INTERIM REPORT

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UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

INTERIM REPORT

CONTENTS

ABSTRACT

Highlights of technical progress during November 1980 are presented for nineteen 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 fo 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 electronics system for the SCTF film and impedance probes was given its initial on-site start up and calibration. All of the installed sensors were connected and "wet" readings were recorded during the hydrostatic pressure test of the vessel. All of the probes except the in-core flags functioned normally. A ground-interaction problem existed on all of the flag probes apparently because of capacitive coupling to vessel ground at the electrodes and in the hard cable termination boxes. A temporary solution to the problem was found by modifying the shield terminations in the boxes. A permanent repair will be performed once the method is verified by testing at ORNL. After the temporary repair the whole system operated normally during the wet calibration.

Fabrication of printed circuit boards and other components for the SCTF/CCTF anti-aliasing filters has been started. It is planned to ship the filters around January 15, 1981.

Bench testing of a prototype electrode-to-ground impedance electronics module was started. Initial indications are that its performance will not be as good as the present electrode-to-electrode modules. It is more sensitive to the effects of long signal cables and does not reject noise as well. This type of electronics is required for the "thumbtack" probe which is being considered for use in the CCTF-II core. At present it is planned to include one "thumbtack" probe along with eleven flag probes for CCTF-II. If the electronics turn out to be unsatisfactory, the "thumbtack" will have to be replaced with a flag probe.

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 differential-type heated thermocouple (HTC), installed in the Thermal Hydraulic Test Facility (THTF), was monitored during the THTF Small Break LOCA Heat Transfer Tests (Series II) run on November 25. Preliminary data review indicates that the HTC reliably detected conditions that led to high rod bundle temperatures. No further experiments are planned in the THTF.

Methods were developed to digitize and replot data taken with chart recorders during the THTF tests. An interim report on the Quasi-Steady-State Film Boiling Tests is being prepared.

A three-element HTC array (3 heated and 3 reference junctions), intended for testing in high-pressure natural convection and in the Instrument Development Loops was fabricated. The HTC electronics designed by J. V. Anderson at INEL were fabricated at ORNL and are being packaged.

A splash shield was installed on the 1-cm-diam heated RTD and the probe was tested in the Air Water Test Facility. The probe was oriented vertically in a 20-cm-diam glass pipe. Tests were run with stagnant air and water, with flowing water and with bubbly and froth flow. It appears that the output signal with that shield could not be clearly related to the flow condition.

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:

NSPP:

The activities of the NSPP during this period were concerned with the conversion of the experimental system to enable aerosol tests to be conducted in steam environments. Several preliminary tests have been conducted to determine maximum operating conditions of temperature and pressure, ability to handle steam condensate in the vessel and transfer from the vessel during operation, capability of the modified aerosol filter sampling systems in steam environments, and operability of a new wall steam condensation rate sampler. On the whole, the tests have shown that the experimental system will operate satisfactorily; however, additional modifications to the aerosol filter sampling systems will be needed to adequately handle the steam condensation.

We plan to adequately demonstrate the serviceability of all sampling systems as well as the experimental system before conduct of the first U_3O_8 aerosol test.

CORE MELT:

Planning has continued for investigations into core-melt release processes.

The possible effects of high density aerosol agglomeration and settling have been reviewed in an invited paper* which was presented in a special session entitled "Realistic Estimates of the Consequences of Nuclear Accidents" at the ANS/ENS International Conference in Washington, D.C. on November 20, 1980.

FAST/CRI-III:

No experiments were performed in the FAST/CRI-III test facility this month. A letter report entitled "Assessment of FAST Water Test

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*Presentation: "Impact of Secondary Effects on the Reduction of Fission Product Source Terms in Class IX Reactor Accidents," presented at a Special Session on Realistic Estimates of the Consequences of Nuclear Accidents at the Annual Conference of the ANS/ENS, in Washington, D.C., November 16-21, 1980.

Results and of the Advisability of Going on to Sodium Tests" was completed and transmitted to the NRC at the end of the month.

ANALYTICAL:

Equations are being developed for a CSMP-program simulation of adiabatic oscillations of a submerged quantity of an ideal gas in one-dimensional geometry. The equations include a simplified model of fluid motion around the gas.

It is felt that solution of this simple model will provide insight in interpreting the FAST results and the results of the more detailed models.

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:

Surveillance System at Sequoyah. Several trips have been made to Sequoyah to examine the status of the surveillance system. The hardcopy unit was returned to Sequoyah on the first trip. These trips were made to measure the signal levels of the various instrumentation signals available to the surveillance system. However, due to start-up testing the plant was not in an operating mode. Because of this start-up testing no data has been obtained from Sequoyah.

The dedicated phone line between Oak Ridge and Sequoyah is being installed by South Central Bell. This phone line should be available soon for transmission of data between Oak Ridge and Sequoyah.

Several software routines have been written for the surveillance system to provide the capability for signal verification and for determining the amplitude of input signal fluctuations. These routines will be used in determining the value of the gain settings for the signals selected for surveillance.

Procurement of Advanced System. The analog system component parts have been placed on order. The circuit board layout has been submitted to drafting. The digital system will be delivered in January, 1981.

DNF/bjm

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 — Mr. Brian W. Pickles of UKAEA visited ORNL on November 3 to review HSST program activities.

Dr. Gustav Katzenmeier of MPA visited ORNL on November 3 to review HSST-MPA cooperative activities in the area of flawed vessel testing.

Dr. Deitmar Sturm of MPA visited ORNL on November 10 and made a presentation on their intermediate-size vessel testing program.

Task 2: Fracture Mechanics and Analysis — The remaining options of the Oak Ridge IBM version of the NOZ-FLAW computer program were fully operational this month. The program is now capable of calculating K-values for flat plates (seven different flaw geometries), cylinders (eight different flaw geometries), and plates with a hole (five different flaw geometries) in addition to nozzle corner flaw configurations. Since the nozzle corner flaw option is only one of twenty-two possible options, the program has been renamed OR-FLAW (Oak Ridge-FLAW) to reflect this generality. OR-FLAW is capable of considering internal pressure loading and/or thermal shock for all options. Either mathematical or arbitrarily shaped flaws may be considered. A users' manual for the nozzle corner flaw option of OR-FLAW has been prepared and will be published in December.

