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Reactor Safety Research Programs

**Quarterly Report
April-June 1983**

Prepared by S. K. Edler, Ed.

**Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute**

**Prepared for
U.S. Nuclear Regulatory
Commission**

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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 April 1 through June 30, 1983, 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. Experimental data and validated models are being used to determine a method for evaluating the acceptance of welded or weld-repaired stainless steel 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. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada; and an instrumented fuel assembly irradiation program is being performed at Halden, Norway. Fuel assemblies and analytical support are being provided for experimental programs at other facilities, including fuel rod deformation and severe fuel damage tests for the Super Sara Test Program, Ispra, Italy; 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

Work on this program was stopped April 23, 1983, due to lack of funding.

INTRODUCTION

The Graphite NDT Research Program was a continuation of previous work at Pacific Northwest Laboratory (PNL) that 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 work for this project was to:

- Modify the eddy-current probe design and obtain calibration constants to permit measurement of oxidation profiles in prototypic high-temperature gas-cooled reactor (HTGR) core support post graphite.
- Continue development of ultrasonic backscattering techniques for evaluation of in-depth oxidation profiles in HTGR core support components.

The objective of this investigation was to demonstrate the feasibility of NDT techniques for in-service monitoring of structural graphite strength in an HTGR.

TECHNICAL PROGRESS

The sweep-frequency eddy-current system was completed with programming to process data at 40 discrete frequencies. The laboratory tests showed good correlation between physical measurements of the oxidation and the eddy-current response. A billet of Stackpole grade 2020 graphite was received from Oak Ridge National Laboratory and samples to be used as oxidation standards were being prepared; the samples were to be used to provide calibration constants (used in the ZFIT computer algorithm) required for processing data from the component flow test loop. Checkout of the new broad-band ultrasonic transducers was postponed pending receipt of anticipated funding.

Work on this program was stopped on April 23, 1983, when PNL learned that the anticipated funding would not be forthcoming. At that time, charges for work on the program had already exceeded the authorized funds.

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

GRAPHITE NONDESTRUCTIVE TESTING (NDT) RESEARCH(a)

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SUMMARY

Work on this program was stopped April 23, 1993, due to lack of funding.

INTRODUCTION

The Graphite NDT Research Program was a continuation of previous work at Pacific Northwest Laboratory (PWL) that 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 new, surface oxidation profiles can be determined from multiwave velocity-current measurements; and 3) the technical feasibility of determining oxidation profiles at greater depths by ultrasonic backscattering techniques. The scope of work for the project was to:

- Modify the eddy-current probe design and obtain calibration constants to permit measurement of oxidized profiles in prototype high-temperature gas-cooled reactor (HTGR) core support post graphite.
 - Continue development of ultrasonic backscattering techniques for evaluation of in-depth oxidation profiles in HTGR core support components.
- The objective of this investigation was to demonstrate the feasibility of NDT techniques for in-service monitoring of structural graphite components in an HTGR.

TECHNICAL PROGRESS

The sweep-frequency eddy-current system was completed with programs to produce data at 40 kHz cycle frequencies. The laboratory tests showed good correlation between depth of measurement of the oxidation and the eddy-current response. A series of six graphite rods 50.8 mm in diameter were received from Oak Ridge National Laboratory and samples to be used as oxidation standards were being prepared. The samples were to be used to provide calibration constants for the ZET computer. Significant progress was made in the development of the HTGR core support post test loop. The design of the new broad-band ultrasonic transducer was postponed pending receipt of anticipated funding.

Work on this program was stopped on April 23, 1993, when PWL learned that the anticipated funding would not be forthcoming. At that time, changes for work on the program had already exceeded the authorized budget.

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

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SUMMARY

Cyclic fatigue testing of the ZB-1 intermediate-scale vessel continued at 290°C, and about one-half of the high-temperature portion of the test matrix has been completed. The test has progressed much more slowly than planned due to test loop problems. Progress towards pattern recognition algorithm retraining has been made as the result of additional waveform analysis. A piece-wise linear decision rule appears to give better classification results. Acoustic emission (AE) monitoring equipment has been installed at the Watts Bar 1 reactor to monitor hot functional testing. Reactor background noise showed a marked reduction with increasing temperature. Tests performed on a pressurized water reactor (PWR) vessel gave encouraging results for detecting AE from belt-line cracking.

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

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

TECHNICAL PROGRESS

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

OFF-REACTOR VESSEL TEST

The 70°C portion of the ZB-1 vessel test consisted of seven steps;⁽¹⁾ the 290°C portion consists of:

- Step 8 - 2000 pressure cycles (100 to 250 bar)
- Step 9 - 4000 pressure cycles (150 to 250 bar)
- Step 10 - hydrostatic test to 290 bar at 70°C
- Step 11 - 2000 pressure cycles (100 to 250 bar)
- Step 12 - 4000 pressure cycles (150 to 250 bar)
- Step 13 - hydrostatic test to 350 bar at 70°C.

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

Step 8 and the first 3000 cycles of Step 9 have been completed. The testing is proceeding more slowly than anticipated due to problems with the test loop. The high-temperature testing imposes a severe strain on the test loop.

Some initial results from the last series of 70°C cyclic tests (Steps 5 and 6) are given in Figures 1 through 4. Source location plots for the acoustic data received in the vicinity of the A533B patch (denoted by solid lines) are shown in Figures 1 and 2. The dashed lines in the figures represent "accept zones" that are centered around each flaw; the zones are created during analysis by establishing source location filters to separate the acoustic events by defect. All subsequent manipulations of the data are then performed on subsets of the total data that correspond to each individual defect. For example, Figure 3 gives the AE count versus cycle count for events received in the zone containing flaw B for Step 5. Although the selection of the zone boundaries is somewhat arbitrary, the boundaries generally represent the accuracy of the source location algorithm. In addition, no zone overlapping is allowed.

Several test conditions are represented in Figure 4. Near the middle of Step 6, the flow noise simulator was turned on for 1000 cycles; and the detection threshold was raised from 1.5 to 3.0 V. Analysis of the data received during this time period was reported in the previous quarterly.⁽¹⁾ In addition, the minimum pressure was raised from 144 to 168 bar during the last 2300 load cycles to investigate the influence of high R-ratio (and correspondingly low ΔK) on AE behavior. A decrease in activity is evident from the data in Figure 4 and is probably due to the decrease in crack growth rate caused by the lower stress intensity factor range.

FIGURE 1. AE Source Indications for Step 5 of ZB-1 Vessel Test

FIGURE 2. AE Source Indications for Step 6 of ZB-1 Vessel Test

FIGURE 3. Acoustic Emission Versus Load Cycles for Flaw B in Step 5 of ZB-1 Vessel Test

FIGURE 4. Acoustic Emission Versus Load Cycles for Flaw B in Step 6 of ZB-1 Vessel Test

An initial test of the AE/flaw severity relationship has been applied to the data from flaw B. Using AE versus cycle curves, AE rates were estimated for 25% of the load cycles in each of the 70°C cyclic pressurization steps. These rates were used in conjunction with the lower-bound flaw severity relations developed from laboratory data to estimate crack growth rates (μ in./s). These rates are given in Table 1 along with crack growth rates computed from $da/dN-\Delta K$ curves for A533B steel specimens tested under similar loading and conditions and from flaw depth estimates obtained from crack opening displacement (COD) gauges.

The results of these calculations indicate that crack growth rates determined from AE data are consistently higher than those obtained from fracture mechanics estimates. This result agrees with the lower-bound analysis concepts developed from laboratory data.

TABLE 1. Fatigue Crack Growth Rates for Flaw B

Step	AE Data, μ in./s	$da/dN-\Delta K$, μ in./s	COD Data, μ in./s
2	2.6	1.4	0.8
3	1.2	0.4	0.4
5	3.2	1.9	1.7
6	1.6(a)	0.5	0.5

(a) One period of high data rate produces an estimated value of 4.1.

Additional pattern recognition analysis has been performed in an attempt to retrain the algorithm for use on the test. To date, five data tapes have been analyzed: CY5 (Step 2), CY7 (Step 3), C17 (Step 6), C36 Tape 4 (Step 9), and a tape of noise from a manhole gasket. The first retraining was performed using data from tapes CY7 and C17. From C17, 45 signals were available that had been screened by the zone isolation system and all 45 signals were plotted (two signals were set aside as electric transients). Since the zone isolator accepted data from only flaw B, all signals were assumed to originate from that flaw.

Exploratory analysis of data from C36 Tape 4 showed four distinct data clusters; one of these clusters was quite tight. A cluster analysis of the two most important features—the natural logarithm of moment 0 of the power spectrum and the value of the autocorrelation function at time lag 37—produced the clusters depicted in Figure 5. A plot of load position versus tag number (Figure 6) with the same labels shows that the tight cluster and its nearest neighbor are in the correct range for crack origin AE signals. Peak load on the load position scale is with minimum load at 0/100. A cursory correlation with zone information generated by monitoring Array 1 supports the idea that these clusters are flaw B data.

A plot of the combined data from CY7, C17, and C36 Tape 4 is shown in Figure 7. The C17 data that were assumed to be crack-origin AE were in fact from a noise source. Source location information and

FIGURE 5. Clustering of Data from C36 Tape 4. In these figures, crack indications are denoted by o and +; noise, by x, Δ, ◊, and ▽; electrical transients by *; and valve closure, by ⊠.

FIGURE 6. Location of C36 Tape 4 Data Clusters on Load Position

analysis of noise from the manhole gasket (see below) tend to support this hypothesis. C17 data passed by the zone isolator appear to have come from a source in the direction of the manhole. The manhole gasket noise analysis results are plotted in Figure 8. In general, the noise does not intrude into the region associated with the data assumed to be crack AE. Pattern recognition results using a new piece-wise linear decision rule classified ~8.5% of the waveforms (66 out of 772) as crack AE, which appears reasonable.

The first 100 out of about 800 CY5 waveforms have been analyzed. Some of the data appears in noise clusters with the rest spread below and to the left of the crack AE cluster shown in Figure 7. Additional analysis of these data will be performed next quarter.

