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Reactor Safety Research Programs

Quarterly Report
April - June 1981

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
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Prepared by
S. K. Edler, Ed.

Pacific Northwest Laboratory
P.O. Box 999
Richland, WA 99352

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ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 1981, for the Division of Reactor Safety Research within the U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL). These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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Core Thermal Model Development

GRAPHITE NONDESTRUCTIVE TESTING[a]

W. C. Morgan, Project Manager
T. J. Davis, Project Manager

M. T. Thomas

SUMMARY

During this quarter the reason for the deviations between measured and calculated eddy current data was identified and corrected by changes in the measurement system. The apparatus for ultrasonic backscattering measurements was assembled, and data are being obtained. Additional information on the elemental composition of impurities in three of the graphites, which were oxidized in previous investigations, has been obtained and is reported herein.

INTRODUCTION

This is a continuation of previous work at Pacific Northwest Laboratory (PNL) that demonstrated the feasibility of monitoring changes in the compressive strength of oxidized graphite by measuring changes in the velocity of an ultrasonic wave propagated through the graphite. The fiscal year (FY)-1981 scope of this project is to:

- develop eddy current techniques for near-surface profiling of oxidation in the Fort St. Vrain PGX core support blocks, including initiation of a prototype system development leading to interpretable indications of graphite strength
- complete assessment of ability to profile oxidation using ultrasonic surface waves; determine technical feasibility of using ultrasonic backscattering techniques to find in-depth oxidation profiles; continue evaluation of high-power ultrasonic tests and acoustic imaging techniques for bulk oxidation measurements as appropriate
- as funds permit, outline development of predictive techniques for oxidation depth profiles under reactor environments to provide strength indications for large graphite components.

The objective of this investigation is to demonstrate the feasibility of nondestructive testing (NDT) techniques for in-service monitoring of structural graphite strength to be applied initially to the Fort St. Vrain reactor.

TECHNICAL PROGRESS

EDDY CURRENT TESTING

During this quarter, we concentrated on improving the computer algorithm (ZFIT) for estimating near-surface graphite oxidation. The ZFIT algorithm uses electromagnetic theory and advanced fitting routines to estimate electrical conductivity versus depth in the material; data from three frequencies are used to deduce the oxidation profile versus depth (current test frequencies are 100 kHz, 1 MHz, and 4 MHz). The energy from each frequency penetrates to a different depth in the graphite, with the low frequency achieving the deepest penetration. The known relationships between electrical conductivity and oxidation in graphite are then used to compute the oxidation profile.

(a) RSR FIN Budget No.: B2101-1; RSR Contact: R. B. Foulds.

Earlier work had shown a discrepancy between measured and predicted eddy current data at the highest test frequencies, even with homogeneous specimens of known conductivity. During this quarter, the source of this discrepancy was found to be electrical resonance of the search coil because of stray capacitance. Coil resonance is generally not a problem in conventional eddy current testing because the systems are directly calibrated to known test samples; thus, the calibration curves include the resonance perturbations. In this case, however, the computer program has to deal directly with resistance and reactance of the search coil in order to employ the necessary electromagnetic theory. The circuit resonance introduced substantial measurement error of these search coil parameters at 4 MHz and some error at 1 MHz. This resonance was caused by stray capacitance in both the coaxial cable connecting the coil to the instrument and the capacitance of the coil (this coil was wound in the normal "layer-on-layer" fashion). Two changes have been made to reduce the stray capacitance:

- The cable capacitance was eliminated by placing a constant current driver in the probe for coil excitation; this results in an extra bonus in that the probe output is a direct function of coil impedance, in contrast to the small nonlinearity introduced by the former bridge circuit.
- The capacitance in the coil was reduced by using a different winding procedure specifically designed for low capacitance. The procedure produces banked windings in which two turns in the first layer are followed by two turns on the second layer. This cycle is repeated until a two-layer coil of the desired inductance is achieved. The result is a three-fold reduction in interturn capacitance over a standard down-and-back two-layer design.

The net result of these changes is that the system resonant frequency is increased from 8 to 20 MHz; this new resonant frequency has negligible effect on the measurements at 4 MHz.

ULTRASONIC TESTING

Work continued to determine the criteria for an ultrasonic backscatter test capable of measuring oxidation in depth. The backscatter approach relies on averaging a number of phase-insensitive waveforms that result from transmitting sound into the specimen and detecting returned energy. The averaging process can be used to measure grain size and detect small reflectors (such as porosity due to oxidation) that would otherwise be undetectable.

A laboratory system for backscatter analysis has been assembled, and analysis of the technique for graphite characterization has begun. The system has been interfaced to a minicomputer system that digitizes the waveforms and performs the averaging process. Video detection of the signals is performed in the computer by taking the Hilbert transform of the waveforms, which results in the needed phase-insensitive data. Estimates of grain size in PGX graphite have been made using the backscatter data, and metallurgical sectioning is being done to determine the degree of correlation. Results indicate that independent waveforms for averaging cannot be obtained by using different test frequencies; slight variations in transducer positioning will be required to obtain independent data. This is significant in that it is timely enough to affect the design of the prototype field system in a positive way.

OXIDATION

Ash residue from the complete in-air oxidation of specimens obtained from samples[a] III-A-10,[b] VI-A-3,[c] and II-F-3[d] was analyzed with x-ray photoelectron spectroscopy (XPS). Since the ash

(a) 1-in. thick, 30-in. diameter disks removed from previously oxidized samples.

(b) III-A-10 was a purified sample of Airco-Speer grade S1090.

(c) VI-A-3 was an "as-received" sample of Pure Carbon grade P3W.

(d) II-F-3 was a purified sample of Stackpole grade 2020.

content from each of these specimens was quite small, a special sample mounting arrangement was designed to allow XPS analysis. The analysis of the spectra is summarized in Table 1, which tabulates the elements detected and their concentration in atomic percent (at.%). Most of the elements are likely in a stable oxide component; the oxygen concentration ranged between 40 and 50 at.%. To remove any contamination from handling, about 30 Å of the surface of the ash was removed by argon ion sputtering. Carbon was also detected; however, some of the oxygen and carbon signal may have originated from the sample mount.

Some of this work was presented at the 15th Biennial Conference on Carbon in Philadelphia, Pennsylvania.(1)

Table 1. Elemental Composition of Ash Samples

Element[a]	Concentration, at.%		
	Sample III-A-10	Sample VI-A-3	Sample II-F-3
Nickel	3.3	4.1	1.8
Iron	1.5	1.8	0.6
Vanadium	8.1	11.0	11.6
Calcium	1.9	1.8	1.9
Sodium	0.3	0.2	0.3
Chlorine	2.0	--	1.0
Silicon	3.9	1.2	5.5
Lead	0.1	0.2	0.1
Potassium	--	1.9	3.0

(a) Carbon and oxygen concentration not reported.

FUTURE WORK

EDDY CURRENT TESTING

- complete testing of the profiling algorithm using large oxidized samples.

ULTRASONIC TESTING

- determine optimum transducer design
- increase test sensitivity
- optimize ultrasonic coupling and near-field criteria
- assess performance of the system on graphite with known oxidation.

OXIDATION

- determine chemical state of impurities during oxidation.

REFERENCES

1. Morgan, W. C., and M. T. Thomas. June 1981. "Catalytic Activity of Impurities in Graphites." In *Proceedings of 15th Biennial Conference on Carbon.* University of Pennsylvania, Pittsburgh, Pennsylvania, pp. 409-410.

ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS[a]

P. H. Hutton, Project Manager
R. J. Kurtz, Assistant Project Manager

J. R. Skorpik
D. K. Lemon
L. J. Busse

SUMMARY

A meeting held at Materialprufungsanstalt (MPA), Stuttgart, West Germany, on June 2, 1981, was effective in finalizing the ZB-1 vessel test plan by resolving several questions. Vessel fabrication is in progress on a schedule that should permit testing to start in February 1982. Wave-guide acoustic emission (AE) sensors and differential preamplifiers for use in the vessel test have been completed and calibrated.

AE sensor locations for monitoring the Watts Bar, Unit 1 reactor have been finalized, and the installation method has been established. A Letter of Agreement contract between the Tennessee Valley Authority (TVA) and Pacific Northwest Laboratory (PNL) covering the AE test effort is in progress.

An AE waveform recorder subsystem has been added to the AE monitor system for integrated testing. This microprocessor-based unit performs the function of recording acoustic waveforms during vessel testing. These waveforms will be used for onsite pattern recognition and for post-test analysis to refine the pattern recognition relationships.

Two unirradiated baseline fracture specimen tests have been monitored for AE at the Naval Research Laboratory (NRL), which is part of a continuing effort to characterize AE from irradiated fracture specimens.

INTRODUCTION

The purpose of this Pacific Northwest Laboratory (PNL) program is to provide an experimental evaluation of the feasibility of detecting and analyzing flaw growth in reactor pressure boundaries on a continuous basis using AE. Type A533B, Class 1 steel is being used in all experimental testing. Objectives of this program are to:

- develop a method to identify crack growth AE signals in the presence of other acoustic signals
- develop a relationship to estimate flaw significance from AE data
- develop an instrument system to implement these techniques
- demonstrate the total concept off-reactor and on-reactor.

Progress relative to these objectives is discussed in the following sections on off-reactor vessel test, reactor installation, AE monitor system development, irradiated fracture specimen tests, and reports. The final section describes work planned for the next quarter.

(a) RSR FIN Budget No.: B2088; RSR Contact: J. Muscara.

TECHNICAL PROGRESS

OFF-REACTOR VESSEL TEST

A meeting held at MPA, Stuttgart, West Germany, on June 2 finalized the ZB-1 vessel test plan. Several questions were satisfactorily resolved as follows:

- test location - Approximately the first half of the test will be performed at MPA, Stuttgart, where superior test facilities are available but the test temperature is limited to 200°F. The second half of the test will be performed at Mannheim where test facilities are less desirable but the test temperature can be extended to 550°F.
- slag inclusion - A slag inclusion will be provided in the installation weld around the A533B vessel wall insert.
- thermal shock - Stainless steel cladding with under clad cracking and poor bonding will be installed on the inside of a 750-mm diameter repair plug. The area will not, however, be exposed to thermal shock.
- hydrotest - A hydrotest sequence interspersed with fatigue loading will be performed to pressure levels ranging from 1.0 to 1.4 times maximum fatigue loading.
- hydraulic noise simulation - Hydraulic background noise will be simulated using a transducer mounted on the inside of the A533B insert and driven by a shaped electronic power source.
- schedule - Vessel fabrication has started with completion expected by January 1, 1982. AE system installation would start about February 1, 1982, with testing to start in mid-February.
- AE systems - There will be one AE monitor system in addition to the PNL system. The second system will be a combination effort by IZFP,[a] Battelle Frankfurt, and KVU.[b]

Fabrication of the A533B insert for the ZB-1 vessel has been completed by MPA up to the final step of fatigue precracking the machined flaws. This was held in abeyance until the June 2 meeting to review methodology. The final configuration of the ZB-1 test vessel is shown in Figure 1, which shows the vessel barrel unrolled to give the relative location of all components.

Thirty-five wave-guide AE sensors have been fabricated for use on the ZB-1 vessel test. They have been calibrated using a helium jet excitation to assure proper fabrication. The sensors have been assigned serial numbers, and a fabrication and calibration record book has been established to document the fabrication history of each sensor and to record its calibration data. One of the sensors has been subjected to temperature tests at temperatures as high as 225°F; this has had no adverse effect on its performance. The differential preamps for use with the wave-guide sensors are nearly completed. These are laid out on a printed circuit board and fabrication is in progress.

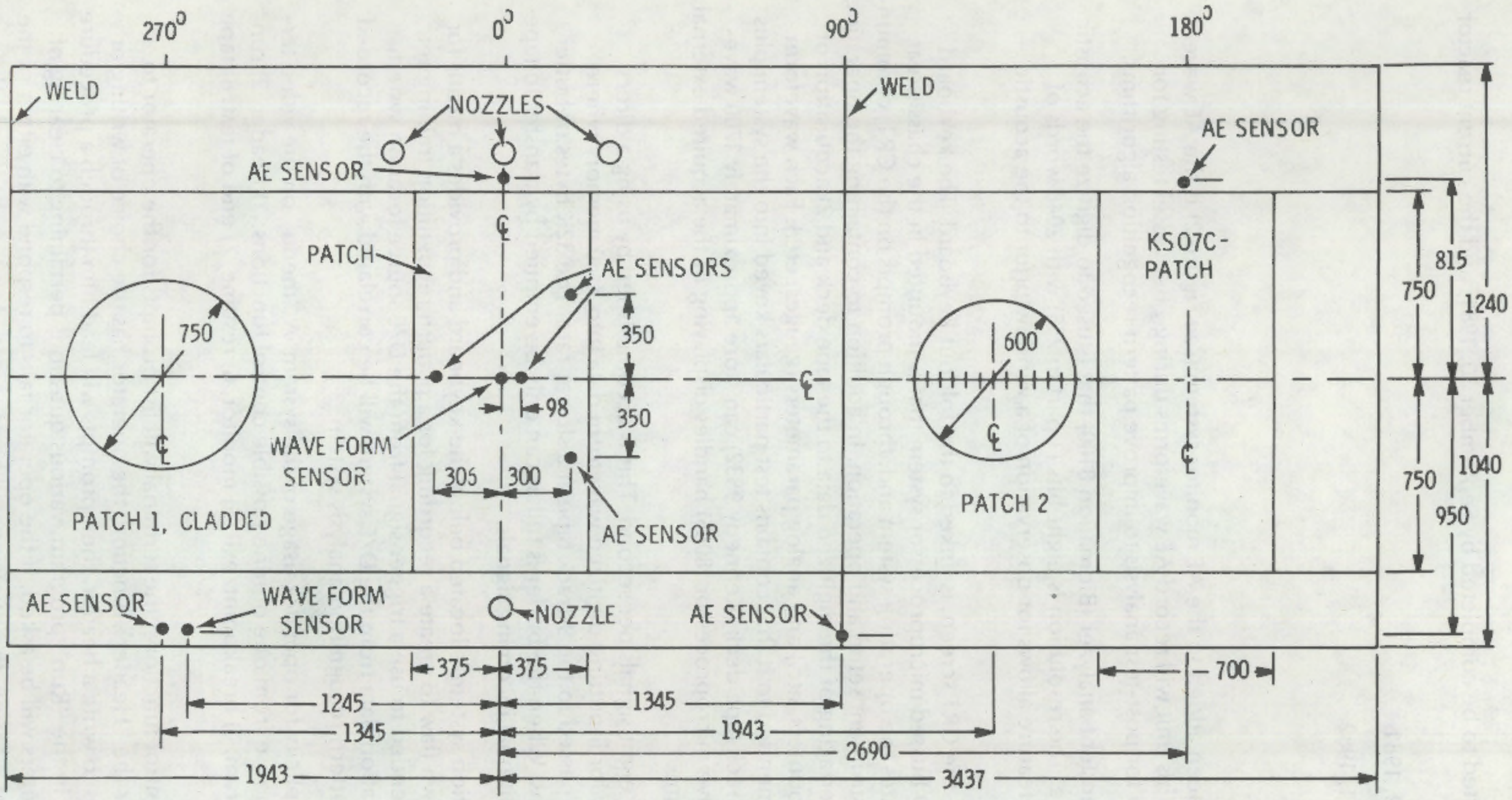
REACTOR MONITORING

Sensor locations for AE system installation on the Watts Bar, Unit 1 reactor have been finalized. A total of 24 sensors will be installed to monitor the No. 2 inlet nozzle, the safety injection pipe adjacent to the No. 2 cold leg, and a 90° segment of the vessel wall. This includes basic detection sensors, waveform recording sensors, and guard sensors where needed.

A written description of the test plan, purpose, and benefits was submitted to TVA on June 1 to provide a basis for generating a "Letter of Agreement" contract between TVA and PNL.

(a) IZFP: Institut fur Zerstorungsfreie Prufuerfaher, Saarbrucken, West Germany.

(b) Kraftwerk Union, Erlengan, West Germany.



NOTE:

1) DIMENSIONS IN MILLIMETERS
2) AN ULTRASONIC NOISE SIMULATION TRANSDUCER WILL BE MOUNTED ON THE INSIDE CENTER OF THE BATTELLE PATCH
3) ALL BATTELLE AE SENSORS WILL BE METAL WAVE GUIDES
4) BATTELLE AE SENSORS ON THE BATTELLE PATCH WILL BE MOUNTED IN DRILLED AND TAPPED HOLES. ALL OTHER BATTELLE SENSORS WILL BE PRESSURE COUPLED USING MAGNETIC MOUNTS

Figure 1. ZB-1 Vessel Test Configuration - Unrolled View of Vessel Barrel

AE system installation is expected to be completed by September 30, 1981; and the current reactor schedule is as follows:

- cold hydro test - October 9, 1981
- hot functional - February 11, 1982
- fuel load - June 1, 1982.

AE MONITOR SYSTEM

The latest subsystem that has been added to the AE monitor system (see Figure 2) is the AE waveform recorder (see Figure 3). This unit will record AE waveforms during ZB-1 vessel testing for onsite pattern recognition and for post-test analysis to improve pattern recognition algorithms.

The top unit in Figure 3 is a transient analyzer (Biomation 8100) that is used to digitize the acoustic signal at a 2-MHz sampling rate. The resolution is eight bits (1 part in 255) with 2048 words of memory storage. A pretrigger feature allows for observation of a window prior to the acoustic event.

The second item from the top, the CRT screen, is linked to the foldout keyboard. The keyboard communicates with a Z-80 STD bussed microprocessor system that is mounted in the chassis rear. The microprocessor incorporates the operating system and, through prompts on the CRT, communicates with the operator for instrument setup and operation. In addition to containing the operating system, the Z-80 handles 1) interfacing of the digitized data to the tape deck and 2) acquisition of various external digital data (counters, etc.) and analog parameters (gauges, etc.). Each waveform written to tape has a header amended to it that contains test particulars keyed into the system plus acquired external data. The 9-track tape deck (Kennedy 9832) can store approximately 1700 waveforms, including headers. A slave microprocessor (8085) handles displaying of the acquired external data on the CRT in near real time.

Waveform acquisition speed is eight signals per second. This is accomplished by using a direct memory access (DMA) process for inputting digitized waveform data into Z-80 memory where waveforms are stacked up and passed to the 9-track tape at a slower rate. The 32K bytes of buffer memory can store 16 waveforms. When the buffer is full, it can only be emptied by transfer to tape and not by flushing due to additional incoming signals.

A three-channel AE zone isolation system is located below the keyboard and provides a means for focusing on signals from a known flaw to create a recording for a pattern recognition training set. Normal system operation is intended to use a trigger signal from the D/E source location system. Through a tagging scheme, location data from the D/E system will be correlated with the recorded waveforms processed by the pattern recognition analysis system.

System design emphasized simplicity for operator usage of the system. A "menu" on the video terminal allows the operator to choose from one of nine possible control functions: 1) header, 2) run program, 3) test/calibrate program, 4) breakpoint/exit, 5) monitor, 6) resume, 7) end of test, 8) tape read, and 9) menu.

After choosing one of these control functions, the terminal will list prompts for the operator to respond to. For example, under the "Header" function, the operator has the choice of writing or listing the header. If he chooses to write a header, then prompts will lead him through a procedure for this with means for editing. In the "Run" program, various questions pertaining to clearing of the total, valid, clock, etc., counters will be asked. If the operator fails to respond with either of the listed choices for clearing or not clearing, the question will be repeated. At no point should the system ever fail to respond back to the operator due to an incorrect operator entry.

Figure 2. AE Monitor/Analysis System with 26-in. Pipe Test Specimen

Figure 3. AE Waveform Recorder Subsystem

The program or function used for acquiring and recording test data is the "Run" mode. "Test/Calibrate" is used for setup and does not involve any recording onto tape; its function is to duplicate the data acquisition part of the "Run" program. The "Breakpoint" function is used to exit from a program; "Resume," in turn, is used to return to the program. "Monitor" is a program used to acquire data (such as total, valid counts) when the material crack is being extended. In the "Monitor" mode, no waveforms are recorded and no data are stored onto tape. The "End of Test" function closes out the data file and rewinds the tape. The "Tape Read" is a method for listing onto the terminal any of the recorded waveforms and their associated headers. No graphics capability exists so the waveform is listed out in hexidecimal rows and columns. "Menu" will list the functions available and their corresponding terminal input control letter.

Immediately upon power up, the system is made to undergo an internal diagnostic check of its RAM memory and it EPROM program memory. The RAM check is performed by verification of writing and reading all low states and then all high states in an alternating sequence. The EPROM check is by calculating a check sum and then comparing it to a self-stored check sum. This will account for bit dropouts or add-ins due to service time. The system also has two debug programs (i.e., Z-80 and 8085 processors) for examination of their internal contents.

IRRADIATED FRACTURE SPECIMENS

Two unirradiated baseline fracture specimens have been AE-monitored during fracture testing at NRL. Preliminary results are shown in Figure 4. Detected AE from these specimens showed an approximately linear relation to crack growth up to load cycle 45. Beyond that point, the AE diminishes while crack growth per load cycle increases. Test data will be analyzed to determine the cause of the transition. The irradiated companions to these specimens will be monitored to complete the minimum effort planned.

REPORTS

- Quarterly Progress Report for the period January 1 to March 31, 1981
- Program Summary for Advisory Committee on Reactor Safeguards (ACRS) review - June 1981.

FUTURE WORK

Plans for the period from July 1 to September 30, 1981, are:

- complete installation of in-containment AE system components at Watts Bar, Unit 1
- revise ZB-1 vessel test plan and obtain approvals
- continue irradiated fracture specimen AE measurement and analysis
- perform AE characterization tests on cast stainless steel piping material
- publish initial evaluation report on AE weld monitoring demonstration results.

