INTERIM REPORT

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for the
DEPARTMENT OF ENERGY

NRC Research and Technical
Assistance Report

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INTERIM REPORT

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ABSTRACT

Highlights of technical progress during June 1979 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.

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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 #B041.,/NRC #60 19 10 01

PKL String Probe testing in air-water has continued in an attempt to correct the previously reported problem of poor coherence at high void fractions. Initial testing was conducted with the probe installed in a 6 inch ID pipe in which the probe body blocked a significant portion of the flow area. It was suspected that this may have contributed to the loss of coherence by introducing additional turbulence. The tests were repeated with the probe installed in the larger IDL (UPTF Instrument Development Loop) test facility however the same results were observed.

In view of the above problem, the following actions are being taken in the string probe development effort. A change in the PKL delivery date (from June to September 79) has been requested. A new probe design which eliminates the electrode support spools and increases the open area will be fabricated and tested. Steam-water testing of the current design has been postponed until the new design is completed. The current design is probably satisfactory for making void fraction measurements. Improved coherence for velocity measurement is desirable but is not a strict requirement for the string probes. The effort to improve the design for PKL is warranted because it will not only provide better velocity meausurements but will greatly reduce the flow blockage area of the string probe. If the new design cannot be completed and tested in time for use in PKL, then the existing probes can be used.

Testing of PKL In-core Flag Probes in steam-water has been completed. A large amount of data has been collected with test conditions covering ranges expected in PKL. This data will serve as a basis for the further development of analysis methods which can be used in reducing data from actual reflood tests. The testing was conducted on a bundle containing 3 probes in the AIRS loop at ORNL and a bundle containing one probe in the 3x3 KWU facility in Germany.

PKL Upper Plenum Flag and Band Probes built into 1 1/2" pipes are being tested in the air-water loop. These are low temperature prototypes whose purpose is to optimize electrode configurations prior to fabrication of a high temperature prototype which will be tested in the steam-water loop.

Testing of PKL Film Probes in steam-water will begin the first week in July. Two film thickness sensors have been completed and will be installed in the steam chamber for preliminary testing. These sensors are of the Lehigh (D-probe) configuration with platinum electrodes installed by chemical deposition. Initial tests have shown that this type of electrode will not be satisfactory for use in PKL because of its susceptibility to severe corrosion when excited with a DC voltage. As a result the PKL sensor electrodes will have to be fabricated by machining which is more difficult than the chemical deposition process. It will require approximately one month to fabricate a complete PKL module containing "D" probes and Electrolysis Potential (EP) probes with machined electrodes. Meanwhile preliminary steam testing of the existing probes will be conducted until corrosion renders them inoperable.

Meetings and Conferences attended by AIRS personnel during June included a Slab Core Design meeting at Tokai, Japan on June 11 and 12 and the 2D/3D Program meeting at Munich during the week of June 25.

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:

Two additional underwater shakedown tests were performed in the FAST facility (FAST 20 and 21). FAST 20 had the highest CDV energy input - 53 kJ - attained to date in underwater tests. FAST 21 was the first test in which xenon gas was included in the test sample (xenon is used to simulate fission product gases). For this test, the CDV energy input was high - 34 kJ - but the steel sample containment tube did not rupture efficiently.

At the end of the month, Kaman pressure transducers were installed in the FAST vessel to permit us to begin the planned FAST underwater test series next month.

In CRI-III, work continued to evaluate the useability of Kaman pressure transducers in FAST underwater tests. CDV 81 was an underwater test in which the sensing element of the transducer was shielded from the pressure pulse produced. Results were encouraging in that there was no evidence of a signal generated at the time of arcing (sample breakup). Tests were also performed in argon and underwater using a "dummy" fuel assembly. These tests showed that the transducer is not sensitive to the magnetic field produced by capacitor discharge. It does seem sensitive to the initiation of capacitor discharge (Ignitron firing), but the signal produced is high frequency (MHz) and appropriate filtering reduces it to acceptable levels.

NSPP:

Analytical data were received from Uranium Oxide Aerosol Exp. No. 205, the second experiment performed with the Plasma Torch Aerosol Generator. The maximum aerosol concentration, at the termination of

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aerosol generation, was 4 gms U_3O_8/m^3. The aerosol concentration achieved during this experiment was the largest using U_3O_8 thus far produced in the NSPP vessel.

