INTERIM REPORT

Accession No.____________

Contract Program or Project Title: Division of Reactor Safety Research-
ORNL Programs

Subject of this Document: October 1979 Monthly Highlights

Type of Document: Monthly Highlight Report

Author: Compiled by Fred R. Mynatt

Date of Document: November 13, 1979

Date Published - November 1979

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

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for the
DEPARTMENT OF ENERGY

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ABSTRACT

Highlights of technical progress during October 1979 are presented for fifteen separate program activities which comprise the ORNL research program for the Office of Nuclear Regulatory Research's Division of Reactor Safety Research.

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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 10 01

The current series of String Probe tests has been concluded and test facilities are being prepared for testing of the SCTF in-core flag probes. With the exception of the small PKL string probe, all of the designs tested gave satisfactory results in measuring void fraction and noise velocity. The small PKL probe measures void fraction satisfactorily but will not measure velocity. A Mod III (with insulators in the side wall) version of the small probe is being fabricated in an attempt to improve the velocity response.

A dual flash photography system has been tested for use in the air/water loop. A 35 mm camera has been modified to operate with two flash units which trigger approximately 10 milliseconds apart. By using different color filters on the flash units, a single exposure will provide two views of the flow pattern displaced in time by 10 milliseconds. This system is patterned after the German system which was used in the 3x3 bundle tests at PKL.

Testing of the second PKL prototype film probe module in steam/water has been started. A series of tests on the film thickness sensor under a wide range of conditions has been completed. The purpose of these tests was to determine the effects of temperature, liquid conductivity, and film velocity upon the response of the ratio circuit. These tests indicate that the Lehigh ratio circuit provides a good measurement of film thickness and is not affected by temperature and velocity over the ranges tested (160°C and 200 cm/sec). The calibration is sensitive to conductivity variations if the conductivity is less than about 1 μseimen/cm. This will occur only at low temperatures (< 50°C) and with very pure water and is not expected to present a problem in any of the Reflood Facilities.

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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

FAST/CRI-III:

Four underwater tests were performed this month (FAST 31 through 34). Two were at a 2.02 MPA (20 atm) argon cover gas pressure. In these two tests, the period between the first two pressure pulses was ∿4 ms, and bubble pulsations continued for at least 80 ms after capacitor discharge. Results of previous tests at .101 MPA (1 atm) cover gas pressure showed the period between the first two pulses to be ∿50 ms. FAST 32 was performed at an intermediate cover gas pressure, ∿.303 MPA (3 atm). As expected, the measured 20 ms period between the first two pulses was between the periods produced in the .101 MPA and 2.02 MPA tests.

NSPP:

Partial data have been received from the chemical analysis of samples from the second mixed oxide aerosol experiment (No. 304). In this experiment, the target mass ratio of uranium oxide to sodium oxide was 10:1. Results in hand indicate that the measured ratio ranged from 8:1 to 9:1 during the first hour of the experiment. The later behavior of the mixed aerosol was very similar to that observed during the first mixed oxide aerosol experiment where the mass ratio of uranium oxide to sodium oxide was in the range 1.2:1 to 2:1. Analysis of this second experiment is continuing.

The third mixed oxide aerosol experiment (No. 305) was conducted on October 29; in this experiment, the target mass ratio of uranium oxide to sodium oxide was 1:10. Data from this experiment should be available by mid-November.

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CRI-II:

The first of a continuing series of mixed oxide (U_3O_8-Na_2O_2) aerosol characterization tests was completed in the CRI-II facility. The aerosol oxides were generated simultaneously by burning the finely divided metals giving an initial concentration of about 20 g/m^3 total aerosol with uranium about a factor of three greater than the sodium. The impactor and spiral centrifuge size distributions were found to nearly coincide, using the redesigned first and second impactor stages. Electrostatically collected TEM and SEM grid samples gave the first series of satisfactory photomicrographs of the uranium oxide chains connected to the large sodium oxide spheres. These were taken with the organic ploymer coating (methyl cyano methacrylate), although the exact effect of the coating is not certain. Additional data are being derived concerning the composition, relative deposition rates and range of sodium-to-uranium concentration ratios over the deposition sequence.

