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Research Program Plan

Reactor Vessels

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

M. Vagins, A. Taboada

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FOREWORD

This document presents a plan for research in Reactor Vessels to be performed by the Materials Engineering Branch, MEBR, Division of Engineering Technology, (DET), Office of Nuclear Regulatory Research. It is one of four plans describing the ongoing research in the corresponding areas of MEBR activity, which are being published simultaneously in four volumes as follows: Vol. 1 Reactor Vessels, Vol. 2 Steam Generators, Vol. 3 Piping, and Vol. 4 Non-Destructive Examination. These plans have been updated and are more detailed expansions of those originally published as part of the Long Range Research Plan for the Office of Nuclear Regulatory Research in NUREG-1080 Vol. 1; for more complete information on background, interfaces, and utilization, the above cited report should be consulted.

These plans were originally written as internal NRC working documents to cover the five year period from FY 1985 through FY 1989, to foster better coordination between the offices of Nuclear Regulatory Research and Nuclear Reactor Regulation, and improve the understanding of the derivation, approach and scope of the research programs. The plans have also been very useful for expanding that circle of understanding of the programs to other parts of the NRC staff, to the ACRS, and to contractors as an important information source and planning base. It is therefore hoped that the readers will benefit from these more clearly delineated objectives, needs, programmatic activities, and interfaces together with the overall logical structure within which these exist.

Publication of these plans will make visible to industry and other interested individuals what our objectives are and how we are approaching the work in these important areas. It is noted that reports of progress in all the areas of MEBR research are published annually in the series of reports "Compilation of Contract Research for the Materials Engineering Branch, Division of Engineering Technology," NUREG-0975 (Vol. 3 Annual Report for FY 1984). It is intended that these plans will periodically be updated; therefore, comments on these plans are welcomed from all quarters. Comments need not be restricted to activities for the five year period covered, but may include comments on omissions or what might be considered for the longer term. Please address comments directly to me.

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Introduction

The ability of the licensing staff of the NRC to make decisions concerning the present and continuing safety of nuclear reactor pressure vessels under both normal and abnormal operating conditions is dependent upon the existence of verified analysis methods and a solid background of applicable experimental data. It is the role of this program to provide both the analytical methods and the experimental data needed. Specifically, this program develops fracture mechanics analysis methods and design criteria for predicting the stress levels and flaw sizes required for crack initiation, propagation, and arrest in LWR pressure vessels under all known and postulated operations conditions. To do this, not only must the methods be developed but they must be experimentally validated. Further, the materials data necessary for input to these analytical methods must be developed. Thus, in addition to methods development and large scale experimental verification this program also develops data to show that slow-load fracture toughness, rapid-load fracture toughness, and crack arrest toughness obtained from small laboratory specimens are truly representative of the toughness characteristics of the material behavior in pressure vessels in both the unirradiated and the irradiated conditions.

Another significant role of this program is to determine the ways and the extent to which the LWR environment (including temperature, stresses, coolant, and radiation) changes and degrades the materials of pressure vessels during their operational life. Thus elements of this program deal with the determination of the sensitivity of the pressure vessel's steel to fracture toughness degradation as a function of irradiation exposure and with methods such as thermal annealing, required to regain this toughness parameter. Also part of this program are studies to improve and standardize dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effects of neutron exposure to the steels. Incorporated into this general area of research is the study of the effect of the LWR environment upon the rate of growth of cracks which might exist in the pressure vessel wall.

1.0 Definition of Issues and Needs

1.1 Analytical Methods

The structural integrity of the reactor pressure vessel is very important. It is the only component of a nuclear power plant for which there is no redundant member. If the pressure vessel fails the results could be catastrophic. This concern has long been recognized. Research started in the late 1960's under the AEC was directed to ensure that such an eventuality did not occur. Intensive research since that time has led to the development and verification of analytic techniques that have been shown to be effective for the prediction of reactor pressure vessel structural performance under a wide set of plant normal operating and upset conditions. Two specific examples of the progress made in this area are the development and application of linear-elastic fracture mechanics (LEFM) and the recent development and application of elastic-plastic fracture mechanics (EPFM) to the analysis of defect containing pressure vessels under certain specific operating load and temperature conditions. While both methodologies are effective in the prediction of vessel behavior over specified vessel temperature domains, there remains a significant vessel wall temperature

range where the application of either technique does not render accurate prediction capability. This is exemplified by the present need to understand, analyze and develop methodologies for a predictive capability for reactor pressure vessels subject to a pressurized thermal shock (PTS) scenario. In such a scenario the temperature variations through the vessel wall are such that the inner portion of the wall is in a frangible or brittle state while the outer portion of the wall remains in a fully ductile condition. There remains a significant middle portion of the vessel wall that is described metallurgically as having toughness properties in the high transition region, neither fully brittle nor fully ductile. While the analytic methods developed to date can do a very effective job in describing the initiation, propagation and arrest of cracks in the inner portion of the vessel wall, as well as initiation propagation and arrest in the outer ductile portion of the wall, there exists no proven analytic techniques for describing the condition of cracks originating in the inner portion of the wall and running through this upper transition region and into the ductile outer portion. Because of this lack, present analytic procedures, as incorporated in the ASME Boiler and Pressure Vessel Code, Sections III and XI, employ only LEFM procedures for application to the PTS problem. While it is presently believed that for most of the materials employed in present day reactor vessels the use of only LEFM methods result in conservative (possibly over conservative) results, there do exist some operating reactor vessels which contain material exhibiting low-upper-shelf- toughness energy for which the use of these procedures and guide-lines as defined in Sections III and XI might result in unconservative findings. Thus, there exists a defined need for new and applicable analytic methods. There are ancillary areas of methods development that are needed before a full predictive capability can be demonstrated. These deal with the structural effect of a pressure vessel's cladding material and a true understanding of the mechanics of crack arrest. Presently that understanding is based completely on limited empirically derived data, making extrapolation of crack arrest methodology to all domains of interest impossible. Further, the need to fully understand and to be able to quantify and apply the observed phenomenon of warm prestressing (WPS) is important to the complete understanding of pressure vessel behavior.

1.2 Experimental Validation

Coupled with this need for the development of predictive analytic capabilities is the essential need to demonstrate their viability and applicability by means of verification experimentation. When the size of the components to which these developed analytic methodologies are to be applied are considered, coupled with the inhibiting effect that large section irradiated properties play in planning adequate experiments, the need for relatively moderate size experimental test beds that still effectively demonstrate real structure applicability becomes apparent.

1.3 Fracture Toughness Data

Presently available and newly developed analytic capabilities are dependent upon the knowledge of material behavior for all domains of applicability. All analytic methods in fracture mechanics require as input such material and toughness parameters as yield stress, σ_y, ultimate strength, σ_u; slow load, plane strain, fracture toughness, K_{IC}; rapid load, plane strain, fracture

toughness, K_{Id}; crack arrest toughness, K_{Ia}; Charpy energy toughness, C_v; and others, including those defining crack initiation and propagation properties in the dynamic domain. Many of these values presently incorporated into the ASME Code, in Sections III and XI, are based on very limited and sparse data. This is particularly true of these data for irradiated material. For instance the fracture toughness and crack arrest criteria in Sections III and XI of the code, K_{IR} and K_{IC}, K_{Ia}, respectively, which define the most conservative limits (or so-called lower bound) of all available valid data on slow-load, rapid-load fracture toughness and crack arrest toughness for specimens up to 12 inches thick are based upon only a few heats of steel, mostly plate and forging material and no weld material at all. Further, the method by which these curves are to be manipulated for application to develop toughness data for irradiated material is based upon very little definitive data. A critical need exists to expand the data base for these curves to the regime of higher toughness values and to verify the procedure for treating irradiated materials, not only for the materials already in the data base but for those, particularly weld material, not presently included. Because of the difficulty and expense of testing large size specimens, particularly irradiated specimens, considerable effort is needed to develop small specimen test procedures that accurately produce toughness values that describe the in-situ behavior of materials in pressure vessels, subject to the geometric and size constraints that actually exist. The process of developing the required data from relatively small test specimens is difficult for unirradiated material. For irradiated material this difficulty is magnified many times by the cost of the irradiation process to develop these specimens and the personnel protection requirements required during the testing of the specimens. Because of these difficulties and because it has been determined that the change in material and toughness properties of steels and weldments are functions of small variations in the materials' constituent chemical elements, the development of an irradiated material properties data base by testing a sufficient number of specimens becomes an almost impossible task. What is clearly needed is a mechanism model, which relates the change of a material's property to the irradiation (neutron) history and its chemical make-up. Such a development will allow the prediction of a material's behavior well beyond the very limited range of the present, empirically derived, data base. This is a particularly defined need as regards the development of predictive capabilities for dealing with the PTS issue and with the general problem of plant aging.

1.4 Dosimetry

Experimentally validated dosimetry methods are needed to significantly improve the accuracy of predictions and measurements of neutron fluence for vessel embrittlement analyses and for updating American Society for Testing and Materials (ASTM) standards that are, or will be, endorsed by regulatory guides. At present neutron flux can be measured with reasonable accuracy in experimental facilities. Based on irradiation in experimental facilities, Charpy V notch specimens are used to establish the empirical relationships between fluence and the reduction of the Charpy specimen's loss in energy toughness for a specific heat of material. In an operating reactor, the core flux leakage calculations form the basis for predictions of the fluence impinging on the vessel wall during the lifetime of the plant. These predictions are validated indirectly by periodically testing the dosimeters from the vessel's surveillance capsules and then using the developed

relationships to establish the effective fluence. This procedure has shown that significant errors exist in the calculational methods used to predict fluence. The research will reduce the error band in this calculational methodology and hence significantly improve the ability to accurately assess the integrity of these structures at any period in their expected life during both normal and postulated accident conditions. It is clear that there is an absolute need for the accurate prediction of fluence if material property values and hence accurate predictive capability for pressure vessel performance is to be established.

