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INSTRUMENTATION AND CONTROLS DIVISION PROGRESS REPORT FOR THE PERIOD SEPTEMBER 1, 1980 TO JULY 1, 1982
VOLUME I

F. R. Mynatt, Director

DISCLAIMER

LaWanda E. Klobe, Editor

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ORNL-5197	Period Ending September 1, 1976
ORNL-5482	Period Ending September 1, 1978
ORNL-5483	Period Ending September 1, 1978
ORNL-5758	Period Ending September 1, 1980
ORNL-5759	Period Ending September 1, 1980

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OVERVIEW

F. R. Mynatt

The ORNL Instrumentation and Controls Division performs basic measurement science research, development and design engineering, specialized instrument fabrication and production, and maintenance services for instruments, electronics, and computers. The I&C Division is one of the largest R&D organizations of its type, and it exists as the result of an organizational strategy to situate ORNL's instrumentation and controls related disciplines in one dedicated functional organization to enhance the Laboratory's expertise and capability in this rapidly expanding, innovative area of technology.

The primary mission of the Instrumentation and Controls Division is to support the programs and projects of the Oak Ridge National Laboratory by applying expertise and capabilities as required to complement the efforts of others in performing basic research and mission-oriented technology development. The contribution of I&C may be a few hours of maintenance service, the fabrication of a special instrument or instrumentation system, the performance of a research and development task in the I&C facilities, or the assignment of from one to many I&C engineers and scientists to a multidisciplinary team working in a specific research area or development project in an ORNL program or division. In its support and maintenance work, the role of the I&C Division is to provide a level of expertise appropriate to complete a job successfully at minimum overall cost and time schedule—a role which involves I&C in almost all ORNL activities. During this reporting period the Division was typically involved in an average of 338 R&D support engineering tasks and 423 maintenance tasks at any given time. During FY 1982 the level of the R&D support engineering effort was $8.2 million and the level of maintenance effort was $5.1 million. In this progress report the I&C Division's R&D support engineering and maintenance work is described in overviews of the engineering sections and the maintenance functions, as well as in specific articles on work in progress and abstracts of published work. A complete list of tasks performed is not provided in order to concentrate on the larger tasks in the Division's work.

The second role of the I&C Division is the performance of basic and applied instrumentation research and development tasks which are in support of ORNL programs and which are of sufficient scope that the I&C effort constitutes a separate program element with direct funding and management responsibility within the Division. The principal funding agencies for such programs are the U.S. Department of Energy, the U.S. Nuclear Regulatory Commission, the Federal Emergency Management Agency, the Department of Defense (U.S. Navy), and the Electric Power Research Institute. During FY 1982 the level of effort in this area was $9.9 million. As might be expected, the scope of the I&C work in this category is comprehensive and therefore suitable for publication. Essentially all major R&D efforts of the Division are represented in this report, either by summaries or abstracts of published reports, articles, and meeting papers, or as summaries of work not yet published.

Figure 1 illustrates the levels of effort in research and development, R&D support engineering, and maintenance over the past five years, and Table 1 contains the dollar amounts and average tasks over the same period.

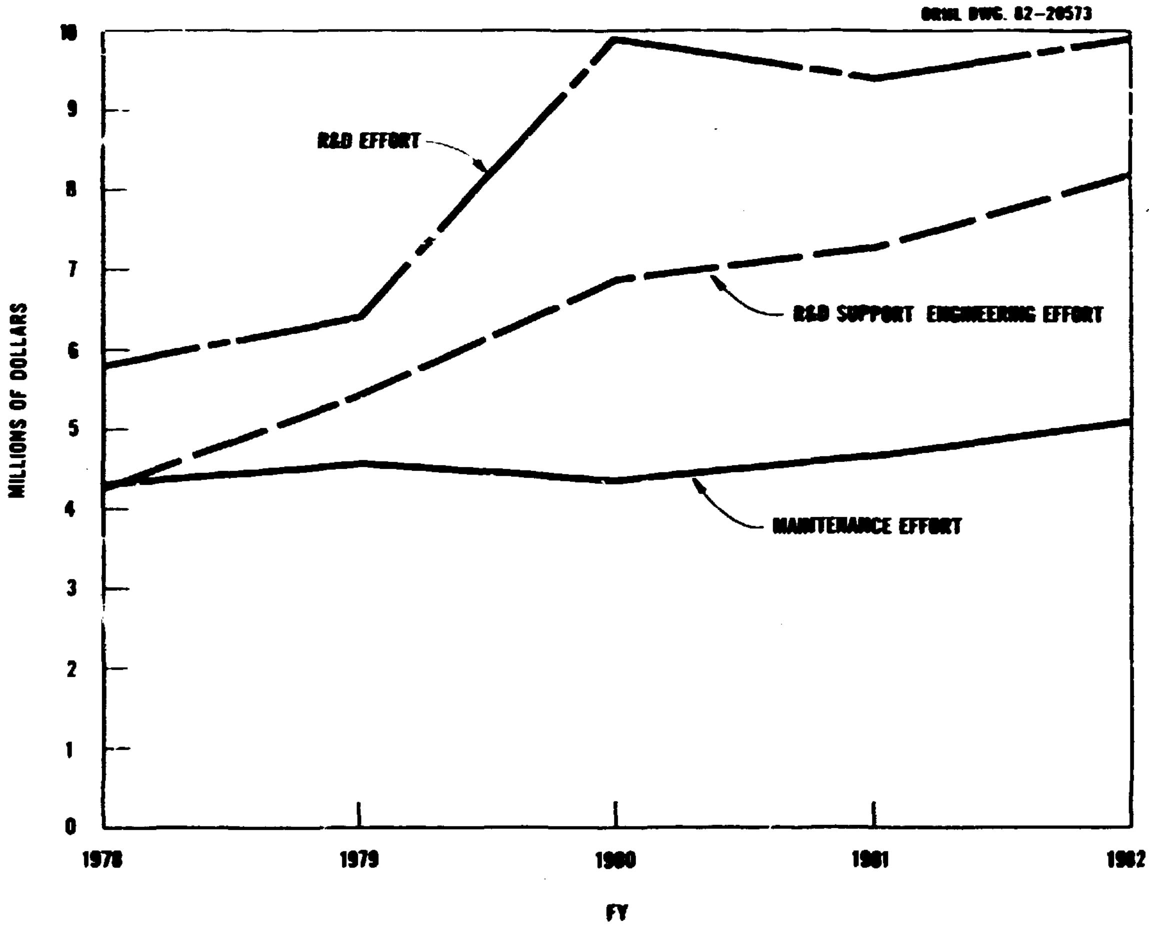

Fig. 1. Level of R & D support engineering effort, maintenance effort, and research and development effort in the Instrumentation and Controls Division, FY 1978 through FY 1982.

Table 1. Level of Effort in Dollars, FY 1978 Through FY 1982

	R&D Support Engineering Effort	Maintenance Effort	R&D Effort
	$	$	$
FY 1978	4,245,000	4,248,000	5,807,718
FY 1979	5,481,000	4,562,000	6,430,999
FY 1980	6,905,000	4,380,000	9,929,000
FY 1981	7,318,000	4,673,000	9,411,000
FY 1982	8,185,000	5,103,000	9,926,000

Because of the explosive, innovative growth of electronics-based technology, I&C faces the difficult challenge of transferring new technology into the Division and merging it with new in-house developments for application to ORNL projects, and also of transferring out the new technology developed in-house. In the case of mission-oriented development, I&C's task is to utilize available technology to the extent possible and augment it with the in-house development required to accomplish the objective. In-house development and fabrication capabilities in I&C, while sufficient, necessarily operate at a very low volume relative to the total instrumentation effort. Maintaining and even improving these capabilities is another major challenge.

Development engineering in the I&C Division is characterized by laboratory-based design, development, and testing. Emphasis is on the combined theoretical, design, and experimental capabilities of each individual or small team, an approach which results in well-integrated development. Also, modern techniques such as computer-aided design and simulations are among the new tools being introduced. In the real-time computer application work a simultaneous hardware/software development approach is proving much more effective than the separate efforts typical of this technical area.

Because of the Division's commitment to the experimental development engineering approach, I&C facilities are characterized by numerous laboratories and laboratory/office combinations. Construction now in progress will provide an additional 7200 sq ft of office and laboratory space to relieve the overcrowded conditions that now exist and thereby improve the effectiveness of our technical staff.

The wide range of activities in the I&C Division and its mixture of primary and supporting roles in research and development result in considerable confusion in determining the areas in which I&C is ahead of and those in which it is following the state of the art. Table 2 lists the areas in which the Division currently is at the leading edge of the technology and plans to maintain its lead. The table does not include every area in which individual I&C staff members have leading expertise, but rather represents the broad areas that are of continuing importance to the Division.

During the past two years the I&C Division has taken a more aggressive position with respect to the pursuit of patents and I•R 100 Awards. In previous years the tendency of the Department of Energy was to seek only those patents necessary to protect the public interest, but now patents are supported on almost all appropriate developments. It is recognized that patents and I•R 100 Awards serve as excellent recognition of achievements in hardware and sometimes software development, and that they complement the role of publications as indicators of the performance of the Division staff. Table 3 lists the patent applications filed and patents assigned during the reporting period, and Table 4 lists the I•R 100 nominations submitted by ORNL on behalf of the I&C Division and indicates awards received. The sizable number of patents and award nominations, along with the publication record represented in this report, demonstrates that the I&C Division has been very productive during this reporting interval.

Table 2. Areas in Which the ORNL Instrumentation and Controls Division Leads the Technology

Thermometry
Radiation Detection
Nuclear Reactor Instrumentation and Controls
Automated Systems for Incipient Failure Detection
Nondestructive Assay of Reactor Fuels
Measurement of Nuclear Subcriticality
Applied Artificial Intelligence
High-Speed Communications
Real-Time Computer Systems for Data Acquisition and Control
Electromagnetic Interference Technology
In-Line Process Analytical Instrumentation
Fiber Optics in Instrumentation
Single- and Two-Phase Flow Instrumentation
Diagnostic Instrumentation
Microprocessor Application in Smart Instruments
Process Measurement and Control
Radiation Pyrometry
Robotics and Teleoperation Development
Ultrasonic Instrumentation
Postition-Sensitive Proportional Detectors
Instruments with Two-Dimensional Imaging
Compact and Portable Complex Instruments

Table 3. Patent Applications Filed and Patents Assigned September 1, 1980 Through July 1, 1982

CNID No.	Names	Description	Patent No. & Date
3741	K. R. Carr	Side Welded Thermocouple	4,251,908 February 1981
3837	D. C. Agouridis R.J. Fox	CdTe Photovoltaic Radiation Detector	4,243,855 January 1981
3859	M. K. Kopp J. A. Hanson	Multianode Cylindrical Position-Sensitive Proportional Counter	4,289,967 September 1981
3915	R. A. Todd	Compensated Count Rate Circuit for Portable GM Survey Meters	4,292,539 September 1981
4014	D. C. Agouridis	Improved CdTe Photovoltaic Radiation Detector	Filed December 1980
4017	R.J. Fox	Solid State Radiation Detector Circuit	Filed September 1981
4035	M. K. Kopp K. H. Valentine	Parallel Plate Fission Counter with Large Plate Area	Filed April 1981
4052	M. K. Kopp K. H. Valentine	Neutron Flux Profile Monitor for Use in Fission Reactor	Filed September 1981
4086	D. R. McNeilly W. R. Miller	Transmission-Medium-Effects Control for Intrusion Systems	Filed January 1982
4092	J. B. Davidson A. L. Case	TV-Based Locating & Measuring System for Two-Dimensional Gels	Filed December 1981
4097	F. W. Manning	Integrated Charger for Self-Contained Dosimeters	Filed March 1982
4113	F. W. Manning	Portable Battery-Free Charger for Radiation Dosimeters	Filed January 1982
4116	V. R. Brantley D. P. Miller	Heat Pump COP Measurement Method and Apparatus	Filed May 1982
4135	R. L. Anderson G. N. Miller	Reactor Coolant Level Sensor	Filed October 1981
4163	P. Angelini* A. J. Caputo* R. E. Hutchens W. J. Lackey* D. P. Stinton*	Method for Forming Microspheres for Encapsulation of Nuclear Waste	Filed March 1982

*ORNL Metals & Ceramics Division.

Table 4. 1982 I-R 100 Awards

Winners

ORNL In-Core Temperature, Density, and Level Measurement System
- G. N. Miller
- R. L. Anderson
- S. C. Rogers

ORNL Inductively Coupled Plasma (ICP) Spectrometer
- J. H. Stewart, Jr. (Analytical Chemistry Division)
- R. T. Roseberry, Sr.

Nominees

ORNL Autoranging Beta-Gamma Survey Instrument
- H. R. Brashear, Jr.
- M. L. Bauer
- R. A. Todd
- R. A. Maples

ORNL Cantilever Coriolis Flowmeter
- W. R. Hamel

ORNL Impedence Probe Two-Phase Flow Measurement System
- B. G. Eads
- J. E. Hardy (Engineering Technology Division)
- M. B. Herskovitz
- R. A. Hess
- J. O. Hylton
- P. A. Jallouk
- W. H. Leavell
- D. B. Lloyd (Engineering Technology Division)
- A. J. Moorhead (Metals & Ceramics Division)
- C. S. Morgan
- J. A. Mullens
- H. R. Payne (UCC-ND Engineering Division)
- M. J. Roberts

ORNL I.M.P. (In-line Multiwavelength Photometer)
- D. D. McCue
- D. T. Bostic (Analytical Chemistry Division)
- R. E. Harper
- J. E. Strain (Analytical Chemistry Division)
- M. L. Bauer

ORNL Multi-tap Transformer Controller
- D. W. McDonald

Section 1

REACTOR SYSTEMS

1.1. Dynamic Analysis

1.2. Surveillance & Diagnostic Methods

1.3. Design & Evaluation

1.4. Detectors

1.5. Facilities Support

1.6. Process Instrumentation Development

1.7. Special Assignments

1.0. OVERVIEW

L. C. Oakes

The Reactor Systems Section (RSS) is committed to the development of advanced measurement and surveillance systems and to the development of instruments, components, systems, and techniques for the control and protection of nuclear reactors. Past efforts have included the design and installation of control and protection systems for all research reactors constructed at ORNL and substantial contributions to the design and fabrication of systems for a number of other research and test reactors throughout the country. The designs were confirmed by total plant simulations and preoperational testing in our laboratories and at other ORNL reactor facilities. The Section has retained the responsibility for maintaining the control and protection systems for ORNL reactors and for upgrading them as components and techniques have improved.

Extensive experience has been acquired by the RSS staff in the design of measurement, control, and protection systems for virtually all reactor types. In the last decade, however, emphasis has shifted from design and installation of reactor instrumentation to development engineering in support of LWR, HTGR, and ABR systems. These activities include simulation and analysis of a wide spectrum of dynamic systems, design analysis, review, and confirmatory research for the NRC, and the development of standards. The Section has likewise become more involved with basic research, development of surveillance and diagnostic systems, sensor systems, subcriticality measurement methods, and special projects in support of the CRBR and FEMA.

The Reactor Systems Section has the responsibility for I&C management of fission reactor instrumentation and control systems for LWRs and LMFBRs. The Section is also responsible for management of a program to integrate various studies of pressurized thermal shock (PTS), as well as the management of ORNL-NRC programs for Reactor Regulation.

The RSS technical staff consists of electronic and electrical engineers, physicists, nuclear engineers, mechanical engineers, draftsmen, and technicians. Their activities are described below by functional group.

Dynamic Analysis Group

This group is responsible for computer analyses of engineered and/or natural systems whose behavior is of concern to development or evaluation projects. Analytical studies are made of system dynamic behavior, instrumentation and control systems designs, and overall stability. Current activities include simulations of gas-cooled power reactor systems, a 300-kW photovoltaic electrical supply, boiling water reactor stability, and PWR plant systems for an in-depth evaluation of control system impact upon safety. In addition, there is a sizable NRC support program to review the transient analysis codes used by reactor vendors and to assist with the reduction of data from experiments on liquid-metal-cooled fuel bundles. A powerful hybrid computer facility is this group's principal tool for performing system simulations.

Surveillance and Diagnostic Methods Group

The activities of this group are directed toward the development and demonstration of new or improved surveillance and diagnostic measurement methods for enhancing the safety, reliability, and operability of nuclear power plants and associated facilities.

Research activities of the group include investigations leading to the development of advanced methods for two-phase flow measurements, monitoring systems for use on rotating machinery in a reprocessing plant, and analysis methods for BWR neutron noise data. The group also carries out several research assessments and consulting activities for the NRC. The principal consultation areas are loose-parts monitoring, neutron and process signal noise analysis, and core-internals vibration monitoring. The research tasks include the advancement of present technology for sizing and locating loose parts, BWR stability monitoring, modeling stochastic processes, and performing automated surveillance and diagnostics. The group is also involved in the demonstration of a prototypic automated noise surveillance and diagnostic system during the first fuel cycle of a commercial PWR.

Design and Evaluation Group

The technical activities of this group are characterized by four major areas: (a) man-machine interactions, (b) design assistance for reactors and experimental facilities, (c) technical assistance to the NRC, and (d) licensing casework review for the NRC. The work performed in these areas is amplified in the following paragraphs.

(a) Studies are made of industry needs and design requirements in the area of man-machine interactions. For example, a program plan was developed for the Department of Energy (DOE) to assist in guiding a large-scale R&D effort to achieve needed improvements in integrating operators into the information channels of nuclear plants. Similarly, assistance is being provided to the NRC Office of Nuclear Regulatory Research in defining functional and safety design requirements for operator aid systems and in establishing needs and measures of improvement for such aids. This effort includes the assessment of Disturbance Analysis and Surveillance Systems (DASS).

(b) Design assistance is provided to ORNL research reactors and experimental facilities. For example, the upgrade of the ORR safety and control systems that will incorporate the latest in safety and control philosophies.

(c) Technical assistance to the NRC is provided in a variety of areas, including reviews for adequacy and reliability, assessment, technique development, and recommendations for regulatory positions and guides. Examples of such assistance are the study of the so-called unresolved safety issues of nuclear power plants, such as station blackouts (dealing with the reliability of emergency ac power systems), and inadequate core cooling (dealing with special instrumentation for measuring in-vessel coolant level).

(d) Licensing case reviews are performed as requested by NRC in the areas of electrical power systems, instrumentation and control systems, and environmental qualification of safety-related (class 1E) equipment.

Detectors Group

This group is responsible for the research and development of advanced detectors and detection systems. Fission counter channels are being developed to replace BF_3 detectors for subcriticality and startup measurements of ABR cores. These channels require fission counters of extremely high sensitivity—10 counts/(nv·s)—that must operate at 150°C in gamma fluxes of up to 10^5 R/h. In addition, a self-contained, remote-signal-transmission neutron counting channel is being developed for subcriticality determinations during the initial core loading of an ABR.

The group is also involved in the development of position-sensitive proportional counters that provide imaging capabilities to greatly reduce data acquisition time and improve spatial resolution in many scientific disciplines—including nuclear medicine, chromatography, space applications, and small-angle scattering experiments with neutrons and photons.

Facilities Support Group

This group maintains the control and protection systems of operating ORNL reactors and associated experiments, modifies these systems when needed or requested, maintains up-to-date system drawings, initiates and maintains test procedures, and keeps records of the failure rates and other historical maintenance data on system components.

Other activities of this group include the review of CRBR balance-of-plant engineering drawings for internal consistency and conformance to the system design descriptions and applicable codes and standards, and the maintenance of a library/repository for CRBR documents.

Process Instrumentation Development Group

Activities of this group include the development and qualification of thermometers—thermocouples, resistance thermometers, and the Johnson noise thermometer—for use in nuclear reactor plants and test facilities, with responsibilities being extended to include other reactor process instruments. Current projects include qualification of gamma thermometers for measuring in-core power, and demonstration of the use of Johnson noise thermometry for *in situ* calibration of the resistance thermometers widely used in reactor plants. Major interests of the group are development of methods for *in situ* calibration and response-time testing of thermometers and other process sensors.

Special Assignments Group

The research, development, and support activities of this group cover a broad range of subjects that include instrumentation for biological systems; measurements of particulate suspensions; radiation survey instrumentation (FEMA); subcriticality measurements for fission reactor startup, spent-fuel storage facilities, and reprocessing plants; nondestructive assay for total fissile mass of spent reactor fuel; plasma diagnostics; fusion energy engineering; and pressurized thermal shock of reactor vessels. Future involvement in the development of power delivery systems and in the evaluation of electromagnetic pulse (EMP) effects is anticipated.

1.0.1. NUCLEAR REACTOR SAFETY RESEARCH SINCE THREE MILE ISLAND

F. R. Mynatt

[Summary of invited paper in *Science* 216(4542), 131–35 (1982)]

The Three Mile Island nuclear power plant accident has resulted in redirection of reactor safety research priorities. The small release to the environment of radioactive iodine—13 to 17 curies in a total radioactivity release of 2.4 million to 13 million curies—has led to a new emphasis on the physical chemistry of fission product behavior in accidents; the fact that the nuclear core was severely damaged but did not melt down has opened a new accident regime—that of the degraded core; the role of the operators in the progression and severity of the accident has shifted emphasis from equipment reliability to human reliability. As research progresses in these areas, the technical base for regulation and risk analysis will change substantially.

1.1. Dynamic Analysis

1.1.1. INITIAL DYNAMIC SIMULATION OF AN HTGR SENSIBLE ENERGY TRANSPORT AND STORAGE PLANT

S. J. Ball N. E. Clapp, Jr.

(Abstract of ORNL/TM-8226, May 1982)

Dynamic models were developed for a General Atomic Company reference design of a high-temperature gas-cooled reactor sensible energy transport and storage (SETS) plant. The resulting computer code uses the IBM simulation language CSMP. The purpose of the program was to investigate the basic dynamic response behavior and controllability. The plant was found to have excellent inherent stability and control features.

1.1.2. SAFETY AND LICENSING ANALYSES FOR THE FORT ST. VRAIN HTGR

S. J. Ball
R. M. Harrington N. E. Clapp, Jr.

(Abstract of paper presented at The Third U.S.-Japan HTGR Safety Technology Seminar, Upton, New York, June 2–3, 1982)

The ORNL safety analysis program for the HTGR includes development and verification of system response simulation codes, and applications of these codes to specific Fort St. Vrain reactor licensing problems. Licensing studies addressed the oscillation problems and the concerns about large thermal stresses in the core support blocks during a postulated accident. Other work includes proposed experiment planning, TMI action plan applicability studies, and a new siting study on the 2240 MW(th) HTGR design.

1.1.3. HTGR SEVERE ACCIDENT SEQUENCE ANALYSIS

R. M. Harrington
S. J. Ball F. C. Kornegay*

(Abstract of paper presented at The Third U.S.-Japan HTGR Safety Technology Seminar, Upton, New York, June 2–3, 1982)

Thermal-hydraulic, fission product transport, and atmospheric dispersion calculations are presented for hypothetical severe accident release paths at the Fort St. Vrain (FSV) high temperature gas cooled reactor (HTGR). Off-site radiation exposures are calculated for assumed releases of 100% of the 24 hour post-shutdown core xenon and krypton inventory and 5.5% of the iodine inventory. The results show conditions under which dose avoidance measures would be desirable and demonstrate the importance of specific release characteristics such as effective release height.

*Energy Division.

1.1.4 SBLOCA OUTSIDE CONTAINMENT AT BROWNS FERRY UNIT ONE—ACCIDENT SEQUENCE ANALYSIS

W. A. Condon* **S. R. Greene**†
R. M. Harrington **S. A. Hodge**†

[Abstract of NUREG/CR-2672 (ORNL/TM-8119/V1), June 1982]

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a postulated Small-Break Loss of Coolant Accident (SBLOCA) outside of the primary containment. The break has been assumed to occur in the Scram Discharge Volume (SDV) piping immediately following a reactor scram that cannot be reset. Every effort has been made to employ the most realistic assumptions during the process of defining the sequence of events for this hypothetical accident. The events before core uncovery are discussed for both the accident sequence without operator action and for the sequences which would occur with operator action. Without operator action, the events after core uncovery would include core meltdown and subsequent containment failure and this event sequence has been determined through use of the MARCH code. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the environment is provided in Volume 2 of this report.

*Department of Nuclear Engineering, The University of Tennessee, Knoxville.
†Engineering Technology Division.

1.1.5. SCRAM DISCHARGE VOLUME BREAK STUDIES PART 1: ACCIDENT SEQUENCE ANALYSIS

R. M. Harrington **S. A. Hodge***

(Abstract of invited paper presented at Tenth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 15, 1982)

This paper describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a postulated small-break loss-of-coolant accident outside of the primary containment. The break has been assumed to occur in the scram discharge volume piping immediately following a reactor scram that cannot be reset. The events before core uncovery are discussed for both the worst-case accident sequence without operator action and for the more likely sequences with operator action. Without operator action, the events after core uncovery would include core meltdown and subsequent containment failure, and this event sequence has been determined through use of the MARCH code. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the environment is provided in the companion paper, Part 2: Fission Product Transport Analysis.

*Engineering Technology Division.

1.1.6. EXPERIMENTAL AND NUMERICAL THERMAL-HYDRAULIC RESULTS FROM A 61-PIN SIMULATED LMFBR SUBASSEMBLY

S. D. Rose* **P. A. Gnadt***
J. F. Dearing* **B. H. Montgomery***
N. E. Clapp **R. H. Morris***
M. H. Fontana* **J. L. Wantland***

[Summary of *Trans. Am. Nucl. Soc.* **34**, 880 (1980)]

The purpose of this work is to provide liquid-metal fast breeder reactor designers with thermal-hydraulic information from a 61-pin electrically heated out-of-core fuel subassembly. The operation of a 61-pin bundle has provided experimental data for computer code verification and provided information that will help to establish confidence in reactor design calculations for the 217-pin assemblies. The bundle was tested in an engineering-scale sodium loop—the thermal-hydraulic out-of-reactor safety (THORS) facility. It was identified as Bundle 9 and was designed to determine characteristics of a 61-pin bundle with Clinch River breeder reactor dimensions operating under normal and off-normal conditions.

The subchannel analysis code SABRE-1 is used to model a full representation of the 914.4-mm heated section of Bundle 9. A wire-wrap representation (by resistance coefficients) is an option used in this version of SABRE. Default values of all thermal-hydraulic correlation parameters are used. A change has been made to the transverse conduction equation to include effects of transverse thermal conduction through the boron nitride fuel pin simulators. Results from several steady-state tests have been compared with SABRE predictions; comparisons from three tests are presented here.

*Engineering Technology Division.

1.2. Surveillance & Diagnostic Methods

1.2.1 A SURVEY OF PROPOSED FUNCTIONAL REQUIREMENTS FOR A DISTURBANCE ANALYSIS AND SURVEILLANCE SYSTEM

W. H. Sides, Jr.
C. B. Oh* P. F. Knight*

[Abstract of NUREG/CR-1760 (ORNL/NUREG/TM-396), October 1980]

A program to enhance the capabilities of operators of nuclear power plants is being pursued by the U.S. Nuclear Regulatory Commission (NRC). The program includes improvements in plant monitoring, diagnostic and corrective action aids, operator-process communication, and operator training. Concerning diagnostic aids, a disturbance analysis and surveillance system (DASS) was considered which would monitor the plant for the approach or occurrence of disturbance conditions and would assist the operator in returning the plant to normal operation or to help mitigate the consequences of a failure condition or misoperation. (A disturbance condition is defined as a condition in which a physical parameter, component, system, or some other element of the plant is outside of or approaching the tolerance limits for a particular mode of operation.)

The NRC had requested Oak Ridge National Laboratory to survey the functional requirements being proposed for a DASS. In fulfilling this task, the proposed requirements were categorized according to whether they could be realized in the short term and backfitted to existing plants or whether they could be realized only in the long term by incorporation into new plant designs. In addition, several recommendations concerning DASS development were made for consideration by the NRC. Finally, the effects of human factors on a DASS were evaluated, and the results are discussed in this report.

The result of the survey indicates that the scope envisioned for a DASS ranges from the display of a few critical plant parameters to the inclusion of extensive cause-consequence analyses and comprehensive plant dynamic models to predict behavior during upset conditions. A principal recommendation from the task is that fundamental decisions be made concerning the role and responsibilities of the operator during plant disturbance conditions, particularly in rapidly developing situations, taking into account the limitations of humans under stress to handle large quantities of information. A licensing question to be answered in the application of a DASS is whether the DASS will be allowed to advise the operator to alter the function or sequence of any safety system or engineered safety feature in the course of a disturbance condition on a time scale which does not allow adequate assessment of the correctness of DASS information. The essential impact on safety is the consequence of DASS failure or misinformation.

*Technology for Energy Corp., Knoxville, Tennessee.

1.2.2. DEMONSTRATION OF AN ON-LINE REACTOR NOISE SURVEILLANCE SYSTEM AT A PWR

N. E. Clapp C. M. Smith

(Abstract of invited paper presented at Ninth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 26–30, 1981)

A brief description of the on-line reactor noise surveillance system at a PWR is given. A list of signals being monitored, system software flow chart, and preliminary data obtained by the system is presented.

1.2.3. ADVANCES IN AUTOMATED NOISE DATA ACQUISITION AND NOISE SOURCE MODELING FOR POWER REACTORS

N. E. Clapp, Jr. F. J. Sweeney
R. C. Kryter J. A. Renier*

[Abstract of *Prog. Nucl. Energy* 9, 493–504 (1981)]

A newly-expanded program, directed toward achieving a better appreciation of both the strengths and limitations of on-line, noise-based, long-term surveillance programs for nuclear reactors, is described. Initial results in the complementary experimental (acquisition and automated screening of noise signatures) and theoretical (stochastic modeling of likely noise sources) areas of investigation are given.

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1.2.4. BASE NEUTRON NOISE IN PWRs

G. Kosály* R. W. Albrecht*
D. J. Dailey* D. N. Fry

(Abstract of invited paper presented at Ninth NRC Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 26–30, 1981)

Considerable activity[1,2] has been devoted in recent years to the use of neutron noise for investigation of problems in pressurized-water reactors (PWRs). The investigators have found that neutron noise provides an effective way to monitor reactor internal vibrations such as vertical and lateral core motion; core support barrel and thermal shield shell modes, bending modes of fuel assemblies, and control rod vibrations. However, noise analysts have also concluded that diagnosis of a problem is easier if baseline data for normal plant operation is available. Therefore, we have obtained ex-core neutron noise signatures from eight PWRs to determine the similarity of signatures between plants and to build a base of data to determine the sources of neutron noise and thus the potential diagnostic information contained in the data.

Preliminary analysis of the frequency spectra of the neutron noise shows spectra features that can be identified with coolant temperature fluctuations, fuel bundle vibrations, core support barrel-pressure vessel relative motion; vibrations induced by the primary coolant pumps and core support barrel shell mode vibrations. The frequencies and amplitudes of these contributions to the neutron noise vary from plant-to-plant and during a fuel cycle. Even so, neutron noise associated with anomalous core support barrel motion can be recognized because it has higher amplitude and different spectral content than any of the signatures obtained in the baseline data acquisition-program.

In summary we conclude that: (1) ex-core neutron noise contains information about the vibration of components in the pressure vessel; (2) baseline signature acquisition can aid understanding of plant specific vibration frequencies and provide a basis for diagnosis of future problems if they occur; and (3) abnormal core support barrel vibration can most likely be detected over and above the plant-to-plant signature variation observed thus far.

*Department of Nuclear Engineering, University of Washington, Seattle.

1. J. A. Thie, "Core Motion Monitoring", *Nucl. Technol.* 45, 5–45 (1979).

2. G. Kosály, "Noise Investigations in Boiling-Water and Pressurized-Water Reactors", *Prog. Nucl. Energy* 5, 149 (1980).

1.2.5 CARDIOGRAMA: A STOCHASTIC, SEMI-EMPIRICAL METHODOLOGY FOR POWER REACTOR SURVEILLANCE AND DIAGNOSTICS

J. A. March-Leuba*
G. de Saussure† R. B. Perez†

[Abstract of *Trans. Am. Nucl. Soc.* 39, 951–52 (1981)]

The utilization of stochastic methods (reactor "noise") for power reactor diagnostics and surveillance applications is by now a relatively well-established technique.[1] In this technique, the power spectral density (PSD) of the fluctuations of a specified state variable is often used to define the reactor's "signature" at a given configuration. Typically, this PSD possesses considerable structure as a function of frequency (i.e., peaks and valleys), and these features have been related with various degrees of success to specific causative mechanisms such as fuel element vibrations, core barrel motions, and thermal-hydraulic resonances. These features also form the basis for surveillance methods whereby changes in the structure of the PSD are detected and then correlated with altered plant conditions.

The purpose of the present work is to address the problem of handling efficiently the substantial amount of information involved in the application of reactor surveillance and diagnostics methods. Specifically, a methodology is described for (a) representing the PSDs parametrically, and (b) detecting changes from the reactor's baseline PSD (normal signature).

To this end, the program CARDIOGRAMA has been implemented to (a) compile reactor PSDs and classify them on a basis of the number and location of their poles in the frequency domain, and (b) provide an algorithm for rapid, sensitive identification of changes in reactor signatures.

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†Engineering Physics Division.

1. D. N. Fry et al., *Nucl. Technol.* 43, 42 (1979); also, Y. Ando et al., *Prog. Nucl. Energy* 1, 163 (1977).

1.2.6. DIAGNOSIS: A PRODUCTION SYSTEM THAT SIMULATES THE DIAGNOSTIC OF ANOMALIES IN THE PRIMARY SYSTEM OF A NUCLEAR POWER PLANT

Eduardo Lavenere Machado*
Adalberto José Soares*

(Abstract of paper presented at Association of Computing Machinery Conference, Gatlinburg, Tennessee, July 1981)

A production system specialized in the diagnosis of anomalies in the primary system of a nuclear power plant is described. The system not only automatically monitors the plant, watching for anomalous situations, but also it can isolate the source of an anomaly, and then explain to a human operator, in English, the problem, the actions to be taken, and its reasoning (how the rules were used in isolating the problem). The simplified model of a nuclear power plant used and the simple set of rules make the program not applicable to complex situations, but show how this type of system can be used as a tool to help plant operators. Some important features and characteristics of the program are discussed.

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1.2.7. SAMPLING CONSIDERATIONS FOR MULTILEVEL CROSSING ANALYSIS

Richard E. Woods* **Rafael C. Gonzalez***

[Abstract of *IEEE Trans. on Pattern Analysis and Machine Intelligence* PAMI-4(2) (March 1982)]

This paper examines the amplitude fluctuations of band-limited functions with bounded zeroth absolute moments, and the problems associated with estimating the level crossing profiles of these functions. Level crossings have received increased attention as features for pattern recognition because of their capability to provide information related to both amplitude and frequency behavior.

A detailed analysis of the average rate of change of band-limited functions is presented, including a derivation of the least upper bound on functions for which the zeroth absolute moments of the functions are bounded. The average rate of change of a function over an interval in which one endpoint of the interval is an extremum of the function is similarly bounded and used to establish a sampling rate which guarantees that between successive samples of a band-limited function with bounded zeroth absolute moment the function itself does not deviate from the amplitude interval defined by the samples by more than some predefined amplitude change. Based on these results, a theorem is developed which defines the sampling rate required to ensure that, in the estimation of level crossing profiles, no more than 2m ($m \geq 1$) level crossings of m levels are missed per extremum of the sampled function. It is shown that the sampling rate defined by this theorem reduces to the well-known Nyquist rate for the special case of zero crossing analysis.

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1.2.8. REDUCING THE SIZE OF A DATABASE BY USING PATTERN RECOGNITION TECHNIQUES

N. E. Clapp, Jr.

(Abstract of paper presented at ASIS 1982 Mid-Year Meeting at The University of Tennessee, Knoxville, June 13–16, 1982)

For some years, Oak Ridge National Laboratory has been engaged in a program to demonstrate continuous on-line surveillance of a nuclear power plant by using noise-related techniques. A computerized data acquisition system for noise-signal processing to obtain power spectral density data was built and temporarily installed at a nuclear power plant. In order to reduce the amount of spectral data acquired by the system, a pattern recognition procedure was developed to provide the capability of saving only the spectral data that were statistically different. The recognition procedure was made adaptive to provide the capability of determining from acquired data the amount of data variation required to be statistically different. The pattern recognition function embodies a number of descriptors that provide a means of recognizing changes.

The automated surveillance system with pattern recognition capabilities provides a valuable tool in obtaining spectral data for the analyst. The system monitors long-term noise behavior and provides concise storage of spectra and the plant status associated with the stored data.

1.2.9. COMMENTS ON "FORMULATION OF THE TWO GROUP STOCHASTIC FEINBERG-GALANIN EQUATIONS FOR HETEROGENEOUS LATTICES"

F. J. Sweeney F. C. Difilippo*

[Reprint of Technical Note in *Ann. Nucl. Energy* 9, 225 (1982)]

In a recent paper by Analytis (1981) the importance of heterogeneous effects in the interpretation of neutron noise in boiling-water reactors (BWRs) was stressed. However, the conclusion reached in this paper is not a new one since several previous works have addressed neutron wave propagation and BWR neutron noise in heterogeneous media.

Quddus *et al.* (1969) have shown that sources in heterogeneous media produce neutron waves composed of various modes. Each mode propagates with a characteristic relaxation length, one of which corresponds to the diffusion length of neutrons in the moderator. An implicit dispersion law was derived for the other modes.

A detailed study of BWR neutron noise by Sweeney (1979) using four neutron energy group numerical transport-theory calculations indicated that homogenization of the fuel and moderator led to predictions of incore neutron detector spatial sensitivity to noise sources which were inconsistent with experimental measurements. This work suggested that 'local' neutron noise may be attributed to coolant density variations perturbing the moderating and attenuating transmission path of neutrons between the fuel and detector locations (i.e. attenuation noise). In a subsequent paper, Sweeney and Robinson (1980) calculated the relative importance of attenuation effects with respect to neutron propagation through fission processes (i.e. reactivity effects) resulting from a moderator density variation. This work led to three important conclusions: (a) 'local' noise in a BWR is the result of neutron propagation primarily in the moderator while 'global' noise results from neutron propagation primarily through fission processes; (b) the relative magnitudes of local and global noise are incorrectly predicted when heterogeneities of the fuel and moderator are not considered; and (c) attenuation and reactivity effects are independent of the number of neutron energy groups assumed. Therefore local and global noise are associated with physical processes rather than being related to the two spatial eigenvalues of the two-group model developed by Behringer and Kosály (1979).

Difilippo (1980) resolved the dispersion law for neutron wave propagation in heterogeneous media into two characteristic relaxation lengths: one corresponding to wave propagation in the moderator [as was found by Quddus *et al.* (1969)], and the other corresponding to wave propagation in the combined fuel and moderator system resulting from the net production of neutrons in the fuel. A significant outcome of these calculations was that local and global noise occurred even when one neutron energy group was assumed in a heterogeneous lattice, however, the local noise disappeared when a homogeneous system was assumed. Again, local and global noise were associated with wave propagation through specific processes: attenuation in the moderator and fission in the fuel.

In summary, we believe that these works along with the recent paper by Analytis (1981) lead to an understanding not only of the importance of heterogeneous effects but also the processes by which noise sources propagate in a reactor.

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1. Analytis G. Th. (1981) *Ann. Nucl. Energy* **8**, 349.

2. Behringer K. and Kosály G. (1979) *Nucl. Sci. Engng* **72**, 304.

3. Difilippo F. C. (1980) *Trans. Am. Nucl. Soc.* **35**, 592.

4. Quddus M. A., Cochran R. G. and Emon E. E. (1969) *Nucl. Sci. Engng* **35**, 342.

5. Sweeney F. J. (1979) *Trans Am. Nucl. Soc.* **33**, 854.

6. Sweeney F. J. and Robinson J. C. (1980) *Trans Am. Nucl. Soc.* **34**, 802.

1.2.10. NEUTRON WAVE PROPAGATION IN HETEROGENEOUS MEDIA AND THE INTERPRETATION OF NEUTRON NOISE IN BOILING WATER REACTORS

F. C. Difilippo*

[Abstract of *Nucl. Sci. Eng.* **80**, 211–17 (1982)]

For a heterogeneous multiplying system, a one-group diffusion model is shown to exhibit two spatial eigenvalues (inverse relaxation lengths). The smallest root, which describes the long-range behavior of the neutron flux, is that eigenvalue that would be exhibited by an equivalent homogeneous system. The largest root corresponds to the inverse relaxation length of the moderator. The existence of these two eigenvalues is relevant to interpretation of neutron noise measurements in boiling water reactors.

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1.2.11. STABILITY MONITORING OF BOILING WATER REACTORS BY TIME SERIES ANALYSIS OF NEUTRON NOISE

B. R. Upadhyaya* **M. Kitamura†**

[Abstract of *Nucl. Sci. Eng.* **77**, 480–92 (1981)]

A method of monitoring stability of boiling water reactors (BWRs) has been developed. The stability parameters were derived from empirical discrete-time modeling of process noise signals and neutron noise signals. Data were taken from an operating BWR-4, and used to perform univariate analysis of average power range monitor (APRM), and local power range monitor signals, and multivariate analysis of APRM and the process signals, reactor pressure, and core flow rate. The parameters such as decay ratio, damping ratio, and characteristic frequency of oscillation, which represent the system stability, were estimated from the impulse response of the system. The impulse response was determined by using the time series models and contains information about the closed loop dynamics of a BWR.

The results indicate the feasibility of using APRM noise analysis for monitoring overall core stability and temporal variations in the stability margin of the reactor. Any significant variation in the stability parameters can be studied using multivariate noise signal algorithms, and cause and effect relationships can be obtained. Because the derived parameters depend on the random noise properties of the signals, this nonperturbing method is most useful for monitoring changes in stability. If an absolute measurement is necessary, a perturbation test must be performed.

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†Visiting research scientist from Tohoku University, Sendai, Japan.

1.2.12. NUMERICAL CALCULATION OF THE GLOBAL AND LOCAL COMPONENTS OF THE NEUTRON NOISE FIELD IN BOILING WATER REACTORS

F. C. Difilippo* **P. J. Otaduy†**

[Abstract of *Nucl. Sci. Eng.* 75, 258–64 (1980)]

A numerical model of the neutron noise field in boiling water reactors (BWRs), which can be readily implemented in existing deterministic computer codes, was formulated. The basis of the model is the assumption of separability of the noise field into local and global components. The application of this modeling was twofold: to determine the frequency range above which cross-correlation techniques can be used to measure steam velocities under normal operating conditions and to evaluate the validity of the point kinetics description of the global component of the neutron noise in BWRs. The model was implemented in the code LAPUR-3 and applied to the Hatch-1 BWR nuclear plant. Comparison with experimental results shows good agreement for frequencies above 6 Hz. At lower frequencies the global noise is overestimated, making apparent the limitation of the point kinetics formulation of the global noise component for this large reactor.

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1.2.13. IN-CORE FLOW VELOCITY PROFILES DURING THE FIRST FUEL CYCLE AT HATCH-1—INFERRED FROM NEUTRON NOISE

J. March-Leuba*

F. J. Sweeney **J. A. Renier†**

W. T. King **R. T. Wood***

(Abstract of EPRI Report NP-2083, Research Project 1754-2, October 1981)

In-core neutron noise data recorded during the first fuel cycle at Hatch-1 were used to infer in-core steam velocity and the presence of boiling in the flow bypass region between fuel boxes. A comparison of velocities thus inferred with calculated fuel-bundle-corner velocities at one local power range monitor (LPRM) position and reactor operating conditions shows that calculated fuel-bundle-corner steam velocities agree with the experimentally inferred velocities. It is likely that bypass boiling occurred at 79% power and 81% core flow at all instrument tube locations when bypass inlet orifices were plugged. No bypass boiling was observed at core elevations lower than the C-LPRM detectors.

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†UCC-ND Computer Sciences Division.

1.2.14. MEASUREMENT OF TRANSIENT TWO-PHASE FLOW VELOCITY USING STATISTICAL SIGNAL ANALYSIS OF IMPEDANCE PROBE SIGNALS

W. H. Leavell **J. A. Mullens**

[Abstract of *Meas. Control Sci. Indust.* 2, 517–30 (March 1981)]

An algorithm was developed to compute the transient, phase-interface velocity in two-phase, steam-water systems during simulated pressurized water reactor (PWR) reflood experiments. From signals produced by two spatially separated impedance probes immersed in a two-phase mixture, the algorithm computes the average transit time of the mixture's phase fluctuations as they move between the two probes. This transit time is computed by first measuring the phase shift between the two probe signals after transformation to the frequency domain and, second, computing the slope of the phase shift by a weighted, linear-least-squares fitting technique. The algorithm, which has been tested with both simulated and real data, accurately tracks velocity transients up to 4 $m \cdot s^{-2}$.

1.2.15. MACHINE RECOGNITION OF VOID FRACTION IN TWO-PHASE FLOWS

Nancy J. Hamilton   Rafael C. Gonzalez*

[Abstract of *IEEE Trans. on Systems, Man, and Cybernetics* SMC-11(11) (November 1981)]

Pattern recognition techniques were applied to the computation of void fraction values in two-phase steam/water mixtures. The void fraction is one of the principal parameters used to characterize the heat transfer properties of two-phase flows in the emergency cooling system of pressurized water reactors. The recognition approach reported here is based on moment descriptors extracted from the amplitude signal of an impedance probe. Using data obtained at the Advanced Instrumentation for Reflood Studies facility of the Oak Ridge National Laboratory, it was shown that void fraction values could be consistently recognized with less than 10 percent error using data segments as short as 0.25 s in duration.

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1.2.16. TRANSIENT TESTING OF AN IN-CORE IMPEDANCE FLOW SENSOR IN A 9-ROD HEATED BUNDLE

J. E. Hardy*   H. Liebert†
W. H. Leavell   J. A. Mullens

[Abstract of NUREG/CR-1909 (ORNL/NUREG/TM-389), February 1981]

An in-core instrumentation package that utilizes impedance sensors has been developed and fabricated to measure void fraction and two-phase flow velocity. The instrumentation package was designed to survive the severe thermal environment generated in the refill-reflood stage of a postulated loss-of-coolant accident and to detect small output signals from the sensors in a very noisy signal environment.

An in-core impedance sensor was tested in a facility capable of simulating in-core, refill-reflood tests. The output signals produced by the sensor were analyzed by a transient algorithm to yield in-core, two-phase void and velocity information.

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†Kraftwerk Union Erlangen (Federal Republic of Germany).

1.2.17. SUMMARY OF STUDIES ON METHODS FOR DETECTING, LOCATING, AND CHARACTERIZING METALLIC LOOSE PARTS IN NUCLEAR REACTOR COOLANT SYSTEMS

R. C. Kryter   F. Shahrokhi*

[Abstract of NUREG/CR-2344 (ORNL/TM-7967), October 1981)]

A research program, largely experimental in character, was conducted to explore the fundamental phenomena and the relationships governing the detection, characterization, and location of loose metallic parts in nuclear reactor coolant systems by means of the sounds produced by their impacts with the reactor vessel walls, internal structures, and piping. Methods for generating impacts of reproducible character and for sensing and analyzing the resultant transient acoustic signals were studied in simple flat-plate geometry in the first phase of this research. In the second phase, deleterious effects on signal detection produced by various sensor misinstallations were quantified, the variation of sensor output with projectile angle of incidence was studied, wave speeds in the medium were measured, and several techniques for ascertaining the location of impacts were developed and tested. In the last phase, tests were performed on a large metallic structure of complex geometry [the reactor vessel of the nonoperating experimental gas-cooled reactor (EGCR) at the Oak Ridge National Laboratory] to assess two impact location techniques under conditions (including simulated interfering background noise) approximating a commercial power reactor.

With regard to possible improvements in performance and extensions of capability for loose-part monitoring systems (LPMSs) installed in present-day commercial reactors, the abundance of detailed information generated by the fundamental studies performed can be condensed into four simple propositions:

1. In contrast to mere detection of impacts from metallic loose parts, impact location and loose-part characterization are vastly more difficult to achieve.
2. Best location/characterization results are achieved when acceleration signals are available from a multiplicity of sensors that are positioned reasonably near ($\lesssim 2$ m) the point of impact and are well coupled acoustically to the monitored structure.
3. Choosing sensor locations on a basis of minimal design impact and/or attaching the sensors to the reactor vessel and related structures by ill-considered or makeshift means are sure ways to compromise the performance potential of any LPMS.
4. A thorough system calibration, including an experimental mapping of sensor responses (generated with either internal or external impacts of various magnitudes and positions), is essential to the realization of superior LPMS performance.

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1.3. Design & Evaluation

1.3.1. RELIABILITY OF THE EMERGENCY AC POWER SYSTEM AT NUCLEAR POWER PLANTS

R. E. Battle

D. J. Campbell* P. W. Baranowsky†

(Abstract of paper presented at ANS Conference on Thermal Nuclear Reactor Safety, Chicago, Illinois, August 29, 1982)

Station blackout at nuclear power plants is an unresolved safety issue because of the past experience with ac power systems and the many safety systems that are dependent upon ac power. Detailed analyses of ac power system design features and operating experience are used to assess the expected frequency of station blackout at operating nuclear plants.

Final Safety Analysis Reports, plant visits, and utility contacts provide details of design features and operating procedures and policies. Licensee Event Reports and utility responses to questionnaires provide operational histories of ac power system equipment.

The histories of the emergency ac power systems are used to estimate the appropriate parameters for reliability studies and to assess design and operational factors. These parameters estimated from the operating experience analyses are used with system fault trees to determine station blackout frequencies.

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*JBF Associates, Inc., Knoxville.

†U.S. Nuclear Regulatory Commission, Washington, D.C.

1.3.2. RELIABILITY OF EMERGENCY AC POWER SYSTEMS AT NUCLEAR POWER PLANTS

R. E. Battle D. J. Campbell*

[Abstract of NUREG/CR-2989 (ORNL/TM-8545), to be published]

Reliability of the emergency onsite ac power system has been questioned within the NRC because of the number of diesel generator failures reported by nuclear plant licensees and because of the likelihood of reactor core damage if the diesels fail during an emergency. Because of these considerations, the NRC classified station blackout, the loss of all ac power at a nuclear plant, is an unresolved safety issue. The NRC requested Oak Ridge National Laboratory to provide a technical basis for resolution of this issue. This report contains the results of a reliability analysis of the onsite ac power system;

it uses the results of a separate analysis of the offsite power systems to calculate the expected frequency of station blackout.

Included is a detailed design and operating experience review. Eighteen plants representative of typical onsite ac power systems and ten generic designs were selected to be modeled by fault trees. Operating experience data were collected from the NRC files and from responses to a questionnaire sent out for this project. There are 1526 events categorized by failure type for 120 diesel generators, and there are data on demands, scheduled maintenance, repair, and modifications for 86 diesel generators.

The important contributors to the onsite power system reliability are the following:

(1) diesel generator failure probability, for which the industry-average is 2.5×10^{-2},

(2) human-error and hardware failure common-cause failure, for which the unavailabilities range from $2.4 \times 10^{-5} - 3.7 \times 10^{-4}$;

(3) scheduled maintenance unavailability for which the industry-average is 6.0×10^{-3};

(4) diesel repair time, for which the median value is 17 h;

(5) plant service-water system unavailability, for which the industry average is 2×10^{-3}.

For the 18 plants modeled, the median probability of failure to start of the onsite power system varies from 2.2×10^{-4} to 4.8×10^{-2}. Sensitivity of the onsite system unreliability to contributors 1–3 is studied. Improvements in the industry-average diesel failure probability will be very small because there is no single subsystem that dominates the failure probability. There are some plants that may be able to improve one or two subsystems for a significant improvement in reliability, but for the industry this cannot be done. However, improvements in the onsite power system reliability will result from improved operating and maintenance procedures and elimination of some design features which have a common-cause failure potential. Plants with two reactors and which require two-of-three diesels to cool both reactors after a loss of offsite power can, by adding a diesel, improve the onsite reliability by a factor between 5 and 10. The approximate cost to add a 3000 kW diesel is $7,000,000. The cost and reliability improvement for other less expensive modifications are included in the report.

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*JBF Associates, Inc., Knoxville, Tennessee.

1.3.3. RELATIVE COST VS RELIABILITY IMPROVEMENT OF NUCLEAR POWER PLANT ONSITE AC POWER

D. J. Campbell* R. E. Battle
J. S. Arendt* P. W. Baranowsky†

(Abstract of paper presented at 1982 Engineering Conference on Reliability for the Electric Power Industry, Hershey, Pennsylvania, June 16–18, 1982)

Emergency onsite ac power systems at nuclear power plants are a major concern in plant risk assessments because of the relatively large frequency of loss of offsite power and the dependence of most other safety systems on ac power. Detailed reviews of onsite ac power systems designs and reviews of experience with diesel generators at U.S. nuclear power plants form the basis of system reliability analyses that show significant improvements in reliability can be obtained at moderate cost for some plants. Onsite ac power system modifications analyzed include procedural modifications, minor equipment modifications and major equipment additions. Relative costs of various modifications are compared with associated system reliability improvements.

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†U.S. Nuclear Regulatory Commission, Washington, D.C.

1.3.4. STATION BLACKOUT AT BROWNS FERRY UNIT ONE—ACCIDENT SEQUENCE ANALYSIS

D. H. Cook*
R. M. Harrington S. A. Hodge†
S. R. Greene† D. D. Yue‡

[Abstract of NUREG/CR-2182 (ORNL/NUREG/TM-455/V1), November 1981]

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to Station Blackout, defined as a loss of off-site power

combined with failure of all onsite emergency diesel-generators to start and load. Every effort has been made to employ the most realistic assumptions during the process of defining the sequence of events for this hypothetical accident. Dc power is assumed to remain available from the unit batteries during the initial phase and the operator actions and corresponding events during this period are described using results provided by an analysis code developed specifically for this purpose. The Station Blackout is assumed to persist beyond the point of battery exhaustion and the events during this second phase of the accident in which dc power would be unavailable were determined through use of the MARCH code. Without dc power, cooling water could no longer be injected into the reactor vessel and the events of the second phase include core meltdown and subsequent containment failure. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the environment is provided in Volume 2 of this report.

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†Engineering Technology Division.

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1.3.5. REVIEW OF THE ESFAS LOAD SEQUENCER AT THE PALO VERDE NUCLEAR GENERATING STATION

P. R. Frey

A design review was performed on the microprocessor-based Load Sequencer Module in the balance-of-plant Engineered Safety Features Actuation System (ESFAS) at the Arizona Public Service Company's Palo Verde Nuclear Generating Station. The hardware and software design documents were reviewed, and the testing procedures and results were evaluated. No design inadequacies were identified; however, a few operational concerns were noted. Among these are a need for (1) a descriptive diagnostic manual to aid in the interpretation of the failure indicators and (2) a software design change procedure to provide for verification and validation of software changes. The results were presented in a letter report to the NRC.

1.3.6. THE ALLOCATION OF FUNCTIONS IN MAN-MACHINE SYSTEMS: A PERSPECTIVE AND LITERATURE REVIEW

Harold E. Price*

Richard E. Maisano* Harold P. Van Cott*

(Abstract of report prepared for ORNL by BioTechnology, Inc., November 1981, under Subcontract 9207)

This report reviews the literature relevant to allocation of functions and presents a procedure for the allocation process applicable to nuclear power plant control rooms. An historical perspective of man's relationship with technology is given as background. Methods and models that have been developed to aid the allocation process are then considered, followed by examples of real-world applications.

The relationship of allocation of function to the system development process is outlined. The report then turns to the proposed procedure of the allocation process. This procedure, conducted in a series of steps, leads to decisions on what roles and functions man and machine will play in the complex man-machine system under consideration. These decisions are based on criteria developed from the literature on human and machine capabilities and limitations. The resultant hypothesized allocations are tested against environmental, system, and psychological constraints. Consideration is also given to human operator acceptance of automation, a crucial question with the increased use of computers and computer-based aids. An example of how this procedure might be applied to one aspect of the nuclear area is detailed. Lastly, suggestions for further work outline what still has to be done to implement the procedure in the evaluation and installation of automated and computer-based aids in nuclear power plants.

*BioTechnology, Inc., Falls Church, Virginia.

1.3.7. FUNCTIONS AND OPERATIONS OF NUCLEAR POWER PLANT CREWS

R. A. Kisner P. R. Frey

[Abstract of NUREG/CR-2587 (ORNL/TM-8257), April 1982]

This report summarizes the results of work performed at Oak Ridge National Laboratory and its subcontractors to define the functions, operations, and organization of nuclear power plant operating crews.

The primary information sources used were ANS and IEEE standards, normal and emergency operating procedures from nuclear power plants, interviews, and literature reviews. The function and organization of operating crews for several plants are discussed generically.

The report covers a wide spectrum of topics including review of standards affecting human factors in the control room, influence of automation on operator functions, classification of operator functions, function of operator at onset of emergency, crew organization, work-induced stress, and operator acceptance of his role.

1.3.8. DEFINING THE ROLE OF THE OPERATING CREW

R. A. Kisner G. F. Flanagan*

(Invited paper presented at the Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 27-31, 1980)

This paper briefly describes a project under way that uses the elements of a systems approach to describe the role of nuclear power plant operating crews under emergency conditions. As much of the work remains to be performed, the application of the results is emphasized, including development of design requirements and review criteria.

*Engineering Physics Division.

1.3.9. A SYSTEMS APPROACH TO DEFINING OPERATOR ROLES

R. A. Kisner G. F. Flanagan*

[Summary of *IEEE Trans. Nucl. Sci.* NS-28(1) (February 1981)]

This paper briefly describes the elements of a systems approach for defining the role of the operating crew of a nuclear power plant. A systems approach aims at solving the multifaceted and interrelated problems produced by complex man-machine systems. Through systems analysis and engineering, an organized and systematic approach to the proper allocation of roles between man and machine can be made. This approach is utilized here to develop a methodology for specifying the optimum role of the crew in a nuclear plant system and also defining the internal structure of the crew for the purpose of ensuring the safe and economic operation of the plant.

*Engineering Physics Division.

1.3.10. ANALYSIS OF THE OPERATORS' ROLE AS DEFINED BY EMERGENCY PROCEDURES DEVELOPED FOR A PWR AND A BWR

C. B. Oh*
E. M. Dougherty* J. L. Hamrick*

(Summary of TEC Report R-81-018, July 19, 1981, prepared for ORNL by Technology for Energy Corp.)

This report is the third of a series of reports examining the operators' role during emergencies. It is part of a program administered by the Oak Ridge National Laboratory (ORNL) which seeks to define the functional design requirements of operational aids for nuclear power plant operators. This report looks at one limited but very important aspect of the duties of the operators—their role in the control room during the onset of emergencies. The accident at TMI showed how human error can contribute significantly to the risks associated with the operators' immediate response to an accident. In addition to the TMI experience, risk studies

have continually shown that human error contributes significantly to the unavailability and unreliability of vital systems relating to plant and public safety.

This report presents a description of the operators' response as defined by emergency procedures for a pressurized water reactor (PWR) and a boiling water reactor (BWR). The operators' response is described in terms of the emergency procedures provided by the respective Nuclear Steam System Suppliers (NSSS). Emergency procedures were used because they portray the ideal operator responses and define the parameters, plant systems, and components of importance with respect to the subject plants.

This report does not attempt to analyze the manner in which the procedures are written. Rather, the logical framework of the operator's role was inferred from the procedures. The inferred framework represents a model for characterizing the operators' role with respect to emergency procedures. The model also serves as a framework within which to consider operator aids. Further analysis and perhaps simulation experiments are needed to test the models against the operators' *in situ* response.

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1.3.11. A CHARACTERIZATION OF THE NUCLEAR POWER PLANT OPERATOR'S ROLE

C. B. Oh*
M. E. Watson* P. F. Knight*
S. V. Asselin* F. E. LeVert*
A. R. Buhl* J. C. Robinson*

(Abstract of TEC Report R-80-022, August 1, 1980, prepared for ORNL by the Technology for Energy Corp.)

