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Reactor Safety Research Programs

Quarterly Report
July - September 1980

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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Reactor Safety Research Programs

Quarterly Report
July - September 1980

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Washington, D.C. 20555
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ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from July 1 through September 30, 1980, for the Division of Reactor Safety Research within the Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining structural graphite strength, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the remaining integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Test assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation and postaccident coolability tests for the ESSOR Test Reactor Program, Ispra, Italy; blowdown and reflood tests in the test facility at Cadarache, France; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL). These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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GRAPHITE NONDESTRUCTIVE TESTING(a)

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SUMMARY

An algorithm has been developed for obtaining near-surface oxidation profiles from eddy current measurements. Low-force dry-coupling materials have been identified for use with ultrasonic backscattering equipment. Information on impurity distributions in oxidized graphites supports our previous conclusions regarding the "inverse oxidation" phenomenon.

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)-1980 scope of this project is to:

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

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

(a) RSR Fin. Budget No.: B2101-0; RSR Contact: R. Foulds

TECHNICAL PROGRESS

Research is being conducted to determine the feasibility of both eddy current and ultrasonic methods for nondestructive measurement of graphite oxidation. The current emphasis of our work is on the use of eddy current measurements for profiling near-surface oxidation and the use of ultrasonic backscattering for obtaining the oxidation profile at greater depths.

EDDY CURRENT TESTING

An algorithm has been developed for estimating the conductivity profile by examining the raw eddy current data obtained at three test frequencies. Further development and testing of the algorithm will be conducted to verify that the method is viable for use on large blocks of oxidized graphite and for developing procedures for transforming the conductivity profile to profiles of density (oxidation) and strength.

We have combined a computer program developed by C. V. Dodd of the Oak Ridge National Laboratory (ORNL) with a parameter-fitting code in a manner that permits the conductivity of the layers to be deduced from multifrequency coil impedance data. The ORNL program was designed to calculate eddy current coil impedance as a function of the conductivity and thickness of several layers of material in the search field. By assuming a layer thickness, it should be possible to calculate an accurate average value of oxidation for each layer.

ULTRASONIC TESTING

Ultrasonic backscattering is being evaluated for assessing oxidation at greater depths than is possible using eddy current techniques. Two materials have been identified that can be used for dry-coupling the ultrasonic signal through the sensor-graphite interface. These are two brands of silicone rubber that will withstand temperatures of 300°C or more. When used as a coupling member, the rubber will provide low attenuation (0.5 to 2 dB) and will require only a few pounds of force to be exerted on the sensor.

INVERSE OXIDATION PROFILES

In our last two quarterly reports^(1,2) we presented data indicating that in some impure graphites the oxidation rate increases toward the interior of the samples. This phenomenon was also linked with an increase in number and size of voids formed during oxidation.

Samples of oxidized PGX have been examined using a Perkin-Elmer model 590 scanning auger microprobe. The resolution (0.2 μm) of this instrument enabled us to analyze the elemental composition of material in the voids as well as along the top surface of the samples. Only carbon and traces of oxygen were found on the surface; but significant amounts of calcium, iron, and sulfur were found inside the voids.

End slabs (3-in. diameter by 1-in. thick disks) that had been removed from oxidized cylindrical samples were oxidized in air at 700°C. The total ash content ranged from 880 ppm (ATJ) to 3820 ppm (PGX) for the five as-received graphites; and purified samples of S1090 and 2020 yielded 17 and 32 ppm, respectively. Both the color and texture of the ash were distinctly different for each grade of graphite. In all of the unpurified samples the primary impurity appeared to be evenly distributed through the sample with individual agglomerations of a different color.

The colors of both the primary ash and the small agglomerates varied with the grade of graphite. The ash from both purified samples also agglomerated into dark fibrous or dendritic clumps, which appeared to be randomly distributed through the sample. We plan to analyze the ash under FY-1981 funding and will attempt to obtain elemental analysis of individual components.

FUTURE WORK

EDDY CURRENT TESTING

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

ULTRASONIC TESTING

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

OXIDATION PROFILES

- determine elemental distributions in the ash samples.

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

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SUMMARY

Preparation of the A533B steel insert for the ZBl German vessel test has been delayed by problems in locating a fabricator. Since fabrication in the United States would require 26 weeks, which would impact the vessel test schedule, a bid request for fabrication in Germany is in process.

Test specimens for acoustic emission (AE) characterization are being fabricated from degraded A508 material received from Materialprüfungsanstalt (MPA), Stuttgart, Germany. MPA plans to insert a piece of this material in the ZBl test vessel. Calculations indicate that external heating of the ZBl vessel to generate cladding-vessel interface stressing for potential acoustic noise is possible but not optimum.

Two of three major subsystems for the breadboard AE monitor system to be applied in the ZBl vessel test are operational, and the third (waveform recording) is in fabrication. Pattern recognition analysis with a composite of three sets of waveforms has demonstrated the need to relate the algorithm to a given AE detection system. The initial method and procedure for applying pattern recognition in the ZBl vessel test has been established.

Arrangements are in progress to install AE monitoring instrumentation on a reactor; and negotiation of a subcontract with GARD, Inc., to demonstrate in-process weld flaw detection by AE is in process.

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

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 used for all experimental testing. Program objectives 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: off-reactor vessel test, AE monitor system development, pattern recognition, reactor installation, weld monitor demonstration, and reports. The final section describes the work planned for the next quarter.

TECHNICAL PROGRESS

OFF-REACTOR VESSEL TEST

Work this past quarter concentrated on locating a manufacturer to fabricate the A533B insert for the ZBl test vessel. Three request for bids (RFBs) were processed and resulted in one response for machining the insert to shape and a companion response for machining required artificial flaws. Precracking of the machined flaws would be performed at Oak Ridge National Laboratory (ORNL). The total estimated cost for preparing the insert by this route is \$85,000; and the estimated time required is 26 weeks from start of fabrication to delivery of the finished insert to MPA, Stuttgart, which would cause a serious delay in starting the test. Negotiations with MPA have indicated that the possibility exists for fabricating the insert in Germany. The preliminary time estimate is 18 weeks; the total estimated cost has not been determined. An RFB to prepare the insert has been submitted to MPA.

A block of degraded A508 steel was received from MPA that is representative of material that will be used to fabricate a test insert (KS07) of identical geometry to the A533B insert. The KS07 insert, however, will not contain artificial defects. The KS07 material is a low-shelf steel containing micro-cracks, which were intentionally produced by off-normal processing procedures. Since current plans call for AE monitoring of the entire vessel, test specimens are being fabricated from the degraded A508 block to enable characterization of AE from this material.

One potential source of acoustic noise that will be investigated during this test is thermal stressing of the base metal-to-weld cladding interface. Since the test conditions at MPA are limited to 90°C, there is a problem in devising a method for simulating this potential acoustic source. One possible solution to this problem--heating of the test vessel from the outside--has been evaluated. Feasibility calculations were performed using simplistic considerations; consequently, the following results only indicate trends.

- Thermal cycling of the vessel outside diameter (OD) at a rate of 2 Hz will be totally ineffective in stressing the inside diameter (ID) stainless steel (SS) clad. Compared to thermal cycling of the vessel ID, stresses will be lower by a factor of about 10^{-14} .
- The desired level of stress in the ID SS clad can be achieved by cycling the OD temperature at a relatively slow rate--about 4 min/cycle versus 2 s/cycle for ID temperature cycling. The resulting AE rate (counts per minute) would be expected to be substantially reduced; however, it should be representative of the type of acoustic signals in a reactor.
- If the ID clad is to be effectively stressed by OD temperature cycling, a substantial area (i.e., more than a wall thickness in dimension) of the OD would need to be heated. This could present experimental difficulties since a significant volume of the vessel would be affected.
- There may be substantial differences in the distribution and senses (tension versus compression) in the SS clad as a function of ID versus OD thermal cycling.

