NUREG/CR-2716
PNL-4275-1
Vol. 1

Reactor Safety Research Programs

Quarterly Report
January - March 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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NUREG/CR-2716
PNL-4275-1
Vol. 1
R3,4,5,GS

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Manuscript Completed: June 1982
Date Published: July 1982

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Prepared for
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FINs B2101, B2088, B2289, B2043,
B2383, B2084, B2372, B2041,
B2277, B2097

ABSTRACT

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from January 1 through March 31, 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.

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GRAPHITE NONDESTRUCTIVE TESTING (NDT) RESEARCH[a]

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SUMMARY

Electrical conductance measurements were made on a series of oxidized PGX samples, and a correlation was obtained between conductance and density. Combining this correlation with the correlation between compressive strength and density of oxidized PGX samples yields a correlation of strength versus conductance. This correlation provides an independent verification of the validity of the eddy current approach for monitoring the strength of high-temperature gas-cooled reactor (HTGR) core support blocks.

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 method for dry coupling 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.

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

TECHNICAL PROGRESS

EDDY CURRENT TESTING

The ZFIT program is set up to minimize the error between measured variables and the corresponding calculated variables in a set of eddy current measurements on laminated conductors. The same program can be used for calibration (varying unknown parameters to match measurements on known lamina) and conductivity profile determination (varying conductivities and thicknesses of layers to match measurements). The measurements consist of two voltages (V_x, V_y) that are, by circuit design, primarily related to the changes in search coil resistance (ΔR) and reactance (ΔX) from air values. Because ΔR and ΔX are calculable if laminar conductivities and thicknesses, coil geometry, and driving frequencies are known, we have tried to use measurements on "known" samples to determine the relationship of (V_x, V_y) to (ΔR, ΔX). Samples of a titanium alloy, 304L stainless steel, and graphite with oxidation levels from 0 to 11.8% were used as calibration standards. The reactance relations seem better determined than the resistance relations, and both seem better behaved at low frequencies (50 to 100 kHz) than at high frequencies (500 kHz to 1 MHz). The correlations are sufficient to confirm the theory but are insufficient to provide the precise calibration needed to allow deduction of subtle conductivity profile changes from the (V_x, V_y) measurements. Conductivities of the oxidized graphite slabs have not been measured independently; they have been calculated, assuming that the change in conductance is a linear function of weight loss during oxidation.

To better determine the change in conductance as a function of oxidation, nine bars (40 x 3.2 x 0.9 cm) were prepared and their conductivities were measured using the standard four-contact method. The electrical conductance (K) of these bars averaged 90.46 ±0.50 kS/m;[a] density (ρ) of the bars averaged 1.750 ±0.008 Mg/m^3; and the specific conductance (K/ρ) averaged 51.70 ±0.17 S • m^2/kg.

Eight of the nine bars were then oxidized at 875°C in a mixture of 80% CO_2 and 20% CO. The maximum weight loss ($\Delta W/W$) was 10.06%; ρ was 1.562 Mg/m^3; and K was 53.37 kS/m. A least squares fit of the data on the oxidized samples to the equation $K = K_o - B\ (\Delta W/W)$ yields K_o = 88.91 and B = 3.474, with a standard deviation (σ) of ±0.56 kS/m. The difference between the derived K_o and the measured K for the unoxidized samples (about 3σ) is highly significant and probably indicates that there is an initial rapid change, followed by the progressive change as a function of oxidation. Similar abrupt changes have been noted in other properties for graphite samples under neutron irradiation, oxidation, or heat treatment.

Although the correlation of conductance versus weight loss is quite good, we have previously observed that changes in other properties (strength, for example) correlate better with density than with weight loss. Table 1 shows compressive strengths measured on nine oxidized PGX cylinders. The strength of sample BA-1 ranks very well on the basis of density but not on the basis of weight loss.

The least squares fit of conductance versus density for the oxidized bars is shown in Figure 1; using the equation $K = A + B\rho$ yields A = -223.91, B = 177.40, and σ = ±0.88. The conductance at the average unoxidized density is lower than the measured average conductance prior to oxidation, again indicating that there is an initial rapid change at very small amounts of oxidation.

A least squares fit of the compressive strength (S) versus density for the data in Table 1 (using the equation $S = A + B\rho$) yields A = -92.00, B = 71.84, and σ = ±1.57. Like the equation for conductance, the calculated strength at the average unoxidized density is also less than the strength measured on the other unoxidized PGX cylinders. By inverting these two equations (to obtain density as a function of conductance and strength) and equating them to eliminate density as a variable, we obtain the relationship:

$$S = 0.4050K - 1.33$$

where S = compressive strength in MPa
K = conductance in kS/m.

(a) Siemens (S), the SI unit for conductance, is numerically equal to the traditional unit mho.

Table 1. Compressive Strength of Oxidized Cylinders

Sample No.	Density, Mg/m³ Original	Density, Mg/m³ Oxidized	Weight Loss, %	Strength, MPa
BA-3	1.744	1.710	1.95	31.67
BA-5	1.745	1.676	3.95	29.33
BA-16	1.759	1.642	6.65	23.02
BA-10	1.760	1.568	10.91	22.96
BA-11	1.760	1.569	10.85	21.29
BA-1	1.684	1.561	7.30	18.51
BA-8	1.739	1.544	11.21	18.44
BA-12	1.766	1.512	14.38	17.48
BA-14	1.754	1.511	13.85	16.11
Average:	1.746	1.588	9.01	22.09
$\pm 1\sigma$:	0.025			

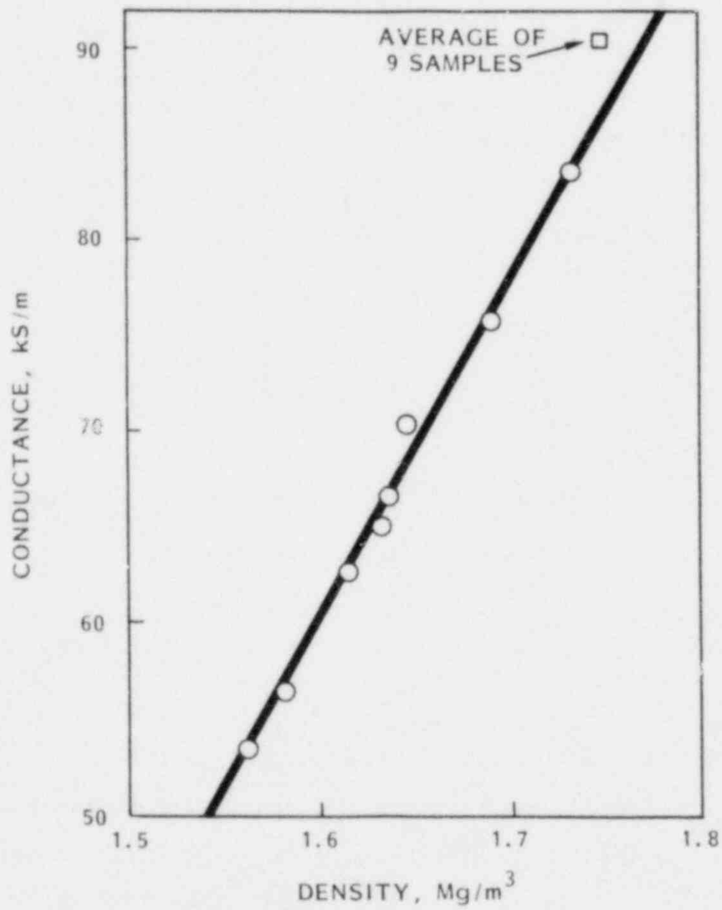

FIGURE 1. Conductance of Oxidized PGX Graphite

This procedure appears to result in a more valid correlation than the correlation against weight loss; however, its applicability to other logs of PGX graphite needs to be verified.

ULTRASONIC TESTING

Ultrasonic measurements are designed to determine the degree of oxidation that may exist within the volume of a material; eddy current measurements can only evaluate the surface of a material and a few millimeters below the surface. Although the technology for relating strength of graphite to ultrasonic velocity has been established and documented, current technology requires access to two parallel surfaces. Because the work at Ft. St. Vrain must be performed with single-side access, techniques to measure and interpret the backscatter ultrasonic energy reflected from voids and grain boundaries are being developed.

Some success has been achieved in correlating the pulse-echo Fourier spectral response with oxidation.[1] The echo from grain boundaries at different depths was analyzed, and the frequency spectral distribution of the echoes was correlated with oxidation or weight loss. Experimentation established that the approach, while technically valid, lacked the reproducibility required to be effective in an in situ reactor test. However, experimentation showed that discrete frequency measurements made with a two-transducer backscatter technique could provide an effective measurement and achieve the reproducibility needed for in-reactor tests. To establish the test frequencies to be used, a series of experiments were performed to determine the attenuation coefficients of graphite specimens as a function of both frequency and oxidation (see Figure 2).

A series of four PGX graphite blocks (3 x 3 x 1/2 in.) were oxidized to 1.8%, 6%, 8%, and 10%. These specimens were coated with a thin layer of lacquer to prevent water from migrating into the specimens during the attenuation coefficient measurements. The specimens were then placed in a holding fixture in a water-filled test tank, and through-transmission ultrasonic tests of attenuation were made. The ultrasonic test frequencies were 520, 770, and 960 kHz and 1.5 MHz. Scans were made along two orthogonal paths to insure that local variations in oxidation did not influence the test results. The test results clearly showed the relationships between oxidation and attenuation. The tests also identified both the test frequencies that would be most effective and the relationship between attenuation and oxidation at any given frequency. It is interesting to note that at 1.5 MHz the attenuation is so high that no effective test can be performed. The data obtained are significant in that the choice of frequencies, test procedure, and analysis requirements can be determined from the curves of Figure 2.

FUTURE WORK

EDDY CURRENT TESTING

- develop an in-field calibration algorithm and procedure
- obtain data using a new and improved laboratory eddy current instrument
- define the conductivity of test specimens more accurately
- determine test parameters such as effective liftoff and probe dimensions more accurately.

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

- determine the chemical state of impurities during oxidation
- obtain additional PGX graphite from other logs
- prepare strength and conductance samples for oxidation and testing.

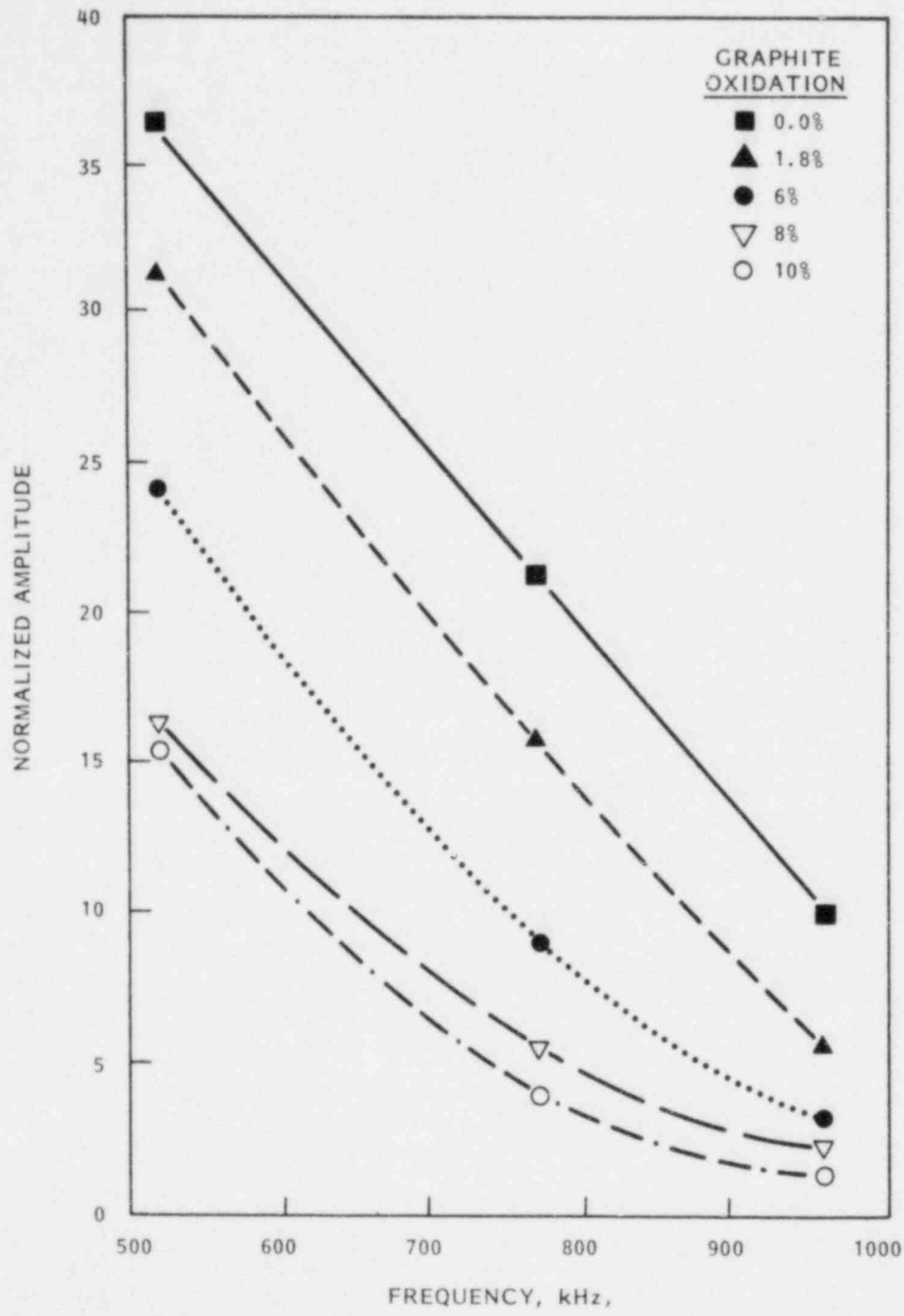

FIGURE 2. Frequency Versus Normalized Amplitude for PGX Graphite Specimens

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

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SUMMARY

Fabrication of the German ZB-1 test vessel has been completed, and all wave-guide acoustic emission (AE) sensors for monitoring the ZB-1 test have been characterized. The ZB-1 test is scheduled to begin in September 1982. AE data obtained from a small slag inclusion area during hydrotesting of the Watts Bar 1 reactor were analyzed and compared to an evaluation of the defect by ASME Code methods. Data obtained from testing irradiated and unirradiated 4T fracture specimens indicated little difference in AE between the two material conditions. Additional data will be obtained from unirradiated specimens for verification. Test specimens have been prepared for A106 ferritic pipe material for AE characterization testing in fatigue crack growth. A Type 304 stainless steel pipe specimen has been prepared for testing to provide AE characterization of stress corrosion cracking (SCC) in this material.

