INTERIM REPORT

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UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

 INTERIM REPORT

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ABSTRACT

Highlights of technical progress during September 1980 are presented for eighteen separate program activities which comprise the ORNL research program for the Office of Nuclear Regulatory Research's Division of Reactor Safety Research.

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: B. G. Eads

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

TECHNICAL HIGHLIGHTS:

SCTF related activities have included shipment of the electronics system, repair of the upper plenum structure film probes, and installation of the upper plenum prong probes. The two latter activities were performed by ORNL personnel at the JAERI facility in Tokai, Japan. Schedules for the installation and check out of all sensors and electronics during October and November have been provided by JAERI. ORNL personnel will begin work on October 10 with completion expected around November 17.

All of the SCTF subroutines for the software package have been completed except the film probe routine. This subroutine is presently being rewritten to incorporate results from the steam-water tests on the SCTF prototype module now in progress. During the month of October the subroutines will be tested on simulated data from a reflood transient. The subroutines and the testing program will be delivered to JAERI at the end of October 1980.

The interface details of the CCTF-II in-core impedance probes were agreed to by ORNL, JAERI, and IHI at the end of September so that the design can now be finalized. This is 1 1/2 months later than requested in the 3D Meeting in July 1980. This only leaves approximately 3 1/2 months to fabricate and test these probes and will probably result in delivery later than the requested date of February 10, 1981. It is very important that the interface details for all of the remaining CCTF-II sensors be finalized by November 1 in order to meet the scheduled delivery date of May 31, 1981.

It has been decided to begin fabrication of the CCTF-II in-core probes using the standard "flag" electrode configuration. If the "thumbtack" electrode configuration can be fabricated and the steam-water test results for the "thumbtack" electrode prove to be as good as the air-water results, then approximately one-third of the CCTF-II sensors could use the new design. The "thumbtack" steam-water tests will be conducted in November 1980.

PROGRAM TITLE: Advanced Two-Phase Instrumentation

PROGRAM MANAGER: K. G. Turnage

ACTIVITY NUMBER: 40 89 55 11 5 (189 #B0401)/ NRC 60 19 11 01

TECHNICAL HIGHLIGHTS:

Testing of heated thermocouple (HTC) coolant sensors continued in the Thermal Hydraulic Test Facility (THTF). Two HTCs, both differential devices, were installed in the THTF upper plenum and monitored during the THTF steady-state film boiling experiments (Test 3.07.9). The HTC provided by Combustion Engineering (C-E) has operated successfully during some 12 periods of flow conditions that resulted in film boiling and high temperatures in the THTF rod bundle. Data were taken at 4.1, 6.2, and 8.3 MPa (600, 900, and 1200 psia). Duirng one period of film boiling at 6.2 MPa, the C-E HTC did not indicate an uncovered (dry) condition; the reason for this is unknown. Information on pressure, temperature void fraction and velocity at the test section outlet, and velocity at the test section inlet was recorded. Rod bundle power and fuel rod simulator maximum temperatures were also obtained. The void fraction at which dryout of the HTC usually occurred appears to have been between 70 and 90% based on the densitometer data.

Additional tests were run with a differential HTC and a miniaturized heated RTD in the natural convection test facility. The data from the ORNL-fabricated HTC support the repeatability of sensor output, particularly in the uncovered state. The 1-cm-diam heated RTD, which was installed for vertical testing, responded correctly and apparently had no problem with rewetting due to condensate or deentrainment.

Several tests were performed with the torsional-extensional ultrasonic probe in high-temperature natural convection to steam. The probe's electrical internals were repaired and earlier tests were repeated to confirm repeatability. Two program members held discussions with personnel from Panametrics, Inc., in Waltham, MA, regarding development of improved high-temperature designs for ultrasonic level sensors.

PROGRAM TITLE: Aerosol Release and Transport from LMFBR Fuel

PROGRAM MANAGER: T. S. Kress

ACTIVITY NUMBER: ORNL # 41 89 55 11 1 (189 #B0121)/NRC # 60 19 20 01

TECHNICAL HIGHLIGHTS:

NSPP/CRI-II:

Two underwater tests (FAST 64 and 66) were performed this month. FAST 64 had a water temperature of ~361 K (about 12 K below the water boiling point). As in previous higher temperature water tests, a small amount of fuel aerosol was transported to the cover gas. An attempt was made to collect some of this aerosol onto electron microscope grids for comparison with previous samples collected in the argon test series.

FAST 66 was performed with the pressure transducers electrically coupled to the test vessel (which is at ground potential). This was an attempt to simulate the conditions that will be present in under-sodium tests, and to confirm that the pressure transducers would provide the appropriate information in under-sodium tests. Pressure results were essentially the same as those produced in previous underwater tests - where the transducers were electrically isolated from the vessel. This confirms that good pressure measurements can be made in the under-sodium tests.

Recent results (in the past few months) from attempts to detect bubbles using pulse-echo acoustic models have not been encouraging. Consequently, present plans are to use pressure measurements in the argon cover gas to detect bubble size changes in under-sodium tests. Additional efforts to develop the acoustic technique for bubble detection are being evaluated.

Presently, a letter report to NRC is being prepared which will critically discuss the underwater experiments and assess the confidence level for going on to the under-sodium experiments.

NSPP/CRI-II:

Planning is underway to make the minor conversions of the NSPP experimental system to enable aerosol tests to be conducted in steam environments as would be expected to occur in LWR accident situations.

A preliminary test (Run 210) was conducted early in this period to provide experience on conducting aerosol experiments in highly moist environments. The run was conducted in a manner similar to that of Run 208 (the first wet U_3O_8 run) except that the mass concentration of U_3O_8 was to be lower. Results are not yet available from the analvtical laboratory.

Efforts are continuing to evaluate the completed series for mixed aerosol experiments. To provide insight into the coagglomerated aerosols, the aerosols - having been lightly deposited onto carbon surfaces - were examined by the EDX (Energy Dispersive X-ray) technique. The EDX technique has been applied to individual sodium clusters averaging about 2 microns in diameter, or the same area covered by the collimated SEM electron beam. In all cases, uranium L-X rays and sodium K-X rays were emitted by the particles thereby supporting our previous conclusion that the sodium seems to cluster around an existing U_3O_8 chain.

ANALYTICAL:

During this period attention was devoted to improving our computer techniques for processing experimental data from the NSPP. A data record report is being prepared for all unreported NSPP tests in the mixed series. In FAST analytical program coding, a new bubble liquid interface formula was derived which eliminates some of the drawbacks of integral balance formulas.

PROGRAM TITLE: Continuous On-Line Reactor Surveillance System

PROGRAM MANAGER: D. N. Fry

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

TECHNICAL HIGHLIGHTS:

Installation of System at Sequoyah 1. The isolation amplifiers to be used to eliminate the pulses introduced by the Sequoyah data acquisition system were installed at Sequoyah. The thermocouple selector switch which provides the capability of selecting either the Sequoyah data system or the Reactor Surveillance system was also installed at Sequoyah.

