4: Data Analysis. Sequoyah neutron signatures obtained over the first fuel cycle were analyzed using funds supplied by NRR/DSI/CPB under FIN B0754. The results of this analysis were reported at a Research Review Group Meeting at NRC on September 20, 1982.

EQUIPMENT PURCHASES:

Purchase of signal conditioning amplifiers for analog tape recording of noise signals was initiated.

Fabrication of computer controlled amplifiers was completed and the amplifiers are currently being tested and calibrated.

System compatibility problems with the Winchester disk were solved and purchase of a unit for the Sequoyah system was initiated.

MEETINGS AND TRIPS:

We met with the NRC Research Review Group (RRG) on September 20-21, 1982 at NRC to report accomplishments during FY 1982 and plans for FY 1983.

REPORTS, PAPERS, AND PUBLICATIONS:

C. M. Smith presented a paper titled, "Performance of an Automated Noise Surveillance System at Sequoyah-1," at RRG meeting stated above. C. M. Smith submitted a paper titled, "Demonstration of a Noise Surveillance System at a PWR," for presentation at the Tenth Water Reactor Safety Research Information Meeting on October 13, 1982. D. N. Fry presented a paper titled, "Neutron Noise Analysis at Sequoyah-1," at the September RRG meeting.

A draft report, A Description of the Hardware and Software of the PSDREC Continuous, On-Line Reactor Surveillance System, was transmitted to NRC for review and comment.

PROBLEM AREAS:

The demonstration of automated noise surveillance at Sequoyah-1 will not resume after Sequoyah refueling unless TVA responds favorably to NRC's request to extend the demonstration.

PROGRAM TITLE: High-Sensitivity Radionuclide Analysis for Internal Dose Assessment

PROJECT MANAGER: J. R. Stokely

ACTIVITY NUMBER: ORNL #41 89 55 14 1 (189 #B0494)/NRC #60 19 31

TECHNICAL HIGHLIGHTS:

Task 1: Survey of Literature and Assessment - A computer search of Nuclear Science Abstracts was made for background information on analytical separation methods used in bioassays of uranium, plutonium, and thorium in human urine. This search was directed toward discovery of information that will be of value in the analytical chemistry of Task 3. The results of the search are being examined. A similar search of Chemical Abstracts is planned.

Task 2: Neutron Activation Methods - Efforts were concentrated on ways to prepare samples of urine for uranium determination by neutron activation analysis (NAA) with delayed neutron counting (DNC). As used at ORNL, the smallest amount of natural uranium that can be determined by DNC is $\sim$25 ng ($\sim$0.2 ng ^{235}U). Since the normal level of uranium in urine has a range of $\sim$0.05 to $\sim$5 ng/ml, the volume necessary for analysis is $\sim$100 to 500 ml. Initial attempts to simply dry samples in plastic bags and irradiate the bag and residue proved impossible because the volume of the resulting sample was too large for the irradiation container (rabbit volume is 1-2 ml). Samples of our "reference" urine have now been prepared by wet ashing 100 ml in a teflon beaker and transferring the residue to rabbits for analysis. Analyses of these samples will soon be made and the results used to help evaluate findings in fission track counting methods in Task 4. The determination of thorium in these samples by NAA is planned to aid in evaluating NAA for thorium determination and to provide a standardized urine sample for use in Task 3.

Task 3: High-Sensitivity Mass Spectrometry and Resin Bead Methodology - Experimental work in this task will begin in November. Present efforts are limited to examination of results of literature searches.

Task 4: Fission Track Etch Determination of Fissile Nuclides of Uranium and Plutonium - After some amount of preparation, we are now in a position to study fission track measurements of uranium in our "reference" urine sample. Samples to be irradiated consist of two Lexan discs that enclose residue from urine evaporation. Preparative work included making a die to cut Lexan discs that would fit our irradiation containers, devising a technique of holding the discs in contact with the residue (and preventing residue loss), cutting and cleaning discs, and setting up a glove box for sample evaporations. Although the fission track method of analysis appears to have adequate sensitivity to measure uranium in nearly all urine samples, solids in the evaporated residue may prevent some fission fragments from reaching the Lexan. We are in the process of evaluating this potential problem by using urine samples containing known quantities of uranium and comparing the results with similiar samples without residue.

It was recently brought to our attention that the Product Certification Division of Y-12 have several Quantimet Image Analyzers. These instruments are computer controlled and have been used for semiautomatic fission track counting. A series of samples of Lexan with a wide range of known track densities have been submitted for track counting. Analysis of a similar series of samples obtained from neutron irradiation of urine samples is planned. If such automated track counting proves successful, our evaluation of fission track methods will be greatly enhanced.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Human Factors in Incident Alert Notification

PROGRAM MANAGER: John H. Sorensen

ACTIVITY NUMBER: ORNL # 41 88 55 05 4 (189 #B0490)/NRC 60 19 31

TECHNICAL HIGHLIGHTS:

Task One: Procedure Review

Work continued on collecting written procedures from utilities. A sample of power plants to be examined was identified for inclusion in the study.

Task Two: Information Discussions

Discussions were held with personnel at the Peach Bottom Power Plant concerning problems and issues in the accident classification and off-site notification procedures. Meetings were being set up with personnel at Davis Beese and North Anna Power Plants.

MEETINGS AND TRIPS:

Jonathan Morell visited the Peach Bottom Power Plant during the second week of September.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Maintenance Error Model

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #41 88 55 03 6 (189 #B0461) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

Approximately 80 of the 144 job analysis questionnaires (56%) for the electrician position have been received. Follow-up communications have been made to nuclear power plants that have not yet returned their questionnaires. Preliminary analysis of this data will be initiated during the next report period.