To summarize the results obtained to date:

- The original set of features seems to be able to discriminate several types of AE signals.
- The two most important features at this time are the natural logarithm of moment 0 of the power spectrum and the value of the autocorrelation function at time lag 37.
- Data from CY7, C17, and C36 Tape 4 exhibit three distinct types of data. Based on load position and source location, one cluster probably corresponds to crack activity from flaw B and the other two correspond to noise (including manhole gasket noise).
- Using the original set of six features (with moment 0 replaced by $\ln$ moment 0), the three data types can be discriminated with better than 90% accuracy.

FIGURE 7. Clustering of Data from CY7, C17, and C36 Tape 4

FIGURE 8. Clustering of Manhole Cover Noise

- The data from the manhole gasket noise tape and CY5 qualitatively support the results described above. Some of the discrepancies may be due to the differences in pressure loading and test temperature for the data tapes.
- A piece-wise linear decision rule appears to give better classification results; this rule has been supplied to the test site in Germany.

REACTOR MONITORING

Sixteen AE sensor channels have been installed at the Watts Bar 1 reactor for monitoring hot functional preservice testing of the reactor. The locations being monitored are the same as previously described: the No. 2 inlet nozzle, the 10-in. safety injection line adjacent to the No. 2 cold leg, and a segment of the reactor vessel wall. A typical high-temperature waveguide sensor installed on the inlet nozzle is shown in Figure 9, waveguides on the safety injection line are shown in Figure 10, and a Westinghouse high-temperature sensor on the bottom of the vessel is illustrated in Figure 11. A compact tension specimen driven by thermal expansion will be installed on the safety injection line to provide a reference for detecting AE from cracking in the environment of reactor background noise.

FIGURE 9. AE Waveguide Sensor Assembly on the No. 2 Inlet Nozzle at Watts Bar 1

FIGURE 10. AE Waveguide Sensors Mounted on No. 2 Safety Injection Line at Watts Bar 1

FIGURE 11. Westinghouse AE Sensor on Vessel Bottom at Watts Bar 1

The Tennessee Valley Authority (TVA) electrical construction staff have been very cooperative in this work. They did an excellent job of installing the necessary permanent cabling from the sensing locations to the auxiliary instrument room where the AE monitoring system is located.

Due to the delays in completing the ZB-1 vessel test in Germany where the primary AE monitoring system is being used, it will be necessary to use a backup system at Watts Bar. This backup consists of a Battelle-owned Dunegan/Endevco 1032D monitoring unit with a Battelle-built waveform recorder for waveform capture. Pattern recognition methods will be applied to these waveforms after the test.

The primary portion of the hot functional test to be monitored is the step-wise increase in temperature and pressure to operating conditions. This increase will proceed as follows:

- ambient to 250°F and 400 psig
- to 350°F and 400 psig
- to 450°F and 1200 psig
- to 557°F and 2235 psig (will be held for about a month).

The increase was started July 2, 1983. The most significant preliminary results concern coolant flow noise. The AE system was seeing a strong noise component at 150°F that would definitely preclude AE detection. By 350°F and 400 psig, the noise had decreased by a factor of 6 to 7—a borderline condition for AE detection. Since this pattern was also observed at the San Onofre reactor, further noise reduction can be expected before operating conditions are reached.

Tests performed on a Washington Public Power Supply System PWR vessel provide encouragement that AE from belt-line cracking should be detectable. The test and the results are summarized in Figure 12. The AE sensors were located at points that would be accessible on an installed reactor vessel. The difference in time of signal arrival at the two sensors indicates a propagation velocity somewhat higher than shear velocity.

FIGURE 12. Signal Detection Tests on PWR Vessel

REPORTS

- Quarterly progress report for the period from January through March 1983.
- Midyear program review, April 22, 1983.

FUTURE WORK

The following work is expected to be completed from July 1 to September 30, 1983:

- Complete the ZB-1 vessel test.
- Complete hot functional test monitoring at the Watts Bar 1 reactor.
- Complete a topical report on the 70°C portion of the ZB-1 vessel test.
- Continue pipe material tests.
- Continue development of the engineering prototype monitor system.
- Continue improvement of the pattern recognition algorithm.

REFERENCES

1. Edler, S. K. July 1983. *Reactor Safety Research Programs Quarterly Report - January-March 1983*. NUREG/CR-3307, Vol. 1, PNL-4705-1, Pacific Northwest Laboratory, Richland, Washington.

REPORTS

- Quarterly progress report for the period from January through March 1983
- Midyear program review, April 22, 1983

FUTURE WORK

The following work is expected to be completed from July 1 to September 30, 1983:

- Complete the ZB-1 vessel test
- Complete hot functional test monitoring at the Waste Ref. reactor
- Complete a topical report on the 20°C portion of the ZB-1 vessel test
- Complete pipe material tests
- Continue development of the engineering prototype monitor system
- Continue improvement of the pattern recognition algorithm

REFERENCES

1. Eiler, S. K. July 1983. Reactor Safety Research Program Quarterly Report - January-March 1983. NUREG CR-3307, Vol. 1, NRC-402-1, Pacific Northwest Laboratory, Richland, Washington.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS(a)

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SUMMARY

During the past quarter, progress was made on a number of tasks. A midyear progress review was held the first of April in Washington, D.C. For the piping task, a working draft of a document for qualifying equipment, procedures, and personnel was presented at a workshop in Seattle, Washington, on June 22 and 23 that was attended by 27 members of the nuclear industry. For the vessel application task, the Matrix II blocks were completed for studying strip, multiple wire, and single wire clad effects on ultrasonic inspection and a first cut on inspection guidelines for the ultrasonic inspection of the near-surface region of vessels. The fracture mechanics (FM) work focused on extending the fatigue crack growth studies to intergranular stress corrosion cracking (IGSCC) and assessing the impact on optimizing the inspection interval and sampling plan.

INTRODUCTION

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

The Integration of Nondestructive Examination (NDE) Reliability and Fracture Mechanics Program at Pacific Northwest Laboratory (PNL) 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 FM analysis, determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to insure a suitably low failure probability
- evaluate the degree of reliability improvement that could be achieved using improved and advanced NDE techniques
- based on material 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, and the results and recommendations are also applicable to Class II piping systems.

TECHNICAL PROGRESS

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

PIPE APPLICATION TASK

Due to an acceleration of the schedule, nearly all piping work in this quarter was directed to Task 4 - Development of Qualification Criteria for Ultrasonic Testing (UT) in ISI. The end product for this task was a draft NRC report that was to be delivered in December 1983. The schedule was accelerated to facilitate a workshop meeting scheduled to follow a symposium on NDE reliability held June 20-21, 1983. Since many of the parties interested in qualification would be attending the symposium, the workshop was scheduled with the assurance of high attendance.

Because of the limited time available for producing the qualification report, a working draft format was chosen. The objectives of the document were to:

- Define criteria, requirements, and processes for qualifying the ultrasonic examinations used for ISI applications on nuclear power plants.
- Provide assurance that applications of UT during ISI can meet application-specific probability-of-detection criteria and application-specific flaw characterization accuracy criteria.

The approach to UT qualification outlined in the document is significantly different from previous qualification schemes in three aspects:

- The three elements of UT (personnel, equipment, and procedures) are involved.
- A statistically significant performance qualification is included.
- Industry-wide uniformity in written examinations from personnel qualification are called for.

Most existing qualification approaches do not address all three elements of UT (personnel, equipment, and procedures). These three elements form an inspection system, interacting with one another in ways that are not currently predictable, and hence cannot be qualified separately.

To quantitatively know the effectiveness of ISI, a performance qualification is required. A performance qualification is a statistically significant demonstration of the ability to meet specified probability-of-detection criteria and characterization accuracy criteria for flaws that may occur in a given component. When the probability of flaw detection and accuracy of flaw characterization is known, the effectiveness of ultrasonic inspections may be accurately included in FM analyses of system reliability. The performance qualification is a test of the entire UT system acting as a unit.

Inspection personnel must complete written examinations to assess their knowledge of test principles and specific problems encountered in nuclear ISI. The inspection personnel must be sufficiently knowledgeable in UT principles and applications to be able to adjust for the wide variety of inspection conditions and configurations present in the field. Currently, written examinations for personnel qualification are composed and administered by the employer. There are many employers of UT/ISI personnel; and, consequently, the length, scope, content, and difficulty of written examinations vary considerably. To achieve a minimum industry-wide base-line level of competence, a uniform written examination must be instituted.

The workshop was attended by representatives of PNL, NRC, ACRS,^(a) EPRI, ASME, ASNT, several nuclear utilities, and three European participants. The general reaction to the document was positive. The most prominent negative reactions were related to concerns over the complexity and expense of implementing this qualification approach. Suggestions were generated for improving the structure and approach of the document and for providing useful technical input to define specific qualification criteria (this input has not been collated). PNL is working on a second draft of the document and is planning more workshops.

VESSEL APPLICATION TASK

The initial thrust of this task has been establishing the effectiveness and limitations of using UT for detecting pressurized thermal shock (PTS)-type flaws. A short-term response to this problem has been published.⁽¹⁾ One of the conclusions in this report is that cladding surfaces have a significant effect on an inspection sound beam.

Research during the last two quarters has quantified the effects of manual single-wire cladding on sound fields. To study the effects of the clad/base metal dilution area, the surface of a manually clad test specimen was ground smooth and the end of the specimen was cut at a 45° angle. Sound beam profile measurements were made by translating an EMAT probe across the angled end of the test specimen. The experimental setup is shown in Figure 1.

The results of shear wave beam profiles at 1.0, 2.25, and 5.0 MHz are shown in Figures 2 through 4. Results from longitudinal beam profiles at the same frequencies are shown in Figures 5 through 7. The data graphically demonstrate that shear wave sound beams are more distorted by cladding than longitudinal beams. Therefore, the data suggest that longitudinal sound waves should be used instead of shear waves for detecting PTS-type flaws when inspecting the near-surface region.

FIGURE 1. Effects of Clad/Base Metal Interface on Beam Profile

(a) ACRS - Advisory Committee on Reactor Safeguards
EPRI - Electric Power Research Institute
ASME - American Society of Mechanical Engineers
ASNT - American Society for Nondestructive Testing.

