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MARTIN MARIETTA

NUREG/CR-4275
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Heavy-Section Steel Technology Program—Five-Year Plan FY 1984–1988

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

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HEAVY-SECTION STEEL TECHNOLOGY PROGRAM — FIVE-YEAR PLAN FY 1984—1988

Prepared by the Staff of the
Heavy-Section Steel Technology Program
Oak Ridge National Laboratory

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Oak Ridge, Tennessee 37831
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CONTENTS

LIST OF FIGURES

LIST OF TABLES

FOREWORD

This report is the second in a series of annual five-year program plan documents for the HSST program. Prior to this series, one early comprehensive program plan document had been prepared. During the intervening years, annual budgetary and management documents were issued which stated the current program. The three multi-year program plan documents issued thus far are listed below.

1. G. D. Whitman et al., *Program Plan — The Heavy-Section Steel Technology Progra‑*, Rev. 0, April 1, 1968, Rev. 1, February 27, 1970 (unnumbered document prepared for the USAEC).

2. HSST staff, *Heavy-Section Steel Tachnology Program — Five-Year Plan, FY 1983–1987*, NUREG/CR-3595 (ORNL/TM-9008), Oak Ridge National Laboratory, Oak Ridge, Tenn. (April 1984).

3. HSST staff, *Heavy-Section Steel Technology Program — Five-Year Plan, FY 1984–1988*, NUREG/CR-4275 (ORNL/TM-9654), Oak Ridge National Laboratory, Oak Ridge, Tenn. (June 1985).

ABSTRACT

This is the second in the current annual series of five-year program plan documents presented for the Heavy-Section Steel Technology program. The program is carried out by the Oak Ridge National Laboratory for the Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research of the U. S. Nuclear Regulatory Commission. The program is aimed at advancing the understanding and validation of materials and structures behavior as they relate to light water reactor pressure vessel integrity. The program has nine technical tasks and a management function. A background statement and a plan-of-action is given for each. Lists of reports published by the program are included as part of the background information. The nine technical tasks address fracture methodology and analysis, materials characterization, crack growth, crack arrest, irradiation effects, cladding evaluations, intermediate-vessel testing, thermal-shock testing, and pressurized thermal-shock experiments.

1. INTRODUCTION AND OVERVIEW

1.1 Objective of Program Plan Document

This document is the second in a series that provides an up-to-date statement of the five-year plan for the Heavy-Section Steel Technology Program. It is intended to be the reference document for management reporting during the forthcoming year. The forthcoming year for this edition is FY 1985. The program funding assumptions and the corresponding workscopes are responsive to guidance from the sponsor, the Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). This document augments other planning instruments, such as the Project and Budget Proposal for NRC work (189) in two important ways. First, milestone charts and plan-of-action statements can be presented here in detail without concern for space limitations. Second, this document can purposefully be drafted at the beginning of the fiscal year, and thereby provide a more accurate (timely) statement of the program plan for the forthcoming year than is possible with the schedule for some other documents, such as the 189.

This plan is a living document and is expected to be updated at the beginning of each year. The five years covered by the (current) plan are the past year (FY 1984), the forthcoming year (FY 1985), and three out-years (FY 1986 through FY 1988). Since a major objective is to provide a reference document for management reporting during the forthcoming year, that year is highlighted in the milestone charts and is emphasized in the plan-of-action sections. The subsequent years are described in a decreasing level of detail without loss of comprehensive coverage. A few identifiable events beyond FY 1988 are included for continuity. Also for continuity, the past year is included in detail, and the milestones that have been completed at the time of this writing are shown by filled symbols.

A complete draft of this edition of the plan was initially submitted to the NRC in December 1984, responsive to their guidance and technical goals. The intervening time was required for publication approval to be granted and for publishing other high priority program reports. Therefore, the contents of the plan are as they were formulated in December 1984, and do not reflect changes that may have occurred since that time.

As part of program background, the reports that have been published since the inception of the program are listed in Appendices A through E. The reports and appendices are categorized according to (A) technical reports, (B) technical manuscript, (C) fabrication reports, (D) progress reports, and (E) program plan documents. These lists are updated continuously and the appendices in this report are through April 1985.

1.2 Objective of the HSST Program

The Heavy-Section Steel Technology Program (HSST) is carried out to advance the understanding and validation of materials and structures behavior as they relate to light water reactor pressure vessel integrity. The program had its beginning in the mid 1960's and has contributed to verifying the applicability of fracture mechanics to vessel integrity assessments, to advancing associated analysis tools, to data generation and correlations development, and to code criteria and rule development. The studies address the determination of the effects of flaws, variations in properties, stress raisers, and residual stresses on the integrity of vessels under combined thermal and mechanical loadings.

The program contains nine technical tasks in fracture methodology and analysis, materials characterization, crack growth, crack arrest, irradiation effects, cladding evaluations, intermediate-vessel testing, thermal-shock testing, and pressurized thermal-shock experiments. The objectives of the various tasks combine to bear on the major issues relating to vessel integrity by providing improved techniques, data bases, and bases of validation. This interrelationship is illustrated by the situation of pressurized thermal-shock loadings where current advancements on fracture methodology and data bases combine with results from previous material and pressure vessel studies to give an assessment for conditions under combined thermal and pressure loadings.

The program budget has increased in recent years to accommodate more complex experiments and increased irradiation work. Concerning complex experiments, a test facility has been prepared for performing tests of pressurized intermediate vessels that are exposed to sharp thermal transients. The first such experiment (PTSE-1) was performed in early FY 1984, and a series of two additional tests is planned.

There is one remaining intermediate vessel for test under pressure-only loading (ITV-10), and it involves a nozzle-to-cylinder configuration and is tentatively scheduled for late FY 1987. Thermal-shock test TSE-7 was completed in FY 1983 and involved a study of finite-length flaw behavior. Further thermal-shock tests are planned to involve experiments with clad specimens. These are to be performed only after further results are available on the behavior of stainless steel cladding in irradiated conditions, and after the NRC provides corresponsing approval.

The major focus of the current irradiation work is on understanding the fracture characteristics of irradiated weld materials. The Fourth Irradiation Series includes four state-of-the-art weld materials that contain less than 0.12% copper. The Fifth Irradiation Series includes two high copper weld materials (Cu = 0.25 and 0.35%) and the Seventh Irradiation Series will be for stainless steel cladding.

1.3 Program Organization

A well-coordinated program has been in effect for more than fifteen years, and it has been instrumental in bringing the national technological base to its current status. The base includes national design codes (e.g., the ASME Boiler and Pressure Vessel Code), standards (e.g., ASTM Standards and USNRC Regulatory Guides), analysis methods, material properties data, and confirmatory structural behavior data. While the program has been mostly carried out at the Oak Ridge National Laboratory (ORNL), it has made use of subcontracted activities to take advantage of special expertise at other locations. The program has also been coordinated with other R and D programs sponsored by the NRC, DOE, EPRI, etc. National and international committee involvements have been maintained to help assure that existing technologies and data are employed and that developments are rationally sound and experimentally verified. In particular, the Pressure Vessel Research Committee (PVRC) and the NRC-appointed Vessel Integrity Review Group (VIRG) have played very important roles.

The program is presently composed of nine technical tasks and a management function. The organization of these tasks is illustrated in the level 2 work breakdown structure for the program shown in Fig. 1.1. The tasks interface with each other and are augmented by documentation and technology transfer efforts. The remainder of this document is organized around this structure with a chapter addressing each technical task. The management task is addressed in the latter part of this chapter. A work breakdown structure has been developed for each task, and they are displayed in the corresponding chapters of this document. It is to be noted that those displays are actually work breakdown structures through level 3, with the fourth level (unnumbered) elements developed to illustrate the scope of work involved. In order to maintain a manageable level of detail, the milestones in each task are written under the level 3 elements of the work breakdown structure and not level 4. Each chapter also includes a brief objective statement for that task, a background discussion, and a plan-of-action section that augment the milestone charts.

1.4 HSST Task H.1 Program Management

1.4.1 Objective

The objective of Task H.1 is to effectively manage the technical tasks undertaken to address priority issues concerning LWR pressure vessel integrity in keeping with NRC approved plans. Management includes planning, integrating, monitoring, reporting, and technology transfer activities.

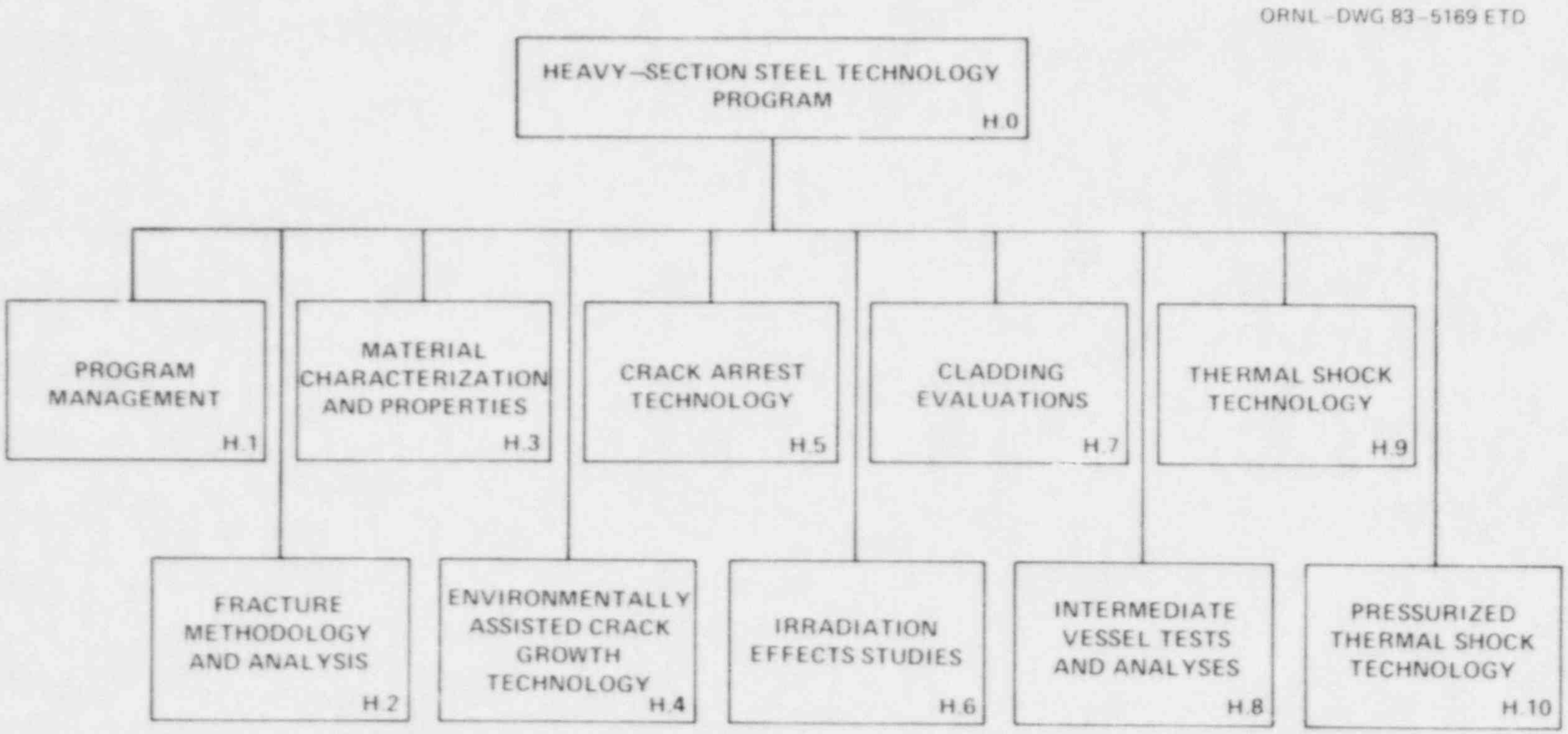

Fig. 1.1 Level 2 work breakdown structure for the Heavy-Section Steel Technology (HSST) program.

1.4.2 Background

The HSST program has been administratively carried out through the Engineering Technology Division of ORNL since its inception in the mid 1960s. A key part of that management function has been the integration of the technical objectives and the efforts of various program participants. The participants have included various divisions of ORNL, but most prominently the Engineering Technology and Metals and Ceramics Divisions, and several universities and industrial organizations through subcontracts. The aim has been and is to utilize capabilities and facilities in a complementing way to meet program objectives. The placement of the program within the ORNL organization is illustrated in Fig. 1.2, which also lists key staff members.

Concerning technical integration, a strong relationship has been maintained with peer groups and committees. These have included the Pressure Vessel Research Committee, ASME Code committees, ASTM Standard groups, and peer advisory groups. Their inputs on problems, priorities, and approaches have been factored into earlier program plans.*[1] In the most recent years, the NRC appointed Vessel Integrity Review Group (VIRG) has served a vital role in this regard, whereas in earlier years, the PVRC was similarly involved. A comprehensive review[2] was conducted in the early stages of the program covering the current practice in RPV design, analysis, materials, fabrication, inspection, and testing. This review and the philosophy that it set in place have assisted in program integration. Additionally such reviews have provided the community with interpretive statements of the state-of-the-art.

Program developments have been transferred to the technical community through progress and topical reports, program review meetings, information meetings, open-literature papers, and committee participation. Technical interactions with foreign countries have been strong, particularly with the European countries. Those countries have acknowledged the leading role that the HSST program has played in developing and verifying methods for assessing pressure vessel integrity.

1.4.3 Plan of Action

The plan for Task H.1 is to continue to manage the program through approved procedures. The level 3 work breakdown structure for this task is shown in Fig. 1.3. The three subtasks are continuations of the management functions that have historically been performed. The specific activities and schedules within the task are given in the following milestone chart. In addition to planning, monitoring, and reporting on the program performance in a timely manner, efforts will continue to be made to maintain effective technology transfer and to continue liaison with peer groups, committees, and programs in foreign countries.

This task will administer the research and development subcontracts and technical consulting arrangements that are required to supplement

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*References are listed in the end of each chapter of this document.

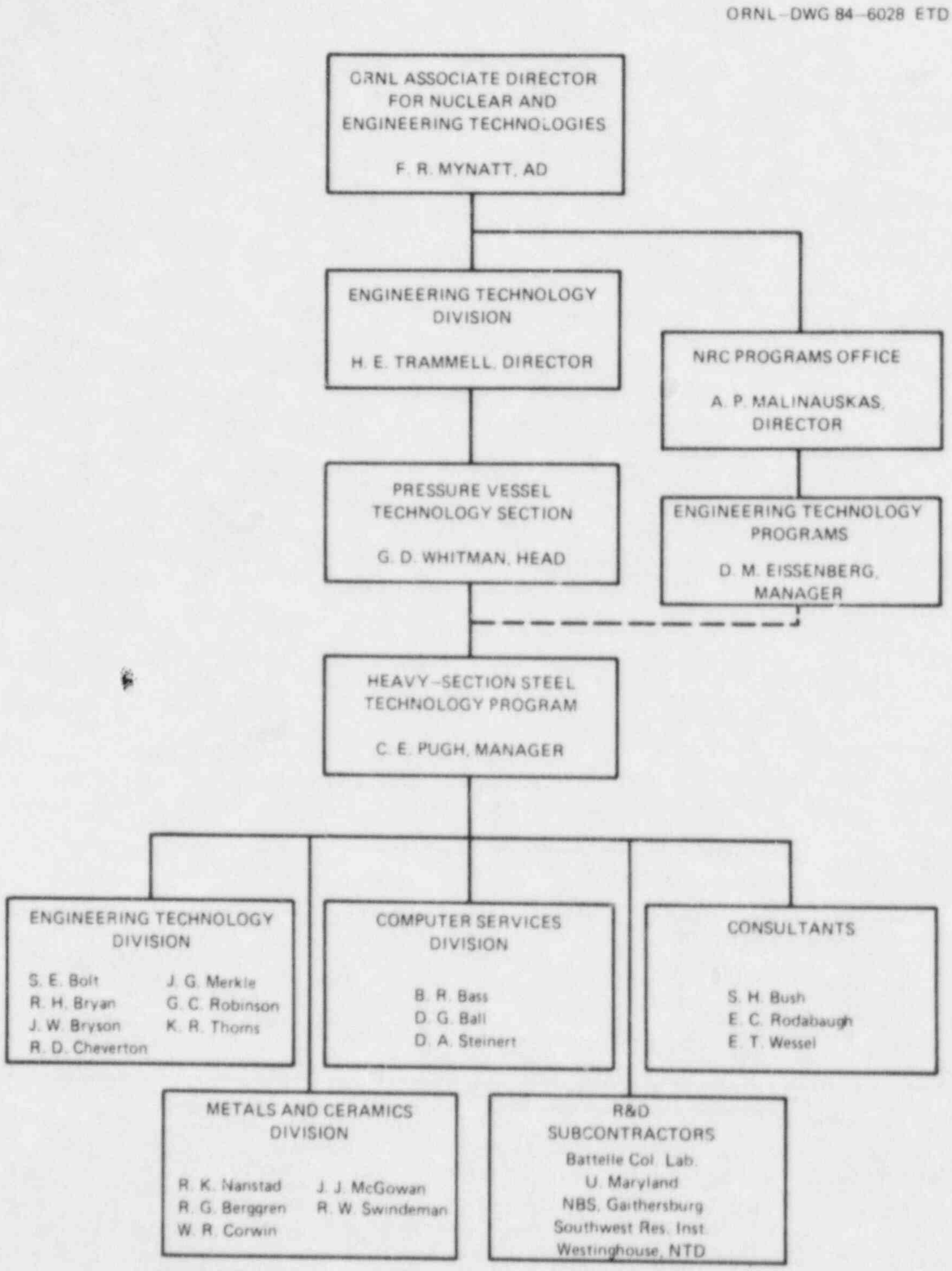


Fig. 1.2 Organization of the Heavy-Section Steel Technology program.

ORNL-DWG 83-5170 ETD

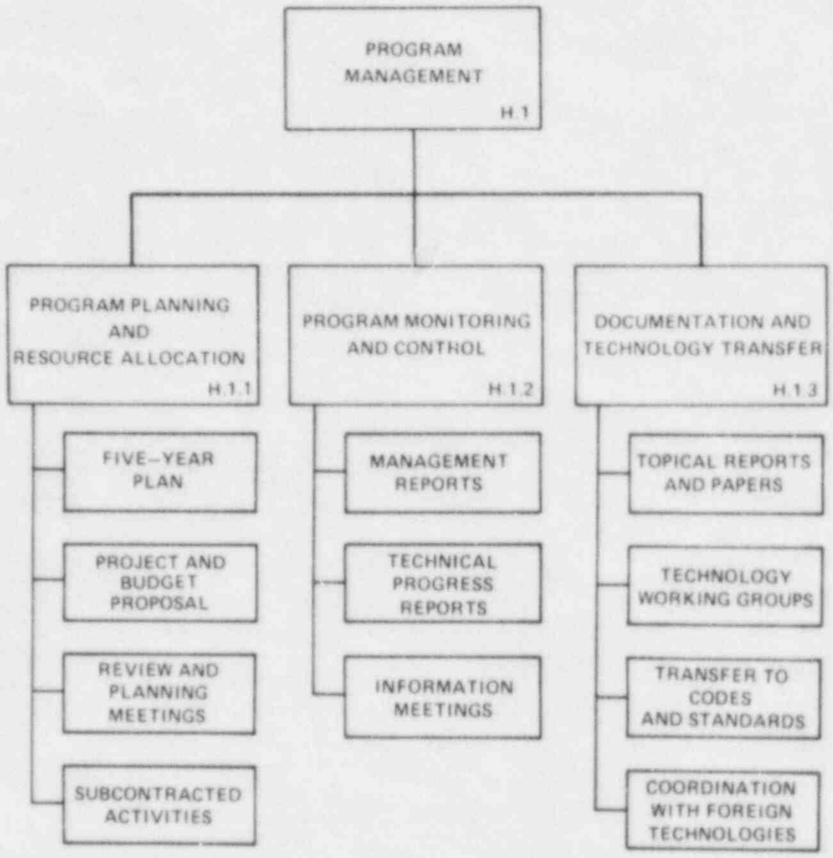


Fig. 1.3 Work breakdown structure for the HSST Task H.1 Program Management.

the ORNL work. The subcontractor progress reports are integrated into the overall program semiannual progress reports. For example, see the FY 1983 and FY 1984 progress reports (Refs. 3–8). The financial burden of progress report publication, all consultants, and program office administration will be born by this task.

Beginning with FY 1984, a revised management reporting system was implemented. This system includes monthly reports from the ORNL program manager to the NRC program manager that have seven parts, (1) a meeting and travel report, (2) a description of technical accomplishments, (3) schedule deviations and impact statements, (4) an identification of anticipated problems, (5) a list of publications, (6) a financial report, and (7) statement of progress towards meeting milestones. An associated part of this reporting procedure is the preparation of an annual edition of the program's five-year plan. The plan will be issued near the end of each fiscal year and will serve as the reference plan document during the next year.

The coordination of technologies with foreign countries often leads to cooperative efforts and assignment of personnel between the various countries. Concerning cooperative efforts, the international round-robin on precracked Charpy testing (Task H.2) will be completed in

FY 1985 and irradiation and further testing of a reactor vessel steel from West Germany will also be carried out in FY 1985 (Task H.3). The program has an invitation to participate in analyzing West Germany's HDR nozzle corner-cracking test results and in the planning and analysis of the Pressurized Thermal Shock Test of the HDR beltline region after insertion of an axial flaw. The HSST program expects to respond to the latter of these invitations to an extent that is in keeping with present budgetary restraints (see Task H.9). The first of these invitations is to be accommodated by ORNL hosting Mr. Stefan Brosi of the Swiss Federal Institute for Reactor Research (SFIRR) for a three-month assignment in FY 1985 to run analyses of HDR nozzles using HSST computer programs.

With regards to other exchange of personnel, Dr. Shafik Iskander of ORNL is to continue his assignment as the NRC representative to the MPA in the FRG through FY 1985 (funded under a separate FIN number). A representative (Dr. Keith Reading) from the UKAEA-Risley was on assignment to ORNL for the first seven weeks of FY 1984 to work with the thermal-shock task (Task H.9). A representative (Dr. H. K. Stamm) from the Institute für Reaktorbauelemente, Karlshrue, is presently assigned to ORNL for seven months, ending in January 1985, to work on dynamic fracture analyses. Also Dr. R. Wanner of the Swiss Federal Institute for Reactor Research is on a one-year assignment to ORNL that will extend through FY 1984, and is working with fracture computer programs such as ADINA.

1.4.4 References

1. G. D. Whitman et al., *Program Plan — The Heavy-Section Steel Technology Program*, Rev. 0, April 1, 1968, Rev. 1, February 27, 1970 (unnumbered document prepared for the USAEC).
2. G. D. Whitman, G. C. Robinson, Jr., and A. W. Savolainen, *Technology of Steel Pressure Vessels for Water-Cooled Nuclear Reactors*, ORNL-NSIC-21, December 1967.
3. G. D. Whitman, C. E. Pugh, R. H. Bryan, *Heavy-Section Steel Technology Program Quart. Prog. Rep. October-December 1982*, NUREG/CR-2751, Vol. 4, (ORNL/TM-8369/V4).
4. C. E. Pugh, *Heavy-Section Steel Technology Program Quart. Prog. Rep. January-March 1983*, NUREG/CR-3334, Vol. 1 (ORNL/TM-8787/V1).
5. C. E. Pugh, *Heavy-Section Steel Technology Program Quart. Prog. Rep. April-June 1983*, NUREG/CR-3334, Vol. 2 (ORNL/TM-8787/V2).
6. C. E. Pugh, *Heavy-Section Steel Technology Program Quart. Prog. Rep. July-September 1983*, NUREG/CR-3334, Vol. 3 (ORNL/TM-8787/V3).
7. C. E. Pugh, *Heavy-Section Steel Technology Program Semiannual Prog. Rep. October 1983-March 1984*, NUREG/CR-3744, Vol. 1 (ORNL/TM-9154/V1).
8. C. E. Pugh, *Heavy-Section Steel Technology Program Semiannual Prog. Rep. April-September 1984*, NUREG/CR-3744, Vol. 2 (ORNL/TM-9154/V2).

1.4.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.1 are given in the following charts.

The symbology used in all the milestone charts in this plan and in the HSST program management reports is given in Fig. 1.4 below. The "n" over of the symbols designates the calendar month when a schedule change is made.

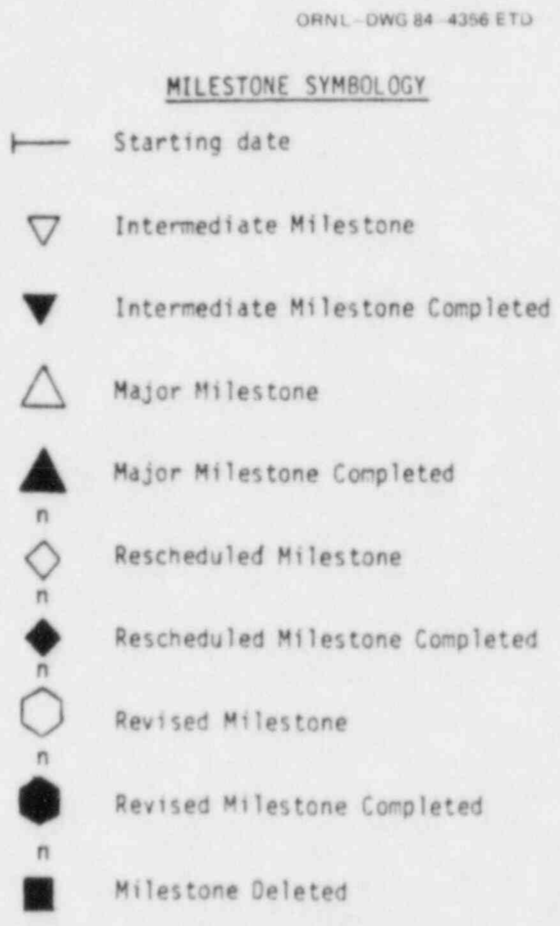


Fig. 1.4 Milestone chart symbology.

MILESTONE STATEMENT AND SCHEDULE

Task: H.1 PROGRAM MANAGMENT

SUBTASK/MILESTONE

H.1.1 <u>Program Planning and Resource Allocations</u>

A. Issue Five-Year PLAN

a. Issue FY-1984 edition

b. Issue FY-1985 edition

c. Issue FY-1986 edition

d. Issue FY-1987 edition

e. Issue FY-1988 edition

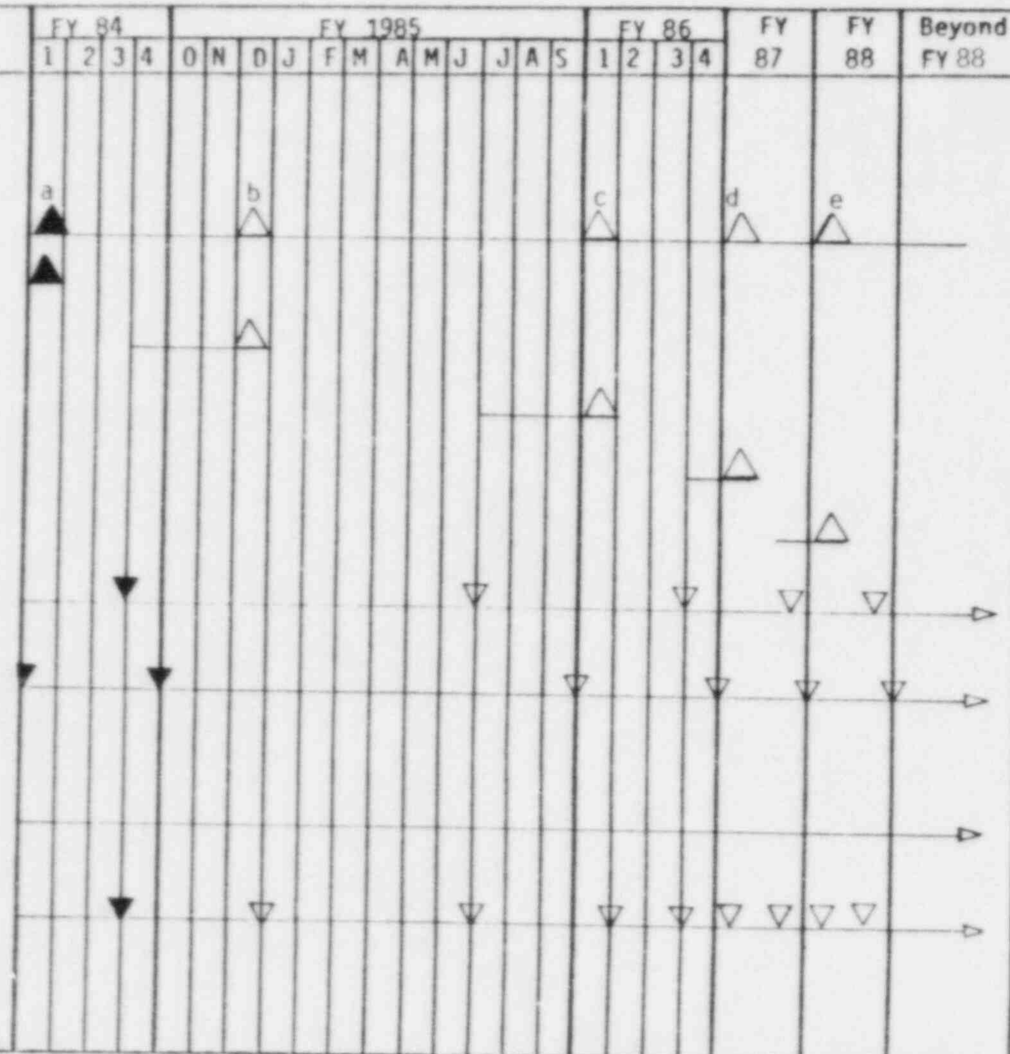
B. Issue Project and Budget Proposal (189)

C. Select and Administer Subcontract Activities on Annual Basis

H.1.2 <u>Program Monitoring and Control</u>

A. Issue Monthly Management Reports (by the end of subsequent month)

B. Issue Seimiannual Technical Progress Reports



MILESTONE STATEMENT AND SCHEDULE

Task: H.1 PROGRAM MANAGEMENT

SUBTASK/MILESTONE

H.1.3 Documentation and Technology Transfer

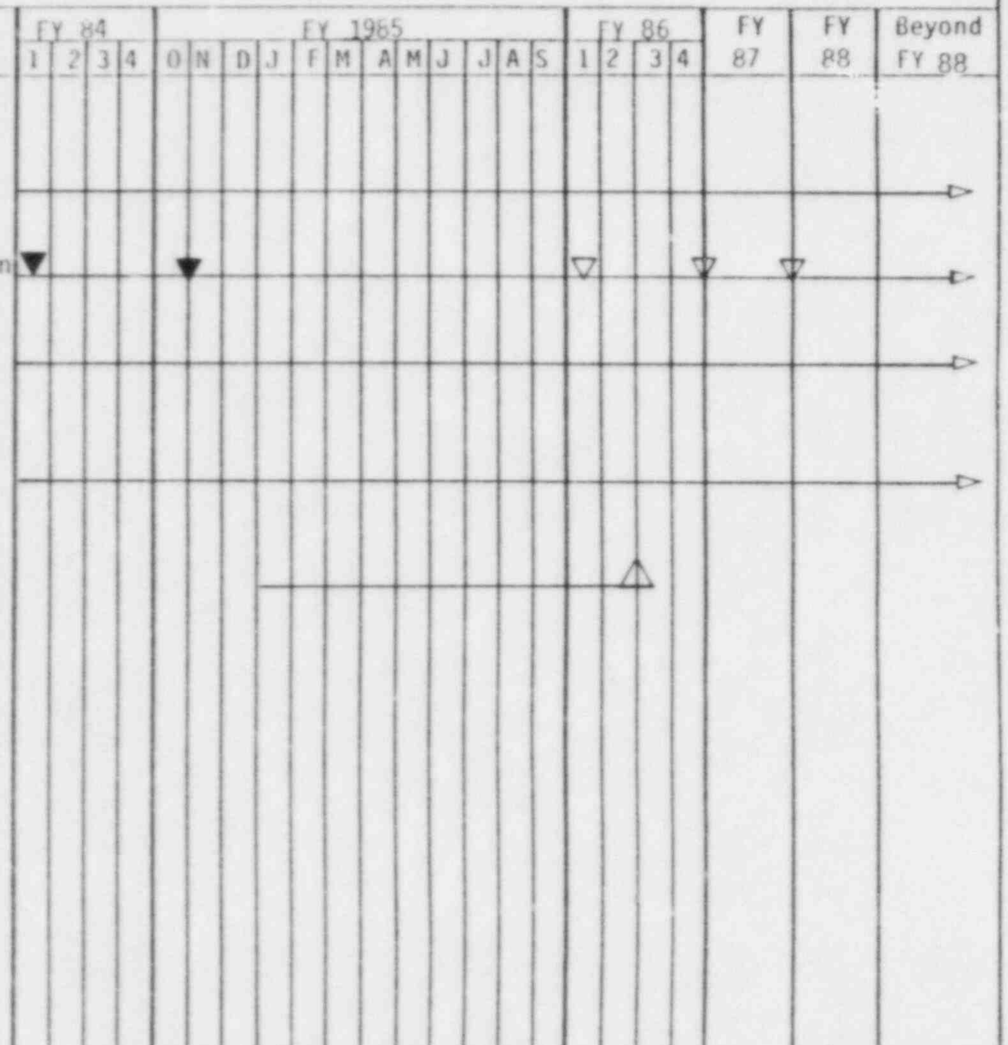
A. Coordinate Issuance of Topical Reports and Papers

B. Participate in NRC Annual Information Meetings

C. Maintain Membership on ASME, ASTM, PVRC, and International Cyclic Crack-Growth Committees

D. Exchange Technology with Foreign Countries as Approved by NRC-NRR

E. Issue Summary Report on Historical Accomplishments of HSST Program



2. HSST TASK H.2 FRACTURE METHODOLOGY AND ANALYSIS

2.1 Objective

The objective of Task H.2 is to develop and evaluate the experimental data and methods of analysis by which the fracture toughness of reactor pressure vessel (RPV) steels can be reliably determined and the results applied with confidence to the design and safety analysis of nuclear vessels.

2.2 Background

The activities covered by this task have generally been and are now organized according to the work breakdown structure shown in Fig. 2.1. The approach has been to assess the applicability of fracture strength analysis methods, with initial methods being based on linear-elastic fracture mechanics, to nuclear vessels on the basis of experimental validation. The HSST Program was the first research effort specifically concerned with high-toughness structural materials that recognized the need for elastic-plastic methods of calculating fracture toughness and analyzing cracked structures. The preceding COD methods considered only crack tip yielding and did not involve elastic-plastic stress analysis. At the outset it was recognized that direct as well as numerical analysis procedures are required, and that experimental programs are vital to the development as well as the validation of analysis methods. The following paragraphs address some of the background in terms of the four subtasks shown in Fig. 2.1.