Progress was made in the definition and operationalism of:

(1) input parameters, value ranges, and default values
(2) task and subtask input data, value ranges, and default values
(3) input menus to be displayed to the model user
(4) pre-processor module for calculation of average subtask duration
(5) selection module for selecting team members for the work group
(6) simulation module including calculation of subtask success probabilities, stress, subtask re-entry, and error detection logic
(7) fatigue and fatigue recovery modules
(8) communication effectiveness module.

MEETINGS AND TRIPS:

Bill Knee, a member of the ORNL staff, met with members of the staff of Applied Psychological Services, Inc. (APS) in Wayne, Pennsylvania on September 14, 1982, to review the progress of the development of the maintenance model.

E. W. Merschoff (NRC/RES), Technical Monitor of this program, reviewed its current status and FY-1983 direction at a working group meeting held at ORNL on September 22, 1982.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Noise Diagnostic Methods for Safety Assessments

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #B0191)/NRC #60 19 11 0

TECHNICAL HIGHLIGHTS:

Task 1 Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. We completed work on assessment of ex-core neutron detector sensitivity to fuel element vibrations and compared code predictions with neutron noise signatures from Sequoyah-1. A draft NUREG report of this work was completed and is receiving in-house review.

Task 2: Surveillance and Diagnostics by Noise Analysis. A program plan for assessment of temperature noise as a core diagnostic tool was transmitted to NRC in accordance with Milestone 2.i.

We continued studies of the effect of pressure sensors and sensing lines on pressure noise signals used for core diagnostics.

Task 3: Primary Water Inventory Surveillance. No activity this reporting period.

Task 4: Evaluate New Surveillance and Diagnostic Methods for Reactor System Fault Detection. We completed initial evaluation of a modelling methodology used to construct dynamic models of primary system components. The modelling method is modular thus providing capability to add models such as those used in our study of pressure, temperature, and neutron noise. The current model of a PWR primary system is being evaluated with data from Sequoyah-1.

We also initiated an assessment of heuristic learning techniques which when used in conjunction with the noise models could greatly aid noise diagnosticians in systematically identifying and monitoring noise sources in the primary system.

MEETINGS AND TRIPS:

Jean Guitton, head of an Electricite de France (EDF) section responsible for noise and loose parts monitoring at 22 EDF PWRs in France, visited ORNL in August to discuss exchange of information between his group and ORNL in the area of reactor diagnostic methods and experience. We hope to initiate an exchange program in FY 1983.

We met with the NRC Research Review Group (RRG) at NRC on September 20-21, 1982 to report accomplishments during FY 1982 and discuss the research plan for FY 1983.

REPORTS, PAPERS AND PUBLICATIONS:

The following papers were presented at the RRG meeting on September 20:

F. J. Sweeney, "Feasibility of Detecting Fuel Element Vibrations and Coolant Boiling Using PWR Ex-Core Neutron Noise";

J. A. Mullens, "Evaluation of Pressure Noise for Nuclear Plant Diagnosis";

R. B. Perez, "Modeling for PWR Noise Surveillance and Diagnostics"; and

F. J. Sweeney, "Temperature Noise for Nuclear Plant Diagnosis".

F. J. Sweeney and J. P. Renier submitted a paper titled, "Sensitivty of Ex-Core Neutron Detectors to Vibration of PWR Fuel Assemblies," to the 7th International Conference on Structural Mechanics in Reactor Technology, to be held August 22-26, 1983 in Chicago, Illinois.

PROBLEM AREAS:

None.

PROGRAM TITLE: NPP Personnel Selection and Training

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #40 10 01 06 (189 #B0466) NRC #60 19 3 1

TECHNICAL HIGHLIGHTS:

Task 1: Program Plan. An outline of the basic evaluation model has been completed. This evaluation model consists of five segments, i.e., evaluation of: 1) training objectives, 2) media selection, 3) course curriculum, 4) internal evaluation, and 5) external evaluation. Development of specific evaluation guidelines and checklists for each of these segments has been initiated.

The first report concerning performance shaping factors was received from Eclectech Associates. This report was reviewed and returned to Eclectech for further work. It was felt that both the data and the report organization needed further development.

The proceedings from the August review group meeting have now been transcribed from the tapes which were made during the meeting. The approximately 100 pages of material obtained from the tapes are being reviewed by the meeting attendees. Once their comments are received, the proceedings will be condensed and summarized before distribution.

Task 2: Evaluation and Upgrading of Simulators. Examination of the Nuclear Safety Information Center (NSIC) data files to develop estimates of frequency of occurrence of various malfunctions continued. These malfunctions are now being categorized by system for easy reference purposes. Approximately 40 systems have been identified.

MEETINGS AND TRIPS:

A meeting was held at ORNL with the NRC Cognizant Monitor. Both present and future work under this program were discussed. Problems with next year's work statement were identified and resolved.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

Program Title: Occupational Radiological Monitoring at Uranium Mills

Program Manager: C. S. Sims

Activity Number: ORNL #41 88 55 04 9 (189 #B0485)/NRC #60 19 31

Technical Highlights:

The first draft of Chapter 4 (External Radiation Monitoring) of the manual entitled "Occupational Radiological Monitoring at Uranium Mills" has been transmitted to the NRC for review.

Review comments on the first draft of Chapter 1 (Introduction) have been incorporated, along with a format change, in the second draft of the chapter and the chapter has been returned to the NRC for additional review.

The drafts of both Chapters were sent to Jack Bell who replaced Steve McGuire as the NRC Project Manager effective September 1, 1982.