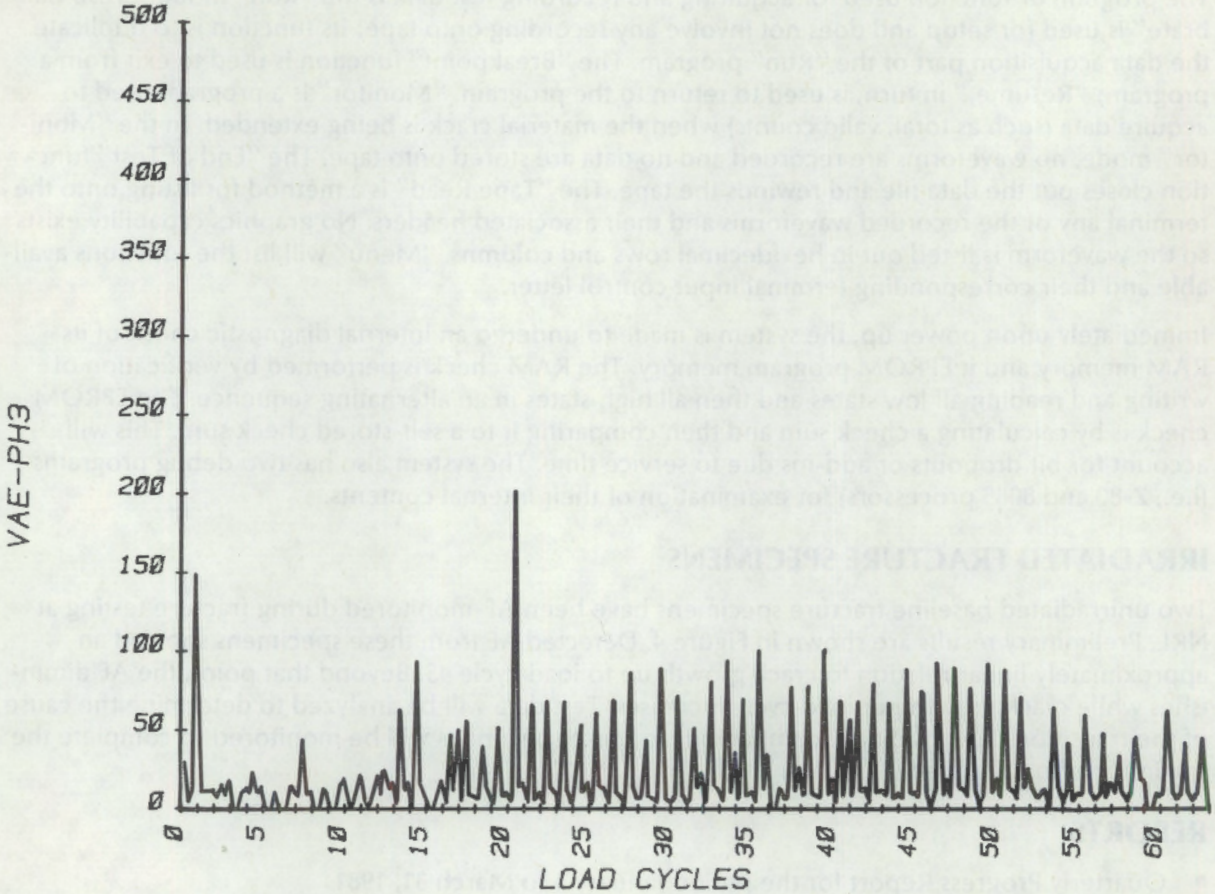


Figure 4. Acoustic Emission Versus Load Cycles for Baseline Fracture Specimen 65W22

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[(a)]

F. L. Becker, Program Manager
S. H. Bush, Project Manager
F. A. Simonen, Project Manager

S. R. Doctor
G. B. Dudder
P. G. Heasler
G. P. Selby

SUMMARY

Activities during the past quarter have been devoted to the piping round robin program. Preparations (sample fabrication, procedure development, and instrument evaluation) were completed in April. The first two of the planned six teams have completed the test matrix on schedule; the third team is scheduled to start on July 13 and finish by July 31. Two pressure vessel activities are in progress in addition to the round robin. Preparations are in progress for a program to determine the reliability of near-surface crack detection (under clad cracks) in reactor pressure vessels. Negotiations are currently in progress with the United Kingdom Atomic Energy Authority (UKAEA) Risley Laboratory to exchange information concerning the reliability of in-service inspection (ISI).

INTRODUCTION

The primary pressure boundaries (pressure vessels and piping) of nuclear power plants are inspected in-service according to the rules of the ASME Boiler and Pressure Vessel Code, Section XI (Rules for In-Service Inspection of Nuclear Power Plant Components). Ultrasonic techniques are normally used for these inspections, which are periodically performed on a sampling of welds.

The Integration of Nondestructive Examination (NDE) Reliability and Fracture Mechanics Program at Pacific Northwest Laboratory (PNL) has been established to determine the reliability of current ISI techniques and to develop recommendations that will assure a suitably high inspection reliability. The objectives of this program are to:

- determine the reliability of ultrasonic ISI performed on commercial light water reactor (LWR) primary systems
- using fracture mechanics analysis, determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to assure a suitably low failure probability
- evaluate the degree of reliability improvement that could be achieved using improved and advanced NDE techniques
- based on material, service, and NDE uncertainties, formulate recommended revisions to ASME Code, Section XI, and Regulatory Requirements needed to assure suitably low failure probabilities.

The scope of this program is limited to ISI of primary systems, and the results and recommendations are also applicable to Class II piping systems.

(a) RSR FIN Budget No.: B2289-0; RSR Contact: J. Muscara.

TECHNICAL PROGRESS

The progress and accomplishments of the past quarter are described below by task.

ROUND ROBIN

The piping round robin is currently in progress, and two of the proposed six teams have completed the test matrix. The test matrix involves 254 tests or measurements on 96 samples of 10-in. Schedule 80 stainless steel; 32.25-in. outside diameter (OD), 2.375-in. wall A106 carbon steel pipe; and 27.5-in. inside diameter (ID), 2.375-in. wall centrifugally cast stainless steel pipe. Each sample contains a field-type weld, and the tests are conducted under laboratory and difficult access conditions. The teams use their own procedures as well as an improved procedure.

Preparations for the round robin tests were completed in April. The samples were characterized to assure they met the flaw size requirements of the test matrix. The preparations also included drafting of an improved procedure and the instrument and search unit evaluation system and procedures.

Each of the first two teams completed the test matrix on schedule (three weeks), which involved 10- and 12-h days for most of the working days. The teams were able to follow and comply with the test protocol; however, the test did require a high degree of supervision to avoid misunderstandings and to assure that all of the required data were reported and collected. A high degree of cooperation was received from the teams.

The results of the inspections and the identity of the teams will not be reported until the completion of the round robin to preserve the confidentiality of the test matrix and to give each team an equal opportunity. However, the results are being continually reviewed to assure that any problems are corrected promptly.

During the test, the time, amplitude, and position were recorded on a four-channel strip chart recorder. The time and amplitude information is obtained from the inspection instrument while the position data are obtained using the search unit tracking and recording system (SUTARS). This method was found to be quite suitable for monitoring test performance and analysis of data. (The test teams do not have access to the recorded data.) The strip chart along with the calibration record, data sheet, and analysis sheet are filed for each inspection.

The results from the analysis sheet are entered into a computer data base that also contains the true state for each sample. The computer analysis program compares the reported and true state values and reports detection or nondetection.

Figure 1 shows an inspection team performing a test on a 10-in. diameter pipe in the difficult access position. The man on the far right is the PNL observer, who monitors the strip chart and time clock (30 min are allowed for each test). The inspector (on the floor) is the Level II who is performing the test. The Level I to his left records the data, and the Level III at the far left analyzes the data and makes the final decisions on the data. In the laboratory condition, the sample and inspection instrument are placed on the bench for easy access.

Only a small portion (location only) of the SUTARS capability has been used to date. The next two teams to be tested use the Sonic Mark I, which is incorporated in SUTARS. These teams will use the full capability of the SUTARS system, which was recently updated to the latest system modification and is now fully operational.

PRESSURE VESSEL APPLICATIONS

Two separate pressure vessel activities are currently in progress: 1) a program to determine the effectiveness of near-surface crack detection (under clad cracks) and 2) arrangements for a cooperative exchange agreement with the UKAEA Risley Laboratory.

Figure 1. Inspection Team Performing an Inspection on a 10-in. Diameter Pipe in the Difficult Access Position

The near-surface crack detection program was initiated to determine the effectiveness of detection techniques. A pressurizer nozzle drop out has been received from Babcock and Wilcox, and cracks will be grown through the clad surface by the thermal fatigue technique. The Pressure Vessel Research Committee, Nondestructive Examination Subcommittee has agreed to include the fabricated sample in the upcoming PISC II round robin test, which has been delayed until the summer of 1982. To obtain information before that time, a separate round robin is being considered. The test sample is scheduled to be completed by October 1981. Upon completion of flaw growth, the block will be evaluated by the most suitable, available techniques; and the results will be reported. This test will allow ISI organizations to demonstrate the effectiveness of their near-surface inspection techniques.

The UKAEA Risley Laboratory is conducting investigations concerning the reliability of ISI of pressure vessels. Two plates containing fabrication and in-service-type flaws, including near-surface cracks, are included in the program. The plates will be inspected by five teams practicing their best available techniques; one team will use an improved ASME technique. The inspections are expected to be complete by November 1981, with preliminary analysis by January 1982 and final analysis with destructive analysis by May 1982.

Discussions are in progress with UKAEA representatives to exchange information on the results of this study and our program on inspection reliability for primary piping systems.

FUTURE WORK

The major objectives of the coming quarter are completion of the round robin for three more teams, fabrication of near-surface cracks in the test plate, and finalization of the cooperative agreement with the UKAEA.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD CODES:

TASK A - IRRADIATION EXPERIMENTS[a]

D. D. Lanning, Program Manager
D. D. Lanning, Task Leader

M. E. Cunningham
R. E. Williford

SUMMARY

This task is concerned with the irradiation of instrumented fuel assemblies (IFAs) for the U.S. Nuclear Regulatory Commission (NRC) at Halden, Norway. The purpose of these tests is to obtain reliable independent data on fuel thermal and mechanical behavior for development of fuel rod modeling computer codes.

Irradiation test IFA-431 is completed. Two other tests (IFA-513 and IFA-527) were removed from the reactor in early April due to fuel failures. IFA-432 was removed from the reactor in early June to permit cooling for late fall shipment to Harwell, U.K.

Anomalous behavior of pressure and temperature in IFA-513 lead Halden and Pacific Northwest Laboratory (PNL) to suspect that slight water inleakage may have occurred in all the rods some time ago. It was recommended during a meeting at Halden (June 22, 1981) that one IFA-513 rod accompany the IFA-527 and IFA-432 rods to Harwell, U.K., for postirradiation examination (PIE) to confirm or deny the presence of water in the rod.

The compliance tests of rods 7 and 8 of IFA-432 at Harwell were further delayed by difficulties in accommodating the test apparatus in the cell. These difficulties are now solved, and the tests are to be conducted in July.

INTRODUCTION

The objectives of the Experimental Support and Development of Single-Rod Fuel Codes Program at PNL are now fourfold:

- collect and analyze in-reactor data on fuel rod thermal/mechanical behavior, especially as a function of burnup
- correlate in-reactor data with postirradiation data and with ex-reactor tests on mechanical and thermal parameters of fuel rods
- integrate the above information into the FRAPCON and FRAP-T series of computer codes
- study the occurrence and mechanisms of fuel cladding failure using controlled experiments with centrally heated simulated fuel pins in a PNL pressurized water loop.

The Halden Boiling Water Reactor (HBWR) in Norway is currently the sole site used by this program for irradiation tests. PIE will be carried out at both Kjeller, Norway, and Harwell, U.K. Task A of the program is concerned with the conduct of the tests and coordination of test design, test fabrication,

(a) RSR FIN Budget No.: B2043; RSR Contact: H. H. Scott.

shipping, PIE, and sample disposal. The test matrix now spans the full range of expected BWR conditions for pelletized UO_2 fuel, including:

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50 to 380 μm (0.002 to 0.015 in.)
- gas compositions ranging from pure helium to pure xenon
- fuel densities of 95% and 92% theoretical density (TD), the latter both stable and unstable regarding in-reactor densification.

IFA-527 is specifically designed to study the progress and variability of fuel cracking and relocation and features xenon-filled rods to magnify thermal effects.

TECHNICAL PROGRESS

On April 5, 1981, the HBWR was restarted after a long shutdown. Pressures in rod 3 of IFA-527, which had been in the 0.2- to 0.3-MPa range, immediately increased to 0.8 to 0.9 MPa. Rod 3 centerline temperatures, which at full power had been 500 to 600K above those in (failed) rods 1 and 2, were seen to be approximately equal to those in rods 1 and 2. Both of these measurements indicated that rod 3 had failed before startup. Meanwhile, rod 6 of IFA-513[a] failed during startup. A predetermined power ramp for IFA-513 was completed; then the reactor was shut down and IFA-513 and IFA-527 were removed due to high coolant activity. It was later decided that these assemblies would not be restarted.

During a brief shutdown in early June, IFA-432 was removed from the reactor to begin cooling before shipment to Harwell in early November for PIE.

The main features of the PIE program for IFA-432 and -527 were decided during meetings between Halden, Harwell, and PNL staff in June. Tables 1 and 2 summarize these features.

Table 1. Postirradiation Examination of Instrumented Fuel Assembly (IFA)-432

	Type of Examination							
Rod Number	Visual	Axial Gamma Scanning	Profilometry	Rod Puncture and Gas Analysis	Optical Metallography	Bulk Fuel Density	Cladding ID	Bulk Burnup
1	X	X	X	X	2	1	1	
2	X		X	X	1			
3	X		X	X	1		1	
4	X				1			
5	X			X	1			1
6	X	X		X	1	1		
9	X			X	2			
Total	7	2	3	6	9	2	2	1

(a) IFA-513 is jointly sponsored by the NRC and the Halden project.

Table 2. Postirradiation Examination of Instrumented Fuel Assembly (IFA)-527

Rod Number	Type of Examination: Visual	Gamma Scanning	Neutron Radiography	Rod Puncture and Pressurize	Optical Metallography
1	X		X		
2	X			X	
3	X		X		
4	X		X	X	X
IFA-513	X		X	X	X
6	X	X		X	
Total	6	1	4	4	2 (6-8 cuts)

FUTURE WORK

Data from IFA-527 and IFA-432 up through the end of life should be received and will be compiled in preparation for an overall report on the series to be completed next year. Data from the compliance tests at Harwell will be analyzed and compared to short-rod tests completed at PNL.

Contractual arrangements for the shipping and PIE of IFA-432 and IFA-527 should be completed.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK B - DATA QUALIFICATION AND ANALYSIS[(a)]

D. D. Lanning, Program Manager
M. E. Cunningham, Task Leader

E. R. Bradley
W. N. Rausch
R. E. Williford

SUMMARY

A major objective of the Experimental Support and Development of Single-Rod Fuel Codes Program is the irradiation of instrumented fuel assemblies (IFAs) to obtain well-characterized data. Task B of this program is responsible for qualifying and analyzing those data. During this quarter fuel rod internal gas pressure data for the months of October and December 1980 and January, May, and June 1981 were received for analysis. Two reports on IFA-527 were completed and forwarded to the U.S. Nuclear Regulatory Commission (NRC) for publication.

INTRODUCTION

The Experimental Support and Development of Single-Rod Fuel Codes Program is a continuation of the Experimental Support and Verification of Steady-State Codes Program (begun in 1974) and is conducted by Pacific Northwest Laboratory (PNL). This program now has the general objectives of collecting and analyzing in-reactor data on fuel rod temperatures, fission gas release, and cladding elongation as a function of irradiation history; correlating postirradiation examination (PIE) with in-reactor data; utilizing ex-reactor testing for a better understanding of fuel rod mechanical behavior; and integrating this information into the FRAPCON computer code series. The qualification and analysis of the data obtained from in-reactor testing of fuel rods is the responsibility of Task B, which has been divided into three subtasks:

- **Subtask B-1 - Data Processing:** This subtask is responsible for receiving, correcting, characterizing, and presenting the data obtained from the fuel assemblies.
- **Subtask B-2 - Data Reports:** This subtask is responsible for preparing reports on the precharacterization of the fuel assemblies, the data obtained from the assemblies, and the postirradiation analysis of the assemblies.
- **Subtask B-3 - Data Analysis:** This subtask is responsible for providing in-depth analysis of the in-reactor fuel rod data. Specific areas of interest for fiscal year (FY)-1981 are analysis of data for inferring fuel relocation and its effect, use of transient temperature data to better understand fuel behavior, analysis of statistical variations and error propagation, and analysis of fuel rod fill gas pressure data for inferring fission gas release.

TECHNICAL PROGRESS

This quarter's activities are discussed below by subtask.

(a) RSR FIN Budget No.: B2043; RSR Contact: H. H. Scott.

SUBTASK B-1 - DATA PROCESSING

During the last quarter, manually collected fuel rod internal gas pressure data from rods 1, 5, and 6 of IFA-432 were received from Halden. These data were for the months of October and December 1980 and January, May, and June 1981.

SUBTASK B-2 - DATA REPORTS

Efforts this quarter centered on completing two reports on IFA-527; both reports were forwarded to NRC for publication and distribution. The first report, "Precharacterization Report for Instrumented Fuel Assembly (IFA)-527," NUREG/CR-2168, discusses the objectives and design of IFA-527. The second report, "Beginning-of-Life Data Report for the Instrumented Fuel Assembly (IFA)-527," NUREG/CR-2167, discusses two periods of operation: normal behavior from July 1 to August 15, 1980; and behavior with failed fuel rods during September 1980. The presence of steam in the failed fuel rods was observed to reduce fuel temperatures relative to the initial xenon fill gas. Both reports were sent to NRC for printing and distributing.

SUBTASK B-3 - DATA ANALYSIS

During this quarter, analysis continued in the areas of fuel cracking and relocation, error and uncertainty analysis, and the general behavior of fuel rods.

Fuel Cracking and Relocation Analysis

A journal article on fuel cracking and relocation was accepted for publication by *Nuclear Technology.* The article is principally based on the work reported in Reference 1.

Error and Uncertainty Analysis

Fuel rods that are identically designed and operated will show some variability in behavior due to the effect of manufacturing tolerances and localized differences in power. Examination of data obtained from replicate rods can be used to quantify the level of variability to be expected from operating rods, provide a comparison to uncertainty estimates, and validate and verify computer codes.

Two sets of replicate fuel rods have been included in the four instrumented test assemblies used in this program. First, five of the xenon-filled rods in IFA-527[a] are not only nominally identical but have also seen the same power history. Second, the standard rods included in each of IFA-431, -432, and -513 are nominally identical but were subjected to different peak powers and operating cycles.

The replicate xenon-filled fuel rods in IFA-527 are being used to study the statistical nature of fuel cracking and relocation. Throughout the first period of irradiation, the observed centerline temperature versus power behavior was highly consistent as shown in Figure 1, where measured centerline temperature versus power from an early-in-life power ascension is presented. The standard deviation of the measured centerline temperature, which ranges from 14 to 36K, was observed to increase as burnup progressed.

Linear propagation of errors has been used to evaluate the uncertainties in computer modeling of fuel rods.[2] By restraining the calculation of temperature to match the data and by assuming uncertainties for dimensions, thermal conductivity, and power of 0.2%, 15%, and 10%, respectively (3σ confidence level), the resulting uncertainty calculations may be interpreted as the expected data variability that may be observed. Although the width of the real data band is fairly constant, the calculated uncertainty band widens with power, which indicates an expected greater variance in the data with power than was actually observed.

(a) The sixth rod is also xenon-filled but has a different fuel-cladding gap size.

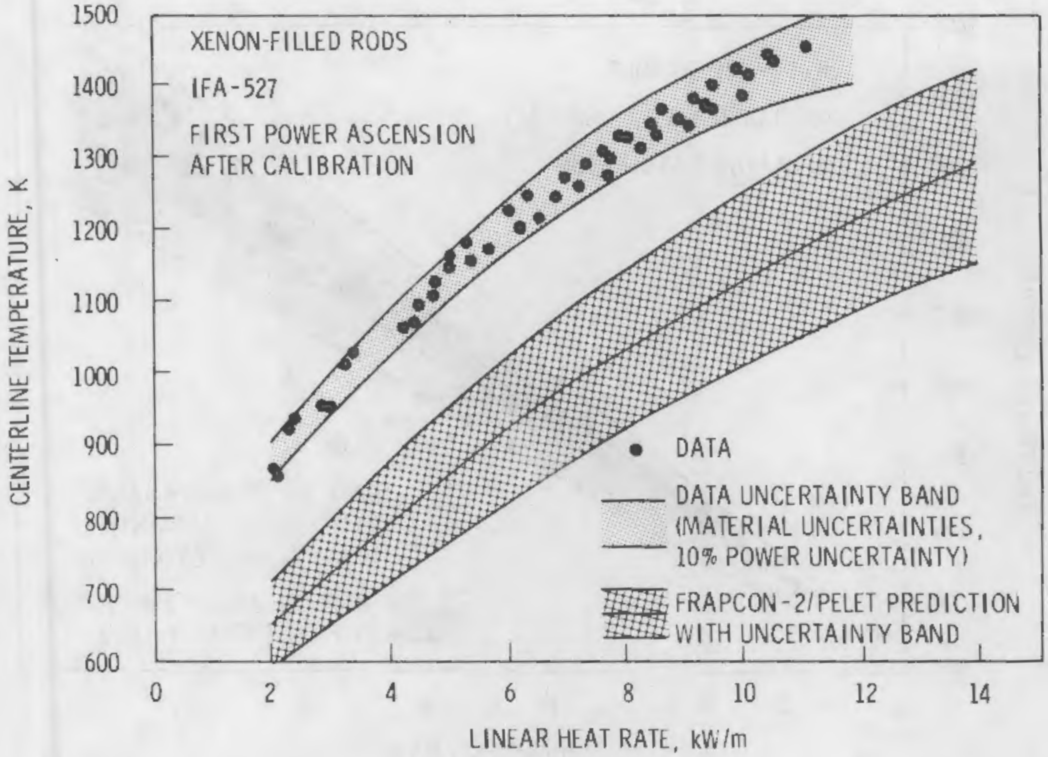


Figure 1. Comparison of Centerline Temperature Data, Code Calculation, and Uncertainty for Xenon-Filled Replicate Fuel Rods

Figure 1 also shows a FRAPCON-2[3] code calculation using the PELET option. PELET significantly undercalculated centerline temperatures: There was no overlapping of the uncertainty bands. One reason for the data/calculation mismatch for this case may be that IFA-527 powers were too low to have caused fuel relocation equal to that in the higher power rods upon which the PELET correlations are based.

Observed centerline temperature versus power, data uncertainty, and a FRAPCON-2/PELET comparison for the replicate helium-filled fuel rods from IFA-431, -432, and -513 are presented in Figure 2. As with the xenon-filled rods, a fairly narrow data band is observed, which indicates greater dependence upon rod design than operating history. The calculated expected uncertainty in the data is greater than the observed variability in the data. The FRAPCON-2 calculation lies along the lower data bound, and there is a substantial amount of overlap between the data and FRAPCON-2 bands. This is also indicative of a good match between the data and the code for these conditions.

In summary, a high degree of consistency has been observed between rods of identical design, the expected uncertainty in centerline temperature is greater than the observed variability, and specific comparisons between data and code calculations can be used to define regions of code applicability.

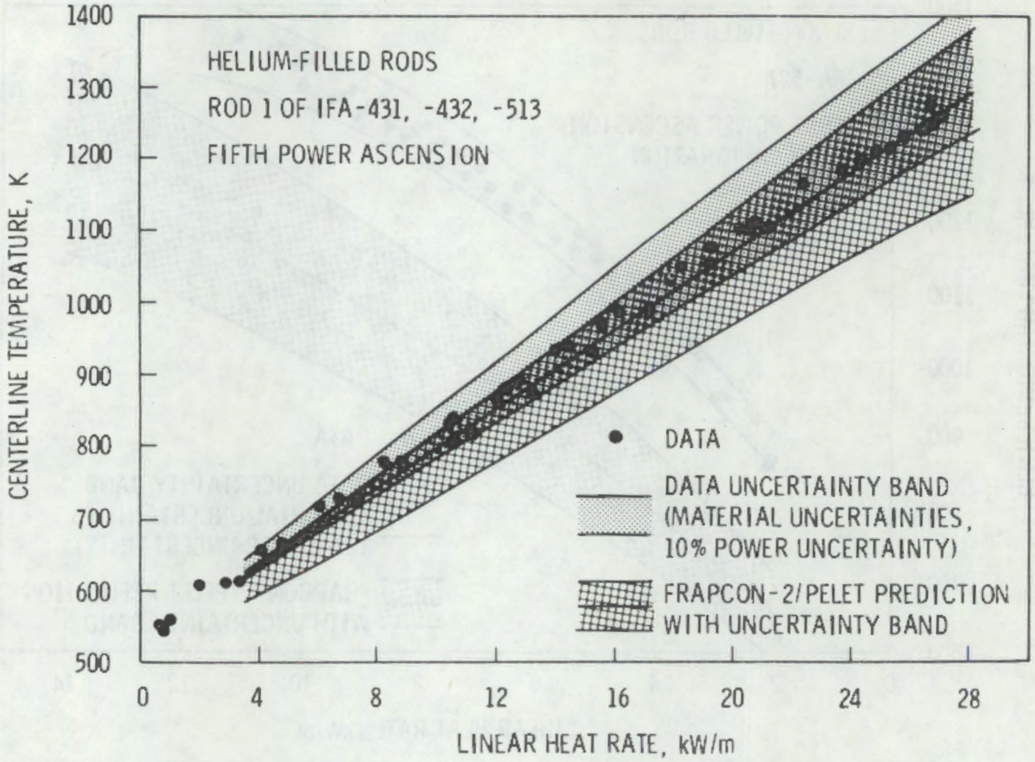


Figure 2. Comparison of Centerline Temperature Data, Code Calculation, and Uncertainty for Helium-Filled Replicate Fuel Rods

FUTURE WORK

Data processing for next quarter will continue as required. A report discussing the irradiation of a fuel rod with densifying fuel will be completed and forwarded to NRC for printing and distribution. Analysis of fuel cracking and relocation, errors and uncertainties, fission gas release, and related subjects will continue.

