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Reactor Safety Research Programs

Quarterly Report
October - December 1981

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
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ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from October 1 through December 31, 1981, for the Division of Accident Evaluation, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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

R. W. Hill
M. T. Thomas

SUMMARY

A procedure has been established for calibrating the eddy current oxidation profiling system; a new coil and some of the required graphite standards have been prepared. Several modifications have been made in the ultrasonic backscattering equipment to improve the signal-to-noise ratio.

INTRODUCTION

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

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

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

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

TECHNICAL PROGRESS

EDDY CURRENT TESTING

A procedure has been developed to calibrate the oxidation profiling system that involves varying test parameters such as liftoff, coil diameter, and sample conductivities. The resultant effective resistivity (ΔR_{eff}) and reactivity (ΔX_{eff}) data (obtained for each set of test parameter changes) are then used with Dodd's theory and the computer algorithm ZFIT to determine the calibration constants. The philosophy at this stage of the program is to supply sufficient data so that the computer algorithms can be accurately fitted to the laboratory eddy current hardware rather than attempting to obtain idealized response signals from the hardware. It is felt that this approach will lead to a practical in-field calibration procedure for the system.

A smaller coil (0.75 in. diameter) has been fabricated, and graphite standards (3 x 3 x 0.5 in. thick) with up to 12% homogeneous oxidation have been prepared for use in the calibration program.

ULTRASONIC TESTING

Several important modifications to the ultrasonic backscattering method are being considered for the laboratory system. These modifications will increase the signal-to-noise ratio and still allow detection of smaller concentrations of oxidation so that the oxidation profile can be determined as a function of distance into the material.

Several experiments are being conducted to determine the ultrasonic signal attenuation as a function of frequency and oxidation for PGX graphite. As the operating frequency of the system is increased, the smaller voids and pores are more easily detected; hence, a more accurate oxidation profile can be deduced. However, attenuation follows the fourth power of frequency, which results in greatly reduced signal-to-noise levels. From these experiments, the optimum frequency range of operation (which is an important parameter for the final ultrasonic backscatter system design) can be determined.

CFTL TESTING

Design information has been supplied to ORNL for NDT examination of the graphite posts that are to be oxidized in CFTL. These tests will take place only if the Fort St. Vrain studies do not supply the necessary demonstration of the testing techniques.

OXIDATION

Graphite density standards with homogeneous oxidation (up to 12% weight loss) have been prepared to calibrate the eddy current equipment. Electrical conductivity standards have been machined and will be homogeneously oxidized to various amounts during the next quarter.

FUTURE WORK

EDDY CURRENT TESTING

- fit Dodd's theory and ZFIT to the laboratory eddy current equipment
- use these fitting techniques to develop an in-field calibration algorithm and procedure.

ULTRASONIC TESTING

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

OXIDATION

- prepare homogeneously oxidized electrical conductivity standards
- determine chemical state of impurities during oxidation.

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

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R. J. Kurtz, Assistant Project Manager

T. T. Taylor

SUMMARY

The A533B insert for the German ZB-1 test vessel was completed and welded into the vessel wall. A Pacific Northwest Laboratory (PNL) pressure vessel will be used to check out the full acoustic emission (AE) monitoring system and the procedures planned for the ZB-1 test.

AE monitoring of a preservice cold hydrostatic test at the Tennessee Valley Authority (TVA) Watts Bar 1 reactor was completed with the following primary results:

- known AE was detected from a crack specimen that was attached to piping
- spontaneous AE was detected from a nozzle area
- coolant pumps generated an unexpectedly high background noise.

A Quick-Look report was issued on the test, and the final report has been prepared for TVA review.

AE data from two irradiated fracture specimen tests were analyzed in terms of J integral and K_J. Data from the two tests compare rationally with each other. Further data will be obtained from a similar unirradiated specimen to complete the evaluation.

INTRODUCTION

The purpose of this 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 and SA351-CF-8A cast stainless, Type 304 wrought stainless, and A106 ferritic piping steels are being used in the experimental testing. Objectives of this program are to:

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

TECHNICAL PROGRESS

Progress relative to the above objectives is discussed in the following sections on off-reactor vessel test, reactor monitoring, AE monitor system development, irradiated fracture specimen tests, and reports.

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

OFF-REACTOR VESSEL TEST

Fabrication of the A533B steel insert for the ZB-1 test vessel was completed and included fatigue precracking three part-circular machined flaws. The German Materialprüfungsanstalt (MPA), who managed insert fabrication and performed flaw precracking, documented the entire process in a report that was delivered to PNL in November 1981. The insert was welded into the ZB-1 vessel, which should now be ready for delivery to the test site.

A conflict between German interests in applying high hydrotest overpressures and U.S. interests in generating consistent fatigue crack growth was amicably resolved. High overpressure hydrotesting (1.4 x operating) will be done at the end of the test matrix.

A 4-ft outside diameter (OD) x 5-1/3-ft long x 3/8-in. thick wall vessel has been temporarily obtained to check out the ZB-1 vessel AE monitor system. The system will be operated at PNL for several weeks in a configuration very similar to that planned for the ZB-1 test.

REACTOR MONITORING

The AE monitoring of cold hydrostatic testing at the Watts Bar 1 reactor has been completed. The No. 2 inlet nozzle, the 10-in. safety injection pipe on the No. 2 cold leg, and a section of the vessel wall were instrumented. Monitoring began October 15, 1981, and ended October 23, 1981. A Quick-Look report on the results was issued on November 16, 1981; and the draft of the final report has been completed and will be reviewed with TVA in mid-January 1982. The following major results were achieved:

- AE was detected from a crack growth specimen that was pressure coupled to the 10-in. accumulator pipe and actuated with a small hydraulic cylinder. This configuration provided a source of known AE to evaluate AE system sensitivity.
- Data filtered by signal duration to remove electrical transients indicated that spontaneous AE was detected from the nozzle area.
- High background noise was noted when all four reactor coolant pumps were operated. This noise may have been significantly influenced by the fact that the vessel contained none of the normal internal structure, which could act to reinforce the pump noise.

This cold hydrostatic test monitoring was performed using temporary signal lead wires. The next step in this effort will be to install permanent signal leads.

AE MONITOR SYSTEM

As part of the continuing testing of the AE monitor system for the ZB-1 vessel test, the pattern recognition algorithm in the PDP 11/03 computer was cross checked with the original algorithm on the VAX 11/780 (see Figure 1). Performance results agree to three digits. Pattern recognition analysis on the PDP 11/03 will require 14 sec/waveform. Although this will be a limitation on the onsite data analysis rate, it should not represent a serious problem. Waveform analysis can proceed in parallel with test monitoring.

IRRADIATED FRACTURE SPECIMENS

Analysis of the AE data obtained during fracture testing of irradiated weld metal specimens 65W-25 and 67W-23 has been completed. Both of these specimens were 4.0-in. thick compact tension specimens with 20% side gooves. Both tests were run at a test temperature of 392°F, and the J-R curves were generated by the unload compliance technique. The average copper content was 0.22% for weld

Figure 1. Acoustic Emission Detection/Analysis System

65W and 0.27% for weld 67W, and the approximate fluence was 3.8×10^{18} n/cm^2 for 65W and $5\ 3 \times 10^{18}$ n/cm^2 for 67W. The J-R curves for these specimens are given in Figure 2. The R-curves exceed the current crack extension limitations in that the region of J dominance has been surpassed. Weld 65W exhibited more ductile behavior than weld 67W, which is not surprising considering its lower copper content and fluence level.

The AE event count versus J integral data are shown in Figure 3. Only AE data obtained during the ramp loading and first hold period of the test sequence were plotted. All AE signals obtained during the ramp unloading and second hold period have been deleted since no crack extension was anticipated during these portions of the test. A roughly linear relationship exists between AE event count and flaw severity as defined by J, and the most acoustically active specimen was also the least ductile.

Converting the J integral fracture parameter to a pseudo-equivalent stress intensity factor (K_J) facilitated a comparison between these data and the data obtained from the Heavy Section Steel Technology Program (HSST) vessel tests. The K_J data were calculated from the following equation:

$$K_J = \left[\frac{EJ_I}{(1 - \nu^2)}\right]^{1/2}$$

where E and ν are Young's modulus and Poisson's ratio, respectively. Caution should be exercised when interpreting the results of this comparison (see Figure 4) since there are significant temperature, specimen geometry, and material property differences between the Naval Research Laboratory (NRL) fracture tests and the HSST vessel tests. For example, Figure 4 suggests that the irradiated material exhibited a lower acoustic response than the unirradiated material, which is opposite to expected results. At this time, the primary significance to be derived from this comparison is further evidence of the influence on AE in moving from a test specimen to a large structure. Fortunately, the influence

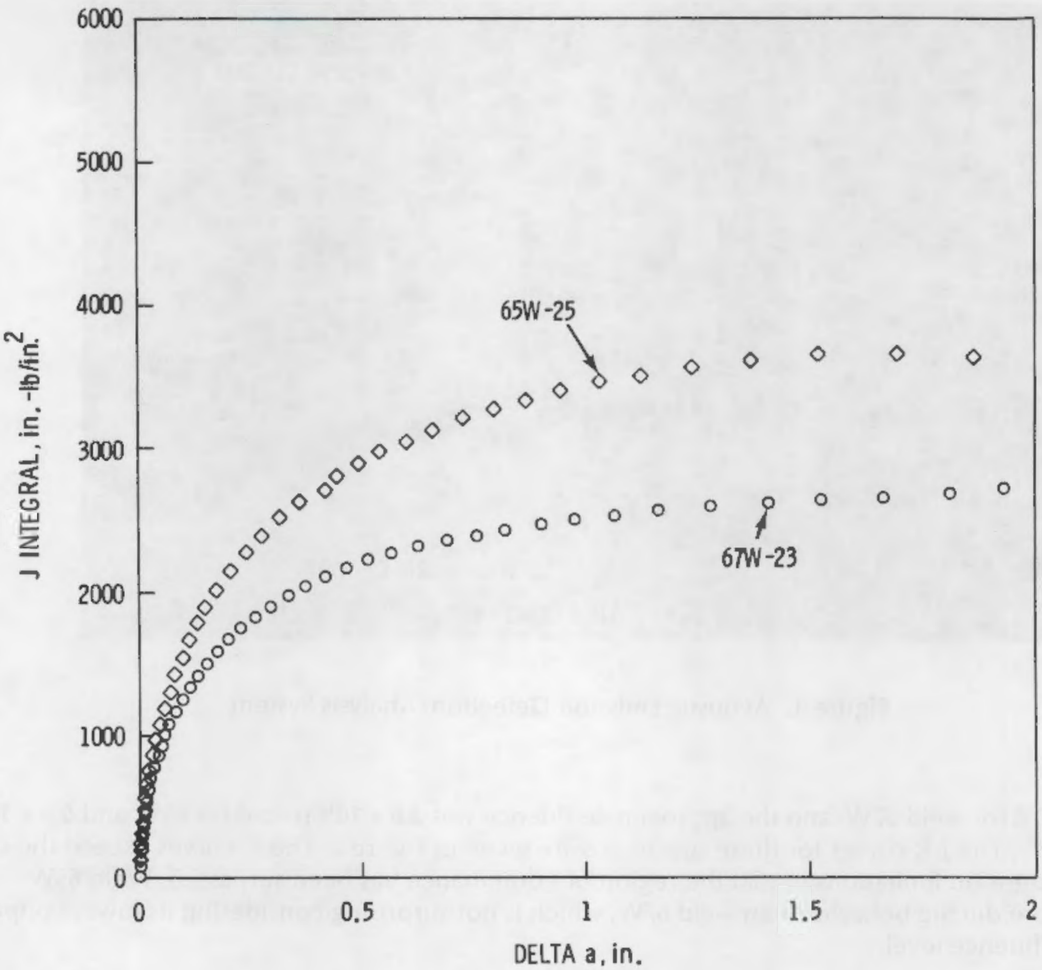

Figure 2. R Curves from Irradiated HSST Welds Tested in the Upper-Shelf Regime with 4T-CT Specimens

appears to be in a direction favorable to structural monitoring. AE data from a similar unirradiated test specimen will provide a more meaningful comparison with irradiated specimen data. Unirradiated data will be obtained when the remainder of the specimens are tested in two to three months.

REPORTS

- Analysis-Before-Test document for the balance of the program
- program review paper presented at Ninth Annual Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 26-30, 1981.
- Quick-Look report, *Acoustic Emission Monitoring of Watts Bar Preservice Cold Hydrostatic Test*
- quarterly progress report for the period July 1 - September 30, 1981.

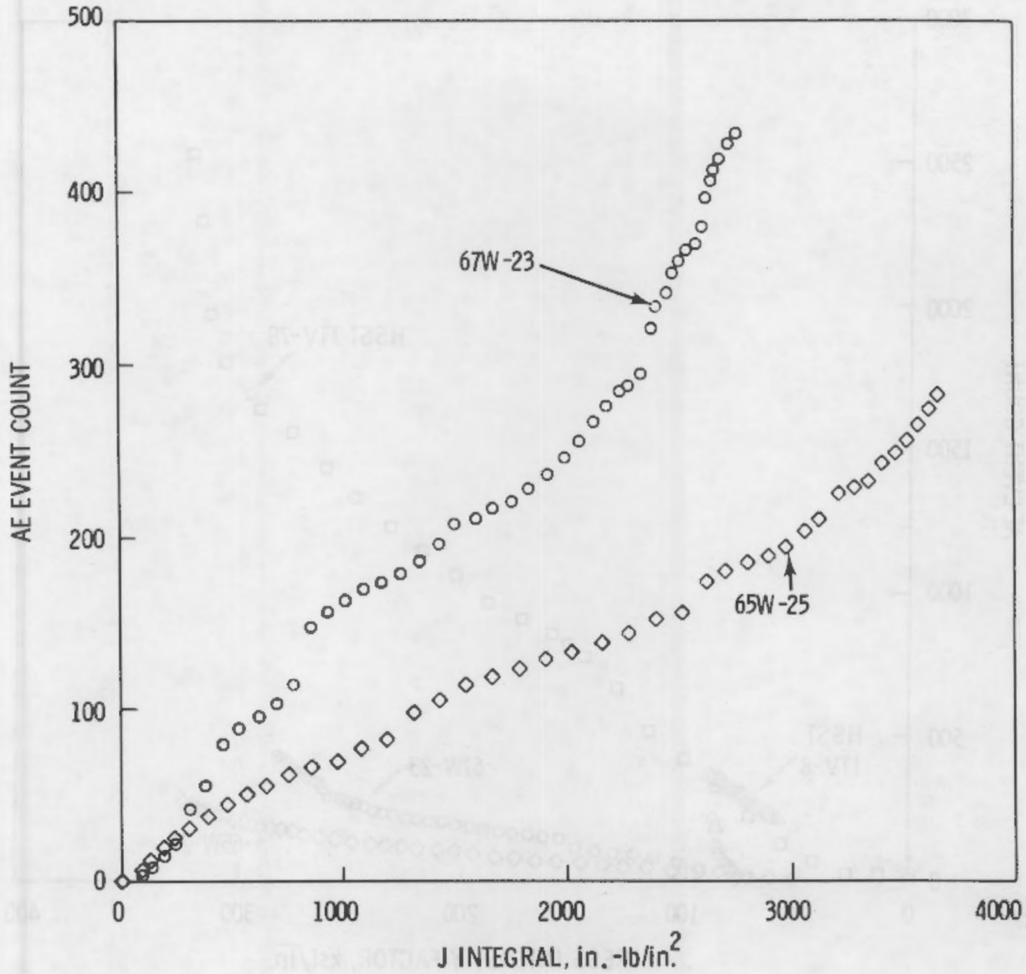

Figure 3. AE Event Count Versus J Integral for Irradiated HSST Welds 67W and 65W

FUTURE WORK

Plans for the period from January 1 to March 31, 1982, include:

- install instrumentation on ZB-1 test vessel in Germany
- start ZB-1 vessel test
- install permanent AE signal leads at Watts Bar 1 reactor
- perform initial pipe material characterization tests
- start design of engineering prototype monitor system.

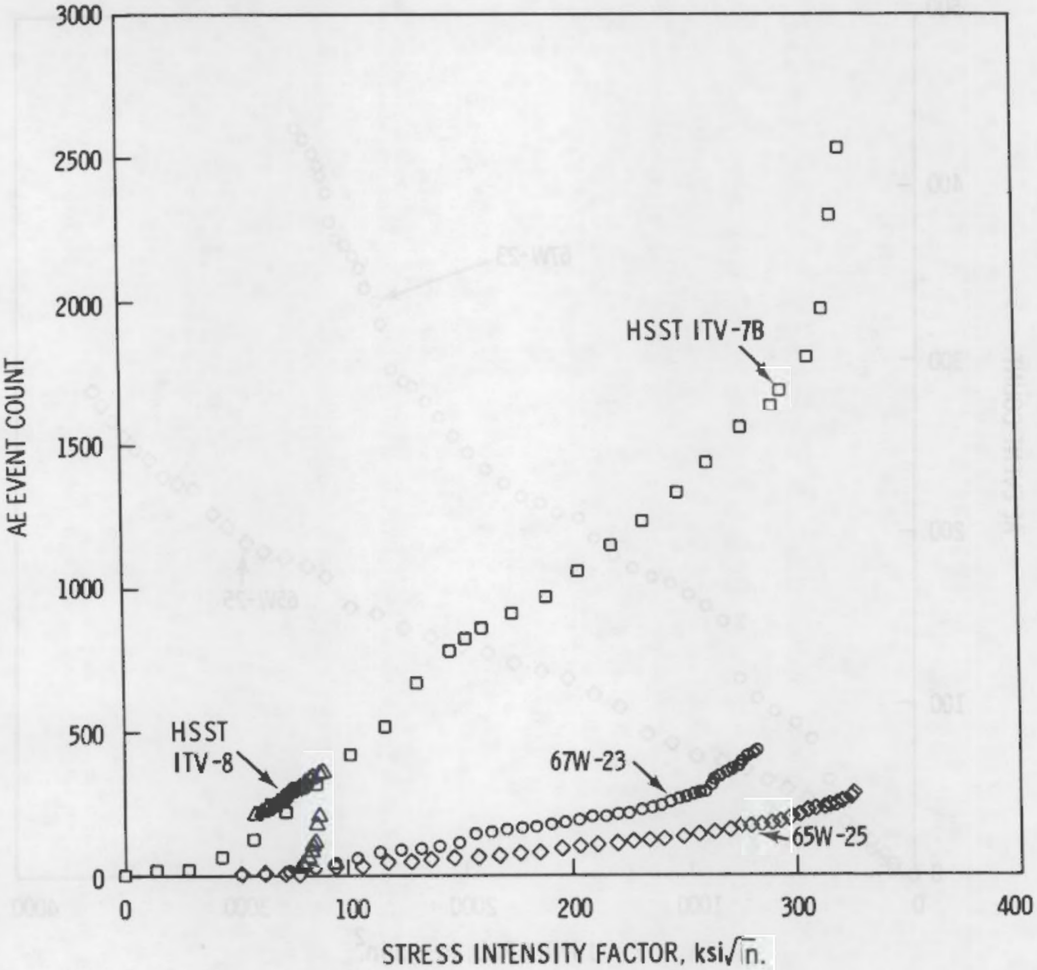

Figure 4. Comparison of AE Event Count Versus Stress Intensity Factor for NRL Irradiated Fracture Tests and HSST Vessel Tests

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[a]

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SUMMARY

During the past quarter, the piping round robin was completed, investigations concerning underclad crack detection were initiated, and deterministic and probabilistic fracture mechanics investigations continued. A preliminary performance evaluation of the first four round robin teams is reported below. Three cracks have been grown in a 4.3-in. thick clad block. Cracks that extended 0.25 in. (smallest crack available) into the base metal were detectable.

A preliminary report on flaw growth rates in pressurized water reactor (PWR) cold-leg piping was completed by Battelle-Columbus Laboratories (BCL). Parametric evaluations of flaw growth were performed to determine the degree of conservatism in the allowable flaw sizes of ASME Section XI. The flaw sizes were generally found to be acceptable except if cyclic stress levels approached the levels permitted by ASME Section III fatigue design. Efforts are in progress to integrate round robin inspection reliability data with probabilistic failure rate investigations being performed by Lawrence Livermore Laboratory (LLL).

INTRODUCTION

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

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

- determine the reliability of ultrasonic ISI performed on commercial light-water reactor (LWR) primary systems
- using fracture mechanics analysis, determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to assure a suitably low failure probability
- evaluate the degree of reliability improvement that could be achieved using improved and advanced NDE techniques

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

- based on material, service, and NDE uncertainties, formulate recommended revisions to ASME Code, Section XI, and Regulatory Requirements needed to assure suitably low failure probabilities.

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

TECHNICAL PROGRESS

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

ROUND ROBIN INSPECTION PROGRAM

Preparations for the piping round robin were completed; the first of six teams completed the examination matrix in May 1981; and the sixth team completed the examination in December 1981. Negotiations are in progress with a seventh organization to supplement data on cast SS samples. The round robin test is described in detail in Reference 1.

Preliminary Round Robin Results

Since the data presented below are preliminary and from just four teams, only limited conclusions can be drawn. In drawing conclusions, one should consider that 1) the data have been scored by computer and the results may improve or be lowered by the manual analysis now in progress and 2) the fatigue cracks that are included are conservative. The intergranular stress corrosion cracks (IGSCCs) are not necessarily conservative.

Two probabilities are reported as a function of crack depth: correct rejection (CR) and detection (DET). CR is the simple probability that a flaw of a given depth will be detected and properly classified as a crack without regard to depth sizing. DET is the probability that the defect was detected but not necessarily properly classified as a crack. The difference between the two is the probability that a crack was called a geometry signal.

The results for both IGSCCs and fatigue cracks in 10-in. Schedule 80 SS pipe are depicted in Figure 1. There was very little difference in test results. For zero size defects, a 37% probability is recorded, which represents the number of blank samples that were improperly called cracks. In Figure 2, the performance of the most and least effective teams is shown. The most effective team had a false call rate (blanks called cracks) of only 8% as opposed to 50% for the least effective team. We believe that a false call rate of less than 15 to 20% for this test is acceptable. In field applications, considerably more analysis will be performed before a pipe is cut out.

