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 INTERIM REPORT

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Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
Operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

NRC Research and Technical Assistance Report

INTERIM REPORT

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ABSTRACT

Highlights of technical progress during May 1981 are presented for nineteen selected ORNL research programs for the Office of Nuclear Regulatory Research.

PROGRAM TITLE: Advanced Instrumentation for Reflood Studies (AIRS)

PROGRAM MANAGER: M. B. Herskovitz

ACTIVITY NUMBER: ORNL #41 89 55 11 8 (189 #B0413)/NRC #60 19 11 01

TECHNICAL HIGHLIGHTS:

The digital data tapes from the SCTF shakedown test of April 24 have been received. We expect to have the analysis completed by mid-June. This initial analysis will take longer than usual, because our procedures and software for handling this large volume of data and reading the Japanese tapes are new and have not been tried out on actual test data.

A preliminary analysis of one flag probe and one wall film probe, data from which were recorded on analog tape, has been performed. The flag probe (UB01821), located 1500 mm below the upper tie plate, was characerized by a rapid quench approximately 70 seconds into the test with subsequent void fraction averaging about 0.6 until complete flooding at 1000 seconds. The only deviation was a dry-out period during which the void fraction stayed around .95 to .99 from approximately 140 seconds to 200 seconds.

The noise velocities measured by this probe showed many direction reversals and fell mostly within the $\pm$ 2 m/s range. Much of the data exhibited very low coherence and appeared to be contaminated with 300-Hz noise, which may be affecting the velocity measurement. We suspect the noise came from the FM recorder rather than the probe. This will be determined when we look at the same data from the JAERI digital tapes.

The film thickness sensors appeared to be working normally and produced signals characteristic of very turbulent wavy films with thicknesses remaining in the 0.1 to 1.0 mm range throughout most of the test. The initial quench was also at 70 seconds.

Sensor fabrication for CCTF-II has been a major effort this month. The upper plenum structural film probes have been shipped and plans are being made for ORNL personnel to travel to Yokohama to install them. Some additional wall film probes for SCTF are also being fabricated to replace units with bad cables. Three probes will be replaced during the SCTF shutdown in June.

Also fabricated this month was an SCTF-style flag probe which will be used for steam-water testing to obtain additional data for the fluid velocity correlation.

PROGRAM TITLE: Advanced Two-Phase Instrumentation

PROGRAM MANAGER: D. G. Thomas

ACTIVITY NUMBER: 40 89 55 11 5 (189 #B0401)/ NRC 60 19 01 30

TECHNICAL HIGHLIGHTS:

The multiple position heated junction thermocouple (MPHJTC) arrived safely at EG&G Idaho. A complication arose, in that to install the probe it would have to be severely bent along its axis. ORNL felt this proposed bending would damage the MPHJTC and possibly diminish the effectiveness of the sensor. An alternative to bending the MPHJTC was to disassemble the upper head on Semiscale. This operation is expensive and time consuming since the upper head supports the entire core. At this juncture, the task is on hold until a solution, which is agreeable to all involved parties, can be worked out.

Effort was continued in attempting to obtain pressure drop data from Semiscale testing of the Westinghouse dP system. Evaluation of that data will begin as soon as it is received at ORNL.

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:

Partial analytical results have been received from U_3O_8 aerosol samples taken during Run 403. This run featured an aerosol mass concentration intermediate between Runs 401 and 402 with low-level steam additions to the vessel atmosphere for 6 hrs rather than 2 hrs as was the case for the first two runs. At the start of aerosol generation the vessel atmosphere was at 0.105 MPa gauge pressure and 377 K; at the end of the 6 hr low-level steam injection period the values measured were 0.114 MPa and 375 K. Over the next 18 hrs the vessel and its contents were allowed to cool.

The aerodynamic behavior of the U_3O_8 aerosol was remarkably similar in all three of the steam-air tests even though the aerosol mass concentration and the scheme of steam addition varied among the tests. The influence of condensing steam seems to be the dominant factor in determining aerosol behavior.

Mixing of the aerosol and steam within the NSPP vessel has not been entirely adequate. Over the first hour, or so, during a run, an aerosol mass concentration gradient exists, varying with height, with the highest concentration in the upper quadrant of the vessel. A small, air-driven, fan will be installed near the bottom of the vessel, centrally-located, and will be used to promote mixing of aerosol and steam during conduct of Run 404.

CORE MELT AEROSOLS:

Using the 1/2 kg melt furnace, several test fuel melts with amounts of Zircaloy-clad fuel varying between 300 and 700 grams have been made to explore methods of providing sufficient crucible water cooling and control of the vigorous metal-water reaction between the metered water supply and the hot Zircaloy. Some vaporization data, mainly showing tin, uranium, and zirconium aerosols vaporizing in that order at about 2400°C, have been obtained.

An opportunity to investigate the importance of the choice of frequency in the kilohertz range (a specification for the new power supply to be purchased) has been afforded by the loan of a Westinghouse rf generator, designed to operate around 400 KHz. This approximates the reported frequency successfully used in France in the CORRECT-II facility. Our present power supply produces rf at only about 200 KHz, and may not be as efficient as the higher frequency.

To provide preliminary information in anticipation of the later NSPP experiments, aerosols of importance to the core-melt source term assessment are being produced using the metal-oxygen dc plasma ignition method in the CRI-II facility. The most important contributor from the structural metal (stainless steel and inconel) has been investigated through a surrogate (pure iron powder) and the physical and aerodynamic properties of the agglomerated Fe_2O_3 determined.

FAST/CRI-III:

Work continued on converting the FAST facility to permit under-sodium tests to be performed.

Welding of stainless-steel wave-guides to the test vessel was completed; the wave-guides permit acoustic transducers to be mounted to the vessel.

Welding needed to connect the sodium fill-and-drain system to the test vessel was completed, and all welds were X-rayed. The vessel head was re-mounted on the vessel, and heaters were re-installed on the vessel surface. Pending an in-house QA audit of the facility, sodium fill and the start of sodium tests should occur next month.

Work on the revised sodium test plan is continuing; however, the plan cannot be completed until a series of calculations using UVABUBL can be performed.

ANALYSIS:

This period has been devoted to work on the UVABUBL code. A number of numerical and conceptual problems have been uncovered in the course of attempts to apply the code to FAST water test conditions, conceivable sodium tests, and large scale reactor accident conditions. These are being resolved with the co-operation of University of Virginia personnel.

Processing of data for NSPP Run 403 was carried out satisfactorily.

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:

Task 1. In-Plant Demonstration of On-Line Surveillance System. The surveillance system experienced several hardware failures during this reporting period. We have not isolated the causes of the failures but suspect that the temperature and dust in the plant computer room may be too high for our system on some occasions. We have increased the frequency of dust filter servicing and designed an over-temperature protect for the system. The temperature protect feature will be installed during the next reporting period.

The system, otherwise, has performed well in obtaining and storing frequency spectra during plant operation. Supplemental data was obtained by recording all patch panel signals on magnetic tape for off-line analysis in the noise analysis laboratory at ORNL. This analysis provided additional insight into the interrelationships between signals by calculation of coherences and phase between signals (the on-line surveillance system monitors only the power spectrum of each signal).

Task 2. Procurement of NRC Surveillance System. Conversion of the DOE system software to make it compatible with the new NRC system hardware is ninety percent complete. The programmable gain amplifiers have been built and are undergoing operational verification. The new system will be installed at Sequoyah during the next reporting period.

Task 3. Experiments in Test Facilities. We have contacted personnel at the Semiscale facility at INEL to determine if future tests there could yield data that can aid our surveillance and diagnostics program. They sent us a preliminary draft of a program plan for natural circulation tests which includes two-phase operation and operation with noncondensible gas under natural circulation conditions.

We plan to submit a proposal to INEL to record data during some of the proposed tests. The tests are planned for July-September 1981.

PROGRAM TITLE: Fission Product Release from LWR Fuel

PROGRAM MANAGER: R. P. Wichner, M. F. Osborne

ACTIVITY NUMBER: ORNL # 41 89 55 10 8 (189 #B0127)/NRC # 60 19 01 40

TECHNICAL HIGHLIGHTS:

1. Furnace Concept Tests

Two additional tests were conducted 1) to compare the initiation of electrical discharge in pure argon vs argon containing 10% nitrogen and 2) to investigate further the relation between RF generator settings and interference with the center thermocouple signal. It was verified that in the power range of interest, pure argon will be suitable for shielding the tungsten susceptor. (Pure helium cannot be used because significant electrical discharge occurs at moderate power input.) Although RF interference with the thermocouple could not be eliminated, suitable test conditions were identified which will allow reliable use of the Pt vs Pt/Rh or W/Re thermocouples to monitor specimen temperature.

