INTERIM REPORT

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for the
DEPARTMENT OF ENERGY

INTERIM REPORT

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ABSTRACT

Highlights of technical progress during May 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:

All of the SCTF in-core flag and film probes have been installed in the heater rod bundles. ORNL personnel were on site at the Okazaki plant in Kobe to supervise and assist with the installation. All of the sensors satisfactorily passed electrical tests after installation.

ORNL personnel also participated in meetings on CCTF and SCTF instrumentation at Tokai while in Japan. Most of the outstanding interface problems relating to CCTF sensor design have been resolved so that design can proceed. A few critical items relating to SCTF electronics and data acquisition still remain undecided, primarily relating to anti-aliasing filters for the data acquisition system and whether they will be provided by JAERI or the USNRC.

Fabrication of the upper plenum prong probes and structural film probes for SCTF has been completed. The film probes were shipped to Japan in mid-May. JAERI has requested that no more shipments be made until after July 1.

The testing of the prototype SCTF in-core flag probe and in-core film probe in steam-water and air-water was completed and the bundle removed from the test stand. The following is a brief summary of the test results:

1. The flag probe in steam/water gave velocities which agreed very well with previous data on PKL style probes. They also matched the previously developed correlation for obtaining liquid and steam velocities.
2. The probe void fractions in steam-water do not agree very well with the gamma-densitometer downstream of the bundle. We believe the probe void fractions are in error but cannot make the required correction because of the problems with the impedance phase angle measurements during these tests. These problems did not become apparent until after the tests were completed.

They also occurred on the previous PKL tests in steam-water. We now know how to correct the problem.

3. The flag probe velocities in air-water do not fit the steam-water velocity correlation.
4. The air-water void fractions agree more closely with the downstream gamma densitometer because the problem mentioned in item 2 is not as serious in low temperature air-water.
5. Some film thickness data were obtained which looks reasonable, however failure of a sensor cable prevented operation during most of the testing.

Steam-water testing of the prototype SCTF upper plenum prong probe has also been completed. The void fraction results look much better than those of the flag probe in relation to the gamma-densitometer. The main reasons for the improvement are the correction of the phase measurement problem and the fact that the prong probe is installed in an open pipe very near the densitometer location as opposed to being in a rod bundle several meters upstream of the densitometer. The prong probe is now being tested in air-water.

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:

Experiments were performed in the TITF pressurizer with a heated thermocouple liquid-level device obtained from the U. S. Navy. The probe is enclosed in a single sheath, with heated and reference thermocouple junctions separated axially by 18 cm. It was mounted vertically in the pressurizer. The measured temperature difference was recorded with the sensor covered with saturated water or steam, with heater powers ranging from 0.4 to 21 watts, and system temperatures ranging from 25 to 300°C. Sensor response was as expected, except for occasional low ΔT readings when the probe was uncovered. Efforts are underway to isolate the cause of this behavior.

A liquid-level sensor that uses differentially-monitored resistance thermometers was tested concurrently with the Navy device. With constant heater power the output ΔT in the uncovered state was observed to decrease with increasing pressure. This is consistent with the expected improvement in heat transfer by natural convection cooling at higher pressures.

The dual-mode ribbon-type ultrasonic level device to be tested at high temperature and pressure was received from the vendor. It was found to work satisfactorily in the room-temperature level facility. A 5-point thermocouple rake was fabricated for use in determining the vertical temperature gradient in the pressurizer when the ultrasonic device is tested. A digital data processing oscilloscope was received. This will be used to process time delay information from the ultrasonic level device.

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 in the FAST facility this month (FAST 53 through 56). The water height above the sample was 710 mm in FAST 53 and 54 and was 1120 mm in FAST 55 and 56. For each of these tests, lower than usual capacitor discharge energy inputs were obtained. FAST 56 was the only test with an energy input greater than 30 kJ. In FAST 55 and 56, a pressure transducer was installed in the argon gas above the water. The data collected from these pressure measurements again indicated (as in FAST 45 and 46) that the pressure changes in the argon gas can be used to estimate the bubble volume as a function of time. FAST 56 was a high-temperature water test (∿359 K) in which movies were taken through the port at the top of the vessel — to try to visually determine if water vapor bubbles transport fuel aerosol to the cover gas.

Four vacuum tests were performed in the CRI-III facility this month (CDV 102 through 105). In each of these, a quartz view "window" was installed to permit observation (by high-speed photography) of fuel pellet heating during preheat and capacitor discharge. In each test at least one layer of UO_2 microspheres was put between the exposed pellet and the view window; it was hoped that by varying the number of microsphere layers and then measuring the pellet temperatures (using the photographic method developed at Sandia) we could extrapolate to the case where no view window was present - the normal sample configuration. CDV 103 and 104 were capacitor discharge tests (with one and two layers of microspheres between the pellet and view window, respectively), while CDV 105 was a "preheat-only" test.