The results for the clad ferritic main coolant pipe are shown in Figure 3. Performance was considerably improved for this case; the CR probability for the improved procedure was 100% for all cracks. This degree of improvement can be attributed to the increased sensitivity—20% distance amplitude curve (DAC) as opposed to 50%—and additional training (approximately 4 hr).

The results for cast SS main coolant pipe are not so encouraging. One team achieved a maximum CR rate of 80%. However, their false call rate on blank sample was 50%. The CR rate of the remaining teams was zero or less than 25%. The flaws in this material are conservative (short and tight); however, they are identical to those in the clad ferritic pipe.

Axial Position Accuracy

A key parameter in distinguishing between cracks and geometrical reflectors is the axial position of the defect relative to the root or other geometric discontinuity. Manual plotting is generally used to

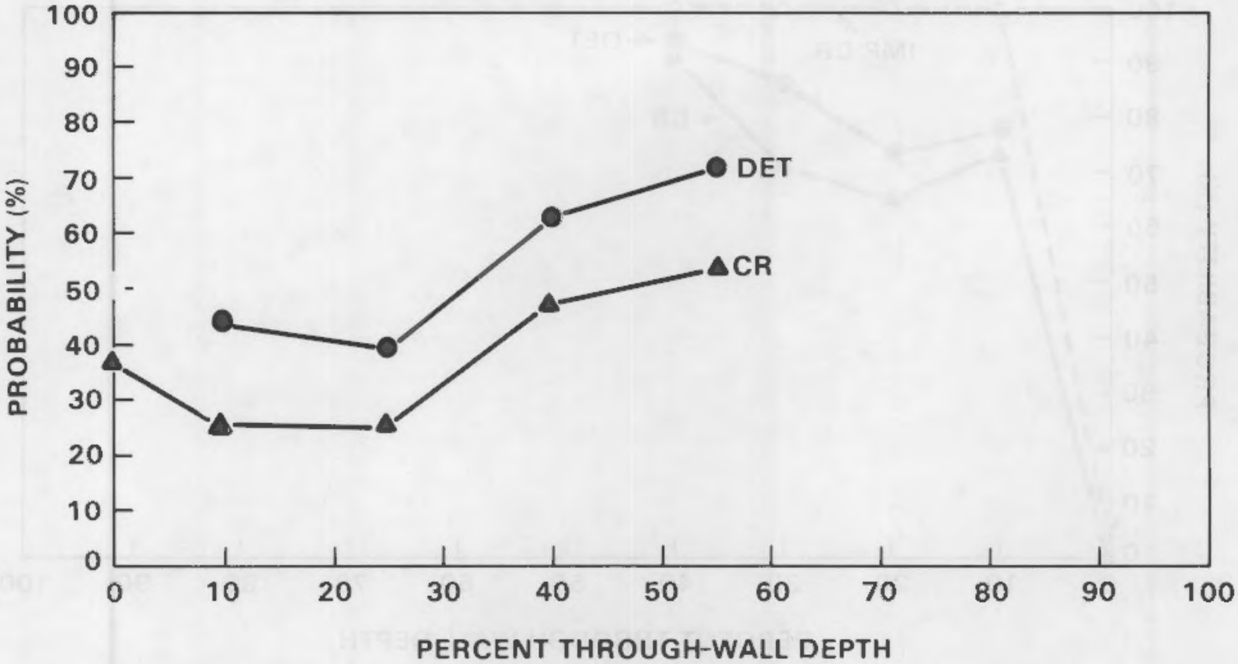

Figure 1. Probability of Detection (DET) and Correct Rejection (CR) of IGSCCs and Fatigue Cracks in 10-in. Schedule 80 SS Pipe as a Function of Through-Wall Depth for Four Teams

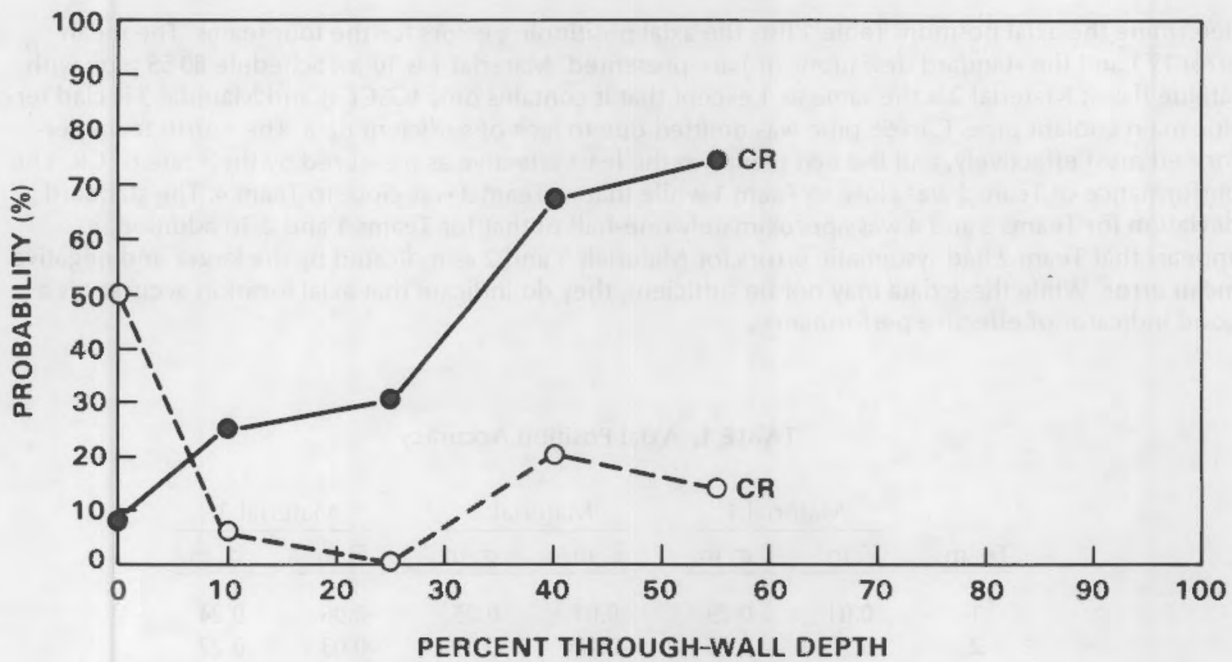

Figure 2. Probability of Detection (DET) and Correct Rejection (CR) of IGSCCs and Fatigue Cracks in 10-in. Schedule 80 SS Pipe as a Function of Through-Wall Depth for Most and Least Effective Teams

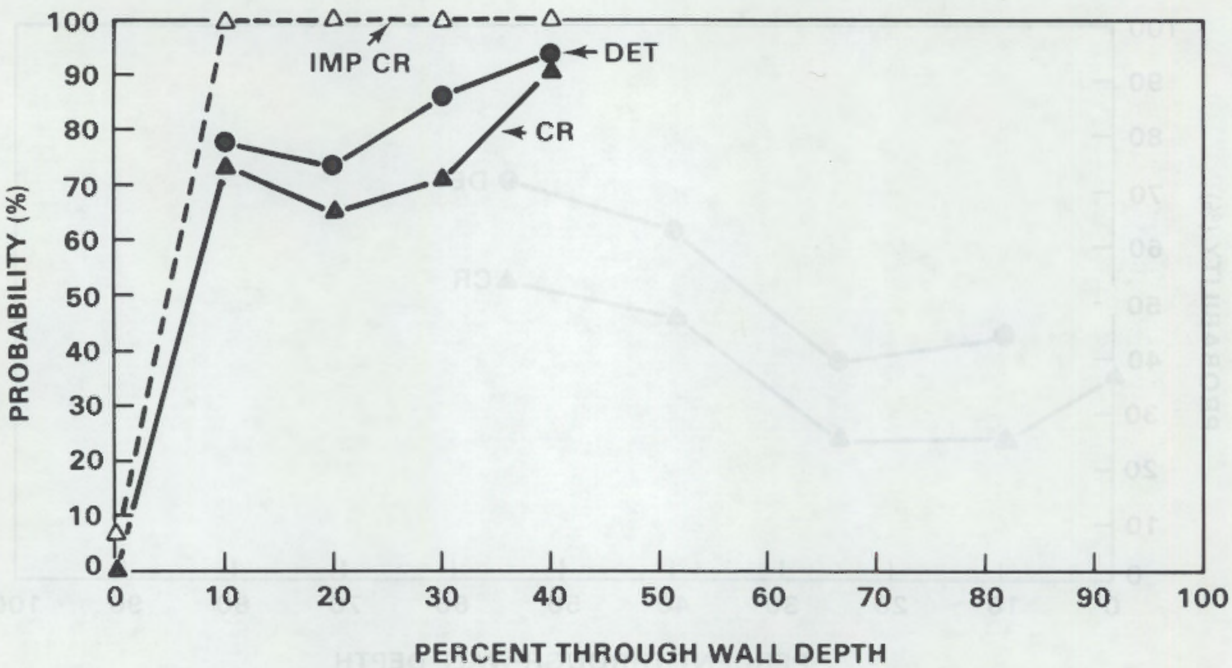

Figure 3. Probability of Detection (DET) and Correct Rejection (CR) of Fatigue Flaws in Clad Ferritic Main Coolant Pipe. CR for the improved procedure (IMP CR) is also shown.

determine the axial position. Table 1 lists the axial positioning errors for the four teams. The mean error ($\bar{X}$) and the standard deviations (σ) are presented. Material 1 is 10-in. Schedule 80 SS pipe with fatigue flaws; Material 2 is the same as 1 except that it contains only IGSCCs; and Material 3 is clad ferritic main coolant pipe. Cast SS pipe was omitted due to lack of sufficient data. The fourth team performed most effectively, and the first team was the least effective as measured by their rate of CR. The performance of Team 2 was close to Team 1 while that of Team 3 was close to Team 4. The standard deviation for Teams 3 and 4 was approximately one-half of that for Teams 1 and 2. In addition, it appears that Team 2 had systematic errors for Materials 1 and 2 as indicated by the larger and negative mean error. While these data may not be sufficient, they do indicate that axial location accuracy is a good indicator of effective performance.

TABLE 1. Axial Position Accuracy

	Material 1		Material 2		Material 3	
Team	$\bar{X}$, in.	σ, in.	$\bar{X}$, in.	σ, in.	$\bar{X}$, in.	σ, in.
1	0.01	0.29	0.03	0.25	-0.06	0.24
2	-0.12	0.25	-0.18	0.29	-0.03	0.22
3	0.002	0.15	0.03	0.17	-0.05	0.12
4	0.06	0.14	0.06	0.12	-0.04	0.10
All	-0.004	0.223	-0.002	0.211	-0.047	0.176

Preliminary Conclusions from the Round Robin

The following conclusions, which are listed in descending order of confidence, can be drawn from this preliminary review of available round robin data:

- Inspection of clad ferritic main coolant pipes can be highly effective if sufficient inspection sensitivity is used.
- Large differences in performance between teams (all meeting Code) were observed.
- The care and accuracy of plotting the axial position of indications appears to be an indicator of performance effectiveness.
- Access to flaws (near side or far side of the weld) had no statistical significance for clad ferritic pipe; flaws could be detected from either side of the weld.
- Procedures to detect cracks in centrifugally cast SS pipes were the least effective of those tested.
- Little difference was noted between the laboratory and difficult conditions as applied in this test.
- Performance improvements resulting from the "improved procedure" were modest, except for the clad ferritic case where improvement was significant.

Conclusions concerning the remaining parameters of the round robin must await further data and analysis.

MEASUREMENT AND EVALUATION

While the past two quarters have been chiefly occupied with the execution of the round robin, measurements have progressed in the inspection of SS welds with single-side access.

Stainless Steel Pipe Welds

Effective ultrasonic inspection of the far side of a SS pipe weld may prove exceedingly difficult. The following two experiments demonstrate this.

A set of electric discharge machined (EDM) notches was fabricated in 10-in. Schedule 80 SS pipe. They were located on the pipe inside diameter (ID) in the weld root and counterbore region. All were identical in size and shape; 25% through-wall depth; and 0.2 aspect ratio (a/ℓ). All were oriented parallel to the weld so as to be detected by an axial ultrasonic scan. One notch was in the center of the root bead, the next was adjacent to the root bead, and the rest were stepped away from the root with each notch 0.050 in. further than the last. With a total of eight notches, the most remote was 0.300 in. from the edge of the root bead.

When inspected from the near side, all notches gave high-amplitude signals; and most of them were several decibels above a 10% notch DAC. When inspected from the far side, however, the amplitudes were extremely low (when the notches could be detected at all). No search unit seemed able to detect all eight from the far side. Flaws of this size are, of course, considered rejectable.

The second experiment consisted of sending an ultrasonic beam through the metal from an angle beam transducer on the outside diameter (OD) using an electromagnetic acoustic transducer (EMAT) to map the beam incident at the pipe ID. A welded section of large-diameter SS pipe was machined so that OD and ID surfaces were flat. A common brand of 1/2-in. 2.25-MHz angle beam transducer was used to transmit a 45° beam from the OD. The first map was taken in base metal on one side of the weld; the transmitter was then incremented toward the weld and across it so that the final map was taken in base metal on the other side of the weld. Using 0.1-in. increments, 25 maps were taken with the results shown in Figure 4 (every other map has been omitted to save space).

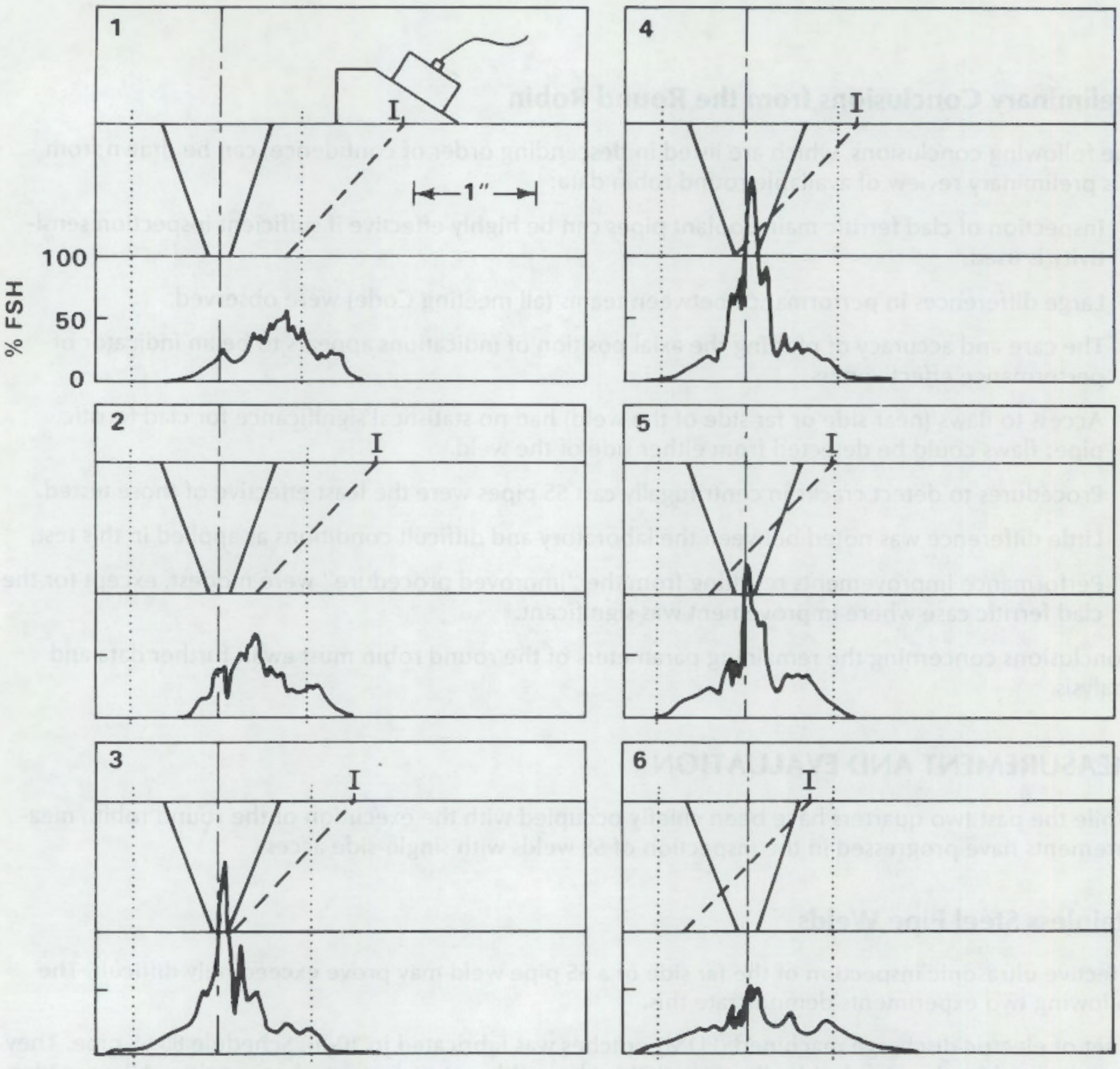

Figure 4. Beam Profile on Inside Diameter of a Welded SS Pipe Segment Using an EMAT Detector. Profile maps 3 through 9 show redirection of the sound beam due to the dendritic structure of the weld.

Clearly, the beam was drawn to the weld root. In maps 3 through 6, the beam index moved 0.6 in.; but the beam peak did not move at all. The diagonal dashed line represents the 43° measured beam angle—the angle an ultrasonic inspector would assume. Maps 9 through 12 show the beam finally "punching through" at its intended 43° angle. The vertical dotted lines mark the outer axial boundaries of the required inspection volume. There was no transducer position capable of sending its beam peak into the far-side required inspection volume.

These two experiments graphically illustrate the severe problems in inspecting the far side of a SS pipe weld with single-side access. Any equipment or procedure touted as capable of such inspection should demonstrate its effectiveness on defects in representative pipe weld zones.

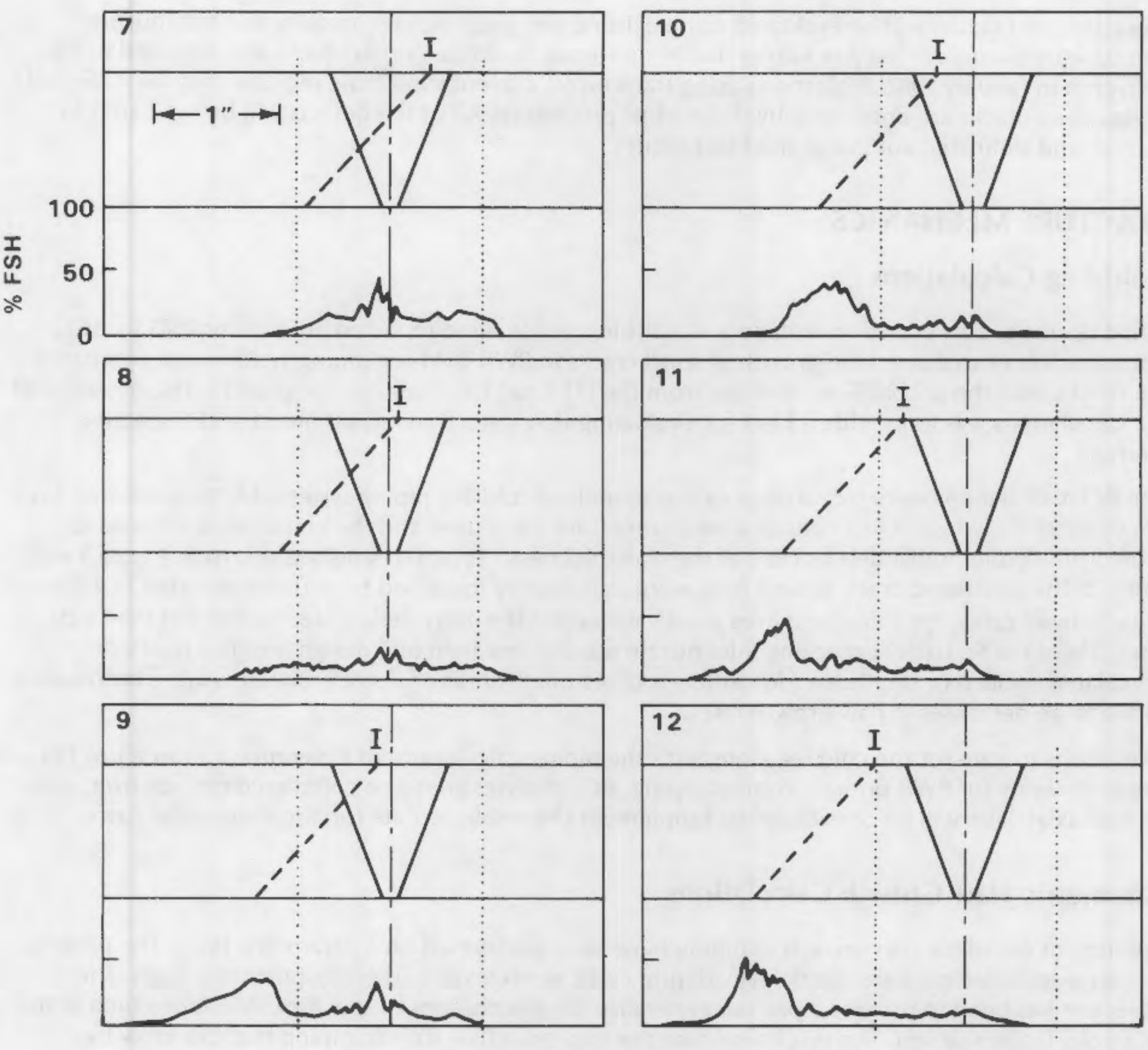

Figure 4. (contd)

VESSEL APPLICATIONS

The major objective of this task is to investigate underclad crack detection reliability. Underclad cracks are of concern in some older reactors in the event of pressurized thermal shock. Information to evaluate techniques that are proposed to meet the requirements of Regulatory Guide 1.150 will also be obtained.

A 21-in. diameter SA553-B, 4.3-in. thick dropout from a pressurizer was received from the Babcock & Wilcox Company. The block was clad with 308L SS with a nominal thickness of 0.25 in. Underclad defects will be grown in the dropout sample by the thermal fatigue technique.[1] Three defects have been grown to depths of approximately 0.25 and 0.5 in. in the base metal. Flaws both parallel and perpendicular to the direction of the cladding are planned.