PROGRAM TITLE: Design Criteria for Piping and Nozzles

PROGRAM MANAGER: S. E. Moore

ACTIVITY NUMBER: ORNL #41 89 55 10 2 (189 #B0123)/#60 19 10 05

TECHNICAL HIGHLIGHTS

Task 1: Moment Loading Parameter Studies for Isolated RPV Nozzles — This task has been completed and the final report has been distributed.

Task 2: Stress Indices and Flexibility Factors for RPV Nozzles and Small Branch Connections — This task has been completed.

Task 3: Documentation and Release of the MULT-NOZZLE Computer Program — This task has been completed.

Task 4: Parameter Studies and Proposed Rules for Closely-Spaced Nozzles — The final report for this task is being written.

Task 5: Research Information Letter on Isolated and Closely-Spaced Nozzles — This task has been completed.

PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: ORNL #41 89 55 01 1 (189 #B0119)/NRC #60 19 10 5

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — Mr. I. Milne of the CEGB visited ORNL on October 1, to review primary system integrity research and development.

On October 2, Dr. Carl Feldbaum, Technical Assistant to Secretary Duncan of the DOE, was shown the HSST exhibits and briefed on the program.

On October 3, J. Repa and J. Allen from LASL were given a briefing on the HSST program vessel tests consistent with their interests in specifying the requirements for vessels to contain explosions.

The program status was reviewed with Ed Zebroski of EPRI on October 18.

G. D. Whitman visited NRC offices in Silver Spring, MD, on October 24, to review fiscal year 1980 activities.

R. G. Berggren and G. D. Whitman participated in a review of a proposal, "In-Service Testing of Nuclear Components," submitted by Science Applications, Inc. of McLean, VA, which will be sponsored by NRC and require coordination with the HSST program.

Task 2: Fracture Mechanics and Analysis — The finite-element computer program NOZ-FLAW was delivered to ORNL on October 24, 1979. Work is currently under way to make the present CDC version of the program operational on UCC-ND IBM equipment. Professor Atluri will submit a draft report of a user manual for the program in early December.

One of the currently emphasized requirements of an acceptable method for elastic-plastic fracture analysis is that it should have a definite analytical basis. With this requirement in mind, the incremental form of Neuber's equation, $K_\sigma K_\varepsilon = K_t^2$, that was used as the starting point for developing the Tangent Modulus Method of inelastic fracture analysis has been derived from a J Integral analysis. In addition, by using Ilyushin's principle of stress and strain scaling for power law stress-strain curves, the usual deformation theory version of Neuber's equation has been derived, thus clarifying the applicability of this well known equation.

Task 4: Irradiation Effects — HEDL completed the first phase of testing on the 0.5T compact specimens from the second 4T series, and the data was transmitted to NRC.

The first capsule of the fourth HSST irradiation experiment (containing specimens of ASTM A533, grade B, class 1 plate) was installed at the BSR for operation. However, when the reactor was brought to power the specimen temperatures rose excessively. The reactor power was reduced from 2 MW to 1 MW and specimen temperatures leveled out at about 270°C. We discontinued operation and conducted a series of tests at low reactor power and also with electrical heat only. These test runs indicate the gamma heating is excessive, and it will be necessary to install a thermal shield of 5 to 6 cm of steel in front of the capsule to achieve proper temperature control. This shield will reduce the fast neutron flux about 40%, thus extending the irradiation time to reach a desired fluence. The design of the shield is in progress.

Machining of specimens for the second capsule is nearing completion. However, the capsule design must now be revised in view of the temperature problem encountered with the first capsule.

