NUREG/CR-1454
PNL-3380-2
Vol. 2

Reactor Safety Research Programs

Quarterly Report
April - June 1980

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
**U.S. Nuclear Regulatory
Commission**

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

Available from

GPO Sales Program
Division of Technical Information and Document Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

Reactor Safety Research Programs

Quarterly Report
April - June 1980

Manuscript Completed: October 1980
Date Published: November 1980

Prepared by
S. K. Edler, Ed.

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Reactor Safety Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN Nos. B2101, B2088, B2289, B2043,
B2084, B2041, B2277, B2097, B2278

ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 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 postaccident coolability tests for the ESSOR Test Reactor Program, Ispra, Italy; blowdown and reflood tests in the test facility at Cadarache, France; 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.

CONTENTS

ABSTRACT	iii
GRAPHITE NONDESTRUCTIVE TESTING	1
ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS	7
INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS	27
EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES: TASK A - IRRADIATION EXPERIMENTS	39
EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES: TASK B - DATA QUALIFICATION AND ANALYSIS	41
EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES: TASK C - CODE COORDINATION AND EX-REACTOR TESTING	51
FUEL SUBASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM	55
CORE THERMAL MODEL DEVELOPMENT	81
LOCA SIMULATION IN NRU	93
STEAM GENERATOR TUBE INTEGRITY	101
RESIDENT ENGINEER AT CADARACHE, FRANCE	115

FIGURES

GRAPHITE NONDESTRUCTIVE TESTING

1	Density Versus Oxidation Depth for 3.0-in. Diameter Sample No. VII-B-A-9	3
2	Density Versus Oxidation Depth for 3.0-in. Diameter Sample No. VI-C-4	4

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

1	Comparison of Two Acoustic Emission Sensors	12
2	Typical Response of Broad Band Wave-Guide to Background Noise	13
3	Least Squares Decision Rule with Five Features	15
4	Least Squares Decision Rule with Six Features	16
A.1	Flaw Locations in A533B Insert (all dimensions in mm)	20

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS

1	Thermal Fatigue Crack--19 mm deep, 64 mm long (0.75 in. deep, 2.5 in. long)--in a Welded Cast Stainless Steel Pipe Sample	29
2	Measured and Theoretical Reduction in Ultrasonic Inspection Sensitivity Resulting from Calibrations According to 1977 Code (notch) Relative to 1974 Code (side-drilled holes-SDH).	31

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES TASK B - DATA QUALIFICATION AND ANALYSIS

1	Gas Pressure Data for Rods 1, 3, and 5 of IFA-513	46
2	Gas Pressure Data for Rods 2, 4, and 6 of IFA-513	47
3	Internal Pressure as a Function of Time During Pressure Spikes in Rod 5, IFA-513	48

FUEL SUBASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM

1	Transverse View of the PBF R-Z TRUMP Model	59
2	Axial View of the PBF R-Z TRUMP Model	60
3	Cladding, Steam, and Shroud Temperatures Calculated for Low-Conductivity Shroud (0.14 W/m-K, 0.083 Btu/hr-ft-°F) for Various Assembly Peak Powers	62
4	Cladding, Steam, and Shroud Axial Temperature Distribution for Low-Conductivity Shroud (0.14 W/m-K, 0.083 Btu/hr-ft-°F) for an Assembly Peak Power of 11.48 kW/m (3.5 kW/ft)	63
5	Illustrative Cladding Temperature Dependence on Heat Transfer Coefficients	65
6	Cladding, Steam, and Shroud Temperature Calculated for High-Conductivity Shroud (1.69 W/m-K, 1.0 Btu/hr-ft-°F) for Various Assembly Peak Powers	67
7	Cladding, Steam, and Shroud Axial Temperature Distribution for High-Conductivity Shroud (1.69 W/m-K, 1.0 Btu/hr-ft-°F) for an Assembly Peak Power of 105 kW/m (32 kW/ft)	68
8	Severe Fuel Damage Program Test Assembly Instrumentation	72
9	Layout of the Super Sara In-Reactor Pipework	74
10	Super Sara LOCA Simulation Loop Arrangement	75
11	Super Sara Small-Break Test Assembly Concept	76

CORE THERMAL MODEL DEVELOPMENT

1	Dartmouth Countercurrent Flow Experiment	88
2	Dartmouth Countercurrent Flow Experiment	89
3	NWU CCFL Experiments, $D_{BUB_{MAX}} = D_{HYD}$	91
4	NWU CCFL Experiments $D_{BUB_{MAX}} = 2 \cdot D_{HYD}$	91

LOCA SIMULATION IN NRU

1	Fuel Powder is Compressed to 95% $\pm$ 1.5% of Theoretical Density	95
2	Sintered Pellets are Ground on a Centerless Grinder to Final Dimensions	96
3	Slotted Pellets are Inserted Between the Thermocouples in the Instrumented Rod	97
4	Lower Rod End Cap is Arc Welded in a Helium Atmosphere	98
5	Steam Probe Mounting Bracket is Attached to the Grid Spacer	99

STEAM GENERATOR RUBE INTEGRITY

1	The Generator as It Is Being Pulled from the Storage Vault at the Surry Nuclear Station	107
2	Loading the Generator on Land Transport at Surry	108
3	Land Transport to the Surry Nuclear Station Dock on the James River	109
4	Generator (from the transition cone end) on the Barge Just Prior to Shipping	110
5	Channel Head End of Generator as It Appeared Just Prior to Shipping	111
6	Generator Docking at Port of Benton (adjacent to the Hanford Project)	112

TABLES

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

1	Revised Test Sequence for Vessel ZB1	9
2	Revised Flaw Geometries - ZB1 A533B Insert	10
A.1	Proposed Test Sequence for Vessel ZB1	21
A.2	Proposed Flaw Geometries - ZB1 (A533B Material Insert)	22

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES
TASK B - DATA QUALIFICATION AND ANALYSIS

- 1 Summary of Pressure Increases and Estimated Gas
Release for IFA-513 49

FUEL SUBASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM

- 1 Summary of Instrument Measurement Needs for the PBF Severe Fuel
Damage Experiments 69
- 2 PBF Severe Fuel Damage Program Test Assemblies - Proposed
Instrumentation Package 70

GRAPHITE NONDESTRUCTIVE TESTING(a)

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

SUMMARY

Oxidation of all PGX samples and the evaluation of the performance of surface-wave equipment have been completed. A strong signal is transmitted at low levels of oxidation; however, rapid attenuation of the signal and distortion of the waveform are observed at higher levels of oxidation. Physical measurements of density profiles confirm previous visual observations that there is an increased oxidation rate below the graphite surface. After evaluation of the effect of this phenomenon on the future program, work was initiated to determine the feasibility of using ultrasonic backscattering to profile the oxidation at greater penetration depths.

INTRODUCTION

This is a continuation of previous work 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-1980 scope of this project is to:

- investigate and develop eddy current techniques for providing interpretable indications of graphite strength
- continue investigation and application of sonic velocity techniques to PGX and ATJ graphites
- recommend an in-service monitoring technique by the end of FY-1980.

The objective of this investigation is to demonstrate the feasibility of non-destructive testing (NDT) techniques for determination of structural graphite strength.

(a) RSR Fin. Budget No.: 82101-0; RSR Contact: R. Foulds.

TECHNICAL PROGRESS

Research is being conducted to determine the feasibility of both eddy current and ultrasonic methods for nondestructive measurement of graphite oxidation. The work has encompassed eddy current testing and three ultrasonic methods.

EDDY CURRENT TESTING

Work continued on the preliminary development of an algorithm for estimating near-surface oxidation profiles using multifrequency eddy current data.

ULTRASONIC TESTING

The ultrasonic techniques that have been or are being evaluated under the program are: through-transmission ultrasonic velocity, surface-wave velocity, and ultrasonic backscattering. Results of the through-transmission tests, which were reported in the previous quarterly, confirmed the correlation between ultrasonic velocity and graphite strength. The test is particularly useful where two sensors can be directly aligned through a maximum of 6 to 10 inches of graphite; greater separations may be possible using elastomer or silicon rubber materials as coupling members.

Evaluation of the surface-wave test apparatus, which was developed under subcontract, has shown a dramatic difference in surface-wave velocity between oxidized and unoxidized PGX. However, for medium to large degrees of oxidation, a surface wave cannot be sustained in PGX. Further evaluation is in progress to determine the limit of oxidation below which satisfactory energy propagation can be sustained.

New data processing techniques have pushed ultrasonic backscatter testing to the forefront as a materials characterization method. We are currently evaluating this method as a means of assessing oxidation in graphite. Backscatter testing uses spatial waveform averaging to distinguish between background noise and the scattered signal from identifiable ultrasonic reflectors, such as grain boundaries, pores, inclusions, and microcracks. Our work has

been directed at characterizing the porosity due to oxidation in graphite using this technique. Currently, dry coupling methods are being developed that will facilitate application of the technique for in-service measurements on core-support graphite components.

OXIDATION PROFILES

It was previously reported⁽¹⁾ that visual observations of sectioned samples indicated that the oxidation rate increased with penetration depth. Figures 1 and 2 show the density profiles, obtained by physical measurement of

FIGURE 1. Density Versus Oxidation Depth for 3.0-in. Diameter Sample No. VII-B-A-9

FIGURE 2. Density Versus Oxidation Depth for 3.0-in. Diameter Sample No. VI-C-4

the volume and weight, of one sample each of P3W and PGX; density could not be obtained for the inner 0.25 in. of the P3W samples because the material was too friable to retain a measurable geometry.

A review of the literature indicates that this "inside-to-outside" oxidation has been observed at Pennsylvania State University⁽²⁾ as well as at Oak Ridge National Laboratory (ORNL).^(a) A report on our observations and interpretation of this phenomenon is being prepared for publication in Carbon.

FUTURE PLANS

EDDY CURRENT TESTING

- complete theoretical analysis for profiling oxidation with multi-frequency eddy current data
- implement and test profiling algorithm.

ULTRASONIC TESTING

- evaluate low-force dry coupling methods for implementing both ultrasonic velocity measurements and ultrasonic backscattering
- evaluate ultrasonic backscattering as a means of quantifying oxidation in graphite
- determine useful limits of ultrasonic surface-wave tests in graphite.

OXIDATION PROFILES

- issue report on observations and cause of inside-to-outside oxidation phenomenon.

(a) Conversations with W. P. Eatherly (ORNL) as reported in our last quarterly (and subsequent conversations).

REFERENCES

1. Hooper, J. L., ed. 1979. Reactor Safety Research Programs Quarterly Report, January 1-March 31, 1979. NUREG/CR-0855, PNL-3040-1, prepared for the Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.*
2. Walker, P. L., Jr., M. Shelef, and R. A. Anderson. 1968. Chem. Phys. Carbon, Vol. 4, Marcel Dekker, New York, (see especially, p. 366).

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE
MONITORING OF NUCLEAR PRESSURE VESSELS^(a)

P. H. Hutton, Project Manager

R. J. Kurtz
J. R. Skorpik
T. P. Harrington
P. G. Doctor

SUMMARY

Plans for participating in a vessel test at Materialprüfungsanstalt (MPA), Stuttgart, West Germany, to evaluate acoustic emission (AE) monitoring concepts were reviewed with appropriate German technical staff. The test matrix and other test details were defined. Preparation of the A533B steel insert for the test vessel is in progress. All components for an AE monitor/analysis system to be used on the German vessel test have been ordered, and fabrication is in progress.

Pattern recognition techniques to separate crack growth AE from noise have been successfully applied to data obtained with metal wave-guides. This is an alternate to expensive high-temperature surface-mounted sensors for monitoring a reactor structure during operation.

Arranging for installation of an AE system in an operating reactor is proving to be a significant problem due to required reactor downtime.

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. Program objectives are to:

- develop a method to identify crack growth AE signals in the presence of other acoustic signals

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

- develop a relationship to estimate flaw significance from AE data
- develop an instrument system to implement these techniques
- demonstrate the total concept both off-reactor and on-reactor.

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

TECHNICAL PROGRESS

OFF-REACTOR VESSEL TEST

This phase of the program is directed toward off-reactor evaluation and demonstration of the concept developed for reactor monitoring and flaw evaluation by AE. The decision was made to monitor vessel testing at MPA, Stuttgart, West Germany,⁽¹⁾ with a target startup date for the test of January 1981.

Work continued this past quarter on developing a suitable test plan for AE monitoring the first MPA vessel test.⁽¹⁾ An initial plan was developed (see Appendix A) and was used in discussions with IzfP^(a) and MPA staff to establish technical details of the test and associated preparation activities. Meetings conducted on June 3 and 4, 1980, in Stuttgart, West Germany, indicated that some revisions to the proposed matrix were necessary to complete the test within the desired 3- to 4-month time period. The major revisions consisted of.

- reducing the total number of pressure cycles by about 33%
- decreasing the cyclic frequency to 1.0 cycle/min
- increasing the test pressure by about 27%

(a) IzfP: Institut für Zerstörungsfreie Prüfverfahren, Saarbrücken, West Germany

- adjusting notch geometries to accommodate the reduced cycles and increased stress.

Tables 1 and 2 summarize the revised matrix.

Since the German vessel is fabricated of A50B steel, a section of A533B steel will be inserted in the 120-mm vessel wall to provide data that are representative of U.S. vessels. The overall dimensions of the A533B insert were changed based upon input from MPA and Oak Ridge National Laboratory

TABLE 1. Revised Test Sequence for Vessel ZB1

<u>Pressure, bar</u>		<u>Number of Cycles</u>	<u>Comments</u>
<u>Maximum</u>	<u>Minimum</u>		
320	32	2,000	--
320	192	20,000	Crack Mark
320	32	2,000	--
320	192	20,000	Crack Mark
320	32	1,500	--
320	192	17,000	Crack Mark
320	32	1,500	--
320	192	13,000	Crack Mark
320	32	1,500	--

NOTES

1. Cyclic frequency shall be at least 1.0 cycle/min and possibly greater during crack marking.
2. Hydrotests shall be performed periodically during the test sequence. The first vessel loading shall be a simulated hydrotest to a pressure of 280 bar at a rate of 1 bar/min. Additional hydrotests, identical to the first, shall be performed following each crack marking. The final hydrotest shall be conducted after all fatigue cycling is completed and pressurization shall be continued to vessel failure.

TABLE 2. Revised Flaw Geometries - ZB1 A533B Insert

Flaw Location	Flaw Depth, (a)	Flaw Length, (a)
	mm	mm
ID ^(b)	21	53
ID	33	85
OD ^(c)	50	200

(a) Dimensions represent the machined notch dimensions

(b) Inside diameter

(c) Outside diameter

(ORNL) staff, and ORNL staff provided the necessary design requirements for precracking the machined notches. The dimensions of the patch were established at 1500x770x120 mm.

Within practical limits, the vessel test conditions will attempt to simulate operating reactor conditions. From the standpoint of AE detection, coolant flow noise is the primary consideration. Simulation of measured reactor background noise by both hydraulic and electronic methods is being developed for application during the vessel test.

Since the ZB1 vessel test at MPA will be run at ~93°C, there is a problem in devising a method for simulating thermal stressing of the stainless steel (SS) cladding-to-vessel interface in an operating reactor vessel. One method suggested is by external local heating of the section of the vessel having inside cladding. The feasibility of this suggestion is being evaluated by Pacific Northwest Laboratory (PNL) heat transfer specialists.

AE MONITOR SYSTEM DEVELOPMENT

The AE monitor system to be used in the German ZB1 vessel test will consist of three subsystems:

- Dunegan/Endevco (D/E) AE detection and source location system
- digitized waveform recorder
- pattern recognition analyzer.

All major equipment items for the waveform recorder and pattern recognition analyzer have been ordered. The 9-track tape recorder, which is common to both systems, has the longest lead time (4 months). The recorders, however, are not critical path items and will not prevent system fabrication progress. The existing Nuclear Regulatory Commission (NRC)-owned D/E 1032 system is presently at Dunegan/Endevco for upgrading and will include a newly designed front end and data acquisition system (DART).