The fabricated cracks will be evaluated using existing and proposed techniques and transducers. Transducers commonly used in Europe for this purpose have been ordered and are expected to be delivered in January 1982. Evaluations using transducers currently available indicate that the 0.25- and 0.5-in. deep cracks are detectable by the manual procedure. All of the defects will be evaluated by manual and simulated automatic field techniques.

FRACTURE MECHANICS

Cold-Leg Calculations

Initial results of flaw growth calculations in cold-leg piping were reported in October 1981 by BCL. The calculations evaluated the growth of small cracks in PWR cold-leg piping systems and compared the results with the probabilistic analyses from the LLL Load Combination Program.(2) The objective of the calculations was to provide a basis for evaluating flaw detection requirements and inspection intervals.

The BCL calculations were based on previous models of cold-leg piping systems.(3) They differed from the original study in that less conservative assumptions were used and the cracks were allowed to grow from smaller initial depths (10% of the wall thickness). Typical results are shown in Figure 5 and Table 2. The calculated crack growth lives were significantly increased from those reported in Reference 3. In all cases, the calculated lives greatly exceeded the 40-yr design life, except that the calculated life of the St. Lucie-1 charging inlet nozzle was still less than one design life. The smaller postulated initial flaw depths and lower levels of assumed vibrational stresses were major contributors to the large decreases in flaw growth rates.

The results to date for the cold leg along with the piping calculations of Reference 3 show a low failure probability for PWR primary coolant piping. BCL analyses are to be completed next quarter; postulated axial flaws will be considered to supplement the results to date for circumferential flaws.

Parametric Flaw Growth Calculations

Predictions of fatigue flaw growth in piping have been performed on a parametric basis. The parameters have included pipe size, initial flaw depth, cyclic stress level, and environmental effects. The objective has been to provide a basis to generalize the conclusions of specific calculations such as the BCL cold-leg evaluation. The results indicate the minimum flaw sizes in piping that can grow by fatigue during an inspection interval and result in a leak or break.

Figure 6 shows typical results that will be presented in a detailed report that is being prepared. The main conclusion is that the present ASME Section XI flaw acceptance standards insure a reasonably uniform fatigue life relative to pipe size and flaw shape. However, the predicted lives are unacceptably short if the cyclic stresses approach the levels permitted by the ASME Section III fatigue design curve.

Probabilistic Fracture Calculations

On November 23-24, 1981, a meeting was held at PNL to discuss cooperative efforts with LLL. The flaw detection data of the PNL round robin inspection will be made available for probabilistic fracture mechanics predictions at LLL. The results of calculations at LLL will in turn be used to evaluate the adequacy of present flaw detection capabilities and inspection intervals. Alternate inspection scenarios will also be addressed to establish the required improvement in flaw detection capability for specific piping systems and failure modes that have been analyzed at LLL.

During this quarter a simple Markov-chain model was developed to aid in applying the LLL failure rate curves. This model will be used to describe the distribution of flaw sizes in a weld as a function of

TABLE 2. Results of the Fracture Mechanics Analyses for Cold Leg Using the Revised Input Data

Analysis	Plant	Joint Location(a)	Component Type(b)	Outside Diameter, in.	Wall Thickness, in.	Material	Initial Crack Size, a_i, in.	Final Crack Size a_f, in.	40-yr Lives	Comments
1	Farley-1	D (BCL Model) 13 (LLL Model)	SP	32.19	2.325	SA351 Grade CF8A	0.25	1.30	20.87	da/dN - ΔK, Figure 16, $\Delta K_{th} = 0$ - Original, Reference 3
							0.25	1.30	52.20	- Trilinear, Upper Bound, Reference 3
							0.25	1.30	>100.00	- Reference 2
2	St. Lucie-1	165B/170TR	E	36.00	3.00	SA516 Grade 7	0.25	1.50	63.90	
3	St. Lucie-1	479TR/465B	SP	36.00	3.00		0.25	1.50	52.40	
4	St. Lucie-1	165B/170TR	SP	35.00	2.50		0.25	1.25	31.47	
5	St. Lucie-1	180TB/170T	SP	36.00	3.00		0.25	1.50	63.39	
6	St. Lucie-1	479TBP	SP	36.00	3.00		0.25	1.50	44.06	
7	St. Lucie-1	453TR	DMW	35.00	2.50		0.25	1.25	14.75	
8	St. Lucie-1	154TH	DMW	35.00	2.50		0.25	1.25	13.35	
9	St. Lucie-1	155T	N	35.00	2.50		0.71	2.36	0.60	40-yr lives for leak

(a) For joint location refer to Reference 3, Figures 7 through 9 for Farley-1 or Figures 3 through 6 for St. Lucie-1.

(b) E - elbow
SP - straight pipe
DMW - dissimilar metal weld
N - nozzle.

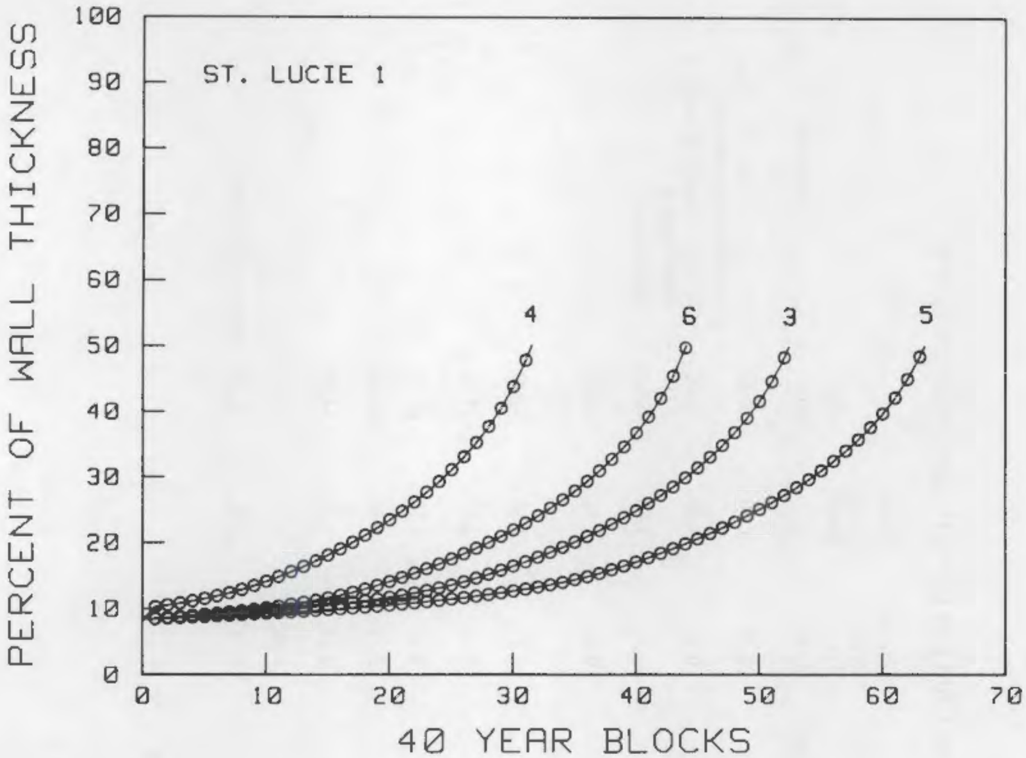

Figure 5. Results of the Crack Growth Analysis for Straight Pipe; St. Lucie-1 (numbers refer to the analysis numbers of Table 2)

time. Data from the LLL probabilistic calculations will be used to estimate the parameters of the Markov-chain so that both models produce approximately the same flaw distributions (over time). The Markov-chain model will allow the effect of inspections to be readily evaluated by variations in the inspection intervals and the probability of inspection curves.

A matrix of calculations has been formulated to apply the results of the LLL analyses for the Zion primary coolant loop. The implications of the PNL round robin flaw detection probabilities will be studied. The impact of alternate weld joint sampling plans will be evaluated with calculations that will estimate the inspection unreliability due to Code sampling plans.

FUTURE WORK

The following areas will be emphasized during the coming quarter.

- The round robin analysis will be completed.
- The plate and cladding for the underclad crack evaluations will be acquired.
- The cold-leg analysis will be completed by BCL.
- The improved and advanced technique evaluation will begin, and the flaw growth in the pressurizer drop-out sample will be completed.

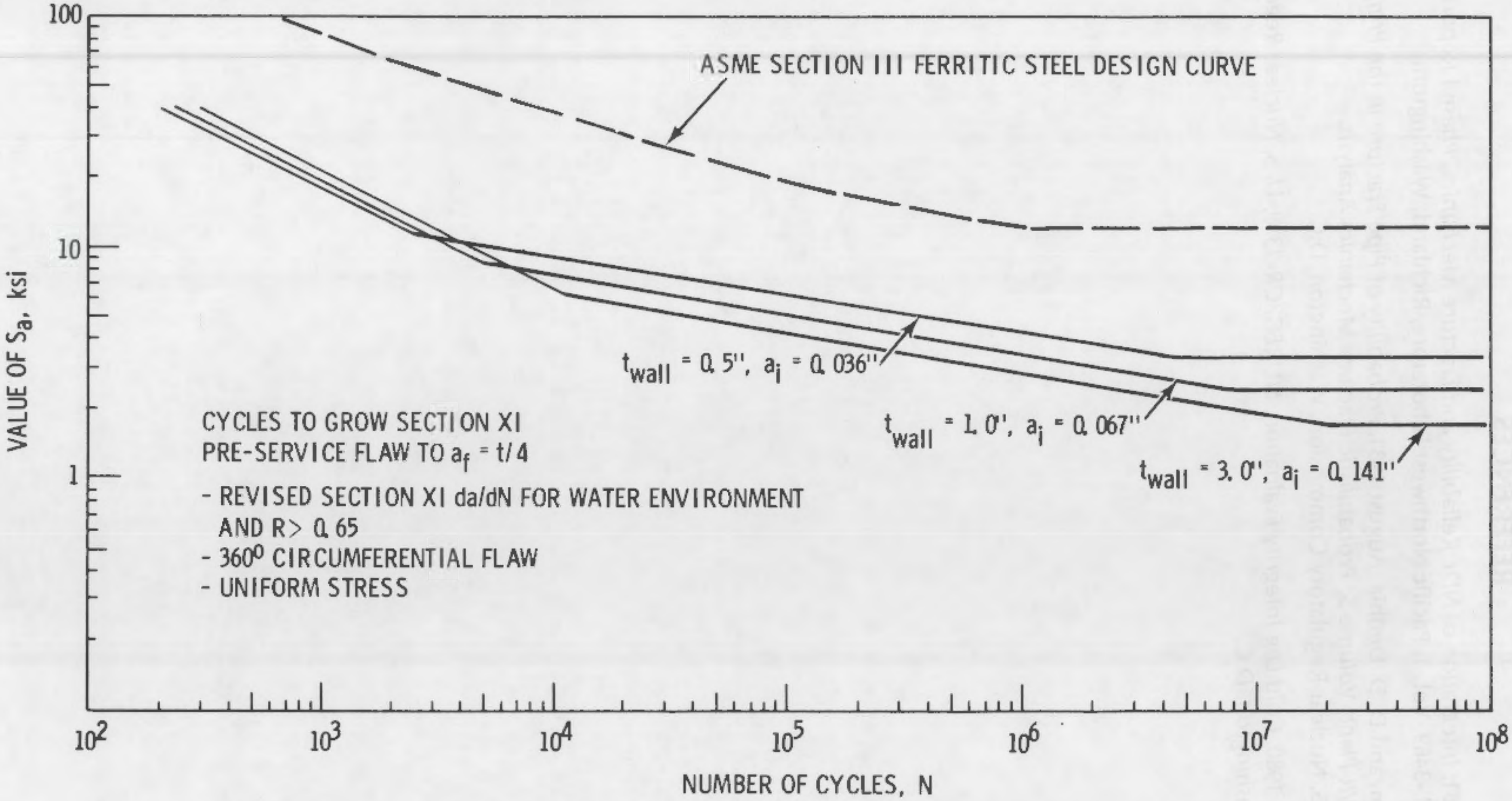

Figure 6. Predicted Life for Defected Ferritic Piping Based on ASME Section XI Preservice Acceptance Standards

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EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

TASK A - IRRADIATION EXPERIMENTS[a]

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

M. E. Cunningham
R. E. Williford

SUMMARY

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

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

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

INTRODUCTION

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

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

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

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

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

TECHNICAL PROGRESS

IFA-432 was removed from the reactor in June 1981 in preparation for shipment to Harwell, U.K., for PIE. The instrumentation in this highly characterized six-rod assembly demonstrated a remarkable survival rate: four fuel thermocouples and three pressure transducers were operating at its discharge burnup of 30 MWd/kgU. It was considered desirable to push this assembly on to a higher burnup of about 50 MWd/kgU; however, this would entail the risk of instrument and rod failure. The following two-part PNL proposal was accepted by the NRC:

- continue irradiation of four IFA-432 rods (2, 3, 5, and 9) to a burnup of 50 MWd/kgU
- send two rods (1 and 6) to Harwell, U.K., with IFA-527 rods for extensive PIE, including radial distribution of retained fission gas, fuel density, and cladding inside diameter (ID) examination.

The shipment to Harwell was scheduled for October 1981, but it was postponed until March 1982 so that rods 1 and 6 could be combined with DOE fuel for cost savings. (The DOE fuel will be sufficiently cooled by March.)

The program also assumed responsibility for the 12-rod DOE (Fuel Performance Improvement Program) assembly IFA-518. Four alternate fuel rod designs are included: 1) sphere-pac rods, 2) rods with annular pellets, 3) rods with annular pellets and graphite-coated cladding, and 4) prepressurized rods with annular pellets and graphite-coated cladding. The rods will be brought to burnups of 20 to 40 MWd/kgU to assess their full life performance. No fuel thermocouples are included in the assembly, but pressure transducers and elongation sensors are provided. IFA-518 is discussed in more detail under Task B of this report (see p. 25).

FUTURE WORK

Contractual negotiations between Halden, Harwell, and PNL relating to irradiation examination and shipping costs and schedules will continue. Current uncertainty in NRC program funding is hampering these negotiations.

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, a paper comparing observed centerline temperature scatter with calculated temperature uncertainty was presented at the American Nuclear Society Winter Meeting. The decision was made to continue irradiating four IFA-432 rods to higher burnup; rods 1 and 6 were removed for destructive postirradiation examination (PIE). Test assembly IFA-518 was transferred from the U.S. Department of Energy (DOE) Fuel Performance Improvement Program to this program to obtain higher burnup data.

INTRODUCTION

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

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

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

TECHNICAL PROGRESS

This quarter's activities are discussed below by subtask.

SUBTASK B-1 - DATA PROCESSING

After IFA-432, IFA-513, and IFA-527 were removed from the reactor in April and June 1981, the decision was made to reinsert IFA-432 to obtain higher burnup data. Rods 1 and 6 were removed from the assembly for destructive PIE; IFA-432 is operating with only rods 2, 3, 5, and 9. It has been estimated that the removal of two rods should have only a minimal effect on the power calibration (beyond reducing the total assembly power to four-sixths of the original). The reactor restarted late in December 1981, and initial power/temperature data (when received) will be used to check for possible power calibration changes.

A Halden test assembly from the DOE Fuel Performance Improvement Program was transferred to this program for continued irradiation to higher burnup. The 12-rod assembly was the third reload of IFA-518[a] and contains a reference design and four alternate fuel designs that are being evaluated for their ability to reduce pellet-cladding interaction (PCI).

The basic design of IFA-518.3 is similar to the other assemblies in this program; a schematic of the assembly is presented in Figure 1. The assembly consists of two 6-rod bundles; the 12 rods are instrumented with either cladding elongation sensors or rod internal gas pressure sensors. A brief summary of the makeup of IFA-518.3 and its irradiation history prior to transfer is presented in Table 1. References 1 and 2 contain precharacterization information for the fuel rods and additional information on instrumentation and assembly design.

SUBTASK B-2 - DATA REPORTS

A paper titled "Predicted Versus Measured Scatter in Centerline Temperature from Replicate Fuel Rods"[3] was presented at the Winter Meeting of the American Nuclear Society (November 29 through December 3, 1981, San Francisco, California). This paper was based on data from three sets of replicate fuel rods that have been irradiated during the course of this program:

- Set 1 - rods 1, 3, and 5 of IFA-513 (identical design and irradiation history); helium-filled rods
- Set 2 - rod 1 of IFA-431, IFA-432, and IFA-513 (identical design but different irradiation histories and peak powers); helium-filled rods
- Set 3 - rods 1 through 5 of IFA-527 (identical design and irradiation history); xenon-filled rods.

Examination of temperature histories showed consistent behavior between rods of identical design with the observed data scatter being less than calculated uncertainties. Centerline temperature data from the helium-filled rods (Sets 1 and 2) compared well with calculated temperatures.

The power difference among the rods in Set 2 manifested itself as a difference in the time to substantial fission gas release and subsequent change in temperature versus power behavior. Prior to the onset of fission gas release, the three rods behaved very similarly even though they had been subjected to different peak powers and irradiation histories.

(a) Designated IFA-518.3.

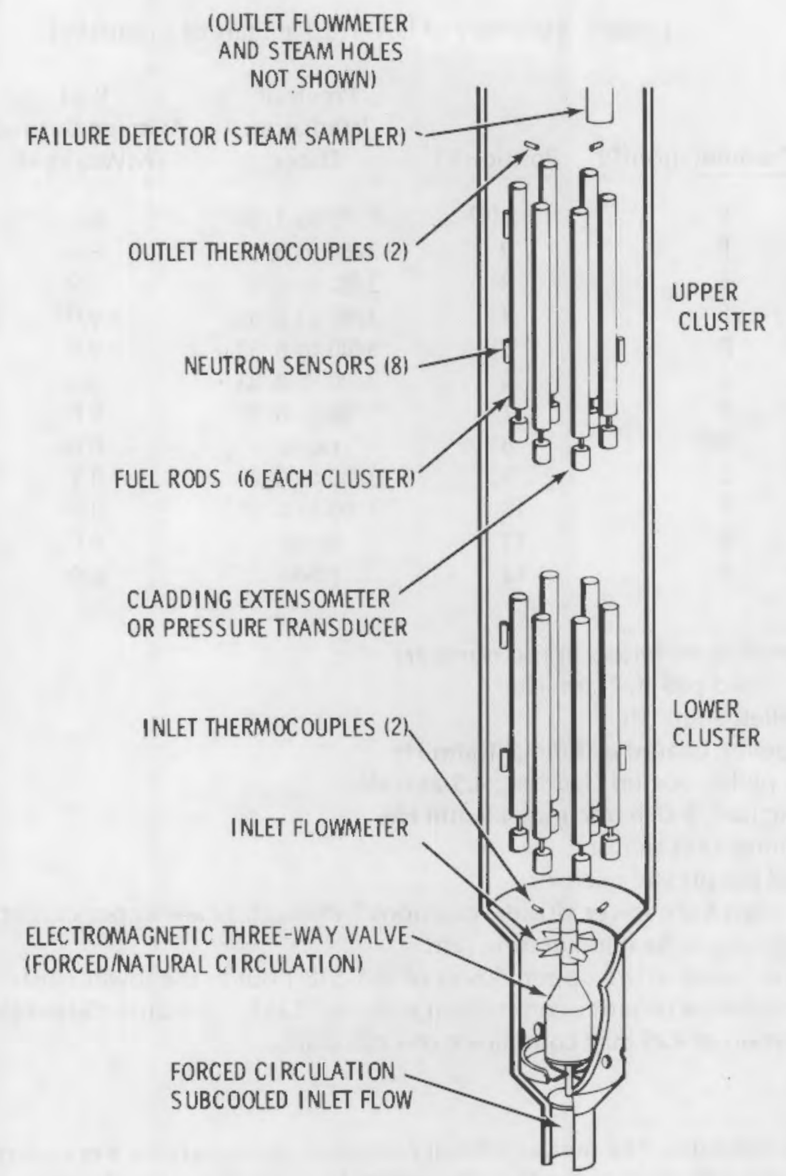

Figure 1. Schematic of Instrumented Fuel Assembly (IFA)-518 Test Rig

SUBTASK B-3 - DATA ANALYSIS

Fission Gas Release in IFA-432

Fuel rods 1, 5, and 6 of IFA-432 were instrumented with pressure transducers to monitor the internal pressure changes that occurred during irradiation. The pressure data (obtained up to 30 GWd/MTU rod average burnup) have been used to estimate fission gas release as a function of burnup. These estimated releases are being compared to calculations based on fission gas release models that are currently being used in the FRAPCON-2[4] computer code.

In calculating fission gas release, the temperature and power histories of the individual fuel rods provided the input for the specific gas release models in order to directly compare the calculated releases

Table 1. Summary of IFA-518.3 at Start of Irradiation

Rod No.[a]	Instrumentation[b]	Position[c]	Previous Irradiation Dates	Rod Average Burnup, MWd/kgM	Rod Average LHGR,[d] kW/m
A7	E	1[e]	7/78 to 1/80	8.5	23
R24	P	2	3/80 to 8/81	8.9	34
SP33	E	3	3/80 to 8/81	8.9	34
R23	E	4	3/80 to 8/81	9.0	34
ACP29	P	5	3/80 to 8/81	9.8	34
ACP28	E	6	3/80 to 8/81	9.9	34
ACP31	P	7	3/80 to 8/81	9.0	30
R26	P[f]	8	none	0.0	—
SP34	E	9	3/80 to 8/81	8.5	30
R25	E	10	3/80 to 8/81	8.4	30
AC12	E	11	none	0.0	—
ACP32	E	12	none	0.0	—

(a) Rod type is specified by letters in rod number:
R = reference - solid pellet, 1-atm He
A = annular pellet, 1-atm He
AC = annular pellet, coated cladding, 1-atm He
ACP = annular pellet, coated cladding, 4.5-atm He
SP = sphere-pac fuel, 3-size fraction, 4.5-atm He.
(b) E = cladding elongation sensor
P = rod internal gas pressure sensor.
(c) Positions 1 through 6 are lower cluster; positions 7 through 12 are upper cluster.
(d) Lifetime average linear heat generation rate.
(e) Rod A7 was positioned in the upper cluster of IFA-518.1 but in the lower cluster of IFA-518.3.
(f) Instrumentation design requires comparison with rod R25 for pressure determination; however, previous irradiation of R25 may complicate this calculation.

to the experimental estimates. The measured fuel centerline temperatures were corrected for irradiation-induced decalibration of the thermocouples by using the relationship:

$$T_{corr} = (T - 20)(0.0054)(BU) + T$$

where T_{corr} = corrected temperature, °C
T = measured temperature, °C (cold junction = 20°C)
BU = burnup, GWd/MTU.

