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ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from October 1 through December 31, 1980, 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 structural graphite strength, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the remaining integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Test 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 and postaccident coolability tests for the ESSOR Test Reactor 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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LOCA SIMULATION IN NRU

GRAPHITE NONDESTRUCTIVE TESTING[a]

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SUMMARY

Our work during this quarter centered on refinement of a computer algorithm for estimating near-surface conductivity profiles from eddy current data. A progress report covering our work in fiscal year (FY)-1980 is being prepared.

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 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.

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

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

Our new computer program for estimating material properties versus depth (ZFIT) is operational on the Brookhaven CDC-7600 computer.

EDDY CURRENT TESTING

The analog computer methods used for data handling in a variety of other multifrequency eddy current tests are not suitable for determining the oxidation profile. We found that a high degree of sophistication is required of a digital computer program for estimating the oxidation profile from multifrequency eddy current data. As a result, an algorithm has been developed for use on a large digital computer.

A new computer program (ZFIT) has been written and is now operational on the Brookhaven CDC-7600 computer. ZFIT combines two programs: the parameter fitting routine MINUIT and the program EDDY. EDDY is based on theory and codes developed by C. V. Dodd and coworkers at Oak Ridge National Laboratory (ORNL); it calculates resistance and reactance of a search coil located above a conducting plane when coil geometry, driving frequency, conductivities, layer thicknesses, and permeabilities of layers in the conducting plane are specified.

ZFIT is essentially an inversion of the EDDY program; it searches for values on a specified list of graphite parameters that will fit a set of multifrequency coil impedance measurements. Work is proceeding to verify that ZFIT will reliably estimate near-surface oxidation profiles and to determine the simplest form of the algorithm that could be implemented on field-portable equipment.

FUTURE WORK

EDDY CURRENT TESTING

- complete development of the profiling algorithm
- test the algorithm using large oxidized samples.

ULTRASONIC TESTING

- optimize the design of the ultrasonic sensors to be used for backscattering measurements
- evaluate the ability of ultrasonic backscattering to accurately quantify graphite oxidation.

OXIDATION PROFILES

- determine elemental distributions in the ash samples.

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

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SUMMARY

During this quarter the problem of arranging for fabrication of the A533B steel insert for the ZB-1 German vessel test was resolved. A contract was negotiated with Materialprufungsanstalt (MPA), Stuttgart, West Germany, to fabricate the insert from material supplied by Pacific Northwest Laboratory (PNL). The feasibility of electronically simulating reactor flow noise for the ZB-1 vessel test has also been demonstrated, and the approach for applying pattern recognition in the ZB-1 vessel test has been finalized. Computation at the test site will be limited to evaluating the effectiveness of the decision rule used; developmental analysis of the data will be done at PNL.

Official agreement has been received from the Tennessee Valley Authority (TVA) to participate in installation of an acoustic emission (AE) test system on one of their reactors.

Hardware portions of the AE monitor system for vessel test and reactor monitoring are essentially complete. Two major work items remain: a signal identification tagging method and interfacing the PDP 11/03 computer to the detection/source location system.

A specialized differential sensor/preamplifier subsystem is being evaluated for benefits in noise rejection.

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

Arrangements and a test plan are being finalized for a demonstration of weld flaw detection with AE for the German technical community. The test is scheduled for mid-February.

Further analysis is being performed in wave-guide waveform data derived from the cylindrical bend test to attempt to refine the algorithm planned for the ZB-1 vessel test.

INTRODUCTION

The purpose of this 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, sensor development, weld monitor demonstration, pattern recognition, and reports. The final section describes work planned for the next quarter.

TECHNICAL PROGRESS

OFF-REACTOR VESSEL TEST

The problem of arranging for fabrication of the A533B insert for the ZB-1 test vessel was resolved this quarter. A firm bid was received from MPA to perform the entire job for $55,000. This includes an initial stress relief heat treatment prior to machining, machining the insert to shape, machining three flaws in the plate surface, and fatigue precracking the machined flaws. The estimated completion time is 20 weeks. A block of A533B base material was shipped by air freight from PNL to the fabrication site in Dusseldorf, Germany, on December 30, 1980. At the present time, completion of the A533B insert is a critical path item in completing the test vessel. An alternative to precracking the machined flaws that would facilitate this step in the finished vessel is being considered, however; and if it is adopted, insert fabrication would no longer be a critical path item. The current estimate for completion of vessel test preparation is July 1981.

Test specimens (two compact tension and two plate tension) have been fabricated from degraded A508 steel supplied by MPA. Since the ZB-1 test vessel will contain an insert of this material, these specimens will be used to determine material AE characteristics.

Two approaches are being investigated to simulate noise in the ZB-1 vessel test: electronic and hydraulic. Feasibility of electronic simulation has been established. A broadband diode noise generator was used in conjunction with shaping filters to produce a good replica of measured reactor noise as seen by an AE sensor. The power input to the transmitting transducer was substantially less than anticipated. Noise simulation by hydraulic means, however, is still unresolved. An initial test that involved throttling water flow through a 12-in. outside diameter (OD) by 2-in. wall by 36-in. long autoclave vessel using an existing globe valve was not successful. A gate valve is being installed in the supply line to determine if it will produce cavitation more effectively during throttling of the flow.

The approach to be used in applying pattern recognition to the ZB-1 vessel test has been finalized. The criteria for constructing the vessel test decision rule assume that there will be many fewer flaw signals compared to innocuous noise in an operating reactor environment. Therefore, it is important to identify all of the flaw signals at the expense of misclassifying a higher percentage of the noise signals. An overall success rate of 75% would be acceptable if close to 100% of the flaw AE were accepted. The noise signals could then be screened out by other means, such as source location. The wave-guide will be the primary sensor used in the test.

Analysis of the composite data has shown that the decision rule should be "calibrated" on data from the sensor that will be used. Therefore, the parameters for the decision function and decision rule will be fit from the available wave-guide data. A block diagram of the pattern recognition implementation method is shown in Figure 1. During vessel testing, there will be neither time nor core space on the computer to do a fundamental pattern recognition analysis of the data (that is, to test features and develop decision rule improvements). Development analyses will be done at PNL. Only the computations needed to evaluate the decision rule in making the flaw growth AE-noise discrimination will be done onsite.

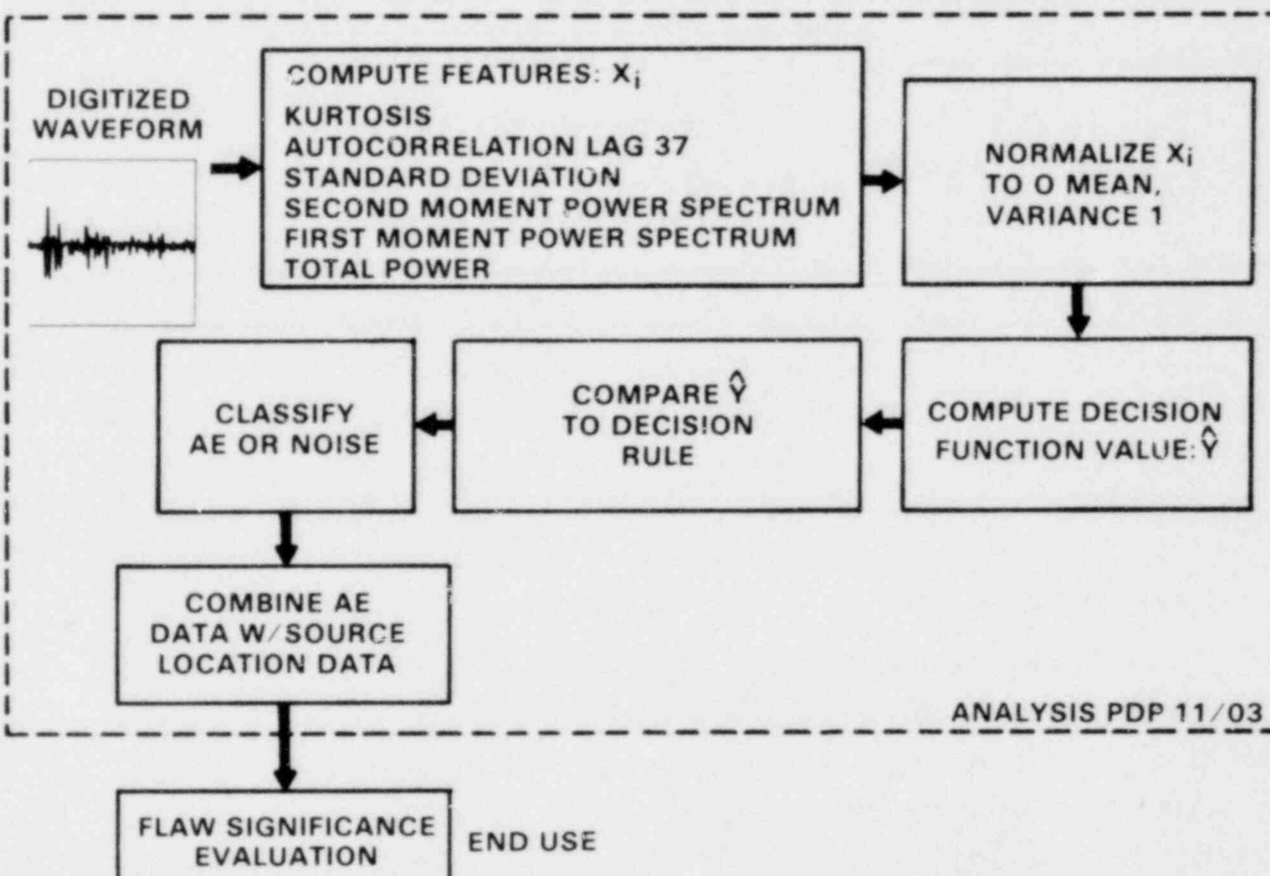

FIGURE 1. Pattern Recognition Implementation

The PNL project manager visited MPA on December 9, 1980, to review the test program details and schedule. The test schedule, insert fabrication, cycle rate, test plan, and flaw precracking method were clarified.

REACTOR INSTALLATION

In preparation for the reactor monitoring phase of the program, discussions have been held with TVA personnel at Knoxville, Tennessee, concerning installation of sensors, preamplifiers, and signal leads out of containment on a TVA reactor. Official agreement has been received from TVA to participate in this effort, and a meeting will be held in January 1981 to finalize details. The current plan is to install the system on the Watts Bar Unit 1 reactor, which is scheduled for full-power operation in late calendar year (CY)-1982. Installation must be completed in April 1981.

AE MONITOR SYSTEM

Fabrication of the AE monitor system for test vessel monitoring and subsequent reactor testing continued this quarter. Work concentrated on the waveform recording subsystem during this period. The waveform recorder controller along with the Biomation 8100 transient analyzer, the Kennedy 9-track magnetic tape recorder, and the video terminal are presently being mounted in a 5x3x2-ft instrument case. All units will be protected by shock mounts and front/rear doors, and the case can be shipped without additional packaging. The waveform recorder can extract the 2K bytes of data from the Biomation 8100 in 4 milliseconds (msec). Another msec is used to obtain the following external parameters for the header: total count, valid count, load position, maximum load, minimum load, load frequency, cycle count, and tag count from the Dunegan/Endevco (D/E) system. Each waveform will have an associated header written on tape with it, and it will take approximately 110 msec to write the header and waveform data to tape. Thus, the maximum waveform recording rate is about 8 signals per second, and each tape will hold around 1200 signals. The microprocessor-based controller has over 32K bytes of buffer memory for stacking the data if the acquisition rate exceeds the record rate of the

Kennedy; this will provide for stacking of 16 signals. Tapes have been written by the waveform recorder system and successfully read and decoded by the PDP 11/03 system.

The following primary items remain to complete the system:

- incorporation of a feature to identification tag each signal as received by the D/E detection/source location system. The same tag will be attached to the recorded waveforms for correlation of all data in analysis. PNL is collaborating with D/E on this item.
- interface of the PDP 11/03 computer to the D/E system for total data analysis.

SENSOR SYSTEM IMPROVEMENTS

A task has been initiated to improve the noise rejection capabilities of the AE sensor/preamplifier subsystem to be used in vessel test and reactor monitoring. Two noise sources commonly encountered in field applications are electromagnetic interference and ground loops. The high common-mode rejection ratio provided by state-of-the-art differential instrument amplifiers coupled with judicious sensor design should provide an improved degree of noise immunity.

Two differential sensors have been designed for evaluation. Figure 2 shows the first of these sensors. Acoustically, this sensor is similar to the wave-guide sensors used in the past; however, some mechanical and electrical improvements have been incorporated. As in the past, a welding rod acts as a wave-guide providing acoustic coupling to test the specimen. This wave-guide is electrically isolated from the sensor housing by means of a Macor® bushing. (Macor is a machinable ceramic material.) An alumina disk (silicon dioxide) that has been silvered on both sides provides an acoustic buffer and also allows for electrical connection. The sensing element in this design is a piezoelectric disk of PZT-5 (0.140 in. diameter). The positive signal lead is soldered directly to the top surface of the PZT-5, and the negative signal lead

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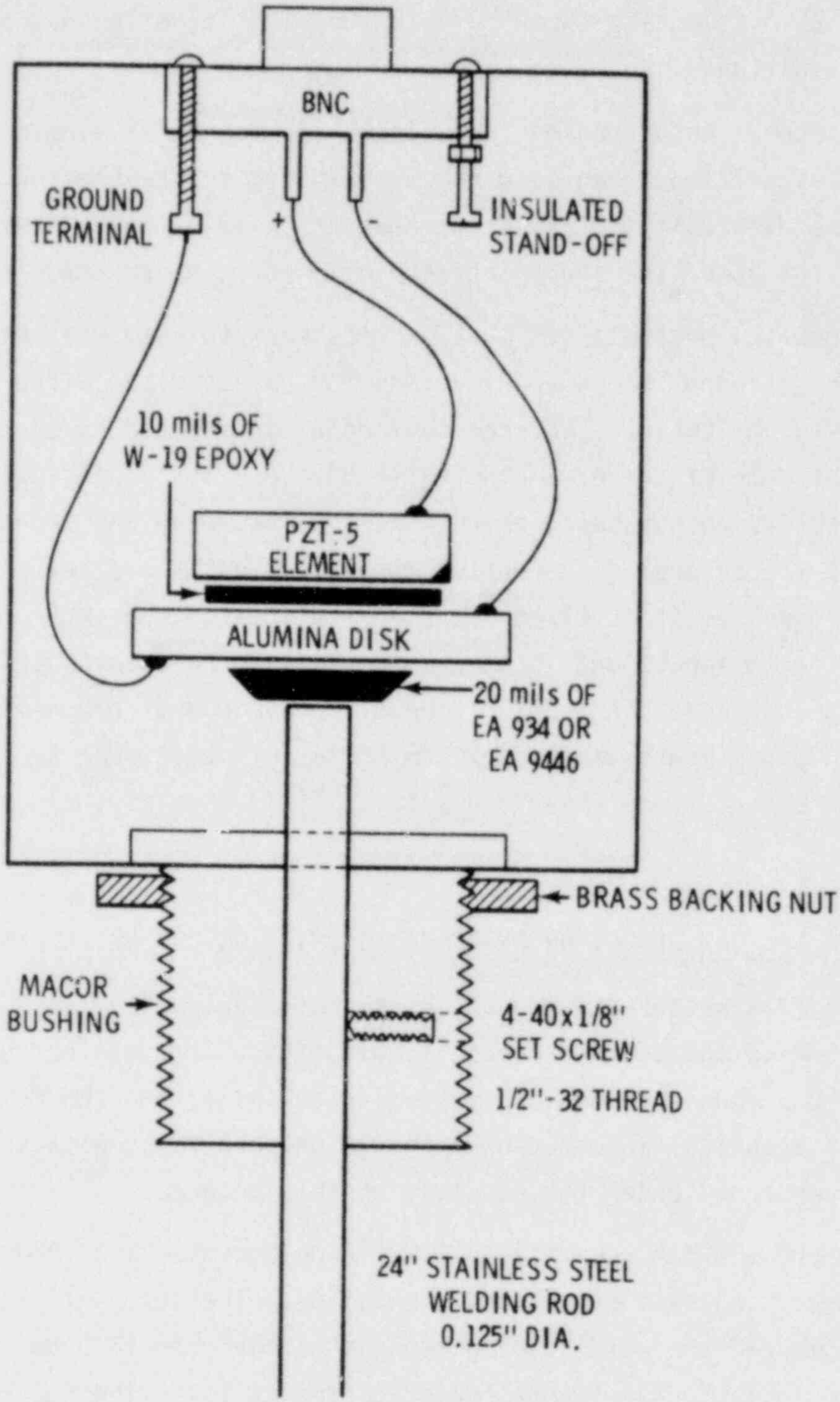

FIGURE 2. Wave-Guide Acoustic Emission Sensor for Differential Operation--Capacitive Connection

is soldered to the alumina disk and relies upon capacitive coupling through the W-19 bond line for signal pickup. One sensor of this design has been fabricated for evaluation.

In the second sensor design (see Figure 3), the PZT-5 element is chamfered so that both signal leads can be directly soldered to the element. This sensor is more difficult to fabricate; however, elimination of the reactive impedance of the bond line should result in improved noise immunity.

A differential preamplifier that is necessary to complete the subsystem has also been designed; the specifications for an optimized differential amplifier are listed in Table 1. Several commercial integrated circuits are available that meet some of these requirements; however, it is difficult to find an amplifier with 5-V output swing as well as the low noise and broad bandwidth required. Of the several differential amplifiers built and tested, the most practical design (see Table 1) consists of a discrete transistor differential pair with buffered inputs and integrated circuit buffered outputs. This circuit, although not ideal in terms of input impedance and common-mode rejection, should show a significant improvement in noise rejection over the present single-ended sensor/preamplifier subsystem. Evaluation of the new concept is in progress.

DEMONSTRATION OF FLAW DETECTION BY ACOUSTIC EMISSION DURING WELDING

The weld flaw AE detection/analysis system developed by GARD, Inc., will be demonstrated to the German technical community. The development was sponsored by the U.S. Nuclear Regulatory Commission (NRC), and the demonstration comes under a technical information exchange understanding with West Germany and will be performed under the auspices of this program.