Work was initiated this month to modify the Oak Ridge version of the ADINA finite-element computer program to evaluate J-integral for plane stress or plane strain problems. A procedure outlined by deLorenzi and Shih of General Electric, Schenectady, was followed for this implementation. The necessary modifications to ADINA have been performed and check cases are presently being run. Since J = G for LEFM, the J-Integral can be used to calculate K for plane stress or plane strain LEFM problems. In fact, many researchers think this may be the most accurate means computationally to obtain K values. The modified Oak Ridge ADINA program gave the handbook value of K for a double-edge crack, uniaxially loaded plate and the J values for several different integration paths reported by deLorenzi were reproduced exactly. Work will be initiated next month to implement a crack growth modeling capability described by deLorenzi into ADINA. In addition, an elastic-plastic J-Integral calculation will be performed for a compact tension specimen and comparisons will be made with Merkle's empirical expression for this specimen.

Task 3: Irradiation Effects — Irradiation of the second capsule of the Fourth HSST Irradiation Study is continuing. This capsule now has about 2500 hours of irradiation at 2 MW reactor power. Disassembly of the first capsule was delayed due to hot cell maintenance problems but should be disassembled soon.

Specimens for the third capsule of this series are being fatigue precracked with completion expected early in December.

Testing of the remaining Charpy specimens from the second and third 4T irradiation experiments was started this month with completion of testing scheduled in December.

The tensile machine will be installed in the hot cell as soon as the Charpy impact tests are completed.

Two 4T-CS specimens from the Second 4T-CT Irradiation Study were shipped to the Naval Research Laboratory for hot cell testing.

Task 4: Thermal Shock — Development of the OCA code was continued, and a draft version was used to perform a parametric analysis of OC-1 with the transient time extended from 20 to 40 min. The draft version of OCA is now available for routine in-house analysis of PWR overcooling accidents.

Tearing instability and fracture-mode conversion (ductile-brittle) as related to lower-bound toughness measurement were investigated using a simple analytical model similar to that considered by Paris et al. It appears, based on this model, that increasing testing-machine compliance to enhance tearing instability would not have reduced the scatter in K_J data obtained with 1T and 2T-CT specimens for TSE-5 and -5A.

Scanning electron microscopy studies of the fracture surfaces from the 1T compact specimens machined from the steel used in TSE-5 and TSE-5A have been initiated. These preliminary examinations have revealed that "islands" of cleavage-mode fracture exist in the predominately dimple-mode fracture regions associated with the ductile tearing that occurred during testing. These embrittled sites appear to be related to the presence of an inclusion. Extensive studies will be made of these fracture surfaces to determine whether or not these areas of cleavage-mode fracture are serving as sites where pop-in can occur, thereby imposing a dynamic, albeit localized, loading condition on the test specimen.

Task 5: Simulated Service Tests — In the special weld development for intermediate test vessel V-8A, the Babcock and Wilcox Company has completed testing of specimens from the last preliminary trial weld (V8Q). This weld was made with a wire of different composition from that with which the nine other preliminary trial welds were made. Improvement in the temperature of the onset of upper shelf behavior, i.e., a small downward shift, did not materialize as had been hoped. Therefore, the earlier choice of preliminary trial weld V822 as the preferred procedure still stands.

Welding and heat treatment of the final trial weld in a 150-mm-thick plate were completed. The V822 procedure with reduced heat input was used. Charpy impact energy characterization will be fully developed, and J-R curve behavior will be determined before approval is given for applying this welding procedure to vessel V-8A.

V-8A test instrumentation work is continuing, and preparations for trials of flawing procedures were initiated.

A series of three-dimensional finite-element analyses of stress intensity factors for an intermediate test vessel under pressurized thermal shock loadings was planned. The finite-element computer code OR-FLAW with thermal input from ADINAT computations will be used to calculate K_I distributions for combined loadings. OR-FLAW is the counterpart of NOZ-FLAW for geometries not involving nozzle-vessel intersections as described under Task 2.

To date OR-FLAW computations for surface flaws in cylinders have been made for the purposes of testing mesh generation subroutines, checking the numerical algorithms of the code, and evaluating inherent limitations of the code. Corrections and additions, made last month, to the program as furnished by Georgia Tech have been checked. Several comparisons have been made of K_I distributions from OR-FLAW and ADINA for pressure loading only. It is evident that some details of crack shape and mesh structure affect the OR-FLAW results. This may be particularly important near intersections of the crack tip with a free surface. This is a situation for which ADINA results are also questionable.

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

Fort St. Vrain (FSV) Reactor Licensing Support: Results of recent calculations of postulated FSV loss-of-forced convection (COFC) plus firewater cooldown (FWCD) accidents were presented at a meeting at NRC-Bethesda on November 7. These calculations are used by LASL to estimate thermal stresses in the core support blocks resulting from large temperature differences between adjacent blocks. Sensitivity studies were included in the presentation which indicated the need for confirmation of code predictions of the reverse flows that occur during the FWCD phase of the accident.

Code Development Work: Further work was done on exercising the ORTURB turbine plant dynamics portion of the ORTAP code to test its response to various postulated transients and equipment malfunctions. A routine was written for the ORECA 3-D core model code which calculates initial conditions for worst-case region outlet temperature dispersions.

Development of Proposed FSV Experiments: Work continued on proposals for several licensing-related FSV experiments, including post Luci lock noise analysis, bypass flow fraction tests, hot streak tests, and reverse flow detection tests.

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 tube-sheet region. We performed an inservice inspection at the Robert E. Ginna Nuclear Power Station. One hundred tubes were inspected during a four hour shift. The tubes were only inspected up to the first support, since this included the region of interest. Although a detailed analysis of the data has not been performed, we only observed a significant amount of thinning in two tubes, with the maximum amount being 5 mils. The thickness reading was offset by 12 mils, which was more than we observed in any of the previous tests. The data can be restored by using the MODCOMP IV to compute the vector corrections for the magnitude and phase readings.

We have also programed the microcomputer in the instrument to make an approximate correction to a standard. Although the calculations showed the 10kHz and 1kHz were slightlv better frequencies, we have changed the frequencies to 20kHz, 100kHz, and 500kHz since the instrumentation, coil design, and stability requirements were not as severe. The experimental tests have shown a small improvement using the new frequency range.

