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Reactor Safety Research Programs

Quarterly Report
January - March 1980

Prepared by C. M. Hagen, Ed.

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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January - March 1980

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Prepared by
C. M. Hagen, Ed.

Pacific Northwest Laboratory
Richland, WA 99352

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ABSTRACT

This document summarizes the work performed by Pacific Northwest Laboratory from January 1 through March 31, 1980, for the Division of Reactor Safety Research within the Nuclear Regulatory Commission. Evaluation of non-destructive examination (NDE) techniques and instrumentation are reported; areas of investigation include demonstrating the feasibility of determining structural graphite strength, evaluating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems, examining NDE reliability and probabilistic fracture mechanics, and assessing the remaining integrity of pressurized water reactor steam generator tubes where service-induced degradation has been indicated. Test assemblies and analytical support are being provided for experimental programs at other facilities. These programs include the loss-of-coolant accident simulation tests at the NRU reactor, Chalk River, Canada; the fuel rod deformation and post-accident coolability tests for the ESSOR Test Reactor Program, Ispra, Italy; the blowdown and reflood tests in the test facility at Cadarache, France; the instrumented fuel assembly irradiation program at Halden, Norway; and the experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory. These programs will provide data for computer modeling of reactor system and fuel performance during various abnormal operating conditions.

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GRAPHITE NONDESTRUCTIVE TESTING(a)

W. C. Morgan, Project Manager
T. J. Davis, Project Manager

SUMMARY

We have completed oxidation of, and all post-oxidation measurements on, the 3-in. dia PGX samples. Both the eddy-current and ultrasonic velocity apparatus, for measurement of the near-surface oxidation profile, have been assembled and tested. Unexpected observations on the results of oxidation versus depth in thick graphite samples have caused us to re-evaluate our future program.

INTRODUCTION

This is a continuation of previous work that demonstrated 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. The FY-1980 scope of this project is to:

1. Investigate and develop eddy-current techniques for providing interpretable indications of graphite strength.
2. Continue investigation and application of sonic velocity techniques to PGX and ATJ graphites.
3. Recommend an in-service monitoring technique by the end of FY 1980.

The objective of this investigation is to demonstrate the feasibility of using nondestructive testing techniques to determine structural graphite strength.

(a) RSR Fin. Budget No.: B2101-0; RSR Contact: R. D. Schamberger

TECHNICAL PROGRESS

Research to develop nondestructive tests for graphite oxidation is being conducted in three technical areas simultaneously: eddy-current testing, through-transmission ultrasonic velocity, and ultrasonic surface-wave velocity. Unexpected (but explainable) results have complicated the development of an algorithm for calculating the oxidation-depth profile from the near-surface eddy-current or ultrasonic-velocity measurements. This is discussed below under "Oxidation Profiles."

EDDY-CURRENT TESTING

Oxidation profiles have been simulated by stacking 3-in. sq and 0.050-in. thick oxidized plates of PGX graphite. Two coil designs and three frequencies have been used to obtain data on the response versus oxidation to a depth of about 0.150 inches.

THROUGH-TRANSMISSION ULTRASONIC VELOCITY

Longitudinal- and shear-wave velocities were measured, in both the axial and diametral directions, on nine oxidized samples of PGX. Because the ultrasonic signal was highly attenuated, maximum oxidation was limited to 14.4% weight loss. Compressive strength measurements were also obtained. The results confirm previous results on other graphites, which show that there is a strong correlation between density and strength.

ULTRASONIC SURFACE-WAVE VELOCITY TEST

A surface-wave test apparatus has been developed under subcontract and is nearly ready for use. It employs one transmitter and two receivers that are coupled ultrasonically to the graphite with RTV coupling members. The test frequency can be varied for control of penetration depth to obtain oxidation profile information. Initial testing shows that coupling is much better than has been obtained with our previous apparatus and that the graphite sustains a large amplitude surface wave.

OXIDATION PROFILES

We had assumed, and to the best of our knowledge all previous investigators have assumed, that the oxidation rate would decrease as a continuous analytic function of distance from the surface.^(a) However, visual examination of oxidized blocks of PGX (6.5 in. x 7.5 in. x 4.5 in.) indicates that there is an area of increased porosity below the surface. The area and distance from the surface appears as an inverse function of the oxidation temperature; physical measurements of density confirm the visual indications that there was an increase in oxidation rate with penetration depth.

Conversations with W. P. Eatherly of Oak Ridge National Laboratory (ORNL), and visual examination of parts of our oxidized 3-in. dia samples, confirmed that this unexpected phenomenon is not isolated to PGX graphite, nor is it isolated to samples oxidized in a mixture of CO_2 and CO .

It appears most likely that the decrease in concentration of the oxidant and increase in concentration of the oxidation-product(s) produces a change in the chemical state (and, thus, a change in the catalytic activity) of one or more of the impurity elements. This phenomenon has been observed in ATJ (the FSVR core support post) graphite, as well as in PGX (the FSVR core support block) graphite.

The implication, from both the Pacific Northwest Laboratory and ORNL observations, is that one could observe, and measure properties of, a hollow shell of a graphite component. We are sure that this condition does not now exist in FSVR; but, because of the possibility that this condition might develop, we believe it warrants further investigation. In order to gain a more thorough understanding of the phenomenon, we have diverted part of our current funding in an effort to identify the catalytic impurity elements. We have also proposed a more comprehensive investigation (beginning in FY 1981), which would include extension of the current computer code technology used in GOP-4^(b) to allow prediction of oxidation profiles from reactor operating

(a) This assumption seems logical based on the fact that the concentration of the oxidant will decrease and the concentration of the oxidation products will increase with increasing penetration depth.

(b) Graphite Oxidation Program, Version 4 - developed at Battelle-Northwest.

data. We have also proposed the investigation of three techniques for directly obtaining the in-depth density profile of thick graphite components.

FUTURE PLANS

Remaining work in the three areas of investigation is as follows:

EDDY-CURRENT TESTING

- Complete theoretical analysis for profiling oxidation with multiparameter eddy-current data.
- Develop and test profiling algorithm.

THROUGH-TRANSMISSION ULTRASONIC VELOCITY

- Evaluate high-power test for propagation over long paths in attenuative graphite specimens.

ULTRASONIC SURFACE-WAVE VELOCITY

- Perform surface-wave velocity tests at selected frequencies.
- Perform destructive determination of oxidation profile.
- Correlate data.

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

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

SUMMARY

The test on the specimen of heavy section cylindrical bend (HSCB) was halted temporarily to permit analysis of existing data and planning for the remainder of the test. Analysis has shown that a pattern recognition technique similar to that previously developed for analysis of data from a laboratory specimen can identify crack growth acoustic emission (AE) from the bend specimen.

A vessel test being performed by Materialprüfungsanstalt (MPA), Stuttgart, West Germany, has been selected for evaluation of AE monitoring techniques under simulated reactor conditions. This will be a cooperative effort between Pacific Northwest Laboratory (PNL), the Nuclear Regulatory Commission (NRC), and MPA; it is designed to meet the objectives of both parties.

A monitoring concept to evaluate identification of AE and determine flaw severity on a vessel test has been developed and the design has been started.

High-temperature AE sensors were installed on the discharge face of "N" Reactor to test the sensors' resistance to an actual reactor environment.

INTRODUCTION

The purpose of this program is to provide an experimental evaluation of the feasibility of using AE on a continuous basis to detect and analyze flaw growth in reactor pressure boundaries. Type A533B, Class 1 steel is being used in all experimental testing. The objectives of this program are to:

- develop a method to identify crack growth AE signals in the presence of other acoustic signals
- develop a relationship to estimate flaw significance from AE data

(a) RSR Fin. Budget No.: B2088; RSR Contact: J. Muscara

- develop an instrument system to implement these techniques
- demonstrate the total concept-off-reactor and on-reactor.

Progress relative to these objectives is discussed in the following sections, which include AE/fracture mechanics testing, off-reactor vessel test, AE monitor system development, high-temperature sensors, and reports. The final section describes the work planned for next quarter.

AE/FRACTURE MECHANICS TESTING

Fatigue crack growth testing of the HSCB specimen continued this past quarter. Additional cycling in room-temperature air was performed until an exterior fatigue crack was discovered in the transition radius between the rectangular and cylindrical portions of the test specimen (see Figures 1 and 2, Reference 1). The test was shut down in mid-January while plans were developed to maximize the data yield for the remainder of the test. In support of this effort, analysis of all of the existing data was begun.

Initial pattern recognition results for characterization of crack growth AE from the bend specimen were based on 57 selected waveforms from a high-temperature, surface-mounted AE sensor. The signal sources included:

<u>Waveform Type</u>	<u>No. of Signals</u>
Valid AE	36
C-Clamp	3
Tapping	14
Pentel	4

The format of the pattern recognition analysis followed the one used in earlier work.⁽²⁾ The waveform features generated were:

Mean
Standard Deviation
Skewness
Kurtosis

- Autocorrelation at Lag 13
- Autocorrelation at Lag 37
- Maximum Frequency Response
- Frequency of the Maximum Response
- Total Power (0 Hz to 1 MHz)
- First Moment of Power Spectrum
- Second Moment of Power Spectrum

With only 57 waveforms, the complete analysis using training and test sets would have little meaning; therefore, the analysis included only the following steps necessary to produce a decision rule:

- Feature Scaling
- Feature Reduction and Orthogonalization
- Least-Squares Decision Rule

The features were autoscaled, and a correlation to property feature reduction algorithm was used. The five features that contained the largest amount of the signal-source information were:

- Kurtosis
- Autocorrelation at Lag 37
- Mean
- Total Power
- First Moment of Power Spectrum

These five features were orthogonalized and fed into the least-squares fitting algorithm to produce a decision function. The value of the decision function is plotted in Figure 1. Waveform number is listed on the abscissa and is arbitrary. The horizontal line represents the value of the decision rule: all points above the line are classified as noise; those below the line, valid AE. The decision rule was located to minimize misclassification of valid AE waveforms as noise. As a result, only one noise was misclassified and no valid AE were misclassified. When the test is resumed, additional waveform data will be gathered to use as a test set for the decision function.

FIGURE 1. Pattern Recognition Segregation of Bend Specimen Acoustic Data

Part of the continuing bend specimen testing will be to test the AE detection and pattern recognition techniques under simulated reactor noise conditions. An existing valve was used to throttle the water flow to the specimen and create simulated reactor flow noise. Initial results show that the frequency spectrum of the noise was reasonably good but the amplitude was low. Installation of a ball valve to obtain the desired amplitude/frequency noise distribution is being considered.

OFF-REACTOR VESSEL TEST

Work continued on the development of a plan to measure and analyze AE data from fatigue crack growth in a heavy section (4 to 6 in.) steel pressure vessel during simulated reactor operation. The original test matrix,⁽¹⁾ which was

based upon a single flaw and many pressure cycles, has been modified. The new test matrix is based upon a limited number of pressure cycles with several flaws of different geometries to obtain a broad range of fatigue crack growth rates.

Efforts have been concentrated on selecting an appropriate site for conducting the vessel test. In addition to the two locations mentioned previously,⁽¹⁾ a third option involves an invitation to PNL by representatives from MPA, Stuttgart, West Germany, under the auspices of the NRC-BMFT Safety Research Exchange Agreement to participate in tests their laboratory is planning to perform on two intermediate-scale pressure vessels,^(a) beginning in early FY-1981. Detailed time and cost estimates were prepared to evaluate the suitability of each testing site in the following categories: 1) vessel preparation, 2) test system preparation, and 3) test performance. The results of these estimates indicate that participation in the MPA Stuttgart tests represents the best overall option.

Since the MPA test vessels are made from A508 steel and this program is focused on A533B, Class 1 steel, the current plan is for PNL to supply an insert of A533B steel with machined and fatigue-presharpened flaws to be welded into the first MPA pressure vessel. The insert will contain three part-circular surface flaws, two on the inside surface of the insert and one on the outside surface. The longitudinal axis of the flaws will be oriented perpendicular to the circumferential stress in the vessel. PNL will perform the AE monitoring and supply all instrumentation needed. Vessel preparation and testing will be performed by MPA. The second MPA vessel will include a defected nozzle, for which monitoring of AE is also planned. Detailed test plans are to be jointly developed by PNL and MPA to satisfy the needs of both laboratories.

AE MONITOR SYSTEM DEVELOPMENT

Design work was begun on a system to test the AE/flaw relationships and pattern recognition techniques under simulated reactor conditions. The planned approach for monitoring the first MPA vessel test involves two multiple sensor

(a) PNL will be participating in the tests

arrays. One array will be located totally on the A533B steel insert to optimize AE detection and source location. The second array will be located to monitor the total cylindrical portion of the vessel. This will provide a more realistic simulation of reactor application. By correlating response of the two arrays to given AE signals, the effects of geometry and distance on the signal and the resulting influence on AE identification and interpretation methods can be investigated. However, it requires a system that can simultaneously acquire data from two arrays. We have determined that the existing NRC-owned, computerized AE monitor system can be upgraded to perform this function and that the modification can be completed within the time limits (by mid-August). Details of the upgrade requirements will be reviewed with NRC as soon as they are available and before action is initiated. The data obtained by this data acquisition system will be analyzed by a system that will be developed at PNL. The system will receive the data, record the signal waveforms digitally processing the waveforms through a pattern recognition algorithm, and categorize signals into "crack growth AE" and "other signal" by pattern recognition. The listing will then be correlated with source location information through common signal identification numbers resulting in a net record of crack growth AE and its source. With this information, an AE/flaw growth relationship can be applied to estimate flaw severity.

HIGH-TEMPERATURE SENSORS

High-temperature sensors were installed on the discharge face piping at N Reactor. These sensors will remain for at least one year to permit evaluation of their performance in the combined environment of high temperature and nuclear radiation. The sensors were installed at the end of February. They will be removed, tested, and replaced when possible during the test period. The sensors were tested previously for 5500 h at 550⁰F in a laboratory oven.

REPORTS

The following reports were prepared during this period:

- Quarterly progress report for the October 1 to December 31, 1979, period

- A report entitled, "Leak Detection and Crack Detection in Light Water Nuclear Reactor Systems by Acoustic Emission Methods."