The Reactor Surveillance System was moved to Sequoyah on August 18, 1980. This system was installed in the computer room at Sequoyah. The selected signals were connected to the Reactor Surveillance System via the patch panel installed by TVA. At this time a verification was made to determine if the isolation amplifiers did remove the pulses introduced by the plant data acquisition system. It was determined that the isolation amplifiers did remove the pulses. A systematic checkout was made to determine if the signal wiring was correct and to verify the operation condition of the Surveillance System. During this checkout the hardcopy unit had a hardware failure, this unit was taken back to ORNL where it has been repaired. The hardcopy unit will be returned to Sequoyah at the time of plant start-up.

Meetings. We presented an overview of the current status of this work at the NRC Surveillance and Noise Diagnostics Research Review Group Meeting in Bethesda on September 22, 1980.

We also discussed our program plan for FY1981 at the NRC Surveillance and Noise Diagnostics Program Review Committee meeting in Bethesda on September 23, 1980.

Procurement of Advanced System. The delivery estimate for the PDP 11/34 with the RL-01 disk is still January 1, 1981. This item is the controlling factor in the schedule for the advanced system. The analog equipment is in the design stage. A choice between using Rockland filters or Precision Filters, Inc. is to be made before procurement.

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 11 05

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration — G. D. Whitman visited Battelle Columbus Laboratories on September 8, to review their activities in support of the HSST program.

R. D. Cheverton and G. D. Whitman attended a meeting at NRC offices in Silver Spring, MD, on September 12, to plan a program to evaluate pressure vessel behavior under pressurized thermal shock transients.

M. Vagins, J. Strosnider, and P. Wu visited ORNL on September 24 and 25, to witness the TSE-5A thermal-shock experiment, tour facilities and review the program.

Dr. Tsutomu Kanai and Mr. Yosuo Hirose of Hitachi, Ltd., Japan, visited ORNL on September 2? and were briefed on the HSST program.

Task 3: Irradiation Effects — Irradiation of the first two capsules of the Fourth HSST Irradiation Series is continuing with excellent temperature control. We expect to complete irradiation of the first capsule in October 1980. This capsule contains tensile, Charpy impact, and 1T compact specimens of A-533B1 steel.

Inspection of specimens for the third capsule of this series is in progress. Capsule components and fatigue precracking of 1T compact specimens have been ordered.

We are preparing for hot-cell tensile testing of specimens from the Third 4T CT Irradiation and the Fourth HSST Irradiation Series.

Task 4: Thermal Shock — Thermal-shock experiment TSE-5A was conducted, and a preliminary analysis of the results indicates that the flaw behaved in accordance with the pretest analysis. It appears that there were four initiation-arrest events and that a fifth initiation event was prevented by warm prestressing. It also appears that the flaw behaved in accordance with lower-bound toughness, a trend that was observed in TSE-5 as well.

Testing was completed of 22 1T compact specimens from thermal-shock prolongation TSP-2 tempered at 680°C for 4 h. Twelve of the specimens were tested at −32°C. The J values at −32°C ranged from 25 to 92 kJ/m^2, K_J (K value calculated from J per equation $K^2 = EJ$) ranged from 72 to 138 MPa $\sqrt{m}$, and the K_{Icd} values ranged from 75 to 145 MPa $\sqrt{m}$. The remainder of the specimens were tested at 38°C. The J at 38°C ranged from 233 to 382 kJ/m^2, K_J ranged from 220 to 281 MPa $\sqrt{m}$, and K_{Icd} ranged from 231 to 296 MPa $\sqrt{m}$.

Six of the specimens tested at 38°C and one specimen tested at −32°C were unloaded prior to fracture and analyzed in accordance with the proposed ASTM standard method for determining the elastic-plastic toughness parameter, J_{Ic}. Most of the results were located too close to the 0.15 mm offset line to permit the construction of an R-curve regression line.

We have also machined 12 additional 1T compact specimens from TSP-2. Six of these specimens were sent to Dr. J. Gudas (David Taylor Naval Ship Research and Development Center) for precracking and testing. The remaining six are being fatigue precracked and will be tested at ORNL.

A preliminary proposal was submitted to NRC for writing a computer program that would calculate crack-propagation behavior for long flaws in PWR vessels for any thermal-pressure transients. The program would include influence functions that would eliminate the necessity for a finite-element fracture-mechanics analysis for each transient.

Task 5: Simulated Service Tests — Babcock & Wilcox Company started work on a 150-mm-thick trial weld, which is intended to be a demonstration of the weld procedure to be used on intermediate test vessel V-8A. This procedure, selected from nine preliminary trials, produced the best combination of results in terms of upper-shelf Charpy impact energy level, transition temperature, and tensile properties.

The selected preliminary submerged-arc weld (V8-22) was made with a copper-clad NiMnMo wire of type SFA 5.23EF2 and a 75% Linde 60, 25% Linde 80 flux mixture followed by a 50-h heat treatment at 566 to 593°C. Upper-shelf energies ranged from 50 to 63 J in preliminary tests, which is a satisfactory level. However, a small percentage of cleavage persisted up to 90°C, which may force us to test vessel V-8A somewhat above 100°C. Since such a high temperature test would be a significant inconvenience, B&W has explored changes in the procedure that should shift the transition temperature downward.

They have recommended and we have agreed to two measures. First, the V8-22 procedure being used on the trial weld will have a lower heat input than used in the preliminary trial. A laboratory trial was made to confirm weldability with the lower heat input. This change, according to estimates, will diminish the transition temperature by ∿15 K. Second, an additional preliminary weld has been made with a different wire. The change in composition is estimated to shift the transition temperature downward by ∿20 K. Test data from these trials are expected in October.

Since intermediate vessel V-5 may be rebuilt to perform a nozzle corner weld repair test, the flaw is being removed to properly preserve the fracture surfaces. The dome of the nozzle has been cut off, and the vessel is in a large boring mill for drilling prior to a combination flame-cutting, machining operation to remove the flaw region.

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:

Fort St. Vrain (FSV) Reactor Licensing Support: Per NRC request, additional ORECA code development work and accident simulations were done in support of the LASL calculations of thermal stresses in the FSV core support block region. High thermal stresses occur in the firewater cooldown (FWCD) phase of a postulated 90-minute loss-of-forced-convection accident. The revised ORECA included enhanced detail in the core support structure and calculation and printout of heat flows in the critical regions. The internodal heat conduction algorithms were also improved substantially. Calculations were done using input data (from GA) for an initial power level of 72% (vs 105%) and a cycle 2 (vs equilibrium) core. The results were forwarded to LASL, PSC, and GA. The 105% power equilibrium core case of the LOFC and FWCD accident was also run using the FLODIS code, which uses a considerably more detailed nodal approximation of the core support region.

ORECA Code Review: A most helpful review of the ORECA code by BNL was used as the basis of a number of corrections and improvements to the code.

Long Range Program Planning: A "white paper" review of HTGR safety code needs was written for internal use to aid in discussions of long range plans. An outline of a proposed experimental program and proposed work on severe accident sequences was developed and sent to NRC.