The waveform recorder system will utilize an existing NRC-owned 8-bit Biomation 8100 transient recorder for signal digitization. Digital data will be transferred to 9-track digital tape via a microprocessor controller. Each waveform will have associated with it an identifier code extracted from the D/E system, which is necessary for matching waveforms with signal source location in the pattern recognition analyzer.

The pattern recognition system is centered around a Digital Equipment Corporation (DEC) PDP 11/03 minicomputer. Data tapes from the waveform recorder will be played back and matched with data input from the D/E system. Specific analysis features will act upon the two data sets for noise versus AE discrimination. The net output expected is a delineation of AE signals verified by pattern recognition and identified by source location. Data tapes will also be shipped to PNL for detailed pattern recognition analysis. If the presently selected features need to be changed, revisions can be sent to Germany and loaded on to the PDP 11/03 system via floppy disk. The possibility of using a large Battelle-Frankfurt minicomputer for the detailed pattern recognition analysis is being considered. A flaw significance evaluation using AE data will also be performed on the PDP 11/03 system. The PDP 11/03 system is a component system that requires minimal assembly time once all component items are received; all items have been ordered.

PATTERN RECOGNITION

Earlier analysis of AE waveforms recorded by wave-guide sensors indicated that insufficient information was available for pattern recognition analysis because the preamps associated with the wave-guides were tuned to a very narrow frequency range. This tuning effectively filtered out much of the discrimination information needed for pattern recognition analysis. Figure 1 compares the spectrum of surface-mounted sensors used in earlier experiments⁽²⁾ with initial data from a wave-guide sensor. The effect of preamp tuning on the shape of the wave-guide spectrum is very apparent.

Since it would be valuable to use wave-guides in the hostile environment of a reactor vessel, data were collected during the cylindrical bend specimen test using a preamp that had been modified to pass a broader band of frequencies. Figure 2 shows a typical spectrum. The noise source in this case is background noise. An important feature to note is that the response still drops sharply below 250 kHz, which would discriminate against reactor background noise.

FIGURE 1. Comparison of Two Acoustic Emission Sensors

FIGURE 2. Typical Response of Broad Band Wave-Guide to Background Noise

The broad band wave-guide sensor data set consisted of 83 waveforms that were divided as follows:

Valid AE	31
C-Clamp	16
Background	20
Rubbing	<u>16</u>
	83

Feature generation and reduction techniques documented previously⁽²⁾ were used to analyze these data, and 10 features were generated for the analysis:

- standard deviation
- skewness
- kurtosis
- autocorrelation at Lag 13

- autocorrelation at Lag 37
- maximum frequency response
- frequency of maximum response
- total power (0 Hz-1 MHz)
- first moment of the power spectrum
- second moment of the power spectrum.

These 10 features were autoscaled, and a correlation-to-property feature reduction algorithm was used. Six features were selected as having the best discrimination capability; they are listed in order of importance below:

- kurtosis
- autocorrelation at Lag 37
- standard deviation
- second moment of the power spectrum
- first moment of the power spectrum
- total power.

This list can be compared to the following set of features that were found to contain the best separation information for AE data gathered with a high-temperature surface-mounted sensor during the same experiment.⁽¹⁾

- kurtosis
- autocorrelation at Lag 37
- mean
- total power
- first moment of the power spectrum.

Four of the five features are common to both sets of data. In addition, two features--kurtosis and autocorrelation at Lag 37--were selected as the most valuable in both cases.

The first five features selected by the feature reduction algorithm for the wave-guide data were input to a least squares fitting algorithm to produce a decision function. The output of this decision function is plotted in Figure 3. The waveform number, which is listed on the abscissa, is arbitrary. The horizontal line represents the value of the decision rule; all points above this line are classified as noise and all those below the line as valid AE. Seven of the 83 waveforms were misclassified giving a 92% correct

FIGURE 3. Least Squares Decision Rule with Five Features

classification. If all six selected features are input to the least squares fitting algorithm, the correct classification rate increases to 99% as shown in Figure 4. These results show that wave-guide sensors may be a viable alternative to the much more expensive high-temperature surface-mounted sensors even when the data are to be used for pattern recognition analysis.

REACTOR INSTALLATION

Installation of an AE system in an operating reactor to test continuous monitoring has been discussed with two utilities. There are three desired monitoring locations: a vessel nozzle, a section of the vessel weld, and a section of primary piping. Obtaining permission to monitor primary piping

FIGURE 4. Least Squares Decision Rule with Six Features

does not appear to be a problem since investigation of any significant AE indications would require only about 2-3 days of reactor downtime. However, obtaining permission to monitor a nozzle and/or a section of vessel weld is definitely a problem (investigation of any AE indications in these locations would require about 35 days of reactor downtime). An NRC waiver of responsibility to investigate AE indications would do little to alleviate the problem because utility policy requires that any indications of a flaw or fault be investigated. This appears to be a common circumstance among the utilities, and it places even greater importance on the off-reactor vessel test to demonstrate the viability of continuous AE monitoring.

REPORTS

A quarterly progress report for the period January 1 through March 31, 1980, was prepared during this quarter as was a program summary for the Advisory Committee on Reactor Safeguards (ACRS).

FUTURE PLANS

Work plans for next quarter (July 1 to September 31, 1980) include:

- fabrication of the A533B steel insert to be welded into the MPA test vessel
- evaluation of hydraulic and electronic methods for simulating reactor coolant flow noise on the MPA test vessel
- preparation of an analysis-before-test document
- testing for AE characterization of degraded A508 steel supplied by MPA
- continued fabrication of the AE monitor/analysis system for vessel test monitoring
- completion of cylindrical bend specimen test
- complete pattern recognition analysis of a composite of AE and noise data from several tests.

REFERENCES

1. Hutton, P. H., and R. J. Kurtz. 1980. "Acoustic Emission-Flaw Relationships for In-Service Monitoring of Nuclear Pressure Vessels." In Reactor Safety Research Programs Quarterly Report, January 1-March 31, 1980, NUREG/CR-1454, Vol. 1, PNL-3380-1, Pacific Northwest Laboratory, Richland, Washington.*
2. Doctor, P. G., T. P. Harrington, and P. H. Hutton. July 1979. Pattern Recognition Methods for Acoustic Emission Analysis. NUREG/CR-0910, PNL-3052, Pacific Northwest Laboratory, Richland, Washington.*

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

APPENDIX A

TEST PLAN FOR U.S.-GERMAN COOPERATIVE RESEARCH TO ACOUSTIC EMISSION MONITOR PRESSURE VESSEL TESTS ZB1 AND ZB2 AT MPA STUTTGART LABORATORY

INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC), Reactor Safety Research Division, has been conducting an extensive research program through one of their contractors--Pacific Northwest Laboratory (PNL)--to develop the application of acoustic emission (AE) to monitor reactor pressure vessels continuously for detection and evaluation of growing flaws. The initial work has been done primarily on laboratory specimens of Type A533-B, Class 1 pressure vessel steel. The program has reached a point where it is recognized that further work must be conducted on heavy section vessels under simulated reactor conditions to evaluate AE signal identification and flaw interpretation techniques adequately. The forthcoming vessel tests, ZB1 and ZB2, to be performed at MPA Stuttgart can meet the technical needs of the NRC program. Cooperative participation by the United States in these tests also appears to offer mutually beneficial technical information exchange.

This test plan is intended to establish a technical and administrative understanding of the bases for NRC participation in the ZB1 and ZB2 vessel tests. Work in behalf of the NRC will be performed by its contractor, PNL.

TEST PLAN

PURPOSE

The purpose of this test is to measure and analyze AE data from fatigue crack growth in a heavy section (100-150-mm thick) steel pressure vessel. The conditions of the test will attempt to simulate reactor pressure vessel (RPV) operation within practical limits. In the ZB1 test, the primary AE data will

be derived from fatigue crack growth in an insert of ASTM A533, Grade B, Class 1 steel in the vessel wall. In the ZB2 test, attention will focus on fatigue crack growth in a nozzle fabricated from degraded A508 material as planned by MPA.

The following test requirements relate to test ZB1 only. A test plan for ZB2 will be developed separately.

TEST VESSEL DESCRIPTION - ZB1

Intermediate pressure vessel ZB1 is a 120-mm thick, 1715-mm outside diameter (OD) cylindrical vessel with hemispherical closure heads. The cylindrical section of the vessel is 2700 mm long; the vessel material is A508 steel. An insert of A533B, Class 1 steel to be welded into the vessel shell will be provided by PNL. Figure A.1 shows the tentative configuration of the insert. The largest insert that can be provided from A533B material on hand is 1550 mm long by 800 mm in the circumferential direction. The insert will contain three part-circular surface flaws that have been fatigue presharpenered by internal notch pressurization. Two of the flaws will be machined on the inside surface of the insert and one flaw will be on the outside surface. The longitudinal axis of the flaws shall be oriented perpendicular to the circumferential stress in the vessel. Final dimensions of the insert will be in accordance with drawings to be provided by MPA.

VESSEL LOADING CONDITIONS - ZB1

In order to simulate RPV loading conditions, the test vessel should be internally pressurized with water at a temperature of about 90°C. The water pressurization system should be capable of providing sinusoidal cyclic mechanical stressing of the vessel at frequencies ranging between 0.5 and 5 cpm. In addition, pressure cycling at R-ratios ($R = \text{minimum pressure}/\text{maximum pressure}$) of between 0.1 and 0.6 should be possible. It is desired to cycle at rates up to 10 cpm during much of the 0.6 R-ratio testing. The proposed test sequence (see Table A.1) is designed to produce measurable crack growth from each of

FIGURE A.1. Flaw Locations in A533B Insert (all dimensions in mm)

TABLE A.1. Proposed Test Sequence^(a) for Vessel ZB1

Pressure, bar		Cyclic Frequency, cpm	Number of Cycles	Comments
Maximum	Minimum			
252	25	5.0	3,000	
252	150	10.0	30,000	Crack mark
252	25	5.0	3,000	
252	150	10.0	30,000	Crack mark
252	25	5.0	2,500	
252	150	0.5/5.0	25,000	Mark, 2K cycles at 0.5
252	25	5.0	2,500	
252	150	0.5/5.0	20,000	Mark, 2K cycles at 0.5
252	25	5.0	2,000	

(a) This test matrix will require approximately 400 hours of cycling time as well as at least 200 hours of preparation, repair, and analysis. Operating on the basis of a 40-hour week, this represents 15 weeks of work. If two shift operations were used, the overall test time could be reduced.

the three flaws, but none of the flaws are expected to propagate through the wall during the loading sequence (see Table A.2).

INNOCUOUS NOISE SOURCE SIMULATIONS - ZB1

One of the important objectives of AE monitoring the vessel test is to evaluate the effect of various reactor background noise elements on AE monitoring. This relates both to basic detection of the AE signal and to identification of crack growth AE signals in the presence of similar noise signals.

One major source of noise is the reactor coolant circulation system. Measurements of this noise taken on operating reactors show it to be very high relative to AE signal levels at low frequencies (<100 kHz). However, with increasing frequency, it diminishes until at about 400-500 kHz it is down near the electronic noise level of the measurement system. To evaluate flow noise

TABLE A.2. Proposed Flaw Geometries - ZB1 (A533B Material Insert)

Flaw Location	Initial Flaw Depth, mm	Initial Flaw Length, mm	Maximum Final Flaw Depth, mm ^(a)
ID ^(b)	24.0	58.9	34.0
ID	47.8	127	71.0
OD ^(c)	59.7	221	89.0

(a) Represents a conservative estimate of the potential crack growth.

(b) Inside diameter

(c) Outside diameter

effects on AE monitoring requires a noise field similar to reactor noise and a growing crack producing AE in that noise field. An approach to simulating reactor flow noise at a localized area requires that the water system for pressurizing the vessel be capable of about 6 m/sec flow rate in the vessel feed water piping. A throttling valve is necessary in the feed water line to increase flow noise to a level comparable with reactor conditions by restricting flow at randomly selected intervals. In conjunction with flow noise generation, a growing crack in a pipe wall near the noise source is needed to produce AE. This could consist of a spool piece inserted adjacent to the valve with a lateral leg containing a crack that can be grown by cyclic bending of the lateral leg. PNL would provide the test spool piece, plus a straight pipe replacement for long-term operation. The approach to be used and the final details are to be jointly developed by PNL and MPA.

Mechanical rubbing of thermal insulation on the exterior of the vessel is a potential noise source to be simulated. To produce this noise, the vessel should be covered with blanket-type glass fiber insulation, which will either naturally or by manipulation have some movement relative to the vessel surface.

Mechanical stressing of a weld slag inclusion may produce noise similar to AE. To investigate this, a slag inclusion is to be incorporated into one of the A533B insert-to-vessel welds. The slag should be about 10 to 20 mm in diameter and near mid-thickness of the vessel wall. The tentative location

and size of the slag inclusion is to be determined prior to pressure cycling of the vessel by nondestructive means (e.g., x-ray and ultrasonic testing) for verification. Following the test, the slag inclusion and any associated crack growth will be characterized by destructive sectioning for correlation with AE results.

Potential base metal-to-weld cladding interface noise must also be investigated. It is proposed to simulate this by depositing a patch of stainless steel (SS) weld cladding approximately 5 mm thick over an area on the inside surface of the vessel of about 0.3 m^2 near the A533B insert. Thermal stressing of the cladding-to-vessel interface is needed to investigate potential resulting AE sources. The method of producing the desired thermal stress will be jointly determined by MPA and PNL. Thermocouples will be installed on each side of the interface to measure actual thermal gradient.

Electrical transients represent the last noise source to be introduced. These transients may be produced by methods such as spark discharge or turning electric motors on and off. These noises will be inserted at random locations on the test sequence.

AE MONITORING - ZB1

A fundamental requirement in AE monitoring is that it be performed by methods that can be directly applied on an operating reactor. This requires the use of sensors capable of monitoring a 290°C surface even though the test temperature will be only about 90°C . Other requirements are to obtain good quality data to be processed for source location, pattern recognition characteristics, and crack growth estimation. In addition, digitized signal waveforms plus conventional AE signal parameters such as signal peak time, amplitude, energy, and duration must be recorded in a fashion that allows identification with test conditions and sequence.

To achieve the above requirements, the planned monitoring arrays include:

- A sensor array located entirely on the A533B insert and encompassing the flaws. This will provide optimized AE data from the growing flaws.

- One or two sensor arrays located to monitor the cylindrical portion of the vessel. These will provide a circumstance more directly related to monitoring a reactor pressure vessel wherein attenuation and geometry effects can be evaluated and related to information from the optimized array.

Each array will involve five high-temperature sensors and each sensor will require about 13 cm^2 for a mounting fixture.

CRACK GROWTH MONITORING REQUIREMENTS - ZB1

Methods other than AE for monitoring the crack propagation are required during the test. Crack opening displacement (COD) and ultrasonic inspection measurements should be used for this purpose. In order to perform COD measurements for ID flaws, COD gauges will be required on the inside surface of the vessel. Thus, access to the interior of the vessel is required, as well as feedthroughs for lead wires. Technique and installation details will be jointly established by MPA and PNL.

After the test is completed, the COD and ultrasonic inspection results will be confirmed by destructive examination of each flaw. It will, therefore, be necessary to remove the A533B insert after the test is completed. Subsequent detailed sectioning of the flaws shall be performed at PNL.

BLIND TESTS - ZB1

Blind tests of the AE monitoring system shall be performed at randomly selected times during the test sequence. In addition to monitoring crack growth in the presence and absence of various innocuous noise sources, it is also necessary to AE monitor the vessel in the presence of background noise when no crack growth is occurring. To produce this situation requires activating the various noise source mechanisms while cycling the vessel pressure between 25 and 126 bar. This pressure cycle should produce little, if any, crack growth and should be applied for only a limited number of cycles (a few hundred) at randomly selected points in the test sequence.

RESPONSIBILITIES

U.S. and German responsibilities in this cooperative test are listed below.

