INTERIM REPORT

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for the
DEPARTMENT OF ENERGY

INTERIM REPORT

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ABSTRACT

Highlights of technical progress during June 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:

The upper plenum prong probes and the downcomer string probes for SCTF have now been shipped. The only remaining sensors to be shipped are the wall mounted film probes (14 modules) and the reference conductivity probe. These sensors are presently being fabricated.

Fabrication of the SCTF electronics system is progressing satisfactorily and is on schedule for the planned shipping date of Sept 1. Preliminary design of the hardware required for the anti-aliasing filters (circuit boards, connector panels, cabinets) has begun. Specifications for procurement of the filters are being prepared. Indications are that delivery of the filters from the manufacturer will be 6-8 weeks from date of order. It should be possible to assemble the hardware in time for a late 1980 shipping date.

Test of the SCTF upper plenum prong probe has been completed. Both axial flow and various directions of cross flow were tested. Probe void fractions for axial flow in air-water and steam-water were in very good agreement with the gamma densitometer. Cross flow results were also good except some directions of flow caused a tendency for the probe to "collect" water and read lower than the densitometer. This effect was most pronounced when the prongs pointed downstream.

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:

A heated thermocouple liquid level sensor was tested in the Advanced Instrumentation for Reflood Studies (AIRS) Test Stand. The facility was used to provide steam and water flow under steady-state conditions at pressures ranging from 310 to 720 kPa (45 to 105 psia). Steam volume fractions ranging from 67 to 100% were obtained. A gamma attenuation densitometer and a turbine flowmeter gave void fraction and velocity information in the system at a location ∿1 m below the test probe. A system for supplying ac power to the probe heater has been installed at the AIRS test stand. Probe power and thermocouple output may be monitored using a single 1000-ft long cable. These components are intended to simulate conditions under which in-vessel level detectors would be used at a nuclear power plant.

The heated thermocouple level sensor obtained from the U.S. Navy was tested in a vertical orientation in the steam-water countercurrent flow loop. The probe remained relatively cool under nearly all flow conditions. It was observed through the transparent test section walls that water droplets were collecting on the probe and running down to quench the heated region. A similar effect probably caused sporadic rewets recorded during natural convection tests with the Navy probe.

A HEATING V computer simulation of a single-sheath level sensor was used to analyze multi-dimensional heat transfer in the probe. The effects of surface heat transfer condition, heater power and thermocouple location were investigated. With the present probe design, there is considerable axial heat transfer and heating of the reference thermocouple, when the surface heat transfer coefficient is low, typical of natural convection in steam.

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:

One under-water test (FAST 57) was performed in the FAST vessel this month. Two acoustic transducers were mounted on the vessel wall as part of the development testing of the system for bubble detection by ultrasonic imaging. This was the first attempt to use the signals from more than one transducer at a time. No meaningful data relative to bubble size was obtained from the acoustic measurements; at this time there is no explanation for this, but the test will be repeated next month.

A new Kaman pressure transducer was installed in the FAST vessel to measure pressure changes in the argon cover gas region. This transducer has a maximum pressure rating of .344 MPa which is ten times less than the rating for the transducer previously used. The new transducer should permit more accurate pressure measurements (and estimates of bubble volumes) to be obtained.

A draft report is being completed that will assess the progress of the under-water tests to date and will establish the need for any additional testing before progressing into under-sodium work.

NSPP:

Run 108, a low concentration sodium oxide aerosol test under dry conditions, was conducted this month. This test completed the planned series on sodium oxide aerosols. The aerosol was generated over a 2-2.5 min period by a pool fire of 0.45 kg of heated sodium. A maximum aerosol concentration of 2.25 $\mu g/cm^3$ was measured at 10 min after the pool fire was initiated (the target concentration was 2 $\mu g/cm^3$). The rate of disappearance of this low concentration aerosol, as expected, was smaller than observed previously for sodium oxide aerosols in the 6-25 μg/cm^3 range. For example, in the higher concentration tests, 90% of the airborne aerosol had settled out or plated out at about 90-100 min after start of the pool fire. In the current test, about 200 min had elapsed before the aerosol concentration was reduced by 90%.

