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Quarterly Report
October-December 1982

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
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Battelle Memorial Institute

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U.S. Nuclear Regulatory
Commission

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Reactor Safety Research Programs

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Reactor Safety Research Programs

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ABSTRACT

This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1982, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities, including loss-of-coolant accident simulation tests at the NRU reactor, Chalk River, Canada; the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

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GRAPHITE NONDESTRUCTIVE TESTING (NDT) RESEARCH(a)

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SUMMARY

The sweep-frequency eddy-current instrument was successfully checked out. A new transducer was designed that will allow measurements to be obtained at close intervals over a wide frequency range. Oxidation rates for the two new logs of PGX graphite have been observed to differ by more than an order of magnitude.

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:

- Modify the eddy-current probe design and obtain calibration constants to permit measurements of oxidation profiles in prototypic high-temperature gas reactor (HTGR) core support post graphite.
- Initiate tests on graphite oxidized in the component flow test loop (CFTL) at Oak Ridge National Laboratory (ORNL) to validate the performance of the sweep-frequency eddy-current technique for obtaining near-surface oxidation profiles in HTGR core support posts.
- Complete development of ultrasonic backscattering techniques for evaluation of in-depth oxidation profiles in HTGR core support components.
- Complete study on the variability within PGX graphite of the critical properties that could affect the strength versus electrical conductance correlation.

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

TECHNICAL PROGRESS

EDDY-CURRENT TESTING

The sweep-frequency eddy-current system checkout was completed, and experimental data indicate that this system is providing highly accurate and orthogonal complex impedance-plane data. Data were obtained on graphite at frequencies ranging from 50 kHz to 10 MHz and compared with computer-predicted data for ideal conditions. This comparison indicated that the nonlinearities in the experimental data are small, and we are ready to complete the ZFIT eddy-current modeling task.

(a) FIN: B2101-3; NRC Contact: R. B. Foulds.

A special eddy-current test probe is being designed to obtain data on oxidized graphite logs from the Core Support Performance Tests (CSPT) at ORNL. The probe design will allow measurements to be obtained from two positions on the surface of oxidized cylinders of 2020 graphite in the CFTL. Additional measurements will be made on the 19.57-cm diameter cylinders after removal from the CFTL; but the in situ measurements will allow direct comparison of changes occurring in those areas during the oxidation run. ORNL has delayed the start of "run zero" for the CFTL; it is now anticipated that the preoxidation measurements on the graphite specimens will be made sometime in March 1983.

ULTRASONIC TESTING

During this quarter, a task and milestone plan was developed for fiscal year 1983; and work resumed on development of the ultrasonic backscattering technique. The design for an improved transducer concept was completed; this design utilizes a plano-concave transducer element that provides a broadband response over the optimum frequency range. The new transducer design will allow data to be taken at many intervals over the frequency range of interest rather than at three or four discrete frequencies and should materially improve the precision of the ultrasonic backscattering measurements. The required plano-concave transducer elements and the material for the waveguides or stand-offs were purchased this quarter. The transducer will be fabricated in February 1983 and data acquisition will proceed.

OXIDATION

During the past quarter, cylindrical samples have been oxidized up to about 9% weight loss; however, several equipment failures have caused a delay in obtaining the data on strength versus density for the two new logs of PGX graphite. There was a very significant difference in the oxidation rates of samples from the two logs; samples from log IX oxidized roughly 12 times faster than samples from log VIII. Some log VIII samples had large areas of rust-yellow coloration after a short time in the furnace, which may indicate high impurity concentrations. Ash samples will be obtained after the strength measurements are completed to check this hypothesis.

The compressive strengths of unoxidized samples from the three logs of PGX graphite correlate very well as a linear function of density. Densities of the 30 samples ranged from 1.690 to 1.790 Mg/m³; their compressive strengths ranged from 29.1 to 45.6 MPa (4220 to 6610 psi); and the mean values are 1.746 Mg/m³ and 38.0 MPa, respectively. The least squares correlation equation is $S = -225.3 + 150.8 \rho$, where S is the compressive strength in MPa and ρ is the density in Mg/m³. This equation can be inverted to obtain the density at which the graphite would no longer be self-supporting: at $S = 0$, ρ ($S=0$) = $225.3/150.8 = 1.494$ Mg/m³.

FUTURE WORK

EDDY-CURRENT TESTING

- complete the ZFIT eddy-current modeling computer algorithm
- complete the design and fabrication of the CSPT eddy-current probe
- make measurements on samples of 2020 graphite
- make preoxidation measurements at ORNL on the graphite to be in test zero of the CSPT series.

ULTRASONIC TESTING

- fabricate plano-concave transducer
- verify the oxidation/backscattering relationship in a two-step procedure: 1) with the graphite immersed in water and 2) using waveguides and dry-coupling techniques.

OXIDATION

- measure strength of oxidized PGX cylinders
- obtain correlations of strength versus density for oxidized PGX
- obtain correlations of electrical conductance versus density for oxidized PGX
- fabricate standards from 2020 graphite sent from ORNL.

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

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SUMMARY

The ZB-1 intermediate-scale vessel test began at Mannheim, West Germany. In the first step of the test matrix (hydrotest to planned operating pressure), acoustic emission (AE) data warned of impending failure in a degraded steel section that had been inserted in the vessel wall by the Germans. The degraded steel section was replaced, and testing will resume by mid-January 1983. Permanent signal lead wires are still being installed for AE monitoring at the Watts Bar 1 reactor; arrangements are being made to AE monitor the hot functional tests (scheduled for April 1983). Early stress corrosion cracking (SCC) in a Type 304 stainless steel (SS) pipe test specimen near a weld produced several thousand detected AE signals. Physical examination showed that this AE came from two separate cracks that were about 0.5 in. long by 0.06 in. deep. An engineering prototype AE monitor system is being designed that will include information needed for leak detection monitoring methods being developed at Argonne National Laboratory (ANL).

INTRODUCTION

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

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

TECHNICAL PROGRESS

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

OFF-REACTOR VESSEL TEST

The major effort this quarter was devoted to the ZB-1 intermediate-scale vessel test at Mannheim, West Germany. As previously described, ZB-1 is a 120-mm thick A508 steel vessel with two inserts in the cylindrical wall. One insert is U.S. A533B steel and the other is a degraded A508 German steel—KS07. The AE monitor system and associated instrumentation—such as a crack opening displacement (COD) measurement system—were shipped to Mannheim and installed at the test site in late September 1982. The test facility is located on the Rhine River at Mannheim; it is part of the Grosskraftwerk Mannheim (GKM) Utility plant site and is administered by GKM. Test instrumentation

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

is located in a trailer outside the entrance to the test bunker. Figure 1 shows the AE instrumentation and the function of the subsystems.

The ZB-1 test vessel is situated about 50 to 60 ft underground on the bottom level of the test bunker. Figure 2 shows the COD gauges and flow noise simulator installed inside the vessel on the A533 insert. The two inside diameter (ID) flaws and the one outside diameter (OD) flaw in the A533 insert are fatigue presharpened.

AE sensors and accessories are situated on both the A533B insert (Figure 3) and the KS07 insert (Figure 4). A primary waveguide AE sensor array (WG 1, 2, 3, and 4) was mounted on each of the inserts by threading the waveguides into drilled and tapped 5 to 40-in. by 3/16-in. deep holes. Lower frequency hydrostatic test monitor sensors are situated on the A533B insert in an array similar to the waveguides (Hydro 1, 2, 3, and 4). These sensors are mounted on the surface with magnets. One other waveguide AE sensor array (not shown in the photos) is installed at each end of the cylindrical portion of the vessel and includes the total vessel cylinder. The waveguides are pressure coupled to the vessel surface to simulate mounting on a reactor vessel.

Step 1 of the vessel test matrix, which was performed in October 1982, consisted of a hydrostatic test intended to reach a maximum of 1.0 times the maximum cyclic pressure planned (240 bar). Figure 5

Figure 1. AE Instrumentation for ZB-1 Test

Figure 2. Inside Surface of A533B Implant in ZB-1 Vessel

shows vessel pressure and AE from the KS07 insert versus time for the first pressurization to 140 bar. This in itself is not particularly noteworthy except for the abrupt increase in AE during the final stage of pressurization. Pressurizing was stopped at 140 bar due to pressure system problems. The added factor that created awareness of a possible problem with the KS07 insert was the predominance of high-amplitude AE signals (≥ 10 V), which showed source locations tending to cluster along a circumferential weld (Figure 6). This created concern that crack growth was taking place even at the relatively low pressure of 140 bar. Low-amplitude (1.5 to 3.0 V) AE was detected from the A533B insert with scattered source indications, which is the expected response to the limited pressurization.