UNITED STATES (PNL)

1. Provide a finished insert of A533B material in accordance with dimensional specification and schedule requirements set forth by MPA. Flaws will be machined into the insert and precracked ready for testing. Insert is to be delivered to the vessel fabrication site.
2. Perform laboratory tests of samples of the degraded A508 material with microcracks prior to vessel testing to evaluate AE characteristics.
3. Provide complete AE monitor/analysis and physical measurement systems at the test site (MPA Stuttgart), install AE sensors, and prepare the system for testing in accordance with the schedule set forth by MPA.
4. Provide physical measurement devices (strain gauges, COD gauges) and signal recording instruments required in support of the AE monitoring as agreed by MPA and PNL.
5. Provide the necessary qualified personnel to conduct AE monitoring and analysis during the course of the test.
6. Record and provide to MPA any AE data from the A508 insert to the extent practical without compromising AE monitoring of the A533 insert.
7. After removal of the A533 insert following the test, transport the insert to PNL and perform destructive examination of the flaws to confirm crack growth characteristics.
8. Prepare a report describing AE monitoring results, physical measurement data, and destructive examination results relevant to the AE

monitoring and the correlations among these. This complete report, plus any supporting data, is to be available to MPA.

GERMANY (MPA STUTTGART)

1. Provide samples of degraded A508 material for laboratory testing by PNL to evaluate AE characteristics prior to the vessel test. The material is to be representative of that being inserted into the ZB1 vessel.
2. Install the A533B insert provided by PNL in the ZB1 vessel including necessary postweld stress relief and slag inclusion.
3. Install SS cladding on section of vessel inner surface.
4. Keep PNL informed on test preparation and testing schedule.
5. Install physical measurement devices (strain gauges, COD gauges, thermocouples) as agreed by MPA and PNL.
6. Provide fixtures and loading apparatus as required by plan jointly agreed on for growing a crack in vessel feed water pipe.
7. Provide vessel test facilities and perform a test sequence in accordance with the test plan described herein. Note: In the event of problems with excessive crack growth rate or lack of growth in either insert, this test plan may be modified by mutual consent of PNL and MPA.
8. Remove the A533B insert from the ZB1 vessel following test completion.
9. Document physical test results and AE results by others in a report to be made available to PNL.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION
RELIABILITY AND FRACTURE MECHANICS^(a)

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

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

SUMMARY

Primary emphasis during the past quarter was directed toward sample preparation for the round robin and measurement of the impact of inspection sensitivity changes resulting from the 1977 Code revision. It is presently expected that the round robin will be delayed from August 1980 to December 1980 due to delays in sample preparation. The reduction in inspection sensitivity resulting from the 1977 Code revision was measured to be from 6- to 16-dB in wall thicknesses from 10 to 61 mm (0.4 to 2.4 in.). Recommendations for corrective measures are in preparation.

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). These inspections are periodically performed on a sampling of pipe joints using ultrasonic techniques.

The Integration of Nondestructive Examination (NDE) Reliability and Fracture Mechanics Program has been established to determine the reliability of current in-service inspection (ISI) techniques and to develop recommendations that will assure a suitably high inspection reliability. The objectives of the program include:

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

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

TECHNICAL PROGRESS

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

TASK 6: SAMPLE FABRICATION

Two major activities are in progress as a part of this task: sample fabrication and sample cracking. The 10-in. Schedule 80S 304 stainless steel (SS) and 27.5-in. inside diameter (ID) cast SS samples have been welded and are currently being cracked by the thermal fatigue process. The A106 carbon steel pipes were delivered in late April 1980, and welding was initiated in early May. After completion of one weld and partial completion of a second weld, numerous cracks were detected near the weld fusion line. Chemical analysis indicated a high carbon content of 0.4% for the material (0.35% is the allowed maximum). The cracking was attributed to lack of sufficient pre-heat and low interpass temperatures that would be unsuitable for material with this high carbon content. The welding procedure has been revised to provide

the necessary preheat and postweld heat treatment. The samples have been remachined; and welding, using the improved procedure, is in progress. It is expected that the welded samples will be delivered by September 1, 1980.

Thermal fatigue cracking of the 10-in. diameter Schedule 80S SS and 27.5-in. ID cast SS pipes is in progress. Two of the six required 10-in. pipes have been completed (five cracks), and two of the 16 required cracks in the 27.5-in. cast pipes have been completed.

Parameter development testing on the 27.5-in. cast material has shown that cracks as deep as 2.54 cm (1.0 inch) can be reliably grown. The crack shown in Figure 1 was grown in a welded 27.5-in. cast SS sample in 21 days. With the currently available fatigue stations (four), it is expected that the cast SS specimens will be completed by October 1, 1980, and the 33-in. diameter ferritic samples will be completed by mid-November 1980. This schedule would allow initiation of the round robin tests by December 1, 1980. Alternatives to shorten this schedule are being investigated.

FIGURE 1. Thermal Fatigue Crack--19 mm deep, 64 mm long (0.75 in. deep, 2.5 in. long)--in a Welded Cast Stainless Steel Pipe Sample

TASK 7: MEASUREMENT AND EVALUATION

Two measurement programs are currently in progress: measurement of the reduction in inspection sensitivity resulting from the use of notch calibration reflectors as opposed to side-drilled holes and the influence of flaw aspect ratio on signal response.

Calibration Sensitivity

The 1977 Revision, Summer 1978 Addenda of the ASME Code Section XI, Appendix III, revised the calibration reflector from a side-drilled hole (1974 Revision) to a notch reflector of depth $d = 0.104 t - 0.009 t^2$ for ferritic steel piping and 10% of wall thickness (t) for austenitic piping. A measurement program was developed to determine the impact on test sensitivity of this Code revision.

The measurement program to determine the impact of the change in calibration reflectors consisted of approximately 540 measurements on 34 piping calibration standards. The standards were from a boiling water reactor (BWR) presently under construction and ranged in wall thickness from 0.237 to 2.343 inches with diameters from 4 to 30 inches. Twenty-six of the samples were ferritic and eight were stainless steel. The samples contained both side-drilled holes and notch reflectors according to the applicable codes. Measurements were performed using 0.25-, 0.5- and 1.0-in. diameter, 2.25-MHz search units with 45° and 60° shear wave contact shoes. The 0.25-in. search unit was used for thicknesses up to 0.75 inch; the 1.0-in. unit, for thicknesses greater than 0.75 inch; and the 0.5-in. unit, for the total range. A Sonic Mark I flaw detector was used for the measurements, and a Nortec 131D flaw detector was used for verification on four of the samples.

The measurements were performed by establishing a distance amplitude correction (DAC) curve for the side-drilled holes and measuring the notch response relative to the DAC curve. Measured results for 45° shear waves--the principal angle used for piping inspection--are shown in Figure 2. The notches produce higher reflected amplitudes than the side-drilled holes. A calibration performed using the notches (1977 Code) will, therefore, provide a less sensitive inspection than the side-drilled hole calibration (1974 Code). The

FIGURE 2. Measured and Theoretical Reduction in Ultrasonic Inspection Sensitivity Resulting from Calibrations According to 1977 Code (notch) Relative to 1974 Code (side-drilled holes-SDH). Error bars (+4 dB) are the $\pm 2\sigma$ measurement error bounds.

reduction in inspection sensitivity is dependent on pipe wall thickness ranging from -6 dB (a factor of 2) at 0.4 inch to -16 dB (a factor of 6.3) at a wall thickness of 2.4 inches. The error bars on the measured curve of Figure 2 are the $\pm 2\sigma$ error bars for the measurements ($\sigma = 2$ dB).

Theoretical calculations were performed according to Ermolov⁽¹⁾ and Werneyer⁽²⁾ to assure that the measured results in Figure 2 were reasonable. These calculations, shown in Figure 2 as Theory, are in close agreement with the measured results. The calculations were performed for the far-field conditions that are satisfied for thicknesses greater than 1.0 inch; however, the differences below 1 inch are not great. It should be noted that the theoretical curve has not been adjusted to fit the data; it is the actual dB difference predicted by the theory--the agreement is remarkable.

For 60° shear wave inspection, there was not a statistically significant difference between the side-drilled hole and the notch calibration sensitivities although the side-drilled hole calibration was qualitatively slightly

less sensitive. This difference between the 45° and 60° shear wave inspection results from mode conversion at the notch reflector. Reflection from the side-drilled hole is always at normal incidence and no mode conversion takes place. For a notch or right angle reflector, two reflections occur: one at the incident and one at the complementary angle since sound is reflected from the back surface and the notch surface. At 45° , both reflections are above the critical angle for longitudinal mode conversion and the sound is totally reflected. However, at 60° incidence, the complementary angle is 30° , which is below the angle of total reflection. At 30° only approximately 15% of the energy is reflected as a shear wave with the remainder being mode converted to a longitudinal wave, which does not return to the search unit.

Three variables in the experiment have not yet been discussed: material, search unit diameter, and instrument. Eight of the samples were stainless steel with a maximum size of 24-in. diameter and 1.140-in. wall. Since this experiment measures the relative amplitudes of reflectors in the same sample, major differences due to material type would not be expected; and this was indeed the case. However, the SS samples did show slightly more scatter due to point-to-point differences in material attenuation. The overall trend, however, was consistent with the ferritic samples. No statistically significant trend difference was noted for changing transducer diameter as long as the correlations were applied within their applicable ranges (up to 0.75 in. thickness for the 0.25-in. search unit and greater than 0.75 in. thickness for the 1.0-in. search unit). Spot checks with the Nortec 131D showed no appreciable differences from results obtained with the Sonic Mark I instrument.

In summary, it can be said that calibration for 45° shear wave inspection will be from 6 to 16 dB less sensitive using the notch reflectors (1977 Code) as opposed to side-drilled holes (1974 Code). Theoretical calculations agree with measured results and can be used in the future as a basis for comparison.

Impact on Inspection Reliability of Inspection Sensitivity

The depth of calibration notches specified in Appendix III - 3430 (Section XI, 1977 Revision) is on the same order as allowable planar indications specified in Tables IWB-3514-2 and -3 for in-service examination (Section XI, 1977 Revision). The calibration sensitivity established by Appendix III is not sufficiently sensitive to detect unacceptably large flaws because:

- The reporting level is set at 100% DAC (primary reference level).
- The "ideal" calibration reflector (notch) is longer than the flaws required to be detected.
- The maximum allowable flaw sizes for preservice examination are smaller than the calibration reflector.
- Cracks occurring in-service are not ideal reflectors and will--due to their roughness, tightness, and orientation--yield lower reflected amplitudes than ideal reflectors (notches).

Measurements and evaluations previously reported^(3,4) indicate that the influence of the factors listed above could be as large as 12 dB. It is, therefore, unrealistic to expect in-service cracks to yield reflected amplitudes exceeding 100% DAC (1977 Code). Based on the above, it is our contention that the reduction in test sensitivity shown in Figure 2, which results from acceptance of the 1977 Code through summer 1978, is not justifiable.

Inspection of Machined Flaws of Varying Aspect Ratios

Under another Nuclear Regulatory Commission (NRC)-sponsored ultrasonic ISI research program,^(a) Pacific Northwest Laboratory (PNL) has acquired specimens of four types of reactor pipe, each containing several machined defects. Inspection of these samples permits observation of the combined influence of several parameters influencing flaw detectability. The four types of pipe are:

(a) RSR Fin. Budget No.: B2157; RSR Contact: George Johnson/Martin Hum

- 4-in. diameter, 0.237-in. wall thickness carbon steel
- 12-in. diameter, 0.688-in. wall thickness 304 SS
- 20-in. diameter, 1.031-in. wall thickness carbon steel
- 20-in. diameter, 1.031-in. wall thickness 304 SS.

The machined reflectors include ID and outside diameter (OD) 1977 ASME, Section XI, calibration end-mill notches and a 1974 ASME side-drilled hole. The other reflectors are electric discharge machined (EDM) notches with three aspect ratios and three angles of orientation (from vertical) for a total of nine EDM defects per pipe sample. The three flaw angles-- 0° (vertical), 15° , and 25° --were selected for their poor reflection characteristics based on earlier results from inspection of angled flaws in flat plates. The three aspect ratios are 0.5 (semicircular), 0.25, and 0.05. The notch depths are those defined as maximum allowable as a function of aspect ratio in Tables IWB-3514-2 and -3 of the Code.

These machined defects were inspected using two ultrasonic instruments that are frequently employed in reactor ISI: a Sonic Mark I and a Nortec 131D. Three transducers were used: 1/4-, 1/2-, and 1-in., 2.25-MHz search units. Inspections were performed at $1/2 V 45^{\circ}$, $3/2 V 45^{\circ}$, and $1/2 V 60^{\circ}$ beam paths. Reflection amplitudes were recorded as decibels relative to the end-mill notch DAC required by the 1977 version of the ASME Code, Section XI.

To date, complete data have been taken for all samples except the 20-in. SS pipe section, which is not included in the following observations and conclusions.

The primary observation is that beam spread seems to be the key to angled-flaw detection, as observed in earlier studies of flat plate samples. Data taken using the Nortec instrument are usually higher in amplitude and fluctuate less with flaw angle changes than data taken using the Sonic unit. The operating frequency of the Nortec unit is some 200 kHz lower than that of the Sonic, leading to more beam spread. Less angle sensitivity and generally higher amplitudes are produced by the smaller of a pair of transducers; that is, the transducer with more beam spread. These observations agree with earlier studies⁽⁴⁾ of angled flaws in flat plates in which it was shown that the

best combination of high amplitude and low fluctuation with angle was achieved using low-frequency (1.5 MHz) dual search units with 3/8-in. square sending and receiving transducers.

A second observation in support of earlier flat plate studies is that the 60° inspection performs better than the 45° inspection for nonvertical flaws because the mode conversion from the calibration notch is at its maximum for a 60° beam, causing a high calibration gain setting. When a nonvertical flaw is inspected, the mode conversion is reduced or absent; therefore, at a 60° beam angle there is more opportunity for the flaw to exceed the notch amplitude and thus become reportable. The greatest advantage of this effect is gained using a large transducer. The reflected beam from a nonvertical flaw may not exit the metal at its entry point (transducer location), but in some cases (most notably for a 15° flaw) the beam exit is close enough to the beam entry point that a large transducer may receive it while a small one might not. Detection of vertical flaws was best achieved by a 45° inspection. Inspection using both beam angles is indicated for higher detection probability.

Increasing pipe wall thickness seems to have a negative effect on reflection amplitudes. The 4-in. pipe data show very little amplitude change as a function of flaw angle, while the 20-in. pipe data fluctuate considerably. This is attributed to two effects. First, the flaws in the 4-in. pipe are about 0.6λ deep, where λ represents the acoustic wavelength; and the 20-in. pipe flaws are about 1.9λ deep. The greater depth causes the reflected beam to be more collimated, meaning that the area of emergence on the pipe surface will be small. Second, the reflected beam in a thick wall has farther to travel to reach the surface. Clearly, if the reflected beam does not retrace the path of the incident beam, then the farther it travels the greater the separation between insertion and emergence points. For the 4-in. pipe, the widespread reflected beam does not have enough room to deviate much from the incident beam path. When it reaches the surface, the reflected beam illuminates an area that is at least partially but probably mostly eclipsed by the transducer, producing a high-amplitude signal. For the 20-in. pipe, the collimated reflected beam has more space in which to diverge from the incident

beam path. Therefore, the reflected beam may illuminate a small area of the pipe surface that may be far from the search unit, resulting in low amplitude and more variation of amplitude with varying flaw angle.

Comparison of aspect ratios yielded few surprises. A longer flaw generally produced an equal or higher amplitude than a similar shorter flaw. There were some exceptions--the longest flaw in a few cases reflected less than the medium length flaw--which at this time defy confident explanation.

Nonvertical flaws of less than maximum length and vertical semicircular flaws rarely proved reportable (100% DAC) in the 45° inspection. The best results were achieved using the 0.25-in. search unit. Many of the data fell below 50% DAC and were, therefore, not even recordable; almost all the data were above 20% DAC; and none were below 10% DAC.