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. 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

OFF-REACTOR VESSEL TEST

The German ZB-1 test vessel was fabricated and shipped to the Institut für Zerstörungsfreie Prüfverfahren (IzfP) in Saarbrücken, West Germany, for pretest nondestructive examination. The vessel includes an A533B steel insert containing precracked machined flaws. During forthcoming fatigue tests, the vessel will yield AE information pertinent to both the A533B steel and the initial vessel material (22 NiMoCr37).

All wave-guide AE sensors for monitoring the fatigue crack growth testing have been characterized. During hydrotest monitoring, the wave-guide sensors will be supplemented by at least one array of commercial AE sensors used for vessel hydrotest surveillance. It is important to be able to relate hydrotest results directly to current commercial practice.

The ZB-1 test schedule has been delayed pending selection of the German team to monitor the test along with PNL. It now appears that the test will begin in August or September 1982.

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

REACTOR MONITORING

Analysis of results from AE monitoring of cold hydrotesting at the Watts Bar 1 reactor continued, and spontaneous AE was detected from the nozzle weld area. Based on results of the preservice nondestructive inspection (ultrasonic and radiographic) and the suggestions of TVA engineering personnel, it is believed that the AE came from a 1/4-in. diameter slag inclusion located near the midwall. This information together with AE data was used to make a first assessment of how the AE/flaw evaluation relationship compared to the accepted Code evaluation of a defect in an actual reactor structure. It is very important at this point to bear in mind that this assessment has **no** implication as to the correctness of Code evaluation procedures; it is totally a first look at how the AE/flaw evaluation relationships respond to actual field information.

Assuming that the flaw indication may be represented as a circular defect, then the stress intensity factor (K_I) during hydrotesting may be determined by the rules and procedures given in Section XI, Article A-3000 of the ASME Code. An estimate of the stress experienced in the nozzle blend radius during the hydrotest was obtained from Reference 1. Using analysis procedures, K_I equals 23 ksi$\sqrt{\text{in.}}$; using the AE/flaw evaluation relationship for hydrotest conditions, K_I equals 55 ksi$\sqrt{\text{in.}}$ Although the difference in the two values is not trivial when applied to crack growth calculations, it is encouraging that the AE/flaw evaluation relationship result is close to the Code value.

TVA has announced a delay of about 1 yr in the startup of the Watts Bar 1 reactor due to steam generator problems. At this point, there is no change in plans to apply AE monitoring even though the schedule is delayed; and installation of permanent leads for AE monitoring is now in progress.

IRRADIATED FRACTURE SPECIMENS

Data obtained from monitoring J-integral fracture tests performed at the Naval Research Laboratory (NRL) on irradiated 4T-CT specimens (originally reported in Reference 2) were further analyzed. Data obtained from unirradiated specimens that were contaminated with noise were also re-examined. It was concluded that data for the loading portion of the test sequence were adequate for a preliminary comparison of AE from irradiated and unirradiated material (see Figure 1). There does not seem to be a great difference in AE between the two material conditions; however, one of the remaining unirradiated test specimens will be monitored to verify this indication.

AE data measured from fracture testing the HSST V-7B and V-8 vessels are also shown in Figure 1. The much increased level of AE from a structure with a surface flaw compared to a laboratory specimen with a through-wall flaw is consistent with our experience, but a total explanation for the difference is still not obvious.

PIPE MATERIAL CHARACTERIZATION

The scope of this program for fiscal 1982 was expanded to include AE characterization of primary piping material. Three plate tension specimens with a 6 x 20 x 1-in. test section have been fabricated from A106 ferritic pipe material. The first of these will be tested in fatigue at room temperature to begin to build a comparison with AE/fatigue crack growth data generated earlier for A533B material.

In another phase of pipe material characterization, a 4-in. diameter, Schedule 80 Type 304 stainless steel pipe specimen to be installed in a loop will be stress corrosion cracked. The objectives are to 1) measure the AE profile for SCC and 2) record AE waveforms from SCC for characterization by pattern recognition techniques.

Since the specimen had undergone about 600 h of crack incubation time with high oxygen water and residual weld stresses present, it was inspected for indications of cracking. A 1/4-in., 2.25-MHz, 45° shear wave ultrasonic search transducer was used with a 10% notch reference. The base and

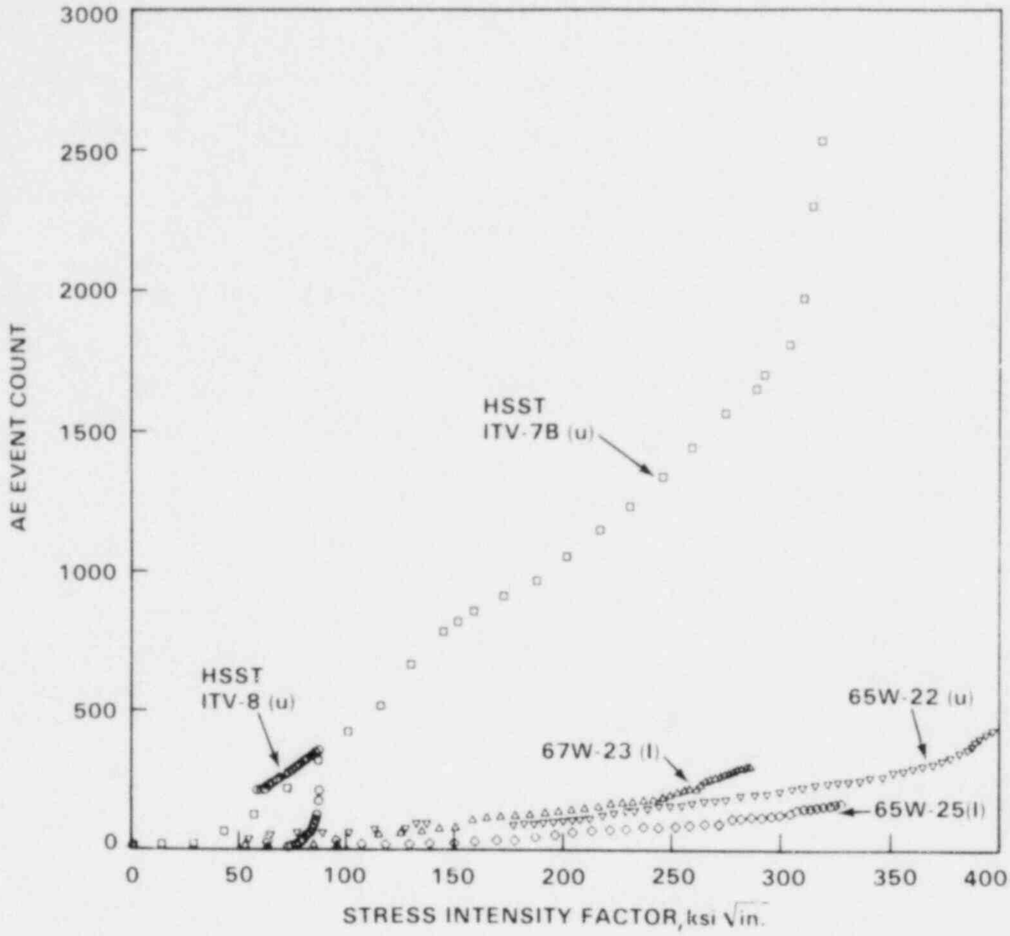

Figure 1. AE Event Count Versus Stress Intensity Factor for NRL Irradiated and Unirradiated Fracture and HSST Vessel Tests

weld metals were inspected from both directions in a normal manner and at a 45° angle to the pipe axis. The highest inside diameter (ID) signal was 17 dB below reference, and the other signals were greater than 20 dB below reference. It was concluded that no cracks were present yet. When cracking is initiated, the electric potential technique will be applied to attempt to monitor crack growth. The test will begin with an external load to 80% of yield.

REPORTS

- quarterly progress report for the period from October 1 to December 31, 1981
- paper titled *Acoustic Emission and Estimation of Flaw Significance in Reactor Pressure Boundaries* for the Conference on Inspection of Pressurized Components, London, U.K., October 12-14, 1981
- program midyear review summary.

FUTURE WORK

Plans for the period April 1 to June 30, 1982, include:

- perform pipe material evaluation tests
- complete installation of permanent signal leads at Watts Bar 1 reactor

- continue development of an engineering prototype monitor system
- issue final report on Watts Bar 1 cold hydrotesting
- visit the ZB-1 vessel test site at Mannheim, West Germany, to resolve final details of the vessel test procedure.

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2. Edler, S. K., ed. *Reactor Safety Research Programs Quarterly Report, October-December 1981.* NUREG/CR-2127, Vol. 4, PNL-3810-4, Pacific Northwest Laboratory, Richland, Washington.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[a]

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S. H. Bush, Project Manager
F. A. Simonen, Project Manager

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

SUMMARY

During the past quarter, the cracking of a 2-ft diameter strip cladding sample was completed; a radiographic evaluation of thermal fatigue cracks in cast stainless steel pipe was conducted; and an evaluation of two dissimilar metal welds was performed. Preliminary evaluation of the underclad cracks indicated that the cracking process was successful; that is, the cracks were close to their intended depths. The radiographic evaluation of thermal fatigue cracks indicated that ultrasonically detectable but tight cracks would not be detected by radiography. The dissimilar metal weld evaluation demonstrated the superiority of optimized longitudinal dual element transducers over conventional shear wave transducers in examining the complex dendritic structure of dissimilar metal welds.

INTRODUCTION

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

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

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

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

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

TECHNICAL PROGRESS

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

UNDERCLAD CRACKS

Cracking of the 2-ft diameter, 4.3-in. thick strip cladding sample was completed in late March 1982; and ultrasonic evaluation of the underclad cracks is now in progress. Detection evaluation of the cracks was performed using 2-MHz dual element 70° longitudinal contact transducers; each of the cracks was detectable. The cracks were sized using 2-MHz dual L-wave 45° and 60° transducers and the crack tip diffraction technique. The indicated depths were sufficiently close to the desired size. After the initial evaluation is completed, the starter notches in the cladding will be repaired; and the final evaluation will be conducted using contact and immersion search units.

RADIOGRAPHY (RT) EVALUATION

A limited investigation was conducted to determine the RT detectability of thermal fatigue cracks in the centrifugally cast stainless steel (CCSS) round robin specimens. Experimental RT data were needed to 1) assess the viability of radiography for this application since the French have indicated that they are using radiography for ISI of the reactor vessel nozzle-to-safe end welds in their pressurized water reactors (PWRs) and 2) provide independent measurements of the relative tightness of these thermally induced fatigue cracks. Both the round robin study and PNL laboratory tests have shown that these cracks are difficult to detect using either conventional or special ultrasonic techniques.

A limited investigation was conducted to evaluate the RT detectability of thermally induced, fatigue cracks using CCSS pipe specimens containing cracks with calculated depths ranging from 10 to 40% of the through-wall thickness (ultrasonically measured crack depths ranged from 12 to 51%). The experiment was designed to evaluate RT crack detectability with respect to an iridium-192 source location (panoramic and inside opposite wall location), source offset angle (in 2.5° steps), direction of offset (±), film type (AA, T, and M), position of control specimen within a nine specimen array, and effect of scatter from adjacent pipe and other surfaces. Both standard ASTM and European (DIN) wire penetrameters were used so that the results could be compared with other U.S. or European data, and a constant geometric unsharpness factor (U_g) was maintained.

The cracks proved to be quite difficult to detect by radiography, even when optimized (rather than conventional field) RT techniques were applied (for example, fine-grained film, long source-to-film distances). In fact, seven of the nine cracks were not initially detected at all; and of the two that were detected, one was detectable only over a very narrow range of technique variables as shown in Table 1.

Subsequent to these tests, two other CCSS pipe specimens with fatigue cracks (B-508 and B-509) were radiographed using optimized RT techniques. In this context, optimized radiography is defined as using a source-to-object distance of 27.5 in. and Kodak T film. The ultrasonically measured crack depths were 35% and 20% through the wall, respectively. These specimens had been made from a different pipe than those listed in the table above, and neither of the cracks could be detected by radiography.

All radiographs exhibited a **readily** visible sensitivity of 2-2T or better (many of the radiographs exhibited 2-1T sensitivity) for both the source side (#40) and the film side (#30) ASTM penetrameters. For comparison, European wire penetrameters (Type DIN FE-6/12) were also used; and at least four wires and often five wires were visible on all of these radiographs. The diameters of the smallest visible wire were either 0.5 or 0.4 mm for the fourth and fifth wires, respectively.

A ferritic steel specimen (B-402) with a thermally induced fatigue crack was also examined using the optimized RT technique. The estimated depth of this crack was 10%, and it was not detectable. This crack was easily detectable by a standard ultrasonic technique.

Table 1. Radiographic Detectability

			SOD(a) = 13.75 in.				SOD = 27.5 in.	
	Crack		Offset Angle					
Block No.	Depth, est. %	UT Depth, %	2.5°	0.0°	-2.5°	-5°	0.0°	0.0°(b)
B-520	10	12	—	—	—	—	—	—
B-517	10	16	—	—	—	—	—	—
B-528	20	23	—	—	D	D	D	—
B-515	20	28	—	—	—	—	—	D
B-521	30	28	—	—	—	—		
B-523	30	38	—	—	—	—		?
B-522	40	40	—	—	—	—		D
B-519	40	51	—	—	—	—		
B-504(b)	40	51	D	D	D	D	D	

(a) SOD = source-to-object distance.
D = detectable by radiography.
? = possibly detectable by radiography.
— = not detectable by radiography.
(b) After a pseudo stress-relief heat treatment (1900°F for 1 h and furnace cooled).

An isotope source (iridium-192) with a 0.125-in. focal spot size was used for all tests, and all film densities were maintained within the fairly narrow range of 2.5 to 3.5 density units. The following conclusions can be drawn from this study:

- Thermal fatigue cracks are tight.
- Radiography is adversely more sensitive to crack tightness than ultrasonic techniques.
- The use of radiography for ISI is questionable if tight cracks are expected.

DISSIMILAR METAL WELDS

Two dissimilar metal welded calibration blocks were borrowed from WNP #2, a nearby boiling water reactor (BWR) that is under construction. One block represents recirculation inlet safe ends; it is fabricated of 316L stainless steel welded to SA508 ferritic steel with Inconel buttering layers and 309 stainless steel weld metal. The outside diameter (OD) is 14 in., and the wall thickness is 1.25 in. The weld grain macrostructure is shown in Figure 1.

