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Heavy-Section Steel Technology Program

Semiannual Progress Report for April - September 1988

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Prepared for
U.S. Nuclear Regulatory Commission

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CONTENTS

LIST OF FIGURES

LIST OF TABLES

PREFACE

The Heavy-Section Steel Technology (HSST) Program, which is sponsored by the Nuclear Regulatory Commission, 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 in April—September 1988. The work performed by the Oak Ridge National Laboratory (ORNL) and by subcontractors is managed by the Engineering Technology Division (ETD) of ORNL. Major tasks at ORNL are carried out by the ETD 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, ORNL/NUREG/TM-194, ORNL/NUREG/TM-209, ORNL/NUREG/TM-239, NUREG/CR-0476 (ORNL/NUREG/TM-275), NUREG/CR-0656 (ORNL/NUREG/TM-298), NUREG/CR-0818 (ORNL/NUREG/TM-324), NUREG/CR-0980 (ORNL/NUREG/TM-347), NUREG/CR-1197 (ORNL/NUREG/TM-370), NUREG/CR-1305 (ORNL/NUREG/TM-380), NUREG/CR-1477 (ORNL/NUREG/TM-393), NUREG/CR-1627 (ORNL/NUREG/TM-401), NUREG/CR-1806 (ORNL/NUREG/TM-419), NUREG/CR-1941 (ORNL/NUREG/TM-437), NUREG/CR-2141, Vol. 1 (ORNL/TM-7822), NUREG/CR-2141, Vol. 2 (ORNL/TM-7955), NUREG/CR-2141, Vol. 3 (ORNL/TM-8145), NUREG/CR-2141, Vol. 4 (ORNL/TM-8252), NUREG/CR-2751, Vol. 1 (ORNL/TM-8369/V1), NUREG/CR-2751, Vol. 2 (ORNL/TM-8369/V2), NUREG/CR-2751, Vol. 3 (ORNL/TM-8369/V3), NUREG/CR-2751, Vol. 4 (ORNL/TM-8369/V4), NUREG/CR-3334, Vol. 1 (ORNL/TM-8787/V1), NUREG/CR-3334, Vol. 2 (ORNL/TM-8787/V2), NUREG/CR-3334, Vol. 3 (ORNL/TM-8787/V3), NUREG/CR-3744, Vol. 1 (ORNL/TM-9154/V1), NUREG/CR-3744, Vol. 2 (ORNL/TM-9154/V2), NUREG/CR-4219, Vol. 1 (ORNL/TM-9593/V1), NUREG/CR-4219, Vol. 2 (ORNL/TM-9593/V2), NUREG/CR-4219, Vol. 3, No. 1 (ORNL/TM-9593/V3&N1), NUREG/CR-4219, Vol. 3, No. 2 (ORNL/TM-9593/V3&N2), NUREG/CR-4219, Vol. 4, No. 1 (ORNL/TM-9593/V4&N1), NUREG/CR-4219, Vol. 4, No. 2 (ORNL/TM-9593/V4&N2), and NUREG/CR-4219, Vol. 5, No. 1 (ORNL/TM-9593/V5&N1).

SUMMARY

1. PROGRAM MANAGEMENT

The Heavy-Section Steel Technology (HSST) Program is arranged into 12 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) crack-arrest technology, (6) irradiation effects studies, (7) stainless steel cladding evaluations, (8) intermediate vessel tests and analyses, (9) thermal-shock technology, (10) pressurized-thermal-shock (PTS) technology, (11) Pressure-Vessel-Research Users' Facility (PVRUF), and (12) shipping cask material evaluations. Progress reports are issued on a semiannual basis, and the report chapters correspond to the tasks.

The work is performed by the Oak Ridge National Laboratory (ORNL) and through a number of research and development (R&D) subcontracts. During the report period, 35 program briefings, reviews, or presentations were made; 19 technical documents were published.

2. FRACTURE METHODOLOGY AND ANALYSIS

During this report period, advances were made in the coordinated effort being conducted under the HSST Program by ORNL and several subcontracting groups to develop the crack-arrest data base and the analytical tools required to construct improved fracture models for reactor pressure vessel (RPV) steel. Additional high-strain rate testing of A 533 grade B class 1 steel was performed for viscoplastic material characterization; rapid-loading fracture testing of small specimens was carried out for dynamic fracture toughness measurements. Analytical efforts have focused on fracture model developments for improved viscoplastic constitutive formulations, moving singular-element techniques for modeling crack propagation, alternative parameters for characterizing inelastic dynamic fracture, constraint effects on crack-tip yielding, and tunneling effects on arrest toughness calculations.

3. MATERIAL CHARACTERIZATION AND PROPERTIES

Posttest material characterization for Wide-Plate Series 2 was performed and involved testing of drop-weight, Charpy impact, tensile, and fracture toughness specimens machined from broken halves of wide-plate specimens WP-2.1 and -2.5. Only minor differences were noted between the posttest results and the pretest characterization.

Scanning electron microscope fractography was performed on wide-plate specimen WP-2.5 and revealed that significant ductile tearing preceded the first cleavage crack-initiation event; furthermore, some ductile tearing was also associated with all subsequent seven events. ORNL participated in the NRC-sponsored cooperative program on the use of dc-potential drop J-R curve testing and submitted results to the coordinating laboratory, David Taylor Research Center.

Finally, preliminary investigations of material potentially suitable for PTSE-4, on low-upper-shelf (LUS) welds, involved fabrication and heat treatments of welds using 2 1/4 Cr-1 Mo weld wire and Linde 80 flux. Charpy impact results revealed transition temperatures from 34 to 60°C and upper-shelf energies from 42 to 48 J, while tensile yield strengths varied from 689 to 726 MPa.

4. SPECIAL TECHNICAL ASSISTANCE

In recognition of the completion of the environmental crack-growth studies within the HSST Program in conjunction with the numerous relatively short-term topics of interest to the Nuclear Regulatory Commission (NRC) that arise on a continuing basis, Task 4 has been redefined to provide coverage of these topics. During this report period, the initial evaluation of the impact of radiation embrittlement on the integrity of light-water reactor (LWR)-vessel supports was completed. The evaluation included specific-plant analyses for Trojan and Turkey Point Unit 3 and indicated that for both plants minimum critical flaw sizes in the supports at 32 EFPY were small enough to be of concern.

In addition, an overall assessment of the special concerns arising from the existence of LUS welds in RPVs was begun with special emphasis on the reevaluation of the applicability of various formulations of the J-integral in assessing large amounts of crack extension.

5. CRACK-ARREST TECHNOLOGY

Sixteen wide-plate, crack-arrest tests have now been completed (three during this report period), which concludes current plans for wide-plate testing. When combined with other large-specimen test results, the wide-plate, crack-arrest toughness values form a consistent trend, showing that arrest can and does occur at temperatures up to and above that which corresponds to the onset of upper-shelf behavior. Also, the calculated K_{Ia} values extend above the limit in Sect. XI of the *ASME Code*.

Preparations continued for testing the stub panel, intermediate-size, crack-arrest specimens that are anticipated as the successors to the wide-plate specimen. Checkout of the loading and heating-cooling-insulation systems was completed as well as development of a draft performance specification for a high-speed data acquisition system.

6. IRRADIATION EFFECTS STUDIES

All testing for the Fifth Series has been completed, results were compiled, and a draft report was completed and submitted to the NRC. Preliminary analyses indicate that the lower-bound K_{Ic} (at 125 MPa·$\sqrt{m}$) curve shift for the higher copper weld is significantly greater than the

Charpy 41-J shift and the postirradiation K_{Ic} curve is of shallower slope than that for unirradiated material. For the Sixth Series, the remote crack-arrest testing device for use in testing irradiated specimens has been thoroughly evaluated, modified, and successfully used to perform unirradiated testing of compact crack-arrest specimens. For the Seventh Series on stainless steel cladding, irradiated 12.5-mm-thick compact specimens (0.5TCS) were tested from −75 to 288°C and revealed consistent decreases in the ductile fracture toughness J_{Ic} in qualitative agreement with observed decreases in Charpy impact energy and lateral expansion.

7. CLADDING EVALUATIONS

Results from clad plate tests have shown that (1) a tough surface layer composed of cladding and/or heat-affected zone (HAZ) has arrested running flaws under conditions where unclad plates have ruptured, and (2) the residual load-bearing capacity of clad plates with large subclad flaws significantly exceeded that of an unclad plate. Regarding flaw characterization studies of clad LWR vessel segments, a topical report and journal paper were completed describing successful examinations on segments from both a boiling- and pressurized-water reactor.

8. INTERMEDIATE VESSEL TESTS AND ANALYSIS

There was no activity in the intermediate vessel tests and analysis task for this period.

9. THERMAL-SHOCK TECHNOLOGY

There was no activity in the thermal-shock technology task for this period.

10. PRESSURIZED-THERMAL-SHOCK TECHNOLOGY

An investigation of the potential benefits of additional PTS experiments was completed during the report period. Recommendations were made to the NRC to perform two additional experiments, PTSE-3 to confirm analytical predictions of effects of cladding and PTSE-4 to clarify mechanisms of tearing of LUS welds.

11. FLAW DENSITY STUDIES FOR PRESSURE-VESSEL-RESEARCH USERS' FACILITY

ORNL undertook an initiative in concert with the NRC and Department of Energy (DOE) to establish a Pressure-Vessel-Research Users' Facility

(PVRUF). The facility is to be centered around a complete PWR pressure vessel and is to provide unique R&D opportunities for a number of organizations.

The overall research plan and conceptual design of the PVRUF to house the vessel will begin when appropriate funding is arranged. Initial R&D activities, however, will proceed with the vessel in its temporary location at the K-25 Plant in Oak Ridge. One early HSST-funded task is to characterize the density, size, location, and orientation of flaws in this vessel for use in probabilistic integrity assessment methods.

During this report period, the modifications of the PVRUF to allow for nondestructive examination of the vessel in its current location and horizontal position was completed, and an overall mission survey was conducted to aid in establishing other research priorities.

12. SHIPPING CASK MATERIAL EVALUATIONS

There was no reportable activity in the shipping cask material evaluations task for this period.

HEAVY-SECTION STEEL TECHNOLOGY PROGRAM SEMIANNUAL PROGRESS REPORT FOR APRIL—SEPTEMBER 1988*

W. R. Corwin

ABSTRACT

The Heavy-Section Steel Technology (HSST) Program is conducted for the Nuclear Regulatory Commission (NRC). The studies relate to all areas of the technology of materials fabricated into thick-section, primary-coolant containment systems of light-water-cooled nuclear power reactors. The focus is on the behavior and structural integrity of steel pressure vessels containing cracklike flaws. The program is organized into 12 tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) crack-arrest technology, (6) irradiation effects studies, (7) cladding evaluations, (8) intermediate vessel tests and analysis, (9) thermal-shock technology, (10) pressurized-thermal-shock (PTS) technology, (11) Pressure-Vessel-Research Users' Facility (PVRUF), and (12) shipping cask material evaluations. During this period, advances were made in the coordinated effort to develop the dynamic materials fracture data and the analytical tools required to construct improved fracture models for reactor pressure vessel (RPV) steels. Analytical efforts included examination of alternative parameters governing dynamic fracture, their corresponding constitutive models and computational implementation, as well as constraint and tunneling effects on crack-arrest calculations. Two areas of NRC topical support were continued: (1) the evaluation of enhanced low-temperature, low-flux irradiation embrittlement on the integrity of RPV supports; and (2) an overall assessment of low-upper-shelf (LUS) welds in RPVs with special emphasis on the reevaluation of the J-integral in assessing large amounts of crack extension. Three additional wide-plate, crack-arrest tests were performed by National Bureau of Standards (NBS), bringing to 16 the total of such tests and concluding the test series. Crack-arrest and other fracture characterization data were obtained for clad-plate and wide-plate series 2 test materials. A draft report was completed describing the Fifth HSST Irradiation Series for the study of K_{Ic} shifts for welds with different copper contents. All irradiated fracture toughness testing was completed for the

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*This report is written in terms of metric units. Conversions from SI to English units for all SI quantities are listed on a foldout page at the end of this report.

second phase of the Seventh HSST Irradiation Series on cladding. A report was drafted describing the nondestructive examinations on segments of clad light-water reactor vessels. A report describing the second series of the clad-plate fracture tests of reactor vessel steels was drafted. Initial studies on the benefits of additional pressurized-thermal-shock experiments were completed and recommendations made to conduct PTSE-3 to confirm analytical predictions of the effects of cladding and PTSE-4 to clarify mechanisms of tearing in LUS welds. The modification of the temporary facility for nondestructive flaw assessment of the PWR pressure vessel in the ORNL PVRUF was completed and additional planning on its overall research thrusts was performed.

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1. PROGRAM MANAGEMENT

W. R. Corwin

The Heavy-Section Steel Technology (HSST) Program, a major safety program sponsored by the Nuclear Regulatory Commission (NRC) at Oak Ridge National Laboratory (ORNL), is concerned with the structural integrity of the primary systems [particularly, the reactor pressure vessels (RPVs)] of light-water-cooled nuclear power reactors. The structural integrity of these vessels is ensured by (1) designing and fabricating RPVs according to standards set by the code for nuclear pressure vessels, (2) detecting flaws of significant size that occur during fabrication and while in service, and (3) developing methods of producing quantitative estimates of conditions under which fracture could occur. The program is concerned mainly with developing pertinent fracture technology, including knowledge of (1) the material used in these thick-walled vessels, (2) the flaw-growth rate, and (3) the combination of flaw size and load that would cause fracture and, thus, limit the life and/or operating conditions of this type of reactor plant.

The program is coordinated with other government agencies and with 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 activities are conducted under subcontract by research facilities in the United States and through an informal cooperative effort on an international basis. Seven research and development (R&D) subcontractors are currently in force.

The program tasks are arranged according to the work breakdown structure shown in Fig. 1.1. Accordingly, the chapters of this progress report correspond to these 12 tasks.

In accordance with guidance from the NRC, reporting emphasis has been changed during this reporting period. Beginning with this edition, the semiannual progress report will provide more of an overview of the

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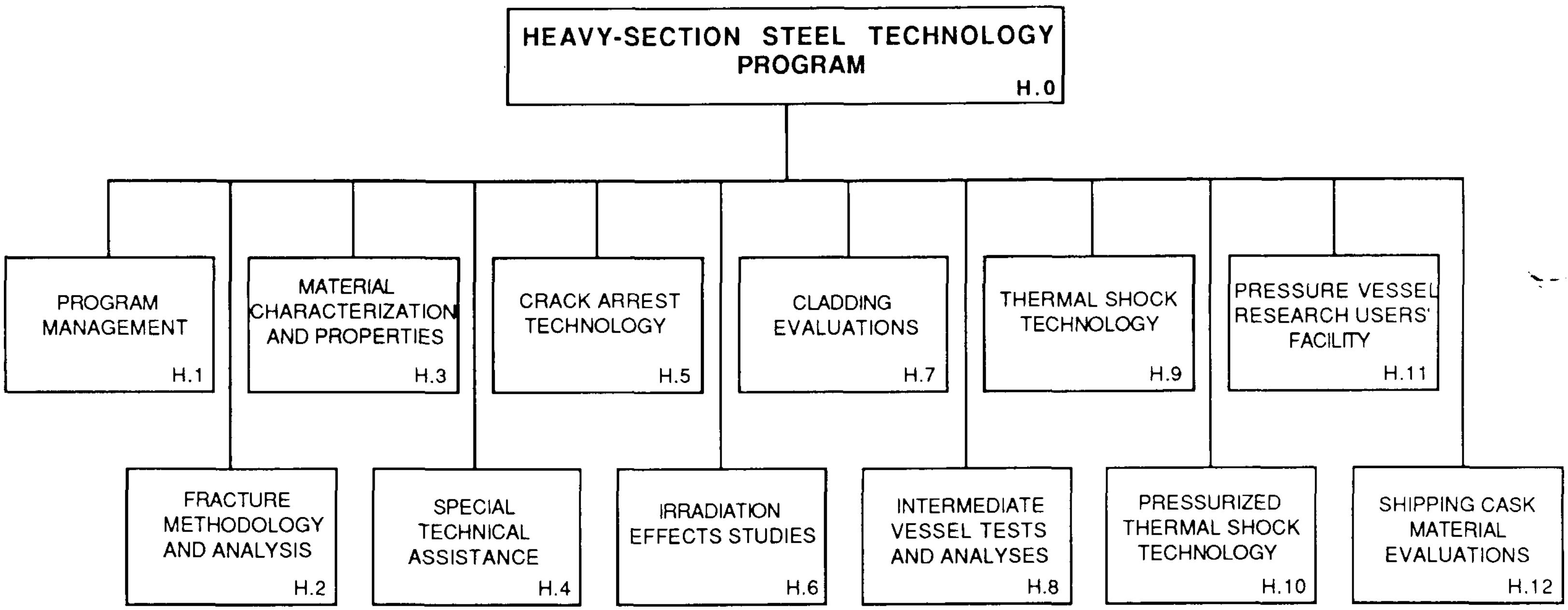

Fig. 1.1. Level-2 work breakdown structure for HSST Program.

work performed during the reporting period. The more detailed technical accounts of the work that have been included in previous progress reports will be covered in an expanded number of program topical reports.