WORK PLANS FOR NEXT QUARTER

Work plans for the period April 1, 1980, to June 30, 1980, include:

- completing the HSCB specimen test
- initiating fabrication of the A533B steel insert to be welded into the first MPA test vessel
- preparing detailed test plans for monitoring the MPA vessel tests and visiting MPA to obtain concurrence
- preparing analysis-before-test and topical summary reports
- continuing the design and fabrication of the AE monitor system.

REFERENCES

1. P. H. Hutton and R. J. Kurtz, "Acoustic Emission-Flaw Relationships for In-Service Monitoring of Nuclear Pressure Vessels," in Reactor Safety Research Programs Quarterly Report, October 1 - December 31, 1979, NUREG/CR-1349, Pacific Northwest Laboratory, Richland, WA 99352.
2. P. G. Doctor, T. P. Harrington, and P. H. Hutton, "Pattern Recognition Methods for Acoustic Emission Analysis," NUREG/CR-D910, PNL-3052, Pacific Northwest Laboratory, Richland, WA 99352, July 1979.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS^(a)

F. L. Becker, Program Manager
S. H. Bush, Project Manager
F. A. Simonen, Project Manager
S. R. Doctor
G. B. Dudder
P. G. Heasler
G. P. Selby

SUMMARY

Measurements performed during the quarter define the influence of flaw aspect ratio and the ineffectiveness of current code requirements for beam overlap. These measurements also indicate that use of Appendix III of the 1977 edition of the ASME Code rather than the Summer 1974 edition of the code will result in a sensitivity decrease of 5 to 16 dB.

Preparation for the round-robin inspections to be initiated in July is proceeding satisfactorily; however, A106 carbon steel pipe samples have not been delivered, which has delayed progress. Initial round-robin tests will proceed without the A106 pipe; that part of the test matrix will be completed at a later date.

INTRODUCTION

The Nuclear Regulatory Commission (NRC) has funded a number of research programs centered around the ultimate goal of minimizing cracks in nuclear reactor primary piping systems. In order to detect flaws, which could lead to cracks, utilities currently inspect these piping systems in service according to the requirements of specific sections of ASME Code. These inspections are performed periodically on a sampling of pipe joints 1) to determine whether cracks are growing in service and 2) to identify whether there may be

(a) RSR Fin. Budget No.: B2289-0; RSR Contact: J. Muscara.

generic-type defects in such plants and systems. Currently, NRC is sponsoring a number of programs to establish both the credibility of in-service inspection (ISI) and requirements necessary to assure continued operating safety. NRC expects that these programs and implementation of their recommendations will provide assurance that ISI will help prevent catastrophic pipe rupture.

The Integration of NDE Reliability and Fracture Mechanics Program was established to determine and improve the reliability of using current in-service techniques to inspect primary piping systems. The objectives of the program include:

- determining the reliability of ultrasonic ISI as used to inspect commercial light water reactor primary piping systems
- determining through fracture-mechanics analysis the impact non-destructive examination (NDE) unreliability would have on system safety and determining the level of inspection reliability that is required for a suitably low failure probability
- evaluating the degree of improvement in reliability that could be achieved by using improved and advanced NDE techniques
- recommending revisions to pertinent code and regulatory requirements to assure continued system operating safety.

Failure history has shown that serious flaws found in service are connected with the I.D. surface. Therefore, this study is directed toward detecting the size and location of the defects (particularly cracks) connected with this surface, with limited emphasis on internal fabrication flaws.

TECHNICAL PROGRESS

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

TASK 6: SAMPLE FABRICATION

Samples are being prepared for the round-robin inspections, Task 8. The six 10-in. Schedule 80S and the two 27.5-in. ID centrifugally cast stainless

steel pipes have been welded. The two 33-1/4-in. O.D. A106 carbon steel pipes have been delivered and are being welded. Completion of welding and cladding of the A106 pipes is expected by June 6, 1980. Cracking of the other pipes has been initiated.

Technique development for thermal-fatigue cracking of the welded pipe samples has now been completed. Thermal-fatigue cracks of the desired sizes have been produced in the pipe samples. Initially, there was a concern as to whether flaws of the required depth (1.0 to 1.25 in.) could be grown in the two heavy-wall pipes. This problem was solved by enlarging the diameter of the cooling nozzle from 1.25 to 2.5 in. The larger nozzle requires a longer cycle time (90 sec compared to approximately 40 sec for the smaller nozzle).

TASK 7: MEASUREMENT AND EVALUATION

Efforts this quarter were directed toward defining the influence of flaw aspect ratio on ultrasonic response and the adequacy of the 10% scan overlap requirement of ASME Section XI, Appendix III, and comparing side-drilled holes and end-mill notches as calibration reflectors. As expected, flaws of aspect ratio 0.5 (depth/length) of unacceptable depths were not detectable using the calibration sensitivity specified by Appendix III. The 10% beam overlap was also found to be inadequate to assure acceptable coverage. Side-drilled holes used for calibrations of inspections performed to the Summer 1974 Addenda of the ASME Code were compared to 10% notches required by the Winter 1977 Revision of the Code. The initial conclusion is that inspections performed to 1977 Code are from 5 to 16 dB (depending on transducer and pipe size) less sensitive than inspections performed to the 1974 Code.

Aspect Ratio

All flaw geometry results reported to date were acquired from inspection of machined flaws having aspect ratios of roughly 0.2. To investigate the effect of changing flaw length, normal vertical flaws were fabricated in the same depth range (40 to 300 mil) with aspect ratios of 0.5 and 0.1. The flaws with aspect ratio 0.1 are saw cuts, and the flaws of 0.5 aspect ratio (semi-circular) are electric discharged machined slots; all are fabricated in 0.6-in. thick 304 stainless steel plate.

The new flaws were inspected using a selection of transducers representative of field units and a variety of beam paths. The beam paths were 45° $1/2$ V, full V, and $3/2$ V, and 60° $1/2$ V. In inspecting machined flaws of aspect ratios 0.5, 0.2, and 0.1, the semicircular flaws (aspect ratio 0.5) are usually much lower in response than the longer flaws, especially at low depth. This points out the fallacy of calibrating on a reflector of (usually) effectively infinite length, when Section XI also requires detection of very short flaws of approximately the same depth. Figure 1 indicates that while three out of the four transducers used produced indications higher than 50% distance amplitude correction (DAC) from the minimum rejectable semicircular flaws, two of the three were just barely above 50% DAC. Since a service defect would not have the ideal reflecting surfaces of the machined flaws, one has to conclude that the same size flaw grown in service would probably not be detected by these transducers. Thus, in service, only the 1/4-in. transducer would have detected the rejectable 75-mil semicircular flaw with a 50% DAC recording criterion.

Scan Overlap

According to paragraph III-2410 of Section XI, Appendix III, scans must overlap by a minimum of 10% of the transducer dimension, measured perpendicular to the scan path. This minimum requirement was evaluated by measuring how far apart scans could be made, while assuring that the flaw response would exceed a specified recording level. This experiment was performed for the 60-mil vertical flaws of 0.5 and 0.2 aspect ratio. Except for the 45° full V, transducers and beam paths used for the aspect ratio study were also used in this evaluation.

The signal is maximized at the desired beam path, and then the transducer is moved perpendicular to the beam direction (sideways) until the signal drops to a certain level relative to DAC, where the transducer position is recorded. The positions of responses 6, 14, and 20 dB below DAC were recorded on both sides of the position of maximum signal, thus defining the maximum scan separation for a certain detectability level.

FIGURE 1. Ultrasonic Response from 0.5 Aspect Ratio Notches in 0.6-in. Thick Stainless Steel Plate Using 1/4-, 1/2-, and 1.0-in. dia, 2.25-MHz Transducers

The results are shown in Table 1. The maximum allowable separation between adjacent scans is expressed in percent of the crystal dimension. When the separation in percent of crystal size is less than 90, the Section XI scan overlap criterion of 10% is inadequate. Observe that the 10% overlap is often insufficient to provide coverage with even a 20% DAC recording criterion.

TABLE 1. Required Inspection Coverage for Detection of 10% Throughwall Flaws

Transducer	Beam		Percent Crystal Dimension					
	path	angle	-6dB(a)	-14dB(a)	-20dB(a)	-6dB(b)	-14dB(b)	-20dB(b)
1/4" 2.25 MHz	1/2	45	36	96	128	100	144	176
	3/2	45	--(c)	152	228	92	176	224
	1/2	60	--	72	112	60	108	140
1/2" 2.25 MHz	1/2	45	--	38	68	40	76	100
	3/2	45	--	56	88	46	84	110
	1/2	60	--	28	54	--	60	84
1" 2.25 MHz	1/2	45	--	--	47	--	50	75
	3/2	45	--	--	45	--	53	78
	1/2	60	--	--	59	--	53	81
3/8"x3/8"(2) dual 1.5 MHz	1/2	45	--	72	96	53	101	128
	3/2	45	--	109	176	67	147	184

(a) Aspect ratio = 0.5

(b) Aspect ratio = 0.2

(c) -- indicates maximum flaw signal below detection criterion

Table 1 also indicates that the required overlap for a given detectability cannot be defined by a single number that will apply to all search units and beam paths. The numbers in the table will also change with pipe-wall thickness and curvature. The obvious inference is that this procedure should be required as part of calibration. Since the smallest surface-connected defect that Section XI requires the inspector to detect is a semicircular flaw with depth 12.5% of wall thickness, one might calibrate on this reflector. The recording level for inspection indications would be set a few dB below a DAC thus established, to allow for the fact that the surface of a service defect would not be as ideal a reflector as the machined surface of the calibration reflectors.

The added value of this semicircular calibration reflector would be the capability of measuring the necessary scan overlap for each inspection. This measurement is neither complex nor time consuming; it also provides assurance of complete coverage.

Side-Drilled Holes Versus Notches

Recent acceptance by NRC of ASME Section XI Code through Winter of 1978, has raised a question as to the adequacy of the inspection sensitivity as defined in Appendix III of Section XI. Previously, most piping inspections were performed or calibrated using the side-drilled holes specified by ASME Section V, Article 5, whereas Appendix III requires the use of surface-connected notches. A program has been initiated to define the difference in inspection sensitivity produced by these two calibration techniques. Limited data is available at this time; however, there is a clear trend indicating notch calibrations yield less sensitive inspections.

TASK 8: ROUND-ROBIN INSPECTIONS

Preparation for round-robin inspections scheduled for initiation in July are proceeding. Review of the procedures submitted by the test teams is in progress as a part of finalizing the round robin test protocol. The "Search Unit Tracking and Recording System" (SUTARS) has been ordered and is scheduled for delivery June 1, 1980. SUTARS will be used to automatically record test data for later error analysis. Use of intergranular stress corrosion samples is being negotiated with the Electric Power Research Institute. We expect to be able to use at least four of these pipes for the round-robin tests.

The statistical test matrix and analysis procedures have been defined. Randomization of flaw locations has also been completed to allow initiation of sample cracking procedures. The statistical analysis procedures will supply information pertaining to correct flaw identification as well as the false rejection probability.

As a result of the late delivery of the 33-1/4-in. A106 carbon steel pipe and a strike involving the welding contractor for this sample, the schedule for obtaining these samples has been delayed. These samples will not be available

for the initial teams. These teams will be called back at a later time to complete the test matrix. It is expected that the A106 samples will be available by mid July.

FUTURE PLANS

The primary emphasis next quarter will be on completing cracking for the available samples and defining the round-robin test protocol and contractual negotiations. Measurements of side-drilled hole (and notch) calibration sensitivities also should be completed during the quarter. An ample data base establishing the relationship of side-drilled hole and notches as calibration reflectors probably will be acquired during the next quarter.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

TASK A - IRRADIATION EXPERIMENTS(a)

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

SUMMARY

Irradiation of IFA-432 and IFA-513 continued. Some instrument failure was experienced. Final arrangements were made to discharge and ship IFA-432 rods 7 and 8 to Harwell for mechanical compliance testing. Fuel assembly IFA-527 is being constructed; it could not be loaded this quarter, however, because of mechanical difficulties. Pretest predictions for this assembly have been made.

INTRODUCTION

The Experimental Support and Development of Single-Rod Fuel Codes Program is a continuation of the Experimental Support and Verification of Steady-State Codes Program (begun in 1974). The current program's general objectives include collecting and analyzing in-reactor data on fuel rod temperatures, fission gas release, and cladding elongation as a function of irradiation history; correlating post-irradiation examination with the in-reactor data; utilizing ex-reactor testing for a better understanding of fuel rod mechanical behavior; and integrating all of this information into the FRAPCON and FRAP-T series of codes used in fuel modeling.

The irradiation tests are being conducted in the Halden Boiling Water Reactor (BWR), Norway. One test (IFA-431) has been completed. Currently, two other test assemblies are being irradiated (IFA-513 and IFA-432). The fourth and final test in the program (IFA-527) is under construction and its irradiation will begin in June 1980.

(a) RSR Fin. Budget No.: B2043; RSR Contact: H. H. Scott

Task A of this program is concerned with conducting the irradiation tests and coordinating the necessary activities associated with testing (test rig design, safety analysis, fuel shipments, post-irradiation examination procedures, fuel disposal, etc.).^(a)

TECHNICAL PROGRESS

Irradiation of the IFA-432 and IFA-513 fuel assemblies continued uneventfully this quarter. However, both assemblies experienced further instrument failure. The instrumentation status is summarized in Table 1.

Construction on IFA-527 is now proceeding. Difficulty with the thermocouple seals prevented loading of the assembly during this quarter. The new target date is the last week of June.

Pretest predictions for IFA-527 have been made with the FRAPCON-2 code that is currently being developed jointly by PNL and EG&G Idaho, Inc. Fuel-stored energy uncertainty obtained through various fuel relocation models is about 30%, as a result of variations in the estimated gap size and fuel surface temperatures. Based on observations of scram behavior of similar rods from IFA-505, we hope to be able to use transient temperature from IFA-527 to resolve the uncertainty in fuel stored energy for these test rods, and hence for the rods they simulate (fission-gas saturated, high-burnup BWR rods).

FUTURE PLANS

The following are this task's planned major events for next quarter:

- startup of IFA-527
- shipment of Rods 7 and 8 (IFA-432)
- compliance testing of Rod 8 at Harwell, UK.

