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for the
DEPARTMENT OF ENERGY

NRC Research and Technical Assistance Report

CONTENTS

# ABSTRACT

Highlights of technical progress during December 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 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:

Two in-core guide tube probes have been installed in a test bundle for the steam-water test facility. One is a CCTF-II style probe of the new single electrode design and the other is a standard flag probe. It is planned to conduct an extensive series of tests covering a range of temperature, pressure, and flow conditions. These will serve as the basis for final verification of the void fraction and velocity algorithms which have been delivered to the Japanese for SCTF.

Component machining for the CCTF-II in-core probes and the PKL-II upper plenum probes has been completed. The assembly and brazing operations are now in progress. The main effort is directed toward meeting the February shipping date for the CCTF-II probes.

It has been tentatively decided to ship eleven standard flag probes and one single electrode combination probe for installation in the CCTF-II core. The combination probe offers several advantages over the standard probe, however, the electronics have not been fully tested at this time. If satisfactory operation is obtained in CCTF-II, then the new design will be an option for use in later facilities.

Other December activities have included low velocity EP probe tests, lab tests on single electrode electronics, and preparation for the SCTF-II/CCTF-II design meeting with JAERI in January.

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 draft interim report on data taken with a heated thermocouple (HTC) sensor during Thermal Hydraulic Test Facility (THTF) Test 3.07.9 was completed. The draft was submitted to NRC/RSR and to Combustion Engineering for review before release.

Data obtained with the HTC during THTF Test 3.07.10 (bundle boiloff and reflood) was copied and transmitted to Combustion Engineering. Bench testing with a six-junction HTC array was completed. A splash shield was mounted over the array and the assembly was installed in the high-pressure test vessel. Nearly steady-state experiments were performed with the assembly in saturated liquid water and steam at pressures ranging from atmospheric to 10.3 MPa (1500 psia). Information on the vertical temperature gradient in the pressure vessel was obtained with a four-element TC array. The device survived the tests and generally performed well.

The INEL HTC electronics package was completed and tested at room temperature conditions. The system's performance was apparently as specified. It is planned to test the electronics in conjunction with a high temperature experiment.

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:

Further modifications to the NSPP aerosol filter sampling systems were made to enable operation in a steam environment. Tests have demonstrated that these filter samplers are now fully operational. Additional testing has continued to determine the response of the full NSPP system under high-temperature, high-pressure steam environments. The first $U_3O_8$ aerosol/steam test is now scheduled for January 1981.

CORE MELT:

Efforts to scale up the small core-melt aerosol experiments using the available 50 kW-250 KHz RF induction generator and the split-wall water-cooled copper crucible skull-melting technique have resulted in our first successful eutectic melt at the level of 0.66/kG $UO_2$ and 0.21 Kg of ZR-4.

This experiment was conducted in a helium cover gas with a cluster of 14 PWR size ZR-4 capsules of about 8 cm in length, each with a 5 mm vented end cap. Zirconium oxide was added to the mixture to form the eutectic both as powdered $ZrO_2$ around the rods and as the material of construction for the crucible, replacing the sintered preformed $UO_2$ crucible which had been used in previous experiments. Since the purpose of the experiment was to test the feasibility of this step in the scale-up toward a maximum 5 Kg objective, no additive aerosol-generating core-melt components were contained in the mixture. Aerosols were observed to be released however, presumably metallic tin from the Zircaloy, which was collected on a high-efficiency filter for later analysis.

Following this experiment, a duplicate eutectic melt will be attempted in which adsorbed water will be provided in a layer of powdered $ZrO_2$ to provide an oxidizing environment for the molten Zircaloy. The released hydrogen will be collected as a measure of the extent of metal water reaction in the meltdown.

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Publication: The paper, "Impact of Secondary Effects on the Reduction of Fission Product Source Terms in Class IX Reactor Accidents," written for the special session sponsored by EPRI and presented at the ANS-ENS Conference has been accepted for publication in Nuclear Technology.

FASTCRI-III:

No experiments were performed in the FAST/CRI-III test facility this month. An "action plan" for conducting an uncertainty analysis using the UVABUBL code developed at the University of Virginia to model the bubble expansion phase of the FAST under-water and under-sodium experiments was transmitted to NRC for comment and approval. The uncertainty analysis described in the action plan, however, will be done by running UVABUBL on the ORNL computer system.

Work continues on developing an analytical adiabatic gas vibration model to simulate some aspects of submerged bubble oscillation. Code test results for a non-fluid bypass version have demonstrated adequacy of the numerical procedures.

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:

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

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: R. P. Wichner

ACTIVITY NUMBER: ORNL # 41 89 55 10 8 (189 #B0127)/NRC # 60 19 11 04 1

TECHNICAL HIGHLIGHTS:

The objective of this new program is to investigate fission product release from irradiated LWR fuel at temperatures up to melting (∿2600°C) in steam. Release data at high temperatures (1600-2600°C) are needed for correlation of existing data at lower temperatures with data obtained from tests at 1500-2800°C using simulant fuel, which were conducted in Germany.

The program will be conducted in two phases. A higher temperature furnace will be developed and incorporated into existing equipment in Phase 1 to perform tests in the 1600-2000°C range. Concurrently, a new experimental system (Phase 2) will be developed with capabilities for testing up to ∿2600°C and for improved collection and identification of fission product species.

Accomplishments to date have included

1. decontamination and testing of the existing equipment and hot cell;

2. preliminary design of a new furnace, using an induction-heated tungsten susceptor, $ZrO_2$ furnace tubes for specimen containment, and low density ceramic insulation to reduce heat loss;

3. procurement of special materials and fabrication of an apparatus to verify this furnace concept;

4. planning and testing procedures for an improved data collection/processing system; and

5. acquisition of suitable unirradiated and irradiated fuel material for use in the initial tests.

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. L. B. Dufour of KEMA, The Netherlands, visited ORNL on December 15 to discuss the BROS program. This work was originally concerned with the evaluation of nozzle corner flaws by linear-elastic fracture mechanics and has now developed to BROS-II which involves development of elastic-plastic fracture mechanics techniques in thick-walled steel construction.

On December 17, R. D. Cheverton and G. D. Whitman visited NRC in Silver Spring, MD, to meet with the Vessel Integrity Review Group to discuss test options for the next thermal shock experiment — TSE-6, development of the overcooling transient analysis code, and pressurized thermal shock experiments.