1.5 Environmentally Assisted Fatigue Crack Growth

A crack or defect in a reactor component will grow longer and deeper with time. The growth will be influenced by the cyclic stresses imposed during normal operational and upset conditions, by the temperature and water chemistry, and by the materials' composition and state. The rate of crack growth must be known accurately for the environmental parameters specific to nuclear plants. A defect, discovered through in-service inspection, may grow to a critical size during a given time into the future. Further the potential for forming a crack through a fatigue action on virgin material must be known accurately for design and inspection purposes. This takes on great importance when the consideration of the limitations on in-service inspections due to radioactive conditions are incorporated into inspection procedures and timing. The recognition that fatigue criteria should be incorporated in the ASME Boiler and Pressure Vessel Code came in the mid-1950's. A design curve for mechanical and thermally-induced stresses was derived which incorporated a method for evaluating mean stresses and cumulative damage. Data were developed and curves were derived based on those data which defined the fatigue life for certain grades of nuclear steels. These curves basically defined the initiation phenomenon of flaws caused by fatigue of an unflawed structure. When these curves were incorporated into Section III of the code as "design curves", factors of two on stress and twenty on load cycles, whichever was more conservative at each point, were applied to the mean curve for the various materials considered. These margins are explicitly not "safety factors", but were intended to cover unknowns of environment, size effects, residual stresses and data scatter. The curves were derived from small, smooth specimens, tested in air environments, at temperatures up to 350°C. Subsequent structural tests of intermediate sized pressure vessels and piping tended to verify the adequacy of the curves, but in many instances the margins built in were used up. None of the specimens tested to define the curves, nor any of the verification tests were conducted in LWR environments. Recently some work has been conducted to define fatigue curves for both smooth and notched specimens for LWR environments, but this work has been extremely limited. However, the sparse data developed indicated that the fatigue curves in use may not be conservative for all cases. In the intervening years fracture mechanics methods and testing for fatigue crack growth measurement (for existing cracks) was developed. Curves developed from these data were incorporated into the code with the inception of Section XI in 1971 and upgraded continually until the present version of Section XI has curves for an air environment and water environments with the water environment curves being dependent on range and type of variation of stress intensity. Though much has been accomplished over the years much remains to be done. A large quantity of data has been developed and much knowledge of the fatigue and particularly environmentally assisted fatigue

phenomena has been generated. Extremely important is the fact that it has not been shown that corrosion fatigue data, developed from laboratory specimens, which formed the bases for the curves in both Sections III and Sections XI, can be applied directly to a structural component, in this case a reactor pressure vessel, subjected to geometric constraints and environmental loading transients which typify an operating nuclear steam supply system. This must be done before unlimited confidence can be placed on the design and analysis curves presently utilized for licensing purposes.

1.6 Thermal Annealing

The provisions of Appendix G of 10 CFR 50 requires an adequate safety margin against the brittle fracture of nuclear reactor vessels. If during the life of a vessel it becomes embrittled by irradiation to a degree that Appendix G requirements are not satisfied it may be necessary to perform a thermal anneal of the reactor vessel beltline region to recover material toughness properties and restore an adequate safety margin. Appendix G requires that reactor vessels for which the predicted value of Charpy upper shelf energy values at end of life is below 50 ft-lbs or for which the predicted value of the adjusted reference temperature (RT_{NDT}) at end of life exceeds 200°F, must be designed to permit a thermal annealing treatment to recover toughness properties. Appendix G provides that such thermal annealing of a reactor vessel may be performed subject to the approval of the Director of the NRR. In the May 27, 1983 amendment to Appendix G of 10 CFR 50, the Commission stated that although the provisions dealing with thermal annealing of reactor vessels have not changed the investigations dealing with pressurized thermal shock effects have prompted some studies of in-situ annealing to resolve possible engineering difficulties and that results of such studies may require further amendments to Appendix G. The methodology for in-situ annealing to recover material fracture toughness properties of irradiated reactor vessel steels needs to be validated in order to develop appropriate licensing criteria, (Appendix G), recommendations for material codes (ASME Code, Section XI), and standards (ASTM) on the subject and to provide a basis for resolution of immediate safety issues, such as USI-A49 (PTS) and USI-A11 (Lower Upper Shelf Energy Materials). Specific items that need to be studied and evaluated include the following:

- o parametric recovery effects of annealing (temperature, time, chemistry of materials, etc.) and reirradiation effects on annealed vessel material;
- o system engineering problems that may develop when using actual heat treatment procedures on actual nuclear plant vessels;
- o residual stresses and possible distortions of the vessels and other components resulting from annealing maximum temperatures and transients;
- o and demonstration of annealing feasibility.

2.0. Description of Research Program

A very significant research program addressing the issues raised in the preceding section is being conducted by many laboratories. These

laboratories include ORNL, NBS (both Gaithersburg and Boulder), MEA, University of Maryland, University of Florida, University of Buffalo, SwRI, HEDL, and BCL. Many consultants are also involved in the studies. Some of these are Dr. Fong Shih, Mr. Everett Rodabaugh, Dr. David Broek, Dr. Edmund Rybicki, Mr. Ed Wessel and others. The size of the effort can be judged by the level of expenditures for FY-84 of approximately 9.5 million dollars. Though the objectives of many of the individual elements of this program are diverse they are all integrated to achieve the end result of establishing the structural integrity of commercial nuclear reactor pressure vessels under normal and upset accident conditions, all as functions of plant aging.

2.1 Analytic Methods Development and Experimental Validation

The preponderance of the work done in this area is being conducted by the staff of the HSST program at ORNL with complementary and integrated work being done by the Southwest Research Institute (SwRI), Battelle Columbus Laboratories (BCL), the University of Maryland (U of M) and the staff of Materials Engineering Associates, (MEA). The emphasis continues to be on providing analytical models for representing the behavior of finite flaws in reactor pressure vessel (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 in the pressure vessel wall, and in time.

2.1.1 Unified Fracture Mechanics (UFM) Methodology

The need to develop a unified fracture methodology to treat different regions of fracture behavior in a single, integrated approach is the specific goal of this research element. Under the current budgetary plan, the level of effort on the unified fracture analysis development is modest during FY-84 with a more vigorous effort in FY-85 when the predictions of candidate theories will be evaluated against the conducted pressurized thermal shock experiments and wide plate crack arrest tests. Inherent in any unified fracture theory is the understanding of the damage done to the material along the crack path in front of the crack face and the quantification of the size of the plastic zone size around the fracture process zone at the crack tip. For this compatible bases for stress analysis (constitutive equations) are required. Time (rate) and temperature dependent inelastic material response will be considered in terms of constitutive equations that do and do not distinguish between plastic and creep strains. The application of current theories of viscoplasticity will be assessed, first in terms of dynamic analysis of fast-running cracks and of arrest events. The laboratory tests that are required to identify and quantify properties that appear in these equations must be performed as a part of implementing the assessments for specific materials of interest. Assessments of the applicability of constitutive equations must 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 stress and strain gradients, (c) compatibility with practical 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. Present plans are for a combined ORNL, SwRI effort along these lines, with

assessments of programs made at the end of each year's period with continuation contingent upon prior progress. It is estimated that it will require 3 years to complete the study.

2.1.2 Upgrade of Analysis Methods and Computer Programs

The emphasis in this element of research is on the upgrading of existing methods and codes that are presently being applied to problem areas. Emphasis is placed on the analysis of the behavior of finite-length flaws in pressure vessels subject to overcooling conditions. A second emphasis is in the completion of the development of dynamic fracture analysis computer codes that are compatible with available fracture theories and constitutive equations. In addition to computer based methods, direct analysis methods are developed 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 report was prepared in FY-84 by the University of Maryland covering the development and qualifications of the dynamic fracture analysis program SAMCR. This code will be bench-marked against a BCL developed (Non NRC funded) dynamic program FRACDYN, and both will be validated by the series of wide plate tests to be conducted in FY-84, 85 and 86. Both these codes are 2-dimensional in format. To extend this capability to 3-dimensional structures, the ADINA-OVIRT system presently on line at ORNL will be modified to include dynamic capabilities. This modified ADINA-OVIRT code will then be used as the vehicle to incorporate the developed viscoplastic constitutive equations to implement the unified fracture methodology with relatively low developmental costs.

2.1.3 Simplified Analyses

A ligament instability analysis has been developed for an externally flawed vessel. An analogous ligament instability analysis will be developed for an internally flawed vessel. Although the theoretical foundations are the same for the two situations, an internally flawed vessel is the specific geometry of concern in overcooling accidents. This work is part of an effort to develop simplified analyses for a wide range of overcooling accidents that are based on direct approaches and not on finite-element or other numerical techniques.

2.1.4 Warm Prestress

Both the HSST staff and MEA will carry out studies to fully understand and quantify the warm prestress effect. A significant input will be made by Dr. Fong Shih as consultant to MEA. Classical theories of plasticity, such as isotropic hardening, kinematic hardening, and combined isotropic-kinematic hardening models will be used with available computer programs, such as ADINA, to examine load history effects. Small scale experiments, including beam and 2T-CT specimens will be carried out to validate the findings of the analytical effort.