During this period, 11 program briefings, reviews, or presentations were made by the HSST staff during program reviews and visits with NRC staff or others. Five topical reports,[1-5] one foreign trip report,[6] and thirteen technical papers[7-19] were published. In addition 24 technical presentations were made: three[20-22] at the International Conference on Computational Engineering Science, held in Atlanta, Georgia, on April 10–14, 1988; two[23,24] at the Joint Meeting of ASTM Task Groups E24.08 and E24.06.02, held in Reno, Nevada, on April 27, 1988; one[25] at the CSNI Fracture Assessment Group Meeting, held at the Materialprüfungsanstalt Stuttgart, FRG, on May 24, 1988; two[26,27] at the IAEA Specialists' Meeting, held at the Materialprüfungsanstalt Stuttgart, FRG, on May 25–27, 1988; twelve[28-39] at the Fourth Annual HSST Program Workshop on Dynamic Fracture and Crack-Arrest Technology, held in Gaithersburg, Maryland, on June 1–2, 1988; two[40,41] at the 14th ASTM International Symposium on Effects of Radiation on Materials, held in Andover, Massachusetts, on June 27, 1988; one[42] at the American Nuclear Society Nuclear Power Plant Meeting, held in Snowbird, Utah, on July 31–August 3, 1988; and, one[43] at the TMS-AIME Fall Meeting and World Materials Congress, held in Chicago, Illinois, on September 27, 1988.

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2. FRACTURE METHODOLOGY AND ANALYSIS

2.1 Introduction

The Heavy-Section Steel Technology (HSST) Program is continuing to improve the understanding of conditions that govern the initiation, rapid propagation, arrest, and ductile tearing of cracks in reactor pressure vessel (RPV) steels. In pressurized-thermal-shock (PTS) scenarios, inner surface cracks in an RPV have the greatest propensity to propagate because they are located in the region of highest thermal stress, lowest temperature, and greatest irradiation damage. If such a crack begins to propagate radially through the vessel wall, it will extend into a region of higher fracture toughness because of the higher temperatures and less irradiation damage. Because crack initiation is a credible event in a PTS transient, assessment of vessel integrity requires the ability to predict *all* phases of a fracture event. These phases include crack initiation, nonisothermal propagation, arrest, stable or unstable ductile tearing, and structural instability. Through the integrated efforts of several laboratory and university research groups, the HSST Program is developing various components of the technology required to treat these phases of a fracture event. The technology includes fracture models, analysis methods, criteria, and data curves and is being developed and validated through small- and large-specimen experiments.

The effect of viscoplasticity in dynamic fracture analysis is a key component of the technology under development. Viscoplastic effects are being included in the dynamic fracture models and computer programs, and their utility is being validated through analyses of carefully controlled experiments. Material properties characterization testing has been performed on A 533 grade B class 1 steel by Ohio State University (OSU),[1] SRI International,[2] and Southwest Research Institute (SwRI),[3] using tensile and split-Hopkinson bar techniques. These data have been used to derive material constants for the Bodner-Partom,[4] the Perzyna,[5] and the Robinson[6] viscoplastic models. These constitutive models, along with crack-propagation techniques and several proposed nonlinear fracture parameters, have been installed in HSST-developed finite-element computer programs. Two computer programs [ADINA/VPF[7-8] from Oak Ridge National Laboratory (ORNL) and VISCRK[9] from SwRI] have been developed independently to evaluate different analysis techniques and to ensure high-quality dynamic solutions. The capabilities of these nonlinear techniques are being compared and evaluated, in part, through applications to the small- and large-specimen crack run-arrest experiments. (A portion of these experiments is described in Chap. 5.)

The following sections describe advances made during this report period in the coordinated effort being conducted under the HSST Program by ORNL and several subcontracting groups to develop the crack-arrest data base and the analytical tools required to construct improved fracture models for RPV steels. Additional high-strain rate testing of A 533 grade B class 1 steel was performed for viscoplastic material characterization; rapid-loading fracture testing of small specimens was carried out for dynamic fracture toughness measurements. Analytical efforts have focused on fracture model developments for improved viscoplastic constitutive formulations, moving singular-element techniques for modeling crack propagation, and alternative parameters for characterizing inelastic dynamic fracture. Three-dimensional effects in fracture modeling were studied, including constraint effects on crack-tip yielding and tunneling effects on arrest toughness calculations.

2.2 Inelastic Fracture Model Development

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The basic postulate of linear-elastic fracture mechanics (LEFM) requires that the inelastic deformation surrounding the crack tip be contained within the K_I-dominant region. Furthermore, it is assumed that rapid crack propagation is governed by a unique *geometry-independent* material property, the dynamic fracture toughness, K_{ID}. Propagation of a running crack occurs under the condition that the applied dynamic stress-intensity factor K_I satisfies $K_I = K_{ID}(\dot{a},T)$, where K_{ID} is taken to be a function of the crack-tip velocity $\dot{a}$ and the temperature T. However, except for very short crack jumps, LEFM assumptions may not be strictly valid characterizations of rapid crack propagation.[10] In particular, a wake of residual plasticity left behind the moving crack tip can violate the K_I-dominance requirement of LEFM. An indication that LEFM conditions are not satisfied occurs when elastodynamic analyses of crack run-arrest data lead to geometry-dependent fracture toughness relations.[11] Recent studies indicate that plasticity and strain-rate effects ($\sim 10^4 \cdot s^{-1}$) can be important for rapid-loading situations, such as cleavage crack-propagation events in ductile RPV steels.[12] Consequently, the HSST Program research efforts at ORNL and several subcontracting sites are supporting the development of viscoplastic-dynamic, finite-element analysis techniques and validating their utility through analyses of carefully controlled experiments.

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2.2.1 Viscoplastic material model characterization

Research efforts at ORNL and several subcontracting sites are directed toward developing strain-rate and temperature-dependent constitutive models for A 533 grade B class 1 steel. As part of this effort, dynamic stress-strain data have been generated by OSU, SRI, and SwRI for A 533 grade B class 1 steel for use in deriving constants for proposed constitutive models. Kanninen et al.[3] obtained dynamic stress-strain data from tensile and split-Hopkinson bar tests for strain rates ranging from 0.001 to 550 s^{-1} and at temperatures of −60 to 150°C. To augment this data base, Giovanola and Klopp[2] performed 15 split-Hopkinson torsion-bar experiments at engineering shear strain rates ranging from 400 to 3000 s^{-1} and at temperatures of −60 to 150°C. Using similar test procedures, Gilat[1] has conducted tests at strain rates of ~800 and 5000 s^{-1} and at temperatures of −150 to 25°C. Results for these tests show that both temperature and strain rate have a significant effect on the material response of A 533 grade B class 1 steel.

During this report period, additional tests were performed by OSU on A 533 grade B class 1 steel at strain rates of ~800 and 5000 s^{-1} and at temperatures of 100, 150, and 240°C. When these data are combined with those previously reported for lower temperatures (−150 to 25°C) (Ref. 1), the results show a decrease in yield and flow stresses with increasing temperature. The rate of this decrease declines with increasing temperature; the strain-rate sensitivity also decreases with increasing temperature. If these trends continue as temperature is increased further, it may lead to a negative strain-rate sensitivity above some temperature for which the stress decreases as the strain-rate increases. Such a phenomenon has been observed previously in steel.

Microscopic (SEM) examinations of the specimens previously tested at −150 and 25°C were also conducted at OSU. The stress-strain curves from tests at −150°C showed an unusual material response. A gradual reduction in the stress was observed following yielding, which ended with a failure of the specimen. In contrast, the room-temperature tests show an upper and lower yield followed by strain hardening. SEM examination of fractured specimens from both temperatures shows a ductile localized failure. The localization region contains numerous voids. In the specimen from the room-temperature test, the voids nucleate and grow only after a substantial amount of plastic deformation. In the specimen from the tests at −150°C, the process of nucleation, growth, and coalescence of voids takes place over a short time immediately after yielding. The nucleation of the voids that led to the failure is believed to be controlled by different mechanisms at different temperatures. At room temperature, large plastic deformation around relatively rigid carbides and other obstacles nucleates voids because compatibility cannot be maintained. At −150°C, a combination of high stress (75% higher than at room temperature) and low fracture toughness nucleates voids at the interface of hard-particle and noncompatible grain boundaries. Thus, the yield point observed in the stress-strain curves of the tests at −150°C is not a yield point in the usual sense beyond which plastic deformation starts. Rather, it is the stress level at which a large number of voids nucleate.

Kanninen et al.[3] used dynamic stress-strain data to derive constants for the Bodner-Partom constitutive model (with isotropic hardening only) appropriate for A 533 grade B class 1 steel at test temperatures ranging from −60 to 150°C. At ORNL, Chang[13] employed stress-strain data from Refs. 2–3 and an extended version of the Robinson model[6] to represent the viscoplastic behavior of A 533 grade B class 1 steel. The Robinson model contains a scalar and a tensor state variable to describe isotropic and kinematic hardening, respectively. Models that combine both isotropic and kinematic hardening are generally more appropriate for representing conditions of unloading, such as those that occur behind a rapidly propagating crack tip.

Studies during this report period at ORNL have shown the versatility of the Robinson model for predicting the strain-aging effect at lower strain rates and the disappearance of this effect at higher strain rates for A 533 grade B class 1 steel. In addition, the Robinson model effectively represents the general trends of the reverse strain-rate effect, which is manifested by the crossing of stress-strain curves of different strain rates. However, improvements are required to better represent the starting point of the yield drop for the lower strain-rate curves. Material constants for the Robinson model to predict test data obtained from A 533 grade B class 1 steel over a range of temperatures will be reported in the near future.

During this report period, numerical implementation of the extended Robinson model into ADINA/VPF following a technique from Hornberger[14] was completed at ORNL. Studies have been initiated to compare predictions of the Robinson model with those of the Bodner-Partom model when applied to viscoplastic-dynamic fracture analysis of A 533 grade B class 1 steels.

2.2.2 Viscoplastic-dynamic analysis methods developments

The predictive capabilities of the nonlinear-dynamic fracture analysis techniques are being evaluated through applications of the ADINA/VPF[7–8] and VISCRK[9] computer programs to analyses of HSST crack-arrest experiments. Recently, viscoplastic-dynamic fracture analyses[15] of the WP-1 series of wide-plate tests[16] were conducted with the ADINA/VPF Program at ORNL using finite-element models with improved mesh refinement near the plane of crack propagation. Results from these analyses were compared with those obtained from other models with different mesh refinements along the crack plane. These combined results indicate that the viscoplastic-dynamic solutions of the wide-plate tests expressed in terms of the inelastic fracture parameters [e.g., T^* (Ref. 17)] have not yet converged for the mesh refinements employed thus far in these studies.

Insight into the difficulties associated with modeling rapid crack-propagation events in RPV steels exhibiting viscoplastic behavior are provided by two recent studies at OSU. Sheu[18] studied the mode I plane-strain problem of dynamic steady-state crack growth in A 533 grade B class 1 steel using the Bodner-Partom model characterized in Ref. 3 and the assumption of small-scale yielding.[18] Sheu resolved the near crack-tip singular fields using a finite-element model with element dimensions

~10^{-3} of the elastic-plastic zone size.[18] (For the Bodner-Partom model, the stress field is $r^{-1/2}$ singular and the elastic strain rates dominate the plastic strain rates near the crack tip.) Furthermore, for A 533 grade B class 1 steel over a temperature range from −60 to 100°C and a crack speed of one-half the Rayleigh wave speed, Popelar* estimates that the zone-of-dominance of the near-tip fields extends from ~5 to 55 μm compared with an inelastic region with dimensions 0.1 to 15 mm. Given that the computational capacity was available to resolve such a small region of an engineering structure using finite elements, it is clear that elements of this size invade the micro-heterogeneity of the material and broach the limits of isotropic continuum analysis.

Several techniques are being explored to circumvent these stringent requirements on crack-tip mesh refinement and related difficulties associated with possible violations of continuum assumptions. Nishioka has proposed an exclusion-zone technique that obviates the need for highly refined crack-tip elements.[19] In this technique, a small rectangular domain of height 2 ε is defined around the crack tip to approximate the finite fracture process zone. During the dynamic analysis, this rectangle is extended in length (but not in height) to include a portion of the plastic wake behind the advancing crack. Nishioka advocates excluding the integration of the volume term of the T^*-integral (see Ref. 17) from this extending exclusion zone. According to a study by Nishioka, the T^*-integral should be essentially invariant with respect to the size of this extending domain provided ε is sufficiently small.[19]

To investigate the potential of the foregoing technique for characterizing fracture behavior, O'Donoghue at SwRI recently performed studies of the geometry independence of the two-component parameter (T^*, ε) using a center-crack panel problem.[20] The panel had dimensions 2 w × 2 h × t (w = 40 mm, h = 20 mm, and t = 25.4 mm) with an initial crack length of a/w = 0.25. A fixed load was applied that produced a nominal stress of 300 MPa and an initial stress-intensity factor of $K_I(T^*) = 77$ MPa·$\sqrt{m}$. The panel was analyzed dynamically as a plane-stress problem using the Bodner-Partom model[3] for a constant crack velocity of $\hat{a} = 1000$ m/s and a series of increasingly refined meshes. Results of these analyses indicate that the time history of T^* was relatively insensitive to mesh refinement for a given height (ε = 1.25 mm) of the exclusion zone. Based on these preliminary calculations, further studies will be conducted by SwRI on the geometry independence of the (T^*, ε) parameter when applied to small- and large-specimen crack run-arrest data.

Moving singular-element formulations represent an alternative technique for achieving convergent solutions for fracture parameters in the context of viscoplastic-dynamic fracture analysis. Two aspects associated with moving singular elements may improve convergence characteristics in inelastic fracture applications. The first is the influence of including singular functions in the solution space of the Ritz-Galerkin approximation. The second, possibly more subtle, aspect is the capability of maintaining a node positioned precisely at the moving discontinuity. (Nodal relaxation techniques currently used in HSST codes to model crack propagation do not specify the exact position of the crack-tip except when the tip is located at an interelement boundary.) In

*C. H. Popelar, Personal Communication to B. R. Bass, Oak Ridge National Laboratory, Oak Ridge, Tn., August 8, 1988.

preliminary work, Thesken and Gudmundsson[21] have implemented a variable-order singular element proposed by Akin[22] into an elastodynamic finite-element formulation and have illustrated its advantages in modeling stationary cracks subjected to dynamic loading.

More recently, the formulation of Thesken and Gudmundsson[21] was extended to incorporate a moving element formulation that allows an adjustable region of convecting elements to be embedded at the crack tip within a finite body. The latter technique permits the order of the crack-tip singularity to be specified by an adjustable parameter for dynamic crack-growth problems. The necessary formulations have been installed in the elastodynamic finite-element program DYNCRACK. During this report period, the DYNCRACK program was implemented at ORNL and fully tested on a number of classical dynamic fracture problems. The program has been shown to perform well, with the variable-order singular element having an extremely favorable effect on the rate of convergence. Work is currently under way in the HSST Program to update the moving element formulation in DYNCRACK to accommodate viscoplastic material behavior. The resultant formulation will be investigated for its potential in resolving the near crack-tip singular fields of the Bodner-Partom constitutive model while remaining in the size regime of a continuum element.

2.3 Three-Dimensional Effects in Fracture Modeling*

C. W. Schwartz†

2.3.1 Constraint effects on crack-tip yielding

The HSST wide-plate test specimens[16] are conventionally treated in finite-element dynamic fracture analyses as two-dimensional, plane-stress problems. Although this approximation is reasonable over most of the structure, it deteriorates close to the crack tip where constraint induced by the near-tip triaxial stress gradients causes deviations from plane-stress yielding conditions. Plane-stress viscoplastic models may thus be in substantial error in the very region where rate effects are expected to be most significant. The substantial computational requirements of viscoplastic-dynamic, finite-element calculations preclude use of more refined three-dimensional models, even on state-of-the-art supercomputers.

Triaxial constraint effects and the transition from plane-stress to plane-strain yielding conditions in the crack-tip region are being investigated at the University of Maryland (UM) through a series of static, nonlinear, three-dimensional analyses of simple fracture geometries. Results compiled to date indicate that through-thickness variations in stresses and displacements are confined to a zone extending from the free surface to a depth of ~5% of the specimen thickness. Furthermore, plane-

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strain yield conditions dominate only for values of the yield factor Fy satisfying Fy < 0.5, while plane-stress conditions dominate for Fy > 3. (The yield factor Fy was defined previously by Hahn and Rosenfeld[23] to quantify the transition from plane-strain to plane-stress yielding as a function of specimen thickness and load level.)

The latter conclusion has implications for viscoplastic-dynamic fracture analyses of the 10- and 15-cm-thick wide-plate tests of A 533 grade B class 1 steel (WP-1 test series[16]). During initiation and most of the propagation phase, the yield factor approaches the small-scale, plane-strain yield domain (0.5 < Fy < 0.9). However, near arrest the yield factor increases well into the plane-stress yield region (3.3 < Fy < 5.0). Until three-dimensional, viscoplastic-dynamic analyses become feasible and/or more realistic two-dimensional yield approximations are developed, the assumption of plane-stress conditions in two-dimensional, viscoplastic-dynamic fracture analyses seems appropriate, particularly near arrest when nonlinear effects are most significant.

2.3.2 Crack tunneling effects

Current studies at UM are aimed at more refined models of the influences of tunneling on fracture toughness in ductile materials. Crack tunneling is commonly observed during the fracture of tough and ductile materials. The loss of constraint near the free surface of the specimen permits development of yielded ligaments that may extend for considerable distances behind the leading edge of the crack front. A significant portion of the apparent fracture toughness measured for a deeply tunneled crack may result from restraining effects of these yielded ligaments.

Considerable tunneling has been observed in several of the WP-1 series[16] of wide-plate tests, raising questions regarding the appropriateness of two-dimensional analytical models that ignore tunneling. Calculations by Popelar[24] suggest that correcting for the restraining effects of tunneling may substantially reduce the actual fracture toughness values inferred from the WP-1 test data, bringing these values closer to — and in a few instances below — the American Society of Mechanical Engineers (ASME) Sect. XI reference toughness curve.