Uranium Oxide Aerosol Exp. No. 206 was performed near the end of this reporting period. The uranium powder feeder was charged with 1 kg of powder, and an estimated 80% was injected into the plasma torch assembly. Although complete analysis of the aerosol samplers will not be accomplished until next month, the small amount of data in hand suggests that a maximum aerosol concentration in the 8-10 gms/m^3 range was achieved.

Preparations are underway for Mixed Oxide Aerosol Exp. No. 303. The aerosol will be produced by simultaneous generation of U_3O_8 (with the Plasma Torch Generator) and Na_2O_x (by a sodium pool fire). Target concentrations are 10 gms/m^3, each, of U_3O_8 aerosol and Na_2O_x aerosol.

CRI-II:

Three exploratory experiments in the generation and characterization of sodium oxide aerosols have been conducted in the CRI-II facility by spraying liquid metallic sodium at 500°C into dry air. Aerosol yields have been relatively high; perhaps up to 65 percent of the sodium sprayed is airborne. Initial concentrations of aerosol of 1, 10, and 50 μg/cc were measured and particle sizes ranging from 2 to 4 μm AMMD were determined by means of the cascade impactors and the sprial centrifuge. X-ray diffraction indicated that the chemical form of the yellow solid is predominantly $Na_2O_2 \cdot 2H_2O$. Characterization of mixed aerosols resulting from the simultaneous generation of both U_3O_8 and Na_2O_2 and their coagglomeration behavior in support of NSPP experiments is of major interest in this program and will be initiated next month at an intermediate level (10 μg/cc of each oxide).

Coagglomeration will be established primarily through the application of the impactors and the spiral centrifuge. Each increment of size distribution will be analyzed for the exact ratio of uranium to sodium over the entire deposition cycle.

Electron microscopy will also be applied to establish the composition of the chain agglomerates and the variability of the identity of adjacent primary particles. Any systematic composition of the agglomerates should be detected in this way.

ANALYTICAL:

The principal requirement for successful finite-difference modeling of the temperature variation at the bubble-liquid interface appear to have been established. Some additional comparisons with analytical results are being made; it should be possible to incorporate the method into FAST bubble rise calculations.

Data reduction has been performed for NSPP Tests 201-203 and 301-302. Figures for these tests will be presented in a data record report.

A paper on NSPP analysis was presented at the 1979 ANS Annual Meeting in Atlanta (June 3-7, 1979).

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 is currently in reproduction.

Task 2: Stress Indices and Flexibility Factors for RPV Nozzles and Small Branch Connections - The final report for this task (ORNL/Sub-2913/10) *Stress Indices and Flexibility Factors for Nozzles in Pressure Vessels and Piping* by E. C. Rodabaugh and S. E. Moore has been published and is ready for distribution. The report contains a complete discussion of stress indices for internal pressure loading, and stress indices and flexibility factors for moment loadings on Class 1 nuclear pressure vessel nozzles and reinforced branch connections in Class 1 nuclear piping. The study shows that the present Code values for maximum stress indices are nonconservative over a significant range of geometric design parameters permitted by the Code rules. New indices and flexibility factors, and the proposed Code rules revisions to implement their use are presented in the report. These new rules have been presented to the ASME Code Committee, and have been accepted at the Working Group level.

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 — All of the planned parameter studies have been completed. Results of the studies show that the current Code design rules permit nozzle spacings that are too close for nozzle-vessel designs which meet the minimum nozzle reinforcement-vessel wall thickness requirements of the Code. On the other hand, the study also shows that the current rules are overly conservative for certain nozzle-vessel designs that exceed the minimum requirements. Proposed rules revisions are currently being drafted that will permit a more balanced design approach.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: A. P. Malinauskas

ACTIVITY NUMBER: ORNL #41 89 55 10 9 (189 #B0127)/NRC #60 19 10 14 1

TECHNICAL HIGHLIGHTS

Test BWR-3 was performed by heating a segment of fuel rod from Peach Bottom-2 reactor to 1200°C in flowing steam for a total effective time of about 25 min. This same segment was pressure-ruptured in test BWR-2 at 850°C, so that the diffusional release which occurred in test BWR-3 was characteristic of release from the type of rupture which would be experienced in a loss-of-coolant accident (LOCA).

The amount of cesium released in test BWR-3 was 1.85% of the total segment inventory. This quantity is nearly identical to that observed to be released with rupture at 850°C in test BWR-2.

