INTERIM REPORT

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for the
DEPARTMENT OF ENERGY

INTERIM REPORT

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ABSTRACT

Highlights of technical progress during July 1980 are presented for sixteen 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:

ORNL personnel travelled to Japan to assist in the installation of the SCTF upper plenum structure film probes. Other SCTF related activities have included: preparation of sensor installation procedures, preparation of initial draft of software manual, and completion of the fabrication of the wall mounted film probes. A JAERI representative (Y. Murao) visited ORNL on July 25 to see our test facilities and discuss various SCTF and CCTF instrument interface problems.

An extra SCTF wall film probe module was fabricated for a series of tests in the steam-water calibration chamber. The main objective of the tests will be to verify EP probe operation at high temperature and to study the structure of the interfaced waves and determine the conditions under which there velocity might be determined by cross correlation. These will be conducted in August.

The FORTRAN subroutines for velocity and void fraction are being modified as required to accommodate the input/output formats of the JAERI data reduction programs. A preliminary partial draft of the user's manual, as mentioned above, has been mailed to JAERI. This is to provide them enough information to begin writing the necessary software to execute the ORNL subroutines. The completed draft will be mailed the latter part of August. The final version of all the subroutines is scheduled for delivery by Oct. 31, 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:

Data obtained with a heated thermocouple coolant sensor in forced convection steam-water flow were analyzed and evaluated. For the range of conditions used, the output temperature differences were found to be nearly independent of the flow stream's velocity and void fraction, in two-phase flow. The readings were influenced by the applied heater power and the presence or absence of a liquid phase.

A perforated-tube splash shield was designed, fabricated and tested with a level sensor in the transparent, counter-current facility. Little improvement in the sensor's performance was noted, compared to results obtained with a bare probe, so the splash shield is being modified.

The Fluid Components, Inc. heated RTD level device was modified by adding a cover over the sensor element. Tests of the device were repeated in the natural convection level test facility, with significantly improved results. Temperature excursions observed in earlier tests, believed to be caused by rewetting of the probe when it was immersed in saturated steam, did not occur with the modified sensor.

A preliminary design of a heated thermocouple sensor for installation in the upper plenum of the Thermal Hydraulic Test Facility at ORNL, was completed. The design is for a vertically oriented probe which incorporates a splash shield.

Three papers were prepared and presented by program members at the Review Group Conference on Advanced Instrumentation for Reactor Safety Research held at Oak Ridge on July 29-31. Papers were given on thermal and acoustic level sensors for use in PWR reactor vessels and on cross-correlation studies with two drag flowmeters.

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:

Three under-water tests (FAST 58, 59, and 60) were performed in the FAST vessel this month. FAST 58 and 59 were part of the effort to evaluate the use of acoustic methods for bubble detection. In FAST 58, no meaningful data relative to bubble size was obtained from the two acoustic transducers mounted on the vessel. FAST 59 was an acoustic listening test, in which three transducers (2.25 MHz, .5 MHz, and .1 MHz units) were mounted on the vessel; with these the acoustic noise due to the preduction of the bubble was recorded. It was found that for 20 milliseconds after CDV discharge, the noise level generated by the CDV discharge was greater than the planned acoustic signal level of the bubble detection transducers. This result probably indicates that no meaningful acoustic measurements can be made until at least 20 milliseconds after the start of bubble formation.

In contrast to the above discouraging results for the acoustic system, the fallback positions of using pressure changes in the argon cover gas as a measure of bubble volume continues to look encouraging. Cover gas pressure changes measured in both FAST 59 and 60 gave reasonable estimates of the maximum bubble sizes. FAST 60 results were particularly interesting; a number of oscillations in cover gas pressure - corresponding to oscillations in pressure measured under water - were observed. FAST 60 was also the first test where the water level was reduced to ~200 mm (8 inches) above the test sample (this was one of the additional test conditions outlined in the draft report, recently transmitted to NRC, on water test progress). As in most previous tests, no aerosol was transported to the cover gas in FAST 60 even with only 8 inches of water above the sample.