FIGURE 2. Shear Wave Beam Profile Through Clad at 1.0 MHz and 45°

FIGURE 3. Shear Wave Beam Profile Through Clad at 2.25 MHz and 45°

FIGURE 4. Shear Wave Beam Profile Through Clad at 5 MHz and 45°

FIGURE 5. Longitudinal Beam Profile Through Clad at 1.0 MHz and 45°

FIGURE 6. Longitudinal Beam Profile Through Clad at 2.25 MHz and 45°

FIGURE 7. Longitudinal Beam Profile Through Clad at 5.0 MHz and 45°

As a result of this work, the following guidelines for ultrasonic inspection of the clad/base metal interface are recommended:

- equipment
 - dual search units using an angled pitch-catch technique
 - nominal inspection frequency of 2.25 to 5 MHz
 - nominal inspection angle of 70° (60° for immersion techniques)
 - longitudinal wave mode
- calibration block
 - clad by same welding procedure as vessel area to be inspected
 - surface roughness representative of the production part
 - calibration reflector should be either a 1/16-in. (1.6-mm) side-drilled hole or a 1/8-in. (3.2-mm) flat-bottom hole at clad interface
 - response from calibration reflector should be adjusted to 80% of full-screen height.
- examination
 - scanning direction both parallel and perpendicular to cladding
 - evaluation of all indications exceeding 50% of the reference level at and below the interface.

FUTURE WORK

During the next quarter, the following activities will be completed:

- The pipe inspection round robin report (including destructive assay of test specimens) will be finished.
- Data analysis of advanced technique studies will be completed.
- Second draft of qualification document will be generated.
- Cladding effects studies on Matrix II blocks containing single wire, multiple wire, and strip clad will be completed.
- Two formal reports—one on the sampling plan and the other on inspection interval optimization for service-induced cracks (IGSCC and fatigue cracks)—will be published.

REFERENCES

1. Taylor, T. T., et al. February 1983. *Detection of Small-Sized Near-Surface Under-Clad Cracks for Reactor Pressure Vessels*. NUREG/CR-2878, PNL-4373, Pacific Northwest Laboratory, Richland,

As a result of this work, the following guidelines for ultrasonic inspection of the clad/bare metal interface are recommended:

- equipment
 - dual search unit using an angled pitch-catch technique
 - nominal inspection frequency of 2.25 to 5 MHz
 - nominal inspection angle of 70° (60° for immersion techniques)
 - longitudinal wave mode
- calibration block
 - clad by same welding procedure as vessel area to be inspected
 - surface roughness representative of the production part
 - reflection reflector should be either a 1.75-in. (4.4-mm) side-drilled hole or a 1/8-in. (3.2-mm) flat-bottom hole at axial interface
 - response from calibration reflector should be adjusted to 80% of full-screen height
- examination
 - scanning direction both parallel and perpendicular to cladding
 - evaluation of all indications exceeding 50% of the reference level at and below the interface.

FUTURE WORK

- During the next quarter, the following activities will be completed:
- The pipe inspection rough table report (including descriptive array of test specimens) will be finished.
 - Data analysis of advanced technique studies will be completed.
 - Second draft of qualification document will be generated.
 - Cladding effects studies on Matrix II blocks containing single wire, multiple wires, and gap clad will be completed.
 - Two formal reports—one on the sampling plan and the other on inspection interval optimization for service-induced cracks (IGSCC and fatigue cracks)—will be published.

REFERENCES

1. Taylor, T. T. et al. February 1983. Detection of Small-Sized Near-Surface Under-Clad Cracks for Reactor Pressure Vessels. NUREC CR-3826, PNL-4523, Pacific Northwest Laboratory, Richland.

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

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SUMMARY

The principal objectives of this program are to obtain in-reactor and out-of-reactor data on thermal and mechanical fuel rod performance and to integrate these data into the FRAPCON-2 computer code. Postirradiation fuel density measurements for Rods 1 and 6 of IFA-432 were received and appear to be consistent with believed changes during irradiation. PARAGRASS was received from Argonne National Laboratory (ANL), and an initial test case was run. Five fuel-cladding mechanical interaction (FCMI) tests using irradiated cladding were conducted in the pellet-cladding interaction (PCI) simulator; cladding ridging was observed.

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 postirradiation examination (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 conducted 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 UO_2 fuel, including:

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50 to 380 μm (0.002 to 0.015 in.)
- 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% of theoretical density (TD), the latter both stable and unstable regarding in-reactor densification
- burnups up to 52 MWd/kgM
- 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.

TECHNICAL PROGRESS

Work that has been completed during this quarter is discussed by task in the following sections.

TASK A - IRRADIATION EXPERIMENTS

IFA-432 and IFA-518 were moved to the outer region of the reactor core (positions 8-24 and 7-16, respectively) in April 1983 to maintain the peak powers at previous levels following a planned increase in the flux level of the central region of the core. The burnup increment for the period from January 2, 1983, to March 14, 1983 (the last reported data) was small—about 0.5 MWd/kgM for both assemblies—due to a long refueling shutdown.

PIE has continued at Harwell, U.K., on Rods 1 and 6 of IFA-432. Bulk fuel pellet density measurements have been completed on samples from the top and bottom of both rods. The results are discussed under Task B.

TASK B - DATA QUALIFICATION AND ANALYSIS

Bulk fuel density measurements have been completed on samples from the top and bottom of Rods 1 and 6 of IFA-432. The results are listed in Table 1; PNL's preirradiation density determinations on the same or sibling pellets are also included. Measuring the irradiated pellets was difficult because a large number of pellet fragments had to be retrieved and individually measured to represent a whole pellet.

At PNL's request, Harwell also measured the density of an unirradiated pellet from Rod 7 of IFA-432 using the same water immersion technique and equipment used to measure the bulk density of the irradiated samples. Harwell and PNL results for this pellet were very similar, indicating little bias between the two laboratories. Thus, direct comparison can be made between the preirradiation (PNL) measurements and the postirradiation (Harwell) results for Rods 1 and 6.

TABLE 1. Harwell Fuel Density Results for IFA-432

Rod Number	Pellet Position	Number of Pieces and Weight Fraction of Pellet	Preirradiation Density, % TD	PIE Density, % TD
1	20	32 and 0.43	95.5	93.5
1	40	33 and 0.50	95.5	94.3
6	17	31 and 0.40	92.2	91.8
6	20	56 and 0.55	92.3	93.8
6	40	36 and 0.50	92.3	94.2
7(a)	-	8 and 0.74	95.5	95.3

(a) Unirradiated.

The postirradiation fuel density in both rods is about 94% TD (excluding the anomalously low value for pellet 17 of Rod 6) (Table 1). A common density for both rods at first seems incredible since the preirradiation densities of the rod were quite different—95% versus 92% TD. However, based on resintering tests for both fuel types, both are estimated to have achieved 96% TD in-reactor.⁽¹⁾ It is also expected that both achieved that density early in life (at less than 5 MWd/kgM burnup), based on PIE density measurements for Rod 6 of IFA-431. Both fuel types were then subject to swelling (due to solid fission product accumulation) through the end of life at 30 to 35 MWd/kgM and achieved a final common density of 94% TD.

This sequence is summarized in Figure 1, where the PIE density results are plotted versus local burnup. It is noteworthy that the results for both rods can be extrapolated back to an early-in-life density of approximately 96% TD, which is the value both are estimated to have reached.

TASK C - FUEL CODE MAINTENANCE AND IMPROVEMENT

The final version of PARAGRASS for FRAPCON-2 Version 1 Modification 5 (V1M5) was received from ANL. One test case was run comparing a stand-alone version of PARAGRASS to a stand-alone version of FASTGRASS. The test case—Rod 6 of IFA-432—showed that the fission gas release was about the same for the two models but that the retained gas radial distribution changed. The FASTGRASS results had an unrealistic, unexplainable “hump” at the midradius in the fuel while the new PARAGRASS model did not.

FIGURE 1. Fuel Density Results for IFA-432

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS

Five FCMI tests were conducted in the PCI simulator using irradiated cladding. The PCI simulator is in an electrically heated water loop with test conditions of 300°C and 1500 psig and utilized an electrically heated simulated fuel rod with an 18-in. heated length. The simulated fuel rod had 0.430 in. OD(a) x 0.373 in. ID cladding, annular-dished UO₂ fuel pellets, and a central tungsten heater rod. The nominal fuel-cladding gap was 0.002 in., and the internal fuel rod helium pressure was 10 psig. The irradiated cladding source was water-filled, nonfueled Zircaloy-4 tubing that experienced neutron fluences equivalent to a burnup of approximately 6 MWd/kgM. During the tests, the axial variation of the cladding diameter was measured at two simultaneous azimuthal orientations that were 90° apart. The test conditions are summarized in Table 2.

The first irradiated cladding test (PCII-1) was a scoping test conducted to 1) determine the power required for initial FCMI (P_i); 2) determine the power where fuel creep rapidly reduces cladding stresses; and 3) if the rod was intact, attempt a power step to induce high cladding stresses and possible failure. The first two objectives were the primary goals for the test so that subsequent tests could be performed over the maximum power differential that would permit high cladding stresses for significant periods and have a higher likelihood of inducing cladding failure. Based on an extrapolation of observed ridge height as a function of power, P_i was determined to be between 1.5 and 2.0 kW/ft. This power was used as the starting point for controlled power ascents in later tests. The power required for rapid cladding relaxation was observed at 8 kW/ft,^(b) just before the fuel rod failed after 28 min. The rod apparently failed due to shorting between the tungsten heater and the cladding through the UO₂ fuel. Later tests were limited to less than 6.5 kW/ft to prevent the shorting problem.

The purpose of the second test (PCII-2) was to induce a cladding failure during the power ascent by straining the cladding at the most sensitive strain rate reported for nonirradiated cladding,^(2,3) $2 \times 10^{-5} \text{ min}^{-1}$. If no failure occurred during the first power ascent, the terminal power was to be held for a significant period and/or attempt to induce additional FCMI by power-cycling the fuel rod. Cladding failure did not occur after three power cycles and an accumulated period of 7.25 h at the terminal power of 6.5 kW/ft.

