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Monthly Highlights Report

ORNL Programs for the NRC Office of Nuclear Regulatory Research

A. L. Lotts

August 1982

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-543-75,
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INTERIM REPORT

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INTERIM REPORT

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ABSTRACT

Highlights of technical progress during July 1982 are presented for ORNL research programs 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:

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.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: ASME Code Section III — Technical Support

PROGRAM MANAGER: G. T. Yah.

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

TECHNICAL HIGHLIGHTS:

Task 2: Piping Support Reactions — A report on piping support restraint loads is being prepared by E. C. Rodabaugh Associates, Inc. This report was originally scheduled to be completed prior to the other report being prepared by E. C. Rodabaugh Associates, Inc., under Task 3. At Mr. Rodabaugh's request, the scheduling of these two reports is being interchanged. Therefore, the first draft of the report on piping support is now scheduled for completion in December 1982.

Task 3: Fatigue Evaluation for Class 2 and 3 Piping Components — E. C. Rodabaugh and Associates, Inc., started writing a report on the Code Class 2 and 3 fatigue evaluation method. The first draft is expected to be completed during the next month.

Task 5: Preloading of Bolted Connections — A revised proposal for future research on the problem of preloading of bolted connections was sent to NRC along with cost and milestone schedules. The proposed program is designed to minimize failures of bolted connections by assuring that bolts are preloaded properly.

Task 7: Evaluation of Section III Acceptance Standards and Fatigue Curves Using A Fracture Mechanics Approach — A fixed price-level of effort subcontract was signed with O'Donnell Consulting Engineers, Inc., on July 6, 1982, for technical assistance to resolve NRC concerns on the relationship between the fatigue design curves, crack growth curves, and the acceptance standards in Sections III and XI of the ASME Boiler and Pressure Vessel Code. The subcontract provides 48 hours of consulting time from Dr. W. J. O'Donnell plus three trips to either Bethesda, Oak Ridge, or New York City.

A sample problem description was developed for the terminal end of the Zion hot leg piping based on information in the series of reports NUREG/CR-2189. This sample problem will be proposed for use in a comparison of fatigue life prediction by the S-N curve approach and by the crack-growth approach.

MEETINGS AND TRIPS

None.

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:

A preliminary list of nuclear power plants of particular interest has been received from the NRC. This list includes examples of Mark I, II, and III type containments; ice condensor containments; and reinforced concrete, prestressed concrete, subatmospheric, and double containments. The availability of leak rate test reports for these plants is being determined by a search of the dockets on file with the Nuclear Safety Information Center. Copies of pertinent reports have been ordered through the center. Additional reports not available through the center will be obtained from the NRC.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

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

PROGRAM MANAGER: D. J. Naus

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

TECHNICAL HIGHLIGHTS:

Typing and makeup of the final report has been completed. Preparation of figures has been completed and they are being placed on mats for reproduction.

An abstract has been completed of a report for presentation at the Seventh International Conference on Structural Mechanics in Reactor Technology to be held at Chicago, Illinois in August 1983.

MEETINGS AND TRIPS:

None.

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 1: Program Management, Fracture Mechanics and Analysis — An axisymmetric heat transfer and stress analysis was conducted on an uncracked ITV using the ADINAT and ADINA finite-element codes. The purpose of this work was to determine the effect of the partially uncooled hemispherical head on the stress profile in the cylinder where a long finite flaw is to be located in the pressurized-thermal-shock test. A finite-element model consisting of 1589 nodes and 490 elements was constructed. In addition to a thermal loading, an internal pressure was applied in the stress analysis. Stress contour plots at one, two, four, six and eight minutes into the transient showed a negligible effect of the hemispherical head on the region of the cylinder where the flaw was located.

Task 2: Irradiation Effects — Irradiation of capsule D of the Fourth HSST Irradiation Experiment was completed. Purchase orders have been placed for plate and weld wire for the 4T-K_{Ic} program, and specifications for weldment fabrication are being prepared. Preparation of specimens for the stainless steel cladding irradiation program is continuing.

Task 3: Thermal Shock — Fracture-mechanics calculations were performed in connection with the OCA limit analysis to determine the effect of the assumed length of the transient and of the expected range in fluid-film heat transfer coefficient on critical values of RTNDT. For most cases the effect was small. This completed the OCA limit analysis, and a final draft of the report was completed.

Additional FM calculations were made for the H. B. Robinson, Rancho Seco, TMI-2 and Ginna recorded accidents. In the process, an error in OCA-II concerning the analysis of circumferential cracks was found and was corrected. The analysis of these accidents was completed and a report submitted to NRC.

The FM probabilistics code has been restructured so that the probabilistics analysis is a subroutine to OCA-II. Efforts are now being made to streamline the output associated with the probabilistics analysis.

An effort was commenced to prepare a users' manual for OCA-II.

Task 4: Intermediate Vessel Test — Preparations for the test of vessel V-8A are continuing on schedule. During the month the vessel assembly was completed and transported to the test cell. All equipment has been put in place for the vessel test. Pretest checks indicate that all systems are functioning properly. The assembly is ready for the final pretest calibration of sensors.

Task 5: Pressurized Thermal Shock — Pressurized-Thermal-Shock Test Facility (PTSTF) design and construction activities continue to remain on schedule and within projected costs. Drawings from the chiller vendor were reviewed and accepted, and performance tests on the main circulation pump were completed successfully.

A PTS experiment test matrix peer review was conducted at the Nuclear Regulatory Commission (NRC) headquarters on July 12–13, 1982. Details of the first two tests were discussed, and test facility, test methods and rationale were delineated. Written responses incorporating major comments were requested by the NRC from all interested parties within one month.

Material from TSE-5A is being considered for the first PTS experiment. The material is an A508 class-2 chemistry quenched and tempered steel cylinder. Extensive pretest characterization was conducted on it prior to TSE-5A tests, and the thermal-shock test resulted in valid, lower bound, fracture toughness data as a function of temperature between −10 and 70°C. The use of this material will save time and expense and provide an accurate data base for analysis. However, OCA-I analysis using the subject material properties indicated that a sink temperature on the order of −45°C was necessary to achieve test objectives. This will require modifications to the PTSTF, which has a design sink temperature of −23°C. The nature and extent of these modifications are being investigated.

Task 6: Cladding Evaluations — An additional test, CP-9, on a plate clad with T309/308 weld metal was performed. A pop-in and arrest was observed at loading conditions between the highest and lowest levels previously examined; however, it appeared to be due to geometry effects rather than effects of cladding. The results of the test will be evaluated further before additional testing will be conducted.

MEETINGS AND TRIPS:

Andre Pellissier-Tanon of Framatome and S. S. Palusamy of Westinghouse Electric Corporation visited ORNL on July 1 to review overcooling accident analysis methods and experiments.

R. E. Schnurstein and Armand T. Onesto of Energy Technology Engineering Center visited ORNL on July 8 to review requirements for the pressurized-thermal-shock facility.

HSST staff members participated in a Pressurized-Thermal-Shock Experiment Planning Review Meeting in Bethesda, MD, on July 12 and 13.

On July 14, G. D. Whitman made a presentation on pressure vessel integrity to the DOE-Sponsored Faculty Institute on Nuclear Power: Fission and Fusion, at the ORAU Professional Training Center.

EPRI/CE visited ORNL on July 20 to present their program of research and development on pressurized thermal shock. M. Vagins of NRC attended this meeting and stayed on July 21 for a general HSST program review.

G. D. Whitman described the pressurized-thermal-shock facility concept and construction plan to the ORGDP superintendent's meeting on July 26.

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:

We have bought and installed a rotary stage scanner in the NRC tube positioner. The scanner is driven by a computer-controlled stepping motor and will allow us to change the lift-off on the pancake probes by rotating the probes over nonconducting shims in the tube. This will allow us to include the effects of internal crud in the tubes among the properties we can measure or reject.

Our summer student is working on experimental tests of our defect theory for multiple tubes using large-scale coils and tubes.

MEETINGS AND TRIPS:

C. V. Dodd and R. W. McClung met with Joe Muscara at ORNL to review the NRC programs at the laboratory.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

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

TECHNICAL HIGHLIGHTS:

Task 1: Benchmark Fields -

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

B. ORR-PSF - The Pressure Vessel Simulator (PVS) Capsule and the Void Box Capsule (VBC) will be moved to the hotcells in August for disassembly. It is expected that the metallurgical specimens and dosimetry capsules will be ready for shipment in early September. All participants must have the following items at ORNL by September 7, 1982:

1) Approved shipping cask and certification papers (non US participants);

2) Purchase order to cover handling charges; and

3) Isotope and Technical Service Order Form (DOE Form EV-391).

All items should be sent to the following address:

Mr Harvey Austin/F. B. K. Kam
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, TN 37830

*A Typographical error has been noted in Table 1 of the previous month. The megawatt hours of irradation should be 422957.42 instead of 387423.90.

C. BSR-HSST - The final dosimetry analysis of capsules A and B is in progress. Maps of exposure parameter values at specimen locations will be provided. Completion is expected in August 1982. Also analyzed is the dosimetry of capsule C.

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

A revised draft of a new "ASTM Standard Recommended Practice for the Analysis and Interpretation of Physis-Dosimetry Results for Test Reactors: has been prepared and distributed for balloting by ASTM subcommittee E10.05.

MEETINGS AND TRIPS: None.

REPORTS, PAPERS, AND PUBLICATIONS: None

PROBLEM AREAS: None

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

PROGRAM MANAGER: J. P. Witherspoon

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

TECHNICAL HIGHLIGHTS:

The direct exposure rate from soils contaminated with deposited radioactivity is a function both of amount and distribution of the radioactivity. Analyses of direct exposure rates from soils on decommissioned nuclear fuel sites have been made using assumptions that the activity is evenly distributed on the surface (1 cm depth) and throughout a 17.8 cm (7 inch) depth. Table 1 gives results of this analysis for a decommissioned PWR. Given the same total activity in the soil (pCi/m^2), the direct exposure is 5.6 times higher if the activity is located in the top cm as opposed to evenly distributed within a 17.8 cm depth. This, of course, relates to measurability of soil contamination for certification surveys.

Contamination values (pCi/m^2) in Table 1 represent those given in NUREG/CR-2241 for a 10 mrem/yr dose level to individuals living in wooden frame houses (residential use scenario) on a decommissioned PWR site. In this case, direct radiation accounts for 97% of the 10 mrem/yr dose when the dose estimate assumes activity is on the soil surface. An exposure rate of 3.8 μR/hr would be measurable with survey instruments against a radiation background in the order of 10 μR/hr. If the activity were distributed evenly throughout a 17.8 cm depth, however, exposure levels (about 0.7 μR/hr) would not be possible to measure accurately with common survey instruments. Based on studies of radioactive fallout, it is probable that most deposited radioactivity in soil will be concentrated within the top 5 cm on an actual site. This would yield an estimated exposure rate in the order of 2 μR/hr.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

Table 1. Comparison of soil surface and soil volume activity and exposure rates for a decommissioned PWR site

Radionuclide		Activity in top 1 cm		Activity in 17.8 cm	
	pCi/m^2	pCi/g	μR/hr	pCi/g	μR/hr
^{54}Mn	8.6E+1	6.4E-3	8.1E-4	3.6E-4	1.5E-4
^{60}Co	7.2E+4	5.3E+0	2.0E+0	3.0E-1	3.7E-1
^{134}Cs	2.1E+4	1.6E-1	4.4E-2	8.8E-3	6.9E-3
^{137}Cs	2.4E+5	1.8E+1	1.8E+0	1.0E+0	3.0E-1
			3.8E+0		6.8E-1

DIVISION OF ACCIDENT EVALUATION

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: M. B. Herskovitz

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

TECHNICAL HIGHLIGHTS:

Analysis for Test Runs 053, 054, and 055 in CCTF-II are continuing.