NSPP/CRI-II:

The final scheduled uranium oxide aerosol test (No. 209) was conducted late this month. This test was an attempt to produce a higher concentration uranium oxide aerosol in the 15-20 μg/cm^3 range. Previous NSPP tests under dry conditions have not exceeded an aerosol concentration of about 6 μg/cm^3. Data from this test will be compared with that from Run 208 which was a high concentration uranium oxide aerosol test conducted under highly humid atmospheric conditions.

Complete analytical data from Run 209 will not be available until next month. However, visual observation of amount of material contained in the total fallout samplers suggests at the objective of this test was achieved.

A new LASL-modified Stöber spiral aerosol centrifuge to be used primarily at the Nuclear Safety Pilot Plant (NSPP) in LMFBR aerosol studies was delivered initially at the CRI-II facility for inspection and primary calibration before transfer to its permanent location.

The new centrifuge has inlet slit spools which represent the best effort to eliminate high inlet losses (for large particles) that other investigators have generally encountered.

Some minor malfunctions were discovered in the control valves and when these were corrected a number of calibration check points were made with standard monodisperse polystyrene latex microspheres.

ANALYSIS:

Pre-test predictions, using HAARM-3, were prepared for NSPP Test No. 209.

The entire series of NSPP experiments through the recently completed mixed aerosol series are being analyzed with HAARM-3 to develop a comprehensive comparison for code validation purposes. To aid in this comparison, improvements were made in the codes plotting capability to allow simultaneous presentation of different cases on the same graph.

Work is continuing on analyzing the heat and mass transfer effects of the under-water FAST experiments.

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 amplifier system for eliminating the interfering pulses emanating from the Westinghouse P250 plant computer has been designed and constructed. The system consists of two standard 19 in. racks containing twelve Burr-Brown model 3456B Isolated Instrumentation Amplifiers each and will be installed in series with the plant computer inputs by TVA. The complement of twenty-four will provide adequate installed spares (8) for system maintenance or expansion. The system will be delivered to the Sequoyah site in early August to coincide with the installation of the signal patch panel by TVA.

In addition to the isolation amplifiers, a method of eliminating the pulses by means of front-end software and hardware modification to the Reactor Surveillance System is being pursued. This capability, if successful, will provide flexibility in handling such pulses and may be implemented in the advanced system currently being procured.

The selector switch for the four core exit thermocouple signals has been designed and constructed. This switch enables the signals to be connected either to the plant computer or to the Reactor Surveillance System. The signals will remain connected to the Reactor Surveillance System except when needed by plant personnel to perform fuel performance calculations (about two hours every two weeks).

The implementation of the automatic power-fail recovery feature in the Reactor Surveillance System has been completed.

The isolation amplifier racks and thermocouple selector switches will be installed in Sequoyah during the first week of August; the Reactor Surveillance System installation is currently planned for the second week.

Procurement of Advanced System

The present estimate from DEC of the delivery time for the PDP 11/34 processor and RK-05 disks is 240 days ARO. By accepting RL-01 disks in lieu of the RK-05 disks the delivery estimate is reduced to 150 days ARO. Accordingly, a request was made to incorporate this modification to the purchase order, making the delivery date approximately January 1, 1981. This item is the controlling factor in the schedule for completion of the advanced system, currently estimated to be the second quarter of FY 81.

The telephone data communications link between Sequoyah and ORNL will be incorporated in the advanced system. Extensive modifications of hardware (memory) and software (monitor) would be required in order to incorporate the communications link in the present Reactor Surveillance System. Because this would significantly delay the deployment of the system at Sequoyah, the link will not be implemented in the present system.

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 — A Vessel Integrity Review Group Meeting was held in Silver Spring, MD, on July 23 and 24. Six members of the HSST program staff made presentations to the group.