Recognizing that PCI failures often occur as a result of relatively rapid reactor power changes, the third test (PCII-3) was conducted at a more rapid cladding strain rate— $12 \times 10^{-5} \text{ min}^{-1}$. It was also intended to maximize the chances of crack growth after reaching the terminal power by holding the power for 40 h. The rod failed after 13 h at 6.5 kW/ft, apparently by short-circuiting from the tungsten electrode to the cladding through the fuel.

Because no PCI-type failure had occurred in the first three tests with as-irradiated cladding, the inner surface of the cladding for the fourth test (PCII-4) was scratched to a depth of approximately 10% of the wall thickness. This rod survived five power cycles to 6.5 kW/ft at a cladding strain rate of $12 \times 10^{-5} \text{ min}^{-1}$ and an accumulated period of 7.25 h at the terminal power.

In an attempt to permit crack growth, the fifth test was conducted by holding the terminal power at 5.5 kW/ft for 27 h after a power ascent at a strain rate of $12 \times 10^{-5} \text{ min}^{-1}$. The cladding did not breach during the test.

Based on diametral measurements, the following results were obtained from the tests:

- P_i repeatedly occurred between 1.5 and 2.0 kW/ft, based on formation of pellet-pellet ridges.

(a) OD - outside diameter; ID - inside diameter.

(b) Note that 8 kW/ft (electrical) is equivalent to 10 kW/ft (nuclear) based on fuel-cladding differential expansion. In terms of cladding deformation, 8 kW/ft in this small-gap simulated fuel rod is equivalent to about 15 kW/ft for a nominal clearance fuel rod at low burnup.

TABLE 2. Summary of PCI Simulator Tests on Irradiated Cladding

Test	Power Cycle No.	Power Ascent(a) Strain Rate, $\times 10^{-5} \text{ min}^{-1}$	Terminal Power, kW/ft	Terminal Power Holding Period, h	Power Descent(a) Strain Rate, $\times 10^{-5} \text{ min}^{-1}$	No. of Axial Profilometry Traces	Results
PCII-1	1	Not controlled; varied between 1 and 100.	8	0.5	—	30	Cladding breach, apparently by electrical shorting.
PCII-2	1	2	6.5	4	-2	44	Cladding intact.
	2	2	6.5	3	-2		
	3	2	6.5	0.25	-2		
PCII-3	1	12	6.5	13	—	16	Cladding breach, apparently by electrical shorting.
PCII-4(b)	1	12	6.5	6	-12	31	Cladding intact.
	2	12	6.5	0.5	>-100		
	3	12	6.5	0.25	>-100		
	4	12	6.5	0.25	>-100		
	5	12	6.5	0.25	>-100		
PCII-5(b)	1	12	5.5	27	-12	21	Cladding intact.

(a) Cladding strain rates were estimated by averaging 1) the overall observed strain including ridges and 2) the ridge strain at several locations from PCII-1 between the power levels of 2 and 6.5 kW/ft.

(b) The inner surfaces of the cladding for PCII-4 and -5 were scratched to a depth of about 10% of the wall thickness.

- Midpellet ridges formed at powers between 3 and 5 kW/ft.
- The ridges increased in size with increasing power, reaching values as high as 0.0007 in. (diametral) at 6.5 kW/ft.
- Permanent ridges from 0.0001 to 0.0002 in. (diametral) were observed in the cladding at zero power.
- Based on only a small sample of 13 ridges, the pellet-pellet ridge height reduced in magnitude about 6% after 27 h at 5.5 kW/ft.

FUTURE WORK

Planned activities for the next quarter include completion of the IFA-432 and the IFA-527 PIE reports and incorporation of PARAGRASS into FRAPCON-2 V1M5.

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ACCELERATED PELLET-CLADDING INTERACTION MODELING(a)

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SUMMARY

The mechanical models describing cracked fuel pellet-cladding interaction (PCI) were implemented in the failure code, and benchmarking is well under way. Three cladding fracture process models were developed to describe the range of possible cladding fracture mechanisms. The first working version of the failure code is scheduled for completion in late July. Task B was completed this quarter.

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 assess 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.

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 Review.

For Task B, PNL has obtained the computer code RADAR(b) 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.

TECHNICAL PROGRESS

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

TASK A - FUEL FAILURE MODELING

Four major components will be constructed to complete the fuel failure code (GT2-F): a transient temperature calculator, a mechanical model for cracked fuel PCI, a cladding damage/fracture model, and a statistical model to describe the probability of fuel failure. The transient temperature calculator was completed and implemented during the first quarter (October-December 1982), and two mechanical models describing the effects of cracked fuel on PCI-induced cladding stresses were completed during the second quarter (January-March 1983). During this quarter, these mechanical models were combined and implemented in the failure code, and benchmarking is currently under way. Progress was also made on the cladding damage/fracture models. Two fracture process models were developed to represent 1) nonchemically assisted (slower) crack processes and 2) chemically assisted (faster) crack processes. These two models are intended to represent the lower and upper bounds of possible cladding cracking mechanisms.

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

(b) Developed by British Nuclear Fuels Limited (BNFL).

A third fracture model was also developed to describe the transition from the slower (nonchemically assisted) to the faster (chemically assisted) fracture process as a function of time-dependent iodine supply (from fission gas release) and time-dependent cladding stress histories. This model treats iodine depletion by chemical reaction to form ZrI_4 , diffusion of ZrI_4 down the crack, adsorption of ZrI_4 at the crack tip and subsequent embrittlement of the Zircaloy cladding, crack initiation time, and crack propagation rate. Preliminary results indicate that chemical reaction rates may dominate during short time period events (moderate to fast fuel rod power transients) while corrosive depletion at the crack tip may dominate at longer time periods. However, once crack growth has initiated, the crack usually propagates very quickly and the cladding fails. This model is currently qualitatively reproducing the well-known iodine-induced strain rate sensitivity of Zircaloy, and benchmarking efforts are in progress to refine these results.

TASK B - RADIAL POWER PROFILES

The technical evaluation of RADAR has concluded that the results from the RADAR model are sound, considering the simplicity of the model. The following parameter values are recommended for use with RADAR: 0.975 for the "fast leakage factor" for all reactors and 0.760 for the "resonance escape probability" for pressurized water reactors. A formulation describing the void-dependent resonance escape probability for boiling water reactor applications was also developed. The modification is based on equations from Reference 1 that have been coded into a new subroutine (VOIDP); three new variables have been added to the FRAPCON-2 input:

- IVOID - flag to specify when VOIDP is to be used. If IVOID = 0, default values of resonance escape probability. If IVOID $\neq$ 0, use VOIDP to calculate resonance escape probability.
- PITCH - pitch of square lattice assembly in cm; used when IVOID $\neq$ 0.
- RHOH2O - fractional density of the water; axial array used when IVOID $\neq$ 0 (not time dependent).

The final report on the RADAR evaluation and incorporation into FRAPCON-2 was issued on June 6, 1983. Task B has been completed, and no further efforts are planned.

FUTURE WORK

The NRC has requested that the first working version of the fuel failure code be completed by July 29, 1983. Plans for next quarter are to meet this goal by completing two activities: 1) benchmarking the mechanical models and 2) implementing the fracture models into the failure code. Work on the statistical models will also continue; completion is expected late in the quarter.

REFERENCES

1. Strawbridge, L. E. September 1963. *Calculation of Lattice Parameters and Criticality for Uniform Water Moderator Lattices*. WCAP-3269-25.

PIPE-TO-PIPE IMPACT(a)

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J. R. Friley

F. A. Simonen

SUMMARY

The limited technical work that was completed during this quarter will be reported next quarter.

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 are being developed and will be compared with current licensing criteria.

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

PIPE-TO-PIPE IMPACT (a)

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SUMMARY

The limited technical work that was completed during this quarter will be reported next quarter.

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 pipe-to-pipe high-energy fluid system piping (CFR) events. Data contained in Standard Review Plan 3.5.2, "Evaluation of Break Location 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 end-grips will be impacted against stationary specimens to achieve required strain levels.

The Pacific Northwest Laboratory (PWL) program involves two main areas: obtaining experimental data and developing predictive models. Preliminary studies to determine significant 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 are being developed and will be compared with current licensing criteria.

EVALUATION AND ACCEPTANCE OF WELDED AND WELD-REPAIRED STAINLESS STEEL FOR LWR SERVICE(a)

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SUMMARY

During the past quarter, efforts continued towards publishing a progress-to-date report and completing the data collection and analysis system assembly by the end of the fiscal year. The analysis-before-test document was sent to the U.S. Nuclear Regulatory Commission (NRC) for review.

INTRODUCTION

The objective of this Pacific Northwest Laboratory (PNL) program is to determine a method for evaluating the acceptance of welded and/or weld-repaired stainless steel (SS) piping for light-water reactor (LWR) service. Based on experimental data, validated models will be developed to predict the degree of sensitization (DOS) and the intergranular stress corrosion cracking (IGSCC) susceptibility in the heat-affected zone (HAZ) of SS weldments. The cumulative effects of material composition, past fabrication procedures, past service exposure, weldment thermomechanical history, and projected component life after repair will be considered.

TECHNICAL PROGRESS

Data analysis continued in two areas: 1) the thermomechanical data generated during the monitoring of a 1.25-in. thick flat plate Type 304 SS feasibility weld and 2) the data generated using the electrochemical potentiokinetic reactivation (EPR) DOS measurement technique.

THERMOMECHANICAL DATA ANALYSIS

Weld HAZ thermomechanical temperature and strain data were taken on a flat plate weld. Temperature readings were recorded for all positions indicated in the instrument domain illustrated in Figure 1. Selected surface strain measurement positions, both parallel and perpendicular to the plate, were also monitored. The instrument domain used to monitor strain and temperature is shown in Figure 2.

Multiple passes were made on the flat plate feasibility weld to assess the amplification, isolation, and filter needs associated with monitoring strain and temperature sensing devices during welding. Selected data were also recorded using a data logger system with a repetition rate of 10 s.

A typical plot of surface strain as a function of arc-on-time and instantaneous temperature is presented in Figure 3. The data indicate that the HAZ material is subjected to compression before any change in temperature occurs. This compressive stress rapidly changes to a tensile stress once the material begins to heat up. The rise time for strain and temperature appears to be approximately the same, and both decrease after the arc passes by.

