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OAK RIDGE NATIONAL LABORATORY

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Monthly Highlights Report

ORNL Projects for the NRC Office of Nuclear Regulatory Research

A. L. Lotts

January 1983

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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Prepared by the
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for the
DEPARTMENT OF ENERGY

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ABSTRACT

Highlights of technical progress during November 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 limited effort during this period has resulted in a few additions to our draft technical memorandum (TM), tentatively entitled *Attenuation Studies in Thick Pressure Vessel Steel.*

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.

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 was prepared and submitted to the NRC regarding the status of this task.

MEETINGS AND TRIPS:

R. G. Berggren attended a meeting of the Metal Properties Council Subcommittee 6, Task Group on Charpy Trend Curves in New York on November 3.

R. K. Nanstad attended ASME Code meetings in New York on November 8 and 9, specifically the Subgroup on Toughness and the Subcommittee on Properties. A special meeting on reevaluation of the drop-weight NDT test method was also attended and a copy of the meeting minutes will be provided to the program monitor.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: ASME Code Section III — Technical Support

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 — R. D. Campbell of Structural Mechanics Associates (SMA) has recently completed a draft report on his first year's work for the PVRC toward the development of dynamic stress criteria for the design of piping systems. The report includes a review of existing information on elastic-plastic response of piping systems to dynamic loads, a discussion of a potential design criterion based on the concept of dynamic-to-static safety margin, the analysis of several very simple piping system models, and a research proposal program plan. As part of our PVRC activities we have been asked to review the report draft and to provide detailed technical comments in time for Dr. Campbell to make any required changes before the PVRC meets again in January 1983.

We also reviewed current ongoing work at EDS Nuclear on the same subject at a PVRC Task Group meeting in San Francisco on November 19, 1982. This work, which is being sponsored by the Electric Power Research Institute (EPRI) and assigned to the PVRC Technical Committee on Piping Systems, is somewhat broader in scope than the SMA work but is less detailed. EDS Nuclear plans to complete a draft report on their work by the end of the year and send it to EPRI for incorporation into a larger document for public release. We will receive a copy of the draft for comment before the final report is published.

Task 2: Piping Support Reactions — E. C. Rodabaugh Associates, Inc., has done some additional work on the effects of rotational and linear displacements of anchors on support loads. They plan to begin writing the first draft of the final report early in December.

Task 3: Fatigue Evaluation for Class 2 and 3 Piping Components — Copies of the draft report *Comparisons of Code Fatigue Evaluation Methods for Class 1 Piping with Class 2 or 3 Piping* by E. C. Rodabaugh and G. T. Yahr was released to the ASME Working Group on Piping Design (WGPD, SCD, ASME III) at their last meeting on November 8, 1982, for use in formulating potential changes in the current ASME Code rules. Our plan is to begin preparing the final report for publication as soon as we receive comments from the NRC reviewers.

Task 5: Preloading of Bolted Connections — Review of the available literature on bolted-joint design continued. Good treatment of fatigue in bolted joints was found in the book *Designing Against Fatigue* by R. B. Heywood and in a paper presented at the 1956 International Conference on Fatigue of Metals by Dr. A. Erker entitled "Design of Screw Fastenings Subject to Repeated Stresses." Both references cite results from many sources, especially in their respective countries of England and

Germany. Both references demonstrate the use of joint diagrams, emphasize the importance of proper preload on fatigue life, and emphasize the importance of the ratio of the bolt stiffness to the stiffness of the clamped parts.

Task 7: Evaluation of Section III Acceptance Standards and Fatigue Curves Using a Fracture Mechanics Approach — Comments were received from W. H. Cullen and S. W. Taggart on the ORNL and EG&G, Idaho, proposals for evaluation of Section III acceptance standards and fatigue curves using a fracture mechanics approach. W. J. O'Donnell has mailed a long detailed discussion of the proposals that will be received next month. Comments from these three individuals and from NRC-RES will help in defining the research that will be pursued.

Four references to French work on fatigue crack initiation were received from Dr. George Slama of Framatome as a result of discussions at the Task Force on Crack Propagation Technology meeting last month. Although the current ORNL and EG&G, Idaho, proposals assume a small crack at the beginning of plant operation, this assumption can be extremely conservative for high cycle fatigue where crack initiation is an appreciable portion of the total life.

MEETINGS AND TRIPS:

S. E. Moore attended the meeting of the ASME Boiler and Pressure Vessel Committee Working Group on Piping Design in New York City on November 8, 1982, as a member.

S. E. Moore attended a working meeting of the PVRC Task Group on Dynamic Allowables, Technical Committee on Piping Systems held in San Francisco, CA, on November 19, 1982. In addition to reviewing the EDS Nuclear work on allowable stresses discussed under Task 1 above, a draft action plan for the T/G was formulated for presentation to the PVRC at its next regular meetings in January 1983.

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:

An evaluation of the formulation of the leak rate equation proposed by EXTRAN was initiated. The fundamental difference between the EXTRAN equation and the ANSI/ANS equation is in the determination of the temperature term. Both equations provide an approximation so an evaluation was begun to determine if the differences were significant.

Only one set of leak rate data has been used in the evaluation so far, but certain trends are observable. In every case but one the difference between the temperature terms in the two equations was approximately two percent or less. In one case the difference was almost twenty percent, but the temperature terms were so small that the twenty percent difference had a negligible effect on the leak rate. In fact, it seems that the only time a significant difference is likely to occur in the leak rates will be when the leak rates are extremely small (approaching zero). These observations are preliminary and may change as additional data are evaluated.

A search of the Nuclear Safety Information Center (NSIC) computer file has been conducted to identify the License Event Reports (LERs) pertaining to Type A leak rate tests. Copies of the LERs will be obtained soon and reviewed. An additional search of the NSIC computer file will be conducted at a later time to identify LERs pertaining to Type B and C leak rate tests.

Due to conflicts in scheduling, one plant visit was canceled. Additional plants that will be conducting leak rate tests in the near future have been identified and will be used to replace any cancelled visits.

MEETINGS AND TRIPS:

On November 30—December 1, 1982, J. R. Dougan and D. J. Naus met with Z. V. Reytblatt of EXTRAN, Inc., and with personnel of Wiss, Janney, Elstner and Associates in Chicago, Illinois.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

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 2: Irradiation Effects — We are preparing specifications for weldments fabrication for the 4T-K_{Ic} Irradiation Program. The specifications must provide for weldments typical of pressure vessel fabrication, in compliance with pertinent parts of the *ASME Boiler and Pressure Vessel Code*, as uniform as practicable with regard to chemical composition and toughness properties, and at as low a cost as practicable. To accomplish these objectives, we have reviewed the ASME Code requirements and the special requirements of this program and are detailing fabrication requirements.

A portion of the specimens for the first capsules of the cladding irradiation study are now on hand and the remainder are being machined.

Details of the testing of specimens from the Fourth HSST Irradiation Study, to be divided between ENSA-MEA and ORNL, were sent to MEA for comment. The MEA reply was received on November 27, and we will immediately compare MEA and ORNL proposals and reach agreement on test temperatures before proceeding with the test program.

Task 3: Thermal Shock — A sensitivity study was conducted to determine the effect of the change in the neutron spectrum through the wall of the vessel on $(\Delta RTNDT_s)_c$. For a typical OCA with $p > 7$ MPa incipient initiation results in failure (no arrest), and for these cases the sensitivity of $(\Delta RTNDT_s)_c$ to a change in the exponential attenuation decay constant from 0.013 mm^{-1} (no change in spectrum) to 0.0095 mm^{-1} (change in spectrum) is very small ($\sim$2°C). For lower pressure cases arrest follows incipient initiation, and thus a larger difference in attenuation is involved. Therefore, the differences in the values of $(\Delta RTNDT_s)_c$ corresponding to incipient failure are also greater but still rather small ($\sim$6°C for $p = 3$ MPa).

Efforts were continued to include finite-length flaws in OCA-II. ORVIRT was modified for the purpose of calculating the necessary influence coefficients, several coefficients were calculated and compared with those of Raju and Newman, and associated mesh convergence studies were conducted. Efforts are continuing to establish the accuracy with which the coefficients can be calculated.

OCA-II is being modified to utilize the coefficients for finite-length flaws, and OCA-P is being modified to accept the new version of OCA-II.

Pretest calculations for TSE-7 are continuing, using ORVIRT to analyze the 3-D flaws. Three segments from prolongation TSP-4 to be used in

TSE-7 have been tempered for 4 h at 649, 677, and 704°C, followed by cooling in air. Charpy V-notch and drop-weight specimens are being machined from each segment.

Task 4: Intermediate Vessel Test — The fracture surfaces of the flaw from vessel V-8A are being mapped through use of a digital mensuration microscope with traveling stage. A series of index marks have been placed on the fracture surfaces and all measurements of crack extension and other prominent features are references to the index marks by a coordinate system.

A purchase order has been let with Babcock and Wilcox to obtain posttest fracture properties from material taken from the V-8A weldment.

Task 5: Pressurized Thermal Shock — Pressurized-Thermal-Shock Test Facility (PTSTF) design and construction activities continue to remain on schedule and within projected costs. The third DOE Level II milestone, completion of Instrumentation and Controls design, was successfully achieved. Design is now complete, all major facility components except the intermediate test vessel shroud or outer test vessel (OTV) are installed, and pipe construction is in progress. The total, directive controlled, task is now approximately half complete.

Investigation of maximum pressurization rates achievable with present pressurization equipment indicated that neither of the existing intensifiers is capable of producing rates envisioned for the PTS-1 experiment. Additionally, examination of probable material properties of the A508 class 2 chemistry, quenched-only steel indicate that fracture toughness properties for OCA analysis are best represented by an RTNDT of 13°C instead of 27°C as originally perceived. The two above factors, as well as 3-D vs 2-D geometry effects reported last period, resulted in an OCA reevaluation of test parameters for the PTS-1 experiment. An attempt was made to minimize the pressurization rate, fracture toughness properties represented by an RTNDT of 13°C were used, and the OCA-generated K_I values were adjusted to represent an 1141-mm crack length.

The preliminary analysis indicated that all PTS-1 test criteria can be achieved with a maximum pressurization rate of 33.1 MPa/min and a maximum initial temperature of 232°C. Initiation and arrest events occur at essentially the same times and conditions as described by the previous analysis. An anticipated stress intensity for the second arrest event of 286 MPa $\sqrt{m}$ at an a/w of 0.5 indicates that upper-shelf behavior can be achieved.

Contract negotiations are continuing with Babcock and Wilcox Company for the preparation of test vessels for shakedown tests of the facility and the first two tests. Original plans were to use ITV V-8 in the unflawed condition for shakedown tests, remove it for flawing and instrumentation, and then use it for PTS-2. However, the estimated time of delivery of this vessel, which was to be prepared with a low-upper-shelf weldment, is now 33 weeks after receipt of contract which is much too late to allow its use as a shakedown test vessel. Therefore we will now delay procurement of the vessel for PTS-2 and have ITV V-8 prepared with the same test material as vessel V-7 which will reduce the delivery time to 21 weeks. This will allow the use of V-8 for shakedown tests and it will have a plug of characterized material making it a possible candidate for a future experiment. A third vessel, ITV V-5, will be prepared with a low-upper-shelf seam weld under a separate contract to be placed in early 1983.

Charpy V-notch testing of the weld metal from low-upper-shelf submerged-arc weldments prepared at Babcock and Wilcox is continuing. Through-thickness Charpy V-notch toughness properties of weldment V813 have been and are being determined for weldments V822 and V823. Additional specimens were machined from welds V822 and V823.

