TASK ACTION PLAN, REVISION 1

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (TASK A-45)

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1. INTRODUCTION

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity via a turbine generator. Following a reactor shutdown, a reactor produces insufficient power to operate the turbine; however, the radioactive decay of fission products continues to produce heat (so-called "decay heat"). Therefore, when reactor shutdown occurs, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressures do not develop which could jeopardize the reactor and the reactor coolant system. It is evident, therefore, that all light water reactors (LWRs) share two common decay heat removal functional requirements, namely: (1) provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink, and (2) maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The reliability of a particular power plant to perform these functions depends on the occurrence frequency of initiating events requiring or jeopardizing decay heat removal operations and the probability that required systems will respond to remove the decay heat.

The results of the Reactor Safety Study (WASH-1400), Ref. 1, showed that the overall frequency of core meltdown in the first generation of large commercial LWRs was probably higher than had been expected (about 5 x 10-5 as compared to 1 x 10-6 per reactor year). Insufficient reliability in the systems required for the decay heat removal function, particularly in response to small loss-of-coolant accidents (LOCAs), was shown to be responsible for a substantial portion of the overall probability of core meltdown.

If it were considered necessary to reduce the risk which is presented to the public by an LWR, there would be a choice between preventive measures to reduce the probability of accidents leading to severe core damage and measures to mitigate the consequences of accidents leading to severe core damage, if they should occur.

Since the probability of failure to remove decay heat is a major contributor to the overall risk, it follows that one of the main aims of preventive measures should be to reduce this probability. However, it must be noted that the scope for risk reduction by this means alone is somewhat limited. For example, if accidents involving failure of decay heat removal contribute 80% of the total risk, then other types of accidents must contribute 20%. Thus a 10-fold improvement in the reliability of the decay heat removal can only reduce the total risk by a factor of about 3.

Thus it follows that, although prevention is fundamentally a sounder solution than mitigation (e.g., the frequency of events which might alarm the public would be reduced and the large investment in the plant would be better protected), nevertheless, mitigation may be more cost-effective.

Consequently, in the Commission's "Severe Accident Rulemaking" proceedings to reduce the risk to the public from LWRs, provision has to be made to compare the relative merits of prevention and mitigation for individual plants. The Task Action Plan, A-45, described here is aimed at ensuring that an appropriate input to the rulemaking process is made in relation to decay heat removal systems (DHRS). In order to do this, quantitative and qualitative acceptance criteria are developed which can be used to test the acceptability of individual designs. Those plants in which assessment shows the DHRS to be inadequate then become candidates for improvement, either by improvement of their decay heat removal capability, or by other means, depending on which is considered to be the more cost-effective.

The principal means for removing the decay heat in a pressurized water reactor (PWR) under normal conditions immediately following reactor shutdown is through the steam generators using the auxiliary feedwater system. In addition to the WASH-1400 study mentioned above, later reliability studies and related experience from the Three Mile Island Unit 2 (TMI-2) accident have reaffirmed that the loss of capability to remove heat through the steam generator is a significant contributor to the probability of a core melt event.

It should be noted that many improvements to the steam generator auxiliary feedwater system were required of the licensees by the NRC following the TMI-2 accident. However, the staff feels that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, this Unresolved Safety Issue (USI) will investigate alternative means of decay heat removal in PWR plants, including but not limited to, using existing equipment where possible. This study will include a representative sample of plant-specific decay heat removal systems evaluations. It will result in recommendations regarding the adequacy of existing decay heat removal requirements and the desirability of, and possible design requirements for, an alternative decay heat removal method, other than that normally associated with the steam generator and secondary coolant system.

This Unresolved Safety Issue program will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).

2. DESCRIPTION OF PROBLEM

A. Nomenclature and Definitions

When a reactor is shut down after operating at power for some time, the effect on the subsequent operating procedures for maintaining safe conditions of four (4) separate heat sources must be taken into account, namely:

(i) the power produced by the fission process while shutting down;

(ii) the sensible heat stored in the fuel;

(iii) the heat due to fission product decay in the fuel; and

(iv) the sensible heat stored in the reactor coolant system (RCS) and in the reactor coolant itself.

These sources are described variously as "residual heat," "decay heat," and "shutdown decay heat," but the term "residual heat" is also used in a more specific sense to mean the fission product heat produced after the reactor has been brought to the "hot shutdown condition." (That is, the initial thermal transients have died out and quasi-steady state has been reached in which reactor coolant temperature and pressure remain constant, at a water temperature of about 300°F in a PWR.) The term "residual heat" will be used in this way in the proposed Revision of Regulatory Guide 1.139 (Ref. 2), which forms part of Task A-45.

Strictly speaking, the term "decay heat removal" could also be considered to include not only the processes used to transfer heat from the reactor to some ultimate heat sink but could also include the processes required to reflood the reactor in the event of a severe loss-of-coolant accident (LOCA). However, in the context of this Task Action Plan, the initial reflooding phase is considered to be a separate issue, whereas the operation in the longer term of the systems used for reflooding in order to assist in the transition to a quasi-steady "hot shutdown" state and their subsequent use in a recirculating mode, are considered in this plan. The auxiliary systems required to achieve and maintain the core in a shutdown condition, notably the coolant chemical volume and control system and depressurization systems, are also considered. A list of the systems which are relevant to the decay heat removal function is contained in Appendix A to this plan. However, not all of these systems will have to be considered in detail.

Thus, the definitions used in this Task Action Plan are as follows:

(a)	Reflood phase - (RFP)	The initial phase of a severe LOCA, when the objective is to reflood the reactor.
(b)	Shutdown decay - heat removal (SDHR) phase	The transition from reactor trip to "hot shutdown," excluding the initial reflooding phase in a severe LOCA.
(c)	Residual Heat - Removal (RHR) phase	The transition from "hot shutdown" to "cold shutdown" and maintaining cold shut down conditions.

(d) Decay Heat Removal (DHR) phase - SDHR and RHR phases combined.

To provide a clear understanding of the terms involving various stages of shutdown, the following definitions will be utilized in this Plan:

	Stage*	Average Coolant Temperature
PWR	Hot Standby	$\geq$ 350°F**
	Hot Shutdown	350°F**>T>200°F
	Cold Shutdown	$\leq$200°F
BWR	Hot Shutdown	>212°F
	Cold Shutdown	$\leq$212°F

DHRS in the context of this Task Action Plan is defined as those components and systems required to maintain primary and/or secondary coolant inventory control and to transfer heat from the reactor coolant system and containment building to an ultimate heat sink following shutdown of the reactor for normal events, off-normal transient events (e.g, loss of offsite power, loss of main feedwater) and the smaller LOCAs, described as "S2" in the Reactor Safety Study (i.e., 1/2" to approximately 2" diameter holes; a diameter of 2" is the largest of the more likely breaks to be expected). DHRS does not encompass those emergency core cooling systems required only to maintain coolant inventory and dissipate heat during the first ten minutes following medium or large LOCAs. However, it is necessary in Task A-45 to consider the supporting systems (e.g., the chemical and volume control system, depressurization systems, and the containment cooling systems) which would be required for successful decay heat removal in various modes. As indicated above, this Task Action Plan covers both the SDHR and the RHR phases.

It should be noted that these definitions are used rigorously in this Task Action Plan (e.g., where the term "DHR" is used, it must be understood that both the SDHR and the RHR phases are involved).

B. The Technical Issues

In a light water reactor there are three broad groups of fault sequences which can lead to severe damage to the fuel, namely:

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*In each stage, the reactivity condition (K_{eff}) is defined to be less than 0.99.

**This temperature is defined as approximately 305°F for some PWRs.

1. Gross failures of vital structures, such as the reactor pressure vessel, which prevent the reactor protection system and the engineered safety features from functioning effectively.

2. Failure of the reactor to shut down correctly (i.e., the ATWS type of fault) in the event of a disturbance which has led to an increase in the ratio of heat produced/heat removed for the fuel.

3. Failure to transfer the decay heat from the fuel to an ultimate heat sink of adequate capacity (e.g., due to loss of primary coolant or lack of auxiliary feedwater).

Studies such as WASH-1400 (Ref. 1) have shown that in general, for LWRs, the major contributor to the probability of severe damage to the fuel stems from failures to remove the decay heat in the SDHR phase, as defined above. However, the existence of the other two fault 2 sequences creates a finite limit to the extent of the improvement in safety which can be achieved by improvement in the performance and/or reliability of the shutdown decay heat removal systems (SDHRS) alone. It can be shown from WASH-1400 and similar studies (Refs. 3 and 4) that, for the stations analyzed, the maximum factor of improvement, in terms of probability of core melt, which could be achieved by improvements to the shutdown decay heat removal systems (including those required in post reflood conditions) alone is about five (5). In other U.S. stations, it is believed that the probability of core melt may be greater, due to lower reliability of their auxiliary feedwater systems (AFWS). Clearly, in those stations, larger reductions in the probability of core melt could be achieved by improvements in the systems required to remove shutdown decay heat. Action has been, or is being, taken to improve the AFWS at those stations.

The existence of this finite limit to the improvement in safety which can be achieved by modifications to the SDHRS alone implies that the cost effectiveness of radical and expensive changes may be low, and therefore, the systematic study delineated herein is required.

The major part of the Task Action Plan is concerned with the first (SDHR) phase, as defined above, but the second (RHR) phase is also covered. In the RHR phase the main problems are (i) to ensure adequate reliability in the electrical and mechanical equipment of the RHRS during prolonged exposure to a hostile environment, such as would be encountered after a LOCA, whether small or large, and (ii) to ensure adequate reliability of the RHRS after being subjected to severely disturbed conditions, such as earthquakes, floods or fires.

In the case of a PWR, it is useful to differentiate between three distinct types of fault sequences which lead to a requirement for shutdown decay heat removal; these are as follows:

(a) Sequences in which there is no loss of primary coolant.

(b) Sequences which commence as in (a) but which degenerate to a state in which the increase in primary coolant pressure causes the relief or safety valves to lift, but reclosure occurs, or isolation is possible.

(c) Sequences in which the initiating event is either:

(i) rupture of the primary coolant circuit,

(ii) failure of RCS pump seals, or

(iii) lifting of a primary circuit relief or safety valve, as in (b), followed by a failure of the valve to re-seat and, in the case of a relief valve, failure of its associated isolating valve to function.

In the first class of sequences, the primary coolant can be kept sub-cooled; in the second state, a controlled blowdown of the primary coolant is possible or alternatively restoration of sub-cooled conditions should be feasible; in the third class of sequence, loss of a large proportion of the primary coolant is inevitable, though restoration of sub-cooled conditions, by continuous injection of fresh water to replace that lost, should be possible if the breach is small (of order one square inch or less).

Thus the problems of shutdown decay heat removal in the type (c) sequences are related mainly to the rate and reliability of injection of emergency cooling water and the rejection of heat from that water to the containment support systems and thence to an ultimate heat sink, whereas in the type (a) and (b) sequences, the problems are related mainly to the transfer of decay heat from the fuel to the primary coolant and the rejection of that heat by circulation through heat exchangers, such as the steam generating units, and from these to an ultimate heat sink.

However, two intermediate cases can be identified for a PWR, namely:

(i) Shutdown decay heat removal by the so-called "feed and bleed" procedure, and

(ii) Shutdown decay heat removal by operation of the steam generating units as reflux condensers.

The existence of those two intermediate cases is taken into account in defining the scope of this Task Action Plan.

In the case of a BWR, improvement of the SDHRS is a less complex problem than in a PWR, since there can be no transition from sub-cooled to saturated conditions in the reactor coolant and boiling in the core is the normal mode of operation. However, the greater simplicity of the BWR tends to reduce the extent to which diversity can be introduced into the design of the SDHRS.

For both PWRs and BWRs, the main technical issues in the RHR phase relate to the reliability of RHR systems, continuity of operation of the RHR system during severely disturbed conditions and the extent to which the components of the RHR system are required to meet requirements for safety grade equipment, including the associated value/impact for existing plants.

C. Background

The TMI-2 accident demonstrated how a relatively common fault, which the operator should have been able to cope with easily, could escalate into a potentially hazardous situation, accompanied by severe financial losses to the utility, owing to difficulties arising in the decay heat removal process.

Other circumstances, of a more unusual nature (e.g., damage to systems by external events such as floods or earthquakes; or by sabotage), which could make removal of the decay heat difficult can also be foreseen.

The question arises therefore whether current licensing design requirements are adequate to ensure that LWRs do not pose unacceptable risk due to failure to remove shutdown decay heat, and whether, at a cost commensurate with the increase in safety which could be achieved, improvements could be made in the effectiveness of shutdown decay heat removal in one or more of the situations described in Section 2.B. above. Resolution of this question is considered to be of sufficient importance to merit raising it to the status of an "Unresolved Safety Issue" (USI).