Many techniques exist to assist the operator in maintaining a safe nuclear power plant. Some have been required by the NRC and others are likely to be required. Unfortunately, a systematic and objective methodology to measure the degree of safety improvement resulting from the addition of an operational aid is lacking. Currently, the tacit assumption behind the requirement of an operational aid is that somehow the operator will be given timely, tractable, and useful information concerning abnormal conditions, providing feedback on the effects of his actions and the status of various reactor and safety systems. This assumption, while desirable, requires development via a methodology to measure quantitatively, if possible, the real improvements to safety which will result from the inception of an operational aid. The basis for establishing such a methodology rests on a systematic understanding of the operator's current role in maintaining a safe plant. Therefore, this evaluation provides a characterization of the operator's role from one perspective: that of the Emergency Operating Instructions (EOI) of a typical Pressurized Water Reactor (PWR). In short, the operator's current role is determined to be the execution of a systematic response relating to each of a predefined set of emergency events which should enable him to cope with any design-[basis] accident. The operator's primary objectives are to insure the proper performance of automatic safety features and to manually control specified parameters if necessary to achieve implicit safety objectives relating to the emergency event(s).

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1.3.12. A TAXONOMY OF THE NUCLEAR PLANT OPERATOR'S ROLE

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A. M. Fullerton* E. M. Dougherty†

(Abstract of paper presented at the Enlarged Halden Reactor Programme Meeting on Process Computer Applications, Fredrickstad, Norway, June 14-19, 1981)

A program is presently under way at the Oak Ridge National Laboratory (ORNL) to define the functional design requirements of operational aids for nuclear power plant operators. A first and important step in defining these requirements is to develop an understanding of the operator's role or function. This paper describes a taxonomy of operator functions that applies during all operational modes and conditions of the plant. Other topics such as the influence of automation, role acceptance, and the operator's role during emergencies are also discussed. This systematic approach has revealed several areas which have potential for improving the operator's ability to perform his role.

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1.3.13. ANALYSIS OF THE OPERATOR'S ROLE DURING THE ONSET OF AN EMERGENCY

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(Summary of TEC Report R-81-004, February 27, 1981, prepared for ORNL by Technology for Energy Corp.)

The research project, "Operational Aids for Reactor Operations," is a multidisciplinary activity administered by the Engineering Physics Division of the Oak Ridge National Laboratory and sponsored by the Nuclear Regulatory Commission (NRC). It focuses on seven areas: (1) operating crew requirements, (2) operator aid functional and safety requirements, (3) methods to evaluate the impact of aids on safety, (4) procedures for assuring consistency in applying the safety impact methodology, (5) assessments of the development of operator aids, (6) the impact on safety of maintaining and testing aids, and (7) modeling of the man-machine interface.

This report primarily addresses area 1—operating crew requirements—by developing a qualitative model of the operator's role during the onset of an emergency. This model can be used as a management, planning, and research tool for nuclear power plant safety programs for improving the understanding of the dynamics of the operator's role.

It is the intent of this work to identify and relate the primary functions of the operator during the initial stages of an emergency. The results of this work should be of interest to regulators, Nuclear Steam System Suppliers (NSSS), designers of instrumentation and controls, operator trainers, and consultants who wish to improve plant safety.

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1.3.14. A CHARACTERIZATION OF THE NUCLEAR POWER PLANT OPERATOR'S ROLE DURING EMERGENCIES

S. V. Asselin* C. B. Oh*

[Abstract of NUREG/CR-1772 (ORNL/Sub-80/13852/1), August 1980, prepared for ORNL by the Technology for Energy Corp.]

Many techniques exist to assist the operator in maintaining a safe nuclear power plant. Some of these techniques [are] required by the Nuclear Regulatory Commission (NRC) and others are likely to be required in the future. Unfortunately, a systematic and objective methodology to measure the degree of safety improvement resulting from the addition of an operational aid is lacking. Currently, the tacit assumption behind the requirement of an operational aid is that somehow it will give the operator timely, tractable, and useful information concerning abnormal conditions, providing feedback on the effects of his actions and on the status of various reactor and safety systems. This assumption, while desirable, requires a methodology to measure quantitatively, if possible, the real improvements to safety that will result from the inception of an operational aid. The basis for establishing such a methodology must rest on a systematic understanding of the operator's current role in maintaining a safe plant.

This evaluation provides a characterization of the operator's role from one perspective: that of the Emergency Operating Instructions (EOI) of a typical Pressurized Water Reactor (PWR).

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1.3.15. TRANSITIONS IN THE ROLE OF THE OPERATOR

E. M. Dougherty*

(Summary of TEC Report R-81-015, June 30, 1981, prepared for ORNL by Technology for Energy Corp.)

The human/machine interface in a nuclear power plant is currently under wide scrutiny. A fundamental factor in this research is the role that the operator assumes during an emergency. TEC Report No. R-81-004 described the operator's role at the onset of an emergency and developed three qualitative models of this role. These models were developed by examining the Emergency Operating Instructions (EOIs) of a typical pressurized water reactor (PWR). The study of the operator's role that led to R-81-004 has been extended to include (1) investigation of the procedural approaches used by the other reactor manufacturers, (2) examination of the transition in operator's role from normal to abnormal conditions and from onset of emergency to long-term emergency management, and (3) review of the cognitive demands that the EOIs place on the operator and their impact on the role of the operator. The second item is the topic of this report.

Among other things, the study of the role of the nuclear power plant (NPP) operator, documented in R-81-004, sought to identify a general framework within which to understand this role. This work was conducted within the confines of certain "interfaces," which delineated the various assumptions made for the analysis. These interfaces included, but were not restricted to:

1. The treatment of the operator crew as a unit
2. The focus on the onset of an emergency only
3. The EOIs as a basis for analyzing the role
4. The operator's behavior above the task level

Interface 1 was based partly on the fact that distinctions in human performance between individuals and teams are not well understood. Also, nuclear reactor operator training allows members of the operator crew to be interchangeable to some degree. Interfaces 2 and 3 were consistent with and motivated by the current focus in the nuclear industry on the lessons learned from the Three Mile Island-2 (TMI) accident. Interface 4 was based on the fact that task analysis is a hotly contested topic, and even if not controversial, requires too much depth for the scope of the analysis, which was to have been as generic as possible.

1.3.16. COGNITIVE DEMANDS ON THE REACTOR OPERATOR (AS INFERRED FROM EMERGENCY OPERATING INSTRUCTIONS)

E. M. Dougherty*

(Summary of TEC Report R-81-014, June 30, 1981, prepared for ORNL by Technology for Energy Corp.)

The human/machine interface in a nuclear power plant is currently under wide scrutiny. A fundamental factor in the associated research is the role that the operator assumes during an emergency. TEC Report No. R-81-004 described the operator's role at the onset of an emergency and developed three qualitative models of this role. These models were developed by examining the Emergency Operating Instructions (EOIs) of a typical pressurized water reactor (PWR). The study of the operator's role has been extended to include (1) investigation of the procedural approaches used by the other reactor manufacturers, (2) examination of the transition in operator's role from normal to abnormal conditions and from onset of emergency to long-term emergency management, and (3) review of the cognitive demands that the EOIs place on the operator and their impact on the role of the operator. The third item is the topic of this report; the other two items are reported separately.

In addressing the operator—the human himself—one realizes three general perspectives of behavior:

1. What the operator can do,
2. What the operator is supposed to do, and
3. What the operator will do.

The concern historically has been with perspective (3). Probabilistic risk assessment, for example, not only requires behavior to be predicted, but quantified as well. This requirement seems intuitively unfeasible (e.g. how can a determination to find strategies that are effective be quantified?). A large body of research into the prediction of human behavior apparently has not diminished this intuition very much.

The predecessor study, reported in R-81-004, analyzed behavior from perspective (2)—what the

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operator is supposed to do. Basically the operator (post-TMI) was seen as goal-directed with a procedure or rule base. The operator was seen as required to return the plant to an interim, safe steady-state. By doing this, he was expected to buy time to diagnose the specific cause of the emergency and then safely shut down the plant. In support of this goal, the EOIs (the rules) were modified post-TMI to allow an operator to move "blindly" but safely, always returning to an earlier stage in the procedure if new knowledge of the plant state warranted it.

The operator's role, according to the EOIs, is what the operator is supposed to do. There remains the question: Can he? A mismatch between this last question, perspective (1), and perspective (2) will clearly impact behavior perspective (3). That is, if the operator cannot do what he is supposed to do, he will not. (This differs from the motivational problem in which the operator does not do what he is supposed to do when he can.) This report investigates what the operator can do, from an examination of the demands the EOIs place upon him.

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1.3.17. UNDERSTANDING HUMAN BEHAVIOR IN OFF-AVERAGE CONDITIONS. FINAL REPORT—VOLUMES I, II, III, AND IV

T. O. Sargent*   R. B. Blum*

(Abstract of ORNL/Sub-7960/1, November 1980, prepared for ORNL by Lund Consulting, Inc.)

To understand the effects of human performance in nuclear power plant operations, specifically, the performance of the operator in the control room, it is necessary to have a clear definition of how the mind works. The Bimodal Theory of Human Behavior in Off-Average Conditions states that behavior will be one of two modes, rigid or flexible, determined by operator response to external conditions. The important contribution of the Bimodal Theory is its statement that in all circumstances, behavior is predictable and can be appropriately directed.

This report will attempt to define and critique the Bimodal Theory, exploring similarities and differences between this and other theories of cognitive behavior. We will use the Bimodal Theory to analyze current nuclear power plant operations and suggest methods of achieving performance and safety improvement.

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1.3.18. A SURVEY OF METHODS FOR IMPROVING OPERATOR ACCEPTANCE OF COMPUTERIZED AIDS

P. R. Frey   R. A. Kisner

[Abstract of NUREG/CR-2586 (ORNL/TM-8236), April 1982]

Methods for assessing and enhancing operator acceptance of computerized aids in various industries were collected from the available literature. These methods were reviewed and are discussed for their application to the computerized aids which may be installed in the control rooms of nuclear power plants. Based on prior work in the data processing field, a method for developing a measure of the user acceptance of these aids was outlined. Techniques for enhancing operator acceptance are also discussed. These techniques may be used during equipment design, operator training, and system operation.

1.3.19. A SYSTEMS APPROACH TO EVALUATION OF CONTROL ROOM STRUCTURE

Martin Becker*   Rodney R. Gay*
Robert C. Block*   Donald R. Harris*
Michael M. Danchak*   John P. Tully*

(Summary of BBH-81-1, May 1981, prepared for ORNL by Becker, Block and Harris, Inc.)

The Three Mile Island accident has led to considerable review of the manner in which nuclear reactors are operated. Much attention has been given to matters affecting the ability of the operating crew to respond to off-normal situations. For the most part, this attention has been to ways in which a conventionally structured crew could be helped to respond effectively. This study has addressed how the structure of the crew itself might be modified so as to improve operational capability.

This study has proceeded by the following: first, a set of functions encompassing the role of the operator was established; next, a set of shortcomings in the performance of these functions in past practice was developed; guidelines for crew structure were then formulated so as to avoid these shortcomings in the future; characteristics of crew structure required to meet the guidelines were then considered, and a particular type, hierarchical systems structure, was noted to be particularly well suited to meeting the guidelines and alleviating the shortcomings of past practice. Additional guidelines were then developed to aid the implementation of the crew structure, i.e. to guide the resolution of the plant into major systems and subsystems. Alternate resolutions were then considered, with one (referred to as the NEP) found to be particularly well suited. Implications of the choices made in actual practice were then considered, including testing for an operational transient, consideration of crew size, training requirements, interfaces with other plant personnel, etc.

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1.3.20. REVIEW OF STANDARDS AND REQUIREMENTS AFFECTING HUMAN FACTORS IN NUCLEAR POWER PLANT CONTROL ROOMS

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R. A. Hedrick* R. W. Starostecki*

(Summary of ORNL #62B-13819C/62X-11, November 1980, prepared for ORNL by Science Applications, Inc.)

The accident at the Three Mile Island-Unit 2 (TMI-2) nuclear power plant highlighted deficiencies in power plant control room design as it relates to accommodating the operators. Inadequacies and design problems affecting human factors had been recognized prior to the event, but TMI-2 provided a dramatic demonstration of lack of accommodation of human information needs under abnormal circumstances.

With the TMI-2 impetus, two general classes of problems with control room design have been identified. The first relates to the design providing its information display and control equipment in an understandable and unambiguous manner. For instance, information displays providing redundant measures of flow rates should be in physical proximity. Likewise, controls which affect the flow should be located close to the flow rate displays. The second category of problem relates to whether the information provided the operator is adequate to support his actions in controlling the plant under abnormal circumstances.

As a result of identification of these problems, activities are underway in the nuclear industry to: (1) evaluate and enhance the design of the control room to support the operator and (2) develop operator aids such as the Disturbance Analysis System to enhance the information provided to the operator. The effort reported here supports both of these efforts by providing a clear and documented understanding of the current requirements and design practices. This serves several purposes:

1. Documented requirement, criteria, and design practices provide basic information needed by designers of new operator aids. This is particularly important as new companies and new designers enter the commercial nuclear power industry. Knowledge of such requirements will make their contributions more timely and effective.
2. There exist known strengths and known deficiencies in the design of nuclear power plant control rooms as they pertain to operator performance and reliability. These must be evaluated against documented requirements and practices in order to develop guidance for cost effective future directions. This evaluation can provide insight into how to correct deficiencies and, as important, how to develop better requirements and design practices.
3. A clear and documented survey of requirements and practices provides the base-line necessary for measuring the impact of advanced operator aids on operator performance and reliability.

The present study approaches these purposes by tabulating codes, standards, and design practices as they relate to operator performance and reliability.

In particular, standards developed under the auspices of the Institute of Electrical and Electronics Engineers (IEEE) and the American Nuclear Society (ANS) are examined. U.S. Nuclear Regulatory Commission Regulatory Guides are reviewed and design practices of the nuclear industry are surveyed.

Current requirements and practices of these services are examined and the planned future directions are identified. The results of this study provide the necessary data base for the new designer entering the nuclear control room design market. It also provides a first supportive step in evaluating the adequacy of these requirements impacting the human operators' interface with the plant.

Chapter 2 of this report describes the state of industry practices in accommodating human factors in control room design. It addresses regulatory attitudes and practices as well as the general practices of industry designers. Chapter 3 presents a review of current standards and regulations. This chapter also describes the significant efforts underway in this area. Finally Chapter 4 presents conclusions and recommendations for means by which the data presented in this [document] can be best utilized.

*Science Applications, Inc., Oak Ridge, Tennessee.

1.4. Detectors

1.4.1. ULTRAHIGH-SENSITIVITY FISSION COUNTER FOR SOURCE-RANGE NEUTRON FLUX MONITORING IN THE CLINCH RIVER BREEDER REACTOR

W. L. Kelly*
K. H. Valentine **M. K. Kopp**

(Abstract of *CRBRP Technical Review*, Fall 1982, pp. 7–24)

Continuing research on advanced methods of thermal neutron detection has resulted in improved neutron counters for ex-vessel flux monitoring in the Clinch River Breeder Reactor Plant. An ultrahigh-sensitivity (50 cps/nv) fission counter system has been developed using a new method of transmission line electrode configuration, a new Ar-CF_4 counter gas, and two-dimensional noise and pileup discrimination electronics. The size and sensitivity of this new counter are comparable to those of BF_3 counter systems, but the fission counter is operable in environments that would destroy BF_3 counters (i.e., up to 500 K temperature and 10^4 R/h gamma radiation field). A prototypic fission counter system of 50 cps/nv sensitivity was built and successfully operated for a test period of one month at 423 K in a gamma radiation field of 4.5×10^4 R/h with no measurable degradation of performance.

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1.4.2. ULTRAHIGH-SENSITIVITY FISSION COUNTER WITH TRANSMISSION LINE ELECTRODE CONFIGURATION

K. H. Valentine
M. K. Kopp **W. T. Clay**
J. A. Harter **G. W. Allin**
G. C. Guerrant **C. E. Fowler**

(Summary of paper presented at 1982 Nuclear Science Symposium, Washington D.C., October 20–22, 1982)

An ultrahigh-sensitivity fission counter (UHSFC) prototype was designed, fabricated, and tested. The objective of this research was development of a fission counter system for ex-vessel, source-range flux monitoring having a neutron sensitivity of >40 counts$\cdot s^{-1}$ [neutrons/($cm^2\cdot s$)]$^{-1}$ (>40 cps/nv) and a size comparable to that of a BF_3 proportional counter system. Also, the

UHSFC system was required to be operable under conditions beyond the capabilities of BF_3 counters: up to a temperature of 450 K and in a gamma radiation field of 7.2×10^{-4} $C(kg\cdot s)^{-1}$ (i.e., 10^4 R/h).

The sensitivity requirement (40 cps/nv) was met by assembling two transmission line fission counters (TLFCs) within a common envelope.[1] The size requirement (<14 cm diameter, <80 cm length) was met by using a high electrode-packing density with parallel, curved electrodes arranged to maximize the ratio of electrode area to sensitive volume.

The measured neutron sensitivity for each TLFC is 23.5 cps/nv at 4.5×10^4 R/h and 450 K. The UHSFC is filled with a recently developed gas[2,3] (80% Ar, 20% CF_4 at 263 kPa pressure) of high electron-drift velocity. The electrode area (5 m^2) is coated with 88 g of enriched uranium (93.15% ^{235}U, 0.99% ^{234}U, balance ^{238}U). The impedance of the lumped element LC transmission lines is 25 Ω, the bandwidth 100 MHz, and the delay 4 ns per node (each TLFC has 50 nodes). Pulse-height and time discrimination is used to mitigate the effects of noise from the alpha pileup current of the uranium coating.

Test results from the UHSFC prototype confirm that TLFCs can be operated with neutron sensitivities >25 cps/nv and that two TLFCs will fit into existing source-range detector positions to achieve a neutron sensitivity of >50 cps/nv. The TLFC instrumentation (beyond the output of the coincidence gate) is identical to the BF_3 counter instrumentation. Thus the neutronic and physical properties of the UHSFC design are compatible with the present design of flux monitoring systems based on BF_3 counters. The operability of each independent TLFC can be verified *in situ* prior to the initial core-loading operation by monitoring the alpha response signature of the inherent ^{234}U activity. Furthermore, the increased gamma immunity of the TLFCs reduces shielding requirements and eliminates the need for a recovery period following exposure to irradiated fuel assemblies, thus increasing the measurable neutron flux and eliminating delays in the fuel loading procedure.

1. K. H. Valentine, M. K. Kopp, and G. C. Guerrant, *Trans. Am. Nucl. Soc.* **39**, 631 (1982).

2. L. G. Christophorou, D. L. McCorkle, D. V. Maxey, and J. G. Carter, *Nucl. Instrum. Methods* **163**, 141 (1979).

3. M. K. Kopp, K. H. Valentine, L. G. Christophorou, and J. G. Carter, "New Gas Mixture Improves Performance of ^3He Neutron Counters," *Nucl. Instrum. Methods*, **201**, 395–401 (1982).

1.4.3. A TEN-FOLD INCREASE IN FISSION COUNTER SENSITIVITY WITH TRANSMISSION LINE ELECTRODE CONFIGURATION

K. H. Valentine
M. K. Kopp G. C. Guerrant

[Summary of *Trans. Am. Nucl. Soc.* **39**, 631 (1981)]

A new method was developed to decrease the effective interelectrode capacitance (hence, to increase the sensitivity) of large fission counters by configuring the uranium-coated electrode area as a lumped element, L-C transmission line. Discrimination against spurious noise pulses (from inherent ^{234}U alpha-activity) was improved by applying the pulses from each end of the transmission line to a time coincidence gate (Fig. 1.4.1). The low capacitance mitigates pileup effects by increasing the bandwidth of the counter, and time-interval discrimination allows the thresholds of the conventional amplitude discriminators to be decreased, thus increasing the countable fraction of fission events. We estimate that sensitivities >20 $counts\cdot s^{-1}$ $(n/cm^2\cdot s)^{-1}$ will be obtainable from a single transmission line fission counter (TLFC). A TLFC, therefore, combines the neutron sensitivity

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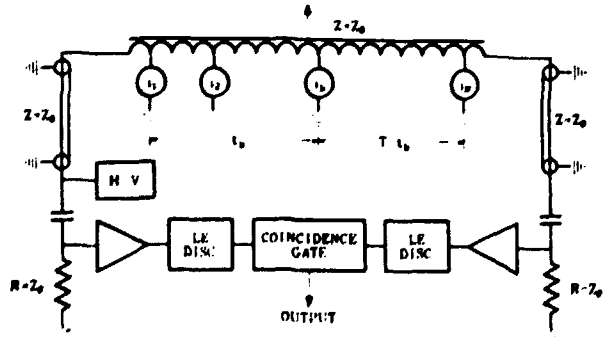

Fig. 1.4.1. The coincidence gate is triggered by correlated pulse pairs arising from fission events, but rejects uncorrelated pileup pulses.

of the BF_3 proportional tube with the ability of the fission counter to operate in higher radiation fields at higher temperatures and at significantly lower voltages. Also, the TLFC is position sensitive, since the position of each fission event is encoded as the differential arrival time of pulses from each end of the line.

A proof-of-principle experiment was performed to verify performance predictions for the TLFC. Sixty-five circular electrodes (10.16 cm in diameter) were assembled into a parallel-plate fission counter. The 32 signal electrodes were interconnected with single-layer, air-core inductors to form a lumped element transmission line having a characteristic impedance of 50 Ω and a total delay of 160 ns. Four ground electrodes were coated with highly enriched UO_2, and 4.0 mCi of ^{241}Am was distributed on the remaining ground electrodes to simulate the inherent alpha-activity of ~130 g of highly enriched, gaseous-diffused uranium. Since the performance of this device was not consistent with Gaussian pileup theory[1] (the countable fission fraction was too low at 0.16), the data were used to develop an accurate mathematical model of non-Gaussian pileup. This model was incorporated into a design optimization algorithm similar to one for conventional fission counters,[2] and then it was used to predict the performance of optimized, TLFC-based neutron counting channels. The results indicate that counters containing ≤50 g of 93.4% enriched UO_2 will have countable fission fractions of >0.8, giving sensitivities of 20 to 30 counts·s^{-1} (n/cm^2·s)$^{-1}$, depending on the amount of neutron absorption in the counter internals.

Transmission line electrode configuration has been used previously to reduce effective interelectrode capacitance.[3] However, two recent developments, low-noise, low-input impedance, wide-band preamplifiers and Ar-CF_4 gas mixtures that have extremely high electron drift velocities[4] (>12 cm/μs), enabled the design of low-impedance TLFCs with bandwidths >100 MHz and electron collection times of 10 to 15 ns. The resulting fast time response with time interval discrimination provides the high degree of pileup rejection required for counters containing large quantities of available and inexpensive gaseous-diffused uranium (93.4% ^{235}U, 1.0% ^{234}U, balance ^{238}U). Pileup effects could be further reduced by decreasing the inherent alpha source through the use of electromagnetically separated uranium (e.g., 99.8% ^{235}U, 0.04% ^{234}U, balance ^{238}U), but the 100X cost factor would make the TLFC economically impractical.

The TLFC could be used as an ex-core, source range flux monitor or as a subcriticality monitor in a fuel reprocessing plant. In either use, the cost of plant construction would be substantially less because cooled instrument thimbles or massive radiation shielding for the detector would not be necessary.

A TLFC is being fabricated for the Clinch River Breeder Reactor Project as a backup to the high-sensitivity BF_3 proportional counter that is presently specified in the reference design of the ex-vessel, source range flux monitor. To achieve a sensitivity of 40 counts·s^{-1} (n/cm^2·s)$^{-1}$ within the available volume (12.7 cm in diameter × 76 cm long), the electrode plates are arranged about a central hub (Fig. 1.4.2). The curvature of the

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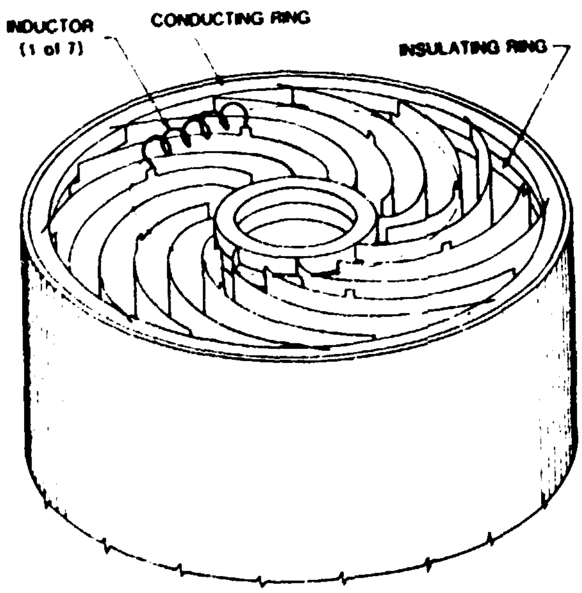

Fig. 1.4.2. New electrode configuration for ion chambers maximizes ratio of electrode area to sensitive volume. For TLFC application, several sections can be stacked together with signal cables routed through the hub.

plates provides a constant interelectrode gap. The sensitive volume is configured as two independent 25-Ω transmission lines, each contributing ≥20 counts·s^{-1} (n/cm^2·s)$^{-1}$ to the total sensitivity.

1. S. O. Rice, "Mathematical Analysis of Random Noise," from selected papers on *Noise and Stochastic Processes*, p. 133, Dover Publications (1954).

2. K. H. Valentine and V. K. Paré, *Trans. Am. Nucl. Soc.*, 33, 696 (1979).

3. W. N. Hess et al., *Nucleonics*, 15, 74 (1957).

4. L. G. Christophorou, *Nucl. Instrum. Methods*, 163, 141 (1979).

1.4.4. POSITION-SENSITIVE NEUTRON FLUX MONITOR

K. H. Valentine

A position-sensitive flux monitor was developed to measure in-core axial flux profiles in light water reactors. This 4-m-long fission counter resolves 11 pixels at typical power range neutron flux values (10^{13} neutrons·cm^{-2}·s^{-1}). Compared to the four-detector assemblies currently used, this monitor provides better spatial resolution and requires only a single signal cable. *(M. K. Kopp, G. C. Guerrant, J. A. Harter)*

1.4.5. CURVED ONE-DIMENSIONAL POSITION-SENSITIVE PROPORTIONAL COUNTER (PSPC) FOR DIFFRACTION MEASUREMENTS USING THERMAL NEUTRONS

G. C. Guerrant

A curved PSPC was developed for large-angle (<130°) neutron diffraction measurements to provide a spatial resolution of 0.2°. A detection efficiency of 50% for 0.13-nm neutrons and count rate linearity up to 10^5 Hz were required. These specifications were met using a multiwire anode with a 75-cm radius of curvature, and LC position-encoding cathode, and a 60% ^{3}He–40% CF_4 counter gas at 600 kPa. *(J. A. Harter, K. H. Valentine, M. K. Kopp)*

1.4.6. NEW GAS MIXTURE IMPROVES PERFORMANCE OF ^{3}He NEUTRON COUNTERS

M. K. Kopp **L. G. Christophorou***
K. H. Valentine **J. G. Carter***

[Abstract of *Nucl. Instrum. Methods* 201, 395–401 (1982)]

Count-rate capability, spatial resolution, and photon discrimination of position-sensitive, proportional neutron counters were improved by using a detector gas mixture of 65% ^{3}He and 35% CF_4. These improvements, relative to previously used ^{3}He-Xe-CO_2 mixtures, were due to the larger electron drift velocity, greater stopping power for the protons and tritons of the ^{3}He(n,p) reaction, and smaller photon cross section of the He-CF_4 mixtures.

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1.4.7. AMPLIFIER FOR ULTRAHIGH-SENSITIVITY FISSION COUNTERS

M. K. Kopp

A preamplifier and a filter/amplifier were developed for near-optimum processing of signals from ultrahigh-sensitivity fission counters having transmission line electrode configuration.

The preamplifier input impedance of 25 Ω, required for the LC-line termination, was realized by a new method of active impedance and dual-feedback loop construction. The thermal noise of the preamplifier at 300 K is only 19.5 $pA/\sqrt{Hz}$, the bandwidth is 150 MHz, and the transresistance is 148 $k\Omega$.

The filter/amplifier is composed of four amplifier stages, interconnected with a passive RCL filter network. The amplifier gain is 4.5 per stage and the bandwidth is 300 MHz (e.g., ORTEC timing amplifier Mod. 574). The output signal in response to an input impulse is a unipolar, 10-ns-wide (fwhm) pulse. The output linear range is ±1 V into a 50-Ω load resistance. *(K. H. Valentine)*

1.4.8. APPLICATIONS OF RESONANCE IONIZATION SPECTROSCOPY TO ULTRALOW-LEVEL COUNTING AND MASS SPECTROSCOPY

S. D. Kramer* M. K. Kopp
G. S. Hurst* T. A. Callcott†
J. P. Young* E. T. Arakawa*
M. G. Payne* D. W. Beekman†

[Abstract of *Radiocarbon* 22(2), 428 (1980)]

In this paper it is shown that the ability to directly detect a daughter atom, using resonance ionization spectroscopy, in delayed time coincidence with the decay of a parent species promises to drastically reduce the background in low-level counting experiments. In addition, resonance ionization can also be used as an ion source for a mass spectrometer system that is capable of discriminating between isobars.

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1.4.9. PROPORTIONAL COUNTER CAMERA FOR THIN-LAYER CHROMATOGRAPHY

J. A. Harter

A prototypic camera for the imaging of spatial distributions of ^{14}C radiolabeled compounds on chromatographic plates and electrophoretic gels was developed and tested. This camera is based on an RC-encoded PSPC (position-sensitive proportional counter) of 10-cm × 10-cm sensitive area. The PSPC is filled with a 95% Xe–5% CO_2 gas at 300 kPa. The thermal noise contribution of the PSPC to spatial uncertainty is <0.5 mm (fwhm). The measured overall spatial uncertainty for ^{14}C particle imaging is <1 mm (fwhm). *(M. K. Kopp, M. A. Meacham, D. D. Schuresco,* W. D. Bostick†)*

*Health and Safety Research Division.
†Chemical Technology Division.

1.5. Facilities Support

1.5.1. CONTROL ROD SERVOMECHANISM AND CONTROLLER FOR THE BULK SHIELDING REACTOR (BSR)

R. E. Battle

A servo system using a programmable controller (PC) was designed, fabricated, tested, and installed on the BSR to control the neutron flux at the demand level set by the reactor operator. The control functions in this system are implemented by a digital rather than an analog process as was done on the old system. In addition to the flux control and the generation of a flux demand level, the PC calculates reactor heat power, which is used to calibrate the power demand readout to the neutron flux. The new system is more reliable, more accurate, and easier to maintain than the analog system.

1.5.2. ENGINEERING REVIEW OF THE CRBRP

K. W. West

The instrumentation and control systems design for the Clinch River Breeder Reactor Project is specified by the System Design Descriptions (SDDs). The SDDs are supported by engineering drawings of three basic types, namely, process and instrument diagrams, control logic diagrams, and elementary diagrams; various equipment tabulations; and performance specifications. The I&C engineering review of the reactor plant is intended to ascertain that (1) the correct information, logic, and data are contained within and consistent among the various documents, and (2) the engineered systems can be expected to function as intended in the SDDs. In executing this review, all appropriate codes, standards, and regulations are applied.

A data base management system has been established to record and recall transactions occurring on the more than 3300 drawings received to date. Thirteen of the requested nineteen balance-of-plant systems described by 1775 drawings have been reviewed since initiation of work in August 1981, and the results have been documented and transmitted to the CRBR Project Office.

1.6. Process Instrumentaton Development

1.6.1. STUDIES OF SHEATHED THERMOCOUPLE CONSTRUCTION AND INSTALLATION IN THERMOWELLS TO OBTAIN FASTER RESPONSE

R. M. Carroll
K. R. Carr R. L. Shepard

(Abstract of paper presented at The 6th Symposium on Temperature: Its Measurement and Control in Science and Industry, Washington, D.C., March 14–18, 1982)

Thermocouple response times were measured as a function of temperature to 650°C. Insulated-junction, stainless-steel-sheathed, MgO-insulated, Chromel-P/Alumel (type K) thermocouples had up to 50% shorter response times when the sheaths were swaged about 8% smaller at the junction. Grounded-junction type K thermocouples were made with nine times shorter response times by using a sheath closure technique that does not melt the thermocouple wires except at the surface. For a thermocouple-thermowell assembly, the response time depended primarily on the material in the annulus between the thermocouple and thermowell. The durability of tip-swaged insulated-junction thermocouples was evaluated by thermal cycling.

1.6.2. RESPONSE OF INSTALLED TEMPERATURE SENSORS

T. W. Kerlin* H. M. Hashemian†
R. L. Shepard K. M. Petersen†

(Abstract of paper presented at The 6th Symposium on Temperature: Its Measurement and Control in Science and Industry, Washington, D.C., March 14–18, 1982)

Two new methods have been developed to permit evaluation of the time response of installed temperature sensors. One permits improved *a priori* estimation of the response that will be obtained upon installation and the other permits *in situ* evaluation of the response of the sensor after installation. The *a priori* method requires response measurements at two or more fluid flow conditions to evaluate an internal (or intrinsic) component of the time constant that is independent of fluid conditions and a surface component that depends on fluid conditions. Estimation of the sensor response in other fluids is obtained from the heat transfer correlations for other fluids. The *in situ* method provides the time response of installed thermocouples and resistance thermometers by passing an electric current through it and measuring the transient response of the sensor following cessation or initiation of the Joule heating. This response can be analyzed to predict the sensor's response to a fluid temperature change. This measurement permits the response time qualification of newly installed sensors and subsequent monitoring of changes in response time with service.

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†Analysis and Measurement Services, Inc., Knoxville, Tennessee.

1.6.3. SURVEY, APPLICATIONS, AND PROSPECTS OF JOHNSON NOISE THERMOMETRY

T. Vaughn Blalock Robert L. Shepard

[Invited paper presented at Sixth International Conference on Noise in Physical Systems, NBS, April 6-10, 1981; *Natl. Bur. Stand. Spec. Publ.* 614, 260-68 (1981)]

Significant progress in the field of Johnson noise thermometry has occurred since the 1971 survey of Kamper. This paper will review the foundation work of Johnson noise thermometry, survey several basic methods of noise thermometry which use conventional electronic signal-processors, and present some applications of noise thermometry in temperature scale metrology and process temperature instrumentation. The important methods of cryogenic noise thermometry which use quantum devices such as Josephson junctions are not included in this survey.

1.6.4. A DECADE OF PROGRESS IN HIGH TEMPERATURE JOHNSON NOISE THERMOMETRY

T. V. Blalock R. L. Shepard

(Abstract of paper presented at The 6th Symposium on Temperature: Its Measurement and Control in Science and Industry, Washington, D.C., March 14–18, 1982)

The theoretical foundation is presented for Johnson noise thermometry. Basic methods for implementation of practical noise thermometers that use conventional electronic signal-processors are surveyed. Applications of noise thermometry in temperature scale metrology and process temperature instrumentation are described. Some conclusions and future prospects for Johnson noise thermometry are derived from the surveyed work.

1.6.5. JOHNSON NOISE POWER THERMOMETER AND ITS APPLICATION IN PROCESS TEMPERATURE MEASUREMENT

T. V. Blalock
J. L. Horton R. L. Shepard

(Abstract of paper presented at The 6th Symposium on Temperature: Its Measurement and Control in Science and Industry, Washington, D.C., March 14–18, 1982)

A Johnson noise power thermometer (JNPT) has been developed and applied to the measurement of temperatures from 400 to 1770 K in nuclear reactors and to the *in situ* calibration of platinum resistance thermometers in nuclear power plants. The JNPT measures the product of the open-circuit noise voltage and the short-circuit noise current from a sensing resistor, from which the absolute temperature and the resistance of the resistor can be calculated. The measurement uncertainty in temperature and resistance is less than 0.5% (99% confidence) for sensing resistors from 50 to 300 Ω over a temperature range from 273 to 1000 K, using signal cables as long as 18 m. The effects of signal cables can be corrected by employing computer algorithms based on either lumped-element or distributed-parameter models. Derivation of these algorithms is discussed. Design considerations are presented for the JNPT noise signal processor which consists of low-noise preamplifiers, bandpass amplifiers, rms-to-dc converters, and a computer-controlled data processor and readout.

1.6.6. NOISE EQUIVALENT CIRCUIT OF LINEAR PASSIVE TWO-PORTS WITH APPLICATIONS TO TRANSMISSION LINES

D. C. Agouridis

[Abstract of *IEEE Trans. Instrum. Meas.* IM-31(2), 119–24 (June 1982)]

A new representation of the thermal noise of linear passive two-ports, expressed in terms of the Thevenin equivalent circuit and convenient to circuit designers, has been derived. This representation is applied to the determination of the contribution of the noise of a transmission line (a special case of a two-port) to the total noise of a noise thermometer system. Preliminary experimental results from measurement of temperature with a noise thermometer connected via a long transmission line encourages further development of practical noise thermometers.

1.7. Special Assignments

1.7.1. INSTRUMENTATION FOR THE FEDERAL EMERGENCY MANAGEMENT AGENCY

F. W. Manning

For many years the Instrumentation and Controls Division has been actively engaged in advising and providing engineering assistance to the Federal Emergency Management Agency (FEMA) or its predecessor organizations on matters relating to radiological detection instrumentation. The first agreement was formalized in 1959, and the effort hovered at the one-man year per year level until FY 1982. As the national program has received more attention, interest in and support for radiological instrumentation has also increased. Under the present interagency agreement funding has risen to a level approaching $750,000.

Summaries of the FEMA tasks accomplished or undertaken in the last two years are listed below in chronological order.

1. A simulated CD V-715 was developed for the training of personnel in the use of radiation survey instruments. In this portable instrument programmed readings are imprinted in the PROM. With proper planning a team can survey a zone and determine the extent and amount of simulated radioactive contamination without the use of radioactive sources.
2. A transfer standard was designed and fabricated to provide instrument calibrations traceable to NBS standards. The device consists of a precision electrometer connected to an ion chamber. These units are calibrated by FEMA against a source traceable to NBS, and are then sent to each of the state shops for use as secondary standards for calibration of radioactive calibrators.
3. A batteryless dosimeter charger has been developed for fallout shelter sets. This device eliminates the logistic problem of periodically replacing dry cells to maintain dosimeters on ready alert. The device is capable of long-term storage. In the near future industry will make a production design and fabricate a small quantity of these chargers from a prototype supplied by ORNL.
4. The I&C Division has provided engineering and management assistance to FEMA to establish a mobilization base production facility for the manufacture of carbon fiber dosimeters. The facility is located in Rolla, North Dakota, and will be operated for FEMA by the Bulova Watch Company, the present operator of the existing William Langer Jewell Bearing Plant. The expertise of four other ORNL divisions contributed to this I&C project.
5. Under the present agreement I&C provides general engineering assistance for the maintenance and upgrading of FEMA instruments in use throughout the nation.
6. A detector assessment program has been established to assist FEMA and the armed services in obtaining satisfactory radiation detectors. FEMA has agreed to finance this program for an interim period, with the hope that its cost will be shared by other agencies in the future.
7. This Division is currently engaged in the development of advanced, low-cost instrumentation for measuring gamma dose rate.
8. Engineering assistance is being provided on a dosimeter program employing radiochromic dyes.

The FEMA project officer sets the priorities and schedules for the various FEMA programs, which may be changed from time to time. At present support for FY 1983 appears to be in the million-dollar range. *(F. W. Manning, H. N. Wilson, G. A. Holt)*

1.7.2. INSTRUMENTATION DEVELOPMENTS FOR TOBACCO SMOKE RESEARCH PROGRAMS

T. M. Gayle

During the reporting period a number of significant instrumentation projects have been carried out in conjunction with Analytical Chemistry Division contracts with The Council for Tobacco Research-USA (CTR) and the National Cancer Institute (NCI). These contracts include various aspects of physical and chemical characterization of cigarette smoke, as well as toxicological studies with large numbers of animals.

One of the major projects involved I&C division efforts to provide complete analytical instrument design and safety control for a large exposure inhalation facility for mice sponsored by CTR and carried out at Microbiological Associates, Bethesda, Maryland. In addition, we were given the responsibility for monitoring analytical procedures and providing documentation during three years of exposures of several thousand animals. Many of the key instruments such as the optical aerosol detectors, thermistor flowmeters, and binary gas analyzers were developed by ORNL to meet the specific demands of the inhalation facility. Data handling and computer processing techniques were also developed during the course of our contract. Approximately 35 field trips to Bethesda, Maryland were made by I&C personnel during the course of the contract in order to carry out the program. A complete Operating and Service Manual for the facility was prepared by I&C and published by ORNL. The two-volume set contains an extensive description of the inhalation facility, complete operating instructions, analytical techniques, and data handling procedures.

The program at Bethesda was completed in the fall of 1981. The equipment was then obtained by the University of Kentucky at Lexington for use in their long-term studies of the effects of cigarette smoke on non-neoplastic diseases in animals. They have recently contracted with ORNL to obtain the services of I&C personnel for operating assistance, design changes, and technical support for the facility. Several field trips have been made to U.K. by I&C personnel to assist in the setup and initial operation of the facility and to train personnel in analytical techniques.

An inhaled smoke dosimeter was developed for The National Cancer Institute to better quantify tobacco smoke delivery to large test animals such as dogs. The dosimeter measures aerosol concentration using the optical sensor developed at ORNL and also measures instantaneous inhaled flow rate with a laminar flow device. The two electronic signals are amplified, multiplied together, scaled, and displayed as dose level. The product signal is digitally integrated and displayed on a solid state counter as accumulated dose. The unit has been successfully used at several NCI-sponsored laboratories.[1] An extension of this basic design is currently under development to provide a cigarette smoke dosimeter for human studies. Both the flow measurement device and the optical aerosol sensor have been miniaturized and are mounted in a small cigarette holder used by the smoker. The unit has been demonstrated to NCI staff and continued development has been authorized. *(R. A. Jenkins,* R. W Holmbert*).*

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*Analytical Chemistry Division.

1. R. A. Jenkins and T. M. Gayle, "An Instrumental Inhaled Smoke Dosimeter for the Quantitative Characterization of Aerosol Exposures," Pulmonary Toxicology of Respirable Materials, 19th Annual Hanford Life Sciences Symposium (December 1980), NTIS CONF-791002.

1.7.3. MILITARY VISUAL OBSCURANT STUDIES

T. M. Gayle

Active support is being given the Analytical Chemistry Division in an extensive program of aerosol technology research and development in conjunction with investigations into the inhalation toxicology of military smokes and obscurants. The several programs involve different materials and cover detailed chemical and physical characterization of the various compounds used, as well as long-term exposures of animals to determine toxicity. In addition to an ongoing investigation of an uncombusted diesel fuel aerosol in cooperation with the Biology Division, we are working on three other obscurant systems: "fog" oil and two incendiary phosphorus formulations. One formulation is composed of red phosphorus containing butyl rubber as a binder, and the other contains white phosphorus impregnated into a wool felt matrix. These represent formulations of current interest to the Department of Defense. The animal exposures for these compounds will be carried out by outside contractors under ORNL supervision.

The instrumentation for aerosol concentration measurements employs a combination LED-phototransistor sensor developed by the I&C Division for cigarette smoke studies in previous programs.[1] The sensor is used with a newly designed remote amplifier-readout unit which displays the concentration, provides a digital integration of concentration (dose), and supplies alarm signals for system control. In the diesel obscurant studies in the Biology Division, eight of these measuring systems are used to document dosage as well as provide automatic shutdown in the event of accidental overexposure.

A high temperature nitrogen purged generator designed by Analytical Chemistry is used to generate the uncombusted diesel aerosol. The diesel fuel is introduced into the high temperature nitrogen atmosphere, adequately simulating the military technique of injecting oil into the exhaust manifold of a tank. The aerosol is then injected into the air inlet flow stream of the animal chambers. Instrumentation for zoned temperature control is included in the aerosol generator, and extensive safety instrumentation is provided in the exposure systems to insure animal safety.

The "fog" oil studies are being designed to simulate a small portable Army field generator that uses a petroleum distillate similar to 10W lubricating oil. Operating limitations preclude use of the actual field generator in these studies, and a generator and monitoring systems similar to those used for diesel oil has been found suitable for "fog" oil. The Environmental Protection Agency (Research Triangle, North Carolina) has been selected to perform the animal exposures under ORNL supervision. We have fabricated eight aerosol measuring systems as well as generator temperature controls for their use.

The red phosphorous and white phosphorous systems are currently under development. The aerosol measuring instruments for use in concentrations up to 10 mg/L will be essentially the same as those used for oil exposures. The generators for the phosphorous compounds require a design entirely different from that used for oil. It has been found that red phosphorous-butyl rubber softened by the additon of hexane can be extruded through a small orifice with a hydraulic press system. The phosphorous burns when it is exposed to air as it exits the orifice, thus producing the aerosol. The hydraulic system can be modulated to control the extrusion rate and thus control aerosol concentration as measured by the LED-phototransistor sensor. Preliminary results indicate that the system may be adapted to the generation of aerosol from the white phosphorous-felt formulation. A complete prototype system has been developed and tested and will be provided to the outside contractor who will conduct the toxicological studies. *(R. W. Holmberg,* J. H. Moneyhum*).*

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*Analytical Chemistry Division.

1. C. E. Higgins, T. M. Gayle and J. R. Stokley, "Sensor for the Detection of Tobacco Smoke Particulates in Inhalation Exposure Systems," *Bieträge Zur Tobakforsching* 9(4), (June 1978).

1.7.4. SOIL MICROBIOLOGY: A MODEL OF DECOMPOSITION AND NUTRIENT CYCLING

O. L. Smith

(Summary of a book, CRC Press, Inc., Boca Raton, Florida, 1982)

An intermediate resolution model of the decomposition of soil organic matter is developed from a comprehensive study of published experimental work. The many organic and inorganic forms of soil N, P, and K are mathematically treated, together with the various transformations between forms. Most of the transformations are moderated by microbes, and the dynamics of the microorganisms are explicitly represented. A simulation is made of a general heterotrophic population using organic C and N for energy, as well as nitrifiers which oxidize nitrogenous compounds chemotrophically. Such explicit treatment of microbe dynamics permits among other things the study of microbe immobilization of important plant nutrients. In addition to the simulation of biological aspects of decomposition, the model treats the physicochemical processes of precipitation, fertilizer and native mineral inputs, leaching loss, sorption of organic and inorganic ions on soil colloids, condensation between organic N and aromatic compounds, and exchange reactions. Model parameters are dealt with in detail in order to base them as firmly as possible on experimental information. The process rates include functional dependence on soil temperature and moisture. For clarity of presentation, the model is divided into four submodels, one each for N, P, and K, and the C energy substrate. The total model is coupled with a plant growth model, and thereby simulates complete element cycles within the plant-soil system.

The model is tested against published experimental results and used to study important soil processes and to offer explanations of well known but incompletely understood experimental observations. The cycles of the three macronutrients N, P, and K through the plant-soil system are treated under both equilibrium and dynamic conditions. The equilibrium equations show that in many ecosystems, the annual-average soil solution concentrations of P and K are independent of the soil biota activity whereas the relative concentrations of the various N-forms in solution remain dependent on related microbe activity. The dynamic equations are compared with and shown to agree closely with a number of experimental observations, including (1) the overall pattern of decomposition and growth of heterotrophs and nitrifiers, (2) the immobilization and mineralization of N as a function of substrate C-N ratio, (3) wastage of substrate by various microbes, (4) the N priming effect, (5) the effect on microbes of oscillating low soil temperatures, and (6) the effect on microbes of soil moist-dry cycles. Among the conclusions drawn from the full model equations are the following: (i) the N priming effect is the consequence of a two-step substrate limitation involving first N and later C because of microbial waste metabolism, (ii) oscillating low soil temperatures result in lower population levels than does the mean temperature at least in part because nonlinearities give a net reduction in growth under oscillating conditions, (iii) moist-dry cycles, lethal to soil organisms, enhance CO_2 evolution in part because of the organisms' self-metabolism, (iv) plant and microbes compete for N to a greater or lesser extent depending on the kind of N available, the C:N ratio, the amount of leaching, and whether or not microbe use of NO_3^- is suppressed, and (v) in fertilizing some crops with NH_4^+, NO_3^-, or organic N, multiple-batch application is preferable with the mineral forms, whereas a single batch is best with the organic form because of differences in microbial immobilization.

1.7.5. THE INFLUENCE OF ENVIRONMENTAL GRADIENTS ON ECOSYSTEM STABILITY

O. L. Smith

[Summary of *Am. Nat.* **106**, 1-24 (July 1981)]

Ecosystems along environmental gradients, one of the most common types of spatial heterogeneity, were evaluated in terms of the linear stability formalism. The more rapidly a perturbed system returns to equilibrium the more stable it is, and a normalized system time constant was developed as a measure of the dynamic properties which aid species in their persistence. This index of relative stability based on log-phase dynamics was shown to be strongly correlated with the behavior of a variety of systems.

Diffusion equations provide a useful representation of the ecosystem characteristics of particular interest in this study. System size, dispersal of seeds, symmetric and asymmetric distributions of growth and mortality, and the consequent spatially varying population profiles are readily treated. From the formalism a number of basic relationships between disturbed parameters and system dynamics were deduced and related to observations in the literature.

The formalism developed here is useful in problems of experimental data analysis and mathematical modeling in which it is desirable to represent a spatially complex system by a simpler average. Two common averaging techniques, one of using the parameter arithmetic average and the other of population weighting the parameter distribution, can result in large errors in such estimates of system behavior. The two procedures may be adequate to represent systems having small edge effects associated with biomass loss from the borders, but, in steep-gradient systems where edge effects may be significant, ignoring them can result in large over- or underestimation (100% or more) of system performance. A better procedure for forming either a data or model average is to formulate a detailed comparison between the actual system equations and the averaging equations and define an equivalent averaging procedure which results in a conceptual match that omits none of the important system properties.

1.7.6. MULTI-WAVELENGTH RAMAN SPECTROSCOPY FOR SPECIES DETERMINATION IN FISSION PRODUCT RELEASE EXPERIMENTS: A PRELIMINARY STUDY*

J. B. Davidson

Laser-Raman Spectroscopy (LRS) is a proposed method for determination of molecular species in a fission product release experiment (Report SAND80-2662). The Sandia (Albuquerque) group has proposed controlled laboratory experiments in which several of the unexpected vapor species would be examined both singly and in combination using a conventional scanning laser excitation and detection system and a high-temperature cell. Raman spectra and concentration data from these experiments were to be extrapolated to an actual fuel element vaporization experiment involving high-temperature steam to determine the feasibility of doing simultaneous multiwavelength, multispecies recording of spectra as close as possible to the point of fission product release.

It was suggested that a wide-range, TV-echelle spectrograph developed in the Instrumentation and Controls Division might be used for recording the spectra. Using this device, a series of integrated exposures of the order of 30 s would be recorded on analog video tape for perhaps 1 h. After the experiment, the data at selected intervals could be digitized and stored in an image memory, and spectra could be identified by comparison with previous calibration runs. Because the whole experiment could be reproduced (spectroscopically) from the tape, it was surmised that the evolution of several species could be studied at leisure.

Experiments performed on the TV-echelle spectrograph indicated that it could be used to obtain Raman spectra, but that sensitivity would have to be increased by additional light intensification and more laser power. It was also determined, however, that the wide spectral range of the TV-echelle system may not be needed and that commercially available multiwavelength Raman systems could be used if adequate signals can be obtained on a conventional scanning system. Experiments being conducted by W. H. McCulla of the ORGDP Enrichment Technology Division are directed to establishing minimum concentrations. He has obtained promising results on CsI and I_2 samples.

Since the spectrograph cannot be placed inside the hot cell in an actual experiment, optical bench tests were made simulating the sample-to-spectrograph distance which would be necessary. It was determined that mirror-lens coupling could relay the signal out of the hot cell without excessive loss.

Assuming adequate funding to obtain state-of-the-art commercial instrumentation, final determination of feasibility awaits further results of the ORGDP experiments. *(R. A. Lorenz,† A. L. Case)*

*Title of an interim report to the Fuel Behavior Branch, U.S. Nuclear Regulatory Commission, April 30, 1982.

†Chemical Technology Division.

1.7.7. RAPID ELECTRONIC AUTOFLUOROGRAPHY OF LABELED MACROMOLECULES ON TWO-DIMENSIONAL GELS

J. B. Davidson A. L. Case

[Abstract of *Science* 215 (March 12, 1982)]

The feasibility of electronically locating and measuring tritium-labeled macromolecules directly on dried electrophoretic gels has been demonstrated. This new procedure eliminates the usual long film exposure in autofluorography and the attendant delay in processing and data reduction. An image intensifier and electronic camera tube are used to integrate the light produced by the tritium interaction with a scintillator incorporated in the gel. Preliminary results show that, compared to film, the exposure is reduced 100 to 1000 times. The response to low activity levels is improved, and spatial resolution is maintained. A proposed instrument could be used for measuring other isotopes as well as fluorescent and visible stains.

1.7.8. A "TABLE TOP" SMALL-ANGLE NEUTRON SCATTERING EXPERIMENT

J. B. Davidson

The typical neutron small-angle scattering instruments are 10–30 m long with large area detectors measuring ~1 m on a side. The size of these devices is determined by several factors—the angular resolution required, the sample size to be used (larger samples scatter more neutrons, making the experiments faster), and the size of the detector's resolution element.

Given a sample area of 1 cm^2 and a detector resolution element of 1 cm^2, the desired angular resolution (~1 mrad), can be gained only by separating the sample and the detector by 10 m or more. However, if a detector having a resolution element size of, say, 1 mm^2 could be used the resolution of 1 mrad can be obtained at a distance of 1 m, thereby shrinking the size of the experimental arrangement by a factor of 10. The smaller resolution element requires that a smaller sample be used and, consequently, requires a longer counting time for a given incident beam intensity.

It was possible to demonstrate this application of a position-sensitive detection and recording system which had been developed for optical spectroscopy. Some experimental time at Port HB-3 at the ORR was obtained through the cooperation of J. W. Cable and H. R. Child of the Solid State Division. This port provides a beam of 4.7 Å neutrons from graphite monochromators. Some shielding was borrowed from the Ames Neutron Group and the detector was set up on a table 2.1 m long in a "small-angle scattering" geometry using collimator-to-sample and sample-to-detector distances of 0.5-1.5 m. Our detector was made neutron-sensitive by means of a 150-mm-diam ^{6}Li-ZnS screen ~0.5-mm thick. The resolution of the detector was ~0.3 mm FWHM for 4.7 Å neutrons. The expected circular scattering patterns were obtained using samples of Ludox particles of 70, 90, and 220 Å in water supplied by R. Triolo of the Chemistry Division.

For some applications not requiring extremely high angular resolution and where longer counting times can be tolerated, a table top system could save hundreds of thousands of dollars usually spent for construction of large vacuum flight paths, trolleys, building extensions, etc. In addition, the cost of the smaller solid phosphor scintillation detector could be one-third to one-half that of the larger gas-filled detectors, which use expensive ^{3}He as the neutron interacting medium.

The table top system might make possible small-angle scattering experiments at university reactors where budgets are smaller and experimental time is cheaper. Alternatively, several such systems could be built at a large reactor for specialized applications. *(A. L. Case)*

1.7.9. TELEVISION-BASED NEUTRON DETECTORS AND APPLICATIONS

J. B. Davidson

(Abstract of paper presented at The Neutron and Its Applications, Cambridge, England, September 13–17, 1982)

A review of TV-based thermal neutron detectors at Oak Ridge National Laboratory is presented. Fiber-optic as well as lens-coupled systems are described. A counting camera sensitive to single events and also having a target integrating mode capable of accumulating data for 1 h is shown. Data with resolution of ~0.3 mm are recorded in an image memory having 512 × 512 × 12 bits of storage. The memory is controlled by an inexpensive microcomputer using the IEEE 488 bus. Single event data are scanned from the camera target and added to memory with a cycle time of 100 ns. Analog, integrated data are digitized to 8 bits at 10^7 pixels/s. Point-by-point background subtraction can be done in 33 ms. Flicker display of two 256- × 256-pixel images provides rapid qualitative comparison. Digital data can be stored on floppy disks or transferred to computer tape for further analysis.

Applications include: in-crucible crystal inspection, display of magnetic domains, section topography of crystals and of directionally solidified materials such as turbine blade alloys, searching for unknown reflections in phase transition studies, and small-angle scattering. A "table top" small angle scattering experiment using ~1 m collimator and sample-to-detector distances is described and some results using LUDOX particles given.

1.7.10. CALCULATION OF THE PROBABILITY OF OVERLAPPING ONE FAMILY OF NUCLEAR LEVELS WITH RESONANCES OF AN INDEPENDENT FAMILY

F. C. Difilippo*

[Reprint of *Trans. Am. Nucl. Soc.* 41, 561–62 (1982)]

Calculations[1] of the resonance integrals of particular isotopes in a mixture of isotopes show that the overlapping of the resonances of one isotope by resonances of other isotopes affects the final values of effective cross sections. The same effect might adversely influence those nondestructive techniques which assay fissile materials on the basis of resonance effects.[2] Of relevance for these applications is the knowledge of the probability of overlapping resonances of a family of nuclear levels (class 1) with resonances of an independent family (class 2). For the sequence of class 1 resonances, we calculate the probability distribution, $p(\delta)$, to find a class 2, first-neighbor resonance at distance (in energy) δ from a class 1 resonance; integration of $p(\delta)$ over the average finite width of the resonances would give the aforementioned probability of overlapping. Because a class 1 resonance can have a class 1 or a class 2 resonance as a first neighbor, the resultant $p(\delta)$ is not given by the distribution[3] of spacings of the composite family (i.e., family 1 plus family 2).

Let $R(\delta)$ and $Q(\beta)$ denote the probability densities to find to one side and a distance δ and β from a class 1 resonance, a class 2 and a class 1 resonance, respectively, as a first neighbor. The probability density that a class 1 resonance has a class 2 first neighbor around δ to either side is given by

$$p(\delta) = 2R(\delta)\int_0^\infty Q(\beta)d\beta + 2R(\delta)\int_\delta^\infty R(\beta)d\beta \ . \quad (1)$$

The first term in Eq. (1) states the probability of having a class 2 resonance to one side and a class 1 resonance to the other side; the second term states the probability of having two class 2 resonances, one to each side of the class 1 resonance. The factor 2 appears because we do not distinguish between class 2 resonances that appear to either side of the class 1 resonance.

The probability distributions $R(\delta)$ and $Q(\beta)$ are related to the level distributions of the two families of resonances, $\pi_1(D)$ and $\pi_2(D)$, as follows:

$$R(\delta) = g(\delta,D_2)\int_\delta^\infty \pi_1(D)dD \quad (2)$$

and

$$Q(\beta) = \pi_1(\beta)\int_\beta^\infty g(\sigma,D_2)d\sigma \ , \quad (3)$$

where D_2 is the average spacing of class 2 resonances, and

$$g(\sigma,D_2) = \frac{1}{D_2}\int_\sigma^\infty \pi_2(D)dD \quad (4)$$

is the probability distribution of the nearest level to a random point.[3]

Assuming the case of a Wigner law[3] for π_1 and π_2 and substituting Eqs. (2), (3), and (4) in Eq. (1), we get

$$p(\delta) = \frac{2}{D_2}\exp\left[-\frac{\pi}{4}\delta^2\left(\frac{1}{D_1^2}+\frac{1}{D_2^2}\right)\right] \times\left\{1-\frac{x}{(1+x^2)^{1/2}}\operatorname{erf}\left[\frac{\sqrt{\pi}}{2}\frac{\delta}{D_1}(1+x^2)^{1/2}\right]\right\} \ , \quad (5)$$

where $x = D_1/D_2$ (D_1 is the average spacing of class 1 resonances) and erf is the error function.

