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TASK ACTION PLAN

PRESSURIZED THERMAL SHOCK (TASK A-49)

Lead Organization:	Division of Safety Technology (DST)
Task Manager:	Roy Woods, Generic Issues Branch (GIB)
Lead Supervisor:	K. Kniel, Chief, GIB, DST
NRR Principal Reviewers:	B. Sheron, E. Throm, P. O'Reilly, A. Oxterth, L. Lois, R. Johnson, R. Klecker, N. Randall, W. Hazelton, G. Vissing, J. Clifford, G. Schwenck
RES Principal Reviewers:	C. Johnson, M. Vagins, P. Baranowsky, J. Strosnider, C. Serpan
Applicability:	Pressurized Water Reactors
Projected Completion Date:	May 1983

DESIGNATED ORIGINAL

Certified By Judy Botts

1. INTRODUCTION AND BACKGROUND

As a result of operating experience, it is now recognized that transients can occur in pressurized water reactors (PWRs) characterized by severe overcooling causing thermal shock to the vessel, concurrent with or followed by repressurization (that is, pressurized thermal shock, PTS). In these PTS transients, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall. This temperature distribution results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature differences across the reactor vessel wall. Effects of this thermal stress are compounded by pressure stresses if the vessel is repressurized.

Severe reactor system overcooling events which could be followed by repressurization of the reactor vessel (PTS events) can result from a variety of causes. These include instrumentation and control system malfunctions, and postulated accidents such as small break loss-of-coolant accidents (LOCAs), main steamline breaks (MSLBs), feedwater pipe breaks, or stuck open valves in either the primary or secondary system.

As long as the fracture resistance of the reactor vessel material remains relatively high, such events are not expected to cause failure. After the fracture toughness of the vessel is reduced by neutron irradiation (and this occurs at a faster rate in vessels fabricated of materials which are relatively sensitive to neutron irradiation damage), severe PTS events could cause crack propagation of fairly small flaws that are conservatively postulated to exist near the inner surface. The assumed initial flaw might initiate and propagate into a crack through the vessel wall of sufficient extent to threaten vessel integrity and therefore core cooling capability.

The Rancho Seco event of March 20, 1978 is believed to represent the most severe (and prolonged) overcooling transient experienced to date. Although the event was considerably less severe than would have been necessary to cause potential failure of the Rancho Seco vessel at the time the event occurred (because of the existing fracture toughness of the vessel), the event nevertheless represents an important precursor for such severe events. That is, had subsequent failures or inappropriate operator actions or lack of proper operator actions occurred, the precursor that did occur could have developed into a more severe (but less probable) PTS event. Similarly, had the Rancho Seco event occurred with a more highly irradiated vessel, vessel integrity could have been jeopardized without the occurrence of additional failures or errors. In the Rancho Seco event, a lightbulb being replaced in the non-nuclear instrumentation/integrated control system (NNI/ICS) panel was dropped and caused a short to occur while the plant was at approximately 70% power. About 2/3 of the instruments that indicate pressure, temperature and level were lost. Furthermore, the operator did not have confidence in the validity of indication or the remaining instrumentation. The reactor tripped, feedwater was lost, the auxiliary feedwater (AFW) pumps started but remained isolated due to the ICS failure, and the once-through steam generators dried out. Subsequent refilling by the AFW and possibly by the main feedwater (MFW) systems caused primary system overcooling and actuation of high pressure injection (HPI) and opening of the AFW isolation valves. Actuation of HPI and MFW caused severe overcooling rates (approximately 300°F/hr) until some of the pumps were shut off by plant operators. Actuation of HPI also caused repressurization of the primary system. Operators did not have what appeared to be a reliable temperature indication, and thus kept AFW and HPI on to maintain core cooling while restoring NNI. During this time, primary system temperature had been reduced to about 285°F.

Since the March 1979 accident at Three Mile Island (TMI), much emphasis has been placed upon the need to run cooling pumps until it is positively determined that they can be turned off without the possibility of core overheating. Such training contributes to the severity of PTS events, however, and may be a factor in making future events of this type even more likely and/or more severe (the Rancho Seco event occurred before TMI).

In view of the above, the program described in this Task Action Plan (TAP) is needed to formulate a regulatory requirement to ensure that the risk of pressure vessel failure from PTS events is sufficiently low through each vessel's design end-of-life. The program that will be conducted to provide firm bases for such a regulatory requirement includes: development of methods for estimating the probability and severity of PTS transients and the operator's role in such events, refinement of methods for determining pressure vessel stresses in the event PTS transients do occur; refinement of methods for determining material properties and failure vulnerability of the vessel due to PTS stresses as a function of vessel exposure to neutron irradiation (and thus as a function of time in plant life); evaluation of potential benefits from potential corrective actions; and development of criteria for acceptability of plant safety margins under postulated PTS events. This program will provide a benchmark to aid NRC in assessing acceptability of several PTS studies currently underway in the industry, as well as forming a basis for recommending acceptance criteria for resolution of the PTS issue.

As stated in Section 3, (Basis for Continued Plant Operation and Licensing Pending Completion), up until the present time we have used a generic method for predicting vessel properties versus irradiation time and have concluded that no event having a significant probability of occurrence could cause any pressure vessel to fail today or in the next few years. However, using those generic methods (which are believed to be conservative) we predict the necessity for some type of corrective action before design end-of-life for several vessels. The results of this program are needed to provide more detailed and realistic (but still conservative) analyses of systems responses, material properties, and risks before decisions are required regarding the nature and timing of the corrective actions.

Potential corrective actions are discussed in Section 2.B.(7) below. They include ways to delay vessel embrittlement by reducing neutron fluence at the critical locations, ways to decrease the probability of PTS events with better control systems and/or operator actions, a way to lessen the consequences of PTS events if they do occur (such as warmer injection water), and a way to improve vessel properties (in-place annealing).