The PNL project manager visited IZFP[a] on December 10, 1980, to resolve questions concerning test format and scheduling. The test weld will be a full thickness submerged arc weld joining two plates that are 1350 mm (53 in.) long by 250 mm (10 in.) thick. Twenty isolated defects involving four defect types

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

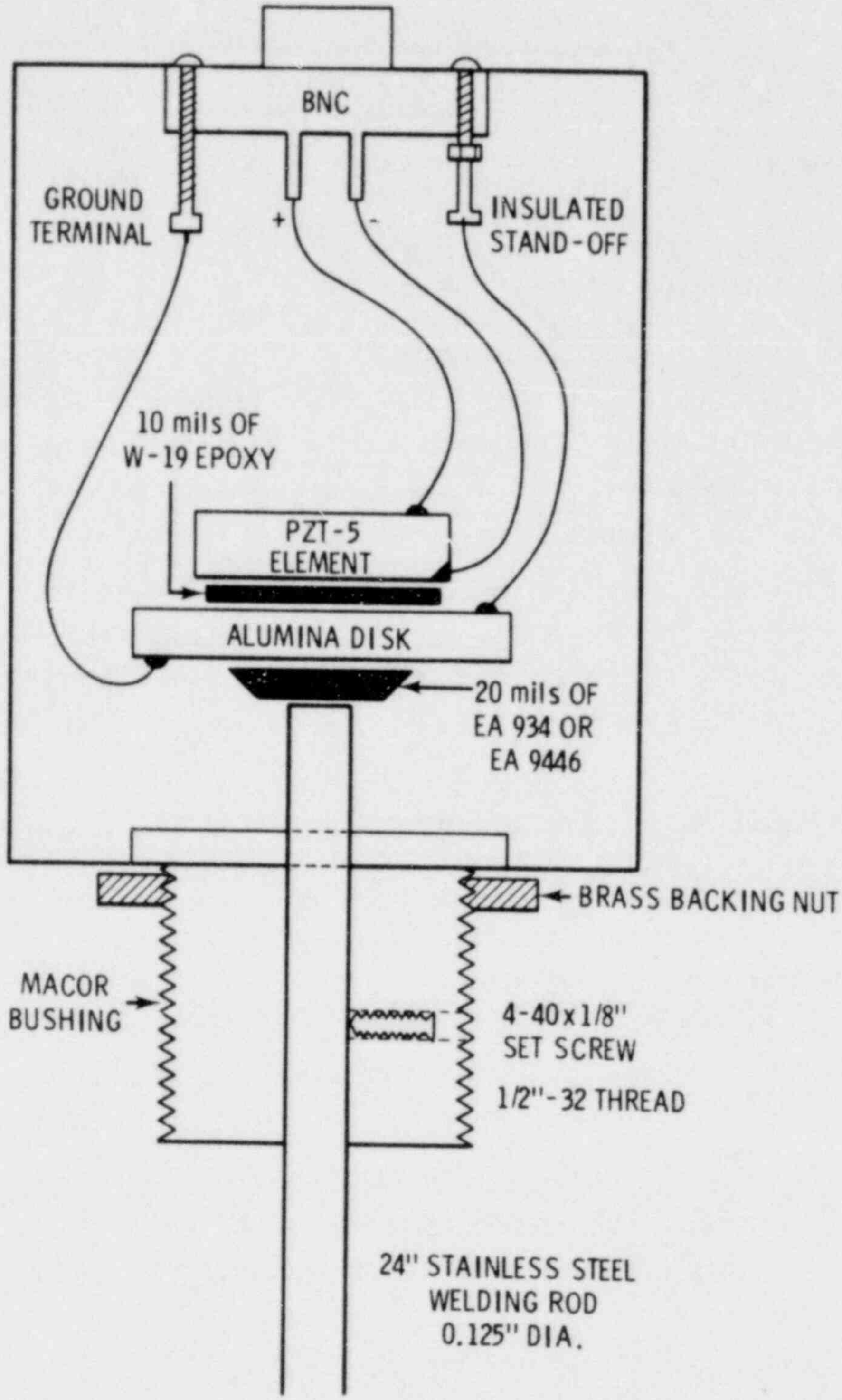

FIGURE 3. Wave-Guide Acoustic Emission Sensor for Differential Operation--Direct Connection

TABLE 1. Differential Amplifier Characteristics

Parameter	Optimal Design	As-Built Design
Gain	20 dB	20 dB
-3-dB Bandwidth	100 kHz to 2 MHz	40 kHz to 2.2 MHz
Output Swing	5.0 V into 50Ω	5.0 V into 50Ω
Input Noise	5.0 μVrms (50Ω)	5.0 μVrms (50Ω)
Input Impedance	>100K	19KΩ at 500 kHz
Common-Mode Rejection Ratio	-40 dB	-29 dB
Power Supply	+24 V	+24 V
Signal Cable	Integral	Integral

will be produced in the weld. AE monitoring will be done during welding and during the first approximately 24 hours of postweld cooling. By present plans, the full weld will be nondestructively inspected following the test. A 200-300 mm (8-12 in.) length of weld will be cut from one end for immediate destructive examination. These results will be used for correlation with AE results. The balance of the weld specimen will be used in the PISC II round robin weld inspection program, which will require about 3 years to complete.

The demonstration is scheduled for the week of February 9, 1981, at a fabrication plant near Dusseldorf, Germany. Negotiation of a subcontract with GARD, Inc., to perform the AE monitoring is in the final stage.

PATTERN RECOGNITION

The final set of wave-guide waveform data derived from the cylindrical bend test has been incorporated in computer storage. The data are being analyzed to look for any indicated refinements to the algorithm planned for the ZB-1 vessel test.

There are a total of 517 usable waveforms: 246 for valid AE and 271 for noise. The noise waveforms can be further broken down by source:

- c-clamp = 41
- tapping = 59
- background = 20
- rubbing = 60
- ultrasonic testing pulse = 13
- electrical = 78.

The following 11 features were generated:

- MN = mean
- SD = standard deviation
- TP2 = total power 200-400 KHz/total power 200-1000 KHz
- FM2 = first moment 200-400 KHz/total power 200-1000 KHz
- SM2 = second moment 200-400 KHz/total power 200-1000 KHz
- TP4 = total power 400-600 KHz/total power 200-1000 KHz
- FM4 = first moment 400-600 KHz/total power 200-1000 KHz
- SM4 = second moment 400-600 KHz/total power 200-1000 KHz
- TP6 = total power 600-800 KHz/total power 200-1000 KHz
- FM6 = first moment 600-800 KHz/total power 200-1000 KHz
- SM6 = second moment 600-800 KHz/total power 200-1000 KHz.

Variance weighting was used to reduce the number of features to seven for input into the decision rule. The seven features, in their order of importance, are SM4, SD, MN, SM2, TP2, TP6, and TP4.

The multilinear least-squares decision rule for categorized data was used to discriminate noise waveforms from valid AE for a training set only. The overall success rate was 79%, with 82% for valid AE and 77% for noise. These results are preliminary, and work is continuing to create training and test sets and to do more extensive feature analysis.

REPORTS

- quarterly progress report for the period July 1-September 30, 1980
- review report, "Acoustic Emission - Flaw Relationship for In-Service Monitoring of Nuclear Pressure Vessels," P. H. Hutton and R. J. Kurtz, October 1980, presented at the Eighth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland.

FUTURE WORK

Plans for the period January 1-March 31, 1981, include:

- fabrication and checkout of AE monitor system
- prepare a complete vessel test plan
- complete AE characterization of A508 German material
- start installation of AE sensing system on TVA reactor
- complete demonstration of weld monitoring with AE in Germany
- AE-monitor irradiated fracture specimen tests at the Naval Research Laboratory.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[a]

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SUMMARY

Primary emphasis during the past quarter was directed toward round robin sample preparation and characterization. Cracking of intergranular stress corrosion crack (IGSCC) samples remains as the major delay in the sample fabrication program; however, we expect that the round robin can be initiated during the coming quarter. Methods for characterizing ultrasonic instruments and search units have been developed and are being implemented. The influence of ultrasonic beam redirection and scattering in austenitic welds is under investigation, and preliminary results indicate that single-side access inspections of austenitic welds are ineffective.

INTRODUCTION

The primary piping systems 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 pipe joints.

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 in-service inspection

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

(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 piping 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 piping systems, and the results and recommendations are also applicable to Class II piping systems. Programs currently in progress concerning inspection reliability of pressure vessels are also being monitored and evaluated.

TECHNICAL PROGRESS

TASK 6 - SAMPLE FABRICATION

Four sample fabrication tasks are in progress. These tasks include thermal fatigue cracking of the 33-in. outside diameter (OD) clad ferritic pipe, 33-in. OD cast stainless steel (SS), 10-in. Schedule 80 SS, and IGSCC of 10-in. Schedule 80 SS pipe. These samples are to be used in the upcoming round robin.

Ferritic Pipe

Fabrication of the two 33-in. OD A106 Grade C welded pipe samples was completed in mid-October, and the samples were delivered to PNL. Thirteen of the 16 samples cut from these pipes have been cracked by the thermal fatigue process; the cracking process is expected to be completed by February 1, 1981.

Cast Stainless Steel

Thermal fatigue cracking of the cast SS samples was completed in late November 1980. However, at least two of the samples appear to be unsuitable due to insufficient crack depth. These samples will be subjected to additional thermal cycling to produce the desired crack depths.

10-in. Schedule 80 Stainless Steel

The thermal fatigue cracking process was completed on six 10-in. Schedule 80 SS pipes in December. However, it appears that many of the cracks are not as large as intended. We believe this condition resulted from a difference in residual stress patterns in these pipes as compared to those that were used for process development. It appears that it will require additional cycling to achieve the desired flaw depths in these samples.

IGSCC Samples

The round robin test matrix requires six 10-in. Schedule 80 pipe samples containing three to four isolated areas of IGSCC in each pipe. It has been our intention to grow these cracks by the graphite wool technique.[a] A sample was successfully cracked in early September 1980; however, attempts to crack the round robin samples have not been successful. These failures may be attributed to several problems, including: less than optimum control of the welding procedure, poor fit of the graphite wool, and a different material chemistry (although presumably more susceptible).

Three samples are currently in the autoclave and are scheduled to be completed by January 23, 1981. A decision will be made at that time as to whether further effort in this area is warranted. The IGSCC samples are the critical path item for beginning the round robin.

(a) The graphite wool technique was developed in Japan by Ishikwajima-Harima Heavy Industries. Transfer of this technology to the United States through PNL was sponsored by the Electric Power Research Institute (EPRI).

TASK 7 - MEASUREMENT AND EVALUATION

Substantial progress has been made toward the development of an instrument and search unit characterization system and initiation of measurements to determine the impact of single-sided access for austenitic welds.

Ultrasonic Instrument Characterization

The purpose of this task is to put together a number of measurement procedures that will allow characterization, in a quantitative manner, of the ultrasonic test (UT) equipment currently being used for ISI of reactor piping. For purposes of characterization, we have conceptually divided the UT instrument into three subsystems: the pulser transmitter, the ultrasonic transducer, and the receiver circuitry and display. Techniques for measuring the electrical and mechanical properties of each of these subsystems are in the process of being implemented; the state of development of each is briefly described below.

Pulser Transmitter. Electrical characterization of the pulser circuitry is accomplished with the system illustrated schematically in Figure 1. The

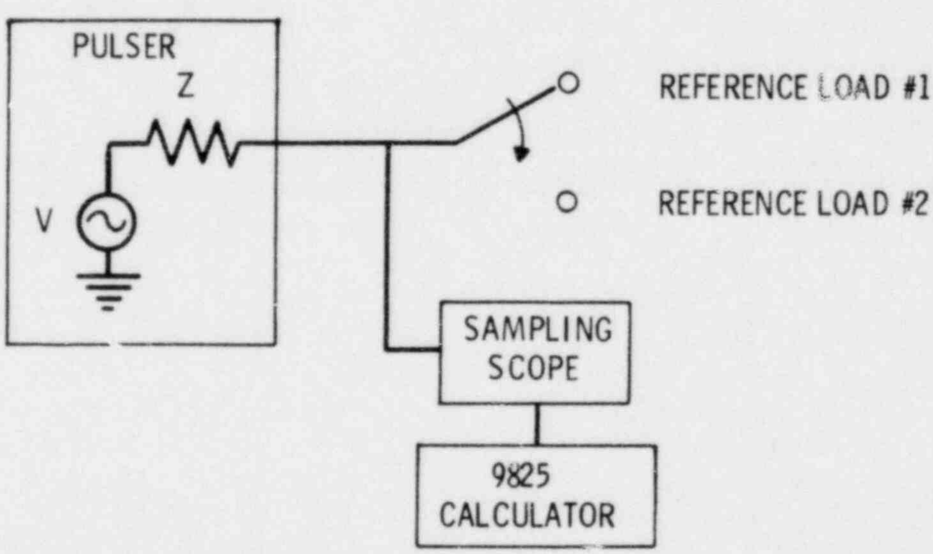

FIGURE 1. Ultrasonic Instrument Transmitter Evaluation System

two properties of the pulser of interest in measuring are its effective output impedance $Z(\omega)$ and the power spectral density of the drive pulse represented by $|V(\omega)|^2$. The system shown in Figure 1 records the time domain signal from the pulser as it drives two different reference loads. These signals are stored in the HP9825A computer. Taking the Fourier transforms of these two signals and performing some point-by-point algebraic manipulations in the frequency domain allows $V(\omega)$ and $Z(\omega)$ to be calculated directly. This system has been implemented and is currently being evaluated on pulsers with known electrical characteristics.

Ultrasonic Transducer. Characterization of the ultrasonic transducer (both electrical and mechanical) is being accomplished with the system illustrated in Figure 2. This system records three temporal waveforms: the transmit pulse into an electrical reference load; the transmit pulse into the transducer; and the receive pulse as measured by the transducer from a perfect reflector. By applying Fourier transforms and linear circuit theory to these signals transducer properties of interest can be calculated. These properties include: the complex electrical impedance of the transducer, the electrical power dissipated in the transducer in the transit mode, the available

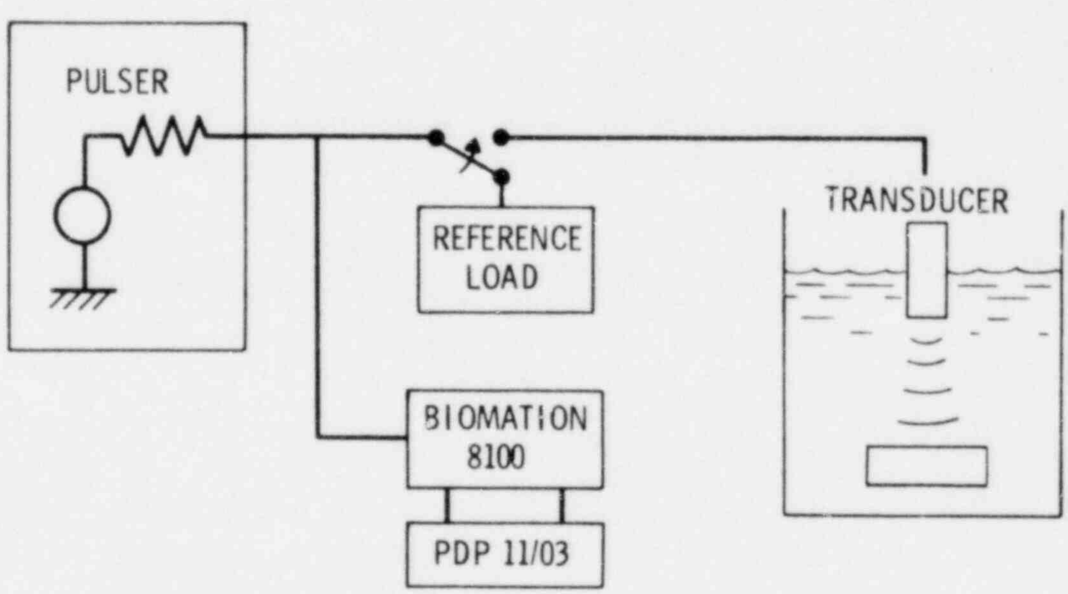

FIGURE 2. Transducer Evaluation System

electrical power generated by the transducer in the receive mode, and the insertion loss (or loop sensitivity) of the transducer. At this time, system software and hardware have been assembled and are being evaluated on a number of transducers.

To fully characterize the performance of an ultrasonic transducer, it is also necessary to determine the beam profile of the device. This task is relatively straightforward for immersion transducers, but it is somewhat more difficult for shear wave contact transducers. For purposes of beam profiling these shear wave contact transducers, a calibration block was built (see Figure 3) that allows either 60^0 or 45^0 shear wave transducers to be mounted on the slanted side surfaces and the sound field produced to be measured by scanning the top surface. The sensor used for scanning will be a noncontacting electromagnetic-acoustic transducer (EMAT) operating at 2.25 MHz.

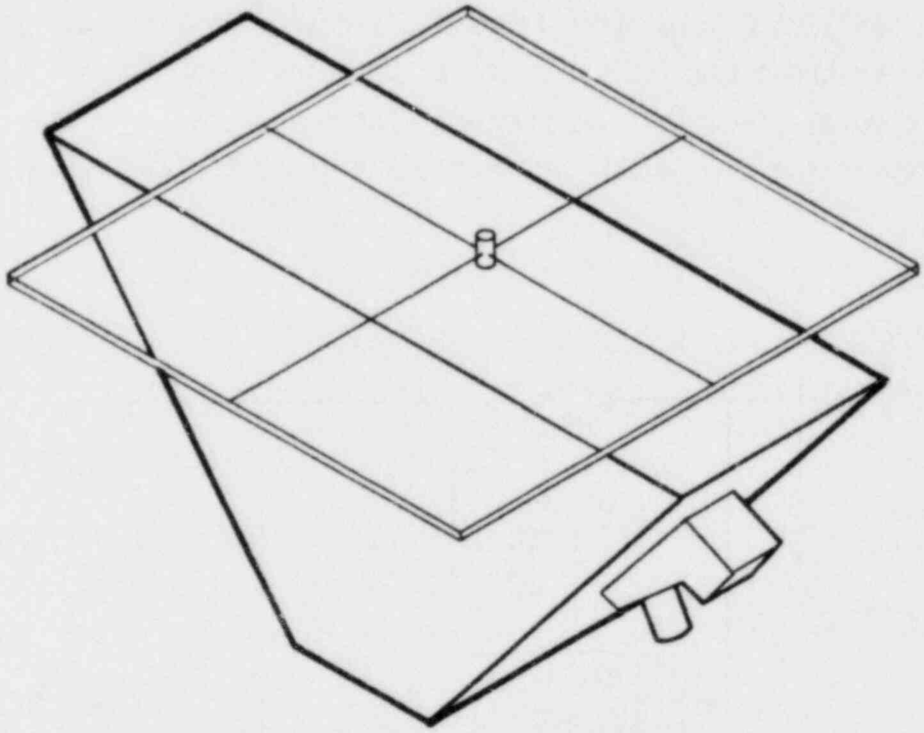

FIGURE 3. Test Block for Measurement of Angle Beam Transducer Sound Field Profiles

At present, the calibration block has been fabricated and an X-Y scanner and controller are being adapted for data collection.

Receiver Circuitry and Display. Electrical characterization of the receiver circuitry and the instrument display is being accomplished with the system shown in Figure 4. This is a semiautomated tone burst system for instrument characterization that incorporates an HP9825A computer for data collection, storage, and display. The following properties of the UT instrument receiver can be measured with this system: receiver linearity, receiver frequency response, root mean square (rms) input noise, and rms input sensitivity. This system is fully implemented at this time.

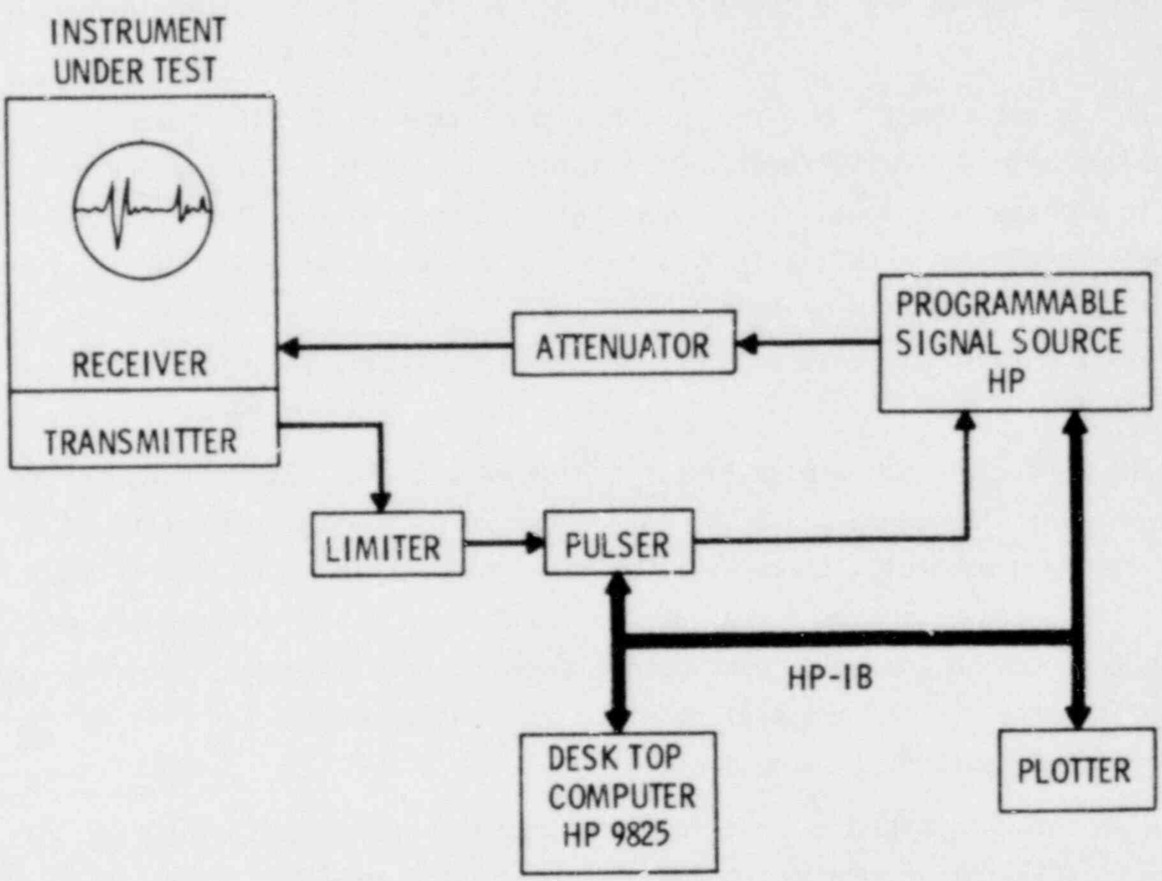

FIGURE 4. Ultrasonic Instrument Receiver Evaluation System

Austenitic Welds

The large dendritic grain structure of austenitic welds presents a major ISI problem when only one side of the weld is accessible. Vertically polarized angle beam shear waves that are normally used for ISI are highly affected by this structure. As a result, the effectiveness of inspection within and beyond the weld metal is highly questionable.

