NUREG/CR-2716
Vol. 3
PNL-4275-3
R3,4,5,GS

Reactor Safety Research Programs

Quarterly Report
July-September 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
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Manuscript Completed: January 1983
Date Published: March 1983

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Prepared for
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Division of Engineering Technology
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 July 1 through September 30, 1982, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor 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 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, 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]

W. C. Morgan, Project Manager
J. M. Prince, Project Manager

D. L. Lessor

SUMMARY

A progress report covering the results of this program was issued. All preoxidation measurements were obtained on the 112 new samples from the two bars of PGX graphite. The new sweep-frequency eddy-current instrument was assembled and is being checked out.

INTRODUCTION

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

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

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

TECHNICAL PROGRESS

A report[1] covering results obtained on this program has been issued. PNL and ORNL coordinated plans to test the new eddy-current instrument on graphites oxidized in the CFTL facility. This cooperative effort should be beneficial to both programs. Other progress completed during this quarter is detailed below by topic.

(a) FIN: B2101-2; NRC Contact: R. B. Foulds.

EDDY-CURRENT TESTING

The laboratory sweep-frequency eddy-current system was fabricated, and it is in the final stages of checkout. The system is designed to operate from 10 kHz to 10 MHz and provide highly accurate and orthogonal complex impedance-plane eddy-current data. It is very important at this stage in the development of the ZFIT program that nonlinearities in the eddy-current processing electronics be minimized to in turn minimize nonlinearities in the ZFIT eddy-current modeling.

Data profiles have been taken with an existing eddy-current instrument at 15 test frequencies ranging from 50 kHz to 1 MHz. The deterioration of the orthogonality of the reactive and resistive data in the complex impedance plane with increasing test frequency excluded data taken above 100 kHz from use in the ZFIT program. It should be noted that the eddy-current model in ZFIT at this point assumes that the impedance plane data are ideal rather than nonlinear. The new sweep-frequency eddy-current system will greatly improve this data characteristic.

ULTRASONIC TESTING

Due to the sudden change in emphasis in the associated U.S. Department of Energy (DOE)-sponsored program, work on the development of the ultrasonic backscattering technique was largely curtailed during this quarter. Staff assignments, however, have been made; and progress should be more rapid next quarter.

OXIDATION

Preoxidation measurements have been obtained on the 112 samples machined from the two new bars of PGX graphite. Ten cylindrical samples (76 mm in diameter by 152 mm long) from each log were compression tested in accordance with ASTM method C695—Method of Test for Compressive (Crushing) Strength of Graphite. The compressive strength of samples from log VIII averaged 41.00 $\pm$2.56 MPa and those from log IX averaged 35.82 $\pm$1.31 MPa, thus bracketing the 37.18 $\pm$2.78 MPa previously measured on samples from log VII. Densities of these samples also bracketed those of log VII (1.743 $\pm$0.013 Mg/m^3); log VIII averaged 1.771 $\pm$0.011 Mg/m^3 and log IX averaged 1.732 $\pm$0.008. It is interesting that the variance in density mirrors the variance in strength.

Many small particles were produced when the samples from log IX broke; this was especially noticeable with the lower density (and lower strength) samples. Less binder was probably used to manufacture log IX than was used for logs VII and VIII.

FUTURE WORK

EDDY-CURRENT TESTING

- finish checkout of sweep-frequency eddy-current system
- begin taking data with the new eddy-current system on old and new graphite specimens
- make preoxidation measurements on CFTL samples.

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

- measure strength of oxidized cylinders
- obtain correlations of strength versus density
- obtain correlations of electrical conductance versus density.

REFERENCES

1. Morgan, W. C., T. J. Davis, and M. T. Thomas. 1982. *Feasibility of Monitoring the Strength of HTGR Core Support Graphite - Part III.* NUREG/CR-2929, PNL-4449, Pacific Northwest Laboratory, Richland, Washington.

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

P. H. Hutton, Project Manager
R. J. Kurtz, Assistant Project Manager

SUMMARY

The acoustic emission (AE) monitor/analysis system for the ZB-1 vessel test has been laboratory tested and is now installed on the test vessel at Mannheim, West Germany. Test checkout and calibration is about 90% complete, and testing is expected to start the first week of October 1982. The permanent AE signal lead wires for reactor monitoring at the Tennessee Valley Authority's (TVA's) Watts Bar 1 reactor are being installed. The wires should be in place by October. A long-term (~6-month) stress corrosion cracking (SCC) test was started August 23, 1982, with 4-in. Schedule 80 Type 304 stainless steel pipe. The results from this test will be compared with those from an accelerated SCC test that is also in progress.

INTRODUCTION

The purpose of this Pacific Northwest Laboratory (PNL) program is to provide an experimental evaluation of the feasibility of detecting and analyzing flaw growth in reactor pressure boundaries on a continuous basis using AE. Type A533B, Class 1 pressure vessel steel, SA351-CF-8A cast stainless, Type 304 wrought stainless, and A106 ferritic piping steels are being used in the experimental testing. The objectives of this program are to:

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

TECHNICAL PROGRESS

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

OFF-REACTOR VESSEL TEST

The ZB-1 vessel test has been the focal point of program work this quarter. The test is scheduled to begin the first week of October at the Grosskraftwerk Mannheim (GKM) test facilities, Mannheim, West Germany. The test vessel is situated in an underground bunker; all controls and instrumentation are located in trailers adjacent to the bunker.

The hardware and software for the AE monitor/analysis system were completed and tested in the laboratory at PNL. The data acquisition/source location system demonstrated the capability to process 15 signals/second from three arrays without saturating. As a safeguard against loss of data, raw data are recorded on a high-density cartridge recorder in parallel with being sent to the source location unit.

All test equipment has been shipped to Mannheim under appropriate U.S. customs export licenses. The AE monitor system and peripherals have been installed and checkout/calibration is ~90%

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

complete. The test will also be AE monitored by at least one German group for added assurance that the AE data generated during this first-of-a-kind test will be adequately monitored and preserved.

REACTOR MONITORING

The permanent AE signal lead wires for reactor monitoring at the Watts Bar 1 reactor are being installed and should be in place by October. The next AE monitoring activity that is planned at Watts Bar will take place during hot functional testing (scheduled for March 1983).

PIPE MATERIAL CHARACTERIZATION

A long-term SCC test began on August 23, 1982. The specimen is a 4-in. Schedule 80 Type 304 stainless steel pipe with postweld sensitization. The test conditions are: internal pressurization with 550°F water at 1400 psig containing 6-ppm O_2. No external load is being applied. Under these conditions, it is expected that 6 months will be required to initiate SCC. Results from this test will provide an opportunity to compare AE data from a long-term SCC test with data from a short-term test that is also in progress.

REPORTS

- Hutton, P. H., et al. July 1982. *Acoustic Emission Monitoring of ASME Section III Hydrostatic Test - Watts Bar Unit 1 Nuclear Reactor.* NUREG/CR-2880, PNL-4307, Pacific Northwest Laboratory, Richland, Washington.
- Quarterly progress report for the period from April 1 to June 30, 1982.

FUTURE WORK

Plans for the period from October 1 to December 31, 1982, include:

- perform the ZB-1 vessel test and associated AE data analysis
- complete installation of permanent AE signal leads at Watts Bar 1 reactor
- continue pipe material tests
- continue development of an engineering prototype AE monitor system
- provide input for the Materials Engineering Branch annual report
- present a review of the program at the conference on Periodic Inspection of Pressurized Components in London, England.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[a]

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S. H. Bush, Project Manager
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T. T. Taylor, Project Manager

G. B. Dudder
P. G. Heasler
G. P. Selby

SUMMARY

During the past quarter, progress was made on a number of tasks. Manual evaluation of all piping round robin test results was completed, and extensive computer analysis of the piping round robin began. U.S. participation in the Programme for the Inspection of Steel Components (PISC) II program began this past quarter, and Pacific Northwest Laboratory (PNL) made preparations to process the data before sending it to Ispra. The results from the search unit tracking and recording system (SUTARS) analysis of piping round robin samples were received. Preliminary results of using the synthetic aperture focusing technique (SAFT) for ultrasonic testing (UT) on centrifugally cast stainless steel (CCSS) are promising; a program to thoroughly evaluate SAFT-UT on CCSS is being prepared. The underclad crack work indicates that the surface roughness of the cladding adversely impacts the sound field and the inspections. All Matrix I specimens have been fabricated and are currently being studied. The probabilistic crack growth calculations have begun using probability-of-detection curves derived from the piping round robin tests. The pressurizer surge line flaw growth calculations and analysis were completed, and a report was prepared for review.

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

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

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

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

TECHNICAL PROGRESS

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

PIPING ROUND ROBIN

All the data recorded by the ISI teams who participated in the round robin were manually checked to eliminate errors, identify error sources, and insure that all data were entered into the computer correctly. Where errors were found, the source was identified and entered into the computer for future analysis. Based on a summary of the corrections, it is not anticipated that there will be any major changes in the results that were reported earlier.[1]

To participate in the piping round robin, ISI team members provided histories of their nuclear reactor inspection experience and UT experience (see Table 1). It is evident from these data that the Level II and Level III participants had extensive experience in both UT and the number of preservice inspections (PSIs) and ISIs in which they had participated prior to the round robin.

Table 1. Summary of ISI Team Members' Qualifications

Team Member Classification	UT Experience, yr		Number of PSIs and ISIs	
	Average	Range	Average	Range
ASNT Level III	10.2	4 to 23	28	7 to 62
ASNT Level II	7.4	2.5 to 13	16.7	2 to 57
ASNT Level I	1.1	0.5 to 2.5	2	0 to 5

PVRC-PISC II

Data recorded by U.S. participation in PISC, which is supported by the U.S. Pressure Vessel Research Committee (PVRC), are being examined by PNL prior to shipment to the Commission of the European Communities, Joint Research Centre, Ispra, Italy. The data are being examined to: 1) check for errors and insure that the data are in the proper reporting format and 2) provide guidance for the work planned in the pressure vessel applications task of the NDE/FM program. Data from the U.S. teams are scheduled to be received by PNL during October 1982.

ADVANCED TECHNIQUES ASSESSMENT FOR PIPING ROUND ROBIN SAMPLES

Southwest Research Institute (SwRI) used the SUTARS on 10-in. Schedule 80 piping round robin specimens. Data were collected during June 1982, and a report of the analysis was received during August. The SUTARS results are currently being manually evaluated and will be scored by the same computer procedures used in the round robin.

Some preliminary testing was performed at the University of Michigan using the SAFT on several CCSS piping round robin specimens. These results appear to be quite encouraging, and a program to thoroughly evaluate SAFT on CCSS is being defined (scheduled for the second quarter of 1983).

UNDERCLAD CRACK DETECTION

The short-term response for the underclad crack detection program was based on results from 1) plates containing cracks that were induced by thermal fatigue, 2) the PNL cracked pressurizer dropout, and 3) machined notches in the Electric Power Research Institute (EPRI) cladded plates.[2,3] Since then, several new samples have been fabricated to complete the Matrix I specimens (see Table 2); these new specimens are being evaluated.

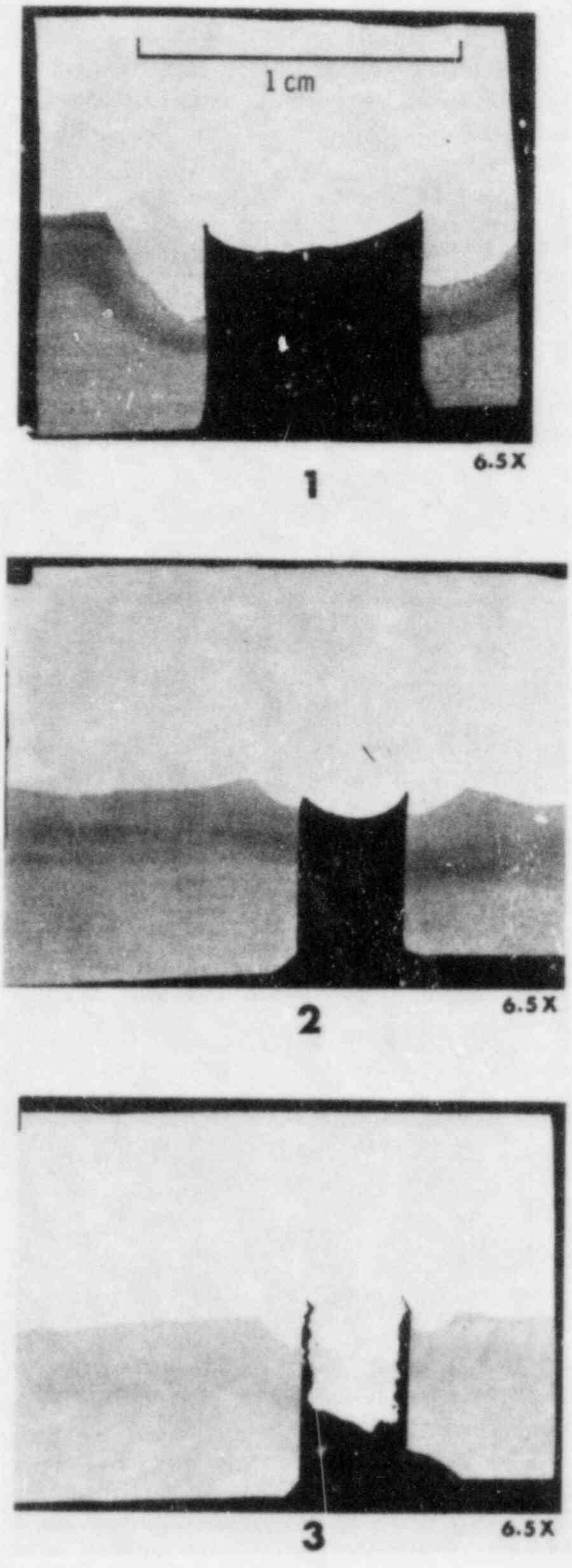
Notches were machined in a carbon steel plate and filled with either tungsten powder or a ceramic powder prior to cladding. Micrographs of cross sections from these notches (see Figure 1) show that the tungsten powder mixed with the overlay metal. Both methods produced a corner near the cladding-carbon steel interface. The tungsten corners were slightly above the interface and were very crack-like. These notches should simulate an actual crack better than an unfilled notch because the cladding metal fills the notch and the top is no longer at the proper height.

Table 2. Flaw Characteristics for Matrix I Specimen

Block	Type of Cladding[a]	Cladding Surface Condition	Type of Flaw[b]	Flaw Direction[c]	Flaw Depth,[d] in.
1	MMA	Smooth	I	Parallel	0.5
	MMA	Smooth	I	Normal	0.5
	MMA	Smooth	II	Parallel	0.5
	MMA	Smooth	II	Normal	0.5
2	MMA	Smooth	III	Parallel	0.5
	MMA	Smooth	III	Parallel	0.5
	MMA	Smooth	III	Normal	0.5
	MMA	Smooth	III	Normal	0.5
3	MMA	Smooth	IV	Parallel	0.5
	MMA	Smooth	IV	Parallel	0.5
	MMA	Smooth	IV	Normal	0.5
	MMA	Smooth	IV	Normal	0.5
4	MMA	Smooth	V	Normal	0.5
	MMA	Smooth	V	Normal	0.5
5	SC or MW	Smooth	VI	As available	0.5

(a) Cladding type: manual metal arc (MMA), single-wire submerged arc (SW), multiple-wire arc (MW), strip cladding (SC).
(b) Refer to Figure 2 for flaw type.
(c) With respect to lay of the cladding.
(d) Aspect ratio (a/ℓ) as close to 1/3 as possible.

Preliminary results[2,3] have shown that 1) smooth cladding surfaces are inspectable and 2) as the cladding surface roughness increases, the inspectability is reduced. The worst case is for unground manual method arc (MMA) cladding. A series of cladding surface profiles for two plates were generated using a mechanical scanner and a linear voltage differential transformer (LVDT) (see Figure 3). The surface was smoothed by hand grinding.



Neg. 82G249-1

Figure 1. Micrographs of Top Portion of Machined Notches (1 and 2 were filled with ceramic powder; 3, with tungsten powder)

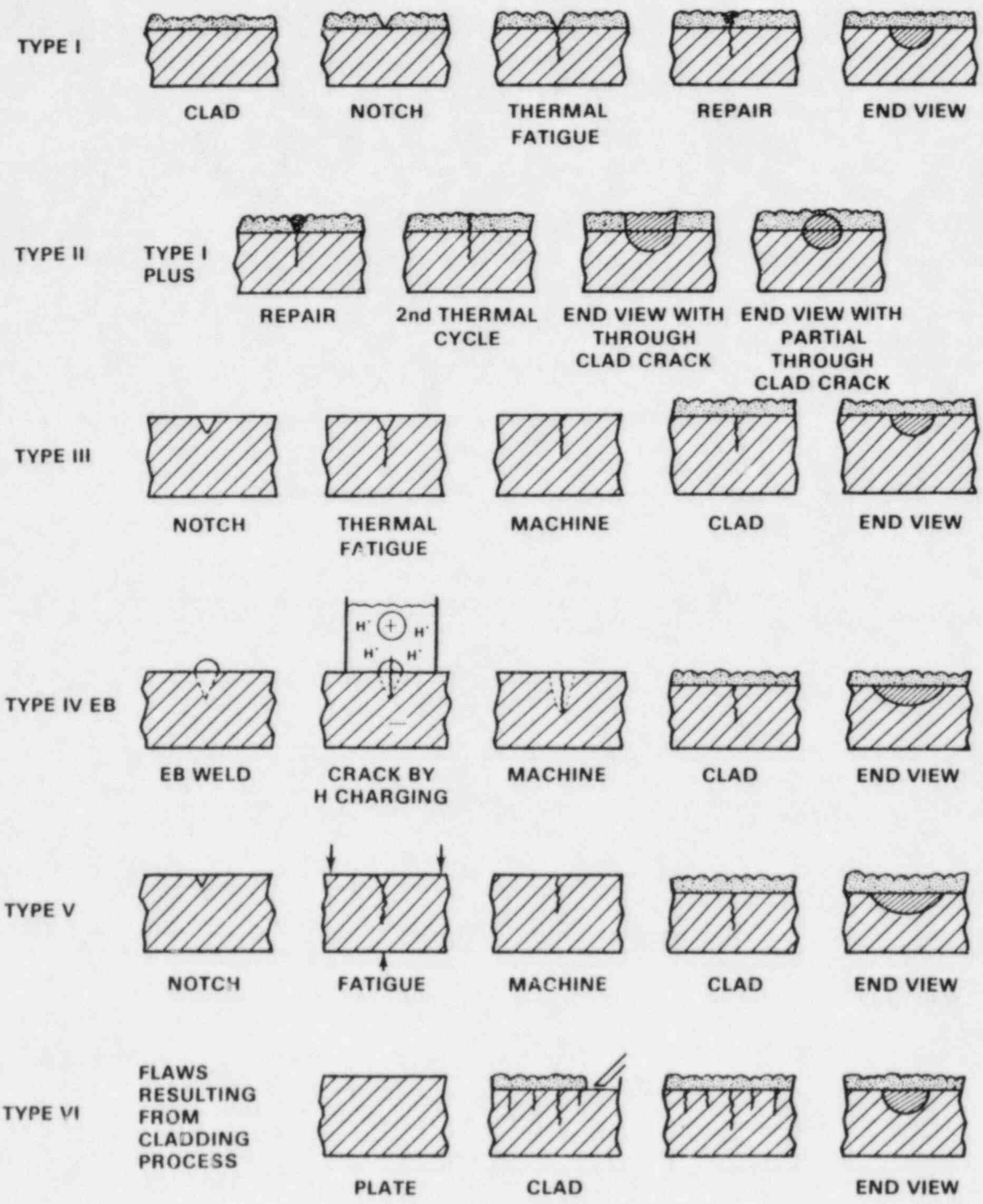
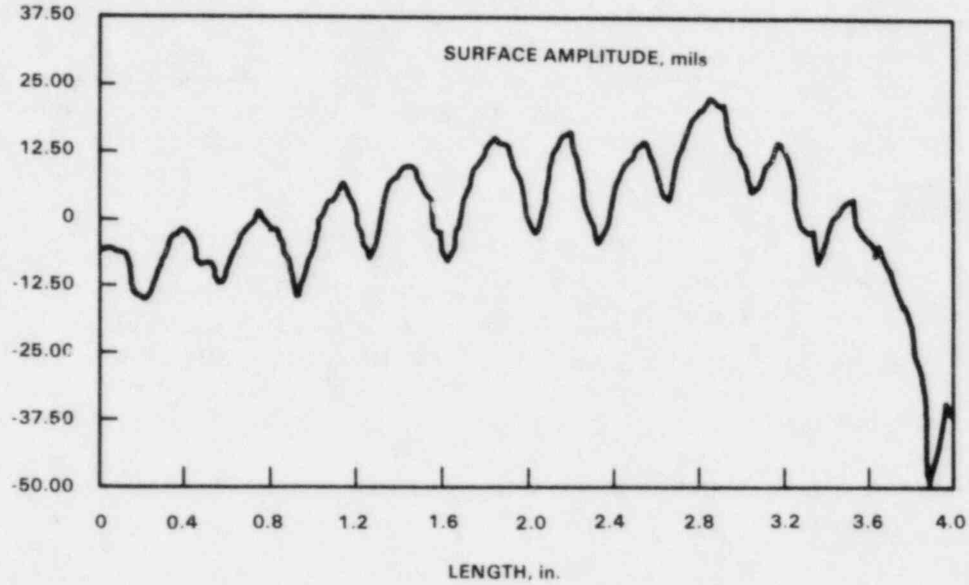
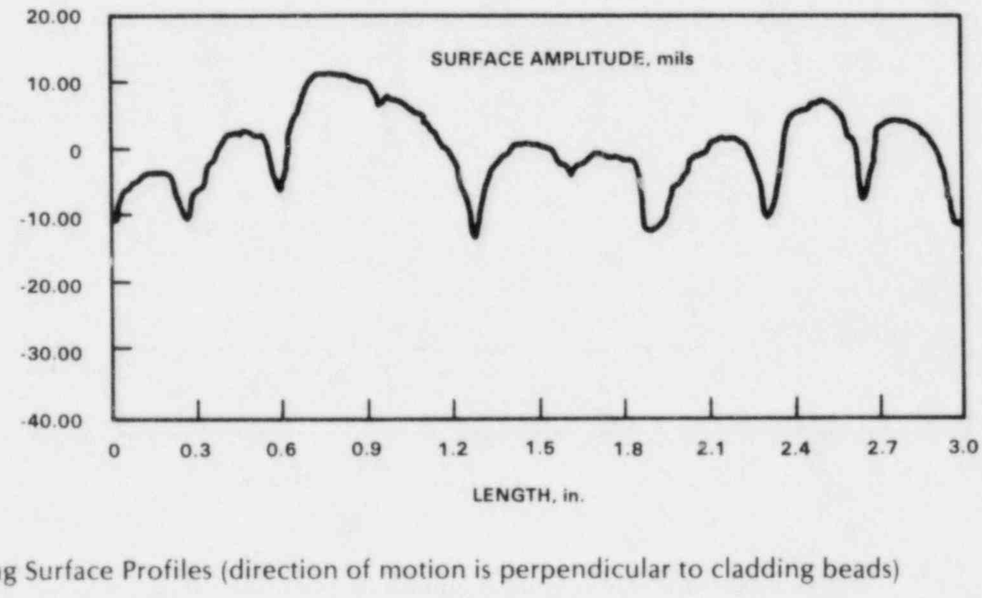


Figure 2. Methods for Fabricating Underclad Cracks



(a) Before Grinding



(b) After Grinding

Figure 3. Cladding Surface Profiles (direction of motion is perpendicular to cladding beads)

It is very apparent that the quality of the inspection deteriorates as the cladding surface roughness increases. To better understand the effect, these surface profiles were used with a ray-tracing computer program to demonstrate their influence. Computer results for the case of a perfectly smooth cladding surface are shown in Figure 4; for an unground MMA cladding surface, in Figure 5; and for a moderately ground cladding surface, in Figure 6. The adverse impact that the surface roughness has on the sound field is apparent. Future work will address methods to characterize the surface roughness and assess the surface smoothness necessary to perform a reliable ultrasonic inspection.