The restraining effects of tunneling have been analyzed at UM for the 10-cm-thick WP-1 test specimens using techniques previously employed by Popelar[25] and by Smith[26]. Popelar[25] considered both strip ligament and parabolic ligament geometries and assumed that the restraining stresses can be averaged across the thickness of the ligaments. Smith[26] had observed earlier that thickness-averaging procedures (such as those employed by Popelar[25]) overestimate the restraining effect of the ligaments for small to moderate amounts of tunneling. Thus, Smith formulates a correction from an integral over the ligament of a solution from Tada et al.[27] for a point force applied to the crack face.

Results from the UM tunneling analyses are summarized in Table 2.1. These analyses have been performed independently of earlier interpretations of the WP-1 tests by Popelar[24]. Popelar predicts a large reduction in the computed arrest toughness, particularly for events WP-1.2B and WP-1.4A, where the tunneling corrections are ~30 and 60%, respectively,

Table 2.1. Tunneling effects on computed crack-arrest toughness values for the WP-1 series of wide-plate tests[a]

Test	$T - RT_{NDT}$ (°C)	K_β (MPa·√m)	Popelar approach[b]		Smith approach[c]	
			K_γ (MPa·√m)	K_α (MPa·√m)	K_γ (MPa·√m)	K_α (MPa·√m)
WP-1.2A	85	440	88	352	15	425
WP-1.2B	115	523	139	384	33	470
WP-1.3	77	243	29	214	22	221
WP-1.4A	52	158	99	59	20	138
WP-1.4B	83	397	56	341	9	388
WP-1.5A	79	229	22	207	3	226
WP-1.5B	95	300	56	244	N/A	N/A
WP-1.6A	77	285	52	233	58	227

[a]K_β = crack-arrest toughness from generation-mode elastodynamic analyses,

K_γ = ligament correction,

$K_\alpha = K_\beta - K_\gamma$ = net crack-arrest toughness.

[b]Method of analysis based on Ref. 25.

[c]Method of analysis based on Ref. 26.

of the uncorrected values. The Smith approach predicts significantly smaller corrections for events WP-1.2B and WP-1.4A that are ~10 and 20%, respectively, of the uncorrected values. For all cases except WP-1.6A, the corrected data based on Smith's approach lie between the uncorrected values and the corrected values from Popelar's approach. None of the Smith-corrected data points lies below the ASME reference curve.

A series of three-dimensional, nonlinear, static analyses of tunneled crack fronts will be conducted at UM to investigate the effects of ligament geometry, material behavior, crack-tip constraint, and arrest load on the inferred arrest toughness for finite-width specimens. Although static analyses neglect the effects of inertia, plastic wake, and possible non-self-similar crack growth, they can provide useful insights into the validity of the analytical approximations.

2.4 Fracture Testing and Cleavage-Fibrous Transition Studies at the University of Maryland*

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2.4.1 Dynamic fracture testing

2.4.1.1 Notched round bar. The impact-loaded, notched, round-bar experiment was developed by UM to simulate the effects of constraint of a very thick specimen with a relatively small round bar. The cylindrical shape should increase the effective thickness of the specimen by a factor of ~3. The specimen configuration consists of a notched bar with a 38.1-mm outside diameter and a machined notch of 19.1-mm diameter. The loading impact system for these specimens delivers 1051 J of energy when the weight of 58.8 kg drops through a distance of 1.83 m.

In the first test series, five specimens of A 508 steel were tested under axial impact loading.[28] All five specimens failed, and data obtained for the determination of the dynamic initiation toughness K_{Id} were found to be consistent and repeatable, yielding an average value of K_{Id} = 54 MPa·√m. A similar evaluation of the lower-bound initiation toughness of A 533 grade B class 1 steel is under way. During this report period, a total of seven A 533 grade B class 1 specimens have been tested. Two specimens were tested near 0°C, and five specimens were tested at room temperature. All specimens failed and strain-gage data were obtained for the determination of K_{Id}. The test results presented in Table 2.2 show greater consistency than more conventional forms of fracture testing [such as with compact crack-arrest (CCA) specimens]. The K_{Id} values from the notched round-bar tests are ~10 MPa·√m lower than the A 533 grade B class 1 K_{Ia} results from the round robin performed in support of the ASTM crack-arrest toughness test method. The fact that the K_{Id} values are lower than the K_{Ia} CCA specimen values is consistent with the multiple initiation sites around the circumference of the notch, giving a better lower bound on the initiation toughness.

In future tests, the specimen diameter will be reduced at the notch to allow for the development of the higher nominal stresses required to initiate fracture at elevated temperatures. Also, the capacity of the drop-weight tower will be increased by adding an initial spring loading to the drop weight.

2.4.1.2 Explosive initiation testing. Feasibility studies at UM demonstrated that strain-gage techniques could be employed to measure K_{Id} in notched short-bar specimens.[28] The loading of the short bar (400 mm

*Work sponsored by the HSST Program under Subcontract No. 7778 between Martin Marietta Energy Systems, Inc., and the University of Maryland.

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Table 2.2. Dynamic initiation toughness of A 533 grade B class 1 steel

Test No.	Temperature (°C)		K_{Id} (MPa·$\sqrt{m}$)
B1	22		80.0
B2	26		79.7
B4	26		77.0
B5	22		80.0
B7	28		79.3
		Av	79.2
B3	3		62.0
B8	3		69.7
		Av	65.8

in length) with its integral dog-bone ends is accomplished with four explosive charges that are detonated simultaneously. Tensile stress waves produced at each end of the bar propagate to the central region of the bar, where they combine to produce a rapidly increasing K_I field that initiates a stationary fatigue-sharpened crack.

Initial experiments of this type were conducted first with a brittle polyester and then with a very brittle steel (4340 hardened to Rc = 51). The current series of experiments utilizes a much more ductile A 533 grade B class 1 steel. In the first experiment on A 533 grade B class 1 steel, the strains generated near the crack tip after detonation were monitored with strain gages and the data interpreted to determine the strains at initiation. These strains were employed in a static K-ε relationship that yielded a K_{Id} of 76.9 MPa·$\sqrt{m}$ at initiation. This value is very close to the lower-bound toughness measurements for A 533 grade B class 1 steel determined from the notched round-bar specimens (see Table 2.2).

The specimen did not fracture into two separate pieces during the dynamic event. Three additional tests were performed with this specimen with progressively larger charges detonated in an attempt to "fail" the specimen. Following the fourth test, it was decided to sever the specimen and examine the crack surface to determine whether the loadings had blunted the crack tip or whether internal damage not evident from a visual examination had been produced. This examination indicated that a localized (pop-in) fracture had actually been produced, presumably during the first experiment. Fractographic studies indicated that a mixed cleavage-fibrous failure developed in the small pop-in produced by the dynamic loading. In the next experiments, face grooves will be used in an attempt to increase the width of the pop-in.

2.4.2 Cleavage-fibrous transition behaviors

Topographic analyses of selected wide-plate fracture surfaces are being performed at UM using stereo-SEM and relative-height measurement techniques. During this report period, optical stereo photographs were made by UM of the initiation region on the fracture surfaces of the WP-1.8 wide-plate specimen.[29] Topology measurements showed that the initiation occurred from separated regions of the crack front at times that were almost simultaneous. The dominant event was adjacent to a segment of the serrated precrack front where the appearance of the precrack leading edge was quite clear. Other regions of the precrack leading edge showed fine scale irregularities. No other features that might pertain to the higher-than-expected initiation load in the WP-1.8 test were observed.

Optical stereo photographs were also made across a large number of the run-arrest events observed in wide-plate test WP-2.2. The purpose of the topology measurements was to see whether measurable abrupt increases of crack opening could be observed following the arrest of general cleavage. Because of surface irregularities, abrupt changes of crack opening smaller than 0.6 mm (a K-value increase of about 100 $MPa \cdot \sqrt{m}$) were not measurable, and no larger changes of crack opening adjacent to crack arrest were observed. In addition, the rate of change of crack opening across cleavage and dimpled rupture regions was closely similar. These results suggested that, across a substantial temperature range adjacent to loss-of-cleavage, the separation resistance of cleavage and fibrous regions was nearly the same. Estimates from thickness reduction measurements provided convincing support for the rapid increase of K that developed with increase of temperature. Details of the above topology measurements will be published in a future report.

2.5 Elastic-Plastic Fracture Studies In Inhomogeneous Materials

B. R. Bass

During this report period, plans were formulated for the HSST Program to participate in a multiyear investigation of elastic-plastic fracture models for inhomogeneous structures. A major product of this investigation will be the development of an elastic-plastic fracture estimation scheme applicable to inhomogeneous structural components. The research will be performed as part of a cooperative effort between the HSST Program and the Atomic Energy Research Committee of the Japan Welding Engineering Society. The Research Committee is chaired by Professor G. Yagawa of the Department of Nuclear Engineering, University of Tokyo, and includes researchers from over 25 industrial and university research organizations in Japan. Through a subcontract agreement with the HSST Program, the Century Research Center (CRC) Corporation (with offices in Toyko and San Jose, California) will serve as liaison between the Research Committee in Japan and the HSST Program. (The elastic-plastic fracture study of inhomogeneous materials is identified as the EPI Program by the Research Committee.)

The research effort in the EPI Program will consist of both experimental and analytical/computational studies divided among several subtasks to be performed by the participating organizations. Commencing in FY 1989, researchers will initiate a survey of material data available from laboratory specimens and engineering structures; an assessment of proposed inelastic fracture analyses techniques is also scheduled. In a follow-on task, crack-growth simulations in homogeneous and inhomogeneous compact tension, three-part bend, and center-crack tension specimens will be performed using computer codes based on incremental plasticity and deformation-theory plasticity formulations. Results from these analyses will be used in the development of estimation schemes for analyzing ductile crack growth in inhomogeneous laboratory specimens. Later research efforts will focus on development of estimation schemes for ductile crack growth in inhomogeneous engineering structures via analyses of through-wall and surface cracks in plates and pipes. Finally, experiments on ductile crack growth in inhomogeneous specimens will be performed for verification of the proposed estimation scheme. Interim reports on developments in the EPI Program will be compiled and issued annually through completion of the program in FY 1992.

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†Available in public technical libraries.

3. MATERIAL CHARACTERIZATION AND PROPERTIES

R. K. Nanstad

Primarily for internal management and budgetary control, the Heavy-Section Steel Technology (HSST) Program created a separate task (Task H.3) for the work on material characterization and properties determinations. However, for the reader's convenience some contributions to this report are placed within other chapters according to the larger tasks that correspond to the particular material studies. For example, in addition to the work reported here, refer to Sect. 7.2 for crack-arrest studies in clad plates and Sect. 7.3 for nondestructive examination studies in support of cladding evaluations.

3.1 Material Characterization for Wide-Plate Series 2

S. K. Iskander E. T. Manneschmidt

The chemistry of the material used in the Wide-Plate Series 2 (WP-2) is in accordance with ASTM Standard Specification for Pressure Vessel Plates, Alloy Steel, Chromium-Molybdenum (A 387), grade 22 class 2 (2 1/4 Cr-1 Mo) steel. It was specially heat-treated to produce low-upper-shelf (LUS) toughness and a high transition temperature. The posttest characterization of this material is being conducted with specimens machined from the broken halves of wide-plate specimens WP-2.1 and -2.5. All characterization has been performed in the T-L orientation, that is, the fracture plane of the test specimens is parallel to the fracture plane of the WP-2 plates.

Drop-weight nil-ductility transition temperature (NDT), Charpy V-notch (CVN), tensile, and fracture toughness testing from both WP-2.1 and -2.5 material has been completed. The results of NDT and CVN testing have been analyzed and are summarized below. The analysis of the remaining tests will be performed during the first half of FY 1989.

For purposes of comparing the results of the present tests, two characterization blocks, PTC1 (Ref. 1) and WPQ1 (Ref. 2), were selected. Block PTC1 is the characterization material used in conjunction with the flaw insert for the pressurized-thermal-shock experiment (PTSE-2) vessel that received a stress-relief heat treatment of 524 ± 14°C. The wide plates and characterization block WPQ1 were heat-treated as one piece at 552 ± 14°C. Block WPQ1 was not representative of the wide plates but has been included solely as a reference to previous results. Thus, the known problem with WPQ1 and the differing heat treatment of PTC1 should be kept in mind.

Some minor differences were found between the results of the present study and those previously reported for the characterization block PTC1. More significant differences were found when the present results are compared with those reported in Ref. 2 for the wide-plate qualification test piece WPQ1.

3.1.1 Drop-weight testing

Drop-weight testing was conducted with standard P-3 specimens per ASTM Standard Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels (E208-85). Although this standard explicitly states that the drop-weight test is insensitive to specimen orientation, the specimens were machined so that the crack on the specimen surface propagates in the T-L orientation and from stock on either side of the midthickness (1/2t) of the plates. The NDTs were determined to be 60 and 55°C for material from plates WP-2.1 and -2.5, respectively.

Noteworthy is that NDT for material from characterization block PTC1 at 1/4t was 49°C; although there was insufficient material for a full determination, a tentative pretest NDT for specimens machined from a portion of WP-2.4 was estimated to be 60°C (Ref. 1). This temperature is in agreement with the posttest value obtained for WP-2.1. The 5°C difference between NDT for plates WP-2.1 and -2.5 is insignificant, considering the variability that this material has exhibited and the reproducibility of the NDT test.

3.1.2 CVN testing

CVN testing has been performed on 12 specimens from the 1/2t depth of both WP-2.1 and -2.5. In addition, 12 specimens were tested from the 1/4t depth of WP-2.1. In the following description, these three sets of specimens will be referred to as originating from "three regions" for brevity. All specimens were in the T-L orientation and were tested at the same temperature intervals. About ten spare specimens are still available if needed. The temperature ranged from 25 to 250°C to provide comparison with previous work.[1,2]

If CVN energy of material from the three regions is compared, the upper-shelf energy (USE) levels are very similar, ranging from 61 to 65 J. There is relatively little difference in energy absorbed in the midtransition region (at 100°C). The greatest differences (±10 J) were exhibited in the lower transition region around NDT. An RTNDT (in accordance with Subarticle NB-2330 of the *ASME Boiler and Pressure Vessel Code*, Sect. III) cannot be made because a CVN impact energy of 68 J (50 ft-lb) was not attained (except for a single specimen from each of the 1/2t-depth material).

The knee of the upper shelf is reached at ~150°C when judged by the attainment of 100% shear fracture appearance. This compares favorably with the results previously reported for PTC1 in Ref. 1. For WPQ1 (Ref. 2), 100% shear at a temperature of 93°C and an USE of 111 J have been reported, in comparison to a range of 61 to 65 J obtained in the present study.

The transition temperature (defined as approximately one-half the USE) from the present study ranges from 76 to 82°C and compares well with that of PTC1, 74 to 81°C. The transition temperature for WPQ1 is 52°C.

3.2 Fractography for Wide-Plate Series 2

I-B. Johansson

Examinations of the fracture surface of wide-plate, crack-arrest specimen WP-2.5 have been performed with a scanning electron microscope. Primarily, cleavage events associated with the first crack-initiation site and each reinitiation site following a crack arrest have been examined.

The fracture surface of wide-plate, crack-arrest specimen WP-2.5 was severely corroded upon receipt; after sectioning into smaller sizes, the following procedure was applied to clean the fracture surface. The specimen was placed in an ultrasonic cleaner with hexamethylene ($C_6H_{12}N_4$) and rinsed thoroughly with propanol; then it was dried and coated with a film of Duo Seal vacuum pump oil. Examination of the fracture surface of specimen WP-2.5 revealed that it had been attacked by hexamethylene and was slightly oxidized. Therefore, it was difficult to locate the specific sites for the cleavage initiation events. It was obvious, however, that significant ductile tearing preceded the first cleavage crack-initiation event. The ductile tearing had a typical dimple structure and extended about 2.2 mm beyond the electron-beam weld. The second cleavage event was also preceded by ductile tearing, and the cleavage initiation site was associated with a particle located a small distance ahead of the ductile tearing region.

Ductile tearing occurred upon all the latter six crack-run arrest events. Of the remaining crack-initiation sites, only the sixth initiation site could be accurately determined, although it was partly concealed by iron oxide. The others appeared to be concealed with iron oxide or the fracture surface was too damaged to trace back to the initiation site. Further examinations will be performed on WP-2.4.

3.3 dc-Potential Drop J-R Curve Round Robin

D. J. Alexander R. K. Nanstad
R. L. Swain

The J-R curves for four specimens, two steel and two aluminum alloys, were determined using a computer-controlled test procedure and a dc-potential drop (dc-pd) technique for monitoring crack extension. The tests were performed as part of a round-robin project on the use of dc-pd coordinated by the David Taylor Research Center (DTRC). Specimens were fabricated and precracked by DTRC and were tested at Oak Ridge National Laboratory (ORNL) without modifications.

The computer-controlled procedure is one that allows the operator to choose unloading compliance or dc-pd as the controlling test method. If unloading compliance is chosen, dc-pd can be monitored simultaneously. For this round-robin project, unloading compliance was selected for test control, and dc-pd was also used to monitor crack extension. This method was chosen to allow direct comparison of the two techniques. A direct current of 10 A was used in each case. During the hold period before each unloading, 60 potential readings are taken and averaged, compared with a reference value, and used to calculate the crack extension based on a predetermined correlation published by DTRC for the specimen configuration used for this project.

Results from all four tests showed fairly good agreement between the compliance and dc-pd crack extension predictions and good agreement of both techniques with the measured crack extensions. Clearly, the selection of normalization voltage is crucial to the final analyses.

The procedure used was based on determination of the initiation of crack growth from a plot of potential voltage vs specimen displacement. For the two steel specimens, agreement between J_{Ic} values and the J-R curves was good. For the two aluminum specimens, a large discrepancy was observed, and subsequent discussions with DTRC point toward specimen orientation differences. Similar differences were observed by other round-robin participants.