G. D. Whitman met with M. Vagins of NRC and T. U. Marston of EPRI on December 18, in Silver Spring, MD, to discuss plans for an information meeting in Oak Ridge in the spring of 1981.

Task 3: Irradiation Effects — Irradiation of the second capsule of the Fourth HSST Irradiation Experiment is continuing. We expect to disassemble the first capsule in January.

The 1T compact specimens for the third capsule have been fatigue precracked and final inspection, including compliance measurements, is in progress. The other specimens, CVN and tensile, are ready for capsule assembly.

Testing of the remaining Charpy specimens from the Second and Third 4T Irradiation Experiments was completed. Lateral expansion and fast-load-drop measurements are in progress. We have found that fracture appearance measurements (percent brittle fracture) are not practical for these submerged-arc welds. In the fracture transition region brittle fracture areas are scattered depending primarily on position of the coarse-grain region of the weld beads.

Work on installing the tensile machine in the hot cell is in progress.

Task 4: Thermal Shock — Fracture-mechanics calculations were made in connection with a possible future thermal-shock experiment, TSE-6. The current purpose of TSE-6 is to achieve very deep penetration of a long axial flaw and to observe the effect of nonlinearities associated therewith. Our recent analysis indicates that appropriate test conditions will include a test cylinder similar to that used for TSE-5 and -5A with a thinner wall (76 mm instead of 152 mm), essentially quench-only toughness properties (similar to TSE-5), and a severe thermal shock using liquid nitrogen (similar to TSE-5A). Under these conditions a shallow crack ($a/w \cong 0.1$) would jump, in a single step, about 90% of the way through the wall, providing another opportunity to investigate dynamic effects at arrest.

Preparation of the OCA code, TSE-5 and -5A topical reports and a SMiRT 6 paper covering TSE-4, -5 and -5A are continuing.

Task 5: Simulated Service Tests — Testing of Charpy V-notch impact and tensile specimens from the final trial weld for intermediate test vessel V-8A have been completed. Two J specimens and a series of drop weight specimens of this material have also been tested. All tests in Phase I of the Babcock and Wilcox Company (B&W) subcontract have been completed except for two J specimens.

The trial weld (V842) is a 152-mm-thick submerged-arc weld made in the Barberton shops of the B&W Nuclear Equipment Division. The materials and welding parameters for this weld were essentially the same as for preliminary trial weld V822, except that a lower heat input was used for the trial weld in the hope that the transition temperature would be diminished slightly. Thirty-three Charpy V-notch impact specimens of this weld were tested in the temperature range from −73°C to 260°C. Twenty-one of these tests were in the range from 66°C to 121°C, in which the onset of upper shelf lies. On the basis of 100% shear fracture appearance, the upper shelf is attained between 93°C and 99°C. On the basis of this information we decided that 120°C would be an appropriate upper shelf temperature for the testing of tensile and J specimens from the trial weld.

Average values of trial weld test results at 120°C are as follows.

| Test | Value |
|---|---|
| Impact energy (J) | 57.6 |
| Tensile: | |
| Yield strength (MPa) | 419.8 |
| Ultimate strength (MPa) | 518.8 |
| Elongation (%) | 24.3 |
| Reduction of area (%) | 53.5 |
| J-integral: | |
| $J_{Ic}$ ($kJ \cdot m^{-2}$) | 63.3 |
| Power law parameters - $J = C(\Delta a)^n$ | |
| $C[kJ \cdot m^{-2}/(mm)^n]$ | 112.224 |
| n | 0.407925 |

These values meet the specifications for the preparation of vessel V-8A. The J-R curve (fit to data in the range $0.15 \lesssim \Delta a \lesssim 13$ mm) lies within the range of 1.6 T data reported by Frank Loss for NRL for the irradiated high copper welds from the HSST program.

Drop weight tests between −40°C and −12°C resulted in failures at all temperatures. Tensile tests of A533 base metal subjected to the post-weld heat treatment given the weld indicated no adverse effect on the base metal.

Almost all data, shop details, and other documents requiring Union Carbide approval before proceeding with the job have been submitted by B&W. Vessel V-8A and associated materials were shipped to B&W. Informal agreement with B&W was achieved on all aspects of the contract for vessel preparation, and the DOE-approved contract document has been sent to B&W.

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: A final letter report was written and submitted to NRC on ORNL calculations of postulated FSV loss of forced convection (LOFC) accidents followed by firewater cooldown (FWCD) for core support thermal stress evaluations. Calculations were made for both the full-power equilibrium core and the cycle 2 core at 72% of full power. Summary results were presented in the report, and detailed outputs from the ORECA code were sent to LASL for use as inputs to their stress analysis codes. New thermal conductivity data received from GA for the core support block PGX graphite was factored into the most recent analysis.

Code Documentation and Development: Work was begun on ORECA code modifications to permit more flexibility in the input specifications for core bypass flow fractions. This applies mainly to the version of ORECA used for comparisons with FSV transient data. A review and update of the HTGR computer codes and other information stored at the ORNL computer center was begun. This "housecleaning" chore is necessary due to recent hardware changes which require transfer of all programs from private disk packs to the new mass storage device.

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 programmed the microcomputer to make linear corrections to the magnitude and phase readings to make the readings on a standard in the field be the same as the standard read back in the laboratory. Later tests run on the ModComp have shown that the linear correction gives about the same results as the vector corrector.) The new system was used for the in-service inspection that we performed at the Point Beach Power Plant on December 13 and 14. The inspection was performed ahead of schedule, and took only three days. The correction program seemed to work quite well and the retest on the ORNL standard immediately after we removed the probe from the steam generator gave excellent results. However, it appears that we are getting a 6- to 8-mil apparent decrease in wall thickness as the probe enters the tubesheet throughout the steam generator. Since this shift did not occur on the ORNL standard test, we have concluded that the cause may be a material difference in the steam generator tubing or the tubesheet itself.

We have written a test program to allow the ModComp to process the data from the Ginna and Point Beach inspection trips. The Ginna data also shows a decrease in apparent wall thickness of 2 to 3 mil, not as great as that from the Point Beach experiments. We shall do further processing of the data and investigate the material difference problems to resolve these differences.