2. Furnace Design

Design of the furnace, and the associated apparatus for installation in the hot cell, was completed. The drawings were thoroughly checked, an Engineering Division QA assessment was submitted, and the drawings were transmitted, certified for fabrication, as follows:

DWG No.	Title
	Fission Product Release Apparatus —
X3E-12505-0003	Furnace Assembly Sheet No. 1
X3E-12505-0004	Furnace Assembly Sheet No. 2
X3E-12505-0005	Detail Sheet No. 1
X3E-12505-0006	Detail Sheet No. 2
X3E-12505-0007	Detail Sheet No. 3
X3E-12505-0008	Detail Sheet No. 4
X3E-12505-0009	Detail Sheet No. 5

3. Fuel Specimens

Nine fuel specimens, as described below, are stored in an adjacent hot cell, ready for testing.

Specimen No.	Reactor	Shutdown date	Burnup (GWD/MT)
B-2	H. B. Robinson 2	May 1974	27.3
B-8A	H. B. Robinson 2	May 1974	28.5
B-9	H. B. Robinson 2	May 1974	25.3
P-7	Peach Bottom 2	March 1976	11.5
P-8	Peach Bottom 2	March 1976	11.8

Specimen No.	Reactor	Shutdown date	Burnup (GWD/MT)
D-1	Dresden-1	Sept. 1975	20.7
D-2	Dresden-1	Sept. 1975	22.2
D-3	Dresden-1	Sept. 1975	23.4
D-4	Dresden-1	Sept. 1975	24.0

In order to assure adequate characterization of these irradiated specimens, samples of each type were selected for burnup analysis and for metallographic examination. These data will be used in the interpretation of release results.

4. Data Acquisition and Analysis

The data acquisition and control system, a disk storage unit for the TP-5000 computer, and a two color optical pyrometer are on order. Software development (for use in data processing with the HP-9825A computer) is underway, and data collected during the concept tests is being analyzed.

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 01 30

TECHNICAL HIGHLIGHTS:

Task 1: Program Administration — R. D. Cheverton, J. G. Merkle and G. D. Whitman, accompanied by M. Vagins of NRC, participated in a meeting on May 21 and 22, at MPA, Stuttgart, Germany, to review cooperative programs in primary system integrity research and development.

G. D. Whitman made a presentation and conducted a tour on May 26 and 27, respectively, for D. F. Ross, Deputy Director, Office of Nuclear Regulatory Research, USNRC.

Preparations were continued for the NRC Vessel and Piping Integrity Review and Workshop which will be held in Oak Ridge, June 1–5.

Task 3: Irradiation Effects — Irradiation of the third capsule of the Fourth HSST Irradiation Study continued through this month. Temperature control remains excellent.

The second capsule of this study was disassembled and neutron dosimeters were submitted for radiochemical analysis. As noted in the April highlights, hot cell tests will not be initiated until the servohydraulic machine is operational in the hot cell.

The specimens being supplied by MPA, Federal Republic of Germany, for the fourth capsule of this study have been shipped from Germany. We expect to begin assembly of this capsule in October.

Task 4: Thermal Shock - Work was continued on TSE-6 with preparations for a thermal-hydraulic test to be performed during the first week of June.

The testing of the 1T compact specimens (1T CS) machined from thermal shock vessel TSC-2 has continued. Fracture toughness (K_J) results from ten specimens tested at 24°C ranged from 198 to 338 MPa $\sqrt{m}$. Values from eleven 1T CS tested at −4°C ranged from 106 to 261 MPa $\sqrt{m}$. The range of toughness values from each test temperature are comparable to those obtained previously from similar tests of specimens from the prolongation of vessel TSC-2 (TSP-2); however, the minimum values from TSC-2 occur at slightly higher toughness for each test temperature indicating that the transition regime of TSC-2 occurs at lower temperatures.

The fatigue precracking of the specimens to be tested at −32°C is in progress. Dynamic testing of precracked Charpy V-notch specimens has begun at 24, −4, and −32°C and the results are being analyzed.

Task 5: Simulated Service Tests — Preparations for the low upper shelf intermediate vessel test V-8A continued. At ORNL the apparatus for fatigue sharpening machined notches in vessels was tried on the V-8A flawing practice weldment. The seal leaked during the first test, and the seal block was modified to provide the correct seal volume.

A mockup of test-vessel filler blocks was used to determine internal instrumentation assembly procedures and cable-length requirement. Strain gage and ultrasonic (UT) transducer specifications were determined. Temperature-compensated weldable strain gages will be used throughout. Six UT transducers, of which one will be a dual crystal type for observing cracks very near the surface, will be installed inside the vessel.

Possible use of electric potential drop measurements on the test vessel was discussed with Battelle Columbus Laboratories (BCL), who have had experience in determining crack growth from such measurements on pipes. It may be feasible to determine the onset of stable tearing by this means.

In vessel and characterization weld preparation, the Babcock and Wilcox Company (B&W) completed the postweld heat treatment of the cylindrical prolongation containing four low upper shelf weld seams. A fifth seam weld was completed to provide sufficient stock for all specified characterization tests. This extra weld will be used in lieu of the one prolongation seam that had radiographic indications of possible defects. Vessel V-8A was prepared for the start of ASA welding the last week in May. The fifth characterization weld seam will be heat treated with the vessel. B&W will proceed with the vessel weld and testing of characterization specimen from the prolongation simultaneously.

Task 6: Pressurized Thermal Shock Studies — The design of the test facility modifications is proceeding with equipment and utility system design and a coolant piping flexibility analysis.

A stress analysis has been initiated to evaluate intermediate pressure vessel seal region deformations under projected thermal and pressure loads during pressurized thermal shock excursions.

PROGRAM TITLE: HTGR Safety Analysis and Research

PROGRAM MANAGER: S. J. Ball

ACTIVITY NUMBER: ORNL #41 89 55 11 2 (189 #B0122)/NRC #60 19 13 02

TECHNICAL HIGHLIGHTS:

Code Development: Assistance was provided to KFA (West Germany) and Public Service Co. of Colorado in implementing the new modification to the Fort St. Vrain (FSV) reactor steam generator simulation code BLAST, as reported last month. This change, which makes use of the Chen correlation for flow nucleate boiling, has improved the convergence characteristics of the BLAST initialization routine.

Severe Accident Sequence Analyses (SASA): Work was begun on an HTGR SASA task, the object of the work being to analyze postulated accident sequences that lead to core damage. The goal is to gain a better understanding of the accident sequences and their consequences, and to generate recommendations for improved designs and procedures, as well as better estimates of emergency planning requirements. To this end, a review of the HTGR Accident Initiation and Progression Analysis (AIPA) was begun. This work purports to be ". . . a comprehensive risk assessment of the HTGR"[1], and, therefore, will be used as an aid in selecting appropriate items for HTGR severe accident studies at ORNL.

Proposed FSV Experiments: The ORECA code, a 3-D thermal hydraulics simulation of the FSV core, was used in developing preliminary procedures for FSV experiments designed to characterize core flow bypass behavior. Series of tests using ORECA are being run to demonstrate the sensitivity of the available measurements (primarily refueling region outlet temperatures, steam generator inlet temperatures, and core ΔP) to various core bypass flow assumptions when changes are made in region orifice positions.

Miscellaneous: A preliminary look was taken at the possibility of using the ORNL Core Flow Test Loop (CFTL), which was originally designed as a GCFR core test facility, for investigating FSV core support post corrosion behavior. Reference material was sent to NRC for review.

[1]Phase II Risk Assessment, Report No. GA-A15000, April 1978.

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 using multiple-frequency techniques.

A series of tests have been run using a 0.060 in. mean radius reflection type coil to inspect the surface of a tube. The coil axis is perpendicular to the surface of the tube, and this type of probe has sometimes been referred to as a pancake probe. The probe was used with the standard 3-frequency instrumentation, but with a slightly modified calibrator module. The same operating frequencies (20 KHz, 100 KHz, and 500 KHz) as the Point Beach inspection were used. In preliminary tests, the multiple frequency system was able to separate and measure the depth of 25% defects on the far wall and 0.0065 in. diam through holes despite several interfering variables. The principal thrust for this activity was for another project but is reported here because of the relevance.