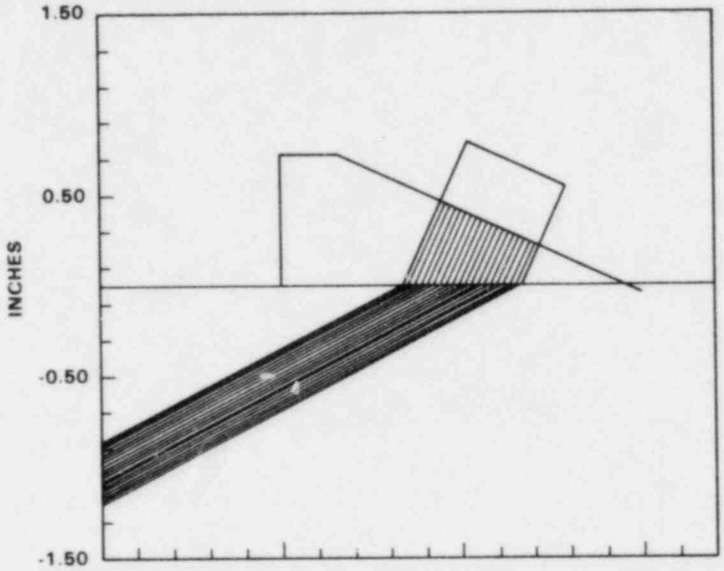


Figure 4. Scatter of Ultrasound Through an Ideal (smooth) Surface

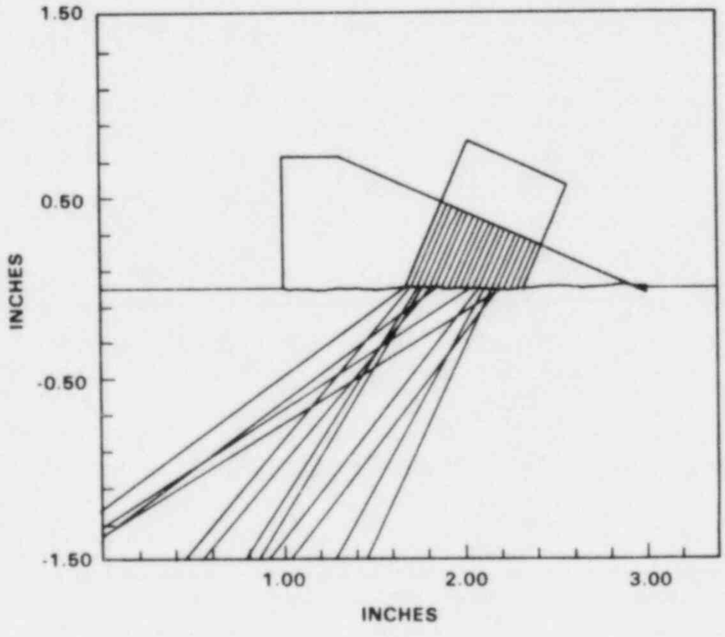


Figure 5. Scatter of Ultrasound Through an Unground Surface

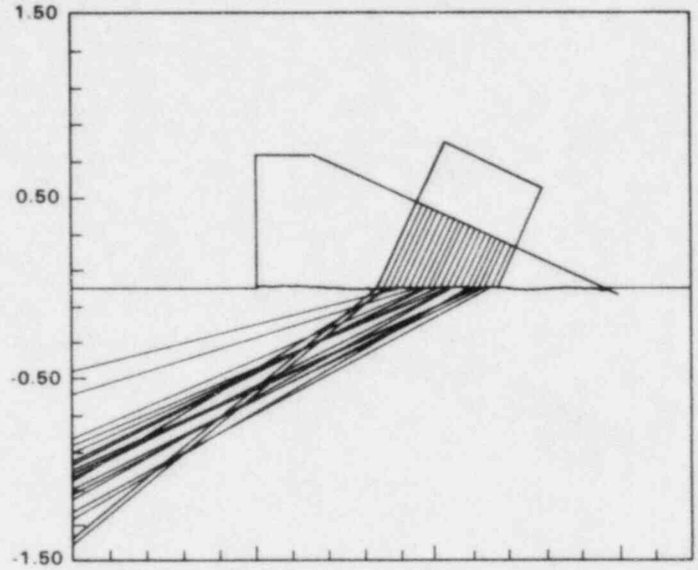


Figure 6. Scatter of Ultrasound Through a Ground Surface

FRACTURE MECHANICS - PROBABILISTIC CRACK GROWTH CALCULATIONS

The main activity during this quarter was the start of probabilistic fracture mechanics calculations through arrangements with Lawrence Livermore Laboratory (LLL). PNL probability-of-detection data are being applied in analyses of alternate inspection scenarios. The LLL model of pressurized water reactor (PWR) steam generator feedwater line cracking has been modified to simulate the growth by fatigue of small surface cracks that initiate by thermal fatigue.

A probabilistic crack growth rate equation has been developed, but it requires further revision. Once this equation is revised, a series of inspection scenarios will be analyzed using the flaw detection curves of Figure 7. These curves are based on the PNL round robin data for clad ferritic piping with extrapolation from the data to:

- treat flaws <5% of the wall thickness
- estimate flaw detection curves for pipe wall thicknesses <2.375 in.
- estimate the enhanced detection capability that can be achieved through improved procedures and technology.

It is expected that LLL will complete the calculations during the next quarter.

PRESSURE SURGE LINE ANALYSIS

Flaw growth calculations were completed for a pressurizer surge line, and a report was prepared. The objective of the calculations was to evaluate ASME Code inspection requirements for piping with a higher level of fatigue usage than exists in PWR primary piping. The results predict that crack initiation should not occur for the design load transients but that small but Codeacceptable flaws can grow by fatigue to become a through-wall leaking crack within 8 yr of operation. Conservatisms in the deterministic crack growth calculations were evaluated. The estimated probability for significant flaw growth, given the existence of a fabrication flaw, is less than $\sim 10^{-2}$. The probability of occurrence of upper bound crack growth rates was combined with the probability of flaw occurrence, and the

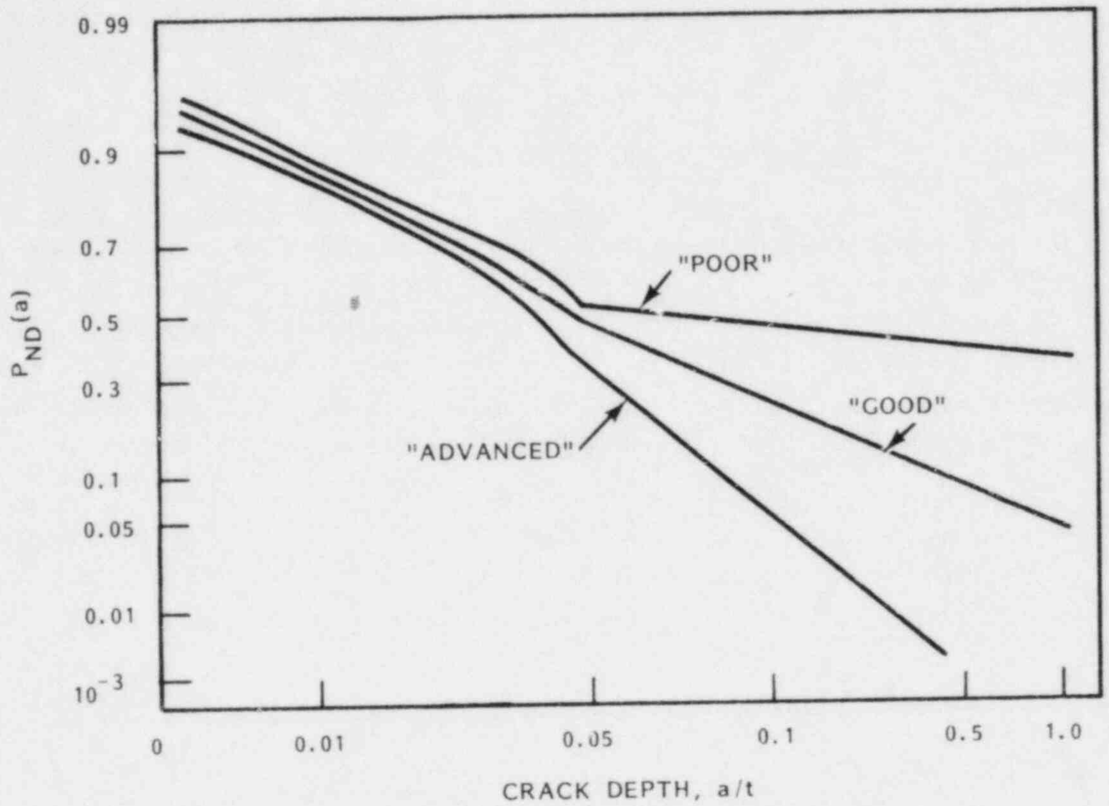


Figure 7. Probability of Flaw Nondetection for Ferritic Piping

estimated leak probability was 5×10^{-5} to 10^{-3}. With the present Code ISI intervals, it was concluded that most flaws should be detected before through-wall cracking and leaks occur.

FUTURE WORK

- analyze all piping round robin data
- prepare the piping round robin report
- destructively assay selected piping round robin specimens
- analyze results from advanced techniques for piping round robin specimens
- review PVRC-PISC II data generated by U.S. ISI teams
- complete data collection and analysis on the underclad crack Matrix I specimens
- begin fabricating the underclad crack Matrix II and III specimens and the specimens for extension of piping round robin results

- complete data collection for the flaw/transducer/instrument interaction matrix
- publish the first volume of the White Paper.

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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, pp. 11-22.

2. Edler, S. K., ed. October 1982. *Reactor Safety Research Programs Quarterly Report, April-June 1982*. NUREG/CR-2716, Vol. 2, PNL-4275-2, Pacific Northwest Laboratory, Richland, Washington, pp. 13-24.

3. Crawford, S. L., et al. September 1982. *Detection of Small-Sized Near-Surface Underclad Cracks in U.S. Pressure Vessels.* NUREG/CR2878, PNL-4373, Pacific Northwest Laboratory, Richland, Washington.

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

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 postirradiation examination (PIE); IFA-513 will remain in Norway. IFA-432 was removed from the reactor in June 1981 in preparation for shipment to Harwell for PIE. However, a proposal for continued high-burnup irradiation was accepted; and four rods from the assembly were reinserted in the reactor. Initial ceramography has been received for rods 1 and 6 of IFA-432 and rod 4 of IFA-527; differences in fuel cracking and relocation between the two assemblies were observed. Helium pressurization of rod 6 of IFA-432 revealed no leaks, thus indicating that no gross leakage of fission gas is likely to have occurred despite damage to the thermocouple leads.

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

INTRODUCTION

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

- collect and analyze in-reactor data on fuel rod thermal/mechanical behavior, especially as a function of burnup and rod design
- correlate in-reactor data with postirradiation data and with ex-reactor tests on mechanical and thermal fuel rod parameters
- 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), Halden, 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:

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

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50 to 380 μm (0.002 to 0.015 in.)
- gas compositions ranging from pure helium to pure xenon
- fuel densities of 95% and 92% theoretical density (TD), the latter both stable and unstable regarding in-reactor densification
- burnups up to 45 GWd/MTM.

TECHNICAL PROGRESS

The examination of IFA-432 (rods 1 and 6) and IFA-527 has continued at Harwell. Examinations completed this quarter include:

- ceramography of both IFA-432 rods (and one comparison cross section from rod 4 of IFA-527)
- "clamshell" cladding slitting on the upper end of rod 1 of IFA-432
- water recovery from IFA-527 rod 4.

Initial conclusions from the preliminary ceramography of the IFA-527/432 rods are that fuel cracking and relocation are similar for rods 1 and 6 of IFA-432 but far less fuel cracking and relocation took place in IFA-527, which operated in a much lower linear heat generation range (12 to 18 kW/m versus 35 to 50 kW/m). The ceramography confirms similar conclusions drawn from in-reactor fuel temperature measurements. Cross sections from lower end regions of rods 1 and 6 of IFA-432 are shown in Figures 1 and 2, respectively; and a comparative cross section from rod 4 of IFA-527 is shown in Figure 3.

The clamshell axial slitting of rod 1 of IFA-432 was done in the hope of leaving behind an area of visible fuel pellet surface, the features of which would then be correlated with features on the cladding inner surface and with the very detailed profilometry performed on the upper half of that rod. The first attempt at slitting the cladding revealed remarkable features on the inner surface of the cladding: traces of pellet-pellet interfaces, traces of major fuel cracks, and galling (see Figure 4). Unfortunately, the shattered pellet surface fragments tended to lift away from the pellet core along with the upper clamshell; therefore, no undisturbed areas of pellet surface could be viewed. The procedure will be repeated with the upper clamshell at 1/3 round rather than 1/2 round.

Small quantities (milligrams) of water were recovered in a cold trap from rod 4 of IFA-527 when it was heated in-cell. The deuterium-to-hydrogen ratio for this recovered water was ~0.038, which definitely indicates the inleakage of heavy water during in-reactor operation, as was also indicated by in-reactor rod pressure and fuel temperature measurements.

The preliminary results on fission gas release for rods 1 and 6 (IFA-432) reported last quarter were confirmed in further analysis. Rod 1 was pressurized with helium to 500 psi and held at that pressure for 24 h with no detectable leaks, which confirms that no gross leakage of fission gas is likely to have occurred for this rod during storage or transit despite damage to the thermocouple leads.

FUTURE WORK

The following examinations remain to be completed at Harwell:

- detailed ceramography to determine grain size and porosity size distributions
- electron probe microanalysis (EPMA) scans and coring in the IFA-432 rods to determine retained fission gas distribution
- bulk burnup and fuel pellet density measurements in the IFA-432 rods

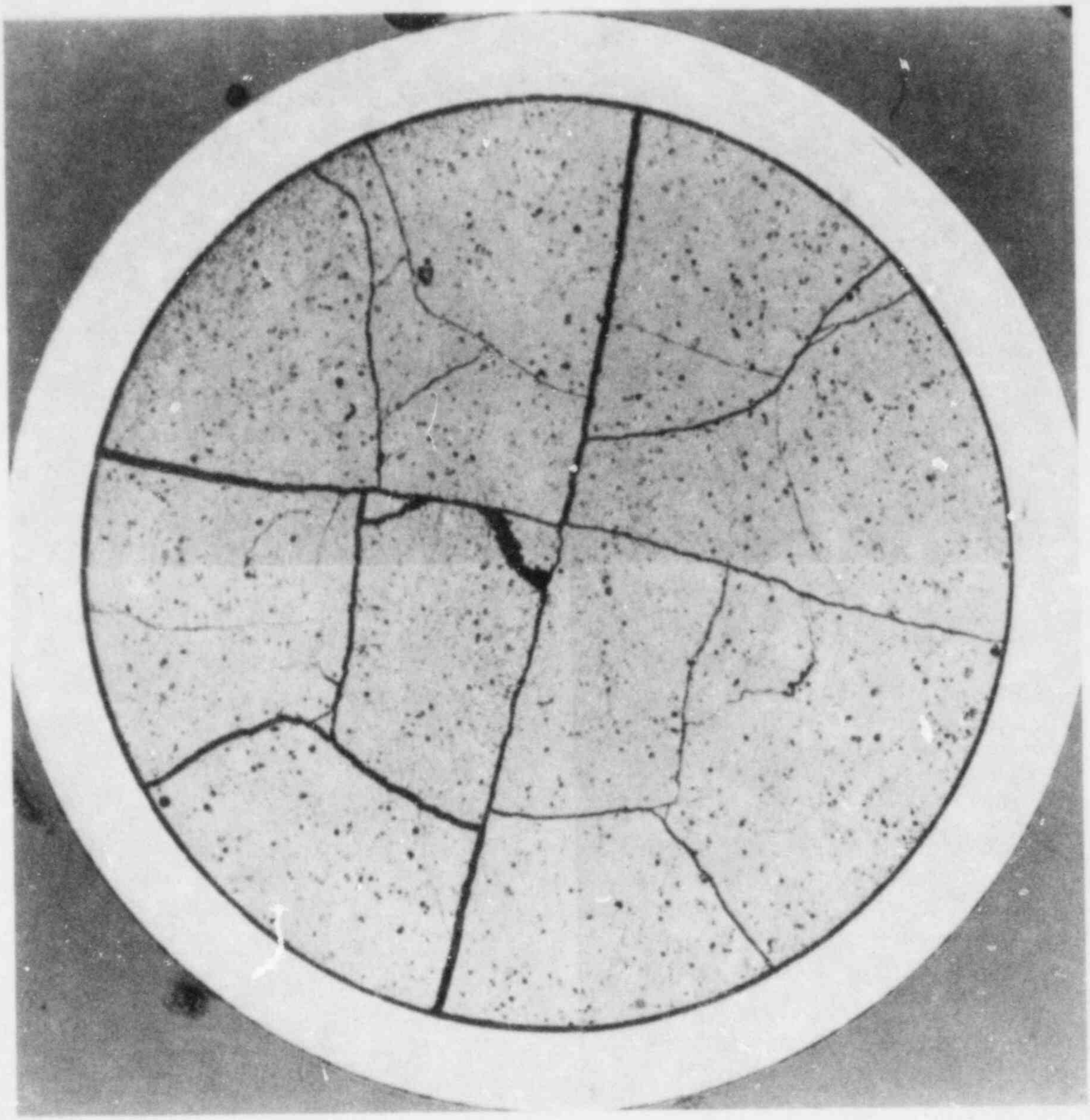


Figure 1. Cross Section from Lower End Region of Rod 1 of I FA-432

- mass spectrometer analyses of several samples along the lower thermocouple of rod 1 of IFA-432 to determine the distribution of transmutation products.

Final data reports for IFA-432 and -527 and an interim program summary report will be prepared after these examinations are completed.