(a) FIN: B2449; NRC Contact: J. Muscara.

FIGURE 1. Instrument Domain Layout

FIGURE 2. Instrument Domain Showing Profilometer Superimposed Over Thermocouple and Surface Marker Movement Detector Layout

FIGURE 3. Simultaneous Surface Strain Perpendicular to the Welding Direction and Temperature 0.6 in. from Weld Centerline

A typical plot of the surface strain perpendicular to the plane of the plate as a function of arc-on-time is presented in Figure 4. A thermally induced "bump" related to arc passage superimposed over a relatively constant rate of plate bending takes place during welding. A profile of the maximum dimensions of a cross section of the bump perpendicular to the welding direction is shown in Figure 5. The bump height is greater than 0.01 in. at the weld centerline and can be resolved 1.5 in. from the weld centerline.

ELECTROCHEMICAL POTENTIOKINETIC REACTIVATION MEASUREMENTS

Standard cell EPR measurements were taken on a series of Type 304 SS specimens heat-treated to different DOS levels. Plots of the data for several alloys are presented in Figure 6. The EPR technique yields self-consistent DOS measurements showing a continuing increase with increasing exposure time for the 600°C and 700°C temperatures and an initial increase and then a decrease at 800°C.

Field cell EPR measurements were taken on a weld HAZ from a 1.1-in. thick, 24-in. diameter Type 304 SS pipe as a function of distance from the weld fusion line and as a function of probe area (Figure 7). The DOS level increased and then decreased with increasing distance from the fusion line. The observed maximum initial increase in DOS increased with decreasing measurement area.

FIGURE 4. Surface Movement Perpendicular to Plate Surface at Weld Centerline

FIGURE 5. Thermal Bump Height as a Function of Distance from Weld Centerline

FUTURE WORK

The progress-to-date report and the data collection and analysis system assembly will be completed.

FIGURE 6. Measured EPR Values as a Function of Isothermal Annealing Time and Carbon Content for Several Type 304 SS Alloys

FIGURE 7. Degree of Sensitization Measurements as a Function of Distance from the Weld Fusion Line for Three Probe Sizes

FIGURE 7. Degree of Sensitization Measurements as a Function of Distance from the Weld Fusion Line for three Probe sizes

SEVERE CORE DAMAGE SUBASSEMBLY PROCUREMENT PROGRAM

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

R. L. Goodman, Project Manager

G. S. Allison	L. L. King
A. J. Anthony	L. J. Parchen
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SUMMARY

The SFD-3 fuel bundle and insulated shroud assemblies were completed and delivered on schedule to the Idaho National Engineering Laboratory (INEL) on June 27, 1983. Final assembly activities are continuing for the SFD-4 fuel bundle and insulated shroud assemblies, which will be delivered to INEL on August 29, 1983. Conceptual and detailed designs for the Power Burst Facility (PBF) Series II SFD test train assembly have begun.

INTRODUCTION

The Severe Core Damage 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, Idaho Falls, Idaho. The objective of this Pacific Northwest Laboratory (PNL) program is to study the behavior of light-water reactor fuel under severe high-temperature, flow-starvation conditions. In Series I, the peak cladding temperature will be limited to 2400K, which includes conditions ranging from those anticipated in a design-basis loss-of-coolant accident to those anticipated through the melting point of Zircaloy. Many portions of the Series I tests should directly benefit the PBF Series II SFD testing and the National Research Universal (NRU) reactor program due to similarities in the experimental objectives and for materials, instrumentation, and fabrication development.

FUTURE WORK

- Complete and deliver the PNL/PBF SFD-4 fuel bundle and insulated shroud assemblies to INEL on August 29, 1983.
- Conduct joint PNL/EG&G working meetings on PBF Series II SFD test train assembly design details and development needs.

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

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

PROCUREMENT PROGRAM

SEVERE CORE DAMAGE SUBASSEMBLY

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SUMMARY

The SFD-1 fuel bundle and insulated shroud assemblies were completed and delivered on schedule to the Idaho National Engineering Laboratory (INEL) on June 27, 1982. Final assembly activities are continuing for the SFD-2 fuel bundle and insulated shroud assemblies, which will be delivered to INEL on August 23, 1982. Conceptual and detailed design for the Power Burst Facility (PBF) Series II SFD test fuel assemblies have begun.

INTRODUCTION

The Severe Core Damage Subassembly Procurement Program includes the design, development of appropriate materials and supporting fabrication process, and complete fabrication of fully instrumented test fuel assemblies for the U.S. Nuclear Regulatory Commission (NRC)-sponsored test program at the PBF Idaho Falls, Idaho. The objective of this Facility Nuclear Laboratory (FNL) program is to study the behavior of light-water reactor fuel under severe high-temperature, low-stress conditions. In Series I, the peak cladding temperature will be limited to 2400K, which includes conditions ranging from those anticipated in a design-basis loss-of-coolant accident to those anticipated through the melting point of Zircaloy. Many portions of the Series I test should directly benefit the PBF Series II SFD testing and the National Research Universal (NRU) reactor program due to similarities in the experimental objectives and for materials, instrumentation, and fabrication development.

FUTURE WORK

- Complete and deliver the PNL/FBF SFD-2 fuel bundle and insulated shroud assemblies to INEL on August 23, 1982.
- Conduct joint PNL/FBF working meetings on PBF Series II SFD test fuel assembly design details and development needs.

(a) FNL 82084-2; NRC Contract R-2740 Houston.

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. Plans are being made for the return of the site representative to the United States.

INTRODUCTION

The Super Sara Test Program (SSTP) was a major European Community effort to study reactor safety during rapid or large-break and slow or small-break loss-of-coolant accidents. The program was to be conducted in the SUPER SARA high-temperature, high-pressure loop in the ESSOR reactor, Ispra, Italy. The objective of the SSTP was to obtain experimental data on fuel rod deformation and postaccident coolability of damaged fuel assemblies after a loss-of-coolant-type accident. The testing program included loop construction and 21 in-reactor experiments simulating 7 large- and 12 small-break conditions in commercial pressurized and boiling water reactors. Funding for the SSTP was terminated by the European Council of Ministers.

FUTURE WORK

During the remaining residence period at Ispra, the NRC site representative will continue to inform the NRC of developments in the areas of interest, particularly the area of postaccident heat removal studies.

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

ESSOR FUEL DAMAGE TEST PROGRAM SUPPORT(a)

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

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) site representative at Jspis is continuing to provide liaison and safety analysis support. Plans are being made for the return of the site representative to the United States.

INTRODUCTION

The Super Data Test Program (SDTP) was a major European Community effort to study reactor safety during rapid or large-break and slow or small-break loss-of-coolant accidents. The program was to be conducted in the SUPER SARA high-temperature, high-pressure loop in the ESSOR reactor, Jspis, Italy. The objective of the SDTP was to obtain experimental data on fuel rod deformation and post-accident condition of damaged fuel assemblies after a loss-of-coolant-type accident. The testing program included loop construction and 31 in-reactor experiments simulating 7 large- and 12 small-break conditions in commercial pressurized and boiling water reactors. Funding for the SDTP was terminated by the European Council of Ministers.

FUTURE WORK

During the remaining residence period at Jspis, the NRC site representative will continue to inform the NRC of developments in the areas of interest, particularly the need of post-accident heat removal studies.

(a) RMC 83373-1; NRC Contact: R. Van Houten.

SEVERE CORE DAMAGE MATERIALS PROPERTY TESTS(a)

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SUMMARY

During this quarter, viscosity measurements were conducted on pure zirconium and Zr-10 mol% UO_2 at temperatures up to 2000°C. Isothermal oxidation studies of Zircaloy in steam were performed at 1400°C.

INTRODUCTION

The objective of this Pacific Northwest Laboratory (PNL) program is to perform high-temperature materials property tests and to provide data that will assist in the planning and analysis of U.S. Nuclear Regulatory Commission (NRC) severe core damage fuel behavior irradiation tests. High-temperature (>1600°C) ex-reactor physical property data and reactor kinetics data are needed on cladding and cladding/fuel/structure reaction products to model rod behavior and to properly account for the melting and refreezing of the cladding. Work is planned to study Zircaloy/ H_2O / UO_2 reaction kinetics and to determine the viscosities of liquefied fuel for several Zr- UO_2 compositions.

TECHNICAL PROGRESS

Kinetic studies on Zircaloy/ H_2O reactions were completed at 1400°C using a laser-heating technique in a flowing superheated (to 500°C) steam environment. The back side of the specimen was heated using a laser. A two-color infrared pyrometer monitored the temperature of the front side of the specimen and controlled the laser power. Heating rates greater than 200°C/s and isothermal temperature control of $\pm 5^\circ\text{C}$ at 1400°C were achieved. Uniform oxide scale layers were observed on the front surface. Micrograph analysis of the front surface of the specimens exposed to steam for up to 1200 s at 1400°C showed that the ZrO_2 scale growth was slightly (~50% thicker) than reported by Cathcart and Urbanic. This increased growth may have been a result of the higher injected steam temperature and the increasing thermal gradient across the sample that was used in this study.

Viscosity measurements on zirconium and Zr-10 mol% UO_2 to temperatures of 2000°C were made in an oscillating cup viscometer (Figure 1). The molten material was contained in a ThO_2 crucible. Although this crucible material was not entirely free of reaction with the sample, it was better than any other known substance. Decay constants were measured experimentally at several temperatures, and the resultant data were used to calculate viscosity. Decay constants increased as the viscosity of the sample decreased. In the Zr-10 mol% UO_2 mixture, the first liquid phase appeared at 1400°C according to the phase diagram; however, the mixture did not behave as a true liquid until the temperatures were above 1800°C. The remaining solid material apparently prevented fluid behavior until it was completely melted.

(a) FIN: B2455; NRC Contact: R. Van Houten.

FIGURE 1. Viscosity Measurement System

FUTURE WORK

- Viscosity measurements of Zr-5 mol% UO_2 and Zr-7.5 mol% UO_2 will be completed.
- The effects of ThO_2 crucible/zirconium reactions on viscosity will be determined, and comparisons will be made between the viscosity of Zircaloy and pure zirconium.
- Kinetic studies of Zircaloy/ H_2O at 1400, 1600, 1700, and 1800°C will be completed.
- Kinetic studies of the Zircaloy reaction in $\text{H}_2/\text{H}_2\text{O}$ environments will be initiated.
- The Nd-Yag laser will be installed and supplemental studies on H_2 blanketing and Zircaloy/ H_2O reaction kinetics will be initiated.