2.2.1 Fracture Toughness Determinations and Strength Methods

2.2.1.1 Small specimen fracture toughness in the cleavage range. On the basis of the large specimen sizes that had to be used to measure LEFM valid values of fracture toughness[1] for HSST Plate 02, it was clear that specimens small enough to be used routinely for material characterization would yield before fracturing. Therefore, early in the program, semiempirical methods,[2,3] and correlations[4] were developed for estimating fracture toughness values from specimens of limited size for the design and analysis of the intermediate test vessels. Following the initial multi-specimen experimental applications[5] of the J Integral[6] for measuring fracture toughness, the HSST Program contributed to the development of single specimen equations applicable first to the notched beam[7] and then to the compact specimen.[8] The latter analysis is now the basis for the measurement of J Integral resistance curves for structural and pressure vessel steels.[9]

Although the existence of size effects in the measurement of fracture toughness has been known for some time, the dependence of these effects on specimen geometry, loading rate, and fracture mode has not been clear, nor the effects consistently recognized. The large scatter

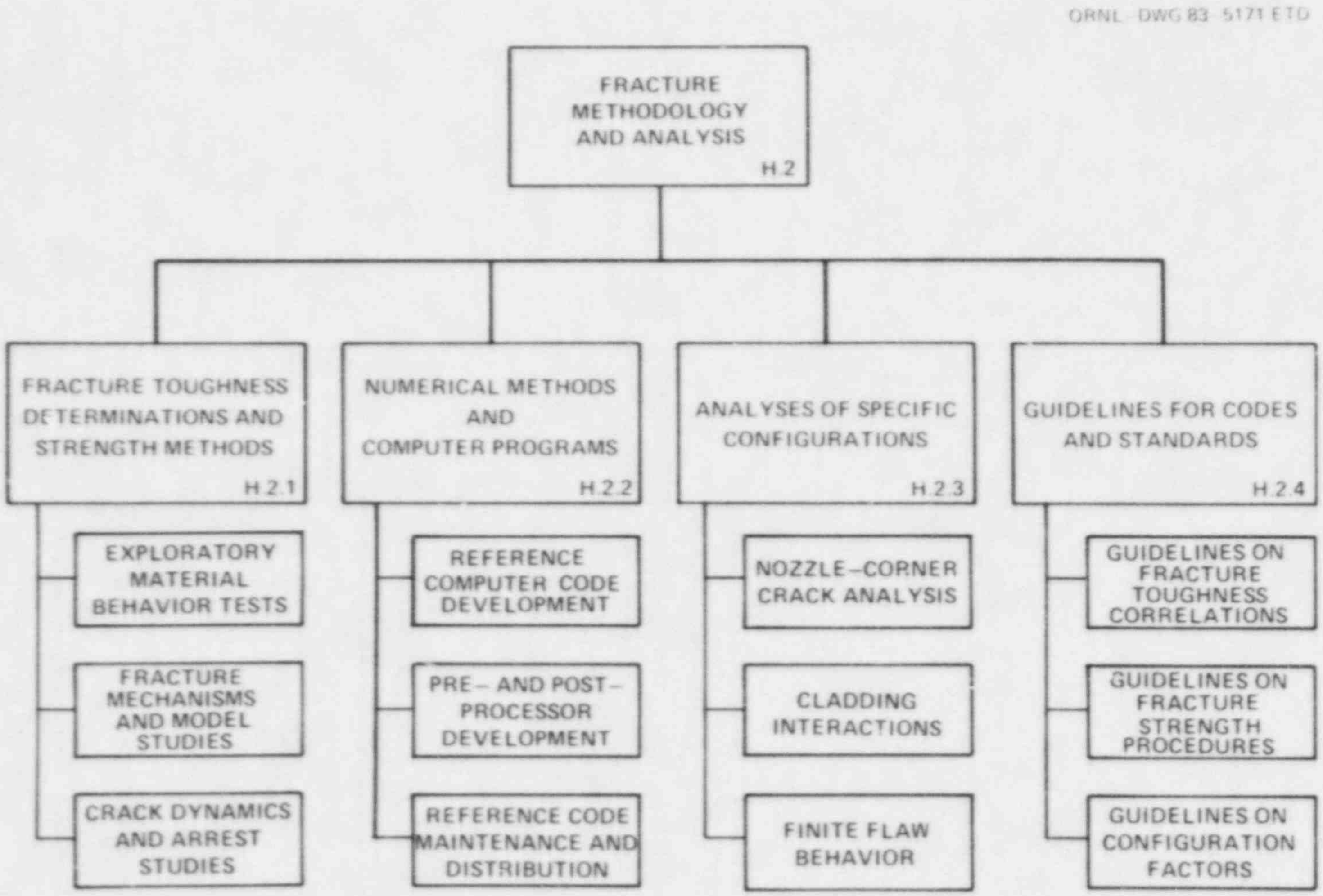


Fig. 2.1 Work breakdown structure for the HSST Task H.2 Fracture Methodology and Analysis.

in data and elevation of small specimen toughness values with respect to large specimen test results[10,11] for TSE-5 and TSE-5A brought this problem to focus, and motivated a deliberate study of the problem.[12] It turns out that a semiempirical method of analysis developed by Irwin[13] in 1960 can be applied to this problem, but that a better understanding of the physical basis for size effects is necessary to justify the application of such a procedure to important problems involving structural safety. The method, as refined by ORNL, is currently being applied to the toughness characterization of the vessels used for thermal-shock and pressurized thermal-shock tests. Recent experience in applying the method to the first pressurized thermal-shock test has indicated that additional developments are needed for temperatures above the RTNDT. At these temperatures, stable crack growth usually precedes cleavage, and strain rate effects caused thereby may affect the test results.

Resistance curve measurements of ductile-tearing toughness currently require the measurement of small amounts of crack extension by somewhat difficult means. This requirement thwarts the use of load-displacement data from specimens not instrumented for Δa measurement for estimating resistance curve values. A method of estimation based only on load-displacement measurements recently developed under the fracture mechanics task helps to circumvent this difficulty.[14] In addition, the proper application of a correction to the J calculation for stable crack extension during an R curve test has a strong effect on the size independence of the resulting R curve and the toughness level implied by the data. An improved correction has been developed by Ernst,[15] and a direct derivation, based on physical reasoning, has been developed and implemented at ORNL, facilitating the direct application of the improved correction to new experimental data.

In the temperature range within which cleavage is the dominant mode of fracture, an increase in loading rate or crack speed tends to lower the toughness toward a minimum level. However, fast running cracks eventually display an increase in toughness with crack speed implying the interaction of multiple physical phenomena. A preliminary analysis has been developed that implies that the observed minimum is the result of the opposite effects of strain rate and entrapped heat in the crack tip plastic zone. This hypothesis will be investigated further, within practical limits, because it offers the possibility of estimating the crack arrest toughness from other more easily measured properties.

2.2.1.2 Cleavage-fibrous transition investigations. The University of Maryland, under an HSST subcontract, has been investigating the physical and metallurgical basis for the conversion of fracture mode with temperature from cleavage to fibrous tearing. This investigation utilizes and extends previous observations of cleavage microcracking[16] and ductile remaining ligaments,[17] and has provided direct supporting information for the study of size effects in cleavage fracture toughness testing (Milestone H.2.1.A). Improved methods of etching and crack profile measurement have been developed, and they promise to facilitate the obtaining of metallurgical information specific to reactor grade steels. An effort has also been underway at Battelle Columbus Laboratories,[18] under another HSST subcontract, in coordination with the work at the University of Maryland.

2.2.1.3 Precracked Charpy V-notch (PCCVN) round robin. Because of the need for reliable small specimen testing methods (Milestone H.2.1.A) and because of periodic proposals and problems concerning the use of PCCVN specimens, a meeting on the subject was held at EPRI in December 1980, sponsored by CSNI.[19] The NRC proposed at that meeting that the HSST Program furnish PCCVN specimens from the TSE-5A cylinder for an international round-robin test program. Eleven laboratories, including ORNL, volunteered to test specimens. To date ORNL and six other laboratories have delivered results, all consistently indicating the same large degree of scatter and upward elevation of toughness values in the transition range. A final evaluation of data received will be performed in FY 1985.

2.2.2 Analysis Methods and Computer Programs

2.2.2.1 Computer code development. Both general and special-purpose computer programs are being developed for use in HSST fracture studies. Table 2.1 gives the status and capabilities of each of the fracture codes developed at ORNL. These programs perform static analysis of brittle or ductile fracture in 2-D or fully 3-D crack geometries. In addition, the SAMCR dynamic fracture analysis code has been developed under subcontract to the University of Maryland; and the dynamic fracture code SWIDAC has been utilized through a subcontract with the Southwest Research Institute. SAMCR (Stress Analysis of Moving Cracks) has been used to perform posttest analyses of TSE-5, with limited success.[20] Improvements have recently been made to the program,[21] and it has been used to analyze the first wide-plate crack-arrest tests (see Task H.5). The dynamic code SWIDAC is a version of the FRACTDYN code developed by Battelle Columbus Laboratories.[22] A copy of SWIDAC was made available to ORNL in 1984, and both SWRI and ORNL have analyzed the first wide-plate crack-arrest tests with this code. Additionally, ORNL has analyzed the first pressurized thermal-shock test (PTSE-1) using SWIDAC.

The OCA series of codes and the ORMGEN-ADINA-OVIRT finite element system have been extensively applied in the design and analysis of HSST experiments. The OCA codes employ a cost-effective influence function approach which when coupled with their post-processing capabilities make them particularly suitable for parametric studies. With the exception of OCA/USA which has an upper-shelf analysis capability and a direct ligament stability analysis, the OCA codes are linear elastic and limited to the analysis of surface flaws in PWR or ITV geometries. The ORMGEN-ADINA-ORVIRT finite element system represents a more general capability, and is a fully 3-D thermo-elastic-plastic finite element system which uses a virtual crack extension technique for the computation of energy release rates at various points along the crack front. Special crack tip elements are employed which introduce the appropriate stress singularity. This system of codes has been given wide distribution both domestically and internationally.

2.2.2.2 Direct analysis techniques. Up to the present time, the analysis of pressurized thermal-shock events in reference model vessels have been performed by elastic analysis, e.g., with the OCA Code.[23]

Table 2.1. Summary of HSST computer programs for fracture analysis

Fracture code	Date developed	Numerical method	Geometry	Nonlinear material behavior	Comments
			ORNL Developed Fracture Codes		
OCA/USA	Under development	Influence function	2-D, limited 3-D	Limited	An enhanced version of OCA-II, designed primarily for pressurized-thermal shock experiments
OCA-P	May, 1984 NUREG/CR-3618 ORNL-5991	Influence function, Monte Carlo	2-D, limited 3-D	No	Probabilistic fracture code employing a Monte Carlo simulation technique
OCA-II	February, 1984 NUREG/CR-3491 ORNL-5934	Influence function	2-D, limited 3-D	No	Outstanding post-processing capabilities, especially suitable for parametric analyses of surface flaws in PWR and ITV geometries
ORVIRT	February, 1983 NUREG/CR-2997 Vol. 2 ORNL/TM-8527/V2	Finite element (Virtual crack extension)	2-D, 3-D	Yes	Versatile and powerful fracture code, interfaces with ORMGEN and ADINA
ORMGEN	December 1982 NUREG/CR-2997 Vol. 1 ORNL/TM-8527/V1	Finite element mesh generator	3-D	N/A	Generates a 3-D finite element model for cracked plates or cylinders in an ADINA compatible format
ORFLAW	April, 1982 NUREG/CR-2494 ORNL/CSD/TM-165	Finite element (Embedded K)	3-D	No	Developed under subcontract to S. Atluri, Georgia Tech, 3-D, linear elastic only, automatic mesh generation
OCA-I	August, 1981 NUREG/CR-2113 ORNL/NUREG-84	Influence function	2-D	No	Strictly 2-D analysis of surface flaws in PWR and ITV geometries
NOZFLAW	March, 1981 NUREG/CR-1843 ORNL/NUREG CSD/TM-18	Finite element (Embedded K)	3-D	No	Developed under subcontract to S. Atluri, Georiga Tech. Addresses nozzle corner flaws only, automatic mesh generation
FMECH	February 1981 NUREG/CR-1499 ORNL/NUREG/CSD/TM-14	Finite element (Virtual crack extension)	2-D	No	First ORNL developed code, limited capabilities relative to more recently developed codes
			Outside Codes that Interface with ORNL Computer Programs		
ORNL-SAMCR	November 1984 NUREG/CR-3891 ORNL/Sub/79-7778/3	Finite element	2D	No	ORNL version of the SAMCR dynamic fracture analysis program developed under subcontract to University of Maryland
SWIDAC	1984	Finite element	2D	No	Elastodynamic fracture analysis code developed at SWRI
BIGIF	1978 EPRI NP-700-1	Influence function	2-D, limited 3-D	No	Developed under EPRI sponsorship, widely used used by utilities
ADINA	1975 (rev. 1978) MIT report 82448-1	Finite element	2-D, 3-D	Yes	A general purpose finite element code for structural analysis developed by K. J. Bathe at MIT

However, direct analyses have been used to show that as the crack becomes deep, the remaining ligament yields under combined bending and tension, causing additional crack opening displacements. Under pressure loading, a crack depth is subsequently reached at which the remaining ligament is yielded under uniform tension, causing large additional crack opening displacements which would eventually lead to ligament necking and tensile instability.[24] The analysis of this phenomenon for an external crack has been included in a modification of the OCA code called OCA/USA[25] (OCA Upper-Shelf Analysis), and was used in the pretest analysis of PTSE-1 (see Task H.10).

2.2.3 Analysis of Specific Configurations

At the outset of the HSST Program it was recognized that the development of fracture mechanics estimating methods for finite length through and part-through surface cracks subject to stress gradients would be an important part of the fracture mechanics task. In fact, the realism and practical applicability of the experimental and analytical results of the HSST program were seen to depend significantly on this factor. Numerical methods were not capable of treating this problem with sufficient speed or accuracy, so semiempirical methods were developed. These methods still prove extremely useful for preliminary estimates, experimental design and the definition of problems for numerical analysis by methods that are now much improved

An experimental approach to the problem for inside nozzle corner flaws was taken, based on the fracture testing of epoxy scale model vessels.[26] Three-dimensional photoelastic studies[27,28] and analytical studies[29] followed. Some differences between these results, with respect to K_I variations around the flaw perimeter for deep cracks, still remain to be resolved.[29]

The burst analysis of through flawed cylinders utilized semiempirical equations developed at Battelle Columbus Laboratories, for which rational derivations were developed at ORNL.[30] These analyses need to be augmented for the purpose of estimating flaw opening areas and leak rates for through flaws, a problem concerning which assistance is being provided to Pacific Northwest Laboratories.

The work on the finite length part-through surface crack subject to a stress gradient began with a review of existing solutions and estimates,[31] followed by the development of a semiempirical equation adjusted to fit three-dimensional photoelastic experimental data.[32] A preliminary version of this equation was used for the initial ORNL study of the thermal-shock problem,[33] and later for the design and analysis of the two thermal-shock experiments involving finite length part-through surface cracks,[34,35] TSE-2 and TSE-7. It was also used for the posttest analysis of the V-8 Test,[36] the flaw design and pretest ductile-tearing instability analysis of the V-8A test,[37] and the design of the flawed stainless steel clad plate tests[38] (see Task H.7). In the latter case a method was developed whereby the shape of arrested cracks propagating in a region of stress and toughness gradient can be estimated. This aspect of the method will be further explored.

In the posttest analysis of the V-8 Test,[36] it was noticed that the back face free surface magnification factor found to be appropriate for a flawed plate produces an overestimate of K_I for a deep crack in a cylinder, because of the circumferential continuity of the cylinder. The use of a different magnification factor based on no change in back surface curvature was found to remedy this problem.[36]

In studying a question concerning the shortest length of a surface crack that could be treated as a continuous crack, for PTS experimental design, it became evident that a considerable difference in K_I values exists between finite length and continuous cracks in pressurized cylinders, for $a/W \geq 0.5$. This is because bending stiffness is not lost beyond the ends of the crack for a finite-length crack. This investigation led directly to the consideration of finite-length flaw effects in PTS analysis[39] (see Tasks H.9 and H.10).

2.2.4 Guidelines for Codes and Standards

2.2.4.1 Fracture-toughness correlations. Current specifications do not require the measurement of complete curves of fracture toughness versus temperature for reactor vessel steels. Therefore correlations are needed in order to facilitate the construction of such curves, which are necessary for performing safety analyses, especially for loadings involving thermal transients. The HSST Program has contributed several useful correlations between impact data and fracture toughness,[4,40,41] as well as the valid data[1,42] on which the present ASME Section III and Section XI reference fracture-toughness curves are based.[43,44]

The HSST Program continues to participate in efforts to develop improved impact energy-toughness correlations, for example through active membership in the PVRC/MPC Task Group on Reference Toughness. This group evaluates the applicability of a statistically augmented correlation between Charpy-impact energy and fracture toughness, based on curve fitting with a hyperbolic tangent equation.[45] ORNL contributed directly to clarifying the mathematical basis of the method, as well as pointing out size effects between the EPRI ITCT toughness data base and the large specimen data for Plates 02 and 03. A problem remains regarding the inability of the method to make consistently accurate predictions of mean values, despite the statistical nature of the method. Additional attention to these problems will be required. We have also participated in the deliberations of ASTM Committee E-24 on Fracture-Toughness Testing concerning the proper application of Equivalent Energy procedures, again pointing out the existence of size effects so as to make the description of the procedure accurate and acceptable. In addition, ORNL will be actively participating in the work of a newly formed task group under ASTM Subcommittee E24.08 on Elastic-Plastic Fracture Mechanics Technology, concerning toughness measurements in the transition range.

2.2.4.2 Fracture strength (analysis) procedures. The development of strength analysis methods applicable to flawed regions in pressure vessels that consider strain gradients and inelastic behavior has been a focal point of the fracture mechanics task since its beginning. As stated earlier, the development of direct as well as numerical iterative methods of analysis has been a guideline within the task. This approach

recognizes the existence of cost and time limitations, the need to determine the significance of specific variables, the need for considering flaw size as a dependent variable in experimental design, the need for independent checks, and the fact that numerical methods are not specified in the ASME Code. Pre and posttest analyses of intermediate vessel tests were solicited from others, and made at ORNL, using as many different methods as could be identified and shown to be applicable to the problem.[46-48] Two of the earliest methods of analysis developed were the Equivalent Energy[49] method and the Tangent Modulus[50] method, the latter of which was originally semiempirical and based on experimental results from surface flawed bars tested to fracture in the elastic-plastic range.[51] Although controversial for some time, the Equivalent Energy method was eventually shown to be relatable to the J Integral, on the basis of an assumed power law stress-strain curve.[52] Furthermore, an extension of this approach resulted in the development of the British Normalized COD Design Curve, thus relating three originally separate fracture analysis procedures.[53] The Tangent Modulus method was subsequently derived theoretically from the J Integral, via the development of an incremental form of Neuber's equation for calculating inelastic stress and strain concentration factors.[54]

In addition to fracture mechanics analyses, it was also necessary to develop direct methods of calculating or estimating the nominal elastic-plastic pressure versus strain curves for vessel cylinders[47] and nozzle corner regions.[55] The former is an exact solution for a trilinear stress-strain curve, and the latter is semiempirical. Both estimates are used to judge the detail necessary for representing the stress-strain curve in a numerical analysis, and the accuracy of the results.

Eventually, a load will be reached at which a flawed structure will fail by necking and tensile instability due to the reduction of load-bearing area caused by the flaw. An analysis of this phenomenon was developed for a part-through surface crack in a vessel cylinder,[46] and it continues to prove useful in analyses such as those for pressurized thermal shock.

It turns out that two of the most difficult fracture mechanics tests to analyze over the entire range of load from elastic to fully plastic are the surface cracked tensile bar and the inside nozzle corner crack. The only analysis method that has so far been successfully applied to both, as well as an ITV cylinder, is the Tangent Modulus method.[50,55,56] The complications in the data are due to such factors as the effects of incremental yielding, transverse contraction, net section eccentricity, strain gradients, and tensile instability. Given the opportunity, numerical methods will be applied to these problems, because they constitute a severe test of the generality of analytical methods.

An important aspect of the HSST Program has been the interpretation of the experimental and analytical results with respect to the expected performance and safety margins of reactor pressure vessels in service. In 1975, a report was prepared concerning the interpretation of the intermediate vessel test results obtained to that date with respect to the safety of actual vessels. It was concluded that only inadequate material properties, large flaws or extreme loading conditions exceeding the values permitted by present codes remain as possible causes of

vessel malperformance under operating conditions.[57] Pressurized thermal-shock transients fall in the latter category, and are presently the subject of intensive investigation (see Task H.10).

The HSST Program staff also participated actively in the work of a recent NRC task group which prepared recommended analysis procedures for determining the safety margins of vessels containing low upper shelf toughness materials.[58] A follow-up report containing comparative example calculations by some of the methods discussed was also presented to the ASME Section XI Working Group on Flaw Evaluation and subsequently published as a NUREG report.[59] The data from intermediate test vessel V-8A relate directly to this subject,[37] and their posttest analysis will provide an opportunity to evaluate the proposed criteria.

2.3 Plan of Action

The plan of action for each subtask is depicted in the milestone chart for Task H.2. A transition from previous accomplishments to the planned activities is provided in many instances by the background discussions given in Section 2.2 above.

2.3.1 Fracture-Toughness Determinations and Strength Methods

The emphasis continues to be on providing analytical models for representing the behavior of finite-length flaws in RPV steels under overcooling accident conditions. Predictions of the initiation, propagation, and arrest of flaws are involved for materials whose fracture characteristics vary in position and time. Consequently the applicability of concepts for a unified fracture mechanics strength methodology to treat different regions of fracture behavior will be developed and/or assessed. In the near term, these efforts will be augmented by the issuance of a report on the cleavage to fibrous transition and continuing work on the improvement of the Irwin B_{Ic} size adjustment procedure. Under the current budgetary plan, the level of effort on the unified theory is modest, with a more vigorous effort planned when the predictions of candidate theories are evaluated against available data from pressurized thermal-shock tests and wide-plate crack arrest tests.

In concert with improved fracture modeling, compatible bases for stress analysis (constitutive equations) are required. Time (rate) dependent inelastic material response will be considered in terms of constitutive equations that do and those that do not distinguish between plastic and creep strains. The applicability of current theories of viscoplasticity will be assessed first in terms of dynamic analyses of fast-running cracks and of arrest events. The laboratory tests that are required to identify and quantify properties that appear in these equations will continue to be performed as a part of implementing the assessments for the specific materials of interest. Assessments of the applicability of constitutive equations include an evaluation of the following aspects: (a) the ability to predict the deformation (rate-dependent) response of laboratory specimens, (b) the ability to predict the response of structural configurations that include multiaxial stresses and stress gradients, (c) compatibility with practicable analysis

tools such as finite-element computer programs, and (d) compatibility of mechanical properties requirements with existing data bases or those that might be developed with reasonable cost. Milestone H.2.1.E as defined in this plan is (through a subcontract) to employ a selected theory of visco-plasticity in an advanced version of the dynamic fracture analysis computer code SWIDAC. A combined ORNL and subcontractor approach is envisaged over the longer run. After the ongoing assessment of current theories of viscoplasticity is completed, further analysis methods developments will be pursued as needed.

In addition to investigating the application of advanced rate-dependent inelastic constitutive equations to fracture mechanics analysis, the effects of the conversion of plastic work to heat near the tip of a running crack will be considered. Since the zone of actual material separation is very near the crack tip, this energy conversion may affect the temperature at the point of incipient microcracking, and this affects the toughness itself. The interaction of this effect with the effects of strain rate will be examined with a view toward making estimates of the crack-arrest toughness when direct measurements cannot be made.

2.3.2 Analysis Methods and Computer Programs

An emphasis in this subtask continues to be on analytical techniques applicable to the analysis of the behavior of finite-length flaws under OCA conditions. A second emphasis is on the provision of dynamic fracture analysis programs that are compatible with available fracture theories and constitutive equations. In addition to computer based methods, direct analysis methods are developed where possible for classes of problems.

The upper-shelf analysis capability that is presently in the computer code OCA/USA will continue to be improved, including an extension to be applicable to clad vessels. A new 2-D dynamic code will be developed at SWRI to be used with viscoplastic constitutive equations. SWIDAC will be used for elastodynamic analyses. A dynamic crack analysis capability will be incorporated into the ADINA-ORVIRT system to give ORNL an in-house general 3-D dynamic analysis capability. This will also serve as the basis for a later incorporation of improved unified fracture models and viscoplastic constitutive equations into the ADINA-ORVIRT system. Considering the current overall capabilities of ADINA, current viscoplasticity theories can be exercised in this finite-element analysis program with relatively low developmental requirements when compared with other large computer codes.

As noted in the background discussions, a ligament instability analysis has been developed for an externally flawed vessel. This method is based on direct analysis approaches and not on finite-element or other numerical techniques. An analogous ligament instability analysis has been developed for an internally flawed vessel, and example calculations will be performed. Although the theoretical foundations are the same for the two situations, an internally flawed vessel is the specific geometry of concern in overcooling accident situations.

2.3.3 Analyses of Specific Configurations

This Subtask (H.2.3) augments Subtask H.2.2 by further qualifying fracture analysis capabilities through the analysis of chosen configurations. Milestone H.2.3.A is aimed at furthering the understanding of warm prestressing (WPS) through the elastic-plastic analysis of sample structures. Since beams have been used in some of the landmark experiments to demonstrate WPS, beams are chosen here for analyses. Classical theories of plasticity, such as isotropic hardening, kinematic hardening, and combined isotropic-kinematic hardening models, will be used with the ADINA computer program to look at loading-history effects on the stress field in the vicinity of a crack. Since versions of such plasticity models are now in ADINA, it offers a good possible tool for this study. Of course, associated extensions are needed to be made to ORVIRT and values for the elastic-plastic properties appropriate to the materials of interest must be inserted into the program. At the same time, existing analytical theories of warm prestressing will be examined both theoretically and by applying them to the results of the pressurized thermal shock tests, the wide-plate crack-arrest tests, and the ORNL/NBS WPS-beam experiments.

2.3.4 Guidelines for Codes and Standards

The plan of action here includes a continuation of contributions of fracture toughness and strength results and data from the program to the goals of relevant technical committees. These committees include the PVRC/MPC Task Group on Reference Toughness, ASTM Committee E-24 on Fracture Toughness Testing, and the ASME BPV Code, Section XI Working Group on Flaw Evaluation. The results are also to be made available to US-NRC Regulatory Guide development and to US-NRC position papers on the resolution of safety issues. Specific issues that are active at this time are identified in the Background Section 2.2.4 above.

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2.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.1 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

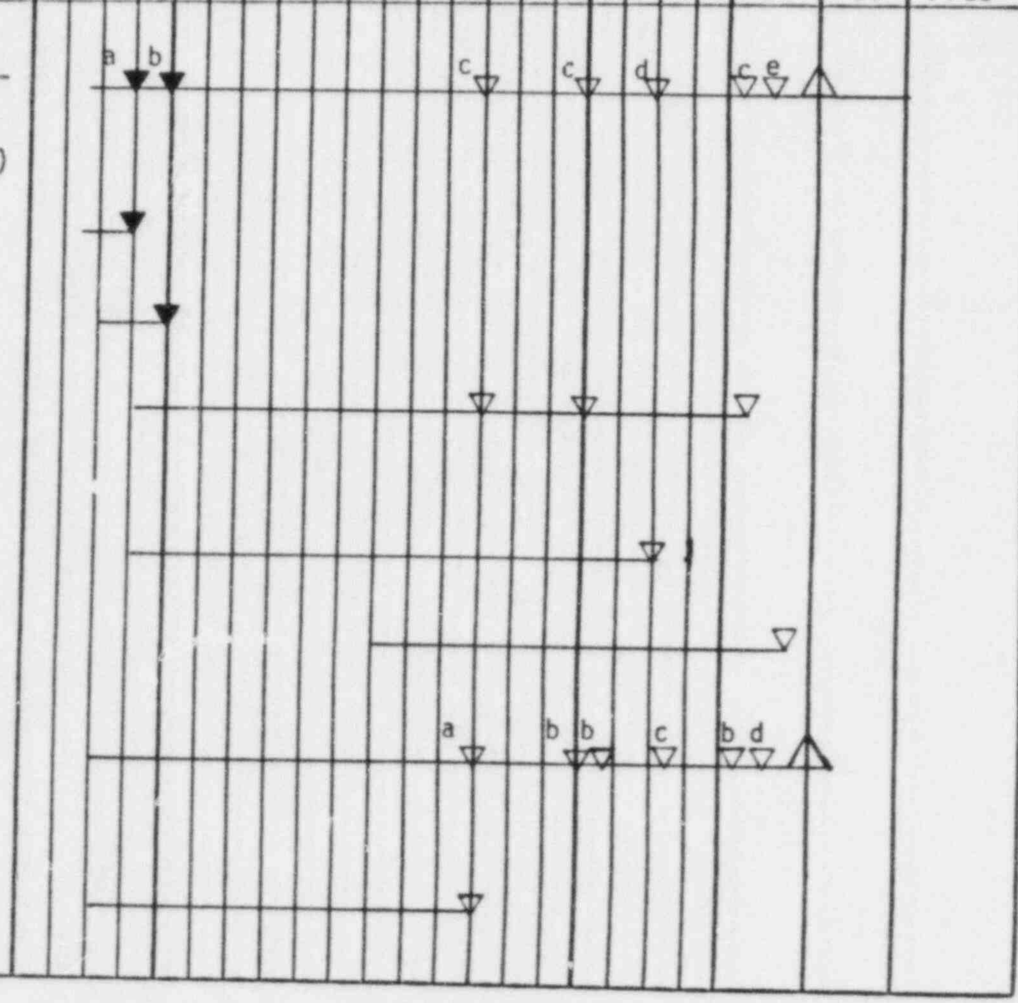
Task: H.2 FRACTURE METHODOLOGY AND ANALYSIS

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY	FY	Beyond
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
H.2.1 Fracture Toughness Determinations and Strength Methods																							
A. Issue Report on Small Specimen Fracture Toughness in Cleavage Range																							
B. Issue Summary Report on Cleavage-Fibrous Fracture Investigations (University of Maryland)	a			b						c													
a. Complete studies for A533 Gr B steel																							
b. Complete studies and comparisons for A503, A514, and A36 steels																							
c. Complete microstructural studies for probabilistic cleavage model																							
C. Complete Exploratory Investigation of High Dynamic Initiation and Propagation Behavior (University of Maryland)																							
D. Issue Report on PCCV Round Robin																							

MILESTONE STATEMENT AND SCHEDULE

Task: H.2 FRACTURE METHODOLOGY AND ANALYSIS (continued)

SUBTASK/MILESTONE	FY 84	FY 1985	FY 86	FY 87	FY 88	Beyond FY 88
	1 2 3 4	O N D J F M A M J J A S	1 2 3 4	87	88	FY 88



E. Complete Assessment and/or Development of Viscoplastic Constitutive Equations for Analysis of Fast Running Cracks in RPV Steels (SwRI)

   a. Identify and screen candidate theories

   b. Determine viscoplastic properties for initial choice

   c. Perform small-scale viscoplastic experiments at low, transition, and upper-shelf temperatures

   d. Identify most appropriate viscoplastic model

   e. Complete application of viscoplastic analysis to wide-plate and PTSE results

F. Complete Assessment and/or Development of a Unified Fracture Model for Use with Viscoplastic Analyses (SwRI)

   a. Identify and screen candidate models

MILESTONE STATEMENT AND SCHEDULE

Task: H.2 FRACTURE METHODOLOGY AND ANALYSIS (continued)

SUBTASK/MILESTONE	FY 84	FY 1985	FY 86	FY 87	FY 88	Beyond FY 88
	1 2 3 4	O N D J F M A M J J A S	1 2 3 4			
F. Complete Assessment and/or Development of a Unified Fracture Model for Use with Viscoplastic Analyses (SwRI) (continued)						
b. Perform small-scale experiments at low, transition, and upper-shelf temperatures						
c. Identify most appropriate unified fracture model						
d. Complete application to available data						
G. Complete Interpretive Assessment of Warm Prestressing Behavior (See Milestone H.2.3.B)	a	b	c			
a. Survey state-of-the-art interpretations						
b. Complete elastic-plastic interpretations						
c. Complete viscoplastic interpretations						

MILESTONE STATEMENT AND SCHEDULE

Task: H.2 FRACTURE METHODOLOGY AND ANLAYSIS (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
H.2.2 Numerical Methods and Computer Programs																							
A. Issue Report on Development of Finite-Element Code SAMCR for Dynamic Fracture Analysis (University of Maryland)					◆																		
B. Incorporate Dynamic Viscoplastic Fracture Models into SWIDAC (or Equivalent Code) (SwRI)													△							△			
C. Complete Development of Analytical Tools for Predicting Finite-Length Flaw Behavior									a ▽	b ▽			c ▽						a ▽	△			
a. Develop OCA-USA to be applicable to clad vessels									▽										▽				
b. Develop ORNOŽL finite-element mesh generator for 3-D nozzle-corner cracks										▽													
c. Develop ORNOZL for HDR analyses (See Milestone H.2.3.C)													▽										

MILESTONE STATEMENT AND SCHEDULE

Task: H.2 FRACTURE METHODOLOGY AND ANALYSIS (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
D. Complete Development of Tools Capable of Predicting Inelastic Fracture Behavior of 2-D and 3-D Structures (ORNL)											a						b			c			
a. Incorporate dynamic crack model into ORNL analysis system based on state-of-the-art techniques																							
b. Incorporate viscoplastic constitutive equations and compatible fracture model into ORNL fracture analysis system																							
c. Complete assessment of ORNL viscoplastic fracture analysis system																							
H.2.3 Analyses of Specific Configurations																							
A. Complete Development of Ligament Instability Analysis for Internally Flawed Vessel																							
B. Support Interpretive Study of Warm Prestressing Phenomena Through Analyses of Sample Structures (See Milestone H.2.1.G)													a						b				
a. Perform classical elastic-plastic analyses of sample scenarios for beams and PTSEs																							

MILESTONE STATEMENT AND SCHEDULE

Task: H.2 FRACTURE METHODOLOGY AND ANALYSIS (continued)

SUBTASK/MILESTONE

B. Support Interpretive Study of Warm Prestressing Phenomena Through Analyses of Sample Structures (See Milestone H.2.1.G) (continued)

b. Perform viscoplastic analyses of sample scenarios for beams and PTSEs

C. Complete Analyses of HDR Crack Configurations

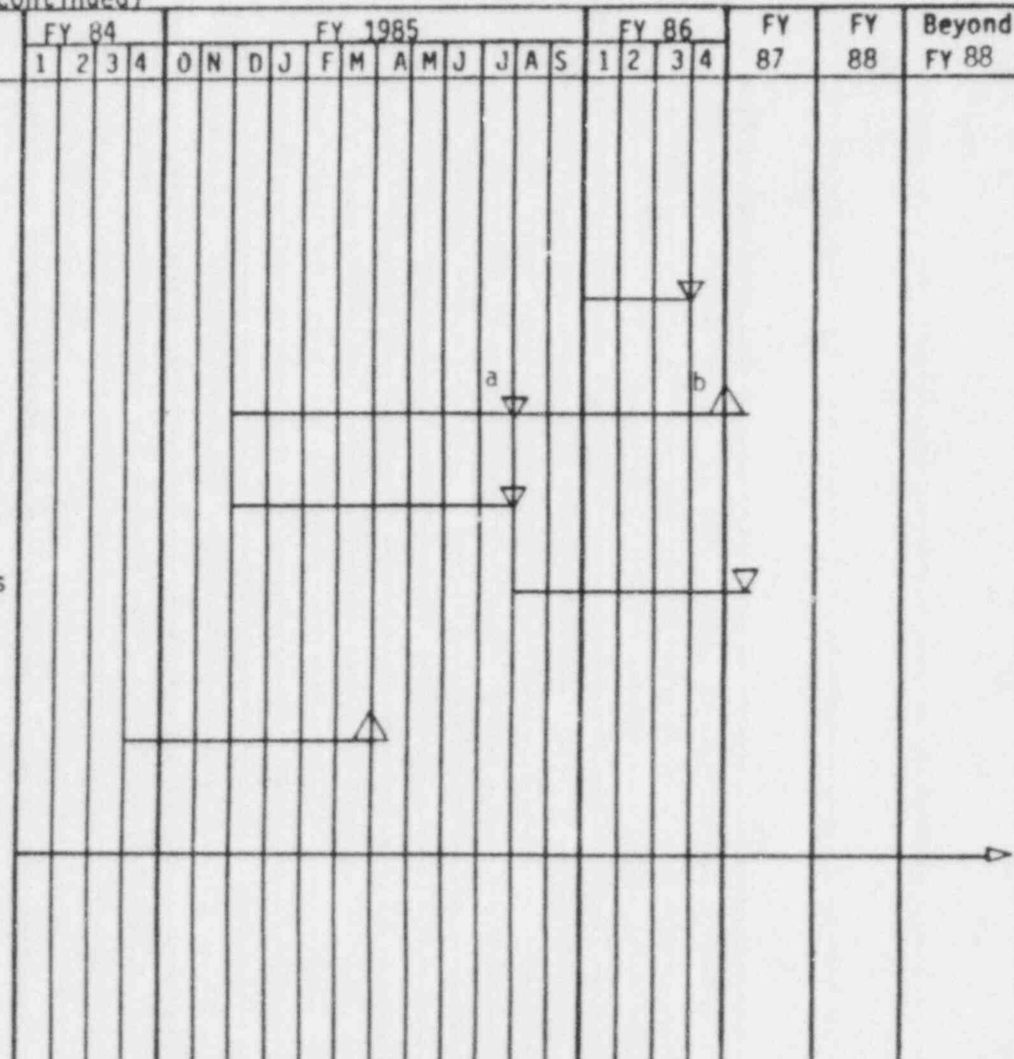
a. Complete nozzle crack analyses (See Milestone H.2.2.C.c)

b. Complete beltline-crack analyses (See Milestone H.9.1.E)

H.2.4 Guidelines for Codes and Standards

A. Organize, Conduct, and Report on CSNI Workshop on Ductile Fracture Mechanics

B. Maintain Membership in PVRC/MPC Task Group on Reference Toughness



MILESTONE STATEMENT AND SCHEDULE

Task: H.2 FRACTURE METHODOLOTY AND ANALYSIS (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
C. Maintain Membership on ASTM Committee E-24 on Fracture Toughness Testing																							
D. Maintain Membership on ASME BPV Code Section XI Working Group on Flaw Evaluation																							
E. Provide Program Results to NRC Regulatory Guides and Position Papers																							

3. HSST TASK H.3 MATERIAL CHARACTERIZATION AND PROPERTIES

3.1 Objective

The primary objective of this task is to characterize the mechanical and physical properties, with emphasis on fracture behavior, of the materials used in the structural experiments which are carried out under other tasks of the HSST Program. The materials currently include those from intermediate test vessels, thermal-shock cylinders, pressurized thermal-shock vessels, wide-plate crack-arrest plates, and clad beams. Other supporting objectives include the development of test methods, active participation in codes and standards activities, and the use and development of metallurgical tools to support all the activities.