The decision was made to proceed with the hydro test; pressurization was incremental with a pressure hold between increments to determine if the AE ceased, which would indicate crack arrest. This approach was followed in Phase 2 of the Step 1 hydro test, which ended at 163 bar due to a leak in the pressure system. The pattern of the AE from both of the inserts was similar to that described above for Phase 1.

Phase 3 of the Step 1 hydro test went to 232 bar, where the test was stopped due to concern for safety of the vessel. Figure 7 shows vessel pressure and AE from both inserts versus time for Phase 3. The AE data from the KS07 insert for the last 28 min of the test (Figure 8) indicated an increasing condition of

Figure 3. Sensors on A533B Implant

continued AE during pressure holds. There were also audible "bangs" from the vessel during this period; the noise was picked up by a microphone placed near the KS07 insert. If AE signal amplitude distributions for the A533B insert and the KS07 insert are compared (Figure 9), it is again evident that most of the AE from KS07 was high amplitude (≥ 10 V) while that from A533B was mostly below 4-V amplitude. Source location information in Figures 10 and 11 shows that the high-amplitude signal sources for the KS07 insert tended to cluster along the circumferential weld while they are well distributed in the A533B insert.

Summarizing the Step 1 test results from ZB-1:

- The data discussed fully supports a decision to replace the KS07 insert.
- AE signal amplitude in the case of heavy section structures appears to be a much more definitive parameter than was indicated in any of the laboratory specimen testing. This parameter will be carefully examined in the forthcoming ZB-1 vessel testing.

The KS07 insert was replaced in late December 1982, and the test will resume by mid-January 1983.

Figure 4. Sensors on KS07 Implant

Figure 5. ZB-1 Vessel Pressure and KS07 Insert AE Versus Time

Figure 6. Source Location for 6- to 10-V Amplitude AE from KS07 Insert (Phase 1 of Step 1 Hydrotest)

Figure 7. AE and Pressure Versus Time for Phase 3 of Step 1 Hydrotest

Figure 8. AE from KS07 Insert and Pressure Versus Time for Last 28 min of Phase 3 of Step 1 Hydrotest

Figure 9. AE Signal Amplitude Distribution for Both Inserts During Phase 3 of the Step 1 Hydrotest

Figure 10. AE Source Location Plots for KS07 Insert for the Final Phase of Step 1 Hydrotest

Figure 11. AE Source Location Plot for Signals ≥ 6 V for A533B During Phase 3 of Step 1 Hydrotest

REACTOR MONITORING

The conduit for the permanent AE system signal leads was installed at Watts Bar 1 reactor, and cable pulling began. The official schedule for hot functional testing is March 29 through April 28, 1983, which conflicts with the ZB-1 test. The ZB-1 test backup AE monitor (modified D/E 1032 system) will be shipped from Germany, and a waveform recorder will be obtained that can be used with the backup monitor.

PIPE MATERIAL CHARACTERIZATION

The test weld in an accelerated SCC test specimen that failed at an installation weld has been investigated. Test definition and the AE detected from the test weld prior to failure are shown in Figure 12. Subsequent sectioning of the specimen showed the evidence presented in Figure 13. This evidence is very positive in the sense that intergranular SCCs of this size produced a discernible AE response of about 4000 event counts.

A second accelerated test specimen with external load to 80% of yield is being tested to check the reproducibility of the above results and to attempt to learn more about parameter changes such as the signal amplitude associated with crack progression. A long-term test specimen with no external load has not yet produced any AE indications after about 4 months of testing.

Figure 12. AE Data from Stress Corrosion Cracking Test

ENGINEERING PROTOTYPE AE SYSTEM

The engineering prototype AE monitor system is being designed. The basic design incorporates 48 AE sensing channels with the capability to expand the number of channels. The ultimate number of data channels in one system will be determined not only by technical limitations in data processing speed but also by the question of how much monitoring capability should be dependent on one processor unit. The 24 sensor channels that are being fabricated will be sufficient for Watts Bar 1 reactor monitoring.

Consideration of leak detection monitoring is also being addressed. As an outgrowth of discussions with ANL regarding their NRC-sponsored leak detection program, a module for measuring signal RMS is included in the monitor system design.

REPORTS

- Hutton, P. H., and R. J. Kurtz. 1982. *Acoustic Emission Estimations of Flaw Significance in Reactor Pressure Boundaries*. PNL-SA-10396, presented at the Conference on Periodic Inspection of Pressurized Components, October 12, 1982, London, England.
- Quarterly progress report for the period from July to September 1982.

a) Dye Penetrant Indications

0.010 in.

0.010 in.

b) Crack Depth at Center of Indication

Figure 13. Stress Corrosion Cracking Associated with the AE in Figure 10

FUTURE WORK

In the period from January 1 to March 31, 1983, the following work will be completed:

- perform the ZB-1 vessel test
- complete preparations for monitoring hot functional testing at the Watts Bar 1 reactor
- perform signal detection and source location tests on an uninstalled reactor vessel, focusing on the belt-line region and nozzle blend radius
- continue pipe material tests
- continue development of the engineering prototype monitor system.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS(a)

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SUMMARY

During the past quarter, progress was made on a number of tasks. The initial results of the detailed pipe inspection round robin (PIRR) analysis were very similar to the preliminary results. Based on the PIRR and underclad work, a presentation was made to the Advisory Committee on Reactor Safeguards (ACRS) on 1) Regulatory Guide 1.150, 2) the effectiveness of ultrasonic testing (UT) to detect underclad cracks—of importance to the pressurized thermal shock (PTS) issue, and 3) the effectiveness/ineffectiveness of UT to detect intergranular stress corrosion cracking (IGSCC) in Nine Mile Point piping. Fracture mechanics (FM) analysis indicated the tradeoff of nondestructive examination (NDE) inspection versus the frequency of inspection for a given level of probability of nondetection.

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

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

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

The scope of this program is limited to ISI of primary systems, but 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.

PIPE INSPECTION ROUND ROBIN

The PIRR has been described in past quarterly reports. Design data were limited to prevent the premature release of information that would compromise the validity of the results. A complete report is now being prepared for release next quarter. The following discussion highlights the most important results generated by the analysis performed to date.

The PIRR was conducted to measure the reliability of crack detection by UT in reactor coolant piping. Six teams from commercial ISI organizations conducted a total of 1500 pipe inspections using their own field inspection procedures and a procedure written by PNL. The inspected pipe specimens included centrifugally cast stainless steel (CCSS) main coolant pipe, ferritic steel main coolant pipe clad with stainless steel on the inner surface (clad ferritic), and 10-in. diameter Schedule 80 stainless steel pipe representative of recirculation and emergency coolant piping. All specimens contained pipe-to-pipe butt welds. Most specimens contained laboratory-induced thermal fatigue cracks; some 10-in. pipes contained laboratory-induced IGSCCs; and some specimens had no defects (blanks). Approximately 80 specimens were used.

An inspection folder for each of the 1500 inspections contained calibration information, raw UT data, PNL's strip chart records of the inspection, and each team's inspection report form. Each inspection folder was manually reviewed to tabulate the types and frequencies of error occurrences. Certain types of errors were corrected; for example, errors due to a misunderstanding of the coordinate system or simple transcription errors. Other errors were corrected but noted separately. These included errors that would have caused, for example, a detected defect to be scored as a missed defect. With the removal of these errors, it is possible to score the teams as if they performed without error and then to apply error rate functions to observe the effect of these errors on detection reliability.

The crack detection performance of the ISI teams is expressed in terms of recording probability, probability of detection, and false call rates. Recording and detection probabilities are presented as functions of crack depth. False call rates can be expressed either as a number of false calls per inch of weld or as the percentage of blank specimens that were called cracked.

Recording probability (RP) is the probability that a signal from a defect will be obtained and recorded by the team's Level I and II inspectors. The RP is primarily a function of the team's procedure, equipment, and how carefully the inspection is performed.

Probability of detection (POD) is the probability that a signal from a defect will be obtained and recorded by the Level I and II team members and will be correctly determined to be a defect signal by the Level III team member.

False calls are noncrack UT indications that are incorrectly determined to be cracks. Other types of error rates, while not specifically calculated, can be estimated from POD and RP plots. For example, a miscall is to record a crack signal and falsely call it a noncrack signal (RP minus POD).