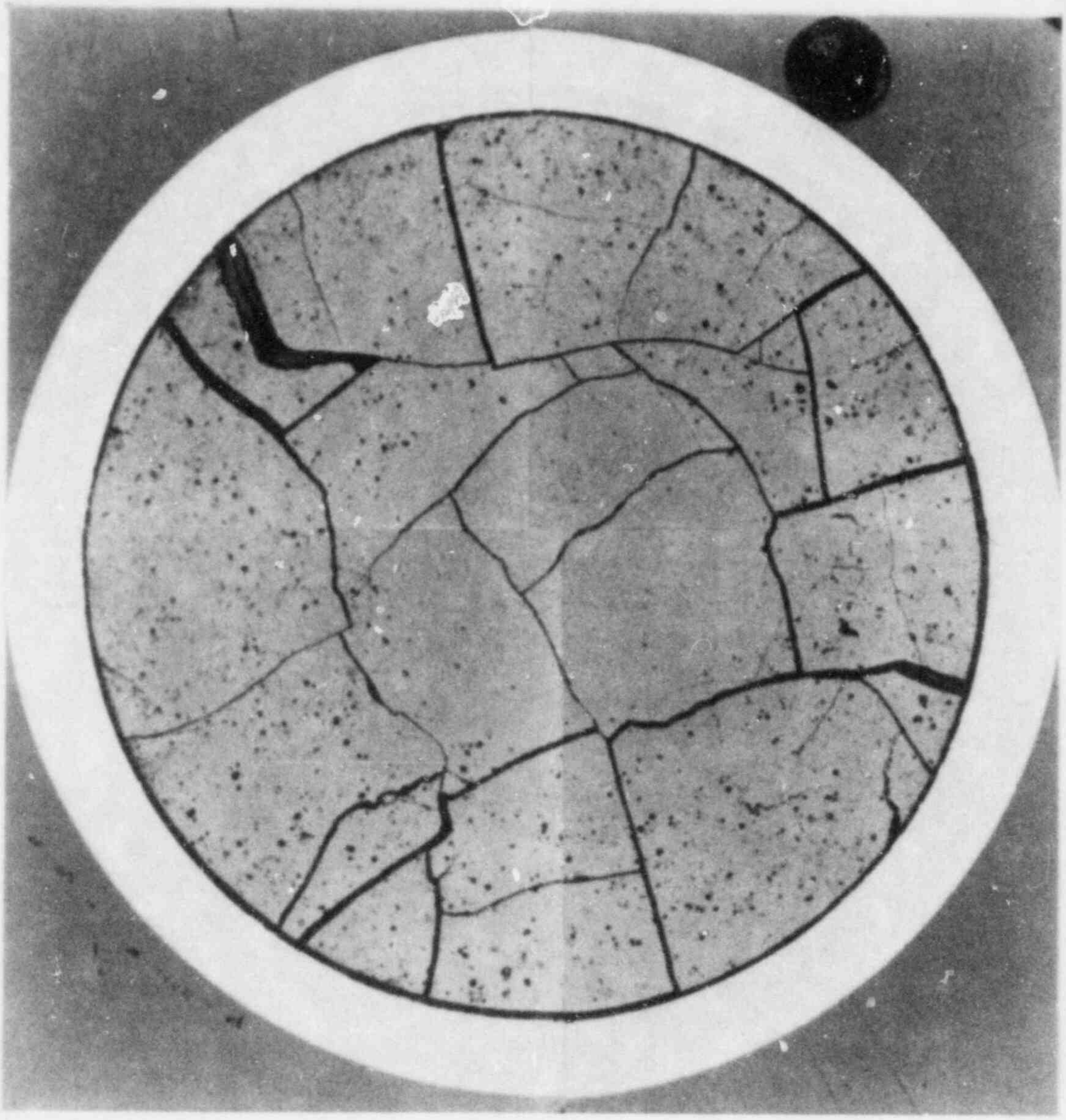


Figure 2. Cross Section from Lower End Region of Rod 6 of IFA-432

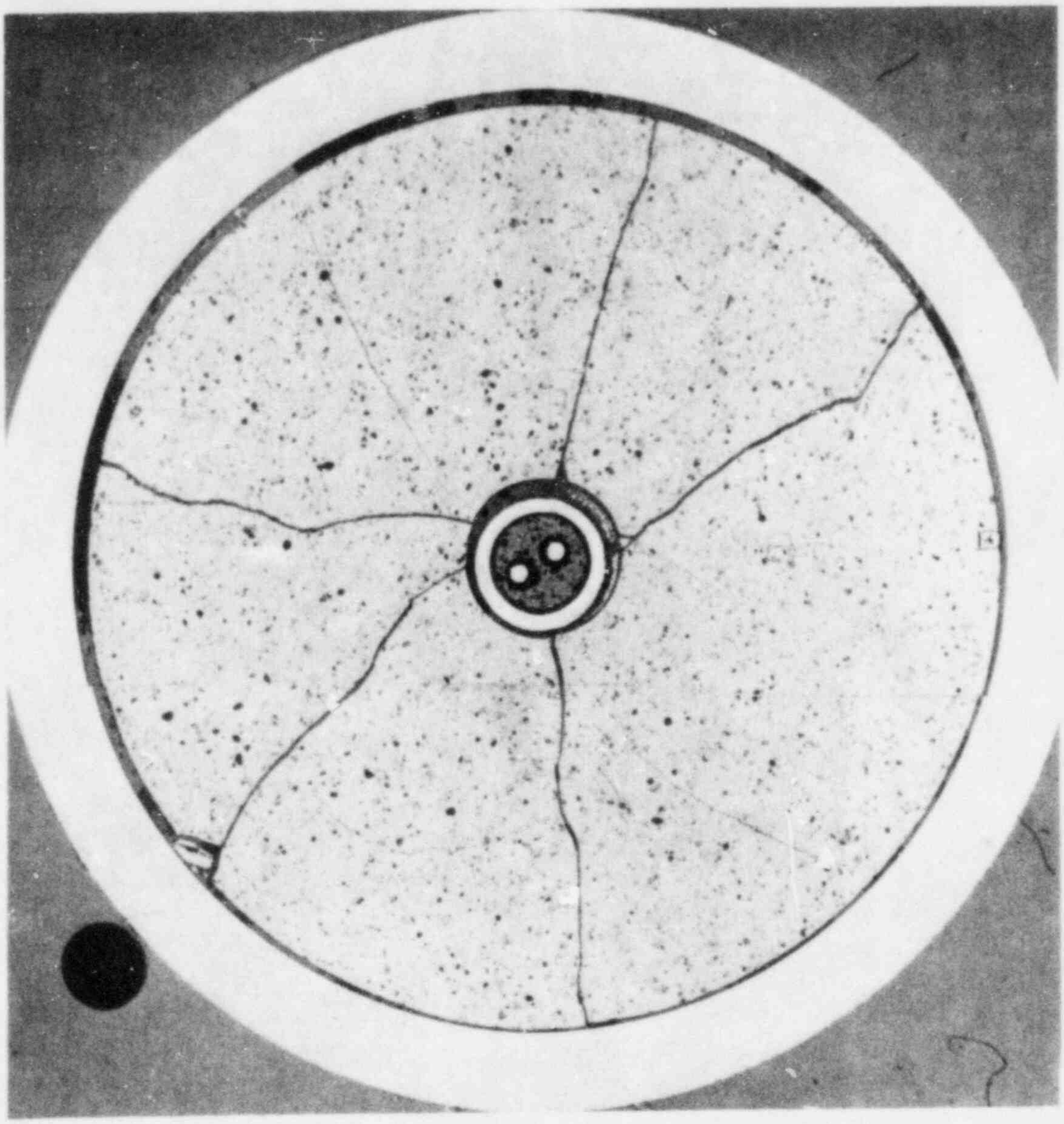
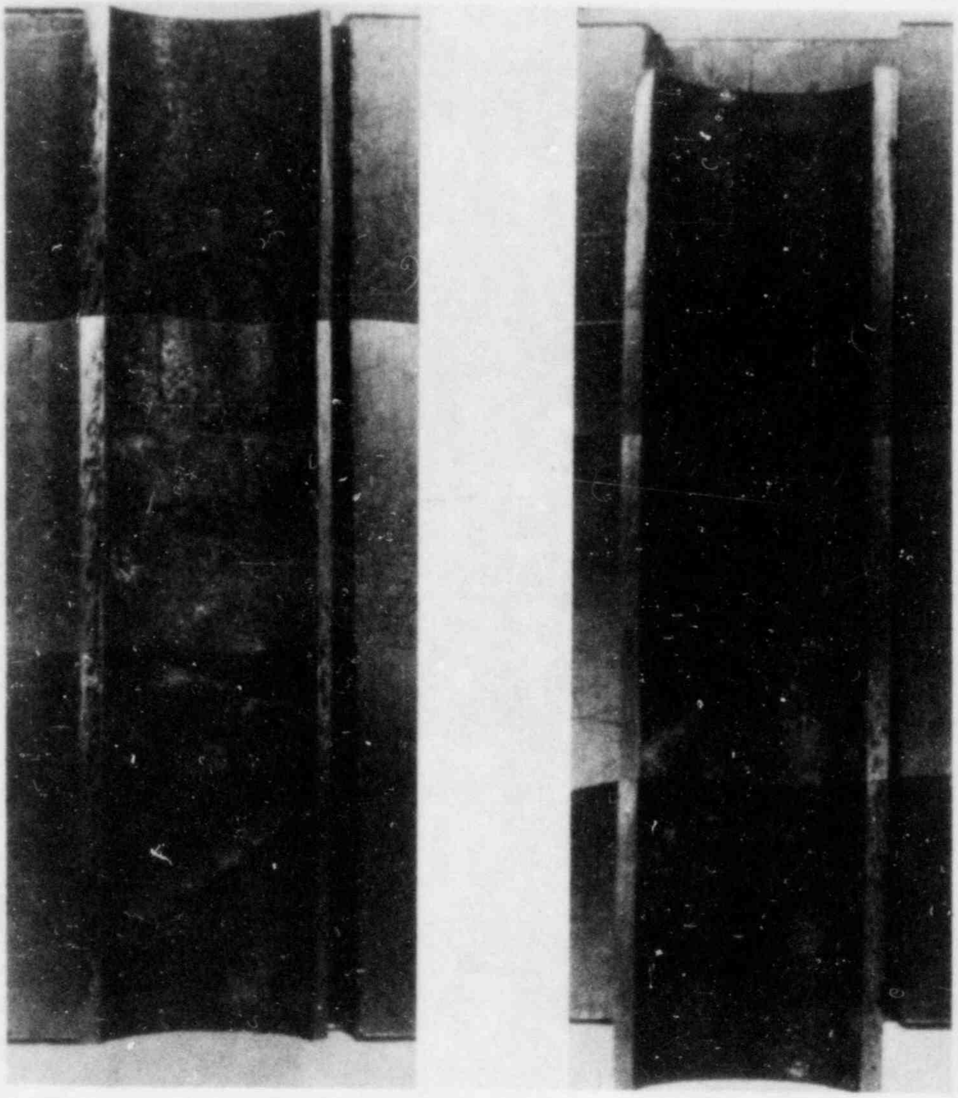


Figure 3. Cross Section from Rod 4 of IFA-527



(a) Upper Clamshell
Without Fuel

(b) Lower Clamshell
Without Fuel

Figure 4. Inner Surface of Rod 1 of IFA-432

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, work began on the final reports for IFA-432 and IFA-527, and the FASTGRASS fission gas release model was compared with fission gas release data from IFA-432.

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 Pacific Northwest Laboratory (PNL) program has the general objectives of 1) collecting and analyzing in-reactor data on fuel rod temperatures, fission gas release, and cladding elongation as a function of irradiation history; 2) correlating postirradiation examinations (PIEs) with in-reactor data; 3) utilizing ex-reactor testing for a better understanding of fuel rod mechanical behavior; and 4) integrating this information into the FRAPCON computer code series. The qualification and analysis of the data obtained from in-reactor testing of fuel rods is the responsibility of Task B, which has been divided into three subtasks:

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

TECHNICAL PROGRESS

This quarter's activities are discussed below by subtask.

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

SUBTASK B-1 - DATA PROCESSING

Rods 1 and 6 of IFA-432 were removed from the assembly for PIE following reactor shutdown in June 1981. In December 1981, the assembly returned to power with only four rods operating (2, 3, 9, and 5). A data tape containing IFA-432 data for the period from December 1981 through April 1982 has now been processed. Calculated local burnups (GWd/MTM) as of April 11, 1982, were:

Rod 2		Rod 3		Rod 9		Rod 5	
Upper	Lower	Upper	Lower	Upper	Lower	Upper	Lower
36.5	27.9	36.4	28.2	10.3	8.1	37.7	29.0

Data from the lower thermocouple of rod 5 of IFA-432 for the period from December 1980 to December 1981 were added to the data file (these data had not been transmitted by Halden previously).

The following IFA-432 instrumentation was still operating when the reactor was restarted: the lower thermocouples for rods 2, 3, and 5; the pressure transducer for rod 5; and the cladding elongation detector for rod 2. A check on the continuity of centerline temperature before and after the assembly change is presented in Figure 1, where thermal resistance for rods 2 and 3 is plotted as a function of time. No significant change occurred in the thermal response of the rods following the removal of rods 1 and 6, which means that the assembly power calibration (after accounting for the removal of two rods) is still applicable.

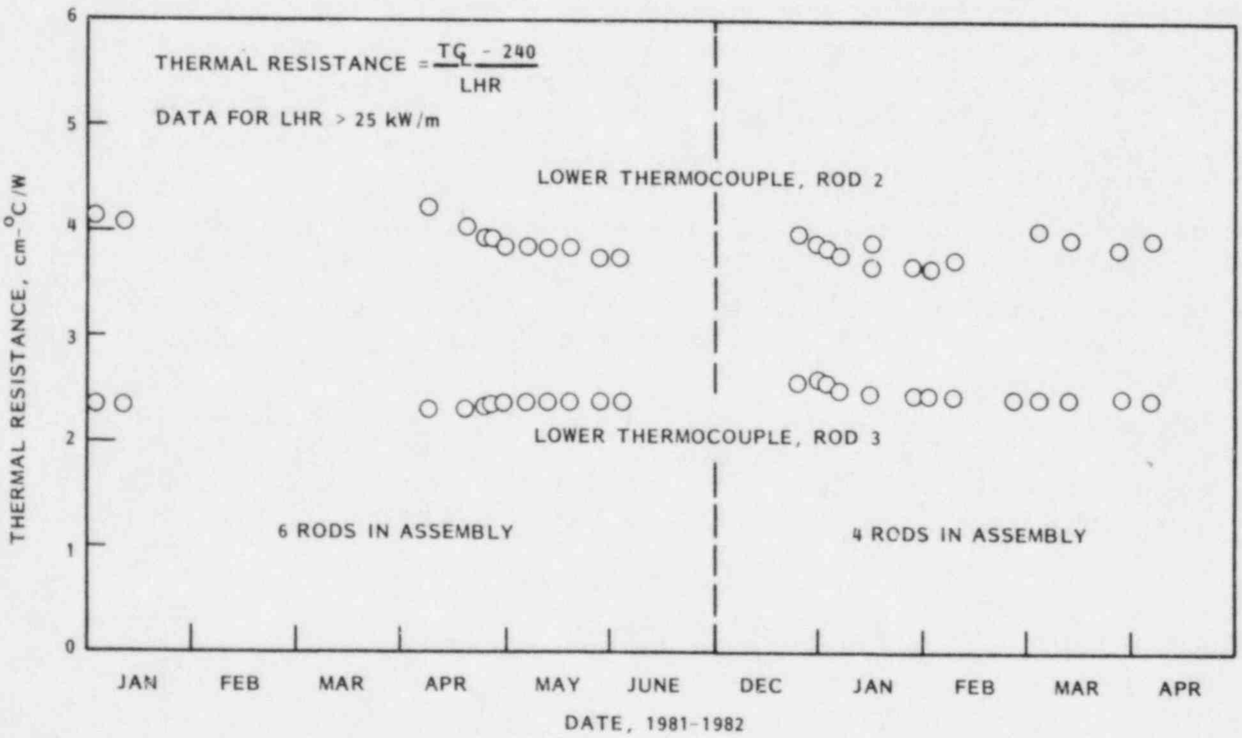


Figure 1. Illustration of Continuity of Power/Temperature Relationship for IFA-432 Rods After Removal of Rods 1 and 6

Although the power/temperature relationships remained constant, the pressure increased significantly for rod 5. Rod 5 pressures had been ~0.9 MPa for hot shutdown conditions and 2.5 MPa at high power (prior to the removal of IFA-432 from the reactor and the removal of two rods). After the assembly was reinserted, the pressures were ~1.6 and 3.3 MPa, respectively. The cause for this pressure increase has not yet been determined.

SUBTASK B-2 - DATA REPORTS

The final reports for IFA-432 and IFA-527 are being prepared. These reports will summarize the preirradiation characterization and the irradiation history and present PIE results. For IFA-432, tabular power/temperature histories will be provided that will be useful in standardizing code-to-data comparisons performed by the U.S. Nuclear Regulatory Commission (NRC) and others. Work on the precharacterization and irradiation sections is well advanced; the reports will be completed when the remaining PIE data are received.

A journal article entitled "Predicted and Observed Scatter in Fuel Rod Temperature of Replicate Rods During Irradiation" was prepared and submitted to *Nuclear Technology*. The paper was accepted and is scheduled for publication in March 1983. The following conclusions were reached in the paper:

- Data from replicate rod sets were consistent; and the observed data scatter was less than would be expected based on a consideration of dimensional tolerances, variation in material properties, and the uncertainty in operating power.
- Differences in behavior in rods from different assemblies are due to the effects of differences in operation. It may reasonably be assumed that rods in different assemblies that are operated approximately the same will behave approximately the same.
- Well-characterized data can be compared with computer code calculations, and uncertainty analysis can be used to identify conditions of code applicability.

SUBTASK B-3 - DATA ANALYSIS

Rods 1, 5, and 6 of IFA-432 were instrumented with pressure transducers to monitor the internal pressure changes that occurred during irradiation. The pressure data (obtained up to 30 GWd/MTM rod average burnup) have been used to estimate fission gas release as a function of burnup, and preliminary rod puncture data are now available from the PIE of rods 1 and 6. These releases are being compared with calculations based on fission gas release models that are currently being used in the FRAPCON-2 computer code.[1]

Fission gas releases for rods 1, 5, and 6 have been calculated using the latest version of the FASTGRASS fission gas release model. In making the calculations, the temperature and power histories were the same as those used previously for comparing the experimental data with other gas release model calculations.[2,3] Initial difficulties were encountered in using FASTGRASS; however, reducing the time step size to 4-h increments enabled successful calculations to be made. This dramatically increased the number of calculations needed; FASTGRASS required over 200 times more computer time than the other gas release models examined.

The calculated fission gas releases are compared with the experimental estimates derived from in-reactor pressure data and PIE measurements in Figure 2. The experimental gas release estimates were derived from the internal pressure data at low power levels by using the ideal gas law and assuming that the internal gas temperature was equal to the moderator temperature. The internal free volume of the fuel rods was varied as a function of burnup by allowing the fuel to densify during the first 2.5 GWd/MTM and swell linearly with burnup thereafter. Fuel densification was taken as 1% for rods 1 and 5 and 5% for the unstable fuel in rod 6. Fuel swelling was estimated from the free volume change in rod 8 of IFA-432 after 22 GWd/MTM burnup. It was also assumed that 1% gas

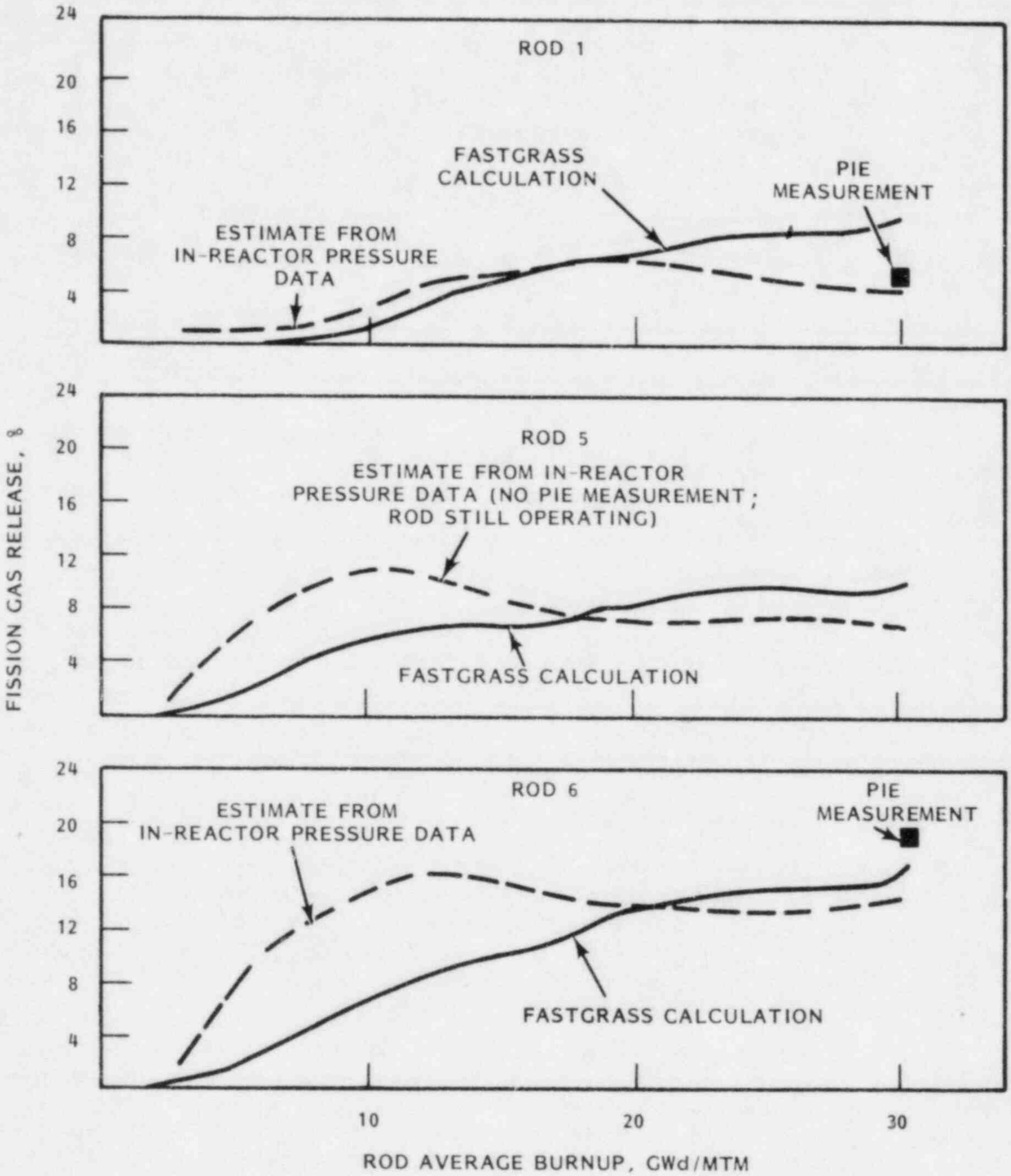


Figure 2. Comparison of FASTGRASS Calculations with IFA-432 Data

release occurred at 2.5 GWd/MTM and that helium production and release were similar to those estimated from the rod 8 PIE measurements.

Uncertainties in the internal free volume of the fuel rods and the measured pressures represent the major contributions to the uncertainties in the fission gas release estimates. The calculated free volumes at 30 GWd/MTM for rods 1 and 6 were both lower than those measured during the PIE of these rods, which tends to shift the experimental estimates upward (<2%). The error bars shown in Figure 2 at 20 GWd/MTM correspond to a ±0.1-MPa uncertainty in the measured pressures.

The FASTGRASS-calculated fission gas releases show reasonable agreement with the experimental data at 30 GWd/MTM for all three fuel rods. The calculations do not show the rapid increase in gas release that was observed in rods 5 and 6 below 10 GWd/MTM; however, none of the models examined followed the experimental gas release as a function of burnup for all three fuel rods. The FASTGRASS calculations show better overall agreement with the experimental data than the other gas release models examined.

FUTURE WORK

Data processing for the next quarter will continue as required. Work will continue on the final reports for IFA-432 and IFA-527; progress will depend on when PIE data are received from AERE-Harwell.

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1. Berna, G. A., et al. December 1980. *FRAPCON-2: A Computer Code for The Calculation of Steady State Thermal-Mechanical Behavior of Oxide Fuel Rods.* NUREG/CR-1845, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

2. 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.

3. Edler, S. K., ed. October 1982. *Reactor Safety Research Programs Quarterly Report, April-June 1982.* NUREG/CR-2716, Vol. 2, PNL-4275-2, Pacific Northwest Laboratory, Richland, Washington.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

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

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

M. E. Cunningham
W. N. Rausch

SUMMARY

Version 1 Modification 4 (V1M4) of FRAPCON-2 has been delayed because of difficulties in linking the FASTGRASS fission gas release model to the code. Progress was made, however, in improving deformation predictions by the PELET/RADIAL mechanical model and thermal predictions by the FRACAS-II thermal/mechanical model.

Two formal reports were issued this quarter. The first report summarizes ex-reactor irradiated/unirradiated fuel rod compliance tests and other tests used to verify the basic parameters of the PELET/RADIAL model in FRAPCON. The second report describes a computerized three-dimensional (3-D) model of the mechanical interaction of fuel pellet fragments with the cladding.

INTRODUCTION

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

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

Task C efforts include: code maintenance, analysis of data from fuel mechanics experiments, and pellet-cladding interaction (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

V1M4 of FRAPCON-2 will not be issued in October 1982 because of difficulties encountered in linking the latest FASTGRASS fission gas release model to the code. Other improvements to be included in V1M4 have been completed and are discussed below.

A long-standing deficiency in the PELET mechanical package has been the failure of the model to predict the cladding creepdown for commercial pressurized water reactor (PWR) rods that is commonly observed. This deficiency stems from the modeling assumption of constant fuel-cladding contact and the consequent mechanical support the fuel is predicted to provide to the cladding, even at low powers and low fuel temperatures. This situation has been remedied by simply lowering the minimum values of radial and axial effective elastic moduli achievable by the model. The code logic still raises these moduli far above the minimum values at nominal-to-high powers, when fuel temperatures are relatively high and differential fuel-cladding thermal expansion reduces the free volume within the rod. However, at the low powers typical of commercial irradiations, the fuel moduli remain low enough to permit cladding creepdown in PWR rods commensurate with what is typically observed. The trend of predicted positive deformations versus power (observed in test rods) has actually improved as a consequence of this change.

Another long-standing deficiency of FRAPCON-2 has been the severe underestimation of fuel centerline temperatures for xenon- or fission gas-filled rods at low powers when the FRACAS-II thermal/mechanical option is used. The predicted temperatures at high power (above 30 kW/m) fall very close to measured values for such rods (especially after 3 to 5 GWd/MTM burnup); but at low power (below 20 kW/m), the calculated temperatures may be less than measured values by several hundred degrees Kelvin. This discrepancy in the shape of the temperature versus power curves for these xenon/fission gas-filled rods was traced to unrealistic FRACAS-II predictions of the variation of gap size with power. These predictions in turn were traced to the particular procedure used in the iteration loop that balances fuel temperatures and gap size (the innermost loop of the code).

At each time step and for each axial node, the iteration on thermal gap size was started from the assumption of zero physical gap size. This so biased the converged solution that small hot gaps and very little gap size variation with power were predicted, which accounted for the correspondence of the code with fuel temperature data at high powers (when true gap sizes are indeed small) and the discrepancy of the code with data at low power (where gap sizes are relatively large). The effect was not noticed when the code was compared with temperatures from helium-filled rods because the thermal resistance of the gap is a much smaller fraction of the total thermal resistance for those rods.

This situation was remedied by starting the gap size iteration at the most recent local estimate of the (unrelocated) hot gap, which led to much more realistic converged gap sizes and variations of gap size with power. Figure 1 shows the improvement in calculated centerline temperatures for the low-power xenon-filled rods in instrumented fuel assembly (IFA)-527.

