Reactor Safety Research Programs

**Quarterly Report
July - September 1981**

Prepared by S. K. Edler, Ed.

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

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ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from July 1 through September 30, 1981, for the Division of Accident Evaluation, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. 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	5
Integration of Nondestructive Examination Reliability and Fracture Mechanics	13
Experimental Support and Development of Single-Rod Fuel Codes:	
Task A - Irradiation Experiments	15
Experimental Support and Development of Single-Rod Fuel Codes:	
Task B - Data Qualification and Analysis	17
Experimental Support and Development of Single-Rod Fuel Codes:	
Task C - Code Coordination and Ex-Reactor Testing	23
Experimental Support and Development of Single-Rod Fuel Codes:	
Task D - Pellet-Cladding Interaction Experiments	27
Pipe-to-Pipe Impact	29
Severe Core Damage Test Subassembly Procurement Program -	
Power Burst Facility Severe Fuel Damage Test Project	33
Severe Core Damage Test Subassembly Procurement Program -	
ESSOR Fuel Damage Test Program Support	41
Severe Core Damage Test Subassembly Procurement Program -	
Nine-Rod Test Train OPTRAN 1-3 Project	49
Core Thermal Model Development	51
LOCA Simulation in NRU	61
Steam Generator Tube Integrity	67

FIGURES

Acoustic Emission/Flaw Relationship for In-Service Monitoring of Nuclear Pressure Vessels

1	Wave-Guide Acoustic Emission Sensor for Watts Bar Reactor	7
2	Watts Bar Crack Growth Specimen	8
3	Test Arrangement for Irradiated Fracture Specimen	9
4	Valid AE Versus Test Time for Irradiated Fracture Specimen 67-W23X	10
5	Valid AE Versus Test Time for Irradiated Fracture Specimen 65-W25	10
6	Valid AE Versus Test Time for Unirradiated Fracture Specimen 67-W22	11

Experimental Support and Development of Single-Rod Fuel Codes:

Task B - Data Qualification and Analysis

1	Comparison of Xenon-Filled Fuel Rod Data and Computer Code Calculations	20
2	Comparison of Data to ANS 5.4 Fission Gas Release Model	21

Pipe-to-Pipe Impact

1	Instrumentation for Pipe-to-Pipe Impact Tests	30
2	Plate Forming Strain Circle Application	31

Severe Core Damage Test Subassembly Procurement Program - Power Burst Facility

Severe Fuel Damage Test Project

1	PBF Severe Fuel Damage Test Train Assembly Sealed Shroud System with ZrO ₂ Fiberboard Insulation and Support Tubes	35
2	PBF Severe Fuel Damage Insulated Shroud Molten Metal Penetration Detector	36
3	Shroud Insulation Penetration Testing with Molten Zircaloy	37
4	Apparent Thermal Conductivity Versus Temperature for Composite Shroud Insulation Using ZrO ₂ Fiberboard and High-Density ZrO ₂ Tube Supports	38
5	Fallback Barrier General Configuration and Instrumentation	39

Severe Core Damage Test Subassembly Procurement Program - ESSOR Fuel Damage Test Program Support

1	Basic Concept of the Liquid Level and Void Fraction Detection Device	42
2	Condition 1: Liquid Level Above Sensing Level	43
3	Condition 2: Liquid Level Below Sensing Level	44
4	Transducer Output	45
5	Multiple Level Detector Using a Single Measuring Orifice	45
6	Liquid Level and Void Fraction Detection Using Multiple Measuring Orifices	46

Core Thermal Model Development

1 Comparison of Predicted and Measured Cladding Temperature History for NRU Test PTH-110 at 4-ft Elevation	53
2 Comparison of Predicted and Measured Cladding Temperature History for NRU Test PTH-110 at 6-ft Elevation	54
3 Comparison of Predicted and Measured Cladding Temperature History for NRU Test PTH-110 at 8-ft Elevation	55
4 Comparison of Prediction to Compartment Pressure History (Room 6) for Battelle-Frankfurt Experiment D1	56
5 Comparison of Prediction to Compartment Pressure History (Room 4) for Battelle-Frankfurt Experiment D1	57
6 Comparison of Prediction to Compartment Pressure History (Room 9) for Battelle-Frankfurt Experiment D1	58
7 Comparison of Prediction to Cladding Temperature History for PKL-K9	59

LOCA Simulation in NRU

1 Test Cruciform in the Rupture Zone at Station 79	63
2 Test Cruciform in the Rupture Zone at Station 80	64
3 Close-Up of Rod Rupture of Center Rod in Figure 1	65

Steam Generator Tube Integrity

1 HVAC System Being Installed in the Steam Generator Examination Facility	70
2 2000-gal Tank for Storage of Liquid Radioactive Wastes	71

TABLES

Experimental Support and Development of Single-Rod Fuel Codes:

Task A - Irradiation Experiments

1 Revised Postirradiation Examination of Instrumented Fuel Assembly (IFA)-527	16
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Experimental Support and Development of Single-Rod Fuel Codes:

Task B - Data Qualification and Analysis

1 Final Status for Test Assemblies	19
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GRAPHITE NONDESTRUCTIVE TESTING^(a)

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SUMMARY

Improvements in the eddy current system and refinements in the ZFIT algorithm have significantly increased the accuracy to which the near-surface oxidation profile can be estimated. The ultrasonic backscattering equipment has been modified to increase detection depth; the transducer design has been optimized; and an improved dry-coupling medium has been tested. A journal article on oxidation has been accepted for publication.

INTRODUCTION

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

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

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

TECHNICAL PROGRESS

EDDY CURRENT TESTING

During this quarter, it was verified that the improved circuitry and the new coil-winding procedure^(b) eliminate the errors caused by circuit resonance. The changes increased the system resonant frequency from 8 to 20 MHz; the new resonant frequency has a negligible effect on the measurements taken at the three frequencies (100 kHz, 1 MHz, and 4 MHz) used in the multifrequency eddy current test.

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

(b) Discussed in last quarterly report.⁽¹⁾

The computer algorithm ZFIT has been significantly refined. Although further work will be performed for verification, we believe that the algorithm can now accurately estimate the near-surface oxidation profile using multifrequency eddy current data.

The success with the ZFIT algorithm during this quarter was based on three factors:

- obtaining improved raw data after eliminating the coil resonance problem
- using sample materials of known conductivity to calibrate the test (previous work dealt primarily with insertion of known impedances in series with the search coil)
- developing additional fitting routines to compensate for remaining measurement error in the eddy current systems.

The algorithm can now accurately predict the conductivity and thickness of individual known test samples. The next phase of testing will involve fixing the thickness of the specimens and predicting the conductivity of two- and three-layered media. Various stacking combinations from the set of 0.050-in. thick oxidized PGX plates will be used for this work.

The algorithm compensates for remaining instrument measurement error by first fitting the a_{ij} coefficients for the following equations:

$$V_x = a_{11}\Delta R_{\text{eff}} + a_{12}\Delta X_{\text{eff}}$$

$$V_y = a_{21}\Delta R_{\text{eff}} + a_{22}\Delta X_{\text{eff}}$$

where V_x and V_y = the supposedly orthogonal voltages generated by the eddy current instrument
 ΔR_{eff} and ΔX_{eff} = the coil resistance and reactance changes as the coil is placed on the specimen.
These values include the effect of parallel damping resistance and stray capacitance.

When this fitting procedure is used with V_x and V_y values resulting from testing specimens of known conductivity, fairly accurate convergence of the ZFIT program occurs. A set of a_{ij} can be determined for which agreement exists between Dodd's theory and actual ΔR and ΔX of the coil itself, which results in the desired final capability—the ability to compute specimen conductivity as a function of V_x and V_y .

A set of coefficients a_{ij} is computed for each frequency. The procedure basically serves to compensate for nonlinearity and/or lack of orthogonality in the analog eddy current instruments. This procedure may prove to be highly beneficial for field measurements where the long distance between coil and instrument may result in additional measurement error. The key to successful use of this preliminary fitting routine appears to be testing samples of known conductivity for calibration purposes instead of switching known impedances in and out of the coil circuit.

ULTRASONIC TESTING

The primary emphasis during this quarter was on improvement of the ultrasonic backscattering method. Several important modifications were made to the laboratory system to increase the signal-to-noise ratio of the backscattered waveforms. Oxidation and large voids can be detected at greater depths in the graphite with this increased ratio.

Several types of piezoelectric transducers were analyzed to determine optimum transducer designs for oxidation testing. Results show that a transducer style manufactured by Panometrics features good performance characteristics for graphite testing. Their broadband "Video Pulse" unit cut for a frequency of 1 MHz was chosen for this application. Clean and highly damped waveforms were achieved that were free of the unwanted radial modes energy and the backing member reflections observed in other units.

A method for dry coupling ultrasound into graphite has been evaluated with positive results. This method uses a specialized mix of two-part RTV silicone rubber, and it is believed that this material can be optimized for use during in-service inspection of the core support blocks. A layer of the material is placed between the transducer and graphite, and pressure is applied to the transducer to place the coupling member in compression. Sonic energy is thus transmitted to and from the graphite without using a liquid coupling.

OXIDATION

An article⁽²⁾ on part of our findings has been accepted for publication in *Carbon*. The article discusses the effect of oxidant/reductant ratios on the catalytic activity of impurity elements, the potential for oxidizing an interior portion of a graphite component more rapidly than the exterior portion, and possible ramifications (for example, the need for in-depth oxidation profiling).

FUTURE WORK

EDDY CURRENT TESTING

- complete verification of the profiling algorithm
- develop a method for in-field calibration of the system
- develop and test a means for mating the algorithm with probe circuits of the type that will be used for field testing.

ULTRASONIC TESTING

- optimize near-field criteria
- assess performance of the system on graphite with known oxidation
- complete development of dry coupling methods
- develop a correlation between backscatter and averaged video ultrasonic waveforms, evaluate radio frequency averaging for detecting wormhole oxidation effects
- determine maximum degree of surface oxidation that may be penetrated satisfactorily by ultrasonic methods.

OXIDATION

- determine chemical state of impurities during oxidation.

REFERENCES

1. Edler, S. K., ed. 1981. *Reactor Safety Research Programs Quarterly Report - April-June 1981*. NUREG/CR-2127, Vol. 2, PNL-3810-2, Pacific Northwest Laboratory, Richland, Washington.
2. Morgan, W. C., and M. T. Thomas. June 1981. "The Inverse Oxidation Phenomenon." PNL-SA-9689, Pacific Northwest Laboratory, Richland, Washington.

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

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SUMMARY

Preparation for the ZB-1 vessel test in Germany is proceeding on schedule. Precracking of machined flaws in the A533B steel insert is nearly finished, and it is still expected that vessel fabrication will be completed in December 1981. The latest version of the test plan is being circulated for approvals.

The acoustic emission (AE) test system on the Tennessee Valley Authority (TVA) Watts Bar, Unit 1 reactor is being installed; and completion is expected by October 10, 1981. Special-purpose wave-guide sensors are being used, and solenoid-driven impactors are being installed to provide a remotely controlled system functional check. A crack growth specimen will be coupled to a location on the piping to input crack growth AE to test the ability of the system to detect AE in the presence of reactor hydraulic noise.

Acceptable operation of the AE monitor system was verified after the Dunegan/Endevco (D/E) detection location subsystem was returned to the factory for repairs.