REFERENCES

1. Williford, R. E., et al. April 1980. *Interim Report: The Analysis of Fuel Relocation for the NRC/PNL Halden Assemblies IFA-431, IFA-432, and IFA-513.* NUREG/CR-0588, PNL-2709, Pacific Northwest Laboratory, Richland, Washington.

2. Cunningham, M. E., et al. October 1980. *Application of Linear Propagation of Errors to Fuel Rod Temperature and Stored Energy Calculations.* NUREG/CR-1753, PNL-3539, Pacific Northwest Laboratory, Richland, Washington.

3. Berna, G. A., et al. January 1981. *FRAPCON-2: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods.* NUREG/CR-1845, EG&G Idaho, Inc., Idaho Falls, Idaho.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK C - CODE COORDINATION AND EX-REACTOR TESTING[a]

D. D. Lanning, Program Manager
R. E. Williford, Task Leader

M. E. Cunningham
W. N. Rausch

SUMMARY

The FRAPCON-2 developmental assessment document was completed and sent to the Idaho National Engineering Laboratory (INEL) and the Fuel Behavior Research Branch (FBRB) of the U.S. Nuclear Regulatory Commission (NRC) for final approval and printing. Some detailed analysis was performed for the fuel rod compliance data to qualify the extent of localized cladding deformations. Equipment modifications and improvements were completed at Harwell, U.K.; and compliance testing of rods 7 and 8 of instrumented fuel assembly (IFA)-432 is ready to resume. Work continued on the three-dimensional (3-D) cracked fuel mechanical model.

INTRODUCTION

The primary objectives of the code maintenance and experimental support efforts are the documentation, maintenance, and improvement of the FRAPCON-2 licensing audit code. Code documentation will result in code descriptions and developmental assessment documents to be done jointly by Pacific Northwest Laboratory (PNL) and INEL. Code improvements include providing experimentally verified models to describe the mechanical interaction between the cracked fuel and the cladding and the quantification of operating conditions that lead to fuel failures with a specified probability.

In fiscal years (FY)-1979 and -1980 thermal-mechanical models were developed that describe the behavior of cracked fuel and were implemented in FRAPCON-2. Fuel cracking causes reduced thermal conductivity and elastic moduli and is presently described by three primary parameters—crack roughness, gap roughness, and crack pattern—that have been inferred from in-reactor data. In FY-1980, ex-reactor data were collected to confirm these parameters. In FY-1981, these experimental efforts will continue in concert with improvement of the cracked fuel model, which represents the driving component for the fuel failure model.

Task C efforts include the following subtasks: code maintenance, fuel mechanics experiments, and pellet-cladding interaction (PCI) model development.

TECHNICAL PROGRESS

Progress that has been made in each subtask during this quarter is summarized below.

(a) RSR FIN Budget No.: B2043; RSR Contact: H. H. Scott.

SUBTASK C-1 - FRAPCON-2 CONTROL AND MAINTENANCE

The final version of the FRAPCON-2 Developmental Assessment was completed and sent to INEL and FBRB for final approval and printing. The new FAST-GRASS fission gas release routine was received from Argonne National Laboratory (ANL), but the link with FRAPCON-2 has not been completed due to manpower and funding deficiencies. FRAPCON-2 test cases were run; and the results were presented at the Enlarged Halden Project Group meeting in Hanko, Norway, in June 1981. Comparisons to other codes and data yielded variable results, as described below.

All the codes (HOTROD, SLEUTHSEER, FEMAXI-III, GAPCON-3, FRAPCON-2, and FRP) did well compared to beginning-of-life (BOL) centerline temperature data from helium-filled rods. FRAPCON-2 was low in comparison to BOL fuel temperatures measured in low-power xenon-filled rods but did fairly well at calculating cladding creepdown and in identifying cases with low gas release. The Beyer-Hann gas release model overcalculated the gas release measured in some high-power Halden assemblies by a factor of over two. Using the FAST-GRASS gas release option in a low-power pressurized water reactor (PWR) rod simulation resulted in anomalous behavior that leads us to question either the correctness or the stability of the link between FAST-GRASS and FRAPCON. This link will be restudied as the latest version of FAST-GRASS is integrated into FRAPCON-2.

SUBTASK C-2 - FUEL MECHANICS EXPERIMENTS

Data analysis continued for the first "long rod" fuel compliance experiment performed by Harwell. This experiment was similar to companion experiments performed by PNL, and both simulate the loading systems that occur in irradiation environments. The fuel column is loaded axially, which places the cladding in axial tension; and cladding axial and diametral deformations are measured simultaneously.

Analog traces from the Harwell experiment were digitized, and a small computer code was written to transform the diametral data into radial deformation data. Significant azimuthal variations of bamboo ridges were found, and the magnitude of these variations appears to correlate with rod bowing measurements that were simultaneously recorded. Detailed analysis of the data is being performed in concert with the 3-D cracked fuel model being developed in subtask C-3 of this program.

Before continuing with compliance testing of rods 7 and 8 of IFA-432, hot cell modifications were needed to accommodate the compliance testing machine at Harwell. These modifications have been completed, the data acquisition system has been improved, and testing is ready to resume.

SUBTASK C-3 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

Work continued on the 3-D cracked fuel PCI model but at a reduced level due to staff commitments. Work began to formulate a system of differential equations to describe the interaction between fuel fragments of arbitrary shape and the cladding. This new model will be used to analyze the compliance data in subtask C-2 of this program.

FUTURE WORK

The following activities are planned for next quarter:

- A FRAPCON-2 "User's Letter" will be issued detailing the anomalies and errors found to date; it will include recommendations for future improvements.
- Harwell will complete the compliance testing of rods 7 and 8 of IFA-432.
- Analysis of PNL and Harwell compliance data should be completed using the 3-D cracked fuel mechanical model, and a report will be issued.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS[a]

D. D. Lanning, Program Manager
R. E. Williford, Task Leader

D. E. Fitzsimmons
R. K. Marshall

SUMMARY

The first instrument design for measuring fuel rod deformation was tested and found to be deficient; thus, a new design with more capabilities is in the fabrication stage. The detailed design of the loop pressure boundary was completed and materials were ordered. A computer code was written for detailed design of the fuel rod samples and analysis of loop operating parameters.

INTRODUCTION

The primary objective of Task D of the Experimental Support and Development of Single-Rod Fuel Codes Program is to collect fuel rod failure data on irradiated cladding under temperature loading conditions typical of those in-reactor, including asymmetrically cracked pellets and coolant external pressures. Fuel-induced pellet-cladding interaction (PCI) will be simulated with cracked annular pellets and an internal heater rod in a pressurized water loop facility at Pacific Northwest Laboratory (PNL). This experimental equipment has the capability for controlled power ramping and load cycling schemes and provides great experimental flexibility at a cost much lower than in-reactor experiments. The relationships between power ramp rate, localized cladding strain rate, and fuel rod relaxation rate will be characterized. The localized cladding deformations will be measured by an instrument especially designed and built for this purpose.

The loop will be proof tested in fiscal year (FY)-1981 with unirradiated cladding, and actual data collection with irradiated cladding will begin in FY-1982. These data complement Task C efforts and provide a means of verifying PCI models.

In Task D-1 a measurement instrument capable of characterizing the elastic deformation of the simulated fuel rod at power within the loop will be designed and produced.

TECHNICAL PROGRESS

Progress that has been made in each subtask during this quarter is summarized below.

SUBTASK D-1 - ROD STRAIN INSTRUMENT DESIGN

The first version of the instrument was fabricated and tested. This instrument was designed to measure cladding diameters at two perpendicular locations simultaneously. The sliding support contacts were found to cause unacceptable surface damage to the cladding sample. A simpler instrument was designed that minimizes contact with the cladding and is also able to simultaneously measure fuel rod bowing found in the compliance experiments (see subtask C-2 of this program). The new instrument is in the fabrication phase, and strain gages and thermocouples have been received.

(a) RSR FIN Budget No.: B2043; RSR Contact: M. L. Picklesimer.

SUBTASK D-2 - LOOP EXPERIMENTS

The detailed design of the loop pressure boundary was completed; and high-pressure pipe, flanges, etc., were ordered. The facility has been designed to permit easy sample changing with a minimum of sample handling and instrument decalibration. This will minimize the cost per sample when the irradiated cladding is tested in FY-1982. A computer code was written for the detailed thermal-mechanical analysis of the cracked fuel-heater rod test samples, and detailed design of the sample configuration is underway.

FUTURE WORK

The following activities are planned for next quarter:

- The new instrument will be fabricated, tested, and calibrated for use in loop experiments.
- Depending on shipping schedules for the required loop pressure boundary materials, the loop components will be fabricated and assembled.
- Fuel rod sample design will be completed using the new computer code, and samples will be prepared for proof testing.

PIPE-TO-PIPE IMPACT[a]

M.C.C. Bampton, Project Manager

J. M. Alzheimer
F. A. Simonen

SUMMARY

Emphasis during the past quarter has been on finalizing the design of the pipe accelerator and rigid supports, selecting a site for testing, procuring needed items, and addressing safety concerns.

INTRODUCTION

The Pipe-to-Pipe Impact Program provides the U.S. Nuclear Regulatory Commission (NRC) with experimental data and analytical models for making licensing decisions regarding pipe-to-pipe impact following postulated breaks in high-energy fluid system piping. Current licensing criteria—as contained in Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with Postulated Rupture of Piping"—will be evaluated. Data will be obtained from a series of tests in which selected pipe specimens with appropriate energies will be impacted against stationary specimens to achieve required damage levels.

This Pacific Northwest Laboratory (PNL) program involves two main areas: obtaining experimental data and developing predictive models. Preliminary analyses to determine significant test parameters and required energies and pipe velocities have been completed. The preliminary test matrix was developed, a system capable of accelerating the pipe was selected, and design of the test facility began. The next phase of the program will encompass construction of the test machines and actual testing. Predictive models will be developed that are analytically based and/or empirical fits of the data. These predictive models will be compared to current licensing criteria.

TECHNICAL PROGRESS

TEST FACILITY

Design of the pneumatic-powered pipe accelerator is complete. Several minor but time-consuming changes were made after the final review by safety personnel. After these changes were incorporated, the design was sent out for bids, which are due back the first week in July. It is expected that the accelerator will be delivered by late August.

An independent review of the accelerator design is being performed to assure that the design is adequate and that nothing has been overlooked in the first stress analysis. This review will be completed before actual testing begins. In addition, procedures will be developed for startup operation of the pipe accelerator to check actual behavior against predicted behavior to provide confidence that the methods used for developing the predictions were correct and that the accelerator is operating smoothly. Figure 1 shows a side view of the pipe accelerator, swinging pipe, target pipe, and rigid support part after the pipe has been accelerated to speed before impact.

For the first series of tests, the target pipe will be rigidly supported below the point of impact. Later tests will be run with the target pipe impacted between two supports. The design of the rigid supports is complete, and fabrication is underway.

(a) RSR FIN Budget No.: B2383; RSR Contact: M. Vagins.

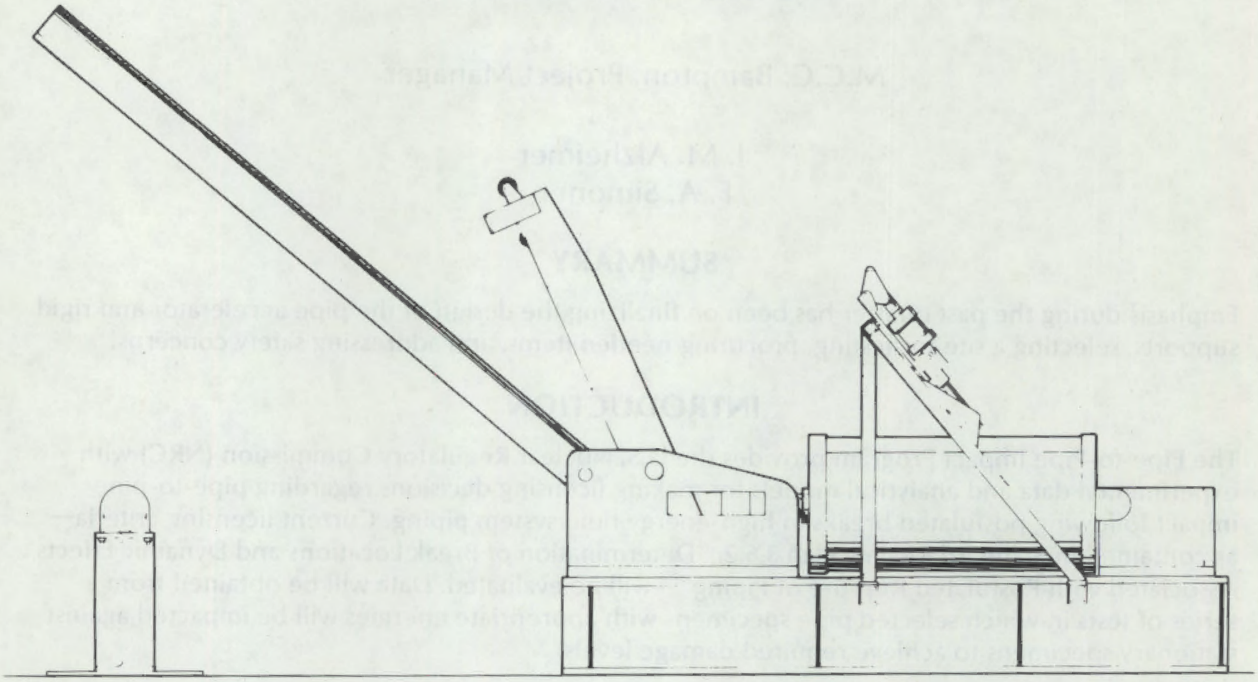


Figure 1. Side View of Pipe Accelerator, Swinging Pipe, Target Pipe, and Rigid Support

The test facility, at least for the first series of tests, will be located on a concrete pad. This site was selected after careful consideration of safety aspects. Experience gained during the first series of tests with rigidly supported pipes will give a better understanding of the response to expect from the simply supported target pipes. These later tests may require moving the facility to a location that will provide greater clearance below the target pipe and permit placing the target pipe in or above a trench. These tests are planned for next year.

Instrumentation for the tests will be associated with the pipe accelerator, the swinging pipe, and the target pipe. The pipe accelerator will include pressure transducers on both ends of the 20-in. diameter pneumatic cylinder and a displacement transducer on the piston stroke. The swinging pipe will have two accelerometers and probably a load pin at the pivot point for some tests. The velocity of the swinging pipe will also be measured just before impact. The target pipe will have a pressure transducer for pressurized tests and strain circles applied to the outside surface. Although it is not currently planned to have force transducers on the rigid supports, they may be included for the simple supports if the information is determined to be useful.

The instrumentation on the pipe accelerator will be used to correlate actual response to that predicted by the kinematics and dynamics computer program. This will provide information to assist in the selection of initial cylinder pressures to obtain needed pipe velocities. Instrumentation associated with the swinging pipe will be used to calculate impact energies and to check the structural integrity of the pivot connection. Data collected on the target pipe will relate to the damage associated with the impact; local strain and gross deformations will be of most interest.

ANALYTIC CONSIDERATION

The magnetic tape containing the ABAQUS-ND computer code has been received and is being installed on the computing system. Predictive models of the first tests are currently being developed; and, hopefully, the pivot forces on the swinging pipe will be estimated before testing starts. In addition, investigations on the effects of internal pressure on the target pipe response and on the effects of support spacings are planned.

An independent review of the pipe accelerator is underway to assure the safety and functionability of the design.

An analysis of the rupturing of the pressurized target pipe is underway to determine what measures will be required to assure safe operation of the tests. This analysis will be used in the design of the simply supported test configuration.

An analysis has begun to establish a separate experiment to investigate the pipe-to-pipe impact of in-core instrumentation lines. This is an area that is of particular concern to the NRC and the Advisory Committee on Reactor Safeguards (ACRS).

FUTURE WORK

The following activities are planned for next quarter:

- complete construction of the pipe accelerator
- complete rigid pipe supports
- develop finite element models of pipe tests
- complete initial tests
- develop in-core instrumentation line pipe-to-pipe impact tests.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

PBF SEVERE FUEL DAMAGE TEST PROJECT[a]

E. L. Courtright, Program Manager
R. L. Goodman, Project Manager

G. S. Allison
L. R. Bunnell
T. M. Fish
L. L. King
R. K. Marshall
M. A. McKinnon
S. D. Miller
L. J. Parchen
J. D. Rising
R. D. Tokarz
J. O. Vining
C. L. Wheeler

SUMMARY

Shroud development testing continued with construction and autoclave testing of a full cross section insulated shroud mockup about 0.30 m (1 ft) long. The mockup was autoclave tested at shroud differential pressures of 3.4 to 13.7 MPa (500 to 2000 psi) in 1.7-MPa (250-psi) increments. The mockup survived the autoclave testing with only minimal shroud inner liner deformation, and no water in-leakage to the insulated region of the shroud was detected. The shroud mockup was subjected to additional testing to examine the deformation and eventual rupture of the Zircaloy inner liner at elevated temperatures. Shroud inner liner failure occurred at 935°C (1715°F) at a shroud differential pressure of 1.4 MPa (200 psi). The inner liner failure occurred at the single site of a previously broken support post.

Testing of the inner liner concept continued with pieces of Zircaloy-2 cladding with a 0.076-cm (0.030-in.) wall being inductively heated with oxygen flowing inside of the tubes at heating rates from 1 to 4°C/s to peak temperatures of around 2000°C. At a 4°C/s heatup rate the oxidation became uncontrolled at approximately 1500°C, whereas specimens were successfully heated to 1920°C at a 1°C/s heatup rate. Even at the lower heating rate (1°C/s), a small amount of melting always occurred. A thick, sound layer of ZrO_2 was formed in these tests that was similar to that observed from electrically heated fuel rods ramp heated in steam at similar heating rates.

Measured thermal conductivity results on the 0.96 g/cc (60 lb/ft^3) ZrO_2 fiberboard were received from Dynatech Research and Development Company, Cambridge, Massachusetts. The measured values of thermal conductivity ranged from 0.094 W/m-K (0.056 Btu/h-ft-°F) at 40°C (104°F) to 0.196 W/m-K (0.116 Btu/h-ft-°F) at 1000°C and are in excellent agreement with vendor-supplied thermal conductivity data.

A formal design review of the fuel bundle and components was held during this reporting period. A major milestone in the design effort was reached with the completion of the formal design review of the instrumented fuel rods and bundle region of the test assembly including the tie place, spacer grids, and lower braze plug. The drawings were approved and released for hardware and component fabrication.

Analytical development focused on extending and improving the existing version of the TRUMP heat transfer computer code. Improvements in the surface-to-steam radiation model were made to

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(a) RSR FIN Budget No.: B2084-1; RSR Contact: R. Van Houten.

the code that allow for evaluations of an overall interchange factor between radiating and reflecting surfaces and between the surfaces and the steam. In addition, the capability to track a dry-out interface during a transient was added to TRUMP to allow the level-tracking model to perform realistic boil-off calculations necessary to support ongoing scoping and design support analysis efforts.

INTRODUCTION

This part of the Severe Core Damage (SCD) Test Subassembly Procurement Program—the Power Burst Facility (PBF) Severe Fuel Damage Test Project—has been divided into two tasks for fiscal year (FY)-1981.

TASK 1 - PBF SEVERE FUEL DAMAGE PROGRAM SUPPORT

The scope of this task includes the design, development of appropriate materials and supporting fabrication processes, and complete fabrication of two fully instrumented test train assemblies. Many portions of the PBF work should directly benefit the ESSOR program due to similarities in experimental objectives, particularly for materials development, instrumentation, and fabrication development. The program is designed to yield important experimental data related to fuel and cladding behavior during small-break accidents as well as to provide information on the postaccident coolability of damaged fuel rod clusters after small-break accidents.

TASK 2 - SMALL-BREAK WATER LEVEL CONTROL AND BOIL-OFF EXPERIMENTS

A small-scale boil-off experiment will be conducted to investigate the sensitivity of water level control to the absolute water level during simulated small-break conditions. The experiment will also examine the effects of water level on the axial temperature profile of the electrically heated rod bundle. Electrical heater rods with approximately 0.46 m of heated length and internal cladding thermocouples will be installed in a nine-rod square array inside a specially designed and fabricated high-pressure test section. The test section will be instrumented to measure inlet and outlet water/steam conditions as well as water level. Water at approximately 295K will be injected into the bottom of the test section at rates sufficient to maintain water levels between 10.2 to 22.9 cm (4 to 9 in.) above the beginning of the heated length. Nominal operating pressure will be about 650 psig, and both water injection rate and heater rod power will be varied over a matrix of test conditions.

TECHNICAL PROGRESS

The following paragraphs detail technical progress made during this reporting period by topic.

SHROUD DESIGN AND DEVELOPMENT TESTING

Autoclave Testing

Shroud development testing continued with construction and autoclave testing of a full cross section insulated shroud mockup about 0.30 m long. In previous small-scale autoclave testing it was found that collapse of the ZrO_2 fiberboard insulation and gross deformation of the shroud inner liner could be prevented by using short lengths of ZrO_2 tubing as support columns in the insulation. Fiberboard ZrO_2 insulation was applied to the shroud mockup of the sealed shroud design with tube support columns. The mockup used a 0.076-cm Zircaloy-2 octagonal inner liner with a 0.96 g/cc zirconia fiberboard insulation. To prevent gross deformation of the inner liner, which would result in crushing of the insulation, about 700 short lengths (0.71 m, 0.280 in.) of support tubing of 0.64 cm (0.250 in.) outside diameter (OD) by 0.48 cm (0.187 in.) inside diameter (ID) were used in the insulation and were located on approximately 1.02 cm (0.4 in.) center-to-center spacings.

The composite insulated shroud system with ZrO_2 fiberboard insulation and tube support posts are shown in Figure 1. The Zircaloy tube support saddles can also be seen. The outer round shroud liner that is welded to top and bottom flanges on the shroud mockup around the Zircaloy tube support saddles is not shown in the photograph.