We have also heat treated four additional segments from prolongation TSP-4 in order to determine the Charpy V-notch and drop-weight toughness after extended exposure times in the temperature range of 552 to 691°C to characterize material for use in the pressurized-thermal-shock experiments. Five conditions are being examined: as-received, 6 h at 552°C, and 17 h each at 552, 566, and 579°C; each heat treatment will be followed by furnace cooling at 28 to 56 K/h. All the heat-treated segments had received a prior heat treatment of 8 h at 524°C, followed by air cooling. The heat treatments have been completed and specimen drawings for Charpy V-notch and drop-weight specimens from each of the five segments and an "as-received" segment from the prolongation (TSP-6) of vessel TSC-6 have also been completed. Specimen machining is in progress.

Task 6: Cladding Evaluations — Remaining clad beams have been weld repaired in preparation for remachining of grooves and electron-beam welding. The remaining unclad beam has been electron-beam welded and instrumented. The beams will be tested to narrow the range of arrest/no arrest conditions.

The crack arrest test facility is ready for testing: 1 x 6 x 6 in. crack arrest specimens have been machined from the base plate material and calibration/characterization of the cooling system has begun.

A purchase requisition has been submitted for obtaining a nozzle cutout from Combustion Engineering for weld clad metal characterization. A purchase requisition and drawings are being prepared to obtain clad beam specimens clad by the three-wire series arc procedure from Combustion Engineering. Characterization material will be made available from this order for the cladding irradiation program as well.

MEETINGS AND TRIPS:

R. D. Cheverton attended a meeting in Hartsville, SC, on November 9, to discuss the integrated pressurized-thermal-shock program with the N. B. Robinson Nuclear Plant personnel.

G. D. Whitman, C. E. Pugh and W. R. Corwin attended a meeting on November 15, at MRL offices in Glenwood, IL, to develop plans for a round robin crack arrest test program.

J. G. Merkle attended the meetings of the PVRC/MPC Task Group on Reference Toughness and the ASME Section XI Working Group on Flaw Evaluation, in Williamsburg, VA, on November 15–16. The Task Group on Reference Toughness took note of the possible size effects existing in its data base of small-specimen cleavage fracture-toughness values, and decided to make trial adjustments of these data by using the Irwin β_{Ic} equation, according to a method recently proposed by ORNL.

G. C. Robinson and W. R. Corwin attended a meeting at Combustion Engineering offices in Chattanooga, TN, on November 16, to review plans for fabricating weld clad specimens.

J. G. Merkle attended an NSF Workshop on Mechanics of Damage and Fracture, held at Stone Mountain, GA, on November 22–24. The ORNL proposal for making a size-effect adjustment to small-specimen cleavage fracture-toughness values was well received, and several other valuable ideas were exchanged. Among these were preliminary analytical results concerning room temperature inelastic strain rate effects on crack tip stress fields and the need for a connection to be made between fatigue-damage theories and the starting flaw sizes used in the fracture-mechanics approach to fatigue crack growth analysis.

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:

A spring-loaded pancake coil has been tested with our new standard that has the matrix of property variations shown in the table below.

Table 1. Property values of new tubing standards

Tubesheet location (in.)	Tubesheet diameter (in.)	Tube wall thickness (in.)	Probe Lift-off (in.)	Defects[a] (%) Holes ID	Holes OD	Notches ID	Notches OD
0.50	0.880	0.050	0.000	0	0	0	0
0.30	0.894	0.040	0.005	25	25	25	25
0.15		0.030	0.010	50	50	50	50
0.10							
0.08							
0.05							
0.02							
0.00							
−0.02							
−0.05							
−0.08							
−0.10							
−0.15							
−0.20							
−0.25							
−0.30							
−0.40							
−0.50							
−0.70							
−1.0							

[a]ID, inside diameter; OD, outside diameter.

This gives a total of 4320 possible combinations if each defect reading set contains one reading on "good" tubing. However, our reading program will only read 1440 values and our fitting program will only allow 1200 values before the programs overflow the MocComp IV memory. A new computer that allows up to 3 megabytes of program storage (compared to 128 kilobytes of program storage in the MocComp IV) is on order. The reading and fitting programs were each run on their maximum number of values. Back plots of the fitted properties showed that the 25% outside diameter notches could be detected and are about the same size as the residual caused by the corner of the tube supports. Running the full data set is expected to give some improvement.

As part of the responsibility for transfer of developed technology to industry, the Nondestructive Testing Development Group conducted a technology transfer meeting at the Knoxville Hilton on November 17, 1982, followed by a tour of laboratory facilities on November 18. Approximately 75 people attended the technical session on the 17th, with over 50 touring the laboratory the following day. A broad spectrum of interests was represented, including utilities, nondestructive testing equipment manufacturers, inspection agencies, material suppliers, universities, government, and other industry. Among the topics discussed were developments of multifrequency eddy current techniques and equipment for in-service inspection of steam generator tubing. Strong interest and appreciation for the meeting was expressed by the attendees. It is anticipated, based on comments, that many attendees will be actively seeking additional transfer of the nondestructive testing technology for their use.

MEETINGS AND TRIPS:

C. V. Dodd gave a presentation, "Multifrequency Eddy Currents for Light-Water Reactor Steam Generator Tubing," at an ORNL Technology Transfer Meeting, "Emerging Advances in Nondestructive Testing," held at the Knoxville Hilton, November 16–18, 1982.

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 - No activity.

C. BSR-HSST - The dosimetry data from capsule D is currently being counted. This data will be included with capsules A, B, and C which make up the 4th series of HSST irradiation experiments.

D. ORR-SDMF - No activity.

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

A. ASTM Standards - No activity.

B. 5th ASTM-EURATOM Symposium on Reactor Dosimetry - J. P. Genthon, EURATOM program chairman, sent a letter indicating that the EURATOM committee will have a proposed format for the symposium during their next meeting scheduled for January 18, 1983. He will forward the format for comments by the U.S. committee.

C. NUREG Reports - Ten NUREG reports have been identified to document the work by various NRC contractors in the LWR-PV Surveillance Program (Table 1). Contributions from this program have been submitted for "NUREG #4." Contributions for "NUREG #1" have been submitted except for Section 7. Experimental data have not been received to complete the analysis.

MEETINGS AND TRIPS: None

REPORTS, PAPERS, AND PUBLICATIONS: None

PROBLEM AREAS: None

Table 1. NUREG Reports Summarizing the Work in NRC's LWR-PV Surveillance Program

W. N. McElroy, Ed., NUREG Report #1: LWR-PV Surveillance Dosimetry Improvement Program: PCA Dosimetry in Support of the PSF Physics-Dosimetry-Metallurgy Experiments, Hanford Engineering Development Laboratory (to be published in 1983).

W. N. McElroy, Ed., "Part I. PSF Physics-Dosimetry Characterization Program," NUREG Report #2: LWR-PV Surveillance Dosimetry Improvement Program: PSF Physics-Dosimetry-Metallurgy Experiments (to be published in 1984).

W. N. McElroy, Ed., "Part II. SSC-1 and SSC-2 Metallurgical Program," NUREG Report #3: LWR-PV Surveillance Dosimetry Improvement Program: PSF Physics-Dosimetry-Metallurgy Experiments (to be published in 1984).

W. N. McElroy and G. L. Guthrie, Ed., NUREG Report #4: LWR-PV Surveillance Dosimetry Improvement Program: LWR Power Reactor Surveillance Physics-Dosimetry Compendium, Hanford Engineering Development Laboratory (to be published in 1983).

W. N. McElroy, Ed., "Part III. PVS and Void Box Physics-Dosimetry Program," NUREG Report #5: LWR-PV Surveillance Dosimetry Improvement Program: PSF Physics-Dosimetry-Metallurgy Experiments (to be published September 1984).

W. N. McElroy, Ed., "Part IV. PVS and Void Box Metallurgy Program," NUREG Report #6: LWR-PV Surveillance Dosimetry Improvement Program: PSF Physics-Dosimetry-Metallurgy Experiments (to be published September 1984).

W. N. McElroy, F. B. K. Kam, E. D. McGarry, Ed., NUREG Report #7: LWR-PV Surveillance Dosimetry Improvement Program: PSF Surveillance Dosimetry Measurement Facility (SDMF) (to be published September 1983).

W. N. McElroy, F. B. K. Kam, E. D. McGarry, Ed., NUREG Report #8: LWR-PV Surveillance Dosimetry Improvement Program: LWR Test Reactor Physics-Dosimetry Compendium (to be published September 1985).

A. Fabry and W. N. McElroy, Ed., NUREG Report #9: LWR-PV Surveillance Dosimetry Improvement Program: VENUS PWR Core Source and Aximuthal Lead Factor Experiments and Calculational Tests (to be published September 1983).

J. Butler, M. Austin, A. Fudge, and W. N. McElroy, Ed., NUREG Report #10: LWR-PV Surveillance Dosimetry Improvement Program: NESDIP PWR Cavity and Azimuthal Lead Factor Experiments and Calculational Tests (to be published September 1984).

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:

For the section on resuspension of soil radioactivity in the addendum to NUREG/CR-2241, radiation doses were recalculated using a resuspension factor of 5×10^{-6} per meter. The most significant results of using this resuspension factor were for the mixed oxide and uranium dioxide fuel fabrication facilities where doses were found to increase by 288 and 36%, respectively, over those given in NUREG/CR-2241. Detectability in soil will be greatly affected at the mixed oxide site; however, the 10 mrem/yr level should still be detectable with Enewetak proportional or imp instruments.

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 31 02

TECHNICAL HIGHLIGHTS:

Analysis of CCTF-II Runs 054 and 055 is continuing. CCTF-II Shakedown Test (053) analysis is complete. SCTF-I Runs 509 and 514 have been analyzed and the data results plotted. Essentially all the analysis for SCTF-I is finished. The results will be compiled and a report published in the future.

A pre-production prototype test plan for the drag body strain gage signal conditioners (AIRS 20 electronic module) is near completion. Two AIRS 20 prototype modules have been fabricated and will be tested in December.

The prototype breakthrough detector was installed in the IDL and room temperature ranging test started.

Design drawings were generated for the dP purge control block and dP electronics block. An acceptance test plan and other documentation were prepared in anticipation of a formal design review at ORNL in December. Two mounting options for the dP measurement block were transmitted to KWU and their response is awaited.

MEETINGS AND TRIPS:

Sandia National Laboratory staff members visited ORNL on November 11 to discuss UPTF program networks.

REPORTS, PUBLICATIONS, AND PAPERS:

None.

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:

November was a time of problem solving. Progress falls into three areas again: (1) torsional wave generation in 1/8-inch diameter material, (2) finalizing the electronic package design, and (3) building a prototype high-energy pulser.

In October, we reported on the disappointment over the new 1/8-inch Remendur magnetostrictive material made in the M&C Division. Some improvement was obtained by heat treating in a hydrogen furnace; however, the torsional wave remained weak (currents up to 50 amperes are required for a strong torsional signal). We then tried a selection of 1/8-inch nickel tubing of various wall thicknesses. After annealing, the thinner tubes were found to generate strong torsional signals with only 10 amperes of axial current. Another bonus of using thin-walled tubing is that the impedance of the tube can be more closely matched to that of a flat section of the same dimensions. This should lead directly to a probe which is testable in the pressurizer and cross-flow test stands.

At each iteration of the electronics, improvements were made in both performance and cost. The high-energy pulser has also had several improvements in cost and performance. The current prototype has led to a final version which will provide about 300 mj of power -- having the parameters of 1 to 12 microsecond pulse widths, up to 800 volts and 20 amperes -- thus fewer turns of larger diameter wire are needed to establish the required magnetic field for stress-pulse generation. As an example, a coil, consisting of 24 turns of #24 B&S Gauge high-temperature-coated wire, resulted in pulses of comparable strength to those obtained with the previous pulsers and coils up to 200 turns.

MEETINGS AND TRIPS:

None.

REPORTS, PUBLICATIONS AND PAPERS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Clad Ballooning Evaluation

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL #41 89 55 10 6 (189 #B0120)NRC 60 19 02 01

TECHNICAL HIGHLIGHTS:

We continued review and expansion of the software used for reducing photographic data to strains and bundle blockage.