2.2 Experimental Verification

2.2.1 Thermal Shock Experiments (TSE)s

Eight thermal-shock experiments (TSE's) have been conducted thus far. The first four (TSE-1, -2, -3, -4) dealt with the investigation of the behavior of shallow flaws and demonstrated initiation and arrest of long and short 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 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 part of the thermal-shock program has been the development of fracture-mechanics models and codes that are used for describing the flaw behavior trends and that are subsequently used for sensitivity studies and assessing the severity of the over cooling accident problem. Fracture mechanics codes developed for the thermal-shock program and which are still extensively used are FMECH, a 2 - and 3 dimensional finite-element fracture-mechanics code; OCA-II, a fracture-mechanics code base on superposition techniques; and OCA-P, a probabilistics fracture-mechanics code that is a combination of OCA-II and a Monte Carlo sampling routine.

2.2.2 Pressurized Thermal Shock Experiments (PTSE)s

Though many questions have been answered, and much of LEFM has been validated under realistic geometric and size constraint conditions during the TSE series, many other questions remain unresolved. Because of the nature of thermal shock loading only, the driving force intensity, K_I, is usually declining by the time a running crack reaches the midwall position of a thick cylinder. This condition leads to arrest conditions wherein the K_I field is usually falling or if rising it is doing so at a very slow rate. Further, the absolute value of the K_I field is never too high, reaching maximums of ~100 to 120 Ksi $\sqrt{in.}$ Under these conditions of test, once WPS occurs, there is no way that it can be overcome. During an actual pressurized thermal shock situation, once a crack is initiated, it propagates through a significantly rising K_I field. By the time the crack reaches mid-wall depth the K_I could reach values as high as 300 Ksi $\sqrt{in.}$ A condition is created where the crack penetrates rapidly to a position in the wall where it encounters vessel material fracture toughness at or above the nominal upper-shelf-toughness as determined by both correlations from Charpy data and from static J_{IC} determinations. Because of the extremely high strain rate at the crack tip and the material temperature, a condition exists where neither LEFM or EPFM analysis is applicable. This is one of the chief justifications for the development of the unified-fracture-mechanics methodology with the concurrent work on viscoplasticity. The analytic program to address these issues has been described earlier. Efforts were initiated in FY-1982 to carry out specific experiments to investigate the undefined parameters involved in the PTS issue. It was decided that this could only be done in an actual pressure vessel that is thermal shocked while containing significant pressure.

Early feasibility studies revealed that experimental practicalities suggested that the test vessel, (an intermediate test vessel (ITV)), should be flawed and thermally shocked on the exterior while pressure is applied normally on its inner diameter. 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 designing the proposed test series the following issues are being addressed:

(1) intervention of the ductile upper shelf in arrest of a fast running crack originating in the frangible zone of the pressure vessel,

(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-83, the pressurized thermal-shock test facility (PTSTF) was completed and four shake down tests were carried out to check out the pressure and thermal shock conditions and all the instrumentation. These tests were carried out with a test vessel identical in geometry and material toughness properties of the actual test vessel except with the absence of the flaw. The first pressurized thermal-shock experiment (PTSE-1) was conducted early in the second quarter of FY-84. This test investigated (a) multiple initiation and arrest in the frangible zone of a relatively long (30 inches) shallow ($a/w = 0.077$) flaw, (b) warm prestressing in a steeply rising K_I field, (c) overcoming warm prestress, reinitiating the crack for a relatively long crack jump, and (d) arrest or no arrest as the crack penetrates the vessel above the ductile upper shelf toughness of the vessel material. The second experiment, PTSE-2, is planned for FY-86. At this time the exact test parameters have not been established and will be so only after the results of the PTSE-1 are evaluated. However, possible experimental parameters to be investigated are (1) upper shelf arrest of a cleavage crack entering the ductile toughness region of a low-upper-shelf-energy weld material, and (2) the influence of cladding on the initiation, propagation and arrest of a short, shallow flaw.

2.3 Fracture Toughness Data Base

The research effort in this area is being carried out by the staff of the HSST program at ORNL and the staff of MEA. Considerable support effort is also being conducted by the National Bureau of Standards (NBS) (both at Gaithersburg and Boulder), BCL, University of Maryland, University of Buffalo, and University of Florida. The emphasis of the work is to expand the fracture toughness data base, both irradiated and unirradiated, to validate the Section III and Section XI fracture toughness curves and the shift procedure for the manipulation of these curves for irradiated material, the development of test and analysis procedures for developing accurate fracture toughness values from small specimens, continuing the basic research

in the determination of the role played by chemical alloying and residual elements of nuclear grade steels in the sensitivity of these materials to radiation embrittlement, and the development of a micromechanism model to fully understand the radiation embrittlement process and thus create a means of predicting this process beyond the empirically developed data base.

2.3.1 Small Specimen Fracture Toughness in the Transition Zone

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 in data and elevation of small specimen toughness values with respect to large specimen test results, particularly with regards to the planning and execution of TSE-5 and TSE-5A brought this problem to focus and motivated a deliberate study of the problem. It turns out that a semiempirical method of analysis developed by George Irwin in 1960, utilizing the β_{IC} factor, can be applied to this problem, but that a better understanding of the physical basis for this application is necessary to justify its use with generality. This method is presently being evaluated utilizing all available data generated from subsize specimens and is being utilized in all NRC ongoing fracture data development programs.

The University of Maryland 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 micro-cracking and ductile remaining ligaments, and is providing direct supporting information for the study of size effects as defined above. In support of this study of size effects the HSST program conducted an international round robin to evaluate the efficacy of the use of dynamic Pre-Cracked Charpy V-Notch (PCCV) specimens to define fracture toughness data. Eleven laboratories, including ORNL, volunteered to test specimens from material derived from the TSE-5A test. To date, ORNL and six other laboratories have delivered results. Though the program is continuing, preliminary results indicate that the PCCV specimens and test procedure render the same large degree of scatter and upwards elevation of toughness in the transition zone as 1 and 2TCT specimens.

All work in this effort is planned for completion by the end of FY-85.

2.3.2 Crack Arrest Technology

2.3.2.1 Development of An ASTM Standard

Compared to fracture toughness data, crack arrest data is very sparse and the technological basis of the crack arrest phenomenon is poorly understood. 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 crack advance. Thus for higher values of crack arrest the use of still higher values of K_o for initiation are required for only moderately high values of crack arrest, K_{Ia}.

Fairly rapidly, large plastic zones are introduced into the specimens at initiation, invalidating results. Regardless of this behavior, this specimen, sometimes called the "Ripling" specimen does give reasonably consistent results in the low shelf to mid transition region of fracture toughness curves. Thus, the University of Maryland under subcontract to ORNL is carrying out an international round robin under the auspices of the ASTM to complete the standardization of a crack arrest specimen and procedure. The University of Maryland will carry out coordination, test material distribution data interpretation and evaluation and prepare a report of its findings, both to ORNL and the ASTM. This round robin testing and reporting will be complete in FY-85.

2.3.2.2 Wide Plate Tests

As discussed above, very little crack arrest data exists for K_{Ia} values above a value of 150 Ksi $\sqrt{in}$. Some measurements have been made by Japanese experimenters using edge cracked, wide-plate (ESSO) specimens. Though these tests have rendered K_{Ia} values as high as 350 Ksi $\sqrt{in}$, and in general have verified the extension of the K_{IR} curve to this value (at least for Japanese nuclear grade pressure vessel steels), the tests have been poorly instrumented and documented making their use in the regulatory process questionable. Under the direction of the HSST program, the ORNL staff, BCL, NBS, University of Maryland and SwRI are involved in a significant program to achieve valid high toughness crack arrest data. This effort has two components; the development of a relatively small specimen that will render high K_{Ia} values; and the carrying out a series of wide plate tests. The key to achieving such high values, and the point of similarity between both efforts is to achieve a rising K_I field with crack advance, and the achievement of a rising fracture toughness resistance or K_{IC} field by means of a temperature gradient. Efforts at developing a relatively small specimen to achieve these results will be carried out by ORNL, BCL and NBS personnel.

Tapered height, tapered side groove and eccentrically loaded specimen configurations will be investigated. It is noted that EPRI is sponsoring a similar effort by Combustion Engineering. This effort is being followed closely. It is believed that very high toughness values for K_{Ia} will be achieved by the planned wide plate tests. These tests will be composed of three series of six tests each. The test specimens are to be roughly 1m. x 1m. x 0.1m, welded between long grips with approximately 10.0 meters between pin holes. The first series will be highly instrumented with thermocouples, strain gauges and timing wires, so that the entire time history of the initiation and run-arrest events are recorded and analyzed completely. These analyses are to verify the achievement of the arrest in a steeply rising K_I field with arrest occurring at or above upper shelf conditions at the point of arrest. If verification can be achieved, the remainder of the tests could be run with minimal instrumentation. Planning for these tests calls for the first series to be carried out with quenched and tempered A533B plate, the second series with a low upper energy shelf plate, and the third series with a A533B material that has been heat treat to simulate a degree of radiation embrittlement. The three series will be carried out in FY 85, and -86, respectively. It should be remembered that the PTSE-1 and PTSE-2 will also render data on crack arrest at high toughness values.