After the thermocouple at the upper (high-power) location failed, the fuel centerline temperatures at this location were estimated from the linear heat generation rate, q, and the fuel rod thermal resistance, R, by the relationship:

$$T = qR + 240$$

A linear extrapolation of the thermal resistance versus power plots from the lower thermocouple locations was used to determine the thermal resistance at the upper thermocouple location as a function of burnup.

Fission gas release calculations have been made using the ANS-5.4 fission gas release model (both with and without the burnup enhancement factor on the diffusion coefficient) and with the Beyer-Hann fission gas release model as it is programmed in the subroutine GASREL from FRAPCON-2 and GAPCON-THERMAL-3.[5] The version of GASREL in FRAPCON-2 uses a different low-temperature gas release model plus the U.S. Nuclear Regulatory Commission (NRC) correction factor for accelerated gas release at burnup levels above 20 GWd/MTU. The results of the calculations at burnup levels of 10, 20, and 30 GWd/MTU are compared to the experimental estimates in Table 2.

The calculations based on the Beyer-Hann model agree quite well with the experimental estimates for all three fuel rods. The small differences in the calculated releases from the two versions of GASREL result from the different low-temperature gas release models and the inclusion of the NRC correction factor in the model from FRAPCON-2. None of the experimental data show any evidence of burnup-enhanced fission gas release. Use of the NRC correction factor did not significantly affect the calculated releases at 30 GWd/MTU and therefore was not observable in the comparisons. At higher burnups, the contribution from the NRC correction factor will become more significant and the subsequent pressure data from rod 5 may be very helpful in defining burnup-enhanced fission gas release.

Calculations based on the ANS-5.4 model without any burnup enhancement also show good agreement with experimental gas release estimates as a function of burnup. However, when the burnup-enhancement factor on the diffusion coefficient is included in the calculations, the model significantly overpredicts the experimental gas release estimates at burnup levels above 10 GWd/MTU. It is interesting to note that the ANS-5.4 model without burnup enhancement predicts gas releases at 30 GWd/MTU, which is very similar to those from the Beyer-Hann model with the NRC high-burnup correction included.

Table 2. Comparison of Experimental Fission Gas Release Estimates from IFA-432 to Calculated Releases Based on the Beyer-Hann and ANS-5.4 Fission Gas Release Models

			ANS-5.4 Model		Beyer-Hann Model (GASREL)	
Rod No.	Burnup, GWd/MTU	Experimental Gas Release Estimates, %	With Burnup Enhancement, %	Without Burnup Enhancement, %	GT-3,[a] %	FRAPCON-2,[b] %
1	10	2 to 4	6	4	4	3
	20	5 to 8	21	9	5	4
	30	4 to 6	29	9	5	9
5	10	8 to 12	12	8	7	5
	20	6 to 9	16	7	8	5
	30	6 to 9	23	7	8	8
6	10	9 to 14	16	9	9	7
	20	9 to 14	26	12	10	8
	30	9 to 15	34	12	10	12

(a) GAPCON-THERMAL-3.
(b) Includes NRC high-burnup correction factor.

FUTURE WORK

Data processing for the next quarter will continue as required. Work will continue on data reports for IFA-432 and IFA-527 that will present the data obtained until their removal from the reactor in June and April 1981, respectively. Analysis of fuel cracking and relocation, fission gas release, and related subjects will continue.

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EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES:

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

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

M. E. Cunningham
W. N. Rausch

SUMMARY

Work continued on establishing a link between FRAPCON-2 and the FASTGRASS fission gas release code with assistance from Argonne National Laboratory (ANL). A new gap conductance model is being prepared and will be included in FRAPCON-2.

The three-dimensional (3-D) pellet-cladding interaction (PCI) code FRAGMT nears completion. The code can model the effects of cracked fuel on cladding ridges, axial elongation, and fuel conditioning/deconditioning. Progress was made on analyzing the fuel rod mechanical compliance data from tests performed at Pacific Northwest Laboratory (PNL) and Harwell, U.K.

INTRODUCTION

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

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

Task C efforts include: code maintenance, analysis of data from fuel mechanics experiments, and PCI model development.

TECHNICAL PROGRESS

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

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

SUBTASK C-1 - FRAPCON-2 CONTROL AND MAINTENANCE

Programming incompatibilities persisted between FRAPCON-2 and the ANL fission gas code FASTGRASS. Attempts to link the codes into a consistent unity revealed that many careful modifications were required. ANL assistance helped to resolve these difficulties; communications have included code exchanges to facilitate test runs.

A link to the INEL computer system was established to reduce computation costs for the larger cases to be run with FRAPCON-2.

A new algorithm for calculating gap conductance was received from the University of Missouri at Columbia and is presently under numerical evaluation in preparation for installing it into FRAPCON-2.

SUBTASK C-2 - PELLET-CLADDING INTERACTION MODEL DEVELOPMENT

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

The convergence algorithm has recently been refined so that the solution reaches equilibrium at the correct choice from the multiple roots of the equilibrium equations. Parametric calculations have begun and will be used to construct distributions of cladding stress concentrations as functions of pellet cracking, axial position in the rod, and power history. These distributions will be used to develop a FRAPCON fuel failure model. Recent calculations with FRAGMT have shown that for a typical fresh boiling water reactor (BWR) rod 0.004- to 0.006-in. axial gaps form between pellets and that cladding axial elongation varies with axial position in the rod. Axial elongation appears to be significantly dependent on pellet-cladding and interfragment friction coefficients.

Analysis of the data from the mechanical compliance tests performed at Harwell, U.K., has shown that there may be a relationship between ridge heights and rod bowing.

FUTURE WORK

The following activities are planned for the next quarter:

- complete the verification of the new FASTGRASS/FRAPCON-2 combination
- issue the Version 1 Mod 3 (V1M3) edition of FRAPCON-2, which will include the changes listed in the "Users' Letter" and the new FASTGRASS link
- begin work on an IBM-compatible version of FRAPCON-2
- complete the analysis of the mechanical compliance test data
- complete the parametric calculations with the 3-D PCI code FRAGMT and construct cladding stress concentration distributions for BWR rods.

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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

Proof tests of the strain-measuring instrument with substitute strain gages attached were essentially completed with good results. The pressure vessel was completed, tested, and received at Pacific Northwest Laboratory (PNL) in December 1981. The instrument and vessel are being readied for installation in the pressurized hot water loop facility following modifications to the loop piping system. The contract for the irradiated cladding neared final approval.

INTRODUCTION

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

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

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

TECHNICAL PROGRESS

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

SUBTASK D-1 - ROD STRAIN INSTRUMENT

The rod strain instrument can simultaneously measure two orthogonal cladding diameters and the fuel rod bow. The instrument was equipped with low-cost strain gages, and proof testing showed that accuracy and repeatability were ± 0.005 mm with no fuel rod bowing. When rod bowing occurred, the

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

readings were offset about 20% but were easily quantified by the bow-measuring strain gage cantilever system. Work is in progress to attach high-temperature strain gages and thermocouples in preparation for installing the instrument in the test loop.

SUBTASK D-2 - LOOP EXPERIMENTS

The pressure vessel was fabricated, and hydrostatic testing was completed at the vendor site. The vessel arrived at PNL in mid-December 1981 and is being preassembled in preparation for installation in the test loop facility. Other design work to modify the loop piping system was completed, and fabrication/procurement is in progress.

Other components for specimen assembly were designed and are being fabricated. One such component is a flange-to-tube connector that will permit reuse of the machined end flanges.

The contract with the vendor for the irradiated cladding is nearly finalized.

FUTURE WORK

The following activities are planned for next quarter:

- High-temperature strain gages will be installed on the rod strain instrument, and the instrument will be readied for installation in the pressure vessel and loop systems.
- Loop piping modifications will be completed.
- The pressure vessel and instrument will be installed in the loop system.
- Two tests with unirradiated cladding will be completed.
- The irradiated cladding for future tests will be delivered to PNL.

PIPE-TO-PIPE IMPACT[a]

M.C.C. Bampton, Project Manager

J. M. Alzheimer
F. A. Simonen

SUMMARY

During the last quarter the first pipe-to-pipe impact tests were conducted. These tests were from the initial group in the matrix and were unpressurized.

INTRODUCTION

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

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

TECHNICAL PROGRESS

Progress that was made during the quarter is discussed below by topic.

PIPE-TO-PIPE IMPACT TESTS

During the past quarter the first tests in the matrix were completed. The first three tests were from Group 1 of the test matrix and were all similar except for differing impact velocities. Pipes used in the tests were from the same heat of SA106 Grade B steel; 10-ft lengths of 6-in. Schedule 40 pipe were swung into 5-ft long, rigidly backed 6-in. Schedule 40 pipe. Pipe tip velocities for the three tests were 100, 134, and 167 ft/sec, respectively. Maximum reduction in impacted pipe diameter was about 50%; no ruptures of the pipe walls occurred. For these rigidly backed tests, the fraction of the energy absorbed by the swinging pipe increased as the impact velocity increased. While there was more deformation of the target pipe as the velocity increased, the increase in the deformation of the swinging pipe was much more noticeable, which is probably related to the deformation modes of the two

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

pipes. The target pipe was deformed by crushing of the cross section, and no deformation in the beam-bending mode was allowed due to the support condition. The deformation of the swinging pipe was almost entirely in the beam-bending mode. Once a plastic hinge was formed at the point of impact, the moment-producing contact force did not have to increase very much to cause continued bending of the swinging pipe. However, a slight increase of the contact force resulted in a correspondingly small increase in deformation of the target pipe.

Although all of the instrumentation was not available for these initial tests, the important data were obtained. Support load cells and strain gages were not installed, and the target pipe pressure transducers were not used since the pipes were open ended and not pressurized. However, the major data of interest were measured: the deformation of the pipe cross sections, the local strains after impact as measured from the strain circles, and the pipe velocity. The transducers used to assess the functioning of the catapult were also installed and operated correctly. No problems were encountered in the operation of the catapult. The pipe velocities were slightly lower for a given pressure than had been predicted, but this difference can be attributed to internal friction and losses that were ignored in the prediction.

One additional test was performed to investigate the effect of an incompressible fluid in the pipe. The first tests were conducted with open-ended target pipes, but for this test the ends of the target pipe were capped. The pipe was filled as full as possible with ambient temperature water that was not initially pressurized above atmospheric pressure. As in previous tests, the target specimen was a 5-ft long section of 6-in. Schedule 40 pipe that was rigidly supported. The swinging pipe for this test was 6-in. Schedule 80 with a tip velocity of 140 ft/sec. No rupture of the pipe wall occurred. Maximum local effective strain as determined from the strain circles was 11% and occurred near the initial point of contact. Maximum internal pressure was very near that required to cause wall yielding due to hoop stress. Based upon the nominal pipe dimension and the reported yield strength, the pressure to yield the wall is 4350 psi. After the impact, the pressure dropped to between 1400 and 1500 psi. Although not validated, the strain gage data indicate strains of over 200 strains/sec in the plastic region.

Specimens for several more tests were prepared. The quality of the strain circles has been improved, and the required application time has been reduced.

Analysis of the accelerometer data has shown a much higher acceleration level and higher frequency content than anticipated. A large part of the signal can be attributed to the ringing of the pipe. These high signal levels have been overloading the conditioning equipment in the accelerometer. Another accelerometer capable of measuring the accelerations accurately is being acquired.

IMPACT TOUGHNESS CHARACTERIZATION TESTS

Two types of impact tests were run on specimens taken from the heat of 6-in. Schedule 40 SA106 Grade B pipe to evaluate the quality of pipe material and to identify the ductile-to-brittle transition temperature (DBTT) for this specific heat. Specimens for each of the test series were taken in two orientations with respect to the rolling direction: longitudinal (LT) and transverse (TL) as described in ASTM E-399.

The Charpy "V" notch tests (ASTM E-23) were performed on a series of subsized specimens. The cross sections of the specimens were 0.250 x 0.394 in. rather than the standard 0.394 x 0.394 in. The results of the Charpy tests for both orientations are illustrated in Figure 1. The LT properties were better than the TL properties, which is expected from pipe products. The LT properties showed a relatively high upper-shelf energy (~55 ft-lb) and a low DBTT of about -10°F. The TL orientation had an upper-shelf value that was less than half the LT value (23 ft-lb). In addition, the DBTT was 40°F, which is near the ambient test temperature of the catapult tests. The TL specimen simulated a crack that exists parallel to the length of the pipe.

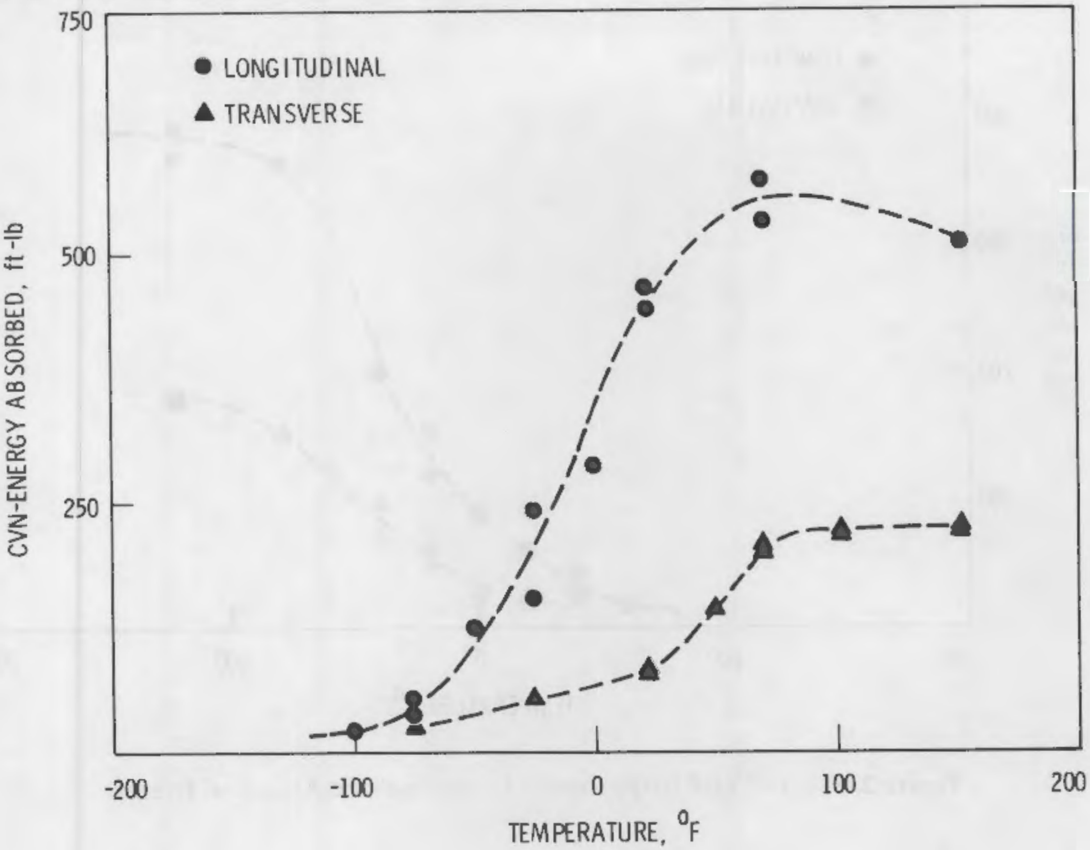

Figure 1. Charpy "V" Notch Results

Standard Charpy V notch tests are sometimes misleading in estimating the DBTT behavior because of the relatively blunt (0.010-in.) V notch and the sensitivity of steels to notch activity.

The second impact test used the same specimen as the Charpy test except that a fatigue crack was induced at the apex of the machined notch to minimize the crack initiation stage of the fracture process. Thus, the precracked Charpy test simulated the conditions under which an existing crack will begin to extend. The precracked Charpy specimens were run as a function of temperature to evaluate the DBTT of the two orientations. Because the precracked lengths were variable, the energy absorbed by the specimens was normalized by the cross-sectional area of the specimen ($\Delta E/A$). The results of the precracked Charpy tests are illustrated in Figure 2. The LT properties are better than the TL properties; the DBTT is 35°F for LT and 40°F for TL as estimated from these curves. It is also possible to evaluate the dynamic fracture toughness (KPCI) from the results of the precracked Charpy tests (see Figure 3). The toughness versus temperature results indicate a DBTT of 20°F for the LT orientation and 60°F for the TL orientation.

In summary, the precracked Charpy results are the most conservative estimate of DBTT. However, if a flaw could exist in the pipe prior to testing, it would be possible to have a brittle failure of the pipe at ambient test temperatures.

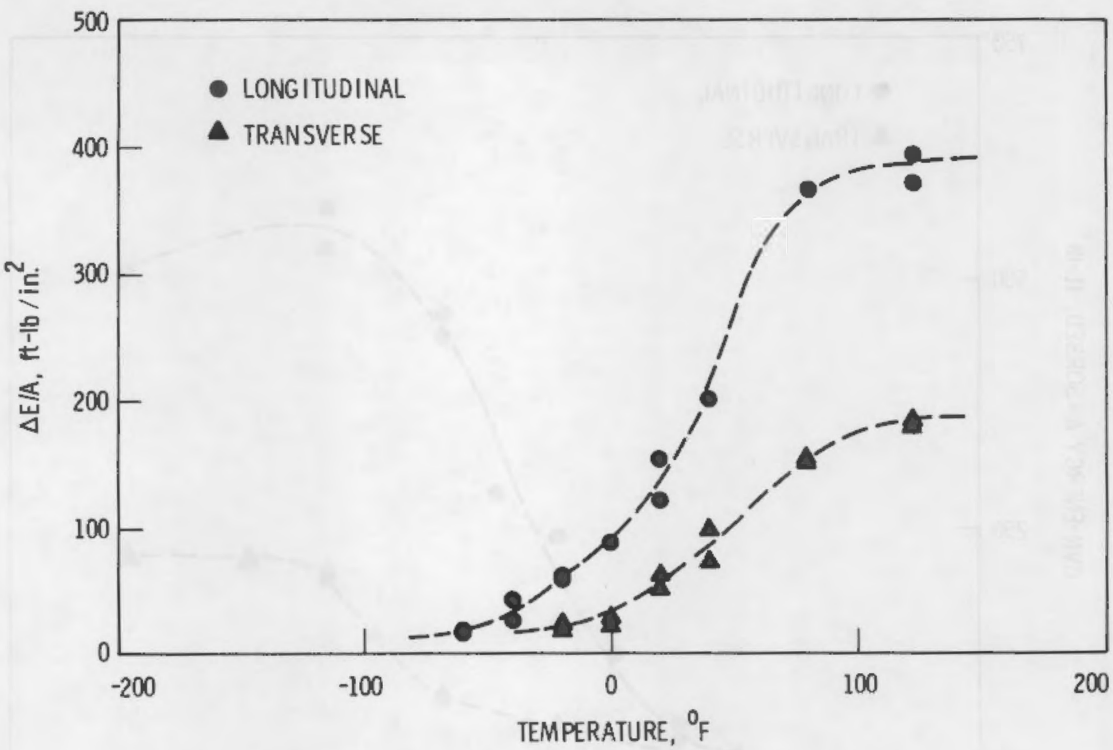

Figure 2. Precracked Charpy Results for Normalized Absorbed Energy

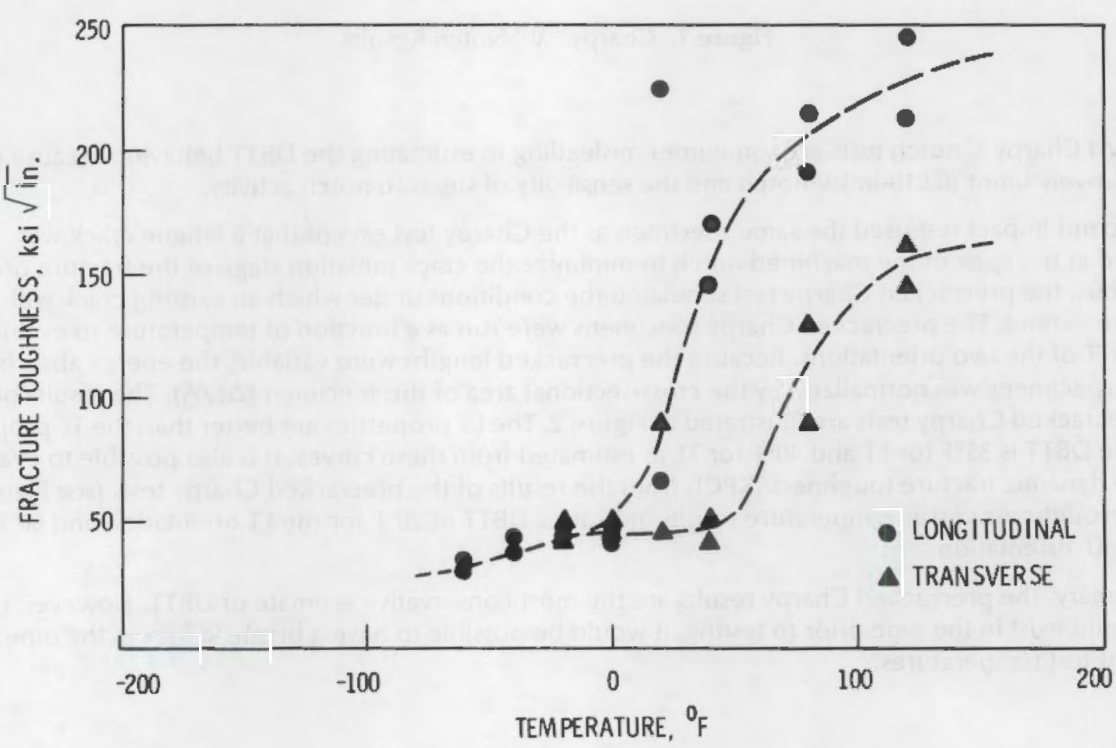

Figure 3. Precracked Charpy Results for Dynamic Fracture Toughness

FUTURE WORK

The following activities are planned for next quarter:

- continue testing and specimen preparation
- continue analysis of test data
- develop failure models.