To some extent the effectiveness of the SDHRS is linked to that of the onsite and offsite electrical supplies; the performance and reliability of those supplies has already been raised to the status of a USI; that is, Task A-44, "Station Blackout." Consequently, the scope of work required herein in relation to the decay heat removal systems is complementary to the Task Action Plan for Task A-44 (Ref. 5). There are a number of other areas (Ref. 6) in which work conducted, or sponsored, by NRC and by other organizations is proceeding that relate to the present Task Action Plan. As discussed in Section 1 above, there is a particularly close relationship between Task A-45 and the work contemplated for Severe Accident Rulemaking.

The above activities have been taken into account in formulating this Task Action Program. In addition, the Task Action Plan embodies elements II-E.3.2, II-E.3.3, II-E.3.4 and II-E.3.5 of the TMI Action Plan, NUREG-0660.

D. Purpose

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shutdown decay heat. This will require the development of a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.

An integrated systems approach to the problem will be employed. Accordingly, quantitative methods will be used, where possible, to develop acceptance criteria for future plants and to measure the effectiveness, and acceptability, of the shutdown decay heat removal systems in existing plants.

In addition, any proposed improvements that are safety related would, of course, have to be consistent with the requirements imposed by the "General Design Criteria" of 10 CFR 50 and would have to take account of any relevant Regulatory Guides and Branch Technical Positions which are already in existence.

3. PLAN FOR PROBLEM RESOLUTION

A. Approach To The Problem

In view of the difference in nature of the technical problems encountered in the SDHR phase and in the RHR phase, the plan presented below is divided into separate sections covering the SDHR phase and RHR phase.

A.I. Shutdown Decay Heat Removal (SDHR)

The approach taken to this phase of the problem comprises the following main elements:

- Development of criteria to judge acceptability of the SDHR function in existing and future plants.

- Development of means for improvement of the SDHR function.

- Assessment of existing plants against the acceptance criteria to identify those in which the SDHR function would require improvement to meet the criteria.

- Development of a plan for implementing proposed new requirements, if any, for SDHR systems required to meet the acceptance criteria above.

Each of these elements constitutes a major Sub-Task, the technical content of which is described in Section 3.B. below. The interrelation of each of the sub-tasks is shown in Figure 1. The relative timing of all work included in this Plan is provided in Part D (Schedule) of this section; more detailed schedules are provided in Appendix B.

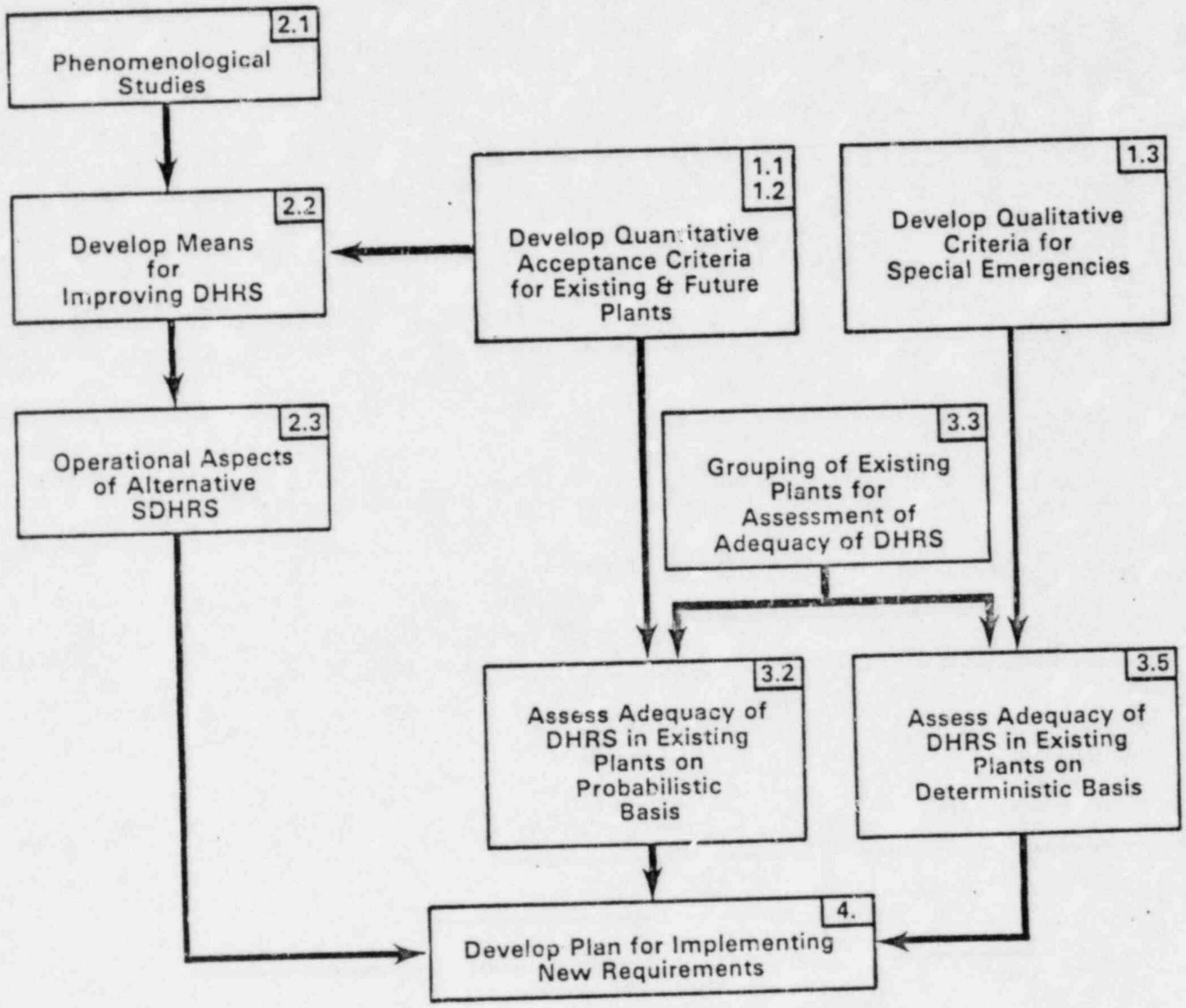
While Task A-45 is in progress, work on possible methods of improving the effectiveness of containment (e.g., filtered venting; hydrogen control and post-accident core retention) will probably be continuing as part of the work in support of the Severe Accident Rulemaking. Both types of work, which can be regarded as "prevention" and "mitigation," respectively, will form inputs to the process of making decisions as to the changes, if any, which should be made to existing plants. They will also provide a basis for deciding, in future plants, the optimum balance between measures to prevent core melt and those to mitigate its effects, if melting should occur.

A.II. Residual Heat Removal (RHR)

The approach taken to this phase of the problem is similar to that for Shutdown Decay Heat Removal, except that little conceptual development is anticipated. The approach is as follows:

- The development of criteria to judge the acceptability of the RHR function will be covered in the same sub-tasks as those for the SDHR function.
- It is not expected that any development of new means for carrying out the RHR function will be required, except perhaps in unusual circumstances (e.g., floods, fires) or in the event of sabotage, particularly for those older plants that have non-safety grade RHR systems.
- The assessment of the adequacy of the RHR system in selected existing plants will be covered in the same sub-task as that for the SDHR systems.
- The development of a plan for implementing proposed new requirements, if any, will be covered in the same sub-task as that for the SDHR system and will include the further development of Regulatory Guide 1.139 (Ref. 2).

Figure 1 Inter-Relation of Sub-Tasks in Task Action Plan A-45



B. Technical Content of Individual Sub-Tasks

SUB-TASK 1. Develop Acceptance Criteria for Assessment of DHR System

For task management purposes, Sub-Task 1 is divided into three parts:

1.1 Develop quantitative acceptance criteria for SDHRS and RHRS in existing plants.

1.2 Develop quantitative acceptance criteria for SDHRS and RHRS in future plants.

1.3 Develop qualitative criteria for acceptance of SDHRS and RHRS in "Special Emergencies" and other cases where the criteria developed in Sub-Tasks 1.1 and 1.2 need to be supplemented.

SUB-TASKS 1.1 and 1.2. Development of Quantitative Acceptance Criteria for DHRS in Existing and Future Plants

At the present time, there are no formally approved quantitative safety goals for nuclear power reactors in the U.S. The problem has been addressed by the Advisory Committee on Reactor Safeguards (ACRS), and a set of trial "Decision Rules" has been suggested for consideration by the Commission (Ref. 7). From time to time in order to provide a basis for urgent licensing decisions, the NRC staff has had to devise their own "decision rules" (e.g., see Refs. 8 and 9). Currently, the Office of Policy Evaluation has published (Ref. 10) for public comment the Commission's Proposed Policy Statement on Safety Goals for Nuclear Power Plants. Pending a decision by the Commission about the adoption of safety goals and their nature, it will be necessary to develop acceptance criteria for use in Task A-45, to provide a basis for decisions concerning the adequacy of DHRS in existing plants. The objectives of this work will be to provide a quantitative "yard-stick," supplemented by deterministic based acceptance criteria. In generating these criteria, consideration will have to be given to the following:

(a) The aforementioned Commission's proposed policy statement.

(b) The ACRS trial proposals -

These include suggested criteria for:

(i) Maximum acceptable frequency of core meltdown.

(ii) Maximum acceptable frequency of large uncontrolled releases to the environment.

(iii) Effects on the individual members of the public and on society as a whole.

(iv) Application of the "As Low As Reasonably Achievable" (ALARA) principle.

(c) Quantitative safety goals proposed by other organizations, such as the Atomic Industrial Forum.

(d) The need to distinguish between the reliability required from the DHRS in normal environmental conditions and for frequent events (loss of main feedwater, loss of offsite power), and in "Special Emergencies" (Ref. 4) due to external events, such as sabotage, floods, or earthquakes, or to internal events, such as cable fires or turbine disintegration, which have low probabilities of occurrence that are difficult to quantify.

(e) Provisional criteria already developed to assist in making licensing decisions by NRC staff (e.g., as in Refs. 8 and 9).

(f) The need to adhere to the ALARA principle and to justify decisions on a "Value/Impact" (i.e., cost/benefit) basis.

For simplicity it is considered that the acceptance criteria should be concerned mainly with the occurrence of large scale fuel melt (more than 30% of the oxide fuel becoming molten), as in the ACRS trial criteria, since this event is closely related to the performance of the DHRS. Therefore, the acceptance criteria for existing and future plants should be based primarily on the frequency of core melt due to DHRS failures and should also define the reliability (in terms of maximum acceptable probability of failure per demand) required from those systems that are part of the DHR function.

In the development of the acceptance criteria, it will also be necessary to recognize that the probability of the unwanted event (i.e., partial fuel melt) depends upon the frequency of demands on the DHRS as well as upon the reliability of the DHRS itself. In this context, it should be noted that as an extension of the "defense in depth" philosophy, it may be worthwhile in some plants to make changes to reduce the frequency of demands, as well as to try to improve the DHRS itself. However, reducing the frequency of demands will not be covered in this Plan.

SUB-TASK 1.3 Development of Qualitative Criteria for "Special Emergencies"

It will also be necessary to supplement the above quantitative criteria to cover the "Special Emergency" situations (e.g., sabotage, fire, airplane crash, vapor cloud explosion), identified by Berry et al. (Ref. 4), which make a contribution to the overall risk that is difficult to quantify. These criteria will cover factors such as separation, redundancy and diversity. The design criteria used in certain foreign countries will be considered in this part of Sub-Task 1.

SUB-TASK 2. Develop Means for Improvement of DHR Function

In this Sub-Task, means for improving DHRS will be examined for certain selective plants or groups of plants. The investigation will cover three distinct aspects of the problem.

(i) A review of the phenomenological aspects, to ensure that the latest available data from "LOFT," SEMI-SCALE," and other test programs is integrated in the engineering studies which form the main part of the Sub-Task. The review will also include examination of the underlying physical processes in any novel solutions and the identification of further analytical and test work that would be necessary to support possible solutions.

(ii) Examination of the engineering aspects of possible means for improving DHRS in order to identify those which are sufficiently promising to warrant consideration for application to existing plants.

(iii) Examination of the operational aspects of alternative means of SDHR as they develop to ensure that the effects on overall system reliability are considered.

For task management purposes Sub-Task 2 is divided into three parts:

Sub-Task 2.1 Phenomenological Studies
Sub-Task 2.2 Conceptual Design Studies
Sub-Task 2.3 Operational Aspects of Alternative SDHR Systems

The technical content of each of these parts is described below.