FUTURE WORK

- Viscosity measurements of $Zr-5 \text{ mol\% UO}_2$ and $Zr-5.5 \text{ mol\% UO}_2$ will be completed.
- The effect of ThO_2 crucible/zirconium reaction on viscosity will be determined, and comparisons will be made between the viscosity of Zircaloy and pure zirconium.
- Kinetic studies of Zircaloy-H₂O at 1400, 1500, 1750, and 1800°C will be completed.
- Kinetic studies of the Zircaloy reaction in H₂O environments will be initiated.
- The Nd-Yag laser will be installed and supplemental studies on H₂ blanketing and Zircaloy-H₂O reaction kinetics will be initiated.

CORE MODEL DEVELOPMENT(a)

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SUMMARY

The COBRA/TRAC code was used to predict Semiscale Test S-UT-5 (a 2.5% communicative cold leg break). The prediction compared well with measured data. COBRA-TF(b) was used to predict two FLECHT-SEASET forced reflood tests. These tests demonstrate the ability to predict reflood transients with low flooding rates and high temperatures. A pretest prediction was made of the HDR water blowdown experiment V21.3 using COBRA-NC. Results will be reported after the test is run and the measured data are released.

INTRODUCTION

The COBRA-TF computer code is being developed for the U.S. Nuclear Regulatory Commission (NRC) to provide better digital computer codes for assessing the behavior of full-scale reactor systems under postulated accident conditions. This Pacific Northwest Laboratory (PNL) project has three main objectives:

- 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).
- Develop a hot bundle/hot channel analysis capability to evaluate the thermal-hydraulic performance of light-water reactor (LWR) fuel bundles during postulated accidents.
- Develop a containment code capable of simulating the steam water blowdown and hydrogen distribution phases of an accident.

The resulting codes—COBRA/TRAC, COBRA-TF, and COBRA-NC—are being used to perform pre- and post-test analysis of LWR components and system effects experiments.

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 at a different velocity. Thus, one can mechanistically treat a continuous liquid core or film moving at a low or possibly negative velocity from which liquid drops are stripped off and carried away by the vapor phase. This 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

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

(b) COBRA-TF: Coolant boiling in rod arrays—two fluid.

in regular Cartesian or cylindrical mesh coordinates; however, since they have a significant impact on the thermal-hydraulic response of the system, these geometries must be modeled with reasonable accuracy.

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

TECHNICAL PROGRESS

COBRA/TRAC APPLICATIONS

COBRA/TRAC⁽¹⁾ was used to predict Semiscale Test S-UT-5.⁽²⁾ This test simulated a 2.5% communicative cold leg break LOCA in a plant with UHI. The calculation was done to assess the ability to predict small-break transients in PWRs equipped with UHI. The vessel module was used to model the vessel, downcomer, and steam generator of the system; and the 1-D TRAC-PD2 components were used for the connecting piping, pressurizer, pumps, and accumulators.

Calculated and measured system pressures are compared in Figure 1. The calculated pressure followed the initial rapid depressurization and then slower rate of depressurization with reasonable accuracy. The collapsed core liquid level measured from the bottom of the heated length is shown in Figure 2. The difference at the beginning of the transient is due to liquid above the top of the core that is included in the measured core liquid level. As the core starts to void, the discrepancies disappear. It can be seen that the calculation compares favorably with the measurement.

FIGURE 1. S-UT-5 System Pressure Versus Time

FIGURE 2. S-UT-5 Core Liquid Level Versus Time

The simulation was carried out to 300 s, when the hot legs began to uncover and the 1-D component drift flux model overestimated the flow resistance inside the hot legs. This, in turn, caused too much depression of the core liquid level. Extending the calculation to 450 s verified this. Replacing the 1-D hot legs with the 3-D two-fluid vessel module should correct this problem and improve the results.

HOT BUNDLE CODE DEVELOPMENT

A joint project is being conducted with the FLECHT/SEASET(3) program to address Appendix K steam cooling/flow blockage rules. The first task of the project is to modify COBRA-TF to accurately predict reflood tests with low flooding rates and high cladding temperatures. Two forced reflood tests from the FLECHT/SEASET 161-rod test series have been selected to evaluate the performance of COBRA-TF. The test conditions are:

Parameter	Test 31805	Test 31203
Pressure, psia	40.0	40.0
Flooding rate, in./s	0.81	1.51
Peak power, kW/ft	0.70	0.70
Initial cladding temperature, °F	1600	1601

Predicted cladding surface temperatures are compared with measured temperature histories at 72 and 120 in. from the bottom of the heated length in Figures 3 through 6. Figures 7 and 8 show the measured and predicted quench front location as a function of time for each of the tests.

FIGURE 3. Comparison of Predicted Cladding Temperatures and Measured Temperatures at 72 in. from Bottom of Heated Length for Test 31805

FIGURE 4. Comparison of Predicted Cladding Temperatures and Measured Temperatures at 120 in. from Bottom of Heated Length for Test 31805

FIGURE 5. Comparison of Predicted Cladding Temperatures and Measured Temperatures at 72 in. from Bottom of Heated Length for Test 31203

FIGURE 6. Comparison of Predicted Cladding Temperatures and Measured Temperatures at 120 in. from Bottom of Heated Length for Test 31203

FIGURE 7. Comparison of Calculated and Measured Quench Front Location for Test 31805

FIGURE 8. Comparison of Calculated and Measured Quench Front Location for Test 31203

The data comparison for Test 31805 at 72 in. is excellent, with a good prediction of the peak temperature, turnaround time, and quench time. At the 120-in. elevation, the peak temperature is $\sim 100^{\circ}\text{F}$ higher than the measured data. More interfacial heat transfer is needed from the grid spacer models at the higher elevations early in the transient to reduce the peak temperature. In Figure 7, the location of the quench front compares very favorably with the measured data from Test 31805.

Predicted cladding temperature histories at 72 in. and 120 in. compare well with the measured data for Test 31203. The peak temperature at 120 in. is slightly higher than the measured peak temperature. The measured quench front elevation indicates that the bundle is quenching from the top as well as from the bottom. The calculation also indicates two quench fronts, both of which compare well with the measured data.

The results show a vast improvement over any previous effort to simulate low flooding rate/high cladding temperature reflood. The reason for the excellent results is the detailed modeling of the heat transfer at the liquid and steam interface. The calculation includes a model to track the movement of a quench front on a grid spacer and a model to break up liquid droplets that hit a grid spacer. These two models affect steam superheat, which in turn determines the cladding surface temperature.

COBRA-NC CONTAINMENT APPLICATIONS

A pretest prediction of the HDR water blowdown experiment V21.3 was completed this quarter and sent to the HDR project before the submission deadline. Results of the COBRA-NC predictions will be compared with experimental data after the experiment is run and the data have been released.

The User's Manual and the Assessment Manual for COBRA-NC are nearly complete. These documents will be sent to NRC for printing and distribution next quarter.

FUTURE WORK

COBRA/TRAC will be applied to a LOCA in a non-UHI PWR. COBRA-TF will be assessed against 21-rod FLECHT-SEASET unblocked bundle data. Flow blockage models will be added to COBRA-TF and assessed against 21-rod FLECHT-SEASET blocked bundle and FEBA data. Development and assessment of COBRA-NC will continue.

REFERENCES

1. Thurgood, M. J., et al. 1982. *COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems*. NUREG/CR-3046, Pacific Northwest Laboratory, Richland, Washington.
2. Jackett, K. E., and L. B. Clegg. November 1981. *Experiment Data Report for Semiscale MOD-2A Small Break Test Series (Tests S-UT-4 and S-UT-5)*. NUREG/CR-2349, EG&G Idaho, Idaho Falls, Idaho.
3. Conway, C. E., et al. December 1977. *PWR FLECHT Separate Effects and System Effects Test (SEASET) Program Plan*. NRC/EPRI/Westinghouse-1.

COOLANT BOILAWAY AND DAMAGE PROGRESSION EXPERIMENTS IN THE NRU REACTOR(a)

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SUMMARY

Work is progressing satisfactorily on the design and procurement activities associated with the first full-length high-temperature (FLHT-1) severe fuel damage (SFD) experiment to be performed in the National Research Universal (NRU) reactor at Chalk River, Ontario. This experiment is scheduled for May 1984. The design of the high-temperature shroud, including instrumentation layout and fuel rods, was completed and is being reviewed. Design of the complete test train and the loop support system is continuing. Purchase orders for materials and components have been placed. The draft experiment plan was completed and reviewed. Calculations to predict the operating parameters for the experiment are in progress, and work on preparing the preliminary safety analysis report (PSAR) for review by Chalk River Nuclear Laboratory (CRNL) is under way.

INTRODUCTION

Two FLHT experiments will be conducted to provide data for evaluating fuel damage progression behavior and fuel debris bed coolability under SFD degraded core conditions. Pacific Northwest Laboratory (PNL) will conduct these experiments in the National Research Universal (NRU) reactor at Chalk River, Ontario. These experiments are the initial tests in the Coolant Boilaway and Damage Progression (CBDP) Program, which is sponsored by the U.S. Nuclear Regulatory Commission (NRC) in support of the SFD program.

The first experiment (FLHT-1) will provide data for investigating the full-length integral effects of a simulated coolant boilaway accident with 12 fuel rods reaching temperatures as high as 2144K (3400°F). The second experiment (FLHT-2) will simulate a pressurized water reactor (PWR) accident where the natural coolant boilaway will result in fuel temperatures up to 2478K (4000°F).

Many of the test train design and fabrication features and testing methods developed as part of the loss-of-coolant accident (LOCA) program conducted in the NRU reactor⁽¹⁾ are being utilized in the CBDP Program; for example, coolant water level measurement using time domain reflectometry, remote pressure sensing to detect fuel cladding failure times, a spray desuperheater to control outlet steam temperature, and the NRU loop facilities that provide thermal-hydraulic control of operating conditions. These capabilities along with the development of high-temperature instrumentation and a hydrogen measuring system will be used to evaluate the degraded core phenomena during the experiments.