Preparations are being made for test BWR-4, in which the Peach Bottom-2 fuel rod gap inventory will be determined by purging the gap space with purified helium while the test segment is heated over the temperature range 700 - 1100°C.

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 10 5

TECHNICAL HIGHLIGHTS

Task 1: Program Administration — On June 11, G. D. Whitman summarized calculations of light-water-reactor primary vessel integrity under overload conditions for Thomas Pigford, member of the President's Commission on the Three-Mile Island Accident.

Dr. Sayta Atluri from Georgia Institute of Technology visited ORNL on June 12, to review the status of his work on flawed nozzle corner analysis.

Messrs. G. Tomassetti and P. Milella from CNEN, Rome, and E. Brutto and G. Prossa from CISE, Milan, visited ORNL on June 25 and 26, to review selected HSST program projects prior to their implementing similar structural tests.

R. H. Bryan presented a paper at the ASME Pressure Vessels and Piping Conference in San Francisco, held June 25—29.

Task 2: Fracture Mechanics and Analyses — Stress intensity factor distributions for four different nozzle corner flaws (a/T = 0.81, 0.53, 0.29, and 0.14) in a BWR geometry were compared using results from NOZ-FLAW, BIGIF (CORTES-EP stress distribution), BIGIF (TMR stress distribution), and photoelastic studies. The two sets of BIGIF calculations were very nearly identical, and subsequent references to the BIGIF results will imply BIGIF using the CORTES-EP stress distribution. Good agreement was obtained between NOZ-FLAW and BIGIF for all but the deepest flaw geometry (a/T = 0.81). Less satisfactory agreement was obtained between NOZ-FLAW and the photoelastic results. Attempts are presently being made at understanding and explaining the discrepancies between the two.

A review of spent fuel shipping cask designs for the prevention of fracture was completed, and a summary letter was sent to the Transportation Branch, Division of Nuclear Materials Safety and Standards, NRC.

Task 4: Irradiation Effects — The capsule design for the fourth HSST irradiation experiment was completed, and the fabrication of capsule parts is in progress. The electrical supply system design changes are on

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schedule with field work to start soon. Required changes in instrumentation wiring design are in progress and on schedule. Purchase of components is on schedule with the delivery of heaters the only tight schedule item. Neutron dosimeter preparation is ahead of schedule.

The 1T CT specimens are presently being fatigue precracked, and we do not anticipate a schedule problem with this item.

HEDL has initiated testing of selected 0.5T CT specimens of the 61W weldment from the second 4T series irradiation.

Task 5: Simulated Service Tests — A representative of TREMIX, Wallingford, PA, utilizing special ultrasonic techniques, investigated the residual stresses in the remaining prolongation from the V-8 in-service weldment. The results indicate the method is not sufficiently accurate for good comparisons with the hole-drilling technique; however, the differences in the stresses from one region to another were in general agreement with previously obtained data when extreme care was taken in positioning and coupling the transducers to the specimen.

Task 6: Thermal Shock — A preliminary three-dimensional fracture mechanics analysis of the actual flaw shapes in TSE-2 was completed. A primary purpose of the study was to determine whether or not warm prestressing was effective in limiting growth of the initially semicircular flaw. Results of the analysis indicate that conditions for warm prestressing existed; however, the calculated nominal values of $(K_I/K_{Ic})_{max}$ for the final crack shape were so close to unity that actual ratios less than unity could have been responsible for the absence of reinitiation.

Analysis of data from the second thermal-hydraulic thermal shock test (LN_2-TSE-TH-2) shows that quenching was very uniform in the area of the lower four thermocouple thimbles, but that surface temperature in the area of the uppermost thimble lagged behind the others by as much as 45 K (80°F) during the transient. This asymmetry is a result of an excessive vapor fraction near the discharge end which is a consequence of an excessive heat removal rate. In a preceding thermal-hydraulic experiment, which more nearly corresponds to the requirements for TSE-5, the heat removal rate was less, and the axial symmetry in quenching was satisfactory. The heat removal rate and thus degree of quenching symmetry is controlled by the thickness of the rubber cement coating. The coating thickness for the most recent experiment was essentially optimum (maximum heat removal rate), while that for the earlier experiment was purposely made greater than optimum.