We have been continuing our discussion with Westinghouse and Point Beach personnel to coordinate our December 16 inspection at their facility. Our truck is scheduled to leave Oak Ridge, December 10, and return December 19.

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:

No report this month.

PROGRAM TITLE: Light Water Reactor Pressure Vessel Irradiation Program

PROGRAM MANAGER: F. B. K. Kam

ACTIVITY NUMBER: ORNL 41 89 55 12 (189 #B0415)/NRC #60 19 10 05

TECHNICAL HIGHLIGHTS

Task 1: Program Administration: Documentation of all PCA experiments and calculations are in progress. A U.S. program committee meeting for the Fourth ASTM-EURATOM International Symposium on Reactor Neutron Dosimetry has been scheduled at Lynchburg, Virginia, Dec. 9-10, 1980. Dr. W. Schneider, co-chairman of the EURATOM committee, will be present to discuss paper selection, format of the Symposium, schedules, etc. A "Call for Papers' will be released in January.

Task 2: Benchmark Fields -

A. PCA - Neutron Field Characterization - Transport Calculations and Dosimetry -

Calculational data and C/E ratios from the PCA "Blind Test" are being compiled and readied for the Blind Test NUREG report.

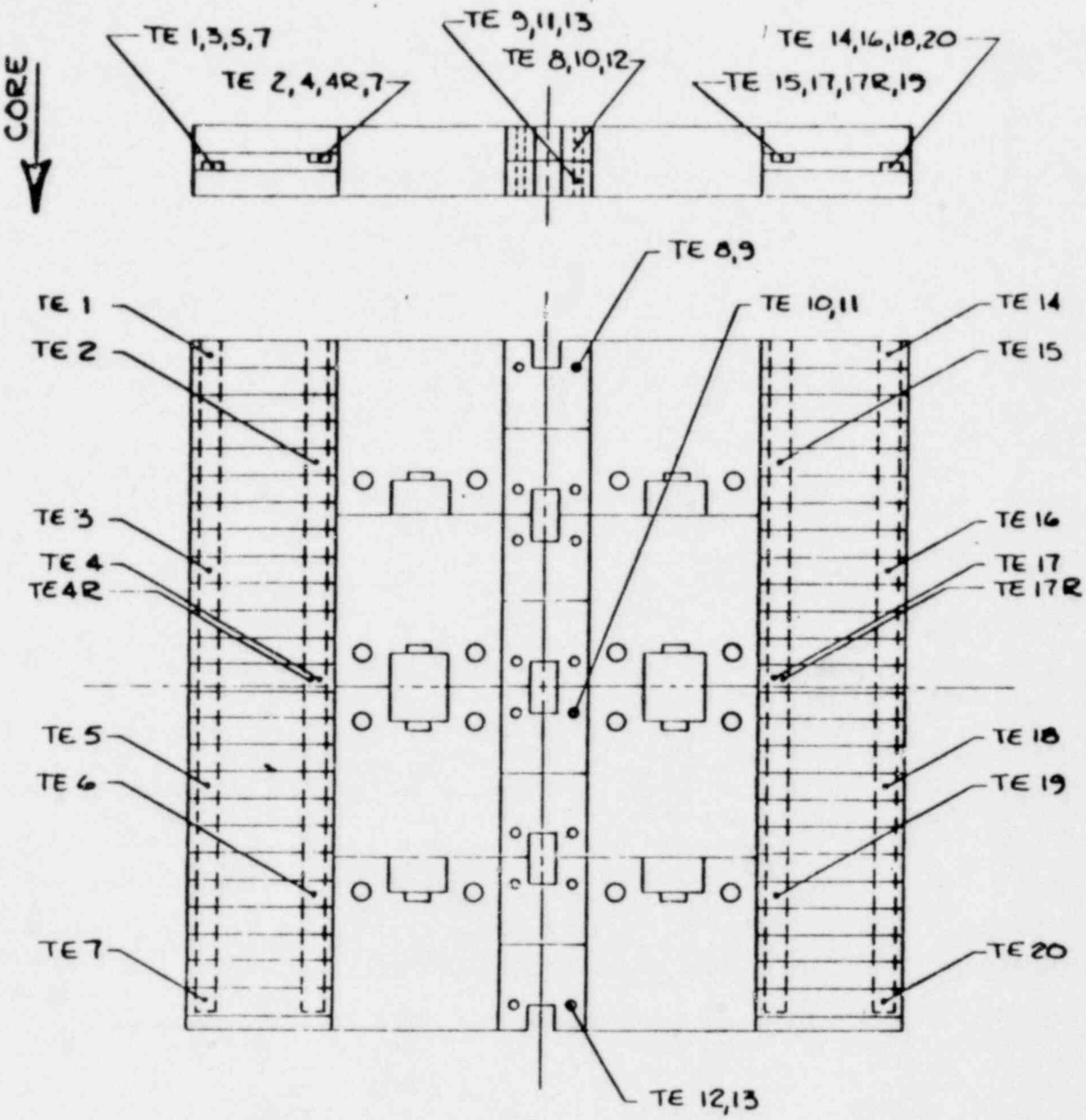
B. ORR-PSF -

Counting data from the gradient wires in the first SSC capsule were analyzed to obtain preliminary estimates of ϕ>1.0 MeV at the locations of the Charpy specimen. The cumulative irradiation history and thermal characteristics for the simulated pressure vessel capsule (SPVC) through the month of August are given in Table 1 and Figure 1.

C. BSR-Surveillance Dosimetry Measurements Benchmark Facility - No report for the month of November.

Task 3: ASTM Recommended Procedures for LWR-PV Surveillance Embrittlement Program -

A revised draft for the neutron transport method guide will be distributed for discussion before the next ASTM E10.05 meeting in Feb. 1980.