NSPP:

Complete data were received from the analytical laboratory for Run 208 which was conducted to investigate the effects of high humidity on the behavior of U_3O_8 aerosol. This experiment was considered as the highest priority additional run to be made if time permitted after completion of the mixed aerosol test matrix. The high humidity in the vessel atmosphere was produced by evaporating water from the sodium burn pan. Three gallons of water (11.3 kg) were evaporated over a period of about 100 min. Approximately 30 min after start of water boiling, the U_3O_8 aerosol was generated over a period of 13.5 min; water evaporation continued for an additional 55 min. A maximum U_3O_8 aerosol concentration of 12.5 $\mu g/cm^3$ was measured about 0.5 min after termination of aerosol generation. The most noticeable effect of high humidity on the aerosol was a change in physical appearance. Electron photomicrographs showed the aerosol to be in the form of nearly spherical agglomerat s as contrasted with chain-like agglomerates found in previous test under dry conditions. The data is being assessed to determine i there are any significant differences in the aerodynamic behavior. Cascade impactor measurements indicated that the aerodynamic mass median diameter increased from 3.3 μ to about 5.5 μ over the first 10 hours of the test.

Preparations have been completed for Run 108 which will be a study of the behavior of a low concentration sodium oxide aerosol under dry conditions for comparison with mixed aerosol behavior. This test will complete the planned tests to scope the mixed aerosol behavior. Conduct of the test is scheduled for June 4.

CRI-II:

Receipt of the new rectangular-slit multistage cascade impactor from the Sierra Corporation prompted its initial test for the size distribution measurement of U_3O_8 chain-agglomerate aerosol.

A special run during which measurements were also made by the use of two other round-jet impactors and the LASL-spiral aerosol centrifuge alternately with the Sierra instrument appear to indicate that it may be a useful device for the aerosol size range of interest. We found some unexpected limitations on the total flow rate that it could accommodate which required that the three finer cut stages be removed. This, however, did not impair its size range capability which we normally encounter in this aerosol. Further comparisons are planned with mixed oxide ($U_3O_8 \cdot Na_2O_2$) aerosol.

A more concerted effort is also being undertaken to obtain more definitive information on the mechanism of the attachment of Na_2O_2 onto the U_3O_8 chain aggregates. All of the data we have obtained so far indicates that the mixed oxide aerosol deposits as if coagglomeration has occurred. Size distribution measurements also show no significant variation between the two phases. Transmission photomicrographs continue to reveal sodium clusters attached to U_3O_8 chains with no unattached sodium oxide particles present. Plans have now been made to examine some of the individual cluster aggregates from the next oxide run for the presence and proportions of both sodium and uranium by means of electron induced X-ray fluorescence analysis (EDAX). Highly dispersed deposits of the aerosol on polished carbon discs will be examined first with the scanning electron microscope and then selected clusters of 1 to 3 microns diameter will be chosen for analysis.

ANALYTICAL:

A pre-test prediction has been made for NSPP Test 108, planned as a low concentration (2.5 g/m^3) sodium oxide aerosol experiment. An investigation of the applicability of integral balance heat conduction theory has been completed leading to simple formulas to be used in analyses of FAST underwater tests. Development of the code structure to describe these tests continues.

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. TVA has made a written request that ORNL provide funds for the installation of a patch panel in the Sequoyah plant computer room, estimating that the cost will not exceed $15,000. At the request of ORNL, DOE/ORO has forwarded to TVA Interagency Agreement No. DE-AI05-80OR20730 which provides an account for this purpose limited to $15,000.

Several methods have been investigated for eliminating pulses emanating from the Westinghouse P250 plant computer. Since many of the input signals to the Reactor Surveillance System are in parallel with the inputs to the plant computer, these interfering pulses render the signals unusable by the Reactor Surveillance System. The most promising method is the use of isolation amplifiers between the inputs to the plant computer and the point at which the signals are routed to the Reactor Surveillance System. The original TVA Design Change Request for the patch panel at Sequoyah has been modified to incorporate the isolation amplifiers and is currently awaiting approval. Each of the four core exit thermocouple signals will be supplied with a selector switch since the isolation amplifiers are not suitable for use with these low-level signals.

An automatic power-fail recovery feature has been incorporated in the Reactor Surveillance System, and system checkout is approximately 80% complete. The isolation amplifier/selector switch unit is currently being designed; Burr-Brown model 3456 Isolated Instrumentation Amplifiers are proposed for use in this unit. Completion of the unit and installation of the system at Sequoyah are scheduled before completion of the natural circulation tests.