Some preliminary tests have shown that if we include the new linear offset correction scheme in the computer programs, it is possible to apply the coefficients generated from the calibration of one coil to another coil of the same type. This means that only one calibration will have to be run for each type of coil and the others can be adjusted to it. Also, post calibrations can be run after a test even if the actual coil used has been discarded.

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: F. B. K. Kam chaired the U.S. Program Committee Meeting for the Fourth ASTM-EURATOM International Smyposium on Reactor Neutron Dosimetry, Dec. 9-10, 1980 at Lynchburg, Virginia. Dr. W. Schneider, co-chairman of the EURATOM committee, was present to present their views. The format and topics for the Symposium were discussed, and the minutes distributed to all committee members a week after the meeting. The announcement for the "Call for Papers" will be made through ANS, ASTM, RSIC, and the EURATOM newsletters.

Discussions of the IAEA Specialists' Meeting, CAPRICE 82, were held with Dr. Schneider and EPRI (by telephone). The meeting has been tentatively set for March 15-19, 1982 at Palo Alto, California. The minutes for these discussions was distributed by W. N. McElroy of HEDL. Dr. Schneider also announced that the proceedings of the 3RD ASTM-EURATOM Symposium and the previous IAEA Specialist Meeting, CAPRICE 79, have finally been published.

Task 2: Benchmark Fields -

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

The compilation and evaluation of experimental and calculational data from the PCA "Blind Test" is being continued for the preparation of the NUREG report.

B. ORR-PSF -

Analysis of the neutron characterization and the testing of the metallurgical specimens from the SSC-1 capsule is continuing. The cumulative irradiation history and thermal characteristics for the simulated pressure vessel capsule (SPVC) through the month of September are given in Table 1.

C. BSR - SDMBF -

No report for the month of December.

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

The ASTM Guide on the Application of Neutron Transport Methods is available in draft form. Modifications based on an internal ORNL review are included. One more internal ORNL review is planned for the first week of Jan. 1981 before the draft version is sent to the appropriate ASTM committee members for further review.

Table 1. Irradiation and Temperature Data Through September 30, 1980

Data for PSF Specimen Set OT
Hours of Irradiation Time = 2780.46
Megawatt Hours of Irradiation = 82857.71

| Thermocouple | Hours of Irradiation | | | | | Average Temperature | Standard Deviation |
|---|---|---|---|---|---|---|---|
| | T<270 | 270<T<280 | 280<T<296 | 296<T<306 | 306<T | | |
| TE101 | 47.23 | 12.65 | 2698.97 | 21.38 | 0.00 | 289.71 | 2.03 |
| TE102 | 46.30 | 8.40 | 2682.86 | 42.66 | 0.00 | 291.59 | 1.47 |
| TE103 | 45.83 | 7.55 | 2726.85 | 0.00 | 0.00 | 289.21 | 1.14 |
| TE104 | 41.76 | 7.95 | 2722.09 | 8.44 | 0.00 | 292.11 | 1.00 |
| TE105 | 4[illegible].35 | 8.65 | 2726.25 | 0.00 | 0.00 | 285.66 | 1.17 |
| TE106 | 41.54 | 8.62 | 2730.06 | 0.00 | 0.00 | 288.63 | 1.06 |
| TE107 | 46.37 | 288.37 | 2445.51 | 0.00 | 0.00 | 281.79 | 1.44 |
| TE108 | 47.86 | 10.48 | 2712.79 | 9.06 | 0.00 | 290.07 | 1.78 |
| TE109 | 47.88 | 11.47 | 2713.08 | 7.78 | 0.00 | 289.52 | 1.74 |
| TE110 | 44.69 | 9.81 | 2712.80 | 12.95 | 0.00 | 290.81 | 1.19 |
| TE111 | | | | | | | |
| TE112 | | | | | | | |
| TE113 | 40.71 | 8.13 | 2731.33 | 0.04 | 0.00 | 290.18 | 1.59 |
| TE114 | 50.06 | 10.85 | 2719.30 | 0.00 | 0.00 | 288.67 | 1.68 |
| TE115 | | | | | | | |
| TE116 | 49.08 | 9.01 | 2723.14 | 0.00 | 0.00 | 290.06 | 1.02 |
| TE117 | 46.21 | 8.11 | 2720.11 | 5.29 | 0.50 | 291.73 | 1.05 |
| TE118 | 48.80 | 10.31 | 2721.18 | 0.00 | 0.00 | 286.10 | 1.13 |
| TE119 | 45.87 | 8.19 | 2726.17 | 0.00 | 0.00 | 286.75 | 1.10 |
| TE120 | 48.54 | 175.42 | 2556.31 | 0.00 | 0.00 | 282.46 | 1.66 |

Data for PSF Specimen Set 1/4T
Hours of Irradiation Time = 2780.46
Megawatt Hours of Irradiation = 82857.71

| Thermocouple | Hours of Irradiation | | | | | Average Temperature | Standard Deviation |
|---|---|---|---|---|---|---|---|
| | T<270 | 270<T<280 | 280<T<296 | 296<T<306 | 306<T | | |
| TE201 | 46.60 | 11.32 | 2719.65 | 2.66 | 0.00 | 290.23 | 1.63 |
| TE202 | 46.87 | 10.68 | 2722.50 | 0.17 | 0.00 | 289.05 | 1.18 |
| TE203 | 46.20 | 8.83 | 2725.23 | 0.00 | 0.00 | 288.44 | 1.19 |
| TE204 | 44.32 | 6.93 | 2728.66 | 0.33 | 0.00 | 289.26 | 0.85 |
| TE205 | 44.15 | 11.18 | 2724.92 | 0.00 | 0.00 | 286.03 | 1.10 |
| TE206 | 42.26 | 11.15 | 2726.83 | 0.00 | 0.00 | 286.66 | 0.97 |
| TE207 | 45.91 | 32.89 | 2701.44 | 0.00 | 0.00 | 282.63 | 1.13 |
| TE208 | 46.26 | 10.38 | 2722.76 | 0.83 | 0.00 | 288.56 | 1.63 |
| TE209 | 46.60 | 11.20 | 2722.43 | 0.00 | 0.00 | 289.16 | 1.49 |
| TE210 | 46.04 | 13.89 | 2720.32 | 0.00 | 0.00 | 286.84 | 0.76 |
| TE211 | 47.13 | 14.36 | 2718.77 | 0.00 | 0.00 | 285.08 | 0.78 |
| TE212 | 41.76 | 5.28 | 2733.18 | 0.00 | 0.00 | 290.76 | 1.17 |
| TE213 | 42.09 | 5.37 | 2732.75 | 0.00 | 0.00 | 289.45 | 1.23 |
| TE214 | 46.87 | 8.84 | 2724.51 | 0.00 | 0.00 | 290.40 | 1.06 |
| TE215 | 47.04 | 12.07 | 2721.10 | 0.00 | 0.00 | 287.56 | 0.81 |
| TE216 | 46.16 | 10.52 | 2723.57 | 0.00 | 0.00 | 287.73 | 0.83 |
| TE217 | 44.82 | 6.21 | 2729.21 | 0.00 | 0.00 | 289.73 | 0.89 |
| TE218 | 44.71 | 11.80 | 2723.76 | 0.00 | 0.00 | 286.57 | 1.13 |
| TE219 | 43.25 | 8.96 | 2728.05 | 0.00 | 0.00 | 286.87 | 1.06 |
| TE220 | 44.33 | 100.18 | 2635.74 | 0.00 | 0.00 | 284.60 | 1.58 |

