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# Heavy-Section Steel Technology Program Quarterly Progress Report for January–March 1978

G. D. Whitman
R. H. Bryan

MASTER

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

OAK RIDGE NATIONAL LABORATORY
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# HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY—MARCH 1978

G. D. Whitman R. H. Bryan

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# PREFACE

The Heavy-Section Steel Technology (HSST) Program, which is sponsored by the Nuclear Regulatory Commission (NRC), is an engineering research activity devoted to extending and developing the technology for assessing the margin of safety against fracture of the thick-walled steel pressure vessels used in light-water-cooled nuclear power reactors. The program is being carried out in close cooperation with the nuclear power industry. This report covers HSST work performed January through March 1978, except for subcontractor contributions which may cover the three-month period ending in February. The work performed by Oak Ridge National Laboratory (ORNL) and by subcontractors is managed by the Engineering Technology Division. Major tasks at ORNL are carried out by the Engineering Technology Division and the Metals and Ceramics Division. Prior progress reports on this program are ORNL-4176, ORNL-4315, ORNL-4377, ORNL-4463, ORNL-4512, ORNL-4590, ORNL-4653, ORNL-4681, ORNL-4764, ORNL-4816, ORNL-4855, ORNL-4918, ORNL-4971, ORNL/TM-4655 (Vol. II), ORNL/TM-4729 (Vol. II), ORNL/TM-4805 (Vol. II), ORNL/TM-4914 (Vol. II), ORNL/TM-5021 (Vol. II), ORNL/TM-5170, ORNL/NUREG/TM-3, ORNL/NUREG/TM-28, ORNL/NUREG/TM-49, ORNL/NUREG/TM-64, ORNL/NUREG/TM-94, ORNL/NUREG/TM-120, ORNL/NUREG/TM-147, ORNL/NUREG/TM-166, and ORNL/NUREG/TM-194.

SUMMARY

## 1. PROGRAM ADMINISTRATION AND PROCUREMENT

The Heavy-Section Steel Technology (HSST) Program is an engineering research activity conducted by the Oak Ridge National Laboratory (ORNL) for the Nuclear Regulatory Commission (NRC) in coordination with other research sponsored by the federal government and private organizations. The program comprises studies relating to all areas of the technology of the materials fabricated into thick-section primary-coolant containment systems of light-water-cooled nuclear power reactors. The principal area of investigation is the behavior and structural integrity of steel pressure vessels containing cracklike flaws. Current work is organized into the following tasks: (1) program administration and procurement, (2) fracture mechanics analyses and investigations, (3) effect of high-temperature primary reactor water on the subcritical crack growth of reactor vessel steels, (4) investigations of irradiated materials, (5) pressure vessel investigations, (6) thermal shock investigations, and (7) foreign research.

The work performed under the four existing research and development subcontracts is included in this report.

Nine program briefings or presentations were made during the quarter, and one technical report was issued.

## 2. FRACTURE MECHANICS ANALYSES AND INVESTIGATIONS

Stress-intensity factor distributions along the tip of nozzle corner cracks in models of intermediate test vessels (ITVs) and boiling-water reactor (BWR) vessels have been measured by photoelastic techniques. Four nozzles with cracks in a plane perpendicular to the cracks previously studied have been tested, and results have been compared quantitatively with earlier results.

## 3. EFFECT OF HIGH-TEMPERATURE PRIMARY REACTOR WATER ON THE SUBCRITICAL CRACK GROWTH OF REACTOR VESSEL STEEL

Testing of specimens in three environmental chambers is continuing in this study of the effects of ramp and hold times and high $\Delta K$. Rapid ramp tests have been completed. The high $\Delta K$ test has been run continuously for nine months and is expected to be completed soon.

Four fatigue specimens were tested in hydrogen sulfide in order to investigate the influence of the environment on the apparent starting $\Delta K$ effect observed in specimens in a PWR environment. In hydrogen sulfide, varying the starting $\Delta K$ did not affect the crack growth rate.

## 4. INVESTIGATIONS OF IRRADIATED MATERIALS

Slow-bend tests were performed on the fatigue precracked Charpy specimens from the second 4T-CTS irradiation experiment. Irradiation of the three capsules of specimens in the third 4T-CTS irradiation experiment was completed with an estimated average fast-neutron fluence in the 4T specimens of $8 \times 10^{18}$ neutrons/cm$^2$ ($E > 1$ MeV).

A hot-cell setup for testing weld tensile specimens of the second 4T-CTS irradiation was completed; the test matrix for the irradiated weld tensile properties was established. Data acquisition system programming for fracture testing is in progress.

The cooperative investigation of unloading compliance techniques for J-integral testing is under way to evaluate the effects of various loading and measurement arrangements and data processing on the sensitivity of the determination of elastic compliance. Three different means of reducing the effects of friction on the loading pins have been tried. Attachment of clip gages for measuring load-line displacement has been made with both a razor blade knife edge and a machined knife edge; the former gave less hysteresis and better linearity in loading and unloading displacements.

## 5. PRESSURE VESSEL INVESTIGATIONS

The block cut from intermediate test vessel V-7B after the test was returned to ORNL and sectioned for examination of the fracture surfaces. It was determined that the initial sharpened flaw was located within the repair weld heat-affected zone (HAZ) as intended. The crack extension generally was outside the HAZ but closely followed the weld interface.

The machining of a notch in vessel V-8 in preparation of the flaw in that vessel was completed. Cyclic pressurization of the notch to sharpen it by fatigue was commenced with ultrasonic and acoustic emission surveillance. Additional work was done to improve the toughness characterization of the submerged-arc fabrication weld in which the flaw will reside. It was determined that $K_{Icd}$ measurements were about the same from WT- and WL-oriented $PCC_V$ specimens.

Flaw C in vessel V-6 was removed from the vessel and split apart to reveal the extension of the crack. This crack had grown stably a small amount prior to failure of the vessel at another flaw.

Two crack-arrest specimens of the MRL type were tested in the course of preparations for participating in the NRC/EPRI cooperative crack-arrest tests.

## 6. THERMAL SHOCK INVESTIGATIONS

Spray procedures for coating the inner surface of the flow test vessel were developed, and the vessel was subjected to two thermal shocks with liquid nitrogen to obtain thermal-hydraulic data. Modifications to the test facility were completed. A parametric analysis of the PWR double-ended-pipe-break LOCA-ECC was completed, and preliminary calculations pertaining to a PWR steam-line break were made.

## 7. FOREIGN RESEARCH

Lists of foreign reports published in *Nuclear Safety* through Vol. 19 (No. 3) have been reviewed to identify topics of interest in the metallurgy and materials areas. Translated copies of reports of interest have been requested, and translations received are being reviewed. A summary of research programs in the Federal Republic of Germany, France, and Japan was prepared.

# HEAVY-SECTION STEEL TECHNOLOGY PROGRAM QUARTERLY PROGRESS REPORT FOR JANUARY—MARCH 1978

G. D. Whitman    R. H. Bryan

ABSTRACT

The Heavy-Section Steel Technology (HSST) Program is an engineering research activity conducted by the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. It comprises studies related to all areas of the technology of the materials fabricated into thick-section primary-coolant containment systems of light-water-cooled nuclear power reactors. The principal area of investigation is the behavior and structural integrity of steel pressure vessels containing cracklike flaws. Current work is organized into seven tasks: (1) program administration and procurement, (2) fracture mechanics analyses and investigations, (3) effect of high-temperature primary reactor water on subcritical crack growth of reactor vessel steels, (4) investigations of irradiated materials, (5) pressure vessel investigations, (6) thermal shock investigations, and (7) foreign research.

Stress-intensity factors are being measured photoelastically for nozzle corner cracks out of the plane of the intersecting cylinder axes. The study of ramp- and hold-time effects in fatigue testing is continuing, and the influence of the environment on the apparent starting ΔK effect was investigated. Precracked Charpy specimens of irradiated weld metal with low ductile shelf toughness were tested in slow bending, and preparations for J-integral testing of larger specimens in this series are continuing. The machined portion of the flaw of vessel V-8 was completed and fatiguing of the notch was started. Further experimental studies of enhanced heat transfer for cryogenic thermal shock tests were made.

## 1. PROGRAM ADMINISTRATION AND PROCUREMENT

G. D. Whitman

The Heavy-Section Steel Technology (HSST) Program, a major safety program sponsored by the Nuclear Regulatory Commission (NRC), is concerned with the structural integrity of the primary systems, particularly the reactor pressure vessels of light-water-cooled nuclear power reactor stations. The structural integrity of these vessels is ensured by designing and fabricating them according to the standards set by the code for nuclear pressure vessels, by detecting flaws of significant size that occur during fabrication and in service, and by developing methods capable of producing quantitative estimates of conditions under which fractures could occur. The program is concerned mainly with developing pertinent fracture technology. It deals with the development of knowledge of the material used in these thick-walled vessels, the rate of growth of flaws, and the combination of flaw size and load that would cause fracture and thus limit the life and/or operating conditions for this reactor plant.

The program is coordinated with other government agencies and the manufacturing and utility sectors of the nuclear power industry in the United States and abroad. The overall objective is a quantification of safety assessments for regulatory agencies, professional code-writing bodies, and the nuclear power industry. Several of the activities are conducted under subcontracts by research facilities in the United States and through informal cooperative efforts on an international basis. Four research and development subcontracts are currently in force.

Administratively, the program is organized into seven tasks, as reflected in this report: (1) program administration and procurement, (2) fracture mechanics analyses and investigations, (3) effect of high-temperature primary water on subcritical crack growth of reactor vessel steels, (4) investigations of irradiated material, (5) pressure vessel investigations, (6) thermal shock investigations, and (7) foreign research.

During this quarter, nine program briefings, reviews, or presentations were made by the HSST staff at technical meetings and at program reviews for the NRC staff or for visitors. One technical report was issued.[1]

## Reference

1. R. H. Bryan et al., *Test of 6-in.-Thick Pressure Vessels. Series 3: Intermediate Test Vessel V-7A Under Sustained Loading*, ORNL/NUREG-9 (February 1978).

# 2. FRACTURE MECHANICS ANALYSES AND INVESTIGATIONS

## 2.1 Stress-Intensity Factors for Nozzle Cracks in Reactor Vessels*

C. W. Smith†
W. H. Peters† T. S. Fleishman†

This chapter is intended to serve as an extension of the progress report[1] for the period October through December 1977. In order to be as brief as possible, reference will be made to that document.

To date two models (four nozzles) with type C cracks (Fig. 2.1) in the range $a/T \approx 0.15$ to 0.55 have been tested and are currently being analyzed. Results indicate that the material in one of the models may have been defective, and this test is being replicated in order to verify this conjecture. Several observations can be made at this point.

1. Type C cracks require approximately twice the pressure of type A cracks in order to cause flaw growth, indicating a strong influence of the vessel wall nominal stress.

2. Type C cracks remain in their initial planes during growth.

3. Type C cracks do not show the flattening in the central region observed in type A cracks. In fact, the moderately deep flaws bulge beyond the quarter-ellipse shape in the central region (Fig. 2.2).

4. Stress-intensity factor (SIF) distributions for cracks in the range $a/T \approx 0.25$ to 0.55 are concave downward like those for type A cracks. Quantitative SIF distributions are currently being obtained.

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*Research performed by the Photoelasticity and Fracture Laboratory in the Engineering Science and Mechanics Department at the Virginia Polytechnic Institute and State University under Subcontract 7015 between Union Carbide Corporation and Virginia Polytechnic Institute and State University.

†Department of Engineering Science and Mechanics, Virginia Polytechnic Institute and State University.

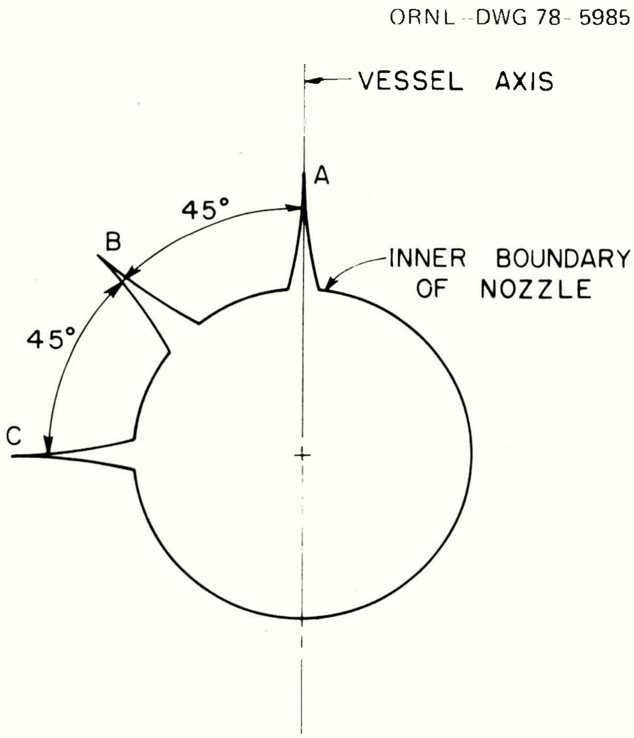


Fig. 2.1. Crack orientation relative to vessel axis (A, phases I and II; B and C, phase III).

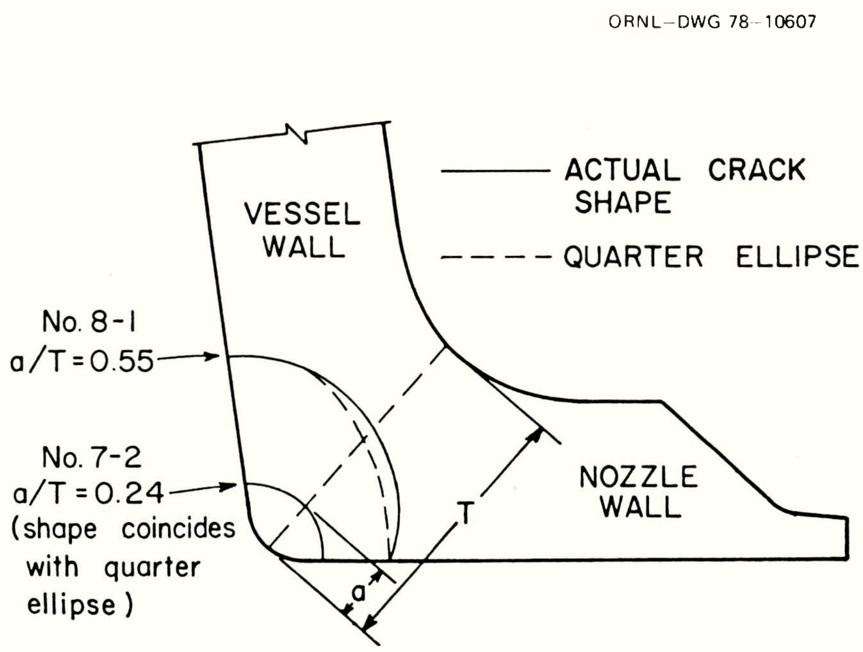


Fig. 2.2. Flaw shapes for type C cracks in a nozzle corner.

## Reference

1. C. W. Smith and W. H. Peters, "Stress Intensity Factors for Nozzle Cracks in Reactor Vessels," *Heavy-Section Steel Technology Program Quart. Prog. Rep. October–December 1977*, ORNL/NUREG/TM-194, pp. 3–24.

# 3. EFFECT OF HIGH-TEMPERATURE PRIMARY REACTOR WATER ON THE SUBCRITICAL CRACK GROWTH OF REACTOR VESSEL STEELS*,†

W. H. Bamford†† L. J. Ceschini††

The objective of this continuing program is to characterize the fatigue-crack-growth rate properties of ferritic vessel steels exposed to PWR primary-coolant environments. Three environmental chambers are being used, and the following areas are being investigated:

| | |
|---|---|
| Ramp- and hold-time effects (2T WOL specimens) | 1 chamber (14 MPa, 288°C)<br>1 chamber (0.14 MPa, 93°C) |
| Crack growth rate at high $\Delta K$ (4T CT specimens) | 1 chamber (14 MPa, 288°C) |

## 3.1 Ramp- and Hold-Time Effects

The investigation of ramp- and hold-time effects continued during the reporting period. Specimens of A508 class 2 forging material are being tested at R ($K_{min}/K_{max}$) = 0.2 in two PWR environments, one at 288°C and another at 93°C. The specimens are 51.8-mm-thick WOL- and CT-type specimens, and the loadings are being applied according to the matrix requirements of Table 3.1. Testing is complete on test type "a" and nearly complete on test type "b" in both environments. As the tests are completed, results are compared with those of the Naval Research Laboratory, where comparable tests are being conducted. As the rise and hold times increase, the tests require progressively more time to complete; therefore, no tests have been completed during this reporting period.

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*Work sponsored by HSST program under UCCND Subcontract 3290 between Union Carbide Corporation and Westinghouse Electric Corporation.

†Conversions from SI to English units for all SI quantities are listed on a foldout page at the end of this report.

††Westinghouse Electric Corporation.

Table 3.1. Projected ramp- and hold-time tests of 2T-WOL specimens in PWR environment, A508 class 2 forging material[a]

| Test | Ramp time (min) | Hold time (min) |
|---|---|---|
| a1 | Rapid, 1 sec | 1 |
| a2 | Rapid, 1 sec | 3 |
| a3 | Rapid, 1 sec | 12 |
| a4 | Rapid, 1 sec | 60 |
| b1 | 1 | 1 |
| b2 | 1 | 3 |
| b3 | 1 | 12 |
| b4 | 1 | 60 |
| c1 | 5 | 1 |
| c2 | 5 | 3 |
| c3 | 5 | 12 |
| c4 | 5 | 60 |
| d1 | 30 | 1 |
| d2 | 30 | 3 |
| d3 | 30 | 12 |
| d4 | 30 | 60 |

[a]Tests a4, b4, c3, c4, d2, d3, d4 (which are very long time tests) will be done after the others are complete.