Procurement of Advanced System. Procurement procedures have begun for the analog signal conditioning equipment and purchase requisitions have been initiated for the communications equipment, processor, and associated digital equipment. An estimate for the processor has been received from DEC, and an order will be placed soon. The present estimate for completion and installation of the new system in Sequoyah is the first quarter of FY 1981.

PROGRAM TITLE: Heavy-Section Steel Technology Program

PROGRAM MANAGER: G. D. Whitman

ACTIVITY NUMBER: ORNL #41 89 55 10 1 (189 #BG119)/NRC #60 19 11 05

TECHNICAL HIGHLIGHTS:

Task 1. Program Administration — Dr. Edward A. Frieman, DOE Assistant Secretary for Energy Research, visited the HSST display area on May 5.

On May 13, Prof. Walter Meyer, Head of Department of Nuclear Engineering, University of Missouri, Columbia, and other members of the department visited ORNL and were briefed on the HSST program.

Dr. K. D. Lathrop, Associate Director for Engineering Sciences, LASL, visited ORNL on May 20 and was briefed on the HSST program.

R. D. Cheverton and G. D. Whitman visited the University of Maryland on May 21 to review work under the subcontract on crack arrest research and development.

On May 27 and 28, P. P. Holz and R. H. Bryan visited the Babcock and Wilcox Company research center in Alliance, Ohio, and their fabrication shops in Barberton, to discuss their work on special weld development for HSST vessel V-8A.

Task 2: Fracture Mechanics and Analysis — A preliminary handwritten draft of a users' manual for NØZ-FLAW, a finite element program for calculating stress intensity factors for nozzle-corner flaws, was received from Georgia Tech. The sample problems described in the manual (a semi-elliptical surface flaw in a cylinder and a slab) were executed successfully by the IBM version of the program by the Computer Sciences Division in Oak Ridge. Only the mesh generation steps could be checked against Georgia Tech results of CDC computations, since complete solutions of the sample problems were not obtained by Georgia Tech. A trial problem has been defined at ORNL for testing the NØZ-FLAW solution against a known theoretical solution for K_I of a semi-elliptical surface crack in a slab.

The latest version of NØZ-FLAW, incorporating the changes necessary for automatic mesh generation for flaws in nozzle corners, has not been sent to ORNL but is expected within a few days. Upon incorporation of the changes in the IBM version of NØZ-FLAW we should be able to check out the full range of potential capabilities promised for this program. Checking of numerical solutions at ORNL has been inhibited by the absence of computer output from corresponding Georgia Tech runs, which may have been discarded.

A series of ADINAT, ADINA, and BIGIF calculations of a pressurized thermal shock in an intermediate test vessel nozzle corner was completed. A refined mesh, discussed in the report for April, was used. The ADINAT finite element heat transfer program was used for calculating the transient temperatures in a moderate thermal shock (similar to that of TSE-1). Stresses for combined pressure and thermal loadings were calculated with ADINA for four times in the transient and for steady state tempetatures. The resultant stress distributions were put into BIGIF calculations of K_I distributions along the nozzle-corner crack front for three crack depths, corresponding to cases studied photoelastically by C. W. Smith at VPI and calculated by S. N. Atluri at Georgia Tech ($a/T = 0.14$, 0.29, and 0.53).

Task 3: Irradiation Effects — Irradiation of the first capsule of the fourth series was continued with excellent temperature control. We expect to reach the target fast neutron fluence of 2×10^{19} neutrons/cm^2 ($E >$ 1 MeV) by the end of August 1980 with the present reactor operating schedule.

Assembly of the second capsule is nearly complete and we expect to start irradiation in June 1980. The thermal shield for this capsule has been installed and neutron dosimeters were irradiated in the dummy capsule to characterize the neutron fluxes in this reactor facility.

Task 4: Thermal Shock — The TSC-2 prolongation was machined (686-mm ID x 991-mm OD) and then tempered at 679 ± 8°C. During the tempering treatment, ten thermocouples were used to obtain temperature profiles in the prolong. This information will help in achieving uniform profiles in TSC-2 when it is tempered.

Following the tempering treatment, residual stress studies were conducted on the TSC-2 prolong by applying strain gages to the inner surface and removing corresponding sections of the near-surface material by sawing. At the completion of these studies two sections were cut from the prolong for further material characterization studies (K_{Ia} and K_{Ic} measurements).

Machining of the TSE-5A test cylinder (TSC-2) was completed with the exception of the three tapered thermocouple-thimble holes. Development of an EDM technique for machining these latter holes has been completed, and preparations for applying 's technique to TSC-2 are under way.