Andre Pellisier-Tanon and Pierre Sollogoub of Framatome and S. S. Palusamy of Westinghouse Electric Corporation visited ORNL on July 25 to exchange information on thermal shock studies and elastic-plastic fracture analysis.

Task 2: Fracture Mechanics and Analysis — This month considerable effort was spent examining and attempting to debug the nozzle corner flaw option of the Oak Ridge IBM version of the NOZ-FLAW computer program. After studying the coding in the mesh generation subroutine, it was concluded that the strategy presently employed in the program would not generate an adequate nozzle-cylinder mesh. An appropriate strategy similar to that used in the MULT-NOZ program is presently being incorporated into NOZ-FLAW. This work is essentially completed and several test problems will be attempted next month.

Task 3: Irradiation Effects — Irradiation of the first capsule of the Fourth HSST Irradiation Series is continuing with excellent temperature control. The average temperature is near the aim 288°C, and the maximum temperature difference after six months is near 9°C.

Irradiation of the second capsule was started June 26. Temperature control is considered good, but not quite as good as for the first capsule of this series. The average temperature after one month of irradiation is 286°C. Two of the thermocouples at the top corners of the specimen array are reading about 11°C below the average, and the remaining thermocouples are reading within 7°C of the average temperature.

Parts are being ordered for the third capsule of this series (specimens have been machined).

Specimens for the fourth capsule are being prepared by MPA, Stuttgart, Germany.

Task 4: Thermal Shock — Preparations for TSE-5A were continued and a full-scale thermal hydraulic experiment was conducted on July 29. The results appear to be quite satisfactory; the severity of thermal shock is approximately the same as that for TSE-5. The TSE-5A test specimen assembly was dismantled and the test specimen prepared for heat treating (temper at 663°C).

Prior to the TSE-5A experiment the inner-surface-spray-coating development work was completed successfully. The final coating technique included application of one coat of 3M-34 followed by five coats of 3M-NF34.

The 22 1T compact specimens machined from thermal shock prolongation TSP-2 tempered at 680°C for 4 h are being fatigue precracked.

Task 5: Simulated Service Tests — Babcock and Wilcox Company tested 15 Charpy V-notch specimens from each of three samples of the V-8A preliminary automated submerged-arc welds. These samples were heat-treated at 593–621°C for 50 hr and represented three different weld compositions. Impact tests were performed at five temperatures from 52–135°C. Charpy energies generally fell within the target range of 47 to 75 J for the three compositions, but the percentage ductile fracture appearance at 93°C was not consistently as high as is required. The weld made with a flux mixture of 90% Linde 60 and 10% Linde 80 showed a transitional behavior in the temperature range of the tests and, consequently, is not likely to be useful. The 60–40 and 75–25 compositions produced 95 to 100% shear at temperatures of 93°C and higher. These show some promise. There appeared to be a systematic variation of properties with location in the weld, a matter that is being investigated further.

Task 6: Pressurized Thermal Shock — The K_I values calculated under Task 2 for various pressurized thermal shock for three flaw depths in an ITV nozzle were compared with the fracture toughness values, K_{Ic} as determined for the forging from which the nozzle in the remaining vessel, V-10, was fabricated. If the material response were governed by "lower bound" fracture toughness values, then the K ratio, K_I/K_{Ic}, would exceed unity. Stable and/or limited unstable crack growth would occur depending primarily upon initial vessel and fluid sink temperatures which would determine whether the material at the crack tip is in the upper transition or upper shelf region. If the material response were governed by average or "upper bound" fracture toughness values, then no crack extension would be predicted.

The results described above were discussed in detail at the Vessel Integrity Review Group Meeting in Silver Spring, MD, on July 23, 1980. After considering both the difficulties in performing and analyzing such an experiment as well as applying the results to other geometries of interest (e.g., irradiated belt-line region), it was decided to investigate pressurized thermal shock experiments in which a cylindrical ITV would be pressurized internally and both flawed and thermally shocked externally. Such an experiment would potentially have the advantages of being experimentally more tractable, more readily analyzable, and provide results applicable to the belt-line region of a reactor pressure vessel.