A request was sent to KfK soliciting PNS cooperation in our evaluation of existing clad ballooning data. We specifically requested published and unpublished REBEKA bundle test information in sufficient detail (design, fabrication, loop operation, test data, results of pretest and posttest examination, etc.) to permit indepth understanding of how the tests were conducted and of the results. Requests for the KfK single rod tests (both out-of-reactor and in-reactor) will be made during a later phase of the evaluation.

A request was also sent to the CEGB Berkeley Nuclear Laboratories soliciting their cooperation. We specifically requested use of their BALON code for calculating clad deformation (with transient temperature-pressure data supplied as input). We also asked if they were still interested in destructive posttest examination of several of our single rod test specimens.

A request was also made to INEL for a stand-alone version of the BALLOON-2 program for use in our evaluation.

Although we have not had formal replies to these requests, we expect each organization will respond favorably.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

A report, entitled "Experiment Data Report for Multirod Burst Test (MRBT) Bundle B-4," was submitted to the Reproduction Department for printing and distribution. The report provides results of the test in sufficient detail to permit analysis of the data by interested users.

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, which has been expected since May, is now projected to arrive at ORNL in January 1983. Because of the uncertainty of receiving this fuel in time for release tests planned in mid-1983, we are investigating procurement of several specimens of Big Rock Point (BWR) fuel through a related test program at Argonne National Laboratory.

During the recent visit at Kernforschungszentrum Karlsruhe (KfK) in Germany, the proposed procurement of simulant fuel (like that used in the SASCHA experiments) was discussed further. The possibility of obtaining specimens of this fuel for tests beginning in late 1983 appear to be good.

2. Fission Product Release Tests and Results

The third test (HI-3) of irradiated fuel was conducted at 2000°C, and all of the test apparatus performed satisfactorily. The conditions for these three tests, each using a 20 cm length of H. B. Robinson fuel of about 30 MWd/kg burnup, are summarized below.

Test No.	Test temperature (°C)	Time at temperature (min)	Average flow rates (L/m STP)		H_2 generated (L)
			Inert gas	Steam	
HI-1	1400	30	0.44 (Ar)	1.01	12.17
HI-2	1700	20	0.33 (Ar)	0.94	13.54
HI-3	2000	20	0.30 (He)	0.36	4.92

The steam flow rate in test HI-3 was reduced to permit melting of the Zircaloy cladding before extensive oxidation, so that the effect of molten cladding on fission product release could be investigated. These conditions resulted in the fuel specimen and its supporting ZrO_2 boat being fused to the ZrO_2 furnace tube. The latter, however, remained intact, so we have not been able to examine the specimen visually. It was sealed in epoxy resin, and will be sectioned radially at a number of locations.

The test apparatus is being disassembled and analyzed by gamma spectrometry. The initial results indicate that about 37% of the ^{85}Kr was released during the test. This value is somewhat lower than the corresponding value in test HI-2. Similarly, gross gamma measurements indicate somewhat less ^{137}Cs release than in HI-2 also. Visible deposits of as yet unidentified material were seen on the first two filters, but no significant reduction in flow rate during the test, as had occurred during test HI-2, was observed.

3. Apparatus Modifications and New Equipment

Preliminary design of apparatus modifications to include installation of (a) ThO_2 furnace ceramics, which are required for tests in the range 2100–2400°C; and (b) multiple collection trains, so that more data may be obtained from each experiment, was begun. A cost estimate for of the required ThO_2 ceramics, which are not available commercially, by Los Alamos Scientific Laboratory has been requested.

Intermittent problems with the TP-5000 multichannel analysis system continue to cause delays in the gamma spectrometry. Bids from three companies for a new MCA/computer system were received and evaluated, and an order for the best system should be placed soon.

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

No new experiments were conducted in this exploratory effort this month. New test apparatus to permit continuous flow analysis of CsI vapor in steam is being constructed. In addition, apparatus for testing the compatibility of CsI vapor with various materials (especially silica and sapphire) under the required test conditions is being assembled and will be in operation soon.

MEETINGS AND TRIPS:

During a trip to KfK and to the Centre d"Etudes Nuclearies de Grenoble (CENG), the results of our fission product release studies were presented and discussed in detail with German and French scientists involved in related reactor safety research. A trip report summarizing these discussions and conclusions was prepared. In addition, our results were presented in a special session on LWR source term development at the Winter 1982 American Nuclear Society Meeting at Washington, D.C.

REPORTS, PAPERS, AND PUBLICATIONS:

ORNL/FTP-1429.

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: Develc.ment continued on the ORECA-FSV code for modeling Fort St. Vrain (FSV) reactor long-term uncontrolled core heat up scenarios. Programming for a model that calculates primary coolant temperatures around the entire primary loop, as well as primary coolant pressure, was completed and successfully executed in an uncontrolled heat up test transient.

BLAST Code Development: In order to provide a benchmark for the BLAST HTGR steam generator dynamics code, BLAST predictions are being compared with plant data obtained from FSV for the November 4, 1978 oscillation transient.

ORTURB Code Development: Work continued on comparing code predictions with measured plant operational data during a turbine runback transient from full power conditions. Additional information from Public Service Co. of Colorado (PSC) will be necessary to fully represent the transient turbine conditions.

2240 MWT HTGR Siting Study: Development work continued on the ORECA (3-D Core) code for predicting severe accident scenarios for the 2240 MWT steam cycle/cogeneration HTGR. A literature search was made on the behavior of PCRV concrete subjected to severe accident conditions. Based on the available data, models were developed and implemented in ORECA for predicting water and CO_2 release from "representative" concrete types. Wide variations in predicted behavior, however, result from wide uncertainty ranges in the assumed concrete properties. The concrete properties for a given plant design will in turn depend on the region of the country in which the plant is built, since material availability varies with locati.n. The models were tested against experimental data from test run at HEDL, and with parameters adjusted to account for the known concrete composition, good correspondence was obtained for the water release transients. Predicted CO_2 release values were considerably higher than measured (for limestone concrete), however. Only about 15% of the potential CO_2 available was actually released, suggesting that CO_2 release may be limited to that necessary to reach its equilibrium pressure within the concrete or the space between the concrete and liner.

Several other model and parameter changes were made in ORECA. A more detailed model was included for the heat transfer from the side reflector to the LCS, incorporating the thermal shield and the upper (Class A, optional fallaway) insulation and the lower (Class B, fixed) insulation regions. An optional model for gray-body thermal radiation in both the upper and lower plenums has been implemented. Test runs accounting for cavity reflection effects were made for comparison with the original model results for assumed emissivities and absorptivities of 0.8. The differences were relatively small, although upper plenum cover plate failure was delayed somewhat. Possibilities of much lower values of emissivity (~0.2) and a "thermal polishing" effect (large reductions in emissivity at high temperatures) are also being investigated. A checkout of code parameters was done to assure that the updated plant design details and geometry values are properly accounted for. Detailed code documentation efforts were also begun.

Mathematical modeling of the fission product transport problem (heat source redistribution) has begun. Each fission product group is being treated as an independent problem of mass transport with effective binary diffusion through the solid. The total heat source redistribution will consist of the combined effects of all the groups.

Fission Product Release from HTGRs: This subtask will lead to predictions of the chemical form and volatilities of fission products and actinides in the fueled core under accident conditions. This subtask will update similar calculations in reports GAMD-9335 published in 1970 by P. Winchell and GA-A12634 published in 1974 by J. H. Norman. Examination of these two reports revealed that only metal-carbon-carbide systems had been investigated, whereas many oxide, hydroxide, and halide gaseous species are known to be important (if not dominant) under HTGR accident conditions. The present work includes all these species.

The U-C-O system is being analyzed currently. The gaseous species included in the calculations are U, C, C_2, C_3, C_n, O, O_2, CO, CO_2, UO, UO_2, UO_3, UC_2, UC_3, UC_4, UC_5, and UC_6, while the condensed (solid or liquid) phases are UO_2, $UC_{1.93}$, and C.

The preliminary results, which are based on very good thermodynamic data from the literature, indicate that the uranium-containing gaseous species have a total pressure up to 1000 times that predicted for the simple $UC_{1.93}$-C system in GA-A12634, even for $UC_{1.93}$ fuels, primarily because the partial pressures of the UO and UO_2 species are often much greater than those for the U species. Graphical and mathematical descriptions of the partial pressures of important U-C-O gaseous species are being developed over hypothetical HTGR accident conditions of temperature, oxygen pressure (or, equivalently, CO/CO_2 and H_2/H_2O ratios), and fueled core composition.

MEETINGS AND TRIPS:

Discussions were held with Dr. N. Kirch of KFA on November 23. Dr. Kirch is KFA's HTR R&D Program Coordinator, and was interested in learning more about ORNL's relationship with NRC, GA, and PSC on FSV licensing problems. KFA is working on selected licensing problems for the THTR program.

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)

Spectrophotometric Measurements (L. M. Toth)

An examination of ionic strength effects on the second stage hydrolysis reaction, $3HOI = IO^{3-} + 2I + 3H^+$, has been performed at pH values of 7 and 10. A solution of 0.231 M H_3BO_3 and 0.5 M $NaClO_4$ with the pH adjusted to either 7 or 10 was used as the constant ionic strength medium. With the concentration of I_2 at 2×10^{-6} M, the growth in I^- was followed as a function of time and then used to determine the rate constant for the second stage reaction using the previously derived relationship between "HOI" and measured I^-. Arrhenius activation energy plots of the rate constants for these reactions at constant ionic strength compared favorably with those in 0.231 M H_3BO_3 alone, but the rates of reaction were found to be ~25% higher at pH = 7 where $NaClO_4$ was included. (No differences were noted for the pH = 10 experiments at constant ionic strength.) Further tests are planned to verify this apparent anomaly at pH = 7.

An examination of the effect of various redox atmospheres on the kinetics of the second stage hydrolysis reaction reveals no discernible effects at this time. Reaction rates for 2×10^{-6} M I_2 in 0.231 M H_3BO_3 at pH = 7 and 65°C have been measured for solutions sparged and equilibrated with atmospheres of N_2, H_2, and O_2, respectively. These have been compared with those previously studied on air-saturated solutions and no differences (within experimental error) in the rate of formation of I^- via the second stage hydrolysis reaction have been observed. Test at higher temperatures will be performed to determine the effect of temperature on the added reagents.

Iodine Volatility Measurements (E. C. Beahm)

Two iodine volatility tests were run at pH = 9 (buffered) and 4.5×10^{-5} g-atom/L, and temperatures of 298 and 323 K. Data analysis for these runs is as yet incomplete. An untraced sample duplicating the conditions of the 298 K test was run using an iodide specific ion electrode to follow the iodide concentration as a function of time. In addition, the iodide content of the 298 K sample traced with ^{131}I was measured after 2 weeks of volatility tests. This iodide content was within 2% of the value expected from the initial iodine concentration as measured by titration with sodium thiosulfate, and assuming the general hydrolysis reaction:

$$3I_2 + 3H_2O \rightarrow 5I^- + IO_3^- + 6H^+ \ .$$

This result indicates that the specific activity of the traced sample had no effect on the stoichiometry of the reaction of iodine and water.

A different volatility test was run on a sample containing 8.5×10^{-3} mol CH_3I/L in pure water. The test started at an initial temperature of 298 K. This resulted in an iodine partition coefficient of 5.9. This value persisted for 4 days of mixing. The system temperature was then raised to 323 K which resulted in a decrease of the partition coefficient then steadily increased reaching a value of 26.1 after 13 days of mixing at 323 K. This test illustrates the ease with which this closed sample system can evaluate iodine partitioning with volatile organic iodine forms.