3.2 Background

The general activities assigned to this task have been an integral part of the HSST Program since its inception. The activities performed are fairly broad in terms of mechanical and physical metallurgy and are primarily directed to the support of the structural testing tasks of the HSST Program. Prior to FY 1984 the activities were treated as parts of the other tasks. Now, however, a distinct task has been established to provide more visibility to the area of materials characterization and to enhance planning and management within ORNL.

The task is carried out according to the work breakdown structure shown in Fig. 3.1 Activities include the microstructural examination of commercial heavy-section steel plates, forgings and weldments, mechanical testing with emphasis on fracture properties, heat treatment studies to produce desired properties for structural tests, welding fabrication, posttest analysis of fracture surfaces using scanning electron fractography, and development of test methods and analytical methods. Most recently, posttest properties studies have been completed for ITV-8A (Task H.8), TSE-7 (Task H.9), and PTSE-1 (Task H.10).

Activities recently undertaken include the development of a crack arrest testing facility, participation in the ASTM round robin on crack arrest, and participation in the HSST-sponsored international round robin on dynamic precracked Charpy testing.

Major accomplishments arising from this task include:

1. Development of tempering treatment procedures for achieving desired material fracture toughness values for vessel and other structural tests.
2. Demonstration of large scatter in transition region fracture toughness and need for lower-bound analysis.
3. Demonstration of size-effects relationship in fracture toughness testing.
4. Development of correlation between onset of Charpy upper-shelf and 100% ductile fracture using a load-drop technique in an instrumented impact test.

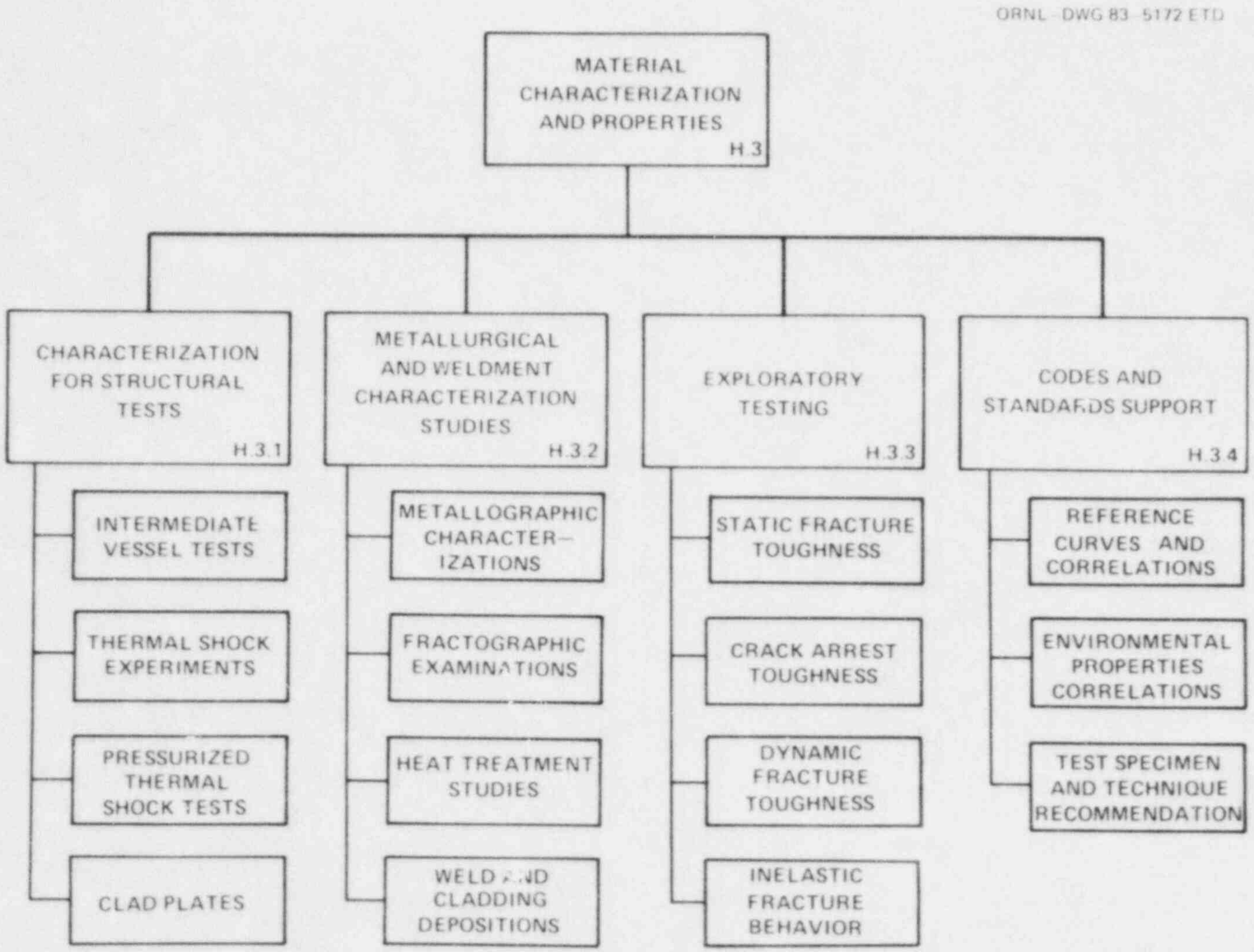


Fig. 3.1 Work breakdown structure for the HSST Task H.3 Material Characterization and Properties.

3.3 Plan of Action

3.3.1 Characterization for Structural Tests

Pretest and posttest characterization have been completed for PTSE-1 as have the tempering studies that provided the heat treatment schedule for the vessel. The test plan for the low upper-shelf test material to be used in PTSE-2 and wide-plate crack-arrest series WP-2 will be similar to that used for the previous pressurized thermal-shock test material. Mechanical properties will be determined prior to the vessel being heat treated to verify the tempering temperature. Posttest characterization will also follow previous plans with completion scheduled for mid FY 1986. In addition to the tensile, CVN and K_J tests, crack-arrest tests will also be conducted to determine K_{Ia} for the upcoming wide-plate and pressurized thermal-shock experiments.

PTSE-3 involves additional characterization because it will be a test with a stainless steel clad vessel. In addition to the usual test plan, mechanical properties will be determined for the stainless steel weld metal cladding. Because the cladding is relatively thin (4 to 6 mm), subsize specimens will be required. Development of test specimens and procedures is continuing in preparation for this characterization as well as for the clad plate tests to be conducted under Task H.7. The pretest investigations will be completed for PTSE-3 by the end of FY 1986.

Characterization testing for the clad-plate tests is similar to that for the clad vessels except that tests will be performed for the three-wire series-arc cladding fabricated by Combustion Engineering (CE), Chattanooga. This project includes the use of weld metal cladding in sufficient thickness to facilitate the removal of standard CVN, tensile, and 0.5T compact specimens for correlation with subsize specimens. Subsize-specimen testing will also be performed with clad nozzle drop-outs procured from CE; this work will be done in FY 1985. Efforts will be continued to obtain drop-outs from other manufacturers.

The testing required for the wide-plate crack arrest tests involves the full range of mechanical property tests, including conventional crack-arrest tests. Initial tests completed in FY 1984 for series WP-1 included tensile, CVN impact, drop-weight, K_{IC} and K_J tests. Complementary development of an alternate crack-arrest test specimen and procedures is underway and will continue in support of the structural tests with particular emphasis on the needs of the wide-plate crack arrest program (Task H.5).

3.3.2 Metallurgical and Weldment Characterization Studies

The activities under this subtask are conducted in parallel with those for the previous subtask. Metallurgical studies involve the use of metallography to examine the microstructure of the materials used for structural tests. Fractography studies involve precision measurements of fracture surfaces and the use of the scanning electron microscope to correlate material deformation behavior with test observations. Scanning electron fractography has been performed for posttest analyses of

TSE-7 and PTSE-1 vessels to examine fracture mode during initiation, propagation and arrest events. When necessary, other metallographic tools such as transmission electron microscope will be used to provide a more detailed examination of microstructural features. This type of investigation has been successfully used to examine the microstructure of the single-wire cladding used for the clad beam tests (Task H.7) and the Seventh HSST Irradiation Series (Task H.6).

3.3.3 Exploratory Testing

The activities of this subtask are related to the needs of the structural testing tasks, but they are better described as separate research and development. The plans and schedules are not well defined because most of those activities are not separately funded and the path of development builds upon previous results. In general, they are performed as time and resources permit.

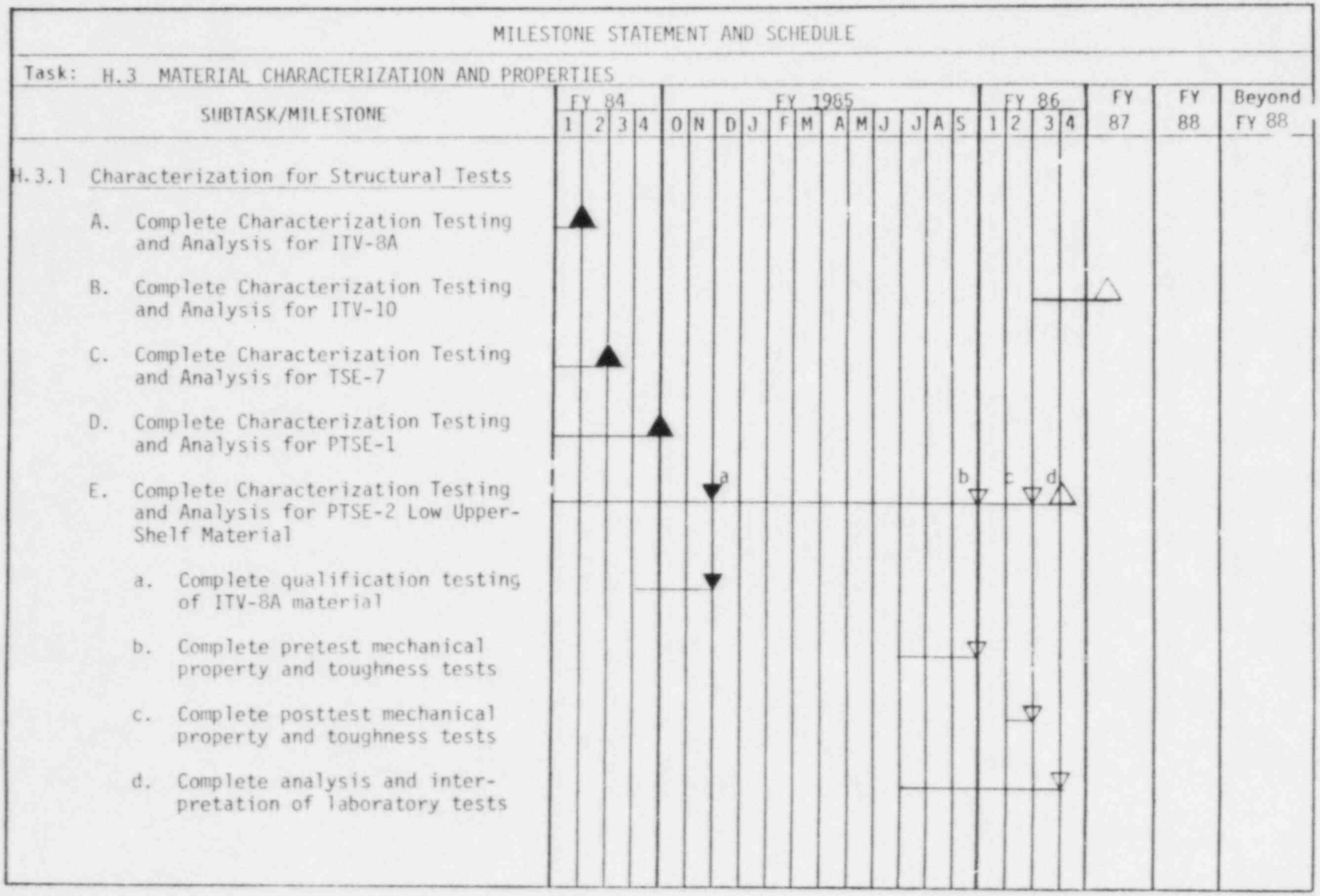
The alternate crack-arrest specimen development is performed in parallel with the development of in-house standard crack-arrest testing capabilities and participation in the ASTM round robin. Exploratory investigations in the area of experimental fracture mechanics include the continued refinement of compliance-type J-R testing procedures, development of a potential-drop system for crack growth monitoring, and fractographic studies and statistical analyses to understand the lower-bound fracture toughness problem. The development of remote crack-arrest testing procedures is directly related to Subtask H.6.5 of the irradiated crack-arrest program. The development of remote testing procedures will be used for hot cell testing of irradiated specimens and is scheduled for completion by the end of FY 1985. The testing of dynamic instrumented precracked Charpy (PCCV) specimens has been completed and provides the ORNL in-house contribution to the international round robin testing program initiated by the HSST Program in 1981 (see Milestone H.2.1.D). This round-robin program is an attempt to understand the relationship between dynamic PCCV fracture toughness results with those obtained from more conventional compact specimen tests and, of course, the results from actual vessel tests. The material was obtained from thermal-shock cylinder 2 (TSC-2) used for TSE-5A.

3.3.4 Codes and Standards Support

Various staff members actively participate in codes and standards organizations such as the American Society for Testing and Materials (ASTM), American Society of Mechanical Engineers (ASME), Pressure Vessel Research Committee (PVRC), and Metals Properties Council (MPC). Participation on various committees of those organizations provides one important forum for transfer of technology gained from HSST Program activities as well as the opportunity to gain feedback from user organizations and other research organizations. It also provides a means for the staff to stay current on codes and standards developments. These activities are performed on a continuing basis and are anticipated to remain active during the life of the HSST Program.

3.4 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.3 are given in the following charts.


MILESTONE STATEMENT AND SCHEDULE
Task: H.3 MATERIAL CHARACTERIZATION AND PROPERTIES
SUBTASK/MILESTONE
FY 84
1 2 3 4
FY 1985
O N D J F M A M J J A S
FY 86
1 2 3 4
FY 87
FY 88
Beyond FY 88
H.3.1 Characterization for Structural Tests
A. Complete Characterization Testing and Analysis for ITV-8A
B. Complete Characterization Testing and Analysis for ITV-10
C. Complete Characterization Testing and Analysis for TSE-7
D. Complete Characterization Testing and Analysis for PTSE-1
E. Complete Characterization Testing and Analysis for PTSE-2 Low Upper-Shelf Material
a. Complete qualification testing of ITV-8A material
b. Complete pretest mechanical property and toughness tests
c. Complete posttest mechanical property and toughness tests
d. Complete analysis and interpretation of laboratory tests
a
b
c
d

MILESTONE STATEMENT AND SCHEDULE

Task: H.3 MATERIAL CHARACTERIZATION AND PROPERTIES (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
F. Complete Characterization Testing and Analysis for PTSE-3																							
a. Complete pretest mechanical property and fracture toughness tests of base metal and clad specimens																							
b. Complete posttest mechanical property and fracture toughness tests																							
c. Complete analysis and interpretation of laboratory tests																							
G. Complete Characterization Testing and Analysis for Clad Plate Tests																							
a. Complete procurement of three-wire clad material (See H.6.6.A. and H.7.2.A.)																							
b. Complete characterization for Phase 1 three-wire clad-plate experiments (See H.7.2.B)																							
c. Complete characterization of commerical three-wire archival cladding (BWR and dropouts)																							
d. Complete characterization for Phase 2 three-wire clad-plate experiments (See H.7.2.C)																							

MILESTONE STATEMENT AND SCHEDULE

Task: H.3 MATERIAL CHARACTERIZATION AND PROPERTIES (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
H. Complete Characterization Testing and Analysis for Wide-Plate WP-1 Material (quenched and tempered A533B steel)																							
I. Complete Characterization Testing and Analysis for Wide-Plate WP-2 Material (low upper-shelf material) (See Milestone H.3.1.E)																							
J. Complete Characterization Testing and Analysis for Wide-Plate WP-3 (A533B steel heat treated to simulate PTSE-1)																							
H.3.2 <u>Metallurgical and Weldment Characterization Studies</u>																							
A. Complete Metallographic and Fractographic Characterizations of Test Vessel Materials (TSE-7, PTSE-1, -2 and -3)			a	b														c			d		
a. Metallography and Fractography for TSE-7																							
b. Fractography for PTSE-1																							
c. Metallography and fractography for PTSE-2																							
d. Metallography and fractography for PTSE-3																							

MILESTONE STATEMENT AND SCHEDULE

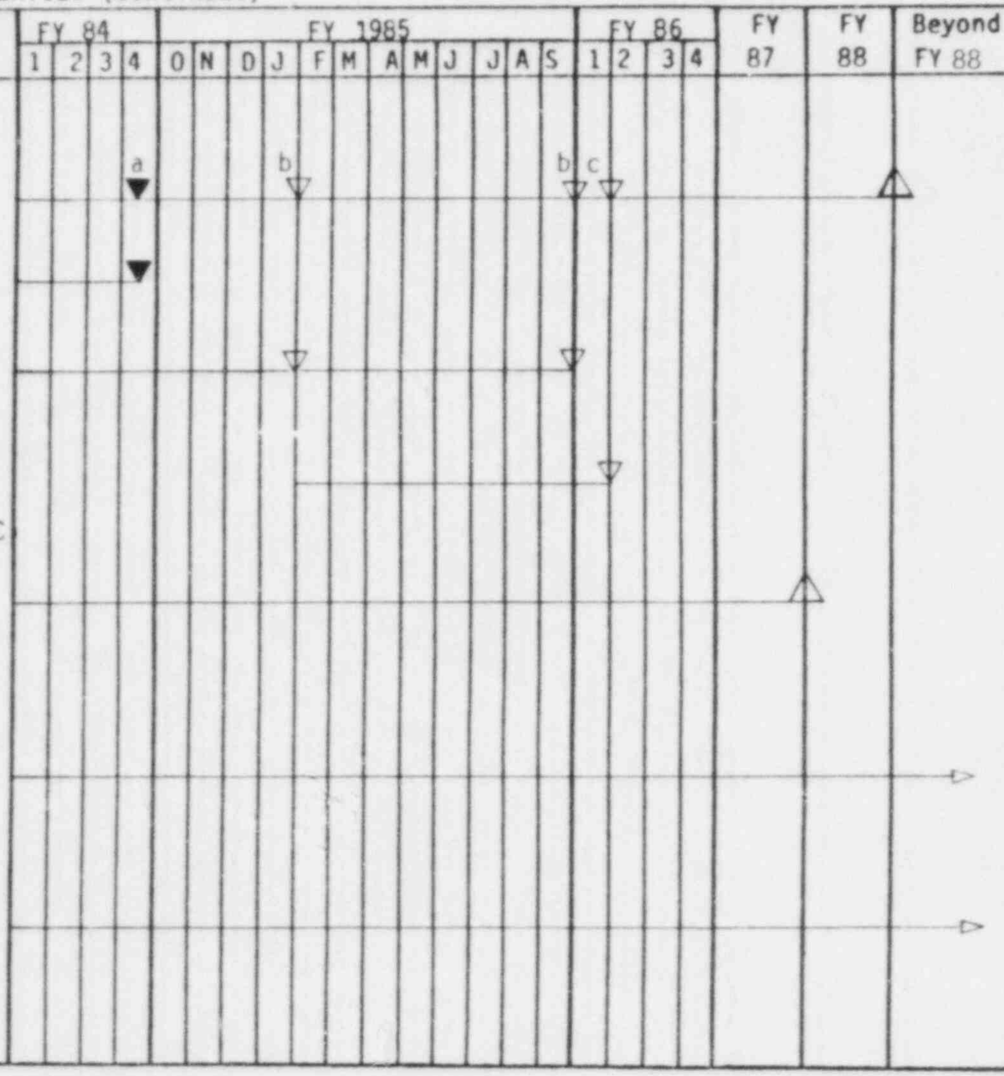
Task: H.3 MATERIAL CHARACTERIZATION AND PROPERTIES (continued)

SUBTASK/MILESTONE	FY 84 (1 2 3 4)	FY 1985 (O N D J F M A M J J A S)	FY 86 (1 2 3 4)	FY 87	FY 88	Beyond FY 88
B. Complete Heat Treatment Studies for PTSE-3 and WP-3						
a. Tempering study for PTSE-3						
b. Tempering study for third series wide-plate specimens, WP-3 (See H.3.1.J.)						
C. Complete Characterization of Weld and Cladding Depositions		a	b c d			
a. Metallography and fractography for phase 1 three-wire clad plates						
b. Metallography for commerical three-wire archival cladding						
c. Metallography for commerical BWR beltline welds and cladding (See Milestone H.3.1.G.c)						
d. Metallography and fractography for phase 2 three-wire clad plates						

MILESTONE STATEMENT AND SCHEDULE

Task: H.3 MATERIAL CHARACTERIZATION AND PROPERTIES (continued)

SUBTASK/MILESTONE



H.3.3 Exploratory Testing

A. Complete Development of Crack-Arrest Test Capabilities

a. Complete ORNL testing in ASTM round robin

b. Complete development of initial alternate crack arrest test methods

c. Complete development of remote crack-arrest specimen and test procedure (See Milestone H.6.5.C)

B. Conduct Exploratory Deformation Investigations to Support Visco-plasticity and Fracture Studies

H.3.4 Codes and Standards Support

A. Advise Standards and Codes Organizations Relative to Changes or Additions Recommended for Reference Curves and Correlations

B. Advise Standards and Codes Organizations Relative to Changes or Additions Recommended for Test Methods

4. HSST TASK H.4 ENVIRONMENTALLY ASSISTED CRACK GROWTH TECHNOLOGY

4.1 Objective

The objective of Task H.4 is to characterize the crack growth rate properties of light-water reactor vessel materials exposed to primary coolant environment, and to provide improved data correlations as appropriate to design codes and regulatory guides.

4.2 Background

In assessing the integrity of nuclear components, the possible presence of cracks or defects is an important consideration. There are two things which can cause cracks to propagate in structures, and both must be considered in design assessments. The first is severe loadings which can cause single or multiple crack jumps, and the second is fatigue loadings, which result from normal plant operation as well as operational transients, and can cause progressive extension of cracks. While the single loadings are essential to consider, the net risk to the structure from these is relatively low, because the high-probability events are low in severity, while the severe events are very low probability. The fatigue loadings result in the highest net risk to the propagation of a flaw in a structure, because they are virtually certain to occur.

This fact was recognized in the early stages of the Heavy-Section Steel Technology Program, and an experimental program was developed to characterize the fatigue crack growth of pressure vessel steels in light-water reactor environments in the late sixties. It was also recognized in the development of flaw evaluation criteria in Section XI of the ASME Code, where fatigue crack growth is one of the key considerations.

This program provided the first verification of Kondo's discovery in 1971 that low-frequency loadings in water environments result in significant acceleration in crack growth rates above those for air environments.[1] The data produced by this program provided the basis for the early incorporation of water-environment effects on crack growth in the ASME Code, in the 1974 edition of Section XI Appendix A.[2] This was the first incorporation of environmental effects on crack growth in a code or standard in the world.

The program continued, with emphasis in the mid-seventies being directed at the effects of environment on plates, forgings, welds, and heat-affected zones, to determine if all behaved similarly. The findings were that they all behaved similarly, but more complete studies revealed that the crack growth rate did not increase in a linear fashion on a logarithmic plot, as with inert environments, but tended to flatten out at a growth rate which was dependent on the frequency of loading. A significant effect of R ratio was also discovered, and verified by other investigators. Again the data produced in this program formed the majority of the data base used to develop a revision to the ASME Code

reference crack growth curves for water environment, which was accomplished in the Winter addendum to the 1980 Code.[3]

Most recent work has resulted in the discovery that the degree of environmental enhancement for pressure vessel steels can be different for different heats of the same steel. It was found that the sulfur content of the steel was the key factor, with high-sulfur levels resulting in significant enhancement, while low-sulfur steels showed only slight enhancement over growth rates observed in air.[4] This was a significant finding, because it shows that the newer, low-sulfur steels used in recent construction will be less susceptible to environmental fatigue.

Another recent discovery of this task was that cracks in pressure vessel steels can grow under constant load in water environments, under certain conditions.[5] The relationship between this mode of subcritical crack growth and fatigue is currently under investigation, and efforts are also being aimed at better defining the conditions under which this growth can occur.

The program is continuing with emphasis now being placed on further understanding the mechanisms involved in the process of environmental enhancement of fatigue crack growth. This understanding is essential to enable accurate predictions of crack growth during service, because it is impossible to model all the types of loadings which can occur. The overall goal remains to develop accurate characterization of subcritical crack growth for use in the assessment of defects found in operating plants.

4.3 Plan of Action

This task continues to be carried out through subcontract with Westinghouse Electric Corporation, Nuclear Technology Division. The work breakdown structure is shown in Fig. 4.1. The first priority continues to be to complete the characterization of the two major known influences on the enhancement of crack growth rates in water environments. These are the influences of the material itself through sulfur content [H.4.1], and the influence of the water environment [H.4.1]. Both of these tasks are now underway.

Another important task is developing a relationship between cyclic-fatigue and static-load crack growth [H.4.2]. This work is interrelated with the mechanisms of environmental enhancement of crack growth, and should lead to considerable improvement in understanding in this area. The key to the ultimate goal of the program is to predict accurately the environmental enhancement of crack growth.

The extension of the characterization studies to piping steels is another key task where the similarities and differences between pressure vessel and piping steels will be investigated [H.4.2]. This must be done to decide whether the present ASME Code reference fatigue crack growth curves are applicable to piping steels. If they are not applicable, a suitable replacement must be developed.

The interrelationships between loading frequency and applied R ratio have been characterized generally by the tests which have already

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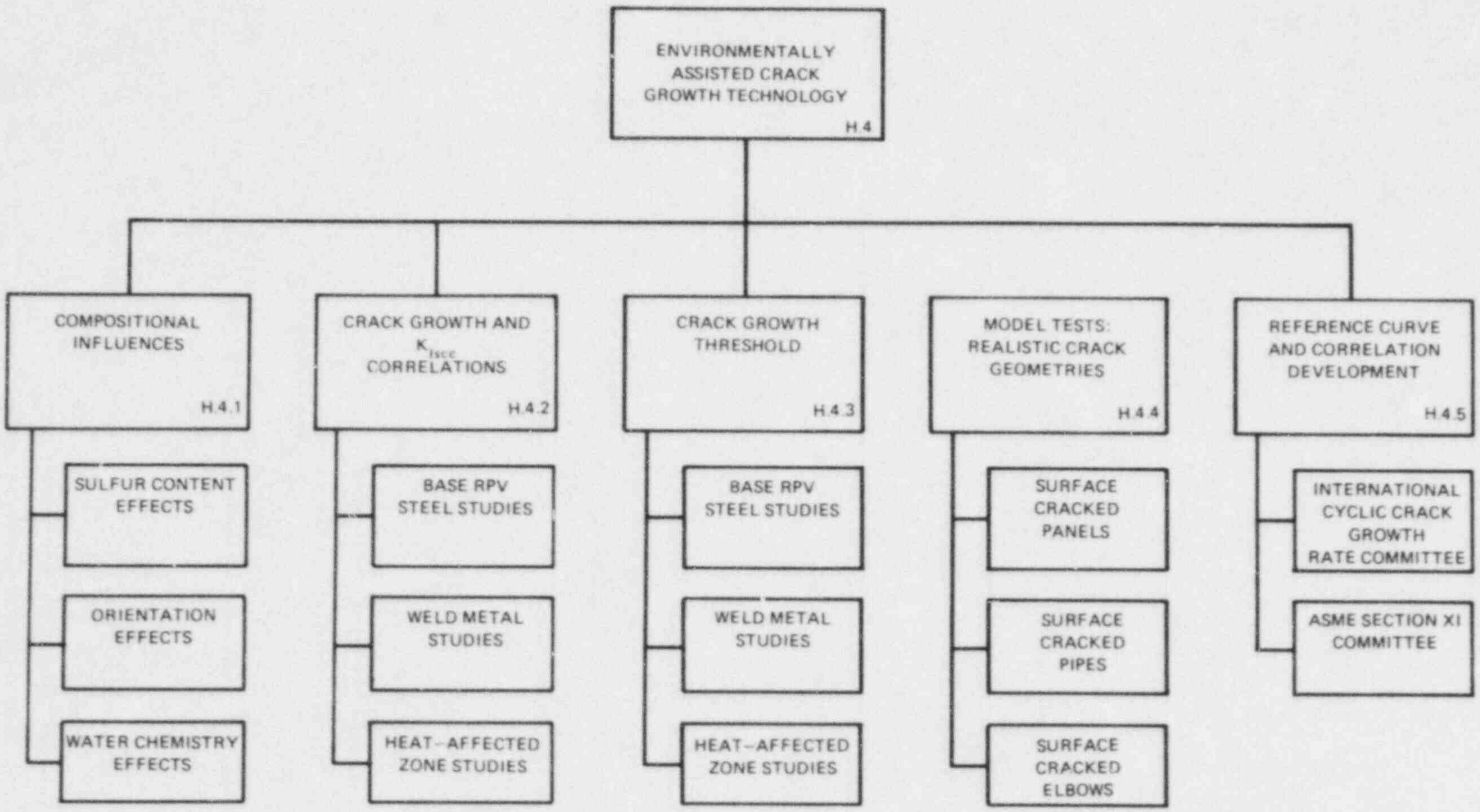


Fig. 4.1 Work breakdown structure for HSST Task H.R Environmentally Assisted Crack Growth Technology.

been completed, but a series of key experiments are needed to complete the characterization. These experiments are needed to define the threshold limits of R ratio and frequency effects, which have not yet been developed. In particular, R ratios greater than 0.75 have seldom (if ever) been attempted in corrosion fatigue testing, and data at frequencies lower than one cycle per minute are very sparse [H.4.3].

The ultimate goal and culmination of the task will be the proposal of revisions to the ASME Code reference crack growth curves for environmental crack growth. Involved with this task is the experimental verification of crack growth predictive methodology through selected model tests. To address the question of the behavior of flaws which might be encountered (and have been encountered) in practice, a series of tests are to be carried out over a three year period. These tests are designed to study the behavior of cracks in a range of ferritic materials, from those which have shown a large effect of environment on crack growth to those which have shown only a small environmental influence. A range of crack shapes will be considered in surface cracked panels, and in later years other geometries and loading types will be considered. In order to attain the range of applied K which can exist in practice, a large test machine will be required, and an MTS test machine with a capacity of 1,000,000 lb is available for dedication to this program. Also under this task is continued participation in the International Cyclic Crack Growth Rate Group, to enable research in this field being performed around the world to be incorporated into any revision of the reference curves.

4.4 References

1. T. R. Mager, D. M. Moon, and J. D. Landes, "Fatigue Crack Growth Characteristics of A533 Grade B Class 1 Plate in an Environment of High-Temperature Primary Grade Nuclear Reactor Water," *Trans. ASME J. Pressure Vessel Technol.*, (November 1976).

2. T. U. Marston (ed.), *Flaw Evaluation Procedures: ASME Section XI*, EPRI Report 719-SR, August 1978.

3. W. H. Bamford, "Technical Basis for Revised Reference Crack Growth Rate Curves for Pressure Boundary Steels in LWR Environment," *Trans. ASME, J. Pressure Vessel Technol.*, 102 (November 1980).

4. W. H. Bamford, "Environmental Cracking of Pressure Boundary Materials, and the Importance of Metallurgical Considerations," in *Aspects of Fracture Mechanics in Pressure Vessels and Piping*, PVP-58, ASME 1982.

5. W. H. Bamford, D. M. Moon, and L. J. Ceschini, "Studies of Statistically and Dynamically Loaded Cracks in Pressurized Water Environment," presented at *Corrosion 83*, Anaheim, California, April 1983, to be published in *Corrosion*.