The other block represents recirculation outlet safe ends and is 1.87 in. thick with a 26-in. OD. Base, buttering, and weld metals are the same as for the other block; but the stainless steel side of the weld preparation is not buttered (see Figure 2).

Both blocks contained ID and OD notches at the weld root along with various side-drilled holes in the base and weld metals. Measurements were taken on these reflectors using 45° sound beams of both shear and longitudinal propagation modes. The shear transducer was a 2.25-MHz, 1/2-in. diameter pulse-echo unit, representative of the search units in most widespread use for pipe inspection except that the resolution was higher than usual. The dual longitudinal search unit operated in a pitch-catch mode at 2 MHz and focused (by roof angle) at a 1-in. depth in the metal.

The difference in performance was drastic. Whenever the sound path to a reflector passed through weld or buttering metal, the reflector was undetectable by 45° shear waves. The 45° L-wave unit, how-

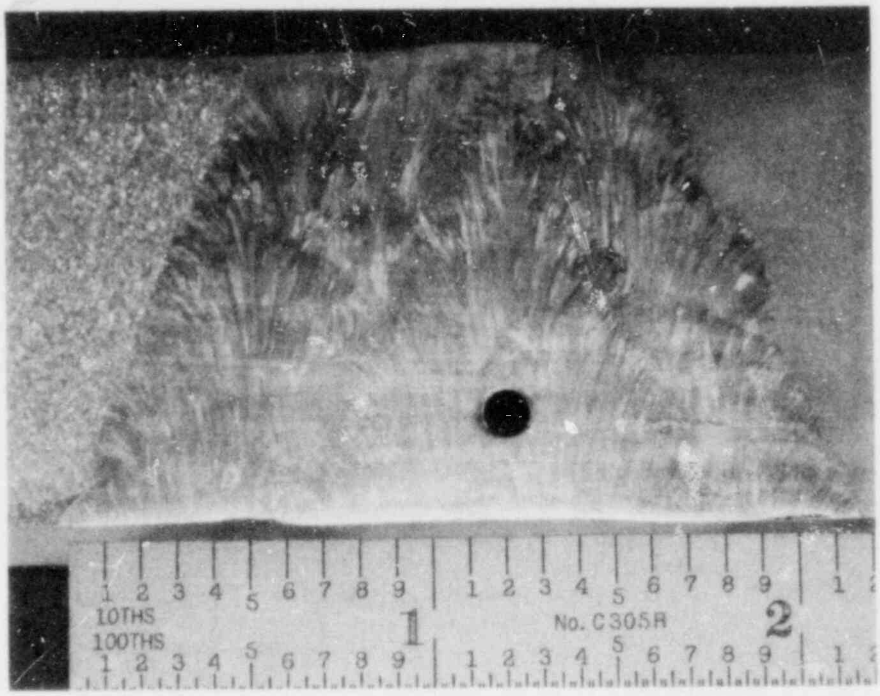

Figure 1. Inlet Recirculation Nozzle Dissimilar Metal Weld

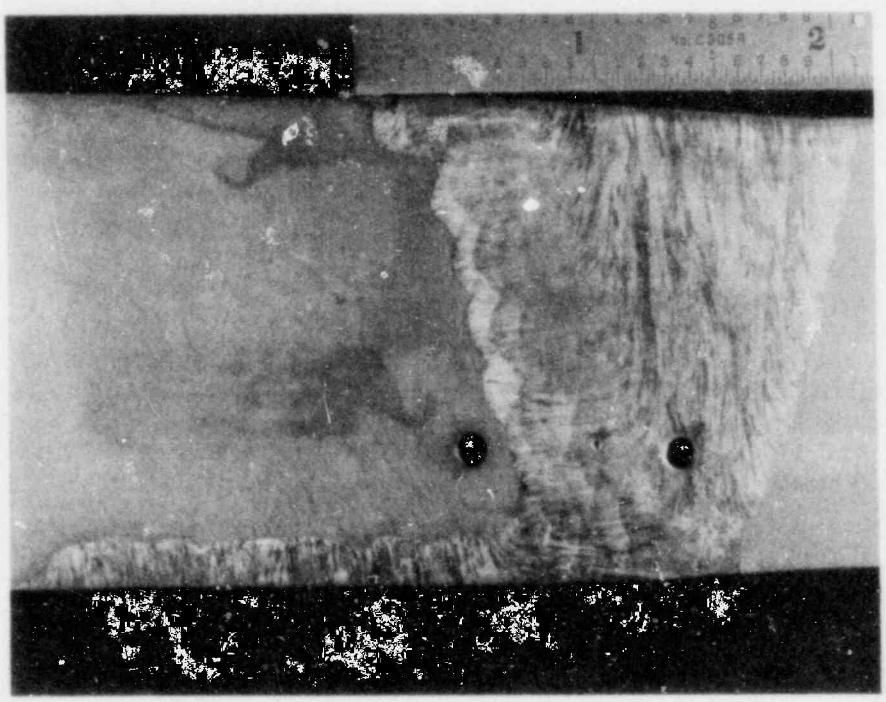

Figure 2. Outlet Recirculation Nozzle Dissimilar Metal Weld

ever, detected every reflector from both directions. The only difficulty encountered with the 45° L-wave unit came with the OD notch, when several signals appeared about the proper metal path and the one correct signal could not be identified. For the other reflectors, there was no unresolved ambiguity, although sometimes more than one signal was observed. This problem is common with refracted L-waves, but it is believed that the zone isolation "focusing" decreases the number of extra signals.

Using the 45° L-wave unit, calibration on the inlet nozzle block ID notch would be either 10 or 15 dB more sensitive than calibration on the 3/4t hole in the stainless steel base metal, depending on the beam direction. In the case of calibration from the stainless steel side (chosen because ISI will be performed from that side), the figure is 15 dB. For the outlet nozzle, the notch calibration is 7 dB more sensitive than the hole calibration.

For the shear wave unit, notch calibration for the outlet nozzle would be 14 dB **less** sensitive than the hole calibration. (The ID notch in the inlet nozzle block was not detectable.) There is a danger that an inspector could note the good ID notch signal and feel that the materials would not degrade the inspection significantly, while in fact the inspection would be valid only up to the first encountered weld metal and no further.

FUTURE WORK

The following areas will be emphasized during the coming quarter:

- complete short-term evaluation of underclad crack detectability
- fabricate further underclad crack samples
- complete evaluation of piping round robin data
- initiate improved and advanced technique evaluation of round robin samples.

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 have been reinserted. The remaining rods were sent to Harwell in March 1982.

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.

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 coordination of test design, test fabrication, shipping, PIE, and sample disposal. The test matrix now spans the full range of expected BWR conditions for pelletized UO_2 fuel, including:

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

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

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

TECHNICAL PROGRESS

Contracts between Halden, Harwell, and PNL were concluded for the examination of IFA-432 (rods 1 and 6) and IFA-527. The contract for IFA-432 now includes electron microprobe scans for radial distribution of fission gas as well as the more standard examinations such as gamma scanning, rod puncture, and profilometry.

Rods 1 and 6 of IFA-432 and the six IFA-527 rods were shipped from Halden to Harwell in late March. Radiography of rods 1 and 4 from IFA-527 revealed no hydriding of the Zircaloy cladding, which is remarkable considering that both rods operated in a failed condition for more than 1 month.

FUTURE WORK

Examination of the IFA-432 and IFA-527 rods will continue. Rod puncture results on the IFA-432 rods are expected in May.

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 substantial progress was made on data reports for IFA-432 and IFA-527. Data qualification efforts dealt with analysis of preliminary data received from the four IFA-432 fuel rods after the assembly was reinserted in the reactor and a comparison of data and analysis between Pacific Northwest Laboratory (PNL) and the Halden Project.

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 collecting and analyzing in-reactor data on fuel rod temperatures, fission gas release, and cladding elongation as a function of irradiation history; correlating post-irradiation examination (PIE) with in-reactor data; utilizing ex-reactor testing for a better understanding of fuel rod mechanical behavior; and integrating this information into the FRAPCON computer code series. The qualification and analysis of the data obtained from in-reactor testing of fuel rods is the responsibility of Task B, which has been divided into three subtasks:

- **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.

SUBTASK B-1 - DATA PROCESSING

Rods 2, 3, 5, and 9 of IFA-432 were reinserted in the Halden Boiling Water Reactor (HBWR) in December 1981. Some preliminary data from the current irradiation cycle have been received, and

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

several points were noted: First, because data had not been received since December 1980 for TF-8 (lower thermocouple of rod 5), it had been concluded that the thermocouple had failed. Data received during the current cycle showed that TF-8 was working again. Halden found that TF-8 had not stopped operating after December 1980 but that the data had not been placed on the data tapes. The missing data will be forwarded to PNL.

Second, prior to reinsertion of IFA-432 there was some limited concern as to what effect the removal of rods 1 and 6 from the assembly would have on the power calibration. A comparison of thermocouple and neutron detector readings from before and after the reinsertion has shown an apparent slight change in calibration (less than 5%). IFA-432, which is now located in a new core position, is now subject to a larger flux tilt across the assembly.

Finally, low-power pressure transducer readings from rod 5 were compared before and after the assembly restart (December 1981). The pressure after the restart increased from 1.32 to 1.67 MPa. No explanation has yet been found for this increase, but more recent data indicate that further pressure increases have occurred.

SUBTASK B-2 - DATA REPORTS

Data reports for IFA-432 and IFA-527 are currently being prepared for publication. The report for IFA-432 will cover the period from June 1980 to June 1981 and will be the final formal data report for the assembly although there will be a report discussing the PIE of rods 1 and 6. Data received from the four rods that are still being irradiated will be presented in future quarterlies. A draft version of the IFA-432 report has been completed and submitted for word processing.

The IFA-527 data report—"End-of-Irradiation Data Report for the Instrumented Fuel Assembly (IFA)-527"—presents the 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 technical review of this report has been completed, and it is being prepared for publication.

SUBTASK B-3 - DATA ANALYSIS

Efforts this past quarter have concentrated on data qualification because the Halden Project had requested a comparison of data. This comparison dealt with fuel temperatures from rods 1 and 3 of IFA-432 and gas pressure/fission gas release from rod 1 of IFA-432. In general, PNL and Halden data on the temperature/power histories for IFA-432 agreed; however, a discrepancy in calculated linear heat rate was uncovered. Since the rods in IFA-432 have different fuel masses because of different fuel dimensions and densities, it is necessary to apply a mass correction factor when calculating linear heat rates. It was found that Halden had been dividing by this factor instead of correctly multiplying (both labs use approximately the same factor). While this caused only a 2% difference for rod 1, it resulted in an almost 8% difference for rod 3. This discovery tends to account for prior observed differences between PNL and Halden data on the irradiation history of rod 3.

FUTURE WORK

Data processing for the next quarter will continue as required. The data reports for IFA-432 and IFA-527 will be completed, and camera-ready copies will be forwarded to NRC for publication. Fuel cracking and relocation analysis will concentrate on a sensitivity study of the effect of crack and gap widths and roughness and applying the relocation model to higher burnups. Fission gas analysis will compare the MATPRO-10 and MATPRO-11 gas release models to the data.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK C - CODE COORDINATION AND EX-REACTOR TESTING[a]

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

M. E. Cunningham
W. N. Rausch

SUMMARY

The link between FRAPCON-2 and the FASTGRASS (Modification 2) fission gas release model was completed. Version 1 Mod. 3 (V1M3) of FRAPCON-2 was delayed to insure that it was compatible with FRAPT-6 restart requirements. The microcracking version of FASTGRASS was received and will form part of FRAPCON-2 V1M4. A new gap conductance model by Dr. S. K. Loyalka showed negligible improvements and was not implemented in FRAPCON-2. The IBM version of FRAPCON-2 and extension of the cracked fuel relocation model to higher burnup were begun.

The three-dimensional (3-D) pellet-cladding interaction (PCI) code FRAGMT has shown that cladding axial elongations can vary appreciably over the length of a fuel rod. The Harwell compliance test data were analyzed; the results may impact allowable axial node lengths in fuel performance codes and may affect in-reactor fuel rod axial relaxation studies. Subtask C-2 experiment verification work was thus completed and reported at the Biannual International Atomic Energy Agency (IAEA) Fuel Modelers Meeting in Preston, England.

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

Task C efforts include: 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

The link between FRAPCON-2 and FASTGRASS Mod. 2 was completed.The decision was made to delay releasing this update because INEL found an error in the FRAP-T6/FRAPCON-2 link. The error was corrected, and an updated version of the code will be released to the National Energy Software Center as V1M3. Other changes to the program (some of which were covered in the August 1981 "Users' Letter") will also be incorporated. The new FASTGRASS link will be issued as V1M4 after additional work is completed to include the microcracking version of FASTGRASS that was recently received from Argonne National Laboratory.

Evaluation of a new gap conductance model from the University of Missouri (Columbia) was completed and was found to yield negligible improvement in gap temperature gradient and gap conductance predictions over the ranges applicable to fuel rods. The model was therefore not implemented in FRAPCON-2. Work also began to extend the PNL cracked fuel relocation model to higher burnups.

SUBTASK C-2 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

Work continued on developing a 3-D mathematical model (FRAGMT) to quantify the effects of asymmetrically cracked pellet fuels on localized cladding stresses. Fuel fragment size and shape effects on conditioning/deconditioning,[a] cladding bamboo ridges and axial elongation, axial gap formation between cracked pellets, and cladding plastic deformations are being modeled.

Recent results show that cladding axial elongation can vary appreciably along the length of the fuel rod, depending on the amount of fuel fragment deconditioning and the existence of axial gaps in the fuel column. Fuel fragment ratcheting was found to be very dependent on interfragment friction, and friction factors greater than about 0.5 tended to severely retard the reconditioning of fragments.

The Harwell compliance test data analysis was completed. These room temperature experiments were designed to simulate in-reactor fuel rod stress systems. Major conclusions were:

- Increased fuel cracking causes more uniform radial dissipation of strain energy over shorter axial lengths so that the maximum allowable axial node length for codes should be reduced.
- Radial-azimuthal-axial interactions for cracked fuel systems are not well understood and can affect fuel failure modeling.
- The viscous nature of axial slipping between cracked fuel and cladding is of sufficient magnitude to affect in-reactor fuel rod axial relaxation studies.
- The predicted reduction in effective elastic modulus for cracked fuel systems was confirmed.