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

The results from chemical analyses of aerosol samples taken during Run 601 were received. Each sample was analyzed for iron, silicon, calcium, magnesium, and aluminum content by ICP Spectroscopy. In this test the aerosol was generated by introducing a mixture of equivalent masses of iron powder and limestone concrete powder into the plasma torch aerosol generator. The preparation of the steam-air test atmosphere and the experimental procedure were the same as those previously used in the U_3O_8 and the Fe_2O_3 aerosol test series. Reduction of the analytical data is not yet complete.

The next scheduled aerosol test in the NSPP vessel will involve study of the behavior of a limestone-aggregate concrete aerosol in a steam environment. Delay in conduct of this test continues because of difficulty in obtaining proper repair parts for the data logger. Steps are being taken to prevent any future delays of this nature.

Concrete aerosol generator upgrade tests were continued in a small test tank as part of the developmental task to produce a suitable generation technique.

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

A significant difference exists between the fission element release fractions determined in the ORNL core-melt experiments and those measured in the earlier, smaller-scale, experiments conducted in the SASCHA project at Karlsruhe. An extensive exchange of information has been initiated in an effort to determine why the SASCHA values are much larger than those measured at ORNL. One possible reason is the difference in core-melt crucible material; thoria (ThO_2) was used in the SASCHA project while zirconia (ZrO_2) has been used at ORNL. Efforts to locate a suppler of thoria crucibles have produced one possible vendor; inhouse efforts to produce thoria crucibles are also pending. The possible effect of crucible materials as well as the differing experimental conditions will be evaluated.

The 1-kg core-melt facility continues to be in part-time operation due to ongoing modifications incorporating a small aerosol holding tank which will permit direct sampling of the released aerosols for particle morphology and size distribution measurements.

AEROSOL GENERATOR SUPPORT EFFORTS: G. W. Parker

Recent requests for design details and drawings on the ORNL plasma torch aerosol generator have been received from Battelle Columbus Labs and from Kraftwerke Union, Federal Republic of Germany. At Battelle Columbus the generator will be used in an EPRI-sponsored, pressure suppression pool aerosol wash-out test program. In Germany a large-scale multiple unit aerosol generation system is being considered to supply test aerosol for the 600 m^3 vessel in the DEMONA program.

The CRI-II facility is being readied for tests of plasma torch aerosol generation under steam conditions as would be expected in the 600 m^3 DEMONA vessel. Initially only low-pressure steam conditions will be used to determine the extent of impairment of the efficiency of aerosol generation. Some modification in sampling techniques and equipment will be necessary to allow for operation in a steam atmosphere.

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

As part of the ABCOVE project, post-test calculations for the AB-5 sodium spray fire test using the QUICK and HAARM-3 codes were performed and results furnished to the other participants in the project. Comparison continues between the pre-test code predictions made at ORNL and those by other participating installations.

A copy of the MSPEC multiple-species aerosol code was received from the Battelle Columbus Labs. Steps to make this code operational at ORNL were initiated.

Tests are continuing to check the performance of the four heat-flux meters attached to the outer surface of the NSPP vessel. It may be possible to use values obtained from these meters in determining the steam condensation rates on the inner wall surfaces during the steam-aerosol tests.

MEETING AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

G. W. Parker, Foreign Trip Report, ORNL/FTR-1428, November 10, 1982.

PROBLEM AREAS:

Delay encountered in conduct of full-scale NSPP aerosol tests due to instrumentation problems.

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. The program is divided into four subtasks: 1) Management and Analysis, 2) Aerosol Transport Tests, 3) Fission Product Transport Tests, and 4) Aerosol Resuspension Tests.

Management and Analysis

This month an attempt was made to use the QUICK and HAARM-3 aerosol codes to predict the results obtained in the fourth "scoping" aerosol test (performed in October). In this experiment iron oxide aerosols were generated for 5 min. The estimated aerosol "residence time" in the pipe (pipe length divided by average upflow velocity) was roughly 60 s. After the test 55.7 g of Fe_2O_3 aerosol was recovered; the location of the recovered aerosol was as follows:

a) Plated on vertical walls: 39.9 g (71%)
b) Settled on floor: 7.9 g (14%)
c) Leaked out of pipe: 8.5 g (25%)

Note that roughly 85% of the collected aerosol was trapped in the pipe. In terms of aerosol plated on the walls, samples taken at the bottom and top of the pipe indicated that the amount of aerosol plateout, per unit surface area, at the bottom of the pipe was roughly four times greater than at the top.

QUICK and HAARM-3 calculations performed for this test assumed that aerosols were generated for 5 min in the pipe at a rate of 0.19 g/s (55.7 g/300 s). The source aerosol distribution was assumed to have a mean geometric radius $r_g = 0.1$ μm and a geometric standard deviation $s_g = 1.8$. In general, preliminary results from HAARM-3 and QUICK calculations did not satisfactorily match the test results. In particular:

a) Large thermal gradients had to be assumed to calculate plated mass values near to those measured. For an assumed 750°C/cm thermal gradient, QUICK calculated ~30 g plated at t = 5 min, while HAARM-3 calculated ~25 g plated at the same time. Since the maximum gas temperature measured 2 ft from the plasma torch was ~250°C, and laminar flow conditions existed in the pipe, a 750°C/cm thermal gradient seems unrealistic.

b) Both codes calculate that at t = 5 min, only a few tenths of a gram of aerosol would settle in the pipe. This is significantly less than the 8.5 g found on the floor after the tests.

We plan to continue to attempt to calculate the results from the scoping aerosol transport tests using QUICK and HAARM-3.

Evaluation of literature related to aerosol resuspension phenomena is now complete, and a letter report related to aerosol resuspension has been initiated. Some preliminary findings include the following:

a) Although aerosol resuspension modeling has been done, there is no model presently available that could be used immediately in TRAP-MELT. This is largely because it is difficult to predict the cohesive forces that will exist for aerosols deposited onto surfaces in accident situations.

b) In previous aerosol resuspension experiments, resuspension seems to have been a turbulent flow phenomena. Since flow in the upper plenum in LWR accidents is likely to be laminar, it is difficult to assess the potential for resuspension without performing experiments.

Aerosol Transport Tests

The fifth scoping test was performed this month; we had planned to generate Fe_2O_3 aerosols for 10 min but had to stop at 6-1/2 min because we were not sure that the plasma torch was still operating. Roughly 77 g of aerosol was recovered in the test; the location of the recovered aerosol was as follows:

a) Plated on vertical walls: 85.5 g (76%)
b) Settled on floor: 10.7 g (14%)
c) Leaked out of pipe: 7.5 g (10%)

In this test roughly 90% of the aerosol was trapped in the pipe. The wall plateout on each of the three pipe segments was as follows: 44.2 g on the lower section, 10.3 g on the center section, and 4.0 g on the top section was roughly 11 times that on the upper section, probably due to the higher temperatures produced near the plasma torch.

An attempt was made to measure the aerosol concentration in the pipe during this test; however, the aerosol sample collected was too small for accurate determination of the airborne concentration. The sampler is being remade to permit larger aerosol samples to be taken.

Fission Product Transport Tests

Progress in this subtask was in the following two areas:

1. A preliminary design for the upper plenum simulator, to be located above the induction furnace, was made.

2. Based on data from previous tests, estimates of the "residence time" produced in the test geometry used in previous experiments (in the NRC B0121 program) were made. Estimated residence times were from 2 to 32 s. The calculations indicate that the height of the upper-core simulator — above the surface of the core-melt furnace — should be 1-2 ft, to permit residence times in the range of 100 s to be produced.

Aerosol Resuspension Tests

An insert to the aerosol transport test section is being fabricated that will permit us to mount 12 collection foils — each being roughly 3 ft long — in the test section. After depositing aerosol generated by the plasma torch onto these foils, they will be individually mounted in a 6-in.-diam Pyrex pipe and preliminary aerosol resuspension tests will be performed. One series of these tests will hopefully be performed in December.

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

The chemical activity of the cesium species makes selection of materials of construction more difficult. Both CsI and CsOH are known to react with quartz, ruling out glassware for the vapor generator and experimental apparatus. Stainless steel has been used in CsI vapor studies, but the CsI may react with stainless steel to release molecular iodine. In addition, molten CsOH and CsOH vapor will react strongly with stainless steel. Possibly, gold or platinum lining of all surfaces expected to come into contact with the vapor may be required. Other alternatives are being explored.

Also, the purity of the carrier gas can prove critical in maintaining the identity of the molecular species with which we wish to work. For example, CsI may be oxidized in the presence of O_2 to release molecular I_2, and Cs reacts with O_2 or H_2O. This is particularly a problem for the set of dynamic experiments involving an oxidic aerosol because O_2 is introduced into the plasma with the metallic powder. Tentatively, an oxygen getter must be installed upstream of the fission product vapor to remove excess O_2 in the aerosol stream.

Attempts to monitor the sorption of I_2 and CsI onto metallic coupons with time using NaI detectors failed in a Battelle study (S. L. Nicolosi and P. Baybutt, NUREG/CR-2713, BMI-2091, R3, R4, January 1982) because of the background count. This method is similar to the one proposed for the static determination of sorption isotherms for deposited aerosols. A rough calculation indicates a ratio of about 100 cm^2 of surface area per cm^3 monitored will give us reasonable sensitivity with regard to background activity. In addition, Clough et al. [Int. J. Air Water Pollution, 9, 769, (1965)] successfully used this technique for measuring sorption isotherms.

MEETINGS AND TRIPS:

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: NSIC Supplemental Bibliographic File

PROGRAM MANAGER: J. R. Buchanan

ACTIVITY NUMBER: ORNL #41 89 55 10 4 (189 #B0126)/NRC 60 19 11

TECHNICAL HIGHLIGHTS:

This program involves the addition of LWR safety information to NSIC base computer file to meet NRC/RES commitment to UKAEA. Work on this program was initiated in the latter part of August 1982 upon receipt of the Program Brief and authorization. Since processing of the LWR information had ceased at NSIC in May 1982 due to redirections in NSIC activities, it was necessary to reestablish the following: document flow, screening, selection, abstracting and indexing, quality review and entry into the computer master file.

A total of 78 document descriptions were added to the computer file in November.

Shipment of tapes to the UKAEA was resumed in November to cover the period of time since shipments were terminated. Two bimonthly tapes were mailed to the Atomic Energy Attache at the British Embassy in Washington, D.C. Including LERs, the tape shipped on November 18 contained 948 document descriptions; the one shipped on November 30 contained 1083 descriptions.

MEETINGS AND TRIPS:

None

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 and the analysis of fission product transport for the scram discharge volume break accident sequence. Preparations have been completed for a major presentation to a forthcoming NRC Inspection and Enforcement (I&E) training seminar. A peer review of the Brookhaven National Laboratory assessment of the Limerick PRA is in progress.

The personnel contributing to the SASA effort at ORNL are divided into three working groups. The individual group reports for progress during November 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.

Chapters 5 and 6 and Appendix B of the draft report concerning loss of DHR accident sequences at Browns Ferry were completed during November. Chapter 5 describes the accident sequence which occurs when the pressure suppression pool cannot be mixed* during heatup, and Chapter 6 extends this consideration to the case of a stuck-open relief valve. Appendix B describes the modifications necessary to the BWR-LACP code to permit analysis of the loss of DHR sequences.

A compendium of the results of the ORNL studies conducted to date has been prepared in the form of a presentation to be given in a training seminar for NRC I&E inspector personnel in early December.

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

Loss of DHR Accident Analyses (S. R. Greene) Chapters 7 and 8 and Appendix C of the draft report were completed during November. The general subject of drywell failure by static overpressurization is discussed in Chapter 7; Chapter 8 describes the results of MARCH runs conducted to study events after core uncovery; Appendix C provides the MARCH code input.

*by the RHR system (which would be operating without cooling water available to the heat exchangers).