The scope of the fiscal year (FY) 1981-1982 program has been expanded to include AE monitoring of piping as well as of the vessel. AE characterization of wrought and cast stainless and A106 steel piping material is planned.

Two irradiated and one unirradiated fracture specimens of A533B weldment were AE monitored at the Naval Research Laboratory (NRL). Results are being analyzed for correlation of AE with fracture mechanics variables.

INTRODUCTION

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

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

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

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

TECHNICAL PROGRESS

OFF-REACTOR VESSEL TEST

Fabrication of the A533B steel insert for the ZB-1 vessel test is nearly complete. Two of the three part-circular machined flaws have been fatigue precracked by cyclic pressurization of the machined notches. The last flaw should be precracked by about mid-October 1981 at which time the A533B insert will be ready for welding into the ZB-1 test vessel. Fabrication of the test vessel is progressing on schedule and is due for completion near the end of December 1981.

The test plan has been revised to reflect agreements reached in the June 2, 1981, meeting at Materialprüfungsanstalt (MPA), Stuttgart, West Germany. The revised test plan is being circulated for approvals.

REACTOR MONITORING

A test plan and drawings for AE system installation at TVA's Watts Bar, Unit 1 reactor have been approved; and seismic qualification determination for the AE system is in process. Installation of AE sensors on the No. 2 inlet nozzle, the 10-in. safety injection pipe adjacent to the No. 2 cold leg, and a section of the vessel wall began on September 24, 1981. Special-purpose wave-guide sensors developed for this purpose are being used; assembled and unassembled views of the sensor are shown in Figure 1. The key features of these sensors are:

- differential operation for high common-mode rejection of electrical noise
- integral sensor-preamp design encased in a stainless steel box
- two-part wave-guide assembly that allows the sensor-preamp unit to be removed or replaced without changing the wave-guide that is attached to the vessel or piping
- electronics components that meet a 250°F operating specification.

Two small solenoid-driven impactors are being installed to provide a reference impulse to check AE system operation.

Temporary signal leads will be strung to a central point from which preservice hydrostatic testing can be monitored. Permanent leads for reactor operational monitoring will be installed later.

A crack growth specimen is also being installed on the 10-in. safety injection pipe. The purpose of the specimen is to provide a known source of AE signals during preservice hydrotesting of the reactor. This will serve the function of evaluating the ability of the AE system to detect crack growth AE in the environment of reactor hydraulic background noise. A schematic of the specimen with attached hydraulic loading cylinder is shown in Figure 2. The specimen was fabricated from 2-in. thick AISI 4340 plate with the plane of the crack oriented in the L-T direction. After machining, the specimen was solution heat treated at 1480°F for 1.75 hr, oil quenched to 150°F, and then tempered at 600°F for 2 hr and air cooled. This heat treatment schedule was specified to produce a material with low fracture toughness and correspondingly maximize the AE. Following heat treatment, the specimen was fatigue precracked at a stress intensity factor range of 20 to 25 ksi $\sqrt{\text{in.}}$. The specimen mounting surface has been contoured to fit flush with the piping and improve coupling of the AE into the pipe.

AE MONITOR SYSTEM

The D/E 1032D system is currently functional after having been returned to the factory for repairs. A successful system checkout was accomplished by monitoring a 26-in. pipe stress corrosion cracking (SCC) test for several days.

Neg. 81G364-2

FIGURE 1. Wave-Guide Acoustic Emission Sensor for Watts Bar Reactor

A preliminary laboratory checkout of the whole AE instrumentation system (i.e., 1032D system, waveform acquisition system, and PDP 11/03 computer system) was performed. As a result of this work, it can be concluded that:

- The 1032D system acquired data properly.
- The waveform tagging scheme performed properly (i.e., the same number appeared on the 1032D data file as appeared on the waveform acquisition system). This is essential to subsequent combining of pattern recognition and source location data.
- Data can be transferred from the D/E 1032D to the PDP 11/03 system. However, data are currently transferred in FORTRAN from the 1032D to the 11/03; therefore, the 11/03 requires six floppy disks to store the data of one 1032D disk. The problem is being solved by editing the program used by D/E to transfer files.
- The pattern recognition software needs further checkout using real AE and noise data.

PIPE MATERIAL CHARACTERIZATION

An element was added to the scope of the FY 1981-1982 program calling for characterization of AE/flaw growth in reactor piping materials. The following progress was made in organizing this effort:

- Arrangements have been made to monitor a 4-in. diameter section of wrought stainless steel pipe during SCC. Two primary functions of this effort are to measure the AE profile as the SCC develops and to collect AE waveforms for pattern recognition analysis.

FIGURE 2. Watts Bar Crack Growth Specimen

- A section of cast stainless steel pipe (31 x 43 x 2-3/8 in.) has been obtained from another U.S. Nuclear Regulatory Commission (NRC) program at PNL. Fatigue crack growth specimens to be machined from this section have been designed. Signal attenuation measurements will be made on the full section of material prior to machining fatigue crack growth specimens.
- A source for immediate delivery of A106 pipe material has been located, and a sufficient quantity of this material has been ordered to fabricate approximately four fatigue crack growth specimens.

IRRADIATED FRACTURE SPECIMENS

Two irradiated and one unirradiated fracture specimens of A533B weldment were AE monitored at NRL this past quarter. The test arrangement is shown in Figure 3; and preliminary results are shown in Figures 4, 5, and 6. The two irradiated specimens show fairly similar AE responses, but the total number of signals is relatively low in both cases. Unfortunately, it was discovered after the unirradiated specimen test that the preamplifier gain was 20 dB lower than in the case of the irradiated specimens. Additional unirradiated specimen tests will be performed in about four months to facilitate obtaining AE data on the unirradiated material. Load, crack opening displacement, and crack length data for all specimens have been requested from NRL to permit correlation of the AE with fracture mechanics variables.

REPORTS

- Quarterly progress report for the period April 1 to June 30, 1981.

FIGURE 3. Test Arrangement for Irradiated Fracture Specimen

FIGURE 4. Valid AE Versus Test Time for Irradiated Fracture Specimen 67-W23X

FIGURE 5. Valid AE Versus Test Time for Irradiated Fracture Specimen 65-W25

FIGURE 6. Valid AE Versus Test Time for Unirradiated Fracture Specimen 67-W22

FUTURE WORK

Plans for the period from October 1 to December 31, 1981, include:

- prepare an analysis-before-test document for the remainder of the program
- monitor cold hydrostatic testing at Watts Bar, Unit 1
- start AE characterization tests on SCC in wrought stainless steel
- finalize preparations for the ZB-1 German vessel test
- publish evaluation report on AE weld monitoring demonstration
- summarize program results for the 9th Water Reactor Safety Research Information Meeting.

FIGURE 2. Valid AE Versus Test Time for Uniaxial Fatigue Specimen W33

FUTURE WORK

- Plans for the period from October 1 to December 31, 1987, include:
- prepare an analysis before test document for the remainder of the program
 - perform cold hydraulic testing at Water Bar Unit 1
 - start AE characterization test on 50C low weight stainless steel
 - finish preparation for the 28-1 Cermet vessel test
 - publish evaluation report on AE weld monitoring demonstration
 - summarize program results for the 5th Water Reactor Safety Research Information Meeting.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS(a)

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SUMMARY

No contribution will be submitted to this quarterly report. At the request of the U.S. Nuclear Regulatory Commission (NRC), an annual report is being compiled on the work conducted in this program in fiscal year (FY) 1981.

INTRODUCTION

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

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

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

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

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

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS(2)

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F. A. Simonen, Project Manager

S. R. Doctor
G. B. Dudder
B. G. Heister
C. P. Selby

SUMMARY

No conclusion will be submitted to this quarterly report. At the request of the U.S. Nuclear Regulatory Commission (NRC), an annual report is being compiled on the work conducted in this program in the year 1987.

INTRODUCTION

The primary mission of the Nuclear Power Reactor Inspection Program is to ensure that the inspection of nuclear power plants is performed in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section XI (Rules for Inspection of Nuclear Power Plant Components). The inspection techniques are normally used for the inspection of components which are periodically performed on a sampling of welds.

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

- determine the reliability of ultrasonic test performed on commercial light water reactor (LWR) primary systems using part of the mechanics analysis determine the impact of NDE uncertainty 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 rational analysis, and NDE uncertainties, formulate recommendations to ASME Code, Section XI, and Regulatory Requirements needed to assure suitably low failure probability.
- The scope of this program is limited to 12 of primary systems, and the results and recommendations are also applicable to Class II piping systems.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK A - IRRADIATION EXPERIMENTS(a)

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SUMMARY

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

Irradiation test IFA-431 is completed. Two other tests (IFA-513 and IFA-527) were removed from the reactor in early April 1981 due to fuel failures. IFA-432 was removed from the reactor in early June to permit cooling for late fall shipment to Harwell, U.K. It has been determined that IFA-513 shall remain in Norway pending eventual return to the United States with other NRC fuel. Puncture and gas analysis of the four intact rods may be carried out at Kjeller, Norway. Compliance tests on rods 7 and 8 of IFA-432 were successfully completed at Harwell. A proposal to continue the irradiation of some IFA-432 rods to high burnup is being considered by the NRC.

INTRODUCTION

The objectives of the Experimental Support and Development of Single-Rod Fuel Codes Program at Pacific Northwest Laboratory (PNL) are fourfold:

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

The Halden Boiling Water Reactor (HBWR) in Norway is currently the sole site used by this program for irradiation tests. Postirradiation examination (PIE) will be carried out at both Kjeller, Norway, and Harwell, U.K. Task A of the program is concerned with the conduct of the tests and coordination of test design, test fabrication, shipping, PIE, and sample disposal. The test matrix now spans the full range of expected BWR conditions for pelletized UO_2 fuel, including:

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50 to 380 μm (0.002 to 0.015 in.)

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

- gas compositions ranging from pure helium to pure xenon
- fuel densities of 95% and 92% theoretical density (TD), the latter both stable and unstable regarding in-reactor densification.

TECHNICAL PROGRESS

Puncture and gas analysis of the four intact fuel rods in IFA-513 (to check for the suspected presence of water) may be carried out at Kjeller, Norway. It has, therefore, been determined that IFA-513 shall remain in Norway pending eventual shipment to the United States with other NRC fuel. This decision resulted in a change in the PIE program for IFA-527 presented last quarter, which showed one IFA-513 rod substituting for rod 5 of IFA-527 in the PIE to be conducted at Harwell. The revised program for IFA-527 is shown in Table 1; rod 5 has been reinstated.

Table 1. Revised Postirradiation Examination of Instrumented Fuel Assembly (IFA)-527

Rod Number	Type of Examination				
	Visual	Gamma Scanning	Neutron Radiography	Rod Puncture and Pressurize	Optical Metallography
1	X		X	X	X
2	X			X	
3	X		X		
4	X		X		
5	X		X	X	X
6	X	X		X	
Total	6	1	4	4	2

Compliance tests conducted at AERE^(a) Harwell, U.K., on rod 7 of IFA-432 (unirradiated) and rod 8 (irradiated) have been completed. In these tests, axial load was applied to the top of the fuel column, and the resulting axial and radial deflections of the cladding were measured. These deflections were remarkably less for rod 8 than for rod 7.