NOTE:

1. TE NUMBERS FOR OT LOCATION ARE 100 + TE # (101 THRU 120)
2. TE NUMBERS FOR 1/4T LOCATION ARE 200 + TE # (201 THRU 220)
3. TE NUMBERS FOR 1/2T LOCATION ARE 300 + TE # (301 THRU 320)

Fig. 1. Thermocouple Locations

Table 1. Irradiation and Temperature Data Through August 31, 1980

Data for PSF Specimen Set OT
Hours of Irradiation Time = 1637.93
Megawatt Hours of Irradiation = 48717.27

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE101	42.91	9.58	2078.54	21.38	0.00	289.61	2.22
TE102	41.98	7.40	2060.37	42.66	0.00	291.67	1.62
TE103	41.51	7.52	2103.39	0.00	0.00	289.25	1.27
TE104	37.77	7.62	2098.60	8.44	0.00	292.21	1.11
TE105	41.36	8.20	2102.87	0.00	0.00	285.90	1.29
TE106	37.55	8.29	2106.57	0.00	0.00	288.79	1.11
TE107	42.38	224.09	1885.97	0.00	0.00	281.95	1.55
TE108	43.54	7.35	2092.43	9.06	0.00	290.23	1.95
TE109	43.56	8.34	2092.72	7.78	0.00	289.50	1.90
TE110	40.36	6.81	2092.31	12.95	0.00	291.08	1.31
TE111							
TE112							
TE113	36.39	8.13	2107.84	0.04	0.00	290.24	1.71
TE114	45.74	7.85	2098.81	0.00	0.00	288.43	1.80
TE115							
TE116	43.76	8.01	2100.65	0.00	0.00	290.10	1.13
TE117	41.89	8.11	2096.62	5.29	0.50	291.86	1.17
TE118	44.48	8.31	2099.69	0.00	0.00	286.05	1.24
TE119	41.55	8.19	2102.68	0.00	0.00	286.92	1.19
TE120	44.22	173.30	1934.93	0.00	0.00	282.39	1.82

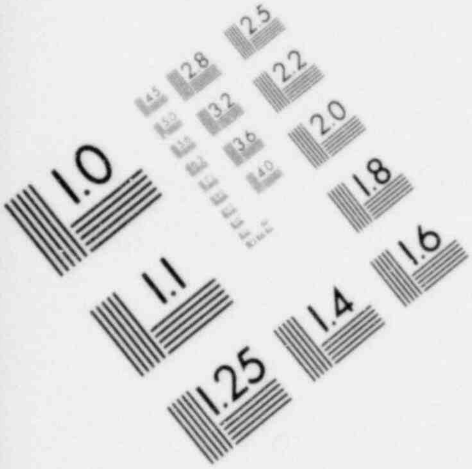
Data for PSF Specimen Set 1/4T
Hours of Irradiation Time = 1637.93
Megawatt Hours of Irradiation = 48717.27

TE201	42.28	8.32	2099.16	2.66	0.00	290.04	1.76
TE202	42.55	7.68	2102.01	0.17	0.00	289.11	1.30
TE203	41.88	8.81	2101.75	0.00	0.00	288.21	1.34
TE204	40.00	6.93	2105.17	0.33	0.00	289.18	0.95
TE205	39.83	11.16	2101.44	0.00	0.00	285.71	1.23
TE206	37.94	11.13	2103.35	0.00	0.00	286.57	1.04
TE207	41.59	31.84	2079.00	0.00	0.00	282.32	1.24
TE208	41.94	7.36	2102.28	0.83	0.00	288.87	1.83
TE209	42.28	8.20	2101.94	0.00	0.00	289.52	1.65
TE210	41.72	10.87	2099.84	0.00	0.00	286.93	0.84
TE211	42.81	11.34	2098.29	0.00	0.00	285.24	0.84
TE212	37.44	5.26	2109.70	0.00	0.00	290.82	1.28
TE213	37.77	5.37	2109.26	0.00	0.00	289.45	1.33
TE214	42.55	5.84	2104.02	0.00	0.00	290.14	1.16
TE215	42.72	9.07	2100.61	0.00	0.00	287.45	0.89
TE216	41.84	9.50	2101.09	0.00	0.00	287.51	0.90
TE217	40.50	6.19	2105.73	0.00	0.00	289.67	0.99
TE218	40.39	11.78	2100.28	0.00	0.00	286.44	1.27
TE219	38.93	8.94	2104.57	0.00	0.00	286.83	1.19
TE220	40.01	100.16	2012.26	0.00	0.00	284.36	1.76

Table 1. (Cont'd)

Data for PSF Specimen Set 1/2T
Hours of Irradiation Time = 1637.93
Megawatt Hours of Irradiation = 48717.27

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE301	41.00	3.60	2068.65	39.18	0.00	290.43	1.42
TE302	42.51	6.88	2103.04	0.00	0.00	286.44	1.11
TE303	39.99	7.82	2104.59	0.00	0.00	287.50	1.15
TE304	37.94	7.09	2106.79	0.58	0.00	290.91	0.87
TE305	37.61	9.07	2105.71	0.00	0.00	287.69	1.16
TE306	39.08	10.35	2102.97	0.00	0.00	286.41	1.03
TE307							
TE308	42.84	5.35	2104.23	0.00	0.00	288.82	1.28
TE309	43.22	5.82	2103.39	0.00	0.00	287.67	1.06
TE310	43.38	10.26	2098.82	0.00	0.00	285.58	1.17
TE311	43.55	10.25	2098.66	0.00	0.00	286.05	1.33
TE312	38.60	3.94	2109.70	0.17	0.00	288.80	1.11
TE313	38.88	4.62	2107.26	1.67	0.00	290.13	1.28
TE314	43.95	5.87	2102.60	0.00	0.00	289.34	1.10
TE315	44.67	6.27	2101.50	0.00	0.00	285.11	0.96
TE316	43.65	3.33	2105.45	0.00	0.00	287.49	0.92
TE317	40.98	5.22	2106.22	0.00	0.00	290.59	1.10
TE318	40.58	5.24	2106.59	0.00	0.00	289.01	1.21
TE319	42.31	10.82	2099.28	0.00	0.00	285.05	1.02
TE320	41.01	9.57	2101.82	0.00	0.00	287.37	1.71