MEETING AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Operational Aids for Reactor Operators

PROGRAM MANAGER: W. H. Sides

ACTIVITY NUMBER: ORNL #41 88 55 02 8 (189 #B0438-9) NRC #60 19 50 01 0

TECHNICAL HIGHLIGHTS:

Task 1 and 2: Evaluation of Engineered Safety Features (ESF) and Criteria for Allocation of Function

Principal effort during this period was on Tasks 2 and 4 described in the Program Plan ("Evaluate ESF's" and "Criteria/Method.") BioTechnology is continuing visits to plants to collect data on the design and operation of engineered safety features. Delays in access are continuing also, as previously reported. B. Paramore, H. Van Cott and J. Debor visited Palo Verde during September 7-18; H. Price and D. Taylor began a series of visits to Duke Power Company, studying design practice. Visits were to HQ Duke Power Company, Charlottesville, VA., Oconee, McGuire and Catawba. A visit to Surrey is now programmed for October 18-25, to Limerick (revisit) November 1-12, and to Rancho Seco November 29 to December 10.

Allocation of functions to man or machine is driven in part by assumptions concerning the relationship of man, plant and computer, relationships which can be represented by a formal model. Definition of this model was aided by the recent NRC workshop on modeling. We now are able to identify an underlying agreement among professionals that suggests a commonly agreed model, which we will use in this project. This model was discussed at a recent ORNL meeting. A paper is now completed in draft and will be presented at the 10th WRSR October 12. The model is based in part on a decision matrix: decisions at any point in the allocation process will be governed by a two-variable model.

Task 3: Effects of Changes in Automation on Operator Performance

A literature survey has been completed and an analysis of it is underway. A report of the findings will be forwarded to NRC early in October. Thus far the open literature has not contained much on the effects of changes in automation on operator performance.

Task 4: Complete the Review and Assessment of Operational Aids

An analysis of the diverse and sparse data received on ten operational aid systems has resulted in the generation of 55 pages of formatted data. After crosscomparing the parameters and characteristics of the aids some generalizations were made. These generalizations will be presented at the 10th WRSR, October 12. A draft report of the data collection and analysis was forwarded to the NRC project monitor September 30. We are hoping to complete the report in November, and it will serve as an interim database.

Task 5: Man-Machine Interface Study

A first draft of the Proceedings of the Cognitive Modeling Workshop will be essentially complete by the end of September. It will consist of the papers (eleven of them, which have been edited for logical consistency, parallel presentation, and grammar) and the working group conclusions and recommendations. Following review by the authors, working group chairmen, and conference chairman, additional drafts will be circulated for review as needed. We are aiming at end of November for a publication data.

MEETINGS AND TRIPS:

September 23: R. A. Kisner and P. R. Frey met with R. Pulliam of BioTechnology to discuss new findings.

REPORTS:

R. A. Kisner, "Review of Operational Aids for Nuclear Plant Operators," draft, September 30, 1982.

PROBLEM AREAS:

None

PROGRAM TITLE: Organizational Interface in Reactor Emergency Preparedness

PROGRAM MANAGER: John H. Sorensen

ACTIVITY NUMBER: ORNL #41 88 55 05 5 (189 #B0491)/NRC #60 19 31

TECHNICAL HIGHLIGHTS:

Task One: Specification of Roles and Responsibilities

Two sets of Emergency Plans were analyzed to determine the consistency of specified roles and responsibilities (North Anna and Ft. St. Vrain). Findings indicate inconsistency in the designation of "Primary" and "Secondary" Responsibilities among plans.

We continued to collect plans from states, local government and utilities to use in the analysis. Federal Agency plans will be analyzed when they become available (probably in November).

Task Two: Dynamics of Interface

The literature review was completed. A draft is available upon request.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

A draft report, "Organizational Behavior and Interactions in Emergencies" was completed by Dennis Mileti.

PROBLEM AREAS:

None.

PROGRAM TITLE: Pressure Sensor/Sensing Line System Evaluation Research

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 88 55 05 2 (189 #B0481)/NRC #60 19 31

TECHNICAL HIGHLIGHTS:

Task 1: Review Status of Pressure Sensor/Sensing Line System Standards, Practices and Technology. We visited five additional utilities in September (ten visited to date) to discuss such topics as type of pressure transmitters used, experience with failures, calibration, quality assurance and installation procedures.

A report summarizing the results of our review of licensing event reports will be submitted to NRC by the end of October.

Our attempt to construct a facsimile of a pressure sensing system has been temporarily halted because of a lack of knowledge on the length of the sensing line. We propose to continue work on simulation of a pressure measurement system by estimating sensing line length and then varying the length to assess the effect of length and bends on response time.

Task 2: Pressure Sensor/Sensing Line Model Development. Work was completed on an empirical study of response characteristics of a Foxboro sensor diaphragm.

We tested a low frequency model of a pressure measurement system with air in the sensing line by comparison with results from the laboratory test facility.

A high frequency model (where standing waves are significant) was formulated and will be tested in the lab.

We developed a technique to assure that all air is removed from the sensing line and sensor before testing in the lab.

Task 3: Evaluate an In-Situ Method for Remote Detection of Pressure Sensor/Sensing Line Degradation. We reviewed and modified a test procedure for remote in-situ response time testing of a Foxboro force-balance pressure sensor. This procedure will be transmitted to NRC for review in October 1982.

A plan for testing noise analysis method to remotely detect pressure sensor/sensing line degradation was transmitted to NRC for review and comment in September 1982.

Tests were conducted on the ORNL pressure test facility to evaluate a pressure perturbation method ("burp" test) for in-situ testing of pressure sensing systems. The results were encouraging so ORNL is preparing a program plan and cost estimate to evaluate this method at PWR primary pressure (~2000 psi) in a test facility and in a commercial PWR.

A Foxboro nuclear qualified NE11GM pressure transducer was received in August 1982. It was calibrated in the ORNL standards lab and tested in the ORNL low pressure test facility.

MEETINCS AND TRIPS:

Trips were made to five utilities in September 1982 to discuss their experience and practices in pressure sensing.

We met with the NRC Research Review Group (RRG) on September 20-21, 1982 at NRC to report accomplishments during FY 1982 and discuss plans for FY 1983.