The magnitude of the problem described in this TAP with pressurized thermal shock was not appreciated during the design stage of currently operating PWRs, although pressure vessel thermal shock had been considered for many years in the context of assuring integrity of the vessel when subjected to cold emergency core cooling water during a large loss-of-coolant accident. Based on a series of thermal shock experiments (unpressurized) conducted at Oak Ridge National Laboratory (ORNL) beginning in 1976 which verified the associated fracture mechanics analyses, it was concluded that a postulated flaw would not propagate through the vessel wall during a large LOCA. Therefore, the vessel's ability to contain water would be maintained during subsequent reflooding which would occur at relatively low pressure due to presence of the large break. However, the possibility of concurrent or subsequent high pressure can negate the above conclusion and will be evaluated in the program described in this TAP.

It should be pointed out that the NRC staff does not believe boiling water reactors (BWRs) have a significant PTS concern, for several reasons. Most importantly, BWRs operate with a large portion of the water inventory inside the pressure vessel at saturated conditions (that is, it exists as a mixture of steam and liquid water at the mixture's boiling temperature and pressure). Any sudden cooling will condense steam and result in a pressure decrease, so simultaneous creation of high pressure and low temperature (necessary to cause a PTS concern) is very improbable. BWR operating experience provides verification that PTS events are very improbable. Although there have been numerous overcooling events, there have been no significant PTS events at any domestic or foreign BWR. Also contributing to the lack of PTS concerns for BWRs is the lower fluence at the vessel inner wall, since BWRs have more water between the core and the vessel wall due to the recirculation flow path (water shields the vessel from the core). Finally, the operating pressure of BWRs is lower, which allows the use of a thinner vessel wall which results in a somewhat lower stress intensity for a postulated crack.

2. PLAN FOR PROBLEM RESOLUTION

A. General Approach to the Problem

An outline of the proposed integrated program to be conducted by the Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES) utilizing the National Laboratories, with input from industry including the PWR owners groups and eight selected utilities is shown in Figure 1. Throughout the program, NRC will obtain and utilize the advice of consultants who are competent in the various technical disciplines relevant to this program, including certain input from the Electric Power Research Institute (EPRI) concerning thermal mixing. Additionally, NRC will work closely with the Advisory Committee for Reactor Safeguards and its consultants.

All work performed through RES and at the National Laboratories will be utilized for input to the NRR licensing decision process, for use as appropriate (and if applicable). It is not the intent that NRR "censor" or overly restrict the course of the research programs. Nor is it the intent that the conclusion of the research projects will be wholly incorporated into licensing requirements without modification. NRR is responsible for developing licensing requirements, and will use the RES and National Laboratory results only as input to the licensing process.

The NRC program consists of the following major sub-tasks. The first two tasks, designated as (a) and (b), are considered to be part of the short-term NRC program to be completed by about June 1982 and are not discussed at length in this TAP which covers the long-term program.

Short-Term Program - Review of Industry Responses

(a) Review of information requested by August 21, 1981 letter to industry groups and eight selected utilities. This will provide a reassessment of the PTS issue by about June 1982. The reassessment will conclude whether or not there appears to be a short-term (within approximately two years) significant problem at any operating plant and will recommend any corrective actions found to be necessary before completion of the program outlined in this TAP. Knowledge gained in these reviews will be utilized to guide the overall NRC program (that is to emphasize work in the areas with the greatest uncertainty). Details of this review can be found, for example, in TAC #47548 for H. B. Robinson, plus sequential TACS for the other seven plants involved.

(b) Draft revision of the trend curves in Regulatory Guide 1.99, Revision 1. This revision will be drafted to reflect new surveillance data and the effects of nickel content on the predicted value of Charpy shift (that is, how irradiated material properties are determined for certain pressure vessel materials).

Long-Term Program - Independent Analysis of PTS

(1) Selection of PTS transients to be analyzed based on systems studies, human factors studies, and probabilistic and risk assessment analyses for three lead plants.

(2) Selection, model improvement and verification of transient codes for use in calculation of the selected transients.

(3) Calculation of the pressure vs. time and the temperature vs. time of the water in contact with critical welds or base metal in the pressure vessel for the selected PTS transients (using the selected and verified codes).

(4) Improvement and experimental verification of a state-of-the-art fracture mechanics code to predict stresses and therefore crack initiation, propagation, and arrest for given pressure-temperature histories at critical welds or base material, including consideration of warm prestress if demonstrated to be applicable. This will include input from near-term fracture mechanics experiments performed by the Heavy Section Steel Technology (HSST) group at ORNL.

(5) Calculation of failure potential vs. irradiation embrittlement (that is, neutron fluence from the operating history) of the pressure vessels at the three lead plants for the selected PTS event sequences using the pressure and temperature vs. time histories from item (3) as input to the item (4) codes. These analyses assume pre-existence of a range of crack sizes infintely long of various depths.

(6) Performance of sensitivity studies to determine changes in predicted vessel failure probability due to uncertainties in such parameters as copper content of the weld, initial crack size, lowest temperature of cooldown, etc.

(7) Development of an understanding regarding feasibility of and benefits to be derived from various proposed corrective actions, including revised fuel loading patterns to reduce fast neutron flux at the vessel wall, increased temperature of safety injection water, improved control and instrumentation systems and/or operator actions to prevent repressurization, and vessel annealing.

(8) Development and publication of a NUREG report recommending a Regulatory Position regarding PTS including appropriate limits (if any) that must be observed at specific classes of plants, and potential corrective actions.

Each of these items constitutes a major sub-task. Many of the sub-tasks are planned to proceed concurrently, but some must be sequential. The accompanying Figure 1 is provided to show an overview of the sub-tasks, including their relationship and schedule. More details of each sub-task are given in the discussion below.

B. Technical Content of Major Sub-Tasks

(a) Review of Requested Information

Full details of this item, which is part of the short-term review leading to a June 1982 reassessment of the PTS issue, can be found in TAC #47548 for H. B. Robinson and sequential TACs for the other seven plants involved. The item is summarized below.