Cascade impactor measurements indicated that the aerodynamic mass median diameter (AMMD) for this low concentration aerosol was not greatly different from those measured during the higher concentration tests. The AMMD value was 3.0 μm at 24 min, 4.1 μm at 57 min, and then decreased to 1.6 μm at about 600 min.

Preparations are underway for Test 209 which will be an attempt to produce a uranium oxide aerosol concentration in the 15-20 μg/cm^3 range. We feel that recent improvements to the plasma torch aerosol generator will make this possible.

CRI-II:

Efforts are continuing to provide additional evidence for co-agglomeration of mixture aerosols through the use of transmission electron photomicrography. The morphology of the mixed-aerosol particles is such that the heterogeneous chains of U_3O_8 appear to be partially encapsulated by the Na_2O_2 particles. An independent evaluation which permits X-ray analysis of individual sodium particle clusters for their content of uranium has been demonstrated with one SEM sample although with relatively poor particle definition. Improvement of this technique is continuing and will include a study of mixed aerosols simultaneously generated and mixing of aerosols generated at different times.

PROGRAM TITLE: Continuous On-Line Reactor Surveillance System

PROGRAM MANAGER: D. N. Fry

ACTIVITY NUMBER: ORNL #41 89 55 12 8 (189 #B0442)/NRC 10 19 11 01

TECHNICAL HIGHLIGHTS:

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

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 — C. P. Debel of Risø National Laboratory, Roskilde, Denmark, visited ORNL on June 9 and 10 and was briefed on the HSST program. He presented the work on fracture at Risø at a seminar.

Bryan Edmondson, Director of Berkeley Nuclear Laboratories of the CEGB in England, visited ORNL on June 23 for discussions on the HSST program.

A technical report, *Flaw Preparations for HSST Program Vessel Fracture Mechanics Testing: Mechanical-Cyclic Pumping and Electron-Beam Weld-Hydrogen Charge Cracking Schemes*, NUREG/CR-1274, by P. P. Holz was published.

Task 2: Fracture Mechanics and Analysis — Several of the mesh generation options in the NØZ-FLAW computer program were successfully executed using the Oak Ridge IBM version of the code. These options include tension-loaded flat plates with central, edge, and double-edge through cracks as well as plates with semielliptical or semicircular surface flaws. Cylinders having straight-edge through cracks and inner-surface semielliptical cracks have also been run successfully for internal pressure loading. Several other options, however, including the nozzle corner flaw option, require further debugging.

Task 3: Irradiation Effects — Data from the Charpy V-notch (C_v) testing from the third HSST irradiation has been evaluated in more detail and the results are presented in Tables 1 and 2. The conclusions as reported in the April Monthly Highlights remain and the important results are as follows. First, weld number 66W had an unirradiated C_v upper shelf of only 76 J. After irradiation the upper shelf energy decreased by about 28% to about 54 J. Although this value is quite low it is above that which is predicted from Regulatory Guide 1.99; 46 J. The second observation is that weld 65W is the first weld (out of seven irradiated in the second and third BSR experiments) that had an actual loss of C_v upper shelf that is greater than that predicted by Regulatory Guide 1.99.

The second capsule of the fourth HSST irradiation experiment was placed in the reactor this month and irradiation was begun on June 26.

Task 4: Thermal Shock — Preparations for TSE-5A were continued. Machining of the three tapered thermocouple-thimble holes and the nine 3-mm-diam holes for COD-gage leads in TSC-2 was completed, thus completing all machining of TSC-2. Assembly of TSC-2 and the insulation cover and assembly of the three tapered thermocouple thimbles was started, and the data system was checked out and repaired as necessary.

Inner-surface-spray-coating development efforts were continued. What appeared to be a satisfactory spray coating of the new coating material (3M-24) was achieved, but the quench time for a 100-mm-diam x 150-mm-long test bar with one end coated was much longer than expected. This work is continuing.