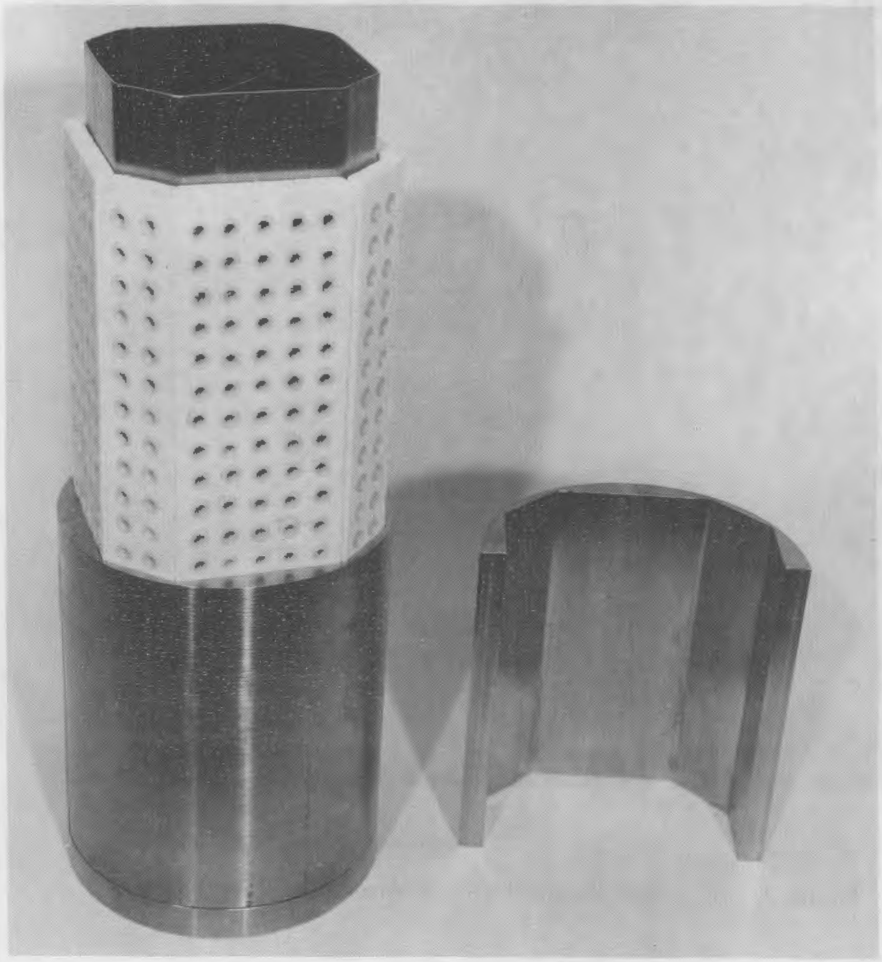


Figure 1. Composite Insulated Shroud System for PBF Severe Fuel Damage Test Train Assemblies

The mockup was autoclave tested at shroud differential pressures of 3.4 to 13.7 MPa in 1.7-MPa increments. The temperature for each test run was saturation temperature at each pressure; for example, 335°C at 13.7 MPa. Each pressure run was held for 3 h, and the mockup was then examined for damage and weighed to detect waterlogging after each test run.

The mockup survived the tests with only minimal deformation of the shroud inner liner between support tube locations. A single support tube broke at about 10.3 MPa (1500 psi) and was noted as an abnormal dent on the shroud inner liner. Post-test radiography confirmed the fracture of the single support tube. No water leakage was detected in the insulation region of the shroud during any of the testing. Figure 2 shows the full cross section insulated shroud mockup after completion of the 13.7-MPa autoclave test run. The autoclave testing verified the inner liner and fiberboard ZrO_2 insulation collapse resistance to 13.7 MPa at 335°C.

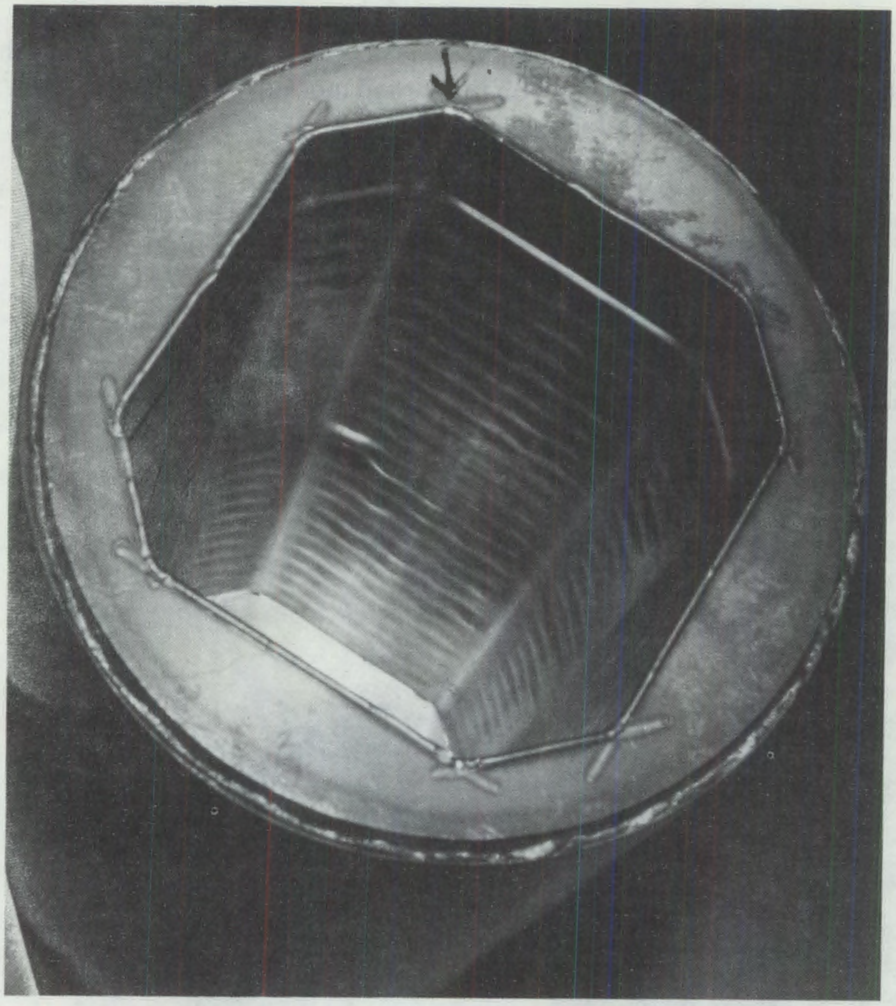
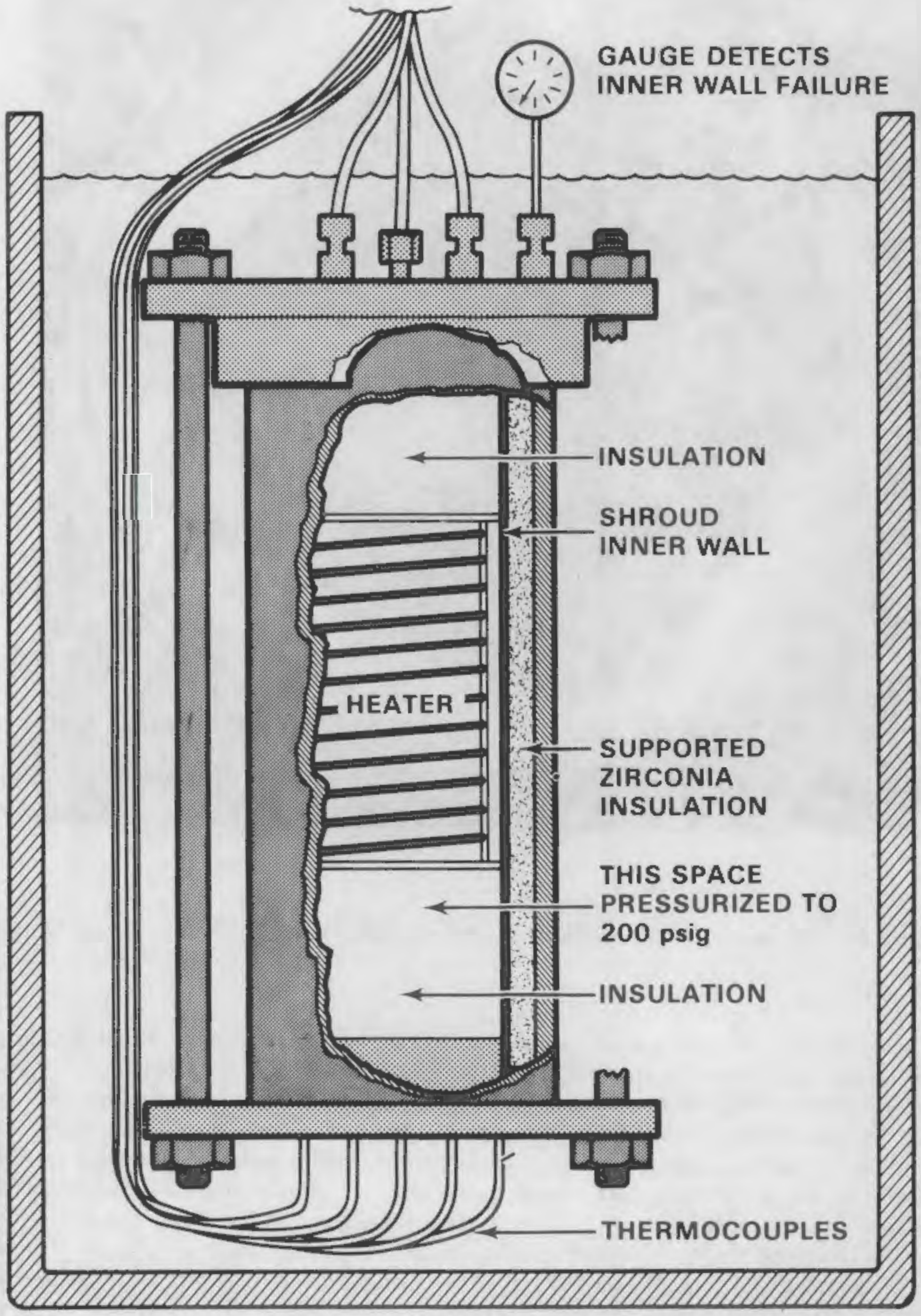


Figure 2. Insulated Shroud Mockup After Autoclave Testing

Elevated Temperature Testing

The full section insulated shroud mockup was subjected to additional testing to examine the deformation and eventual rupture of the 0.076-cm thick Zircaloy inner liner. A schematic view of the test setup for the inner liner elevated temperature testing of the full section shroud mockup is shown in Figure 3. A slip-in heater that conformed to the octagonal shape of the inner liner was used to create a steep axial thermal gradient in the shroud mockup. The total shroud structure was sealed and submerged in a water bath. A shroud differential pressure of 1.37-MPa gas pressure was applied inside the inner liner in the slip-in heater region of the mockup, and the inner liner temperature was ramped until a pressure gauge connected to the insulation region of the shroud signaled inner liner failure. The shroud inner liner failed at 935°C at a shroud differential pressure of 1.4 MPa; the failure occurred at the single site of a previously broken support post. Figure 4 shows the post-test condition of the full section shroud mockup after the inner liner elevated temperature testing. The arrow on the shroud top flange in Figure 4 indicates the approximate circumferential location of the failure of the inner liner. Inner liner failure occurred at approximately the axial midpoint of the inner liner. Figure 5 presents the axial temperature profile of the shroud



PROCEDURE: HEAT WITH DIFFERENTIAL PRESSURE UNTIL FAILURE,
EXAMINE FOR DISTORTION

Figure 3. Schematic of Test Apparatus for Inner Liner Elevated Temperature Testing

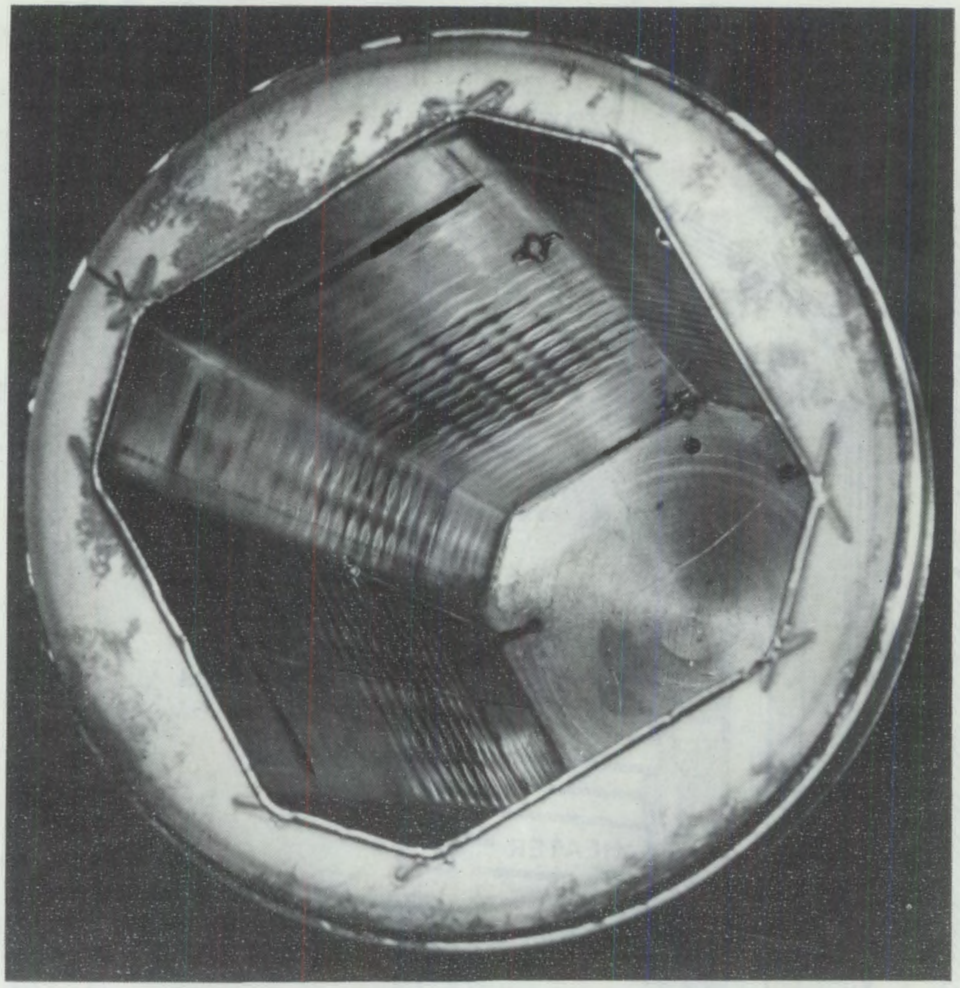


Figure 4. Insulated Shroud Mockup After Inner Liner Elevated Temperature Testing

inner liner as a function of relative axial position for a peak power of 5100 W, which was attained in the mockup heater at the time of inner liner failure. The direct relationship between the inner liner temperature profile and the permanent deformation of the inner liner can be seen from both Figures 4 and 5. The overall liner in the highest temperature location deformed a maximum of 0.11 cm (0.045 in.) radially between support posts at a shroud differential pressure of 1.4 MPa. No other outstanding effects on the inner liner were noted. Based on the results of the elevated temperature testing, it is seen that a simple pressure balance system should be used to keep the full system pressure of 6.9 MPa (1000 psi) from crushing the ZrO_2 fiberboard insulation. A simple pressure balance system of approximately 0.16 to 0.32 MPa (25 to 50 psi) below system pressure is desirable for the insulated shroud assembly during in-reactor testing.

A specimen of 0.96 g/cc ZrO_2 fiberboard was submitted to Dynatech Research and Development Company, Cambridge, Massachusetts, for thermal conductivity measurements. Apparent thermal conductivity measurements have been completed, and a brief description of the testing method as well as the results is presented below. The fiberboard sample, which was tested over a temperature range from 40 to 1100°C, was circular and approximately 20.3 m (8.0 in.) in diameter by 0.76 m (0.30 in.) thick.

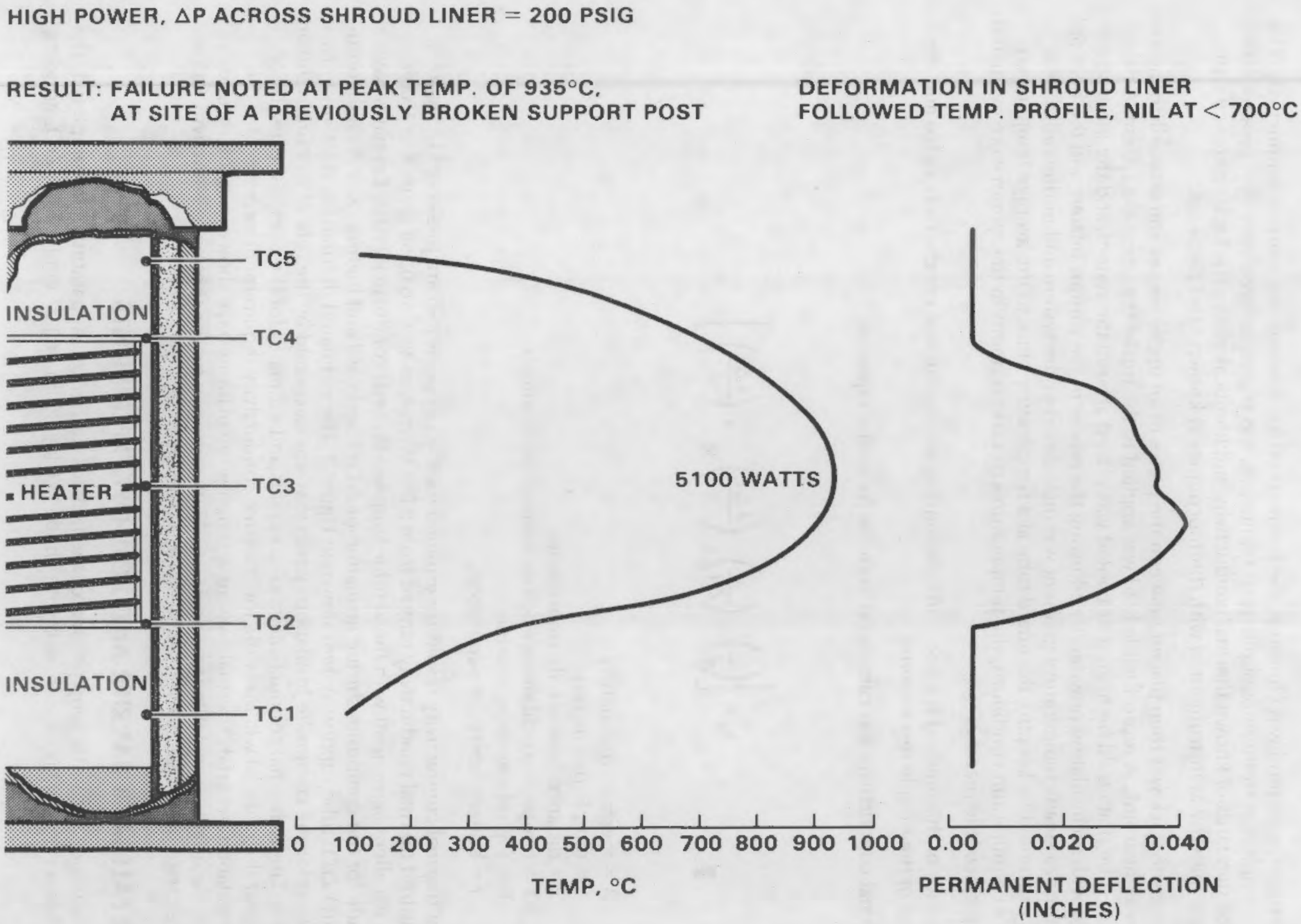


Figure 5. Shroud Inner Liner Axial Temperature Profile and Deformation for a Heater Power of 5100W

The comparative method was chosen as the test method for determining thermal conductivity. The sample, complete with thermocouple instrumentation, was placed between two Pyroceram® 9606 reference standards of known thermal conductivity and identical geometry. Each reference standard (heat meter) was instrumented with thermocouples at known fixed distances.

The composite stack was then placed between the plates of an upper heater and an auxiliary heater and a lower heat sink. A reproducible load was applied to the top of the complete system. A thermal guard tube that could be heated or cooled was placed around the system, and the inner space was filled with an insulating powder. By adjusting the power in the various heaters and the heat sink temperature, a steady temperature gradient was maintained in the system and undue radial heat loss was prevented by keeping the guard tube at a temperature close to the average temperature sample. At equilibrium conditions, the temperatures of various points in the system were evaluated from thermocouple readings.

The accuracy of this method is ±5 to ±10% depending on the thermal conductivity range and the condition of the sample test material.

The thermal conductivity was calculated from the following equation:

$$\lambda_s = \left[\left(\frac{1}{2}\right) \left(\frac{x}{\Delta T}\right)_s \left(\frac{\lambda \Delta T}{x}\right)_R + \left(\frac{\lambda \Delta T}{x}\right)_r\right]$$

where λ = thermal conductivity
s = sample parameters
x = distance between thermocouples
ΔT = temperature difference across material of distance x
R = top reference parameters
r = bottom reference parameters.

Apparent thermal conductivity results determined for the sample tested are presented in Figure 6. The measured thermal conductivity ranged from 0.094 W/m-K at 40°C to 0.196 W/m-K at 1000°C and is in excellent agreement with the vendor-supplied thermal conductivity data. Estimates have been made for the composite shroud insulation of ZrO_2 fiberboard and high-density (95% theoretical density) ZrO_2 tube support system shown in Figure 1. The estimated thermal conductivity versus temperature for the composite insulation system that was calculated by the rule of mixture is shown in Figure 6. These thermal conductivity measurements and estimates for the composite insulation indicate that the material will meet the low thermal conductivity requirement necessary for the shroud insulation material. Thermal-hydraulic scoping calculations have shown that the shroud insulation material is acceptable from a test feasibility standpoint if the thermal conductivity value lies in the range of 0.14 to 1.69 W/m-K (0.083 to 1.0 Btu/h-ft-°F).

WATER FALLBACK BARRIER AND BOIL-OFF EXPERIMENTS

All field welds on the water loop portion of the fallback barrier test apparatus were completed. The fallback barrier test section was loaded with about 13.6 kg (30 lb) or 1.27-cm (1/2-in.) ZrO_2 grinding

®Trademark of Dynatech Research and Development Company, Cambridge, Massachusetts.