In summary, stressing of the weld clad-to-base metal interface by OD thermal cycling appears possible. The need for OD thermal cycling may be avoided, however, if tentative test plans calling for the last portion of the test to be performed in a test loop at Mannheim, West Germany, are approved. The Mannheim test loop can provide test temperatures to 290°C and would allow thermal stressing of the clad by cold water thermal shock from the vessel ID.

AE MONITOR SYSTEM

The AE monitor system breadboard for vessel monitoring and subsequent reactor testing is shown in Figure 1. It is termed a breadboard system because it uses as much existing equipment as possible to demonstrate a function. A spectrum of data are recorded to facilitate post-test analysis to improve monitoring methods. Ultimately, we expect to design an engineering prototype that would optimize the instrument system and eliminate a number of peripheral features such as waveform recording. At that point, a pattern recognition analysis will be performed in essentially real time.

This monitor system concept utilizes two arrays for the German vessel test: One will concentrate on the A533B insert to maximize AE data acquisition from that area, and the other will monitor the total cylindrical section of the vessel to better simulate in-service monitoring. The upgraded Nuclear Regulatory Commission (NRC)-owned D/E 1032 AE monitor will be used to determine AE source location and associated signal parameters such as peak time and duration. This information will be stored on disk together with the parametric features, and it will go to a PDP 11/03 computer for current analysis.

Digitized signal waveforms will be recorded on tape in parallel with processing through the D/E 1032 system. These waveforms will then be played back to the PDP 11/03 for pattern recognition analysis.

The PDP 11/03 will also be used to sort pattern recognition information on the basis of source location derived from the D/E 1032 system. A common waveform numbering will be used to correlate pattern recognition results with source location. Data accepted by pattern recognition and associated with a specific location will be used to estimate flaw severity.

FIGURE 1. AE Monitor/Analysis System Concept

As a contingency backup for the vessel test, a zone isolation subsystem is also provided. In the event that the number of noise signals is so great that the waveform recorder is essentially totally occupied recording these at the sacrifice of AE signals, the zone isolator can be used to restrict waveform acceptance to a specified area. It will also serve to identify AE from individual flaws in the A533B insert if the source location system does not provide adequate source resolution.

Upgrading of the NRC-owned D/E 1032 has been completed, and the pattern recognition/flaw evaluation subsystem has been assembled. All major components for the balance of the system are on hand and design/fabrication is in process. The zone isolation subsystem is an existing instrument.

PATTERN RECOGNITION

The work done during this quarter included analyzing the composite data set and planning the onsite pattern recognition analysis during the vessel test.

Analysis of Composite Data

The composite data are atypical of any given AE monitoring circumstances because different sensors and different gain values exist within the data. The purpose of this analysis is to estimate the effect of sensor and gain changes on reaching a common algorithm to separate AE from noise. The composite data are from three specimen-sensor combinations:

- 1-in. thick laboratory specimen with a conventional surface-mounted sensor (BNW4)
- 3-in. thick wall cylindrical bend specimen with a high-temperature surface-mounted sensor (Westinghouse)
- 3-in. thick wall cylindrical bend specimen with a broad band waveguide sensor.

Valid AE and various types of noise waveforms were obtained for each specimen-sensor combination; a total of 649 waveforms were obtained. The number of waveforms for each sensor and AE source is given in Table 1.

TABLE 1. Waveforms Obtained with Each Specimen-Sensor from Various AE Sources

<u>Source</u>	<u>Specimen-Sensor Combination</u>		
	<u>BNW4</u>	<u>Westinghouse</u>	<u>Wave-Guide</u>
Valid AE	146	132	38
C-Clamp	17	29	16
Tapping	31	27	4
Rubbing	41	66	17
Background	22	31	20
Hydraulic	--	12	--

The analysis used 11 features, which are independent of signal amplitude; they are listed below.

mn = mean

sd = standard deviation

tp1 = total power 200-400 KHz/total power 200-1000 KHz

fm1 = first moment 200-400 KHz/first moment 200-1000 KHz

sm1 = second moment 200-400 KHz/second moment 200-1000 KHz

tp2 = total power 400-600 KHz/total power 200-1000 KHz

fm2 = first moment 400-600 KHz/first moment 200-1000 KHz

sm2 = second moment 400-600 KHz/second moment 200-1000 KHz

tp3 = total power 600-800 KHz/total power 200-1000 KHz

fm3 = first moment 600-800 KHz/first moment 200-1000 KHz

sm3 = second moment 600-800 KHz/second moment 200-1000 KHz

The feature reduction analysis resulted in 10 selected features, which are listed in their order of importance: sd, tp2, sm3, tp3, fm3, sm2, fm2, sm1, mn, and fm1.

A least-squares analysis on a training set produced a decision rule with an overall correct valid AE-noise classification rate of only 78%. It is not surprising that this is lower than the success rates that were obtained for the individual specimen-sensor data sets since they ignored any specimen and sensor differences. A closer look at the results shows that the success rates

differ for the valid AE and noise waveforms (74% and 82% for valid AE and noise, respectively). The successful classification rates for each sensor are listed in Table 2.

TABLE 2. Classification Rates for Each Sensor

<u>Source</u>	<u>BNW4</u>	<u>Westinghouse</u>	<u>Wave-Guide</u>
Valid AE, %	85	62	71
Noise, %	66	88	95

The success rates vary considerably, but no one sensor is clearly better. Comparing these results with those obtained using individual specimen-sensor data sets provides strong evidence that the pattern recognition technique must be calibrated to a given AE detection system.

Plans for the Vessel Test

The criteria for constructing the vessel test decision rule assume that there will be many fewer valid AE signals than innocuous noise signals in an operating reactor environment; therefore, it is important to identify all valid signals at the expense of misclassifying a higher percentage of the noise. An overall success rate of 75% would be acceptable if close to 100% of the valid AE were accepted. The noise signals could then be screened out by other means (such as source location).

During vessel testing, there will be neither time nor core space on the computer to do a complete pattern recognition analysis of the data. The onsite analyses will include the essentials needed to evaluate the decision rule, and the wave-guide will be the primary sensor to be used in the test. Analysis of the composite data has shown that the decision rule should be calibrated on data from the sensor that will be used; therefore, the parameters of the decision rule will be fitted from available wave-guide data.

The procedure for evaluating the decision rule will be to:

- compute the features
- scale the features

- compute the value of the least-squares decision function
- compare the computed value to the decision rule value.

These analyses will implement the decision rule and provide a classification of waveforms, which will then be combined with data from the source location system. Detailed post-test analysis of recorded waveforms will be performed at PNL.

REACTOR INSTALLATION

Based on discussions with Tennessee Valley Authority (TVA) personnel at Knoxville, Tennessee, it appears to be technically and administratively feasible to install an AE monitor system on a TVA reactor for testing and evaluation. Installation of sensors, preamplifiers, and signal leads out of containment will probably be completed by January 1981; however, full test monitoring will not start until early 1982 due to the reactor completion schedule.

DEMONSTRATION OF FLAW DETECTION BY AE DURING WELDING

Demonstration of the weld flaw AE detection/analysis system developed by GARD, Inc., (sponsored by NRC) to the German technical community at MPA, Stuttgart, is planned. The demonstration comes under the technical information exchange understanding with West Germany and will be performed under the auspices of this program. Negotiation of a subcontract with GARD, Inc., to perform the demonstration is in progress.

REPORTS

- Quarterly progress report for the period April 1-June 30, 1980.
- Trip Report: Saarbrücken and Stuttgart, Germany, June 2-4, 1980.

FUTURE WORK

In the period October 1-December 31, 1980, we plan to:

- complete fabrication and assembly of the breadboard AE monitor for the vessel test

- fabricate the A533B steel insert for the MPA vessel test
- prepare an analysis-before-test document
- complete tests for AE characterization of degraded A50B German material
- complete evaluation of reactor flow noise simulation
- finalize arrangements for installation of an AE-sensing subsystem on a reactor
- AE-monitor irradiated fracture specimen tests at the Naval Research Laboratory.