We have accepted an invitation to use our equipment to inspect an intergranular attack sample from the Ginna steam generator. The inspection is scheduled for June in Lynchburg, Virginia, and is being coordinated by Steve Brown.

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 attended the 7TH LWR-PV Surveillance Dosimetry Program Meeting at Babcock and Wilcox in Lynchburg, VA, May 19-22, 1981. In addition to the regular discussion of activities, status, and scheduling for the program, considerable time was spent in reviewing the NUREG report on the PCA experiments and "Blind Test."

On May 27-28, 1981 Denwood Ross, Deputy Director of the Office of Nuclear Regulatory Research, was at ORNL to review the various NRC programs on site. F. B. K. Kam presented an overall view of the LWR-PV Surveillance Dosimetry Program. Dr. Ross commented that some of the results from this program can be used immediately.

Andy Thomas of Rolls-Royce Associates was at ORNL for approximately two days of discussions, May 27-28, 1981. A tour of the HSST displays at the Y-12 plant was arranged by R. Bryan in spite of his busy schedule preparing for the upcoming HSST Seminar and Workshop, June 1-5, 1981.

Participants in the program are reminded that the experimental results from the ORR-PSF low- and high-powered start-up experiments, the Westinghouse perturbation experiment, and the SSC-1 dosimetry and metallurgical experiment are required by August 1, 1981 for inclusion into the 1981 Annual NRC Report. Russ Hawthorne has published preliminary results on the SSC-1 metallurgy, but has been delayed from completing the testing of the Charpys and tensiles specimens due to lack of information from participants. Participants are urged to contact Russ if they have not already done so.

Task 2: Benchmark Fields -

A. PCA - Transport Calculations and Dosimetry

Final review of the NUREG report will be completed.

B. ORR-PSF

Temperatures and reactor power data from the pressure vessel irradiation capsule were processed through Feb. 28, 1981. Results are reported in Table 1.

Experimental data from tests made on the second surveillance capsule (SSC-2) have been analyzed and a discrete-time optimal control law was obtained. Appropriate modifications for the computer control algorithm software have also been made.

C. BSR-HSST

The dosimetry in Capsule B of the 4TH HSST irradiation series is being removed in the hot cells.

D. IAEA REAL-80 Project

The data from the IAEA REAL-80 Project were processed with the WINDOW code and the results were compared with the previously obtained results from the LSL least squares adjustment code. The WINDOWS code tends to make much larger adjustments on the differential fluxes, but the results from the integral parameters are quite similar, well within the calculated uncertainty bounds. The results are being formatted for submission to the IAEA.

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

A TM-report on the general theory for least squares adjustment and curve fitting procedures has been prepared. This paper is intended as a reference document for use in the ASTM E10.05.01 Task Group on Uncertainty Analysis and the Standard on Adjustment Procedures and other standards for the LWR-PV Surveillance Embrittlement Program.

Table 1. Cumulative Characterization Data for the Pressure Vessel Capsule Through February 28, 1981.

Data for PSF **Specimen Set OT**
Hours of Irradiation = **5313.69**
Megawatt Hours of Irradiation = **150426.33**

Thermocouple	Hours of Irradiation					Average	Standard
	T<270	270<T<280	280<T<296	296<T<306	306<T	Temperature	Deviation
TE 101	59.95	22.29	5210.07	21.38	0.00	289.62	1.79
TE 102	56.48	17.20	5197.32	42.66	0.00	291.38	1.33
TE 103	56.05	15.58	5242.05	0.00	0.00	289.21	1.01
TE 104	50.91	13.01	5240.32	9.44	0.00	292.02	0.97
TE 105	54.71	18.73	5240.27	0.00	0.00	285.72	1.09
TE 106	50.79	13.91	5248.96	0.00	0.00	288.84	1.00
TE 107	56.06	376.13	4881.51	0.00	0.00	282.21	1.38
TE 108	63.69	17.38	5223.51	9.06	0.00	289.29	1.52
TE 109	63.75	18.34	5223.80	7.78	0.00	288.88	1.55
TE 110	55.94	17.22	5227.59	12.95	0.00	290.08	1.09
TE 111	131.68	20.23	5161.75	0.00	0.00	287.86	1.53
TE 112	1.75	0.00	5311.91	0.00	0.00	288.00	0.00
TE 113	49.86	14.52	5247.23	0.04	2.00	290.23	1.64
TE 114	67.22	17.08	5229.36	0.00	0.00	288.73	1.53
TE 115	4.08	0.00	5309.58	0.00	0.00	288.00	0.00
TE 116	62.50	12.58	5238.60	0.00	0.00	290.00	0.58
TE 117	56.36	15.61	5235.90	5.29	0.50	291.20	0.92
TE 118	59.08	21.56	5233.10	0.00	0.00	286.44	1.04
TE 119	56.15	19.11	5238.42	0.00	0.00	286.53	1.00
TE 120	61.70	198.48	5053.56	0.00	0.00	283.09	1.47

Data for PSF Specimen Set 1/4 T
Hours of Irradiation Time = 5313.69
Megawatt Hours or Irradiation = 150426.33

TE 201	60.78	17.32	5232.94	2.66	0.00	290.27	1.51
TE 202	61.05	15.83	5236.63	0.17	0.00	288.87	0.99
TE 203	59.68	13.34	5240.69	0.00	0.00	288.59	1.06
TE 204	56.77	14.30	5242.30	0.33	0.00	289.35	0.86
TE 205	56.60	21.31	5235.79	0.00	0.00	286.30	1.03
TE 206	54.71	19.52	5239.46	0.00	0.00	286.80	0.90
TE 207	59.41	84.24	5170.04	0.00	0.00	282.95	1.07
TE 208	61.85	13.70	5237.29	0.83	0.00	288.13	1.34
TE 209	62.23	18.93	5232.53	0.00	0.00	288.72	1.27
TE 210	61.26	25.61	5226.82	0.00	0.00	286.53	0.96
TE 211	64.42	28.82	5220.46	0.00	0.00	284.62	0.84
TE 212	54.18	9.21	5250.29	0.00	0.00	290.79	1.05
TE 213	54.54	9.77	5249.35	0.00	0.00	289.47	1.17
TE 214	62.40	12.09	5239.17	0.00	0.00	290.46	0.96
TE 215	62.63	18.49	5232.53	0.00	0.00	287.44	0.71
TE 216	61.75	12.85	5239.10	0.00	0.00	287.85	0.79
TE 217	58.27	10.28	5245.14	0.00	0.00	289.61	0.96
TE 218	58.19	16.27	5237.25	2.0[illegible]	0.00	286.84	1.02
TE 219	56.73	13.43	5243.54	0.00	0.00	287.01	0.90
TE 220	57.81	106.65	5149.23	0.00	0.00	285.26	1.25

Table 1. (Cont'd)

Data for PSF Specimen Set 1/2 T
Hours of Irradiation Time = 5313.69
Megawatt Hours of Irradiation = 150426.33

Thermocouple	Hours of Irradiation					Average Temperature	Standard Deviation
	T<270	270<T<280	280<T<296	296<T<306	306<T		
TE 301	59.77	8.98	5205.79	39.18	0.00	290.20	1.37
TE 302	62.02	14.21	5237.47	0.00	0.00	286.45	0.[illegible]3
TE 303	59.09	10.84	5243.76	0.00	0.00	287.27	0.91
TE 304	54.68	10.95	5247.47	0.58	0.00	290.99	0.79
TE 305	54.37	12.03	5247.29	0.00	0.00	287.63	0.91
TE 306	57.17	17.08	5239.43	0.00	0.00	286.58	0.83
TE 307	1.75	0.00	5311.91	0.00	0.00	288.00	0.00
TE 308	61.69	9.58	5242.41	0.00	0.00	289.14	1.13
TE 309	63.07	12.46	5238.18	0.00	0.00	287.69	0.88
TE 310	64.89	29.83	5218.99	0.00	0.00	285.31	1.05
TE 311	65.06	28.79	5219.88	0.00	0.00	285.72	1.13
TE 312	57.00	8.40	5248.11	0.17	0.00	288.72	0.94
TE 313	56.95	8.41	5246.67	1.67	0.00	290.05	1.02
TE 314	64.13	11.61	5237.94	0.00	0.00	289.15	1.01
TE 315	65.85	12.53	5235.32	0.00	0.00	285.08	0.89
TE 316	62.80	6.04	5244.86	0.00	0.00	287.58	0.72
TE 317	56.22	11.51	5245.96	0.00	0.00	290.77	0.88
TE 318	55.82	11.53	5246.33	0.00	0.00	289.31	0.97
TE 319	60.73	17.69	5235.26	0.00	0.00	285.28	0.79
TE 320	58.77	13.71	5241.20	0.00	0.00	287.74	1.25