Task 6: Thermal Shock — The feasibility of conducting a thermal shock experiment that will accomplish what was intended for TSE-5 is being examined. Fracture mechanics calculations were made using a toughness-vs-temperature curve that exhibits the desired transition temperature but lower-shelf toughness that is about twice that for TSC-1 (A508 tempered at 613°C). [Based on a recent review of A508 data it appears that such a curve would be characteristic of A508 class 2 material (tempered at ∿700°C).] The results of the analysis indicate that the maximum K ratios would be too small. Increasing RT_{NDT} increased the K ratios but drove the point of incipient warm prestressing (IWPS) deeper which resulted in no net gain in K ratio for IWPS. Thus, it is concluded that a toughness curve similar to that used for the design of TSE-5 must be used, or the severity of the thermal shock must be increased. The TSE-5 design toughness curve was based on HSST Plate 02 data.

The severity of the thermal shock can, in principle, be increased by more nearly optimizing the thickness of the rubber cement coating on the inner surface of the test cylinder and/or by applying heat to the outer surface of of the cylinder before and/or during the quench in LN_2 so as to increase the positive temperature gradient in the cylinder wall without exceeding the maximum permissible temperature of the rubber cement (∿130°C). Calculations and small-scale heat transfer experiments are being conducted at this time to determine the actual benefit to be derived from heating the outer surface.

In preparation for a VIRG meeting that was scheduled for the latter part of October, additional fracture mechanics calculations were made for a steam-line-break problem that had originally been analyzed and submitted to NRC in March 1978.

During TSE-5, Ailtech SG 125 strain gages were used for detecting initiation and arrest events by sensing sudden changes in COD. These gages also indicated absolute values of COD plus a bias associated with temperature effects and differential expansion. After making corrections for the bias, the measured and calculated COD's did not agree very well. One explanation for the discrepancy is an inaccurate effective gage factor. Thus, additional gage calibrations are under way.

Characterization of A508 material in connection with TSE-5 and a possible future thermal shock experiment is continuing.

Twelve 1T compact specimens machined from a radial segment of prolongation TSP-2, which was tempered at 650°C for 4 h, have been received, and seven have been tested between −18 and 93°C. Four test temperatures (−18 10, 38 and 93°C) and two depth locations (0.25 and 0.59t where 0t is the inner surface) were used. The fracture toughness values ranged from 85 to 246 MPa $\sqrt{m}$ at −18 and 93°C, respectively. The remainder of the specimens will be tested shortly.

We have also received the Charpy and tensile specimens from vessel TSC-1 and prolongation TSP-1 (both were tempered at 613°C for 4 h) and have begun testing the tensile specimen.

The 2T, 1T, and C_vT compact specimens and the drop-weight specimens from prolongation TSP-1 are due, and the precracking of the compact specimens will start immediately after receipt of the specimens.

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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 20 02

TECHNICAL HIGHLIGHTS

Development of the ORTAP code for FSV: The implementation of the new steam turbine model subroutine (ORTURB) in the ORTAP code has been completed. Sample cases representing rod withdrawal transients have been executed and compare favorably with earlier ORTAP runs. Additional cases representing reduction in electrical demand to 25% power have been executed and the resultant steam turbine calculations agree very well with published heat balances.

The segments of the ORTAP coding that were consuming large amounts of computer time have been rewritten. ORTAP now uses far less computer time than the earlier version and the cost of running ORTAP has decreased by a factor of ten.

Attempts to match actual FSV operating data with calculated ORTAP results have been made. Allowances for core bypass helium flow and regenerative heating in the steam generator need to be implemented.

FSV Upper Plenum Reverse-flow plume experiments: The intermediate plume experiment has been modified to give greater accuracy and repeatability in the determination of heat transfer coefficients (Nusselt numbers Nu). The on-line computer program was modified to take more samples of the calorimeter temperature to determine its rate of rise, and insulation was added to the top of the platform (simulated cover plates) to reduce the effects of ambient temperature on heat losses. The durations of the runs are controlled by the statistics of the results; i.e. data are collected until the results are accurate to a prescribed Nu tolerance for a given confidence level.

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 10 05

TECHNICAL HIGHLIGHTS

We are continuing our task to improve the detection of defects in the presence of other property variations.