Impact of Machined Flaw Data on Inspection Reliability

Inspection of machined notches of varying angle and aspect ratio in pipe specimens indicates poor recording and reporting probability. The flaws were of maximum allowable (minimum rejectable) depth, and inspections were carried out using an ASME Section XI, 1977 DAC.

The following major conclusions were reached from the inspection:

- 60° beam inspection will detect nonvertical flaws more reliably than 45° inspection. 45° inspection is generally better for 0° flaws. Use of both beam angles is indicated.
- Detection probability of nonvertical flaws is enhanced by high beam spread, which may be achieved by low frequency and/or low transducer size.
- Nonvertical flaw orientation becomes more of a problem as pipe wall thickness increases. In inspection of thin wall pipe, countermeasures may not be necessary.
- Reliable detection of nonvertical flaws, especially flaws of high aspect ratio (short flaws), will require an increase in inspection sensitivity through changes in calibration reflector and/or recording and reporting levels.

The data set will be complete with the inspection of the 20-in. SS pipe specimen. In addition, these samples will be inspected with a 1.5-MHz dual search unit in hopes of reducing detectability dependence on flaw orientation. The spectra of beams reflected from certain flaws will be examined to check the theory that low frequencies enhance nonvertical flaw detection.

TASK 8: ROUND ROBIN PREPARATION

Preparations for the round robin inspection are proceeding although sample preparation (see Task 6) has delayed the work by several months. Contractual negotiations with inspection agencies are in progress. The test protocol has been written and will be sent to the inspection agencies for comment. The SUTARS (Search Unit Tracking and Recording System) has been delivered and system installation and checkout is in progress.

FUTURE PLANS

An attempt will be made to shorten the projected delay due to sample preparation. Efforts are continuing to define the impact of decreased inspection sensitivity resulting from the 1977 Code revisions and the formulation of recommendations for correcting this problem area. A revised program schedule and redirection of effort will be defined during the month of August.

REFERENCES

1. Ermolov, I. N. April 1972. "The Reflection of Ultrasonic Waves from Targets of Simple Geometry." Nondestructive Testing 5:2(87-91).
2. Werneyer, R., F. Walt, and M. Klein. 1977. Mathematisches Model Zur Fehlerrekonstruktion bei der Ultraschall - Impulsecho und Tandimprung und Ergebnisse aus Modellversuchen. Report No. 770212-TW, Institut Fur Zerstörungsfreie prüfverfahren (IZfP), Saarbrücken, West Germany, p. 6.
3. Hooper, J. L., ed. 1979. Reactor Safety Research Programs Quarterly Report, April 1-June 30, 1979. NUREG/CR-0962, PNL-3040-2, prepared for the Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.*

4. Hooper, J. L., ed. 1980. Reactor Safety Research Programs Quarterly Report, July 1-September 30, 1979. NUREG/CR-1009, PNL-3040-3, prepared for the Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.*

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

TASK A - IRRADIATION EXPERIMENTS(a)

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

M. E. Cunningham
R. E. Williford

SUMMARY

Irradiation of instrumented fuel assembly (IFA)-432 and IFA-513 continued until May 5, 1980, when the reactor was shut down for fuel repositioning and removal of leaking rods. Rod 5 in IFA-513 was later found to be leaking and was removed; no replacement rod was made. IFA-527 was completed and loaded into the reactor; its actual startup was delayed until July 2, 1980.

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). The current program's general objectives include 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 with in-reactor data; utilizing ex-reactor testing for a better understanding of fuel rod mechanical behavior; and integrating this information into the FRAPCON and FRAP-T series of codes used in fuel modeling.

Irradiation tests are being conducted in the Halden Boiling Water Reactor (HBWR) in Norway. One test (IFA-431) has been completed; and two other test assemblies are currently being irradiated (IFA-432 and IFA-513). The fourth and final test in the program (IFA-527) is under construction, and its irradiation will begin in July 1980.

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

Task A of this program is concerned with conducting irradiation tests and coordinating the necessary activities associated with testing (test rig design, safety analysis, fuel shipments, postirradiation examination procedures, fuel disposal, etc.).

TECHNICAL PROGRESS

Irradiation of IFA-432 and IFA-513 continued. During the reactor shutdown from May 5 to July 2, 1980, rod 5 of IFA-513 was determined to have a small leak. It is believed that this leak occurred on October 10, 1979, when the lower thermocouple and the pressure transducer simultaneously failed and elongation was abnormally high. The rod was removed from the assembly; however, no replacement rod was made.

The transfer of IFA-432 rod 7 (unirradiated) and rod 8 (irradiated) from Halden to Harwell, UK, was prepared this quarter. The transfer will permit fuel axial compliance tests to begin at Harwell. The actual shipment was delayed until the first week in July.

FUTURE PLANS

The following major events are scheduled for this task in the coming quarter:

- completion of calibration and startup ramps for IFA-527
- compliance testing and fission gas release tests at Harwell, UK.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

TASK B - DATA QUALIFICATION AND ANALYSIS(a)

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

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

SUMMARY

A major objective of the Experimental Support and Development of Single-Rod Fuel Codes is the irradiation of instrumented fuel assemblies (IFAs) to obtain well-characterized data. Task B of this program is responsible for qualifying and analyzing these data. During this quarter three papers were presented at the 1980 Enlarged Halden Program Group Meeting. Work also continued on the four major areas of analysis: fuel relocation, transient temperatures, modeling and data errors, and fission gas release.

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). 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 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. The qualification and analysis of the data obtained from in-reactor testing of fuel rods is the responsibility of Task B of this program.

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

Task B work is divided into three subtasks:

- Subtask B-1 - Data Processing: This subtask is responsible for receiving, correcting, characterizing, and presenting data obtained from 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 obtained. Specific areas of interest for FY-1980 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 and are organized by subtask.

SUBTASK B-1 - DATA PROCESSING

Activities during this quarter were related to preparing data for the data analysis subtask.

SUBTASK B-2 - DATA REPORTS

The 1980 Enlarged Halden Program Group Meeting was held in Lillehammer, Norway, from June 1-6, 1980. Three papers were presented based on program data and analysis. The first paper, "Status of the USNRC/PNL Halden Test IFA-432," discussed the thermal changes that have been observed in IFA-432 during its irradiation. Estimates of fission gas release, based on analysis of fill gas pressure data, were also presented. The second paper,

"Interpretation of Fuel Centerline Thermocouple Response to Reactor Scrams," discussed data obtained from a reactor scram on August 23, 1979. It was concluded that decalibration of the thermocouples (TCs) has resulted in measured full-power temperatures that are up to 20% lower than the true temperatures. In addition, the decalibration is of a type that cannot be detected by the linear power drop operation. The third paper, "Fuel Cracking and Relocation," discussed the application of temperature and cladding elongation data in the development of the Pacific Northwest Laboratory (PNL) fuel relocation model. This model related reductions in fuel thermal conductivity and elastic modulus to the stress state of the fuel.

CRAKR is a two-dimensional (r, θ) finite difference code that was developed at PNL in 1977-1978. This code analyzes heat transfer within a fuel rod as a function of crack geometry and eccentricity.⁽¹⁾ During this quarter work proceeded on documenting the code. Computer coding was improved, an option for nonsymmetrical cladding was added, and a rough draft of the code description and user's manual was begun.

In March 1980, PNL was asked to contribute to a document being prepared by the Committee on the Safety of Nuclear Installations (CSNI), which reports to the Organization for Economic Cooperation and Development (OECD). This document examines fuel rod behavior during loss-of-coolant accident (LOCA) conditions. The PNL contribution is an appendix discussing fuel rod thermal calculations and includes methods for calculating steady-state and transient temperatures and a review of the current data base for thermal and mechanical models and properties. The appendix was forwarded during this quarter to CSNI.

SUBTASK B-3 - DATA ANALYSIS

During this quarter data was analyzed in these areas of interest: fuel relocation, transient temperature response, modeling and data errors, and fission gas release.

Fuel Relocation Data Analysis

A report on the PNL fuel relocation model was published.⁽²⁾ The model uses a description of the crack geometry to couple the fuel rod thermal and mechanical behavior. The major conclusion of this work is that fuel cracking

and relocation cause significant reductions in the effective elastic modulus and effective thermal conductivity for the fuel. In addition, the operating cracked fuel does not appear to obey the solid cylinder models that have traditionally been used in fuel performance codes. Preliminary results indicate that these conclusions are unchanged at higher burnups (up to 860 GJ/kgU).

Additional efforts have focused on the qualification of cladding elongation data and the investigation of relaxation phenomenon observed from in-reactor data. The qualification of cladding elongation data is still in progress, with instrument response and sensitivity information being collected from Halden. A first analysis of relaxation phenomena from IFA-432 revealed a dependency of elongation rate on power ramp rate for rod 3 (small gap rod), which simulates swelled fuel. Rods 1 and 2 (standard and large gap, respectively) showed negligible ramp rate dependencies at beginning-of-life (BOL).

IFA-527 startup data are eagerly awaited because of the assemblies inverted power profile (lower power at the top of the rod). This experiment, when compared with previous assemblies, will indicate the importance and existence of axial slipping between fuel and cladding. The assumption of no axial slipping has been used in fuel relocation and mechanical model development.⁽²⁾

Transient Temperature Data Analysis

An abstract was submitted to the American Nuclear Society (ANS) for their November 1980 meeting to be held in Washington, D.C. The submittal discusses the observation that steady-state and transient temperature data can be correlated. An observed steady-state thermal resistance drop for rod 6 of IFA-513 between August 1979 and January 1980 is supported by transient temperature data obtained from scrams on August 23, 1979, and January 6, 1980.

Modeling and Data Error Analysis

The objective of this work is to evaluate errors and uncertainties in computer code predictions of steady-state fuel behavior. The current work concentrates on evaluating stored energy uncertainty for different fuel rod designs (boiling and pressurized water reactors), examining the problem of validating theoretical models and computer applications, and evaluating the effect on temperature prediction uncertainty when using a cracked fuel model.

During this quarter a qualitative discussion on model validation and verification was prepared for inclusion in the upcoming report. The following three definitions are an important part of this discussion.

- code verification - A process to check that a theoretical model has been correctly implemented on a computer.
- code analysis - Operation of the model to gain a better understanding of the model.
- code validation - Comparison of output from the computer code implementation of the models to data obtained from an observed physical situation.

The use of specific definitions is necessary for a common understanding of these facets of developing confidence in computer modeling.

The process of model validation is the most difficult to quantify and is dependent on the code in question and its intended use. It should be recognized that differences between the code and reality are not always due to inadequate models. Differences can also arise because of errors in observed data or errors in the input to the code. Therefore, it is always necessary to state what level of agreement between code and reality is required for validation of the code.

Fission Gas Release Data Analysis

During the past quarter, the internal gas pressure data from IFA-513 were examined to estimate fission gas release during irradiation. The internal pressure data for six fuel rods are shown in Figures 1 and 2 as a function of rod average burnup. In general, internal pressures increase gradually with irradiation, but there are regions where the data are inconsistent with the expected data trends. This is especially evident in rod 2 where the pressures decrease and then increase from 270 to 400 GJ/kgU. It is believed that this is due to electronic problems since pressure measurements from rod 2 taken after 400 GJ/kgU are consistent with earlier data.

Anomalous pressure measurements are also seen for the other rods from ~300 to 350 GJ/kgU. An abrupt pressure drop of about 0.22 MPa is observed during this period, which is not consistent with the measured fuel rod power

FIGURE 1. Gas Pressure Data for Rods 1, 3, and 5 of IFA-513

levels or temperatures. This pressure drop is assumed to be associated with the data acquisition system because all of the fuel rods showed identical behavior. Except for rod 5, pressure data taken after 350 GJ/kgU are consistent with earlier data and therefore the problem has apparently been corrected.

Pressure data from rod 5 show three pressure spikes during the first 350 GJ/kgU of irradiation; and extremely high pressures (~2 MPa) thereafter (see Figure 3). This rod was recently identified as being defective and was removed from the assembly. The extremely high pressures recorded since the

FIGURE 2. Gas Pressure Data for Rods 2, 4, and 6 of IFA-513

start of the fourth reactor operating cycle (350 GJ/kgU) indicate that the rod failed during the first power ramp of this cycle. However, the three prior pressure spikes suggest a small defect may have been present through most of the irradiation.

All three of the pressure spikes occurred after the reactor was shut down for an extended period of time and the moderator cooled to about 350K. Figure 3 shows internal pressure as a function of time immediately after each shutdown period. These data were taken after the moderator had been heated to 513K and with the reactor operating at a low power level. As can be seen,

FIGURE 3. Internal Pressure as a Function of Time During Pressure Spikes in Rod 5, IFA-513

the majority of the pressure increase occurred during the initial reactor heatup as evidenced by the high pressures recorded at time zero. The pressures continued to increase as the reactor was taken to full power and then decreased to values consistent with the initial fill gas pressure. The time required to reach equilibrium was ~15 hours.

The characteristics of the pressure spikes are consistent with a small amount of water entering the fuel rod during the extended shutdowns. Upon

heating to 500K, the water would vaporize and produce the high pressure reading observed at time zero. The subsequent pressure increase is directly related to the reactor power level, while the pressure decrease may be attributed to chemical reactions with the fuel and cladding that remove the water vapor and its components from the gas phase. A detailed kinetic study has not been attempted, but 15 hours seems a reasonable time to remove a small amount of water vapor from the system.

To estimate fission gas release from internal pressure measurements, it is desirable to use data taken at zero reactor power where temperature gradients along the fuel rod are minimized. Reliable data for zero power operation are not available, thus low power data are used to estimate fission gas release. Using these data and assuming that internal temperatures equal the moderator temperature produces a slightly higher estimate for fission gas release.

The pressure increase at low reactor power and the estimated release fractions are given in Table 1. Rod 5 data have been omitted because of the extremely high pressures that were observed in this rod after October 1979.

The estimated release fractions were based on the difference between two pressure measurements; thus, the repeatability of the measurements is of concern in determining the uncertainty of the estimates. Halden personnel have suggested the repeatability of the pressure transducers is $\pm 1.5\%$ or ~ 0.003 MPa at 0.2 MPa. However, examination of the data obtained during calibration of the pressure transducers in IFA-513 shows differences of up to 0.024 MPa between repeat pressurizations and somewhat greater differences during

TABLE 1. Summary of Pressure Increases and Estimated Gas Release for IFA-513

<u>Rod No.</u>	<u>ΔP, MPa</u>	<u>Fission Gas Release, %</u>
1	0.0	0.0
2	0.069	2.2
3	0.031	0.9
4	0.043	1.4
6	0.063	2.0

depressurization. These differences are thought to be associated with the mechanical behavior of the bellows and could decrease with continued operation. However, each transducer was given at least 10 pressurization cycles prior to calibration, which should have minimized these effects. Therefore, repeatability of the measurements is assumed to be ~ 0.02 MPa.

The total pressure increases at the end of the irradiation period (Table 1) do not greatly exceed the assumed repeatability of the measurements. This leads to large uncertainties in the estimated fission gas release, which in some cases exceed 100% at the 3σ confidence level. These large uncertainties preclude making meaningful comparisons between the fuel rods in IFA-513. The fission gas release appears to be in the range of 1-2%, which is reasonable for fuel temperatures below 1700K.

REFERENCES

1. Williford, R. E., and C. R. Hann. June 1978. "KRAKR-Steady-State Asymmetric Temperature Distributions in Nuclear Fuel Rods." Trans. ANS 28, June 18-22, 1978, San Diego, California, pp. 422-423.
2. Williford, R. E., et al. April 1980. The Analysis of Fuel Relocation for the NRC/PNL Halden Assemblies IFA-531, IFA-432, and IFA-13. NUREG/CR-0588, PNL-2709, Pacific Northwest Laboratory, Richland, Washington.*

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

TASK C - CODE COORDINATION AND EX-REACTOR TESTING(a)

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

M. E. Cunningham
W. N. Rausch

SUMMARY

The Pacific Northwest Laboratory (PNL) mechanics package (PELET) was incorporated into the joint PNL-Idaho National Engineering Laboratory (INEL) version of FRAPCON-2 as the code neared completion. PNL contributions were completed for the Code Development and User's Manual and for the Developmental Assessment documents. Fuel densification and swelling were incorporated into the PNL fuel relocation model. The fuel rod compliance test frame entered the proof test and calibration phase. Coding of the advanced relocation kinetics algorithm neared completion. Experimental support efforts yielded data confirming assumptions used in the fuel relocation and mechanical response models.

