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Reactor Safety Research Programs

Quarterly Report
April-June 1982

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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ABSTRACT

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 1982, for the Division of Accident Evaluation and the Division of Engineering Technology, 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, including 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.

ABSTRACT

The document summarizes work performed by Pacific Northwest Laboratory (PNL) from April 1 through June 30, 1982 for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission (NRC). Evaluation of nondestructive examination (NDE) techniques and instruments are reported. Areas of investigation include demonstrating the feasibility of determining the strength of structural systems, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probability 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 piping pipe impacts followed by postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide benchmark data 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, including loss-of-coolant accident (LOCA) simulation tests at the NRC Reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident capability tests for the EBR reactor super test program, PWR tests; the instrumented fuel assembly installation program at Idaho, 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.

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GRAPHITE NONDESTRUCTIVE TESTING (NDT) RESEARCH(a)

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SUMMARY

The results of tests of the ZFIT code using homogeneously oxidized graphite standards are very satisfactory. Significant progress was made on the ultrasonic backscattering technique; the operating frequencies, test procedure, and engineering criteria are now defined as a result of two experimental investigations. Samples are being prepared from two additional logs of PGX graphite; data from these samples will considerably improve estimates of the precision that can be obtained when base-line information is lacking for in-reactor components.

INTRODUCTION

The Graphite NDT Research Program is a continuation of previous work at Pacific Northwest Laboratory (PNL) that has demonstrated: 1) 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, 2) that near-surface oxidation profiles can be determined from multifrequency eddy-current measurements, and 3) the technical feasibility of determining oxidation profiles at greater depths by ultrasonic backscattering techniques. The scope of this project is to:

- Continue development of eddy-current techniques for near-surface profiling of oxidation in the Fort St. Vrain PGX core support blocks, with particular emphasis on: 1) detailed design and assembly of a carefully controlled eddy-current probe to protect the signal, 2) additional testing of the algorithm for calculating electrical conductivity and density, and 3) development of a technique for in-reactor calibration of the probe.
- Continue development of ultrasonic backscattering techniques to evaluate oxidation profiles at greater depths; in particular, 1) continue development of a dry coupling method to provide sufficient signal quality, 2) develop correlation between backscatter signal and density, and 3) develop the software to interpret the signals.
- Determine appropriate arrangements to test the applicability of the above techniques in the Oak Ridge National Laboratory (ORNL) CFTL and carry out such testing as desirable, considering Fort St. Vrain testing schedules and funding limitations.
- As funds and schedules permit, outline and conduct portions of the work program for the development of techniques to predict oxidation depth profiles in reactor environments, providing strength indications for large graphite components.

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

TECHNICAL PROGRESS

Progress completed during this quarter is detailed below by topic.

(a) FIN: B2101-1; NRC Contact: R. B. Foulds.

EDDY-CURRENT TESTING

The main effort during the last quarter centered on constructing an improved eddy-current apparatus that will allow greatly improved repeatability for measurements taken at different frequencies. In addition, calculations were performed to check the workability of the ZFIT program.

Conductivity versus density relationships were determined by separately fitting the measurements on unoxidized and oxidized samples. The measurements from nine unoxidized graphite bars and eight bars oxidized from 1.5 to 10.1% each fit a relationship of the form

$$K = A + B\rho \quad (1)$$

where K is conductivity in kS/m and ρ is density in Mg/m^3 . The following values for Equation (1) were obtained:

- $A = -1.99$ and $B = 52.85$ for unoxidized bars only
- $A = -223.91$ and $B = 177.42$ for oxidized bars only.

These conductivity versus density relationships were used to deduce the conductivities of six graphite slabs used for eddy-current voltage response calibration. The changes in resistance and reactance from air values were calculated for the search coil placed on the graphite surface. This calculation was made at frequencies from 50 kHz to 1 MHz using the theory of C. V. Dodd et al., (1) which relates impedance of a driven coil to conductivities and thicknesses of an adjacent laminar medium. The measured voltages (V_x and V_y) from the search coil monitoring circuit with the coil against the graphite slabs were assumed to be related to resistance (ΔR) and reactance changes (ΔX) by the linear relationships:

$$V_x = b_1 \Delta R \quad (2)$$

$$V_y = b_2 \Delta X \quad (3)$$

The coefficients b_1 and b_2 were determined at each frequency to minimize the sum of the square error for the six graphite slabs. Measured V_y values versus calculated reactance changes (ΔX) from air values are plotted in Figure 1 for the 70-kHz frequency.

Figure 1. Measured (reactance-like) Voltage V_y for 0.5-in. Graphite Slabs with Calculated Reactance Change Using Dodd's Theory (Reference 1)

Two tests of the ZFIT code were performed using these sets of (b_1 , b_2) values and measurements on the D4-6 (0.42% oxidized) sample. Measured (V_x , V_y) values at 50, 60, 70, 80, 90, and 100 kHz were used. In Test 1, the code was instructed to vary a single parameter (conductivity) of a 0.5-in. graphite slab to match the measured V_x and V_y by the calculated $b_1\Delta R$ and $b_2\Delta X$ in Equations (2) and (3). In Test 2, both slab thickness and conductivity were determined by ZFIT. The calculated ΔR and ΔX values are functions of conductivities and thicknesses of lamina in the medium.

The results of the two ZFIT tests are presented in Table 1. The table also gives the conductivity calculated from density using the correlation determined for oxidized graphite. The results suggest that the linear conductivity versus density relationship determined for oxidized graphite can be used down to the low oxidation level of sample D4-6 (0.42%) but that a physical process associated with oxidation causes a dramatic change in the conductivity for very small amounts of oxidation.

The results of a one-parameter fit on sample D4-3 (4.46% oxidized) are also shown in Table 1 and support the hypothesis of a different conductivity versus density relationship for unoxidized and oxidized graphite. The conductivities of D4-6 and D4-3 given by Equation (1) at their preoxidation densities with the A and B values for unoxidized graphite were 89.60 and 89.44 kS/m, respectively.

The two-parameter determination of conductivity and thickness on sample D4-6 is quite satisfactory; however, it required over 2 min of computation on the CDC 7600. The estimated slab thickness for sample D4-3 is about 18% larger than the true value. Considering that the sensitivity of the eddy-current measurements diminishes rapidly with depth (and that the equipment is not intended for profile measurements below about 0.15 in. into the block), this result is more than satisfactory.

ULTRASONIC TESTING

Two experimental investigations have been completed. The first defined the ultrasonic attenuation coefficient of the materials as a function of frequency, and the second provided the laboratory validation for the ultrasonic backscatter measurements. The second experiment is of particular significance because the results provide the basis for making oxidation measurements on materials that have only single-surface access such as those in operating high-temperature gas-cooled reactors (HTGRs). The combination of data from the attenuation and backscatter experiments defines the operating frequency and the test procedure that will be used in the actual tests. These data also define engineering criteria for the ultrasonic test instrument.

The results of the backscatter experiments (see Figure 2) indicate that low oxidation produces a high level of backscattered energy while heavily oxidized specimens have substantially less backscatter.

Table 1. Comparison of Conductivity Correlation Predictions and ZFIT Results on 0.5-in. Graphite Slabs

Sample Number	Mean Density, kS/m	Conductivity Inferred from Eq. (1), kS/m	ZFIT Conductivity, kS/m		ZFIT Calculated Thickness for Test 2, in.
			Test 1	Test 2	
D4-6(a)	1.726	82.3	81.8	81.8	0.50
D4-3(b)	1.653	69.4	70.2	70.8	0.59

(a) 0.42% oxidized.

(b) 4.46% oxidized.

Figure 2. Ultrasonic Backscatter Measurements

These results are consistent with backscatter measurements in other materials but are the opposite to what was originally hypothesized. From the tests, it appears that measurements made at two frequencies (e.g., 0.5 and 0.75 MHz) would provide the data required to interpret the degree of oxidation existing below the surface of the graphite block. These tests are most encouraging in that the results show that the ultrasonic instrument can be less complicated than initially anticipated. Additional tests that are necessary before the electronic design can be finalized will be performed when new oxidized specimens are available.

OXIDATION

Quarter-round segments (11 in. long) from two additional logs of PGX graphite were obtained from General Atomic Company. A total of 56 samples are being machined from each bar for nondestructive and destructive measurements following various amounts of oxidation. Using samples from different logs of PGX will allow a better estimate of the variability that is introduced in the correlations of strength versus the measurable parameters by variability of the material in the logs. For example, a correlation of compressive strength (S) versus electrical conductance (K) will be possible for each bar, similar to the relationship reported last quarter.⁽²⁾

FUTURE WORK

EDDY-CURRENT TESTING

- obtain data using a new and improved laboratory eddy-current instrument
- complete fabrication of improved eddy-current equipment
- refine instrumentation parameters for ZFIT using new equipment.

ULTRASONIC TESTING

- design and develop the ultrasonic transducer network for the frequencies chosen for the test
- validate the frequency spectrum reflected from the various oxidized specimens
- complete development of the dry couplant procedure to be used in the reactor environment
- initiate design of the test system to be incorporated into the prototype model.

OXIDATION

- prepare strength and conductance samples from two other logs of PGX graphite
- measure strength of unoxidized and oxidized samples.

REFERENCES

1. Dodd, C. V., et al. 1973. *The Analysis of Reflection-Type Coils for Eddy-Current Testing*. ORNL-TM-4107, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
2. Edler, S. K., ed. July 1982. *Reactor Safety Research Programs Quarterly Report, January-March 1982*. NUREG/CR-2716, Vol. 1, PNL-4275-1, Pacific Northwest Laboratory, Richland, Washington.

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

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SUMMARY

The German ZB-1 vessel test is scheduled to begin in September 1982 at Mannheim, West Germany; a new acoustic emission (AE) data acquisition system to be used for the test is being assembled and tested. A topical report on AE monitoring the cold hydrotest at the Watts Bar 1 reactor is ready for publication. Analysis of AE data from fatigue crack growth in A106B pipe material at room temperature indicates that the AE rate versus crack growth rate fits within the limits for similar data from A533B steel. Scatter for the A106B data is greater, however, than for the A533B data. AE monitoring of a Type 304 stainless steel pipe being tested for stress corrosion cracking (SCC) indicated that cracking has started. Ultrasonic verification will be attempted prior to waveform recording for pattern recognition evaluation.

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 pressure vessel steel, SA351-CF-8A cast stainless, Type 304 wrought stainless, and A106 ferritic piping steels are being used in the experimental testing. The 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.

TECHNICAL PROGRESS

Progress relative to the objectives stated above is discussed in the following sections on off-reactor vessel test, reactor monitoring, pipe material characterization, and reports.