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:

Code Development Activities: An improved simulation of the steam lines has been implemented in ORTAP. This improvement consisted of reformulating the governing differential equations for the state of the steam, and then solving these differential equations using a Runge-Kutta method. However, the new simulation suffers, as the previous simulation did, from being a mathematically explicit formulation with respect to time. This limits the overall computational time step to small values (~0.1 s). Modifying the steam line simulation to be implicit with respect to time would involve modifying the entire plant steam system (steam generator and reheater, steam lines, electrical turbines, and helium circulator turbine) as one system of differential equations. This would involve a considerable effort, and we do not plan to do this.

Frequent discussions were held with Public Service Co. of Colorado (PSC) about detailed design data for the FSV steam generators. A problem with the steady-state search technique for BLAST has been uncovered, and an investigation of this problem has been initiated.

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.

We have changed the program in the microcomputer so that the raw data is stored on the magnetic tape rather than the calculated data. This will allow additional and different processing of the data to be performed if we discover a property variation that was not originally anticipated. The microcomputer will still calculate and display the properties on a strip-chart recorder in real time as the tube is scanned. In addition, another microcomputer that will be at the remote operator station in the truck can also read the data, convert it to the properties, and plot the data.

We are continuing to install the instrumentation in our remote inspection station in the truck. Cabling is being constructed for the remote operations and the instrumentation has been installed in special shock-mounted chassis. Two large reels have been constructed to hold the large cable for the camera and positioner control and the instrument control. These reels will be permanently mounted in the truck and we can roll out the length of cable needed at any inspection site.

A new tube standard with 5 thicknesses, 3 inside diameters, and 20 different hole sizes has been constructed. It has been run with TUBRDG, TUBFIT, and PLTRDG. The runs thus far have shown some improvement, and we are arranging our properties in different manners to get additional improvements.

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. A series of tests were run in the three-module air/water loop with the tie plate turbine relocated to the midplane of the upper core support plate. Comparison of the results from these tests with earlier results with the turbine located 4 mm above the tie plate showed that the data scattered around a line based on the ratio of the flow areas with a maximum deviation of ±30%. However, the results for air only indicated the same velocity (for a given volumetric flow rate) for the turbine located at both positions.

The thermal shock tests in the steam/water loop have been completed. These tests were conducted with the hot leg extended 35 in. downward toward the upper-core support plate. Data reduction has not been completed. However the most severe shocks that have been observed so far were $-140\ lb_f$ and $+50\ lb_f$. These results were obtained with a steam inlet flow rate of 7.5 ft^3/sec, 9.7 gpm of core spray and 52 gpm of hot leg injection with a subcooling of 180°F. Since the large negative force exceeded the design limits for the transducer, there was a zero shift. However when the transducer was removed and recalibrated there was no change in linearity of response to load and no change in the temperature response of the transducer. There are two important points about these results: (1) the production models of the transducer will contain stops to prevent over-ranging and (2) even when there were no stops to prevent over-ranging there was no damage to the transducer and since the zero shift was readily identifiable there was no loss in data.

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: Followup calculations of the PCA Blind Test have been received from four participants to date. Technical Newsletter No. 5 has been distributed to request that the participants try to get their results to F. B. K. Kam early in September 1980. Information received after this date will be analyzed and processed, but the information will not be available for insertion into the first NUREG report due Nov. 30, 1980.

Task 2: Benchmark Fields -

A. PCA - No highlights.

B. ORR-PSF -

A computer program for analyzing temperature data from the ORR-PSF experiment has been written, but it has not been tested. The program computes a daily summary of the irradiation history for each capsule. Average temperature (with variances), irradiation time, integral power, and integral power with selected temperature ranges are to be obtained for each thermocouple.