All the data developed above and all prior and ongoing publicly available crack arrest data are being incorporated into a computerized crack arrest data bank for the NRC by BCL.

2.3.3 Irradiated Fracture Toughness

2.3.3.1 HSST Irradiation Series 1, 2, 3, and 4

The Welding Research Council (WRC) published WRC Bulletin 175 in August 1972 and established a reference stress intensity curve (K_{IR}) that was constructed as a lower bound to K_{IC}, K_{Id}, and K_{Ia} data then available for A533 Grade B Class 1 and A508 steels. This curve was incorporated into the ASME Code, Section III and is used as a guideline for operation of reactors to provide protection against nonductile fracture of reactor pressure vessels. The effects of irradiation on initiation fracture toughness (K_{IC} and K_{Id}) were not so well understood. The HSST program began a series of irradiation experiments to improve this understanding. The first irradiations, Series 1 examined the static and dynamic fracture toughnesses of irradiated 4T-CT specimens. Both base plate and submerged-arc weld metal specimens were used. The results of Series 1 showed that fracture initiation toughness versus test temperature curves shifted to higher temperatures by an amount approximated by the shift in C_v test results for the same materials. However, there were too few specimens to examine the shape of the shifted curves and so few large specimens were tested that a statistical analysis was not possible. Series 2 and 3 carried out irradiations and testing of a significant number of specimens, CT's, C_v's and tensiles, of low upper shelf energy weldments. All testing was conducted at the upper shelf temperatures and resulted in a comprehensive evaluation of the effects of irradiation on the upper shelf ductile fracture toughness of these materials. Series -1, -2, and -3 are complete. Series 4 is also a ductile upper shelf study except that submerged-arc welds fabricated with low copper content and current practice welding procedures are being examined as well as the same plate material used in Series 1. (Low upper-shelf material from the Federal Republic of Germany are included in this series.) Series 4 includes only 1TCT specimens, in addition to Charpy and tensile specimens. 240 CT specimens have been irradiated, and the testing of these specimens (K_{IC}, J_K) is presently underway with testing being split about evenly between ORNL and MEA. There are enough specimens to give statistically meaningful data as well as ensuring the elimination of the "one lab bias" syndrome. It is planned that testing, evaluation and reporting of the 4th Series will be complete in FY-85.

2.3.3.2 HSST Irradiation Series 5

The Series-5 irradiation program was initiated in early FY-83, with the ordering of 90-lineal feet of two weldments (45 ft. each). These weldments are fabricated with weld rod containing .25% and .35% copper, respectively. These materials are being fabricated into a large number of compact, Charpy, tensile and drop weight specimens. Specimens as large as 8TCT unirradiated and 4TCT irradiated will be tested. The target fast-neutron irradiation fluence will be $1.5 \times 10^{19} n/cm^2$ ($E > 1$ MeV). The objectives of this series are:

1. Develop unirradiated and irradiated K_{IC} data for two copper content weldments with valid K_{IC}'s in excess of 130 Ksi $\sqrt{in}$. Lower valid data are to be achieved with smaller specimens. Higher, invalid data are to be developed and adjusted by use of the β_{IC} correction method.

2. Tensile properties of unirradiated and irradiated materials will provide data for both determining test parameters for the fracture toughness tests and for analyzing the developed data.

3. 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 C_v 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 curves for both the irradiated and unirradiated materials.

4. The results of the experiments should establish, at least for reactor vessel weldments, not only the correlation on temperature shifts between Charpy curves at the 41J energy value but the actual shape of the irradiation shifted K_{IC} curve.

The irradiations will be carried out in the Oak Ridge Reactor (ORR) commencing in February 1984 and continue until the end of FY-85. Testing of the specimens, both unirradiated and irradiated commenced in March of 1984 and will continue until the end of FY-86, with completion of reporting to be accomplished during FY-87.

2.3.3.3 MEA Split Melt Irradiation

A much smaller, complementary study will be conducted solely by MEA using 5 split melt plates of A533B and A302B nominal compositions, where the nickel, phosphorous and copper contents will all be varied. Irradiated and unirradiated C_v, tensile and 0.5TCT specimens will be employed as well as unirradiated drop weight specimens. Though this program is basically directed to determine the effect of alloying and residual elements of the steels on the correlation of C_v 41J and 100 MPa $\sqrt{m}$ shift temperature, it will also give much information on the shape of the shifted fracture toughness curves for plate materials. This will be accomplished by developing K_{Jc} and K_{Jd} data from the 0.5 TCTs and converting those data using the β_{IC} correction methodology. Irradiation of specimens began in the last quarter of FY-84 with the first series of testing complete by the end of the second quarter of FY-85. Irradiations and testing will continue until the end of FY-86 with full reporting in FY-87, thus paralleling the HSST program.

2.3.3.4 HSST Irradiation Series -6 and -7

The same material used for the HSST Series-5 irradiation program will be used for the Series-6 program. This series has as objectives the validation of the C_v transition shift correlation and determination of the K_{Ia} curve shift shape, thus paralleling the objectives of the Series 5 program for the crack arrest curve. However, because of the ongoing work in the crack arrest

technology this program has been delayed until some of the uncertainties involved in K_{Ia} testing have been resolved. Thus, though the material has been procured, it is anticipated that specimen fabrication will not commence until late in FY-85, with irradiations and testing not to be completed until late in FY-87. The HSST Series 7 irradiation program involves the determination of irradiation effects on A308 and A309 stainless steel cladding material. The first 3 capsules of specimens, mostly C_v's and tensiles with some 0.5CT have been irradiated and testing was complete by the end of the 4th quarter of FY-84. This material is either A308 or A309, deposited by use of a "one-wire" method on base material of A533B. Sufficient material was deposited to allow the fabrication of the specimens from the deposited material and also to allow enough specimens to be developed through the thickness of deposit to determine the effect of weld and base plate material diffusion. 3-wire common current practice weld deposits have been ordered from Combustion Engineering and will be delivered early in FY-85, and will be fabricated into C_v, tensile and 0.5CT specimens by mid FY-85. Irradiations will commence during FY-85 and testing will be concluded in FY-86.

2.3.3.5 MEA Stainless Steel Irradiation

MEA will conduct a relatively modest complementary program to determine the ability of the stainless steel cladding to inhibit or mitigate the initiation and propagation of small flaws through the cladding of reactor pressure vessels. This program involves the fabrication of clad, part-through cracked plates, and testing them in tension, in both the unirradiated and irradiated conditions to determine the apparent K_{IC}'s. Also to be tested are unclad part-through (or surface) cracked plates. Thus allowing establishment of K_{IC}'s for unclad plates, unirradiated plates and irradiated plates, giving insight into what would occur in an actual reactor. Clad specimens will be fabricated using the same 3-wire process as for Series-7, by C.E. All unirradiated testing will be complete by the end of FY-85 and irradiated testing by late FY-86, again paralleling the HSST program.

2.3.3.6 Dose Rate Effect

Recent comparisons of embrittlement data from long-term low flux rate sources (power reactors) and short-term, high flux rate sources (test reactors) tend to suggest that there is a time-temperature dependency of the embrittlement process. Damage rate can be thought of in terms of net defect production, i.e., the number of defects generated, minus the thermal self annealing losses. Thus, net damage production may be a composite function of damage rate, irradiation time, irradiation temperature, and composition. Since the preponderance of irradiation damage data has been and, for the foreseeable future, probably will continue to be generated in test reactors, it is necessary to define this net damage phenomenon. MEA is carrying out this investigation. The plan is to irradiate two plate materials, A302B and A533B and two submerged-arc weldments, A533B with Linde 80 and A533B with Linde 0091 flux, all at three neutron flux rate levels ($\sim 7 \times 10^{10}$, $\sim 4 \times 10^{11}$, and $\sim 8 \times 10^{12}$ n/cm^2/sec), using the reflector region, the core edge region and an in-core location in the NSTF reactor of the University of Buffalo. Core-edge and in-core irradiations are to be made to 0.5, 1.0, and 2.0×10^{19} n/cm^2 (E >1Mev) thus giving 3 fluence levels at each location. The

capsules at the reflector location, which receives a flux rate of 7×10^{10} n/cm2/sec will be irradiated to a fluence level of only 0.5 x 10^{19} n/cm^2 because of the long time involved in achieving this level. At all positions spectrum data will be obtained. Post irradiation test determinations for K_{IC}/K_{JC} fracture toughness, C_v values and tensile values will be carried out to quantify any variation of the change in these values as a function of the dose rate at a given temperature. Some irradiations are underway, however the program will not be completed until mid FY-87.

Another approach to determination of the existence of a dose rate effect is based on the use of experimental data from long-term service in a power reactor. This is being done through measurements of the fracture properties of the steel pressure vessel wall of the decommissioned German Gundremmingen KRB-A reactor. Some 15 4-inch diameter trepans have already been removed from the beltline region of the vessel and are being machined at the Material Testing Laboratory, MPA, University of Stuttgart FRG where they will also be tested. These data will provide the long-term (about 12 years) slow irradiation rate comparison basis. The faster rate test reactor comparison will be provided from irradiations being done by MEA in the NSTF in typical accelerated irradiation positions, and also by the UK AERE Harwell in two in-core positions and one other flux converter position in a heavy water reactor. The neutron spectra for the KRB-A power reactor and all the test reactor positions will be carefully calculated and verified, as possible, with radiometric measurements. In addition to providing key information on the dose rate effect in pressure vessel steels, the KRB-A study will provide the first set of data on through-thickness embrittlement from an operating power reactor wherein the low irradiation rate, high operating temperature (550°F) and operating stress level (1000 psi) were constantly applied over the long time period. The five-inch thick vessel wall is just thick enough to provide validation of the radiation damage attenuation models predicted by the dpa damage-fluence criterion.