PROGRAM TITLE: LWR Severe Accident Sequence Analysis (SASA)

PROGRAM MANAGER: S. A. Hodge

ACTIVITY NUMBER: ORNL # 41 89 55 13 4 (189 #B0452)/NRC 60 19 01 3 0

TECHNICAL HIGHLIGHTS

The personnel contributing to the SASA program at ORNL are divided into three working groups. The mission and progress of each of these groups will be summarized in individual sections below The Severe Accident sequence currently under study is Station Blackout at Browns Ferry Unit 1 (Loss of Offsite Power and Failure of the Onsite Diesel-Generators to start and load).

Most of the effort within each group during May has been devoted to completion of the applicable sections of the report "Station Blackout at Browns Ferry Unit One," NUREG/CR-2182. A draft of this report was delivered to the NRC technical monitor on 28 May, and copies have subsequently been distributed to the SASA principal investigators at the other participating laboratories as well as to interested parties at ORNL. Comments have been requested by July 15.

By prior agreement with the NRC technical monitor, this draft has been distributed before completion of all sections of Chapter 12, which reports the results of the fission product transport analysis. (The remainder of Chapter 12 is expected to be completed before August 1 and will be distributed for comment as soon as available.) Since the fission product transport analysis must in all cases await the completion of the accident sequence analysis, it is proposed that the fission product transport results obtained during future ORNL SASA studies be documented in the form of a separate report.

The next BWR Severe Accident sequence to be analyzed has been selected as a non-isolatable small-break LOCA outside of containment at the Browns Ferry Nuclear Plant Unit One. It was originally considered* that the accident initiator should be a hypothetical break in the scram discharge volume piping which occurs following a scram that cannot not be reset. However, this accident sequence has been the subject of recent extensive review** and it is believed that a SASA study at this point would be redundant. Accordingly, a main steam line break combined with gross containment isolation valve leakage accident initiator is currently being considered for the non-isolatable small-break LOCA outside of containment sequence.

The individual group reports for progress during April are presented below, with a brief initial statement of the purpose of each group.

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*See the Monthly Highlights Report for April 1981.

**GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks, NEDO-24342 (April 1981).

Group I: determines and analyzes the events of the accident sequence which would occur prior to core uncovery, using efficient coding. An efficient set of "mechanized hand calculations" specific to Browns Ferry Unit One has been developed at ORNL for use in obtaining quick-plotted studies of the plant response to hypothetical operator actions. This program for plant simulation (BWR-LACP) is used to provide answers wherein the level of detail provided by the more sophisticated codes does not justify their added expense, nor the additional time required to prepare input and obtain the results. As an example, the MARCH code would require extensive internal modification to make it capable of modeling plant response to postulated operator actions during the early stages of a Station Blackout sequence.

Activities during May were directed toward completion of a comparison between the ORNL BWR-LACP calculations and those of RELAP-IV and the TVA Browns Ferry Simulator. A transient energy balance was added to the reactor fuel model to see if the extra heat capacity would lengthen the periods between safety relief valve (SRV) actuations, thereby improving the agreement between BWR-LACP and the other two codes. However, this modification did not produce a significant change in predicted SRV actuation.

The progress achieved in BWR-LACP development and the results of the Station Blackout calculations were presented at the SASA program technical review meeting in Silver Spring, MD on 28 May.

Group II: determines and analyzes the events of the accident sequence which would occur following core uncovery, including core melt and containment failure, using the sophisticated codes.

Local MARCH Capability. Previously, all MARCH runs have been performed using a CDC version of the code on the computers at Brookhaven National Laboratory (BNL). This has been satisfactory for the short term, but has involved some undesirable delays and inefficiencies. Accordingly, effort is underway to implement an IBM version of MARCH which has been developed and made available by TVA.

The Aerosol Release and Transport (ART) program manager at ORNL (Tom Kress) also desires a local MARCH capability for the benefit of the ART project, and has agreed to bear one-half of the cost for the implementation of a local IBM version.

A magnetic tape containing IBM versions of MARCH, CORRAL-2, and KORLIN was obtained from TVA on May 7. The tape also contained the JCL file which TVA uses to run the MARCH/KORLIN/CORRAL job string, and the TVA Browns Ferry sample problem input data file.

Most of the JCL changes which are necessary to run MARCH on ORNL's computer system have been implemented. Both the "G" and "Q" compilers are employed for program compilation. An unforseen problem was encountered when it was determined that the roll size employed by ORNL's G compiler was insufficient for compilation of the MACE subroutine.

The problem was resolved by obtaining an expanded version of the G compiler from the K-25 plant and loading it into ORNL's computer system. Remaining JCL problems are related to the difference in the TVA and ORNL system architectures and should be resolved shortly. MARCH should be operational on ORNL's computer system in early June.

After the code becomes operational, several sample problems will be analyzed and compared to similar calculations from BNL's version. Following this validation effort, all modifications necessary to ensure compatibility between MARCH 1.4B and ORNL's version will be implemented.

Group III: determines the magnitude and timing of fission product release from the fuel, establishes the various pathways for fission product release to the atmosphere, and performs the fission product transport calculation for each Severe Accident sequence analyzed.

Over the past several months, the services of two of the members of this group (R. P. Wichner and R. A. Lorenz) have been largely unavailable to the ORNL SASA program due to pre-emption for work on the iodine state-of-technology report. As a result, the draft report for Station Blackout initially submitted for peer review contains some labeled but otherwise blank pages in this area. This NRC-ORNL agreed-to procedure was adopted to avoid unnecessary delay in the distribution of this otherwise 95%-completed report. The currently unavailable material will be distributed for review as soon as possible; the deadline for distribution of the final report (1 Oct 1981) should be unaffected.

Release of Fission Products from Fuel. Section 12.2, "Fission Product Release from the Fuel: Fission Gas, Iodine, and Cesium," was written and submitted for inclusion in the draft report Station Blackout at Browns Ferry Unit One. The section contains fission product release models for these species from rod rupture up to fuel melting.

The task in progress is to estimate the rate of formation of aerosols over melted fuel. Our target is to formulate a semi-empirical model based on experimental data extrapolated to larger scale and containment pressure.

Fission Product Pathways Calculation. A data interface between the MARCH code and the noble gas transport calculation has been provided. The interface was required to transmit temperature and flow data generated by MARCH in about 100 core control volumes as input to the transport rate coding.

An initial noble gas transport calculation for the Browns Ferry complete station blackout sequence has been completed. Though several minor alterations are required, the calculation has incorporated the following elements: (1) an improved noble gas release from fuel model, (2) flow and temperature histories as input developed from sequence analysis, (3) incorporation of containment vessel and reactor building failures modes, and (4) an estimate of transport and holdup in the reactor building.

Tables 1 and 2 list Kr and Xe (respectively) inventories calculated for the station blackout sequence for five representative control volumes plus total remaining amounts of the release. The three events noted in the heading of each table are predicted by the MARCH code to occur at the times indicated below:

Event	Approximate time following loss of injection capability
Core slump	2 1/2 hr
Vessel failure	3 hr
Containment failure	4 1/2 hr

For the station blackout sequence, the loss of injection capability occurs when the 250 Volt DC system is lost due to battery exhaustion; this is assumed to occur four hours after the inception of the blackout. The fourth results column in Tables 1 and 2 contains the predicted noble gas inventories one-half hour after containment failure.

Representative plots of xenon released from the fuel, released from the primary containment, inventory in the refueling area of the reactor building, and total amount released are shown in Figs. 1 to 4. The time values on the abscissa in each of these figures proceed from the 4-hour point, when injection capability is assumed lost.