Both TSC-1 (TSE-5 test cylinder) and its prolongations were tempered at 613°C (1135°F) for four hours. Following this operation, a full-length EB weld was made in the prolong and then hydrogen charged to produce a crack. This flaw will be removed and inspected for conformity prior to making a similar flaw in TSC-1.

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 ORECA code for the Fort St. Vrain (FSV) reactor: Investigations were made of the ORECA calculations of the postulated worst-case 90-min loss-of-forced-convection (LOFC) accident followed by firewater cooldown (FWCD). Comparisons with previous FLODIS predictions had shown large differences both in maximum reverse-flow temperatures during the LOFC (ORECA was higher) and the maximum forward-flow temperatures during FWCD (ORECA was lower). Two changes made in the ORECA calculation reduced the differences significantly. First, the assumed total full-flow core pressure drop was reduced from 10 psi (per FSAR) to 6.24 psi by assuming wider-open refueling region orifices. This is more realistic in view of the current operating procedures, which attempt to minimize the core flow resistance to inhibit oscillations. Reducing the orifice resistances makes the flow redistributions more sensitive to coolant channel friction losses. The second ORECA change was to improve the accuracy of the region gas viscosity calculations, which had previously been based on two-point averages of the region gas temperatures. The ORECA calculation of the new maximum reverse-flow temperature (at 90 min) is 1671°F (vs. 1724°F previously), and the new maximum forward-flow temperature (at 180 min) is 2263°F (vs. 2216°F previously). Maximum fuel and lower plenum hot streak temperatures are both still well below damage limits.

Investigation of the FSV Oscillation Problem: A letter was written to NRC summarizing our confirmatory calculations of GA's "Jaws" model for explaining FSV oscillations. ORECA calculations of refueling region flow transients during the Nov. 4, 1978 oscillation event indicated that the jaws model may not be adequate for explaining the large changes in observed region outlet temperatures. ORECA-derived flow variations were as large as ~±75%, compared with GA's calculation of 38% as a maximum change due to jawing.

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FSV Upper Plenum Reverse-Flow plume experiments: The conceptual design of the intermediate plume heat-transfer experiment was completed, and the required components are being acquired.

Development of the ORTAP code for FSV: The present FORTRAN version of ORTAP at the ORNL computer site was copied to tape and transferred to the ORGDP computer site. Computer calculations of a sample problem run at both sites using identical coding and input data showed significant differences. The reasons for the differences, probably undefined variables, are presently being investigated. A new computer model of the high pressure turbine is under development.

PROGRAM TITLE: Improved Ed Current In Service Inspection for Steam Genera.or Tubing

PROGRAM MANAGER: Robert W. McClung

ACTIVITY NUMBER: ORNL #41 89 55 12 1 (189 #B0417-8)/NRC #30 19 10 05

TECHNICAL HIGHLIGHTS

We are continui ; our task to impiove the detection of defects in the presence of other property variations.

We h e constructed a new a/d converter module that has eight separate integ.ating a/d converter chips. This allows a conversion to be started by a single pulse, and then the computer can be processing and s ing previous data until the new readings are ready. The readings are the result of integrating the signals over a 10 msec time interval, which should result in a much lower noise level, even at a higher inspection rate. The module is now being tested. Additional laboratory samples of tubing containing flaws are being obtained.

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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 — Fabrication of the end box baffles and control rod guide tube internals for the air-water flow visualization loop is 90% complete. Fabrication of the low energy gamma densitometer for that loop was completed. Installation will be made concurrently with the baffles. Two turbine meters were installed in the upper plenum; the first, a 3/8-in.-diam bidirectional meter, was located just above the tie plate in the end box. The second turbine meter, a 1-in.-diam unidirectional meter, was located just above the round hole in the upper core support plate.

The AIRS program conducted tests of the current version of the PKL-type string probe in the IDL air/water loop. Although the results will be reported in detail elsewhere, it is of interest that, when the probe was located ~7 in. vertically above the upper core support plate and near the center line of the center module, the phase plot indicated an upward velocity of 5 ft/sec even when there was no net carryover.

Task 2: Construction — Fabrication of the steam/water loop pressure vessel was started. Procurement of components for the loop was continued.

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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 - Table 1 shows the total specimen receipts at the end of the current reporting period. It is not expected that all of the specimens will be at ORNL before Aug. 1, 1979.