SUB-TASK 2.1 Phenomenological Studies

Part I - PWR

At the present time there appear to be some alternative means for removal of shutdown decay heat from PWRs, that appear to be technically feasible, but the use of which has not yet been formally approved, even for emergency situations, in either existing or future plants. For example:

(i) Transfer of heat from the reactor core to the steam generators by two-phase natural circulation.

(ii) Operation of the steam generating units as reflux condensers, as an alternative to true natural circulation.

(iii) The use of a high-pressure residual heat removal system which could, in emergency, be brought into use before stable "hot shutdown" conditions have been established.

(iv) Application of the "feed and bleed" concept.

(v) Operation of a shut down PWR with limited boiling in the core.

Several of these possible methods have the advantage that they could, in principle, provide means of removing decay heat that do not require complex systems with large power supplies, and/or provide diversity in the means of removing decay heat.

Of these potential means of decay heat removal, item (i) and (ii) provide a substantial extension to the range of cases in which transfer of decay heat to the secondary coolant would be possible; items (iii) and (iv) provide genuinely diverse alternative methods of removing decay heat, which do not rely on the use of the steam generators; item (v) is of practical importance in the application of (iii) and (iv); items (ii) and (iv) may be able to provide the operator with a useful extension of the time available to deal with a situation such as loss of all feed water to the steam generators.

On-going programs on the thermal-hydraulics of PWRs will probably provide the data requested to determine whether it is practicable to make use of one or more of the above means. However, a review of some aspects of this work may be necessary, as part of Task A-45, and it is possible that some crucial analytical and/or test work may be identified, which could be carried out as part of one of the existing programs, to substantiate some of the assumptions on which alternative modes of operation are based.

In addition, preliminary thermal-hydraulic analyses of possible alternative SDHRS solutions selected for development in Sub-Task 2.2 will be carried out as part of Sub-Task 2.1 in order to establish major parameters such as flow, power, and instrumentation requirements and to identify any test work required to substantiate system performance.

Part II - BWR

The work content of this part of Sub-Task 2 will be similar to that of Part I, but there appears to be less scope for the introduction of alternatives to the means already employed in existing BWRs. However, besides the decay heat removal system options discussed in Reference 4, other possible alternatives for BWRs are:

(i) For existing plants, the provision of a secondary suppression pool, or an isolation condenser, since in the more recent designs decay heat removal depends upon the continued integrity of the suppression pool.

(ii) For existing plants, the provision of a higher capacity RHR heat exchanger for cooling the suppression pool.

SUB-TASK 2.2. Conceptual Design Studies

In the Sandia study of alternative DHR systems, a number of possible schemes have been identified (Ref. 4). In the final phase of the Sandia program, which was originally part of the generic water reactor safety research program, but is now part of the Severe Accident Research Program, the engineering feasibility of six possible schemes for PWRs and three possible schemes for BWRs has been examined by an architect/engineering organization with appropriate experience.

The objective of the program, as stated by Sandia, is "...to perform an initial screening (based on engineering experience and judgment) of the concepts on the basis of:

(1) ability to backfit
(2) feasibility
(3) state-of-the-art
(4) cost
(5) independence
(6) ability to meet emergencies

For those concepts which can be shown to be most promising, the contractor should develop a preliminary design involving major components and support systems. These designs should then be analyzed for their compatibility with several existing and new power plant designs. This analysis should focus upon identifying:

(1) costs
(2) interface requirements
(3) operational problems
(4) unresolved technical issues."

It is expected that the Sandia program on alternative DHR systems will provide a valuable input to Task A-45, but depending on the outcome of the review work described under Sub-Task 2.1 above, it may be necessary to examine the engineering feasibility of some other possible solutions, on the same lines as in the Sandia program. These may include the following:

(a) PWR

(i) Feed and bleed, with the minimum of additional equipment.
(ii) More intensive utilization of natural circulation of both primary and secondary coolant, in all modes (i.e., single phase, two phase and reflux condensation).
(iii) Deliberate operation with limited boiling in the core.

(iv) Further examination of a high pressure recirculation system (similar to that used in the RHR phase).

(b) BWR

More intensive utilization of natural circulation, including reflux condensation, to reduce the emergency power requirements and to simplify systems.

In addition, greater emphasis than may be intended in the Sandia program will be given in Sub-Task 2.2 to factors such as simplicity of emergency operating procedures, increased diversity in the means of removing decay heat, reductions in the demand for emergency power and the possibility of improvising "last ditch" methods for removal of decay heat. The scope for increasing the reliability of the DHR function with the minimum of equipment will also be investigated.

As part of this Sub-Task, the merits of alternative decay heat removal systems as utilized in certain foreign countries will be evaluated. The evaluation of foreign LWRs reported by Sandia in Reference 4 provides a good starting point for this study.

Based on value/impact (i.e., cost/benefit) evaluations, as part of Sub-Task 2.2, the alternative systems that can be shown to meet the acceptance criteria developed under Sub-Task 1 will be ranked in terms of their suitability to substantially increase the plants' capability to deal with a broader spectrum of transient and accident situations.

Sub-Task 2.2 will also include the conceptual development of designs for separate, dedicated DHRS capable of functioning in "Special Emergency" situations for existing plants, to a point at which feasibility is established and the cost of such dedicated systems can be estimated. Use will be made of previous work in this area, e.g., that described in References 4 and 13, including the alternative DHR systems utilized in some foreign LWRs.

The scope of this Sub-Task 2.2 also includes performing value/impact evaluations to determine to what extent existing plants should have the capability to achieve and maintain cold shutdown using safety grade equipment. As part of this, Sub-Task 2.2 will consider the adequacy of reliability and performance criteria and standards for systems that are required to achieve and maintain cold shutdown conditions. The results of this part of the Sub-Task will be of major importance in the future revisions of Regulatory Guide 1.139 (Residual Heat Removal System) and the associated section of the Standard Review Plan, Section 5.4.7, which forms part of the scope of Sub-Task 4.

SUB-TASK 2.3 Operational Aspects of Alternative SDHR Systems

An important aspect of the reliability of the SDHR function is the practicability and simplicity of the operating procedures which are required. The two main factors are:

(a) The time available to take action at each part of the operating sequence and the extent to which the time factor necessitates automation of the operating procedures.

(b) The degree of similarity between the operating procedures required for the various accident scenarios which lead to a requirement for SDHR.

In order to carry out the assessment of the adequacy of the DHRS in Sub-Tasks 3.2 through 3.5 below, some appreciation of both of the factors described above is required. An investigation to provide data on both aspects is therefore included in the study of means for improving the SDHRS. Once hot shutdown conditions have been established, the operator has a lot more time to consider his actions, thus it has not been considered necessary to extend the scope of Sub-Task 2.3 to include the operation of the RHRS.

The technical content of Sub-Task 2.3 is as follows:

Part I - Time Available for Operator Action (PWR and BWR)

(i) Review available data relating to the time scale of events for each of the means of SDHR examined in Sub-Task 2.2 and determine those areas, if any, where additional information is necessary.

(ii) Formulate a program of work to provide the additional information identified as necessary in (i) above, for consideration by NRC.

(iii) Evaluate the time available to the operator for a set of cases which will be defined by NRC.

Part II - Definition of Outlines of Operating Procedures (PWR and BWR)

(i) Review the available information relating to operating procedures for decay heat removal in LWRs, including an estimate of the probability of error on the part of the operator.

(ii) Define, in outline form, the operating procedures required for alternative means of SDHR and assess the probability of operator error.

(iii) Confirm that the instrumentation and controls required for the most effective operating procedures have been or are scheduled to be, installed on existing plants.

SUB-TASK 3. Assessment of Adequacy of DHRS in Existing LWRs

For task management purposes, Sub-Task 3 is divided into three parts (Note below that Sub-Tasks 3.1 and 3.4 were deleted from a previous version of the Plan, and Sub-Task number designations of 3.2, 3.3 and 3.5 were maintained to avoid confusion):

Sub-Task 3.2	Assess Adequacy of DHRS in Selected Existing LWRs
Sub-Task 3.3	Grouping of Other Existing Plants for Assessment of Adequacy of DHRS
Sub-Task 3.5	Assess Adequacy of DHRS in Existing Plants on a Deterministic Basis

SUB-TASK 3.2 - Assess Adequacy of DHRS in Selected Existing LWRs on a Probabilistic Basis

The Reactor Safety Study (WASH-1400), the Reactor Safety Study Methodology Application Program (RSSMAP) and the Interim Reliability Evaluation Program (IREP) will provide risk and reliability assessments for about ten specific plants (Surry, Peach Bottom, Sequoyah, Grand Gulf, Oconee, Calvert Cliffs, Browns Ferry, ANO-1, Millstone-1, Crystal River) in the near future. Risk assessments for four other plants (e.g., Zion, Indian Point, Limerick and Big Rock Point) should also become available in time to be of use in this Task Action Plan. Nevertheless, risk assessments may not be available for some types of plants (e.g., Westinghouse 2-loop reactors), and a different approach will become necessary, as described in Sub-Task 3.5, below.

For those existing plants where a risk and/or reliability assessment has been made, or will be available in a useful time the contributions to the overall core melt frequency from DHR system failures can be compared with the acceptance criteria. If the criteria are met, then no major change in design would be necessary, although application of the ALARA principle might suggest some minor changes. If the selected criteria are NOT met, some design changes should be considered but it will not be immediately apparent whether upgrades to existing DHR systems or a separate, dedicated system should be recommended. That is, a plant in this category would only be a "possible candidate" for improvement of its DHRS.

For those plants which are possible candidates for improvement, the usefulness of improving the various subsystems of the DHRS will be investigated by examining the effects on the overall core melt frequency of an arbitrary improvement (e.g., by factor 10) of each of the sub-systems' unavailability. This investigation will help to establish the relative priority to be given to each of the "possible candidate" plants.

The priority for the development of conceptual designs for improved DHRS for a specific plant (see Sub-Task 2.2) will depend on the estimated core melt frequency due to that plant and on the effectiveness of improvement in the DHRS as a means of reducing that frequency.

For existing plants, the need to provide any additional "dedicated" DHRS system, primarily to cope with "Special Emergency" situations, will be reviewed in the light of the criteria derived in Sub-Task 1.3.

SUB-TASK 3.3 - <u>Grouping of Other Existing Plants for Assessment of Adequacy of DHRS</u>

For those existing plants where a risk and/or reliability assessment is not expected to be available within a useful time, it will be necessary to extrapolate the results, obtained in Sub-Task 3.2 above, for those plants which they resemble most closely. Based on the specific design features of systems which perform the decay heat removal function for the plants noted in Sub-Task 3.2 above, it will be determined whether it is feasible to divide the operating U.S. commercial plants into groups. The groups will be defined such that evaluations and subsequent regulatory actions with regard to the decay heat removal function would apply (with perhaps minor modifications within a group) to all plants within the group. Thus the regulatory action for each member of the group should be the same, or very nearly so, as that for the parent member of the group.

Accordingly, those plants will be identified which may be expected to have similar design characteristics to the ten to fourteen plants noted in Sub-Task 3.2 above, and an initial determination will be made of the extent to which grouping of plants for the purposes of the overall Task A-45 program is possible. In this respect, use will also be made of any reliability assessments which have been carried out on parts of the DHRS (e.g., investigation of AFWS reliability, Refs. 11 and 12 and of on-going work such as the investigation of reliability of emergency power systems as part of Task Action Plan A-44, "Station Blackout", Ref. 5). Thus, on completion of this part of Sub-Task 3, it is hoped that each LWR will have been allocated to one or other of ten to fourteen parent groups. Regulatory action will then be based on the characteristics of these parent groups.

SUB-TASK 3.5 - <u>Assess Adequacy of DHRS in Existing Plants on a Deterministic Basis</u>

It is recognized that performing Task A-45 solely on a probabilistic risk assessment (PRA) based approach will not be completely sufficient. As delineated above in Sub-Task 1.3, qualitative acceptance criteria will need to be developed to cover those "Special Emergency" situations which make a contribution to overall risk but are difficult to quantify. In addition, a PRA based approach does not provide a thorough understanding of the time sequence of events, such as steam generator dryout time or time to core uncovery, which are important elements in assessing the operational aspects of alternative SDHR systems by more mechanistic analyses as delineated above in Sub-Task 2.3. Therefore, a parallel approach that utilizes both PRA and deterministic methods will be utilized in this Plan.