A proposal to alter the irradiation/PIE schedule for IFA-432 is under consideration by the NRC. Under the proposed plan, two rods (1 and 6) would go to Harwell in October 1981 and the remaining four rods (1, 3, 5, and 9) would continue to operate and accumulate burnup in the HBWR for up to 2 yr until they reach a rod average burnup exceeding 40,000 MWd/MTM. This plan would yield fission gas release data at high burnup, and some of the resources saved could be spent in fiscal year (FY) 1982 for extended PIE on rods 1 and 6.

FUTURE WORK

Contractual arrangements between Halden and Harwell for shipping and PIE of IFA-432 and -527 will be completed, and the assigned rods will be shipped to Harwell. Optical metallography on one section of rod 8 will be performed, and a final report will be prepared as the final work in the compliance test contract. Both items are scheduled for completion by mid-December 1981.

(a) Atomic Energy Research Establishment.

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

The irradiation of instrumented fuel assemblies (IFAs) to obtain well-characterized data is a major objective of the Experimental Support and Development of Single-Rod Fuel Codes Program. Task B of this program is responsible for qualifying and analyzing those data. During this quarter, the final in-reactor data for IFA-432, -513, and -527 were received. A report discussing the in-reactor behavior and analysis of postirradiation examination (PIE) for rod 6 of IFA-431 (densifying fuel) was completed and forwarded to the U.S. Nuclear Regulatory Commission (NRC) for publication and distribution.

INTRODUCTION

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

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

TECHNICAL PROGRESS

This quarter's activities are discussed below by subtask.

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

SUBTASK B-1 - DATA PROCESSING

IFA-432 was removed from the reactor in June 1981 to permit cooling for shipment, and IFA-513 and -527 were removed from the reactor in April 1981 due to failed fuel rods. The final data tapes for these three assemblies were received this quarter, and the data were processed for use in analysis. Current rod burnups and status of the four test assemblies used in this program (IFA-431, -432, -513, and -527) are listed in Table 1.

The W/Re thermocouples used to measure fuel centerline temperatures are subject to alteration from neutron-induced transmutation and a subsequent decrease in the emf^(a) of the thermocouple for a given temperature difference. During this quarter, a correction factor of 1.75% per 10^{24} n/m² thermal neutron fluence (at the thermocouple) was applied to the temperature data for IFA-432, which resulted in an end-of-life (EOL) temperature correction of approximately 13.6%.^(b) For example, a measured temperature of 1525K was corrected to 1695K.

SUBTASK B-2 - DATA REPORTS

A report discussing the irradiation behavior and PIE results for rod 6 of IFA-431 was sent to NRC for printing and distribution. This report is titled "Observation of Porosity Reduction in a Densification-Prone Test Fuel Rod: Data and Analysis," NUREG/CR-2042.

Rod 6 contained fuel that was unstable with respect to densification and had an initial density of 92% TD. The rod exhibited temperatures higher than rod 1 (same dimensions but 95% TD stable fuel) but lower than rod 2 (95% TD stable fuel with dimensions equivalent to instantaneous densification of 92 to 96.5% TD). Thus, the assumption of instantaneous densification was found to result in conservative fuel temperatures relative to actual densification. Similarly, comparison of NRC fuel codes to the temperature data resulted in calculated temperatures that were generally higher than observed temperatures.

PIE of the fuel yielded a final density of 96.5% TD. The observed microstructural changes were consistent with the density increase; there was a marked depletion of submicrometer diameter pores and total pore volume.

SUBTASK B-3 - DATA ANALYSIS

Error and Uncertainty Analysis

Recent comparisons between fuel rod temperature data and FRAPCON-2 calculations have indicated relatively good agreement for helium-filled rods at beginning of life (BOL).⁽¹⁾ However, there are some questions regarding the adequacy of FRAPCON-2 for rods filled with xenon gas; the code underpredicted the measured temperatures at BOL.⁽²⁾ A reexamination of the data and computer code calculations has helped to explain the differences.

First, there was an apparent difference in fuel temperatures between xenon-filled rod 4 of IFA-431/432 and the xenon-filled rods of IFA-527. This difference is particularly apparent after several power ascensions. Comparing temperature versus power for the first power ascension (Figure 1) shows that the initial temperatures were fairly close. The lower temperature for rod 4 of IFA-431 (relative to IFA-527) may be partially attributed to the oversize fuel pellets that were used in rod 4 to constrain the fuel pellets containing the thermocouple.

(a) Electromotive force.

(b) This correction is in broad agreement with Halden's own current estimates. The whole subject is under current experiment, research, and review.

Table 1. Final Status for Test Assemblies

Rod	Dates of Operation	Rod			Final Status	
		Average Burnup, MWd/MTU	UTC(a)	LTC	PT	CE
IFA-431						
1	06/08/1975 - 02/13/1976	4,316	S	S	S	S
2	06/08/1975 - 02/13/1976	4,461	S	S	-	S
3	06/08/1975 - 02/13/1976	4,621	S	S	-	S
4	06/08/1975 - 02/13/1976	4,479	F	S	-	S
5	06/08/1975 - 02/13/1976	4,334	S	S	S	S
6	06/08/1975 - 02/13/1976	4,242	S	S	S	S
IFA-432						
1	12/08/1975 - 06/07/1981	29,202	F	S	S	F
2	12/08/1975 - 06/07/1981	29,344	F	S	-	S
3	12/08/1975 - 06/07/1981	29,460	F	S	-	F
4	12/08/1975 - 02/13/1976	2,493	F	F	-	S
5	12/08/1975 - 06/07/1981	30,399	F	F	S	F
6	12/08/1975 - 06/07/1981	30,517	F	F	S	F
7	not irradiated	—	—	—	—	—
8	06/09/1976 - 01/06/1980	22,099	—	—	—	S
9	03/26/1980 - 06/07/1981	5,856	—	—	—	S
IFA-513						
1	11/28/1978 - 04/08/1981	10,656	S	S	S	S
2	11/28/1978 - 04/08/1981	10,496	S	F	S	S
3	11/28/1978 - 04/08/1981	10,207	S	F	S	S
4	11/28/1978 - 04/08/1981	10,078	S	S	S	F
5	11/28/1978 - 05/24/1980	7,567	S	S	S	S
6	11/28/1978 - 04/08/1981	10,527	S	S	S	S
IFA-527						
1	07/01/1980 - 04/08/1981	1,158	F	F	F	F
2	07/01/1980 - 04/08/1981	1,080	S	F	S	S
3	07/01/1980 - 04/08/1981	991	S	S	S	S
4	07/01/1980 - 04/08/1981	973	S	S	S	S
5	07/01/1980 - 04/08/1981	1,045	S	F	S	S
6	07/01/1980 - 04/08/1981	1,174	S	S	F	S

rod failed
rod failed
rod failed
rod failed
rod failed

(a) UTC = upper thermocouple.

LTC = lower thermocouple.

PT = pressure transducer.

CE = cladding elongation.

S = survived.

F = failed.

rod failed
rod failed

rod failed
rod failed
rod failed
rod failed
rod failed

FIGURE 1. Comparison of Xenon-Filled Fuel Rod Data and Computer Code Calculations

A comparison of first-ramp and low-burnup temperatures for rod 4 of IFA-431 may also be seen in Figure 1. The decrease in temperature is attributed to fuel relocation, which increased the fuel-cladding gap conductance. In contrast, IFA-527 temperatures did not decrease as significantly from BOL. This has been attributed to the low powers reached by IFA-527 (15 kW/m compared to 35 kW/m for IFA-431) and thus a reduced driving force for fuel cracking and relocation.

A comparison between a FRAPCON-2 calculation using the FRACAS-II mechanical option⁽³⁾ and measured data may be seen in Figure 1. Data from the high-powered xenon-filled fuel rod and the code calculation are in fairly good agreement at 3600 MWd/MTM. It is therefore expected that the code will give reasonable results in situations where fission gas release severely contaminates the initial fill gas since this can only happen after sufficient burnup, swelling, and relocation have occurred to bring the gap size in line with the results in a xenon mixture.

Fission Gas Release in IFA-432

Fuel rods 1, 5, and 6 in IFA-432 were instrumented with pressure transducers to monitor the internal pressure changes that occurred during irradiation. The pressure data (obtained up to 30-GWd/MTU rod average burnup) have been used to estimate fission gas release as a function of burnup, and estimated releases from rod 1 were compared to calculations based on the ANS 5.4 fission gas release model.

Fission gas release was estimated from the pressure data by using the same assumptions given in Reference 4 except that fuel swelling and helium release from the fuel were included in the calculations. (a) The estimated release fractions for all three rods increased with burnup to a maximum value and then decreased with further irradiation. The maximum value occurred between 10 and 20 GWd/MTU and was directly related to the temperature history of the individual fuel rod. At 30 GWd/MTU, the estimated fission gas release was less than 15% in all rods with rod 6 (which operated at higher temperatures) having the largest release.

Comparing the estimated gas release from rod 1 to calculations based on the ANS 5.4 fission gas release model indicated that "burnup-enhanced" fission gas release did not occur. Experimental estimates compare favorably with the release fractions calculated when the model was used without the burnup enhancement factor on the diffusion coefficient (see Figure 2). With the burnup enhancement factor included, the model greatly overestimated the experimental data even when the volume-average temperature of the fuel was lowered by decreasing the thermal conductivity of the fuel 25% (the outer limit of its uncertainty).

FIGURE 2. Comparison of Data to ANS 5.4 Fission Gas Release Model (with and without burnup enhancement)

(a) Linear extrapolations with burnup were made from measured data of rod 8 of IFA-432 (22-GWd/MTU burnup).

FUTURE WORK

Data processing for the next quarter will continue as required. Work will begin on data reports for IFA-432 and IFA-527 that will present the data obtained until their removal from the reactor. Analysis of fuel cracking and relocation, fission gas release, and related subjects will continue.

REFERENCES

1. Berna, G. A., D. D. Lanning, and W. N. Rausch. June 1981. *FRAPCON-2 Developmental Assessment*. NUREG/CR-1949, U.S. Nuclear Regulatory Commission, Washington, D.C.
2. Edler, S. K., ed. 1981. *Reactor Safety Research Programs Quarterly Report - April-June 1981*. NUREG/CR-2127, Vol. 2, PNL-3810-2, Pacific Northwest Laboratory, Richland, Washington.
3. Berna, G. A., et al. December 1980. *FRAPCON-2: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods*. NUREG/CR-1845, Idaho National Engineering Laboratory, Idaho Falls, Idaho.
4. Bradley, E. R., et al. 1979. *An Evaluation of the In-Pile Pressure Data from Instrumented Fuel Assemblies IFA-431 and IFA-432*. NUREG/CR-1139, PNL-3206, Pacific Northwest Laboratory, Richland, Washington.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

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

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

M. E. Cunningham
W. N. Rausch

SUMMARY

The FRAPCON-2 developmental assessment document was published and distributed. The FAST-GRASS gas release code from Argonne was incorporated into FRAPCON-2. Corrections to Version 1 Mod 2 (V1M2) of FRAPCON-2 were issued in a "Users' Letter."

The mechanical compliance tests on unirradiated and irradiated instrumented fuel assembly (IFA)-432 rods were completed at Harwell, and data analysis has begun. A three-dimensional (3-D) cracked fuel pellet-cladding interaction (PCI) model has been programmed, and initial calculations have been performed.