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

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SUMMARY

Primary emphasis during the past quarter was directed toward sample preparation for the round robin inspections that will begin in January 1981. Significant accomplishments include: satisfactory completion of fabrication of 33-in. outside diameter (OD) clad ferritic pipe samples, development and demonstration of an intergranular stress corrosion cracking (IGSCC) process for cracking 10-in. Schedule 80 pipe, progress on thermal fatigue cracking of pipe samples, and completion of the Phase I report for the program. In addition, experiments designed to define the most suitable procedure improvements are in progress, and development of a system for measuring and evaluating transducer and instrument characteristics has been initiated.

INTRODUCTION

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

The Integration of Nondestructive Examination (NDE) Reliability and Fracture Mechanics Program at Pacific Northwest Laboratory (PNL) has been established to determine the reliability of current in-service inspection

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

(ISI) techniques and to develop recommendations that will assure a suitably high inspection reliability. The objectives of this program are to:

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

The scope of this program is limited to ISI of primary piping systems, and the results and recommendations are also applicable to Class II piping systems. Programs currently in progress concerning inspection reliability of pressure vessels are also being monitored and evaluated.

TECHNICAL PROGRESS

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

TASK 1 - WHITE PAPER

The White Paper is a critical review of available literature concerning the reliability of NDE. Chapters 1-6 and 9 have been completed and distributed. Chapter 10, "Failure Statistics and Flaw Significance," has been completed; Chapter 11, "Relevant Statistical and Probabilistic Models," is in final review; Chapter 12, "Probabilistic Fracture Mechanics," has been completed; and Chapter 13, "Probability Models - Reliability of Flaw Detection - Statistical Data Confirming Detection Reliability," is in review. It is expected that Chapters 10-13 will be distributed in November, that the

remaining chapters will be completed in fiscal year (FY)-1981, and that the complete document will be published in 1982.

TASK 6 - SAMPLE FABRICATION

Three sample fabrication activities are in progress: 33-in. OD ferritic pipe weldments, IGSCC samples, and thermal fatigue crack growth samples.

Ferritic Pipe Weldments

The two 33-in. OD A106 grade C pipe samples were welded and ultrasonically and radiographically examined; they were acceptable. Stainless steel (SS) cladding was applied on the inside diameter (ID) of the pipes. It is expected that the pipe samples will be delivered to PNL by October 1, 1980.

IGSCC Samples

A 10-in. Schedule 80 304 SS pipe was cracked (IGSCC) at PNL during the past quarter using the graphite wool technique. This technique involves placing graphite wool pads over the heat-affected zone (HAZ) adjacent to the weld and placing the pipe in an autoclave at $\sim 288^{\circ}\text{C}$, 8-ppm oxygen for 600 hours. The character (tightness, orientation, location, and depth) of the flaws produced depends on the residual stress distribution degree of sensitization and length of exposure. The flaws produced in the 10-in. Schedule 80 pipe were predominantly circumferential, linear, and located within 2 mm of the weld root; the depth is not known at this time.

Material was purchased for six 10-in. diameter Schedule 80 pipe samples for IGSCC to be used in the round robin tests. Three of the samples have been welded and are ready for exposure. The fabrication and treatment processes that were applied are expected to yield cracks that are more typical of those that have occurred in the field (i.e., more random in orientation and discontinuous rather than linear). When these samples are placed in the autoclave and exposed for 600 hours, a range of crack sizes is expected. The next three samples will be processed and exposed for times that are expected to yield the desired range of flaw sizes.

The development of a capability to grow IGSCC-type cracks in the United States is a significant achievement; the technique was originally developed in Japan by Ishikwajima-Harima Heavy Industries. It has been suggested that the best method for assuring effective and reliable detection of IGSCCs in service is to require that the inspection techniques be qualified using pipes that contain defects typical of those found in the field. This requirement is now practical.

Thermal Fatigue Cracking

Thermal fatigue cracking of the 10-in. Schedule 80S SS and the 27.5-ID cast SS pipes is in progress. Cracking of two of the six 10-in. pipes has been completed, and work is in progress on a third pipe. Eleven of the 16 cast SS samples have been cracked, and work is in progress on three more. It is expected that cracking of all samples, including the clad ferritic 33-in. OD samples, will be completed by mid-December 1980.

TASK 7 - MEASUREMENT AND EVALUATION

Major efforts during this quarter have been in trying to make the SUTARS data analysis computer program function properly and in defining and preparing to resolve the measurement and evaluation open items presented in the Research Information Letter (RIL) of September 9, 1980, which is currently under review by Nuclear Regulatory Commission (NRC) staff.

Search Unit Tracking and Recording System (SUTARS)

The SUTARS hardware--control unit, ultrasonic instrument, sensor array, 4-track recorder, 4-track playback unit--seems to be working well. This will be verified by making a SUTARS data tape from a sample with known machined reflectors and shipping it to Southwest Research Institute (SWRI) for processing.

PNL personnel have been unsuccessfully attempting to discover why the SUTARS data analysis program is not working; SWRI, the vendor of the program, has sent troubleshooters to investigate the problem. Although they did not solve the problem, they are of the opinion that the difficulty is in the data

analysis system hardware--probably in the reading function of the 9-track magnetic tape unit. SWRI believes that it will be difficult and time-consuming to remedy the situation. Currently, a PNL computer scientist is investigating the system and will enlist the assistance of a computer maintenance/troubleshooting expert, the computer vendor, and the tape unit vendor as necessary.

Side benefits of this correction effort have been the addition and modification of hardware to allow successful operation of the Gould 5005 printer/plotter and hardware modifications that corrected communication problems between the Kennedy 4-track magnetic tape reader and the Data General Nova 4S computer.

Measurement and Evaluation Open Items

Review of the ASME Code, Section XI, 1977, current literature, and PNL measurement and evaluation results has revealed several areas of deficiency in current inspection requirements as described in the RIL of September 9, 1980. These areas are:

- calibration reflector and sensitivity
- inspection angle
- sizing
- surface conditions and contour
- scan parameters
- coverage of inspection volume
- transducer and instrument performance
- differences between inspection of ferritic welds and inspection of austenitic or dissimilar metal welds
- differences in attenuation and beam angle between calibration block and pipe base metal
- allowance for austenitic weld metal attenuation and beam steering in single-side access conditions.

Some of these items must be resolved before the round robin begins since they will influence the improved procedure that is to be implemented. Others are not related to the round robin (for example, the round robin includes no dissimilar metal welds); therefore, their resolution is not needed as quickly.

The calibration reflector used for the round robin improved procedure will be a high aspect ratio (short) flaw of maximum allowable depth as defined in Tables IWB-3514-2 and -3 of ASME Code, Section XI, 1977. Precisely which flaw--from the preservice table, the in-service table, or another of our own design but derived from these tables--is yet to be resolved.

Record/report sensitivity levels must be set on the basis of a large mass of data (not all of which is available yet) including crack geometry effects, roughness and tightness effects, and base and weld metal attenuation variables as well as the desired probability of crack detection.

The round robin improved procedure will require that both 45° and 60° beam angles be used, based primarily on investigation of the influences of crack and counterbore geometry.

Crack length sizing can be performed fairly accurately by a simple probe motion technique. Depth sizing, however, will not be part of the round robin improved procedure because we have found no reliable technique for making this measurement. Fortunately, depth sizing is not as important as crack detection; in practice, when a crack is found it is almost always removed regardless of depth. If a crack is detected and depth determination is deemed necessary, we would recommend fabrication of natural and machined reflectors in samples as nearly identical as possible to the cracked component and careful qualification of sizing techniques on these reflectors.

In the round robin, all samples will have well-ground weld crowns and pipe surfaces so that surface roughness, crown noise, and restriction of search unit movement due to the weld crown will not be present. There will, however, be surface contour due to diametrical shrinkage, possibly aggravated by crown removal. This contour could reduce coupling and change both the beam shape and angle, thus reducing reflection amplitude and causing errors in flaw location. Contoured plates are now being designed for measurements verifying and quantifying these effects.