Finite-element mesh-convergence studies were conducted for the 3-D LEFM analysis of the final crack front from TSE-5. The results indicate that the original mesh was adequate. For another TSE-5 analysis a 90° model, as opposed to the usual 180° model, was used for 2-D calculations to see what error such a simplification might make. The error is substantial. The reason for concern is that the University of Maryland has been using the 90° model in their crack-arrest analysis of the second crack jump in TSE-5.

A segment of thermal-shock prolongation TSP-2 is being machined into twenty-two 1T compact specimens. The segment has received the same heat treatments as the thermal-shock test cylinder TSC-2 including a temper of 680°C for four hours.

Task 5: Simulated Service Tests — Work at Babcock & Wilcox Company on the preparation of intermediate test vessel V-8A is progressing. The three preliminary trial weldments were made, and each was cut into three segments so that each composition could be given three different heat treatments. One heat treatment was completed on three weld segments, and specimens are being machined from these pieces. Room temperature tensile tests, Charpy V-notch impact tests in the range 52°C to 149°C and chemical analysis will be made for each segment.

The K_I values at nozzle corner flaws for an intermediate test vessel under pressurized thermal shock loading calculated under Task 2 were compared with estimates of the relevant fracture toughness of a typical nozzle material. While uncertainties exist, it appears that the K ratio approaches unity in the structure and may exceed it near the surface.

Table 1. Summary of unirradiated Charpy V-notch impact properties from welds in third BSR experiment

Weld number	% Cu (nominal)	Temperature[a]		Upper shelf (J)	Energy (ft-lbs)
		°C	°F		
65W	0.35	40	105	101	74
65W	0.22	0	30	112	82
66W	0.43	70	160	76	56
67W	0.27	−5	25	103	76

[a]Temperature at which 68 J (50 ft-lb) energy obtained.

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 Develop.... Activities: A sample transient representing a turbine runback from 100% to 25% power has been successfully accomplished with ORTAP, incorporating the revised turbine model (ORTURB). The final results agree very well with published heat balances and plant conditions at 25% power.

A slight modifi tion of the input cards for BLAST, the steam generator simulator, has been found to decrease computation costs. Work on improving the simulations for the steam lines that connect the main plant components has been initiated.

Code implementation and verification activities: Frequent discussions with Public Service Co. of Colorado (PSC) have been held regarding their implementation of the ORTAP code. Detailed design data for the FSV steam generators was received from PSC. This information is considerably more complete than that which had been previously supplied to ORNL. This information is being included into the BLAST code input deck.

Background information and code development notes are being supplied to BNL to assist with their review of the ORECA code.

Table 2. Summary of effect of irradiation in weld metal Charpy V-notch impact properties

Weld metal	Fluence neutrons/cm^2 x 10^{-18} (> 1 MeV)	Irradiation temperature (°C)	(°F)	Shift in Charpy V-notch reference temperature[a] Predicted[b] (°C)	(°F)	Actual (°C)	(°F)	Loss of Charpy V-notch upper shelf energy (%) Predicted[b]	Actual
64W	3.7–4.6	244	472	116	210[c]	∿83	150	37[c]	∿31
	3.7–4.3	272	522	116	210	∿83	150	37	∿31
	3.4–4.1	288	550	111	200	∿83	150	37	∿31
65W	3.0–3.7	273	524	67	120	∿61	110	29	∿41
	3.4–4.0	276	528	67	120	∿61	110	29	∿41
	3.2–4.0	283	542	67	120	∿61	110	29	∿41
66W	5.0–6.2	268	514	139	250	∿97	175	40	∿28
	4.2–5.4	280	535	128	230	∿97	175	38	∿28
67W	3.6–4.5	269	517	89	160	∿86	155	33	∿21
	3.8–4.5	282	540	89	160	∿86	155	33	∿21

[a] 41 J (30 ft-lbs).

[b] Regulatory Guide 1.99.