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

POWER BURST FACILITY SEVERE FUEL DAMAGE TEST PROJECT[a]

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SUMMARY

Additional experiments were conducted to test the proposed shroud molten metal penetration device for the Power Burst Facility (PBF) severe fuel damage (SFD) test train assemblies. Testing of the composite shroud insulation for penetration by molten Zircaloy continued with the completion of a new water-cooled hearth plate. Test results indicated that the Zircaloy was cooled sufficiently by the copper hearth to solidify just above the molten metal penetration detector.

The final assembly of a prototypic 32-rod depleted-UO_2 fuel bundle in a ~0.3-m long insulated shroud mockup was completed. EG&G Idaho, Inc., will use the mockup and fuel bundle to investigate neutron radiography computer-assisted tomography (CAT) techniques that may be used to examine pre- and post-test fuel conditions in the SFD test train assemblies.

Internal design reviews were held at Pacific Northwest Laboratory (PNL), Richland, Washington, during December 1981 for the interface, outlet assembly, and general arrangement drawings for the PBF SFD-ST and SFD-1 test train assemblies. All test train design drawings except for the water fallback barrier assembly 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 these two test trains will be held in a joint PNL/EG&G meeting in early February 1982.

Five composite billets of Zircaloy-to-stainless steel were made to provide transition piece stock for both the National Research Universal (NRU) and PBF severe fuel damage programs. Eight prototypic shroud transition pieces were coextruded, machined, vacuum leak checked, and autoclave pressure leak tested. All tests were successful with no evidence of leakage or corrosion of any of the test pieces.

A Zircaloy component called a "saddle" is needed to transform the inner octagonal bundle configuration into a round outer shroud containment. Initial problems with forming were solved by preheating the blanks to 538°C. The U-shaped stock material was used for the final machining of the shroud saddle halves. All of the saddle halves necessary to complete two insulated shroud assemblies are now in-house at PNL.

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

INTRODUCTION

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

TECHNICAL PROGRESS

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

SHROUD DEVELOPMENT TESTING

Recent testing of the insulated shroud has addressed the question of how well the insulator structure would resist attack by molten Zircaloy in the event that a large portion of the cladding should melt. Testing results indicate that at temperatures around 2273K, the ZrO_2 fiberboard insulation will dissolve in the Zircaloy to an extent depending on the amount of Zircaloy present. (The ZrO_2 fiberboard is not resistant to attack by molten Zircaloy.) The oxide is dissolved fairly readily in the metal, and a large amount would be dissolved before the Zircaloy would solidify at 2300K. From the Zr-O phase diagram,[1] it can be seen that the first oxide in equilibrium with the melt at 473K would be obtained at an average composition containing about 43 at.% oxygen. As the two-phase region is reached, the metal should cease to dissolve oxide. To contain 43 at.% oxygen, each gram of Zr would dissolve approximately 0.7 g of ZrO_2 because the Zr in the ZrO_2 must be considered. On a volume basis, 1 cc of metal would dissolve about 6.7 cc of the low-density ZrO_2 fiberboard in reaching 43 at.% oxygen. Theoretically the molten metal could consume a large quantity of the ZrO_2 fiberboard insulation at 2273K.

Several tests in the PBF severe fuel damage test program will involve rapid fuel bundle heating to temperatures of about 2300K. If the heating is rapid enough, much of the cladding will liquify at its melting point (2133K) or at the Zr-UO_2 eutectic (~1700K) instead of oxidizing. This molten metal may become a hazard to test operations depending on the quantity present. A molten metal penetration detector will detect any penetration of the shroud so that corrective action can be taken. Additional experiments have been conducted to test the proposed penetration device, and testing of the composite shroud insulation for penetration by molten Zircaloy continued with the completion of a new water-cooled hearth plate.

Tests were conducted using an arc melting furnace to melt a slug of Zircaloy above a full-thickness section of shroud insulation that included the high-density ZrO_2 support tubes. Figure 1 shows the testing arrangement. The insulation was placed in a depression in a water-cooled copper hearth of a nonconsumable electrode arc melting furnace, and a pool of molten Zircaloy was maintained above the insulation. The Zircaloy temperature was estimated to approach 3273K. Three additional liquid metal penetration tests were performed in the new water-cooled hearth plate. Power supply limitations prevented testing at the power predicted for PBF in-reactor tests (~ 0.8 kW/in.^{2}); therefore, the minimum available power was used (3 kW/in.^{2}).

The molten fuel penetration detector signaled failure only during the first test; the Zircaloy penetrated the edge of the insulation because of a poor fit between it and the water-cooled copper hearth. A better fit eliminated this problem, and the molten Zircaloy was maintained above the insulation for

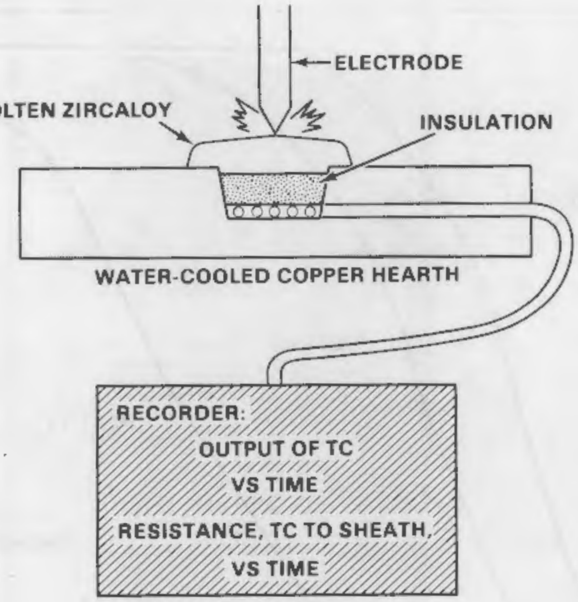


Figure 1. Shroud Insulation Penetration Testing with Molten Zircaloy

about 10 min in each of the next two tests. Since a sheathed thermocouple with its junction in the center of the insulation disc was used as a molten fuel penetration detector, its signal was monitored. During the third test, comparison of the temperature versus time for five heatups clearly showed the increase in thermal conductivity of the insulation as it was consumed by the molten Zircaloy. Results of the tests indicated that the thermocouple junction reached a very high temperature (900°C) during the testing but then apparently attained equilibrium since the temperature was absolutely constant for about 1.5 min. Temperature measurements with the molten metal penetration detector thermocouple from the simplified arc melt tests are shown in Figure 2. Apparently, the Zircaloy in heats 2 and 4 was cooled sufficiently by the copper hearth to solidify just above the molten metal penetration detector.

A short (~10.2-cm) full-section insulated shroud is being built for the final and most prototypic liquid metal penetration test (see Figure 3). After final assembly, a pool of Zircaloy approximately 6.4 cm deep that contains some UO_2 will be kept molten for a target time of 1 hr or until the molten metal penetration detector signals failure.

PROTOTYPE SHROUD AND BUNDLE

The final assembly of a prototypic 32-rod depleted-UO_2 fuel bundle in a 0.3-m long insulated shroud mockup was completed. A typical fuel bundle grid spacer, the inner octagonal Zircaloy liner, and the two outer round Zircaloy liners are shown in Figure 4. The completed prototypic 32-rod fuel bundle and insulated shroud mockup are shown in Figure 5. EG&G Idaho, Inc., will use the insulated shroud mockup and fuel bundle to investigate neutron radiography CAT scanning techniques that may be used to examine the pre- and post-test conditions of the fuel in the PBF severe fuel damage test train assemblies.

TEST TRAIN HARDWARE COMPONENTS

Five Zircaloy-to-stainless steel composite billets were fabricated to provide transition piece stock for both the NRU and PBF severe fuel damage programs. These billets were then coextruded into

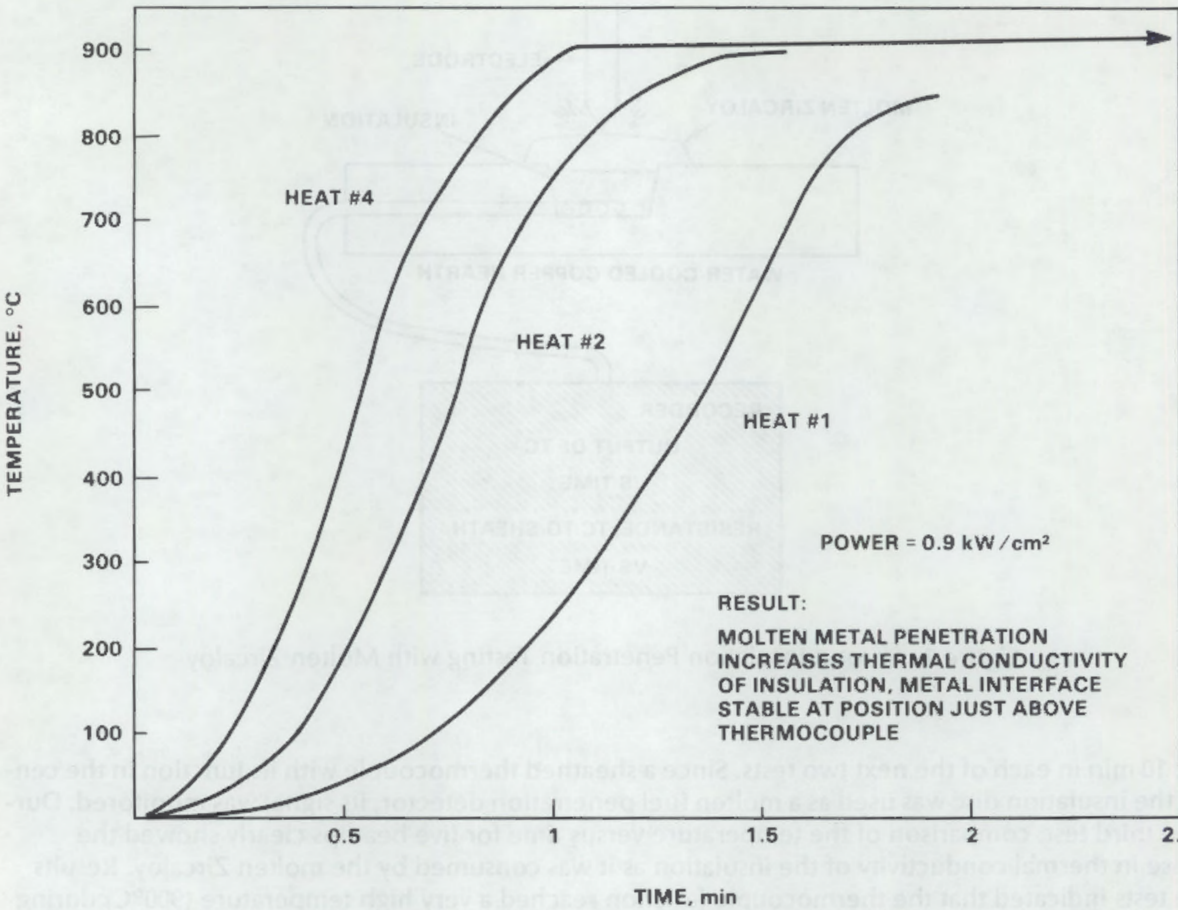


Figure 2. Thermocouple Response in Simplified Arc Melt Tests

transition stock. One extrusion was used for the NRU fuel rod Zircaloy-to-stainless transition pieces for thermal-hydraulic test bundle TH-2 that was recently assembled at PNL. The remaining four pieces of transition stock will provide Zircaloy-to-stainless transition pieces for the thermocouple penetrations on the upper Zircaloy flange of the insulated shroud for the PBF severe fuel damage test train assemblies. Dimensional checks, metallography, and die penetrant tests of the transition stock material were completed and all were acceptable.

Eight prototypic Zircaloy-to-stainless shroud transition pieces were machined from PNL extruded stock and subjected to vacuum leak check and autoclave pressure leak testing. Autoclave testing consisted of five 8-hr cycles at 545°F and 1000-psi pressure. The tests were successful with no evidence of leakage or corrosion on any of the test pieces. Elevated temperature testing with internal pressure is yet to be completed on the eight prototype transition pieces.

As part of the development activities for the insulated shroud, Zircaloy plate was hot rolled from a 1.91 1.91 cm to a 1.59 cm thickness and cut into 25.4-cm blanks for bending into U-shaped stock material for the shroud saddle halves. The shroud saddles are the parts of the insulated shroud assemblies that are needed to transform the inner octagonal bundle into a round outer shroud containment. Initial bending attempts at room temperature resulted in crushed pieces of Zircaloy; however, the bends were successfully made by preheating the blanks to 811K. The bending sequence is shown in Figure 6. This material was used for the final machining of the shroud saddle halves by an offsite vendor. Completed saddle halves for two complete insulated shroud assemblies are now at PNL.

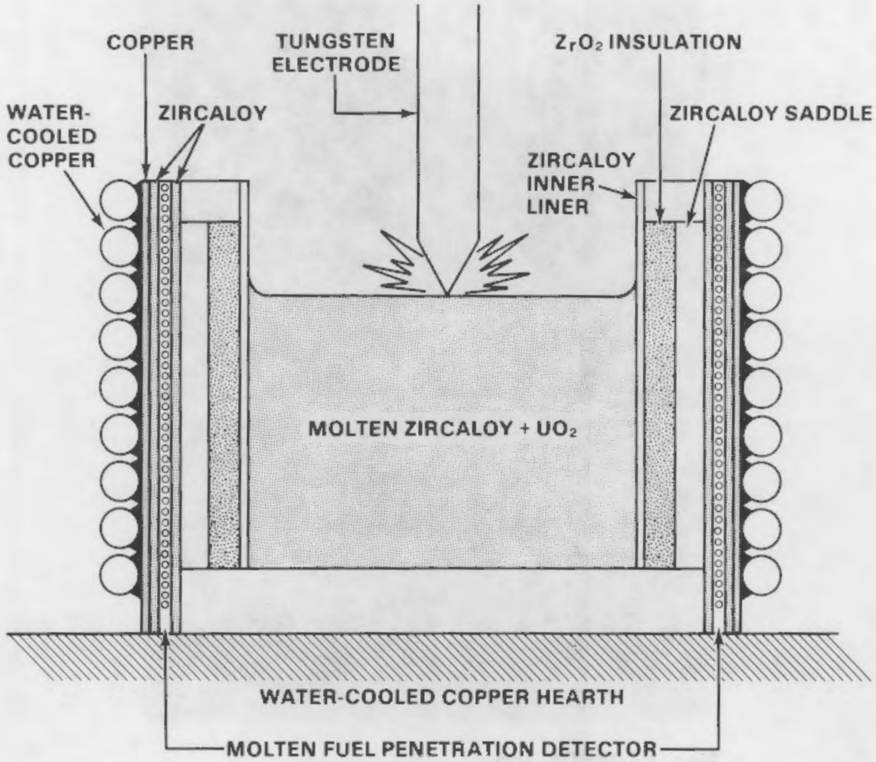
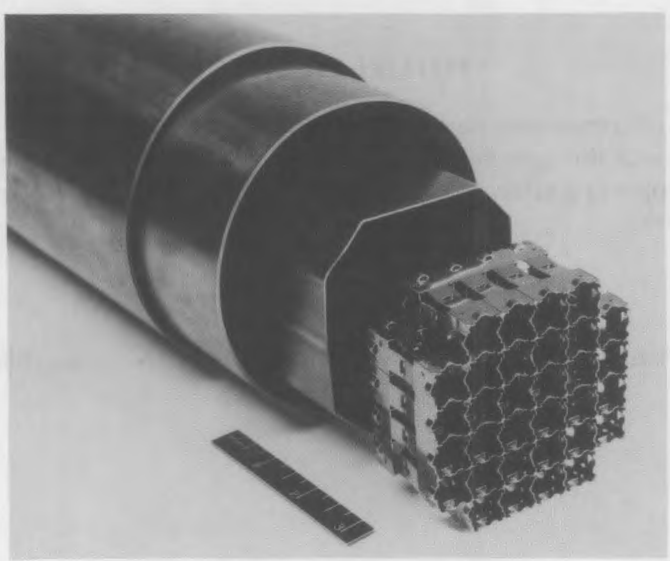


Figure 3. Full-Section Shroud Molten Metal Penetration Test (sectional view)



Neg. 8106129-2

Figure 4. Insulated Shroud Mockup Component Parts

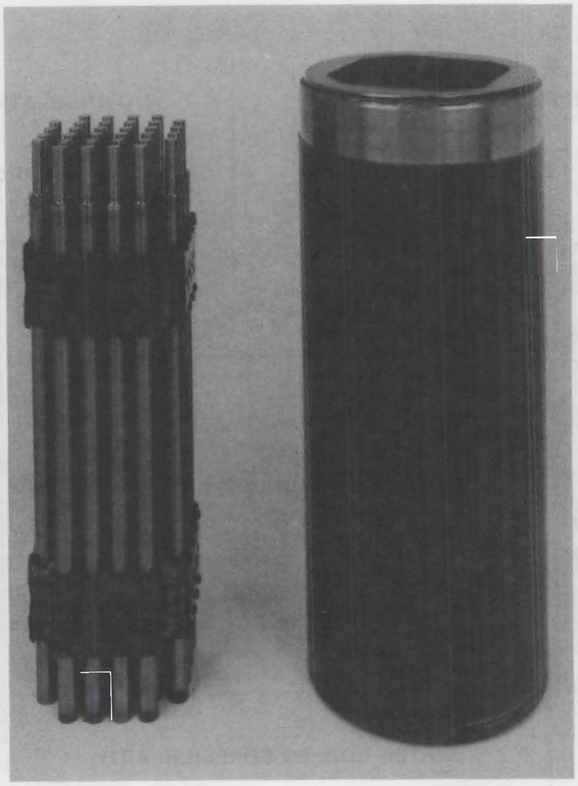


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Figure 5. Prototypic 32-Rod Depleted-UO_2 Fuel Bundle and Insulated Shroud Mockup

FUTURE WORK

Fabrication and assembly of component parts for the SFD-ST and SFD-1 test train assemblies will continue. A final design review of the completed design drawings for the test train asembly will be held with EG&G Idaho, Inc. Supporting safety-related analysis and documentation required for the SFD-ST and SFD-1 tests will continue.

REFERENCES

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

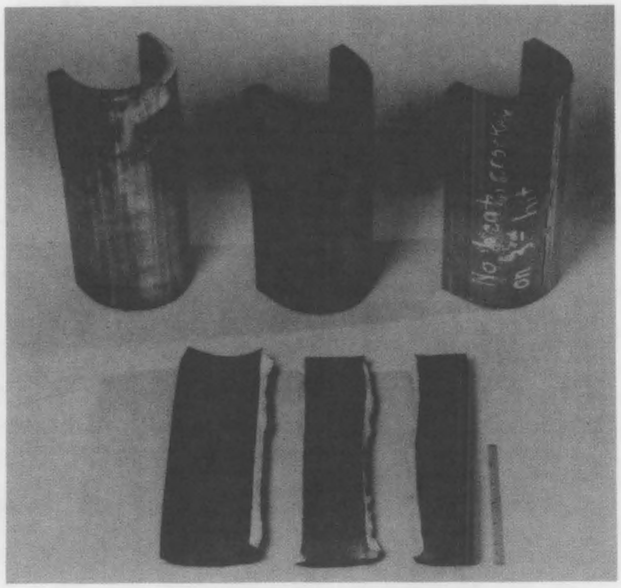

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Figure 6. Development Sequence for Zircaloy U-Shaped Stock Material for Insulated Shroud Saddle Halves

SEVERE CORE DAMAGE TEST SUBASSEMBLY PROCUREMENT PROGRAM

ESSOR FUEL DAMAGE TEST PROGRAM SUPPORT[a]

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SUMMARY

The major area of activity during the past quarter centered on the preparation for and attendance at a Joint Research Centre (JRC), Ispra-U.S. support team meeting held October 21 through 27, 1981, at Ispra, Italy. Highlights of that meeting are reported below.

INTRODUCTION

The Super Sara Test Program (SSTP) is a major European Community effort to study reactor safety during rapid or large-break and slow or small-break loss-of-coolant accidents (LOCAs). The program will use the SUPER SARA high-temperature, high-pressure loop in the ESSOR reactor at Ispra, Italy. The SSTP is designed to yield important experimental data on fuel rod deformation and postaccident coolability of damaged fuel assemblies after they experience a loss of coolant. The complete testing program currently includes loop construction and 21 in-pile experiments to simulate 7 large- and 14 small-break conditions in commercial pressurized water reactors (PWRs) and boiling water reactors (BWRs).

TECHNICAL PROGRESS

JRC, ISPRA-U.S. SUPPORT TEAM MEETING

Six U.S. engineers visited the JRC at Ispra, Italy, October 21 through 27, 1981, as part of the U.S. Nuclear Regulatory Commission (NRC) support to the SSTP. The main objective of the visit was to identify areas of safety analysis where NRC would support the SSTP licensing effort.

Seven items were identified where the United States could provide information for input to the Progetto di Massima (PM)—the initial licensing document. The seven items included: 1) molten material behavior, 2) bundle grid support, 3) steam explosions, 4) high-temperature Zircaloy oxidation kinetics

(a) FIN: B2372-1; NRC Contact: R. Van Houten.
(b) EG&G Idaho, Inc., Idaho Falls, Idaho.

in steam, 5) hydrogen production rates, 6) hydrogen explosion effects, and 7) temperature measurements to satisfy safety needs. Pacific Northwest Laboratory (PNL) responded to these items in November 1981; these responses are included as an appendix to this contribution.

JRC staff stated that the SSTP licensing would be augmented if the Zr-2.5% Nb in-pile pressure tube could become an ASME Nuclear Pressure Vessel Code Material (Section III, Class 1). An agreement was made to check on the feasibility of getting the material approved for use in Section III.