INTRODUCTION

The primary objectives of code development and experimental support efforts are to provide verified analytical models for implementation into FRAPCON-2 and FRAP-T and to coordinate code structure improvements and verification between PNL and INEL.

FY-1979 efforts resulted in fuel performance models that more closely represent physical reality in a preliminary code structure to facilitate the implementation of these models. The fact that operating nuclear fuel is cracked results in significant reductions in effective thermal conductivity and elastic moduli from those for solid UO_2 . The state of the cracked fuel was described by three primary mechanical parameters: crack roughness, gap

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

roughness, and crack pattern. However, those parameters are presently inferred from thermal in-reactor results. Efforts in FY-1980 have focused on establishing a firm physical foundation for the cracked fuel model and the implementation of these models into a combined PNL-INEL version of FRAPCON-2.

This task is structured to reflect the increased emphasis on mechanistic relocation and fuel-cladding mechanical interaction (FCMI) models that have resulted from the analysis of in-reactor data. There are four subtasks: code coordination and development, mechanical testing of short rods, FCMI model development, and model verification.

TECHNICAL PROGRESS

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

SUBTASK C-1 - CODE COORDINATION AND DEVELOPMENT

The PNL mechanics routine PELET was merged with the INEL version of FRAPCON-2, and both versions were subsequently merged into one code. PNL written contributions for the Code Development and User's Manual and for the Developmental Assessment documents were completed and forwarded to INEL. Test cases for the developmental assessment were prepared and forwarded to INEL. Papers were presented at the annual Halden Project Group Meeting and at the Joint German-Japanese-American meeting. Fuel densification and swelling were included in the PNL relocation model.

SUBTASK C-2 - MECHANICAL TESTING OF SHORT RODS

These experiments are designed to obtain out-of-reactor data on the diametral and axial FCMI behavior of short-length UO_2 -Zircaloy rods. Emphasis is placed on loading systems present during irradiation.

Fabrication of test frame components was completed. The apparatus was assembled and proof testing began. Enough components of the instrumentation and electronics package arrived to assemble and begin calibration. An

equipment design review was completed for the comparison tests on irradiated rods being conducted at Harwell, UK. A letter of experimental objectives was also prepared and forwarded to Harwell.

SUBTASK C-3 - FCMI MODEL DEVELOPMENT

An advanced model is under development to economically model the effects of asymmetric pellet cracking that result in fuel column anisotropy and FCMI. Programming of the relocation kinetics algorithm (FRAGMT) is nearly complete. The algorithm uses a linear perturbation technique to simultaneously ratchet all irregular fuel fragments to their thermal-mechanical equilibrium positions at each point of the input power-center temperature history. The highly non-linear fragment ratcheting motions and asymmetric gap evolution are expected to be representable as a stepwise linear approximation in real time. This algorithm will permit numerical verification of the parameters and assumptions of the Task B relocation model as a function of fuel fragment geometry, plus provide information relevant to the study of localized fuel-cladding mechanical interactions.

SUBTASK C-4 - MODEL VERIFICATION

Experimental efforts are yielding data that verify the three primary mechanical parameters of the cracked fuel relocation and mechanical response model. An integrated set of experiments is being conducted at PNL to provide data quantifying the crack pattern, crack roughness, and gap roughness parameters. Surface interaction experiments conducted on single-crack systems have shown a dependence of effective surface roughness on stress level that was caused by rotation of the fragments as the load increased.

FUTURE PLANS

The following activities are planned for the next quarter.

- Developmental assessment test cases for the completed PNL-INEL joint version of FRAPCON-2 will be run and included in the document.

- A real-time simulation of fuel fragment relocation will be calculated with the advanced relocation kinetics code FRAGMT.
- Data from the first series of short rod mechanical compliance tests will be obtained.
- Compliance testing on irradiated IFA-432 rods will commence at Harwell, UK.

FUEL SUBASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM(a)

C. L. Mohr, Program Manager
E. L. Courtright, Project Manager
R. E. Schreiber, Deputy Project Manager
9-Rod Test Train
R. L. Goodman, Deputy Project Manager
PBF Severe Fuel Damage Test Trains
F. E. Panisko, Deputy Project Manager
Super Sara Test Program

G. S. Allison	J. King
L. M. Brown	L. L. King
L. R. Bunnell	R. K. Marshall
G. M. Hesson	M. S. Quigley
R. L. Hjelm	J. D. Rising
D. E. Hurley	R. D. Tokarz

SUMMARY

Progress was made in the areas of thermal-hydraulic analysis and test assembly instrumentation evaluation and selection in support of the Power Burst Facility (PBF) Severe Fuel Damage Program. A two-dimensional (R-Z) model of the proposed 32-pin test assembly for the PBF tests was developed to aid in experiment design evaluations and parametric studies and will be used with the TRUMP heat transfer computer code. A review of the measurement requirements for the PBF program as given in the Experiment Requirements Document for the Severe Code Damage Assessment Program was completed, and a proposed instrumentation package was outlined for the first two PBF test train assemblies. Preliminary design of the test train assemblies for the PBF tests was begun as was the detailed design of the lower portion of the test train assemblies.

Progress on the Super Sara Test Program (SSTP) centered on establishing the technical feasibility of achieving the desired testing conditions based on the Pacific Northwest Laboratory (PNL) conceptual test train design, thermal-hydraulic and instrumentation analyses, and the proposed test operations.

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

INTRODUCTION

The Fuel Subassembly Procurement and Irradiation Test Program is divided into four tasks for FY-1980 with the following objectives:

- Task A - PBF Test Train Hardware: To provide instrumented test train hardware for selected PBF experimental programs. Assembly hardware will include replacement quadrants for reactivity-initiated accident (RIA)-type four-rod test trains and a fully instrumented nine-rod test assembly for the RIA 1-7 and operating transient (OPTRAN) 1-3 experiments. In addition, this project will develop test train concepts consistent with the stated experimental objectives of the proposed PBF small-break loss-of-coolant accident (LOCA) test program.
- Task B - Gap-Conductance Test Analysis: To provide testing methodologies and analysis techniques for obtaining fuel rod thermal performance from PBF experiments and provide post-test analysis and data interpretation of experimental results.
- Task C - Dual-Purpose Assemblies: To transfer the instrumented fuel assembly (IFA)-513 dual-purpose test rods currently under steady-state irradiation in Halden to the PBF for subsequent transient testing.
- Task D - Multirod Test Trains - SSTP: To supply fully instrumented and fueled test assemblies, analytical services, and engineering support for the SSTP in the ESSOR reactor at Ispra, Italy. The program is designed to yield important experimental data related to fuel rod deformation and postaccident coolability of damaged fuel rod clusters after small-break, medium-break, and large-break LOCAs.

TECHNICAL PROGRESS

PBF TEST TRAIN HARDWARE

4X Replacement Quadrants

No work was performed on the final assembly of the 4X replacement quadrants. All component parts, including the high-pressure connector parts, and all instruments necessary to complete the final assembly are on hand; however, progress has been delayed pending future design modifications that may be required to meet revised experimental objectives. Information concerning the revised experimental objectives is not yet available because modifications to existing 4X hardware must be made.

Nine-Rod Test Train

Design and fabrication for the abnormal OPTRAN tests 1-3 and RIA tests 1-7 are proceeding well with minor adjustments in schedule and scope. All materials were acquired and provided to the various fabrication vendors. Final assembly and flow testing will be conducted at Idaho National Engineering Laboratory (INEL) by EG&G Idaho. Remote-handling equipment design is nearly complete and draft procedures are written.

PBF Severe Fuel Damage Program

Scoping thermal-hydraulic analytical calculations were completed, and instrumentation packages were proposed for the first two test train assemblies. Bundle power requirements and heat losses were determined from analytical calculations; in addition, shroud-insulation criteria for the high-temperature PBF tests were established. Instrumentation packages consistent with the stated experimental objectives of the PBF program were developed and transmitted to EG&G for their review and comment.

Scoping Thermal-Hydraulic Analytical Results

A two-dimensional (R-Z) model of the proposed 32-pin test assembly for the PBF severe damage experiments was developed for use with the TRUMP heat transfer computer code. An R-Z model was developed because the R- θ -Z model was slow and expensive to use in performing parametric evaluations of various

test assembly geometries. In addition, the axial resolution in the R- θ -Z model was very poor due to limitations on the total number of nodes. By neglecting temperature variations in the θ (azimuthal) direction, a smaller model with superior axial resolution was developed. The assumption of azimuthal symmetry is reasonable for use with the low-conductivity insulating shroud concept. For higher conductivity shrouds, the R-Z model gives information relating to overall energy balance and average steam cladding as well as shroud temperatures. The R-Z model temperature predictions for high-conductivity shrouds are not as accurate as the temperature predictions for the low-conductivity shroud; however, they are adequate for design support calculations and scoping analysis purposes.

Figure 1 shows a transverse view of the R-Z TRUMP model developed for the PBF scoping analysis. The 32 fuel pins are lumped together in three concentric fueled rings with steam flow through channels between the three rings; the doubly contained insulated shroud is also assumed to be a concentric ring. Each fuel ring is divided into three radial UO_2 nodes and two Zircaloy cladding nodes; the steam channels have one radial node; and the shroud has four radial nodes. Figure 2 shows an axial view of the PBF R-Z model. As can be seen, only the bundle region above the water level, which is divided into 10 axial nodes, was modeled in these scoping calculations.

The geometry of this model provides that UO_2 volume, cladding volume, cladding surface area, and pellet-cladding gap are in correct proportion for each fuel ring. The doubly contained insulated shroud is modeled to its proposed thickness, although it is round instead of the proposed hexagonal shape. The total steam flow area is correct, and the steam rates in the channel are adjusted to yield a uniform ratio of steam flow rate to cladding heat transfer area as exists in the proposed PBF 32-pin fuel bundle. For radiation heat transfer calculations, the inner fuel rings can "see" the outer fuel rings and shroud using the view factors calculated for a 32-pin bundle geometry. With the simple concentric ring geometry, radiation from cladding to steam can be included as an important part of the model.

FIGURE 1. Transverse View of the PBF R-Z TRUMP Model

The R-Z model can be used for both PBF and Super Sara small-break calculations with the only differences being the fueled length of the bundle and the heat transfer characteristics beyond the shroud. The R-Z model can be modified for Super Sara by lengthening the axial nodes (or adding more axial nodes) and by adding radial nodes to represent the pressure tube, the CO₂ gas coolant flow between pressure tube and safety tube, the safety tube, and the reactor moderator coolant.

Preliminary thermal-hydraulic scoping calculations were completed. A simulation of the boiled-off coolant situation expected for PBF tests was performed at PNL using the R-Z TRUMP model for the proposed 32-pin test

FIGURE 2. Axial View of the PBF R-Z TRUMP Model

assembly. Additional independent scoping calculations were completed at the General Electric Vallecitos Nuclear Center with the THTD computer code.

Two sets of shroud calculations were completed with the only difference between the two being the assumed thermal conductivity of the shroud material. In the first run, the shroud thermal conductivity was 0.14 W/m-K (0.083

Btu/h-ft-⁰F); and in the second run, it was 1.69 W/m-K (1.0 Btu/h-ft-⁰F). The thermal conductivity value used for the low-conductivity shroud calculations (0.14 W/m-K) was supplied by the manufacturer of the felted or woven ZrO₂ cloths that are proposed as insulation for the PBF test assemblies. Work is being done at PNL to verify the thermal conductivity values of these two materials.

The R-Z model assumes a stable liquid level and a constant, uniform steam flow through the bundle. It also assumes that a 558K (545⁰F) heat sink exists immediately outside the shroud. For initial runs, neither the energy release from the metal-water reaction nor the heat transfer by radiation to the steam was included in the analysis.

In all cases, the calculations were run until a steady state was reached. Additional assumptions and conditions common to all of the runs are:

- 32-pin test bundle, 0.91 m (3 ft) long
- 0.23-m (~9-in.) water level
- axial power profile based on a peak-to-average of 1.35
- radial power profile across the bundle based on uniform 3% UO₂ enrichment
- test assembly water inlet temperature of 303K (86⁰F)
- 6.89-MPa (1000-psi) test assembly pressure
- cladding and shroud emissivities of 1.0.

Initial calculations were made with the model at a water level of 0.10 m (4 in.); however, it was noted that small variations in the water level about that point resulted in significant variations in the steam flow rate and test pin temperature. For this reason, it was believed that a 0.23-m (9-in.) water level would be a much better boundary condition for the test, especially with the proposed low-conductivity shroud material.

The results of the low thermal conductivity shroud case ($k = 0.14$ W/m-K) are presented in Figures 3 and 4. The curves in Figure 3 show that peak

FIGURE 3. Cladding, Steam, and Shroud Temperatures Calculated for Low-Conductivity Shroud (0.14 W/m-K, 0.083 Btu/hr-ft-°F) for Various Assembly Peak Powers

cladding temperatures in the desired range can be obtained with very modest test section peak powers even with a 0.23-m water level. The uniform steam flow rate for this case is 0.65 g/s (5.19 lb_m/h). The curves also show that there is a temperature difference of less than 111K (200°F) between the peak cladding and shroud temperatures; thus, the test pin-to-pin temperature variation is very small. For a detailed examination of the pin-to-pin temperature variations, the R-θ model should be run.

Figure 4 presents the axial temperature distribution for 11.48 kW/m peak test section power and the 0.23-m water level. Since the steam and shroud

FIGURE 4. Cladding, Steam, and Shroud Axial Temperature Distribution for Low-Conductivity Shroud (0.14 W/m-K, 0.083 Btu/hr-ft-°F) for an Assembly Peak Power of 11.48 kW/m (3.5 kW/ft).

temperature follow the cladding temperature closely, the radial heat losses are small, which indicates that the system is well insulated radially. A system heat balance shows that of the 9.08 kW generated above the liquid level, 2.24 kW (25%) go to the steam and 6.84 kW (75%) are lost radially. The average outlet steam temperature is 1981K (3106°F). The metal-water reaction would be expected to increase temperatures well above those reported from these initial runs. The effect of the radiation to the steam is not expected to have a strong influence on temperatures, and it is believed that the addition of radiation to steam will not greatly affect the results. This is based on two premises. The first is the hyperbolic nature of the temperature-heat transfer coefficient relationship:

$$T_c = T_s + \frac{Q/A}{h}$$

where T_c = clad temperature
 T_s = steam temperature
 Q/A = heat flux
 h = heat transfer coefficient.

The steam temperature at any given axial location is basically determined by an energy balance; that is, the integrated net power up to the point in question divided by the product of the steam mass flow rate and its effective heat capacity. The radiant heat transfer to steam enters the argument in that it affects the value of the heat transfer coefficient and, therefore, the cladding temperature.

The conditions of the proposed test assembly with the low-conductivity shroud can be used for a semiquantitative illustration of this argument. The peak heat flux in this assembly late in the transient when quasi steady-state conditions are reached is approximately $1.58 \times 10^4 \text{ W/m}^2$ (5000 Btu/h-ft^2). When this value of heat flux is entered in the temperature-heat transfer coefficient relationship, the cladding-to-steam temperature difference can be calculated as a function of the heat transfer coefficient as shown in Figure 5.

From the initial calculations, it was found that for the calculated steaming rates the convective heat transfer coefficient in the test assembly varied with time and location; however, a value of $170 \text{ W/m}^2\text{-K}$ ($30 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$) is calculated as a reasonable average. Locating this value (h) in Figure 5 gives a cladding-to-steam temperature difference of 94K (170°F). If a radiative component is added, a larger effective heat transfer coefficient results: $h = h_c + h_r$.