4.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.4 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE																							
Task: H. 4 ENVIRONMENTALLY ASSISTED CRACK-GROWTH TECHNOLOGY																							
SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
H.4.1 Compositional Influences (Westinghouse)																							
A. Complete Study of Sulfur Effects on Crack-Growth Rates in A533B Steel	a			b																			
a. Complete study of high-sulfur plates																							
b. Complete study of low-sulfur plates																							
B. Complete Study of and Submit Report on Boron Effects on Fatigue Crack Growth																							
C. Complete Development of Electro-Chemical Potential Measurement Systems to Monitor Water Environment		a								b													
a. Complete installation on fatigue chambers																							
b. Submit experience report																							

MILESTONE AND STATEMENT SCHEDULE

Task: H.4 ENVIRONMENTALLY ASSISTED CRACK-GROWTH TECHNOLOGY (continued)

SUBTASK/MILESTONE

H.4.2 Crack Growth and K_{Iscc} Correlations

A. Complete Tests to Correlate K_{Iscc} and Cyclic-Crack Growth

a. Complete tests on two heats of A533 Gr B steel

b. Complete tests on weld metal and heat-affected zone materials

c. Complete tests of two additional heats

d. Complete constant extension rate tests (CERT) for the additional tests

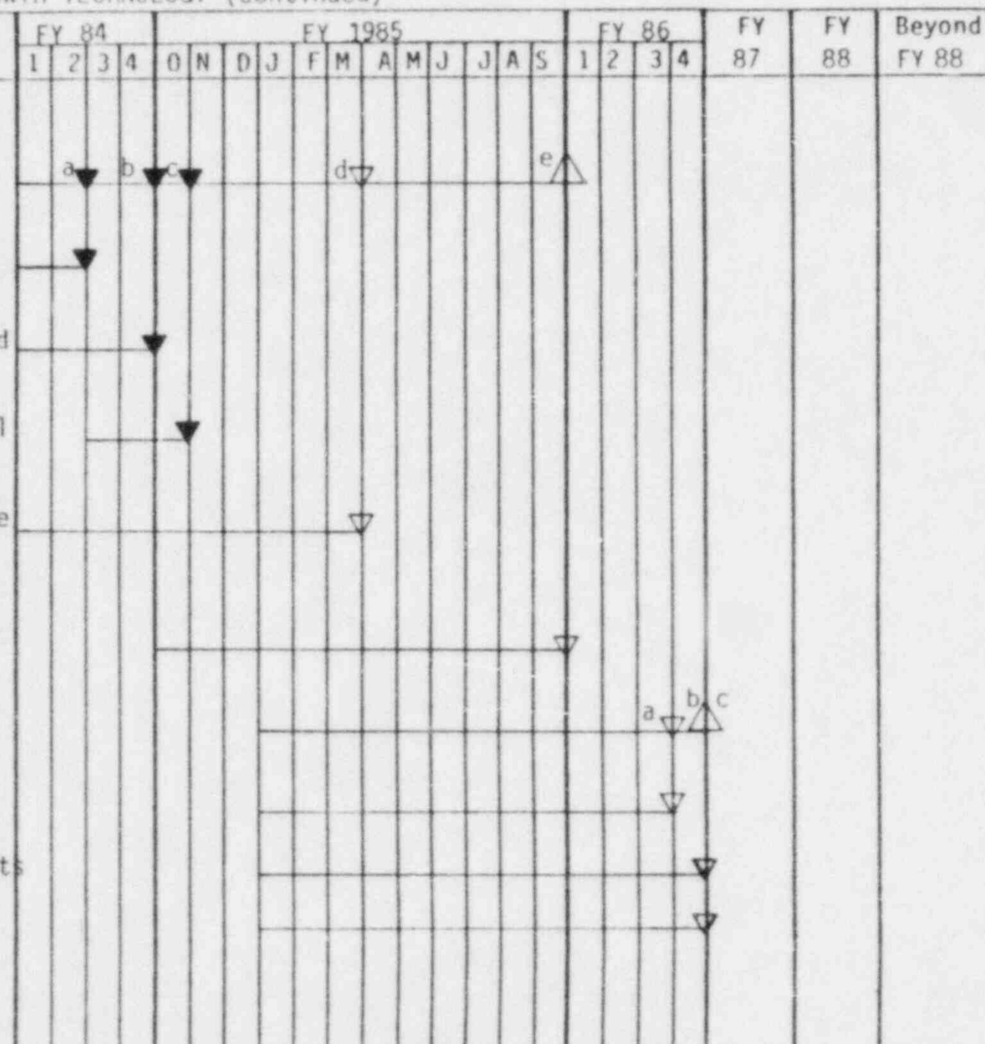
e. Issue report on available K_{Iscc} vs fatigue data

B. Complete Tests for Three Heats of Piping and Structural Steels

a. Complete CERT tests

b. Complete cyclic crack-growth tests

c. Complete comparison of results with RPV steel data



MILESTONE STATEMENT AND SCHEDULE

Task: H.4 ENVIRONMENTALLY ASSISTED CRACK-GROWTH TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84 1	FY 84 2	FY 84 3	FY 84 4	FY 1985 O	N	D	J	F	M	A	M	J	J	A	S	FY 86 1	FY 86 2	FY 86 3	FY 86 4	FY 87	FY 88	Beyond FY 88
H.4.3 Crack-Growth Threshold																							
A. Issue Report on High R Ratio (R≥ 0.7) Crack-Growth Studies																	a ▽		b ▽	△			
a. Complete tests																	▽						
b. Complete saturation assessments																			▽				
B. Issue Report on Low-Frequency Crack-Growth Studies																	a ▽		b ▽	△			
a. Complete tests																	▽						
b. Complete tests and threshold assessments																			▽				
C. Complete Development of Relation Between High R Ratio and Frequency Effects																					△		
H.4.4 Model Tests: Realistic Crack Geometries																							
A. Issue Report on Five Series of Simple Model Tests to Verify Observed Trends																	a ▽	b ▽			c ▽ d ▽ e ▽	f ▽ △	
a. Develop technical plan																	▽						
b. Set up test system for large specimens																		▽					

MILESTONE STATEMENT AND SCHEDULE

Task: H. 4 ENVIRONMENTALLY ASSISTED CRACK-GROWTH TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
A. Issue Report on Five Series of Simple Model Tests to Verify Observed Trends (continued)																							
c. Series 1 tests of high susceptibility material																							
d. Series 2 tests of low susceptibility material																							
e. Series 3 flow rate effects																							
f. Series 4 water chemistry effects																							
g. Series 5 pressurized pipe tests																							
H.4.5 <u>Reference Curve and Correlation Development</u>																							
A. Complete Development of Revised Crack-Growth Rate Curves for RPV Steels																							
B. Maintain Membership on International Cyclic Crack-Growth Rate Committee																							

5. HSST TASK H.5 CRACK ARREST TECHNOLOGY

5.1 Objective

The objective of Task H.5 is to provide crack-arrest toughness data over ranges of temperature and materials to support validation of structural assessment methods, to define structural tests, to validate ASTM test procedures, and to develop procedures for remote testing.

5.2 Background

The pressurized thermal shock issue for Pressurized Water Reactors (PWRs) involves the broadest range of fracture phenomena. The issue addressed the potential for crack initiation at temperatures near the RT_{NDT} in material that has been degraded by irradiation exposure (see Task H.6). This task (H.5) is concerned with understanding the ability of vessel materials to arrest such running cracks as they propagate into regions of tougher material and conditions of increasing values for the stress intensity factor. Postarrest behavior is also important to understanding the entire dynamic fracture process. The overall HSST program is providing data from small specimens and large thermally and pressure loaded vessels to establish the validity of the crack-arrest concept for application to the PTS problem. Task H.5 focuses on generating data in sufficient quantities and over sufficient ranges of conditions to quantify the concepts for representative vessel materials. The elastic, elastic-plastic, and viscoplastic fracture mechanics concepts developed in Task H.2 are essential building blocks for Task H.5.

Calculations of pressure vessels containing a long axial crack and subjected to pressurized thermal-shock (PTS) loading show that high levels of crack-arrest toughness, at temperatures approaching the upper shelf, are needed to ensure vessel safety.[1] In addition, the first intermediate vessel PTS test (see Task H.10) has shown that crack-arrest values higher than previously measured with compact specimens are required for predicting the test results.[2] Few crack-arrest toughness measurements have been made above 175 MPa$\sqrt{m}$, except for those made by Japanese investigators using edge cracked wide-plate (ESSO) specimens.[3,4] Tests of a similar nature have been planned using HSST program materials, and dynamic fracture analysis procedures based on the SWIDAC[5] and SAMCR[6] computer codes are being developed. These analysis procedures have been applied to selected existing Japanese data in order to aid in the planning of HSST tests, and to the results of the first test which was performed in FY 1984. The crack-arrest toughness value of over 300 MPa$\sqrt{m}$ was obtained from this test.

The current procedure for making crack arrest toughness measurements is based on the use of a transverse wedge loaded rectangular compact specimen. A disadvantage of this specimen is that K_I decreases with a/W, forcing the use of high values of K for initiation, with attendant large plastic zones, in order to obtain only moderately high values of K_{Ia} for sufficiently long crack jumps. Despite lubrication,

friction effects between the bottom surface of the specimen and the supporting base complicate the interpretation of data. Inverted wedge cones or partial shims may remedy this problem, but they may also eliminate a compensating phenomenon, which is the delayed release of strain energy stored near the split pins that might otherwise cause displacement oscillations and reinitiation after arrest. These phenomena require further investigation.

5.3 Plan of Action

The plan of action is described below in terms of the three major subtasks that comprise Task H.5. The associated work breakdown structure is shown in Fig. 5.1.

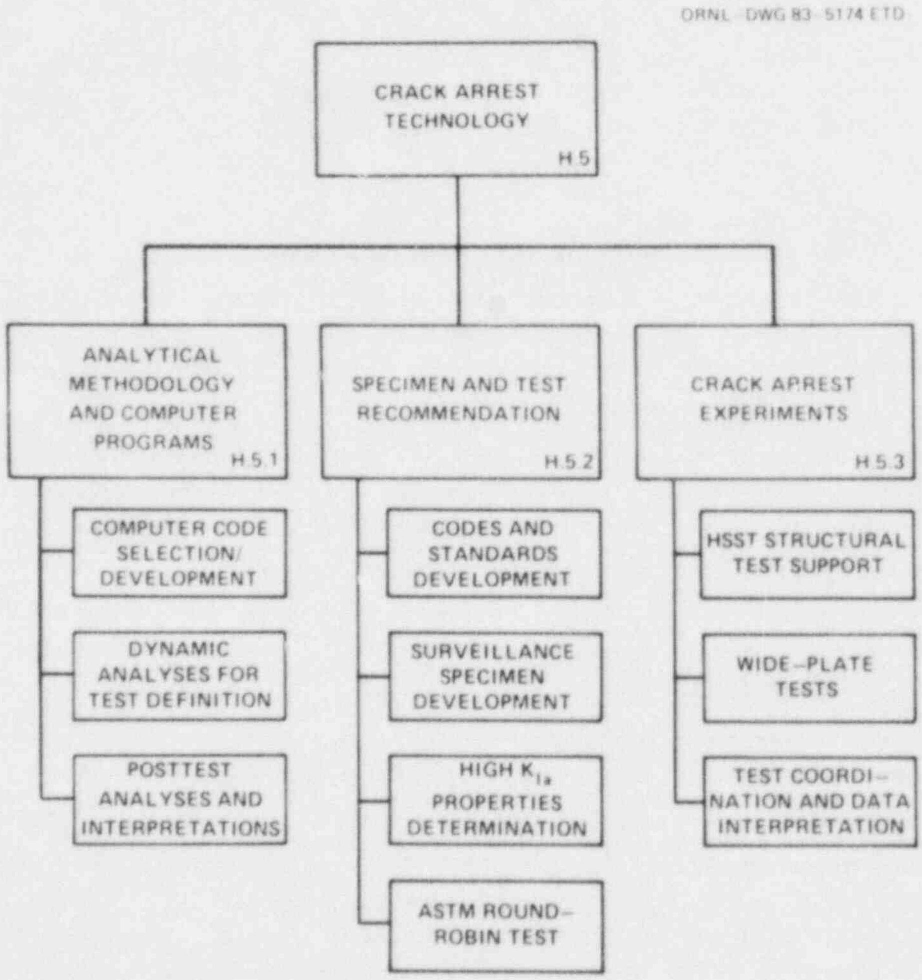


Fig. 5.1 Work breakdown structure for HSST Task H.5 Crack Arrest Technology.

5.3.1 Subtask H.5.1 Analytical Methodology and Computer Programs

The emphasis in this subtask is on analyzing the crack arrest experiments performed in Subtask H.5.3. This is accomplished in concert with computer program development efforts in Task H.2. The planned efforts are aimed mostly at wide-plate tests covering three RPV materials. Pretest analyses are to be performed for test definition. Results from posttest analyses are to be compared with test results to either validate or improve analysis methods. The pretest analyses for the first series of wide-plate crack-arrest tests are using state-of-the-art dynamic analysis procedures. The code SWIDA is being used by the Southwest Research Institute (SWRI) and the SAMCR code by the University of Maryland. ORNL is using both codes. Later analyses may use more advanced theoretical bases, if warranted by the results of Subtask H.2.1, such as viscoplasticity.

During FY 1985, pretest and posttest analyses will be performed for the last five tests in the WP-1 series. Pretest analyses will be performed for the first tests in the second set, WP-2. Additional static and dynamic analyses will be performed of "duck-bill" specimens to explore their potential for use as a small crack-arrest specimen that can yield K_{Ia} values in excess of 200 MPa $\sqrt{m}$ under conditions of a rising K_I field.

5.3.2 Subtask H.5.2 Standardized Specimen and Test Recommendation

Much of the activity in this subtask is being carried out under subcontract with the University of Maryland. One important goal is the development of an ASTM standard on crack arrest testing. The international round-robin testing is being coordinated, including test material distribution, data interpretation, and evaluation of the draft standard, by the University of Maryland under this subtask.

Geometric modifications to the CT specimen may help to eliminate or alleviate the decrease of K_I with a/W. These include variable depth side grooves and a reverse tapered profile. Other specimen designs are being investigated to explore their potential use to generate significant amounts of data at relatively high temperatures and in rising K_I fields.

5.3.3 Subtask H.5.3 Crack Arrest Experiments

This subtask has two emphases. The first relates to the generation of small-specimen crack-arrest data in support of structural tests and analyses. Four materials were under investigation in FY 1984: low upper-shelf weld from ITV-8A, A508 steel from TSE-7, steel from PTSE-1, and A533B steel from the first series of wide-plate tests. During FY 1985 data will be generated on low upper-shelf material to be used in PTSE-2, and second series of wide-plate tests. This will then be followed by tests on the base material to be used in PTSE-3. Much of this K_{Ia} testing is performed at Battelle Columbus Laboratories under subcontract, and a computerized K_{Ia} data base is also maintained at BCL for

publicly held data. A report is to be prepared on the data base in FY 1985.

The second emphasis is to perform three series (six tests each) of crack arrest tests on wide-plate specimens (approximately 1m × 1m × .1m) under tensile load and a transverse temperature gradient. The goal is to measure crack-arrest toughness near or above upper-shelf conditions and with the crack propagating into a rising K_I field. The three test materials are quenched and tempered A533B steel, low upper-shelf weldment, and A533B that has been heat treated to simulate the material used in the first pressurized thermal-shock experiment (PTSE-1). In order to be compatible with the budget assumptions used in this plan, the tests of the first material, which was initiated in FY 1984, will be completed in FY 1985, the second material is to be initiated in FY 1985, and the six tests for the third material scheduled for FY 1986. The testing effort will be performed under an interagency agreement with the National Bureau of Standards, Gaithersburg, Maryland, who was equipment capable of applying the required loads (up to about six million pounds).

5.4 References

1. A. Sauter, R. D. Cheverton, and S. K. Iskander, *Modification of OCA-1 for Application to a Reactor Pressure Vessel with Cladding on the Inner Surface*, ORNL/TM-8649, Oak Ridge National Laboratory.

2. R. H. Bryan, *Pressurized Thermal-Shock Test of 6-in.-Thick Pressure Vessels. PTSE-1: Investigation of Warm Prestressing and Upper-Shelf Arrest*, ORNL- , Oak Ridge National Laboratory, 1985.

3. Y. Nakano, "Stress Intensity Factor During Brittle Crack Propagation and Arrest in ESSO Specimens," *18th National Symposium on X-Ray Study on Deformation and Fracture of Solids*, The Soc. of Mat'l. Science, Japan, July 13-14, 1981.

4. A. R. Rosenfield et al., "BCL HSST Support Program," pp. 7-39 in *Heavy-Section Steel Technology Program Quart. Prog. Rep. for July-September 1982*, ORNL/TM-8369/V3, Oak Ridge National Laboratory.

5. J. Ahmad et al., "Elastic-Plastic Finite Element Analysis of Dynamic Fracture," *Engineering Fracture Mechanics*, Vol. 17, No. 3, pp. 235-246, 1983.

6. C. W. Schwartz, et al., *SAMCR: A Two-Dimensional Dynamic Finite Element Code for the Stress Analysis of Moving Cracks*, NUREG/CR-3891 (ORNL/Sub/79-7778/3) University of Maryland under Subcontract with Oak Ridge National Laboratory, November, 1984.

5.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.5 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

Task: H.5 CRACK-ARREST TECHNOLOGY

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY	FY	Beyond
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
H.5.1 Analytical Methodology and Computer Programs																							
A. Complete Dynamic Analysis of Japanese Wide-Plate K_{Ia} Specimens for Test Definition	▲																						
B. Complete Analysis of A533B Base-Metal Wide-Plate Tests and Comparisons with Data (WP-1)				a ▼			b ▽			c ▽					d △								
a. Complete pretest (state-of-art) analyses for definition of WP-1 test conditions				▼																			
b. Complete posttest analyses for WP-1.1							▽																
c. Complete preliminary visco-plastic analyses of WP-1.1										▽													
d. Complete posttest analyses and comparisons between test data and analyses															▽								
C. Complete Analyses of Low Upper-Shelf Wide-Plate K_{Ia} Tests (WP-2)															a ▽				b ▽△				
a. Complete pretest analyses and definition of test conditions															▽								

MILESTONE STATEMENT AND SCHEDULE

Task: H.5 CRACK-ARREST TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
C. Complete Analyses of Low Upper-Shelf Wide-Plate K_{Ia} Tests (WP-2) (continued)																							
b. Complete posttest analyses and comparisons between test data and alternative analyses (see Milestone H.5.3.C)																							
D. Complete Analyses of Wide-Plate K_{Ia} Tests with A533B Steel Heat Treated to Simulate Degraded Material (WP-3)																			a		b		
a. Complete pretest analyses and definition of test conditions																							
b. Complete posttest analyses and comparisons between test data and alternative analyses (See Milestone H.5.3.D)																							
H.5.2 Specimen and Test Recommendation																							
A. Publish Report on ASTM Round Robin on K_{Ia} Testing (University of Maryland)	a							b		c													
a. Distribute material for ASTM round robin																							
b. Obtain test results from round-robin participants																							

MILESTONE STATEMENT AND SCHEDULE

Task: H. 5 CRACK-ARREST TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84	FY 1985	FY 86	FY 87	FY 88	Beyond FY 88
	1 2 3 4	O N D J F M A M J J A S	1 2 3 4	87	88	FY 88

A. Publish Report on ASTM Round Robin On K_{Ia} Testing (University of Maryland) (continued)

c. Complete analysis of data and evaluate applicability of method

B. Complete Selection of a Laboratory Specimen and Procedures for Obtaining K_{Ia} Data at High Toughness in Rising Toughness Field

a. Identify candidate specimens

b. Complete analytical and experimental evaluation of candidate specimens (e.g., duckbill)

c. Develop data base for WP-1 type material

d. Complete comparisons between available laboratory and base specimen data

C. Complete Engineering Feasibility Study for Locating a 24,000 kip-Tensile Machine at ORNL

MILESTONE STATEMENT AND SCHEDULE

Task: H.5 CRACK ARREST TECHNOLOGY

SUBTASK/MILESTONE

H.5.3 Crack-Arrest Experiments

A. Complete Small-Specimen Crack-Arrest Tests to Support Structural Tests and Analyses (BCL and ORNL)

a. Low upper-shelf weld for ITV-8A and A533 Gr B steel for PTSE-1

b. A508 steel for TSE-7

c. A533B steel for WP-1

d. Low upper-shelf weld for PTSE-2

e. Base material for PTSE-3

B. Complete Series of Six Wide-Plate K_{Ia} Tests to Near the Upper-Shelf Temperature for Quenched and Tempered A533B Steel (Series WP-1)

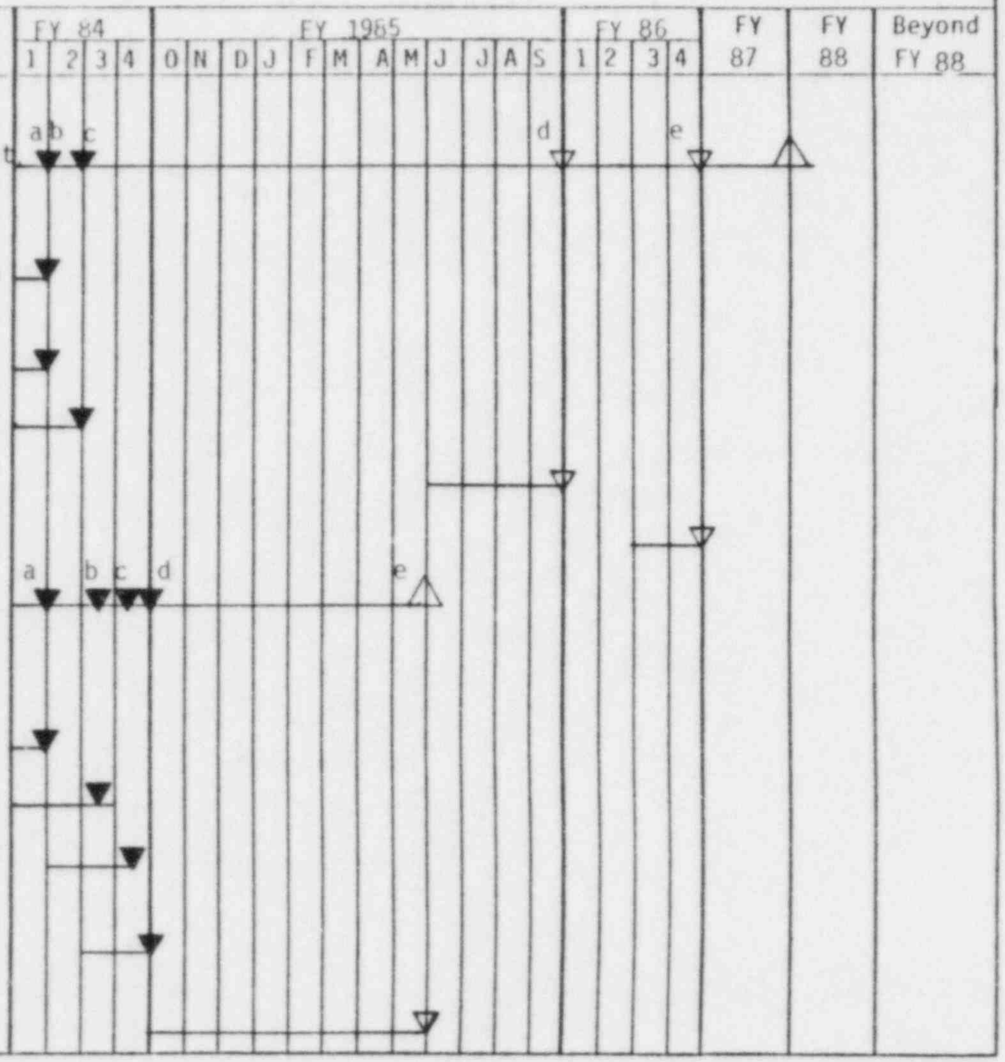
a. Select test site

b. Complete specimen preparations

c. Complete development of NBS facilities and pull plates

d. Complete first highly instrumented tests

e. Complete five remaining tests



MILESTONE STATEMENT AND SCHEDULE

Task: H.5 CRACK-ARREST TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84 1	FY 84 2	FY 84 3	FY 84 4	FY 1985 O	N	D	J	F	M	A	M	J	J	A	S	FY 86 1	FY 86 2	FY 86 3	FY 86 4	FY 87	FY 88	Beyond FY88
C. Complete Series of Wide-Plate K_{Ia} Tests Near the Upper-Shelf Temperature for Low Upper-Shelf Weld (WP-2)												a				b							
a. Procure low upper-shelf material and fabricate test specimens (see Milestone H.10.4.B.b)																							
b. Complete two series of three tests																							
D. Complete Series of Six Wide-Plate Tests Near the Upper-Shelf Temperature for A533B Steel Heat Treated to Simulate PTSE-1 (Series WP-3)																							
E. Maintain Computerized Crack-Arrest Data Base (BCL)				a									b			c							
a. Enter data through FY 1984																							
b. Enter data through FY 1985																							
c. Issue summary report																							

6. HSST TASK H.6 IRRADIATION EFFECTS STUDIES

6.1 Objective

The objective of Task H.6 is to determine the effects of neutron irradiation on the fracture toughness properties of typical nuclear reactor pressure vessel (RPV) materials, including plate, forgings, welds, and stainless steel cladding. The properties of interest include fracture initiation toughness (K_{Ic} and J_{Ic}), crack arrest toughness, (K_{Ia}), ductile tearing resistance (dJ/da), Charpy V-notch (CVN) impact energy, drop-weight NDT, and tensile properties.

6.2 Backgr und

In 1972, the Heavy-Section Steel Technology Program began the first irradiation series in response to the need for information regarding the effects of neutron irradiation on the mechanical properties, particularly fracture toughness, of light-water nuclear reactor pressure vessels. Much research had already been performed in the area of linear elastic fracture mechanics (LEFM) and the effects of specimen size and temperature were known.

The Welding Research Council (WRC) published WRC Bulletin 175 (Ref. 1) in August 1972 and established a reference stress intensity (K_{IR}) curve that was constructed as a lower-bound to K_{Ic}, K_{Id}, and K_{Ia} data available for A533 grade B class 1 and A508 steels. This curve was incorporated into the ASME Code and is used as a guideline for operation of reactors to provide protection against nonductile fracture. The effects of irradiation on fracture toughness were not so well understood. A summary of the HSST irradiation series that have been or plan to be undertaken to improve such understanding is shown in Table 6.1. The work breakdown structure for this task is shown in Fig. 6.1, and it is organized around those series.

Series 1 examined static and dynamic fracture toughness with 100-mm (4TCS) compact specimens. That size was determined to be the maximum that could be efficiently used relative to neutron damage symmetry, gamma heating, etc. Both base plate and submerged-arc weld metal were used. The results of Series 1 showed that fracture toughness versus test temperature curves shifted to higher temperatures by an amount approximated by the shift in CVN test results and that fracture toughness reached high values at the higher test temperatures even after irradiation. However, few specimens of the larger sizes were irradiated and tested; and statistical analysis was not possible.

Series 2 and 3 were conducted to examine the effects of irradiation on the ductile-shelf toughness of submerged-arc welds fabricated with high copper levels and older commercial processes that resulted in a fairly low Charpy upper-shelf energy. The motivation for looking at the upper shelf is the requirement in Appendices G and H of Part 10 CFR50 that the Charpy upper-shelf energy must not fall below 68 J (50 ft-lb) as determined from surveillance specimens. In addition to Charpy impact

Table 6.1. Summary of heavy-section steel technology

HSST irradiation series number	Objective	Materials	Total specimen complement	Reactor i... da...
1	Upper transition, fracture toughness of plate and weld metal	A5333 grade B class 1 (plate 02), submerged-arc weld metal	4TCS — 6 CVCS — 140 CVN — 154 Ten — 34	Battelle ... Reactor, ... Ohio, We... conducte... to 12/13...
2	Ductile shelf, fracture toughness of low CV shelf material	Low shelf submerged-arc weld metal, 61W, 62W, 63W, Cu: 0.29, 0.21, 0.30%	4TCS — 6 1.6TCS — 6 0.8TCS — 12 0.5TCS — 117 CVN — 207 Ten — 27	Bulk Shie... Reactor, ... Tenn., 1... 3/3/77
3	Ductile shelf, fracture toughness of low CV shelf material	Low shelf submerged-arc weld metal, 64W, 65W, 66W, 67W, Cu: 0.35, 0.22, 0.42, 0.27%	4TCS — 6 1.6TCS — 6 0.8TCS — 12 0.5TCS — 117 CVN — 207 Ten — 27	Bulk Shie... Reactor, ... Tenn., 1... 3/29/78
4	Ductile shelf, fracfracture toughness of state-of-the-art weld weld material	A533 grade B class 1 (plate 02), current practice submerged-arc weld metal, 68W, 69W, 70W, 71W, Cu: <0.10, two FRG materials	1TCS — 240 CVN — 348 Ten — 52	Bulk Shie... Reactor, ... Tenn., 1... 7/25/82
5	K_{Ic} curve shift, compare with CVN curve shift; K_{Ic} values high as possible	Submerged-arc weld metals, Cu: 0.25, 0.35%, no copper-coated electrodes, copper added to melt	4TCS — 16 2TCS — 28 1TCS — 60 CVN — 76 Ten — 24 DWT — 16	Oak Ridge ... Reactor, ... Tenn.; i... began 5/... expected ... of 9/85
6	Crack arrest toughness	Submerged-arc weld metals, Cu: 0.25, 0.35%, no copper-coated electrodes, copper added to melt	Preliminary: 2TCA — 8 1TCA — 16 0.5TCA — 30	Planned: ... Research... Oak Ridg...
7	Stainless steel cladding, fracture toughness of submerged-arc stainless steel cladding	309/308 single-wire oscillating and 308 three-wire series arc	Planned: CVN — 110 Ten — 30 0.5TCS — 48	Nuclear S... and Tech... Facility... New York...

(HSST) irradiation program — December 1984

adiation s	Neutron fluence, n/cm^2 (E > 1 MeV)	Irradiation temperature, °C (°F)	Comments
serach olumbus, inghouse 10/20/72 3	$2.2—7.0 \times 10^{19}$	270—300 (515—570)	Static and dynamic fracture toughness tests conducted. Status: program completed.
ing ak Ridge, 15/76 to	$0.4—2.1 \times 10^{19}$	233—343 (450—650)	Temperature extremes and lower fluences were for smaller specimens. Status: testing completed, final report in preparation.
ing ak Ridge, 19/77 to	$0.4—1.2 \times 10^{19}$	233—310 (450—590)	Temperature extremes and lower fluences were for smaller specimens. Status: testing completed, final report in preparation.
ing ak Ridge, 18/79 to	$0.5—2.7 \times 10^{19}$	288 (550)	Status: Irradiations completed, bulk of testing completed.
esearch ak Ridge, adiations /84, ompletion	Target: 1.7×10^{19}	Target: 288 (550)	Large (8TCS, 6TCS) unirradiated specimens tested to obtain high K_{Ic} values. Status: irradiations began 5/11/84.
ak Ridge eactor, Tenn.	Target: 1.7×10^{19}	Target: 288 (550)	Status: Irradiations begin about 10/85.
ence logy Buffalo,	Target: 1, 2, and 5×10^{19}	Target: 288 (550)	Status: Single-wire cladding irradiation and testing completed. Three-wire will be irradiated in FY 1985 and FY 1986.

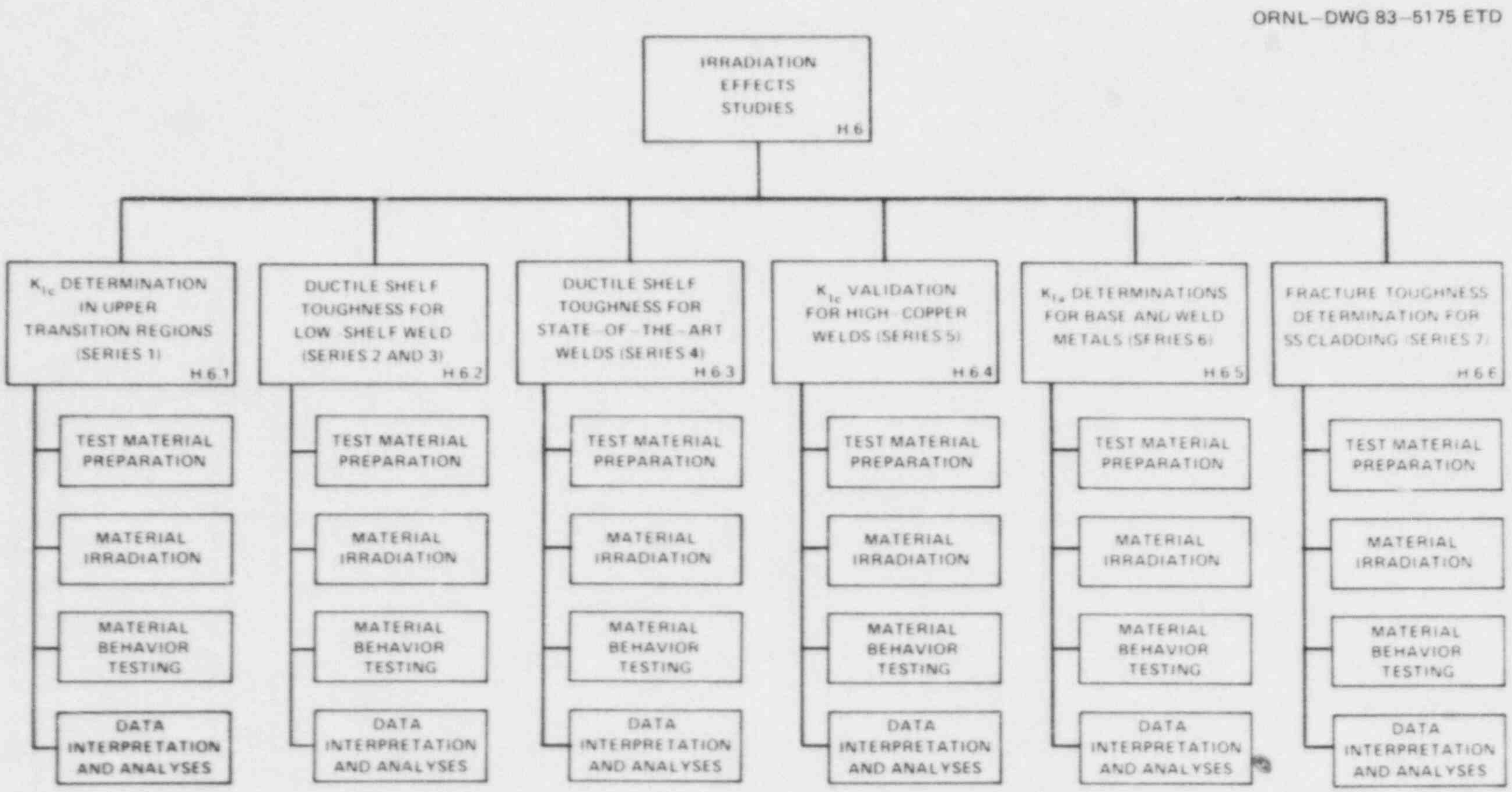


Fig. 6.1 Work breakdown structure for HSST Task H.6 Irradiation Effects Studies.

and tensile specimens, compact specimens ranging from 12 to 100 mm were tested to determine toughness using elastic-plastic test and analysis techniques. The results of Series 2 and 3 showed that the tearing modulus, as measured by elastic-plastic test and analysis, can decrease to very low values after irradiation and can be approximated by the decreases in Charpy impact test upper-shelf energies. These analyses and correlation efforts are still in progress.