Two Shakedown Reflood Tests were recorded from PKL-II and analysis of these tests are continuing. The results will be discussed with PKL personnel in October, 1982.

Discussion of the UPTF Interface Design of the DP system, Breakthrough Detector, and Drag Body was conducted between ORNL and UPTF personnel.

Two Prototype Breakthrough Detectors were fabricated to accomodate the new hole size in the end box.

A prototype electronic transducer for the Tie Plate Drag Body and the Breakthrough Detector was designed and fabricated. This unit will be tested with changes being made and followed by additional pre-production prototypes.

MEETING AND TRIPS:

R. A. Hess, W. L. Zabriskie, H. R. Payne, and J. E. Smith attended a UPTF Instrumentation Interface Meeting at Erlangen, FRG.

A UPTF Software meeting is planned for the middle of September, 1982.

REPORTS, PUBLICATIONS, AND PAPERS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Advanced Two-Phase Instrumentation

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: 40 89 55 11 5 (189 #B0401)/NRC 60 19 31 02

TECHNICAL HIGHLIGHTS:

This was a transition month for ultrasonic sensor program personnel. Building on the past successes of the program, we were able to meet and exceed several of the program goals: (1) generate high-energy pulses capable of being transmitted over 400' thus giving strong and fast stress pulses in the magnetostrictive rod; (2) construct transducer coils allowing closer coupling to the magnetostrictive material and operation at greater temperatures; (3) develop the electronics to allow time-resolution of less than 10 ns.

In particular, the July milestones of transmission of acoustic pulses over long, curved wave guides and the effective sectioning of the probe were met. These accomplishments allow us to pursue the major program goals: high resolution temperature, density, and level measurements; long, sectioned probes operating in the range and environment necessary for PWR installation; and the ability to remotely excite and read the probes.

MEETINGS AND TRIPS:

None.

REPORTS, PUBLICATIONS AND PAPERS:

None.

PROBLEM AREAS:

None.

PROJECT TITLE: Evaluation of Bundle Heat Transfer Models and Correlations

PROJECT MANAGER: C. B. Mullins

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

TECHNICAL HIGHLIGHTS:

Analysis of the intermediate flow heat transfer test series is continuing. Quality assurance is in progress. The error analysis is ∿80% completed.

Screening of film boiling, void fraction, and steam cooling data recently acquired is continuing and ∿30% complete.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Fission Product Release from Fuel

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

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

TECHNICAL HIGHLIGHTS:

1. Fuel Procurement and Characaterization

No additional information regarding the irradiation data or the expected arrival of the shipment of fuel from five different LWRs has been received.

Further correspondence with H. Albrecht at Kernforschungszentrum Karlsruhe about supplying simulant fuel for our tests has resulted in general agreement of the composition, specimen size, total activity, number of specimens, and time schedule for the fabrication. Two significant questions remain to be resolved; a suitable transport carrier has not been identified, and the total cost has not been estimated. Based on the information currently available, we do not expect cost to be a major problem.

2. Fission Product Release Tests and Results

Following conduct of test HI-2 (1700°C for 20 min in steam), significant contamination of the hot cell and test facility was encountered. This contamination appeared to be associated with ceramic dust released from the furnace during apparatus disassembly, and was eliminated with reasonable effort.

Because of the relatively high fission product release in this test (about 50% of the ^{85}Kr was released, compared to 2.83% in the previous test at 1400°C), the thermal gradient tube, filter package, and some of the furnace components required entirely remote handling. A second hot cell is being used in the disassembly, examination, and preliminary sampling/leaching of these components.

The initial counts for gamma spectrometry of all test HI-2 components have been completed, and leaching with basic solution for iodine removal (and subsequent activation analysis) is in process. Because of the high levels of radioactivity, resulting primarily from ^{134}Cs and ^{137}Cs, changes in counting equipment and technique have been required. Lower level samples have been counted in room 227B, where the multichannel analyzer is located. For the higher level samples, a second detector, located in room 226 and connected to the MCA by long signal cables, has been put into operation. Counting distances up to 12 m, with or without lead attenuation, are employed with this detector, which has required a significant effort for detector calibration.

The operating and release results of the two tests with LWR fuel are summarized below:

Test No.	Temperature (°C)	Time (min)	Zr oxidation[a] (% ZrO_2)	Fission product release (%)		
				^{85}Kr	^{137}Cs	^{129}I
HI-1	1400	30	80	2.83	1.75	2.04
HI-2	1700	20	90	50.0	*b*	*c*

[a]Based on amount of H_2 produced during test.

[b]Accurate ^{137}Cs analyses of all components are not yet available, but preliminary data indicate Cs release was more than 10 times greater than HI-1.

[c]No iodine data currently available.

Following test HI-2, the fuel specimens from both experiments were cast in epoxy resin and transferred to the High Radiation Level Examination Laboratory (HRLEL) for subsequent examination. Transverse sections were cut from each specimen and are being polished for metallographic examination of the fuel and oxidized cladding, and comparison with an untested specimen of the same fuel. Following these studies, selected samples will be prepared and shipped to Argonne National Laboratory for additional investigation of the test effects.

3. Species Identification by Laser-Raman Spectroscopy

Tests with molecular iodine (I_2) have been extended to include the effect of water vapor on the strength of the laser-induced fluorescence signal obtained from I_2. The presence of 2 to 3 atm of water vapor in the 500-700°C temperature range decreased signal strength some, but not sufficiently to preclude detection of very low I_2 concentrations. Because of problems with impurities in some grades of silica, special silica cells have been obtained to conduct additional tests with CsI vapor in vacuum and in other atmospheres.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

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

TECHNICAL HIGHLIGHTS:

BLAST Code Development: The code modifications to delete the dynamics of the helium conservation equations from the BLAST steam generator/reheater system of differential equations have been implemented into a stand-alone version of BLAST. This particular version of BLAST is structured so that it will be easily incorporated into the plant dynamics code ORTAP. This version also incorporates the nodal outlet enthalpy sensitivity to state variables. Preliminary results are promising in that stable computations have been achieved at time-steps two orders of magnitude larger than the minimum steam nodal transport time without artificially increasing superheat nodal volumes. However, accuracy does suffer as the time-step is increased. Testing of the quasi-static helium side modifications is continuing.

Core Code Development: The simplified version of CORTAP used to simulate the dynamic behavior of the FSV core was modified to simulate the 2240 MWT SC/C HTGR. The results were compared with the results of CORTAP for an ATWS loss of coolant transient. The comparison revealed some differences in both dynamic and steady state results. The reasons for the differences are being studied. The reason for most of the steady state difference is that the core coolant flow is split in CORTAP between core channels and side reflectors, whereas the simplified model does not include the side reflector flow.

2240 MWT HTGR Siting Study: Further development work was done on the ORECA code (3-D core) version of the 2240 MWT design. The model for calculating primary system pressure was developed and incorporated into ORECA, along with provisions for inducing and optional depressurization when the PCRV relief valve pressure limit is reached during an uncontrolled core heatup accident (UCHA). Sensitivity studies were run to determine the effects of various assumed natural circulation rates on the pressure excursions. It was found that with only relatively modest increases over the nominal expected values of reverse flow, the pressure relief limit is not reached, and a much faster cooldown is achieved. Various accident scenarios were investigated for up to 90-hr UCHA transients, noting the sensitivity of the maximum core temperatures and time-to-peak values to different model and operation assumptions. A dynamic model for the reference CAHE design was developed and tested. A CAHE "design program" was also written to generate the simulator parameters from the design point operating data. A 3-D core temperature plotting program was written which draws core "temperature surfaces" for a chosen axial slice at a given time. An investigation of fission product transport effects on afterheat redistribution during a UCHA was begun. The method used in the GA CORCON code is being reviewed as a possible approach to be used in the ORECA code.

Fission Product Release from HTGRs: An activity was initiated to define fission product source terms appropriate for severe accident sequence analysis of the Fort St. Vrain reactor. In conducting a review of the data-base information on fission product release from HTGR fuel in general, and Fort St. Vrain in particular, some complexities or potential limitations seem to be encountered. These are: technological evolution, experimental empiricism, and chemical complexity. The continuing fuel particle technological evolution may constrain the development of a comprehensive understanding of fission product release during hypothetical accidents in the FSV reactor. Much of the experimental development of the 1970s may be of limited direct applicability to FSV since work with throium dicarbide fuels appears to have become a lower priority research item during that period. Most of the recent work is on oxide or oxycarbide fuel types, and direct translation of these experimental findings to the earlier thorium dicarbide fuel in FSV may not be totally straightforward. The problems of experimental empiricism exist because a portion of the published articles or reports describing fission product release from HTGR fuel particles are empirical in nature. That is, a given article may carefully describe how a certain fuel was prepared, how it was tested, and what data was collected. Sometimes lacking, however, is a discussion which attempts to relate those results to the previous information on similar or different fuels. Thus, in evaluating information which may be specifically relevant to the FSV reactor, perhaps only tests with actual FSV fuel particles can be unequivocably accepted as being rigorously representative of the fission product release behavior to be expected in hypothetical FSV accidents. Chemical complexity is also a problem because while individual HTGR fuel particles are small in size, each represents a sealed and isolated chemical environment which undergoes many complex chemical reactions during fuel burn-up. The chemical behavior of the fuel and fission products and their attack on the coating layers (which can lead to their eventual release from failed fuel) is dependent upon a number of chemical parameters. When this chemical complexity is combined with the experimental empiricism, it is difficult to compile information on any fuel other than actual FSV thorium dicarbide fuel into an understanding of the fission product release to be expected in hypothetical FSV severe accidents and the mathematical representation of this behavior.

MEETINGS AND TRIPS: None.

REPORTS, PAPERS, AND PUBLICATIONS: A paper summary entitled "Dynamic Computer Simulation of the Fort St. Vrain Turbines" by J. C. Conklin was submitted for the 1983 Power Plant Dynamics, Control, and Testing Symposium.

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)

The characterization of the species giving rise to the weak shoulder at 270 nm has been completed. Although the species had been tentatively assigned to HOI, an examination of the band as a function of pH did not produce the expected shifts in intensity expected for an equilibrium between it and the dissociated form: $HOI = H^+ + OI^-$; i.e., this band should have decreased with the subsequent growth of the OI^- band. Instead, it grew in intensity, leading us to conclude that it is merely an additional OI^- band.

Furthermore, no evidence for HOI has been found in the vapor. We, therefore, conclude that it is not detectable via spectrophotometric means in aqueous solutions (most probably due to a combination of low concentration and low molar absorbtivity) and does not have enough volatility to be found in the vapor.