[c] Irradiation temperature is too low. Regulatory Guide 1.99 is valid for nominal irradiation temperature of 288°C.

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 are in the process of installing our instrumentation in a truck so that we can transport it to a reactor site and then perform the inspection from the rear of the truck. We are continuing to modify the instrumentation so that it can be operated remotely from the back of the truck to the steam generator.

We are continuing with our attempt to get permission to inspect a steam generator during a reactor outage.

A review of the program was presented to the ACRS Subcommittee meeting on metal components on June 17 in Washington.

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. New tie plate turbines were received from INEL and installed in both the air/water and steam/water loops Both turbines operated significantly better in both single- and two-phase flow in both loops and their range was better matched to the flows existing in the loops. Both loops now have a full complement of UCSP instrumentation and final instrument calibration has begun.

In order to obtain information bearing on the choice between long line DP and tie plate drag body, over 125 tests were run with various combinations of core spray flow rates and core spray plus hot leg injection flow rates. Although data reduction is still incomplete, to date neither instrument appears to have a significant technical advantage over the other. As a further test of their ability to withstand thermal shock the hot leg injection line was extended 35 in. downward toward the upper-core support plate. Both instruments survived the first series of tests with hot leg injection water subcooled 105°F even though condensation shocks were quite severe.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiat. n 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: F. B. K. Kam, L. F. Miller, and F. W. Stallmann attended the ASTM E10.05 Subcommittee Meeting at Savannah, Georgia on June 1-5, 1980. Updates of the unfolding and transport practices were discussed. The uncertainty task group (E10.05.01) held one session to discuss its role relative to the LWR standards, practices, and guides. Discussions were held relating to the Fourth ASTM-EURATOM Symposium. The minutes of this meeting will be distributed to all participants.

Task 2: Benchmark Fields -

A. Dosimetry Measurements

Two months of PCA experiments have been scheduled and finalized for October and November, 1980. A description of the experiments was given in the minutes of the LWR-PVS Dosimetry Program Meeting at NBS, May 19-21, 1980.

B. PSF Irradiation Experiments -

1) The SSC-1 irradiation capsule was removed at 12:53 PM, June 23, 1980 after approximately 44 days of irradiation. The dummy SSC capsule was inserted and the irradiation of the SPVC and void box capsule continued.

2) Process Control System - A draft copy of the documentation of the control algorithm is completed. Typing and editing are in progress.

A problem with occasional failures in the firing of the solid state relays which regulate electrical heater power has been identified and corrected. Drift in the clock which controls the computational time interval relative to the clock which controls the one-shots was the source of the difficulty. It was corrected by identifying the failure and shifting the time base of the computational time interval.

C. BSR Benchmark Field Facility* -

Plans are underway to design and build a benchmark field referercing facility at the Bulk Shielding Reactor (BSR) for the standardization and certification of LWR surveillance dosimetry. This facility will provide a permanent mockup for current surveillance capsules in which dosimetry methods applied by vendors and service laboratories can be:

i) validated and certified;

ii) improved by development of supplementary experimental data; and

iii) evaluated in terms of actual uncertainties.

Participants are requested to submit to B. K. Kam dimensions and drawings of typical surveillance capsules used in power reactors.

Task 3: Neutron Field Characterization - Transport calculations to characterize the PSF neutron environment have been delayed to FY-81 due to lack of funding.

Task 4: Dosimetry and Damage Correlation - The minutes of the PCA Blind Test workshop were distributed to the participants for comments and for additional information to try to resolve discrepancies between the measurements and calculations. Followup information was received from B&W and Japan. The distribution of all followup information will be made available to the participants by means of the technical newsletter.

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*For more information, see Attachment #5 May 19-23, 1980 "5th LWR-PVS Dosimetry" and "PCA Blind Test" Meeting results.

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:

R. H. Chapman attended the US-NRC/FRG-PNS/Japan-JAERI Annual Safety Research Information Exchange Meeting and Workshop in Karlsruhe, Germany, during the week of June 9. Recent MRBT single rod heated shroud test results and a quick-look report on the B-5 (8 X 8) test were presented at the meeting.