We have written driver programs to position the tube and the tube supports with respect to the probe and each other. The driver programs read the desired location of the tube and the tube supports, calculate the number of steps the new position is from the old position, and issue the required number of pulses to the proper motor controller. We have received the two stepping motor-driven positioners and one of the motor controllers, and are now testing the system with the single controller.

We have received the digital-to-analog converters to output the properties to the strip-chart recorder. We have tested the converter board and found that we need to add buffers on the computer output to display all five properties. However, we can drive three digital-to-analog converters without buffers.

Additional contacts have been established with Battelle Northwest Laboratory personnel for obtaining tubing samples that contain stress-corrosion cracking.

Arrangements have been made with a reactor operator to visit and observe the environment and performance of commercial eddy-current examination of steam generator tubing.

Preparations were made for a presentation of program progress at the Seventh Water Reactor Safety Research Information Meeting to be held in Gaithersburg, Maryland, November 5-9, 1979.

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PROGRAM TITLE: Instrument Development Loop

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: ORNL #41 89 55 12 5 (189 #B0427)/NRC #60 19 10 01

TECHNICAL HIGHLIGHTS

Task 1: Operations — Scoping tests of AIRS string probes were conducted in the IDL three-module air-water loop. The latest version of the large PKL-II string sensor was located $\sim$7 in. above the upper core support plate; the smaller PKL-II string sensor was located in an end box $\sim$1/2 in. above the tie plate. The void fractions from the large string probe compared very well to those from the low energy gamma densitometer located in the upper plenum. The velocities from both string sensors will be compared to velocities from turbo probes located in similar locations.

Task 2: Construction — Fabrication of the steam-water loop pressure vessel is 99% complete. Facility construction by the CPAF contractor is now 33% complete.

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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 05

TECHNICAL HIGHLIGHTS

Task 1: Program Administration - Thirty-two Charpy-V specimens for the void box capsule have not been received as of October 31, 1979 from Fracture Control Corporation.

F. B. K. Kam, F. W. Stallmann, and R. E. Maerker attended the IAEA Specialists' meeting at Jülich on Accuracies in Correlation between Property Change and Exposure Data from Reactor Vessel Steel Irradiation, September 24-27, 1979. Three invited papers were presented at the meeting. During October 1-5, 1979, they attended the 3rd ASTM-EURATOM International Symposium on Reactor Dosimetry at Ispra, Italy. Six papers were presented at the Symposium.

The perturbation experiment scheduled for October at the PCA by HEDL and Westinghouse was postponed until June 1980. The experiment is an investigation of the perturbation effects in the vessel resulting from the replacement of water by steel. An actual power reactor surveillance capsule is to be used in the perturbation study.

Task 2: Benchmark Fields -

A. Dosimetry Measurements

1) PCA-PVF Dosimetry Measurements

a) Fission chamber and radiometric measurements were performed on the 4/12, 8/7, 4/9, 8/12 and 12/13 configurations. The 4/12 configuration was measured since it is a mockup of the configuration to be used for the ORR metallurgical irradiation. Results for the 8/7 and 12/13 configurations were obtained since they are to be evaluated by the NRC computational "Blind Test".

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2) ORR-PSF Dosimetry Measurements

a) The PSF low power and mid-power ($\sim$15 MW) startup characterization experiments were completed in 15 runs during October 23-29, 1979. Radiometric sensors and damage detectors were employed for the measurements. A 16 day run at 30 MW is in progress and will complete the startup experiments.

B. Mechanical Design and Installation Activity

1) PSF Installation and Checkout

a) An out-of-pool checkout of the assembled PSF indicated satisfactory operations. It was then installed and tested in the ORR pool. The support facility and dosimetry capsule assembly functioned properly. In-pool installation of the water and gas lines was also completed. Final revisions of the drawings to reflect as-built conditions is in progress.

2) SSC and IIC design and fabrication

a) Fabrication of parts for the SSC and IIC is complete except for some final machining that has to be done at a later stage of the assembly.