Figure 1 presents the average POD curve for all six teams obtained from the inspection of clad ferritic pipe with near-side access using field procedures.^(a) With current ISI practice, the POD is at least 70% for cracks 15% through-wall or deeper. The point on the plot at the 0% through-wall depth represents the false call rate; of all the blank clad ferritic specimens inspected, only about 4% were called cracked. The results for far-side access were quite similar to those for near-side access, indicating that the weld metal does not degrade the inspection.

The average POD curve using the PNL improved procedure on the clad ferritic pipe with near-side access is shown in Figure 2. The performance is nearly perfect: only one crack was missed, and the false call rate was still low. The improved procedure basically corresponds to Section XI Code Case N-335, which is a complete rewrite and enhanced version of Section XI Appendix III. It was concluded that ISI reliability can be very high for ferritic pipe welds.

Figure 3 presents the RP and POD curves for one team that used their field procedure on CCSS pipe. The curves seem high, but the false call rate is also very high; this team achieved a high detection rate by essentially calling all the material cracked. The other teams did not take that approach, and their results cluster around the 20% to 30% level with 20% to 30% false call rates. (Two teams declared it a "no test," stating that they had no confidence in their ability to effectively inspect CCSS pipe.) The PNL improved procedure, which centered around the use of an optimized transducer, was equally ineffective. It was concluded that CCSS pipe should be considered uninspectable using current manual UT techniques.

Figure 1. PIRR Results for Clad Ferritic Pipe; Near-Side Access; Field Procedures

(a) Called "Code" in the plots; actually, most procedures were enhanced beyond Section XI minimum.

Figure 2. PIRR Results for Clad Ferritic Pipe; Near-Side Access; Improved Procedure

Figure 3. PIRR Results for One Team on CCSS Pipe; Field Procedure

Figure 4 presents the POD curves for each team obtained from the inspection of 10-in. stainless steel pipes containing IGSCC with near-side access using field procedures. The heavy black line is the average POD. (False call analysis for the 10-in. pipes is not complete; any data points at zero crack depth should be ignored.) The curve with square data points represents the performance of a team using a minimum Code procedure. The five other teams used procedures enhanced in one or more ways beyond the minimum.

Figure 4. PIRR Results for IGSCC in 10-in. Pipe; Near-Side Access; Field Procedures

Figure 5 is similar to Figure 4 but includes all 10-in. stainless pipes (i.e., both IGSCC and thermal fatigue cracks) with near-side access and field procedures. Again, square data points represent the performance of a minimum Code procedure; the spread between the curves is quite striking.

Figure 6 shows the performance of the six teams on all 10-in. pipes with near-side access using the PNL improved procedure. The improvement in average performance (heavy line) is not large because most of the field procedures were already enhanced in similar ways. There was substantial improvement over the team that had used the Code minimum and that improvement would probably have been greater if there had been more time to get accustomed to the improved procedure. The most striking feature of this plot is the variability in crack detection performance between teams using identical procedures and ultrasonic transducers.

Figure 7 presents the average POD curve for all six teams for all 10-in. stainless steel pipe using field procedures in the far-side access condition (when the sound beam must pass through the weld metal). The spread in the data is shown for illustration (raw data were removed from the other 10-in. pipe plots for the near-side access condition for clarity). Since some of the data points represent more inspections than others, the line was fit using a "maximum likelihood" technique. The defect detection performance for far-side access is clearly very poor. The false call rate for this plot is ~22%, which is roughly equal to the POD.

It was concluded that for 10-in. stainless steel pipe inspection ideal performance (such as shown in Figure 2) cannot be achieved in manual UT with current technology. Even with identical equipment and procedures, personnel variables introduce a large spread in detection performance. When near-side access is impossible, inspection is ineffective. Equipment and procedures to be used for inspection of stainless steel pipe welds should be assessed for suitability by the NRC, and inspection personnel should undergo a "blind test" of crack detection performance to qualify their capability. NDE reliability, thus quantified, can be used as input to FM calculations, leading to more effective ISI sampling plans.

Figure 5. PIRR Results for IGSCC and Thermal Fatigue in 10-in. Pipe; Near-Side Access; Field Procedure

Figure 6. PIRR Results for IGSCC and Thermal Fatigue in 10-in. Pipe; Near-Side Access; Improved Procedure

Figure 7. PIRR Results for IGSCC and Thermal Fatigue in 10-in. Pipe;
Far-Side Access; Field Procedures

The following conclusions have resulted from the PIRR:

- UT inspection of clad ferritic main coolant pipe can be 100% effective if adequate sensitivity is used. Section XI minimum sensitivity (50% of the notch amplitude) is not adequate.
- Inspection of clad ferritic pipe is equally effective with and without weld metal in the sound path.
- UT inspection of centrifugally cast stainless steel is ineffective using current manual techniques.
- Section XI minimum requirements do not provide effective inspection of wrought stainless steel pipe welds. Increased sensitivity and selection of optimized transducers improves detection reliability.
- When the sound beam must pass through the butt weld in wrought stainless steel pipe, UT inspection using current field techniques is ineffective.
- Even when using identical equipment and procedures, detection reliability varies significantly.
- Crack detection reliability in UT inspection of stainless steel pipe welds should be qualified.

PROGRAM OUTPUT FOR REGULATORY ISSUES

Based on the experience and knowledge gained from the PIRR and the underclad crack detection work (reported last quarter), the NRC requested information on several specific issues. The ACRS requested that a presentation be made on the research activities as they relate to:

- failure to detect IGSCC with UT at Nine Mile Point
- Regulatory Guide 1.150 on the inspection of reactor pressure vessels
- effectiveness of ultrasonic techniques for the inspection of U.S. nuclear pressure vessels for small-sized near-surface cracks of interest to the PTS issue.

In addition, discussions were held with licensing staff to provide input for revisions to IEB 82-03. The PIRR formed the basis for the recommendations to this bulletin, and a letter containing the recommendations and rationale was mailed in December to Joe Collins of I&E.

FRACTURE MECHANICS TASK

The objective of this task is to apply deterministic and probabilistic FM to guide the development of ISI requirements. During this quarter, preliminary flaw detection curves from the PNL round robin were applied in piping reliability calculations. The results indicate the potential of improved NDE procedures to increase the reliability of a piping system subject to service-induced cracking.

Results of calculations being performed at Lawrence Livermore Laboratory (LLL) will be available early next quarter. These calculations simulate the cracking that has been experienced by PWR steam generator feedwater lines. The crack growth and failure rates are being treated in a probabilistic fashion. Alternate scenarios of crack detection curves and inspection intervals are being simulated.

Deterministic FM calculations are also being applied to establish optimum combinations of flaw detection capability and inspection interval (see Figure 8). These results are based on an SCC growth predic-

Figure 8. Effect of Inspection Interval on Detection of IGSCC

tion for 10-in. diameter piping as described in a recent report from the Electric Power Research Institute.⁽¹⁾ The crack detection curves cover the range of poor to good as observed for ferritic piping in the PNL round robin. The improved curve is an estimate of the enhanced flaw detection capability that can be expected through field application of improved procedures and technology. Clearly, the probability of detecting a flaw prior to leakage increases significantly with improved NDE procedures and with more frequent inspections over the life of the component. As the flaw detection capability increases from "poor" to "good" to "improved," the probability of failure (leak) can be decreased without unreasonably short inspection intervals. For example, the "good" curve is at least as effective with two inspections as the poor curve with 20 inspections.

Calculations during the next quarter will expand the scope of the calculations. Flaw detection curves will be selected to span the entire range of conditions of the PNL round robin, and the effect of uncertainties in crack growth rates will be treated.

FUTURE WORK

- complete analysis of PIRR data
- fabricate test blocks for calibration reflector studies
- fabricate test blocks for cladding effects studies.

REFERENCES

1. General Electric Company. July 1982. *The Growth and Stability of Stress Corrosion Cracks in Large-Diameter BWR Piping*. NP-2472, prepared for BWR Owners' Group and Electric Power Research Institute.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES(a)

D. D. Lanning, Program Manager

D. D. Lanning, Task A Leader
M. E. Cunningham, Task B Leader
W. N. Rausch, Task C Leader
J. O. Barner, Task D Leader

E. R. Bradley
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SUMMARY

The principal objectives of this program are to obtain in-reactor and ex-reactor data in thermal and mechanical fuel rod performance and to integrate these data into the FRAPCON-2 computer code. During this quarter, optical ceramography of the instrumented fuel assembly (IFA)-432 and IFA-527 rods undergoing postirradiation examination (PIE) was completed. Data for IFA-432 and IFA-518 for the period from May 12 to August 24, 1982, were received, processed, and summarily examined. Programming of Version 1, Modification 4 (V1M4) of FRAPCON-2 was completed. The second nonirradiated pellet-cladding interaction (PCI) test was successfully completed.