INTRODUCTION

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

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

Task C efforts include the following subtasks: code maintenance, fuel mechanics experiments, and PCI model development.

TECHNICAL PROGRESS

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

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

SUBTASK C-1 - FRAPCON-2 CONTROL AND MAINTENANCE

The FRAPCON-2 developmental assessment report (NUREG/CR-1949) was published and distributed. A new version of FASTGRASS was received from Argonne National Laboratory and has been modified to run as a subroutine of FRAPCON-2.

Some errors were found in the V1M2 edition of FRAPCON-2: Stored energy calculations for annular pellets, FRACAS-II cladding creep calculations, FRACAS-II internal gas pressure calculations, and fuel swelling model results were found to be in error. The corrections for these errors were announced in a "Users' Letter" that was issued in August 1981.

SUBTASK C-2 - FUEL MECHANICS EXPERIMENTS

Harwell completed the mechanical compliance tests on the unirradiated rods and the irradiated rod 8 of IFA-432. This experiment was similar to companion experiments performed by PNL, and both simulate the loading systems that occur in irradiation environments. The fuel column is loaded axially, which places the cladding in axial tension; and cladding axial and diametral deformations are measured simultaneously.

A preliminary data analysis showed that the unirradiated rod 7 performed in a manner similar to the prototype steatite pellet rods tested at PNL and Harwell. However, the cladding radial and axial deformations of irradiated rod 8 were substantially less than for the unirradiated rods. A time-dependent axial compression of the irradiated fuel was also observed. The relaxation rates appear to correspond with in-reactor data, but further analysis is required to verify this.

SUBTASK C-3 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

Work continued on the 3-D cracked fuel PCI model. Equations for the fuel and cladding deformations were written, and solutions were obtained for the thermal deformations of two fundamental fuel fragment shapes. Variations of these shapes provide asymmetric fuel geometries needed for the 3-D model. A shell model in the cladding that accounts for the effects of axial stresses on the bamboo ridge heights was developed.

The highly coupled and nonlinear equations were combined into a computer code (FRAGMT). Equations for cracked fuel interfragment shear strengths were formulated and included to model fuel relocation, conditioning, and deconditioning as a function of fragment shape and size. Fuel creep equations were included, and the first calculations showed that fuel creep was dependent on temperatures and fragment geometry.

Improvements were made in the axial compatibility equations at the cracked pellet interfaces. Further calculations produced reasonable ridge heights that showed the effects of axial forces at high rod powers. Cladding elongation predictions were qualitatively correct but of too great a magnitude for the initial pellet-cladding gap used.

FUTURE WORK

The following activities are planned for the next quarter:

- complete the verification of the new FASTGRASS/FRAPCON-2 combination
- issue the Version 1 Mod 3 (V1M3) edition of FRAPCON-2 that includes the changes listed in the "Users' Letter" and new FASTGRASS link and modifications to the PELET/RADIAL mechanical option

- begin work on an IBM-compatible version of FRAPCON-2
- analyze the compliance test data
- complete improvements in the axial compatibility equations of the 3-D PCI model, and perform calculations to quantify the burnup dependence of fuel relocation/conditioning, ridge heights, and cladding elongation.

- begin work on an IBM-compatible version of FRAPCON-2
- analyze the compliance test data
- complete improvements in the axial compatibility editions of the DFCI model, and perform calculations to quantify the pump response of fuel relocation/conditioning, ridge height, and cladding elongation

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS(a)

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

D. E. Fitzsimmons
R. K. Marshall

SUMMARY

The new cladding deformation measurement instrument with simultaneous ridge/bow measurement capability was fabricated and assembled and is now undergoing proof tests and calibration. The pressure vessel design for the loop tests with irradiated cladding was completed, approved, and sent out for bids. A bid for providing irradiated unfueled cladding samples was received and is under negotiation to minimize costs. Initial calculations were performed to optimize simulated fuel rod design so that failure thresholds may be bracketed in the experiments.

INTRODUCTION

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

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

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

TECHNICAL PROGRESS

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

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

SUBTASK D-1 - ROD STRAIN INSTRUMENT DESIGN

The new cladding deformation instrument was fabricated and assembled. The new instrument is capable of simultaneously measuring two orthogonal cladding diameters and the fuel rod bow. Initial testing showed that an adjustment of relative stiffness between the strain gage cantilever systems was required. Adjustments were made, and the instrument is now undergoing proof testing and calibration.

SUBTASK D-2 - LOOP EXPERIMENTS

The pressure vessel design was completed, design review was conducted, the drawings were approved, and the package was sent out for bid to fabricate. Fabrication of other necessary components was begun onsite.

A supplier for irradiated cladding samples was located, and drawings and specifications were forwarded. A proposal to supply the irradiated cladding was received, and PNL requested that the supplier resubmit an abbreviated proposal to minimize costs.

Radiation Work Procedures for handling irradiated cladding onsite were reviewed; and the necessary filters, shields, and holding tanks were identified and included in cost estimates for FY 1982.

Calculations with the sample design code written for this task revealed that the pellet-cladding gap must be minimized to insure that the experiments reach the failure regime for cladding stresses. However, the localizing effect of bamboo ridges was not accounted for in the calculations. Estimates for fuel fabrication were also received.

FUTURE WORK

The following activities are planned for next quarter:

- The subcontract for pressure vessel fabrication will be awarded, and the vessel will be fabricated and delivered to PNL for inspection and assembly.
- The subcontract for the irradiated cladding will be awarded.
- The cladding deformation instrument will be installed in the pressure vessel for final proof tests and calibrations.
- Sample design calculations will be refined to include the bamboo ridge model from Subtask C-3.
- Fuel fabrication for the irradiated cladding tests will begin.

PIPE-TO-PIPE IMPACT(a)

M.C.C. Bampton, Project Manager

J. M. Alzheimer

F. A. Simonen

SUMMARY

During the past quarter the pneumatic-powered pipe accelerator was fabricated and installed. The test facility was readied for testing, and test specimens were prepared.

INTRODUCTION

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

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

TECHNICAL PROGRESS

TEST FACILITY

During the past quarter the test facility was readied for the first series of tests. The pneumatic-powered pipe accelerator was fabricated offsite and has been delivered and anchored in place at the test site. The pipe support for the first tests was fabricated and installed. The large pneumatic cylinder used to power the system arrived just as the quarter ended and has been installed on the accelerator. The cylinder has been shown to function properly. All major components of the system are in place and ready for pretest checkout.

INSTRUMENTATION

All of the instrumentation for the initial tests is on hand and has been or will be installed shortly. Figure 1 shows the locations of the various transducers to be used. All transducers will be installed before the first test except for the target pipe pressure transducer, the support load cells, and the pipe pivot pin load transducer. Since the initial target pipes will not be pressurized, the pressure transducer is not needed. Because the decision to include the load transducers was made at a fairly late date, there was not enough time to install them.

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

ALL MEASUREMENTS WILL BE DYNAMICALLY RECORDED EXCEPT FOR STRAIN CIRCLES

FIGURE 1. Instrumentation for Pipe-to-Pipe Impact Tests

Several of the measurements will be used to compare experimental test results with analytic predictions. These include strain data obtained from strain circles and strain gages, accelerometer data, support loads, and the final deformed shape. All data, with the exception of the strain circles, will be recorded on a 14-channel data tape recorder. An example of strain circle usage for the shaping of a thick steel plate is shown in Figure 2.

SPECIMENS

All of the pipe needed for the test matrix has been received, and all specimens are of A106-B. The material properties of the steel are being characterized; static tensile properties and chemical composition will be checked and impact toughness will be characterized. Specimens for the first several tests have been prepared.

ANALYSIS

An independent review of the accelerator design to assess its structural integrity has been completed. This review provides the needed reassurance that the mechanism will not be overstressed during operation.

The ABAQUS-ND computer code has been used to model the initial test of the matrix. A rather simple model was used with only a limited number of beam elements. Results of the experiment will be compared to the ABAQUS-ND predictions to determine where model refinements are needed.

Neg. 81H271-1

FIGURE 2. Plate Forming Strain Circle Application

IN-CORE INSTRUMENTATION LINE TESTS

Four impact tests were performed on pipes similar to those used for in-core instrumentation line piping. The impacted pipes had outside diameters (ODs) of 1.05 in. and wall thicknesses of 0.154 in. The material was seamless 304 stainless steel. Two of the test specimens were rigidly supported behind the impact point while two were simply supported at 9 in. on either side of the impact point for a total span of 18 in. No pipe failures were observed. The test data are currently being analyzed to determine the amount of energy absorbed.

FUTURE WORK

The following activities are planned for next quarter:

- complete the first series of tests
- compare test results with analytic predictions
- start the development of failure models
- build pipe break hazards barrier
- prepare for pressurized tests.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

POWER BURST FACILITY SEVERE FUEL DAMAGE TEST PROJECT(a)

E. L. Courtright, Program Manager
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T. M. Fish	L. J. Parchen
L. L. King	J. D. Rising
R. F. Klein	R. D. Tokarz
R. K. Marshall	J. O. Vining
M. A. McKinnon	C. L. Wheeler

SUMMARY

Recent testing of the insulated shroud has addressed the question of how well the insulator structure would resist attack by molten Zircaloy in the event that a large portion of the cladding should melt. Results indicate that at temperatures around 2000°C the ZrO₂ fiberboard will dissolve in the Zircaloy to an extent that depends on the amount of Zircaloy present. Specimens are now available that have been in contact with molten Zircaloy at estimated temperatures greater than 2500°C for 1, 3, and 6 min. Additional experiments are in progress to test the proposed shroud molten metal penetration device for the Power Burst Facility (PBF) severe fuel damage (SFD) test train assemblies.

A formal design review of the closure head, flow tube liner, bundle repair, and hanger region of the PBF severe fuel damage test train assembly was held with EG&G Idaho, Inc. Minor changes were made to some components of the test train assembly hardware and incorporated into the design drawings to facilitate the post-test handling of the assembly in the PBF canal and hot cells. An internal design review of the insulated shroud assembly and details was also held during this reporting period. Design changes were made to the upper shroud flange hardline seals for instrument hardlines that are routed from the insulated region of the PBF shroud. The seals are no longer of the grayfoil design but utilize a Zircaloy-to-stainless transition piece similar to that used on the instrumented fuel rods for routing instrument hardlines through fuel rod end caps.

Measured thermal conductivity results for two full-section samples of 0.96 g/cc ZrO₂ fiberboard containing high-density ZrO₂ tubing to be used for the shroud insulation material indicated that the material will meet the low thermal conductivity requirements necessary for the shroud insulation material. Thermal-hydraulic calculations performed at both Pacific Northwest Laboratory (PNL) and EG&G Idaho, Inc., have shown that the shroud insulation material is acceptable from a test feasibility standpoint if the thermal conductivity value is from 0.14 to 1.69 W/m-K.

Pressure drop testing of the proposed fallback barrier configuration for the PBF severe fuel damage test train assemblies has been completed. Various steam pickup and removal line termination locations were evaluated in water fallback tests completed with a preheated fallback barrier bed (ZrO₂

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

grinding cylinders) at temperatures to 593°C. Results of the testing show that countercurrent steam flow should be used in conjunction with the heated ZrO_2 bed to prevent water breakthrough and cooling of the fallback barrier.