The completion of the compliance tests represents the completion of subtask C-2 experimental verification efforts for FY 1982. These results were presented at the IAEA Fuel Modelers Meeting in Preston, England, March 14-19, 1982.

FUTURE WORK

The following activities are planned for the next quarter:

- issue the V1M3 edition of FRAPCON-2, which will include the changes listed in the "Users' Letter" and other correction, including the FRAPT-6 restart requirements
- continue work on an IBM-compatible version of FRAPCON-2
- complete a formal report to NRC describing the experimental verification results and the compliance test results

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(a) Conditioning refers to gap closure due to fuel pellet cracking.

- complete a formal report to NRC describing the 3-D PCI code FRAGMT
- begin implementing FRAGMT in FRAPCON-2
- complete the extension of the cracked fuel relocation model to higher burnups (20,000 MWd/MTM)
- present FRAPCON-2 comparisons to data at the Enlarged Halden Program group meeting at Lahti, Finland, June 1982.
- issue the V1M4 edition of FRAPCON-2, which will include the new microcracking FASTGRASS routines.

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 measurement instrument and the data acquisition system are about 50% completed; the loop piping and electrical modifications are about 90% completed. Two unirradiated samples have been prepared, and one is instrumented and ready for testing. Unirradiated experiments for fiscal year (FY) 1982 are scheduled to begin in the third quarter. The irradiated samples for FY 1983 experiments are expected to arrive in May.

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 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. 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. Methods for welding the high-temperature strain gages to the Zircaloy cantilevers were developed, and the welding operations were completed. Methods for routing the hard lines to the pressure vessel boundary were also developed. The instrument and its drive system are about 50% completed, as is the electronics and data-gathering system. Two strain gages that were broken during assembly were reordered, and delivery is expected in May. The first stages of the experiment will proceed with a 20% reduction in data acquisition.

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

SUBTASK D-2 - LOOP EXPERIMENTS

Reductions in funding have reduced the FY 1982 Task D scope to demonstrating the capabilities of the facility with two unirradiated cladding samples. The irradiated cladding samples are due to arrive in April and will be used in FY 1983 experiments.

Loop facility piping and electrical modifications are about 90% complete. Two unirradiated samples have been prepared, and one is presently instrumented and ready for testing. The pressure vessel preassembly is completed and awaits the instrument for installation in the loop.

FUTURE WORK

The following activities are planned for next quarter:

- The pressure vessel and instrument will be installed in the loop system.
- Two tests with unirradiated cladding will be completed.
- The irradiated cladding for future tests will be delivered to PNL.
- Data analysis and reporting will begin.

PIPE-TO-PIPE IMPACT[a]

M.C.C. Bampton, Project Manager

J. M. Alzheimer
F. A. Simonen

SUMMARY

One additional pipe-to-pipe impact test was run during the last quarter. Further testing was temporarily suspended to decide if possible flaws in the specimen should be characterized and if so which nondestructive examination (NDE) techniques should be used. The consequences of rupturing specimens under pressurized water reactor (PWR) conditions were given additional attention due to the potential safety hazards.

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 design of the test facility has been completed. 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

The first simply supported test from the matrix was completed using unpressurized specimens at room temperature. Compressive strains of over 55% and tensile strains of over 25% were present on the impacted pipe. Significantly more deformation and higher local strains were observed for the target pipe than for a similar energy level (impact velocity) with the rigidly backed test.

After testing, examination of the target pipe revealed several cracks on the outside surface in areas of relatively high tensile strain. Since it was not possible to determine the extent of the through-wall penetration of these cracks from surface examination, a portion of the pipe wall containing the most extensive crack was sectioned. Microphotographs of the wall cross section showed that the crack penetrated at a very shallow angle with the surface. The distance along the crack was ~ 0.040 in., but the maximum depth from the surface was only ~0.007 in. There was no indication that the crack would have penetrated the wall even if more pipe deformation had occurred.

A concern has been raised that the specimens should be nondestructively examined before additional testing to characterize any flaws that could possibly cause a rupture of the specimen. If the size and configuration of the flaw could not be determined after rupture, doubt would exist as to the actual cause of the rupture. It is not likely that any significant flaws exist in the walls in the region of impact;

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

A concern has been raised that the specimens should be nondestructively examined before additional testing to characterize any flaws that could possibly cause a rupture of the specimen. If the size and configuration of the flaw could not be determined after rupture, doubt would exist as to the actual cause of the rupture. It is not likely that any significant flaws exist in the walls in the region of impact; however, to erase any doubt about the validity of the data should a rupture occur, a series of NDE techniques have been proposed to locate and characterize any possible significant flaws. Surface-connected, longitudinal, and delamination flaws—the types of flaws that are thought to be due to the method by which the pipe is made—are being emphasized. The NDEs will include magnetic particle, dye penetrant on the outer surface, and hand-held ultrasonic scans normal and at an angle to the outside surface. The magnetic particle examination should detect longitudinal flaws of 0.5 in. or longer; the dye penetrant test should detect surface-connected cracks of 0.050 in. or longer; and the hand-held ultrasonic scan should detect delaminations of 0.25 in. or longer. The angled ultrasonic scan is intended to detect flaws 0.020 in. deep and 0.060 in. long and larger. Radiography of the specimen is not planned because it is most sensitive to through-wall cracks, which are not expected due to the manufacturing process. In addition, through-wall cracks of significant size should be detected by the angled ultrasonic scans.

SAFETY CONSIDERATIONS FOR TESTS CONDUCTED UNDER PWR CONDITIONS

Information received from Battelle-Columbus Laboratories indicates the potential for damage associated with the rupture of a pipe filled with water under PWR pressure and temperature conditions. Damage curves were presented that correlate well with the few actual test points and these predictions have been confirmed by independent analyses. It has been concluded that if pipes filled with water under PWR conditions are to be tested a much stronger containment system will have to be constructed and/or the test site will have to be moved. An alternate approach would be to use compressed gas to pressurize the pipe. The pressure waves associated with the rupture of such a pipe would be nearly as strong as for a water-filled pipe.

A third approach under consideration is to perform the tests using oil as the pressurizing fluid. An oil has been located that will not boil, ignite, or burn at 500°F if the pipe were to rupture. The compressibility of the oil is probably similar to that of water; however, no data at temperature are available. If oil were used and a rupture occurred, the pressure in the pipe would drop suddenly and would not propagate the rupture. For our purposes, a failure is any rupture of the pipe wall and the extent of the opening is not of concern.

It should be noted that not all safety-related piping in nuclear power plants is as hot as 550°F; emergency core cooling system lines, for example, may have standard operating temperatures of 150°F. It has not been determined that the most conservative tests are those at 550°F, and tests at lower temperatures are being considered.

SPECIMEN PREPARATION AND INSTRUMENTATION

All specimens have been prepared except for the welding of end caps and the application of strain gages, which were not added because of possible interference with the NDE.

The support load measuring system has been evaluated in several tests. The impulses as determined from the load/time histories are in good agreement with the momentum change of the swinging pipe.

ANALYSES

An ABAQUS-ND model was developed to examine the effects of temperature-induced variations in material properties. It was hoped that an assessment of the need to conduct tests at 550°F could be made. Although a working model was developed and run through several time steps, the effort was abandoned due to the extremely high costs that would be required to run the model to a significant deformation level for just one set of material properties.

Using the EPRI Elastic-Plastic Fracture Mechanics Handbook, J versus length curves were plotted for constant load conditions. With a superimposed J_R curve, a complete description of specimen deformation and crack growth behavior should be possible. J_R curves have been obtained for A106 Grade B steel; and further data regarding fracture, strain rate, and stress-strain properties as functions of temperature for A106 Grade B steel in Schedule 40, 80, and 120 are expected.

FUTURE WORK

The following activities are planned for next quarter:

- characterize any pipe flaws
- select pressurizing fluid and test temperatures
- continue testing
- analyze data and develop predictive models.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

POWER BURST FACILITY SEVERE FUEL DAMAGE TEST PROJECT[a]

R. L. Goodman, Project Manager

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

SUMMARY

Internal design reviews of the water fallback barrier assembly drawings for the Power Burst Facility (PBF) SFD-ST and SFD-1 test train assemblies were held at Pacific Northwest Laboratory (PNL), Richland, Washington, during January 1982. All test train design drawings (including the water fallback barrier assembly drawings) have been completed, issued as approved drawings, and released for fabrication, assembly, and outfitting activities.

The final formal design review of the entire drawing package for the two test trains was held in a joint PNL/EG&G Idaho, Inc./U.S. Nuclear Regulatory Commission (NRC) meeting in Idaho Falls, Idaho, on February 10, 11, and 12, 1982. The PNL presentations of the final design were well received and there were no major problem areas. A small number of action items for both PNL and EG&G resulted from the meetings.

Additional experiments using a full cross section insulated shroud mockup were conducted to test the proposed molten metal penetration detector (MMPD) for the PBF severe fuel damage (SFD) test train assemblies. To date the insulated shroud mockup has been through two heating cycles with a total time at temperature (molten Zircaloy) of about 10 min. Additional testing is in progress to accumulate ~1 h at temperature to further examine the behavior of both the insulated shroud and the MMPD.

Many hardware items for various component assemblies of the overall test train assembly have been fabricated and assembled. A total of 55 fuel rods (38 standard and 17 instrumented) have now been completed; 64 rods are required for the SFD-ST and SFD-1 test bundles. All of the 32 fuel rods for the SFD-ST fuel rod bundle have been completed and pressurized to 550-psi internal pressure. The fuel rod bundle for the SFD-ST test is now ~50% complete. The first two octagonal shroud inner liners were successfully formed from Zircaloy sheet stock, machined, welded, stripped from the welding mandrel, and leak checked. The PNL/PBF insulated shroud assembly for the SFD-ST test train assembly was completed; and work began on the insulated shroud assembly for the SFD-1 test train assembly.

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

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

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.

TECHNICAL PROGRESS

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

SHROUD DEVELOPMENT TESTING

Recent testing of the insulated shroud addressed the question of how well the insulator structure would resist attack by molten Zircaloy if a large portion of the cladding should melt. Previous test results[1] indicated that the sheathed thermocouple used as a MMPD in the simplified shroud insulation penetration testing showed an increase in the thermal conductivity of the shroud insulation as it was consumed by the molten Zircaloy in successive heats. The results of the simplified arc melt tests also indicated that the thermocouple junction reached a very high temperature (900°C) during the testing but then apparently attained equilibrium since the temperature was absolutely constant for about 1.5 min. Apparently, the Zircaloy was cooled sufficiently by the copper hearth to solidify just above the MMPD.

For the final and most prototypic of the liquid metal penetration tests, a short (~10.2-cm) full cross section insulated shroud was built. Figure 1 shows a schematic sectional view of the shroud, and Figures 2 and 3 are photographs of the insulated shroud. In Figure 3 the outer round shroud liner is in place around the support saddle halves, but the MMPD cable wrap and the second outer round shroud liner that surrounds and contains the cable wrap are not shown.

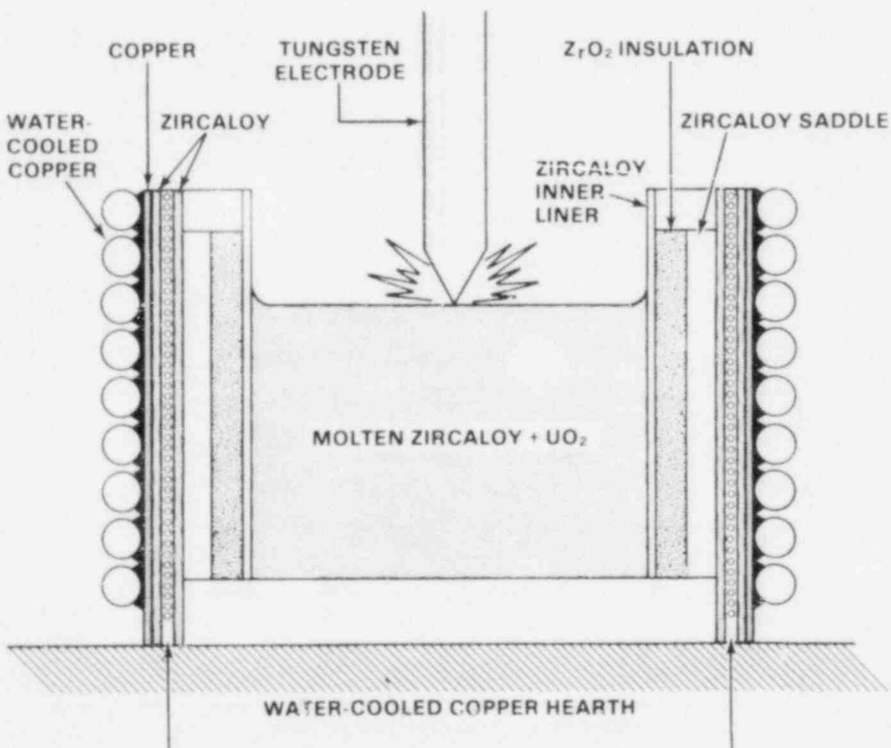

Figure 1. Full-Section Shroud Molten Metal Penetration Test (sectional view)