FRAPCON-2 and other fuel performance codes were compared with Halden test fuel data at the June 1982 Enlarged Halden Program Group meeting, as reported under Task B last quarter. That comparison will be much improved by the corrections noted above.

SUBTASK C-2 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

Two formal reports were completed this quarter: *Experimental Verification of a Cracked Fuel Mechanical Model* (NUREG/CR-2872, PNL-4377) and *A Model for the Localized Mechanical Behavior of Cracked Nuclear Fuel Pellet Systems Status Report* (NUREG/CR-2931, PNL-4451). A few minor errors will be corrected in NUREG/CR-2872, and it will be sent to the NRC by mid-October for printing and distribution. NUREG/CR-2931 has been forwarded to the Fuel Behavior Branch for comment and is scheduled for printing in November. These reports complete the two remaining FY 1982 mile-

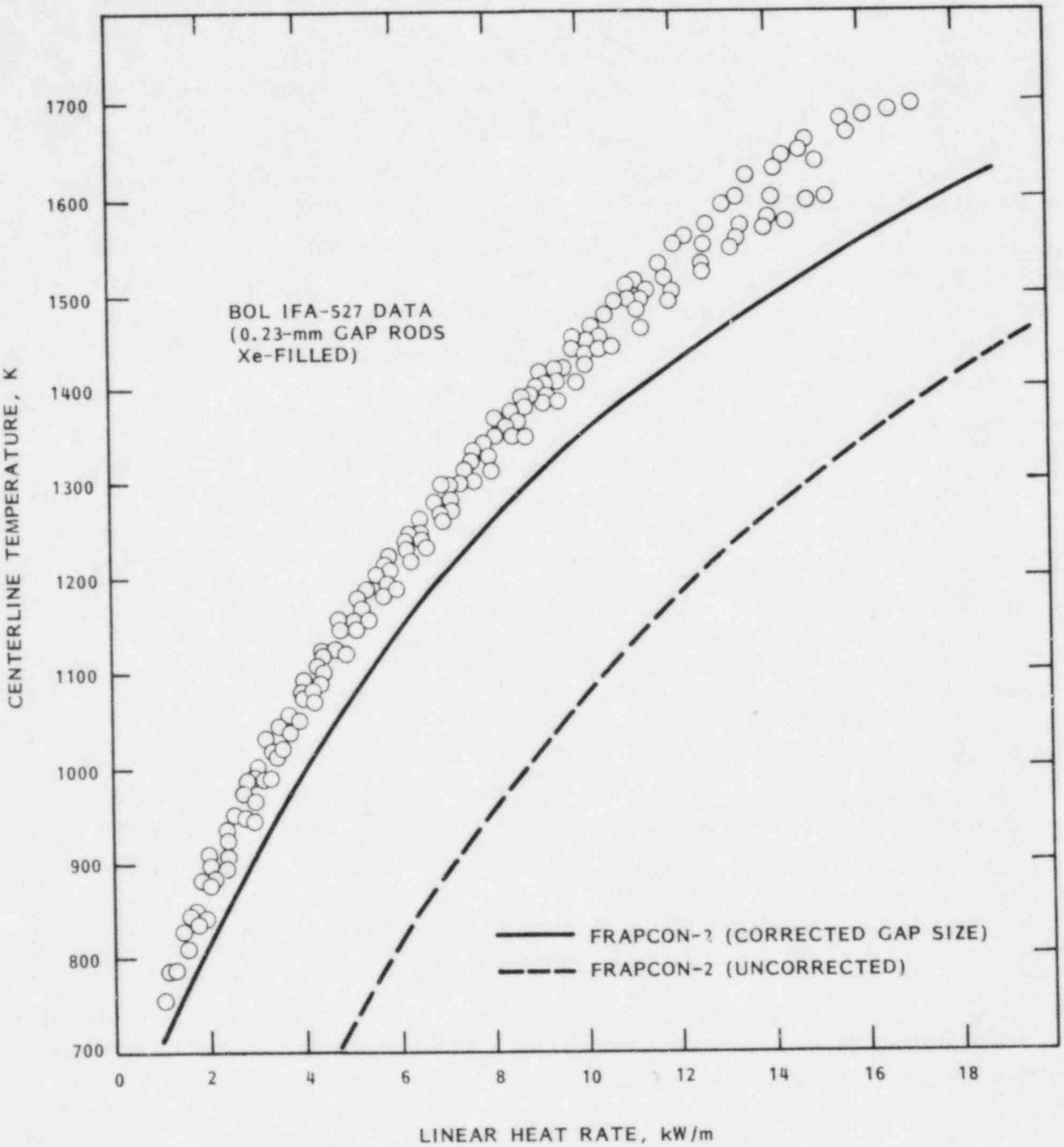


Figure 1. Comparison of FRAPCON-2 (FRACAS-II) Calculations Before and After Correction of the Gap Size Iteration Versus Fuel Centerline Data from IFA-527

stones for Subtask C-2, and future work to complete a fuel failure model will be conducted under a separate project. A brief description of each report follows.

NUREG/CR-2872 describes the results of a series of laboratory experiments conducted to independently verify a model that describes the rod average nonlinear mechanical behavior of cracked fuel in pelletized UO_2/Zircaloy nuclear fuel rods under normal operating conditions. The analytical model is briefly described, and each experiment is discussed in detail.

The experiments were conducted to verify the general behavior and numerical values for the three primary independent modeling parameters (effective crack roughness, effective gap roughness, and total crack length) and to verify the model predictions that the effective Young's moduli for cracked fuel systems were substantially less than those for solid UO_2 pellets.

In general, the model parameters and predictions were confirmed, and new insight was gained concerning the complexities of cracked fuel mechanics. Other significant findings were that:

- Increased fuel cracking may cause an axial localization of fuel-cladding mechanical interaction (FCMI).
- Fuel-cladding axial slipping may affect in-reactor fuel rod relaxation studies.
- Effective crack roughnesses depend on crack surface morphology (waviness).
- Large cladding stress concentrations may be caused by asymmetric fuel fragment ratcheting during relocation.
- An apparently anomalous relationship exists between fuel rod bowing and cladding diametral deformations.

NUREG/CR-2931 reviews the current developmental status of a computerized mathematical model (FRAGMT) that simulates the **localized** mechanical interactions that occur between the cracked UO_2 pellets and the Zircaloy cladding of light-water reactor fuel rods during normal operation. This mechanical interaction causes the diametral ridges and axial elongations in the cladding that determine fuel rod performance and affect rod failure propensity. FRAGMT models the simultaneous occurrence of cladding plastic deformations, axial slipping between the cracked fuel column and the cladding, and fuel fragment ratcheting (radially inward or outward discontinuous rigid body motion) for a single fuel fragment over a half-pellet length.

The results of two test cases are presented. Significant conclusions include:

- Fragment ratcheting is important for calculating cladding deformations and depends primarily on the normal and friction forces at the radial fuel cracks, which in turn depend on fragment size.
- Smaller fragments appear to be more difficult to recondition (inward ratcheting); 100% reconditioning is not possible.
- Circumferential fuel cracks tend to persist, even at higher rod powers.
- Axial slipping can result in rod behavior that resembles in-reactor fuel rod relaxation data.
- Friction coefficients between the cracked fuel columns and the cladding are relatively large (1.0 to 1.5).
- FCMI can vary axially along the rod length.

FUTURE WORK

The following activities are planned for the next quarter. V1M4 of FRAPCON-2 will be issued, including coding changes, user's letter, and results of verification runs. The development of FASTCON—a trimmed-down and faster-executing version of FRAPCON-2—will begin.

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 first unirradiated cladding tests in the pressurized loop were conducted. The tungsten heater rod failed due to a possible chemical reaction, which necessitated additional design work. Ultrasonic cleaning equipment for the irradiated cladding samples was received, and weld development for attaching unirradiated cladding extensions to the irradiated cladding was completed.

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 considerable experimental flexibility at a cost much lower than in-reactor experiments. The relationships among 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, the strain instrument, and the fuel simulators will be proof tested in fiscal year (FY) 1982 using unirradiated cladding; actual data collection with irradiated cladding will begin in FY 1983. These data will complement Task C efforts and provide a means of verifying PCI models. In Task D-1, a measurement instrument capable of characterizing the elastic-plastic deformations of the simulated fuel rod at power within the loop will be designed and produced.

TECHNICAL PROGRESS

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

SUBTASK D-1 - ROD STRAIN INSTRUMENT

The rod strain instrument is designed to simultaneously measure two orthogonal cladding diameters and the fuel rod bow. Work on this subtask has now been completed.

SUBTASK D-2 - LOOP EXPERIMENTS

During the last quarter, the first power-on hot-water pressurized loop tests on unirradiated cladding were performed. The first test was a hydrostatic test (zero rod power) to test instrument performance

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

and check for leaks in the loop piping. The instrument drive was found to have friction problems at the pressure seal; a high-pressure, high-temperature lubricant at the drive shaft seals solved this problem. The instrument readings were subject to some noise from coolant flow, temperature effects, and instrument drive motors; solutions to these problems are being studied.

The second test on unirradiated cladding used a rod power equivalent to 23 kW/m and room temperature coolant. Fluid bouyancy effects may have caused hot spots at the top of the rod that induced a temperature/resistivity/power instability in the internal tungsten heater rod. As the power was decreased, the heater rod fractured. Post-test examination of the fuel rod components revealed evidence of a tungsten-UO_2 chemical reaction. At the localized failure point, a significant reduction in the tungsten heater rod diameter occurred and a slight diameter reduction was also observed over most of the length of the tungsten rod. The localized nature of the failure point suggests a chemical reaction assisted by a contaminant (for example, water or a hydrocarbon). These possibilities are being studied.

Work has progressed on preparing irradiated cladding samples. The originally received equipment for ultrasonic cleaning of the cladding was defective, but replacement equipment has now been received. A pellet design for the tests was completed, and pellet fabrication is nearly finished. Weld development for attaching unirradiated cladding extensions to the irradiated cladding samples was also completed.

FUTURE WORK

In the coming quarter it is expected that the design and materials problems associated with the fuel simulator and the strain measurement instrument will be resolved. Further tests with unirradiated cladding are planned to verify the solutions. The irradiated cladding samples should be ready for testing (separate simulators constructed) by the end of the quarter.

PIPE-TO-PIPE IMPACT[a]

M.C.C. Bampton, Project Manager

J. M. Alzheimer
F. A. Simonen

SUMMARY

Testing with specimens at pressurized water reactor (PWR) conditions continued. At least one test was completed from all except one group in the matrix. Two pipe ruptures occurred with heavier pipes impacting lighter pipes, which was not unexpected based on current U.S. Nuclear Regulatory Commission (NRC) licensing criteria.

INTRODUCTION

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

This Pacific Northwest Laboratory (PNL) program involves two main areas: obtaining experimental data and developing predictive models. Preliminary analyses to determine significant test parameters and required energies and pipe velocities have been completed. The test matrix has been developed; a system capable of accelerating the pipe has been built; and the test facility has been designed. The current phase of the program encompasses the actual testing. Predictive models that are analytically based and/or empirical fits of the data will be developed and compared with current licensing criteria.

TECHNICAL PROGRESS

Testing with specimens at PWR conditions continued. At least one test was completed from each group in the matrix except for one group (see Table 1). The two group 4b tests with the highest velocities resulted in ruptures of the impacted pipes. No other ruptures occurred even though some specimens had significant deformation.

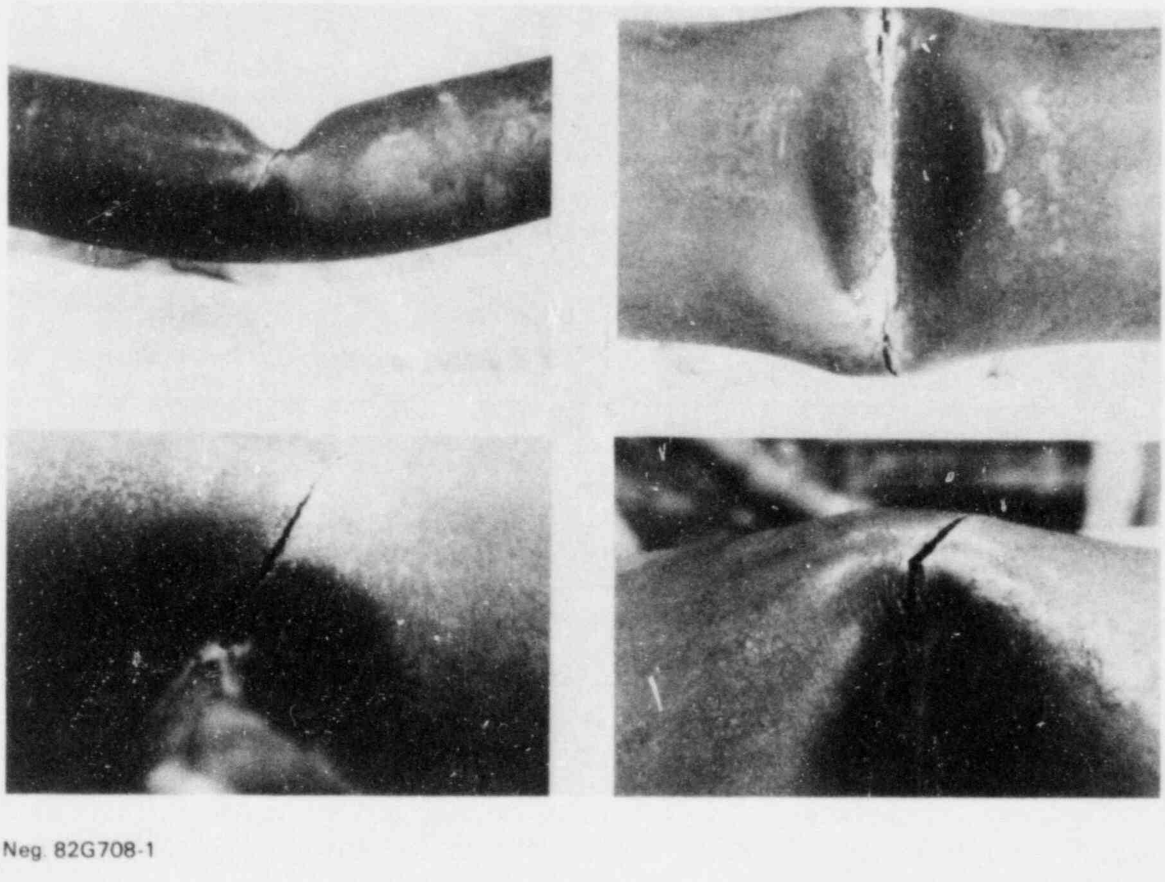
The two ruptures were for pipe combinations for which current NRC criteria would have required a rupture to be postulated. Current NRC licensing criteria (Standard Review Plan 3.6.2) stipulate that the failure of the impacted pipe must be assumed if it is smaller in diameter or thinner in wall thickness than the swinging pipe. The two ruptures in group 4b were for impacted pipes of the same diameter but thinner wall thicknesses. The first pipe rupture occurred with a pipe tip velocity of 180 feet per second (fps). The rupture was a circumferential 1-in. long crack and was in an area of high wall bonding. The second rupture occurred with a pipe tip velocity of 172 fps. The pipe did not rupture upon the initial impact but upon the second impact after the swinging pipe had rebounded ~3 ft, which indicates that the pipe was very close to failure from the initial impact (a very small amount of energy is associated with the second impact). The pipe ruptured at two cracks that were each ~4 in. long (see Figure 1). These cracks were much larger than those in the other ruptured pipe even though the

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

impact velocity had been lower. This result may be due to a slightly higher test temperature or material variations. For all of the tests from group 4b, the swinging pipe was virtually undeformed; therefore, almost all of the energy was absorbed in deforming the target pipe.

Table 1. Pipe-to-Pipe Impact Test Matrix I

Group	Swinging Pipe	Target Pipe	Tests Completed	Test Velocities, fps
1	6-in. Schedule 40	6-in. Schedule 40	all	various
2	6-in. Schedule 40	6-in. Schedule 40	1	185
3	6-in. Schedule 80	6-in. Schedule 80	4	180, 172, 149, 140
4a	6-in. Schedule 40	6-in. Schedule 80	1	217
4b	6-in. Schedule 80	6-in. Schedule 40	3	180, 172, 159
5a	6-in. Schedule 120	12-in. Schedule 60	0	—
5b	6-in. Schedule 80	3-in. Schedule 160	1	138



Neg. 82G708-1

Figure 1. Cracks in 6-in. Schedule 40 Pipe Resulting from Impact by a 6-in. Schedule 80 Pipe

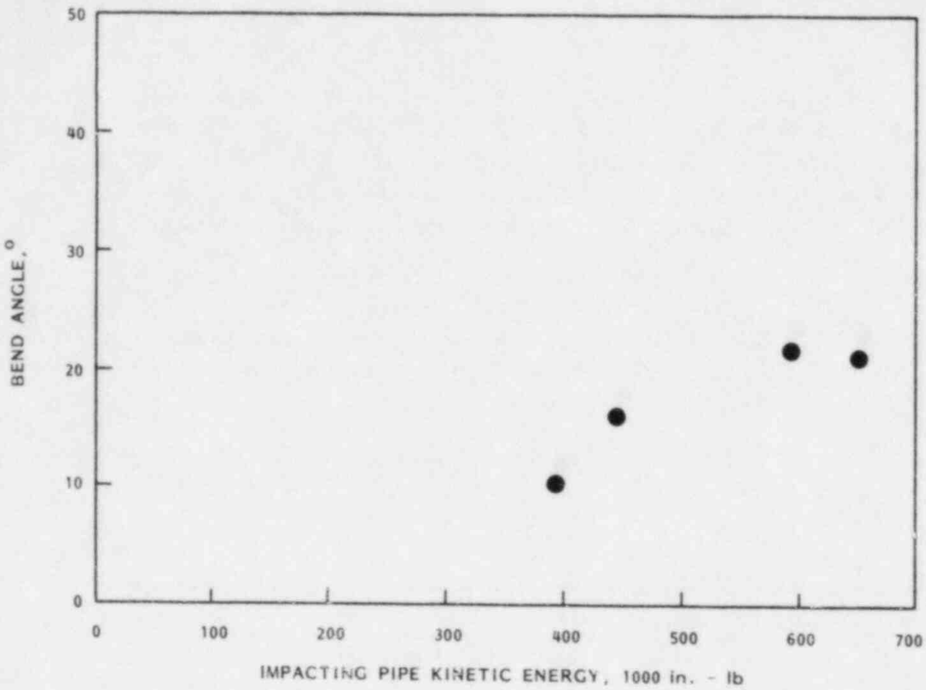
Following the second rupture, the sunflower seed oil that was being used as a pressurizing fluid was blown through the cracks and atomized. The vapor cloud ignited shortly thereafter causing a brief but intense fire that was quickly extinguished. The damage from the fire was contained to a few burnt instrument leads and a few burnt slats in the fence. The system for pressurizing the pipe was modified to significantly reduce the amount of oil that will escape following a rupture.

From the data trends for tests from groups 2 and 3, (see Figures 2 and 3) there appears to be some scatter in both the diameter changes and the bend angles. However, a straight line drawn through the origin would result in a good first-cut correlation of the data. Data from the strain gages, strain circles, and support loads are being analyzed. Unfortunately, the strain circles on the ruptured pipes were destroyed by the fire.

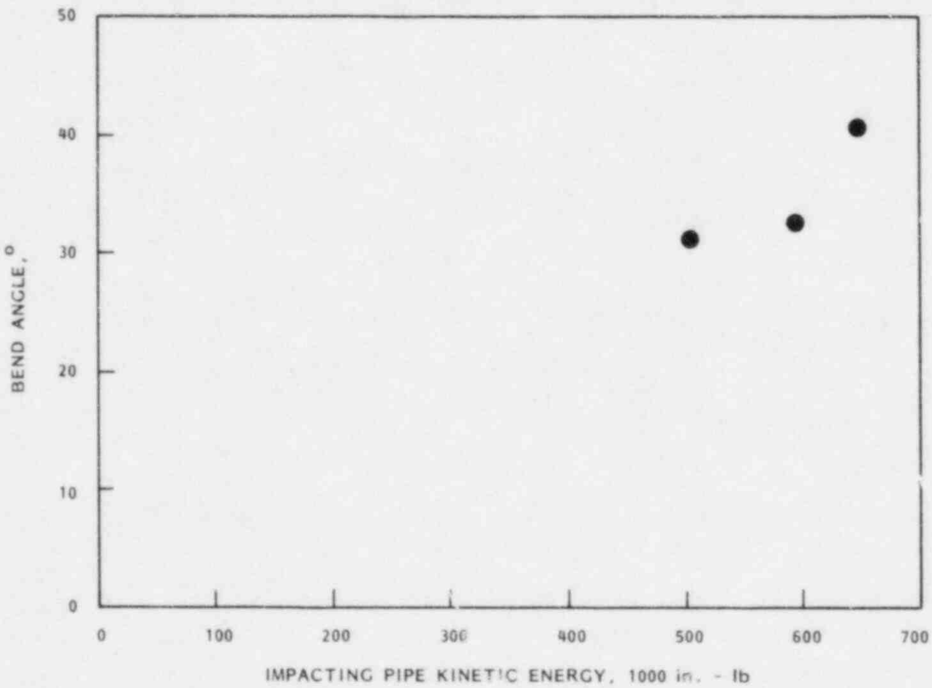
FUTURE WORK

The following activities are planned for the next quarter:

- complete remaining hot oil tests
- report results of tests and analysis.