Table 1. (Cont'd)

Data for PSF Specimen Set 1/2T
Hours of Irradiation Time = 2780.46
Megawatt Hours of Irradiation = 82857.71

| Thermocouple | Hours of Irradiation | | | | | Average Temperature | Standard Deviation |
|---|---|---|---|---|---|---|---|
| | T<270 | 270<T<280 | 280<T<296 | 296<T<306 | 306<T | | |
| TE301 | 45.32 | 5.62 | 2690.13 | 39.18 | 0.00 | 290.24 | 1.27 |
| TE302 | 46.83 | 9.90 | 2723.52 | 0.00 | 0.00 | 286.37 | 1.00 |
| TE303 | 44.31 | 7.84 | 2728.07 | 0.00 | 0.00 | 287.38 | 1.05 |
| TE304 | 42.26 | 7.11 | 2730.27 | 0.58 | 0.00 | 291.03 | 0.79 |
| TE305 | 41.95 | 9.07 | 2729.19 | 0.00 | 0.00 | 287.67 | 1.04 |
| TE306 | 43.42 | 11.38 | 2725.41 | 0.00 | 0.00 | 286.56 | 0.93 |
| TE307 | | | | | | | |
| TE308 | 47.16 | 6.37 | 2726.71 | 0.00 | 0.00 | 289.04 | 1.19 |
| TE309 | 47.54 | 8.84 | 2723.87 | 0.00 | 0.00 | 287.60 | 0.97 |
| TE310 | 47.70 | 13.31 | 2719.26 | 0.00 | 0.00 | 285.68 | 1.05 |
| TE311 | 47.87 | 13.27 | 2719.14 | 0.00 | 0.00 | 286.08 | 1.19 |
| TE312 | 42.92 | 4.96 | 2732.18 | 0.17 | 0.00 | 288.71 | 1.00 |
| TE313 | 43.20 | 4.64 | 2730.74 | 1.67 | 0.00 | 290.01 | 1.16 |
| TE314 | 48.27 | 8.89 | 2723.08 | 0.00 | 0.00 | 289.20 | 0.99 |
| TE315 | 48.99 | 9.29 | 2721.98 | 0.00 | 0.00 | 285.11 | 0.91 |
| TE316 | 47.97 | 3.35 | 2728.93 | 0.00 | 0.00 | 287.51 | 0.84 |
| TE317 | 45.30 | 5.24 | 2729.70 | 0.00 | 0.00 | 290.77 | 0.99 |
| TE318 | 44.90 | 5.26 | 2730.07 | 0.00 | 0.00 | 289.15 | 1.09 |
| TE319 | 46.65 | 13.85 | 2719.72 | 0.00 | 0.00 | 285.16 | 0.92 |
| TE320 | 45.35 | 9.57 | 2725.30 | 0.00 | 0.00 | 287.41 | 1.53 |

PROGRAM TITLE: LWR Severe Accident Sequence Analysis (SASA)

PROGRAM MANAGER: M. H. Fontana

ACTIVITY NUMBER: (189 #B0452)/NRC 60 19 01 30

TECHNICAL HIGHLIGHTS:

OPERATIONS AND SYSTEMS:

A compendium of information concerning the Reactor Core Isolation Cooling (RCIC) system has been distributed to the SASA team members. This document and the previously distributed compendium for the High Pressure Coolant Injection (HPCI) system are being circulated for comment within TVA.

Additional reference material for Brown's Ferry was obtained from the Nuclear Safety Information Center. This includes:

(1) FSAR Amendments 40, 42 through 59, 62, 63, and 66

(2) Radiological Emergency Plan

(3) Responses to AEC questions dated October 12, 1971 and December 6, 1971

(4) Final Summary Report – Unit 1 Startup (and a similar document for Unit 2)

(5) TVA Draft Environmental Statement Supplements and Additions

(6) Emergency Core Cooling Systems Low Pressure Coolant Injection Modifications for Performance Improvement – Units 1 and 2 (dated Octover 1977 – and a similar document for Unit 3)

(7) Torus Support System and Attached Piping Analysis for the Brown's Ferry Nuclear Plant Units 1, 2, and 3 (September 1976)

(8) Torus Relief Valve Piping and Vent Header Support Modification (October 1975)

(9) Reactor Containment Building Integrated Leak Rate Test Report – Unit 1 (and a similar document for Unit 2)

(10) Safety Evaluation of the TVA Brown's Ferry Nuclear Plants Units 1, 2, and 3 (by the AEC, dated June 1972)

(11) Technical Specifications and Bases for Brown's Ferry Nuclear Plant Unit 1 (dated June 1977, and a similar document for Unit 2)

(12) Index of Licensee Event Reports (LERs) applicable to Brown's Ferry Unit 1 for the following systems:

Main Coolant System
HPCI System
RCIC System
Containment
Shutdown Cooling Systems
Electrical Power Systems (includes several cases of loss of offsite power)

(13) Index of "Responses to NRC Questions" concerning the same systems listed in item (12)

The material indexed in items (12) and (13) is available in complete form either as hard copy or microfiche at the Nuclear Safety Information Center. (Copies of the complete LERs concerning loss of offsite power during ice storms in early 1980 are being obtained now.)