Series 4 is also a ductile shelf study except that submerged-arc welds fabricated with low copper content and current practice welding procedures were examined as well as the same plate material (HSST plate 02) used in Series 1. Low upper-shelf materials from the Federal Republic of Germany are included in that study. Series 4 includes only 25-mm-thick compact specimens (1TCS) in addition to the Charpy and tensile specimens. Sufficient specimens exist to perform a statistical analysis of the results, a factor which was lacking in the previous series. The irradiations were completed in 1982 and testing is almost completed. Preliminary results show that low copper and nickel contents result in virtually no effect of irradiation on fracture toughness as measured by K_{IC} and J_{Ic}. The testing program is a cooperative effort between ORNL and Materials Engineering Associates (MEA).

Series 5 was conceived to validate the amount and shape of the K_{Ic} curve shift as a consequence of neutron irradiation. Currently, estimates of the K_{Ic} curve shift are based on results from Charpy impact testing with the assumption that the shift of the Charpy toughness curve to higher temperatures can be applied directly to the K_{Ic} curve. This is much the same as the objectives of Series 1 except that sufficient specimens of various sizes up to 100 mm thick will be tested to allow for statistical analysis. Also, drop-weight specimens will be included in the irradiation program. It is not yet known where they will be tested. Submerged-arc welds have been fabricated for inclusion in this series with two weld metals containing different, high-copper contents. The copper was added to the melt before being drawn into wire to provide a uniform copper content in the weld metal, compared to the less-uniform distribution obtained with copper-coated wire.

Series 6 will determine the effect of irradiation on the pressure vessel material's ability to arrest a rapidly propagating flaw. The long-awaited development of an ASTM test method for determining crack arrest fracture toughness, K_{Ia}, has now led to a draft standard; and this has prompted the planning for Series 6 as a natural follow-on to Series 5 on crack initiation. The submerged-arc weldments used for Series 5 will also be used for Series 6. Crack arrest toughness is considered by many to represent the minimum toughness of a material, and it is essential that the effects of irradiation on that property be understood.

Series 7 is designed to determine the effects of irradiation on pressure vessel stainless steel cladding. Cladding is applied to the reactor vessel to minimize corrosion products in the coolant. Analyses of certain thermal-shock scenarios have been inhibited by a lack of information regarding the fracture resistance of the cladding. The little information available in the literature indicates that stainless steel cladding may undergo severe embrittlement as a consequence of neutron irradiation. The plan for Series 7 includes submerged-arc cladding

applied by the single-wire oscillating and the three-wire series arc processes. Charpy impact, precracked Charpy, tensile, and 12.5 mm compact specimens (0.5TCS) will be utilized. The single-wire cladding CVN and tensiles irradiation to 2×10^{23} neutrons/m^2 has been completed with results showing irradiation effects to be dependent on the degree of base metal dilution during fabrication.

6.3 Plan of Action

6.3.1 Series 1

This program is completed.

6.3.2 Series 2 and 3

All planned testing on these series have been completed. A few tensile specimens from each weld of Series 3 will be retained for testing with extensometry to obtain stress-strain information in addition to determinations of strength and ductility. A comprehensive report will be prepared to describe the results and analyses.

6.3.3 Series 4

The irradiations were completed in July 1982. The testing is under way at both ORNL and MEA in accordance with the test plan described below.

In response to Nuclear Regulatory Commission requests, testing of specimens from capsules A, B, and C of the Fourth HSST Irradiation Study are being conducted by two laboratories, ORNL and MEA. Testing by MEA is conducted at the Nuclear Science and Technology Facility of the State University of New York at Buffalo. The plan provides for an approximately equal division of testing between the two facilities.

The primary objective of the Fourth HSST Irradiation Series is to provide statistical data on elastic-plastic fracture toughness (upper-shelf toughness) of nuclear pressure vessel steels and welds. The plan provides for at least half of the testing to be conducted at "upper-shelf" temperatures. In order to provide data for statistical analyses, the test plan also provides for five Charpy impact tests and five 1TCS tests of each material at each of six selected test temperatures unless fewer specimens are available.

The Charpy impact tests from capsules A, B, and C have been completed by ORNL and MEA. Statistical analyses have been conducted and show excellent agreement between laboratories.

Test temperatures for 1TCS specimens were chosen as follows: (1) NDT temperature [from Charpy 41-J (30-ft-lb) transition temperature] or less such that a valid K_{Ic} is obtained, (2) high in the "transition" range (J-R test), (3) "upper knee," and (4) three equally spaced temperatures on the "upper shelf," the highest test temperature being 288°C (550°F).

Two tensile tests have beeen conducted at each test temperature. The irradiated tests were conducted from about the NDT temperature (Charpy 30-ft-lb transition temperature) to 288°C (550°F). Additional test pairs will be conducted at temperatures spaced approximately equally between the minimum and maximum test temperatures. Unirradiated tests were conducted at temperatures as low as necessary to provide the strength information needed for the analysis of K_{Ic} tests.

The testing sequence is as follows:

1. Preliminary Charpy impact tests to determine the probable 41-J transition temperature and "upper-knee" temperature: This has been completed for HSST-02 (A533B-1) and the four welds, both irradiated and control.
2. Complete Charpy impact tests: These have been completed.
3. Tensile tests at temperatures selected on the basis of the CVN results: Tensile tests have been completed.
4. Control (unirradiated) 1TCS tests:
   a. valid K_{Ic} tests
   b. "upper-shelf" tests (J-R tests)
   c. "upper-transition" tests (K_J tests)
   d. "upper-knee" tests (J-R tests)
5. Irradiated 1TCS tests:
   a. valid K_{Ic} tests
   b. "upper-shelf" tests (J-R tests)
   c. "upper-transition" tests (K_J tests)
   d. "upper-knee" tests (J-R tests)

The test plan for unirradiated 1TCS tests have been completed and most irradiated tests are completed. A few unirradiated and irradiated specimens of each material are available for testing following analyses of the results obtained to date.

6.3.4 Series 5

6.3.4.1 Objectives. The primary objective of this program is to obtain valid fracture toughness (K_{Ic}) curves for two nuclear pressure vessel materials irradiated at 288°C (550°F). The largest practical compact specimen that can be irradiated is a 4TCS. In the irradiated condition, a 4TCS specimen of these materials can measure K_{Ic} to a level of about 130 MPa$\sqrt{m}$ (120 ksi$\sqrt{in.}$). Smaller specimens (2T and 1TCS) would be employed to measure valid K_{Ic} values down to about 50 MPa$\sqrt{m}$ (45 ksi$\sqrt{in.}$).

In support of the primary objective, there are several supporting objectives necessary to realizing the potentials of this program. These supporting objectives include the following:

1. Unirradiated K_{Ic} data for the two irradiated materials. Large specimens (8TCS) will be required for measurement of a valid K_{Ic} of 130 MPa$\sqrt{m}$ in the unirradiated material. Smaller specimens (6T, 4T, 2T, and 1TCS) would be employed for the lower K_{Ic} levels.

2. Both unirradiated and irradiated 1T and 2TCS specimens would be tested by the J-integral method to provide K_J data above the valid K_{Ic} capacity of these specimen sizes. These data would help assure valid test results for the more expensive (and valuable) larger specimens and would also provide a much needed comparison of K_J results with K_{Ic} results.
3. Tensile properties of unirradiated and irradiated materials will provide data both for determining test parameters for the fracture toughness tests and for analysis of the fracture toughness data.
4. Charpy V-notch impact tests of unirradiated and irradiated materials will provide data to guide the setting of fracture toughness test parameters and correlate radiation-induced shifts of CVN transition temperature with radiation-induced shifts of the K_{Ic} curves. Drop-weight tests will be used to index the nil-ductility temperature (NDT) of the Charpy curve for unirradiated and irradiated materials.

The test plan will provide for statistical analysis of the results insofar as is practical.

6.3.4.2 <u>Materials</u>. A radiation-induced temperature shift of toughness properties (CVN transition temperature and K_{Ic} curve) of 85°C (150°F) will provide a significant and meaningful separation of the unirradiated and irradiated properties. The radiation-induced shifts of RPV materials are functions primarily of fast-neutron fluence, irradiation temperature, and copper content of the steels. Since an irradiation temperature of 288°C (550°F) has been chosen as most significant for both the industry and regulatory bodies, the most practical fast-neutron fluence and copper contents must be selected. The target fast-neutron fluence will be 1.7×10^{23} n/m^2 (E > MeV). Two submerged arc weldments with nominally A533 grade B composition are being fabricated for this program; the target copper contents are 0.25 and 0.35%. The weldments will be fabricated and stress-relieved according to commercial practice in 216-mm-thick plate. About 14 lin m of each weldment will be fabricated for the program.*

6.3.4.3 <u>Specimen complement</u>. The valid K_{Ic} measuring capacity of a compact specimen is a function primarily of yield stress and specimen thickness. The yield stress for the unirradiated submerged-arc weldments should be about 480 MPa (70 ksi) and about 620 MPa (90 ksi) for the irradiated material. Valid fracture toughness tests will be conducted at five target toughness levels and, therefore, five estimated test temperatures relative to the NDT. The specimen sizes (K_{Ic} and K_J) and number of specimens for each toughness level (and estimated test temperature) are presented in Table 6.2 for unirradiated and irradiated materials. The complement of Charpy V-notch, miniature tensile (MT) and drop-weight specimens is presented in Table 6.3.

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*This amount of weldment will also provide material for the Series 6 program on irradiation effects on crack arrest (K_{Ia}); see Section 6.3.5.

Table 6.2. Compact specimen complement for the K_{Ic} curve shift program

Test/specimen types	Number of specimens per material							
	Target K_{Ic} (MPa·$\sqrt{m}$) and Estimated T-NDT (°C)	49 −56	66 −14	88 3	110 11	132 19	Spare specimen	Totals
		Unirradiated						
K_{Ic}, 1T-CS[a]		10					*b*	10
K_{Ic}, 2T-CS[a]			4				*b*	4
K_{Ic}, 4T-CS[a]				4			*b*	4
K_{Ic}, 6T-CS[a]					2		2	4
K_{Ic}, 8T-CS[a]						2	2	4
K_J, 1T-CS			6	6	6	6	*b*	24
K_J, 2T-CS			4	4	4	4	*b*	12
K_J, 4T-CS					4[c]	4[c]	*b*	8[c]
		Irradiated						
K_{Ic}, 1T-CS[a]		4	6				2	12
K_{Ic}, 2T-CS[a]				6			4	10
K_{Ic}, 4T-CS[a]					4	4		8
K_J, 1T-CS				6	6	6		18
K_J, 2T-CS					4	4		8

[a] Valid tests.

[b] Material for additional specimens is available.

[c] These tests would be valuable if material is available.

The number of specimens listed in Tables 6.2 and 6.3 are per material: the total numbers of specimens are twice the numbers listed. The numbers of specimens specified are based on a consideration of statistical requirements and within the constraints of probable irradiation facilities.

6.3.4.4 Test plan. Two organizations, MEA and ORNL, will participate in the testing program. Because of testing equipment limitations, all testing of irradiated 4TCS specimens and unirradiated 6TCS and 8TCS specimens will be conducted by MEA and all tensile tests, unirradiated and irradiated, will be conducted by ORNL. Due to cost considerations (MEA does not have out-of-cell test facilities), about 75% of the Charpy testing will be conducted at ORNL. All material characterization will be conducted at ORNL. A fraction of the unirradiated specimens, CVN, 1T-K_{Ic}, and 1T-K_J, will be retained for testing with the corresponding irradiated specimens.

Table 6.3. Charpy V-notch impact, drop-weight, and tensile specimens for K_{Ic} curve shift program

Estimated test temperature relative to NDT (T-NDT) (°C)	Number of specimens per material					
	Charpy-V specimens		Drop-weight specimens		Tensile specimens	
	Unirradiated	Irradiated	Unirradiated	Irradiated	Unirradiated	Irradiated
Scoping tests[a]	16	13	12	16		
−56					6	3
−28	5					
−14					6	3
0	10	10				
+24					6	3
+40	10	5				
+70	10	5			6	3
+111	10					
+167	10	5				
+222	10					
Spares		18				4
Totals (per material)	81	56	12	16	24	16

[a]Varied temperatures to determine NDT and "upper-knee" temperatures.

The testing sequence is as follows:

1. Materials inspection and characterization, including ultrasonic inspection, chemical analyses, metallurgical structure studies, standard tensile tests, drop-weight tests, and preliminary CVN tests.
2. Testing of a major fraction of unirradiated specimens:
   a. CVN tests,
   b. tensile tests,
   c. J_{Ic} (K_J) tests of 1TCS specimens,
   d. K_{Ic} tests of 1TCS specimens,
   e. J_{Ic} (K_J) tests of 2TCS specimens,
   f. K_{Ic} tests of 2TCS specimens,
   g. J_{Ic} (K_J) tests of 4TCS specimens,
   h. K_{Ic} tests of 4TCS specimens, and
   i. K_{Ic} tests of 6TCS and 8TCS specimens.
3. Testing of irradiated specimens and remainder of unirradiated specimens:
   a. CVN tests,
   b. tensile tests,
   c. drop-weight tests,
   d. J_{Ic} (K_J) tests of 1TCS specimens,
   e. K_{Ic} tests of 1TCS specimens,
   f. J_{Ic} (K_J) tests of 2TCS specimens,
   g. K_{Ic} tests of 2TCS specimens, and
   h. K_{Ic} tests of 4TCS specimens.

All irradiations are planned for the Oak Ridge Research Reactor poolside facility. A prototype 4T capsule was designed and tested to provide information for dosimetry analyses and thermal shield design. The first four capsules (eight 4TCS) have been irradiated.

6.3.5 Series 6

6.3.5.1 Objectives. The primary objective of this series is to obtain valid crack arrest toughness (K_{Ia}) curves for two nuclear pressure vessel materials irradiated at 288°C. Supporting objectives include the following:

1. The existing K_{Ia} data base for unirradiated and irradiated materials will be enlarged.
2. The upward temperature shift and the shape of the K_{Ia} curve, relative to RT_{NDT}, due to irradiation will be verified.
3. The K_{Ia} results will complement the K_{Ic} results obtained with the same well characterized material.
4. Lower-bound data will be obtained for analysis of the K_{IR} curve under irradiated conditions.

6.3.5.2 Materials. The materials for this series are the same submerged-arc weldments used for the K_{Ic} program of Series 5.

6.3.5.3. Specimen complement. As with K_{Ic}, the valid K_{Ia} measuring capacity of a compact crack arrest (CCA) specimen is a function primarily of yield stress and specimen thickness. The yield stress for the

unirradiated submerged-arc weldments should be about 480 MPa (70 ksi) and about 620 MPa (90 ksi) for the irradiated material. A 50-mm-thick CCA specimen (2TCCA) can provide a "valid" K_{Ia} measurement up to about 187 MPa$\sqrt{m}$ (170 ksi$\sqrt{in.}$) for the irradiated material and about 145 MPa$\sqrt{m}$ (132 ksi$\sqrt{in.}$) for the unirradiated material. A 3TCCA specimen can measure up to about 190 MPa$\sqrt{m}$ for unirradiated material, about the same as that for the irradiated 2TCCA specimen.

The specimen complement is preliminary, but currently we envision three capsules will be irradiated in two series. The first capsule will contain the 0.5TCCA (30) and 1TCCA specimens while the second and third capsules will contain the 2TCCA (9 each) specimens. The small specimen capsule will be irradiated first by itself, and the large specimen capsules will be irradiated together. This complement should provide enough specimens to allow for statistical analyses in the same manner as Series 5. The number of specimens given for each size is the total to be divided as equally as possible for two materials.

6.3.5.4 Test plan. All material characterization will be performed by ORNL as part of the Series 5 program. All irradiations are planned for the Oak Ridge Research Reactor poolside facility. The testing plan is yet to be formulated, but it will generally follow the concept of testing small specimens to establish likely test temperatures for the larger specimens. Test procedures will follow the latest ASTM standard available at the time.

6.3.6 Series 7

6.3.6.1 Objectives. The objective of this series is to obtain toughness properties for two types of stainless steel cladding in the unirradiated and irradiated conditions. The properties to be obtained include tensile, Charpy V-notch impact, and J-integral toughness (using precracked Charpy specimens and 0.5TCS specimens). The goal is to evaluate irradiated weld-metal cladding representative of that used in early PWRs that are being evaluated for their fracture resistance under overcooling situations.

6.3.6.2 Materials. The materials for this study are stainless steel claddings, nominally type-308, deposited on A533, grade B, class 1 steel plate using two weld cladding procedures. The cladding and stress relief treatment will duplicate commercial procedures as closely as possible. However, to permit fabrication of mechanical test specimens of the cladding, cladding thickness will be about 15 mm, obtained by multilayer deposition of the cladding. This is thicker than the usual pressure vessel cladding thickness (4 to 6 mm), but it should represent multi-layer cladding. Materials will be characterized by welding parameters, chemical composition, metallographic examination, and mechanical properties.

The two weld cladding procedures chosen for this study are the single-wire oscillating procedure and the three-wire series-arc procedure. The primary differences between these procedures is in heat input and the resulting amounts of base metal dilution of the stainless steel cladding.

In the single-wire oscillating procedure, the first layer of cladding is deposited using type-309 weld wire and additional layers are

deposited using type-308. In the current instance, this procedure used high heat input during welding and resulted in considerable base metal dilution of the lower layers of cladding. Both the cladding near the base metal (higher dilution) and the top layers of cladding (little dilution) have been studied. This material has been fabricated at ORNL with the identical weld wire, flux, welding conditions, and heat treatment as used for fabrication of the clad beams for the "cladding evaluation program" (Task H.7). Irradiation and testing of the first two groups of specimens has been completed. Charpy impact and tensile tests have shown that the highly diluted cladding layer (adjacent to base metal) experienced substantial radiation damage, while the cladding layers not diluted with base metal showed almost no irradiation effects (Ref. 2).

In the three-wire series-arc procedure, weld wires of different chemistries, e.g., types 308, 309, and 312 stainless steel, are independently fed into the welding arc. Feed rates for each wire are adjusted to obtain the desired cladding composition. This procedure uses low heat input and results in low base metal dilution of the stainless steel cladding. A contract has been placed with Combustion Engineering Corporation for preparation of the three-wire series-arc clad test plate with delivery expected in early FY 1985.

6.3.6.3 Specimen complement. Charpy V-notch, precracked Charpy, miniature tensile, and 0.5TCS specimens will be fabricated. Irradiation capsules will contain either 20 Charpy (V-notch or precracked) and six tensile specimens or twelve 0.5TCS specimens.

6.3.6.4 Test plan. Table 6.4 presents the cladding irradiation program summary. The first two capsules contained CVN and tensile

Table 6.4. Cladding irradiation program summary

Capsule	Type cladding	Specimen complement	Target fluence (n/m^2)
A	1-wire, type 309	20 CVN, 6 tensile	2×10^{23}
B	1-wire, type 308	20 CVN, 6 tensile	2×10^{23}
C	3-wire and 1-wire, type 309	12 0.5TCS (6 ea.)	2×10^{23}
D	3-wire	20 CVN, 6 tensile	2×10^{23}
E[1]	3-wire	12 0.5TCS	2×10^{23}
F	3-wire	20 CVN, 6 tensile	1×10^{23}
G	3-wire	12 0.5TCS	1×10^{23}
H	3-wire	20 CVN, 6 tensile	5×10^{23}

Note:

[1]Capsule E included in original plan, 3-wire compacts are now contained in capsule C in lieu of 1-wire type 308.

specimens, type 309 (bottom layer) in one capsule and type 308 (top layer) in the other, and they were irradiated to a fluence of 2×10^{23} n/m^2. The irradiations are conducted by MEA in the Nuclear Science and Technology Facility (NSTF) reactor at the University of Buffalo. The NRC currently has a dosimetry program under way for core positions B4 and C2 in the NSTF, and it is those core positions that are being utilized for the cladding irradiations. Charpy impact specimens are tested to obtain a full toughness curve. Tensile specimens are tested from room temperature to 288°C. A capsule containing 0.5TCS specimens from the layer of singe-wire clad material is also planned and will be irradiated because the C_v results indicate significant degradation of toughness.

The three-wire cladding from Combustion Engineering will be available in early FY 1985. Current plans for irradiations include C_v and tensile specimens to be exposed to fluences of 1, 2, and 5×10^{23} n/m^2. The irradiation at 2×10^{23} n/m^2 should be completed by mid-FY 1985 with testing completed by the end of FY 1985. We have also planned for irradiation of small compact specimens (0.5TCS) from the three-wire cladding to fluences of 1 and 2×10^{23} n/m^2. The capsule irradiation sequence will be planned so that the results at 2×10^{23} n/m^2 can be used to assess the need for data at 1×10^{23} n/m^2. The irradiation sequence is presented in Fig. 6.2. A complementary laboratory testing program will be conducted to include the investigation of subsize specimens. The objective of that effort is to ascertain the feasibility of irradiating and testing specimens sectioned from "regular" cladding (i.e., one- or two-layer cladding as normally applied to a vessel) for comparison with the full-size specimens sectioned from the "built-up" cladding. Additionally, a similar study will be performed with composite specimens sectioned to allow for notch or crack-tip placement at specific regions within the clad or heat-affected zone.

ORNL–DWG 85–4548 ETD

CVN and Tensile	←---- A ----→	←---- D ----→		←-- F --→
CVN and Tensile	←---- B ----→	←---------------- H ----------------	----→	
0.5TCS		←---- C ----→		←-- G --→

Fig. 6.2 Cladding irradiation sequence. [Capsules F and G (1×10^{19} n/m^2) will be irradiated if results of capsules D and E (2×10^{19} n/cm^2) justify].

The recommended testing sequence is as follows:

1. Materials inspection and characterization.
2. Test unirradiated specimens.
   a. tensile tests,
   b. CVN tests,
   c. dynamic PCCVN tests (about half of the PCCVN specimens),
   d. static bend tests of PCCVN specimens (remainder of PCCVN specimens), and
   e. J-R tests on 0.5TCS specimens.
3. Test irradiated specimens.
   a. tensile tests,
   b. CVN tests
   c. dynamic PCCVN tests,
   d. static PCCVN tests, and
   e. J-R tests on 0.5TCS specimens.

A metallurgical study to characterize the material as well as scanning electron fractography to aid in understanding fracture mode under all test conditions are planned in conjunction with the irradiation and testing activities. A clad reactor vessel nozzle cutout has been procured from Combustion Engineering, and the suitability of that material for the testing and irradiation programs will be determined.

6.4 References

1. PVRC Ad Hoc Task Group on Toughness Requirements, *PVRC Recommendations on Toughness Requirements*, Welding Research Council, WRC Bulletin 175, August 1972.

2. W. R. Corwin, R. G. Berggren, and R. K. Nanstad, *Charpy Toughness and Tensile Properties of a Neutron-Irradiated Stainless Steel Submerged Arc Weld Cladding Overlay*, NUREG/CR-3927 (ORNL/TM-9309), Oak Ridge National Laboratory, September 1984.

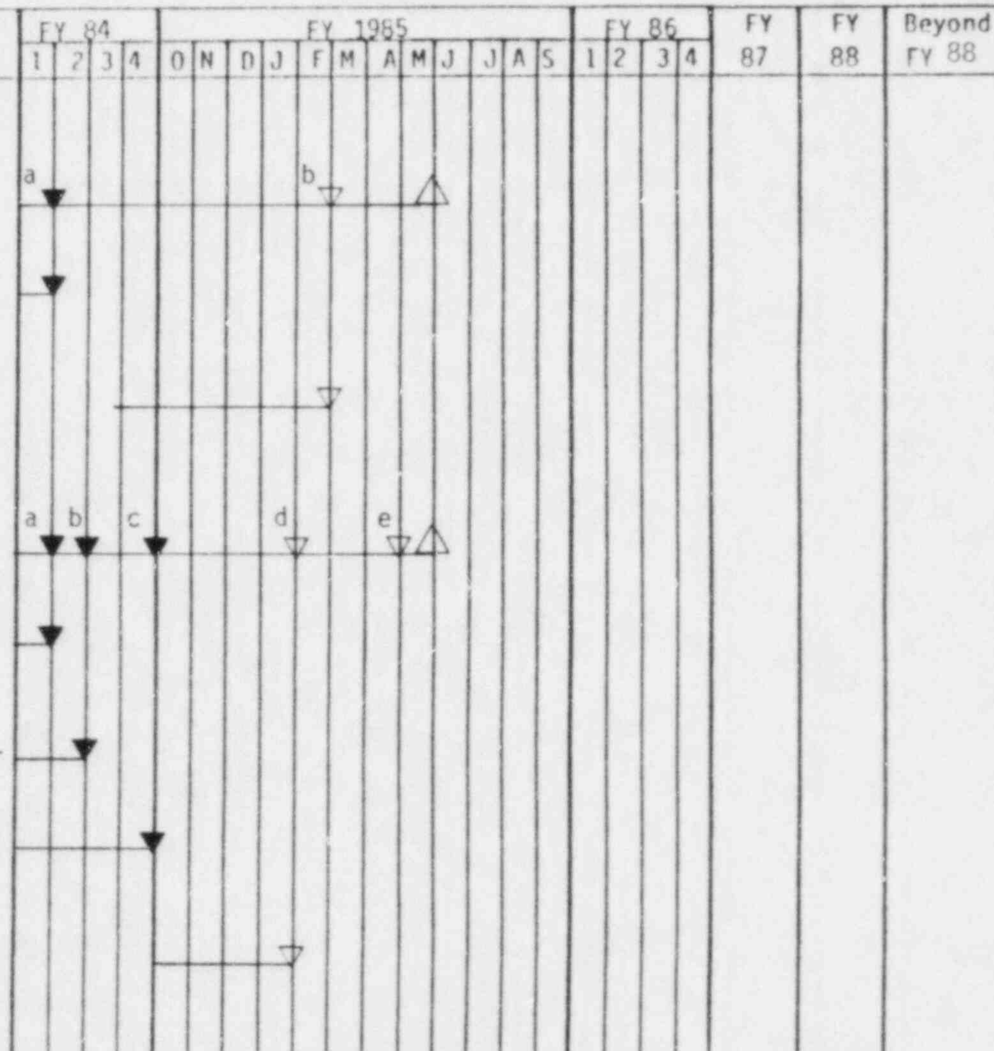
6.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.6 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

Task: H. 6 IRRADIATION EFFECTS STUDIES

SUBTASK/MILESTONE



H.6.2 Ductile Shelf Toughness for Low-Shelf Welds (Series 2 and 3)

A. Issue Report on Low-Shelf Weld Irradiations

a. Complete tensile testing for Second and Third Irradiation Series

b. Complete draft report

H.6.3 Ductile Shelf Toughness for State-of-the-Art Welds (Series 4)

A. Complete Specimen Testing (Tensile, CVN, and 1TCT) (Capsules A-D)

a. Complete testing of tensile specimens - capsules A, B, and C

b. Complete testing CVN specimens - capsule D (FRG)

c. Complete Phase I testing of 1T compact specimens - capsules A, B, and C

d. Complete Phase II testing of 2T compact specimens - capsules A, B, and C

MILESTONE AND STATEMENT SCHEDULE

Task: H.6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE	FY 84	FY 1985	FY 86	FY 87	FY 88	Beyond FY 88
	1 2 3 4	O N D J F M A M J J A S	1 2 3 4			

A. Complete Specimen Testing (cont'd)

- e. Complete testing of 1T compact and tensile specimens - capsule D (FRG)

B. Issue Report on Series 4 - Capsules A, B, and C

- a. Complete analysis of CVN data
- b. Complete analysis of 1TCS data
- c. Compare predictions and data
- d. Complete interpretations
- e. Complete draft report

H.6.4 K_{Ic} Validation for High-Copper Welds (Series 5)

A. Complete Test Material Preparation (12 Capsules)

- a. Complete fabrication of welds
- b. Fabricate specimens - capsules 1 through 4
- c. Fabricate specimens - capsules 5 through 12

MILESTONE STATEMENT AND SCHEDULE

Task: H. 6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE

B. Complete Material Irradiations (Tensile, CVN, T, 2T, and 4T)

   a. Complete 4TCS dummy capsule verification test

   b. Complete 4TCS capsule design

   c. Complete small specimen capsule design

   d. Capsules 1 through 4 (4TCS)

   e. Complete 2TCS capsule design

   f. Capsule 5 and 6 (CVN, tensile, 1TCS)

   g. Capsules 7 and 8 (2TCS)

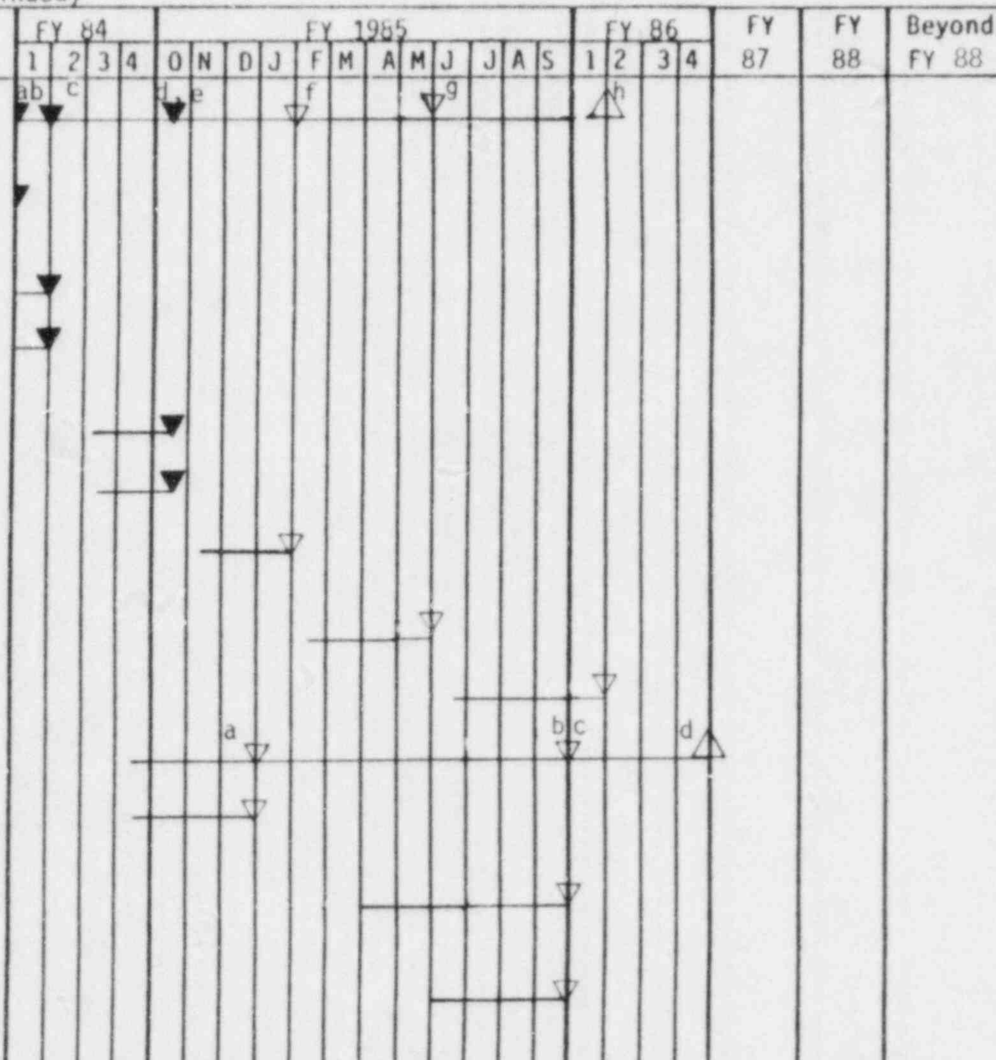
   h. Capsules 9 through 12 (4TCS)

C. Complete Specimen Testing

   a. Complete unirradiated scoping CVN, tensile, and DWT tests

   b. Complete irradiated CVN, tensile, and DWT tests

   c. Complete unirradiated compact specimen tests - 1T, 2T,4T, 6T, and 8T



MILESTEONE STATEMENT AND SCHEDULE

Task: H. 6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE

C. Complete Specimen Testing (continued)

d. Complete irradiated compact specimen tests - 1T, 2T, and 4T

D. Issue Final Report on Series 5

a. Complete analysis of CVN and DWT data

b. Complete analysis of compact specimen data

c. Compare predictions and data

d. Complete interpretations

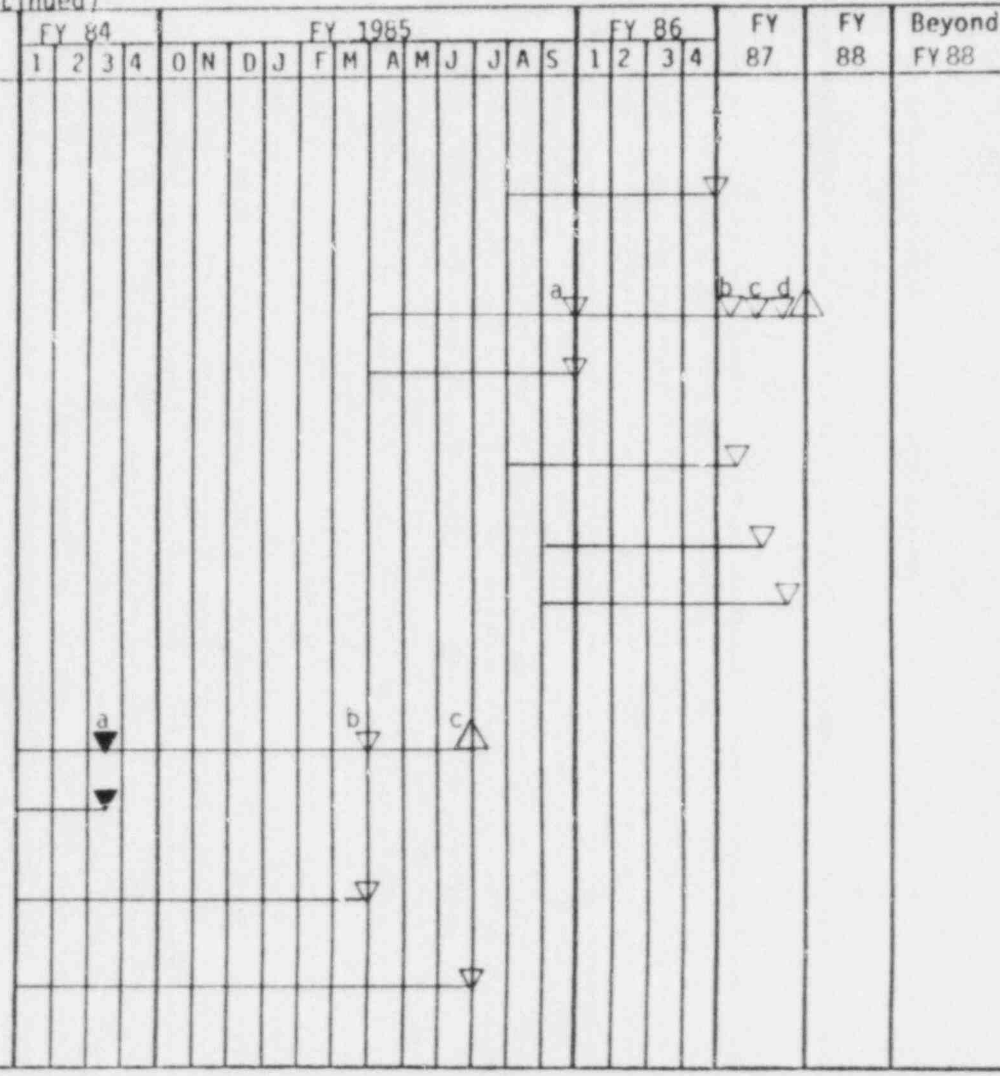
H.6.5 K_{Ia} Determinations for Base and Weld Metals (Series 6)

A. Complete Test Material Preparation

a. Complete fabrication of welds (See H.6.4.A.a)

b. Select CCA specimen design and complement

c. Fabricate CCA specimens



MILESTONE STATEMENT AND SCHEDULE

Task: H.6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
B. Complete Material Irradiations (Two Capsules)											a ▽						b ▽	c △					
a. Complete CCA capsule design											▽												
b. Capsule A																	▽						
c. Capsules B and C																		▽					
C. Complete CCA Specimen Testing																a ▽		b ▽		c △			
a. Complete development of remote CCA test facility																▽							
b. Complete unirradiated CCA tests																		▽					
c. Complete irradiated CCA tests																				▽			
D. Issue Report on Series 6																					a ▽ b ▽ △		
a. Complete analysis of CCA data																					▽		
b. Compare predictions and data																					▽		
H.6.6 Fracture Toughness Determination for Stainless Steel Cladding (Series 7)																							
A. Complete Test Material Preparation							a ▽		b ▽	c △													
a. Complete procurement of three-wire clad (See Milestone H.3.1.G)							▽																

MILESTONE STATEMENT AND SCHEDULE

Task: H. 6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE	FY 84 1	2	3	4	FY 1985 O	N	D	J	F	M	A	M	J	J	A	S	FY 86 1	2	3	4	FY 87	FY 88	Beyond FY 88
A. Complete Test Material Preparation (continued)																							
b. Fabricate compact specimens of one-wire clad - capsule C									▽														
c. Fabricate CVN, tensile, and compact sepcimens of three-wire clad - capsules C, D, F, G, and H										▽													
B. Complete Material Irradiations (Three Capsules of One-Wire and Four Capsules of Three-Wire Clad)	▼ a											▽ b					▽ c			▽ d △ e			
a. Capsules A and B (one-wire, CVN, and tensile) - 2×10^{23} n/m^2	▼																						
b. Capsules D (three-wire CVN and tensile) - 2×10^{23} n/m^2 and C (one-wire and three-wire compacts)												▽											
c. Capsule H (three-wire CVN and tensile) - 5×10^{23} n/m^2																	▽						
d. Capsule F (three-wire CVN and tensile) - 1×10^{23} n/m^2																				▽			
e. Capsule G (three-wire compacts) - 1×10^{23} n/m^2																				▽			

MILESTONE STATEMENT AND SCHEDULE

Task: H.6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE	FY 84 (1 2 3 4)	FY 1985 (O N D J F M A M J J A S)	FY 86 (1 2 3 4)	FY 87	FY 88	Beyond FY 88
C. Complete Specimen Testing	a b	c	d e f g h	i		
a. Complete unirradiated one-wire CVN and tensile tests						
b. Complete irradiated one-wire CVN and tensile tests - 2×10^{23} n/m^2						
c. Complete unirradiated three-wire CVN and tensile tests						
d. Complete irradiated three-wire CVN and tensile tests - 2×10^{23} n/m^2						
e. Make decisions concerning necessity for irradiation of capsules F and G						
f. Complete unirradiated one- and three-wire compact tests						
g. Complete irradiated one-wire and three-wire compact tests - 2×10^{23} n/m^2						
h. Complete three-wire CVN and tensile tests - 5×10^{23} n/m^2						

MILESTONE STATEMENT AND SCHEDULE

Task: H.6 IRRADIATION EFFECTS STUDIES (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
C. Complete Specimen Testing (continued)																							
i. Complete irradiated three-wire compacts, CVN, and tensile tests - 1×10^{23} n/m^2																					▽		
D. Issue Final Report on Series 7			a ▼	b ▼ c ▼													d ▽			e ▽	f ▽	△	
a. Complete analysis of one-wire CVN and tensile data - 2×10^{23} n/m^2			▼																				
b. Complete interpretation of one-wire CVN and tensile data - 2×10^{23} n/m^2				▼																			
c. Issue interim report on irradiation effects on one-wire cladding - 2×10^{23} n/m^2				▼																			
d. Complete analysis and interpretation of three-wire CVN and tensile data - 2×10^{23} n/m^2																	▽						
e. Issue interim report on irradiation effects on three-wire wire cladding																				▽			
f. Complete analysis and interpretation of all results																					▽		

7. HSST TASK H.7 CLADDING EVALUATIONS

7.1 Objective

The objective of Task H.7 is to demonstrate the effect of stainless steel weld cladding on the extent of crack propagation, for small surface cracks subjected to stress gradients similar to that produced by a thermal shock.