SSC-1 has been transferred from the ORR-PSF pool to the hot cells. Equipment to cut the capsule will be moved into the hot cells the first week in August. Decapsulation of SSC-1 should be completed in August.

C. Surveillance Dosimetry Measurements Benchmark Facility (SDMB) at the Bulk Shielding Reactor (BSR) -

-sign of the BSR-SDMB has begun and it is essential that engineering drawings of typical surveillance capsules used in power reactors be available to interface capsules with the facility. A letter will be initiated to request these drawings from the vendors.

Task 3: Neutron Field Characterization - Transport Calculations and Dosimetry -

Additional data from the PCA Blind Test calculations have been received and are being processed. Several hypotheses for the explanation of the discrepancies between calculations and experimental data are being tested.

Task 4: ASTM Recommended Procedures for LWR-PV Surveillance Embrittlement Programs - No highlights.

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:

Retrieval and reduction of the B-5 test data continued during this report period. Report quality data plots were produced to show the transient behavior of the test parameters. Problems with the software prevented recovery of the data in tabular format, which is necessary to define burst temperatures, pressures, and times with the desired accuracy. These problems have been resolved and retrieval of the data in this format will be accomplished early next month.

The subcontract for hydraulic testing of the B-5 bundle has not yet been finalized, due to extensive reviews by the legal experts. Delays associated with the process have caused about a three month delay in the availability of this important data and will cause a similar delay in destructive examination of the bundle to obtain detailed strain data.

Fabrication of the fuel pin simulators for the next bundle test (a 6 X 6 array identified as B-4) is on schedule. Design drawings of the test assembly are nearly complete, and fabrication of the components will be initiated as soon as the drawings are released.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 11 4 (189 #0191)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

Loose-Part Detection Systems. A detailed summary report on the loose-parts monitoring work was transmitted to NRC. A condensed version of this report will be issued next year as a NUREG report.

PWR Baseline Signature Acquisition. In our previous report, we stated that recently acquired neutron noise signatures from a new CE plant were lower in amplitude than other CE plant signatures in our baseline library. We have since found out that the new plant has a different design which may account for the lower amplitude neutron noise (CE and plant personnel have stated that they also observe less neutron noise in the new plant).

A letter report is being redrafted after receiving technical review.

BWR Stability Monitoring. We have completed our assessment of the feasibility of monitoring stability of BWR using neutron noise. We submitted a paper, "Stability Monitoring of Boiling Water Reactors by Time Series Analysis of Neutron Noise," to Nuclear Science and Engineering. A final report on our assessment of stability monitoring using noise analysis is being prepared.

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:

Activities in relation to the Reactor Safety Research Data Repository have been suspended in order to reallocate funds and manpower to make a determination of how research information management might be improved to better meet NRC's overall needs.

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 July, the staff of the Nuclear Safety Information Center (a) processed 728 documents, (b) responded to 57 inquiries (of which 31 involved the technical staff and 8 were for commercial users), and (c) made 18 computer searches. The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 147 times between June 2 to 30 making it the fifth most utilized of the 28 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 6 visitors and participated in 4 meetings.

One NSIC report is in reproduction: *Annotated Bibliography on the Transportation and Handling of Radioactive Materials* (ORNL/NUREG/NSIC-168). 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); *Bibliography of Microfiched Foreign Reports Distributed under the NRC Reactor Safety Research Foreign Technical Exchange Program, 1979* (ORNL/NUREG/NSIC-177); *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); and *Nuclear Power Plant Operating Experience — 1979 Annual Report* (ORNL/NUREG/NSIC-180).

During the month of July, we received 14 foreign documents (2 Italian, 6 German, and 6 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 July 31, 1980, to H. H. Scott).

NSIC's Selective Dissemination of Information (SDI) is available to paying users (as well as exempt users). During the month of July we added 2 paid and one exempt users which, with other withdrawals and renewals, leaves the SDI service at a total of 391 users.