The objective of these measurements is to determine the achievability of full coverage of the required inspection volume. This volume, defined by IWB-2500-8 of ASME Section XI 1977, extends axially one-fourth in either side of the weld crown and from the inside diameter (ID) up to one-third of the pipe wall thickness. Measurements were made to determine the amount of ultrasonic signal that reaches the ID surface over the length of the required inspection volume.

In the experiment a 0.5-in. 2.25-MHz transducer with a 45^{0} shear shoe is coupled onto the OD surface as if inspecting the weld. This search unit is used in the transmit mode. The sound field incident on the ID surface is mapped by scanning an EMAT axially along the ID. Received amplitude is charted versus ID axial position on an X-Y recorder. Thus, a one-dimensional map of the ID sound field for each position of the transmitting transducer is obtained.

The first such map was obtained in base metal. The transmitting transducer was then incremented 0.1-in. toward the weld for each subsequent map until the weld was fully traversed; the final map was in base metal on the other side. The data showed that as the sound field began to enter the weld, it was diverted to the weld root. Since the weld root is small, this was also a focusing effect; the magnitude received here was often greater than at the center of the base metal sound field.

When the transmitter's beam index reached the approximate center of the crown, part of the beam began to "punch through" the weld and reach its expected (45^{0}) exit point; the rest of the beam was still being diverted straight down to the root. With further increments of the transmitter, the signal at the weld root finally disappeared and all the sound went to the 45^{0} exit area.

At no transducer position did any large part of the beam reach the ID surface within the required inspection volume of the far side of the weld. The sound was diverted to the root; and when forward incrementing finally allowed part of the beam to penetrate, it overshot the volume of interest.

In addition to the impact on coverage, this redirection of energy results in additional unexplainable indications, which are generally called "metallurgical reflectors" as there appears to be no discontinuity responsible for the reflection. From our data we expect that in many cases sound redirected toward the root and the crown is responsible for these extraneous signal responses.

This seems to indicate that 45^{0} shear waves cannot be used successfully in single-side access inspection of austenitic SS welds; however, the data are not yet complete. The EMAT senses only one component of the particle vibration; for the data described above, the EMAT was sensing vibration in the pipe-axial direction. To complete the data, the pipe-circumferential and pipe-radial vibration components must also be measured. It is expected that these measurements will support the conclusions drawn from data in hand, and the remaining data will be acquired as other round robin activities permit. Similar samples have also been fabricated from extra sections of 10-in. Schedule 80S welded pipe, and the same experiment can be performed on one or more of them.

TASK 8 - ROUND ROBIN PREPARATION

Round robin activities during the past quarter included: performance of a "mini round robin" or trial run, documentation of samples, and development of data collection and analysis software.

The trial run test matrix of 65 measurements, which concentrated primarily on 10-in. Schedule 80 pipe with a small sample of cast SS, was completed in early December 980. The objective of the trial run was to establish the effectiveness and applicability of the test protocol as well as provide some information concerning the probability of detection. The trial run also provided training for PNL personnel who will administer the tests. Significant results of the tests include:

- Time required for one inspection (one-fourth of a 10-in. pipe from one side) was 20 min at 1977 Section XI sensitivity; this includes approximately 10 min of setup time.
- Lowering the reporting level by 6 dB from the above level increased the inspection time by 20 min.
- Time required for cast SS samples was approximately 15 min.
- As expected, cast SS proved to result in a low probability of detection.
- It may be desirable to minimize the size of certain experiments, such as defects on the far side of austenitic welds and cast SS, that will result in very low probabilities of detection.
- The analog outputs of the Search Unit Tracking and Recording System (SUTARS) along with time and amplitude outputs from the inspection instrument were recorded on a four-channel strip chart and proved to be both highly useful and accurate.

Data collection and analysis software has been designed and partially tested. We have attempted as much as possible to maintain a format similar to the PISC Trials. The results of the inspection are directly entered into the computer from the inspection report form, which requires only 2 to 3 min per inspection. Scoring results will be available as the test is in progress and will assist in early identification of errors that could invalidate test results.

Scoring of the test results requires an accurate description of the flawed samples. This documentation is in progress. The data required on each round robin sample include:

- sample ID
- maximum response in dB relative to ASME XI distance amplitude correction

- circumferential location of maximum UT "hot spot"
- circumferential location of crack endpoints
- axial location of crack
- crack depth.

The UT data in this list (maximum response and circumferential location of maximum UT hot spot) are obtained by manual angle beam inspection using an appropriate search unit and instrument. Circumferential location of the endpoints of a crack and axial location of a crack are obtained by dye penetrant examination. The method of crack depth measurement varies according to material.

FUTURE WORK

Primary emphasis in the coming quarter will be on completing round robin preparations. It is expected that all sample fabrication will be completed by early March; the round robin will be initiated at that time.

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-432 and IFA-513) are still under irradiation as is the final test, IFA-527. All the IFAs are heavily instrumented six-rod clusters.

The Halden Project, NRC, and Pacific Northwest Laboratory (PNL) mutually agreed to allow further operation of IFA-527, even though at least three of its six rods have suffered failure of the pressure boundary (i.e., cladding or end closure) and have some inleakage of water. The assembly was successfully brought back to full power from cold shutdown in late December with no detectable fission product leakage and no further fuel rod failures.

Postirradiation examinations (PIEs) of rod 8 of IFA-432 being conducted at Harwell, UK, are nearly complete.

(a) RSR Fin. Budget No.: B2043; RSR Contact: G. P. Marino.

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-1 series of computer codes
- study the occurrence and mechanisms at fuel cladding failure via controlled experiments with centrally heated simulated fuel pins in a PNL pressurized water loop.

The Halden reactor in Norway is currently the sole site used by this program for irradiation tests. PIEs will be carried out at both Kjeller, Norway, and Harwell, UK. Task A of the program is concerned with the conduct of the tests and coordination of test design, test fabrication, shipping, PIE, and sample disposal. The test matrix now spans the full range of expected boiling water reactor (BWR) conditions for pelletized UO_2 fuel, including

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50-380 μm (0.002-0.0015 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

Irradiation experience this quarter was uneventful because the reactor was shut down for most of the period. No further rod or instrument failures occurred. A joint decision was made by NRC, PNL, and Halden to continue operation of IFA-527 even though three of its six rods have suffered failure of the pressure boundary (cladding or end closure) and inleakage of water. The decision to continue operation was based on the following considerations:

- During one month of full-power operation with failed rods (September 12 to October 3, 1980), IFA-527 released no detectable fission products. Therefore, the prospects are good for continued successful operation.
- At least two rods show no evidence of failure and their continued operation helps fulfill initial objectives for the assembly (tracing the speed and thermal effect of fuel pellet cracking and relocation).
- One rod has evidenced water inleakage at the top but not at the bottom, as determined by fuel centerline thermocouple readings in both ends of the rod. This behavior may be rare experimental evidence of the ability of the fuel to block water migration from a failed to an unfailed region.

IFA-527 was restarted from cold shutdown to full power (18 kW/m peak linear heat generation rate) on December 20-23, 1980. There were no further instrument or rod failures and no detectable fission product release.

Preliminary data has been obtained from the in-cell apparatus for measuring fuel column axial compliance developed at Harwell, UK. Only simulated fuel columns (drill rod and cracked alumina) have been tested so far.

FUTURE WORK

More data will be gathered on simulated rods from the compliance test at Harwell, UK, and compared to similar data from tests at PNL. Upon favorable comparison (hopefully, by late January 1981) Harwell will test the compliance of rod 7 (unirradiated) and rod 8 (irradiated).

We anticipate uneventful operation of IFA-527 until February 1981 at which time a reactor outage is scheduled. During that outage one of the failed rods will be removed. Hopefully, later PIEs will indicate the progression of the cladding damage in the failed rods. All assemblies should restart in March and continue operation until August or September 1981, at which time they will be permanently discharged to meet cooling and PIE requirements before the mandated close of the program in September 1982.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:
TASK B - DATA QUALIFICATION AND ANALYSIS[a]

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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 that data. During this quarter data for IFA-527 was received, corrected, and made available for analysis; thermal resintering and thermal conductivity tests were run on IFA-513 and IFA-527 archive fuel pellets. A paper on fuel cracking and relocation was presented at the November 1980 American Nuclear Society meeting held in Washington, D.C.

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:

(a) RSR Fin. Budget No.: B2043; RSR Contact: G. P. Marino.

- 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.

SUBTASK B-1 - DATA PROCESSING

During the last quarter, test fuel data report (TFDR) tapes for IFA-432, -513, and -527 were received for the period July 1 to October 4, 1980. The IFA-527 data transmittal was found to have an incorrect value for the total assembly power that initially caused some disagreement between Halden and PNL estimates of local power. Good agreement on local power was obtained after correcting the error.

SUBTASK B-2 - DATA REPORTS

Measurements of the thermal conductivity and thermal resintering behavior of the fuel used in IFA-513 and IFA-527 have been made. A laser "flash" technique was used to determine the thermal conductivity of thin disks cut from archive pellets. Results showed close agreement between the two fuel batches.

Comparison to other thermal conductivity data shows good agreement with Lyons[1] and MATPRO,[2] while the IFA-431 and -432 fuel had a higher measured thermal conductivity.[3] Thermal conductivity may be described by the following equations.

- IFA-513:

$$K(T) = 7.495 - 9.153 \times 10^{-3}T + 3.637 \times 10^{-6}T^2 + 8.075 \times 10^{-9}T^3$$
$$- 1.293 \times 10^{-11}T^4 + 7.214 \times 10^{-15}T^5 - 1.441 \times 10^{-18}T^6$$

- IFA-527:

$$K(T) = 8.657 - 1.990 \times 10^{-2}T + 4.605 \times 10^{-5}T^2 - 7.474 \times 10^{-8}T^3$$
$$+ 7.190 \times 10^{-11}T^4 - 3.664 \times 10^{-14}T^5 + 7.587 \times 10^{-18}T^6$$

where K(T) is in W/m-^oC and T is in ^oC ($200 < T < 1400^oC$).

One IFA-513 pellet and three IFA-527 pellets were thermally resintered for 24 hours at 1973K. Immersion density measurements revealed a density increase of approximately 0.32% theoretical density (TD). The average density increase for the 95% TD stable fuel used in IFA-431 and -432 was approximately 0.25% TD. The thermal resintering and thermal conductivity results support the conclusion reached in the IFA-513 precharacterization report[4] that the IFA-513 fuel should be similar in behavior to IFA-431 and -432 fuel.

Efforts continued this quarter on the analysis of the 92% TD unstable fuel used in rod 6 of IFA-431 and IFA-432. Uncertainty bands for the measured temperature data were defined on the basis of instrument error and by linear propagation of errors.[5] Fuel performance computer code predictions were refined and the following results were obtained:

- FRAPCON-1 temperature predictions lie above the uncertainty band.
- GAPCON-THERMAL-3 (GT3) temperature predictions are below the band if no densification is allowed; however, assuming a final fuel density of 96.5% TD results in temperatures above the uncertainty band.
- FRAPCON-2 temperature predictions lie within the uncertainty band.

SUBTASK B-3 - DATA ANALYSIS

During this quarter analysis continued in the areas of fuel relocation and cracking and fission gas release.

Fuel Relocation and Cracking Analysis

Sensitivity analysis has been performed on the PNL-developed fuel cracking and relocation model.[6] Two principal conclusions have been reached. First, the inferred radial elastic modulus of the cracked fuel appears to be relatively insensitive to cladding elongation data for a particular rod of specific initial gap. Second, varying the crack pattern (7-11 cracks) and deduced crack roughness (multipliers of 0.8-1.3) can yield calculated fuel-cladding interfacial stresses (at 30 kW/m) ranging from 2.7 to 27.6 MPa (400-4000 psi). At the same time, thermal conductivities and gap conductances remain within their uncertainty bounds. Since fuel crack pattern would be expected to vary axially along the rod and since the effective crack roughnesses are probably related to the crack patterns, this implies that the localized stress state of the cladding is quite variable. This is enhanced by the effects of fuel surface discontinuities (gap roughness) that cause stress concentrations at the inner surface of the cladding. The impact on modeling of cladding failures could be appreciable.

A paper entitled "Thermal-Mechanical Properties of Cracked UO_2 Pellets" was presented at the November 1980 American Nuclear Society meeting in Washington, D.C.; and a journal article discussing the PNL fuel cracking and relocation model is being prepared for submittal to Nuclear Technology.

Transient Data Analysis for IFA-527

Fuel centerline thermocouples have continued to operate in both failed and unfailed IFA-527 rods. A major concern is whether the thermocouples operating in the failed rods, which contain water or at least steam, are still giving reliable temperature readings. Bias possibly exists from electrical shunting or thermocouple wire and sheath degradation due to the presence of steam. The correspondence of the relative transient and steady-state behavior of the failed versus unfailed rods leads us to believe that the operating thermocouples in both types of rods are reliable.

Specifically, indicated temperatures (at a given power) in failed rods stand intermediate between the unfailed nominal and small gap rods; similarly, their transient response to rapid power decreases is intermediate between the other two rods. This is qualitative confirmation of the correctness of the temperature readings.

For a more quantitative confirmation of failed rod fuel temperatures, we simulated the transient behavior with the small code MWRAM (tuned to the apparent steady-state data of various rods). The correspondence of calculated and measured slopes of temperature versus time in a linear power decrease constitutes confirmation of initial thermocouple readings (see Table 1).

TABLE 1. Measured and Calculated Slopes of Normalized Temperature Versus Time for IFA-527 Rods (September 22, 1980)

Rod Number	Condition	Temperature at 17 kW/m, K	Measured Slope, %/sec	Calculated Slope, %/sec
1	Failed	1033	0.77	0.77
2	Resealed	1674	0.34	0.39
3	Unfailed	1801	0.34	0.36
4	Failed	1000	0.77	0.78
6	Unfailed (small gap)	819	0.81	0.81

Fission Gas Release Analysis

The results from gas sampling rod 8 of IFA-432 have been received from Harwell and were used to estimate fission gas release with the following results:

- pressure = 0.94 MPa at 273K
- gas composition = 85.3% xenon and krypton and 13.7% helium
- gas release fraction = 9.2%.

Rod 8 replaced rod 4 of the original assembly and was removed from the assembly and sent to Harwell for PIE after achieving an average burnup of 1890 GJ/kgU (22,000 MWd/MTM). Rod 8 was not instrumented with thermocouples; therefore, a temperature history is not available.

Rods 1 and 8 of IFA-432 were of identical design and operating history and should have had similar temperature histories. Figure 1 compares the measured gas release fraction for rod 8 with the deduced release fractions for rod 1. The measured value for rod 8 is within the estimated range for rod 1. However, the estimates for rod 1 do not include the effects of fuel swelling or helium production and release during irradiation; and the rod 8 data indicates that both of these effects should be included in the analysis. If these effects were included, the estimated release fraction for rod 1 would drop below the measured release fraction for rod 8. This difference could also be due to higher fuel temperatures for rod 8 or to the uncertainties in estimating fission gas release from in-reactor pressure data.

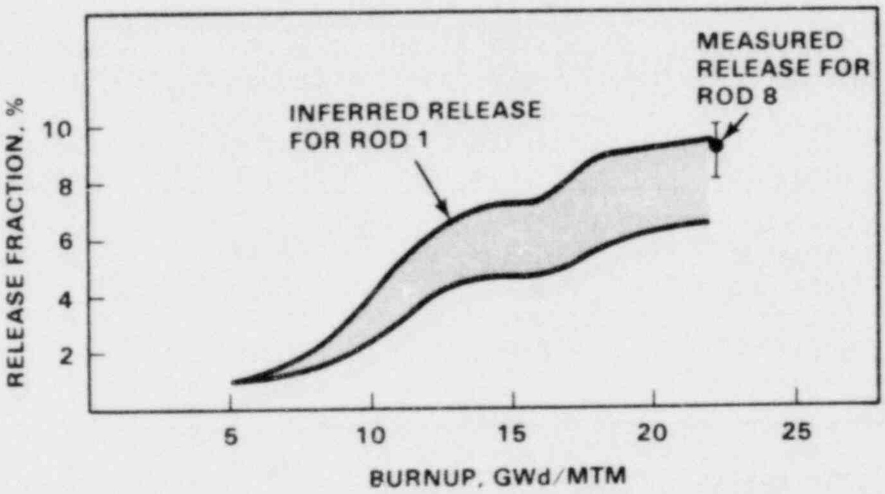

FIGURE 1. Comparison of Fission Gas Release in Rods 1 and 8 of IFA-432

FUTURE WORK

Data processing for next quarter will continue as required. Data reports for IFA-432 and IFA-513 for the period from April 1978 to January 1980 will be published. Reports discussing the precharacterization and startup of IFA-527 will be completed when necessary information is received from Halden. Analysis of fuel cracking and relocation, fission gas release, and related topics will continue.

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*Available for purchase from the National Technical Information Service, Springfield, VA 22161.

**Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES: TASK C - CODE COORDINATION AND EX-REACTOR TESTING[a]

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R. E. Williford, Task Leader

S. O. Bates
M. E. Cunningham
W. N. Rausch

SUMMARY

The FRAPCON-2 code development and user's manual document was completed and sent to the U.S. Nuclear Regulatory Commission (NRC) for publication, and the Developmental Assessment document is nearing completion. Five cracked fuel-cladding compliance experiments were performed, and the detailed axial-diametral deformation data is under analysis. Three-dimensional (3-D) cracked fuel model development was accelerated in preparation for the development of a fuel failure model for FRAPCON-2.

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 description and developmental assessment documents in coordination with the Idaho National Engineering Laboratory (INEL). Code improvement includes 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 those models were implemented in FRAPCON-2. Fuel cracking causes reduced thermal conductivity

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(a) RSR Fin. Budget No.: B2043; RSR Contact: G. P. Marino.

and elastic moduli and is presently described by three primary parameters--crack roughness, gap roughness, and crack pattern--that were inferred from in-reactor data. In FY-1980, ex-reactor data was 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.

SUBTASK C-1 - FRAPCON-2 CONTROL AND MAINTENANCE

The FRAPCON-2 code development and user's manual document was completed and forwarded to NRC for publication. The Developmental Assessment document is in editing at INEL, and PNL expects to receive a copy for review by mid-January. Some of the assessment results were presented at the 8th Water Reactor Safety Research Meeting, October 27-30, 1980.

SUBTASK C-2 - FUEL MECHANICS EXPERIMENTS

Five fuel rod compliance experiments have been performed. Detailed data was simultaneously collected for fuel axial load and strain, cladding axial strain, and diameter traces of the cladding. Each sample was subjected to four loading ramps, and fuel cracking occurred primarily on the first two ramps. When the fuel column was removed from the cladding, an axial variation in the number of cracks was observed that is thought to be the cause of the axial variation in the bamboo ridge heights. This data has been digitized for further computer analysis. Cracked fuel column-cladding friction experiments were also performed, and friction coefficients between 0.5 and 1.5 were calculated from the data. Some preliminary cracked fuel-cladding relaxation experiments were also performed.

SUBTASK C-3 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

Meetings with the Fuel Behavior Research Branch (FBRB) of the NRC resulted in some changes in emphasis for this subtask. Two-dimensional (2-D) cracked fuel model efforts will be applied toward an accelerated development of the 3-D model, which will become part of a newly defined fuel failure model. Present efforts are focused on the development of a stochastic model for predicting the cladding inner surface stress distributions caused by the asymmetrically cracked fuel. The basic approach for the fuel failure model was outlined in a letter to FBRB.