7.2 Background

The Cladding Evaluations Task of the HSST Program was initiated in FY 1982 to study the interaction of stainless cladding with flaws initiated in and propagating in base metal. From the designer's viewpoint stainless cladding is primarily viewed as a corrosion- and crud-prevention material in light-water reactor vessel design and except for its effect upon fatigue in thermal transients, its effect upon structural integrity has heretofore been largely disregarded. With the more recent focus of safety studies upon LOCA scenarios that emphasize the behavior of small flaws, it has become evident that stainless cladding may have a key role in the propagation and/or arrest of propagating flaws. Complicating factors which seriously affect an understanding of the role stainless cladding plays in flaw propagation are its fracture toughness as a function of temperature and irradiation dose and the influence of several fabrication processes that have been used in vessel fabrication. Meager data exist in both of these areas. The initial phase of this study has attempted to address this question by testing stainless-clad specimens that had been subjected to heat treatments to simulate "beginning-of-life" and "end-of-life" toughness conditions to fast-running cracks. The work breakdown structure for this task has been organized around the type of cladding being investigated as shown in Fig. 7.1.

A survey of fabrication processes employed on reactor vessels revealed that the majority of light-water reactor vessels have employed either three-wire or strip-clad processes with the three-wire process being predominantly used on early vessels, and the strip process on later vessels. Because of the pressing need for data, the mothballing by vendors of their three-wire equipment, and the attendant difficulty in obtaining timely contracts for vendor preparation of specimens, initial specimens used in the HSST program tests were prepared at ORNL by using a single-wire welding procedure.

The specimens were designed as rectangular parallelepipeds with stainless cladding on one face. Grooves were machined in the cladding with the intent to provide a plane surface at the bottom of the groove at the stainless-base metal interface. An electron beam (EB) weld was then applied to the bottom groove surface. Specimens were cooled to the test temperature and were loaded by constant-moment loading to the stress state required. Hydrogen charging of the EB weld was initiated and presented the stainless cladding with a relatively fast-running

ORNL-DWG 83-5176 ETD

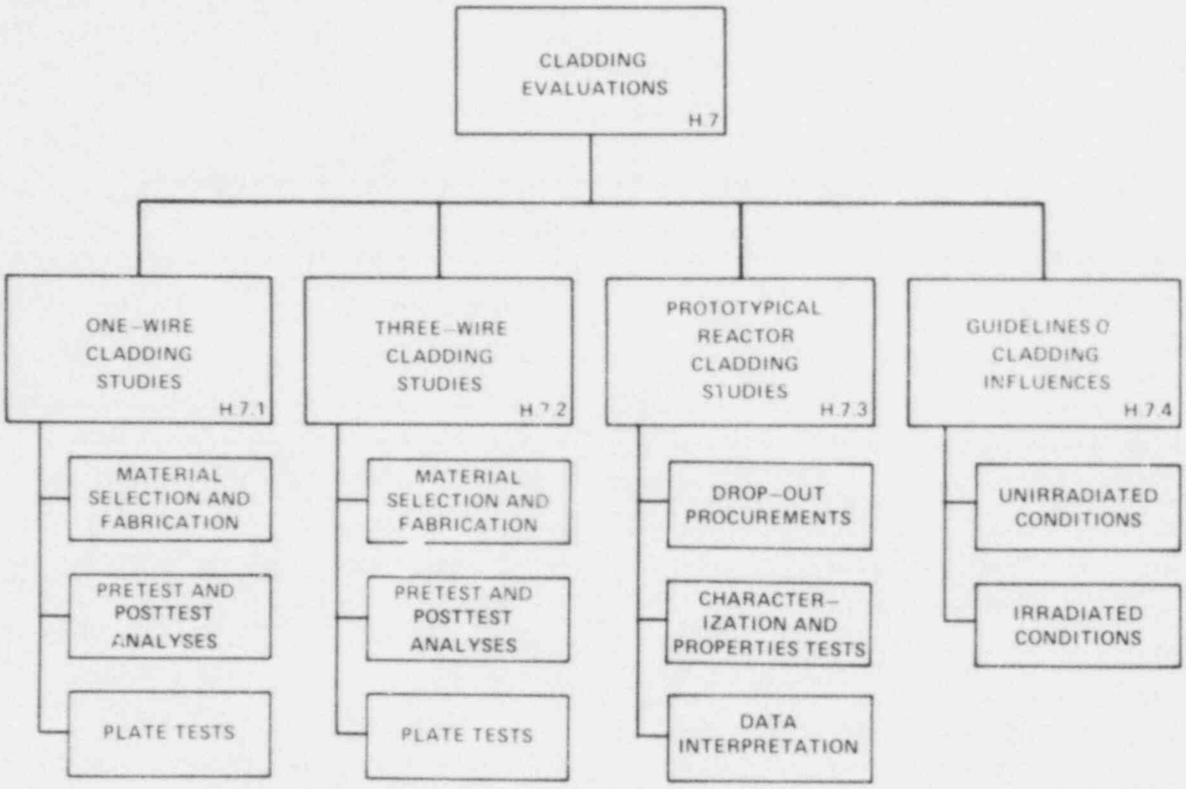


Fig. 7.1 Work breakdown structure for HSST Task H.7 Cladding Evaluations.

crack. A matrix of specimens was planned that varied the parameters: flaw size, run distance from EB weld to cladding, cladding type, and stress state in order to elucidate cladding arrest behavior.

Problems were experienced with groove machining to obtain the stainless base metal interface, in some cases the groove was too shallow, in others too deep. On specimens where stainless remained below the groove, premature popping of the EB weld prior to hydrogen charging was a common phenomenon, preventing a proper control of the stress state. On specimens with too deep grooves, the geometry caused premature arrest and prevented the flaw from running to the cladding. In addition, the specimens prepared by sigma-phase heat treatment were too brittle and, based on limited data, are not representative, as intended, of end-of-life conditions.

The tests completed to date under the initial phase of this study indicate that the cladding employed to represent beginning-of-life conditions has sufficient arrest toughness to stop running cracks, but the upper and lower bounds of crack arrest are not yet determined. Analyses of the tests by two approximate techniques and by the ORVIRT finite-element method have not been completely consistent because of stress concentration effects of the very deep grooves in the specimens. The fabrication techniques employed for this first series of tests have

resulted in conditions that have prevented control of the stress state at pop-in of the hydrogen-charged EB welds. Consequently, bounding of the arrest toughness of the stainless cladding has been prevented. A report has been drafted that covers this Phase I study of cladding behavior.

Preparations are now under way to redesign and fabricate a new series of specimens that will eliminate the problems presented by the groove/EB weld design of the first series. In addition, this series will employ material prepared by a three-wire weld cladding technique typical of many early reactor vessel designs.

A nozzle dropout having a nominal 4.76 mm (3/16 in.) layer of 3-wire stainless cladding was purchased from Combustion Engineering Co., Inc. Tensile and Charpy-impact values were obtained and photomicrographic studies made.

7.3 Plan of Action

A special fabrication subcontract was placed with Combustion Engineering Co., Inc., in July 1983, to produce specimen blanks and characterization material employing a three-wire weld cladding technique typical of many early reactor vessel designs. A special heat treatment procedure was developed and applied to this material with the purpose of raising the NDT of the base plate while maintaining the room temperature yield strength between 414 and 690 MPa. In addition, the Charpy-impact energy of the cladding shall be at least 54.2 J (40 ft-lb) at a temperature either -7°C (20°F) below the NDT of the base plate or at which the Charpy-impact energy is 27.1 J in the base plate, whichever is lower.

The combination of these material properties will substantially improve the capability of obtaining the crack arrest toughness for this type of cladding for beginning-of-life conditions. The conditions requisite for obtaining end-of-life cladding arrest toughness values will be based on the results from the Seventh Irradiation Series (see Ref. 2 in Task H.6) where testing of irradiated specimens of single-wire stainless cladding was completed during FY 1984, and of three-wire cladding is scheduled for FY 1986.

Three clad-plate specimens and one unclad specimen will be machined and tested in FY 1985 from the blanks furnished by the Combustion Engineering contract. A second set of three-clad and one-unclad plate specimens will be tested in FY 1986. In addition, the properties of single layer and multilayer cladding will be characterized.

Two segments of a BWR pressure vessel from Hope Creek, N.J. were obtained for flaw and properties characterization. One of these pieces is from the beltline region and it includes segments of longitudinal and circumferential welds. The second includes a recirculation nozzle. The base material is A533B steel, and both have stainless steel cladding. NDE and destructive examinations will be performed in FY 1985 to determine flaw distributions in the base metal, cladding, and weld metal. Properties values and variations will be obtained for each material. Negotiations are continuing for the procurement of a nozzle dropout with

single-wire, six-wire and strip cladding types. Tensile and Charpy studies will be performed on these materials.

A collection of the data from these various sources will provide a basis for more substantive direction of this activity.

7.4 References

1. W. R. Corwin, et al., *Effect of Stainless Steel Weld Overlay Cladding on the Structural Integrity of Flawed Steel Plates in Bending*, NUREG/CR-4015 (ORNL/TM-9390), Oak Ridge National Laboratory April 1985.

7.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.7 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

Task: H.7 CLADDING EVALUATIONS

SUBTASK/MILESTONE

H.7.1 One-Wire Cladding Studies

A. Issue Report on First Series of One-Wire Clad-Plate Tests

H.7.2 Three-Wire Cladding Studies

A. Purchase Clad-Plate Material from RPV Manufacturer (see Milestone H.3.1.G.a)

B. Issue Report on First Series of Three-Wire Clad-Plate Tests

a. Machine test specimens

b. Complete test of three-clad and one-unclad specimens

c. Complete posttest analyses

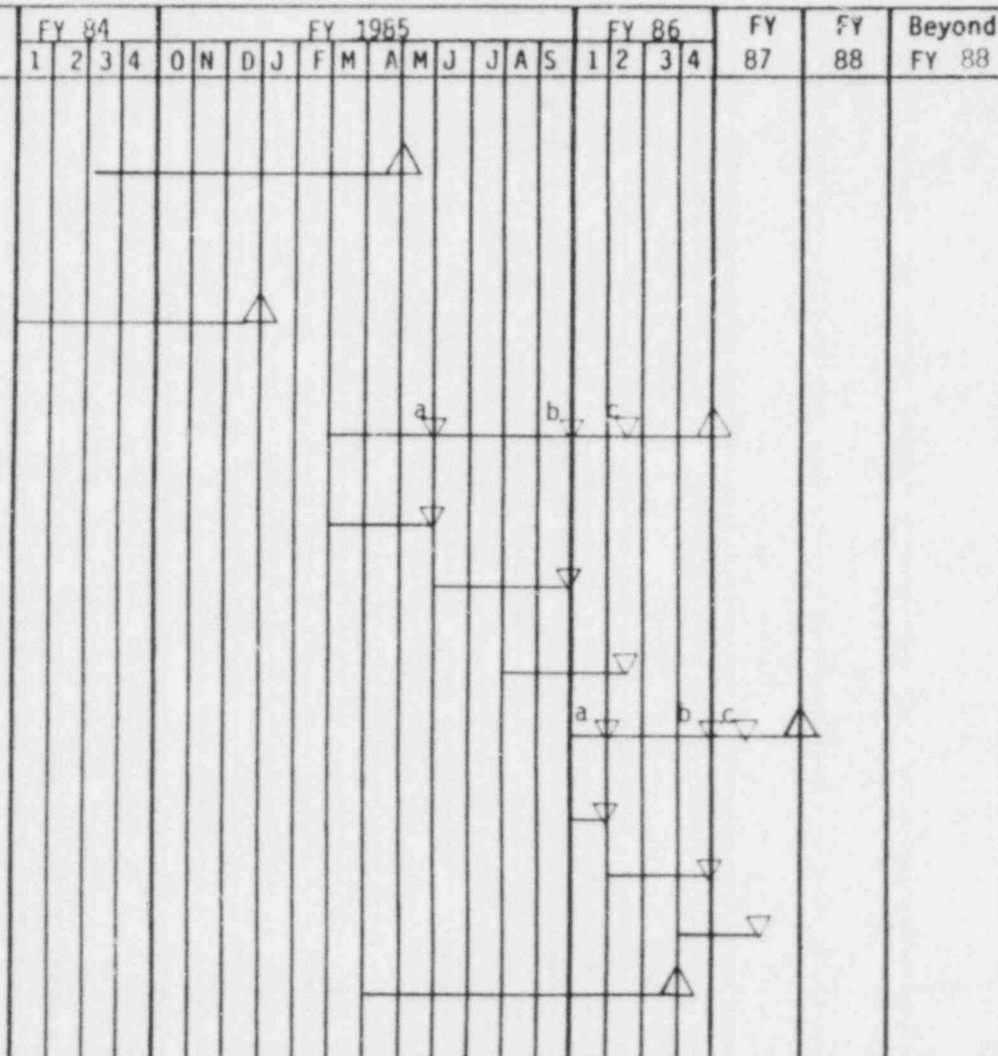
C. Issue Report on Second Series of Three-Wire Clad-Plate Tests

a. Machine test specimens

b. Complete series of three tests

c. Complete posttest analyses

D. Complete Study of Residual Stresses



MILESTONE STATEMENT AND SCHEDULE

Task: H.7 CLADDING EVALUATIONS (continued)

SUBTASK/MILESTONE

H.7.3 Prototypical Reactor Cladding Studies

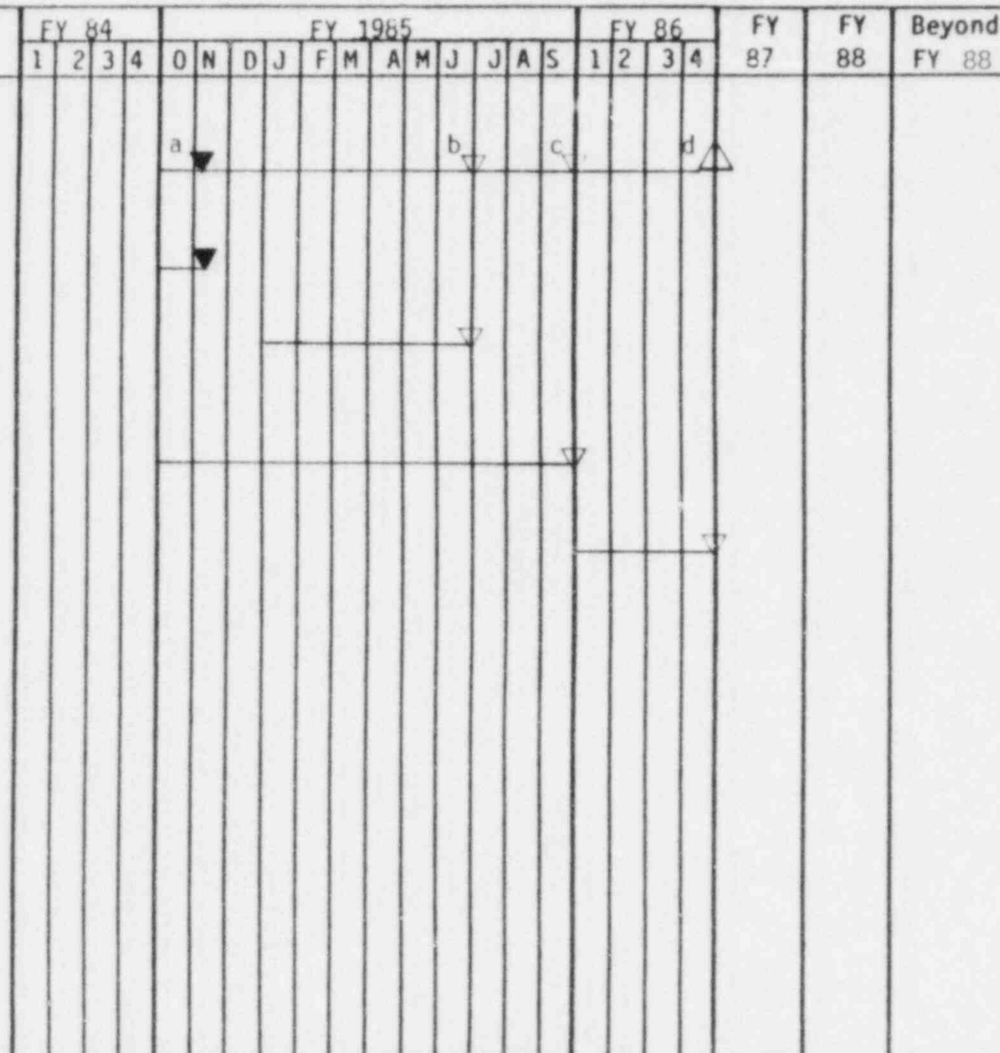
A. Complete Characterization Study of RPV Drop-Outs

a. Procure BWR pressure vessel beltline-clad material

b. Complete,NDE, metallographic and properties characterization of BWR material

c. Procure dropouts from RPV manufacturers

d. Complete interpretation and comparison of clad material results



8. HSST TASK H.8 INTERMEDIATE VESSEL TESTS AND ANALYSES

8.1 Objective

The objective of Task H.8 is to expose flaws to specific stress states and conditions to (1) demonstrate actual behavior under conditions found in reactor pressure vessels (RPV's) and (2) determine the applicability of theoretical methods to predict the fracture behavior of RPV's. This is accomplished through testing and analyzing thick pressurized vessels over a full range of fracture characteristics.

8.2 Background

The original plan of the HSST program contemplated that simulated service tests would comprise one of the principal program tasks.[1] Simulated service tests were conceived as fracture tests of specimens or structures of dimensions large enough to develop stress states in the region of interest like those in reactor pressure vessels. The general purpose was to demonstrate whether or not fracture mechanics concepts that had been verified experimentally on a small laboratory scale were valid and useful in the fracture analysis of full-size vessels.

The main factors that were expected to be important in relating laboratory-scale fracture to that of large-scale structures were:

(1) Wall thickness
- (a) transverse restraint
- (b) variability of properties
- (c) flaw location
- (d) flaw orientation

(2) Material-embrittling factors
- (a) temperature
- (b) irradiation
- (c) strain aging
- (d) reversed plastic strain
- (e) heat treatment, temperature embrittlement, and composition

(3) Stress analysis
- (a) crack location, configuration, orientation, and number
- (b) diameter-to-thickness ratio
- (c) crack depth-to-thickness ratio
- (d) biaxiality and stress concentration
- (e) thermal and residual stresses

(4) Crack configuration
- (a) depth
- (b) shape
- (c) preparation techniques

The validation of methods for fracture predictions was to proceed in steps from laboratory specimens to intermediate-scale vessels to full-size pressure vessels. First, the simple transition temperature and fracture mechanics results obtained from more or less standard specimens were to be generalized to thick specimens with varying flaws subjected to tensile loading. Such specimens were to be examined both for tough and frangible behavior. Next, the complexities of pressure vessel geometry were to be introduced on small epoxy and steel models in preparation for tests of flawed intermediate-scale vessels. Finally, tests were to be performed on vessels having the full-scale features of reactor pressure vessels. Practical considerations of limited monetary and other resources eliminated full-scale tests and tests of irradiated models from the plan.

As actually implemented, the intermediate vessel test series (Task H.8) and the thermal-shock (Task H.9) and pressurized thermal-shock (Task H.10) tests constitute the simulated service tests. The test specimens (cylinders or vessels) have been chosen to be thick and long enough to produce stress states in the flawed region that represent the conditions that would be obtained in a reactor pressure vessel. Intermediate test vessels (ITV) are ~150-mm thick, which is the nominal thickness of a boiling-water-reactor vessel. With this thickness and the 1300-mm length of the test section, the ITV's are also excellent fracture-mechanics specimens. Large surface flaws in the beltline region of these vessels are essentially as well restrained as they would be in a reactor pressure vessel. It is important to remember now that, at the inception of the HSST program, there were doubts and apprehension about whether a thick structure would exhibit a "transition temperature" separating frangible and ductile behavior. While that concern was resolved, transverse restraint, and therefore thickness, in fracture behavior is as important a factor as ever. Only in intermediate vessel and thermal-shock tests has transverse restraint been great enough to produce plane-strain conditions with high K_I values in pressure vessel steels.

The program of intermediate vessel tests was, therefore, planned to fulfill the broad objectives of the simulated service tests over the wide variety of conditions that constitute the real operating environments. The main effort of this task was to determine which factors or conditions were important and then to carry out tests with the particular combinations of conditions that would contribute significant information on behavior of structures in service. Factors of concern include material properties; flaw location, orientation, shape, size, and sharpness; and loading and environmental conditions. If expense, resources, safety, and time were of no importance, the significance of these factors could be studied under actual service conditions imposed on vessels in a large number of experiments with vessels as large as, or larger than, reactor pressure vessels. Since such an approach is completely impractical, each factor is studied separately or in combination with others on as simple a scale as practicable.

The objective of each simulated service test has been to provide data from which the ability of analytical methods to predict the fracture behavior of a flawed structure under known conditions of material properties and loading could be assessed. In the planned progression of

tests, analytical methods are confirmed, improved, or their limitations revealed. The testing of intermediate vessels in conjunction with flawed tensile specimens of similar material permits consideration of many variables, such as flaw size, section thickness, temperature, and stress state. The effects of differences in transverse restraint, toughness, plastic strain, biaxiality, and stress concentration can also be observed and analyzed.

The original objectives emphasized in the simulated service tests were (1) to demonstrate capability to predict the "vessel transition temperature" for a selected crack configuration using the material of interest (ASTM A533, grade B, class 1 plate; ASTM A508, class 2 forging); (2) to demonstrate, for the materials of interest, the capability to predict various combinations of load (pressure), temperature, and crack configuration in full-thickness walls (152 mm or more) that will not cause fracture, and finally a combination that will cause fracture for both frangible and tough fracture conditions.

The intermediate vessel tests have been divided into four series:

(1) flaws in cylindrical vessels, A508, class 2 forging steel — two vessels;
(2) flaws in cylindrical vessels with longitudinal weld seams, A508, class 2 forging steel, submerged-arc welds — three vessels;
(3) flaws in cylindrical vessels with longitudinal weld seams, A533, grade B, class 1 plate steel, submerged-arc welds — two vessels;
(4) cylindrical vessels with radially attached nozzles, vessels of A508, class 2 forging steel and A533, grade B, class 1 plate steel; nozzle of A508, class 2 forging steel — three vessels.

Test conditions were selected to produce fracture toughnesses ranging from the low transition to the fully ductile upper shelf. The variations of test conditions, flaw configurations, materials, and stress states are summarized in Table 8.1. The scope of the activities necessary to carry out the tests and analyses is illustrated in the work breakdown structure, which is shown in Fig. 8.1.

Most of the tests in the original plan have been performed, including the supporting research.[2-5] An extensive series of epoxy and steel model tests was performed with ~1/7- to 1/14-scale models of the ITV test section,[2-5] and instrumented epoxy models were tested to obtain K_I shape factors for surface flaws and stress concentration factors for nozzle corners. Tensile specimens 152 mm thick with surface flaws prototypic of ITV flaws were tested over a range of temperatures from lower shelf to upper shelf.[6-9] Flawed tensile specimens scaled to match the small-scale steel vessels were tested. Information from the small-scale to full-scale tests was used to estimate the effect of section thickness on the "effective transition temperature" and to determine the effect of specimen size on apparent fracture toughness.

The experiments have been accompanied by several types of analyses.[2-5,10,11] Structural analyses were made to define the elastic and elastic-plastic states of stress in the flawed region and to understand the influence of the test vessel geometry on the stresses in this

Table 8.1 Summary of test conditions for the Intermediate Vessel Test Program

Vessel	Test temperature (°C)	Material type	Toughness state	Flaw type[a]	Stress	Loading[b] state	Fracture mode
V-1	54	A508, Cl 2	Upper transition static upper shelf	Part circular outside surface	E&EP[c]	Pressure	Ductile tearing and mode conversion to cleavage
V-2	0	A508, Cl 2	Low transition	Part circular outside surface	E&EP	Pressure	Cleavage
V-3	54	Weld	Upper transition static upper shelf	Part circular outside surface	E&EP	Pressure	Ductile tearing and mode conversion to cleavage
V-4	24	Weld[d] A508, Cl 2	Low transition	Part circular outside surface (2 flaws)	E&EP	Pressure	Cleavage
V-5	88	A508, Cl 2	Dynamic upper shelf	Nozzle-corner notch	Stress concentration E&EP	Pressure	Ductile tearing; leak but no failure
V-6	88	A508, Cl2 Weld[d]	Dynamic upper shelf	Part circular surface flaws outside[d] (2) inside (1)	EP	Pressure	Ductile tearing and full shear tear
V-7	91	A533, Gr B, Cl 1	Dynamic upper shelf	Long, deep notch	EP	Pressure	Ductile tearing with leak but no failure
V-7A	88	A533, Gr B, Cl 1	Dynamic upper shelf	Long, deep notch	EP	Pressure, pneumatic, sustained load	Ductile tearing with leak but no failure
V-7B	87	HAZ[e]	Dynamic upper shelf	Long, deep notch	EP	Pressure, residual stress sustained load	Ductile tearing and leak with incipient tearing instability
V-8	-23	Submerged arc weld	Low transition	Semielliptical outside surface	E	Pressure, residual stress[f]	Cleavage initiation and arrest (twice) with leakage but no failure
V-8A	157	Low-upper shelf weld	Dynamic upper shelf	Semielliptical outside surface	EP	Pressure	Stable and unstable ductile tearing terminated without leakage
V-9	24	A508, Cl 2	Low transition	Nozzle-corner notch	Stress concentration EP&P	Pressure	Ductile tearing with mode conversion to cleavage
V-10		A508, Cl 2					
A[g]	>>RTNDT		High transition	Shallow notch in nozzle corner	E	Cyclic pressure	Fatigue crack growth
B	<<RTNDT		Low transition	Nozzle corner notch	E	Pressure	Cleavage

[a]Initial flaws in V-1 to V-9 were sharpened by fatigue or by EB welding (V-7 Series) and were oriented in a radial-axial plane.

[b]Mechanical loads were hydrostatic except for V-7A, which was pneumatic.

[c]E = elastic, EP = elastic-plastic.

[d]Failure site.

[e]Heat-affected zone of Section XI repair weld.

[f]Residual stress from Section XI repair weld.

[g]This testing mode is not included in milestone schedule.

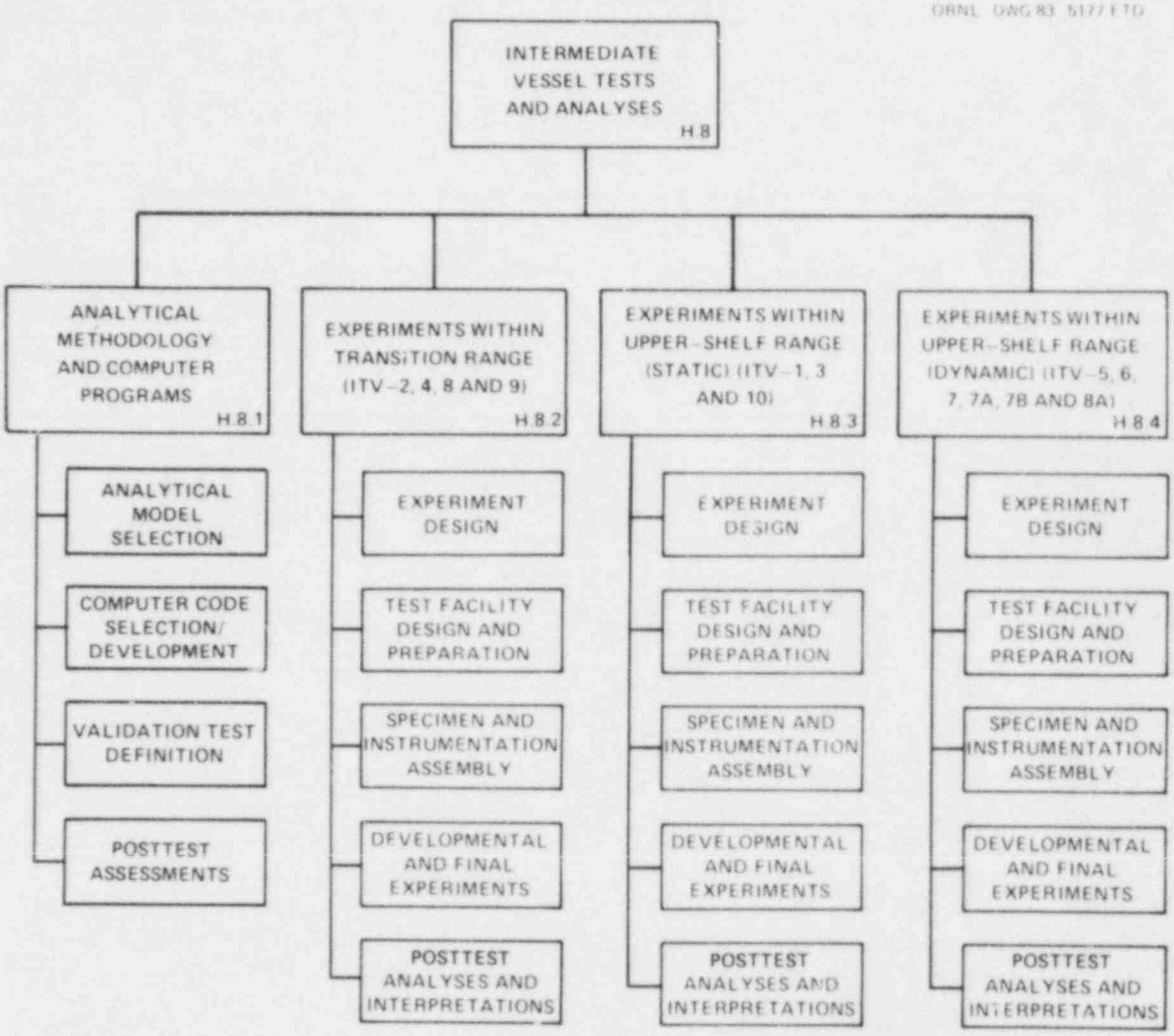


Fig. 8.1 Work breakdown structure for HSST Task H.8 Intermediate vessel tests and analyses.

region. In view of the relative simplicity of analysis of unflawed structures, much of the fracture analysis was based upon superposition of stress fields of unflawed complex structures and simple flawed bodies. Models of failure that compete with fracture, such as tensile instability, were also investigated by analysis, since they are particularly important in cases in which flaws are large or toughness is high.

Every ITV test involves material properties determinations, generation of a flaw of desired geometry and acuity, pretest evaluation of flaw behavior and estimation of modes and conditions of fracture, and measurement during the test of the variables that define the state of stress of the vessel and geometry of the flaw. Twelve ITV tests have been performed successfully, so that methods of preparing and testing vessels have been well established. Four tests not originally planned

have been conducted to study pneumatic pressurization (sustained loading), repair welds, residual stresses, low upper-shelf toughness, and tearing instability.[11-14] The demand for special test conditions and more precise observation of flaw behavior will require continuing work to develop the techniques for future tests.