Accordingly, in this Sub-Task, more conventional engineering or so-called "deterministic" evaluations of DHR systems will be performed. For certain selected existing plants as determined by the grouping effort described above in Sub-Task 3.3, an assessment of the adequacy of SDHR and RHR systems against the interim qualitative acceptance criteria developed in Sub-Task 1.3 will be made by deterministic evaluations. These evaluations will examine specific plant DHR system specifications, including but not limited to, flow schematics, piping and instrumentation diagrams (P&IDs), plant general arrangement drawings, etc. to determine features such as the adequacy of separation, redundancy, independency, diversity, accessibility of plant for inspection and testing, and freedom from potential common mode faults. Plant walk-throughs will also constitute a vital part of this evaluation. Limited reliability analyses of the major systems required for the DHR function may also be included in this "deterministic" evaluation.

It is anticipated that useful input to this Sub-Task will be provided from the existing Systematic Evaluation Program (SEP). For eleven currently operating plants licensed prior to 1970, the SEP will provide assessments of the extent to which each facility meets current criteria used by the Regulatory staff for licensing new facilities.

SUB-TASK 4. Development of Plan for Implementing New Licensing Requirements for DHR Systems

Besides developing a plan for implementing new licensing requirements for DHR systems, this Sub-Task will include overall project management, technical direction and integration for the entire Task A-45 program, including selection and management of sub-contractors. For organizational purposes, this Sub-Task is further divided into the SDHR and RHR phases as described below.

B.I. Shutdown Decay Heat Removal (SDHR)

When Sub-Tasks 1 through 3 are nearing completion, a plan will be developed for implementation of proposed new licensing requirements, if any, for SDHR systems. This plan may, for example, recommend that the licensees be required to assess their SDHR systems against the acceptance criteria developed in this plan and to develop SDHR system changes needed to meet the criteria. Technical Specification modifications for shutdown decay heat removal will form part of the plan for implementation. The plan will be published in the form of a NUREG report containing recommendations for Rulemaking, Regulatory Guide(s), and/or Standard Review Plans, as appropriate.

The plan of implementation will also include a comprehensive and consistent set of proposed design criteria and requirements for SDHRS in future plants.

B.II. Residual Heat Removal (RHR)

As explained in Part A.II of Section 3 above, the technical work required to develop a plan for implementing proposed new requirements, if any, for the RHR systems is basically the same as that for the SDHR systems and will be organized and managed under the same Sub-Task headings. The information presently available suggests that the main effort will need to be directed towards gaining a better understanding of the variety of ways in which the RHR function could be performed with the systems already installed, and of the suitability and adequacy of reliability of those systems which are required to operate for prolonged periods in a hostile environment or following "Special Emergencies." This additional understanding should then provide a basis for recommending the extent to which any of the various systems that can be used in the RHR role need to be upgraded, in order that the provisions for performing the RHR function can meet the General Design Criteria of 10 CFR 50 and the quantitative and qualitative acceptance criteria derived as part of Task A-45.

C. Management of Work

The responsibility for preparing and implementing a program to resolve this USI is with the Generic Issues Branch (GIB), Division of Safety Technology (DST), Office of Nuclear Reactor Regulation (NRR). A Task Manager in the GIB will provide overall management of all work identified in this Task Action Plan, including outside technical assistance contract work and coordination of all work performed by other divisions and branches, both within NRR and RES. The Task Manager will also provide close coordination with the Advisory Committee on Reactor Safeguards (ACRS). NRR will have the responsibility of taking licensing-related actions on decay heat removal issues, in both the SDHR and the RHR phases, during the conduct of this program.

D. Schedule

The following schedule has been developed for the completion of the major tasks of this program. A more detailed schedule breakdown for all work included in this Plan is provided in Appendix B.

Sub-Task Number	Title	Reporting Date
1	Develop acceptance criteria for assessment of DHRS	Draft August 1982 Final May 1983

2	Develop means for improvement of DHR Function	Draft May 1983 Final April 1984
3	Assessment of adequacy of DHRS in existing LWRs	Draft July 1983 Final April 1984
4	Development of plan for implementing new requirements	Draft April 1984 Final NUREG/CR Report Nov 1984

4. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION OF TASK

The auxiliary feedwater (AFW) system is a very important safety system in a PWR in terms of providing a heat sink via the steam generators to remove core decay heat. The TMI-2 accident and subsequent studies have further highlighted the importance of the AFW systems. As previously indicated, the NRC staff required certain upgrading of the auxiliary feedwater systems for all LWRs following the TMI-2 accident. Although this USI will investigate alternative means of decay heat removal, it is the NRC staff's view that in general (not on a plant-specific basis) if the licensees comply with the upgrading of requirements for the AFW systems, the action taken following the TMI-2 accident justifies continued operation and licensing pending completion of this USI. Further discussion and the bases for this view are provided below for each type of LWR.

A. TMI-2 Accident

The accident at TMI-2 on March 28, 1979 involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system, and subsequent operator intervention to severely reduce flow from the safety injection system. The resulting severity of the ensuing events and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt action to: (a) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2-type events, and (b) investigate the potential generic implications of this action on other operating reactors.

The Bulletins & Orders Task Force (B&OTF) was established within the NRC Office of Nuclear Reactor Regulation (NRR) in early May 1979 and completed its work on December 31, 1979. This task force was responsible for reviewing and directing the TMI-2-related staff activities associated with the NRC Office of Inspection and Enforcement (IE) Bulletins, Commission Orders, and generic evaluations of loss-of-feedwater transients and small-break loss-of-coolant accidents for all operating plants to assure their continued safe operation. Reference 14, NUREG-0645, "Report of the Bulletins and Orders Task Force," summarizes the results of the work performed.

B. Generic and Plant-Specific Studies

For B&W-designed operating reactors, an initial NRC staff study was completed and published in Reference 12, NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company." This study considered the particular design features and operational history of B&W-designed operating plants in light of the TMI-2 accident and related current licensing requirements. As a result of this study, a number of findings and recommendations resulted which are now being pur. d.

Generally, the activities involving the B&W-designed reactors are reflected in the actions specified in the Commission Orders. Consequently, a number of actions have been specified regarding transient and small-break analyses, upgrading of auxiliary feedwater reliability and performance, procedures for operator action, and operator training. The results of the NRC staff review of the B&W small-break analysis is published in Reference 15, NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants."

Similar studies have been completed for operating plants designed by Westinghouse (W), Combustion Engineering (C-E), and General Electric (GE). Those studies, which also focus specifically on the predicted plant performance under different accident scenarios involving feedwater transients and small-break loss-of-coolant accidents, are published in Reference 11, NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants"; Reference 16, NUREG-0635, "General Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant-Accidents in Combustion Engineering-Designed Operating Plants; and Reference 17, NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications."

Based on the review of the operating plants in light of the TMI-2 accident, the NRC staff reached the following conclusions:

(1) The continued operation of the operating plants is acceptable provided that certain actions related to the plants' design and operation, and training of operators identified in Reference 14, NUREG-0645 are implemented consistent with the recommended implementation schedules.

(2) The actions taken by the licensees with operating plants in response to the IE Bulletins (including the actions specified in Reference 18, NUREG-0623, "Generic Assessment of Delayed

Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the B&OTF independently confirmed the safety significance of those related actions recommended by other NRR task forces as discussed in Reference 14, NUREG-0645.

C. Pressurized Water Reactors (PWRs)

The primary method for removal of decay heat from pressurized water reactors is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feed-water or auxiliary feedwater (AFW) systems, and is rejected to either the turbine condenser or the atmosphere via the secondary coolant system safety/relief valves. As previously indicated, following the TMI-2 accident, the importance of the AFW was highlighted and a number of improvements were made to improve the reliability of the AFW (see Reference 14, NUREG-0645, "Report of the Bulletins and Orders Task Force"). It was also required that operating plants be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any AC power source; that is, if both offsite and onsite AC power sources are lost.

As discussed in Reference 19, some pressurized water reactors potentially have at least one alternate means of removing decay heat if an extended loss of feedwater is postulated. This method is known as "feed and bleed" and uses the high pressure injection (HPI) system to add water coolant (feed) at high pressure to the primary system. The decay heat increases the system pressure and energy is removed through the power-operated relief valves (PORV) and/or the safety valves (bleed), if necessary. It should be noted that some PWRs incorporate HPI pumps that cannot operate at full system pressure (cutoff head about 1500 psi). For those cases, the PORVs can be manually opened, thereby reducing the system pressure to within the operating range of the HPI. Limited vendor analyses have shown that the core can be adequately cooled by this means, provided that the operator takes the appropriate action in time and containment pressure can be controlled to a safe level.

At low primary system pressure (below about 200 psi), the long-term decay heat is removed by the residual heat removal system to achieve cold shutdown conditions.

D. Boiling Water Reactors (BWRs)

The principal means for removing decay heat in boiling water reactors while at high pressure is via the steam lines to the turbine condenser. The condensate is normally returned to the reactor vessel by the main feedwater system; however, the steam turbine-driven reactor core isolation cooling (RCIC) system is provided to control primary system inventory, if an abnormal event occurs where AC power is not available. If the condenser is assumed unavailable, energy can be removed via the safety/relief valves to the suppression pool. Also, a high pressure coolant injection (HPCI) system or high pressure coolant spray (HPCS) system is provided on most BWRs as a backup to the RCIC system. These systems can recirculate fluid to the reactor vessel from either the condensate storage tank or the suppression pool.

When the primary system is at low pressure, the decay heat is removed by the residual heat removal (RHR) system. If the RCIC system and HPCI/HPCS systems are unavailable, so that primary system pressure must be reduced, the pressure can be lowered by the automatic depressurization system (ADS) which opens the safety/relief valves and rejects energy to the suppression pool. At low pressure, long-term cooling in the RHR mode is initiated to achieve cold shutdown conditions.

In some earlier BWRs, an RCIC system was not provided. For those cases, an isolation condenser was provided as a passive backup means for removing decay heat while at high system pressure.

E. Conclusion

In summary, because of the upgrading of current decay heat removal systems that was required following the TMI-2 accident, it is concluded that, in general, plants may continue to be licensed and operated before the ultimate resolution of this generic issue without endangering the health and safety of the public. However, licensee compliance with the upgrading of decay heat removal system requirements must be examined by the staff on an individual case basis.

Notwithstanding, this USI will evaluate the benefit of providing alternate means of decay heat removal which could substantially increase the plants' capability to handle a broader spectrum of transients and accidents. The study will include a number of plant-specific DHR systems evaluations and will result in recommendations regarding the desirability of, and possible design requirements for, improvements in existing systems or an alternative decay heat removal method, if the improvements or alternatives can significantly reduce the overall frequency of core melt in a cost effective manner.

5. ASSISTANCE REQUIRED FROM NRR

A. Division of Licensing (DL)

Provides the coordination necessary to expedite the collection of required operating reactor experience and design data. Information needs will be related to shutdown decay heat and to residual heat removal systems' reliability and risk assessments, design characteristics, and plant walk-throughs. DL will provide assistance to the Task Manager for A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to Task A-45 for which DL has responsibility, as identified in Reference 6. DL will assist in coordinating the implementation program for operating reactors and license reviews, including the reviews of requests for information, working closely with the Task Manager in the Generic Issues Branch. DL will also contribute to the formulation, review, and approval of interim and final licensing positions.

Manpower Requirements* -

Operating Reactors Branch No. 1	0.05 my
Operating Reactors Branch No. 2	0.05 my
Operating Reactors Branch No. 3	0.05 my
Operating Reactors Branch No. 4	0.05 my
Systematic Evaluation Program Branch	0.05 my
Operating Reactors Branch No. 5	0.05 my
Licensing Branch No. 1	0.05 my
Licensing Branch No. 2	0.05 my
Licensing Branch No. 3	0.05 my
Standardization and Special Projects Branch	0.05 my

B. Division of Systems Integration (DSI)

Provides review and comment on the technical evaluations provided by the Task Manager in the areas of reactor and auxiliary systems, instrumentation and control, electrical and power systems, containment heat removal, and systems interactions. DSI will provide assistance in the identification of design and operational characteristics of AC power supplies and systems required for decay heat removal. DSI will provide assistance to the Task Manager for A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to Task A-45 for which DSI has responsibility, as identified in Reference 6. In addition, DSI will contribute to the formulation, review, and approval of interim and final licensing positions, including the development of a comprehensive and consistent set of decay heat removal system requirements.

*All the manpower requirements provided below are estimates on an annual basis.