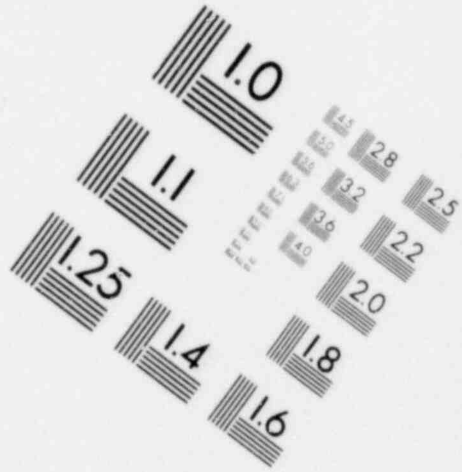
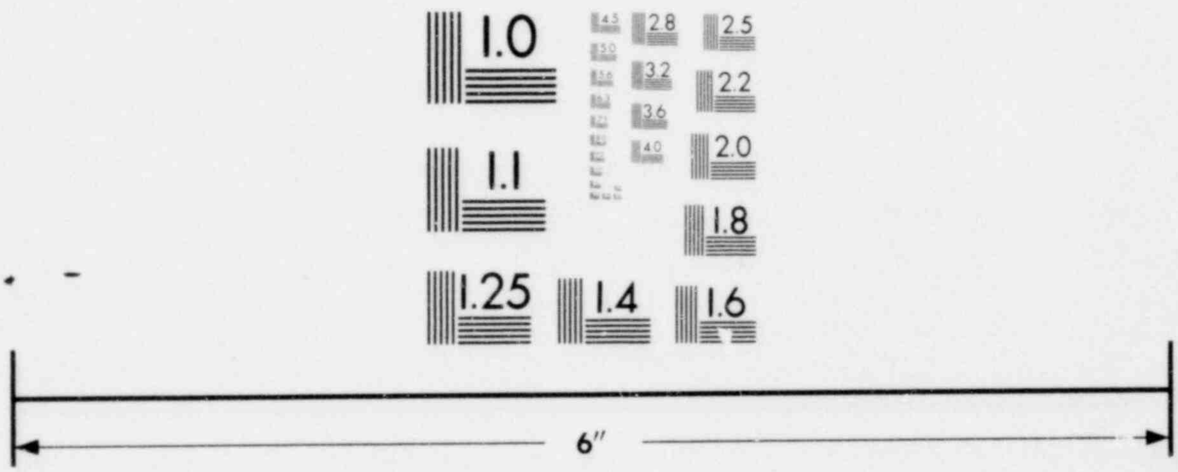
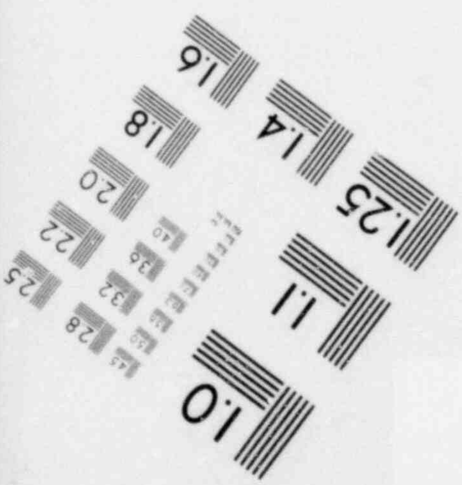


IMAGE EVALUATION
TEST TARGET (MT-3)





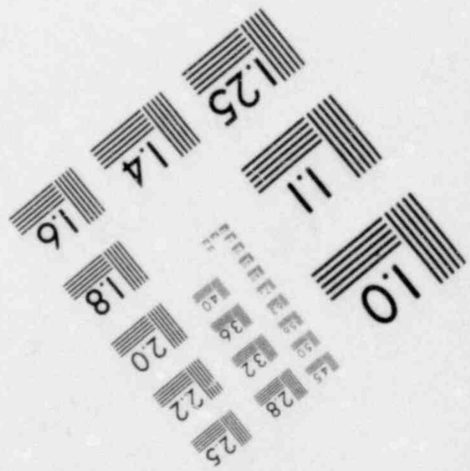


IMAGE EVALUATION
TEST TARGET (MT-3)

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PROGRAM TITLE: LWR Severe Accident Sequence Analysis (SASA)

PROGRAM MANAGER: M. H. Fontana

ACTIVITY NUMBER:

TECHNICAL HIGHLIGHTS:

GENERAL:

A meeting was held at TVA on November 5, 1980 to discuss specific areas of work that ORNL will do in the SASA program, related work to be done by TVA and specific areas of assistance that TVA could provide to the SASA program.

ORNL agreed to make available to TVA a work breakdown and schedule chart of the SASA program, identifying three areas where TVA assistance would be desirable. The chart is complete in draft form and has been internally reviewed.

An ORNL representative was invited to meetings at Brown's Ferry and at the Sequoyah Simulator Center. Also, ORNL was invited to a planning session with TVA and Packard, Lowe, and Garrick. Specifics are described in the Operations and Systems section, below.

PHENOMENOLOGY:

A memo on ATWS for GE BWR's as a result of reviewing GE proprietary reports at NRC was completed. ATWS is being considered as an additional sequence for the Brown's Ferry SASA Program.

Several MARCH runs were prepared for the cases of with and without ECCS injection following the complete station blackout.

The sequences of events following the complete station blackout for the cases of with and without ECCS injection were reviewed.

The CORRAL-02 code at the BNL CDC-7600 computer facility was installed.

The FRAP-T4 code was obtained from the Argonne Code Center.

RELAP4/MOD6 verification of SASA sequence, for both with and without ECCS injection, were started.

OPERATIONS AND SYSTEMS

Liason has been made with the Probabilistic Risk Assessment (PRA) team for Brown's Ferry Unit 1. This team represents a joint effort between TVA and the firm of Packard, Lowe and Garrick (P,L,&G). The objectives of the study are:

- *to provide a probabilistic risk assessment for Brown's Ferry Unit 1*

— *to provide a management tool for*

a) identification of beneficial safety improvements

b) assessment of improvements recommended by others

— *to teach TVA personnel how to conduct probabilistic risk assessments*

— *to enhance the station staff response to abnormal plant behavior*

A significant portion of the PRA study involves the sequencing of events and systems analysis of the individual plant systems. A SASA team member has been invited to participate in these two efforts. This affords the opportunity to avoid duplication of effort in the basic gathering of information concerning the design, operation, and accident response of the Brown's Ferry Unit 1 systems. A list of the 60 accident initiating events developed by TVA and P,L,&G for the PRA analysis was obtained.