The following areas of support were also identified:

- supply information on Power Burst Facility (PBF)[a] SFD-ST and SFD-1 tests, including a short visit to Ispra by a key test engineer
- assess current Super Sara loop design for off gas and other waste treatment capabilities
- provide SFD-ST TRAC calculations
- provide rubble bed properties, geometry, and expected behavior
- provide stress analysis for the Super Sara in-pile tube:
  a) for combined and dynamic loads, especially high-energy or explosive loading
  b) to analyze tube stress for a partially oxidized tube of zirconium alloy
  c) to determine what calculational tools should or could be used and what is their availability
- provide detailed drawings of PBF control and transient rod systems
- provide details on the PBF test train
- provide an assessment of the JRC SFD test train design and work to achieve **one** consensus design
- provide details on PBF fission product detection system to be used with SFD tests
- assess the utility of a gamma densitometer as a substitute for neutron tomography including software.

The reference low- and intermediate-temperature SFD tests could encounter operational difficulties with liquid level control. An alternate approach would be to run the tests without a liquid level in the bundle and supply the bundle at the inlet region with controllable steam. This arrangement would maximize the length of the bundle at the desired test temperatures. The type C tests would be run using a liquid level in the bundle region.

Three SSTP pressure tubes and two safety tubes are on order from Atomic Energy of Canada Limited (AECL).

It is expected that Italian licensing approval will be almost on a test-by-test basis rather than blanket approval for all the tests. Project schedules will be generated after the beginning of 1982 when eight working groups have completed their planning.

U.S. support team members provided guidance to the JRC staff in many areas; key areas where one-to-one discussions were held included:

- loop design fabrication and operation
- loop waste system treatment
- PBF safety analysis
- PBF safety experience
- PNL reactor safety experience.

(a) Idaho Falls, Idaho.

Excellent presentations that generated much interest were made by Dick McCardell and Steve Atkinson of EG&G Idaho, Inc. Dick's topic covered safety analysis for testing in PBF; he discussed the PBF safety envelope, the new mode of PBF operation, analysis for SFD testing, and a guide for preconditioning. Steve presented a summary of PBF safety analyses including safety limits and goals, analysis approach, and SFD analysis status.

APPENDIX

RESPONSES TO SAFETY CONCERNS

A brief analysis was telexed to JRC, Ispra in response to seven safety items of immediate interest. These items were requested by the Ispra staff for the SSTP Progetto di Massima. The item and the response provided are presented below:

ITEM: Behavior of support grid during thermal transients due to irradiation, molten fuel contact, or cold water shock; choice of grid materials and diverse supporting options.

RESPONSE: Initial SFD tests currently planned for the PBF all use a liquid level above the bottom of the test train. Thus, the support grid immersed in water is expected to see no harmful environment. If the liquid level should accidentally fall below the bottom of the support grid, emergency measures such as test termination, reactor scram, and added emergency water would occur.

If the support grid is not protected by water (like that being considered in SSTP), then suitable materials (refractory metals, alloys, and ceramics) used in an optimum design and tested out-of-pile will be incorporated into the test assemblies. It should be noted that since the support grid is out of the active core region grid material with low neutron cross section will not be required.

If suitable materials cannot be found to withstand expected conditions (including molten Zircaloy contact), the test and/or test bundle will be designed to protect the support grid. Analysis will also be performed to show that loss of a portion or all of the support grid will not produce an unsafe condition.

ITEM: Water/Zircaloy oxidation reaction - kinetics.

RESPONSE: Calculations using currently available Zircaloy/steam oxidation data(1) for a SFD test with limited radial and axial in-pile heat transfer will yield very high (runaway) temperatures. These calculations do not account for any inhibiting effects at high temperature and high reaction rates. Out-of-pile tests such as those performed by Dr. S. Hagen at KFK show that the runaway is self-limiting to acceptable temperatures. One of the main inhibitors is the rapid downward movement of liquid Zircaloy away from the high-temperature region where it cools and solidifies; its movement stops the high-temperature ramp.

The Hagen experiments showed that the peak temperature reached due to the exothermic oxidation reaction can be controlled by varying the amount of low-temperature (below approximately 1200°C) Zircaloy oxidation; the greater the amount of Zircaloy oxidized below 1200°C, the lower the peak temperature during the rapid oxidation above approximately 1500°C. Of course, the rapid temperature increase stops once all the Zircaloy has oxidized. Since the initial SFD tests in the SSTP will follow a slow heatup rate, all the Zircaloy will be oxidized before its melting point is reached. These early tests as well as those run in the PBF will demonstrate the controllability and safety of subsequent faster heat rate tests. It is expected that each subsequent experiment will be shown to be safe based on the results of previous experiments.

Another possible rate-inhibiting occurrence shown by work at Argonne National Laboratory (ANL) for the Nuclear Safety Analysis Center (NSAC) and the Electric Power Research Institute (EPRI) was that when the mole fraction of hydrogen in steam extends above 40%, the Zircaloy oxidation rate may decrease significantly.

ITEM: Hydrogen production rates.

RESPONSE: The actual hydrogen production rate in and of itself is not a safety problem because it will never be excessive; however, the total amount of hydrogen produced, its distribution (mainly outside the loop), and its concentration do require safety analysis (see the hydrogen explosion effects section). Some safety credit may be taken in retarding the Zircaloy-steam reaction rates inside the test section at very high temperature due to locally high hydrogen concentrations resulting from large hydrogen production rates associated with Zircaloy oxidation.

ITEM: Hydrogen explosive effects (conditions).

RESPONSE: The effects (mechanical stress and strain) of hydrogen explosions are not addressed because it is possible that the conditions for such an explosion can be 1) defined and 2) avoided by design. The reactor-related conditions describing hydrogen behavior are summarized in two U.S. reports: one by NSAC-EPRI and the other by NRC/Sandia National Laboratories. These reports provide conditions that can create hydrogen safety problems. The SSTP design and analysis will show how such problem areas are expected to be avoided.

ITEM: Steam explosions.

RESPONSE: A steam explosion is a type of vapor explosion that involves water flashing to steam. As a result of potential liquid metal fast breeder reactor (LMFBR) safety problems, much work was performed on vapor explosions (including some steam explosions). After the Three Mile Island (TMI)-2 accident, nuclear safety research is now concentrating on steam explosions and little work is being performed by the U.S. nuclear industry on other types of vapor explosions. An excellent summary of nuclear-related vapor explosive phenomena is given in Reference 2. The five stages of the vapor explosion are:

- initial stable film boiling so that vapor film separates the two liquids and permits coarse premixing without excessive energy transfer
- breakdown of film boiling due to pressure or thermal effects
- fuel-coolant contact upon breakdown of film
- rapid vapor production, causing shock-pressurization due to 1) spontaneous vapor bubble nucleation and fine-scale fragmentation and intermixing or 2) a large effective heat transfer surface as a result of fine-scale fragmentation and intermixing
- adequate physical and inertial constraints to sustain a shock wave.

Current research in this area is being done in the United States at Sandia Laboratories.

Safety analyses of the SSTP will initially determine the boundary conditions using material-stored energies. An assessment can then be made of the possibility of vapor explosions of varying degrees of thermal-to-mechanical work conversion efficiency. Based on the results of such analyses, appropriate safety systems and test plans can be incorporated to satisfy the ultimate safety objectives.

ITEM: Temperature measurements for safety-trip circuitry development.

RESPONSE: The design and development of safety limits and reactor trip set points for experiments like those conducted in NRU by PNL that simulate a LOCA are dependent on safety analysis predictions. The reliability of these limits is indicated by the predictive capability of the analytic model to represent similar test operating conditions. Ideally, that capability is demonstrated on initial, benign operations using the same reactor and test train assembly; however, predictions for similar experiments and facilities will provide the qualification for initial safety assessment.

LOCA safety analyses are performed with simplified, yet mechanistic computer codes like COBRA to evaluate the fuel rod and structure temperatures during experiments that produce fuel rod failure and fuel/structure debris. The peak fuel rod temperature and structure temperature histories are used to relate the expected fuel rod failure with resultant (probably worst-case) effects. Throttled coolant flow represents the dilated fuel cladding effects (or in extreme cases, the fuel debris bed blockage), and that throttled coolant flow results in peak structure temperature history predictions. The most important information to be gained is the time-temperature relationships between the test fuel rods and the critical structures modeled in the safety analysis. These comparable temperature histories of the test assembly fuel and the test assembly structures will permit one or more to be instrumented, while the temperatures of another can be reasonably deduced and/or monitored to assure that the experiment remains within safe operating conditions.

Structural temperature (and strength) limits will be identified as the basic limitation to Super Sara experiment operating conditions. The pressure tube is the primary pressure boundary of the test assembly; hence, a temperature limit profile of the tube is required to identify its pressure (and time) design limits. Because of the extended duration of the Super Sara experiments, temperature limits on the pressure tube take creep rupture into account. Existing available Zr 2.5% Nb materials property data may be used to provide initial safety assessment information; however, the Super Sara design must provide the final temperature limit(s) for the pressure tube.

The final design of the test train assembly to be used in Super Sara experiments will establish the instrument sensor types, their locations, and the routing of their instrument leads. Those instruments identified as safety-trip circuit sensors will be designed to survive the experiment with high confidence. Instrument technology presently offers the capability of locating thermocouples in either fuel rods or structural members. Sensor survivability is much greater in the structural member, where instrument lead protection can be guaranteed in the outer coolant channels. This location also provides a more comparable monitor site for measured temperatures, which serve as the basis for deduced pressure tube temperatures and comparison with their limits. Safety analysis will also provide the basis for selecting sensor axial elevations to monitor the peak temperature and temperature profile of the test assembly structure.

Reliability is designed into the instrumentation that provides the safety-trip circuitry. The final test train design will incorporate redundant thermocouples at several isothermal locations rather than using single sensors. The signals from redundant thermocouples in each of these isothermal regions or elevations will be averaged so that individual sensor failures can be discarded with signal logic, thereby maintaining reliable safe-trip circuit operation throughout the experiment.

To complete the development of safety-trip circuits for the Super Sara experiments, the time delay and temperature differences between measured data and pressure tube limits will be accounted for. The circuit time evaluation will include the difference between event timing (like temperature peaking) for measured and deduced components, the sensor response time constant, the data-averaging time requirement, and the trip circuit reaction time. Analyzed temperature differences, offsets, and a safety margin will also be used to establish the trip set points for each safety-trip circuit. These temperatures and their time histories will be used to establish the conservative trip set point temperature for each measuring circuit that will insure that the pressure tube temperature cannot reach within the safety margin of its design temperature limit.

Steve Atkinson of EG&G also provided analysis on questions related to molten experimental fuel bundle behavior in the sodium loop safety facility and the PBF, oxidation reaction modeling, steam explosion, and temperature measurements for safety trips.

FUTURE WORK

Because of redirection from NRC there will be a major shift in the scope of the program. The support to JRC, Ispra is expected to be reduced except for the support that is currently provided by the NRC site representative at ESSOR. PNL will also assist in the SSTP licensing efforts as funding permits.

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1. Urbanic, V. F., and T. R. Heidrick. 1978. *J. Nucl. Matl.* 75:251-261.

2. Cronenberg, A. W., and R. Benz. 1980. "Vapor Explosion Phenomena with Respect to Nuclear Reactor Safety Assessment." In *Advances in Nuclear Science and Technology, Vol. 12,* Plenum Press, New York.

CORE THERMAL MODEL DEVELOPMENT[a]

M. J. Thurgood, Project Manager

T. E. Guidotti

SUMMARY

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

INTRODUCTION

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

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

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

The code also features flexible noding, which allows modeling of the complex geometries encountered in reactor vessel internals, such as slotted control rod guide tubes, jet pumps, and core bypass 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) FLECHT = full-length emergency cooling heat transfer.
(c) FEBA = flooding experiment in blocked arrays.

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.

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

TECHNICAL PROGRESS

A COBRA/TRAC calculation of a 200% cold-leg break LOCA in a PWR equipped with UHI was performed to exercise the code using a full-scale, four-loop system model with a 3-D vessel. This COBRA/TRAC simulation, which is the largest performed to date, serves as the shakedown calculation for future large-scale simulations. The results should be considered as preliminary because the calculation began before the data assessment of COBRA/TRAC was completed.

SYSTEM DESCRIPTION

The UHI system is an additional emergency core cooling (ECC) system designed by Westinghouse Electric Corporation to deliver subcooled liquid from a passive high-pressure accumulator to the upper head. The vessel has four injection ports in the upper head, hollow support columns and guide tubes that provide hydraulic flow paths from the upper head to the core outlet, and upper head cooling jets that connect the downcomer to the upper head. The injection ports discharge the UHI liquid from an 1800-ft^3 accumulator filled with borated water. Check valves on the delivery lines open when the pressure in the upper head falls below the set point pressure of 1250 psia, and isolation valves close on a low accumulator water level signal after 1000 ft^3 of water is delivered to the upper head. Flow injected into the upper head exits through the guide tubes and support columns, which have the same hydraulic resistance, and is delivered to the top of the fuel assemblies. There is one assembly beneath each guide tube or support column so that a uniform flow distribution can be delivered to the core outlet plenum. The remainder of the primary system is similar to other non-UHI Westinghouse designs with the exception of the cold-leg accumulator set point pressure, which has been reduced from 600 to 400 psia. For a more detailed description of the UHI system, see Reference 1.

Figure 1 illustrates the COBRA/TRAC vessel nodalization. There are 548 hydrodynamic cells to model the fluid volume; 249 heat slabs to represent the stored energy of the vessel internals; and 10 rods to model the conduction heat transfer in the nuclear fuel.

CALCULATIONAL RESULTS AND DISCUSSION

Five periods of core cooling were observed in the calculation as shown by the cladding temperature response in Figure 2. The first cooling period occurred at 6 sec when flashing in the lower plenum led to a liquid level swell that entrained water into the core. This event resulted in a period of effective cooling, lowering the cladding temperatures for the first time. After the drops evaporated, the temperatures began to rise until the first UHI water delivery at 6 to 12 sec lowered the cladding temperatures for a second time. The core again dried out until a second UHI delivery occurred from 19 to 27 sec. However, it did not provide enough heat transfer to decrease the cladding temperatures. The upper head began to drain at 32 sec, which provided moderate cooling-leveling of the midplane cladding temperature until the beginning of bottom reflood at 58 sec. During bottom reflood, the cladding temperatures turned around as cooling continued to decrease rod temperatures until the end of the calculation.

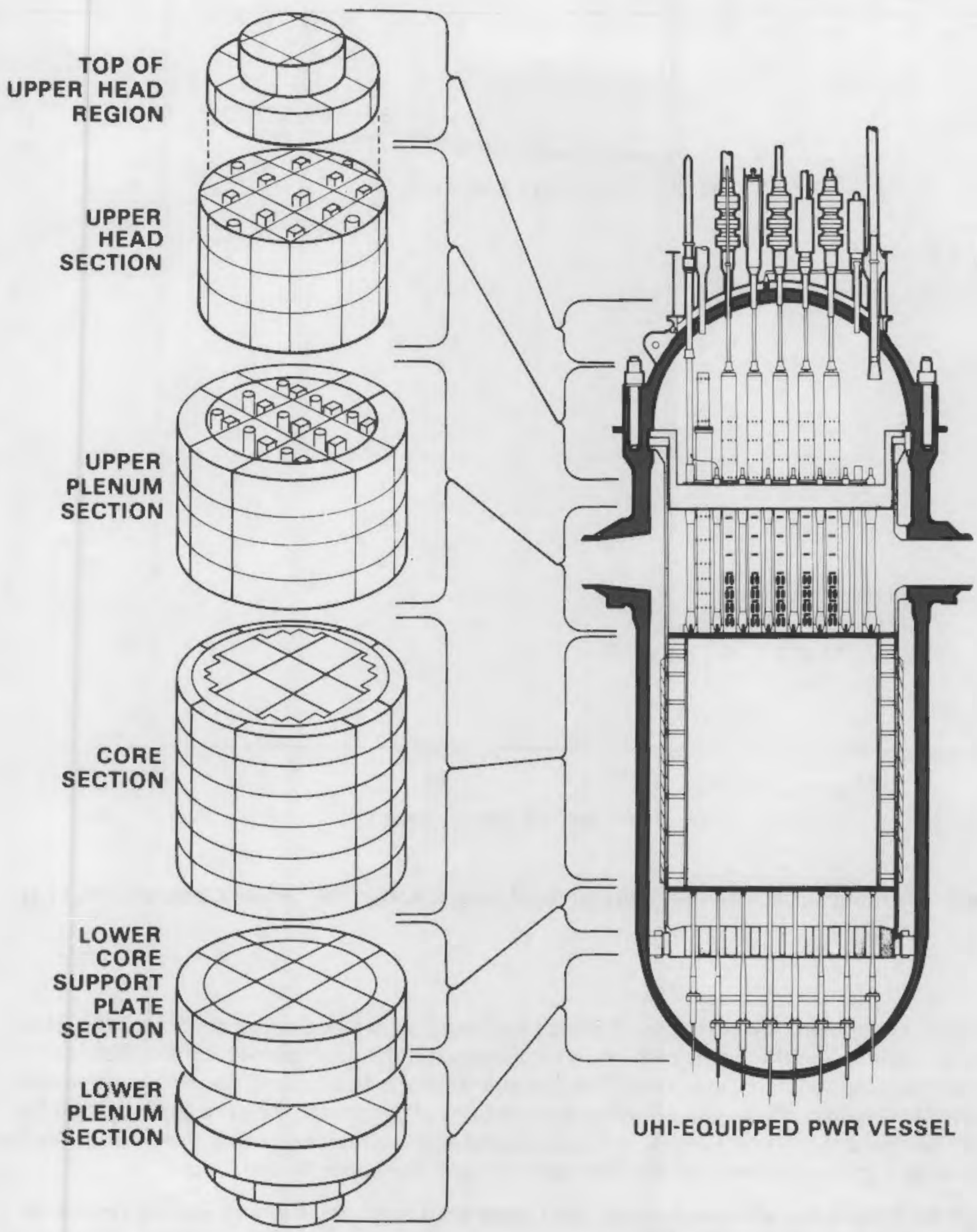

Figure 1. COBRA/TRAC UHI-Equipped Pressurized Water Reactor Vessel and Nodalization

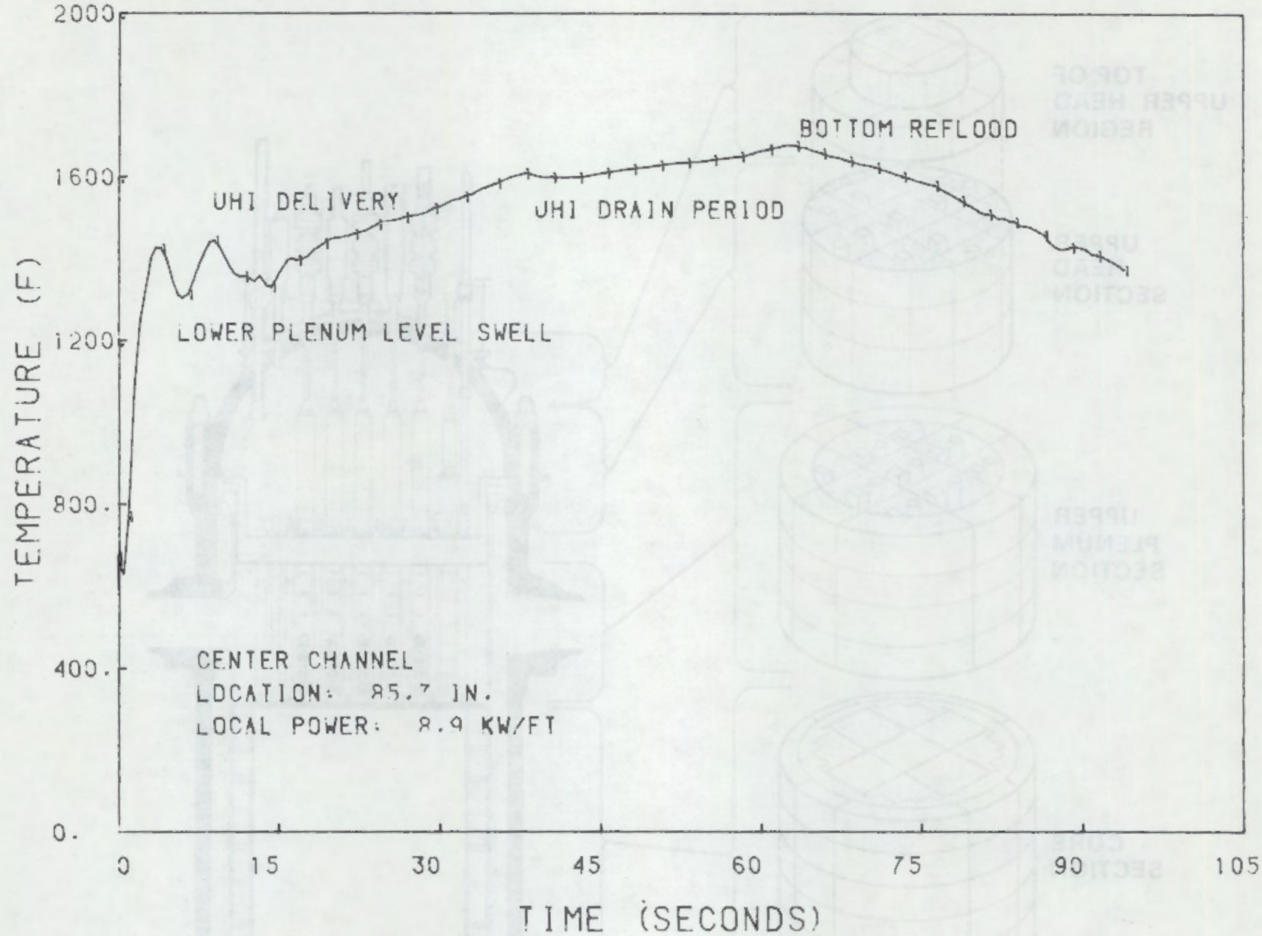

Figure 2. Cladding Surface Temperature of the Average Rod in the Center Channel at the 7-ft Elevation

After subcooled blowdown, the core flow direction became positive beginning at about 2 sec. This was the result of flashing in the lower plenum, which increased the core inlet-to-broken hot-leg pressure difference and led to large vapor velocities through the core. In addition, flashing in the lower plenum raised the liquid froth height, allowing entrainment of drops into the core as evidenced by the core void fractions in Figures 3 and 4. The cooling that was maintained during this period was the result of the large vapor velocities and the desuperheating of the vapor by the drops.