Manpower Requirements -

Reactor Systems Branch	0.25* my
Auxiliary Systems Branch	0.25* my
Instrumentation and Control Systems Branch	0.05 my
Power Systems Branch	0.05 my
Containment Systems Branch	0.05 my

C. Division of Engineering (DE)

Provides review and comment on those technical issues/evaluations provided by the Task Manager involving fire protection, environmental qualification, mechanical/structural integrity, and materials considerations as related to decay heat removal systems. DE will provide assistance to the Task Manager for A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to Task A-45 for which DE has responsibility, as indicated in Reference 6. In addition, DE will contribute to the development of a consistent and comprehensive set of decay heat removal system requirements.

Manpower Requirements -

Chemical Engineering Branch	0.05 my
Equipment Qualification Branch	0.05 my
Mechanical Engineering Branch	0.025 my
Structural Engineering Branch	0.025 my
Materials Engineering Branch	0.025 my

D. Division of Human Factors Safety (DHFS)

Provides review and comment on those technical issues/evaluations involving man/machine interfaces. In this area, DHFS will contribute to the development of a consistent and comprehensive set of decay heat removal system requirements. Any upgrade to existing DHR systems or any new dedicated systems will have to have operator procedure guidelines developed, as part of Sub-Task 2.3, Operational Aspects of Alternate SDHR Systems; and DHFS will have a major role in this activity.

Manpower Requirements -

Human Factors Engineering Branch	0.025 my
Procedures and Test Review Branch	0.025 my

*Reflects RSB and ASB responsibility directly related to reactor and auxiliary systems required for decay heat removal.

E. Division of Safety Technology (DST)

Provides overall management of program to resolve this USI. Provides liaison between NRR and RES and provides coordination of activities performed within NRR which are part of this Task Action Plan. DST has primary responsibility for the review of draft licensing recommendations and for coordination of the internal management and public review process required to adopt final licensing requirements and positions. DST will provide review, comment, and technical support on those issues/evaluations provided by the Task Manager involving reliability and risk assessments and cost/benefit assessments related to decay heat removal systems. DST will provide assistance to the Task Manager for A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to Task A-45 for which DST has responsibility, as indicated in Reference 6. DST will also coordinate the formal revision and publication of licensing documents (i.e., Rules, Regulatory Guides, Standard Review Plans) with the Office of Nuclear Regulatory Research.

Manpower Requirements -

Generic Issues Branch	0.75* my
Reliability and Risk Assessment Branch	0.25* my
Licensing Guidance Branch	0.05 my
Safety Program Evaluation Branch	0.10* my
Research and Standards Coordination Branch	0.025 my

6. ASSISTANCE FROM RES DIVISIONS

Since RES has the lead role on related programs (e.g., RSSMAP, IREP), very close coordination and cooperation will be required on Task A-45 between NRR and RES. RES assistance will be required from the Divisions of Risk Analysis, Accident Evaluation, Engineering Technology, and Facility Operations. The Division of Risk Analysis will provide technical input from their Sandia Laboratory Programs on Alternate Decay Heat Removal Concepts, and Severe Accident Research, technical evaluations relative to reliability and risk assessment for decay heat removal systems, and input from Task A-44**, "Station Blackout," relative to decay heat removal systems. The Division of Accident Evaluation will provide technical input relative to the response of existing and improved shutdown decay heat

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* Reflects GIB overall management rsponsibility, technical support from RRAB in the area of reliability and risk assessments on decay heat removal systems, and cost/benefit evaluations from the SPEB on alternative, dedicated heat removal systems.

** Task A-44 is an Unresolved Safety Issue that is managed by the Generic Issues Branch, DST, but the Task Manager for this task is a member of the Reactor Risk Branch, RES.

removal systems to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support improved decay heat removal systems behavior under transient and accident conditions. The Division of Engineering Technology will provide assistance in the preparation and publication (i.e., Rules, Regulatory Guides, Standard Review Plans) of a consistent and comprehensive set of shutdown decay heat and residual heat removal requirements. The Division of Facility Operations will provide technical input from their Sandia Laboratory Program on Nuclear Power Plant Design Concepts for Sabotage Protection. RES will provide assistance to the Task Manager for A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to Task A-45 for which RES has responsibility, as identified in Reference 6.

Manpower Requirements -

Division of Risk Analysis	0.50 my
Division of Accident Evaluation	0.20 my
Division of Engineering Technology	0.10 my
Division of Facility Operations	0.05 my

7. TECHNICAL ASSISTANCE

Direct technical assistance contract work in support of the program will be required for nearly all sub-tasks. It is anticipated that most of the funding will be provided by the Office of Nuclear Reactor Regulation. Table 1 provides a summary of the total estimated technical assistance program requirements. A brief description of the technical assistance required for this program is also provided below. The scope of work for the individual technical assistance contracts will be developed in more detail as part of implementing the Task Action Plan.

Sub-Task 1

A. Develop Acceptance Criteria for Assessment of DHRS.

1. Contractor - UCLA (through December 1981; after this date, contractor is to be selected)

2. NRC Managing Organization - DST/NRR

3. Scope

   Assist in the development of criteria which can be used to judge the acceptability of the DHRS in existing and future LWRs.

   The criteria for existing and future plants should be based primarily on frequency of core melt due to failure of DHR systems and should also define the reliability (in terms of

Table 1 Summary of Technical Assistance Program Requirements for Task A-45

Sub-Task No.	Sub-Task Title	Contractor	Estimated Effort (MY) FY 81	FY 82	FY 83	FY 84	FY 85	Total	Estimated Costs* ($1000)	Remarks
1	Develop Acceptance Criteria for Assessment of DHRS 1.1 Existing Plants 1.2 Future Plants 1.3 Develop Qualitative Criteria for Special Emergencies	UCLA thru 12/81; after this date, contractor to be selected	0.1 See Remarks	0.5	0.5	--	--	1.1	135	FY81 - Some input available from previous UCLA program, under DSI; supplemented contract by $10K for work on A-45
2	Develop Means for Improvement of DHR Function									
	2.1 Phenomenological Studies	To Be Selected	See Remarks	0.25	1.75	--	--	2	250	FY81 - Input to S/T 2.2 from previous program at Sandia under RES sponsorship.
	2.2 Conceptual Design Studies			0.25	3.0	3.75	--	7	875	
	2.3 Operational Aspects of Alternative SDHR Systems			--	0.75	0.75	--	1.5	187.5	
3	Assess Adequacy of DHRS in Existing LWRs									
	3.2 Assess Adequacy of DHRS in Selected Existing Plants on Probabilistic Basis	To Be Selected	See Remarks	0.2	2.3	--	--	2.5	312.5	FY81 - Some input from previous program at Sandia under RES sponsorship. Obligated $100K at BNL (FINA-3381) to work on S/T 3.3.
	3.3 Group Other Existing Plants for Assessment of Adequacy of DHRS	BNL	0.4	0.4	--	--	--	0.8	100	
	3.5 Assess Adequacy of DHRS on Deterministic Basis	To Be Selected	--	--	3	4	--	7	875	

Table 1 (Continued)

Sub-Task No.	Sub-Task Title	Contractor	Estimated Effort (MY) FY 81	FY 82	FY 83	FY 84	FY 85	Total	Estimated Costs* ($1000)	Remarks
4	Develop Plan for Implementing New Requirements	Sandia Lab.	--	1.2	0.7	2.1	--	4	500	On May 3, 1982, obligated $400K at Sandia (FINA-1309) as the Lead Lab/Program Manager to start work on A-45.
	TOTAL EFFORT (MY)		0.5	2.8	12	10.6	--	25.9		
	TOTAL COST ($1000)		60	350	1500	1325			3235	Obligated $60K in FY 81 and $450K in FY82 for work on A-45.

[a] Based on $125K/MY over a 3 year period to account for labor escalation, travel, and computer costs.

maximum acceptable probability of failure per demand) required from those systems that are part of the the DHR function.

In developing the criteria, account must be taken of:

- The Commission's proposed policy statement on safety goals for nuclear power plants (see Reference 10).
- The estimated risks from existing LWRs derived in WASH-1400.
- The "trial" set of quantitative safety goals suggested to the Commission by the ACRS, Reference 7.
- Provisional quantitative criteria already developed and used by NRC.
- The need to adhere to the ALARA principle.

A set of criteria of a more deterministic nature will be developed for assessing the adequacy of the SDHRS and RHRS in those plants to which the available PRAs and IREP studies cannot be extrapolated with sufficient confidence to include them in one of the "groups" established in Sub-Task 3.3.

It will also be necessary to develop a set of criteria of a more qualitative nature to assess the adequacy of DHRS in "Special Emergency" situations arising from internal and external hazards, such as cable fires, earthquakes, and sabotage, even where an extensive PRA is available for the plant. In addition, the criteria used in certain foreign countries will be considered.

4. Funding Requirements

$135,000

Sub-Task 2

Development of Means for Improvement of DHR Function.

A. Sub-Task 2.1. Phenomenological Studies

1. Contractor - To be selected.

2. NRC Managing Organization - DST/NRR

3. Scope

A review is required of the phenomenological aspects of possible means for improvement of SDHRS, including the following:

Part I - PWR

(a) Effect of limited boiling in the core (when shut down) on RCS pumps and on natural circulation, including possibility of "vapor locking" in steam generator tubes;

(b) Effect of secondary coolant level and temperature on natural circulation;

(c) Minimum "feed and bleed" rates necessary to maintain safe conditions in the core:

(i) with sub-cooled conditions,
(ii) with limited boiling;

(d) Effect of excessive bleed rate on conditions in the core and thermal stresses in primary circuit;

(e) Rate of rise of containment pressure and temperature in a "feed and bleed" regime without containment cooling;

(f) Effect of prolonged "bleed" without "feed";

(g) Feasibility of maintaining safe conditions in the core by use of the steam generators as "reflux condensers," after loss of a substantial fraction of the primary coolant inventory; as part of this study, the effect of noncondensible gases should be considered;

(h) Effect of departure from sub-cooled conditions on feasibility of using a high pressure version of the conventional "Residual Heat Removal System;"

(i) Identification of fault conditions in which rapid blow-down of secondary circuit would be advantageous; and

(j) Review possible methods for defining the system parameters for the coolant chemical and volume control system.

In each case, the effect of failure or maloperation of the pressurizer should be considered.

Part II - BWR

(a) Feasibility of maintaining safe conditions in the core by a reflux condenser external to the reactor vessel, after losing a substantial fraction of the primary coolant inventory;

(b) Thermal-hydraulic parameters of an isolation condenser for a 1000 MW(e) BWR; and

(c) Thermal-hydraulic aspects of any novel methods for SDHR identified in foreign BWR designs as part of Sub-Task 2.2.

Part I of the review should include consideration of the latest available experimental data on two phase flow in reactor accident conditions (e.g., "LOFT" small break tests). The effect of more rapid and more complete separation of steam and water phases in some two-phase flow situations should be considered in relation to the problems enumerated above and to any previous analytical solutions. Where appropriate, additional test work on a laboratory scale or on existing test facilities should be defined.

It will be necessary to consider separate plant configurations which are typical of those employed by each of the U.S. LWR vendors.

4. Funding Requirements

$250,000

B. Sub-Task 2.2. Conceptual Design Studies

1. Contractors - To be selected

2. NRC Managing Organization - DST/NRR

3. Scope

The objectives of this technical assistance program are as follows:

(1) Preliminary development of new concepts for improvement of SDHRS in LWRs based on maximum utilization of existing equipment.

(2) Further development of the most promising concepts evolved in Section (1) of this scope, including associated value/impact (or cost/benefit evaluations).

(3) Further development of the more promising schemes for improvement of SDHRS in existing LWRs evolved in the RES sponsored Sandia program on alternate DHR concepts, taking account of the results of the thermal/hydraulics work identified in Sub-Task 2.1.

(4) Develop conceptual designs for separate, dedicated DHRS capable of functioning in "Special Emergency" situations, for existing plants, to a point at which feasibility is established and the cost of such dedicated systems can be estimated. Use will be made of previous work in this area,

such as that reported in References 4 and 13, including the alternative DHR systems utilized in some foreign LWRs.

(5) Perform value/impact evaluations to determine to what extent existing plants should have the capability to achieve and maintain cold shutdown using safety grade equipment. Inlude an assessment of the adequacy of reliability and performance criteria and standards for systems that are required to achieve and maintain cold shutdown conditions.

4. Funding Requirements

$875,000

C. Sub-Task 2.3. Operational Aspects of Alternative SDHR Systems

1. Contractor - To be selected

2. NRC Managing Organization - DST/NRR

3. Scope

The objectives of this technical assistance program are as follows:

Part I - Time Available for Operator Action (PWR and BWR)

(i) Review available data relating to the time scale of events for each of the means of SDHR examined in Sub-Task 2.2 and determine those areas, if any, where additional information is necessary.