## 3.2 Crack Growth at High $\Delta K$

Testing continued on specimen F-24, a 101.6-mm-thick compact specimen of A508 class 2 forging material at 1 cpm and $R = 0.2$. The size of this specimen allows testing to be conducted at higher values of applied $\Delta K$ than have previously been possible, but the added ligament area of the specimen also means that the testing time is longer. This test has continued for nine months and is still not complete. The results thus far indicate that the test will be completed early in the next reporting period.

## 3.3 Starting Condition Effects

It was reported earlier[1] that the initially applied loadings on a specimen in a water environment can significantly affect the amount of environmental enhancement produced. If the specimen is started at an applied stress-intensity factor range $\Delta K$ that is too high, the crack growth rate results will show less enhancement than that of identical specimens started at a lower $\Delta K$ value. This behavior is illustrated in Fig. 3.1, which shows results[2] from several specimens of pressure vessel steel tested in a PWR environment at $R = 0.2$. These specimens were started at values of $\Delta K$ ranging from approximately 21 to 44 $MN \cdot m^{-3/2}$ and show markedly different crack growth behavior.

This phenomenon has been observed, but not identified as a separate effect, for several other low-alloy steels exposed to a water environment.[3-5] The effect is also strikingly similar to data obtained for water-induced cracking under static loads for higher strength steels.[6,7] This phenomenon may be due to non-steady-state crack growth that is controlled by the kinetics of the steel-water system. To investigate the degree to which the kinetics control this process, a series of pressure vessel steel specimens was tested in a hydrogen sulfide environment. Earlier tests had shown that this environment produced crack growth rates for this steel very similar to those produced in a much slower test in a water environment. An example is shown in Fig. 3.2. The test frequency is not important for the hydrogen sulfide tests because the kinetics of the environmental interaction are so fast.

To study the effects of starting conditions, four specimens of HSST plate 04 were tested in an environment of room-temperature hydrogen sulfide at 414 kPa. The initial applied $\Delta K$ was varied from 19 to 72 $MN \cdot m^{-3/2}$ through different applied loads; and as a separate check, the initial $\Delta K$ for one specimen was elevated by a longer starting crack length rather than higher load. In all cases the R ratio was very low, 0.01 to 0.07. Results of this series of tests, summarized in Fig. 3.3, show that there is no effect of starting conditions for this environment. This leads to the conclusion that starting conditions can have an effect on the crack growth rate data only for material-environment combinations where the kinetics of the system are relatively slow. Thus far, such effects have only been observed in water environments, where slow loadings (around 1 cpm) have been applied. Studies of these effects are continuing and results will be reported as available.

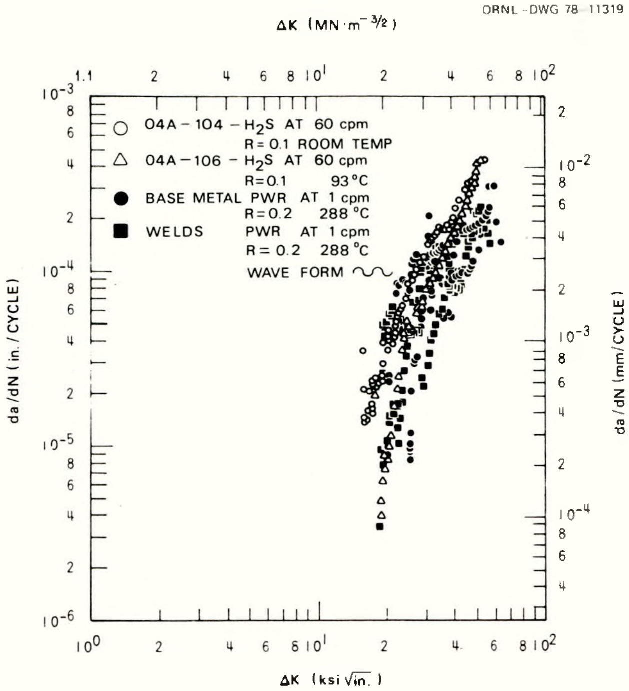


Fig. 3.1. Example of the effect of starting conditions on fatigue crack growth in pressure vessel steel in PWR environment (1T-CT data from Ref. 2).

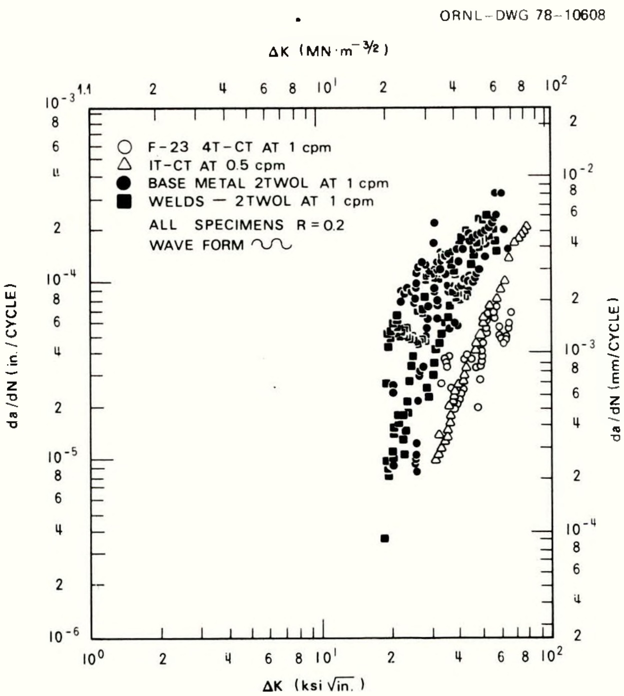


Fig. 3.2. Comparison of fatigue crack growth in pressure vessel steels in hydrogen sulfide and PWR environment — low R ratio.

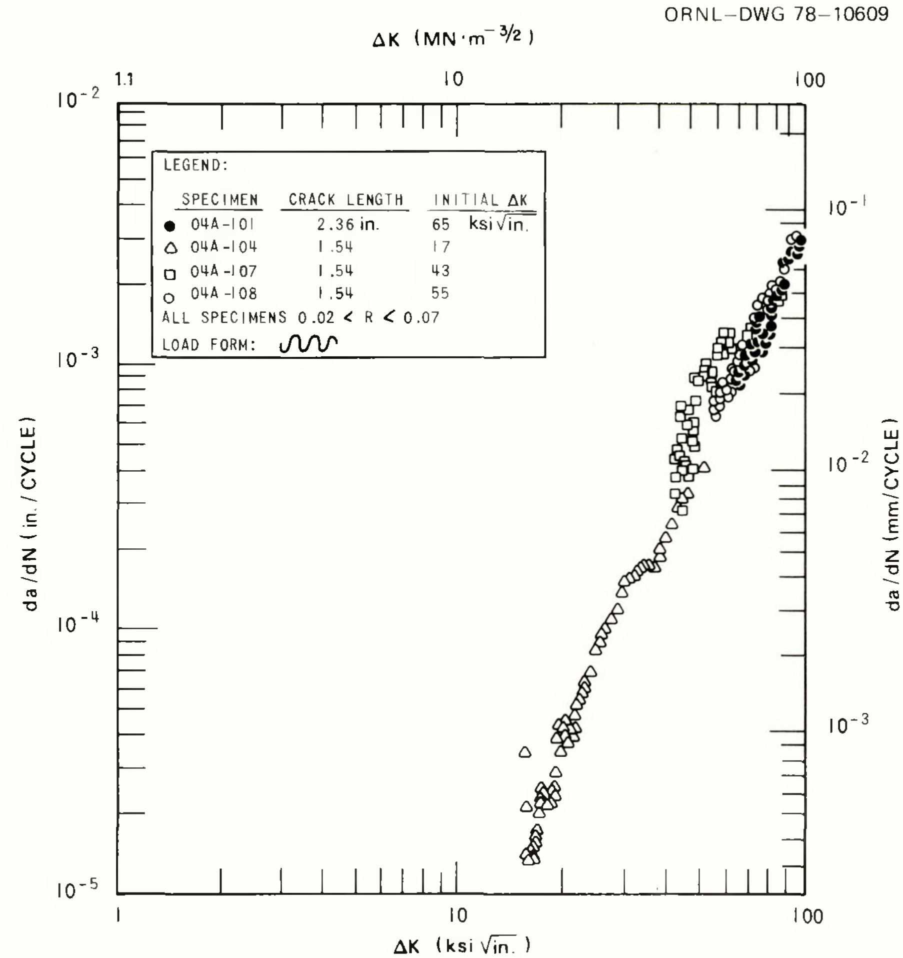


Fig. 3.3. Effect of starting conditions on fatigue-crack-growth rates in reactor pressure vessel steel in hydrogen sulfide at room temperature.

## References


1. W. H. Bamford, D. M. Moon, and L. J. Ceschini, "Crack Growth Rate Testing in Reactor Pressure Vessel Steels," *Proceedings of Fifth Water Reactor Safety Information Meeting, Gaithersburg, Md.* (November 1977).

2. H. E. Watson, F. J. Loss, and B. H. Menke, "Fatigue Crack Propagation in LWR Materials," *Structural Integrity of Water Reactor Pressure Boundary Components — Progress Report Ending May 31, 1977*, Naval Research Laboratory, Washington, D.C.

3. P. M. Scott and D. R. V. Silvester, "The Influence of Mean Tensile Stress on Corrosion Fatigue Crack Growth in Structural Steel Immersed in Seawater," United Kingdom Atomic Energy Authority Harwell Corrosion Service, Interim Technical Report UKOSRP 3/02 (May 1977).

4. G. A. Miller, S. J. Hudak, and R. P. Wei, *J. Testing Evaluation* 1, 524 (1973).

5. P. S. Pao, W. Wei, and R. P. Wei, "Effect of Frequency on Fatigue Crack Growth Response of AISI 4340 Steel in Water Vapor," Lehigh University Report IFSM-77-85 (October 1977).

6. J. D. Landes and R. P. Wei, "The Kinetics of Subcritical Crack Growth Under Sustained Loading," *Int. J. Fracture* 9 (September 1973).

7. R. P. Wei, S. R. Novak, and D. P. Williams, "Some Important Considerations in the Development of Stress Corrosion Cracking Test Methods," *Mat. Res. Std.* 12(9) (1972).

# 4. INVESTIGATION OF IRRADIATED MATERIALS*

## 4.1 Toughness Investigations of Irradiated Materials

R. G. Berggren J. W. Woods
T. N. Jones D. A. Canonico

### 4.1.1 Second 4T-CTS irradiation study

Slow-bend tests were conducted on the fatigue precracked Charpy specimens from the second 4T-CTS irradiation experiment, and results are being analyzed. Final neutron dosimetry analyses will be completed in the next quarter.

### 4.1.2 Third 4T-CTS irradiation study

Irradiation of capsules A and B of the third 4T-CTS irradiation experiment was completed February 17, 1978, after 1012 hr of irradiation. Irradiation of capsule C was completed March 28, 1978, after 1581 hr. These irradiation times for the three capsules should provide average fast-neutron fluences in the 4T-CT specimens of about $8 \times 10^{18}$ neutrons/cm$^2$ ($E > 1$ MeV). Disassembly of these capsules is planned for the next quarter.

## 4.2 Effects of Irradiation on Pressure Vessel Steels†

J. A. Williams††

### 4.2.1 Tensile testing of irradiated welds from second 4T irradiations

The hot-cell test setup for conducting irradiated tensile tests was completed. An extensometer system, which is active during the total specimen elongation, was developed as part of the test setup; direct

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*Conversions from SI to English units for all SI quantities are listed on a foldout page at the end of this report.

†Research performed under Purchase Order 11Y-50917V for the Oak Ridge National Laboratory, operated by Union Carbide Corporation under contract to the U.S. Energy Research and Development Administration (now DOE).

††Hanford Engineering Development Laboratory.

computerization of test results is possible through the use of this type extensometer. The in-cell test setup is shown in Fig. 4.1. The extensometer system measures deformation over a 1.27-cm-long gage length of the miniature specimen. Dual LVDTs are algebraically summed to cancel anomalous readings due to possible bending moments during the initial portion of the test. Calibrated accuracy of the extensometer is better than 0.001% linearity.

A verification of the extensometer in actual test application was conducted by high-resolution recording of the elastic load-deflection curve; linearity of the recorded curve was better than 1% and no hysteresis was observed on unloading. The elastic modulus of the load deflection was observed to be $2.04 \times 10^5$ MPa; the modulus determined is within the known range for this class of material.

A tensile test matrix was developed to maximize the tensile information obtained from the available specimens. Table 4.1 gives the distribution of specimens by irradiation temperature and neutron fluence. The specimens in parentheses have irradiation temperatures in more than one range for a significant period of the irradiation time. The specimen test temperatures are shown in brackets for each specimen. The test temperatures will yield a range of data which can be treated by a rate-temperature parameter to maximize property characterization; the tensile test strain rate will be 0.01 $sec^{-1}$. Tests will be conducted during the next quarter.

### 4.2.2 Fracture test system computerization

Test system computerization is being developed for fracture testing. The fluid nature of ductile fracture testing and analysis demands that a practical method of data storage, retrieval, and reprocessing be available for future reanalysis; second, data of sufficiently high resolution and quantity must be obtained for statistically accurate determinations of ductile fracture toughness. The only reasonable way to obtain this information is by sophisticated data acquisition systems and programming.

An on-line acquisition system currently being interfaced and programmed at HEDL is shown in Fig. 4.2. Figure 4.3 gives a functional block diagram of the acquisition system and its capabilities. Programming of

ORNL Photo 1636-78

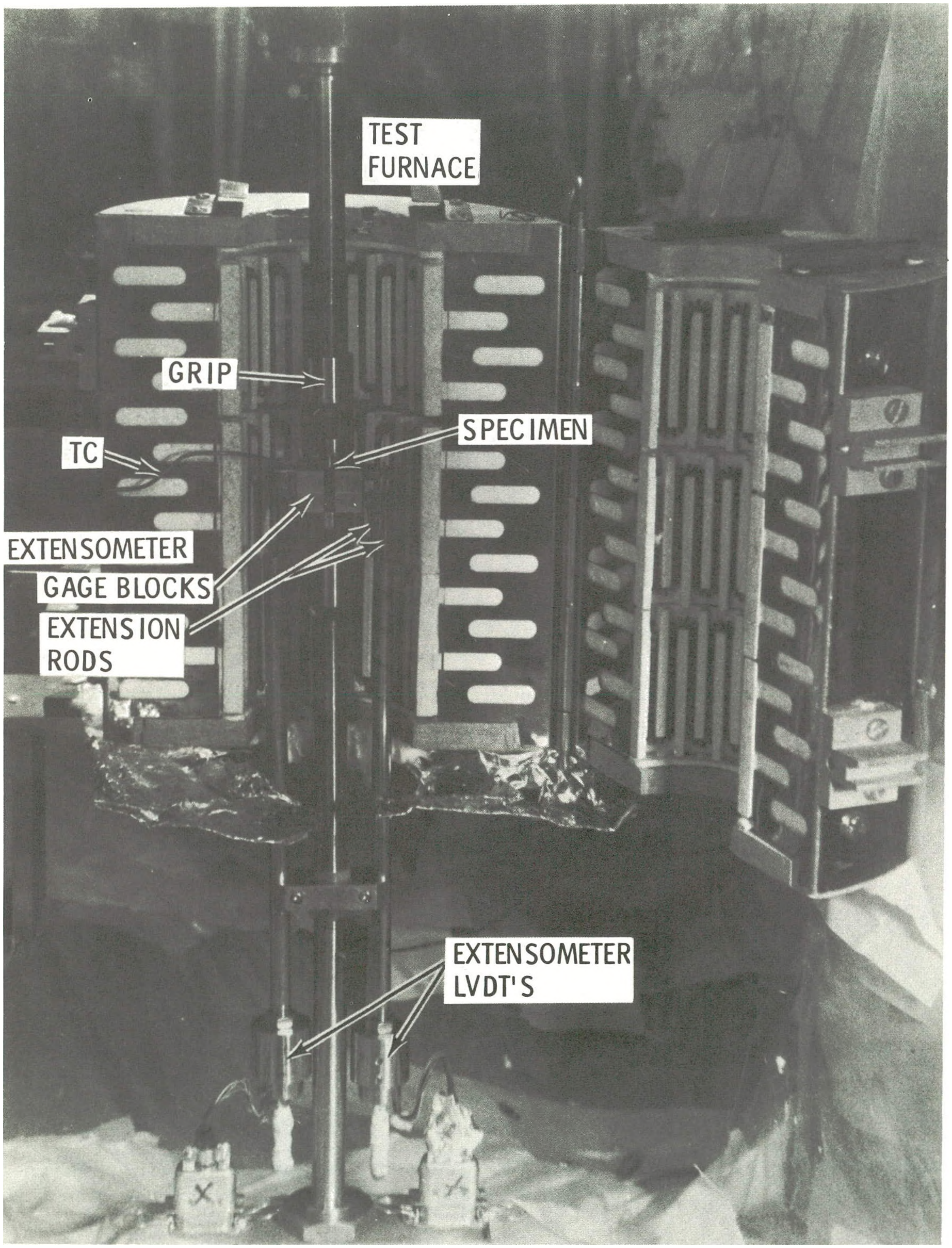


Fig. 4.1. Hot-cell test setup for irradiated tensile testing of specimens from 4T irradiations.