The fraction f of the total number of class 1 resonances having a class 2 first neighbor at distance between 0 and Δ is

$$f\left(\frac{\Delta}{D_1},x\right) = \int_0^\Delta p(\delta)d\delta$$

$$= \frac{x}{(1+x^2)^{1/2}}\operatorname{erf}\left[\frac{\sqrt{\pi}}{2}\frac{\Delta}{D_1}(1+x^2)^{1/2}\right]$$

$$\times\left\{2 - \frac{x}{(1+x^2)^{1/2}} erf\left[\frac{\sqrt{\pi}}{2}\ \frac{\Delta}{D_1}(1+x^2)^{1/2}\right]\right\}\ . \quad (6)$$

The fraction F of the total number of class 1 resonances having a class 2 first neighbor at any distance is

$$F = \int_0^{\infty} p(\delta)d\delta =$$

$$\frac{x}{(1+x^2)^{1/2}}\left[2 - \frac{x}{(1+x^2)^{1/2}}\right]\ . \quad (7)$$

Figure 1.7.1 compares calculated results from Eqs. (6) and (7) with experimental data from Ref. 4 for different values of the parameter Δ/D_1. The intercomparison shows agreement within the indicated confidence limits that result from the finite number of class 1 resonance data that are available for analysis. It is evident that even for small values of Δ/D_1, the overlapping effect is important for large values of x, since f and F approach unity as x approaches infinity.

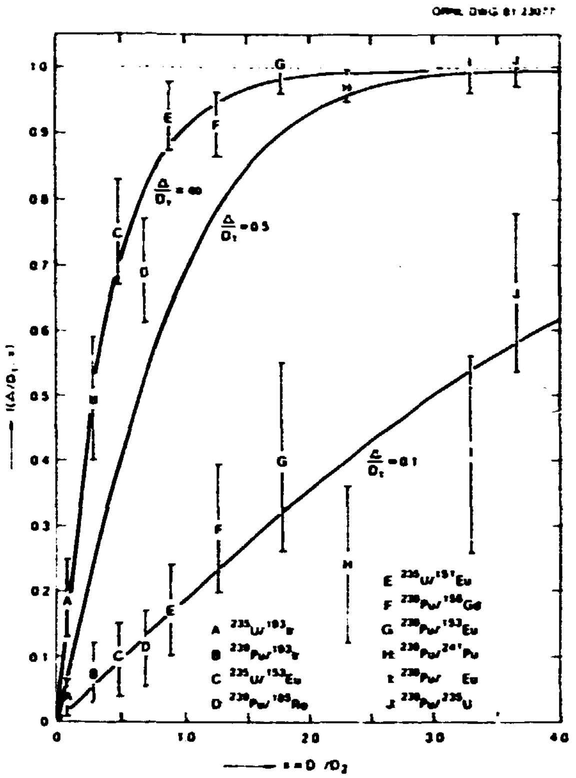

Fig. 1.7.1. Fraction of class 1 resonances with a class 2 first neighbor at a distance $\leqslant\Delta/D_1$. The continuous line represents theoretical results, Eqs. (6) and (7). Experimental data are plotted for ΔD_1 = 0.1 and ΔD_1 = ∞, and family 1 is the first member of each pair.

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1.7.11. EVALUATION OF PRESSURIZED THERMAL SHOCK

R. C. Kryter — T. J. Burns*
R. D. Cheverton† — R. A. Hedrick‡
F. B. K. Kam** — C. W. Mayo‡

[Abstract of NUREG/CR-2083 (ORNL/TM-8072), October 1981]

This report evaluates the threat to reactor vessel integrity posed by pressurized thermal shock (overcooling) events. The study focuses on pressurized-water reactors manufactured by the Babcock and Wilcox Company, as typified by the Oconee Unit 1 plant. The thermal-hydraulic transient data used as input to the linear elastic fracture-mechanics analysis of vessel flaw propagation were calculated by others, using the IRT and TRAC computer codes. In some hypothetical transients, vessel failure due to pressurized thermal shock is predicted early in reactor life, but identified calculational approximations, modeling deficiencies, and uncertainties in the probabilities of occurrence for the overcooling transients studied suggest that the quantitative results obtained should be regarded as preliminary estimates having substantial associated uncertainties. These areas of uncertainty are evaluated and recommendations are made for further work.

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1.7.12. CALCULATIONS OF THE CRBR INITIAL LOADING MOCKUP EXPERIMENTS

D. L. Selby* J. T. Mihalczo

[Summary of *Trans. Am. Nucl. Soc.* 39, 930 (1981)]

An initial loading experiment was performed in the Zero Power Plutonium Reactor[1] (ZPPR) to evaluate the Clinch River Breeder Reactor (CRBR) initial loading procedure. This evaluation includes confirmation of calculational methods for predicting the response of detectors for monitoring initial loading. Changes in the response of the source range flux monitor (SRFM) to an extraneous neutron source were also examined as a means for enhancing low SRFM count rates during the early stages of the loading. The experimental mockup included the core regions and an ~60-deg sector representing the materials from the radial blanket through the SRFM block. The one exception is that the steel matrix of the ZPPR facility introduced steel in nonsteel regions and precluded the mockup of the cavity region. Fuel was loaded symmetrically from the center outward, with some effort in the later stages of the loading to equalize fuel subassembly reactivity worths.

Analysis of the analytical and experimental results yielded the following conclusions:

1. Inverse count rate curves constructed from experimental data were very similar to those constructed from calculations.
2. The use of transport theory was necessary to calculate absolute count rates at the SRFM location.

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1.7.13. PRELIMINARY INVESTIGATION OF ^{252}Cf-DRIVEN NEUTRON NOISE ANALYSIS FOR SUBCRITICAL FUEL SOLUTION SYSTEMS

J. T. Mihalczo
R. C. Kryter W. T. King

[Summary of *Trans. Am. Nucl. Soc.* 38, 359 (June 1981)]

A method for determining the reactivity of highly subcritical systems of fissile material, using neutron-noise power spectral densities in conjunction with a ^{252}Cf source, had previously been tested in two fast reactor critical assemblies (a mockup of the Fast Flux Test Facility reactor and unreflected enriched uranium metal assemblies[1]) and one thermal reactor (a light water moderated and reflected lattice of Oak Ridge Research Reactor fuel elements[2,3]). The last-mentioned test demonstrated the effectiveness of the method in water-moderated systems and thereby prompted the present study of its application to facilities for fuel preparation, reprocessing, and storage.

To investigate the applicability of this method to facilities for fuel preparation, reprocessing, and storage, limited experiments were performed with a uranyl fluoride solution. The Los Alamos National Laboratory SHEBA facility,[4] an unreflected cylindrical tank (56 cm diam), was partially filled with a solution containing 5 wt% ^{235}U-enriched uranium [ρ(U)~1 g/cm^3, H/U atomic ratio ~550].

These preliminary investigations show the ^{252}Cf-driven noise measurement to be a promising technique for on-line monitoring of the subcriticality of fuel solution tanks in operating facilities.

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1.7.14. FEASIBILITY OF LWR SUBCRITICAL REACTIVITY MONITORING USING THE ^{252}Cf-DRIVEN NEUTRON NOISE METHOD

J. T. Mihalczo
W. T. King J. A. Renier*

[Summary of *Trans. Am. Nucl. Soc.* 41, 619–20 (1982)]

Calculations were undertaken to ascertain if the subcriticality of large arrays of light water reactor

(LWR) fuel pins can be measured by the ^{252}Cf-driven neutron noise method.[1] Such arrays are typical of those encountered in the initial fuel loading of LWRs and in spent fuel storage pools.

Both static and kinetic, two-dimensional diffusion calculations estimated the quantities that would be measured. In addition, the Monte Carlo method determined the spatial distribution of the fission multiplicative processes and thereby qualitatively ascertained how the longer fission chains contribute to the counts registered by the detector.

These calculations indicate that subcriticality measurements by the ^{252}Cf-driven neutron noise method are feasible. Such measurements would be independent of detector location over a large section of the core in which the detection efficiency would be sufficient for adequate statistical precision. Application to the highly poisoned arrays of LWR fuel pins encountered during the initial loading of PWRs and in fuel storage pools may also be feasible. For the nearly infinite systems, either multiple ^{252}Cf sources and detector pairs or a single source-detector pair that could be moved around the core may be required to monitor the reactivity of the total system. Further evaluation of the applicability of this method to arrays of LWR fuel pins should be done experimentally in a mockup of a partial core or fuel storage pool.

*UCC-ND Computer Sciences Division.

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1.7.15. BWR SUBCRITICAL REACTIVITY MONITORING USING THE ^{252}Cf SOURCE DRIVEN NEUTRON NOISE METHOD

J. T. Mihalczo
W. T. King ***J. A. Reiner****

(Abstract of invited paper presented at Ninth NRC Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 26-30, 1981)

A calculational program was undertaken to ascertain if the shutdown margin of a BWR after the initial loading of fuel can be measured by the ^{252}Cf driven neutron noise analysis method. The ^{252}Cf driven neutron noise analysis method obtains the reactivity from measured ratios of power spectral densities, $(G_{12}^{*} \cdot G_{13})/(G_{11} \cdot G_{23})$ where the subscript 1 refers to an ionization chamber containing ^{252}Cf placed in the core that provides source neutrons to initiate the fission chain multiplicative process and the subscripts 2 and 3 refer to a pair of detectors also in the core which detect neutrons from the source multiplied fission chains. The theory of this measurement method has been verified in experiments with fast critical assemblies. Recent measurements with a plate type research reactor at Oak Ridge National Laboratory (ORNL) and with a uranyl fluoride solution tank at Los Alamos demonstrated its applicability to water moderated systems. The uranyl flouride experiment also demonstrated the advantage of this method over the conventional inverse count rate method of monitoring initial loading to criticality since one of the main advantages of the method is that the ratios of spectral densities does not depend on detection efficiency which usually changes with fuel loading.

The methods of calculation used in this analysis were: (1) Monte Carlo to examine the spatial extent of the various size fission chains; (2) static 2-dimensional diffusion theory to calculate the detection efficiencies for this type of measurements; and (3) a kinetics code using 2-dimensional diffusion theory, JPR kinetics, to obtain the cross power spectral densities from which the appropriate ratios of spectral densities can be obtained.

The results of this investigation indicate that measurements of the shutdown margins of BWR's by this technique are feasible. We recommend that an experiment with LWR fuel pins be performed to test the method. The experiment should be performed with a partial core mockup where the systematics of spatial effects would be evaluated experimentally and provide a benchmark for calculations. This experiment would test the ability of this method to monitor the subcriticality during startup operations and could also be designed to test this method for fuel storage applications.

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1.7.16. A REVIEW OF SUBCRITICALITY MEASUREMENTS USING ^{252}Cf-DRIVEN POWER SPECTRAL DENSITY MEASUREMENTS

J. T. Mihalczo

[Abstract of *Trans. Am. Nucl. Soc.* 39, 517 (December 1981)]

The ^{252}Cf neutron-driven noise analysis method has two advantages over other methods of reactivity determination: (a) the subcriticality of a multiplying system can be determined without knowledge of its neutronic properties at delayed criticality, and (b) the interpretation of the measured data to obtain the subcriticality does not depend on knowledge of the inherent source strength or the detection efficiency. Thus, this method can be used in the initial fuel loading of systems where determination of subcriticality dependent on some calibration near critical is not possible. It can also be used in determining the reactivity of assemblies where sufficient material to achieve criticality is not available or where loading to critical is undesirable.

The ^{252}Cf method of measurement has potential application in (a) initial loading of reactors (light water reactors and liquid-metal fast breeder reactors), (b) refueling of reactors, (c) fuel processing and reprocessing facilities, (d) fuel storage facilities, (e) post-accident situations, and (f) basic experiments where sufficient fuel to achieve criticality is not available. Before practical application is implemented, however, additional experiments are required to investigate the following areas: the lower practical limit of measurement, restrictions on placement of detectors, spatial effects, and measurement times.

1.7.17. CALCULATED RATIOS OF SPECTRAL DENSITIES FOR ^{252}Cf-DRIVEN NEUTRON NOISE SUBCRITICALITY MEASUREMENTS WITH A 5%-^{235}U-ENRICHED URANYL FLUORIDE SOLUTION

J. T. Mihalczo
W. T. King J. A. Renier*

[Abstract of Trans. Am. Nucl. Soc. 41, 588 (June 1982)]

A method that uses ratios of neutron-noise power spectral densities in conjunction with a ^{252}Cf source[1] for determining the reactivity of a highly subcritical system of fissile materials was demonstrated in experiments with an unreflected, ^{235}U-enriched (5%), uranyl fluoride solution tank.[2]

The results from these experiments[2] show the applicability of this method for monitoring the subcriticality of fuel solution tanks in fuel processing or reprocessing facilities. This paper describes calculations of these spectral densities and the spectral density ratio as a function of frequency using the JPRKINETICS code[3] and compares the calculated with measured results.

The good agreement between the calculated and measured parameters validates this calculational method for unreflected, low-enriched, fuel-solution systems with a central ^{252}Cf neutron source. Further comparisons between measurement and calculations for a variety of solution systems are required before acceptance of this calculational method for these types of experiments.

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*UCC-ND Computer Sciences Division.

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1.7.18. ^{252}Cf-SOURCE-DRIVEN NEUTRON NOISE METHOD FOR MEASURING THE SUBCRITICALITY OF SUBMERGED HFIR FUEL ELEMENTS

J. T. Mihalczo W. T. King

[Summary of *Trans. Am. Nucl. Soc.* 43, 408 (1982)]

The ^{252}Cf-source-driven neutron noise analysis method[1] was tested to determine whether it could be applied to measure the subcriticality of High Flux Isotope Reactor (HFIR) fuel elements[2] submerged in water. If successful, this method would have significant advantage over methods currently in use for this purpose. Presently the reactivity of each assembled fuel element in water is measured after shipment from the fuel fabricator and prior to insertion into the HFIR by performing critical experiments using additional uranium fuel and/or absorber to achieve delayed criticality. Since this new ^{252}Cf-driven subcriticality measurement method would not require critical experiments, it would be time and cost effective. To determine the accuracy of the measured subcritical reactivities, the ^{252}Cf noise method was compared with the results from another measurement method, the break-frequency noise analysis (BFNA) method.[3]

The HFIR fuel element tested consisted of involute fuel plates arranged as two concentric annuli, the inner one 12.9 cm o.d. and the outer one 43.5 cm o.d. When submerged, the center of the element was water-filled, and a >15-cm-thick reflector of water covered the top, bottom, and side. Additional uranium fuel plates were placed in the central water-filled region and boron-stainless steel strips were placed in the outer annulus to adjust the reactivity.

An ionization chamber containing a ^{252}Cf neutron source (detector 1), which also furnished neutrons for the fission chain multiplicative process, was placed in the central water region along the axis of the annuli at midplane height. Two additional neutron detectors (detectors 2 and 3), which responded to particles from fission chains, were placed opposite one another and adjacent to the outer surface of the annuli at the midplane. In measurements with calibrated ^{235}U plates in the central water-filled region, all three detectors were placed adjacent to the outer surface of the annulus, 120° apart and at midplane height.

In Fig. 1.7.2 the ratio of spectral densities $G^*_{12}G_{13}/G_{11}G_{23}$ obtained from the ^{252}Cf method is plotted vs reactivity obtained from the BFNA method previously described,[3] which required a measurement at a known reactivity (in this case, ten cents subcritical) as a reference. This reference reactivity was obtained from inverse kinetics rod drop measurements,[3] in which the reactivity was reduced by decreasing the top reflector thickness. Negative reactivities were varied below 2.8 dollars by adding boron-stainless steel strips to the outer annulus, and above 2.8 dollars by adding uranium

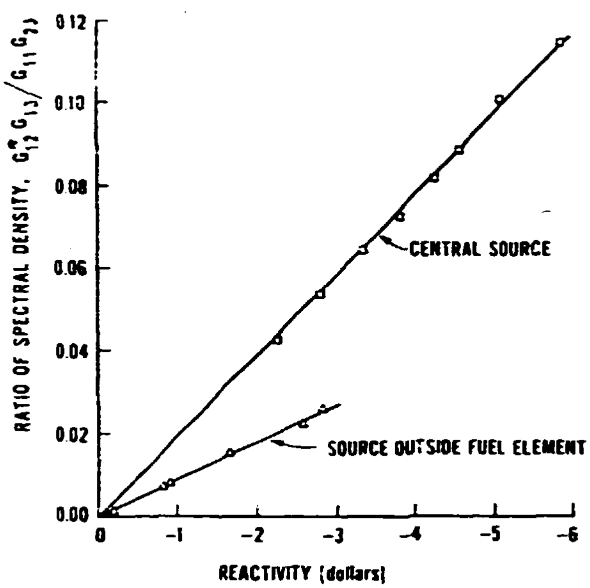

Fig. 1.7.2. Ratio of spectral densities, $G^*_{12}G_{13}/G_{11}G_{23}$ versus reactivity from BFNA for HFIR fuel elements submerged in water.

fuel plates to the water-filled central region. The dependence of the ratio on source position seen in Fig. 1.7.2 is due to the difference in the importance of neutrons born at the center vs the importance of those born at the outer surface of the fuel element. However, for each source location the linearity of the ratio of spectral densities as a function of reactivity shows that the proportionality constant between the ratio of spectral densities and reactivity is (as expected) not reactivity dependent. Thus the proportionality constant obtained from measurements with one submerged fuel element can be used to obtain the reactivity in measurements with other submerged fuel elements.

Subsequent measurements by the ^{252}Cf method with ten other HFIR fuel elements (Table 1.7.1) yielded subcritical reactivities between 2.22 and 2.77 dollars. In all cases, the values agreed within six cents of the values from the independent BFNA method. This agreement validates the ^{252}Cf method for this application.

The results of both the ^{252}Cf and BFNA methods, corrected for reactivity effects of the detectors, show that the reactivity of a submerged fuel element is ~25 cents less subcritical than that determined by the critical experiment method.[4]

Table 1.7.1. Comparison of submerged HFIR fuel element reactivities from ^{252}Cf-driven neutron noise analysis, standard breakfrequency noise analysis, and critical experiment methods

HFIR fuel element no.	Subcritical reactivity (dollars)[a]		
	^{252}Cf method[b]	Breakfrequency noise analysis[b]	Critical experiment[c]
68	2.22	2.17	2.14
222	2.77	2.79	2.89
229	2.69	2.66	2.71
230	2.55	2.56	2.61
231	2.77	2.74	2.80
233	2.55	2.58	2.49
234	2.61	2.61	2.70
235	2.70	2.67	2.69
236	2.62	2.62	2.70
237	2.72	2.66	2.67
238	2.68	2.72	2.76

[a]Measurement precision is 1%.

[b]Values not corrected for detector reactivity effects, which are ~25 cents in all cases.

[c]Reference 4.

This latter method makes use of the calibrated reactivity worth of fuel plates positioned in the central, water-filled region to add sufficient reactivity to make the assembly critical or moderately supercritical. Any excess reactivity is compensated in part by calibrated boron-stainless steel strips positioned between fuel plates in the outer annulus; any remaining excess reactivity is determined by positive period measurements. The reactivity worths of the individual fuel plates were measured in critical experiments[2] with other fuel plates present in the central water-filled region; the plates were thus located in a higher flux region than would be present in the normal HFIR fuel element submerged in water. This caused the reactivity worth of the extra fuel plates to be overestimated, resulting in overestimation of the subcritical reactivities of the submerged HFIR fuel elements.

The ^{252}Cf-source-driven neutron noise analysis method is a viable technique for subcriticality determination of HFIR fuel elements submerged in water. With existing signal processing and analysis, the ratio of spectral densities (thus the reactivity) can be measured to an accuracy of 1 in 100 parts in 15 min. Since this method obviates the need for a critical experiment with each new fuel element prior to its insertion into the HFIR, it is a less costly method.

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1.7.19. NONDESTRUCTIVE ASSAY OF SPENT BOILING WATER REACTOR FUEL BY ACTIVE NEUTRON INTERROGATION

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G. L. Ragan **G. G. Slaughter**‡

(Abstract of paper presented at the 22nd Annual Meeting of the Institute of Nuclear Materials Management, San Francisco, California, July 13–15, 1981)

Spent boiling water reactor (BWR) fuel from Dresden I was assayed for total fissile mass, using the active neutron interrogation method. The nondestructive assay (NDA) system used has four Sb-Be sources for interrogation of the fuels; the induced fission neutrons from the fuel are counted by four lead-shielded methane-filled proportional counters biased above the energy of the source neutrons.

Spent fuel rods containing 9 kg of "heavy" metal (~1% fissile) were chopped into 5-cm segments and loaded into three 1-liter cans (~3 kg per can). (Because of mechanical constraints in handling the fuel in the reprocessing test facility, the fuel could not be assayed as entire lengths of fuel rods.) The three cans were assayed in seven combinations of one, two, or three cans, enabling an evaluation of the precision and accuracy of the NDA system for different amounts of fissile material. The fissile mass in each combination was determined by comparing the induced-fission-neutron count rate with the counts obtained from a known standard comprising chopped segments of unirradiated Dresden fuel. These masses were compared to the masses determined by chemical analyses of the spent fuel.

The results from the nondestructive assays agreed with results from the chemical analyses to within 2–3%. Similar agreement was obtained when two combinations of canned spent fuel were used as standards for the nondestructive assays.

The assay of BWR spent fuel served as a test of the NDA system which was developed at the Oak Ridge National Laboratory for the assay of spent liquid metal fast breeder reactor (LMFBR) fuel subassemblies at the head-end of a reprocessing plant. Results of previous experiments and calculations reported earlier using simulated LMFBR fuel subassemblies indicated that the NDA system can measure the fissile masses of spent fuel subassemblies to within an accuracy of 3%. Results of the assays of spent BWR fuel reported herein support this conclusion.

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1.7.20. TWO-DIMENSIONAL IMAGING OF X-RAY EMISSION DENSITY IN THE ISX-B TOKAMAK

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J. L. Dunlap‡ **C. W. Nestor***

[Abstract in *Bull. Am. Phys. Soc.* **26**(7) (September 1981)]

Three arrays with a total of 80 slit-collimated semiconductor soft x-ray detectors view the ISX-B plasma in a poloidal plane. The detector currents, each proportional to a line integral of emitted x-ray power density, are amplified, recorded by a computer data acquisition system, and processed by a generalized Abel inversion method[1] to yield maps of the emission density in the plane. Since contours of constant emission density are assumed to be on magnetic surfaces, the internal magnetic geometry can be seen. We show samples of maps produced in the course of operating with a variety of plasma configurations.

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‡Fusion Energy Division.

1. A. P. Navarro, V. K. Paré, and J. L. Dunlap, "Two-Dimensional Spatial Distribution of Volume Emission from Line Integral Data," submitted to *Rev. Sci. Instrum.*

1.7.21. MHD INSTABILITY STUDIES IN ISX-B

V. K. Paré

A. P. Navarro* **J. D. Bell**‡

J. L. Dunlap† **W. R. Wing**†

[Abstract in *Bull. Am. Phys. Soc.* **25**(8), 976 (1980)]

Signals to Mirnov loop and to collimated soft x-ray diagnostics change gradually with injected beam power (or equivalently, plasma beta). Levels up to approximately 0.5–0.7 MW enhance the amplitude and period (up to 12 ms) of the sawtoothing behavior. Near-maximum power levels (1.7 MW) also give rise to a relaxation oscillation. Primitive features of it (inside/outside $m = 0$, and odd-m precursor oscillation) are like those of the classical sawtooth, but the period (1–2 ms) is much reduced. Intermediate powers result in prolonged, even continuous, activity at the "precursor" frequency (10–20 kHz). Mirnov signals couple to the precursor activity even at low injected power. They exhibit a dominant $m = 2$, $n = 1$ symmetry and apparent poloidal propagation in the ion diamagnetic direction. The latter implies that these modes are Doppler shifted due to beam driven (toroidal) plasma rotation. The extent to which balooning modes are involved is not clear.

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[†]Fusion Energy Division.
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1.7.22. HIGH BETA STUDIES IN THE ISX-B TOKAMAK

G. H. Neilson et al.[*,†]   V. K. Paré

(Abstract of paper presented at the 1982 International Conference on Plasma Physics, Goteborg, Sweden, June 9–15, 1982)

Experimental results from the ISX-B tokamak (major radius $R_o = 0.93$ m. minor radius $a = 0.26$ m, plasma current $I_P = 230$ kA, elongation $\kappa = 1.1 - 1.6$, toroidal field $B_\phi \leq 1.5$ T, density $\bar{n}_e \leq 1.1 \times 10^{20}$ m^{-3}, neutral beam power $P_b \leq 2.5$ MW) at volume-averaged beta ($\langle\beta\rangle$) values up to 2.5% are described. Two aspects of these studies are presented: (1) empirical scaling of beta and of confinement time, and (2) MHD equilibrium analysis of ISX-B plasmas. The main points which are made are, respectively: (1) Global confinement time τ_E^* exhibits a strong positive dependence on plasma current ($I_p^{3/2}$), a negative dependence on beam power ($P_b^{-2/3}$), little or no dependence on $\langle\beta\rangle$ or density, and no correlation with variations in the observed ($m = 1$; $n = 1$ dominated) MHD activity. (2) Profile analysis is coupled with an MHD equilibrium solver to obtain a model of the plasma consistent with profile, magnetic probe, and soft x-ray data, and with the boundary conditions imposed by the poloidal coil currents.

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[*]Fusion Energy Division.

[†]Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.23. HIGH BETA RESULTS IN ISX-B WITH INTENSE NEUTRAL BEAM INJECTION

P. H. Edmonds et al.[*,†]   V. K. Paré

[Abstract of invited paper in *Heating in Toroidal Plasmas*, Vol. I, pp. 3–14 (1982)]

Experiments on the ISX-B device show a deterioration in confinement at high beam power. In particular the electron energy confinement time falls catastrophically with increasing beam power. The maximum volume averaged beta values achieved are <2.5%; this is much less than would be predicted by extrapolating the low power data. Elongation has not been observed to have any significant effect on the maximum attainable beta, perhaps due to the limited range of both internal and external elongation.

There are two likely candidates for the loss of confinement. The phenomena may be caused by the gradual onset of resistive MHD pressure driven modes producing deteriorating confinement through fluctuations in the poloidal magnetic field. Alternatively the phenomena may be specific to the method of heating, neutral injection, being caused, for example, by plasma rotation, where the rotation speed approaches the ion thermal velocity. Experiments are in progress to investigate both of these possibilities.

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[*]Fusion Energy Division.

[†]Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.24. MAGNETOHYDRODYNAMIC INSTABILITY WITH NEUTRAL-BEAM HEATING IN THE ISX-B TOKAMAK

J. L. Dunlap et al.*,† V. K. Paré

[Abstract of *Phys. Rev. Lett.* 48(8) (February 22, 1982)]

This paper describes observations of magnetohydrodynamic instability with neutral-beam heating in the ISX-B tokamak and the theory specifically developed to support these experiments. The observed magnetohydrodynamic activity is explained by the resistive model presented but is not responsible for the observed degradation of confinement. Increasingly important $n > 1$ pressure-driven modes are predicted by the theory for the higher experimental β_p values, but there is no experimental verification of their presence.

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†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.25. TWO-DIMENSIONAL SPATIAL DISTRIBUTION OF VOLUME EMISSION FROM LINE INTEGRAL DATA

A. P. Navarro*
V. K. Paré J. L. Dunlap†

[Abstract of *Rev. Sci. Instrum.* 52(11) (November 1981)]

A method is presented for determining the two-dimensional (2-D) spatial distribution of the volume emission intensity $[E = f(r,\theta)]$ in a plasma from line integral data obtained by arrays of collimated detectors looking through the plasma along different chords. Subject to the assumption that E can be represented by $\sum_m [f_m(r)\cos m\theta + g_m(r)\sin m\theta]$ with only a few terms m, expressions are developed that separate harmonic contributions to the line integrals, a numerical procedure is outlined for inversion of these contributions to obtain the factors $f_m(r)$ and $g_m(r)$, and the formulas that result are presented for $m = 0$ through $m = 5$. This method was developed for application to data from arrays of soft x-ray detectors on the Impurity Study Experiment (ISX-B) tokamak. One use would be comparison of the soft x-ray volume emission contours with other measures of plasma shape. Simulations of these shape determinations were performed, using as input, a pressure profile representing a calculated MHD equilibrium for a D-shaped plasma in ISX-B. The results of these simulations are shown and the determination of the poloidal structure of MHD perturbations in the plasma is also illustrated.

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1.7.26. LERFCM—A COMPUTER CODE FOR SPATIAL RECONSTRUCTION OF VOLUME EMISSION FROM CHORD MEASUREMENTS IN PLASMAS

A. P. Navarro*
V. K. Paré J. L. Dunlap†

(Abstract of ORNL/TM-7499, January 1981)

Local Emissivity Reconstruction From Chord Measurements (LERFCM) is a package of computer programs used to determine the two-dimensional spatial distribution of the emission intensity of radiation in a plasma from line integral data, which represents signals from arrays of collimated detectors looking through the plasma along different chords in a plane. The method requires data from only a few detector arrays and assumes that the emission distribution in the plane of observation has a smooth angular dependence that can be represented by a few low-order harmonics. The intended application is a reconstruction of plasma shape and MHD instabilities, using data from arrays of soft x-ray detectors on the Impurity Study Experiment Tokamak.

In another paper by Navarro, Paré, and Dunlap entitled "Two-Dimensional Spatial Distribution of Volume Emission From Line Integral Data," the algorithms are derived in detail, and the results of simulation tests of the LERFCM package are shown. In this report the algorithms are summarized, and the functions and relationships of the programs are described, with the aid of a flow chart. The programs that were written specifically for the reconstruction calculations are listed fully.

The package is written in FORTRAN and runs on a Digital Equipment Corporation PDP-10 computer. Modification of input/output statements would, in most cases, be sufficient to enable the programs to be run on another computer; however, the user would have to add the graphics capability appropriate to the facilities available to him.

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1.7.27. HIGH-BETA INJECTION EXPERIMENTS ON THE ISX-B TOKAMAK

D. W. Swain et al*,†
J. T. Mihalczo V. K. Paré

[Abstract of *Nucl. Fusion* 21(11), 1409 (1981)]

Neutral-beam injection of up to 2.5 MW into plasmas in the ISX-B tokamak (R_o = 0.93 m, a = 0.27 m, B_T = 0.9 − 1.5 T, I_p = 70 − 210 kA, $\bar{n}_e$ = 2.5 − 10 × 10^{13} cm^{-3}) has created plasmas with volume-averaged beta of up to ~2.5%, peak beta values of up to ~9%, and root-mean-square beta values of up to ~3.5%. Energy confinement time is observed to decrease by about a factor of two as beam power goes from 0 to 2.5 MW; the decrease is caused predominantly by the electron confinement time falling below the predictions of 'Alcator scaling' by a factor of 3–4 at high beam power. An empirical relationship of the form $\beta_p I_p^{1/2} = f(P_b)$ fits our measurements over a wide range of plasma parameters. The function $f(P_b)$, where P_b is the beam power, is linear for $P_b \leqslant 1.2$ MW but tends to saturate for 1.2 MW $\leqslant P_b \leqslant 2.5$ MW. Although the equilibria attained in ISX-B are predicted to be above the threshold for the ideal magnetohydrodynamic (MHD) ballooning instability, no evidence of these modes is observed.

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†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.28. HIGH BETA STUDIES WITH BEAM-HEATED, NONCIRCULAR PLASMAS IN ISX-B

E. A. Lazarus et al.*,†
J. T. Mihalczo V. K. Paré

(Abstract in *Proc. 10th European Conference on Controlled Fusion and Plasma Physics, Moscow, September 14–19, 1981; Vol. 1, Contributed Papers*, p. A-4)

We have expanded the high β experiment in ISX-B to include elongated, moderate q plasmas and developed profile analysis methods consistent with both MHD equilibrium theory and details of other supporting measurements.

The major results of high beta experiments in *circular* discharges (with elongation factor <1.2) are as follows: (a) with injection powers up to 2.5 MW, we have achieved β ~2.5%; (b) β increases more slowly with beam power above ~1.5 MW; and (c) the saturation results from a degradation of energy confinement, primarily due to decreasing electron energy confinement time as beam power increases.

Measurements to date with elongation factor ≃1.5 have not resulted in higher β values than those achieved in circular plasmas; however, we are not prepared to conclude that further experiments cannot demonstrate improvement due to elongation.

*Fusion Energy Division.
†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.29. INFLUENCE OF NEUTRAL-BEAM INJECTION ON IMPURITY TRANSPORT IN THE ISX-B TOKAMAK

R. C. Isler et al.*,† V. K. Paré

[Abstract of *Phys. Rev. Lett.* 47(9) (August 31, 1981)]

Observations of radiation from iron and from argon used as a test gas indicate that co-injection inhibits impurity accumulation in the interior of ISX-B (impurity study experiment) tokamak

discharges, but counter-injection enhances accumulation. These results agree qualitatively with recent theoretical calculations.

*Fusion Energy Division.

†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.30. HIGH-β PROFILE ANALYSIS OF ISX-B PLASMAS

R. M. Wieland et al.*,† V. K. Paré

[Abstract in *Bull. Am. Phys. Soc.* 26(7) (September 1981)]

The time-independent analysis of high-β ISX-B plasmas using Thomson electron scattering profiles and magnetic loop data has been augmented by the addition of self-consistent models for both MHD equilibrium and ion power balance. An interior flux surface geometry is obtained from a fixed boundary "moments" solution of the Grad-Shafranov equation. In this model, the toroidal current profile is functionally parameterized so as to yield a q profile which matches the position of the $q = 1$ surface as deduced from chordal soft x-ray measurements. The ion component of the required pressure profile comes from a detailed treatment of the ion power balance using a neoclassical χ_i multiplier to yield a central T_i in agreement with ion temperature diagnostics. Results will be presented.

*Fusion Energy Division.

†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.31. ENERGY CONFINEMENT IN BEAM-HEATED ISX-B PLASMAS

M. Murakami et al.*,†
J. T. Mihalczo V. K. Paré

[Abstract in *Bull. Am. Phys. Soc.* 26(7) (September 1981)]

Energy confinement in high power ($P_b \lesssim$ 3 MW) beam-heated discharges with circular [$\kappa(a) \sim 1.1$] and elongated [$\kappa(a) \lesssim 1.6$] cross sections is examined. Internal magnetic geometries are determined by a data analysis code (ZORNOC), consistently with MHD equilibrium theory and details of diagnostic measurements [e.g., $T_e(R,z)$, $n_e(R,z)$, $\int n_e dz$, etc.]. Initial results from elongated discharges, with $q_\psi \gtrsim 3$ and $P_b \leq 2$ MW, showed little difference in energy confinement from those of circular discharges. However, because of the internal elongation [$\kappa(a/2) \lesssim 1.3$] is rather modest, the expected geometrical gain is small $\lesssim$15%) and difficult to detect. Operation at lower q_ψ and higher P_b is expected to increase internal elongation and to provide more definitive information concerning improvement of confinement with elongation.

*Fusion Energy Division.

†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.32. MHD ACTIVITY IN THE ISX-B TOKAMAK

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V. K. Paré E. T. Blair‡
J. D. Bell‡ W. R. Wing*

[Abstract in *Bull. Am. Phys. Soc.* 26(7) (September 1981)]

We continue to investigate the MHD instabilities in high β neutral beam heated discharges, using Mirnov coils and collimated soft x-ray detectors. The latter have been augmented recently by the addition of three arrays mounted in the same poloidal section. High n ballooning instability is not observed. We show that the interior of the plasma is dominated by $m = 1$ instability, and that the poloidal field fluctuations at the edge are predominantly $m/n = 2/1$. These and other more detailed features agree with a theoretical mode that is a high β_p distortion of the $m/n = 1/1$. We also describe variations of instability behavior as a function of plasma parameters (single parameter scans and shaped vs nonshaped cross sections) and speculate on the role the instability may play in the response of β to increasing neutral injection power.

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1.7.33. HIGH BETA AND SHAPING EXPERIMENTS IN ISX-B

G. H. Neilson et al.*,† V. K. Paré

[Abstract in *Bull. Am. Phys. Soc.* 26(7) (September 1981)]

High beta experiments in ISX-B have been extended to study the effects of plasma cross section. The elongation is variable over the range 1.1–1.65 by means of poloidal winding reconnections, plus a small quadrupole-like field for fine control. Axisymmetric vertical displacements are stabilized by a feedback-controlled radial field, up to $\kappa \simeq 1.65$. The external shape is determined from poloidal magnetic field measurements and the coil currents, and the internal geometry from a self-consistent analysis which uses measured profiles, rational surfaces, and shape, together with MHD equilibrium theory. No shape-dependent enhancement in $\langle \beta \rangle$ has yet been observed; this might be explained by a lack of significant internal elongation: for $q_\psi \simeq 3$ and $\kappa = 1.55$ the inner surfaces' elongation is only 1.3.

*Fusion Energy Division.

†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

1.7.34. HIGH BETA STUDIES ON ISX-B WITH NEUTRAL BEAM INJECTION

J. Sheffield et al.*,†
J. T. Mihalczo V. K. Paré

(Abstract of paper presented at 2nd Joint Grenoble-Varenna International Symposium on Heating in Toroidal Plasmas, Como, Italy, September 3–12, 1980)

Injection of H^0 into D^+ plasmas with beam power P_b of up to 1.7 MW has produced rms betas of ~4%, volume-averaged betas of ~3%, and central betas of ~10% in the ISX-B tokamak. Although theoretical calculations indicate that the observed equilibria may be unstable to ballooning modes, no catastrophic loss of confinement has been observed, and beta continues to increase with injection power. In these beam-dominated, high-beta discharges the electron and ion energy confinement times are still similar to those obtained with ohmic heating: ion energy confinement is neoclassical within a factor of ~2, and electron energy confinement follows the usual Alcator scaling. In high-power injection discharges the character of the magnetohydrodynamic (MHD) behavior changes, the particle confinement time decreases, and the inward impurity transport appears to be inhibited. These effects, however, may not be linked directly to beta.

*Fusion Energy Division.

†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

Section 2

RESEARCH INSTRUMENTS

2.0. OVERVIEW

C. D. Martin

The Research Instrument Section (RIS) is responsible for instrument design, development, and fabrication in support of ORNL divisions and programs, national fusion and fission energy programs, and the national defense. The Section is generally considered to be the product-oriented arm of the Division because it is concerned with instrument design and the development and fabrication of production prototypes. Emphasis is on development of radiation detection and measurement instruments that are not commercially available.

Because of the particular technical expertise of its staff, the Research Instrument Section is frequently asked to consult on the purchase of such items as computers, pulse height analyzers, and radiation detectors for others in UCC-ND, in DOE, and even in other government agencies. This year the Section has been asked to act as consultants to the Federal Emergency Management Agency in an assessment of the capability of American industry to consistently produce high quality Geiger-Mueller tubes. The Section is also providing technical support to the UCC-ND Engineering Division in the implementation of a master plan for DOE radio communications in the Oak Ridge area.

The Section is organized into six groups: Circuits and Detector Systems; Computer Systems; Product Design, Fabrication, and Drafting; Mechanical Development; Special Electronics; and Accelerator Physics. However, because most instruments are developed by teams consisting of members from more than one group, little evidence of group affiliation will be found in the following discussion.

U.S. Navy-ORNL RADIAC Program

The most significant product from the Research Instrument Section during this reporting period is the new line of Navy-ORNL RADIAC (Radiation Detection, Indication, and Computation) instruments. These instruments were developed as part of an ongoing program to investigate detector systems, to develop instrumentation methodologies, and to design engineering models of advanced radiation measurement instruments in support of U.S. Navy nuclear propulsion, weapons, and disaster preparedness programs. Projects for the period include a low-range beta-gamma survey instrument, a remote calibrator for low-range survey instruments, a single-switch auto-ranging low-range beta-gamma survey instrument, a single switch high-range beta-gamma survey instrument, an aqueous tritium measurement system, and a tritium-in-air monitor.

The key to the development of single-switch auto-ranging survey instruments is the use of a liquid crystal display to replace the conventional electro-mechanical D'Arsonval meter that, together with a range-changing switch, indicated the radiation field strength on previous instruments. In the present demonstration model of the low-range instruments (Fig. 2.1), a microprocessor provides GM tube selection, performs statistical averaging at low count rates, and displays the calculated field strength in analog form on a liquid crystal replica of the face of a standard meter movement.

ORNL Photo No. 6078-82

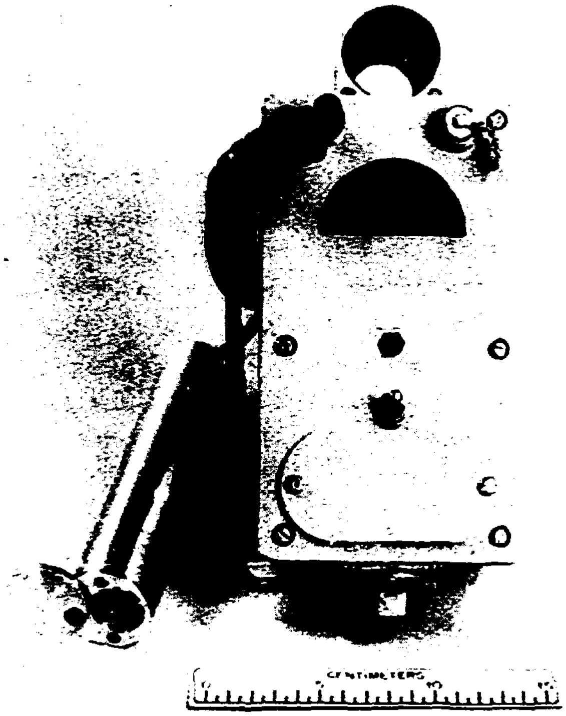

Fig. 2.1. Auto-ranging beta-gamma survey instrument.

Auto-ranging scale indication is used to avoid ambiguities. In another single-switch RADIAC which uses an ionization chamber as the radiation detector, a custom LSI circuit replaces the microprocessor and performs the necessary functions of data processing and data encoding for the liquid crystal display. The auto-ranging RADIAC instrument shown in Fig. 2.1 was one of the ORNL nominations for an IR•100 award.

Neutron Detectors

The Research Instrument Section continues to help develop better neutron detectors. In particular, this Section engineered and fabricated the ultrahigh-sensitivity fission counter developed in the Reactor Systems Section. This counter has a sensitivity of greater than 40 cps/nv and will be used to replace BF_3 detectors which are often trouble-prone and sensitive to gamma-ray background. For advanced reactors, the Section is developing an experimental in-vessel fission counter and in-vessel electronics, both of which must operate in sodium at 230°C. In-vessel electronics will transmit the data ultrasonically to an ex-vessel receiving station. In-vessel monitoring is expected to give a better measure of subcriticality than the ex-vessel detectors currently planned.

A 3¾-carat diamond was one of the more unusual neutron detectors used by members of the Research Instrument Section. This detector was used to identify, by pulse shape discrimination, the various high energy neutron interactions in diamonds. The experiment was only partially successful.

Circuit Development

Electronic circuits, both analog and digital, are essential to modern instrumentation. To maintain state-of-the-art support for ORNL in this area, the Section is in the process of obtaining a computer-aided design facility. This facility will improve the productivity of the discrete circuits and hybrid microcircuits produced in-house. To obtain custom LSI circuits, the Section is using so-called "silicon foundries" to implement in-house designs.

Enviromental Monitoring

The Section is deeply involved with the development of detectors and systems for environmental monitoring. Most of the instrumentation is focused on the measurement of radioactive hazards, but a multiplexer was also developed and installed for dissolved-oxygen measurements in a cluster of artifical ponds filled with water from such diverse sources as coal mines and sewage treatment plants.

Position-Sensitive Detectors

Position-sensitive radiation detectors continue to be important to the Section. During the reporting period, the Circuits and Detector Systems Group constructed eight large-area multiwire proportional counters and nine drift chambers for use by others. In addition, they assumed responsibility for the maintenance of the position-sensitive counters in the ORNL Small-Angle Neutron Scattering Facility. This assumption of responsibility required the assembling of special equipment to recover the $10,000 worth of ^{3}He gas that must be removed from the counter whenever it is opened.

Stand-Alone Computer Use

The Section continues to be active in interfacing computers to other hardware for control of experiments. During this reporting period, a single-stepping motor controller was designed and built to operate up to 128 motors at the ORNL port at the National Synchrotron Light Source. Also, a new distributed processing and experiment control facility was designed and built for the neutron-scattering facility at the High Flux Isotope Reactor. Both of these projects use PDP-11 computers, but personal computers are being used more and more to control experiments and to collect data. It is expected that this trend will continue as the capabilities of these machines increase.

Plant Security

The Section is involved in two aspects of plant security. One aspect centers on detecting unauthorized entry and the other on the new UCC-ND badge system.

Engineering Support for Fusion Energy Division

Engineering support for the Fusion Energy Division was continued. During this reporting period the focus was on the Elmo Bumpy Torus (EBT) experiments. Several diagnostics were improved or installed, and a second generation of gyrotrons is being installed. When operational these gyrotrons will supply 400 kW of continuous rf power to the EBT.

Engineering Support for Accelerator Physics

Engineering support for accelerator physics also continued. The Radiation Safety System for the Holifield Heavy Ion Research Facility was checked for proper operation with a radioactive check source and turned over to the operating staff as a fully operational system. Further responsibility for the safety system falls to the ORNL Physics Division. In addition, various improvements were made in the Oak Ridge Electron Linear Accelerator, in the Holifield Heavy Ion Research Facility, and in the Oak Ridge Isochronous Cyclotron.

Other

Interspersed and integrated with the major activities are numerous small jobs in response to requests for help from researchers with experimental setups. The requests are generally for nonstandard (not commercially available) items of instrumentation to complete their experimental data collection systems. For the most part, these items are "one-time" developments employing state-of-the-art techniques and devices. A wide range of expertise that is abreast of the explosive growth of the electronics field and new detector developments is maintained to adequately meet these needs.

2.1. NAVY-ORNL RADIAC DEVELOPMENT PROGRAM

The Navy-ORNL RADIAC[1] (Radiation Detection Indication And Computation) development program was initiated in 1976 between the Instrumentation and Controls Division and the Navelex Nucleonics Branch of the U.S. Navy. Its purpose is to investigate detector systems, develop instrumentation methodologies, and design engineering models of advanced radiation measurement instruments and systems in support of Navy nuclear propulsion, weapons, and disaster preparedness programs. Projects for this reporting period include a low range beta-gamma survey instrument, a remote calibrator for the low range survey instrument, an auto-ranging low range beta-gamma survey instrument, an auto-ranging high range beta-gamma survey instrument, an aqueous tritium measurement system, a tritium-in-air monitor, a thermal luminescence dosimeter reader/computer interface, and a universal RADIAC diagnostic analyzer. In addition, staff members involved in the program provide technical consultation to the Nucleonics Branch on radiation measurement and instrument development problems.

The low range beta-gamma survey instrument uses two GM tubes in a single housing to cover five decades of gamma radiation measurement (0.1 to 1000 mR/h full scale) and two decades of beta radiation measurement (0.1 to 10 mR/h). A conventional D'Arsonval meter movement and range changing switch are used for radiation field indication.

Calibrating the low range RADIAC instrument was a problem. It has a separate calibration potentiometer for each of the five ranges, and each potentiometer may have to be adjusted during each calibration on an open-source range. To speed this process, a remote control screwdriver adjustment and range change switch selector were developed for use with the RADIAC during calibration.

Navy testing of previous versions of the low range RADIAC revealed problems with mechanical shock and vibration and with excessive sensitivity to electromagnetic interference. To eliminate these problems, a new probe housing and probe cable have been designed. Five low range RADIACS that incorporate these new designs are to be delivered to the Navy for testing. In addition, documentation sufficiently detailed to enable construction will be made available to the Navy.

The auto-ranging low range beta-gamma survey instrument is an advanced version of the low range RADIAC with the same measurement ranges. The range-changing mechanism and display meter have been replaced with advanced microcircuits, including a low power microprocessor, and a direct reading multidecade analog liquid crystal display with auto-ranging scale indication to avoid ambiguities. The microprocessor provides GM tube selection based on count rate and does statistical averaging at low count rates. The display is all solid state for reliable service and allows range changing and indication without mechanical movements. A demonstration model in a military package has been developed, and a more advanced model is under development for Navy evaluation.

A high range auto-ranging RADIAC is being developed to measure four decades of gamma radiation from 1 to 1000 R/h full scale. An inexpensive, sealed, 100-cc ionization chamber with one standard atmosphere nitrogen and a thin aluminum beta window is designed for containment within the housing of the instrument. A unique thick-film hybrid electrometer covers six decades of input current (10^{-13} to 10^{-8} A). Through the use of log and antilog circuits, the electrometer covers the entire input current range and outputs a frequency proportional to ionization currents from 1 to 100,000 cps. An ORNL-designed custom CMOS gate-array integrated circuit (I.C.) converts this frequency to the necessary signals to drive an analog display type liquid crystal display (LCD) that presents measurement data with automatic scale indication. A triple output, high efficiency dc-dc converter supplies power from two D cells for the electrometer, the display driver I.C., and the ion chamber. The components (hybrid electrometer, driver I.C., and LCD) and subsystems (dc-dc converter and ion chamber) are being characterized over the required operating temperature range, −40 to 60°C. A special heater system is being developed for low temperature (−40 to 0°C) operation of the LCD.

An advanced engineering model fixed/portable tritium-in-air monitor was developed for measuring gaseous tritium concentrations for 1 to 10,000 $\mu Ci/m^3$. The instrument employs a fully gamma compensated spherical flow through ion chamber, air pump, linear electrometers, 4½-digit digital voltmeter, dc-to-dc converter high voltage supply, and battery charger. The instrument provides digital readout of the tritium measurement over the full range without gain switching, and operates from line continuously or on batteries for up to four hours of portable use. Audible and visual alarms are provided. One prototype instrument has been fabricated and demonstrated to the Navy and is being fully documented for continuing development when requested by Navelex.

As part of the development of an aqueous tritium measurement system, a conceptual bench scale model of a room temperature liquid scintillation system was designed, built, and tested. The system consisted of two RCA 4501 photomultiplier tubes, in-house designed amplifiers, discriminators, coincidence gates, and a photomultiplier tube (PMT) balance circuit. A microprocessor was designed into the instrument to collect and process data. The microprocessor also interacts with the operator, minimizing his work load and thereby reducing the chance of possible operator-induced errors. Urine or water samples with tritium concentrations from 1 to 5 $\mu Ci/L$ can be measured to within 20% uncertainty with typical counting times ranging from 40 s to 3 min, depending on the background count rate, the sample's activity, and the degree of quench. Tritium concentrations above 5 $\mu Ci/L$ are measured to within 10% uncertainty with typical counting times varying less than 3 min. Water samples with tritium concentrations of 0.1 $\mu Ci/L$ can be measured to within 20% uncertainty with a typical counting time of 30 min and to within 30% uncertainty with a typical counting time of less than 10 min. The system employs a novel PMT balance circuit that eliminates the necessity of using matched photomultiplier tubes. The system also employs a fast coincidence gate, which reduces the probability of counting extraneous phosphorescence pulses and therefore allows the operator to place the sample directly into the instrument without waiting for a prolonged period of time to elapse.

To help automate thermoluminescence dosimeter reading, an interface was built to enable an Apple II Plus computer to communicate with up to five thermoluminescence dosimeter (TLD) readers. This interface was a modification of the microprocessor-based data logger developed during the last reporting period. A conceptual design of a more advanced interface for up to seven TLD readers was completed and reported to the Navy. Serial number, ship number, TLD number, and dates are entered via the Apple, and data from the TLD reader are accumulated automatically and either logged on disk or transmitted to a larger computer system.

Finally, a study was begun to investigate a universal RADIAC Diagnostic Analyzer for use by RADIAC repair facilities and Navy Tender operations. During this study, which is in its very early stages, we will try to integrate RADIAC repair needs with overall Navy automatic test equipment systems.

Because of the Research Instrument Section's participation in this program, the U.S. Navy has obtained and will continue to obtain advanced radiation measurement instruments at minimum cost and without having to develop its own expertise in this area. *(R. K. Abele, G. W. Allin, M. L. Bauer, H. R. Brashear, W. L. Bryan, R. E. Cooper, J. T. De Lorenzo, M. S. Emery, R. A. Maples, V. C. Miller, J. E. Phelps, D. R. Patek, R. A. Todd, D. E. Smith)*

1. H. R. Brashear et al., *Instrumentation and Controls Division Biennial Progress Report, September 1, 1978, to September 1, 1980,* ORNL-5758, pp. 199–202.

2.2. ADVANCED GAMMA AND X-RAY DETECTOR SYSTEMS

During this reporting period work has continued on advanced detector systems that use both gas-filled and solid-state detectors.

Investigation of the gas scintillation proportional counter and its possible applications were briefly discussed in an earlier report.[1] Project funding ended at that time, but the investigation continued although greatly limited in scope.

Gas scintillation proportional counter detectors are important because the energy resolution of these detectors, if the detectors are properly designed and operated, is approximately a factor of two better than that of conventional proportional counters. Improved resolution is obtained because the counter is operated slightly below that potential at which the electron multiplication process begins, and thus the statistical spread due to that process is eliminated. Instead of detecting the amplitudes of pulses that result from the electron multiplication process, the uv photon pulses resulting from the accelerated-electron-induced excited states are detected and analyzed. This is done by using a uv-sensitive photomultiplier optically coupled to the counter. The output pulse height of the photomultiplier is proportional to the ionization produced by the ionizing photon or charged particle.

The best resolution achieved to date for ^{55}Fe was 9.7% full-width half-maximum. This was obtained by coupling the photomultiplier tube (RCA C-31000-M) directly to the counter, thus eliminating the spectrosil quartz counter window, and repurifying the xenon counter gas several times. However, a resolution of 7.5% has been reported in the literature[2] for a xenon-filled gas scintillation proportional counter of complicated design but having a spectrosil window and operated under special conditions.

Because lack of funds made it necessary to limit this investigation mainly to working with counter parts already available, the spectrasil counter window was used in all subsequent tests. Various electrode arrangements were tried but did not produce any major effect on resolution, although high-voltage breakdown problems were greatly decreased. Of the several different gas mixtures tested, a mixture of xenon + 10% helium improved the resolution by increasing the electron drift velocity (without absorbing the uv), which for xenon is in the 1500-1950 Au region, peaking at about 1750 Au. It was also noted that discharging the counter for short periods improved the resolution, probably because of a change in gas impurities—both in concentration and composition. Some loss in resolution was noted when an EMI-9656 KQR photomultiplier tube was used instead of the RCA C-31000-M tube. Neither tube was optically flat over the entire tube window, the EMI tube being less so than the RCA tube. No optical coupling fluid may be used between the windows because even a very thin film of the fluid would absorb all of the short wavelength uv.

The most difficult problem in achieving good resolution is caused by the degrading effects of gas contaminants even in very small amounts such as a few parts per million. These impurities degrade the resolution because of uv absorption and/or electron attachment. The impurities are present in "pure" xenon and increase due to counter outgassing. A "dirty" vacuum system is another source of counter contamination. On the basis of the gas-purifying procedures tried and the results of counter gas analysis, it appears that impurities other than oxygen and water vapor are the major problem impurities and that these are organic and/or organic decomposition products. Most investigators use either hot calcium purification systems or SAES-gathering systems as an integral part of the detector. These methods remain to be evaluated. *(J. B. Davidson, R. E. Zedler)*

A novel circuit was designed and tested to measure the exposure rate (mR/h) of a gamma radiation field using a solid-state compensated, chlorine-doped cadimum-telluride crystal. Cadmium-telluride is used because its high atomic number allows efficient detection of gamma radiation. The circuit is the second generation of a novel circuit developed earlier in this Division.[3] In the second generation circuit, a potentiometer can be adjusted in order to make the reading of the instrument independent of the gamma energy. The potentiometer setting needed to achieve a flat energy response of the detector system's sensitivity varies from one detector to another. This is because the method of doping of the cadmium-telluride crystals causes the mobilities and lifetimes for the charge carriers to vary from one detector to another. A charge injection system now being developed will allow easy calibration for each detector without the use of a number of gamma sources. The injection scheme will allow rapid measurement of the mobilities and lifetimes of the charge carriers.[4] An injection method based on the use of point contacts was initially tested, but it failed because reasonable current could not be detected due to the high resistivity of the crystal. A second method of charge injection, based on light injection, has been conceptually designed but has not been implemented. *(G. A. Colman, R. J. Fox, R. A. Todd)*

From the above report it is clear that, despite funding reductions, members of the Research Instrument Section, in cooperation with others in the I&C Division, have been able to continue research on the next generation of low-energy X-ray detectors and gamma-ray detectors—the gas scintillator proportional counter and the cadmium-telluride detector.

1. R. E. Zedler and J. B. Davidson, ORNL-5758 (June 30, 1981), p. 165.

2. Peacock et al., *Nucl. Instrum. Methods* **169**, 613–25 (1980).

3. R. J. Fox, D. C. Agouridis, U.S. Patent Document 4,243,885.

4. W. Shockley, *Electrons and Holes in Semiconductors*, D. Van Nostrand Co. Inc., New York (1950).

2.3. NEUTRON DETECTION AND SUBCRITICALITY MEASUREMENTS

Neutron detection is essential for the development and operation of fission and fusion reactors as well as for the safe processing and reprocessing of fuel for reactors. The Research Instrument Section provided mechanical design support and circuit development support for the ultrahigh-sensitivity fission counter developed for the source-range channel of advanced nuclear reactors. In particular, a high-speed dual discriminator for use with the ultrahigh-sensitivity fission counter (see Section 1.4) was designed, constructed, and tested. The front-end circuitry featured MECL III logic devices with an improved latching scheme. Fast NIM logic outputs were provided at front panel connectors for both single-channel analyzer channels and a coincidence output (both TTL and fast NIM). Coincidence output was triggered by pulses in each single-channel analyzer occurring within 100 300 ns of each other to account for the delay line characteristics of the counter. The etched wiring board layout used ground plane and stripline techniques, with all signal lead lengths kept to a minimum. The unit was housed in a three-wide NIM module along with two amplifier filter boards. Front panel controls included upper- and lower-level discriminator settings and an external dc control option for the lower lead discriminator. *(R. A. Todd)*

A significant achievement was the use of empirical springback data to determine the design of the forming die used in making the electrodes for the ultrahigh-sensitivity counter. The first derivative of the curve equation gave the desired as-formed radius at any point. Test samples of the material used, half-hard 3003 aluminum alloy, were formed around different mandrel radii and allowed to spring back to establish as-formed radii. The ratios of mandrel radii to as-formed radii were then used to modify the curve equation to produce the data points for machining the forming die on a numerical control milling machine. The rubber-die-formed electrodes produced as a result of the first attempt exceeded expectations regarding accuracy and consistency. *(G. W. Allin)*

Special coaxial transmission lines with stainless steel exterior, magnesium oxide dielectric, and a characteristic impedence of 25 Ω were fabricated at ORNL for the ultrahigh-sensitivity counter. *(M. M. Chiles)*

The ultrahigh-sensitivity fission counter is designed to operate ex-vessel at ambient temperature. However, an experimental wireless in-vessel neutron monitor[1-3] is being developed to measure the subcriticality of an advanced nuclear reactor as the core is loaded. In-vessel monitoring is expected to give a better measurement of subcriticality. The monitor will consist of a fission counter and an electronics package that will transmit the neutron data to the base station outside the reactor by means of ultrasonic waves. The environment for the detector and electronics will be liquid sodium at a temperature of ~220°C, with negligible gamma or neutron radiation. The overall electronics package is functionally organized in a hybrid thick-film module and includes (1) an amplifier-filter-discriminator, (2) a dc-dc converter, (3) a 2-MHz oscillator, and (4) a gated 2-MHz driver. The transmitter circuitry will be assembled in discrete form to permit more effective heat sinking of its transistors.

The fission counter for this wireless monitor will be a Reuter Stokes RSN10A, modified for low interelectrode capacitance and for operation at a low bias voltage. A special high-purity electrode coating of ^{235}U and a gas mixture of Ar -0.01% CO_2 (at $\sim$133 kPa absolute pressure) were selected to ensure adequate performance at the much-lower-than-normal counter bias potential of 10 V. Most fission counters require more than 200 V for satisfactory operation.

Experience with the demonstration unit revealed that inadequate bakeout can cause a severe degradation of the neutron pulse height at 230°C. A four-day bakeout under vacuum at 270°C prior to filling of the counter is needed to guarantee no outgassing of its internal structure at elevated temperatures. However, we see no fundamental limitation after tests at 230°C for >2000 h.[2,3] The neutron sensitivity is $\sim$1 count/s-nv, and the potential data transmission rate is $\sim 10^4$ counts/s. Some possible problem areas were pointed out by this demonstration test. These problems related to the instability of solid-aluminum electrolytics and failures of aluminum wire bonds to the gold metallization of hybrid thick-film circuits when conducting currents >100 mA. They have been resolved and have not seriously delayed the assembly of the prototype.