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

PROGRAM MANAGER: Robert W. McClung

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

TECHNICAL HIGHLIGHTS:

We are continuing our task to improve the inspection of steam generator tubing with emphasis on intergranular attack in the tubesheet region.

Work is continuing on the preparation of our equipment for field testing at a reactor site. The tape system has now been improved so that it will allow multiple tubes to be recorded on a single tape. The program will let the operator skip over a number of tubes, or will search the tape for a given row and column number. The tape can be positioned at the last tube and additional recordings made at that point.

Sealable boxes have been constructed to transport any contaminated equipment without contaminating the rest of the truck.

We have been in contact with personnel at the Robert E. Ginna Nuclear Power Plant to intermesh our inspection with their inspection. There are additional improvements that can be made in the calibrations, data acquisition, and data reduction. We will make as many of these changes as we can before our inspection trip. We will be able to make the rest of the changes after the trip, and no data should be lost.

We have inspected additional Inconel 600 tubing for Bob Clark, using a pancake probe. The object of this inspection was to obtain the best possible data for the defects in the tubes rather than test our circumferential coil system. The results are included in Table 1.

Table 1. I-600 Steam Generator Tubing From Battelle

Tube Number	Chart Divisions	Defect Depth From Curve (mils)
B45-9	1.6; 0.7	21.8; 18
B49-4	3; 1.2	31.7; 19.9
B45-2	2.7; 2.1	28.8; 24.8
B61-8	1.6; 1.7; 1.6	21.8; 22.3; 21.8
B63-9	1.6; 0.8	21.8; 18.3
B46-10	1.2; 0.6	19.9; 17.7
B34-4	1.2; 2.8; 1.2; 0.6	19.9; 29.4; 19.9: 17.7
B46-9	2.6; 1.8	28; 22.9
B63-10	2.8 (many)	29.4
B63-2	3.1 (many	31.3

PROGRAM TITLE: Instrument Development Loop

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: ORNL #41 89 55 12 5 (189 #B0427)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Task 1: Operations. The IDL Steam/Water Loop shares the DAS with THTF. Because of operational problems experienced by the THTF, the IDL loop could not be operated on its planned schedule. It now appears that all calibration tests will be completed by the second week in October. The tie plate turbine bearings failed after 100 hr of operation; the bearings were replaced and it is now working satisfactorily.

A tie-plate drag body transducer has been fabricated with the strain gage bridge completed inside the transducer housing. This represents a major step forward since it will minimize lead wire temperature effects. This transducer will be installed in the IDL Steam/Water Loop as soon as the calibration tests are completed.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

ACTIVITY NUMBER: ORNL 41 89 55 12 0 (189 #B0415)/NRC #60 19 10 05

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration: The SSC-1 metallurgical capsule from the ORR-PSF irradiation experiment has been decapsulated and the Charpy V, tensile, J_{1c} 1T, and 1/2T-CT specimens shipped to J. R. Hawthorne at NRL. A letter was sent to all participants requesting that preirradiation properties of the specimens, chemistry of the specimens, and flux data (if available) be sent to Russ at NRL with a copy for my files. Dosimetry specimens were also shipped to HEDL, KFA, and B&W, who provided casks for return shipment as specified in the minutes of the past three LWR-PVS Dosimetry Review Meetings. Although the handling cost of each shipment is relatively small, the overall cost is not. Each participant has been requested to provide a purchase order to F. B. K. Kam to take care of their own cost. This cost must include the cost of nonreturnable containers if a cask was not provided.

Task 2: Benchmark Fields -

A. PCA - Neutron Field Characterization - Transport Calculations and Dosimetry

An evaluation of the PCA Calculational Blind Test Results, ORNL/TM-7565 (to be published), has been prepared for presentation at the NRC Eighth Safety Research Information Meeting at NBS, Oct. 27-31, 1980. Recently received data from $^{6}Li(n,\alpha)$ and gas proton recoil spectroscopy measurements are included.

Two months of PCA experiments have been scheduled and the gamma spectroscopy measurements by Ray Gold and Bruce Kaiser are in progress. E. D. McGarry from NBS has set up the run-to-run fission monitor to normalize all experiments.

B. ORR-PSF

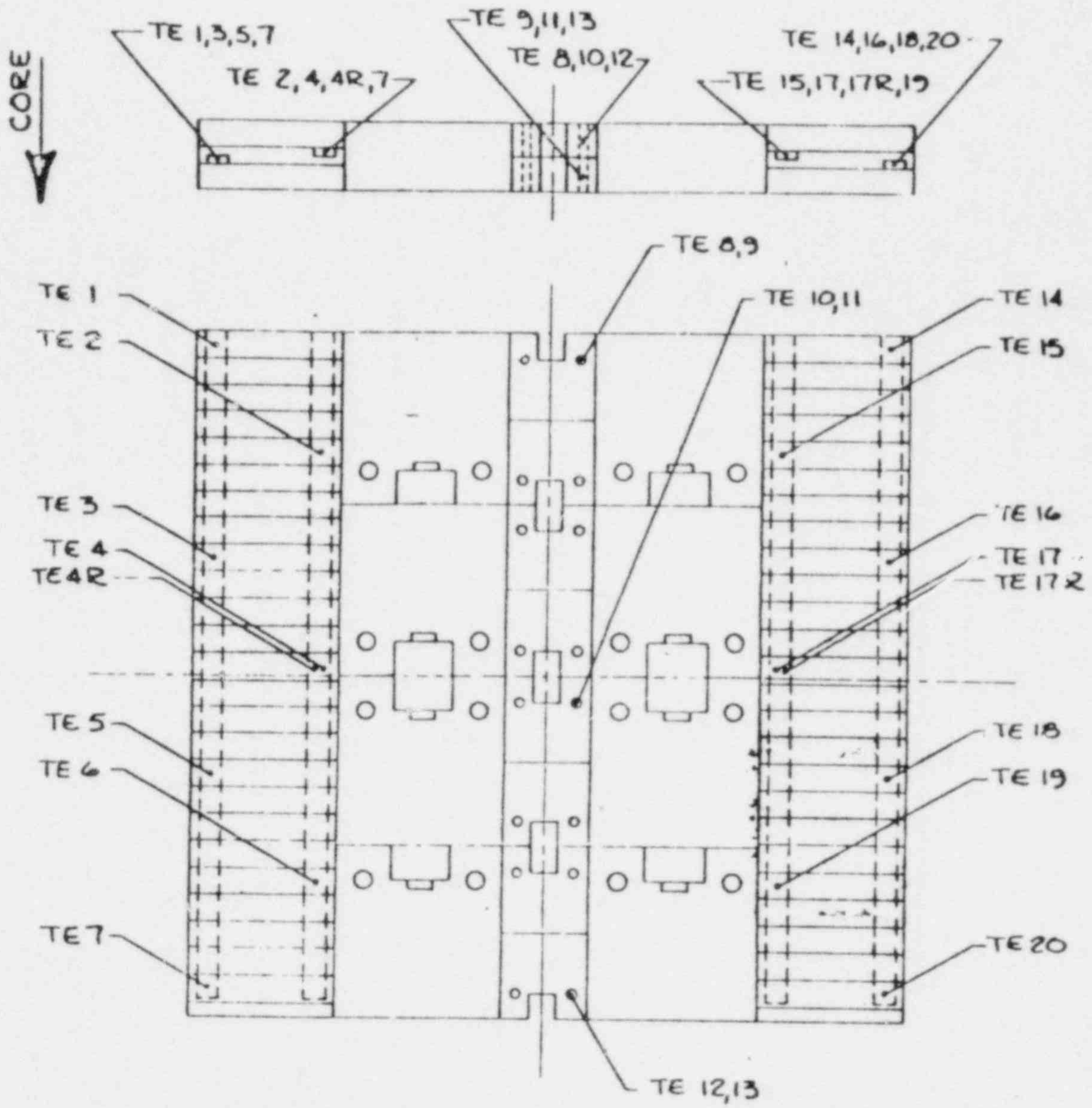
The computer program which evaluates the ORR-PSF irradiation data has been modified to compute a histogram of irradiation time for five temperature intervals rather than a histogram of megawatt hours for three temperature intervals. A time histogram is more useful in the analysis of mechanical property data from the irradiation test specimens than a megawatt-hour histogram. Data previously processed has been processed again with the modified computer program, and typical results for the 1/4T position is presented in Table 1 for the month of May. Any comments concerning the format of the data should be addressed to F. B. K. Kam. Fig. 1 shows the location of the thermocouples.