MARCH 2.1 Model Development (L. J. Ott) The basic version of the new MARCH meltdown models A and B has been completed. The model features separate nodal temperatures for the BWR fuel assembly shrouds and control rods for each zone of the reactor core. Oxidation of the Zircalloy fuel assembly shroud material is included in the model.

It is anticipated that the new models will be delivered to Battelle Labs (BCL) by December 13 for ultimate incorporation into MARCH 2.1. Preliminary indications are that application of the new meltdown models will generally result in significantly slower BWR fuel heatup rates than are predicted with the current MARCH meltdown models A and B.

MARCH 2.1 Model Development at RPI (R. T. Lahey, M. Podowski) The pace of the detailed core meltdown model development effort was slowed significantly during November due to the unexpected temporary absence of a key research team member because of a death in the family. With the delay, this model will be delivered to BCL in early December.

MARCH 2.1 BWR Modeling Needs Assessment (S. R. Greene) A meeting with members of the Systems Engineering Division of the General Electric Company was held in San Jose on November 4 for the purpose of information exchange concerning experience in the application of MARCH to BWR severe accident analysis. Discussion was open and informative, yielding important information for inclusion in ORNL's BWR modeling needs assessment and MARCH 2.1 critique. The topics discussed include MARCH 1.1 BWR modeling limitations, generic BWR severe accident modeling needs, and the results of GE's independent analysis of severe BWR core heatup transients.

The GE analysis indicates that the temperature difference between the hottest fuel pin in an assembly and the surrounding fuel assembly shroud wall would be less than 10°F, one hour after fuel temperature reaches 3000°F. This result is in substantial agreement with the preliminary results from ORNL's modified MARCH meltdown models A and B, and highlights the importance of modeling to represent the radiative heat transfer between the large number of structural components in the BWR core.

Pressure Suppression Pool (PSP) Modeling (D. H. Cook) The lumped parameter PSP model has been completed. This model predicts PSP response to safety-relief valve (SRV) discharge and the thermal stratification which persists after relief valve closure, and is described in Appendix D of the draft Loss of DHR Accident Sequences report.

The PSP dynamics during T-quencher discharge have been modeled and verified by comparing code results with the suppression pool temperatures measured during single SRV discharge tests previously conducted by GE.

The results of the ORNL PSP analysis have been made available to GE during a November visit to San Jose (funded by GE).

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.

Fission Product Transport Calculations and Draft Report Progress (R. P. Wichner, C F. Weber, S. J. Niemczyk) Completion of the draft report of fission product transport for the Browns Ferry small-break LOCA outside containment accident sequence is now projected for December 31. Unanticipated demands by other ORNL programs upon the services of key analysis personnel needed for the completion of the transport calculation have caused a three-month delay.

An approach to fission product transport analysis for the loss of DHR accident sequence at Browns Ferry has been developed. The important new feature of this sequence is long-term boiling of the suppression pool.

MEETINGS AND TRIPS:

S. R. Greene visited GE at San Jose November 4 to obtain information concerning vendor experience in the application of MARCH to advanced boiling water reactor designs.

Upon GE request and funding, D. H. Cook visited the Nuclear Energy Business Operations Center, General Electric Company (GE), San Jose, California, on November 11 and 12 to present the ORNL SASA program modeling efforts for the thermal-hydraulic response of the pressure suppression pool under accident conditions.

S. A. Hodge attended a meeting at Bethesda, MD on November 16 to discuss the future role of the ORNL SASA effort with representatives of NRR. The ORNL SASA team will peer-review the BNL assessment of the Limerick utility-sponsored PRA.

REPORTS, PAPERS, AND PUBLICATIONS:

The draft report concerning loss of DHR accident sequences at Browns Ferry has been submitted to the NRC technical monitor and the other SASA laboratories for peer review.

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:

Work continues on the review and revision of three NUREG reports:

(a) Health and Safety Standards: Theoretical Rationale and Application to Safety Goals for Nuclear Power

(b) Modeling the Societal Impact of Multiple Fatality Accidents

(c) Evaluation of Mortality Risks for Institutional Decisions

All three reports are in advanced stages of revision and the NUREGs should be ready for distribution sometime in January.

MEETINGS AND TRIPS: None

REPORTS, PAPERS, AND PUBLICATIONS: See above

PROBLEM AREAS: None

PROGRAM TITLE: Analysis of Reliability Data from Nuclear Power Plants

PROGRAM MANAGER: R. J. Borkowski

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

TECHNICAL HIGHLIGHTS:

Task 1: Reports. The pump report is in the final stages of publication. Copies of the final version were sent to the sponsor.

Task 2: Data Collection and Encoding. Data collection efforts are being planned for Plant 6. The encoding of population, failure and repair records on pumps and valves in Plants 1-4 is complete. Data base preparation and final editing of these records are proceeding.

Task 3: Data Analysis. The evaluation of the valve records to determine an appropriate analysis strategy is continuing. The initial outline of the valve report is complete.

Task 4: Cooperation with Other Data Systems. Contacts have been made with three European data systems as well as U.S. systems, (CREDO, NPRDS, and GADS).

Task 5: Human Error Analysis. Proceeding as planned. Initial taxonomy has been developed.

MEETINGS AND TRIPS:

R. J. Borkowski presented the IPRD program at the winter meeting of ANS in Washington, D.C. on November 17. He also met with subcontractors J. R. Fragola, SAI, and D. Shurman, ASA, to discuss the Human Factors Analysis.

REPORTS, PUBLICATIONS, AND PAPERS:

"The In-Plant Reliability Data System: A Data Bank for Components in Commercial Nuclear Power Plants," by J. P. Drago, R. J. Borkowski (ORNL), and J. R. Fragola, (SAI), paper at ANS winter meeting.

PROBLEM AREAS:

Funding has not arrived in time to avert delays and additional subcontracting costs.

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:

During November 1 - 4, 1982, D. J. Campbell, B. C. Ellison and D. P. Wagner visited the Arkansas Nuclear One, Unit 1 plant. The purpose of the visit was to obtain specific information about component locations, possible sources of root causes of common cause failure, and existing barriers to these root causes. The ANO-1 staff was most cooperative in providing tours and detailed discussions on all areas of interest. All objectives of the plant visit were accomplished.

All components that appear in the IREP ANO-1 fault trees related to the Emergency Feedwater System accident sequence analysis (Task 2) have been assigned specific locations within the plant based on the plant inspection and supporting information supplied by Sandia and AP&L. These fault trees describe the following systems:

1. Emergency Feedwater System,
2. High Pressure Injection System,
3. Reactor Building Spray Injection System,
4. Reactor Building Cooling System,
5. Engineered Safeguards Actuation System,
6. Service Water System,
7. Emergency AC Power System,
8. DC Power System,
9. Battery and Switchgear Emergency Cooling System, and
10. Emergency Feedwater Initiate and Control System.

This component location information, along with component susceptibilities and plant barrier information, is being prepared for input to the computer-aided analysis.

Common cause failure analysis (CCFA) procedure development (Task 3) is proceeding on schedule. The IREP (NUREG/CR-2728), NREP (NUREG/CR-2815), and PRA Procedures Guides (NUREG/CR-2300) have been reviewed to identify appropriate interfaces for the CCFA procedures. These procedures are being developed as the actual plant analysis progresses.

Detailed development of the ANO-1 rodded scram system fault tree has been completed for the scram system analysis (Task 4). This fault tree is being transmitted to ANO-1 staff for review. The plant visit described above provided location information for the scram system components. The completed fault tree and location information, along with component susceptibilities and plant barrier information, is being prepared for input to the computer-aided analysis.

MEETINGS AND TRIPS:

Project staff visited the Arkansas Nuclear One plant site, as described in the "Technical Highlights" section of this report.

REPORTS, PAPERS AND PUBLICATIONS:

None in this reporting period.

PROBLEM AREAS:

No new problems in this reporting period.

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 1E. Three additional tables were developed for inclusion into the December interim report. A tabular summary of postulated UF_6 source term information reported by NRC licensees for eight NRC-selected facilities was developed. This summary includes release amounts, release rates, release durations, and assumptions made in developing source terms for postulated UF_6 releases. Some of the larger UF_6 release events at the eight NRC-licensed facilities were summarized in a second table. To expand this table, an attempt was made to access a NRC/ANL data base for radiation events in the commercial nuclear fuel cycle from 1950 through 1978, but unfortunately it was found to be inoperative. In a third table, UF_6 handling process parameters retrieved in our study such as temperatures, pressures and vessel inventories were tabulated.

Task 1I. The December interim report was begun and will include some of the information presented to NRC in a meeting on November 23, 1982.

Task 2. Identification of Event-Controlling Parameters

Task 2A. Several methods of grouping the accident scenarios reported in the September progress report have been considered in an attempt to identify a fewer number of "generic scenarios." Scenarios can be grouped according to zones of UF_6 inventory affected by an accident, especially with respect to the UF_6 production facilities. At least four major zones of UF_6 inventories have been identified at one or more of the NRC-licensed facilities. These inventories are associated with (1) fluorination, (2) distillation, (3) cylinder operations, and (4) cylinder feed operations. Little is openly known about distillation inventories.

MEETINGS AND TRIPS:

J. Dykstra, R. A. Just, and M. Siman-Tov made a presentation to the NRC Research Review Group in Washington, D.C., on November 23, 1982. The presentation reported progress made on the project to date as well as a qualitative evaluation of analytical methods available or needed for developing UF_6 source terms. NRC pointed out that it is interested in utilizing any readily available methods for developing UF_6 source terms, recognizing that developing these tools may be beyond the scope of the current effort.

REPORTS, PAPERS AND PUBLICATIONS:

None

PROBLEM AREAS:

The NRC/ANL data base on radiation events at commercial facilities was inoperative.

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. Documentation of the event trees and branch probabilities is underway. Twelve sequences to be calculated by INEL and LANL have been specified. These include:

1. Main steamline break
2. SBLOCA (stuck-open PORV)
3. Steam Generator Overfeed (no ICS runback of main feedwater, no HPI throttling)
4. Oconee-3 turbine trip of March 14, 1980
5. Turbine bypass valve failure (2 TBV's)
6. Turbine bypass valve failure (4 TBV's)
7. SBLOCA (2" diameter hole in hot leg)
8. SBLOCA (4" diameter hole in hot leg)
9. Maximum sustainable main feedwater overfeed to steam generator without FW pump trip
10. Hot standby turbine bypass valve failure
11. Overfeed of steam generator on emergency feedwater (similar to Rancho Seco)
12. SBLOCA with low decay heat

A probabilistic fracture mechanics model is being built for the Oconee study. Due to schedular restraints, the model will consider only infinite flaws in the axial direction and 360° flaws in the circumferential direction. The ORNL models are being modified to consider finite length flaws but the modifications will not be completed in time for the Oconee study.

Thermal-Hydraulic Modeling

INEL has completed calculations on sequences #1, 2, 3, 4 listed above. Comparisons with plant data indicate that the INEL model calculates overall trends in plant response satisfactorily. Calculations for sequences #1, 2, 3 indicate no severe thermal shock to the reactor vessel wall. For the sequence #1 (main steamline break), the vent valve flow warms the fluid in the downcomer substantially. The lowest calculated downcomer fluid temperature was 481K (407°F); including estimated uncertainty, 427K (309°F). The lowest cold leg fluid temperature was 390K (242°F); including estimated uncertainty, 369K (205°F).

The LANL model of Oconee using TRAC is operational. Calculations of the sequences assigned to LANL have begun.

Calvert Cliffs - Unit 1

Task 1 - Modeling the plant - ORNL has experienced some difficulty obtaining some of the needed plant data. As a result, the assistance which ORNL was to give LANL in modeling the plant has not progressed as rapidly as planned. LANL had asked ORNL to accelerate this effort because they had predicted a one month delay (until April 1) due to anticipated difficulties in control system modeling. This acceleration has not taken place, but no new schedule slippages are anticipated.