Task 2: Benchmark Fields -

A. LWR Dosimetry Pressure Vessel Benchmark Field (PCA) -

1) The support fixture for the NBS run-to-run monitor was installed. Core rearrangements and other preparations of the PCA were completed and operation was begun for the experiment program on June 6, 1979.

Unirradiated fuel elements were brought from Y-12 to reload a core with low gamma background, for the spectrometry measurements.

A leak, apparently at the flanged gasket, was discovered in the tube in the void box. Repair will be made before the facility is needed.

B. LWR Metallurgical Pressure Vessel Benchmark Field (PSF) -

1) PSF Support Structure

a) Detailing, checking, and revising engineering drawings of components of the ORR pressure vessel simulator continue. A rig to measure the as-installed position of the simulator at the ORR poolside has been designed.

b) Installation of the simulator and verification of positions relative to the reactor should be completed by Sept. 15, 1979, the scheduled date.

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2) Process Control System (PCS)

a) The computer, memory, hard disk, and terminal for the computer system have been installed and partially tested. Memory exercises $XM and processor tests and exercises $XP execute satisfactorily. RT-11 operating system software, which will be used only for initial testing, is also running. The final operating system will be RSX-11S. The analogue and digital subsystems, scheduled for delivery June 22, have not yet arrived. The delay may create problems in maintaining the progress schedule. Work continues on the system task software. Emphasis is being placed upon prompt documentation of completed tasks.

b) The machine tool work necessary to modify the temperature characteristic simulation experiment is nearly completed. The modified experiment will be run in July.

3) Instrumented Irradiation Capsule (IIC)

a) A prototype cooler-plate manifold was assembled, welded, and brazed. Some modifications in the assembly will be made.

b) The fabrication of the IIC and two surveillance specimen capsules were ordered from Continental Tool and Engineering, Inc., Oak Ridge, Tenn.

c) More testing has been done on the test heater and cooler plate assemblies. Performance of the front gas gap is being evaluated. A three-dimensional heat transfer model is needed to obtain more accurate information from the tests. However, no major problem was encountered, other than the requirement for an additional heater region.

Task 3: Neutron Field Characterization - Several preliminary calculations evaluating the importance of the accuracy of the fixed source specification were completed. Results from an eigenvalue calculation compare well with results from a fixed source calculation using a cross section library with the χ_g's (fission spectrum) zeroed. Differences exist only in the very low energy groups.

Additional calculations for several fixed source distributions will be performed, in order to further establish the required accuracy of the fixed source distribution specification.

Calculations for a Heavy Section Steel Technology experiment in the Bulk Shielding Reactor are also in progress. Cross section preparation is nearly completed, but neutronics calculations have not begun. A fixed source will be calculated using diffusion theory, while the flux in the experiment will be obtained from a transport theory calculation using the fixed source.

Task 4: Dosimetry and Damage Correlation - Perturbation measurements in the PCA 4/9 configuration, made with miniature fission chambers, have been analyzed.

Fluxes and spectral indices for a variety of thermal shield-pressure vessel simulator configurations have been calculated. These calculations, along with recent results from damage analyses in power reactor surveillance capsules, may dictate modifications in the configuration and schedule of the PSF metallurgical irradiation experiment in order to obtain the maximum amount of information.

Table 1. Total Specimen Receipts at End of Reporting Month.

SOURCE	CHARPYS	TENSILES	THERMAL SLEEVES	IT CT	1/2T CT	SMALL SPECIMENS	DOSIMETRY
Rolls Royce, Ltd.	76--3/2/79	14--3/2/79	12	NOT APPLICABLE		22 6/79	All-5/79
EPRI	72--3/2/79	12--3/2/79	All 6/79	NOT APPLICABLE			
MOL/Cockerill	84--4/5/79 16--5/29/79	*	*	NOT APPLICABLE		50 CC** 5/29/79	5/29/79
KFA	*	*	*	NOT APPLICABLE			6/14/79
FCC	*	NOT APPLICABLE					
HEDL	NOT APPLICABLE					*	*
NRC/NRL	178--4/3/79	38--4/3/79	6/5/79	40--5/7/79	80--5/7/79	NOT APPLICABLE	
CE	NOT APPLICABLE						6--6/8/79
GE	NOT APPLICABLE						6--6/8/79
B & W	NOT APPLICABLE						5--6/26/79
Westinghouse	NOT APPLICABLE						***

*Scheduled, not received.
**Incomplete shipment.
***Dosimetry to be performed by HEDL

Small specimens consist of compression cylinders, hardness disks, TEM disks, and damage monitors.