INTRODUCTION

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

TASK 1 - PBF SEVERE FUEL DAMAGE PROGRAM SUPPORT

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

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

A small-scale boil-off experiment will be conducted to investigate the sensitivity of water level control to the absolute water level during simulated small-break conditions. The experiment will also examine the effects of water level on the axial temperature profile of the electrically heated rod bundle.

Electrical heater rods with approximately 0.046 m of heated length and internal cladding thermocouples will be installed in a 9-rod square array inside a specially designed and fabricated high-pressure test section. The test section will be instrumented to measure inlet and outlet water/steam conditions as well as water level. Water at approximately 295K will be injected into the bottom of the test section at rates sufficient to maintain water levels between 10.2 to 22.9 cm (4 to 9 in.) above the beginning of the heated length. Nominal operating pressure will be about 650 psig, and both water injection rate and heater rod power will be varied over a matrix of test conditions.

TECHNICAL PROGRESS

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

SHROUD DEVELOPMENT TESTING

Several tests in the PBF severe fuel damage test program will involve rapid fuel bundle heating to temperatures of about 2300K. If the heating is rapid enough, much of the cladding will liquify at its melting point (2133K) or at the $Zr-UO_2$ eutectic (~1700K) instead of oxidizing. This molten metal may become a hazard to test operations depending on the quantity present. A molten metal penetration detector will detect any penetration of the shroud so that corrective action can be taken. Schematic view of the overall cross section of the insulated shroud is shown in Figure 1.

The molten metal penetration detector (Figure 2) is a fairly simple device that is designed to be built from stainless steel-sheathed thermocouples (TCs). The operating principle is that molten Zircaloy at high temperature will produce a short circuit between the outer sheath and either of the inner element wires and signal the need for corrective action. Redundancy is provided by using four independent sheathed TCs in parallel including four independent monitoring circuits (see Figure 2). If one of the four sheathed TCs should become inoperative prior to or during the early stages of testing, the adjacent sheathed TC will provide coverage within about 1 mm of the inoperative hardline.

FIGURE 1. PBF Severe Fuel Damage Test Train Assembly Sealed Shroud System with ZrO_2 Fiberboard Insulation and Support Tubes

Recent testing of the insulated shroud has addressed the question of how well the insulator structure would resist attack by molten Zircaloy in the event that a large portion of the cladding should melt. Testing results indicate that at temperatures around 2000°C , the ZrO_2 fiberboard insulator will dissolve in the Zircaloy to an extent depending on the amount of Zircaloy present. (The ZrO_2 fiberboard is not resistant to attack by molten Zircaloy). The oxide is dissolved fairly readily in the metal, and a large amount would be dissolved before the Zircaloy would solidify at 2300K . From the Zr-O phase diagram,⁽¹⁾ it can be seen that the first oxide in equilibrium with the melt at 200°C would be obtained at an average composition containing about 43 at.% oxygen. As the two-phase region is reached, the metal should cease to dissolve oxide. To contain 43 at.% oxygen, each gram of Zr would dissolve approximately 0.7 g of ZrO_2 because the Zr in the ZrO_2 must be considered. On a volume basis, 1 cc of metal would dissolve about 6.7 cc of the low-density ZrO_2 fiberboard in reaching 43 at.% oxygen. Theoretically the molten metal could consume a large quantity of the ZrO_2 fiberboard insulation at 2000°C .

Tests were conducted using an arc melting furnace to melt a slug of Zircaloy above a full-thickness section of shroud insulation that included the high-density ZrO_2 support tubes. Figure 3 shows the testing arrangement. The insulation was placed in a depression in a water-cooled copper hearth of a nonconsumable electrode arc melting furnace, and a pool of molten Zircaloy was maintained above the insulation. The Zircaloy temperature was estimated to approach 3000°C .

After only ~ 1 min about 80% of the insulation was dissolved at an applied power of 10 to 20 kW or about 0.93 to 1.71 kW/cm^2 to the test sample. Two additional samples were tested that had been in contact with molten Zircaloy at estimated temperatures greater than 2500°C for 3 and 6 min. The 6-min specimen is essentially completely penetrated. However, the energy involved in the arc melting tests ($\sim 1.6 \text{ kW/cm}^2$) is well above the 0.12 kW/cm^2 power calculated to result from nuclear heating

ELECTRICAL CIRCUIT USED TO MONITOR LMPD

A = MEASURES CURRENT TO VERIFY INTACT CONDUCTORS

R = MEASURES SHEATH-TO-CONDUCTOR RESISTANCE

FIGURE 2. PBF Severe Fuel Damage Insulated Shroud Molten Metal Penetration Detector

plus the Zircaloy metal-water reaction for the actual in-reactor tests. Additional shroud penetration tests will be run at lower powers, which are more prototypic of the actual calculated powers planned for the PBF severe fuel damage tests.

A prototypic specimen of composite shroud insulation was submitted to the Dynatech Research and Development Company for measurement of apparent thermal conductivity. The composite shroud insulation specimen was tested over a temperature range from 40 to 1000°C with repeat measurements at 800 and 1000°C after achieving the highest temperature measurement. The specimen was circular in cross section with a diameter of approximately 200 mm, a thickness of 13.2 mm, and a density of 88.7 kg/m³.

Results of the apparent thermal conductivity measurements for the composite shroud are presented in Figure 4. The measured thermal conductivity ranged from 0.294 W/m-K at 40°C to 0.588 W/m-K at 1000°C. Figure 4 shows an additional plot of estimated thermal conductivity versus temperature that was calculated using the "rule-of-mixtures." The calculated thermal conductivity plot used actual measured thermal conductivity data for the ZrO₂ fiberboard insulation and literature data for the high-density (100% of TD) ZrO₂ tube supports. The ZrO₂ tube supports used in the calculations were spaced the same as for the actual composite shroud insulation sample that was sent to Dynatech for apparent thermal conductivity measurements.

FIGURE 3. Shroud Insulation Penetration Testing with Molten Zircaloy

The results of the thermal conductivity measurements on the composite shroud insulation indicate that the material will meet the low thermal conductivity requirements necessary for the shroud insulation material. Thermal-hydraulic calculations performed at both PNL and EG&G Idaho, Inc., have shown that the shroud insulation material is acceptable from a test feasibility standpoint if the thermal conductivity value is in the range of 0.14 to 1.69 W/m-K.

WATER FALLBACK BARRIER AND BOIL-OFF EXPERIMENTS

Pressure drop testing of the proposed fallback barrier design for the PBF severe fuel damage test train assemblies has been completed for preconditioning water flow rates to 1.02 kg/sec at a system pressure and temperature of 8.3 MPa and 277°C, respectively. In addition, various steam pickup and removal line termination locations were evaluated in water fallback tests conducted with a preheated fallback barrier bed (ZrO₂ grinding cylinders) at temperatures to 593°C. Two sizes of ceramic ZrO₂ grinding cylinders were also evaluated in the testing.

A pressurized hot water and steam supply system was used to conduct the fallback barrier tests. The general configuration of the water fallback barrier bed including TC locations is presented in Figure 5.

Preliminary analysis of the data indicated that the pressure drop through the fallback barrier is approximately 7583 Pa/cm of bed length for the 0.95-cm diameter ZrO₂ grinding cylinders and approximately 5515 Pa for the 1.3-cm diameter ZrO₂ grinding cylinders for a water flow rate of 1.02 kg/sec at a temperature of 277°C. The cross-sectional flow area of the barrier bed was 66.9 cm². The measured pressure drop for either type of grinding cylinder presents no problems in the present design configuration for overall pressure drop for the test train assemblies.

FIGURE 4. Apparent Thermal Conductivity Versus Temperature for Composite Shroud Insulation Using ZrO₂ Fiberboard and High-Density ZrO₂ Tube Supports

Examination of the data from the water fallback barrier testing indicates that the effectiveness of a packed bed of ZrO₂ grinding cylinders as a water fallback barrier depends on the ZrO₂ cylinder size and the bed temperature relative to the water saturation temperature. For barrier bed temperatures typical of PBF operating conditions, water falling on top of the barrier bed will film boil much like water falling on a hot stove. From the test results, the water fell on the hot ZrO₂ spheres and appeared to break into small droplets; and the small droplets then fell completely through the bed. It is thought that the small water droplets are blanketed by a vapor film that prevents significant heat transfer to the droplets from the hot barrier bed. For either the 0.95- or 1.3-cm diameter ZrO₂ cylinders heated to approximately 538°C with water fallback flow rates as low as 1.9 kg/hr and no countercurrent steam flow, water will completely pass through approximately 50.8 cm of barrier bed in slightly less than 6 sec.

Based on results of the fallback barrier testing it was concluded that the barrier bed must do two things to be successful in preventing water fallback through the bundle. First, it must break the water into small droplets; and second, it must retain the droplets within the bed material long enough for vaporization or provide a means to remove them from the barrier bed. When sufficient countercurrent steam flow is maintained with a heated fallback barrier bed, both of the above requirements are

FIGURE 5. Fallback Barrier General Configuration and Instrumentation

met. Steam flow rates of 60 lb/hr and greater through the barrier bed successfully prevented water breakthrough for water flow rates in excess of 0.71 kg/sec.

Based on test results obtained from the fallback barrier testing, the water fallback barrier assembly for the SFD test train assemblies will use either 0.95- or 1.3-cm diameter ZrO₂ cylinders with a bed height of approximately 33 cm. In addition, the steam pickup and removal line will be centered in the bed; and the termination point of the collection end of the steam pickup and removal line will be located approximately one-third the overall length of the ZrO₂ bed from the bottom end of the fallback barrier assembly.

FUTURE WORK

Fabrication and assembly of component parts for two complete test train assemblies for the SFD-ST and SFD-1 tests for the PBF severe fuel damage program will continue. Boil-off testing with heater rods in the boil-off test apparatus will be completed. A final design review of the completed test train assembly for the tests will be held with EG&G Idaho, Inc. Supporting safety-related analysis and documentation required for the PBF SFD-ST test will be outlined and prepared.

REFERENCES

1. Lustman, B., and F. Kenze, Jr. 1965. *The Metallurgy of Zirconium*. McGraw Hill, New York, p. 355.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

ESSOR FUEL DAMAGE TEST PROGRAM SUPPORT(a)

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S. Crowell
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SUMMARY

A unique concept for liquid level and void fraction detection is described. The detector can help control and analyze severe fuel damage (SFD) tests.

INTRODUCTION

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

TECHNICAL PROGRESS

LIQUID LEVEL AND VOID FRACTION DETECTION SYSTEM

The unique concept that has been developed by PNL is composed of state-of-the-art components that can provide a real-time readout of the liquid level in a fuel assembly. The basic detection system is defined below, the two configurations that were tested are described, and the results of prototype testing are presented.

Definition of Detection System

The basic concept consists of an orifice (I) located in the test assembly, a connection tube or pipe through the closure head to a condenser section and then to a second orifice, followed by a flow control valve (see Figure 1). The second orifice includes appropriate pressure taps to enable the use of a differential pressure transducer to measure the mass flow rate.