Task 2 - Identify event sequences - The first draft of the system state trees for the Calvert Cliffs study is nearly complete. This task is on schedule.

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

Task 9 - Review models and results with plant owner - The last review was on September 21, 1982. The next reviews are planned for December 2, 8, 9, 1982.

HB Robinson - Unit 2

Task 1 - Modeling the plant - Several Carolina Power and Light employees briefed the PTS study team on HB Robinson - Unit 2 onsite. Written information transfers are beginning. INEL will model the plant with RELAP. ORNL will model the plant with RETRAN to support the probabilistic risk analyses. This task is on schedule.

Task 2-8 - Not scheduled to begin until later dates.

Task 9 - Review models and results with plant owner - A review of the program plant for HB Robinson-2 was presented to Carolina Power and Light personnel at the HB Robinson site.

MEETINGS AND TRIPS:

November 9, 10 (HB Robinson): reviewed program plan concerning HB Robinson - Unit 2; identified data needs; reviewed proposed plant modeling and fracture mechanics techniques; toured the site; briefed by HB Robinson staff regarding plant geometry and operation.

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 11 05

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 8 tasks. During November, activities continued in the following areas:

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.

Task 8. Development of a computer code to model precursor event trees has been initiated.

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: The Long Range Planning Group at ORNL is interested in setting priorities of 23 R & D Areas, based on the opinions of 11 judges. Uppuluri met with Truman Anderson and Gerald Payne of ORNL and discussed about the applicability of the computer programs developed here based on Saaty's methods.

Paul Slovic, Decision Research, Eugene, Oregon, sent an analysis of risk perception data based on the use of a version of Saaty's method. Uppuluri made some comments and suggestions on this analysis.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS AND PUBLICATIONS:

A letter report on the NRC/ORNL Workshop on Propagation of Uncertainties, held at Oak Ridge during October 7-8, 1982 was mailed to NRC.

PROBLEM AREAS:

None.

PROGRAM TITLE: Probabilistic Fire Risk Analysis

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL 41 88 55 05 3 (189 #B0493) NRC #60 19 51

TECHNICAL HIGHLIGHTS:

Work proceeded this month on improving the current fire growth models and on developing improved generic fire suppression models.

The fire growth code, COMPBRN, recently completed and documented, was used to simulate one of the Sandia cable tray fire experiments which investigated the ability of enclosure fires to defeat physical separation of components. The results of the simulation runs are currently being analyzed; these results should be helpful in fine-tuning the empirical constants used in the code's fire plume and hot gas layer models.

[illegible]ral analytical framework for the progression of a fire event se[illegible]as developed. This framework involves the definition of fire detection/suppression states, e.g., "automatic detectors activated", and the transitions between states. Given the stochastic distributions of the times of transition between states, the distribution of the time required to reach a particular state via a specified sequence of transitions can be computed. The actual time to reach the state is the minimum of the various path times, hence, competing risks analysis is appropriate.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Risk Analysis Evaluations

PROGR MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #40 10 01 06 5 (189 #B0465) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

Task 1: Zion PRA Audit Analysis - The technical description of the Zion Probabilistic Safety Study (PRA) was prepared. The final report is in preparation. This technical description of the Zion PRA presents the logic structure of the analysis. This logic structure explains what variables were calculated to describe public risk and how they were calculated.

Task 2: Using PRA in the Regulatory Process - No activity.

Task 3: Multiattribute Risk Descriptions - No activity.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS AND PUBLICATIONS:

Task 1 - Scope and Approach.
Task 2 - Audit Analysis of the Zion Probabilistic Safety Study.

PROBLEM AREAS:

None.

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 #60 19 5 1

TECHNICAL HIGHLIGHTS:

We continued the effort of applying our method of the Cut set Size Truncation (CST) methodology on one of the dominant sequences of Calvert Cliffs IREP. This effort was stopped due to errors in the files that contained merged fault trees of Calvert Cliffs IREP. A new set of files is requested from Sandia National Lab which are made and being sent to us.

Meanwhile we further developed the CST methodology to account for errors which occurred due to the employment of the rare event approximation method that is used to sum the contribution of cut sets. We also used the Nuclear Engineering Department's Tektronix 4054 Computer to plot the error which occurred in Probability Calculation versus different cut set sizes by using the CST methodology.

MEETINGS AND TRIPS:

A meeting was held at the NRC's office of risk analysis on November 22, 1982 between D. Rasmuson and M. Modarres and the progress of this project was discussed. Some strategies for future calculations were adopted.

REPORTS, PAPERS, AND PUBLICATIONS: None

PROBLEM AREAS:

The IREP files obtained from SAI through the University Computing Company (UCC) have errors in them and thus we are unable to run SETS. A new set of files are made and are being sent to us from Sandia National Labs.

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. Rand received formal reviews on WD-1671-ORNL, "Dealing With Uncertainty Arising Out of Probabilistic Risk Assessment," by Solomon, Kastenberg, & Nelson from two reviewers. By December 10, 1982, Rand will respond to these review comments and submit a revised manuscript to ORNL and NRC.

Rand has also asked ORNL for comments, but as of this date, none were received.

B. Regarding the value impact portion of Rand's work, Rand is still awaiting draft NRC documents from P. Rathbun. These documents are instrumental in the development and application of the value impact approach.

Rand has developed a detailed value impact approach for assessing a number of generic fixes. In preliminary applications of this approach, Rand finds that for moderate size accidents (e.g., TMI) and larger (e.g., ATWS), the health costs from person-rem releases make up only a small fraction of the total accident impact. Property damage, business costs, and secondary costs make up the major financial impact. A large cost resulting from a postulated accident at a specific site is absorbed by the nuclear industry as a whole.

C. Rand is assessing -- in a highly preliminary fashion -- incentives for more extensive self regulation by the nuclear industry. As the industry costs associated with potential nuclear accidents increases, the incentive for greater regulation by the utilities/industry consortium is increased.

Rand has some preliminary calculations to support the incentives for self regulation.

More detailed data on the cost and scope of generic fixes and reactor accidents would permit more extensive value impact calculations.

MEETINGS AND TRIPS:

On November 24, 1982, K.A. Solomon (Rand) met with A.S. Benjamin (Sandia) at UCLA and also at Rand to discuss both Rand's progress on the value impact work as well as Rand's data needs.

REPORTS, PAPERS, PUBLICATIONS:

A. On November 18, 1982, K.A. Solomon presented at the winter meeting of the American Nuclear Society, "The Costs of Closing Nuclear Power Plants Under Both Voluntary and Involuntary Conditions." The presentation displayed the method for calculating the costs and applied this method in detail to the potential voluntary closure of Indian Point and discussed the application for estimating the costs associated with the accident at Three Mile Island.

B. In a paper related to but not directly in support of Rand's tasks, M. Meyer and K.A. Solomon published "Risk Management Practices In Local Communities,", Rand P-6821.

In summary, this paper discusses the current state of risk management practices in local communities in the U.S. and offers some alternatives to present policies, which are mainly implicit rather than explicit attempts to limit overall risks of death and injury due to technological and natural causes.

This work is directly applicable to the NRC tasks under study by Rand in that this work maps out five distinct alternatives for managing risk in the local communities. The alternatives, as discussed in the paper, are directly applicable to risk management policy alternatives within the NRC.

PROBLEM AREAS:

A. As of this date, Kuljian Corporation (Rand's subcontractor) has failed to respond to Rand's review of Kuljian's draft document.

B. Rand is anxiously awaiting draft value impact studies from P. Rathbun in order that Rand may integrate these studies in their work.

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 report entitled "Metabolic Data and Retention Functions for the Intracellular Alkali Metals" by R. W. Leggett was given to Dr. Allen Brodsky of ORPP during his November 22 & 23 visit at ORNL. Dr. Leggett's attention has now turned to metabolic models for plutonium and other transuranics to serve the needs of bioassay procedures. Dr. Brodsky met with Dr. Leggett to discuss both the alkali metal effort and the present effort on plutonium.

Dr. Brodsky also met with Dr. Bernard with regard to metabolism of uranium. Copies of the first draft of several sections of a report were discussed. Further work is underway using the power function model in conjunction with an analytical procedure of the late W. S. Snyder.

Further efforts will be addressed toward cobalt and the alkaline earths.

MEETINGS AND TRIPS:

Dr. A. Brodsky met the the staff of the Metabolism and Dosimetry Group on Nobember 22 & 23.

REPORTS, PAPERS, AND PUBLICATIONS:

A draft of "Metabolic Data and Retention Functions for the Intracellular Alkali Metals" by R. W. Leggett is now in laboratory review.

PROBLEM AREAS:

None.

PROJECT TITLE: Continuous On-Line Reactor Surveillance System Evaluations

PROJECT MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 12 8 (189 #B0442)/NRC #60 19 31

TECHNICAL HIGHLIGHTS:

Task 1: In-Plant Demonstration of On-Line Noise Surveillance System. Due to refueling of Sequoyah Unit 1, no additional data has been taken. However, system evaluation and continued analysis of data has been continuing. Documentation of all signatures collected during the first fuel cycle is almost complete.

Task 2: Abnormal Operating Condition Data Collection. Permission has been received from DOE to participate in future LOFT tests. Special test conditions were reviewed by LOFT personnel.

Task 3: On-Line Noise Surveillance System Upgrade. Development of algorithms for having the on-line analyzer perform cross power spectral densities (CPSD) has begun. These algorithms will be tested and then incorporated into the analyzer.

EQUIPMENT PURCHASES:

The Winchester disk for the analyzer has not yet been received. The subsystem vendor reports problems in obtaining the Winchester drive from his supplier.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

Purchase of Winchester Disk Drive (see above).

PROGRAM TITLE: Development of Human Reliability Model for NPP Maintenance Personnel

PROGRAM MANAGER: P. M. Haas

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

TECHNICAL HIGHLIGHTS:

The "Job Analysis of the Maintenance Supervisor and Instrument and Control Supervisor Positions for the Nuclear Power Plant Maintenance Personnel Reliability Model," NUREG/CR-2668, ORNL/TM-8299, received final editing and was sent to publication. The report will be disseminated early in the next report period (December, 1982).

The analysis of the data obtained from the job questionnaires for the electrician position was initiated. The final return rate for the questionnaires was 67%.

The logic for the following simulation model modules were completed during this report period, and programming of these modules was initiated: (1) fatigue, (2) radiation stress, (3) communication effectiveness, and (4) accessibility effects. The development of the model logic for the following aspects continues: (a) simulation of emergency conditions, (b) method of combining the various contributors to total stress, and (c) methods of calculating risk to public. In addition, the programming of menu generation programs and information flow within the model continues.

Arrangements have been made with maintenance supervisory personnel at a PWR plant for the conduct of detailed task analyses. Three different maintenance positions will be addressed with respect to several important tasks as identified through previous job analyses. The task analysis data will be used to support the formulation and development of the model, will have input for information required for validation, and may be part of a 5-6 task library that may be used for instructional purposes regarding the running of the model. Efforts are currently underway to make arrangements for similar task analyses at a BWR facility.

A Peer Review Group Meeting for this program has been scheduled for December 7-8, 1982 at the NRC offices in the Rockville, Maryland area. The members of the Review Group invited to this meeting are: Bill Askren, USAF Human Resources Laboratory; John Garrick, Pickard, Lowe and Garrick; Bob McDermand, EBASCO Services; Dave Meister, USN Personnel R&D Center; Jack Parris (or representative), EPRI; Ken Strahm, INPO; and John Wreathall, NUS Corporation.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

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 Assessment - No additional survey was made this month; some results of assessments appear in the other task area.