A one-half day visit of a SASA team member to the TVA Power Operations Training Center in company with TVA and P,L,&G personnel provided familiarization with the Control Room layout for Brown's Ferry Unit 1. Copies of the four-volume set of Operator Training manuals for Brown's Ferry were obtained and are now available at ORNL. These training manuals provide systems information in far greater detail than does the FSAR.

A follow-on 1 1/2 day visit to the Brown's Ferry Station provided opportunity for familiarization with the actual plant layout. Copies of the three laminated poster special operator aids which have been affixed to the Unit 1 control panels were obtained and are now available at ORNL. These are:

a) RHR Crosstie Capability - a layout of the RHR systems showing the interconnections between the three Brown's Ferry units.

b) Vessel Level Instrument Ranges - a reference drawing showing the important items within the reactor vessel with heights relative to vessel zero. The various reactor vessel water level indication system ranges and the inter-relation between them is also shown.

c) Unit 1 Relief Valve Discharges - a diagram showing the pressure suppression pool and the location within of the reactor vessel relief valve tailpipe discharges. Similar diagrams applicable to Unit 2 and Unit 3 were also obtained.

A compendium of information concerning the High Pressure Coolant Injection (HPCI) system has been assembled and is in distribution. A similar document for the Reactor Core Isolation Cooling (RCIC) system is in preparation.

FISSION PRODUCT RELEASE AND TRANSPORT

a) Rate of Release from Fuel

The literature is being reviewed with the objective of obtaining a release rate function for each of the seven fission product groups in the general form:

(release rate) = (burst release) + f(t).

The burst release occurs at the calculated time of cladding failure (which must be provided) and the previous history of the rod, primarily its heat rating. The value of the second term, f(t), depends on the temperature at time, t, and probably the rod heat rating as well. The data base for determination of the release functions is quite small (except for noble gases), therefore we expect that a relatively simple empirical expression will suffice.

b) Accident Sequence Input Information

Progress for the month of November has centered upon the task of identifying the necessary input information required for fission product transport during core meltdown, and accessing the available computational tools for the purpose of obtaining this input (i.e., thermal-hydraulics, chemical species concentrations, deposition sites, etc. in the vessel and containment). A letter was written that details these input requirements.

At the present time, the MARCH code seems to be the best tool for providing the above input information because of its availability, low computational cost, and sufficiently detailed output information. This code breaks the reactor vessel into 500 core control volumes (10 radial channels and 50 axial nodes), 4 volumes in the upper vessel head, and 5 volumes in the lower vessel head. Sufficient control volumes are also included for the drywell and wetwell.

c) Auxiliary Systems Interactions

The following Brown's Ferry systems have been examined (in a preliminary fashion) to determine possible interactions with the fission product pathways during an accident: primary containment the isolation valves, secondary containment basic ventilation system and standby gas treatment system, the HPCIS isolation valves, the RHR heat exchangers, the liquid and gaseous radwaste systems, the reactor building closed cooling water system, the raw cooling water system, the RHR service water system, the heating, ventilation and air-conditioning systems, the condensate storage system.

A list of questions have been formulated regarding the behavior of these systems during a station blackout. A meeting has been scheduled with TVA systems personnel to discuss these items.

d) Fission Product Transport Calculation

Work during November has centered around the fission product transport (f.p.t.) in the primary coolant system. The main thrust has been a literature search of reactions and mechanisms of f.p.t. and retention in the pressure vessel. Within the last week, the MARCH mock-up of BF-SB1 has been accomplished. We have already begun the task of superimposing the species transport on the MARCH thermal-hydraulic results.

We have decided to dispense with the projected initial hand-calculation and proceed directly to employing a modified MARCH as the framework for the calculated results.

SIMULATION MODEL DEVELOPMENT

Development of the BWR Loss of Heat Sink Simulation continued. The calculation of reactor vessel hydrogen/steam temperature, partial pressures, and flow through the relief valves was programmed and debugged. An improved calculation of cooling of uncovered fuel was incorporated. Equations were developed for the containment model.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

ACTIVITY: ORNL #41 89 55 10 6 (189 #B0120/NRC #60 19 11 04 1)

TECHNICAL HIGHLIGHTS:

Mr. T. Furuta of the JAERI Laboratories at Tokai, Japan, visited ORNL on November 3 for discussions on Zircaloy oxidation and deformation research. The JAERI deformation research program is quite similar to the MRBT program with respect to fuel pin simulator design and test conditions. In spite of the close similarities, the JAERI bundle test results are not in very good agreement with MRBT bundle results. As yet the cause of the disagreement has not been ascertained, although some design features and test conditions have been identified as contributing to the differences.

One single rod heated shroud test was performed during this reporting period. Preliminary results indicate burst conditions of 1028°C and 1065 kPa; the heating rate during the time of deformation was $\sim$5 K/s. The burst strain was about 68%, considerably greater than would be expected by the audit curve proposed in NUREG-0630 for low ramp rates.

Final assembly of the B-4 test array is in progress. The assembly will be moved to the test facility about mid-December, where it will be installed in the test vessel. It is anticipated that the burst test will be performed in mid-January.

A. W. Longest, of the MRBT staff, visited the B&W Alliance Research Laboratories, Alliance, Ohio, on November 20 to discuss and give approval of the B&W Technical Plan for performing the flow tests on the B-5 test array. Since budgetary constraints precluded exercising an option for detailed velocity profiles (using a laser doppler velocimeter system), we gave B&W permission to seek financial support elsewhere for these tests on the condition that if support is obtained the test data must be published in the open literature. The bundle is currently scheduled to be returned to ORNL in late April 1981, at which time it will be destructively examined to obtain detailed deformation data.

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:

PWR In-core Vibration Detection Study. Neutron kinetic calculations to determine the sensitivity of ex-core neutron detectors to in-core vibrations are continuing. Trial computer runs with the JPRKINETICS code show good convergence properties. Several modifications have been performed to adapt the code to treat vibration problems. A 3-D plotting routine has also been installed as an aid in understanding and visualizing the JPRKINETICS results.