Technical Review Meeting Presentation. The current status of the fission product transport analysis for the Station Blackout and the preliminary results for the noble gas releases were presented at the SASA program technical review meeting in Silver Spring, MD on 28 May.

Table 1. Preliminary noble gas results

Location	Krypton inventories (PBq)[a]			
	At core slump	At vessel failure	After failure of containment	30 minutes later
Total present	1010	880	670	610
Fuel rods	330	270	200	190
Wetwell water	2	29	2	1
Wetwell air	29	470	11	1
Drywell	2	110	19	1
Refueling floor	0	0	130	1
Atmosphere	0	0	38	420

[a] 1.0 petabecquerel = 27,000 curies.

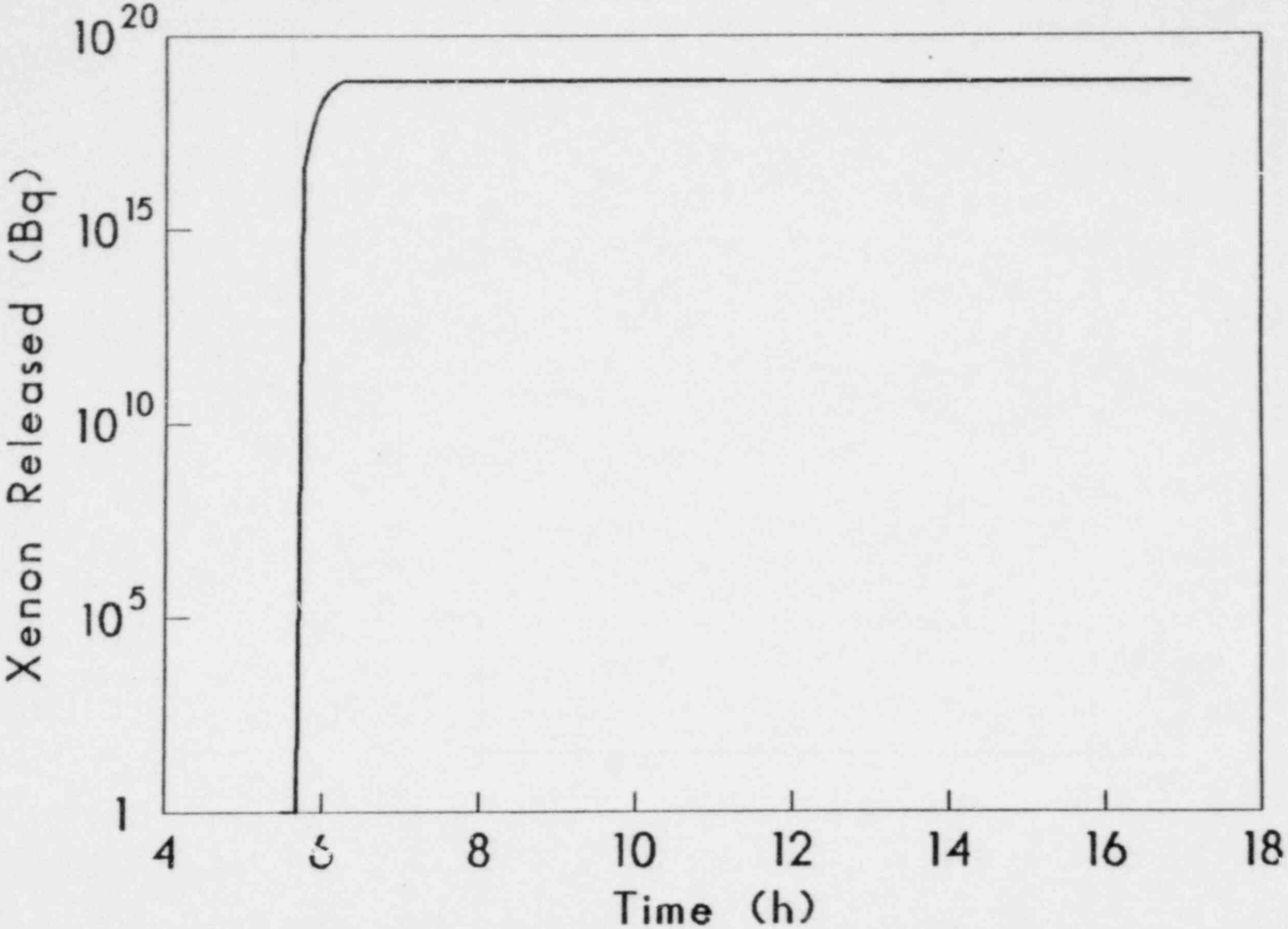
Table 2. Preliminary noble gas results

Location	Xenon inventories (PBq)[a]			
	At core slump	At vessel failure	After failure of containment	30 minutes later
Total present	9820	9800	9720	9690
Fuel rods	3260	3050	3030	3020
Wetwell water	24	460	130	72
Wetwell air	260	5060	170	33
Drywell	20	1170	270	12
Refueling floor	0	0	1820	29
Atmosphere	0	0	520	6450

[a] 1.0 petabecquerel = 27,000 curies.