OFF-REACTOR VESSEL TEST

The German ZB-1 vessel test is scheduled to begin early in September 1982. Final details will be resolved at a meeting in Germany on July 12-14, 1982.

The original commercial AE monitor system was replaced with another system developed by Battelle-Northwest (BNW)(b) to enhance monitoring continuity for the vessel test. The entire ZB-1 vessel monitor system is shown in Figure 1. The waveform recording instrument and the analysis computer are unchanged from an earlier description.⁽¹⁾ The system has 12 channels that are subdivided into three arrays, and each array can have up to four sensors. Delta times and total counts are measured per sensor for all three arrays. These data, along with other acquired data, are passed over a high-speed serial link to a minicomputer-based system for source location computation. Data included are: delta times; test number; calendar/clock time; peak time, pulse height, and duration for one channel per array;

(a) FIN: B2088; NRC Contact: J. Muscara.

(b) Battelle, Pacific Northwest Laboratories, a division of Battelle Memorial Institute.

Figure 1. AE Monitor System for ZB-1 Vessel Test - Functional Diagram

eight parametrics; seven load parameters; tag number; and an external digital number. The system can send all data or selected portions; a maximum of 40 words can be sent at one time for each array.

Due to the serial link, data cannot be passed as fast as it is processed, which can create an undesirable situation with burst data. To circumvent the problem, data are stacked in a buffer at the processing rate and passed at nonactive moments. The buffer size is 800 events for collection of all 40 pieces of data.

Data are also backed up on a cartridge tape with mass storage capabilities (67 Mbytes) and a high-speed parallel interface and displayed on a video terminal. Two modes are possible: The itemized mode furnishes numerical data values while the graphics mode shows signal source locations by zone. Cylindrical, quad, and linear sensor arrays can be accommodated.

For correlation with the waveform recorder, a tag number is generated and sent to the waveform system. This tag number ultimately provides a means for matching a digitized waveform with its computed source location.

Software has been written for the digital PDP 11/03 minicomputer to receive output from the acquisition system. Data are stored on floppy disk and interpreted by a location algorithm to yield location coordinates of the AE sources. These operations occur concurrently with data acquisition. The location coordinates are used to construct a graphical representation of AE activity as a function of the position on the test specimen. A hard copy of the plot can also be made. The location coordinates for each event are combined with other event parameters and stored on floppy disk in a specified format for subsequent analysis.

REACTOR MONITORING

A topical report on the results from AE monitoring the cold hydrotest of the Watts Bar 1 reactor vessel has been reviewed and submitted to NRC for publication.

PIPE MATERIAL CHARACTERIZATION

The scope of the fiscal 1982 work was expanded to include AE characterization of primary piping materials, including A106B ferritic steel and wrought cast stainless steels.

A106B Pipe Specimen

A fatigue crack growth test on an A106B pipe specimen was completed. The specimen was machined from a 24-in. Schedule 160 carbon steel pipe with the chemical and tensile properties given in Table 1. The single-edge-notch test specimen geometry is shown in Figure 2. The longitudinal axis of the test specimen was parallel to the longitudinal axis of the pipe. A notch was saw-cut into the edge of the specimen 1 in. from the gage section centerline. The specimen was then polished to facilitate optical

Table 1. Chemical and Tensile Properties of A106B Pipe Material

Heat No.	Concentrations, wt%				
	C	Si	Mn	P	S
443549	0.24	0.26	0.97	0.029	0.011

Yield strength: 41.2 ksi
Tensile strength: 76.9 ksi
Elongation (2-in. gage): 29.0%

crack length measurements. Surface-mounted AE sensors were bonded to each end of the specimen gage section and calibrated prior to testing. The specimen was then loaded into an MTS test machine using a 1-Hz sine load waveform. The initial R-ratio for this test was 0.1, and it was increased to 0.6 after ~125,000 cycles by raising the minimum load. All testing was performed in air at room temperature. A plot of the cumulative crack length and AE event count versus load cycles is shown in Figure 3.

Figure 2. Geometry for Single-Edge-Notch Specimen A106B-1

Figure 3. Crack Length and AE Event Count Versus Load Cycles for Specimen A106B-1

An initial comparison of the AE event rate (counts/cycle) versus fatigue crack growth rate ($\mu\text{in.}/\text{cycle}$) is shown in Figure 4. The data generally fall between the upper and lower bounds for room temperature A533B data, indicating that the two materials have similar AE response behaviors. The scatter for A106B data is greater than for A533B data, particularly for the $R = 0.6$ loading condition. No explanation for this result has been determined at this time. Additional analysis of these data is planned to determine if peak time, pulse height, and load position information can be used to reduce the scatter.

Type 304 Stainless Steel Specimen

AE from SCC in Type 304 stainless steel is being investigated using 4-in. diameter Schedule 80 pipe specimens. Testing began with a specimen that was already welded into the test manifold and that had experienced 800 h of earlier testing. This test will result in a set of data from an accelerated SCC test conducted at minimal cost. The data can be used for subsequent comparison with data from a long-term SCC test, which should give insight to the effect of accelerated testing. From the results obtained thus far (see Figure 5), it appears that SCC started after ~ 1200 h of testing. The specimen will be ultrasonically examined during the next unload-reload cycle to attempt to verify the presence and

Figure 4. AE Event Rate Versus Fatigue Crack Growth Rate for Specimen A106B-1

Figure 5. AE Data from Stress Corrosion Cracking Test

location of SCC. Waveforms will also be recorded for pattern recognition evaluation. The fixtures required to install a second specimen in the test system are being fabricated with the objective of starting SCC after about 6 months testing.

REPORTS

- Quarterly progress report for the period from January 1 to March 31, 1982.

FUTURE WORK

Plans for the period from July 1 to September 30, 1982, include:

- continue pipe material characterization tests
- start the ZB-1 vessel test
- install permanent signal leads at the Watts Bar 1 reactor
- continue development of an engineering prototype AE monitor system
- issue the final report on Watts Bar 1 cold hydrotest monitoring.

REFERENCES

1. Edler, S. K., ed. September 1981. *Reactor Safety Research Programs Quarterly Report, April-June 1981*. NUREG/CR-2127, Vol. 2, PNL-3810-2, Pacific Northwest Laboratory, Richland, Washington.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS(a)

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SUMMARY

During the past quarter, progress was made on a number of tasks. New cold leg fracture mechanics (FM) calculations support the present ASME Code inspections as being conservative for pressurized water reactor (PWR) primary piping except for the nozzle location. The probabilistic FM model is being modified to apply the Pacific Northwest Laboratory (PNL) piping round robin probability-of-detection data in a situation of high failure probability due to thermal fatigue. Flaw growth calculations for a pressurizer surge line suggest that inspections should be performed more frequently than the typical ASME Code inspection interval of 10 yr. Manual evaluation of the piping round robin data shows the most common error is the large variability in the reported dB crack amplitude response. Evaluation of improved and advanced ultrasonic testing (UT) techniques for pipes began for two automated methods. Preliminary studies show that near-surface defects (6 mm and greater) under strip-type cladding can be detected with a high degree of confidence; the effectiveness of crack detection through manual cladding remains to be demonstrated.

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 PNL was established to determine the reliability of current in-service inspection (ISI) techniques and to develop recommendations that will insure a suitably high inspection reliability. The objectives of this U.S. Nuclear Regulatory Commission (NRC) program are to:

- determine the reliability of ultrasonic ISI performed on commercial light-water reactor (LWR) primary systems
- using FM analysis, determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to insure 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 insure suitably low failure probabilities.

(a) FIN: B2289-0; NRC Contact: J. Muscara.

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.

TECHNICAL PROGRESS

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

FRACTURE MECHANICS

Progress made on cold leg calculations, probabilistic fracture calculations, and a pressurizer surge line analysis is described in this section.

Cold Leg Calculations

Flaw growth calculations in PWR cold leg piping have been completed at Battelle-Columbus Laboratories (BCL). The calculations evaluated the growth of small cracks in PWR cold leg piping systems and compared the results with the probabilistic analyses from the Lawrence Livermore Laboratory (LLL) Load Combination Program.⁽¹⁾ The objective of the calculations was to provide a basis for evaluating flaw detection requirements and inspection intervals.

Table 1 summarizes the results of the recently completed calculations and compares the results with those of the previous BCL cold leg study.⁽²⁾ The present calculations used less conservative assumptions, and the cracks were allowed to grow from smaller initial depths (10% of the wall thickness). In all cases (except for the charging inlet nozzle), the revised calculations predicted fatigue lives greater than the 40-yr design life. It was therefore concluded that present ASME Code inspections are conservative for PWR primary piping (except for the nozzle location). A report describing the cold leg calculations is being prepared.

Table 1. Results of BCL Revised Cold Leg Evaluation

Component	Predicted Life, yr	
	Original Study	Revised Results
Straight pipe (circumferential flaw)	0 to 120	500 to 4000
Elbows (axial flaw)	0 to 120	150 to 1200
Dissimilar metal weld (circumferential flaw)	0 to 1	600
Nozzle (corner flaw)	0.4 to 16	20

Probabilistic Fracture Calculations

Arrangements were completed for probabilistic analyses at LLL. In this work, LLL will expand and apply their probabilistic FM model of PWR steam generator feedwater line cracking. The objective is to apply the PNL probability-of-detection data from the piping round robin in a situation of high failure probability due to thermal fatigue. The analysis at LLL will consist of:

- refinement of the existing model to treat service-induced flaws and to develop a probabilistic distribution for the da/dN curve
- calculations for a matrix of inspection scenarios that will address two plants (high and moderate failure probabilities), four inspection intervals, and three detection probabilities (poor, good, and advanced)
- calculations of transition matrices for the Markov-chain inspection model being developed at PNL.

The three detection probability curves were provided to LLL in June. The curves were based on extensions of the PNL piping round robin data for clad ferritic piping. The extensions involved: 1) extrapolating detection data for flaw depths of less than 5% of the wall thickness and 2) estimating detection curves for pipe wall thicknesses less than the 2.375-in. thick specimens of the round robin.

The calculations at LLL should be completed during the coming quarter. Computer simulations will begin once the data base for the da/dN distribution is compiled and analyzed.

Pressurizer Surge Line Analysis

Flaw growth calculations have been performed for a pressurizer surge line; the details of the piping design and the design transients were based on data for an actual PWR. The objective of the calculations is to evaluate ASME Code inspection requirements for piping with a higher level of fatigue usage than exists in PWR primary piping.