To explain the functional characteristics of this device, it is assumed that there is no void fraction generated in the assembly and that the assembly pressure is constant at 7 MPa. With these assumptions in

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

FIGURE 1. Basic Concept of the Liquid Level and Void Fraction Detection Device

mind, only two conditions can exist for the sensing device. For calculational purposes, a 1.4-MPa pressure drop from the pressure vessel to the flow control valve is arbitrarily chosen. Condition 1—when the vessel is full of liquid or at least the sensing orifice (I) is in liquid—is depicted in Figure 2. Both orifices have been arbitrarily chosen to be 1 mm in diameter.

With these assumptions, the conditions that exist in the sensing system can be calculated. The 1.4-MPa pressure drop is shared equally by the two orifices while the condenser is cooling the liquid. The differential pressure transducer is seeing a nominal 0.7-MPa pressure drop, and the mass flow rate through the sensor is nominally 345 kg/hr.

For Condition 2 (see Figure 3), all conditions are the same as for Condition 1 except that the liquid level inside the pressure vessel is below the sensing orifice; therefore, the sensing orifice sees only water vapor. The condenser cools and condenses the vapor to liquid, and the second orifice still sees liquid at a fixed temperature. Because the condenser reduces the volume of the vapor, the pressure drop of the sensing orifice (I) increases to a nominal 1.35 MPa while the measuring orifice drops to 0.05 MPa due to the much reduced mass flow rate.

This basic device then provides a definitive real-time output that shows the presence or absence of liquid at the sensing location. A characteristic curve can be defined that assumes that no void fraction exists inside the pressure vessel. The differential pressure transducer output versus increasing fluid temperature is plotted in Figure 4. The first portion of the curve from 25°C to T_1 is solely a linear function of the change in the density of the liquid as it heats.

T_1 is defined by the pressure drop from the vessel to the flow control valve and the sizes of the orifices. It is the temperature and pressure at which flashing begins in the first orifice. As flashing occurs and is subsequently condensed downstream, the pressure drop in the system becomes a more complex function of liquid and vapor densities and becomes nonlinear from T_1 to $T_{\text{saturation}}$. It should be noted that if the liquid level drops below the sensing orifice at any point along this curve, the mass flow rate and the pressure drop across the measuring orifice will drop to the nominal 0.05-MPa value, indicating the drop in liquid level.

CONDITION I

ORIFICE I & II EQUAL (1 mm)

MASS FLOW RATE NOMINALLY 344 kg/hr

PRESSURE DROP ORIFICE I = 0.7 MPa

ORIFICE II = 0.7 MPa

(WATER FLOW IN BOTH DRIFICES)

FIGURE 2. Condition 1: Liquid Level Above Sensing Level

If the temperature of the liquid goes to $T_{\text{saturation}}$ and the presence of a void fraction is ignored or suppressed, then point P_1 in Figure 4 is a definitive value that represents the condition of the liquid at the sensing level in the vessel. Without the existence of a void fraction, the sensor has only two conditions— P_1 and P_2 . P_2 provides a distinctive output indicating that the liquid level is below the sensing elevation. From P_2 , the steam becomes superheated as the temperature increases and the transducer output becomes a linear function primarily related to the density change of the steam as its temperature increases.

Design Configurations

Liquid Level Detection

The previous description of the basic function of the detector referenced a sensor with one detector orifice and one measuring orifice. To better measure liquid level, two improved configurations can be used. One of these configurations (see Figure 5) consists of multiple detector orifices and one measuring orifice with a single differential pressure transducer. The second configuration (see Figure 6) consists of both multiple detector and measuring orifices, each having a differential pressure transducer.

In Figure 5 the individual orifices (labeled I, II, III, IV, and V) are connected to a single line through the closure head and ultimately to the measuring orifice and differential pressure transducer. The output for this device is shown in Figure 5. Each level change will cause an appropriate drop in the

CONDITION II

ORIFICE I & II EQUAL (1 mm)

MASS FLOW RATE NOMINALLY LESS THAN 45 kg/hr

PRESSURE DROP ORIFICE I = 1.35 MPa

ORIFICE II = 0.05 MPa

SATURATED STEAM FLOW IN ORIFICE I

WATER FLOW IN ORIFICE II

FIGURE 3. Condition 2: Liquid Level Below Sensing Level

total mass flow rate in the line and the associated incremental drop in the transducer output. Similarly, the multiple measuring device shown in Figure 6 will indicate the appropriate liquid level.

Liquid Level and Void Fraction Detection

If a void fraction exists, the output of the multiple detector/single measurement device (Figure 5) will give a somewhat ambiguous reading.

If the orifices at levels II, III, and IV are in a void fraction, level V is in liquid and level I is in vapor. The single transducer will see a mass flow rate that corresponds to the sum of the liquid from level V plus the sum of the liquid and vapor collected by levels IV, III, and II and the vapor collected by level I. The transducer output for this condition would probably settle between levels II and III, and the output would be noisy due to condensing vapor. However, there are many possible void fraction conditions that may occur, and a single reading cannot define the exact conditions. This detection system can be useful if certain conditions are expected to exist in the system, which may be possible in the PBF and ESSOR tests.

To completely define the void fraction, a multiple detector/multiple measuring device must be used. Figure 6 depicts five separate sensing systems, each with its own condenser, orifice, and differential pressure transducer. With this configuration, each level orifice has three distinct possible conditions. Each orifice is either in a liquid, a vapor, or a void fraction.

FIGURE 4. Transducer Output

FIGURE 5. Multiple Level Detector Using a Single Measuring Orifice

FIGURE 6. Liquid Level and Void Fraction Detection Using Multiple Measuring Orifices

ESSOR Liquid Level Detection

In the SFD tests scheduled for ESSOR, the desired liquid level and the likely area of void fraction can be reasonably defined for each test. A very real constraint of these tests is bringing tubes out of the closure head. In addition, the space used for tubes in the closure head limits the available space for instrument leads from the test assembly.

These constraints favor the single tube/multiple orifice device. If this device were to be used, the configuration shown in Figure 5 is recommended: a 6-mm diameter line through the closure head, passing through the bypass flow to the bottom of the fuel assembly, then into the fuel assembly area. Five orifices are equally spaced along the tube from zero elevation to 1 ft. The 1/4-in. line is then terminated in a 2-ft length of alumina or zirconia tube with a fixed and appropriately sized internal diameter. In this configuration, the first orifice is at the top of the active core and there are enough orifices in the liquid level area to provide sufficient information on the movement of the level as well as added information to recognize and anticipate the effect of a change in control settings.

Prototype Testing

This concept was initially tested on a package steam generator with a sight glass. The pressure chamber held about 3.8 l of water, and the steam generator had 3 kW of heat input. The heaters were activated by a settable pressure switch. The sensing system was made from 6-mm stainless steel tubing with 0.5-mm orifices. Rather than using a differential pressure transducer, the downstream side of the measuring orifice was exhausted to atmosphere.

The single sensing orifice was placed in the pressure vessel through a pipe fitting in the top of the steam generator, and the steam generator pressure was set to 0.3 MPa. The condenser consisted of a coil of 6-mm stainless steel tubing immersed in an ice bath. Tests were conducted with the sensing system as the only bleed on the steam generator, followed by tests while bleeding liquid from the blow-down valve, and then by bleeding steam from the pressure relief valve. These three types of tests should have resulted in a liquid level being generated without a void fraction, with a minimum void

fraction, and with a high void fraction. In each case, the sensing system accurately defined the liquid level. In the second and third cases, the system also indicated the existence of the void fraction prior to liquid level. The void fraction was indicated by a reduction in the mass flow rate seen by the measuring orifice and by the increase of noise on the flow signal. However, even in the high void fraction case, the difference in the mass flow rate in the void fraction and the liquid level was clear.

The problem with these tests arises from two factors: One cannot observe the void fraction, and constant pressure of the steam generator system cannot be maintained during the tests. Because of these problems, a void fraction generator that operates with Freon was fabricated.

In addition to laboratory testing, two liquid level sensors are being installed in the boil-off tests being conducted at PNL in support of the SFD program. The test system is being inserted in the 189D high-pressure water loop. The test assembly consists of nine electrically heated rods in a vertical section, and the tests will be performed at 7 MPa and 285°C water. The water will be boiled off slowly using the heated rod assembly; steam temperatures are expected to reach 650°C during the test. Each liquid level device consists of a single 3-mm diameter stainless steel tube with three orifices of 0.5-mm diameter; the orifices are positioned 15 cm apart on the rods.

FUTURE WORK

The development work presently under way at PNL on the liquid level device may be sufficient to allow its use in the SFD tests. It may also be capable of discrete measurement of void fraction.

to action, and with a high void fraction in each case, the sensing system accurately defined the liquid level in the second and third cases. The system also indicated the existence of the void fraction prior to liquid level. The void fraction was indicated by a reduction in the mass flow rate of the measuring orifice and by the increase of noise on the flow signal. However, even in the high void fraction case, the difference in the mass flow rate in the void fraction and the liquid level was clear.

The problem with these tests arises from two factors. One cannot observe the void fraction and constant pressure of the steam generation system cannot be maintained during the test. Because of these problems, a void fraction generator that operates with fixed flow was fabricated.

In addition to laboratory testing, two liquid level sensors are being installed in the boiler off-sets being conducted at PNL in support of the SFD program. The test system is being installed in the 1600 high-pressure water loop. The test assembly consists of nine electrically heated rods in a vertical section and the test will be performed at 7 MPa and 327°C water. The water will be boiled off slowly using the heated test assembly. Steam temperature was expected to reach 520°C during the test. Each liquid level device consists of a single 3-mm diameter stainless steel tube with three orifices and 2-mm diameter. The orifices are positioned 15 cm apart on the rods.

FUTURE WORK

The development work presently under way at PNL on the liquid level device may be sufficient to allow its use in the SFD test. It may also be capable of discrete measurement of void fraction.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

NINE-ROD TEST TRAIN OPTRAN 1-3 PROJECT(a)

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SUMMARY

The final design report is being prepared, and the final quality assurance certificate was prepared. The project was terminated at the end of September 1981.

INTRODUCTION

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

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

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

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

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM NINE-ROD TEST TRAIN OF TRAIN 1-3 PROJECT(a)

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R. E. Schreiber, Project Manager

R. E. Falowski
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SUMMARY

The final design report is being prepared and the final quality assurance certificate was prepared. The project was terminated at the end of September 1987.

INTRODUCTION

The purpose of the Pacific Northwest Laboratory (PNL) project was to provide instrumented test train hardware for Power Burst Facility (PBF) experimental program. A nine-rod test train was chosen for the standard operating transient simulation (OPTSIN 1-3).

In support of this experiment, a test train was designed with a replaceable core region and shroud so that after loss of a single rod, it could be performed with the basic assembly. The design allowed for insertion and removal of one rod following the precondensing phase and had built-in features to facilitate assembly and repair.

The OPTSIN 1-3 experiment involved a series of transients in the PBF. Since the instrumented fuel bundle is composed of 18 rods, it is necessary to have a test train being water cooled (WTR) service. It was necessary to develop procedures and equipment for removal, assembly and the test train. The assembly work was done by EG&G Idaho, Inc., at the Idaho National Engineering Laboratory (INEL).