(a) Considerable coordination is required because post-irradiation examination will be performed at Kjeller, Norway, and Harwell, UK, and all test fuel rods will be returned to the United States either for transient testing or disposal.

TABLE 1. Instrument Status(a)

<u>Assembly Rod</u>	<u>Pressure Transducers</u>	<u>Upper Thermocouples</u>	<u>Lower Thermocouples</u>	<u>Elongation Sensors</u>	<u>Neutron Detectors</u>
Rod 1	R	X	R	X	All satisfactory except co-detector
Rod 2	-	-	X	R	
Rod 3	-	U	R	X	
Rod 4 (8)	-	-	-	R	
Rod 5	R	X	R	X	
Rod 6	X	X	X	X	
<u>IFA-513</u>					
Rod 1	R	R	R	R	All satisfactory
Rod 2	X	X	R	R	
Rod 3	R	X	R	R	
Rod 4	R	R	R	X	
Rod 5	X	X	X	R	
Rod 6	X	R	R	R	

(a) Symbols: R = Reliable
 U = Unreliable
 X = Not Functioning
 - = Not Installed

(a) Symbols: R = Reliable
 U = Unreliable
 X = Not functioning
 - = Not installed

Assembly Rod	Pressure Transducers	Upper Thermocouples	Lower Thermocouples	Elongation Sensors	Neutron Detectors
IA-513					
Rod 1	R	R	R	R	All satis- factory
Rod 2	X	X	R	R	
Rod 3	R	X	R	R	
Rod 4	R	R	R	X	
Rod 5	X	X	X	R	
Rod 6	X	R	R	R	
IA-512					
Rod 1	R	R	R	R	All satis- factory
Rod 2	X	X	R	R	
Rod 3	R	X	R	R	
Rod 4	R	R	R	X	
Rod 5	X	X	X	R	
Rod 6	X	R	R	R	
IA-511					
Rod 1	R	X	R	X	All satis- factory except co- detection
Rod 2	-	-	X	R	
Rod 3	-	U	R	X	
Rod 4 (B)	-	-	-	R	
Rod 5	R	X	R	X	
Rod 6	X	X	X	X	

TABLE 1. Instrument Status (a)

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

TASK B - DATA QUALIFICATION AND ANALYSIS(a)

D. D. Lanning, Program Manager
M. E. Cunningham, Task Leader
E. R. Bradley
W. N. Rausch
R. E. Williford

SUMMARY

This quarter, a data tape covering the period October 1979 - January 1980 was received and corrected, two reports and one paper were published, and work continued on the four major areas of analysis: fuel relocation, transient temperatures, error propagation, and fission gas release.

INTRODUCTION

A major objective of the Experimental Support and Development of Single-Rod Fuel Codes is the irradiation of instrumented fuel assemblies (IFAs) to obtain well-characterized data. Task B of this program is responsible for qualifying and analyzing this data.

This task is divided into three subtasks:

- Data Processing - includes receiving, correcting, characterizing, and presenting the data obtained from the fuel assemblies.
- Data Reports - includes preparing reports on the precharacterization of the fuel assemblies, the data obtained from the assemblies, and the post-irradiation analysis of the assemblies.
- Data Analysis - includes providing in-depth analysis of the in-reactor fuel rod data obtained. Specific areas of interest for FY 1980 are analysis of data for inferring fuel relocation and its effect, use of transient temperature data to better understand fuel

(a) RSR Fin. Budget No.: B2043; RSR Contact: H. H. Scott

behavior, analysis of statistical variations and error propagation, and analysis of fuel rod fill-gas pressure data for inferring fission gas release.

TECHNICAL PROGRESS

This quarter's activities are discussed below and are organized by subtasks.

SUBTASK B-1: DATA PROCESSING

During this quarter, one data tape covering the period October 1, 1979 - January 6, 1980 was received, corrected, and made available for analysis.

SUBTASK B-2: DATA REPORTS

A report discussing the pre-irradiation characterization of IFA-513 was published this quarter (Bradley et al. 1979). This report discusses the objectives and design of IFA-513, and presents data on fuel, cladding, and instrumentation.

SUBTASK B-3: DATA ANALYSIS

This quarter, substantial work was completed involving analysis of data in these areas of interest: fuel relocation, transient temperature response, modeling and data errors, and fission gas release. Details of this work follow.

Fuel Relocation Data Analysis

A formal report on the basic concepts used in fuel relocation and mechanical response modeling is now being prepared for publication (Williford et al. 1980).

The primary conclusions can be summarized as follows:

- (1) Fuel cracking and relocation cause a redistribution of the free area (initial gap) so that part is occupied by the cracks and part is occupied by the gap.

- (2) The redistribution of free area (formation of cracks) causes power-dependent reductions in the effective thermal conductivity and the effective elastic moduli of the cracked fuel column.
- (3) Equilibrium requires constant contact between the cracked fuel and the cladding; thus, the traditional solid cylinder model is not adequate for predicting fuel response.
- (4) The primary thermal-mechanical feedback is due to crack closure effects on the fuel thermal conductivity, rather than to the gap closure effects on gap conductance.

The primary thrust of the Pacific Northwest Laboratory fuel relocation and mechanical response model is to relate the thermal and mechanical behaviors of the fuel rod via a simple description of the cracked fuel geometry. The three basic parameters in this description are: the crack pattern (total crack length), the effective crack roughness (microroughness and waviness of the fuel fragment surfaces), and the effective gap roughness (fuel surface microroughness and the macroroughness caused by the misalignment of the fuel fragments).

Figure 1 illustrates the validity of the assumption that the cracked fuel and cladding are in constant contact. The gap size versus power is plotted for both the solid cylinder and for the cracked fuel models. Note that the solid cylinder model includes only thermal expansion, and no fuel relocation is assumed. Since the effective gap roughness is greater than the calculated gap size of the cracked fuel model, it is evident that the fuel and cladding are in constant contact.

There are also strong indications that the fuel rod mechanical response to power changes is dominated by the effective radial elastic modulus of the cracked fuel at low powers and by the effective axial modulus at higher powers.

The use of the cracked fuel model in FRAPCON-2 is shown in Figures 2 and 3. The improvement in the simultaneous prediction of thermal and mechanical response is apparent.

FIGURE 1. Fuel-Cladding Gap Sizes Inferred from Data via Both Cracked Fuel and Solid-Cylinder Models

Transient Temperature Data Analysis

On August 23, 1979, a deliberate full power SCRAM was performed at the Halden Boiling Water Reactor. During this last quarter, thermocouple data from that SCRAM was analyzed for the purposes of deducing thermocouple decalibration and inferring fuel rod thermal-mechanical behavior.

Initial observation of the SCRAM data from rods 1 and 3 of IFA-432 (20 GWd/MTM burnup) showed a thermal response that was slower than expected when compared to similar rods from IFA-513 (8 GWd/MTM) and IFA-505 (4 GWd/MTM). This is indicative that the measured resistance for the IFA-432 rods is lower than reality; i.e., as a result of thermocouple decalibration, the measured temperatures were lower than reality. By comparing a small computer code (Cunningham, Lanning and Montgomery 1979) with the data, it is possible to estimate the amount of decalibration that has occurred. Figure 4 illustrates that a decalibration of approximately 20% is necessary for the transient temperature calculator to match the data.

FIGURE 2. Fuel Center Temperatures for Rod 6, IFA-513

In previous analysis of 20% power drops (Lanning and Cunningham 1979), a "cracked pellet" model of fuel behavior has consistently given better results than the conventional "solid pellet" model. These two views of fuel rod heat transfer have also been compared against the SCRAM data. As seen in Figure 5,

FIGURE 3. Cladding Elongations for Rod 6, IFA-513
(Corresponds to the temperature data
in Figure 2)

the cracked pellet model gives significantly better results over the entire SCRAM time range than does the solid pellet model. This means that during the SCRAM, the gap conductance is approximately constant (the fuel does not pull away from the cladding), while the effective fuel thermal conductivity decreases with decreasing power (internal fuel cracks open up as temperature decreases).

Modeling and Data Error Analysis

The objective of this work is to evaluate errors and uncertainties in computer code predictions of steady-state fuel behavior. The current work concentrates on evaluating stored energy uncertainty for different fuel rod designs (boiling and pressurized water reactors), examining the problem of

FIGURE 4. Comparison of Calculations to Data for Rod 1, IFA-432 (230 μm gap)

validating theoretical models and computer applications, and evaluating the effect on temperature prediction uncertainty when using a cracked fuel model.

During this last quarter, a paper discussing the use of linear propagation of errors, as applied to steady-state fuel performance codes, was published by Nuclear Technology (Cunningham, Hann and Olsen 1980).

The rod-type comparison has shown, as expected, differences in the thermal resistance between rod designs. However, uncertainty analysis has shown that

FIGURE 5. Comparison of Cracked and Solid Pellet Models to Scram Data for a High-Resistance Rod (IFA-5D5, TF6)

the beginning-of-life relative uncertainty for centerline temperature and stored energy is nearly the same for all designs (if the same input uncertainties are applied to each design). Figure 6 is a parametric plot based on this conclusion. This figure presents relative uncertainty for centerline temperature and stored energy as a function of power uncertainty. The three different curves are based on different uncertainties for fuel thermal conductivity, specific heat, and flux depression. Notice that when the uncertainty for power

FIGURE 6. Centerline Temperature and Stored Energy Uncertainty as a Function of Power Uncertainty

exceeds 20%, the uncertainties for centerline temperature and stored energy are nearly linear with power uncertainty. This indicates the strong dependence that these two variables have upon power.

Fission Gas Release Data Analysis

The interim report on the internal pressure data from test assemblies IFA-431/432 (Bradley et al. 1979) was published this quarter. The report analyzes the pressure data taken through January 1979, estimates the gas release as a function of fuel burnup, and compares the estimated gas release to the predictions from three gas release models (Beyer 1975; Rausch and Panisko 1979; Reymann 1978) that are currently being used in Nuclear Regulatory Commission audit codes.

As expected, the fission gas release was found to be directly related to the fuel temperatures. For the highest temperature rod (Rod 6, IFA-432), the

maximum estimated release fraction was 15% at approximately 1100 GJ/kgU (13 GWd/MTM). The fuel temperatures decreased following this burnup and the maximum estimated release fraction decreased to 13% at approximately 1700 GJ/kgU (20 GWd/MTM) in Rod 6, IFA-432. The maximum estimated gas release fractions for rods 1 and 5 of IFA-432 were less than 10% at 1700 GJ/kgU.

The primary conclusions of the analysis are as follows:

- Fuel temperature is the predominant factor in fission gas release.
- Fuel density has only a minor influence on fission gas release.
- Burnup-enhanced fission gas release is not significant for exposures up to 1700 GJ/kgU.

Additional pressure data has now been obtained from rods 1 and 5 of IFA-432 up to a burnup of approximately 1900 GJ/kgU (22 GWd/MTM). The estimated release fractions deduced from these data suggest no significant increases in gas release have occurred during this period. The pressure transducer in rod 6, IFA-432 failed during this operating period and no additional data were obtained.

FUTURE PLANS

The work planned for next quarter includes the following:

DATA REPORTS

A report, The Analysis of Fuel Relocation for the NRC/PNL Halden Assemblies IFA-431, -432, and -513, written by R. E. Williford et al., April 1980, will be published as NUREG/CR-0588 (PNL-2709).

DATA ANALYSIS

- The pretest predictions for IFA-527 will be completed.
- Data from IFA-432 and IFA-513 at higher burnups will be analyzed.
- The effects resulting from axial power profile shapes will be quantified.

- A qualitative statement will be prepared regarding the process of validating theoretical models and computer applications.
- Fuel thermal conductivity uncertainty for cracked fuel pellets will be examined.

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EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES

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

O. D. Lanning, Program Manager
R. E. Williford, Task Leader
W. N. Rausch
M. E. Cunningham
J. Hofweber
E. Greenwell

SUMMARY

Refinements in the relocation model have been implemented in FRAPCON-2 and are being evaluated against in-reactor data. Code structure improvements are nearing completion as progress toward the combined Pacific Northwest Laboratory (PNL)-Idaho National Engineering Laboratory (INEL) version of FRAPCON-2 continues. Experimental support efforts are yielding data that is confirming the assumptions used in the fuel relocation and mechanical response models.

INTRODUCTION

The primary objectives of the code development and experimental support efforts are to provide verified analytical models for implementation into FRAPCON-2 and FRAP-T and to coordinate code structure improvements and verification between PNL and INEL.

During FY 1979, fuel performance models were defined that more closely represent physical reality and that are in a preliminary code structure to facilitate the implementation of these models. The fact that operating nuclear fuel is cracked resulted in significant reductions in effective thermal conductivity and elastic moduli from those for solid UO_2 . The state of the cracked fuel was described by three primary mechanical parameters: (1) crack roughness, (2) gap roughness, and (3) crack pattern. However, those parameters are presently inferred from thermal results. Efforts in FY 1980 will focus on

(a) RSR Fin. Budget No.: B-2043; RSR Contact: H.H. Scott

establishing a firm physical foundation for the cracked fuel model and the implementation of these models into a combined PNL-INEL version of FRAPCON-2.

This task is structured to reflect the increased emphasis on mechanistic relocation and pellet-cladding interaction (PCI) models that have resulted from the analysis of the in-pile data. There are four subtasks: code coordination and development, mechanical testing of short rods, PCI model development, and model verification.

TECHNICAL PROGRESS

The following summarizes the progress that has been made in each subtask during this quarter.

SUBTASK C-1 - CODE COORDINATION AND DEVELOPMENT

Refinements in the relocation model have been implemented into the joint PNL-INEL version of FRAPCON-2. Code structure changes have been completed to improve convergence characteristics and to reduce code requirements. Test cases have been chosen for the developmental assessment of FRAPCON-2, and preparation of input has begun. The Code Documentation and User's Manual is partially written.

SUBTASK C-2 - MECHANICAL TESTING OF SHORT RODS

Equipment fabrication is nearing completion for experiments designed to obtain out-of-pile data on the diametral and axial fuel-clad mechanical interaction (FCMI) behavior of short-length UO_2 -Zircaloy rods. Emphasis is placed on simulation of in-pile loading conditions. Simultaneous experiments are being conducted at PNL on unirradiated rods and at Harwell, UK, on unirradiated and irradiated rods.