The piping configuration and the results of the crack growth predictions are given in Figure 1. Fabrication flaws were assumed to be equal in depth to those permitted by the ASME Section XI preservice acceptance standards. These postulated flaws were predicted to grow at much greater rates than similar flaws in PWR primary coolant piping due to the occurrence of thermal fatigue transients in the surge line. The unsafe region of Figure 1 corresponds to the flaw growth to a depth greater than one-fourth of the wall thickness. The crack growth factor plotted in Figure 1 is similar to a fatigue usage factor; a predicted factor greater than 1 implies an unsafe condition. The flaw growth analyses for the surge line predicted failure in about one-seventh of the design life of 40 yr, which suggests that inspections should be performed more frequently than the typical ASME Code inspection interval of 10 yr.

MANUAL EVALUATION OF THE PIPING ROUND ROBIN DATA

The data collected during the piping round robin are being reviewed to identify all sources of error. Errors that are clearly of PNL origin (for example, incorrectly entering data into the computer) will be corrected. All other error sources will be identified, and their frequency will be established by recording each occurrence. Currently, all of the clad ferritic specimens and half of the 10-in. Schedule 80 pipe specimens have been reviewed.

The most common error is the large variability in the reported dB crack amplitude response. The next most common error is in plotting the location of defects, and the third most common type of error is in making physical measurements (i.e., misreading the ruler or using the wrong response). The errors are currently being classified to facilitate determination of the scoring strategy. The probability of detection data will probably be reported with and without error corrections; all dB responses, however, will be corrected.

EVALUATION OF IMPROVED AND ADVANCED ULTRASONIC TESTING TECHNIQUES

Although the data analysis from the piping round robin⁽³⁾ is preliminary, there is clear evidence that substantial improvements are required for reliable inspection of centrifugally cast stainless steel (CCSS)

(a) Surge Line Geometry

(b) Crack Growth Factor Versus Location on Surge Line

Figure 1. Crack Growth Analysis of Surge Line

and Type 304 stainless steel piping. A number of ultrasonic techniques have been proposed by various researchers to provide the inspection improvements for the aforementioned materials.

During the quarter, two improved techniques were used to inspect some of the round robin pipe specimens. The first technique—the automatic ultrasonic test system (AUTS)—was developed by the Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. The AUTS operates at its noise limit and thus provides inspections that are far more sensitive than conventional ultrasonic inspections. Images are made of the inspection area, providing C-scan views of the weld root and any anomalies. The use of an automated scanner should also reduce measurement errors. The AUTS was used to scan 10-in. Schedule 80 stainless steel pipes and CCSS; results will be evaluated next quarter.

The second technique—the search unit tracing and recording system (SUTARS)—was developed by Southwest Research Institute, San Antonio, Texas. The SUTARS is a manual scanner consisting of an automated ultrasonic data logger and a position recorder. The value of this system lies in being able to establish a base line from which changes can be discerned on subsequent inspections. The SUTARS was used to record base-line data on nine of the 10-in. Schedule 80 pipes prior to cracking. During this quarter, these pipes were reexamined, again using the SUTARS.

NEAR-SURFACE CRACK DETECTION IN NUCLEAR PRESSURE VESSELS

The belt-line welds in several U.S. reactors have suffered serious radiation damage, resulting from high copper content in the welds. In the event of an overcooling transient, it has been postulated⁽⁴⁾ that underclad cracks as small as 6.0 mm deep could result in vessel failure. A combination of conditions would be required for this type of failure to occur, including: existing cracks, high radiation damage, and high (approaching operating) pressure at low temperature. This generic safety issue is identified as pressurized thermal shock (PTS).

In order for NDE to provide confidence in the integrity of the vessel, it is necessary to demonstrate exceedingly high probabilities of detection for 6.0 mm and larger cracks. Cracks both parallel and perpendicular to the cladding, weld defects within the first 25 mm, and cracks resulting from cladding deposition are of interest. Most circumferential welds have been clad using the manual metal arc (MMA) process while longitudinal welds are clad by either the multiple wire or the strip cladding submerged arc process. While some areas are ground, very few would meet European standards for smoothness.

European techniques^(5,6) using 70° L-waves are effective in detecting 3.0 mm or smaller underclad cracks under ideal conditions (smooth cladding and cracks predominantly perpendicular to the cladding). It remains to be shown what inspection effectiveness can be achieved under the conditions present in older U.S. reactors for cracks running both perpendicular and parallel to the cladding.

Efforts are currently in progress by the NRC, the Electric Power Research Institute (EPRI), and ISI vendors and utilities to address this problem area. The objective of this work is to advise the NRC on the effectiveness of underclad crack detection for cracks of concern in the event of a PTS event. These data would then be used to evaluate the impact of ISI in reducing the failure probability under postulated PTS conditions. The evaluation of techniques for the detection and sizing of underclad cracks is divided into three phases:

- flaw detectability
- indication characterization
- technique effectiveness.

Flaw Detectability

Flaw detectability experiments have been performed on strip cladding, single-wire submerged arc cladding, and manual cladding. Both ground and unground surfaces were evaluated. The test blocks

used for this evaluation included a 750-mm diameter strip cladding dropout (see Figure 2), two 600-mm square blocks with strip and single-wire cladding with one side ground and the other as deposited,^(a) and two small blocks with ground and unground manual cladding. The dropout sample contained through-cladding notches as well as thermal fatigue cracks under the cladding. The two EPRI blocks contained notches under the cladding, and the two manual cladding samples each contained two reference reflectors and were used to evaluate the general noise level.

The measurements were taken using a 2-MHz dual beam longitudinal (SEL) 70° transducer with 10 x 15-mm elements and a focal crossover point of 17 mm. This unit was considered optimum for the cladding conditions and thicknesses (6 to 9 mm) tested. All measurements were performed manually.

Figures 3 through 7 show the relative signal amplitudes for: a 3-mm diameter flat bottom hole (FBH) at the interface, an ASME-type through-cladding notch, thermal fatigue underclad cracks, underclad notches, base-line noise, and indications from within the cladding. The base-line noise is the level that would yield a nearly continuous recording. Occasional indications arise from the cladding itself and are particularly prevalent when propagation is perpendicular to unground strip cladding at the overlaps.

The samples in Figures 3 through 5 each exhibit similar responses and good signal-to-noise (S/N) ratios. Calibration gain was within ± 2 dB for each of the three samples. The noise level in Figure 3 is slightly higher even though the cladding is the same thickness and was deposited to the same specifications as sample 2 in Figure 4. The S/N ratio for the ground single-wire cladding of Figure 6 is slightly

Figure 2. Strip Cladding Dropout Sample

(a) Access to these samples was provided by J. R. Quinn, EPRI.

Figure 3. Relative Amplitude Response for Sample in Figure 2. Strip cladding surface is ground smooth.

Figure 4. Relative Amplitude Response of Ground Strip Cladding Sample with Underclad Notches (sample provided by EPRI)

Figure 5. Relative Amplitude Response of Unground Strip Cladding Sample with Underclad Notches (sample provided by EPRI)

Figure 6. Relative Amplitude Response of Ground Single-Wire Submerged Arc Cladding Sample with Underclad Notches (sample provided by EPRI)

less, but still satisfactory. This single-wire submerged arc cladding has a bead width of ~ 12 mm. An additional gain of 6 dB was required for calibration of the sample in Figure 6 over that used in Figures 3 through 5. A 14-dB additional gain was required for the unground single-wire cladding of Figure 7; the S/N ratio for this condition is marginal for an automatic examination.

Indication Characterization

It would be desirable to design an automatic testing system that would record only defects of interest and ignore all other indications, however, this is not always possible, particularly for rough cladding and propagation perpendicular to the cladding. Characterization involves separating crack indications from innocuous cladding indications and crack sizing. In general, cladding indications peak earlier than underclad cracks. For cracks 6 mm and deeper, this difference is easily resolved if the approximate cladding thickness is known. In addition, the amplitude locus curve has characteristics that aid in this determination.

Crack sizing may be accomplished by the crack tip diffraction technique (see Figure 8). In the pulse-echo application of this technique, signals are received from the face of the crack and diffracted from the tip. Relative amplitude locus curves for 70° , 60° , and 45° L-waves are depicted in Figure 9. For 70° , the crack face reflection tends to overwhelm the tip-diffracted signal. For 45° , the face and tip signals are sharp and easily resolvable. While any of these angles can be used for sizing, 45° is consistently more reliable.

A 2-MHz dual 45° transducer with 10 x 15-mm elements and a focal crossover point of 25 mm was chosen for the sizing applications. This unit exhibits good near-surface resolution and good crack tip

Figure 7. Relative Amplitude Response of Unground Single-Wire Cladding Sample with Underclad Notches (sample provided by EPRI)

Figure 8. Crack Tip Diffraction Technique for Sizing Underclad Cracks

Figure 9. Relative Amplitude Locus Curves for Crack Tip Diffraction Sizing of Underclad Cracks

sensitivity. The results of these experiments are shown in Figure 10. Most of the data applies to notches; only four real cracks are included. However, many real cracks have been measured, and a measurable signal that corresponded to the expected depth has always been achieved. Unfortunately, only a few of these cracks have been destructively measured. A higher frequency and shorter focal length transducer may yield better resolution; however, satisfactory results can be achieved using this 2-MHz unit.

Technique Effectiveness

The effectiveness of techniques actually applied in the field will depend on a number of factors, including:

- calibration and recording sensitivity
- suitability of calibration standard (cladding type, thickness, roughness, and type of reflector)
- scan pattern increment
- contour following for immersion techniques.

For the highest reliability, it is recommended that the recording level be set no more than 6 dB above the persistent noise level and at least 6 dB below a 3-mm FBH (with face perpendicular to the clad-

Figure 10. Measured Versus Actual Depth for Cracks and Notches Measured by the Crack Tip Diffraction Technique (45° L-waves at 2 MHz)

ding). For the most part, this procedure will avoid the problem of cladding variability and provide a suitable sensitivity for smooth-ground manual cladding over circumferential welds. The scan pattern increment should be small enough for the shortest flaw of interest to exceed the recording threshold at least twice. The effectiveness of the technique should be demonstrated on realistic mockups covering the range of variables involved in the examination.

The effectiveness of optimized near-surface examination techniques can be very high for smooth-ground strip cladding, based primarily on S/N ratios for a small but representative selection of cladding conditions (see Table 2). To be considered optimized, a system should exhibit at least a 12-dB S/N ratio. An estimate of relative underclad crack detectability for optimized systems is also given in Table 2. S/N ratios may vary for other equipment and particular samples.