Another recent accomplishment was the preparation of a technical specification for the primary power source, an 8-V, 1.6-W radioisotopic generator with a plutonium ($^{238}Pu_2O_3$) heat source. This power source will supply power for the electronics located with the detector in the reactor. Requests for quotes were issued and one quote was received from Teledyne Energy Corporation. However, the purchase of the unit will be delayed until the monitor is needed for initial core-loading tests.

Work on the signal-transmitting package for the wireless monitor and associated electronics recently reached the stage of final assembly for testing. Design and bonding techniques have resulted in transducers with good efficiency and control of resonant frequency to within $\pm 0.5\%$ at 230°C. A gated, 2-MHz Class B power amplifier drives the transducer and can supply 2 W with about 50% efficiency. With 30 μs-wide gating pulses from the system pulse discriminator, the power amplifier bias requirement is typically 15 mA (at 8.5 V) when the average transmission rate is 1000 cps. The power amplifier input is a 2-MHz sine wave supplied by a crystal-controlled oscillator whose crystal is matched to that of the transducer in order to obtain close frequency matching and tracking. The structure and character of the ultrasonic beams projected by the transducers and a conical reflector were studied with a Schlieren optical system.[4] The detector is designed to be contained within a dummy fuel element and positioned within the reactor core. No restrictions exist other than those imposed by the subcriticality measurement, which confine the detector's core position.

The completion of the wireless monitor is scheduled by the end of FY 1982, and a qualification test for a minimum of 24 h at 230°C is scheduled. Prior to this, all electronic circuits including the ultrasonic transducer will receive a "burn-in" of one week at 230°C. The qualification test will be made with the entire unit submerged in an oil bath to permit the extraction of ultrasonic signals. Neutron sources external to the oil bath vessel will produce the input to the monitor. Power to the monitor will be obtained from an external 8-V supply designed to simulate the Teledyne radioisotopic generator. Future testing will include the undersodium testing of the transmitting head at 230°C to verify both its mechanical and electrical capabilities. *(G. W. Allin, T. V. Blalock,* M. M. Chiles, J. T. DeLorenzo, E. J. Kennedy,* J. T. Mihalczo, V. C. Miller, A. C. Morris, J. M. Rochelle,* K. H. Valentine)*

For the next generation of advanced reactors, development work on a high-temperature fission chamber was started. The goal is to have a chamber that will operate at 550°C and survive temperature excursions to 650°C. Because materials and counting gas compatibility are of major concern, a prototype chamber, FC-3-FG, is being constructed to test the compatibility of these constituents. The fabrication of chamber parts is virtually complete with the exception of appropriate cables and end seals. When these parts are on hand the final assembly will commence and counting gas mixtures will be tested. One of the prime candidates for the counting gas is 20% CF_4 and 80% Ar. This mixture has been tested at lower temperatures by Christophorou et al., of ORNL with good success. *(A. C. Morris, R. L. Shepard, K. H. Valentine)*

In support of continued development of external-source-driven subcriticality measurements, subcriticality experiments were performed in uranyl fluoride solutions at LASL, in mixed plutonium-uranium nitrate solutions at Battelle Northwest Laboratory (for CFRP), and with HFIR reactor cores submerged in water for the Reactor Operations Division at ORNL. Special ^{252}Cf fission chambers[5] were fabricated for the neutron driving source, and several other special detectors, including Li glass scintillation and commercial ^{3}He proportional chambers, were procured and supplied for these experiments. *(M. M. Chiles, J. T. Mihalczo)*

To enable measurements of neutron cross sections, a fission counter with .005-in.-thick aluminum windows and multiple plates was fabricated. Five plates of .0005-in.-thick aluminum were coated with the following isotopes: ^{252}Cf, ^{235}U, ^{241}Pu, ^{10}B, and ^{241}Pu. Efforts were made to eliminate crosstalk between the five plates. A second counter, containing only two plates and ^{10}B coated foils, was fabricated for the same experiments. *(W. T. Clay, F. E. Gillespie, V. C. Miller)*

A rugged, spherical-electrode fission chamber was designed to serve as a standard for the measurement of the average number of neutrons per fission. The thickness of the stainless steel wall was increased, and the welding technique was improved by the use of properly designed fixtures to control the heat generated during the welding process. Final assembly and testing is awaiting the deposition of 0.17 μg of ^{252}Cf onto the center electrode, which is being achieved by the transfer method from a mother source. *(M. M. Chiles, F. E. Gillespie)*

The present scintillation-type burst detector installed at the Health Physics Research Reactor has limited sensitivity when the reactor is operated below its designed yield of 10^{16} fissions per burst. Some proposed experiments require fission bursts as low as 10^{13} fissions, but, when the reactor is operated at these lower activity rates, the rise and fall of the fission neutron flux is much slower. A detector that will exhibit a time profile of the burst over several decades of activity is desired. A ^{3}He gas ionization chamber was therefore operated with an operational amplifier to drive an oscilloscope display of the neutron burst, and preliminary tests indicate that this monitor will perform over several decades. During the lower activity burst the increase of the neutron flux was slow enough (about 150 ms) to activate the scram system before the burst was completed. Tests have been discontinued until the scram circuit can be modified to allow bursts at 10^{13} fissions. *(M. M. Chiles, J. T. DeLorenzo, L. Holland†)*

A small-scale investigation of the use of a diamond scintillation detector for high energy neutrons was conducted. A triangular, commercial quality 3¾-carat diamond was suspended on a fine nickel wire at the geometric center of a cross-shaped beam tube at the ORELA Facility. Two opposing photomultiplier tubes were positioned to detect the scintillation light pulses and amplify the electron signal. Neutrons in the 5- to 20-MeV range were used in the investigation, and it was demonstrated that the diamond material will scintillate due to the neutron-carbon interaction. Further tests will be required to determine if this type of detector can discriminate between the neutron energies within the test energy range. *(A. C. Morris, J. H. Todd)*

A new rc current pulse integrator was designed for pulse-mode noise analysis in subcriticality measurements. The new model has several major improvements over old models.[6] The input of the integrator is designed to accept a standard fast NIM logic pulse from a discriminator and has a 50-Ω characteristic impedance; hence, it does not need to be close-coupled to the discriminator. The output current pulse width is virtually independent of the input signal shape. Current amplitude of 1.75 mA to 28 mA can be selected by the user from the front panel. *(J. T. DeLorenzo, M. S. Emery, W. T. King)*

A modular timing filter amplifier for pulse shape discrimination of neutron and gamma pulses was designed and constructed using RLC passive Nowlin filter networks between wideband, linear amplifier stages. Several filter networks were built for bipolar pulse durations ranging from 75 to 600 ns to experimentally determine zero crossing time difference for each set of filters. Computer simulation of the system's transfer function indicated less zero crossing time difference than was experimentally observed using published decay time constants for NE213 scintillators. The amplifiers have a gain of 7.4 and a bandwidth of 32 MHz with no difference in large and small signal rise time. The amplifiers will drive ±10 V unterminated or ±5 V into 50 Ω, thus providing at least 200:1 dynamic range at the amplifier output. For the 150-ns filter, the zero crossing time difference was >1.0 ns; thus, this instrument, along with a fast zero-crossing discriminator and a flash ADC to divide the spectrum into 16 groups, is capable of pulse shape discrimination at counting rates in excess of 5 MHz over a wide range of neutron energies. *(R. A. Todd, G. A. Colman)*

By designing and constructing numerous improved neutron detectors and associated electronic equipment, Research Instrument Section personnel are making a major contribution to the development and operation of fission and fusion reactors. In particular, mechanical design support and circuit development support have been provided for the ultrahigh-sensitivity fission counter for nuclear reactors. For more advanced reactors, an experimetal in-vessel detector and in-vessel electronics are being developed, both of which will operate at 220°C. For still more advanced reactors, another in-vessel neutron detector now under development will operate at 550°C. In addition, various specialized neutron detectors have been built to support reactor development. One special detector was built to measure neutron cross sections, and another was built as a standard for measurement of the number of neutrons produced per fission. Finally, a preliminary evaluation was made of diamond as a neutron detector, and two special electronic circuits were designed for use with neutron detectors. One of these circuits is a new rc current pulse integrator for subcriticality measurements. The other circuit is a new timing-filter amplifier for the identification, by means of pulse shape discrimination, of neutron-caused pulses in a background of gamma-ray-caused pulses. This timing-filter amplifier will be used with neutron detectors in fusion energy experiments.

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*Department of Electrical Engineering, The University of Tennessee, Knoxville.

†Operations, Research and Development.

1. FY 1980 work on this project is described in greater detail in an ORNL letter report entitled "Design and Demonstration of a Remote Signal Transmission Neutron Monitor" and in The University of Tennessee reports ORNL/Sub-7685/7, ORNL/-Sub-7685/12, and ORNL/Sub-7685/11.

2. K. H. Valentine et al., *High Temperature Electronics for a Wireless, Initial Core Loading Neutron Monitor*, ORNL-5759 (June 1981).

3. J. T. De Lorenzo et al., "Wireless, In-Vessel Neutron Monitor for Initial Core-Loading at Advanced Breeder Reactors," Paper presented at IEEE Conf. on High Temp. Electronics, Tucson, Arizona, March 25–27, 1981.

4. Schlieren equipment is the property of the ORNL Metals and Ceramics Division and the assistance of Mr. William Simpson is acknowledged.

5. "Double Containment ^{252}Cf Fission Chamber," *Instrumentation and Controls Division Progress Report for Period Ending September 1, 1971*, ORNL-4734, p. 15.

6. J. T. De Lorenzo, N. J. Ackermann, Jr., and R. C. Kryter, *RC Current Pulse Integrator for Pulse-Mode-Type Analysis in LMFBR Subcriticality Studies*, ORNL-4734 (September 1971).

2.3.1. FISSION CHAMBER ASSEMBLY AND MATCHING PREAMPLIFIER

J. T. DeLorenzo

(Abstract of ORNL/TM-8212, to be published)

The preamplifier for the wide-range counting channel at the High Flux Isotope Reactor was redesigned. This new design incorporates current state-of-the-art components as replacements for the obsolete components of the earlier design. To accommodate the new design, a test and maintenance manual was prepared for use by qualified technicians and field engineers. The test procedures described herein do not require special test equipment and have been successfully applied to the prototype design.

2.3.2. WIRELESS, IN-VESSEL NEUTRON MONITOR FOR INITIAL CORE-LOADING OF ADVANCED BREEDER REACTORS

T. V. Blalock E. J. Kennedy*
M. M. Chiles J. M. Rochelle*
J. T. DeLorenzo K. H. Valentine

(Abstract of paper presented at IEEE Conference on High Temperature Electronics, Tucson, Arizona, March 25–27, 1981)

An experimental wireless, in-vessel neutron monitor is being developed to measure the reactivity of an advanced breeder reactor as the core is loaded for the first time to preclude an accidental criticality incident. The environment is liquid sodium at a temperature of ~220°C, with negligible gamma or neutron radiation. With ultrasonic transmission of neutron data, no fundamental limitation has been observed after tests at 230°C for >2000 h. The neutron sensitivity was ~1 count/s-nv, and the potential data transmission rate was $\sim 10^4$ counts/s.

*Department of Electrical Engineering, The University of Tennessee, Knoxville, Tennessee.

2.4. CIRCUIT DEVELOPMENT

Electronic circuits, both analog and digital, are essential to modern instrumentation. To maintain state-of-the-art support for ORNL in this area, a computer-aided design facility will be obtained for circuit development. I&C Division requirements for computer-aided design (CAD) of printed circuit boards, hybrid microcircuits, and instrument packaging were established, and possible implementations were investigated. A turnkey system was determined to be the lowest risk method to implement CAD in the I&C Division, and Instrument Specification #273 has been written for the competitive acquisition of a two-station CAD system. *(W. L. Bryan)*

The I&C Division is already using computer-aided engineering in the form of the circuit simulation computer program SPICE. In particular, the SPICE computer program was used to model and analyze the Machine Drive Package (MDP) driver circuit that powers the three-phase hysteresis motors used in centrifuge enrichment facilities at K-25. Component failures in the MDP circuit have been higher than predicted and primarily result in "blown" power transistors in the output drivers and the switching regulator circuit. A computer analysis of the time response of the system determined the instantaneous operating points of the output devices, and these points were plotted on graphs with the Safe Operating Area (SOA) curves. It was found that the driver circuit connected to the −dc rail in a common-emitter configuration did not have the drive capability of the common-collector configuration that connected to the +dc rail. It was recommended that the −dc driver level-shifter circuitry be redesigned to provide more current drive to the output. In principle, this difficulty could have been discovered by careful voltage and current measurements on the actual equipment which had already been built. However, such techniques failed to locate the trouble, and it was finally necessary to use computer simulation to determine the cause of the frequent transistor failures. *(R. A. Todd)*

Circuit designs continue to be implemented by either standard printed circuit (P.C.) or hybrid microcircuit techniques. The hybrid microcircuit technique is used whenever there is a need for small size, a wide bandwidth, or a hermetic seal to protect the circuit against the environment. An example of a circuit design in which implementation benefited from hybrid fabrication is a very sensitive, low-noise, charge-sensitive amplifier that was needed for applications where space is very limited. To obtain small size the amplifier was made as a thick-film hybrid circuit, and all components were attached to the substrate by solder reflow, ensuring a reliable and simple connection. Typical performance is 2.5×10^{12} V per Coulomb and equivalent noise charge of 2×10^{18} Coulomb per square root Hertz. Rise time is 60 ns and fall time is 0.8 ms. The charge-sensitive stage may be followed by a voltage gain stage, with the degree of gain specified by the customer. The entire unit (charge sensitive and voltage gain stages) fits on a substrate measuring only 1 × 0.65 in. *(R. E. Cooper, M. S. Emery)*

Through the use of computer-aided circuit simulation and the hybrid microcircuit fabrication facility, the Research Instrument Section makes state-of-the-art circuit development available to ORNL researchers.

2.5. POSITION-SENSITIVE DETECTORS

Eight multiwire proportional counters and nine drift chambers were constructed *(H. R. Brashear, H. O. Cohn,* V. C. Miller)* for ORNL researchers studying high energy particle interactions in a bubble chamber at Fermi National Laboratory. Both types of chambers are position sensitive along only one direction. Three of these one-dimensional chambers are located at each station and are rotated 120 deg in relation to each other to provide two-dimension information. The multiwire chamber has a fixed region of uncertainty in the position readout, whereas the drift chamber has a variable region of uncertainty in the position readout that, in principle, can be made much smaller than that for the multiwire chamber. However, in spite of the superior resolution for the drift chamber, both chamber types were needed. Failure of a single amplifier for the multiwire chamber has limited effect on the chamber performance, whereas failure of an amplifier in the drift chamber leaves a 2-in.-wide dead space in the chamber.

The nine drift chambers, 1.2 × 1.2 m, have 0.0008-in.-diam. gold-plated tungsten anode wires spaced 2 in. apart and biased at a positive potential of 1500 VDC. Directrix wires of 0.004-in.-diam. berylium-copper are in the anode plane and are placed 1 in. on either side of the anode wires to define the drift space unique to each anode wire. The directrix wires are biased at a negative potential of 4000 VDC. Cathode plane wires are of 0.004-in.-diam. berylium-copper and are spaced 0.1 in. apart. A divider network of ten 22-MΩ resistors partitions the negative high voltage from −4000 VDC on the wires directly opposite the directix wires to ground potential on the wires directly opposite the anode wires. The windows are 0.003-in.-thick mylar coated with aluminum on one side and are used both as an electrostatic shield and to contain the flow gas.

Two of the eight multiwire proportional counters have a 36-in. × 36-in. active area. The other six counters have an active area of 44 in. × 44 in. Anode wires are gold-plated tungsten 0.0008 in. diam. and are spaced 0.08 in. (2 mm) apart. Cathodes of the counters are constructed of 1/8-in.-thick Hexcel material made of epoxy-fiberglass, and the inner surface is coated with conducting paint. Windows are 0.003-in.-thick mylar to contain the flow gas.

Two-dimension position-sensitive proportional counters of still another type, one that uses resistance-capacitance delay line position encoding, were built a few years ago *(W. T. Clay, C. E. Fowler, M. A. Meacham)* for the ORNL small-angle X-ray scattering facility and small-angle neutron scattering facility.[1] The I&C Division is now responsible for engineering support and maintenance of the counters.

The resistance-capacitance counter at the neutron facility contains 60 atmosphere liters of ^{3}He gas, which is valued at approximately $10,000. Because these counters must be occasionally opened for repair, a gas transfer system was built to transfer, purify, and store the ^{3}He gas when the counter must be opened. It has been suggested that the addition of carbon tetrafluoride gas may improve the performance of ^{3}He neutron counters,[2] and the transfer system was therefore designed to purify and store that gas also if necessary *(C. E. Fowler)*. Improvements to the neutron facility during the reporting period include a new sample changer for the specimen chamber, reliable beam-stop operation, and buffer memory for data collection. The X-ray facility has been modified to provide experiment or data collection timing that is functionally the same as that of the neutron facility, without altering its original characteristics. *(W. C. Koehler,† R. T. Roseberry)*

From this discussion it is clear that the Research Instrument Section staff is active in the construction and maintenance of various types of position-sensitive proportional counters needed for ORNL research. During the next few years, the level of activity is expected to increase as more researchers find a need for special-purpose position-sensitive counters.

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*Physics Division.

†Solid State Division.

1. R. K. Abele, G. W. Allin, W. T. Clay, C. E. Fowler, M. K. Kopp, *IEEE Trans. Nucl. Sci.* NS 28, 811 (1981).

2. M. K. Kopp, K. H. Valentine, L. G. Christophorou, and J. G. Carter, "New Gas Mixture Improves Performance of ^{3}He Neutron Counters," *Nucl. Instrum. Methods* 201, 395–401 (1982).

2.6. STAND-ALONE COMPUTERS

The Research Instrument Section has used various PDP-11 computers, sometimes in conjunction with CAMAC instrumentation, to acquire data and to control experiments for ORNL experimenters.

In particular, a new distributed processing and experiment control facility has been installed for the Solid State Division to replace four old and inadequate neutron spectrometer controllers located in the beam room at the HFIR. The new facility broadens the experimental capabilities of the many ORNL users as well as visiting scientists. The facility provides flexibility in handling experiment control and programming and simple and intermediate data processing tasks as well as flexibility in preparing new experiments with new programming tasks and new data reduction routines.

The data acquisition system consists of four PDP-11/23 satellite computers and one PDP-11/34 host computer. The operating system software for the five computers is DEC RSX-11M, which allows multiuser, multitask real-time data acquisition and program development. DECnet software provides communication between the host computer and each satellite computer.

The host computer has 256 Kbytes of memory, four 5.2-Mbyte disk drives, a VT100 video terminal, a DECwriter III, a Tektronix 4014 graphics terminal, and a Versatec 1200 printer/plotter. Three dedicated communication lines link the host system to the ORNL Computing Center: an IBM 2780 protocol communication line links the host computer to the IBM 3033, an asynchronous line links the host computer to the PDP-10, and a switch-selectable line links the Tektronix 4014 to the PDP-10 instead of to the host computer.

Each satellite computer provides an independent data acquisition system for one spectrometer. Each satellite computer has a floating point processor, 256 Kbytes of memory, two 5.2-Mbyte disk drives, a DECwriter II, a Tektronix 4006 graphics terminal, and a CAMAC crate which interfaces to the experiment. The CAMAC crate provides users with a wide range of selectivity for experiment setup and control. The Tektronix 4006 graphics terminal is interfaced to one of two Tektronics 4611 hard-copy units in the systems.

The old neutron spectrometer controllers were replaced by the satellite computer systems at the rate of approximately one every six weeks, with about one week of experiment downtime for each installation. Results of testing and evaluating each satellite showed the system to be highly reliable. The entire system implementation was operational by April 1982.

Another PDP-11/23 computer-based data acquisition system is being used to replace a PDP-8E-based computer system at the Chemistry Division's neutron diffractometer located at the High Flux Isotope Reactor. The new system consists of a PDP-11/23 computer running under a DEC RSX-11M operating system, two 5.2-Mbyte disk drives, 256 Kbytes of memory, a DECwriter II, and a CAMAC crate.

The CAMAC crate has a 1-MHz histogramming memory module which accepts the neutron diffractometer data from a Northern Scientific ADC. An up-counter module in the CAMAC crate is used to accumulate monitor counts from a monitor detector located in the beam line of the neutron diffractometer. A real-time clock module in the CAMAC crate records the ADC live time. The user has the option to run the experiment against either monitor counts or ADC live time. During the experiment the user has the capability to interrogate the data without interfering with data acquisition.

The operating system has minimal overhead time on the running task of the experiment after the experiment is initialized and started, because most of the data acquisition is handled in the CAMAC crate. This frees the system for other tasks such as program development or data reduction.

Finally, a number of custom CAMAC modules have been designed to allow acquisition, processing, and storage of pulse data from high-intensity sources such as a synchrotron. This project was undertaken to provide a computer-independent means for handling data at rates in excess of one million events per second. The work was carried out in cooperation with the ORNL Metals and Ceramics and Solid State Divisions and the Los Alamos National Laboratory.

A previous report (ORNL TM-7325) described an external bus that permits data acquisition at rates higher than those allowed by CAMAC cycles while allowing CAMAC control of acquisition parameters. Work at ORNL during this report period involved checkout, documentation, and system integration of the previously designed components as well as design of a new CAMAC module. This module provides the interface between the data acquisition system and the analog-to-digital converter that will be used at the National Synchrotron Light Source. Two complete systems have been readied for operation, one of which will be used by the Solid State Division at ORNL. The second system will be used by the Metals and Ceramics Division at the National Synchrotron Light Source.

The original prototype system was built and checked out at LANL. Two additional systems were built at ORNL and were checked out on a MODCOMP computer. Special diagnostic software was required and was written in FORTRAN by the Metals and Ceramics Division. Documentation was updated to reflect both changes in the LANL prototype and additional design changes necessary to meet system performance specifications. After the two systems were fully checked out on the MODCOMP, the system designated for NSLS was moved to the PDP-11 computer that will be used at NSLS. The analog-to-digital converter was selected and a custom interface module was designed for it. The ADC and interface were checked out by writing a machine language program which can serve as a data acquisition backup in the event of failure of the data memory on the fast data acquisition system.

All hardware is now ready for use at NSLS. The MODCOMP FORTRAN diagnostic programs have been revised to run on the PDP-11 computer.

This work has shown the feasibility of building a reproducible fast data acquisition system capable of handling events at rates in excess of 1,000,000/s. Furthermore, this system can be installed on any computer which has a CAMAC interface. *(D. E. Smith, D. E. McMillan, R. A. Willems,* A. T. Habenschuss*)*

A third PDP-11 computer has been used as part of a stepping motor control system capable of operating up to 128 motors, which has been designed and built for use with ORNL's port at the National Synchrotron Light Source. This work was undertaken for the Metals and Ceramics Division, and the system was developed to allow computer control of the beam while saving the expense, space, and power required for individual stepping motor controllers. The system has been assembled and tested and has been integrated with the PDP-11 computer.

The motor control system was designed to provide flexibility, low demand on the host computer, and fail-safe motor operation. Flexibility is provided by modular construction. A controller chassis can accept from one to sixteen motor controller boards, and each controller board may be cabled to a motor driver chassis equipped for one to eight similar motors.

Demand on the host computer is minimized by using commercially available microprocessor motor controllers on each motor control board and using a small microcomputer system to interface the controllers to the host computer. Fail-safe motor operation is implemented by providing a motor driver board with limit-switch sensing for each motor. Motor travel may be stopped immediately at a limit without action by the host computer.

The motor control system is rack mounted; a 6-ft rack can house all components, including motor power supplies, for 72 motors.

The motor driver board is capable of handling four-phase stepping motors with winding currents up to 5 A. Resistors and power supplies in the motor driver chassis adapt driver boards to different motors. Stepping rates of 1000 full steps/s and 2000 half-steps per/s have been achieved with several different motor types, hence higher rates can be utilized with smaller motors.

The microprocessor motor controller accepts commands for motor speed and acceleration, and motion is accomplished by issuing a command that includes the desired number of steps. The motor controllers are interfaced to the host computer by a Z80 microcomputer system and STD bus hardware. A command interpreter using 4 Kbytes of ROM was written. A serial RS232 interface is used between the Z80 and the host, which allows the control system to be easily connected to any computer system. Since commands and responses are in ASCII characters, interface software can be implemented through operating system terminal drivers. This interface scheme allows substitution of any RS232 terminal for the host computer whenever manual command generation can be used. Hardware design goals have been verified by system testing using a terminal for command entry.

This stepping motor control system may be compared with the motor controls required for four motors used for sample and detector positioning on the same beam line: The four motors must operate at speeds up to 10,000 half-steps per second. This rate requires special drive techniques, and commercial translators are used. Also,

- Interface to the computer requires four modified CAMAC stepping motor modules.
- Inclusion of limit switches for fail-safe operation required the design of a special interface between the CAMAC outputs and the translators, and an additional CAMAC module is required to allow the computer to sense limit status.
- The translators and limit switch interface require 35 inches of rack space.
- The computer interface occupies nine CAMAC positions.

This comparison reveals the distinct advantages of the ORNL stepping motor control scheme when a large number of motors must be operated at moderate speeds. *(G. W. Turner)*

*Metals and Ceramics Division.

2.7. ENVIRONMENTAL MONITORING—DETECTORS AND SYSTEMS

A prototype of an environmental monitoring data collection system was installed for the Department of Environmental Management in the ORNL Industrial Safety and Applied Health Physics (IS&AHP) Division. The system includes a local air monitoring station (LAM), which was designed to accommodate all of the instruments and functions required at any future LAM station, and a prototype host computer capable of handling up to four LAMs similarly equipped.

Data collection is accomplished by a microcomputer that converts LAM data into engineering units, organizes the converted data (along with status and alarm conditions) into convenient records, and transmits those records to the host computer upon request from the host.

The prototype host is a computer-based pulse height analyzer without analog-to-digital converters or other normal data input hardware. The host receives data from the LAM upon request and writes it into the pulse height analyzer memory. A header file is then generated and the analyzer memory transferred as a data file to flexible disc for archiving.

Spectral data collected by a basic pulse height analyzer at the LAM is received by the host and transferred into its analyzer memory for quantitative analysis. All of the software for performing the necessary spectral analysis is included with the computer-based analyzer.

A continuous sampler for collecting radioactive iodine and particulates from air was developed as part of this new LAM. The sampler consists of a filter paper for collection of particulates and an activated charcoal cartridge for collection of radioactive iodine; an assembly that positions the sample near the end of a germanium detector, channels sample air to and from the filter and cartridge, and includes a manual filter-cartridge changer; and an external pump for drawing the sample through the assembly. The germanium detector and sampling assembly are located inside a lead shield.

During operation of the LAM, greater detection sensitivity is obtained by simultaneous collection of sample and gamma energy spectra. As a result, for longer-lived activities the energy peak heights are proportional to the square of the simultaneous collection time. *(C. C. Hall, R. T. Roseberry, R. L. Shipp)*

A new linear count rate meter, ORNL Model Q-5851, has been designed for use in either alpha or beta-gamma constant air monitors. The instrument is a low-cost, solid-state replacement unit for existing vacuum tube models that have been in service since the 1950s. The instrument is an ac-powered, rack mount unit that will work well with either GM tubes or alpha scintillation probes. The circuitry features an adjustable high voltage supply, a current-sensitive input amplifier that permits the use of long cables between the detector and amplifier, a discriminator, a count rate circuit, an aural indication of counting rate with a volume control knob, alarm circuitry with external alarm relay contacts, and drive for the front panel recorder. The instrument rear panel connectors are pin-compatible with the 250 existing vacuum tube versions now in service; therefore replacement of modules in the field will be straightforward. *(R. A. Todd, H. N. Wilson)*

A different radiation detection system has been designed for the Y-12 Product Certification Division to count the alpha and beta activities of air and smear samples. The system incorporates two modified card readers that will position IBM cards, on which the sample is placed, under a silicon surface barrier detector. The system includes a Berkeley stack sample handler that counts the alpha activity present on stack sample discs and uses two photomultiplier tubes with ZnS scintillators as detectors. Pulse height discrimination is used for all detectors in order to distinguish various isotopes and reject radon and thorium counts. The data are reduced in three HP-87A personal desktop computers, which are also used for control of the modified card readers and the Berkeley handler. The processed data is sent from the three HP-87A computers to a PDP-11/24 processor for data storage and concentration. A degree of redundancy has been designed into the data processing system in order to ensure its continuous operation. *(G. A. Colman)*

Health Physics Division. Its purpose will be to differentiate between the widely varing background of naturally occuring alpha activity and the maximum permissible concentration (MPC) levels of plutonium. With minor modification and a microcomputer, this monitor may be incorporated into the air monitors presently in use. The methodology for it was developed by E. D. Gupton of the IS&HP Division.

This development is currently in its third phase. The first phase was an analog approach, which worked to some degree but was not easily adaptable to new environments and was limited as to further development potential. The second phase incorporated an RCA CMOS, CDP-1802, microprocessor in an all-digital system that allowed the method to be refined via a main frame computer. The third phase incorporated an Apple II Plus computer that enabled the monitor to be real-time with the data plotted and printed on a CRT and data storage on floppy disk.

Further plans include interfacing several air monitors to one microcomputer such that a complete facility would have the capability of monitoring MPC levels of plutonium. *(H. R. Brashear, A. C. Morris, D. R. Patek)*

Because the detection of radioactivity in soil is as important as in air, a phoswich scintillation detector consisting of a thin CaF_2(Eu) and a 1-in.-thick NaI(Tl) scintillator was procured to evaluate as a detector of radioactive contaminants in soil samples. The phosphorescence decay rates of these two scintillators are different and, therefore, with a pulse shape analyzer the pulses from events occurring in each scintillator can be separated. An alpha or beta spectrum can be acquired from the scintillations in the CaF_2(Eu) crystal, and a gamma spectrum can be acquired from those occurring in the NaI(Tl) crystal. Also, by operating the elctronic circuit that processes the alpha or beta pulses from the CaF_2(Eu) in an anticoincidence mode with the NaI(Tl) pulses, the background pulses caused by gamma photons in the CaF_2(Eu) can be rejected. This allows an improved spectrum of alpha or beta radiation in high gamma background. Detailed information show this sys-

Because properly calibrated radiation detectors are important to civil defense, a new instrument for calibrating the Civil Defense CDV781 Aerial Survey Monitor has been designed to replace the older, bulky calibrator The standard integrated circuits incorporated in the new design reduced the size and the number of components, which, along with rugged construction, increased the reliability of the unit. A prototype of the new instrument was sent to Civil Defense for evaluation. They subsequently ordered sixty units, which were assembled in-house and shipped to the Federal Emergency Management Agency. *(M. S. Emery)*

The Research Instrument Section upgraded the data collection system used by the Environmental Sciences Division to measure dissolved oxygen (O_2) in 20 ecosystems ponds. An Intel 8085 microprocessor-based system was implemented for sequencing, accepting, and recording O_2 data from the probes presently in use. An improvement to the probe-switching arrangement was also made by substituting reed relay switches for the stepping relay previously used. *(W. H. Andrews, T. R. Barclay, A. C. Morris)*

As part of the effort to protect the public against radiation hazards, a low power, battery-operated data logging system was assembled. This system will be used by the Environmental Sciences Division for recording the water level in wells drilled around radioactive waste burial grounds. The unit employs a Memodyne 821 Data Logger fitted with transient voltage protection and timing circuitry designed in this Division. The system can accept up to fifteen inputs in addition to a reference channel for calibration purposes. The actual water level measurement is made by portable liquid level recorders manufactured by Belfort Instrument company. The level recorder is modified by adding a potentiometer to provide an output voltage proportional to the water level. This voltage is monitored by the data logger and recorded on cassette tape for subsequent computer analysis. The system has been very successful thus far, and two more units are currently being assembled. *(M. S. Emery)*

Two compact versions of a third generation derivative uv absorption spectrometer were designed and fabricated in cooperation with the Health and Safety Research Division to monitor atmospheric or waste water contaminants. Both instruments utilize photodiodes rather than photomultiplier tubes, which reduces the weight and eliminates the need for a high-voltage supply in the instruments. After amplification of the photodiode current, virtually all of the signal processing is done by the software program on a single-board computer in the module. The air monitor features a unique and compact multi-pass sample chamber built into the instrument chassis, and the waste water monitor includes a compact, light-tight sample cell holder for liquid samples. An open-frame, microprocessor-controlled power supply for the deuterium lamp was designed and constructed for these units. *(A. R. Hawthorne,*,† J. C. Moyers, C. E. Stevenson, R. A. Todd)*

Two mechanical devices were developed to produce aerosols for laboratory study by environmentalists. First, to study the differences in aerosols produced by cigarette smokers having wide variations in their "puff profiles," a device was developed to provide puff profiles with variable flow rate and total volume. This device can produce puff with a total volume up to 100 cc, and the flow rate can be controlled by a programmable stepping motor which drives a teflon-sealed piston by means of a ball screw. Preliminary testing of this device is under way. Second, a motorized device has been designed and fabricated for extruding materials to be burned to produce aerosols under study. An extruder with a 5000-pound force capacity capable of low speed, accurate extrusion rates was needed to handle red phosphorus/butyl rubber compounds. The variable-speed, motor-driven ball screw device can extrude at rates of .12 to 6 in. per min with a speed regulation of ±3%. *(G. W. Allin, T. M. Gayle)*

From the above discussion it is clear that the Research Instrument Section staff has developed detectors, electronics hardware, and data acquisition systems to minimize health hazards in the environment. In particular, development of a new local area monitor for radioactive iodine and particulates has been accomplished. The alpha/-beta/gamma constant air monitor which was designed in the 1950s is being updated. This monitor measures the radioactivity of an air sample that is continuously drawn through it, while another system has been designed to automatically measure the radioactivity of previously collected air and smear samples. Still another measures small concentrations of airborn plutonium, and a special scintillation detector with pulse shape discrimination has been developed to measure radioactive contamination of soil samples. The Section has also developed a new electronic calibrator for the Civil Defense Aerial Survey Monitor. Work areas less directly connected with the measurement of radioactivity included development and installation of a microprocessor-based multiplexer for the O_2 probes located in a cluster of artificial ponds which are filled with water from such diverse sources as coal mines and sewage treatment plants. In addition, the Section has assembled a battery-operated data logging system for measuring the water level in the areas around a radioactive waste burial ground, has developed the electronics for a third generation uv absorption spectrometer for chemical, non-radioactive carcinogens, and has developed two mechanical devices to produce aerosols for study by environmentalists. Although most of these instruments were developed to measure radioactivity in the environment, a surprisingly large fraction were developed to measure chemical pollutants.

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2.7.1. AN UPDATED SYSTEM FOR MOBILE GAMMA-RAY SCANNING

T. E. Myrick*
M. S. Blair E. T. Loy*
R. W. Doane* W. H. Shinpaugh*

[(Abstract in *Health Physics* 43(1), 156 (July 1982)]

The mobile gamma-ray scanning system developed and operated for the Department of Energy by Oak Ridge National Laboratory (ORNL) has been upgraded to improve sensitivity and reliability, and to provide continuous radionuclide-specific analyses. The gamma-ray detection system consists of three 4 × 4 × 16-in. NaI(Tl) Polyscin* log crystals, each with an integral 3.5-in. photomultiplier tube. The crystals are housed in a lead-shielded steel frame to provide a 12 × 16-in. detector surface area for acceptance of gamma-rays through one side of the survey van. The detector and shield height can be varied with a hydraulic lift mechanism to optimize the detector field of view.

The detector output is transferred to a computer-controlled eight-channel discriminator and interface, designed and fabricated at ORNL. This unit provides for continuous analysis of data inputs for correlation of system location with count rate information. Six separate energy regions-of-interest are analyzed and a ^{226}Ra-specific algorithm is employed to identify locations containing residual radium-bearing materials. Data on other naturally occurring radionuclides (such as ^{40}K and ^{232}Th) are obtained for comparison as part of the analysis. Multichannel analysis capabilities are included in the system for additional qualitative radionuclide identification. The system is operator-controlled through keyboard instructions to an on-board minicomputer. Data output is provided on the computer video screen, dual stripchart recorders, and a graphic printer. Data storage is provided by a dual floppy disc system.

This mobile laboratory is being used primarily in support of DOE's radiological survey program in the vicinity of uranium mill tailing sites and locations formerly utilized under government contract in operations involving radioactive materials. Typically, upon completion of the mobile scanning, comprehensive on-site surveys are conducted by DOE on those properties identified as containing contaminated materials. A description of the system will be presented as well as typical results from recent mobile scans.

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*Health and Safety Research Division.

2.8. PLANT SECURITY

The I&C Division is involved in two aspects of plant security, detecting unauthorized people who are trying to enter the area unnoticed and undetected, and developing and implementing a new Union Carbide Nuclear Division (UCC-ND) badge system.

As part of ORNL's ongoing program to improve security, additions have been made to the closed-circuit television (CCTV) systems for Building 6000 to increase surveillance levels at its three main entrances. Enhanced security is especially important during off-shift hours when building population is low, but the accelerator facilities must be accessible to operators and research personnel around the clock.

Three separate CCTV cameras are involved in this new installation—two situated out-of-doors to survey front and rear entrances, and the third located indoors to monitor a secondary front entrance. The two outside cameras are mounted in environmental housings which are temperature-controlled. The cameras are provided with both electronic and mechanical iris controls that can compensate for very large variations in ambient lighting levels. Moreover, "silicon target" type camera tubes were specified for the camera procurement to reduce the possibility of the vidicon "burn-in" effects caused by constant, or exceptionally bright, image scenes.

Two sets of three CCTV monitors were installed within the building. One set was located within the Receptionist's Office to provide a simultaneous view of all three entrances being surveyed. The second set of three CCTV monitors is mounted within the Accelerator Control Room. These monitors enable the accelerator operator to identify any persons seeking to enter at unusual hours.

After cameras and indoor monitors were adjusted to achieve the proper framing, focus, and lighting contrast for each entrance area, the new CCTV system was demonstrated to appropriate Physics Division personnel. It has been operating successfully since September 1981.

In addition to CCTV, the protection of both the physical plant perimeter and interior spaces requires a variety of detection devices and alarm signal transmission systems. The design of a security system for a perimeter or an interior space begins with a survey to determine the physical and environmental conditions in the area of interest. When the conditions are defined, appropriate detection sensors are specified. More than one type of sensor may be used in the same area to provide for unusual or irregular physical or environmental conditions and to reduce the likelihood of defeat. Examples of sensor types available for use include passive audio (sound listening), proximity (capacitance), vibration (motion), passive infrared (motion), ultrasonic (motion), and microwave (motion) detectors. *(T. R. Barclay, A. C. Morris, J. A. Russell)*

In addition to the ongoing ORNL security improvement program, the Safeguards and Security Upgrade Project was established in 1976 in response to a directive from DOE on improving the protection of special nuclear material.[1,2] Project tasks include the following:

- Building improvements
- Enlargement of the ORNL Emergency Control Center
- Security fencing
- A new storage vault
- Improved area lighting
- Perimeter and interior intrusion alarms
- A computerized alarm processing system
- A radiochemical pilot plant inventory system
- A TV monitoring system.

Most of the work performed by the I&C Division during this reporting period involved instrumentation in the new vault, the rf cable TV transmission system,[3] the security alarm monitoring and access control system (SAMACS), installation of TV cameras, and connection of security alarms to the alarm processor.

The TV monitoring system consists of 40 intermediate- and low-light-level television cameras, a broadband multichannel rf cable transmission system, monitors, motion detectors, and videotape recorders. The TV cameras are installed at strategic locations designated by the Laboratory Protection Division. Interior cameras provide video signals which are monitored by scanning digital video motion detectors, thereby supplementing conventional motion detecting methods. Transmitted TV pictures provide an immediate means of assessing alarms, thus permitting rapid identification of false or nuisance alarms, enabling assessment of real alarms, and providing additional information upon which to base response actions. Due to the constant motion characteristics and variable contrasts of outdoor scenes, video motion detection monitoring for the outdoor cameras is not contemplated. Outdoor TV pictures are useful in assessing the cause of alarms and determining the need for and the type of response required. Three cameras are equipped with remote pan and tilt controls to extend visual coverage.

The rf cable system that ties the network of TV cameras to the ORNL Emergency Control Center is similar in many respects to a community antenna television system (CATV).[4] It uses cable, amplifiers, and hardware designed for CATV systems. The main difference from a conventional CATV system is that this system has multiple "head ends" and only one "customer" end. It was designed and installed by a CATV contractor to meet ORNL specifications. The contractor experienced considerable difficulty during construction due to signal level adjustments substantially different from normal CATV systems. I&C engineers and maintenance technicians conducted extensive testing to verify compliance with specifications and to provide assistance to the contractor in identifying the cause of the problems. The rf system has been accepted and is now operational, and all TV cameras are operational and are now in use as monitoring devices. The video motion detection system for the indoor cameras is only partially operational at this time.

The installation and testing of the security alarm monitoring and access control system[5] (SAMACS) for the ORNL Laboratory Protection Division was completed in mid-April 1982. The delay of one year from a projected completion date of April 30, 1981 was caused by the vendor's internal organization problems.

Factory testing in a temporary installation in Building 3500 was completed in March 1981. The field interface devices (FID) were then moved to their permanent locations in ten buildings, where the field wiring was connected by a second subcontractor. The central processor and its peripherals were installed in the Communications Center and were connected to the FIDs by redundant telephone pairs. Preliminary verification of the field wiring and operability of the FIDs was performed in the early summer of 1981. A system checkout was attempted in late July and early August, but communication errors and software bugs resulted in failure of the preliminary test.

Communication errors were corrected by repairing several communication line driver modules in August, followed by the correction of software bugs, both known and newly discovered, in September. A 30-day (720-hour) acceptance test was performed during October and November, at which time the software acceptance tests were completed by a demonstration of the execution of 54 operator commands. Without stopping the acceptance test clock, the system was made available during the day shift to the vendor's programmer for correction of detected software errors and training of the communications officers. The final data base was also entered into the system during the acceptance test. The acceptance test was completed on November 29, 1981.

Preoperational testing was started in December 1981 and completed in April 1982. The alarms for this final test were divided into three groups: Volumetric Intrusion Alarms (VIA) in Building 2500, alarms in Building 3019, and alarms in seven other buildings. After all of the buildings were tested to verify operation of the modules in the FIDs, the field wiring from the FIDs to the individual alarm points in all of the buildings except 3019 and 2500 was verified. The alarms in Building 3019 and the VIAs of Building 2500 required on-line transfer from the old system to the new system on a daily schedule beginning the last week of January 1982.

The SAMACS system polls eleven FIDs that collect and store alarm information from the respective intrusion detectors which they monitor. If the main computer is inoperative an alternate reporting channel is used, the central computer generates an audible tone, displays alarm information on the operator's CRT terminal, and prints a log of all transactions. Upon request, the processor will produce reports sorted by date, time, type of alarm, type of transaction, alarm number, or any combination of these.[6] I&C personnel participated in the checkout of hardware and software and the installation and testing of the complete system. Connection of alarms to the FIDs was done by a subcontractor. The alarm monitoring system is in service and routinely operated by the communications officers of the ORNL Laboratory Protection Division. A final SAMACS system software modification was completed in May 1982 to correct a problem caused by the VIA control at the time the alarms were connected.

Storage vault instrumentation designed by I&C was installed by I&C personnel after acceptance of the new building. Two criticality monitors, an evacuation horn, and evacuation beacons were installed and connected to the evacuation alarm system for Buildings 3012 and 3044. Other instrumentation included a health physics continuous air monitor, television monitors, and intrusion alarms.

The safeguards and security upgrade project was closed March 31, 1982. *(R. I. Crutcher, C. C. Hall, J. W. Reynolds, J. A. Russell)*

A badge-operated portal entry and controlled door entry system has been developed for deployment at the UCC-ND plants. The program has been directed by the UCC-ND Security and Safeguards Department, and has involved participation by members of many disciplines within the Union Carbide Nuclear Division.

The UCC-ND badge system was developed to modernize the Nuclear Division's plant and building entry authorization and control; to eliminate the need for the daily distribution and handling of large numbers of cover badges; to provide immediate invalidation of terminated, lost, and suspended badges; to provide superior badge audit capability; to provide more flexible visitor entry authorization; to implement the UCC-ND employee badge as a key for door control; and to lower badging operations costs.

The security management and control badge system is software driven throughout. The host computer is a PDP 11/34 with 1-Mword memory, three 67-Mbyte disk drives, a magnetic tape drive, 16 EIA RS232 lines, 64 20Ma current loop lines, CRT terminals, and console printing terminals. The applications software is written in FORTRAN, running under the RSX-11M-Plus executive. The data base consists of approximately 20,000 UCC-ND employees, 10,000 non-Carbide badged persons (RUST, DOE, etc), and presently a 3,000-person ORGDP visitor file. A common data base and common software and hardware modules make the system installable at any of the four UCC-ND plants. All badge reader controllers are connected to the host computer over in-plant dedicated telephone lines. The badge reader system is a security management system which at present can detect, log, and report as many as 97 different badge system operational and error transactions via its CRT activity monitors, statistician display, and hard-copy reports.

Important features of the badge reader system include modular, standardized, and flexible hardware and software which can be easily installed for many types of applications, including employee portal entry, security officer/system interaction, visitor authorization, door and rotogate control, entry and exit badge authorization and logging, printing of visitor authorization cards, and on-line area population monitoring and logging. Other broad areas which offer potential for UCC-ND badge reader usage include stores issues, timekeeping, special materials control, and medical and library logging.

A computer-controlled UCC-ND badge reader portal entry system went into operation at the Oak Ridge Gaseous Diffusion Plant (ORGDP) on June 26, 1982. The ORGDP application of the system was the first operational stage of a personnel entry and control system that may eventually be used in one form or another at each of the four Nuclear Division plants. Badge-reader-controlled door entry is scheduled to be operational in three reactor buildings (7900, 3042, and 3010) at the ORNL site in December 1982.

A Biology Division area entry rotogate and an Engineering Technology Division special area are to be under badge-controlled entry in February of 1983.

The UCC-ND badges designed to be used with this system were issued for use at all four plants of the Nuclear Division in April 1980.

Continuing improvements have been made to upgrade the mechanical aspects of the badge design and improve the component bonding. Also, the badge fabrication procedure was revised to tighten the quality control on the shelf life of the components coated with urethane adhesive.

A new badge was designed for use with the "flimsy" type temporary badge. It contains an encoded, machine-readable module, and its artwork consists of a white background with a red stripe around its border and a red diagonal slash (the international designation) across both faces. "Not Valid Without Photo Badge" is printed on the front face. This new badge was needed to implement the ORGDP badge reader system.

The master control for the ORGDP badge reader system is operated with an extensive data base that includes the security-related parameters of all UCC-ND employees and contractor personnel. To monitor entry request communications at a portal, the computer system reads a badge, searches the master authorization file, and transmits a response to both the employee and the portal security officer regarding the validity of the badge. Hardware and procedures have been installed to permit authorized plant protection personnel to alter designed fields in the master security files and thus immediately change portal entry authorizations for any employee or visitor.

Each portal entry lane requires a badge reader controller, along with all or some of the following equipment: a badge reader with message display to be used by the employee seeking admittance; a badge reader with message display to be used by the guard; and a keypad unit for use by the guard for entry authorization of visitors and of employees who have forgotten their badges. Each controller can also support a printer configured to print visitor authorization cards. Generally only one printer and one keypad unit is required per portal regardless of the number of entry lanes open at the portal. The 20 entry lanes at 9 portals initially implemented at ORGDP required 42 badge readers, 9 printers, and 11 keypad units. Locked security cradles are used for mounting all employee badge readers because they are generally not located entirely within the confines of a secure portal area. Weather shields are mounted on exposed security cradles.

To expedite the testing of many controllers during the fabrication and assembly of the badge reader equipment, a semi-automated loop-back tester was developed to qualify and troubleshoot the initial run of approximately 30 microprocessor-based control boards for the ORGDP personnel access control system upgrade.

The badge reader control board contains a microprocessor, ROM, RAM, and various interface circuits. This board interacts with external devices through four connectors with a total of 20 parallel outputs, 11 parallel inputs, and 3 full-duplex serial lines. A thorough check of all inputs and outputs is required before a control board can be installed in a working badge reader station.

In normal operation, the control board requires connections to a power supply and two external reader-display units, as well as a communication link to the host computer. The several operational modes of the control board each require a different host computer program. Programming a computer-driven general-purpose board test facility would have been prohibitively expensive, even if one had been available. Manual testing was also precluded due to the large number of input/output channels and the required response time. Separation of hardware and software testing would allow the substitution of a custom test program for the control board microprocessor. Since the hardware design was already fixed while the control software was still undergoing developmental changes, such a division of testing was the most realistic and flexible approach.

The greatest possible degree of automation in testing was desired: to achieve this, an external test fixture was designed to interconnect board inputs and outputs. Serial lines were interconnected in a staggered mode so that intentional data errors could be introduced and detected. The test fixture has a light-emitting diode (LED) indicator for each parallel output, the purpose of which is twofold. The indicators allow visual initial verification of operation of the control board outputs. Once output operation has been verified, the indicators then serve as a display to show progress through the test program and to provide diagnostic information about any failures detected.

Test software is written in two sections, the first of which advances through the outputs, activating one at a time. This section of the test is automatically repeated so that proper output operation can be verifed by observing the indicator light sequence. Once proper operation of the outputs has been verified, activation of a switch on the test fixture panel advances to the second section of the test program, which is fully automatic. It tests operation of pertinent microprocessor instructions, exercises RAM memory, then tests all inputs by operating them through the previously tested outputs. The serial lines are tested by sending messages containing errors from one line to the next. Once proper error detection is established, good blocks of data are transmitted to verify that all data patterns can be accommodated. After a positive indication is provided at the conclusion of all tests, the automatic cycle is repeated so that a board may be "burned in" under test conditions.

Successful operation of this test system requires that the microprocessor be functional and capable of executing a basic subset of instructions. Devices connected to the processor's external bus must not overload the bus. Experience has shown that these constraints are not severe, and boards with failures in the basic components are easily identified by the total lack of response in the test fixture.

The sample system has been used successfully to check the initial set of boards as well as the set of boards for reactor access control at ORNL. It will also be used to check the approximately 70 boards for planned installation at Y-12 and ORGDP. Diagnostic tables included with the test procedure have made it possible to isolate the location of failures to within one to three components in almost every case. Duplication of the test system is now in progress to provide adequate routine maintenance of control boards at the various Nuclear Division plants.

Two types of badge reader control boxes have been designed, both with identical control boards. The portal entry control boxes use 25-conductor standard EIA RS-232 cables with 'D' type connectors for all interconnections. The door control boxes have terminal strips for termination of conductors pulled through conduit.

A badge reader controller used for door control can support either an entry reader and an exit reader or entry readers at two different doors. *(E. Madden, R. G. Affel,* G. W. Allin, Y. H. Etheridge,† D. E. McMillan, C. R. Mitchell, J. W. Simmons,† A. A. Smith)*

Research Instrumentation Section personnel have made a significant contribution to cost-effective security at ORNL and throughout the Nuclear Division. This has been accomplished by implementing various electronic intrusion detection devices, including sound vibration and motion detectors as well as closed-circuit television. Alarms for the detectors alert the guards to potential intrusions, and the closed-circuit television enables the guards to determine immediately the seriousness of each alarm. I&C Division personnel also contributed to cost-effective security through the development and partial deployment of the UCC-ND badge system. The implementation of intrusion alarms, closed-circuit television, and the badge system will enable better protection of government property without an increase in the security force.

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*UCC-ND Security and Safeguards Department.

†UCC-ND Computer Sciences Division.

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5. J. W. Reynolds, E. Madden, and F. E. Wetzel, *Instrumentation and Controls Division Biennial Progress Report for Period September 1, 1978, to September 1, 1980*, ORNL-5758 (June 1981), p. 194.

6. J. W. Reynolds report on alarm processor, to be published.

2.8.1. SECURITY TELEVISION MONITORING USING A WIDEBAND RADIO FREQUENCY CABLE SYSTEM

J. A. Russell R. I. Crutcher
A. L. Case F. E. Wetzel*

(Abstract of invited paper presented at 1981 Carnahan Conference on Crime Countermeasures, University of Kentucky, Lexington, May 13–15, 1981)

The Oak Ridge National Laboratory (ORNL) was equipped with a multichannel, bidirectional rf cable television system for security assessment. The multichannel cable system was selected over a more conventional video cable system that has separate cables to each camera. Primary considerations for selection of the rf cable system were initial cost and ease of modification or additions to the system. Two 300 MHz cables, having a capacity of ~60 channels, and modulators, located in buildings or building complexes, receive video signals from 40 TV cameras. These signals are transmitted as rf signals by the cable system to a centralized emergency control center (ECC) where they are demodulated, processed, and displayed by video equipment. TV monitors, digital video motion detectors, and recorders enable the dispatcher in the ECC to evaluate and document the video information.

The vulnerability of a multichannel cable system to loss of all signals by cable damage or amplifier failure was minimized by designing branch systems and by routing most of the cables within the main ORNL security perimeters. The video signals are used for security monitoring and assessment. Since the video motion detectors require control of the time relationship of the incoming TV signals, a system sync signal is produced at the ECC and distributed to all cameras on one of the outgoing channels. Sync signals are delayed at each camera so that the proper time relationship of each video signal at the ECC is set.

This paper covers the justification for a TV system and the reasons for selecting an rf cable system. It includes a discussion of the design criteria, installation, and expansion capabilities of the system.

*UCC-ND Engineering Division.

2.9. ENGINEERING SUPPORT FOR FUSION ENERGY DIVISION

The I&C Division provides several on-site engineers to the ORNL Fusion Energy Division, located at Y-12. Most of the I&C effort during this period was in support of the Elmo Bumpy Torus (EBT) experiment. Installation of the Neutral Beam Diagnostic on EBT and preliminary operation and testing of the system were completed. Extensive redesign of the system components, originally developed and utilized in the ORMAC facility, was required. Additional new components were developed in order to complete the system requirements: a vacuum system control and interlock panel, a 2000-V, 200-A transient-protected rectifier stack for the ion source arc supply, various motor-driven ion beam diagnostic probes providing digital readout of ion beam parameters and probe position, and the ion beam transport system.

Additional electrical and mechanical engineering design support was given for the following plasma diagnostics: (1) A neutral particle time-of-flight spectrometer for investigating neutral particles leaving the plasma confinement region, (2) A rotating low energy X-ray scanner, (3) A 70-GHz microwave spectrometer, (4) The SCAT PAK II laser-based Thompson scattering diagnostic for measuring the plasma electron temperature and density on the ISX-B machine, (5) The measurement of the uv light intensity from the EBT plasma and the fluctuations of that intensity, and (6) the measurement of the X-ray spectrum from the EBT plasma. In addition, a master interlock and interconnection control system was designed for the ion-cyclotron resonance heating antenna, and a CAMAC waveform generator was designed for Tokamak plasma position control and gas-puff control.

A second gyrotron and all supporting main, body, and gun magnet power supplies are being designed and built to ORNL specifications for installation on EBT-S. This new gyrotron will double the microwave power available to EBT-S, and will alternately provide a microwave source to the EBT-P test stand for component development. A high voltage switch has been developed to switch the beam and gun voltage of the present power supply between the two gyrotrons. Gyrotron tubes to minimize cross coupling between the two gyrotrons are also under development. At the conclusion of the program, one 27.7-GHz gyrotron and one 28-GHz gyrotron will together supply 400 kW of continuous rf power for electron cyclotron heating of the plasma in the EBT-S experiment. Because each gyrotron costs approximately $300,000 and creates potentially dangerous conditions, extensive interlocks have been designed and installed to protect both the operators and the gyrotrons. The estimated cost of the project is in excess of $1.5 million. *(D. D. Bates, D. W. Bible, J. M. Madison, J. E. Phelps, R. E. Wintenberg)*

2.10. ENGINEERING SUPPORT FOR ACCELERATOR PHYSICS

The I&C Accelerator Physics Group has broad responsibilities for operational support of the Oak Ridge Electron Linear Accelerator (ORELA), the Holifield Heavy Ion Research Facility (HHIRF), the Oak Ridge Isochronous Cyclotron (ORIC), the system electronics for physics experiments, and accelerator and detector developments. Furthermore, by means of cooperative experiments between researchers at these facilities and at other national laboratories, this support extends beyond ORNL.

ORELA is operated more than 5000 h per year. Significant tasks this year included the changeout of Section One of the accelerator due to a failure in its cooling system, repair of large vacuum leaks in Section Four, complete mechanical realignment of the accelerator, and replacement of the four SF_6 cooling gas blowers with two blowers that have higher capacity and improved oil seals. In addition, several of the diode sputter-ion vacuum pumps were replaced with triode units because of their easier starting; a larger diameter beam scraper orifice was installed downstream from the electron gun; and the gamma-flash beam shape detector monitor was repositioned to give better statistics on the beam pulse width.

The beam output capability of the 300-kV Cockroft-Walton deuteron linear accelerator has recently been increased by a factor of 100, and a crossed field analyzer to eliminate unwanted ions from the beam is under development, as is a system for pulsing the beam with or without synchronization with ORELA pulsing. Associated with this accelerator is the development of a precision (0.1%) positioning system for detector position within a volume of 3 × 3 × 5 m. *(T. A. Lewis, G. K. Schulze, H. A. Todd, J. H. Todd)*

Experience in the operation of the Oak Ridge Electron Linear Accelerator (ORELA) has proven that improvement of the electron beam injector is the key to sustained and improved facility performance. ORELA presently produces electron guns with increasing reliability, but appropriately "clean" facilities should improve reliability and performance. A laboratory of approximately 2400 ft^2 planned and proposed as a second-floor extension of Building 6010 will provide space for improved production facilities and equipment to allow a detailed characterization of the gun output not presently possible. This laboratory space will also provide appropriate facilities for integrated tests of the gun, the gun driver, and the entire complex prebuncher system, permitting design improvements and new components to be tested without consuming accelerator time. The overall goal is to significantly improve the practical output while maintaining the required 5000–6000 h of annual operation. *(T. A. Lewis)*

A program is under way to improve the present electron gun pulser, and to later change to a high-power planar triode pulser with digital control. The design goal of this pulser is 5000 V output into 50 Ω with rise-times and fall-times of 1 ns, and pulse widths variable from 3 to 50 ns at rates up to 1 kHz *(T. A. Lewis, G. K. Schulze)*

This group also developed a gated 800-kHz rf oscillator capable of 20 μs bursts of power at an 800-W level for driving a crystal transducer for slow neutron defraction. This equipment is for the purpose of gating slow neutrons for a Solid State Division experiment at the HFIR. Six of these circuits and their frequency control are contained within the volume of one NIM crate, while one commercially available unit is twice that size. *(G. K. Schulze)*

As assistance to the ORELA experimenters, engineering support is provided for nuclear measurements of the neutron multiplicity of ^{252}Cf,

^{233}U, ^{235}U, and ^{239}Pu and for neutron fission cross sections of ^{233}U, ^{235}U, and ^{239}Pu. Responsibility for the nuclear measurements extends from the detectors to and including the data acquisition systems and the development or procurement of special instrumentation which is rapidly extending into the use of CAMAC. *(R. W. Ingle, J. H. Todd)*.

Also at ORELA, programs for investigating neutron detectors for use as flux monitors and for using NaI detectors as gamma multiplicity detectors are continuing. In addition, a gamma-ray detector system and associated electronics for measuring capture cross sections of high background fission products is currently under development. *(R. W. Ingle, J. H. Todd)*

In June 1982 the tandem accelerator contractor officially turned over operation of the tandem accelerator to the HHIRF staff. An effort will be made to coordinate facility development with the need to provide beam time to experimenters.