2.3.4 Irradiation Damage Mechanism Model

From all the above description of ongoing and planned irradiation effects programs it is clear that continuing on the empirically based method path of determining radiation effects on nuclear grade steels and weldments, has been, is, and will continue to be expensive, time consuming and limited in information extrapolation to only the data base developed. What is needed is the development of a micromechanism model, based upon first principals, that would relate the steels metallurgical factors, net damage, flux, temperature and possibly other parameters, to the change in observed fracture toughness. Significant efforts have been expended in the past to accomplish just this, but have been largely unsuccessful. However, a more thorough understanding of the role played by metallurgical factors, gained through empirical testing and recent advances in analytical electron microscopy offer renewed hope that a significant breakthrough might now be possible. This new program will be conducted by MEA under the guidance of J. R. Hawthorne. The preponderance of work will be conducted by Dr.'s Hren, Bates and Newkirk at the University of Florida. The general plan initially is to study reference (binary and tertiary) iron alloys in the unirradiated and irradiated conditions. This will be followed by controlled melts of A533B and A302B nominal chemical compositions, where each constitutive chemical element is closely controlled.

The work will focus on advanced microscopy techniques for purposes of (a) confirmation of the basic mechanism by which copper content contributes to radiation sensitivity, (b) definition of the mechanism or means by which nickel alloying reinforces the copper content contribution, and (c) isolation of the mechanism by which phosphorous content contributes to the radiation sensitivity level. In addition to the above (primary) research goals the studies will explore, for purposes of identification, mechanistic models for other impurity element/alloying element interactions and, on the basis of all its findings, will assess the relative levels of contribution of the copper and phosphorus mechanisms to the overall radiation damage process. This element of the overall research program will be initiated early in the second quarter, FY-84 and should be complete in mid FY-87.

2.4 Surveillance Neutron Dosimetry

The experimental programs are designed to produce validated, standard methods for the prediction, calculation and measurement of neutron flux and fluence parameters, and the correlation to embrittlement and degradation of mechanical properties of pressure vessel steel in LWRs. The result is achieved by establishing a series of benchmarks for use by vendors, laboratories, utilities, or others producing fluence and embrittlement values, and by NRC for review of submissions and for preparation of regulatory positions, such as Reg. Guide 1.99 on fluence and embrittlement relations for irradiated vessel steel and the rule on pressurized thermal shock. The program contains the elements of Benchmarks, Test and Power Reactor Validation Irradiation, Embrittlement-Damage Analysis, and Standards. The Standard practices, methods, etc., to be produced under this program are shown in Figure 1.

2.4.1 Benchmarks

These are a very carefully controlled series of experiments representing reactor configurations the results of which provide a calibration mark for the proof of accuracy of future predictions and analyses. In each case, theoretical calculations are made of the configuration (frequently by several laboratories using different methods and data bases) followed by extensive measurements of dosimetry both during the irradiation and after the experiment is shut down. Further, determination of changes in metallurgical properties are often included. Discrepancies between calculations and measurements are resolved, and the entire experiment is documented for future use. Documentation includes the as-built physical dimensions and configuration, the physical constituents (fuel loading, purity, chemical compositions, etc.) calculation methods and results, dosimetry materials, counting results and analyses, and metallurgical data (pre- and post-irradiation if part of the experiment). The benchmarks which have been or are being produced by the program are described below:

- PCA (Pool Critical Assembly): A physics-dosimetry mock-up of a LWR thermal shield/water gap/vessel wall that allows for reactor physics calculations to predict neutron flux and spectrum in this environment and extensive dosimetry measurements to verify the calculation. The thermal shield was 1-inch thick stainless steel, and the vessel was 8.0-inch thick carbon steel; water gaps of several different dimensions were used to simulate different distances of the thermal shield from the core and the shield from the vessel. This benchmark is complete.

- PSF (Pool-Side Facility): A physics-dosimetry-metallurgy mock-up that is a duplicate of the PCA except that surveillance capsules were placed at locations in front of and behind the thermal shield, and capsules were substituted in the vessel wall at the inner surface, 1/4T and 1/2T locations; contained within the capsules were metallurgical test specimens of typical pressure vessel steels and welds, so that following irradiation, the change in properties of the steel at the different locations could be measured. Dosimetry and embrittlement measurements are complete, and a blind test of embrittlement predictions by different persons both in the USA and abroad is essentially complete.

- SDMF (Surveillance Dosimetry Measurement Facility): A series of mock-ups of LWR surveillance capsules within the PCA experimental configuration to determine the effect of the surveillance capsule itself upon the resulting dosimetry values measured from the irradiation. Typical Westinghouse surveillance capsules have already been studied, and B&W and CE types are nearing completion.

- VENUS: A program undertaken by the Belgian Nuclear Laboratory at Mol, Belgium, with support provided by the NRC SDIP program, to predict and to measure the flux spectrum and intensity continuously from the source in the last row of fuel elements completely out into the vessel wall. Current practice reactor physics calculations are made in the core, stopping at the edge of the core; vessel dosimetry analysts then assume that calculation to be correct and use it as a source for flux and spectrum degradation through the thermal shield, water and vessel. This benchmark makes no such assumption, but rather couples these two calculations to provide one continuous analysis. The experimental work is essentially complete.

- NESDIP (Nestor Dosimetry Improvement Program): A program undertaken by the UKAEA, Winfrith Laboratory using the Nestor reactor facilities, to establish benchmarks for dosimetry in the cavity between the vessel wall and the biological shielding. Cavity dosimetry is gaining increasing interest because it permits data to be obtained from a reactor wherein conventional in-core and surveillance dosimetry cannot be obtained (for example, when surveillance capsules cannot be reinserted). USA contributions to this program are in a support role. This benchmark will consist of theoretical calculations validated by measurements. The experimental work will be completed in 1986.

2.4.2 Test and Power Reactor Validation Irradiations

In addition to the irradiations conducted in the above noted benchmarks, additional work is being conducted in carefully selected reactors which offer unique opportunities to establish comparisons of measurements and calculations (predictions) in the realistic environments which are simulated by the more simplified benchmarks. To be noted is the fact that the benchmarks are close to ideal configurations especially for calculations and for analysis; thus, it is necessary to assure that the measurement and calculation procedures which are developed from the benchmarks can be successfully applied in the non-ideal configurations

of real, operating reactors. Thus, irradiations of dosimetry measurement materials, and occasionally metallurgical materials as possible, are placed in such diverse reactors as Maine Yankee, H. B. Robinson-2, BR-3, Point Beach 2, Arkansas 1 and 2, Crystal River 3, McGuire 1, Brown's Ferry 3, Tihange 1, University of Buffalo Research Reactor, ORR and BSR, as well as the Venus and Nestor reactor noted above.

For the irradiations, dosimetry material packages are prepared that will allow the detection and counting of all neutrons in all the important energy ranges of interest to dosimetry. The materials include, for radiometrics, Fe, Ni, Co, and the fission materials, ^{238}U, ^{237}Np, etc.; solid state track recorders; stable monitors such as Helium Accumulation Fluence Monitors; Damage Monitors such as sapphire. Furthermore, in certain instances, direct neutron and gamma counting and time of flight measurements are carried out using specialized equipment. Following irradiation, precision counting and analysis of all the foils is conducted.

A crucial part of the program connected directly with the dosimetry materials irradiations is the calculation of the neutron and gamma fields before irradiation to help size the foils and determine exposure times, followed by reconciliation of the calculations with measurements. Transport theory methods are typically used, but other methods such as Monte Carlo are also used at times by the program and by other participants.

The capability to analyze dosimetry counting data to yield accurate fluence and spectrum information from program irradiations has a spin off in that the same can be done for surveillance irradiations conducted in reactors and the original data analyzed by test laboratories. Thus, the program has reevaluated the dosimetry data on all surveillance irradiations conducted in US power reactors to date and provided it for update of Reg. Guide 1.99.

2.4.3 Embrittlement-Damage Analysis

This is a small part of the program because materials irradiations are typically done elsewhere in the RES program. The results of other such irradiations are reviewed and analyzed, but some few critical experiments are also conducted to look at details which cannot be covered in the large irradiations of massive specimens. Studied here are the effects upon radiation embrittlement of specific chemical elements for adjustment of the coefficients on specific elements in the equations relating embrittlement and fluence in Reg. Guide 1.99. (Particular emphasis has been placed on copper, nickel, and phosphorous taken singly and in combinations contained within small volume melts of steel.) Evaluation of the metallurgical specimens from the PSF experiment is conducted under this topic, as well as the study of hardness, and microstructure at the TEM level for specimens from selected materials and irradiations. Emphasis is on damage correlation to test current embrittlement prediction methodologies.

2.4.4 Standards

The result of all the experimental and calculation studies and analysis is the basis for validation of a set of standards as shown in the matrix in Figure 1. A schedule showing the progress of approval of these standards is shown in Figure 2.