It has been determined that there are 12 different approaches to loss of makeup coolant capability following a complete loss of AC power; these paths are shown in Figure 1 and described below:

<u>Paths to Loss of Coolant Makeup Capability</u>

1) Most likely. Boiloff to core uncovery starting with depressurized, normal level RV at time 5 hours. (The assumed time for loss of DC power may prove to be a significant parameter.)

2) Boiloff to core uncovery starting with depressurized, normal level RV when CST water exhausted, after injection of at least 135,000 gallons using RCIC. The amount injected may prove to be a significant parameter (i.e., there could be as much as 375,000 gallons available for injection by RCIC).

3) Boiloff to core uncovery starting with fully pressurized, normal level RV at time 5 hours.

4) Boiloff to core uncovery starting with fully pressurized, normal level RV when CST water exhausted, after injection of at least 135,000 gallons using RCIC.

5) Same as (1), except HPCI used in place of RCIC so that RV level cycles between 476 1/2" and 582" unitl DC power lost. Thus, RV level can be anywhere between these limits when boiloff starts.

6) Boiloff to core uncovery starting at time PSP level reaches +7" since HPCI injection capability lost due to high PSP temperature. Level between 476 1/2 and 582" and RV depressurized when boiloff starts.

7) Same as (2), except HPCI used in place of RCIC until CST water exhausted. (Probably not necessary to run this case) [ this case assumes operator takes action to prevent the shift in HPCI suction].

8) Same as (3), except HPCI used in place of RCIC until DC power lost at 5 hours. (Probably not necessary to run this case.)

9) Same as (6), except that RV is fully pressurized when boiloff starts. (Time at which PSP level reaches +7" may be significantly later for this case.)

10) Same as (4), except HPIC used in place of RCIC. (Probably not necessary to run this case.) [ This case assumes operator takes action to prevent the shift in HPCI].

11) Case of no injection, but operator depressurizes when PSP temp = 120°F.

12) Case of no injection, no depressurization.

## SIMULATION MODEL DEVELOPMENT

Development of the BWR Loss of Heat Sink (LOHS) simulation continued in December with completion of programming for the containment mass/energy balance model.

The following improvements to the simulation were completed: installation of a rudimentary wetwell model, a change of integration methods to reduce CPU time, a restart capability to facilitate calculation of multiple event branching points, and development of a plotting capability.

Analysis was performed of the cases presented in the "Operations and Systems" Section. This code has restart capability so that the individual cases can be run beginning from the applicable branch points.

A run has been completed for the joint cases 3 and 4 to the 5-hour branch point. Plots of Downcomer Level, Steam Pressure, Steam Flow (via steam reliefs and to RCIC turbine), RCIC system Injection Flowrate, Total Injected Flow (to reactor vessel), Suppression Pool Temperature vs time for this case are available.

PHENOMENOLOGY

The core uncovery and meltdown times for the Loss of AC power event were found to depend greatly on the choice of time after shutdown. In the MARCH code, for time less than 5 seconds, the current power level was assumed to be equal to the reactor operating power level. This would yield unnecessarily conservative results. Furthermore, the decay heat curve from ANS-5.1 (1973) which MARCH had adopted was unnecessarily conservative. The revised decay heat curve from ANS-5.1 (1979) had incorporated several changes so that the decay heat curves were more realistic. A subroutine ANSQ79 was completed, based on ANS-5.1 (1979) with 2-sigma uncertainty, to modify the current MARCH code. With this revised decay heat curve, the core meltdown time was expected to be long delayed when HPCI/RCIC injections were available.

Sequence of events tables were revised and distributed for complete station blackout both with and without HPCI/RCIC injections up to the core uncovery time. Based on information contained in Question 4.2, FSAR Volume 6, the HPCI failure time was chosen to be approximately 3 hours. This was found to be in agreement with thermodynamic calculations of HPCI on and off times after the water level dropped to Level 2. This also agreed with ∿428 seconds total HPCI turned-on time from some GE internal documents.

The possibility of using the 3-D vessel subroutine of the TRAC code to model the thermal stratification in the Mark I pressure suppression chamber resulting from the S/RV steam discharges was investigated. Thermal stratification in the Mark I suppression pool would affect the performance of ECCS systems which would draw water rom the pool. It would also affect the amount of subcooling available for steam condensation before condensation oscillations grew in amplitude and frequency. This problem is expected to become worse in stuck-open relief valve and ATWS events.

A potential limit-cycle oscillation problem in the core related to the cyclings of the HPCI system was identified. A sudden pressure drop inside the pressure vessel would result from the HPCI injections; this would in turn cause a sudden increase in voids and possibly, the expulsion of water from the fuel bundles in the core. Although voids would increase the void reactivity feedback and tend to reduce the neutron flux during an ATWS, excessive voids beyond boiling transitions would change the heat transfer characteristics by wetting and rewetting the surfaces. It is possible that large flow and pressure oscillations in the core could result from the HPCI cyclings. How much additional damages could this cause the fuel and claddings needs to be studied, especially for the ATWS event. Multi-channel calculations are needed to understand the amplitude and phase changes occurring in individual channels and their effects on the fuel behavior.

A possible way to attack this problem would use the COBRA/TRAC code to compute the localized thermal hydraulics in individual channels and use the results as inputs to the FRAP-T6 and/or COMETHE codes to examine the fuel behavior.

As for the rod dimensional changes, that is, the potential effects of oscillating mechanical loads on wasted and collapsed cladding in the presence of high neutron flux spikes and on center-melted $UO_2$ fuel pellets, it appears that some code development work will be needed. This could be outside the scope of the SASA program.

The WECHSL code on corium-concrete interactions was obtained from SANDIA.

The RELAP4/MOD6 verification effort on SASA sequences were continued for both with and without HPCI/RCIC injections.

## FISSION PRODUCT PATHWAYS ANALYSIS

### Fission Product Release from Fuel

The correlation of Chapman (NUREG/CR-1023) appears to provide a basis for estimating ductile cladding failures (expansion or ballooning followed by rupture). This correlation includes temperature, pressure differential, and rate of heatup. The pressure differential is that measured at the instant of rupture of the ballooned cladding; it is less than the peak pressure because of the volume increase resulting from the ballooning. We will attempt to modify the correlation in order to obtain the pressure equivalent for unexpanded fuel cladding. This should enable us to calculate the time of ductile failure by using a fixed void volume and the temperature of the plenum (where most of the gas resides) for the internal pressure calculation. Brittle cladding failures such as might occur with prolonged film boiling must be determined differently.