(a) PNL Budget No. 82034-1 NRC Contract R. Van Houten

CORE THERMAL MODEL DEVELOPMENT(a)

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SUMMARY

Developmental assessment of the COBRA/TRAC computer code continued during the past quarter. Reasonable predictions of various experimental data were made, including reflood and SEMISCALE experiments. The noncondensable gas field was incorporated into COBRA-TF. This new version of the code has been compared with the Committee on the Safety of Nuclear Installation (CSNI) Numerical Bench Mark Problem and has been used to predict some of the Battelle-Frankfurt D-series containment experiments. A simulation of a double-ended cold leg break loss-of-coolant accident (LOCA) in an upper head injection (UHI)-equipped pressurized water reactor (PWR) was completed up until 35.0 sec into the transient; the remainder of the transient will be completed next quarter.

INTRODUCTION

The COBRA-TF computer code is being developed for the U.S. Nuclear Regulatory Commission (NRC). The purpose of this work is to provide better digital computer codes for computing the behavior of full-scale reactor systems under postulated accident conditions. The resulting codes are being used to perform pre- and post-test analysis of light water reactor (LWR) components and system effects experiments. This Pacific Northwest Laboratory (PNL) project has three 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 UHI-equipped PWRs
- to apply COBRA-TF with a noncondensable gas field to containment analysis.

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

(a) FIN Budget No.: B2041; NRC Contact: S. Fabric.

(b) FLECHT = full-length emergency cooling heat transfer.

(c) FEBA = flooding experiment in blocked arrays.

regions. These geometries cannot be modeled easily in regular Cartesian or cylindrical mesh coordinates; however, since they have significant impact on the thermal-hydraulic response of the system, these geometries must be modeled with reasonable accuracy.

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

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

A noncondensable gas field was incorporated into the code during the past quarter by adding another continuity equation for the mixture of noncondensable gases. Additional mass diffusion equations were solved to obtain the volume fraction for each component in the gas mixture, which may consist of any number of gases. Eight different gases have been included in the current coding.

TECHNICAL PROGRESS

NRU LOSS-OF-COOLANT ACCIDENT SIMULATION

This LOCA test series is being run in the National Research Universal (NRU) reactor, Chalk River, Ontario, Canada. The tests are being conducted by PNL for the NRC and will evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length nuclear rod bundle during the heat-up, reflood, and quench phases of a LOCA. The results from the experiments are also being used to assess the ability of the COBRA/TRAC computer code to predict the temperature and fluid conditions during only the reflood and quench phases.

Description of Experiment

The NRU test section consisted of thirty-two 12-ft long nuclear fuel rods with a 0.374-in. outside diameter (OD) surrounded by a stainless steel shroud. The entire test train was inserted into a 25-ft long pressure tube inside the NRU reactor. The experimental test conditions were obtained in four steps: 1) a steady-state phase, 2) a heatup phase, 3) a reflood phase, and 4) a quench or scram phase. Temperature, power input, and mass inlet flow measurements were recorded during the entire transient.

COBRA/TRAC Model

Because of the physical dimension changes, the entire NRU test section was modeled using one channel and 27 vertical nodes. Two 18-in. and eleven 1-ft nodes were used to model the heated length, while one 6-in. and thirteen 1-ft nodes were used to model the area above the test section. The 12 test rods and 20 guard rods were modeled by two rods; and the hanger, pressure tube, and shroud were modeled by three heat slabs. The region above the test section was modeled because of the possibility of water fallback. A variable flow rate and enthalpy was specified at the bottom of the test section, and two TRAC components were used to specify a 40-psia boundary condition at the top of the section.

Discussion of Results

Figures 1 through 3 compare the temperature results from the COBRA/TRAC simulation to experimental data for prototypic thermal-hydraulic test 110 (PTH110). Results have been presented for the

FIGURE 1. Comparison of Predicted and Measured Cladding Temperature History for NRU Test PTH-110 at 4-ft Elevation

4-, 6-, and 8-ft elevations for the simulation and the 4.03-, 6.36-, and 8.11-ft elevations for the experimental data. The predictions were generally accurate until the peak temperature was reached; the results then had a tendency to drift below those measured for the lower elevation and above those measured for the higher elevations. These discrepancies have been attributed to the liquid carryover model.

BATTELLE-FRANKFURT D-SERIES CONTAINMENT TEST

COBRA-TF with the noncondensable gas field was used to simulate test D1 of the Battelle-Frankfurt D-series containment blowdown experiments⁽¹⁾ to assess the code's ability to predict the pressurization rate of the various rooms in a containment during blowdown.

FIGURE 2. Comparison of Predicted and Measured Cladding Temperature History for NRU Test PTH-110 at 6-ft Elevation

Description of Experiment

The Battelle-Frankfurt tests involve blowdown experiments performed with a scale-model containment. The geometric scale for the interior subcompartments and vent areas is 1:64 and is based on a 1200-MW PWR (Biblis A). This geometric scaling allows a 1:1 scaling on the pressure histories so that test results can be extrapolated to a full-size containment. The model containment is constructed with nine inner subcompartments that are interconnected by 57 vent ducts with orifice plates of various sizes and types. Thus, the system may be configured in a variety of geometries. The D-series tests are characterized by the single-phase blowdown of approximately 69.6-bar saturated steam. The data from test D1 were chosen for comparison with the COBRA-TF prediction. Test D1 consisted of three rooms connected in series: rooms 6, 4, and 9.

FIGURE 3. Comparison of Predicted and Measured Cladding Temperature History for NRU Test PTH-110 at 8-ft Elevation

COBRA-TF Model

The three rooms in test D1 were modeled with a 1-D mesh for the hydrodynamics. The heat transfer to the wall was also modeled in one dimension. All walls and surface areas were assumed to be made of 0.25-m thick concrete since the exact information on structures within each room was not readily available. A heat transfer coefficient of $2000 \text{ W/m}^2\text{-K}$ was specified between the structure and the containment atmosphere. The steam blowdown source was modeled by specifying a mass flow rate into the first mesh cell of the first room.

Discussion of Results

The predicted pressures for each of the three rooms are shown in Figures 4 through 6. The comparisons are fairly good. The pressure in the first room (room 6) is slightly overpredicted, which is probably a result of not having all of the heat transfer surfaces within the room correctly specified (some steel surfaces were present).

PKL-K9 REFLOOD TEST

Experiments simulating the refill and reflood procedure after a LOCA in a 1300-MW PWR primary coolant system were performed at the KWU test facility in Erlangen, Germany. COBRA/TRAC was used to model experiment K9 of the PKL series (completed December 1979). PKL-K9 was characterized by a double-ended guillotine break in the cold leg with emergency core cooling (ECC) water injected into the cold legs.

FIGURE 4. Comparison of Prediction to Compartment Pressure History (Room 6) for Battelle-Frankfurt Experiment D1

FIGURE 5. Comparison of Prediction to Compartment Pressure History (Room 4) for Battelle-Frankfurt Experiment D1

Description of Test

The PKL test facility was designed to simulate a 1300-MW PWR. The main components were a vessel, an external pipe downcomer, and three primary coolant loops (one with double capacity). Each coolant loop contained a U-tube steam generator and pump resistance in the cold leg.

COBRA/TRAC Model

The vessel and downcomer were modeled with the COBRA-TF vessel module. The hot and cold leg pipes and the steam generators were evaluated with TRAC 1-D components. The three coolant loops of PKL-K9 were modeled in COBRA-TRAC with two loops, which provided equivalent volumes and hydraulic diameters.

The initial rod temperatures as well as the break pressure and ECC injection histories were applied as boundary conditions. The initial system flow was zero, and the pressure was set to 4.6 bars.

FIGURE 6. Comparison of Prediction to Compartment Pressure History (Room 9) for Battelle-Frankfurt Experiment D1

Discussion of Results

The cladding temperature profile at 6.4 ft is shown in Figure 7. The peak cladding temperature at approximately 25 sec is accurately predicted as is the general shape of the curve. Quench is predicted approximately 25 sec early, but the reported quench front range at this level is from 130 to 210. At elevations below 6.4 ft, temperatures and quench fronts are in good agreement. Above the 6.4-ft level, the peak temperature and initial curves are in good agreement; but the quench fronts are not well predicted due to an error in the falling film quench front logic.

Collapsed water level measurements showed fairly good agreement with the PKL reported data, but after 150 sec it appeared that the level was low. Break pipe pressures also matched the reported data well.

FIGURE 7. Comparison of Prediction to Cladding Temperature History for PKL-K9

FUTURE WORK

A COBRA-TF model of the SEMISCALE steam generators has been set up. SEMISCALE test UT-2 will be rerun with this new steam generator model during the next quarter. The UHI simulation will also be completed. Work will continue on the development of the hot-bundle version of COBRA-TF and the application of COBRA-TF to containment analysis.

REFERENCES

1. Bolstad, J. W., et al. 1980. *Comparison of Compare Mod-1 Subcompartment Calculations with Battelle-Frankfurt D-Series Test Results*. NUREG/CR-1817, Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

LOCA SIMULATION IN NRU(a)

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SUMMARY

The second materials test (MT-2) was conducted at Chalk River Nuclear Laboratories (CRNL) in late July 1981. A series of pretransient reflood tests were conducted to characterize the reflood loop response to variable reflood rate commands (not in original design criteria) that will be required for the joint U.S. Nuclear Regulatory Commission (NRC)-United Kingdom cladding balloon tests in late September 1981. The MT-2 test train was the first to reuse a previously irradiated instrumented shroud/guard rod assembly, thereby saving approximately \$1 million in fuel, instrumentation, and hardware costs. The MT-1 test cruciform was removed from the split shroud/guard rod assembly under 5 ft of water in the fuel bay. The MT-2 test cruciform was installed, and the instrument lead seal was successfully reconstituted at poolside. The peak cladding temperature obtained during the tests was 1650°F with a reflood decay time of 35 sec and variable reflood rates between 0.5 to 5.5 in./sec.

Post-test examination of the test cruciform on the disassembly, examination, reassembly machine (DERM) revealed that 7 of the 11 pressurized rods had ruptured. The DERM was upgraded to perform bundle profilometry as well as single-rod profilometry. Photographic, video, and profilometry data were obtained for the test cruciform. Examination of the shroud/guard rod assembly was completed, and the decision was made to use this assembly for the NRC-U.K. thermal-hydraulic test series (TH-2) scheduled for September 1981, thereby considerably reducing the costs for this test series.

INTRODUCTION

The objective of the Pacific Northwest Laboratory (PNL) LOCA Simulation in NRU Program is to provide information on the heatup, reflood, and quench phases of a loss-of-coolant accident (LOCA). The tests are designed to give information on the quench-front velocities within a fuel bundle, the liquid entrainment [10 CFR 50, App. K (Sec. ID 2)], and the heat transfer coefficients [10 CFR 50, App. K (Sec. ID 5)] for full-length pressurized water reactor (PWR) fuel as a function of reflood rate and delay time before reflood starts. A total of six test trains will be prepared for the program. The first test train will be devoted to thermal-hydraulic behavior; more than 25 tests using this assembly were conducted in the National Research Universal (NRU) reactor at Chalk River, Ontario, Canada. The test loop in the NRU will accommodate a full-length (12-ft long), 32-rod bundle on a 6 x 6 array with the corner pins removed. The bundle design uses commercial enrichments, cladding materials and dimensions, and grid spacers.

Since future materials tests will have 12 pressurized fuel rods in the test cruciform, the rods will deform and rupture during testing. These materials tests will evaluate the effects of ballooning and rupture on reflood flow restrictors, quench front velocities, and associated heat transfer coefficients.