The scan parameters category includes scanning speed, overlap, and search unit skew.

The 6-in./s maximum scan speed presently in force is considered too high, and the improved procedure will reduce this to 1-2 in./s for manual inspection. Since skewing of the search unit up to 45° from axial improves detection of IGSCC, it will be required for probable IGSCC conditions.

The present Section XI 10% scan overlap requirement (adjacent scans overlapping by a minimum 10% of transducer diameter) has been found to be insufficient. Furthermore, it was determined that no single number can assure adequate but not excessive scan overlap. Measurement of optimum scan overlap will be part of calibration in the round robin improved procedure and will be performed using the high aspect ratio calibration reflector.

Assurance of full coverage of the inspection volume will depend on the resolution of some of the above items, notably surface condition and contour and scan overlap. There are special problems in some cases, such as change in beam direction in anisotropic weld metal (the beam can actually follow a curved path). Further, the inspector should be required to make an accurate sketch of the weld cross section to aid in inspection analysis and insure that the inspector knows exactly what he is inspecting.

An instrumentation system is under development to characterize transducer and instrument performance. It will be used to characterize round robin inspection equipment and the influence of variations in these characteristics on system performance. The objective of these measurements is the identification of key parameters and development of measurement techniques to assure effective control of the inspection process. The parameters to be measured include:

- transducer - insertion loss, percent bandwidth, frequency bandwidth center, impedance as a source, impedance as a receiver
- instrument - output impedance, output voltage with 50Ω load, pulse spectrum with 50Ω load, receiver input impedance, receiver amplifier bandwidth

- transducer-instrument combination - pulse spectrum at transducer and shear wave beam profile in steel as a function of distance.

The measurement system is to be interfaced with a small computer that will produce report-quality test results.

It is not expected that all of the parameters listed above will have to be measured to assure the effectiveness of tests performed in the field. However, these parameters will allow measurement and calculation of synergistic effects of transducers, cables, and instrument and control settings. It is expected that the measurement system will be operational by early December 1980.

Code Section XI fails to mention some of the crucial differences between inspection of ferritic and austenitic welds. (Dissimilar metal welds, which are not part of the round robin, have not yet been considered.) The code makes no mention of the sometimes wide variability of attenuation in austenitic steel, which can vary between the calibration block and the pipe base metal, between the base metal and the weld metal, and even between different locations within the base metal. Most of these variations can be mapped using fairly simple techniques (single-V pitch-catch or even normal beam L-wave). Samples are being fabricated that will allow measurement of sensitivity loss in inspection performed through austenitic weld metal, which must be known in the case of single-side search unit access. They will also allow measurement of beam steering effects.

Some austenitic stainless steels are so attenuative and "noisy" that record/report criteria expressed as an amplitude in terms of distance amplitude correction (DAC) may not be practical; a signal-to-noise (S/N) ratio may be more appropriate. In the very worst cases (for example, in the case of centrifugally cast SS piping) crack indications (even for large cracks) sometimes just barely exceed the mean noise level and are themselves exceeded by noise peaks. Inspectors may have to simply report what "looks" like a crack with no attention to amplitude or S/N ratio.

In some austenitic steels, ultrasonic velocity may vary between the calibration block and the pipe base metal. Since beam angle is a function of ultrasonic velocity, this condition could cause error in data analysis and flaw location.

TASK 8 - ROUND ROBIN

The round robin inspection is scheduled to begin in early January 1981 and will be completed in June 1981. It is expected that the remaining test samples will be completed by early December 1980. Contractual negotiations are in progress with the six test teams, and a draft questionnaire and test protocol have been submitted to the test teams for their review and comment. A trial run on a selected test sample will be conducted in November 1980 to establish the effectiveness of the test protocol and allow revisions prior to the round robin.

FUTURE WORK

Key tasks to be completed during the coming quarter include:

- completion of Chapters 10-13 of the White Paper
- completion of sample fabrication and cracking, including six IGSCC pipe samples
- finalization of an improved procedure for use in the round robin
- conduct of a round robin trial run to test the effectiveness of the test protocol and procedures.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD CODES

TASK A - IRRADIATION EXPERIMENTS(a)

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

M. E. Cunningham
R. E. Williford

SUMMARY

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

Irradiation test IFA-431 is completed. Two other tests (IFA-432 and IFA-513) are still under irradiation. The final test, IFA-527, began irradiation this quarter. All the IFAs are heavily instrumented six-rod clusters.

Fuel failures that occurred this quarter in IFA-513 (1 rod) and IFA-527 (3 rods) are under investigation. Postirradiation examination (PIE) and compliance testing of two rods from IFA-432 at Harwell, UK, progressed satisfactorily.

INTRODUCTION

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

- 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

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

- integrate the above information into the FRAPCON and FRAP-T series of computer codes.

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

- powers up to 50 kW/m (16 kW/ft)
- gap sizes of 50-350 μm (0.002-0.0025 in.)
- gas compositions ranging from pure helium to pure xenon
- fuel types of 95% to 92% theoretical density (TD), respectively stable and unstable regarding in-reactor densification.

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

TECHNICAL PROGRESS

Irradiation of IFA-432 continued this quarter with no further instrument failure, and one rod (rod 5) failed in IFA-513 due to cladding or seal failure. IFA-527 began irradiation on July 1, 1980. This six-rod assembly contains five nominal-gap rods (230 μm) and one small-gap (60 μm) rod. All rods are filled with xenon gas to magnify thermal effects and all contain 95% TD stable fuel. Rods 1-5 will assess the speed and variability of fuel relocation, while rod 6 will provide a crosscheck on rod powers and an upper boundary on the conductance of a fission gas saturated rod. The rod powers are held very low (15 kW/m peak) by placing the assembly at the extreme edge of the core. This area has significant flux tilt, which provides a spectrum of power levels for the replicate rods.

Each rod has a continuous readout pressure transducer (bellows type), an elongation detector, and two centerline (top and bottom) thermocouples (TCs). On initial startup, all instruments functioned with the exception of the rod 6 pressure transducer, which failed immediately. All other instrumentation functioned well during an irradiation period extending into mid-August 1980. When the reactor started on September 14, however, three pressure transducers (rods 1, 4, and 5) went off-scale (approaching system pressure), the upper TC in rod 5 failed, and the lower TC in rod 2 became intermittent. This signaled the pressure-barrier failure of rods 1, 4, and 5 and possibly rod 2. Rods 3 and 6 appeared to continue normal operation. No fission product leakage could be detected.

Discussions are continuing with Halden over the probable cause of the rod failures and the possible future of IFA-527. The reactor will be shut down for most of the coming quarter.

The calibration of IFA-527 was successfully accomplished just after startup. All of the nominal-gap rods registered a thermal resistance increase of 5-10% after the first rise to power, which could be due to fuel densification; resintering tests on the fuel are underway. Further data analysis will appear in next quarter's report.

Gamma scanning, profilometry, and rod puncture have been completed at Harwell, UK, on uninstrumented rod 8 from IFA-432 (a 230- μ m gap, helium-filled rod containing 95% TD stable fuel pellets). It had an average burnup of 22,000 MWd/MTM upon removal. Major PIE results were:

- Cumulative fission gas release fraction was no more than 9%.
- Cladding creepdown was about 0.05 mm (-0.4% diametral strain).
- Interpellet ridging was definite; peak-to-valley ridge heights were about 0.05 mm.
- Interpellet deposition of cesium was apparent at the upper (high power) end of the rod.
- Gross gamma intensity had an axial profile in line with the power axial profile.