FUTURE WORK

The following activities are planned for next quarter:

- The FRAPCON-2 Developmental Assessment document will be completed and published.
- The cracked fuel-cladding compliance, friction, and relaxation experimental data will be analyzed in detail; and report writing will begin.
- The 3-D cracked fuel model will near completion, and development will begin on the fuel failure model for FRAPCON-2.
- Compliance testing of an irradiated rod at Harwell, UK, will resume.

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 objectives and work schedule were defined for this task. Subtask D-1 focuses on developing an instrument to measure strain over small gage lengths (1 mm), and subtask D-2 involves adapting a pressurized water loop in a Pacific Northwest Laboratory (PNL) facility where centrally heated instrumented simulated fuel pins will be run to failure under various conditions.

A concept review for the strain-measuring instrument design reduced the number of options to two, and preliminary configurations have been established for both cases. Design and radiation shielding reviews for the loop facility were performed, and electronics components were ordered. In addition, the preliminary pressure boundary design was begun.

INTRODUCTION

The primary objective of Task D is to collect fuel rod failure data on irradiated cladding under loading conditions typical of those in-reactor, including asymmetrically cracked pellets and coolant external pressures. The 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 PNL. This experimental equipment has the capability for controlled power ramping and load cycling schemes and provides great experimental flexibility

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

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. This data complements Task C efforts and provides a means of verifying PCI models.

TECHNICAL PROGRESS

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

SUBTASK D-1 - INSTRUMENT DESIGN

Discussions with the Fuel Behavior Research Branch (FBRB) of the U.S. Nuclear Regulatory Commission (NRC) in October 1980 defined the scope and objectives of this subtask. An instrument will be designed to measure localized cladding strains and strain rates in the loop tests with flexibility for other laboratory tests. Background information was collected, and a survey of instrument concepts began. Options were narrowed to either strain gage or eddy current methods, and a preliminary configuration was established for each case.

SUBTASK D-2 - LOOP EXPERIMENTS

Subtask objectives and a schedule were defined in October 1980. A design review established the maximum loop pressure boundary inner diameter (test section ID) as 76 mm (3 in.). This is the design envelope for subtask D-1. A review of the loop radiation shielding for FY-1982 experiments revealed that existing equipment is adequate for operation with irradiated cladding, but improvements will be necessary for the sample handling and assembly stages. Electronics packages for the loop power supply/control and safety systems have been ordered, and a preliminary design of the pressure boundary has begun.

FUTURE WORK

The following activities are planned for next quarter:

- An instrument concept will be chosen for the in-loop strain gage, detailed design will begin, and manufactured components will be ordered.
- Detailed design of the pressure boundary will begin, and materials will be ordered for fabrication. Electronics components should arrive near the end of the quarter.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

PBF SEVERE FUEL DAMAGE TEST PROJECT[a]

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J. O. Vining
C. L. Wheeler

SUMMARY

Detailed mechanical design of the lower inlet and fuel pin regions of the test train assembly for the first two Power Burst Facility (PBF) severe fuel damage (SFD) tests was completed during this quarter. Major portions of both the design and development efforts in support of the program were directed towards the insulated shroud design for the PBF severe fuel damage test assemblies. In the current design for the insulating shroud the ZrO_2 insulation is in a hermetically sealed octagonal container consisting of a relatively massive Zircaloy outer structure and a thin sacrificial Zircaloy inner structure.

A specimen of shroud material was submitted to Dynatech Research and Development Company, Cambridge, Massachusetts, for thermal conductivity measurements. Thermal conductivity measurements were completed, and the results are being reviewed and evaluated at Pacific Northwest Laboratory (PNL).

Final design of the fallback barrier test apparatus was completed during this quarter. In addition, the final design of the boil-off test apparatus is near completion (scheduled to be finished in early January 1981). The boil-off test apparatus will utilize the same inlet/outlet piping as the fallback barrier test apparatus, thereby saving a substantial amount of time and money.

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

Several prototype temperature profile detectors (TPDs) were designed, fabricated, and successfully tested in a horizontal orientation at temperatures of 570K, 917K, and 1197K. One additional prototype TPD was successfully tested in a vertical orientation at 1725K. Preparations are under way for other tests of vertically oriented devices at higher temperatures since the desired operating temperatures for this device are 1845K and 2100K.

INTRODUCTION

This part of the Severe Core Damage (SCD) Test Subassembly Procurement Program--the 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 work scope of this task includes the design effort, 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 the 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 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 1 m of heated length and internal cladding thermocouples will be installed in a 7-rod circular 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

Detailed mechanical design of the lower inlet and fuel pin regions of the test train assembly for the first two PBF SFD tests was completed. The design for the lower inlet region for the test assembly was completed (including the inlet region in-pile liner seal interface, the lower braze plug assembly, the shroud bypass water turbine flowmeter, and the fuel rod tie plate). In addition, the fuel pin design in the core region was completed; and the instrument types and locations were detailed for each of the 32 fuel pins in the bundle.

TEST ASSEMBLY INSTRUMENTATION

Table 1 summarizes the instrument types and the relative number of each instrument type in the test train assembly. Fuel cladding, steam, and shroud temperatures will be measured as will fuel rod and system pressure histories. A PNL design of a fuel rod failure indicator will be used on portions of the fuel rods within the bundle to indicate the approximate time of individual fuel rod failure during the tests. Additional instrumentation within the test train assembly will include eddy current transducers and eddy current or pressure taps for both plenum pressure and coolant measurements. Self-powered neutron detectors (SPNDs) will be located within the bundle to determine test assembly powers during the tests and will possibly be used to indicate fuel movement within the test bundle during the later stages of the tests. The SPNDs planned at this time will be made of an Inconel 600 collector and sheath with a cobalt emitter and MgO insulation. Coolant water flow rate for both assembly and bypass water will be measured with turbine flowmeters that will be integral to the test train assembly. Finally, a wire mesh fuel and molten metal penetration detector for the test train assembly shroud and TPDs that will indicate

TABLE 1. PBF Severe Fuel Damage Test Assembly Instrumentation

Level	Inlet Coolant Temperature TCs[a]	Bypass TCs	Differential TCs	ID Clad TCs	Fuel Centerline TCs	Inside Shroud TCs	Middle Shroud TCs	Outside Shroud TCs	SPNDS	Steam Probes	Total
1	2	2	4 (Lower Leg)								8
2[b]				7		2		4	2		15
3[c]				10		2	4	4	2	5	27
4[d]				7	6	2	4	4	2		25
5									2		2
6			4 (Upper Leg)				2		2	5	13
7[e]										3	3
8[f]										3	3
Total	2	2	8	24	6	6	10	12	10	16	96

(a) TCs = thermocouples.
(b) 0.35 m above bottom of fuel column.
(c) 0.50 m above bottom of fuel column.
(d) 0.70 m above bottom of fuel column.
(e) 0.2 m above top of fuel column.
(f) 0.4 m above top of fuel column.

Notes:
2 temperature profile (2-1/8-in. tube through braze plug)
2 mesh (2 hardlines)
2 flow turbines (2 hardlines)
4 coolant pressure transducers (8 hardlines)
5 pressure switches (8 hardlines)
5 plenum pressure transducers (5 hardlines)
Total hardlines through lower braze plug = 54 (including 2-1/8-in. tube for temperature profile)
Total hardlines through upper braze plug = 116 (including 2-1/8-in. tube) (including 4 differential hardlines)

peak temperatures within the bundle after conventional thermocouples no longer function will be used in the assembly instrumentation package.

Details of the radial and axial array of the placement of the instruments within the test train assembly are shown in Figure 1. The figure shows the entire 32-pin bundle cross section for the test assembly; the various levels correspond to the instrument types and relative numbers of each instrument type as presented in Table 1. It should be noted that there are grid spacers at approximately levels 3, 5, and 7 within the test train bundle region.

SHROUD DEVELOPMENT EFFORTS

Major portions of both the design and development effort in support of the program have been directed toward the insulated shroud design for the PBF severe fuel damage test assemblies.

Early in the development work a shroud insulator material was selected on the basis of melting point, compatibility with steam, low nuclear cross section, availability of material, and fabricability. The material that appeared to best meet these requirements was a ZrO_2 fabric bonded with ZrO_2 binder and sintered to a rigid shape. The fabric shrinks about 10% between 1000^oC and 2000^oC with all but approximately 0.5% of the shrinkage occurring between 1000^oC and 1600^oC. Fabrication procedures were developed to allow for shrinkage and still produce parts of the required dimensions. The current design for the insulating shroud is for the ZrO_2 insulation to be in a hermetically sealed octagonal container, composed of a relatively massive Zircaloy outer structure and a thin sacrificial Zircaloy inner layer.

During operations at loop pressures up to 15.5 MPa (2250 psia), the Zircaloy on both sides of the insulated shroud will be forced against the insulation with about equal pressure. Subsequent testing indicated that the composite insulation made from ZrO_2 fabric bonded with ZrO_2 will not withstand this pressure; it will crumble, as was easily demonstrated using a hydraulic press. If the inner Zircaloy layer is eliminated, the insulation will be wet with hot high-pressure water during portions of the testing. Autoclave testing showed that the fabric insulation is destroyed by such treatment. Although

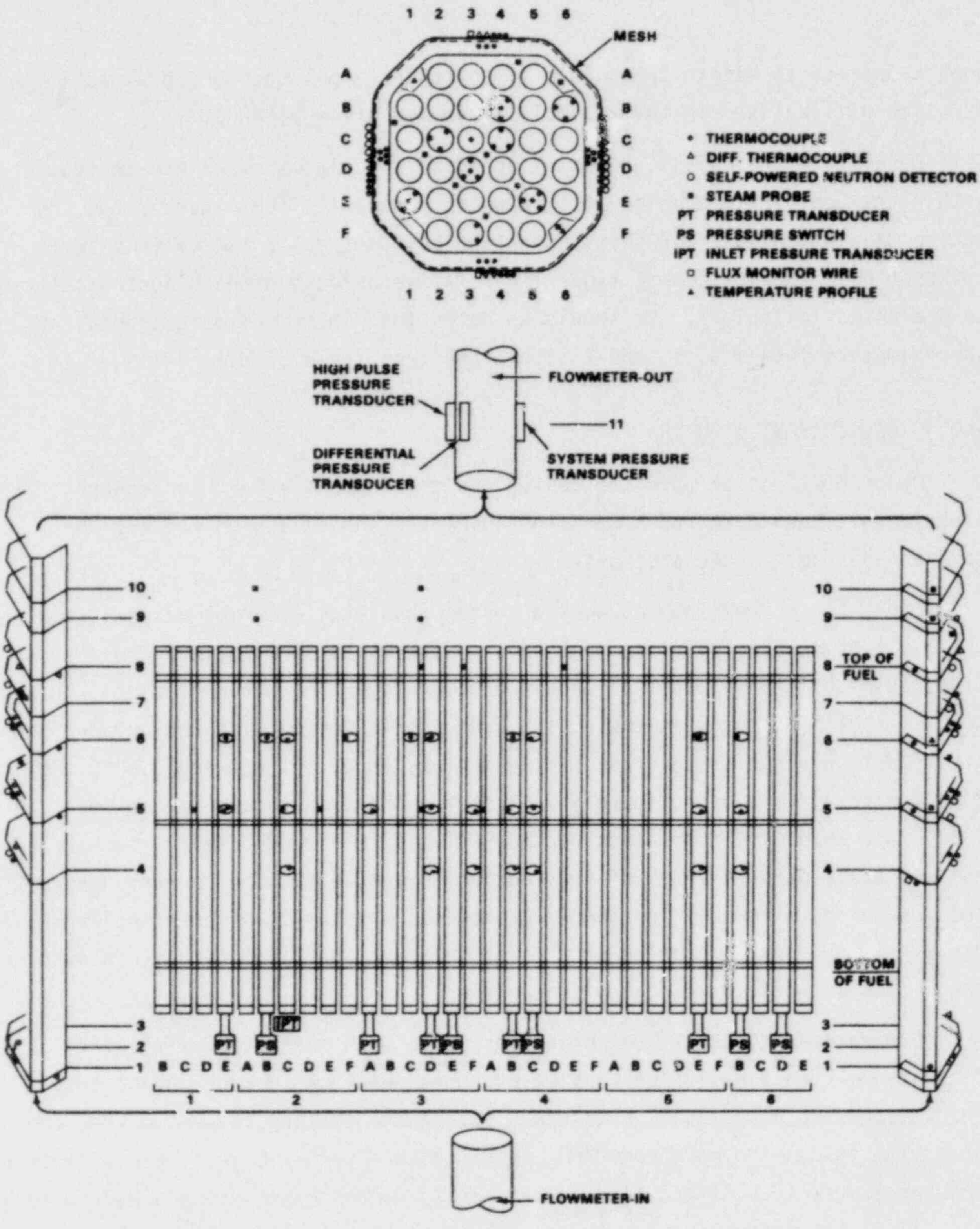

FIGURE 1. Bundle Instrumentation for Severe Fuel Damage Tests in PBF

ZrO_2 is very stable under such conditions, the small diameter (4-micron) fibers are attacked, thus disrupting the overall composite structure and destroying the fabric insulation.

Concepts were developed for supporting the differential pressure between the inner and outer layers of the shroud by the use of rigid insulation between the layers. It was necessary to choose another form of ZrO_2; the present choice is a ZrO_2 honeycomb material used so that the honeycomb cells are parallel to the heat flow path. The open cells can be filled with hollow ZrO_2 bubbles, permitting heat flow by conduction only. The honeycomb is very strong (compressive strength >22,000 psi parallel to the open cells) and quite thermal shock resistant when cut or formed into thin sections. Samples have been obtained, and the feasibility of filling the cells with bubble ZrO_2 forms has been demonstrated. The bubbles are dimensionally stable; firing of the structure including the bubbles to 1600°C causes no shrinkage that would loosen them from the honeycomb.

A specimen of the ZrO_2 shroud insulation material was submitted to Dynatech for thermal conductivity measurements, which are required to determine the proper shroud insulation thickness. Preliminary tests have been completed, and a brief description of the testing method as well as results are presented below. A sample consisting of several bonded ZrO_2 fabric layers was submitted for apparent thermal conductivity and tested over a temperature range from 40°C to 1100°C. Approximate dimensions were 63 x 5.9 mm (2.48 x 0.23 in.), and the density was 1830 kg/m^3 (114 lb/ft^3). 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 of 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

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

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 and surroundings were filled with an insulating powder. By means of adjustments to 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 of the sample. At equilibrium conditions, the temperatures of various points in the system were evaluated from the 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:

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

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.

The apparent thermal conductivity results determined for the sample tested are presented in Table 2 and Figure 2. These results 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/hr-ft-^{0}F).

TABLE 2. Results of the Apparent Thermal Conductivity of a ZrO_2 Fabric Sample

Temperature		Apparent Thermal Conductivity	
^oC	^oF	W/m-K	Btu/hr-ft^oF
40	104	0.250	0.148
200	392	0.278	0.165
400	752	0.307	0.182
600	1112	0.331	0.196
800	1472	0.352	0.209
1000	1832	0.372	0.221
1100	2012	0.382	0.226
800[a]	1472	0.358	0.212
1100[a]	2012	0.393	0.233

(a) Repeat tests after the highest temperature measurements.

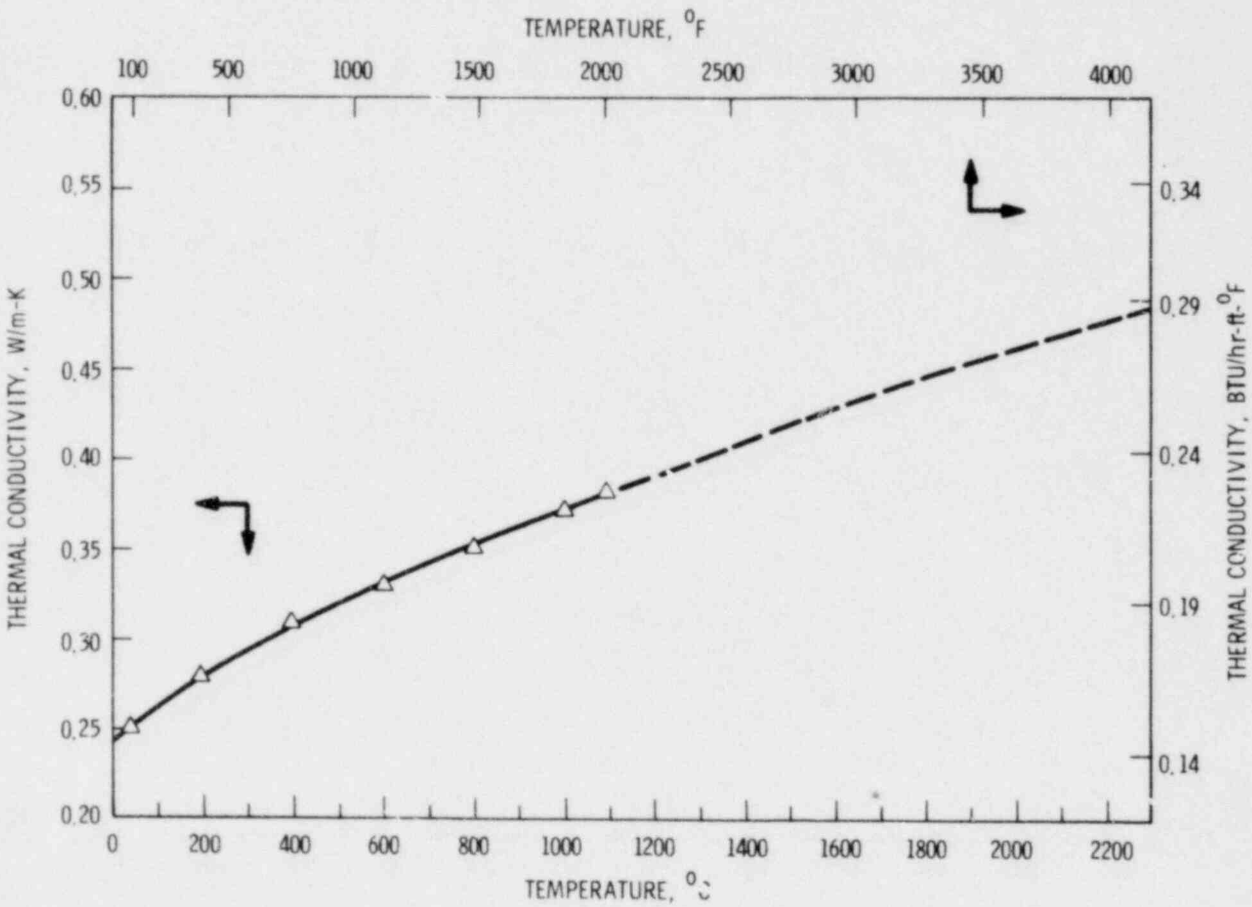

FIGURE 2. Apparent Thermal Conductivity Versus Temperature for ZrO_2 Fabric Insulation

FUTURE WORK

Detailed design of test train assembly No. 1 for the PBF Severe Fuel Damage Test Program will continue. Development work will continue at an accelerated rate on the selection and evaluation of alternate forms of shroud insulation material including the determination of methods for fabrication and assembly of the shroud. Work will be initiated to develop a suitable Zircaloy thermocouple to seal the shroud, such as a braze, a Zircaloy-to-stainless transition joint, or a mechanical seal. The higher temperature-sensing capabilities of TPDs will be extended to the goal temperatures of 1845K and 2100K through further testing. Design, fabrication, and testing of prototypes for the wire mesh molten fuel and molten metal penetration detector for the test assembly shroud that have been started will continue.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

ESSOR PROJECT[(a)]

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F. E. Panisko, Project Manager

J. E. Tanner
R. D. Tokar
C. L. Wheeler

SUMMARY

Efforts have been directed towards demonstrating the technical feasibility of performing small-break pressurized water reactor (PWR)-type tests in ESSOR. Feasibility can be demonstrated if satisfactory analytical results can be obtained for the desired test conditions using representative test assembly characteristics. Existing codes are being evaluated to find the optimum one to modify for part of our feasibility analysis. Instrumentation needs for test operation and control are being determined. Alternate shroud materials are being evaluated as a backup to the current concept to be used in Power Burst Facility (PBF) severe fuel damage (SFD) tests.