As an alternative, we have determined the partition coefficient for the chlorine analog, HOCl, since both the HOCl and OCl^- produce well-defined spectra. We find that the HOCl partition coefficient is $16600 \begin{cases} +0 \\ -5000 \end{cases}$ at 22°C and $5000 \begin{cases} +0 \\ -2900 \end{cases}$ at 100°C. Further refinement of these values is expected. Note that Henry's Law partition coefficients for the elements Cl_2, Br_2, and I_2 are calculated to be 2.4, 26, and 95, respectively, at 20°C.

A simple procedure has been developed to convert the iodide ion in NaI into elemental iodine. In this procedure, dilute NaI solutions ($\leq$0.01 N NaI) traced with ^{131}I are treated with iodic acid to give a pH of about 2.0. The iodine released is carried by argon into a cylinder containing pure water. This iodine is then used to introduce known quantities of trace elemental iodine into our sample tubes. Using this procedure to produce iodine traced with ^{131}I, we are now equilibrating a sample to measure iodine partitioning. The sample contains 8×10^{-5} g-at. I/L at a pH of 7.0 and at a temperature of 298 K. An iodine volatility determination under these conditions has just been completed. Data analysis for this run is as yet incomplete.

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 NSPP facility is currently undergoing a maintenance and repair program. Due to aging, exposure to decontamination chemicals, and thermal cycling during the ten steam/aerosol tests much of the gasketing material in the piping networks and vessel flange assemblies has deteriorated and replacement with a high-temperature, chemical resistant, silicone material is in progress. All major valves are being cleaned and inspected. Sampling devices and instrumentation are being inspected and necessary repair/maintenance/calibration is being performed. Completion of this maintenance program is expected in early August.

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

The behavior of the high-fission-yield, alkali metal element, cesium in a core-melt environment is of importance because of its expected volatility and the high probability of its interaction with radioiodine vapor. The well known high-temperature instability of cesium compounds offers a means for observing both cesium release rate and chemical form during a meltdown experiment. This first experiment utilized Zircaloy-clad fuel tubes containing a mixture of cesium uranate (Cs_2UO_4), at the appropriate concentration, and powdered UO_2. The experiment was conducted by heating the fuel tube bundle in three successive steps from about 1400 to 1850°C. The immediate appearance of a dark metallic aerosol over the hot fuel tubes indicated that cesium was being released as the element; however, during transport in a hydrogen-steam atmosphere it changes to a colorless form, presumably $CsOH \cdot XH_2O$. During a total heating time of about 8 min up to a maximum temperature just above the Zircaloy clad melting point (1850°C), the cesium was completely vaporized; further heating to partial fuel tube melting produced no further vapor or aerosol.

Future tests of this type will include mixtures of fuel, SrI_2, Cs_2UO_4 and control rod material. Interaction of the vapors/aerosols from these materials will be of special interest.

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

The QUICK and HAARM-3 codes are being modified for the special requirements of the ABCOVE study. The AEROSIM code has also been given some attention, but it is difficult to determine at this point whether it will be possible to use this code within the time limitations of the ABCOVE study.

Wall-heat-flux data from the NSPP experiments are being analyzed to determine whether this information can be used to predict the rate of steam condensation at the vessel walls. If a simple wall-condensation model can be developed from these data, it may be possible to determine the importance of diffusiophoresis in NSPP steam/aerosol tests. Current efforts are directed towards calibrating the thermoelectric heat-flux sensors.

MEETINGS AND TRIPS:

T. S. Kress attended conferences to discuss aerosol behavior at the following three locations, July 7-14, 1982: Studsvik Energiteknik, Nykoeping, Sweden; Battelle Institute, Frankfurt, FRG; KfK, Karlsruhe, FRG.

REPORTS, PAPERS AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL #40 89 55 10 6 (189 #B0120)/NRC #60 19 13 01

TECHNICAL HIGHLIGHTS:

Evaluation and analysis of B-4, B-5, and B-6 test results continued. This effort is important in that it leads to greater understanding and insight to the deformation behavior in bundles.

MEETINGS AND TRIPS:

R. H. Chapman visited Pacific Northwest Laboratories July 6-8 for discussions on quick-look results of the Battelle MT-4 NRU test and the KfK REBEKA-5 test. A separate meeting was held on July 8 to discuss MRBT, NRU, and LOFT experience with small diameter sheathed thermocouples.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Near-Term TRAP-MELT Verification

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

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

TECHNICAL HIGHLIGHTS:

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

A draft of the preliminary work plan for the TRAP-MELT Verification Test Program was sent to NRC this month. This work plan will be issued as an ORNL interim report.

In work related to planning for aerosol resuspension tests, a survey of the literature on aerosol resuspension theory and experiments was initiated this month.

In preparation for performing a series of preliminary aerosol experiments, the aerosol test section was mounted in the CRI-II facility. Modifications to the test section were completed, and a safety summary was prepared and is now in the process of being reviewed. Because of this, the first experiment will not be performed until sometime in early August.

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

All revisions and improvements to the draft report SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis (NUREG/CR-2672, Vol. I) in response to the comments of peer reviewers have been completed and the report has been submitted to the publications office.

Two additional personnel have joined the ORNL SASA team in response to the need for an early additional effort in support of the MARCH upgrade work. Larry Ott is currently working half-time on this task and will expand his participation to full-time by November 1; George Lawson will work half-time on this project from August 1 through January 31.

Work continues on the preparation of the draft report for Vol. II of NUREG/CR-2672, which concerns the study of fission product transport for the scram discharge volume break analysis. Work is also underway on the determination of the accident sequence for the third Browns Ferry Severe Accident scenario, loss of decay heat removal (DHR) events.

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

The final draft of Section 5 of NUREG/CR-2672, Vol. I, has been completed during July. This material pertains to the effect of operator action during the scram discharge volume break accident sequence.

For the loss of decay heat removal accident sequence at Browns Ferry:

(1) Investigated drywell coolers; found they may be a significant heat sink if the atmosphere is steam-rich.
(2) Investigated RHR heat exchangers; found that only one of the four heat exchange/pump circuits must be operating in order to prevent pool temperature from exceeding 200°F in an extended post-shutdown decay heat removal sequence.
(3) Investigated the need for further information; determined that information on RCIC/HPCI/RHR/CS pump suction piping arrangement will be necessary in order to calculate accurate net positive suction head (NPSH) values.

<u>Group II</u>: Determines and analyzes the events of the accident sequence which would occur following core uncovery, including core melt and containment failure.

<u>Final Work on NUREG/CR-2672, Vol. I</u> (S. R. Greene) During July the final revisions to Chapter 6 and Appendices B, C, and F of NUREG/CR-2672 were completed.

<u>Work in Support of NUREG/CR-2672, Vol. II</u> (S. R. Greene) Discussions with the SASA fission product transport group (Group III) have been conducted to identify the input data requirements (MARCH output) for the upcoming fission product transport analysis of the scram discharge volume break accident at Browns Ferry, Unit I. The majority of the MARCH modifications necessary to accommodate the Group III requests have been made at this time.

<u>MARCH Upgrade Project</u> (S. R. Greene) As a result of previous work in defining MARCH BWR application problems, ORNL-SASA has been asked to participate as a co-contractor with BCL in a short-term MARCH improvement project with the goal of delivering an improved MARCH version (version 2.1) for BWR applications to the NRC by January 31, 1983. A significant portion of the model development effort associated with this project will be subcontracted effort with Dr. Lahey et al., at RPI.

It is BCL's position that MARCH 2.1 will be the final MARCH version developed prior to release of the follow-on MELCOR code (currently scheduled for preliminary release by October 1984). Since the MARCH 2.1 development schedule requires internal release of the code by January 31, 1983, it is evident that some of the most significant BWR MARCH modeling concerns will not be addressed by this version. Briefly, the final version of the MARCH 2.1 development plan calls for ORNL to:

(1) Identify those analytical capabilities which are both necessary and desirable for realistic BWR severe accident evaluation;
(2) Based on (1), perform an assessment of the capabilities and limitations of the MARCH 2.1 code for BWR application and issue a report thereof;
(3) Develop a simplified channel box model (based on RPI's recommendations) which is compatible with the existing MARCH core meltdown models;
(4) Develop realistic BWR ECCS pump flow models for inclusion in MARCH 2.1.

In anticipation of these efforts the Group II staffing level has been increased with the addition of Larry Ott and George Lawson.

Work at RPI on the detailed BWR core melt model continued during July. While testing the computer program which calculates the fuel pin and cladding temperature distribution, an unexpected phenomena was encountered. The results of the calculations indicate that due to the heat generation of the Zr-H_2O reaction, the unoxidized Zirc can begin melting <u>before</u> melting occurs at the inner cladding surface. Two new models were developed to simulate this phenomena. Work also was initiated on the detailed channel box and control rod heatup and oxidation models.

Pressure Suppression Pool Modeling (D. H. Cook) Pressure suppression pool (PSP) response modeling involves consideration of energy addition through the downcomers and energy addition through the T-quenchers. The downcomer modeling was improved during July in support of the ongoing SASA study of loss-of-DHR sequences involving LOCAs within the drywell.

As discussed in the monthly report for June, the computer program PACMAN* has been written to predict PSP response to low- and medium-mass flux steam addition through the downcomers. PACMAN is a nine-node lumped parameter model of one PSP bay coupled to a single node representing the airspace above. This code is essentially a simplified analysis tool for scoping use until the major code development effort is complete. During July, work was performed to verify the approach used in PACMAN and to improve the method of energy addition to each node.

A more rigorous model of the PSP is being performed for the major effort. During July, a five-equation set which models 2D geometry, transient transport of mass, momentum, and energy in the PSP was adopted. Also, a finite difference algorithm was chosen for solving the coupled set of nonlinear PDEs. Future efforts will be in programming the algorithm.

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 and Aerosol Collection in the Standby Gas Treatment System (SGTS) (R. A. Lorenz) During the scram discharge volume (SDV) break base case accident, the SGTS is expected to operate for only 30 min following failure of the drywell electrical penetration seals. Significant amounts of molecular iodine (I_2) and methyl iodide (CH_3I) will be trapped by the impregnated charcoal, and larger amounts of fission products associated with aerosol particles will be collected on the prefilters and HEPA filters in the SGTS. The SGTS flow system is assumed to fail because the high inlet gas temperature will deteriorate the electrical insulation or other electrical components or will cause bearing failure. Although the aerosol concentration and transport calculations have not yet been performed for the SDV base case accident, it is expected that after 30 min operation collecting the core melt-concrete interaction aerosol will plug the HEPA filters.

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*Pool And Containment Mixing ANalysis.

For example, tests with several different types of aerosols show that ~3000 g of particles will plug (cause a drastic increase in flow resistance) a standard 1000-cfm HEPA filter either distributed between a prefilter and a HEPA filter or on the HEPA filter alone. If the core melt-concrete interaction produces 150 g/s aerosol and half of this reaches the SGTS, the flow rate through the system will decrease to less than 50% of normal after 18 min of operation because of filter backpressure.