Conclusions

Near-surface defects (6 mm and greater) under strip-type cladding can be detected with a high degree of confidence. The effectiveness of crack detection through manual cladding remains to be demonstrated. These preliminary results are the first stage of an in-depth program that will specifically evaluate:

- underclad cracking methods for growing cracks
- different cladding types and surface conditions
- immersion techniques
- underclad crack sizing methods
- calibration procedures.

Table 2. Estimate of Relative Detectability of Underclad Cracks Deeper than 6 mm (optimized system)

Type of Cladding	Finish	Flaw Orientation	S/N Ratio for 6-mm Crack, dB	Detectability Confidence
Strip	Smooth	Perpendicular	18 to 24	Very high
Strip	Smooth	Parallel	16 to 24	Very high
Single-wire	Smooth	Perpendicular	16 to 22	Very high
Strip	Unground	Perpendicular	18 to 20	Very high
Single-wire	Smooth	Parallel	16 to 20	Very high
Strip	Unground	Parallel	14 to 18	Very high
Manual	Ground	Perpendicular and parallel	10 to 16	High
Single-wire	Unground	Perpendicular and parallel	10 to 14	Moderate
Manual	Unground	Perpendicular and parallel	6 to 10	Low

FUTURE WORK

The following areas will be emphasized during the coming quarter:

- complete manual evaluation of piping round robin data
- begin statistical analysis of piping round robin data
- draft piping round robin report
- complete fabrication of test matrix specimen No. 1 and UT for determining "best" underclad cracking method
- evaluate results of advanced techniques on the piping round robin specimens.

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FUTURE WORK

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EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK A - IRRADIATION EXPERIMENTS(a)

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

M. E. Cunningham
R. E. Williford

SUMMARY

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

Irradiation test IFA-431 is completed. Two other test assemblies (IFA-513 and IFA-527) were removed from the reactor in April 1981 due to fuel failures. IFA-527 was sent to Harwell, U.K., in March 1982 for examination; IFA-513 will remain in Norway for eventual disposition along with other NRC fuel. IFA-432 was removed from the reactor in June 1981 in preparation for shipment to Harwell for postirradiation examination (PIE). However, a proposal for continued high-burnup irradiation was accepted; and four rods from the assembly were reinserted. Preliminary fission gas release results for the two remaining rods, which were sent to Harwell in March 1982, are reported in this quarterly.

This program has assumed responsibility for IFA-518—a 12-rod U.S. Department of Energy (DOE) assembly that contains fuel rods with alternate fuel designs. These rods will be taken to medium or high burnup to assess the performance of the design alternatives. Irradiation of these rods has continued uneventfully throughout the quarter.

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 and rod design
- 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. 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 the 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:

(a) FIN: B2043; NRC Contact: H. H. Scott.

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

TECHNICAL PROGRESS

During June 1982, M. E. Cunningham traveled to the Atomic Energy Research Establishment (AERE)-Harwell, U.K., to discuss the PIE of IFA-432 and IFA-527 and to Vaaksy, Finland, to attend the Enlarged Halden Program Group Meeting. The Halden meeting is discussed under Task B; the status of the PIE at AERE-Harwell is summarized below.

- The nondestructive examinations (visual, neutron radiography, gamma scanning) of both assemblies have been completed. Visual examination revealed that the thermocouple and gas pressure transducer leads were damaged during transportation. Neutron radiography of rods 1 and 4 of IFA-527 revealed some slight anomalies in the pellet stacking arrangement but no immediately apparent cladding internal hydriding. Very little pellet fragmentation was evident. The axial gamma scans of rods 1 and 6 of IFA-432 revealed cesium peaks in pellet-pellet interfaces, which is commensurate with the high fuel temperatures these rods were known to have experienced.
- Destructive examinations have begun with the fission gas sampling of rod 6 of IFA-432. The remainder of the examinations (metallography, retained fission gas, density, burnup) are scheduled to be completed during fiscal year 1982.

Preliminary results from puncture and gas collection on both IFA-432 rods have been received and are reported under Task B. Gas pressures matched or slightly exceeded those measured in-reactor, indicating that gross gas leakage did not occur; however, there is evidence of preferential loss of helium, especially in rod 1. This evidence alone does not compromise the fission gas release estimates for the rods (5 to 6% for rod 1 and 19 to 20% for rod 6). Void volume measurements indicate net fuel volume reduction in both rods; but much more is indicated in rod 6 (which contains unstable, densifying fuel) than in rod 1 (which contains stable UO_2 fuel at 95% TD).

FUTURE WORK

Examination of IFA-432 and IFA-527 rods will continue.

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, preliminary postirradiation examination (PIE) fission gas release measurements were received, and the data reports for IFA-432 and IFA-527 were completed. In addition, a Pacific Northwest Laboratory (PNL) staff member attended the Enlarged Halden Program Group Meeting.

INTRODUCTION

The Experimental Support and Development of Single-Rod Fuel Codes Program is a continuation of the Experimental Support and Verification of Steady-State Codes Program (begun in 1974). This PNL program has the general objectives of 1) collecting and analyzing in-reactor data on fuel rod temperatures, fission gas release, and cladding elongation as a function of irradiation history; 2) correlating PIE data with in-reactor data; 3) utilizing ex-reactor testing for a better understanding of fuel rod mechanical behavior; and 4) 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 in-reactor fuel rod data. Specific areas of interest are analysis of data for inferring fuel relocation and its effect, use of transient temperature data to better understand fuel behavior, analysis of statistical variations and error propagation, and analysis of fuel rod fill gas pressure data for inferring fission gas release.

TECHNICAL PROGRESS

This quarter's activities are discussed below by subtask.

(a) FIN: B2043; NRC Contact: H. H. Scott.

SUBTASK B-1 - DATA PROCESSING

A tape was received from Halden containing data for IFA-432 and IFA-518 from December 1981 through April 1982. The data for the lower thermocouple of rod 5 of IFA-432 that were inadvertently left off the tapes received during 1981 were also included.

Preliminary results have been obtained from the fission gas sampling of rods 1 and 6 of IFA-432. Those results are:

	Rod 1	Rod 6
Initial void volume, cm ³	5.0	5.1
Final free volume, cm ³	5.56	7.24 ± 1.5%
PIE gas pressure at 0°C, MPa	0.44	1.22
Final in-reactor pressure at 0°C, MPa	0.423	1.04
Gas mixture:		
% helium	3.0	5.35
% krypton	12.2	15
% xenon	84.8	80.45
% hydrogen	0.2	0.09
Fission gas release, %	5 to 6	19 to 20

The agreement between in-reactor and hot cell pressure measurements indicates no great overall loss of gas; however, there appears to have been preferential loss of helium. It is not believed that this loss affects the fission gas release. The void volume changes correspond to a 1.0% fuel densification in rod 1 and a 4% densification in rod 6. These values are reasonable but do not seem to indicate any fuel swelling. Hopefully, the fuel density measurements to be made on these rods will clarify the question of fuel density/porosity variations.

SUBTASK B-2 - DATA REPORTS

Two data reports were completed during the last quarter. The first report—"End-of-Irradiation Data Report for the Instrumented Fuel Assembly (IFA)-527," NUREG/CR-2600—presents data collected during the irradiation of IFA-527 (July 1980-April 1981) and provides some analysis of the behavior of the assembly. The rods behaved with a high level of similarity during the first operating cycle, and all six rods had failed by the end of the irradiation. The unfailed temperatures were greater than those observed for rods irradiated to fission gas saturation (for example, rod 1 of IFA-432), and temperatures decreased when steam replaced the original xenon fill gas.

The second report—"Final Data Report for the Instrumented Fuel Assembly (IFA)-432," NUREG/CR-2567—presents centerline temperature, power, cladding elongation, and gas pressure data for IFA-432 from June 1980 through June 1981. Centerline temperature versus time figures that were corrected for thermocouple decalibration are included.

SUBTASK B-3 - DATA ANALYSIS

Computer Code Comparison

During June 1982, M. E. Cunningham traveled to the Atomic Energy Research Establishment (AERE)-Harwell, U.K., to discuss the PIE of IFA-432 and IFA-527 and to Vaaksy, Finland, to participate in the Enlarged Halden Program Group Meeting. The discussions held at AERE-Harwell are summarized in Task A. The principal topic of the Enlarged Halden Program Group Meeting was the comparison of four selected data sets and a number of fuel performance computer codes. PNL contributed FRAPCON-2 calculations for the first three cases: temperature and fission gas release as a function of

burnup; solid and hollow fuel temperatures and cladding diametral deformation; and thermal and mechanical behavior during and following a power ramp.

In general, codes that may be considered current (for example, FRAPCON-2, FEMAXI-3, HOTROD, FRP) did reasonably well in the comparison while older codes (for example, GAPCON-THERMAL-3) demonstrated known weaknesses. Comparison of FRAPCON-2 to the data indicated some need to improve effective fuel properties as a function of burnup.

Fission Gas Release in IFA-432

Rods 1, 5, and 6 of 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 preliminary rod puncture data are now available from the PIE of rods 1 and 6. These releases are being compared to calculations based on fission gas release models that are currently used in the FRAPCON-2⁽¹⁾ computer code.

Fission gas release calculations have been made using the FGASRL subcodes from MATPRO-10 and MATPRO-11. In making the calculations, the temperature and power histories were the same as those used to compare the experimental estimates to the ANS 5.4 and Beyer-Hann fission gas release model calculations.⁽²⁾ The results of the present calculations at 10, 20, and 30 GWd/MTU are compared with the experimental estimates derived from the pressure and PIE data in Table 1.

The experimental gas release estimates derived from the pressure and PIE data show good agreement for rod 1, but the PIE data for rod 6 indicate an ~30% higher release than estimated from the pressure data. This higher release estimate is caused by 1) the internal pressure being ~16% higher than indicated by the pressure transducer and 2) the internal free volume of the rod at 30-GWd/MTU burnup

Table 1. Comparison of Experimental Fission Gas Release Estimates from IFA-432 with Calculated Releases from the FGASRL Subcodes in MATPRO-10 and MATPRO-11

Rod Number	Burnup, GWd/MTU	Experimental Gas Release Estimates, (a) %	Gas Release from PIE, %(b)	Calculated Gas Release, %	
				MATPRO-10	MATPRO-11
1	10	2 to 4		5.9	0.9
	20	5 to 8		11.2	4.4
	30	4 to 6(c)	5 to 6	13.3	4.3
5	10	8 to 12		16.0	5.8
	20	6 to 9		17.1	4.1
	30	6 to 9		17.6	3.3
6	10	9 to 14		18.2	8.2
	20	9 to 14		22.6	8.9
	30	9 to 15(d)	19 to 20	23.8	7.9

(a) From in-reactor pressure data.