REPORTS, PAPERS AND PUBLICATIONS:

The following papers were presented at the RRG meeting on September 20:

"Degradation of Pressure Sensing Systems" by J. A. Mullens

"Transient Performance of Pressure Sensors" by T. W. Kerlin

"Pressure Measurement System Modeling and Testing Including Sensing Line Effects" by J. A. Mullens, J. A. Thie and T. W. Kerlin

PROBLEM AREAS:

None.

PROGRAM TITLE: Safety Implications of Control Systems

PROGRAM MANAGER: R. S. Stone

ACTIVITY NUMBER: ORNL #41 88 55 03 8 (189 #B0467)/NRC #40 10 01 06 5

TECHNICAL HIGHLIGHTS:

Task A: Program Planning - A formal program planning session under the direction of M. Becktell of Sandia National Laboratories is scheduled for Nicholson Lane on October 14-15.

Task B: Generic Study (work applicable to all plants) - Work continued on coupling together the thermodynamics, hydraulics, neutronics, pump, and water properties routines of the primary loop. Several additions were made to the water properties packages to improve mathematical efficiency; these include the derivatives of density with respect to pressure and internal energy for single-phase flow, and derivatives of internal energy and density with respect to pressure for two-phase flow. The specific heat, density, and thermal conductivity of Inconel-600 (the steam generator tube material) were added.

Following the approach of RETRAN, the pressurizer was mathematically divided into two regions, vapor and liquid. Equilibrium is assumed in each region, with nonequilibrium allowed between regions to simulate the important nonequilibrium characteristics of pressurizer transients.

Task C: Babcock and Wilcox Analysis - Development of the Integrated Control System model focused on the turbine and turbine bypass control portions of the Integrated Master Subsystem. Lack of plant specific information continued to slow progress and necessitate speculation on ICS design and operation. Turbine control is assumed to regulate the throttle valve according to a linear relationship between valve area and stem position. Stem position is controlled by an error signal that is a function of pressure, load, and frequency errors. The gain of this loop is unknown; a gain to achieve a rate of change not to exceed 2%/min is assumed.

The bypass control system provides a means of relieving high turbine pressure and is based on an error signal that is biased by either 50 psi or 125 psi depending on plant conditions. This modified error signal is used as input to a proportional plus reset controller whose output is then compared with the error in steam generator pressure. The larger of these pressure errors is used to control the bypass valve, with flow assumed proportional to stem position. The overall gain for this loop is unknown; a value that allows a 10 psi/min change is assumed.

To improve computer running time, the six-feedwater heater system of the Oconee plant is being lumped into two equivalent heaters in ORTURB. Computer runs were made with both the six- and two-feedwater heater models to establish criteria for lumping. Heater groups (A, B, and C) and (D, E, and F) were collapsed, respectively, into a single high and a single low pressure heater. This natural grouping follows the coupling of the A, B, and C heaters to the high pressure turbine, and D, E, and F to the low pressure turbines. Steady state flows in the two-heater model were adjusted to reproduce the following parameters of the six-heater model: outlet and inlet steam generator flow, inlet steam generator temperature, pressures and enthalpies along the high and low pressure turbines, flashing flow from the flash tank, and feedwater temperature at the exit of heater D. Vapor holdup times in the high and low pressure heaters were adjusted to reproduce the stored masses of vapor. The lumping criteria were tested in transients with a 20% reduction from full power in one minute, followed by 45 seconds at 80% power and then return to full power in one minute. During the transients, the six-heater and two-heater feedwater temperature and pressure at the steam generator agreed within 2 psi and 1°F. The criteria appear to achieve an equivalent feedwater heater system when viewed at the steam generator.

The matrix exponential and the LSODE methods were used to solve the seven coupled differential equations in each feedwater heater. Upon completion of the comparison, the faster of the methods will be chosen. Extensive runs were made, varying the duration of the time step and the holding time of the vapor in the shell side of the low pressure heater, both of which strongly influence computer running time. Although developed in the context of Oconee-1, decisions made here will apply generically to future plant modeling.

Task D: Study of Second PWR - Talks with Baltimore Gas and Electric continue. See Meetings and Trips section.

Task E: Criteria - No activity this month.

MEETINGS AND TRIPS:

On September 1-3 R. S. Stone remained in Sandia for the final portion of the electrical system program planning session which began on August 31, and which was reported last month. Work flow diagrams and computer printouts concerning the results of program planning for the plant electrical system have been received from Sandia's Project Management Support Division.

On September 21 N. E. Clapp met with SAI personnel and with visiting analysts from LANL, here for a joint meeting with Baltimore Gas & Electric staff members and with ORNL engineers on the Pressurized Thermal Shock program. The objective of the Clapp-SAI involvement was to resolve problems with the controls portion of LANL's Oconee-1 model, on which LANL has requested assistance. (Ned Clapp is doing the ICS modeling for the Safety Implications of Control project.)

Despite considerable progress made at the Oak Ridge work session, LANL still wants Clapp's presence in Los Alamos at an early date. He will probably leave October 4 and will remain the rest of that week. Our major concern is that the non-controls portion of the TRAC program be running beforehand, so as to make effective use of Clapp's expertise. Despite heavy LANL dedication to that task, it is not clear to us that goal will be met by October 5.

On September 27 R. S. Stone joined D. L. Basdekas, K. R. Goller, E. C. Wenzinger, and A. J. Szukiewicz of NRC for a meeting with Baltimore Gas and Electric (BG&E) personnel at BG&E Headquarters in Baltimore. The purpose of this meeting was to discuss the cooperation of BG&E in a review of the safety aspects of control systems in the Calvert Cliffs nuclear power station. Ten members of the utility staff attended the meeting, chaired by R. C. L. Olson of BG&E's Licensing And Analysis Unit. Stone made a commitment to provide complete details on one category of data requests. This will give BG&E more insight into just what they are agreeing to supply if they decide to cooperate.