(ii) Formulate a program of work to provide the additional information identified as necessary in (i) above, for consideration by NRC.

(iii) Evaluate the time available to the operator for a set of cases which will be defined by NRC.

Part II - Definition of Outlines of Operating Procedures (PWR and BWR)

(i) Review the available information relating to operating procedures for decay heat removal in LWRs, including an estimate of the probability of error on the part of the operator.

(ii) Define, in outline form, the operating procedures required for alternative means of SDHR and assess the probability of operator error.

(iii) Confirm that the instrumentation and controls required for the most effective operating procedures have been or are scheduled to be, installed on existing plants.

4. Funding Requirements

$187,500

Sub-Task 3

A. Assessment of Adequacy of DHR Systems in Existing LWRs

1. Contractors - To be selected

2. NRC Managing Organization - DST/NRR

3. Scope

This sub-task consists of the following parts (As previously indicated, Sub-Tasks 3.1 and 3.4 were deleted from a previous version of the Plan, and Sub-Task number designations 3.2, 3.3 and 3.5 were maintained to avoid confusion):

Sub-Task 3.2	Assess Adequacy of DHRS in Selected Existing Plants on a Probabilistic Basis
Sub-Task 3.3	Group Other Existing Plants for Assessment of Adequacy of DHRS
Sub-Task 3.5	Assess Adequacy of DHRS in Existing Plants on a Deterministic Basis

For contractual purposes, the various Sub-Tasks may be divided up, but this determination has not been made yet.

The objectives of this technical assistance program are as follows:

(1) Obtain an assessment of the contribution to risk and/or core melt probability from decay heat removal system (DHRS) failures for the specific plants analyzed in the Reactor Safety Study (WASH-1400), the Reactor Safety Study Methodology Application Program (RSSMAP), and the Interim Reliability Evaluation Program (IREP), and compare with the acceptance criteria (see Section 3.B, Sub-Task 3.2 for more detail).

(2) For the specific plants analyzed in WASH-1400, RSSMAP, and IREP, the contractor shall make a determination whether it is feasible to classify all other currently operating U.S. commercial LWRs into groups based on whether they have similar DHRS design characteristics to

the plants covered in item (1) (see Section 3.B, Sub-Task 3.3 for more detail).

(3) For the groups of plants defined under item (2) above, the contractor shall assess their DHR systems to determine whether they meet the probabilistic acceptance criteria developed under Sub-Task 1.1.

(4) As it seems unlikely that it will be possible to assess the adequacy of DHR systems for all existing plants solely by a risk or reliability based approach, the contractor shall assess the adequacy of SDHR and RHR systems in certain selected existing plants by deterministic methods. The criteria developed in Sub-Task 1.3 will be used for this purpose, possibly including the design criteria as utilized in certain foreign countries.

4. Funding Requirements

$1,287,500.

Sub-Task 4

A. Development of Plan for Implementing New Licensing Requirements

1. Contractor: Sandia Laboratories

2. NRC Managing Organization - DST/NRR

3. Scope

The contractor for this Sub-Task will provide overall project management, technical direction and integration for the entire Task A-45 program, including selection and management of sub-contractors.

When Sub-Tasks 1 through 3 are nearing completion, the contractor will assist the staff in developing a detailed plan for implementing any proposed new licensing requirements stemming from Task A-45. This plan will define in detail what assessments the licensees should perform to determine whether their DHR systems meet the acceptance criteria, and what action they should take in terms of developing proposed changes if their DHR systems do not meet the acceptance criteria. The plan of implementation will also include a comprehensive and consistent set of proposed design requirements for DHR systems, including Technical Specification modifications (see Section 3.B, Sub-Task 4, for more detail).

4. Funding Requirements

$500,000

8. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

Interaction with outside organizations, including establishing a peer review group, could include AIF, EPRI, NSAC, INPO, FERC, FAA, utilities, NSSS vendors, A&Es, and foreign development agencies, regulators, and manufacturers of nuclear power stations.

Peer review will also be conducted through periodic ACRS briefings and issue of draft documents for public comment.

9. POTENTIAL PROBLEMS

The potential problem areas which have been identified are outlined below:

A. Obtaining sufficient NRR manpower to work on Task A-45.

B. Annual program funding must be approved and obtained.

C. Development of appropriate reliability or quantitative goals for Task A-45 and translation of probabilistic results into licensing requirements.

D. Obtaining necessary design information and operating experience on DHR systems, including the most current information resulting from post-TMI changes.

E. Uncertainty in the quality of information that will be available from ongoing and planned reliability and risk assessments, on what schedule, and the extent to which the information can be extrapolated to all operating plants.

F. The number of plants that need to be assessed may be significantly greater than the plants that will have a risk or reliability study performed.

Each of the above potential problem areas could delay the program.

10. REFERENCES

1. USAEC. "Reactor Safety Study," WASH-1400 NUREG 75/012, October 1975.

2. NRC, Office of Standards Development, "Proposed Revision 1 to Regulatory Guide 1.139. "Guidance for Residual Heat Removal from Hot Shutdown Conditions to Achieve and Maintain Cold Shutdown," (to be published for public comment).

3. BIRKHOFER. "The German Risk Study for NPP," IAEA Bulletin 22 No. 5/6, October 1980.

4. BERRY, D.L. and BENNETT, P.R. "Study of Alternate Decay Heat Removal Concepts for LWRs - Current Systems and Proposed Options," SANDIA Report SAND 80-0929, NUREG/CR-1556, April 1981.

5. RES/NRC, "Task Action Plan A-44, Station Blackout." July 1980.

6. Memorandum, A. R. Marchese to T. E. Murley, "Activities Related to Task A-45, Shutdown Decay Heat Removal Requirements," April 8, 1981.

7. ACRS "An Approach to Quantitative Safety Goals for Nuclear Power Plants," NUREG-0739, October 1980.

8. NRC, "Evaluation of Explosions Postulated to Occur on Transporation Routes near NPP," Regulatory Guide 1.91, Revision 1, 1978.

9. Memorandum, R. M. Bernero, RES/PAS to R. J. Mattson, NRR/DST, "Interim Quantitative Action Criteria," July 29, 1980.

10. NRC, Office of Policy Evaluation, "Safety Goals for Nuclear Power Plants, Discussion Paper," NUREG-0880, February 1982.

11. NRC, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, January 1980.

12. NRC, "Staff Report on the Generic Assessment of Feed-Water Transients in PWRs Designed by B&W Co.," NUREG-0560, May 1979.

13. EBERSOLE, J. C. and OKRENT, D. "An Integrated Safe Shut-Down Heat Removal System for LWR," UCLA Eng-7651, May 1976.

14. NRC, NUREG-0645, "Report of the Bulletins and Orders Task Force," January 1980.

15. NRC, NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants," January 1980.

16. NRC, NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant-Accidents in Combustion Engineering-Design Operating Plants," January 1980.

17. NRC, NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Design Operating Plants and Near-Term Operating License Applications," February 1980.

18. NRC, NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979.

19. Memorandum, B. W. Sheron to K. Kniel, "Status of Feed and Bleed for Emergency Decay Heat Removal," March 31, 1981.

APPENDIX A

LIST OF SYSTEMS* RELEVANT TO DECAY HEAT REMOVAL (DHR)

I. PWR

A. Systems Relevant to Frequency of Demand for DHR

Spurious operation or failures in the following systems result in reactor scram, or shutdown and a demand for DHR:

1. Reactor Scram System (Spurious Operation)
2. Main Feed System
3. Power Generation System (Turbo/Generator, Condensate System)
4. Reactor Coolant System
5. Off-Site Electrical Systems

B. Shutdown Heat Removal Phase (i.e., Reactor Scram to Hot Shutdown)

(i) Principal systems contributing to SDHR function:

1. Auxiliary Feedwater System
2. High Pressure Injection System
3. Low Pressure Injection System
4. High Pressure Recirculation System
5. Low Pressure Recirculation System
6. Containment Spray Injection System
7. Containment Spray Recirculation System
8. Containment Heat Removal System
9. Chemical and Volume Control System
10. Containment Isolation System and Sodium Hydroxide Addition System

(ii) Support systems for the above "functional systems":

1. Electrical Power (AC and DC)
2. Steam and Compressed Air Supplies
3. Control and Instrumentation Systems (including primary and secondary coolant blow-down systems, reactor vessel level control instrumentation and primary coolant circuit venting systems)
4. Lubrication and Cooling Systems for the "Functional" Systems
5. Suction sources of water (e.g., CST, RWST)

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*Not all of these systems will be covered in detail in Task A-45.

C. Residual Heat Removal Phase (i.e., hot shutdown to cold shutdown and thereafter)

(i) Principal Systems Contributing to RHR Function:

1. Residual Heat Removal Systems
2. Low Pressure Recirculation System

(ii) Support Systems, as in I.B(ii)

II. BWR

A. Systems Relevant to Frequency of Demand for DHR

Spurious operation or failures in the following sytems result in reactor scram, or shutdown and a demand for DHR:

1. Reactor Scram System (Spurious Operation)
2. Main Feed System
3. Power Generation System (Turbo/Generator, Condensate System)
4. Reactor Coolant Recirculation System
5. Off-Site Electrical System

B. Shutdown Heat Removal Phase (i.e., Reactor Scram to Hot Shutdown)

(i) Principal systems contributing to SDHR function:

1. Vapor Suppression System
2. High Pressure Coolant Injection System
3. Low Pressure Coolant Injection System
4. Core Spray Injection System
5. Reactor Core Isolation Cooling System
6. Low Pressure Coolant Recirculation System
7. Core Spray Recirculation System
8. High Pressure Service Water System
9. Secondary Containment System
10. Isolation Condenser System

(ii) Support systems for the above "functional" systems:

1. Electrical Power (AC and DC)
2. Steam and Compressed Air Supplies
3. Control and Instrumentation Systems (including automatic primary containment vacuum breaker system and coolant depressurization systems)
4. Lubrication and Cooling Systems for the "Functional" Systems
5. Suction sources of water (e.g., suppression pool)

C. Residual Heat Removal Phase (i.e., Hot Shutdown to Cold Shutdown and thereafter)

(i) Principal Systems Contributing to RHR Function:

1. Residual Heat Removal System
2. Core Spray System
3. High Pressure Service Water System

(ii) Support Systems (as in II.B(ii))

APPENDIX B

DETAILED SCHEDULAR BREAKDOWNS FOR TASK A-45 WORK

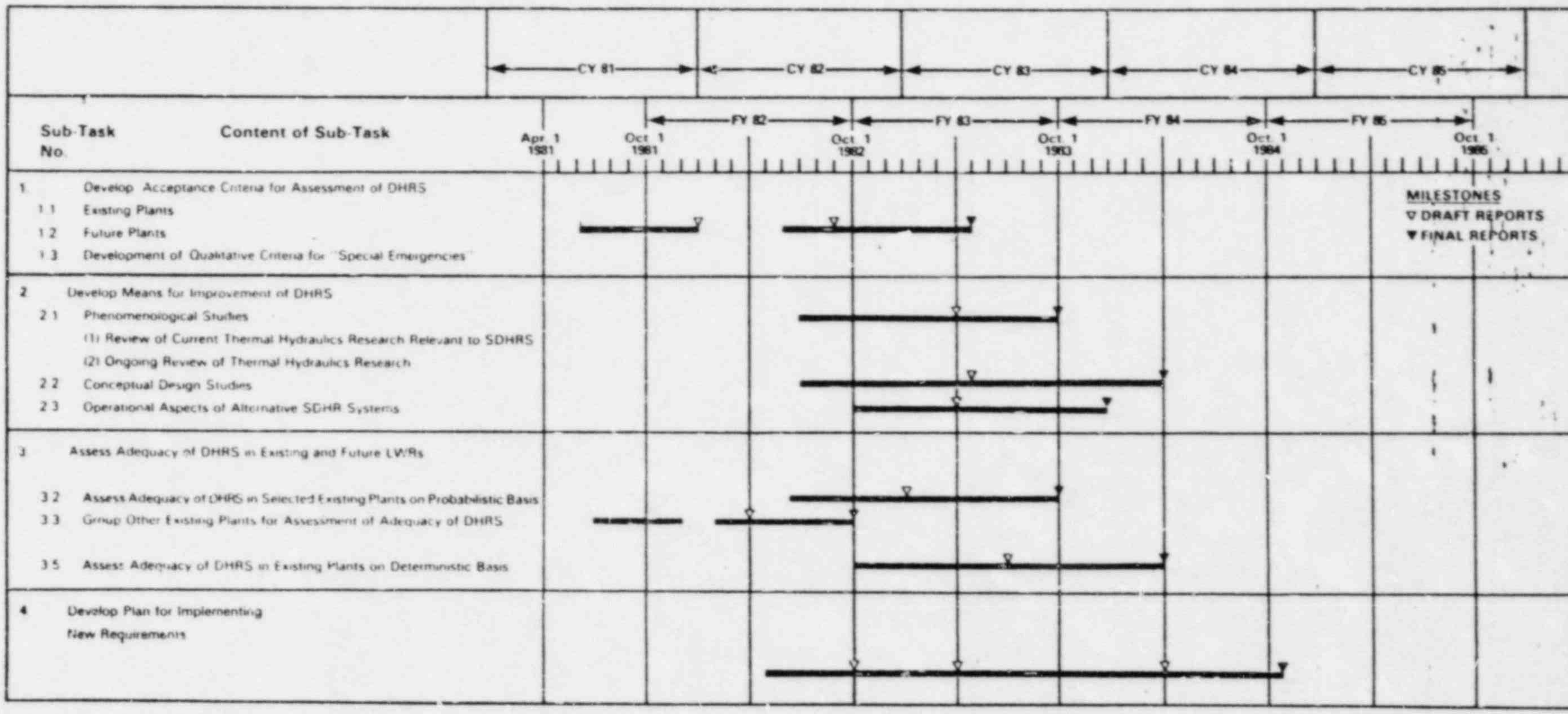
Table B-1

DETAILED SCHEDULE FOR TASK A-45
"SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"