The collection of literature on noble gas release is nearing completion. There are a number of models (computer programs) for calculating fission gas release during normal reactor operation. We would like to find the results of calculations covering the core of a Brown's Ferry type BWR that would provide initial fission gas distribution within the fuel rods for three locations: the plenum and open voids, the grain boundaries, and within the grains. We are looking at the possibility of treating the release from each of these regions independently in order to provide a simple yet satisfactory accident release model. (This phase of the work is experiencing some delay due to the requirements of the NRC Iodine Status Report.)

## Physical, Chemical Properties and Precursor Effects

Calculations have been performed to determine which chemical and radioactive species should be significant contributors to the noble gas transport calculation. In chemical and radioactive terms, fission product krypton and xenon are the only noble gases which will be present in significant amounts. They contribute ~98% of the moles of noble gases in the fuel with the remainder being helium. They also comprise >99.9% of the radioactivity with only minor contributions from the other noble gases.

For krypton and xenon, the important chemical species are stable isotopes. The radioactive isotopes are Kr-83m, 85, 85m, 88 and Xe-131m, 133, 133m, 135, 135m. Halogen precursors are not important chemically, but they are a significant source of the radioactive noble gas isotopes. Thus, to provide completeness, the noble gas transport calculation will have to follow the stable krypton and xenon, the important radioactive isotopes, and the important halogen isotopes.

Because this initial calculation is intended to follow only the noble gases, the behavior of the halogen precursors will not receive much emphasis. Initially, the precursors will be treated as if they were noble gases although they will not be considered in noble gas saturation or equilibrium calculations. Later, when halogen behavior has been studied more fully, reanalysis of the noble gases will provide a better estimate for the precursor effects.

## Brown's Ferry Auxiliary Systems Interface

A meeting was held with TVA Systems Design personnel in Knoxville, Tennessee as a step towards identification of possible nonstandard fission product leakage pathways via (1) pressure vessel penetrations, (2) dry-well penetrations, (3) operations of the Basic Ventilation System or Standby Gas Treatment System, (4) failures in the Standby Cooling System, particularly the RHR, (5) the Radwaste System, plus (6) other Auxiliary Systems. All these added pathways will be included in the leakage pathway calculation. (This task is experiencing some delay due to work required for the NRC Iodine Report.)

## Fission Product Transport Calculation

We are currently in the process of setting up a simple transport calculation procedure designed to couple as closely as possible with the MARCH code. This procedure performs a mass balance on each selected control volume on species in the gas, liquid phases and for surface deposition. It is essential to set up this procedure because the MARCH code does not contain provision for an adequate number of control volumes outside the core.

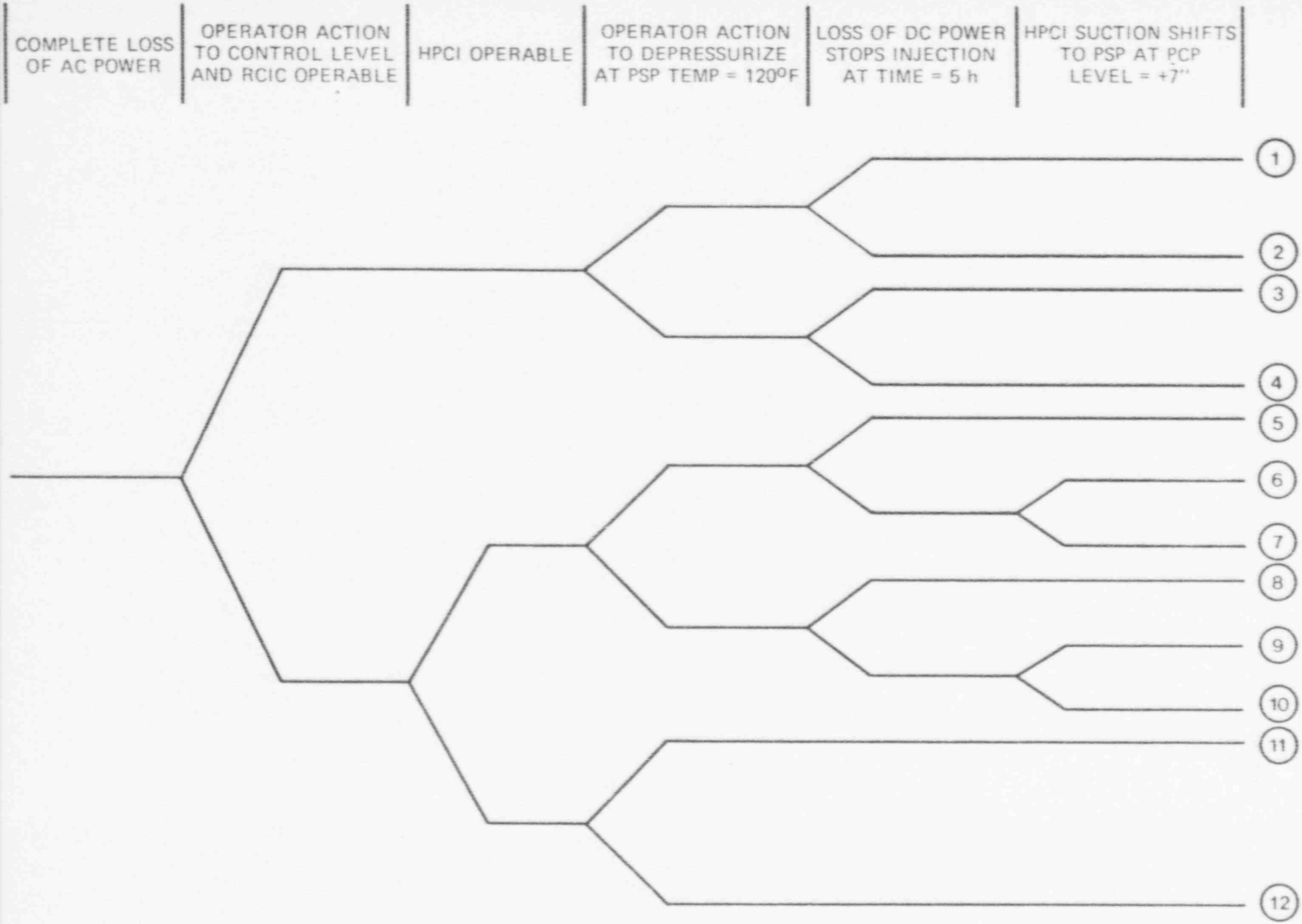


Fig. 1. Paths to Loss of Coolant Makeup Capability

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:

R. H. Chapman and J. L. Crowley visited the NRC/RSR Offices in Silver Spring, MD on December 16 and 17 for technical and programmatic discussions. Test objectives and conditions for the forthcoming B-4 (6 X 6) bundle test were confirmed. It was also agreed that the last bundle test (B-6) currently planned in this program will be a 6 X 6 array and that its test conditions will be based on data derived from the NRU in-reactor bundle tests now in progress and on the results of several MRBT single rod tests to be conducted after the B-4 bundle test.