First assembly of the specimens in the frames was completed. The assemblies will need to be ground to within required flatness. Cooler parts, as well as the back filler pieces, are ready for assembly. Preparation of thermocouples for use in the capsules was completed. The heaters are checked and are being prepared for use in the heater plates.

A partial weldment of the IIC box was used for a trial fit into the PVS before the PVS was installed in the pool. The final dimensions that the box has to be machined to were determined.

C. SSC and IIC Heat Transfer Analysis

a) An independent review of the parameters affecting temperature controls in the capsules was initiated. This includes a review of the heat transfer calculations, heater and cooler capacities.

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A 4/12 configuration gamma heating calculation was run on XSDRNPM. No major difference is seen in the IIC.

A 3-D XYZ heat transfer model of the SSC simulator heating test was successfully run with the updated heat generation rates in the iron. Temperatures in the specimen simulator are 2.5% higher than in the experiment. This model will also be used to simulate the heating test with neon as filler gas.

D. Process Control System

a) The development of the control algorithm is complete. This includes the experiments for determining the heat transfer characteristics of the capsule as well as the simulation studies which are necessary to evaluate the controller performance. In progress is the documentation for this effort.

Software development is more than 90% complete. Since the last reporting period, the data acquisition routines have been completed. Currently, an operations manual describing this software is being written.

Task 3: Neutron Field Characterization - Effort related to neutron flux characterization during September and October was directed toward analyzing transport calculations and completing Monte Carlo calculations.

Results from the BSR-HSST transport calculations have been plotted and documented; however, it will be necessary to perform these calculations again since modifications must be made to the experiment facility. Monte Carlo results are available for the PCA-PV and the BSR-HSST facilities but have not been analyzed. Comparisons of transport calculations with measurements in the PCA-PVF have been completed for spectral indices and for absolute reaction rates.

Task 4: Dosimetry and Damage Correlation - Several papers have been completed and presented. Two papers were presented at the IAEA Specialists' Meeting in Jülich, September 24-27, 1979 and five were presented at the 3rd ASTM-EURATOM Symposium in Ispra, Italy, October 1-5, 1979.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

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

TECHNICAL HIGHLIGHTS

R. H. Chapman visited the NRC Office of Nuclear Reactor Regulation in Bethesda, MD, on October 30–31 and November 1 to discuss MRBT experimental results and interpretation of the data in terms of expected cladding performance during postulated accidents.

Five single rod tests, all with heated shrouds, were conducted during this reporting period. This brings the total to 17 tests (14 with and 3 without shroud heating) conducted in this series of scoping tests to investigate the importance of this parameter. Two or three additional tests are planned. The results of this test series, along with the results of an earlier series with the shroud unheated, will be evaluated and used as a basis for planning a test series to elucidate the effect of the significant test parameter(s) on deformation behavior in the temperature range of interest.

Although the results of this latest series of tests have not been evaluated in detail, preliminary results indicate significantly greater deformation can be expected, particularly at the lower heating rates used in the comparison. Also, the importance of thermal-hydraulic (i.e., steam flow) conditions is becoming apparent. It appears that the presence of the heated shroud drastically affects the fuel pin simulator heat losses (and hence the rod input power requirements) and resultant temperature uniformity, giving rise to the observed large deformation. In this latest test series the cladding temperature rate of increase was carefully controlled, causing the input power to vary as required by the thermal hydraulic conditions. In all of our previous tests, the input power was held constant, causing the cladding temperature rate of increase to vary, particularly near the end of the test when the heat losses became significant. These effects may necessitate redesign of the test equipment in order to perform tests that better simulate reactor accident conditions.

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Fabrication of the B-5 (8 X 8) bundle fuel pin simulators continues to show good progress; approximately 65% of the required number have been completed.

All the instrumentation required for expanding the test facility to accommodate the 8 X 8 bundle test is on order and most of the orders have been delivered. The order for the pressure transducer power supplies has been delayed, pending evaluation of some available units from unused onsite equipment. Installation of the instrumentation is scheduled to commence in mid-December, when the craftsmen become available from other tasks. This is later than originally scheduled but will not impact on the test schedule for the B-5 test.