2.5 Environmentally Assisted Fatigue and Fatigue Crack Growth

This major subelement of the integrated research program for nuclear reactor pressure vessels is directed at the evaluation of environmentally affected fatigue crack initiation and environmentally assisted crack growth for a variety of materials and service conditions. The technical progress in environmentally assisted crack growth has increased enormously in the past few years, due largely to (1) an increased data base, (2) the subsequent definition of critical variables and their degree of influence, and (3) the formation of tentative models for the micromechanistic process of environmentally assisted crack growth. In view of the progress which has been made on these fundamental studies research is now realistically beginning on the application of these and future basic results to reactor-typical components, including the effects of "real" defect geometry, service loadings, stress concentration and gradients, environments and materials constituent elements. The research program in this area contains elements addressing various aspects of crack initiation and propagation in PWR typical environments. The program provides for the continuation of the existing fundamental studies and for expansion into new areas of initiation and environmentally assisted elastic-plastic fracture. This fundamental work is coupled with research that will lend itself directly to modification and/or extension of appropriate chapters, paragraphs and appendices of Sections III and XI of the ASME Code.

The major portion of this work will be carried out by MEA with input from Westinghouse and assistance by other consultants.

2.5.1 Development of S-N Curves for Nuclear Grade Steels in PWR Environments

General Electric, under sponsorship of the EPRI, has completed a 3-year program entitled "BWR Environmental Cracking Margins for Carbon Steel Piping." One significant element of this program dealt with the investigation of the fatigue behavior of carbon steel components in high-temperature, BWR environments. In that work, both smooth and notched specimens of SA-333-Gr 6 and SA-106-Gr B material were tested in air and with two levels of relatively high (0.2 ppm and 8.0 ppm) oxygenated water at primarily 550°F. Loading conditions included fully reversed as well as zero-to-tension cases, and the data were evaluated using conventional code procedures. The data in air at 550°F showed a factor of 2 (on cycles) decrease in failure lifetime as compared to the Code mean data curve for testing at room temperature in air. In oxygenated water environments, a further reduction in fatigue lifetime was observed. In general, high amplitude/ low cyclic frequency loading results in the most pronounced environmental effects.

ASTM STANDARDS FOR SURVEILLANCE OF LWR NUCLEAR REACTOR PRESSURE VESSELS AND SUPPORT STRUCTURES.

FIGURE 1. ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels and Support Structures.

FIGURE 2. Preparation, Validation, and Calibration Schedule for LWR-PV and Support Structure Surveillance Standards.

The NRC program is to carry out a similar study for a select number of piping and pressure vessel steels and welds under simulated PWR environments. Particular emphasis is placed on specimen surface finish to the degree that the specimens tested will have code allowable surface finish and contain code allowable surface defects. A sufficient number of specimens will be tested at different strain amplitude levels and different frequencies to establish an S-N fatigue curve that minimizes the uncertainties that arise from expected data scatter. The end objective of this work is to develop an S-N fatigue curve for steels and weldments in PWR environments. Inherent in this study, and using the previously developed GE data, is a determination of the differences, if any, in the developed S-N curves for BWR and PWR environments.

2.5.2 Continuation of Environmentally Assisted Fatigue Crack Growth Data Development

These tests have formed the basis of the subcritical crack growth program of the NRC for the last decade. Results from this program, together with other work carried out by Westinghouse and General Electric formed the basis for the water line curves now incorporated in Section XI of the ASME B&PV Code.

The scope of this work involves testing both unirradiated and irradiated A533B and A508 material, 1 and 2 TCT specimens, both base material, weld metal and HAZ. This is the completion of an ongoing data base development program which is providing crack growth properties, particularly da/dN vs WK curves, for inclusion in the data base support of the environmentally assisted fatigue curves for Section XI.

2.5.3 The Effect of "Real" Crack Geometry and Constraints

The vast preponderance of environmentally assisted fatigue crack growth data has been developed using specimens of the compact tensile type. Such specimens are characterized by relatively large crack dimensions, straight crack fronts, constant K_Is along the crack front and a condition where the corrodent surrounds and wets all surfaces of the specimen. This is markedly different from those conditions which prevail in the case of "real" crack geometries in cylindrical bodies. For this latter condition initial crack dimensions are usually small, the crack front is not straight, K_I is not constant along the crack front, and the corrodent washes only the surface of the body that contains the crack. Presently, environmentally assisted fatigue crack growth assessments of defects found in real structures are carried out using the compact tensile specimen derived data.

The specific objectives of this subtask are to 1) develop a test procedure or test procedures that will allow the determination of environmentally assisted fatigue crack growth of "real" geometry (semi-elliptical or half-penny) cracks, under 3D constraints and PWR environments; 2) develop a matrix of tests that will generate sufficient data to allow a thorough evaluation of the applicability of existing compact tensile specimen derived data to the analysis of "real" defects in PWR components; 3) carry out the testing; and 4) carry out the called for evaluation.

The program consists of two steps. Carrying out environmentally assisted fatigue crack experiments on PTC panels in the autoclave simulating PWR environments, followed by final verification tests of cylinders containing "real" semielliptical or half-penny cracks on their inner surface, where the interior of the cylinders are subject to PWR chemistries, 550°F temperatures and various pressure conditions leading to effective K's. The data developed from these two portions of the program will be compared to similar (that is da/dn vs σ K curves) developed from compact tensile tests. Based on this comparison the utility of the present data for application to pressure vessel and piping conditions will be established.

2.5.4 Effects of Cladding

Many components which comprise the pressure retaining boundary of PWR primary systems are fabricated from stainless steel clad carbon steel. The principal examples of this are the reactor pressure vessels. No environmentally assisted fatigue crack growth can occur in the carbon steel substrate until a crack penetrates through the cladding and exposes the substrate to the water environment. This program piggy-backs the effort defined in program element 2.5.3 defined above. It is limited to testing clad PTC specimens in the PWR environment simulated in the autoclaves. These series of experiments when coupled with the results of the unclad PTC specimens should give a realistic appraisal of the effects of the cladding.

2.5.5 Development of a Micromechanism Model

In recent years, several different micromechanism models have been proposed which attempt to characterize environmentally assisted fatigue crack growth from the viewpoints of the steel's constitutive chemical elements, the water chemistry, the crack tip strain rate, crack tip blunting, corrosion product build up in the crack, and crack closure phenomena. A generally validated micromechanism model is needed to allow extrapolation of existing data to other materials and loading conditions. Without such a model experimentally derived data must be relied upon and such data are time consuming and costly to generate.

The specific scope of this subtask is as follows: 1) document, review, and evaluate all existing proposed micromechanism models for environmentally assisted fatigue crack growth; 2) based upon the evaluation conducted in 1) either select one that is judged to be correct or depending upon their domains of applicability, select more than one; 3) develop a matrix of tests to experimentally validate the model(s); and 4) carry out these tests and complete the evaluation.

2.5.6 Development of a Cumulative Damage or Usage Factor

Almost all environmentally assisted fatigue crack growth data has been generated under constant load amplitude or constant K_I values. These data have led to the presently defined air and water lines for the fatigue crack growth curves in Section XI for the ASME B&PV Code. Since these data are presented as relationships between da/dN and σ K it should be theoretically possible to predict the growth of a crack when subject to a spectrum of loads. Section XI does not present a method for analyzing fatigue crack

growth under a variable loading spectrum where load interactions are considered. Rather, a linear damage accumulation scheme is suggested. The linear damage accumulation scheme simply sums increments of crack growth on a cycle by cycle basis. Such a procedure is suspect particularly in environmentally assisted crack growth. Effects such as crack closure caused by overload cycles may be particularly important and should not be neglected. The development of a cumulative damage or usage factor for environmentally assisted fatigue crack growth is required.

The scope of this subtask is to develop and carry out a test matrix of compact tensile type specimens for testing under variable amplitude loading conditions in PWR environments. The material selection for the specimens has been limited to two, one pressure vessel steel and one piping steel. The number of tests being performed is sufficient to indicate trends and allow the evaluation of any proposed cumulative damage rule.

2.6 Annealing of Reactor Pressure Vessels

The primary objective of this program element is to develop the proper background information to establish licensing criteria for the development of standards to be applied to proposed in-situ annealing procedures for commercial nuclear reactor vessels, and to identify those technical areas that need to be resolved before such criteria can be established. Studies of the parameters that enter into the annealing process, temperature, time and material constitutive elements have been underway for a considerable time and are still being studied by MEA under NRC sponsorship and by Westinghouse under the sponsorship of EPRI. In 1982 a program was initiated at EG&G/Idaho Falls with the following tasks:

o Review past and current research dealing with the parametric recovery effects of annealing, especially with regard to actual fracture toughness recovery in high copper content weld metals.

o Review past and current programs dealing with system aspects of in place thermal annealing of reactor pressure vessels, particularly the EPRI/ Westinghouse Study.

O Identify aspects of proposed in-place annealing procedures for which there are no or insufficient technical bases.

o Identify system aspects of proposed inplace annealing procedures which are generic and those which are clearly plant specific in nature; identify aspects which would be governed by current ASME Boiler and Pressure Vessel Code and/or NRC specifications and those for which no specifications or standards exists.

o Develop criteria to be met by any proposed inplace annealing procedure to ensure the continued safe operation of the reactor pressure vessel and the remainder of plant components.

o Identify areas of research needed to allow full development of annealing criteria.

o Continue active involvement with ASME and ASTM task groups dealing with thermal annealing considerations.

MEA is conducting a rather extensive program to clarify most of the remaining issues in doubt on parametric effects in annealing. These deal with the chemical composition effects on annealing, efficacy of high temperature annealing and reembrittlement path for irradiated post anneal steels.