INTRODUCTION

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

- **Task A** - collect and correlate in-reactor and PIE data on fuel rod thermal/mechanical behavior, especially as a function of rod design and burnup
- **Task B** - qualify, organize, and analyze the fuel performance data and report the data, trends, and conclusions
- **Task C** - integrate the above information into the FRAPCON series of computer codes
- **Task D** - study the occurrence and mechanisms of cladding failure using controlled experiments with centrally heated simulated fuel pins in a pressurized water loop at PNL.

The Halden Boiling Water Reactor (HBWR), Halden, Norway, is currently the sole site used by this program for irradiation tests. PIE is being carried out at the AERE-Harwell^(b) laboratories in the United Kingdom. The in-reactor test matrix now spans the full range of normal BWR conditions for pelletized UO₂ fuel, including:

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50 to 380 μm (0.002 to 0.015 in.)
- initial gas compositions ranging from pure helium to pure xenon

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

(b) Atomic Energy Research Establishment.

- fuel densities of 95% and 92% theoretical density (TD), the latter both stable and unstable regarding in-reactor densification
- burnups up to 52 GWd/MTM
- alternate fuel designs (annular fuel pellets, coated cladding, and sphere-pac fuel).

Five instrumented test assemblies have been irradiated thus far in the program. Two of the assemblies—IFA-432 and IFA-518—are still under irradiation.

TASK A - IRRADIATION EXPERIMENTS

Optical ceramography on rods 1 and 6 of IFA-432 and rod 4 of IFA-527 has now been completed at Harwell. The IFA-432 rods will be redone for the final report for improved quality and clarity. The micrographs reveal much greater cracking and relocation in IFA-432 rods than in equivalent fuel types in IFA-527 (which operated at less than half of the linear heat generation rate) as reported last quarter.⁽¹⁾ A lack of columnar grain growth was also observed in the micrographs for rods 1 and 6 of IFA-432, which raises the question of whether the assumed thermocouple decalibration rate⁽²⁾ for the IFA-432 assembly is correct. Using the decalibration rate that had been assumed for the assembly and extrapolating thermal resistance from the operating lower end thermocouples, long-term operating temperatures in excess of 2000°C had been assumed for the high-power upper ends of the rods. Such a temperature should have resulted in columnar grain growth, especially for the initially small grain fuel in rod 6.

Gamma scanning for Zr, Nb, and Cs was performed on rods 1 and 6 of IFA-432; the resulting profiles are presented in Figure 1. An anomalous depression or flattening of the gamma activity was observed (and verified in repeated scans) for rod 6. The added line on the gamma scans is a linear interpolation between the calculated local linear heat rates at the thermocouple positions over the last few months of irradiation. It would be expected that the gamma scan would nearly match the linear heat rate profile at end-of-irradiation. As seen in Figure 1a, the activity deviated 10% to 15% below the expected relative variation for the upper end of rod 6. No such deviation was seen for rod 1 (Figure 1b). Since rods 1 and 6 were adjacent in the assembly, it is difficult to imagine a sufficiently strong perturbation to the local thermal neutron flux (from control rods or neighboring absorber shields) that would affect the power production of one rod and not the other. Strong end-to-end differences in the relative fissile mass in rod 6 were ruled out from the detailed fabrication and precharacterization records. This matter is still under investigation. Migration of cesium to the pellet-pellet interfaces is definitely indicated by the series of peaks in the cesium scans for both rods. The axial temperature gradient is reflected by the axial variation of peak heights.

TASK B - DATA QUALIFICATION AND ANALYSIS

A tape containing data for the period from May 12 to August 24, 1982, was received for IFA-432 and IFA-518. The IFA-432 data have been processed; peak burnups as of August 24, 1982, were 37.6, 37.4, 38.9, and 11.7 MWd/kgM for rods 2, 3, 5, and 9, respectively. A file system was organized for the IFA-518 data, data processing routines were updated, and all IFA-518.3 data were filed.

A quick analysis of the IFA-518.3 data indicated that the internal gas pressure for reference rod R24 continues to increase and the calculated current fission gas release is 18%. Pressure data from the annular coated-cladding rods indicate significantly less fission gas release than from the reference rods (~7% release at a burnup of 15 MWd/kgM). Reference rods R23 and R25 continue to show evidence of fuel-cladding mechanical interaction but do not appear to have any permanent cladding axial elongation.

Work has continued on the PIE and final reports for IFA-432 and IFA-527. Appendices for the IFA-432 report were prepared that 1) discuss the thermocouple decalibration correction that has been applied

Figure 1. Gamma Scan Profiles for Rods 1 and 6 of IFA-432

to the temperature data and 2) present standardized power/temperature histories as a function of elapsed time at power.

TASK C - FUEL CODE MAINTENANCE AND IMPROVEMENT

The newest version of FASTGRASS was linked to V1M4 of FRAPCON-2. After receipt of a dynamically dimensioned version of FASTGRASS from Argonne National Laboratory, testing revealed that many of the variables used in the coding had been redimensioned so that they were incompatible with previous usage. This redimensioning required modifying many portions of the interface between FRAPCON-2 and FASTGRASS, often in a nonobvious manner.

Other changes to FRAPCON-2 that were in the V1M4 UPDATE include: fixing the printout for times in excess of 900 days; allowing up to 200 time steps; printing the fuel column axial expansion; printing porosity and dish volume when PELET is used; fixing the printout of coolant temperature and film coefficient; and fixing the printout of stress and strain when FRACAS-I/II are used.

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS

The second heater assembly test with nonirradiated cladding was successfully completed. An electrical heat rating of 26.9 kW/m (8.2 kW/ft) was attained, which is equivalent to 42.7 kW/m (13 kW/ft) from a diametral fuel thermal expansion standpoint. This successful test indicates that the problems associated with the failure of the first nonirradiated cladding test⁽¹⁾ have been solved.

The diametral measuring instrument was found to be too sensitive to spurious mechanical and electrical signals during the first and second heater assembly tests with nonirradiated cladding. This original instrument was designed to measure diametral changes in the test rod cladding in both the axial and azimuthal directions. A simpler instrument has been designed and is being fabricated that measures only axial variations in diameter. This device, which uses mechanical amplification of diametral changes, is expected to be less sensitive to mechanically induced vibrations. An electrical frequency cutoff has been identified; electrically induced noise can be eliminated above the cutoff by using a suitable filter; thus, significant improvement in signal stability is also expected.

Swipable beta-gamma contamination of four irradiated Zircaloy cladding tubes for the PCI simulation tests has been reduced from >10,000 cpm to ~500 cpm by ultrasonic cleaning. This relatively low swipable contamination level is expected to have a very positive effect on test assembly fabrication, handling, and subsequent PCI testing. The radiation field associated with each of the four cleaned tubes was 2 rem/h at contact and 30 mrem/h at a distance of 1 m. Although these radiation levels are of concern with respect to personnel exposure, it is expected that adequate shielding and procedures can be prepared that will permit handling and test assembly fabrication activities (except welding) to be accomplished outside a hot cell.

FUTURE WORK

Planned activities for next quarter include: continued analysis of the IFA-432 and IFA-527 PIE data; completion of the IFA-432 and IFA-527 final and PIE reports; work on a time- and space-conserving version of FRAPCON-2 (FASTCON); and work on preparation of PCI simulators using irradiated cladding.

REFERENCES

1. Edler, S. K. 1983. *Reactor Safety Research Programs - July-September 1982*. NUREG/CR-2716, Vol. 3, PNL-4275-3, Pacific Northwest Laboratory, Richland, Washington.
2. Edler, S. K. 1982. *Reactor Safety Research Programs - July-September 1981*. NUREG/CR-2127, Vol. 3, PNL-3810-3, Pacific Northwest Laboratory, Richland, Washington.

ACCELERATED PELLET-CLADDING INTERACTION (PCI) MODELING(a)

R. E. Williford, Project Manager

M. E. Cunningham

D. D. Lanning

SUMMARY

The basic structure of the fuel failure code (GT2-F) was defined, and the fuel rod transient temperature calculator was completed. A new mechanical model describing pellet-cladding interaction (PCI) induced by cracked fuel is nearly completed; the model will be implemented in the fuel failure code next quarter. Work also began on implementing a burnup-dependent pellet radial power profile model in the FRAPCON-2 code.

INTRODUCTION

This Pacific Northwest Laboratory (PNL) program is divided into two tasks with the following objectives:

- **Task A** - to complete a PCI-related fuel failure model for U.S. Nuclear Regulatory Commission (NRC) policy use, to provide an assessment of the model, and to report the results.
- **Task B** - to obtain or develop an improved burnup-dependent model for predicting pellet radial power profiles to be used in the NRC fuel behavior code FRAPCON-2.