Table 4.1. Irradiation fluence and temperature distribution for tensile specimens from second 4T irradiations[a]

| | Fluence (neutrons/cm$^2$ × $10^{18}$) | Irradiation temperature (°C) | | | | |
|---|---|---|---|---|---|---|
| | | 232–260 | 260–293 | 293–316 | 316–343 | 343–371 |
| Capsule A, weld 61W | 3 | | | | 61W-9 [RT] | 61W-8 [RT] |
| | 7 | | | 61W-5 [RT]<br>61W-6 [288] | 61W-7 [RT] | |
| | 15 | | 61W-3 [RT]<br>61W-4 [149] | | | |
| | 19 | | 61W-1 [RT]<br>61W-2 [288] | | | |
| Capsule B, weld 62W | 7 | (62W-2) [RT] | (62W-13) [149]<br>(62W-19) [288] | (62W-2)<br>62W-3 [RT] | | |
| | 14 | | | (62W-1) [RT] | | |
| | 16 | | 62W-10 [RT]<br>62W-11 [288] | | | |
| | 21 | | 62W-8 [RT]<br>62W-9 [149] | | | |
| Capsule C, weld 63W | 6 | | | (63W-14) [RT]<br>(63W-15) [RT] | (63W-15) | |
| | 7 | | 63W-5 [RT]<br>63W-9 [288] | | | |
| | 10 | | (63W-10) [RT] | | (63W-10) | |
| | 13 | | 63W-3 [RT]<br>63W-4 [149] | | | |
| | 16 | | 63W-1 [RT]<br>63W-2 [288] | | | |

[a]Brackets indicate test temperatures (°C) at a strain rate of 0.01 $sec^{-1}$; parentheses indicate specimens with multiple irradiation temperatures.

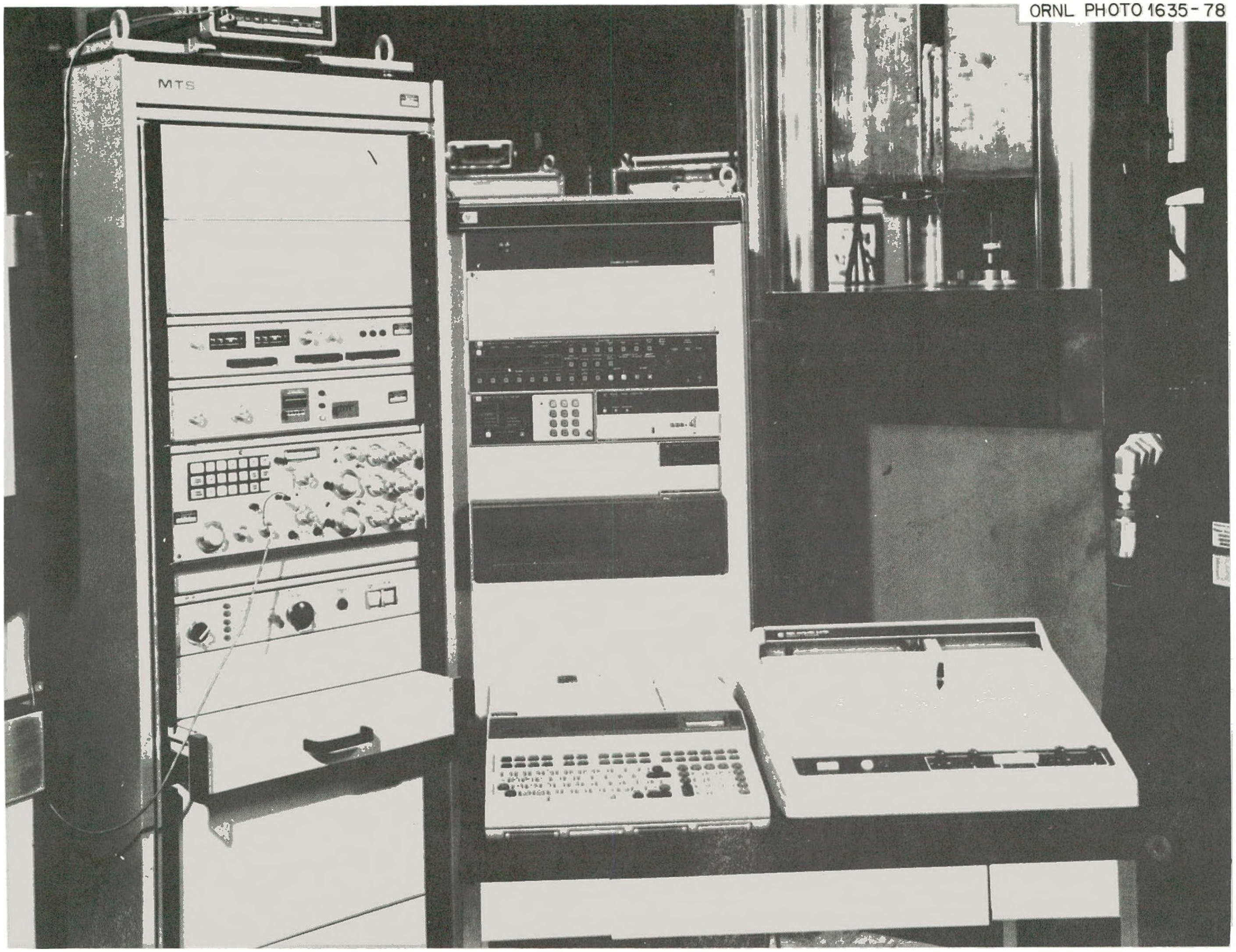


Fig. 4.2. Fracture test data acquisition and processing system.

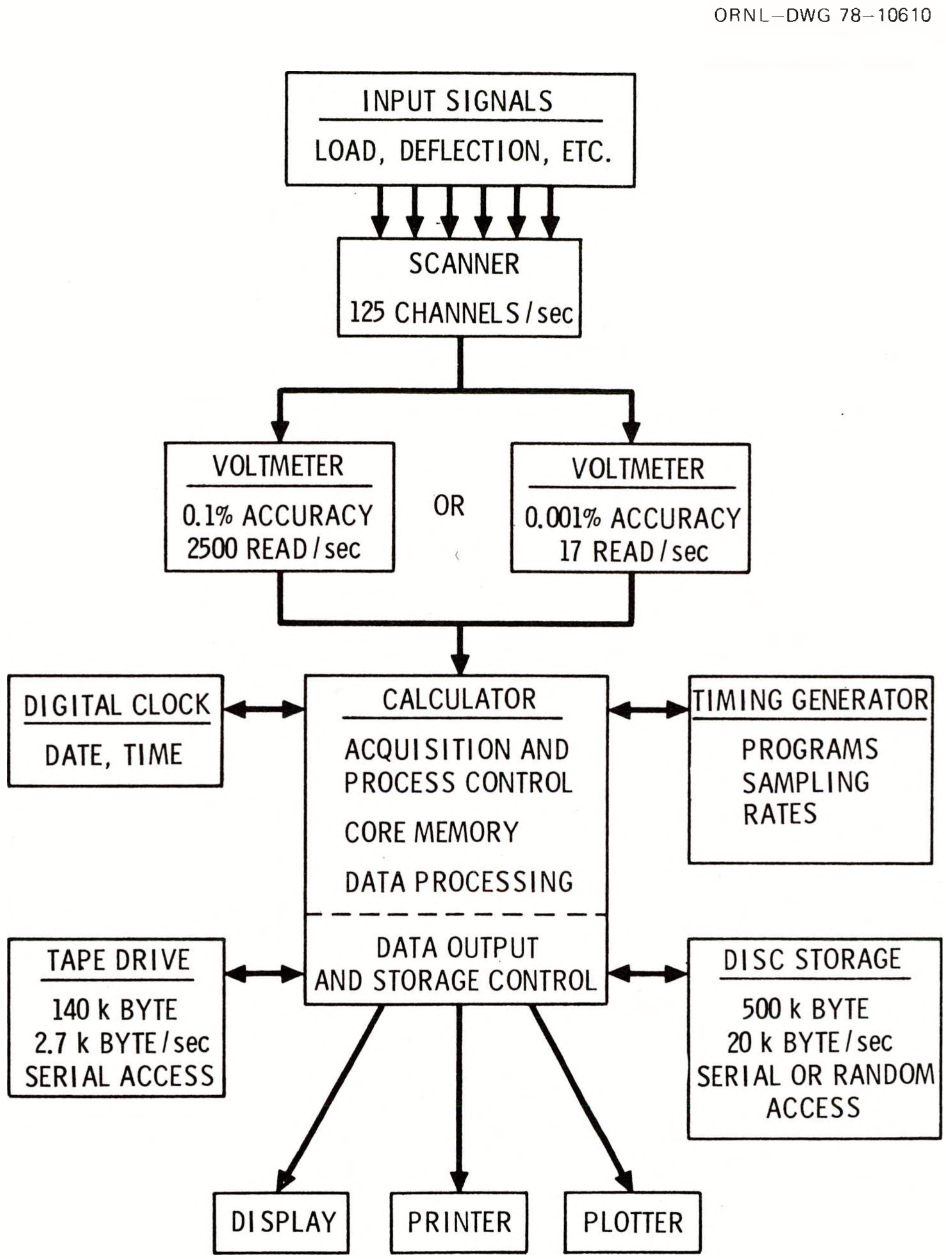


Fig. 4.3. Block diagram of data acquisition and processing system, showing general functions and capabilities.

fracture toughness calculations has been completed, debugged, and verified with actual experimental data. Real-time data acquisition and processing programs are currently being developed.

### 4.3 Evaluation of the Unloading Compliance Technique for Single Specimen J-Integral Testing*

G. A. Clarke† J. D. Landes†

#### 4.3.1 Introduction

The goal of the first test series of this program was to evaluate the unloading compliance test by computerized methods. Testing was conducted on an A508 material using 1T compact specimens to evaluate the effects of various types of loading pin arrangements and knife edges on the load-displacement records used in the unloading compliance method. In order to evaluate the sensitivity required for the unloading compliance values necessary for an accurate determination of $J_{Ic}$, a simple analysis of the data generated on the A508 material was completed.

While the applicability of the Merkle-Corten[1] correction factors has been experimentally verified,[2] questions still remain as to the form of the equations to use. The more general form of the equation used to calculate the correction factors required the separation of the J integral into its elastic and plastic parts, $J_e$ and $J_p$, respectively. A more approximate form of the equation utilizes the total J-integral value, that is, $J_T = J_e + J_p$. The applicability of this simplified form of the Merkle-Corten equation in the elastic region has been shown by an analysis comparing the alternative forms. It was concluded that, for $a/w > 0.5$, the approximate form of the equation can be used without appreciable loss in accuracy. The following sections describe the work completed in this program to date.

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*Work sponsored by HSST Program under UCCND Subcontract 7394 between Union Carbide Corporation and Westinghouse Electric Corporation.

†Westinghouse Electric Corporation.

### 4.3.2 Computerization of J-integral techniques

The computerization of the testing techniques originally created a number of previously unknown problems. The principal problem was the noise from the hydraulic and the electrical systems. Considerable improvement in the program output was made by using a combination of both active bandpass analog filters and a digital filtering system in the computer. Further evaluation of the computer method is presently under way.

### 4.3.3 Sensitivity required for $J_{Ic}$ testing

In order to determine the sensitivity required for an accurate $J_{Ic}$ value, an acceptable scatter band on J was required. The authors assumed that a scatter band of ±10% on $J_{Ic}$ would be considered adequate. It was determined that if the crack extension were measured within 0.38 mm of the actual value of stable crack growth for the A508 test material, the value of $J_{Ic}$ could be determined within ±10%. While this value of crack extension is well within the present sensitivity of the 1T test procedure, it is not known whether we have enough sensitivity to measure within the required accuracy for a 4T specimen. Tests will be performed shortly to determine this.

### 4.3.4 Effects of loading pins and knife edges on the load-displacement record

In the past, considerable attention has been given to the type of loading pin arrangements necessary to eliminate the effects of friction on the loading pins. The methods used to eliminate friction include roller bearings in the loading clevis or flat bottom holes in the clevis with undersized round loading pins. These methods were evaluated by amplifying the load-displacement records for each test using the various loading arrangements. There was no significant difference in the results obtained from either the roller bearing method or the flat bottom hole method. A wider hysteresis was found in the load-vs-displacement curve on the test using the undersized pins. This may indicate a higher degree of friction than found in the other two methods.

The normal compact specimen design for the unloading compliance test employs a razor blade knife edge. The thin, sharp point of the razor blade reduces friction between the knife edge and the clip gage. Previously, knife edges on the load line of the specimen were machined into the specimen. In order to determine the difference between the razor blade knife edge and the machined-in knife edge, tests were conducted on both specimen types. The load vs load line displacement record for the razor blade knife edge is shown in Fig. 4.4. The relatively small amount of hysteresis and the parallel nature of both the loading and the unloading slopes should be noted. The load vs load line displacement curve for the specimen with a machined-in knife edge is shown in Fig. 4.5. The fact that the unloading curve has a distinctly different slope than the loading curve indicates a considerable amount of friction between the clip gage and the machined-in knife edge.

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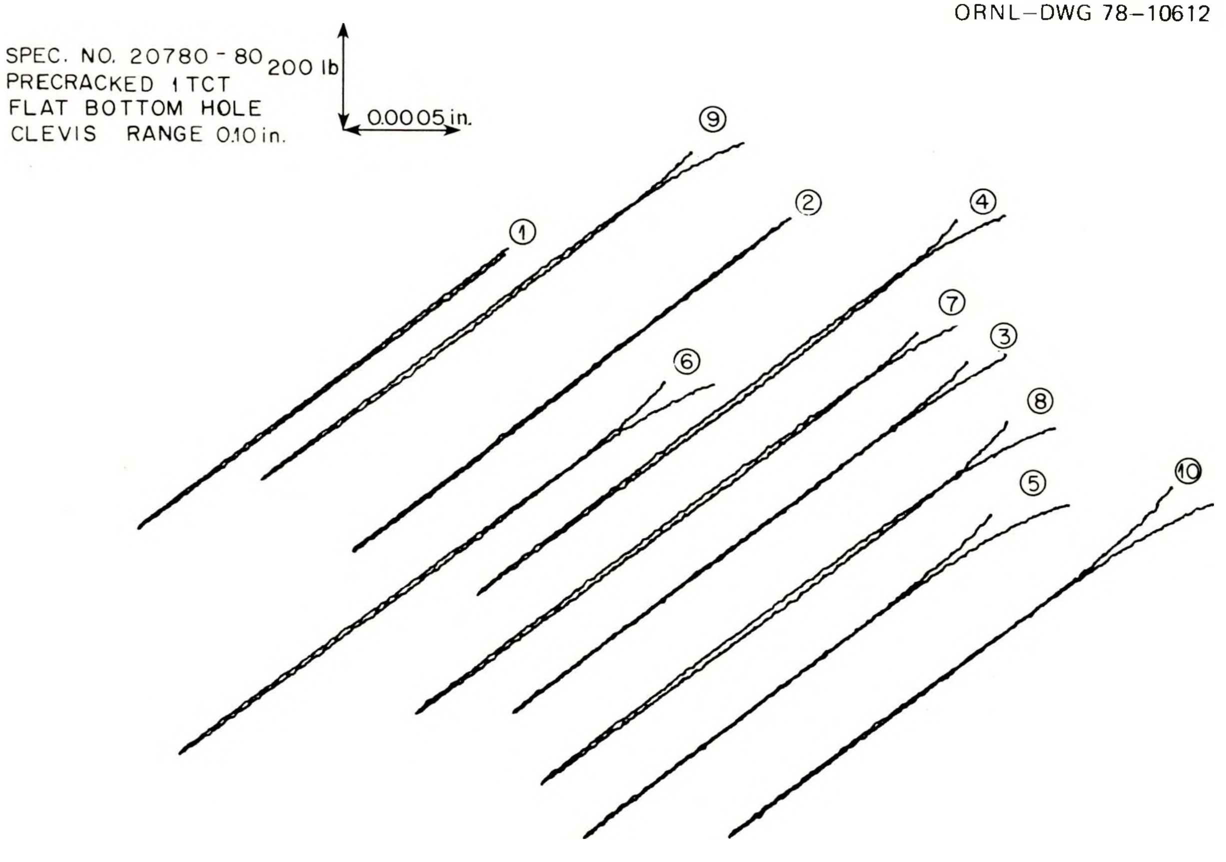


Fig. 4.4. Amplified load-displacement record for specimen with machined-in knife edge.

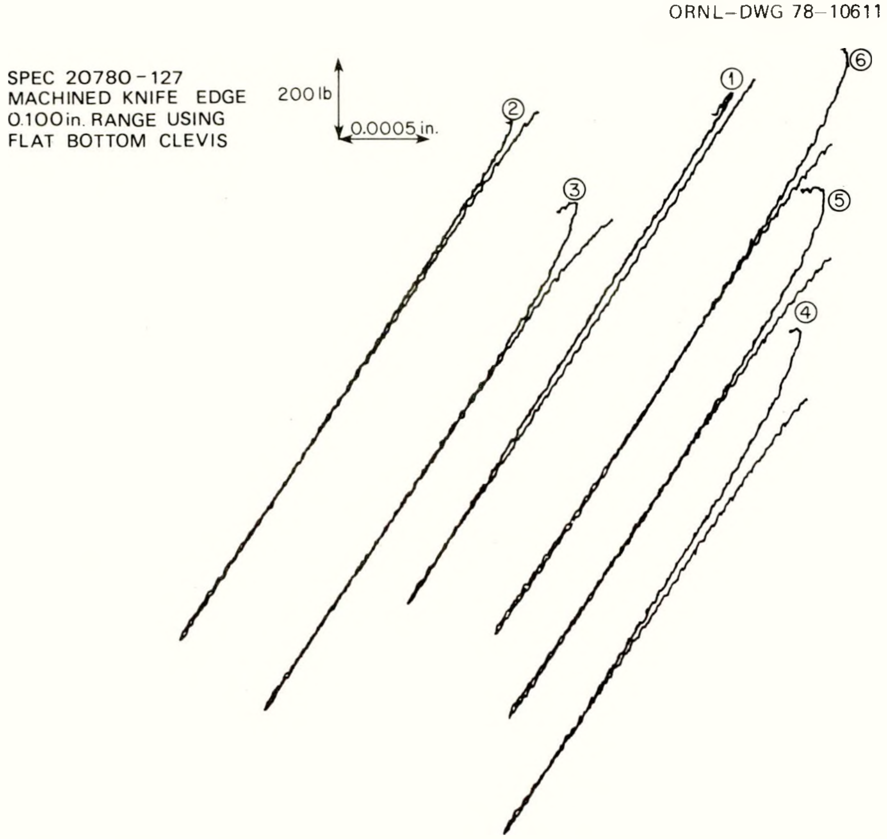


Fig. 4.5. Amplified load-displacement record for specimen with razor blade knife edge.