C. BSR - Surveillance Dosimetry Measurements Benchmark Facility - No activity this month.

Table 1. Irradiation and Temperature Distribution Data from April 30-May 31, 1980

Data for PSF Specimen Set SPVC-1/4T
Hours of Irradiation Time = 664.49
Megawatt Hours of Irradiation = 19860.24

Thermocouple	Hours of Irradiation					Average	Standard
	<T<270	270<T<280	280<T<296	296<T<306	306<T	Temperature	Deviation
TE201	19.52	1.94	640.37	2.66	0.00	291.53	2.05
TE202	19.79	1.50	643.04	0.17	0.00	290.55	[illegible].71
TE203	20.16	2.00	642.34	0.00	0.00	289.31	1.39
TE204	19.28	2.88	641.99	0.33	0.00	289.79	1.04
TE205	19.45	5.62	639.42	0.00	0.00	286.19	1.20
TE206	19.12	4.79	640.59	0.00	0.00	286.54	1.19
TE207	20.65	18.65	625.21	0.00	0.00	282.35	1.42
TE208	19.18	2.11	642.37	0.83	0.00	289.54	2.57
TE209	19.52	1.77	643.21	0.00	0.00	289.89	2.01
TE210	19.62	4.22	640.67	0.00	0.00	286.45	0.96
TE211	19.96	3.54	641.02	0.00	0.00	285.12	0.87
TE212	18.95	2.03	643.50	0.00	0.00	289.50	1.55
TE213	18.95	2.03	643.50	0.00	0.00	288.01	1.50
TE214	19.79	1.50	643.20	0.00	0.00	289.74	1.56
TE215	19.96	2.52	642.00	0.00	0.00	287.04	1.22
TE216	19.79	2.54	642.17	0.00	0.00	287.13	0.99
TE217	19.45	2.54	642.49	0.00	0.00	289.16	1.32
TE218	19.62	4.67	640.23	0.00	0.00	285.75	1.93
TE219	19.29	3.38	641.85	0.00	0.00	285.71	1.79
TE220	20.32	88.74	555.44	0.00	0.00	282.62	2.70



NOTE:

TE NUMBERS FOR 1/4 T LOCATION ARE 200 + TE # (201 THRU 220)

Figure 1.

PROGRAM TITLE: LWR Severe Accident Sequence Analysis (SASA)

PROGRAM MANAGER: M. H. Fontana

ACTIVITY NUMBER:

TECHNICAL HIGHLIGHTS:

Specific objectives for the ORNL SASA Program were defined and, subsequent to a meeting at NRC-RSR on September 3 and 4, 1980, guidance was obtained from NRC-RSR directing ORNL to assume responsibility for the Brown's Ferry Unit 1 BWR SASA assessment. ORNL's charter now includes overview of the entire sequence starting from the initiation of the accident through the end point. A work breakdown structure was established and key staffing completed.

Agreement was reached with TVA for cooperation and assistance. Two meetings were held at TVA and three were held at ORNL, with TVA representation. TVA has agreed to provide us with pertinent information including design parameters, operating parameters, and operating procedures; they also agreed to have ORNL participate in a MARCH/CORRAL "Course" to be presented at TVA by BMI-Columbus.

A preliminary assessment of the phenomenological response of the plant was made for the case of complete loss of offsite and onsite power without RCIC and HPCI operation and a draft report was prepared. The results were shown on a Sequence Progression Time-Line Chart, devised by ORNL to facilitate the display and comprehension of events, often occurring concurrently, as a function of time; to emphasize important events and time available for potential corrective action; and to guide SASA work subtasks. Assessments are continuing for the cases where emergency coolant injection can occur and on the effect of limited time of DC power availability.

Work was started on developing a simple dynamic model that would be used to assist in the development of the time-line charts, the assessment of operator perception of events, and the assessment of potential emergency action. Specific aspects of the simple model would be checked against detailed calculations by codes such as RELAP, TRAC, RETRAN, and MARCH/CORRAL as required. Other laboratories would be requested to perform these calculations, when appropriate. The capabilities of RELAP and TRAC for performing the needed calculations are being assessed. INEL has been requested to perform a RELAP calculation for the case of unmitigated loss of AC power; the results are expected momentarily. The MARCH/CORRAL code was obtained; because it is written for CDC computers, ORNL will operate it on either the CYBERNET system or the remote linkage with the BNL computer system. Because of limited use anticipated, conversion for use on ORNL's IBM system probably is not justified.

The Brown's Ferry emergency operating procedures were obtained. In conjunction with an operator licensing meeting at TVA, discussions were held on the feasibility of manual operation of the HPCI and RCIC systems. It was found that the necessary valves could not be manually operated in the absence of DC power. Estimates of the time period over which DC power can be assured are being made.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY NUMBER: ORNL #41 89 55 10 6 (189 #B0120)/NRC #60 19 11 04 1

TECHNICAL HIGHLIGHTS:

Two single rod heated shroud tests were conducted in the beta temperature region. These tests were performed under constant power (voltage) conditions to eliminate difficulties discussed in the previous report with rapid power changes in tests conducted with constant heating rate control as the tube exits the two phase temperature region.

In the first of these two tests, SR-78, the voltage was adjusted on the basis of meager data to a value expected to produce a nearly constant heating rate of about 10 K/s at the end of the transient. In reality, the selected voltage was too high, and the heating rate was about 18 K/s. Burst conditions were 1065°C, 1135 kPa, and 91% circumferential elongation at the failure. This strain is considerably greater than anticipated in NUREG-0630 for slow ramps ($\leq$ 15 K/s) and is consistent with that expected for fast ramps ($\sim$ 30 K/s).