REPORTS, PAPERS, AND PUBLICATIONS:

Papers by O. L. Smith and R. S. Stone were submitted for the Tenth Water Reactor Safety Research Information Meeting, October 12-15, 1982.

PROBLEM AREAS:

Access to detailed design data on each of the plants to be studied depends upon a working relationship with the utility concerned. Such a relationship does not presently exist for any plant.

Most of our present milestones have slipped by 2-3 months, and under present conditions may continue to slip. Time taken from productive work to seek data, inefficient programming because of missing information, unscheduled meetings and trips of key people, subcontractor problems, all have contributed to slowing the work pace. A best effort has been made and has resulted in substantial accomplishments. Some of the obstacles are removable, some may not be. Future scheduling should make ample provision for undefined but inevitable nonproductive activities. This issue should be addressed at the program planning meeting in October.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #40 10 01 06 01 (189 #B0421) NRC #60 19 11 01 2

TECHNICAL HIGHLIGHTS:

Task 1: Operator Response Time. The revised report on initial BWR simulator experiments (NUREG/CR-2534) was completed and transmitted for Laboratory approval and printing. The General Physics (GP) draft of the report on calibration of initial simulator results with field data has been reviewed and revised. It is ready for final typing. A final draft of the informal report on BWR field data collection is in preparation and is expected to be transmitted to NRC in October. Data reduction and analysis of the FY-82 PWR and BWR simulator experiments and preparation of a draft report by GP is still in progress. Plans were developed with GP for continued field data collection and completion of the FY-83 program. Final contract negotiations will have to be delayed until an authorization for funding is received.

Task 2: BWR Task Analysis. A rough draft of a NUREG/CR report summarizing the results of the task analysis and potential applications of the data was received from GP.

MEETINGS AND TRIPS:

E. W. Merschoff, the NRC Cognizant Monitor for this program visited ORNL September 22, 1982 to review this and other human factors programs under his direction.

REPORTS, PAPERS, AND PUBLICATIONS

None.

PROBLEM AREAS:

None.

DIVISION OF HEALTH, SITING AND WASTE MANAGEMENT

PROGRAM TITLE: CONCEPT/OMCOST Code Development

PROGRAM MANAGER: H. I. Bowers

ACTIVITY NUMBER: ORNL No. 41 88 55 03 4 (189 No. B0454)/NRC No. 60 19 01 30

TECHNICAL HIGHLIGHTS:

1. Nuclear Power Plant O&M Costs and Estimating Guidelines. The final report was printed and distributed.

2. Trends in Power Plant Capital Investment Cost Estimates. Preparation of tables and graphs for Section 3 — Review of NRC/DOE Capital Investment Cost Studies — 1976 to 1982 continues at a very low level of effort.

3. Validation of Cost Models in the CONCEPT Code. No activity this month.

MEETINGS AND TRIPS: Howard Bowers and Jerry Delene met with NRC staff in Bethesda on September 15 to discuss the FY 1983 effort on fuel cycle economics.

REPORTS, PAPERS, AND PUBLICATIONS: NUREG/CR-2844, ORNL/TM-8324, Nonfuel Operation and Maintenance Costs for Large Steam-Electric Power Plants — 1982, was published.

PROBLEM AREAS: United Engineers & Constructors (UE&C) has notified us that completion of the January 1982 update of the Energy Economic Data Base has been re-scheduled to mid-November. Therefore, there will be some delay in completing Tasks 2 and 3. We will re-examine the schedule when we receive the needed information from UE&C.

<u>PROGRAM TITLE</u>: Environmental Dose Indices

<u>PROGRAM MANAGER</u>: K. F. Eckerman

<u>ACTIVITY NUMBER</u>: ORNL 40 10 01 06 (189 B0477-2)/NRC 60-82-117

<u>TECHNICAL HIGHLIGHTS</u>:

Elemental transfer parameters for the terrestrial foodchain have been adopted from the work of Baes *et al*. (ORNL-5786, to be published). In that document, the soil-to-plant factors are expressed in terms of dry weight (Reg. Guide 1.109 used a wet-weight base for it's tabulations). Baes *et al*. have constructed factors for both protected vegetation (fruits, seeds, tubers, etc.) and unprotected vegetation (foliar portion). The following factors permit conversion of the transfer factors to a wet-weight basis:

Exposed produce	- 0.13
Protected produce	- 0.22
Grains	- 0.89

Tabulated below is a comparison of the newer values with those currently being used in the Reg. Guide 1.109.

Soil-to-Plant Transfer Factors (wet-weight)

Element	New Values			R.G. 1.109
	Exposed	Protected	Grains	
Co	2.5×10^{-3}	1.6×10^{-3}	6.2×10^{-3}	9.4×10^{-3}
Sr	3.2×10^{-1}	5.6×10^{-2}	2.2×10^{-1}	1.7×10^{-3}
I	1.3×10^{-1}	2.2×10^{-1}	8.9×10^{-1}	2.0×10^{-2}
Cs	1.0×10^{-2}	6.7×10^{-3}	2.7×10^{-2}	1.0×10^{-2}
Np	1.3×10^{-2}	2.2×10^{-3}	8.9×10^{-3}	2.5×10^{-3}

As seen from this tabulation, the values for strontium, as well as those for neptunium, are higher than those currently used in the Reg. Guide. These increases are generally recognized as the current thinking, particularly for strontium.

Transfer factors for meat and milk have also been recommended by Baes *et al*. The transfer to milk factors were largely based on the recent extensive review by Ng *et al*. (UCRL-S1939). The transfer to meat factors are based on review of the literature. Tabulated below is a comparison for selected elements of the new values with those of Reg. Guide 1.109. Note the increased transfer of cesium to meat, an increase indicated by other analyses. Note also, that many of the other changes are generally within a factor of two.