The temperature of the gas flowing into the SGTS may approach 200°C (392°F) before failure of the flow system. Fission product decay heat from radioactive iodine on the charcoal and other fission products collected on the filters will also cause heating. Heat from oxidation of the charcoal will contribute additional heat equivalent to 1 cal/min per gram of charcoal in the 200–225°C range. Iodine desorbs from charcoal when heated, and a high radiation field further enhances desorption. Desorption depends on the temperature, flow rate of gas through the charcoal bed, and duration of flow. Experiments show relatively little dependence on charcoal type at high temperature in a radiation field. The following table illustrates the magnitude of the problem for a 75-min desorption. A flow velocity of 20 cm/s is representative of normal flow.

Radioactive iodine desorbed after 75 min flow (%)

Temperature [°C (°F)]	Superficial velocity measured at 25°C (cm/s)		
	0.2	2	20
100 (212)	5.7×10^{-7}	9.7×10^{-6}	4.8×10^{-2}
200 (392)	3.6×10^{-5}	1.0×10^{-3}	7.9
300 (572)	7.1×10^{-4}	3.1	99.5

Because of failure of the SGTS flow system in the accident under study, we do not expect large amounts of iodine to be desorbed.

Aerosol Transport Analysis (A. L. Wright) The HAARM-3 aerosol code will be used, as in the Station Blackout sequence, to calculate aerosol behavior in the drywell and in the reactor building/refueling floor for the present SBLOCA sequence. The programs previously used for the Station Blackout calculations were modified slightly and are again operational. The HAARM-3 input data needed to calculate aerosol behavior in the drywell for the SBLOCA sequence is now being prepared.

Methyl Iodide Formation Rate (J. W. Nehls) We are reexamining the methyl iodide formation rate model employed in the earlier study on complete station blackout. The earlier model was based on a linear production rate over a 4-h period up to an equilibrium level predicted to depend on the iodine concentration in the gas phase. The equilibrium concentration was obtained from an earlier review.

Our reexamination of the available formation rate data indicates that while the 4-h formation period (up to the equilibrium org-I level) is valid, the rate is not linear with time. Initial rates are more rapid than the linear rate, with the predicted equilibrium level slowly approached at ~4 h. In addition, we find that the equilibrium level is somewhat temperature dependent as well as I-concentration dependent.

Using this data review, we have revised our organic iodide formation rate model.

Fission Product Transport Calculation (C. F. Weber) During the past month, we have begun implementing necessary modifications in the fission product transport code. These changes have been concerned with:

1. enhancing and revising the coding used in the station blackout calculations so as to be more accurate and efficient,
2. implementing necessary data and coding alterations to handle the new SDV leak sequence.

Modifications in the first category are essentially complete, and those of the second category should be complete within a few weeks. As soon as this is done, certain changes in the fission product chemistry models will be implemented. Among such changes will be the inclusion of transport models for cesium.

MEETINGS AND TRIPS: On July 15, R. M. Harrington, S. A. Hodge, R. P. Wichner, R. A. Lorenz, and J. A. Nehls attended a meeting at TVA headquarters, Knoxville, to discuss the Browns Ferry Reactor Building and Standby Gas Treatment System (SBGTS) response to the Severe Accident conditions imposed by the postulated Scram Discharge Volume (SDV) break accident sequence. This information is necessary to the fission product transport analysis, and a great deal of useful information was exchanged. In particular, TVA personnel pointed out that the reactor building fire protection system would spray into the Reactor Building under the harsh conditions imposed by this accident; this factor is now being incorporated into the ORNL reactor building code analysis models.

S. R. Greene traveled to Brookhaven (BNL) and met with Trevor Pratt and other members of the BNL Nuclear Energy Department, Accident Analysis Group, on July 20. Pratt's group is responsible for NRC/NRR's analysis of the Limerick and GESAR PRAs. Greene visited BNL at the request of Dr. James F. Meyer (NRC/NRR) to present a summary of his findings regarding the problems associated with application of MARCH to BWR severe accident analysis. Pratt has indicated that he will forward a written summary of BNL's BWR MARCH application concerns to the ORNL SASA team in the near future. Listings of BNL's MARCH input data sets from the BNL Limerick studies conducted to date were also obtained during the visit.

Also on July 20, S. A. Hodge and D. H. Cook attended a review board meeting at the University of Tennessee to establish the candidacy of the latter for his BWR Pressure Suppression Pool Analytical work to gain doctoral status. Cook's presentation and thesis defense accomplished the objective.

S. R. Greene, S. A. Hodge, and L. J. Ott met at Pittsburgh with representatives of NRC, BCL, RPI, and BNL on July 26 to establish final planning for the MARCH 2.1 development effort.

REPORTS, PAPERS, AND PUBLICATIONS: Volume 1 of the report NUREG/CR-2672, SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis has been submitted for publication.

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:

One paper, Health and Safety Standards: Theoretical Rationale and Application to Safety Goals for Nuclear Power," has been distributed for review. One review has been recieved to date. Much of our work this month has been applied toward writing our report on risk aversion. A draft of this report, titled "On the Societal Impact of Multiple-Fatality Accidents," should be completed soon.

MEETINGS AND TRIPS:

Baruch Fischhoff has been working recently at the Applied Psychology Unit of the Medical Research Council in Cambridge, England. On July 2nd, at the invitation of the United Kingdom Atomic Energy Agency (UKAEA), Fischhoff presented a full day seminar in which he described the work being done at Decision Research on topics relevant to nuclear power safety regulation. Specifically, Fischhoff gave three lectures, each 1-1/2 hours long and each followed by 1/2 hour of discussion. The titles of these lectures were:

1. Judgmental Aspects of Probabilistic Risk Analysis
2. Setting Safety Goals for Nuclear Power
3. Public Perceptions of Risk

The lectures were arranged by the National Centre of Systems Reliability, a unit of the UKAEA Safety and Reliability Directorate. The importance attached to these talks is indicated by the fact that special busses were arranged to transport people to the lectures from outlying sites. More than 70 persons attended the day-long meeting.

REPORTS, PAPERS AND PUBLICATIONS: See above.

PROBLEM AREAS: None

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

PROGRAM MANAGER: K. F. Eckerman

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

TECHNICAL HIGHLIGHTS:

Further review and examination of calculational models and their information needs have been underway. An additional paragraph, at the request of R. Rawl (DOT), was prepared and forwarded to Rawl. We expect to be receiving comments from other participating countries shortly for review.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Analysis of Reliability Data from Nuclear Power Plants

PROGRAM MANAGER: J. P. Drago

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

TECHNICAL HIGHLIGHTS:

Task 1: Reports. The general methodology report NUREG/CR-2641 (ORNL/TM-8271) was published. A draft of the pump report was completed and will be submitted to NRC for comment in the first week . August.

Task 2: Data Collection and Encoding. The ORNL subcontractor, SAI, completed grading additional valve-related failure and repair records for Plants 1 through 4. A total of 3128 new valve failure and repair record pairs from Plant 1 were transmitted to ORNL. Encoding of failure and repair data from Plants 2, 3, and 4 is in progress.

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

Task 4: Data Analysis. Maintenance frequency and failure rate calculations for remaining pumps in Plant 1 were completed. Calculations for Plants 2, 3, and 4 will be completed in August.

Task 5: Comparison with Other Data Bases. This task is complete.

Task 6: Human Error Data. No activity this month.

MEETINGS AND TRIPS:

A trip was made by J. P. Drago on June 12 to Northeast Utilities in Hartford, CT to discuss the objectives of the In-Plant Reliability Data program and invite NU to participate in the program.

REPORTS, PUBLICATIONS, AND PAPERS:

Published the In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report, ORNL/TM-8271.

A paper proposed for the 4th Euredata Conference describing the IPRDs has been accepted.

PROBLEM AREAS:

None.

PROGRAM TITLE: Common-Cause Evaluation (CCE) in Applied Risk Analysis

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #41 88 55 03 7 (189 #B0459) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

A final report has been completed by subcontractor and will be published at ORNL as an ORNL/TM document. The report is in final editing at ORNL.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Common Cause Failure Analysis Procedure

PROGRAM MANAGER: G. F. Flanagan

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

TECHNICAL HIGHLIGHTS:

Programming efforts for the common cause screening program are approximately 90% complete. Program testing has begun using several sample problems that will exercise all options of the screening program.

The following information has been received for use in the Arkansas Nuclear One Unit 1 analysis:

1. ANO-1 IREP Main Report and

2. Appendix B.5 - Emergency Feedwater System.

The remaining appendices to the IREP report are still needed for analysis of other systems that appear in the accident sequences.

A preliminary ANO-1 analysis scope has been forwarded to G. F. Flanagan. Comments or changes are needed within two weeks in order to complete the problem definition.

A draft copy of the NREP procedures guide has been received and is being reviewed by the project staff.

No information has been received for the Browns Ferry IREP analysis.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS AND PUBLICATIONS:

D. P. Wagner has been invited to present a paper on past and current common cause failure analysis work at the Tenth Water Reactor Safety Research Information Meeting in October, 1982.

PROBLEM AREAS:

Prompt delivery of additional information for the plant analyses is required in order to maintain project schedules.

PROGRAM TITLE: Definition of Scenarios and Evaluation of Methodologies 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:

The final 189 for the project was completed and submitted to NRC by M. Siman-Tov on August 13, 1982. The title of the project has been slightly modified to emphasize evaluation of methodologies for analysis, and some task numbers have been rearranged. Task numbers used below reflect these modifications in the final 189.

Task 1. Literature Review and Scenario Identification

Task 1A. NRC accident information will be utilized for identifying bounding UF_6 releases at NRC facilities. Approximately 95% of the NRC documents requested from the NRC project contact, Steve Bernstein (NRC-RES), and the NRC Public Document Room were received by mid-July. Review of the documents (about 5,000 pages) reveals that detailed accident scenario information for UF_6 releases is generally not available in NRC public documents.

Accidents postulated by NRC licensees focus on UF_6 cylinder rupture or cylinder valve rupture/failure as bounding or maximum release events. With respect to postulated UF_6 incidents, the NRC-licensed facility documents rarely include considerations of health effects of UF_6 releases.

To gather additional information on potential UF_6 accident scenarios, visits by a three-man team from UCC-ND to four NRC-licensed facilities have been arranged with NRC assistance:

Facility	Location	Date
Allied Chemical Corporation UF_6 Production Plant	Metropolis, IL	8/10/82
Kerr-McGee Corporation UF_6 Production Plant	Gore, OK	8/12/82
Nuclear Fuel Service Fuel Fabrication Plant	Erwin, TN	8/19/82
Westinghouse Electric Company Fuel Fabrication Plant	Columbia, SC	8/24/82

Task 1C. UCC-ND project team members, J. Dykstra and J. L. Gamble, prepared a list of potential UF_6 accidents at UF_6 handling facilities which they compiled based on diffusion plant experience and safety evaluations. As in the NRC-literature review, cylinder integrity is the principal safety concern.

Task 3. <u>Review of Analytical Models</u>

Task 3A. D. D. Holt completed a letter for internal distribution entitled "Preliminary Review of NRC Accidental Analysis Handbook (AAH) Models for Accidental UF_6 Releases Simulation," which was distributed internally on July 14, 1982.