In the second test, SR-79, the voltage was adjusted to a value expected to produce a heating rate of about 28-30 K/s. In reality the observed rate was 35 K/s. Burst conditions were 1108°C, 1105 kPa, and 103% circumferential elongation at the failure. This strain is consistent with, but greater than, that anticipated in NUREG-0630 for fast ramps.

Preparations for the B-4 (6 X 6) bundle test remain on schedule. All the simulators have been fabricated and are undergoing final acceptance inspection. Fabrication of the necessary test components is underway. Prospects continue to appear favorable for conducting the test in February as scheduled.

The subcontract for hydraulic testing of the B-5 bundle has been signed and will be effective October 1, following a protracted delay for extensive reviews by the legal experts. The bundle will be shipped to the research facilities of the subcontractor in early October for the scheduled tests. We expect the bundle to be returned in April, at which time destructive examination will be initiated to determine the extent of deformation.

At the request of the Division of Safety Systems, R. H. Chapman attended a meeting of the ACRS Subcommittee on Fuel Behavior in Washington, D.C., on September 3. The purpose of the meeting was to discuss the Cladding Swelling and Rupture Models for LOCA Analysis proposed in NUREG-0630. Many of the proposed models are based on MRBT test results.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #B0191)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Fiscal Year 1980 Accomplishments. We performed the following tasks during fiscal year 1980:

1. Completed a BWR stochastic model and extended technique to include in-core vibrations
2. Completed loose parts monitoring experiments in the EGCR and issued a report summarizing the results of these studies
3. Completed stability analysis of Peach Bottom noise data supplied by EPRI
4. Obtained baseline signatures from B&W, CE and W plants
5. Evaluated time-series analysis methods for application to reactor surveillance and diagnostics
6. Evaluated the technical merit for extension of the baseline signature program to include signals other than neutron noise
7. Concluded our participation on the ASME Subcommittee on Pipe Vibration Monitoring by summarizing our method verification calculations at a national ASME meeting

Publications During Fiscal Year 1980

1. F. J. Sweeney, "Modeling the Local Component of In-Core Neutron Detector Noise in a BWR," Trans. Amer. Nucl. Soc. 33, 854 (November 1979).

Publications During Fiscal Year 1980 (continued)

2. F. J. Sweeney, "Sensitivity of Detecting BWR Control Rod Vibrations Using Neutron Noise," Trans. Amer. Nucl. Soc. 33, 793 (November 1979).

3. B. R. Upadhyaya and M. Kitamura, "Monitoring BWR Stability Using Time Series Analysis of Neutron Noise," Trans. Amer. Nucl. Soc. 33, 342, (November 1979).

4. M. Kitamura, "Detection of Sensor Failures in Nuclear Plants Using Analytic Redundancy," Trans. Amer. Nucl. Soc. (June 1980).

5. W. H. Sides, Jr., and K. R. Piety, "Automated Pattern Recognition System for Noise Analysis," Trans. Amer. Nucl. Soc. (June 1980).

6. F. J. Sweeney and J. C. Robinson, "Relative Importance of Attenuation and Reactivity Effects in Explaining Local and Global BWR Neutron Noise," Trans. Amer. Nucl. Soc. (June 1980).

7. J. E. Stoneking and R. C. Kryter, "Screening Procedures for Vibrational Qualification of Nuclear Plant Piping," Proc. Fourth National Congress on Pressure Vessel and Piping Technology, San Francisco, August 1980.

8. F. J. Sweeney, A Theoretical Model of Boiling Water Reactor Neutron Noise, Ph.D. dissertation, The University of Tennessee (1980).

9. B. R. Upadhyaya and M. Kitamura, "Stability Monitoring of Boiling Water Reactors Using Time Series Analysis of Neutron Noise," submitted for publication in Nuclear Science and Engineering.

Publications During Fiscal Year 1980 (continued)

10. R. C. Kryter et al., "Application of Noise Analvsis to Safety-Related Assessment and Reactor Diagnostics," Proc. International Meeting on Thermal Reactor Safety, Conf-800403, Knoxville, April 1980.

11. D. N. Fry, "Application of Noise Analysis to Safety-Related Diagnostics and Assessments," presented at the NRC Seventh Water Reactor Safety Research Information Meeting, National Bureau of Standards, November 1979.

12. F. Shahrokhi, Metalic Loose-Parts Detection and Characterization, Ph.D. dissertation, The University of Tennessee (1980).

13. M. Kitamura and B. R. Upadhyaya, "An Improved Time Series Modeling Approach for Diagnosis and Surveillance of Reactors," Proc. International Meeting on Thermal Reactor Safety, Conf-800403, Knoxville, April 1980.

14. J. C. Robinson and D. N. Fry, "Diagnostics at TMI Using Noise Analysis," Proc. International Meeting on Thermal Reactor Safety, Conf-800403, Knoxville, April 1980.

15. F. J. Sweeney et al., "Stochastic Modeling of BWR Neutron Noise," 13th Informal Meeting on Reactor Noise, Cadarache, France, May (1980).

16. W. H. Sides, Jr., "An Automated Noise Analysis System for Improving Plant Performance," NRC-IEEE Conference on Advanced Electrotechnology Applications to Nuclear Power Plants, Washington, DC, January 1980.

Publications During Fiscal Year 1980 (continued)

17. R. C. Gonzalez, "Application of Pattern Recognition Techniques to Nuclear Reactor Surveillance," Proc. IEEE Conf. Decision and Control, Vol. 2, p 1069-1071, December, 1979.

Meetings. We presented a summary of the results of our research during the last year at the NRC Surveillance and Noise Diagnostics Research Review Group Meeting in Bethesda on September 22, 1980.

We also discussed our program plan for FY1981 at the NRC Surveillance and Noise Diagnostics Program Review Committee Meeting in Bethesda on September 23, 1980.

A paper was prepared and submitted for presentation at the Eighth Water Reactor Research Information Meeting to be held at the National Bureau of Standards on October 27-31, 1980.

PROGRAM TITLE: NRC Reactor Safety Research Data Repository (RSRDR)

PROGRAM MANAGER: Betty F. Maskewitz

ACTIVITY NUMBER: ORNL #41 89 55 11 9 (189 #B0402)/NRC # 60 19 10 01 2

TECHNICAL HIGHLIGHTS:

There is no report for the month of September.

PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: W. B. Cottrell

ACTIVITY NUMBER: ORNL #41 89 55 10 4 (189 #B0126)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

During the month of September, the staff of the Nuclear Safety Information Center (a) processed 788 documents, (b) responded to 64 inquiries (of which 45 involved the technical staff and 13 were for commercial users), and (c) made 17 computer searches. The RECON System, which now has over 200 remote terminals, reports that the NSIC data file was accessed 169 times between August 1 to 29 making it the fourth most utilized of the 26 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 10 visitors and participated in 4 meetings.