2.6.1 Composition Effects on Annealing

Studies of heat treatment for the mitigation of radiation induced embrittlement are finding increasing evidence of an effect of steel (or weld) composition on annealing recovery. The large scatter observed in percentage recovery with 750°F annealing is an indication of compositional dependency of some type. This program is intended to clarify this issue. This investigation is being performed with 8 plates from laboratory melts having statistical variations in copper, nickel and phosphorous contents. Charpy-V specimens will be irradiated in two experiments at 550°F to 2×10^{19} n/cm^2 (E > 1 MeV) in the NSTF reactor. Specimen tests will be made to establish notch ductility properties for the as-irradiated and annealed conditions. Annealing will be carried out at 750°F for 168 hours. Compositional influences will be determined from comparisons of transition temperature recovery at the 41J energy level and residual transition temperature shift. The effects of composition will also be evaluated from comparisons of percent upper shelf energy recovery and residual upper shelf energy loss. Out-of-reactor aging tests at 550°F followed by annealing at 750°F for 168 hours will be performed for comparison with the irradiated /annealed data.

2.6.2 Irradiation Reembrittlement Path

The NRC's irradiation(I) anneal(A) reirradiation(R) program, now in its second phase, represents a systematic effort to qualify the annealing method for the control of radiation induced embrittlement to commercial vessel materials. IAR behavior is being assessed, both from the standpoint of transition temperature elevation and from the standpoint of upper shelf toughness reduction. From the completed Phase 1 study two generalized observations were made: (1) there is a greater rate of embrittlement in heat treated (annealed) material compared to non-annealed material, and (2) a significant difference between upper shelf energy and transition temperature shift recovery occurs as a response to a post-irradiation anneal. The experimental phase of IAR Phase 2 investigations will be completed by the construction and irradiation of one, three capsule specimen assembly (Capsules A, B, & C) in the NSTF reactor. Each assembly will contain C_v specimens of two welds in approximately equal numbers, and two tensile specimens (one for each weld). Capsule C will be irradiated at 550°F to 2.5×10^{19} n/cm^2 (E 1 MeV). After irradiation, a portion of the Capsule C specimens will be annealed at 750°F for 168 hours. Capsules A and B will be irradiated at 550°F to 2.0×10^{19} n/cm^2 , annealed at 750°F for 168 hours, then reirradiated to 550°F. The reirradiation fluence for Capsule A will be 0.25×10^{19} n/cm^2 and for Capsule B will be 0.5×10^{19} n/cm^2. C_v tests will employ an instrumented impact tester and will establish I and IA condition properties for Capsule C and IAR conditions for Capsules A and B. These data, together with prior Phase 2 results will be analyzed to qualify the reembrittlement path and its dependence on first cycle fluence level and the flux type used in the weld preparation. In addition the correlation of C_v and CT test observations will be examined for the I condition at 1.0×10^{19} and 1.5×10^{19} n/cm^2 and for the IAR condition at 1.5×10^{19} n/cm^2.

2.6.3 High Temperature Annealing

Prior annealing studies conducted for the NRC evaluated the efficacy of post-irradiation annealing temperatures of 650°F and 750°F. A recently completed study carried out by Westinghouse under the sponsorship of EPRI indicated that annealing temperatures of 850°F for 168 hours showed very high percent recovery (80 to 100%) in all lost toughness properties. Of greater interest, they reported that reembrittlement after such an high anneal temperature took place at a slower rate than the embrittlement of the virgin material. This is in direct contrast to the results of all NRC (and others) sponsored research for annealing temperatures up to 750°F for 168 hours. The number of specimens tested and the variation in chemical composition of those specimens was very limited in the Westinghouse study. Exploratory studies carried out by MEA suggest that responses to 850°F anneal is dependent, in part, on material composition. It is necessary to confirm or to further investigate this 850°F anneal phenomenon.

In this study four different copper content submerged arc weld materials fabricated with Linde 80 and Linde 0091 welding fluxes will be fabricated. The welds are to depict two levels (high/low) of nickel content. Standard C_v and tensile test specimens of materials are to be irradiated at 550°F to 1.2×10^{19} n/cm^2 (E> 1 MeV), annealed to 850°F for 168 hours and reirradiated at 550°F to 0.3×10^{19} and 0.7×10^{19} n/cm^2, using the NSTF reactor. Notch ductility and tensile properties will be established for each I, IA, and IAR condition. The testing of these specimens will render data allowing the determination of the interrelationship of: annealing temperature vs recovery vs reembrittlement rate vs nickel content vs weld flux type. Rates of embrittlement are to be compared to those generated from virgin material at two fluence levels. The C_v 41J temperature would be used for comparing transition behaviors; upper shelf energy comparisons will be conducted at 440°F.

2.6.4 System Engineering Problems and Feasibility of Annealing

An industry survey and review of possible candidates for an irradiated vessel annealing demonstration were completed at the end of calendar year 1982. The results from the EPRI/Westinghouse program were reviewed, and discussions were conducted with nuclear personnel worldwide concerned with potential annealing of reactor vessels. The results from this task indicate that annealing of an in-place reactor vessel is feasible, but solvable engineering problems do exist. None of the potentially available irradiated pressure vessels (Indian Point-1, Shippingport, and Humboldt Bay in the United States, KRB-A, in West Germany, and BR-3 in Belgium) are similar enough in material, size, geometry, vessel support, and radiation embrittlement to the large, commercial pressurized water reactors of concern to warrant an irradiated vessel demonstration effort. In addition, an irradiated vessel annealing demonstration could cost \$30 - \$50 million. Thus, such a demonstration is not cost effective. If a nuclear vessel more typical of those currently under review for the pressurized thermal shock problem became available, an annealing demonstration using this plant would have significantly greater applicability. However, the material response, logistics, and engineering aspects of annealing are highly plant specific, and no guarantee of success or failure for annealing can be implied from such an annealing demonstration.

Work has been initiated to define the usefulness and cost of performing an anneal on a system mockup using an unirradiated vessel.

The EPRI/Westinghouse study presented a heat treating approach for dry annealing of a typical Westinghouse vessel. An EG&G Idaho subcontractor, Cooperheat, one of the world's largest heat treaters, has developed a state-of-the-art review of large vessel heat treating at temperatures much higher than 850°F; Cooperheat has also prepared alternative heat treating procedures for a typical commercial pressurized water reactor vessel. The development of these procedures has required further consideration of health physics and system engineering problems. One critical area of concern being addressed by Cooperheat is temperature measurement and control during the annealing cycle.

One nontrivial problem with a localized heat treatment is the degree of distortion that may occur after the annealing cycle. The extent of residual stresses is also an important consideration. The EPRI/ Westinghouse program only superficially addressed this aspect of annealing. An EG&G contractor, Combustion Engineering, Inc., performed a thermal and structural analysis of a reactor vessel specifically evaluating distortions and residual stresses (taking into account primary creep deformation).

The two subcontract efforts (Cooperheat and Combustion Engineering, Inc.) were completed early in FY-1984. A report was written summarizing the state of knowledge with respect to annealing recommending further research needs. The successful wet annealing program being undertaken in Belgium on the BR-3 reactor vessel (at a lower temperature of 650°F) was closely monitored. This annealing project is the first since the SM-1A anneal.

2.6.4.1 Criteria for Annealing

An American Society for Testing and Materials task group is in the process of upgrading and revising guide ASTM E509-74, entitled "In-Service Annealing of Water-Cooled Nuclear Reactor Vessels." Emphasis in this effort is on the material properties and surveillance aspects of annealing rather than system engineering problems. Preliminary drafts of the revised ASTM Standard Guide were written within the task group during FY-1983. The system safety issues would most likely be covered by American Society of Mechanical Engineers Boiler and Pressure Vessel Code subcommittees. Currently, the ASME Section XI Subcommittee is establishing a Subgroup on Requalification which would have in its charter the issue of pressure vessel annealing. Other subgroups within Section XI should contribute to any Code requirements which may be developed. EG&G Idaho involvement within ASTM and ASME will continue. Involvement within ASTM and ASME will continue, and a final revision to ASTM E509 should be completed within FY-1986.

2.6.4.2 Engineering Demonstration to Validate Methodology

In cooperation with industry the methodology (engineering design and instrumentation) will be developed for a full scale vessel anneal experiment, a full scale vessel anneal will be engineered and performed and appropriate tests performed to demonstrate the engineering aspects of in-situ annealing.

3.0 Schedule

A network showing major milestones and tasks to be completed by fiscal year is enclosed in FIGURE 3. Milestone charts for specific program elements can be provided upon request.

4.0 Coordination

Coordination within NRC to ensure that research is directed towards and produces products for regulatory needs is ensured by close one-on-one ties between staff members of MEBR, and MTEB and GIB of DST, both of NRR. It is further augmented by periodic reviews of all reactor vessel programs by the Vessel Integrity Review Group (VIRG), and by reviews by the ACRS.

Coordination of reactor vessel research with organizations outside the NRC is achieved in several ways. An augmented VIRG, with members from the U.S. technical community, NSSS vendors, U.S. nuclear utilities and representatives from European and Japanese laboratories and governments meet periodically to review and comment on the efficacy and applicability of the program. Each year the program is presented for national and international visitors to the annual NRC Water Reactor Safety Research Information Meeting. Personal contact by MEBR personnel is maintained with their counterparts of EPRI with each group attending the program reviews of the other. Extensive coordination and cooperation is achieved with European research groups through formal cooperative research agreements established between the NRC and European governmental agencies. This is enhanced by onsite visits to the European laboratories by MEBR personnel each year and by foreign representative visits to U.S. laboratories conducting research for the NRC. A prime example of this is the activities centered about the HSST program. Finally, extensive participation by both MEBR personnel and contractor personnel in professional society activities such as ASTM and ASME and International groups such as the CSNI, the IAEA and the International Cyclic Crack Growth Rate Group (ICCGR), ensure that the work being done is timely, effective, and applicable to nuclear safety issues.