SUBTASK C-3 - PCI MODEL DEVELOPMENT

An advanced model is under development to economically model the effects of asymmetric pellet cracking that result in fuel column anisotropy and PCI. Efforts expended on fuel fragment and crack shape descriptions, compatibility

equations, and ratcheting mechanisms are beginning to yield a consistent description of the behavior of cracked fuel systems. Understanding gained from these efforts has been useful for designing model verification experiments.

SUBTASK C-4 - MODEL VERIFICATION

Experimental efforts are yielding data that verify the three primary mechanical parameters of the cracked fuel relocation and mechanical response model. An integrated set of experiments is being conducted at PNL to provide data quantifying the crack pattern, crack roughness, and gap roughness parameters. The value for total crack length that was assumed for analyzing the in-reactor data was 3.5 fuel diameters, whereas the total crack length measured from PIE photos was 3 to 6-1/2 diameters. The calculated crack and gap roughnesses for in-reactor data ranged between 0.2 to 2.0 mils and 0.1 to 1.0 mil, respectively. Laboratory experiments yielded crack roughnesses that were 1.0 to 2.0 mils, and gap roughnesses averaging 0.88 mils.

FUTURE PLANS

During the next quarter, the following will be done:

- Developmental assessment test cases for FRAPCON-2 will be completed.
- The actual writing of the Code Documentation and User's Manual should be nearly complete.
- The advanced FCMI algorithm will be coded.
- Data from the short-rod mechanical tests will commence.
- Model verification experiments will be completed.

CORE THERMAL MODEL DEVELOPMENT(a)

D. S. Trent, Program Manager
M. J. Thurgood, Project Manager

K. L. Basehore
K. R. Crowell
J. M. Cuta
T. L. George
T. E. Guidotti
J. M. Kelly
A. S. Koontz
G. A. Sly

SUMMARY

A pretest simulation of the JAERI Slab Core Test Facility (SCTF) gravity reflood experiments has been performed with COBRA/TRAC. The effects of reflood oscillations, steam binding, entrainment, thermal nonequilibrium, a flow blockage, and the boiling curve shape at the quench front were all included in the simulation. The results of the calculation are discussed in the report and areas requiring further development of assessment are pin pointed.

INTRODUCTION

The COBRA-TF computer code is part of the analysis techniques being developed by the Nuclear Regulatory Commission's (NRC) Water Reactor Safety Research Program. The purpose of this work is to provide better digital computer codes to be used in computing the behavior of full-scale reactor systems under postulated accident conditions. The resulting codes are presently being used to perform pre- and post-test analysis of the effects of such accident conditions on light water reactor components and systems. This project has two main objectives:

- to develop a hot bundle/hot channel analysis capability that will be used to evaluate the thermal-hydraulic performance of light water reactor fuel bundles during postulated accidents

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- to develop the capability to simulate a water reactor primary system for modeling complex internal vessel geometries such as those encountered in upper head injection-equipped pressurized water reactors (PWRs).

COBRA-TF is formulated to model three-dimensional, two-phase flow using a three-field representation. The fields are the vapor field, the continuous liquid field, and the droplet field. The model allows thermal nonequilibrium between the liquid and vapor phases and allows each of the three fields to move with different velocities. Thus, one can mechanistically treat a continuous liquid core or film moving at a low, or possibly negative velocity, from which liquid droplets are stripped off and carried away with the vapor phase. This is an essential feature in the treatment of the hydrodynamics encountered during the reflooding phase of a loss-of-coolant accident. Because of this model, it has been possible to predict liquid carryover in the FLECHT low reflood series of experiments.⁽¹⁾ The treatment of the droplet field is also essential for prediction of other phenomena such as counter-current flow limiting and upper plenum de-entrainment and fall back.

The code also features flexible noding, which permits the modeling of complex geometries encountered in reactor vessel internals, such as slotted control rod guide tubes, jet pumps, core bypass regions, etc. These geometries are not easily modeled in regular Cartesian or cylindrical mesh coordinates; but because of their significant impact on the thermal-hydraulic response of the system, reasonably accurate modeling is essential.

The fuel rod heat transfer model utilizes a rezoning mesh that automatically reduces the rod heat transfer mesh size in regions of high heat flux or steep temperature gradients and increases the mesh size in regions of low heat flux. This model has proven very effective for resolving the boiling curve in the region of the quench front.

COBRA/TRAC REFLOOD ANALYSIS OF THE JAERI SLAB CORE
TEST FACILITY WITH A 60 PERCENT FLOW BLOCKAGE IN TWO BUNDOLES

COBRA/TRAC pretest simulations of the JAERI SCTF were run at NRC request. The purpose of these simulations is to provide information concerning the thermal-hydraulic behavior of the test facility that will be useful in test planning and will be helpful in determining the appropriate location and range of instrumentation to be placed in the vicinity of the flow blockage. This report contains the results^(a) of a reflow simulation with a 60% flow blockage in the third and fourth bundles from the center of the core. Modeling uncertainties and deficiencies that may have a significant effect on the results of the simulation are also discussed.

DESCRIPTION OF EXPERIMENT

The purpose of the SCTF is to investigate the radial flow distribution in a PWR vessel during the reflow phase of a loss-of-coolant accident (LOCA). The SCTF consists of an eight-bundle slab vessel one bundle wide. The facility simulates a radial slice of a PWR that begins at the core center and passes radially outward through the downcomer and vessel wall. Bundles intended to simulate those at the center of a PWR core are referred to as the center (or inner) bundles. Those located near the downcomer are referred to as the outer bundles. The core has a chopped cosine axial power distribution as well as a radial power distribution. Beginning with the center bundle, the radial power factors are 1.12, 1.12, 1.29, 1.29, 1.05, 1.05, 0.85, and 0.85. A blockage was introduced into the third and fourth bundles from the center bundle. The blockages are co-planar near the center of the axial length, covering about 60% of the bundle flow area. In addition to the eight-bundle core, the vessel contains a downcomer, lower plenum, core bypass, and upper plenum (and associated internals). The vessel connects via a hot leg and intact cold leg to a steam-water separator, which in turn connects to a containment tank at controlled

(a) The results of the simulation should be viewed with caution, as the code has not been assessed with data from similar, large-scale, core gravity feed reflow experiments.

pressure. A broken cold leg, which connects the vessel downcomer directly to the containment tank, is also included. Emergency core cooling (ECC) water may be injected into the intact cold leg and/or into the upper plenum.

COBRA/TRAC MODEL

A COBRA/TRAC system component schematic of the SCTF is shown in Figure 1. The vessel and separator are modeled using the COBRA-TF vessel module

FIGURE 1. Slab Core Test Facility System Component Schematic for the COBRA/TRAC Model

(component 8); while the hot- and cold-leg pipes are modeled using TRAC 1-D valve, pipe and tee components. The pipes leading away from the separator and the broken cold leg are connected to break components 3 and 5, which model the containment back pressure of 3.0 bars. The intact cold leg was modeled as a tee, the branch of which was connected to a fill (components 6 and 7) allowing the specified injection of ECC water at 330K, through a flow area of 0.0552 m^2 . ECC was injected at the rates given below.

<u>Time, sec</u>	<u>Velocity, m/sec</u>
0	0.5
2	1.65
25	1.03
30	1.21
60	1.03
120	0.83
165	0.75
180	0.57

A loss coefficient of 25 was specified for the pump flow resistance simulators in the intact cold leg and broken hot leg.

The COBRA-TF nodalization of the vessel is shown in Figure 2. Thirty-one channels were used in the simulation providing a total of 181 mesh cells in the vessel. A channel was used to model each bundle in the core. Upper plenum injection and injection of extra core steam were ignored in this calculation. All walls and internal structure, including the rod bundles, were thermally modeled, with the exception of horizontal surfaces in the vessel.

The initial conditions for the simulation were representative of the conditions postulated to exist at the beginning of reflood in a LOCA. The rod temperatures were initialized using a temperature profile taken from a TRAC model of the SCTF. The temperature profile for each bundle was corrected for the radial power profile specified for the SCTF. Axial and radial power profiles specified for the facility were used in the simulation and the initial power to each bundle was based on the specified total power of 8.4 MW. The

FIGURE 2. Nodalization Diagram for the Two-Dimensional Vessel Component

standard ANS 20% power decay curve at the beginning of reflood was specified for the transient. The lower plenum was filled with saturated water up to the level of the bottom of the core barrel; the remainder of the vessel was filled with saturated steam. Initial flows in the vessel and piping were zero. The initial vessel pressure was set equal to the specified pressure of 6 bars.

The transient simulation was initiated by opening the valves on the broken hot and cold legs at a linear rate from 0.0 to 2.0 seconds while simultaneously initiating ECC injection in the intact cold leg at the rate given previously.

RESULTS OF CALCULATION

Steam flow through the broken cold and hot legs began as the valves were opened. Initially all of the ECC water injected was carried out of the broken cold leg. During this period, some of the water in the lower plenum was carried up the downcomer and out the broken cold leg. Steam velocities in the downcomer decreased as the vessel depressurized. A sufficient amount of sub-cooled water began accumulating in the top of the downcomer at about eight seconds to condense all of the steam flowing up the downcomer plus that generated by the hot downcomer walls. The downcomer then filled from the top down.

Eventually the water near the bottom of the downcomer became saturated, which stopped condensation, and hence vapor flow, in the downcomer. The weight of liquid in the downcomer forced the steam to flow up through the core, and the liquid in the downcomer dumped into the lower plenum, filling it with water and introducing liquid at the bottom of the core. During this period, condensation of steam in the downcomer and cold leg caused the vessel pressure to drop below the containment pressure of 43.5 psia, causing backflow of steam from the containment tank. As liquid entered the core, vapor generated by heat transfer in the core initiated recovery of the vessel pressure and caused the downcomer to refill with water. Vapor generation in the core forced water from the core in to the lower plenum and downcomer. The hydrostatic head in the downcomer then forced water back into the core as the vapor generation rate decreased. This U-tube oscillation between the core and downcomer persisted throughout the transient, increasing in amplitude as the liquid level approached the higher temperatures at the center of the core. These

oscillations are reflected in a plot of the liquid fraction in the outer bundle (channel 13) as a function of time (Figure 3). This figure also shows that the rate at which the liquid level rises in the core diminishes as it approaches the core midplane. This is a direct result of the increased power level in the center of the core and the diminishing driving head caused by smaller liquid level differences between the core and downcomer. The amplitude of the liquid level oscillations also increases, which may also be attributed to the above factors. The oscillation amplitude was smaller in the blocked bundles than in unblocked bundles as a result of the increased flow resistance caused by the blockage. Large amounts of entrainment were computed during positive liquid flow into the core. Some of this liquid was de-entrained in the upper plenum, which formed a liquid pool above the upper core plate. The liquid fraction in the upper plenum is shown in Figure 4. The liquid level in the upper plenum increased slowly throughout the transient. The presence of liquid in the upper plenum helped drive the core oscillations as it increased the flow resistance

FIGURE 3. Bundle Liquid Fraction in the Outer Bundle, Channel 13

FIGURE 4. Liquid Fraction in the Upper Plenum

for steam leaving the core. As the pressure in the core increased, part of the liquid in the upper plenum was forced out the hot leg by the steam flow. Recent work with COBRA/TRAC has indicated that flow of steam through a liquid pool above the upper core plate may not be adequately treated in the code.

A large fraction of the steam generated in the core was condensed in the intact cold leg by the ECC water. During the portion of an oscillation when a low vapor generation rate existed in the core, all of the core steam was condensed in the cold leg and the pressure dropped below the containment pressure and caused the vapor to backflow into the separator. As liquid re-entered the core, dropping the liquid level in the downcomer, vapor entered the downcomer through the broken cold leg as a result of condensation.

The amount of liquid injected in the cold leg was observed to have an important influence on the behavior of the system and may have been an additional driving mechanism for the computed flow oscillations. The vapor generated in the core was greatest when the liquid level was at its highest point during an oscillation. This resulted in an increase in pressure that

forced the liquid level back down. At the same time, the liquid level had dropped below the elevation of the cold leg nozzles in the downcomer, and vapor condensation resulted in a lower pressure there. Thus, the low pressure above the liquid level in the downcomer augmented the high pressure in the core in accelerating the liquid flow out of the core into the downcomer. As the liquid in the core reached its lowest level during an oscillation, the vapor generation rate fell off and the core pressure decreased. At the same time, the liquid level in the downcomer reached the top of the downcomer and liquid was forced out the broken cold leg. This increased the pressure in the top of the downcomer, which, combined with the low pressure in the core, forced liquid back into the core from the downcomer.

The cladding temperature histories at several elevations for the outer bundle and a blocked bundle are shown in Figures 5 and 6. The peak cladding temperature rise was terminated shortly after the bottom of the core was reflooded, as a result of precooling provided by the large amount of entrainment resulting from the flow oscillations. The entrainment model used in the code has not been tested against gravity reflood data and may give too much entrainment for oscillatory flow conditions. Top quenching was computed in all bundles as a result of rewet caused by the entrained liquid and as a result of liquid that occasionally entered the core from the upper plenum. Both quench fronts propagated at a faster rate in the outer bundles than in the inner ones. The quench front in the blocked bundle moved at the same velocity as in surrounding bundles until the liquid level approached the blockage elevation. After that time, the increased flow resistance at the blockage apparently caused a slower propagation of the quench front in the blocked bundle. The cladding temperature within the blockage was lower than that on either side of the blockage as a result of the increased thermal resistance between the cladding and heater rod caused by the larger gap width in the ballooned portion of the rod. The 4.5-, 6.8-, and 9.1-ft elevations in the blocked bundle were not quenched during the period covered by this calculation. The quench front propagates at a diminishing rate as a result of the higher power levels at the center of the core.

FIGURE 5. Cladding Surface Temperature in the Outer Bundle, Channel 13

The time-averaged value for the heat transfer coefficient in the blocked bundle at the plane of the blockage was 18.6 as compared to 19.3 for surrounding bundles. The vapor flow rate at the blockage plane decreases in time as a result of the diminishing difference in liquid levels between the core and downcomer. The vapor flow rate in the blocked bundle is about 50% of the flow rate that is in surrounding bundles. The entrainment flow rates were about 60% of those in surrounding bundles.