The J_{Ic} (modified Ernst) results were 30.1 and 57.8 kJ/m^2 for the aluminum and 152.3 and 170.0 kJ/m^2 for the steel. The J-integral values were determined by the Merkle-Corten (J_{MC}), the deformation theory (J_D), and the modified-J (J_M) methods, while the J_{Ic} values given were determined using only the data between the exclusion lines. If all the data are used to determine J_{Ic}, the values change as much as 18% for the steel and 10% for the aluminum. Regarding the J-R curves, the ordering from highest to lowest was J_M, J_{MC}, and J_D for the steel, but J_{MC}, J_M, and J_D for the aluminum.

3.4 LUS Weld Material Investigations

D. J. Alexander R. K. Nanstad
G. M. Goodwin

In preparation for the conduct of the Fourth Pressurized-Thermal-Shock Experiment (PTSE-4), 2 1/4 Cr-1 Mo weld metal and Linde 80 flux were procured for the fabrication of a submerged-arc weld intended to have low CVN impact energy on the ductile shelf and a high transition temperature. The target ranges of properties are yield strength of 520 to 760 MPa (75 to 110 ksi), a CVN ductile shelf energy of 54 to 68 J (40 to 50 ft-lb), and a CVN transition temperature (average of USE and lower shelf energy) of 90 to 140°C (194 to 284°F).

The weldment was prepared in 150-mm-thick (6-in.) plate and sectioned into slabs of sufficient thickness for machining of CVN and tensile specimens. Some slabs were given postweld heat treatments (PWHTs) of 510°C (P510) or 565°C (P565) for 6 h followed by furnace cooling to 150°C. Additional steel was placed above and beneath the weld slices in an attempt to more closely simulate the thermal mass of a vessel insert. Test specimens were fabricated from the center 100-mm weldment thickness and centered across the weld so the specimen orientations are transverse with the CVN crack propagation in the direction of welding. Specimens were also fabricated from as-welded (AW) slabs.

Tensile results for the AW, P510, and P565 conditions gave average yield strengths of 726, 707, and 689 MPa, respectively. The ultimate strengths were similar, ranging from 804 to 821 MPa. Total elongations range from 11 to 13%. Charpy impact results revealed transition temperatures of 40, 34, and 60°C with USEs of 48, 42, and 42 J, respectively. Furthermore, the lateral expansion values on the upper shelf are ~0.81 mm (0.032 in.).

These preliminary results show the desired yield strength, but CVN transition temperatures and ductile-shelf values were lower than desired. Chemical analysis of the weld is under way, as are additional PWHTs, to define a wider treatment vs behavior window.

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4. SPECIAL TECHNICAL ASSISTANCE

4.1 LWR Vessel Supports

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G. C. Robinson R. K. Nanstad

4.1.1 Introduction

Comparison of radiation-induced embrittlement data from the High-Flux Isotope Reactor (HFIR) vessel surveillance program with similar data from materials testing reactors (MTRs) indicates a fluence-rate effect; that is, the lower the fast neutron flux, the greater the embrittlement per neutron.[1,2] Furthermore, it appears that the HFIR data are applicable to an evaluation of light-water reactor (LWR) vessel supports because the exposure conditions (temperature, neutron flux, and materials) are similar. As a result, the Nuclear Regulatory Commission (NRC) requested that Oak Ridge National Laboratory (ORNL) determine the impact of radiation-induced embrittlement on the integrity of LWR vessel supports, considering the HFIR data. An initial study and a report were to be completed by September 30, 1988, and that milestone was met. Preliminary studies are discussed in Ref. 2, and the completed initial study is reported in Ref. 3.

4.1.2 Objective and scope of initial study

The primary objective of the study was to determine if the integrity of any of the LWR vessel supports would be in jeopardy before 32 effective full-power years (EFPY) as a result of radiation embrittlement. The scope of the study included development of correlations for applying the HFIR data to LWR supports, a survey and cursory evaluation of all LWR vessel supports, selection of two LWR plants for specific-plant analysis, and the specific-plant analyses.

4.1.3 HFIR vessel surveillance data

The HFIR surveillance data are discussed in Ref. 2 and are summarized herein.

The existence of a rate effect is indicated by a comparison of typical high-flux, low-temperature [<100°C (<200°F)] MTR ΔNDTT-vs-dpa data with the HFIR surveillance data, which correspond to essentially the same irradiation temperature [50°C (120°F)], materials (A212-B, A105-II, and A350-LF3), and fast-neutron energy spectrum but have a fast-neutron flux that is much less (2×10^8 compared with 1×10^{13} neutrons/cm^2·s for E > 1.0 MeV). As shown in Fig. 4.1, the HFIR data fall far to the left of the MTR data, indicating a fluence-rate effect.

Although differences in the spectra for the two reactors were considered small, they were accounted for by correlating the data with dpa (E > 0.1 MeV) rather than fluence (E > 1.0 MeV). Furthermore, possible

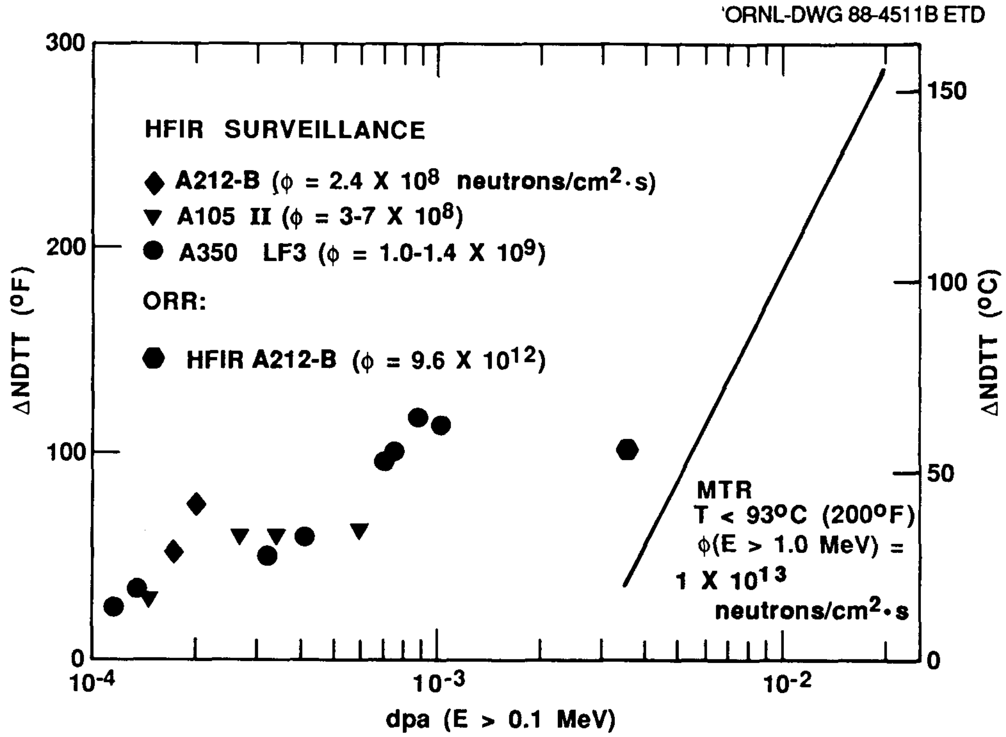

Fig. 4.1. Comparison of HFIR vessel surveillance data and MTR data [ΔNDTT vs dpa (E > 0.1 MeV)].

significant differences in chemistry were considered by irradiating HFIR A212-B archive material in the Oak Ridge Research Reactor (ORR), a typical high-flux, low-temperature MTR. As shown in Fig. 4.1, the data point falls close to the other MTR data, indicating that the large difference between the MTR and HFIR data is not the result of differences in chemistry and/or spectrum.

4.1.4 Correlation of HFIR data for application to LWR vessel supports

Application of the HFIR data to the evaluation of LWR vessel supports requires extrapolation of the HFIR data and thus correlations between ΔNDTT, dpa, and dpa rate. A first attempt to correlate the data is discussed in Ref. 2, and the more recent correlation is shown in Fig. 4.2. To obtain the correlations, data representing ΔNDTT vs dpa for the different irradiation rates in the HFIR and MTR irradiations were reduced to best-fit straight lines in log-log space. To permit extrapolation and interpolation of ΔNDTT relative to dpa rate, the correlations were cross-plotted as shown in Fig. 4.2. The correlations indicated by Fig. 4.2 are considered to be more accurate than similar correlations discussed in Ref. 2. However, in either case, the uncertainty associated with the extrapolation is considered to be rather large.

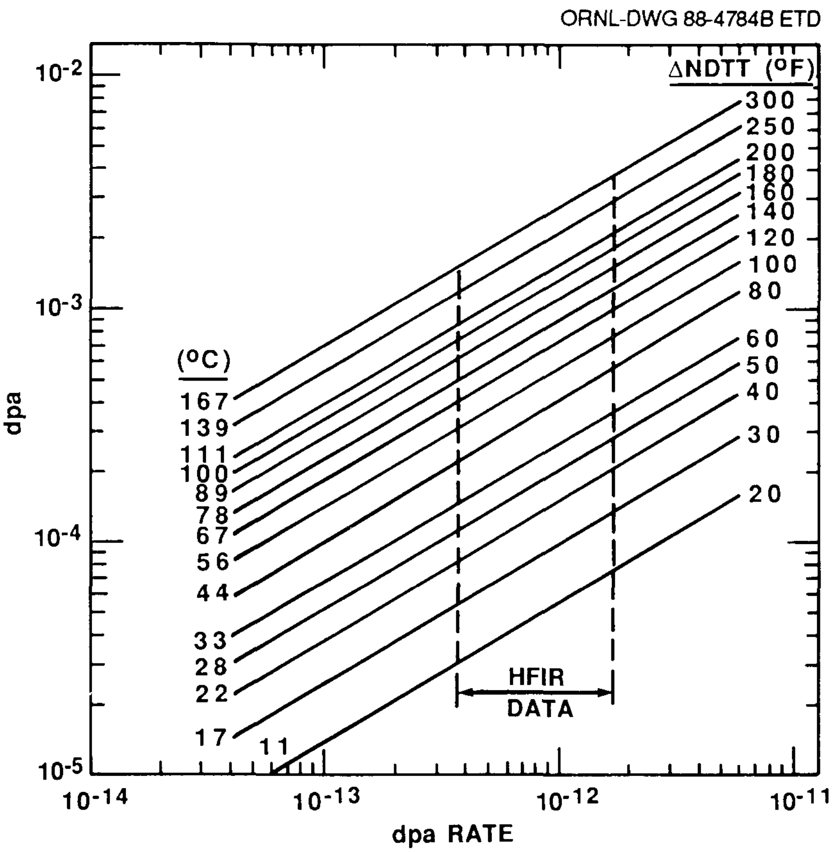

Fig. 4.2. dpa vs dpa rate with ΔNDTT as parameter.

4.1.5 Preliminary applications of HFIR data

Using the radiation-damage trend curves in Ref. 2 and Fig. 4.2 and also the MTR data, ΔNDTT values were calculated for "typical" General Electric (GE), Babcock & Wilcox (B&W), Westinghouse (W), and Combustion Engineering (CE) plants, for which multigroup fluxes were available, assuming vessel supports to be located at midheight of the core in the cavity between the vessel and biological shield. The results (Table 4.1) indicate much larger values based on the HFIR data than on the MTR data. It is also apparent that the two HFIR correlations result in about the same value.

4.1.6 Specific-plant analysis

The results in Table 4.1 are indicative of a substantially increased potential for propagation of flaws. However, an accurate assessment requires a detailed specific-plant, fracture-mechanics analysis because of the wide variation in support designs. After a survey and cursory evaluation of all LWR vessel support designs and the development of criteria for selecting plants for detailed specific-plant evaluation, Trojan (Portland General Electric) and Turkey Point Unit 3 (Florida Power and Light) were selected. The vessel supports for these plants include

Table 4.1. Vessel support ΔNDTT values corresponding to 32 EFPY and midheight of core ("typical" LWR plants)

NSSS designer (type reactor)	ϕ (E > 1 MeV) (neutrons/cm^2·s)	dpa rate (E > 0.1 MeV) (s^{-1})	dpa	ΔNDTT (°C)		
				MTR data	HFIR data	
					A[a]	B[a]
GE (BWR)[b]	2.9×10^7	5.8×10^{-14}	5.8×10^{-5}	0	*c*	*c*
B&W (PWR)[d]	2.0×10^8	6.1×10^{-13}	6.1×10^{-4}	11	100	72
W (PWR)	5.9×10^8	3.9×10^{-12}	3.9×10^{-3}	28	133	122
CE (PWR)	1.8×10^9	4.5×10^{-12}	4.5×10^{-3}	39	139	122

[a]ΔNDTT vs dpa (E > 0.1 MeV) correlations A and B (preferred).

[b]Boiling-water reactor.

[c]ΔNDTT not estimated for dpa < 10^{-4}.

[d]Pressurized-water reactor.

steel cantilever beams that are partially embedded in the concrete biological shield and extend into the cavity between the shield and vessel, where the "vessel" load is applied to the beam (Fig. 4.3). This design feature is not used for many plants but is of particular interest because the beam is in bending, and this introduces tensile stresses that are a necessary ingredient for propagation of flaws.

The Trojan beam is located just below midheight of the core, where the fluence is a maximum, and the upper and lower flanges of the beam contain a grout hole, directly above the innermost pedestal, that was introduced by flame cutting. The existence of the hole increases the primary stresses, and the flame-cutting operation introduces the possibility of quench cracks, residual stresses, and a localized area of reduced fracture toughness.

The design of the Trojan support is such that only vertical loads are applied to the cantilever beam. The most severe load considered included that associated with an auxilliary-line-break loss-of-coolant accident (LOCA). For the fracture analysis, dynamic fracture toughness data consistent with the dynamic response of the vessel/support system were used.

The Turkey Point beam is located near the upper end of the core, where the fluence is about one-half that at midheight, and there is no geometric discontinuity such as that in the Trojan beam. However, the design is such that the beam is subjected to both horizontal and vertical loads, and the most severe load considered included that associated with a large-break LOCA. Because of a lack of fracture toughness and vessel/support dynamic-response data, the American Society of Mechanical

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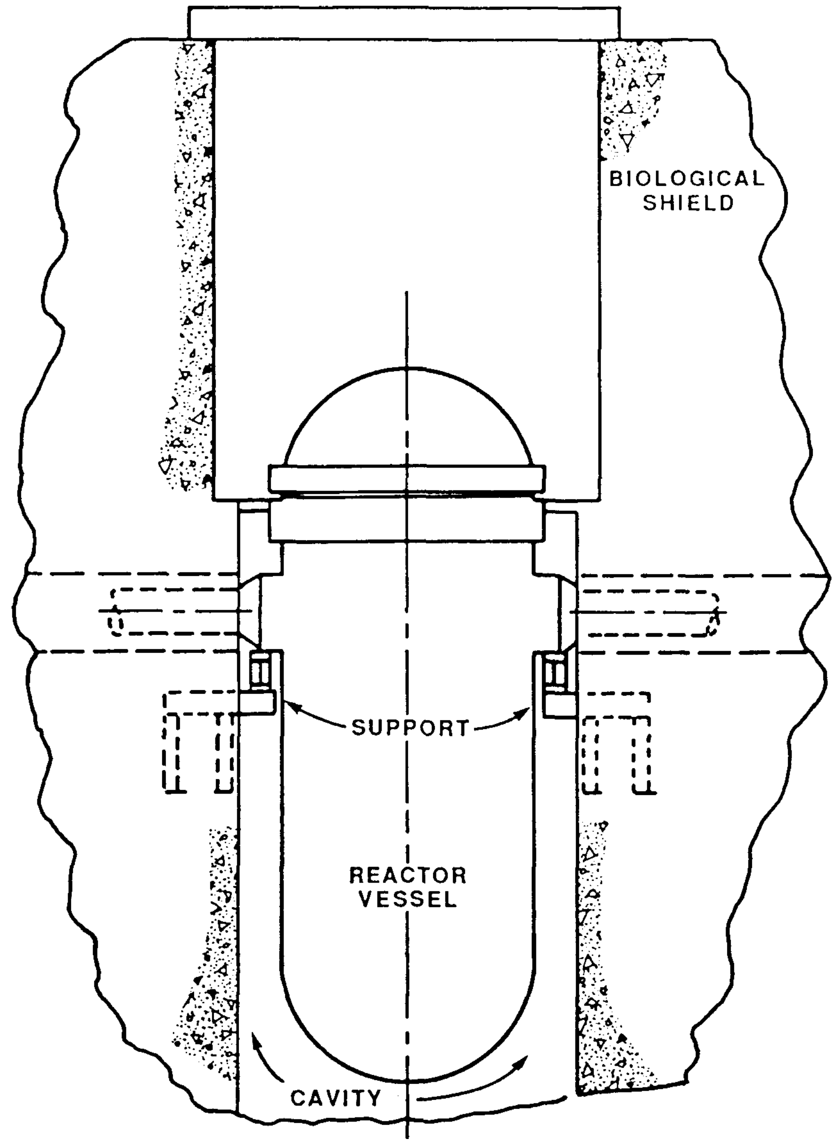

Fig. 4.3. PWR vessel support that includes partially embedded steel cantilever beam.

Engineers (ASME) lower-bound dynamic fracture toughness curve (K_{IR} vs $T - RT_{NDT}$) was used. Thus, with regard to the most severe loading condition and the fracture toughness of the beam material, the Turkey Point evaluation was conservative relative to that performed for Trojan.