The Task A fuel failure modeling draws upon previous work performed under FIN B2043, where cracked fuel mechanical behavior models were developed. With the addition of cladding damage and statistically oriented models, the resulting code will predict the probability of fuel rod failure under normal reactor operating conditions and for events described by Chapter 15 of the Safety Analysis Report.

For Task B, the computer code RADAR, which was developed by British Nuclear Fuels Limited (BNFL), has been obtained by PNL for use in FRAPCON-2 or related codes. RADAR calculates the changing pellet radial power profile as a function of ^{235}U burnup and ^{239}Pu production. The purpose of Task B is to verify RADAR results and implement the code in FRAPCON-2.

TECHNICAL PROGRESS

The progress that has been made in each task during this quarter is described below.

TASK A - FUEL FAILURE MODELING

The basic structure of the NRC fuel failure code was defined. The GAPCON-THERMAL-2 code will serve as a foundation for the failure code, which is temporarily labeled GT2-F. Submodels in the code will be updated and modified as necessary. The primary new additions to GT2-F are: a transient temperature calculator, a new mechanical model for cracked fuel PCI, a cladding damage model, and a statistical model to describe the probability of fuel failure. The end result will be a code that can economically calculate burnup-dependent failure probabilities.

(a) FIN: B2452; NRC Contact: H. H. Scott.

The output of the transient thermal calculator was verified for a range of transient power histories. In order to adequately calculate cladding damage, improvements were needed in the mechanical models so that PCI effects induced by cracked fuel pellets were correctly addressed. To this end, a new mechanical model (PCI2) was developed that addresses azimuthal cladding bending, radial cladding bending (ridge formation), cladding axial elongation, the number of fuel cracks, and radial relocation of the fuel fragments. The solution method is constructed so that the cracked fuel follows the stress-effective modulus path documented in Reference 1. Some components of the fuel fragment ratcheting models from Reference 2 are also used. The new mechanical model (PCI2) is currently in the final debugging stages and will be implemented in GT2-F next quarter.

TASK B - RADIAL POWER PROFILES

Work began on inserting the RADAR burnup-dependent radial power profile model into Version 1 Modification 4 of FRAPCON-2.

FUTURE WORK

The following activities are planned for the next quarter. The mechanical model (PCI2) will be completed and implemented in the failure code (GT2-F), and code performance testing will commence. Detailed work on the cladding damage model will also begin. RADAR will be implemented in FRAPCON-2, and verification of RADAR's neutronics submodels should near completion.

REFERENCES

1. Williford, R. E. December 1982. *Experimental Verification of a Cracked Fuel Mechanical Model*. NUREG/CR-2872, PNL-4377, Pacific Northwest Laboratory, Richland, Washington.
2. Williford, R. E. March 1983. *A Model for the Localized Mechanical Behavior of Cracked Nuclear Fuel Pellet Systems: Status Report*. NUREG/CR-2931, PNL-4451, Pacific Northwest Laboratory, Richland, Washington.

PIPE-TO-PIPE IMPACT(a)

M.C.C. Bampton, Project Manager

J. M. Alzheimer

F. A. Simonen

SUMMARY

The initial matrix tests were completed. Tests were run with the impacted pipes at pressurized water reactor (PWR) pressure and temperature, and oil was used as the pressurizing fluid. Model development continued for correlation of the data.

INTRODUCTION

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

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

TECHNICAL PROGRESS

The test matrix for the hot oil tests was completed. Three or four tests at different impact velocities were completed for each group as originally planned except for Groups 4a and 5a. There was no significant damage to the target pipe at the maximum catapult energy level for either of these groups. Further tests for these groups would have given no additional useful data on deformation characteristics.

The only ruptures that occurred were the two reported in the last quarterly report.⁽¹⁾ Another test with the same size pipe was run with a velocity between those that ruptured. The pipe did not rupture even though the surface had numerous deep cracks. The deformation of the specimens was very near that expected from the previous tests.

The Group 5b tests (3-in. diameter target pipes) were modified because of the large deformations that occurred. For the first test from the group, the supports were 108 in. apart. However, to keep the pipes from impacting the ground, the supports were moved to 84 in. and 42 in. apart, respectively, for the next two tests. For the test with supports 42 in. apart, the pipe bent at the point of impact and at the inside edge of each support.

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

Since completion of the hot oil tests, various methods have been investigated for correlating the data and for developing analytical models. Hopefully models will be developed that will allow the extension of available data to other impact energy levels and other pipe sizes. New criteria for determining when pipe-to-pipe impact must be avoided will result from these models.

FUTURE WORK

The following activities are planned for the next quarter:

- continue development of analytical models
- determine scope of additional tests and prepare for testing.

REFERENCES

1. Edler, S. K. 1983. *Reactor Safety Research Quarterly Report - July-September 1982. NUREG/CR-2716, Vol. 3, PNL-4275-3, Pacific Northwest Laboratory, Richland, Washington.*

**SEVERE CORE DAMAGE SUBASSEMBLY
PROCUREMENT PROGRAM**

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

R. L. Goodman, Project Manager

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

SUMMARY

The Pacific Northwest Laboratory (PNL)/Power Burst Facility (PBF) SFD-2 fuel bundle and insulated shroud assemblies were completed and delivered on schedule to the Idaho National Engineering Laboratory (INEL) on December 1, 1982. All design drawings for the PNL/PBF SFD-3 and SFD-4 fuel bundle and insulated shroud assemblies were approved and released for fabrication activities in early October 1982. The final formal design review of the SFD-3 and SFD-4 fuel bundle and insulated shroud assemblies was held in joint PNL/EG&G Idaho, Inc., meetings in Idaho Falls, Idaho, on October 28 and 29, 1982. The PNL design presentations were well received by EG&G, and only two action items resulted from the final design review meetings. All materials and machined components for the fuel bundle and insulated shroud assemblies for both the SFD-3 and SFD-4 test train assemblies have been completed and are now in-house and awaiting quality assurance receiving inspection.

INTRODUCTION

The Severe Core Damage (SCD) Subassembly Procurement Program includes the design, development of appropriate materials and supporting fabrication processes, and complete fabrication of fully instrumented test train assemblies for the U.S. Nuclear Regulatory Commission (NRC)-sponsored test program at the PBF. Many portions of the PBF work should directly benefit the ESSOR and National Research Universal (NRU) reactor programs due to similarities in the experimental objectives, particularly for materials development, instrumentation, and fabrication development. The program is designed to yield important experimental data related to fuel and cladding behavior during small-break accidents as well as to provide information on the postaccident coolability of damaged fuel rod clusters after such accidents.

FUTURE WORK

- complete the insulated shroud assembly for the SFD-3 test train assembly
- assemble the fuel rods for the SFD-3 and SFD-4 fuel bundle assemblies
- begin assembly of the SFD-3 fuel bundle assembly.

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

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

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

SUMMARY

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

INTRODUCTION

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

FUTURE WORK

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

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

SEVERE CORE DAMAGE SUBSYSTEM PROCUREMENT PROGRAM ESSOR FUEL DAMAGE TEST PROGRAM (SUPPORT)

E. F. Finko, Project Manager
J. W. Lipton, ESSOR Site Representative

SUMMARY

The ESSOR Fuel Damage Test Program (SUPPORT) is a representative of the ESSOR Fuel Damage Test Program (SUPPORT) and is a representative of the ESSOR Fuel Damage Test Program (SUPPORT).

INTRODUCTION

The ESSOR Fuel Damage Test Program (SUPPORT) is a representative of the ESSOR Fuel Damage Test Program (SUPPORT) and is a representative of the ESSOR Fuel Damage Test Program (SUPPORT). The ESSOR Fuel Damage Test Program (SUPPORT) is a representative of the ESSOR Fuel Damage Test Program (SUPPORT) and is a representative of the ESSOR Fuel Damage Test Program (SUPPORT). The ESSOR Fuel Damage Test Program (SUPPORT) is a representative of the ESSOR Fuel Damage Test Program (SUPPORT) and is a representative of the ESSOR Fuel Damage Test Program (SUPPORT).

FUTURE WORK

The main effort on this project is currently being provided by the ESSOR Fuel Damage Test Program (SUPPORT) and is a representative of the ESSOR Fuel Damage Test Program (SUPPORT). The ESSOR Fuel Damage Test Program (SUPPORT) is a representative of the ESSOR Fuel Damage Test Program (SUPPORT) and is a representative of the ESSOR Fuel Damage Test Program (SUPPORT).