A suitable site representative was identified for the first two-year assignment at Ispra, Italy.

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 21 tests

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

to simulate various large- and small-break conditions in PWRs and boiling water reactors (BWRs). Pacific Northwest Laboratory (PNL) will supply three fully instrumented and fueled test assemblies, analytical services, and engineering support for the SSTP.

TECHNICAL PROGRESS

Efforts have been primarily directed at demonstrating the technical feasibility of performing small-break PWR-type tests in SUPER SARA. Analytical capabilities are being re-evaluated, test instrumentation and control are being investigated, and alternate shroud materials are being pursued.

Computer codes and other analytical techniques do not currently exist to model small-break LOC events. We are currently evaluating codes with the intent of modifying the optimum existing code; a version of COBRA is currently being evaluated. Part of our evaluation compares COBRA output with that from the TRUMP code. TRUMP is primarily a heat transfer code in which all hydraulic-related phenomena are treated via user input. Therefore, when the heatup rate and peak clad temperatures are controlled by flow rate, it is necessary to track the liquid level, which can only be done using a code that calculates local fluid conditions. To set inlet flow maps to obtain the desired heatup rate and to study the test assembly temperature response to change in inlet flow, it is necessary to develop and use tools that predict hydraulic as well as thermal phenomena.

Preliminary analytical results plus analysis of potential test operation methods have led to the conclusion that small-break high-temperature testing in SUPER SARA may be feasible if the earlier concept of a spray desuperheater is replaced with test section bypass coolant flow. This change will insure proper cooling of the pressure vessel and key instruments as well as the test bundle steam. In addition, we are proposing a separate exit steam line to provide a means of controlling the loop pressure and measuring the hydrogen and steam generation rates. Post-test condensable fission product measurements can also be performed with this system.

The fuel assembly shroud is the most important component in demonstrating the feasibility of small-break tests in SUPER SARA. The current design consists of low-density, low thermal conductivity ZrO_2 sandwiched with Zr metal inner and outer liners. If a backup material is needed for the Zr liners, few possibilities exist. The properties of a commercial molybdenum alloy, TZM, may be a suitable replacement when properly coated for protection from high-temperature steam. This alloy is being evaluated.

FUTURE WORK

During the next quarter the major portion of the technical feasibility study should be completed. The evaluation of existing codes will continue, and the physical testing of some TZM samples will begin.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

9-ROD TEST TRAIN FOR OPTRAN 1-3 PROJECT[a]

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R. E. Schreiber, Project Manager

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

SUMMARY

The 9-rod test train hardware (see Figure 1) was described in the report for the first quarter of 1980. All major components and instruments are now on hand.

Remote-handling techniques that were developed earlier are being verified. Work during this reporting period focused on the remote-handling equipment to be used in-cell and in-basin and in the transfer of the instrumented bundle of preirradiated rods from the cell to the Materials Testing Reactor (MTR) basin for final assembly of the test train.

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 9-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.

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

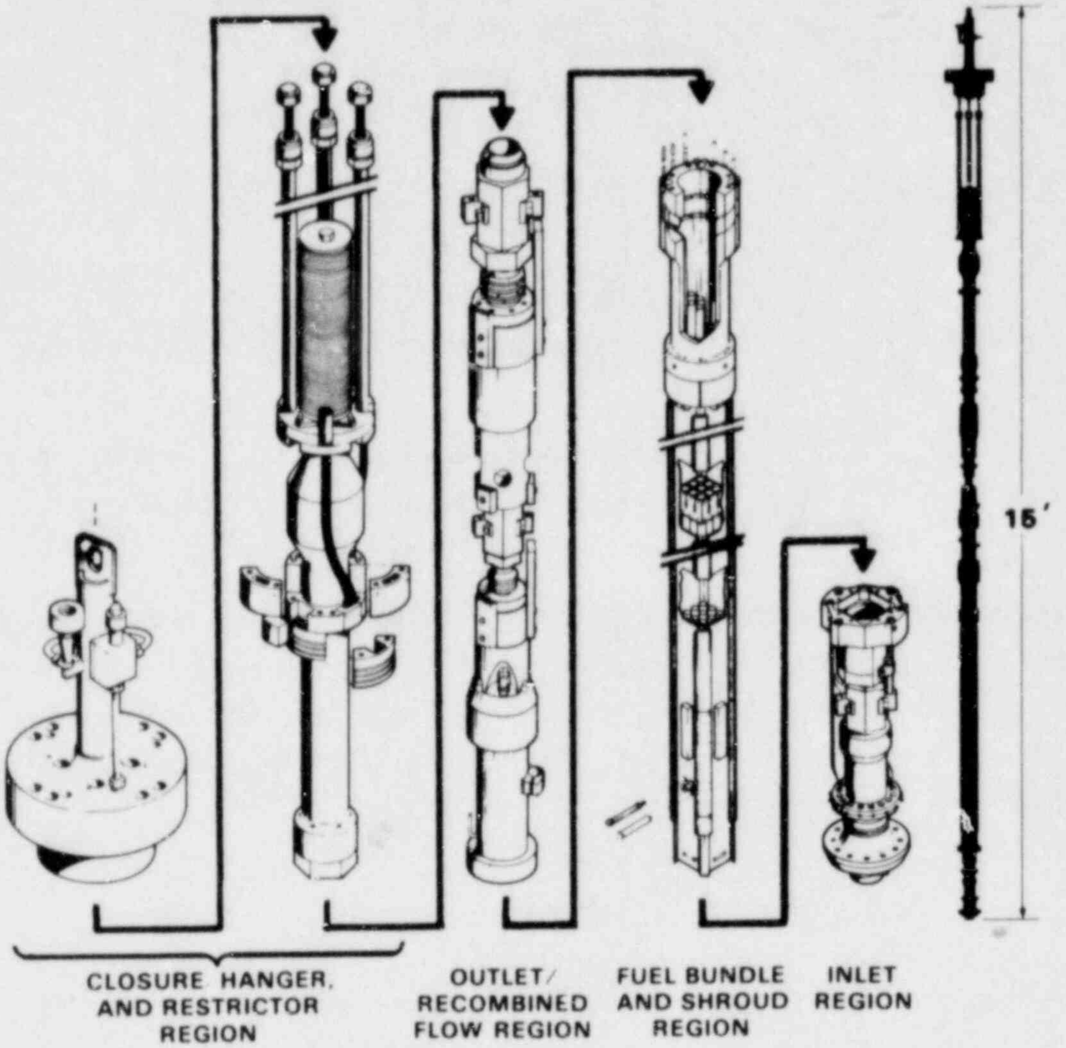

FIGURE 1. 9-Rod Test Train Assembly

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).

TECHNICAL PROGRESS

Special tools, fixtures, and procedures have been developed for assembling this test train in the hot cell and the assembly basin and for transfer among the assembly areas and the PBF test reactor. Much of the hot cell methodology has already been worked out by EG&G. PNL has developed new techniques that involve attaching pressure transducers and applying strain gages to irradiated fuel rods, attaching special fuel rod extensions and marking them (Figure 2), forming the modified rods into fuel bundles held together by spacer grids of commercial design (Figure 3), and then loading the bundle into a split shroud assembly (Figure 4). The split shroud facilitates disassembly for postirradiation examination (PIE). Additional neutron and gamma detection equipment is attached to or located around the shroud in the final test train assembly.

A special support fixture or "paddle" (Figure 5) was built to allow the completed bundle and shroud assembly to be transferred to the MTR assembly basin in a shipping cask called the "White Elephant." A strongback, shown in Figures 6 and 7, is vertically located on a wall of the basin and holds the entire test train. The inlet region of the test train is mounted in the bottom of the strongback, and the bundle and shroud assembly is attached with clamps and screws. Next the outlet region of the test train is placed in the strongback, and an outlet filter screen is mounted on top. The core region instrumentation assemblies and shroud hardline supports are mounted around the shroud. The lower portion of the test train is rotated to the open position, and a cover is placed over the open fuel bundle assembly to prevent foreign objects from falling into it. (This step creates sufficient slack in the hardlines for later reopening). Flow restrictor plugs (part of the seal system) are installed on the bundles of hardlines, which are then clamped to the outlet region at the restrictor seal elevation. Retainer nuts and braze plugs are then mounted on the bundles of hardlines at the closure head elevation. Four bundles of hardlines go up through the hanger bars (tubes); separate bundles are split out and led through adjacent holes or through the center penetration of the closure head. All braze plugs are seated, the hanger bars are attached to the head, and the fuel bundle assembly is rotated to the closed position.