Flaw locations considered were in the top flange of the beam at radial locations corresponding to the inner surface of the shield (maximum fluence), point of maximum bending moment (outboard of the shield), and for Trojan, the grout hole (beyond the point of maximum bending moment). Critical flaw sizes were calculated for each of these locations for operating times corresponding to late 1988 and to 32 EFPY. Flaw types included semielliptical surface flaws, corner cracks (Turkey Point), and edge notches (Trojan grout hole). Results of the analyses (Tables 4.2 and 4.3) indicate that for the most severe loading condition, the minimum critical flaw depth for late 1988 is 22 mm (0.9 in.) for Trojan and 8 mm (0.3 in.) for Turkey Point. For 32 EFPY, the values are 11 mm (0.4 in.) and 8 mm (0.3 in.), respectively. The flaw with a depth of 22 mm (0.9 in.) traverses the width of the beam [410 mm (16 in.)] and thus may have very low probability of existence. However, the others are small enough to be of concern.

The critical flaw sizes given in Tables 4.2 and 4.3 are best-estimate values in the absence of residual stresses. The inclusion of estimated residual stresses around the grout hole reduces the 32 EFPY minimum critical flaw depth from 11 to 6 mm (0.4 to 0.2 in.).

Table 4.2. Summary of critical flaw depths for Trojan

Location on beam	Loading condition[a]	Flaw type	Critical flaw depth (mm)					
			7.48 EFPY[b]		32 EFPY			
			a/ℓ[c]		a/ℓ			
			0	0.1	0	0.1	0.2	0.3
Shield	A	Surface	29	>32	21	>32		
inner	B	semi-	27	>32	20	32	>32	
surface	C	ellipse	23	>32	17	25	>32	
Maximum	A	Surface	29	>32	19	30	>32	
bending	B	semi-	27	>32	18	26	>32	
moment	C	ellipse	22	>32	15	20	28	>32
Flange	A	Twin	>50		41			
grout	B	edge	>50		30			
hole	C	cracks	>50		11			

[a]A = Dead weight (DW) + thermal (T) + operating basis earthquake (OBE)
B = DW + T + safe shutdown earthquake (SSE)
C = DW + T + small-break loss-of-coolant accident (SBLOCA)

[b]Corresponds to late 1988.

[c]Ratio of maximum depth (a) to surface length (ℓ)

Table 4.3. Summary of critical flaw depths for Turkey Point

Location on beam	Loading Condition[a]	Critical flaw depth (mm)					
		11.8 EFPY[b]			32 EFPY		
		Flaw[c]			Flaw		
		E	F	G	E	F	G
Shield inner surface	B	>64	>64	43	>64	>64	30
	D	41	33	10	30	25	8
Maximum bending moment	B	53	>64	25	48	>64	23
	D	28	23	8	20	18	8

[a]B = DW + T + SSE, D = DW + T + large-break LOCA (LBLOCA).

[b]Corresponds to late 1988.

[c]E = edge crack
F = full-width surface crack
G = quarter-circular corner crack

4.2 LUS Weld Evaluation

J. G. Merkle

Since 1979, the NRC and the nuclear industry have recognized the need to develop engineering procedures and criteria for dealing with the problem of LUS toughness welds in nuclear reactor pressure vessels (RPVs). In 1980, the NRC convened a task force that recommended analytical procedures but did not perform trial calculations or propose criteria. The ASME Section XI Working Group on Flaw Evaluation was asked by the NRC to develop criteria. This group has compared analytical procedures and developed proposed criteria. However, these criteria are partly based on tearing resistance characterization procedures that need further verification with respect to large-scale tests. Meanwhile, vessels in service continue to accumulate neutron fluence. Consequently, the NRC has asked the HSST Program to prepare a basis document for evaluating current efforts and planning the resolution of the LUS weld problem. The objective of this subtask is to provide such a document, the scope of which will include background, significance, status of research efforts and code activities, and a list of needed NRC and industry actions.

The current HSST Program effort on the LUS weld issue has two major thrusts: (1) participation in an ad hoc task group convened by the NRC to examine both current and improved controlling parameters for ductile crack extension and (2) compilation of the state-of-the-art assessment of the entire LUS issue.

The specific problem concerns the continued operation of pressurized-water reactors (PWRs) with LUS weld toughness. Already, five units have fallen below the legal screening limit of 68 J, and it is anticipated that an additional ten units will do so before reaching the end of their original design lives. It is expected that continued operation below the 68-J limit can be justified using an instability (J-T) analysis demonstrating that unstable crack propagation will not occur. A draft ASME document using this approach suggests that crack extensions up to 18 mm may be required to develop the necessary toughness. Furthermore, it is proposed that a parameter called J_{mod} should be used as the parameter controlling ductile crack extension rather than the conventional deformation J (J_D). However, the use of J_{mod} may be premature, because some "J-delta a" data are known to exhibit an upward curvature that is not physically reasonable.

The HSST Program has been active in the series of ad hoc working group meetings through participation of the staff as well as that of Professor J. Landes of the University of Tennessee and Professor H. Ernst of Georgia Institute of Technology and Alpha Research, who were both placed under subcontract for this purpose. The meetings have been held with two objectives: (1) to determine which J parameter is the more correct and define its range of validity, in particular, the allowable amount of crack extension and (2) to determine how small-specimen data may be extrapolated and applied to structures. Conclusions reached so far can be summarized as follows:

1. Considerable uncertainty exists regarding which J-like parameter is fundamentally more correct and its range of applicability.
2. At present, for conservatism, it would appear prudent to continue with deformation theory J, although with further advances in defining its limits of applicability J_{mod} may be preferred.
3. For many materials, existing crack-growth (delta a) limits in testing standards now appear unduly conservative: 30% of the remaining ligament appears more appropriate.
4. In testing high-toughness materials, it is more difficult to obtain "valid" data covering a fixed amount of crack extension. For these materials, limitations on J are likely to be more restrictive than crack extension limits; thus, it is unlikely that any increase in allowable crack extension would be of particular benefit. This is likely to be the cause with modern high-purity RPV steels and austenitic stainless steels.
5. There is a continuing need for development of elastic-plastic fracture-mechanics (EPFM) testing techniques to assist in the evaluation of different EPFM parameters, their ranges of applicability, and means of extrapolation to larger crack extensions.
6. Current approaches to defining allowable J and delta a limits are incomplete because they do not include a material sensitivity factor relating crack growth to changes in crack-tip constraint.

An outline of the evaluation report on the LUS issue has been compiled, and a presentation on the subject was made at the Sixteenth NRC Water Reactor Safety Information Meeting in October 1988. The report will cover a number of items: the causes, history, scope, and status of the problem; the historical and fracture-mechanics bases for the current 68-J (50 ft-lb) lower limit for Charpy upper-shelf energy; the code and existing regulatory guidance and an appraisal of its adequacy; an incorporation of the initial evaluation of appropriate fracture parameters from the ad hoc task group; and a description of relevant materials data for irradiated LUS welds. It is anticipated that the evaluation report will provide a basis for the recommended courses of action necessary to reach an improved regulatory position on the continued operation of vessels obtaining LUS welds.

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5. CRACK-ARREST TECHNOLOGY

D. J. Naus

5.1 Background

Current light-water-reactor (LWR), pressure-vessel, safety assessment methods are based in large measure on Sects. III and XI of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC)*. In pressurized-thermal-shock (PTS) scenarios, flaws on the inner surface of a reactor pressure vessel (RPV) have the greatest propensity to propagate because they are in the region of highest thermal stress, lowest temperature, and greatest irradiation damage. If such a flaw begins to propagate radially through the vessel wall, it will extend into a region of higher fracture toughness because of the higher temperatures and less irradiation damage. Although the thermal stresses may decrease with propagation depth, the stress-intensity factor caused by the elevated-pressure loading will be increasing. Assessment of the integrity of an RPV under such a postulated crack run-arrest scenario requires prediction of the arrest location, potential reinitiation, stable and unstable ductile crack growth, and structural instability of the remaining vessel wall ligament.

The fracture toughness correlations contained in the *ASME B&PVC* embody the position that one cannot assume a crack-arrest toughness value K_{Ia} above 220 MPa$\cdot\sqrt{m}$ for LWR pressure-vessel steels. The imposition of this limit is based in part on the fact that no K_{Ia} data existed at or above this level and because Charpy tests showed that impact energy levels exhibit an upper-shelf behavior. Therefore, the nature of crack-arrest behavior and K_{Ia} extrapolations to temperatures higher than that at which this limit occurred could not be presumed.

The ASME limit does not impose difficulties in making assessments for LWR pressure vessels undergoing thermal shock transients with low accompanying pressure levels. However, PTS scenarios could lead to conditions where the driving force on a propagating crack increases to levels well in excess of the current ASME limit. Thus, safety assessment methods for this type of condition would require an understanding of the following points.

1. If the driving force on a crack exceeds 220 MPa$\cdot\sqrt{m}$ by a significant margin, can the material exhibit crack-arrest behavior?
2. If the materials do exhibit high K_{Ia} values with increasing temperature, what is the relationship between K_{Ia} and temperature? That is, does a temperature limit exist above which cleavage crack propagation cannot continue regardless of how high the driving force?
3. If crack arrest does occur at high temperatures where the material behavior is typically dominated by ductile behavior, then what interactions exist between the various fracture modes, including arrest, stable crack growth, unstable crack growth, and tensile instability?

5.2 Objective

The primary objective of the crack-arrest studies is to generate data and associated analysis methods for understanding the crack-arrest behavior of prototypical RPV steels at temperatures near and above the onset of the Charpy upper-shelf region. Program goals include (1) extending the existing K_{Ia} data bases to values above those associated with the upper limit in the *ASME B&PVC*; (2) clearly establishing that crack arrest occurs before fracture-mode conversion; and (3) validating the predictability of crack arrest, stable tearing, and/or unstable tearing sequences for ductile materials.

5.3 Scope

Wide-plate tests and analyses are being used to provide bases for obtaining and interpreting dynamic fracture data (with relatively long crack runs) and bases for validation of viscoplastic fracture models and analysis methods. Supplemental tests are also planned using intermediate-size, stub-panel, crack-arrest specimens. During this report period three wide-plate, crack-arrest tests were completed. Also, checkout of the loading and heating-cooling systems for the stub-panel, crack-arrest specimens was completed as well as development of a draft performance specification for a high-speed data acquisition system.

5.4 Wide-Plate, Crack-Arrest Testing*

R. deWit† S. R. Low†
R. J. Fields†

5.4.1 Specimen preparation

Each test specimen is fabricated by welding a precracked, single-edge-notch test article (1 × 1 × 0.1 m or 1 × 1 × 0.15 m) to pull plates having the same cross-section geometry as the test article (Fig. 5.1). Up to 40 thermocouples and 25 strain gages are positioned on each specimen, with the concentration of instrumentation being adjacent to the plane of crack propagation. The specimen is then placed into the National Institute of Standards and Technology testing machine, and the heating–cooling–insulation systems are installed as well as the balance

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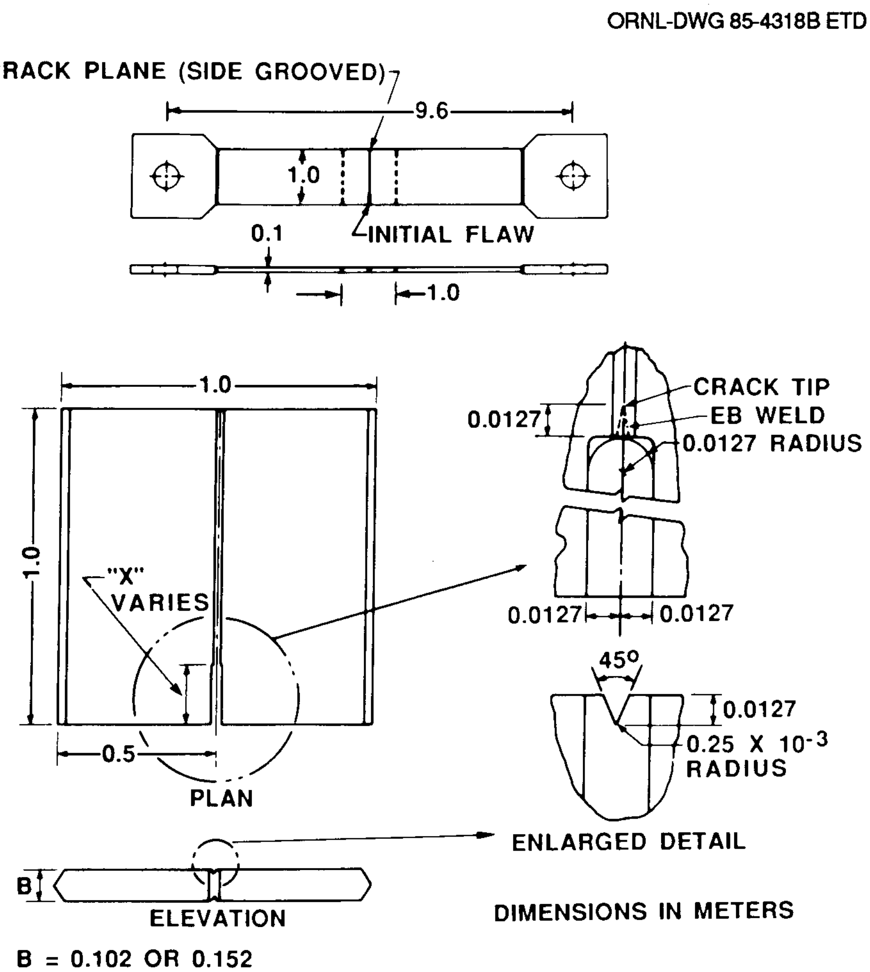

Fig. 5.1. Schematic of HSST wide-plate, crack-arrest specimen.

of the instrumentation (crack-opening-displacement gages, acoustic-emission transducer, and displacement transducer). More details on specimen preparation can be obtained from Ref. 1.

5.4.2 Testing procedure

After checkout of the instrumentation systems to demonstrate operability, a thermal gradient is imposed across the plate by liquid-nitrogen cooling of the notched edge while heating the other edge of the plate. When the desired thermal gradient is established, final calibrations on the instrumentation systems are completed, and tensile load is applied to the specimen until the crack run-arrest events occur. Figure 5.2 presents a specimen under test.

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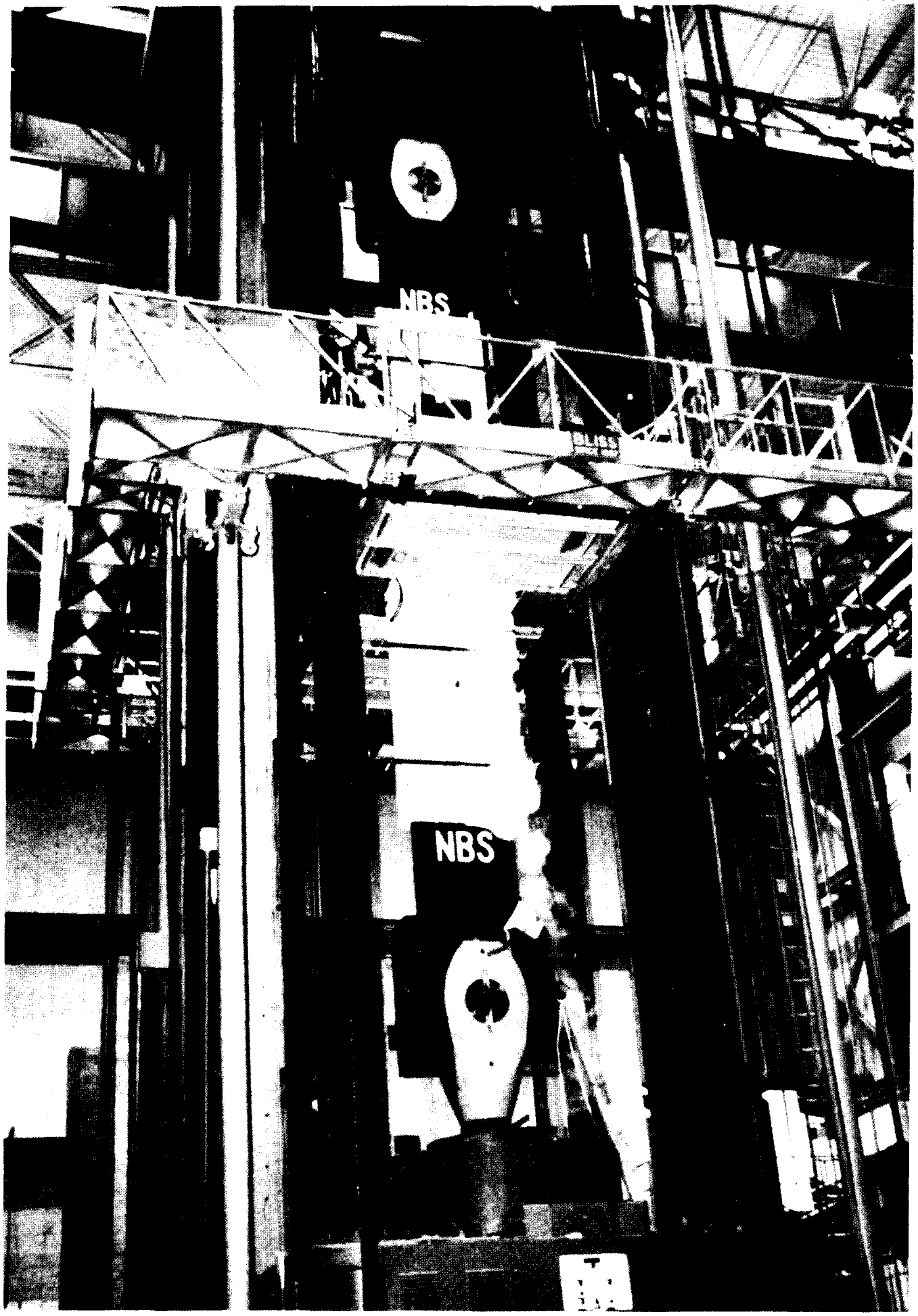

Fig. 5.2. Wide-plate, crack-arrest specimen under test.