Preparations were ated for further investigations of inner-surface-coating application techniques and performance.

We are examining the through-thickness variation of K_{Icd} using 0.394T compact specimens (C_VCS) from prolongation TSP-1 after receiving the same temper treatment as test cylinder TSC-1 (613°C for 4 h, air-cooled). The tips of the fatigue cracks in the C_VCS are at six depth locations from the inner surface (0.06, 0.17, 0.33, 0.50, 0.67, and 0.83T) of the 152-mm-thick wall. A minimum of six specimens from each depth location are ing tested at −18°C. We will also test six C_VCS at 10 and 38°C and compare the results from previous 1T and 2T CS at the same test temperatures and depth locations to determine the size effect.

Task 5: Simulated Service Tests — The Babcock and Wilcox work on preparation of a special low-upper-shelf toughness seam weld for intermediate test vessel V-8A was initiated in May. In some pre-contract trials, B&W, on the basis of a small number of specimen tests, determined that the prescribed Charpy impact energy and tensile properties were attainable. Preliminary automatic submerged-arc welds with SFA 5.23 type EF-2 electrodes and three different flux combinations (mixtures of Linde 60 and Linde 80) are being made. A portion of each of the three welds will be given about 50 hours of heat treatment at one of three different temperatures. A set of tensile and Charpy V-notch specimens will be tested for each of the nine resulting combinations; the data will be used in selecting the particular process that will be used for the special seam weld in vessel V-8A. All materials have been procured for the trial welds and the first preliminary weldment was made.

Preliminary planning and analysis for pressurized thermal shock testing of intermediate test vessels are continuing. Program tasks are being defined; conceptual arrangements of nozzle-corner test sections are being studied; and transient temperature, stress, and K_I calculations are being made. The ADINAT-ADINA-BIGIF series of calculations discussed under Task 2 was performed to help in defining test facility design parameters.

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: The revised turbine model (ORTURB) has been successfully inserted into the ORTAP-FSV code, replacing the previous turbine model. This version of ORTAP uses the original version of the BLAST code steam generator model. Improvements on BLAST suggested by RWTUV are being incorporated as appropriate to model the dynamic response of the FSV plant. The first draft of the formal documentation of ORTURB (TM report) has been completed.

Code Verification Activities: The comparison of CORTAP calculations with data from the Fort St. Vrain control rod influence tests was completed. CORTAP calculations of reactor power are in good agreement with the plant data for both rod insertions and rod withdrawals.

A request to General Atomic for additional information concerning the FSV oscillation transient of 11/4/78 was prepared. This information is needed for a detailed comparison of BLAST predictions vs. measured plant data for this transient. Additional information concerning measurements of steady state temperature profiles within the highly instrumented steam generator module (module B-2-3) which would be quite useful in verifying BLAST steady state predictions was also requested.

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.

The utility has refused to grant us permission to inspect the Point Beach steam generator during their July outage, and suggested that we contact them later for their November outage. As a result, we are making contact with personnel at the H. B. Robinson and Robert E. Ginna plants to see if we can obtain permission to test our system on their generators. (All three plants are experiencing the deep crevice attack in Westinghouse Model 44 type steam generators.)

We have remeasured a sample of tube 2073 that we obtained from the Point Beach Unit 1 steam generator, using a through-transmission measurement. The tube measured 1.16 to 1.20 mm (0.0457 to 0.0472 in.), with an average value of 1.18 mm (0.0464 in.). The average reading from the previous measurements with a reflection coil was 1.19 mm (0.0467 in.). We will loan a through-transmission instrument to Westinghouse and allow them to test their other tubing samples. Westinghouse has not been making absolute wall thickness measurements on the samples in their laboratory. There is some speculation that the decrease in wall thickness may have been caused by the pulling operation.

We have rewritten the programs TUBRDG and TUBFIT to allow them to contain larger arrays. Both programs will now allow up to 1250 different sets of properties to be measured and fitted. This allows a smoother and better fit to be made to the experimental data. In addition, TUBFIT will fit specified properties from the sets of readings. For instance, the tubesheet inside diameter can be fitted to only those readings taken when the probe was inside the tubesheet, and the defect location can be fitted only to those readings taken while centered on the defect. The same set of calibration readings will be used for all the readings.

A switching power supply has been installed in one of the instruments. This supply has saved about 4.5 kg of mass (10 lb of weight) and the instrument runs about 10°C cooler than with the linear power supply.