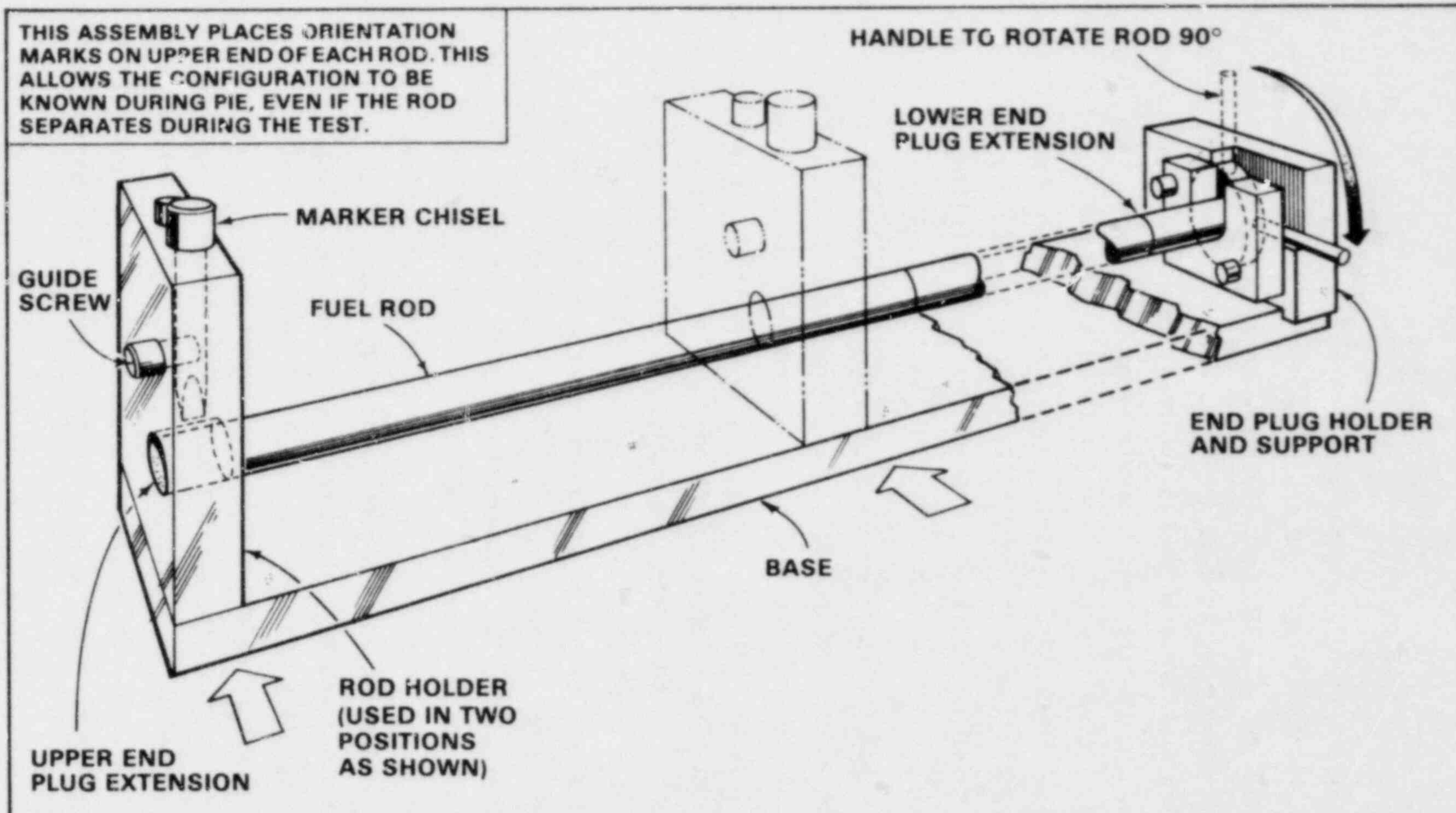

FIGURE 2. Fiducial Marker Fixture

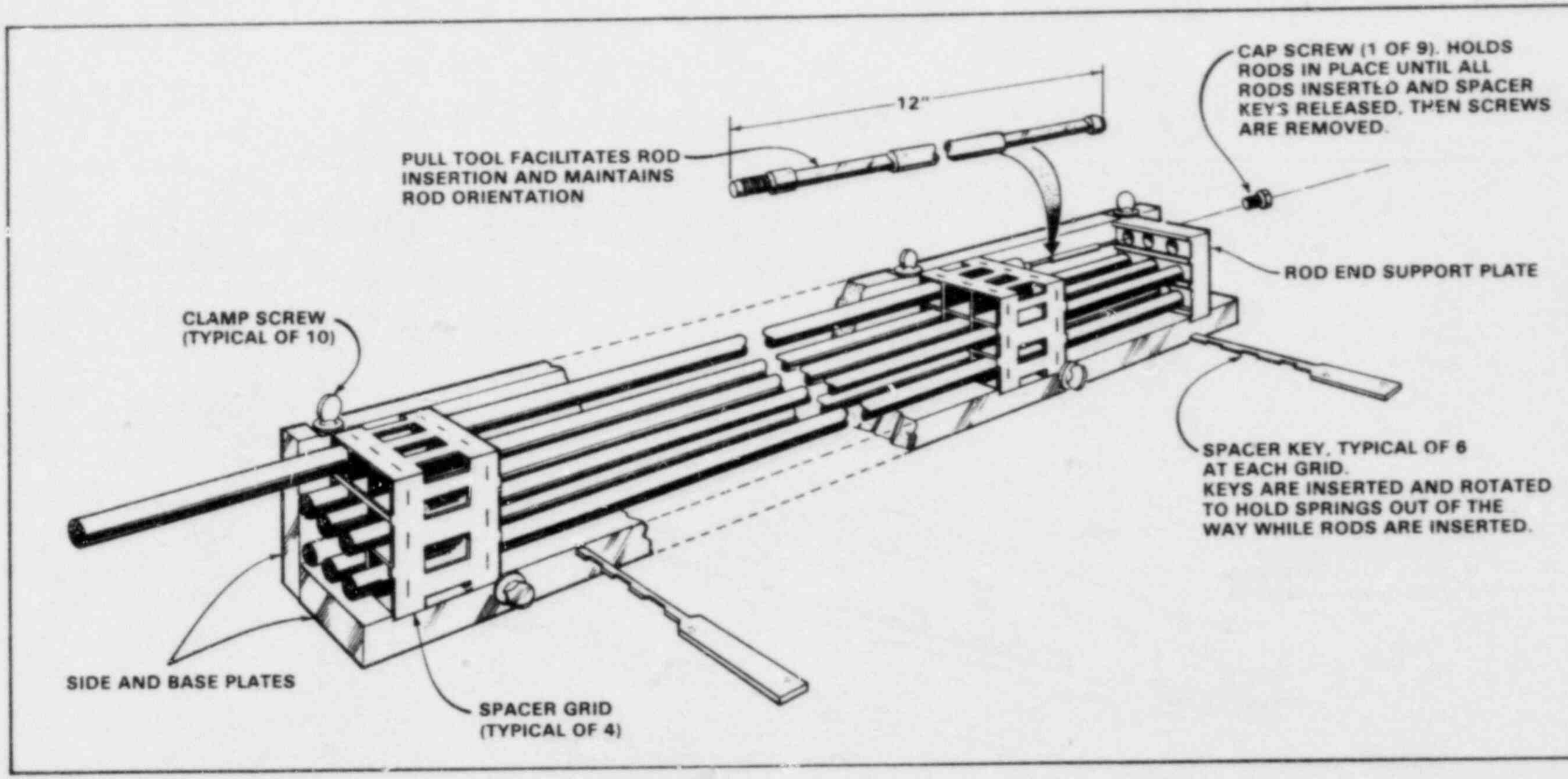

FIGURE 3. Fuel Bundle Loading Fixture

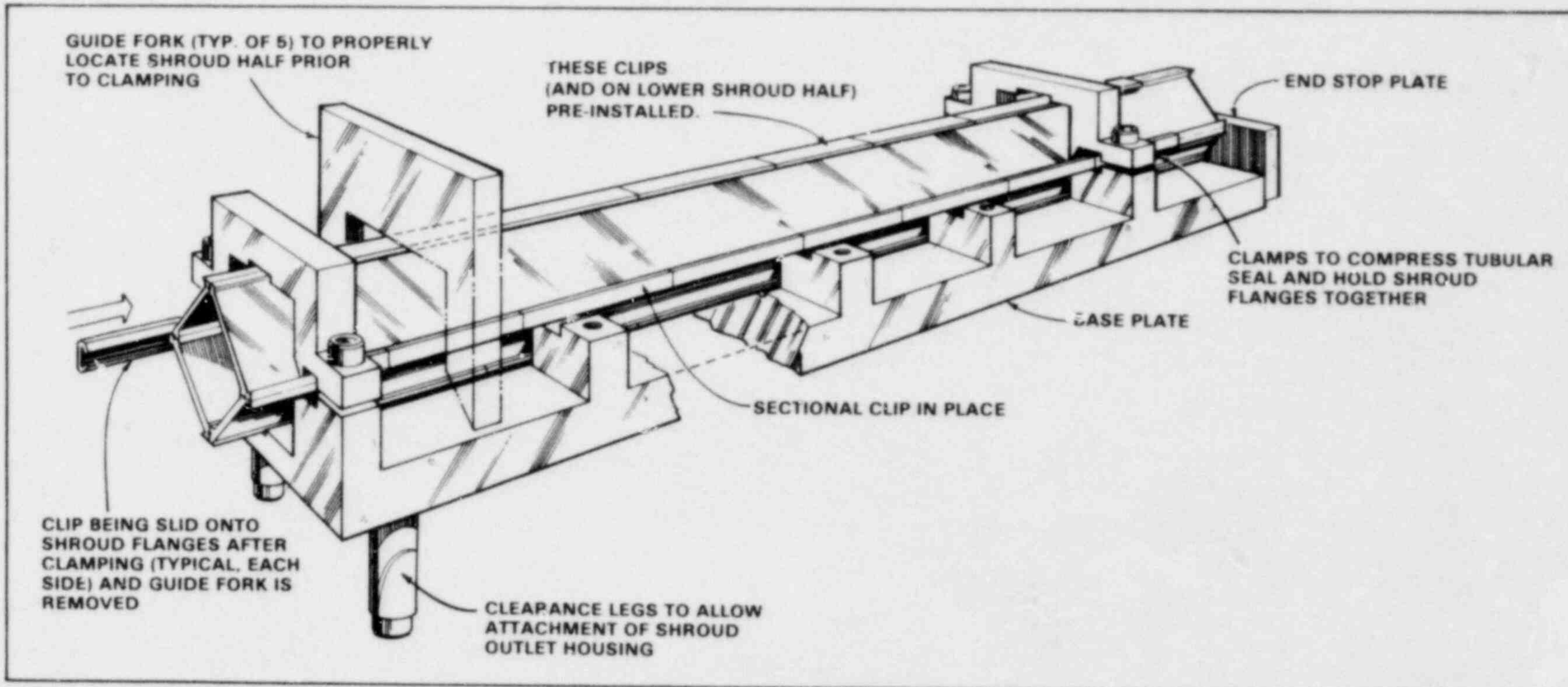

FIGURE 4. Shroud and Fuel Bundle Assembly Fixture

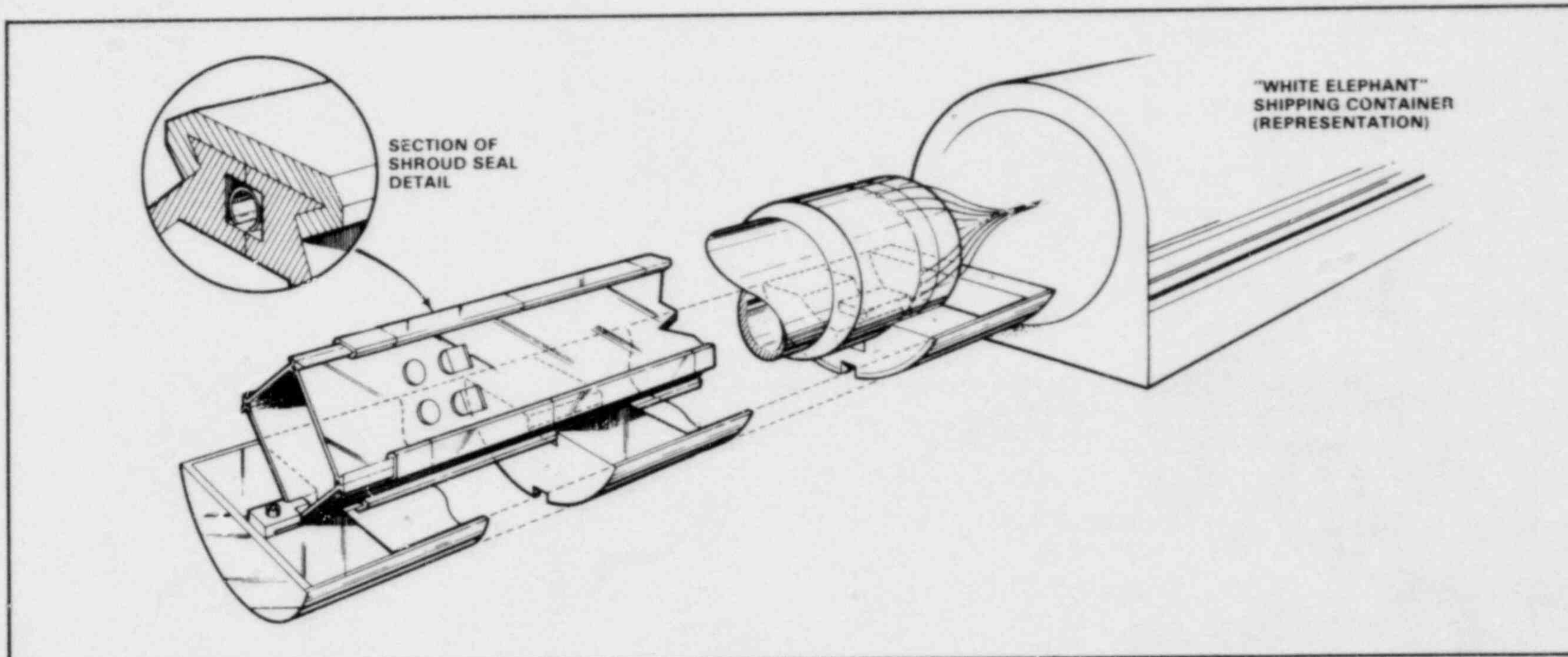

FIGURE 5. Shipping Fixture to Transfer Bundle Containing Irradiated Rods from Hot Cell to Basin

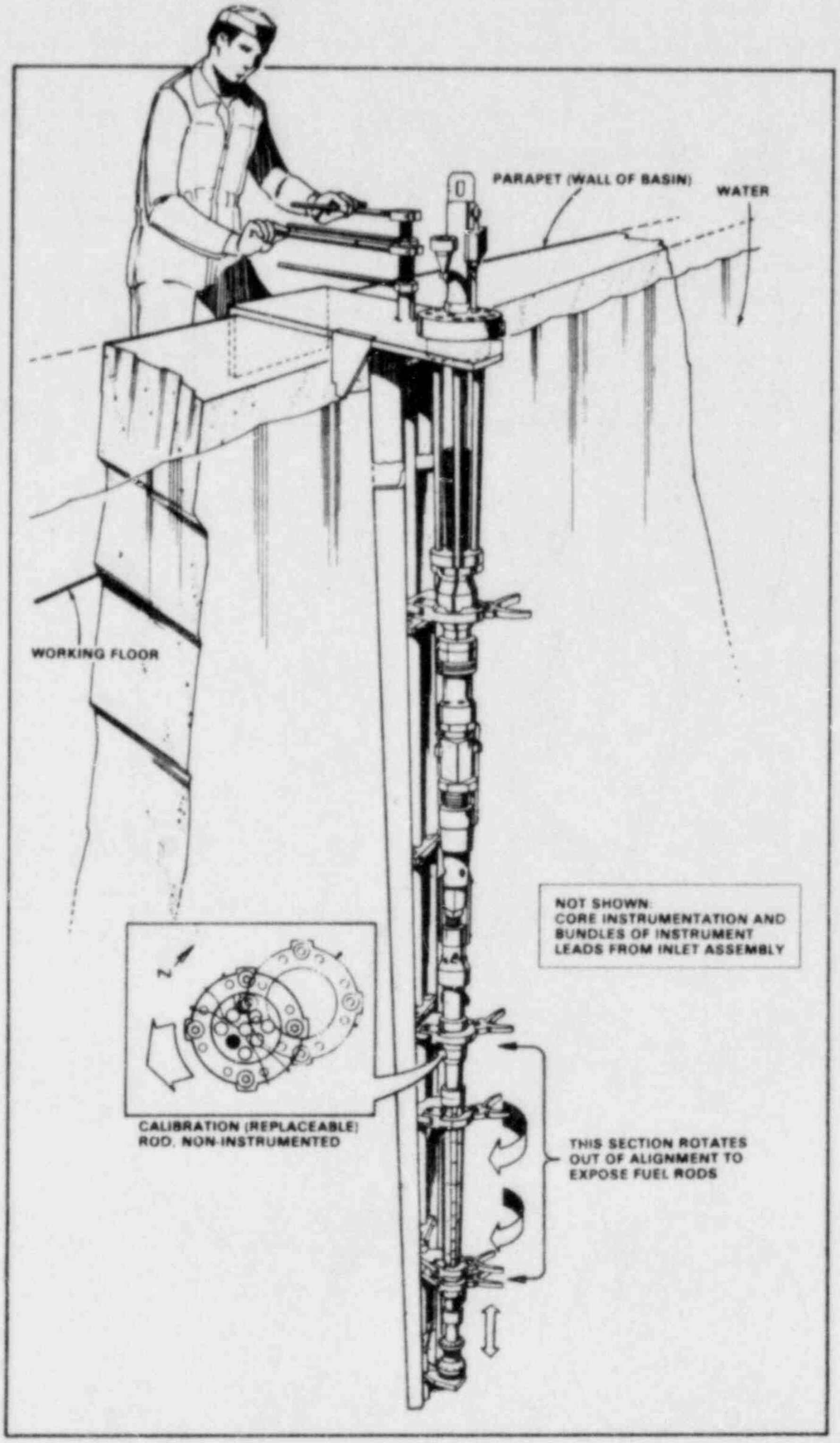

FIGURE 6. Test Train Strongback in Operating Position

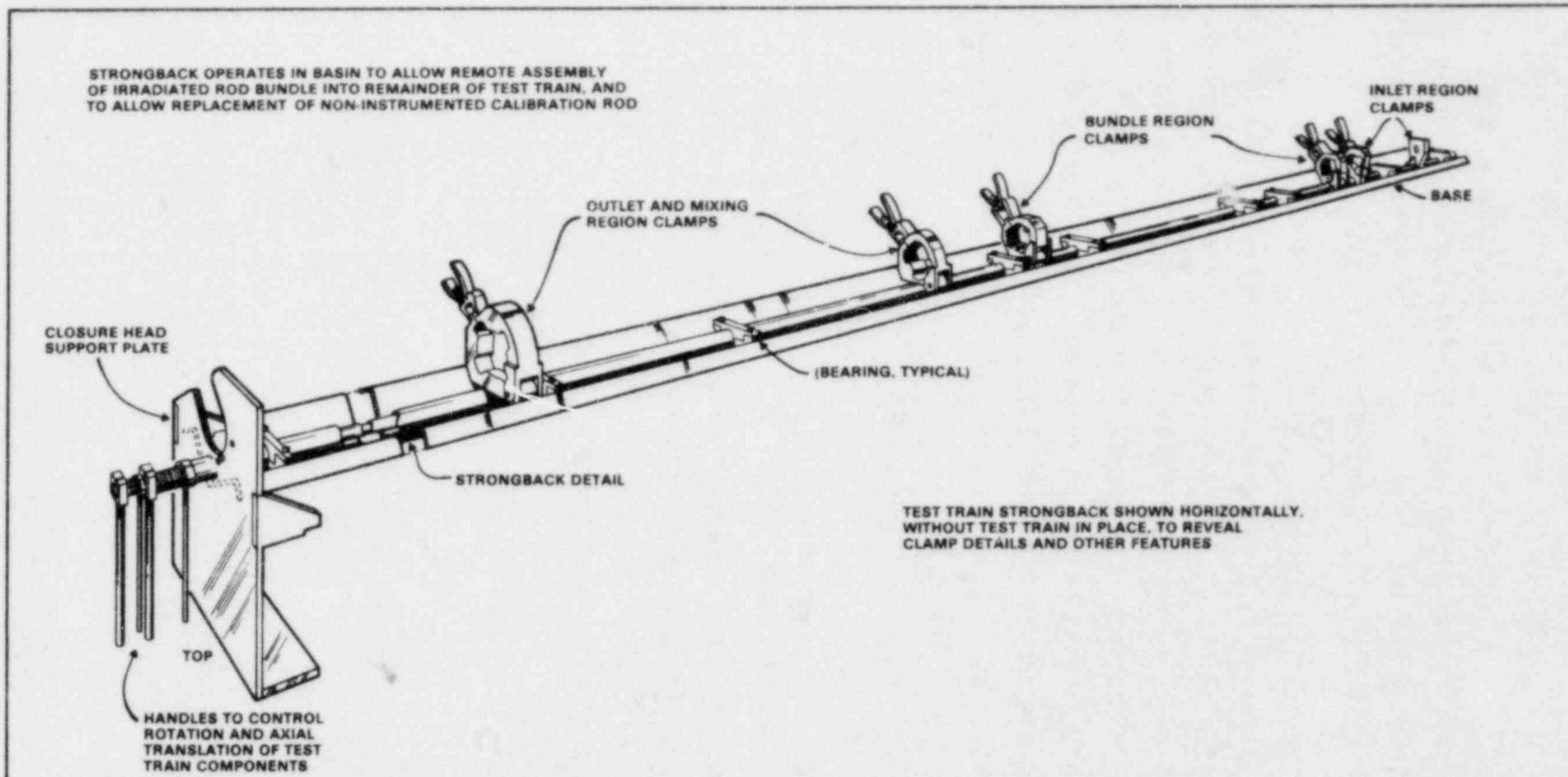

FIGURE 7. Test Train Strongback

The fuel assembly is now attached to the outlet assembly and seal rings are installed at the restrictor elevation. The final step is to tie down all hardlines before releasing the test train from the strongback and raising it into the transporter cask for its journey to the PBF pool.

A feature incorporated in the test train design is the ability to exchange a rod without major disassembly by placing the strongback in the PBF pool, clamping the test train in place, and then detaching the core instrumentation and hardline support tubes so the fuel (with the inlet region) can be unbolted and rotated to provide access to the bundle from the top. A simple handling tool is indexed with a fitting so it can reach down and screw into the end of a noninstrumented rod, extract it from the bundle, and replace it so the test train can be reassembled and returned to the reactor with minimum downtime.

FUTURE WORK

All fixtures and documentation will be provided to EG&G as soon as the remote-handling equipment is checked out. Final fittings on the test train are nearly complete, and test train components and instruments will be shipped when completed. Calibration of neutron and gamma detectors will be done by EG&G. PNL will assist EG&G with strain gage development. PNL will also support assembly and flow tests of the test train by EG&G.

CORE THERMAL MODEL DEVELOPMENT[a]

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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, countercurrent flow limiting (CCFL), void distribution, downcomer, subcooled boiling, nucleate boiling, and post-critical heat flux (CHF) heat transfer experiments. Results obtained thus far indicate that additional work is required in the areas of predicting subcooled CCFL and top reflood.

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

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(a) RSR Fin. Budget No.: B2041; RSR Contact: S. Fabic.

- 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 carry-over in the FLECHT low reflood 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.

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

RENSSELAER POLYTECHNIC INSTITUTE FLAT PLATE PHASE SEPARATION EXPERIMENT

COBRA-TF simulations have previously been reported[1] for two of the RPI air/water phase separation experiments.[2] Both runs had a total inlet mass flow rate of 206.53 lb_m/min and an inlet quality of 0.257% ($\alpha_v = 0.6058$). One run was made with 24 simulated fuel rods (unheated) in the test section; the other run was made without rods. The steady-state void distribution predicted by the code did not agree with the data for either run. This deficiency was attributed to the lack of viscous and turbulent stress modeling in the code. Theoretical work by Drew et al.[3] and Lahey et al.[4] has shown a direct relationship between the local vapor void fraction and the local liquid phase turbulence structure for fully developed two-phase flows in simple geometries. These results imply that a reasonable turbulence model is necessary if two-fluid codes are to predict phase distribution phenomena accurately.

Turbulent stresses have been added to the liquid momentum equations; and turbulent heat flux, to the liquid energy equation. The turbulent terms are modeled using a 3-D mixing length theory. Although it is still in the initial testing stage, this model has greatly improved the code's prediction of RPI data. The RPI "rods in" case was simulated to test the effect of the new turbulent terms. The results are shown in Figure 1, which is a plot of lateral void distribution at four axial levels along the test section. Code predictions are shown as solid lines, and the average value for the data is shown as a dashed line. The overall results are quite good, considering a constant value of mixing length was used for the entire test section. The improvement over the prediction without turbulence is quite noticeable at levels 3 and 4.

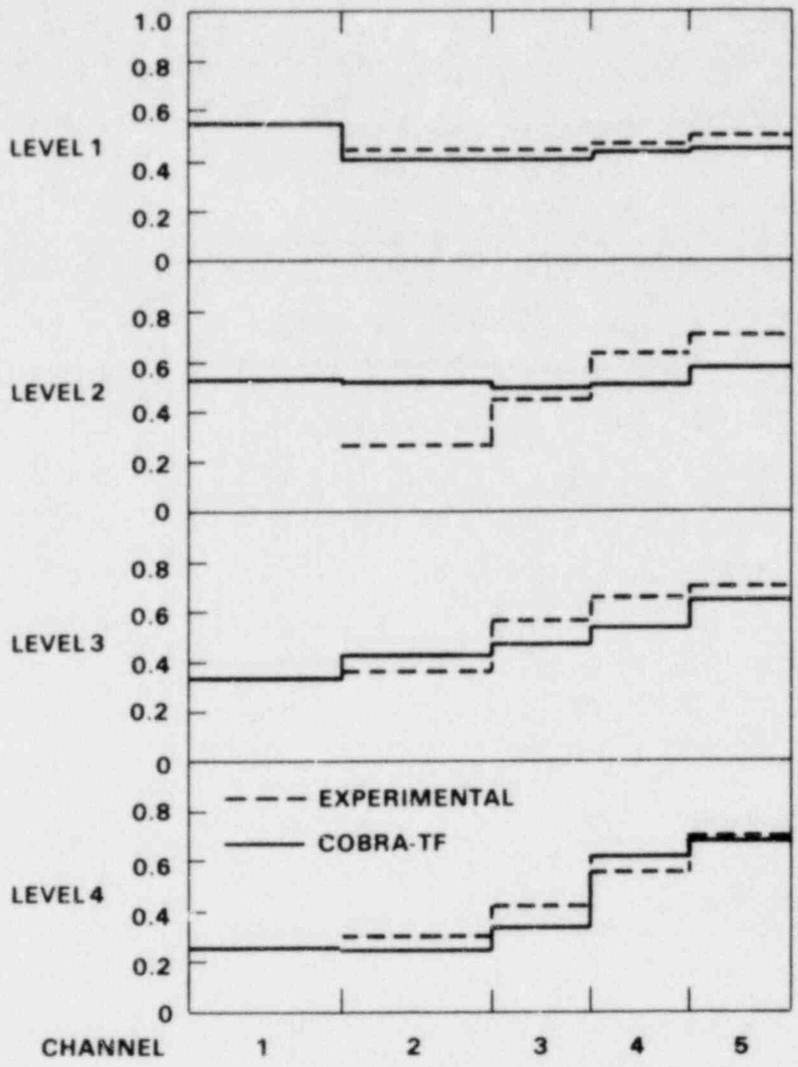

FIGURE 1. Void Distribution with Turbulence

FRIGG BOILING CHANNEL EXPERIMENT

The FRIGG experimental facility consists of an 8-MW loop containing a 36-rod electrically heated bundle. The loop can be operated at both forced and natural coolant circulation. The two-phase pressure drops, axial and radial void distributions, burnout, the natural circulation mass velocities, and stability limit of the loop were measured. COBRA/TRAC predictions of some of the experimental runs were made to assess the code's ability to predict the two-phase pressure drop, axial void distribution, and subcooled boiling measured in the experiment.

A comparison of the code prediction with the measured axial void distribution for a high subcooling, forced flow run is shown in Figure 2. The first 3 meters of the bundle contain subcooled water. It can be concluded that the code has done a very good job predicting the void fraction during subcooled boiling in this experiment. Figure 3 shows the code's prediction of the axial void distribution for a low subcooling, high heat flux natural circulation run. Again, very good agreement between the prediction and the data is observed. This gives an indication of the code's ability to predict the slip between the vapor and liquid phases over a wide range of void fractions. The lower bundle elevations consist of bubbly flow while a drop-film regime exists at the top of the bundle. The code's prediction of the correct two-phase pressure drop and void fraction is reflected in the comparison with the experimental mass velocity versus heat flux curve (see Figure 4).

CREARE DOWNCOMER EXPERIMENTS

Predictions of CREARE downcomer water penetration data have been reported previously.[1] Predictions of higher liquid injection rate data and subcooled water data were completed this quarter. The prediction of a high flow subcooled liquid injection case is shown in Figure 5, which shows that the code has done a very reasonable job in predicting the steam velocities at which complete bypass and complete penetration occur. Prediction of the steady-state downcomer data in general is quite good. The transient downcomer data of Battelle-Columbus is now being simulated.