Two NSIC reports are in reproduction: *Annotated Bibliography on the Transportation and Handling of Radioactive Materials* (ORNL/NUREG/NSIC-168) and *Bibliography of Microfiched Foreign Reports Distributed under the NRC Reactor Safety Research Foreign Technical Exchange Program, 1979* (ORNL/NUREG/NSIC-177). Several other NSIC reports are in various stages of preparation, including *Nuclear Power and Radiation in Perspective* (ORNL/NUREG/NSIC-161); *Role of Probability in Risk and Safety Analysis* (ORNL/NUREG/NSIC-167); *Annotated Bibliography on Fire and Fire Protection in Nuclear Facilities* (ORNL/NUREG/NSIC-172); *Summary and Bibliography of Safety-Related Events at Boiling Water Nuclear Power Plants as Reported in 1979* (ORNL/NUREG/NSIC-178); *Summary and Bibliography of Safety-Related Events at Pressurized Water Nuclear Power Plants as Reported in 1979* (ORNL/NUREG/NSIC-179); *Nuclear Power Plant Operating Experience — 1979 Annual Report* (ORNL/NUREG/NSIC-180); *Summary Report on Light Water Reactor Systems Survey* (ORNL/NUREG/NSIC-181); and *Accident Sequence Precursors of Potential Severe Core Damage: Status Summary Report* (ORNL/NUREG/NSIC-182).

During the month of September, we received 23 foreign documents (1 Dutch, 6 German, 11 Japanese, and 5 UK). In accordance with the arrangements effective January 1, 1979, a copy of each of these have been sent to Steve Scott (NRC) for microfiche processing. In addition, the foreign language documents were reviewed for translation (see letters of September 30, 1980, to H. H. Scott).

During the month of September, NSIC's Selective Dissemination of Information (SDI) added one paid user which, with other withdrawals and renewals, leaves the SDI service at a total of 386 users.

All technical articles for *Nuclear Safety* 22(1) were completed and mailed to NRC, DOE and TIC on September 19th. The "current events" material (covering events which occurred during July and August) for *Nuclear Safety* 21(6) were completed by September 19th (except for the data on operating power reactors which was not yet available from NRC). Most technical articles for *Nuclear Safety* 22(2) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 21(5) were received from the printer (via TIC) on September 18th.

We have been advised that the NRC support for NSIC in FY-1981 will be 500 K rather than the 620 K previously projected. Inasmuch as even the 620 K represented a reduction in support over that provided the last two years, a budget of 500 K will result in significant curtailment of the NSIC program. This problem was addressed in a letter of September 15th to W. S. Farmer entitled "Changes in NSIC Program Required by Reduced FY-1981 Funding." In that letter we propose to curtail our "cost-recovery" activities, the preparation of reports, and the processing of selected literature, with an expected savings of 150 K.

TABLE 1 RECON DATA BASE ACTIVITY FROM 08-01-80 TO 08-29-80
(21 OPERATING DAYS)

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	CITATIONS PRINTED
EDB	(TIC) DOE ENERGY DATABASE	4342	6380	160637
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	652	959	11332
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	230	575	10062
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	177	230	1795
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	169	313	12773
GAP	(DOE) GENERAL AND PRACTICAL INFO.	150	134	196
EMI	(EMIC) ENV. MUTAGENS INFO.	137	395	3405
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	117	663	4597
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	108	175	464
ESI	(EIC) ENV. SCIENCE INDEX	93	217	390
WRE	(WRSIC) WATER RESOURCE RESEARCH	67	192	909
EIA	(EIC) ENERGY INFO. ABSTRACTS	65	125	334
NRC	(LC) NATIONAL REFERRAL CENTER	50	76	896
IPS	(TIC) ISSUES AND POLICY SUMMARIES	45	34	56
PRD	(TIC/NRC) POWER REACTOR DOCKETS	36	40	114
CIM	(DOE) CENTRAL INVENTORY OF MODELS	31	35	82
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	31	42	221
TUL	(U. TULSA) TULSA DATA BASE	31	76	299
SLR	(FRANKLIN) SOLAR DATA BASE	27	25	110
API	(API) AMER. PETROLEUM DATA BASE	26	64	438
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	25	21	109
NES	(NESC) NATIONAL ENERGY SOFTWARE	24	51	20
RSI	(RSIC) RADIATION SHIELDING INFO.	23	60	552
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	20	11	3
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	12	8	1
RSC	(RSIC) RADIATION SHIELDING CODES	9	4	-

PROGRAM TITLE: Operational Aids for Reactor Operators

PROGRAM MANAGER: G. F. Flanagan

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

TECHNICAL HIGHLIGHTS:

During the month of September, staff members met with several potential subcontractors to discuss work related to operational aids. Several new tasks are being considered; three of which are

1. Caveats for Operational Aid Designs - This task would entail outlining those functions, objectives, and techniques associated with operational aids that are undesirable from a plant safety perspective. Some examples of improper use of an operational aid would be (a) compensating at the operations level for errors and gross deficiencies in plant design; (b) substituting one kind of error for another by failing to account for system interactions or by arbitrarily shifting responsibility from machine to man or vice versa; and (c) masking, instead of repairing, plant design problems.

2. Modeling the Operator as a Supervisor - This task would determine the feasibility of simulating the combined human-process system. The study would specifically determine (1) the usefulness (especially as related to safety) of modeling the operator as a supervisory element, (2) the availability of data to support the simulation effort, (3) the adequacy of existing models of the nuclear power plant processes, and (4) the direction that an R&D program should take. This feasibility study could give rise to a long-term modeling and simulation program.

3. Operator Acceptance of New Equipment and Procedures - This task would identify potential impediments to operator acceptance of improved control rooms and operational aids.

In addition, the problem of over-reliance on operator aids will be addressed. On this task, we anticipate close collaboration with INPO.

Draft reports were received from subcontractors this month: (1) A. A. El-Bassioni, R. A. Hedrick, and R. W. Starostecki, "Review of Standards and Requirements Affecting Human Factors in Nuclear Power Plant Control Rooms," principal investigator: J. R. Penland, Science Applications, Inc., Oak Ridge. This document will become available as an ORNL/TM. (2) T. O. Sargent and R. B. Blum, "Understanding Human Behavior in Off-Average Conditions," Lund Consulting, Inc., Mohegan Lake, New York. This report will also become available as an ORNL/TM. Both documents are in review.

R. A. Kisner attended a course on failure detection and identification offered during Tenenssee Industries Week at The University of Tennessee. The course reviewed stochastic systems, noise analysis, and Kalman Filtering as they apply to surveillance and diagnostic systems. Many advanced operational aids of the future will be developed using stochastic systems theory.

ORNL discussed the possibility of obtaining a human factors engineering perspective from a reputable human factors consulting firm. Their contribution would be sought in preparing the operator aid acceptance criteria as well as review of the role of the operator.

R. A. Kisner and G. F. Flanagan prepared papers and viewgraphs for the WRSR meeting (NRC) and the IEEE 1980 symposium on nuclear power systems.