## References

1. J. G. Merkle and H. T. Corten, "A J Integral Analysis for the Compact Specimen, Considering Axial Force as Well as Bending Effects," *Trans. ASME*, pp. 286–92, November 1974.

2. J. D. Landes, H. Walker, and G. A. Clarke, "Evaluation of Estimation Procedures Used in J Integral Testing," presented at the ASTM Symposium on Elastic Plastic Fracture Mechanics, Atlanta, Ga., November 1977.

# 5. PRESSURE VESSEL INVESTIGATIONS*

## 5.1 Posttest Examination of V-7B Flaw

P. P. Holz    R. H. Bryan

The block of metal containing the flaw in intermediate test vessel V-7B was cut from the vessel for ultrasonic and destructive examination.[1] The block was examined by the Southwest Research Institute and the University of Michigan and returned to ORNL during the reporting period.

The block was saw cut transversely in 16 places as shown in Fig. 5.1. These cuts exposed cross-sectional views of the initial crack and the crack extension at 14 planes (28 faces). The etched faces of these cross sections show that the initial electron-beam weld crack lies entirely within the heat-affected zone (HAZ) of the weld repair as had been desired (see Fig. 5.2).

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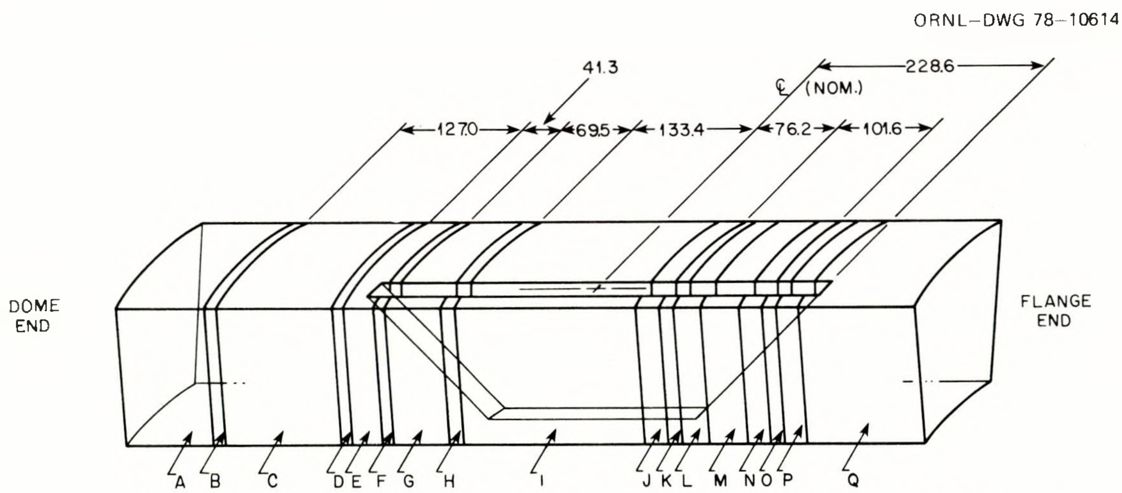
*Conversions from SI to English units for all SI quantities are listed on a foldout page at the end of this report.




| SLAB | NOMINAL THICKNESS |
|---|---|
| B, D, F, H, K, O | 10 |
| J, L, N, P | 13 |

Fig. 5.1. Block cut from intermediate test vessel V-7B containing entire repair weld and fracture zone. Slabs A through Q were cut after all ultrasonic examinations were completed.


Fig. 5.2. Face $K_f$ (the face of slab K nearest the flange end of the vessel) of cross section of V-7B fracture.

Segments J, K, and L (Fig. 5.1) fell in two pieces, since they were completely fractured during the test. Segments C, M, and Q were chilled in liquid nitrogen and broken open to expose the fracture surfaces. The two segments including the crack tips at both ends of the flaw (C and Q) are shown in Figs. 5.3 and 5.4, respectively. The visual examination of these pieces confirms the location of the crack front determined ultrasonically by K. K. Klindt.[1] It also appears that the crack extension lies entirely in the base metal in some regions, in weld metal in others, and in the HAZ only at crossover points. Figure 5.4 shows that the crack front curvature conforms to the interface of the weld and base metal. The location of the crack extension surfaces relative to the three types of metal (base, HAZ, and weld) will be determined by additional metallographic examination.

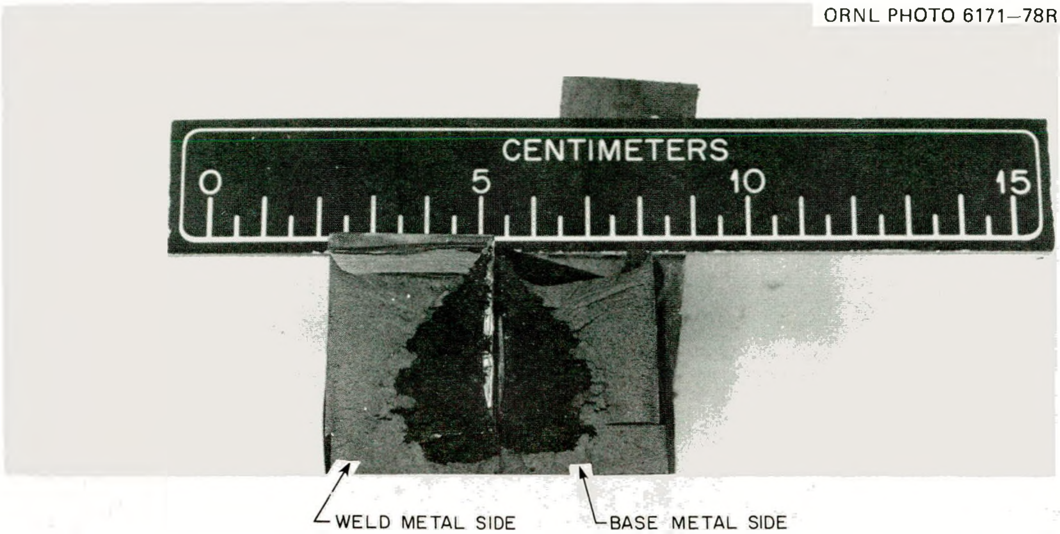



Fig. 5.3. V-7B fracture surfaces in slab C exposed by splitting the chilled slab.

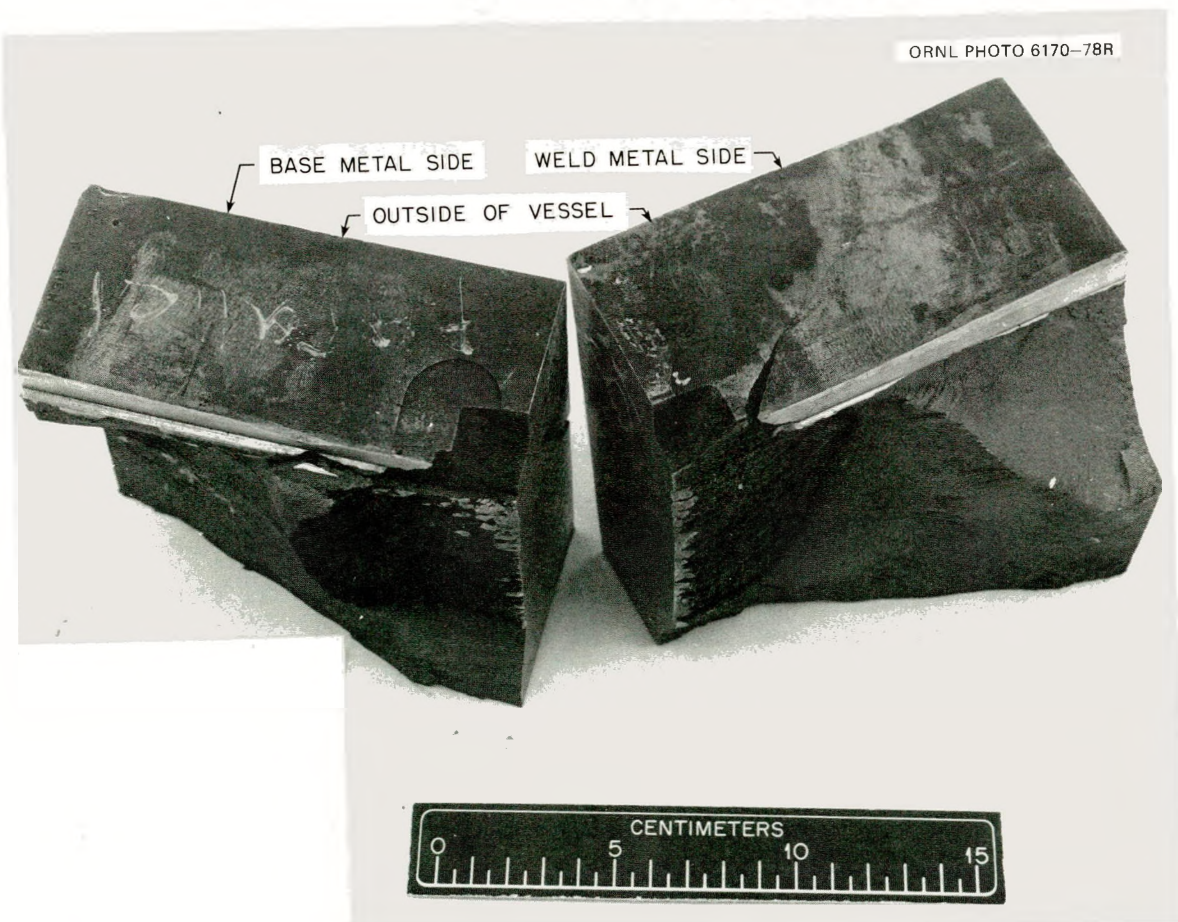



Fig. 5.4. V-7B fracture surfaces in slab Q exposed by splitting the chilled slab.

## 5.2 Preparations for Intermediate Test Vessel V-8 Test

P. P. Holz

A 6.35-mm-wide longitudinal slot was machined into the center cylinder submerged-arc fabrication-seam weld of V-8, a distance of 19 mm from the approximately half-wall-thickness weld-repair edge, as shown in Fig. 5.5. A slitting saw cutter ground to a 30° included angle with a 0.127- to 0.191-mm tip radius was used for the final cuts to provide the tip of the notch with a sharp edge. A special stainless steel insert provided

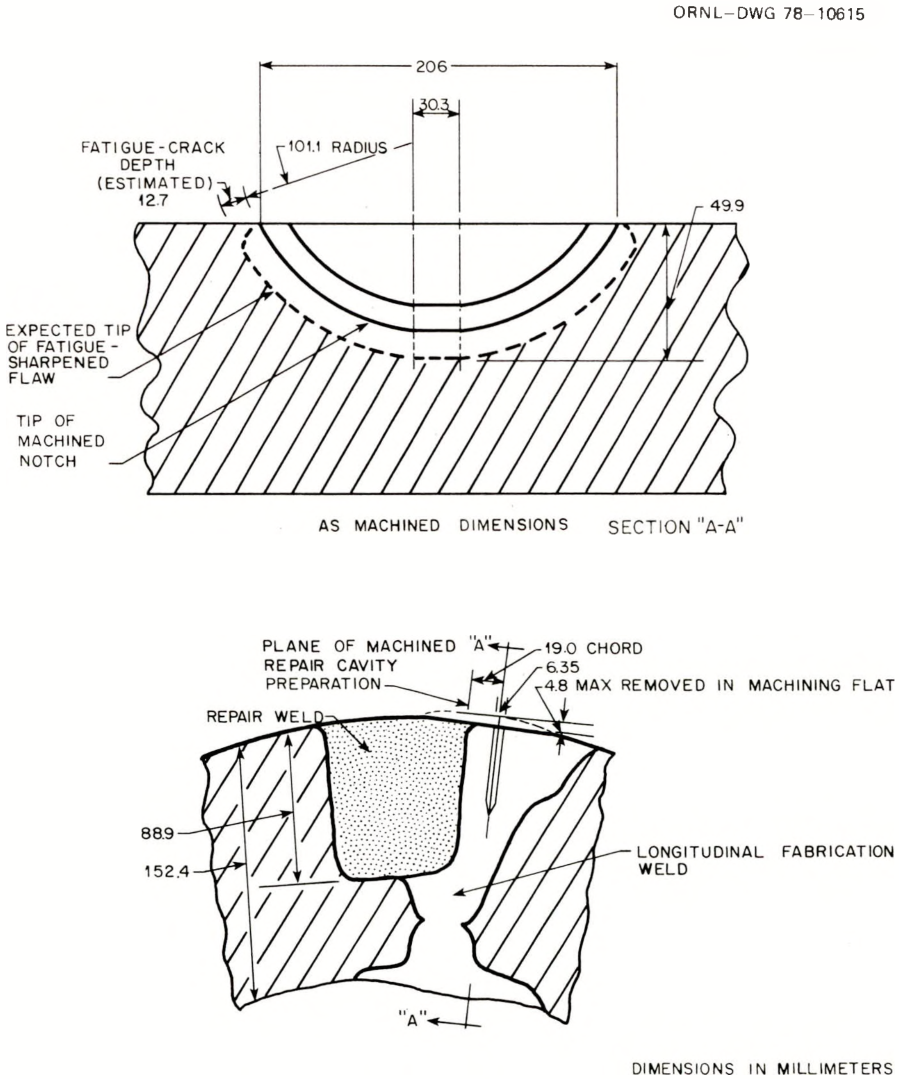



Fig. 5.5. Flaw details for intermediate test vessel V-8.

a tight-fitting plug for the slot and channeled the hydraulic pressurization fluid to the notch tip perimeter. Details of the machining and flaw pump setup duplicated the V-9 mockup previously used for prototype testing,[2] except that a large 152-mm Giddings and Lewis vertical boring mill was used for slotting the horizontally placed vessel as shown in Fig. 5.6.

Fifty strain gages were installed on the vessel in the region of the flaw to monitor changes in strain that occur as a result of residual stress relieved as the flaw is introduced. Tensile residual stresses were observed on the strain gages along the cut plane as the slitting saw first penetrated into the vessel wall. Later, during initial cyclic hydraulic pressurization of the notch to grow the flaw, strain readings taken in similar places implied that the original residual stress noted during slotting was considerable.

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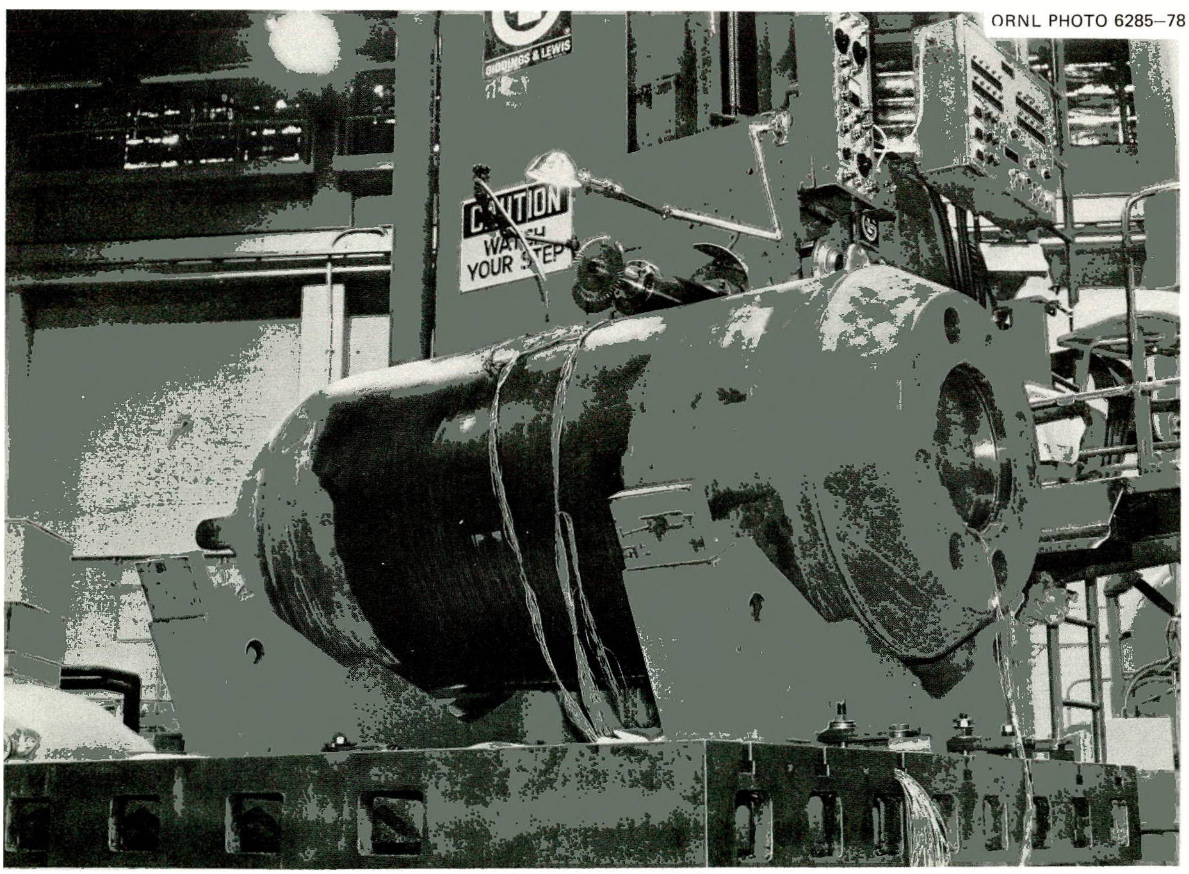


Fig. 5.6. Vertical boring mill slot machining for intermediate test vessel V-8.