NRR has requested plant-specific information from eight selected licensees regarding material properties, operator procedures, and systems interactions that can cause PTS events and the probability of such events. NRR will review this information (the "60-day" and "150-day" responses to the August 21, 1981 letters to the eight licensees) along with other (generic) input from the three PWR owner's groups (and EPRI) to provide a reassessment of the PTS issue to the Commission by about June 1982. The reassessment will conclude whether or not there exists a PTS problem at any plant significant enough to warrant immediate corrective action, and will recommend those corrective actions, if any, that must be initiated before completion of the program described in this TAP. Knowledge gained from these short-term reviews will be utilized as appropriate (for example, as a starting point) in the programs described in this TAP, and will guide the NRC program to emphasize the areas where the most uncertainty exists.

NRR has also initiated an effort through the Division of Human Factors Safety to improve operating procedures to lessen the probability of a severe PTS event. The near-term program will result in identification (by each licensee) of a recommended method or "pathway" to avoid both overcooling events (with concurrent or subsequent pressurization) and overheating events. Plant operating procedures will be put in place or revised as needed to facilitate the operator's task in maintaining plant safety, along with appropriate operator training in those procedures and their underlying technical basis. Generic guidelines for updated procedures will be completed by mid-1982. Plants that require immediate corrective action can have plant-specific procedures in place, and all training regarding those procedures complete by the end of 1982 if required to deal specifically with PTS events.

In addition, a task force has been formed to audit procedures that deal with potential PTS events, and to audit operator training regarding those procedures and regarding PTS phenomena. These audits will be completed for the eight selected plants by June 1982. A second task force has been formed to accelerate consideration of methods that could significantly reduce flux at the vessel wall.

(b) A revised Regulatory Guide 1.99 will be drafted. Based on preliminary analyses of the PWR surveillance data base, which was gathered as part of the thermal shock studies, it appears that the formulas for the trend curves for Charpy shift in Regulatory Guide 1.99 should have a new nickel-dependent term included. This will be done in the draft Regulatory Guide. The new term will sharply reduce the observed overprediction when Regulatory Guide 1.99, Revision 1 is applied to low nickel material such as A302B steel. For high nickel material, the new term will have little effect. In addition, the planned draft revision to Regulatory Guide 1.99, Revision 1 will update the data base and will put the trend curves on a statistical basis from which both mean curves and upper bound curves will be derived.

The remaining items discussed below are the long-term PTS program, the principal topic of this TAP.

(1) Determination of Event Sequences to be Considered

Three major sub-tasks are involved in selecting the transients to be considered.

(1-a) Preliminary Development and Quantification of Event Trees for Transients Which Could Result in Overcooling.

NRR is performing a preliminary probability study of PTS initiating events (precursors) including MSLB, large break loss-of-coolant accident (LBLOCA), small break loss-of-coolant accident (SBLOCA), core shutdown cooling by safety injection with flow out the pressurizer safety valves and no feedwater (such as, "feed and bleed" core cooling), and feedwater transients in which increased feedwater is supplied to the steam generators (SGs) combined with steam flow out of the SGs through open dump or relief valves. This study includes multiple failures and multiple operator errors. This study will be performed for three lead plants (one from each PWR Nuclear Steam Supply System vendor) selected as the optimum available combination of typicality (vessel materials and control systems) and worst irradiation embrittlement. This study will incorporate information obtained in the responses to the August 21, 1981 NRC letters sent to eight representative plants.

(1-b) Development and Quantification of Event Trees for PTS Events Including Review of Control and Safety Systems.

Results of item (1-a) will be input into a RES program with ORNL to Perform a study of detailed control and safety system design at the three lead plants. That contract is to provide details of control and safety system functions and failure modes that may lead to PTS event sequences. Owners of the three lead plants will provide to ORNL control, feedwater, and safety system functions pertinent to PTS event sequences. ORNL will define about twelve event sequences in sufficient detail to provide input to Los Alamos National Laboratory (LANL) and Idaho Nuclear Engineering Laboratory (INEL) calculations of reactor coolant pressure and temperature vs. time in the downcomer region. The event sequences specified will include consideration of multiple failures and multiple operator errors. Discussions will be held with licensees of the three lead plants as PTS studies progress, and areas of disagreement between ORNL, the NRC staff, and the licensee (for example, credit for operator action or control system performance and consideration of multiple failures) are to be indicated in the initial reports along with a justification of the final position.

(1-c) Human Factors Studies

An additional ORNL research project, managed by the Human Factors Branch of RES, will address required operator actions for the transients being considered and result in an assessment of the probability and the effect of human errors on the likelihood of occurrence and severity of overcooling transients. The NRC will develop human error probabilities from this information.

The above results will be jointly used by NRC and ORNL to determine which PTS events are the major risk contributors, and these events will be used in sub-tasks 3, 5, 6 and 8 below (refer to Figure 1). The results will also be used to review new procedures that will be adopted by PWRs to help prevent PTS events and to lessen the severity of those that do occur.

(2) Transient Model Development and Verification

Concurrent with sub-task 1, LANL and INEL will be developing and obtaining data to verify the TRAC and RELAP5, and SOLA codes which will be used to calculate $P(t)$ and $T(\overline{r},t)$ for the selected PTS events. The three codes need some model improvement and verification by comparison with data. Code improvements are needed for the pressurizer model, for thermal mixing in the cold leg and downcomer regions, and to model the secondary (steam-feedwater) system. Data on thermal mixing in the downcomer will be obtained from an ongoing EPRI program and will be used to verify the SOLA code. Brookhaven National Laboratory (BNL) will perform a QA function for the input decks and completed calculations.

(3) Calculation of $P(t)$ and $T(\overline{r},t)$

These calculations will be performed at LANL and INEL for the Transient event sequences identified in sub-task 1 using the improved codes developed and verified in sub-task 2.

(4) Improvements in Methods and Data for Fracture Mechanics Calculations

Several different types of experiments are being planned or are underway to provide data needed for methods improvement. These tests are planned as part of the HSST program at ORNL. The experiments are designed to improve our understanding of flaw initiation, propagation, and arrest so that fracture mechanics calculations will be more relevant to PTS conditions. Planned tests include a series designed to further our understanding of the warm-prestress phenomenon and the limits of its applicability. Ultimately it is hoped that the methods can be extended beyond the presently accepted linear elastic fracture mechanics methodology to include elastic-plastic fracture mechanics methods. In particular, these programs will focus on obtaining theoretical and emperical information on the effects of cladding and the potential benefits of warm prestressing. Consideration will also be given to crack propagation into material still on the upper shelf, thus integrating A-49 with A-11.