(b) Preliminary estimates and subject to possible change.

(c) End-of-life best estimate = 4.1%.

(d) End-of-life best estimate = 14.7%.

being ~8% larger than previously estimated. The reason for these differences has not yet been determined, but the forthcoming PIE results from fuel density and cladding deformation measurements are expected to help identify the cause of the volume discrepancy.

There are significant differences in the gas releases calculated by the two versions of FGASRL. FGASRL from MATPRO-10 shows reasonable agreement with the PIE data for rod 6 but overpredicts the experimental estimates for rods 1 and 5. Conversely, FGASRL from MATPRO-11 shows reasonable agreement with the experimental estimates for rod 1 but underpredicts the data from rods 5 and 6. Both versions of FGASRL are based on the MacDonald-Weissman fission gas release model. MATPRO-11 uses a two-step gas release process wherein fission gas is first released to grain boundaries and subsequently released to the internal free volume of the fuel rod. For IFA-432 fuel rods, only a small amount of gas was calculated to be retained at the grain boundaries and the large difference in gas release calculated by the two versions of FGASRL is primarily related to differences in the constants used for the MacDonald-Weissman gas release model.

FUTURE WORK

Data processing for the next quarter will continue as required. Work will begin on the final reports for IFA-432 and IFA-527. Fuel cracking and relocation analysis will concentrate on a sensitivity study of the effect of crack and gap widths and roughness and on applying the relocation model to higher burnups.

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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

Version 1 Modification 3 (V1M3) of FRAPCON-2 was released to the National Energy Software Center at Argonne National Laboratory (ANL). A preliminary version of the FASTGRASS fission gas release subcode (with microcracking) was received, and work was performed to link it to FRAPCON-2. Pacific Northwest Laboratory (PNL) is awaiting delivery of the final version of the FASTGRASS microcracking code, which will be included in V1M4 of FRAPCON-2.

The Harwell report on the mechanical compliance testing of rods 7 and 8 of IFA-432 was received, reviewed, and returned for final preparation. Work is nearly completed on a formal report describing the experimental verification of the U.S. Nuclear Regulatory Commission (NRC)/PNL cracked fuel mechanical model in FRAPCON-2. Reporting of advanced pellet-cladding interaction (PCI) models also began.

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 PNL and the 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 into 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: code maintenance, analysis of data from 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: B2043; NRC Contact: H. H. Scott.

SUBTASK C-1 - FRAPCON-2 MAINTENANCE

Version 1 Mod. 3 (V1M3) of FRAPCON-2, which contains coding modifications and corrections, was released to the National Energy Software Center. An updated Users' Letter accompanied the new code. One major correction was the inclusion of cladding creepdown in the FRACAS-II model, which previously failed to function due to a coding error. The effect of adding creepdown to the calculation of centerline temperature in a pressurized water reactor (PWR) rod is demonstrated in Figure 1. Another major change was to include densification/swelling in the free volume calculation in PELET/RADIAL, which influences the fuel moduli. The effect of that change is shown in Figure 2, where the fuel temperatures predicted by V1M3 rise in response to early-in-life densification.

V1M3 did not contain a new version of the FASTGRASS fission gas release submodel; the inclusion of this option was delayed to ensure compatibility of FRAPCON-2/V1M3 with FRAP-T6. This option will be included in V1M4, which will be completed when the final version of FASTGRASS with the micro-cracking features is delivered.

SUBTASK C-2 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

Harwell's report on the mechanical compliance tests for rod 7 (unirradiated) and rod 8 (irradiated) of IFA-432 was received, reviewed, and returned for final preparation. Harwell expects to deliver the report in late July.

Figure 1. Change in Centerline Temperature from Cladding Creepdown Calculated by FRACAS-II (standard PWR rod operating at a constant peak power of 50 kW/m)

Figure 2. Centerline Temperature for a BWR Rod Delineating the Difference Between the V1M2 and V1M3 PELET/RADIAL Options

Work is over 90% complete on a formal report to the NRC describing the experimental verification of the NRC/PNL cracked fuel mechanical model. This report describes the laboratory experiments that verify the model parameters (effective crack roughness, effective gap roughness, total crack length) and predicted results (reduced effective moduli for cracked fuel systems). Experiments were conducted on single cracks, crack systems, and cracked fuel-cladding systems.

Work also began on a report describing an advanced three-dimensional (3-D) PCI model, which will be used in FY 1983 to complete development of a fuel failure model. Some additional technical information was found in the literature that will help improve the definition of interfragment friction factors for the fuel conditioning/deconditioning models.

FUTURE WORK

The following activities are planned for the next quarter:

- issue the V1M4 version of FRAPCON-2, which will include the FASTGRASS microcracking model
- complete the report describing the experimental verification of the cracked fuel model
- complete the report describing the advanced (3-D) PCI model for cracked fuel
- begin implementing the 3-D PCI model (FRAGMT) into FRAPCON-2
- complete extension of the cracked fuel mechanical model to higher burnups.

Figure 2. Crankshaft Temperature for a BWR Rod Deforming the Difference Between the V102 and V101 Crankshaft Options

Work is now being completed on a formal report to the NRC describing the experimental verification of the NRC's cracked fuel mechanical model. This report describes the laboratory experiments that verify the model parameters (effective crack roughness, effective gas roughness, total crack length) and the model's predicted results (reduced effective modulus for cracked fuel systems). Experiments are conducted on single cracks, crack systems, and cracked fuel-cladding systems.

Work also began on a report describing an advanced three-dimensional (3-D) PCI model, which will be used in FY 1993 to complete development of a fuel failure model, some additional mechanical analysis, and to improve the definition of intervention action factors. Work found in the literature that will help improve the definition of intervention action factors for the fuel conditioning/conditioning model.

FUTURE WORK

The following activities are planned for the next quarter:

- Issue the V102 version of FRAPCON-3, which will include the FASTORAS neutron kinetics model.
- Complete the report describing the experimental verification of the cracked fuel model.
- Complete the report describing the advanced 3-D PCI model for cracked fuel.
- Begin implementing the 3-D PCI model (FRAPCON-3).
- Complete extension of the cracked fuel mechanical model to higher burnups.

**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
M. A. McKinnon

SUMMARY

The rod strain instrument was completed and transported to the loop facility for calibration and installation. Calibration is nearly complete, and the first experiment will be completed in late July.

Cladding preparations for the irradiated tests have resumed. Because of the unexpectedly high radiation levels in the samples (received in May), cladding preparation costs have increased. The test matrix has been adjusted accordingly.

INTRODUCTION

The primary objective of Task D of the Experimental Support and Development of Single-Rod Fuel Codes Program at Pacific Northwest Laboratory (PNL) 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 PNL. This experimental equipment has the capability for controlled power-ramping and load-cycling schemes and provides great experimental flexibility at a cost much lower than in-reactor experiments. The relationships between power-ramp rate, localized cladding strain rate, and fuel rod relaxation rate will be characterized. The localized cladding deformations will be measured by an instrument especially designed and built for this purpose.

The loop will be proof tested in fiscal year (FY) 1982 using unirradiated cladding followed by actual data collection with irradiated cladding. 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-plastic deformations of the simulated fuel rod at power within the loop will be designed and produced.

TECHNICAL PROGRESS

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

SUBTASK D-1 - ROD STRAIN INSTRUMENT

The rod strain instrument is designed to simultaneously measure two orthogonal cladding diameters and the fuel rod bow. Assembly of the strain gages and routing of the hard lines were completed, and the instrument was transported to the loop facility for characterization and calibration. After some electronics problems were overcome (noise, ground loops), instrument repeatability was found to be

(a) FIN: B2043; NRC Contact: H. H. Scott.

$\pm 5 \mu\text{m}$ (± 0.0002 in.). The calibration is nearly complete, and the first hot test on unirradiated cladding is scheduled for mid-July.

SUBTASK D-2 - LOOP EXPERIMENTS

The irradiated cladding samples were received at the hot cell in May; and, due to funding restorations, sample preparation for the irradiated cladding experiments has proceeded. Unexpectedly high radiation levels in the samples have substantially increased preparation costs; thus, the experimental matrix has been reduced to 5 or 6 samples (10 had been originally planned). Some capital expenditures were also required to equip the hot cell to handle the samples. A proof test using unirradiated cladding is scheduled for late July, and testing of the irradiated samples will begin in October.

FUTURE WORK

The following activities are planned for next quarter:

- Subtask D-1 will be completed.
- The proof test on the unirradiated cladding sample will be completed.
- A report will be issued documenting the instrument, loop facility, and the first experiment.
- Preparation of the irradiated cladding samples will near completion, and testing will begin on the initial samples.

PIPE-TO-PIPE IMPACT(a)

M.C.C. Bampton, Project Manager

J. M. Alzheimer

F. A. Simonen

SUMMARY

All specimens to be used for target specimens were nondestructively examined. Oil was selected as the pressurizing fluid for the hot tests, primarily for safety considerations. Testing was resumed, and the first two hot pressurized tests were completed.

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 test matrix has been developed; a system capable of accelerating the pipe has been built; and the test facility has been designed. The next phase of the program will encompass the 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

All of the specimens to be used as target pipes had at least the center 8 in. examined using various nondestructive examination (NDE) methods, and all of the pipes were examined using ultrasonic inspection. The inspections indicated that the pipes were defect free with the exception of three of the four 12-in. diameter pipes, which had several cracks on the outside surface. The cracks were sized for depth using an electrical resistivity gauge (ERG); the cracks with the largest ultrasonic response also had the largest ERG reading (estimated crack depth = 0.050 in.).

The four 12-in. and one 6-in. diameter Schedule 40 pipes were also inspected using magnetic particles. Indications on the 12-in. pipes were marked using white ink; no indications were present on the 6-in. pipes. Only one 6-in. pipe was examined with magnetic particles because no indications were found during ultrasonic inspection. A dye penetrant examination was attempted on a few of the pipes, but the coating used to protect the strain circles sealed any possible cracks and negated its usefulness.

Since no flaws were uncovered in the 3- or 6-in. diameter pipes, these pipes can be tested without regard to orientation. The 12-in. pipes with flaw indications will still be used; however, the flaws will be situated so that they will not be in areas of anticipated high tensile strains. Flaw locations have been marked for future reference should a rupture occur.

(a) FIN: B2383; NRC Contact: G. Weidenhamer.