FIGURE 1

BF-SB1. XENON RELEASED FROM FUEL



DRAFT

FIGURE 2

BF-SB1. XENON RELEASED FROM PRIMARY CONTAINMENT

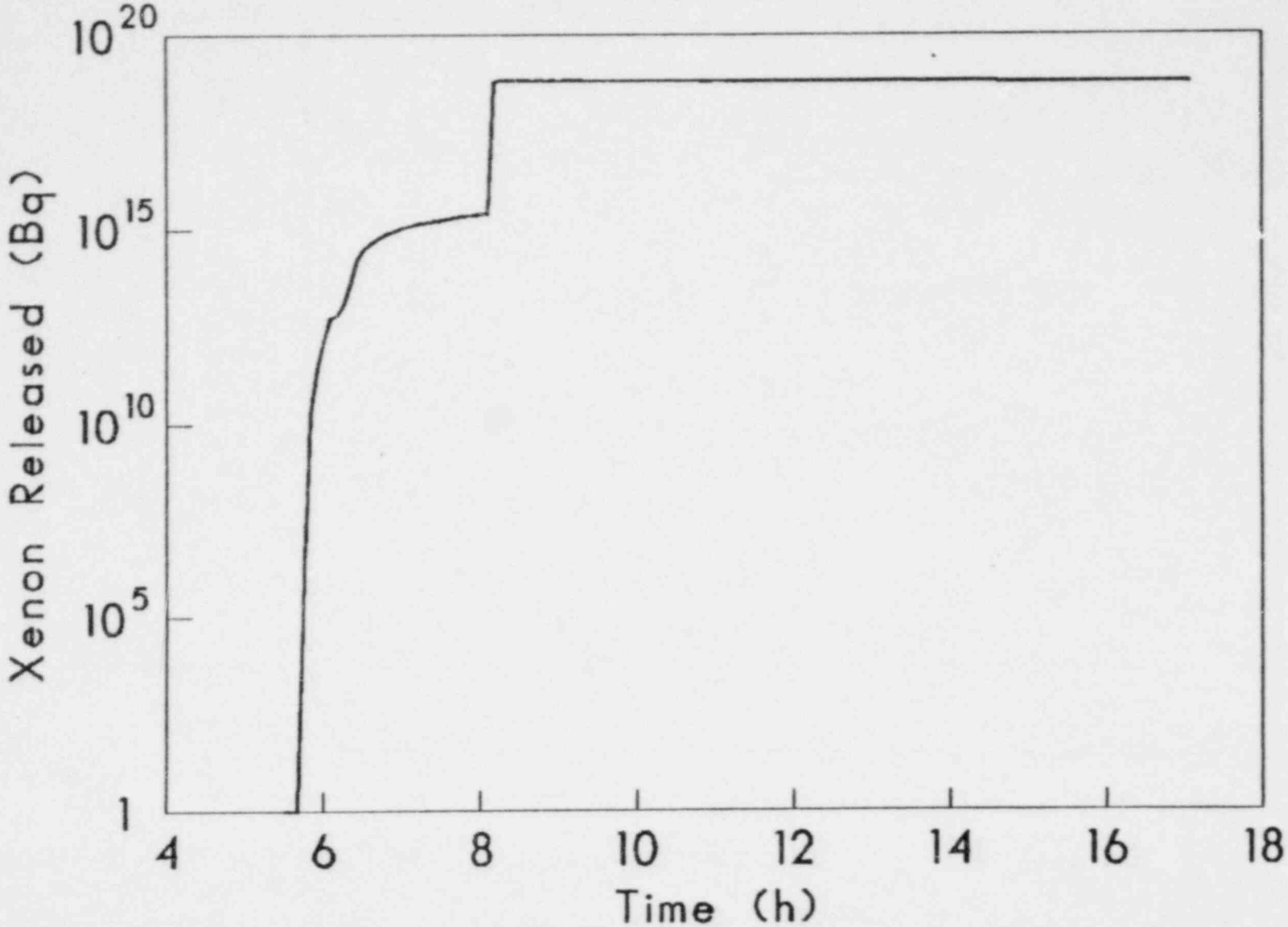


FIGURE 3

BF-SB1. XENON INVENTORY IN REFUELING AREA

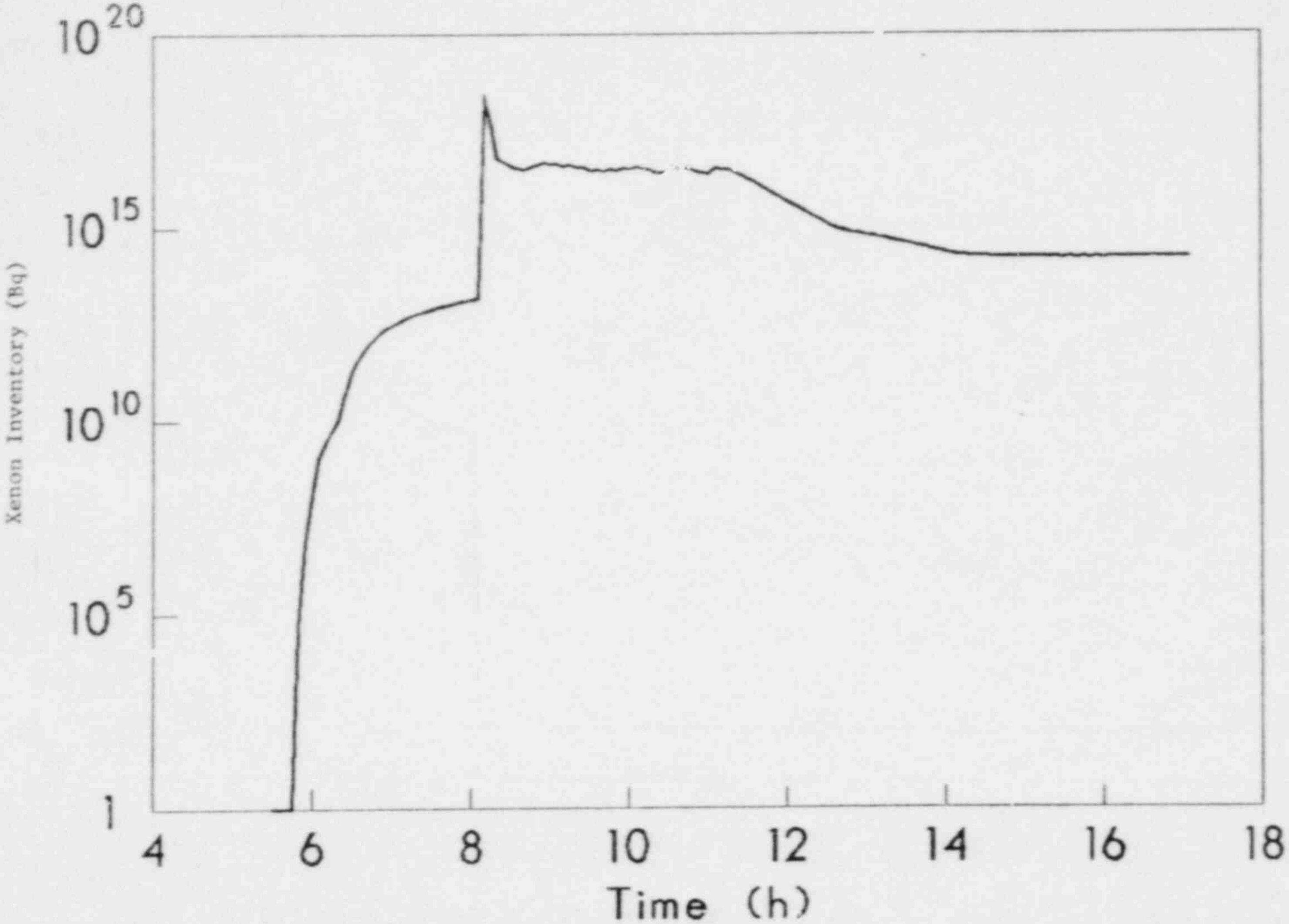
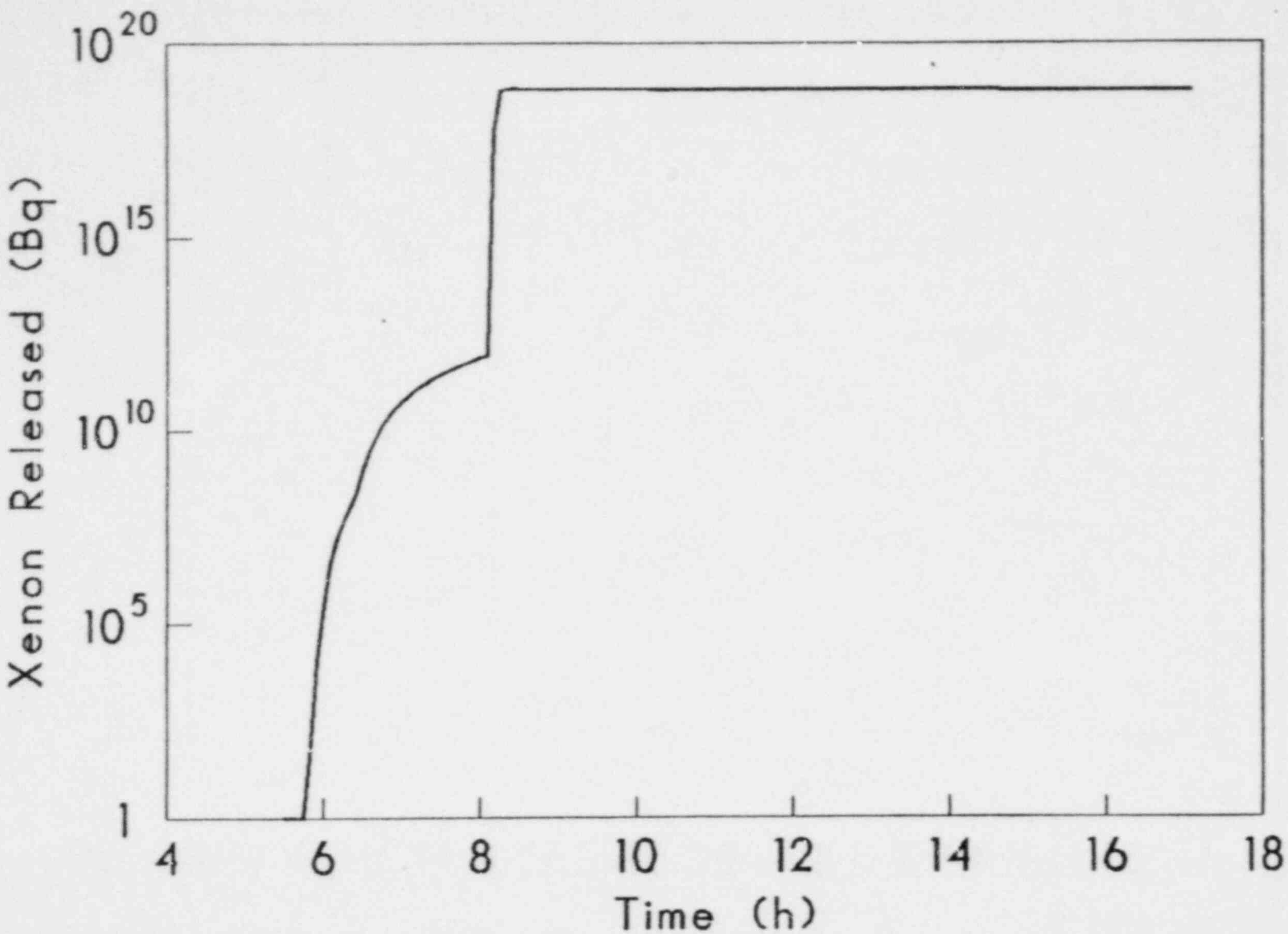


FIGURE 4

BF-SB1. XENON RELEASED TO THE ATMOSPHERE



24

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PROGRAM TITLE: Maintenance Error Model

PROGRAM MANAGER: P. M. Haas

ACTIVITY NUMBER: ORNL #41 88 55 03 6 (189 #B0461) NRC #60 19 03 10

TECHNICAL HIGHLIGHTS:

Review of the information gathered during ORNL and Applied Psychological Services (APS) staff informal interviews with NPP maintenance personnel has led to identification of some of the key variables likely to be of concern in the proposed model and typical outputs of use to potential users. More specific information on user requirements will be obtained by way of survey questionnaires and structured interviews. A total of approximately 10-15 interviews are currently planned with representative types of user organizations.