Task 3B. D. D. Holt (UCC-ND) is researching NRC literature for UF_6 release models. His review has identified only cursory and unsubstantiated modeling of UF_6 releases. Details of several studies performed by NRC licensees were not available at NRC; therefore, such details have to be obtained from the NRC licensees.

MEETINGS AND TRIPS:

J. Dykstra and M. Siman-Tov attended the Fuel Cycle Facility Safety Research Program Review Group Meeting #8 on July 21-22, 1982, at Norwood, Massachusetts. This meeting provided us an overview of the ongoing efforts of other NRC AAH contributors.

REPORTS, PAPERS AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Evaluation of Pressurized Thermal Shock

PROGRAM MANAGER: R. C. Kryter

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

TECHNICAL HIGHLIGHTS:

Probabilistic Risk Analyses

Work continued both on refining probabilities for "generic" events and on establishing probabilities for those event branches that require specialized analyses. In the latter category, probabilistic analyses for the MFW runback and for the MFW pump trip on high SG level events were completed and documentation of the methods and assumptions used was begun.

Documentation of the methodology used in constructing the Oconee PTS event trees, both detailed and reduced, was also continued. In addition, specification of a second set of four overcooling scenarios for detailed T-H analysis by LANL and INEL was initiated, in order to mesh with calculational schedules agreed upon by DRA and DAE.

Establishment of data needs and review of the Calvert Cliffs FSAR were begun in connection with our upcoming meeting with the plant owner, Baltimore Gas and Electric.

Thermal-Hydraulic Modeling

"Scoping study" code. Under subcontract to Oak Ridge, SAI continued their development of a capability for performing scoping-level calculations. The status of the various work areas is as follows:

- RELAP-5 installation--solved problems with excessive I/O count and dynamic loading error; are now restructuring logic functions to reduce code running time on IBM. All sample problems have been run and we have started running Oconee stand-alone feedtrain cases.

- Property table verification--received code cycle 18 updates and incorporated them into program. We plan to run revised code with stand-alone steam line model next.

- Control systems modeling--completed last month; issued additional explanatory information in late July to improve understanding of other project participants.

- Feedtrain modeling--reported results from stand-alone runs on July 15, 20, and 30; final input stream was provided to INEL, LANL, and BNL on July 30.

- Primary system modeling--completed interface with secondary system; debugging continues.

- Aux-FW and MSL modeling--scheduled for development and interfacing in August.

"Production" codes. At Los Alamos, a stand-alone model of the Oconee secondary was developed and tested for its steady-state performance. This model provides a closed-loop representation of the secondary systems and includes steam lines, valves, condenser hotwell, and feedtrain components. The secondary model was then combined with the previously developed primary model, and a calculation of steady-state operating conditions was performed on the CRAY-1 computer. Results are presently being assessed.

Using only the primary system model, a transient test problem was also executed independently on both the LANL CDC-7600 and CRAY-1 computers; the results agreed quite well.

At INEL, studies were performed on the modeling of the steam generator secondary to improve the calculated exit quality. Alterations to the noding and code model did not significantly improve the results; therefore, the present model will be used.

The final SAI model of the feedtrain was delivered to INEL on July 30. This model will be reviewed and will form the basis of the feedtrain model used for the INEL calculations. Final information on the SAI model of the B&W integrated control system was received on July 26. INEL's review of this ICS model for inclusion in the overall plant simulation was begun immediately.

MEETINGS AND TRIPS:

July 15 (Oak Ridge): Reviewed program progress and results, schedular slippage, programmatic concerns raised by R. B. Minogue on June 29, etc. with Carl E. Johnson, NRC:RES:DRA.

July 30 (Silver Spring, MD): Presented models of B&W Integrated Control System (ICS) and Oconee feedtrain developed for ORNL by SAI, justified simplifying assumptions and embodied data, and shared stand-alone results obtained to date with other PTS study participants.

REPORTS, PAPERS, AND PUBLICATIONS:

July 1: Transmitted to Duke Power Company, per their request, a copy of the ORNL-developed OCA-I (overcooling analysis) computer code.

July 6: Transmitted to Los Alamos, per their request, a copy of the Oconee-1 ICS manual, noding diagrams for the models then being developed by SAI, input streams for four active model types, and boundary conditions and results for one 1000-timestep case of interest.

July 8: Transmitted to LANL, INEL, BNL, Duke Power, and NRC documentation (word description, block diagram, computer input stream, etc.) of the SAI-developed Oconee ICS model for the guidance of, and/or direct incorporation by, the groups responsible for performing PTS T-H transient calculations.

July 20: Transmitted to LANL, INEL, BNL, and NRC initial stand-alone feedtrain results from MSLB and turbine trip simulations.

July 23: Transmitted to recipients of July 8 documentation additional clarifying information on the ICS model suitable for PTS studies.

July 30: At request of INEL, provided them with same information sent to LANL on July 6. Also, provided all PTS study participants with finalized input streams for ICS and feedtrain models and with graphical representations of results obtained by SAI through stand-alone operation of these models.

<u>PROBLEM AREAS</u>:

Programmatic concerns (particularly schedular slippage) raised by R. B. Minogue during his June 29, 1982 review of the FIN B0468 integrated assessment program were addressed in a letter dated July 27, 1982 from A. L. Lotts (ORNL) to R. M. Bernero (NRC:D/DRA), which presented for NRC's consideration a possible alternative work arrangement.

Misunderstandings regarding the responsibilities of and the interfaces among the various organizations participating in the T-H modeling of Oconee surfaced at the July 30 meeting in Silver Spring. Steps will be taken to prevent a recurrence of such ill-defined interfaces in the FY 1983 studies directed at understanding PTS in representative CE and <u>W</u> plants.

PROGRAM TITLE: LWR Accident Sequence Precursor Study

PROGRAM MANAGER: Wm. B. Cottrell

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

TECHNICAL HIGHLIGHTS

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

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

Task 3: In-depth review of 1980 LERs was completed and the in-depth review of the 1981 LERs is nearing completion.

MEETINGS AND TRIPS

1. Program Review at NRC, Wm. B. Cottrell and J. W. Minarick met with F. M. Manning and P. W. Baranowsky, July 13, 1982.

2. ASP Presentation for NRR, Wm. B. Cottrell and J. W. Minarick, July 14, 1982.

3. ASP Presentation for NRC Commissioners, R. M. Bernero and J. W. Minarick, July 20, 1982.

REPORTS, PAPERS, AND PUBLICATIONS

None

PROBLEMS

None

PROGRAM TITLE: LWR System Survey for PRA

PROGRAM MANAGER: Wm. B. Cottrell

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

TECHNICAL HIGHLIGHTS

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

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

NRC forwarded $15,000 to permit the publication of an updated report, which will include the six Task B studies as appendixes. Work will be initiated on this revised report in August.

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 4: A new task to conduct a workshop on the Propagation of Uncertainties (POU) was added to the FY82 program brief. Plans are underway to conduct a workshop on Models in POU, during October 7-8, 1982.

Technical Assistance: There are three technical groups interested in bidding on a new project on Accident Sequence Precursor Methodology, sponsored by NRC. We are in the process of identifying one of the groups to assign the task.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS AND PUBLICATIONS:

A report: Survey of Error Propagation in Systems, by V. R. R. Uppuluri and W. Kuo is in publication ORNL/CSD/TM-190.

PROBLEM AREAS:

None.

PROGRAM TITLE: Risk Analysis Evaluations

PROGRAM MANAGER: G. F. Flanagan

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

TECHNICAL HIGHLIGHTS:

Task 1: Audit Analysis - A draft description of the audit analysis for the Zion Probabilistic Safety Study was prepared. The report of the Zion Study was received from the utility and an initial review of the document initiated.

Task 2: Regulatory Decision Making - No activity.

Task 3: No activity.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS AND PUBLICATIONS:

None.

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:

The Computer Program SETS was obtained from the Sandia National Lab and the effort to implement the code on the SNL computer was initiated. It is expected that during the month of August the code will be put in full operational status.

The dominant sequences of the IREP study for Calvert Cliffs-I Power Plant were assessed and all relevant systems in these sequences were identified. The computer files for these systems were obtained from the Science Applications, Inc.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Utilization of Risk Analysis and Risk Criteria

PROGRAM MANAGER: C. F. Flanagan

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

TECHNICAL HIGHLIGHTS:

A. K. A. Solomon (Rand) reviewed on behalf of ORNL and NRC "Health and Safety Standards: Theoretical Rationale and Applications to Safety Goals for Nuclear Power," Baruch Fischhoff of Decision Research of Eugene, Oregon. This review contained six general comments and seventy specific comments. The general comments are summarized below:

1) This draft is far too wordy. It often takes a few pages to get to the point. Sometimes it is difficult to identify the point that it is trying to be made; it is hidden among lower level or unrelated information.

To tighten up the draft, the author should first list the points that he is trying to make and should then express those points in the beginning of the appropriate sections. The balance of the section should then be devoted to verifying those points. Those "interesting but non-supportive facts" should either be moved to an appendix or eliminated completely.

2) The sections are poorly structured and often do not flow together. In particular, the two primary sections - theoretical and application - seem almost to be stand alone documents.

Many of the sections and the paragraphs suffer the same problem of appearing disjoint.

To help overcome this problem, the author might wish to strengthen the Introduction by adding a "roadmap" of the chapters to follow. This map should conceptually resemble a fault tree showing how the sections and subsections relate and follow one another. The purpose of the overall paper and each of the sections should be stated as well.

3) Many of the assertions and/or findings are self serving. For example, "... more research on risk perception is needed."

Closely related to that problem is that it is difficult for the reader to distinguish between what is an assertion and what is a conclusion. Some of these assertions/conclusions require better documentation. Some of the conclusions appear somewhat independent of any of the supportive arguments made before. Some seem merely like good common sense.

Many arguments are passed over too quickly, are totally unsupported, and are too qualitative or too soft to be of extreme use.

4) The document as a whole raises more issues than it solves. In that sense it is descriptive or the problem and not prescriptive of the solution. While there is nothing wrong with a tutorial, it should be packaged as exactly that.

5) The last several pages appear to be devoted to a review of NUREG 0880. If this draft were to eventually become a NUREG as well, then such a review would not be appropriate because it reflects only a single position. What might be appropriate is a brief summary and analysis of the primary reviews of NUREG 0880 (EPRI, IEEE, ScubedF, etc.).

6) In my judgment, the phrase "acceptable risk" is often misused. An agreed upon goal in no way implies that if the goal is met, then the risk is acceptable. It merely means that the risk is below some prespecified standard.

On July 28th K. A. Solomon received a note from B. Fischhoff acknowledging receipt of these comments.

During August or September 1982 K. A. Solomon will meet directly with B. Fischhoff in order to assist in providing a constructive rewrite of this draft.

B. In a letter dated June 16th, Kuljian Corp. agreed to provide a draft report by July 15, 1982. As of July 30th, such a draft has not arrived at Rand; nor has any written progress report arrived in several months. By the first week in August, Rand will notify Kuljian in writing of Kuljian's violation of the Kuljian-Rand sub-subcontract and Rand's intention to immediately terminate said.