FIGURE 6. Cladding Surface Temperature in the Blocked Bundle, Channel 18

COMPARISON WITH PREVIOUS CALCULATIONS

A COBRA/TRAC simulation of the SCTF with a 60% flow blockage in the third and fourth bundles from the center was reported previously.⁽²⁾ The main differences between the current calculation and the previous one are that, respectively, the intact cold leg connection to the steam separator was not modeled, the core power was 10 MW rather than 8.5 MW, the initial vessel pressure was 43.5 psia rather than 87.0 psia, and the initial lower plenum liquid level was the bottom of the core rather than the bottom of the core barrel.

The most significant of these differences is the cold leg connection to the steam separator. All of the steam generated in the core was forced to flow through the broken hot leg to the containment tank in the first calculation. This resulted in a significant amount of steam binding and the reflood rate was limited by the back pressure caused by the flow resistance in the hot leg. The downcomer filled with water early in the transient and remained full throughout the calculation. The bulk of ECC water injected back flowed out the broken cold leg. "Rolling wave"-type oscillations across the core dominated the flow behavior with occasional flow reversals occurring in the downcomer.

The calculation reported here modeled the cold leg connection to the steam separator. Steam entering the separator from the vessel could flow into the intact cold leg where it was condensed by the ECC water injected there. During periods of lower vapor generation rates in the core, all of the steam produced in the core was condensed by this means. Even at peak vapor generation rates, the bulk of the vapor was condensed in the intact cold leg. This gave a minimum back pressure on the vessel. This calculation was characterized by large-amplitude, forced U-tube oscillations between the downcomer and core. The downcomer did not remain full of water but had a liquid level when water was entering the core from the downcomer. These oscillations provided large liquid flow rates to the core, which resulted in significant carryover fractions. The liquid thus entrained caused peak cladding temperatures to turn around shortly after bottom of core recovery.

The ECC injection rate specified for this calculation was of about the right magnitude for accumulator injection but was an order of magnitude too high for LPIS injection. It can be expected that under actual LOCA conditions the reflood behavior may go between the two extremes covered by these calculations as the ECC system switches from accumulator injection (little or no steam binding) to LPIS injection (high steam binding). In addition, steam generated in other primary system components such as steam generators can be expected to contribute to steam binding. Steam coming from these components is not modeled in the SCTF.

CONCLUSIONS

The simulations of reflood in the JAERI SCTF have provided the following insights into behavior that might be expected during reflood tests conducted in the facility and parameters that may significantly influence that behavior:

- Both U-tube oscillations between the core and downcomer and rolling wave oscillations across the bottom of the core have been computed. The nature of these oscillations appears to be strongly coupled to the steam path flow resistance.
- The amount of ECC water injected in the cold leg has a direct effect on the steam path flow resistance as it determines how much steam will be condensed in the intact cold legs, decreasing the amount of steam required to flow out the broken hot leg.
- Large liquid carryover rates were computed in this simulation. This resulted in early turnaround of peak clad temperatures due to dispersed flow film boiling. A liquid pool was also formed in the upper plenum as a result of structural de-entrainment of the entrained drops.
- The following effects of blockage have been observed:
 - The amplitude of oscillations in the blocked bundles are damped relative to those in surrounding bundles.
 - The heat transfer coefficients in the blocked regions are lower than in surrounding regions. The decrease was less significant in this calculation than in the first. This may be attributed to the larger entrainment flow rates in this calculation.
 - Vapor flow rates in the blockage are about 50% of those in surrounding bundles.
 - The quench-front velocity in the blocked bundle decreases more rapidly than in unblocked bundles as the blockage plane is approached.

- The cladding temperature in the blocked region decreases faster than that above or below the blockage as a result of the increased gap width between the heater rod and cladding.
- Vapor velocities are too high to allow any significant accumulation of liquid drops around the blockage.

FUTURE WORK

COBRA/TRAC has not been assessed against any similar gravity reflood experiments. The code should be assessed against such data before confidence can be placed in the results of these calculations. In particular, the following models should be assessed:

- the entrainment rate model for oscillating reflood
- the treatment of the liquid pool above the upper core plate.

For this purpose, a simulation of tests conducted in the Cylindrical Core Test Facility is planned. A model of the test facility is being set up to assess the code by comparing its calculational results with data taken in that facility.

REFERENCES

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2. M. J. Thurgood, et al., COBRA/TRAC Reflood Analysis of JAERI Slab Core Test Facility, Battelle, Pacific Northwest Laboratory, FATE-80-103, 1980.

LOCA SIMULATION IN NRU(a)

C. L. Mohr, Project Manager
J. P. Pilzer, Deputy Project Manager
P. N. McDuffie, Project Administration

G. M. Hesson	G. E. Russcher
L. L. King	C. Nealley
R. K. Marshall	R. L. Goodman
G. W. McNair	W. D. Meitzler
L. J. Parchen	R. E. Schreiber

SUMMARY

Fabrication of the first of six test train assemblies has been initiated. This assembly will be used to test thermal hydraulic behavior during the heat up, reflood, and quench phases of a large-break loss-of-coolant accident (LOCA) and will be available for testing in October 1980. Modifications to the U-2 loop to accommodate transient testing have also been initiated. Major procurements are complete, and pre-fabrication of the modules to be placed in the reactor during summer shut downs is near completion. The data acquisition system has been programmed with the appropriate software for test operations, system debug, and data reduction.

The Safety Analysis Report (SAR) was completed for presentation to the Canadian reactor safety board. The primary emphasis of this report was associated with assuring pressure tube temperatures of $<800^{\circ}\text{F}$ during the transient. As part of this work, detailed calculations were also performed using the Combustion Engineering code, THERM, the Idaho National Engineering Laboratory code, FRAP-T5, and the Pacific Northwest Laboratory (PNL) code, GAPCON-FLECHT.

INTRODUCTION

The objective of the LOCA Simulation in NRU Program is to provide information on the heat up, reflood, and quench phases of a LOCA. The tests are

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designed to give information on the quench-front velocities within a fuel bundle, the liquid entrainment [10 CFR 50, App. K (Sec. ID, 2)], and the heat transfer coefficients [10 CFR 50, App. K (Sec. ID, 5)] for full-length pressurized water reactor (PWR) fuel as a function of reflood rate and delay time before reflood starts. A total of 6 test trains will be prepared for the program, with the first test train devoted to thermal hydraulic behavior. A total of 18 tests are planned for the first test train. The remaining 5 test trains will be used for only one test each. These test trains will have prepressurized fuel rods and, as a result, the rods will deform and rupture during the test. These tests, called "materials tests," will evaluate the effects of ballooning and rupture on the quench-front velocities and associated heat-transfer coefficients.

The NRU reactor, located at Chalk River, Canada, has a test loop that will accommodate a 12-ft long, 32-pin bundle on a 6 x 6 array with the corner pins removed. The bundle design uses commercial enrichments, cladding dimensions, and grid spacers.

TECHNICAL PROGRESS

The fabrication of the major components for the test train has been completed, including fuel pellets, grid spacer components, shrouds, and much of the assembly tooling. Significant progress on the fabrication of the U-2 loop modification has also been made. Major portions of the loop piping components are being prefabricated into modules, which will be inserted into the reactor piping system during scheduled reactor shut downs. Delivery of purchased components is the limiting factor in both the test train and loop fabrication; major suppliers of all types of items have had great difficulty in meeting scheduled delivery dates and providing components that meet technical specifications.

The test train design uses a combination of unique instrumentation techniques for full-length PWR fuel pins, and a significant development effort is required to perfect these techniques. The materials test design concept calls for the thermocouples to be spot welded to the cladding inside surface at

locations up to 8 ft (244 cm) from one end of the fuel rod. The fuel pellets will be slotted to house the thermocouples. Some fuel rods will have up to four thermocouples at various elevations. These instrumentation techniques are intended to improve the accuracy of measuring the temperature of the fuel bundle during the reflood and quench phases of the experiment. By placing thermocouples inside the fuel cladding, uncertainties about flow perturbation are eliminated, and a more accurate estimate of cladding temperatures can be made.

The first assembly to be used for the thermal hydraulics test series will use grooved fuel pellets in unpressurized rods with the thermocouples unwelded to the cladding ID surface. Subsequent tests will use welded thermocouple fabrication techniques for all fuel rods.

Fabrication for the major fuel rod component parts is complete. The stainless steel shrouds have been completed, and flow tests of the "C" seals used to seal the halves of the shroud have been performed. The seals, which use a self-energizing concept to maintain seal surface pressure, have performed very well with essentially no leakage. Fabrication procedures for the grid spacer for the guard rods and cruciform rods are complete. These spacers are currently being fabricated for all of the test trains.

The progress on loop modifications for the U-2 loop in the NRU reactor has also been significant. In order to meet the project target test time of mid-October 1980, the Chalk River Nuclear Laboratories (CRNL) staff has developed a modular fabrication procedure using pre-fabricated major piping component modules and connecting them with mechanical connectors instead of welding. This procedure reduces the radiation exposure to the workers and speeds the assembly during reactor shut downs. The CRNL staff has currently completed the loop instrumentation and control phases of the loop modification and are now involved in installation and pre-fabrication activities.

The safety analysis and safety review presentations have been formally made to the Canadian National Safety Advisory Committee, who are charged with approving the SAR made by PNL for the test train and for the loop operation by CRNL. Final approval is expected in August.

FIGURE 1. Cladding Temperature, 5-in./s Reflood Rate, No Delay Time Before Reflood

FIGURE 2. Cladding Temperature, 2-in./s Reflood Rate, 20-s Delay Time Before Reflood

FIGURE 3. Cladding Temperature, 2-in./s Reflood Rate, 44-s Delay Time Before Reflood

FIGURE 4. Fuel Centerline Temperature, 2-in./s Reflood Rate, No Delay Time Before Reflood

FIGURE 5. Cladding Dilatation, 2-in./s Reflood Rate, 20-s Delay Time Before Reflood

FIGURE 6. Fuel Stored Energy, 2-in./s Reflood Rate 44-s Delay Time. Before Reflood

Predictions of fuel pin behavior for various test conditions were made using the THERM, FRAP-T5, and GAPCON-FLECHT codes. Comparisons were made for the peak power point along the fuel rod length, and Figures 1 through 6 show the results of the comparison calculations between these codes. The independent variables shown in these figures are reflood rate and delay time to initiate reflood. The calculations are based on the test operating procedure for the transient phase of operation, which starts at a temperature of 800°F (700K) and then begins a heatup in steam until the reflood water is introduced. Both the GAPCON-FLECHT and FRAP-T5 thermal hydraulic quench models are based on the FLECHT data. The THERM model is based on using FLECHT as well as additional proprietary information and is used in licensing-oriented analyses.

The differences in the calculated results shown by the various models are basically the differences in the entrainment parameters in the thermal hydraulic models. The results show significant differences for computing quench times for the bundle; however, they are very similar for predicting peak temperatures. The FRAP-T5 Code and the GAPCON-FLECHT code had deformation models, whereas the THERM code did not provide any mechanical data. In most cases, the GAPCON-FLECHT code seems to be mid-span between the FRAP-T5 and the THERM calculations.

FUTURE PLANS

The first test train will be assembled and the loop modifications to the U-2 loop will be completed. A test operations document will be assembled and a detailed test matrix will be selected.

REFERENCE

Code of Federal Regulations, Title 10 (Energy), Part 50 Domestic Licensing of Production and Utilization Facilities. Revised January 1, 1980. U.S. Government Printing Office, Washington, D.C.

RESIDENT ENGINEER AT CADARACHE, FRANCE(a)

D. S. Trent, Project Manager
C. L. Wheeler, Resident Engineer

SUMMARY

The remaining minor start-up tests, required by licensing personnel, were finished, and an additional modification to the loop was made.

INTRODUCTION

The objective of this program is to facilitate the exchange of information between the U.S. and France. To this end, a resident engineer is being supported at the PHEBUS program in France. The PHEBUS experimental program for in-pile studies is operated by the Service D'Essais de Surete (Safety Tests Service) of the French Atomic Energy Commission. The facility, located at Cadarache, consists of a closed test loop driven by a pool-type reactor that develops about 60 mW for a 20-minute test period. The test loop, which passes through the center of the core, contains an 80-cm test section and is capable of operating at pressurized water reactor (PWR) conditions. Current plans call for blowdown and reflood type testing on rod bundles containing either single-rod or 25-rod assemblies having geometries typical of those found in large PWRs.

CURRENT STATUS

The three remaining start-up tests were finished in February. These three tests consisted of 1) studying the effect of xenon on operation and shutdown with the 25-rod test configuration; 2) verifying the ability of the DRG (clad rupture detector) to function properly; and 3) verifying radio protection. The facility modification was completed by the end of March. This modification was necessary because the original design did not maintain the proper regulation

(a) RSR Fin. Budget No.: B2278; RSR Contact: W. V. Johnson

of the force tube temperature. In the original design, cold water ($\sim 40^{\circ}\text{C}$) at a low flowrate was used to cool the force tube. After leaving the force tube, the force tube coolant is mixed with the primary loop flow just above the core where, in principle, it regulates the temperature of the hot-leg coolant to 320°C . However, in order to achieve the desired hot-leg temperature in the single-rod tests, a very low coolant flow rate had to be used, which proved difficult to regulate. Therefore, the design was changed; hot water ($\sim 300^{\circ}\text{C}$), taken from the cold leg of the main loop at a relatively high flow rate, is now used to cool the force tube.

FUTURE WORK

The experimental program was not initiated this quarter because of the loop modification. The program is expected to begin next quarter with the following four experiments:

1. 25-rod hydraulic (non-nuclear) cold-leg break
2. 25-rod cold-leg break with heating rate of 27 kW/m
3. 25-rod hydraulic hot-leg break
4. 25-rod hot-leg break with heating rate of 27 kW/m.

STEAM GENERATOR TUBE INTEGRITY(a)

R. A. Clark, Project Manager
V. F. FitzPatrick, Deputy Project Manager

J. M. Alzheimer

R. L. Burr

P. G. Doctor

G. H. Lyon

C. J. Morris

K. R. Wheeler

INTRODUCTION

The Steam Generator Tube Integrity Program (SGTIP) is a multiphase, multi-task laboratory program. The program's principle objective is to provide the Nuclear Regulatory Commission (NRC) with validated information on the remaining integrity of pressurized water reactor (PWR) steam generator tubes where service-induced degradation has been indicated. An additional objective is to evaluate nondestructive instrumentation/techniques for examining defects in piping or tubing that serve as the reactor primary system pressure boundary.