The boilaway transient for FLHT-1 will be stabilized at several temperature levels by controlling the flow rate of the makeup coolant water in the assembly. Maximum temperatures to be achieved for the 12-rod fuel bundle are 2144K (3400°F). Steady-state data such as coolant flow rate, coolant level rate of change, steam production, and hydrogen generation rate will be collected as temperatures are increased to evaluate degraded core behavior.

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

The second experiment (FLHT-2) will simulate a PWR degraded core accident where the natural boil-away of the coolant will result in fuel temperatures up to 2478K (4000°F). Operating conditions and design requirements for FLHT-2 will be developed from information obtained during FLHT-1. The operating performance of high-temperature instrumentation, safety features, and the hydrogen measurement and steam/off-gas velocity measurement systems are of particular interest. FLHT-2 design features will be similar to those for FLHT-1 except that different materials may be used for the higher temperature operation of the insulated shroud and thermocouple sensors during continuous boilaway of the test assembly coolant.

TECHNICAL PROGRESS

Objectives were developed for the CBDP experiments to obtain integrated degraded core behavior information under prototypic conditions. An experiment plan for the FLHT test series was drafted and reviewed in detail with CRNL staff. The initial test (FLHT-1) is scheduled for May 1984 using a full-length 12-rod PWR test bundle at temperatures as high as 2144K (3400°F) in the NRU reactor. PNL and CRNL staff concurred on the experiment feasibility and major technical features of the experiment and its operations.

In addition to work on the experiment plan, the following design analysis and fabrication progress was made:

- The high-temperature shroud design was completed and is being reviewed. The shroud has the following features:
 - low-conductivity, high-strength ZrO_2 insulation
 - Zr-Ta thermocouples mounted in the insulation cavity
 - a molten material penetration detector located on the outside of the insulation zone.
- Design of the fuel rods was completed and is being reviewed.
- Design of the test train is continuing with emphasis on the inlet region, the steam plenum, and the outlet region. Preliminary drawings of instrument locations and wiring diagrams for the test train were completed and are being reviewed.
- A bundle makeup coolant control system module, which also provides for hydrogen measurement, is being designed. This system requires extensive interface with the NRU.
- The piping design between the top of the test train and the makeup control system module is being evaluated. The current design emphasis is focused on evaluating and reducing the radiation exposure to personnel during disassembly and removal of the test train from the NRU loop facility.
- Two meetings were held with CRNL to further define design requirements and interface points.
- Purchase orders were placed for the Zircaloy stock needed to fabricate the shroud and steam plenum components and the rigid zirconia insulation for the shroud. The special stainless steels (alloy 630 and Nitronic 50) were ordered and have been received. Long lead components for the high-temperature thermocouples were ordered.
- Preliminary feasibility calculations were performed that showed that a boildown test can be performed and controlled within the capabilities of the NRU loop facility.
- A revised TRUMP-BD model was developed that describes the latest design features of the test assembly, including the hanger and upper plenum assembly components.
- Upgraded data acquisition and control system electronic components and hardware were procured. A preliminary control system design based on the current experiment plan was initiated.

- Preliminary calculations were completed to determine radiation shielding requirements, both during and after the FLHT experiments. The results of these calculations are being reviewed.
- Initial design information was presented and discussed with CRNL design and safety engineers. Using this and other analytical data, CRNL safety engineers are preparing for the initial review by the Canadian Nuclear Safety Advisory Committee.
- Work began on the PSAR; the report will include reactor physics, test neutronics, and thermal-hydraulic calculations for accident hazard analyses.
- The NRU large-break LOCA test data have been analyzed, and the results were presented to the Advisory Committee on Reactor Safeguards fuels subcommittee.

FUTURE WORK

Work will continue in essentially all the areas mentioned above; the following specific work will be completed:

- The design of the inlet region, outlet region, and steam plenum of test train for FLHT-1 will be completed.
- Shroud and fuel rod designs for FLHT-1 will be reviewed.
- The designs of the makeup coolant control and the hydrogen measurement system will be completed.
- Mockups of the inlet and outlet seal regions will be fabricated to confirm design geometries.
- The experiment plan will be updated based on discussions with CRNL.
- The PSAR will be issued for review.
- All long lead items for the test assembly will be procured.

REFERENCES

1. Mohr, C. L., and G. M. Hesson. August 1983. "LOCA Rupture Strains and Coolability of Full-Length PWR Fuel Bundles. In *Proceedings of the Seventh International Conference on Structure Mechanics in Reactor Technology, SMIRT-VII*, Chicago, Illinois.

- Preliminary calculations were conducted to determine radiation shielding requirements, both during and after the EHT experiment. The results of these calculations are being reviewed.
- Initial design information was presented and discussed with CRNL design and safety engineers. Using this and other analytical data, CRNL safety engineers are preparing for the initial review by the Canadian Nuclear Safety Advisory Committee.
- Work begun on the PSAR. The report will include reactor physics, test geometries, and thermal-hydraulic calculations for accident hazard analysis.
- The NRU large-break LOCA test data have been analyzed, and the results were presented to the Advisory Committee on Reactor Safeguards test subcommittee.

FUTURE WORK

- Work will continue in essentially all the areas mentioned above; the following specific work will be completed:
- The design of the inlet region, outlet region, and steam generator of test train for EHT-1 will be completed.
 - Steam and fuel rod designs for EHT-1 will be reviewed.
 - The design of the makeup control system and the hydrogen measurement system will be completed.
 - Mockups of the inlet and outlet test regions will be fabricated to confirm design geometries.
 - The experiment plan will be updated based on discussions with CRNL.
 - The PSAR will be issued for review.
 - All long lead items for the test assembly will be procured.

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1. Mohr, C. J., and G. McFerson. August 1983. "LOCA Response Studies and Capability of Full-Length Fuel Rod Bundles." In *Proceedings of the Seventh International Conference on Structural Mechanics in Reactor Technology*, SMIR-7/VII, Chicago, Illinois.

STEAM GENERATOR GROUP PROJECT(a)

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SUMMARY

The Technical Advisory Group (TAG) met at the Electric Power Research Institute (EPRI), Palo Alto, California, on April 12 and 13, 1983. Participants then traveled to Richland, Washington, for direct observation and discussions of Steam Generator Group Project (SGGP) progress.

A large greenhouse was constructed around the channel head of the Steam Generator Examination Facility (SGEF) for contamination control during tube unplugging. Unplugging 470 of the 748 plugged tubes in the steam generator began May 26, 1983, and was successfully completed without incident on June 15, 1983, by a team from Westinghouse Steam Generator Services Division. The plugs were mechanically drilled from the tubes without lubricant or cutting oil. Significant quantities of water, sometimes filling the tube, were found and removed from 102 tubes. Five samples have been analyzed; one was primary water and four were secondary water. Sludge was found in 30 of the tubes as deep as 15 in.

Zetec, Inc., was selected as the subcontractor for the base-line eddy-current study. At the end of the quarter, the channel head and the greenhouse were clean and ready for the base-line study, which is scheduled to begin on July 11, 1983. The data acquisition and analysis system was being tested in its final configuration.

INTRODUCTION

The SGGP was initiated in January 1982 and is a continuation and expansion of the Steam Generator Tube Integrity Program (SGTIP): a multiphase, multitask laboratory program conducted at Pacific Northwest Laboratory (PNL). 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 six years of service was judged suitable for this research. It was transported by ocean-going barge through the Panama Canal from Virginia to Hanford, Washington, where it was installed in a specially designed containment facility, the 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.

Consortia of government and private organizations from Italy, France, Japan, and the United States have joined with the U.S. Nuclear Regulatory Commission (NRC) as program sponsors. Data obtained

(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.

from service-induced specimens will help establish and support licensing and regulatory requirements for operating nuclear steam generators as well as offer potential benefits in reliability. Research efforts on the Surry II generator will emphasize the following areas:

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

Representatives from all of the participating consortia and from the NRC and its contractors met at EPRI in Palo Alto, California, on April 12 and 13, 1983, for their regular semiannual meeting. EPRI program managers reported accomplishments of their steam generator programs, and contractor and vendor representatives also described their programs. On the second day of the meeting, Dr. Muscara described the NRC program and Dr. Clark gave an overview of SGGP progress. PNL task leaders gave status reports on Tasks 3, 6, 7, 9, and 10. Drafts of topical reports were distributed for review prior to publication. Plans for future subcontracts and for tubing and tube sheet sample removal were discussed. Participants then traveled to PNL, Richland, Washington, to observe the work in the SGEF directly. Inspection devices and techniques, data acquisition and analysis systems, decontamination specimens, health physics analysis instruments and techniques, and preparations for tube unplugging were demonstrated. The next TAG meeting will be held the week of October 17, 1983, in Tokyo and Kobe, Japan.

Tube Unplugging (Task 8)

The task of opening 470 plugged tubes in the steam generator was successfully completed June 15, 1983. These tubes are now accessible from both ends for insertion of nondestructive probes (Tasks 7 and 9). A total of 969 plugs were drilled out by a 16-man team from Westinghouse Steam Generator Services Division; 46 PNL technicians and engineers assisted. The air-driven drill used for the unplugging is shown in Figure 1.

FIGURE 1. Air-Driven Drill Used to Remove Plugs from Steam Generator

Twenty-nine of the plugs were drilled out to allow the drilling fixture to be positioned in some areas of the tube sheet. Five tubes could be opened only on one end. In one case, a drill bit broke in the tube and could not be removed. In two other tubes, the stub ends of the plugs had been blown off when they were installed and could not be retrieved. One hole was not drilled straight but may be correctable later. The fifth tube is obstructed by an unknown object, and it is not considered worth the time and cost necessary to retrieve it. All five of these tubes are accessible from only one end.

The unplugging operation proceeded smoothly with no lost time for injury or other incidents. It was completed in forty 10-h shifts, as planned, on schedule and within budget. A few equipment breakdowns caused minor delays. Broken shafts had to be replaced several times in the air-driven drill motors, but the time lost was not excessive. An automatic positioning mechanism being developed by the subcontractor was used with a hydraulic drill in the hot leg side but had to be removed from service about halfway through the program, due to hydraulic problems. When it functioned properly, the automatic equipment worked rapidly and conserved personnel exposure.