The September progress report from Lund Consulting follows:

INTRODUCTION

This project will document and develop the bimodal model of cognitive behavior and use bimodal analysis to both develop and apply criteria for analysis and design for hardware, software and management systems in nuclear power plant operations. Specifically, the results of this work will provide information on the effects of stress on performance of reactor operators. The objective of this project is for the development of preliminary analysis and design criteria for hardware, software and management systems in nuclear power plant operations. This project was authorized under Subcontract #7960. The duration of this project is estimated to be from September 1, 1980, to November 1, 1980. The progress report we are submitting is for the work performed within the month of September, 1980.

PROJECT REPORT

The goal of this task is to, through the use of bimodal analysis, produce criteria for analysis and design of hardware, software management systems in nuclear power plant operations. Our plan of action included preliminary reports on the Bimodal Theory and one on Bimodal Analysis. These reports will be supplemented by case histories and a literature search. Lastly, two final reports will be produced: one report on preliminary analysis criteria for man/machine interface in nuclear plant operation for hardware and software application as well as preliminary design criteria for man/machine interface in nuclear plant operation for hardware and software application. As this project began September 1, 1980, there was no prior progress to report. During this time, beginning September 2, 1980, progress on this progress has produced a preliminary report on Bimodal Theory and a preliminary listing of the case histories. Work has been continuing on the development of the literature search as well as Bimodal Analysis. The task on Bimodal Analysis (Task 2) is not due until October 12, but we are hoping to receive comments at our October 3 meeting with Union Carbide staff, which will help us develop the document further. During the month of October, we will complete the work on bimodal analysis, the literature search, any additional case studies, and spend the final part of the month developing the preliminary analysis and design criteria. Finally, in the last week of October we will prepare the final report.

SUMMARY

Progress was made during the month of September with the production of a preliminary report on Bimodal Theory and a listing of case studies. At the present time, work is going according to schedule. However, after our meeting on October 3, we will know if there will be any reason for a delay in our estimated completion time of November 1.

The August-September progress Report from Becker, Block, and Harris follows:

INTRODUCTION

This activity will have as an objective the specification of efficient relationships between operating staff and plant so that situations that may threaten essential plant functions are likely to be identified readily and accurately. This activity will seek a basis for assigning system concepts to plant operation. The initial approach will be to use concepts of hierarchical systems. This project was authorized under purchase order number 40X-40454V. The duration of this project is estimated to be from September 1, 1980, to January 1, 1981. The progress report we are submitting is for the work performed within the month of September, 1980.

PROJECT REPORT

An important task to be addressed early was the selection of a representative pressurized water reactor plant type. After some deliberation, we determined to use the Combustion Engineering plants for reference purposes. This determination was based on two considerations:

(1) In view of the small fraction of existing plants built by Babcock and Wilcox, and in view of the important differences between Babcock and Wilcox plants and those of Westinghouse and Combustion Engineering, we decided that the reference plant be from either Westinghouse or Combustion Engineering.

(2) In view of the published work from Combustion Engineering on operator roles, safety functions and related matters, we decided to use Combustion Engineering systems for reference purposes. (This statement should not be viewed as prejudicial regarding states of the art within competing companies. The statement reflects a judgment relative to available open literature).

In developing hierarchical system descriptions of nuclear plants, we chose a course of action that would be (1) compatible with the Combustion Engineering plant design, (2) capable of being generalized to other plant designs, and (3) applicable to future activities.

We began the construction of an initial hierarchical system structure. This structure then will be compared with the safety functions postulated by Combustion Engineering. This comparison then serves as a basis for modifying, refining, and extending the initial hierarchical system description.

We conducted a brief literature review of hierarchical systems theory. This Bibliography was not intended to be exhaustive.

Simultaneously, we began a review of safety-related regulations that apply to plant operation. This will identify the restrictions that apply to implementation of the hierarchical structure we are considering.

In order to apply a system description to a specific transient, we are giving consideration to operational transient tests conducted at the LOFT facility in Idaho. Particular attention is being given to a loss-of-feedwater transient. Some general information regarding this test has been reviewed, and arrangements have been made to acquire more detailed information. We will review the type of hierarchical system description that would fit the LOFT facility, and determine the similarity that it has with commercial power plants.

The LOFT transient tests are being reviewed in anticipation that a documented review will describe power plant transients more completely and objectively than literature on transients which have occurred at commercial power plants; however, a well documented transient at a commercial power plant would be preferred.

SUMMARY

Progress was made during the month of September with the selection of a PWR, the construction of an initial hierarchical operations structure, the initiation of a literature review (including a review of pertinent regulations), and a review of applicable LOFT transient tests.

As of the end of this reporting period, the pace of progress is on schedule.

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: J. D. White

ACTIVITY NUMBER: ORNL #41 89 55 10 3 (189 #B0125)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Task 1: Single-Rod Loop Testing — This task has been completed. No further work is planned at this time.

Task 2: Analysis — Data Management. Work on the data reports for Tests 3.03.6AR and 3.05.5B has been completed on schedule. Process tapes for these tests have been generated using calibrated engineering units data.

Work on the Data Report for the Bundle Uncovery/Recovery Tests C through H has been completed on schedule. This included plots of the transient (recovery) and steady-state (uncovery) data for each test in the series.

Processing of the Hot Test Section Fill Test, performed on 8/25/80, has been completed. The results were forwarded to I&C personnel for appropriate action.

Electric Pin Analysis. Model development for the THTF core shroud wall inverse computations (needed for the quasi-steady-state, boiloff, and uncovery/recovery tests) has been completed. Program development with full interfacing has been completed. The power drop calibration run on August 29, and Test 3.06.6B have been used to determine the number of forward time steps required for stabilization of the computational and numerical technique.

ORTCAL, ORINC, and the preprocessor have been modified to handle the additional 15 thermocouple levels in the refurbished bundle 3. Preliminary runs by ORTCAL, ORINC, and the preprocessor have been completed for THTF Test 3.06.6B.

Three papers have been written (and accepted) for the International Symposium on Fuel Rod Simulators-Development and Application to be held in Gatlinburg in October.

Thermal-Hydraulics Analysis. Refinements in the COBRA/TRAC model and input boundary conditions are being made for a recalculation of the Upflow

Film Boiling Test 3.03.6AR. In addition, a binary file at BNL consisting of level average surface fluxes and temperatures was created for Test 3.03.6AR. This information will provide experimental data for debugging the version of COBRA/TRAC to be used in local fluids calculations. PNL estimates the completion of this work in late October.

A quick-look report for THTF Test 3.06.6B (Upflow Film Boiling) was issued. A preliminary report on heat transfer from this test is in progress and is approximately 25% complete.

Steady-state film boiling heat transfer experiments were run. Work on this test series is just beginning.

A draft of the final report for the first series of bundle uncovery tests was completed. Pretest planning for the second series of bundle uncovery tests has continued and is close to completion. The pretest planning for the uncovery tests and reflood tests is complete. The planning for the boiloff tests is still tentative as we are awaiting guidelines from NRC as to the depressurization conditions that are desired.