CORE THERMAL MODEL DEVELOPMENT(a)

M. J. Thurgood, Project Manager

J. M. Kelly
T. E. Guidotti
D. E. Debellis
S. H. Bian

SUMMARY

Documentation of the COBRA/TRAC code was completed during the past quarter. The five-volume set will be printed and distributed as NUREG/CR-3046. Models were added to COBRA-TF under the hot bundle program, and additional developmental assessment of COBRA-NC was performed.

INTRODUCTION

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

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

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

The code features flexible noding, which allows modeling of such complex geometries as slotted control rod guide tubes, jet pumps, and core bypass regions. These geometries cannot be easily modeled in regular Cartesian or cylindrical mesh coordinates; however, since they have significant impact on the thermal-hydraulic response of the system, these geometries must be modeled with reasonable accuracy.

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

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

The fuel rod heat transfer model uses a rezoning mesh to reduce the rod heat transfer mesh size automatically in regions of high heat flux or steep temperature gradients and to increase the mesh size in regions of low heat flux. This model has proven very effective in resolving the boiling curve in the region of the quench front.

TECHNICAL PROGRESS

COBRA APPLICATIONS

The documentation of COBRA/TRAC was completed this quarter. In addition, advances were made toward a small-break LOCA analysis capability by development of a one-dimensional (1-D) PWR/UHI model and a choked flow model.

COBRA/TRAC Documentation

The COBRA/TRAC code documentation⁽¹⁾ was sent to the NRC for printing and distribution. Volume 1 of the 5-volume report describes the basic three-field conservation equations and constitutive models used in the vessel component (COBRA-TF). Volume 2 contains the finite-difference equations for the vessel and describes the numerical techniques used to solve these equations and the coupling between the TRAC-PD2 equations and the COBRA-TF vessel equations. Volume 3 is the Users' Manual, which consists of line-by-line input instructions for COBRA/TRAC and guidance for using the code. Volume 4—the Applications Manual—contains the results of simulations run to assess the performance of the code. Volume 5 is a Programmers' Guide.

PWR/UHI 1-D Model

A 1-D version of the best-estimate PWR/UHI input model was derived by collapsing the 3-D input model. The 3-D model had 556 hydrodynamic cells, 10 nuclear fuel rods, and 249 heat slabs; the 1-D model uses 66 cells, 2 rods, and 26 slabs to model the same geometry. In addition, the 1-D version combines the three intact loops into one modeled loop.

Checks are being performed to assure that the two versions use exactly the same values for volume, metal mass, power, loss coefficients, etc. The same version of COBRA/TRAC used in the 3-D calculation will be used in the 1-D calculation. Any differences that are noted between the calculations can be attributed to nodalization and not to model differences. The 1-D calculation will be completed next quarter.

Choked Flow Model

Analytical modeling of choked flow for the COBRA/TRAC code will improve computing efficiency by avoiding fine mesh noding and improving the convergence properties.

The choked flow model from TRAC-PF1⁽²⁾ was used in COBRA/TRAC; this model is basically identical to that used in RELAP5.⁽³⁾ It was chosen because of its proven capability to simulate blowdown transients.

The analytical model was developed by Ransom and Trapp⁽⁴⁾ and is based on a two-fluid model with thermal equilibrium between vapor and liquid. The choked velocity is derived from four conservation equations—one mass conservation for the mixture, two momentum equations for both vapor and liquid phases, and one entropy balance equation for the mixture. These equations may be written in a characteristic form by setting source terms to zero:

$$A \left(\frac{\partial \bar{u}}{\partial t} \right) + B \left(\frac{\partial \bar{u}}{\partial x} \right) = 0$$

By solving this equation, the perturbation wave velocities (choked velocities) may be obtained.

In subcooled choking, where the fluid changes from subcooled single phase to two phase at the discharge throat, the choking velocity can be calculated by applying Bernoulli's equation. For the case where the transition to a two-phase mixture occurs upstream of the throat, the choked velocity is identical to that obtained from the two-phase choked model mentioned previously. To determine which model is used, both choked velocities are calculated; and the larger was used as the choking velocity to be imposed at the throat.

To evaluate the model, COBRA/TRAC with the choked flow option was used to simulate the Chen blowdown test.⁽⁵⁾ Data from the tests revealed that no partial vaporization occurred in the upstream neighborhood of the apertures even though the local static pressure of the fluid dropped significantly below the vapor pressure. Water remained in a superheated metastable liquid state.

The apertures selected have the largest diverging section (60° divergence) so that they can be closely simulated by the code, which allows the water to be discharged into the ambient pressure immediately at the throat.

In Figure 1 the calculation and the data are compared for the Aperture 7 and 8 tests. Both apertures have a 1/2-in. inside diameter (ID) opening but different discharge lengths. The code-calculated mass fluxes are lower than the measured values; however, as the pressure increases (up to 200 psia in the test), the differences decrease.

Figure 1. Comparison of Calculation and Data for Aperture 7 and 8 Tests

Flow separation can occur in certain situations especially for sharp-edged openings. The flow area was reduced 5% in the RELAP5 choked flow calculation⁽⁴⁾ to account for this phenomenon. If the same approach were used in this work, the flow would be reduced by ~5% (because the choked flow is in a subcooled regime); and the comparison between calculation and measurement would be closer.

In summary, comparison between the COBRA/TRAC calculation and the Chen blowdown test indicate that the code can provide reasonably good predictions of discharge flow rate under a subcooled choked condition. The calculations also proved the capability of using a coarse noding at the break location instead of the fine noding used previously. More testing is needed to qualify the two-phase choked flow model in the code; therefore, these models were not included in the current (cycle 13) release of COBRA/TRAC.

HOT BUNDLE CODE DEVELOPMENT

The objectives of the hot bundle version of COBRA-TF are to:

- perform best-estimate coupled thermal-hydraulic/rod deformation calculations
- perform detailed subchannel calculations to predict the effect of flow blockage upon reflood heat transfer.

Coupled thermal-hydraulic/rod deformation calculations require models for both blockage heat transfer and fuel rod deformation. The blockage heat transfer models will be simplified versions of those developed in Task 2. In addition, a statistical/empirical model to predict the onset and extent of flow blockage will be developed and implemented in the code. This approach replaces the original task of incorporating FRAP-T6⁽⁶⁾ into COBRA-TF as the rod deformation model. After development is complete, assessment will be performed using the NRU⁽⁷⁾ materials test data.

A joint project (Task 2) is being conducted with the FLECHT-SEASET⁽⁸⁾ program to address the Appendix K steam cooling/flow blockage rules. This program consists of the following tasks:

- modify COBRA-TF to predict low flooding rate/high temperature reflood tests
- assess against 161- and 21-rod and FLECHT-SEASET unblocked bundle data
- implement flow blockage models
- assess against 21-rod FLECHT-SEASET blocked bundle and FEBA⁽⁹⁾ data
- modify flow blockage models
- fix models and perform assessment with FLECHT-SEASET 163-rod data.

During this quarter, efforts have concentrated on modifying COBRA-TF; the following models were added to the code:

- thermal radiation to fluid and structures
- azimuthal conduction in heater rods
- grid spacer rewet
- grid spacer loss coefficient
- droplet breakup at grid spacer
- two-phase enhancement for dispersed flow heat transfer.

Refinements to the models for interfacial shear and heat transfer for the dispersed flow and transition/froth flow regimes were also made.

COBRA-NC CONTAINMENT APPLICATION

Further assessment of COBRA-NC was conducted this quarter by post-test calculations of two Battelle-Frankfurt model containment pressurization tests—C-13 and C-15. The facility consists of nine rooms with a total volume of $\sim 20,000$ ft³. Blowdown of high pressure, subcooled water was initiated in room 1 (central-most room) by rupture of burst diaphragms located on the two high-energy lines ending in room 1. The only difference between tests C-13 and C-15, aside from differences in the blowdown histories, was that the flow area between rooms 1 and 3 was halved in test C-15.

A lumped parameter model with 16 channels in four sections was formulated to represent the facility, and 22 unheated flat plate conductors were used to model the concrete and metal surfaces available for condensation. Blowdown into room 1 was achieved by establishing cell boundary conditions on the proper channel for mass flow, pressure, and mixture enthalpy versus time.