8.3 Plan of Action

Explicit preliminary plans have been made for testing the one untested vessel, V-10. A schedule is presented for a hypothetical fracture test of this vessel with a nozzle. This test would be performed well below the transition temperature and would be the first attempt to test the precise application of linear-elastic fracture mechanics (LEFM) to a flaw in a complex stress state. Earlier tests of vessels with nozzles were performed at such high temperatures that general yielding of the vessels preceded failure. A test under linear-elastic conditions is needed to show how LEFM should be applied to regions of stress concentration, a point on which earlier tests may have been interpreted by some people in an unconservative way.

Since V-10 is the last ITV with a nozzle, some consideration has been given to preceding the test described above, which will be a destructive test, by a fatigue crack growth test. This is shown in Table 8.1 as V-10A. Schedules and costs, however, are estimated only for the fracture test identified in the table as V-10B.

8.4 References

1. *Program Plan, The Heavy-Section Steel Technology Program*, prepared by the Staff of the HSST Program, Oak Ridge National Laboratory, April 1, 1968 (revised February 27, 1970).

2. R. W. Derby, J. G. Merkle, G. C. Robinson, G. D. Whitman and F. J. Witt, *Test of 6-Inch-Thick Pressure Vessels. Series 1: Intermediate Test Vessels V-1 and V-2*, ORNL-4895, Oak Ridge National Laboratory, February 1974.

4. J. G. Merkle, G. C. Robinson, P. P. Holz, J. E. Smith, and R. H. Bryan, *Test of 6-In.-Thick Pressure Vessels. Series 3: Intermediate Test Vessel V-7*, ORNL/NUREG-1, Oak Ridge National Laboratory, August 1976.

5. J. G. Merkle, G. C. Robinson, P. P. Holz and J. E. Smith, *Test of 6-In.-Thick Pressure Vessels. Series 4: Intermediate Test Vessels V-5 and V-9 with Inside Nozzle Corner Cracks*, ORNL/NUREG-7, Oak Ridge National Laboratory, August 1977.

6. S. C. Grigory, *Tests of 6-In.-Thick Flawed Tensile Specimens, First Technical Summary Report, Longitudinal Specimens Numbers 1 through 7*, HSSTP-TR-18, Southwest Research Institute, June 1972.

7. S. C. Grigory, *Tests of 6-Inch-Thick Flawed Tensile Specimens, Third Technical Summary Report, Longitudinal Specimens Numbers 14 through 16, Unflawed Specimen Number 17*, HSSTP-TR-22, Southwest Research Institute, October 1972.

8. S. C. Grigory, *Tests of 6-Inch-Thick Tensile Specimens, Fourth Technical Summary Report, Tests of 1-Inch-Thick Flawed Tensile Specimens for Size Effect Evaluation*, HSSTP-TR-23, Southwest Research Institute, June 1973.

9. S. P. Ying and S. C. Grigory, *Tests of 6-Inch-Thick Tensile Specimens, Fifth Technical Summary Report, Acoustic Emission Monitoring of one-Inch and Six-Inch-Thick Tensile Specimens*, HSSTP-TR-24, Southwest Research Institute November 1972.

10. J. G. Merkle, G. D. Whitman, and R. H. Bryan, *An Evaluation of the HSST Program Intermediate Pressure Vessel Tests in Terms of Light-Water-Reactor Pressure Vessel Safety*, ORNL-TM-5090, Oak Ridge National Laboratory, November 1975.

11. R. H. Bryan, T. M. Cate, P. P. Holz, T. A. King, J. G. Merkle, G. C. Robinson, G. C. Smith, J. E. Smith and G. D. Whitman, *Test of 6-In.-Thick Pressure Vessels. Series 3: Intermediate Test Vessel V-7A Under Sustained Loading*, ORNL/NUREG-9, Oak Ridge National Laboratory, February 1978.

12. R. H. Bryan, P. P. Holz, J. G. Merkle, G. C. Smith, J. E. Smith and W. J. Stelzman, *Test of 6-In-Thick Pressure Vessels. Series 3: Intermediate Test Vessel V-7B*, NUREG/CR-0309 (ORNL/NUREG-38), Oak Ridge National Laboratory, December 1979.

13. R. H. Bryan, D. A. Canonico, P. P. Holz, S. K. Iskander, J. G. Merkle, J. E. Smith and W. J. Stelzman, *Test of 6-In.-Thick Pressure Vessels. Series 3: Intermediate Test Vessel V-8*, NUREG/CR-0675 (ORNL/NUREG-58), Oak Ridge National Laboratory, December 1979.

14. R. H. Bryan, S. E. Bolt, J. G. Merkle and G. D. Whitman, *Quick-Look Report on Test of Intermediate Vessel V-8A — Tearing Behavior of Low Upper-Shelf Material*, ORNL/SST-4, Oak Ridge National Laboratory, August 25, 1982.

8.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.8 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

Task: M.8 INTERMEDIATE VESSEL TESTS AND ANALYSIS

SUBTASK/MILESTONE

H.8.2 <u>Experiments Within Transition Range</u>

A. Issue Report on Test of Vessel with Nozzle-Corner Crack (Test ITV-10)

a. Complete test definition

b. Complete pretest analyses and pressure tests

c. Complete posttest analyses

H.8.4 <u>Experiments Within Upper-Shelf Range</u>

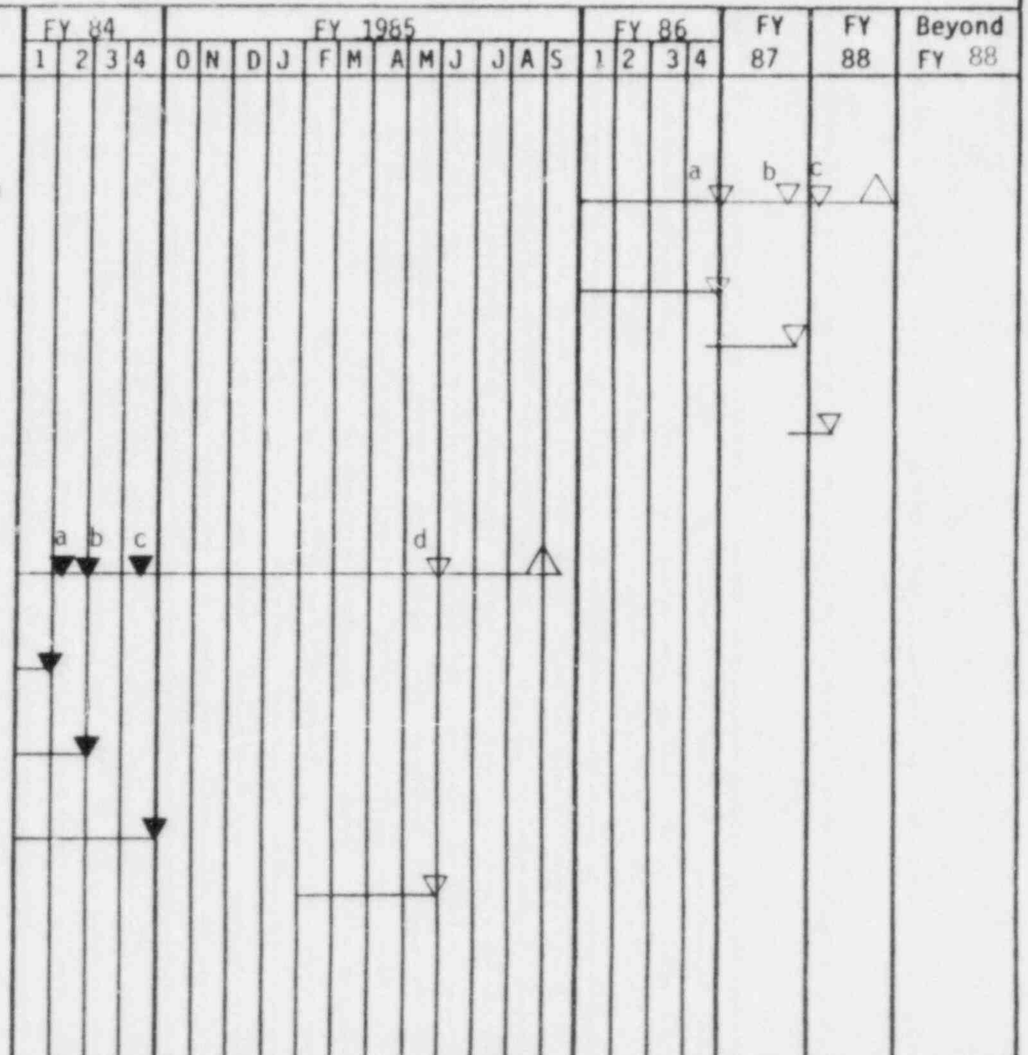
A. Issue Report on Test of Vessel with Low Upper-Shelf Weld (ITV-8A)

a. Complete posttest interpretations of flaw geometry data

b. Complete evaluation of posttest properties data

c. Complete posttest analyses

d. Complete report draft



9. HSST TASK H.9 THERMAL-SHOCK TECHNOLOGY

9.1 Objective

The objectives of Task H.9 are to (1) investigate the behavior of inner-surface flaws in pressure vessels during severe thermal-shock loading conditions, and (2) evaluate severity of the pressurized thermal-shock problem to the extent necessary for defining an appropriate program for investigating flaw behavior.

9.2 Background

Thermal shock to PWR pressure vessels during postulated transients such as the large-break loss-of-coolant accident (LBLOCA) was identified as a possible challenge to vessel integrity as early as 1968. Preliminary studies conducted prior to 1973 indicated the possibility of crack initiation during a LBLOCA, but the ability of the crack to arrest could not be properly assessed because of a lack of appropriate crack-arrest toughness data. In 1973 the HSST staff recommended that a thorough study of the thermal-shock issue be undertaken, and that the effort should include, among other things, thermal-shock experiments designed to investigate the validity of the methods of analysis [primarily linear elastic fracture mechanics (LEFM)] as applied to thermal-shock loading conditions.

The HSST Thermal-Shock Task was established in 1973, and soon thereafter a finite-element method of analysis, based on linear-elastic fracture mechanics was developed for analyzing deep as well as shallow flaws. This tool was used to predict the behavior of two-dimensional flaws during LBLOCA loading conditions. Indications were that for the case of typically high-copper material and high fluence, very shallow flaws would initiate, and in a series of initiation-arrest events the flaw would penetrate deep into but not entirely through the wall. Furthermore, crack arrest had to take place with the stress intensity factor (K_I) increasing with crack depth, and in some cases warm prestressing, with K_I decreasing with time, could limit the depth to which the flaw would propagate.

It was decided that each of the above flaw behavior trends should be investigated experimentally under conditions similar to those expected for the PWR's. This required the use of thick-walled steel test cylinders, that would be flawed and thermally shocked on the inner surface, and the achievement of approximately the same potential for crack propagation as predicted for PWR vessels under LBLOCA thermal-shock loading conditions. Since the toughness of the test-cylinder material would not be reduced by radiation damage, as it is in a PWR vessel, it was necessary to use special heat treatments and/or a more severe thermal shock than expected for the PWR.

Eight thermal-shock experiments have been conducted thus far. The first four (TSE-1, -2, -3, -4) were relegated to an investigation of the behavior of shallow flaws and demonstrated initiation and arrest of long

and short shallow flaws in good agreement with LEFM. The remaining experiments (TSE-5, -5A, -6, -7) were conducted with larger diameter test cylinders so that deeper penetration of the flaw and other desired characteristics could be achieved. The results included demonstrations of multiple initiation-arrest events, deep penetration of the flaw, arrest in a rising K_I field, warm prestressing, large surface extension of short flaws, a lack of significant dynamic effects at arrest, and the inability of a flaw to completely penetrate the cylinder wall under thermal-shock loading conditions only. Each of these demonstrations validated the methods of analysis.

An important aspect of the experimental program is the determination of the fracture-toughness properties of the test-cylinder material, using small-size lab specimens, because in many cases the validity of LEFM is established on the bases of good agreement between lab toughness data and similar data deduced from the thermal-shock experiments. An important discovery in the process of obtaining the lab data was the existence of very large scatter in the crack initiation toughness. This has led to a reconsideration of the methods used to obtain valid data from a reasonable number of lab specimens and of the expected behavior of flaws in the test cylinders and PWR vessels. Task H.3 includes the characterization effort for TSE materials.

Another important part of the thermal-shock program has been the development of fracture-mechanics models and codes that are used for discovering the flaw behavior trends and that are subsequently used for sensitivity studies and assessing the severity of the OCA problem. Fracture-mechanics codes developed as a part of Task H.9 and which are still used extensively are FMECH, a two- and three-dimensional finite-element fracture-mechanics code; OCA-II, a fracture-mechanics code based on superposition techniques; and OCA-P, a probabilistics fracture-mechanics code that is a combination of OCA-II and a Monte Carlo routine.

Although the thermal-shock experiments have been devoted to the investigation of flaw behavior under thermal-shock-loading conditions only, a continuing analysis of OCA's has revealed the fact that the greatest challenge to vessel integrity comes from transients that involve high pressure as well as thermal shock. Pressure loading has little effect on the propagation of shallow flaws, but thermal-shock can drive the initially shallow flaw deep enough for pressure effects to be substantial. For some postulated OCA's the state-of-the-art methods of analysis indicate that following initiation of a shallow flaw there may be no arrest of the propagating flaw, the flaw finally penetrating the wall as a result of plastic instability (excessive pressure loading) in the remaining ligament. The actual behavior of the flaw under combined thermal and pressure loading is being examined experimentally as a part of Task H.10, along with assessments of predictive methods.

A summary of the experiments that have been conducted as a part of Task H.9 is presented in Table 9.1. References 1—2 are reports that have been prepared on completed tests. Reports on tests TSE-5, -5A, -6, and -7 are under preparation.

9.3 Plan of Action

The work breakdown structure for Task H.9 is shown in Fig. 9.1. The experimental efforts have been organized by test and these are supported by a common analytical subtask.

Under combined thermal-shock and pressure loading conditions an initially shallow flaw will begin to propagate in the frangible (brittle) toughness regime and from then on will encounter increasingly tougher and higher temperature material as the crack moves radially through the wall at very high speed. At some point, presumably there will be a transition from frangible fracture to ductile tearing with a considerable reduction in crack speed and perhaps arrest of the crack. This particular combination of postulated events is referred to as arrest on the upper shelf, and it has not yet received much attention. In the fracture-mechanics analysis of OCA's it has generally been assumed that crack arrest will not take place if a crack arrest toughness greater than ~200 MPa $\sqrt{m}$ is required. This may or may not be a reasonable approach.

Three experimental approaches are employed by the HSST program in investigating arrest on the upper shelf: experiments similar to the thermal-shock experiments with both thermal and pressure loading (Task H.10); wide-plate tests with a toughness gradient across the width of the plate (Task H.5); thermal-shock experiment so an experiment with the initial flaw located in a region having low Charpy energy at upper-shelf temperatures (low-upper-shelf material), designated as TSE-8, was tentatively planned to be conducted in FY 1985. However, it has been deferred indefinitely since the Thermal Shock Test Facility could not apply an appropriate test transient in its current configuration and since both wide-plate tests and a pressurized thermal shock test were planned.

PWR vessels are clad on the inner surface with an initially austenitic stainless steel material. Dilution with the base material and heat treating during application of the cladding leave the cladding material in an undetermined state with regard to fracture toughness and radiation-damage resistance. It is possible that the cladding resistance to crack propagation will be high enough to prevent surface extension of short flaws that extend through the cladding into the base material. If this is the case, and if it is reasonable to assume that initial flaws will not be long, it appears that failure of a PWR pressure vessel during an OCA will not be possible because a short flaw that cannot extend on the surface presumably cannot extend through the wall.

The effect of cladding on flaw behavior is being studied in Task H.7 using clad plates, and studies that are being carried out by other programs. Two experiments, TSE-9 and TSE-10, were tentatively scheduled here beyond 1987. Thermal-shock experiment TSE-7 demonstrated the ability of a short flaw to extend on the surface to become a very long flaw in the absence of cladding and serves as a base case for the clad TSE experiments if they are performed.

Before the cladding-effects thermal-shock experiments can be considered further, it is necessary that the studies of irradiation effects on the clad material be completed, as well as the pressurized thermal

Table 9.1. Summary o

Experiment identification	TSE-1	TSE-2	TSE-3	
Test cylinder				
Identification number	TSV-1	TSV-2	TSV-1	TS
Dimensions, mm (in.)				
Inside diameter	242 (9.5)	242 (9.5)	242 (9.5)	24
Outside diameter	533 (21)	533 (21)	533 (21)	53
Length	914 (36)	914 (36)	914 (36)	91
Material				
Heat treatment	Quench only from 871°C (1600°F)			
RTNDT, °C (°F)				
Flaw (initial)				
Orientation	Axial	Axial	Axial	Ax
Length, mm (in.)	914 (36)	38 (1.50)	914 (36)	91
Depth, mm (in.)	11 (0.42)	19 (0.75)	11 (0.42)	11
Generation Tech				
Thermal shock				
Cylinder temperature (initial), °C (°F)	288 (550)	289 (552)	291 (555)	29
Sink temperature, °C (°F)	4 (40)	-23 (-10)	-23 (-10)	-2
Quench medium	Water	Alcohol and water	Alcohol and water	Al wa
Flaw behavior demonstrated	Non-initiation	Initiation-arrest w/moderate surface extension	Initiation-arrest	In ar

[a]Intended flaw.

[b]Inadvertent flaw.

f thermal-shock experiments

TSE-4	TSE-5	TSE-5A	TSE-6	TSE-7
V-2	TSV-1	TSC-2	TSC-3	TSC-4
2 (9.5)	686 (27)	686 (27)	838 (33)	686 (27)
3 (21)	991 (39)	991 (39)	991 (39)	991 (39)
4 (36)	1220 (48)	1220 (48)	1220 (48)	1220 (48)
A508 class-2 chemistry				
	Tempered 613°C, 4 h	Tempered 679°C, 4 h	Tempered 613°C, 4 h	Tempered 704°C, 4 h
	66 (150)	10 (50)	66 (150)	-1 (30)
ial	Axial[a] Circumferential[b]	Axial	Axial	Axial
4 (36)	1220 (48)[a] ~6 (0.25)	1220 (48)	1220 (48)	38 (1.50)
(0.42)	16 (0.63)[a,b]	11 (0.42)	7.6 (0.30)	~15 (0.60)
Electron-beam weld				
1 (555)	96 (205)	96 (205)	96 (205)	93 (200)
5 (-13)	-196 (-320)	-196 (-320)	-196 (-320)	-196 (-320)
cohol and ter	LN_2	LN_2	LN_2	LN_2
itiation- rest	Series of initiation-arrest events; negligible dynamic effects; extensive surface extension of short flaw	Series of initiation-areas events; warm prestress; arrest in rising K_I field	Long crack jump w/neg. dynamic effects; arrest in rising K_I field; inability of flaw to penetrate wall	Extensive surface extension of short flaw

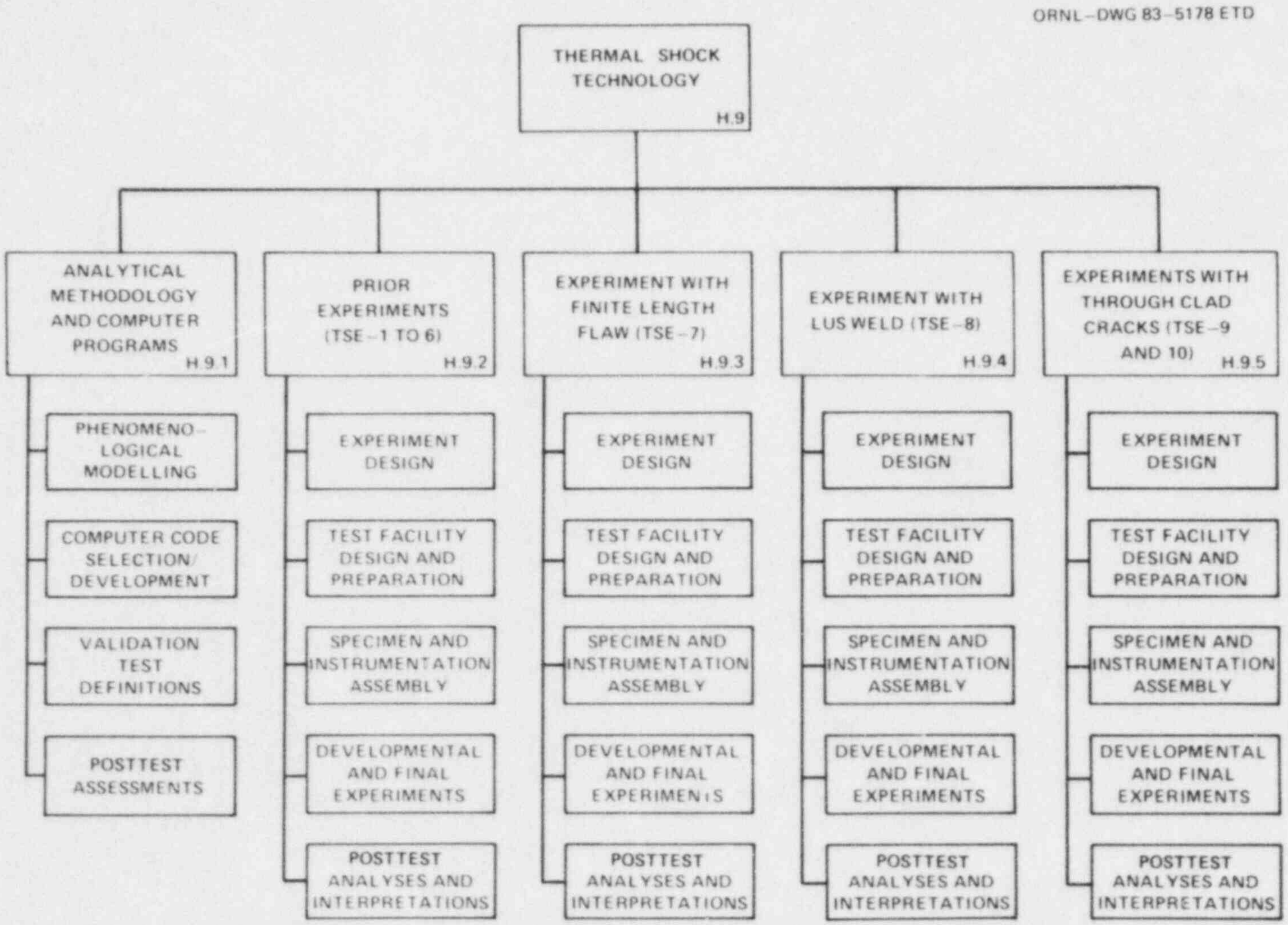


Fig. 9.1 Work breakdown structure for HSST Task H.9 Thermal Shock Technology.

shock test PTSE-3. These are being done as part of Task H.6 and H.10, respectively.

During FY 1985, the activity in this task will be directed at issuing reports on work previously completed. In particular, reports will be issued on experiments TSE-5, -5A, -6, and -7. A report will also be issued on the development of the influence coefficients incorporated into the OCA series of computer codes to capture 3-D aspects of flaw behavior. Finally a report will be issued on parametric analyses involving 2-D and 3-D flaw representations.

In preparation for potential future tests of clad vessels, a LN_2 spray technique will be designed and conceptually demonstrated to have the capability for applying an enhanced thermal shocks to TSE cylinders.

9.4 References

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2. R. D. Cheverton and S. E. Bolt, *Pressure Vessel Fracture Studies Pertaining to a PWR LOCA-ECC Thermal Shock: Experiments TSE-3 and TSE-4 and Update of TSE-1 and TSE-2 Analysis*, ORNL/NUREG-22, Oak Ridge National Laboratory, December 1977.

3. R. D. Cheverton, S. K. Iskander, and S. E. Bolt, *Applicability of LEFM to the Analysis of PWR Vessels Under LOCA-ECC Thermal Shock Conditions*, NUREG/CR-0107 (ORNL/NUREG-40), Oak Ridge National Laboratory, October 1978.

4. R. D. Cheverton and S. K. Iskander, *Application of Static and Dynamic Crack Arrest Theory to Thermal Shock Experiment TSE-4*, NUREG/CR-0767, (ORNL/NUREG-57), Oak Ridge National Laboratory, June 1979.

5. R. D. Cheverton, *Quick-Look Report for HSST Thermal-Shock Experiment TSE-5*, TSP-1006, Oak Ridge National Laboratory, August 1979.

6. R. D. Cheverton et al., "Thermal Shock Investigations," pp. 52-80 in *Heavy-Section Steel Technology Program Quart. Prog. Rep. for July-September 1979*, NUREG/CR-1197 (ORNL/NUREG/TM-370), Oak Ridge National Laboratory.

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13. R. D. Cheverton, *Heavy-Section Steel Technology Thermal-Shock Program Quick-Look Report for TSE-6*, TSP-1008, Oak Ridge National Laboratory, December 1981.

14. R. D. Cheverton et al., "Thermal Shock Investigations," pp. 52-92 in *Heavy-Section Steel Technology Program Quart. Prog. Rep. for October-December 1981*, NUREG/CR-2141, (ORNL/TM-8252), Oak Ridge National Laboratory.

15. R. D. Cheverton et al., "Fracture Mechanics Data Deduced from Thermal-Shock and Related Experiments with LWR Pressure Vessel Material," *Trans. ASME J. Pressure Vessel Technol.*, 105(2) (May 1983).

16. R. D. Cheverton, *Heavy-Section Steel Technology Thermal-Shock Program Quick-Look Report for TSE-7*, TSP-1009, Oak Ridge National Laboratory, June 1983.

17. R. D. Cheverton, S. K. Iskander, G. D. Whitman, "The Intensity of PWR Pressure Vessels During Overcooling Accidents," *Proceedings of the Internatinal Meeting on Thermal Nuclear Reactor Safety*, NUREG/CP-0027, 1 (February 1983).

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19. F. J. Loss, A. A. Gray, Jr., and J. R. Hawthorne, *Significance of Warm Prestress to Crack Initiation During Thermal Shock*, NRL/NUREG-8165, Naval Research Laboratory, Washington, D.C., September 1977.

20. S. K. Iskander, *Two Finite-Element Techniques for Computing Mode I Stress Intensity Factors in Two- and Three-Dimensional Problems*, ORNL/NUREG/CSD/TM-14, Oak Ridge National Laboratory, February 1981.

21. D. G. Ball et al., *OCA-II, A Code for Calculating the Behavior of 2-D and 3-D Surface Flaws in a Pressure Vessel Subjected to Temperature and Pressure Transients*, ORNL-5934, February 1984.

22. D. G. Ball and R. D. Cheverton, *OCA-P, A Probabilistics Fracture-Mechanics Code for Application to Pressure Vessels*, ORNL 5991, Oak Ridge National Laboratory.

9.5 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.9 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

Task: H.9 THERMAL-SHOCK TECHNOLOGY

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
H.9.1 <u>Analytical Methodology and Computer Programs</u>																							
A. Issue Report on OCA-II		▲																					
B. Issue Report on OCA-P and Coordinate with IPTS Studies			▲																				
C. Issue Report on Influence Coefficients								△															
D. Issue Report on (6/1, 2-m) Flaw Parametric Study																△							
E. Issue Report on Analyses of HDR Beltline Thermal-Shock Experiments																a ▽		b ▽			c ▽ △		
a. Complete a feasibility study in keeping with NRC directions																▽							
b. Complete analysis of design																		▽					
c. Complete posttest analysis																					▽		
H.9.2 <u>Prior Experiments</u>																							
A. Issue Report on TSE-5, -5A, and -6														△									

MILESTONE STATEMENT AND SCHEDULE

Task: H.9 THERMAL-SHOCK TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4	87	88	FY 88
H.9.3 Experiment with Finite Length Flaw (TSE-7)																							
A. Issue Report on TSE-7																△							
H.9.5 Experiments with Through-Clad Cracks																							
A. Complete Development of a LN2 Spray Technique																						△	
B. Issue Report on Tests of Clad Specimens Containing 3-D through-clad flaws (TSE-9 and 10)																							△ △

10. HSST TASK H.10 PRESSURIZED THERMAL-SHOCK TECHNOLOGY

10.1 Objective

The objective of Task H.10 is to test and evaluate the fracture response of thick vessels under thermo-mechanical stress fields and with fracture characteristics realistic to overcooling accident conditions to (1) demonstrate flaw initiation and arrest behavior under frangible and non-frangible conditions, and (2) determine the applicability of theoretical methods of fracture mechanics.

10.2 Background

This task was undertaken to provide experimental data to validate analytical methods for assessing the response of reactor pressure vessels under conditions representative of an overcooling accident. Fracture analysis capabilities had been developed to this point on the basis of extensive fracture properties tests and on assessments of structural tests that had been performed in support of nuclear and non-nuclear technology development. Many of the developments in support of nuclear applications had been carried out by other tasks of the HSST program. Whereas Task H.8 and H.9 had provided data and assessments on the fracture behavior of pressurized vessels and thermally shocked cylinders separately, prototypic tests had not been performed under the combined thermal and pressure loadings. Therefore, preparations were initiated for tests and analytical assessments of the response of intermediate test vessels (ITV's) under combined loadings to validate the applicability of fracture methodology.

Efforts were initiated in FY 1982 to carry out specific experiments to investigate the initiation, growth, and arrest behavior a shallow crack in thermally-shocked pressurized ITV's. Early feasibility studies revealed that experimental practicalities suggested that the test vessel should be flawed and thermally shocked on the exterior. As far as the technical objectives are concerned, this test configuration gives the same basis for methods validation as if the flaw and thermal shock were on the inside. In defining tests and in designing the test facility, several issues were to be addressed:

(1) intervention of the ductile upper shelf in arrest of a cleavage fracture
(2) effectiveness of a variety of types of warm prestressing, including marginal conditions and antiwarm prestressing
(3) behavior of short flaws in vessels with and without cladding and in regions with toughness and K_I gradients
(4) upper shelf tearing instability and
(5) arrest in high positive K_I gradients

In FY 1983 a major effort was devoted to preparing the pressurized thermal-shock test facility (PTSTF) in Building K-702, the abandoned

power house building at K-25 in Oak Ridge. The engineering firm (Rust Engineering) completed the basic construction of the facility in May 1983 on schedule and within budget. The facility was then turned over to ORNL for integration with the other test preparations. Other preparations for the first test (PTSE-1) that were concurrently and subsequently pursued in FY 1983 included (1) fabrication of two vessels for use in the shakedown experiment (PTSE-0) and PTSE-1, (2) design and installation of the instrumentation for vessels to be used in PTSE-0 and PTSE-1, (3) design and fabrication of specimen installation fixturing at the PTSTF, (4) vessel ballast, (5) design and assemblage of data acquisition system, (6) integration of all test systems, (7) material properties tests, and (8) performance of pretest analysis for test transient definition.

The shakedown experiment demonstrated the operability of the facility, and the first test series (PTSE-1A, -1B, and -1C) was performed in early FY 1984. This test of SA508 steel, which was heat treated to give a high RT_{NDT} (91°C), produced two run-arrest events that gave K_{Ia} values near 300 $MPa\sqrt{m}$.

10.3 Plan of Action

Within this five-year program plan, a series of three PTS experiments is envisaged. Each test is to assess the fracture characteristics associated with a representative PWR material, configuration, and loading. The first test, which was conducted in FY 1984, used a base RPV material, SA508 forging steel; the second test is to employ a low-upper-shelf steel, and the third test vessel is to be clad with a typical stainless steel cladding. The second test (PTSE-2) is scheduled for mid-FY 1986, and the third for FY 1987. The objectives of these three tests have been generally defined as:

(1) PTSE-1 investigated warm-prestressing and anitwarm prestressing effects on cleavage initiation and demonstrated the upper-shelf arrest of a crack initiated in cleavage.
(2) PTSE-2 is to demonstrate the propagation of a crack through the transition from cleavage to ductile tearing prior to arrest.
(3) PTSE-3 is to investigate the influence of stainless steel cladding on the initiation and growth of a short flaw.

The work breakdown structure for this task is shown in Fig. 10.1 and is centered around these three tests, their analyses, and the preparation of the required test facility. The principal criteria on which the test program is based are:

(1) The tests will be designed to challenge the predictions of analytical methods that are applicable to full-scale reactor pressure vessels under combined loading.
(2) The scale of the tests will be large enough to attain effectively full-scale restraint of the flawed region.

ORNL-DWG 83-5179 ETD

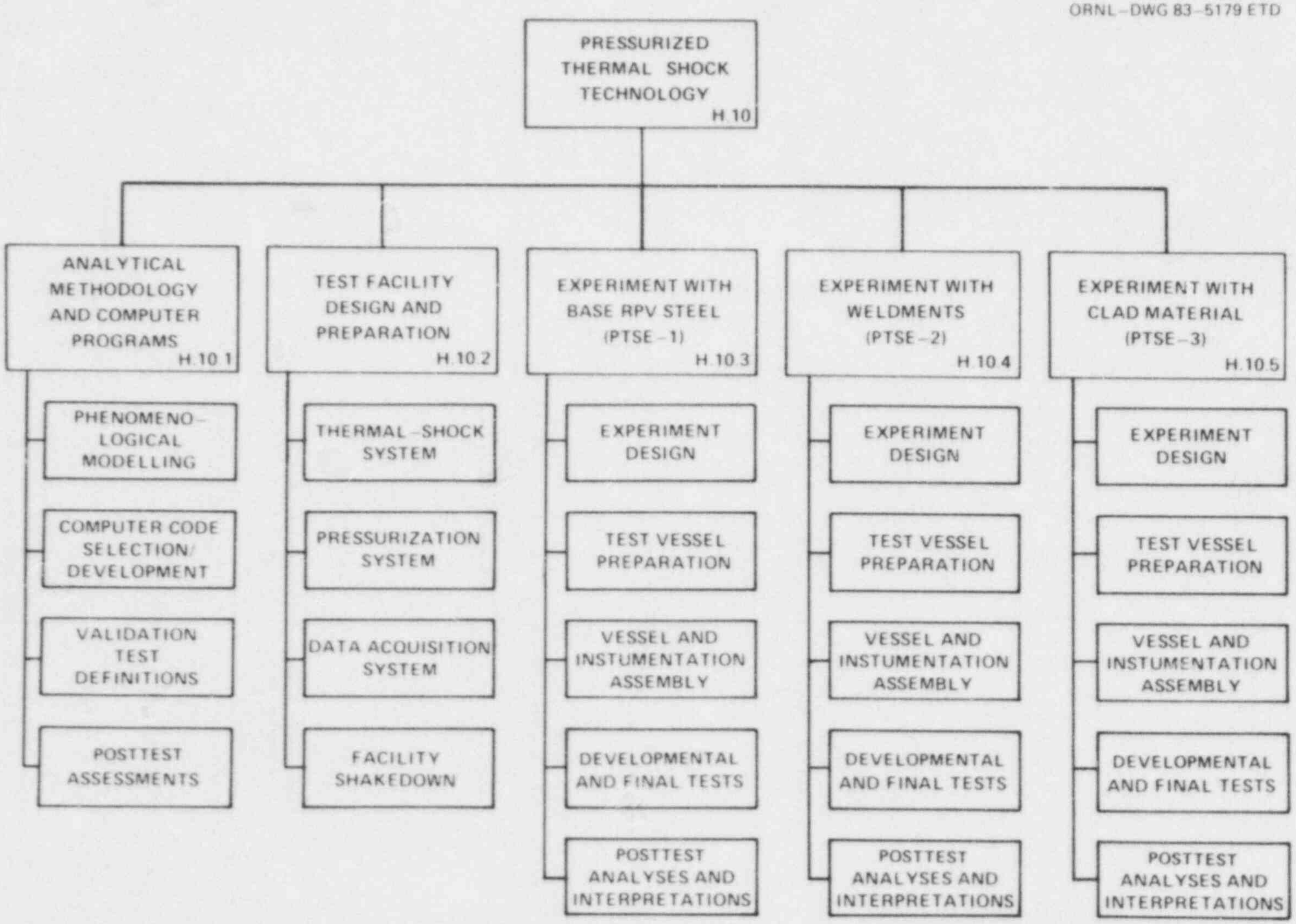

Fig. 10.1 Work breakdown structure for HSST Task H.10 Pressurized Thermal Shock Technology.