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

Early work showed that conventional, single-frequency, eddy current evaluation of steam generator tubes, as used for in-service inspections, could be

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improved. Thus, program efforts were expanded to include new eddy current measurement techniques, the effects of different calibration standards, and a more complex statistical analysis of NDT data.

The first two phases of the program involved the study of mechanically (Phase I) and chemically (Phase II) defected tubing. Phase III of the program includes correlating the mathematical models developed in Phases I and II with actual service-flawed tubing. Lack of suitable specimens led to the redirection of Phase III into the Surry Generator Program, a subprogram of SGTIP. The Surry Generator Program is responsible for relocating a nuclear steam generator removed from service at Surry, Virginia, to Hanford, Washington. Once at Hanford, the steam generator will be housed in a specially designed facility to allow nondestructive as well as destructive examination of the unit. Research will be conducted in the areas of NDT technique verification and instrument development, defect matrix identification, profiling of defect types, determination of the extent and location of defects, identification of deposits and sources of corrosion, and health physics. The generator will also become a source of specimens with service-induced flaws for various NRC programs. Potential research phases include simulated operation of a portion of the generator to assess long-term effects on generator serviceability of chemical cleaning or decontamination procedures that may be proposed to NRC for licensing. Study of the recovery and reuse of materials in decommissioned reactor components is also a potential research task.

TECHNICAL PROGRESS

The following paragraphs detail progress on program tasks active this past quarter. Almost all efforts concerned preparation and licensing activities for transporting a retired-from-service nuclear steam generator from Surry, Virginia, to Hanford, Washington. Research-related activities were limited to initial work on a series of leak-rate experiments.

LEAK-RATE DETERMINATIONS FOR THROUGH-WALL DEFECTED INCONEL 600 STEAM GENERATOR TUBES

At the request of NRC licensing, a task was added to SGTIP this past quarter to determine resultant leak rates from several different through-wall defect configurations. Defected specimens are to be pressurized at PWR operating temperatures and leak rates determined as a function of pressure differential across the specimen. Leakage will be into the atmosphere with maximum primary side pressurization of 2500 psig at 600°F. The maximum leak-rate capacity of our system will be 300 gallons per minute.

The following test matrix was agreed upon for this task:

1. Uniform Thinning and Stress Corrosion Cracking (SCC)

Study 1/4-in., 1/2-in., 1-in. and 1-1/2-in. SCC with uniform thinning twice the crack length and at the following depths: 0%, 35 to 45%, 55 to 60%, 75 to 85%. (Number of tests = 16)

2. Long Through-wall Cracks

Study 2-in. and 3-in. cracks with no uniform thinning. (Number of tests = 2)

3. SCC Through-wall Circumferential

Study circumferential cracks of 45°, 90°, 120°, 180°, 270°. (Number of tests = 5)

4. Leak in Simulated Tube Sheet Crevice

Test a single SCC and multiple SCC with and without packing. (Number of tests = 4)

5. Collapse in Simulated Tube Sheet Crevice

Collapse test SCC tube and uncracked tube, each in a simulated tube sheet, to 3500 psi. (Number of tests = 2)

6. Leak Under Dent

Study tubes with 1/8-in. and 1/4-in. cracking on each side of support plate. (Number of tests = 2)

All the above tests are to be conducted on 0.875-in. O.D. x 0.050-in. wall tubing. In addition:

7. Leakage in 0.625-in. x 0.035-in. Tubing

Spot check leakage in 0.625-in. O.D. x 0.035-in. wall tubing with no thinning and varying crack lengths. (Number of tests = 4)

The SGTIP will also fabricate and supply two simulated tube sheet specimens to an NRC-sponsored NDT program at Oak Ridge. These specimens will be comprised of a defected Inconel tube placed into a carbon steel tube sheet simulation. One specimen will have a tight corrosion product packing in the tube-tube sheet annulus, the other will not.

Figure 1 shows the intended configuration of simulated tube-sheet specimens. This quarter we have ordered necessary materials for the test matrix. Specimen fabrication will be initiated the coming quarter.

SURRY GENERATOR PROJECT

This project, a subprogram of SGTIP, entails retrieving a nuclear steam generator removed from service at the Surry II plant at Surry, Virginia, and transporting the generator to Hanford, Washington. Once at Hanford, the generator will be located in a specially designed temporary facility to allow non-destructive and destructive examination.

Research on a removed-from-service steam generator originated out of Phase III of SGTIP. Phase III involved obtaining and testing service-defected tubing. These tests were to confirm models of remaining tube strength, developed in Phases I and II on mechanically- and chemically-simulated tube defects. An inability to obtain representative service-defected tubing was a prime motivation for proposing research on a removed-from-service generator.

Tasks on the Surry Steam Generator Program this past quarter involved finalizing procedures and contracts for shipping the steam generator. Institutional requirements for shipment were completed. Subcontracted work was initiated on long-lead items necessary for shipment. Title I design of the Steam

PROBLEM THAT EXISTS IN SERVICE:

PLANNED SIMULATION:

FIGURE 1. Tube-Sheet Simulation for Leak-Rate Tests

Generator Examination Facility (SGEF) was completed and Title II design initiated. A temporary storage facility to hold the generator until completion of the SGEF was constructed.

Temporary Storage

Figure 2 is an artist's rendition of the interim storage facility (ISF) for the steam generator. The generator will sit on two concrete supports located beneath the transport cradles. Also shown in the figure is a structure enclosing the channel head end of the generator. Although we do not currently anticipate any research activity on the generator until it is housed in the SGEF, it may be desirable to conduct limited NDT. This work would require access through the manways in the waterbox under the tube sheet. The structure pictured around the channel head end of the generator represents a rigid greenhouse that would be used for any early NDT. This past quarter, the concrete support cradles were cast and a concrete pad was poured to serve as the floor for the rigid greenhouse. We poured this pad to avoid the possibility of having to construct forms and pour concrete beneath the radioactive generator. Detailed design and construction blueprints for the greenhouse were also completed this quarter.

Steam Generator Examination Facility

This quarter, Title I design (preliminary design) and cost estimating was completed for the SGEF. A Title I report has been approved, issued, and forwarded to NRC. The cost estimate for the SGEF at the Title I design level is \$1,500,000. This is divided between \$176,000 for engineering (including \$21,000 for escalation and contingency) and \$1,324,000 for construction (including \$121,000 for escalation and \$293,000 for contingency). The \$1,500,000 total compares very favorably with the cost range \$1,493,700 to \$1,906,600 established in our original feasibility study. The Title I cost estimate (\$1,500,000) is \$240,000 less than the estimate at the conclusion of conceptual design (\$1,740,000). At the same time, contingency funds, an expression of uncertainty in the estimate, dropped from \$381,300 at the conceptual design level to \$310,000 at the Title I design level. Title II design (definitive design) was initiated and approximately half completed this quarter. Title II design will be finalized in the next quarter.

Transportation Licensing of a Surry Generator

Transportation of the generator hinged on a licensing decision by the Department of Transportation (DOT). We submitted a request to allow shipment

FIGURE 2. Interim Storage Facility for Steam Generator

of the steam generator as a low specific activity (LSA) package. This required an exemption from the limit of 0.001 millicuries per square centimeter for an LSA shipment. Our exemption request was based on consideration of the 3-in. thick carbon steel steam generator shell as the package. The generator has about 0.003 millicuries per square centimeter activity, essentially uniformly distributed in the tenacious oxide coating on the inside of the steam generator tubes. Typically, an LSA package can consist of a cardboard container; therefore, an exemption from the 0.001 millicuries per square centimeter limit should be justified.

A detailed Operational Plan was submitted to the DOT in support of our exemption request. After two public comment periods, the DOT granted our exemption on March 28, 1980. Transport of the generator will thus be allowed as an LSA cargo.

Transportation Preparation of the Surry Generator

All open issues that remained at the end of last quarter have been favorably resolved this quarter. In March, Pacific Northwest Laboratories (PNL) and Virginia Electric and Power Company (VEPCO) signed a Steam Generator Purchase Agreement and Site Use Cooperative Permit. These documents provided transfer of ownership of the generator; established insurance liability during preparation, handling, and shipping operations; and provided the basis for PNL personnel and PNL subcontractors to operate on VEPCO's Surry Nuclear Power Station site. The most significant aspect of the agreement between VEPCO and PNL was the provision by the Department of Energy (DOE) of Price-Anderson Act nuclear indemnity coverage. This coverage flows through PNL, a DOE prime contractor, to VEPCO as a subcontractor to PNL.

Several other shipping preparation activities were concluded this quarter. Williams Crane and Rigging, the subcontractor in charge of Surry site handling activities, was authorized to fabricate the concrete shields to be placed on the barge. These slabs were cast to allow 35 days curing prior to placement on the barge. All structural steel to anchor the generator and concrete shielding was also acquired.

A contract was signed with Foss Tug and Barge to transport the generator. Foss activated the tug and barge out of Seattle and proceeded to Port Arthur, Texas, with a cargo of lumber. From there, the barge will proceed to Surry in early April to pick up the generator.

Pre-Shipment Generator Examination

Prior to expending funds to prepare and ship the generator, an examination was conducted to assure that the generator was in transportable condition. A subcontract was let to Combustion Engineering for conducting a secondary side examination. The examination consisted of cutting 12-in. square holes at three locations in the generator. The secondary side of the generator was visually examined along with boroscope and fiber optic viewing. Participation in the examination included expert representatives from NRC, PNL, Westinghouse, Babcock and Wilcox, Combustion Engineering, and Kraftwerk Union. Two major conclusions were sought, that the generator was not so service degraded that it could not withstand transport to Hanford without damage and that interim storage at Surry had not resulted in corrosion that could obviate use of the generator as a research vehicle. The general conclusion reached by all examiners was that the generator should be transportable without resulting transport damage. In addition, very little Hematite or red-rust was visible on carbon steel components. This indicated that there was little moisture in the generator during Surry site storage and very little corrosion occurred during storage. It was concluded that the generator remained as it was at the time of removal from service and would serve as a useful research vehicle.

In conducting the secondary side pre-shipment examination, the opportunity was taken to document the condition of the generator. Measurements of internal component locations were established to allow comparison after transport. Sludge samples and corrosion-product scrapings were obtained for analysis. Color photographic documentation of accessible regions was provided. Figures 3 through 10 show typical views of the generator secondary side. Figures 3 and 4 show a tube support plate nearly closed by "hourglassing." Almost all deformation has occurred from the hot-leg side. The Inconel 600 tubes are fairly clean with a few mils of gray-colored deposit that readily flakes off. The light spots on the tubing are tentatively identified as copper but have not

FIGURE 3. Tube Support Plate with "Hourglassed" Flow Slot

FIGURE 4. Close-up Figure 3 Showing Crack in Support Plate

FIGURE 5. Underside of Support Plate in Figure 3 (Note closure of flow slot)

FIGURE 6. Segment From Support Plate Found Lying in Generator

FIGURE 7. Detail of Loose Support Plate Segment

FIGURE 8. Below Topment Tube Support Plate With Inner Row U-Bends Visible

FIGURE 9. Sixth Support Plate Viewed From Above Showing Bowing of Hot Leg Tubes

FIGURE 10. Detail of Inner-Row U-Bends

yet been analyzed. A crack is noted in the support plate, but this could be traced only a short distance in the available visual path. A sludge deposit is noted on the outer edge of the support; however, in general, very little sludge accumulation was apparent. No tube could be found that had a clean annulus between tube and tube support plate. Figure 5 is another view of the same support plate (taken from the underside) shown in Figures 3 and 4; the extent of hourglassing is graphically depicted. In Figure 6, a loose piece of metal is seen. This was later identified as a piece of tube support plate that had fallen from an upper support plate after all four ligaments had failed (Figure 7). Figure 8 is taken below the upper-most support plate. The inner-most row of U-bends is visible through the flow slot. Also apparent is a bowing of tubes into the flow lane. Figure 9 shows that this bowing results in the hot leg tubes coming within about 1/8 in. of the cold leg tubes between the 6th and 7th support plates.

Figure 10 shows the inner row of U-bends, where considerable ovalization was visible. A consensus could not be reached by the examiners if cracking was visible, as some thought cracks were present and others believed we were looking at mandrel marks.

At the conclusion of this examination, the cuts in the steam generator were closed by replacing the removed sections and welding them back into place. In addition, a 3/4-in. plate was lapped over the cut and welded to the vessel.

ENVIRONMENTAL IMPACT ASSESSMENT

DOE is deeply involved in this project. Work will be conducted at the DOE's Hanford site, and DOE will take ownership of the generator and provide insurance during the shipment. DOE requested that PNL provide an Environmental Impact Assessment (EIA) of the program. This included transport of the generator, research activities at Hanford, and decontamination/decommissioning of Hanford facilities upon program completion. A draft EIA was submitted for DOE comments at the end of December. After addressing all comments, an expanded EIA was resubmitted. On March 27, 1980, DOE accepted the EIA for publication and ruled that an Environmental Impact Statement would not be required. This

means that based on the EIA, the proposed generator move is not "a major Federal action significantly affecting the quality of the human environment."

Accelerometer

This task was added to the program because we judged it prudent to document all accelerations the generator experienced during the barge transport from Surry to Hanford. The program subcontracted the design and fabrication of a three axis accelerometer to be mounted on the steam generator. A transmitter on the barge will continuously radio accelerometer output information to a receiver placed on the tug where both a magnetic tape of accelerations and a strip chart display will record data. Diehl and Lundgaard, Inc., Marine and Mechanical Engineers has been awarded a subcontract to provide this equipment.

MILESTONES

- EIA accepted by DOE with a negative declaration of need for an EIS
- VEPCO-PNL Cooperative Permit and Transfer of Ownership Agreement signed
- DOT ruling in favor of allowing request for shipment of the generator as an LSA item
- Foss Tug and Barge contract established
- Accelerometer design/fabrication contract placed
- Title I design of the SGEF completed
- Pre-shipment examination of the generator established that no storage damage has occurred and the unit is transportable.