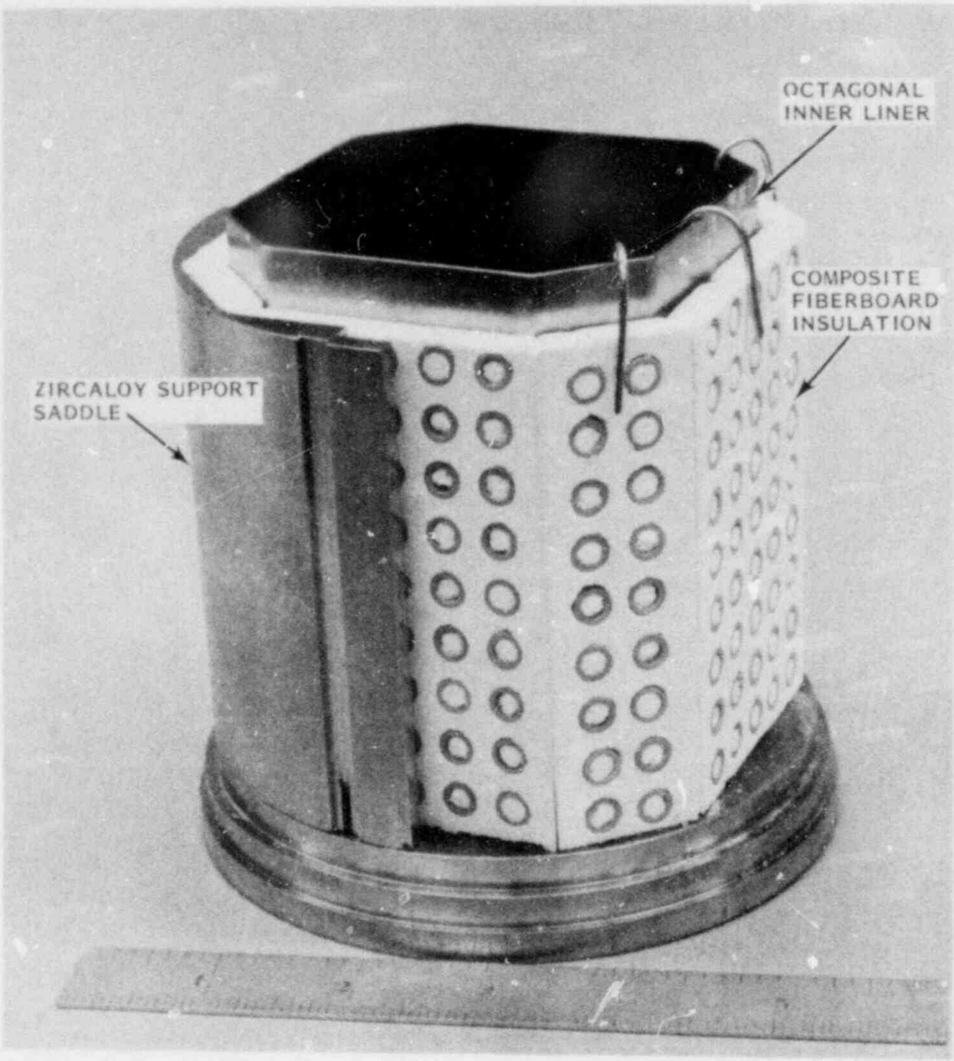

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Figure 2. Component Parts of Full Cross Section of PNL/PBF Insulated Shroud for MMPD Testing

After final assembly of the full-section shroud, a molten Zircaloy/20% UO_2 alloy was poured into the central shroud cavity and allowed to cool. An electric arc melter then supplied power to the central cavity to obtain and maintain a molten Zircaloy/UO_2 mixture. The goal of the testing was to maintain the central portion of the cavity in a molten state for about 1 h or until there was a failure indication from the MMPD.

To date the insulated shroud has been through two heating cycles with a total time at temperature (molten Zircaloy) of ~10 min. During the first heating, the electric arc melter supplied power to the central cavity on five separate runs. The power and time at power were gradually increased to a maximum of 22.5 kW for 1 min. Approximately 75% of the top surface was molten, and post-test radiography determined the depth of the melt. A thermocouple placed at the outer surface of the insulation indicated a maximum temperature of 300°C. The MMPD showed no change in resistance during the first heating cycle. By appearance, the molten Zircaloy partially melted a portion of the inner shroud liner during this test.

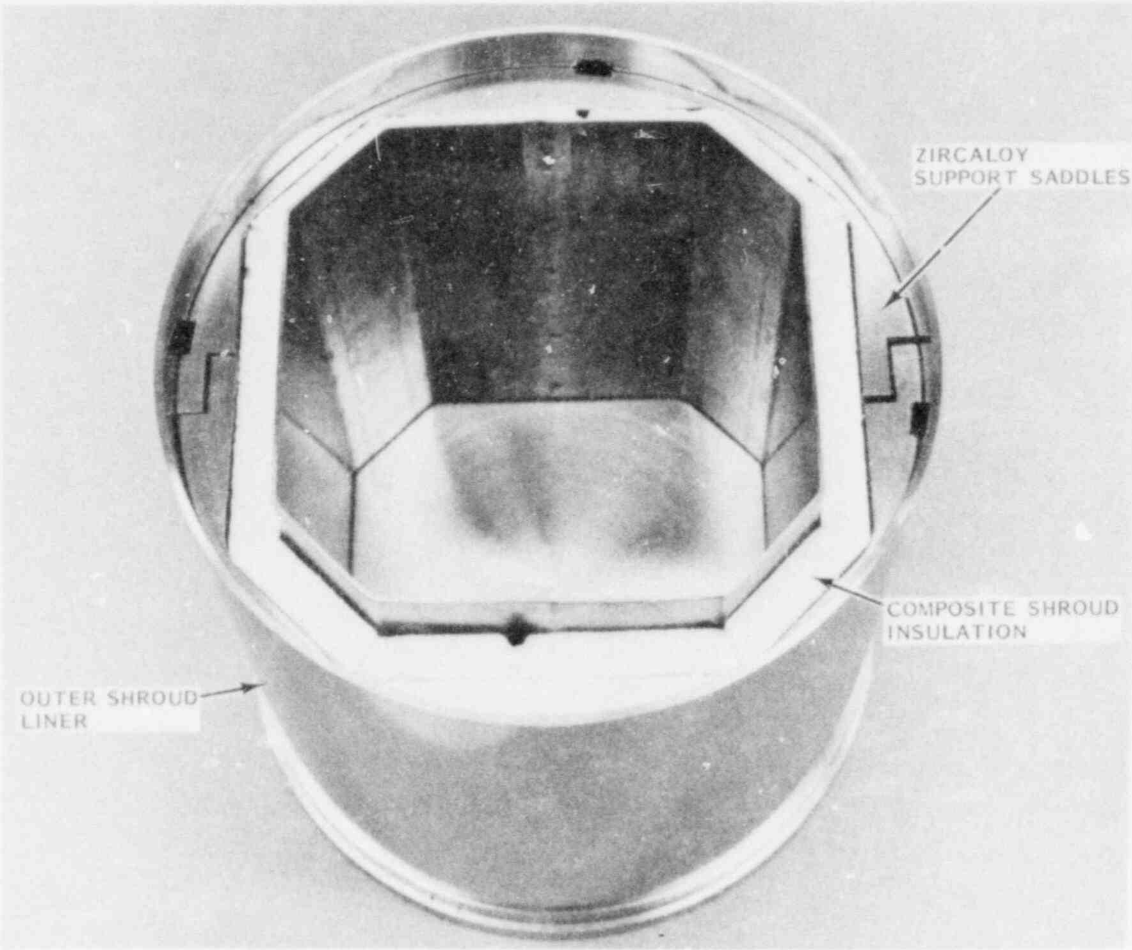

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Figure 3. Full Cross Section of PNL/PBF Insulated Shroud for MMPD Testing

The second heating cycle of the insulated shroud brought the total time at temperature to ~10 min. Post-test radiographs indicated that ~3.8 cm of Zircaloy and UO_2 were melted and that the molten metal substantially penetrated the insulation in a 90° sector of the shroud cross section. In preliminary tests previously conducted on a smaller cross section of shroud insulation, the metal froze at this point because of the cooling ability of the water jacket around the hearth plate. For the full cross section insulated shroud, however, it will be necessary to accumulate more time at temperature to determine this behavior. To date, the MMPD has shown no measurable change in resistance or continuity during the testing. Testing will continue to hopefully accumulate ~1 h at temperature for the central shroud cavity or until there is a failure indication from the MMPD.

TEST TRAIN ASSEMBLY COMPONENT HARDWARE

Many hardware items for various component assemblies of the overall test train assembly have been fabricated and assembled. A total of 55 fuel rods (38 standard and 17 instrumented) have now been completed out of a total of 64 required for the SFD-ST and SFD-1 test bundles. All of the 32 fuel rods for the SFD-ST fuel rod bundle have been completed and pressurized to 550-psi internal pressure. The fuel rod bundle for the SFD-ST test is now ~50% complete. The first two octagonal shroud inner liners were successfully formed from Zircaloy sheet stock, machined, welded, stripped from the welding mandrel, and leak checked. The PNL/PBF insulated shroud assembly for the SFD-ST test train assembly was completed, and work began on the assembly for the SFD-1 test train.

Figures 4 through 7 are photographs of the actual component subassemblies for the overall insulated shroud assembly that will be used for the PBF test program. The sequencing of the figures shows the approximate order of component fabrication and assembly. Figure 4 shows the upper Zircaloy flange subassembly for the shroud. The Zircaloy-to-stainless steel transition pieces through which instrument hard lines are routed and brazed are shown on the right side, and the transition from an octagonal inner shroud liner to a round tube that connects to the shroud upper flange is shown on the left. Figure 5 shows the ~2.4-m long octagonal inner shroud liner welded to the shroud upper flange. In Figure 6, the MMPD cable wrap is being wound onto the inner round shroud liner. After the cable wrap was installed (~91.4 m for each of four cables), the outer round shroud liner was welded to the top and bottom flanges of the insulated shroud assembly. The completed PNL/PBF insulated shroud assembly is shown in Figure 7. A total of 20 instrument hard lines and two pressure equalization/sensing lines are routed through the transition pieces on top of the shroud and brazed in place.

FUTURE WORK

Fabrication and assembly of component parts for the SFD-ST and SFD-1 test train assemblies will continue. The assembly and outfitting of the test train assembly for the SFD-ST test will be completed, and the test train assembly will be shipped to INEL in early May 1982.

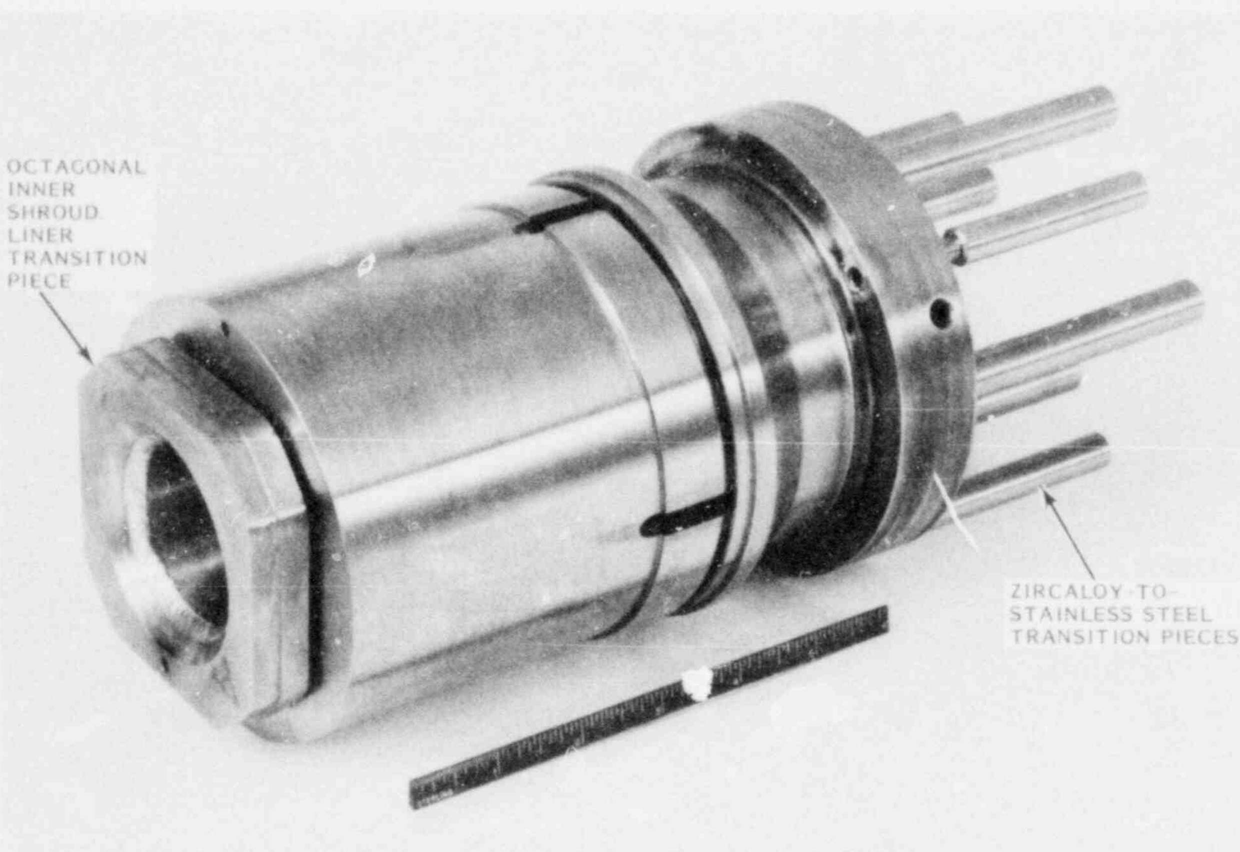

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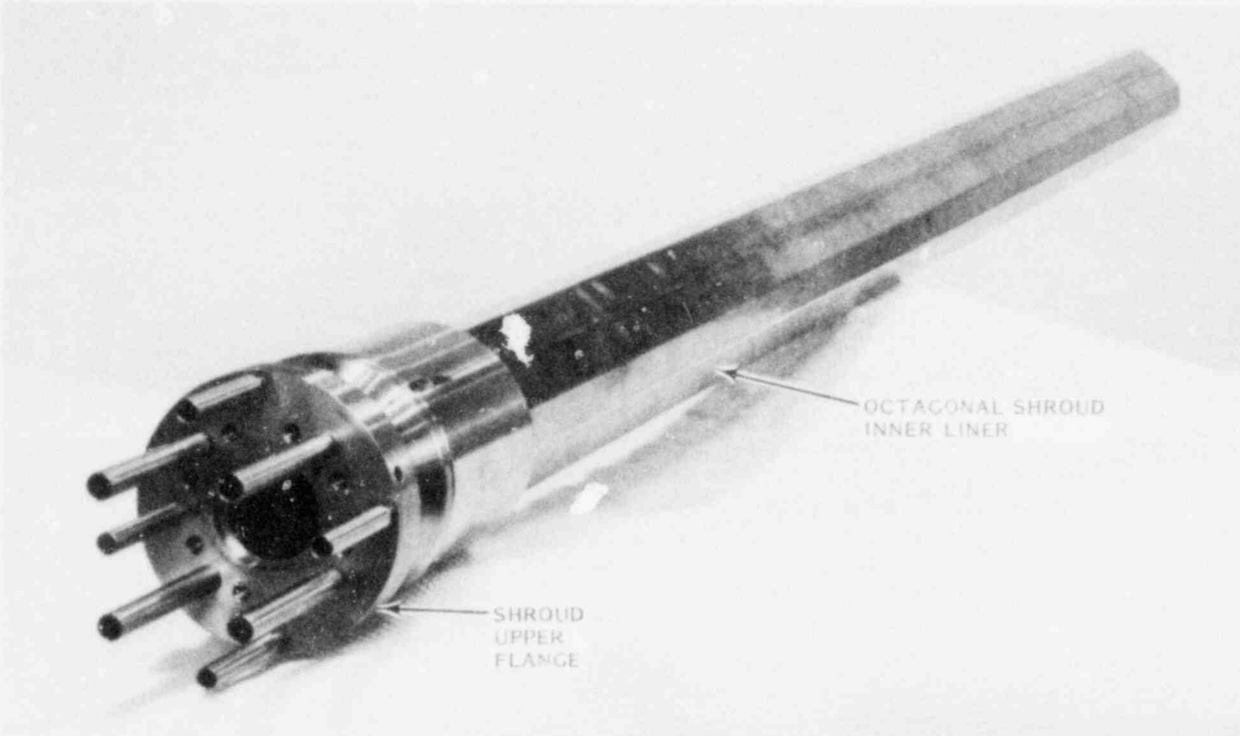
Figure 4. PNL/PBF Insulated Shroud Upper Flange Assembly



Neg 8200430-4

Figure 5. Insulated Shroud Octagonal Inner Liner Welded to Upper Flange

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1. Edler, S. K., ed. March 1982. *Reactor Safety Research Programs Quarterly Report, October-December 1981.* NUREG/CR-2127, Vol. 4, PNL-3810-4, Pacific Northwest Laboratory, Richland, Washington.