Loose-Parts Monitoring. A NUREG report is being drafted in which our studies and assessment of loose-part detection, location, and characterization methodologies used by the nuclear industry are summarized.

BWR Stability Monitoring. A NUREG report is being drafted to summarize our assessment of neutron noise as a means of monitoring BWR stability.

Non-Neutron Signals for Safety Assessment. We are preparing a letter report that provides justification for using non-nuclear signals in the performance of noise diagnostics. Initial emphasis will be placed on primary pressure and temperature signals, and the report will outline our plan of attack for assessing the feasibility of diagnosing particular safety problems with the aid of these signals.

Primary Coolant Inventory Monitoring. A study has been initiated to determine the feasibility of the method for continuous monitoring of the primary coolant inventory in PWRs, using existing plant sensors and instrumentation. In connection with this study, the Tennessee Valley Authority has agreed to provide us with information on the instrumentation installed at the Watts Bar Plant.

Meetings: We presented a summary of program accomplishments at the NRC Eighth Water Reactor Research Information Meeting held on October 29, 1980 at the National Bureau of Standards in Gaithersburg.

DNF/bjm

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 0! 2

TECHNICAL HIGHLIGHTS:

There is no report for the month of November.

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 November, the staff of the Nuclear Safety Information Center (a) processed 417 documents, (b) responded to 64 inquiries (of which 44 involved the technical staff and 10 were for commercial users), and (c) made 14 computer searches. The RECON System, which now has over 200 remote terminals, reports that the NSIC data file was accessed 199 times between October 1 to 31 making it the sixth most utilized of the 26 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 19 visitors and participated in 7 meetings.

Several 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 November, we received 53 foreign documents (2 French, 43 German and 8 UK). In accordance with the arrangements effective January 1, 1979, a copy of each of these have been sent to R. S. Scott (DDC) for microfiche processing. In addition, the foreign language documents were reviewed for translation (see letter of November 26, 1980, to H. H. Scott, RSR).

During the month of November, NSIC's Selective Dissemination of Information (SDI) was providing service to a total of 366 users, including 4 new users. It should be noted that consistent with our FY-81 budget, "cost-recovery" services have been eliminated so that as "paid" subscriptions expire they are not renewed. However, this and other services to NRC, DOE and their subcontract personnel are not affected.

All technical articles for *Nuclear Safety* 22(2) were completed and mailed to NRC, DOE and TIC on November 21st. The "current events" material (covering events which occurred during September and October, for *Nuclear Safety* 22(1) were completed by September 21 (except for the data on operating power reactors which was not yet available from NRC). Most technical articles for *Nuclear Safety* 22(3) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 21(6) were received from the printer (via TIC) on November 12th.

TABLE 1 RECON DATA BASE ACTIVITY FROM 10-01-80 TO 10-31-80
(24 OPERATING DAYS)

DATA BASE IDENT.	DATA BASE NAME AND SUPPORTING INSTALLATION IDENTIFICATION	NO. OF SESSIONS	NO. OF EXPANDS	CITATIONS PRINTED
EDB	(TIC) DOE ENERGY DATABASE	4360	6076	139645
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	735	1011	8906
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	332	922	20108
EMI	(EMIC) ENV. MUTAGENS INFO.	235	407	9669
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	209	242	2858
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	199	325	9607
GAP	(DOE) GENERAL AND PRACTICAL INFO.	155	247	918
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	135	175	1448
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	111	140	3741
ESI	(EIC) ENV. SCIENCE INDEX	90	159	1361
EIA	(EIC) ENERGY INFO. ABSTRACTS	66	85	517
IPS	(TIC) ISSUES AND POLICY SUMMARIES	57	80	580
WRE	(WRSIC) WATER RESOURCE RESEARCH	49	103	620
SLR	(FRANKLIN) SOLAR DATA BASE	38	74	623
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	34	33	105
NES	(NESC) NATIONAL ENERGY SOFTWARE	29	42	123
PRD	(TIC/NRC) POWER REACTOR DOCKETS	29	54	430
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	28	17	3
API	(API) AMER. PETROLEUM DATA BASE	27	54	350
TUL	(U. TULSA) TULSA DATA BASE	27	54	707
NRC	(LC) NATIONAL REFERRAL CENTER	26	35	9
CIM	(DOE) CENTRAL INVENTORY OF MODELS	22	34	26
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	17	38	-
RSI	(RSIC) RADIATION SHIELDING INFO.	13	9	61
ARF	(EMIC/ETIC) AGENT REGISTRY FILE	10	2	-
RSC	(RSIC) RADIATION SHIELDING CODES	7	1	16

PROGRAM TITLE: Operational Aids for Reactor Operators

PROGRAM MANAGER: G. F. Flanagan

ACTIVITY NUMBER: ORNL #41 88 55 02 8 (189 #B0438-9)NRC #60 19 50 01 0

TECHNICAL HIGHLIGHTS:

Work continued during the month of November as staff members reviewed several proposals for work that would contribute to the operational aids program.

R. A. Kisner and G. F. Flanagan presented a paper at the 1980 IEEE Symposium on Nuclear Power Systems at Orlando, Florida, November 7, 1980. The paper, "A Systems Approach to Defining Operator Roles," describes the elements of a systems approach for defining the role of the operating crew of a nuclear power plant.

G. F. Flanagan participated in the INEL program review at Idaho Falls, Idaho. He presented a summary of the work underway in the Operational Aids for Reactor Operators Program.

R. A. Kisner attended the IEEE Working Group 5.5 Human Factors meeting in Washington, D.C., November 18, 19. The Working Group is divided into three task groups: Human Performance Evaluation, Program Plan, and Colors/Codes/Symbols. The program plan task group is modifying Mil-H-46855B to provide a guide for a systematic, reproducible, and tractable evaluation of human performance in Nuclear Power Plants. Lewis F. Hanes of Westinghouse is chairman of WG5.5 and Leonard C. Pugh, Sr. of General Electric is head of the Program Plan task group.

No new contracts were let this month, and work at ORNL was reduced to minimum because of unavailability of FY81 funds. This situation will be rectified by January 1, 1981.