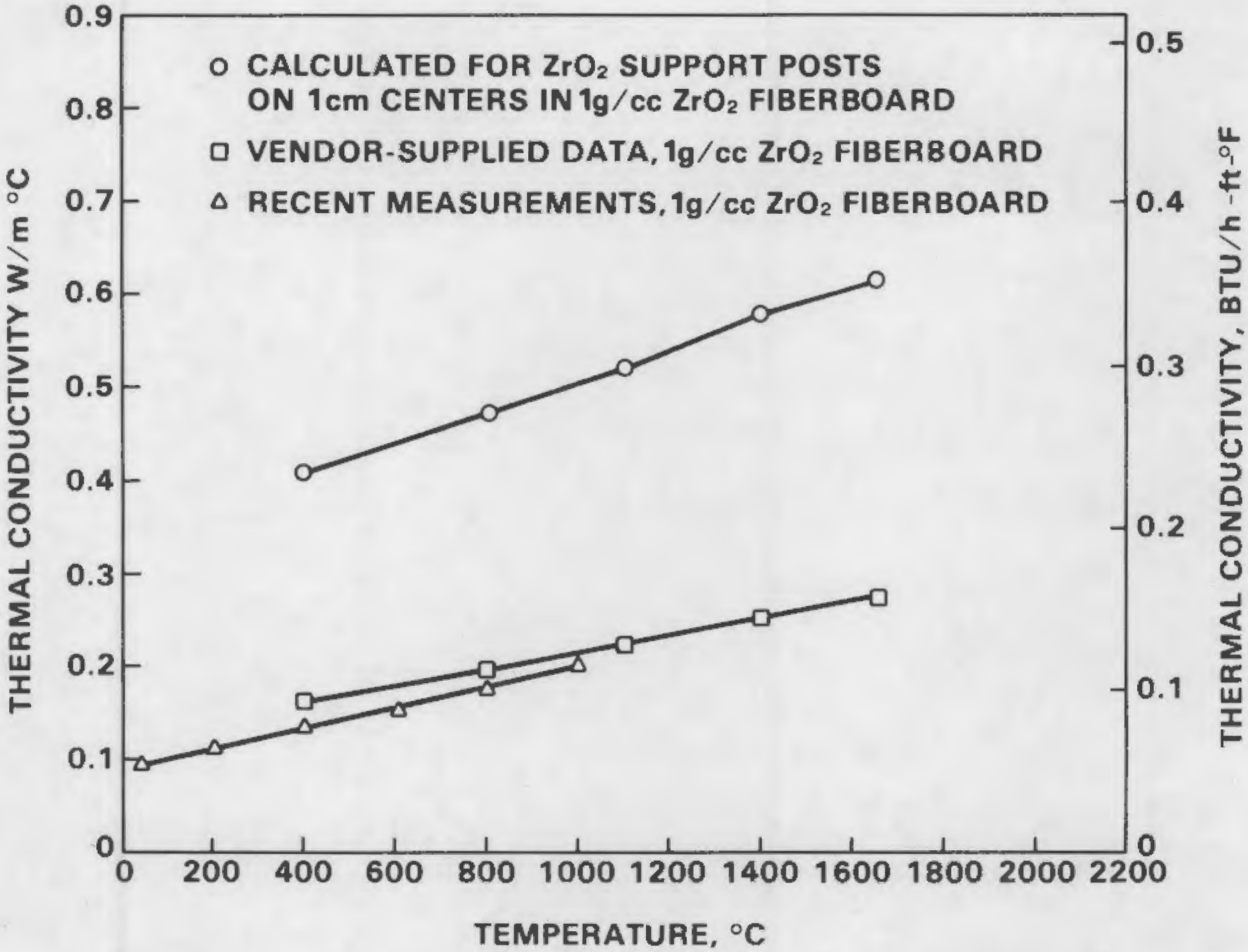


Figure 6. Apparent Thermal Conductivity Versus Temperature for 0.96-g/cc ZrO_2 Fiberboard

cylinders and instrumented with 16 thermocouples. Two 2.54-cm (1-in.) thick honeycomb ceramic disks of circular cross section were used to support the grinding cylinders. The test section has been installed in the loop, and the water loop with its related pressure manifolds has been hydrostated to 17.1 MPa (2500 psi). Loop temperature controllers for the fallback barrier and boil-off tests, pressure transmitters, and electrical and mechanical tubing connections have been installed. Present plans call for fallback barrier testing to begin in mid-July 1981.

The Kay Ray liquid level detection system and steam superheater needed to support the boil-off tests were received during this reporting period. The superheater and boiler were installed in the steam portion of the loop and were successfully acceptance tested. Heater rods for the boil-off tests are scheduled to be received by July 10, 1981.

FUTURE WORK

Fabrication and assembly of component parts for the instrumented fuel rods for the PBF severe fuel damage tests will continue. Detailed design of the insulated shroud region of the test train assembly will be completed, and a design review of the assembly will be held with EG&G Idaho, Inc. Pressure drop testing of the proposed fallback barrier system will be completed for a range of typical preconditioning flow rates. Boil-off testing with the heater rods in the boil-off test apparatus will be started.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

ESSOR FUEL DAMAGE TEST PROGRAM SUPPORT[a]

E. L. Courtright, Program Manager
F. E. Panisko, Project Manager

G. S. Allison	L. J. Parchen
L. R. Bunnel	R. L. Shaub
J. M. Creer	J. E. Tanner
U. P. Jenquin	R. T. Tokarz
L. L. King	J. W. Upton, Jr.
N. J. Lombardo	C. L. Wheeler

SUMMARY

Progress is reported in the area of program management including technical information exchange between Pacific Northwest Laboratory (PNL) and the Joint Research Center (JRC) at Ispra, Italy, and in the area of experimental predesign thermal analysis.

INTRODUCTION

The Super Sara Test Program (SSTP) is a major European community effort to study reactor safety during rapid or large-break and slow or small-break loss-of-coolant (LOC) events. The program will use the SUPER SARA high-temperature, high-pressure loop in the ESSOR reactor at Ispra, Italy. The SSTP is designed to yield important experimental data on fuel rod deformation and postaccident coolability of damaged fuel assemblies after they experience a loss of coolant. The complete testing program currently includes loop construction and 21 in-pile experiments to simulate 7 large- and 14 small-break conditions in commercial pressurized water reactors (PWRs) and boiling water reactors (BWRs). PNL will supply 3 of the 14 fully instrumented and fueled small-break test assemblies and analytical and engineering support as the U.S. Nuclear Regulatory Commission (NRC) stipend to the SSTP.

TECHNICAL PROGRESS

PROGRAM MANAGEMENT/TECHNICAL EXCHANGE

A significant effort was expended in planning/scheduling/costing PNL's effort as the NRC representative to the SSTP. Numerous options were explored consistent with the needs of the NRC and available funding while trying to provide optimum support to JRC-Ispra. Some of these plans were revised due to a four-month delay on the European side of the program. Initial efforts were directed toward demonstrating severe fuel damage (SFD) testing feasibility that included developing satisfactory analytical capabilities. Efforts on test feasibility led to conceptual test apparatus design, especially the fuel bundle shroud; and currently, progress is being made in shroud development and demonstration plus supplying technical support to JRC-Ispra in the areas of safety, licensing concerns, loop construction, and related SFD testing.

(a) RSR FIN Budget No.: B2372-1; RSR Contact: R. Van Houten.

Various technical meetings were held both at PNL and Ispra, Italy, with JRC-Ispra staff. Dr. R. Nijsing (Ispra) visited PNL and discussed our current analytical capabilities and limitations plus our expected future improvements. At a meeting held at PNL, J. Bourdon and O. Simoni presented the JRC-Ispra current large-break test train design as well as some ideas for SFD shroud design. Related work at PNL for NRU large-break testing, Power Burst Facility (PBF) nine-rod test train design and fabrication, plus PBF severe fuel damage test apparatus design, construction, and analysis were presented and discussed. J. Upton, Jr., (PNL) and R. McCulloch (Oak Ridge National Laboratory) participated in an Ispra workshop for the development of electrically heated fuel rod simulators for SSTP out-of-pile testing. Dr. Upton made preliminary arrangements for a 2-yr assignment in Ispra that will begin early this fall. T. Doyle (Ispra) visited PNL to discuss our quality assurance system relative to the needs of JRC-Ispra for the SSTP test trains.

EXPERIMENTAL APPARATUS PREDESIGN THERMAL ANALYSIS

During this quarter, analytical development focused on extending and improving our version of the TRUMP code. The effort consisted mainly of improving the surface-to-steam radiation model and in adding the capability of tracking a dry-out interface during a transient. In addition, several minor changes were made to enhance the versatility and readibility of the code.

The changes to the surface-to-steam radiation model were made because the previous approximation only considered absorption from radiation between any two surfaces. The new model uses the method of Hottel(1) for enclosures of gray surfaces containing a gray gas of uniform temperature. Hottel's method is used to calculate an overall interchange factor between radiating and reflecting surfaces and between the surfaces and the steam. The foundation for this model was an existing in-house routine written for the NRU program, which generates TRUMP radiation model input cards when absorption by the medium is neglected.

The liquid level tracking mode that was added to TRUMP now allows realistic boil-off calculations for our design analysis. Model power and/or inlet flow histories are inputted, and a dry-out level is then calculated using a steady-state energy balance to find the elevation when all of the fluid vaporizes. This model neglects the energy required to vaporize the liquid as the level moves from one elevation to another; however, the error will be small for slow transients typical of those expected from the severe core damage tests. (The error in the total energy deposited in the steam in the example problem subsequently discussed is less than 0.5%.) For calculations when the inlet coolant flow is zero or the boil off is rapid, it will be necessary to modify the above approach by inputting a liquid level history.

A sample problem using the PBF severe core damage test train geometry was chosen to evaluate this model. Since the same problem is being analyzed by EG&G Idaho, Inc., we will be able to compare our model to theirs. The input for this problem is as follows:

- The rod average heating rate linearly increases from 1.2 to 5.4 kW/m in 3600 s.
- The inlet flow is held constant at 0.0336 kg/s with an inlet subcooling of 40K.
- The system pressure is held constant at 6.89 MPa.

The metal-water reaction is modeled using the Cathcart(2) correlation. Sample results are given in Figures 1, 2, and 3. In Figure 1, typical temperatures for clad, steam, and shroud are given for the duration of the transient. The discontinuous appearance results from a combination of the way the heat transfer coefficient at the interface and the heat from metal-water reactions are handled. In the current model the heat transfer coefficient of the axial node containing the interface is held constant at a relatively high value until the water/steam interface crosses the centerline of the

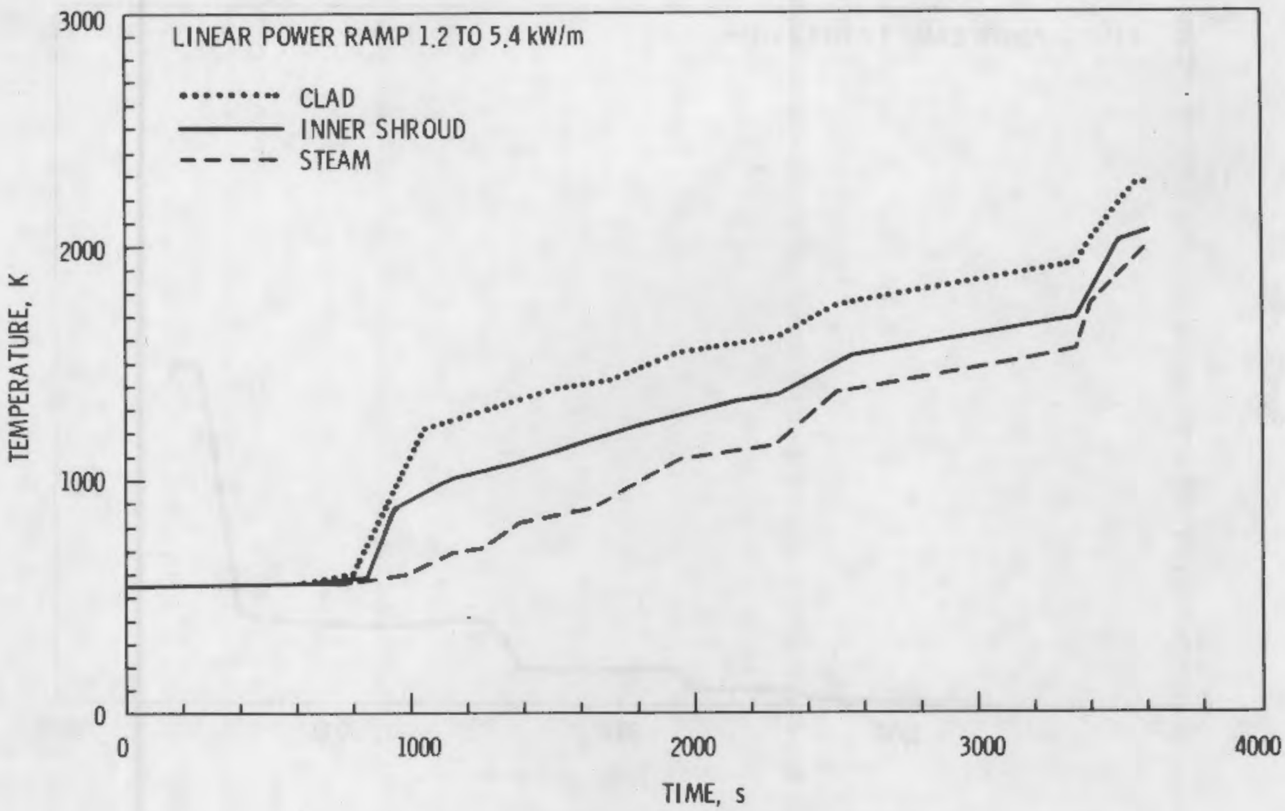


Figure 1. PBF Sample Problem - Temperature Versus Time

node. Then the coefficients suddenly drop to a value defined by the Nusselt number correlation:

$$Nu = (7.86)Pr^{0.333}$$

where Pr is the Prandtl number.

This produces a rapid increase in the local clad temperature at the node containing the interface and allows for rapid oxidation of the clad over the full length of the node. A large fraction of the energy from the heat of oxidation is deposited in the steam, which is then convected downstream and accelerates the oxidation in downstream nodes. A curve showing the rod average heating rate from zirconium dioxide formation is given in Figure 2, while the axial distribution of Zircaloy consumed in the shroud inner liner is given in Figure 3.

FUTURE WORK

- SSTP planning and scheduling will continue as needed to respond to NRC and JRC needs during these early stages of loop and test train construction and Comitato Nationale per l'Energia Nucleare (CNEN) licensing submittals. Informal technical information exchanges will take place as needed. The first PNL-supplied site representative—Dr. J. W. Upton, Jr.—and one or two other U.S. experts will reside at Ispra to support JRC staff. In addition, other U.S. experts on safety analysis, experiment licensing, and test operations will support the JRC staff to maintain program schedule and direction.

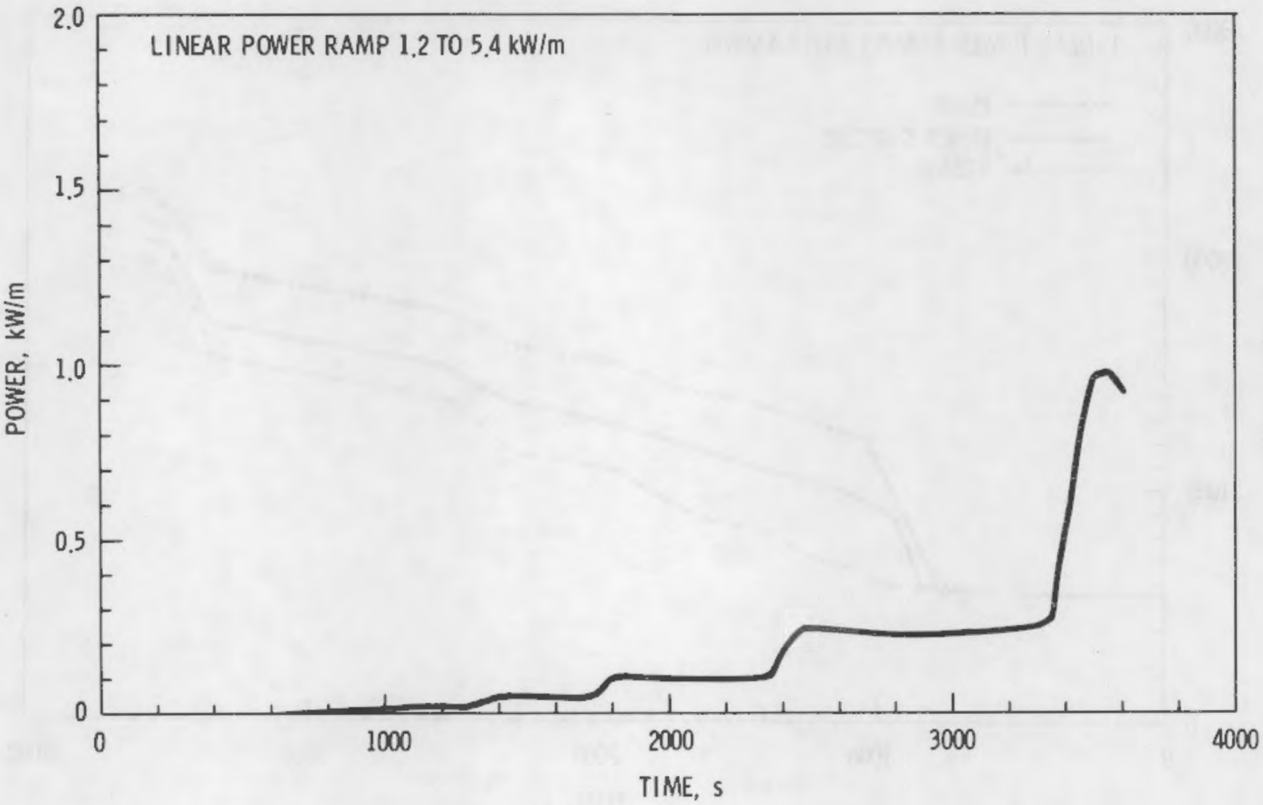


Figure 2. PBF Sample Problem - Rod Average Power Resulting from Metal-Water Reaction

- Evaluation of the TRUMP model will continue by comparing our results to those of EG&G Idaho, Inc., using PBF geometry, General Electric calculations using SUPER SARA geometry, and the results of various in-house mini-verification tests.
- Shroud development efforts, including the evaluation of promising high-temperature ceramic protective coatings, will continue. Various combinations of shroud Zircaloy and ZrO_2 fiberboard components will be tested in inert gas and oxygen environments at high temperatures in the presence of liquid Zr, Zr(O), and UO_2. Inert gas is used to maintain furnace integrity.
- A simple liquid level detection system will undergo demonstration testing. Basic proof-of-concept tests were successfully completed, and a patent application was filed. This system shows promise for two-phase composition (quality) measurements as well as liquid detection.

REFERENCES

1. McAdams, W. H. 1954. *Heat Transmission*. McGraw-Hill Book Company, Inc., New York.
2. Hagrman, D. L., and G. A. Reymann. 1979. *MATPRO—Version 11: A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior*. NUREG/CR-0497, EG&G Idaho, Inc., Idaho Falls, Idaho.

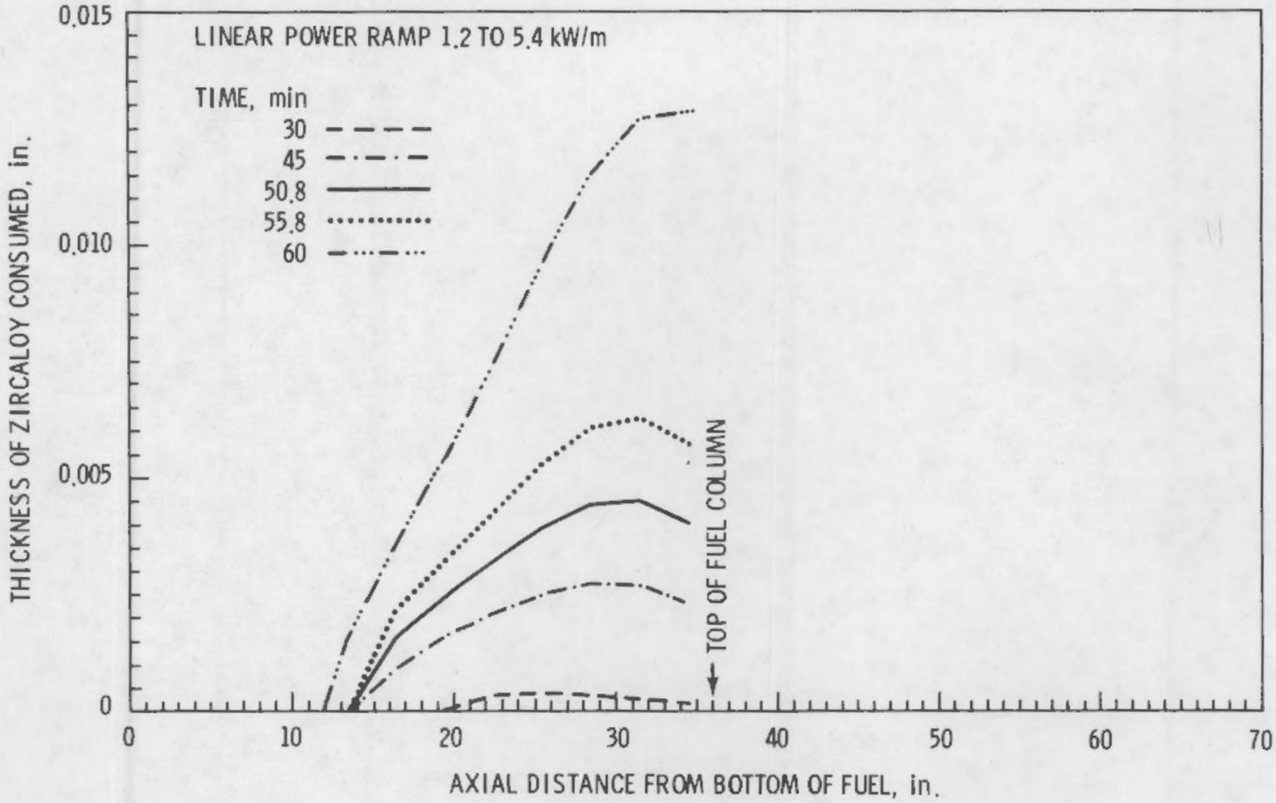


Figure 3. PBF Sample Problem - Zircaloy Consumed Versus Elevation for Shroud Inner Liner

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

NINE-ROD TEST TRAIN OPTRAN 1-3 PROJECT[a]

E. L. Courtright, Program Manager
R. E. Schreiber, Project Manager

R. E. Falkoski
D. E. Hurley
L. L. King

SUMMARY

Since only minimal work was done on the project this quarter, complete details of final work will be given in the quarterly report for July-September 1981.

INTRODUCTION

The purpose of this Pacific Northwest Laboratory (PNL) project is to provide instrumented test train hardware for Power Burst Facility (PBF) experimental programs. A nine-rod test array was chosen for the abnormal operating transient simulation (OPTRAN 1-3).

In support of this experiment, a test train has been designed with a replaceable core region and shroud to allow other tests of a similar nature to be performed with the basic assembly. The design allows for insertion and removal of one rod following the preconditioning phase and has built-in features to facilitate disassembly and repair.

The OPTRAN 1-3 experiment involves a series of transients in the PBF. Since the instrumented fuel bundle will be composed of irradiated rods that have seen normal boiling water reactor (BWR) service, it has been necessary to develop procedures and equipment for remotely assembling the test train. The assembly work is to be done by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL).

(a) RSR FIN Budget No.: B2034-1; RSR Contact: R. Van Houten.

CORE THERMAL MODEL DEVELOPMENT[a]

M. J. Thurgood, Project Manager

J. M. Kelly
R. J. Kohrt

SUMMARY

Development assessment of the COBRA/TRAC computer code continued during the past quarter. Reasonable predictions of various two-phase flow phenomena that are important in reactor safety were made, including comparisons with reflood and critical heat flux experiments. Comparison of code predictions with experimental data is in general very good. The code was also released to the Argonne Code Center this quarter.

INTRODUCTION

The COBRA-TF computer code is being developed as part of the U.S. Nuclear Regulatory Commission (NRC) Water Reactor Safety Research Program in the area of analysis development. The purpose of this work is to provide better digital computer codes for computing the behavior of full-scale reactor systems under postulated accident conditions. The resulting codes are being used to perform pre- and post-test analysis of light water reactor (LWR) components and system effects experiments. This Pacific Northwest Laboratory (PNL) project has two main objectives:

- to develop a hot bundle/hot channel analysis capability that will be used in evaluating the thermal-hydraulic performance of LWR fuel bundles during postulated accidents
- to develop a water reactor primary system simulation capability that can model complex internal vessel geometries such as those encountered in upper head injection (UHI)-equipped pressurized water reactors (PWRs).

COBRA-TF is formulated to model three-dimensional (3-D), two-phase flow using a three-field representation: the vapor field, the continuous liquid field, and the droplet field. The model allows thermal nonequilibrium between the liquid and vapor phases and allows each of the three fields to move with different velocities. Thus, one can mechanistically treat a continuous liquid core or film moving at a low or possibly negative velocity from which liquid drops are stripped off and carried away by the vapor phase. This is an essential feature in the treatment of the hydrodynamics encountered during the reflooding phase of a loss-of-coolant accident (LOCA). This model allows the prediction of liquid carryover in the FLECHT and FEBA[b] series of experiments. The treatment of the droplet field is also essential in predicting other phenomena such as CCFL and upper plenum deentrainment and fallback.