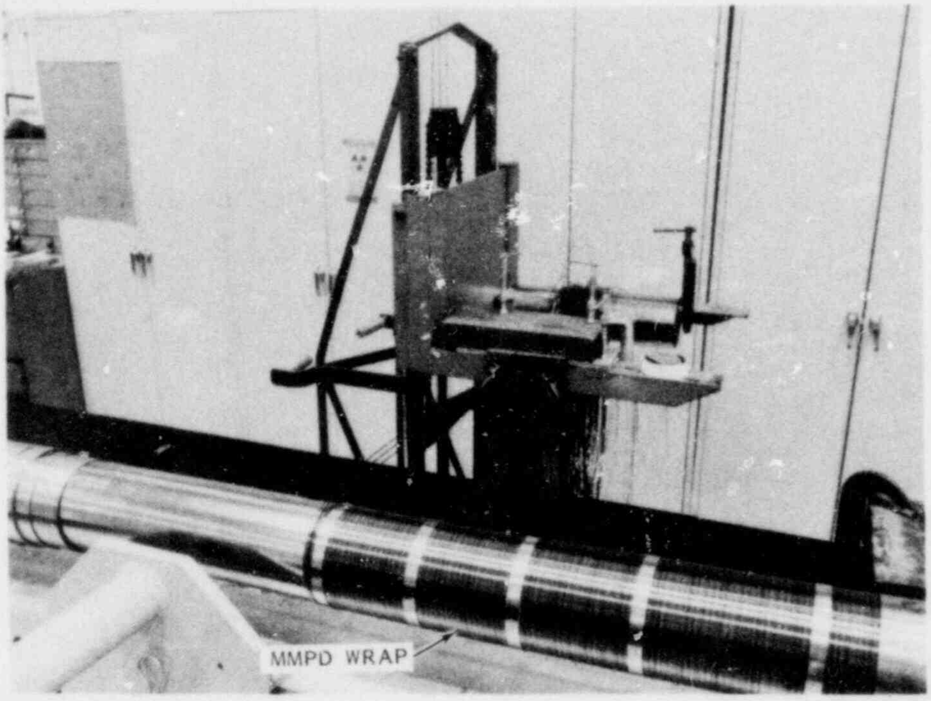

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Figure 6. Molten Metal Penetration Detector Wrap Being Installed in Insulated Shroud Assembly

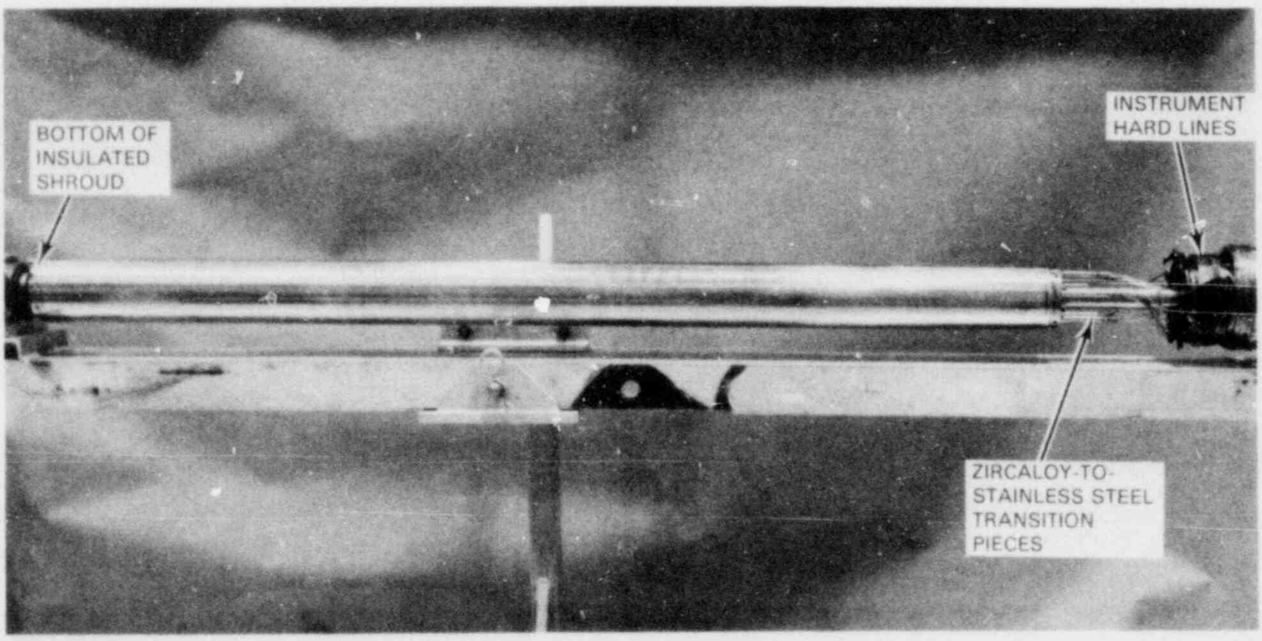

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Figure 7. Completed PNL/PBF Insulated Shroud Assembly

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

A joint Pacific Northwest Laboratory (PNL)-Joint Research Centre (JRC), Ispra Super Sara Test Program (SSTP) severe fuel damage test train design review was held at PNL, Richland, Washington, during the week of March 22, 1982. Oscar Simoni, Roland Zeyen, and Luigi Ferrario visited PNL to discuss test train shroud design, instrumentation, and in-reactor pressure tube design. The U.S. Nuclear Regulatory Commission (NRC) site representative at Ispra, Italy, is continuing to provide liaison and safety analysis support.

INTRODUCTION

The 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 14 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

T. E. Guidotti

SUMMARY

During the last quarter, the released version of the COBRA/TRAC computer code was assessed and code applications continued. A best-estimate simulation of a loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR) equipped with upper head injection (UHI) was performed. The calculation demonstrated that COBRA/TRAC is capable of performing detailed, full-scale system simulations.

INTRODUCTION

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

The code 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) FLECHT = full-length emergency cooling heat transfer.

(c) FEBA = flooding experiment in blocked arrays.

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

TECHNICAL PROGRESS

A COBRA/TRAC calculation of a 200% cold-leg break LOCA in a PWR equipped with UHI was performed using a full-scale, four-loop system model with a 3-D vessel.

BEST-ESTIMATE PWR/UHI CALCULATION

In the last quarterly report[1] a preliminary PWR/UHI calculation was discussed. That calculation was performed before the data assessment of COBRA/TRAC was completed and included some preliminary assumptions. This quarter, a best-estimate PWR/UHI 200% cold-leg break was simulated using COBRA/TRAC.

Both the input model and the code were modified since the October 1981 calculation. The significant changes include:

- An error in the gap conductance calculation was corrected, and a fuel relocation model was added.
- The Cathcart[2] metal-water reaction heat source was added to COBRA/TRAC.
- The falling film quench-front model was modified as the result of comparisons with the Westinghouse G-2 top reflood experiments.
- Best-estimate power peaking factors were used. The peaking factors that were used in the previous calculation were more typical of evaluation model values.
- The 1979 ANSI/ANS-5.1 decay heat standard was used instead of the ANS +20% values.
- The injectable liquid volume in the cold-leg accumulators was increased.
- Noding improvements in the upper and lower plenums were incorporated.

The hot rod peak cladding temperatures for the first 52 s of the calculation are plotted in Figure 1. As the core dried out, the temperature rapidly increased to 1075°F and then dipped beginning at 3.5 s because of flashing in the lower plenum. At 8 s, the peak cladding temperature of 1150°F was reached; simultaneously, the upper head began to flash, forcing liquid down the support columns and guide tubes. By 14 s, this UHI water quenched the entire core and the upper head began to condense, which drew the flow up the guide tubes and support columns and interrupted water delivery to the core. Condensation refilled the upper head with liquid by 18 s, allowing the continued accumulator injection to force liquid down the support columns to requench the core. The UHI accumulator isolation valves closed at 23.2 s, stopping upper head injection and temporarily interrupting liquid flow to the core. The temperature rose briefly before the upper head drain period (starting at 35 s) sent additional water into the core. The calculation will continue during next quarter until the end of bottom reflood.

CONCLUSIONS

This calculation demonstrated COBRA/TRAC's ability to simulate full-scale 3-D systems. Many phenomena were predicted, including upper head flashing, downcomer bypass, and lower plenum level swell. Condensation and CCFL played key roles in the UHI water delivery behavior. Condensation in the upper head limited the UHI delivery and separated each of the three delivery periods while CCFL at the upper tie plate nozzle delayed the initial UHI delivery and prevented UHI penetration during bottom reflood. The peak cladding temperature for this calculation was 1180°F.

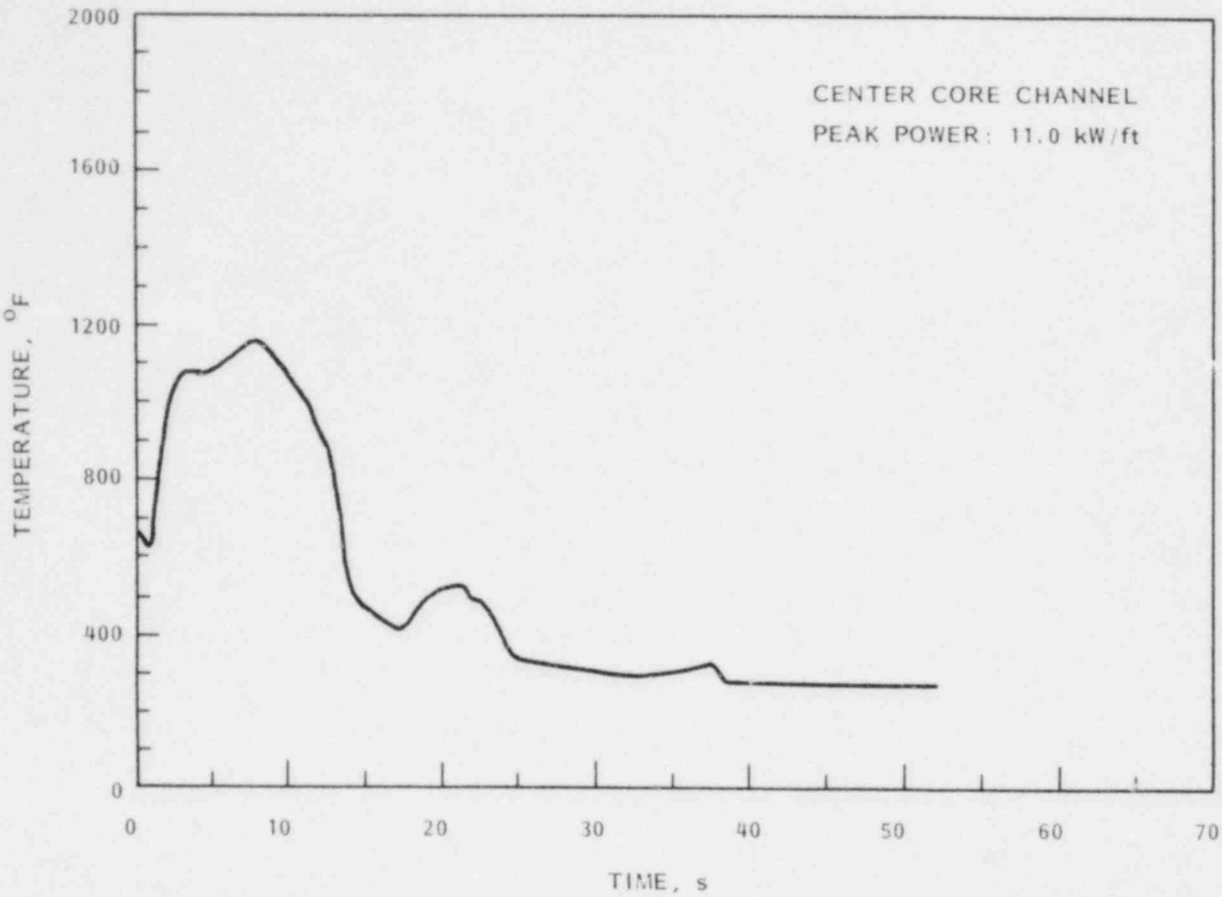

Figure 1. Hot Rod Peak Cladding Temperature

FUTURE WORK

This simulation will be discussed in greater detail in the next quarterly report. A calculation using evaluation model power levels is also planned for next quarter.

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1. Edler, S. K., ed. March 1982. *Reactor Safety Research Programs Quarterly Report, October-December 1981.* NUREG/CR-2127, Vol. 4, PNL-3810-4, Pacific Northwest Laboratory, Richland, Washington.

2. Cathcart, J. V. August 1976. *Quarterly Progress Report on the Zirconium Metal-Water Oxidation Kinetics Program.* ORNL/NUREG/TM-41, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

LOCA SIMULATION IN THE NRU REACTOR[a]

C. L. Mohr, Program Manager

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

SUMMARY

The past quarter was devoted to report preparation and sponsor presentations. The test configuration for the fourth materials experiment (MT-4) was redefined to a 12-rod pressurized cruciform assembly instead of the previously discussed 32-rod fully pressurized test train. Conceptual test train designs for 2100 to 2500°F and 2500 to 3600°F tests were evaluated, and possible test loop modifications for these high-temperature tests were discussed with Chalk River Nuclear Laboratories (CRNL). A paper titled *Ballooning and Flow Blockage for High Alpha LOCA Conditions* was presented by C. L. Mohr at the March 14-19, 1982, International Atomic Energy Agency (IAEA) Specialists' Meeting in Water Reactor Fuel Element Computer Modeling.