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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

Preparation of the fuel simulators for the B-5 (8 X 8) bundle continued. The grooving operation is completed on 70 of 75 simulators, while coating (ZrO_2) and infrared scanning continue to maintain a sufficient stock of acceptable simulators for the assembly of the fuel pin simulators for B-5. About six fuel pin simulators have been assembled thus far for B-5. An additional six were assembled for use in single rod tests, which presently have priority status.

Thermocouples continue to be a problem due to late delivery by the vendor. In-house fabrication of thermocouples from basic sheath material has been initiated to maintain the assembly of fuel pin simulators.

Three additional single rod tests were performed this month utilizing the heated shroud. The shroud temperature followed rod temperature satisfactorily in all but one test which experienced controller problems. Of several shakedown problems concerning the new single rod flange assembly, the most serious is that of the split-tube shroud which sometimes opens during or after the burst. The cause and solution of this problem will be investigated before additional burst tests are performed.

On June 21, E. D. Hindle of the UKAEA Springfields Fuel Laboratory visited ORNL for discussions on the MRBT program.

For the week of June 25, R. H. Chapman and D. O. Hobson participated in the Fourth Japanese-German-American Fuel Modeling Workshop held in Idaho Falls. An informal presentation was made of the latest B-3 analysis, B-5 design criteria, and some preliminary results of single rod tests with heated shroud.

PROGRAM TITLE: Noise Diagnostics for Safety Assessment

PROGRAM MANAGER: R. S. Booth

ACTIVITY NUMBER: #41 89 55 11 4 (189 #B0191)/NRC #60 19 10 01

TECHNICAL HIGHLIGHTS

There is no report this month.

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

PROGRAM MANAGER: Betty F. Maskewitz

ACTIVITY NUMBER: ORNL #40 89 55 11 9 (189 #B0402)/NRC 60 19 10 02 2

TECHNICAL HIGHLIGHTS

A paper, coauthored by the RSRDR program manager and Stanley F. Bankert of EG&G Idaho, was prepared for the proceedings of SMIRT-5/CAFEM-5 to be held in Berlin, FRG August 13 - 22, 1979.

Data received from the NRC/RSR Data Bank Processing Group at INEL in late June is currently in the RSRDR processing system and will be made available in transportable packages in the near future. Included are measurement data and computed parameter data for the following tests:

Semiscale S-07-2 Short
Semiscale S-07-3 Short
Semiscale S-07-6 Short
Semiscale S-07-6 Long
Semiscale S-07-2 Long
Semiscale S-07-3 Long
TFB Test PCM01
TFB Test PCM05
Marviken Test 03-06 Average Data
Marviken Test 03-11 Average Data
Marviken Test 03-08 Average Data
Marviken Test 03-10 Average Data

The results of the Marviken tests are processed as "protected" data, and restricted in distribution due to international agreements. All requests for these data will be cleared through NRC/RSR. Distribution will be made only upon their written authorization to RSRDR.

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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 June, the staff of the Nuclear Safety Information Center (a) processed 733 documents, (b) responded to 65 inquiries (of which 41 involved the technical staff and 3 were for commercial users), and (c) made 12 computer searches. The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 227 times during the previous month making it the fourth most utilized of the 25 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 9 visitors and participated in 1 meeting. Joel Buchanan, Assistant Director of NSIC, is still on temporary assignment to EPRI in connection with their Three Mile Island assessment.

Two NSIC reports were issued during June: *Bibliography on Common Cause — Common Mode Failures* (ORNL/NUREG/NSIC-148) and *Annotated Bibliography on Analytical Techniques for Stress Analysis of the Nuclear Steam Supply System* (ORNL/NUREG/NSIC-157). Three other NSIC reports are in reproduction, including *Index to Nuclear Safety, A Technical Progress Review, by Chronology, Permuted Title, and Author, Vol. 11, No. 1 Through Vol. 19, No. 6* (ORNL/NUREG/NSIC-162); *Bibliography of Microfiched Foreign Reports Distributed Under the NRC Reactor Safety Research Technical Exchange Program 1978* (ORNL/NUREG/NSIC-163); and *Summary and Bibliography of Operating Experience with Valves in Light Water Reactor Nuclear Power Plants* (ORNL/NUREG/NSIC-171). 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); *Annotated Bibliography of Licensee Event Reports in Boiling-Water Nuclear Power Plants as Reported in 1978* (ORNL/NUREG/NSIC-164); *Annotated Bibliography of Licensee Event*

Reports in Pressurized-Water Nuclear Power Plants as Reported in 1978 (ORNL/NUREG/NSIC-165); *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); and *Annotated Bibliography on Fires and Fire Protection in Nuclear Facilities* (ORNL/NUREG/NSIC-172).