C. K. A. Solomon and P. Rathbun agreed on a final version of the FY82 workscope. The tasks agreed upon are summarized below:

<u>Task 1: Review and Methodology Development</u>

A. Review

Review PLG-0209; Vol. 12 of Indian Point Probabilistic Safety Study PRA Procedure Guide; Vol. 1 and Vol. 0/2 (aka NUREG/CR-2300) and other mutually agreed upon studies. Specifically examine the treatment of uncertainty.

B. Develop Method

To provide guidelines for NRC decisions involving risk and develop a preliminary approach for specifying the degree of confidence required on information involving risk values. This preliminary approach is to be based on experimental literature, de facto experience, and Delphi interviews. As noted above, the method will be extended to other situations including analysis of various consequence categories, including some CLASS 9 accidents.

C. Report Approach

Within 2-1/2 months of start date (September 1, 1982) brief Task 1 and follow with a written text of the briefing.

Task 2: Detail Method

A. Detail Method

Sub-task 1 - Develop a taxonomy of potential uses of risk analysis results in the NRC regulatory and licensing processes. Potential uses should be explicit with examples along with information requirements needed to support licensing decisions.

Sub-task 2 - Review selected PRA results chosen by NRC and by Rand and develop appropriate specific methodology for the formal potential applications identified in Task 1. The PRA's selected for review will include those that provide (1) only point estimates, (2) point estimates and associated confidence limits, and (3) point estimates and Bayesian limits. Particular emphasis should be given to the uncertainty in information in the proposed methodologies for the decisionmaking process.

Sub-task 3 - Risk assessment study results are often provided in terms of a number of different measures, e.g., early fatalities, latent cancers, property damage. This task requires the development and documentation of multi-attribute procedures for utilizing these varied outputs in decision making. The procedures should be applied on a selected PRA for demonstration.

Sub-task 4 - Develop methodology to utilize PRA results to determine if the proposed deterministic criterion in NUREG 0880 is met with a high degree of confidence. Evaluate proposed risk criteria using selected PRA results (propose alternative criteria if risk assessment results cannot be used to evaluate proposed criteria).

B. Performance measures

Develop performance measures for deciding what constitutes a sufficient band of information in deciding the appropriate confidence intervals and various consequence categories; i.e., in deciding what is a reasonable methodology. The performance measures will include but not be limited to:

- Nontime variance; i.e., for fixed levels of information, the results of applying the method remain consistent.

- Simplicity

C. Decision Rule

To the extent possible, develop a decision rule for what constitutes reasonable and sufficient information in judging the tolerable level of uncertainty; i.e., specify the appropriate confidence levels.

Task 3: Form of PRA Output

Identify format of PRA output and uncertainty analysis consistent with the application of Task 2-C. Identify the nature of distribution needed, the level of detail, and the form of the data.

Task 4: Provide Value Impact Guidance

Work with Los Alamos and Sandia, where applicable, to provide constructive review dealing with value impact, secondary costs, business costs, and related topics.

Task 5: Other Guidance

Where applicable and when agreed upon, provide consulting, guidance, and review to NRC staff.

During this reporting period, Rand has made substantial progress on Task 1 by reviewing a number of documents; on Task 2 by developing and detailing an aggressive approach for establishing an uncertainty decision rule; on Task 3 by identifying the qualitative nature of the data that NRC needs regarding risk and uncertainty in order to apply the decision rule; and on Task 4 by providing to David Strip (Sandia) a written set of notes dealing with how to estimate primary and secondary and direct and indirect costs of a reactor accident.

D. K. A. Solomon and P. Rathbun met with Al Benjamin (Sandia) and David Strip (Sandia) on July 27th. Details of this meeting are discussed under Meetings and Trips.

MEETINGS AND TRIPS:

On July 26 and 28 K. A. Solomon met with P. Rathbun to detail the project's progress. A set of notes was provided to P. Rathbun at this time. P. Rathbun is currently in the process of revising Task 4 - Value Impact, to reflect a more specific and detailed scope.

On July 27th both K. A. Solomon and P. Rathbun met with Al Benjamin and David Strip to exchange information on Sandia's and Rand's NRC supported work. K. A. Solomon provided to Al Benjamin and David Strip written material. Benjamin and Strip agreed to forward written documentation to Rand. Rand's specific needs are:

(1) A listing of the 180 generic safety issues (from Roger Blond, NRC)

(2) Any event and fault tree analysis of one or several of these issues (from Sandia)

(3) A costing of generic fixes along with estimated reductions in risk for one or several of these issues (from Sandia)

D. Strip was provided with a review of his draft NUREG/CR-2723, and with a Rand Working Draft outlining Rand's approach at estimating secondary and indirect costs following a nuclear power plant accident (WD-1575-ORNL, "Classifying the Costs Associated with Nuclear Reactor Accidents"), July 23, 1982.

REPORTS, PAPERS, PUBLICATIONS:

Forwarded to ORNL was a revised set of mats of Rand N-1808-ORNL. ORNL will either approve the format of these mats or provide specific comment. ORNL may forward one or more copies to NRC for their comment on format.

PROBLEM AREAS:

See item B under Technical Highlights.

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:

Documentation on the uranium and alkali metal bioassay models is being prepared. About 2 man-months of effort are required to document our findings at this stage of this project.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Continuous On-Line Reactor Surveillance System Evaluations

PROGRAM MANAGER: D. N. Fry

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

TECHNICAL HIGHLIGHTS:

Task 1: In-Plant Demonstration of On-Line Noise Surveillance System.
Magnetic tape recordings of 14 Sequoyah noise signals were obtained on June 8 and July 17. The system performed well during June and July although storage area is becoming limited.

The first draft of a hardware and software description of the automated noise surveillance system was completed.

NRR contacted TVA licensing office regarding extension of the demonstration at Sequoyah. NRR is preparing a formal letter request to TVA.

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

Task 3: Experiments in Test Facilities.
Because of funding limitations and the need to complete on-going projects, we have decided not to participate in any additional LOFT tests this fiscal year.

Task 4: Data Analysis.
We are continuing to analyze the changes in Sequoyah noise signatures and the behavior of the noise descriptors used to detect changes in the signatures.

EQUIPMENT PURCHASES:

Bids were received for the much needed Winchester disk drive for Sequoyah. This disk will relieve the storage problems, but has not been purchased because of on-going investigations with the manufacturer concerning possible system compatibility problems.

MEETINGS AND TRIPS:

C. M. Smith attended an Engineering Foundation conference on signature analysis in Rindge, NH, July 19-23 to assess the state-of-the-art of signature analysis in areas other than nuclear power.

REPORTS, PAPERS, AND PUBLICATIONS:

C. M. Smith presented a paper titled "Demonstration of an Automated On-Line Surveillance System in a Commercial Nuclear Plant" at the conference stated above.

PROBLEM AREAS:

None.

PROBLEM AREAS:

We have not obtained approval from TVA for extension of the Sequoyah-1 demonstration into the second fuel cycle. As stated above NRR is requesting TVA to permit the demonstration to continue.

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 documents from utilities outlining control room unusual event procedures. Documents obtained to date were reviewed by the project staff. Methods of analyzing the procedures were discussed. Data on reactor facilities were assembled to provide a basis for the selection of plants to be included in the study.

Task Four: Case Study

Work was initiated on a case study(ies) of incident alert notifications. Preliminary investigation of incidents at Ginna and Vermont Yankee power plants began.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Maintenance Error Model

PROGRAM MANAGER: P. M. Haas

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

TECHNICAL HIGHLIGHTS:

The draft report detailing the methods, procedures and results of the job analysis of the I&C technician was sent to ORNL by Applied Psychological Services, Inc. (APS). The report will be reviewed and prepared for publication as a NUREG/CR.

The initial task list for the electrician position was reviewed by supervisory personnel at one BWR and one PWR facility. These reviews resulted in a final task list that will be utilized in formulating the job questionnaire for the electrician position. Preliminary contacts have been made with 20 nuclear power plants to whom the questionnaire will be distributed. Additional contacts are currently being made.

Development continues on the fatigue, stress, communication, decision making and radiation effects subroutines. In addition, the analytic work defining the effect of actual and required ability on the probability of subtask success was completed. A list of modules, variables, features and outputs was prepared as well as a preliminary block diagram of 8 of the 16 modules. Fortran was chosen as the modeling language, and the simulation model will be developed to run on the IBM system.

MEETINGS AND TRIPS:

A presentation entitled "Human Reliability Model for LWR Maintenance" was given at the Engineering Physics Division Information Meeting on July 29. The presentation summarized the work in the program and discussed the areas of future work.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Noise Diagnostic Methods for Safety Assessments

PROGRAM MANAGER: D. N. Fry

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

TECHNICAL HIGHLIGHTS:

Task 1: Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. A final report of our assessment of ex-core neutron detector sensitivity to fuel element vibrations was initiated. A draft of this report will be completed by September 30, 1982.

Task 2: Surveillance and Diagnostics by Noise Analysis. We have not been able to obtain a consistent correlation between changes in the 10-20 Hz pressure noise at Sequoyah-1 and events taking place in the plant.

Pressure noise data from LOFT were used to understand the relationship between pressurizer bubble size and pressure noise. Tests were performed in an ORNL test facility to determine the effects of sensing line dimensions and restrictions on observed pressure noise.

We are preparing a program plan for assessment of temperature noise as a diagnostic tool. This plan will be presented and discussed at the NRC Research Review Group meeting in September.

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

Task 4: Evaluate New Surveillance and Diagnostic Methods for Reactor System Fault Detection. Due to a request by NRC that we participate in the IAEA Specialists' Meeting on Early Diagnosis of Failures in Primary System Components of Nuclear Power Plants, we will not have sufficient FY 1982 funds to sponsor the University of Washington to evaluate their proposed method for isolating the sources of Sequoyah-1 ex-core neutron noise.

We have tested the expert system methods on laboratory generated data. This method will ultimately be evaluated using Sequoyah-1 data.

MEETINGS AND TRIPS:

J. A. Mullens and D. N. Fry debriefed NRC on June 3rd on Mullens' attendance at the 15th Informal Meeting of Reactor Noise Specialists and his visits to reactor noise research laboratories in the Netherlands, Germany and France.

MEETINGS AND TRIPS (Cont'd.):

Noise specialists from Sweden and France visited ORNL in June and July to discuss areas of mutual interest and possible exchange of noise data.

J. A. Thie attended the IAEA Specialists' Meeting on Early Diagnosis of Failures in Primary System Components of Nuclear Power Plants held June 21-25 in Prague, Czechoslovakia.

REPORTS, PAPERS AND PUBLICATIONS:

1. Report of Foreign Travel of J. A. Thie.

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 first segment of this project, which has been involved with data collection, is nearly complete. Two organizations identified for site visits in the original work plan remain to be scheduled. All rating lists of performance shaping factors have been returned from previous visits and a compilation of this information is in the process of being prepared.

The first review group meeting which was planned for late July has been rescheduled for August 30 due to schedule conflicts on the part of several group members.