Tasks 2 and 4: Neutron Activation and Fission Track Methods for the Determination of Uranium in Urine - The highlights of tasks 2 and 4 have been combined this month to report the results of a survey that are applicable to both tasks. A survey was made to determine what reactor irradiations facilities are available in the United States for commercial applications of delayed neutron and fission track counting methods to the determination of uranium in urine. Seven private and nine university laboratories were contacted. Those contacted were listed either in the 1982 Buyers Guide of Nuclear News under the category of neutron activation analysis services or in DOE/TIC-8200-R4S which lists nuclear reactors at universities for research and training. All but two universities and two private companies reported that they provide neutron activation analysis as a commercial service either on a part-time or full-time basis. Most have delayed neutron counting facilities and several have conducted fission track counting for measuring uranium in water and other materials. The survey was not exhaustive and it is estimated that an additional 15-20 reactor facilities exist in the U.S. that might provide such commercial services. It appears to be a common practice for private companies to use nearby university reactors for such services. Most of these reactors have neutron fluxes that are high enough to provide the sensitivity necessary for measurements of uranium in urine by either delayed neutron or fission track counting. We have therefore concluded that availability of reactor irradiation facilities does not represent a practical hindrance to the use of these methods for bioassay of uranium in urine.

Currently we believe that no further experimental studies of delayed neutron counting is necessary to evaluate its use for bioassay. The fission track counting method, on the other hand, has been studied much less and seems to warrant additional experimental study before it can be adequately evaluated.

Task 3: High-Sensitivity Mass Spectrometry and Resin Bead Methodology - Work has begun to experimentally evaluate this method for the determination of U and Pu in urine. At the present time, thorium is not being included in the analysis. Based on experience and literature information, the following method has been adopted to isolate U and Pu from urine. To 10 ml of urine, 20 ml of concentrated HCl, 5 ng of ^{233}U, and 1 ng of ^{239}Pu are added. After a period of time for digestion, the solution is run through an anion resin column to sorb the U and Pu which are then eluted with a solution that is 0.4 M HCl and 0.01 M HF. The volume of the solution is reduced to 0.2 ml and the U and Pu are loaded on anion beads for isotope dilution mass spectrometry measurements. The simplicity of the method should effectively minimize contaminants from glassware and reagents.

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 obtaining written procedures from utilities. Work on the analysis of procedures was begun.

Task Two: Informal Discussions

Discussions were held with personnel at Public Service and Gas of New Jersey concerning problems and issues in the accident classification and off-site notification procedures. The discussions focused on non-technical factors which influenced the classification of transients into the four emergency action level categories. A synthesis of the results of the discussions to date was drafted.

Task Four: Case Study

Arrangements were made with NRC to develop a catalog of incidents which led to a declaration of one of the four emergency classifications. This will be used to select case studies and in developing a simulation experiment.

MEETINGS AND TRIPS:

John Sorensen met with the project manager, Mike Jamgochian, on November 3, at NRC. J. Morrel met with personnel at Public Service and Gas in Salem, N. J.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

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:

No effort this month.

MEETINGS AND TRIPS:

Dr. Allen Brodsky of ORPP met with various staff members on November 22 & 23.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROJECT TITLE: Noise Diagnostic Methods for Safety Assessments

PROJECT MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #B0191)/NRC #60 19 31

TECHNICAL HIGHLIGHTS:

Task 1: Assessment of Neutron Noise Surveillance and Diagnostic Techniques. A report containing the results of baseline neutron noise measurements in LWRs will be drafted in December 1982. An outline of this report was approved by the NRC Research Review Group at the September 1982 meeting.

Task 2: Assessment of Pressure Noise. Plans were made for a trip to a LOFT experiment in early 1983. A paper was accepted for a conference at the University of Tennessee early in 1983.

Task 3: Assessment of Temperature Noise Surveillance and Diagnostic Techniques. Two papers were presented at the ANS Winter Meeting held in Washington, D.C. November 14-18, 1982. See REPORTS, PAPERS AND PUBLICATIONS below.

Task 4: Loose Parts Monitoring Systems (LPMS) Assessment. Plans are being formulated for an exchange of information (theoretical developments, laboratory experiments, field experience) on loose-parts monitoring with research and applications specialists actively engaged in such work in Europe. We are presently corresponding with groups in France (Electricite de France, Saint-Denis) and Germany (GRS, Garching) regarding an exchange of published material and a visit to their research facilities and reactor installations (probably during the spring or summer of 1983) to gain first-hand information.

Task 5: Evaluate New Surveillance and Diagnostic Methods for Reactor System Fault Detection. We received proposals from the University of Washington and North Carolina State University for investigation of new methods for extracting diagnostic information from reactor noise signals. The proposals are being reviewed and subcontracts will be initiated in the next reporting period if the proposed work is deemed important to the goals of this task which is to assess new surveillance and diagnostics methods for detection of faults in nuclear plant systems.

EQUIPMENT PURCHASES:

None.

MEETINGS AND TRIPS:

F. J. Sweeney attended the ANS Winter Meeting held in Washington, D.C. November 14-18, 1982.

The NRC course on PWRs (RP104) was attended by J. A. Mullens.

REPORTS, PAPERS AND PUBLICATIONS:

The following papers were presented at the ANS Winter Meeting, Washington, D.C., November 14-18, 1982:

"Calculation of Kinetic Spatial Weighting Factors in Power Reactors," F. J. Sweeney and J. P. Renier.

"Behavior of Core Exit Temperature Noise RMS in PWRs", F. J. Sweeney.

"Measurement of Core Coolant Flow Velocities in PWRs Using Temperature-Neutron Noise Cross Correlation," F. J. Sweeney and B. R. Upadhyaya.

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. The training systems evaluation model is very near completion. Seventy-five percent of the evaluation criteria and check lists have now been completed. These items have been distributed to subject matter experts external to this program for review. It is expected that all evaluation materials will be completed by the end of December and the initial review of the materials will be completed by mid-January.

A project review group meeting was held in Washington, D. C. on November 17, 1982. Minutes for that meeting will be distributed in the near future. The evaluation model outline along with some examples of the criteria, check lists, and guidelines were presented for review purposes. A structured meeting debriefing will be mailed to all participants to formally collect their expert opinions concerning materials presented at the meeting.

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. A review of 25 of the 40 plant systems has now been completed. An evaluation of level of criticality and level of difficulty has been initiated for each of the malfunctions in these 25 systems. The three rating factors - frequency, criticality, and difficulty - will then be combined in a systematic manner in order to determine what malfunctions should be trained.

MEETINGS AND TRIPS:

A project review group meeting was held in Washington on November 17. D. L. Selby, L. H. Gray, and P. M. Haas from ORNL attended the meeting. Minutes from the meeting will be distributed in the near future.

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:

Work is in progress on the first draft of Chapter 5 (Surface Contamination Surveys) of the manual entitled "Occupational Radiological Monitoring at Uranium Mills". When this Chapter is complete, the whole manual will be in draft form.

MEETING AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Operational Aids for Reactor Operators

PROGRAM MANAGER: W. H. Sides, Jr.

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

TECHNICAL HIGHLIGHTS:

FY 1982

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

R. Pulliam of BioTechnology met at ORNL with R. A. Kisner, P. R. Frey, and W. H. Sides on November 15. Two options were discussed for an example of a documented design process to be used in a demonstration application of BioTechnology's allocation of function methodology: 1) existing data on the design of a HVAC system for a PWR, and 2) data on the redesign of the control room of North Anna 2 by the Virginia Electric Power Co. (VEPCO). BioTechnology will review the documents for the HVAC system and will contact VEPCO concerning the control room redesign.

A draft report by R. Pulliam and R. Maisano entitled "Cognitive Models of NPP Control: A Summary and Interpretation of the Literature" will be incorporated in the proceedings of the Workshop on Cognitive Modeling (see Task 4, below).

A meeting with BioTechnology and the NRC is scheduled for December 16 to discuss the schedule for the remainder of the allocation of function project.

Task 2: Effects of Changes in Automation on Operator Performance

Task was completed in FY-82.

Task 3: Complete the Review and Assessment of Operational Aids

Task was completed in FY-82.

Task 4: Man-Machine Interface Study

The draft report on cognitive modeling provided by BioTechnology (see Task 1, above) is being incorporated in the proceedings of the Workshop on Cognitive Modelling. The proceedings have been assembled and will be in reproduction by mid-December.

FY 1983

Task 5: Methods and Criteria for Evaluation of Operator Aids

A plan for accomplishing this task was devised, and a review of pertinent NRC documents and standards has begun. A decision will be reached in December concerning the need for assistance by subcontract for a portion of this task. If subcontracting is necessary, a Statement of Work will be prepared in December.

Task 6: Review and Assessment of Operational Aids

Data collection is resuming to expand the data base begun in FY-82 under Task 3.

MEETINGS AND TRIPS:

None.

REPORTS:

R. Pulliam, R. Maisano, "Cognitive Models of NPP Control: A Summary and Interpretation of the Literature", Working Draft, October 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

Six additional sets of Emergency Plans were analyzed to determine the consistency of specified roles and responsibilities, bringing the total to ten.

We continued to collect plans from states, local government and utilities that did not respond to the initial contacts.

Task Two: Dynamics of Interface

A set a discussion topics for the informal discussion was developed from the literature review.

MEETINGS AND TRIPS:

J. Sorensen met with the project manager, M. Jamgochian, at NRC on November 3, and with M. Sanders and R. Jaske of FEMA in Washington, D.C. on November 2.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROJECT TITLE: Pressure Sensor/Sensing Line System Evaluation Research

PROJECT 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. A draft report summarizing visits to pressure sensor manufacturers and nuclear plants was completed and is undergoing review.

Another report containing the results of our review of licensing event reports was completed and will be transmitted to NRC for review in December.

Task 2: Pressure Sensor/Sensing Line Model Development. Equations for a low-frequency pressure perturbation test model were completed. The equations predict the type of behavior observed in our simple pressure perturbation tests. Further testing is necessary to determine their accuracy.

Task 3: Evaluate an In-Situ Method for Remote Detection of Pressure Sensor/Sensing Line Degradation. We completed the test procedure for remote in-situ response time testing of Foxboro force balance pressure sensors. The procedure is being evaluated by applying it to both nuclear qualified and standard Foxboro sensors. The procedure and results of tests will be transmitted to NRC in December.

We prepared a cost estimate for utilization of the ORNL Transient Instrument Test Facility (TITF) for evaluating in-situ methods for measuring response time of pressure sensing systems. TITF is a circulating water loop capable of operating up to 1600 psig pressure and 600°F temperature.

MEETINGS AND TRIPS:

A meeting between ORNL, their consultants and E. C. Wenzinger was held in Knoxville on October 7, 1982 to review program status and discuss future directions. The group concluded that assessment of noise analysis methods should be halted and more effort put on evaluation of a pressure perturbation method for measuring response time of pressure sensing systems.

T. W. Kerlin met with L. Finkelstein of City University of London to discuss Finkelstein's methods for modeling pressure transducers.

Kerlin also held discussions with Electricité de France (EdF) regarding EdF's sensor response program and possible use of an EdF test loop for testing U.S. pressure transducers at LWR operating conditions.

REPORTS, PAPERS AND PUBLICATIONS:

None.

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 T. S. Oglesby of Sandia National Laboratories was held at ORNL on November 9-10. In attendance were T. Oglesby and E. Postenrieder of Sandia, D. L. Basdekas, E. C. Wenzinger, and A. J. Szukiewicz of NRC, J. S. Woods of the Union Carbide Engineering Division, and ORNL Safety Implications project staff.

Task B: Generic Study (work applicable to all plants) - A preliminary version of the pressurizer model was developed, FORTRAN coded, and partially debugged. The methodology, based on RETRAN, partitions the pressurizer into potentially non-equilibrium liquid and vapor regions. Two test runs at operating temperature and pressure were made to checkout the vapor-region mass, energy, and volume, and the liquid-region mass, energy, momentum, and volume equations. In the first run, an imposed 10% step-increase in primary system pressure induced an insurge of fluid into the pressurizer. In the second run, system pressure was reduced by 10%, resulting in outsurge. The simulations appeared physically realistic, and inspection of mathematical truncation errors indicated that the Taylor's seried approximations used in state relationships were acceptable, permitting the same non-iterative solutions used elsewhere in the system hydraulics.