During the period of this report, a Perkin-Elmer 8/32D computer was integrated with the tandem accelerator control system computer. The 8/32D shares a memory bank in common with the primary control computer, making it capable of monitoring accelerator operating parameters, monitoring the control system operation, and performing higher level control functions. At the present time programs are run on the 8/32D to record tandem operating parameters for scanning the mass-analyzing magnet following the ion source in order to determine the ions being produced and their intensities, to set up tandem parameters for a new beam, and to automatically change beam energy. During this period assistance was also provided to rewrite the system software and aid in the debugging of the data acquisition system hardware. *(R. C. Juras)*

The Research Instrument Section provides experienced scientists and engineers to support the various particle accelerators at ORNL. Their efforts have resulted in the reduction of scheduled downtime at the accelerators as well as continual improvements in the operation of the various facilities.

2.10.1. POSITIVE PION-NUCLEUS ELASTIC SCATTERING AT 30 AND 50 MeV

B. M. Preedom et al.*,† N. W. Hill

[Abstract of *Phys. Rev.* 23(3), 1134 (March 1982)]

We present measured angular distributions for π^+-elastic scattering at 30 and 50 MeV from selected targets with $A = 12$ to 208. These angular distributions were analyzed using a phenomenological optical potential of first-order form. The mass dependence of the potential strength parameters displays the isospin dependence expected from the free πN interaction.

*University of South Carolina, Columbia, South Carolina.

†Because of the large number of coauthors, only the principal author and any coauthors from the Instrumentation and Controls Division are listed here.

2.10.2. A MAJORITY LOGIC SCINTILLATION DETECTOR FOR keV ENERGY SCATTERED NEUTRONS

R. R. Winters* N. W. Hill

[Abstract in *Bull. Am. Phys. Soc.* 27(5), 629 (May/June 1982)]

Inelastically scattered neutrons can be separated by time-of-flight techniques from elastically scattered neutrons by using a plastic scintillator coupled to three photomultiplier tubes. The neutron detector is operated in a majority-logic mode to allow discrimination below the single photoelectron response. Using this technique, low sensitivity to discrimination bias and to small pulse height noise is achieved.

*Denison University, Granville, Ohio.

2.10.3. A CRYOGENIC IONIZATION CHAMBER FOR HIGH-RESOLUTION FISSION CROSS-SECTION MEASUREMENTS

Richard C. Extermann*
George F. Auchampaugh* Clayton E. Olsen*
John D. Moses* Nat W. Hill

[Abstract of *Nucl. Instrum. Methods* 189, 477–84 (1981)]

This paper describes a multiple-plate gas ionization chamber designed for cross-section measurements of neutron-induced fission in strongly radioactive nuclei. The requirements of high resolution are discussed, in particular the need for cooling the sample to reduce Doppler broadening, and the selection of the gas mixtures that will enable fast counting at a low temperature. The observed gain in resolution with cooled samples is in good agreement with theoretical predictions.

*Los Alamos National Laboratory, Los Alamos, New Mexico.

2.10.4. THE RATIO OF THE $^{10}BF_3$ AND $^{3}He(n,p)$ CROSS SECTIONS BETWEEN 0.025 eV AND 25,000 eV

C. D. Bowman* R. Gwin†
J. W. Behrens* J. H. Todd

[Abstract in *Nuclear Cross Sections for Technology*, NBS Spec. Publ. 594, p. 97 (1980)]

The $^{10}BF_3(n,\alpha)$ and $^{3}He(n,p)$ cross sections have been compared in the energy range from 0.025 to 25,000 eV. Measurements at NBS using filtered beams gave results at 2 and 25 keV. At the Oak Ridge National Laboratory using the ORELA facility measurements were completed between 0.025 and 10,000 eV. Normalizing the ratio of BF_3/He to 1 at 0.03 eV, the ratio increases by 1% at 10 eV, by 2% at a few hundred eV, by 4% at 2 keV, and by 16% at 25 keV. The large deviations at the higher energies are expected purely from the nuclear parameterization of the cross sections. However, the deviations from 1/v in the ratio below 100 eV are surprising and perhaps might have their origin in the molecular binding for ^{10}B in the $^{10}BF_3$ system.

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†Engineering Physics Division.

2.10.5. A COMPARISON OF (n, α) CROSS SECTION MEASUREMENTS FOR $^{10}BF_3$ AND SOLID ^{10}B FROM 0.5 TO 10,000 eV

A. D. Carlson*
C. D. Bowman* R. G. Johnson*
J. W. Behrens* J. H. Todd

[Abstract in *Nuclear Cross Sections for Technology*, NBS Spec. Publ. 594, p. 89 (1980)]

The (n,α) cross sections of $^{10}BF_3$ gas and solid ^{10}B are compared in order to study possible influences of binding effects on reaction cross sections. It is shown that for $^{10}BF_3$ a deviation from a 1/v cross section of 8% is expected. The ratio of the measured cross sections does not show this deviation.

*National Bureau of Standards, Washington, D.C.

2.10.6. A HIGH RESOLUTION FOCAL PLANE DETECTOR FOR HEAVY IONS

T. P. Sjoreen* F. E. Bertrand*
J. L. C. Ford* E. E. Gross*
J. L. Blankenship D. C. Hensley*
R. L. Auble* M. V. Hynes†

[Abstract in *Bull. Am. Phys. Soc.* 27(4), 520 (April 1982)]

A magnetic spectrograph focal plane detector with 0.5 mm resolution is needed to exploit the high-resolution heavy-ion beams from HHIRF, because the Elbek and Enge spectrographs have small dispersion (~2). For this reason, a vertical drift chamber (VDC), similar to the MIT detector,[1] was designed and constructed with an active length of 38 cm. Signals from each of the 51 sense wires are amplified and detected by leading-edge discriminators and encoded onto 1 of 5 tapped delay lines. Start signals are taken from a plastic scintillator. Measurement of elastic scattering from a 40 $\mu g/cm^2$ ^{197}Au target of 129 MeV ^{12}C ions from ORIC yielded a position resolution using 0.65 mm FWHM ($\Delta E/E = 4.3 \times 10^{-4}$). Most of the line width is estimated to be due to components other than the VDC. An inexpensive vacuum bulkhead feedthru for 75 coax cables was developed.

*ORNL Physics Division.
†Los Alamos National Laboratory.

1. W. Bertozzi et al., *Nucl. Instrum. Methods* 141, 457 (1977).

2.10.7. THE ^{187}Os (n,n') CROSS SECTION AT $E_n = 63$ keV

R. L. Hershberger*
M. T. McEllistrem* R. L. Macklin†
M. Balakrishnan* N. Hill

[Abstract in *Bull. Am. Phys. Soc.* 27(4), 543 (April 1982)]

The importance of the ^{187}Os inelastic cross section at $E_n = 30$ keV for the application of the ^{187}Re-Os cosmochronometer has been pointed out by Woosley and Fowler.[1] Winters *et al.*[2] made a measurement at $E_n = 30$ keV but have only tentative results with large uncertainties. To avoid some of the problems encountered at 30 keV, the experiment was redone with $E_n = 63$ keV at the University of Kentucky Van de Graaff Laboratory. The neutrons were produced via the $^7Li(p,n)$ reaction using a pulsed proton beam. The scattered neutrons were detected with special coincident neutron detectors and were energy analyzed using time of flight. Preliminary analysis gives $\sigma(n,n') = 1.8 \pm 0.6$ b at $E_n = 63$ keV. A discussion of the measurement, the extrapolation to $E_n = 30$ keV, and the implications for the ^{187}Re-Os cosmochronometer will be given.

*University of Kentucky.
†Physics Division.

1. S. E. Woosley and W. A. Fowler, *Astrophys. J.* 233, 411 (1979).
2. R. R. Winters, K. Kappeler, K. Wisshak, B. Berman, and J. Browne, *Bull. Am. Phys. Soc.* 24, 854 (1979).

2.10.8. NEUTRON TRANSMISSION MEASUREMENTS OF $^{187}Os + n$

R. F. Carlson* J. A. Harvey†
R. R. Winters‡ N. W. Hill

[Abstract in *Bull. Am. Phys. Soc.* 27(4) 543 (April 1982)]

The neutron total cross section for ^{187}Os, in the energy range, 27 eV to 200 keV, has been measured at the ORELA facility by the neutron-time-of-flight technique, utilizing a 2.0 gm Osmium sample ($n = 0.008401$ Os-nuclei/barn) enriched to 70.38% ^{187}Os. Measurements were performed at an 80-m flight station with an energy resolution, E/E, of 0.1% using a ^6Li glass scintillator.

Resolved resonances have been analyzed by a Reich-Moore multilevel code (SAMMY) to obtain parameters for 85 resonances up to 500 eV. Preliminary determinations of the level spacing (5 eV) and s-wave strength function (3.9×10^{-4}) for ^{187}Os are in agreement with recent analysis[1,2] of the Osmium isotopes, made in connection with the use of the Re-Os chronometer for estimating the duration of stellar nucleo synthesis.

*Middle Tennessee State University.
†Physics Division.
‡Denison University.

1. J. C. Browne and B. L. Berman, *Phys. Rev. C* 23, 1434 (1981).
2. R. R. Winters, R. L. Macklin and J. Halperin, *Phys. Rev. C* 21, 563 (1980).

2.10.9. S-WAVE NEUTRON STRENGTH FUNCTION FOR $^{204}Pb + n$ REACTION

D. J. Horen*
J. A. Harvey* N. W. Hill

[Abstract in *Bull. Am. Phys. Soc.* 26(8), 1139 (October 1981)]

High resolution neutron transmission measurements have been made on ^{204}Pb using the 80-m flight station at ORELA. Below $E \sim 110$ keV about 40 s-wave resonances are observed which give a neutron s-wave strength function of $S_0 \approx 0.9 \times 10^{-4}$. Both the s-wave level density and neutron strength function below $E \sim 110$ keV for $^{204}Pb + n$ are about ten times larger than the respective quantities determined[1] for $^{206}Pb+n$. A possible explanation for the large difference in the relative strength functions for these two systems which have the same neutron separation energy will be discussed in terms of doorway states arising from particle-core excitations.

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*Physics Division.

1. D. J. Horen, J. A. Harvey, and N. W. Hill, *Phys. Rev. C* **20**, 478 (1979).

2.10.10. NEUTRON TOTAL CROSS SECTION AND RESONANCE PARAMETERS OF ^{231}Pa

Abdel-Razik Z. Hussein* **N. W. Hill**
J. A. Harvey* **J. R. Patterson***

[Abstract of *Nucl. Sci. Eng.* **78**, 370–76 (1981)]

Time-of-flight measurements of the neutron total cross section of ^{231}Pa were carried out, in the energy range 0.01 to 10 000 eV, on two sample thicknesses using the Oak Ridge Electron Linear Accelerator as the pulsed neutron source. The ^{231}Pa sample material was in the form of Pa_2O_5 from which two samples were made for the transmission measurements with thicknesses of 3.35×10^{-4} and 7.91×10^{-4} atom/b, respectively. Measurements were made for both thicknesses using an 18-m flight path and a neutron energy resolution of ~0.3%. Transmission data were also obtained on the thick sample using the 80-m flight path with an energy resolution of ~0.08%. The ^{231}Pa samples were cooled with liquid nitrogen to reduce the Doppler broadening of the resonances. The transmission data have been analyzed to obtain the resonance parameters for all observed resonances up to 120 eV. The multilevel *R* matrix code MULTI, which includes instrumental resolution and the Doppler broadening, has been used to fit the data. Resonance energies and neutron widths were determined for a total of 137 resonances. The radiation widths of 17 resonances below 12 eV were obtained based on a determination of the effective temperature of the sample from the analyses of resonances at higher energies where Doppler broadening is dominant. The average radiation width was determined to be 40 ± 2 meV. The average observed level spacing was computed to be 0.47 ± 0.05 eV for the resonances up to 23 eV. The s-wave strength function up to 70 eV is $(0.90 \pm 0.10) \times 10^{-4}$. Good agreement was obtained with earlier fast chopper data of ^{231}Pa resonance parameters in the 0.01- to 70-eV energy region. The neutron widths of the ^{231}Pa resonances are needed to determine the fission widths of the resonances from fission cross-section data and to reevaluate the neutron-induced reactions on this isotope.

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2.10.11. SOME SPECTROSCOPIC PROPERTIES OF FINE STRUCTURES OBSERVED NEAR THE ^{231}Pa(*n*, *f*) FISSION THRESHOLD

S. Plattard* **G. de Saussure†**
G. F. Auchampaugh‡ **J. A. Harvey****
N. W. Hill **R. B. Perez†**

[Abstract of *Phys. Rev. Lett.* **46**(10), 633 (March 9, 1981)]

The ^{231}Pa neutron-induced fission cross section from 140 to 400 keV was resolved into finer structures. For some of the fractionated vibrational resonances in this energy region, the assignment of spectroscopic parameters may support evidence for an asymmetrically deformed third minimum in the ^{232}Pa fission barrier. Also, for the first time, narrow fission resonances are observed above 1.3 eV exhibiting an average fission width $\langle \Gamma_f \rangle_{obs} = 8 \mu eV$.

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2.10.12. K COMPONENTS FOR THE ~1.4-, ~1.6-, AND ~1.7-MeV STRUCTURES IN THE FISSION OF ^{232}Th+n

G. F. Auchampaugh* G. de Saussure†
S. Plattard‡ R. B. Perez†
N. W. Hill J. A. Harvey**

[Abstract of *Phys. Rev. C* 24(2), 503 (August 1981)]

Neutron-induced angle-integrated fission cross sections of ^{232}Th were measured from 1.3 to 1.8 MeV with a nominal neutron energy resolution of 0.15 ns/m. Data were taken for the angular intervals 0° to 23.4°, 0° to 33.7°, 0° to 51.7°, and 0° to 90°. The structure at ~1.4, ~1.6, and ~1.7 MeV were interpreted in terms of rotational bands with K = 1/2, 3/2, and ≥5/2. The approximate relative fission strengths for the K bands are in the proportion 1.7:24:1.0, 0.0:2.6:1.0, and 1.0:2.8:0.0 for the three structures, respectively.

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2.10.13. MEASUREMENT OF NEUTRON TRANSMISSION SPECTRA THROUGH ^{232}Th FROM 8 meV TO 4 keV

D. K. Olsen* R. W. Ingle

(Abstract of ORNL/TM-7661, April 1981)

Neutron transmission spectra through room-temperature ^{232}Th samples have been measured using the time-of-flight technique, the ORELA pulsed neutron source and a 1-mm thick Li-glass detector. The measurement and data reduction are described in detail. The 40-m transmission spectra through eight samples are directly compared from 15 to 4000 eV with resolution-broadened transmission spectra calculated from the ENDF/B-V total cross section. Two sets of 22-m transmission spectra through five samples are combined into one total cross section from 0.008 to 15.0 eV and compared with the ENDF/B-V evaluation.

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2.10.14. PRECISE MEASUREMENT AND ANALYSIS OF NEUTRON TRANSMISSION THROUGH ^{232}Th

D. K. Olsen*
R. W. Ingle J. L. Portney†

(Abstract of Proceedings of the Conference "1980 Advances in Reactor Physics and Shielding," pp. 743–54)

Three sets of transmission time spectra through up to eight samples of ^{232}Th have been measured for neutron energies from 6.0 meV to 0.1 MeV using a flight-time technique over 22- and 40-m path lengths, the ORELA pulsed neutron source, and a 1-mm thick lithium glass detector. The resulting total cross section from 0.1 to 20.0 eV seems to be smaller than that contained in the ENDF/B-V evaluation. Least-squares analysis of the transmissions from 9 to 440 eV using a multilevel Breit-Wigner formalism results in neutron widths consistent with those previously reported. An average radiation width of 25.2 meV is obtained for 19 low-energy s-wave resonances.

*Engineering Physics Division.

†Summer Student, Oak Ridge Associated Universities, 1980.

2.10.15. THE RADIATIVE CAPTURE YIELD OF THORIUM-232 FROM 100 to 4000 eV

R. B. Perez*
G. de Saussure* J. Halperin†
R. L. Macklin‡ N. W. Hill

[Abstract of *Nucl. Sci. Eng.* 80(1), 189–98 (January 1981)]

The neutron capture yield in two ^{232}Th samples (0.0008 and 0.0027 atom/b, respectively) was measured with the Oak Ridge Electron Linear Accelerator time-of-flight facility over incident neutron energies from 100 to 4000 eV. A detailed

comparison of the measured capture yields with calculations based on ENDF/B-V resonance parameters suggests that above 500 eV the evaluation needs additional work; in particular, the average capture appears systematically underestimated.

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†Chemistry Division.
‡Physics Division.

2.10.16. NEUTRON CAPTURE CROSS SECTION OF NEPTUNIUM-237

L. W. Weston* J. H. Todd

[Abstract of *Nucl. Sci. Eng.* 7[illegible] 184 (1981)]

The neutron capture cross section of ^{232}Np was measured from 0.01 eV to 200 keV. The capture cross section was normalized at 0.0253 eV to a value of 180 ± 6 b derived from previous total cross-section measurements in the resonance region of neutron energies and the shape of the present data from 0.0253 eV to the resonance region. Resonance parameters were derived for the neutron energy region from 0.01 to 100 eV. Agreement with ENDF/B-V is poor in the thermal region (6.4%), excellent in the resonance region (~2%) except for the 0.491-eV resonance, and good (~5%) in the keV neutron energy region. An uncertainty analysis including a correlation matrix for group-averaged cross sections is presented. These results are important both for thermal and fast reactor applications and the calculation of ^{238}Pu production, an intense alpha emitter.

*Engineering Physics Division.

2.10.17. URANIUM-238 INELASTIC NEUTRON SCATTERING AT 82 keV

R. R. Winters* J. A. Harvey†
N. W. Hill D. K. Olsen‡
R. L. Macklin† G. L. Morgan**

[Abstract of *Nucl. Sci. Eng.* 78, 147–53 (1981)]

Using a thick iron filter to produce an 82-keV group of nearly monoenergetic pulsed neutrons from the Oak Ridge Electron Linear Accelerator white neutron source, the differential and integrated neutron inelastic scattering cross sections from the first excited state of ^{238}U have been measured. We find that the angular distribution is forward peaked, and we obtain estimates of the Legendre coefficients of P_0, P_1, and P_2. The measured integrated inelastic cross section is 381 ± 21 mb, in good agreement with the ENDF/B-V evaluation and with other statistical and optical model calculations.

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**Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

2.10.18. A MEASUREMENT OF THE AVERAGE NUMBER OF PROMPT NEUTRONS FROM SPONTANEOUS FISSION OF CALIFORNIUM-252

R. R. Spencer*
R. Gwin* R. Ingle

[Abstract of *Nucl. Sci. Eng.* 80, 603–29 (1982)]

The Oak Ridge National Laboratory large liquid-scintillator detector was used in a precise determination of $\bar{\nu}_p$, the number of neutrons emitted promptly, from spontaneous fission of ^{252}Cf. Measurements of the detector efficiency over a broad energy region were made by means of a proton-recoil technique employing the Oak Ridge Electron Linear Accelerator "white" neutron source. Monte Carlo calculation of the detector efficiency for a spectrum representative of ^{252}Cf fission neutrons was calibrated with these elaborate measurements. The unusually flat response of the neutron detector resulted in elimination of several known sources of error. Experimental measurement was coupled with calculational methods to correct for other known errors. These measurements lead to an unusually small estimated uncertainty of 0.2% in the value obtained, $\bar{\nu}_p = 3.773 \pm 0.007$.

*Engineering Physics Division.

2.10.19. MEASUREMENT OF THE AVERAGE NUMBER OF PROMPT NEUTRONS EMITTED PER FISSION OF ^{233}U RELATIVE TO ^{252}Cf FOR THE ENERGY REGION 500 eV TO 10 MeV AND BELOW 0.3 eV

R. Gwin*

R. R. Spencer* R. W. Ingle

(Abstract of ORNL/TM-7988, November 1981)

The energy dependence of the average number of prompt fission neutrons emitted per fission, $\bar{\nu}_p(E)$, has been measured for ^{233}U relative to $\bar{\nu}_p$ for ^{252}Cf over the neutron energy ranges 500 eV to 10 MeV and below 0.3 eV. A large Gd-loaded liquid scintillator was used to detect neutrons and the samples of ^{233}U and ^{252}Cf were contained in fission chambers. The present results for $\bar{\nu}_p(E)$ for ^{233}U are in accord with the experimental results of Boldeman and the evaluated results of Lemmel in the thermal energy range, but in the neutron energy region between 100 keV and 1 MeV the present data are 1% or more larger than other experimental values.

*Engineering Physics Division.

2.10.20. MEASUREMENT OF THE AVERAGE NUMBER OF PROMPT NEUTRONS EMITTED PER FISSION OF ^{235}U RELATIVE TO ^{252}Cf FOR THE ENERGY REGION 500 eV TO 10 MeV

R. Gwin*

R. R. Spencer* J. H. Todd

R. W. Ingle H. Weaver*

(Abstract of ORNL/TM-7148, January 1980)

The averge number of prompt neutrons emitted per fission, $\bar{\nu}_p(E)$, has been measured for ^{235}U relative to $\bar{\nu}_p$ for the spontaneous fission of ^{252}Cf over the neutron energy range from 500 eV to 10 MeV. The samples of ^{235}U and ^{252}Cf were contained in fission chambers located in the center of a large liquid scintillator. Fission neutrons were detected by the large liquid scintillator. The present values of $\bar{\nu}_p(E)$ for ^{235}U are about 0.8% larger than those measured by Boldeman. In earlier work with the present system, it was noted that Boldeman's value of $\bar{\nu}_p(E)$ for thermal energy neutrons was about 0.8% lower than obtained at ORELA. It is suggested that the thickness of the fission foil used in Boldeman's experiment may cause some of the discrepancy between his and the present values of $\bar{\nu}_p(E)$. For the energy region up to 700 keV, the present values of $\bar{\nu}_p(E)$ for ^{235}U agree, within the uncertainty, with those given in ENDF/B-V. Above 1 MeV the present results for $\bar{\nu}_p(E)$ range about the ENDF/B-V values with differences up to 1.3%.

*Engineering Physics Division.

2.11. COMMUNICATIONS: RADIO, CLOSED-CIRCUIT TELEVISION, AND COMPUTER

In the mid-1970s the Telecommunications Branch of the Department of Energy initiated a study of the Oak Ridge Operations radio networks. The purpose of the study was to determine compliance with existing Office of Telecommunications policy regulations and to evaluate the impact of proposed new regulations. A telecommunications consultant was engaged by DOE to study the needs of Oak Ridge Operations and its contractors and to prepare a master plan for adequate communications and compliance with regulations.

The consultant produced a Master Radio Plan, and in early 1980 the plan was accepted by the responsible telecommunications organizations. The consultant's services did not include a detailed design package that would meet existing and known requirements for the immediate future, which the contractors could implement. He was available only for consulting services and technical guidance. The UCC-ND Engineering Division was given design responsibility, and the I&C Division was asked to provide technical support on system design.

In the Master Plan, primary emphasis was given to radio frequency assignments, radio frequency channeling, and equipment compliance with frequency tolerances and levels of spurious emissions.

The consulting engineer selected network frequencies, obtained preliminary frequency approval, and prepared preliminary equipment requirement and cost estimates. After approval of the plan the need for additional voice privacy was recognized. Digital voice privacy capability was then expanded from four fixed locations on two networks to multiple fixed locations and mobile, plus portables on six networks. These changes complicated both the technical and financial aspects of the project. The need for additional voice privacy arose because individuals and organizations not affiliated with the government routinely monitor the Oak Ridge Operations radio networks. When unusual radio activities are intercepted they cause inquiries which create operational complications. Such unauthorized monitoring violates the privacy provisions of the Communications Act, but since these provisions are unenforceable, increased voice privacy capability is necessary.

The Master Radio Plan involves the radio networks used by DOE, three UCC-ND plants, several contractors, and one other government agency. A total of 15 networks are involved, with 6 repeater station locations. A large number of single-base stations, mobiles, and portables are assigned to each network. There are six multiple-base-station control locations.

I&C has participated in establishing equipment requirements, preparing specifications and bills of material, evaluating bids for equipment, and performing some of the activities originally delegated to the consultant. Significant areas of involvement included the following:

1. Identification of possible intermodulation and desensitization problems.
2. Designs of rf configurations to reduce these problems.
3. Designs of wireline controls.
4. Assistance in development of a work plan.

One very interesting task involved the determination of intermodulation problems using a computer program developed by I&C. The desensitization study involved obtaining manufacturers' data for spurious transmitter emissions, receiver sensitivity, combiner and multicoupler frequency characteristics, free space path loss, receiver input selectivity, and other factors. Using this information, filter networks were designed to reduce the degradation of service caused by the proximity of radio transmitters and adjacent frequency assignments. The use of filters reduces system performance on some networks because of the unavoidable attenuation of desired signals.

Another interesting activity still in progress is the development of a work plan for converting the frequencies of existing networks with minimum disruption of service. Although the job is complicated by the large number of units to be converted, the project is scheduled for completion in 1984. *(J. A. Russell)*

In addition, a job study and preliminary engineering has been completed for a broadband communications (Broadcom) system between X-10 and Building 9201-2 at Y-12. This work is in response to a need long recognized by Laboratory management for video teleconferencing, seminar relay, and an increase in the speed of remote computer terminals.

This project will provide a bi-directional, high-speed, broadband cable communications system between the ORNL facilities at X-10 and Y-12, and will in time replace the existing low-speed, narrowband, twisted pair cabling. The system will utilize proven, readily available commercial CATV components and have sufficient communications capacity to serve as a link in a possible future network encompassing the DOE Technical Information Center and K-25.

The Broadcom system will consist of two 0.75-in. coaxial cables of 75-Ω impedance equipped for 300-MHz bandwidth. The head-end or translator will be located in Building 4500 at X-10. Stations connected to the system will transmit on frequencies in the 5-MHz to 116-MHz range and will receive from the head-end on frequencies in the 156-MHz to 300-MHz range.

Electronic components will be provided for one cable only, for which a cascade of wideband CATV amplifiers will be used. The nominal output of the amplifiers will be +32 dbmv, and the nominal loss due to load division and line loss will not exceed 22 db before an amplifier is used to boost the signal a like amount. The purpose is to keep the amplifier performance optimized and minimize the deterioration of carrier-to-noise ratio, cross-modulation, second-order intermodulations, and composite triple-beat products.

This system is an FY 1982 GPP project, and it is anticipated that as funds permit the Broadcom cable will be extended to other buildings in Y-12 and X-10 to provide Broadcom services to users in more conference rooms, offices, and laboratories. *(J. L. Lovvorn)*

Closed-circuit television (CCTV) is also used many ways within X-10. In support of safe operation of hot cells at X-10, an rf transmission line/coupler system has been developed for the transfer of video and digital data signals along a crane rail within a hot cell. The device accommodates two video channels and a data channel from a crane-mounted manipulator and supplies a control data channel back to the manipulator. The system uses a slotted transmission line mounted stationary on the crane rail and a movable directional coupler pickup on the crane carriage. This communications system functions as a prototype for the larger systems planned for fuel recycle facilities.

In addition, a number of television systems have been evaluated for use in maintenance activities within a hot cell. The cameras tested include standard resolution monochrome, high resolution monochrome, and standard resolution color. Also, some evaluation and development was done for a three-dimensional television system. Various lens combinations and viewing angles were tested to define the system requirements. A flat-screen rear-projection television is being evaluated to determine if it offers any advantage over conventional monitors. *(R. I. Crutcher)*

Finally, the design and installation of conventional CCTV systems, as well as audiovisual systems is a support function of the I&C ivision. Many video systems have been designed and installed in conference rooms and auditoriums. The systems include TV monitors, video tape players, and sometimes large screen TVs. Audiovisual systems installed in auditoriums and conference rooms include projection screens, overhead projectors, microphones, amplifiers, and speakers. *(T. R. Barclay)*

In these ways, Research Instrument Section personnel experienced in the design of radio and TV communications systems are enabling DOE to maintain communications within ORNL, between ORNL and Y-12, and within the entire Oak Ridge Operations area.

Section 3

MEASUREMENT AND CONTROLS ENGINEERING

MEASUREMENT AND CONTROLS ENGINEERING

3.0. OVERVIEW

C. A. Mossman

The work of the Measurement and Controls Engineering Section covers an extremely broad spectrum which includes basic metrology research, original instrument development in a diversity of fields, and the application of all types of commercial instruments, controls, and digital computers. The Section has had many years of highly successful experience in the design, development, and implementation of instruments and control systems in a range of fields from biomedical analysis to nuclear fuels reprocessing. A very significant contribution has been made in the instrumentation, control, data acquisition, and analysis for a large number of electrically simulated nuclear reactor safety experiments and for the Atomic Vapor Laser Isotope Separation (AVLIS) program. The Metrology Research and Development Laboratory provides calibration services for all types of instruments for the entire Laboratory.

Growth, Diversity, Trends and Future Directions

Growth.—During the reporting period the Section staff peaked in numbers and began to decline because of limited hiring and competition from industry and academia for highly qualified staff members.

Despite this decrease in professional staff, the Section has grown in technical depth. The topics reported here reflect a measure of that technical growth, and attention is especially directed to the reports on the Electromagnetic Interference (EMI) Task Group activities, the Atomic Vapor Laser Isotope Separation (AVLIS) Program, and Distributed Microprocessor Control for Advanced Servomanipulators, which represent considerable extension and deepening of Section technical contributions. The demonstrated need for MACES engineers to apply their expertise in engineering and science to projects outside the traditional confines of X-10 indicates that these skills are basic to the overall Oak Ridge mission, and that the staff is both motivated and adaptable to diverse requirements.

Diversity.—For the first time ever, the MACE Section made a concerted effort to provide intensive measurement and control engineering support for UCC-ND projects and groups outside the X-10 Site. This effort was conceived primarily as a method of improving long-term retention of staff members by offering a greater amount and diversity of interesting and challenging work as well as additional opportunities for growth and advancement. In order to demonstrate that the Division has much to offer the other UCC-ND installations, instrument development support was undertaken on three projects. These initial projects were the Large Coil Test Facility in the ORNL Fusion Energy Program, the ORGDP Separations Systems Division centrifuge development project, and the Y-12 Atomic Vapor Laser Isotope Separation (AVLIS) process development project. Although the effort to provide expertise for these projects required stretching engineering resources severely, the work has had a healthy effect on morale because it has given the technical staff diverse, challenging work. Also, this increase in the workload has fulfilled another objective, that of insulating the Section from most of the budget uncertainties that have occurred.

Involvement in the AVLIS project has been especially challenging. MACES engineers have been involved in three major areas of work in this project: (1) The core program to develop methods and hardware for future pilot plant stages, (2) EMI investigation to ensure that data gathered in the experiments is free of EMI contamination and adequate to permit accurate control, and (3) development of instruments and distributed computer control for the material-handling development module (MHDM). The MACES staff have become highly valuable contributors to these projects in the few months they have been involved.

Trends and Future Directions.—A two-year report cycle represents an appreciable portion of the current engineering knowledge half-life. It is thus possible to see many effects of the very rapid pace of technological advancement within this report period. Also, measurement and controls technology is one of the first areas to take advantage of new devices and techniques. The Section has encouraged the staff to study and apply appropriate portions of new technology in its work.

New Areas of Development and Former New Areas, Now Routine

Computers.—Both large minicomputers and microcomputers continue to challenge staff ingenuity with new hardware and new software. The advent of the 16-bit microcomputers has produced excitement about the tremendous increase in computer power. At the same time the lack of available software for the new 16- and 32-bit micros has been frustrating. However, it is anticipated that good operating systems and applications programs will be available soon.

Robotics and Artificial Intelligence.—In this endeavor it has been a challenge to broaden concerns to include vision and pattern recognition. The work has also resensitized the staff to the needs of human operators for perception aids and information filters. One outgrowth of the robotics work has been a study of how artificial intelligence research may help provide solutions to problems that have proven intractable up to now.

Sensors, Effectors, and EMI.—New sensors, based both on new integrated circuit technology and on innovative application of old principles, continue to provide opportunities for improved process measurement and control. At the other end of the sensor-effector spectrum, larger power control devices are increasingly being applied in ways that ignore their EMI consequences. The resulting compromise of measurement and control signals has required mounting a significant effort for EMI analysis, test, and control.

Simulation, Modeling, and Analysis.—The ever-increasing cost of doing research has focused attention on and renewed the commitment to improved simulation, modeling, and analysis work. An increase in this work has in turn called attention to the use and availability of tools for this area. Interest in high-resolution computer graphics and the benefits promised by computer-aided design machines has been stimulated, and these techniques will continue to be tested and evaluated.

Implementation of New Technology.—The pace of new developments has made it necessary to devise methods for making former frontier technologies more routine. For example, programmable controllers are now considered a standard application tool. Standardized designs have been built, and standardized programming and documentation procedures have been devised. The same is true for the 8-bit microcomputers that are now routinely embedded in many of the instruments developed.

Business Systems and Support Efforts

Business Systems.—This report period has seen many of the methods and systems developed by this Section for job cost accounting, staff time accounting, and word processing adopted throughout the I&C Division. Many of the technical staff have begun routinely using computer terminals for generating draft reports and correspondence. These documents are then accessed by the Section secretaries or by the Information Processing Center, either directly from the computer files or via flexible disk media, for editing and final printing. The lack of common file and communications structures is presenting only minimal impediments to pursuing the full "Office Of The Future" operation described in numerous recent magazine articles. The goals of this Section include the achievement of the necessary common formats and protocols, but progress has been slowed by a lack of available staff to work on this function.

Support Efforts.—Much of the technical effort of the section is involved in the support of the overall I&C Division research effort. Occasionally this support is extended to other areas of the ORNL system, which may themselves be support functions. Such is the case for the significant effort now being provided in the medical area. For several years, because of its real-time computer systems expertise, this Section has handled the electrocardiogram acquisition and analysis screening system for the four-plant employee medical service. This support, expanded to cover employee medical records and clinical laboratory test results, is summarized elsewhere in this report.

Educational Support.—It is believed that many types of support make good sense for the overall research effort and for the mutual benefit of employees and the organization as a whole. An example of this philosophy is seen in the educational functions being pursued. Two programs are underway: The Engineering Co-Op Program for undergraduates, and a new effort, Measurement and Controls Engineering graduate studies for staff members. The Co-Op Program is successfully running at the level of 7+ students, but difficulties have been encountered in getting the graduate studies program under way.

The graduate studies program was initially conceived as a means of upgrading the staff and retaining the young top-performing engineers who had been inclined to take university research assistantships in order to facilitate their further education. In response I&C joined with administrators and faculty of The University of Tennessee in developing a joint ORNL-University Graduate Program in Measurement and Controls Engineering. The program is in its second year and is operating on a limited basis as part of the UCCND-ORAU Graduate Studies Program. It is hoped that the program can be expanded and that more flexible arrangements for in-house graduate studies can be developed in future years.

Section Organization

The activities of the section are carried out by six working groups and three major programs which report to section management:

Real-Time Computer Systems Group.—The objective of the Real-Time Computer Systems Group is to provide the research, development, and application of computer-based systems for real-time measurement, control, and information processing necessary to meet ORNL on-line experimental data requirements. This group performs functions that encompass all facets of hardware and software development from definition of requirements and concepts through equipment procurement or fabrication and software implementation.

This group presently uses off-the-shelf minicomputer technology to build on-line real-time data acquisition and analysis systems for all types of experimental work throughout ORNL. Data systems have been built for reactor safety research experiments, aquatic ecology research, and bioinstrumentation research, to name a few. This group has the expertise to do complete computer system jobs; its state-of-the-art expertise in low level and high speed analog input systems is especially noteworthy.

Instrument Development Group.—The objective of the Instrument Development Group is to perform instrumentation systems research, development, and application engineering. Special measurement systems are developed over the entire range of classical physical phenomena in virtually all physical regimes.

This group is presently performing development work in several major areas, including enriched uranium recovery, diagnostic instrumentation for centrifuge development, instrumentation and control for the AVLIS program, gas composition measurement and control for the Core Support Performance Test (CSPT), special instrument development for the AIRS program, and many others. The group has lead responsibility for embedded microprocessor expertise within the Section and also is playing a key role in robotics development in nuclear fuel reprocessing.

Measurements Research Group.—The basic objectives of the Measurements Research Group are to improve and extend sensor and instrument ranges through research and development and to provide instrument and measurement evaluation, calibration, and consultation services.

This group performs continuing activities related to sensor evaluations and development. This work encompasses a wide range of physical properties measurements, including (1) thermometry, (2) pressure and vacuum, (3) heat and energy, (4) fluid flow, and (5) liquid level and density. Sensor development and evaluation covers all aspects of sensor performance and implementation considerations such as materials, fabrication methodology, error sources, and failure modes.

Process Systems Development Group.—The overall objective of the Process Systems Development Group is to provide systems engineering in the development, application, and support of measurement and automatic control systems for physical process systems in concert with the research and development needs and goals of ORNL. The functions necessary to meet this objective encompass problem definition, concept synthesis, design and implementation, startup, and follow-on maintenance.

This group uses commercially available hardware and software where possible as a basis for building measurement, control, actuator, and operator interface subsystems throughout ORNL. The majority of group staff members presently perform work for the Consolidated Fuel Reprocessing Program. Other, smaller efforts in support of the Enriched Uranium Recovery and AVLIS Programs are also continuing.

Process Systems Applications Group.—The objective of the Process Systems Applications Group is to perform instrumentation system research, design, development, and application engineering in support of a wide variety of projects being conducted by the Engineering Technology Division of ORNL. To meet this objective expeditiously, the group is composed of professionals and engineering assistants working in the Y-12 area, the site of the Engineering Technology Division's principal projects, and representing a broad spectrum of I&C expertise.

The group performs instrument engineering work in support of projects such as Thermal Hydraulic Test Facility, Core Support Performance Test, Pressurized Thermal Shock Test Facility, Section Steel Technology, Multi Rod Burst Test studies, and Large Coil Test Facility.

Measurement Systems & Analysis Group.—The objective of the Measurement Systems and Analysis Group is to facilitate the use of modern analytical methods by the section staff. Computer simulation of process dynamics; finite element analysis of mechanical, fluid, or electrical systems;

and spectral analysis of sensor signals are examples of analysis techniques that are frequently useful in the section's work.

In many cases the software and analytical techniques are available from various industries or have been developed by staff members on previous jobs. The main function of the group with regard to analytical methods is to obtain and make available the tools so that they can be used conveniently.

The group is also responsible for instrumentation and systems engineering support of experimental and research projects conducted by other divisions, as well as the routine engineering services required by Laboratory operations, utilities, and radioisotope processing facilities.

Atomic Vapor Laser Isotope Separation Program.—The instrumentation-related objectives of this program are to provide measurement, control, data acquisition, and analysis for all appropriate process parameters, primarily those concerned with materials handling issues such as injecting uranium into the process, generating an atomic vapor with acceptable properties, separating product from tails, and withdrawing product and tails from the process.

Work is now in progress on three research vessels which together comprise the Core Program. Longer range work and planning are in progress for the Materials Handling Demonstration Module (FY 1984) and the Development/Demonstration Module (FY 1989).

Consolidated Fuel Reprocessing Program.—The overall instrumentation-related objective of this program is the development of an improved technology base for instrumentation, control, and data or information handling functions associated with all aspects of nuclear fuel reprocessing.

Long-range plans for this program have been developed and work is in progress for the development of: (1) improved process control and information management techniques using digital technology, (2) new remote process measurement systems, and (3) automated (robotic) remote maintenance viewing and handling systems. Improved operator interfacing concepts using video terminals and colorgraphics are being studied and implemented. Successful research in in-line analytical instruments will have significant impact on process operations and safeguards.

Advanced Instrumentation for Reflood Studies (AIRS) Program.—The objective of this program is the development and implementation of two-phase flow measurement systems to be used in joint U.S., Japanese, and West German pressurized water reactor safety research, under the sponsorship of the U.S. Nuclear Regulatory Commission. This program is managed by the I&C Division and is supported by significant work in the Engineering Technology Division and Metals and Ceramics Division of ORNL and by UCC-ND Engineering.

Instrumentation for measuring velocity and void fraction in water/steam two-phase regimes is undergoing concurrent development, fabrication, and delivery to West German and Japanese installations. The development phase is expected to continue through 1983, with last hardware deliveries and on-site support extending to 1988. Additional work investigating special instrumentation problems occurring in the overall joint program is also being performed.

3.1. Atomic Vapor Laser Isotope Separation (AVLIS) Program

3.1.0. OVERVIEW

D. W. McDonald

The AVLIS program was chosen by DOE in April 1982 for continued funding as the advanced isotope separation process which offered the greatest likelihood of technological success combined with potential for significant cost reductions over competing processes. Prior to selection AVLIS was one of three competing advanced isotope separation technologies. All of these processes were supported by UCC-ND. The Molecular Laser Isotope Separation (MLIS) process was under the direction of Los Alamos National Laboratory and involved the separation of uranium in the molecular gaseous state (UF_6). The Plasma Separation Process (PSP) was under the direction of TRW and involved atomic vapor excited by rf fields in a very strong magnetic field. The Atomic Vapor Laser Isotope Separation (AVLIS) process is under the direction of Lawrence Livermore National Laboratory (LLNL) and involves the selective photo-ionization of ^{235}U in the atomic vapor state.

The AVLIS effort at UCC-ND is primarily concerned with material handling issues, i.e., injecting uranium into the process, generating an atomic vapor with acceptable properties, separating product from tails, and withdrawing product and tails from the process. This effort is presently carried out in three research vessels which together comprise the Core Program and which are referred to as MTU, STU, and EB-I. The Materials Test Unit (MTU) is used to address material compatability issues, the Source Test Unit (STU) is used to address vapor generation issues, and the EB-I facility is used as the integrating facility addressing all material handling issues in a single system. The contribution of MACES engineers in support of the Core Program is summarized in a section below.

A new plant-scale facility is under construction at the K-25 site for operation by UCC-ND and is scheduled for startup in the first quarter of FY 1984. The Material Handling Development Module (MHDM) will be used to develop plant-viable hardware, operation philosophies, and control and measurement techniques related to material handling problems. MACES support for this facility is summarized below.

All AVLIS research and development efforts are directed toward developing a baseline design for the Demonstration/Development Module (DDM), a laser system and material handling system integrated into a plant prototypic system capable of full-scale enrichment and throughput. It will be used to verify performance projections, refine operation philosophies and control techniques, develop maintenance procedures and strategies, and essentially complete the transfer of AVLIS technology from the development phase to the operational environment. The DDM is expected to be a FY 1984 line item, with startup scheduled for FY 1989. MACES contributions to this effort are also described below.

3.1.1. THE INTEGRATED CONTROL AND MEASUREMENT (ICAM) SYSTEM FOR THE AVLIS PROCESS

W. W. Manges

The development of the Integrated Control and Measurement (ICAM) System for the Atomic Vapor Laser Isotope Separation (AVLIS) process represents a logical progression in the use of computer-based instrumentation in a complex experimental facility. As the MACES engineers on the program became familiar with the processes involved, it became apparent that an integrated instrumentation system was needed. The integration of what are traditionally referred to as the "local instrumentation" and the "data acquisition and control system" (DACS) has led to a philosophy that can support the complexity of the process and expand to meet the requirements of the anticipated scale-up to the plant.

Anticipating the problems of an integrated approach, including the interaction required between the instrumentation engineer and the computer system development engineer as well as the reliability issues of utilizing computer systems in local instrumentation, we began the careful development of an overall instrumentation philosophy. Since the program managers are interested in the long-term cost and efficient operation of the facilities being planned, they welcomed our interest in understanding the entire process so that we could integrate the instrumentation effectively. This has helped us get involved in the decisions that are necessary to develop and implement an overall instrumentation philosophy. Since the program is currently involved in a laboratory-scale system with plans for a full-scale prototype and finally a demonstration/development facility before the plant-scale facility is built, we have the opportunity to test our concepts on smaller and less expensive systems before making the more expensive step to a plant-wide system. The goal is to have an integrated control and measurement system available for the Materials Handling Development Module (MHDM) currently being constructed at the K-25 site. The system being developed for the EB-I laboratory upgrade permits the testing of some of the concepts being developed for MHDM. The lower levels of the hierarchical system are being implemented in the EB-I experiment area, with additional levels being added as we progress through the MHDM. Final refinements to the ICAM system are anticipated during implementation of the Development/Demonstration Module (DDM) with complete utilization available for the plant.

3.1.2 AVLIS CORE PROGRAM INSTRUMENTATION TASK

M. L. Bauer
R. W. Tucker, Jr. W. W. Manges

The Core Program of the AVLIS project exists to develop methods and hardware for future utilization in prototypic pilot plants. The instrumentation task involves upgrading the data acquisition and analysis capability of the present facility, as well as developing advanced instrumentation for future analytical and operational needs.

Special Experimental Support

As part of the I&C effort, we have been supplying instrumentation expertise to the AVLIS program. This has involved building special-purpose instruments, improving the design of existing equipment, and system integration. An example of a newly designed unit is the E-Beam (Electron Beam Gun) Emission Controller for controlling the power produced by the E-Beam system. Another unit was an isolation amplifier for reading thermocouples at elevated voltages. Because of the high voltages and the accuracies needed at high temperatures, special efforts had to be made to account for errors and drifts in the system. A continuing effort has also been put into improving existing equipment such as the regulator (Tetrode) carts used to regulate E-Beam voltage, and improving flow instrumentation to enable better calorimetry on the melt crucible for source efficiency experiments.

Instrumentation Upgrade

A major effort in support of the Core Program has been the upgrade of the measurement and control systems for the MTU, STU, and EB-I facilities. The three primary goals of this effort are to increase the operational reliability of the facilities, to increase the integrity of the experimental data collected, and, where possible, to test and prototype concepts for future AVLIS facilities. While improvements have been made to each facility, the primary emphasis has been on EB-I, which provides the best environment for the development of an integrated control and measurement system for the AVLIS process.

In addition to the application of appropriate sensors and experienced field engineering, the upgrade of the EB-I instrumentation has reduced the prominence of traditional panelboard instrumentation in favor of a "soft" operator's interface. The use of a programmable controller for discrete control functions has allowed further integration of the instrumentation with the data system, as well as provided for the automation of some routine system functions. All sensors and signal conditioners have been carefully chosen to maintain the overall system philosophy.

Integrated Control and Measurement (ICAM) System

The EB-I experiment offers the first opportunity to implement some of the philosophy developed for the ICAM system for the AIS program. This implementation represents the first level, process interface, of the hierarchy. However, because no additional levels will be implemented in this installation, some of the functions performed at the higher levels must also be performed at this level. Therefore the touch panel user interface as well as some CAMAC process interface concepts and hardware will be tested here. A memory mapped graphics processor is another area where some new hardware implementations are being tried. The experience and feedback obtained during this initial implementation supplies valuable input into the overall philosophy for support of the AVLIS process.

Electromagnetic Interference (EMI) Investigation

A study was undertaken to characterize the EMI problems associated with the Core Program vessels. Measurements were made of radiated as well as conducted EMI, and the interference levels were recorded for both steady state operation and transient, arc-down operation. Steady state EMI was not found to be a problem for any of the signals presently used, but the transients created by arcs were very large in some cases. Recommendations were developed for ways to reduce the EMI produced by arc-downs, and a report was written detailing these ideas. Other suggestions included techniques for the treatment of low level signal lines as well as concepts for use in future systems, e.g., the use of three-phase power on the heaters and filament power supplies. *(W. H. Andrews)*

3.1.3. MATERIAL HANDLING DEVELOPMENT MODULE (MDHM) INSTRUMENTATION TASK

D. W. McDonald

MHDM will be used to develop material handling hardware, to demonstrate thermal control, to demonstrate liquid uranium flow, to develop a reliability data base, and to develop operations procedures and personnel training for the DDM. The instrumentation task is committed to assuring the integrity of data, assuring safe and reliable operation, maintaining flexibility for future development, utilizing plant viable methods, and demonstrating integrated control. The operating environment associated with this facility is extremly harsh to instrumentation. Sensors inside the module must be compatible with a high vacuum environment, operate at high temperatures, and be able to withstand uranium vapor deposition. Outside the vessel the instrumentation must withstand the electromagnetic interference generated by arcing of the electron beam system. The AVLIS I&C task team has generated an AVLIS I&C philosophy document which addresses these and other special problems associated with the AVLIS process. This document sets the framework for an integrated control and measurement system. MHDM will represent the fist totally integrated (from sensor to mainframe computer) control system developed by MACES. *(A. F. Johnson, M. L. Bauer, R. W. Tucker, Jr., W. W. Manges)*

MHDM I&C Design

The MHDM ICAM system is being implemented through close interaction between MACES engineers and the UCC-ND Engineering Division. The cooling system, vacuum system, magnetic field generating system, and uranium injecting system are complete and are compatible with the AVLIS I&C philosophy. Thus the data from every sensor and the status of every switch, pump, and valve is available to the data system for display on the operator consoles. Every signal entering the control room is either over an optical fiber link or an isolated current loop. This mitigates the effects of field-generated electromagnetic interference. *(D. W. McDonald, R. W. Tucker, Jr., W. W. Manges)*

MHDM Thermometry Development

The severe operating environment within the MHDM vessel and the accuracy requirements necessary for thermal control require that special consideration be given to the thermal sensor. An investigation of several candidate sensors and life tests performed on them under simulated AVLIS conditions led to a recommended sensor. *(R. L. Anderson)*

MHDM Tetrode Control Development

The AVLIS process employs electron beam systems to generate atomic vapor. These high voltage systems arc due to process-generated phenomena. MACES engineers were asked to redesign the control electronics for the high voltage regulator (tetrode) to allow it to better accomodate arcs without introducing deleterious transients into the process. The new control electronics have demonstrated improved stability, lower overshoot, faster response, and enhanced tetrode regulation. *(M. L. Bauer, W.H. Andrews, J. T. Greer*)*

*AIS Division.

MHDM Programmable Controller (PC) Specification

The ICAM system requires that all PCs be integrated into the overall system. This demands that special functions and capabilities be programmed into the PC over and above those required for its sequential control functions. Developing a system specification and being able to communicate that specification to others can become a problem rivaling that of data system specification. Therefore a modified version of the YOURDON method of data system specification has been adapted to the specification of PC systems. This allows a symbolic representation of our PC system for review and documentation purposes. *(D. W. McDonald, A. A. Shourbaji)*

3.1.4. AVLIS DEMONSTRATION/DEVELOPMENT MODULE (DDM)

D. W. McDonald

A significant effort was exerted in generating the instrumentation and power conditioning portions of the Conceptual Design Report (CDR) for the AVLIS DDM. The CDR is the basis for the cost estimate for a full-scale AVLIS demonstration facility, and it was one of the factors considered in the selection of the AVLIS process over competing processes. The CDR incorporates the instrumentation philosophy developed from experience with the AVLIS process and provides direction for current development, design, and prototyping activities. Continuing support will be provided as the CDR is updated to reflect refinements in the process.

3.2. Gas Centrifuge Enrichment Technology Support

3.2.0. OVERVIEW

N. C. Bradley

Instrumentation and Controls Division participation in this area began in June 1981 with staff members assisting UCC-ND Engineering in the completion of prototype facilities being built for verification of design concepts for the Gas Centrifuge Enrichment Plant to be located at Portsmouth, Ohio. Expertise in the design and development of real-time computer-based data systems and instrumentation for the measurement and control of plant processes was applied to help resolve problems which were impacting on the timely completion of these facilities. Attention was given also to the assessment of the effects of electromagnetic interference (EMI) on the operation of plant components.

Among the work activities in which I&C personnel were involved, several typical tasks are reported below.

3.2.1. REAL-TIME COMPUTER SYSTEMS APPLICATIONS

J. T. Hutton R. M. Tate

There was a need to interface a computer-based data acquisition system with microcalculator-controlled mass assay systems at the Centrifuge Plant Demonstration Facility (CPDF). Principal activities included (1) determination of a method for execution of the assignment, (2) preparation of software documentation capable of transmitting data between the mass assay systems and the data acquisition system, and (3) implementation and demonstration of the system. A serial data link was selected to carry mass assay data from the output of Hewlett-Packard desktop computers, which control and monitor the mass assay systems, to the facility computers. The CPDF computer system was upgraded to provide for expanded data acquisition capacity and to develop switchover capability between the two main CPDF computers. Detailed user notes describing the software for the desktop computer/mass spectrometer system were written. The interface software was satisfactorily tested to demonstrate the capability of the system to insert mass spectrometer data into the supervisory computer data base.

3.2.2. CENTRIFUGE ACCELEROMETER

W. H. Andrews
R. E. Hutchens D. R. McNeilly

An instrument was designed to measure the acceleration/deceleration of a gas centrifuge. Using a one pulse per revolution signal from the machine's internal tachometer, the instrument provides digital readout of the rate of change of rotational speed with a resolution of 1×10^{-5} revolutions per second per second and updates the reading every 10 s.

3.2.3. PHASE MEASUREMENT IN HIGH NOISE ENVIRONMENT

D. R. McNeilly

A microprocessor-based phase monitoring instrument was developed to measure the phase relationship between two parts of a rotating system. The instrument developed was required to extract information from signals having signal-to-noise ratios as low as 1:80. A patent disclosure has been filed on the unique lock-in amplifier designed for this application.

3.2.4. MACHINE DIAGNOSTIC UNIT

D. R. McNeilly

A new-generation microprocessor-based diagnostic unit was developed to replace an earlier instrument of limited capabilities. The new instrument provides a wide variety of detailed information about the condition and operation of a centrifuge. A front-panel keyboard is provided for operator interaction.

3.2.5. ELECTROMAGNETIC INTERFERENCE (EMI) STUDIES

W. H. Andrews
J. L. Horton M. J. Roberts

Special test equipment components were purchased, screen-room equipment tests were conducted, and on-site surveys were performed to characterize the performance of centrifuge control equipment with regard to electromagnetic interference (EMI) susceptibility and generation. Development has begun on the software for the control of EMI testing hardware during standard emission and susceptibility tests.

Technical assistance has been provided to UCC-ND Engineering in the review of a series of reports from an outside contractor describing evaluation studies of EMI problems to be expected in the Portsmouth Gas Centrifuge Enrichment Plant (GCEP). Assistance has also been provided in the evaluation of vendor-supplied preliminary radiated emission test data on equipment for the GCEP.

3.3. Advanced Instrumentation for Reflood Studies (AIRS) Program

3.3.0. OVERVIEW

M. B. Herskovitz   B. G. Eads

The AIRS Program is a part of a large international research and development program to evaluate loss-of-coolant accidents (LOCA) in pressurized water reactors. The United States (U. S. Nuclear Regulatory Commission) provides advanced instrumentation and analysis, and The Federal Republic of Germany (F.R.G.) and The Japanese Atomic Energy Research Institute (JAERI) provioe experimental facilities and conduct tests to increase understanding of reactor behavior and to improve codes that describe the accident. ORNL has the responsibility for developing new instruments to measure void fraction, flow velocity, liquid film thickness and velocity, mass flow between upper plenum and core, breakthrough of water from the upper plenum into the core, and fast differential pressure response above and across the reactor tie plate. All of the testing is performed in a non-nuclear environment. The program is managed by the Instrumentation and Controls Division and is supported by significant work in the Engineering Technology Division and Metals and Ceramics Division of ORNL and by UCC-ND Engineering. Other participating U.S. National Laboratories are Idaho National Engineering Laboratory, Los Alamos National Laboratory, and Sandia National Laboratory.

The ORNL program started in late 1977 with a simultaneous development and production effort to meet the initial instrument delivery to meet the F.R.G. PKL-II Facility requirements in 14 months. Key items in the early completion of instruments for the F.R.G. PKL-II facilities included the development of a direct braze alloy for fastening ceramics and metals, a platinum-based cermet resistant to thermal shock, design of sensors, and the development of sensitive electronics to measure small capacitance changes (10 femtofarads). The design criteria called for instruments to operate in steam-water environments at temperatures up to 900C and under severe transients of −300°C per second. All of the effor is performed on a best-effort basis.

Instrumentation systems have been installed in the F.R.G. Primar Kreislauf (PKL-II) Test Facility, the Japanese Slab Core Test Facility-I (SCTF-I), and the Japanese Cylindrical Core Test Facility-II (CCTF-II) with mixed results. Early data were obtained from SCTF-I; but, due to cable failures, difficulties in obtaining data were experienced later. Data have been obtained from both the PKL-II and CCTF-II and are now being analyzed.

In another F.R.G. facility, the Upper Plenum Test Facility (UPTF), ORNL has the responsibility for providing measurement systems to evaluate mass flow from the upper plenum into the core. This facility is a full-size model of the upper plenum of a typical pressurized water reactor. ORNL developments include the following:

1. Fast response pressure drop measurement systems,
2. A non-intrusive steam- and water-force measuring, strain-gauge-based system called the Tie Plate Drag Body Measurement System, and
3. Another strain-gauge-based system, known as the Breakthrough Detector, installed below the tie plate to measure the passage of water from the upper plenum into the core region of the facility reactor vessel.

Approximately 200 strain gage amplifiers, known as AIRS 20, will be supplied.

This program is unique in that the Laboratory is responsible not only for the development, design, and fabrication of instruments but also for supervision of the installation in the field, participation in shakedown tests, and analysis of data from both the F.R.G. and Japanese facilities. We further have the responsibility for developing the mass flow algorithm. This aspect is both interesting and challenging, as the implementation of software routines is subject to the uniqueness of the machine language instruction set of the data system used at different locations.

All work must be coordinated with the project schedules of both the F.R.G. and Japanese facilities. As of this writing, ORNL has met all project commitments.

3.3.1. MASS FLOW INSTRUMENTATION FOR AIRS

J. E. Smith[*] T. M. Cate
D. G. Thomas[*] B. A. Denning
H. R. Payne[†] R. A. Hess
L. V. Wilson[†] D. R. McNeilly

Tie Plate Drag Body Transducer.—In order to measure mass flow of steam and water between the upper plenum and the core, a non-intrusive drag body was developed using existing hardware, specifically the tie plate, as the drag element. To measure the force on the tie plate, a strain gauge transducer was designed to be enclosed in existing end box hardware (Fig. 3.3.1). The strain gauges were of the weldable type with special configurations and options to minimize thermal and lead wire effects. Prototypes were designed and fabricated for use in the Instrument Development Loop (IDL). During this period the units were calibrated and successfully tested under a wide variety of single- and two-phase flow conditions. New end box internals have been designed for use in the UPTF and SCTF-III experiments. Forty units will be fabricated in FY 1983.

Breakthrough Detectors.—An instrument to detect water breakthrough from the upper plenum into the core region was needed for the UPTF experiments in F.R.G. In response to this need, an instrument concept, based on momentum flux data obtained during drag body testing in the IDL, was proposed and accepted by NRC. A drag flag transducer (Fig. 3.3.2) was designed to be mounted just under the end box to measure momentum flux of fluid passing through two holes in the tie plate. A strain-gauge sensor with electronic characteristics similar to those of the tie plate drag body was incorporated into this unit. Prototypes have been built and are being tested.

Electronics for Tie Plate Drag Body and Breakthrough Detectors.—In order to interface the strain-gauge-based transducers described above to the unique requirements of the F.R.G. and Japanese facilities, a special instrumentation package was designed to provide remote excitation control, calibration capability, and amplification. The provision for dual-buffered outputs was an additional requirement. Three prototypical units have been fabricated and tested and the final configuration is now being designed.

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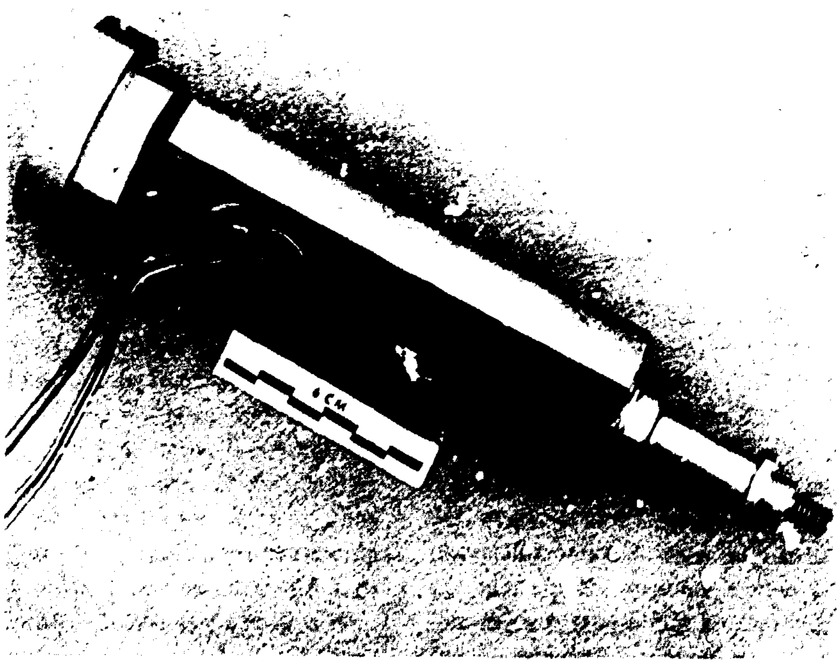

Fig. 3.3.1. Tie plate drag body transducer.