REFLOOD EXPERIMENTS

Simulations of reflood experiments conducted in the FEBA, FLECHT, and CCTF facilities have been conducted during the past quarter. In general, the code predicted higher cladding temperatures in the upper portions of the bundles than were measured. This was attributed to two main deficiencies in the code:

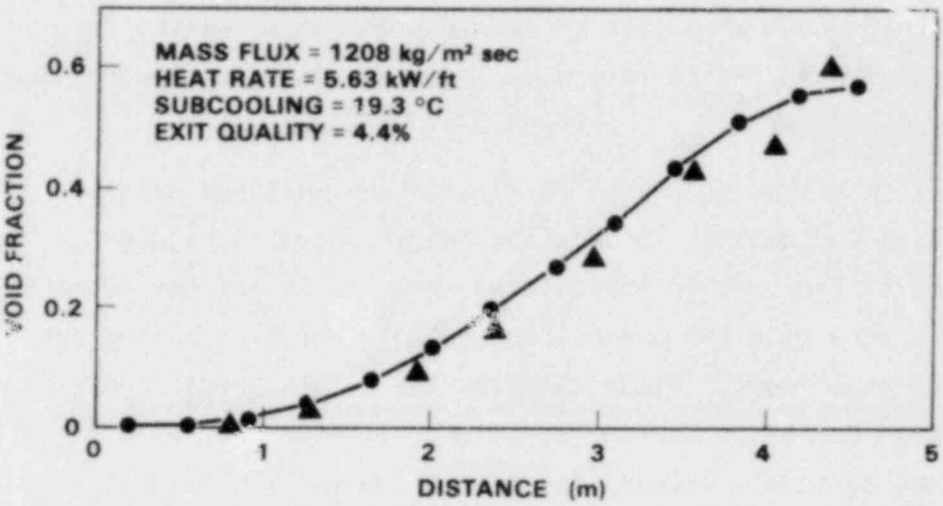

FIGURE 2. FRIGG Axial Void Distribution - High Subcooling

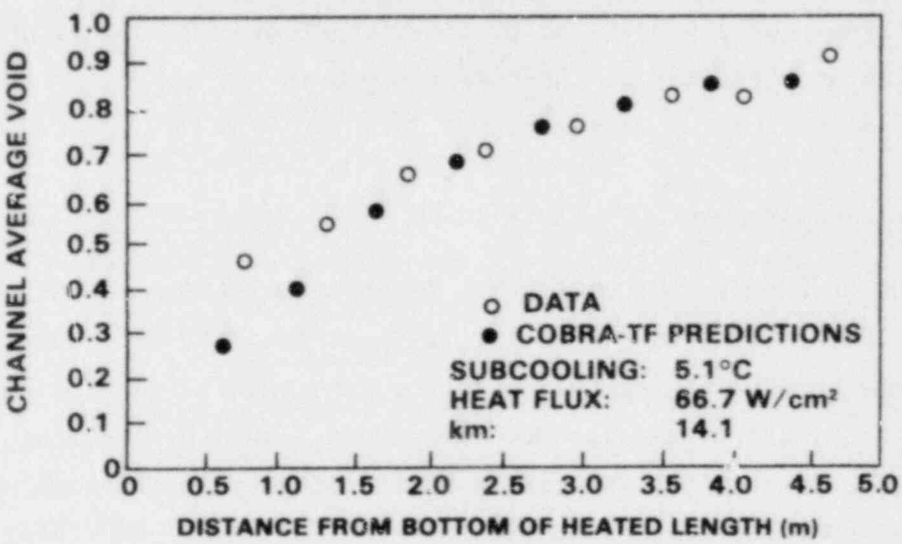

FIGURE 3. FRIGG Axial Void Distribution - High Heat Flux

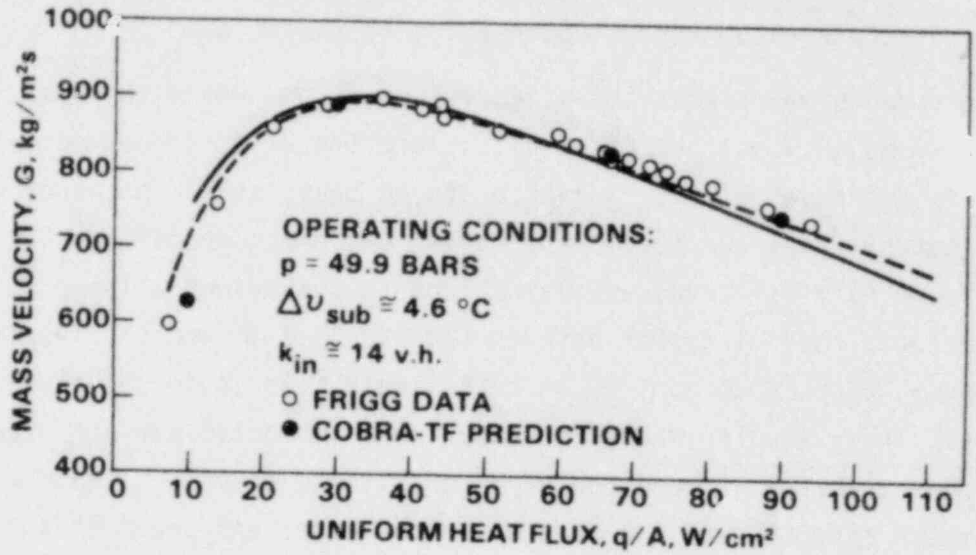

FIGURE 4. FRIGG Mass Velocity Versus Heat Flux

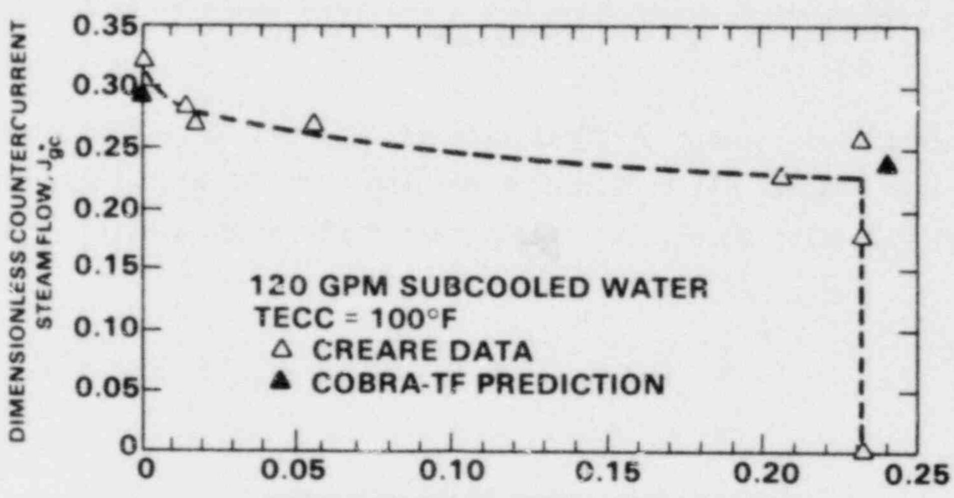

FIGURE 5. CREARE Liquid Penetration Versus Steam Flow

the lack of a rod-to-wall and rod-to-steam radiation model and the effect of grid spacers on drop breakup and desuperheating of the steam.

The grid spacers may affect the steam superheat and hence the clad temperatures in two major ways. First of all, since the grids do not generate heat internally and have relatively little stored heat, it can be expected that they will be wet by drops carried in the steam. As drops deposit on the grid spacers, a liquid film is formed and maintained that provides a large surface area for interfacial heat transfer between the liquid film and the superheated vapor. Secondly, drops that impinge on this liquid film cause splattering of smaller drops. These smaller drops increase the interfacial surface area and enhance the heat transfer between the vapor and liquid phases. Simple models for both of these effects were implemented in the code, and the FEBA and FLECHT simulations were rerun. The predicted clad temperatures at all elevations were in much closer agreement with the data.

FUTURE WORK

Developmental assessment of COBRA-TRAC is expected to be completed during the next quarter. Work will continue on merging COBRA-TF with FRAP, implementing a thermal radiation model, and incorporating the noncondensable gas field.

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LOCA SIMULATION IN NRU[(a)]

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

SUMMARY

The thermal-hydraulic test series was successfully completed during October 1980. A total of 28 tests were conducted within the parameters given below. The draft of the Quick-Look report was released in December 1980.

- reflood delay times - 3 to 66 sec
- reflood rates - 0.7 to 10.5 in./sec
- peak cladding temperatures - 1220^{o}F to 2000^{o}F.

Component fabrication and assembly of the next materials test train continues. Chalk River Nuclear Laboratories (CRNL) estimates the next available test window will occur in March 1981.

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

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

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 6x6 array with the corner pins removed. The bundle design uses commercial enrichments, cladding dimensions, and grid spacers.

TECHNICAL PROGRESS

The first in-reactor tests at CRNL were conducted from October 21 to October 30, 1980. The test train was preconditioned by cycling the reactor power from 5% to 100% in preplanned sequences to induce fuel cracking and check the instrumentation. The test loop was then converted to the pretransient-transient configuration, i.e., a steam-cooling mode for pretransient operation and a controlled pressurized water reflood mode for quench. A typical transient test sequence would be:

- stabilize test train temperature in the steam flow mode
- shut off steam; initiate transient
- initiate reflood water at a controlled rate after preselected delay time.

Preliminary results from the draft Quick-Look test report are shown in the following tables and figures. Table 1 shows the peak cladding temperatures that were measured and predicted for the various reflood rate and delay time combinations selected for the test series. Table 2 compares the predicted and measured quench times.

Figure 1 shows the location of the instruments that were used. Peak power was between levels 13 and 15. Figures 2 through 5 are representative of the results of two of the test series: test 129 and test 107. The peak values and

TABLE 1. Experimental Peak Cladding Temperature

Test Number	Reflood Rate, in./sec	Delay Time, sec	Peak Clad Temperature: Transient, °F	Peak Clad Temperature: Reflood, °F	Peak Clad Temperature at Turnaround: Measured, °F	Peak Clad Temperature at Turnaround: Predicted FLECHT-TRUMP, °F	Peak Clad Temperature at Turnaround: Predicted Therm, °F
101	3.8	28[a]	871	881	1403	1350	1365
104	3.8	37	853	1336	1487	1400	1445
105	1.9	7	858	907	1364	1400	1370
106	10.5[b]	19	873	1101	1223	1100	1150
107	1.9	19	891	1154	1578	1500	1420
108	1.4	11	891	1010	1676	1700	1500
109	1.3	22	865	1158	1881	1800	1580
110	1.9	30	895	1314	1665	1600	1525
111	1.4[c]	11	817	962	1696	1700	1500
112	3.8	37	843	1330	1589	1400	1425
113	7.6	37	845	1408	1526	1400	1395
114	7.6	32	858	1368	1477	1300	1300
115	9.5	66	795	1666	1758	1800	1720
116	3.8	51	836	1500	1707	1600	1605
117	3.8	66	817	1599	1788	1800	1800
118	2.9	52	844	1480	1756	1700	1675
119	2.9	46	862	1451	1673	1600	1620
120	5.9	51	847	1460	1611	1600	1580
121	3.8	36	833	1304	1579	1400	1425
122	7.6	52	866	1486	1611	1600	1575
123	2.9	51	848	1532	1788	1700	1675
124	5.9	52	861	1556	1688	1600	1580
125	1.4	20	872	1138	1802	1800	1565
126	1.2	3	797	800	1644	1700	1530
127	1.0	3	943	966	1991	1900	1650
128	2.0	50	911	1604	1991	1800	1735
129	1.4	32	940	1371	1898	1900	1670
130	0.7[d]	5	929	998	2040		

(a) Unplanned delay caused by problems in prefill.
(b) Malfunctioning equipment caused greater reflood rate than planned.
(c) First 2 sec of data missing.
(d) Reactor tripped at approximately 1850°F.

TABLE 2. Comparisons of Predicted and Measured Quench Times

Test Number	Reflood Rate, in./sec	Delay Time, sec	Quench of Peak Point		Quench of Bundle	
			Measured Time, sec	Predicted Time, sec	Measured Time, sec	Predicted Time, sec
101	3.8	28(a)	85	--	120	180
104	3.8	37	101	150	138	200
105	1.9	7	125	255	155	780
106	10.5(b)	19	42	50	--	--
107	1.9	19	154	290	173	800
108	1.4	11	189	410	276	1800
109	1.3	22	268	440	292	1700
110	1.9	30	173	315	193	810
111	1.4(c)	11	205	410	235	1800
112	3.8	37	109	150	138	200
113	7.6	37	82	115	93	120
114	7.6	32	73	185	83	280
115	9.5	66	107	140	107	160
116	3.8	51	148	180	159	240
117	3.8	66	150	210	163	270
118	2.9	52	170	230	172	320
119	2.9	46	141	220	165	310
120	5.9	51	113	150	120	170
121	3.8	36	122	150	126	200
122	7.6	52	100	1[illegible]5	108	140
123	2.9	51	157	230	165	320
124	5.9	52	113	150	122	170
125	1.4	20	219	440	269	1700
126	1.2	3	207	450	257	>2000
127	1.0	3	344	625	344	>2000
128	2.0	50	162	355	220	830
129	1.4	32	216	470	287	1700
130	0.7(d)	5				

(a) Unplanned delay caused by problems in prefill.
(b) Malfunctioning equipment caused greater reflood rate than planned.
(c) First 2 sec of data missing.
(d) Reactor tripped at approximately 1850°F.

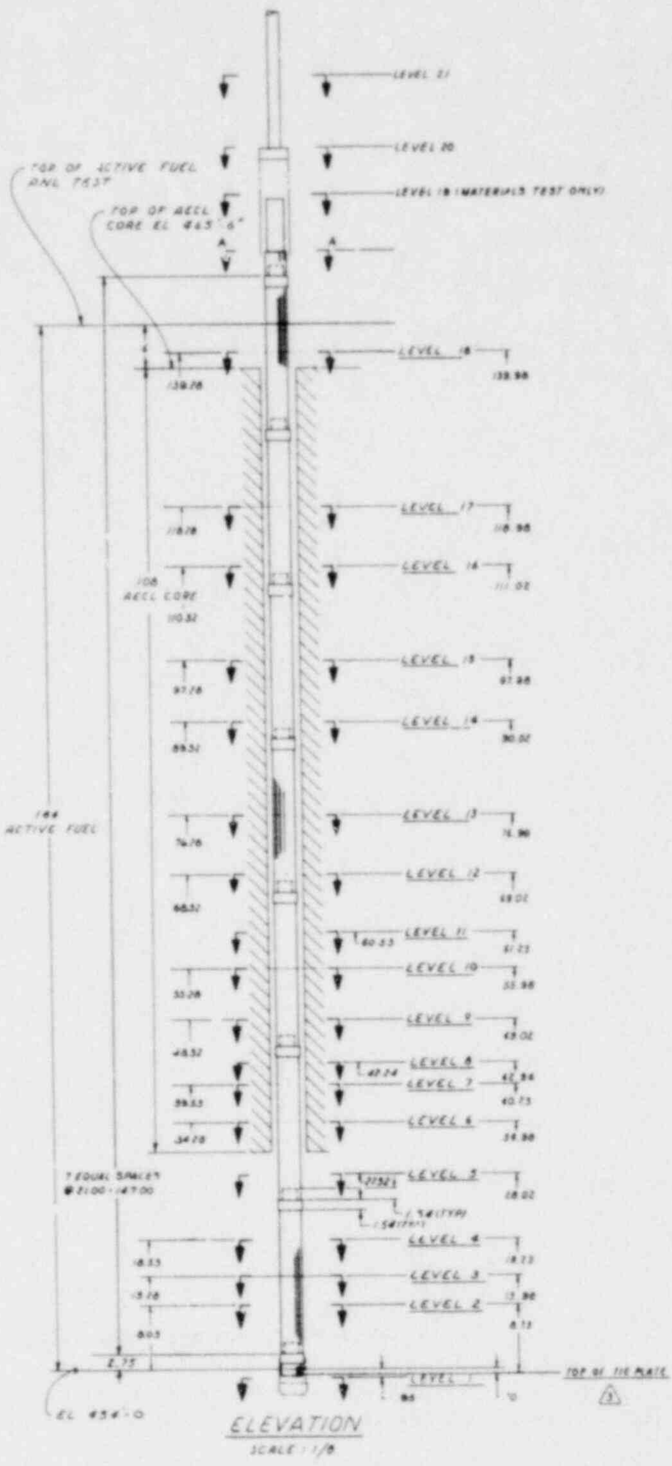

FIGURE 1. Test Train Instrumentation Layout

POOR ORIGINAL

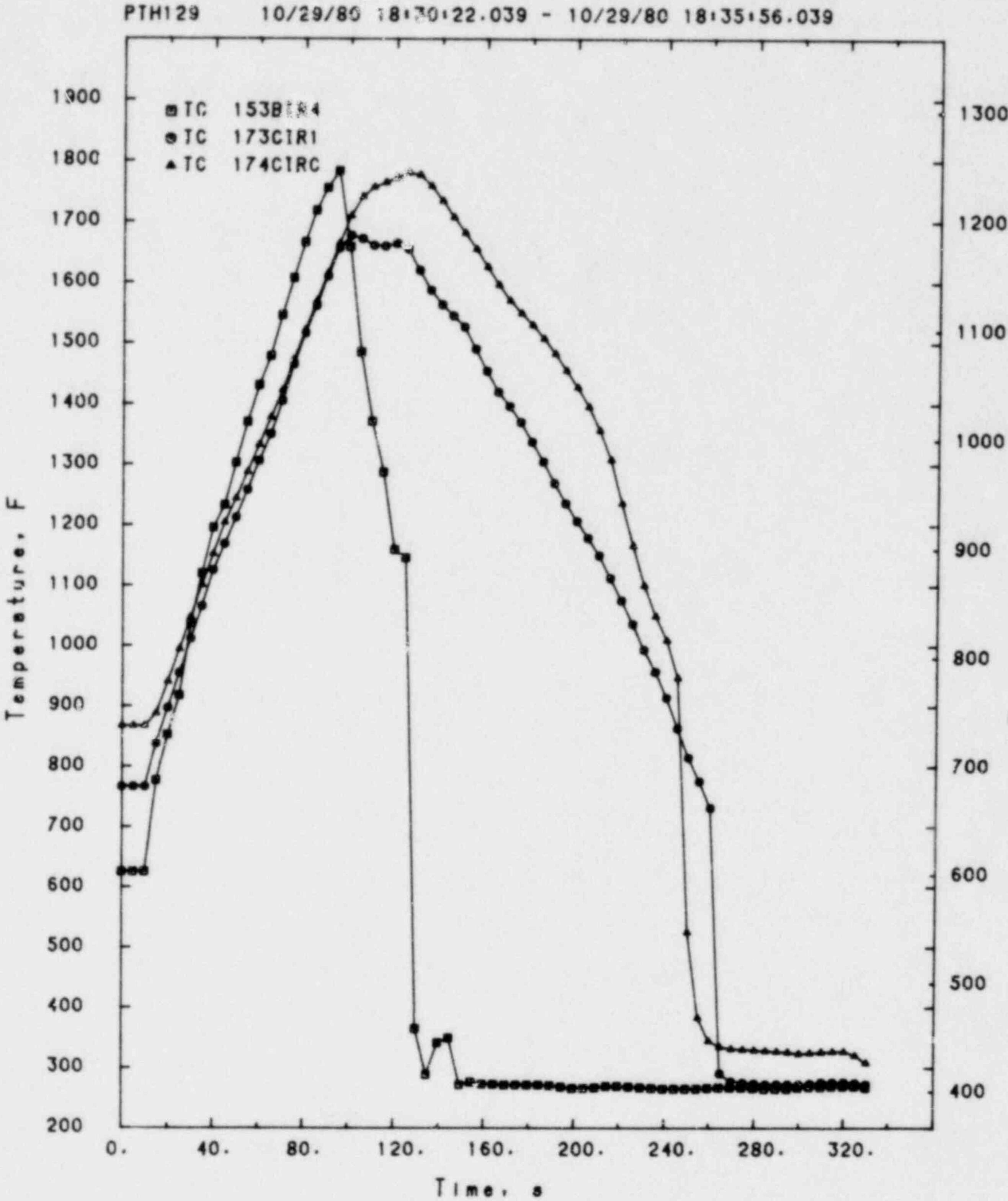

FIGURE 2. Test Fuel Rod Center and Interior Cladding Temperature Histories During Transient PTH129