At the midyear review, test temperatures, support configurations, pipe sizes, and possible pressurizing fluids were discussed. Due to the safety problems associated with testing using water-filled pipes at pressurized water reactor (PWR) conditions, an alternate pressurizing fluid was proposed. The concept of using an oil as the pressurizing fluid was generally accepted, at least for determining when a rupture would initiate. Further testing may be required with hot water to determine the extent of rupture. Sunflower seed oil was selected as the pressurizing fluid because it does not boil or burn at 550°F and it is inexpensive.

A comparison was made using sunflower seed oil and water at 550°F with the following results:

- **State:** At 500°F and 2000 psi, both the water and the oil are liquid. If the pipe would rupture, the water would flash to steam while the oil would remain liquid.
- **Density:** Over the range of available data (70 to 450°F), the density of vegetable oil is 0.90 to 0.94 times that of saturated liquid water.
- **Compressibility:** The compressibility of vegetable oil is about 0.005 vol% per atmosphere at room temperature while that of water is 0.0046. Data on the compressibility of oil at high temperatures were not available.
- **Sonic Velocity:** The speed of sound in the two fluids is within 10% for the range of data available.

Testing resumed using oil as the pressurizing fluid after adding a heating system to the pipe impact facility and some instrumentation modifications to permit testing at the higher temperature. One test from Group 2 and one test from Group 3 of the test matrix were completed. Group 2 tests involve 6-in. Schedule 40 pipe impacting 6-in. Schedule 40 pipe, and Group 3 tests involve 6-in. Schedule 80 pipes impacting 6-in. Schedule 80 pipes. Both tests were run with the target pipe pressures and temperatures at 2100 psi and 550°F, and the tip velocities of both swinging pipes were ~185 ft/s.

No ruptures of the pipe walls occurred, but deformation modes of the two target pipes were significantly different. The Schedule 40 pipe appeared to have undergone almost pure beam bending with very little reduction in the cross section. The Schedule 80 pipe had much more reduction in the cross section that was similar to but not as extensive as that observed for the unpressurized specimens. The internal pressure in the pipes did not increase by more than 50 psi during impact. All instrumentation operated satisfactorily during the tests with the exception of the strain gages, where the lead pulled loose due to the severe dynamic loading. The strain gage system is being modified for future tests. Test data are currently being evaluated.

FUTURE WORK

The following activities are planned during the next quarter:

- complete hot oil tests
- evaluate and analyze data
- develop predictive models.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

POWER BURST FACILITY SEVERE FUEL DAMAGE (SFD) TEST PROJECT(a)

R. L. Goodman, Project Manager

G. S. Allison	R. F. Klein	L. J. Parchen
L. R. Bunnell	R. K. Marshall	R. D. Tokarz
T. M. Fish	S. D. Miller	J. O. Vining
L. L. King	R. E. Page	

SUMMARY

The Pacific Northwest Laboratory (PNL)/Power Burst Facility (PBF) SFD-ST test train assembly was completed and delivered to the Idaho National Engineering Laboratory (INEL) on April 28, 1982, on a chartered DC-3 aircraft; the delivery date was 18 days ahead of schedule. A PNL representative participated in the final assembly and outfitting activities for the SFD-ST test train assembly from May 2 to 14, 1982, at INEL. No problems were encountered.

All of the hardware items for various component assemblies for the SFD-1 test train assembly have been fabricated and assembled. The PNL/PBF insulated shroud assembly was completed, and final assembly of the nine remaining instrumented fuel rods for the SFD-1 fuel rod bundle will be completed when the remaining high-temperature thermocouples to be used as fuel cladding inner diameter thermocouples and fuel bundle steam probe assemblies are delivered.

INTRODUCTION

This part of the Severe Core Damage (SCD) Test Subassembly Procurement Program—the PBF Severe Fuel Damage Test Project—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 to provide information on the postaccident coolability of damaged fuel rod clusters after small-break accidents.

FUTURE WORK

The SFD-1 test train assembly will be delivered to INEL by September 15, 1982.

(a) FIN: B2084-2; NRC Contact: R. Van Houten.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM POWER BURST FACILITY SEVERE FUEL DAMAGE (SFD) TEST PROJECT(a)

R. L. Goodman, Project Manager

G. S. Allison R. A. Klein L. I. Burdick
J. R. Bunnell R. K. Marshall R. D. Tokarz
D. M. Eick S. D. Miller J. O. Yining
L. L. King R. E. Page

SUMMARY

The Pacific Northwest Laboratory (PNL) Power Fuel Facility (PFF) SFD-27 test train assembly was completed and delivered to the Idaho National Engineering Laboratory (INEL) on April 28, 1983, on a leased DC-3 aircraft. The delivery date was 15 days ahead of schedule. A PNL representative participated in the final assembly and checkout activities for the SFD-27 test train assembly from May 2 to 14, 1983. At INEL, no problems were encountered.

All of the hardware items for each component required for the SFD-27 test train assembly have been fabricated and assembled. The PNL PFF installed around assembly was completed, and final assembly of the remaining instrumentation required for the SFD-27 fuel rod bundle will be completed. When the remaining high-temperature thermocouples to be used as fuel cladding under dynamic heat transfer and fuel bundle steam flow assemblies are delivered.

INTRODUCTION

This part of the Severe Core Damage (SCD) Test Subassembly Procurement Program—the SCD Severe Fuel Damage Test Project—includes the design, development of appropriate materials and supporting fabrication processes, and complete fabrication of two full instrumentation test train assemblies. A primary objective of the PFF work is to directly benefit the ESDOR program due to knowledge in the experimental objective, particularly for material development, instrumentation, and fabrication development. The program is designed to yield important experimental data related to fuel and cladding behavior during small-break accidents as well as to provide information on the postaccident coolability of damaged fuel rod clusters after small-break accidents.

FUTURE WORK

The SFD-27 test train assembly will be delivered to INEL by September 15, 1983.

(a) ENEC B2084-2, NRC Contract, R. Van Ikonen.

**SEVERE CORE DAMAGE TEST SUBASSEMBLY
PROCUREMENT PROGRAM
ESSOR FUEL DAMAGE TEST PROGRAM SUPPORT(a)**

E. L. Courtright, Program Manager
F. E. Panisko, Project Manager
J. W. Upton, ESSOR Site Representative

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) site representative at Ispra, Italy, is continuing to provide liaison and safety analysis support.

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 accidents (LOCAs). 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 12 small-break conditions in commercial pressurized water reactors (PWRs) and boiling water reactors (BWRs).

FUTURE WORK

The main effort on this project will be provided by the NRC site representative. Additional PNL support will be limited due to reduced NRC funding for SSTP support.

(a) FIN: B2372-1; NRC Contact: R. Van Houten.

CORE THERMAL MODEL DEVELOPMENT(a)

M. J. Thurgood, Project Manager

J. M. Kelly
T. E. Guidotti

SUMMARY

Work continued this quarter on the development of the hot bundle code. The containment code was assessed against data from the Hanford Engineering and Development Laboratory (HEDL) hydrogen migration standard problems and with CASP standard problem #2. The HDR reactor pressure vessel blowdown for tests V42-V44 has been completed, and the nodalization of the HDR containment for test V43 is nearly complete.

INTRODUCTION

The COBRA-TF^(b) computer code is being developed for the U.S. Nuclear Regulatory Commission (NRC) to provide better digital computer codes for assessing 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 water reactor primary system simulation capability that can model complex internal vessel geometries such as those encountered in upper head injection (UHI)-equipped PWRs
- 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 containment application capability.

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 permits each of the three fields to move with a different velocity. Thus, one can mechanistically treat a continuous liquid core or film moving at a low or possibly negative velocity from which liquid drops are stripped off and carried away by the vapor phase. This is an essential feature in the treatment of the hydrodynamics encountered during the reflooding phase of a loss-of-coolant accident (LOCA). 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 features flexible noding, which allows modeling of such complex geometries as slotted control rod guide tubes, jet pumps, and core bypass regions. These geometries cannot be easily modeled 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 uses 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.

(a) FIN: B2041; NRC Contacts: J. T. Han and T. Lee.

(b) COBRA-TF = coolant boiling in rod arrays-two fluid.

TECHNICAL PROGRESS

Progress made during the past quarter in each of the three program areas is reported below.

COBRA/TRAC SYSTEM SIMULATION CAPABILITY

A report on the best-estimate UHI simulation has been completed. The input model for the UHI system simulation with conservative evaluation model values for the heat generation rates and power factors has been completed and the steady state is being run. This simulation will be completed during the next quarter.

COBRA/TRAC: TOP QUENCH-FRONT MODEL

The fine mesh-rezoning model is the essential component of the COBRA/TRAC quench-front model. This model allows computation of quench-front propagation due to both radial heat transfer and axial conduction and integration of the resultant boiling curve over the hydrodynamic mesh cell to provide the phasic heat inputs for the fluid energy equations.

To correctly predict the quenching rate, the correct heat transfer coefficients must be specified for the fine mesh nodes. This is most critical in the transition boiling region (~1 to 2 cm in length for a fully developed quench front). Unfortunately, this is also the region of greatest uncertainty, especially for top quench fronts. There is no general consensus on an appropriate heat transfer correlation for this region of the boiling curve that spans all the conditions expected in a LOCA.

As a consequence of this correlation gap and to provide a continuous boiling curve, the method proposed by Bjornard and Griffith⁽¹⁾ has been adopted. The transition boiling heat flux (q''_{TB}) is given by:

$$q''_{TB} = \delta \cdot q''_{CHF} + (1 - \delta) \cdot q''_{min}$$

$$\delta = \left(\frac{T_{wall} - T_{min}}{T_{CHF} - T_{min}} \right)^2$$

where δ = fraction of wall wetted
 q''_{CHF} = critical heat flux (CHF)
 q''_{min} = heat flux at the minimum film boiling temperature
 T_{wall} = wall surface temperature
 T_{min} = minimum film boiling temperature
 T_{CHF} = temperature at the CHF point.

For bottom reflood, Griffith's modification of Zuber's⁽²⁾ pool CHF correlation was used to specify the CHF point as follows:

$$q''_{CHF} = \text{maximum} (0.2, 1 - \alpha) \cdot q''_Z \\ = q''_{NB}(T_{CHF})$$

where α = vapor volume fraction
 q''_Z = Zuber's CHF correlation, function of pressure only
 q''_{NB} = nucleate boiling heat flux from CHEN correlation.

Thus, q''_{TB} for bottom reflood is:

$$q''_{TB} = \text{maximum} (0.2, 1 - \alpha) \cdot \delta \cdot q''_Z + q''_{FB}$$

where q''_{FB} = film boiling heat transfer.