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PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. S. Booth

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #B0191)/NRC #60 19 10 1

TECHNICAL HIGHLIGHTS

No technical highlights this month.

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

PROGRAM MANAGER: Betty F. Maskewitz

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

TECHNICAL HIGHLIGHTS

Seven magnetic tapes were received from the Data Bank System containing test data for TFTF tests 151, 152, 166, 167, 168, and 169 and FLECHT-SET test 15134. Attempts to copy these data were unsuccessful. Analyses of 12 previous tapes revealed an unsuccessful attempt to create these IBM binary format tapes on a CDC machine. Each of the 19 bad tapes were returned for rewriting.

ORNL Foreign Trip Report 691, a record of Betty F. Maskewitz's participation in meetings and technical discussions in Germany and France for the period August 11-29, 1979 was issued. She was invited to represent the NRC Data Bank program at the 5th International Conference on Structural Mechanics in Reactor Technology (SMIRT-5), August 13-17, 1979, Berlin (West), and to participate in a post-conference seminar on the Computational Aspects of the Finite Element Method (CAFEN-5). Ms. Maskewitz gave two invited papers: "The USA NRC/RSR Data Bank System and the Reactor Safety Research Data Repository (RSRDR)" and "A Survey of Major Data Management Systems." She also served as a consultant on data management. FTR 691 summarizes these conferences and other technical visits.

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PROGRAM TITLE: Nuclear Safety Information Center

PROGRAM MANAGER: W. B. Cottrell

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

TECHNICAL HIGHLIGHTS

During the month of October, the staff of the Nuclear Safety Information Center (a) processed 971 documents, (b) responded to 81 inquiries (of which 51 involved the technical staff and 11 were for commercial users), and (c) made 19 computer searches. The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 71 times between September 12 to 28 making it one of the most utilized of the 25 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 14 visitors and participated in 3 meetings.

One NSIC report was issued during October: *Annotated Bibliography of Licensee Event Reports in Boiling-Water Nuclear Power Plants as Reported in 1978* (ORNL/NUREG/NSIC-164). Several other NSIC reports are in various stages of preparation, including *Radiation in Perspective* (ORNL/NUREG/NSIC-161); *Annotated Bibliography on the Safeguards Against Proliferation of Nuclear Materials* (ORNL/NUREG/NSIC-160); *Breeder Reactor Safety: Review of Current Issues and Bibliography of Literature* (ORNL/NUREG/NSIC-166); *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); *Reports Distributed Under the NRC Reactor Safety Research Foreign Technical Exchange Program* (ORNL/NUREG/NSIC-170); *Annotated Bibliography on the Transportation and Handling of Radioactive Materials* (ORNL/NUREG/NSIC-168); *Bibliography of Reports on Research Sponsored By the NRC Office of Nuclear Regulatory Research* (ORNL/NUREG/NSIC-169); and *Description of Selected Incidents Which Have Occurred in Nuclear Facilities* (ORNL/NUREG/NSIC-176).

During the month of October, we received 31 foreign documents (27 German and 4 UKAEA). In accordance with the arrangements effective January 1, 1979, a copy of each of these have been sent to Steve Scott for microfiche processing. In addition, the foreign language documents were reviewed for translation (see letter of October 31, 1979, to G. L. Bennett).

NSIC's selective dissemination of information (SDI) is available to paying users (as well as exempt users). During the month of October we added 2 exempt which, with other withdrawals and renewals, leaves the SDI service at a total of 400 users.

The regular *Nuclear Safety* staff meeting was held on October 3. Minutes of that meeting and a tentative outline for the next several issues of *Nuclear Safety* were distributed shortly thereafter. TIC sent the final galleys of *Nuclear Safety* 20(6) to the printer on October 23rd. The technical content of *Nuclear Safety* 21(1) is in final composition at TIC, but that issue awaits the September–October "current events" material (submission deadline - November 15th). All technical articles for *Nuclear Safety* 21(2) have been peer-reviewed and are being edited for submission to NRC, DOE, and TIC.