Concurrently with the lower plenum flashing, the upper head exhibited flashing and condensation behavior. At 5 sec, water was flashing to steam in the top of the upper head, forcing flow down the guide tubes and support columns. The UHI accumulator began injecting subcooled (80°F) water at 1.9 sec, which mixed with saturated water in the lower region of the upper head. By 10 sec, the liquid level was still dropping and was just above the UHI ports. The temperature in the lower portion of the upper head was decreasing and liquid was continuing to flow down the guide tubes and support columns. At 11.7 sec, the liquid level had reached the elevation of the UHI ports and allowed the jet to carry steam into the subcooled liquid, which initiated condensation and caused the pressure in the upper head to decrease. As the pressure decreased, the flow in the guide tubes and support columns

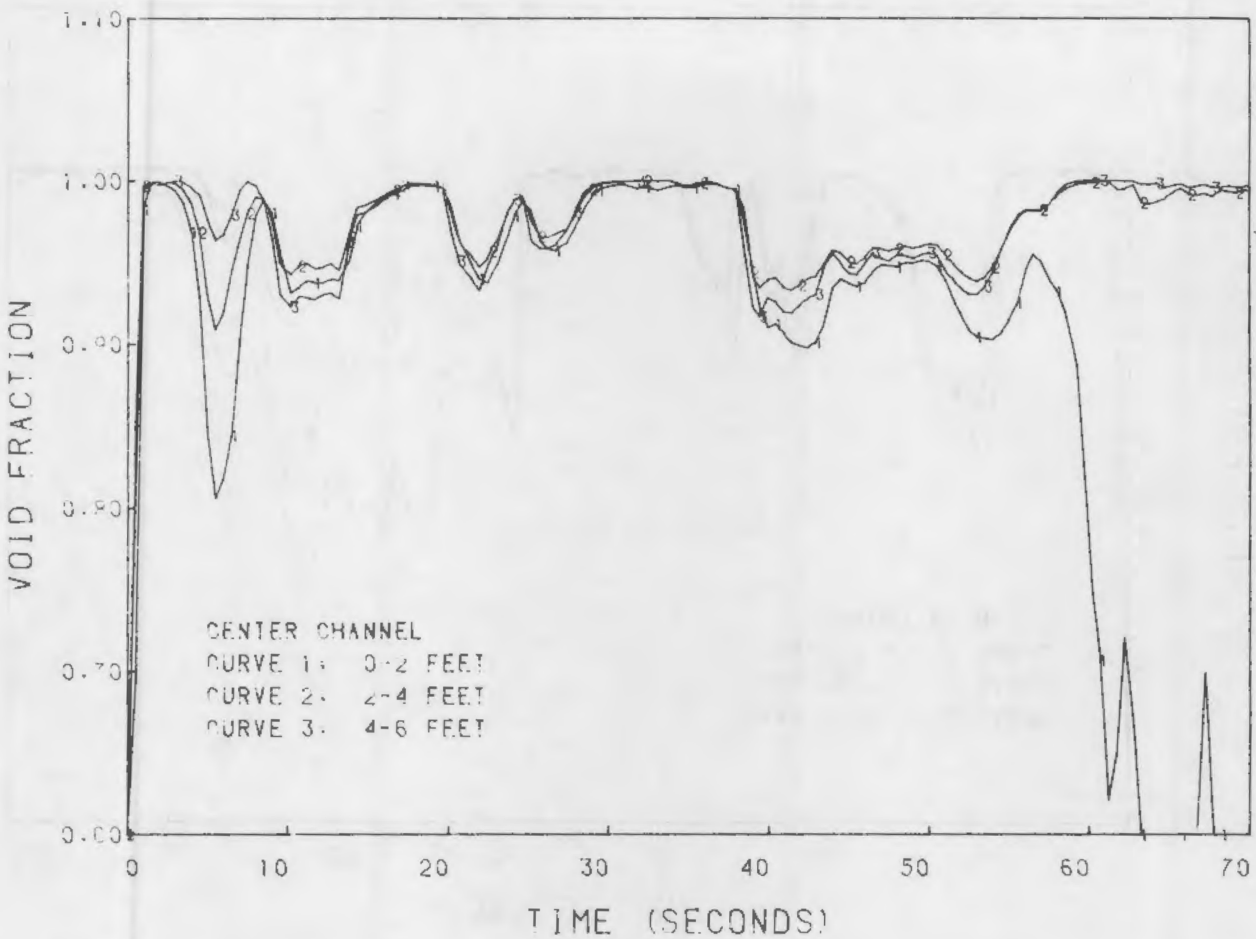

Figure 3. Core Void Fractions Versus Time in the Bottom Three Hydrodynamic Cells

began flowing upward at 13.3 sec and provided another source of steam and increased the condensation rate until the upper head was completely refilled with liquid by 16 sec.

The flashing in the upper head forced liquid down the guide tubes and support columns and initiated the first of three UHI water delivery periods. Liquid began accumulating above the core at 3 sec; however, large steam velocities generated during lower plenum flashing prevented liquid from penetrating the upper tie plate nozzle. This CCFL continued until 8 sec when a decreased vapor flow rate allowed the liquid to penetrate and flow into the core. The liquid quenched the top of the core; and some of it evaporated, creating large downward vapor velocities, while the remainder fell as liquid chunks and drops as shown by the void fractions in Figures 3 and 4 during the 10 to 15 sec period.

It is known from recent data assessment calculations of the Westinghouse G-2 facility that the dispersed flow heat transfer during top reflood was underpredicted. Since the drop-to-wall contact heat transfer correlation was inadequate, the heat transfer that occurred during UHI delivery was primarily due to forced convection of steam. The drops desuperheated the steam, which coupled with the large vapor velocities provided enough steam-forced convection heat transfer to turn rod temperatures around as shown in Figure 2. Had the correct amount of drop-to-wall heat transfer been present, additional core cooling would have occurred. Currently, a new drop-wall heat transfer model is being

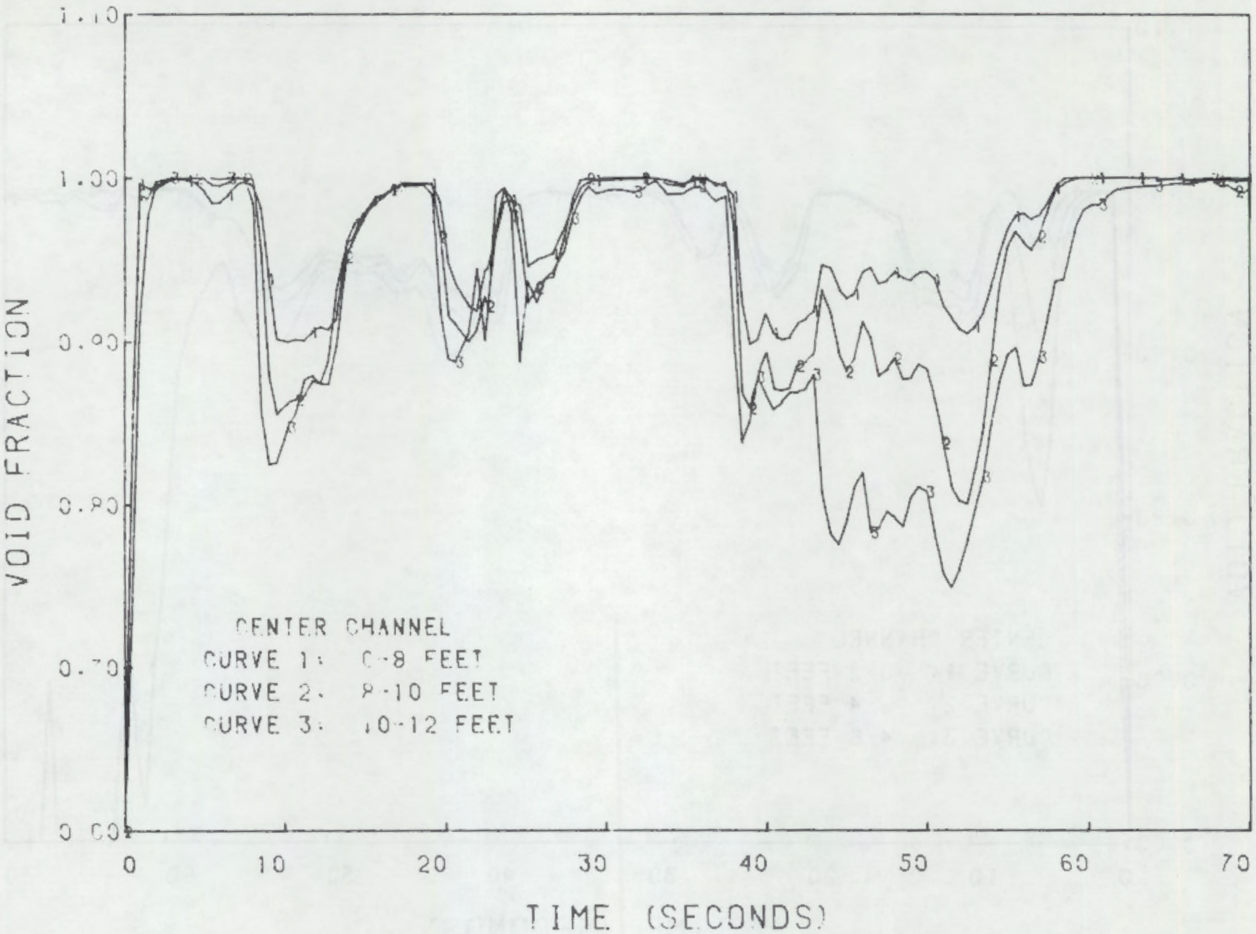

Figure 4. Core Void Fractions Versus Time in the Top Three Hydrodynamic Cells

assessed using the G-2 data; the new model will be used in future UHI simulations. This UHI delivery period ended at 15 sec when condensation in the upper head forced flow up the guide tubes and support columns.

By 18 sec, the period of upper head condensation ended and the second UHI water delivery began as continued UHI accumulator injection forced liquid down the support columns. At the same time, vapor was flowing up the guide tubes to condense on subcooled liquid inside the top of the guide tubes. Upon reaching the core, the UHI water, which was not impeded by CCFL, immediately penetrated the upper tie plate nozzle, generating vapor and a stagnation point. The stagnation point moved down through the core, and as it did a transition to partial and then complete CCFL occurred at the upper tie plate. This UHI delivery period had a low cooling effect because of the low vapor velocities associated with the stagnation point and the underprediction of the drop-to-wall heat transfer. At 21.8 sec the UHI accumulator shut off, decreasing the liquid flow down the support columns.

Following the second delivery period at 27 sec, condensation in the guide tubes increased, which lowered the pressure in the upper head enough to stop the delivery of liquid down the support columns. The condensation continued until depressurization and condensation heating gradually increased the temperature in the upper head to saturation by 35 sec. Upon reaching saturation, the upper head pressure increased, allowing the liquid in the upper head to drain down the support columns while vapor flowed up the guide tubes. This drain period began at 32 sec and ended by 56 sec; it delivered a large amount of liquid to the core (Figures 3 and 4) but provided only moderate cooling.

Bottom reflood began at 58 sec, causing the vapor generation rate in the core to increase and establish CCFL at the upper tie plate. The high vapor velocities provided enough forced convection to steam that the cladding temperatures began to decrease and continued to do so until the calculation was stopped at 94 sec. Table 1 summarizes the calculation.

Table 1. Summary of COBRA/TRAC Calculation

Event	Time, sec
End of subcooled blowdown	<0.5
UHI accumulator "On"	1.9
Upper head begins to flash	2.0
Lower plenum level swell	6.0
First UHI water delivery	6 to 12
Upper head condensation begins	14.0
Cold-leg accumulators "On"	14.3
Second UHI delivery period	19 to 27
UHI accumulators "Off"	21.8
Complete ECC bypass	15 to 20
Partial ECC bypass	20 to 36
Third UHI delivery (drain) period	32 to 56
Beginning of bottom reflood	58.0
Calculation terminated	94.0

CONCLUSIONS

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

Three problem areas emerged in the calculation. First, the dispersed flow heat transfer during UHI delivery was underpredicted because the drop-to-wall contact heat transfer was in error. Second, the initial stored energy of the rods was too high as a result of low gap conductance values. Third, the initial enthalpy in the top of the upper head was too warm, causing the upper head to flash early. Because of these anomalies, the results of this calculation should be used judiciously. A future simulation will incorporate desirable noding improvements learned from this calculation and changes to eliminate the problem areas.

FUTURE WORK

A best-estimate PWR/UHI calculation will be performed, and assessment of the containment version of COBRA-TF will continue. The hot bundle code will be assembled and assessment will begin.

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1. Young, M. Y., et al. 1976. *Westinghouse Emergency Core Cooling System Evaluation Model Application to Plants Equipped with Upper Head Injection.* WCAP 8479-P, Vol. 1, Westinghouse Electric Corporation (proprietary), Monroeville, Pennsylvania.

LOCA SIMULATION IN NRU[a]

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SUMMARY

The past quarter was devoted to completing the joint U.S. Nuclear Regulatory Commission (NRC)-United Kingdom (U.K.) test series composed of TH-2, TH-3, and MT-3. The second thermal-hydraulic test (TH-2) was successfully completed. The purpose of TH-2 was to establish and validate the test parameters for the third materials test (MT-3) and the data acquisition and control system (DACS) automatic variable rate reflood loop control system. All test objectives were accomplished.

A new test train was modified to meet the U.K. instrumentation requirements; it was assembled and shipped to Chalk River Nuclear Laboratories (CRNL), Chalk River, Ontario, for the MT-3 test. A brief thermal-hydraulic test series (TH-3), which reused the TH-2 test train, was conducted to calibrate loop response. MT-3 was then successfully conducted. The peak cladding temperature was held in the 1400 to 1525°F range for 180 sec. Post-test disassembly and examination revealed extensive cladding ballooning over a considerable length of the fuel rods (more than 60 in.) before rupture. All parties involved in the tests were very satisfied with the results.

INTRODUCTION

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

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

TECHNICAL PROGRESS

The TH-2 test cruciform (12 unpressurized fuel rods) was successfully installed in the previously used MT-2 guard rod/shroud assembly. Installation was accomplished on the disassembly, examination, reassembly machine (DERM) table under 5 ft of water in the fuel bay. The test train was inserted in the reactor loop, and a series of scoping tests were run to establish and validate the parameters and DACS control functional response for MT-3.

(a) FIN: B2403; NRC Contact: R. Van Houten.

Analysis of TH-2 test data indicated that certain control parameters could be improved to refine the loop control response and thereby increase the time-at-temperature to 180 sec. These changes were incorporated in the control system.

Before subjecting the MT-3 test train to the new parameters, an additional thermal-hydraulic test series (3 runs) was conducted prior to the MT-3 test by reusing the TH-2 test train. The TH-2 assembly was modified to include a desuperheater to reduce the exit steam temperature above the assembly. Additional instrumentation was added to monitor the exit steam temperature at the upper plenum area of the fuel rods more effectively. The improved performance parameters were validated, and MT-3 was conducted the following day. A peak cladding temperature from 1400 to 1525°F was maintained for 180 sec during the extended transient period. Pressure transducers indicated that all rods had ruptured.

DERM examination after the test confirmed all ruptures and extensive cladding ballooning over ~40% of the length of the rods. Single-rod profilometry revealed that ballooning (and ruptures) occurred between locations 70 to 130 in. on all rods. Figures 1 through 4 show the cladding profiles for the fuel rods in the "C" sector of the cruciform. Onset of ballooning occurred between 60 and 70 in. for each rod and extended to 130 to 135 in.

FUTURE WORK

The MT-4 test train is being assembled for the May 1982 test window at CRNL. All 32 fuel rods will be pressurized, and this test train will contain the prototype time domain reflectometry (TDR) liquid level sensor.

Arrangements are being made to obtain some previously irradiated (25-MWd) fuel rods for inclusion in the MT-5 test cruciform.