Task 2: Evaluation and Upgrading of Simulators. Review of literature on techniques for assessment of simulator training effectiveness has continued. A conceptual framework for selection of malfunctions to be included in training is being suggested from previous work which would rank possible malfunctions according to their frequency of occurrence, criticality and difficulty. In a systems approach, the information necessary to make the ranking would be provided from a comprehensive systems analysis (including job/task analysis). In the interim, the rankings could be made on a subjective basis. The structure should not have to be revised as more objective data is obtained.

As a first step toward providing the necessary data from the Nuclear Safety Information Center (NSIC) files are being examined and categorized to develop estimates of frequency of occurrence of various malfunctions for "similar" plants.

MEETINGS AND TRIPS:

Visits to Pilgrim Station, Arkansas Power and Light Company, Arkansas Nuclear One and Connecticut Yankee were made to obtain information from subject matter experts on current approaches to selection, training evaluation, etc. and on key performance shaping factors to be addressed in selection and training.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Nuclear Plant Management Appraisals

PROGRAM MANAGER: Wm. B. Cottrell

ACTIVITY NUMBER: ORNL #41 88 55 02 (189 #0486-2)/NRC 60 19 31

TECHNICAL HIGHLIGHTS

This program provides technical assistance to the Division of Program Development and Appraisal of I&E and to the Division of Facility Operations of Nuclear Regulatory Research with regard to management performance appraisals of licensed nuclear power facilities. The program includes work on three different tasks as follows:

Task 1: Review of SALP Evaluation Guidance

Task 2: Evaluation of Performance Appraisals

Task 3: Development of Quantitative Appraisal Criteria

Tasks 1 and 2 are completed. Task 3 was drafted during the month and will be transmitted to the NRC early in August following completion of our review.

MEETINGS AND TRIPS

None

REPORTS, PAPERS, AND PUBLICATIONS

Evaluation of Performance Appraisals of Eighteen U.S. Nuclear Power Plants by the Nuclear Regulatory Commission, W. R. Casto, E. W. Hagen, and E. G. Silver, ORNL/NSIC-211 (June 1982).

PROBLEM AREAS

None

PROGRAM TITLE: Occupational Radiological Monitoring at Uranium Mills

PROGRAM MANAGER: C. S. Sims

ACTIVITY NUMBER: ORNL #41 38 55 04 9 (189 #B0485)/NRC #60 19 31

TECHNICAL HIGHLIGHTS:

The first draft of Chapter 1 (Introduction) of the manual entitled "Occupational Radiological Monitoring at Uranium Mills" has been written, typed, and transmitted to the NRC for review.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

PROGRAM TITLE: Operational Aids for Nuclear Power Plant Operators

PROGRAM MANAGER: J. L. Anderson

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

TECHNICAL HIGHLIGHTS:

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

During the recent visit to a BWR project, BioTechnology staff remarked on the difficulty of detecting and characterizing problem solving behaviors. These behaviors appeared during both emergency and routine operations, with much greater than expected frequency. They are not reliably detectable from documentation and may escape notice during actual operations. Nevertheless, observers were able to identify repeated, almost constant problem solving behaviors during operations such as startup.

In the repeated performance of a task it was observed that minor variances in operating sequence, plant conditions, and control timing quickly resulted in non-standard responses by the plant. Result: no task sequence was exactly that described in written procedures and no two sequences of the same task were identical. Operators performed constant diagnostic problem solving, in order to understand differences in display configuration and to relate them to plant status. When prompted, operators were able to verbalize the mental questions which were raised by each change in display configuration, and to partially describe their mental processes of hypothesis formulation and problem solving. Study of this issue continues by H. Van Cott, B. Paramore, and J. Hill of BioTechnology staff.

BioTechnology staff members plan site visits to training or operational facilities as shown by the accompanying table which also recapitulates site visits already performed. Projected are visits to control system design review activities of the Duke Power Company which has three units under design. BioTechnology is participating in that design review.

Task 3: Effects of Changes in Automation on Operator Performance

No technical work on this task performed during July.

Task 4: Complete the Review and Assessment of Operational Aids
Data are still being collected from several organizations.

Table. Site Visit Plan

Date	Site	Location	Type	Vendor*	Activity
6 May	National Bureau of Standards	Darnestown, Md.	Research	–	Control Room Plant Control System
4-7 June	Limerick	Pottstown, Pa.	BWR	GE	Simulator Task Analysis
24 June	Design Branch, Combustion-Engineering	Windsor, Conn.	PWR	C-E	Simulator Displays Design Activity
19-23 Aug.	Palo Verde	Wintersburg, Ariz.	PWR	C-E	Control Rooms Simulator Task Analysis
Aug.	(Possible visit in ORNL area)		–	–	Simulator
Sept.	Surrey	Gravel Neck, Va.	PWR	WH	Simulator Control Room
Sept.	Design Branch, Duke Power Co.	Charlotte, N.C.	PWR	–	Design Activity
Oct.	McGuire	Cowans Ford Dam, N.C.	PWR	WH	Control Room Design Activity Operator Interviews
Oct.	Limerick	Pottstown, Pa.	BWR	GE	Control Room Simulator Task Analysis
Oct.	Rancho Seco	Clay Station, Calif.	PWR	B & W	Control Room Operator Interviews
Nov.	Oconee	Seneca, S.C.	PWR	B & W	Training Activity Operator Interviews
Nov.	Catawba	Clover, S.C.	PWR	WH	Training Activity Operator Interviews
Nov.	Clinton	Clinton, Ill.	BWR	GE	Control Room Task Analysis

*B & W – Babcock & Wilson
GE – General Electric
C-E – Combustion-Engineering
WH – Westinghouse.

Task 5: Man-Machine Interface Study

Preparations for the Workshop on Cognitive Modeling of Nuclear Plant Control Room Operators (August 15-18) are well underway. Eleven papers will be presented. Participants will attempt to supply informed, written responses to the following theme questions:

1. Does the NRC need information about cognitive models for regulation of nuclear facilities and what are the major purposes of these models?

2. What reactor operator cognitive behaviors do we need to predict? Can these behaviors be listed in order of importance?

3. How would such predictions or other outputs from cognitive models be applied by the NRC in the areas of:

   a. plant design
   b. procedures
   c. management
   d. training
   e. licensing or other

4. What are the principal cognitive modeling approaches in use now, and how are they used in other applications or fields?

5. At what stage of development do models become useful?

6. What taxonomies apply? (e.g., normative vs. descriptive, control vs. decision, mathematical vs. verbal, deterministic vs. stochastic.)

7. What data are presently available to use for generating models? What data are missing? Do we have means of collecting needed data?

8. What sources of data can be used to calibrate or validate cognitive models of nuclear power plant operators? (e.g., anecdotal data, LERs, simulators, etc.)

9. What are the current obstacles to:

   a. developing the status of cognitive modeling of nuclear power plant operators
   b. calibrating and validating the models
   c. putting the models to use
   d. learning from experience.

One of the principal approaches, which will be discussed, is that of the Bolt, Beranek, and Newman team. Their report, which is scheduled to be published shortly as a NUREG/CR, is being used as background material for the workshop. It describes the results of a study aimed at determining the feasibility of applying a supervisory control modeling technology to the study of critical operator-machine problems in the operation of a nuclear power plant. The report includes brief overviews of various alternative approaches to the modeling of human performance, and different perspectives on the roles of operators in process control activities like those represented in a power plant.

The result of the study is a conceptual model that incorporates the major elements of the operator and of the plant to be controlled. The operator portion of the model is developed at a block diagram level and includes several algorithms that are considered suitable for use with various model components. The plant portion of the model is developed from literature available on plant dynamics and is of the "first-principles" type. Both models are presented in detail sufficient for demonstrating the feasibility of developing a quantitative supervisory control model, for outlining the requirements for data to build and operate such a model, and for discussing its potential applications.

MEETINGS AND TRIPS:

July 21: R. A. Kisner met with Thomas Sheridan of MIT to discuss details of the upcoming Cognitive Workshop.

REPORTS:

None.

PROBLEM AREAS:

A higher than planned contract cost for Tasks 1 and 2 and additional effort early in the fiscal year to provide a composite report of FY81 results has led to a cost/budget problem. As a result, the effort will be reduced on Tasks 3 and 4 for the remainder of the fiscal year. All major milestones are expected to be accomplished, however. Because of late funding authorization and consequent delay in subcontract arrangements, Tasks 1 and 2 are scheduled for completion in February 1983.

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

Work continued on collecting utility, state, and local plans for a radiological emergency at a nuclear power plant. Sampling criteria for selecting plants to include in the study were developed. An initial screening identified 23 sites for possible analysis. A preliminary framework for abstracting and analyzing data from the plans was developed.

Task Two: Dynamics of Interface

An outline for the literature review on organizational behavior and interaction in emergencies was developed with our consultant, Dennis Mileti. The literature review will cover theoretical research on organizational behavior and interorganizational relations, and applied research on organizations in disaster and in nuclear emergencies. The conclusions will summarize this research, suggest an ideal emergency organizational arrangement and discuss the implications for emergency preparedness regulations.

MEETINGS AND TRIPS:

John Sorensen attended a workshop on hazard management in Boulder, Colorado on July 19 and 20.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

Obtaining utility, state, and local emergency plans is proving to be more time consuming than anticipated.

PROGRAM TITLE: Pressure Sensor/Sensing Line System Evaluation Research

PROGRAM MANAGER: D. N. Fry

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

TECHNICAL HIGHLIGHTS:

Task 1: Review Status of Pressure Sensor/Sensing Line System Standards, Practices and Technology. Visits were made to Rosemont, Foxboro, Barton and Westinghouse pressure sensor factories. We discussed, with each manufacturer, such topics as field experience, quality assurance programs, sensing line limitations, accuracy and range specifications, transient response capabilities and calibration procedures.

We also held meetings with representatives from five different utilities to discuss such topics as type of pressure transmitters used, experience with failures, calibration, quality assurance and installation procedures. Additional visits to three plants are being scheduled for September.

We completed a review of pressure sensing licensing event reports and are preparing a report summarizing the results of this study.

Task 2: Pressure Sensor/Sensing Line Model Development. A water column facility is being used to study the characteristics of a Foxboro sensor diaphragm. This work will be completed during the next reporting period.

Tests were performed in the ORNL pressure s sor test facility to verify a sensing line-sensor diaphragm mode. obtained from the literature. This model will be extended to include the Foxboro sensor model and tested with the laboratory facility during the next reporting period.

Task 3: Evaluate an In-Situ Method for Remote Detection of Pressure Sensor/Sensing Line Degradation. We completed a draft of a test procedure for remote in-situ response time testing of a Foxboro force-balance pressure sensor.

A plan for testing noise analysis methods to remotely detect pressure sensor/sensing line degradation was complete and is currently being reviewed.

We performed preliminary tests to demonstrate that air in a sensing line reduces the bandwidth of pressure noise sensed with that line.

We have decided not to perform response time tests at LOFT in August because the proposed methods for in-situ testing have not been fully developed and the nature of the August tests at LOFT is not consistent with requirements for our tests.

The Foxboro model NE11GM pressure transducer is scheduled to be delivered the week of August 9.

MEETINGS AND TRIPS:

A meeting was held at the University of Tennessee on July 14th to review program status and future directions. G. S. Lewis, NRC project manager, participated in this meeting.