Currently underway are a set of tests with small flaws in several square-foot, 2 inch thick plates that are stressed by four point bending (that is, no thermal or pressure stress). These tests will involve through-clad cracks, under-clad cracks, degraded cladding, and no cladding. Later, irradiated samples will be used.

Also currently underway are a set of tests using cylinders approximately 3 feet in diameter and 4 feet long with various flaw geometrics which are tested using liquid nitrogen (but without pressure stress). Some of these cylinders will be clad on the cooled surface to determine cladding effects.

A pressurized thermal shock test is being planned which will be pressurized cylinder that will be thermally shocked to simulate both types of PTS stresses (thermal and pressure-induced).

Fracture mechanics codes (OCA-1 at ORNL and the NRC codes) will be further developed utilizing the above experimental results plus analytical work in the areas of: effect of cladding; treatment of through-clad cracks; treatment of warm prestress; three-dimensional effects; and size and shape of pre-existing cracks. More precise fluence/materials data and properties information will be obtained and developed for use as input to these calculations. Results of this sub-task will remove known conservatisms where possible in the fracture mechanics codes.

(5) Vessel Failure Analyses

Calculations will be performed using the methods and data from sub-task 4 and the $P(t)$ and $T(\overline{r},t)$ results from sub-task 3 for PTS events. This sub-task's results will include the occurrence probability of each PTS event from sub-task 1 and the consequences of each event (that is, crack initiation, propagation, arrest, or through-wall penetration) at various times in the vessel life. These results will be used to provide a prediction of reactor vessel failure as a function of effective full power years (EFPY) of operation for the PTS events. A range of crack depths are assumed to pre-exist for these calculations. Extension of any of those pre-existing cracks into a through-wall crack penetration will be assumed to produce vessel failure. Considering that sub-task 1 also produced an estimate of the frequency of each transient considered, the last output of this sub-task will be a "best" estimate (somewhat conservative) of vessel failure probability vs. effective full power years for the three (typical) lead plants. These results will be condisered by NRR and used as appropriate on one of the inputs into the licensing decision process.

(6) Sensitivity Studies

There are many uncertainties in the overall program (sub-tasks 1 through 5). The effect of those uncertainties on sub-task five's results will be evaluated. Examples are: initial crack size, fluence and/or material properties, copper and nickel content of the welds, temperature at the weld, cooling rate, and pressure. Sensitivity of the program results described above to credible variations in these parameters (individual or varying in multiple combinations simultaneously) must be assessed before a Regulatory Position can be determined. This will be done in two diverse ways:

(a) A series of $P(t)$, $T(\overline{r},t)$ and fracture mechanics calculcations for several combinations of different input parameters, will be performed to determine the effects of variations in the input on outputs of sub-task 5.

(b) NRC has developed a statistical, Monte Carlo-based computer code that will allow calculation of a response surface resulting from a statistical variation of many input parameters. A statistical result can be obtained giving the mean value of risk due to PTS events, and variance in that risk, with consideration for the uncertainties.

Results of both methods will be utilized to arrive at a determination of risk from PTS events at the representative three lead plants. Since representative plants were selected, the results can in principle be generalized to obtain an approximate value for risk at other PWRs. Extrapolation, approximation, or engineering judgment may have to be used for specific plants that differ significantly from the "typical" lead plants selected.

(7) Benefits/Practicality of Corrective Actions

Several potential corrective actions are possible, and will be considered. These include:

(a) Reducing the neutron flux at the pressure vessel. For example, some of the outermost fuel elements in the core could be replaced with partially loaded or reflector elements or a fuel management program adopted that places partially depleted fuel elements near the vessel.

(b) Annealing the reactor pressure vessel in-situ to restore some or all of the fracture toughness lost by neutron irradiation. Although annealing is feasible from a metallurgical standpoint, and studies made to date have not revealed any damaging side effects, it would be expensive and would require a long down time.

(c) Reducing the thermal shock during some transients by raising the temperature of the emergency core cooling system injection water.

(d) Reducing the probability of the event by new procedures, new control systems, new instrumentation systems or a combination of all three to prevent repressurization or give clearer indication to the operator that a situation is developing that has potential PTS concerns. These corrective actions would provide automatic actions or allow operator actions with a higher degree of reliability to prevent repressurization.

The programs described below will provide the information needed to assess the benefits to be derived from, and the practicality of, the various proposed corrective actions.

ORNL will provide consultation to the NRC staff in evaluating the effectiveness of the various corrective actions as part of their ongoing contract with NRC. In addition, BNL will evaluate effectiveness of the fuel rearrangement or fuel removal corrective actions designed to reduce fast neutron flux of the vessel wall.

As part of licensee responses to the August 21, 1981 NRC request, the eight licensees have been asked to comment on the effectiveness and practicality of the various proposed corrective actions.

EPRI is sponsoring a program to evaluate the effectiveness of proposed corrective actions. They have already presented preliminary results of these studies regarding benefits to be derived from warmer safety injection water, and they have also presented results of long-term benefits to be derived from annealing irradiated pressure vessel materials at various temperatures, as well as a preliminary study by Westinghouse regarding the feasibility of in-place pressure vessel annealing. These results were presented at the Ninth Water Reactor Safety Research Meeting, October 26-30, 1981, held at the National Bureau of Standards in Gaithersburg, Maryland.

(8) Regulatory Position

Utilizing all of the above described information, particularly the risk vs. EFPY from sub-task 5 and the effectiveness of proposed fixes from sub-task 7, the NRR staff will propose a Regulatory Position for Commission approval and issuance for public and industry comment. This proposed Regulatory Position will be compatible with the NRC's safety goal position currently under development. After resolution of the comments, an implementation position will be recommended to the Commission. We anticipate that the implementation position will contain: (1) required plant-specific limits; (2) suggested corrective actions for plants that exceed those limits; and (3) a justification of the acceptability of plants not exceeding those limits.