Values of h_r will range from 341 to $454 \text{ W/m}^2\text{-K}$ (60 to $80 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$) at the temperature levels of interest. If $397 \text{ W/m}^2\text{-K}$ ($70 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$) is added to the $170 \text{ W/m}^2\text{-K}$ ($30 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$) convective component, a total effective heat transfer coefficient of $568 \text{ W/m}^2\text{-K}$ ($100 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$) results. At this value Figure 5 shows a cladding-to-steam temperature difference of 28K (50°F). In other words, adding radiation to the steam

FIGURE 5. Illustrative Cladding Temperature Dependence on Heat Transfer Coefficients

decreases the cladding temperature by 67K (120°F), which is not a large amount at the temperature levels under consideration.

The second premise arises from radial heat loss considerations and is peculiar to the test assembly, which is surrounded by a relatively low-temperature sink. Radial heat losses are large and due primarily to surface-to-surface radiation. This is a parallel heat path to the steam, and the heat flux to the steam should be correspondingly reduced. Calculations for the

PBF assembly show that approximately one-quarter of the heat generated within the bundle is carried out with the steam. A second curve, shown in Figure 5, is based on a heat flux one-quarter of that used in previous calculations and emphasizes the prior conclusions: Inclusion of surface-to-steam radiation in the calculations will change the results very little.

The results for the high thermal conductivity ($k = 1.69 \text{ W/m-K} = 1.0 \text{ Btu/h-ft-}^{\circ}\text{F}$) shroud material are presented in Figures 6 and 7. The uniform steam flow rate for this case is 6.39 g/s ($51.05 \text{ lb}_m/\text{h}$), and the radial heat losses for these cases are an order of magnitude larger than the low-conductivity shroud. A test section power of 98.4 to 114.8 kW/m is required to produce the required test temperature. In addition to the undesirably high power levels required to attain the test conditions, the high-conductivity shroud results in a large pin-to-pin temperature gradient as evidenced by the more than 444K (800°F) difference between cladding and shroud temperatures. Therefore, the R- θ model must be run to determine the maximum fuel pin temperature with any accuracy. The peak cladding temperatures presented in Figures 6 and 7 actually represent a sort of average temperature at any given axial location. The actual peak fuel pin temperature would be a few hundred degrees hotter, and the minimum fuel pin temperature would be a few hundred degrees cooler. Temperature predictions for the high-conductivity shroud are not as accurate using only the R-Z model as the predictions for the low-conductivity shroud; however, they are adequate for comparison purposes.

It can be concluded that the temperatures required by the PBF Severe Fuel Damage Program can be reached with a 0.23-cm water level if the test can be designed using a low thermal conductivity shroud material. Obviously, a far superior overall test can be designed using the low thermal conductivity shroud material because of the ease of obtaining the test conditions in existing test reactors, the degree of simulation of the postulated power reactor accident, and the degree of analytical sophistication required to predict the test performance.

Test Train Assembly Instrumentation

A detailed review of the measurement requirements (see Table 1) for the PBF program was completed based on the Experiment Requirements Document. A

FIGURE 6. Cladding, Steam, and Shroud Temperature Calculated for High-Conductivity Shroud (1.69 W/m-K, 1.0 Btu/hr-ft-°F) for Various Assembly Peak Powers

proposed instrumentation package was outlined for detailed discussions with EG&G for the first two test train assemblies that includes a consolidated listing of the results of the instrument selections (including a definition of the instrument types and the relative numbers of each instrument type that are proposed for the first two PBF Severe Fuel Damage Test Assemblies). Fuel, cladding, and steam temperatures are all important parameters to be measured and recorded during the experiments. Fuel rod and system pressure histories data are also desirable. Additional items that were identified as necessary for the experiments include a liquid level indicating and control device, self-powered neutron detectors (SPNDs) to measure bundle power and possibly indicate fuel movement during the tests, on-line sample systems to provide

FIGURE 7. Cladding, Steam, and Shroud Axial Temperature Distribution for High-Conductivity Shroud (1.69 W/m-K , $1.0 \text{ Btu/hr-ft-}^\circ\text{F}$) for an Assembly Peak Power of 105 kW/m (32 kW/ft)

information on hydrogen gas evolution from the metal-water reaction and provide data on the steam/hydrogen proportions, and possible use of melt profile devices to indicate peak temperatures within the bundle after conventional thermocouples (TCs) no longer function.

Table 2 presents the proposed instrumentation package; three basic types of TCs will be used. High-temperature TCs to be used in the assemblies will be one of two types: either the Re-augmented W sheath with W/Re TC and BeO insulation (which has an upper temperature measurement of approximately 2473K) or the Zr sheath W/Re or Pt/Rh TC with BeO insulation as developed by EG&G (which has an upper temperature measurement of approximately 2100K). The

TABLE 1. Summary of Instrument Measurement Needs for the PBF Severe Fuel Damage Experiments

Measurement	Instrumentation
Cladding Temperature	External/Internal Thermocouples (TCs)
Peak Cladding Temperature	Internal TCs
Fuel Temperature	Centerline TCs
Steam Temperature	Steam Probe
Cladding Oxidation	Eddy Current Probe
Fuel Rod Pressure	Pressure Transducers (PXDs)
System Pressure	PXDs
Bundle (axial) ΔP	PXDs or Pressure Tube Taps to Out-of-Reactor Sensors
Liquid Level	PXDs/Heated TCs/Fission Detectors/Perforated MgO Cable
Power	Self-Powered Neutron Detectors (SPNDs)/Self-Powered Gamma Detectors (SPGDs)/Fission Chambers
Fuel Movement	Fission Chambers
H ₂ Evolution	On-Line Gas Sampler
Steam/H ₂ Versus Time	Gas Samples
Incipient Melting	Bi-Metallic Strips

Re-sheathed TCs^(a) should be acceptable for the PBF tests, except for one design modification: To improve the TC response time, TC assemblies will be back-filled with helium rather than argon gas. The Zr sheath TCs will be resistance welded at the hot junction end to the inner surface of the cladding tube; consequently, the hot junction end must be designed to accommodate the resistance weld without destroying the integrity of the sheath. In addition, the Zr sheath TCs should be designed with a grounded junction. The remainder of the temperature measurements made by TCs within the test train assembly will be made with Inconel-600 sheath TCs with chromel-alumel elements and MgO insulation. The Inconel-600 TCs will be used for the lower temperature

(a) Designed by Hanford Engineering Development Laboratory (HEDL), Richland, Washington, and built by Kaman Science Corporation, Colorado Springs, Colorado.

**TABLE 2. PBF Severe Fuel Damage Program Test Assemblies -
Proposed Instrumentation Package**

Measurement Requirement	Instrument	Number of Sensors	Range	Remarks
Clad Surface Temperature (inside)	Thermocouple	15	280-2100K	Zr sheath, W-Re or Pt-Pt/Rh TC with BeO insulation.
Clad Surface Temperature (outside)	Thermocouple	8	280-1600K	Inconel-600, Cr-Al TC, MgO insulation, located in lower region of assembly.
Plenum Temperature	Thermocouple	1	280-2100K	Zr sheath, W-Re or Pt-Pt/Rh TC, BeO insulation.
Fuel Centerline Temperature	Thermocouple	4	280-2475K	Re-augmented W sheath, W-Re TC, Hafnia insulation.
Shroud Temperature (inside)	Thermocouple	16	280-2100K	Zr sheath, W-Re or Pt-Pt/Rh TC, BeO insulation.
Shroud Temperature (outside)	Thermocouple	16	280-1600K	Inconel-600, Cr-Al TC, MgO insulation.
Coolant Temperature	Thermocouple	12(a)	280-2100K	Zr sheath, W-Re or Pt-Pt/Rh TC, BeO insulation. Steam probes.
Plenum Pressure	Eddy current transducer	4	0.1-8.6 MPa	Must be mounted at bottom of rod because of sensor temperature limitations.
Coolant Pressure	Eddy current or pressure taps	3	0.1-15.5 MPa	Must be mounted on bottom of rod because of sensor temperature limitations.
Neutron Detectors	SPND	16	1×10^{14} n/cm ² -s	Cobalt emitter, Inconel-600 collector and sheath, MgO insulation.
Coolant Flow Rate	Turbine flowmeter	2	18-180 gpm	Preconditioning flow rate.
Fuel Rod Failure Indicator	Self-powered with Cr-Al TC	7	---	Cr-Al TC with ground interrupt; must be located at bottom of rod.

(a) 3 levels of 4

measurements (below 1600K). All TCs will be designed with an in-line seal to prevent fission product migration up the cable in the event the probe/sheath fails during the experiment.

Additional instrumentation within the test train assembly will include eddy current pressure transducers and eddy current or pressure taps for both plenum pressure and coolant pressure measurements (see Table 2). SPNDs will be located within the bundle to determine test assembly powers and may 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 an Inconel-600 collector and sheath with a cobalt emitter and MgO insulation. In addition, a PNL-designed 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. Last of all, the coolant flow rate during the water flow preconditioning portion of the test will be measured with turbine flowmeters that will be integral to the test train assembly.

Radial and axial arrays showing the placement of the instruments within the test train assemblies are shown in Figure 8. The entire 32-pin bundle cross section for the proposed test assembly is shown, and the proposed instrument placements given in Table 2 are detailed. There are grid spacers at levels 3, 5, and 7 (approximately) within the test train assembly. For the initial instrumentation packages the total number of penetrations through the closure head at the top of the test train assembly has been limited to approximately 100 hard lines.

Additional conceptual instruments that will require development were identified for the PBF Severe Fuel Damage Tests and include a possible liquid level detection and control device, a fuel bundle temperature profile instrument, steam quality and hydrogen generation measurement instruments, and fuel bundle plugging and fuel motion monitor instruments.

SUPER SARA TEST PROGRAM

The scope of this project includes provisions for supplying three fully instrumented and fueled test train assemblies and analysis and engineering support for irradiation in the ESSOR reactor as part of the SSTP. The program

FIGURE 8. Severe Fuel Damage Program Test Assembly Instrumentation

objective is to determine the behavior of typical light water reactor (LWR) fuel rods experimentally under large- and small-break LOCA conditions. Figure 9 shows the ESSOR test reactor Super Sara in-pile loop with once-through coolant and 2-m fuel length. The loop arrangement is shown in Figure 10 and includes an out-of-pile twin thermal-hydraulic test facility; Figure 11 depicts the PNL conceptual test train design. The spray desuperheater cools the fuel bundle exit steam, which prevents excessive loop piping temperatures. The ZrO_2 spheres above the fuel bundle vaporize any water that falls from the top of the test train.

Since the main emphasis of the program is to analyze fuel bundle behavior during small-break LOCA test conditions, 13 small-break LOCA tests are currently proposed that are divided into four types based on the fuel rod damage mechanisms as defined below.

- Type 1 - Cladding ballooning and rupture without oxidation
- Type 2 - Cladding oxidation without ballooning and rupture, with no constraint on T_{max} cladding
- Type 3 - Cladding ballooning and rupture without oxidation (Three Mile Island type), followed by a continued temperature rise leading to external and internal cladding oxidation
- Type 4 - Cladding ballooning and rupture with simultaneous and/or prior oxidation, with no constraint on T_{max} cladding.

Work last quarter showed the need for a high thermal resistance shroud, especially for higher temperature tests. Investigation of achieving such a suitable shroud led to various zirconia (ZrO_2) products of high and low theoretical density with thermal conductivities ranging from approximately 0.14 to 1.73 W/m-K. Preliminary thermal calculations using the computer code TRUMP indicate that the desired fuel rod temperatures (as high as 2170K) can be reached while maintaining low rod and bundle power ratings.

Candidate instruments and their placement within the fuel bundle were examined. Different instrument packages will be needed for large- and small-break tests; however, the instrument layout will be similar to that proposed for the PBF Severe Fuel Damage test train given early in this section.

FIGURE 9. Layout of the Super Sara In-Reactor Pipework

FIGURE 10. Super Sara LOCA Simulation Loop Arrangement

FIGURE 11. Super-Sara Small-Break Test Assembly Concept

Results indicate that:

- State-of-the-art instrumentation will not survive the high steam temperature at the top of the bundle from small-break tests.
- Cladding surface TCs will not survive the maximum temperatures expected in the small-break tests, and reliance will have to be placed on internal TCs.
- At low steaming rates, the option of venting bundle steam through a separate tube and measuring the total released hydrogen may be preferable to an in-loop desuperheater.
- The feasibility of collecting and measuring hydrogen is being evaluated.
- Several methods of monitoring liquid level appear feasible. Selection criteria should consider minimizing the number of hard lines and the production of other useful data. The need and feasibility of monitoring liquid level during the tests require evaluation.

FUTURE PLANS

TASK A - PBF TEST TRAIN HARDWARE

Many major items--i.e., shrouds, grids, fuel rod components, and most instruments for the test train--are deliverable in the third quarter. Functional and calibration checks will be conducted.

In-cell remote-handling equipment will also be completed during the third quarter. Poolside remote-handling equipment will be completed in the fourth quarter, and all equipment and technology will be transferred to EG&G.

All nonirradiated assembly work will be completed during the fourth quarter with final assembly of all modules scheduled for completion in the first quarter of 1981.

Detailed design of the test train assembly No. 1 for the PBF Severe Fuel Damage Program will be started. Shroud feasibility studies and thermal-hydraulic experimental evaluations that are in progress will continue. Requests for quotes packages for critical instrument purchases and purchase orders for instruments will be placed. Ongoing feasibility studies on additional conceptual instrumentation concepts will be continued.

TASK B - GAP-CONDUCTANCE TEST ANALYSIS

A review of the preliminary PR-1 results will be performed to determine if useful fuel rod thermal-resistance data can be obtained.

TASK C - DUAL-PURPOSE ASSEMBLIES

The IFA-513 assembly is currently being irradiated at Halden. No further work is planned until this assembly is ready for discharge near mid-1981.

TASK D - MULTIROD TEST TRAINS - SSTP

Efforts will continue to evaluate and modify the conceptual test train design as required by refinements in program objectives and scope. Analysis techniques will be improved to model bundle test conditions more accurately.

Test train design and proposed testing conditions, including requisite preconditioning treatments, will be presented and discussed at planning and directional meetings to be held in Karlsruhe, Germany; Cadarache, France; and Ispra, Italy. The effects of fuel rod length, preoxidized cladding, and various shroud thermal resistances will be analyzed.

Test results design and proposed testing conditions, including realistic pressure, vibration, and thermal treatments, will be presented and discussed as planning and direct effects are to be held in Karlsruhe, Germany, Karlsruhe, France, and Japan. The effects of fuel rod length, pressurized cladding, and various thermal resistances will be analyzed.

The test results design and proposed testing conditions, including realistic pressure, vibration, and thermal treatments, will be presented and discussed as planning and direct effects are to be held in Karlsruhe, Germany, Karlsruhe, France, and Japan. The effects of fuel rod length, pressurized cladding, and various thermal resistances will be analyzed.

The test results design and proposed testing conditions, including realistic pressure, vibration, and thermal treatments, will be presented and discussed as planning and direct effects are to be held in Karlsruhe, Germany, Karlsruhe, France, and Japan. The effects of fuel rod length, pressurized cladding, and various thermal resistances will be analyzed.

The test results design and proposed testing conditions, including realistic pressure, vibration, and thermal treatments, will be presented and discussed as planning and direct effects are to be held in Karlsruhe, Germany, Karlsruhe, France, and Japan. The effects of fuel rod length, pressurized cladding, and various thermal resistances will be analyzed.

CORE THERMAL MODEL DEVELOPMENT(a)

D. S. Trent, Program Manager
M. J. Thurgood, Project Manager

SUMMARY

Developmental assessment of the COBRA/TRAC code has been initiated. Improvements to the code are being made as comparisons with data indicate they are needed. The prediction of countercurrent flow limiting (CCFL) is important in reactor safety applications. Code predictions of CCFL experimental data and the changes made in the code to make these predictions are presented in this report.