The code also features flexible noding, which allows modeling of the complex geometries encountered in reactor vessel internals, such as slotted control rod guide tubes, jet pumps, and core bypass regions. These geometries cannot be modeled easily in regular Cartesian or cylindrical mesh coordinates; however, since they have significant impact on the thermal-hydraulic response of the system, these geometries must be modeled with reasonable accuracy.

(a) RSR FIN Budget No.: B2041; RSR Contact: S. Fabic.
(b) Flooding experiment in blocked arrays experiment.

The fuel rod heat transfer model utilizes a rezoning mesh to reduce the rod heat transfer mesh size automatically in regions of high heat flux or steep temperature gradients and to increase the mesh size in regions of low heat flux. This model has proven very effective in resolving the boiling curve in the region of the quench front.

COBRA-TF has been implemented into TRAC-P1A as the vessel module, providing a system simulation with the capabilities described above. The resulting code, referred to as COBRA-TRAC, is being assessed by comparing its predictions of various two-phase flow experiments with the measured data from the experiments. Several such simulations have been completed during the past quarter.

TECHNICAL PROGRESS

BENNETT CRITICAL HEAT FLUX EXPERIMENTS

Ten of the experiments conducted by A. W. Bennett et al.[1] concerning heat transfer to steam-water mixtures have been simulated by COBRA/TRAC. The objectives of the simulations are to demonstrate the code's ability to calculate dry out of a liquid film and heat transfer to the liquid deficient region resulting from power input exceeding the critical heat flux (CHF).

Description of Experiment

Bennett uses a 219-in. electrically heated nimonic tube of 0.497-in. inside diameter (ID) at 1000 psia. Nimonic has a low-temperature coefficient of electrical resistibility that allows for a uniform heat flux with large axial variation in temperature. The CHF condition is attained by increasing the power to the test section until a rapid increase in temperature is observed. When the test section reaches steady state, the temperatures, power input, mass flow, and exit pressure are recorded.

COBRA/TRAC Model

The test section was modeled with twenty 1-ft long vertical nodes. The flow rate and enthalpy were specified at the bottom of the test section and two TRAC components were used to specify a 1000-psia boundary condition at the top.

Discussion of Results

Table 1 summarizes the test conditions and compares the COBRA-TF calculated dry-out points with those measured experimentally. The maximum difference between measured and calculated dry-out points is 22 in. Considering that the COBRA-TF nodes are 12 in. apart and that the distance between thermocouples on the test section is 6 to 13 in., a difference of only 22 in. is good.

Rod temperature profiles for tests 5359 and 5397 are shown in Figures 1 and 2. The dry-out point is indicated by a rapid increase in temperature. Above the dry-out point heat is transferred from the wall to a flowing mixture of steam and water droplets. In test 5359 rod temperature continued to increase with distance above the dry-out point. When the mass flow rate was increased, as in test 5397, rod temperature decreased with distance above the dry-out point. Generally, COBRA-TF temperature profiles compared well with the shape of the measured profile. The maximum difference between measured and calculated rod temperature was 200°F; these results were obtained without the use of an empirical CHF correlation. The CHF location occurred at the point where the liquid film was depleted, and film dry out occurred as a result of entrainment of liquid drops and evaporation from the film.

Table 1. Summary of Bennett Tests

Test Number	Power, kW/ft	Mass Flow Rate, lbm/s	Inlet Enthalpy, Btu/lbm	Dry-out Point, in. Experiment	COBRA Simulation
5359	6.58	0.1085	464.6	140.0	138.0
5336	9.92	0.1834	484.6	140.0	138.0
5273	11.12	0.2807	489.6	140.0	162.0
5250	11.07	0.3742	494.6	164.0	186.0
5294	13.26	0.5389	500.6	164.0	186.0
5310	13.64	0.7036	500.6	185.0	198.0
5379	20.66	1.048	517.6	140.0	162.0
5397	22.25	1.430	510.6	164.0	174.0
5313	14.52	0.6998	498.6	170.0	186.0
5368	15.73	1.063	516.6	207.0	198.0

JAERI CYLINDRICAL CORE TEST FACILITY

COBRA/TRAC was used to simulate a gravity feed reflood experiment conducted in the JAERI cylindrical core test facility (CCTF)(2) to assess the code's ability to predict gravity feed reflood transients.

Description of Experiment

The large-scale reflood experiment CCTF test C1-2 (Run 011) was chosen for this simulation. The CCTF was designed to model a full-height core section and the intact and broken loops of a PWR. The facility contained a non-nuclear core of 2000 electrically heated fuel rod simulators arranged in a rectangular array and placed within a cylindrical vessel. The test vessel includes a downcomer, lower plenum, core region, and upper plenum with associated internals. The intact loop represents the three intact loops of a PWR and included a steam generator and pump simulator with associated piping. The broken loop also contained a steam generator and pump simulator. The test was initiated by turning on the power to the core with the vessel filled with steam with the exception of 0.86 m of saturated water in the lower plenum, allowing the rods to undergo a nearly adiabatic heatup. Fifty-three seconds after power was applied to the fuel rod simulators, accumulator injection into the lower plenum was initiated. When the liquid level reached the bottom of the core at 69 s, the accumulator injection was switched to the cold leg for the remainder of the transient. The core power decay was initiated at 67 s, and at 78 s accumulator injection ended and low pressure coolant injection (LPCI) was initiated. All heater rods were quenched by 588 s.

COBRA/TRAC Model

The vessel was modeled using a one-dimensional representation of the core, lower plenum, downcomer, and upper plenum. The flow paths through support columns and guide tubes in the upper plenum were also modeled. One-foot long nodes were used in the core. The heater rods were modeled using a single average rod heat transfer model. The loops were modeled with TRAC one-dimensional components. The containment was modeled as a constant pressure boundary condition.

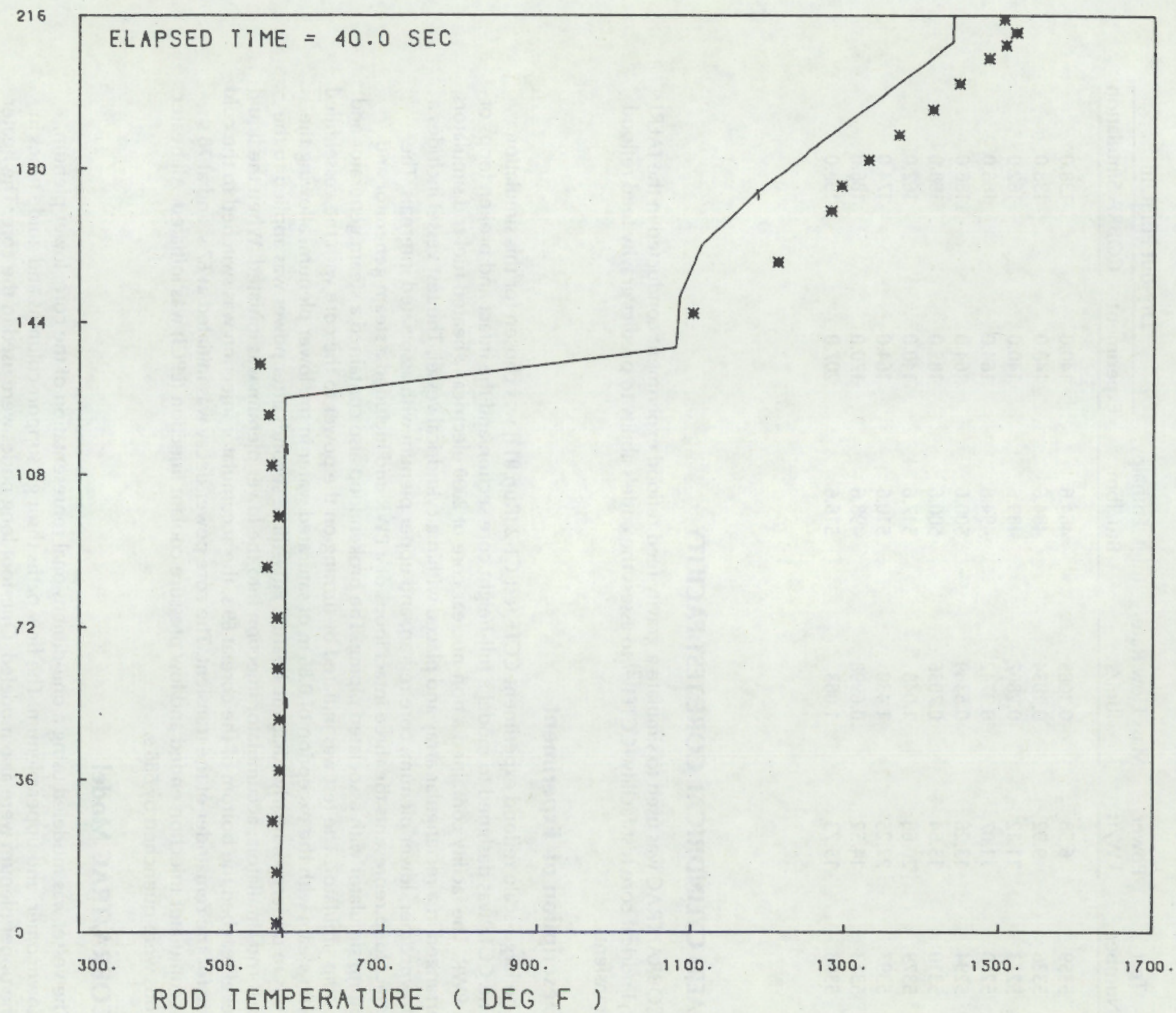


Figure 1. Bennett Test 5359

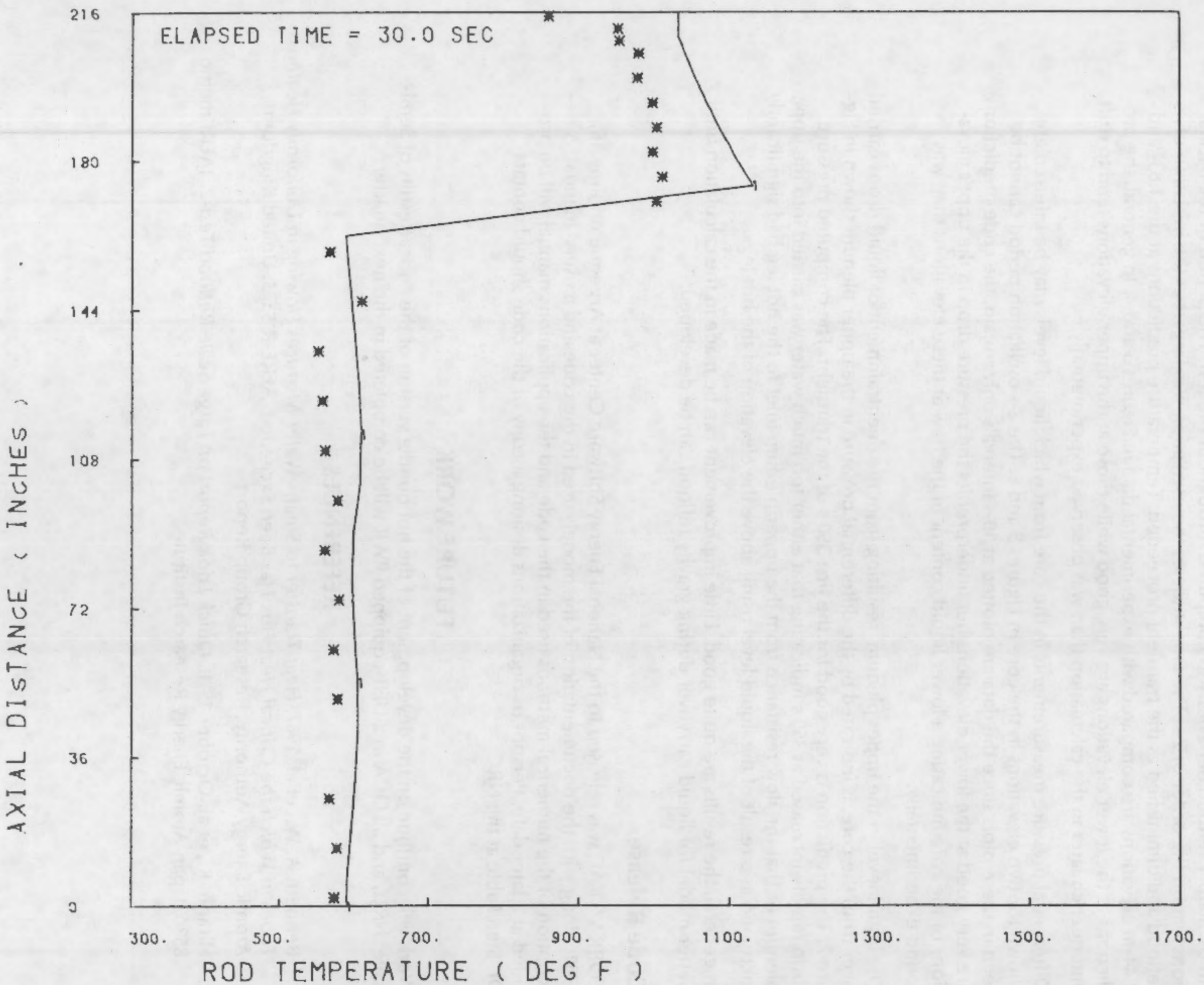


Figure 2. Bennett Test 5397

Discussion of Results

An oscillatory reflood behavior was computed throughout the test. The period of the oscillations was on the order of 2 s. The amplitude of the oscillation was largest at the initiation of reflood and diminished as the transient proceeded. Temperature predictions at the 1.83- and 2.44-m elevations are compared with experimental data in Figures 3 and 4. In general, the predictions at the lower elevations are very good while those at the upper elevations tend to drift and quench later in the calculation than was observed experimentally.

Differential pressure measurements in the core from which liquid levels may be inferred are shown for two elevations in the core in Figures 5 and 6. The 2-s oscillation period cannot be seen in these plots since the data are plotted at 10-s intervals only. Again, the code predictions are very good at the lower elevations but underpredict the pressure drop in the upper elevations of the core indicating a lower liquid content of the flow at these elevations than was found experimentally.

The liquid level in the upper plenum, resulting from the deentrainment of liquid drops carried over from the core, is indicated by the differential pressure in the upper plenum shown in Figure 7. The prediction is very good for the first 350 s of the transient. The computed pressure drop rapidly increased at 350 s indicating that either too much water was carried into the upper plenum or that the flow resistance from the upper plenum inlet to the hot leg had significantly increased as a result of the liquid level rising above the elevation of the hot legs.

In general, the results are quite good. Little improvement can be made in the calculation until a better model for liquid carryover during gravity reflood can be developed.

Code Release

COBRA/TRAC was released to the National Energy Software Center at Argonne on June 16, 1981, along with the documentation of the models used in the code and a users' manual. A description of the numerical methods used in the code and the applications manual will be provided at a later date. Persons or organizations desiring a copy of the code should contact Dr. Stan Fabic at the NRC.

FUTURE WORK

Work will continue on the development of the hot bundle version of the code. Results of Semiscale, NRU, and a LOCA in a UHI-equipped PWR will be completed in the next quarter.

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1. Bennett, A. W., et al. 1967. *Heat Transfer to Steam-Water Mixtures Flowing in Uniformly Heated Tubes in Which the Critical Heat Flux Has Been Exceeded.* AERE-R 5373, United Kingdom Atomic Energy Authority, Research Group Report.

2. Hirano, K., et al. October 1970. *Quick Look Report on Large Scale Reflood Test-2.* JAERI memo 8530, Japan Atomic Energy Research Institute.

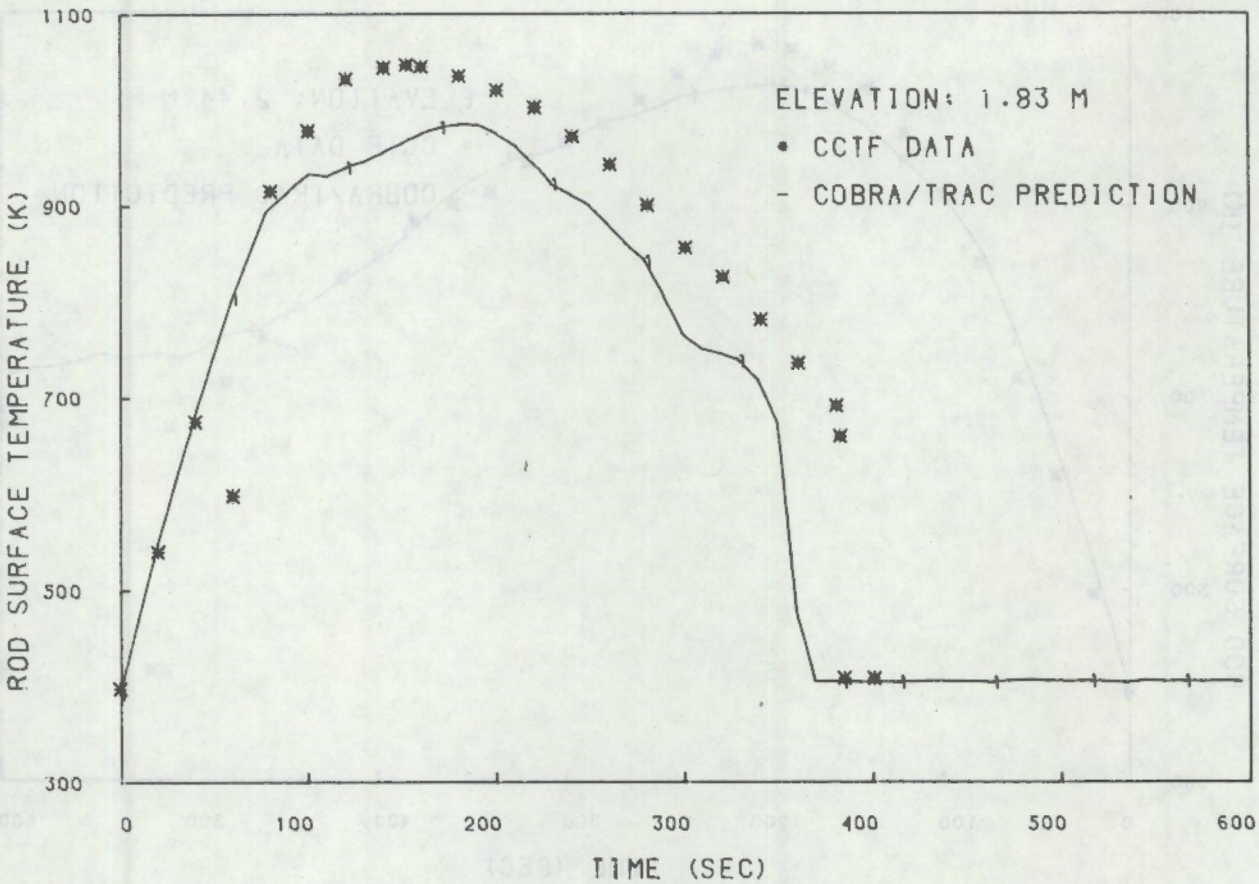

Figure 3. CCTF Test C1-2 Heater Rod Temperature at 1.83-m Elevation

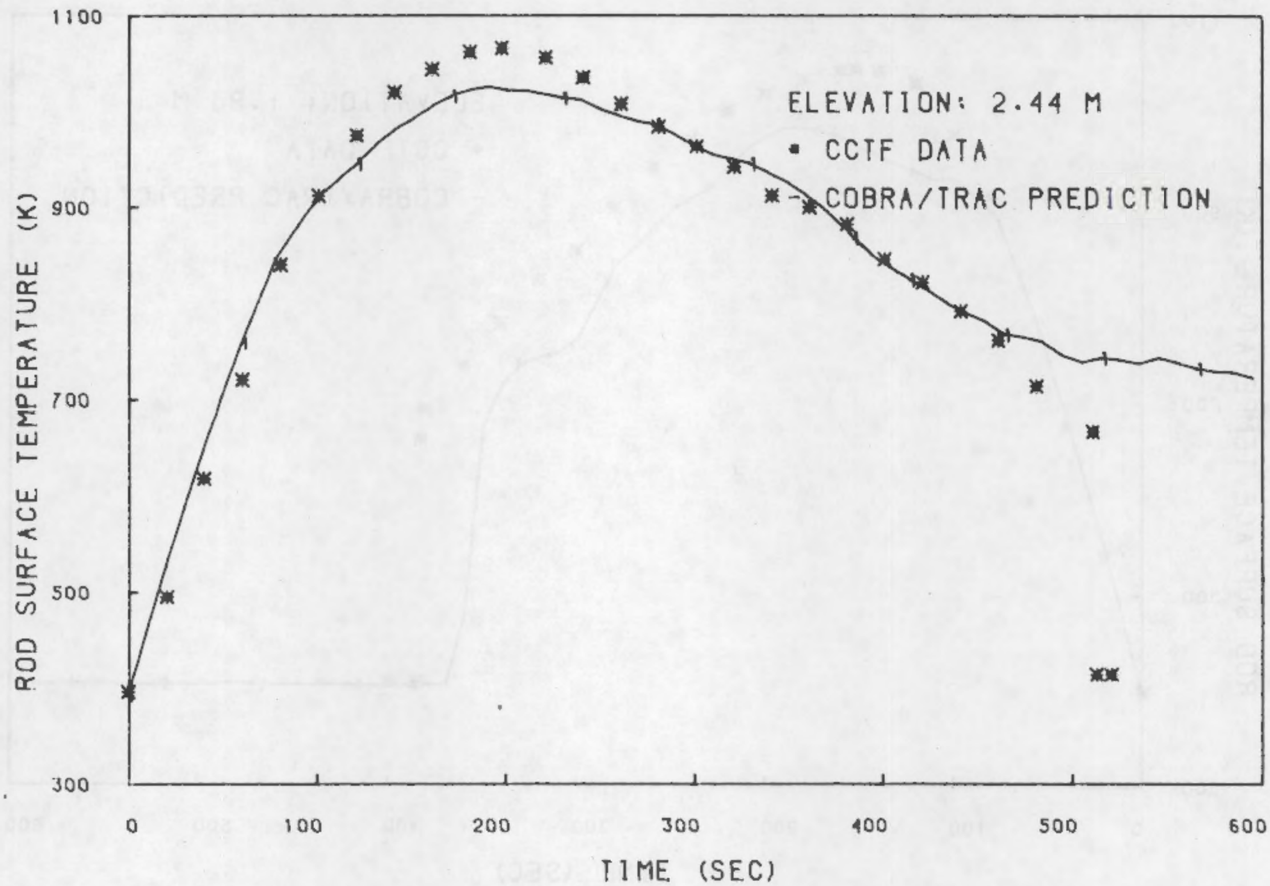

Figure 4. CCTF Test C1-2 Heater Rod Temperature at 2.44-m Elevation

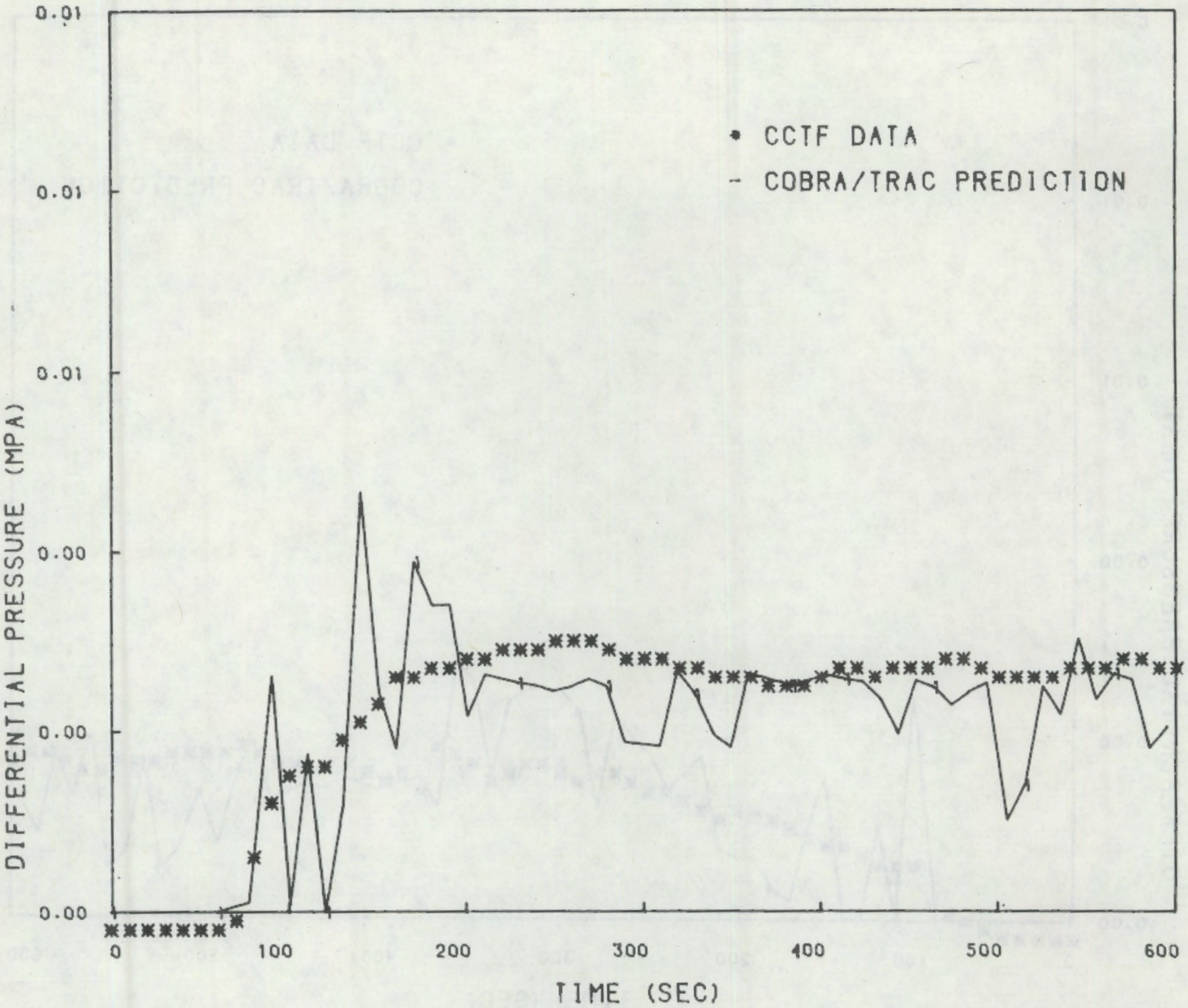

Figure 5. CCTF Test C1-2 Differential Pressure Between 2- and 4-ft Elevation in the Core