In view of the increasing interest and volume of literature on risk, reliability, and accident probabilities (topics which had previously been included in the NSIC information category No. 1 - "General Safety Considerations"), these topics will henceforth be included in their own category No. 23 entitled "Risk, Reliability, and Probabilistics." The scope of this new category is defined as "Covers documents that may be useful in performing reliability, cost-benefit, probabilistic and/or risk analysis on all or portions of the nuclear fuel cycle. In addition, it includes selected information on the comparison between nuclear and other energy technologies."

All technical articles for *Nuclear Safety* 21(6) were completed and mailed to NRC, DOE and TIC on July 23rd. The "current events" material (covering events which occurred during May and June) for *Nuclear Safety* 21(5) were completed by July 18th (except for the data on operating power reactors which was not yet available from NRC). Most technical articles for *Nuclear Safety* 22(1) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 21(4) were received from the printer (via TIC) on July 16th.

TABLE 1 RECON DATA BASE ACTIVITY FROM 06-02-80 TO 06-30-80
(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	3991	6600	168320
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	559	814	9576
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	348	929	22804
EMI	(EMIC) ENV. MUTAGENS INFO.	173	322	8627
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	147	341	5874
GAP	(DOE) GENERAL AND PRACTICAL INFO.	144	126	2124
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	139	248	14290
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	132	181	1937
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	98	151	783
ESI	(EIC) ENV. SCIENCE INDEX	93	201	1135
IPS	(TIC) ISSUES AND POLICY SUMMARIES	79	101	42
EIA	(EIC) ENERGY INFO. ABSTRACTS	63	122	523
WRE	(WRSIC) WATER RESOURCE RESEARCH	54	88	1353
SLP	(FRANKLIN) SOLAR DATA BASE	40	38	31
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	36	60	220
NES	(NESC) NATIONAL ENERGY SOFTWARE	31	66	7
API	(API) AMER. PETROLEUM DATA BASE	30	41	950
PRD	(TIC/NRC) POWER REACTOR DOCKETS	25	31	795
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	24	20	390
NRC	(LC) NATIONAL REFERRAL CENTER	23	21	-
CIM	(DOE) CENTRAL INVENTORY OF MODELS	22	60	-
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	18	26	-
RSI	(RSIC) RADIATION SHIELDING INFO.	18	9	522
TUL	(U. TULSA) TULSA DATA BASE	16	40	258
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	11	5	-
RSC	(RSIC) RADIATION SHIELDING CODES	10	9	20
NER	(EIC) NATIONAL ENERGY REFERRAL	7	4	-

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 Operation - No single-rod loop activity was planned for this month.

Task 2: Analysis - Nuclear Pin Simulation Analysis. Posttest analysis of THTF Test 3.05.5B is proceeding on schedule. A letter to the NRC Program Manager which included plots of selected instrument responses and the results of some preliminary analysis was issued eight working days after the test was run. The quick-look report is at this writing 90% complete, and will be issued within the next week. Analysis which will be reported in the next report (to be issued on September 1) is underway.

Electric Pin Analysis. The preliminary run on THTF Test 3.05.5B (reactor simulation test) by the preprocessor program for ORINC and ORMDIN has been completed (on schedule). Model and program development for the THTF core shroud inverse computations (needed for the quasi-steady-state and uncovery/recovery tests) has been started.

Thermal-Hydraulic Analysis. An interim report on the results of THTF Test 3.03.6AR, Upflow Film Boiling, was completed on schedule on July 21, 1980. The report presents preliminary comparisons of experimentally determined heat transfer coefficients to those produced by film boiling correlations available in RELAP4 Mod 5 Update 2. The bundle fluid conditions used to evaluate the correlations were calculated by a homogeneous, thermodynamic equilibrium computer code (a locally-modified version of RELAP4). In the correlation comparisons, the Dougall-Rohsenow correlation was high (by factors of 1.5 to 2) while the Groeneveld 5.7 and 5.9 correlations were closer to the data and were both high and low at times. Preliminary analysis of the test suggests that a significant degree of thermal nonequilibrium existed. Thus correlation comparisons based on bundle fluid conditions produced by an equilibrium code cannot be used as a basis for any final or definitive judgments of the correlations. The applicability of RELAP4 with these correlations can be assessed, however.