Cyclic notch pressurization is in progress with both ultrasonic (UT) and acoustic-emission (AE) instrumentation being used to observe fatigue crack growth. Information from the UT instruments will indicate crack depth, while the AE instruments should indicate when crack extension occurs. The peak pressure used in the initial cyclic loading for fatiguing is being held to less than 83 MPa so that the $K_I$ of the machined notch is not greater than 24 $MN \cdot m^{-3/2}$ in the absence of residual stress, thereby minimizing overload on the fatigue crack because of superimposed residual stress.

## 5.3 Characterization of the Repair Weld in Vessel V-8

### 5.3.1 Fracture toughness investigations of the fabrication weld
(W. J. Stelzman and D. A. Canonico)

We have continued to determine the static fracture toughness ($K_{Icd}$) of the ITV-8 fabrication weld. Test results from WL-oriented (crack propagating in the welding direction) precracked Charpy-V specimens ($PCC_V$) are shown in Fig. 5.7. Also shown are the results from WT-oriented $PCC_V$ specimens from the V-8 prolongation fabrication weld and WL-oriented 1T and 2T compact specimens (CS) from the V-9 prolongation fabrication weld. These data have been previously reported.[3,4]

A comparison of the $PCC_V$ $K_{Icd}$ results from WL-oriented specimens with the results from WT-oriented specimens show that nearly all the results from the WL-oriented specimens fall within the scatter band of the WT-oriented results in the −46 to 10°C test range. Only one specimen, tested at −18°C, fell outside the WT-oriented scatter band. Therefore, based on the limited number of $PCC_V$ specimens tested, we conclude that the effect of orientation (W or L) on $K_{Icd}$ is minimal. This similarity in toughness is important since the vessel flaw will advance in either or both of these directions during testing.

Additional weld metal fracture toughness specimens, both $PCC_V$ and CS, from the ITV-8 and ITV-9 prolongation fabrication welds are being prepared.

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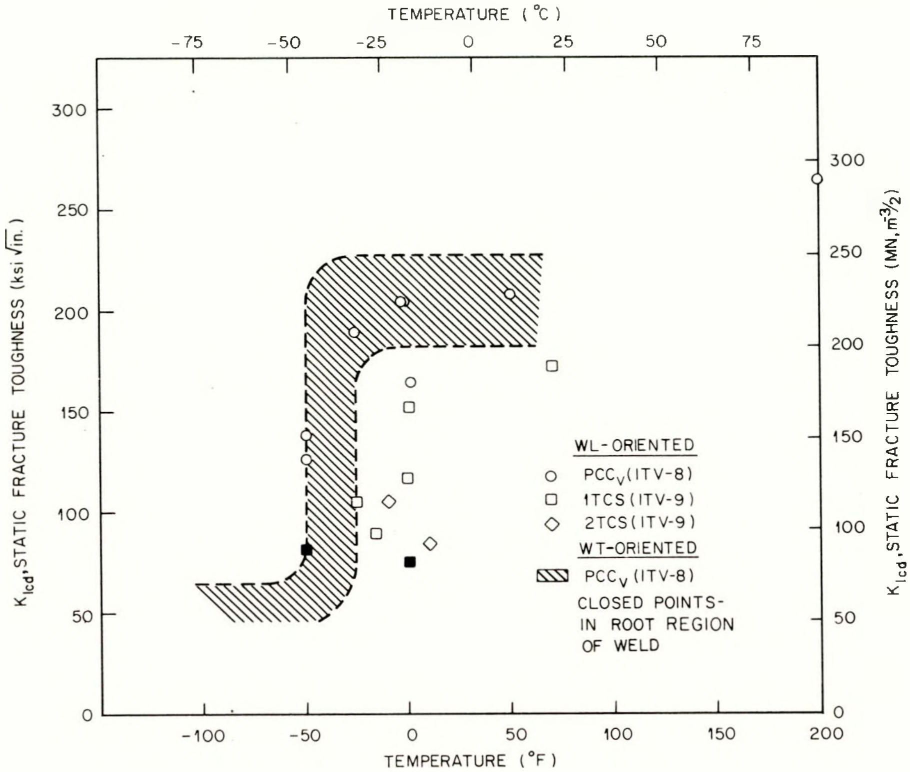


Fig. 5.7. Variation of static fracture toughness of the fabrication weld metal in the ITV-8 and ITV-9 prolongations from precracked Charpy and 1T and 2T compact specimens.

### 5.3.2 Fractographic examination of ITV-8 fabrication weld (D. A. Canonico and R. S. Crouse)

Previously,[5] we reported a relationship between fracture toughness and the amount of dimple fracture at the fatigue crack tip in $PCC_V$ specimens. Subsequently,[4] a relationship was established among the microstructure at the $PCC_V$ crack tip, the test temperature, and the fracture toughness. An example of this relationship is shown in Fig. 5.8. The narrow band of dimple fracture at the initiation site is $1 \times 10^{-3}$ mm. The fracture toughness of this specimen (44 MPa $\sqrt{m}$ at −46°C) was among the lowest obtained for the fabrication weld from the ITV-9 prolongation. Similar comparative studies are being conducted for $PCC_V$ specimens that exhibited higher fracture toughness values at the −46°C test temperature.

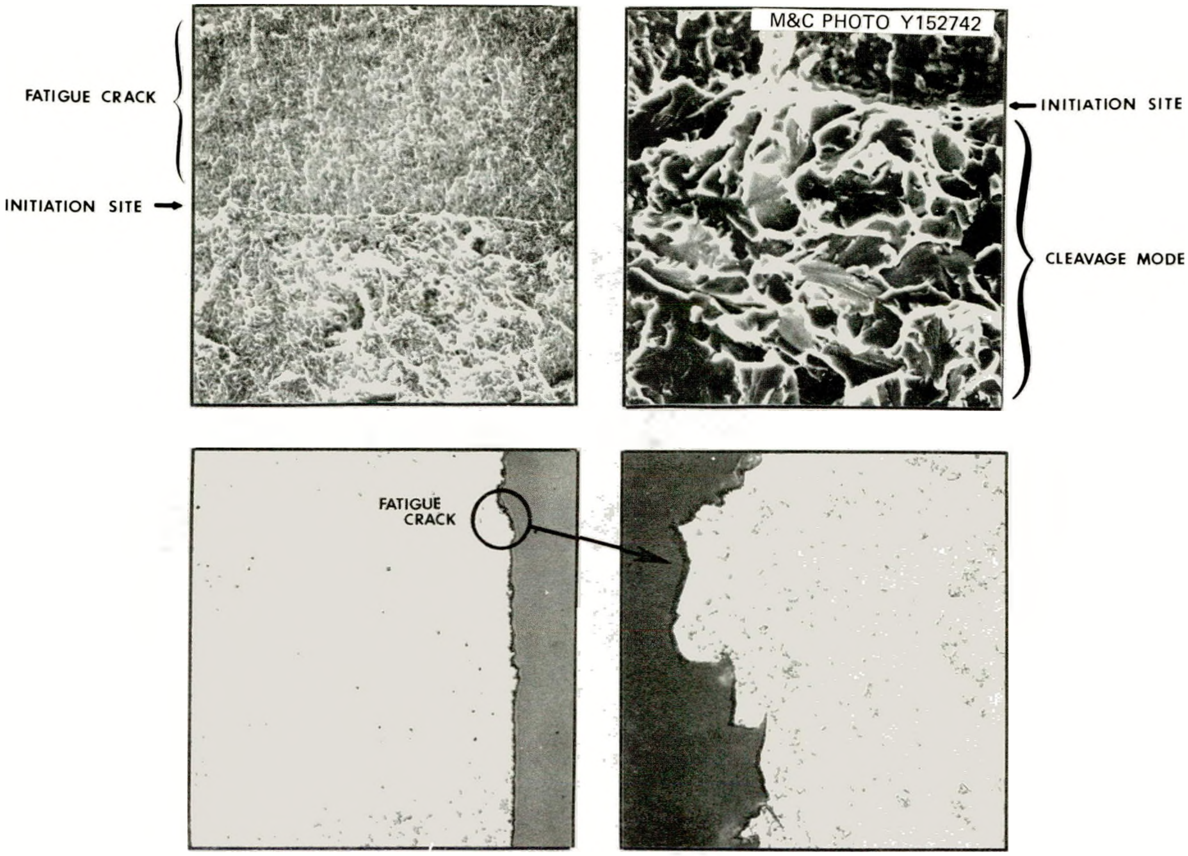


Fig. 5.8. Relationship between the microstructure at the fatigue crack tip in a precracked Charpy V-notch specimen; the amount of dimple fracture at the crack tip and the fracture toughness is illustrated. This specimen, V8W58, exhibited very poor toughness (44 MPa $\sqrt{m}$) at −46°C. The width of the dimple region at the initiation site is very narrow ($1 \times 10^{-3}$ mm) and fracture is by cleavage mode. (Original reduced 31.5%)

## 5.4 Investigation of Fracture Surfaces of ITV-6 Inside Surface Flaw

P. P. Holz

A block was removed from ITV-6 containing flaw C, an inside surface flaw in the seam weld of the vessel cylinder.[6] The block was chilled in liquid nitrogen and broken open along the plane of the crack to expose the fracture surfaces. The crack was examined to determine the amount of stable crack extension developed during the test of the vessel.

Figure 5.9 shows both crack faces; Fig. 5.10 shows one crack face and the profile of the crack at its deepest point obtained by cutting

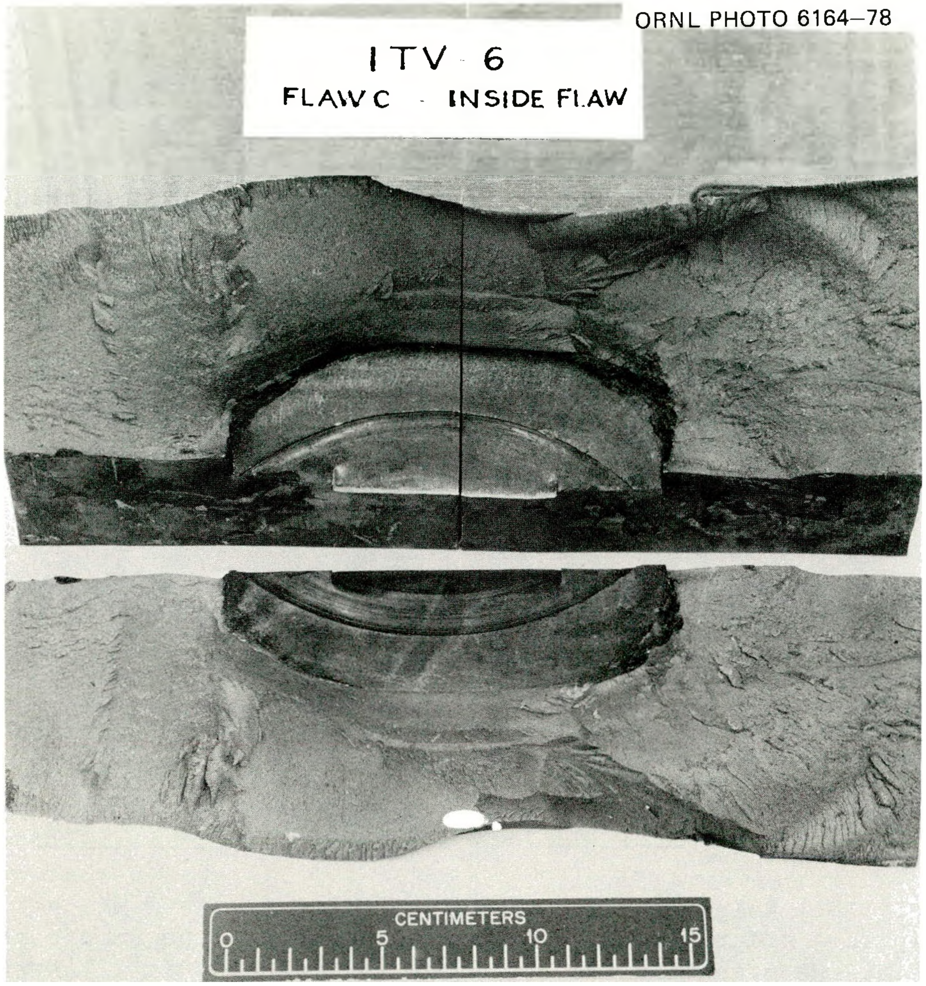


Fig. 5.9. Fracture surfaces of flaw C from intermediate test vessel V-6.

the other crack face piece in half. At the deepest point of the crack, the stable crack growth of the crack beyond the fatigue crack tip measures 6.4 mm. There is about the same amount of stable crack growth at other positions, but the plane of the stable crack growth changes from the fatigue crack plane at the deepest point to perpendicular to the crack plane at the surface. Vessel V-6 was the first intermediate test vessel to exhibit crack arrest. At a test temperature of 88°C the vessel failed at about 220 MPa. Final fracture was a full shear extending approximately 0.5 m from each end of flaw A, a similar flaw on the outside surface of the vessel.

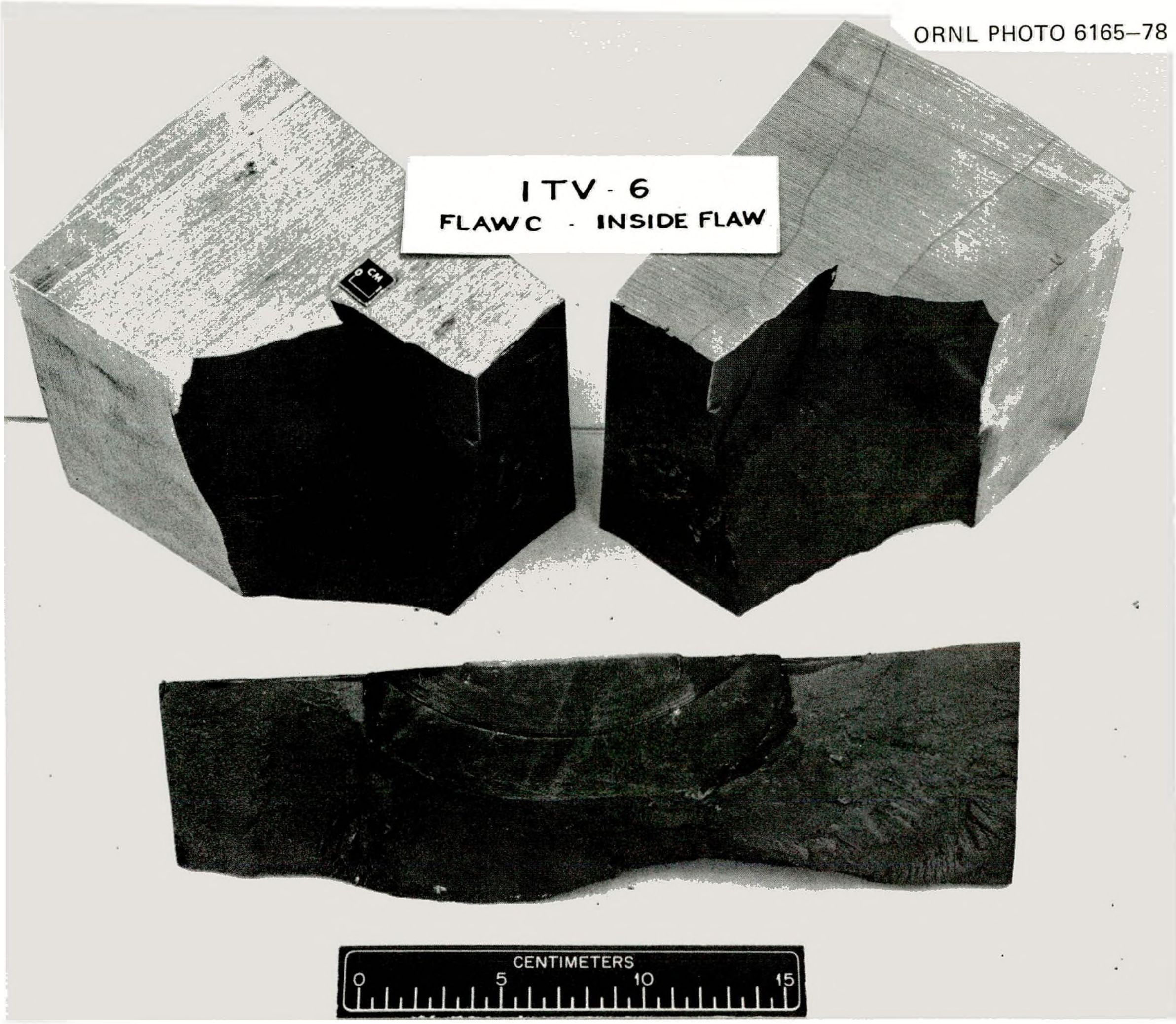


Fig. 5.10. Fracture surfaces and flaw profile of flaw C from intermediate test vessel V-6.