PROBLEMS

- No Hanford handling contract is yet in place, due primarily to lack of agreement on a suitable price with the prime-contender subcontractor.

FUEL SUBASSEMBLY PROCUREMENT AND IRRADIATION TEST PROGRAM(a)

C. L. Mohr, Program Manager
E. L. Courtright, Project Manager
R. E. Schreiber, Deputy Project Manager
9-Rod Test Train
D. D. Lanning, Deputy Project Manager
Gap-Conductance Analysis
R. L. Goodman, Deputy Project Manager
Multi-Rod Test Trains

C. H. Allen	R. A. Lundgren
J. S. Barnhart	R. K. Marshall
L. R. Bunnell	G. W. McNair
M. E. Cunningham	D. R. Montgomery
R. E. Falkoski	C. Nealley
G. M. Hesson	F. E. Panisko
R. L. Hjelm	J. P. Pilger
D. E. Hurley	M. S. Quigley
J. King	J. D. Rising
L. L. King	G. E. Russcher
R. F. Klein	R. D. Tokarz
R. R. Lewis	

SUMMARY

The 9-rod design for abnormal operating transient (OPTRAN) test 1-3 and reactivity initiated accident (RIA) test 1-7 was formally reviewed and approved, and the ordering process for major work has been initiated. Potential delays in instrument deliveries may require final assembly at Idaho National Engineering Laboratory to meet scheduled test dates. Current efforts are directed toward the development of remote handling equipment. Preliminary procedures have been prepared and some hot cell fixture designs are nearly complete.

Substantial progress has been made in the areas of thermal-hydraulic analysis, conceptual assembly design, test instrumentation, and experimental control in support of the SUPERSARA test program. Preliminary radiation heat transfer calculations have been performed to determine bundle power requirements. These analyses include: metal-water reaction, convective steam heat

(a) RSR Fin. Budget No.: B2084; RSR Contact: R. Van Houten

transfer, heat losses from the bundle, bundle peak temperatures, and maximum pressure vessel and shroud temperatures. A concept for a test train capable of performing the full spectrum of large- and small-break loss-of-coolant accident (LOCA) tests in ESSOR, as well as the high-temperature, severe-damage tests in the Power Burst Facility (PBF) was developed. The design features a common shroud and bundle subassembly and changeable instrumentation packages. Tests sequencing and control philosophies for conducting and controlling the SUPERSARA experiments have been formulated and evaluated.

INTRODUCTION

The Fuel Subassembly Procurement and Irradiation Test Program has been divided into four tasks for FY 1980. The objectives of these tasks are as follow:

- Task A - PBF Test-Train Hardware. The purpose of this task is to provide instrumented test-train hardware for selected PBF experimental programs. Assembly hardware will include replacement quadrants for RIA-type four-rod test trains, and a fully-instrumented 9-rod test assembly for the RIA 1-7 and OPTRAN 1-3 experiments.
- Task B - Gap-Conductance Test Analysis. This task will be directed at providing testing methodologies and analysis techniques for obtaining fuel rod thermal performance from PBF experiments. Post-test analysis and data interpretation of experimental results will be provided.
- Task C - Dual-Purpose Assemblies. This task is responsible for transferring IFA 513 dual-purpose test rods currently under steady-state irradiation in Halden to PBF for subsequent transient testing.
- Task D - Multi-Rod Test Trains - ESSOR Project. The scope of this project includes provisions for supplying fully instrumented and fueled test assemblies, analytical services, and engineering support for the ESSOR Reactor Test Program at Ispra, Italy. The program is designed to yield important experimental data related to fuel rod

deformation and post-accident coolability of damaged fuel rod clusters after small-break, medium-break, or large-break LOCA. In addition, this project will also develop test-train concepts consistent with the stated experimental objectives of the proposed PBF small-break LOCA test program.

PBF TEST-TRAIN HARDWARE

4X QUADRANTS

The copper-constantan thermocouples were received and the differential temperature assemblies were fabricated and calibrated. The subassembly of the 4X replacement quadrants was completed. All component parts, including high-pressure connector parts and all instruments necessary to complete the final assembly of the 4X replacement quadrants, are on hand; however, final assembly of the quadrants has been stopped, pending future design modifications that may be required to meet revised experimental objectives.

9-ROD TEST TRAIN

The 9-rod test-train hardware, Figure 1, is designed to accommodate testing of nine previously irradiated rods and contains appropriate instrumentation to monitor fuel rod behavior during the course of the imposed transients. The test assembly includes a removable, replaceable bundle section with associated fuel-rod instrumentation. The fuel bundle for each test is enclosed by a flow shroud that permits half the cooling water to bypass the bundle. A recombined flow region allows the two-phase steam/water mixture generated within the bundle to be combined with the bypass water to quench the steam. Flows, temperatures, pressures, and motions are recorded along with neutron and gamma flux during both RIA and OPTRAN tests.

There are two interchangeable fuel rod bundles, one designed specifically for each test. The bundles are also designed so one noninstrumented rod can be removed and replaced following the calibration phase of the experiment. The test train includes four other sections: the inlet region, the fuel bundle and shroud region, the recombined flow region, and the closure, hanger,

FIGURE 1. 9-Rod Test-Train Assembly

and restrictor region. For clarity, the core instrumentation assembly, which is mounted around the outside of the fuel bundle and shroud region, has been omitted in Figure 1. Each modular section has its own package of instrumentation and hardline cabling. Each package of cables exits the closure head through a braze plug. The number of lines in each plug is limited to about a dozen to facilitate brazing.

Figure 2 shows the inlet assembly. The lower ring with the holes provides the appropriate flow restriction between the test train and the flow tube to give the 1:1 bypass-to-shroud flow split. Pipe plugs are removed from the holes to fine tune the flow. A flow orifice plate, held by a large nut at the base, controls the flow into the test train. The cylindrical section contains a crossed-plate flow straightener. The collar around the straightener housing contains the lower (upstream) set of differential thermocouples. Above that is the turbine flowmeter with the hexagonal-shaped housing containing pickup coils and above the flowmeters is the turbine-to-shroud adapter. A 10-K pulse pressure transducer is outside the adapter housing.

FIGURE 2. Inlet Region

Figure 3 shows the principal features of the fuel bundle and shroud region. The rods are held in place with four grids patterned after a Babcock and Wilcox design: keys are used to hold the springs out of the way while the rods are inserted, and once the rods are aligned, the keys are removed. This method of rod loading protects the thermocouples and hardline hold-down straps from scraping against the grids during insertion. The four Zircaloy shroud plates are held together at each corner in a dovetail fashion with a sealing gasket (steel tube) between each set of plates to be clamped by heavy clips, which slide down from the ends. This provides a strong, leak-tight shroud that can resist substantial pressure pulses. The fuel-rod end-plug extensions reach up into the shroud outlet housing where they come into contact

FIGURE 3. 9-Rod Fuel-Bundle Assembly

with the linear variable differential transformer (LVDT) extension rods. Hardline leads from thermocouples, strain gages, and rod pressure transducers are conducted out between the conical collars of the split outlet housing. This arrangement keeps the hardlines conveniently out of the way during rod replacement operations. The hardlines from the bundle are all external to the test train above the shroud outlet housing. The lines are bundled and passed through various restrictor seals and braze plugs to the top of the test train. Some of the core instruments are not mounted on the shroud. These can be moved aside when the fuel bundle assemblies are changed. In addition, all hardlines that are not capable of being spliced are grouped and separated from lines that can be spliced to minimize the risk inherent in repair procedures. The shrouded fuel-bundle assembly with all of its hardlines will be replaced between the RIA 1-7 and OPTRAN 1-3 tests.

The outlet/recombined flow region, Figure 4, contains many pieces of equipment in addition to performing flow mixing. The lower housing contains

FIGURE 4. Outlet/Recombined-Flow Region

the LVDT assemblies, the other leg of the differential thermocouples, a pressure tap, and a mounting that holds a pulse pressure transducer. The housing above that point contains a flow straightener, leading to a turbine flowmeter. The axial load on the test train is conducted around the flowmeter by load bars. These adjustable bars and the bellows that seal the flowmeter housing to the test train are used to make final adjustments for alignment of the bundle fuel centerline with the PBF core centerline.

The recombiner housing (holes) leads to a flow straightener, more thermocouples, and another turbine flowmeter of similar design (load bars and bellows) but which has twice the capacity of the others.

The hanger assembly (Figure 5) is composed of a flow tube, a four-piece flow restrictor to force the flow through the recombiner, seal penetrations for

FIGURE 5. Hanger Assembly

each bundle of hardlines, a fuel particle catch screen (0.035-in. opening) and catch cup, and four hanger tubes containing hardline bundles. Braze plugs are mounted to the hanger tubes. The catch cup hangs down inside at the elevation of the bulbous section. Other bundles of hardlines also have braze plugs at the elevation of the closure head.

The closure head, Figure 6, is similar to earlier versions used on the PBF. There are actually nine penetrations for braze plugs: eight of two different sizes in a ring around the lifting tube and one inside the central tube. Connectors to soft lines are to be attached (by EG&G) a few feet above the head. The other hardware shown in the figure is a multiple valve and hose disconnect system for nitrogen purging and air bleeding the loop.

FIGURE 6. Closure Head

Figures 7 and 8 are schematic layouts of the instrumentation for the two tests. The reentrant flow is split between the bundle and the shroud bypass. The bidirectional inlet flowmeter senses the normal bundle flow as well as the momentary flow reversal that may occur during the transient. The high-pressure pulse transducer will detect the pulse wave as it propagates downward in the test train. The two sets of thermocouples give the absolute and differential inlet coolant temperatures. Around the outside of the shroud are the steady-state and transient neutron flux detectors. The self-powered neutron detectors (SPNDs) are arranged to follow the axial variation of flux with time. The flux wires give the total dose; the fission chamber gives an absolute measure of flux. The self-powered gamma detectors (SPGDs) are used to correct the signals from the SPNDs, which tend to measure both neutrons and gammas. The 30-in. SPNDs provide an axial average flux. Some of the instruments are mounted directly on the shroud while others are held away from it on support tubes. The core instrumentation that is not mounted on the shroud is moved out of the way for bundle replacement and reinstalled for the second test.

At the center of the shroud, a tap leads to a pulse pressure transducer. At the outlet from the bundle, both absolute and differential thermocouples monitor the temperature of the coolant exiting the bundle. The exit flow is also measured. A high-pressure pulse transducer in the same area monitors the propagating wave traveling up the test train. After the bypass flow reenters the test train and quenches any steam, the mixed mean absolute, as well as differential temperatures, are measured along with the total flow. Backup instrumentation includes three externally mounted pressure transducers. Lines from the instruments tap into the top and bottom of the shroud, as well as the center. The pressure drop across the bundle and the absolute pressure at the center outlet can be recorded.

The fuel-rod instrumentation is listed in Table 1 for RIA 1-7 and Table 2 for OPTRAN 1-3. All of these instruments are intended to provide data on the steady-state and transient thermal and mechanical loads on the cladding. These tests will be conducted to study the nature of cladding deformation and rupture, as well as the consequences of rupture.

FIGURE 7. Axial Schematic of Test-Train Assembly Showing Assembly Instrumentation Placement for OPTRAN 1-3

FIGURE 8. Schematic Representation of the Test RIA 1-7 Test-Train Assembly Showing Relative Locations of the Test-Assembly Instrumentation

TABLE 1. Measurement Requirements for Test RIA 1-7 Fuel-Rod Instrumentation

<u>Rod^(a)</u>	<u>Measurement</u>	<u>Instrument</u>	<u>Instruments Per Rod</u>	<u>Instrument Locations^(b)</u>
807-1	Cladding Surface Temperature	Thermocouple	2	22.83 in. at 315° 26.77 in. at 135°
	Internal Gas Pressure	Pressure Transducer	1	Upper Plenum
	Cladding Elongation	LVDT	1	Top of Fuel Rod
807-3	Cladding Elongation	LVDT	1	Top of Fuel Rod
807-5	Cladding Surface Temperature	Thermocouple	2	22.83 in. at 315° 26.77 in. at 135°
	Internal Gas Pressure	Pressure Transducer	1	Upper Plenum
	Cladding Elongation	LVDT	1	Top of Fuel Rod
807-6	Cladding Surface Temperature	Thermocouple	2	22.83 in. at 135° 26.77 in. at 315°
	Internal Gas Pressure	Pressure Transducer	1	Upper Plenum
	Cladding Elongation	LVDT	1	Top of Fuel Rod
807-8	Cladding Elongation	LVDT	1	Top of Fuel Rod
807-9	Cladding Surface Temperature	Thermocouple	2	26.77 in. at 315° 21.65 in. at 135°

(a) Rods 2, 4, 7, and 10 are not instrumented

(b) From bottom of fuel stack and 0° reactor north

TABLE 2. Fuel-Rod Instrumentation Requirements for OPTRAN 1-3

Rod ^(a)	Measurement	Instrument ^(b)	Instruments Per Rod	Instrument Locations ^(c)
903-1	Internal Gas Pressure	Pressure Transducer	1	Upper Plenum
	Cladding Surface Temperature	Thermocouple	2	19.72 in. at 315° 22.68 in. at 135°
903-3	Cladding Strain (Hoop)	Strain Gage	2	6.93 in. at 135° 14.80 in. at 135°
	Cladding Strain (Axial)	Strain Gage	2	6.93 in. at 135° 14.80 in. at 135°
903-5	Cladding Strain (Hoop)	Strain Gage	2	6.93 in. at 135° 14.80 in. at 135°
	Cladding Strain (Axial)	Strain Gage	2	6.93 in. at 315° 14.80 in. at 315°
903-6	Cladding Surface Temperature	Thermocouple	2	19.72 in. at 315° 22.68 in. at 315°
903-8	Cladding Strain (Hoop)	Strain Gage	2	6.93 in. at 135° 14.80 in. at 135°
	Cladding Strain (Axial)	Strain Gage	2	6.93 in. at 315° 14.80 in. at 315°

(a) Rods 2, 4, 7, and 9 are not instrumented

(b) Each instrumented rod has an LVDT

(c) From bottom of fuel stack and 0° reactor north

A special feature, a rod locking mechanism to permit easy replacement of the calibration rods, is illustrated in Figure 9. The calibration rods (RIA 1-7) give the steady-state flux profile (gamma scan). As indicated in Figure 9, the side notches on the lower end plugs of the fuel rods engage the square support bars. This creates the reference plane that both orients the rods in the bundle and determines the axial position of the fuel with respect to the PBF core. The rod retainers lock all rods in position axially.