Trips were made to four pressure sensor manufacturers and meetings were held in Knoxville and at ORNL with representatives from five utilities to discuss pressure measurement methods and experiences.

REPORTS, PAPERS AND PUBLICATIONS:

A paper "Dynamic Analysis of a Foxboro Force Balance Pressure Transducer" was submitted to the American Nuclear Society for publication in the society transactions.

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 revised 189 has been submitted. The program plan is being brought up-to-date and a schedule flow chart prepared.

Task B: Because of the lack of detailed information needed to implement function elements in the ICS, an attempt is being made to parameterize these elements from information now available. Estimates were made of two functions in the steam generator-feedwater control loop: (1) the BTU limit, which is determined in the ICS by a weighted sum of reactor outlet temperature, feedwater temperature, and steam generator pressure times normalized reactor coolant flow; and (2) the feedwater demand, which, to reduce effects of changing water temperature on plant state, is altered by the difference between actual and desired feedwater temperatures. The desired temperature was parameterized as a piecewise continuous function of flow rate.

Task C: The heat transfer package was coded and debugged and consists of 16 heat transfer correlations and supporting properties including water viscosity, conductivity, and specific heat. A subprogram for computing friction factors was also coded and debugged. It uses a simplified homogeneous model that was shown to be comparable in accuracy to more detailed models.

The water properties package previously described was added to the thermohydraulics routines for the core and piping.

Task D: Criteria - The criteria position paper is under revision.

MEETINGS AND TRIPS:

On July 23, R. S. Stone, O. L. Smith and A. L. Lotts met with NRC personnel at the Nicholson Lane headquarters to reply to contentions that the large digital blowdown code RELAP-5 would be a more suitable vehicle for this control system study. We expressed our opinion that the hybrid development is the obvious way to do the job and that use of RELAP-5 would be an expensive and inadequate misuse of a code developed for other purposes. The issue was resolved in favor of continuing with the hybrid machine for such power plants as are analyzed by ORNL.

On July 27, R. S. Stone and O. L. Smith joined K. R. Goller, W. M. Morrison, E. C. Wenzinger, D. L. Basdekas, A. J. Szukiewicz, and P. Norian of NRC in a meeting with Baltimore Gas and Electric management and staff at the BG&E headquarters in Baltimore. ORNL and NRC personnel presented details of the Safety Implications of Control Systems Program, emphasizing the benefits to the cooperating utilities and urging BG&E's cooperation. Mr. Goller expressed the view that the study of Calvert Cliffs would probably proceed with or without BG&E's help, and that the study would benefit the utility far more with their active help and especially with their participation. BG&E undertook to discuss the matter internally and to get back to NRC with the results of that discussion.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

Access to detailed design data on each of the plants to be studied depends upon a working relationship with the utility concerned. Such a relationship does not presently exist for any plant, and this lack continues to impact progress in most phases of this project. Discussions continue. (See Meetings and Trips above.)

Raising of a "conflict of interest" question in regard to our subcontractor, the Oak Ridge Office of Science Applications, Inc. (SAI), has prevented extension of our contract with SAI, and is slowing progress on the failure modes portion of the project. An early resolution of this problem is needed.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

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

TECHNICAL HIGHLIGHTS:

Task 1: Operator Response Time. Data collection for "Week 2" of the PWR simulator training program is complete. Data has been collected for fourteen teams of operators on twelve different accident scenarios. Eight of these scenarios are identical to those used in "Week 1" training earlier in the year. Demographic data and subjective data obtained from operators has been reduced for statistical analysis. The analysis has been initiated. Approximately 120 runs have enough data to permit statistical analysis for response time and error data.

The BWR data collection has also been completed, but restrictions on collection of demographic and subjective data have limited the "richness" of the data collected. Sufficient data for statistical analysis have been obtained for five events with four to six runs per event.

Re-writing of the draft of NUREG/CR-2534 on the initial BWR simulator experiments is still in progress. The General Physics Corporation draft of the NUREG/CR report on the simulator-field calibration has been reviewed internally and by an independent reviewer. Comments and requirements for revision will be transmitted to GPC in August, and a final draft is expected to be ready for publication in September.

Task 2: BWR Task Analysis. Reduction and analysis of the task analytic data is complete. A draft report is being prepared by GPC which summarized the task analytic data and presents conclusions on how such data can be used to help structure criteria for safety-related operator actions. The draft report is expected to be transmitted to ORNL by mid-August.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

DIVISION OF HEALTH, SITING AND WASTE MANAGEMENT

PROGRAM TITLE: CONCEPT/OMCOST Code Development

PROGRAM MANAGER: H. I. Bowers

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

TECHNICAL HIGHLIGHTS:

1. Nuclear Power Plant O&M Costs and Estimating Guidelines. The report has been edited, typed on mats, and proofread and is now being corrected. Publication is scheduled for mid-September.

2. Trends in Power Plant Capital Investment Cost Estimates. A draft of a literature review of 31 documents on cost-size scaling was completed and submitted to NRC with our letter of July 16.

This review indicates that the consensus in the electric utility industry is that the cost-size scaling exponent for nuclear units is about 0.4 to 0.6 and for coal units is about 0.7 to 0.9. The scaling relationships being used in the CONCEPT code fall within these ranges.

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

MEETINGS AND TRIPS: None.

REPORTS, PAPERS, AND PUBLICATIONS: None.

PROBLEM AREAS: None.

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:

The review of parameter values is now completed. The computer code to carry out the necessary compilations is being prepared.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

None.

PROBLEM AREAS:

NONE.

PROGRAM TITLE: Evaluation of Atmospheric Dispersion Models

PROGRAM MANAGER: F. C. Kornegay

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

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration: F. C. Kornegay spoke with R. F. Abbey, R. A. Kornasiewicz, I. Van Der Hoven, and L. L. Beraton by conference call on 7/30/82. At that time, I expressed my opinion that the short-term SF_6 concentrations are the most important data of the INEL field program.

Task 3: Meteorological Data: The appropriate sounding data for use in setting mixing layer depths for the remaining INEL test days, except tests 1 and 2, are now at ORNL.

Task 4: Atmospheric Models - A copy of the MESODIF code will be sent from INEL to ORNL.

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	
3. Update Integrated System	9/82	

TECHNICAL HIGHLIGHTS:

Task 1. We completed updating the National Data Base and are now preparing to reestimate SLED.

Task 2. We mailed copies of the report on the Integrated Forecasting System to the persons to whom we have transferred the model.

MEETINGS AND TRIPS:

On July 15, Bob Shelton and Dan Hamblin met with Clark Prichard, Darrel Nash and John Stewart. They discussed the upcoming NRC review of the Model Transfer.

REPORTS, PAPERS AND PRESENTATIONS:

Wen S. Chern, Richard E. Just and Hui S. Chang, A Varying Elasticity Model of Electricity Demand With Given Appliance Saturation, ORNL/NUREG/TM-438, Oak Ridge National Laboratory, June 1982.

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.

MEETINGS AND TRIPS:

None.

REPORTS, PAPERS, AND PUBLICATIONS:

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: 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:

Comments on the site availability document and the population distribution document were received from the sponsor. Additional comments on the "final" draft of the protective actions document were received from the sponsor. Several data bases for the major societal resources work were obtained.

MEETINGS AND TRIPS:

One member of the sponsor's office visited ORNL for two days to discuss the major societal resources work.

REPORTS, PAPERS AND PUBLICATIONS:

None.

PROBLEM AREAS:

None.

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

PROGRAM MANAGER: R. C. Tepel

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

TECHNICAL HIGHLIGHTS:

Task 1: Initial Methodology Development and Salem II Analysis in its Present Ownership and Institutional Structure -- This task is an evaluation of the cost and the distribution of the cost caused by the delay in operation of the Salem II Nuclear Power Plant. Initial results from PRODCOST and RAM, production cost and utility financial models respectively, have been developed and presented to NRC. Comments and suggestions from NRC will be incorporated in the refinement and analysis of model results.

Task 2: Completion of Methodology Development and Case Studies -- This task will utilize the framework established in Task 1 to develop an analysis of costs of delay for utilities with different financial and operational characteristics typical of the ownership and regulatory framework within which the utility operates.

MEETINGS AND TRIPS:

On July 28, 1982, R. C. Tepel, L. J. Hill, and J. W. Van Dyke, accompanied by R. B. Shelton met with NRC in Washington to present initial results and analysis of work on Task I. Attending the meeting for NRC were Clark Prichard, Argil Toalston, and Sid Feld. The initial results from PRODCOST and RAM were discussed. Clarifications were suggested and cases for investigation in Task II of the project were also discussed.

REPORTS, PAPERS, AND PUBLICATIONS:

None

PROBLEM AREAS:

None

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

PROJECT MANAGER: R. O. Chester

PRINCIPAL INVESTIGATOR: D. C. Kocher

ACTIVITY NUMBER: ORNL #41 88 54 32 0 (189 #A9041)/NRC #10 19 03 03 3

TECHNICAL HIGHLIGHTS:

Task 1: Assessments of Uncertainties — The title and author list for the summary report, NUREG/CR-2506, ORNL-5838, has been changed to *Uncertainties in Geologic Disposal of High-Level Wastes — Groundwater Transport of Radionuclides and Radiological Consequences*, by D. C. Kocher, C. S. Bard, and A. L. Sjoreen. These changes more nearly reflect the content of the report and the significant contributors to the effort. Extensive revisions of the document are in progress but have not been completed.

Task 3: Support for 10 CFR Part 60 Rulemaking — Work on this task has mainly involved discussions among ORNL staff concerning uncertainties in the geosciences and the effectiveness of Part 60 in compensating for the significant sources of uncertainty. R. E. Meyer and G. D. O'Kelley of the Chemistry Division have agreed to make a formal statement on a chemist's perspective on uncertainties. More informal input has been obtained from F. W. Dickson of the Chemistry Division, who is former chairman of the Geology Department at Stanford University and currently a member of the BWIP review panel, and E. D. Smith, a geologist/hydrologist in the Environmental Sciences Division with considerable experience in problems of hazardous waste disposal.

Task 4: Geologic Records and Future Site Evolution — The first draft has been completed of a report summarizing current modeling capabilities for predicting geologic processes and their effects on waste isolation. Before completing this task, we intend to review a book we have just received entitled *Predictive Geology*, ed. by G. de Marsily and D. F. Merriam, and incorporate relevant information into our report.

Task 5: Technical Assistance for Developing HLW Radiological Criteria — Comments on our discussions on a system performance standard for high-level waste disposal held at the NRC on June 28 have been received from M. R. Knapp and D. J. Fehringer of NMSS, but no additional progress has been made on this task.

Task 6: Short-Term Technical Assistance — A major effort for this month was an in-depth review of the NRC document *Rationale for Technical Portion of 10 CFR Part 60, Disposal of High-Level Radioactive Waste in Geologic Repositories*. The review was transmitted to M. R. Knapp of NMSS.

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 55 04 0 (189 #B0462)/NRC #60 10 02 20

TECHNICAL HIGHLIGHTS:

This program is reported quarterly per agreement with the NRC technical monitor.

MEETINGS AND TRIPS:

None

REPORTS, PAPERS, AND PUBLICATIONS:

None

PROBLEM AREAS:

None

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