ORNL-DWG 83-8219

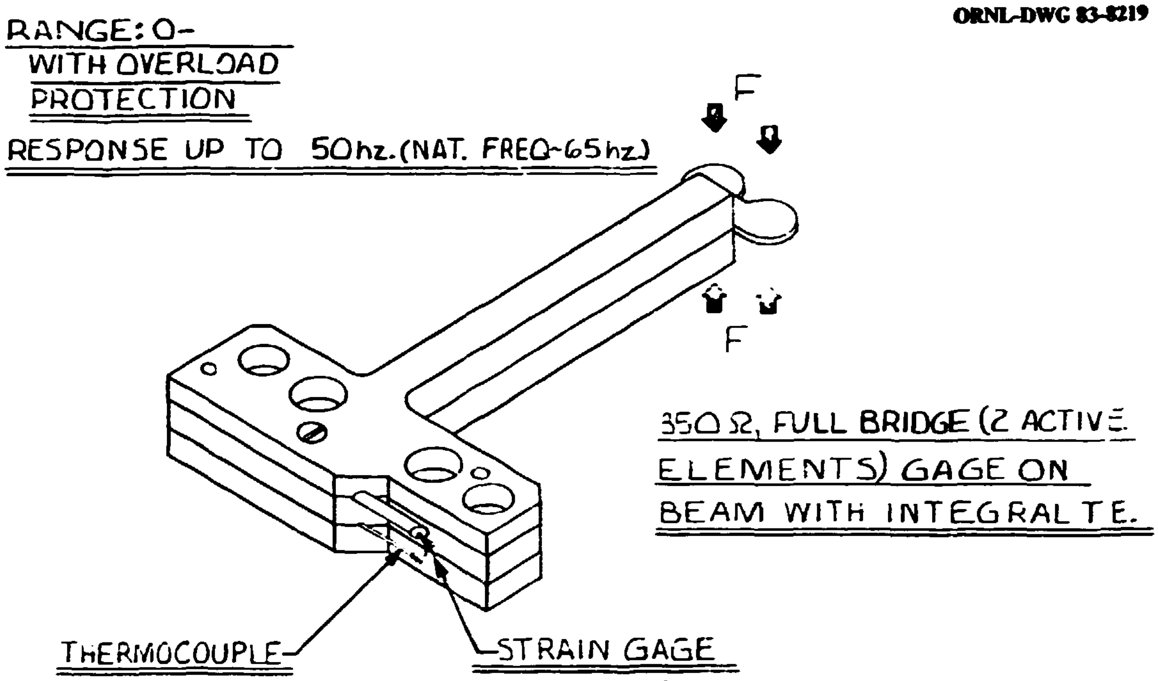

Fig. 3.3.2. Breakthrough detector.

Electronic transduction equipment is being developed, such that one electronic design can be used for both the Tie Plate Drag Body and the Breakthrough Detector.

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*Engineering Technology Division.

†UCC-ND Engineering Division.

3.3.2. DEVELOPMENT OF FILM THICKNESS AND VELOCITY PROBES

W. F. Bethmann

R. A. Hess C. J. Remenyik*

In response to a loss-of-coolant accident (LOCA) in a water-cooled nuclear reactor, the reactor is reflooded with water from standby storage tanks to carry off heat that would otherwise damage fuel rods, vessel walls, and other structures. During the later phases of the reflooding, a mixture of steam and water flows between the structures and a flowing film of water adheres to the surfaces. To gain some knowledge about the behavior of these films and to aid heat transfer calculations in designing reactors, engineers have developed instruments for the measurement of liquid film thickness, flow velocities, and mass flow rate in reactors and their simulators. Basic concepts for these instruments were developed at Lehigh University by Dr. John Chen and coworkers and adapted at ORNL to operate in high temperature steam and water environments and in the presence of high levels of electromagnetic interference.

The film thickness probe is one such instrument designed to measure the thickness of the condensate film on the structural surfaces (Fig. 3.3.3). The principle of operation of this type of probe is based on the measurement of the electrical admittance of the fluid in the vicinity of a set of electrodes. The magnitude of the electrode-to-electrode admittance and electrode-to-ground admittance varies with the thickness of the liquid film passing over it. The ratio of the two signals effectively cancels out all water properties and is a function of film thickness and sensor geometry only. Tests show that the output from the sensor electronics is stable and reproducible for film thicknesses greater than 0.1 mm and less than 8.0 mm.

The electrolysis potential probe is another instrument designed to measure the average liquid velocity inside the water films (Fig. 3.3.3). It was

ORNL-DWG 83-8220

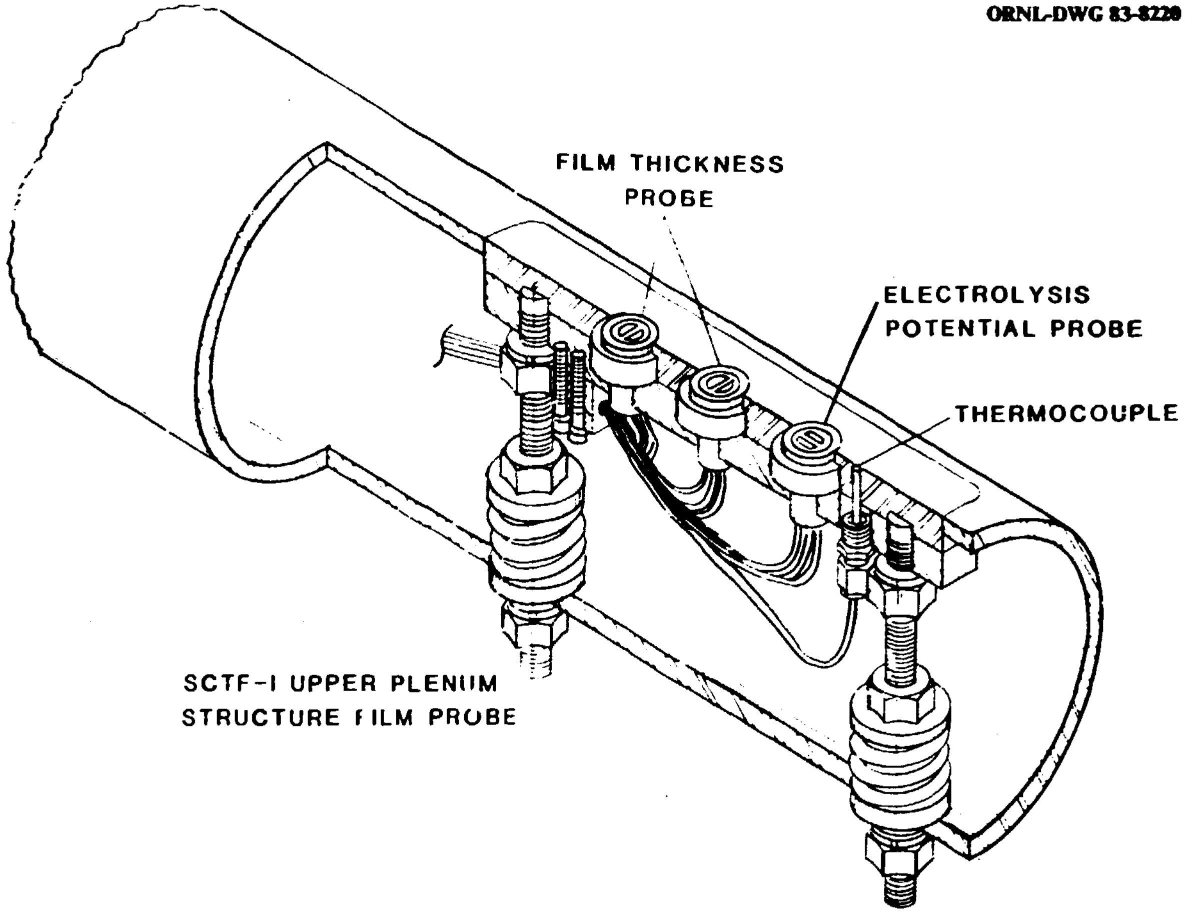

Fig. 3.3.3. Typical film probe assembly.

discovered by Chen of Lehigh University that if two electrodes were placed in flowing water and excited by a dc voltage of sufficient magnitude to produce visible electrolysis, the electric current through the sensor decreased with increasing water velocity. This measurement is dependent upon the conductivity of the water and the thickness of the film passing over it and is sensitive to the ion concentration, temperature, dielectric constant, diffusion coefficient, and various hydrodynamic parameters. Laboratory testing under carefully controlled conditions indicated that film velocities in the range expected in LOCA tests could be measured. Preliminary results, however, from the LOCA facilities have not been encouraging because of the inability to compensate for the sensor's high sensitivity to local ion concentrations in the film.

Another concept for flow velocity is the measurement of the wave velocity on the flowing film. This is achieved by the use of two film thickness probes, one downstream from the other. A cross-correlation technique is used in which the signals from each sensor are converted into a fast Fourier Transform and a delay time obtained from the two functions along with a coherence value. Since the distance between the two probes is known along with the delay time of the signal, the wave velocity becomes the result.

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3.3.3. SOFTWARE DEVELOPMENT FOR AIRS INSTRUMENTATION

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The data reduction software for the AIRS program consists of a series of subroutines written by ORNL which operate under control of a main program developed by the operators of the test facilities in Japan and the Federal Republic of Germany. ORNL also wrote a main program to run these subroutines on an in-house computer (PDP 11/34). These subroutines were all written in FORTRAN ANSI X3.9.

Six different subroutines were supplied to calculate the required two-phase flow parameters from the raw probe data. The main programs are responsible for the overall data management, sensor identification, timing, and branching as required to call the subroutines in the correct sequence. Each of the subroutines is designed to perform a certain type of analysis on a set of data from one or more sensors associated with a measurement location. A brief description of each subroutine is given below.

Subroutine REFCON.—This subroutine analyzes the data from the reference conductivity probe. It computes the liquid conductivity, the steam pressure, and the saturation temperature. This data is required as input data for some of the other subroutines; therefore, the reference conductivity probe data must be analyzed first.

Subroutine NOVA.—This subroutine calculates an estimate of the transport time between two adjacent sensors by cross-correlation of the two signals.

Subroutine INNOVA.—Subroutine NOVA requires certain input parameters which are based on data sampling rates, data array sizes, and user judgment and experience. INNOVA is used to determine values of the parameters which have given satisfactory results at ORNL. Since the parameters normally do not change throughout the test, INNOVA is executed only once at the beginning of the main program.

Subroutine VOIDFR.—This subroutine calculates the void fraction from a set of impedance magnitude and phase data from an impedance probe.

Subroutine FILMPR.—This subroutine accepts the estimates of transport time and void fraction produced by NOVA and VOIDFR and calculates a noise velocity, a steam velocity, and a liquid velocity.

Subroutine FILMPR.—This subroutine analyzes the data from a film probe module to produce an estimate of the average film thickness and velocity.

A typical reflood test will last from 600 to 1000 seconds. During the test, approximately 100 impedance sensor signals are recorded on magnetic tape in high density PCM format. Software to manage these large volumes of data and to display and store analysis results was developed partly by I&C staff members and partly by Foster Miller Associates under subcontract with ORNL.

3.3.4. TWO-PHASE FLOW ANALYSIS AND MASS FLOW ALGORITHMS

J. E. Hardy* J. O. Hylton
R. A. Hess R. N. McGill*
C. T. Hsu† D. G. Thomas*

The AIRS program has developed a variety of impedance probes for measurement of local two-phase flow phenomena. In particular, void fraction and flow velocity are the parameters of interest. These sensors were tested at ORNL under a wide range of flow conditions and test configurations. The results of the experiments were analyzed to determine measurement ranges of the various sensors. In addition, an algorithm was developed which can calculate the separate phase velocities by employing a void fraction and the velocity measured by an impedance probe. Mass flow models have been formulated to monitor flow rates in several regions of an experimental reactor test facility. These regions include the downcomer, vent valves, and the upper core support plate.

The magnitude and phase of the impedance probe output were used to calculate the local void fraction. These probe-determined values for void were compared to those measured by gamma densitometers. For the range of voids expected in the experimental test facility, there was good agreement between the two measurements. Test conditions included air and water and steam and water environments and a variety of temperatures and pressures.

The flow velocity is obtained by applying noise analysis techniques to the signals produced by two axially separated impedance sensors. A single velocity is determined which is equal to the dominant fluctuations in the two-phase flow. These fluctuations may be bubbles, drops, slugs, or a combination. As a result, the velocity is not necessarily the velocity of either the liquid or the vapor phase. Since the phase velocities are the quantities required, correlations were developed to determine phase velocities from the impedance probe velocity. These algorithms were developed by first formulating the governing two-phase flow equations (continuity, momentum, and energy) for both air-water and steam-water systems and then subjecting the equations to a scaling analysis. Using the significant parameters as determined by scaling and experimental data obtained during tests at ORNL, generalized correlations were found for liquid and vapor phase velocities. These generalized correlations may be used in a wide variety of test configurations and fluid combinations. The algorithms have been able to produce phase velocities that agree to within 30% of values predicted by the separated flow model. This accuracy level holds for a wide range of temperatures and pressures as well as for different bundle configurations. This is an important feature because the in-core sensors are installed in various facilities that operate over wide pressure and temperature ranges.

Flow models have been developed for monitoring the mass flow rate in a downcomer and upper plenum vent valve region. These models employ the output from two instruments. In the downcomer, an ORNL string probe and a drag body produced by Idaho National Engineering Laboratory (INEL) are used to measure mass flow into and out of the vessel via the downcomer. An ORNL string probe and an INEL turbine meter are combined to yield a mass flow rate through a vent valve in the upper plenum.

A major problem in the 2D/3D program is coupling the tests to be conducted in the Slab Core Test Facility in Japan and the Upper Plenum Test Facility in the Federal Republic of Germany. One approach is to measure the flows at the core/upper plenum interface (the common boundary between the two facilities) and attempt to match them as closely as possible. The following instruments have been proposed to make the necessary measurements:

1. Tie plate drag body transducer or differential pressure transducer to measure dP across the tie plate
2. Free-field turbine meter located above the tie plate
3. Temperature transducer
4. Pressure transducer
5. Differential pressure transducer to measure collapsed liquid level.

Development of the upper core support plate mass flow algorithm is complicated by the expected diverse nature of the flows. In some tests a saturated steam/water mixture flows from the core region; in other tests, in addition to the flow from the core, saturated or subcooled water flows into the upper plenum region from the hot leg. Depending on the flow rates of the individual streams and the degree of subcooling, a variety of flow regimes may occur. These include (1) all liquid down, (2) countercurrent flow in which gas (or vapor) goes up and liquid goes both up and down, and (3) concurrent flow in which both gas (or vapor) and liquid go up.

The approach followed in developing the mass flow algorithm was to perform calibration tests with individual sources for the input flows, move to the next degree of complexity and calibrate with two sources for the input flows, and finally perform tests with three sources for the input flows. Thus, in the order of complexity, calibration tests were performed for (a) core steam, (b) hot leg water, (c) core steam and core liquid spray, (d) core steam and hot leg water, and (e) core steam, core liquid spray, and hot leg water. Conditions a, b, and c were steady-state tests, and mass flow rate correlations and switching criteria were developed for all of the flow regimes that occurred. Conditions d and e, however, produced transient test conditions because of the injection of highly subcooled hot leg water. Nevertheless, in a few tests conditions were of a quasi-steady nature, and the results agreed well with the steady state correlations for high-flow upflow and all water down.

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3.3.5. ELECTRICAL IMPEDANCE STRING PROBES FOR TWO-PHASE VOID AND VELOCITY MEASUREMENTS

J. E. Hardy* **J. O. Hylton**

[Abstract of NUREG/CR-2505 (ORNL/TM-8172), May 1982]

An instrumentation scheme has been developed to measure two-phase flow velocity and void fraction during the refill/reflood stages of a loss-of-coolant accident in experimental test facilities. The instrumentation's principle of operation was based on measurement of the electrical impedance of two-phase mixtures. Two-phase velocity is estimated by time-of-flight analysis of signals from two spatially separated sensors. A relative capacitive technique was employed to measure void fraction.

The impedance sensor consists of a pair of stainless steel wires strung back and forth across a stainless steel frame. This sensor was dubbed "string" probe for this reason. The string probe was designed to withstand temperatures of 350°C, thermal transients of 300°C/s, and severe fluid- and condensation-induced shocks.

Void measurements from developed string probes were compared with gamma attenuation densitometer values; velocity measurements by the string probe were compared with calculated phase velocities and turbine meter velocities. In large open-flow areas (such as an upper plenum or end box), good agreement was found between densitometer void values and string sensor voids. Flow velocities determined by the string probe yielded reasonable agreement when compared with turbine and phase velocities. Generally, the string probe instrumentation (1) proved to be durable in air/water and steam/water flows and (2) demonstrated an ability to measure a wide range of flow velocities (1 to 15 m/s) and void fractions (0 to 0.99+).

*Engineering Technology Division.

3.3.6. MEASUREMENT OF VELOCITY AND VOID FRACTION IN STEAM-WATER MIXTURES WITH ELECTRICAL IMPEDANCE PROBES

J. O. Hylton **R. N. McGill***

(Abstract of paper presented at the Third CSNI Specialists' Meeting on Transient Two-Phase Flow, Pasadena, California, March 23–25, 1981)

High-temperature electrical impedance probes have been developed at the Oak Ridge National Laboratory to make local measurements of two-phase flow parameters in steam and water mixtures. These sensors are designed to survive a simulated pressurized water reactor reflood transient. They have been built in several different configurations and are intended primarily for insertion into test vessels to make measurements where other methods cannot be used. Application of these sensors has been made to the measurement of velocity and void fraction in the spaces between simulated fuel rods. Fluid impedance signals from two adjacent sensors are analyzed to estimate the void fraction and the transit time for random flow fluctuations to travel between the sensors. Analytical and empirical correlations have been developed to correct for flow regime effects and relate the measured transit time to the fluid phase velocities.

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3.3.7. INSTRUMENTATION DEVELOPMENT FOR LOW RANGE, LONG LINE DIFFERENTIAL PRESSURE MEASUREMENTS

W. L. Zabriskie **G. N. Miller**
S. C. Rogers **K. G. Turnage***

[Abstract of *ISA Trans.* **20**(4), 61–75 (1981)]

A major experimental facility planned for the investigation of the reflood and refill phases of a loss-of-coolant accident in a pressurized water reactor (PWR) is a full-scale model of a PWR upper plenum, designated as the Upper Plenum Test Facility (UPTF). One of the goals of the test program is to determine the characteristics of the dynamic steamwater flow at the core/upper plenum interface. To meet this need, an instrumentation system designed to measure small differential pressures generated by this two-phase flow has been developed, fabricated, and evaluated.

*Engineering Technology Division.

3.3.8. THERMAL-SHOCK-RESISTANT CERMET INSULATORS

C. S. Morgan*

[Abstract of NUREG/CR-2363 (ORNL/TM-8038), November 1981]

Sensors capable of tolerating high-temperature steam and severe thermal shock have been produced for measurement of three-dimensional two-phase (steam and water) fluid flow parameters in two German and two Japanese nonnuclear reactor reflood test facilities. To satisfy the insulator requirements for sensors, we developed an Al_2O_3-Pt cermet material, which contains 0.5 to 1.5 vol% finely dispersed platinum. Repetitive water-quenching tests of these high-density (>97% of theoretical) insulators from 520°C followed by helium permeation measurements indicate that the cermets have high thermal shock resistance (TSR). Additionally, the cermets are unaffected by long-term exposure to high-temperature steam.

The cermets are produced by hot pressing an Al_2O_3-Pt powder containing finely dispersed small-particle-size platinum nodules obtained by *in situ* reduction of a platinum chloride compound. The metal particles increase the TSR by impeding and diverting cracks, thus increasing the energy required to propagate cracks through the material. The metal nodules may also effect a mechanical locking of ceramic grains, thus further increasing the TSR.

Alumina-chromium cermets prepared by a similar *in situ* reduction procedure also possess improved TSR properties. The less expensive metal could make cermets applicable in more cost-conscious situations.

*Metals and Ceramics Division.

3.3.9. LIQUID VELOCITY MEASUREMENTS WITH AN ELECTROLYSIS POTENTIAL PROBE

W. F. Bethmann

(Abstract of report to be published)

In response to a loss of coolant (LOCA) in a water cooled nuclear reactor, the reactor vessel is reflooded with water from standby storage tanks to carry off heat that would otherwise damage fuel rods, vessel and other structures. During the latter phases of the reflooding, a mixture of steam and water flows between the structures and a flowing film of water adheres to the surfaces. To gain some knowledge about the behavior of these films and to aid heat transfer calculations in designing reactors, instruments have been developed for the measurement of film thickness, flow velocities, and mass flow rate in reactors and their simulators. The Electrolysis Potential (EP) Probe is one such instrument and is designed to measure average liquid velocity inside the films. It was discovered that if the EP Probe was excited with a constant voltage of sufficient magnitude to produce electrolysis, the current through the sensor decreased in a repeatable manner with increasing average film velocity.

3.4. Instrumentation Development Support for Fuel Reprocessing

3.4.1. AN IN-LINE MULTIWAVELENGTH PHOTOMETER FOR THE DETERMINATION OF HEAVY METAL CONCENTRATIONS

D. T. Bostick*
D. D. McCue M. L. Bauer
J. E. Strain* D. M. Dixon*

(Abstract of paper presented at the 25th Conference on Analytical Chemistry in Energy Technology, Gatlinburg, Tennessee, October 6–8, 1981)

An in-line photometer has been developed for continuous monitoring of uranium and plutonium concentratons in high radiation environments of nuclear fuel reprocessing plants. The instrument is equipped with multiple narrow band interference filters to monitor sample transmission in the 400 to 800 nm range. The filters are mounted in a rotating filter wheel which is located in front of a stationary tungsten halide light source. The monochromatic light from the respective optical filters is transmitted through a fiber optic cable of up to 10 m in length to the in-line sample flow cell located within the reprocessing area. A similar length of cable returns the optical signal to the photometer where the light intensity is detected with a photomultiplier tube, amplified, and processed with an LSI-11 computer system. The combined use of a rotating filter wheel and fiber optic cables provides a nearly simultaneous multiwavelength, multielement analysis directly in the reprocessing stream, while isolating the radiation-sensitive optical and electronic components in accessible, protected locations.

This report will describe the design and operation of the individual components of the prototype photometer. The performance of the photometer as an in-line monitor for the determination of heavy metal concentration in both aqueous and organic process solutions will also be discussed.

*Analytical Chemistry Division.

3.4.2. EVALUATION OF FIBER OPTICS FOR IN-LINE PHOTOMETRY IN HOSTILE ENVIRONMENTS

M. L. Bauer
D. A. Bostick* J. E. Strain*

(Abstract of paper presented at the 25th Annual International Symposium and Exhibit of the Society of Photo-Optical Instrumentation Engineers, San Diego, Calif., Aug 24–28, 1981)

Commercial fiber optics cables, both bundled and single-fiber, were evaluated for application in an in-line photometer being developed for monitoring uranium and plutonium concentrations in high radiation environments in nuclear fuel reprocessing plants. The relative attenuation of the optical signals resulting from both the radiation damage and the couplings between lengths of optical cable was determined for specimen cables. An ultraviolet-enhanced fiber bundle demonstrated good radiation resistance to a total dose of 10^8 rads, which is the dose estimated to be received during a one-year lifetime of the in-cell portion of the photometer. The loss in signal transmission in cables with several junctions was much greater in the fiber-bundle cable than in a 1-mm-diam, silica, single-fiber cable: a loss of 6 dB across each junction in the fiber bundle cable vs 3 dB in the single-fiber cable. Coloration of the single-fiber after a dose of 10^6 rads was 2.5 times greater at 500 nm compared with that of a fiber bundle of similar length. However, the single-fiber cable was still usable at a dose greater than 10^7 rads. The photometer was designed to use a single-fiber optical cable with adequate radiation shielding.

*Analytical Chemistry Division.

3.4.3. IN-LINE INSTRUMENTATION FOR CONTROL OF NUCLEAR FUEL REPROCESSING-I: IN-LINE NEUTRON POISON AND FREE ACID MONITOR

J. E. Strain*　　D. D. McCue
D. T. Bostick*　　R. E. Harper

[Reprint of *Trans. Am. Nucl. Soc.* 43, 275-76 (1982)]

The Consolidated Fuel Reprocessing Program (CFRP) at the Oak Ridge National Laboratory (ORNL) is involved in developing technology for reprocessing spent nuclear fuel. The successful operation of a reprocessing plant depends on the availability of dependable real-time in-line sensors that measure metal and acid concentrations, flow rate, liquid level, etc., in certain parts of the process. Sensors will be required to operate in both aqueous and organic streams and to transmit the analytical results to a central data acquisition system for storage and display and for possible computer control of some reprocessing operations.

The Analytical Chemistry and Instrumentation and Controls Divisions of ORNL are jointly involved in the development, design, and evaluation of specific in-line chemical sensors. The constraints on the sensor design are severe and necessitate the development of very rugged and dependable instruments. In general, these sensors must be accurate, modular in design for easy replacement by remote manipulators, have a high reliability, require infrequent maintenance, provide relatively rapid response, and also be capable of operating in the presence of acid fumes and high radiation levels. Two of the in-line sensors that have been developed are described in this paper, and two are described in the companion paper[1] to illustrate the considerations that must be included in their design.

The continuous dissolution of chopped fuel elements in a rotary dissolver poses a special problem and necessitates the monitoring of a neutron absorber that is added to the nitric acid for criticality control. The neutron poison monitor that we have developed incorporates an ^{241}Am-Be neutron source, a ^{10}B-lined neutron detector, and a flow-through sample compartment to both moderate and absorb neutrons. The instrument is calibrated in terms of "molar barns" so that any neutron absorber, e.g., boron, cadmium, or gadolinium, can be used with the same calibration. The range of the neutron poison monitor can be modified by a change of internal sample volume. The neutron source and detector are built as a single unit and can be replaced by a remote manipulator. The neutron detector is capable of operating in a radiation field of 10^5 R/h.

The adjustment of nitric acid concentration is necessary at many points in the process and in most cases can be monitored with a radiation resistant conductivity cell. If, however, the acid level is too high, the conductivity becomes insensitive to small changes in acidity, and at 5.8 *M* HNO_3, the conductivity begins to decrease with increasing acidity. To satisfactorily cover the required range of 2 to 10 *M*, it was necessary to develop an instrument utilizing another technique. This was accomplished by heating a circulating side stream to a constant temperature and measuring the ratio of water and nitric acid in the vapor. A schematic diagram of the monitor is shown in Fig. 3.4.1. It was found that this ratio is proportional to the acidity up to 10.8 *M* HNO_3, at which point the acid concentration in the vapor reaches 5.8 *M*. Solutes in the sampled stream have relatively little effect because the vapor pressures of both HNO_3 and HO_2 are similarly depressed by solutes. Problems due to the hydrolysis of uranium and plutonium are also avoided because the analyzed stream is not diluted

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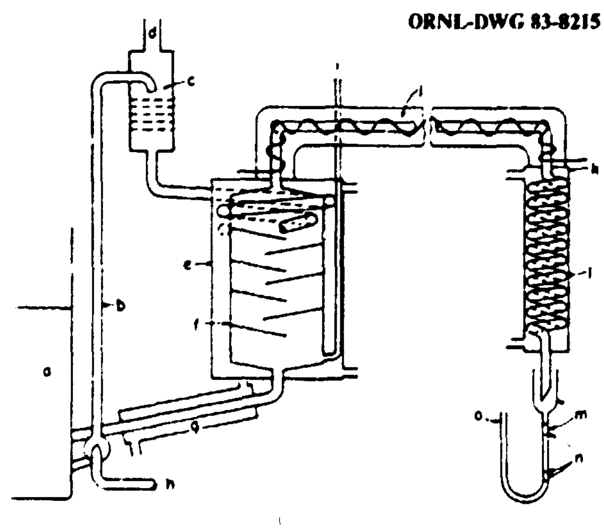

Fig. 3.4.1. Schematic diagram of monitor.

or neutralized. The entire instrument is modular and requires only air and electricity. There are no moving parts except for the flowing sample. Calibration can be checked and modified as needed by off line sample analysis.

*Analytical Chemistry Division.

1. D. T. Bostick et al., "In-Line Instrumentation for Control of Nuclear Fuel Reprocessing-II: In-Line Heavy Metals and Oxidant-Reductant Monitors." *Trans. Am. Nucl. Soc.* 43, 276 (1982).

3.4.4. IN-LINE INSTRUMENTATION FOR CONTROL OF NUCLEAR FUEL REPROCESSING—II: IN-LINE HEAVY METALS AND OXIDANT-REDUCTANT MONITORS

D. T. Bostick*

J. E. Strain* R. E. Harper

D. D. McCue M. L. Bauer

[Reprint of *Trans. Am. Nucl. Soc.* 43, 276–77 (1982)]

Continuous reprocessing of nuclear fuels requires real-time information regarding the oxidation state and concentration of uranium and plutonium throughout the recycling procedure. In-line filter photometers have been used to follow process stream concentration of a single metal. The optical transmission at a selected wavelength is dependent on the concentration of the given heavy metal in a specific oxidation state. An in-line photometer has recently been developed that simultaneously monitors U(VI), Pu(III) and Pu(IV) concentrations continuously in reprocessing streams (Fig. 3.4.2.). The instrument is equipped with multiple narrow band interference filters to measure the transmission of U(VI) at 416 and 426 nm, Pu(III) at 602 nm, Pu(IV) at 660 nm, and stream background transmission at 580 nm. The filters are mounted in a rotating filter wheel located in front of a stationary tungsten halide light source. The monochromatic light from the respective optical filters is alternately reflected into either a sample or a reference light path. The sample light path consists of a single strand of quartz fiber that transmits the monochromatic light into and out of an in-line sample flow cell located within the reprocessing area. The reference light path, constructed of similar optical material, does not penetrate the process-

ORNL-DWG 83-8216

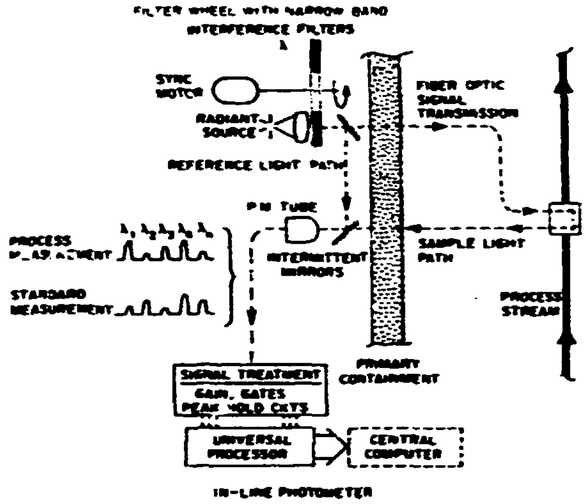

Fig. 3.4.2. The multi-wavelength photometer, utilizing minimum in-cell hardware.

ing area. The optical signals from either the sample or the reference light path are detected with a photomultiplier tube, amplified, and processed with an LSI-11 computer system.

The design of the prototype photometer permits the in-line analysis of 1 to 200 g/L U(VI) and 1 to 100 g/L Pu(III) and Pu(IV) in aqueous and organic process streams, while locating the bulk of the instrumentation offline. Such a design protects the radiation sensitive optical and electronic components and locates the moving parts in easily accessible locations. This feature simplifies instrument maintenance, replacement, and calibration procedures. The rapidly rotating filter wheel provides for a multielement determination while simultaneously correcting for stream background transmission. Signal averaging and discrimination of transmission data can be incorporated into computer software to improve sampling reproducibility by correcting for spurious fluctuations in stream composition. The reference fiber corrects for fluctuations in light source intensity and photomultiplier drift and provides an indication of the integrity of the individual optical filters and sample fiber-optic light path.

The valence state of plutonium is adjusted at several points in the reprocessing procedure to effect a separation of plutonium from uranium and fission product contaminants. A typical flowsheet calls for the continuous reduction of Pu(IV) to

Pu(III) with an acidic solution of either hydrazine (HYN) or a mixture of hydrazine and hydroxylamine (HAN). An oxidant, such as nitrite, is added downstream in the process to destroy excess reductant. Excess nitrite is, in turn, destroyed by an oxygen sparge. The presence or absence of these various redox reagents in the stream signifies completion of the redox reactions. In-line cyclic voltametry has been used to provide an indication of excess HYN, HAN, and nitrite. A stainless steel electrochemical cell was fabricated containing a platinum working electrode and a vitreous carbon quasi-reference electode. The cell body performed as the counter electrode. The internal volume of the cell was 0.4 ml; the voltage was scanned at 2 V/s to avoid convection effects in the flowing stream. The presence of HAN, HYN, and nitrite are observed at 1.1, 0.4, and 0.04 V, respectively, versus a standard calomel electrode. Peak grabber circuitry can be employed to automatically locate the oxidative waves and detect the presence of excess oxidant or reductant. Although accurate quantitative measurement of the redox reagents is not possible, the in-line electrochemical monitor can sense the presence of the above redox reagents at concentrations >1 m*M*.

*Analytical Chemistry Division.

3.4.5. ELECTROMECHANICAL THREE-AXIS DEVELOPMENT FOR REMOTE HANDLING IN THE HOT EXPERIMENTAL FACILITY

B. J. Bolfing **P. E. Satterlee**
J. Garin* **S. M. Babcock**

(Abstract of paper presented at 29th Remote Systems Technology Division Conference, American Nuclear Society, San Francisco, California December 1981).

A three-axis closed-loop position control system has been designed and installed on an overhead bridge, carriage, tube hoist for automatic positioning of manipulation at a remotely maintained work site. The system provides accurate (within 3 min) and repeatable three-axis positioning of the manipulator. The position control system has been interfaced to a supervisory minicomputer system that provides teach-playback capability of manipulator positioning and color graphic display of the three-axis system position.

*Fuel Recycle Division, on loan from Westinghouse Electric Corporation, Advanced Reactors Division, Pittsburgh, Pennsylvania.

3.4.6. DISTRIBUTED DIGITAL PROCESSING FOR SERVO-MANIPULATOR CONTROL

H. L. Martin
P. E. Satterlee **B. J. Bolfing**

Reprint of *Trans. Am. Nucl. Soc.* 43 752-753 (1982)]

Oak Ridge National Laboratory (ORNL) is performing work on the conceptual design of a pilot-scale plant for nuclear fuel reprocessing. One feature of the concept is the incorporation of completely remote operation and maintenance of the process equipment within a large sealed hot cell. Servo-manipulators will be used to perform the majority of the equipment maintenance. Efficient maintenance in a large volume hot cell to which human access is not permitted requires improvements in servo-manipulator reliability, flexibility, and cable handling. As an essential ingredient in the development of reprocessing technology, the Consolidated Fuel Reprocessing Program is engaged in a remote maintenance development program. One goal of this program is to improve the total performance of servo-manipulators for use in future plants.

A distributed digital control system is being developed to monitor and control a remotely operable two-arm manipulator and integral closed-circuit television system. The manipulator/camera system is shown in Fig. 3.4.3. The manipulator system is a Central Research Laboratory Model M-2 manipulator implemented with digital controls to enhance operational flexibility.

The decision to use a digital control system was motivated by several system requirements. The large volume of coverage anticipated for this system requires long cable runs. A significant reduction in cable handling is accomplished through use of a multiplexed communication path. The number of communication conductors was

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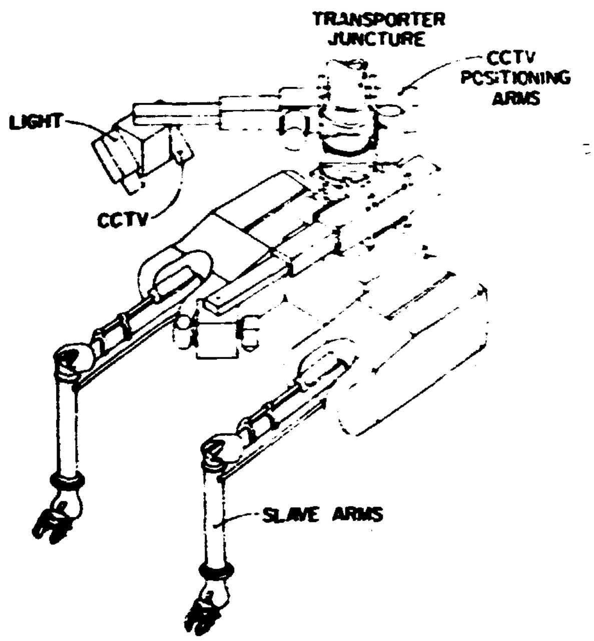

Fig. 3.4.3. In-cell portion of the industry standard servomanipulator system.

reduced from more than 100 per arm to 2 per arm. System performance is enhanced by the immunity to long-term drift and noise, which is characteristic of a digital system. Multi-mode operation is more easily accomplished digitally than with analog electronics. The desire for detailed diagnostics capability and a 120-Hz control update rate motivated the use of individual microprocessors for operation and monitoring of each joint.

A block diagram of the control hardware is shown in Fig. 3.4.4. The digital control system is composed of three major subsystems: the system monitor, the communication controller, and the joint controller. Each controller consists of a dedicated, programmable microprocessor with peripheral circuits for necessary input and output. This digital intelligence results in monitoring and control flexibility that is not possible with analog systems. Operational status, including current, velocity, and position, can be requested by the operator at any time. Supervisory monitoring of nonstandard conditions of high temperature and excessive current are automatically reported to the operator.

A single-loop system is presently operational, including motor, amplifier, controls, and software. Design of the communication hardware and man-machine interface are complete, and fabrication is progressing. Present efforts are directed toward final fabrication and system integration. The system should be operational in late 1982.

The digital control system will be capable of teach/play-back, supervisory control, and multiple operating modes. Future development activities will include applying innovative master-slave techniques to reduce operator fatigue, and automatic control for camera/end-effector coordination. The system will be used to demonstrate and develop techniques for remote maintenance of process equipment related to the reprocessing of nuclear fuels.

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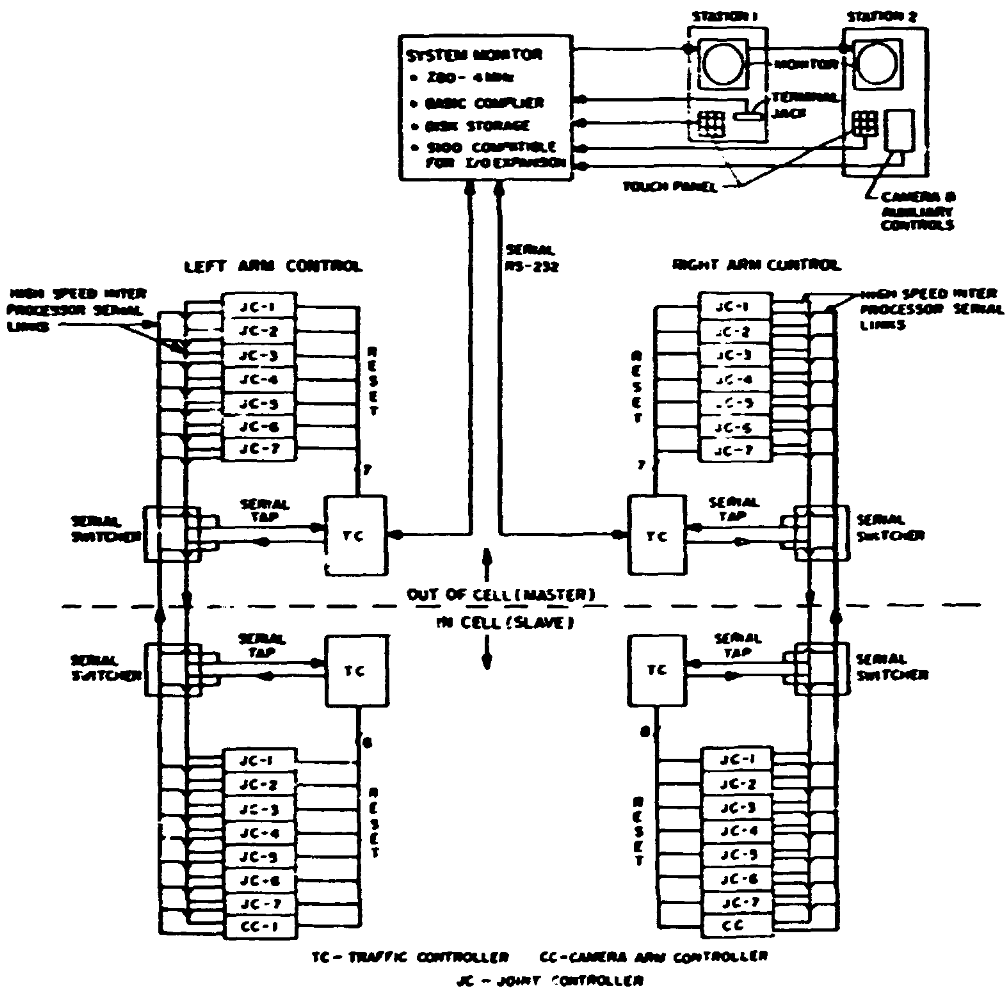

Fig. 3.4.4. General diagram of control hardware arrangement.

3.5. In-Core Experiments and Reactor Systems

3.5.1. INSTRUMENTATION SUPPORT FOR BLOWDOWN HEAT TRANSFER PROGRAM—THERMAL HYDRAULIC TEST FACILITY

S. S. Gould

The Blowdown Heat Transfer Program is an experimental separate-effects study of the principal variables involved in loss-of-coolant accident (LOCA) analysis for pressurized water reactors. Results of tests conducted from early FY 1980 through early FY 1981 may be used in the analysis of non-LOCA type, pressurized water reactor accidents (ATWS Accident Transient Without Scram). Primary test results are obtained from the thermal hydraulic test facility (THTF), a non-nuclear, pressurized water loop incorporating an electrically heated rod bundle. Final testing of a 64-rod bundle, with a threefold instrumentation increase over earlier bundles, concluded the THTF program.

Instrumentation support for the BDHT program consisted primarily of operations support (including in-depth calibration and maintenance) at the THTF and an error analysis to define the steady-state and transient error uncertainty bands of the measurements. The manpower support was approximately four instrument engineers and technicians for final bundle testing. Seven series of tests were conducted during this final testing period, and the test results have been published.*

A significant amount of effort was expended in the generation of instrument uncertainty band analysis for all critical THTF instrumentation (turbine flowmeters, gamma densitometers, strain gage pressure cells, differential pressure cells, Type E thermocouples and resistance thermometer devices, strain gage drag disks, and rod power instrumentation). The analyzed instruments and their minimum, steady-state, and 2–5 error bands (PSS, 95% confidence interval) are described in detail. The assumptions and judgments employed in the estimation of transient contributions to the uncertainties are well documented, and the steady-state and transient contributions to the uncertainty are listed separately in tables to allow the user to estimate the transient contribution if he so desires. Most of the data available on specific sources of instrument errors were obtained in single-phase flow, which was utilized with engineering judgment to combine and extrapolate to two-phase flow error bands.

Software development requirements were centered in the area of user utilities and graphics data display. Utilities were developed to aid in the correction of the official instrument description data base as well as calibration of instrumentation. The graphics data display program provided a means of viewing data of a previous testing series and simultaneously seeing the simulated physical changes in the loop in color picture format.

Additional support included the maintenance of a 1200-instrument data base and associated files to accommodate new loop instrumentation configurations prior to each testing sequence, and the processing of over 300 data tapes. *(R. L. Durall, B. J. Veazie, W. Ragan, A. G. Sutton, A. F. Johnson)*

*See following abstracts.

3.5.2. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 1. ORNL SMALL BREAK LOCA TEST SERIES I: EXPERIMENTAL DATA REPORT

T. M. Anklam* D. K. Felde*
D. F. Hunt* A. G. Sutton
M. S. Thompson* S. S. Gould

[Abstract of NUREG/CR-2525, Vol. 1 (ORNL/NUREG/TM-407/V1), April 1982]

The report presents experimental data and calculated steady-state and transient instrument uncertainties from Oak Ridge National Laboratory Small Break Loss of Coolant Accident (LOCA) Heat Transfer Test Series I. The subject test series was composed of six high-pressure, low-flow, quasi-steady-state heat transfer tests and six high-pressure reflood tests. The test series was designed to obtain data under conditions similar to those expected in a small break LOCA. In addition to the experimental data, calculated inlet and outlet mass flows and rod powers are presented.

*Engineering Technology Division.

3.5.3. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 2. THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 3.03.6AR—TRANSIENT FILM BOILING IN UPFLOW

C. B. Mullins* S. S. Gould
D. K. Felde* D. G. Morris*
A. G. Sutton J. J. Robinson*

[Abstract of NUREG/CR-2525, Vol. 2 (ORNL/NUREG/TM-407/V2), April 1982]

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) Test

3.03.6AR. This test was conducted by members of the Oak Ridge National Laboratory (ORNL) Pressurized-Water-Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on May 21, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small- and large-break loss-of-coolant accidents.

Test 3.03.6AR was conducted to obtain transient film boiling data in rod bundle geometry under reactor accident-type conditions. The primary purpose of this report is to make the reduced instrument responses for THTF Test 3.03.6AR available. Included in the report are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

*Engineering Technology Division.

3.5.4. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 3. THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 3.06.6B—TRANSIENT FILM BOILING IN UPFLOW

C. B. Mullins* S. S. Gould
D. K. Felde* D. G. Morris*
A. G. Sutton J. J. Robinson*

[Abstract of NUREG/CR-2525, Vol. 3 (ORNL/NUREG/TM-407/V3), May 1982]

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) Test 3.06.6B. This test was conducted by members of the Oak Ridge National Laboratory Pressurized-Water-Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on August 29, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small and large break loss-of-coolant accidents.

Test 3.06.6B was conducted to obtain transient film boiling data in rod bundle geometry under reactor accident-type conditions. The primary purpose of this report is to make the reduced instrument responses for THTF Test 3.06.6B available. Included in the report are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

3.5.5. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 4. ORNL SMALL BREAK LOCA HEAT TRANSFER TEST SERIES II: EXPERIMENTAL DATA REPORT

T. M. Anklam*
D. F. Hunt* A. G. Sutton
D. K. Felde* S. S. Gould
M. S. Thompson* C. R. Hyman*

[Abstract of NUREG/CR-2525, Vol. 4 (ORNL/NUREG/TM-407/V4), June 1982]

This report presents experimental data and calculated steady-state and transient instrument uncertainties from the Oak Ridge National Laboratory Small Break LOCA Heat Transfer Test Series II. The subject test series was composed of six combined heat transfer and mixture level swell tests, six additional mixture level swell tests, five high-pressure reflood tests, and five high-pressure boiloff tests. Also, the data and uncertainties are reported from two supplemental mixture level swell tests that were not part of Test Series II. Calculated inlet and outlet mass flows and fuel rod simulator power levels are reported in the report appendices.

*Engineering Technology Division.

3.5.6. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 5. THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 3.08.6C—TRANSIENT FILM BOILING IN UPFLOW

C. B. Mullins*

D. K. Felde*	S. S. Gould
A. G. Sutton	D. G. Morris*
K. N. Schwinkendorf*	J. J. Robinson*

[Abstract of NUREG/CR-2525, Vol. 5
(ORNL/NUREG/TM-407/V5), May 1982]

Reduced instrument responses are presented for Thermal-Hydraulic Test Facility (THTF) Test 3.08.6C. This test was conducted by members of the Oak Ridge National Laboratory Pressurized-Water-Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on October 1, 1980. The objective of the program was to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small and large break loss-of-coolant accidents.

Test 3.08.6C was conducted to obtain transient film boiling data in rod bundle geometry under reactor accident-type conditions. The primary purpose of this report is to make the reduced instrument responses for THTF Test 3.08.6C available. Included in the report are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

*Engineering Technology Division.

3.5.7. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 6. THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST 3.05.5B—DOUBLE-ENDED COLD-LEG BREAK SIMULATION

C. B. Mullins*

D. K. Felde*	D. G. Morris*
A. G. Sutton	J. J. Robinson*
S. S. Gould	K. N. Schwinkendorf*

[Abstract of NUREG/CR-2525, Vol. 6
(ORNL/NUREG/TM-407/V6), June 1982]

Thermal-Hydraulic Test Facility (THTF) Test 3.05.5B was conducted by members of the Oak Ridge National Laboratory (ORNL) Pressurized-Water Reactor (PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on July 3, 1980. The objective of the program is to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small- and large-break loss-of-coolant accidents.

Test 3.05.5B was designed to provide transient thermal-hydraulics data in rod bundle geometry under reactor accident-type conditions. Reduced instrument responses are presented. Also included are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

*Engineering Technology Division.

3.5.8. ORNL ROD BUNDLE HEAT TRANSFER TEST DATA, VOLUME 7. THERMAL-HYDRAULIC TEST FACILITY EXPERIMENTAL DATA REPORT FOR TEST SERIES 3.07.9—STEADY-STATE FILM BOILING IN UPFLOW

C. B. Mullins*	S. S. Gould
D. K. Felde*	D. G. Morris*
A. G. Sutton	J. J. Robinson*

[Abstract of NUREG/CR-2525, Vol. 7
(ORNL/NUREG/TM-407/V7), May 1982]

Thermal-Hydraulic Test Facility (THTF) test series 3.07.9 was conducted by members of the Oak Ridge National Laboratory Pressurized-Water Reactor (ORNL-PWR) Blowdown Heat Transfer (BDHT) Separate-Effects Program on September 11, September 18, and October 1, 1980. The objective of the program is to investigate heat transfer phenomena believed to occur in PWRs during accidents, including small- and large-break loss-of-coolant accidents.

Test series 3.07.9 was designed to provide steady-state film boiling data in rod bundle geometry under reactor accident-type conditions. This report presents the reduced instrument responses for THTF test series 3.07.9. Also included are uncertainties in the instrument responses, calculated mass flows, and calculated rod powers.

*Engineering Technology Division.

3.5.9. DEVELOPMENT OF A WIDE RANGE VORTEX SHEDDING FLOWMETER FOR HIGH TEMPERATURE HELIUM GAS

S. P. Baker
R. M. Ennis, Jr.* P. G. Herndon

Abstract of ORNL/TM-7794, July 1981)

The Core Flow Test Loop (CFTL) is a high temperature, high pressure helium gas circulating system, designed for study of the thermomechanical performance of gas-cooled fast reactor components at prototypic steady-state and transient operating conditions.

A single flowmeter was required for helium gas measurement in the CFTL. A volumetric flow accuracy of 1.0% of reading was required over a flow range of 166:1 and pressure range of 45:1 at a temperature of 350°C. In addition, a time response of <0.1 s and a minimal pressure drop across the flowmeter were required. No commercially available flowmeter could provide this performance.

This report describes results of both a development and Titles I and II design effort to provide the required flowmeter. A commercial vortex shedding flowmeter (VSFM) design was modified to provide a remotely located thermal vortex sensor. Experiments were conducted to determine empirically the relationships among thermal sensor signal, sensor connecting tubing geometry, and flow conditions for both air and helium gas flow. The geometry for remotely locating the sensor was then optimized to provide wide flow rangeability for high temperature helium gas flow.

Four VSFMs, one 6 in. and three 4 in., were fabricated and calibrated over the full CFTL range of use and beyond. All four meters fulfilled the CFTL requirements.

*TRW, Inc., Oak Ridge, Tennessee.

3.5.10. APPLICATION OF A VORTEX SHEDDING FLOWMETER TO THE WIDE RANGE MEASUREMENT OF HIGH TEMPERATURE GAS FLOW

S. P. Baker
R. M. Ennis, Jr.* P. G. Herndon

[Abstract of *ISA Trans.* 21(1), Instrument Society of America (1981)]

A single flowmeter was required for helium gas measurement in a gas cooled fast breeder reactor core flow simulator. Volumetric flow accuracy of 1.0% of reading was required over the Pipe Reynolds Number range 6×10^3 to 1×10^6 at pressures from 0.2 to 9 MPa (29 to 1305 psia) and at 350°C (660°F) temperature. A vortex shedding flowmeter with a remotely located thermal sensor was employed. The relationships among thermal sensor signal, sensor connecting tubing geometry, and flowing conditions were determined empirically for both air and helium gas flow. The geometry for remotely locating the sensor was optimized to provide wide flow rangeability for high temperature helium gas flow. Volumetric flow accuracy was measured over a range of 348:1 in flow and 46:1 in density. Volumetric flow accuracy of ±1.75% of reading was demonstrated over a range of 288:1 in flow.

*TRW, Inc., Oak Ridge, Tennessee.

3.6. Energy, Conservation, and Electric Power Systems

3.6.1. ENERGY UTILIZATION AND CONSERVATION

J. O. Hylton

Instrumentation and Controls Division personnel are participating in the Energy Utilization and Conservation Research Programs conducted by the Laboratory's Energy Division. These programs are directed toward making improvements in commonplace technology used in commercial and residential buildings and in industry. There are several ongoing experimental facilities in operation in which various concepts, both old and new, are tested. The major facilities include the following:

1. An Energy Conservation Laboratory for testing equipment such as home appliances, heat exchangers, heat pumps, and various types of energy storage and conversion systems.
2. An earth-sheltered passive solar dormitory for guest researchers at the Holifield Heavy Ion Accelerator. The building includes an extensive instrumentation and data acquisition system for evaluation of its energy utilization performance.
3. Experimental residential houses located in the Karns area and at the UT Tech Site in Blount County for evaluation of new concepts in heating and cooling.
4. A solar tracking test facility for photovoltaic cells. This facility provides the capability to measure short- and long-term cell performance characteristics.

Much of the instrumentation in these facilities is for the purpose of determining energy flow and balance in various systems. These quantities are inferred or calculated from measurements of fluid velocity, density, and temperature made by carefully placed sensors. Other measured parameters include electrical power, humidity or dew point, soil moisture, heat flux, solar radiation, and weather data.

In most cases the measurements are a straightforward application of commercially available technology. There were, however, some interesting problems and pitfalls which had to be overcome and are worthy of mention here:

1. The selection of reliable hardware and measurement methods which would provide "laboratory accuracy" in a field environment.
2. The development of automatic data acquisition and measurement systems which would operate continuously in the field unattended for periods of several days. The achievement of less than 5% downtime on such a system over the course of a one-year experiment is not easy.
3. The selection of data storage and retrieval methods and software that would be capable of readily accessing data collected for one year for computer analysis.
4. The development of a "hardened" data system which would survive natural and man-made electrical transients. Protection from ground potential rise with five houses feeding one data system through buried cables, to mention just one solution needed.

In addition to the above, these projects have resulted in the development of a low cost, relatively high accuracy, heat power flow (btu) meter which can be quickly installed on a residential heat pump. This device will determine the coefficient of performance of such equipment in a matter of minutes when used by an unskilled operator. *(G. S. Canright, M. E. Buchanan, J. M. McIntyre, D. R. Miller)*

3.6.2. THERMAL ENERGY STORAGE TEST FACILITY

C. Brashear

The thermal energy storage test (TEST) facility is an engineering-scale test loop for cycling residential-size, thermal energy storage units through conditions that simulate solar or off-peak electrical power applications to evaluate their performance. The test loop uses hot or cool liquids as the thermal transport fluid for cycling the TEST devices through a series of charge and discharge modes. Design and installation of the instrumentation, data acquisition, and control systems for the liquid loop were completed in FY 1980.

This system furnishes a high degree of versatility and controllability for producing a variety of input conditions; specifically, it permits control of loop parameters, such as temperature and flow rate input schedules, in step-function, sinusoidal, or stochastic patterns to fully simulate actual conditions and usage of solar or off-peak electrical power storage units.

A PDP-11/03 minicomputer was installed to generate the desired input control loop parametric functions and provide for data acquisition and storage. The supervisory application software utilized a high-level, process-oriented language (PRO by STAFF Computer Technology Inc.) to accommodate the sophisticated testing sequences. Software development was completed at the close of FY 1981. *(S. S. Gould)*

3.6.3. EARTH THERMAL STORAGE-ASSISTED HEAT PUMP

C. Brashear

The purpose of this program is to quantify the seasonal improvements in the heat pump efficiency and the energy savings which would result if the heat pump used the heat or "cool" stored in the earth under a house with a crawl space. To accomplish this objective three "identical," unoccupied houses, located in relatively close proximity, were leased for use in a controlled test. Two of the houses are configured to use the crawl space preheating concept, one for single pass and one for total recycle. In the single-pass mode, outdoor air is drawn through the crawl space, passed over the outside coil of a heat pump, and finally discharged to the atmosphere. In the total recycle mode, air discharged from the heat pump is recirculated through the crawl space and then returned to the outside coil. A conventional heat pump installation in the third house serves as the control for the experiment. It was our responsibility to provide instrumentation and data acquisition, storage, and retrieval support for the three houses.

Instrumentation was provided and installed to monitor temperature (crawl space and outside air and ground at various locations, room temperature, carpet, insulation, etc.); humidity; weather conditions (radiation, wind speed and direction); BTUs of heat delivered; and watt-hour readings (total heat pump use, I^2R use, and defrost cycle use).

An Accurex Autodata TEN/10 data logger was installed in each of the three houses with Techtran Model 817 Data Cassette recorders for data storage and Racal-Vadic 3451 triple modems for data retrieval. A Techtronix 4006-1 graphics terminal, triple modem, and data cassette recorder were installed at the control station in each house for data retrieval, reduction, display, and archival storage in the ORNL DEC System 10 User Service Center.

To protect the remote locations from fire and vandalism, a self-dialing burglar-fire alarm security system was installed in each of the test houses. *(R. L. Durall, S. S. Gould, D. R. Miller, L. D. Hunt, and R. L. Hansard)*

3.6.4. DATA ACQUISITION SYSTEM FOR ENERGY CONSERVATION DEMONSTRATION COMPLEX

D. R. Miller G. N. Miller

(Abstract of paper presented at 27th International Instrumentation Symposium, Instrument Society of America, Indianapolis, Indiana, April 27-30, 1981)

A data acquisition system (DAS) was developed for unmanned operation at an energy conservation engineering test facility to monitor an Annual Cycle Energy System (ACES) house, a solar wet-plate system house, a standard air-source heat-pump house, and an air-heating-type solar collector in parallel with an air-to-air heat pump experiment. The DAS scans 160 channels each hour and records instantaneous temperature readings, integrated heat flows, weather data, and electric power consumed. The data system comprises standard, commercially available data system components where feasible. Special attention was given to selection of devices yielding high resolution and stability at the liquid flow rates and temperatures typical of such systems.

System control by a proved computer-controller with an uninterruptible power source has resulted in greater than 99% availability of the DAS.

3.6.5. POWER SYSTEMS TECHNOLOGY PROGRAM

D. W. McDonald R. K. Adams

The Power Systems Technology Program was established at ORNL by the DOE Office of Electrical Energy Systems to conduct research and to perform technical management for DOE contracts in fields related to the nation's electrical power needs. The ORNL effort was subdivided into two programs to best address the two distinct technological thrusts of the Office of Electrical Energy Systems organization. The High Voltage Technology Program was given responsibility for the technical management and the research and development effort in areas related to the implementation of high voltage (1200 KVAC and 1500 KVAC) equipment and systems. The Electrical

Systems Program was concerned with the application of new technologies in the electric utility industry.