C. Management of Work

The responsibility for preparing and implementing a program to resolve this Unresolved Safety Issue is with the Generic Issues Branch, Division of Safety Technology (DST), Office of Nuclear Reactor Regulation. A Task Manager in the GIB will provide overall management of all work identified in this Task Action Plan, including coordination of all work performed by other divisions and branches, both within NRR and RES. NRR will have the responsibility of taking licensing-related actions on pressurized thermal shock issues during the conduct of this program.

D. Schedule

The following schedule estimates have been developed for the completion of the major tasks of this program.

Tentative Schedule

	Sub-Task	Estimated Completion Date
(a)	Review of Requested Information	June 1982
(b)	Draft of Revised Reg. Guide 1.99	June 1982
(1)	Determination of Events	May 1982
(2)	Transient Model Development	May 1982
(3)	$P(t)$ and $T(\overline{r},t)$ Calculation	August 1982
(4)	Fracture Mechanics Code Development	September 1982
(5)	Fracture Mechanics Calculation	October 1982
(6)	Sensitivity Studies	January 1983
(7)	Benefits of Corrective Actions	November 1982
(8)	Regulatory Position	May 1983

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION

The staff has made a preliminary evaluation to determine whether any immediate licensing action is necessary. This evaluation included: (1) the types of transients or accidents that could lead to overcooling of the reactor system; (2) experience to date with transients that have occurred at PWRs in the United States; (3) the probability that such overcooling events will occur; (4) initial and irradiated material properties; and (5) the capability of reactor vessels to withstand these transients based on fracture mechanics calculations. Items 4 and 5 focused on the likelihood of a flaw existing in a reactor vessel, material properties of the vessel, the copper content of reactor vessel welds, and the extent of reactor vessel irradiation (fluence).

A. Background

Severe reactor system overcooling events which could occur under pressure or be followed by repressurization of the PWR reactor vessel (PTS events) can result from a variety of causes. These include instrumentation and control system malfunctions and postulated accidents such as SBLOCAs, MSLBs, or feedwater pipe breaks. Rapid cooling of the reactor vessel internal surface induces a temperature gradient across the reactor vessel wall. The temperature gradient induces thermal stresses, with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature differences across the reactor vessel wall. Effects of this thermal stress are compounded by the pressure stress if the vessel is repressurized.

As long as the fracture resistance of the reactor vessel material remains high, such transients (except for extremely severe events) will not cause failure. After the fracture toughness of the vessel is reduced by neutron irradiation, severe thermal transients could initiate crack propagation from fairly small flaws near the inner surface and result in significant cracking. The vessels of most concern are those with high radiation exposure in materials of relatively high sensitivity to radiation damage (such as those made with welds of high copper content).

For failure of the reactor pressure vessel to occur, a number of contributing factors must be present. These factors are: (1) a flaw of sufficient size to initiate and propagate; (2) a level of irradiation (fluence) and properties and composition sufficient to cause significant embrittlement of the material (the exact fluence is dependent upon trace elements present, that is, high copper content causes embrittlement to occur more rapidly); (3) a severe overcooling transient with repressurization; and (4) the crack must be driven to a size and location such that the vessel fails.

B. Evaluation

The staff preliminary review of overcooling events and their probabilities included a review of the staff's study on the frequency of overcooling events at Babcock & Wilcox (B&W) plants (Ref. 1), a survey of operating experience on Westinghouse (W) and Combustion Engineering (CE) plants (Ref. 2); a review of available accident analyses in Final Safety Analysis Reports and in vendor topical reports; and a preliminary probabilistic analysis performed by DST (Ref. 3). The preliminary results of these evaluations indicate that there is a probability of about 10^{-3} per reactor year that a B&W-designed plant will experience a severe overcooling transient similar to or worse than that experienced at Rancho Seco on March 20, 1978. The Rancho Seco transient was the most severe overcooling transient experienced by any PWR in the United States. This probability of 10^{-3} per reactor year includes contributions from steam generator control system malfunctions (the dominant contributor); SBLOCAs; main steamline or feedwater line breaks; and complete loss of feedwater flow. The staff estimated that the probability of such an overcooling event in CE or W-designed reactors is lower, perhaps by an order of magnitude, than for B&W-designed reactors. This difference is based on design differences and on operating experience.

In the 1978 Rancho Seco transient, reactor pressure was maintained at a fairly high level (1500 psig to 2100 psig) throughout the cooldown. The minimum temperature of the reactor coolant (280°F) during the transient was high enough so that material toughness of the reactor vessel was adequate. This evaluation leads the staff to believe that if this transient were to be repeated at Rancho Seco or any other B&W-designed facility within the next few years, the reactor vessel failure would still be unlikely. Nonetheless, the possibility of vessel failure as a result of an overcooling event cannot be completely ruled out. If an overcooling event such as that at Rancho Seco were to occur, even for the vessel with the most limiting material properties in existance today, the staff would not expect a failure.

The staff conclusion is supported by the ORNL analyses of the Rancho Seco event (Ref. 4). Reference 4 analyses and later ORNL analyses (Refs. 5 and 6) indicate that the threshold irradiation level for crack initiation (that is, small cracks growing to larger ones assuming conservative initial material properties such as RT_{NDT}=40°F and copper content of 0.35%) would be in the range of 10^{19} neutron/cm^2. The highest fluence to date in a B&W-designed facility is less than half the minimum value listed above. It would, therefore, be several years before any B&W-designed facility reached its threshold irradiation level.

Some reactor vessels in CE and W facilities have somewhat higher fluences; however, other mitigating factors--such as lower values of initial RT_{NDT}--provide a significant margin of failure should an overcooling event similar to that at Rancho Seco occur.