5.4.3 Test summary

The wide-plate, crack-arrest tests utilized three materials: (1) prototypical A 533 grade B class 1 material from HSST plate 13A (WP-1 series), (2) a second heat of A 533 grade B class 1 material supplied by Combustion Engineering, Inc. (WP-CE series), and (3) degraded (simulated) low-upper-shelf base materials (WP-2 series). Properties of these materials are presented in Refs. 1–3. To date, 16 tests have been completed, with 3 having been conducted during this report period: WP-1.8, WP-CE-2, and WP-2.2.

5.4.3.1 Test WP-1.8. Test WP-1.8 was the eighth test in the WP-1 series and the second in this series to use a 152-mm-thick plate. After obtaining a satisfactory thermal gradient, the specimen was loaded at 20 kN/s until a load of 24.5 MN was obtained. A crack run-arrest event had not occurred, so the specimen was rapidly unloaded in an effort to resharpen the crack tip before application of a second load cycle.

Two changes were made in the testing procedure before application of a second loading cycle. The crack-tip temperature was lowered ~11°C, and the specimen loading rate was increased to 312 KN/s. At the load of 26.5 MN, the crack run-arrest events and ductile tearing initiated and lasted ~34 ms. Examination of the fracture surface and strain-gage records indicated that three cleavage crack run-arrest events occurred. More details on this test are presented in Ref. 4.

5.4.3.2 Test WP-CE-2. Before conducting test WP-CE-2, the specimen was warm prestressed by loading the specimen to 14 MN at 25°C, holding the load at 14 MN for 5 min, and slowly unloading to 5 MN. While maintaining the load at 5 MN, the temperature gradient was developed. Having established a satisfactory thermal gradient, the specimen was loaded at 9.6 kN/s. At a load of 14.6 MN, the crack run-arrest events initiated. Examination of the fracture surface and strain-gage records indicated that three cleavage crack run-arrest events occurred.

5.4.3.3 Test WP-2.2. Specimen WP-2.2 was also warm prestressed before testing by preheating the specimen to 120°C, applying a load of 16 MN, holding the load at 16 MN for 5 min, and slowly unloading to 3 MN. While maintaining the load at 3 MN, the thermal gradient was developed. After establishing the desired thermal gradient, the specimen was loaded until the cleavage crack run-arrest events initiated at ~17 MN. Preliminary indications are that multiple (more than three) crack run-arrest events occurred during this test.

5.5.4 Wide-plate analysis at ORNL (J. Keeney-Walker, B. R. Bass)

Posttest analyses were conducted for each wide-plate, crack-arrest test to investigate the interaction of parameters (plate geometry, material properties, temperature profile, and mechanical loading) that affect the crack run-arrest events. Three-dimensional, static, finite-element analyses were performed to determine the stress-intensity factor at the time of crack initiation using the ORMGEN/ORVIRT fracture analysis system[5,6] in conjunction with the ADINA finite-element code.[7] Quasi-static analyses used the Oak Ridge National Laboratory (ORNL) code WPSTAT

(Ref. 8) to evaluate the static stress-intensity factors as a function of crack length and temperature distribution across the plate. Elastodynamic analyses (application mode or generation mode) were carried out using the ADINA/VPF dynamic crack analysis code.[9]

Crack-arrest toughness values computed by both static and dynamic analyses as well as those determined using handbook techniques are presented in Ref. 10. Figure 5.3 presents results to date for the wide-plate tests as well as data obtained from other large-scale tests.[11–22] Collectively these data show that arrest can and does occur at temperatures up to and above that which corresponds to the onset of Charpy upper-shelf behavior, and the measured K_{Ia} values extend above the limit included in Sect. XI of the *ASME Code*.

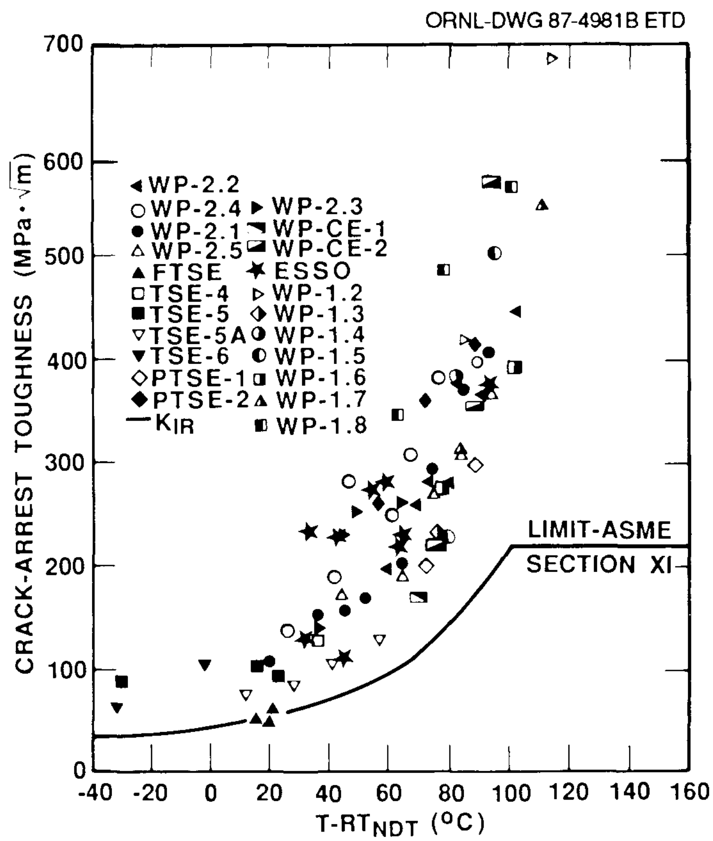

Fig. 5.3. High-temperature, crack-arrest toughness data vs temperature ($T - RT_{NDT}$) for wide-plate and other large specimen tests.

5.5 Stub-Panel, Crack-Arrest Testing

C. B. Oland G. C. Robinson

Studies to evaluate the usefulness of a relatively small panel specimen (Fig. 5.4) for crack-arrest experiments have continued at ORNL.

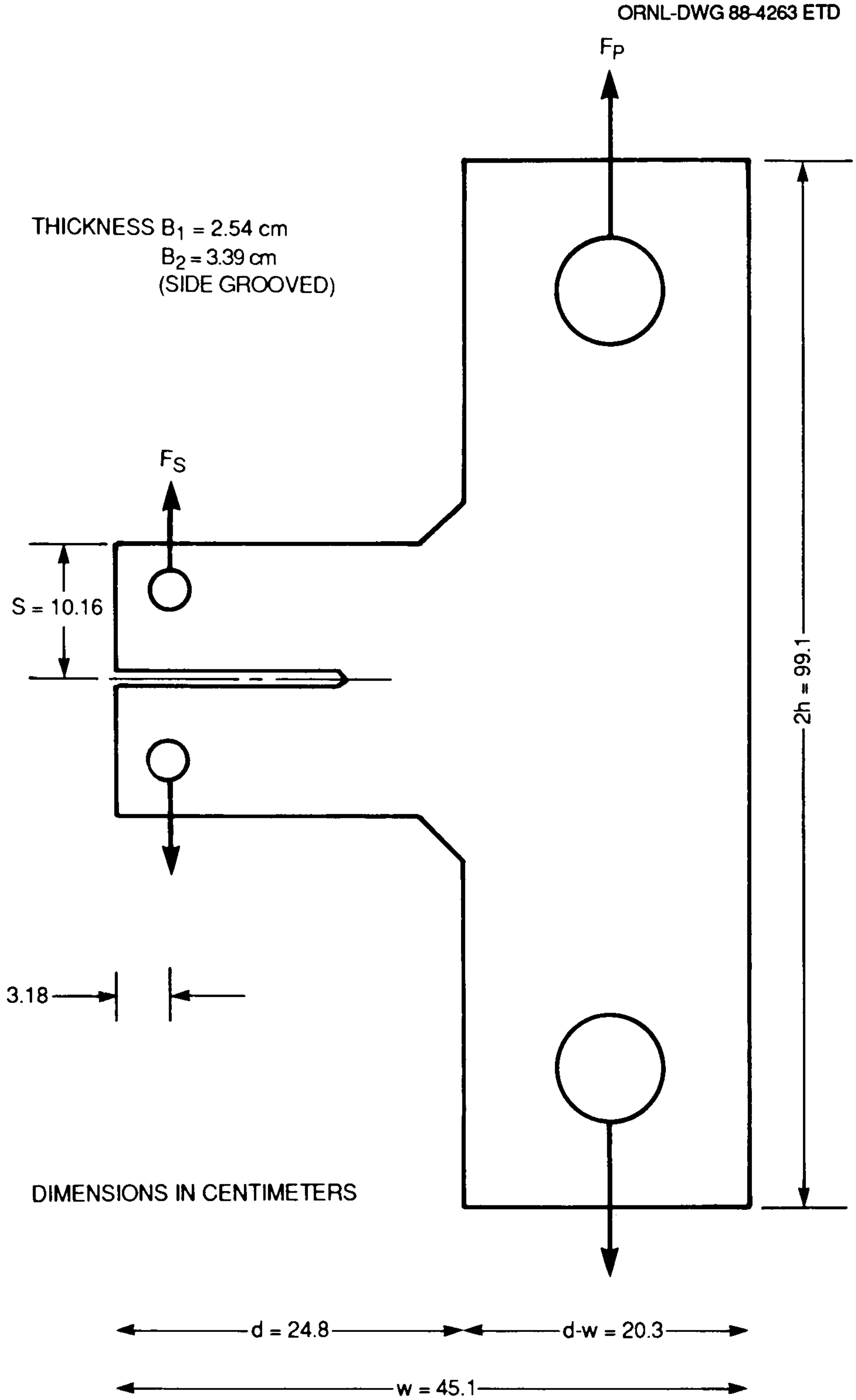

Fig. 5.4. Schematic of stub-panel, crack-arrest test specimen.

During this report period activities were related to development of the test equipment and drafting of a performance specification for a high-speed data acquisition system.

5.5.1 Testing equipment

All testing hardware components required for applying load to the specimen have been designed, procured, fabricated, and received. Hydraulic power supplies for the primary and auxiliary loading systems have been installed, and demonstration of operability of the loading systems has been completed. Thermocouple and strain-gage cables connecting the specimen to the data acquisition systems have been fabricated and intalled. Cable and signal-conditioning circuit fabrication for the two load cells has been completed. Four linear variable displacement transducers have been fabricated and calibrated.

A checkout of the heating-cooling-insulation systems used to develop the desired thermal gradient in the specimen was completed under representative test conditions; that is, primary and auxiliary loading hardware were installed on the specimen. Results of this evaluation show that the desired temperatures at the specimen crack tip (−40°C) and back edge (180°C) can be attained, with steady-state conditions established in ~4 h.

5.5.2 High-speed data acquisition system

A draft specification for a 40-channel, high-speed data acquisition and signal-conditioning system has been prepared. Basic components of the system include digital waveform recorders, a digital waveform storage oscilloscope, strain-gage signal conditioners, interconnecting cables, and software.

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6. IRRADIATION EFFECTS STUDIES

R. K. Nanstad

The Heavy-Section Steel Technology (HSST) Irradiation Effects Task (Task H.6) consists of a number of projects concerned with the effects of neutron irradiation on the fracture toughness and mechanical properties of reactor pressure vessel (RPV) materials. The task currently involves seven designated series of experiments; the first four are completed. The active series described here include (1) the Fifth and Sixth Series, which will characterize the shifts and shapes of the irradiated K_{Ic} and K_{Ia} curves, respectively, and (2) the Seventh Series, concerned with the irradiation resistance of stainless steel cladding.

6.1 Fifth Irradiation Series

R. K. Nanstad R. L. Swain
F. M. Haggag T. N. Jones

The objectives of the Fifth Series are to determine the K_{Ic} curve shifts and shapes of two high-copper, 0.23 wt % (72W) and 0.31 wt % (73W), submerged-arc welds. All planned unirradiated and irradiated testing for the Fifth Irradiation Series has been completed. The irradiations were performed by Oak Ridge National Laboratory (ORNL) at the Oak Ridge Research Reactor at a nominal temperature of 288°C to average fluences of ~1.6×10^{19} neutrons/cm^2 (>1 MeV). Tests performed included tensile, Charpy V-notch (CVN), impact, drop-weight (DWT), and fracture toughness. Unirradiated compact specimens of 25.4-, 50.8-, 101.6-, 152.4-, and 203.2-mm thickness (1TCS, 2TCS, 4TCS, 6TCS, and 8TCS, respectively) were tested, while irradiated testing was limited to 1TCS, 2TCS, and 4TCS.

Results show that the CVN 41-J temperature and DWT nil-ductility transition (NDT) temperature shifts are very similar for both welds following adjustment for fluence differences. The CVN 41-J temperature shifts were 72 and 82°C, while the adjusted DWT NDT temperature shifts were 70 and 83°C for welds 72W and 73W, respectively. Room-temperature yield strength increases were about 25 and 30% for 72W and 73W, respectively.

For fracture toughness tests, the scatter in both unirradiated and irradiated fracture toughness results using elastic-plastic fracture mechanics is large. Preliminary analyses of the data, that is, without statistical analyses, indicate that the lower-bound K_{Ic} curve shift (at 125 MPa·$\sqrt{m}$) for weld 73W is significantly greater than the CVN 41-J shift and the postirradiation K_{Ic} curve is of shallower slope than the curve for unirradiated material. The change in shape, however, is similar to the shape change shown by the CVN results. Final analyses of all results will be dependent on the interpretation of cleavage pop-ins. Additionally, various statistical analyses and empirical adjustment schemes are being investigated, all aimed toward the development of a rational scheme for constructing irradiated K_{Ic} curves.

6.2 Sixth Irradiation Series: Crack Arrest

S. K. Iskander   E. T. Manneschmidt
R. K. Nanstad   R. L. Swain

The primary objective of the Sixth Irradiation Series is to determine the effect of irradiation on the shift and shape of the K_{Ia} vs $(T - RT_{NDT})$ curve, where K_{Ia} is the plane-strain, crack-arrest fracture toughness, T is the temperature, and RT_{NDT} is the reference nil-ductility transition temperature. At present, the entire curve is shifted horizontally without change by an amount equal to the shift of the 41-J CVN impact energy level. As is well known, irradiation of RPV ferritic steels to fluences of the order of 2×10^{19} neutrons/cm^2 (>1 MeV) can cause changes in the CVN curve geometry. Thus, it is necessary to determine whether similar changes can occur in the K_{Ia} curve, particularly if such changes could lead to nonconservative determinations of the irradiated K_{Ia}. Although the data are still being evaluated, preliminary results[1] from the Fifth Series indicate that this is indeed the case, with the shift and shape of the K_{Ic} curve being similar to that of the CVN curve. The slope of the irradiated K_{Ic} curve is somewhat flatter than the unirradiated curve, similar to slope changes exhibited by CVN curves.

The materials used in the Sixth Series are identical to those used for the Fifth, namely two weldments with high copper levels of 0.23% and 0.31%, and designated 72W and 73W, respectively. The results of the Sixth Series will therefore complement the K_{Ic} results of the Fifth, using the same well-characterized material.

The specimen complement is given in Table 6.1. Two capsules, each containing 30 specimens of the two weldments have been irradiated at 288°C to a fluence of ~2×10^{19} neutrons/cm^2 (>1 MeV). Both capsules

Table 6.1. Compact crack-arrest specimen complement for each of the 72W and 73W weldments irradiated in the Sixth HSST Irradiation Series

Type	Dimensions (mm)	Quantity
Weld embrittled	25.4 × 76.2 × 76.2	8
Weld embrittled	24.4 × 152.4 × 152.4	7
Weld embrittled	33.0 × 152.4 × 152.4	3
Duplex	33.0 × 152.4 × 152.4	12

have been disassembled, the dosimeters retrieved and submitted for analysis, and the specimens transferred to storage. A preliminary dosimetric analysis has been performed, and a report will be prepared in FY 1989. Testing of the specimens is scheduled to commence in FY 1989.

Most of the unirradiated specimen controls have been tested, analyzed according to ASTM Test for Determining Plane-Strain Crack-Arrest Fracture Toughness, K_{Ia}, of Ferritic Steels (E1221-88), and reported.[2] More unirradiated control specimen testing needs to be performed but had to await the fabrication of additional weldments. A total of 3.6 m from each of the 72W and 73W weldments has now been fabricated and delivered to ORNL, with a letter report detailing the fabrication procedure. It is estimated that <1 m of each weldment will be required to complete the unirradiated testing, with the remainder to be held as archival material.

The highest valid value of K_{Ia} obtained during testing of unirradiated controls of both welds is ~125 $MPa \cdot \sqrt{m}$ at 30 and 50 K above NDT for 72W and 73W, respectively. The number of valid data points in this range is sparse, and more valid data at NDT + 40 K and above are needed. Experience has shown that in this temperature range, 50-mm-thick, duplex, crack-arrest specimens are more likely to yield test results that are valid according to E1221-88. Such specimens will be fabricated and tested. However, the production of successful 50-mm duplex specimens from weld material is a difficult process.[3] Although much progress has been made, the procedure still needs to be optimized.

The remotely operated fixture developed for use in testing crack-arrest specimens in the hot cell has been described previously.[4] Before its use in the hot-cell, it has undergone some testing in the laboratory[5] using specimens of the same size as the irradiated ones of the Sixth Series. The uniformity of temperature distribution in the test specimen for temperature ranges in excess of those envisioned to be used inside the hot cell has been evaluated. The accuracy of the fixture's temperature-indicating device, as compared with temperatures measured by thermocouples fixed to a specimen, has been verified. Moreover, this device has been used to perform some unirradiated K_{Ia} testing,[6] and the remote device has functioned without major problems, although some minor ones still need to be addressed.