FUTURE WORK

The test apparatus for determining the axial compliance of rod 8 has been fabricated. Commissioning tests, measurements on unirradiated rod 7, and in-cell measurement of rod 8 are expected to be completed next quarter.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

TASK B - DATA QUALIFICATION AND ANALYSIS(a)

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

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

SUMMARY

A major objective of the Experimental Support and Development of the Single-Rod Fuel Codes Program is the irradiation of instrumented fuel assemblies (IFAs) to obtain well-characterized data. Task B of this program is responsible for qualifying and analyzing that data. During this quarter a report on error analysis was completed and a journal article on transient temperature analysis was published. Work also continued on the analysis of transient temperatures, modeling and data errors, and fission gas release.

INTRODUCTION

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

(a) RSR Fin. Budget No.: 82043; RSR Contact: H. H. Scott

- Subtask B-1 - Data Processing: This subtask is responsible for receiving, correcting, characterizing, and presenting the data obtained from the fuel assemblies.
- Subtask B-2 - Data Reports: This subtask is responsible for preparing reports on the precharacterization of the fuel assemblies, the data obtained from the assemblies, and the postirradiation analysis of the assemblies.
- Subtask B-3 - Data Analysis: This subtask is responsible for providing in-depth analysis of the in-reactor fuel rod data. Specific areas of interest for fiscal year (FY)-1980 are analysis of data for inferring fuel relocation and its effect, use of transient temperature data to better understand fuel behavior, analysis of statistical variations and error propagation, and analysis of fuel rod fill gas pressure data for inferring fission gas release.

TECHNICAL PROGRESS

This quarter's activities are discussed below by subtask.

SUBTASK B-1 - DATA PROCESSING

During the last quarter, a test fuel data report (TFDR) tape for the period March to May 1980 was received for IFA-432 and IFA-513. The data was corrected and made available to the other subtasks for analysis. Preliminary data (July 2-4, 1980) was also received on the startup of IFA-527. This data was sufficient to prepare the power calibration routines for the handling of IFA-527 TFDR tapes, but questions still exist concerning the power calibration of the assembly.

SUBTASK B-2 - DATA REPORTS

Rod 6 of IFA-431 and IFA-432 contained 92% theoretical density (TD) fuel that was unstable with respect to in-reactor densification. During the PIE of rod 6, IFA-431⁽¹⁾ it was found that the fuel densified to 96.5% TD after an

irradiation of 366 GJ/kgU (4240 MWd/MTM) (preirradiation thermal resintering tests gave a final density of 95.3% TD). Efforts were begun this quarter to report on the observed microstructural changes and the observed thermal behavior of the rod and to provide some analysis of the densification behavior.

Figure 1 presents the thermal resistance^(a) behavior for the high power region of rod 6 of IFA-431 during its early irradiation. The rapid resistance increase during the first 4.3-6.5 GJ/kgU (50-75 MWd/MTM) is indicative of densification. The resistance increase was completed by approximately 43 GJ/kgU (500 MWd/MTM).

FIGURE 1. Early-in-Life Resistance Behavior of Rod 6, IFA-431

(a) Thermal resistance is analogous to electrical resistance and is defined as the difference between the fuel centerline and coolant temperature divided by the local linear heat rate.

Data from this rod is also being compared to predictions by the codes GAPCON-THERMAL-3 (GT3)⁽²⁾, FRAPCON-1⁽³⁾, and FRAPCON-2^(a). This work has just begun, and the following general results have been obtained:

- GT3 - No densification underpredicts temperatures while densification overpredicts temperatures.
- FRAPCON-1 - Overpredicts temperatures.
- FRAPCON-2 - The PELET option underpredicts temperatures while the FRACAS option overpredicts temperatures.

SUBTASK B-3 - DATA ANALYSIS

During this quarter analysis continued in the areas of transient temperature response, modeling and data errors, and fission gas release.

Transient Temperature Data Analysis

Transient temperature data from the first 3 days of irradiation for IFA-527 was received this quarter. An initial analysis of this data (using preliminary estimates of power) indicates transient temperature behavior midway between the response of a solid pellet and a fully cracked pellet. This is similar to the behavior that was observed for IFA-513 at beginning-of-life (BOL).

An article entitled "Use of Fuel Thermocouple Transient Response for Data Verification and Fuel Rod Modeling" was printed in the mid-August issue of Nuclear Technology.⁽⁴⁾

Modeling and Data Error Analysis

A report entitled Application of Linear Propagation of Errors to Fuel Temperature and Stored Energy Calculations⁽⁵⁾ was completed this quarter. This report discussed uncertainty analysis for different rod designs, the process of model validation, and the effect of cracked pellet fuel models upon temperature uncertainty. The work that was performed in these three areas may be briefly summarized as follows:

(a) FRAPCON-2 is not yet publicly available, October 1980.

- Using a postulated power history, the uncertainty for the predicted response of boiling water reactor (BWR) and pressurized water reactor (PWR) fuel rods was evaluated. BOL relative uncertainty for BWR and PWR fuel rods is approximately the same while end-of-life (EOL) uncertainty differs for BWR and PWR fuel rods.
- Determining the validity of modeling relative to reality is discussed in qualitative terms. Validity is dependent upon verifying that the code correctly implements the model and that satisfactory agreement is found between the model and reality. Validity is a subjective concept and will vary to fit the circumstances.
- Fuel modeling codes are now using cracked pellet fuel models, which result in decreased fuel surface temperatures. Estimated stored energy is lowered, but its relative uncertainty is increased. In general, however, the absolute upper uncertainty bound for stored energy is lower for a cracked pellet model than for a solid pellet model.

Fission Gas Release Data Analysis

Internal pressure data from IFA-432 and IFA-513 through May 1980 were received from Halden and examined at PNL. The rod average burnup for IFA-432 is currently greater than 2300 GJ/kgU (26 GWd/MTM), and the estimated fission gas release is less than 10% for the two fuel rods with operating pressure transducers. The estimated release fractions have remained fairly constant for the last 600 GJ/kgU (7 GWd/MTM), which is not consistent with a strong burnup dependence on fission gas release. However, a detailed analysis of the fuel temperatures and gas release histories is required to determine if a weak burnup dependence exists; this analysis will be made after IFA-432 is discharged from the reactor.

The pressure data from rod 6 of IFA-513 showed a pressure spike during the first startup after the spring shutdown for maintenance and refueling. The characteristics of the pressure spike were similar to those observed previously for rod 5 of IFA-513, which was identified as defective and subsequently

removed from the assembly. The similarities in the pressure spikes suggest that rod 6 may have developed a small defect.

A data report for IFA-513 was written and is currently being reviewed at PNL. The report contains in-reactor data collected from IFA-513 for the irradiation period from November 1978 to January 1980. Fuel temperatures, cladding elongation, internal pressures, and power levels are presented as a function of time; and the general trends in the data are discussed. This report will be published in the next quarter.

FUTURE WORK

The data processing subtask will continue to handle data as it arrives. Corrections will be made to the IFA-513 computer routines to account for the removal of rod 5. After receipt of IFA-527 precharacterization and startup data, those two reports will be completed. The analysis of rod 6 of IFA-431 and the documentation of CRAKR will also be completed. The analysis being performed under the data analysis subtask will continue into FY-1981. The transient temperature analysis will principally support the relocation analysis. A topic of interest under the error analysis work will be the evaluation of the identical rods in IFA-513 and IFA-527.

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

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

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

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

M. E. Cunningham
W. N. Rausch

SUMMARY

The FORTRAN coding for FRAPCON-2 has been completed, and developmental assessment cases are being evaluated. Publication of the Code Description and User's Manual is forecast for December 1, 1980. The first mechanical compliance tests were performed on short rods, and the Harwell (UK) apparatus for compliance testing of instrumented fuel assembly (IFA)-432 rods is nearly completed. A relocation kinetics (fuel ratcheting) algorithm was completed, and the first calculations were performed based on the cracked fuel geometry from rod 6 IFA-431 postirradiation examination (PIE) photomicrographs.