It should be pointed out that the NRC staff does not believe BWRs have a significant PTS concern, for several reasons. Most importantly, BWRs operate with a large portion of the water inventory inside the pressure vessel at saturated conditions, (that is, it exists as a mixture of steam and liquid water at the mixture's boiling temperature and pressure). Any sudden cooling will condense steam and result in a pressure decrease, so simultaneous creation of high pressure and low temperature (necessary to cause a PTS concern) is very improbable. BWR operating experience provides verification that PTS events are very improbable since there have been no significant PTS events at any domestic or foreign BWR (that is, significant pressurization during or after a severe overcooling has not occurred). Also contributing to the lack of PTS concerns for BWRs is the lower fluence of the vessel inner wall, since BWRs have more water between the core and the vessel wall due to the recirculation flow path (water shields the vessel from the core). Finally, the operating pressure of BWRs is lower, which results in a lower stress intensity at the bottom of a postulated crack.

C. Conclusions and Recommendations

As a result of its evaluations to date, the staff has concluded that the probability of a severe overcooling transient (similar in magnitude to the Rancho Seco event) is relatively low. For B&W-designed reactors this probability is estimated to be about 10^{-3} per reactor per year, and for W- and CE-designed reactors, it is lower, perhaps by an order of magnitude. In addition, the staff has concluded that, based on present irradiation levels at operating reactors, reactor vessel failure from such an event is unlikely. Accordingly, the staff believes that no immediate licensing actions are required on operating reactors pending resolution of this issue. For plants not yet licensed, licensing can proceed for all of the above reasons. Also, the long-term PTS resolution will be produced by this TAP before irradiation history at those new plants is large enough to cause a significant PTS concern.

4. TECHNICAL ORGANIZATIONS INVOLVED

A. Generic Issues Branch, Division of Safety Technology, Office of Nuclear Reactor Regulation

Manpower Requirements: 1982 1-1/4 man-year
1983 1-1/4 man-year

(See Section 2.C) - Overall coordination and direction of the effort will be provided by GIB.

B. Office of Nuclear Reactor Regulation (Other Branches)

A significant portion of the work on this project will be performed by contractors as discussed throughout this TAP and as summarized in Figure 1. The contracts will be administered by RES, but the appropriate NRR personnel will be used to closely monitor and direct the various technical disciplines involved in the contract work as it progresses to assure that the work produced satisfies the licensing needs. In addition, several Technical Assistance programs will help with this work (see Section 4.D). Also, the various contractor efforts (reports) will be reviewed when submitted. Manpower estimates are given below in the form (x, y) where x is the branch's professional staff-year estimate for FY-1982 and y for FY-1983. See also Table 1 for further summary of efforts involved. The effort indicated on Figure 1 and in the paragraph below does not include the short-term PTS program described in items (a) and (b) above concerning the eight plants that received the August 21, 1981 letters, and the Regulatory Guide 1.99 Draft revision. See TAC #47548 and the other seven sequential TACS for the item (a) separate manpower request, or see the summary given in Table 1 of this TAP which shows a line entry for each item. The estimates and schedules below are for the long-term program described in this TAP.

This TAP will involve: the Materials Engineering Branch (2, 2) (that is, 2 man-years in FY-1982, 2 man-years in FY-1983) for materials properties and fracture mechanics direction and support; the Probability and Risk Assessment Branch (1/2, 1/2) for support in the estimation of probabilities for several PTS events and quantification of the event trees; the Reactor Systems Branch (1/2, 1/2); for direction of control system studies and transient code development and verification; the Instrumentation and Control Systems Branch (1/2, 1/2) for direction of control system studies and transient code development and verification; the Instrumentation and Control Systems Branch (1/6, 1/6) for direction of control system studies; Core Performance Branch

(1/4, 1/3) for fluence studies and studies of corrective actions involving fuel removal or re-arrangement to reduce flux at the vessel wall; the Division of Human Factors Safety (1/3, 1/3) for direction of studies on operator errors, procedures and training; and the Division of Licensing (1/2, 1/2) for coordination of requests to licensees. A breakdown by branch showing when the manpower will be required is shown in Figure 2.

C. Office of Nuclear Regulatory Research (2, 2)

RES resources will be utilized to administer the various contracts, and in addition they will provide consultations and guidance to the various technical review disciplines in NRR. NRR is responsible for review milestones and licensing decisions, and time indicated for RES groups in this TAP are not to be construed as assignments. They are estimates of the time that will be spent as described above.

One of the two approaches to the sensitivity studies will be performed using methods developed by the Materials Engineering Branch of the Division of Engineering. See description under sub-task 7 above.

The contracts will be:

ORNL will analyze event sequences leading to PTS and will estimate the probability of vessel failure at one "lead" plant for each PWR vendor.

LANL and INEL will improve and verify transient analysis codes and will calculate $P(t)$ and $T(\overline{r},t)$ for use in the ORNL fracture mechanics analyses. BNL will study fluence to the pressure vessel and assist in evaluation of proposed corrective actions involving fast neutron flux reduction.

RES plans to participate in the EPRI/CREARE experiments to obtain certain data needed for code development such as thermal mixing in the downcomer and cold legs.

Section B.b.4 describes the HSST program at ORNL that is also a part of the RES program being applied to the PTS concern.

D. Technical Assistance (also see Table 2)

The Reactor Systems Branch of the Division of Systems Integration, NRR will utilize Technical Assistance contracts at INEL and LANL to review several thermal hydraulics codes used by the licensees to calculate pressure and temperature history as a function of time for the selected event sequences.

The Core Performance Branch of the Division of Systems Integration, NRR will utilize technical assistance at BNL to benchmark the DOT 3.5 fluence code.

The Generic Issues Branch of the Division of Safety Technology, NRR will utilize a contract with Pacific Northwest Laboratory (PNL) to form a functional multi-disciplinary group to investigate PTS. The functional group will contain one or more experienced professional persons in: probability and risk assessment systems (PRA), thermal hydraulics, materials, fracture mechanics, and non-destructive examination. The PNL effort will also utilize nationally known consultants in the various fields as necessary.

5. POTENTIAL PROBLEMS

A. Close coordination and unity of purpose is required between NRR and RES.

B. Close cooperation is needed between ORNL and the licensees of the three "lead" plants.

C. Close supervision of ORNL is needed from a combined "NRR/RES" group.

D. NRC and ORNL must see that LANL, BNL and INEL remain closely coordinated with the overall effort.

E. Coordination and cooperation must be maintained with industry to provide analyses and data for NRC studies.

REFERENCES

1. "Insights on Overcooling Transients in Plants with the B&W NSSS," M. Taylor to S. Fabic, dated October 29, 1980.