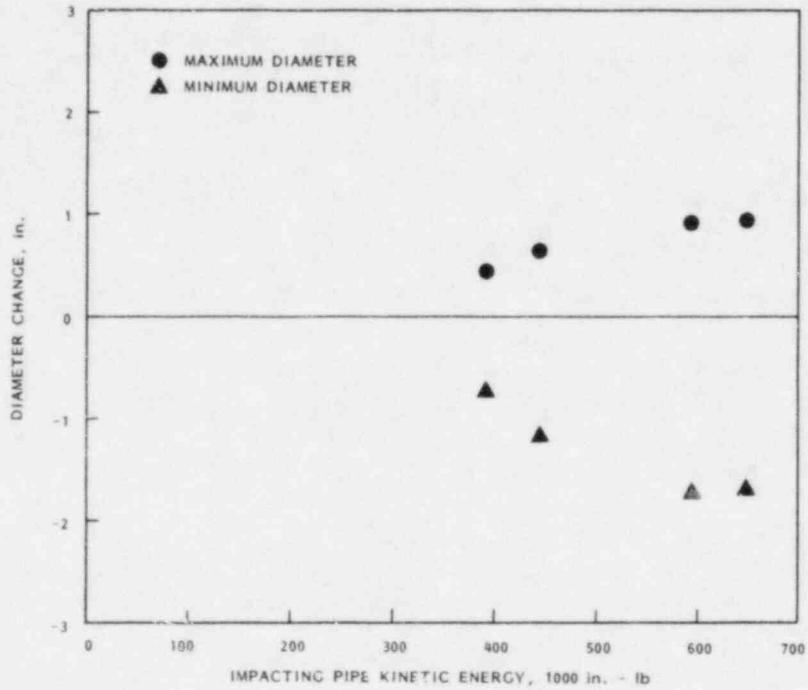


a) 6-in. Schedule 80 Pipe Impacting 6-in. Schedule 80 Pipe

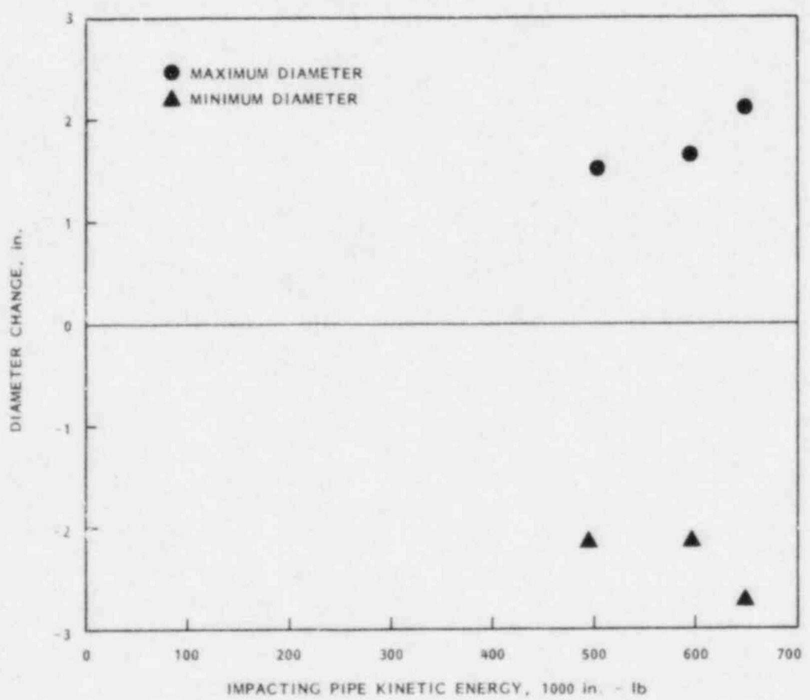


b) 6-in. Schedule 80 Pipe Impacting 6-in. Schedule 40 Pipe

Figure 2. Effects of Impact Energy on Impacted Pipe Bend Angle



a) 6-in. Schedule 80 Pipe Impacting 6-in. Schedule 80 Pipe



b) 6-in. Schedule 80 Pipe Impacting 6-in. Schedule 40 Pipe

Figure 3. Effects of Impact Energy on Impacted Pipe Deformation

SEVERE CORE DAMAGE SUBASSEMBLY PROCUREMENT PROGRAM

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

R. L. Goodman, Project Manager

G. S. Allison, T. M. Fish, A. J. Anthony, S. O. Bates, L. L. King, L. J. Parchen, C. E. Bigelow, R. F. Klein, J. O. Vining, L. R. Bunnell, R. K. Marshall

SUMMARY

The Pacific Northwest Laboratory (PNL)/Power Burst Facility (PBF) SFD-1 test train assembly was completed and delivered on schedule to the Idaho National Engineering Laboratory (INEL) on September 15, 1982.

PNL has agreed to fabricate, assemble, and outfit the fuel bundle and insulated shroud assemblies for the PBF SFD-2 test train assembly. Final assembly of the standard fuel rods for the SFD-2 fuel bundle assembly and fabrication of the majority of the instrumented rods were completed; and insulated shroud fabrication, assembly, and outfitting began. Arrangements are being made to deliver the SFD-2 fuel bundle and insulated shroud assemblies to INEL on December 1, 1982.

Internal design reviews were held at PNL, Richland, Washington, during September 1982 for the SFD-3 and SFD-4 fuel bundle and insulated shroud assemblies. All component hardware design 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 SFD-3 and SFD-4 assemblies will be held in a joint PNL/EG&G Idaho, Inc., meeting in late October 1982.

INTRODUCTION

The Severe Core Damage (SCD) Subassembly Procurement Program includes the design, development of appropriate materials and supporting fabrication processes, and complete fabrication of fully instrumented test train assemblies for the U.S. Nuclear Regulatory Commission (NRC)-sponsored test program at PBF. Many portions of the PBF work should directly benefit the ESSOR and National Research Universal (NRU) reactor programs 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 such accidents.

TECHNICAL PROGRESS

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

(a) FIN: 2084-2; NRC Contact: R. Van Houten.

TEST TRAIN ASSEMBLY COMPONENT HARDWARE

Final assembly of component hardware for the SFD-1 test train assembly was completed. The test train assembly was delivered to INEL on September 15, 1982, on a chartered DC-3 aircraft (see Figure 1). Figure 2 shows the completed test train assembly hanging vertically from the assembly straightness fixture. Specialized instrumentation includes cladding inner diameter thermocouples; fuel centerline thermocouples; steam probe assemblies; fuel rod pressure transducers; fuel rod pressure switches; inlet, outlet, and differential coolant thermocouples; shroud inner liner and mid-insulation thermocouples; a shroud molten metal penetration detector (MMPD); and a specially developed temperature profile detector (TPD) assembly.

SFD-3 AND SFD-4 TEST ASSEMBLY DESIGN

In early July 1982, PNL, NRC, and EG&G Idaho, Inc., established the work scope and experiment design basis and criteria to prepare instrumented fuel element test assembly modules for the NRC-sponsored SCD fuel behavior irradiation tests in the PBF. PNL will design the fuel bundle assemblies to accommodate preirradiated rods and simulated instrumented control rod assemblies and will fabricate, assemble, and outfit the SFD-3 and SFD-4 test fuel bundle and insulated shroud assemblies for the PBF tests.

The SFD-3 fuel rod bundle will consist of 26 preirradiated test fuel rods, two fully instrumented fresh fuel rods, and four control rod guide thimbles. The fuel rod bundle for the SFD-4 test will be identical except that it will contain simulated control rod assemblies in the four control rod guide thimble locations. The instrumentation array for the SFD-3 and SFD-4 assemblies is shown in Figure 3.

Internal design reviews were held at PNL for the SFD-3 and SFD-4 fuel rods, control rods, and fuel bundle and insulated shroud assemblies. All design drawings have been completed, approved, and released for fabrication, assembly, and outfitting activities.

An intermediate design review in support of the SFD program was held in Idaho Falls with EG&G Idaho, Inc., on August 18-20, 1982. The design details for the preirradiated fuel bundle assembly were reviewed, and the preliminary design for the insulated shroud and control rod assemblies was presented. A definition of design interfaces was also discussed for the SFD-3 and SFD-4 test train assemblies, and the proposed remote assembly/disassembly operations for the fuel bundle and insulated shroud assemblies based on the PNL design were also reviewed.

The approved designs for the PBF SFD-3 and SFD-4 fuel bundle assembly components are presented in Figures 4 through 7.

FUTURE WORK

- The PNL/PBF SFD-2 fuel bundle and insulated shroud assemblies will be delivered to INEL on December 1, 1982.
- The final formal design review of the entire drawing package for the SFD-3 and SFD-4 fuel bundle and insulated shroud assemblies will be held in a joint PNL/EG&G meeting.
- Material and fabricated component hardware procurement for the SFD-3 and SFD-4 fuel bundle and insulated shroud assemblies will continue.

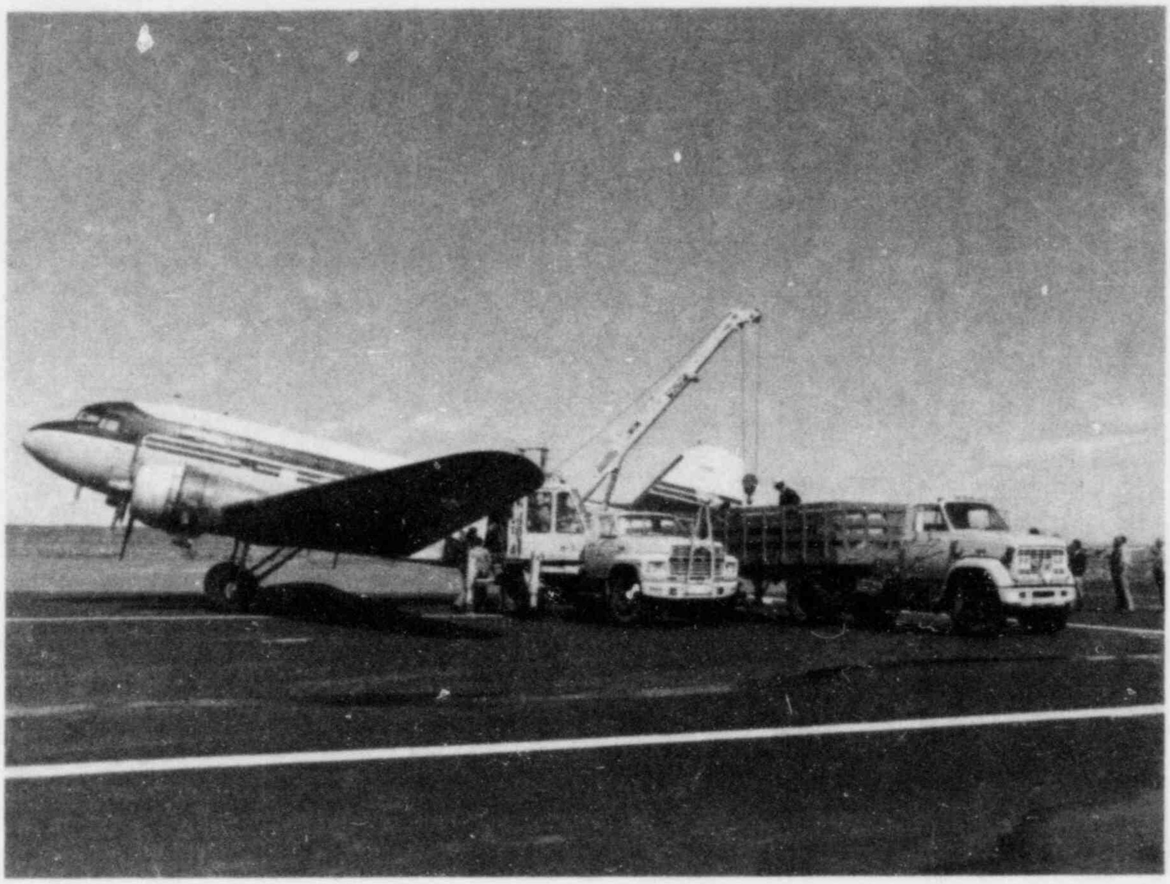


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Figure 1. Loading PNL/PBF SFD-1 Test Train Assembly for Shipment to INEL

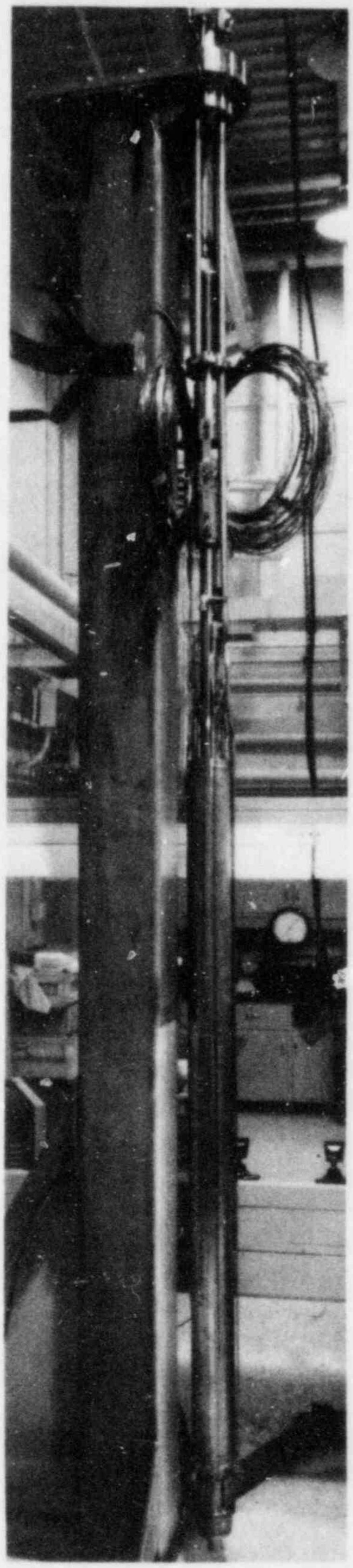

Figure 2. Final PNL/PBF SFD-1 Test Train Assembly

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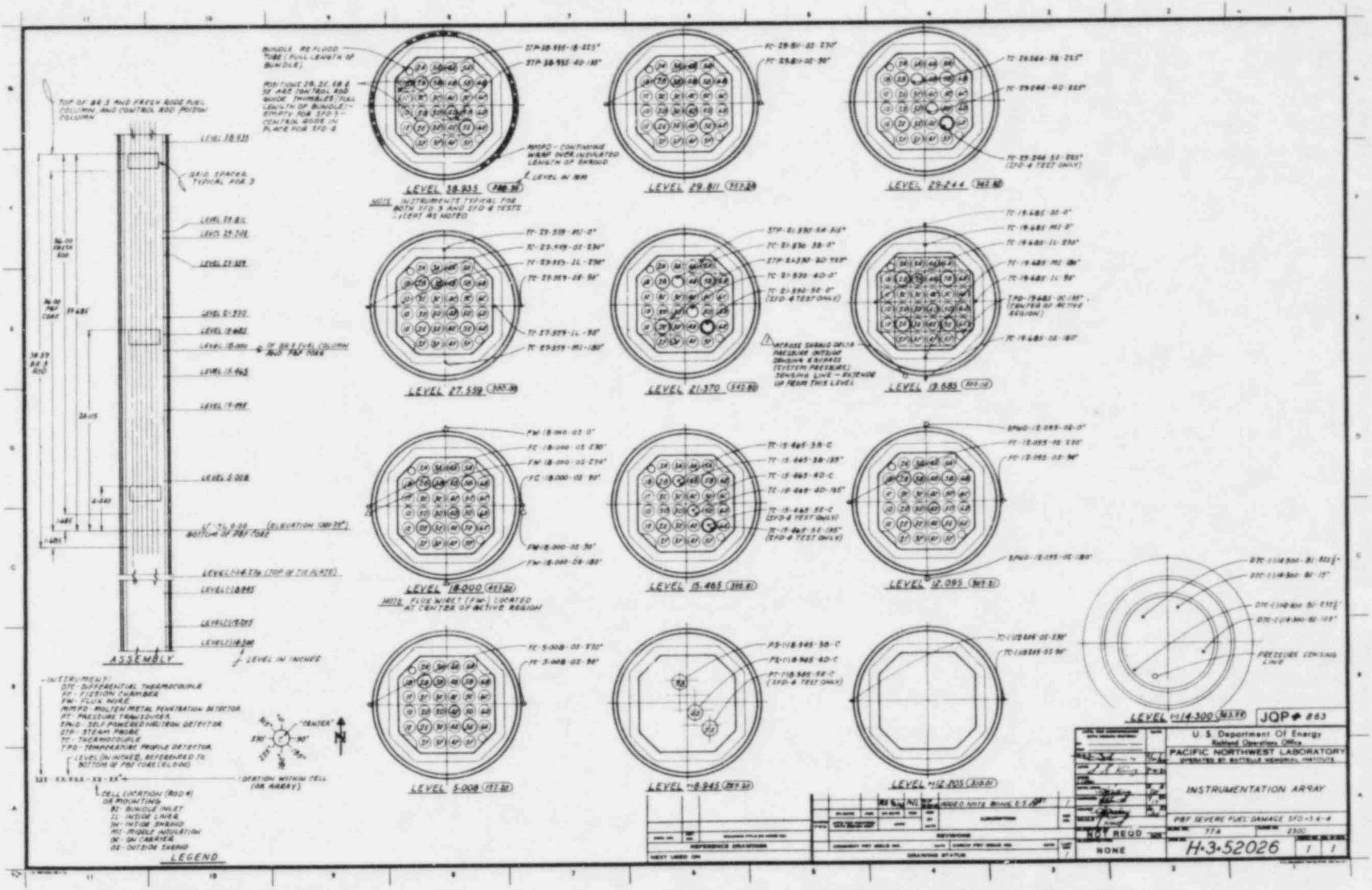

Figure 3. SFD-3 and SFD-4 Fuel Bundle and Insulated Shroud Assembly Instrumentation Array