INTRODUCTION

The Loss-of-Coolant Accident (LOCA) Simulation in the NRU Reactor progam is being conducted in the National Research Universal (NRU) reactor at 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 and 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.
- 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 reliable cladding temperature control of a simulated LOCA to insure 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.

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

The test loop in the NRU will accommodate a full-length (12-ft), 32-rod bundle on a 6 x 6 array with the corner pins removed. The bundle design uses commercial enrichments, cladding materials and dimensions, and grid spacers.

TECHNICAL PROGRESS

Data analysis and disassembly, examination, reassembly machine (DERM) data reduction were completed for the TH-2, TH-3, and MT-3 experiments. Data reports for each of these experiments and associated final experiment operations plans are being prepared.

Post-test examination of the MT-3 guard rod assembly indicated that it could be reused; therefore, the MT-4 experiment was redefined to be a 12-rod pressurized cruciform in the MT-3 shroud assembly. The proposed MT-4 test should not damage the guard rods, and the shroud assembly will be reused for the MT-5 test.

Conceptual designs using the current shroud with an insulated liner and 21 fuel rods (see Figure 1) were evaluated for the MT-6 (2100°F), MT-7 (2300°F), and MT-8 (2500°F) experiments. Discussions were initiated with CRNL on possible required loop modifications; for example, increasing the loop back pressure to 400 to 800 psi and potential off-gas control systems.

Design concepts for a 12-rod test train with thick insulation that fits in the current NRU loop were evaluated. These test trains would explore the material effects at 2500 to 3600°F in a four-test series (see Figure 2).

FUTURE WORK

The MT-4 experiment will be conducted during the week of May 24, 1982. Licensing and shipping procedures for the previously irradiated fuel rods for MT-5 and MT-6 instrumentation requirements will be defined.

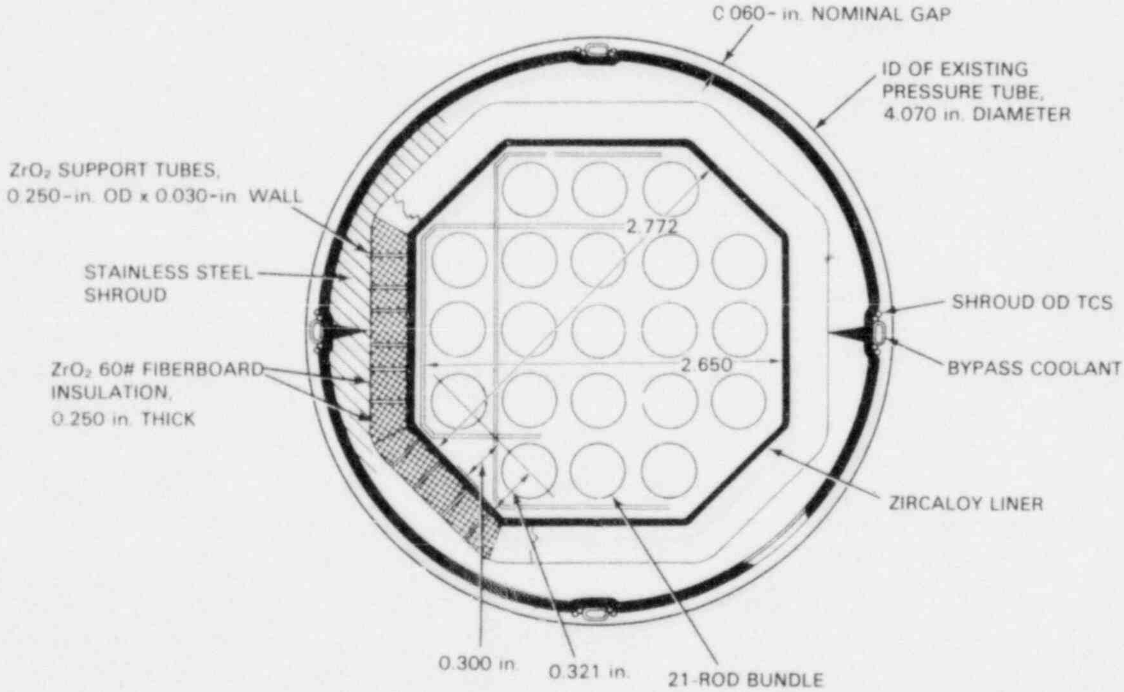

FIGURE 1. Cross Section of NRU Insulated Shroud for 21-Rod Bundle Experiments

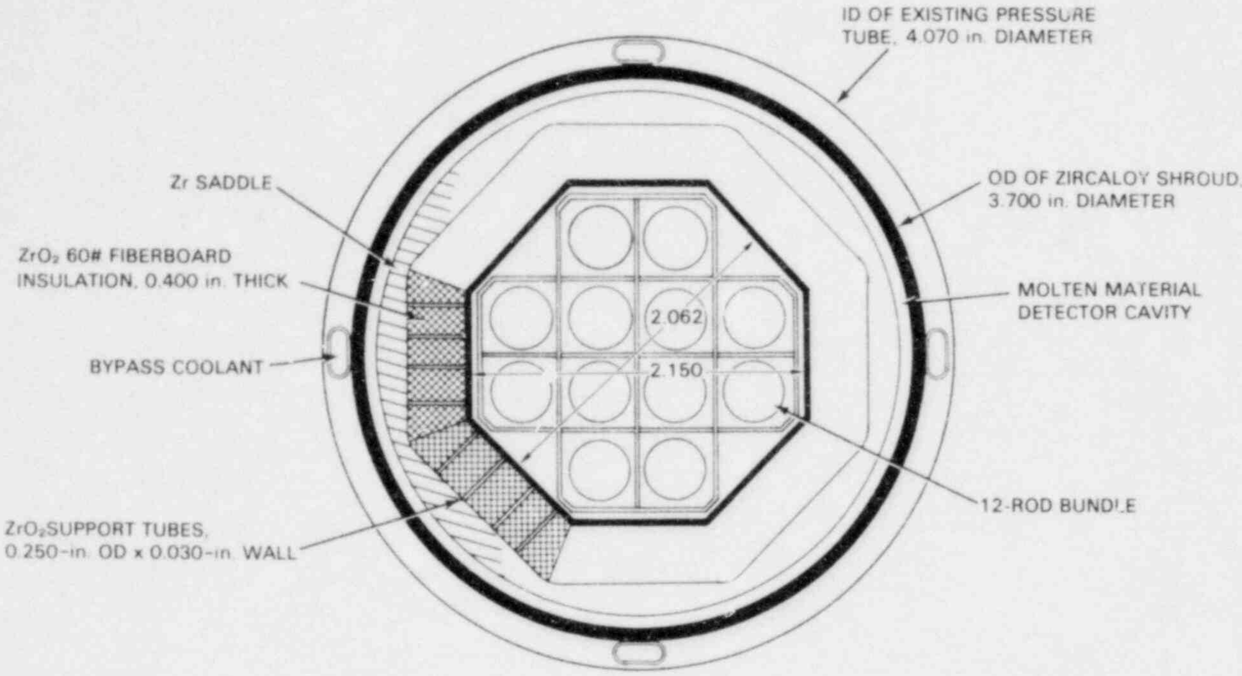

FIGURE 2. Cross Section of NRU Shroud with Thick Insulation for 12-Rod Bundle Experiments

STEAM GENERATOR GROUP PROJECT[a]

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SUMMARY

The Steam Generator Group Project (SGGP) was initiated when the removed-from-service Surry IIA steam generator was positioned in the Steam Generator Examination Facility (SGEF). An organizational meeting of potential SGGP sponsors was held, and marketing of the project also continued with technical presentations in Europe. Task 5, the reopening of preshipment penetrations to re-examine the generator secondary side, was completed. Advertised announcement and RFP packages were prepared for Tasks 6 and 8 (decontaminating the channel head and removing tube plugs, respectively). Radiation field mapping of the generator and the SGEF is progressing; in addition, planning and scheduling of base line eddy current in-service inspection (ISI), nondestructive testing (NDT) round robin, and decontamination tasks are proceeding. Secondary side nondestructive examinations were initiated, and research continued on stress corrosion crack (SCC) characterization and leak rate determinations.

INTRODUCTION

The SGGP, which was initiated this quarter, 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 comple-

(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.

tion of the specially designed containment facility (the 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 NRC's request, PNL is seeking to establish interest from other parties to join in program participation. Potential exists for research and development in chemical cleaning, decontamination, corrosion product identification, corrosion mechanism studies, and materials recovery under alternate sponsors. 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 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, 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.

TECHNICAL PROGRESS

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

STEAM GENERATOR GROUP PROJECT

The SGGP was initiated on January 11, 1982, when the Surry IIA steam generator was placed in the SGEF. An organizational meeting of SGGP participants and potential participants was held in Richland, Washington, February 3, 4, and 5. Twenty-one people representing 20 organizations from 6 countries attended the meeting.

Further marketing was conducted in Europe where three technical presentations were made to potential program participants. A participation contract has been finalized, and contracts have been provided for signatures.

Steam Generator Examination Facility (Task 1)

The SGEF was completed in December 1981 (see Reference 1).

Position the Generator in the SGEF (Task 2)

The 220-ton service-degraded steam generator was lowered through a removable roof panel in the SGEF on January 11, 1982; it is now positioned in its normal vertical operating position. The move to the SGEF from an interim storage site and the lift into the SGEF were performed under subcontract by Neil F. Lampson Co. Figure 1 shows the generator being moved from its interim storage site to the SGEF; and Figures 2, 3, and 4 show the generator being lifted into the SGEF.

Health Physics (Task 3)

The health physics task provides procedures for personnel exposure monitoring, control, and documentation. Health physics research activities will include radiologic mapping, determination of the effectiveness of decontamination efforts, and evaluation of waste and waste disposal problems associated with various operations. Efforts this quarter included exposure control of the steam generator move into the SGEF. Radiation control procedure preparation and review were conducted for initial research efforts involving reopening of preshipment penetrations through the steam generator shell (Task 5) and entering the primary side through the channel head manways in preparation for Tasks 6, 7, 8, and 9. A major portion of the past quarter's effort involved radiation field mapping in the SGEF tower and in the steam generator through shell and primary side penetrations (see Figures 5 and 6). Radiation measurements between 11 and 12 R/h were recorded inside the steam generator on dosimeter trains passed into the unit through shell penetrations. The channel head region had radiation levels of 4 to 5 R/h. In addition, research staff members were trained to operate inside a radiation zone during this quarter.

Data Management System (Task 4a)

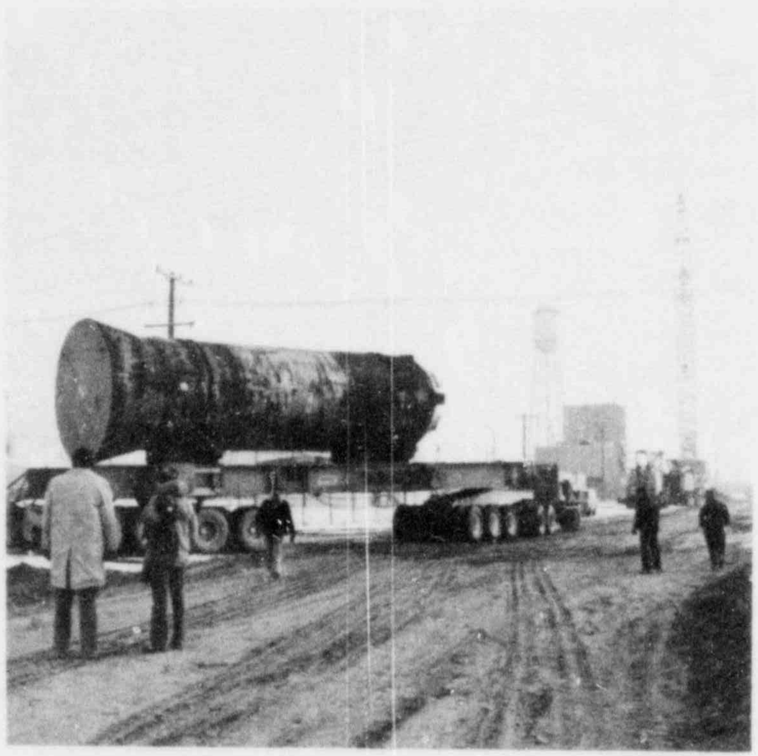
Development of software to interface the computerized data acquisition system with NDT (mainly eddy current) devices providing data inputs was the major activity conducted under this task. In addition, procurement and assembly of components for a remotely operated, computer-controlled NDT probe pusher-puller were pursued. This PNL-designed pusher-puller will automatically provide NDT probe position information to the computer acquisition system in parallel with normal NDT data.

An analysis and compilation of historical ISI eddy current tapes and other historical operating data are proceeding. The historical data base includes ISI data, water chemistry information, and operating records from Westinghouse, Virginia Electric & Power Company, and the NRC.

Reopen Preshipment Penetrations (Task 5)

The preshipment examination involved cutting three penetrations through the generator shell while the unit was stored at the Surry nuclear station. Corrosion product and dimensional data were acquired as was an assessment of general transportability and condition of the unit. The preshipment examination determined that the unit, as stored at Surry, was in a condition representative of the final condition in service and that the unit was not so service degraded as to preclude successful shipment for research purposes.

After the generator was placed in the SGEF, the first task was to reopen the preshipment penetrations and assess any change of condition or damage that had occurred during barge transport to Hanford, interim storage at Hanford, or placement in the SGEF. Dimensional data taken showed no change in measurements from those acquired at Surry; visually, corrosion product coloring looked unchanged, and there was no evidence that water damage had occurred during transport. There was, as expected, some indication that loose debris had relocated within the secondary side, including loose scale/sludge and metallic parts that had previously broken away from the secondary structure. The only noted mechanical change was that several tight inner row U-bends that were cracked during the preshipment examination had now opened. No dislocation in the upper support plate was measured. Figures 7 through 10 depict results of this task.