INTRODUCTION

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

During fiscal year (FY)-1979 fuel performance models were defined that more closely represent physical reality in a preliminary code structure to facilitate the implementation of these models. The fact that operating nuclear fuel is cracked results in significant reductions in effective thermal conductivity and elastic moduli. The state of the cracked fuel is described by three primary mechanical parameters: crack roughness, gap roughness, and crack

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

pattern. However, those parameters have only been inferred from thermal in-reactor results. In FY-1980 we hope to establish a firm physical foundation for the cracked fuel model and implement these models in a combined PNL-INEL version of FRAPCON-2.

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

TECHNICAL PROGRESS

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

SUBTASK C-1 - CODE COORDINATION AND DEVELOPMENT

The FORTRAN coding for FRAPCON-2 has been completed, and the code's developmental assessment cases are being evaluated. Any obvious errors will be corrected, and the code will be released for general use. The draft of the Code Description and User's Manual is nearly completed with a forecasted publication date of December 1, 1980. The developmental assessment document is being compiled as computer information becomes available.

SUBTASK C-2 - MECHANICAL TESTING OF SHORT RODS

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

A test apparatus for pushing axially on the fuel column and measuring resultant diametral and axial strain of the rod has been built at PNL, and a similar apparatus for testing irradiated rods has been built at Harwell, UK. The remainder of the electronics package for the PNL test apparatus was received during this quarter; the test apparatus was calibrated, samples were

prepared, and the first tests were conducted. Diameter traces were good, but rod bowing due to misalignments gave erroneous cladding elongation data. For the first series of tests, a substitute ceramic (steatite) was found to simulate fuel cracking more closely than alumina. Profilometry and gamma scan data for rod 8 of IFA-432 were received from Harwell, where companion compliance tests on this irradiated rod will be conducted.

SUBTASK C-3 - FCMI MODEL DEVELOPMENT

An advanced model is under development to characterize the effects of asymmetric pellet cracking that result in fuel column anisotropy and FCMI. Programming of the relocation kinetics algorithm (FRAGMT) was completed, and the first calculations were performed. The highly nonlinear response characteristics of the cracked fuel system during fragment ratcheting resulted in an ill-conditioned response matrix with resulting unstable solutions. This was corrected through a normalization procedure and some coding modifications. A case was run based on the geometry from rod 6 IFA-431 PIE photomicrographs. The average discontinuity in the fuel surface profile due to misalignment of adjacent relocated fragments was 0.015 mm; the largest discontinuity was 0.025 mm. The average and largest misalignment angles between the fragments and the cladding inside diameter (ID) were 0.26° and 0.51° , respectively. Irregular gap geometries are indications of stress concentrations occurring at the cladding inner surface during normal operation.

SUBTASK C-4 - MODEL VERIFICATION

Experimental efforts are yielding data that verify the three primary mechanical parameters of the cracked fuel relocation and mechanical response model. Efforts to quantify crack patterns and the response of single crack systems are nearly complete; the only remaining experiment is the bulk modulus test for the prototype cracked fuel pellets. This will be done in a triaxial rock mechanics facility being built at PNL.

FUTURE WORK

The following activities are planned for the next quarter.

- The Code Description and User's Manual will be published, and FRAPCON-2 will be released.
- The developmental assessment of FRAPCON-2 will be completed.
- A presentation on FRAPCON-2 will be made at the Water Reactor Safety Meeting in October.
- The short rod mechanical compliance tests should be completed.
- Compliance testing on IFA-432 rods at Harwell will near completion.
- The two-dimensional FRAGMT relocation code results will be compared with in-reactor results⁽¹⁾ for cracked fuel effective moduli and conductivities.
- Two-dimensional relocation modeling should be completed, and three-dimensional work will begin.
- Bulk modulus testing of a cracked fuel prototype is scheduled to begin.

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

LOCA SIMULATION IN NRU(a)

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P. N. McDuffie, Project Administration

SUMMARY

Final assembly and checkout of the thermal-hydraulic test train was completed. The unit was shipped on schedule to Chalk River Nuclear Laboratories (CRNL), Chalk River, Canada, to test the thermal-hydraulic behavior of a full-length fuel bundle during heatup, reflood, and quench phases of a large-break loss-of-coolant accident (LOCA). The test series is currently scheduled to begin October 21, 1980.

Modifications to the U-2 loop at CRNL are almost complete. The data acquisition system was checked out, and the test operations plan was completed.

Component fabrication and assembly of the next two materials test trains began. CRNL anticipates that the next test window will be available February 1981.

INTRODUCTION

The objective of the Pacific Northwest Laboratory (PNL) LOCA Simulation in NRU Program is to provide information on the heatup, reflood, and quench phases of a LOCA. The tests are designed to give information on the quench-front velocities within a fuel bundle, the liquid entrainment [10 CFR 50, App. K (Sec. ID 2)], and the heat transfer coefficients [10 CFR 50, App. K (Sec. ID, 5)] for full-length pressurized water reactor (PWR) fuel as a function of reflood rate and delay time before reflood starts. A total of six test trains will be prepared for the program; the first test train will be devoted to thermal-hydraulic behavior and more than 40 tests are planned. The remaining five test trains will be used for only one test each. These test

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

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

The test loop in the NRU reactor will accommodate the 12-ft long, 32-pin bundle on a 6x6 array with the corner pins removed. The bundle design uses commercial enrichments, cladding dimensions, and grid spacers.

TECHNICAL PROGRESS

Final assembly and checkout of the thermal-hydraulic test train was completed. The unit and all supporting checkout and handling equipment were shipped to CRNL, and the test operations plan as amended to incorporate internal review comments was published.

Modifications to the U-2 loop at CRNL are progressing. PNL is assisting CRNL in obtaining needed components (particularly gray-loc fittings, flowmeters, and pressure relief valves) to meet the targeted test date of October 21, 1980.

A program review meeting was held on September 9, 1980, with the sponsor and several user groups. As a result of this meeting, PNL will attempt to increase the number of thermal-hydraulic tests from the originally planned 18 to more than 40 tests. The additional tests were requested by the user groups to obtain data at delay times and reflood rates not specifically covered in the original test matrix.

The major components of the disassembly, examination, reassembly machine (DERM) are in various phases of fabrication. The DERM control computer was received and software development began. This unit will be installed in the fuel pool at CRNL for disassembly and examination of the test trains and for loading the new fuel units for materials tests 3, 4, and 5.

Figures 1-7 depict the various stages of final assembly of the thermal-hydraulic test train.

FIGURE 1. Guard Rod Assembly (foreground) and Replaceable Test Bundle (background) Assembled Separately; Test Bundle is Then Placed in Guard Rod Bundle

FIGURE 2. Shroud Instrument Leads Installed in Lower Shroud Half

FIGURE 3. Thermocouple, Steam Probe, and Self-Propelled Neutron Detector Leads Routed Through the Seal (background); Involves 182 Instrument Leads

FIGURE 4. Three Connectors Installed to Connect Instruments to Data Acquisition Control System

FIGURE 5. Lower End View of Assembled Test Train Shroud

FIGURE 6. Shroud Assembly Inserted in Hydraulic Test Assembly

FIGURE 7. Completed Test Train Installed in Shipping Container

FUTURE WORK

The thermal-hydraulic test series will be conducted at Chalk River, Canada, starting October 21, 1980. Fabrication and assembly of the next two materials test trains will continue, and DERM component assembly will begin.

CORE THERMAL MODEL DEVELOPMENT(a)

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SUMMARY

No contribution was submitted for this quarterly report.

INTRODUCTION

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

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

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

FUEL ASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM(a)

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SUMMARY

An additional subtask has been added to the Fuel Subassembly Procurement and Irradiation Test Program in support of the Power Burst Facility (PBF) Severe Fuel Damage Task. It consists of a small-scale boil-off experiment that will investigate the sensitivity of water level control to absolute water level during simulated small-break conditions. In addition, the experiment will examine the effects of water level on the axial temperature profile of the electrically heated rod bundle. During this quarter, pretest analytical simulations were completed to determine the feasibility of the experiments and to examine possible operating conditions and the temperatures of various components within the testing system. Detailed design of the fuel pins and core regions of the test train assemblies was started, and detailed design of the lower portion of the test train assembly was completed and is being reviewed.