5.0 Capabilities of NRC Staff

MEBR staff members acting as managers for the reactor vessel research programs are Mr. Milton Vagins and Mr. Alfred Taboada.

Mr. Vagins received his B.A. and B.M.E. degrees in Mechanical Engineering (1952 and 1960, respectively) from the City College of New York. During the three years (1963-1966) that Mr. Vagins taught courses in the Engineering Mechanics Department of the Ohio State University he completed all requirements for and entered into candidacy for the Ph.D. degree in Engineering Mechanics. Mr. Vagins was employed by the Battelle Memorial Institute, both at the Columbus and Richland laboratories where he was sequentially, research scientist, program manager and line manager from 1960 to 1979. He joined the Materials Engineering Branch, RES in April of 1979 and has had the responsibility for the direction of the HSST program and other fracture mechanics related programs. He was instrumental in bringing the P.T.S problem to the attention of the NRC and played a key role in the timely achievement of the interim resolution to USI-A49. He

acts as the resident expert in fracture mechanics for RES. He currently directs the HSST program, the Structural Integrity of Light Water Reactor Boundary Components program, the Degraded Piping Program, Piping related research programs at the DTNSRDC, and the Naval Academy, and controls the activities of MEBR's Resident Engineer in Germany.

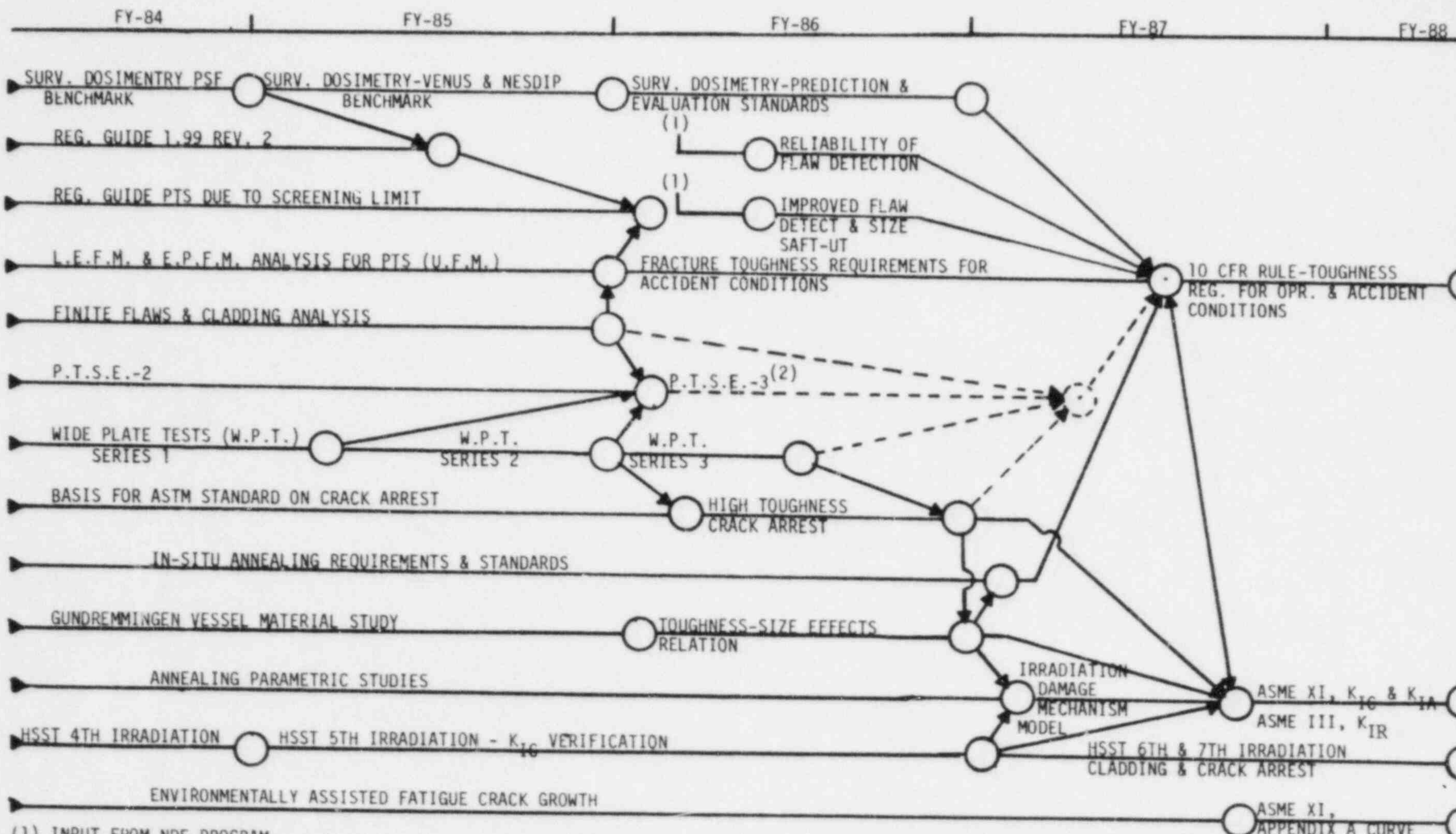
Mr. Taboada received his B. Chem. Eng. degree from Newark College of Engineering and his M. Met. Eng. degree from Renssalaer Polytechnic Institute. He has worked in the NRC for the past 9 years as a Program Manager, Standards Development staff, and Operating Reactor Reviewer, and in the A.E.C. for 9 years as a Senior Materials Engineer, managing engineering materials development work for advanced reactors. Prior to that he worked at ORNL for 12 years as a Group Leader and Project Metallurgist in the design, development, fabrication, and NDE of the following reactor programs: ANP, MSRP, HFIR, HRP, and SNAP-8. He is a member of the ASME B&PV Code Committee, Subgroup on Materials and ASTM-E10 Executive Committee, and has participated in the NRC Pipe Crack Study Groups and the NRC Piping Review Committee. He is currently managing the SCC of PWR Steam Generator Tubing Program at BNL, the Materials Engineering Program at ORNL, the RV Thermal Annealing Program at INEL and the Dosimetry Programs at HEDL, NBS and ORNL.

6.0 Closure of Technical Issue

The most important area in reactor vessels that will be closed in the FY 1986-87 time frame from a research viewpoint is that of pressurized thermal shock. The licensing position, in the form of the Screening Criterion, was established in FY 1983 with considerable assistance and background from the research program; tests and experimental results will have continued to accrue subsequently that will validate the licensing position taken earlier. A corollary issue will also be closed in the FY 1987-88 time frame, that of establishment and validation of the Post-Screening Limit Criterion. Here will be set forth the requirements for actions once a utility's vessel actually reaches the screening criterion. Another aspect of the PTS evaluation issue is the need for validated elastic-plastic fracture mechanics analysis procedures for vessels; these procedures will be in place by this time. As irradiation embrittlement has been a continuing critical issue to reactor vessels, the research program should result in closure of several items in this area as well. First, verification will be achieved for the shape and radiation-induced shift in the K_{IC} initiation curve included in ASME Code Section XI for base and weld metal. This will greatly aid the confidence of the staff in accepting evaluations of the toughness of reactor vessels based on surveillance capsule test results. Next, adequate results should be available to set out the criteria for sensitivity of different steels to embrittlement, i.e., why one steel embrittles more than another, and what is the mechanism for radiation embrittlement. Such information is necessary for determining predicted rates of embrittlement in the future, and thus, whether or not a vessel will have adequate toughness during its lifetime. Finally, these predictions of lifetime embrittlement require accurate neutron dosimetry procedures; these will also be completed in the FY 1986-87 time frame. Among the facets of this issue is the effect of the dose rate, whereby predictions are typically

made based on surveillance or test reactor irradiations both of which are accelerated in time. However, the embrittlement of a vessel accrues at a much slower rate, thus raising the issue. To settle this issue, several experi- ments will be complete, as well as direct measurements of embrittlement from decommissioned power reactor vessels. Thus, to be complete in the FY 1987-88 period will be the issue of correlation of power reactor vessel properties to experimental test data.

FIGURE 3 REACTOR VESSEL RESEARCH PLAN



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14 ABSTRACT (200 words or less)

This document presents a plan for research in Reactor Vessels to be performed by the Materials Engineering Branch, MEBR, Division of Engineering Technology, (DET), Office of Nuclear Regulatory Research. It is one of four plans describing the ongoing research in the corresponding areas of MEBR activity, which are being published simultaneously in four volumes as follows: Vol. 1 Reactor Vessels, Vol. 2 Steam Generators, Vol. 3 Piping, and Vol. 4 Non-Destructive Examination. These plans have been updated and are more detailed expansions of those originally published as part of the Long Range Research Plan for the Office of Nuclear Regulatory Research in NUREG-1080 Vol. 1.

15a KEY WORDS AND DOCUMENT ANALYSIS

Nuclear Reactor Pressure Vessels
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Neutron Dosimetry

15b DESCRIPTORS

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