(3) Material in the flawed region will be characterized by specimen tests prior to each vessel test.
(4) Test conditions and materials will be selected to produce:
   a. realistic RPV stress fields and gradients around the flaw (with general primary membrane stress intensities less than ASME Code allowables) and
   b. realistic fracture-toughness conditions in the zone of action
(5) Loading conditions and controls will be used to prevent bursting the vessel (except when desired) to minimize damage to the test facility
(6) The test facility will be capable of producing (with realistic stresses) a variety of fracture possibilities:
   a. cleavage initiation of small flaws
   b. cleavage initiation and arrest below the upper shelf
   c. cleavage initiation with arrest on the upper shelf
   d. arrest in a high positive K_I gradient
   e. warm and nonwarm prestressing states in succession
   f. progressive (upper-shelf) tearing, tearing instability, and restabilization

As noted in the Background section above, pressurized thermal-shock tests are an extension of the large-scale experimental fracture mechanics studies carried out as parts of the thermal-shock (H.9) and intermediate vessel test (H.8) tasks. A particularly important test condition attainable only in the PTS facility is the contemporaneous toughness gradient and flexible loading history. This capability makes it possible to consider simulating, in the neighborhood of a crack, both the toughness regimes and stress states that are obtained in severe hypothetical overcooling accidents. Furthermore, fracture phenomena of importance in evaluation of overcooling accidents (for example, antiwarm prestressing and upper-shelf arrest) can be investigated in thick sections only in the PTS experiments.

The specific test conditions for PTSE-2 and -3 will not be finalized until information is available from other activities. Associated wide-plate crack-arrest tests (Task H.5) will be very relevant to defining PTSE-2. Results from the Seventh Irradiation Series studies from Task H.6 will be used in the definition of PTSE-3. The schedule shown on the milestone charts for PTSE-2 is based upon a presumption that a reworked test vessel will be delivered to ORNL at the time indicated. Excessive costs were proposed earlier by a prospective subcontractor, and this necessitated a relaxation of specifications and preparation of a new request for competitive bids. If further delays develop, it may become desirable to pursue the PTSE-3 test before PTSE-2.

10.4 Milestone Statement and Schedule

The statement and schedule for the milestones in Task H.10 are given in the following charts.

MILESTONE STATEMENT AND SCHEDULE

Task: H.10 PRESSURIZED THERMAL-SHOCK TECHNOLOGY

SUBTASK/MILESTONE	FY 84 (1 2 3 4)	FY 1985 (O N D J F M A M J J A S)	FY 86 (1 2 3 4)	FY 87	FY 88	Beyond FY 88
H.10.1 Analytical Methodology and Computer Programs						
A. Complete Pretest Analyses for PTSE-1						
B. Complete Pretest Analyses for PTSE-2 (LUS Material)			b c d			
a. Complete feasibility and design analyses						
b. Complete interpretation of characterization data						
c. Complete parametric analyses over ranges of conditions						
d. Identify test conditions						
C. Complete Pretest Analyses for PTSE-3			a	b c d		
a. Complete feasibility analyses for PTSE-3 (Clad Vessel) (See Milestone H.2.2.C.a)						
b. Complete interpretation of characterization data						
c. Complete parametric analyses over ranges of conditions						
d. Identify test conditions						

MILESTONE STATEMENT AND SCHEDULE

Task: H-10 PRESSURIZED THERMAL-SHOCK TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
H.10.2 Test Facility Design and Preparation																							
A. Complete Preparation of Thermal-Shock System	▲																						
B. Complete Preparation of Pressurization System	▲																						
C. Complete Preparation of Data Acquisition System	▲																						
D. Complete Facility Shakedown Evaluations	▲																						
H.10.3 Experiment with Base RPV Steel (PTSE-1)																							
A. Complete Experiment Design	▲																						
B. Complete Test Vessel Preparation	▲																						
C. Complete Performance of PTSE-1		▲																					
D. Complete Posttest Evaluations and Issue Report on PTSE-1			a ▼	b ▼ c ▼							△												
a. Complete data qualificaton and interpretation			▼																				
b. Complete posttest fracture examinations				▼																			
c. Complete fracture analyses and comparisons with data				▼																			

MILESTONE STATEMENT AND SCHEDULE

Task: H.10 PRESSURIZED THERMAL-SHOCK TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84 1	FY 84 2	FY 84 3	FY 84 4	FY 1985 O	N	D	J	F	M	A	M	J	J	A	S	FY 86 1	FY 86 2	FY 86 3	FY 86 4	FY 87	FY 88	Beyond FY 88
H.10.4 Experiment with Low Upper-Shelf Steel (PTSE-2)																							
A. Complete Design of PTSE-2 Using a Low Upper-Shelf Steel			a, b ▼	c ▲																			
a. Specify test material			▼																				
b. Establish test objectives			▼																				
c. Establish experimental requirements				▼																			
B. Complete Test Vessel Preparation				a ▼					b ▽							c ▽	d e ▽ f △						
a. Complete design of modified vessel				▼																			
b. Contract LUS steel fabrication (See Milestone H.5.3.C.a)									▽														
c. Complete material characterization (See Milestone H.3.1.E.b)																▽							
d. Complete procurement of thermocouple thimbles																	▽						
e. ORNL receive fabricated test vessel																	▽						
f. Generate flaw in vessel																	▽						

MILESTONE STATEMENT AND SCHEDULE

Task: H.10 PRESSURIZED THERMAL-SHOCK TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
C. Complete Performance of PTSE-2																		a, b, c					
a. Complete instrumentations of vessel																							
b. Establish prototype transient and system parameters																							
c. Complete test																							
D. Complete Posttest Evaluations and Issue Report on PTSE-2																			a, b	c			
a. Complete data qualification																							
b. Complete posttest fracture examinations																							
c. Complete fracture analyses and comparisons between predictions and data																							
H.10.5 Experiment with Clad Material (PTSE-3)																							
A. Complete Design of PTSE-3 Using a Stainless Steel Clad Vessel																a	b	c					
a. Establish test objectives																							
b. Specify test vessel materials																							
c. Establish experimental ranges																							

MILESTONE STATEMENT AND SCHEDULE

Task: H.10 PRESSURIZED THERMAL-SHOCK TECHNOLOGY (continued)

SUBTASK/MILESTONE	FY 84				FY 1985												FY 86				FY 87	FY 88	Beyond FY 88
	1	2	3	4	O	N	D	J	F	M	A	M	J	J	A	S	1	2	3	4			
B. Complete Test Vessel Preparation																a ▽	b ▽			c ▽ d ▽ e △			
a. Design test vessel																▽							
b. Contract vessel cladding																	▽						
c. Characterize material																				▽			
d. ORNL receive clad vessel																				▽			
e. Generate flaw in vessel																				▽			
C. Complete Performance of PTSE-3																					a ▽ b ▽ c △		
a. Complete vessel instrumentation																					▽		
b. Establish prototype transient and system parameters																					▽		
c. Complete test																					▽		
D. Complete Posttest Evaluations and Issue Report on PTSE-3																					a ▽ b ▽	c ▽ △	
a. Complete data qualification and interpretation																					▽		
b. Complete posttest fracture examinations																					▽		
c. Complete fracture analyses and comparisons with data																						▽	

APPENDIX A

LIST OF TECHNICAL REPORTS PUBLISHED UNDER THE HSST PROGRAM

The major topical reports that have been published by the Heavy-Section Steel Technology (HSST) Program since the late 1960's have been tered into a numbered list entitled Program Technical Reports. The list is published in the front of each report for the convenience of readers. The current list follows:

1. S. Yukawa, *Evaluation of Periodic Proof Testing and Warm Prestressing Procedures for Nuclear Reactor Vessels*, HSSTP-TR-1, General Electric Company, Schenectady, New York (July 1, 1969).

2. L. W. Loechel, *The Effect of Testing Variables on the Transition Temperature in Steel*, MCR-69-189, Martin Marietta Corporation, Denver, Colorado (November 20, 1969).

3. P. N. Randall, *Gross Strain Measure of Fracture Toughness of Steels*, HSSTP-TR-3, TRW Systems Group, Redondo Beach, California (November 1, 1969).

4. C. Visser, S. E. Gabrielse, and W. VanBuren, *A Two-Dimensional Elastic-Plastic Analysis of Fracture Test Specimens*, WCAP-7368, Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (October 1969).

5. T. R. Mager and F. O. Thomas, *Evaluation by Linear Elastic Fracture Mechanics of Radiation Damage to Pressure Vessel Steels*, WCAP-7328 (Rev.), Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (October 1969).

6. W. O. Shabbits, W. H. Pryle, and E. T. Wessel, *Heavy-Section Fracture Toughness Properties of A533 Grade B Class 1 Steel Plate and Submerged Arc Weldment*, WCAP-7414, Westinghouse Electric Corporation, PWR Systems Divison, Pittsburgh, Pennsylvania (December 1969).

7. F. J. Loss, *Dynamic Tear Test Investigations of the Fracture Toughness of Thick-Section Steel*, NRL-7056, Naval Research Laboratory, Washington, D.C. (May 14, 1970).

8. P. B. Crosley and E. J. Ripling, *Crack Arrest Fracture Toughness of A533 Grade B Class 1 Pressure Vessel Steel*, HSSTP-TR-8, Materials Research Laboratory, Inc., Glenwood, Illinois (March 1970).

9. T. R. Mager, *Post-Irradiation Testing of 2T Compact Tension Specimens*, WCAP-7561, Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (August 1970).

10. T. R. Mager, *Fracture Toughness Characterization Study of A533, Grade B, Class 1 Steel*, WCAP-7578, Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (October 1970).

11. T. R. Mager, *Notch Preparation in Compact Tension Specimens*, WCAP-7579, Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (November 1970).

12. N. Levy and P. V. Marcal, *Three-Dimensional Elastic-Plastic Stress and Strain Analysis for Fracture Mechanics, Phase I: Simple Flawed Specimens*, HSSTP-TR-12, Brown University, Providence, Rhode Island (December 1970).

13. W. O. Shabbits, *Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B Class 1 Steel Plate*, WCAP-7623, Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (December 1970).

14. P. N. Randall, *Gross Strain Crack Tolerance of A533-B Steel*, HSSTP-TR-14, TRW Systems Group, Redondo Beach, California (May 1, 1971).

15. H. T. Corten and R. H. Sailors, *Relationship Between Material Fracture Toughness Using Fracture Mechanics and Transition Temperature Tests*, T&AM Report 346, University of Illinois, Urbana, Illinois (August 1, 1971).

16. T. R. Mager and V. J. McLoughlin, *The Effect of an Environment of High Temperature Primary Grade Nuclear Reactor Water on the Fatigue Crack Growth Characteristics of A533 Grade B Class 1 Plate and Weldment Material*, WCAP-7776, Westinghouse Electric Corporation, PWR Systems Division, Pittsburgh, Pennsylvania (October 1971).

17. N. Levy and P. V. Marcal, *Three-Dimensional Elastic-Plastic Stress and Strain Analysis for Fracture Mechanics, Phase II: Improved Modelling*, HSSTP-TR-17, Brown University, Providence, Rhode Island (November 1971).

18. S. C. Grigory, *Tests of 6-in.-Thick Flawed Tensile Specimens, First Technical Summary Report, Longitudinal Specimens Numbers 1 through 7*, HSSTP-TR-18, Southwest Research Institute, San Antonio, Texas (June 1972).

19. P. N. Randall, *Effects of Strain Gradients on the Gross Strain Crack Tolerance of A533-B Steel*, HSSTP-TR-19, TRW Systems Group, Redondo Beach, California (June 15, 1972).

20. S. C. Grigory, *Tests of 6-Inch-Thick Flawed Tensile Specimens, Second Technical Summary Report, Transverse Specimens Numbers 8 through 10, Welded Specimens Numbers 11 through 13*, HSSTP-TR-20, Southwest Research Institute, San Antonio, Texas (June 1972).

21. L. A. James and J. A. Williams, *Heavy Section Steel Technology Program Technical Report No. 21, The Effect of Temperature and Neutron Irradiation Upon the Fatigue-Crack Propagation Behavior of ASTM A533 Grade B, Class 1 Steel*, HEDL-TME 72-132, Hanford Engineering Development Laboratory, Richland, Washington (September 1972).

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39. J. G. Merkle, G. D. Whitman, and R. H. Bryan, *An Evaluation of the HSST Program Intermediate Pressure Vessel Tests in Terms of Light-Water-Reactor Pressure Vessel Safety*, ORNL-TM-5090, Oak Ridge National Laboratory, Oak Ridge, Tennessee (November 1975).

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52. R. D. Cheverton, S. K. Iskander, and S. E. Bolt, *Applicability of LEFM to the Analysis of PWR Vessels Under LOCA-ECC Thermal Shock Conditions*, NUREG/CR-0107 (ORNL/NUREG-40), Oak Ridge National Laboratory, Oak Ridge, Tennessee (October 1978).

53. R. H. Bryan, D. A. Canonico, P. P. Holz, S. K. Iskander, J. G. Merkle, J. E. Smith, and W. J. Stelzman, *Test of 6-in.-Thick Pressure Vessels, Series 3: Intermediate Test Vessel V-8*, NUREG/CR-0675 (ORNL/NUREG-58), Oak Ridge National Laboratory, Oak Ridge, Tennessee (December 1979).

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55. J. A. Williams, *Tensile Properties of Irradiated and Unirradiated Welds of A533 Steel Plate and A508 Forgings*, NUREG/CR-1158 (ORNL/SUB-79/50917/2), Hanford Engineering Development Laboratory, Richland, Washington (July 1979).

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57. P. P. Holz, *Flaw Preparations for HSST Program Vessel Fracture Mechanics Testing; Mechanical-Cyclic Pumping and Electron-Beam Weld-Hydrogen Charge Cracking Schemes*, NUREG/CR-1274 (ORNL/NUREG/TM-369), Oak Ridge National Laboratory, Oak Ridge, Tennessee (May 1980).

58. S. K. Iskander, *Two Finite Element Techniques for Computing Mode I Stress Intensity Factors in Two- or Three-Dimensional Problems*, NUREG/CR-1499 (ORNL/NUREG/CSD/TM-14), Computer Sciences Div., Union Carbide Corporation, Nuclear Division (February 1981).

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60. S. N. Atluri, B. R. Bass, J. W. Bryson, and K. Kathiresan, *NOZ-FLAW: A Finite Element Program for Direct Evaluation of Stress Intensity Factors for Pressure Vessel Nozzle-Corner Flaws*, NUREG/CR-1843, (ORNL/NUREG/CSD/TM-18), Computer Sciences Div., Oak Ridge Gaseous Diffusion Plant, Oak Ridge, Tennessee (March 1981).

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62. S. K. Iskander, R. D. Cheverton, and D. G. Ball, *OCA-1, A Code for Calculating the Behavior of Flaws on the Inner Surface of a Pressure Vessel Subjected to Temperature and Pressure Transients*, NUREG/CR-2113, (ORNL/NUREG-84), Oak Ridge National Laboratory, Oak Ridge, Tennessee (August 1981).

63. R. J. Sanford, R. Chona, W. L. Fourney, and G. R. Irwin, *A Photoelastic Study of the Influence of Non-Singular Stresses in Fracture Test Specimens*, NUREG/CR-2179, (ORNL/Sub-7778/2), University of Maryland, College Park, Maryland (August 1981).

64. B. R. Bass, S. N. Atluri, J. W. Bryson, and K. Kathiresan, *OR-FLAW: A Finite Element Program for Direct Evaluation of K-Factors for User-Defined Flaws in Plate, Cylinders, and Pressure-Vessel Nozzle Corners*, ORNL/CSD/TM-165, NUREG/CR-2494 (April 1982).

65. B. R. Bass and J. W. Bryson, *ORMGEN-3D: A Finite Element Mesh Generator for 3-Dimensional Crack Geometries*, NUREG/CR-2997, Vol. 1 (ORNL/TM-8527/V1), Oak Ridge National Laboratory, Oak Ridge, Tennessee (December 1982).

66. B. R. Bass and J. W. Bryson, *ORVIRT: A Finite Element Program for Energy Release Rate Calculations for 2-Dimensional and 3-Dimensional Crack Models*, NUREG/CR-2997, Vol. 2 (ORNL/TM-8527/V2), Oak Ridge National Laboratory, Oak Ridge, Tennessee (February 1983).

67. R. D. Cheverton, S. K. Iskander, and D. G. Ball, *PWR Pressure Vessel Integrity During Overcooling Accidents: A Parametric Analysis*, NUREG/CR-2895 (ORNL/TM-7931), Oak Ridge National Laboratory, Oak Ridge, Tennessee (February 1983).

68. D. G. Ball, R. D. Cheverton, J. B. Drake, and S. K. Iskander, *OCA-II, A Code for Calculating Behavior of 2-D and 3-D Surface Flaws in a Pressure Vessel Subjected to Temperature and Pressure Transients*, NUREG/CR-3491 (ORNL-5934), Oak Ridge National Laboratory, Oak Ridge, Tennessee (February 1984).

69. A. Sauter, R. D. Cheverton, and S. K. Iskander, *Modification of OCA-I for Application to a Reactor Pressure Vessel with Cladding on the Inner Surface*, NUREG/CR-3155 (ORNL/TM-8649), Oak Ridge National Laboratory, Oak Ridge, Tennessee (May 1983).

70. R. D. Cheverton and D. G. Ball, *OCA-P, A Deterministic and Probabilistic Fracture-Mechanics Code for Application to Pressure Vessels*, NUREG/CR-3618 (ORNL-5991), Oak Ridge National Laboratory, Oak Ridge, Tennessee (MAY 1984).

71. J. G. Merkle, *An Examination of the Size Effects and Data Scatter Observed in Small Specimen Cleavage Fracture Toughness Testing*, NUREG/CR-3672 (ORNL/TM-9088), Oak Ridge National Laboratory, Oak Ridge, Tennessee (April 1984).

72. C. E. Pugh et al., *Heavy-Section Steel Technology Program — Five-Year Plan FY 1983—1987*, NUREG/CR-3595 (ORNL/TM-9008), Oak Ridge National Laboratory, Oak Ridge Tennessee (April 1984).

73. D. G. Ball, B. R. Bass, J. W. Bryson, R. D. Cheverton, and J. B. Drake, *Stress Intensity Factor Influence Coefficients for Surface Flaws in Pressure Vessels*, NUREG/CR-3723 (ORNL/CSD/TM-216), Oak Ridge National Laboratory, Oak Ridge, Tennessee (February 1985).

74. W. R. Corwin, R. G. Berggren, and R. K. Nanstad, *Charpy Toughness and Tensile Properties of Neutron Irradiated Stainless Steel Submerged-Arc Weld Cladding Overlay*, NUREG/CR-3927 (ORNL/TM-9709), Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 1984).

75. C. W. Schwartz, R. Chona, W. L. Fourney, and G. R. Irwin, *SAMCR: A Two-Dimensional Dynamic Finite Element Code for the Stress Analysis of Moving CRacks*, NUREG/CR-3891 (ORNL/Sub/79-7778/3), University of Maryland, College Park, MD (November 1984).

76. W. R. Corwin, G. C. Robinson, R. K. Nanstad, J. G. Merkle, R. G. Berggren, G. M. Goodwin, R. L. Swain, and T. D. Owings, *Effects of Stainless Steel Weld Overlay Cladding on the Structural Integrity of Flawed Steel Plates in Bending, Series 1*, NUREG/CR-4015 (ORNL/TM-9390), Oak Ridge National Laboratory, Oak Ridge, Tennessee (April 1985).

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78. R. D. Cheverton, D. G. Ball, S. K. Iskander and R. K. Nanstad, *Pressure Vessel Fracture Studies Pertaining to the PWR Thermal-Shock Issue: Experiments TSE-5, TSE-5A and TSE-6*, NUREG/CR-XXXX (ORNL-6163), Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory Oak Ridge, Tenn. (in preparation).

APPENDIX B

LIST OF TECHNICAL OR PROGRAMMATIC MANUSCRIPTS PUBLISHED UNDER THE HSST PROGRAM

Reports published under the HSST program that present interim data or results of limited studies that support larger goals are entered and numbered in a list entitled Program Technical or Programmatic Manuscripts. The list is published in the front of each report for the convenience of readers. The current list follows:

1. A Guide for Material Control and Data Control for the Heavy Section Steel Technology Program, Oak Ridge National Laboratory, June 15, 1968 (prepared by ORNL Inspection Engineering Department).

2. C. L. Segaser, System Design Description of the Intermediate Vessel Tests for the Heavy Section Steel Technology Program, USAEC Report ORNL-TM-2849, Revised, Oak Ridge National Laboratory, July 1973.

3. HSST Intermediate Vessel Closure Analysis, Teledyne Materials Research Company, Waltham, MA, Technical Report E1253(b), March 25, 1970.

4. C. L. Segaser, Feasibility Study, Irradiation of Heavy Section Steel Specimens in the South Test Facility of the Oak Ridge Research Reactor, USAEC Report ORNL-TM-3234, Oak Ridge National Laboratory, May 1971.

5. D. A. Canonico, Transition Temperature Considerations for Thick-Wall Nuclear Pressure Vessels, USAEC Report ORNL-TM-3114, Oak Ridge National Laboratory, October 1970.

6. F. J. Witt and R. G. Berggren, Size Effects and Energy Disposition in Impact Specimen Testing of ASTM A 533 Grade B Steel, USAEC Report ORNL-TM-3030, Oak Ridge National Laboratory, August 1970.

7. G. D. Whitman and F. J. Witt, Heavy Section Steel Technology Program, USAEC Report ORNL-TM-3055, Oak Ridge National Laboratory, November 1970.

8. D. A. Canonico and R. G. Berggren, Tensile and Impact Properties of Thick-Section Plate and Weldments, USAEC Report ORNL-TM-3211, Oak Ridge National Laboratory, January 1971.

9. J. G. Merkle, L. F. Kooistra and R. W. Derby, Interpretations of the Drop Weight Test in Terms of Strain Tolerance (Gross Strain) and Fracture Mechanics, USAEC Report ORNL-TM-3247, Oak Ridge National Laboratory, June 1971.

10. J. G. Merkle, A Review of Some of the Existing Stress Intensity Factor Solutions for Part-Through Surface Cracks, USAEC Report ORNL-TM-3983, Oak Ridge National Laboratory, January 1973.

11. N. Krishnamurthy, Three-Dimensional Finite Element Analysis of Thick-Walled Vessel-Nozzle Junctions With Curved Transitions, USAEC Report ORNL-TM-3315, Oak Ridge National Laboratory, July 1971.

12. C. E. Childress, Manual for ASTM A-533 Grade B Class 1 Steel (HSST Plate 03) Provided to the International Atomic Energy Agency, USAEC Report ORNL-TM-3193, Oak Ridge National Laboratory, March 1971.

13. G. C. Robinson, Discussion of SwRI Model Parametric Tests, USAEC Report ORNL-TM-3313, Oak Ridge National Laboratory, June 1971.

14. F. J. Witt, The Equivalent Energy Method for Calculating Elastic-Plastic Fracture (canceled).

15. R. W. Derby and C. L. Segaser, Quality Assurance Program Plan, Intermediate Vessel Test Facility (HSST Program), USAEC Report-TM-3373, Oak Ridge National Laboratory, May 1971.

16. C. W. Hunter and J. A. Williams, Fracture and Tensile Behavior of Neutron-Irradiated A533-B Pressure Vessel Steel, Hanford Engineering Development Laboratory, Richland, Washington, Report No. HEDL-TME 71-76, February 6, 1971.

17. A. A. Abbatiello and R. W. Derby, Notch Sharpening in a Large Tensile Specimen by Local Fatigue, USAEC Report ORNL-TM-3925, Oak Ridge National Laboratory, November 1972.

18. S. A. Legge, Effects on Fracture Mechanics Parameters of Displacement Measurement Geometry for Varying Specimen Sizes, Westinghouse Electric Corporation, Pittsburgh, PA, Report No. WCAP-7926, June 1972.

19. F. J. Witt and T. R. Mager, A Procedure for Determining Bounding Values on Fracture Toughness K_{Ic} at Any Temperature, USAEC Report ORNL-TM-3894, Oak Ridge National Laboratory, October 1972.

20. J. G. Merkle, An Elastic-Plastic Thick-Walled Hollow Cylinder Analogy for Analyzing the Strains in the Plastic Zone Just Ahead of a Notch Tip, USAEC Report ORNL-TM-4071, Oak Ridge National Laboratory, January 1973.

21. K. K. Klindt and D. A. Canonico, Evaluation of Discontinuities in HSST Twelve-Inch-Thick Plate, USAEC Report ORNL-TM-4155, Oak Ridge National Laboratory, June 1973.

22. S. A. Legge, Analysis and Experimental Verification of the Thermal Behavior of a Four Inch Steel Section Undergoing Nuclear Heating, Westinghouse Electric Corporation, Pittsburgh, PA, Report No. WCAP-8022, December 1972.

23. R. W. McClung, K. K. Klindt and K. V. Cook, An Evaluation of the PVRC and EEI-TVA Programs for Pre- and In-Service Nondestructive Examination of Nuclear Pressure Vessels, June 1973 draft transmitted to USAEC with cover letter of 7/17/73 from G. D. Whitman to Director, RRD, subject: Transmittal of Report, An Evaluation of the PVRC and EEI-TVA Programss for Pre- and In-Service Nondestructive Examination of Nuclear Pressure Vessels.

24. G. C. Robinson, J. G. Merkle and R. W. Derby, Fracture Initiation Aspects of the Loss of Coolant Accident for Water Cooled Nuclear Reactor Pressure Vessels, draft copy transmitted to USAEC by letter of 9/28/73 from D. B. Trauger to H.J.C. Kouts, subject: Thermal Shock Report — HSST Program.

25. W. K. Wilson and J. A. Begley, Variable Thickness Study of the Edge Cracked Bend Specimen, Westinghouse Electric Corp., Pittsburgh, PA, Report No. WCAP-8237, November 1973.

26. J. A. Williams, Some Comments Related to the Effect of Rate on the Fracture Toughness of Irradiated ASTM A533-B Steel Based on Yield Strength Behavior, HEDL-SA 797, December 1974.

27. S. C. Grigory, Heavy Section Steel Program Tests of 6-Inch-Thick Tensile Specimens, Sixth Technical Summary Report, Southwest Research Institute, San Antonio, TX, SwRI Project 03-2520, April 19, 1974.

28. H. J. Bellucci, Three-Dimensional Elastic-Plastic Stress and Strain Analyses for Fracture Mechanics: Complex Geometries, MARC Analysis Researcf Corp., Palo Alto, CA, Report No. 09177 (TR 75), November 1975.

29. Richard Smith, Weld Repair of Heavy Section Steel Technology Program Vessel V-7, Electric Power Research Institute, Palo Alto, CA, EPRI NP-179 (ORNL/Sub/88242-76/1) August 1976, (prepared by Combustion Engineering, Inc., Chattanooga, TN, W. D. Goins and D. L. Butler).

30. C. W. Smith, M. Jolles, W. H. Peters, Stress Intensities for Nozzle Cracks in Reactor Vessels, Virginia Polytechnic Institute and State University, Blacksburg, VA, VPI-E-76-25 (ORNL/SUB/7015-1), November 1976.

31. C. W. Smith, W. H. Peters, W. T. Hardrath, and T. S. Fleischman, Stress Intensity Distributions in Nozzle Corner Cracks of Complex Geometry, Virginia Polytechnic Institute and State University, Blacksburg, VA, VPI-E-79-2 (NUREG/CR-0640, ORNL/SUB/7015-2), January 1979.

32. G. A. Clarke, An Evaluation of the Unloading Compliance Procedure for J Integral Testing in the Hot Cell, Final Report, Westinghouse Electric Corp., Research and Development Center, Pittsburgh, PA, NUREG/CR-1070 (ORNL/Sub-7394/1) October 1979.

33. W. R. Corwin, Assessment of Radiation Effects Relating to Reactor Pressure Vessel Cladding, NUREG/CR-3671 (ORNL-6047), Oak Ridge National Laboratory, July 1984.

34. H. A. Domian, Vessel V-8 Repair and Preparation of Low Upper-Shelf Weldment, NUREG/CR-2676 (ORNL/Sub/81-85813/1), Babcock & Wilcox Company, Alliance, Ohio, June 1982.

35. H. A. Domian, Vessel V-7 and V-8 Repair and Characterization of Insert Material, NUREG/CR-3771 (ORNL/Sub/82-52845/1), Babcock & Wilcox Company, Alliance, Ohio, May 1984.

36. J. J. McGowan, Tensile Properties of Irradiated Nuclear Grade Pressure Vessel Welds for the Third HSST Irradiation Series, NUREG/CR-4086 (ORNL/TM-9477), March 1985.

37. J. J. McGowan, Tensile Properties of Irradiated Nuclear Grade Pressure Vessel Plate and Welds for the Fourth HSST Irradiation Series, NUREG/CR-3978 (ORNL/TM-9516), January 1985.

APPENDIX C

LIST OF DOCUMENTARY REPORTS PUBLISHED UNDER THE HSST PROGRAM

An important concept in the HSST program has been the wide use of carefully characterized and documented test materials. In support of documenting fabrication information, separate topical reports have been prepared on some of the most significant materials used in the program. Those reports are listed below.

1. C. E. Childress, Fabrication History of the First Two 12-in. Thick ASTM A-533 Grade B, Class 1 Steel Plates of the Heavy-Section Steel Technology Program, Documentary Report 1, USAEC Report ORNL-4313, Oak Ridge National Laboratory, February 1969.

2. C. E. Childress, Fabrication History of the Third and Fourth ASTM A-533 Steel Plates of the Heavy-Section Steel Technology Program, USAEC Report ORNL-4313-2, Oak Ridge National Laboratory, Februay 1970.

3. C. E. Childress, Fabrication Procedures and Acceptance Data for ASTM A-533 Welds and a 10-in.-thick ASTM A-543 Plate of the Heavy-Section Steel Technology Program, Documentary Report 3, USAEC Report ORNL-4313-3, Oak Ridge National Laboratory, January 1971.

4. C. E. Childress, Fabrication and Mechanical Test Data for Acceptance of the First Six 6-Inch-Thick Intermediate Test Vessels of the Heavy-Section Steel Technology Program, USAEC Report ORNL-TM-4351, Oak Ridge National Laboratory, October 1973.

5. C. E. Childress, Fabrication and Mechanical Test Data for the Four 6-Inch-Thick Intermediate Test Vessels Made from Steel Plate for the Heavy-Section Steel Technology Program, Documentary Report 5, ORNL-TM-5074, Oak Ridge National Laboratory, January 1976.

APPENDIX D

LIST OF PROGRESS REPORTS PUBLISHED UNDER THE HSST PROGRAM

Since its inception, the HSST program has issued periodic progress reports. A listing of the numbers of all prior progress reports is included in the front of each issue for the convenience of the readers. The progress reports issued to date are given below.

1. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1967, USAEC Report ORNL-4176, Oak Ridge National Laboratory, January 1968.

2. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending February 29, 1968, USAEC Report ORNL-4315, Oak Ridge National Laboratory, October 1968.

3. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1968, USAEC Report ORNL-4377, Oak Ridge National Laboratory, April 1969.

4. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending February 28, 1969, USAEC Report ORNL-4463, Oak Ridge National Laboratory, January 1970.

5. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1969, USAEC Report ORNL-4512, Oak Ridge National Laboratory, March 1970.

6. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending, February 28, 1970, USAEC Report ORNL-4590, Oak Ridge National Laboratory, October 1970.

7. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1970, USAEC Report ORNL-4653, Oak Ridge National Laboratory, April 1971.

8. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending Febraury 28, 1971, USAEC Report ORNL-4681, Oak Ridge National Laboratory, December 1971.

9. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1971, USAEC Report ORNL-4764, Oak Ridge National Laboratory, April 1972.

10. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending February 29, 1972, USAEC Report ORNL-4816, Oak Ridge National Laboratory, October 1972.

11. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1972, USAEC Report ORNL-4855, Oak Ridge National Laboratory, April 1973.

12. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending February 28, 1973, USAEC Report ORNL-4918, Oak Ridge National Laboratory, February 1974.

13. Heavy-Section Steel Technology Program Semiannual Progress Report for Period Ending August 31, 1973, USAEC Report ORNL-4971, Oak Ridge National Laboratory, July 1974.

14. Quarterly Progress Report on Reactor Safety Programs Sponsored by the Division of Reactor Safety Research for April-June 1974, II. Heavy-Section Steel Technology Program, ORNL-TM-4655.

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52. Heavy-Section Steel Technology Program Semiannual Progress Report for October 1983-March 1984, NUREG/CR-3744, Vol. 1 (ORNL/TM-9154/V1) (June 1984).

53. Heavy-Section Steel Technology Program Semiannual Progress Report for April-September 1984, NUREG/CR-3744, Vol. 2 (ORNL/TM-9154/V2) (December 1984).

54. Heavy-Section Steel Technology Program Semiannual Progress Report for October 1984-March 1985, NUREG/CR-4219, Vol. 1 (ORNL/TM-9593/V1) (in preparation).

APPENDIX E

LIST OF PROGRAM PLAN DOCUMENTS PUBLISHED UNDER THE HSST PROGRAM

This report is the second in a series of annual five-year program plan documents for the HSST program. Prior to this series, one early comprehensive program plan document had been prepared. During the intervening years, annual budgetary and management documents were issued which stated the current program. The three multi-year program plan documents issued thus far are listed below.

1. G. D. Whitman et al., *Program Plan — The Heavy-Section Steel Technology Program*, Rev. 0, April 1, 1968, Rev. 1, February 27, 1970 (unnumbered document prepared for the USAEC).

2. HSST staff, *Heavy-Section Steel Technology Program — Five-Year Plan, FY 1983–1987*, NUREG/CR-3595 (ORNL/TM-9008), Oak Ridge National Laboratory, Oak Ridge, Tenn. (April 1984).

3. HSST staff, *Heavy-Section Steel Technology Program — Five-Year Plan, FY 1984–1988*, NUREG/CR-4275 (ORNL/TM-9654), Oak Ridge National Laboratory, Oak Ridge, Tenn. (June 1985).

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The second in an annual series of five-year program plan documents is presented for the Heavy-Section Steel Technology program. The program is carried out by the Oak Ridge National Laboratory for the Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research of the U. S. Nuclear Regulatory Commission. The program is aimed at advancing the understanding and validation of materials and structures behavior as they relate to light water reactor pressure vessel integrity. The program has nine technical tasks and a management function. A background statement and a plan-of-action is given for each. The nine technical tasks address fracture methodology and analysis, materials characterization, crack growth, crack arrest, irradiation effects, cladding evaluations, intermediate-vessel testing, thermal-shock testing, and pressurized thermal-shock experiments.

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