Work began on installing the RETRAN code, previously obtained by license agreement with EPRI, on the ORNL IBM-3033 computer. This code may be used for part, or all, of the testing of the hybrid model.

Task C: Babcock and Wilcox Analysis - Work continued on revising the hybrid model of the ICS to reflect recent information extracted from the ZELAP5 computer model of Oconee Unit 1 (available from EG&G Idaho).

The core hydraulics model was expanded to explicitly simulate both loops of the Oconee Unit 1.

We are currently conducting an FMEA with respect to the steam generator overfill problem. The problem separates rather naturally into two parts:

1. mechanisms leading to overfill

2. consequences of overfill.

We are directing our initial attention almost exclusively to the first of these. Analysis is required for important parts of (2). In particular, the case of the overfilled steam generator, with liquid water in the main steam line merits analytic attention. This problem is important because

a. it may directly damage secondary components,

b. a steam line break initiated from these conditions may be much more severe than is usually estimated,

c. these conditions may cause a steam line break.

For point (1) above, the extensive work of B&W has been reviewed. Additional scenarios have been added to those proposed by B&W. In some cases we feel that the B&W event sequences may be ambiguous. Where applicable, postulated component failures have been related to control system causation. Scenarios especially requiring simulation are being selected on the bases of seriousness, unpredictability of outcome, and unavailability of results from other sources.

Task D: Study of Second PWR - No activity this month.

Task E: Criteria - Criteria for selection of systems potentially contributory to overcooling, overfilling and core damage, and for selection of broad FMEA outputs for computer analysis were prepared and sent to NRC.

MEETINGS AND TRIPS:

On November 9-10 ORNL Personnel met in Oak Ridge with NRC staff and with members of the Sandia Project Management Support Division to conduct a network analysis of the proposed plant electrical system program and its interface with the Safety Implications of Control program. See Task A: Program Planning.

On November 16-17, F. H. Clark, R. E. Battle, and W. T. Jewell of ORNL, together with D. L. Basdekas of NRC, met with Sandia personnel at Sandia National Labs (SNL) to gather information developed by Sandia for the Safety Implications of Plant Electrical Systems project. This task is being transferred from SNL to ORNL; it was desired that the transition should take advantage of all information gathered and progress made at Sandia.

REPORTS, PAPERS, AND PUBLICATIONS:

None

PROBLEM AREAS:

Access to detailed design data on each of the plants to be studied remains a problem. Design data for the Oconee-1 model have been obtained from a variety of sources in a laborious and time-consuming fashion. Through the set is complete for development of the plant model, the values used have not been confirmed by Duke Power. Plant specific transient data for model verification are not at this time available, though experiments of the required type have been run by Duke. Continued failure to obtain these data will restrict validation to benchmark program comparisons between our model and a currently approved systems code. The same lack of data holds true for Calvert Cliffs, both for design parameters and for validating transient experiments. A search for these numbers is starting, but lack of data remains a problem area.

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: Criteria Development. Planning for SROA criteria development continued. A general approach to criteria development has been structured and will be reviewed at a GPC/ORNL staff working meeting December 2. The working plan integrates results of this program with the new program of training simulator experiments to begin in FY-1983. Field data from two BWR units collected by MSU has been reduced and tabulated. Forty-five occurrences spread across six types of abnormal events were identified. Final iteration of revisions to the report on calibration of initial simulator results (NUREG/CR-3092, ORNL/TM-8599) are completed and the report is in makup, in preparation for printing.

Task 2: BWR Task Analysis. The review of the GPC draft of the NUREG/CR report on the results of the FY-1982 BWR task analysis by ORNL staff was completed. The document was returned to GPC, and a final draft is expected in mid-December. The application of task analysis to SROA criteria was included in the program planning for FY-1983.

MEETINGS AND TRIPS:

P. M. Haas presented a summary of SROA and other ORNL/NRC human factors programs at the DOE Man-Machine Interface Seminar in Santa Clara, California on November 1 and 2, 1982.

REPORTS, PAPERS, AND PUBLICATIONS:

"Criteria for Safety-Related Nuclear Power Plant Operator Actions: Iniitial Boiling Water Reactor (BWR) Simulator Exercises," A. N. Beare, et al., NUREG/CR-2534, ORNL/TM-8195.

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:

Trends in Power Plant Capital Investment Cost Estimates. Preparation of draft material continues at a very low level of effort.

Validation of Cost Models in the CONCEPT Code. We have developed a more detailed Statement of Work for this task based on conversations with NRC staff on September 15 and will be transmitting this to NRC in early December.

MEETINGS AND TRIPS: None.

REPORTS, PAPERS, AND PUBLICATIONS: None.

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 a second time to mid-December. Therefore, there will be some delay in completing the capital investment cost study and validation of cost models in the CONCEPT code. We will re-examine the schedule when we receive the needed information from UE&C.

PROGRAM TITLE: Environmental Dose Indices

PROGRAM MANAGER: K. F. Eckerman

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

TECHNICAL HIGHLIGHTS:

As discussed in the October monthly report, we are now going to include consideration of the dose equivalent associated with response to the deposits on activity on the groundplane. Calculations of the effective dose equivalent rate (weight sum of the dose equivalent rate in various tissues of the body) for the groundplane exposure have been completed. This data and information on the nuclear decay chains are now being added to the computer code.

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: A statement of work for 1983 fiscal year was received. A 189 to reflect this change in the project scope and budget is in preparation.

Task 3: Meteorological Data: C. C. Gilmore of ORNL continued to examine and correct the GRID 3 tower data from the 1981 INEL field program. These data, once processed, will provide additional insight into the near-field turbulence during the field tests.

Task 4: Atmospheric Models - All available data from INEL have been used in the model evaluation study. As additional test days from INEL become available, these cases will be incorporated in our study.

Task 5: Conference Review - A review of the findings of the major conferences and symposia on data needs and uncertainties has begun.

MEETINGS AND TRIPS:

None.

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

MAJOR MILESTONES ANTICIPATED AND ACCOMPLISHED:

	Date Anticipated	Date Attained
1. Data Base Maintenance	7/82	7/82
2. Model Transfer	9/82	9/82
3. Update Integrated System	2/83	

TECHNICAL HIGHLIGHTS:

Task 1. Work continued on the estimation of SLED. Preliminary results for three more regions have been obtained.

Task 2. Typing continued on the Proceedings of the Workshop on Electricity Demand Forecasting by State Agencies.

Task 3. In a phone conversation on November 15, Colleen Rizy spoke with Stan Hvostik of the Florida Public Service Commission. Stan said the commission would like to have SLED transferred to them when it has been updated.

Task 4. We transferred SLED to T. Gary Williams at the Mississippi Department of Energy and Transportation. They were unable to run the USAD model because it required an IMSL routine to which they are not subscribers. Since then, Brady Holcomb has located a routine which is adaptable to numerous types of computer systems. We are transferring this to Gary and will be transferring it in the future to others. This routine makes USAD adaptable to virtually any system. It is anticipated that this will significantly reduce our role as computer consultants in the transfer process.

Task 5. We continued to consult with Ed Hudspeth.

MEETINGS AND TRIPS:

None

REPORTS, PAPERS AND PRESENTATIONS:

None

PROBLEM AREAS:

None

PROGRAM TITLE: Pathogenic Microorganisms in Closed Cycle Cooling Systems

PROGRAM MANAGER: Webster Van Winkle

PRINCIPAL INVESTIGATOR: R. L. Tyndall

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

TECHNICAL HIGHLIGHTS:

Technical highlights are reported only the last month of each quarter of the fiscal year, per agreement with our Project Manager at NRC, Paul Hayes.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Residual Activity Criteria

PROGRAM MANAGER: R. O. Chester

ACTIVITY NUMBER: ORNL #40 10 01 06 (189 #B0498)/NRC #6C 19 42

TECHNICAL HIGHLIGHTS:

Task 1: Scenario Definition -

A radionuclide Daughter Inventory Generator (DIG) computer code has been outlined and coded, and is being debugged. This code will facilitate the construction of disposal site inventory radionuclide data sets for the PRESTO transport code. The DIG code accepts a table of radionuclides initially present in a waste stream and produces a tabulation of radionuclides present after a user-specified elapsed time. This resultant radionuclide inventory characterizes wastes that have undergone daughter ingrowth before leaching and transport, and lists daughter radionuclides that should be considered for inclusion in the pollutant source term. Output of the DIG code also summarizes radionuclide decay constants, which are required in the PRESTO code input data set.

Data sets have been prepared and stored on disk, containing the nuclide-specific radionuclide concentrations in various waste streams (PWR compactable trash, BWR compactable trash, and PWR secondary resins). These data sets will undergo pre-processing using the DIG code, and the resulting nuclide tables will be used for generation of radioactive waste source terms.

A modification has been made to the PRESTO code to allow independent specification of water taken from surface and sub-surface sources for each of several possible end uses: human consumption; irrigation; and consumption by beef and dairy animals.

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:

None.

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:

Work continued on revisions to population distribution document per suggestions by the sponsor. Revised versions of the protective actions document and the site availability document are under review by the sponsor. Extended tables related to the Major Societal Resource Task were delivered to the sponsor.

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. G. 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 - Revisions of the chapter on geochemical uncertainties in the major report on uncertainties in geologic waste disposal have been completed. The remaining chapters requiring revision before submission of a revised draft report to NRC include the executive summary, introduction, and summary and conclusion.

Task 3: Support for 10 CFR Part 60 Rulemaking - Revisions of the letter report on the role of geochemistry and its uncertainties in the regulation of geologic waste disposal have been completed, but the revisions have not yet been incorporated into the computer file. Every effort will be made to complete this work during the coming month and to send three reports on this task to the NRC.

Task 4: Geologic Records and Future Site Evolution - Transmittal to the NRC of the letter report on modeling capabilities for predicting geologic processes and their effects on waste isolation is awaiting completion of the geochemistry letter report in Task 3.

Task 5: Technical Assistance for Developing HLW Radiological Criteria - The first draft of a letter report on technical issues for environmental radiation standards for high-level waste disposal has been completed and is presently being typed into a computer file for editing and revision. An outline of an environmental impact statement for a high-level waste standard has been prepared by R. E. Thoma of the Energy Division and will shortly be sent to the NRC.

Task 6: Short-Term Technical Assistance - A review of Chapter 16 of the Hanford Site Characterization Report was begun. This review has also necessitated study of several other chapters in the report. A letter report summarizing this review will be sent to the NRC by December 10.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Valence Effects on Adsorption

PROGRAM MANAGER: R. E. Meyer

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

TECHNICAL HIGHLIGHTS:

During the past month, our new controlled atmosphere box was delivered, and we are in the process of installing it. Besides continuing our usual sorption measurements, we are trying to find accurate methods of determining the residual dissolved oxygen in our solutions and the concentration of carbonate species in solutions. For measuring oxygen, we are using a coulometric electrochemical method, and for determination of carbonate we are experimenting with titration with acid making use of an automatic titrator.

MEETINGS AND TRIPS:

None

REPORTS, PAPERS, AND PUBLICATIONS:

Our topical report, "VALENCE EFFECTS ON ADSORPTION - A Preliminary Assessment of the Effects of Valence State Control on Sorption Measurements", NUREG/CR-2863 ORNL-5905, is now in the Technical Publications Department and will shortly be sent to the Reproduction Department for printing.

PROBLEM AREAS:

None

Internal Distribution

1. R. E. Adams
2. S. I. Auerbach
3. S. J. Ball
4. V. B. Baylor
5. E. C. Beahm
6. J. T. Bell
7. W. F. Bethman
8. R. J. Borkowski
9. H. I. Bowers
10. J. C. Brown
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