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: Three-Module Air/Water Loop - Two turbine meters (one for air-water and one for steam-water) from INEL were delivered by INEL staff. After installation of the turbine meter in the three-module loop, tests were conducted to evaluate the instrument in typical operating conditions. The turbine meter did not prove satisfactory for the application in the air-water system, thus, the INEL staff returned to Idaho with the instrument for more development work. The performance of a cut-out tie-plate drag-body was evaluated in pure air upflow and in pure water downflow over the flow ranges of the system. Also, the separator for the three-module air-water loop arrived and will be installed to increase the capability to handle carryover water existing the hot leg during two-phase flow conditions.

Task 2: Steam/Water Facility - Installation of the UCSP flow measurement instrumentation was completed. Preliminary evaluation indicates erratic operation of the INEL turbine meter.

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 11 05

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration: F. B. K. Kam and F. W. Stallmann attended a LWR-PVS Dosimetry Program Meeting at NBS on May 19-21, 1980. This semiannual meeting summarizes the activities, status, and schedules for the program. F. B. K. Kam, R. E. Maerker, and F. W. Stallmann attended the NRC PCA Blind Test Meeting at NBS on May 22-23, 1980. F. W. Stallmann and R. E. Maerker presented papers at this meeting. The minutes from this meeting will be distributed to all participants.

Task 2: Benchmark Fields -

A. Dosimetry Measurements

All scheduled PCA and PSF dosimetry experiments have been completed for FY-80. Dosimetry experiments scheduled for FY-81 will be noted in the minutes of the LWS-PVS dosimetry program meeting at NBS, May 19-21, 1980.

B. PSF Irradiaton Experiments

1) The irradiation of the SSC-1 and SPVC is proceeding on schedule. The irradiation time for the SSC-1 was determined to be 44 days based on the results of the PSF startup characterization program and calculations. This increase of irradiation time (30-44 days) will mean that the specimens will be available for shipping July 15, 1980 instead of July 1, 1980. Preparations are now in progress for the removal from the reactor and decapsulation of the SSC-1 in the hot cells. Participants in the dosimetry program are urged to ship an approved shipping cask to ORNL at the earliest possible date.

The second surveillance capsule (SSC-2) is scheduled for irradiation in FY-81 (see minutes of LWR-PVS Dosimetry Program Meeting). A dummy surveillance capsule will be inserted to maintain the same neutron flux environment on the SPVC.

The void box capsule will be irradiated for the same length of time specified for the SPVC.

2) Process Control System (PCS) - The ORR-PSF irradiation capsules have been under automatic control in the ORR irradiation field for essentially all of the month of May. During the first fuel cycle of the capsule irradiation, proportional plus integral control of the average temperature was employed. The simpler control method was first used so that manual adjustments of the variacs could be more easily implemented. Considerable time was expended to obtain flat temperature distributions in both capsules while also ensuring that sufficient electrical energy was input to each region so that automatic control could be accomplished. Subsequent startups of the reactor with the PSF experiment in place have employed the optimal integral control law which regulates each variac individually. The optimal control law maintains temperatures within ±5°C, and the method which controls only the average temperature maintained temperatures within ±10°C. Additional improvement in the temperature distribution control would require considerable effort.

Data for incorporating the reactor power into the control law were obtained but have not yet been incorporated. Inclusion of this option into the control algorithm should improve the transient response during rapid changes in reactor power.

Documentation of the control algorithm and associated calculations is in progress.

Task 4: Dosimetry and Damage Correlation - The first phase of the PCA Blind Test comparison has been completed. Calculated over Experimental (C/E) ratios were determined for absolute reaction rates, equivalent fission fluxes, and spectral indices as far as certified experimental data were available. The results were presented at a meeting on May 22-23, 1980 at NBS and were discussed with the participants of the Blind Test. It was noted that one-dimensional calculations tend to overestimate fluxes whereas two-dimensional ones result in flux values which are too low. Part of this problem may be due to leakage corrections and it was suggested to compare some calculations after removal of leakage corrections. Additional information was requested from the participants of the flux-spectrum at the core center and of fluxes >1.0 MeV and >0.1 MeV at additional mesh point in order to determine more accurately traverses and gradients along the centerline. On the experimental side, problems concerning photofission corrections have not yet been resolved, particularly for the $^{238}U(n,f)$ reaction.

A comprehensive document containing all available information about the PCA Blind Test comparison will be published as part of a NRC-NUREG report on Sept. 30, 1980.

The dosimetry measurements from the PSF-PV startup program were evaluated in connection with new transport calculation to finalize the irradiation times for the metallurgical capsules. It was determined that the irradiation time for the three capsules in the PV simulator will be 570 days and for the first SSC capsule 44 days to achieve the same dpa as the capsule at the 1/4 T position in the SPVC.