Preliminary analysis of temperature-pressure data from the recent 8 X 8 (B-5) test were discussed. These data suggest significant axially extended ballooning of the interior 4 X 4 array will be revealed when the bundle is destructively examined. If this proves correct, rod-to-rod mechanical interactions (after the deforming rods contact) must be included in the deformation models if one expects to predict burst pressures and deformation. This appears to be a formidable, but necessary, undertaking.

Final assembly of the B-4 test array was completed, and the array was installed in the test facility on December 10. Instrumentation and electrical hookups and system checkout procedures are in progress. It is expected that the burst test will be conducted in mid-January.

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 December, 1980.

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:

The purpose of NSIC is to collect, evaluate, and disseminate information on the safety of nuclear facilities to the nuclear community through reports and the bimonthly technical progress review, *Nuclear Safety*, and to NRC staff members and their contractors through SDI, searches, and consultation. (The work reported herein is that with which NRC is especially concerned and for which the NRC RFS Program supported with 500 K, i.e., 66%.)

During the month of December, the staff of the Nuclear Safety Information Center (a) processed 747 documents, (b) responded to 94 inquiries (of which 50 involved the technical staff), and (c) made 21 computer searches. The RECON System, which now has over 200 remote terminals, reports that the NSIC data file was accessed 122 times between November 3 to 26 making it the fifth most utilized of the 31 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 10 visitors and participated in 2 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 December, we received 44 foreign documents (3 Italian, 29 French, 2 German and 10 Japanese). 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 December 31, 1980, to H. H. Scott, RSR).

During the month of December, NSIC's Selective Dissemination of Information (SDI) was providing service to a total of 342 users, including 12 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. (Note: There are 12 paid subscriptions yet to expire.) However, this and other services to NRC, DOE and their subcontract personnel are not affected.

The regular *Nuclear Safety* staff meeting was held December 3rd. 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* 22(1) to the printer on December 23, 1980. The technical content of *NS* 22(2) is in final composition at TIC, but that issue awaits the November—December "current events" material (submission deadline to TIC - January 15). All technical articles for *Nuclear Safety* 22(3) have been revised by the authors and are being edited for submission to NRC, DOE and TIC.

TABLE 1 RECON DATA BASE ACTIVITY FROM 11-03-80 TO 11-26-80
(18 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 | 3864 | 5331 | 123366 |
| NSA | (TIC) NUCLEAR SCIENCE ABSTRACTS | 536 | 841 | 7584 |
| WRA | (WRSIC) WATER RESOURCES ABSTRACTS | 275 | 633 | 13450 |
| RIP | (DOE) ENERGY RESEARCH IN PROGRESS | 213 | 238 | 2350 |
| EMI | (EMIC) ENV. MUTAGENS INFO. | 188 | 262 | 7124 |
| NSC | (NSIC) NUCLEAR SAFETY INFO. CENTER | 122 | 176 | 4447 |
| GAP | (DOE) GENERAL AND PRACTICAL INFO. | 114 | 63 | 1456 |
| FED | (DOE/EIA) FEDERAL ENERGY DATA INDEX | 109 | 106 | 71 |
| ESI | (EIC) ENV. SCIENCE INDEX | 78 | 109 | 1187 |
| ETI | (ETIC) ENVIRONMENTAL TERATOLOGY | 73 | 61 | 1015 |
| EIA | (EIC) ENERGY INFO. ABSTRACTS | 59 | 67 | 611 |
| IPS | (TIC) ISSUES AND POLICY SUMMARIES | 55 | 90 | 143 |
| WRE | (WRSIC) WATER RESOURCE RESEARCH | 42 | 97 | 297 |
| NLS | (NESC) NATIONAL ENERGY SOFTWARE | 36 | 33 | 8 |
| ERG | (BERC) ENHANCED OIL AND GAS RECOVERY | 35 | 50 | 1147 |
| PRD | (TIC/NRC) POWER REACTOR DOCKETS | 32 | 53 | 213 |
| SLR | (FRANKLIN) SOLAR DATA BASE | 31 | 31 | - |
| GID | GOVERNMENT & INDUSTRY DATA EXCHANGE | 29 | 31 | - |
| TUL | (U. TULSA) TULSA DATA BASE | 27 | 32 | 292 |
| EIS | (TIRC) EPIDEMIOLOGY INFO. SYSTEM | 25 | 29 | - |
| SUP | (TIC) THESAURUS SUPPLEMENT | 24 | 46 | - |
| CIM | (DOE) CENTRAL INVENTORY OF MODELS | 22 | 10 | 22 |
| API | (API) AMER. PETROLEUM DATA BASE | 21 | 36 | 1315 |
| NTB | (NASA TECH BRIEF FILE | 17 | 37 | 98 |
| SER | (TIC) SERIAL TITLES DATA BASE | 16 | 35 | - |
| NRC | (LC) NATIONAL REFERRAL CENTER | 13 | 10 | - |
| NSR | (NDP) NUCLEAR STRUCTURE REFERENCE | 13 | 15 | - |
| RSI | (RSIC) RADIATION SHIELDING INFO. | 12 | 8 | 266 |
| OGR | OIL AND GAS RESERVE FILE | 11 | 8 | - |
| RSC | (RSIC) RADIATION SHIELDING CODES | 5 | 1 | - |
| ARF | (EMIC/ETIC) AGENT REGISTRY FILE | 3 | 12 | 6 |

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 December; however, no new contracts were let because of delay in availability of FY81 funds.