During the month of June we received 57 foreign documents (19 German and 38 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 German language documents were reviewed for translation (see letter of June 22, 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 June we added 1 exempt and 1 paid subscriptions bringing the SDI service to a total of 396 users.

The regular *Nuclear Safety* staff meeting was held on May 30th. Minutes of that meeting and a tentative outline for the next several issues of *Nuclear Safety* were distributed early in June. TIC sent the final galleys of *Nuclear Safety* 20(4) to the printer by the end of the month. The technical content of *Nuclear Safety* 20(5) is in final composition at TIC, but that issue awaits the May—June "current events" material (submission deadline July 15th). All technical articles for *Nuclear Safety* 20(6) have been peer-reviewed and are being edited for submission to NRC, DOE, and TIC.

During the month we received FY 1979 Program Brief (Midyear Revision) and FY 1980 Program Assumptions. Although the work descriptions for both years are identical, the FY 1980 funding is significantly lower. The probable impact of this decreased funding was sent in a letter to G. L. Bennett on May 25th. Similar information was sent formally to ORO by the end of June.

TABLE 1 RECON DATA BASE ACTIVITY FROM 05-01-79 TO 06-01-79
(22 OPERATING DAYS)

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	CITATIONS PRINTED
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	740	1294	23057
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	183	307	3993
API	(API) AMER. PETROLEUM DATA BASE	37	90	366
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	316	1004	22884
GAP	(DOE) GENERAL AND PRACTICAL INFO.	166	191	4047
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	18	23	531
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	24	38	306
EMI	(EMIC) ENV. MUTAGENS INFO.	203	208	8523
EDB	(TIC) DOE ENERGY DATABASE	3530	6027	140725
PRD	(TIC/NRC) POWER REACTOR DOCKETS	48	71	2307
NBI	(NBIC) NATL BIOMONITORING INV.	9	1	0
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	56	101	1104
ESI	(EIC) ENV. SCIENCE INDEX	92	200	1545
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	227	406	20491
WRE	(WRSIC) WATER RESOURCE RESEARCH	63	155	951
NRC	(LC) NATIONAL REFERRAL CENTER	44	52	513
NER	(EIC) NATIONAL ENERGY REFERRAL	28	48	120
RSI	(RSIC) RADIATION SHIELDING INFO.	14	11	0
EIA	(EIC) ENERGY INFO. ABSTRACTS	87	125	1376
RSC	(RSIC) RADIATION SHIELDING CODES	7	2	0
ESR	(DOE) ENERGY ENVIRONMENT & SAFETY	38	68	400
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	132	154	1802
TUL	(U. TULSA) TULSA DATA BASE	53	123	3017
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	65	76	51
CIM	CENTRAL INVENTORY OF MODELS (DOE)	23	21	70

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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 2: Analysis — Nuclear Pin Simulation. Plans for improvements to increase the versatility of the PINSIM computer code are being made. These improvements will include incorporation of the capability to use local fluid conditions from other computer codes in PINSIM's heat transfer calculations, incorporation of improved heat transfer correlation switching logic, and development of a multiple-axial-level back-calculation capability. A generalized inverse conduction scheme, similar to the specialized scheme currently used in ORINC, is also planned. The new configuration of PINSIM, incorporating these improvements, will be referred to as PINSIM-MOD2.

Electric Pin Simulation. An evaluation of the cross-sectional microphotographs of bundle 3 production heater FRS-038 has been completed. A memorandum is being prepared stating the results of the study and suggesting the diametrical measurements of the sublayers of the fps to be used in future models of the fps (i.e., RELAP, ORINC, PINSIM, etc.).

A fully comprehensive water-steam physical properties package for use on the PDP-11 and -10 and the IBM-91 and -195 has been developed and tested.