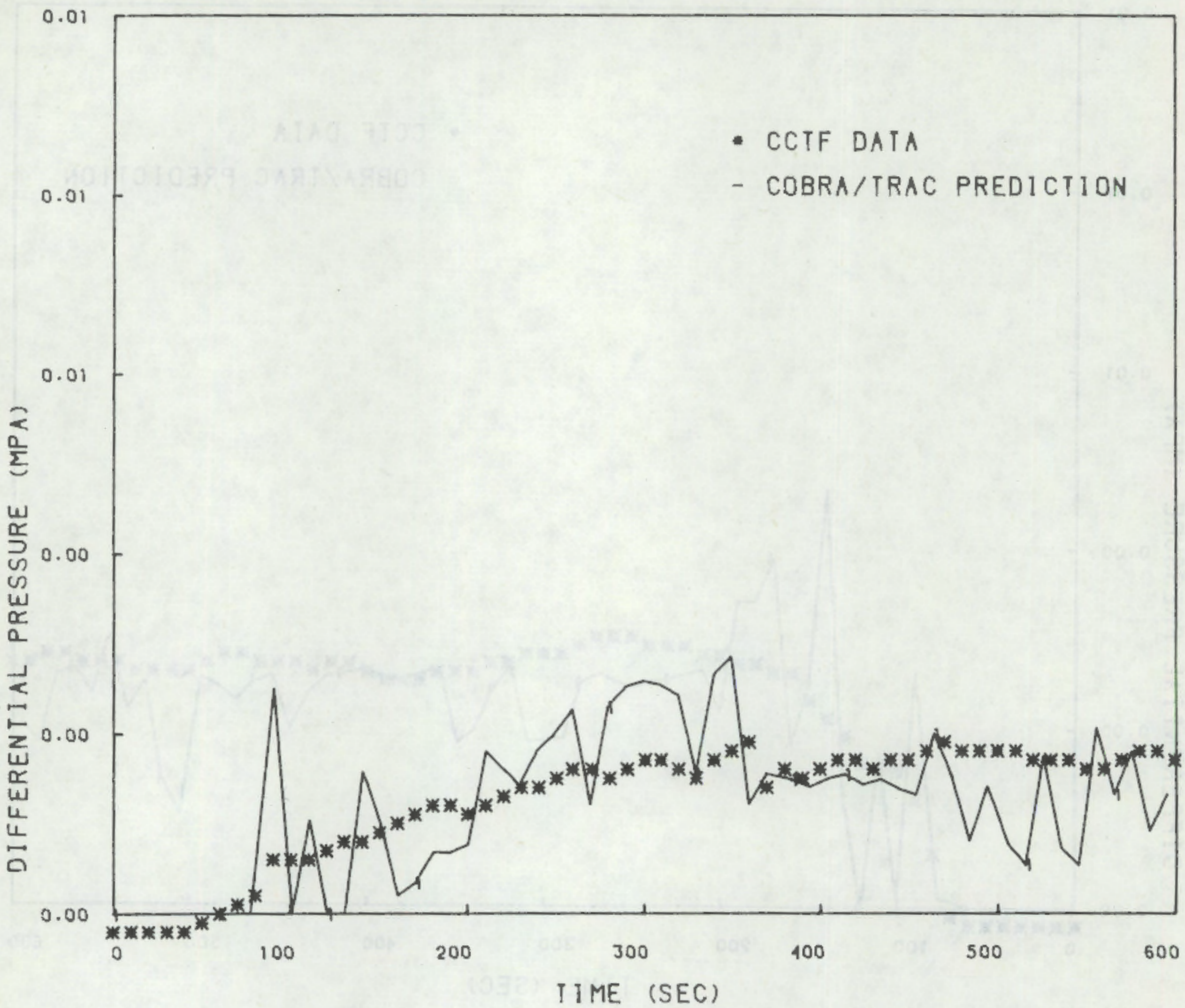

Figure 6. CCTF Test C1-2 Differential Pressure Between 4- and 6-ft Elevation in the Core

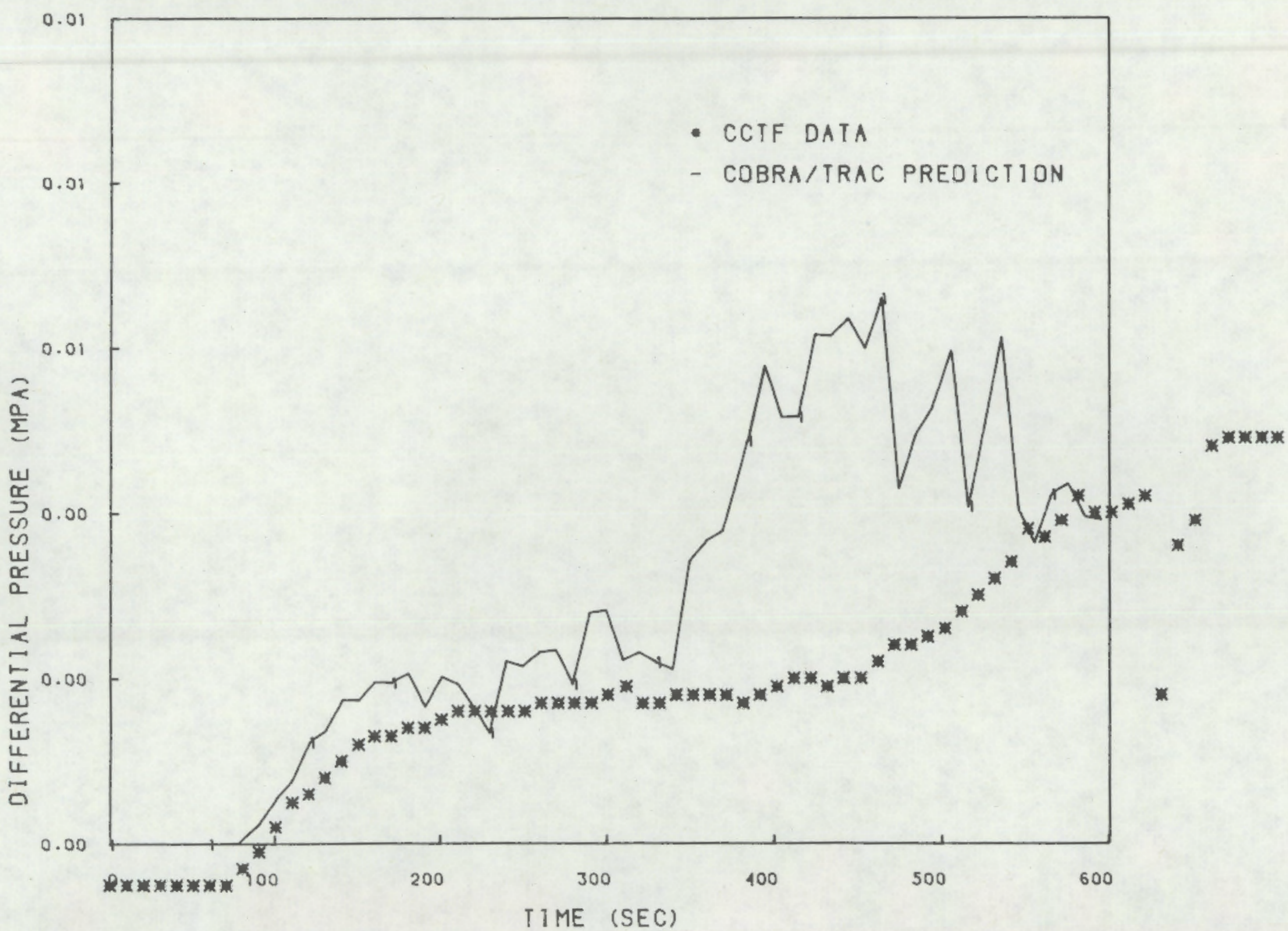

Figure 7. CCTF Test C1-2 Differential Pressure in the Upper Plenum

LOCA SIMULATION IN NRU[a]

C. L. Mohr, Project Manager

J. P. Pilger, Assistant Project Manager
P. N. McDuffie, Project Administration

SUMMARY

The first Materials Test (MT-1) was conducted at Chalk River Nuclear Laboratories (CRNL) in early April 1981. Post-test examination, using the disassembly, examination, reassembly machine (DERM), indicated that 5 of the 11 pressurized fuel rods ruptured during the test sequence. Test results and video coverage of the test bundle were presented to the sponsor in early June.

A joint agreement was finalized with the British and the U.S. Nuclear Regulatory Commission (NRC) for a cooperative test designed to meet their needs. This NRC-UK test is tentatively scheduled for the September 1981 test window, using the modified NRC MT-3 test train.

The second Materials Test (MT-2) will use a new pressurized rod cruciform test bundle installed in the MT-1 test train. MT-2 is scheduled for the late July 1981 test window.

INTRODUCTION

The objective of the Pacific Northwest Laboratory (PNL) LOCA Simulation in NRU Program is to provide information on the heatup, reflood, and quench phases of a loss-of-coolant accident (LOCA). The tests are designed to give information on the quench-front velocities within a fuel bundle, the liquid entrainment [10 CFR 50, App. K (Sec. ID 2)], and the heat transfer coefficients [10 CFR 50, App. K (Sec. ID 5)] for full-length pressurized water reactor (PWR) fuel as a function of reflood rate and delay time before reflood starts. A total of six test trains will be prepared for the program. The first test train will be devoted to thermal-hydraulic behavior, and more than 25 tests are planned. The remaining five test trains will be used for only one test each. These test trains will have prepressurized fuel rods; and, as a result, the rods will deform and rupture during the test. These materials tests will evaluate the effects of ballooning and rupture on quench-front velocities and associated heat-transfer coefficients.

The test loop in the NRU reactor (Chalk River, Canada) will accommodate the 12-ft long, 32-pin bundle on a 6 x 6 array with the corner pins removed. The bundle design uses commercial enrichments, cladding dimensions, and grid spacers.

TECHNICAL PROGRESS

MT-1 was conducted at Chalk River on April 2, 1981. The following test conditions were selected:

- peak cladding temperature - 1600°F
- reflood delay time - 30 s
- reflood rate - 2-in./s.

(a) RSR FIN Budget No.: B2277; RSR Contact: R. Van Houten.

Post-test visual examinations using the DERM indicated that 5 of the 11 pressurized fuel rods ruptured and the remainder experienced varying degrees of ballooning. Single-rod profilometry was performed on this test bundle. The DERM proved to be an invaluable aid for test train disassembly and detailed examination.

Figures 1, 2, and 3 show the MT-1 test train after it was opened for inspection. MT-1 test results are included in the quick loop report, which has been released for publication.

The MT-2 test cruciform was completed and shipped to CRNL on May 29, 1981. This will be the first attempt to reconstitute a previously used test train shroud and guard rod assembly with a new test cruciform. MT-2 is tentatively scheduled for the July 20-24, 1981, test window.

A joint agreement was finalized with the British and the NRC for a cooperative test and is tentatively scheduled for the September 1981 test window. The MT-3 test train will be modified to meet British instrumentation requirements for this test. Test conditions will be specified by the British.

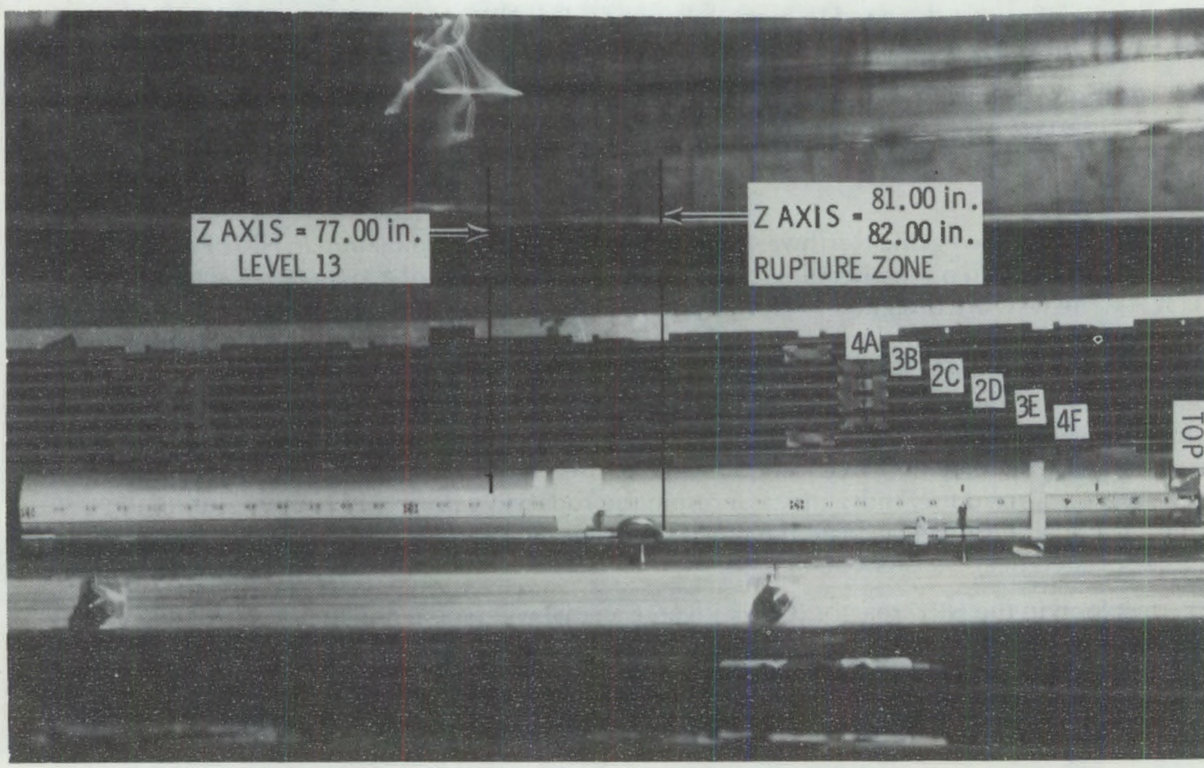

Figure 1. Center Third of Test Fuel Rod Bundle and Rupture Region

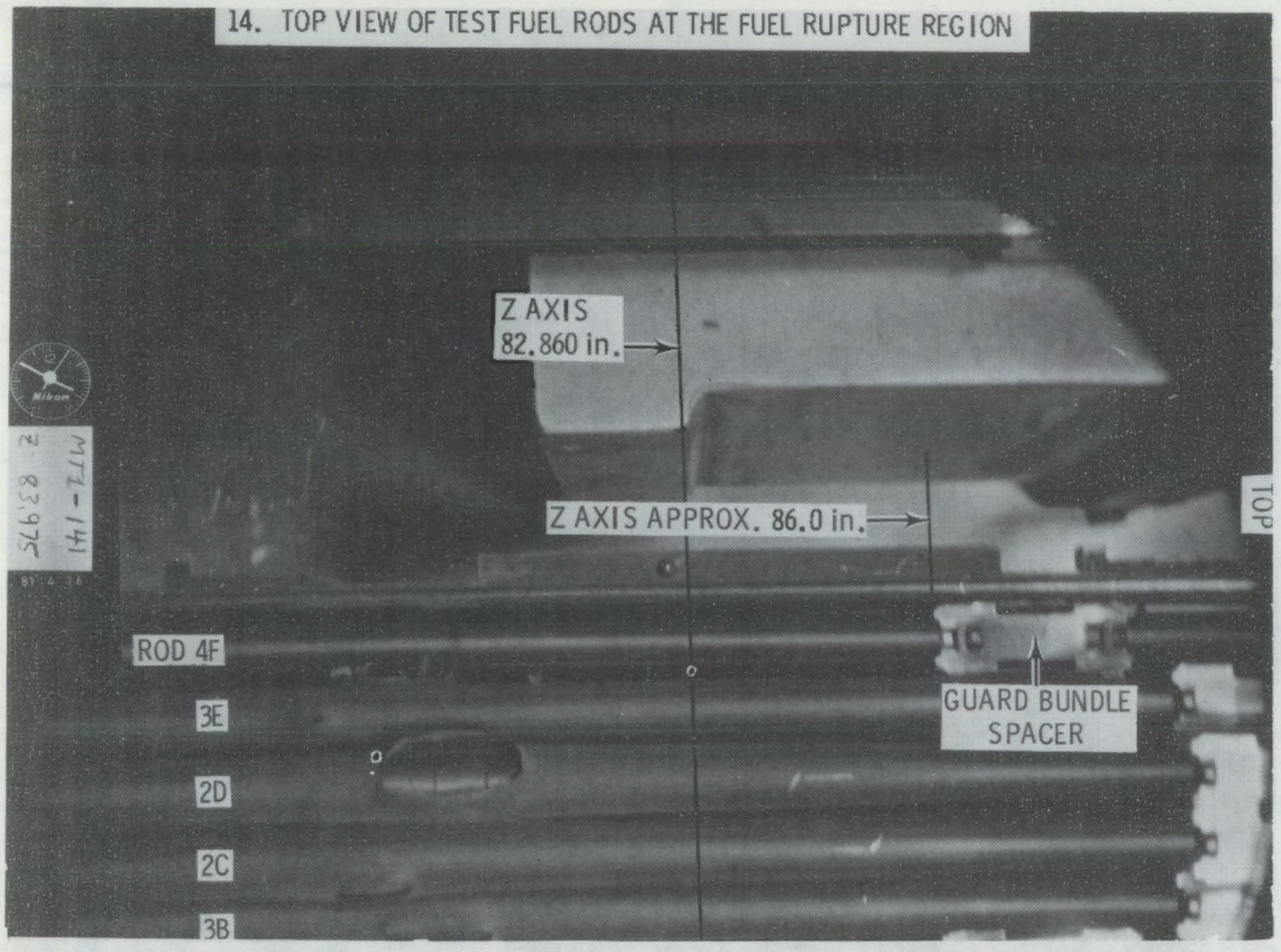

Figure 2. Top View of Test Fuel Rods at the Fuel Rupture Region

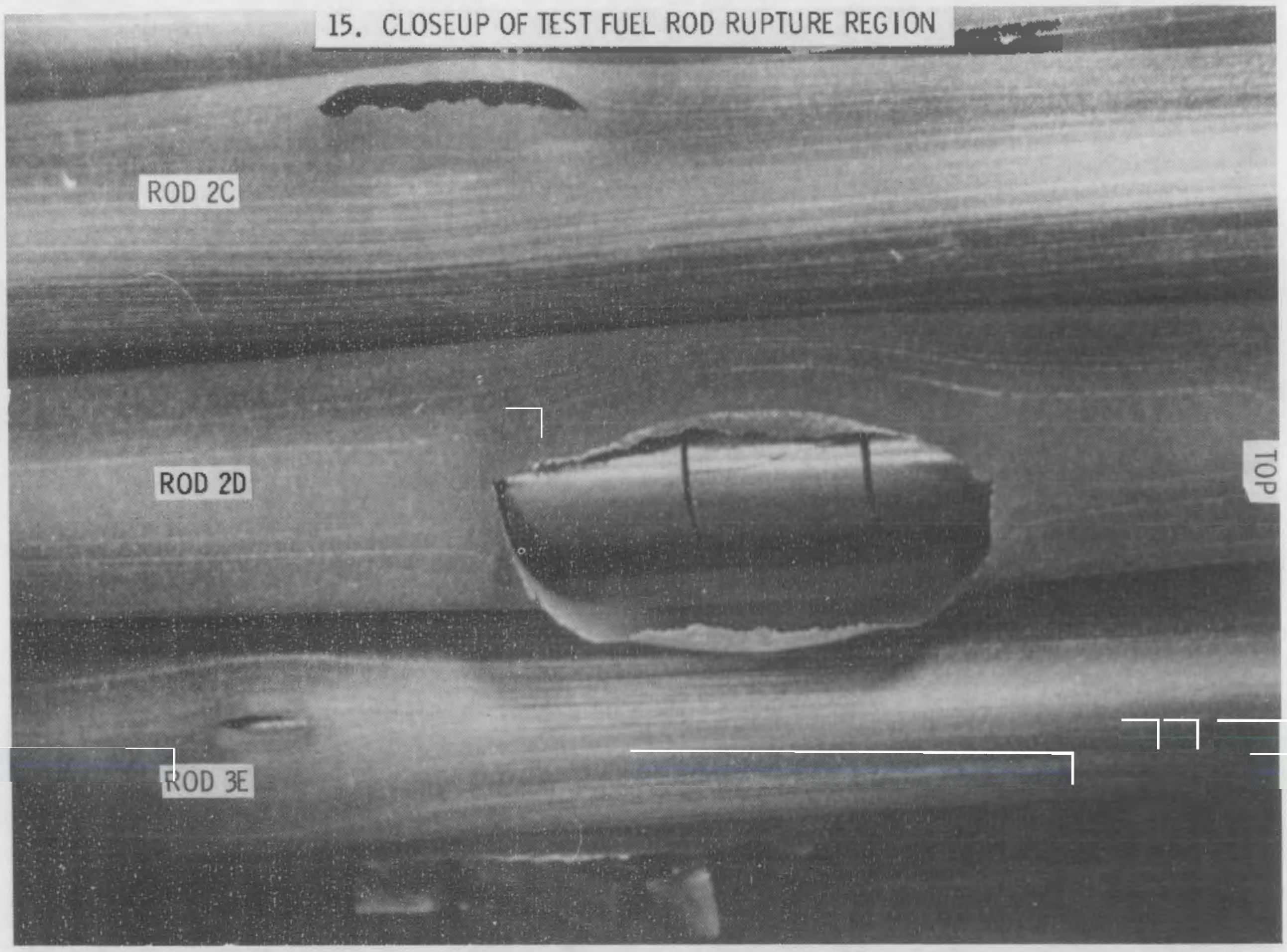

Figure 3. Close-up of Test Fuel Rod Rupture Region

STEAM GENERATOR TUBE INTEGRITY[a]

R. A. Clark, Project Manager
V. F. FitzPatrick, Deputy Project Manager

J. M. Alzheimer G. H. Lyon
A. S. Birks C. J. Morris
R. L. Burr M. E. Olson
P. G. Doctor E. B. Schwenk
G. R. Hoenes K. R. Wheeler

SUMMARY

The Steam Generator Examination Facility (SGEF) is 68% complete; construction remains six weeks ahead of schedule. Efforts to establish a multisponsored Steam Generator Group Project included technical marketing presentations in Europe and Japan. Programmatic tasks that precede research on the Surry generator continued per plan, the data acquisition task has completed the design and procurement of equipment, and a radiation training program and personnel monitoring system is being established under the health physics task. In addition, Surry generator research tasks on tube unplugging, baseline primary side nondestructive characterization, the nondestructive testing (NDT) round robin, and secondary side access were initiated this quarter. Research continued on stress corrosion crack (SCC) characterization and leak rate determinations.