6.3 Seventh HSST Irradiation Series

F. M. Haggag — T. N. Jones
R. K. Nanstad — E. T. Manneschmidt
R. L. Swain

6.3.1 Phase 1

In this phase, stainless steel cladding applied by the single-wire, oscillating, submerged-arc method was evaluated and most of the work reported earlier.[7] Results of the Charpy impact and tensile test specimens irradiated to 2×10^{19} neutrons/cm^2 (>1 MeV) at 288°C are given in Ref. 7. Eight irradiated 0.5TCS fracture toughness specimens, each from types 308 and 309 stainless steel cladding, were tested recently in the hot cell using the single-specimen unloading compliance technique. Also, unirradiated 0.5TCS specimens were precracked and then tested over the same test temperature range of the irradiated specimens, namely, from −75 to 288°C. Final crack length measurements are being made, and test data will be analyzed to determine the effect of irradiation on the initiation fracture toughness (J_{Ic}) and on the tearing modulus when applicable. Furthermore, six irradiated precracked CVN specimens, each from 308 and 309 stainless steel cladding, will be tested dynamically to obtain K_{Id} toughness data.

6.3.2 Phase 2

In the second phase, which is almost completed, a commercially produced, three-wire, series-arc cladding was evaluated under similar irradiation and testing conditions as in Phase 1. The three-wire, series-arc procedure, developed by Combustion Engineering, Inc., Chattanooga, Tennessee, produced highly controlled weld chemistry, microstructure, and fracture properties in all three layers of the weld. These three layers of cladding were required to provide enough thickness for the fabrication of tensile, CVN, and 0.5TCS from the cladding. The tensile and CVN specimens were irradiated to fluences of 2 and 5 × 10^{19} neutrons/cm^2 (>1 MeV), while the eight 0.5TCS were irradiated only to 2 × 10^{19} neutrons/cm^2 (>1 MeV). Test results of the tensile and CVN specimens were reported earlier in Ref. 8; however, the results are summarized in the following section. Preliminary results of the 0.5TCS are also discussed. Final analysis of all fracture toughness specimens will be discussed in the next progress report.

6.3.3 Results and discussion

Metallographic examination of the three-wire, stainless steel cladding showed a distribution of delta-ferrite in an austenitic matrix quite typical of microstructures seen in good practice, commercial-weld overlay cladding in RPVs.

6.3.3.1 Irradiation history. The specimens were irradiated in three capsules in the core of the 2-MW pool reactor at the Nuclear Science and Technology Facility in Buffalo, New York. The tensile and CVN specimens were irradiated in two capsules each containing 20 CVN and 6 miniature tensile specimens. The third capsule contained the 24 0.5TCS specimens of the two phases of the Seventh Irradiation Series. All capsules were instrumented with thermocouples and dosimeters, and the specimens resided in a mixed helium and air environment during the irradiation. Irradiation temperatures were maintained at 288 ± 11°C. Each capsule was rotated 180° at least once during its irradiation exposure for side-to-side fluence balancing. The average fluences for the three capsules were 2.14 ± 8%, 5.56 ± 5%, and 2.36 ± 6% × 10^{19} neutrons/cm^2 (>1 MeV), respectively. These fluences are for a calculated spectrum based on iron, niobium, and cobalt dosimeter wires.

6.3.3.2 Effect of irradiation on tensile properties. The yield strength of the three-wire, stainless steel cladding increased because of irradiation exposure. The effects were greater at room temperature and below. At the fluence level of 2×10^{19} neutrons/cm^2, the yield strength increased by 9, 20, and 28% at test temperatures of 288°C, room temperature, and −125°C, respectively. At the higher fluence level of 5×10^{19} neutrons/cm^2, the yield strength increased by 6, 16, and 34% at the test temperatures of 288°C, room temperature, and −125°C, respectively. Thus, increasing the fluence by a factor of 2.5 resulted in only small effects on yield strength. The effects of irradiation on the ultimate strength and ductility were very small.

6.3.3.3 Effect of irradiation on Charpy impact properties. Irradiation of the three-wire, series-arc, stainless steel cladding specimens at 288°C to fluence levels of 2 and 5×10^{19} neutrons/cm^2 (>1 MeV) decreased the CVN upper-shelf energy (USE) by 15 and 20% and increased the 41-J transition temperature by 13 and 28°C, respectively. Irradiation also degraded the CVN lateral expansion significantly; the upper shelf was reduced by ~40% for both fluence levels. Furthermore, the 0.38-mm (0.015-in.) transition temperature shifts were 41 and 46°C for the low and high fluences, respectively. These results generally agree with those for the single-wire cladding produced with good weld practice.[7]

6.3.3.4 Effect of irradiation on fracture toughness of three-wire cladding specimens. Preliminary analysis of the 0.5TCS fracture toughness specimens fabricated from three-wire, series-arc stainless steel cladding indicated that irradiation exposure to an average fluence of 2.41×10^{19} neutrons/cm^2 (>1 MeV) resulted in a consistent decrease in the initiation fracture toughness (J_{Ic} or J_Q) at test temperatures of −75°C, room temperature, 120°C, and 288°C. This is in agreement with the reduction in both the CVN USE and lateral expansion discussed above. However, the percent reduction in initiation toughness of the 0.5TCS is greater than that of the CVN impact energy but closer to the percent reduction of the CVN lateral expansion. Irradiation effects on the tearing modulus will be reported later following the final analysis of all fracture toughness specimens. Both unirradiated (control) and irradiated specimens exhibited the same trend of slightly increased initiation toughness from −75°C to room temperature followed by a significant toughness decrease from room temperature to 288°C.

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7. CLADDING EVALUATIONS

7.1 Objective

The objective of Task H.7 is to demonstrate the effect of stainless steel weld cladding on the resistance to and extent of crack propagation for small surface cracks subjected to stress gradients similar to that produced by a thermal shock. The occurrence, size, and distribution of such small cracks in representative segments from actual boiling-water reactor (BWR) and pressurized-water reactor (PWR) pressure vessels is also being studied to broaden the data base used in probabilistic studies.

7.2 Fracture Mechanics Testing of Clad Plates

S. K. Iskander B. C. Oland
G. C. Robinson D. J. Alexander
K. V. Cook

7.2.1 Background

A small crack near the inner surface of clad nuclear reactor pressure vessels (RPVs) is an important consideration in the safety assessment of structural integrity. The behavior of such flaws is relevant to the pressurized-thermal-shock (PTS) scenario and to the plant life extension issue. There is a dearth of information on the behavior of small flaws in the presence of cladding. Considerable experimental results have shown that, in the absence of cladding, a small surface flaw in an embrittled material subjected to severe thermal shock will become a long flaw. Thus, some RPV integrity studies[1] have assumed infinitely long flaws (although small flaws are certainly more credible). The question remains about the role that a tough surface cladding will play in preventing the propagation of small flaws along the surface. Furthermore, the flaw could tunnel beneath the cladding, in which case the residual strength of the structure needs to be determined.

The first series (Series 1) of such experiments indicated that the cladding employed may have had sufficient arrest toughness to stop running cracks.[2] The bounding of the contribution of the stainless steel cladding on the arrest toughness was not realized.

7.2.2 Accomplishments to date

A special plate specimen (900 mm long by 400 mm wide and 50 mm thick) has been developed to investigate the cladding effects. Six clad and two unclad plate specimens have been tested in four-point bending to simulate steep stress gradients that arise during PTS accident scenarios. A site for a sharp flaw to initiate is introduced into the base metal by means of an autogenous electron-beam (EB) weld. It is approximately elliptic in shape, about 70 mm long along the surface, and 15 mm deep. At the time a flaw is required, the EB weld is hydrogen charged.

Six plates were commercially clad using the three-wire, series-arc technique (used on some of the older RPVs) and stainless steel type 308, 309, and 312 weld wires. The base metal on which the cladding was applied, an ASTM A 533 grade B chemistry steel, was heat-treated to raise its transition temperature so that it would be brittle at temperatures at which the cladding is tough. Two similar plates were left unclad to provide comparison data.

Two different testing procedures were followed. In one of those, used with one unclad and five clad plates, the test specimen was loaded to a specific target surface strain, and while the load was maintained constant, it was hydrogen-charged to cause crack initiation. The effect of the different surface strains is to store varying levels of energy in the plate. A flaw initiates in the EB weld site and propagates until it either arrests when it encounters the clad region or leads to complete rupture of the plate. The purpose of this test is to provide information about the load-arresting capacity of cladding. Plates that do not rupture are heat-tinted to define the arrested flaw shape and are then reloaded until either another pop-in or plate rupture occurs. The purpose of the reloading is to provide data on the load-carrying capacity of plates with large flaws. In the second procedure, the flaw was initiated beforehand in one clad and an identical unclad plate. They were then loaded to provide an indication of the differences in the initiation load-carrying capacity of flawed, clad, and unclad plates.

7.2.3 Results

Results from testing at room temperature have shown that (1) a tough surface layer composed of cladding and/or heat-affected zone (HAZ) has arrested running flaws under conditions where unclad plates have ruptured, and (2) the residual load-bearing capacity of the clad plates with large subclad flaws significantly exceeded that of an unclad plate.

The results of considerable material characterization studies have shown that the HAZ associated with the cladding has a higher toughness (in the temperature range in which the plates were tested) than either base metal or cladding. The toughness was measured by Charpy V-notch impact energy. Thus the HAZ may have played a dominant role in the enhanced load-carrying capacity of the plates.

An interim report, "Effect of Stainless Steel Weld Overlay Cladding on the Structural Integrity of Flawed Steel Plates in Bending — Series 2," by S. K. Iskander et al., has been written and is under review.

7.2.4 Further work recommendations

It is not clear at this time whether it was the cladding or HAZ that had the dominant role in increasing the load bearing capacity of the clad plates. Furthermore, it is assumed that neutron embrittlement would degrade the HAZ. More work is necessary to clarify the effect of cladding alone on the structural integrity of an operating RPV. A relatively inexpensive test to determine what role the HAZ alone may have played in arresting a running flaw could be performed on available broken halves of previously tested plates. The cladding would first be machined off and the pieces would be EB welded together to form a test specimen of approximately the same dimensions as those of the plates previously tested. The test procedure used in testing the clad plates would be followed to determine the load-carrying capacity of the plate with HAZ only.

Moreover, to transfer the results of laboratory testing of the clad plates to RPVs, the necessary analytical tools will need to be developed because available state-of-the-art techniques may not be applicable. The large plastic strains and displacements, as well as the lack of contained plasticity occurring under load in the thin layer of cladding, do not fulfill the assumptions of elastic-plastic fracture mechanics.

Small-scale testing of "Jo-block specimens" could provide a means of gaining insight into methods that will allow the transfer of the results from the clad plate tests to the analysis of the structural integrity of an RPV at reasonable cost. A possible approach would be to use a hypothetical stress-strain curve of cladding averaged out over some finite length opposite the "flaw" of Jo-block specimens. As a first step, such a curve would be used as "material properties" of a cladding region (corresponding in geometry to that from which it was obtained) in a large strain/displacement finite-element analysis of the structural integrity of the clad plates with the various flaws encountered during the testing of these plates. If such an approach could be verified, it could be applied to RPVs using hypothetical flaws and transient loads.

7.3 Flaw Characterization Studies of Clad LWR Vessel Material

K. V. Cook R. W. McClung
R. A. Cunningham, Jr.

Nondestructive testing tasks on segments of light-water reactor (LWR) vessels were completed. A paper[3] was published in *Nuclear Engineering and Design*.

A topical report, "Detection and Characterization of Indications in Segments of Reactor Pressure Vessels," by K. V. Cook, R. A. Cunningham, Jr., and R. W. McClung, has been prepared and reviewed and will be published soon. This report describes the studies that have been conducted to estimate flaw density in segments cut from LWR pressure vessels as part of the Oak Ridge National Laboratory's Heavy-Section Steel Technology (HSST) Program. Objectives were to evaluate these LWR segments for flaws with ultrasonic and liquid-penetrant techniques. Both objectives were successfully completed on segments from both a BWR and a PWR design.

The topical report describes in detail one significant indication detected in a BWR seam weld by ultrasonic techniques and characterized by further analyses terminating with destructive correlation.

Also described in the topical report are results from the PWR segments. The PWR segments contained relatively little weldment; thus, the ultrasonic examination was limited to the cladding and subcladding regions. Only one indication of note was detected ultrasonically at the clad-to-base metal interface.

Fluorescent liquid-penetrant inspection of the cladding surfaces for both LWR segments detected no significant indications [i.e., for a total of ~6.8 m^2 (72 ft^2) of cladding surface].

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8. INTERMEDIATE VESSEL TESTS AND ANALYSIS

No activity in the intermediate vessel tests and analysis task during this period.

9. THERMAL-SHOCK TECHNOLOGY

No activity in the thermal-shock technology task during this period.

10. PRESSURIZED-THERMAL-SHOCK TECHNOLOGY

R. H. Bryan

10.1 Summary

An investigation of the potential utility and feasibility of additional pressurized-thermal-shock experiments was initiated during the previous report period at the request of the Nuclear Regulatory Commission (NRC). The investigation, which has now been completed, involved the consideration of two distinct classes of fracture problems: the effects of stainless steel cladding on crack behavior and the behavior of cracks in low-upper-shelf welds. Results and conclusions were presented to the NRC in August 1988, and it was then recommended that two additional experiments be performed: PTSE-3 to confirm analytical predictions of the effects of cladding and PTSE-4 to clarify mechanisms of tearing of low-upper-shelf welds.

10.2 An Experiment on Effects of Cladding on Crack Propagation

During the conception of the pressurized-thermal-shock experimental program, effects of cladding on crack propagation were recognized as an uncertain factor in vessel integrity evaluations with the possibility that conclusions drawn from overcooling accident analyses could be made better or worse. Accordingly, in 1981 cladding investigations involving clad beam tests, thermal-shock experiments, irradiation-effects experiments, and a pressurized-thermal-shock experiment were considered as potentially useful elements of a research plan. Clad plate tests[1,2] and irradiation studies[3] were initiated in 1981 and 1982, respectively, as the preliminary steps of the plan. The clad plate tests were devised as a simple means of determining whether cladding could serve as a potential deterrent to longitudinal extension of short surface cracks. The objective of the irradiation studies was to measure the effects of irradiation on cladding toughness as a contribution to resolving questions about whether the properties of irradiated cladding could promote or inhibit crack propagation.

The clad plate tests have been completed and showed that the cladding had some capacity for limiting crack propagation.[4–6] The respective contributions of fracture toughness and ductility to cladding behavior in these tests are not yet well-known.

Early studies of irradiation effects of austenitic stainless steels showed that toughness can be reduced by fast neutron fluence levels that are pertinent to pressurized-water-reactor pressure vessels.[7,8] More recent studies were made specifically of the effects of irradiation on cladding. In the HSST program seventh irradiation series,[3] cladding produced by an automated single-wire, submerged-arc welding process indicated that very little degradation in toughness would be expected in good quality cladding, but toughness was appreciably reduced in highly diluted cladding.[9] In the second phase of the seventh irradiation series, three-wire, series-arc cladding is being investigated. Irradiation of this cladding, which is typical of some commercial light-water reactor materials, resulted in a decrease in toughness, an increase in yield strength, and no substantial change in ultimate tensile strength and ductility.[10]

With the conclusion of these preliminary cladding studies, the principal cladding issue of 1981 remains unresolved. It has not yet been determined whether the presumption of an infinitely long flaw is always more conservative than consideration of a short flaw.

In one comparison of these alternative assumptions Yang and Bamford analyzed a surface crack in a clad vessel under pressurized-thermal-shock loading in which the variation in flaw shape was taken into account.[11] They concluded that, for a particular loading sequence, accounting for shape changes leads to more favorable consequences than are predicted by NRC-accepted procedures[12] and that, including the effects of cladding, it gives slightly more favorable results.

It is now understood that some loading sequences could produce more severe consequences if one starts with a short crack, even though the initial effect may be to inhibit crack propagation. Therefore, the next logical step should include establishment of a credible experimental basis for verifying analytical methods that now exist or may be developed for evaluating propagation of short cracks in cladding.

To support the investigation of the utility and feasibility of a clad-vessel experiment, modifications to the OCA/USA computer program were made, and the SHAPE program was developed.[13] The SHAPE program uses stress-intensity factors calculated by OCA/USA for a pressurized-thermal-shock transient to generate the modes of crack shape changes that are attainable. Three-dimensional, finite-element analyses were performed to generate the sets of influence coefficients needed for this task.[14,15]

Potential experimental configurations for a clad vessel test (PTSE-3) have been explored with OCA/USA and SHAPE analyses. Cladding parameters and initial crack geometries that are compatible with the capabilities of the pressurized-thermal-shock test facility[16] have been identified. The recommended experiment has an initially short axial surface crack that penetrates the cladding layer. In the transient experiment the crack will be made to propagate depthwise and then axially in a predictable succession of shape changes. Tunneling beneath the cladding will occur during the latter phase of propagation. The crack will finally arrest within the test section of the vessel.

The vessel for this experiment has already been fabricated, except for cladding. This vessel (originally ITV-7) and the vessel for PTSE-1 (Ref. 16) were prepared by the Babcock & Wilcox Company with identical welded-in inserts of specially tempered material.[17] The base material for the clad vessel test is fully characterized by small-specimen tests and by the PTSE-1 experiment itself. Therefore, more pertinent toughness data are available for planning PTSE-3 than for planning any previous thick-vessel fracture test. In the current PTSE-3 plan, cladding with properties representative of irradiated cladding will be applied to the area where the flaw will be located. It is expected that the influence of the cladding will depend more on its strength and ductility than on intrinsic toughness. The experiment will be designed to reveal the dependence of crack behavior on these cladding characteristics.