INTRODUCTION

The COBRA-TF computer code is being developed as part of the 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) component and system effects experiments. This project has two main objectives:

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

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

COBRA-TF is formulated to model three-dimensional, 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. 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 in the vessel described above. The resulting code is referred to as COBRA/TRAC and is being used to simulate several experiments, including the FLECHT, SEMISCALE, and JAERI test facilities.

During the past quarter predictions of a variety of two-phase flow experiments were initiated to assess the models used in the code. Significant improvements have been made to the code that allow prediction of CCFL in a variety of geometries. These improvements and the comparison of code predictions with CCFL experimental data are described below.

TECHNICAL PROGRESS

COUNTERCURRENT FLOW LIMITING

CCFL has a significant influence on the behavior of a reactor system during a LOCA. This phenomenon limits the rate at which emergency core cooling (ECC) water penetrates to the lower plenum (as in the case of cold leg injection) or the rate at which water can enter the core from above (as in the case of UHI or upper plenum, hot leg, or core spray injection). CCFL may be important to other parts of the system as well. It is therefore essential that a thermal-hydraulic computer code intended to predict the behavior of a reactor system during an accident be able to predict CCFL reliably. If this code is to be used to obtain best-estimate extrapolations into sizes, geometries, or conditions not covered by experimental data, it is necessary that the CCFL phenomenon be modeled by the code from a mechanistic rather than an empirical point of view.

Significant progress has been made toward understanding the physical mechanisms leading to CCFL.⁽¹⁻⁶⁾ The insight gained from these studies has been used to improve the physical models used in COBRA-TF to obtain a mechanistic prediction of CCFL. The following major changes were made to model CCFL:

- The falling film flow entrainment model was improved.
- The correlation for the size of drops formed from the falling liquid film was improved.
- The flow regime transition criterion for annular flow was improved.
- Nodalization for the pool was improved.

Downward Film Flow Entrainment Model

CCFL or flooding in tubes has its origin in interfacial instability and entrainment.^(1,3,7) As the liquid film flows down the tube wall, waves are created on the film surface. At sufficiently high vapor velocities and for sufficiently large wave amplitudes these waves become unstable and burst toward the center of the tube. The liquid drops formed from this wave breakup will be carried away by the vapor if the vapor velocity is high enough to lift the

drops against gravity. Since the amplitude of the waves is proportional to the film thickness, this process of wave instability leads to a limit on film thickness.

It has been proposed that droplet removal occurs because the pressure variation over the wave due to the vapor flow exceeds the stabilizing force of surface tension.^(3,7) The ratio between the two forces that will result in wave breakup can be given as:

$$C_1 \leq \frac{\rho_v V_r^2 \lambda a}{\lambda \sigma} \quad (1)$$

where C_1 contains the effects of wave shape on the surface tension and pressure forces. If a simple drag formulation is used for the pressure force and the wave amplitude is assumed to be on the order of four times the film thickness,⁽³⁾ the following expression is obtained:

$$0.5 \leq \frac{\rho_v V_r^2 \delta}{\sigma} \quad (2)$$

Equation (2) can be used to define a critical film thickness above which wave instability will occur.

$$\delta_{\text{CRIT}} = \frac{0.5\sigma}{\rho_v V_r^2} \quad (3)$$

Excess liquid flowing in the film resulting in a film thickness greater than the critical thickness will be removed from the film and carried away by the vapor.

Drop Size

Tatterson et al.⁽⁸⁾ have developed a correlation for the size of drops formed from the wave instability described above. Their correlation for the Sauter mean diameter is:

$$d = \frac{1.12 \times 10^{-2}}{\left[\frac{\rho_v V_v^2 \left(\frac{fs}{2} \right)}{\sigma D_H} \right]^{1/2}} \quad (4)$$

where $\frac{fs}{2} = \frac{0.046}{Re_v^{0.20}}$

D_H = tube diameter.

Annular Flow Regime Transition

Up to the present time, most advanced two-phase flow computer codes (including COBRA-TF) have used the vapor volume fraction as a criterion for transition to or from annular flow. This criterion is not adequate in the presence of high vapor velocities. That is, if annular flow exists and the vapor velocity is sufficiently high, it is impossible to go through a transition in flow regime to a bubbly flow even though the vapor volume fraction has decreased below a value at which annular flow is generally observed. Dukler et al.⁽⁵⁾ used this idea in developing a criterion for the transition to annular flow in vertical cocurrent flow. They argue that annular flow would exist if the vapor velocity were sufficient to lift liquid droplets entrained from the film. The velocity at which this occurs can be obtained from a simple balance between drag and gravity forces acting on the drop.

$$V_{CRIT} = \left[\left(\frac{8}{6} \right) \frac{d \Delta \rho g}{C_D \rho_v} \right]^{1/2} \quad (5)$$

If the vapor velocity exceeds V_{CRIT} and the relative velocity is sufficiently high to cause wave instability (Equation 3), then annular flow exists regardless of the vapor volume fraction. Annular flow is still assumed to exist at low vapor velocities and high vapor volume fraction as is the case for free-falling films.

Nodalization for Pools

In reactor safety applications, the inlet condition for the tube in which flooding occurs is often a liquid pool. It is therefore essential to model the flow in the liquid pool correctly to obtain accurate inlet conditions for the flooding tube or orifice. The most important aspect to model is the velocity and void fraction distributions in the pool. When a high-velocity vapor enters a pool from a tube or orifice, it can be expected that a high-velocity, high-void fraction region will exist in the pool directly above the tube or orifice relative to the fluid in the pool surrounding the inlet tube or orifice. One might expect a two-phase jet consisting of vapor and water drops at higher vapor velocities and water and vapor bubbles at lower vapor velocities. Liquid in the two-phase jet above the inlet can be expected to be rising while that surrounding the inlet may be falling.

The void and velocity distribution in the pool can be modeled by two-fluid computer codes if a sufficiently small mesh is used to resolve the gradients. However, this would require a very large number of mesh cells to model the liquid pool in the upper plenum of a PWR, which has several vapor jets entering the pool through orifices in the top nozzles of the fuel assemblies.

The optional subchannel formulation of the momentum equation and channel-splitting capability available in COBRA-TF can be used to model the pool with a reasonable number of nodes. This is done by modeling the pool in each region of the upper plenum with two subchannels: one for the area directly above the holes and the other for the area surrounding the holes. The channel directly above the holes has the same flow area and hydraulic diameter as the holes. This provides a two-region model of the pool, which has been successfully used to predict CCFL in a variety of geometries.

CCFL PREDICTIONS

Dartmouth Countercurrent Flow Experiments

Simulations were made of Dartmouth College⁽³⁾ air-water countercurrent flow flooding experiments in vertical tubes to evaluate the physical models and nodalization described above.

The test facility consisted of a vertical, transparent tube test section 2, 6, or 10 in. in diameter and 40-48 in. long and an upper and lower plenum. A 55-gal drum was used to construct the upper plenum where water entered through a 2-in. pipe at rates up to about 250 gpm. An overflow was cut out of the side of the upper plenum allowing a pool height of 18-20 inches. Air was injected in the lower plenum via a 10-in. pipe. The liquid penetration rate was determined by measuring the liquid accumulation rate in the lower plenum.

The experimental procedure was as follows:

- Airflow was set at a high enough value to stop all liquid penetration into the flooding tube.
- The water supply was then turned on, and a liquid pool was allowed to develop in the top plenum.
- Airflow was decreased, allowing water to penetrate the tube.
- Water accumulation in the lower plenum was measured.
- Airflow was checked for constancy.

The COBRA-TF nodalization was used to simulate these experiments. The test section was modeled with a single column of mesh cells 10 cells high. The bottom two mesh cells were used to model the collection of water in the lower plenum. The upper plenum was modeled with two columns of mesh cells. The first was located directly above the test section and had the same flow area and hydraulic diameter as the test section. The second modeled the flow area in the remainder of the upper plenum. The subchannel formulation of the momentum equation was used for the cross flow between these two channels. Air was modeled with steam at a suitable pressure (~30 psi) to give the correct density and was injected into the third mesh cell of the upper plenum. An overflow was modeled with a pressure boundary condition in the fourth cells of the upper plenum, which provided the correct liquid head in the pool. A node length of 6 inches was used in this simulation.

Experimental results were presented on plots of dimensionless gas flux, $J^{*1/2}$, as a function of dimensionless liquid penetration flux, $j_l^{*1/2}$, where for the i th phase:

$$J_i^* = \frac{\rho_i^{1/2} J_i}{[gD(\rho_f - \rho_g)]^{1/2}} \quad (6)$$

where D is the diameter of the test section.

COBRA-TF predictions of the 2-in. tube experimental results are shown in Figure 1. The comparison is quite reasonable. At high airflow rates film stability and entrainment in the test section or at the tube inlet limited the rate at which liquid could penetrate to the lower plenum. At lower airflow rates liquid penetration was limited by the bubbly flow in the pool above the test section inlet, which also limited the rate at which liquid could enter the test section.

COBRA-TF predictions of the 10-in. tube experimental data are shown in Figure 2. Again, the comparison is very reasonable and the limiting behavior was similar to that for the 2-in. test section.

FIGURE 1. Dartmouth Countercurrent Flow Experiment

Northwestern University CCFL Experiments

Air-water countercurrent flow experiments in a vertical test section containing an orifice plate were conducted at Northwestern University (NWU). The

FIGURE 2. Dartmouth Countercurrent Flow Experiment

purpose of the experiments was to study the penetration of liquid through the holes of the orifice plate against the upflow of air through the holes. Various hole sizes and numbers of holes were investigated. Air was injected below the orifice plate, and water was injected above it through a nozzle designed to eliminate the downward momentum of the water. An overflow pipe was connected to the upper plenum to maintain the pool height, and an air outlet pipe was connected above that.

The experimental procedure was as follows: For a given orifice plate, liquid injection nozzle height, and water flow rate:

- water flow was initiated in the upper plenum
- airflow was turned on to the desired rate
- the liquid penetration rate was measured
- the procedure was repeated for higher airflow rates.

Experimental data were simulated for an orifice plate containing fifteen 10.5-mm holes arranged in a 3x5 square array. The liquid pool above the orifice plate was modeled with two columns of mesh cells using the subchannel approximation for the lateral flow between the two. The first channel had a flow area equal to the total area of the 15 holes and a hydraulic diameter

equal to the diameter of a single hole. The second channel modeled the area surrounding the holes. A node length of 5 inches was used.

Two channels were used below the orifice plate: The lower one served as a liquid collection volume, and the top one modeled the region below the orifice plate into which air was injected. Liquid was injected into the fifth axial node in the upper plenum without axial momentum.

Experimental results were presented on plots of dimensionless gas flux, $H_g^{*1/2}$, as a function of dimensionless liquid penetration flux, $H_l^{*1/2}$, where for the i th phase

$$H_i^* = \frac{\rho_i^{1/2} j_i}{[g W_8 (\rho_l - \rho_g)]^{1/2}} \quad (7)$$

where W_8 is a characteristic dimension that is an interpolation between D_H used in j^* scaling and the wavelength of the Taylor instability, $\{\sigma/[g(\rho_f - \rho_g)]\}^{1/2}$, used in k^* scaling.

COBRA-TF predictions of the data are shown in Figure 3. The predictions are quite reasonable with liquid film instability and entrainment limiting penetration at higher vapor velocities and bubbly flow in the pool limiting liquid penetration at lower vapor velocities. At lower vapor velocities less liquid penetration was predicted than was measured experimentally.

The bubble size is limited in the calculation to the hydraulic diameter of the holes. With holes located so close together it is probable that bubbles coalesced and that larger bubble diameters are possible. The limit on bubble sizes was arbitrarily set to twice the hole diameter and the calculation was repeated. As can be seen in Figure 4, this change improved the data comparison in the bubble limiting region.

FIGURE 3. NWU CCFL Experiments, $DBUB_{MAX} = D_{HYD}$

FIGURE 4. NWU CCFL Experiments $DBUB_{MAX} = 2.D_{HYD}$

FUTURE PLANS

Developmental assessment calculations will continue with model improvements being made as comparisons with data indicate the need. The predictions of CCFL will be extended to include subcooled CCFL.

REFERENCES

1. Tien, C. L., et al. December 1979. Flooding in Two-Phase Countercurrent Flows. EPRI NP-1283, Electric Power Research Institute, Palo Alto, California.
2. Richter, H. J., et al. June 1979. "Effect of Scale on Two-Phase Countercurrent Flow Flooding." NUREG/CR-0312.*
3. Lovell, T. W. June 1977. The Effect of Scale on Two-Phase Countercurrent Flow Flooding in Vertical Tubes. Master's Thesis, Thayer School of Engineering, Dartmouth College.
4. Pushkina, O. L., and Y. L. Sorokin. September 1969. "Breakdown of Liquid Film Motion in Vertical Tubes." Heat Transfer, Soviet Research 1:5.
5. Dudder, A. E., et al. January 1980. Two Phase Interaction in Countercurrent Flow. Annual Report, University of Houston.
6. Hsieh, C. October 1979. Countercurrent Air/Water and Steam Water Flow Above a Perforated Plate. Master's Thesis, Northwestern University.
7. Dullman, J. C., et al. 1978. "Interpretation of Entrainment Measurements in Annular Gas-Liquid Flows." Seminar of the International Center for Heat and Mass Transfer, Dubrovnik, Yugoslavia, Hemisphere Pub., Vol. 2, September 4-9, 1978, pp. 681-693.
8. Tatterson, D. F., et al. January 1977. "Drop Sizes in Annular Gas-Liquid Flows." AIChE Journal 23:1.

*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Springfield, VA 22161.

LOCA SIMULATION IN NRU(a)

C. L. Mohr, Project Manager
J. P. Pilger, Deputy Project Manager
Design and Fabrication

SUMMARY

Fabrication of the major components of the thermal-hydraulic test train is well underway. This unit consists of a full-length 32-rod bundle--31 unpressurized, instrumented fuel rods and an instrument thimble that will be used to conduct a series of thermal-hydraulic tests designed to obtain data on a typical pressurized water reactor (PWR) fuel bundle during a large-break loss-of-coolant accident (LOCA). Modifications to the U-2 loop of the Chalk River (Canada) NRU reactor are progressing satisfactorily. The data acquisition system is in final checkout and should be ready for shipment in early July.

The Safety Analysis Report is being reviewed by the Canadian Reactor Safety Board. No problems are anticipated.

INTRODUCTION

The objective of the LOCA Simulation in NRU Program is to provide information on the heatup, reflood, and quench phases of a 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 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, using unpressurized fuel rods, will be devoted to thermal-hydraulic behavior. A total of 18 tests are planned for the first test train. The remaining five test trains will be used for only one test each. These test trains will have

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

pressurized fuel rods; and, as a result, the rods will deform and probably rupture during the test. These materials tests will evaluate the effects of ballooning and rupture on the quench-front velocities and associated heat-transfer coefficients.

The test loop in the NRU reactor 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 grid spacers have been fabricated and accepted by quality assurance (QA). Fuel rod components have been received and/or fabricated and accepted. The fuel pellets have been slotted and are ready for rod assembly. There are 12 types of fuel pellets ranging from solid pellets to pellets with three external thermocouple slots and a center hole for the centerline thermocouple. Approximately 12,000 pellets are required for the test train. Figures 1-5 depict the fuel assembly procedures.

The Safety Analysis Report of the NRU program was presented to the Canadian review board on reactor safety, which has the responsibility for approving proposed reactor tests in the NRU. Their review of the LOCA simulation project has identified several areas where additional information is required. As a result of this presentation a final safety review is scheduled for August.

A detailed test operations plan document is currently being prepared that describes in detail the actual test operation including control settings, sign-off procedures, and QA checks on techniques for recording testing information. Several iterations with Chalk River Nuclear Laboratories (CRNL) staff will be required to complete this document. It is planned that this document will not be completed until the current modifications to the U-2 loop by the CRNL staff are finished.