Code predictions of tests C-13 and C-15 were good for both the short (2 s) and long (40 s) time intervals. Figures 2 and 3 show the long-term pressurization response of room 1 calculated by COBRA-NC for tests C-13 and C-15, respectively. The data agree within 12% for test C-13 and 8% for test C-15. The increased peak pressure incurred in test C-15 due to the decreased flow area between rooms 1 and 3 is evident. In the latter portion of the simulation, however, the break room responses for both tests are almost identical, indicating that dynamic effects lose predominance as blowdown ends. The pressure decay rate calculated by COBRA-NC after blowdown ends is less than that shown by the data. This is the result of insufficient steam being condensed by the model. Possible reasons for the low-pressure decay rate could be either 1) a deficiency in the wall heat transfer modeling of the code or 2) heatup of the heat slabs used for the wall surfaces. An assumed average wall thickness was used in the model since this information was not available in the data reports. Consequently, some or all of the heat slabs representing the walls could be inadequately thick.

Figure 2. Long-Term Response for Room 1 in Test C-13

Figure 3. Long-Term Response for Room 1 in Test C-15

FUTURE WORK

During the next quarter, cycle 13 of COBRA/TRAC will be released to the Argonne code center after final assessment calculations are made on the "frozen" version. The 1-D PWR/UHI calculation will be completed and work will begin on a PWR calculation without UHI. Models in the hot bundle version of COBRA-TF will be refined to improve the prediction of low flooding rate reflood data, and assessment against the 21-rod unblocked bundle data will be made. In addition, documentation of COBRA-NC will be completed and a jet impingement test will be simulated.

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LOCA SIMULATION IN THE NRU(a)

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SUMMARY

The design was completed for the first 21-rod coolant boilaway and damage progression test—materials test 6 (MT-6), and the revisions to the Safety Analysis Report (SAR) were approved by Atomic Energy of Canada Ltd. (AECL). Fabrication techniques were developed for installation of the high-density ZrO_2 insulation material in the test train. Due to budget restraints, fabrication of the MT-6 test train was deferred and resources were concentrated on the full-length source term (FLST) test train design. The FLST design uses the high-density ZrO_2 insulation in a 12-rod configuration for testing up to 3200°F. For testing from 3600 to 5000°F, a thoria insulation material is being developed.

Instrumentation leads will be brought into the test train from the bottom (cool) end of the test loop. The time domain reflectometry (TDR) liquid level detector was tested in the laboratory with the seal at the bottom (under water) and gave reliable indications of water level during simulated reflood. Design work was initiated on a bypass coolant system for controlling the loop pressure tube temperature during the high-temperature tests.

INTRODUCTION

The Loss-of-Coolant Accident (LOCA) Simulation in the NRU Program is being conducted in the National Research Universal (NRU) reactor at Chalk River Nuclear Laboratories (CRNL), Chalk River, Ontario, Canada, by Pacific Northwest Laboratory (PNL). The program is sponsored by the U.S. Nuclear Regulatory Commission (NRC) to evaluate the thermal-hydraulic and mechanical deformation behavior of a full-length, 3% enriched light-water reactor (LWR) fuel rod bundle during the heatup, reflood, and quench phases of a LOCA. The tests are designed to give information on the quench-front velocities within a fuel bundle, the liquid entrainment [10 CFR 50, App. K (Sec. 1D 2)], and the heat transfer coefficients [10 CFR 50, App. K (Sec. 1D 5)] for LWR fuel as a function of reflood rate and delay time before reflood starts. A series of thermal-hydraulic and materials deformation tests has been successfully completed using full-length fuel bundles over temperature ranges of 1400 to 2000°F. The program is now being expanded to include one series of FLST tests at 2600 to 3200°F and another series of tests at 3200 to 5000°F to support the severe fuel damage test program.

TECHNICAL PROGRESS

The design package for the MT-6 test train was completed. Fabrication techniques for handling and installing the high-density ZrO_2 insulation were developed. Revisions to the MT-6 SAR were approved by AECL.

Due to budget constraints, it was decided to defer fabrication of the MT-6 test train and concentrate resources on the design of the FLST test train. The FLST design will use the high-density ZrO_2 insulation in a 12-rod configuration for tests up to 3200°F. The design will be modified to use a thoria liner and thoria insulation in a 12-rod configuration for tests up to 5000°F (see Figure 1). A pressurized

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

Figure 1. Cross Section of FLST Test Train

bypass coolant system will be designed to cool the shroud/pressure tube interface at these higher temperatures. Revisions to the SAR required for the FLST were identified and discussed with CRNL. The use of irradiated rods and control rod materials is being considered for later tests in the FLST series.

Hydrogen monitoring and fission product monitoring/capture system requirement are being defined. A development program was initiated to devise techniques to fabricate thoria fibers for insulation.

FUTURE WORK

The FLST design will be accelerated, and development work on the thoria fibers will continue. Revisions to the SAR for the FLST series will be prepared and coordinated with CRNL. Long lead items will be ordered. The computer for the FLST loop control and hydrogen monitoring will be ordered, and control loop/safety software development will be initiated. The loop bypass coolant system will be designed.

bypass coolant system will be designed to cool the strip pressure tube inlets at these higher temperatures. Revisions to the SAR required for the FTSI were identified and discussed with CNSC. The use of insulated rods and control rod materials is being considered for later tests in the FTSI series.

Hydrogen monitoring and fission product monitoring capture system requirement are being defined. A development program was initiated to develop techniques to facilitate the design for insulation.

FUTURE WORK

The FTSI design will be accelerated, and development work on the FTSI will continue. Revisions to the SAR for the FTSI series will be prepared and coordinated with CNSC. Long lead items will be ordered. The computer for the FTSI loop control and hydrogen monitoring will be ordered, and control loop test software development will be initiated. The loop bypass coolant system will be designed.

STEAM GENERATOR GROUP PROJECT(a)

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SUMMARY

Task 6 (channel head decontamination) was completed. Two subcontractors were selected from among six bidders, and each decontaminated half of the channel head using proprietary wet chemical processes. Each operation was monitored with several radiation measuring techniques. Decontamination factors (DFs) of 6 to 7 were achieved in the channel head space (tube "shine" still contributed significantly), and surface contamination of the stainless steel was reduced by a factor of more than 30. A computer facility was completed; the data acquisition computer system will be installed after final shakedown tests. Minimum diameters were measured in 96 unplugged tubes that were selected to represent the condition of the steam generator tube bundle. Probe sizes for the base-line eddy-current in-service inspection (ISI) may now be selected. Bids that were received in response to the second tube unplugging (Task 8) RFP were again not completely acceptable. Additional information was requested from the bidders; it is currently being evaluated. Several options are being considered to begin Task 8 in March 1983.

INTRODUCTION

The Steam Generator Group Project (SGGP), which was initiated in January 1982, is a continuation and expansion of the Steam Generator Tube Integrity Program (SGTIP): a multiphase, multitask laboratory program conducted at Pacific Northwest Laboratory (PNL). Under the SGTIP, mechanically and chemically produced defects were placed in steam generator tube lengths to simulate service degradation. Specimens with defects were then nondestructively characterized and destructively tested to determine the remaining integrity under burst or collapse failure modes. Constitutive equations were subsequently established relating defect morphology and severity to remaining tube integrity. Other SGTIP objectives included studying the reliability and accuracy of nondestructive flaw characterization by eddy-current testing of steam generator tubes. Experiments were, and are currently being, conducted to determine the consequences of tube failure in terms of leak rate. The stability of through-wall flaws is also a consideration in these experiments.

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

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

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

ocean-going barge through the Panama Canal from Virginia to Hanford, Washington, where it was installed in a specially designed containment facility, the Steam Generator Examination Facility (SGEF). The SGEF is equipped for both nondestructive examination (NDE) and physical sectioning of the generator and includes capabilities to perform chemical cleaning and decontamination.

Because of the potentially unique opportunities presented by this generator, a broadened research program has been developed that is of interest and need to U.S. and foreign government agencies, private organizations, and vendors. Data will be obtained from service-induced specimens to help establish and support licensing and regulatory requirements for operating nuclear steam generators. Potential exists for research and development in chemical cleaning, decontamination, corrosion product identification, corrosion mechanism studies, and repair techniques under alternate sponsors. Consortia of government and private organizations from Italy, France, Japan, and the United States have joined with the U.S. Nuclear Regulatory Commission (NRC) in program sponsorship of the SGGP. Research efforts on the Surry II generator will emphasize the following areas and will help establish and support regulations and guidelines in these areas:

- validation studies of primary side NDT techniques and instrumentation
- verification of remaining integrity of service-defected steam generator tubes
- assessment of the secondary support structure integrity
- health physics - ALARA control of radiation exposure associated with maintenance, repair, ISI procedures, and waste handling
- defect matrix profiling and identification
- long-term operating effects of secondary side cleaning and primary side decontamination
- nondestructive ISI of the secondary side
- development and testing of innovative NDE devices and techniques
- demonstration of proposed repair techniques with assessment of reliability and safety aspects of the repair
- demonstration of decontamination and cleaning methods; and assessment of effectiveness, waste generation, and potential tube or support structure damage.