(a) FIN Budget No.: B2403; NRC Contact: R. Van Houten.

TECHNICAL PROGRESS

The DERM was successfully used under 5 ft of water to remove the MT-1 test cruciform and install the MT-2 test cruciform in the previously irradiated shroud/guard rod assembly. The DERM software for bundle profilometry was validated, and the MT-2 test bundle was characterized. Photographic and video data were obtained. Single-rod profilometry was performed on all test rods. The shroud and guard rods were examined in detail and deemed suitable for use in the NRC-U.K. thermal-hydraulics test series, thereby considerably reducing test costs.

The joint NRC-U.K. test parameters for TH-2 were defined. The DACS software was modified to incorporate the variable reflood rates required to hold peak cladding temperatures at $\sim 1450^{\circ}\text{F}$ for 120 sec or longer to allow for maximum ballooning.

The TH-2 test cruciform (12 unpressurized fuel rods) was assembled and shipped to CRNL for installation in the MT-1 shroud/guard rod assembly.

A prototype of the time domain reflectometry (TDR) liquid level sensor was successfully tested in the laboratory with demineralized water. If the special coaxial cable connector can be obtained in time, this unit will be used in MT-3.

The following photographs show some of the ruptured MT-2 test rods.

FUTURE WORK

The joint NRC-U.K. thermal-hydraulic test series will be conducted to establish loop performance and control parameters required to attain the MT-3 objectives of holding peak cladding temperatures at $1450 \pm 50^{\circ}\text{F}$ for more than 150 sec.

Neg. B1H640-1

FIGURE 1. Test Cruciform in the Rupture Zone at Station 79

Neg. 81H640-2

FIGURE 2. Test Cruciform in the Rupture Zone at Station 80

Neg. 81H640-4

FIGURE 3. Close-Up of Rod Rupture of Center Rod in Figure 1

FIGURE 3. Close-up of Red Rump of Green King in Figure 1.

STEAM GENERATOR TUBE INTEGRITY(a)

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SUMMARY

The Steam Generator Examination Facility (SGEF) is 92% complete; construction remains ahead of schedule. Efforts to establish a multiparticipant Steam Generator Group Project continued with technical marketing presentations and proposals to several foreign and domestic concerns. Programmatic tasks that precede research on the Surry generator continued on schedule. Design, procurement, and assembly of a data acquisition system has been completed; a comprehensive health physics program has established necessary personnel monitoring and radiation training procedures and obtained required instrumentation. In addition, generator research tasks continued with planning, scheduling, and costing on reopening preshipment penetrations, channel head decontamination, baseline eddy current in-service inspection (ISI), steam generator tube unplugging, a nondestructive testing (NDT) round robin, and secondary side access. Research also continued on stress corrosion crack (SCC) characterization and leak rate determinations.

INTRODUCTION

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

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

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

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

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

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

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

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

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

Because of the potentially unique opportunities presented by the availability of the removed-from-service Surry generator, a broadened research program has been developed that should be of interest and need to government agencies, private organizations, and vendors. At NRC's request, PNL is seeking to establish interest from other parties to join in program participation. Potential exists for research and development in chemical cleaning, decontamination, corrosion product identification, corrosion mechanism studies, and materials recovery under alternate sponsors.

TECHNICAL PROGRESS

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

SURRY GENERATOR PROJECT

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

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

Steam Generator Examination Facility (Task 1)

At the end of this quarter construction of the SGEF was 92% complete and slightly ahead of schedule; the facility is scheduled for completion by December 7, 1981. Building subsystem [HVAC,^(a) cranes, etc.] acceptance test procedures will begin in mid-October. Current efforts include completion of wiring, interior painting, and remaining finishing work.

During this quarter the HVAC system was completed, change room facilities were completed, hoods and casework were installed in the laboratory, the breathing air compressor was installed, and most interior painting was done. A request has been issued for an Operational Readiness Review, which must be satisfactorily completed prior to building startup; and startup of the SGEF must be done before the service-degraded steam generator can be placed in the facility. As part of the Operational Readiness Review process, several documents have been completed: an Operational Readiness Plan, a building emergency plan, a fire plan, acceptance test procedures for building subsystems, and a radiation work training plan. Building construction contingency funds have been used to improve fire-proofing over insulating material installed in the SGEF tower and to run several underground conduits from the SGEF to the 314 building. The conduits will allow transmittal of experimental data and building alarm signals to the 314 building where the project computer system is located and project personnel have offices. Figure 1 shows work in progress on the HVAC system, and a 2000-gal tank installed to handle liquid radioactive waste is pictured in Figure 2. Additional space has been provided for later tank additions for cleaning and decontamination solutions.

Positioning the Generator in the SGEF (Task 2)

The 220-ton service-degraded steam generator will be lowered through a removable roof panel in the SGEF and positioned vertically. A Commerce Business Daily announcement sought expressions of interest from subcontractors, and a statement of work package has been prepared that provides for a two-stage selection of the subcontractor. First, a technical proposal will be submitted and then technically qualified responders will be requested to submit a price. The generator move is scheduled for early December.

Generator Research Tasks

The data management system task (task 4) will provide computer software and arrange for use of computer hardware to handle information derived from research on the Surry generator. Analog signals will be transmitted by underground cable approximately 500 ft from the SGEF to a computer facility in the 314 building. The analog data will be stored on tape and will also be digitized using a PDP 11/44 computer system that will also control the NDT probe pusher-puller. An indexing device will be provided to automatically input probe position information into the computer along with NDT signals. Digitized data will be stored on discs, and information will be transferred to a VAX 11/780 system with software for analyzing large data sets. Digitized data will also be archived. Both computer systems will allow terminal data input of visual observations, analytical results, etc. A graphics capability is also being provided that will plot generator cross sections and show summary information about the cross section.

All computer hardware has been received and installed for system verification. The software for various input/output devices has been completed, and software packages are available for statistical analysis. Demonstration of the data management system will utilize historical ISI eddy current tapes of the Surry generator provided by the Virginia Electric and Power Company (VEPCO) and Westinghouse. A topical report on the data management/analysis system has been prepared for submission to NRC.

(a) Heating, ventilating, and air conditioning.

Neg. 8105614-13cn

FIGURE 1. HVAC System Being Installed in the Steam Generator Examination Facility

Neg. 8105614-9cn

FIGURE 2. 2000-gal Tank for Storage of Liquid Radioactive Wastes

The health physics task (task 3) provides procedures for personnel exposure monitoring, control, and documentation. Health physics research activities will include radiologic mapping, determination of the effectiveness of decontamination efforts, and evaluation of waste and waste disposal problems associated with various operations. Efforts this quarter include continued inputs into the SGEF Operational Readiness Plan and development of a training program for SGEF personnel.

The first research task to be done once the generator is placed in the SGEF will be to reopen preshipment penetrations (task 5). The preshipment examination involved cutting three penetrations through the generator shell while the unit was stored at the Surry nuclear station. Corrosion product and dimension data were acquired as was an assessment of general transportability and condition of the unit. Reopening these penetrations will provide information on any changes that may have occurred during shipment. The examination will be conducted by PNL personnel with a subcontractor providing cutting operations. Detailed task planning, radiation work procedures, and a statement of work for the subcontract activities were completed this quarter. The examination is scheduled for December 1981.

Discussions were held with two potential program participants concerning decontamination of the channel head (task 6) to minimize radiation exposure during primary side access. The availability of the channel head is being offered as an opportunity for research related to development/demonstration of decontamination techniques. Scheduling of this task will depend on whether the task is conducted via a subcontract or by a program participant.

Baseline eddy current ISI (task 7) will be conducted for a 100% examination of all generator tubes and to establish the best possible nondestructive definition of the generator tube condition. Detailed milestone and quality assurance plans have been developed for this task. Discussions have been held with Intercontrol concerning its participation.

The objective of task 8 is to unplug those generator tubes that were plugged during the service life of the generator. Initially planned as a selective process based on evaluation of the historical ISI tapes, tube plugging criteria at the time of plugging, and the results of task 7, it is now anticipated that all plugged tubes will be unplugged prior to task 7. A prebid process was completed this quarter to establish costs and potential vendors. This task has been rescheduled for early 1982 completion. The subcontractor bid process and selection will be conducted during the coming quarter.

In task 10—secondary side access—penetrations will be made through the generator shell for inspections and sample removal. Early coordination on this task is necessary to establish how and where cuts will be made and in which order. Task planning has been initiated regarding penetration size, geometry, location, and timing. In coordination with task 5, the best method for cutting is being explored and subcontractor work statements are being prepared.

Management Activities

Efforts continued to broaden participation in and technical support of research on the Surry IIA steam generator via a Steam Generator Group Project. Because of the unique research potential provided by the availability of the service-degraded steam generator, NRC is allowing other parties to join all or portions of the program. This quarter presentations were made to Japanese utility interests, the Taiwanese utility, and several private nuclear service industry concerns. Letters of interest have been received from four parties to date.

Details of an operating agreement to define the mechanisms for additional participants to join the NRC program were established by PNL, NRC, and the U.S. Department of Energy.

Other management activities included preparation of operating documents toward startup of the SGEF and detailed planning of fiscal year (FY) 1982 task actions.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

Stress Corrosion Cracking Characterization

A round robin test consisting of eddy current characterization of 10 laboratory-produced intergranular stress corrosion cracked (IGSCC) Inconel 600 steam generator tubes has been completed. The specimens will shortly be subjected to confirmatory destructive assay. Results of the round robin will be available the first quarter of FY 1982.

Leak Rate Tests

A series of leak rate tests were initiated on through-wall defected steam generator tubes. The defects consisted of laboratory-induced axially oriented IGSCCs of various lengths. Tubes were pressurized with 600°F water, and leakage to ambient air was measured. Conditions leading to crack growth and massive tube failure were examined as were leak rate determinations. The test series will continue with cracks of different orientation. Difficulties in producing circumferentially oriented tight SCCs are still being encountered.

MILESTONES

The data acquisition/analysis system was completed, and a topical report was prepared.

PROBLEMS

Difficulties in fabrication of leak rate specimens continue to delay this task.

FUTURE WORK

During the coming quarter the following activities will be pursued:

- continued marketing of the Steam Generator Group Project
- complete the SGEF
- position the generator in the SGEF
- conduct a destructive assay of round robin SCC specimens and issue a topical report
- complete leak rate tests
- complete the Phase II report, Part A
- issue a topical report on the data acquisition/analysis system.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

Stress Corrosion Cracking Characterization

A round robin test consisting of eight current characterization of 18 laboratory produced metal samples. These samples included 180°C/1000 psi and 200°C/1000 psi. The samples were subjected to constant stress with the results of the round robin will be available in the second half of 1992.

Leak Rate Tests

A series of leak rate tests were initiated in the spring of 1991. The tests consisted of 180°C/1000 psi and 200°C/1000 psi. The tests were conducted with the use of a leak rate test facility. The results of the tests will be available in the second half of 1992.

ANALYSES

The data acquired from the tests was analyzed and a final report was prepared.

PROBLEMS

During the execution of the tests, several problems were encountered.

FUTURE WORK

During the coming year, the following activities will be completed:

- complete the testing of the steam generator samples
- complete the testing of the 180°C/1000 psi samples
- complete the testing of the 200°C/1000 psi samples
- complete the testing of the 180°C/1000 psi samples
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