FIGURE 9. Rod Locking Mechanism

Underwater, the tabs are pulled out about 1/4 in. and rotated as shown. The removable rods may now be extracted through the top of the shroud outlet housing and replaced. These two rods are not instrumented.

All of the instruments for the two tests, with the exception of the strain gages, are on order. Economic conditions have caused instrument delivery schedules to be delayed from anticipated deliveries in August to as late as November.

Now that the test-train design is complete and acquisition of the hardware is underway, greater effort has been applied to the development of remote handling equipment. Preliminary procedures for remote handling have been prepared and some of the hot-cell fixture designs are nearly complete.

The installation of strain gages on fuel rods for OPTRAN 1-3 may prove difficult as a result of materials incompatibility problems. The process involves welding stainless steel to Zircaloy, and the relative thermal expansion of these two metals may interfere with the desired elastic strain measurement. Currently, we are evaluating temperature compensation methods, including redesign of the gages.

GAP-CONDUCTANCE TEST ANALYSIS

The purpose of this task is to assist in evaluating PBF thermal data from which source information of fuel-cladding gap conductance may be extracted. During the past quarter, EG&G transmitted to Pacific Northwest Laboratory (PNL) the first actual digitized PBF transient data, consisting of fuel thermocouple steady-state and scram data (and power calibration data) from the two fresh, simulated pressurized water reactor rods in test LOC-3 (Rods 1 and 3).

Figure 10 shows the fuel temperature-vs-power curves for both rods and their centerline thermocouple response to reactor scram. The slight, apparent difference in thermal resistance between the two rods is confirmed by their transient response. Rod 3, which has a slightly lower slope of temperature versus power (hence lower thermal resistance), does exhibit relatively faster scram-temperature decrease, as was theorized, given the identical geometry and initial power of both rods.

FIGURE 10. Data From PBF, LOC-3 (Scram, Left; Steady State, Right)

The inferred gap conductances for both rods (at 40 kW/m) are about $10 \text{ kW/m}^2\text{-K}$, with the conductance for Rod 3 being about 8 to 10% higher than that for Rod 1. Since Rod 3 differed from Rod 1 only in helium pressurization (4.83 MPa in Rod 3 vs. 2.21 in Rod 1), it is tempting to ascribe the slight conductance difference to the pressurization. However, this is difficult to imagine. The inferred gap size for these rods at 40 kW/m is about 25 microns, whereas the temperature jump distances are a few tenths of microns (because of the high helium pressure), so the pressure effect on temperature jump distance is just not significant in this case. There is a slight increase in helium conductivity with gas pressure, but data by Ho and Liendenfrost⁽¹⁾ would indicate that this effect could account for only about 1/4 of the observed difference in inferred conductance. It is concluded, then, that there are simply slight differences in the effective gap sizes, which happen to favor Rod 3 in this case.

MULTI-ROD TEST TRAINS - ESSOR PROJECT

Work in support of the ESSOR Reactor Test Program completed during the quarter includes two main areas of technical interest: thermal-hydraulic analytical calculations and test-train design concepts. From the analytical calculations, bundle power requirements and heat losses were determined and shroud-insulation criteria were established. In addition, test-train design and instrumentation concepts consistent with the stated experimental objectives of the ESSOR and the proposed PBF High-Temperature Severe Damage programs were developed.

THERMAL-HYDRAULIC ANALYTICAL RESULTS

Three series of calculations have been performed in support of the ESSOR Reactor Test Program. The first involved the use of the TRUMP heat transfer code to assess radial heat losses due to radiation heat transfer only. The results of these calculations are summarized in Table 3. They show that quite large specific powers are required to sustain the peak temperatures planned for the ESSOR small-break tests. Only modest improvements can be achieved through

TABLE 3. Assembly Average Powers Required to Achieve a Given Peak Assembly Temperature for Various Postulated Shroud Types

Peak Assembly Temperature		Assembly Average Power		Case Examined
$^{\circ}\text{F}$	$(^{\circ}\text{C})$	kW/ft	(kW/m)	
3500	(1927)	14.30	(46.9)	Base Case; With No Radiation Shields
3000	(1649)	8.90	(29.2)	
3500	(1927)	12.80	(42.0)	Base Case With One Radiation Shield
3000	(1649)	7.71	(25.3)	
3500	(1927)	11.4	(37.4)	Base Case With Two Radiation Shields
3000	(1649)	6.71	(22.0)	
3500	(1927)	9.51	(31.2)	Pyrolytic Graphite Shroud
3000	(1649)	6.89	(22.6)	

the use of radiation shields if emissivity values are high in the range of 0.9 - 0.95. Greater improvements at the higher temperature will result from the use of low-conductivity insulation in the shroud. The calculations have shown that the improvement is much less at the lower temperature, which is a reflection of the fact that the improvement is due to a lower thermal conductance rather than a radiation shielding effect.

The second series of calculations was directed towards assessing the magnitude of heat removal caused by the small, but finite, steam flow axially transporting the heat. These calculations demonstrate that the steam temperature is a direct function of the distance above the liquid interface for a given axial power profile if radial heat losses are ignored (i.e., the radial-radiant and axial-convective solutions were not coupled). The results of the steam-temperature and steaming-rate calculations are shown in Table 4. The steaming rate was calculated in terms of a mass flow rate and also in terms of a reflood rate or inlet velocity. The lowest power, 3.4 kW/ft (10.3 kW/m) of assembly peak power or 2 kW/ft (6.56 kW/m) of assembly average power, is about the 1% decay heat level estimated for the Three Mile Island core at the time it was uncovered.

TABLE 4. Steaming Rates and Outlet Temperatures for ESSOR Small-Break Test Configurations

Assembly P_{peak} kW/ft(kW/m)	P_{total} (kW)	P_{above} Interface (kW)	Steam-Water Interface Level ft(m)	Steaming Rate			h_{vapor} Btu/lb(J/g)	T_{out} °F(°C)	Inlet Water Velocity	
				(g/s)	(lb/s)	(lb/hr)			(in./s)	(cm/s)
3.14(10.30)	12.00	9.00	2(0.61)	1.97	0.00434	15.6	1968(4577)	3660(2016)	0.0335	0.0851
3.14(10.30)	12.00	6.00	3(0.91)	3.94	0.00868	31.2	656(1526)	1595(868)	0.0669	0.170
5(16.40)	19.10	14.32	2(0.61)	3.14	0.00691	24.9	1968(4577)	3660(2016)	0.0533	0.135
5(16.40)	19.10	9.55	3(0.91)	6.27	0.138	49.7	656(1526)	1595(868)	0.106	0.269
10(32.81)	38.20	28.65	2(0.61)	6.27	0.138	49.7	1968(4577)	3660(2016)	0.106	0.269
10(32.81)	38.20	19.19	3(0.91)	12.50	0.276	99.4	656(1526)	1595(868)	0.213	0.541
15(49.21)	57.30	42.98	2(0.61)	9.40	0.207	74.5	1968(4577)	3660(2016)	0.160	0.406
15(49.21)	57.30	28.65	3(0.91)	18.80	0.0415	149.0	654(1526)	1595(868)	0.320	0.813
20(65.62)	76.40	57.30	2(0.61)	12.50	0.0276	99.4	1968(4577)	3660(2016)	0.213	0.541
20(65.62)	76.40	38.20	3(0.91)	25.10	0.0553	199.0	655(1526)	1595(868)	0.426	1.080

The third and final series of thermal-hydraulic calculations in this reporting period combined the steaming-rate calculations with the radiation-loss calculations. The results are shown in Figure 11, which gives the axial-rod temperature and steam-temperature distributions for a case of an average assembly power of 20 kW/ft (65.62 kW/m). Figure 11 also shows the axial temperature distribution for the case of an average assembly power of 13 kW/ft (42.65 kW/m) under a flow condition, i.e., losses by radiation only.

FIGURE 11. Sample Calculated Axial Temperatures

The foregoing simple calculation shows that an assembly with an average power of 20 kW/ft (56.62 kW/m) with a cosine axial power distribution and a steam-water interface at 1.54 meters above the bottom of the fuel column will reach peak clad temperatures slightly above 220K (3500°F). The temperatures reach a peak around one meter above the water interface because metal-water reaction has not been included in these calculations.

Preliminary results for calculations completed by General Electric Vallecitos Nuclear Center using the THTD code indicate that somewhat higher assembly powers, perhaps as much as 30 kW/ft (98.43 kW/m) average power, may be required to sustain the same temperatures. In general, however, the TRUMP and THTD results compare favorably. The assembly powers can be roughly divided by 32 to obtain an indication of rod average power.

TEST-TRAIN CONCEPTUAL DESIGN

Test-train conceptual designs have been developed for both the ESSOR and the proposed PBF test programs. The ESSOR-SUPERSARA Program⁽²⁾ involves both Small- and Large-Break LOCA Experiments while the PBF⁽³⁾ Small-Break Test Series is directed at High-Temperature Severe Damage Experiments.

Test Trains for ESSOR

The basic test assembly concept for the SUPERSARA Program is shown in Figure 12. It is designed to accommodate both large- and small-break experiments. Features common to both types of experiments are:

- arrangement of baffles and stabilized ZrO_2 spheres to prevent water falling back into test bundle
- doubly-contained insulated shroud
- instrumented fuel bundle assembly with the instrument hardlines exiting the bundle from the bottom of the assembly
- hardline routing up to top closure head on the outside of the insulated shroud between the shroud and pressure tube
- fuel particle and debris catcher at bottom of assembly.

FIGURE 12. ESSOR Large- and Small-Break Test-Assembly Concept

Three other features could be added to accommodate experimental conditions unique to the more severe small-break tests. Starting at the bottom of the test train assembly, there are devices to catch debris and fragments and split the bypass water flow to maintain the required flow around the outside of the insulated shroud and the inside of the pressure tube. Next are the hardline outlet housings, which provide the path for the instrument lines to exit from the bottom of the test assembly. The doubly-contained insulated shroud encloses the two-meter fuel bundle region, which includes pressure switches, pressure transducers, and the stabilized bed of ZrO_2 spheres. The proposed 32-fuel pin arrangement and the high-temperature insulated shroud is shown in greater detail in the cross-section view of Figure 12. Above the insulated shroud in the test assembly is a water-spray steam desuperheater and a series of baffles to help keep water from falling back into the test bundle. The desuperheater will only be needed in the higher-temperature, small-break tests. A biological shield plug is contained in the region above the spray desuperheater and below the top closure head.

PBF Test Train

A test-train conceptual design for the PBF High-Temperature Severe Damage test was also developed. The design incorporates many of the features discussed above and shown in Figure 13. These include the doubly-contained insulated shroud, the 32-pin fuel bundle region, and the stabilized bed of ZrO_2 spheres.

The major components included in the proposed PBF High-Temperature, Small-Break LOCA test assembly are:

- inlet and outlet fuel particle and debris catch screens
- inlet and outlet turbine flow meters
- arrangement of stabilized ZrO_2 spheres to prevent water fallback into the test bundle
- doubly-contained insulated shroud
- instrumented fuel-bundle assembly with instrument hardlines exiting the bundle from the bottom of the assembly

- hardline routing up to the top closure head on the outside of the insulated shroud and in the water downcomer channel between the shroud and in-pile liner
- possible device to indicate and control the liquid level
- possible probe to measure hydrogen-gas evolution.

Figure 13 includes an overall view of the test-train assembly concept. The use of a 32-rod test bundle, as illustrated in the cross-section portion of the figure, will provide a commonality in design between the ESSOR and PBF test-train assemblies and should make it easier to compare the experimental results of both assemblies. Beginning at the bottom, below the assembly, is the debris and fragment catcher attached to the lower end of the in-pile tube liner. At the bottom of the assembly proper is the flow splitter, which maintains the required bypass water flow between the outer shroud and the in-pile tube liner. The remaining major components of the assembly below the fuel bundle region are the reflood lines that come from the top closure head, a flow straightener, a flow turbine, and the hardline outlet housings that provide the path for the instrument hardlines to exit from the bottom of the test assembly.

The one-meter fuel bundle region includes the pressure switches, pressure transducers, and the bed of stabilized ZrO_2 spheres that prevent water fall-back. All are enclosed within the doubly-contained insulated shroud. Above the shroud is a noninsulated hanger tube, a flow turbine, and a porous-metal filter. This filter will allow steam to flow from inside the test assembly and mix with the bypass water, while preventing water from flowing back into the test assembly. Above the porous-metal filter is the hardware that connects to the hanger tube and suspends the test train from the top closure head.

FUTURE PLANS

TASK A - PBF TEST-TRAIN HARDWARE

A strain-gage development program will be initiated in the second quarter and will be directed towards obtaining solutions to some of the problems that have been identified.

FIGURE 13. PBF High-Temperature, Small-Break Test-Assembly Concept

All procurement contracts will be placed in the second quarter. Some instruments and most hardware should be available by the end of the third quarter for check out and assembly.

All nonirradiated assembly work will be completed during the fourth quarter with final assembly of all modules scheduled for completion in the first quarter of 1981.

TASK B - GAP-CONDUCTANCE ANALYSIS

A review of the preliminary PR-1 results will be performed to determine if useful fuel-rod thermal-resistance data can be obtained.

TASK C - DUAL-PURPOSE ASSEMBLIES

The IFA-513 assembly is currently being irradiated at Halden. No further work is planned until this assembly is ready for discharge near mid-CY 1981.

TASK D - MULTI-ROD TEST TRAINS - ESSOR PROJECT

A report will be issued on the small-break test-train concepts. Program objectives and schedules will be discussed in meetings with the ESSOR staff at PNL. The feasibility of new instrumentation concepts will be evaluated. Preliminary material studies on shrouds and test-train insulation and structural components will be performed. Lastly, a report will be issued on the thermal analysis performed to date including the power requirements for the SUPERSARA test matrix.

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Department of Atmospheric
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