The Committee on the Safety of Nuclear Installation (CSNI), which is a committee of the Nuclear Energy Agency, plans to publish a state-of-the-art report (SOAR) on *Fuel Behavior During a LOCA*. Each chapter of the report will be written by an editor, based on his knowledge, published reports, and solicited contributions. R. H. Chapman attended (as an editor) a meeting at CSNI in Paris on June 19 to discuss preparation of the report.

Participation in these meetings afforded an opportunity for Chapman to visit several European nuclear research laboratories to discuss Zircaloy cladding deformation research. In particular, visits were made to the Euratom Joint Research Center at Ispra, Italy, the CEGB Berkeley Nuclear Laboratories, at Berkeley, England, and the UKAEA Nuclear Development Laboratories in Springfields, Windscale, and Winfrith, England. Much useful information on current and planned research was obtained. Also, since the MRBT program is well known at all the sites visited, a short presentation was made at each, using materials prepared for the Karlsruhe meeting. Based on the questions and discussions generated, the presentations were well received.

Preliminary visual examination of B-5 was completed in preparation for the hydraulic characterization, which will be performed under subcontract by an outside research laboratory as soon as contractural arrangements can be finalized. Problems were encountered in the removal of the internal heaters. Eleven of the 64 heaters could not be removed with ordinary force and were left in the tubes to avoid the possibility of disturbing the deformed bundle. While the burst location and orientation in the tubes without heaters could be determined accurately by use of a borescope, the burst locations for those 11 tubes with heaters in place could be determined only approximately by observing the bundle for the exit point of injected smoke.

Photographs and other external dimensional documentation were made to complete the preliminary examination of B-5. Further examination, including sectioning and determining deformation profiles, will be performed after the hydraulic characterization.

A quick-look report, based on data recorded during the test and the posttest examination described above, was prepared. The report will be reproduced and given limited distribution in early July.

The design is underway for the next bundle (B-4) test which will be similar to B-5, except that it will be a 6 × 6 array instead of 8 × 8. Certain design improvements will be made for B-4 that were observed to be necessary from the temperature distribution in the B-5 test.

Three single rod heated shroud tests were performed this month. Two of the tests were conducted to investigate constant internal pressure as a parameter. The other burst test was performed at a low heat rate in the $\alpha \rightarrow \beta$ transition temperature range to investigate this low deformation region. Results from the tests are not yet available.

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:

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

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

PROGRAM MANAGER: Betty F. Maskewitz

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

TECHNICAL HIGHLIGHTS:

Work in relation to the RSRDR task has currently been held in abeyance pending an official decision as to future directives of the program. Personnel assigned to the task are currently involved in work related to making a survey of data needs within NRC programs prior to consideration of establishing a generic data bank as a technical resource to those programs.

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 June, the staff of the Nuclear Safety Information Center (a) processed 906 documents, (b) responded to 71 inquiries (of which 35 involved the technical staff and 7 were for commercial users), and (c) made 16 computer searches. The RECON system, which now has over 200 remote terminals, reports that the NSIC data file was accessed 132 times between May 1 to 30 making it the sixth most utilized of the 28 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 10 visitors and participated in 4 meetings.

One NSIC report was issued: *Index to Nuclear Safety, Vol. 11 through Vol. 20* (ORNL/NUREG/NSIC-175). One other 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); *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 June, we received 20 foreign documents (2 French, 7 German, and 11 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 June 27, 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 June we renewed 3 exempt users which, with other withdrawals and renewals, leaves the SDI service at a total of 396 users.

The regular *Nuclear Safety* staff meeting was held June 4th. Minutes of that meeting and tentative outlines for the next several issues of *Nuclear Safety* were distributed shortly thereafter. TIC sent the final galleys of *Nuclear Safety* 21(4) to the printer on June 27, 1980. The technical content of *NS* 21(5) is in final composition at TIC, but that issue awaits the May—June "current events" material (submission deadline to TIC - July 15th). All technical articles for *Nuclear Safety* 21(6) have been revised by the authors and are being edited for submission to NRC, DOE and TIC. A Guide to Section Editors was completed and distributed to all Section Editors and their line supervision.