Preliminary task lists for maintenance jobs have been developed, and brief questionnaires have been sent to 10 representative nuclear power plant managers to determine variability of maintenance job titles across the industry.

A working outline of the proposed program plan for model development and validation was prepared by APS, and some of the basic model requirements have been delineated. A major consideration identified is that the model should be as self-contained as possible, that is, it should require a minimum of detailed user-input data and should not require extensive development of a data bank to be of use. In order to accomplish this, techniques will be examined which will permit relative rather than absolute quantitative estimates of performance. Thus, overall task or job performance could be estimated on the basis of a relatively few input data.

PROGRAM TITLE: Multirod Burst Tests

PROGRAM MANAGER: R. H. Chapman

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

TECHNICAL HIGHLIGHTS:

R. H. Chapman visited the NRC/RES offices in Silver Spring, MD, on April 30–May 1 to participate in another Fuel Testing Task Force meeting. The Task Force continues to consider research needs to support the degraded core cooling rulemaking process.

R. H. Chpaman traveled to Japan during the last half of May. Informal meetings were held May 14–15 with JAERI research staff personnel at Tokai to discuss MRBT evaluation of JAERI fuel pin simulators and how differences in experimental test equipment and procedures might explain the lack of agreement between JAERI and MRBT bundle test results. Two prepared papers were presented to the US-NRC/FRG-PNS/Japan-JAERI Ballooning Information Meeting and Workshop in Tokai the week of May 18. A meeting was also attended May 25 at JAERI Headquarters in Tokyo for informal discussions with senior NRC, PNS, and JAERI staff on severe core damage research.

J. L. Crowley visited the Canadian Chalk River Research Laboratories May 12–13 to observe posttest examination of the full length, nuclear fueled, test array tested recently by BNWL under LOCA conditions.

The B-5 (8 × 8) bundle was cast in epoxy and marked off for sectioning at approximately 60 axial positions. The bundle was delivered to the metallography laboratories for making the cuts and polishing and photographing the sections for strain determinations.

Two FLECHT-SEASET 21-rod bundles were cast in epoxy so that they can be sectioned by that program to determine distortion caused in recent thermal-hydraulic tests.

Preparations continue for the B-6 (8 × 8) test. Approximately 45% of the required simulators have been fabricated. Grids were obtained and are being modified for the test array. Drawing revisions are complete, and fabrication of other test components continues on schedule.

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:

Task 1: Monitoring Methods to Detect and Quantify Flow-Induced Vibrations of In-Vessel Components. We are completing the assessment of neutron noise for detection of internal vibrations in BWRs by performing two-dimensional calculations of the background noise caused by boiling. The background noise thus obtained will be compared with earlier calculations of in-core neutron detector response to postulated control rod vibrations to assess the sensitivity for vibration detection. This work should be completed on schedule by the end of the next reporting period.

The verification of advanced calculational methods for PWR systems was continued by comparing one-dimensional (TASK) calculations of ex-core neutron detector response to postulated fuel assembly vibrations with experimental data reported in the open literature. Our results indicate that the TASK calculations overpredict the detector response. We are currently studying the sensitivity of TASK results to the cross section generation techniques used. Work on this task is on schedule.

Task 2: Loose-Parts Monitoring Systems. This work is completed and a final report will be issued in August 1981.

Task 3: Surveillance and Diagnostics by Noise Analysis. We are continuing to acquire and analyze baseline data from Sequoyah-1.

Task 4: Primary Coolant Inventory Monitoring. TVA has not responded to our request for information about the availability and accuracy of instrumentation required for inventory monitoring. Therefore, we have not been able to complete the study of the feasibility of monitoring coolant inventory.

Meetings. We met with G. S. Lewis and other members of the new NRC Instrumentation and Control Branch on May 27 to review our FY1981 program and the program assumptions for FY1982.

We submitted titles for six papers to be presented at the Ninth Water Reactor Safety Research Information Meeting at the National Bureau of Standards on October 28, 1981.

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 and DOE 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 RES Program supported with 500 K, i.e., 66% of the total NSIC budget; the balance is from DOE).

During the month of May, the staff of the Nuclear Safety Information Center (a) processed 906 documents, (b) responded to 62 inquiries (of which 36 involved the technical staff), and (c) made 18 computer searches. The RECON System, which now has over 600 remote terminals (not 200 as I had reported earlier), reports that the NSIC data file was accessed 145 times between April 2 to 30 making it the fourth most utilized of the 30 data bases on RECON (see attached Table 1). During the past month, the NSIC staff received 7 visitors and participated in 2 meetings.

One NSIC report was issued in May [*Index to Nuclear Safety, Vol. 11(1) through Vol. 21(6)*, ORNL/NUREG/NSIC-186]. Several other NSIC reports are in various stages of preparation, including *Nuclear Power and Radiation in Perspective* (ORNL/NUREG/NSIC-161); *Role of Probability in Risk and Safety Analysis* (ORNL/NUREG/NSIC-167); *Annotated Bibliography on Fire and Fire Protection in Nuclear Facilities* (ORNL/NUREG/NSIC-172); *Summary and Bibliography of Safety-Related Events at Boiling Water Nuclear Power Plants as Reported in 1979* (ORNL/NUREG/NSIC-178); *Summary and Bibliography of Safety-Related Events at Pressurized Water Nuclear Power Plants as Reported in 1979* (ORNL/NUREG/NSIC-179); and *Nuclear Power Plant Operating Experience — 1979 Annual Report* (ORNL/NUREG/NSIC-180).

During the month of May, we received 5 foreign documents, all from the Federal Republic of Germany. 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 May 29, 1981, to H. H. Scott, RSR).

During the month of May, NSIC's Selective Dissemination of Information (SDI) was providing service to a total of 357 users, including 1 new user. 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; also the free distribution list has been screened. (Note: There are 8 paid subscriptions yet to expire.) However, this and other services to NRC, DOE and their subcontract personnel are not affected.

All technical articles for *Nuclear Safety* 22(5) were completed and mailed to NRC, DOE and TIC on May 22nd. The "current events" material (covering events which occurred during March and April) for *Nuclear Safety* 22(4) was completed by May 15 (except for the data on operating power reactors which was not yet available from NRC). Most technical articles for *Nuclear Safety* 22(6) have been received, submitted to peer review, and are in various stages of preparation. Final copies of *Nuclear Safety* 22(3) were received from the printer (via TIC) on May 27th.