Neg. 8200094-22cn

Figure 1. Moving the Generator from Interim Storage to the SGEF (seen in background)

Neg. 8200094-83cn

Figure 2. Tilting the Generator into a Vertical Position

Neg. 8200094-97cn

Figure 3. Lifting the Generator into the SGEF

Neg. 8200094-114cn

Figure 4. View of the Generator As It Entered the SGEF Through the Removable Roof Panel

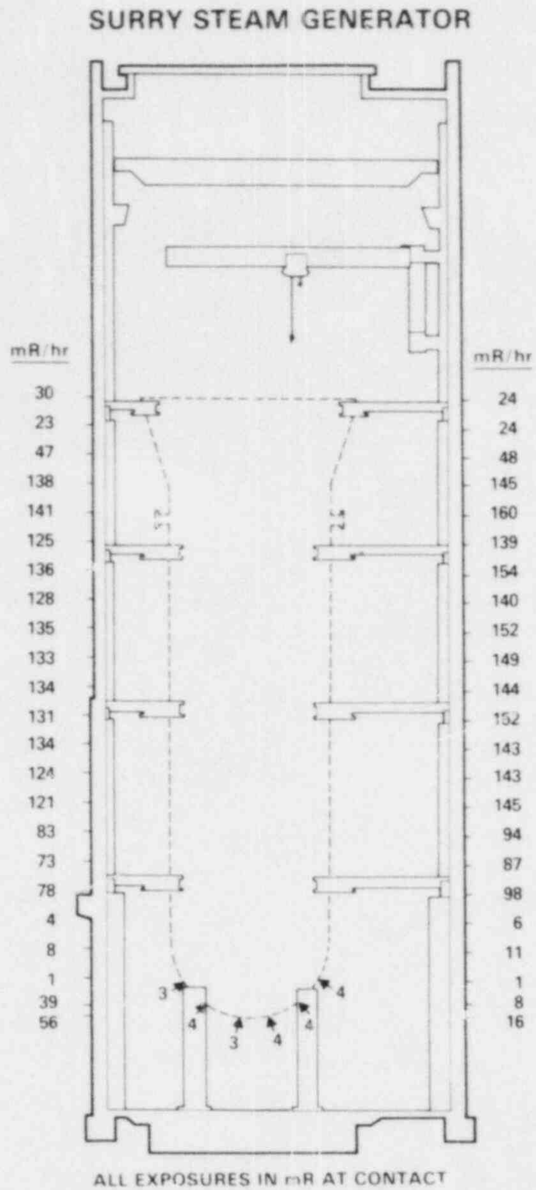


Figure 5. Radiation Levels at Contact of the Generator in the SGEF

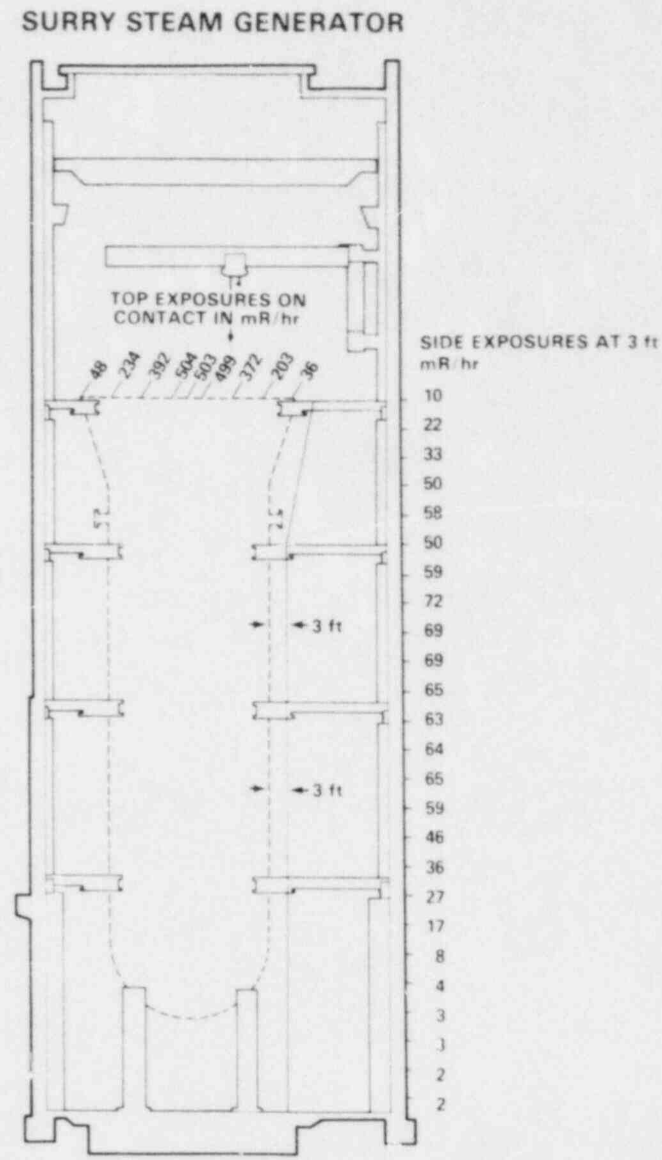
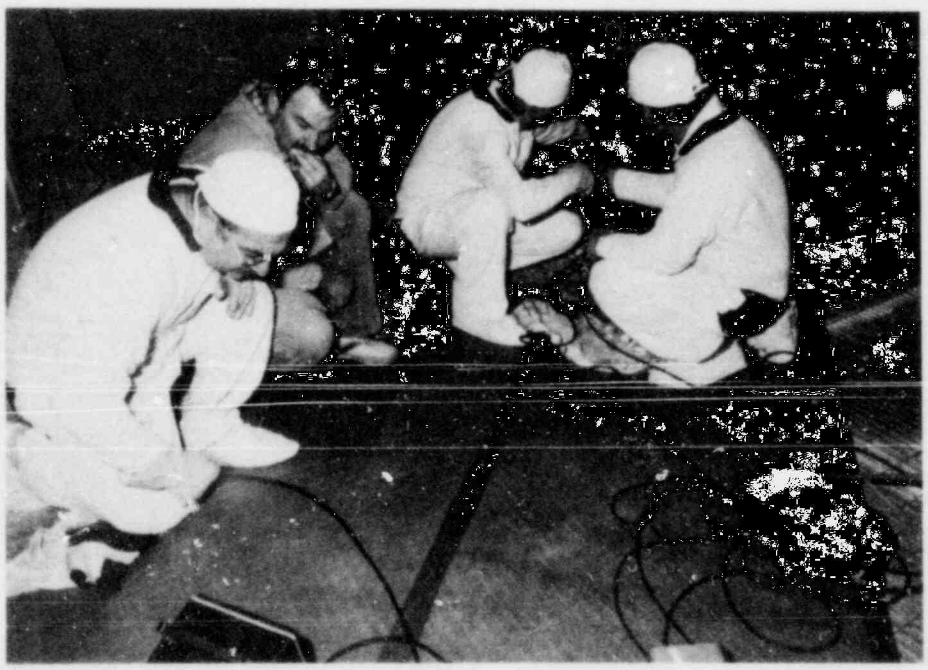


Figure 6. Radiation Levels at 3-ft Distance from the Sides of the Generator

Neg 8201521-7cn

Figure 7. Reopened Preshipment Penetration Showing Reusable Shielding Door and Added Side Shielding

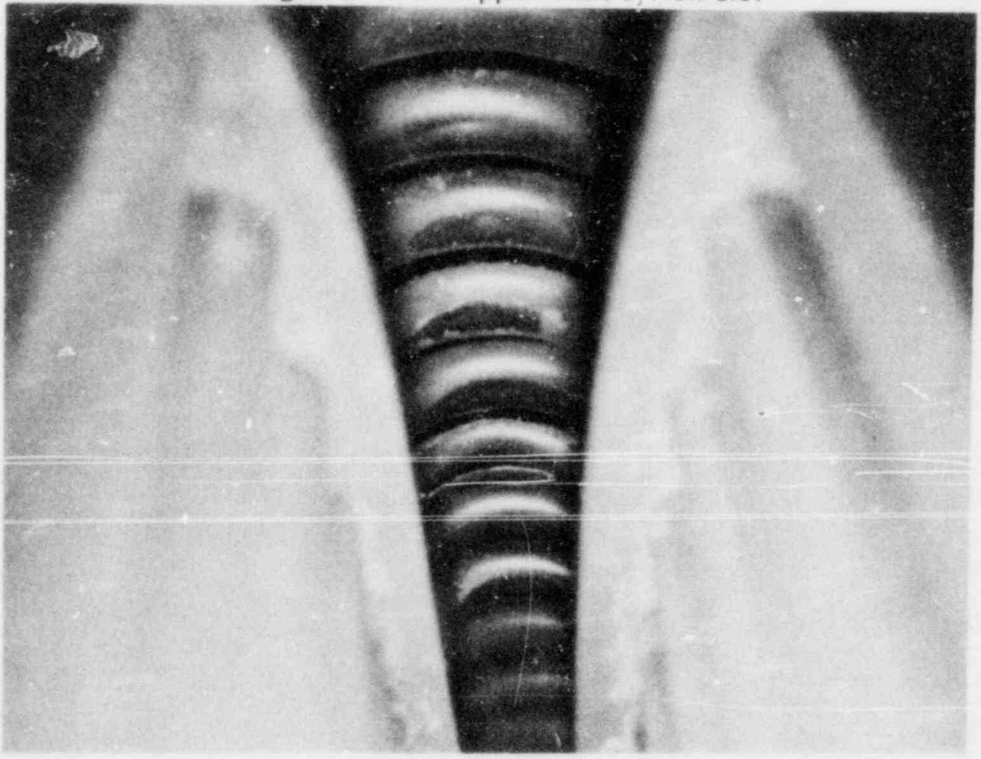


Neg. 8201521-42cn

Figure 8. Video Tape Documentation of Generator Secondary Side Condition Using Miniaturized Camera

Neg. 8200722-8cn

Figure 9. First Support Plate at Flow Slot



Neg. 8201673-15cn

Figure 10. U-Bend Region Through Upper Support Plate Flow Slot

Decontaminate Channel Head (Task 6)

Channel head decontamination is mainly an effort to minimize radiation exposure during primary side access. The availability of the channel head is being offered as an opportunity for research-related development/demonstration of decontamination techniques. To date the cold-leg manway cover has been removed and base line exposure data have been taken; preparations are under way to remove small portions of the stainless steel and Inconel surfaces to examine corrosion product films. These films will be compared with similar postdecontamination surface analysis. During this quarter a *Commerce Business Daily* (CBD) announcement sought sources for the decontamination; a RFP package was prepared and is currently being distributed to interested parties.

Base Line Eddy Current ISI (Task 7)

Examination of 100% of the generator tubes will establish the best possible nondestructive definition of their condition. Prior to the base line examination, tube plugs will be removed (Task 8) from many of the tubes to increase the available defect matrix and allow comparison with historical ISI data. Plans are to use an EM3300 single-frequency system to repeat historical ISI measurements and Zetec M1Z 12 and Intercontrole systems for advanced multifrequency characterizations. A RFP package has been prepared, and subcontract sources will be sought.

Tube Unplugging (Task 8)

Removal of most or potentially all plugs from the 748 plugged tubes is being planned to increase the availability of defects and for correlation with historical ISI data. A CBD announcement seeking subcontract sources has been submitted for publication, and the RFP package is being prepared.

Secondary Side Access (Task 10)

The objective of this task is to conduct nondestructive characterization of the secondary side of the generator, including tubes and support structure. The task will also characterize sludge pile location, and corrosion product sampling will be conducted. Efforts were initiated as an extension of Task 5 activities through the reopened preshipment penetrations, and initial activities focused on extending the areas characterized through the preshipment penetrations. Efforts are concentrating in the inner row U-bend area and at the upper tube sheet surface. Equipment for performing this task has been identified and ordered and includes miniature cameras, a fiberscope, and articulated sampling devices.

Other Tasks

Detailed task planning has been initiated on Tasks 9 (NDT Round Robin), 12 (Specimen Removal), 15 (Secondary Side Cleaning), 16 (Primary Side Decontamination), and 17 (Establish Mockup Tube Bundle).

Management Activities

The organizational meeting of the SGGP was held February 3, 4, and 5 in Richland, Washington, and included representatives of participants and potential participants in this joint research effort. Presentations by technical and program management staff outlined the proposed research, detailed the capabilities to perform this research, and provided information on expertise in the various program areas. Discussions of technical content and program participation mechanisms were held. Additional technical marketing presentations were made in Madrid, Spain; Rome, Italy; and Brussels, Belgium, to participants and potential participants. Additional marketing was conducted with Taiwan and Japan. Detailed task planning was initiated subsequent to overall technical agreement on program content at the organizational meeting.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

An extensive eddy current round test was conducted on 10 laboratory-manufactured SCC defect tube specimens. These specimens were metallographically sectioned this past quarter. A topical report will be issued next quarter on the round robin results.

MILESTONES

- steam generator placed in SGEF
- organizational meeting of SGGP
- Task 5 (Reopen Preshipment Penetrations) completed.

PUBLICATIONS/PRESENTATIONS

- *A Steam Generator Odessey,* which discusses the transport of the Surry generator to Hanford, was presented at the March 25 meeting of the Tri-Cities Chapter ASM.

FUTURE WORK

During the coming quarter the following activities will be pursued:

- establish the contractor for and initiate channel head decontamination task
- establish the contractor for the tube unplugging task
- issue topical report on Task 5
- issue topical report on Task 4a
- continue secondary side inspections
- issue topical report on radiation level mapping
- issue topical report on the eddy current round.

REFERENCES

1. Edler, S. K., ed. March 1982. *Reactor Safety Research Quarterly Report, October-December 1981.* NUREG/CR-2127, Vol. 4, PNL-3810-4, Pacific Northwest Laboratory, Richland, Washington.

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4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Reactor Safety Research Programs Quarterly Report January - March 1982	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) S.K. Edler, Editor	5. DATE REPORT COMPLETED MONTH June YEAR 1982
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Pacific Northwest Laboratory Richland, WA 99352	DATE REPORT ISSUED MONTH July YEAR 1982
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Accident Evaluation Division of Engineering Technology Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. FIN NO. B2101, B2088, B2289, B2043, B2383, B2084, B2372, B2041, B2277, B2097
13. TYPE OF REPORT Quarterly	PERIOD COVERED (Inclusive dates) January - March 1982
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT (200 words or less)

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from January 1 through March 31, 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.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report) Unclassified	21. NO. OF PAGES
Unlimited	20. SECURITY CLASS (This page) Unclassified	22. PRICE $

NRC FORM 335 (11-81)

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