TABLE 1 RECON DATA BASE ACTIVITY FROM 05-01-80 TO 05-30-80
(20 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	3531	5276	124144
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	637	920	8370
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	258	621	18066
EMI	(EMIC) ENV. MUTAGENS INFO.	183	285	7981
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	161	243	2492
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	132	245	14801
GAP	(DOE) GENERAL AND PRACTICAL INFO.	129	129	2352
ESI	(EIC) ENV. SCIENCE INDEX	119	189	1373
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	111	125	3601
EIA	(EIC) ENERGY INFO. ABSTRACTS	83	170	909
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	71	79	519
IPS	(TIC) ISSUES AND POLICY SUMMARIES	56	160	52
WRE	(WRSIC) WATER RESOURCE RESEARCH	52	104	632
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	41	79	609
PRD	(TIC/NRC) POWER REACTOR DOCKETS	36	14	14123
SLP	(FRANKLIN) SOLAR DATA BASES	29	64	-
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	28	12	221
NES	(NESC) NATIONAL ENERGY SOFTWARE	26	37	-
CIM	(DOE) CENTRAL INVENTORY OF MODELS	23	53	32
NER	(EIC) NATIONAL ENERGY REFERRAL	22	24	23
NRC	(LC) NATIONAL REFERRAL CENTER	22	23	169
API	(API) AMER. PETROLEUM DATA BASE	21	41	432
RSI	(RSIC) RADIATION SHIELDING INFO.	21	12	3312
TUL	(U. TULSA) TULSA DATA BASE	17	10	166
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	11	12	182
RSC	(RSIC) RADIATION SHIELDING CODES	9	11	-
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	1	6	-

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 - No activity this month.

Task 2: Analysis - Data Management. Data reduction is continuing on the THTF Upflow Film Boiling Test. Modifications to the Instrument Data Base (IDB) have been identified. A process tape has been generated which includes additional quantities such as composite density calculated from a triple-beam densitometer, individual rod powers, and total bundle power. Processing of data from this test is about 60% complete.

Data reduction has begun on the THTF downflow film boiling test. An uncalibrated engineering units tape has been generated. A process tape containing quantities of composite densities from three triple-beam densitometers, individual rod powers and total bundle power has been generated. This task is approximately 25% complete.

Documentation of the computer code DACREP (Data Conversion and Reduction Program) has been started and is presently in draft form. Figures and tables will be compiled for this publication in the near future.

Nuclear Pin Simulation Analysis. Pretest analysis associated with the upcoming double-ended blowdown test (Test 3.05.5B) is complete. Both the hydraulics analysis (to determine appropriate rupture disk orifice sizes and operating conditions) and the power programming analysis were completed soon enough to allow the test to be run two weeks ahead of schedule.

Final review of the report describing the nuclear pin simulation analysis of THTF Test 105 has been completed. The draft has been modified to incorporate the reviewers' comments and has been forwarded for publication. The report describing the PINSIM-MOD2 verification study is undergoing review prior to publication.

Electric Pin Analysis. The preliminary run on THTF Test 3.03.6AR (Upflow Film Boiling Test) by the preprocessor program for ORINC and ORMDIN has been completed (on schedule). 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 — thus limiting these codes to computational type work.

Debugging of ORINC (Version 3.2 — for THTF bundle 3 tests) has been completed. The preliminary run of ORINC on THTF Test 3.03.6AR has been completed and reviewed (on schedule).

ORMDIN has been updated to interface with the preprocessor tape, and this version of the code has been debugged (on schedule). The code is now fully automated to process BDHT tests in the THTF. A test run has been made at four locations in the THTF core for Test 3.03.6AR.