TABLE 1 RECON DATA BASE ACTIVITY FROM 09-12-79 TO 02-28-79*
(12 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	2,099	3,314	90,141
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	337	527	10,348
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	196	718	21,357
EMI	(EMIC) ENV. MUTAGENS INFO.	141	225	7,290
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	102	137	4,608
GAP	(DOE) GENERAL AND PRACTICAL INFO.	98	214	898
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	79	155	5,347
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	71	137	10,932
ESI	(EIC) ENV. SCIENCE INDEX	69	169	4,623
EIA	(EIC) ENERGY INFO. ABSTRACTS	44	79	628
PRD	(TIC/NRC) POWER REACTOR DOCKETS	30	56	946
API	(API) AMER. PETROLEUM DATA BASE	27	32	274
WRE	(WRSIC) WATER RESOURCE RESEARCH	27	96	456
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	26	24	5
NRC	(LC) NATIONAL REFERRAL CENTER	21	42	506
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	17	32	14
NER	(EIC) NATIONAL ENERGY REFERRAL	17	21	-
CIM	(DOE) CENTRAL INVENTORY OF MODELS	14	17	45
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	13	9	46
NBI	(NBIC) NATL BIOMONITORING INV.	12	7	40
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	11	12	-
RSC	(RSIC) RADIATION SHIELDING CODES	9	2	-
RSI	(RSIC) RADIATION SHIELDING INFO.	9	9	20
TUL	(U. TULSA) TULSA DATA BASE	8	13	175
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	2	-	-

*Due to the many Programming changes required to implement cost recovery, the statistics for the first six working days in September were inadvertently lost. In order to project the totals for the month the totals shown should be multiplied by 1.46. We are sorry for the inconvenience.

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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 10 01

TECHNICAL HIGHLIGHTS

Task 1: THTF Operations — The design and planning of modifications for the quasi-steady-state bundle uncovery tests, test # 3.03.6A was completed. Work is now in progress for preparation of test # 3.01.5A, isothermal blowdown: cable trays have been added and new positive cables installed; water was circulated twice for instrument DAS verification; new spool pieces were instrumented; and the signal conditioning system was completed.

Task 2: Two-Phase Instrument Development — A mass flow code, AMICON, is being tested and debugged for use on THTF engineering units tapes. The code calculates mass flows at the spool pieces from three homogeneous models and the Aya model using various combinations of spool piece data. The code also produces an estimate of the best model based on steady-state uncertainties which are carried through the mass flow calculations.

Fabrication of three-bladed full flow drag disk targets has been completed and the three-bladed targets have been installed in the THTF spool pieces.

Ranging and acquisition of instrumentation required for the bundle uncovery tests are being undertaken. Full flow drag disk targets for the 2-in. outlet spool piece steam flow measurements are being fabricated. Low range differential pressure cells for liquid level indication across the test section have been acquired.

Task 3: FCTF Testing — Testing of a fuel rod simulator (FRS) in the FCTF is currently underway to determine limits for safe operation of the bundle uncovery tests. The FRS is being heated in air at temperatures of 1000, 1300, and 1600°F followed by calibration tests to determine the effects of uncovery at high temperatures on the mechanical integrity of the rod.

Task 4: Analysis — Thermal-Hydraulic Analysis. Pretest analysis for the isothermal blowdown (test # 3.01.5A) is completed. The test will have an outlet break area of 0.0081 ft^2 and an inlet break area of 0.00675 ft^2. The pressurizer surge line will be orificed to 0.25-in.-ID. The average test section temperature and pressure will be 545°F and 2250 psi.

Pretest analysis for the vertical upflow film boiling test (test # 3.03.6B) and the first powered reactor simulation test (test # 3.04.5B) is continuing.

Pretest analysis of the quasi-steady-state bundle uncovery tests (test # 3.02.6A) is underway. An algorithm to estimate steam outlet temperatures and peak clad temperatures during the uncovery phase of the tests has been derived and programmed. Preliminary results indicate that rod surface temperatures in the range of 1000 to 1500°F are likely to be encountered. Current work is centering on prediction of quench front velocities during the recovery stage of the tests.