For top quenching, the void fraction can be very large (0.95 to 0.99) and yet still produce significant quench rates. Since the void fraction remains near unity, the $(1 - \alpha)$ modifier applies too restrictive a constraint upon the transition boiling heat flux. The proposed fix for UHI heat transfer simply removes the $(1 - \alpha)$ modifier for a region near the top quench front (the so-called sputtering region.) An exponential decay as a function of distance was employed in the UHI calculation:

$$\zeta = \exp [-0.299(\Delta Z - 1.2)]$$

$$\zeta = \text{minimum} [1.0, \text{maximum} (\zeta, 1 - \alpha)]$$

where ζ = exponential modifier
 Δz = distance below a top quench front (in.).

Thus,

$$q''_{TB} = \zeta \cdot \delta \cdot q''_Z + q''_{FB}$$

The transition boiling heat flux at a top quench front is not enhanced above the reasonable value of $(\delta \cdot q''_Z)$, but rather the amount by which it is decreased as a function of void fraction is reduced.

The large quenching rates present in the Westinghouse G-2 data are real and are primarily a function of the high pressure at which these tests were conducted. A marked difference in quenching behavior exists between the high-pressure tests (200 and 500 psi) and the low-pressure tests (20, 60, and 100 psi). In fact, at 20 psi, hardly any quenching occurs. The excellent data comparisons of COBRA/TRAC with the G-2 constant pressure tests, despite the extreme variation in pressure and quenching rates, attest to the reasonableness of this approach. Due to the proprietary nature of the G-2 data, no data comparison plots can be provided.

HOT BUNDLE CODE DEVELOPMENT

The hot bundle analysis concept is illustrated in Figure 1. First, a simulation of the primary system response to a designated transient would be performed with a best-estimate two-fluid system code. From this calculation, which employs a somewhat coarse nodding of the core region, the transient boundary conditions (upper and lower plenums) would be defined. Using these boundary conditions, several hot bundle transients using a much finer mesh in the core region may be run varying parameters such as local peaking factors. Local subchannel fluid conditions and rod temperatures will be calculated by COBRA-TF, and fuel rod analysis will be performed by FRAP-T6.⁽³⁾ An active link will permit feedback of the current fuel rod state (material properties, gap conductance, and cladding ballooning) from FRAP-T6 to the rod temperature and hydrodynamic solution algorithms of COBRA-TF and vice versa. This procedure will provide a hot bundle analysis capability for coupled thermal-hydraulic/rod deformation calculations.

Concurrent with this effort, a joint project with the FLECHT/SEASET program⁽⁴⁾ has begun. This cooperative effort between Westinghouse and PNL will produce a version of COBRA-TF capable of addressing Appendix K concerns about reflood heat transfer with flow blockages. As outlined in Figure 2, this program has three distinct phases:

- assess the ability of COBRA-TF to predict reflood heat transfer in an unblocked rod bundle (FLECHT/SEASET 21 rod—unblocked)

Figure 1. Hot Bundle Analysis Concept

Figure 2. Schematic of Westinghouse/PNL Flow Blockage Program

- incorporate flow blockage models and assess reflood heat transfer in blocked bundles (FLECHT/SEASET 21 rod—blocked)
- perform quantitative assessment of flow blockage heat transfer capability (with a fixed version of COBRA-TF) for the FLECHT/SEASET 163-rod bundle (blockage islands with bypass).

To accomplish these objectives, the following tasks are in progress:

- radiation model development - thermal radiation to fluid and structure
- turbulence model development - single-phase mixing and void drift formulation
- COBRA-FRAP linkage
- conduction model enhancement - addition of azimuthal conduction and sleeve blockage model
- flow blockage heat transfer models
- developmental assessment.

Radiation Model

A radiation heat transfer model is being developed for the COBRA-TF hot bundle code with the capability of performing subchannel and lumped channel radiation calculations. The model will include automatic view factor calculations for bundle geometries and three modes of radiation heat transfer: rod to rod, rod to vapor, and rod to droplet. The above features will be developed for both optically thin and thick participating media. To date, the coding for the three modes of radiation heat transfer for an optically thin participating media has been developed and is currently being assessed against GÖTA/and FLECHT/SEASET data.

Turbulence Model: General Electric (GE) Nine-Rod Tests

Simulation of the GE nine-rod tests⁽⁵⁾ was performed to assess single-phase and two-phase turbulent mixing models. Single-phase turbulent mixing is calculated using the traditional intersubchannel mixing coefficient approach, where for the *i*th subchannel:

$$W'_{ij} = \beta \cdot s_{ij} \cdot \bar{G}_{ij}$$

$$\nabla \cdot q^T_i = \sum_{k=1}^{NK} W'_{ik} \cdot (h_j - h_i) \cdot \Delta x$$

where NK = number of gaps connected to subchannel *i*.

A similar term appears in the axial momentum equations, but no terms appear in the mass equations since *W'* represents a fluctuating cross flow with no net mass transfer.

The two-phase mixing model adds the effect of lateral "void drift," which contains a mass transport term, to the fluctuating cross flow. A simple formulation based on the work of Lahey⁽⁶⁾ and Kelly⁽⁷⁾ is employed.

$$\left(\frac{\epsilon}{\bar{q}}\right)_{ij} = \frac{W'_{ij}}{[s_{ij} \cdot (\langle \bar{\rho}_i \rangle - \langle \bar{\rho}_j \rangle)]}$$

$$(G^T_V)_{ij} = \left(\frac{\epsilon}{\bar{q}}\right)_{ij} \left\{ [(\alpha \rho_v)_i - (\alpha \rho_v)_j] - [(\delta \rho_v)_i - (\delta \rho_v)_j] \right\} \text{EQ}$$

where the equilibrium void distribution is approximated by

$$[(\delta\rho_v)_i - (\delta\rho_v)_j] \text{ EQ} = \frac{(G_i - G_j)(\alpha_i + \alpha_j)}{\bar{G}_{ij}}$$

Terms accounting for the momentum and energy transfer associated with this mass transfer are added to the conservation equations.

The GE nine-rod test section consisted of a nine-rod bundle (3x3) with a uniform heated length of 6 ft and a development length of 4 ft. The rod diameter was 0.570 in., and the pitch-to-diameter ratio was 1.295. Rod position was maintained by 1/8-in. diameter steel pins located every 12 in. in the axial direction (the last set was 10 in. upstream from the measurement station). Both subchannel mass fluxes and qualities were isokinetically sampled at the exit plane. Figure 3 shows the one-quarter symmetry section (one corner, two-edge, and one central subchannel) model. Ten axial levels ($\Delta x = 12$ in.), constant flow, and enthalpy at the inlet and pressure at the outlet (1000 psia) were used. The spacer pins were modeled as loss coefficients that were adjusted to yield the correct total single-phase pressure drop.

For these preliminary data comparisons, a constant value of the mixing parameter, $\beta \approx 0.01$, was selected. The test conditions are listed in Table 1; the single-phase results, in Table 2; and the two-phase results, in Table 3. A good comparison with the measured mass fluxes for the single-phase tests (see Table 2) was achieved with a constant value of the mixing parameter ($\beta = 0.01$). However, it appears that this agreement is due more to a careful specification of the loss coefficients for the spacer pins than to a correct modeling of the turbulent mixing. The importance of the diversion cross flow due to the loss coefficients was illustrated in a debug run where no losses were input for the pins. For this case, the mixing coefficient was increased by an order of magnitude before a reasonable data comparison was seen.

The results of the two-phase mixing tests (see Table 3) indicate an improvement over the calculations without mixing. The trend for vapor to "drift" from the low mass flux corner subchannel to the higher mass flux regions is also shown. However, for the higher void fraction cases (tests 2G2 and 2E3), the mixing is severely underpredicted and the central subchannel vapor flow rate decreases. These deficiencies illustrate the need to enhance the mixing coefficient as a function of void fraction and develop more sophisticated equilibrium void distribution models.

Figure 3. Quarter Section of Symmetry for the GE Nine-Rod Tests

Table 1. Summary of GE Nine-Rod Test Conditions

Test Number	$G/10^6, \text{lb}/\text{ft}^2\text{-h}$	Power, kW	Subcooling, Btu/lb	Exit Quality
1C	0.990	0	504.6	—
1D	1.510	0	504.6	—
1E	1.970	0	504.6	—
2E1	1.080	1064	142.9	0.035
2E3	1.060	1064	29.1	0.215
2G2	1.080	1596	189.8	0.090

Table 2. GE Nine-Rod Single-Phase Data Comparisons

	G_{corner}	G_{edge}	G_{center}
Test 1C			
Data	0.701	0.939	1.150
COBRA-TF (no mixing)	0.682	0.932	1.144
COBRA-TF ($\beta = 0.01$)	0.748	0.941	1.120
Test 1D			
Data	1.095	1.441	1.690
COBRA-TF (no mixing)	1.046	1.427	1.737
COBRA-TF ($\beta = 0.01$)	1.159	1.441	1.690
Test 1E			
Data	1.620	1.910	2.190
COBRA-TF (no mixing)	1.377	1.864	2.258
COBRA-TF ($\beta = 0.01$)	1.523	1.884	2.197

Table 3. GE Nine-Rod Two-Phase Data Comparisons

	Corner		Edge		Center	
	G	X	G	X	G	X
Test 2E1						
Data	0.950	0.004	1.102	0.026	1.162	0.051
COBRA-TF (no mixing)	0.616	0.118	1.108	0.028	1.169	0.041
COBRA-TF ($\beta = 0.01$)	0.814	0.005	1.083	0.034	1.146	0.042
Test 2E3						
Data	0.965	0.160	1.081	0.185	1.126	0.249
COBRA-TF (no mixing)	0.616	0.301	1.072	0.191	1.157	0.223
COBRA-TF ($\beta = 0.01$)	0.722	0.239	1.003	0.323	1.245	0.201
Test 2G2						
Data	1.000	0.020	1.111	0.068	1.130	0.110
COBRA-TF (no mixing)	0.516	0.231	1.154	0.064	1.126	0.100
COBRA-TF ($\beta = 0.01$)	0.841	0.110	1.094	0.078	1.116	0.096

COBRA/FRAP Linkage

FRAP-T6 is currently running at Los Alamos National Laboratory. Results of standard problem #1 compare exactly with a microfiche copy of results sent with the transmittal tape from EG&G Idaho, Inc. A driver code is currently being used to call FRAP as a subroutine. COBRA creates a tape of fuel rod conditions as a function of time, and the driver code uses the tape to simulate an active link between COBRA and FRAP. Subroutines replaced by COBRA can be removed from FRAP to reduce core space and allow the codes to be actively linked. The driver code provides verification as unused subroutines are eliminated.