Sub-Task No.	Content of Sub-Task	Reporting Date	
1	Develop Acceptance Criteria for Assessment of DHRS		
1.1	Existing Plants		
1.2	Future Plants	Draft	Aug 82
1.3	Development of Qualitative Criteria for "Special Emergencies"	Final	May 83
2	Develop Means for Improvement of DHR Function		
2.1	Phenomenological Studies		
	(1) Review of Current Thermal-Hydraulics Research Relevant to SDHRS	Draft on (1)	Dec 82
	(2) On-going Review of Thermal-Hydraulics Research	Draft on (2)	Apr 83
		Final on (1&2)	Oct 83
2.2	Conceptual Design Studies	Draft	May 83
		Final	Apr 84
2.3	Operational Aspects of Alternative SDHR Systems	Draft	Apr 83
		Final	Jan 84
3	Assess Adequacy of DHRS in Existing LWRs		
3.2	Assess Adequacy of DHRS in Selected Existing Plants on Probabilistic Basis	Draft	Jan 83
		Final	Oct 83
3.3	Group Other Existing Plants for Assessment of Adequacy of DHRS	Draft	Apr 82
		Final	Sept 82
3.5	Assess Adequacy of DHRS in Existing Plants on Deterministic Basis	Draft	July 83
		Final	Apr 84
4	Develop Plan for Implementing New Requirements (e.g., Prepare NUREG, Reg. Guide)	Outline of Plan	Oct 82
		1st Draft	Apr 83
		2nd Draft	Apr 84
		Final NUREG/CR Report	Nov 84

Figure B-1

DETAILED SCHEDULE FOR TASK A-45,
"SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS"



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. Oxfurth, 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

PDR

~~8206710120~~

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 pu[illegible] of a NUREG report recommending a Regulatory Position [illegible] PTS including appropriate limits (if any) that must b[illegible] 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 gain from these short-term reviews will be utilized as appropriate (for exam 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 identifica (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 plac or revised as needed to facilitate the operator's task in maintaining safety, along with appropriate operator training in those procedures a their underlying technical basis. Generic guidelines for updated procedures will be completed by mid-1982. Plants that require immedia corrective action can have plant-specific procedures in place, and all training regarding those procedures complete by the end of 1982 if req to deal specifically with PTS events.

In addition, a task force has been formed to audit procedures that dea with potential PTS events, and to audit operator training regarding th procedures and regarding PTS phenomena. These audits will be complete for the eight selected plants by June 1982. A second task force has been formed to accelerate consideration of methods that could signific 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 applicabili[ty]. Ultimately it is hoped that the methods can be extended beyond the presentl[y] 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 effect[s] 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 ac..ions 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 sched 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 transien code development and verification; the Instrumentation and Control Systems Branch (1/6, 1/6) for direction of control system studies; Core Performance Br

(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.

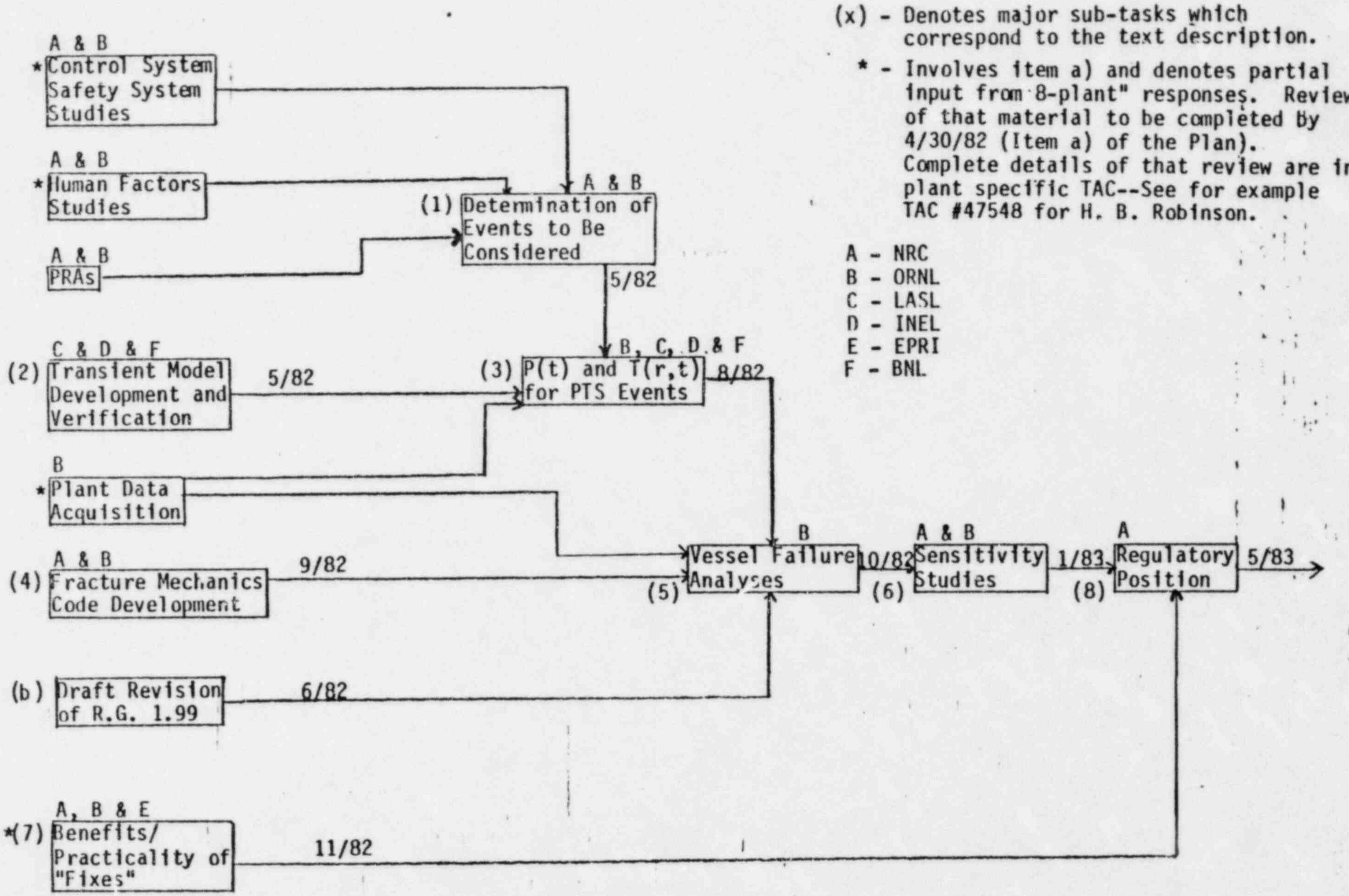
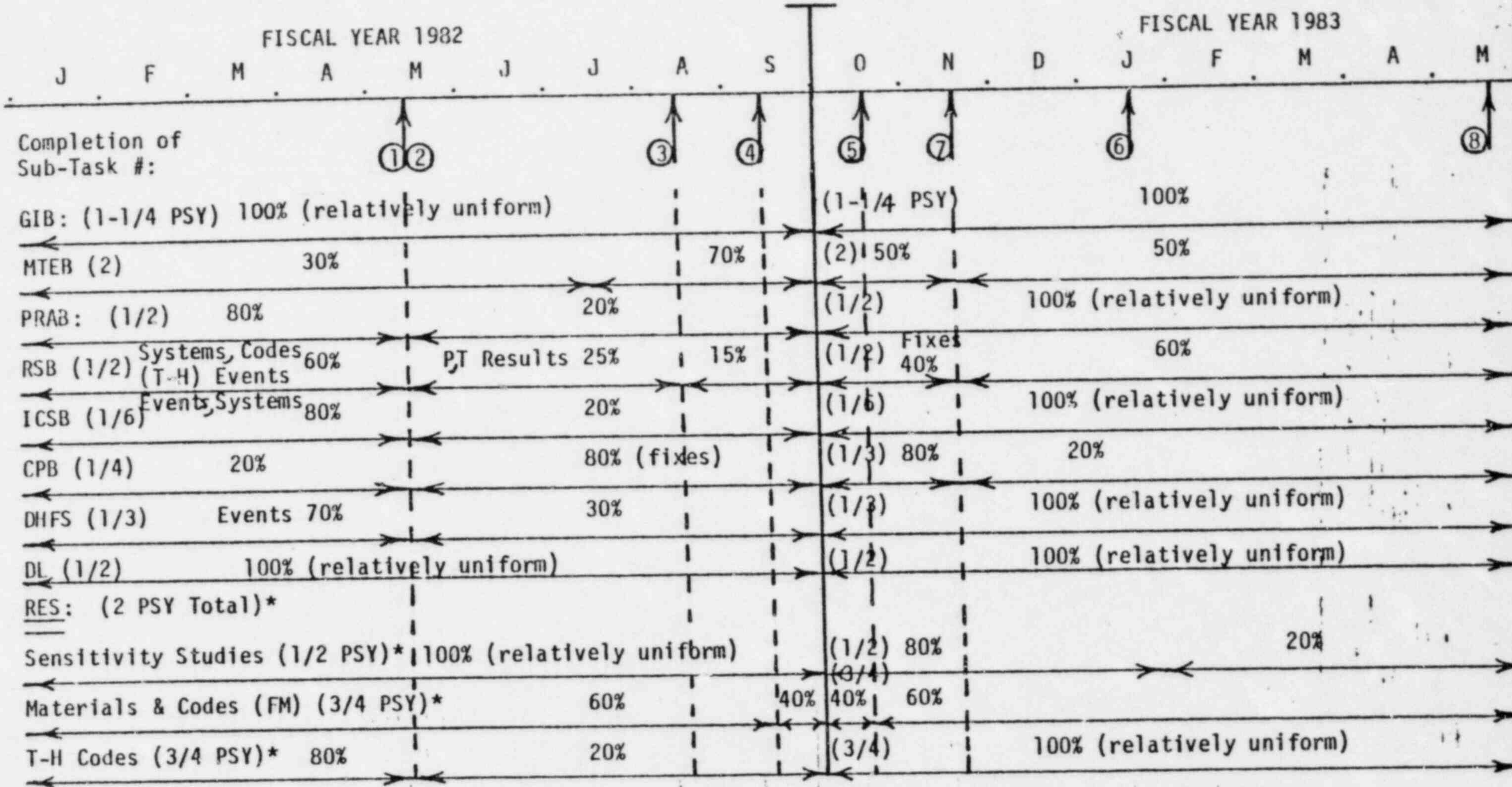


FIGURE 1

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 (See Section 2.A.a and 2.B.a above)	DST/GIB DST/RRAB DL/ORB DL/ORB	0.50 0.25 0.25 0.17	0 0 0 0
	DSI/RSB DSI/CPB DHFS/PTRB DE/MTEB RES/MEB	0.25 0.17 0.17 0.75 0.42	0 0 0 0 0
TOTAL SHORT TERM	NRR RES	2.51 0.42	0 0
LONG TERM PROGRAM			
(Reference Draft TAP for A-49)	DST/GIB DST/RRAB DL/ORB DSI/CPB DSI/RSB DSI/ICSB	1.25 0.50 0.50 0.25 0.50 0.50	1.25 0.50 0.50 0.33 0.50 0.50
	DHFS/PTRB DE/MTEB RES/MEB RES/Johnson RES/Shotkin	0.33 2. 1 0.5 1	0.33 2. 1 0.5 1
TOTAL LONG TERM	NRR RES	4.58 3.5	5.60 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

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	800	700
A6047	Code Assessment and Applications	Shotkin	INEL		
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.