Milk and Meat Transfer Factors

Element	Meat Baes	Meat 1.109	Milk Baes	Milk 1.109
Co	2×10^{-2}	1.3×10^{-2}	2×10^{-3}	1.0×10^{-3}
Sr	3×10^{-4}	6×10^{-4}	1.5×10^{-3}	8.0×10^{-4}
I	7×10^{-3}	2.9×10^{-3}	1.0×10^{-2}	6.0×10^{-3}
Cs	2×10^{-2}	4.0×10^{-3}	7.0×10^{-3}	1.2×10^{-2}
Np	5.5×10^{-5}	2.0×10^{-4}	5.0×10^{-6}	5.0×10^{-5}

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Evaluation of Atmospheric Dispersion Models

PROGRAM MANAGER: F. C. Kornegay

ACTIVITY NUMBER: ORNL #41 89 55 13 1 (189# B0446)/NRC #60 19 12 01

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration: F. C. Kornegay discussed the progress of the Model Evaluation Program with R. F. Abbey. The lack of analyzed data from the INEL and SEADEX programs will cause some delay in our analyses. If the delays become significant, changes in the program schedule will be required.

Task 3: Meteorological Data: C. Gilmore of ORNL was sent to INEL to assist the NOAA office in analyzing data from the field tests. The data tapes were all checked and corrected so that a final, complete data tape from the Grid 3 tower can be made.

Task 4: Atmospheric Dispersion Models: All model runs for the available INEL field data are complete, and have been received at ARAP where pattern recognition testing is almost complete.

MEETINGS AND TRIPS:

C. Gilmore to INEL to assist in data analysis, 9/16-9/21/82.

REPORTS, PAPERS, PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Forecasting Electricity Demand by States

PROGRAM MANAGER: D. M. Hamblin

ACTIVITY NUMBER: ORNL #41 88 55 01 3 (189 #B0190)/NRC #60 19 32 02

TECHNICAL HIGHLIGHTS:

Task 1. We continued to work on reestimating Version II of SLED (to be called Version III) using the updated data base and the new employment variables in the industrial sector. Because of problems we have encountered, we are about one month behind on this task.

Task 2. We continued to consult with Ed Hudspeth, of the Alabama Department of Energy, concerning USAD. Plans have been made for Hudspeth and his colleague, Jan Slatton, to visit Colleen Rizy at ORNL on October 7.

MEETINGS AND TRIPS:

Dan Hamblin and Colleen Rizy participated in the National Regulatory Research Institute's CERES & SLED Workshop in Columbus from September 27 to September 29. Details of the Workshop can be found in the trip report, transmitted under separate cover.

REPORTS, PAPERS AND PRESENTATIONS:

None.

PROBLEM AREAS:

We are about one month behind on the reestimation of SLED because of time consumed in preparation for the workshop.

PROGRAM TITLE: Internal Dose for Specific Occupational Exposure Conditions

PROGRAM MANAGER: K. F. Eckerman

ACTIVITY NUMBER: ORNL 40 10 01 (189 B0475-2)/NRC 60-82-103

TECHNICAL HIGHLIGHTS:

Committee 2 of the ICRP did not tabulate secondary limits for natural uranium in it's Publication 30. However, such guidance was contained in the 1959 report of the Committee, i.e., Publication 2. The results of our calculations of the secondary limits for natural uranium are tabulated below in the format of Publication 30.

Annual Limits on Intake,
ALI(Bq) and Derived Air Concentration,
DAC(Bq/m^3) for Natural Uranium

Quantity	Oral		Inhalation		
			Class D	Class W	Class Y
	f_1=0.05	f_1=2x10^{-3}	f_1=0.05	f_1=0.05	f_1=2x10^{-3}
ALI	5x10^5	7x10^6	5x10^4	3x10^4	2x10^3
	(7x10^5)	-----	(7x10^4)	-----	-----
	Bone Surf.	-----	Bone Surf.	-----	-----
DAC			20	10	1

Note in the above tabulation if the ALI was based on the nonstochastic constraint, we have shown in parenthesis the intake permitted by the stochastic constraint and below that value the organ involved in the nonstochastic consideration. A becquerel of natural uranium is taken to consist of 0.5031, 0.0222, and 0.4747 Bq of U-234, U-235, and U-238, respectively.

Frequently, questions have been raised as to the particle size dependence of the secondary limits. Tabulated below are the ALI's for natural uranium as a function of particle size and clearance class.

Annual Limit on Intake, ALI(Bq) for inhalation of natural uranium as a function of particle size.

AMAD (Micron)	Clearance Class (f1) D(0.05)	W(0.05)	$Y(2x10^{-3})$
0.1	$3x10^4$ns	$1x10^4$	$6x10^2$
0.3	$4x10^4$ns	$2x10^4$	$9x10^2$
0.5	$5x10^4$ns	$2x10^4$	$1x10^3$
1.0	$5x10^4$ns	$3x10^4$	$2x10^3$
3.0	$4x10^4$ns	$4x10^4$	$3x10^3$
5.0	$4x10^4$ns	$6x10^4$	$4x10^3$
10.0	$4x10^4$ns	$8x10^4$	$7x10^3$

NOTE: ns indicates ALI is based on nonstochastic limit for bone surfaces.

Class D: soluble compounds (UF_6, UO_2F_2, and $UO_2(NO_3)_2$).

Class W: less soluble compounds (UO_3, UF_4, and UCR_4).

Class Y: insoluble compounds (UO_2 and U_3O_8).