2. Nuclear Power Experience 1980, Bernard J. Verra, Publisher: Nuclear Power Experience, Inc. Encino, California.

3. Frequency of Excessive Cooldown Events Challenging Vessel Integrity, A. Thadani to G. Lainas, dated April 21, 1981.

4. Parametric Analysis of Rancho Seco Overcooling Accidents, ORNL letter, R. D. Cheverton to M. Vagins (NRC, RES), March 3, 1981.

5. Evaluation of Pressurized Thermal Shock, Oak Ridge National Laboratory, NUREG/CR-2083, October 1981.

6. Staff review of ORNL Report on Pressurized Thermal Shock, Memorandum for the Commissioners from W. Dircks, EDO, October 30, 1981.

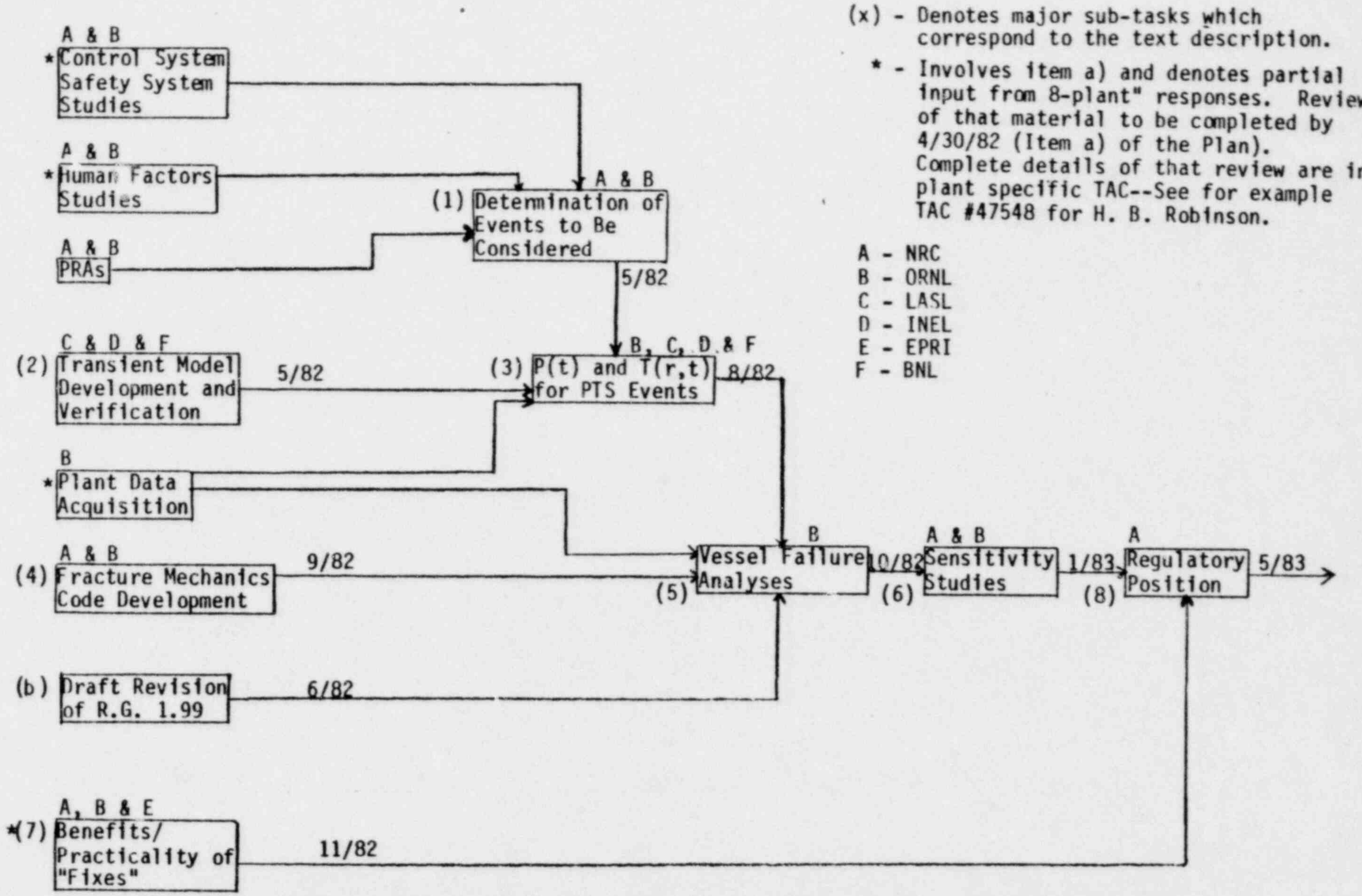
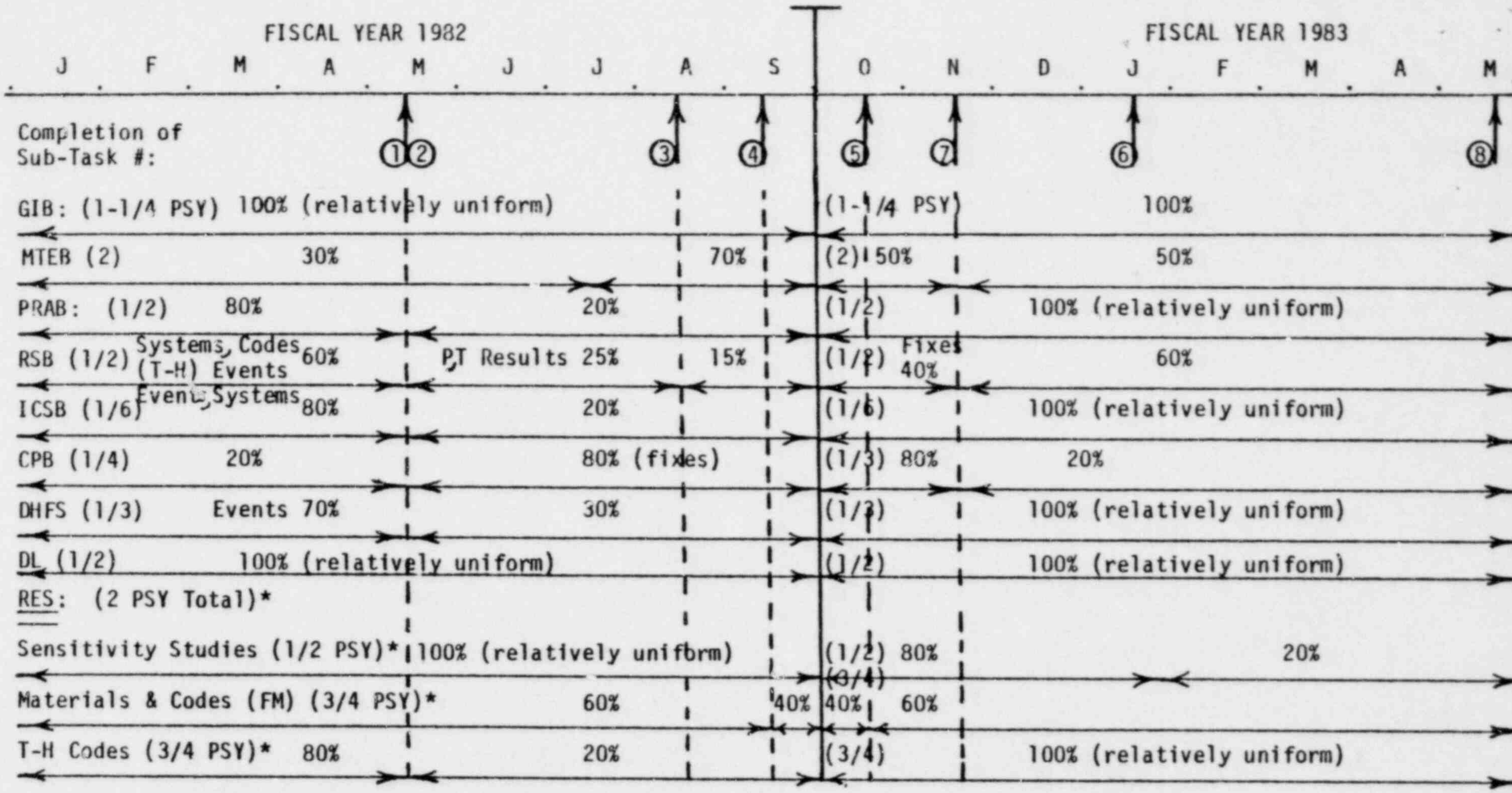