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:

All programmatic activities during this reporting period were associated with preparations for the B-5 (8×8) bundle burst test which was successfully performed at 1750 hours on May 30. This completes milestone 40445. Completion of this milestone two days ahead of the scheduled completion date, which was established over a year ago, gave the MRBT staff a feeling of great satisfaction.

Test conditions were selected to duplicate those used in the B-3 (4×4) test so that the effect of array size on deformation could be evaluated. Consistent with this objective, initial conditions were established to cause the tubes to burst at ~765°C, using a heating rate of ~10 K/s. Although the bundle has not yet been removed from the test facility for visual examination, quick-look evaluation of the data recorded during the test indicates the bursts occurred in the expected temperature range.

Several hours prior to the test, power was applied to the array for ~9 s to ascertain proper functioning of all the systems and that the performance was as expected. Prior to the short transient, all the simulators were leak tight (less than 10 kPa pressure loss per min at the initial pressure level of 12,200 kPa). Based on quick-look data obtained during the short transient, an adjustment was made to the voltage to increase the ramp rate to the desired rate. During the several hours of waiting for the temperature to equilibrate at the desired initial condition, one of the simulators developed a gross leak. Since this simulator was located in the outer ring of the simulators (one of the guard heater simulators) and the leak was so large that its effect could not be nullified by admitting helium at the rate of loss, a decision was made to conduct the test with this simulator unpressurized. When thermal equilibrium was established, the remaining simulators were pressurized to the desired initial level and isolated from the system to provide a constant gas mass inventory in each during the test. All the simulators (including the non-pressurized one) were heated at the same ramp rate. Since the non-pressurized simulator is in the outer ring near a corner, its effect is not expected to be significant. The first tube burst after approximately 44.0 s of heating and the last pressurized one burst about 5.6 s later. Power was on the bundle approximately 48.1 s and was terminated after 60 tube bursts, about 1.5 s before the 63rd burst. Forty-five of the bursts occurred in the 2-s interval from 45.0 to 47.0 s after power-on.

Plans are underway to remove the bundle from the test facility for detailed visual examination and a posttest photography. Later, detailed hydraulic characterization of the array will be performed in a series of flow tests by an outside research laboratory. These tests are scheduled for completion by January 1, 1981. Following the flow characterization tests, the bundle will be cast in epoxy and sectioned transversely at frequent axial positions to obtain detailed deformation data. These results are expected to be available by October 1, 1981.

Plans are underway for the next bundle test, which will be a 6×6 array (B-4). This test is currently scheduled for about February 1, 1981, and it will be performed at about 800°C, using a ramp rate of 5 K/s. Two of the simulators will be unpowered to investigate the effect of cold rods in a highly deformed bundle.

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:

Loose-Part Detection Systems. The final draft of a comprehensive summary report on the loose parts monitoring work was completed.

Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. Modules for collapsing multi-group kinetic data were completed and tested. Two-dimensional calculations to determine the sensitivity for detecting boiling in a PWR are being run using JPRKINETICS and preliminary results are being evaluated.

Time Series Analysis Methods. A letter report is being prepared that summarizes our evaluation of time series analysis methods.

PWR Baseline Signature Acquisition. Neutron and process signals from a B&W plant and neutron signals from a CE plant were analyzed over three frequency bands: 0-0.3HZ; 0-1.3HZ and 0-31.3HZ. The results show that the variation between the power spectral density (PSD) signatures of different neutron detector signals in the CE unit is greater than for the neutron detectors in the B&W unit. We've also observed that the neutron noise signatures from this CE plant have a significantly (a factor of 10) lower amplitude than signatures we obtained at two other CE plants last year. This result could alter our previous conclusion that generic neutron noise signatures can be developed for PWRs.

We've also observed a statistically significant spectral structure at low frequencies (0.005-1HZ) in both neutron and process signals. From this, we conclude that there may be useful diagnostic information at frequencies lower than 0.1HZ (the lower limit of our previous baseline signature analysis).

We have also computed the amplitude probability distribution (APD) of the neutron and process noise signals. The results indicate that the signal noise amplitudes have a normal distribution, i.e. no significant skewness in the APDs.

Analysis of recorded baseline data is continuing in an attempt to identify the sources of neutron noise and the relationships between neutron and process noise signals. A letter report summarizing the results is being drafted.

BWR Stability Monitoring. The feasibility of using neutron noise to monitor for changes in BWR stability was studied by comparing predictions of a BWR dynamic model with the results of noise analysis of neutron signals from Peach Bottom 2 (data supplied by EPRI).