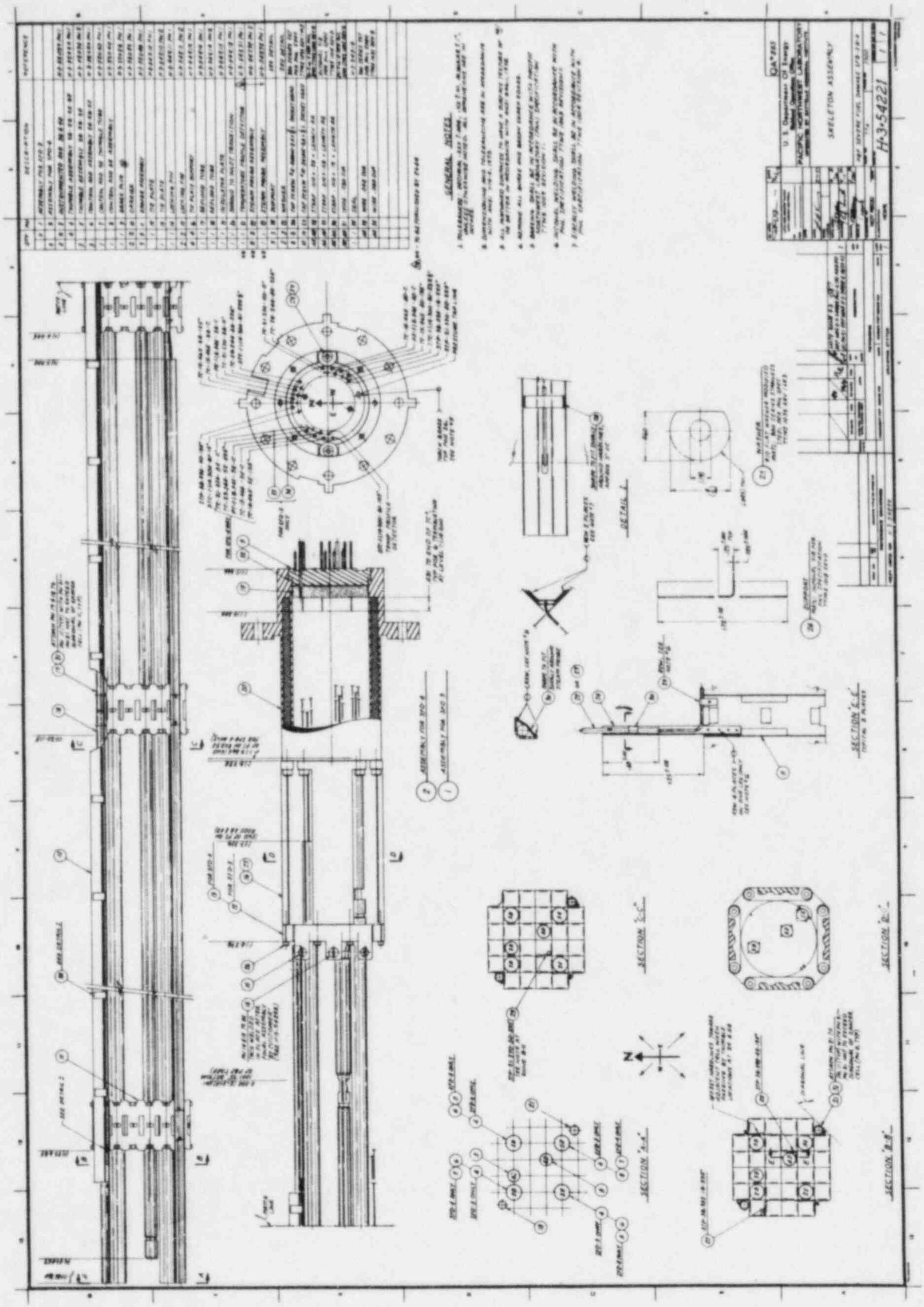

Figure 4. Fuel Bundle Skeleton Assembly

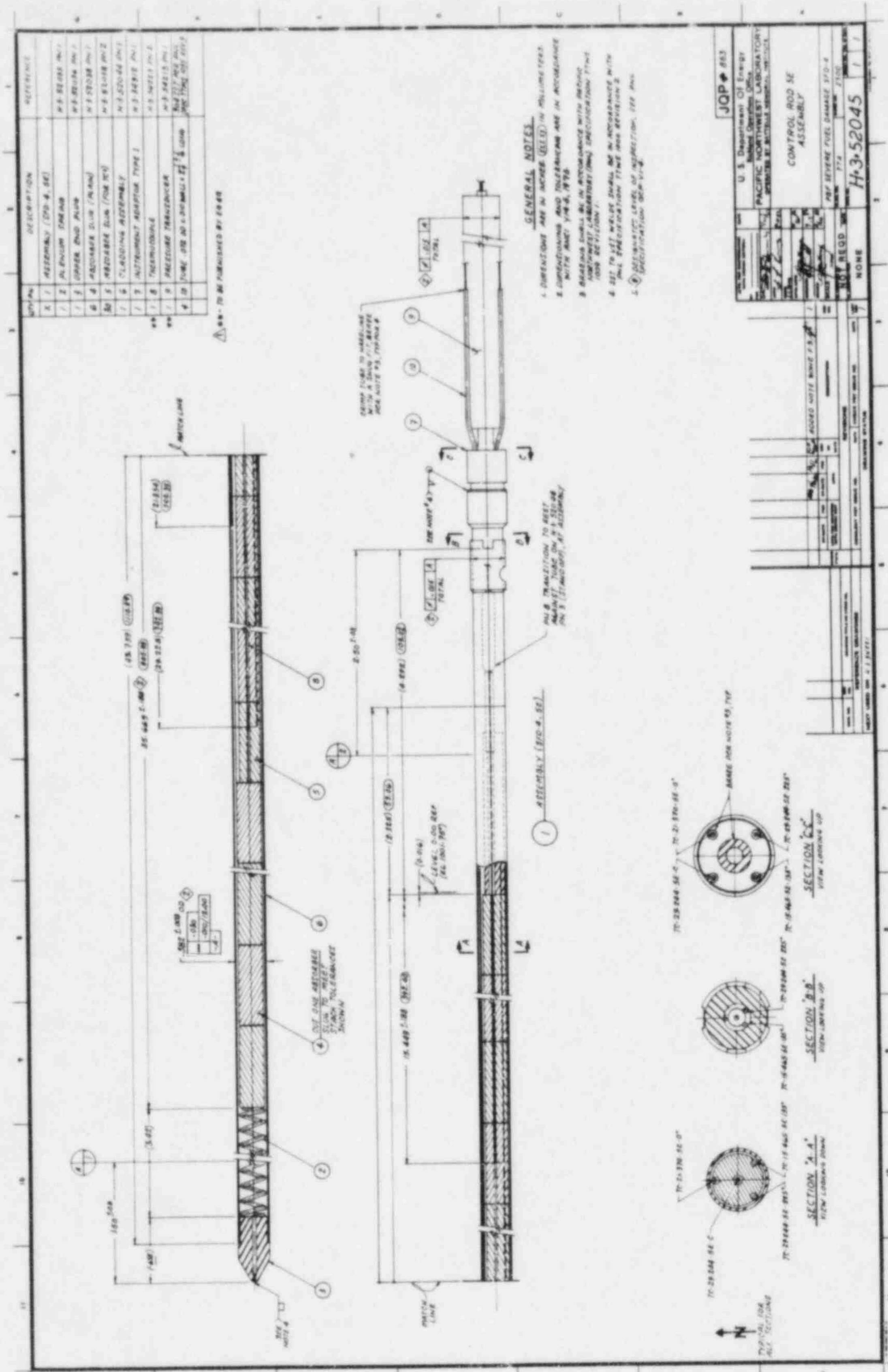

Figure 5. SFD-4 Control Rod Assembly

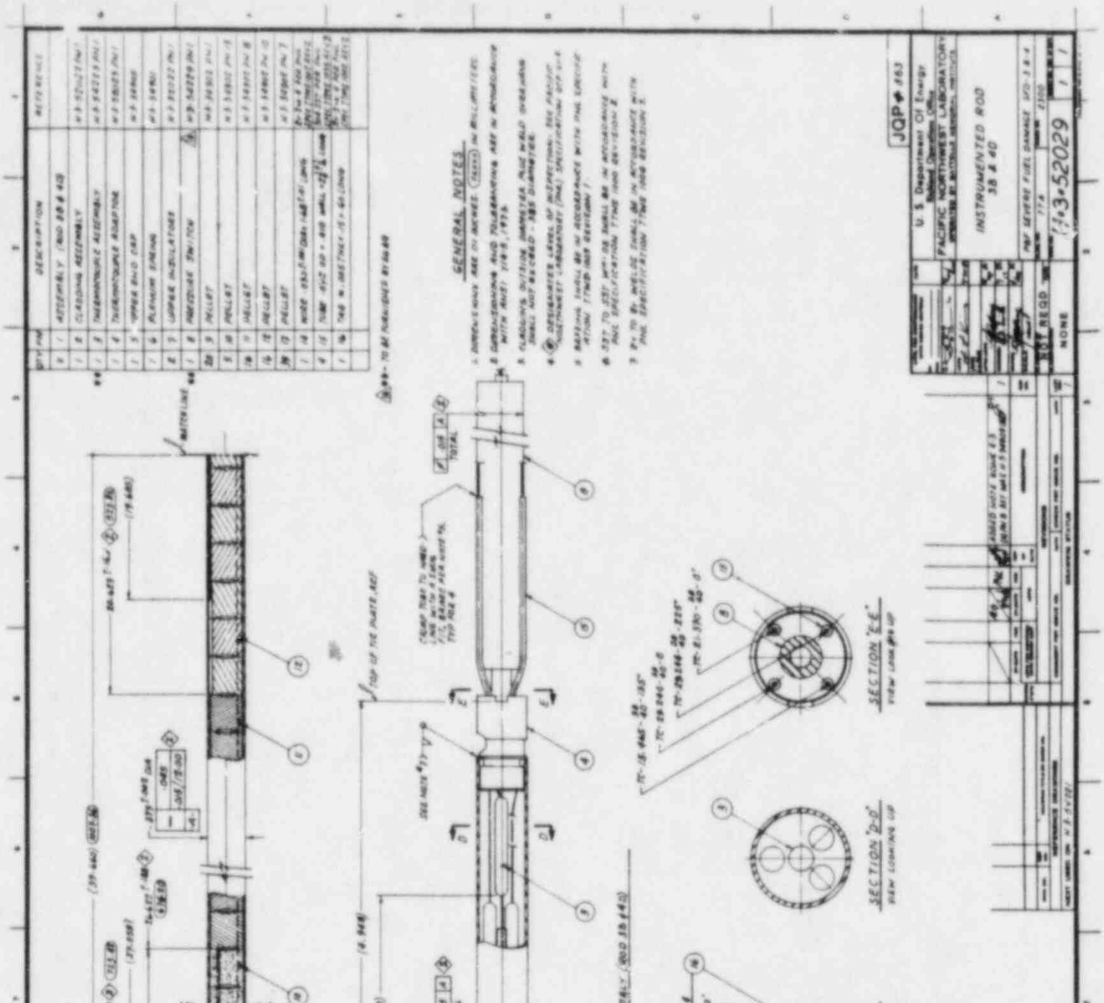

Figure 6. SFD-3 and SFD-4 Instrumented Rod Designs

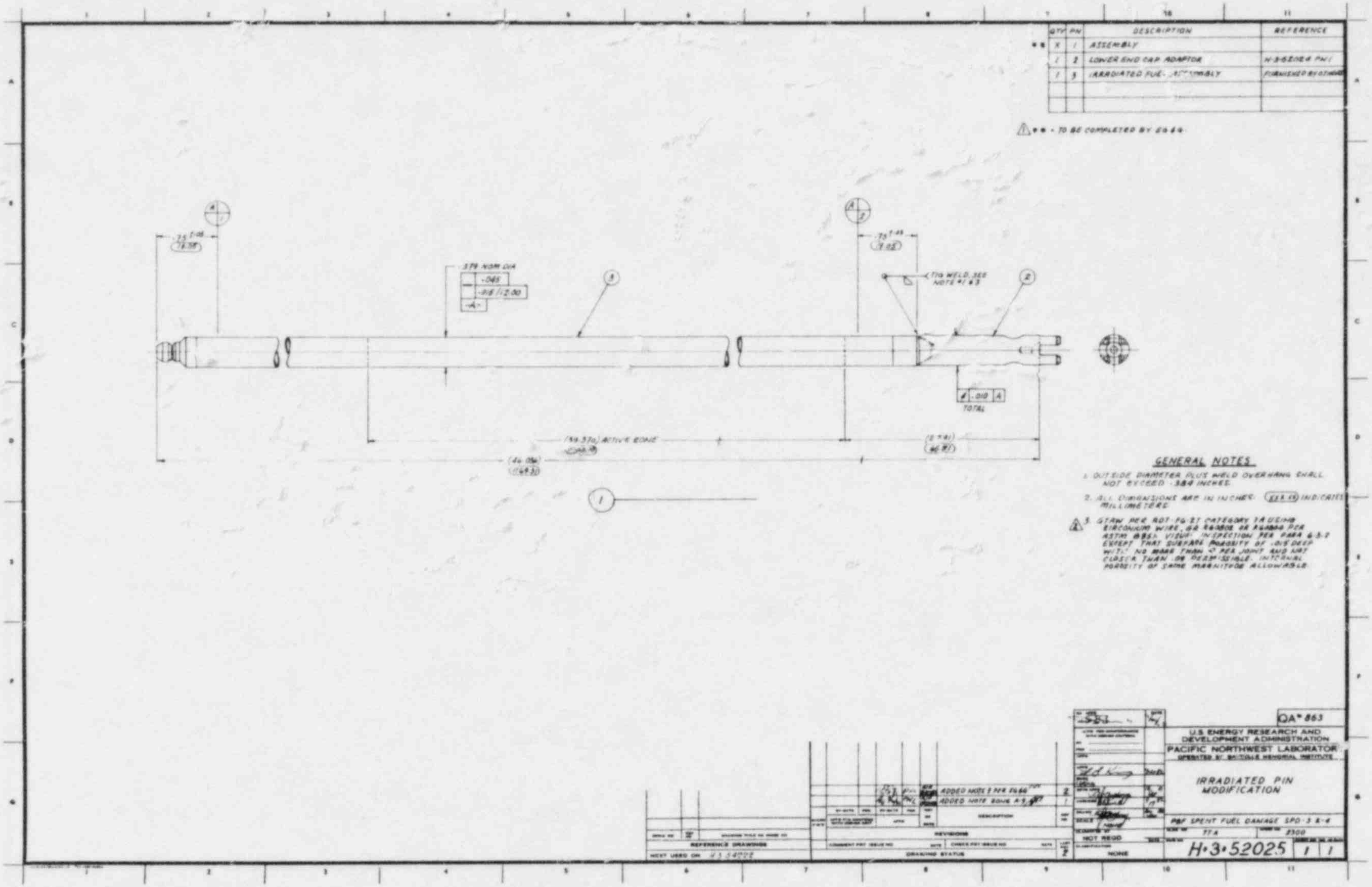

Figure 7. SFD-3 and SFD-4 Irradiated Pin Modifications

SEVERE CORE DAMAGE SUBASSEMBLY PROCUREMENT PROGRAM

ESSOR FUEL DAMAGE TEST PROGRAM SUPPORT[a]

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

SUMMARY

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

INTRODUCTION

The Super Sara Test Program (SSTP) is a major European Community effort to study reactor safety during rapid or large-break and slow or small-break loss-of-coolant accidents (LOCAs). The program is being conducted in the SUPER SARA high-temperature, high-pressure loop in the ESSOR reactor, Ispra, Italy. The objective of the SSTP is to obtain important experimental data on fuel rod deformation and postaccident coolability of damaged fuel assemblies after they experience a loss of coolant. The testing program currently includes loop construction and 21 in-reactor experiments to simulate 7 large- and 12 small-break conditions in commercial pressurized water reactors and boiling water reactors.

FUTURE WORK

The main effort on this project is currently being provided by the NRC site representative. Additional Pacific Northwest Laboratory support is limited to safety and licensing concerns.

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

CORE THERMAL MODEL DEVELOPMENT[a]

M. J. Thurgood, Project Manager

J. M. Kelly
T. E. Guidotti
R. J. Kohrt

SUMMARY

An upper head injection (UHI) 200% cold leg break simulation with evaluation model (EM) power was completed during this quarter using the COBRA/TRAC code. Work also continued on the development of the hot bundle code. The containment code was assessed against data from the Hanford Engineering and Development Laboratory (HEDL) hydrogen migration standard problems and with some Battelle-Frankfurt hydrogen distribution tests. The pretest prediction of the HDR steam blowdown test V-43 was also completed.

INTRODUCTION

The COBRA-TF[b] computer code is being developed for the U.S. Nuclear Regulatory Commission (NRC) to provide better digital computer codes for assessing the behavior of full-scale reactor systems under postulated accident conditions. The resulting codes are being used to perform pre- and post-test analysis of light-water reactor (LWR) components and system effects experiments. This Pacific Northwest Laboratory (PNL) project has three main objectives:

- to develop a water reactor primary system simulation capability that can model complex internal vessel geometries such as those encountered in UHI-equipped pressurized water reactors (PWRs)
- to develop a hot bundle/hot channel analysis capability to evaluate the thermal-hydraulic performance of LWR fuel bundles during postulated accidents
- to develop a containment code capable of simulating the steam water blowdown and hydrogen distribution phases of an accident.

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 feature is essential in the treatment of the hydrodynamics encountered during the reflooding phase of a loss-of-coolant accident (LOCA). The treatment of the droplet field is also essential in predicting other phenomena such as countercurrent flow limiting (CCFL), upper plenum deentrainment and fallback, and two-phase jet impingement.

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.

(a) FIN: B2041; NRC Contacts: J. T. Han and T. Lee.
(b) COBRA-TF = coolant boiling in rod arrays-two fluid.

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.

TECHNICAL PROGRESS

Progress during the past quarter is detailed below by topic.

PWR/UHI CALCULATION WITH EVALUATION MODEL POWERS

The UHI system simulation with EM power levels was completed. This calculation used the same input and code version as the best-estimate (BE) simulation;[1] only the power levels were different. The EM power calculation used 102% of total power, 120% of ANS-5.1 decay heat,[2] a 17% higher total peaking factor, and the Baker-Just metal/water reaction heat source.[3] These parameters gave the EM power calculation a 43% larger peak heat flux during bottom reflood.

The temperature response of the two calculations differed dramatically. The BE calculation exhibited an early quench of the entire core at 14 s when the water from the upper head was forced into the core. The hottest bundles of the EM calculation, however, remained above the minimum film boiling temperature—1200°F at high pressure[4]—during UHI delivery and did not quench until bottom reflood. The peak cladding surface temperatures for the hot rods are plotted in Figure 1, where curve 1 shows the BE temperature and curve 2 shows the EM power temperature. Although the center region of the core remained unquenched, the rest of it quenched during UHI water delivery as

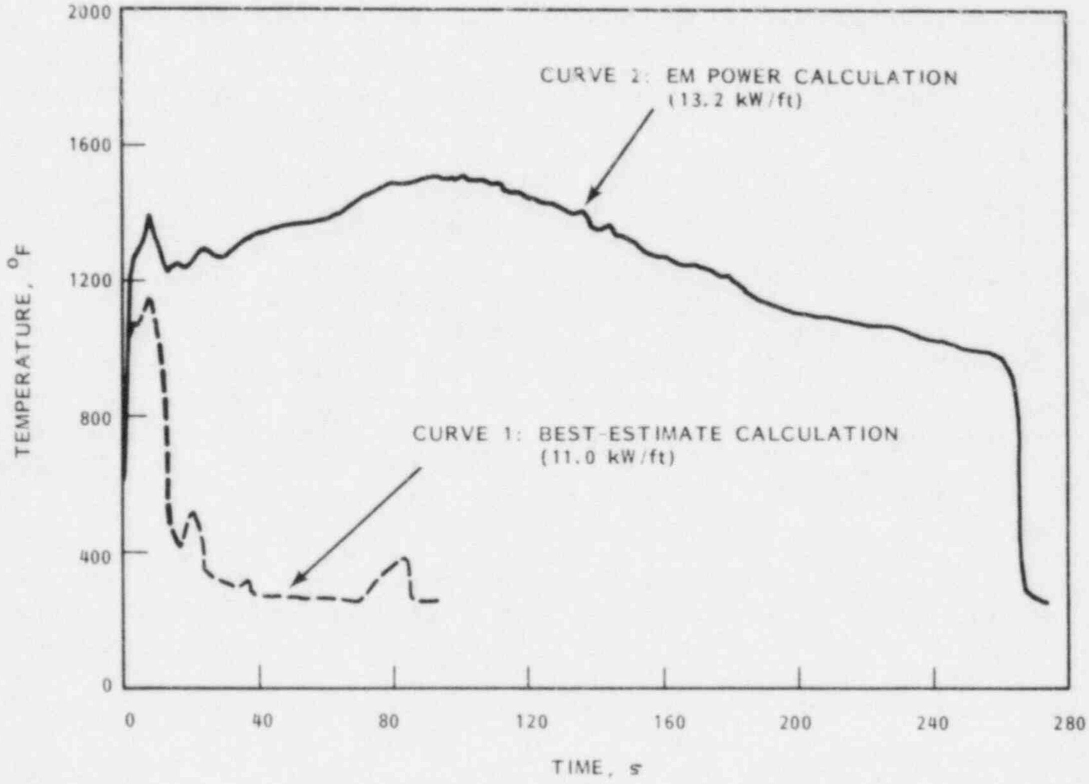

Figure 1. Peak Cladding Surface Temperatures for the Hot Rods

plotted on curve 2 in Figure 2, which shows the peak cladding temperature in an outside channel (one of eight outside channels). Curve 1 of Figure 2 shows the peak cladding temperature of the hot rod in the center channel of the core.

The hydrodynamic behavior in the EM power calculation was similar to the BE calculation. The collapsed liquid levels for both calculations are plotted in Figure 3, which indicates that the liquid was delivered to the core on four occasions. The BE calculation (curve 1) allowed slightly more liquid in the core and allowed it to enter at earlier times than the EM power calculation. The difference occurred because the lower power levels and, hence, lower vapor generation rates led to less CCFL in the BE calculation.

During the first core delivery, liquid came from two sources. Beginning at 3 s, flashing in the lower plenum forced liquid into the core and caused the liquid inventory to peak at 5 s. At ~6 s, additional liquid was provided as flashing in the upper head and accumulator injection forced the liquid down the support columns and into the core. The upper head was filling with cold water from the UHI accumulator during this time (since 2.4 s). Eventually, the cold water mixed with the hot water in the upper head and interrupted the flashing, which led to steam condensation (at 14 s). The condensation lowered the upper head pressure and caused the flow in the support columns to reverse. The upper head had refilled with water by 17 s. Liquid delivery to the core was then interrupted, allowing the core to empty by 18 s because the flow in the support columns was upward.

Once the upper head was refilled, continued accumulator injection forced liquid down the support columns and into the core once again. This second forced UHI water delivery ended when the accumulator reached the low-level set point and was shut off at 23.2 s (BE calculation).

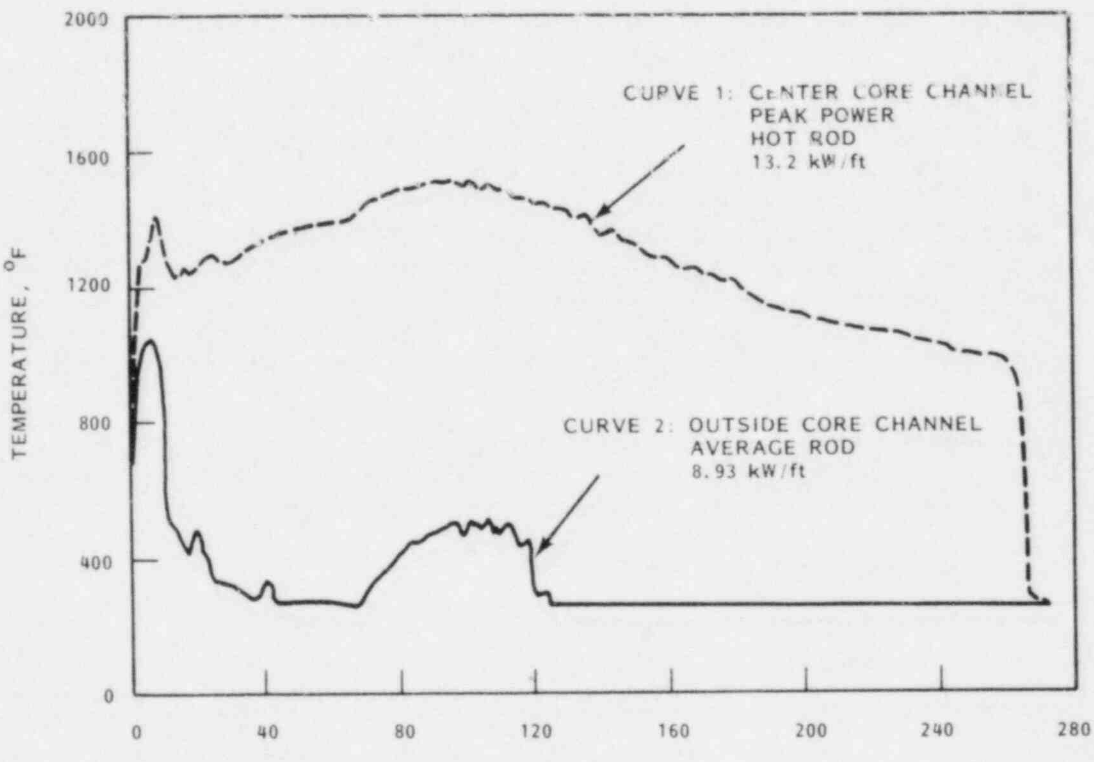

Figure 2. Peak Cladding Temperatures from Evaluation Model Power Calculation Versus Time

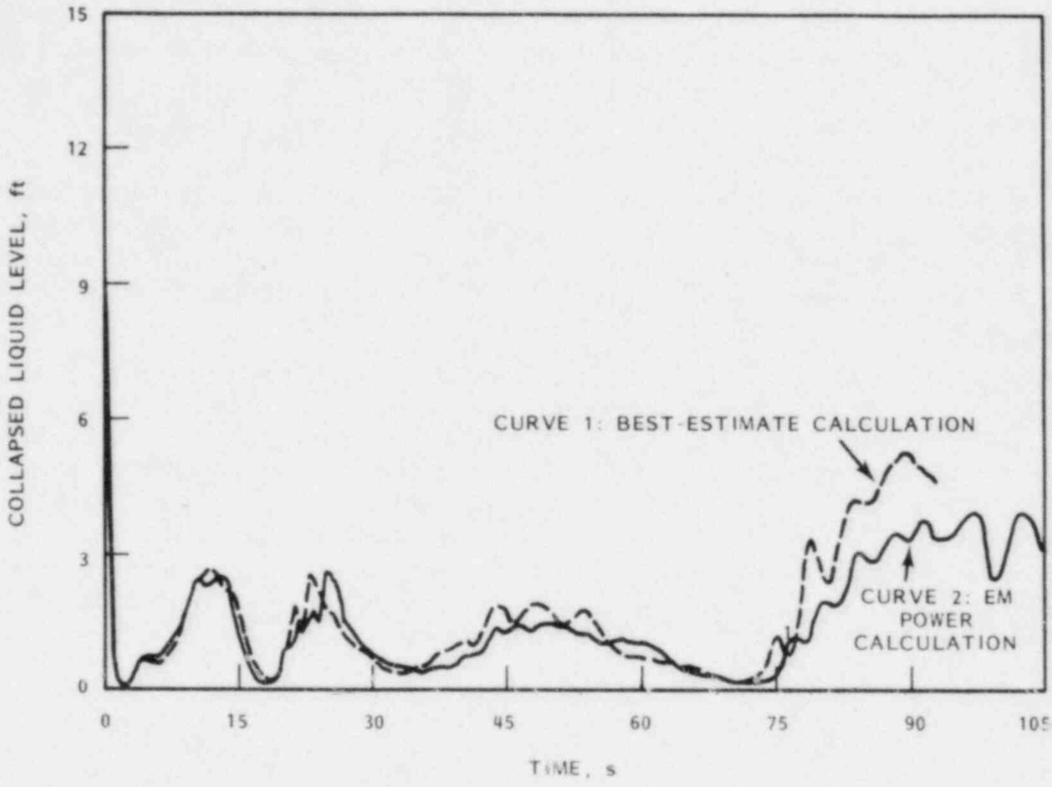

Figure 3. Core Liquid Level Versus Time

Without forced injection, the pressure in the upper head fell below the upper plenum pressure because of condensation. The support column downflow was diminished until the gravity head was enough to overcome the decreasing pressure drop. This event marked the beginning of upper head drain, which provided liquid to the core for the third time. During the upper head drain, large vapor generation rates in the core led to large downward vapor velocities that prevented bottom reflood. Once the upper head emptied, however, the core began to dry out, which lowered the vapor generation rate and allowed the liquid accumulated in the downcomer to collapse into the lower plenum and core inlet. Bottom reflood began at 73 s in the BE calculation and at 75 s in the EM power calculation.

Peak cladding temperatures were 1150°F at 8 s for the BE calculation and 1513°F at 96.1 s for the EM power calculation. The entire core quenched at 14 s in the BE calculation. All but the center of the core had quenched by then in the EM power calculation, and it quenched at 266 s.

HOT BUNDLE CODE DEVELOPMENT

The hot bundle version of COBRA-TF is being developed to evaluate the thermal-hydraulic performance of LWR fuel bundles during postulated accidents. A joint project is also being conducted with the FLECHT/SEASET program[5] to address Appendix K issues concerning reflood heat transfer with flow blockages. The first phase of development is to modify COBRA-TF to predict reflood heat transfer on a subchannel basis in an unblocked rod bundle. Code modifications include the addition of:

- thermal radiation to fluid and structures
- azimuthal conduction in heater rods

- grid spacer rewet
- grid spacer loss coefficient
- droplet breakup at grid spacer.

Radiation Model

The radiation model has been incorporated into the COBRA-TF hot bundle code. The model calculates radiative heat transfer in a rod bundle with an optically thin participating media. Three modes of heat transfer are possible: rod to rod, rod to vapor, and rod to droplet. The model can analyze heat transfer for each subchannel or for a group of subchannels (one lumped channel) in the bundle.

Rod-to-rod heat transfer has been assessed against data from the GOTA experiments. Results show good agreement between calculated and experimental rod surface temperatures. The rod-to-vapor, rod-to-droplet, and channel-lumping capabilities are currently being assessed against the 21-rod FLECHT/SEASET test bundle data.