The correlation comparison program has been completed. It includes 51 correlations and combinations of correlations which are used to define the boiling curves used in RELAP4/MOD5, RELAP4/MOD7, TRAC PD2, and COBRA/TRAC.

Nuclear Pin Simulation Analysis. The 60-day interim report describing THTF Test 3.05.5B, which fell behind schedule because of difficulties encountered in obtaining calculated test section fluid conditions, was completed and distributed 30 days late. The report presented preliminary transient local fluid conditions and FPS surface conditions, the results of a best-estimate posttest RELAP4 simulation of Test 3.05.5B, and comparisons of bundle fluid conditions and FPS responses from the test with core fluid conditions and nuclear fuel pin responses predicted by RELAP4 to have occurred in a PWR during a LOCA.

Task 3: THTF Operations — The bundle was operated on three occasions during September, each of which were attempts to complete the Quasi-Steady-State/Inverted Annular Film Boiling Test Series 3.07.9 and an Upflow Film Boiling Blowdown Test 3.06.6C. During each attempt, mechanical problems developed including:

1. Rupture of the blowdown rupture disk due to high test section pressure during the inverted annular film boiling data test.
2. Severe leakage at the seals of the primary circulating pump.
3. Broken or loose fittings (causing leaks) on primary water lines due to system pressure fluctuations and loop dynamic instabilities.

Although some valuable experimental data was obtained, the entire test series has not been completed. Plans are presently being made to complete this test series and to make loop modifications necessary to successfully complete the upcoming Bundle Uncovery Test Series 3.06.10I-N.

Task 4: Two-Phase Instrument Development — The in-bundle gamma densitometer system has been tested in the Upflow Film Boiling Test (3.06.6B) and Quasi-Steady-State Test (3.07.9). Initial results look promising. Voltage differences of 1.0-1.5 V between empty bundle and full bundle have been recorded with signal noise at approximately 5% of this voltage difference. From initial results, power on the bundle and power changes do not appear to affect the instrument signal. The effect of temperature on calibration is being investigated. Checkout of the motor drive system has been completed and the system operated during testing. Use of the system in a scanning mode is being investigated.

Design, procurement, and fabrication of an outlet orifice manifold for steam flow measurements has been completed and it will be installed for the Bundle Uncovery Tests.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #40 10 01 06 01 (189 #B0421-8) NRC #60 19 11 01 2

TECHNICAL HIGHLIGHTS:

Preliminary results from the first series of simulator exercises were transmitted by General Physics Corporation (GPC). Table 1 summarizes data on the time required for operators to initiate the first manual action in each of the seven events used in these PWR exercises. Discussion and conclusions will await the complete report from GPC, which is in preparation at this time. Some quantitative field data have also been obtained. Table 2 illustrates the kind of data it has been possible to retrieve.

A paper summarizing the safety related operator actions program and initial results obtained was written and transmitted for presentation at the Eighth Water Reactor Safety Information Meeting in October.

A meeting was held at GPC offices to discuss with personnel from the Risk & Operations Branch, Division of Systems Reliability Research of NRR and Sandia Laboratories the possible cooperation between the two programs. Sandia Laboratories is being funded by the Risk & Operations Branch to collect and assess data on human error rates to support and/or supplement the "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Application," which was prepared by A. D. Swain and H. E. Guttmann of Sandia. It was concluded that the data collected in this program can be of direct benefit to the Sandia effort, and that Sandia has the capability to support the safety-related operator actions program in a number of ways. It was suggested that the Sandia staff examine in detail the accident scenarios that have been carried out in the ORNL program to identify specific operator actions which would be applicable to the handbook format as well as specific possibilities for cooperative studies. Detailed information has been transmitted to Sandia.

Table 1.

Initial Simulator Data - PWR

Time (Seconds) to Initiate First Manual Action

Shift	Rod Drop	SGTR	RTD Failure	MFP Trip Flux Tilt	NI Failure	Small LOCA	Inadv. S.I.
	Event						
A1	32	100	10	875	9	802	505
A2	202	-51*	7	114	14	194	208
A3	48	84	7	556	8	835	317
A4	41	102	14	299	9	852	149
A5	1170	34	6	35	7	588	---
A6	97	53	10	54	14	464	197
A7	223	---	19	505	13	---	---
A8	71	170	33	137	23	2038	1023
C1	49	30	10	456	6	547	186
C2	57	57	9	408	5	---	274
MEAN	199	67.8	12.5	344	10.8	790	357
(Std. Dev.)	(347)	(51.5)	(8.2)	(268)	(5.4)	(551)	(291)

* The times listed are referenced to the time of the specific event alarm (Blowdown Stream Radioactivity). In this case, the operators performed the initial action (start charging pump) prior to the alarm.

Table 2. Typical Field Data: Dropped Rod

Shift	Event Alarm	Clock Time: Action 1 T_i	Action 1 T_c	Action 2 T_i	Action 2 T_c	Action 3 T_i	Action 3 T_c	Action 4 T_i	Action 4 T_c
A1	1933	1933	1947	--	--	1947	2002	--	--
A3	0420	0420	0446	--	--	0448	0509	--	--
A4	1319	1320	1349	--	--	1349	1410	--	--
A5	1346	1346	1417	--	--	1435	1458	--	--
A6	0957	0959	1020	--	--	1036	1107	--	--
A11	1254	1315	--	--	--	--	--	--	--
A13	0958	0958	1057	--	--	1100	1111	--	--
A15	1226	1226	1256	--	--	1315	1328	--	--
A19	1930	1930	2005	--	--	2008	2032	--	--
A20	0234	0234	0246	--	--	0247	0305	--	--
A22	0349	0349	--	--	--	0455	0507	--	--
A23	1725	1725	1739	--	--	--	1739	--	--
A24	1735	1737	1813	--	--	1820	1833	--	--
A25	2237	2237	2248	--	--	--	2314	--	--
A26	0821	0821	0850	--	--	0913	0927	--	--
A27	0825	0827	0855	--	--	--	0910	--	--
A28	0120	0120	0143	--	--	0150	0223	--	--
A29	1735	1737	1813	--	--	1820	1833	--	--
A31	1245	1245	1330	--	--	1418	1430	--	--
C1	1052	1052	1105	--	--	--	--	--	--

T_i = Time action initiated.

T_c = Time action completed.

Action 1 = Reduce turbine power.
Action 2 = Shift to manual group rod control.
Action 3 = Pull rod to bank position.
Action 4 = Shift to automatic rod control.

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL # 41 89 55 11 7 (189 # B0124/NRC # 60 19 11 04 1

TECHNICAL HIGHLIGHTS:

The draft of the final report of the Zircaloy Fuel Cladding Creepdown Program is almost completed. An hour-by-hour chronology of the test was received from ECN-Petten this last month and was necessary to complete the analysis of HOBBIE-8.

The final report will contain the creepdown data, test descriptions, and analyses of HOBBIE tests 2 through 8 and will compare the data and analyses with creep testing results from other workers.

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