Although a direct comparison of model and noise analysis results is not possible at this time (model input data corresponds to end of fuel cycle 2 whereas the noise data was acquired at the end of cycle 3), the results are encouraging in that the model and noise analysis estimates of decay ratio agree to within $\sim$ 15%. New model estimates will be obtained when cycle 3 core operating data corresponding to the noise measurements is obtained from EPRI.

However, the agreement between noise analysis and model predictions suggests that an on-line stability monitor can be implemented in BWRs without installing new sensors or perturbing normal operation.

Meetings. A trip report (ORNL/FTR-843) summarizing our attendance at the 13th Informal Meeting on Reactor Noise in Cadarach, France was issued. Topics pertinent to ORNL research included: vibration and boiling modeling, core internals vibration measurement, temperature noise modeling, and time series modeling. The trip provided valuable information which directly relates to the achievement of our research milestones.

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:

No technical highlights this month.

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 May, the staff of the Nuclear Safety Information Center (a) processed 845 documents, (b) responded to 72 inquiries (of which 45 involved the technical staff and 6 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 162 times between April 1 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 4 visitors and participated in 2 meetings.

Two NSIC reports are in reproduction: *Annotated Bibliography on the Transportation and Handling of Radioactive Materials* (ORNL/NUREG/NSIC-168) and *Index to Nuclear Safety, Vol. 11 through Vol. 20* (ORNL/NUREG/NSIC-175). 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); and *Nuclear Power Plant Operating Experience — 1979 Annual Report* (ORNL/NUREG/NSIC-180).

During the month of May, we received 27 foreign documents (18 French and 9 German). 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 April 30, 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 May we added 4 exempt users which, with other withdrawals and renewals, leaves the SDI service at a total of 398 users.

All technical articles for *Nuclear Safety* 21(5) were completed and mailed to NRC, DOE and TIC on May 23rd. The "current events" material (covering events which occurred during March and April) for *Nuclear Safety* 21(4) were completed by May 16th (except for the data on operating power reactors which was not yet available from NRC). Most technical articles for *Nuclear Safety* 21(6) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 21(3) were received from the printer (via TIC) on May 23rd.

There has been no further information from DOE regarding TIC's continued participation in the publication of *Nuclear Safety*. We have, however, been advised by J. G. Coyne of TIC (by letter dated May 9th) that the dedicated RECON terminals (such as the one now owned by NSIC) will have to be replaced with Telenet service and ASCII/CRT terminals. No date has been set for this change.

TABLE 1 RECON DATA BASE ACTIVITY FROM 04-01-80 TO 04-30-80
(21 OPERATING DAYS)

DATE BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	CITATIONS PRINTED
EDB	(TIC) DOE ENERGY DATABASE	3355	5020	112560
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	607	869	8785
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	292	854	18991
EMI	(EMIC) ENV. MUTAGENS INFO.	198	390	7285
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	162	294	6944
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	156	393	3241
GAP	(DOE) GENERAL AND PRACTICAL INFO.	124	163	17500
ESI	(EIC) ENV. SCIENCE INDEX	117	238	622
EIA	(EIC) ENERGY INFO. ABSTRACTS	103	150	845
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	103	136	906
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	90	51	7186
NES	(NESC) NATIONAL ENERGY SOFTWARE	59	135	1125
IPS	ISSUES AND POLICY SUMMARIES (TIC)	43	69	8
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	42	33	16
API	(API) AMER. PETROLEUM DATA BASE	36	58	1554
WRE	(WRSIC) WATER RESOURCE RESEARCH	36	55	82
SLP	(FRANKLIN) SOLAR DATA BASES	33	39	89
NRC	(LC) NATIONAL REFERRAL CENTER	31	51	74
CIM	(DOE) CENTRAL INVENTORY OF MODELS	30	38	64
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	29	48	718
TUL	(U. TULSA) TULSA DATA BASE	29	61	147
NBI	(NBIC) NATL BIOMONITORING INV.	25	42	-
PRD	(TIC/NRC) POWER REACTOR DOCKETS	23	18	138
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	20	30	165
NER	(EIC) NATIONAL ENERGY REFERRAL	18	41	186
RSI	(RSIC) RADIATION SHIELDING INFO.	17	10	880
RSC	(RSIC) RADIATION SHIELDING CODES	10	14	35
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	7	7	-

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 - No single-rod tests were performed during this month.

Task 2: Analysis - Thermal-Hydraulic Analysis. COBRA/TRAC pretest calculation of test 3.03.6AR was initiated. A steady-state calculation was performed and the transient was initiated from a restart dump. The transient calculation proceeded to 2.45 sec before failing. Discussion with PNL is underway to remedy the problem. This work is approximately 60% complete.

Pretest planning for a downflow film boiling test is continuing and approximately 90% complete.