R. A. Kisner and J. L. Anderson attended the EPRI conference on computerized operator support systems held at Tampa, Florida, on December 16 and 17. The conference gave utilities a chance to report on their work in developing operator aid systems and also a chance to voice their opinions to a large audience.

The following paragraph outlines our work with subcontractors in FY80:

(1) Completed subcontracts:

SAI - Review of Standards and Requirements Affecting Human Factors in Nuclear Power Plant Control Rooms

Lund - Human Behavior in Off-Normal Conditions

TEC - Role of the Operator in Emergency Conditions

(2) Subcontracts in place:

TEC - a) to use senior reactor operators and shift technical advisors to further the development of the role of the operator. This includes a mini-task analysis by the STA at Turkey Point.

b) to examine at least one of the emergency operating instructions from each of the other vendors and from different utilities to assure their operational role description is somewhat generic.

c) develop an algorithm such as might be used by a manufacturer to estimate cost and schedule for development, research, demonstration, installation manufacture and verification of a generic automated device. This is simple and easy to use so as to allow ORNL to objectively compare various devices being proposed as to cost and time to availability.

Becker, Block and Harris -

complete the work on crew structure. This final phase of the work will examine how individuals on the operating staff would relate to the systems organization including training requirements, and procedure structure and producing a final report.

(3) Subcontracts under consideration:

(a) Feasibility and utility of modeling the nuclear power plant operator

(b) Legal restrictions on use of proposed operator aids, training and restructuring crews, etc.

(c) Role of the operator from the training perspective

(d) Operator acceptance of new equipment and procedures.

ORNL has been requested by NRC to provide a report from research performed so far that would generate a series of technical issues associated with DASS and would indicate how regulatory might address them during the review process.

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.

Task 2: Analysis — Fuel Rod Simulator (FRS) # 058, which was removed from THTF bundle 3 during the refurbishment of the bundle in July and August, was cross-sectioned, microphotographed and the internal dimensions measured. The results were received from the Metals and Ceramics Division this month. The measurements have been analyzed and the results have been distributed in a BDHT memorandum. There was some obvious stress relief and swelling in the FRS during its lifetime. These diametrical changes have been incorporated into ORINC (used to calculate FRS surface temperatures and heat fluxes) and ORTCAL (described below).

ORTCAL is a computer program which determined fuel rod simulator thermophysical properties from data taken with the rod bundle installed in the THTF. Due to the FRS diametrical changes, ORTCAL-Part 2 regression runs have been redone. The final runs of ORTCAL-Part 3 have been made.

The ORMDIN report has been published as ORNL/NUREG/CSD/TM-17. ORMDIN performs two-dimensional calculations of FRS surface temperatures and heat fluxes.

Documentation of several computer programs used in FRS analysis is continuing and is ∿20% complete.

An interim report to NRC covering preliminary results of Test Series 3.07.9 (steady-state film boiling) is in preparation and over 50% complete. Thus far the analysis supports the results of the transient film boiling tests run at ORNL in that the Dougall-Rohsenow correlation is shown to overpredict the heat transfer coefficient.

Analysis of THTF transient film boiling test 6C is in progress and approximately 25% complete.

A revised version of the final report on heat transfer during the uncovered phase of the first Small Break LOCA Test Series (tests 3.02.10 C-H) has been completed and is currently undergoing review. The review process is expected to be completed by February 1, 1981. Work was started on several papers to be submitted to the 20th National Heat Transfer Conference. Preliminary drafts are expected by January 15, 1981. Work is continuing on a letter report summarizing the second Small Break LOCA Test Series; completion is expected by January 26, 1981.

A review of NRC/NRR requests for research in support of licensing was completed and two series of tests to be run in the THTF were recommended. The recommended test series would address high pressure pool boiling heat transfer and intermediate mass flux film boiling ($1.2 \times 10^4 \leq G \leq 2 \times 10^5$ $lb_m/hr \cdot ft^2$).

Task 3: THTF Operations — During December, the Thermal Hydraulic Test Facility was being returned to the original design 4-in. pipe configuration in preparation for mothballing the facility for the termination of operations. Work is now being initiated for the writing and publication of a final facility description report. This task is scheduled to continue through the next several months.

Task 4: Two-Phase Instrument Development — Evaluation of the in-bundle densitometer system is continuing. Results from the Upflow Film Boiling Test 3.06.6B indicate nonphysical trends in the densitometer for fluid temperatures exceeding approximately 315°C (600°F). Leakage currents due to decreasing resistance with increasing temperature across the ion chamber and/or tri-ax cable insulation material are swamping the radiation induced ion chamber current for fluid temperatures exceeding approximately 332°C (630°F). Due to the high fluid temperatures [exceeding 482°C (900°F)] experienced in the uncovered portion of the latest small break LOCA tests (3.09.10 I-N), steady-state data from the in-bundle densitometers is not available. Transient in-bundle densitometer test data for the boiloff and reflood portions of these tests (3.09.10 O-X) has not been analyzed at this time.

PROGRAM TITLE: Safety Related Operator Actions

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL# 41 89 55 12 3 (189 #B0421-8) NRC #60 19 11 01 2

TECHNICAL HIGHLIGHTS:

The draft report summarizing the results of the initial PWR simulator exercises was submitted to ORNL by General Physics Corporation (GPC) and forwarded to NRC. After review by NRC and ORNL staff, it will be published as an ORNL/TM report.

Analysis of PWR field data continued. Memphis State has accumulated site records for the following events (numbers of events in parentheses):

Inadvertent Safety Injection (25)
Flux Tilt (35)
Small LOCA (6)
Dropped Rod (14)
Nuclear Instrument Failure (20)
RTD Failure (2)
Steam Generator Tube Leak (2)

It is expected that records for approximately 25 more occurrences identified in previous docket searches will be obtained during January, and that will complete the initial field data collection effort for PWRs. The extent to which applicable time response data can be derived from these records is not yet clearly established. Some of the occurrences certainly do not follow the sequence outlined for the simulator exercises and quantitative data will be limited.

The team of consultants working on the simulator response characteristics task had its second meeting at ORNL on December 2. An outline for the preliminary report has been established. A draft of the report is expected to be completed by Franklin Research Center (with input by GPC) in mid-January.

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:

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

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 December.

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