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

TECHNICAL PROGRESS

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

STEAM GENERATOR GROUP PROJECT

The SGGP was initiated on January 11, 1982, when the Surry IIA steam generator was placed in the SGEF. Consortia from Italy, France, and Japan have formally become program participants; and the Electric Power Research Institute (EPRI) has joined the program. A Technical Advisory Group (TAG) consisting of representatives of these organizations met at Cadarache, France, on October 20-22, 1982, to review program accomplishments and plan future activities.

Health Physics (Task 3)

The health physics task provides procedures for personnel exposure monitoring, control, and documentation. Health physics research activities include radiologic mapping,⁽²⁾ determination of the

effectiveness of decontamination efforts, and evaluation of waste and waste disposal problems associated with various operations. Previous efforts included exposure control of the steam generator move into the SGEF. Radiation control procedure preparation and review were conducted for initial research efforts involving reopening of preshipment penetrations through the steam generator shell (Task 5) and entering the primary side through the channel head manways in preparation for Tasks 6, 7, 8, and 9. The channel head region had radiation levels of 4 to 5 rem/h prior to decontamination (Task 6). After decontamination, radiation levels measured with thermoluminescent detectors (TLDs) fell to 0.4 to 0.7 rem/h. Figures 1 through 6 show initial and final radiation readings on each side. Other methods of radiation measurement were also used and confirmed the TLD results. These methods included: 1) external monitoring of the channel head with shielded NaI and hand-held Ge detectors and 2) gamma spectroscopy of contaminated corrosion specimens with a GeLi detector.

Table 1 shows accumulated personnel radiation exposures by quarters for 1982. Major activities during each quarter are shown for a rough correlation.

Statistics - Data Acquisition, Management, and Analysis (Task 4)

Software was developed to interface the computerized data acquisition system with nondestructive testing (NDT) (mainly eddy-current) devices that provide data input. In addition, a remotely operated, computer-controlled NDT probe pusher-puller was assembled and tested. This PNL-designed pusher-puller will automatically provide NDT probe position information to the computer acquisition system in parallel with normal NDT data. The system is completed and undergoing final shakedown tests.

Figure 1. Initial Radiation Readings for Cold Leg (rem/h)

Figure 2. Final Radiation Readings for Cold Leg (rem/h)

Figure 3. Change in Radiation Readings for Cold Leg (rem/h)

Figure 4. Initial Radiation Readings for Hot Leg (rem/h)

Figure 5. Final Radiation Readings for Hot Leg (rem/h)

Analysis and compilation of historical ISI eddy-current tapes and other historical operating data are almost completed. The historical data base includes ISI data, water chemistry information, and operating records from Westinghouse, Virginia Electric & Power Company, and the NRC. Experimental data from Tasks 5 and 6 are being input to the system. Plots and analyses of chemistry data are being made to assess functional relationships between several chemistry parameters.

A computer facility in a trailer adjacent to the SGEF was completed. It has a self-contained temperature and humidity control system and a Halon fire prevention system. The computer will be installed in the facility early in 1983.

Figure 6. Change in Radiation Readings for Hot Leg (rem/h)

Table 1. Steam Generator Personnel Radiation Exposures for 1982

Personnel	Total Exposure, mrem				Total
	Jan-Mar	Apr-Jun	July-Sept	Oct-Dec	
PNL engineers and technicians	2,472	831	4,351	4,151	11,805
PNL crafts personnel	4,125	975	2,933	629	8,662
Visitors	5	79	93	0	177
Subcontractor personnel	0	0	1,620	634	2,254
Total	6,602	1,885	8,997	5,414	22,898

Note: The following major tasks took place in each of the above quarters:

Jan-Mar Move steam generator into SGEF.
Reopen and inspect preshipment openings.

Apr-Jun Inspect secondary side of steam generator.

July-Sept Decontaminate channel head cold leg.

Oct-Dec Decontaminate channel head hot leg.
Profilometer 96 tubes.

Decontaminate Channel Head (Task 6)

The channel head was decontaminated to minimize radiation exposure during primary side access during future tasks. The availability of the channel head was offered as an opportunity for research-related development/demonstration of decontamination techniques. Two subcontracts were let; each subcontractor decontaminated half of the channel head. The cold leg side was decontaminated by

London Nuclear Services, Inc., using a proprietary Canadian process. The hot leg side was decontaminated by Quadrex Corporation using another proprietary process that was developed in Great Britain. Both operations were successful in reducing radiation levels by factors of 6 to 7. Most of the remaining radiation in the channel head is due to "shine" from the tubes above. Surface contamination on the stainless steel was reduced by a factor of 30.

Base-Line Eddy-Current ISI (Task 7)

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

In order to select eddy-current probes in a proper array of sizes to perform the ISI, diametric measurements of a statistically selected sample of 96 steam generator tubes were made. A 0.600-in. diameter eddy-current probe was passed through both ends of each unplugged tube. A Zetec Model MIZ-15 eddy-current apparatus measured the signals. Most of the denting was detected on the hot leg or inlet sides of the tubes. Three tubes were found to be obstructed (<0.600 in.) at the first, third, and fourth support plates, respectively, prohibiting evaluation beyond those points. Eleven readings below 0.700-in. were obtained, and two tubes were found to be free of denting.

On the cold leg side, no obstructions were encountered. Only three measurements were below 0.740-in. (nominal tube inside diameter is 0.775-in.). Several probes will be required to accommodate the range of diameters observed on the hot leg side. The distribution of denting in the cold leg side was minimal. Based on these results, we expect to find very few problems in the cold leg, and full-size probes with adequate fill factor can be applied in almost all cases. The hot leg tubes may require at least two separate examinations. Based on the concentration of deformations at the second support plate of the inlet side, access to the hot leg may have to be gained by passing a flexible probe from the outlet side over the bend and down into the upper reaches of the hot leg on occluded tubes.

Tube Unplugging (Task 8)

An RFP to remove the plugs from 461 of the 748 plugged tubes was issued. Unplugging the tubes will increase the availability of defects for correlation with historical ISI data. Responses to a fixed price RFP were nonresponsive. Two bidders have responded to a second RFP on a time and materials basis; however, some of the conditions were unacceptable. Additional information has been received from the bidders and is being evaluated. Several options are being considered to accomplish the task within available funding and as close as possible to the original schedule.

Secondary Side Access (Task 10)

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

Tube Sheet Section Removal (Task 11)

A large section of the tube sheet will be removed for analysis of operating effects on the tube sheet, especially at the tube-tube sheet crevices. The section to be removed has been identified as rows 7 to

17 and columns 70 to 94. The tubing included in the section plus 6-in. lengths above the tube sheet upper surface will be removed along with the section. Cutting the large shell opening to accomplish this task is being done jointly with Task 10.

Technical Advisory Group (TAG)

Task Action Plans for all currently active tasks were presented to the TAG meeting in Cadarache, France, on October 20-22, 1982. Results of both decontamination operations were also presented. Strong interest was expressed in Task 4 because of a need to computerize data at all operating plants, and there was some concern that Task 11 will be very expensive. Perhaps Task 11 should be undertaken only if Task 10 uncovers some form of tube sheet damage. The next TAG meeting is scheduled for the week of April 11, 1983, at EPRI in Palo Alto, California.

PHASE II - STEAM GENERATOR TUBE INTEGRITY PROGRAM

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

MILESTONES

- Formal SGGP participation was approved by Italy, France, Japan, and EPRI.
- Task 5 (reopen preshipment penetrations) was reported.
- Task 6 (decontaminate channel head) was completed on both sides.
- 96-tube profilometry test was completed.

FUTURE WORK

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

- select the contractor for and begin the tube unplugging task
- issue topical report on Task 5
- issue topical report on Task 4a
- continue secondary side inspections
- issue topical report on the eddy-current round robin
- issue topical report on Task 6
- activate data acquisition computer system.

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16. ABSTRACT (200 words or less) <p> This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1982, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Evaluations of nondestructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining the strength of structural graphite, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the integrity of pressurized water reactor steam generator tubes where service-induced degradation has been indicated. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Core thermal models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Fuel assemblies and analytical support are being provided for experimental programs at other facilities, including loss-of-coolant accident simulation tests at the NRU reactor, Chalk River, Canada; the ESSOR reactor Super Sara Test Program, Ispra, Italy; the instrumented fuel assembly irradiation program at Halden, Norway; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. </p>					
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