Work on the test 3.03.6AR Quick Look Report and Preliminary Local Fluid Conditions Report has begun and is 10% complete.

Work on the "correction comparison" code is approximately 50% complete. Work will continue through the next month.

Work on analysis of the Bundle Uncovery Test Series (3.02.10A-H) is continuing and is approximately 65% complete. In May, work centered on the separation of the convective and radiative components of the FRS surface heat flux. This will allow for a direct comparison between existing convective heat transfer correlations and Bundle Uncovery Test Series data.

Nuclear Pin Simulation Analysis. Preliminary plans for conducting the pretest and posttest analyses associated with the upcoming double-ended blowdown test (3.05.5B) have been completed. The pretest hydraulics analysis to determine appropriate THTF rupture disk orifice sizes is continuing on schedule. PINSIM problem models for determining an appropriate electric pin power program have been developed and tested.

The expansion of the FLIP (Fluids Interface for PINSIM) code which allows access to ORINC-calculated parameters has been completed and verified. Plans for beginning the posttest PINSIM analysis of THTF Test Series 4 are being developed.

Electric Pin Analysis. Debugging and verification of a preprocessing program for ORINC and ORMDIN has been completed. This program basically restructures and combines the coefficient data tape and engineering units tape for a given THTF test into a single one-pass tape input for ORINC and ORMDIN. With the preprocessed tape, ORINC and ORMDIN will be limited to computational type work.

ORINC has been modified for THTF bundle 3 tests. Debugging is continuing with the aid of a preprocessor tape of a simulated THTF power drop.

The final segment of the ORTCAL calibration package for THTF bundle 3 has been coded and programmed. Debugging has started.

THTF test 3.03.6AR (upflow film boiling test) has been processed by parts 1 and 2 of ORTCAL.

Data Management. Data from two hot test section fill tests, which were run during this month, have been reduced and used to verify the locations of thermocouples in the THTF test section.

Data reduction has begun on the THTF upflow film boiling test. An uncalibrated engineering units tape has been generated and is being used in the analysis of this test.

Task 3: THTF Operations - Refurbishment of bundle 3 was completed this month. This included the removal of each FRS from the bundle and checking for and repairing ground faults if found. The faults were caused by moisture contamination due to pin holes in the sheath where the "O" ring support sleeves were welded into place. The "O" ring support was redesigned and each FRS was modified with the new design and replaced into bundle 3.

Also completed this month was the installation of additional differential pressure gages in the test section. Ten penetrations in the test section barrel and shroud box were made. These were spaced on the north side along the entire heated length of the test section.

The bundle was operated on three occasions this month. The first was a checkout of the facility which climaxed with a 90 kW/FRS power drop. The second was an attempt to conduct the upflow film boiling test (3.03.6A). The system operated satisfactorily through the above except during the blowdown. Power was tripped prematurely by a generator voltage safety circuit. This was set off by vibrations in the facility during the blowdown. The third operation was the successful completion of the upflow film boiling test. Again the facility operated satisfactorily.

Task 4: Two-Phase Instrument Development. Final design of the in-bundle densitometer system is near completion. Procurement and fabrication of the source and electronics is expected on schedule. Procurement of the ion chambers is slightly behind schedule but should not impact the schedule for initial testing of the system.

The pretest mass flow uncertainty analysis methodology using the RELAP4 thermal hydraulics code prediction and the mass flow code AMICON has been performed for the upflow film boiling test (3.02.6A).

Design has begun on a new thermocouple array rod for in-bundle fluid temperature measurements and also on modification of the test section barrel and shroud box to provide thermocouples for shroud wall temperature and energy storage measurements. Both measurement improvements are intended for the upcoming 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:

Simulator experiments being performed by General Physics Corporation at the TVA Training Center continued. A total of five groups of operators have now performed the exercises in the five modules comprising the experiment. Software to extract event-specific information from the Performance Monitoring System (PMS) raw data tapes has been completed for several events.

Memphis State University Center for Nuclear Studies has continued collection and analysis of field data. One additional plant was visited this month. At each site plant logs and available computer output related to specific events of interest is copied for analysis by CNS staff. Field-data collection also includes completion of written questionnaires by plant operators regarding abnormal events they have experienced. This information will supplement data obtained from site records and from simulator studies.

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

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

TECHNICAL HIGHLIGHTS:

HOBBIE-8, the final in-reactor creepdown test in the joint NRC/ECN Petten test series, has completed the creepdown phase of the test. Pressure was shifted successfully from the external to the internal pressurization mode and the cladding specimen is now creeping out. The test will terminate at the end of the present reactor cycle on June 30, 1980.

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