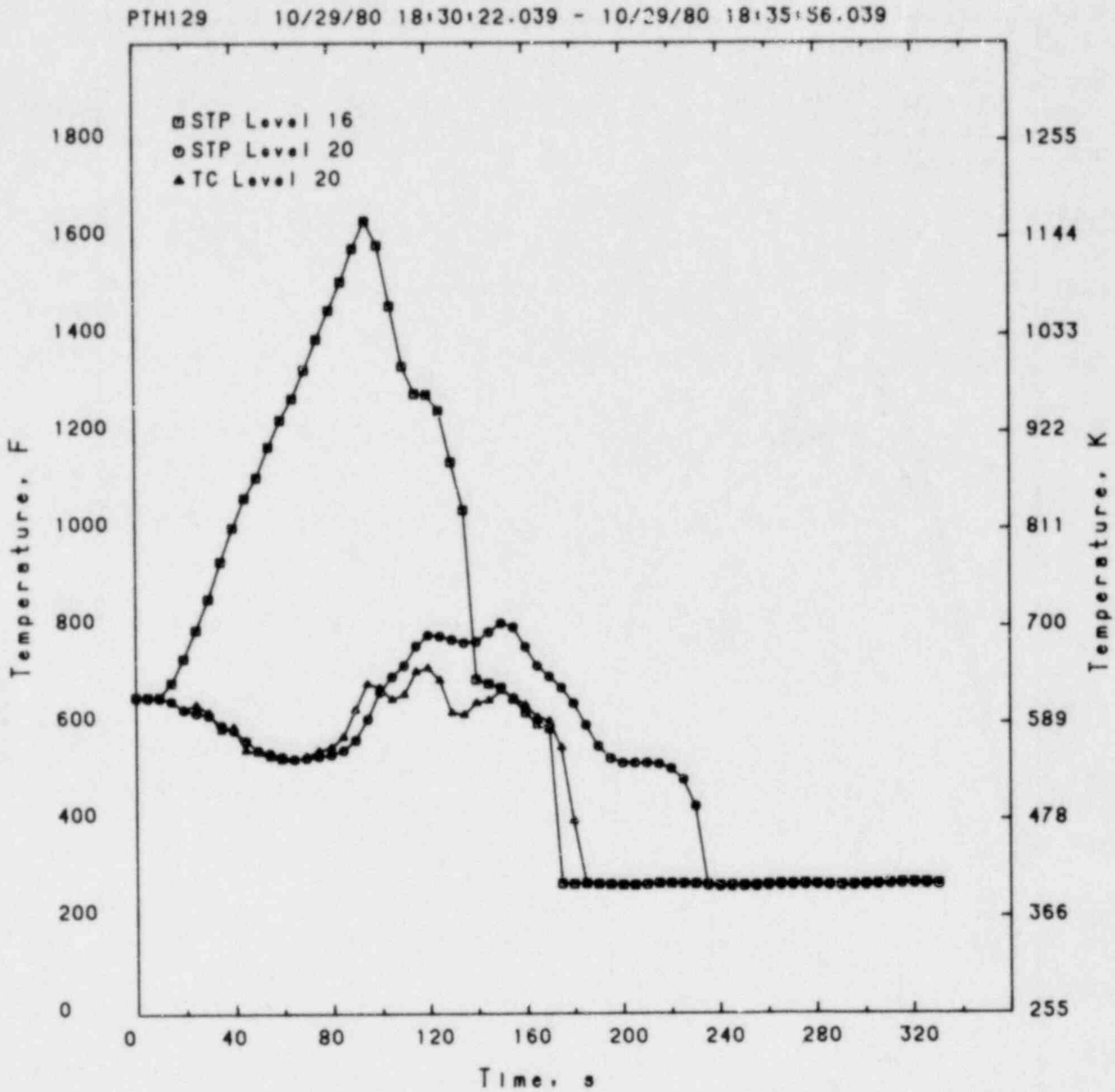

FIGURE 3. Steam Temperature Probe History During Transient PTH129

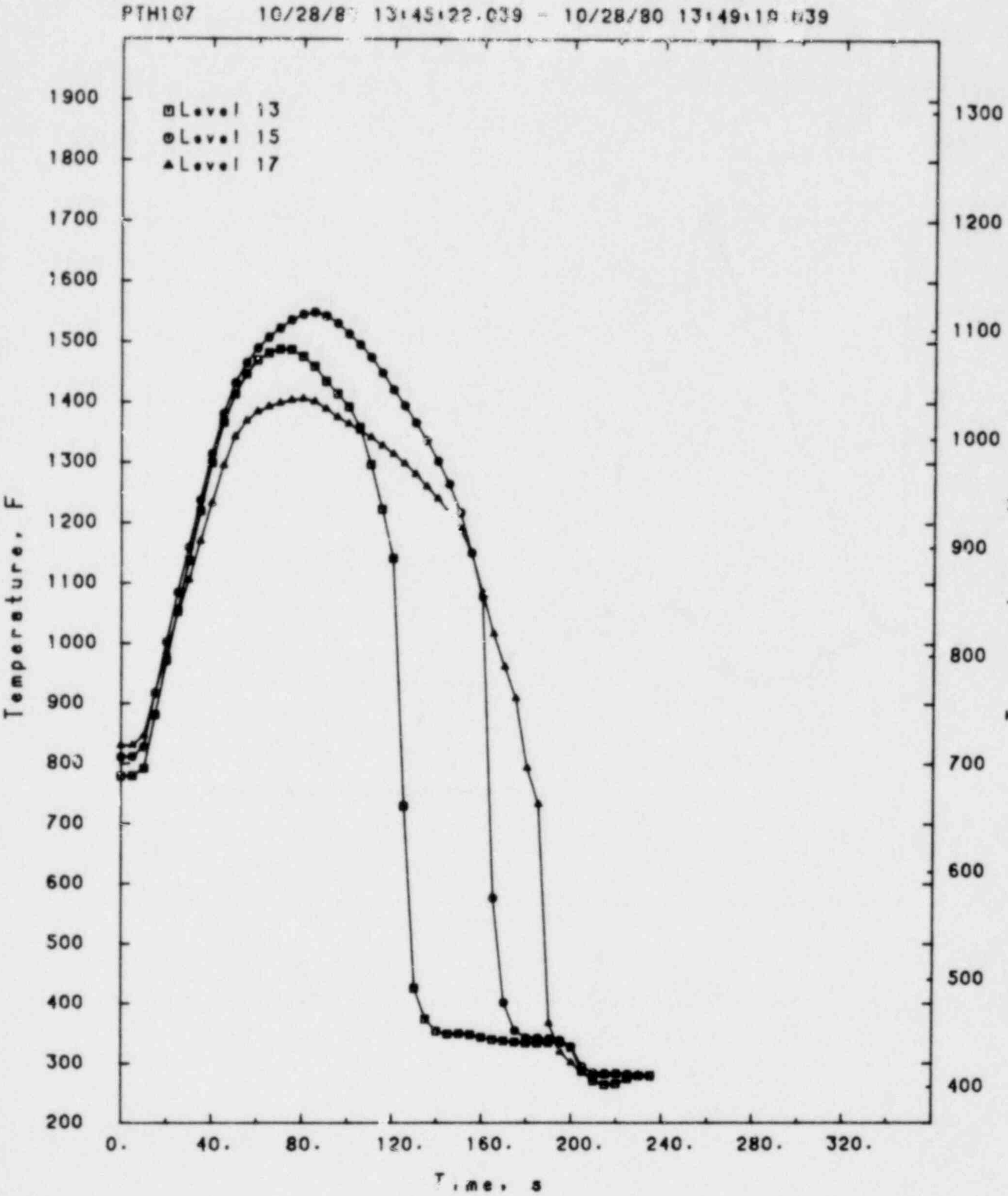

FIGURE 4. Average Guard Fuel Rod Cladding Temperature Histories (interior thermocouples) During Transient PTH107

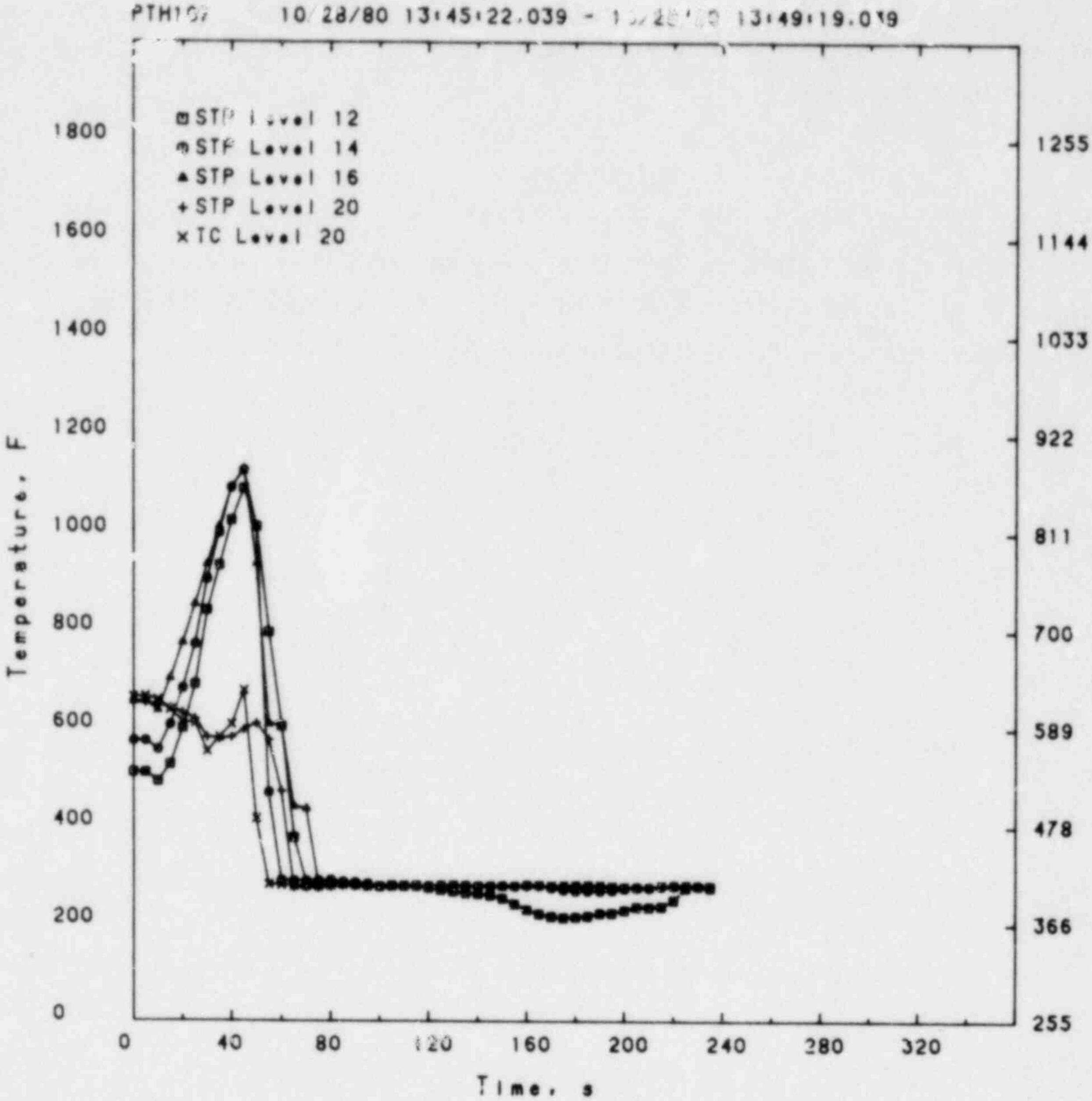

FIGURE 5. Steam Temperature Probe History During Transient PTH107

quench times do not correspond with the values stated in Tables 1 and 2 because the instrument readings selected for the graphs were not located at the peak temperature locations.

FUTURE WORK

The test train for Materials Test 1 is being fabricated and will utilize pressurized fuel rods in the cruciform bundle. The assembly and checkout of the disassembly, examination, reassembly machine (DERM) will continue.

STEAM GENERATOR TUBE INTEGRITY[a]

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SUMMARY

The construction contract for the Steam Generator Examination Facility (SGEF) was let to George A. Grant, Inc.; and ground was broken for construction. Efforts were initiated to establish a Steam Generator Project Office at Pacific Northwest Laboratory (PNL). The program task on providing a data management system was started. Research continued on specimens for stress corrosion crack (SCC) characterization, intergranular attack in the tube sheet region, 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 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

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

fully nondestructively characterized by means of positive replication and various nondestructive testing (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 subsequently 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.

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 6 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 in the SGEF.

(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.

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. Research will be conducted in the following areas:

- NDT technique verification and instrument development
- defect matrix identification
- profiling of defect types, extent, and locations
- identification of deposits and sources of corrosion
- verification of component integrity
- health physics.

The generator will also become a source of specimens with service-induced flaws for various NRC programs. Potential future research phases include simulated operation of a portion of the generator to assess long-term effects of possible chemical cleaning or decontamination procedures on generator serviceability that may be proposed to NRC for licensing. Study of the recovery and reuse of materials in decommissioned reactor components is another potential research task.

TECHNICAL PROGRESS

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

CHEMICALLY PRODUCED STEAM GENERATOR TUBE DEFECT SIMULATIONS

Tube Sheet Crevice Specimen

A specimen was designed to simulate defects in the tube sheet crevice region and will be used in NDT instrumentation validation studies and leak rate tests. A specimen consists of a steam generator tube section with SCCs, regions of intergranular attack, and machined defects along its length. This tube section is inserted into a machined tube sheet simulation and the end is rolled into place. The innovative aspect of the specimen involved producing a region of intergranular attack on the Inconel 600 steam generator tubing while retaining the grains in place. This simulates intergranular attack on steam generator tubes in the tube sheet crevice, where it is postulated that corrosion product and/or sludge buildup prevent the grains from dropping out. Conventional eddy current NDE, which depends on metal loss for signal generation, is thus relatively ineffective at detecting this type of defect. Two specimens were shipped this quarter to Oak Ridge National Laboratory (ORNL) for use on another NRC-NDT program.

Stress Corrosion Cracked Specimen

Efforts are continuing at definitive characterization of laboratory-produced intergranular SCC (IGSCC) in Inconel 600 steam generator tubes. A group of 10 round robin specimens was nondestructively characterized in Germany and at ORNL this past quarter. After one further quarter of examinations the tubes will be destructively assayed for comparison with NDT data.

Leak Rate Specimen

Tube specimens with through-wall SCCs of various lengths and orientations are being fabricated for leak rate tests. The specimen matrix of axial cracks is essentially completed, and efforts are concentrating on producing a circumferentially oriented IGSCC. Figures 1 and 2 show the device designed to stress the specimen to produce circumferential cracking using our caustic corrosion autoclaving technique.[1]

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.

Steam Generator Examination Facility (SGEF)

George A. Grant, Inc., was awarded a 360-day fixed price ($1,444,000) contract to construct the SGEF. The completion date will therefore be on or before November 19, 1981. SGEF groundbreaking occurred on December 22, 1981; Figure 3 shows the SGEF site at groundbreaking, and Figure 4 shows the construction status at the end of the quarter. The Department of Energy-Richland

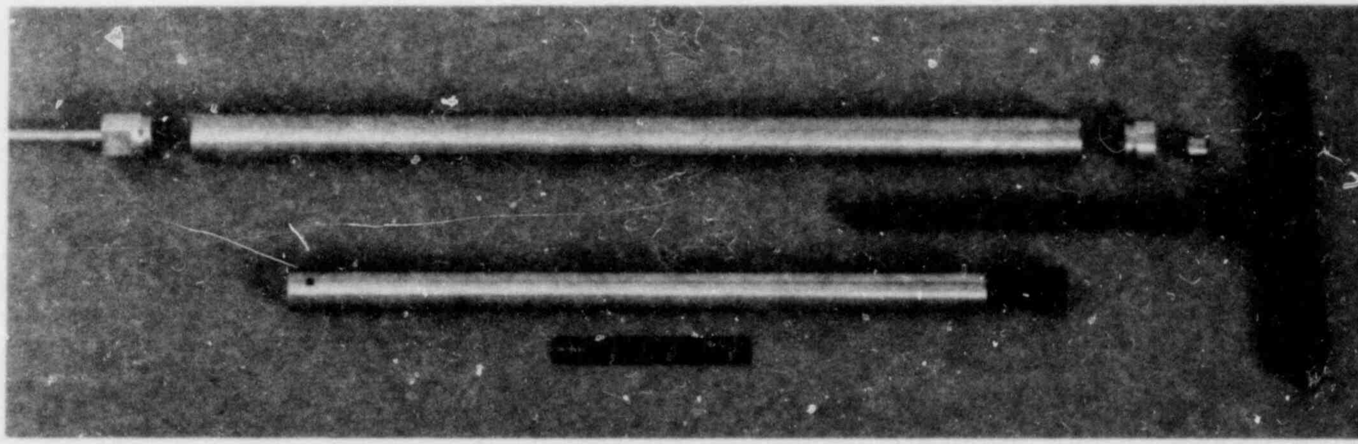

FIGURE 1. Disassembled Components for Stressing a Steam Generator Tube to Produce Circumferential Stress Corrosion Cracking

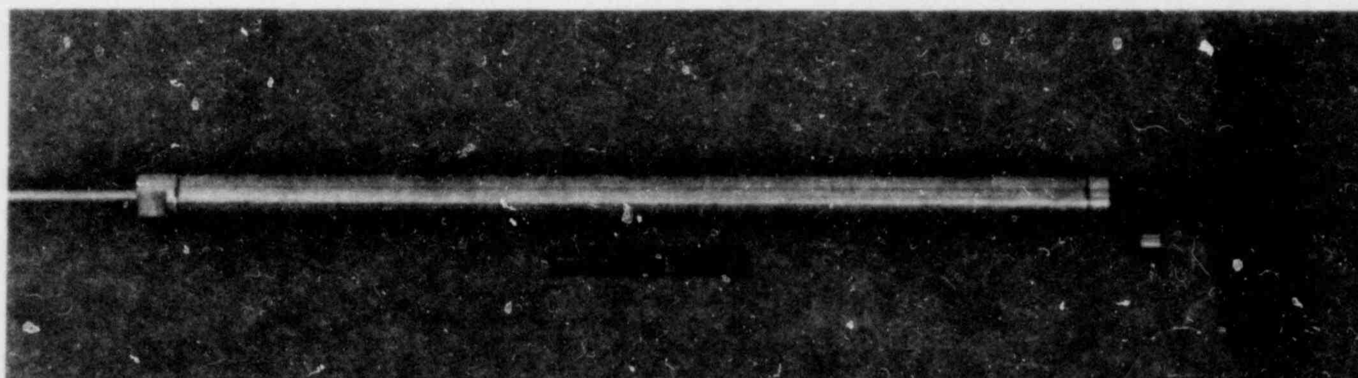

FIGURE 2. Assembled Components from Figure 1 Showing Application of Loading Prior to Seal Welding Specimen

FIGURE 3. Excavation Progress for Steam Generator Examination Facility as of December 22, 1980

FIGURE 4. Excavation Progress for Steam Generator Examination Facility as of December 28, 1980

Operations (DOE-RL) has negotiated the Title III (as-built) design contract with Facilities System Engineering Corporation (FSEC).

Generator Research Tasks

Work was initiated on the data management system task, which is to provide computer software and arrange for use of computer hardware to handle information derived from research on the Surry generator. The completed data system is to be tested prior to initiation of research on the generator. The system will be designed to allow easy access for statistical evaluation of data and will incorporate computer graphics to allow data to be presented in sectional locations through the generator.

Management Activities

Contacts were made with several parties to initiate efforts to obtain a broadened sponsorship through a Steam Generator Project Office located at PNL. Representatives of industrial and government organizations from West Germany, France, the United Kingdom, and Japan were contacted; responses from these parties are expected early in the first quarter of 1981.

A draft of the contract proposed for use by the Steam Generator Project Office was forwarded to NRC for comment. This draft contract establishes how the research project would be operated under multisponsorship.

Basic ordering agreements are being negotiated with domestic PWR vendors to provide efficient access to vendor-developed software and hardware that could be used in research aspects of the project. It is our intent not to develop hardware for performing tasks where such equipment exists and is available in a timely and economic manner.

The first part of the two-part Phase II Program Report has been completed in draft and is undergoing internal review. NUREG/CR-1626, "Eddy Current Inspection of Inconel-600 Steam Generator Tubes at the Tube Sheet," was published this quarter.

Milestones

- Title III design contract placed with FSEC.
- SGEF construction contract placed with George A. Grant, Inc.
- SGEF groundbreaking.

Problems

Delay in letting of the construction contract for the SGEF has resulted in an approximate 5-month slippage from the originally scheduled completion date. This will result in an approximately 2-month slippage in planned availability of the generator for research.

FUTURE WORK

During the coming quarter the following activities will be pursued:

- continued discussions with potential sponsors for joint participation in the Surry generator examination
- a talk on the Surry generator research program to be presented at the Golden Gate Metals Conference
- continued specimen preparation and testing activities on leak rate and SCC specimens
- activities leading to establishment of a Steam Generator Project Office
- the first section of the Phase II Report will be forwarded to NRC.

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RESIDENT ENGINEER AT CADARACHE, FRANCE[a]

D. S. Trent, Project Manager
C. L. Wheeler, Resident Engineer

This program is an ongoing reactor safety experimental program being conducted by the French Atomic Energy Commission; however, Pacific Northwest Laboratory's participation in the program ceased as of September 30, 1980. The resident engineer has returned, and no further contributions will be made to this report.

(a) RSR Fin. Budget No.: B2278; RSR Contact: G. P. Marino.

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16. ABSTRACT (200 words or less)

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from October 1 through December 31, 1980, for the Division of Reactor Safety Research within the Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining structural graphite strength, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the remaining integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Test 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 and post-accident coolability tests for the ESSOR Test Reactor 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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