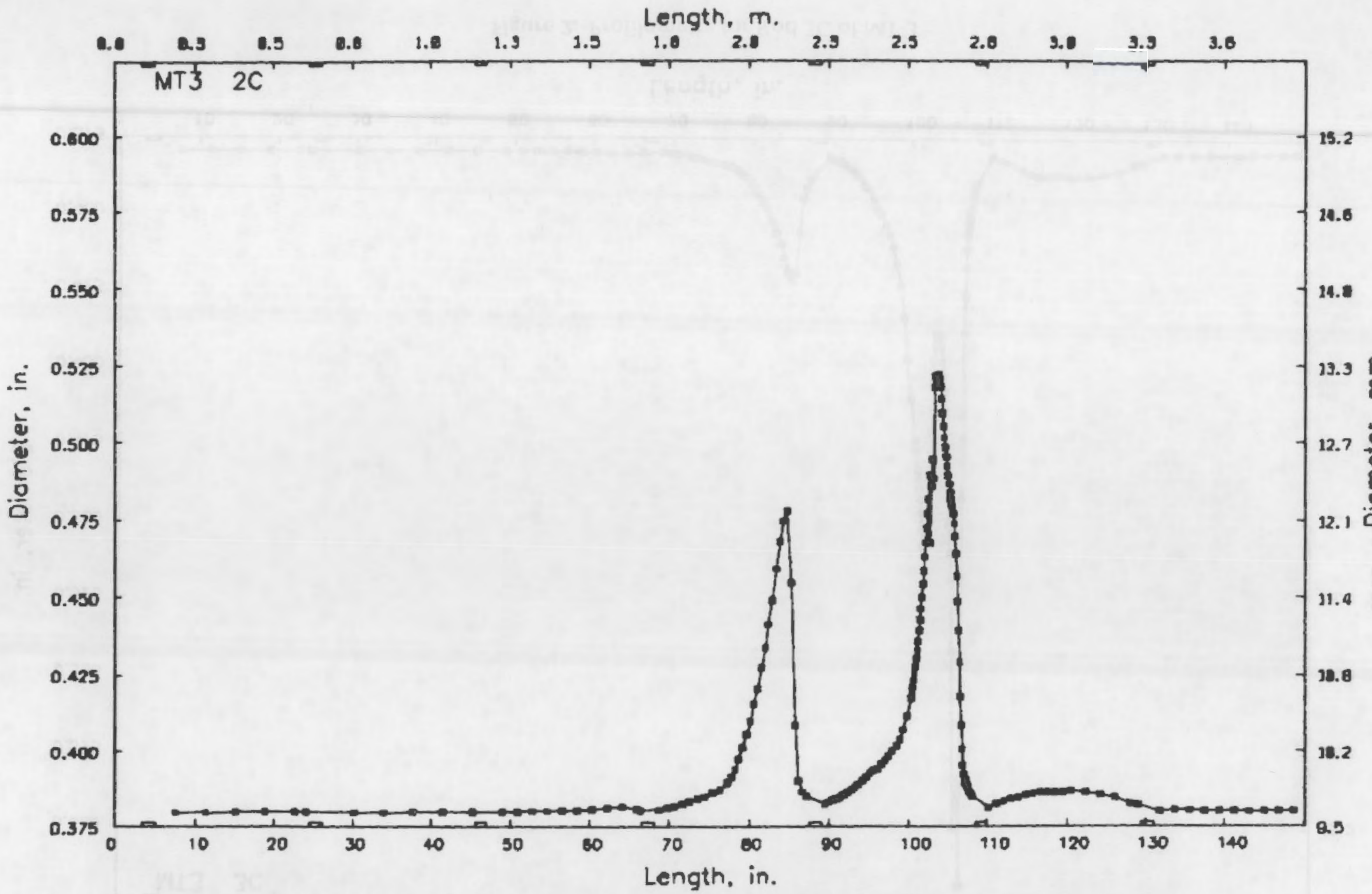

Figure 1. Profilometry for Rod 2C of MT-3

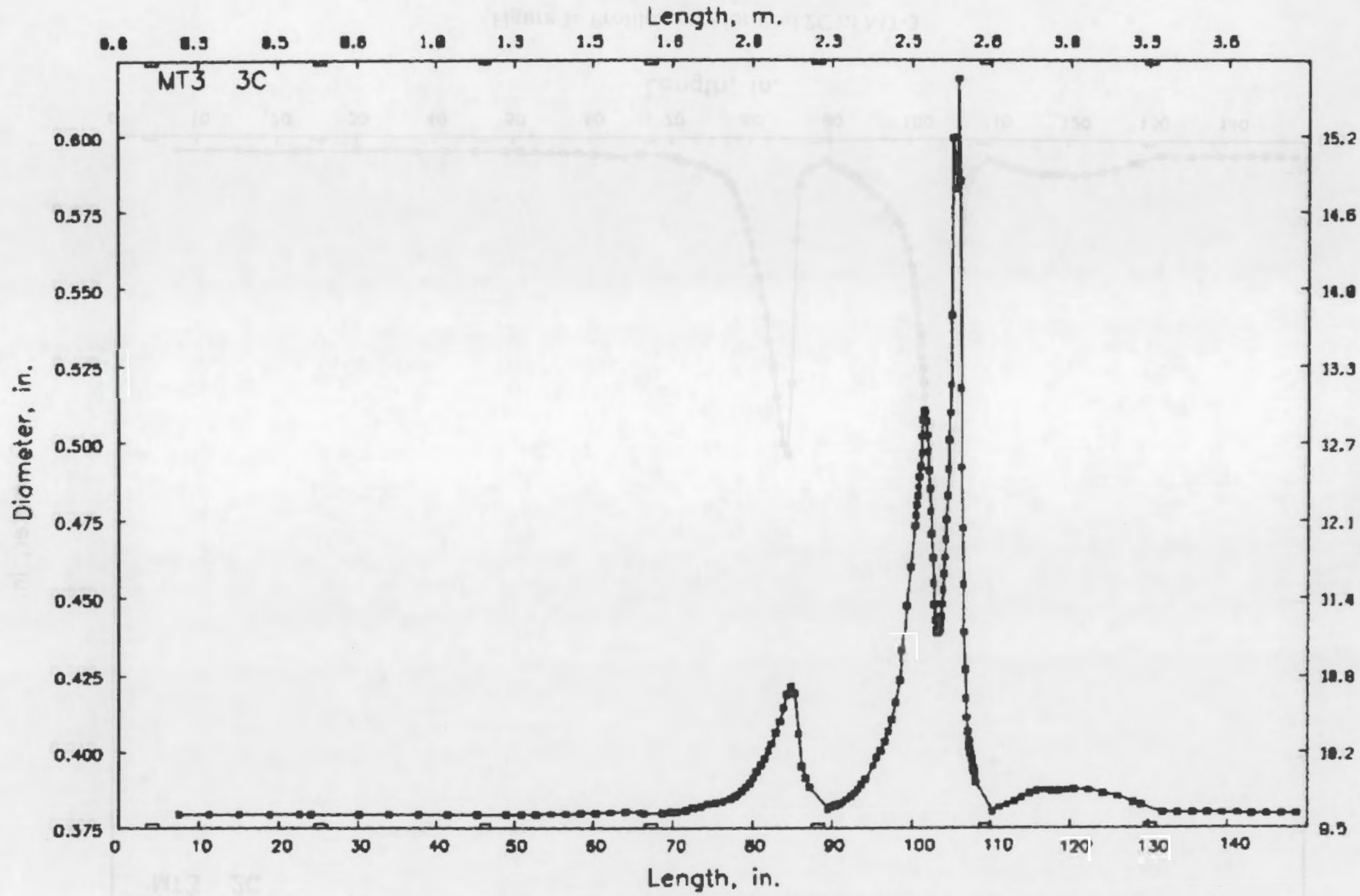

Figure 2. Profilometry for Rod 3C of MT-3

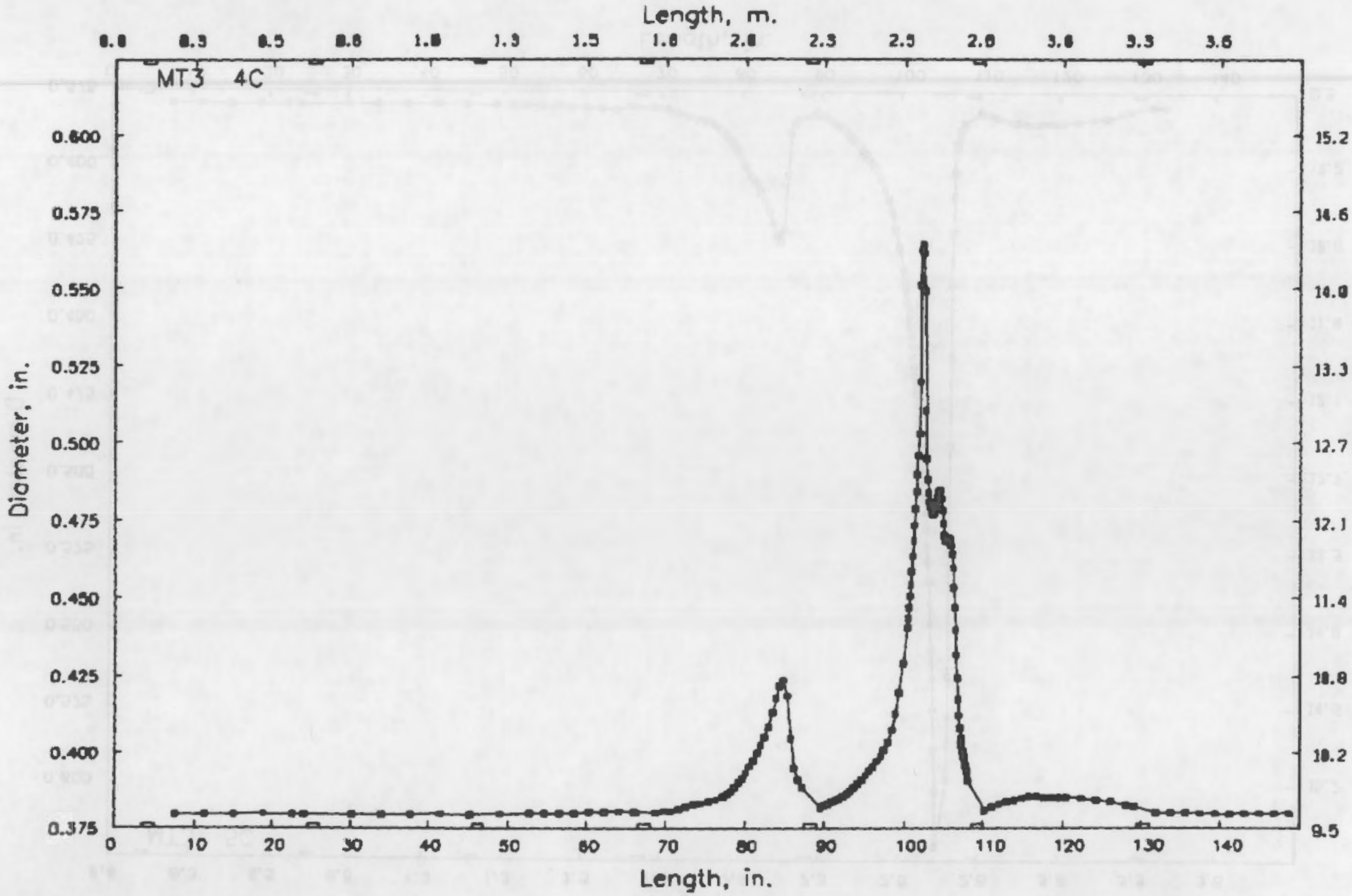

Figure 3. Profilometry for Rod 4C of MT-3

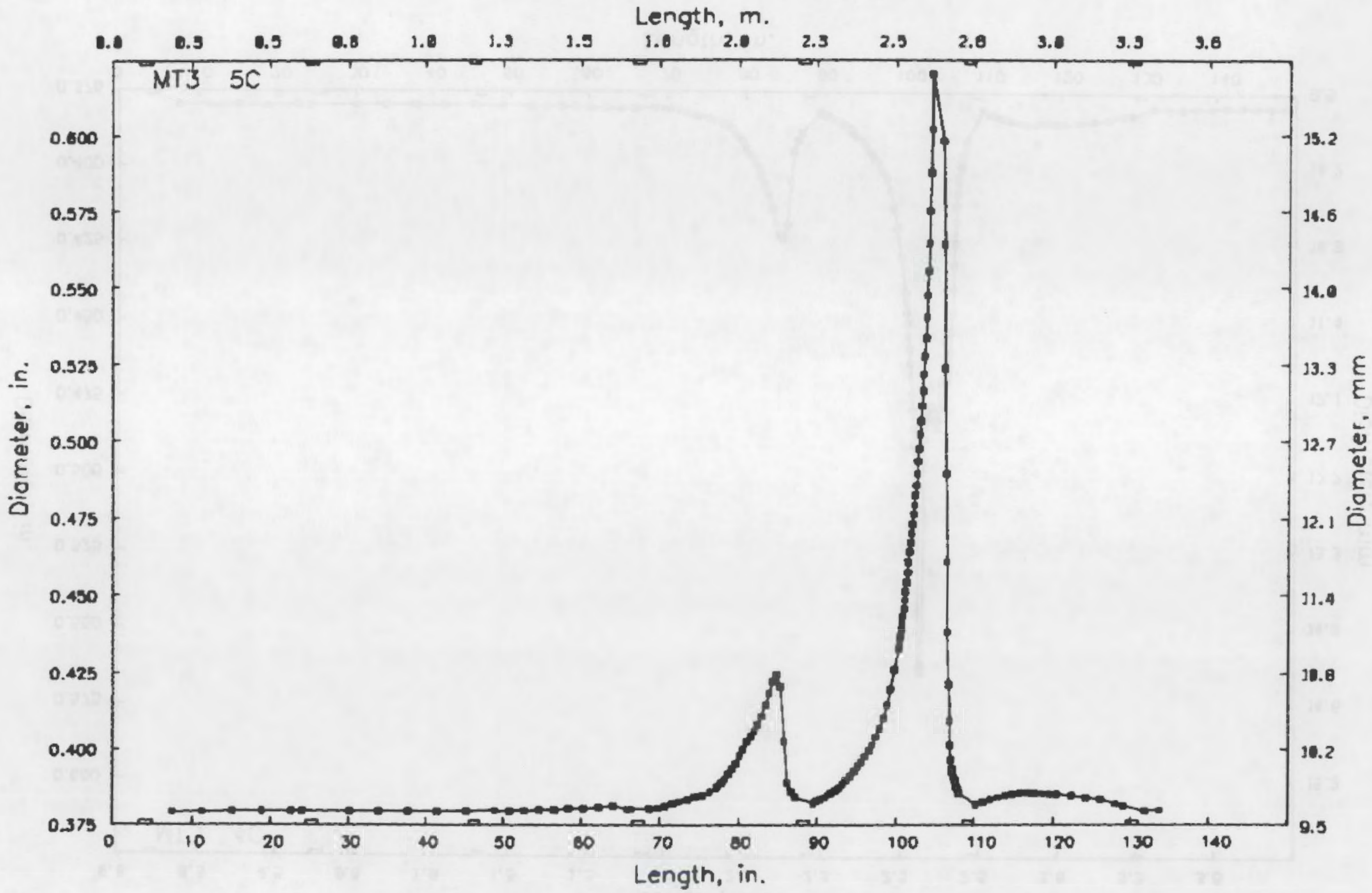

Figure 4. Profilometry for Rod 5C of MT-3

STEAM GENERATOR TUBE INTEGRITY[a]

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SUMMARY

The Steam Generator Examination Facility (SGEF) was completed, and operational control documents for use of the SGEF were produced. Efforts to establish a multiparticipant Steam Generator Group Project (SGGP) continued with technical marketing presentations and proposals to several concerns. A contract was placed for lifting the Surry generator into the SGEF; lifting equipment has been assembled onsite. In addition, generator research tasks for planning, scheduling, and costing to reopen preshipment penetrations; channel head decontamination; baseline eddy current in-service inspection (ISI); steam generator tube unplugging; a nondestructive testing (NDT) round robin; and secondary side access continued. Research also continued on stress corrosion crack (SCC) characterization and leak rate determinations.

INTRODUCTION

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

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

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

The first two phases of the program involved the study of mechanically (Phase I) and chemically (Phase II) defected tubing. Phase III of the original program included correlating the mathematical models developed in Phases I and II with actual service-flawed tubing. However, a lack of suitable specimens led to the redirection of Phase III into an effort to conduct extensive nondestructive and destructive

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

evaluations on a retired-from-service nuclear steam generator. A generator[a] removed from the Surry II nuclear plant (Surry, Virginia) after six years of service was judged suitable for this research.

Initial efforts on the Surry generator were concerned with licensing and transport activities to bring the unit from Virginia to Hanford, Washington. Since its arrival at Hanford, the generator has remained on a storage pad awaiting the completion of the specially designed containment facility (the SGEF). The SGEF is equipped to allow both nondestructive examination (NDE) and physical sectioning of the generator. Capabilities to perform chemical cleaning and decontamination are also included.

Research efforts on the Surry IIA generator will be initiated shortly after the generator is placed in the SGEF, the move is scheduled for January 1982. NRC-sponsored research will emphasize the following areas:

- validation studies of primary side NDT techniques and instrumentation
- verification of remaining integrity of service-defected steam generator tubes
- assessment of the secondary support structure integrity
- health physics - ALARA Control of Radiation Exposure associated with maintenance, repair, 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.

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

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

TECHNICAL PROGRESS

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

SURRY GENERATOR PROJECT

The Surry IIA generator remains at an access-controlled interim storage site on the Hanford Reservation. The generator is maintained under an argon gas purge that keeps it at a 1/2-psi positive pressure relative to the surrounding atmosphere. At the end of this reporting period, preparations were under way to lift the unit onto a trailer for transport to the SGEF.

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

Steam Generator Examination Facility (Task 1)

The SGEF was completed and accepted from the construction contractor in December 1981. Figure 1 shows a view of the tower portion of the facility. The generator will be set in its normal vertical operating position on the stand visible at the bottom of Figure 1 and shown in Figure 2. The liquid radioactive waste handling system is also shown in Figure 2. Space has been provided in the stainless steel catch basin for later installation of tankage that may be necessary for chemical cleaning and decontamination demonstration tasks. The rigid portable "greenhouse," which can be attached to the steam generator during cutting and sectioning operations to control the spread of contamination, is shown in Figure 3. In the figure, craftsmen prepare for a practice lift between floor elevations in the SGEF tower. A portion of the equipment room that contains HVAC[a] systems, the breathing air compressor, a heat recovery system, a vacuum system for radioactive sample stations, and multiple high-efficiency particulate air (HEPA) filters is shown in Figure 4.

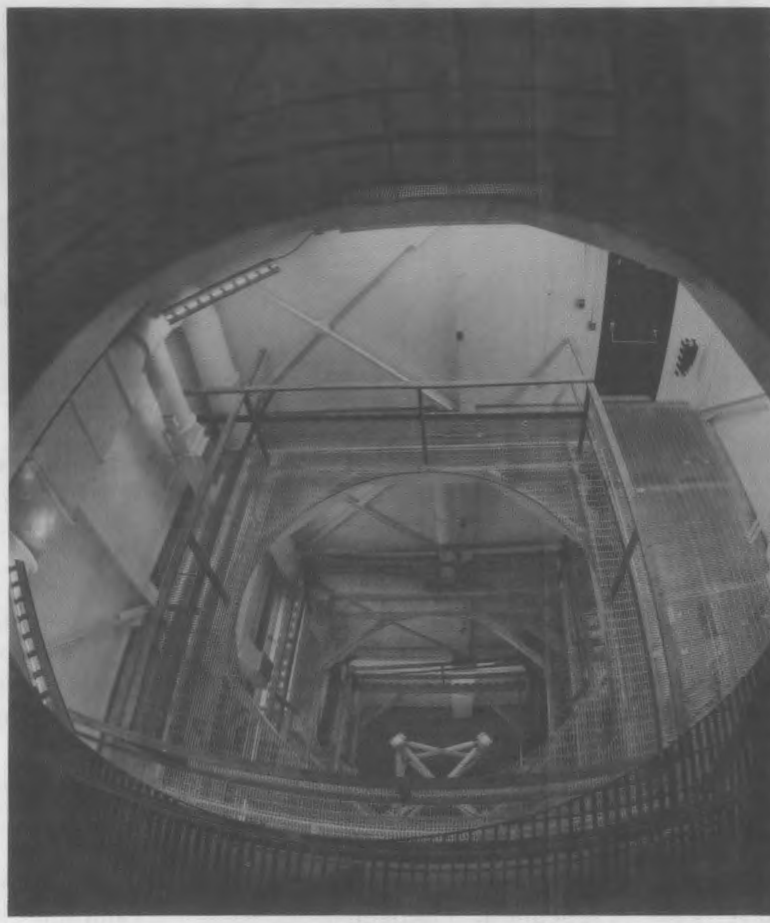

Figure 1. Steam Generator Examination Facility Viewed from the Uppermost Floor

(a) Heating, ventilating, and air conditioning.

Figure 2. View of the Stand for Supporting the Steam Generator in the SGEF Tower. The liquid radioactive waste handling system is shown in the background.

All documents in support of an Operational Readiness Review (ORR) were completed. The ORR must be satisfactorily completed before the SGEF can be used. The ORR board recommended and DOE-RL[a] concurred upon further review that the facility was ready for loading the Surry generator. A few ORR open items remain before experimental activities can begin. These items are mainly concerned with operating personnel training and training documentation.

Positioning the Generator in the SGEF (Task 2)

The 220-ton service-degraded steam generator will be lowered through a removable roof panel in the SGEF and positioned vertically. A subcontract has been issued to Neil F. Lampson Company to move the generator from the interim storage site and complete the required lift into the SGEF. The subcontract calls for task completion by January 15, 1982. At the end of this quarter, necessary equipment had arrived and was being assembled. This task, which was originally scheduled for December completion, was delayed due to weather conditions.

(a) U.S. Department of Energy - Richland Operations.

Figure 3. Rigid Portable Greenhouse that Connects to the Generator During Sectioning Operations

Generator Research Tasks

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

Demonstration of the data management system was conducted using historical ISI eddy current tapes of the Surry generator provided by the Virginia Electric and Power Company (VEPCO) and Westing-

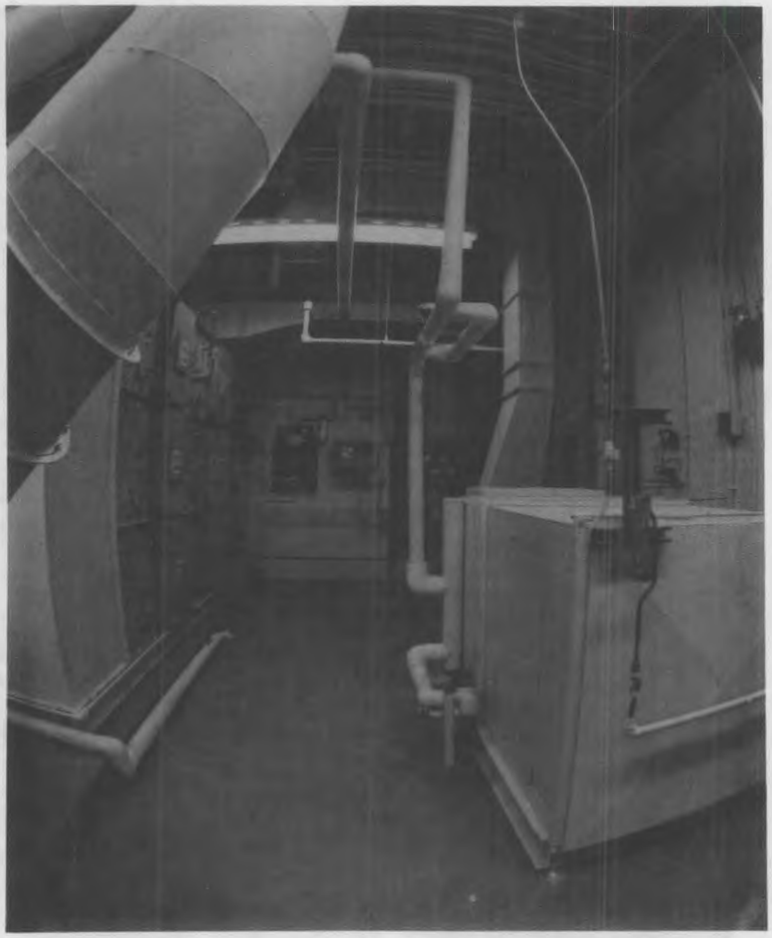

Figure 4. SGEF Equipment Room

house. Modifications to the 314 building (adjacent to the SGEF) have been initiated, where the PDP 11/44 for on-line data acquisition will be housed. The system should be established in its final location by February 1982.

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

The first research task to be done once the generator is placed in the SGEF will be to reopen preshipment penetrations (task 5). The preshipment examination involved cutting three penetrations through the generator shell while the unit was stored at the Surry nuclear station. Corrosion product and dimension data were acquired as was an assessment of general transportability and condition of the unit. Reopening these penetrations will provide information on any changes that may have occurred during shipment. The examination will be conducted by PNL personnel with in-house crafts providing cutting operations. The examination is scheduled for January and February 1982.

Task 6—decontaminating the channel head—is included in an effort to minimize radiation exposure during primary side access. The availability of the channel head is being offered as an opportunity for research-related development/demonstration of decontamination techniques. Discussions continue with a number of parties interested in demonstrating their techniques.

Baseline eddy current ISI (task 7) will be conducted for a 100% examination of all generator tubes and to establish the best possible nondestructive definition of the generator tube condition. Activities this quarter involved discussions with potential subcontractors; a statement of work has been generated. Equipment to be obtained by the SGGP for continued use in the generator channel head has been identified, and procurement was initiated. These items consist of the NDE probe pusher-puller, probe positioning device, and templates. The project is obtaining these items to avoid radiation exposure of participants in the later NDE round robin activities by eliminating the need for successive round robin teams to place their own similar devices in the channel head.

Management Activities

Efforts continued to broaden participation in and technical support of research on the Surry IIA steam generator. Because of the unique research potential provided by the availability of the service-degraded steam generator, NRC is allowing other parties to join in all or portions of the program. This quarter's contacts included Ontario Hydro (Canada), Korea Electric, MITI (Japan), and domestic PWR vendors. The SGGP organizational meeting will be held February 3-5, 1982, in Richland, Washington.

Other management activities have been concerned with preparations for the ORR of the SGEF. A number of publications and presentations were also completed this quarter.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

Leak Rate Tests

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

Preliminary results of this task were presented at the Ninth Water Reactor Safety Research Information Meeting. Data and videotape documentation of tests were forwarded to NRC.

MILESTONES

- The SGEF was completed.
- The subcontract for the steam generator move was signed.

PUBLICATIONS/PRESENTATIONS

- "Planned Research Program on the Retired Surry IIA Steam Generator." Presented at the ANS winter meeting, November 30, 1981, San Francisco, California.

- "NDE Round Robin Possibilities on a Retired-from-Service PWR Steam Generator." Presented at the Ninth Annual Water Reactor Safety Research Information Meeting, October 26-30, 1981, Gaithersburg, Maryland.
- "Leak Rate Studies on Laboratory SCC Steam Generator Tubes." Presented at the Ninth Annual Water Reactor Safety Research Information Meeting, October 26-30, 1981, Gaithersburg, Maryland.
- "Steam Generator Integrity Research." Presented to Portland, Oregon Chapter of American Nuclear Society, November 19, 1981.
- *A Description of the Data Acquisition and Statistical Analysis Computer Systems for the Steam Generator Group Project.* Submitted for review.

FUTURE WORK

During the coming quarter the following activities will be pursued:

- organizational meeting of the SGGP; continued project marketing
- place the generator into the SGEF
- conduct a destructive assay of round robin SCC specimens and issue a topical report
- continue leak rate tests
- conduct task 5—reopen preshipment penetrations.

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16. ABSTRACT *(200 words or less)*

This document summarizes the work performed by Pacific Northwest Laboratory (PNL) from October 1 through December 31, 1981, for the Division of Accident Evaluation, U.S. Nuclear Regulatory Commission (NRC). Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision-making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities. These programs include loss-of-coolant accident (LOCA) simulation tests at the NRU reactor, Chalk River, Canada; fuel rod deformation, severe fuel damage, and postaccident coolability tests for the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho. These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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