RELAP4 MOD6 Update4 calculations for tests 165, 166S, 167R, and 177 have been performed. Comparisons with engineering units and ORINC data are being made.

The pressurizer surge line in tests 165 and 166S appears to be about 20°F (11.1°C) colder than test section outlet line temperatures. Since the volume of this water is about 0.9291 ft^3 (0.0263 m^3), and since the volume of the heated length of the core is 0.7620 ft^3 (0.0215 m^3), significant additional cooling capacity existed which would not have been available if the surge line had been at outlet temperature conditions. A quantitative estimate of this additional cooling capacity has not yet been obtained. Since homogeneous equilibrium is assumed in RELAP, this density gradient is smeared out. Although not as severe, this interface presents the same difficulty as the "hot-cold-hot" interface existing downstream of the main heat exchangers prior to moving the pressurizer to its present location. One possible way to alleviate this complication is to place heaters on the pressurizer-surge line so that it remains at the same temperature as the test section outlet fluid.

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Nuclear Pin Simulation Analysis. Development of the local fluids condition interface code for PINSIM-MOD2 has been completed; debugging of the transient calculational control routines continues.

Nuclear pin simulation analysis of THTF test 105 continues. All of the calculations invalidated by a recently-discovered error in PINSIM have been repeated, and preparation of the report is again underway.

Data Management. Debugging of the THTF Bundle 3 MOD 2 Data Reduction Code (DACREP) is continuing. The routines which format the transient output file for both blowdowns and power drops will be debugged when appropriate data becomes available.

A FORTRAN callable subroutine has been written to be used in the preprocessor to ORMDIN. This preprocessor requires a list of CCDAS channel numbers for both sheath and middle thermocouples to be processed. This subroutine will search the Instrument Data Base (IDB) file and create a list containing all sheath and middle thermocouple channel numbers. It then determines which thermocouples are questionable or failed and deletes those channel numbers from the list before returning to the calling routine.

Data reduction has been completed for FCTF tests 79-10-1, 79-10-2, and 79-10-3.

Electric Pin Analysis. Development of the "transfer function" for bundle 3 multi-dimensional inverse calculations is continuing.

Development of the computational techniques and code for the one-dimensional solution of the inverse problem with respect to bundle 3 FRSs is continuing.

A HEATING5 R-Z model of a bundle 3 FRS from the THTF test section outlet to the upper extent of the air duct chimney has been developed and is now being debugged.

The development of a preprocessing program for ORINC for bundle 3 inverse computations has started. This program will hopefully remove the I/O burden from ORINC, thus leaving just the computational-cpu type work for ORINC.

The development of the heat transfer coefficient comparator program has started.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #41 89 55 12 3 (189 #B0421-8)NRC #60 19 10 01

TECHNICAL HIGHLIGHTS:

The initial draft of the Memphis State University Center for Nuclear Studies (MSU/CNS) report summarizing conclusions and recommendations from the survey of existing U. S. nuclear power plant simulators was received and reviewed by ORNL. A meeting was held at ORNL October 19 during which the results of the study were reviewed by ORNL staff including Operations Division personnel. Recommendations for modifications and additions to the material and to the report are being acted upon by MSU/CNS. It is planned to have the revised draft prepared by mid-November.

Planning for simulator experiments to be started by late FY 80 was initiated.

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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 10 04 1

TECHNICAL HIGHLIGHTS

The HOBBIE-7 test was shipped to ECN-Petten on November 5, 1979, and will be installed immediately following removal of HOBBIE-6. Since HOBBIE-7, -8, and -9 are each scheduled to be dual creepdown/creepout stress reversal tests, it was decided to continue HOBBIE-6 past its planned shutdown and to reverse the stress state by internally pressurizing the cladding specimen. This reversed stress state will be held for approximately 400 h, and the results will be precursors to the later tests.

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