10.3 A Proposed Experiment with a Low-Tearing-Resistance Weld

The second pressurized-thermal-shock experiment, PTSE-2 (Ref. 18), was concerned with behavior of a crack in material with low tearing resistance, which is one aspect of the NRC pressurized-thermal-shock issue.[12] The principal objective of the PTSE-2 experiment was to observe the transitional behavior of steel with low tearing resistance to elucidate the interplay of cleavage and ductile modes of fracture. The experiment produced four phases of stable tearing and two of unstable tearing.[18] It is obvious from the results that the PTSE-2 material had both cleavage and tearing resistances much higher than the limiting value of 220 $MPA \cdot \sqrt{m}$ implied by Sect. XI of the *ASME Boiler and Pressure Vessel Code*[19] and used in developing the NRC pressurized-thermal-shock rule.[20,21] This conclusion confirmed that the limiting value may be quite conservative. However, the incremental depths of stable tearing observed in both PTSE-1 (Ref. 16) and PTSE-2 were inconsistent with calculations based on small-specimen, J-resistance tests and deformation-plasticity analysis.

Several factors may contribute to this inconsistency. Two important possibilities are that a deformation-plasticity model may not be adequate, particularly for some phases of PTSE-2 or that structural and material characteristics more complex than a J_I-resistance curve relationship may govern tearing behavior.

Confirmation of a reliable means of predicting the amount of stable tearing is important, because this type of tearing could in some situations inhibit cleavage crack progagation and in others promote propagation. Furthermore, the influence of warm prestressing on initiation of cleavage fracture is very sensitive to the extent of stable tearing.

The experiment (PTSE-4) proposed for clarifying this problem is similar to PTSE-2. However, the initial crack in PTSE-4 will be placed in an insert of low-tearing-resistance weld metal. Certain types of welds were noted by the NRC as having relatively high sensitivity to radiation damage in terms of shifting the brittle-ductile transition temperature and reducing the Charpy V-notch upper-shelf energy.[22] Thus the PTSE-4 test will examine flaw behavior in the type of material of greatest concern among reactor pressure vessel steels.

To perform the PTSE-4 experiment, the PTSE-2 vessel must be repaired with a long insert of weld metal with appropriate properties. Trial weldments have been fabricated with a 2-1/4 Cr-1 Mo weld wire with suitable composition. Tensile and impact properties of the welds are being determined for a range of tempering treatments so that optimal procedures for vessel preparation can be specified.

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11. PRESSURE VESSEL RESEARCH USERS' FACILITY

11.1 Facility Development

W. E. Pennell   C. E. Pugh

Industry experience with operating pressurized-water reactor (PWR) pressure vessels suggests that long-term research and development (R&D) programs will be required to ensure the safe and efficient operation of these critical components. Support needs fall into the broad categories of (1) plant life extension, (2) in-service inspection, and (3) maintenance and repair. Recognition of this need prompted a Nuclear Regulatory Commission (NRC)-supported Oak Ridge National Laboratory (ORNL) initiative to create a Pressure-Vessel-Research Users' Facility (PVRUF), with a broad capability to meet both the near-term and long-term nuclear industry needs for PWR pressure vessel R&D support.

The basic approach adopted in development of the PVRUF concept was to focus on those PWR pressure vessel R&D needs that require the use of a full-scale prototypic test article to generate high-quality, directly usable program output. Primary elements of this approach to defining the PVRUF Facility concept are summarized in Fig. 11.1. The essential step

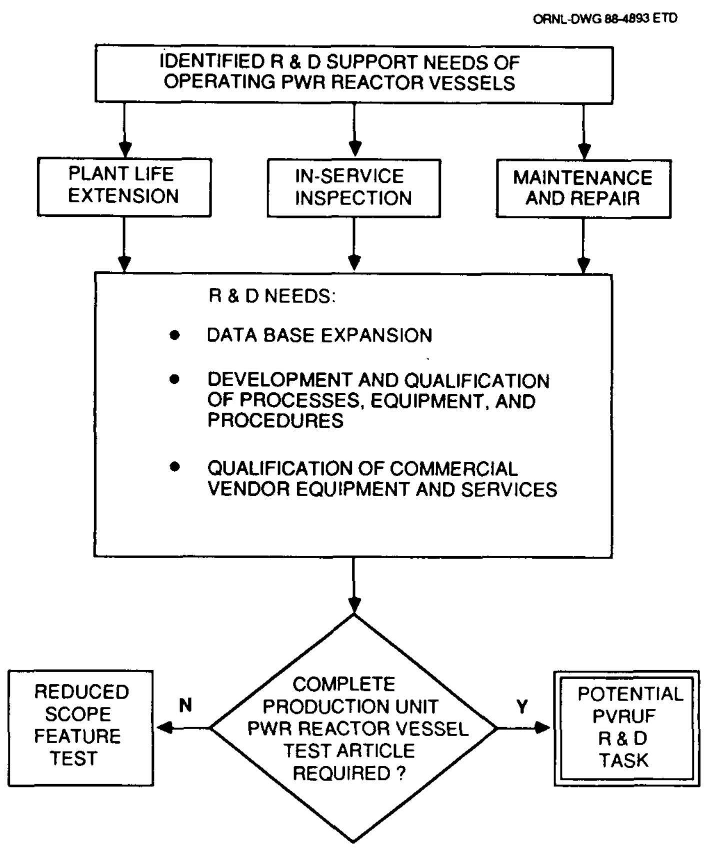

Fig. 11.1. PVRUF concept.

of procuring a production unit PWR pressure vessel for use in the facility was taken in 1987 and has been previously reported.

Initial development of the program plan for PVRUF is proceeding and following the logic defined in Fig. 11.2. Essential elements of this logic are (1) the use of a mission survey to determine the high-priority PVRUF R&D tasks; (2) development of functional requirements, conceptual designs, and cost and schedule estimates for PVRUF based on the mission survey results; and (3) preparation of a proposal and sponsor interaction leading to a definition of the time-phased funding commitment for the project.

Actions on the initial phase of the PVRUF mission survey are well advanced. This phase of the survey used appropriate segments of the ORNL technical community to evaluate and rate potential PVRUF R&D tasks. Fourteen R&D tasks were included in the review package. Each of the candidate R&D tasks was defined by a brief written statement of its scope and end product, together with an identification of any feasibility issues. Reviewers were asked to rate those candidate R&D tasks that fell within their area of expertise in each of five rating categories. The

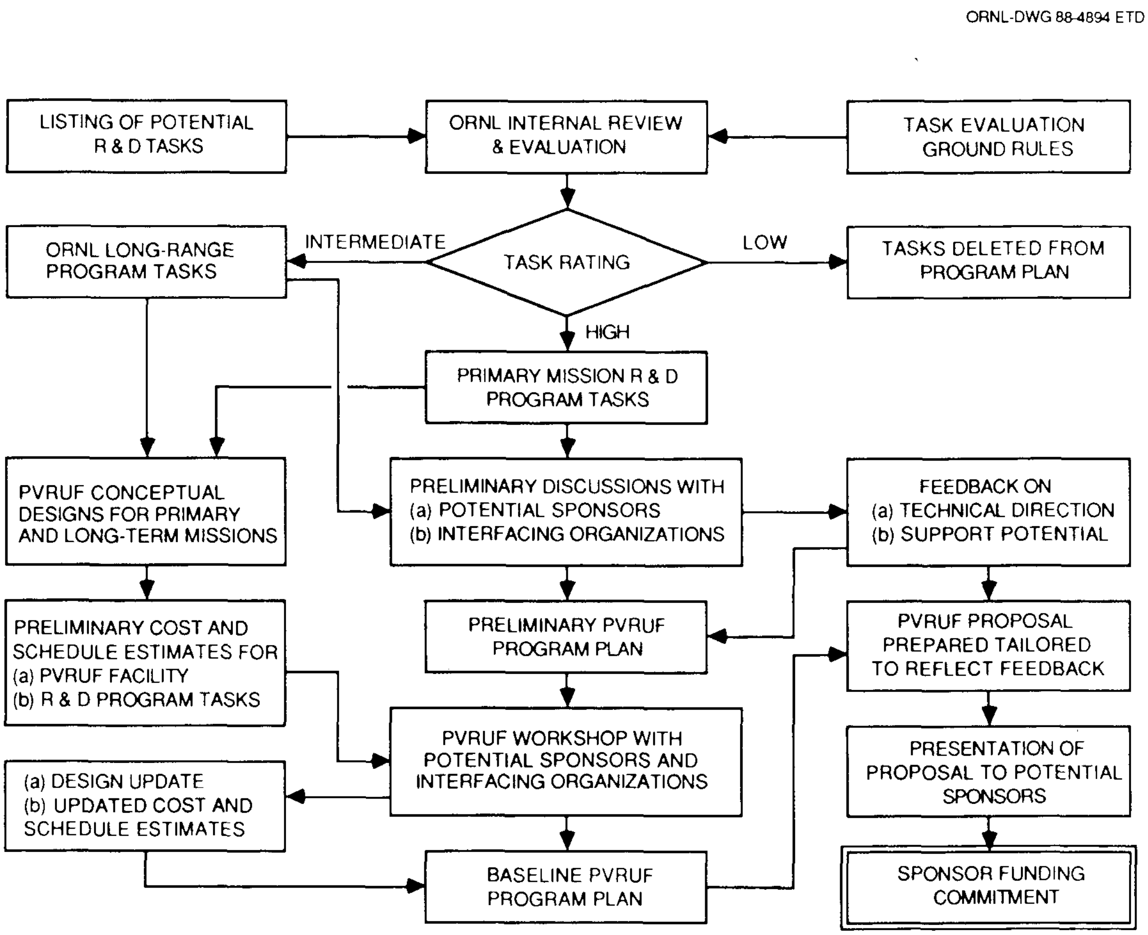

Fig. 11.2. PVRUF initial program development logic.

most important of the task rating categories were (1) the need for the output from the candidate R&D task and (2) the need for PVRUF to generate the required R&D task output.

Response to the PVRUF mission survey has been good; 75% of the anticipated responses has now been received. Spreadsheets have been prepared for a statistical analysis of the numerical R&D task rating data.

Progress on the PVRUF program development task has been substantially reduced during the current report period because of large-scale diversion of PVRUF program personnel to work on the higher priority evaluation of the flaw tolerance of irradiated reactor vessel supports.

11.2 Flaw Studies for PVRUF

K. V. Cook R. W. McClung
R. A. Cunningham, Jr.

11.2.1 Nondestructive testing introduction

As reported in the previous semiannual,[1] a plan was formulated for the nondestructive evaluation (NDE) of the PVRUF vessel for flaw density. An updated draft plan for the proposed NDE activities on PVRUF was prepared using comments from Pacific Northwest Laboratory (PNL) personnel (S. Doctor) and from the program for inspection of steel components (PISC) personnel (S. Crutzen). Phase 1 studies completed during this period were (1) familiarization of personnel with drawings and inspection data as available; (2) preparation of vessel for internal inspection (i.e., craft work directed by Engineering Technology personnel); and (3) initiation of the NDE of the cylindrical portion of the PVRUF vessel.

11.2.2 Accomplishments and plans

A plan was written to guide NDE activities on PVRUF. Ultrasonic contact angle-beam search units (transducers) were procured for initial manual inspection of zones contained in the cylindrical section of the vessel. The protective coating on the inner (clad) surface was removed from portions of the area in preparation for nondestructive inspection of the cladding for flaws. Figure 11.3 shows the inside of the vessel after the partial removal of coating. Note the mobile metal platform provided as a working surface for the NDE efforts. This moveable platform was designed, fabricated, and installed by Energy Systems personnel. Also note that the uncovered surface (exposed clad) is on the upper-half of the vessel's circumference. The coating removal begins near the nozzles and extends along the cylindrical length. Although not shown in Fig. 11.3, the removal ends very near support lugs attached at the bottom (dome section) of the vessel.

A small area near the support lug position was used to demonstrate that a fluorescent penetrant inspection could be performed successfully. This demonstration also established procedures that can be used for penetrant examination. Preparation was begun on detailed written quality assurance (QA) plans.

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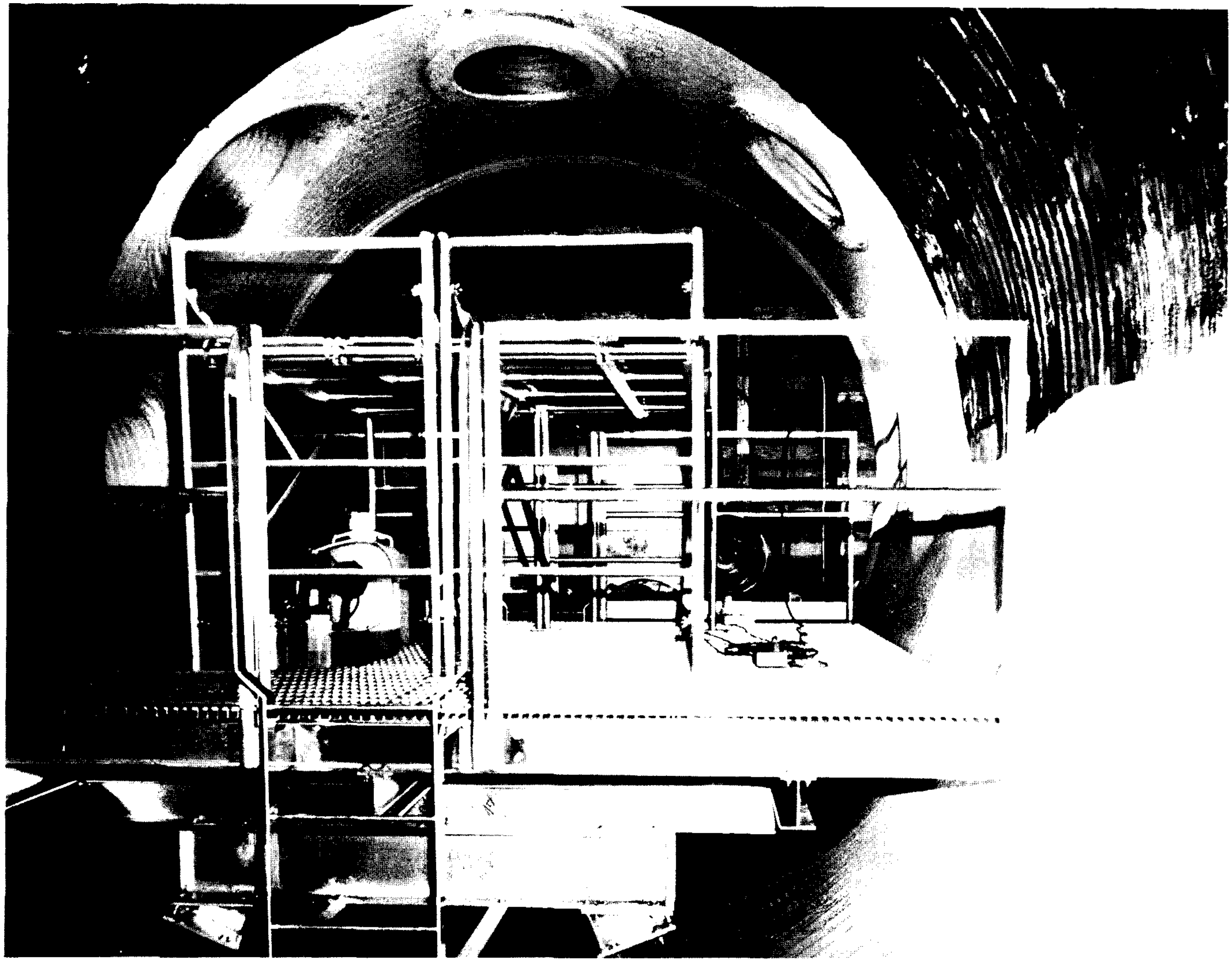

Fig. 11.3. Inside view of PVRUF vessel showing mobile work platform from bottom of vessel. The stainless clad surface is also evident on upper-half of vessel's circumference because protective coating has been removed from this area.

Reference

1. R. W. McClung, R. A. Cunningham, Jr., and K. V. Cook, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., "Nondestructive Testing Plans," pp. 246–47 in *Heavy-Section Steel Technology Program Semiann. Prog. Rep. October 1987–March 1988*, NUREG/CR-4219, Vol. 5, No. 1 (ORNL/TM-9593/V5&N1).

12. SHIPPING CASK MATERIAL EVALUATIONS

No activity in the shipping cask material evaluations task during this period.

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
kN	lb_f	224.809
kPa	psi	0.145038
MPa	ksi	0.145038
$MPa \cdot \sqrt{m}$	$ksi \cdot \sqrt{in.}$	0.910048
J	ft·lb	0.737562
K	°F or °R	1.8
kJ/m^2	$in.\text{-}lb/in.^2$	5.71015
$W \cdot m^{-2} \cdot K^{-1}$	$Btu/h \cdot ft^2 \cdot °F$	0.176110
kg	lb	2.20462
kg/m^3	$lb/in.^3$	3.61273×10^{-5}
mm/N	$in./lb_f$	0.175127
$T(°F) = 1.8\ T(°C) + 32$		

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

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13. ABSTRACT (200 words or less)

The Heavy-Section Steel Technology (HSST) Program is an engineering research activity conducted by the Oak Ridge National Laboratory for the Nuclear Regulatory Commission. The Program comprises studies related to all areas of the technology of materials fabricated into thick-section primary-coolant containment systems of light-water-cooled nuclear power reactors. The investigation focuses on the behavior and structural integrity of steel pressure vessels containing cracklike flaws. Current work is organized into twelve tasks: (1) program management, (2) fracture methodology and analysis, (3) material characterization and properties, (4) special technical assistance, (5) crack-arrest technology, (6) irradiation effects studies, (7) cladding evaluations, (8) intermediate vessel tests and analysis, (9) thermal-shock technology, (10) pressurized thermal-shock technology, (11) Pressure Vessel Research Users' Facility, and (12) shipping-cask material evaluations.

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Cladding
Flaws
Thermal shock
Fracture mechanics
Crack arrest
Crack growth

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