TABLE 1 RECON DATA BASE ACTIVITY FROM 04-02-81 TO 04-30-81
(23 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	3849	4433	98557
NSA	(TIC) NUCLEAR SCIENCE ABSTRACTS	546	541	4674
WRA	(WRSIC) WATER RESOURCES ABSTRACTS	239	593	11273
NSC	(NSIC) NUCLEAR SAFETY INFO. CENTER	145	139	5449
GAP	(DOE) GENERAL AND PRACTICAL INFO.	135	72	1243
RIP	(DOE) ENERGY RESEARCH IN PROGRESS	132	130	1011
FED	(DOE/EIA) FEDERAL ENERGY DATA INDEX	109	53	75
EMI	(EMIC) ENV. MUTAGENS INFO.	97	102	6737
ESI	(EIC) ENV. SCIENCE INDEX	68	107	839
EIA	(EIC) ENERGY INFO. ABSTRACTS	58	49	294
ETI	(ETIC) ENVIRONMENTAL TERATOLOGY	57	54	1515
IPS	(TIC) ISSUES AND POLICY SUMMARIES	47	41	79
WRE	(WRSIC) WATER RESOURCE RESEARCH	37	66	554
ERG	(BERC) ENHANCED OIL AND GAS RECOVERY	30	37	86
PRD	(TIC/NRC) POWER REACTOR DOCKETS	25	30	-
TUL	(U. TULSA) TULSA DATA BASE	25	32	800
NRC	(LC) NATIONAL REFERRAL CENTER	22	29	22
SLR	(FRANKLIN) SOLAR DATA BASE	21	39	481
EIS	(TIRC) EPIDEMIOLOGY INFO. SYSTEM	20	32	4
GID	GOVERNMENT & INDUSTRY DATA EXCHANGE	20	80	1
NSR	(NDP) NUCLEAR STRUCTURE REFERENCE	19	27	43
SER	(TIC) SERIAL TITLES DATA BASE	17	12	-
RSI	(RSIC) RADIATION SHIELDING INFO.	15	49	446
OGR	OIL AND GAS RESERVE FILE	14	13	-
NES	(NESC) NATIONAL ENERGY SOFTWARE	13	8	-
SUP	(TIC) THESAURUS SUPPLEMENT	13	37	1296
CIM	(DOE) CENTRAL INVENTORY OF MODELS	12	57	-
NTB	(NASA) NASA TECH BRIEF FILE	12	36	-
RSC	(RSIC) RADIATION SHIELDING CODES	2	1	-
API	(API) AMER. PETROLEUM DATA BASE	1	-	-

PROGRAM TITLE: Operational Aids for Reactor Operators

PROGRAM MANAGER: J. L. Anderson

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

TECHNICAL HIGHLIGHTS:

During May, staff members continued work on describing the operating crew's role. A generalized taxonomy of the operator's function has been developed using a top-down approach. The operator's functions have been divided into three broad categories: supervising plant operations, maintaining equipment and coordinating support actions. Each of these categories has been subdivided further with particular attention given to supervising plant operations.

Technology for Energy Corporation has completed the comparison of the Emergency Operating Procedures of three NSSS. The narrative describing this comparison and the resulting definition of the operator's role under emergency conditions is currently being prepared. In addition, an investigation of some of the excessive cognitive demands on the operator as determined from the emergency guidelines has been completed and is being documented.

Many of the results of the investigation of the operator's role are being prepared for presentation at the Halden Conference in Norway, June 15-19.

The literature search on operator acceptance and utilization of computerized aids had provided a sizable amount of information on the user oriented design of interactive systems. Additional sources are being sought for data on other aspects of acceptance such as training and operation. From the search to date, it appears that no quantitative measures for acceptance have been developed.

Biotechnology, Inc. began work on the allocation of function in the nuclear power plant control room. The project objectives and general approach were discussed in the initial meeting with ORNL staff on May 12. The work during the first weeks of the subcontracts has been related to reviewing available data and assessing the applicability of previous work in the aerospace industry.

ORNL staff presented a summary of the Operational Aids Program to Denwood F. Ross, Deputy Director of RES, NRC.

PROGRAM TITLE: Postaccident Iodine and Tellurium Chemistry

PROGRAM MANAGER: R. P. Wichner, L. M. Toth

ACTIVITY NUMBER: ORNL # 41 89 55 13 5 (189 #B0453-1)/NRC # 60 19 01 10

TECHNICAL HIGHLIGHTS:

A means of releasing I_2 into a previously equilibrated solution of 2500 ppm B as boric acid held at 150°C has been developed in order to study the kinetics of I_2 hydrolysis as a function of pH and concentration parameters. With this system we are able to examine the chemical reaction by following either reactant or product absorption spectra as a function of time and then determine the overall stoichiometry of the reaction. The current experiments have been performed at pH's of 6.5 and 7.0 for $[I_2] = 1 \times 10^{-4}$ M. This very fast reaction goes to 90% completion within one second after release of I_2 (by rupture of a glass capsule containing a weighed amount) via

$$3I_2 + 3H_2O \approx IO_3^- + 5I^- + 6H^+ .$$

It is not possible to follow the kinetics of this reaction at times less than one second because the spectrometer instrumentation is limited by the response time of the recorder. Nevertheless, we presently know 1) that through an analysis of the mass balance of reactants and products, the overall stoichiometry of the reaction is as written above and therefore, 2) the titanium cell will be satisfactory for studying these kinetic reactions at pH values >6 because there is no indication of corrosion reactions that would otherwise alter the above stoichiometry.

PROGRAM TITLE: PWR Blowdown Heat Transfer-Separate Effects

PROGRAM MANAGER: W. G. Craddick

ACTIVITY NUMBER: ORNL #41 89 55 10 3 (189 #B0125)/NRC 60 19 11 01

TECHNICAL HIGHLIGHTS:

Task 1: Single-Rod Loop Testing — This task has been completed.

Task 2: Analysis — A computer code has been developed for analysis of the quasi-steady-state tests (test series 3.07.9) in the Thermal Hydraulic Test Facility (THTF). The code determines the test section heat balance and local fluid conditions and then [after reading fuel rod simulator (FRS) thermophysical property data from a coefficient data tape] computes the local FRS surface temperature and heat flux at each FRS sheath thermocouple position. The code has been developed, debugged, and tested on THTF test 3.07.9J). The code also computes the uncertainties in the local FRS surface temperature and heat flux at each FRS sheath thermocouple position.

The interim report for Small Break LOCA Heat transfer Test Series II (3.09.10I-X) has been completed. Preparations are underway to ship data tapes for tests 3.09.10I-X to the INEL data bank. Shipment is expected to be complete by July 1. Work on the data report for Small Break LOCA Heat Transfer Test Series I (3.02.10C-H) is 80% complete. A draft report will be issued in June.

Task 3: THTF Operations — Work is continuing on the facility description document. Work on test specific instrument application drawings is 70% completed. Work on test specific isometric system drawings is 90% completed.

Task 4: Two-Phase Instrument Development — Work on providing calculated mass flows for THTF test data reports is 90% completed. The mass flow code AMICON is used to generate the time dependent mass flows at the instrumented spool pieces.

Work is continuing on estimating mass flux uncertainties in two-phase flow for THTF spool piece data. Mass flux models using the spool piece instruments (turbine meter, drag disk, and gamma densitometer) are being applied to the outlet spool piece data from the steady-state upflow film boiling tests (3.07.9). The test section subcooled inlet flow provides a reference standard for model comparisons and uncertainty estimates. The work is 80% completed.

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 and Field Data Collection. Data collection by GPC on the BWR (Browns Ferry) simulator is proceeding on schedule. For "Utility A", Phase-1 Requalification Training is complete (14 operator crews). Data collection for 12 of 14 crews in the Phase-2 training of Utility A and for 10 of 12 groups from Utilities B and C have been completed. Data reduction is complete for Utility A, Phase-1, and is approximately 50% complete for the remaining data tapes that have been collected.

Field data collection was completed by MSU/CNS for two additional utilities (4 units). The report on the analysis of the PWR data has not yet been completed.

Task Analysis Pilot Study. A literature survey of task/analysis formats was conducted to identify common elements and differences. Berliner codes were reviewed for redundancy or lack of clarity in terminology for "specific behaviors". A list of terms and definitions applicable to nuclear plant operations is being prepared.

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:

Spectral densities as a function of detector positions were formed using the calculated detector responses from the JPRKINETICS code. The quantities of interest, the ratios of spectral densities, were then calculated and plotted as a function of detector position. These results are encouraging because they show that correct results can be obtained with the detectors placed in many positions covering a large region in a reactor core. They also show that erroneous results are obtained when two detectors are placed adjacent to each other - a behavior observed in some previous experiments.

At present, the JPRKINETICS results are being further evaluated to verify that they have been interpreted correctly.

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54. M. L. Picklesimer, RES-NRC
55. J. Pidkowicz, DOE/ORO
56. L. N. Rib, RES-NRC
57. C. Z. Serpan, RES-NRC
58. M. Silberberg, RES-NRC
59. H. Sullivan, RES-NRC
60. L. S. Tong, RES-NRC
61. M. Vagins, RES-NRC
62. R. W. Wright, RES-NRC
63-65. Office of Reactor Research Coordination, DOE
66. Assistant Manager for Energy Research and Development, DOE-ORO
67-68. Technical Information Center
69-70. Division of Technical Information and Document Control, NRC