FIGURE 1. Fuel Powder is Compressed to 95% $\pm 1.5\%$ of Theoretical Density

FIGURE 2. Sintered Pellets are Ground on a Centerless Grinder to Final Dimensions

FIGURE 3. Slotted Pellets are Inserted Between the Thermocouples
in the Instrumented Rod

FIGURE 4. Lower Rod End Cap is Arc Welded in a Helium Atmosphere

FIGURE 5. Steam Probe Mounting Bracket is Attached to the Grid Spacer

FUTURE PLANS

The first test train will be assembled and all instrumentation will be checked. Internal fuel rod thermocouples will be accurately located using an eddy current probe developed by Pacific Northwest Laboratory (PNL). The DACS system will be shipped to CRNL for installation and checkout. The draft test plan and detailed test matrix will be circulated for comment. Loop modifications will continue. Target shipment date of the test train from PNL to CRNL is still mid-September 1980.

STEAM GENERATOR TUBE INTEGRITY(a)

R. A. Clark, Project Manager

V. F. FitzPatrick, Deputy Project Manager

J. M. Alzheimer

R. L. Burr

P. G. Doctor

G. H. Lyon

C. J. Morris

K. R. Wheeler

SUMMARY

Almost all efforts during the past quarter concerned preparation activities and transport of a retired-from-service nuclear steam generator from the Surry II nuclear plant (Surry, Virginia) to Hanford, Washington. Other research-related activities included preparation of specimens for a leak rate test matrix, a letter report on tube defect characterization in the tube sheet region, and continuing efforts at nondestructive characterization of tight stress corrosion cracks (SCCs) in steam generator tubing.

INTRODUCTION

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

Initial program tasks included producing a matrix of steam generator tube specimens with mechanically or chemically induced flaws simulating defects found in nuclear steam generator service. These flawed specimens were then fully nondestructively characterized by means of positive replication and

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

various nondestructive testing (NDT) techniques (mainly eddy current testing). The tube specimens were next tested to failure at PWR steam generator operating temperatures. The failure strength, actual flaw dimensions, and NDT-indicated flaw dimensions were utilized to derive mathematical relationships. These relationships were 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 program includes correlating the mathematical models developed in Phases I and II with actual service-flawed tubing. Lack of suitable specimens led to the redirection of Phase III into the Surry Generator Project (SGP), a subprogram of the SGTIP. The SGP is comprised of tasks to relocate a nuclear steam generator removed from service at Surry, Virginia, to Hanford, Washington. Once at Hanford the steam generator will be housed in a specially designed facility to allow nondestructive and destructive examination of the unit. Research will be conducted in the areas of NDT technique verification and instrument development; defect matrix identification; profiling of defect types, extent, and locations; identification of deposits and sources of corrosion; and health physics. The generator will also become a source of specimens with service-induced flaws for various NRC programs. Potential later research phases include simulated operation of a portion of the generator to assess long-term effects on generator serviceability of chemical cleaning or decontamination procedures that may be proposed to NRC for licensing. Study of the recovery and reuse of materials in decommissioned reactor components is also a potential research task.

TECHNICAL PROGRESS

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

LEAK RATE DETERMINATIONS FOR THROUGH-WALL DEFECTED INCONEL 600 STEAM GENERATOR TUBES

Fabrication of tube specimens with SCCs of various lengths was begun this quarter based on the matrix of specimens listed in the January-March 1980 quarterly. In addition, simulated tube sheet sections have been machined. A flowmeter has been ordered that is capable of measuring flows from 30 to 400 gpm at 650°F and pressures up to 2500 psig.

EDDY CURRENT INSPECTION AT THE TUBE SHEET

Steam generator tube sections of Inconel 600 with single machined defects were individually placed into a simulated carbon steel tube sheet. Uniform thinning simulated a large volume defect; elliptical wastage, a medium volume defect; and an electro discharge machined (EDM) notch, a small volume defect. Each defected tube was examined with single and multifrequency differential eddy current tests and a single-frequency absolute eddy current test. The defect was sequentially repositioned in the simulated tube sheet and reexamined. In this manner information on defect detectability was developed for three flaw volumes as a function of flaw location relative to the top of the tube sheet using commonly accepted eddy current techniques. Results of this research are being published next quarter as NUREG/CR-1626.

SURRY GENERATOR PROJECT

This subprogram of the SGTIP entails retrieving a nuclear steam generator removed from service at the Surry II plant at Surry, Virginia, and transporting it to Hanford, Washington. Once at Hanford the generator will be located in a specially designed temporary facility to allow nondestructive and destructive examination.

This subprogram originated out of Phase III of the SGTIP, which involved obtaining and testing service-defected tubing. These tests were to confirm models of remaining tube strength that were developed in Phases I and II on mechanically and chemically simulated tube defects. Because of the inability to obtain representative service-defected tubing, Pacific Northwest Laboratory (PNL) proposed research on a removed-from-service generator.

This past quarter Surry generator 2A (removed from service in February 1979) was successfully transported to Hanford as detailed below.

Preshipment Examination

The previous quarterly report presented details on the preshipment examination that determined that the generator was not so service degraded as to preclude successful transport. It was also established that interim storage at Surry had not resulted in damage that would compromise future research results. During the preshipment examination deposits from several secondary side locations were obtained, including scrapings of the oxide scale on the generator tubing, material pried out of the annulus between tubes and tube support plates, and sludge found on top of support plates. X-ray analysis using the Debye-Scherrer technique was conducted this quarter on samples from five separate regions. All of the samples yielded very similar results: A spinel-type oxide of the form $X^{+2}Y_2^{+3}O_4$ plus elemental copper existed in approximately the same ratio in all samples.

Temporary Storage

Temporary storage pads were cast last quarter; this quarter the roadway to the temporary storage area was widened and compacted to allow transport of the 220-ton generator on a rubber-tired trailer.

Interim Storage Facility

Provision has been made to provide a rigid greenhouse over the channel head end of the steam generator while in temporary storage. Design of this greenhouse (designated the Interim Storage Facility) has been completed. Construction will commence only if it is decided that early NDT data are desirable; it can be constructed within a month of the decision to do so.

Steam Generator Examination Facility (SGEF)

This quarter work was substantially completed on the Title II (definitive) design of the SGEF. Title II blueprints and cost estimates were issued for J. A. Jones and PNL review. Several design revisions were identified during the review process; and these comments have been returned to Facilities Systems Engineering Corporation (FSEC), the architect-engineer subcontractor, for inclusion in the final Title II report. The Title II design was approved by PNL after incorporation of the Design Review comments. J. A. Jones prepared the bid package to advertise for a construction contractor; it will be issued early next quarter. The SGEF is on schedule and within budget.

Transportation of the Steam Generator

The A generator from the Surry II nuclear plant was transported to Hanford via barge from Surry down the James River to the Atlantic Ocean, through the Carribean Sea and the Panama Canal, and up the west coast of North America to the mouth of the Columbia River. The barge then proceeded up the Columbia River to the Port of Benton, located just south of the Hanford reservation. Barge loading at Surry required 6 days; the voyage from April 20 to May 25, 34 days; and unloading at Hanford was accomplished by May 28, 1980. The entire shipment went extremely smooth per the Department of Transportation-approved Operational Plan.

After unloading at Hanford the generator was moved on a rubber-tired trailer to the interim storage pad. After it was positioned on the pad an argon gas purge was established to maintain a 1/2-psi positive pressure relative to the surrounding atmosphere. A 1/4-in. hole was drilled through the shell at the lowest point on the horizontally positioned generator to determine if the generator was dry after transit. Some water was present and the generator was completely drained. Analysis of the water showed it contained 50-ppm chloride and trace amounts of elements found in sea water. The water could be from condensation during the voyage, or it possibly was present in the transition cone at Surry before shipment. During the transport 12 bottles of argon ($\sim 2650 \text{ ft}^3$) were bled into the generator; thus, there was little potential for an oxidizing reaction. Figures 1 through 6 show various aspects of the transport operation.

Accelerometer

A three-axis accelerometer was attached to the generator during the barge transport. Acceleration information was logged in the towing tug boat with an FM radio transmission link and is currently being analyzed to determine maximum accelerations experienced by the generator during shipment.

Management Activities

A presentation was made at a Metal Components Subcommittee meeting of the Advisory Committee on Reactor Safeguards (ACRS). Proposed research activities on the Surry generator were outlined. Other management activities included issuance to DOE-Richland of safety analysis documents for operation of the SGEF. Work on the detailed Research Program Plan is also proceeding.

Milestones

The following milestones for the Surry Generator Project were met during the past quarter:

- Contract with Lampson established for off-loading the Surry generator at Hanford.
- Interim Storage Facility design and access roadways to temporary storage completed.
- Surry II generator A successfully shipped to Hanford, Washington.

Problems

Program funding for fiscal year (FY)-1980 will not allow letting of a construction contract for the SGEF, which will result in slippage of the construction schedule. If FY-1981 funds are received promptly in early October 1980, the 2-3 week delay should have no programmatic impact.

There is currently a construction union crafts strike at Hanford. If this strike continues beyond the date funding for the SGEF becomes available, the construction start date will be delayed until the strike is settled.

Program funding for FY-1980 is exhausted after the generator move (per the current fiscal plan). Further activity on most tasks will await fiscal FY-1981 funding.

FIGURE 1. The Generator as It Is Being Pulled from the Storage Vault at the Surry Nuclear Station

FIGURE 2. Loading the Generator on Land Transport at Surry

FIGURE 3. Land Transport to the Surry Nuclear Station Dock on the James River

FIGURE 4. Generator (from the transition cone end) on the Barge Just Prior to Shipping. The 10-in. thick concrete radiation shielding walls are shown.

FIGURE 5. Channel Head End of Generator as It Appeared Just Prior to Shipping. The anchoring stop and an antenna used to transmit data from a three-axis accelerometer on the generator are visible.

FIGURE 6. Generator Docking at Port of Benton (adjacent to the Hanford Project). The red tarpaulin was removed in the Panama Canal Zone because it became dislodged during the first leg of the trip.

FUTURE PLANS

Remaining tasks this fiscal year include completion of Title II (definitive) design of the SGEF. The SGEF bid package will be issued with construction bids opened on September 24, 1980. A draft copy of the research program plan will be issued for comment during the next three months. Work is also continuing on Phase II SGTIP tasks including NDT characterization of SCC in steam generator tubing and leak rate determinations associated with through-wall tubing defects.

RESIDENT ENGINEER AT CADARACHE, FRANCE(a)

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

SUMMARY

There was no progress to report during the current quarter.

INTRODUCTION

The objective of this program is to facilitate the exchange of information between the United States and France. To this end, a resident engineer is being supported at the PHEBUS program in France. This experimental program for in-pile studies is operated by the Service D'Essais de Surete (Safety Tests Service) of the French Atomic Energy Commission. The facility, located at Cadarache, consists of a closed test loop driven by a pool-type reactor that develops about 60 mW for a 20-minute test period. The test loop, which passes through the center of the core, contains an 80-cm test section and is capable of operating at pressurized water reactor (PWR) conditions. Current plans call for blowdown and reflood type testing on rod bundles containing either single-rod or 25-rod assemblies having geometries typical of those found in large PWRs.

(a) RSR Fin. Budget No.: 82278; RSR Contact: W. V. Johnson

DISTRIBUTION

No. of
Copies

No. of
Copies

OFFSITE

	A. A. Churn DOE Patent Division 9800 S. Cass Avenue Argonne, IL 60439	R. Foulds Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555
788	U.S. Nuclear Regulatory Commission Division of Technical Information and Document Control 7920 Norfolk Avenue Bethesda, MD 20014	H. H. Scott Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555
2	DOE Technical Information Center R. F. Abbey, Jr. Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555	R. Van Houton Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555
	S. Fabric Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555	M. A. Wolf Department of Atmospheric Sciences Oregon State University Corvallis, OR 97330
	D. A. Hoatson Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555	L. Agee Electric Power Research Institute P.O. Box 10412 Palo Alto, CA 94304
	W. V. Johnston Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555	B. R. Sehgal Electric Power Research Institute P.O. Box 10412 Palo Alto, CA 94304
10	J. Muscara Reactor Safety Research Division Nuclear Regulatory Commission Washington, DC 20555	F. Shakir Department of Metallurgy Association of American Railroads 3140 S. Federal Chicago, IL 60616

No. of
Copies

SM-ALC/MMET
Attn: Capt. John Rodgers
McClellan AFB, CA 95652

Dr. Sotirios, J. Vahaviolos
Western Electric, ERC
P.O. Box 900
Princeton, NJ 08540

Mr. Jerry Whittaker
Union Carbide Company
Oak Ridge National Laboratories
Y-12
Oak Ridge, TN 37830

Mr. L. J. Anderson, B2402
Dow Chemical Company
Texas Division
P.O. Drawer K
Freeport, TX 77541

Mr. M. C. Jon
Western Electric, ERC
P.O. Box 900
Princeton, NJ 08540

W. L. Pearl
Nuclear Water & Waste Technology
P.O. Box 6406
San Jose, CA 95150

FOREIGN

P. Caussin
Vincotte
1640 Rhode-Saint-Genese
Belgium

W. G. Cunliffe
Building 396
British Nuclear Fuels Ltd.
Springfields Works
Salwick, Preston
Lancs. PR40XJ
U.K.

No. of
Copies

ACE Sinclair
Research Division
Berkeley Nuclear Laboratories
Berkeley
Gloucestershire, GL 13 9 PB
U.K.

Don Birchon
Admiralty Materials Laboratory
Holton Heath Poole
Dorset, England
020-122-2711

ONSITE

50 Pacific Northwest Laboratory

J. M. Alzheimer
M. C. Bampton
F. L. Becker
T. D. Chikalla
R. A. Clark
E. L. Courtright
J. M. Cuta
J. F. Dawson
R. L. Dillon
S. K. Edler
C. E. Elderkin
J. E. Garnier
R. K. Hadlock
C. R. Hann (3)
A. J. Haverfield
T. W. Horst
P. H. Hutton (3)
J. M. Kelly
R. J. Kurtz
P. T. Landsiedel
D. D. Lanning
R. P. Marshall
C. L. Mohr
W. C. Morgan
C. J. Morris
R. D. Nelson
P. W. Nickola
L. T. Pedersen
G. J. Posakony
E. B. Schwenk (2)

No. of
Copies

J. R. Skorpik
A. M. Sutey
M. J. Thurgood
G. L. Tingey
D. S. Trent (3)
C. L. Wheeler
Technical Information (5)
Publishing Coordination Fe(2)

*U.S. GOVERNMENT PRINTING OFFICE: 1980-O-341-742/573

NRC FORM 335 (7-77)		U.S. NUCLEAR REGULATORY COMMISSION BIBLIOGRAPHIC DATA SHEET		1. REPORT NUMBER (Assigned by DDC) NUREG/CR-1454, Vol. 2 PNL-3380-2	
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Reactor Safety Research Programs Quarterly Report April - June 1980				2. (Leave blank)	
7. AUTHOR(S) S. K. Edler, Editor				5. DATE REPORT COMPLETED MONTH: October YEAR: 1980	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Pacific Northwest Laboratory Richland, WA 99352				DATE REPORT ISSUED MONTH: November YEAR: 1980	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555				10. PROJECT/TASK/WORK UNIT NO.	
13. TYPE OF REPORT				11. CONTRACT NO. FIN Nos. B2101, B2088, B2289, B2043, B2084, B2041, B2277, B2097, B2278	
15. SUPPLEMENTARY NOTES				14. (Leave blank)	
16. ABSTRACT (200 words or less) <p> This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 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 ESSDR Test Reactor Program, Ispra, Italy; blowdown and reflood tests in the test facility at Cadarache, France; 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. </p>					
17. KEY WORDS AND DOCUMENT ANALYSIS					
17a DESCRIPTORS					
17b. IDENTIFIERS/OPEN-ENDED TERMS					
18. AVAILABILITY STATEMENT Unlimited				19. SECURITY CLASS (This report) Unclassified	
20. SECURITY CLASS (This page) Unclassified				21. NO. OF PAGES 22. PRICE \$	