The I&C Division was responsible for the technical management of the High Voltage Program. This included several existing DOE contracts, the implementation of new contracts (by UCCND for DOE) and in-house research. The I&C Division effort centered around the development of new laboratory equipment and techniques, the development of *in situ* measurement instrumentation, analytical modelling, and investigating new applications of solid state technologies, for both single crystal and polycrystalline materials.

In the Electric Systems Program, I&C personnel provided technical support in the area of measurement and data analysis for several experimental projects relating primarily to automated load management and studies of the national electrical distribution grid. These projects are discussed in separate articles in this report.

3.6.6. CHARACTERISTICS OF THE EASTERN INTERCONNECTION LINE FREQUENCY

R. K. Adams

J. W. McIntyre **F. W. Symonds***

[Abstract of *IEEE Trans. on Power Apparatus and Systems*, Vol. PAS-101(12) (ISSN 0018-9510), pp. 4542–47 (December 1982)]

This paper reports the results of recent efforts to characterize the frequency on the Eastern Interconnection. The power system line frequency has been used routinely in the control and management of generation. A data collection and analysis program was undertaken to enable the characterization of this line frequency over time spans from five minutes to twenty-four hours. The results shown here reveal the range of frequencies that appear on the Interconnection, the nature of the variation about the 60 Hz set point, the spectrum of periodic components present, and the influence on the measured frequency of the averaging time used for measurement.

*The University of Tennessee.

3.6.7. MICROCOMPUTER USE FOR STUDYING INTERCONNECTED ELECTRIC SYSTEM FREQUENCY

R. K. Adams

J. M. McIntyre **R. W. Rochelle***

(Abstract in *The Best of the Computer Faires, Vol. VI: Conference Proceedings of the 6th West Coast Computer Faire*, San Francisco, California, April 3–5, 1981, pp. 309–10

Interconnected electric system frequency continually varies about the value of 60 Hz in response to electric load changes and as generation is adjusted to match load. The need for power system load management due to generation constraints has prompted us to make a detailed study of this system frequency. This work is based on much earlier studies of system frequency stemming from our overall interest in precision measurement and control. The use of microcomputers allows a wider variety of measurements of power system frequency than was previously feasible. We have used microcomputers both to record the output of frequency measuring instruments and to directly measure system frequency. Although applied here to power system frequency, the techniques we have used are generally applicable to the accurate measurement and monitoring of a wide range of frequencies.

*Department of Electrical Engineering, The University of Tennessee.

3.6.8. DATA ACQUISITION AT A RESIDENTIAL PHOTOVOLTAIC SYSTEM

J. M. McIntyre **G. N. Miller**

(Abstract of paper presented at 28th ISA International Instrumentation Symposium, Las Vegas, Nevada, May 3–6, 1982)

This paper describes the techniques employed for data collection and analysis in the study of a residential photovoltaic system in Phoenix, Arizona. A prototypical model home incorporated a photovoltaic array on the roof to serve as a power source for the dwelling. The array produces electric power directly from sunlight; however, the output of a solar array is direct current (dc) that must be con-

verted to alternating current (ac) for use in a normal home. This installation was unique in that the converted output was connected directly to the local electric utility. Several problem areas were investigated, particularly the electrical harmonics caused by the conversion to ac. An unusual data acquisition system was used at the site to record voltage and current waveforms and perform spectral analysis, total harmonic distortion, power and power factor analysis in the field immediately following each measurement. Using this means, investigators were able to monitor and evaluate the effects of perturbations on the system as they occurred.

3.7. Computer Systems

3.7.1. A MICROPROCESSOR-BASED ALPHANUMERIC THERMOCOUPLE SWITCH INDICATOR

D. R. Patek

An 8085 microprocessor-based microcomputer was utilized as a man-machine interface on panel instrumentation at the TRU facility. The instrumentation is three banks of 50 switches that connect thermocouples to three Doric Electronic temperature indicators. The microcomputer alternately scans each bank and reports to the operator a 1–8 character alphanumeric description of each thermocouple selected for temperature readout. When two or more switches in a bank are inadvertently selected, the numbers (1–50) of the switches are given to the operator instead.

The single-board microcomputer on a board 4.5 in. square was purchased from Sea Data Corporation. The microcomputer is connected via a ribbon cable to three Sea Data 4-in. × 4.5-in. alphanumeric display boards. The program, with alphanumeric descriptions of each of the 150 thermocouples, is located on a single 2-kilobyte UV erasable read-only memory (EPROM). The EPROM is readily reprogrammable in the event operators wish to alter the thermocouple descriptions. The entire unit, which was fabricated in I&C's Instrumentation Shop, occupies a space only 3 in. high × 18 in. wide × 10 in. deep in the facility control panel. *(M. L. Bauer, A. A. Shourbaji)*

3.7.2. USE OF AN INTELLIGENT DATA LOGGER AS A FRONT END FOR A DATA ACQUISITION SYSTEM

W. W. Manges

The intelligent data logger has become more and more attractive for use in data acquisition systems in recent years. Some offer very good hardware and well-developed processing capabilities. Some of the major limitations of these devices have always been data display and system integration capabilities. These data loggers, if they offer any outside access at all, have only an RS232 serial port. This offers almost no flexibility at all for data display but makes system integration at least possible. We have developed software modules to interface an Acurex Autodata-10 intelligent data logger to the minicomputer-based Oak Ridge Research Reactor Data Acquisition and Control System (ORRDACS).

The development of this software allows us to display all the data accessed by the data logger using the same sophisticated real-time graphics and tabular displays developed for the major experiments linked to the ORRDACS. The operator can then do all the checking, limit modifying, and even control parameter adjustment from a remote site using the serial line. The software can completely reprogram the data logger from the data base stored in the host computer and, therefore, can permit initialization without local intervention. We use pattern checking and a supervisory task to help ensure the integrity of the communication with the host computer.

The success of this integration of the Autodata-10 into a data acquisition system has led to its use in other programs. By acquiring the data base and user interaction software from ORRDACS and installing it in a PDP-11 minicomputer with an Autodata-10 and a Tektronix 4006 compatible graphics terminal, we can bring up a small data acquisition system with good graphics and user interface very rapidly and inexpensively. This procedure has been used in the Atomic Vapor Laser Isotope Separation (AVLIS) Laboratory at Y-12 as an interim system until a full upgrade to an integrated system is completed.

Although this software was originally developed with the idea that the data logger would be a small portion of a larger system, the size of the data base developed was 600 points. Some applications do not need this many points, however, and a smaller data base would be sufficient. Current development is aimed at reducing the size of the entire package to run in a microcomputer with relatively severe memory restrictions.

In summary, we have developed software to allow an engineer to bring up a good, reliable, user-friendly data acquisition system in a very short time. Also, because the interconnection is via a RS232 line, the data logger can be linked by any telephone line and be located at virtually any distance from the host computer. The planned reduction in computer resource requirements should make the package even more attractive. The system has the limitation of speed, however, and can be used only on relatively slow processes. In such systems, however, it could be used to implement distributed control as well as simple data acquisition. *(R. R. Bentz)*

3.7.3. CONTROL OF TEMPERATURE IN CAPSULE IRRADIATION EXPERIMENTS

W. W. Manges

In specimen irradiation experiments, it is important not to subject the specimens to high temperature for an extended period of time. The conventional technique used is to vary the composition, and thus the thermal conductivity, of the gas that flows in the annulus between the specimen chamber and the exterior wall. An operator routinely adjusts the flow valves of two gases to maintain some reasonable specimen temperature. The concept of using this technique to more tightly control specimen temperature led the Engineering Technology Division to build a test capsule equipped with voltage-controlled valves. Because manual control of the test setup proved to be less than satisfactory, we interfaced the capsule setup to the Oak Ridge Research Reactor Data Acquisition and Control System (ORRDACS) computer system and developed a software control algorithm to adjust the gas flows.

By first modeling the process and the proposed controller, it was possible to investigate the system response characteristics under various process and controller configurations. The most significant challenges were in handling the dead time in the response (caused by the gas lines) and in accommodating tremendous fluctuations in the overall gain of the process. The gain change requirement was necessary to simulate the variations in thermal absorption of the specimens as the control rods were adjusted in the reactor core. Also, because the capsule consisted of two independently controlled zones, the coupling between them had to be considered.

Application of the compensated rate feedback controller developed from the modeling work proved capable of controlling the measured specimen temperature to within 2°C over a 20% variation in input power to the capsule. Further development is necessary on a full power test setup. Also, a technique for bumpless transfer is needed so that the operator may switch between control modes without seriously perturbing the system.

3.7.4. MEDICAL SYSTEMS SUPPORT

R. L. Simpson

Continuous support for the ORNL Health Division began in 1971 when equipment was installed for analog recording and digitizing the results of electrocardiogram (ECG) tests. The Smith-Mayo

Electrocardiogram Analysis Program was obtained from the Mayo Clinic and installed at the ORNL Computing Center to analyze ECGs and aid physicians in selecting those tests needing examination by a cardiologist. The ORNL experience was so successful that additional equipment was purchased, and the service was expanded to all four Nuclear Division plants in 1974. In 1982 the system has been upgraded by leasing and installing a new IBM Electrocardiogram Analysis Program which is expected to significantly enhance ECG analysis capabilities.

A four-plant Medical Computer System was installed in Building 3500 in 1979. Medical data collected from CRT units in all four plants are entered and stored on disk; once each week the data are formatted and written on magnetic tape for subsequent entry into the Nuclear Division Medical Data Base. During this reporting period a number of additions and improvements have been made to the system, which now consists of a PDP-1134A minicomputer with 128K words of MOS memory, four disk drives with a total capacity of 276 million bytes of bulk storage, a nine-track, 800 bpi magnetic tape drive, a sixteen-port multiplexer, a hard-copy terminal, and a 300-lmp line printer. The system software is a core-resident operating system which consists of the following subsystems: Executive, I/O monitor, ANSI Standard MUMPS interpreter, and data base supervisor. The system has a CRT unit for data entry at each of the four plants and an LSI-11 microprocessor at the ORNL Medical Center for automatic data acquisition from hearing tests, pulmonary function tests, and chemical analyses of blood and urine.

In addition to the routine acquisition and processing of medical data, a number of additional special programs have been set up on the Four-Plant Medical Computer system:

1. A program for the Industrial Hygiene Department to facilitate recall and recordkeeping for employees who require periodic custom fitting of respiratory masks.
2. Programs for both the X-10 and Y-12 dieticians for recall and recordkeeping on employees participating in medically advised diet programs.
3. A program for the Y-12 Medical Department for recording and reporting mortality data for employees and former employees.
4. An inventory file for the Industrial Hygiene Department to maintain current information on all lasers at ORNL (type, responsible investigator, location), with periodic reporting.
5. An inventory file for the Industrial Hygiene Department to monitor locations, contents, status, and condition of all emergency supply cabinets at ORNL, with biannual reporting. *(S. M. Odom)*

3.7.5. LARGE-COIL TEST FACILITY

S. S. Gould

The Large-Coil Test Facility (LCTF) is used for the investigative testing of half-scale superconducting toroidal field coils in a six-coil toroidal array. The coils are nominally defined as noncircular, superconducting, operating at a peak field of at least 8 T, and with horizontal and vertical bore dimensions of ~2.5 m × 3.5 m, respectively. The six coils will be supplied by General Electric, General Dynamics, Westinghouse, Japan Atomic Energy Research Institute, Swiss Institute for Nuclear Science, and German Nuclear Research Center at Karlsruhe. The test results and their evaluation will provide a basis for the design and fabrication of toroidal field coils for Tokamak fusion experiments. Our involvement centers around three main areas, namely, the Data Acquisition System, sensor out-of-tank cabling, and signal conditioning patching.

Data acquisition system design for coil diagnostics consists of computer-based hardware and software to record and display data from coil sensors and coil-related facility sensors. The signals are classified into two categories fast and slow according to the sampling rate requirements. The fast signal channels are voltage taps, coil currents, thin film temperature sensors, pickup coils, strain gages, and displacement transducers. These sensors are to be sampled at rates up to 1000/s. Slow signals include the remaining temperature sensors, strain gages, displacement transducers, and field strength sensors. Sampling rates for the slow signals are to be 1 s per channel or slower.

The hardware purchased to acquire and provide display capability for these data is a Digital Equipment Corporation (DEC) PDP-11/60 minicomputer-based system. DEC LSI-11/23 microcomputers are used as front-end processors. Four identical LSI-11/23 units are used for acquiring the fast signals. Each of these processors has 64 analog input channels with RAM memory capable of ring buffering up to 2 s of data at 1000 samples/second/channel. Software has been developed to perform ring buffering of the data while transmitting a small portion of the averaged data to the PDP-11/60 over a DMA link for historical evaluation. In the event of a pulsed field experiment or a random trigger from the normal zone detection circuits, the ring buffering is terminated under program control, and the memory contents, which contain data acquired both before and after the trigger, are transferred to the PDP-11/60. The fast front ends do not utilize any operating system, and all software programs must be down-line loaded from the Host PDP-11/60.

For the slow data, an additional LSI-11/23 was purchased to acquire and transfer data over a DMA link to the [illegible]-11/60 similarly to that of the fast units. The analog input system uses a 500-channel Analogic ANDS 5400 system with DMA link to the LSI-11/23. This front end uses an RSX-11S multi-tasking operating system. Software is being developed to transmit the channel data to the PDP-11/60 at periodic intervals and perform additional functions such as offset correction, averaging, ring buffering, channel rate grouping, and input calibration and maintenance. A task in the PDP-11/60 has been developed to control each front-end processor by initializing the operating parameters and synchronizing data acquisition as required for each experiment.

The PDP-11/60 hardware provides on-line data storage through a 40-megabyte disk as well as archival storage on a nine-track magnetic tape drive and a high speed network link to the Fusion Energy Division PDP-10 User Service Center. These storage units were selected in order to accommodate the data rate, >26 megabytes/24 h, and on-line data retrieval requirements. A satellite LSI-11/23 system was provided to control a CAMAC crate (used for automatic control of voltage compensation gain settings) and to drive a DEC VT-11 graphics display terminal (used to display real-time diagnostic data). A Tektronix 4114 graphics display terminal and TEK 4611 hard copy unit were purchased for display of general diagnostic data. A Versatec printer/plotter and Vector-to-Raster converter have been purchased for obtaining hard copies of graphs and data. Using the RSX-11M operating system, the software for this system acquires and writes to disk the data from the five front-end processors. The display programs access the disk, calibrate, and convert to engineering units before outputting to the appropriate device. Other software being developed perform the communication, archival, data and sensor documentation, calibration, and maintenance functions.

Facility diagnostic sensors utilize an 800-channel FX System 2000 data scanner in conjunction with the slow front-end processor via a serial link. This scanner is a stand-alone subsystem with a very slow sampling rate (one scan per minute maximum).

Our main role in the data acquisition task has been the procurement, installation, and checkout of hardware subsystems as well as assessment and management of the software development effort performed by the UCC-ND Computer Sciences Division (although we also made a significant contribution to the software development in the areas of sensor calibration and maintenance, sensor documentation, and user software utilities).

A total of 114 drawings were prepared to show the sensor out-of-tank cables. The starting point for these cables was the hermetic feedthrough at the vacuum vessel wall for each inside-vessel sensor. Cables continued to either a group of patch panels (appropriate sensors are selected for monitoring during particular tests) or specific instruments located in the control room. Cables from the patch panels to other control room instruments were also covered. Connectors were specified for each end of the cables that would mate on one end with the hermetic feedthroughs in the tank wall and with the chassis connectors at the control room instruments or the patch panels. The experimenters used different circuits even for similar sensors, such as strain gages. Care was exercised throughout to maintain circuit integrity when matching cable conductors to the connector contacts.

Signal conditioning patching in the control room was accomplished by preparing 9 new drawings and review and revision of 12 drawings prepared by others. These provided detail for interface cabling, panels, connections, and signal distribution subassemblies for the Data Acquisition System. *(J. W. Cunningham, M. E. Greene, R. L. Hansard, W. A. Bird)*

3.7.6. COMPUTER NETWORK COMMUNICATIONS

R. W. Hayes

(Abstract of ORNL/TM-8087, June 1982)

An in-depth study of Digital Equipment Corporation's network software package (called DECnet) was completed to determine its performance characteristics. It was determined that the software provided many desirable features and capabilities while maintaining good flexibility. However, it was also determined that DECnet requires a significant amount of system resources such as disk storage and memory. The data transmission throughput was determined for communication between two minicomputers and between a minicomputer and a microcomputer over RS-232 communication lines and coaxial cable. Other characteristics were examined such as error detection and recovery and data flow and control. It was discovered that the restrictions due to DECnet, although not insurmountable, should be important factors in the design and specification of computer systems on which it is to be used.

3.7.7. DISTRIBUTED RESOURCE MANAGEMENT: DEADLOCKS

R. W. Hayes

(Abstract of ORNL/TM-8086, June 1982)

Many applications require data base management without the need for an entire data base management system. Also, most simple locking mechanisms are not adequate for real-time data acquisition and control applications. This report presents a basic resource management mechanism which acts as a "traffic cop" to control access to shared data bases in a real-time system. This mechanism makes use of a deadlock avoidance algorithm to resolve allocation conflicts.

The concept of resource management is developed and the problem of system deadlocking is described. Alternatives to the deadlock problem are presented for real-time systems and a modified deadlock avoidance algorithm is developed to provide efficient resource allocation. This algorithm is implemented on an RSX-11M system and is designed for easy expansion to distributed systems. A basic approach to distributed resource management is also presented.

3.7.8. COMPUTER-GENERATED SPEECH

Y. Aimthikul*

(Abstract of Master's Thesis, The University of Tennessee, December 1981)

The need for communication between man and machine is increasing as the use of computers pervades the industrial and information processing worlds. Since speech is perhaps the most natural means of communication, computer speech output is well-suited to applications in which an operator's eyes and hands are busy monitoring situations. The Instrumentation and Controls Division of the Oak Ridge National Laboratory (ORNL) sponsored this project because of the potential need to use computer speech output in numerous applications throughout ORNL. The specific objectives were (1) to establish the development tools necessary to readily implement a speech output capability in a computer system and (2) to evaluate commercially availabale speech modules for quality of speech and flexibility.

This thesis reviews the essential aspects of speech synthesis and distinguishes between the two prevailing techniques: compressed digital speech and phonemic synthesis. It then presents the hardware details of the five speech modules evaluated. FORTRAN programs were written to facilitate message creation and retrieval with four of the modules driven by a PDP-11 minicomputer. The fifth module was driven directly by a computer terminal. The compressed digital speech modules (T.I.

990/306, T.S.I. Series 3D and N.S. Digitalker) each contain a limited vocabulary produced by the manufacturers while both the phonemic synthesizers made by Votrax permit an almost unlimited set of sounds and words. A text-to-phoneme rules program was adapted for the PDP-11 (running under the RSX-11M operating system) to drive the Votras Speech Pac module. However, the Votrax Type'N Talk unit has its own built-in translator.

Comparison of these modules revealed that the compressed digital speech modules were superior in pronouncing words on an individual basis but lacked the inflection capability that permitted the phonemic synthesizers to generate more coherent phrases. These findings were necessarily highly subjective and dependent on the specific words and phrases studied. In addition, the rapid introduction of new modules by manufacturers will necessitate new comparisons. However, the results of this research verified that all of the modules studied do possess reasonable quality of speech that is suitable for man-machine applications. Furthermore, the development tools are now in place to permit the addition of computer speech output in such applications.

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3.8. Measurements Research

3.8.1. THERMOMETRY IN THE MULTIROD BURST TEST PROGRAM

R. L. Anderson
K. R. Carr T. G. Kollie*

[*Abstract of* NUREG/CR-2470 (ORNL/TM-8024), March 1982]

A temperature measurement error analysis was performed for the Type S (0.25-mm-diam, bare-wire) and Type K (0.71-mm-diam, sheathed) thermocouple circuits used to measure the temperature of the Zircaloy-clad, electrically heated fuel-rod simulators in the Multirod Burst Test program (MRBT) at Oak Ridge National Laboratory (ORNL). An important objective of the MRBT program is to improve the understanding of the behavior of the Zircaloy cladding of nuclear fuel rods under conditions postulated for a large-break, loss-of-coolant accident.

Eight categories of error sources were studied both analytically and experimentally: thermal shunting; electrical shunting and leakage; thermocouple calibration; thermocouple decalibration in service; thermoelectric properties of thermocouple extension wire, plugs, and jacks; thermocouple reference junction; data acquisition system; and electrical noise. Nonroutine temperature measurement verification techniques applied and described or referenced (some related to measurements other than the fuel-rod simulator application) include: experimental verification of the error due to the thermocouple cold work during installation; experimental studies to determine probable error due to the thermal shunting caused by thermocouples; special on-site verifications of proper operation and accuracy of the temperature-measurement systems; the use of two lots of thermocouple wire in a single thermoelectric circuit with proper compensation via the use of calibration coefficients; verification of proper thermoelectric material in thermocouple plugs and jacks via application of an instrument developed at ORNL; and temperature measurement with thermocouple sensing junctions welded directly to a test article with dc potential of 100 V or more from ground potential.

The analysis produced the following estimates for the total maximum errors in the range 300 to 1000°C:

> Type K thermocouples (worst case of two test facilities) exclusive of thermal shunting error, which remains to be estimated by mathematical modeling: ±12.5°C in addition to −1.7°C due to thermocouple cold work. Type S thermocouples: ±10.6 in addition to −1.4°C due to thermocouple cold work.

*Y-12 Development Division.

3.8.2. TESTING OF THERMOCOUPLES FOR INHOMOGENEITIES A REVIEW OF THEORY, WITH EXAMPLES

C. A. Mossman
J. L. Horton R. L. Anderson

(Abstract of paper presented at Temperature: Its Measurement and Control in Science and Industry, 5, Pittsburgh, Pennsylvania, March 1982)

The effect of inhomogeneities on the behavior of thermocouples is examined in detail. Methods of testing for inhomogeneities are outlined, and the methods employed at the Oak Ridge National Laboratory are described. Examples are given of the uses of an inhomogeneity Test Facility as a diagnostic tool to determine the cause of temperature measurement errors.

3.8.3. FAILURE OF SHEATHED THERMOCOUPLES DUE TO THERMAL CYCLING

R. L. Anderson R. L. Ludwig*

(Abstract of paper presented at Temperature: Its Measurement and Control in Science and Industry, 5, Pittsburgh, Pennsylvania, March 1982)

Open circuit failures (up to 100%) in small-diameter thermocouples used in electrically heated nuclear fuel rod simulator prototypes during thermal cycling tests were investigated to determine the cause(s) of the failures. The experiments conducted to determine the relative effects of differential thermal expansion, wire size, grain size, and manufacturing technology are described. It was concluded that the large grain size and embrittlement which result from certain common manufacturing annealing and drawing procedures were a major contributing factor in the breakage of the thermocouple wires.

*Y-12 Development Division.

3.8.4. DECALIBRATION OF SHEATHED THERMOCOUPLES

R. L. Anderson
J. D. Lyons W. H. Cristie*
T. G. Kellie† R. Eby*

(Abstract of paper presented at Temperature: Its Measurement and Control in Science and Industry, 5, Pittsburgh, Pennsylvania, March 1982)

The decalibration of sheathed thermocouples at high temperatures has been studied by a variety of methods including inhomogeneity testing, calibration, metallography, and an ion microprobe mass analyzer. The results clearly show that the sheathed thermocouple is a complex system at elevated temperatures and that the thermoelements can be contaminated by the diffusion of materials through the insulation from the sheath. Decalibration of base-metal thermocouples is more severe in stainless steel sheaths than in high-nickel alloy sheaths such as Inconel 600. Decalibration tests to 1200°C have been run on type K and Nicrosil versus Nisil in vacuum and in argon atmospheres with sheathed assemblies. The decalibrations of both types of thermocouples are similar under these conditions.

*Analytical Chemistry Division.
†Y-12 Development Division.

3.8.5. AUTOMATED TEMPERATURE MEASUREMENTS FROM −183 to 2300°C

M. H. Cooper, Jr.
R. L. Anderson C. A. Messman

(Abstract of paper presented at Temperature: Its Measurement and Control in Science and Industry, 5, Pittsburgh, Pennsylvania, March 1982)

Precision temperature measurements were formerly a labor intensive activity. With the advent of relatively inexpensive desk-top computer controllers and high quality digital voltmeters, much of the data acquisition in precision temperature measurements can be assumed by an automated system. The experience over the past eight years at the Oak Ridge National Laboratory in applying automated instrumentation to temperature measurements is related. Making precision measurements with automated data systems requires considerable care beyond the simple recording of data. Redundancy and self-checking techniques have been developed to ensure reliable and accurate measurements. The accuracy of an automated system is shown to be equal to that of a manually operated laboratory potentiometer.

3.8.6. IN-CORE THERMOCOUPLE PERFORMANCE UNDER SIMULATED ACCIDENT CONDITIONS

L. A. Banda*
D. G. Cain† R. L. Anderson

[Abstract of *IEEE Trans. Nucl. Sci.* NS-28(1) (February 1981)]

The accident at Three Mile Island, Unit 2, Nuclear Station on March 28, 1979, resulted in reactor core conditions that far exceeded normal operating ranges of reactor in-core instrumentation. The behavior of these instruments under high temperature, super-heated conditions had not been previously characterized and the significance of the data obtained during the accident was not easily interpreted.

The purpose of these high temperature experiments on standard in-core thermocouples was to obtain some insight into their behavior under simulated core melt conditions, to determine if any improvements could be made to the basic design of the instrument, and to attempt to correlate experimental results with TMI 2 data.

*Combustion Engineering, Inc., Windsor, Connecticut.
†Nuclear Safety Analysis Center, Palo Alto, California.

3.8.7. ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY-MARCH 1980

K. G. Turnage* C. E. Davis*
R. L. Anderson G. N. Miller

[Abstract of NUREG/CR-1647 (ORNL/NUREG/TM-403), September 1980]

Work performed to develop reliable liquid level sensors for in-vessel use in pressurized water reactors is described. During this period, an improved heated thermocouple level sensor was fabricated and techniques for separating thermocouple output signals from superimposed ac voltages were developed. Experiments that were performed with a thermal-type level sensor in natural convection to saturated water and steam are described. Pressures in those tests ranged from 0.1 to 10 MPa (15 to 1500 psia). Acoustic techniques using pulse transit times along a waveguide were studied. A single waveguide was used to obtain accurate measurements of liquid level at room temperature and to compensate for temperature variations over a 10°C temperature range.

*Engineering Technology Division.

3.8.8. ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR JULY–SEPTEMBER 1980

K. G. Turnage* G. N. Miller
R. L. Anderson R. W. McCulloch*
C. E. Davis* S. C. Rogers

[Abstract of NUREG/CR-1903 (ORNL/NUREG/TM-430), March 1981]

Work performed to develop and evaluate liquid level sensors for in-vessel use in pressurized-water

reactors is described. Techniques used to fabricate heated thermocouple (HTC) probes are presented. Experiments were performed with five thermal-type liquid level sensors in natural convection to saturated water and steam. Pressures in those tests ranged from 0.1 to 8.6 MPa (15 to 1250 psia). Test goals were to study sensor repeatability and optimize splash shield designs. An HTC was installed in the upper plenum of the Thermal Hydraulic Test Facility and tested during steady-state film boiling tests. The HTC performed well except at high test-section outlet velocities.

Tests intended to prove the principle of the temperature-compensated ultrasonic liquid level instrument were performed in high-temperature saturated liquid water and steam. The mean densities obtained using the ultrasonic device agreed with those from pressure difference measurements, within experimental error.

*Engineering Technology Division.

3.8.9. TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR OCTOBER–DECEMBER 1980

K. G. Turnage* G. N. Miller
R. L. Anderson R. W. McCulloch*
C. E. Davis* S. C. Rogers

[Abstract of NUREG/CR-2007 (ORNL/TM-443), May 1981]

Work performed to develop and evaluate liquid level sensors for in-vessel use in pressurized water reactors (PWRs) is described. Thermal-type and acoustic-type devices were tested under conditions similar to a hypothetical PWR loss-of-coolant accident. A heated thermocouple (HTC) and a pressure difference instrument were tested in the Thermal-Hydraulic Test Facility during high-pressure rod-bundle uncovery and reflood experiments. Both methods provided useful indications of the system coolant inventory. A three-element HTC was fabricated and tested in a pressurizer over a wide range of conditions. Improvements were made in a ribbon-type ultrasonic liquid level device. Data obtained from the ultrasonic device in the pressurizer were reduced and evaluated.

*Engineering Technology Division.

3.8.10. TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY–MARCH 1981

J. E. Hardy* S. C. Rogers
C. E. Davis* R. W. McCulloch*
K. G. Turnage* G. N. Miller

[Abstract of NUREG/CR-2204, Vol. 1 (ORNL/TM-7877), July 1981]

Development and evaluation of liquid level sensors for in-vessel use in pressurized-water reactors are described. A four-element heated thermocouple (HTC) level device was designed and fabricated by Oak Ridge National Laboratory to withstand loss-of-coolant accident thermal-hydraulic conditions. The multiple-position heated junction thermocouple was tested in a pressurizer over a wide range of conditions and will be installed and tested under a variety of accident conditions in the Semiscale Test Facility at EG&G Idaho. A single-element HTC was operated under high-void-fraction air and water flows to determine cooling effects of a mist. Data showed detectable cooling at surprisingly high void levels (~ 0.98). A standpipe arrangement was tested in air and water to study the relationship between liquid level in the standpipe and flowing void of the surrounding media. Conceptual designs for a pressure seal technique in a reactor vessel for the ultrasonic level device (ULD) were discussed with a reactor vendor. Continued improvements, including interfacing the probe output with a Hewlett-Packard HP85 calculator, were made with the ULD, thus improving the quality of the data produced.

*Engineering Technology Division.

3.8.11. ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR APRIL–JUNE 1981

J. E. Hardy*
S. C. Rogers G. N. Miller

[Abstract of NUREG/CR-2204, Vol. 2 (ORNL/TM-8010, October 1981)]

Preliminary evaluation of a reactor level instrumentation system located at the Semiscale Test

Facility at EG&G Idaho was completed. This instrumentation system was designed to monitor core cooling status by using differential pressure (dP) transmitters to give an indication of in-vessel collapsed liquid level. The dP system is required to yield a reliable and unambiguous indication of water level for the full range of steady-state and transient conditions. To aid in evaluating data from the liquid level instrumentation, dP data taken from the Thermal Hydraulic Test Facility were analyzed. Results showed that pressure-drop cell output provided information relative to core liquid level in some cases. However, problem areas that might lead to incorrect reactor operator response were observed where pressure-drop data indicated collapsed liquid levels above the core quench front.

*Engineering Technology Division.

3.8.12. ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR JULY–SEPTEMBER 1981

J. E. Hardy* S. C. Rogers
G. N. Miller W. L. Zabriskie

[Abstract of NUREG/CR-2204, Vol. 3 (ORNL/TM-8162), January 1982]

The performance of the Westinghouse Reactor Vessel Level Indicating System (RVLIS) in the S-UT-3 test (a communicative break in the cold leg of Semiscale) was analyzed. The Westinghouse RVLIS gave similar indications to Semiscale Test Facility instrumentation measuring the same phenomena [differential pressure (dP)] over equal spans. The Westinghouse measurement is apparently conservative when compared with the two-phase froth level. These dP measurements appear to be nonconservative estimates of level, however, when the measurement system spans the upper core support plate. Level measurement errors of up to 150 cm (60 in.) were observed during S-UT-3. Westinghouse claims that these differences are caused by differences between Semiscale and Westinghouse Reactors. A recommendation for resolving these differences is made.

*Engineering Technology Division.

3.8.13. ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR OCTOBER–DECEMBER 1981

J. E. Hardy* G. N. Miller
S. C. Rogers W. L. Zabriskie

[Abstract of NUREG/CR-2204, Vol. 4 (ORNL/TM-8231), March 1982]

The performance of the Westinghouse Reactor Vessel Level Indicating System (RVLIS) during tests S-UT-6 and S-UT-7 (5% cold-leg breaks in the Semiscale Test Facility) was analyzed. The RVLIS, a system employing differential pressure (dP) cells, gave estimates of vessel level similar to those of Semiscale level instrumentation when measuring over equal spans. These RVLIS measurements are conservative to vessel coolant levels for both S-UT-6 and S-UT-7. At times, the RVLIS indications are greater than the vessel collapsed liquid level measured by Semiscale instrumentation. During S-UT-6, level estimate differences between RVLIS and Semiscale dPs of up to 211 cm (85 in.) were observed. These discrepancies may be explained by differences in Semiscale and Westinghouse pressurized-water reactor internal designs. Excellent agreement was noted between Semiscale and Westinghouse vessel levels for S-UT-7, an upper-head injection test.

*Engineering Technology Division.

3.8.14. INADEQUATE CORE COOLING INSTRUMENTATION USING HEATED JUNCTION THERMOCOUPLES FOR REACTOR VESSEL LEVEL MEASUREMENT

R. L. Anderson
J. L. Anderson G. N. Miller

[Executive Summary of NUREG/CR-2627 (ORNL/TM-8268), March 1982]

This document presents a technical review for the NRC staff of Inadequate Core Cooling Instrumentation incorporating a Reactor Vessel Level Monitoring System using Heated Junction Thermocouples proposed by Combustion Engineering's response to NRC requirements for an evaluation of the need for additional instrumentation to detect the approach to inadequate core cooling and in particular to evaluate means for measuring reactor vessel water level.

Emphasis was placed on evaluating the Inadequate Core Cooling (ICC) Instrumentation system as a whole which includes, besides the heated junction thermocouple reactor vessel level measurement, the saturation margin monitor, the core exit thermocouples, the safety parameter display system, and the critical functions monitor system.

The definition of Inadequate Core Cooling (ICC) is discussed in detail. Measurable variables are defined for evaluation of the ICC instrumentation system. Following is a discussion of the evaluation considerations used in the review. The existing plant instrumentation which can be used for ICC detection was reviewed, followed by an analysis of the need for additional instrumentation. Combustion Engineering's proposed heated junction thermocouple water level measurement system is described in some detail and along with the testing done by ORNL and Combustion Engineering. Analyses of small break accidents performed by Combustion Engineering are summarized and are related to reactor vessel level measurements. The heated junction thermocouple level measurement system was evaluated with respect to the NRC requirements and for possible ambiguities.

The use of the ICC instrumentation is limited to those transients which progress relatively slowly and for which operator action is required to prevent ICC. The system may not be used during a rapid depressurization. It is believed that such transients would be of short duration (≈100 s), relative to the response time required for the operator to take action in the regime of small breaks of 0.1 sq-ft and smaller; it is not expected to be used for very rapid transients associated with large breaks, since recovery actions should be initiated automatically without operator intervention. The system can be used, however, to monitor the recovery from large breaks.

The conclusion of this evaluation is that the Inadequate Core Cooling Instrumentation system which includes the Heated Junction Thermocouple Reactor Level Measurement System proposed by Combustion Engineering will meet the NRC requirements to provide plant operators with an unambiguous indication of the approach to inadequate core cooling during slow transients. For cases in which the reactor vessel is filled with a two-phase mixture, experimental evidence indicates that the system will indicate collapsed liquid level or the trending of the reactor vessel coolant inventory. Furthermore, the system will provide the plant operator a valuable indication of the effect of the recovery measures from both slow and fast transients.

3.8.15. INADEQUATE CORE COOLING INSTRUMENTATION USING DIFFERENTIAL PRESSURE FOR REACTOR VESSEL LEVEL MEASUREMENT

G. N. Miller

J. L. Anderson R. L. Anderson

[Executive Summary of NUREG/CR-2628 (ORNL/TM-8269), March 1982]

This document presents a technical review of the Inadequate Core Cooling Instrumentation with a Reactor Vessel Level Monitoring System using a differential pressure measurement system proposed by Westinghouse, Inc., for pressurized water reactors. This system is Westinghouse's response to requirements of NUREG-0737 to evaluate the need for additional instrumentation to detect the approach to inadequate core cooling and in particular to evaluate means for measuring reactor vessel water level. Because the question of the need for reactor vessel measurement instrumentation has been a somewhat controversial issue, this report includes a great deal more material than is normally found in a technical review of this nature. It is the intention to provide in one document, coverage of all of the relevant material that has accumulated since the accident at TMI-2.

Emphasis was placed on evaluation of the generic Inadequate Core Cooling (ICC) Instrumentation system as a whole which includes, besides the differential pressure reactor vessel level measurement, the saturation margin monitor, the core exit thermocouples and the display system (either the 7300, an analog display, or the microprocessor based system with a plasma panel display). Westinghouse refers to this complete system as the Reactor Vessel Level Instrumentation System (RVLIS). The system was evaluated on the basis of documentation supplied by Westinghouse

Inc., tests run at ORNL and tests run at INEL (SEMISCALE).

The RVLIS is a differential pressure measurement system. There are two trains of differential pressure measurement. On each train there are three differential pressure transmitters. Two transmitters are connected from the bottom of the vessel to the top of the vessel via tapping existing penetrations [on an Upper Head Injection (UHI) plant, the top connection is to the Hot Legs]. One of these two transmitters referred to as the narrow range unit is set up to measure the collapsed level in the vessel (0 to 100%) with the pumps off. With the pumps on, the narrow range unit is off scale high. The second transmitter, referred to as the wide range unit, is scaled to read 0% vessel empty and pumps off, and 100% with vessel full and all pumps on. With pumps off and vessel full, the wide range unit reads about 33% (15% on a UHI plant). The third transmitter of one train is connected between the hot legs and the top of the vessel and is referred to as the upper range unit. Westinghouse says this unit is not to be used except during head venting. When the vessel is full and pumps are off, the instrument indicates 100%, the unit indicates off scale in the 0% direction with vessel full and pumps on due to the frictional differential pressure.

These transmitters are connected to the vessel by armored capillary tubing. The transmitters are outside the containment wall, and there are two isolators in each capillary tube, one close to the vessel penetration tap point and one close to the containment wall. The capillary tubes are vacuum filled with demineralized deaerated water. The isolators have switch closures that indicate loss of capillary tubing water. Further, the isolators have valve stops that prohibit excessive fluid transfer. Problems in this area can be confirmed by the other train measurements and by the switch closures. Temperature measurements are made on any vertical run of these capillary tubes to compensate for density variations.

The generic display system for either plant could be an analog processor with panel meter readout (7300 system) or a microprocessor-based system with a plasma panel readout (microprocessor system). Either system is supplied with a strip chart recorder for trending the analog outputs. Either system compensates for density variations between reference legs and the vessel. When the pumps are on, the 7300 system has a light telling the operator to disregard the narrow range and the upper range indications. The microprocessor unit has a status indication for these measurements to indicate when they are to be disregarded.

Analyses have been presented by the Westinghouse Owner's Group in WCAP-9753, of the system behavior with 1- and 4-in.-diam breaks. Summary reports describing the generic analog and microprocessor based differential pressure level measurement system together with the Saturation Margin monitor and core exit thermocouples assert that these systems are adequate for detecting an approach to inadequate core cooling for breaks up to 4-in. diam. Tests of the differential pressure system were added to the regular testing program at SEMISCALE and the results reported in EGG-SEMI-5494 and EGG-SEMI-5552. Additional analysis of these results are forthcoming in ORNL-TMs. Some differences between indications of the Westinghouse system and the SEMISCALE differential pressure level system were noted in the upper head. Westinghouse claims that this difference is mainly a result of differences in the configurations between the full-sized Westinghouse reactor and SEMISCALE upper head regions. Indications of other Westinghouse differential pressure level measurements were in good agreement with the SEMISCALE instruments in the same range. A repeat test was performed with the configuration of SEMISCALE modified to simulate a Westinghouse reactor. The dp level measurements in this test were in good agreement (less than 5% error) with SEMISCALE instrumentation. On August 8, 1981, the NRC requested additional information from the utilities proposing to use the Westinghouse differential pressure system. Most of these questions have been resolved to the staff's satisfaction, but a few outstanding questions remain to be answered. The generic description of the system along with the clarification supplied appear to be adequate for approval of the system for trial installation and use. Plant specific features, however, will still require review on a plant-by-plant basis.

In summary, the systems proposed by Westinghouse do provide an unambiguous indication of water level above the core when, in fact, such a level exists. During rapid transients, ambiguous indication may occur, but are expected to be of brief duration. For cases in which the reactor vessel is filled with a two-phase mixture, experimental evidence indicates that the differential pressure systems will indicate collapsed liquid level or the trending of the reactor vessel coolant inventory. The conclusion of this evaluation is that the Inadequate Core Cooling Instrumentation system which includes the differential pressure reactor vessel level indicating system (RVLIS) proposed by Westinghouse will meet the requirements of NUREG-0737 to provide the plant operator with an unambiguous indication of the approach to adequate core cooling in small break LOCA transients. Furthermore, the system will provide the plant operator with a valuable indication of the effect of the recovery measures.

3.8.16. EX-CORE NEUTRON DETECTORS FOR REACTOR VESSEL LEVEL MEASUREMENT

J. L. Anderson

R. L. Anderson G. N. Miller

[Executive Summary of NUREG/CR-2626 (ORNL/TM-8267), March 1982]

This report documents the evaluation of an application by Alabama Power Company and National Nuclear Corporation for a Non-Invasive Coolant Level Monitor for the Farley Power Station, Units 1 and 2.

The principle of the noninvasive coolant level monitoring system is based on the detection of 2.2 MeV photoneutrons produced by the interaction of high energy γ-rays emitted by fission products in the core, in particular ^{140}La, with deuterium in the coolant water inside the vessel. The system utilizes neutron detectors located at the top and at the bottom of the reactor vessel. Since both the bottom and top signals arise from photoneutrons, the ratio of bottom counts to top counts should be indicative of the ratio of the masses of water below and above the core. After calibration, a decrease in this ratio would be assumed to indicate a decrease in the mass of water and hence the water level above the core.

The system tested at the Farley Unit One Nuclear Power Plant of Alabama Power Co. consisted of externally mounted $^{10}BF_3$ neutron detector assemblies. Four sets of double chamber detectors were installed on the top of the reactor vessel. In the complete system another set of eight neutron detectors is intended to be installed below the reactor vessel. These were not installed, however, for the November 1980 tests. The exact configuration has not been decided. A 1-cm-thick lead plate is provided between the vessel and neutron detectors for γ-ray shielding. Each detector assembly consists of two stainless steel, 5 cm diam, 60 cm long, $^{10}BF_3$ filled, lead shielded detector tubes encased in a plastic block for neutron moderation. Because of the large difference in photoneutron production between power and shutdown conditions, a smaller, less sensitive set of ^{10}B lined detectors will be provided for use during power operation.

Because of this large difference in photoneutron production, it is estimated that there will be a transition period of the order of 10 min associated with delayed neutron decay immediately after shutdown during which uncertainty in water level indication may exist.

On approaching inadequate core cooling, the reduction in water density will cause an increase in neutron streaming, a reduction in the γ-driven neutron source, and a decrease in the core neutron multiplication. The streaming effect is expected to predominate, thus the bottom to top count ratio is expected to decrease steadily as the water level falls, but a possibility for ambiguity exists. Experiments have shown that the ratio's decrease is not very significant until the water level drops below about 150 cm above the core. There, sensitivity to level changes increases substantially. When the core is uncovered the upper detector count rate increases dramatically. However, as the level drops below the core top, the reduced photoneutron generation and decreased neutron multiplication could cause the top readings to decrease. A possibility of ambiguous indication exists in this level range since a decreasing count rate could be interpreted as an increasing water level.

The limiting conditions for the applicability of the NICLM system are: 1) the inability of the monitor to accurately measure level in the vessel for levels greater than five ft above the top of the core, 2) the inability of the monitor to measure level when the level is very near or below the top of the core, 3) the low counting rate of the system, 4) sensitivity of count rate vs level calibration to other radioactive sources, such as fission products in the coolant, and 5) present lack of test data for normal power operation of top detector monitors and the normal operation and shutdown operation of the top-to-bottom ratio monitoring system.

While the various experiments and calculations have shown the ability of the approach to detect changes in water level under some very limited circumstances, a viable and reliable system for measuring water level under all necessary conditions has *not* been demonstrated.

Based on the information available, we believe that the ex-core neutron detector level measurement system is not a likely short-term solution for reactor vessel level monitoring. Development of the techniques has not progressed to the point where it could be of use to reactor operators for detection of ICC conditions. It is concluded that the instrument system does not conform to minimum requirements for accident monitoring instrumentation, or the minimum requirements of NUREG-0737.

3.8.17. EVALUATION OF THERMAL DEVICES FOR DETECTING IN-VESSEL COOLANT LEVELS IN PWRs

J. E. Hardy* C. E. Davis*
K. G. Turnage* R. L. Anderson

(Abstract of NUREG/CR-2673 (ORNL/TM-8306), August 1982]

From investigations conducted immediately after the Three Mile Island nuclear power plant accident, some safety areas needing improvements were identified. One new Nuclear Regulatory Commission requirement was the unambiguous detection of the approach of inadequate core cooling. Designs to meet this requirement have generally included new instrumentation to monitor the coolant level in the reactor vessel. Thermal sensors proposed for use in pressurized-water reactor (PWR) vessels were tested and evaluated. The thermal devices tested use pairs of K-type thermocouples or resistance temperature detectors to sense the cooling capacity of the medium surrounding the device. One sensor of the pair is heated by an electric current, while the unheated one senses the ambient fluid temperature. The temperature difference between the heated and unheated sensors provides an indication of the cooling capacity of the surrounding fluid. Experiments that simulated the thermal-hydraulic conditions of a postulated PWR loss-of-coolant accident (LOCA) were run, including both natural- and forced-convection two-phase flow tests. Results suggest that thermal level devices generally indicate the existence of poor cooling conditions in LOCA environments. In some cases, however, the indication of the thermal devices may not be a direct measurement of water level. Shielding and separator tubes have been devised to ensure that the thermal sensors indicate the collapsed liquid level inside the separator tube. Preliminary evaluation of these protection systems is given.

*Engineering Technology Division.

3.8.18. AUTOMATED FACILITY FOR TESTING AND CALIBRATION OF DIFFERENTIAL PRESSURE TRANSMITTERS

G. N. Miller R. L. Anderson
R. F. Spille W. L. Zabriskie

(Abstract of *Proc. 27th International Instrumentation Symposium*, Indianapolis, Indiana, April 27–30, 1981, 443–50)

An automated test facility for differential pressure transmitters was designed and constructed at Oak Ridge National Laboratory (ORNL). This facility enables transmitters of standard ranges from 4 to 1000 in. H_2O to be characterized, calibrated, and tested to determine the effects of many external variables. The absolute accuracy of the system (expressed as a percentage of the span of the transmitter under test) is fixed by the accuracy of the pressure regulation loop which varies from 0.055% for zero-based spans to 0.286% for typical worst-case, elevated spans.

3.8.19. HIGH TEMPERATURE, HIGH PRESSURE WATER LEVEL SENSOR

G. N. Miller
R. L. Anderson
S. C. Rogers
L. C. Lynnworth*
W. B. Studley*
W. R. Wade*

[Abstract of *IEEE Ultrasonics Symposium*, pp. 877-81 (1980)]

A sensor was developed to measure water level over a range of 750 mm with an uncertainty of ±20 mm at a temperature from 20 to 250°C and pressure up to 15.2 MPa. The sensor is type 304, flattened stainless steel rod. Its cross section is 1.6 X 3.2 mm, and its measured torsional transit time is a function of water density, ; level, *L*; and temperature, *T*. To minimize the influence of *T*, the extensional transit time is also measured in the same sensor. To interrogate the sensor with both modes, Joule and Wiedemann transducers are multiplexed in an alternating sequence. Experimental results, problems, and remedies are discussed.

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3.8.20. ULTRASONIC LEVEL, TEMPERATURE, AND DENSITY SENSOR

S. C. Rogers G. N. Miller

[Abstract of *IEEE Trans. Nucl. Sci.* NS-29(1) (February 1982)]

A sensor has been developed to measure simultaneously the level, temperature, and density of the fluid in which it is immersed. The sensor is a thin, rectangular stainless steel ribbon which acts as a waveguide and is housed in a perforated tube. The waveguide is coupled to a section of magnetostrictive material which is surrounded by a pair of magnetic-coil transducers. These transducers are excited in an alternating sequence to interrogate the sensor with both torsional ultrasonic waves, utilizing the Wiedemann effect, and extensional ultrasonic waves, using the Joule effect. The measured torsional wave transit time is a function of the density, level, and temperature of the fluid surrounding the waveguide. The measured extensional wave transit time is a function of the temperature of the waveguide only. The sensor is divided into zones by the introduction of reflecting surfaces at measured intervals along its length. Consequently, the transit times from each reflecting surface can be analyzed to yield a temperature profile and a density profile along the length of the sensor. Improvements in acoustic wave dampener and pressure seal designs enhance the compatibility of the probe with high-temperature, high-radiation, water-steam environments and increase the likelihood of survival in such environments. Utilization of a microcomputer to automate data sampling and processing has resulted in improved resolution of the sensor.

3.8.21. CALORIMETER FOR MEASURING AC MAGNETIC LOSSES IN SMALL SAMPLES

G. S. Canright R. L. Anderson

(Abstract of paper presented at The Third Joint Intermag—Magnetism and Magnet Materials Conference, Montreal, Canada, July 20-23, 1982)

A calorimeter has been developed to measure the total core loss in small samples of magnetic materials. The calorimeter is similar to the helium boil-off calorimeter used in some cases to measure ac losses in superconductors.[1,2] Power losses in the sample are determined by measuring the rate of boiling of a fluid in which the sample is immersed. Mass flow of the vaporized fluid is obtained using a volumeter, cathetometer, and timer. AC magnetic excitation is provided by an air solenoid.

The method has been verified using a known resistive power input. Uncertainty of the method is estimated at 3%; however, in principle the method is capable of considerably greater precision. The operating temperature is easily changed by a change in the calorimetric fluid and/or operating pressure. Both liquid nitrogen and Freon-114 have been used as the calorimeteric fluid, giving loss measurements at 77K and 248K. To date measurements have been made at flux densities up to 1.7T, and at power frequency (60 Hz). However, the method may be used at any frequency and flux density up to a total core loss of 0.5 W.

1. "1982 Annual Book of ASTM Standards," Part 44, ASTM, Philadelphia, 1982, pp. 47-60, 68-107, and 155-63.

2. D. A. Ball and H. O. Lorch, *J. Sci. Instrum.*, 42, 90 (1965).

3.8.22. ANALYSIS OF A CANTILEVER CORIOLIS MASS FLOWMETER CONCEPT

W. R. Hamel

A new flowmeter concept which simultaneously measures density, mean velocity, and mass flow rate has been conceived, analyzed, and experimentally verified. The concept is based on the dissipative effect of Coriolis inertial forces present in a vibrating hydroelastic cantilever. The concept also demonstrates: (1) the use of dynamic governing equations as a basis of measurement and, (2) the value of microprocessor algorithm implementation.

The cantilever Coriolis flowmeter concept was analyzed using literature associated with hydroelastic system stability as a basis. The governing equation is a fourth-order non-self-adjoint partial differential equation for which no closed-form solution is known. The boundary value problem associated with the cantilever geometry was solved numerically. Approximate single-mode analytical solutions were obtained using perturbation theory and a related simplified technique. In the range of interest, all three solutions agreed very well.

The theoretical analysis showed that the dynamic response of the cantilever flow sensor with static deflection motion excitation is essentially a second-order damped response in which the decay constant and frequency of vibration are dependent on the flowing fluid density and mean velocity. For this form of motion excitation, the cantilever Coriolis flowmeter operates as a sampled-data sensor. The analytical results were used to develop algorithms which transform the recorded transient waveform decay and frequency characteristics into fluid density, mean velocity, and mass flowrate.

A proof of principle experiment was performed to verify the analytical results. Using a microprocessor-based data acquisition and computation system three cantilever flow sensors were evaluated over a range of 0 to 1000 kg/h using a simple water circulation loop. Each cantilever Coriolis flowmeter was compared against an industrial thermal flowmeter. Thin wall tubing sensors worked best in spite of difficulties with multimodal response characteristics. The mass flow measurement tracking between the cantilever Coriolis and the thermal flowmeters was good with relative linearity in the range of +5% to +10%. Density measurement results between the Coriolis density calculation and the known water density were +1% and +5%. Follow-on work will include prototypic experimentation and evaluation of several enhancements.

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*Doctoral Dissertation, The University of Tennessee, December 1981.

3.8.23. AN INVESTIGATION OF THE EFFECT OF THE URANYL ION ON PROTON SPIN RELAXATION TIMES IN AQUEOUS SOLUTIONS

Amanda W. Renshaw

(Abstract of thesis for the Master of Science Degree, The University of Tennessee, 1982; also ORNL/TM-8539, to be published)

The purpose of this project was to determine the effect of the uranyl ion on the spin-lattice relaxation time of protons in aqueous solutions.

A flowmeter using nuclear magnetic resonance (NMR) techniques is being designed for monitoring the flow rate of aqueous solutions which are acidic, contain plutonium and uranium ions, chemical waste, and radioactive waste materials. The spin-lattice relaxation time is critical to the design of a flowmeter using nuclear magnetic resonance techniques.

Experimental studies showed that spin-lattice relaxation time did not change for different molar concentrations of sample solutions of uranyl nitrate and uranyl acetate prepared from pure laboratory chemicals dissolved in distilled water. Therefore, it was concluded that the uranyl ion does not contribute to the relaxation process for protons in water.

3.9. Fossil Energy Studies

3.9.1. STATE OF THE ART ASSESSMENT OF COAL PREPARATION PLANT AUTOMATION

K. R. Carr — J. M. Jansen, Jr.
G. O. Allgood — L. N. Klatt*
R. L. Anderson — D. D. McCue
W. H. Andrews — R. L. Moore
N. C. Bradley — J. C. Moyers†
C. H. Brown‡ — L. E. Ottinger
G. S. Canright — E. R. Rohrer
G. M. Caton** — G. V. Seaver††
W. R. Hamel — D. S. Walia‡‡
J. T. Hutton — G. R. Wetherington

(Abstract of ORNL-5699, February 1982)

As the basis for a continuing program to optimize the performance of coal preparation plants through the application of modern instrumentation and controls technology, the state of the art in the industry was assessed. Coal preparation literature was surveyed comprehensively, coal preparation facilities were visited, and discussions were held with workers in the industry. The object was to determine the feasibility of and need for increased automation, to identify areas where instrument development is needed, and to analyze characteristics of the coal industry relevant to further automation.

The salient conclusions of the study are: (1) the coal preparation industry in the United States derives little of the benefit available from instrumentation and controls technology, (2) rapid recovery of capital investment will result from automation, and (3) an appreciable savings of a valuable energy resource can be effected through the orderly development of an automation program.

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**Information Division.
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3.9.2. POTENTIAL ECONOMIC BENEFITS FROM PROCESS CONTROL OF COAL PREPARATION PLANTS

N. C. Bradley
G. O. Allgood — J. C. Moyers*

(Abstract of ORNL/TM-5736, October 1981)

Significant quantities of usable coal are annually discarded as waste from coal cleaning plants; in 1978, for example, 25 million tons of coal (valued at $600 million) were discarded as waste in 96 million tons of refuse. Recovery of clean coal from the refuse is not economically feasible. To minimize such losses, maximum recovery of clean coal during initial processing of the run-of-the-mine material is vital.

The purpose of this study was to evaluate the cost-effectiveness of process contol in achieving maximum recovery of clean coal from these plants. Data obtained from coal cleaning plants showed that manual control of operations will not yield maximum recovery of clean coal. On the basis of 1978 production figures, it is estimated that a $10 million cost for development of measurement sensors and process control techniques for coal preparation could be recovered through conservation of clean coal in less than five years.

*Engineering Technology Division.

3.9.3. DYNAMIC MODELING AND SIMULATION OF FROTH FLOTATION AND VACUUM FILTRATION UNITS

G. O. Allgood — G. S. Canright
C. H. Brown, Jr.* — W. R. Hamel

(Abstract of paper presented at Twelfth Annual Pittsburgh Conference on Modeling and Simulation, University of Pittsburgh, Pittsburgh, Pennsylvania, April 30–May 1, 1981)

Dynamic mathematical models are developed to simulate the operation of froth flotation and vac-

uum filtration units as found in the coal preparation industry. A set of ordinary differential and algebraic equations is derived for each system based on mass balances. The models are then implemented using IBM's Continuous System Modeling Program (CSMP). The open-loop response of the systems is presented.

*Chemical Technology Division.

3.9.4. TRANSIENT MODELING OF FROTH FLOTATION AND VACUUM FILTRATION PROCESSES

C. H. Brown, Jr.* G. S. Canright
G. O. Allgood W. R. Hamel

(Abstract in *Proceedings of the 1981 Symposium on Instrumentation and Controls for Fossil Energy Processes*, San Francisco, California, June 8–10, 1981; Argonne National Laboratory Report, January 1982)

Transient models of the froth flotation and vacuum filtration processes as applied to fine-coal beneficiation are presented. The models consist of sets of simultaneous ordinary differential and algebraic equations which were derived based on the principles of conservation of mass. The froth flotation process model was developed by drawing the analogy between a single flotation stage and a continuous stirred-tank reactor. Flotation kinetics were described by a model that is first order in displacement of solids concentration from equilibrium. The kinetic parameters were obtained by analysis of published experimental data. The vacuum disc filter model was developed by subdividing the filtration cycle into filtration, drying, and blowoff. The filtration and drying portions of the operating cycle were described by classical representations of these two unit processes. Equations that describe the filter vat dynamics are also presented, thus completing the mathematical representation of filtration.

The application of modern process control technology in the coal preparation industry has become more important due to the projected increase in coal usage and costs, the stiffer environmental regulations, and the fact that available coal generally requires more stringent cleaning than in the past. Mathematical analysis of coal preparation processes facilitates detailed studies in the areas of process control and economics and evaluation of existing and potential instrumentation devices.

In a typical coal preparation plant, run-of-mine coal is initially crushed to a maximum size of 10 cm (4 in.). This coarse material is subsequently split, using a combination of vibrating screens and cyclones, into a coarse fraction, 10 cm by 0.6 cm (4 in. by ¼ in.); a medium size fraction, 0.6 cm by 0.06 cm (¼-in. by 28 mesh); and fine coal that is less than 0.06 cm in diameter (28 mesh by 0). Cleaning operations used in the coarse coal circuit are jigs and heavy media washers.

*Chemical Technology Division.

3.9.5. DYNAMIC MODELING AND CONTROL ANALYSIS OF FROTH FLOTATION AND CLEAN COAL FILTRATION AS APPLIED TO COAL BENEFICIATION

G. S. Canright G. O. Allgood
C. H. Brown, Jr.* W. R. Hamel

(Abstract of ORNL/TM-8015, November, 1981)

Dynamic models were developed for coal beneficiation plant froth flotation and vacuum disk filtration processes to perform comparative analyses of manual and automatic control techniques and determine if implementation of automatic control would be cost effective. The froth cell simulator was based on a tanks-in-series model utilizing first-order flotation kinetics. The vacuum disk filter model was based on classical representations of tank mixing, filtration, and drying. Both models were implemented on the Continuous Systems Modeling Program on the IBM-360 computer system.

Three types of control were implemented on the froth flotation simulator: a manual operator control, fixed reagent flow-rate control, and automatic feedforward control. Results indicate a definite economic incentive for implementation of automatic control in these flotation units. Calculated times to recover the cost of a typical system range from 5 months to 2 years, depending on the frequency of the process disturbances and the cost of coal.

A manual control routine and an automatic control system, based on fixing the volumetric feed rate via a flow dampening surge tank and control valve, were implemented on the filter model to determine if automatic control would improve cake moisture control. The results indicate that significant improvements were not realized with this type of automatic control. It is possible, however, that second-order effects, such as reduced operator requirements and reduced maintenance, could make automatic control desirable for filtration. An evaluation of these factors is beyond the scope of this report.

*Chemical Technology Division.