ALI's have been calculated as a function of enrichment. Complete checking of the calculations has not been completed as of this writing; these data will be reported in the annual progress report.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Pathogenic Microorganisms in Closed Cycle Cooling Systems

PRINCIPAL INVESTIGATOR: R. L. Tyndall

PROGRAM MANAGER: Webster Van Winkle

ACTIVITY NUMBER: ORNL #41 88 55 01 9 (189 #B0418)/NRC #60 19 30 02

TECHNICAL HIGHLIGHTS: Three field trips to two northern sites were carried out. Air samples were collected for two days on each of the three trips. One large volume and one impinger type sampler were used to collect the down wind test samples on each trip. Legionnaires' Disease Bacteria (LDB) were not detected in test air samples at one of the two test sites. The concentration of LDB in the tower water at this site was 10^5 per liter of water. Legionnaires' Disease Bacteria were detectable in test air samples at the second site. The concentration of LDB per one-hundred liters of air was approximately 2.3×10^3. The concentration of LDB in the cooling tower water was approximately 4×10^7 per 100 liter. A Legionella isolate from water obtained from a post condenser water box was analyzed serologically and found not to cross react with antisera made against known species of serotypes of Legionella. The isolate is being examined by personnel at CDC for DNA relatedness with known Legionella to determine if it is a new species of Legionella.

MEETINGS AND TRIPS: None

REPORTS, PAPERS AND PUBLICATIONS: A paper entitled "Concentration, Serotypic Profiles, and Infectivity of Legionnaires' Disease Bacteria Population in Cooling Towers" by R. L. Tyndall was published in the Journal of the Cooling Tower Institute. A paper entitled "Effects of Cocultivation of *Legionella pneumophila* and Free-Living Amoeba" by R. L. Tyndall and E. L. Domingue was accepted for publication in *Applied and Environmental Microbiology*.

PROBLEM AREAS: None

PROGRAM TITLE: Residual Activity Criteria

PROGRAM MANAGER: R. O. Chester

ACTIVITY NUMBER: ORNL #40 10 01 06 (189 #B0498)/NRC #60 19 42

TECHNICAL HIGHLIGHTS:

Task 1: Scenario Definition - Data are being collected for the physical characterization of three example waste disposal sites. The PRESTO computer code was revised to permit more complete specification of parameters describing regions of subsurface flow.

D. Fields contacted R. Dayal of Brookhaven National Laboratory, seeking chemical exchange data for shallow land burial sites. Dayal has such data and will send reports containing Kd values for the Barnwell, South Carolina, and the West Valley, New York, disposal sites.

Task 2: Dose Calculation - No activity on this task during this month.

Task 3: Develop Dose Recommendations - No activity on this task during this month.

Task 4: Develop Activity Recommendations - No activity on this task during this month.

MEETINGS AND TRIPS:

D. E. Fields attended the NRC-sponsored "Symposium on Low-Level Waste Disposal: Facility Design, Construction, and Operating Practices", held in Washington, D.C. on September 29-30.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Technical Assistance for NEPA Activities in Support of Siting Rulemaking

PROGRAM MANAGER: H. E. Zittel

ACTIVITY NUMBER: ORNL #41 88 54 32 3 (189 #A-9043)/NRC 10 19 03 07 1

TECHNICAL HIGHLIGHTS:

During the month a representative of the sponsor visited ORNL one day to discuss work on this program. At this time, tables representing some major societal resources surrounding nuclear plant sites were delivered to the sponsor. Work continued on revisions to the three documents being prepared for the sponsor. Revised graphic materials for the population distribution document were delivered to the sponsor.

MEETINGS AND TRIPS:

None

REPORTS, PAPERS AND PUBLICATIONS:

None

PROBLEM AREAS:

None

PROGRAM TITLE: The Distribution of Impact in Delaying the Operation of a Nuclear Power Plant

PROGRAM MANAGER: R. C. Tepel

ACTIVITY NUMBER: ORNL #41 88 55 04 3 (189 #B0472)/NRC #60 19 02 10

TECHNICAL HIGHLIGHTS:

Task 1: Initial Methodology Development and Salem II Analysis in its Present Ownership and Institutional Structure-- This task was to evaluate the cost and distribution of cost caused by the delay in operation of the Salem II Nuclear Power Plant. This task is now completed.

Task 2: Completion of Methodology Development and Case Studies-- This task utilized the framework developed in Task 1 to develop an analysis of costs of delay for utilities with different financial and operational characteristics typical of the ownership and regulatory framework within which the utility operates. The results of this task are now being written.

MEETING AND TRIPS

None

REPORT, PAPERS, AND PUBLICATIONS

None

PROBLEM AREAS:

None

PROGRAM TITLE: Threadfin Shad Impingement: Population Response

PROGRAM MANAGER: S.G. Hildebrand/P. Kanciruk

ACTIVITY NUMBER: ORNL 41 88 55 015 (189 #B0406)
NRC # 60 19 02 10

TECHNICAL HIGHLIGHTS:

Task 1: Acquire hydroacoustic literature. The literature acquisition phase of the project is completed. The computer data base contains about 425 papers (many abstracted) on hydroacoustic biomass estimation techniques. Over 225 papers are on hand.

Task 2: Evaluate existing downscan hydroacoustic systems. Analysis of relevant literature is complete. The report is being placed on mats and will go to reproduction soon.

MEETINGS AND TRIPS:

None

REPORTS, PAPERS, AND PUBLICATIONS:

None

PROBLEM AREAS:

None

PROJECT TITLE: Uncertainties in Assessment of Long-Term Collective Dose and Health Effects from Geologic Disposal of High-Level Waste

PROJECT MANAGER: R. O. Chester

PRINCIPAL INVESTIGATOR: D. C. Kocher

ACTIVITY NUMBER: ORNL #41 88 54 32 (189 #A9041)/NRC #10 19 03 03 3

TECHNICAL HIGHLIGHTS:

Task 1: Assessments of Uncertainties - Extensive revisions of the major report on uncertainties in geologic waste disposal have continued, with most of the effort focusing on the discussions of geochemical uncertainties. It should be emphasized that the revisions of this report, which are still in progress, have not changed the conclusions contained in the draft version previously sent to the NRC.

Task 3: Support for 10 CFR Part 60 Rulemaking - The first draft of a letter report on the role of geochemistry and its uncertainties in the regulation of geologic waste disposal has been completed and is undergoing internal review.