## 5.5 Crack-Arrest Tests

G. C. Smith

During the reporting period, preparations were made for the ORNL participation in the NRC/EPRI crack-arrest cooperative test effort. Methods of installing ladder-type gages on side grooved specimens were tried, and a satisfactory technique was achieved. Two crack-arrest specimens of the MRL type were machined from A533 material and tested at ORNL. At room temperature, $K_{Ia}$ values of 93 and 90 $MN \cdot m^{-3/2}$ were determined. These values fell very near the center of the scatter band reported by Crosley and Ripling.[7] The crack velocities measured during those tests were 560 and 470 m/sec. One problem that was encountered during the tests had to do with the diameter of the split ring used to load the specimen. For the first test, a split ring with the recommended 25.4 mm diameter was used and the ring fractured during the test. While this did not adversely affect the test result, it did require the machining of an additional split ring with an enlarged diameter of 38.1 mm. With the resulting larger net section coupled with an improved lubricant (i.e., molybdenum disulfide), the second ring did not fail during the test.

During the reporting period, the draft report covering the three crack arrest model tests was completed and circulated for comment.

## References

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2. P. P. Holz, "Trial Preparation of V-8 Type Flaw," *Heavy-Section Steel Technology Program Quarterly Prog. Rep. October—December 1977*, ORNL/NUREG/TM-194, pp. 44—48.

3. W. J. Stelzman and D. A. Canonico, "Characterization of the Repair Weld in Vessel V-8," *Heavy-Section Steel Technology Program Quart. Prog. Rep. July—September 1977*, ORNL/NUREG/TM-166, pp. 90—108.

4. W. J. Stelzman, D. A. Canonico, and R. S. Crouse, "Characterization of the Repair Weld in Vessel V-8," *Heavy-Section Steel Technology Program Quart. Prog. Rep. October—December 1977*, ORNL/NUREG/TM-194, pp. 48—62.

5. W. J. Stelzman and D. A. Canonico, "Characterization of the V-7B and V-8 Repair Weldment," *Heavy-Section Steel Technology Program Quart. Prog. Rep. January—March 1977*, ORNL/NUREG/TM-120, pp. 40—44.

6. R. H. Bryan et al., *Test of 6-In.-Thick Pressure Vessels. Series 2: Intermediate Test Vessels V-3, V-4, and V-6*, ORNL-5059, p. 17 (November 1975).

7. P. B. Crosley and E. J. Ripling, "Plane Strain Crack Arrest Characterization of Steels," *Journal of Pressure Vessel Technology*, pp. 291—98 (November 1975).

# 6. THERMAL SHOCK INVESTIGATIONS*

R. D. Cheverton    S. E. Bolt
S. K. Iskander

## 6.1 Introduction

During this reporting period for the Thermal Shock Program, spray procedures for coating the inner surface of TSV-F in the new vertical spray facility were developed; TSV-F was subjected to two liquid nitrogen ($LN_2$) thermal shocks in the $LN_2$ test facility to obtain thermal-hydraulic data; the test facility was redesigned and construction of new components completed; a fracture-mechanics parametric analysis of the PWR double-ended-pipe-break LOCA-ECC was completed; a first draft of the LOCA-ECC thermal shock "significance" report was completed; and preliminary fracture-mechanics calculations pertaining to the PWR steam-like-break accident were performed.

## 6.2 Cryogenic Quenching

ORNL is attempting to develop a capability for imposing thermal shocks on steel cylinders using $LN_2$ as the heat sink. The $LN_2$ quench has certain advantages over other thermal shock techniques considered, not the least of which is lower cost, and is appropriate for proposed future thermal shock experiments.[1]

A complication associated with cryogenic quenching is the formation of a nitrogen vapor blanket (film-boiling regime) that drastically retards heat transfer until the film superheat (surface temperature — $LN_2$ temperature) drops to a few degrees. At this point the nucleate boiling regime is established, and a rapid quench is achieved — but too late. This difficulty is circumvented by applying a thin layer of insulating material to the metal surface. The insulating effect limits the heat flux initially, allowing the nucleate boiling regime to be established immediately.

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*Conversions from SI to English units for all SI quantities are listed on a foldout page at the end of this report.

Under these circumstances a severe thermal shock can be achieved as discussed in Refs. 1 to 5.

The insulating material we are using is an adhesive (3M-brand NF-34) similar to rubber cement. When properly applied, it not only provides the necessary insulation but also provides a high density of nucleation sites that further enhances heat transfer. An appropriate application method is spraying, and several spray techniques have been investigated. A technique suitable for coating the inner surface of the 533-mm-OD thermal-hydraulic test specimen (TSV-F) involves rotating the test cylinder about its longitudinal axis while moving the spray gun in and out of the test cylinder bore. The cylinder is mounted vertically on a motor-driven rotary table, and the gun is attached to a guided mandrel that is raised and lowered with a hoist. The facility is shown in Fig. 6.1.

An acceptable coating must have good adhesion, uniform thickness, the desired thickness (there is an optimum), and a uniform and high density of nucleation sites. Several spray tests were conducted in the vertical spray facility using a 910-mm length of 250-mm pipe for a test specimen. A combination of specimen rotational speed, gun-travel speed and distance, adhesive viscosity, spray gun adjustments, and angle of spray was finally arrived at that resulted in a satisfactory coating. With this done, TSV-F was coated with ∿0.46 mm of adhesive in preparation for an $LN_2$ quench in the $LN_2$-TSTF.

Two thermal shock experiments were conducted in the $LN_2$-TSTF (see Ref. 1 and Fig. 6.2) with TSV-F to determine the degree of symmetry in quenching and the heat transfer coefficients. Previous tests with smaller specimens in a different test facility resulted in satisfactory performance.[4] However, the greater length of TSV-F and the different type of test facility required for the larger specimen introduced some new problems.

The first of the two experiments [$LN_2$-TSE-F(2)] was conducted with an initial test specimen temperature of 21°C. Temperature-vs-time curves for points in the wall about 0.8 mm from the inner surface, presented in Figs. 6.3 and 6.4, show substantial axial gradients. At least part of this undesirable result is attributed to vapor binding in the circulating pump (see the pump curve in Fig. 6.3) and to problems in transferring $LN_2$ from the 1500-ℓ $LN_2$ dewar to the test specimen. Aside from the asymmetry

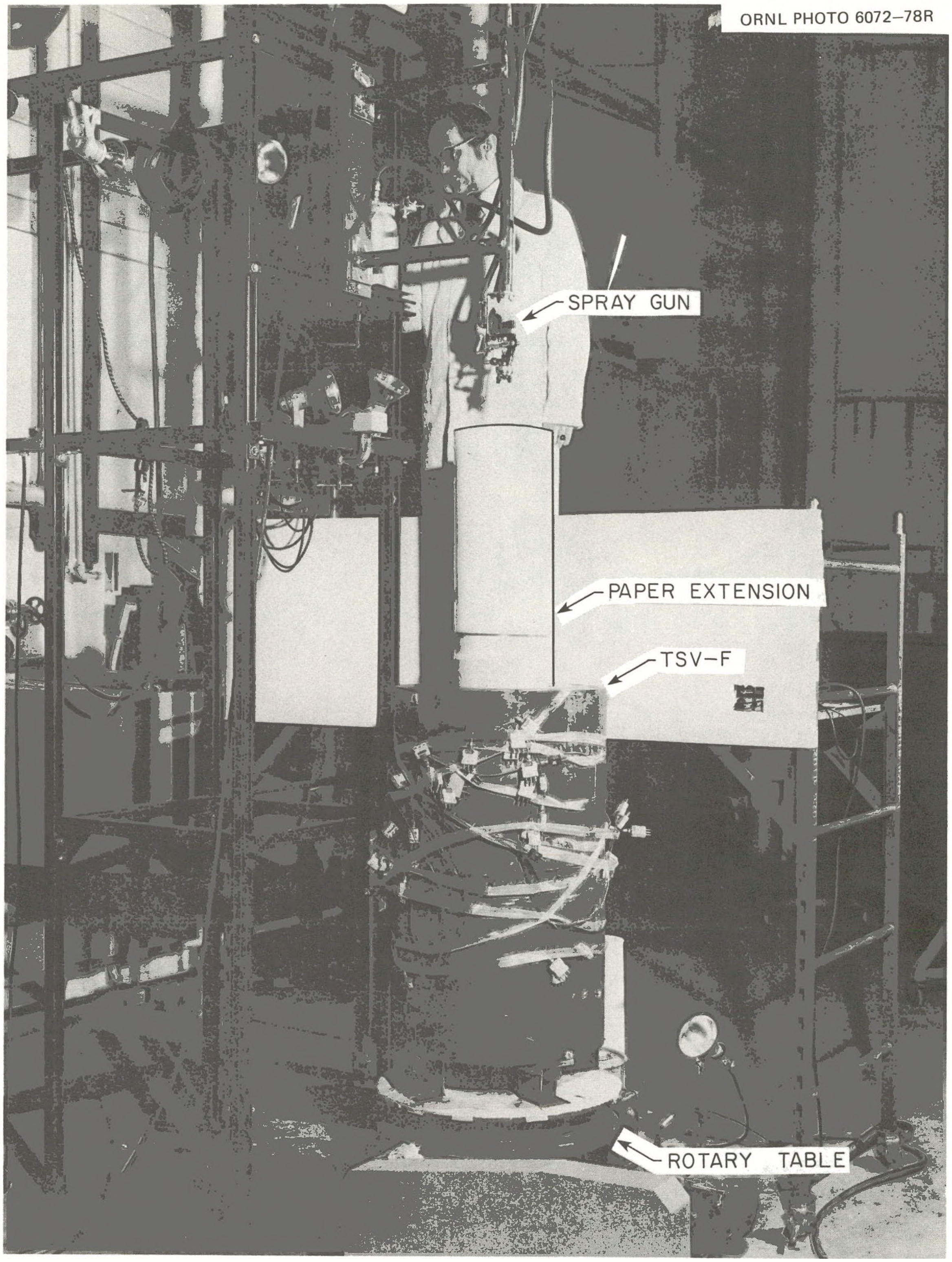


Fig. 6.1. Vertical spray facility.

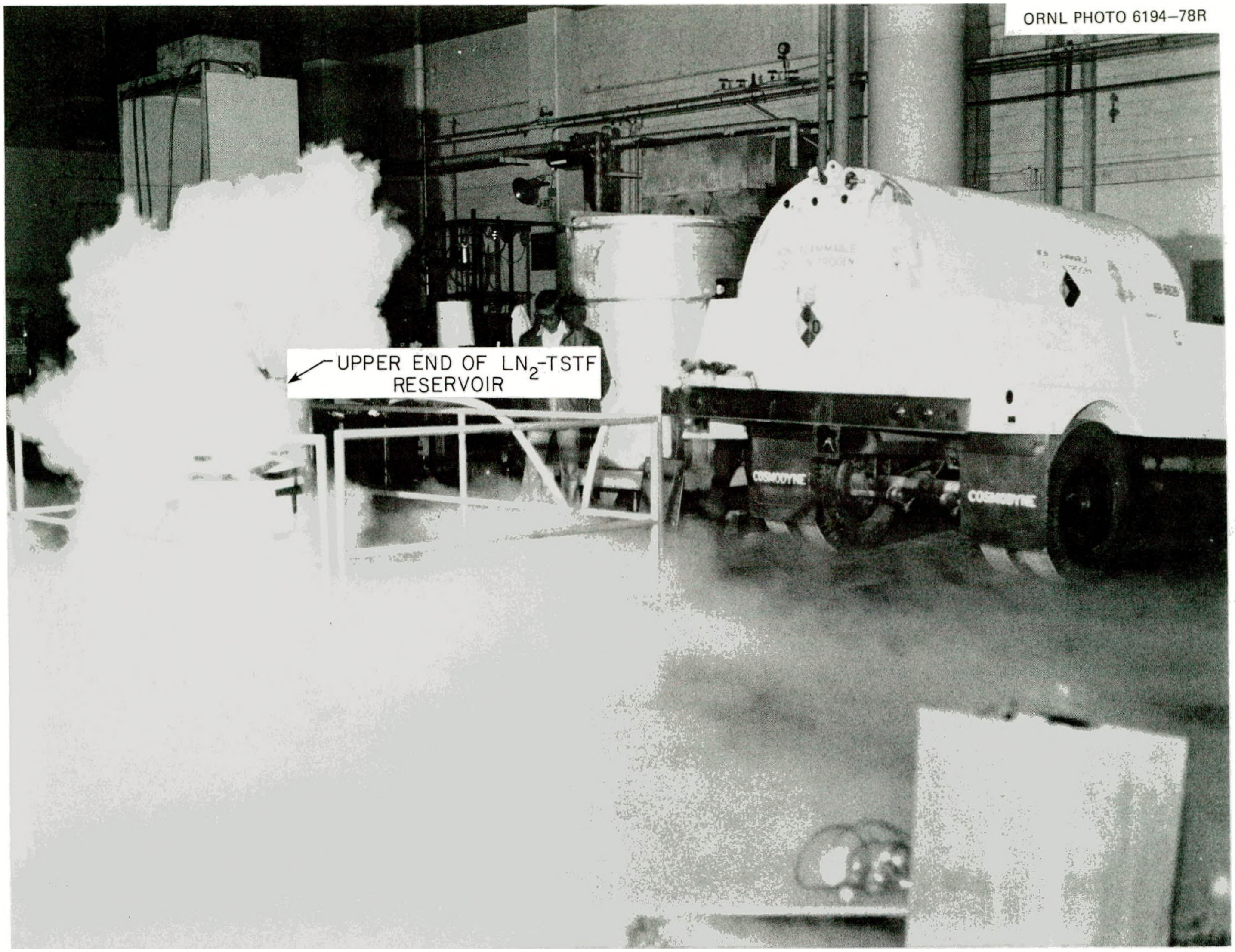


Fig. 6.2. Photograph of $LN_2$-TSTF taken during $LN_2$-TSE-F(2).

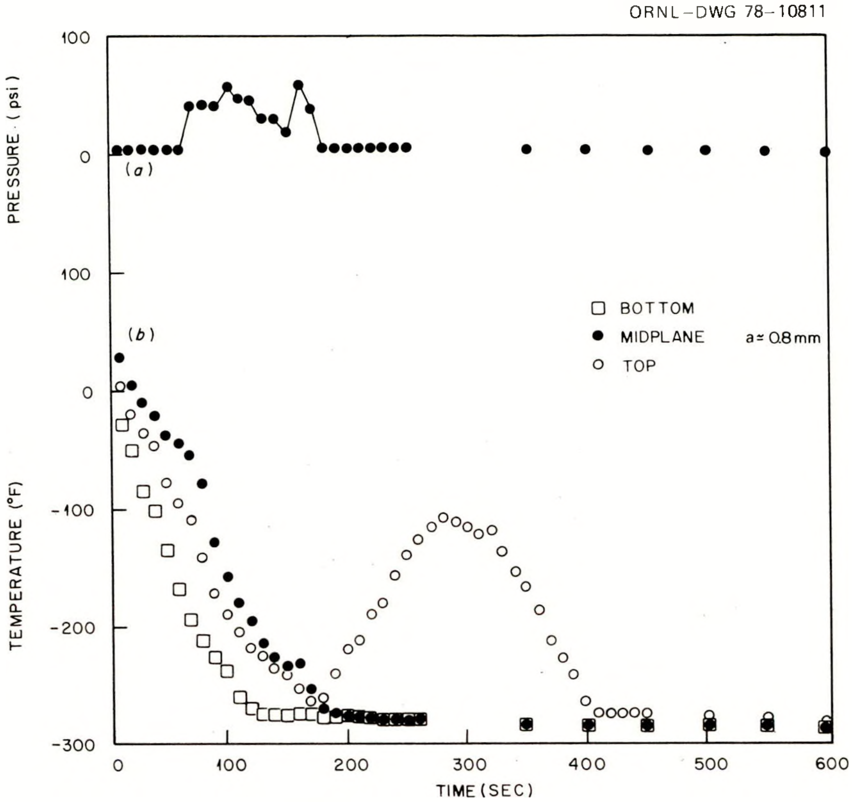


Fig. 6.3. Pump discharge pressure (*a*); temperature vs time for points ∿0.8 mm from inner surface during quench of TSV-F with $LN_2$ initial temperature = 21°C (*b*). Test $LN_2$-TSE-F(2).

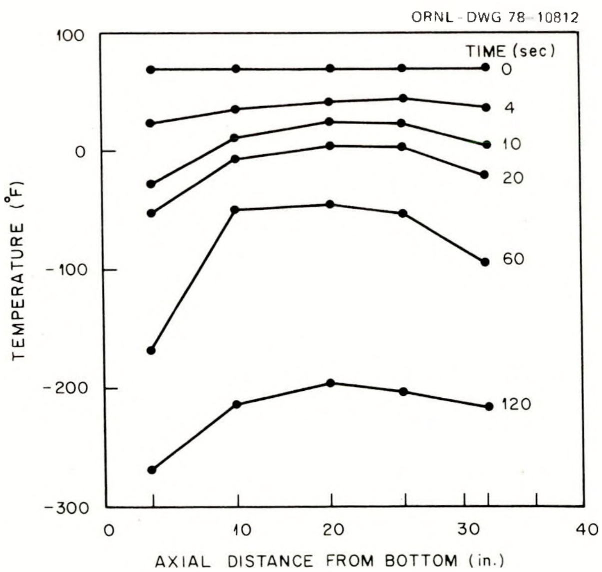


Fig. 6.4. Axial temperature profiles during $LN_2$-TSE-F(2) for points ∿0.8 mm from inner surface.

problem, the results looked good in that the heat transfer coefficient was acceptable. Figure 6.5 shows the coefficient as a function of metal surface temperature and compares the data points with two curves that were derived from numerous experiments conducted with the smaller specimens and that have been found via fracture mechanics calculations to be satisfactory for the proposed future $LN_2$-quench thermal shock experiments. As indicated, the data points are closer to the higher of the two curves.