Becker, Block, and Harris, Inc. has completed Phase I of their work. An interim report has been delivered to ORNL for review. In the first phase of work guidelines for grouping plant systems were established. Several structural arrangements for a heirarchy of plant systems have been reviewed for both normal and abnormal operation. As a result of this review a multilevel structure was chosen. A PWR system was decomposed and fitted to this multilevel structure. Then the progression of an operational transient was studied using this structure.

Technology For Energy, Inc. has begun expanding their work on defining the role of the nuclear power plant operator. They are soliciting help from utilities, procedure writers, plant designers, and plant operators in enlarging the scope of their original work (published as NUREG/CR-1772).

Lund Consulting, Inc. has submitted for review a final draft of their report, "Understanding Human Behavior in Off-Average Conditions." This report, if accepted, represents the completion of their contract.

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: W. G. Craddick

ACTIVITY NUMBER: ORNL #41 89 55 10 3 (189 #B0401)/NRC 60 19 11 01

TECHNICAL HIGHLIGHTS:

Task 1: Single-Rod Loop Testing — This task has been completed. No further work is planned at this time.

Task 2: Data reduction is continuing for the steady-state film boiling tests (THTF Test Series 3.07.9) and the second Small Break LOCA Heat Transfer Test Series (THTF Test Series 3.09.10).

FORTRAN codes written for use in the analysis of the first small break test series have been modified to utilize the new format of the data from the second test series. This includes processing of the special FRS thermocouples and shroud box thermocouples that were added to the facility. Small Break LOCA Heat Transfer Test Series I data tapes are being prepared for the INEL data bank.

All ORTCAL-Part 1 runs for the refurbished THTF Bundle 3 have been completed. ORTCAL-Part 2 regression runs have also been completed. The initial run of ORTCAL-Part 3 has also been made. ORTCAL is a computer program which determines fuel rod simulator (FRS) thermal properties from data taken with the rod bundle installed in the THTF. Generation of FRS thermal properties should be completed in December. Documentation of several computer programs used in FRS analysis is continuing and is 10% complete.

An interim report describing results of analysis of THTF Test 3.06.6B (Film Boil[illegible] Upflow) was completed and distributed to NRC and BDHT Review Group [illegible]bers. The results of analysis performed on the test indicate that (1) the Dougall-Rohsenow correlation generally overpredicts the heat transfer, (2) the Groeneveld 5.7, Groeneveld 5.9, and Condie-Bengston IV correlations perform better than the Dougall-Rohsenow correlation, and (3) the Groeneveld-Delorme correlation is overly conservative in its predictions of film boiling heat transfer. Results of analysis for Test 3.06.6B concur with those found in analysis of THTF Test 3.03.6AR (Film Boiling in Upflow) and the preliminary analysis of steady-state film boiling tests (Test Series 3.07.9) recently completed at ORNL. An abstract covering analysis of Tests 3.06.6B and 3.03.6AR has been submitted to the 20th National Heat Transfer Conference (August 1980).

Study of Test Series 3.07.9 is underway and is approximately 15% complete. An abstract describing analysis performed on these steady-state film boiling tests has also been submitted to the 20th National Heat Transfer Conference.

Small Break LOCA Heat Transfer Test Series II has been completed. A preliminary survey of the data indicates that 17 of the 23 requested tests were successfully completed. Testing at pressures above 1200 psia was not done due to the possibility of overpressurizing the outer seal cooling system. Overpressurization was possible because the inner O-rings were damaged during the early portion of the testing, which allowed communication between the test section and outer seal coolant system. Leakage problems extant during the tests were mitigated by isolating the test section annulus and allowing pressure between the annulus and test section to equalize before heat transfer data was taken.

Task 3: THTF Operations — Additional modifications were made to the facility during November to improve the instrumentation for the second Small Break LOCA Heat Transfer Test Series (THTF Test Series 3.09.10). At present, some leakage occurs at both the top of the shroud box and at the inner O-ring seal areas. Additional instrumentation was installed in an attempt to measure these leakage rates.

THTF Test Series 3.09.10 was successfully completed on November 25. Plans are presently being completed to return the facility to the original 4-in. pipe configuration and to shut down the system.

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 was completed for two additional groups of PWR operators with the seven PWR accident events previously used. The proposed BWR exercises have been reviewed with TVA staff, and a tentative experimental design has been developed by General Physics. This design will be reviewed by Memphis State and ORNL. BWR data collection will begin on January 12.

Memphis State has continued with the field data collection effort. A total of 18 visits to 13 power plants (11 PWR and 2 BWR) have been made. During this month additional data has been collected on incidences of rod drops, feedwater failures, inadvertent safety injections, and small LOCAs.

Simulator Response Characteristics

Franklin Research Center has continued with collection and analysis of information on specification and verification of simulator response in non-nuclear simulators. From the literature search and personal contacts some 75 documents ranging from acceptance test plans to research papers have been selected as particularly relevant and have been summarized.

General Physics has made contact with Gould, EAI, CE and B&W during this month. As initial comparisons between methods/practices in nuclear vs. non-nuclear industries are made, a basic difference first noted is that the nuclear industry does not employ a highly structured approach to the specification of simulator requirements. The verification process is in general quite similar, though in some cases verification may be more quantitative in non-nuclear applications, and the updating requirements may be more rigorous.

PROGRAM TITLE: Subcritical Reactivity Monitoring by the Californium-252 Source Driven Neutron Noise Method

PROGRAM MANAGER: C. W. Ricker

ACTIVITY NUMBER: ORNL #41 89 55 13 2 (189 #B0451)/NRC 60 19 11 01

TECHNICAL HIGHLIGHTS:

The JPRKINETICS code has been successfully modified to perform detector response calculations up to 10 kHz on the full shutdown ZIMMER core. Preliminary results were reasonable and a more comprehensive set of calculations have been initiated. Monte carlo calculations have also been started to determine the spatial distribution of different length fission chains in the reactor.

CWR/bjm

PROGRAM TITLE: Zircaloy Fuel Cladding Creepdown Studies

PROGRAM MANAGER: D. O. Hobson

ACTIVITY NUMBER: ORNL #41 89 55 11 7 (189 #B0124)/NRC # 60 19 11 04 1

TECHNICAL HIGHLIGHTS:

There is no report for the month of November.

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64-65. Technical Information Center

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