Task 4: Geologic Records and Future Site Evolution - The letter report on modeling capabilities for predicting geologic processes and their effects on waste isolation has completed internal review and revision, and should soon be ready for transmittal to the NRC.

Task 5: Technical Assistance for Developing HLW Radiological Criteria - Informal discussions with ORNL and NRC staff and reviews of relevant literature have continued, and preparation of a letter report should begin shortly. Most of the opinions and points-of-view expressed in the outline presented to the NRC in June have not changed as a result of subsequent work, except for the realization that utilizing uranium ore bodies in a standard is not likely to be as advantageous as previously believed.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

An abstract entitled "Environmental Radiation Standards for Geologic Disposal of Highly Radioactive Wastes," by D. C. Kocher, has been submitted for presentation at the Waste Management '83 Symposium to be held in Tucson, Arizona, on February 27-March 3, 1983.

PROBLEM AREAS:

None.

PROGRAM TITLE: Valence Effects on Adsorption

PROGRAM MANAGER: R.E. Meyer

ACTIVITY NUMBER: ORNL #41 88 55 04 0 (189 #B0462) #60 19 02 20

TECHNICAL HIGHLIGHTS:

Task 1: Apparatus and Materials. In June we received the Tc-95m which we had ordered late in 1981. Our first task was to insure that it was in the proper valence state initially, i.e. as Tc(VII), and therefore to look for procedures that could make the distinction between Tc(VII) and other valence states. Examination of the literature showed that the pertechnetate ion, TcO_4^-, could be extracted quantitatively by a solution containing tetrephenylarsonium-chloride and that other valence states would not be extracted. Experiments showed that the Tc which we received was entirely in the VII state as we expected. Other experiments demonstrated that extraction could be carried out in HCl solutions from 1E-4M to 4M. Electrochemical experiments in our recirculating flow cell showed that as the potential was made more reducing, less of the technetium was extracted, presumably because lower valence states were formed. There is contradictory evidence in the literature concerning the speciation of the lower valence states, but in neutral and alkaline solution in the absence of complexing agents, oxides or hydroxides of Tc(IV) are probably formed. The solubilities of these oxides and hydroxides are not known with certainty, but we find that Tc will essentially disappear from solution when Tc(VII) is reduced in neutral or alkaline solutions.

In the course of these reduction experiments with Tc we found out that if our entire apparatus is enclosed in the controlled atmosphere box, the reduction occurs more easily, i.e. at less reducing potentials. Probably a small amount of oxygen leaks into the plastic tubing used in the circulating loops of the apparatus and this causes some oxidation. The observed reducing potentials are therefore probably modified by the action of the oxygen. For this reason, we have decided to conduct all anoxic experiments in the atmosphere box. Since we have only one atmosphere box, we have requested and received capital money for an additional box. The order has now been placed and we hope to receive it in a few months.

Task 2: Sorption Measurements (with valence state control). We have conducted two types of sorption experiments this past quarter. In one type, a complete recirculating column coupled with an electrochemical cell was used, and in the other the materials were put into a beaker in the atmosphere box and the solid separated from the solution by means of a dialysis membrane. In each experiment, valence state analysis was used before, during, and after the experiment to follow the course of the reaction.

Experiments with recirculating column. We have described in early reports experiments with Np(V) sorption on alumina. The first experiments were designed to maintain Np in the V state and then to measure its sorption characteristics on alumina, an adsorbent which has well-defined sorption properties and no redox properties. These experiments showed that when the Np was kept in the V state, the distribution coefficients were reproducible and reversible, and the values increased with pH. However, when we attempted to carry out the same experiment at reducing potentials, i.e. with Np the IV state, we were unable to produce Np(IV) at potentials where thermodynamically it should be reduced. We later found that when we changed from a platinum electrode (our initial choice) to a silver electrode, we were able to produce some Np(IV). During the past quarter, we have found that putting the entire apparatus in the controlled atmosphere box greatly improves our ability to form Np(IV). With this apparatus, we were able to study Np adsorption on alumina at reducing potentials. As expected, most of the Np sorbed on the alumina was in the IV state ($\sim$60%) but some was also in the V state ($\sim$40%). All of the remaining Np in the solution was in the V state and some Np(IV) was found on the cell walls. Thus with this apparatus we were able to show that we could produce Np(IV) in solution, that it could be transported in the loop, and that we could determine the valence state distribution of the Np on the alumina with reasonable certainty. To determine the valence state on the alumina, we leached the material in the controlled atmosphere box with 1 N HCl and immediately conducted valence state analysis by solvent extraction. In these experiments, we assume that there is no significant valence change on leaching, and this is a good assumption since the leaching is carried out in the atmosphere box.

The other type of experiment was conducted with two different basalt samples and two different solutions, 0.1 M NaCl and a synthetic groundwater. Two samples of basalt were used, one a sample obtained from the WISAP(WRIT) program for use in the controlled sample program. This sample was found to contain ferromagnetic material, most probably a little metallic iron. We also obtained a sample of Umtanum Basalt and ground it ourselves to make sure that it was not contaminated. The results showed that almost all of the Np adsorbed on the two types of basalt was in the IV state, and the remainder adsorbed as Np(V). There was no significant difference between the two groundwaters but there was much more sorption on the contaminated basalt even though ultimately both solutions in contact with the basalt showed little difference in iron composition.

MEETINGS AND TRIPS:

R. E. Meyer attended a NRC Research Program Planning Workshop on Nuclear Waste Management Research on Geochemistry of HLW Disposal, Ann Arbor, Michigan, September 22-23, 1982.

REPORTS, PAPERS AND PUBLICATIONS:

A report entitled "VALENCE EFFECTS ON ADSORPTION. A preliminary Assessment of the Effects of Valence State Control on Sorption Measurements," NUREG/CR-2863, ORNL-5905 was completed and is in the process of publication.

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