In preparation for the second test [$LN_2$-TSE-F(3)], modifications were made to the $LN_2$ transfer system, and as a result there was some improvement in performance during the second test. For this test the specimen initial temperature was 93°C, the initial temperature corresponding to a proposed future experiment. The quench was more violent, as would be expected, and an excessive amount of $LN_2$ was expelled from the $LN_2$ reservoir. Once again the result was a large axial gradient in quenching rate. Even so, the heat transfer coefficients during the times of $LN_2$ sufficiency were satisfactory, as shown in Fig. 6.5. Quench curves for this case are shown in Figs. 6.6 and 6.7. As indicated in these figures, initially the upper portion of the specimen "starves" but eventually quenches at a high rate as the $LN_2$ makeup catches up.

These experiments indicated that the desired heat transfer coefficients can be achieved. However, a test facility design change would be required before the question of symmetry could be studied further. Since the observed degree of asymmetry in quenching was unacceptable, a decision was made to modify the facility and conduct an additional test. The new design after modifications is shown in Fig. 6.8. In this modified facility, the initial $LN_2$ injection rate will not be as rapid but the expulsion rate should be much less.

## 6.3 PWR Double-Ended-Pipe-Break LOCA-ECC Parametric Analysis

As a finale to the first phase of the Thermal Shock Program, a "significance" report was written covering the four thermal shock experiments (TSE-1 to -4) and their relation to the analysis of reactor pressure vessels. A section of this report constitutes a brief fracture mechanics

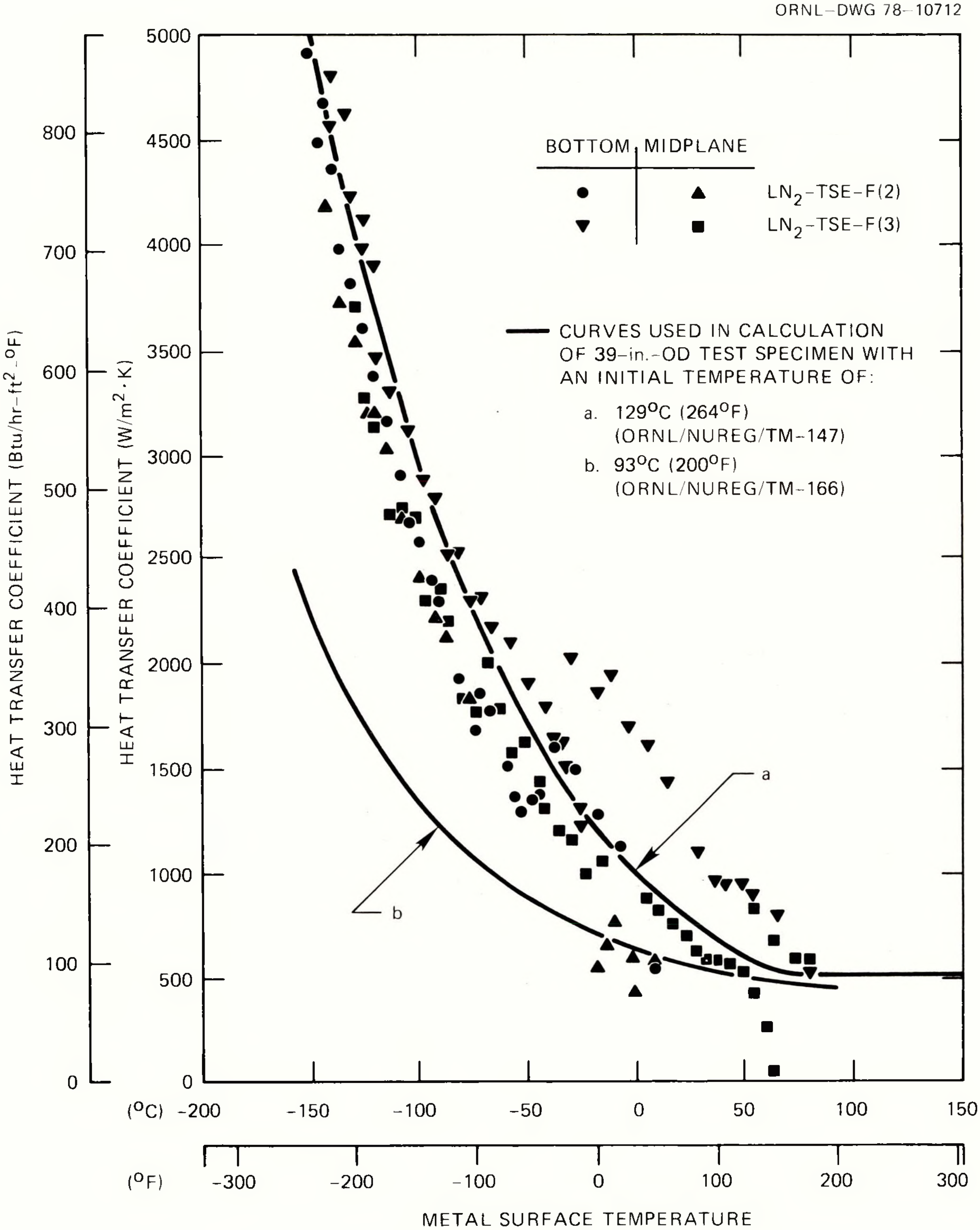


Fig. 6.5. Heat transfer coefficient vs metal surface temperature for $LN_2$ quench of TSV-F from 21°C and 93°C.

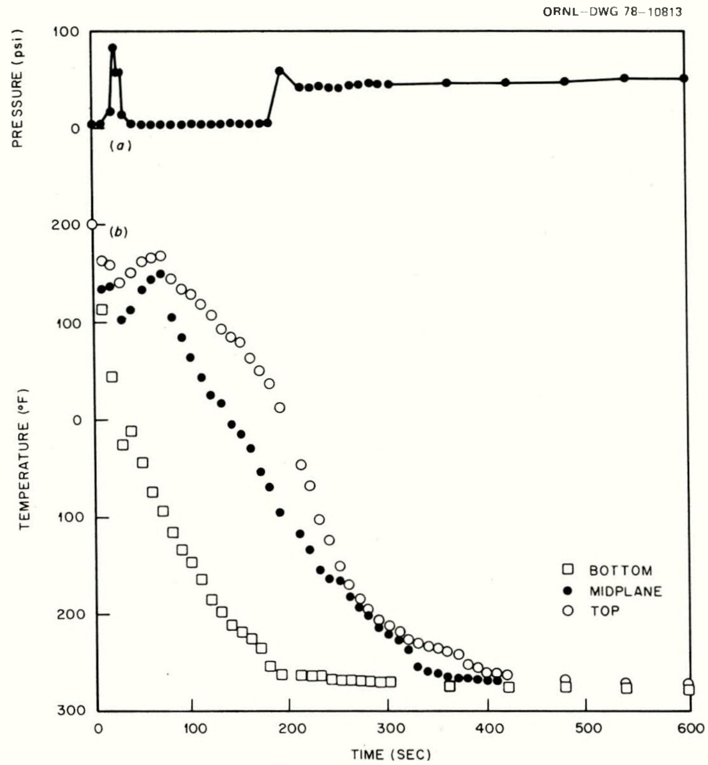


Fig. 6.6. Pump discharge pressure (*a*); temperature vs time for points ~0.8 mm from inner surface during quench of TSV-F with $LN_2$ initial temperature = 93°C (*b*). Test $LN_2$-TSE-F(3).

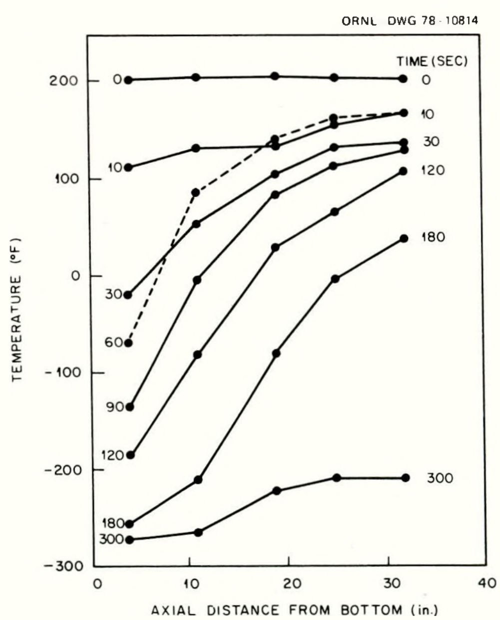


Fig. 6.7. Axial temperature profiles during $LN_2$-TSE-F(3) for points ~0.8 mm from inner surface.

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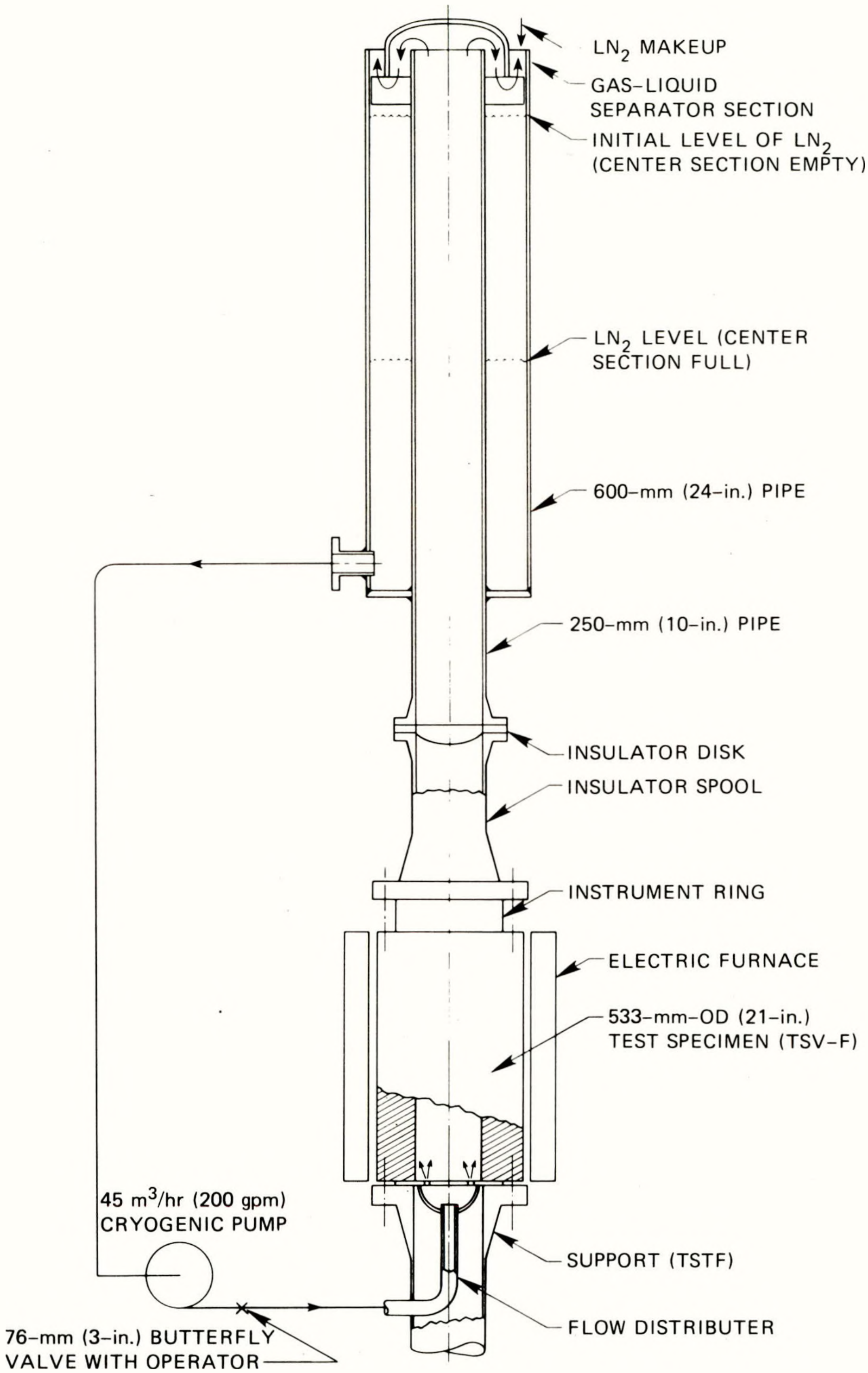


Fig. 6.8. Liquid nitrogen thermal shock test facility [modified design for $LN_2$-TSE-F(4)].

parametric analysis associated with the PWR double-ended pipe break LOCA-ECC. The parametric nature of the analysis consisted of variations in $RT_{NDT}$, fast-neutron fluence, and temperature of the emergency core coolant; the case matrix is shown in Table 6.1. The analysis is based on the PWR reference calculational model described in Ref. 6 except that somewhat different curves for $K_{Ic}$ and $K_{Ia}$ vs $T - RT_{NDT}$ and $\Delta RT_{NDT}$ vs fluence and copper concentrations were used. Reference 7 indicates that the $K_{Ic}$ and $K_{Ia}$ curves given in Section XI of the ASME Code may include an unnecessary 40°F shift, at least from the standpoint of an accident analysis. The 40°F shift was removed for the parametric analysis. The new $\Delta RT_{NDT}$ curve is that specified in Regulatory Guide 1.99, Rev. 1, Sept. 16, 1966. The flaw geometry was limited to long axial and continuous circumferential (two-dimensional models).

Table 6.1. Cases considered in parametric analysis

| $RT_{NDT}$ (°F) | $F_0$ ($10^{19}$ neutrons/cm$^2$) | | |
|---|---|---|---|
| | 2 | 4 | 6 |
| 0 | | X | |
| 40 | X | X[a] | X |
| 80 | | X | |

[a] $T_s$ = 70, 90, 130°F.

The results of the parametric analysis are presented in the form of $K_I$ vs time curves with crack depth as a parameter and curves of fractional critical crack depth ($a_c/w$) vs time for $(K_I/K_{Ic}) = 1$ and for $(K_I/K_{Ia}) = 1$. The first set of curves indicates the minimum crack depth for which warm prestressing is effective, and the second set provides an estimation of the maximum depth of crack penetration with and without the beneficial effects of warm prestressing. Selected data from these curves are included in Table 6.2.

Table 6.2. Summary of selected results of the RCM double-ended-pipe-break LOCA-ECC parametric analysis

| Flaw | Sink temp. (°F) | Surface fluence (neutrons/cm$^2$) | $RT_{NDT}$ (°F) | Copper concentration | WPS[a] threshold (a/w) | Maximum arrest depth: With WPS (a/w) | Maximum arrest depth: Without WPS (a/w) | Maximum initiation depth without WPS (a/w) |
|---|---|---|---|---|---|---|---|---|
| Axial | 70 | $4 \times 10^{19}$ | 0 | Low | | | | NI[b] |
| | | | 0 | High | 0.25 | 0.34 | >0.8 | >0.6 |
| | | | 40 | Low | | | | NI |
| | | | 40 | High | 0.30 | 0.40 | >0.8 | >0.6 |
| | | | 80 | Low | <0.04 | NI-WPS[c] | 0.47 | 0.18 |
| | | | 80 | High | 0.37 | 0.50 | >0.8 | >0.6 |
| Circumferential | | | 0 | Low | | | | NI |
| | | | 0 | High | 0.25 | 0.34 | 0.60 | 0.48 |
| | | | 40 | Low | | | | NI |
| | | | 40 | High | 0.30 | 0.40 | 0.70 | 0.57 |
| | | | 80 | Low | <0.03 | NI-WPS | 0.36 | 0.17 |
| | | | 80 | High | 0.37 | 0.46 | 0.79 | 0.67 |
| Axial | 70 | $2 \times 10^{19}$ | 40 | Low | | | | NI |
| | | $2 \times 10^{19}$ | | High | 0.17 | 0.26 | >0.8 | $\sim$0.6 |
| | | $4 \times 10^{19}$ | | Low | | | | NI |
| | | $4 \times 10^{19}$ | | High | 0.30 | 0.40 | >0.8 | >0.6 |
| | | $6 \times 10^{19}$ | | Low | <0.05 | NI-WPS | 0.35 | 0.12 |
| | | $6 \times 10^{19}$ | | High | 0.40 | 0.52 | >0.8 | >0.6 |
| Axial | 70 | $4 \times 10^{19}$ | 40 | Low | | | | NI |
| | 70 | | | High | 0.30 | 0.40 | >0.8 | >0.6 |
| | 90 | | | Low | | | | NI |
| | 90 | | | High | 0.30 | 0.40 | >0.8 | >0.6 |
| | 130 | | | Low | | | | NI |
| | 130 | | | High | 0.25 | 0.35 | >0.8 | 0.60 |

[a] Warm prestressed.

[b] No initiation since $(K_I/K_{Ic})_{max} < 1$ for all a/w.

[c] No initiation since all cracks with $K_I/K_{Ic} = 1$ are warm prestressed.

In order to establish the sensitivity of the results to variations in the several parameters considered, a base case was selected with the following parameter values: sink temperature = 70°F, surface fluence = $4 \times 10^{19}$ neutrons/cm$^2$ and $RT_{NDT}$ (for zero fluence) = 40°F. As shown in Table 6.2, results for this case indicate that for low-copper vessels the maximum K ratios for the two crack geometries (long axial and continuous circumferential) are less than unity (no crack initiation). For the high-copper vessel the threshold fractional crack depth for warm prestressing is 0.30, and the maximum depth of crack penetration, assuming warm prestressing to be effective, is 0.40. If warm prestressing is not effective, the deepest point of arrest would be 0.70 for the circumferential crack and >0.80 for the long axial crack.