Planning sessions were held to define the large-break and severe fuel damage tests to be run on the Super Sara loop in the ESSOR reactor at Ispra, Italy. The current matrix contains 21 tests consisting of pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assembly geometries subjected to large-break and severe fuel damage conditions.

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

INTRODUCTION

The Fuel Subassembly Procurement and Irradiation Test Program at Pacific Northwest Laboratory (PNL) is divided into four tasks for fiscal year (FY)-1980. The objectives of these tasks are as follows:

- Task A - Multirod Test Trains - PBF Severe Fuel Damage Test Program: To supply two fully instrumented test train assemblies for the PBF Severe Fuel Damage Test Program. The scope of work includes the design effort, development of appropriate materials and supportive fabrication processes, and complete fabrication of the assemblies. Many portions of the PBF work should directly benefit the Super Sara Test Program (SSTP) due to similarities in experimental objectives-- particularly for the materials development, instrumentation, and fabrication development required to support the development of the PBF test train assemblies. 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 B - Gap-Conductance Test Analysis: To provide testing methodologies and analysis techniques for obtaining fuel rod thermal performance from PBF experiments and provide post-test analysis and data interpretation of experimental results.
- Task C - Dual-Purpose Assemblies: To transfer the instrumented fuel assembly (IFA)-513 dual-purpose test rods, which are currently under steady-state irradiation in Halden, Norway, to the PBF for subsequent transient testing.
- Task D - Multirod Test Trains - SSTP: To supply fully instrumented and fueled test assemblies, analytical services, and engineering support for the SSTP in the ESSOR reactor at Ispra, Italy. The program is designed to yield important experimental data related to fuel rod deformation and postaccident coolability of damaged fuel rod clusters after small-break, medium-break, and large-break loss-of-coolant accidents (LOCAs).

TECHNICAL PROGRESS

MULTIROD TEST TRAINS - PBF SEVERE FUEL DAMAGE PROGRAM

4X Replacement Quadrants

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

Small-Scale Boil-Off Experiment - Feasibility Study Results

A small-scale boil-off experiment will be conducted at PNL to investigate the sensitivity of water level control to the absolute water level during simulated small-break conditions. The experiment will also examine the effects of water level on the axial temperature profile of the electrically heated rod bundle. Electrical heater rods with approximately 1 meter (39.4 in.) of heated length and internal cladding thermocouples (TCs) will be installed in a seven-rod array inside a specially designed and fabricated high-pressure test section. The heater rods will have linear axial and radial power profiles and will consist of a central Nichrome heating element surrounded by MgO insulation and contained within an Inconel sheathing material. The section will provide visual access to the boiling/water level interface zone and will be instrumented to measure the inlet and outlet water/steam conditions as well as water level. Water at approximately 21⁰C will be injected into the bottom of the test section at rates sufficient to maintain water levels between 10.2-22.9 cm (4-9 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.

Pretest analytical simulations of the seven-rod array within the high-pressure test section were completed. The feasibility of the suggested experiment has been established, and suggestions as to the possible operation of the testing sequence were made.

The TRUMP heat transfer computer code was used to perform the scoping pre-test calculations, and the heater rod array was simplified for the TRUMP analysis. The center heater rod was modeled in its true geometry, but the outer six rods were lumped together in a cylinder around the central heater rod. Figure 1 shows the cross-sectional geometry of the simplified heater rod test

FIGURE 1. Schematic Top View of System Analyzed for Seven-Rod Boil-Off Test

array. The total volume of MgO insulation and Inconel sheath material of the outer heated cylinder in the TRUMP model contains the volume of materials equivalent to the total of the six individual heater rods in the array. The total sheath heat transfer area for the outer heater cylinder in the TRUMP model is equivalent to that for all six heater rods. Radiation heat transfer is accounted for in the model by allowing the central heater rod as well as the outer heated cylinder to radiate to the shroud.

Figure 1 also illustrates the remainder of the test apparatus as modeled in the TRUMP code. Two steam channels exist within the containment shroud with mass flow rates weighted to maintain a correct ratio of sheath heat transfer area-to-steam flow rate. Outside the quartz shroud is an Al_2O_3 insulator surrounded by a stainless steel (SS) pressure tube. The quartz is nearly transparent; therefore, the model assumes that the rod array radiates directly to the Al_2O_3 insulator. Two gaps of stagnant steam exist between the shroud, insulator, and pressure tube.

In the TRUMP model, only the assembly above the liquid-steam interface is considered; and a quiescent interface is assumed with no frothing. The steaming rate is computed by performing an energy balance on the water below the interface. The components of the assembly shown in Figure 1 were modeled to run the full axial length of the bundle.

Initial calculations showed that the outer pressure tube temperatures would be high with only free convection cooling on its outer boundary. As a result, a cooling jacket with 367K (200°F) water will surround the pressure tube and help to keep the temperature of the pressure boundary within lower design limits.

Two TRUMP simulations of the boil-off test were performed: one with a 10.2-cm (4-in.) water level and the other with a 22.9-cm (9-in.) water level. Both simulations operated at 900 W/m-rod (900 W/rod) power for the heater rods. The initial temperature throughout the system was 558K (545°F), and the calculations were terminated when a steady state was reached within the test rods. Figures 2 and 3 show the steady-state axial temperature profiles for the two water levels. The water level had very little effect on system temperatures or

FIGURE 2. Seven-Rod Test Bundle Simulation - Axial Temperature Profiles at the 10.2-cm Water Level

on the shape of the profile because the steaming rates were so small for both cases that the energy loss was primarily through the shroud. An energy balance shows that 93.8% of the energy input was lost radially for the 10.2-cm water level and 85.9% was lost radially for the 22.9-cm water level. Thus, the axial power profile is the most significant factor in determining the shape of the axial temperature profile.

Simulation results indicate that overheating the pressure tube will not be a problem if the cooling water flow is maintained. The peak pressure tube temperature did not exceed 394K (250°F) in either simulation.

Figure 4 illustrates the heatup time required to achieve steady-state conditions. Initially, the entire system was at 558K (545°F) for the two simulations; and after about 1000 seconds, the rods should be at a steady-state temperature.

FIGURE 3. Seven-Rod Test Bundle Simulation - Axial Temperature Profiles at the 22.9-cm Water Level

MULTIROD TEST TRAIN - SUPER SARA TEST PROGRAM

A conceptual test train design review was held at PNL with several Ispra, Italy, staff. The advantages and disadvantages of using the single PNL test train design for both large-break and severe fuel damage tests were discussed. One of the main disadvantages is that the fuel assembly instrument leads would unnecessarily lead out of the bottom of the fuel assembly and then up along the outside of the flow shroud. A simpler design for the large-break tests would route the instrument leads up and out of the fuel assembly. The main advantages of using just one test train design are cost reductions in design, procurement, fabrication, assembly, installation in the reactor, and postirradiation examination (PIE) and more data consistency among the tests.

Several technical planning (brainstorming and task force) sessions were held to define the SSTP. The current matrix consists of 4 large-break PWR fuel

FIGURE 4. Seven-Rod Test Bundle Simulation - Heatup from 558K Temperature Versus Time

assembly geometry tests, 2 large-break BWR tests, 11 severe fuel damage PWR tests, 2 severe fuel damage BWR tests, and 2 undefined tests. The test schedule calls for about six tests per year starting in mid-1983. The tests cover the peak cladding temperature range from 1070 to 2300K.

Information and exchange meetings were held with EG&G, Idaho,^(a) staff who are involved with PBF-Severe Fuel Damage tests and with the fuels subcommittee of the Advisory Committee on Reactor Safeguards (ACRS).

(a) EG&G, Idaho, Inc., Idaho Falls, Idaho.