It had been predicted that 24 plugs could be removed per shift; during the operation, this number was frequently exceeded. PNL personnel worked three or four 10-h shifts in pairs (2 h on and 2 h off), positioning drills and replacing drill bits (Figure 2). Total radiation exposures were 15 rem for Westinghouse personnel and 46.8 rem for PNL personnel. Maximum individual exposures were 1.27 rem for a Westinghouse worker and 1.87 rem for a PNL worker.

Water was released from 102 tubes when the pilot hole broke through the stub ends of the plugs. The amount of water varied greatly from tube to tube; a few tubes were apparently full of water. Whenever possible, samples were taken of the water; 71 water samples were collected. Preliminary results from five water samples indicate that one tube contained primary water (from high boron content) and the other tubes contained secondary side water. None contained more than 20 ppm sodium, indicating there was no seawater intrusion.

FIGURE 2. Technician Changing Drill Bits

After the Inconel plugs were removed, about 30 tubes were still plugged by a sludge deposit. Sludge deposits varied in depth from a few inches to about 15 in. and ranged from the consistency and color of mud to a hard, black deposit. These deposits were later drilled out with a hand drill using extensions as required. Samples are being photographed and analyzed.

Contamination control during the drilling operation was much simpler than expected. Frequent removal of turnings and swabbing of spilled liquid kept the channel head safe and relatively clean. No significant contamination escaped the greenhouse. The facility is now ready for the base-line eddy-current study (Task 7) to begin.

Health Physics (Task 3)

A detailed report of the first year's results, including descriptions of the radiation measuring techniques, was presented to the TAG meeting. A topical report on the radiological mapping task is being reviewed by participants.⁽¹⁾

Analysis and consultation on the tube unplugging task yielded several recommendations for ALARA savings and general radiation safety. For example, about 2000 lead plugs were installed in an optimum pattern of tube ends; the radiation field near the tube was lowered by 10%. When the divider plate was electropolished, the radiation field near the divider plate was lowered by 20% or more. Near the center of the channel head bowl, the field was lowered by only 10%. These reductions were achieved in both legs of the channel head. In addition, there were a few hot spots at the base of the divider plate on the cold leg side. Thermoluminescent detector (TLD) measurements indicate that the field in the vicinity of these hot spots was reduced by a factor of four.

A training program was devised for the channel head workers so that they would perform safely and efficiently. A computer terminal installed in the SGEF at the radiation monitor's station allowed real-time tracking of personnel exposure during Task 8.

The computer program for tracking personnel exposures during the tube unplugging operation worked very well. Although about 70 personnel were involved, it was possible to call up the exposure history of any individual as it was needed. Exposure control of this many people was relatively simple. Maximum individual exposures were 1.87 rem for PNL and 1.27 rem for Westinghouse. Total PNL exposure (46 people) was 46.8 rem; total Westinghouse exposure (16 people), 15 rem. In addition to routine monitoring, special TLDs were placed on the heads, right and left wrists, front and back torsos, and right ankles of eight channel head workers. When these data are analyzed, exposures can be correlated with time-motion models of the work that was performed. Environmental monitoring in the vicinity of the SGEF continues to show a maximum exposure rate at the fence line of 0.7 mrem/h.

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

Software was developed to interface the computerized data acquisition system with NDT (mainly eddy-current) devices that provide data input. A PNL-designed, remotely operated, computer-controlled NDT probe pusher-puller will automatically provide NDT probe position information to the computer acquisition system in parallel with normal NDT data. The completed system was successfully tested and demonstrated, and it was installed in the 377 computer facility in March 1983. A software modification to make probe positioning more like a manual controller was completed and tested. A manual written for the computer operators included a calibration procedure for verifying accurate computer acquisition of the data. This procedure will be performed routinely. A topical report describing the system was issued.⁽²⁾

A draft of the historical data report is being prepared. 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. Color cross-sectional plots of the following data have been generated:

- plugging reason by year
- tubes inspected for denting by year
- size of eddy-current indications by year
- highest support plate passed for each probe size by year for the denting inspection program
- sludge heights before and after lancing treatments.

In addition, time plots were made of the blowdown chemistry data from the Westinghouse data base. The effects of the changeover in secondary side water chemistry from phosphate to AVT are readily apparent. Patterns of the different variables are related to outages and other events as documented in plant operating reports to the NRC.

Decontamination Channel Head (Task 6)

The channel head was decontaminated to minimize radiation exposure during primary side access. The availability of the channel head was an opportunity for research-related development/demonstration of decontamination techniques. Two subcontracts were let: the cold leg side was decontaminated by London Nuclear Services, Inc., using a proprietary Canadian process; and the hot leg side was decontaminated by Quadrex Corporation using another proprietary process that was developed in Great Britain. Both operations reduced 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.

Radiation levels were further reduced around the manway openings and on the divider plate by electropolishing. A detailed report of the entire decontamination effort has been completed in draft form for review by program participants.⁽³⁾

Base-Line Eddy-Current ISI (Task 7)

During this quarter, all PNL Task 7 equipment for the base-line study was installed and tested. A sub-contract with Zetec, Inc., was negotiated to provide personnel and equipment to perform the base-line study on a five 10-h shifts-per-week basis for about six weeks beginning the week of July 11, 1983. Two complete scans will be taken of all accessible tubes. A single-frequency scan at 400 kHz using the Zetec EM 3300 apparatus will duplicate ISIs taken at Surry and will identify changes that may have occurred since then. The other scan will use the multifrequency Zetec MIZ-12 apparatus, representing the current state of the art. Negotiations are still in progress with Intercontrol for a third 100% base-line ISI to be performed in the fall of 1983.

A cooperative effort between PNL and Oak Ridge National Laboratory (ORNL) to apply pattern recognition techniques to multifrequency eddy-current instruments is under way. ORNL will provide technical information on the eddy-current parameters that are important for detection and sizing, based on the physical principles of eddy-current technology. PNL will develop relationships between signal processing parameters and physical parameters. It will then be possible to use the signal processing technique of data processing to speed up the ORNL eddy-current analysis system. PNL will also develop decision-making algorithms for the ORNL processed data.

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.

All of the inner row U-bends have been characterized by borescopic examination. Of the 94 tubes inspected, 49 are either creased or cracked; some are opened on the top and bottom of the U-bend. There appears to be some of the same cracking in second row tubes, although this is difficult to observe. A direct correlation was observed with the physical appearance of the tubes. Every tube that was creased or cracked (plus another 17) was shiny in appearance, with pronounced flow patterns. None of the dull tubes (ones with corrosion product deposits) showed any evidence of mechanical damage. There was also an apparent correlation with position relative to the flow slots in the top support plate. The dull tubes were generally above the hard spots between flow slots, while the shiny ones were over the slots. What appear to be beads of resin matrix are present on the support plate; some beads will be removed for analysis. Similar observations have been made by others during inspections of other steam generators.

Probing through 2-in. diameter holes above the tube sheet has shown that the minimum spacing between tubes is 5/16 in. (0.79 cm). This dimension sets the maximum diameter for the new radiation-resistant fiberscope that will replace the one that darkened after 30-rem exposure. Quartz fiber fiberscopes, which were not available before, are now available up to 100-ft lengths (30.5 m). A small-diameter borescope is also being considered for this application. Photographic results using subminiature cameras through the four 2-in. holes just above the tube sheet have been highly satisfactory.⁽⁴⁾ The cameras, shown in Figures 3 and 4, are very inexpensive (only a few dollars each); and the cost of the 2-in. holes is also low (approximately \$500 each). Mechanical drilling of this size hole, with careful vacuuming of chips during drilling, minimizes the amount of debris on the secondary side. Furthermore, the debris is localized directly below each hole, and its carbon steel metallic nature makes it easy to identify and remove from secondary side samples such as sludge and corrosion products. The holes can be oriented to allow exploration between columns of tubes or between rows. The 2-in. diameter size assures access to at least one intertube space. Such holes are planned at strategic places around the steam generator shell: at the tube sheet, at each tube support plate, and in the U-bend region.

FIGURE 3. Subminiature Pinhole and Lensed Cameras for Intertube Photography in Steam Generator

FIGURE 4. Subminiature Cameras Mounted for Use in Steam Generator

Tube Sheet Section Removal (Task 11)

Earlier reports discussed removal of a large section of the tube sheet (~200 tubes) for analysis. Because this task is extremely difficult and expensive, it is now suggested that, at least initially, cores 1.5-in. in diameter by 22-in. long be removed at selected tube locations. The tubes would be internally cut above the tube sheet and removed with the core. These long cylinders of tube sheet and tube would then be removed to a laboratory for further sectioning and metallography. This method is considered more practical and economical, and it provides more flexibility to sample the tube sheet over a wider area. When there is insufficient clearance under a tube, it may be necessary to core out a section of the channel head bowl for access. Electrical discharge machining is a likely candidate process for this task. Other methods are being considered but may require more personnel exposure or larger quantities of radioactive waste.

MILESTONES

- The tube unplugging task was completed.
- Zetec, Inc., was selected as the contractor for the base-line eddy-current ISI.
- A topical report on Task 6 was issued.
- A report on radiological mapping in and around the steam generator (Task 3) was issued.

FUTURE WORK

- The base-line eddy-current ISI will be performed.
- The Task 4 historical data report will be issued.
- The Task 5 pre- and postshipment inspection report will be issued.
- An RFP for tube sheet sectioning will be issued.
- Task 9, the NDT round robin, activities will begin.

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16. ABSTRACT (200 words or less) This document summarizes work performed by Pacific Northwest Laboratory from April 1 through June 30, 1983, 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. Experimental data and validated models are being used to determine a method for evaluating the acceptance of welded or weld-repaired stainless steel 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. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada; and an instrumented fuel assembly irradiation program is being performed at Halden, Norway. Fuel assemblies and analytical support are being provided for experimental programs at other facilities, including fuel rod deformation and severe fuel damage tests for the Super Sara Test Program, Ispra, Italy; and experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho.				14. (Leave blank)	
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