A sensitivity study of ORMDIN input variables has started. An FCTF test case has been run using the R-θ version of ORMDIN.

A heat balance program has been started for use on the PDP-11 for future bundle 3 tests. The plotting portion of the code is complete.

Thermal-Hydraulic Simulation. In the past month a plan to perform pretest analysis for THTF Test Series 5 has been developed. Test Series 5 is a set of "reactor core simulation" tests which begin in the last quarter of 1979. Pretest analysis will be performed to specify a loop configuration and power program that will produce core conditions that are typical of what might be expected in a large scale loss of coolant accident.

Actual pretest analysis of Test Series 5 will be performed with a TRAC-P1A system model or a RELAP4 MOD5 system model of the THTF. The code selection will be based on a comparative analysis of the ability of TRAC-P1A and RELAP4 MOD5 to compute the local fluid conditions, rod surface temperatures and rod surface heat fluxes extant in THTF test 177.

A RELAP4 MOD5 system model for use in the comparative analysis exists and a TRAC-P1A model is under development.

Work on a RLPSFLUX core model of THTF MOD2 is now 50% complete. Work is currently centering on determination of pressure loss coefficients for bundle 3 spool pieces and grid spacers.

A COBRA-TF model of THTF MOD1 is now 90% complete. Upon completion the model will be sent to PNL where a THTF MOD1 test case will be run. Analysis of the results of this test case will aid in the selection of a computer code for local fluid condition calculations in THTF MOD2 testing.

Analysis of the predictive capability of RELAP4 MOD6 is now 20% complete. Currently, work is centering on development of a set of programs to allow plotting of heat transfer correlations as a function of local fluid conditions.

Task 3: THTF Operations — The assembly of bundle 3 was completed this month and was bolted into the THTF test section barrel. This bundle has 60 ORNL fabricated fuel rod simulators; 4 instrumented rods and approximately 1200 thermocouples. Completion of the installation is awaiting fabrication of the inlet plenum. The plenum was rejected earlier due to out of tolerance dimensions of the drag disk standoffs. This should be completed the first week in July.

Installation of the rebuilt MG set (S-6) was halted this month because General Electric could not deliver the two generator armatures. Delivery of the armatures is scheduled for 7/2/79.

The software needs for the PDP-11/34 data acquisition system have been defined and efforts are underway to see that these needs are met. In particular, (1) an overall system design for this software has been developed, (2) specific tasks within these needs have been defined, (3) the principle data plotting program is in the final testing stages (it will be used in conjunction with other programs to generate

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the quick look reports on the PDP-11), (4) the plotting package for the loop heat balance program is operational, and (5) considerable progress has been made in the heat balance program, the engineering units program and the steady-state thermocouple verification program.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: T. F. Bott and P. M. Haas

ACTIVITY NUMBER: ORNL #40 10 01 06 01 (189 #BO421-8)/NRC #60 19 10 01 2

TECHNICAL HIGHLIGHTS:

During the month, work was initiated through Memphis State University on an interim report summarizing nuclear plant simulator capabilities. Special attention was given to the ability to treat multiple failures, out-of-position equipment, saturated c... s in the primary plant, and natural circulation. The interim report will be released to NRC the second week in July. Bid specifications and technical descriptions for several simulators were collected and arrangements to visit several simulators were made. Some contacts were made with simulator experts in other fields of technology in order to assess the relative state-of-the-art in nuclear plant simulation.

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-5 capsule, which was started up in the HFR reactor at ECN-Petten on May 10, suffered a series of anomalous coil failures during the first month of operation. Ten of the 20 eddy-current coils showed open circuits when checked for resistance. This is in contrast with HOBBIE-4, which completed approximately 1200 hours with no failures. Preliminary hot cell examination by the Dutch has not yet disclosed the reason for the failures.

We suspect that the failures, which were abrupt and random, were due to breakage of the fragile aluminum coil ribbons which were potted in a Sauereisen compound. Such a procedure was used, however, in the successful HOBBIE-4 capsule. The HOBBIE-6 capsule, which was fabricated except for final calibration and pressure vessel welding, is being disassembled and will have its coil/coil holder assembly replaced with the HOBBIE-7 assembly. The HOBBIE-7 assembly will have much less potting compound around the coils. The HOBBIE-6 assembly will be stripped, rebuilt, and used in HOBBIE-7.

The HOBBIE-1 data report has been published.

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