FUTURE WORK

TASK A - MULTIROD TEST TRAINS - PBF SEVERE FUEL DAMAGE PROGRAM

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

TASK D - MULTIROD TEST TRAIN - SUPER SARA TEST PROGRAM

Conceptual test train designs will be modified as required by analysis and thus become preliminary designs. Instrumentation and shroud concepts will continue to be developed to satisfy severe fuel damage test requirements.

STEAM GENERATOR TUBE INTEGRITY(a)

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SUMMARY

Detailed design of the Steam Generator Examination Facility (SGEF) was completed. The bid package for construction of the SGEF was issued; seven bids were received in response. A draft research program plan covering planned research on the Surry generator was submitted to the Nuclear Regulatory Commission (NRC) for comment. Research continued on stress corrosion crack (SCC) characterization and leak rate determinations.

INTRODUCTION

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

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

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

various nondestructive testing (NDT) techniques, mainly eddy current testing. The tube specimens were next tested to failure at PWR steam generator operating temperatures. The failure strength, actual flaw dimensions, and NDT-indicated flaw dimensions were used to derive mathematical relationships; and these relationships were subsequently plotted to provide, within a statistical certainty band, the remaining mechanical integrity of a steam generator tube as a function of its flaw type and size as indicated by eddy current testing.

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

The first two phases of the program involved the study of mechanically (Phase I) and chemically (Phase II) defected tubing. Phase III of the original program included correlating the mathematical models developed in Phases I and II with actual service-flawed tubing. However, a lack of suitable specimens led to the redirection of Phase III into an effort to conduct extensive nondestructive and destructive evaluations on a retired-from-service nuclear steam generator. A generator^(a) removed from the Surry II nuclear plant (Surry, Virginia) after 6 years of service was judged suitable for this research.

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

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

Research efforts on the Surry 2A generator will be initiated shortly after the generator is placed in the SGEF, which is scheduled for completion in June 1981. Research will be conducted in the following areas:

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

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

TECHNICAL PROGRESS

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

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

Tube specimens with through-wall SCCs of various lengths and orientation are being fabricated, and approximately half of the specimen matrix has been completed. The high-temperature flowmeter (capable of measuring flows to 400 gpm at 650°F and 2500 psig) that is necessary for leak rate determinations has been received. We are currently attempting to establish a high-temperature, high-pressure water reservoir larger than the existing 150-gal unit; a capacity of 1000 to 3000 gallons is desirable.

SURRY GENERATOR PROJECT

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

Steam Generator Examination Facility (SGEF)

Title II (definitive) design of the SGEF was completed. After incorporation of J. A. Jones and PNL review comments, Facilities Systems Engineering Corporation (FSEC)--the architect-engineer subcontractor--issued the final Title II design report. J. A. Jones issued a bid package for a construction contractor, and seven bids were received in response. George Grant was the low bidder; however, contract award awaits commitment of fiscal year (FY)-1981 funding.

Management Activities

A draft research program plan for the Surry generator was prepared and submitted to NRC for comment, and a revised version of the program plan was prepared based on the comments received. NRC and PNL program managers held a meeting with the Electric Power Research Institute (EPRI)-Steam Generator Owners Group management and technical leaders. Discussions centered around potential EPRI or Steam Generator Owners Group participation in the program. Negotiations are in progress between NRC and PNL to form a Steam Generator Projects Office, which would allow PNL to establish and manage a multisponsor research effort on the Surry generator.

Milestones

- Title II design of SGEF completed.
- Bid package issued for SGEF.
- Draft research program plan submitted to NRC.

Problems

FY-1981 funding must be in hand before the SGEF construction contract can be let; any delay in receipt of funding will delay the building schedule. Completion of Phase II work of the original program is falling further behind

schedule due to difficulties with nondestructive characterization of tight SCCs. The FY-1980 annual report is behind schedule but is close to completion.

FUTURE WORK

During the next quarter the following activities will be pursued.

- The construction contract for the SGEF will be let, and ground will be broken for the SGEF.
- PNL and NRC program management will contact potential sponsors for joint participation in the Surry generator examination.
- Specimen preparation and testing activities on the leak rate test matrix will continue.
- A task to develop a computerized data storage and retrieval system for the Surry generator examinations will be initiated.
- A task to establish health physics procedures for operation of the SGEF will be initiated.
- Activities on Phase II SCC specimens will continue.

RESIDENT ENGINEER AT CADARACHE, FRANCE(a)

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

SUMMARY

Fill tests from the test matrix defined in February 1980 were run. "Quick look" reports for the first four tests were completed, and translations were sent to Nuclear Regulatory Commission (NRC) personnel. Complete documentation is currently underway.

INTRODUCTION

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

TECHNICAL PROGRESS

Tests 206, 207, 208, 209, and 101 have been run; the tests, which were defined by the test matrix of February 1980, are summarized in Table 1. There were no surprises from any of the tests, and the general results and important characteristics are described below.

(a) RSR Fin. Budget No.: B2278; RSR Contact: W. V. Johnson

TABLE 1. Summary Description of Tests Completed to Date

Test Number	Test Train Number	Number of Rods	Heating Rate, kW/m	Break Location	Initial Pressure, bar	Inlet Temperature, °C
206	Hydraulic	25	0	Hot Leg	155	320
207	25.01	25	27	Hot Leg	155	320
208	25.01	25	0	Cold Leg	155	320
209	25.01	25	27	Cold Leg	155	320
101	Hydraulic	1	0	Hot Leg and Cold Leg	155	320

Departure from nucleate boiling (DNB) occurred during both of the heated tests; however, clad temperatures did not rise above the initial temperature in either test. For the cold-leg break (test 209) DNB occurred when the flow through the assembly stagnated as it changed from forced flow through the loop to reverse flow exiting through the break. The rod then immediately quenched as the reverse flow was established. In the hot-leg break (test 207) DNB occurred late in the blowdown; and the clad temperature rise was arrested by the programmed accumulator injection, which was automatically initiated by the loop pressure trip at 15 bar.

Refill water was injected into the cold leg at the bottom of the test train in all of the tests. In the cold-leg break tests (208 and 209) the refill and reflood proceeded in a steady, orderly manner; but both of the hot-leg break tests (206 and 207) were characterized by large amplitude oscillatory reflood with a period of about 7 seconds.

Tests 207, 208, and 209 were run using the same test assembly, and in general the test train instrumentation functioned as expected. Test 206 used the hydraulic test train that was used for the reactor and loop commissioning tests. During this test approximately 1/3 of the assembly thermocouples (TCs) registered transient temperatures clearly below saturation. This same problem occurred during test 101 where all of the assembly TCs registered transient temperatures below saturation. The problem has been traced to the connectors between the TCs and hard lines that exit the test train; the insulation

surrounding the connection apparently absorbs water vapor and loses resistance. This problem is under study and will be eliminated before proceeding to test 102.

Quick-look reports for the first four tests were completed, and translations were sent to NRC personnel. Complete documentation is underway.

FUTURE WORK

After solving the problem with the TC connections, the program will continue with tests 102, 103, and 104. The primary parameters for those tests are given in Table 2.

TABLE 2. Summary Description of the Next Three Tests for the PHEBUS Loop

<u>Test Number</u>	<u>Test Train Number</u>	<u>Number of Rods</u>	<u>Heating Rate, kW/m</u>	<u>Break Location</u>	<u>Initial Pressure, bar</u>	<u>Inlet Temperature, °C</u>
102	01.01	1	57	Hot Leg and Cold Leg	155	320
103(a)	01.01	1	57	Hot Leg and Cold Leg	155	320
104(a)	01.03	1	57	Hot Leg and Cold Leg	155	320

(a) For these tests, refill will be delayed so that target clad temperatures of 1000°C and 1204°C will be reached for tests 103 and 104, respectively.

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16. ABSTRACT (200 words or less)

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from July 1 through September 30, 1980, for the Division of Reactor Safety Research within the Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining structural graphite strength, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the remaining integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Test assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation and post-accident coolability tests for the ESSOR Test Reactor Program, Ispra, Italy; blowdown and reflood tests in the test facility at Cadarache, France; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL). These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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