FIGURE 1

LONG TERM PLAN



Note: Data shown are % of Professional Staff Years (PSY) time commitment - PSY shown in ( ).

*See Section 4.C. RES times indicated are estimates of consulting time and contract monitoring time that will be used but are not to be considered commitments to the review effort aimed at generic licensing or Regulatory Requirements.

FIGURE 2. SCHEDULE DETAILS

TABLE 1

PRESSURIZED THERMAL SHOCK
NRC PROFESSIONAL STAFF YEARS

DESCRIPTION	BRANCH OR PERSON	FY 82 PSY	FY 83 PSY
SHORT TERM PROGRAM	DST/GIB	0.50	0
	DST/RRAB	0.25	0
(See Section 2.A.a and 2.B.a above)	DL/ORB	0.25	0
	DL/ORB	0.17	0
	DSI/RSB	0.25	0
	DSI/CPB	0.17	0
	DHFS/PTRB	0.17	0
	DE/MTEB	0.75	0
	RES/MEB	0.42	0
TOTAL SHORT TERM	NRR	2.51	0
	RES	0.42	0
LONG TERM PROGRAM			
(Reference Draft TAP for A-49)	DST/GIB	1.25	1.25
	DST/RRAB	0.50	0.50
	DL/ORB	0.50	0.50
	DSI/CPB	0.25	0.33
	DSI/RSB	0.50	0.50
	DSI/ICSB	0.50	0.50
	DHFS/PTRB	0.33	0.33
	DE/MTEB	2.	2.
	RES/MEB	1	1
	RES/Johnson	0.5	0.5
	RES/Shotkin	1	1
TOTAL LONG TERM	NRR	4.58	5.60
	RES	3.5	2.67
RELATED PROGRAMS			
Reg Guide 1.99 Revision P. Randall (See Section 2.A.b and 2.B.b above)		0.25	0.2

TABLE 2

PRESSURIZED THERMAL SHOCK - RESEARCH AND TECHNICAL ASSISTANCE

FIN #	DESCRIPTION	NRC CONTACT	CONTRACTOR	FY 82 K$ (1)	FY 83 K$ (1)
B0119	HSST	Vagins	ORNL	4595	4677
B8133	LWR Pressure Boundary Integrity	Vagins	ENSA	500	600
B5988	Surveillance Dosimetry	Serpan	HEDL	762	980
B0415	Pressure Vessel Simulation	Serpan	ORNL	569	300
B6224	Dosimetry Meas. Data Base	Serpan	NBS	128	200
B7026	JR Curve	Vagins	USNA	60	70
A3215	Code Assessment and Application	Shotkin	BNL		
A6047	Code Assessment and Applications	Shotkin	INEL	800	700
A7027	Analytical Res. in LWR Safety	Shotkin	LANL		
A7217	TRAC Calc Assistance	Shotkin	LANL		
B0468	Pressurized Thermal Shock	C. Johnson	ORNL	500	300
A7272	Reactor Systems Support of Operating Reactors Action Item	Throm	LANL	235	100
B0763	Review of LOFTRAN and MARVEL	Throm	ORNL	35	0
A3381	Pressure Vessel Irradiation Embrittlement	Lois	BNL	180	200
-	USI A-49 at PNL (Review group and individual consultants)	R, Woods	PNL	400	400
TOTAL - PTS PROGRAM				8764	8527

(1) Dollars shown are for the portion of the FIN which is for PTS. The only two FINS which are exclusively PTS are B0468 and the Undesignated FIN for USI A-49.