Azimuthal Conduction Model

The results of a simple transient conduction problem using the azimuthal conduction model were compared with the same problem solved with the TEMPEST[6] computer code. The results compare exactly, indicating that the model is working properly.

Grid Spacer Model

A new grid spacer model is being added to COBRA-TF that includes propagation of a rewetting front along the length of the grid spacer. Grid rewetting is an important factor in calculating heat transfer downstream of the grid. The net effect of a grid spacer is to desuperheat the vapor flowing through it, causing enhanced heat transfer from the rods downstream of the grid. The model consists of a rewetting front velocity as described by Yamanouchi[7] and a heat balance on the dry portion of the grid to solve for the grid temperature.

The equation for the rewetting front velocity approximates the heat transfer coefficient in the dry part of the grid as zero and the temperature of the liquid film as the saturation temperature. The grid is assumed to rewet at 100°F above the saturation temperature. Heat transfer from the wet portion of the grid is limited by the critical heat flux (CHF) and by the heat of vaporization of the liquid film. The minimum of the heat transfer coefficient derived from Zuber's CHF correlation and the heat transfer coefficient assuming the entire liquid film evaporates is used as the heat transfer coefficient for the wet portion of the grid.

The rewetting front velocity is substituted into the time derivative of temperature in the heat balance equation. The resulting second order differential equation is solved for rewetting front velocity assuming the initial temperature is large.

$$U^{-1} = \rho C_p \sqrt{\frac{t}{h \cdot k}} \left(\frac{T_\infty - T_o}{T_o - T_f} \right)$$

where U = rewetting velocity
ρ = density
C_p = specific heat
t = grid thickness
h = wet portion of heat transfer coefficient
k = grid thermal conductivity
T_∞ = grid temperature, dry portion
T_o = grid rewetting temperature (T_f + 100°F)
T_f = saturation temperature.

The grid temperature above the rewetting front is approximated by a constant temperature. The heat balance used to derive the temperature includes radiation and conduction to the grid. Radiation heat transfer is calculated from the rod to the grid and from the vapor to the grid. All three of the bodies participating are assumed to be gray. Emissivity of the vapor is calculated using the Sun model.[8] Convection heat transfer is calculated using the same heat transfer coefficient as the heater rods at the axial level of the grid spacer.

Grid Spacer Loss Coefficient Model

The geometry of the rod bundle grid spacers is complex; therefore, the pressure loss coefficient is best determined from an empirical correlation. Yao[9] has developed a correlation for grid spacers in rod bundles as a function of a Reynolds number and blockage ratio.

$$K_G = C \times \begin{cases} 196.0 \cdot Re^{-0.33}\,\varepsilon^2 & 10^3 < Re < 10^4 \\ 41.0 \cdot Re^{0.16}\,\varepsilon^2 & 10^4 < Re < 10^5 \\ 6.5 \cdot \varepsilon^2 & 10^5 < Re \end{cases}$$

where C = multiplier

Re = mixture Reynolds number

ε = blockage ratio.

The blockage ratio is the fraction of flow area blocked by the grid space. The Reynolds number is calculated using mixture viscosity, density, and velocity. The correlation was developed for grid spacers with a rounded leading edge. A multiplier can be used to apply this correlation to grid spacers in other rod bundles.

Droplet Breakup Model

Desuperheating of vapor flowing through a grid spacer is also caused by the breakup of entrained liquid droplets that impinge on the grid spacer. Smaller drops significantly increase droplet surface area and evaporate much faster than larger drops. Evaporation is a source of saturated vapor, which results in desuperheating of the vapor downstream of the grid spacer. The number of small drops generated at each grid spacer is controlled by an empirical grid efficiency, which is multiplied by the flow area blocked by the grid spacer and the droplet mass flow rate to obtain a small droplet mass generation rate. The size of the small drops depends on the Weber number of the large drops.

COBRA-NC CONTAINMENT APPLICATION

The assessment of COBRA-NC against a variety of containment test data continued during this quarter. In particular, post-test calculations of the HEDL hydrogen distribution test and two Battelle-Frankfurt hydrogen distribution tests were made, and a blind pretest calculation of the HDR steam blowdown test V-43 was completed.

A post-test calculation was made of the HEDL horizontal jet test (Jet A) to determine the cause of the oscillation observed in the hydrogen concentration at the middle and upper elevations in the blind pretest prediction. The oscillation was caused by an error in the restart dump; when the code was run without restarting, no oscillations occurred. A coarser mesh was used in the post-test calculation. No other changes were made to the input model. The hydrogen concentration at the 125° top elevation is shown in Figure 4.

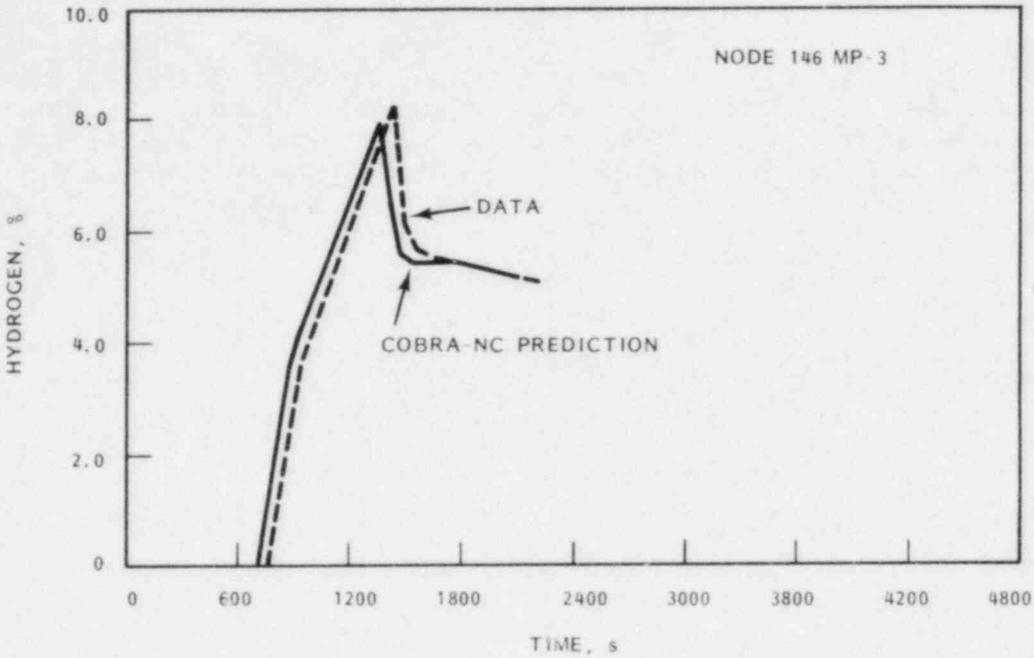

Figure 4. Hydrogen Concentration at the Top of the 125° Location—Post-Test Calculation

One of the features of COBRA-NC is that it can be used in either a multidimensional finite difference mode or in a lumped parameter mode. Test 12 of the hydrogen distribution series conducted in the Battelle-Frankfurt model containment was simulated to assess the lumped parameter capability.

A schematic of the Battelle facility is shown in Figure 5. Rooms 1, 2, and 5 through 8 were used during test 12 and all other rooms were sealed off, resulting in the test section shown schematically in Figure 5.

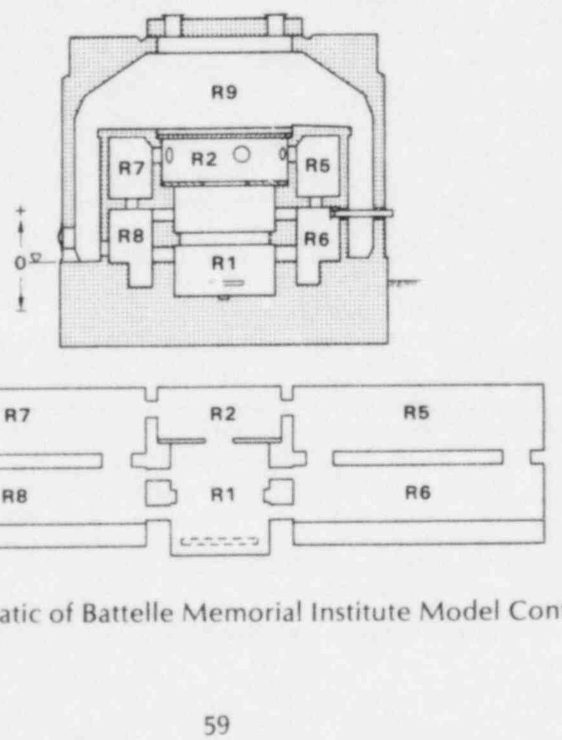

Figure 5. Schematic of Battelle Memorial Institute Model Containment

The gas source was in Room 1, and all rooms were initially at the same temperature. The nodalization and the boundary conditions used in the calculation are shown in Figure 6. The code predictions presented in Figures 7 through 10 for sensors in Rooms 1, 2, 7, and 6, respectively, are very good for the isothermal test using a lumped parameter approach. The solution should be tested for accuracy by varying the time step size when using the lumped parameter approach because some sensitivity to time step size was observed in this calculation. A maximum time step size of 10 s was used, and the calculation required 5 min of CPU time on a CDC 7600 for the 23.6-h transient.

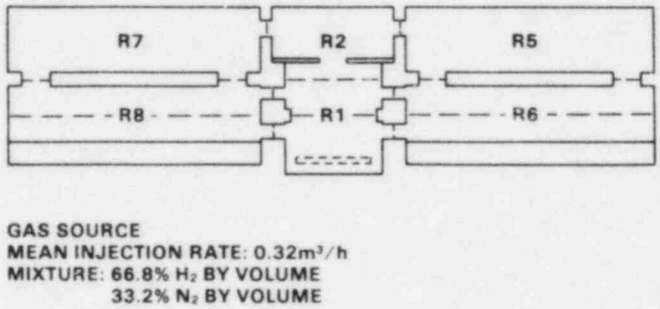

Figure 6. Nodalization and Boundary Conditions for Battelle-Frankfurt Test 12

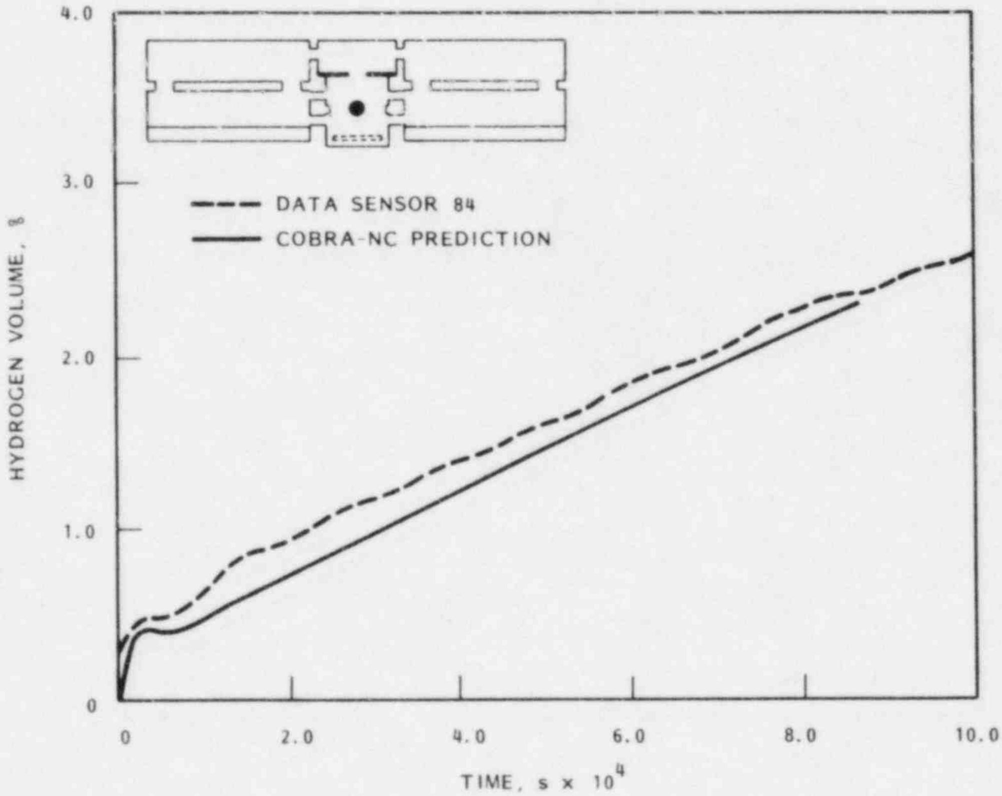

Figure 7. Hydrogen Concentration in Room 1 for Test 12

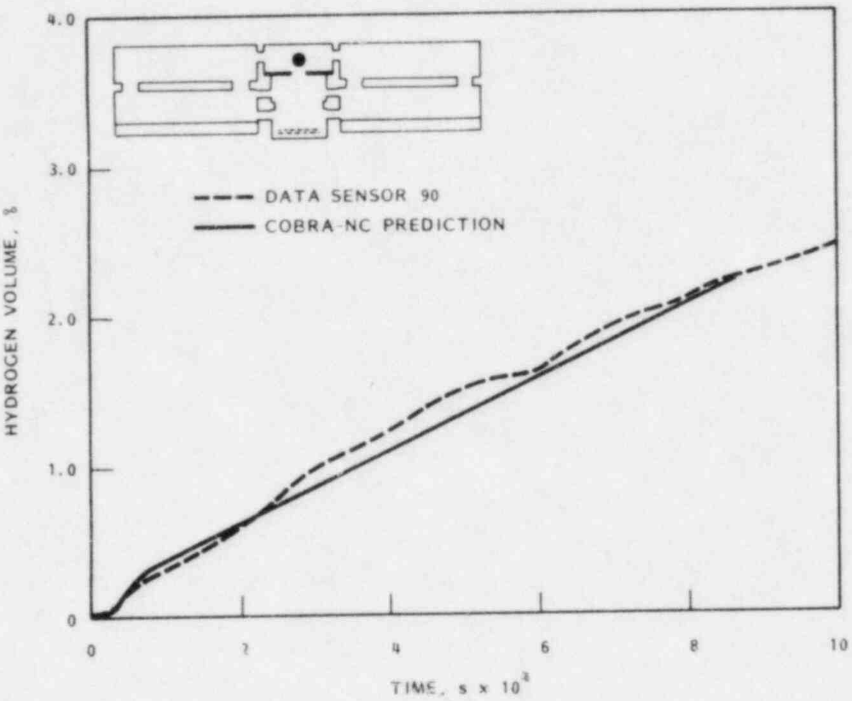

Figure 8. Hydrogen Concentration in Room 2 for Test 12

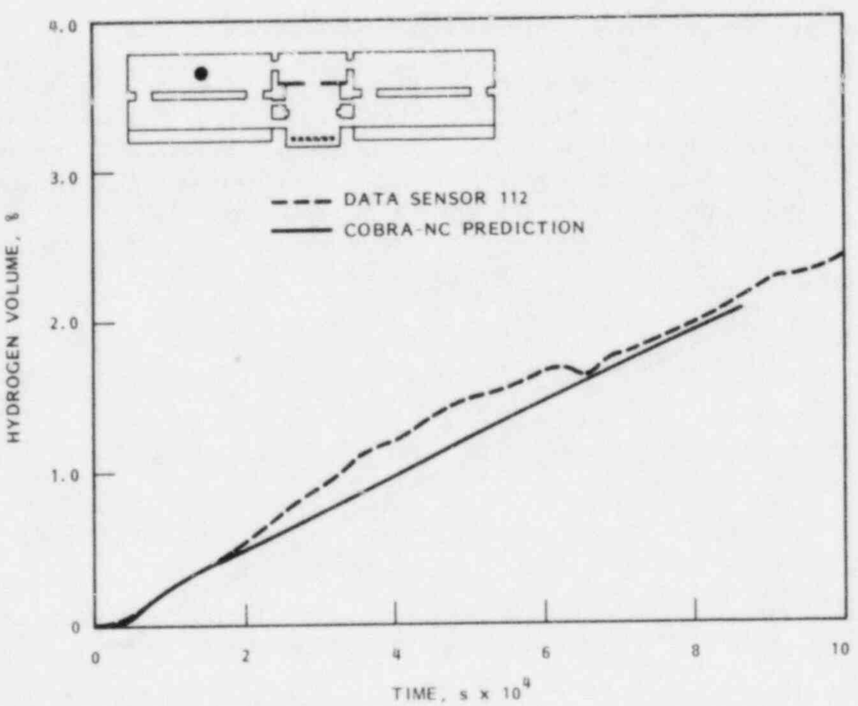

Figure 9. Hydrogen Concentration in Room 7 for Test 12

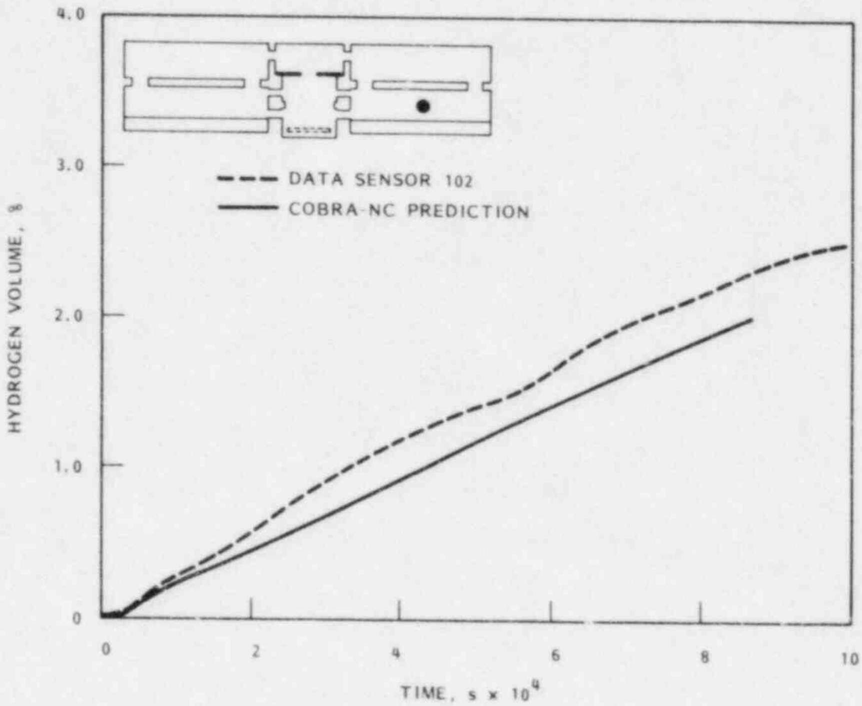

Figure 10. Hydrogen Concentration in Room 6 for Test 12

FUTURE WORK

A 1-D model for the UHI reactor vessel will be set up during the next quarter to assess the validity of simpler noding schemes that could reduce computation time if they are sufficiently accurate. Development of the hot bundle code will also continue during the next quarter. The preliminary assessment of the containment code will be completed during the next quarter, and the code will be documented and released to the public shortly after the beginning of the calendar year.

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LOCA SIMULATION IN THE NRU[a]

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SUMMARY

Due to the success of the materials test series (MT-1 through MT-4), the objectives of the program have been expanded to include a series of coolant boilaway and damage progression tests using previously irradiated fuel rods in a redesigned 21-rod test train. The MT-6 and MT-8 experiments will be 21-rod configurations in the new high-temperature shroud. MT-6 will be the proof-of-concept test, using unirradiated rods. A peak cladding temperature of 1478K (2200°F) will be used to validate the National Research Universal (NRU) reactor loop liner cooling system design and to establish parameters for the MT-8 experiment. MT-8 will be the first test train to contain previously irradiated fuel rods (maximum of three rods with burnups from 20,000 to 30,000 MWd/MTM). A fission product/hydrogen release monitoring and capture system is being designed. The peak cladding temperature for MT-8 will be 1811K (2800°F). Experiments MT-9 through MT-11 will be 12-rod superinsulated test trains with two or three irradiated rods; the tests will explore the 1922 to 2255K (3000 to 3600°F) maximum cladding temperature range.

The MT-6 test train is being fabricated. The high-density ZrO_2 insulation has successfully undergone thermal tests. Design concepts for the MT-8 test train and the fission product/hydrogen release monitoring systems are being developed.

INTRODUCTION

The Loss-of-Coolant Accident (LOCA) Simulation in the NRU Program is being conducted in the NRU reactor at Chalk River Nuclear Laboratories (CRNL), Chalk River, Ontario, Canada, by Pacific Northwest Laboratory (PNL). The program is sponsored by the U.S. Nuclear Regulatory Commission (NRC) to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length, 3% enriched light-water reactor (LWR) fuel rod bundle during the heatup, reflood, and quench phases of a LOCA. The tests are designed to give information on the quench-front velocities within a fuel bundle, the liquid entrainment [10 CFR 50, App. K (Sec. ID 2)], and the heat transfer coefficients [10 CFR 50, App. K (Sec ID 5)] for LWR fuel as a function of reflood rate and delay time before reflood starts. The program is composed of several thermal-hydraulic and cladding materials deformation experiments. The initial thermal-hydraulic experiment (TH-1) was conducted in October 1980 and provided a data base for predicting the quenching characteristics of Zircaloy fuel rods under various reflood conditions. Since that time, several other experiments have been conducted:

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

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

- 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 precise reflood rate control to obtain the cladding temperatures desired in 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.
- TH-4 and MT-4 (May 1982) used the MT-3 shroud and guard rod assembly and a 12-rod externally pressurized test cruciform. The external pressurization system allowed the TH-4 scoping tests to be conducted with unpressurized rods in the test cruciform. The rods were then pressurized, and MT-4 was conducted. Flux wires determined the flux profile, and a PNL-developed time domain reflectometry (TDR)-type liquid level detector provided real-time measurement of the reflood water level.

TECHNICAL PROGRESS

Post-test examination of the MT-4 fuel rods was completed. The United Kingdom requested that selected rods from MT-3 be reexamined with the new disassembly, examination, reassembly machine (DERM) rupture zone mapping head and that photomicrographs be prepared for certain rupture locations. This work was completed in late September 1982.

The MT-6 high-temperature shroud design with a 21-rod fuel bundle was finalized. The design is identical to that shown in Figure 1, but irradiated rods will not be used. The high-density ZrO_2 insulation passed thermal tests, and fabrication techniques are being devised. The shroud Zircaloy-to-stainless steel transition piece was successfully tested at temperature and pressure. All instrumentation is on order or in various stages of acceptance testing. The shroud-pressure tube interface will be cooled by a new pressurized bypass coolant system instead of the currently used leak rate system. Revisions to the safety analysis report (SAR) were completed and forwarded to Atomic Energy of Canada Limited (AECL).

Performance criteria for the MT-8 fission products release and hydrogen generation monitoring and retrieval systems were drafted. Gamma spectroscopy techniques are being investigated for fission product detection. Raman spectroscopy techniques are being considered for hydrogen monitoring. Capture systems that can be readily removed (remote handling and shielded containers required) from the U-2 loop piping system are being investigated.

FUTURE WORK

Fabrication of the MT-6 test train will continue. The MT-6 test plan will be developed to include all MT-6 goals and scoping tests for the MT-8 test where possible.

Detailed design analysis efforts for MT-8 will begin. Revisions to the SAR for MT-8 will be drafted and coordinated with CRNL. Sources and costs for obtaining and shipping the irradiated fuel rods will be sought. Design of the fission products release monitoring and capture systems and the hydrogen monitoring system design will be initiated. Meetings will be held with CRNL staff to devise an acceptable capture system.