The Quick Look report for THTF Test 3.04.7R, Downflow Film Boiling, was completed on schedule on July 16, 1980. An expanded version of a code to compare experimental and correlation heat transfer coefficients is approximately 85% completed. This work is behind schedule due to the necessity of repeatedly having to divert manpower to other tasks. Documentation of the code is 10% completed.

Application of the COBRA/TRAC computer code to THTF Test 3.03.6AR is continuing. Calculations of the first 4 sec of the 30-sec transient have produced slightly early predictions of departure from nucleate boiling and somewhat high predictions of rod surface temperatures at 3 sec.

Data Management. Preliminary data reduction and process tapes have been generated for the Upflow Film Boiling Test (3.03.6AR), the Downflow Film Boiling Test (3.04.7), and the Reactor Simulation Test (3.05.5B). Questionable instruments and required modifications to the Instrument Data Base (IDB) have been identified for each of these tests.

Processing of power drop tests performed on 5/3/80 and 5/15/80 have been completed. Data reduction tapes for these tests have been generated for use in bundle characterization.

Task 3: THTF Operations - The bundle was operated on one occasion this month, the double-ended LOCA simulation test 3.05.5B. For this blowdown, the nominal break areas were 32 and 28% at the inlet and outlet, respectively. At blowdown, the test section inlet and outlet temperatures were ~550 K (530°F) and ~603 K (625°F), the test section flow rate was 3.28×10^{-2} m^3/s (520 gpm), and the heater rod power was 126.6 kW/rod. After blowdown, the rod power was cycled automatically by the power programmer system. Although the test section was operated at rather extreme conditions, it performed quite satisfactorily, thus allowing successful completion of the test.

Modifications are now being made in preparation for the remaining tests scheduled for FY 80. These modifications include:

1. Installation of shroud wall differential thermocouples at 12 test section locations.

2. Fabrication and installation of two new in-bundle thermocouple array rods.
3. Fabrication and installation of two in-bundle gamma densitometers and a motor drive system.
4. Range change and calibration of the 10 in-bundle differential pressure transducers in preparation for the upcoming bundle uncovery tests.
5. Pipe and loop modifications, including instrumentation, for future tests.

Plans are being evaluated and finalized for the upcoming August and September tests.

Task 4: Two-Phase Instrument Development - The ion chambers for the in-bundle densitometer system have been received. Fabrication of one of the two Gd_2O_3 sources has been completed and the second will be available on schedule. Fabrication of the source-ion chamber linkage components has been completed. Assembly and checkout of components have begun. Installation is expected prior to the next series of tests.

The mass flow code AMICON has been run for the Reactor Simulation Test (3.05.5B) and the results applied to Analysis. Development of a methodology for determining enthalpy uncertainties for use with AMICON has been completed. Work has begun on including the methodology in AMICON.

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:

Development of software for extraction of event-specific data from raw data tapes produced during simulator experiments has been completed, and tabulation of results has been initiated. A videotape of operators (General Physics instructors) performing the simulator exercises has been made to demonstrate experimental procedures and illustrate operator actions and simulator response. Editing of the tape is in progress.

Collection and analysis of field data is continuing. Three site visits were completed in July. A preliminary summary was prepared of routine statistical techniques potentially applicable to derivation of the calibration between simulator results and field data. A list of candidate events for the BWR simulator exercises to be conducted in FY 1981 was prepared.

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:

There is no report for the month of July, 1980.

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