3.9.6. DYNAMIC SIMULATION OF THE COAL FROTH FLOTATION AND FILTRATION PROCESSES

G. S. Canright G. H. Brown, Jr.*
G. O. Allgood W. R. Hamel

(Abstract in *Proceedings of the 1981 Symposium on Instrumentation and Control for Fossil Energy Processes*, San Francisco, California, June 8-10, 1981)

A digital computer simulation is presented for two unit operations from the fine-coal circuit of coal preparation plants. Transient mathematical models of the flotation process and the fine-coal filter were the basis for the simulation, which was carried out using the IBM Continuous System Modeling Program (CSMP). Control strategies representative of those currently in use, as well as a proposed automatic control, were modeled and applied to each unit. The fine-coal filter performance was expressed in terms of cake moisture content at discharge. Performance of the flotation unit was quantified in terms of overproduct ash content and salable coal lost to the tailings. A standard feed disturbance was input to the flotation process and carried through the overflow to the filter. Performance of the various control schemes was compared according to the above criteria. The two filtration controls showed no significant difference in performance; however, the flotation results gave a definite ranking of the three control strategies. A simple economic analysis (payout period) was performed for the two flotation control alternatives that yielded the best performance. The analysis indicated that substantial savings could be realized by the automatic control of froth flotation.

*Chemical Technology Division.

Section 4

MAINTENANCE MANAGEMENT

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4.0. OVERVIEW

P. W. Hill

The purpose of the Maintenance Management Department is to further the mission of Oak Ridge National Laboratory by maintaining an effective maintenance program capable of providing maintenance, fabrication, modification, installation, calibration, testing, and operation of instruments and controls in support of research and development efforts.

The Maintenance Management Department provides products and services to divisions throughout ORNL as well as to the three other sections within the Instrumentation and Controls Division. A computerized maintenance information system (MAINS) is utilized to provide historical data, preventive maintenance scheduling, backlog summaries, service documents, and other information related to the maintenance of instrumentation. Department manpower consists of 22 management and staff personnel, 12 technical support personnel, and 101 instrument technicians. The Department includes forty shops, with staffs varying from one to twenty, strategically located throughout the Laboratory. The Department's annual expenditure for maintenance and fabrication is over $5 million, providing service for 35,000 documented instruments with a total value in excess of $51 million.

Recognizing that improvement of maintenance management is a continuous process, it is necessary to maintain progressive attitudes and the active involvement of personnel. A results-based program has been initiated to establish objectives and achieve goals within a reasonable time frame. Maintenance Management Department changes in direction shall be carried out in a planned and systematic manner by use of a documented management plan and a computer-based job control system, which has been designed to operate in connection with the maintenance information system (MAINS), already established.

4.1 MAINTENANCE MANAGEMENT PLAN FOR THE MAINTENANCE MANAGEMENT DEPARTMENT

D. G. Prater

The development of a management plan for the Maintenance Management Department is under way. Its purpose is to upgrade the maintenance program through the continuous, progressive application of management skills in order to gain better utilization of the maintenance dollar.

The plan will outline the elements of effective management deemed necessary for planning, organizing, leading, and controlling a successful maintenance organization. Management by objectives (MBO) techniques will be employed to establish realistic goals and develop a plan of action to attain those goals within a reasonable time frame. The plan will identify policy, define responsibilities, establish measures of performance, and describe activities to organize and control the maintenance function. It will be designed to reveal both deficiencies and opportunities, as well as establishing objectives and determining priorities.

Important elements of the plan will include an interactive computerized job control system, a computer-based maintenance information system (MAINS), training activities, measures of performance, and a priority system. These elements will be used to develop short- and long-range plans, document actions and changes to correct deficiencies, and take advantage of opportunities. The plan will address problems concerning quality, procurement, material control, safety, and document controls. Human factors will be addressed through employee selection, labor relations, public relations, and affirmative action.

4.2 DEVELOPMENT OF A JOB CONTROL SYSTEM FOR THE MAINTENANCE MANAGEMENT DEPARTMENT

D. G. Prater

A job control system is a management tool used to assist and document work planning, work scheduling, cost estimation and collection, backlog reporting, job status, and work performance.

To meet these requirements, an I&C Maintenance Work Request has been developed to collect the needed information. The proposed job control system will be interactive, allowing access from each supervisor location. The system is built around the 1022 data base management program maintained in the PDP DEC-10 computer. Maintenance information normally stored by an IBM batch card system will be transferred to the 1022 data base on a weekly basis for further analysis and job control. The Maintenance Information System (MAINS) will continue to provide a maintenance history of each instrument. The Job Control System will offer such features as unique job identification, accrual of material and labor costs as they occur, work order justification, a printed copy of labor and material estimates, exception reporting of backlog, job scheduling, estimate overruns, etc. Scheduled recall for preventive maintenance and calibration will be maintained in the computer-based maintenance information system (MAINS). A file of document numbers will identify the proper maintenance document associated with each instrument. A spare parts inventory will be maintained, indicating the availability and location of spare parts. This system will be compatible with the Job Control System currently used by the P&E Division, with the added ability to identify material costs as they accrue and flag instruments overdue for scheduled preventive maintenance or calibration.

4.3 WORKMANSHIP STANDARDS—DEVELOPMENT AND TRAINING

C. T. Stansberry

A Workmanship Standards Program was developed and presented in training sessions conducted with the overall objective of improving the quality of workmanship by establishing standards for electrical and mechanical fabrication and maintenance techniques for work performed both in the field and in-house.[1]

The I&C Division engineering, research, and maintenance staff reviewed current practices and procedures of private industry and other government agencies such as the National Aeronautics and Space Administration (NASA) to establish standards that would support the overall objectives of the Maintenance Management Department for the improvement of workmanship. Additional information was solicited within the Division and collected for review and consideration. From the material collected, a Workmanship Standards Manual was prepared. After review for correctness and accuracy by key personnel, the necessary procedural steps were taken for adoption of the manual as a quality assurance document of the Instrumentation and Controls Division, identified as Operating Procedure IPD-13, *Workmanship Standards, Electrical and Mechanical.*

The IPD-13-manual documents the requirements established in the following three areas: (1) minimum requirements for the layout, etching, and fabrication of printed circuit (PC) boards; (2) design and inspection requirements for artwork to be used for fabricating rigid, single-layered, etched wiring boards with a conductive pattern on one or both sides; and (3) design and inspection requirements for artwork for photographed metal tags and plates of anodized aluminum. The manual also establishes a minimum level of workmanship for the fabrication and maintenance of instrumentation, including field maintenance, and clarifies acceptance and rejection criteria for workmanship for those cases in which the basic measure of quality is largely subjective.

Nine training sessions were conducted, divided into three major areas:

1. The history and organization of the Workmanship Standards Manual was explained, the training was outlined, and the need for documentation with respect to quality assurance was identified.
2. Subjective criteria were described for three work performance levels: preferred, acceptable minimum, and rejectable.
3. The technicians received "hands-on" experience of printed circuit soldering and desoldering techniques.

All line supervisors and technical support personnel and 97% of the instrument technicians completed the training program. Through class feedback and discussions, additional information relating to working conditions, tools, and situations existing in Department shops was received and documented for further development to improve the quality of workmanship. *(B. A. Denning, A. A. Smith, D. G. Prater).*

1. C. T. Stansberry, Workmanship Standards Training Report, ORNL/CF-82/85, June 1, 1982.

4.4 A MICROPROCESSOR-BASED TERMINAL INTERFACE FOR SPECIAL CHARACTER CONVERSION

R. J. Bradford

The code converter developed for special character conversion is based on the Intel 8080 microprocessor and has one kilobyte of read-only memory, one kilobyte of read/write memory, and two serial input/output ports. It is designed to convert control codes from various terminals into codes acceptable to minicomputers. The converter may also be used to convert the ASCII character set to the EBCDIC format.

Code conversion is accomplished with a look-up table stored in a read-only memory. When power is applied (or during reset), the table is automatically transferred to the read/write memory. For each control character to be converted, the operator activates code conversion with a command to the terminal. The converter receives the code from the terminal, replaces it with the proper conversion code from the look-up table, and transmits it to the minicomputer.

The converter is designed to allow bi-directional communication between terminals and minicomputers. Its baud rate is switch-selectable between 110 and 9600 bps, and word configuration is accomplished by a program stored in a read-only memory.

This device has greatly increased the degree of flexibility possible between communication devices and minicomputers. *(W. W. Engle, Jr.*)*

*Engineering Physics Division.

4.5 REPORT ON SHOPS

The Maintenance Information System (MAINS) is a computer-based system which provides a maintenance history and a recall system for the preventive maintenance and calibration of instruments at ORNL. The system has been designed to improve the efficiency of the Division's maintenance support and to meet the requirements of Department of Energy (DOE) directives, UCC-ND standard practice procedures, and the ORNL quality assurance program. Although the system is utilized principally by the Instrumentation and Controls Division in its maintenance function, it is available to other divisions of the Laboratory upon request.

A Maintenance Document System has been developed for the preservation of maintenance documentation using microfiche as the storage medium. Reported during the previous period, this system continues to expand and now has the appropriate document numbers listed in the MAINS inventory with each instrument for faster access to necessary maintenance information. A MAINS document listing is issued periodically.

The Research Instrument Maintenance Section provides support through matrix organization to the Research Instrument Section.

The technical support persons are usually Engineering Technologists, who were promoted from instrument technician based on requirements, ability, work habits, and other motivational factors. Engineering Technologists generally work with instrument technicians in teams dedicated to service of specific types of equipment. Their tasks include further training of instrument technicians, maintaining the replacement parts inventory, collection and preservation of maintenance documentation, and supervisory relief.

Training, adaptability, and adjustments to constantly changing technology and support requirements in the operations and research areas have been stressed. The section has been restructured to improve cost-effectiveness and productivity in fabrication, support of engineering, onsite research, and operation maintenance. The section is responsible for such diverse maintenance responsibilities as health physics instrumentation, test equipment, oscilloscopes, computers, terminals, pulse height analyzers, two-way communications, audio and video systems, security systems, and teletypes.

A bench stock inventory and standardized parts storage system has been implemented for the department. Seven shop areas are presently included, and the system will soon be department wide.

The Electronic Instrument Fabrication Group supports I&C Engineering, providing services in areas of module construction, prototypes, bread boards, modifications, and radiation detector construction/repair. The group is responsible for the construction of engineering models, prototypes, limited production runs of instruments not built by commercial companies, and for the production of etched wiring boards and photometal products.

The etched board and photometal operation was moved to a larger facility and upgraded with new equipment. Continued improvement has been realized in the production of plated through-hole and flexible etched wiring boards. Chemical milling and photometal work constitutes 15% of the work load in this area.

Approximately 80% of the group's fabrication effort was directed toward research instruments designed by engineers of the Instrumentation and Controls Division, and the remainder applied to special projects of other divisions and in-house fabrication of instruments and cables for ORNL stores. A special commitment was made to the inspection of store stock items in electronic stores.

Interfacing with all engineering groups is accomplished on an engineer/technician basis for most fabrication work while maintaining matrix responsibility with the Product Design Group of the Research Instrument Section.

The Accelerator Maintenance Group provides on-site maintenance support for the Physics and Engineering Physics Divisions at the Holifield Heavy Ion Research Facility (Building 6000) and the Oak Ridge Electron Linear Accelerator (Building 6010). Accelerator and research instruments are maintained by a group of instrument technicians on two shifts.

North and West Area Maintenance Shops: Each of the three area shops of the north and west quadrants of ORNL is staffed by an instrument technician reporting to an area supervisor.

The Solid State Division's maintenance support is provided by a shop located in Building 3001. A shop in Building 2000 provides maintenance support for the Department of Quality Assurance and Inspection.

The Environmental Sciences Division is served by a shop located in Building 1505 providing maintenance support in that division's laboratories and in the ORNL environmental study areas.

These shops derive engineering support from the Special Electronics Group of the Research Instrument Section. Additional instrument technician support is coordinated by a supervisor assigned to the Pulse Height Analyzer Group. This group is located in Building 3001 and is engaged in the maintenance of pulse height analyzers, liquid scintillation counters, teletypes, and other specialized equipment. Their service area is plant wide, and the shop personnel are divided into teams responsible for specific equipment types.

Teletype usage at ORNL is on the decline, and presently only one instrument technician is assigned this task. An effective preventive maintenance program continues to hold downtime to a minimum.

The acquisition of pulse height analyzers, both hard-wired and computer-based, continues, and some of the older technology units are being replaced. This has maintained a stable equipment level through this report period. A technical support person and three instrument technicians are assigned the maintenance task for these instruments and for the Spiral Reader System. This is a dual computerized bubble chamber film reading and analysis system.

The Research Instrument Section also supplies engineering support for the Computer and Terminal Maintenance Group. Maintenance is supplied for plotting equipment, modems, multiplexors, communications concentrators, and associated devices utilized with hard-wired and dial-up systems. Fault isolation is performed on all communication lines and for two communication computers used to control the inputs to the IBM computers. These inputs are from batch station, RECON, and dial-up phone lines.

Maintenance for computer terminals and modem/acoustic couplers, DEC computer systems, and personal computers is provided through this shop. Both areas are expanding at a rapid rate.

Remote Job Entry or batch stations, Hewlett-Packard computers, and scientific calculators are maintained by a team assigned to this shop.

The remaining team in this group has the greatest diversity of equipment in SEL computers, numerical control equipment, the PDP-15 computer systems, and the computerized equipment transferred from Ames Laboratory to the ORR.

The Computer and Terminal Maintenance Group continues to expand, the most significant growth being in computer terminals, which have doubled in numbers since 1980.

The supervisor for the computer maintenance group is responsible for both the Chemistry Division shop in Building 4500N and the Test Equipment Maintenance shop in Building 3500. These groups formerly interchanged personnel on a routine basis but now are identifiable entities with distinct training and work responsibilities for specific work areas. This is expected to improve productivity.

An automated calibration system is being installed which will dramatically reduce labor costs incurred in the calibration of oscilloscopes and plug-ins at ORNL.

The Monitoring Maintenance Group provides maintenance of portable survey and fixed station radiation and other monitoring equipment for the Health and Safety Research Division. The Operations Division's waste monitoring instruments are also serviced by this group.

A shop in Building 2007 provides maintenance of portable survey radiation instruments, and a shop in Building 3026 has maintenance responsibility for local air monitoring instruments at the ORNL plant site and remote area locations. A 100-m meteorological tower with three instrument levels and two 30-m meteorological towers with two instrument levels have been added to the work load of this group. The instrumentation for each tower must be checked and calibrated at three-month intervals. Forty-eight local air monitors and an off-gas stack monitor are presently being upgraded with microprocessor-based equipment.

Technicians assigned to a shop in Building 3130 are working with the Operations Division's waste program. The Intermediate Level Waste Facility, Tank Farms, Shale Fracture, and Central Control Room are primary work locations.

The Audio-Visual Group in Building 3587 provides maintenance for three areas of specialized instrumentation and audio-visual services for ORNL at scientific and administrative meetings. Support service was provided for both in-plant meetings and out-of-plant meetings.

Additional efforts include maintenance support for building sound systems and projection equipment at ORNL. The video equipment, including closed circuit TV and 25 miles of CATV for security surveillance, are also maintained by this team.

Other security instruments and systems are serviced by a technician located in a secure area of Building 3587.

The Laboratory has five radio networks for two-way communications in security, fire protection, monitoring and system checkout, work control, and management requirements. There is also a radio paging system at ORNL with a transmitter and 200 paging receivers. The Radio Shop is staffed and equipped to maintain and certify all radio equipment.

The radio systems are currently undergoing an upgrading program.

The Measurement and Control Section maintains process instrumentation, leak detectors, thermocouples, gas chromatographs, spectrophotometers, valves, induction heaters, and analytical instrumentation. The group also fabricates "heavy" panel boards.

The Process Instrument Group is responsible for instrument maintenance at the Transuranium Processing Plant, the Radiochemical Processing Pilot Plant, the Steam Plant, and the Consolidated Fuel Reprocessing Plant. Efforts of this group were also devoted to providing maintenance support for customer instrument requirements, calibrations for quality control, fabrication of instruments, inspection of store stock items, and maintenance of leak detectors. They also fabricated, tested, and inspected thermocouple parts and assemblies. Approximately 30% of the maintenance work was for research instrument support for the Instrumentation and Controls Division, and 70% was utilized by other divisions throughout ORNL.

Tests of the Gunite Sludge Removal Facility instrumentation have been completed. Test and installation of instruments for the CFRP and CEUSP programs are continuing.

The Analytical Instrument Maintenance Group provides maintenance of chemical analyzers and induction heaters, as well as instruments for the tobacco smoke research, Walker Branch watershed, and coal gasifier groups.

The Metals and Ceramics Maintenance Group provides maintenance and small-scale design support to the Metals and Ceramics Division. Several closed-loop electrohydraulic fatigue test machines were modified to provide for complex tests of both strain and load control. Further, because all of the tests are conducted at elevated temperatures and for long periods of time, the control and failsafe devices were integrated to protect tests in progress from line transients and power outages. The group also provided day-to-day maintenance services for this equipment.

The Panelboard Fabrication Facility fabricated transmitter panels for the integrated test facility of the Consolidated Fuel Reprocessing Plant. Panelboard construction for the Consolidated Edison Uranium Solidification Program continued in addition to the routine repair work on used valves for other ORNL facilities.

Consolidated Fuel Reprocessing Facility maintenance continued in the areas of fabrication, installation, calibration, and testing of instruments associated with the various systems and experiments.

Three technicians are responsible for the maintenance of a broad range of instrumentation such as microprocessor-controlled digital pressure sources, pneumatic instruments, analytical instruments, and microprocessor-based telemetry systems for the transmission of data.

The Reactor Controls Group continues to provide maintenance support to all operating reactors at ORNL. This crew is unique in that each maintenance technician is certified for reactor controls work.

C. G. Allen	*J. H. Day*	*A. J. Millet*
C. C. Barringer	*B. A. D.,ming*	*J. Miniard*
J. D. Blanton	*R. P. Eppler*	*E. G. Price*
R. J. Bradford	*J. M. Farmer*	*R. P. Rosenbaum*
H. L. Brant	*J. A. Goan*	*C. A. Smith*
R. H. Brown	*J. A. Keathley*	*C. T. Stansberry*
J. Campbell	*C. W. Kunselman*	*C. W. Tompkins*
B. L. Carpenter	*D. J. Marshall*	*B. A. Tye*
R. M. Childs	*J. W. McNeillie*	*W. H. Williams*
C. R. Cinnamon	*R. L. McKinney*	

APPENDIX

SUPPLEMENTARY ACTIVITIES

The Instrumentation and Controls Division maintains liaison with industry and the academic community through its Advisory Committee and consultants, and through student and faculty research and training programs carried on within the Division.

Advisory Committee—1982

C. Lester Hogan, Fairchild Camera & Instrument Corporation, 464 Ellis Street, Mountain View, CA 94042

J. Venn Leeds, Jr. (Chairman), Professional Engineer, 10807 Atwell, Houston, TX 77096

Charles A. McKay, Corporate Director of Technology, The Foxboro Company, 38 Neponset Avenue, Foxboro, MA 30205

Paul W. Murrill, Board Chairman and Chief Executive Officer, Gulf States Utilities Company, Post Office Box 2951, Beaumont, TX 77704

Bernard M. Oliver, Vice President, Research & Development, Hewlett-Packard Company, 1501 Page Mill Road, Palo Alto, CA 94304

Herbert E. Trammell, Director, Engineering Technology Division, Oak Ridge National Laboratory, Post Office Box Y, Oak Ridge, TN 37830

Division Consultants

J. D. Allen
C. J. Borkowski
E. P. Epler
R. F. Saxe
M. A. Schultz
J. A. Thie

Oak Ridge Associated Universities Trainees

M. J. Goeckner
A. L. Johnston
M. A. Seaman
R. Y. Liu

Co-op Students

C. Cornils
M. S. Cosson
J. H. Debusk
W. R. Ely
A. L. Fox
A. P. Frissora
T. E. Julian
J. C. Moyers, Jr.
W. L. Oen
M. N. Smith
M. W. Wendel

Engineering Career Opportunity Program

P. D. Ewing
J. Johnson
E. Merriweather

SCIENTIFIC AND PROFESSIONAL ACTIVITIES, ACHIEVEMENTS, AND AWARDS

September 1, 1980 - July 1, 1982

R. K. Adams

Fellow, Instrument Society of America

Member, ORNL Proposal Review Committee

Registered Professional Engineer

D. G. Agouridis

Member, Institute of Electrical and Electronics Engineers

Member, Gamma Alpha

Member, Eta Kappa Nu

U.S. Patent No. 4,243,885, *CdTe Photovoltaic Radiation Detector*, D. C. Agouridis and R. J. Fox, January 1981

Patent filed: *Improved CdTe Photovoltaic Radiation Detector*, D. C. Agouridis, December 1980

G. O. Allgood

Member, Institute of Electrical and Electronics Engineers

A. H. Anderson, Jr.

Member, Instrument Society of America; Secretary, Oak Ridge Section

Member, Institute of Electrical and Electronics Engineers

Member, National Society of Professional Engineers

Registered Professional Engineer

John L. Anderson

Member, American Nuclear Society; Member, Standards Committee Subcommittee ANS-4, Reactor Dynamics and Control

Member, ORNL Engineering Physics Division Safety Review Committee

Registered Professional Engineer

R. L. Anderson

Metric Coordinator, Instrumentation and Controls Division

ORNL Delegate, National Conference of Standards Laboratories

Member, Institute of Electrical and Electronics Engineers

Member, American Vacuum Society

Member, American Society for Computing Machinery

Member, Sigma Xi

Industrial Research & Development 1982 I•R 100 Award, "ORNL In-Core Temperature, Density, and Level Measurement System," G. N. Miller, R. L. Anderson, and S. C. Rogers

Patent Application Filed: *Reactor Coolant Level Sensor*, R. L. Anderson and G. N. Miller, October 1981

S. M. Babcock

Member, American Society of Mechanical Engineers

Best Paper Award: "Electromechanical Three-Axis Development for Remote Handling in the Hot Experimental Facility," J. Garin, B. J. Bolfing, P. E. Satterlee, and S. M. Babcock, presented at *29th Remote Systems Technology Development Conference of the American Nuclear Society*, San Francisco, California, December 1981

Steven P. Baker

Member, Instrument Society of America

Registered Professional Engineer

Invited paper: "Application of a Vortex Shedding Flowmeter to the Wide Range Measurement of High Temperature Gas Flow," S. P. Baker, R. M. Ennis, Jr., and P. G. Herndon, presented at *Second International Symposium on Flow: Its Measurement and Control in Science and Industry*, St. Louis, Missouri, March 23-26, 1981; *Proceedings*, p. 219

S. J. Ball

Member, Society for Computer Simulations

Chairman, ORNL Reactor Operations Review Committee (RORC)

T. R. Barclay

Member, ORNL Director's Laser Safety Review Committee

Registered Professional Engineer

D. D. Bates

Member, Fusion Energy Division Committee on Development of Grounding System, Diagnostics, and Power System for EBT-P

Member, Fusion Energy Division Committee on Major Electrical Equipment Review

Member, Fusion Energy Division Safety Review Committee

Member, Fusion Energy Division Committee to Investigate the Failure of EOV that Resulted in Damage to EBT-S

Ronald E. Battle

Ad Hoc Member, ORNL Reactor Operations Review Committee

Registered Professional Engineer

Martin L. Bauer

Member, American Vacuum Society

Invited paper: "Evaluation of Fiber Optics for In-Line Photometry in Hostile Environments," M. L. Bauer, D. T. Bostick, and J. E. Strain, presented at *25th Annual International Symposium and Exhibit of the Society of Photo-Optical Instrumentation Engineers*, San Diego, California., August 24-28, 1981

Invited paper: "An In-Line Multiwavelength Photometer for the Determination of Heavy Metal Concentrations," D. T. Bostick, D. D. McCue, J. E. Strain, M. L. Bauer, and D. M. Dixon, presented at *25th Conference on Analytical Chemistry in Nuclear Technology*, Gatlinburg, Tennessee, October 6-8, 1981

W. F. Bethmann, Jr.

Member, Instrument Society of America

T. V. Blalock

Member, Transducers Committee of the Industrial Electronics and Control Instrumentation Society of the IEEE

Haliburton Engineering Professorship, The University of Tennessee, 1981

Invited paper: "Survey, Applications, and Prospects of Johnson Noise Thermometry," T. Vaughn Blalock and Robert L. Shepard, presented at *Sixth International Conference on Noise in Physical Systems,* National Bureau of Standards, Gaithersburg, Maryland, April 6-10, 1981

B. J. Bolfing

Best Paper Award: "Electromechanical Three-Axis Development for Remote Handling in the Hot Experimental Facility," J. Garin, B. J. Bolfing, P. E. Satterlee, and S. M. Babcock, presented at *29th Remote Systems Technology Development Conference of the American Nuclear Society,* San Francisco, California, December 1981

R. S. Booth

Member, American Nuclear Society

Member, ORNL Wigner Fellowship Selection Commttee

O. W. Burke

Member, American Nuclear Society

K. R. Carr

Member, Instrument Society of America

U.S. Patent No. 4,251,908, *Side-Welded Fast Response Sheathed Thermocouple,* 1981

Radford M. Carroll

Member, American Nuclear Society

Ned E. Clapp, Jr.

Member, Instrument Society of America

Howard E. Cochran

Registered Professional Engineer

Member, Four-Plant Instrumentation Engineering Standards Committee

R. I. Crutcher

Registered Professional Engineer

Invited paper: "Security Television Monitoring Using a Wideband Radio Frequency Cable System," J. A. Russell, A. L. Case, R. I. Crutcher, and F. E. Wetzel, *1981 Carnahan Conference On Crime Countermeasures,* Lexington, Kentucky, May 1981; *Proceedings,* pp. 19-21

J. W. Cunningham

Member, I&C Division Design and Drafting Standards Committee

J. B. Davidson

Member, Sigma Xi

Patent Application Filed: *TV-Based Locating & Measuring System for Two-Dimensional Gels*, J. B. Davidson and A. L. Case, December 1981

J. H. Day

Senior Member, Instrument Society of America

Member, I&C Division Maintenance Information System Committee

B. G. Eads

Member, Instrument Society of America

Invited paper: "In-Vessel Instrumentation for High-Temperature Transient Two-Phase Flows," presented at *Eighth Water Reactor Safety Information Meeting*, Gaithersburg, Maryland, October 27-31, 1980

P. D. Ewing

Member, Institute of Electrical and Electronics Engineers

D. M. Fleischer

Member, Institute of Electrical and Electronics Engineers

R. J. Fox

Member, Instrument Society of America

U.S. Patent No. 4,243,855, *CdTePhotovoltaic Radiation Detector*, D. C. Agouridis and R. J. Fox, January 1981

Patent Application Filed: *Solid State Radiation Detector Circuit*, September 1981

Dwayne N. Fry

Member, American Nuclear Society

Tom M. Gayle

Registered Professional Engineer

R. C. Gonzalez

Chairman, IEEE Technical Committee on Pattern Recognition and Image Processing of the System, Man and Cybernetics Society, 1982

IBM Professorship, The University of Tennessee, 1981

Invited paper: "Real-Time Image Enhancement," R. E. Woods and R. C. Gonzalez, *Proc. Symposium on Very High Speed Computing Technology*, pp. I.45-I.66 (1980)

Invited paper: "Real-Time Digital Image Processing," R. E. Woods and R. C. Gonzalez, *Proc. IEEE* 69(5), 643-54 (1981)

Invited paper: "Real-Time Digitizer for Stereo Image Processing," R. C. Gonzalez and R. E. Crossley, *J. Pattern Recognition* 14(1-6) 289-96 (1981)

Invited paper: "The Use of Activity in Testing Digital & Analog Systems," B. M. E. Moret, M. G. Thomason, and R. C. Gonzalez, *Proc. Comput. Soc. Automatic Test Program Generation*, 120-27 (1981)

Invited paper: "Spatial Image Processing Masks from Frequency Domain Specifications," E. R. Meyer and R. C. Gonzalez, *Proc. SEG/USN Shear Waves and Pattern Recognition Symposium*, 250-60, (1980)

Invited paper: "An Overview of Digital Image Processing and Recognition," R. C. Gonzalez, *Proc. 40th Annual Symp. Electron Microscopy Soc. of Amer.*, 700-703 (1982)

Invited paper: "Computer Vision Techniques for Industrial Applications," R. C. Gonzalez and R. Safabakhsh, *Computer* 15(12) (1982)

J. M. Googe

Member, Institute of Electrical and Electronics Engineers

Member, American Society for Engineering Education

Registered Professional Engineer

G. C. Guerrant

Member, Iota Lambda Sigma

Edward W. Hagen

Fellow, Instrument Society of America; Managing Editor, Oak Ridge Section *Oak Ridge Recorder*; Control and Instrumentation Section Editor, *Nuclear Safety Journal*

Member, Institute of Electrical and Electronics Engineers; Secretary, Nuclear Power Engineering Committee Subcommittee 7—Human Factors; Newsletter Editor

Member, International Electrotechnical Commission Subcommittee 45A, Reactor Instrumentation

Member, American Nuclear Society; Member, Human Factors Group

Registered Professional Engineer

Gerald Hamby

Member, National Laboratory Electronic Services Committee

William R. Hamel

Member, American Society of Mechanical Engineeers; Member, Oak Ridge Section College Affairs Committee

Member, Institute of Electrical and Electronics Engineers

Member, Sigma Xi

Ph.D. Dissertation: "An Analysis of a Cantilever Coriolis Mass Flowmeter Concept," The University of Tennessee, 1981

R. M. Harrington

Member, ORNL Reactor Operations Review Committee

Invited paper: "Scram Discharge Volume Break Studies, Part I - Accident Sequence Analysis," R. M. Harrington and S. A. Hodge, presented at *Tenth Water Reactor Safety Research Information Meeting*, Gaithersburg, Maryland, October 15, 1982

J. A. Harter

Member, Institute of Electrical and Electronics Engineers

Robert W. Hayes

Member, Digital Equipment Corp. Users' Society; Chairman, East Tennessee Local Users' Group

K. M. Henry

Member, ORNL Burst Reactor Experiment Review Committee

P. G. Herndon

Member, Institute of Electrical and Electronics Engineers

Member, I&C Division Design and Drafting Standards Committee

Invited paper: "Application of a Vortex Shedding Flowmeter to the Wide Range Measurement of High Temperature Gas Flow," S. P. Baker, R. M. Ennis, Jr., and P. G. Herndon, presented at *Second International Symposium on Flow: Its Measurement and Control in Science and Industry*, St. Louis, Missouri, March 23-26, 1981; *Proceedings*, p. 219

M. B. Herskovitz

Fellow, Instrument Society of America

Member, Institute of Electrical and Electronics Engineers

Member, American Society for Testing and Materials; Member, Committee E-40, Technical Aspects of Products Liability Litigation; Member, Committee E-20, Temperature Measurement; Secretary, Subcommittee E-20.4, Thermocouples

Chairman, ORNL Electrical Safety Committee

Registered Professional Engineer

N. W. Hill

Member, U.S. NIM-CAMAC Committee

Paul W. Hill

Member, I&C Division Maintenance Information System Committee

J. H. Holladay

Member, Instrument Society of America

John L. Horton

Member, Instrument Society of America

Member, Institute of Electrical and Electronics Engineers

Registered Professional Engineer

J. O. Hylton

Member, Instrument Society of America

Member, Sigma Xi

Richard E. Hutchens

Patent Application Filed: *Method for Forming Microspheres for Encapsulation of Nuclear Waste,* Peter Angelini, Anthony J. Caputo, Richard E. Hutchens, Walter J. Lackey, and David P. Stinton, March 1982

J. M. Jansen, Jr.

Member, Institute of Electrical and Electronics Engineers

Member, ORNL Computer Steering Committee; Member, Subcommittee on Future Technology of Computing; Chairman, Subcommittee on Cable Communications System Building Coordinators

E. B. Johnson

Fellow, American Nuclear Society; Member, Professional Divisions Committee, 1981-82; Member, Publications Steering Committee, 1982-85; Chair of Special Session, "Role of Women in Plant Operations," ANS 1981 Winter Meeting; Chair, NCSD Program Committee and Representative to ANS National Technical Program Committee, 1980-81; Chair, Nuclear Criticality Safety Division, 1981-82; Member, Oak Ridge/Knoxville Section Bylaws Committee; Member, ANS Standards Committee; Secretary, Subcommittee 8, Fissionable Materials Outside Reactors

Member, American National Standards Institute; Member, Group N16, Nuclear Criticality Safety

Member, Sigma Xi; Oak Ridge Branch Treasurer, 1979-81

Member, New York Academy of Sciences

Member, American Physical Society

Member, American Association for the Advanceent of Science

Chair, Audit Committee for the 1982 Triennial Review of the ORNL Criticality Committee

J. A. Keathley

Chairman, I&C Maintenance Information System Committee

W. T. King

Member, American Nuclear Society

Roger A. Kisner

Member, Audio Engineering Society

Member, IEEE Nuclear Power Engineering Committee Human Factors Working Group SC 5.5; Task Group Discussion Leader, 1981 IEEE Standards Workshop on Human Factors and Nuclear Safety

Member, National Society of Professional Engineers

Registered Professional Engineer

Program Committee Chairman, NRC Workshop on Cognitive Modeling of Nuclear Plant Control Room Operators, Dedham, Massachusetts, August 15-18, 1982

Speaker, ORAU Traveling Lecture Program, "General Review of the Human Factor in Nuclear Power Plant Operators," Virginia Polytechnic Institute & State University, March 9, 1981

Invited paper: "Defining the Role of the Operating Crew," R. A. Kisner and G. F. Flanagan, presented at the Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 27-31, 1980

Invited paper: "A Taxonomy of the Nuclear Plant Operator's Role," R. A. Kisner, A. M. Fullerton, P. R. Frey, and E. M. Dougherty, presented at the OECD Enlarged Halden Programme Group Meeting, Fredrikstad, Norway, June 1981

Manfred K. Kopp

Member, Institute of Electrical and Electronics Engineers; Member, Program Committee for the IEEE Nuclear Science Symposium

Member, Sigma Xi

Registered Professional Engineer

U.S. Patent 4,289,967, *Multianode Cylindrical Proportional Counter for High Count Rates*, M. K. Kopp and J. A. Hanson, 1981

Patent Application Filed: *Parallel Plate Fission Counter with Large Plate Area*, M. K. Kopp and K. H. Valentine, April 1981

Patent Application Filed: *Neutron Flux Profile Monitor for Use in Fission Reactor*, M. K. Kopp and K. H. Valentine, September 1981

Robert C. Kryter

Member, American Nuclear Society

Member, ORNL Proposal Review Committee

Joseph Lewin

Member, American Nuclear Society

James L. Lovvorn

Senior Member, Institute of Electrical and Electronics Engineers

Member, I&C Division Maintenance Information System Committee

E. Madden

Member, UCC-ND ADP Equipment Maintenance Review Committee

Wayne W. Manges

Member, Institute of Electrical and Electronics Engineers; Member, IEEE Computer Society

F. W. Manning

Life Member, Institute of Electrical and Electronics Engineers; Member 1984 IEEE Nuclear Science Symposium Committee; Member, IEEE Nuclear and Plasma Science Society Standing Committee on Radiological Instrumentation

Member, American Nuclear Society

Member, American National Standards Institute; Member, Group N42, Nuclear Instrumentation

Member, I&C Design and Drafting Standards Committee

Registered Professional Engineer

Registered Professional Land Surveyor

Patent Application Filed: *Portable Battery-Free Charger for Radiation Dosimeters*, January 1982

Patent Application Filed: *Integrated Charger for Self-Contained Dosimeters*, March 1982

K. P. Manoni

Member, Institute of Electrical and Electronics Engineers

C. D. Martin, Jr.

Member, Instrument Society of America

Member, Institute of Electrical and Electronics Engineers

Member, American Management Associtiom

H. L. Martin

Member, American Society of Mechanical Engineers

Associate Member, National Society of Professional Engineers

D. D. McCue

Member, American Society for Testing and Materials; Member, Committee C.26, Nuclear Fuel Cycle; Chairman, Subcommittee C26.10, Instrumentation

Invited paper: "An In-Line Multiwavelength Photometer for the Determination of Heavy Metal Concentrations," D. T. Bostick, D. D. McCue, J. E. Strain, M. L. Bauer, and D. M. Dixon, presented at *25th Conference on Analytical Chemistry in Nuclear Technology*, Gatlinburg, Tennessee, October 6-8, 1981

D. W. McDonald

Member, Institute of Electrical and Electronics Engineers

Julia M. McIntyre

Member, Institute of Electrical and Electronics Engineers

D. E. McMillan

Registered Professional Engineer

D. R. McNeilly

Member, Institute of Electrical and Electronics Engineers

Member, Eta Kappa Nu

Member, Tau Beta Pi

Patent Application Filed: *Transmission-Medium-Effects Control for Intrusion Systems,* D. R. McNeilly and W. R. Miller, January 1982

J. T. Mihalczo

Member, American Nuclear Society; Member, Engineering & Technology Accreditation/Registration and Professioanl Development Committee

Chairman, ORNL Burst Reactor Experiment Review Committee

Invited paper: "A Review of Subcriticality Measurements Using ^{252}Cf-Driven Power Spectral Density Measurements, presented at *1981 Winter Meeting of the American Nuclear Society,* San Francisco, California, December 3, 1981; *Trans. Am. Nucl. Soc.* **39**, 517-18 (1981)

D. R. Miller

Member, Instrument Society of America

Member, I&C Division Maintenance Information System Committee

Award for Significant Contribution to Success of Annual Cycle Energy System House (one of ten Outstanding Engineering Achievements of 1979), November 1980

Patent Application Filed: *Heat Pump COP Measurement Method and Apparatus,* V. R. Brantley and D. R. Miller, May 1982

G. N. Miller

Member, Institute of Electrical and Electronics Engineers

Member, Instrument Society of America

Member, National Society of Professional Engineers

Industrial Research 1982 I•R 100 Award, "ORNL In-Core Temperature, Density, and Level Measurement System," G. N. Miller, R. L. Anderson, and S. C. Rogers

Patent Application Filed: *Reactor Coolant Level Sensor,* R. L. Anderson and G. N. Miller, October 1981

R. L. Moore

Member, Instrument Society of America

Member, Institute of Electrical and Electronics Engineers

Member, ORNL Reactor Experiment Review Committee

A. C. Morris, Jr.

Registered Professional Engineer

C. A. Mossman

Member, Instrument Society of America

R. C. Muller

Member, Institute of Electrical and Electronics Engineers

F. R. Mynatt

Recipient, E. O. Lawrence Award, 1981

Fellow, American Nuclear Society

Chairman, Triennial Audit of ORNL Reactor Experiment Review and Reactor Operations Review Committees

Member, ORNL Computer Steering Committee

Member, ORNL GPE Committee

Member, UCC-ND High Speed Communication Committee

Member, Industry Degraded Core Rulemaking Steering Group

Invited paper: "Nuclear Reactor Safety Research Since Three Mile Island," *Science* **216**, 131-35 (April 9, 1982)

C. H. Nowlin

Member, Institute of Electrical and Electronics Engineers

Lester C. Oakes

Fellow, Institute of Electrical and Electronics Engineers; Secretary, Administrative Committee of the IEEE Nuclear and Plasma Society; Chairman, Awards Committee; Chairman, Reactor Instrumentation Committee

Member, American Nuclear Society

Member, Technical Evaluation Group, Instrumentation and Electrical, TMI-2

Member, National Engineering Simulator Study Group—DOE

Registered Professional Engineer

Pedro J. Otaduy

Member, American Nuclear Society

Member, American Association for Artificial Intelligence

Member, American Association for Computing Machinery

E. P. Wigner Fellowship, ORNL, 1980

A. R. Boynton Memorial Award, University of Florida, 1981

D. G. Prater

Member, American Society for Quality Control

Member, Institute of Certification of Engineering Technicians

Member, UCC-ND Maintenance Committee

J. A. Ray

Member, Institute of Electrical and Electronics Engineers

Amanda W. Renshaw

Affirmative Action Coordinator, Instrumentation & Controls Division

M.S. Thesis, "An Investigation of the Effect of the Uranyl Ion on Proton Spin Relaxation Times in Aqueous Solutions," The University of Tennessee, 1982

Charles W. Ricker

Member, American Physical Society

Member, Sigma Xi

M. J. Roberts

Senior Member, Institute of Electrical and Electronics Engineers

Member Tau Beta Pi

Member, Phi Kappa Phi

Special Assignment: U.S. Nuclear Regulatory Commission On-Site Representative to the Federal Republic of Germany on 2D/3D Programs, 1980–81

S. C. Rogers

Industrial Research & Development 1982 I•R 100 Award, "ORNL In-Core Temperature, Density, and Level Measurement System," G. N. Miller, R. L. Anderson, and S. C. Rogers

R. T. Roseberry, Sr.

Industrial Research & Development 1982 I•R 100 Award, "ORNL Inductively Coupled Plasma (ICP) Spectrometer," J. H. Stewart, Jr. (Analytical Chemistry Division) and R. T. Roseberry, Sr.

John A. Russell

Member, American Society for Testing and Materials; Member, Committee F-12, Security Systems and Equipment; Chairman, Task Group F- 12.40, Detection and Surveillance Systems and Devices

Member, Executive Committee of Carnahan Conference on Security Technology

Chairman, Nuclear Security Engineering Seminar

Member, UCC-ND Radio Advisory Committee

Member, National Society of Professional Engineers

Member, Tennessee Society of Professional Engineers

Registered Professional Engineer

Chairman, ORNL Accelerators and Radiation Sources Safety Review Committee

Member, I&C Division Design and Drafting Standards Committee

Invited paper: "Security Television Monitoring Using a Wideband Radio Frequency Cable System," J. A. Russell, A. L. Case, R. I. Crutcher, and F. E. Wetzel, *1981 Carnahan Conference On Crime Countermeasures*, Lexington, Kentucky, May 1981; *Proceedings*, pp. 19-21

Paul E. Satterlee, Jr.

Member, Eta Kappa Nu

Best Paper Award: "Electromechanical Three-Axis Development for Remote Handling in the Hot Experimental Facility," J. Garin, B. J. Bolfing, P. E. Satterlee, and S. M. Babcock, presented at *29th Remote Systems Technology Development Conference of the American Nuclear Society*, San Francisco, California, December 1981

Gerald K. Schulze

Member, Society for Information Display

Member, U.S. NIM-CAMAC Committee

R. L. Shepard

Senior Member, Instrument Society of America; Lecturer on Thermometry (Short Courses); Session Chairman and Editorial Sixth ISA Temperature symposium

Member, American Society for Testing and Materials; Chairman, Committee E-20, Thermometry; Chairman, Subcommittee E-20.11, Sheathed Heaters

Session Chairman, 1982 IEEE Nuclear Science Symposium; Member, Program Committee

Member, Sigma Xi

A. A. Shourbaji

Member, Instrument Society of America; Member, Education Committee; Member, Speakers' Directory Advisory Committee

W. H. Sides

Member, American Nuclear Society

Member, Sigma Xi

Member, ORNL Reactor Operations Review Committee

R. L. Simpson

Member, Digital Equipment Corp. Users' Society

Member, I&C Division Maintenance Information System Committee

David E. Smith

Member, American Society of Clinical Pathologists

O. L. Smith

Member, Sigma Xi

Robert S. Stone

Member, American Nuclear Society

Member, Society for Computer Simulation

J. E. Swift

Member, International Information/Word Processing Association

Member, American Business Women's Association

UCC-ND Career Planning Consultant

Chair, I&C Division Library Committee

R. M. Tate

Member, Institute of Electrical and Electronic Engineers

L. H. Thacker

Member, Phi Beta Kappa

Member, ORNL Graduate Fellow Selection Panel

H. A. Todd

Registered Professional Engineer

Member, Engineering Physics Division Safety Review Committee

J. H. Todd

Registered Professional Engineer

Chairman, ORNL Stores Stock Advisory Committee

R. A. Todd

U.S. Patent No. 4,292,539, *Compensated Count Rate Circuit for Portable GM Survey Meters,* September 1981

K. H. Valentine

Member, American Nuclear Society

Patent Application Filed: *Parallel Plate Fission Counter with Large Plate Area,* M. K. Kopp and K. H. Valentine, April 1981

Patent Application Filed: *Neutron Flux Profile Monitor for Use in Fission Reactor,* M. K. Kopp and K. H. Valentine, September 1981

K. W. West

Member, Three Mile Island Instrumentation and E. E. Survivability Planning Group

Chairman, I&C Division Design and Drafting Standards Committee

Member, I&C Division Maintenance Information System Committee

H. N. Wilson

Member, ORNL Reactor Operations Review Committee (HPRR)

W. D. Zuehsow

Member, Institute of Electrical and Electronic Engineers

PUBLICATIONS

Numbers shown in parentheses following titles refer to abstracts in this report. The respective abstracts note the affiliation of those authors who are not members of the Instrumentation and Controls Division.

D. C. Agouridis, "Noise Equivalent Circuit of Linear Passive Two-Ports with Applications to Transmission Lines," *IEEE Trans. Instrum. Meas.* **IM-31**(2), 119–24 (June 1982). (1.6.6)

Y. Aimthikul, "Computer-Generated Speech," Master's Thesis, The University of Tennessee, December 1981. (3.7.8)

J. L. Anderson, R. L. Anderson, and G. N. Miller, "Ex-Core Neutron Detectors for Reactor Vessel Level Measurement," NUREG/CR-2626 (ORNL/TM-8267), March 1982. (3.8.16)

R. L. Anderson, J. L. Anderson, and G. N. Miller, "Inadequate Core Cooling Instrumentation Using Heated Junction Thermocouples for Reactor Vessel Level Measurement," NUREG/CR-2627 (ORNL/TM-8268), March 1982. (3.8.14)

R. L. Anderson, K. R. Carr, T. G. Kollie, "Thermometry in the Multirod Burst Test Program," NUREG/CR-2470 (ORNL/TM-8024), March 1982. (3.8.1)

T. M. Anklam, D. F. Hunt, D. K. Felde, M. S. Thompson, A. G. Sutton, S. S. Gould, and C. R. Hyman, "ORNL Rod Bundle Heat Transfer Test Data, Volume 4. ORNL Small Break LOCA Heat Transfer Test Series II: Experimental Data Report," NUREG/CR-2525, Vol. 4 (ORNL/NUREG/TM-407/V4), June 1982. (3.5.5)

T. M. Anklam, D. F. Hunt, M. S. Thompson, D. K. Felde, A. G. Sutton, and S. S. Gould, "ORNL Rod Bundle Heat Transfer Test Data, Volume 1. ORNL Small Break LOCA Test Series I: Experimental Data Report," NUREG/CR-2525, Vol. 1 (ORNL/NUREG/TM-407/V1), April 1982. (3.5.2)

S. V. Asselin and C. B. Oh, "A Characterization of the Nuclear Power Plant Operator's Role During Emergencies," NUREG/CR-1772 (ORNL/Sub-80/13852/1), August 1980. (1.3.14)

G. F. Auchampaugh, S. Plattard, N. W. Hill, G. de Saussure, R. B. Perez, and J. A. Harvey, "K Components for the 1.4-, 1.6-, and 1.7-MeV Structures in the Fission of ^{232}Th$+n$," *Phys. Rev. C* **24**(2), 503 (August 1981). (2.10.12)

S. P. Baker, R. M. Ennis, Jr., and P. G. Herndon, "Development of a Wide Range Vortex Shedding Flowmeter for High Temperature Helium Gas," ORNL/TM-7794, July 1981. (3.5.9)

S. J. Ball and N. E. Clapp, Jr., "Initial Dynamic Simulation of an HTGR Sensible Energy Transport and Storage Plant," ORNL/TM-8226, May 1982. (1.1.1)

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M. Becker, R. C. Block, M. M. Danchak, R. R. Gay, D. R. Harris, and J. P. Tully, "A Systems Approach to Evaluation of Control Room Structure," BBH-81-1, Becker, Block and Harris, Inc., May 1981. (1.3.19)

N. C. Bradley, G. O. Allgood, and J. C. Moyers, "Potential Economic Benefits from Process Control of Coal Preparation Plants," ORNL/TM-5736, October 1981. (3.9.2)

G. S. Canright, C. H Brown, Jr., G. O. Allgood, and W. R. Hamel, "Dynamic Modeling and Control Analysis of Froth Flotation and Clean Coal Filtration as Applied to Coal Beneficiation," ORNL/TM15, November 1981. (3.9.5)

K. R. Carr, G. O. Allgood, R. L. Anderson, W. H. Andrews, N. C. Bradley, C. H. Brown, G. S. Canright, G. M. Caton, W. R. Hamel, J. T. Hutton, J. M. Jansen, Jr., L. N. Klatt, D. D. McCue, R. L. Moore, J. C. Moyers, L. E. Ottinger, E. R. Rohrer, G. V Seaver, D. S. Walia, and G. R. Wetherington, "State of the Art Assessment of Coal Preparation Plant Automation," ORNL-5699, February 1982. (3.9.1)

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J. B. Davidson and A. L. Case, "Rapid Electronic Autofluorography of Labeled Macromolecules on Two-Dimensional Gels," *Science* **215** (March 12, 1982). (1.7.7)

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E. M. Dougherty, "Transitions in the Role of the Operator," R-81-015, Technology for Energy Corp., June 30, 1981. (1.3.15)

E. M. Dougherty, "Cognitive Demands on the Reactor Operator (As Inferred from Emergency Operating Instructions)," R-81-014, Technology for Energy Corp., June 30, 1981. (1.3.16)

J. L. Dunlap and V. K. Paré, "Magnetohydrodynamic Instability with Neutral-Beam Heating in the ISX-B Tokamak," *Phys. Rev. Lett.* **48**(8) (February 22, 1982). (1.7.24)

Richard C. Extermann, George F. Auchampaugh, John D. Moses, Clayton E. Olsen, and Nat W. Hill, "A Cryogenic Ionization Chamber for High-Resolution Fission Cross-Section Measurements," *Nucl. Instrum. Methods* **189,** 477-84 (1981). (2.10.3)

P. R. Frey and R. A. Kisner, "A Survey of Methods for Improving Operator Acceptance of Computerized Aids," NUREG/CR-2586 (ORNL/TM-8236), April 1982. (1.3.18)

R. Gwin, R. R. Spencer, and R. W. Ingle, "Measurement of the Average Number of Prompt Neutrons Emitted per Fission of ^{233}U Relative to ^{252}Cf for the Energy Region 500 eV to 10 MeV and Below 0.3 eV," ORNL/TM-7988, November 1981. (2.10.19)

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J. E. Hardy and J. O. Hylton, "Electrical Impedance String Probes for Two-Phase Void and Velocity Measurements," ORNL/TM-8172 (NUREG/CR-2505), May 1982. (3.3.5)

J. E. Hardy, W. H. Leavell, H. Liebert, and J. A. Mullens, "Transient Testing of an In-Core Impedance Flow Sensor in a 9-Rod Heated Bundle," NUREG/CR-1909 (ORNL/NUREG/TM-389), February 1981. (1.2.16)

J. E. Hardy, G. N. Miller, S. C. Rogers, and W. L. Zabriskie, "Advanced Two-Phase Flow Instrumentation Program Quarterly Progress Report for July–September 1981," NUREG/CR-2204, Vol. 3 (ORNL/TM-8162), January 1982. (3.8.12)

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J. E. Hardy, K. G. Turnage, C. E. Davis, and R. L. Anderson, "Evaluation of Thermal Devices for Detecting In-Vessel Coolant Levels in PWRs," NUREG/CR-2673 (ORNL/TM-8306), August 1982. (3.8.17)

R. W. Hayes, "Computer Network Communications," ORNL/TM-8087, June 1982. (3.7.6)

R. W. Hayes, "Distributed Resource Management: Deadlocks," ORNL/TM-8086, June 1982. (3.7.7)

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R. C. Isler and V. K. Paré, "Influence of Neutral-Beam Injection on Impurity Transport in the ISX-B Tokamak," *Phys. Rev. Lett.* **47**(9) (August 31, 1981). (1.7.29)

W. L. Kelly, K. H. Valentine, and M. K. Kopp, "Ultrahigh-Sensitivity Fission Counter for Source-Range Neutron Flux Monitoring in the Clinch River Breeder Reactor," *CRBRP Technical Review,* 7–24 (Fall 1982). (1.4.1)

R. A. Kisner and G. F. Flanagan, "A Systems Approach to Defining Operator Roles" *IEEE Trans. Nucl. Sci.* **NS-28**(1) (February 1981). (1.3.9)

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M. K. Kopp, L. G. Christophorou, K. H. Valentine, and J. G. Carter, "New Gas Mixture Improves Performance of ^{3}He Neutron Counters," *Nucl. Instrum. Methods* **201**, 395–401 (1982). (1.4.6)

S. D. Kramer, M. K. Kopp, G. S. Hurst, T. A. Callcott, J. P. Young, E. T. Arakawa, M. G. Payne, and D. W. Beekman, "Applications of Resonance Ionization Spectroscopy to Ultralow-Level Counting and Mass Spectroscopy," *Radiocarbon* **22**(2), 428 (1980). (1.4.8)

R. C. Kryter, T. J. Burns, R. D. Cheverton, R. A. Hedrick, F. B. K. Kam, and C. W. Mayo, "Evaluation of Pressurized Thermal Shock," NUREG/CR-2083 (ORNL/TM-8072), October 1981. (1.7.11)

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J. March-Leuba, F. J. Sweeney, W. T. King, J. A. Renier, and R. T. Wood, "In-Core Flow Velocity Profiles During the First Fuel Cycle at Hatch-1—Inferred from Neutron Noise," EPRI Report NP-2083, Research Project 1754-2, October 1981. (1.2.13)

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J. O. Hylton and R. N. McGill, "Measurement of Velocity and Void Fraction in Steam-Water Mixtures with Electrical Impedance Probes." (3.3.6)

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M. L. Bauer, D. A. Bostick, and J. E. Strain, "Evaluation of Fiber Optics for In-Line Photometry in Hostile Environments." (3.4.2) (Invited)

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E. A. Lazarus, J. T. Mihalczo, and V. K. Paré, "High Beta Studies with Beam-Heated, Noncircular Plasmas in ISX-B," p. A-4. (1.7.28)

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N. E. Clapp, Jr., and C. M. Smith, "Demonstration of an On-Line Reactor Noise Surveillance System at a PWR." (1.2.2)

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J. T. Mihalczo, "A Review of Subcriticality Measurements Using ^{252}Cf-Driven Power Spectral Density Measurements," pp. 517–18. (1.7.16) (Invited)

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K. H. Valentine, M. K. Kopp, and G. C. Guerrant, "A Ten-Fold Increase in Fission Counter Sensitivity with Transmission Line Electrode Configuration," pp. 631–32. (1.4.3)

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R. L. Anderson and R. L. Ludwig, "Failure of Sheathed Thermocouples Due to Thermal Cycling." (3.8.3)

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C. A. Mossman, J. L. Horton, and R. L Anderson, "Testing of Thermocouples for Inhomogeneities. A Review of Theory, with Examples." (3.8.2)

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T. V. Blalock, J. L. Horton, and R. L. Shepard, "Johnson Noise Power Thermometer and its Application in Process Temperature Measurement" (1.6.5)

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Carroll, R. M., K. R. Carr, and R. L. Shepard, "Studies of Sheathed Thermocouple Construction and Installation in Thermowells to Obtain Faster Response." (1.6.1)

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J. M. McIntyre and G. N. Miller, "Data Acquisition at a Residential Photovoltaic System." (3.6.8)

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S. J. Ball, R. M. Harrington, and N. E. Clapp, Jr., "Safety and Licensing Analyses for the Fort St. Vrain HTGR." (1.1.2)

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Clapp, N. E. Jr., "Reducing the Size of a Database by Using Pattern Recognition Techniques." (1.2.8)

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Campbell, D. G., J. S. Arendt, R. E. Battle, and P. W. Baranowsky, "Relative Cost vs Reliability Improvement of Nuclear Power Plan Onsite ac Power." (1.3.3)

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G. S. Canright and R. L. Anderson, "Calorimeter for Measuring ac Magnetic Losses in Small Samples." (3.8.21)

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R. E. Battle, D. J. Campbell, and P. W. Baranowsky, "Reliability of the Emergency ac Power System at Nuclear Power Plants." (1.3.1)

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J. B. Davidson, "Television-Based Neutron Detectors and Applications." (1.7.9)

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R. M. Harrington and S. A. Hodge, "SCRAM Discharge Volume Break Studies. Part I: Accident Sequence Analysis." (1.1.5)

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K. H. Valentine, M. K. Kopp, J. A. Harter, G. C. Guerrant, W. T. Clay, G. W. Allin, and C. E. Fowler, "Ultrahigh-Sensitivity Fission Counter with Transmission Line Electrode Configuration." (1.4.2)

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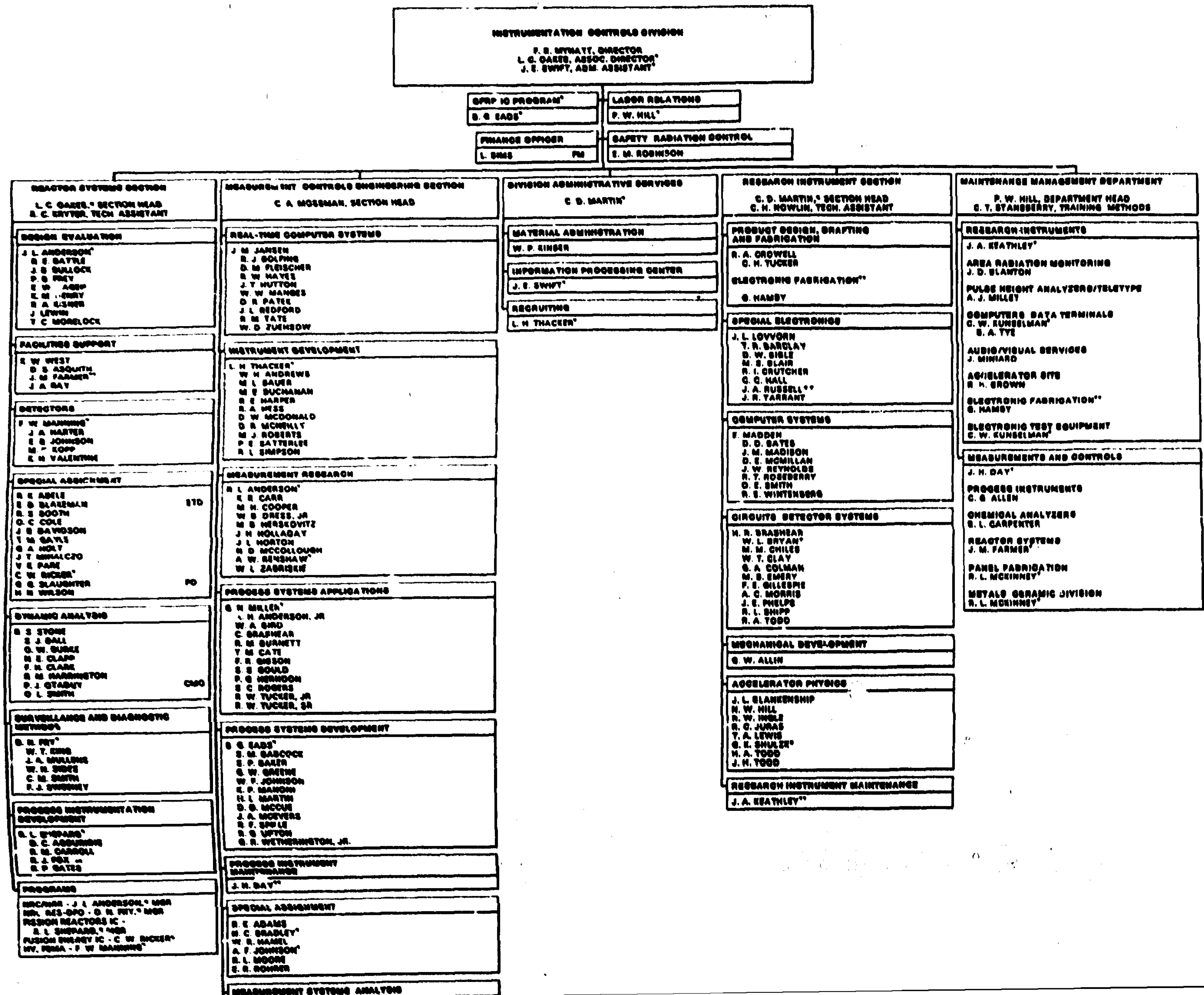
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