As shown in Table 6.2, considering a range of $RT_{NDT}$ from 0 to 80°F does not change the results substantially. For the low-copper vessels with $RT_{NDT}$ = 80°F, there is a small range of crack depth for which the K ratio is greater than unity. However, if warm prestressing is effective, its threshold is so low that flaws will not initiate. In the absence of warm prestressing, the maximum arrested crack depths for the two cracks are 0.47 and 0.36.

Reducing $RT_{NDT}$ from 40 to 0°F reduces the maximum arrested crack depth from 0.40 to 0.34 for the high-copper warm-prestress case, but without warm prestressing the maximum arrested crack depth is still greater than 0.8.

The sensitivity of the calculated results to fast-neutron fluence was investigated by considering a range of $2 \times 10^{19}$ to $6 \times 10^{19}$ neutrons/cm$^2$ for the case of a long axial flaw. As shown in Table 6.2, this range in fluence had about the same effect as the above range in $RT_{NDT}$ (0 to 80°F). The effect of variations in sink temperature over the range 70 to 130°F was found to be quite small.

It is apparent from this sensitivity analysis that the ranges of parameters considered do not substantially change the situation associated with the base case. That is, if warm prestressing is effective, crack penetration is limited to ∿50%; if warm prestressing is not effective, crack penetration for long axial flaws in high-copper vessels will be in excess of 80%, while that of continuous circumferential cracks will be limited to ∿80%. Low-copper vessels may experience crack propagation in the absence of warm prestressing, but the penetration is limited to ∿50%.

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# 7. FOREIGN RESEARCH

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The objective of this task is to systematically collect, maintain, and review products of foreign research that are applicable to safety of LWR primary systems. The areas covered are fracture mechanics, metallurgy, welding, and structures fabrication. The validity and usefulness of foreign results for application to safety and licensing of LWRs are to be identified to NRC, and recommendations are to be made concerning the application of pertinent well-founded research results.

Lists of foreign reports published in *Nuclear Safety* through Vol. 19, No. 3 (May—June 1978) have been reviewed to identify topics of interest in the metallurgy and materials areas. A total of 23 foreign language research reports have been identified, and requests for translated copies have been submitted to NRC. Seven translated documents have been received, two of which were not on the request lists. These 7 plus 12 English language documents have been reviewed to date. However, the reviews conducted have not revealed new results to be brought to the attention of NRC.

A combined summary of light-water reactor research programs in the Federal Republic of Germany, France, and Japan was prepared from published program descriptions.[1–4] The combination of abbreviated summaries is given in Table 7.1, which provides a basis for comparing the individual programs and indications of cooperative and/or complementary study areas. The table is structured to focus on study categories, components and systems involved, and the specific activities in each country.

Several installations and facilities are used in each country to carry out the individual safety programs. However, in each case, a particular project and associated facility are emphasized or given particular attention. These special projects and facilities are briefly described in Table 7.2.

Table 7.1. Outline of light-water reactor safety programs in the Federal Republic of Germany, France, and Japan

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| Response to power fluctuations | Fuel elements and core structures | France | Studies of power fluctuation-induced failures are underway. These are being done in Strip Program and are planned for the CEA experimental reactor. | Strip Program, Studsvik (Sweden) |
| Degradation of failed cladding and release of fission products | | | Laws governing release and deposition of fission products in primary circuit are being determined. (Bouffon Program) | |
| Fatigue and vibrational behaviors | | | Fatigue endurance tests are being conducted on fuel elements and core structures under conditions of actual flow, temperature and pressure. Responses to flow-induced vibrations are also under examination. | |
| Reactivity initiated accident | | Japan | The following are being considered: transient behavior of fuel before rupture under high rates of power change; deformation and rupture of cladding under various accident conditions (experiments are to be done in pulsed reactors); and forces due to fuel rupture and to dispersed-fuel particle heat-up and their consequences. Computer codes are to be developed for describing the various phenomena. | |
| Power-coolant-mismatch | | | These studies are to be conducted under the Japanese Nuclear Safety Research Reactor (NSQR-2) program to obtain information on fuel failure as a function of changing power density, coolant flow rate, etc. | Complementary to U.S. power-coolant-mismatch experiments in PBF. |
| Loss-of-coolant accident | Fuel elements | Federal Republic of Germany | Failure mechanisms and thresholds are to be determined. Influences of fuel rod deformation on emergency core cooling efficiency and effects of flow blockages are to be established. | |
| | System internals | | Blowdown discharge rates are being determined experimentally; measuring techniques for two-phase flow are being developed, tested and calibrated. Discharge rates are to be examined in HDR experiments. Also to be studied are pressure-relief waves and forces on internal structures, including reactor core, core barrel, and core support structure. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| Loss-of-coolant accident (contd.) | Core | | Core cooling during blowdown and reflooding is under study. Blowdown heat transfer in PWRs is under investigation, and preparations are being made to examine the influences of flow circuits. | Large-scale experiments are to be conducted in close cooperation with U.S. NRC and Japan. |
| | | | Reflood heat transfer experiments are also underway and preparations are being made to investigate influences of flow circuits on refilling and flooding phases. Hot-leg injection and combinations of hot- and cold-leg injections are being studied. | |
| | | | Core meltdown as the result of core cooling failure following a loss-of-coolant accident is being studied. Computer programs are under development to consider melting down of fuel rods, the state and behavior of the molten mass in the reactor vessel, the path of the molten mass through the containment building, fission product release from the molten mass, and the possibility of (as well as the mechanisms for) a containment failure. Small primary cooling system leaks as well as reduced or delayed injection of emergency cooling are also to be addressed. Experimental studies in support of model development are planned. Concrete destruction rates are being investgiated. | |
| | Containment | | Full-pressure containment experiments are being done on a scale model. Analysis methods are under development for heat transfer and compartment pressure computations. Results from HDR experiments will be used for verification. | |
| | | | Pressure-suppression system tests are being done in large facilities, including a decommissioned reactor containment structure. Dynamic load shapes are being determined in a simulated cell sector. Under study are the vibrational behavior of the system, flexibility of the vessel wall, and the random nature of the condensation phenomena. Additional experiments are for examining dynamic behavior of the pressure-suppression system structure and to determine locations and magnitudes of critical loads. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| Loss-of-coolant accident (contd.) | Fuel element | France | Cladding deformation and possible shattering are being examined in out-of-pile tests (Edgar Program) to enable evaluation of these factors during a loss-of-coolant accident. Studies are planned (Phebus Program) on overall fuel element behavior under a simulated loss-of-pressure accident following initiation of emergency injection. | Phebus Program tests will supplement LOFT and PBF tests. |
| | Core | | Two-phase fluid flow tests (Canon, Moby Dick, Super Moby Dick, Marviken CFT, Rebeca) are being conducted to examine phenomena involved at critical flow rates. Blowdown (Omega Loop) and reflood heat transfer (Ersec Loop and Grenoble) are under study. Other areas of study are cladding deformation effects on fluid flow, effects of thermohydraulics on cladding deformation and rupture, and interaction between injection water and steam-water mixture (Epis 1 and 2). | |
| | | | Computational models and programs for accident analysis are under development. Global experiments (Phebus installation) are to be done in connection with this development work. | Tests are to be organized within the framework of PBF, the in-pile large-scale LOFT system test, the Blowdown Ispra out-of-pile test, and ROSA II. |
| | System internals and primary cooling system | | Studies are being made on behaviors of internal structures during slow decompression and on mechanical effects during a pipeline rupture, including reaction force of the jet, hammering of the pipes, and jet impact force. | |
| | Pressure vessel | | Effects of cold shock during emergency injection are being examined. | |
| | Pumps | | Pump behaviors under accident conditions are under investigation. | |
| | Fuel element | Japan | Integral (ROSA) and separate effects tests are being carried out and associated computer programs developed. The blowdown and reflood processes are being examined as well as heat transfer and fuel and fission product behaviors. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study description | Interfacing projects |
|---|---|---|---|---|
| Loss-of-ccolant accident (contd.) | Containment | | Containment response studies include effects of heat capacity of containment wall, temperature distribution in the contained atmosphere, and flow resistance for multicompartment containers. | |
| Normal and accident conditions | Primary circuit | Federal Republic of Germany | Failure types and consequences for pressurized components are under examination. Fracture formation, fracture velocity, discharge phenomena, acceleration of fragments, and pressure distributions and histories are to be studied. Dynamic responses of structures surrounding failed components, including concrete, other primary circuit components, and the containment are to be addressed. | |
| | | France | Reactor Pressure Vessel<br>Analytical studies are addressed to verification of builder design computations and to testing validities of simplified formulas for linear elastic fracture analyses. Vessel steel and weldments are characterized for strength, fatigue behavior, and behavior under irradiation. Also under study are cracking tendencies under environmental conditions and annealing temperatures for removal of irradiation effects. Low cycle fatigue tests are being conducted on vessel material, and tests on structures are planned in connection with analysis method development. | |
| | | | Steam Generators<br>A major program is underway on metallurgical, hydraulic, and vibration studies. Other (unspecified) studies are also to be undertaken. | |
| | | Japan | Irradiation effects and fracture behaviors of materials are main concerns. Corrosion and stress-corrosion-cracking studies are planned. Other areas of interest are quality assurance, fatigue testing, and stress and dynamic response analyses. | |
| | | | Crack propagation in base metal and weld joints of pressure vessels is to be investigated. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| Normal and accident conditions (contd.) | Primary circuit (contd.) | Japan (contd.) | Pipe rupture accidents and effects on surrounding structural components are being investigated. A computer program is being developed to treat flawed structures. | |
| | Containment | France | Studies were done on entrapment of iodine in concrete, the propagation of a mixture of water, steam and air in the casement of the containment (Rebeca Program), and the thermal exchange of an air-steam mixture on a wall under transient conditions (Ecorta Program). | |
| Quality assurance and component safety | Steam supply system | Federal Republic of Germany | Quality Assurance<br>Ultrasonic methods are being investigated for use in automatic inspection of the entire reactor pressure vessel in the installed condition. Expansion to the entire primary circuit is to be addressed. An eddy current system was developed for surface and thin-part examinations. Acoustic emission techniques are under study for crack detection and interpretation in continuously monitored components. The nondestructive methods are to be critically examined in the HDR program. | |
| | | | Component Safety<br>Influences of deviations from material quality (cracks, pores, segregation, and internal stresses) which result from manufacturing and processing are under investigation. Also being investigated are influences of irradiation, aging, and cyclic embrittlement. Analytical tools for prediction of crack growth, failure thresholds, and failure modes are being improved. | |
| | | France | Quality Assurance<br>Inspection methods [ultrasound, acoustic emission, Foucault (eddy) current] are being certified and improved. Acoustic emission is targeted for continuous monitoring; steam generator pipe inspection is by eddy current. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| Quality assurance and component safety (contd.) | Steam supply system (contd.) | Japan | Quality Assurance<br>Nondestructive test method work is concentrated on quantitative determinations of sizes, numbers, and distributions of flaws in weld and base metal that are compatible with fracture analysis capabilities. | |
| External impacts | Reactor system installation | Federal Republic of Germany | Earthquake<br>Investigations are to be done to verify correctness of computer code developments and material representations. Essential parts of these investigations are included in the HDR program. | |
| | | | Chemical Explosions<br>Effects of pressure waves from explosions (such as from detonation of released gases) are being studied. Possible pressure wave fields and their respective load spectra for various building configurations are to be catalogued. | |
| | | | Airplane Crash<br>Instrumented concrete units are to be impacted by instrumented large projectiles. From such tests, a model is to be developed to describe the supporting capacities of reinforced-concrete load-bearing slabs under shock-type loads. | |
| | | France | Earthquake<br>Studies address definitions of response spectra, nonlinear behaviors of structures, interactions between ground and foundations, structural responses to earthquakes, and development of a seismic-tectonic map of France. | |
| | | | Chemical Explosions<br>Consequences of accidental ignition of a cloud of inflamable gas are being investigated. Pressure waves from gas explosions are being modeled, and the effects of these waves on power plants and the layout of buildings is being studied. Both detonation and slow combustion are to be considered in the program. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| External impacts (contd.) | Reactor system installation (contd.) | France (contd.) | Projectiles (Including Aircraft) Effects of missile impacts on reinforced concrete are being studied and equations for design developed. | |
| | | Japan | Earthquakes Ground motions and their characteristics are under study. Also being studied are nonlinear behaviors of systems and components and interactions between buildings and foundations. Experiments are to be done for critical components and structures, and seismic sensors for use in nuclear environments are to be developed. | |
| Radiological aspects | | Federal Republic of Germany | Radiation exposure under normal operating conditions, including routine inspection and maintenance work, as well as during repair work and decommissioning is being analyzed for minimization. Investigations are being done on distribution of fission and activation products in the primary circuit and containment, atmospheric dispersion, retainment systems for fission products, and development of decommissioning concepts. | |
| | | France | Being studied are transfers of active products and levels of contamination in the power plant and its enviorns during normal operation and in cases of accidents. Calculational models have been developed. | |
| | | Japan | Technology is being developed to control the level of reactivity to levels proposed by the U.S.-NRC. Normal and accident conditions for reactor systems are included, and annual inspection considerations are being addressed. The investigations include corrosion product generation and behavior, and control and removal of radioactive corrosion products. Fuel reprocessing plant effluent control and liquid radwaste control from both reactor and fuel reprocessing plants are under study. | |

Table 7.1 (continued)

| Study category | Component or system | Country | Study details | Interfacing projects |
|---|---|---|---|---|
| Risk and reliability | Nuclear reactor installation | Federal Republic of Germany | Computer codes are being developed for determination of failure probabilities for complex systems. Information on failure rates and reliability data are to be collected by monitoring nuclear power plants. | |
| | | France | Rupture probabilities for reactor vessels are being studied using statistical and probabilistic approaches. Experimental studies are done in support of theoretical work. | |
| | | | The probabilistic approach is also being developed for safety analyses of nuclear installations. Reliability data are being collected on equipment and components. Availabilities of systems as functions of time are being calculated and improvements made in computational methods. Potential failure initiating events and accident consequences are under study. | |
| | | Japan | Probabilistic safety assessment methods are under development. A reliability research center is to be established to collect data on abnormal operations and failure. Reliability assessment methods are also being developed. | |

Table 7.2. Special projects and facilities

| Project | Scope | Facility | Remarks |
|---|---|---|---|
| HDR (Heissdampfreaktor-Sicherheitsprogramm, Federal Republic of Germany) | Experiments are to be conducted on: (1) the behavior of a nuclear reactor installation during simulated earthquake type loads, (2) the load on the primary circuit and containment during loss-of-coolant accidents, (3) the reliability of available nondestructive examination techniques for primary circuit components, (4) the safety margins of the reactor pressure vessel in the case of previous weakening. | Decommissioned 100-MW(t) HDR Experimental Reactor. Original large components of the installation will be used. | |
| Phebus (France) | In-pile studies are to be conducted to investigate behaviors of fuel elements throughout core blowdown and reflood phases of a LOCA. Specific objectives are: (1) validate computer programs describing blowdown and reflooding, (2) study of emergency injection from a modeling standpoint, (3) study of physico-chemical and mechanical behavior of fuel pins and assemblies during an accident, (4) study of behavior of fuel pins when the limits set by core safety criteria are reached. | Facility located at Cadarache Nuclear Research Center. Nuclear power is supplied by a driver core of the pool type, which develops a power of ∿60 MW(t) during the 20-minute test period. A test loop passes through the core and maintains the test assembly at the operating conditions of a pressurized water reactor. External circuits make it possible to perform depressurization and emergency injections as in the case of a PWR LOCA. | Major projects most closely comparable to Phebus are LOFT, Semiscale MOD 1, PBF, BLOWDOWN (Ispra), and ROSA II. |
| ROSA (Rig of Safety Assessment, JAERI, Japan) | Examinations of LOCA consequences, including thermohydraulic behavior during blowdown, reflooding and heatup, are to be carried out. The studies are divided into three phases: ROSA-I, -II, and -III. The second is directed to PWRs while the third is directed to BWRs. Safety assessments are to be made and computer programs developed. | ROSA-I (  to 1973) Pressure vessel fitted with rupture discs for studying thermodydraulics due to loss of coolant. A pressurizer, preheater, feed-water piping and coolant-discharge piping are attached to the vessel. There is no emergency-core cooling system. | Interfaces with LOFT Project. |

Table 7.2 (continued)

| Project | Scope | Facility | Remarks |
| --- | --- | --- | --- |
| ROSA (contd.) | | ROSA-II (For PWR) (1973-1976)<br>This is a modified ROSA-I facility for out-of-pile experiments. The electrical power is 224 MW; there are two coolant loops, designed to study the thermohydraulics of emergency-core cooling with system effects. Each loop has a steam generator and a coolant circulation pump. One loop has valves and equipment for simulating single-ended and double-ended breaks. Simulated vessel internals are used; these include the core, core barrel, flow mixer plate, and core plates.<br>ROSA-III (For BWR) (1977-1979)<br>The ROSA facility will be modified to include a spray system for emergency core cooling. | |

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Conversion factors[a]

| SI unit | English unit | Factor |
|---|---|---|
| mm | in. | 0.0393701 |
| cm | in. | 0.393701 |
| m | ft | 3.28084 |
| m/s | ft/s | 3.28084 |
| kPa | psi | 0.145038 |
| MPa | ksi | 0.145038 |
| $MN \cdot m^{-3/2}$ ($MPa \cdot \sqrt{m}$) | $ksi\sqrt{in.}$ | 0.910048 |
| J | ft-lb | 0.737562 |
| K | °F or °R | 1.8 |
| $kJ/m^2$ | $in.-lb/in.^2$ | 5.71015 |
| $W \cdot m^{-2} \cdot K^{-1}$ | $Btu/hr\text{-}ft^2\text{-}°F$ | 0.176110 |
| T(°F) = 1.8 T(°C) + 32 | | |

[a]Multiply SI quantity by given factor to obtain English quantity.

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