Conduction Model

An azimuthal conduction model has been added to COBRA-TF to divide any heater rod into as many as eight azimuthal segments. Conduction between each segment is computed explicitly. The model is currently being assessed.

Flow Blockage

Work is currently under way to specify a consistent set of models for both flow blockage and grid spacer heat transfer effects. Phenomena that will be included are:

- droplet breakup
- drop impingement heat transfer
- boundary layer separation and attachment
- grid spacer rewet
- two-phase enhancement of single-phase forced convection.

COBRA-NC Containment Application

The COBRA-NC code is being assessed against a variety of containment test data, including room pressurization, differential pressure between rooms, jet impingement loads, pressure wave loads, and hydrogen migration. During this quarter, code predictions were made of CASP standard problem #2, the HEDL standard problems on hydrogen mixing and distribution, and the HDR reactor pressure vessel blowdown.⁽⁸⁾ The input model for the HDR containment steam line break test V43 is nearly complete. A comparison with the CASP standard problem #3 is also in progress. CASP #2 and #3 are post-test data comparisons, and both tests have indicated a need for a better wall condensation heat transfer coefficient.

The HEDL test comparisons were done in a "blind" mode for both the horizontal and vertical jet tests. The code predictions compared favorably with the measured data from both tests. The code prediction for the horizontal jet indicated some oscillations in hydrogen concentration in the upper elevations that were not observed experimentally. The perturbation appears to be initiated in the vicinity of the jet and may be a result of an erroneous prediction of steam condensation. The cause of the perturbation will be investigated further. The fluid temperature prediction for the horizontal jet test was very good. Both the hydrogen concentration and temperature were reasonably well predicted for the vertical jet. The concentrations at the bottom of the test section were underpredicted, which was attributed to the fact that the full jet momentum was not modeled for the vertical jet.

FUTURE WORK

The UHI system simulation with evaluation model heat generation rates will be completed during the next quarter. Development and assessment of the hot bundle code will also continue. The COBRA-NC code will be used to perform blind test predictions of the HDR steam line break test V43.

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LOCA SIMULATION IN THE NRU REACTOR(a)

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SUMMARY

The 12-rod test cruciform for the fourth materials experiment (MT-4) was installed in the previously used MT-3 guard rod/shroud assembly. Several additional systems were incorporated in the test train to improve the quality and usefulness of the data acquired. Added features included an external fuel rod pressurization system, flux wires, a time domain reflectometry (TDR)-type liquid level sensor, and a differential pressure-type liquid level detector. An additional capability for detailed rupture zone mapping was added to the disassembly, examination, reassembly machine (DERM) post-test examination equipment. The MT-4 test was successfully conducted from May 25-26, 1982. Peak rupture strains of up to 95% were obtained during the test. Subsequent thermal-hydraulic testing using a full range of reflood rates and boildown conditions showed no degradation in cooling of the deformed fuel bundle.

INTRODUCTION

The Loss-of-Coolant Accident (LOCA) Simulation in the NRU Reactor Program is being conducted in the National Research Universal (NRU) reactor at Chalk River Nuclear Laboratories (CRNL), Chalk River, Ontario, Canada, by Pacific Northwest Laboratory (PNL). The program is sponsored by the U.S. Nuclear Regulatory Commission (NRC) to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length, 3% enriched light-water reactor (LWR) fuel rod bundle during 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 LWR fuel as a function of reflood rate and delay time before reflood starts. The program is composed of several thermal-hydraulic and cladding materials experiments. The initial thermal-hydraulic experiment (TH-1) was conducted in October 1980 and provided a data base for predicting the quenching characteristics of Zircaloy fuel rods under various reflood conditions. Since that time, several other experiments have been conducted:

- MT-1 (April 1981) used a pressurized cruciform of 11 test rods, 1 water tube, and 20 unpressurized guard rods. The delay time and reflood rate were selected to reach a peak fuel cladding temperature of 1144K (1600°F); 6 of the 11 rods ruptured.
- MT-2 (July 1981) used the same guard rod and shroud assembly as MT-1. One objective of the experiment was to perform a low-temperature—1089K (1500°F)—test using variable reflood rates. A test loop malfunction allowed higher temperatures than desired; 8 of the 11 rods ruptured.
- TH-2 (September 1981) used a new unpressurized 12-rod thermal-hydraulic test cruciform in the MT-1/MT-2 guard rod and shroud assembly. Temperatures were held above 1033K (1400°F) for up to 280 s.

(a) FIN: B2277; NRC Contact: R. Van Houten.

- TH-3 (November 1981) used the same test bundle as TH-2 with the addition of several thermocouples and a spray desuperheater. The experiment was performed just prior to MT-3 to develop more precise reflood rate control to obtain the cladding temperatures desired for the success of MT-3.
- MT-3 (November 1981) used a new test assembly with 12 pressurized test rods. All 12 rods ruptured during the test; and the average peak bundle diameter strain was 36%, which corresponds to an average flow area reduction of 68%. Fuel cladding temperatures were maintained above 1033K (1400°F) for 180 s.

TECHNICAL PROGRESS

The fourth materials experiment (MT-4) was successfully conducted in the NRU reactor at Chalk River, Ontario, from May 25-26, 1982. Peak rupture strains of up to 95% were obtained with the total time in the transient of ~1200 s. Peak bundle strains of 67% were obtained. All of the rods burst within 52 to 58 s of the initiation of the transient during the adiabatic heatup prior to the initiation of the reflood. The heat transfer coefficients during this period were between 1.5 and 2 Btu/h-°F-ft². During subsequent periods of the transient in alternate reflood and boildown phases, heat transfer coefficients of 1.0 and 20 to 50 Btu/h-°F-ft² were measured.

Several new systems were incorporated in this test train to improve testing capabilities and provide additional data. An external (out-of-reactor) fuel rod pressurization system was used for the first time. With this system, scoping thermal-hydraulic tests (1000°F maximum peak cladding temperature) could be conducted prior to the transient test using unpressurized fuel rods to validate the loop control parameters and establish set points for the transient test. The rods were then pressurized using the pressurization system located on top of the reactor. When the transient test was conducted, all pressurized fuel rods ruptured between 52 and 58 s, as had been predicted. The peak cladding temperature was 1155K (1620°F).

Flux wires were added in this test train to better define the flux profiles during the test. A PNL-developed TDR-type liquid level detector was also used (see Figure 1). The detector is designed for continuous measurement of the coolant liquid level during reflood. The performance of the detector was deemed outstanding by the test observers (see Figures 2 and 3). Indications of the changing liquid level, slugging, and entrained water are evident in the test data. An experimental differential pressure-type liquid level detector was also used for the first time. Preliminary data analyses are inconclusive as to its performance.

A new six-probe rupture zone mapping head was developed for the DERM post-test examination of the fuel rods. This system will allow more accurate location and characterization of the rupture zone of the fuel rod. It will be used on the MT-4 rods and for re-examination of the MT-3 rods.

The NRC and CRNL reached agreement in principle to extend their cooperative agreement for use of the NRU reactor for 4 yr after the September 1982 expiration date of the current contract.

FUTURE WORK

The MT-4 experimental results will be analyzed, and a test report will be prepared. A post-test review meeting will be held at PNL for sponsor and user groups.

Due to funding limitations, the previously irradiated rod test (MT-5) will be postponed. All efforts will be directed toward the coolant boilaway and damage progression tests: MT-6 through MT-8.

Attempts will be made to establish correlations between the MT-4 TDR liquid level data and other MT-4 sensors so that the actual liquid levels in prior tests can be more accurately determined.

Figure 1. Schematic of Time Domain Reflectometry-Type Liquid Level Detector

Figure 2. Time Domain Reflectometry Liquid Level Detector Display

MT4.04 5/27/82 23:32:28.000 5/27/82 23:58: 9.000

Figure 3. Time Domain Reflectometry Liquid Level Output Plotted with Reflood Rate

STEAM GENERATOR GROUP PROJECT(a)

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SUMMARY

Due to the group participation in this project, progress will be reported in a separate report.

INTRODUCTION

The Steam Generator Group Project (SGGP) is a continuation and expansion of the Steam Generator Tube Integrity Program (SGTIP): a multiphase, multitask laboratory program conducted at Pacific Northwest Laboratory (PNL). Under the SGTIP mechanically and chemically produced defects were placed in steam generator tube lengths to simulate service degradation. Specimens with defects were then nondestructively characterized and destructively tested to determine the remaining integrity under burst or collapse failure modes. Constitutive equations were subsequently established relating defect morphology and severity to remaining tube integrity. Other SGTIP objectives included studying the reliability and accuracy of nondestructive flaw characterization by eddy-current testing of steam generator tubes. Experiments were, and are currently being, carried out to determine the consequences of tube failure in terms of leak rate. Stability of through-wall flaws is a consideration in these experiments.

Models of remaining tube integrity that were developed using defect simulations during the SGTIP will be verified using actual service-defected tubing. To obtain the necessary specimens and to address increasing concerns on various other aspects associated with steam generator integrity, a retired-from-service nuclear steam generator was acquired. A generator^(b) removed from the Surry II nuclear plant (Surry, Virginia) after 6 years of service was judged suitable for this research.

Initial efforts on the Surry generator were concerned with licensing and transport activities to bring the unit from Virginia to Hanford, Washington. The unit was temporarily stored awaiting the completion of the specially designed containment facility—the Steam Generator Examination Facility (SGEF). The SGEF is equipped to allow both nondestructive examination (NDE) and physical sectioning of the generator and includes capabilities to perform chemical cleaning and decontamination.

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 the U.S. Nuclear Regulatory Commission's (NRC's) request, PNL has established interest from other parties to joined in program participation. Potential exists for research and development in chemical cleaning, decontamination, corrosion product identification, corrosion mechanism studies, and materials recovery under alternate

(a) FIN: B2097; NRC Contact: J. Muscara.

(b) 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.

sponsors. Several indications of intent to join with NRC in program sponsorship led to the formation of the SGGP. Research efforts on the Surry II generator will emphasize the following areas:

- validation studies of primary side nondestructive testing (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, in-service inspection (ISI) procedures, and waste handling
- defect matrix profiling and identification
- long-term operating effects of secondary side cleaning and primary side decontamination
- nondestructive ISI of the secondary side
- development and testing of innovative NDE devices and techniques
- demonstration of proposed repair techniques with assessment of reliability and safety aspects of the repair
- examination of decontamination and cleaning methods; and assessment of effectiveness, waste generation, and potential tube or support structure damage.

The generator will also become a source of specimens with service-induced flaws for use in various programs operated by SGGP participants.

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