INTRODUCTION

The Steam Generator Tube Integrity Program (SGTIP) is a multiphase, multitask laboratory program conducted at Pacific Northwest Laboratory (PNL). The principal objective is to provide the U.S. Nuclear Regulatory Commission (NRC) with validated information on the remaining integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. An additional objective is to evaluate nondestructive instrumentation/techniques with which to examine defects in piping or tubing that serves as the reactor primary system pressure boundary.

Initial program tasks included producing a matrix of steam generator tube specimens with mechanically or chemically induced flaws that simulated defects found in nuclear steam generator service. These flawed specimens are then fully nondestructively characterized by means of positive replication and various NDT techniques, mainly eddy current testing. The tube specimens are next tested to failure at PWR steam generator operating temperatures. The failure strength, actual flaw dimensions, and NDT-indicated flaw dimensions are used to derive mathematical relationships; and these relationships are plotted to provide, within a statistical certainty band, the remaining mechanical integrity of a steam generator tube as a function of its flaw type and size as indicated by eddy current testing.

Early work showed that conventional, single-frequency, eddy current evaluation of steam generator tubes as used for in-service inspections (ISIs) could be improved. Thus, program efforts were expanded to include new eddy current measurement techniques, the effects of different calibration standards, and a more complex statistical analysis of NDT data.

(a) RSR FIN Budget No.: B2097; RSR Contact: J. Muscara.

The first two phases of the program involved the study of mechanically (Phase I) and chemically (Phase II) defected tubing. Phase III of the original program included correlating the mathematical models developed in Phases I and II with actual service-flawed tubing. However, a lack of suitable specimens led to the redirection of Phase III into an effort to conduct extensive nondestructive and destructive evaluations on a retired-from-service nuclear steam generator. A generator[(a)] removed from the Surry II nuclear plant (Surry, Virginia) after six years of service was judged suitable for this research.

Initial efforts on the Surry generator were concerned with licensing and transport activities to bring the unit from Virginia to Hanford, Washington. The generator is now at Hanford and will remain on a storage pad until a specially designed containment facility (the SGEF) is completed. The SGEF will be equipped to allow both nondestructive examination (NDE) and physical sectioning of the generator. Capabilities to perform chemical cleaning and decontamination will also be included.

Research efforts on the Surry IIA generator will be initiated shortly after the generator is placed in the SGEF, which is scheduled for completion in November 1981. NRC-sponsored research will emphasize the following areas:

- validation studies of primary side NDT techniques and instrumentation
- verification of remaining integrity of service-defected steam generator tubes
- assessment of the secondary support structure integrity
- health physics - ALARA Control of Radiation Exposure associated with maintenance, repair, and ISI procedures
- defect matrix profiling and identification
- long-term operating effects of secondary side cleaning and primary side decontamination
- nondestructive ISI of the secondary side.

The generator will also become a source of specimens with service-induced flaws for various NRC programs.

At NRC's request PNL is conducting a marketing effort to broaden the sponsorship of the program. Potential exists for research and development in chemical cleaning, decontamination, corrosion product identification, corrosion mechanism studies, and materials recovery under alternate sponsors.

TECHNICAL PROGRESS

The following paragraphs detail progress of program tasks active this past quarter.

SURRY GENERATOR PROJECT

The Surry IIA generator remains at an access-controlled interim storage site on the Hanford Reservation. The generator is maintained under an argon gas purge that keeps it at 1/2-psi positive pressure relative to the surrounding atmosphere.

(a) The Surry II generators were among the first removed from service in the United States; they contain evidence of most of the degradation mechanisms identified in steam generators and have features that are common to many similar units.

Steam Generator Examination Facility

Figures 1 and 2 illustrate the construction status of the SGEF as of the end of this reporting period; construction is 68% complete and six weeks ahead of schedule. Current efforts include inside finishing work, wiring, and ventilation system completion. The building is structurally complete, the plumbing is finished, and most ventilation sheet metal is in place. An Operational Readiness Plan has been drafted for the SGEF and is currently undergoing comments. Other documents in progress include a building emergency plan, fire plan, and preventative maintenance plan. No net cost affecting change orders or work delays has occurred.

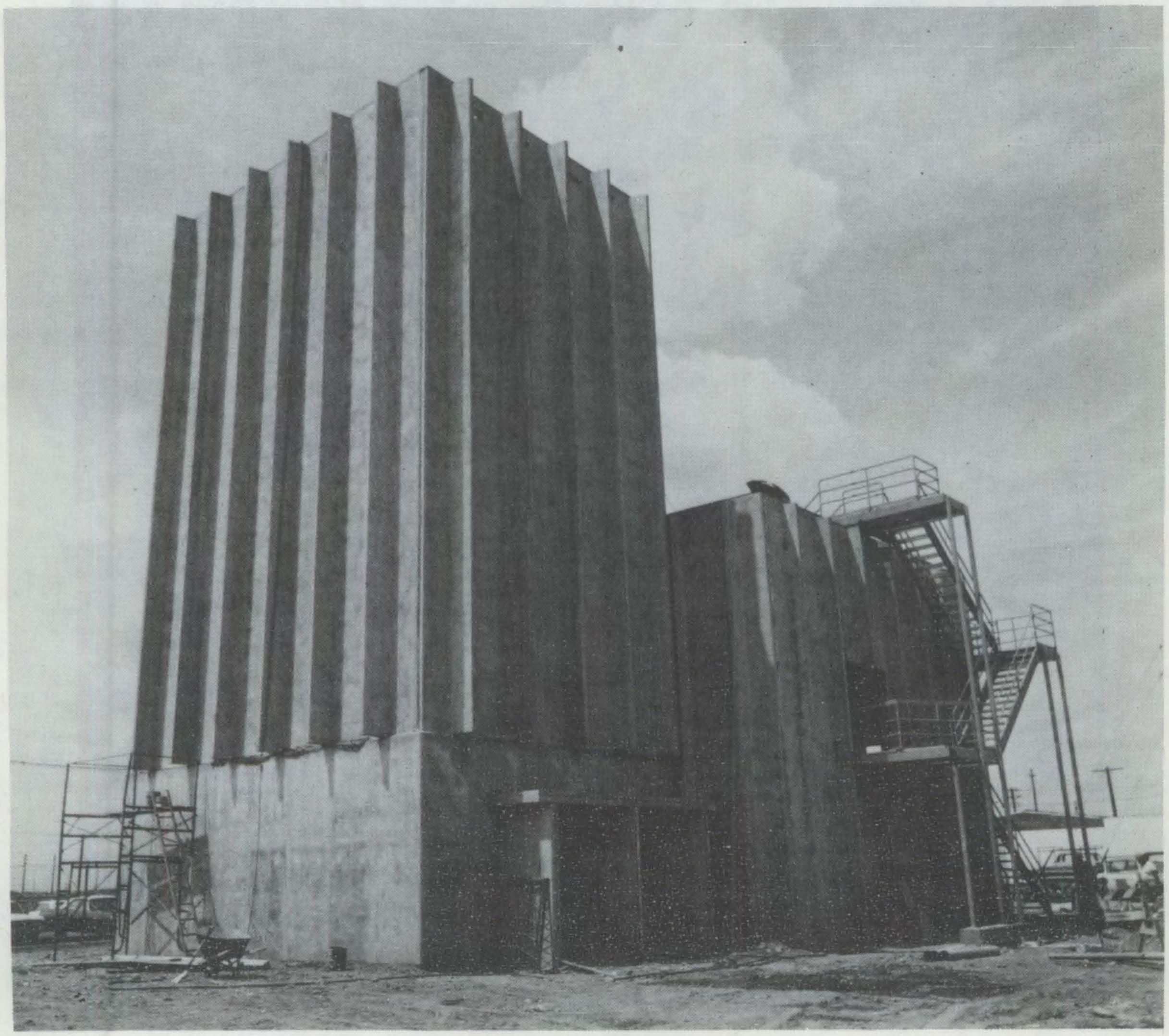

Figure 1. Steam Generator Examination Construction Status as of June 1981

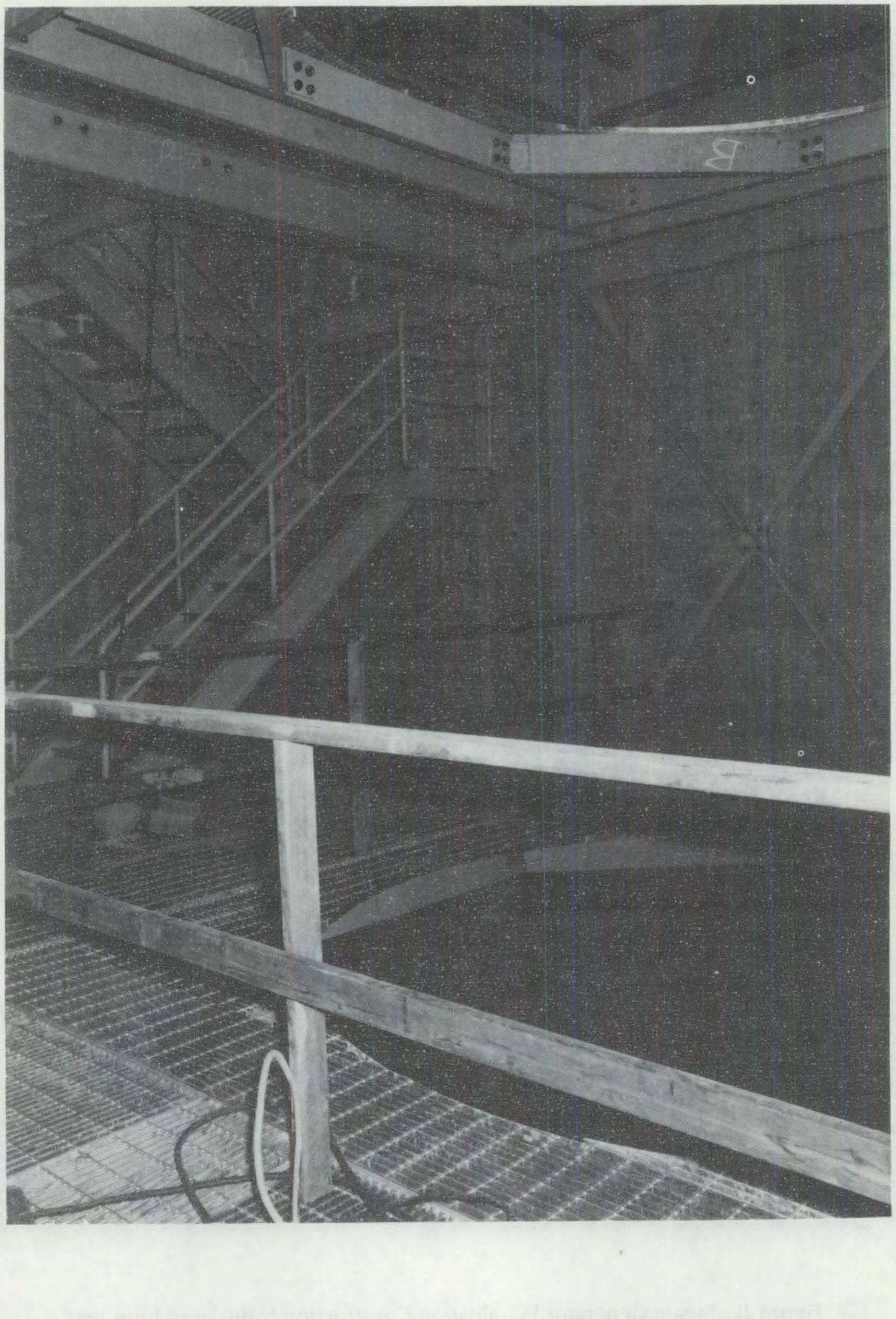

Figure 2. Inside of Steam Generator Examination Facility Tower Showing Gratings and Space for Steam Generator

Generator Research Tasks

The data management system task will provide computer software and arrange for use of computer hardware to handle information derived from research on the Surry generator. Figure 3 illustrates the data management system. Analog signals will be transmitted by underground cable approximately 500 ft from the SGEF to a computer facility in the 314 building. Analog data will be stored on tape and will also be digitized using a PDP 11/44 computer system, which will also control the NDT probe pusher-puller. A pusher-puller will be modified to automatically provide probe position information into the computer along with NDT signals. Digitized data will be stored on discs, which will be transferred to a VAX 11/780 system that has software for analyzing large data sets. Digitized data will also be archived. Both computer systems will allow terminal data input of visual observations, analytical results, etc. A graphics capability is also being provided to plot generator cross sections and show summary information about the cross section. All computer hardware is installed with the exception of an 8-track drive and a disc drive that have not yet arrived. Software for performing analytical operations and graphics is being developed.

This task has also provided capability to digitize the historical ISI eddy current tapes provided by VEPCO. The data will be utilized during the coming quarter to test the computer systems. Arrangements have also been made to acquire other historical service data on the Surry IIA steam generator.

The health physics task provides procedures for personnel exposure monitoring, control, and documentation. Health physics research activities will include radiologic mapping, determination of the effectiveness of decontamination efforts, and evaluation of waste and waste disposal problems associated with various operations. Efforts this quarter included preparing portions of the SGEF Operational Readiness Plan involving operations in a radiation zone, identifying and initiating purchase of monitoring instrumentation for the SGEF, and acquiring components for radiologic mapping activities. Work also continued on the computer-based dosimetry records system and on a training program for SGEF personnel.

Work was initiated on task 7—baseline eddy current inspection. In conjunction with this effort, task 8—tube unplugging—and task 9—NDT round robin—were begun early. All three tasks are in a detailed planning stage. Efforts include prebid costing of subcontracted services (tube unplugging, baseline ISI) and drafting of detailed task milestone schedules and costs. The baseline eddy current examination (task 7) will involve 100% inspection of all unplugged steam generator tubes. At least two eddy current systems will be used: a single frequency system identical to that used in the last Surry ISI and at least one widely accepted multifrequency system. In addition, a baseline tube profilometry measurement is under consideration. Task 8 efforts (tube unplugging) have concentrated on costing/scheduling development for fiscal year (FY)-1982 budget purposes. Letters have been sent to several potential subcontractors to develop information on the costs associated with unplugging 200 tubes and all 748 plugged tubes. Consideration is being given to conducting unplugging operations before baseline ECT, which would avoid a segmented baseline NDT effort. Initial plans were to use early baseline data along with analysis of historical ISI data to unplug only selected tubes and then complete the baseline. This method would probably result in unplugging fewer tubes. The cost/time/radiation exposures are being compared for each scenario.

Efforts continued on task 5—reopening preshipment examination penetrations—which was initiated at the end of last quarter. This task will assess any physical or chemical changes in the generator between the preshipment examination and placing the generator in the SGEF at Hanford. The task naturally leads into task 10—secondary side access—which provides for secondary side nondestructive characterization. Task 5 is at a detailed planning stage with preparation of work statements for cutting operations. Task 10 was initiated six months ahead of schedule to provide additional planning time and optimal use of staff.

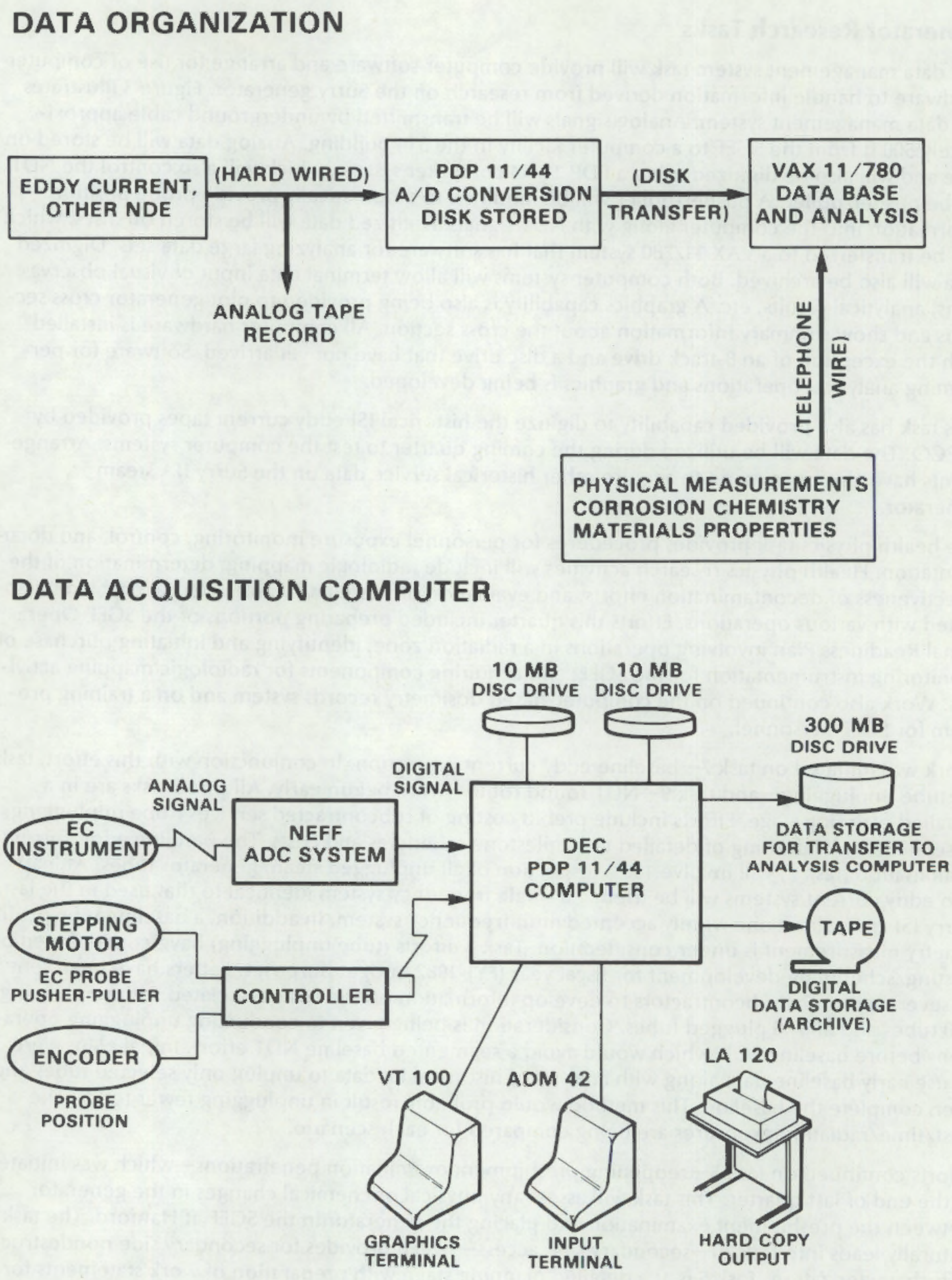

Figure 3. Data Acquisition/Retrieval Systems and Organization of Data Flow

A Commerce Business Daily announcement has been submitted for a subcontractor for task 2—positioning the generator in the SGEF. The subcontractor will move the Surry generator approximately 2 km from the interim storage site to the SGEF in November. A 30-m lift will then vertically position the generator in the SGEF through a removable roof panel.

Management Activities

Efforts continued to broaden sponsorship and technical support of research on the Surry IIA steam generator via a Steam Generator Group Project. Five presentations were made in Europe to parties representing the French government, nuclear vendor, and utility; the Italian government and vendor; the Swiss government, vendor, and utility; German vendors and utilities; and the British utility. In addition, presentations were made to Japanese vendors and utilities. Responses to program participation are scheduled for the coming quarter.

Prior to the foreign marketing trips, a contract was established at NRC with PNL inputs that provides the conditions for sponsors joining the Surry generator program. Definitions were established on program control/management, funding mechanisms, and opportunities for participation by nongovernment sponsors.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

Stress Corrosion Cracking Characterization

Efforts are continuing at definitive nondestructive characterization of laboratory-produced intergranular stress corrosion cracking (IGSCC) in Inconel 600 steam generator tubes. A group of 10 round robin specimens was nondestructively characterized by the ConAm Division of Nuclear Engineering Services and by Adaptronics. Confirmatory destructive assay of these specimens has been delayed due to interest by additional parties in participating in the round robin. Destructive assay is now planned for the current quarter.

Leak Rate Tests

Production of the matrix of test specimens with through-wall SCCs of various lengths and orientations continues. We have encountered difficulties in fabricating long through-wall and circumferential cracks. Leak rate tests on available specimens will be conducted during July. All necessary equipment is available onsite.

MILESTONES

- Baseline eddy current task initiated.
- Tube unplugging task initiated.
- NDT round robin task initiated.
- Secondary side access task initiated.

PROBLEMS

There remains a delay in publishing reports due to technical staff manpower limitations. Difficulties in fabrication of leak rate specimens will delay completion of this test matrix. The decontaminate channel head task (task 6) is not making reasonable progress. Staff changes are being evaluated to correct this situation.

FUTURE WORK

During the coming quarter the following activities will be pursued:

- continued marketing of a Steam Generator Group Project
- establish contracts with Group Project participations
- conduct destructive assay of round robin SCC specimens
- conduct leak rate tests
- establish a subcontract for placing the Surry generator into the SGEF
- establish subcontract(s) for channel head decontamination.

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16. ABSTRACT This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 1981, for the Division of Reactor Safety Research within the U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL). These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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