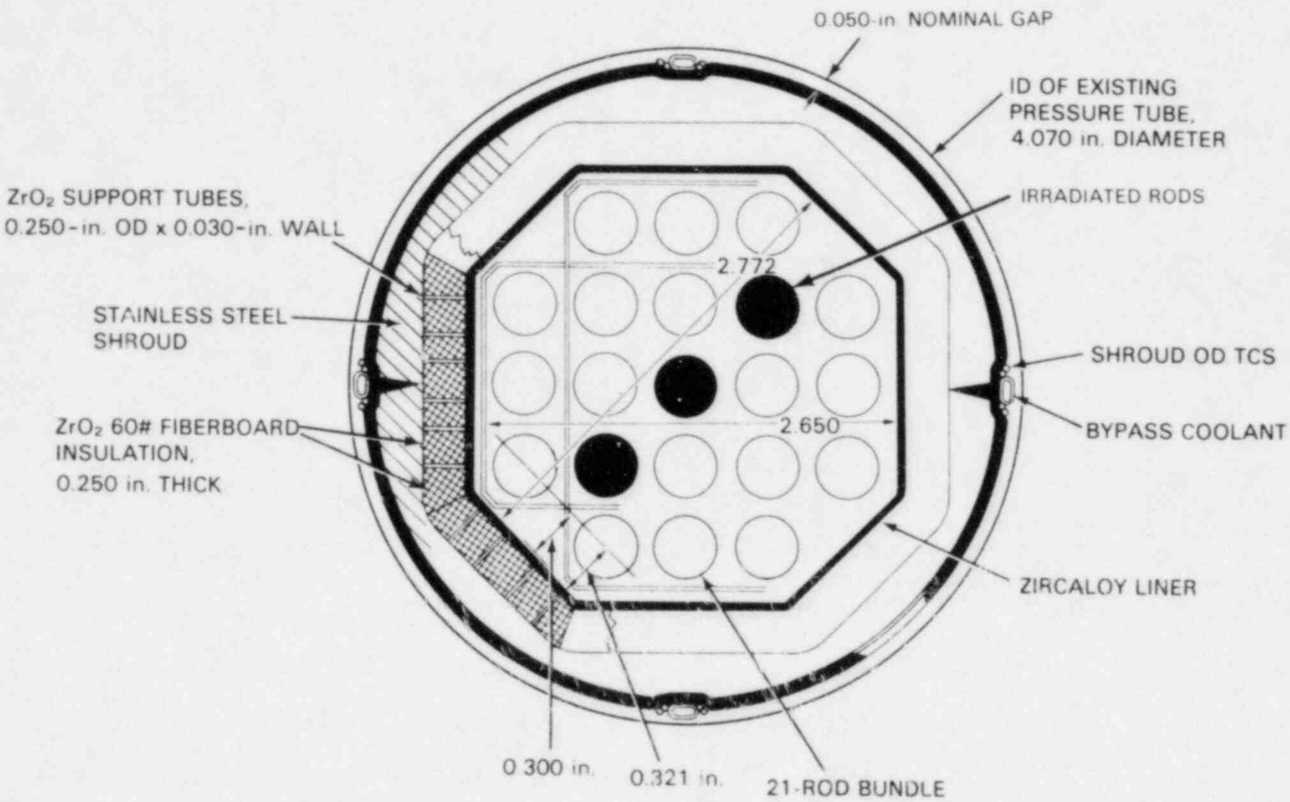

Figure 1. Test Train Concept for MT-6 (unirradiated rods) and MT-8 (irradiated rods)

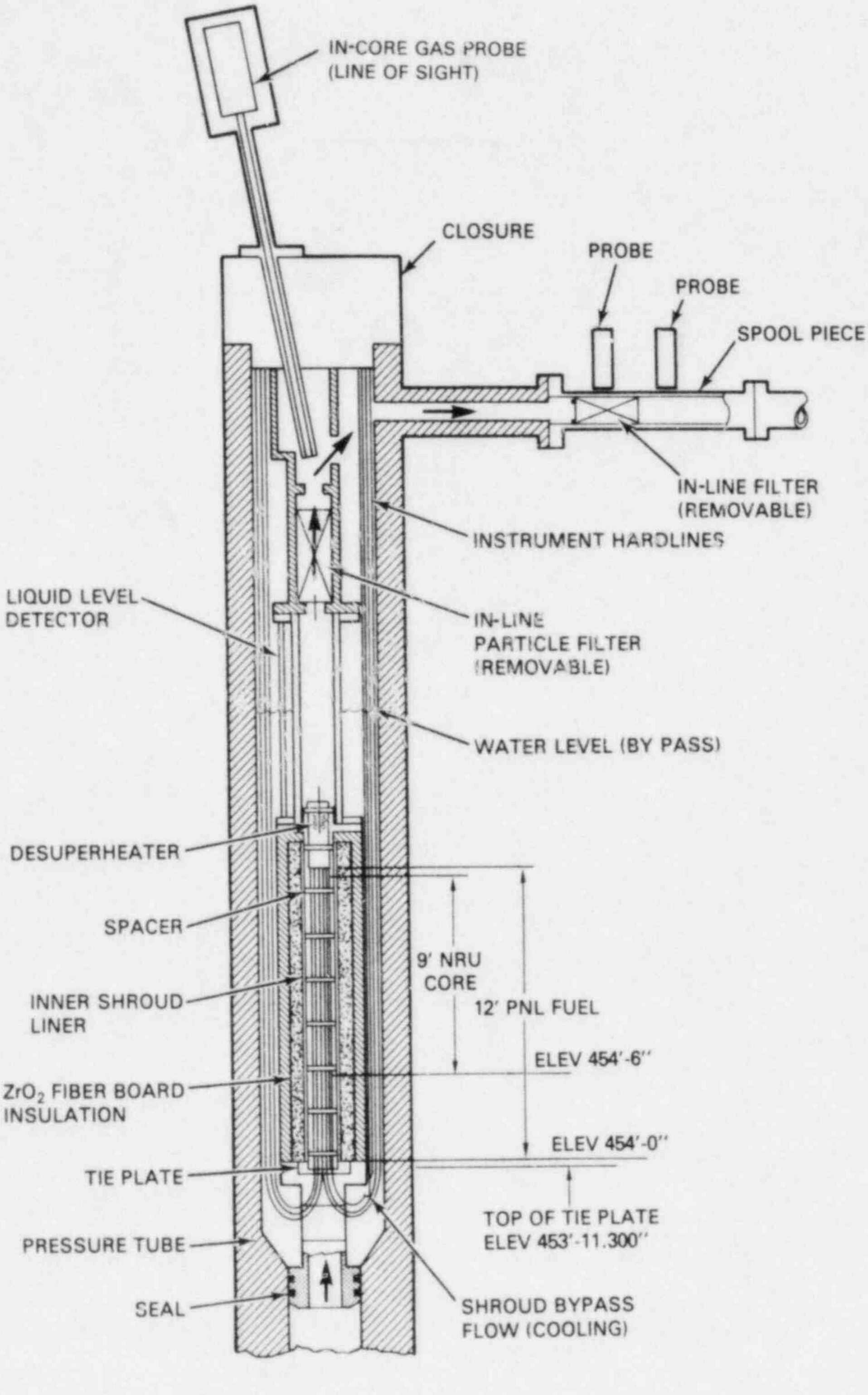

Figure 2. MT-8 Test Train Concept with Fission Product Detection System

STEAM GENERATOR GROUP PROJECT[a]

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SUMMARY

Task 5 (reopening of preshipment penetrations) was completed. The steam generator was found to be in essentially the same condition as it was prior to shipment except that the inner row U-bend cracks that had been observed at Surry had opened up in a couple of instances. Advertised announcement and RFP packages were prepared for Tasks 6 and 8 (decontaminating the channel head and removing tube plugs). Radiation field mapping of the generator and the Steam Generator Examination Facility (SGEF) is progressing; in addition, planning and scheduling of the base-line eddy-current in-service inspection (ISI), nondestructive testing (NDT) round robin, and decontamination tasks are proceeding. Secondary side nondestructive examinations (NDEs) were initiated, and research continued on stress corrosion crack (SCC) characterization and leak rate determinations. The cold leg side of the channel head was decontaminated using a proprietary process applied by a subcontractor.

INTRODUCTION

The Steam Generator Group Project (SGGP), which was initiated this year, 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, conducted to determine the consequences of tube failure in terms of leak rate. The stability of through-wall flaws is also 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

(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.

Hanford, Washington. The unit was temporarily stored awaiting the completion of the specially designed containment facility: the SGEF, which is equipped for both 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 has established interest from other parties to participate in the program. Potential exists for research and development in chemical cleaning, decontamination, corrosion product identification, corrosion mechanism studies, and repair techniques under alternate sponsors. Several indications of intent to join with the U.S. Nuclear Regulatory Commission (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
- demonstration 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; 21 people representing 20 organizations from 6 countries attended the meeting. During the past quarter, marketing presentations were made to potential program participants in Japan and Taiwan. Consortia from Italy, France, and Japan have formally become program participants; and the Electric Power Research Institute (EPRI) has joined the program.

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. With minor adaptations to the work at hand, the facility has served well for Tasks 5 and 6 (see Reference 2).

Health Physics (Task 3)

The health physics task provides procedures for personnel exposure monitoring, control, and documentation. Health physics research activities include radiologic mapping,[2] determination of the effectiveness of decontamination efforts, and evaluation of waste and waste disposal problems associated with various operations. Previous efforts 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. The channel head region had radiation levels of 4 to 5 R/h prior to decontamination (Task 6). After decontamination of the cold leg side of the channel head, radiation levels fell to 0.6 to 2 R/h.

Data Management System (Task 4a)

Software was developed to interface the computerized data acquisition system with NDT (mainly eddy-current) devices providing data inputs. 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.

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. Initial experimental data, especially from Task 5, are being input to the system.

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.[2] No dislocation in the upper support plate was measured. A preliminary report of these findings was prepared (PNL-SA-10501) and presented to the annual ANS meeting. A detailed topical report for all participants is being prepared.

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 was offered as an opportunity for research-related development/demonstration of decontamination techniques. The cold leg manway cover was removed, and base-line exposure data were taken. Small samples of the stainless steel manway insert were cut, and core samples were drilled from both sides of the channel head stainless steel strip cladding (Figure 1) to examine corrosion product films. These films will be compared with similar postdecontamination surface analysis.

Responses to the RFP for decontamination services were evaluated. Two subcontracts were let, each to decontaminate one-half of the channel head. The cold leg side was decontaminated by London Nuclear Services, Inc., using a proprietary process. An average decontamination factor (dF) of ~8, but variable over the surface, was achieved (see Figures 2, 3, and 4). The hot leg side will be decontaminated by Quadrex Corporation next quarter using another proprietary process that was developed in Great Britain.

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. An RFP package has been prepared, and subcontract sources will be sought.

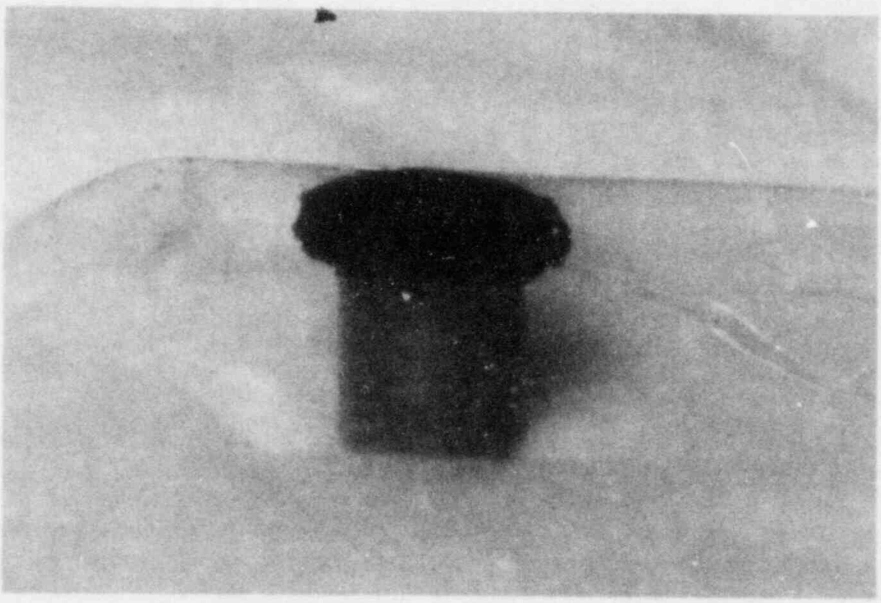

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Figure 1. Core Sample Drilled from Steam Generator Channel Head

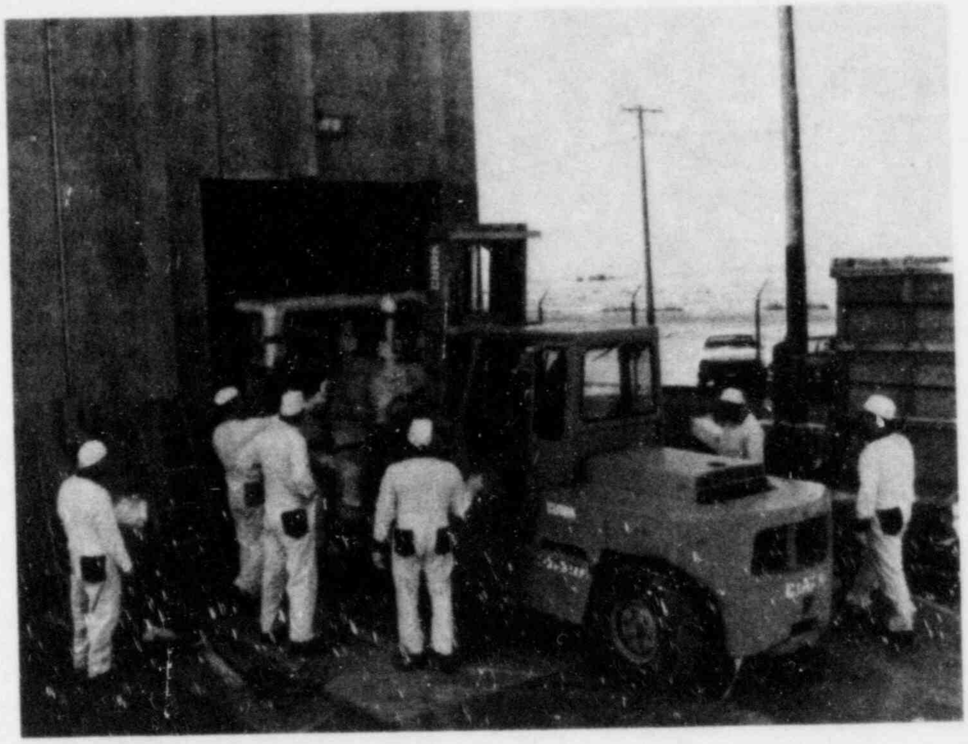

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Figure 2. Loading London Nuclear Services Equipment into SGEF Truck Lock

Tube Unplugging (Task 8)

An RFP to remove the plugs from 461 of the 748 plugged tubes was issued. Unplugging the tubes will increase the availability of defects for correlation with historical ISI data. Responses to a fixed price RFP were nonresponsive; a time and materials RFP has gone out to bidders.

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. Sludge pile locations will be characterized, and corrosion product sampling will be conducted. Initial activities focused on extending the areas characterized through the preshipment penetrations, and current efforts are concentrating in the inner row U-bend area and at the upper tube sheet surface. Equipment for performing this task has been received and includes miniature cameras, a fiberscope, and articulated sampling devices. An RFP to open several larger shell openings for more extensive inspection and removal of samples is being issued to contractors.

Tube Sheet Section Removal (Task 11)

A large section of the tube sheet will be removed for analysis of operating effects on the tube sheet, especially at the tube-tube sheet crevices. The section to be removed has been identified as rows 7 to 17 and columns 70 to 94. The tubing included in the section plus 6-in. lengths above the tube sheet upper surface will be removed along with the section. Cutting the large shell opening to accomplish this task is being done jointly with Task 10.

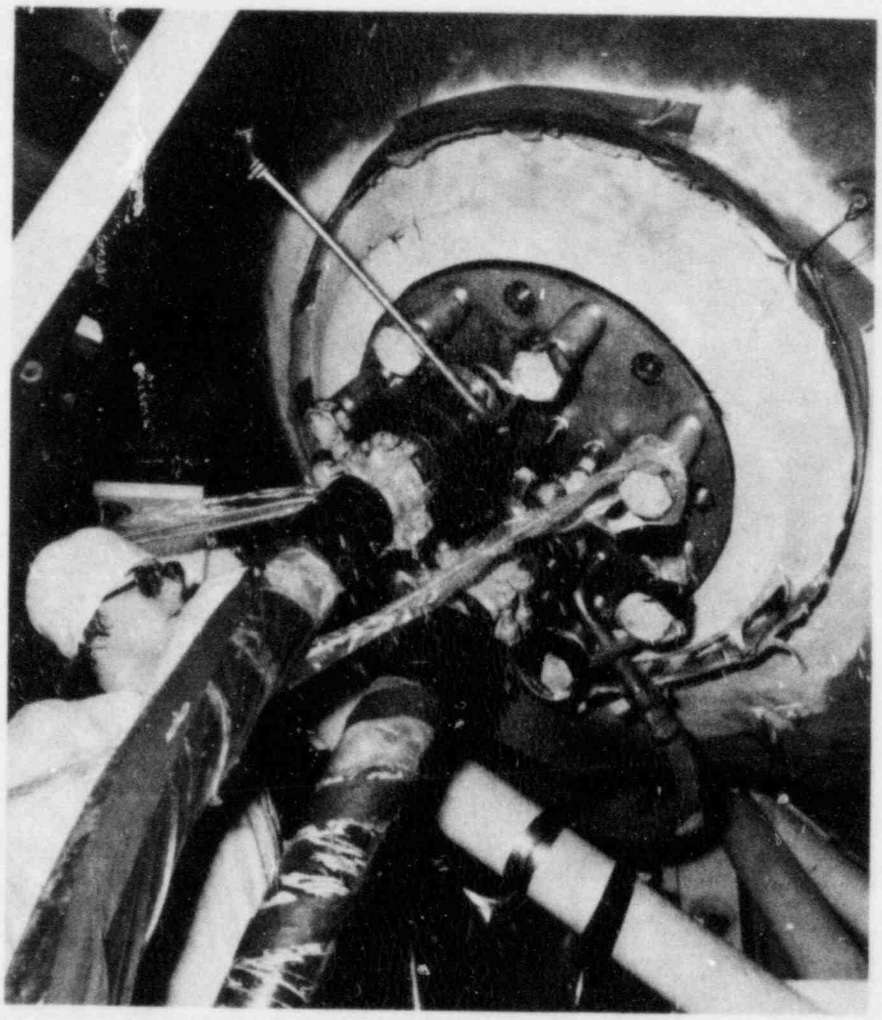

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Figure 3. Substitute Manway Flange with Fittings for Decontamination Solutions

Other Tasks

Task Action Plans were prepared for all other tasks that begin in the coming fiscal year and will be presented to the Technical Advisory Group (TAG) in Cadarache, France, in October. Participant inputs as a result of the TAG meeting are expected to influence planning and implementation.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

An extensive eddy-current round robin test was conducted on 10 laboratory-manufactured SCC defect tube specimens. Analysis of the results shows that present eddy-current techniques exhibit deficiencies in the reliability of detection and accuracy of sizing for low-volume defects, particularly SCC.

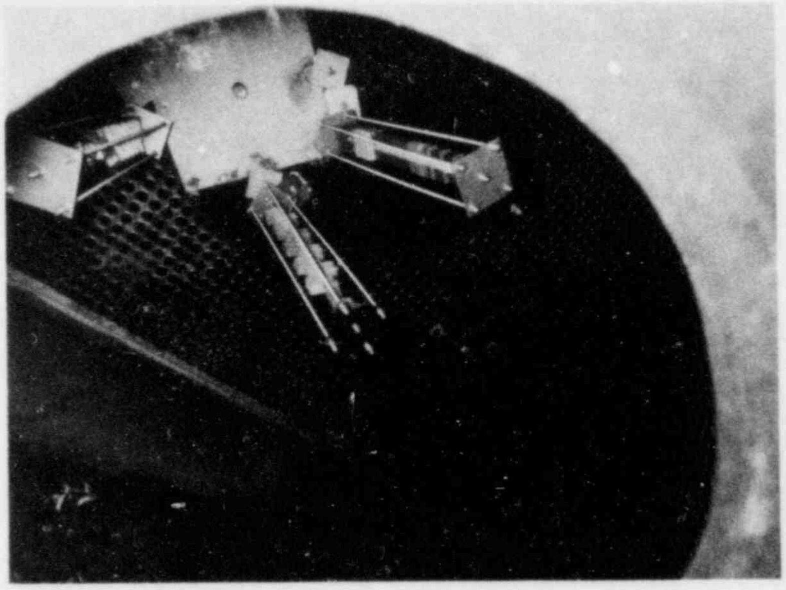

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Figure 4. View of Cold Leg Side of Channel Head After Decontamination, Showing Corrosion Samples

MILESTONES

- Formal participation in SGGP approved by Italy, France, Japan, and EPRI.
- Task 5 (reopen preshipment penetrations) reported.
- Task 6 (decontaminate channel head) completed on cold leg side.

PUBLICATIONS/PRESENTATIONS

A description of the SGGP was presented to a June 15-16, 1982, conference of utility representatives sponsored by Power Cutting, Inc., in Chicago, Illinois. The technology of repairing and replacing nuclear steam generators was presented.

"Initial Inspection of a Service-Degraded Steam Generator Removed from Service," PNL-SA-10501, was presented to the American Nuclear Society Annual Meeting in San Francisco, June 6-10, 1982.

FUTURE WORK

During the coming quarter, the following activities will be purused:

- attend the TAG meeting scheduled for October 18, 1982, at Cadarache, France
- complete decontamination of the hot leg side of the channel head
- 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 robin.

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BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-2716, Vol. 3
PNL-4275-3

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Reactor Safety Research Programs
Quarterly Report
July-September 1982

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
S. K. Edler, Editor

5. DATE REPORT COMPLETED
MONTH January | YEAR 1983

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Pacific Northwest Laboratory
P.O. Box 999
Richland, WA. 99352

DATE REPORT ISSUED
MONTH March | YEAR 1983

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Accident Evaluation
Division of Engineering Technology
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U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO. B2101, B2088, B2097, B2289, B2043, B2383, B2084, B2372, B2041, B2391, B2277

13. TYPE OF REPORT
Quarterly

PERIOD COVERED (Inclusive dates)
July-September 1982

15. SUPPLEMENTARY NOTES

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16. ABSTRACT (200 words or less)

This document summarizes work performed by Pacific Northwest Laboratory (PNL) from July 1 through September 30, 1982, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor 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 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, 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

quarterly progress
core thermal modeling
loss-of-coolant accident
severe fuel damage
FRAPCON computer code
fracture mechanics
pipe-to-pipe impact

17a. DESCRIPTORS

nondestructive examination
in-service inspection
postirradiation testing
in-reactor testing
steam generator

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)
Unclassified

21. NO. OF PAGES
75

22. PRICE
$

NRC FORM 335 (11 81)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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NUREG/CR-2716, Vol. 3

QUARTERLY REPORT – JULY-SEPTEMBER 1982

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