The final segment (Part 3) of the ORTCAL calibration package for THTF bundle 3 has been debugged. ORTCAL-Part 3 essentially determines the effective thermal diffusivity of the FRS boron nitride core. This work has been completed prior to receiving actual data from operations.

Thermal Hydraulics Analysis. Analysis of the bundle uncovery/recovery test continued in June. Comparison of heat transfer coefficients predicted by existing correlations to experimentally determined heat transfer coefficients has been done. Results indicate that none of the correlations follow data trends well. Results of bundle recovery testing indicate the development of inverted annular flow in some high inlet flooding rate tests.

The Quick Look Report for the Upflow Film Boiling Test, 3.03.6AR, was completed on June 20, 1980. Analysis of the data is underway and is 25% complete. Planning for the next Upflow Film Boiling Test (3.06.6B) is underway and is approximately 40% complete. Work on the correlation comparison program is continuing and is approximately 65% complete.

Task 3: THTF Operations - The bundle was operated on two occasions this month. The first was an attempt to conduct the Downflow Film Boiling Test (3.04.7). The system operated satisfactorily up to steady-state conditions at 37 kW/rod and a test section outlet temperature of 604 K (627°F). For an unknown reason, the power to the primary pump tripped, resulting in a rapid test section pressure and temperature increase. Promptly, bundle power was manually tripped, but the loop was already depressurizing due to a break in the 12.7-mm (0.5-in.) rupture disk located at the top of the pressurizer.

The downflow film boiling test was successfully completed during the second attempt. This time, the system operated satisfactorily throughout the entire test. Blowdown (inlet only) was initiated and the bundle was ramped to full power, resulting in an excursion to film boiling in the test section.

Presently, plans are being finalized for completing modifications necessary for future test operations, including the installation of new thermocouple array rods, shroud thermocouples, and an in-bundle gamma densitometer.

Task 4: Two-Phase Instrument Development — Fabrication of the in-bundle densitometer system components is near completion. Delivery of the ion chambers is expected in early July. Development of the fabrication technique for the semi-annular source capsule has been completed and tested. The sources will be available on schedule. The design and fabrication of the electronics and signal conditioning for the ion chambers is also on schedule. Assembly and initial testing of the densitometer system (excluding the motor drive system) is scheduled for mid-July. The motor drive system is being fabricated on schedule for installation prior to the Bundle Uncovery Test Series.

Work on updating instrument uncertainties is continuing. Work on model errors for single- and triple-beam gamma densitometers using the TU PHASE computer code has begun. The code simulates inputted flow regimes and determines model errors resulting from sampling finite beam paths. The mass flow code AMICON has been run for the Upflow Film Boiling Test (3.03.6AR) and the results supplied to the Thermal Hydraulics Analysis Group.

Design is nearing completion for the shroud wall thermocouple additions. The thermocouples will be axially located near FRS thermocouple levels along the heated length and near grid spacer thermocouple levels along the upper half of the heated length.

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 data collection has been completed for the first eight groups of operators, from "utility A". Exercises with four additional groups from a second utility were planned for July, but that utility postponed its training program until FY 1981. At least two groups from a third utility have been scheduled during the remainder of FY 1980 (beginning in August). Development of software for extraction of event-specific data has continued, and processing of raw data tapes has been initiated.

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL # 41 89 55 11 7 (189 # BO124)/NRC # 60 19 11 04 1

TECHNICAL HIGHLIGHTS

HOBBIE-8, the final test in the joint NRC/ECN-Petten program for in-reactor testing of Zircaloy fuel cladding, was completed on June 30 after approximately 1500-1600 h of irradiation. As in the two previous tests, a pressure reversal was included in the test procedure. After approximately 500 h of external pressurization (18.6 MPa), the test was switched to internal pressurization (5.2 MPa) to simulate a high buildup of fission gas products.

As soon as the data tape for HOBBIE-8 is received, analysis will start. The data tape for the internal pressurization portion of the HOBBIE-7 test has still not arrived. A copy of the external portion was sent by mistake earlier, but the Dutch have now mailed a new copy.

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