December 8, 1981

SECY 81-687

POLICY ISSUE
(Notation Vote)

For: The Commissioners

From: William J. Dircks
Executive Director for Operations

Subject: DESIGNATION OF PRESSURIZED THERMAL SHOCK AS AN UNRESOLVED SAFETY ISSUE

Purpose: To obtain Commission approval for designation of the pressurized thermal shock safety concern as an Unresolved Safety Issue.

Discussion: The issue of pressure vessel thermal shock has been considered for many years in the context of assuring integrity when cold emergency core cooling water is injected into the reactor vessel following a large loss of coolant accident. A series of thermal shock experiments was conducted at Oak Ridge National Laboratory starting in 1976. Utilizing the results of these experiments with an unpressurized vessel along with fracture mechanics analyses which supported the experiments, it was confirmed that a postulated flaw would not propagate through the reactor vessel wall during a large LOCA. Therefore, it was concluded that vessel integrity would be maintained during reflooding with cold water at relatively low pressure following the large LOCA. However, repressurization following a LOCA was not considered in this early work.

As a result of operating experience including the overcooling transient that occurred at the Rancho Seco nuclear plant on March 20, 1978, it was recognized that it was necessary to consider severe overcooling transients in pressurized water reactors followed by repressurization of the primary system. In these

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pressurized thermal shock transients, reactor vessels would be subjected to pressure stresses superimposed on thermal stresses resulting from the overcooling transient.

Vessel failure could result from a pressurized overcooling event only if several conditions exist simultaneously.

(1) The reactor vessel steel (particularly the weld material) has suffered "embrittlement," or more correctly, loss of fracture toughness, due to neutron irradiation. The rate at which the loss of toughness occurs is dependent on the specific material properties (e.g., the copper content of weld materials) and weld locations and is thus quite plant-specific. A significant increase in the Reference Temperature for the Nil Ductility Transition, RT_{NDT}, is one specific numerical indication of the more general phenomenon of loss of fracture toughness, and the whole phenomenon is often simplistically stated in those terms, i.e., "significantly elevated RT_{NDT}."

(2) An overcooling transient occurs which cools the inner surface of the vessel to a temperature where the steel has inadequate fracture toughness (often simplistically stated as "cooled to a temperature below the RT_{NDT}").

(3) The overcooling persists until a steep temperature gradient exists through a large fraction of the pressure vessel's thickness (a quick cooling of the inner wall without time for the heat removal to reach a significant depth will not create the necessary conditions).

(4) A flaw of a certain critical size range is present at the location of high thermal stress and at the same area of the vessel where the fracture toughness has been reduced by neutron irradiation.

(5) The vessel is repressurized to (or has remained at) a significant fraction of its operating pressure, either during or after the steep temperature gradient exists across the vessel's thickness.

Preliminary efforts to define what conditions would be necessary to propagate a flaw through the entire vessel thickness, thus potentially failing the vessel, were initiated in early 1980. These included: (1) definition of transients and accidents that could result in overcooling with subsequent pressurizatio and their probability of occurrence; (2) development of analytical techniques to perform pressurized overcooling transient and fracture mechanics analysis; (3) a survey of operating plant reactor vessel material properties at present integrated radiation fluence levels; and (4) planning for conducting pressurized thermal shock experiments as part of the Heavy Section Steel Technology (HSST) program at Oak Ridge National Laboratory (ORNL).

Beginning in early 1981, the staff initiated an intensive investigation of the pressurized thermal shock issue which has involved meetings with PWR Owners Groups, reactor vendors, and the ACRS, and letters requesting further plant-specific information from eight licensees. The Commission has been briefed on several occasions regarding the issue and the status and planning of staff activities, most recently on November 24, 1981 (see SECY-81-286, 81-286A, and the October 30, 1981 memorandum from W. J. Dircks to the Commissioners regarding the ORNL report concerning PTS at Oconee).

In view of the importance of this issue, and the significant commitment of staff and contractor resources involved within NRR and RES, the staff has concluded that consideration should be given to designating this issue as an Unresolved Safety Issue (USI) under Section 210 of the Energy Reorganization Act of 1974, as amended in December 1977. The staff has applied the screening and selection criteria approved by the Commission in connection with the designation of four new USIs on December 24, 1980 to the issue of pressurized thermal shock (see memorandum from Samuel J. Chilk to William J. Dircks regarding SECY-80-325, Special Report to Congress Identifying Unresolved Safety Issues, December 24, 1980; and memorandum from Edward J. Hanrahan to Chairman Ahearne regarding Screening/Selection Criteria for Unresolved Safety Issues, November 25, 1980.) The results are presented in Enclosure A.

Recommendation: That the Commission:

1. Approve designation of "Pressurized Thermal Shock" as Unresolved Safety Issue A-49.

2. Note

   a. That a detailed Task Action Plan is being developed to define the scope, resources and schedules for resolution of this issue. A preliminary discussion of the proposed scope is provided in Enclosure B.

   b. That the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs, the Subcommittee on Energy Conservation and Power of the House Committee on Energy and Commerce, the Subcommittee on Environment, Energy and Natural Resources of the House Committee on Government Operations, and the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works will be informed.

William J. Dircks
Executive Director for Operations

Enclosures:
A. Application of USI Screening Criteria
B. Plan for Problem Resolution

Commissioners' comments should be provided directly to the Office of the Secretary by c.o.b. Monday, December 28, 1981.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT December 18, 1981, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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ENCLOSURE A

Enclosure A

Application of USI Screening Criteria to
the Issue of Pressurized Thermal Shock

The process for selection of Unresolved Safety Issues involves applying the screening criteria presented in NUREG-0705 and SECY-80-325 and is restated below.

Initial Screening Criteria

An issue or recommendation should be screened out from further consideration for designation as an Unresolved Safety Issue if it meets one or more of the following:

1. The issue or recommendation is not related to nuclear power plant safety, e.g., transportation of radioactive materials.

2. A staff position on the issue or recommendation has been developed or could be developed within six months. The purpose of this criterion is to eliminate those issues that are near resolution and, therefore, do not constitute truly "unresolved" issues. Such issues do not warrant the attention and resources normally associated with an Unresolved Safety Issue.

3. The issue is not generic.

4. The issue or recommendation is only indirectly related to nuclear power plant safety, e.g., recommended changes in the licensing process, NRC organization, etc.

5. Definition of the issue requires long term confirmatory or exploratory research. The basis for this criterion is that investigative studies of matters for which no clearly defined safety deficiency or improvement has been identified, although an appropriate regulatory activity, do not warrant designation as Unresolved Safety Issues.

6. The issue or recommendation is related to one already being addressed as a USI and can reasonably be or already is included in the current program.

7. The issue or recommendation requires a policy decision rather than a technical solution. The purpose of this criterion is to eliminate those issues that only require a management decision and do not represent potential deficiencies in existing safety requirements that require development of a resolution. In some cases, the results of these policy decisions may require designation of new Unresolved Safety Issues.

In addition to the criteria above, additional criteria were suggested by E. J. Hanrahan in a memorandum to Chairman Ahearne dated November 25, 1980. These criteria were adopted by the Commission in a memorandum from Samuel J. Chilk to William J. Dircks dated December 24, 1980 and are stated as follows:

An issue or recommendation should be screened out from further consideration as a USI if it meets one or more of the initial screening criteria listed in SECY-80-325 or one of the additional criteria listed below:

1. The issue is related to safety improvements where existing protection is adequate.

2. The issue includes progammatic matters involving implementation of issue resolutions already achieved.

3. The issue includes collections of related issues in lieu of focused critical issues. (In this regard, an attempt should be made to define the issue so that matters extraneous to the issue are eliminated.)

Application of the above initial screening criteria should result in identification of sharply focused issues where the basic adequacy of the technical basis for existing safety requirements is in doubt and new knowledge must be developed to resolve that doubt.

The issue of pressurized thermal shock was not eliminated from further consideration by applying the seven initial screening criteria or the three additional criteria suggested by Mr. Hanrahan.

Selection Criteria Proposed by E. J. Hanrahan and Adopted by the Commission

Issues which pass the initial screening criteria and are then presented to the Commission for consideration as USIs should address the following questions:

1. What is the known and/or potential deficiency in the technical basis of existing staff guides/requirements?

2. What present safety requirements appear to be inadequate or in doubt?

3. What new knowledge must be developed to either confirm the adequacy of the technical bases which support existing requirements or to define new requirements that would restore adequate protection?

4. What actions are being taken on operating reactors pending development of new knowledge necessary to resolve the issue?

A discussion of these questions follows:

1. What is the known and/or potential deficiency in the technical basis of existing staff guides/requirements?

   Response: (1) No Design Basis Accident considered during plant design resulted in adequate consideration of the pressurized thermal shock issue; (2) Present review methods based on "single failure of a safety system" may not be adequate to encompass proper review of the pressurized thermal shock issue (the Rancho Seco event involved several errors/failures); (3) Operating instructions and plant safety and control system designs do not adequately address the need to limit repressurization following an accident. A small LOCA or a MSLB with introduction of cold HPI water has the potential for causing growth of a pre-existing small crack in the RPV by a thermal shock mechanism; thus, subsequent repressurization at low RPV metal temperatures (near NDT) could complete the fracture and fail the vessel which could uncover the core and result in core melt.

2. What present safety requirements appear to be inadequate or in doubt?

   Response: Nuclear plant safety is dependent upon the assumption that the RPV integrity will be maintained under all conditions (including all transients and accidents). However, repressurization or concurrent pressure after a thermal cooling transient of sufficient severity could rupture a vessel which has accumulated extensive neutron exposure. Present regulatory requirements do not provide adequate assurance that RPV integrity will be maintained for full plant lifetime.

3. What new information must be developed to either confirm the adequacy of the current technical bases or to define new requirements that would restore adequate protection?

   Response: More precise information and analyses are needed in two basic areas: fracture mechanics and transients/accidents.

   Fracture mechanics information and analyses need to be developed to obtain a more realistic understanding of what conditions the pressure vessel can be expected to survive. The probability that a critical size crack exists at a critical location in a pressure vessel, temperature and fluence dependence of material properties for various weld materials, effects of stainless steel cladding, and effects of warm pre-stressing, are areas where a more definitive understanding is needed.

Transients and accident scenarios must be developed to assess what challenges to the pressure vessel can realistically be expected, and with what probability. This is a major effort, involving control and safety system failure modes, understanding of thermal hydraulic mixing inside the vessels, operating experience assessment, and human factors considerations. Complete resolution of this issue will not be achieved until it is realistically understood what sequences must be postulated, what subsequent failures must be assumed, and what credit/penalty must be assumed for operator behavior including his instructions and his available instrumentation after failures have occurred.

4. What actions are being taken (if any) on operating plants pending resolution of the issue?

Response: The matter of instructions to operating plants currently is a part of the continuing dialogue between the NRC and NSSS vendors under NUREG-0737, Item II.K.2.13. Operators at all operating plants have been alerted to the potential for severe overcooling during certain transients and the importance of ensuring the primary system is not repressurized. In addition, studies are underway by EPRI, the PWR Owners Groups, the eight licensees that received the NRC August 21 requests, and the NRC staff. These near-term activities may result in further specific requirements for operating plants pending complete resolution of the issue.

ENCLOSURE B

Enclosure B

Plan for Problem Resolution

An outline of the proposed integrated program to be conducted by NRR and RES utilizing the National Laboratories, with input from licensees is shown in Figure 1. It consists of the following main elements.

(1) Selection of PTS events 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 events (using the selected and verified codes).

(4) Development of a state-of-the-art fracture mechanics code to predict crack initiation and arrest for given pressure-temperature histories at critical welds or base material.

(5) Calculation of failure potential vs. irradiation embrittlement (i.e., neutron fluence from the operating history) of the pressure vessel at the three lead plants for the selected PTS events.

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

(7) Development of an understanding regarding feasibility and benefits to be derived from various proposed corrective actions, including

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 documenting the regulatory position regarding PTS including appropriate limits that must be observed at specific classes of plants (if any), and potential corrective actions.

Each of these items constitutes a major sub-task. More details of each sub-task will be given in the Task Action Plan. Many of the sub-tasks are planned to proceed concurrently, but some must be sequential. The accompanying figure is provided to show an overview of the sub-tasks, including their relationship.

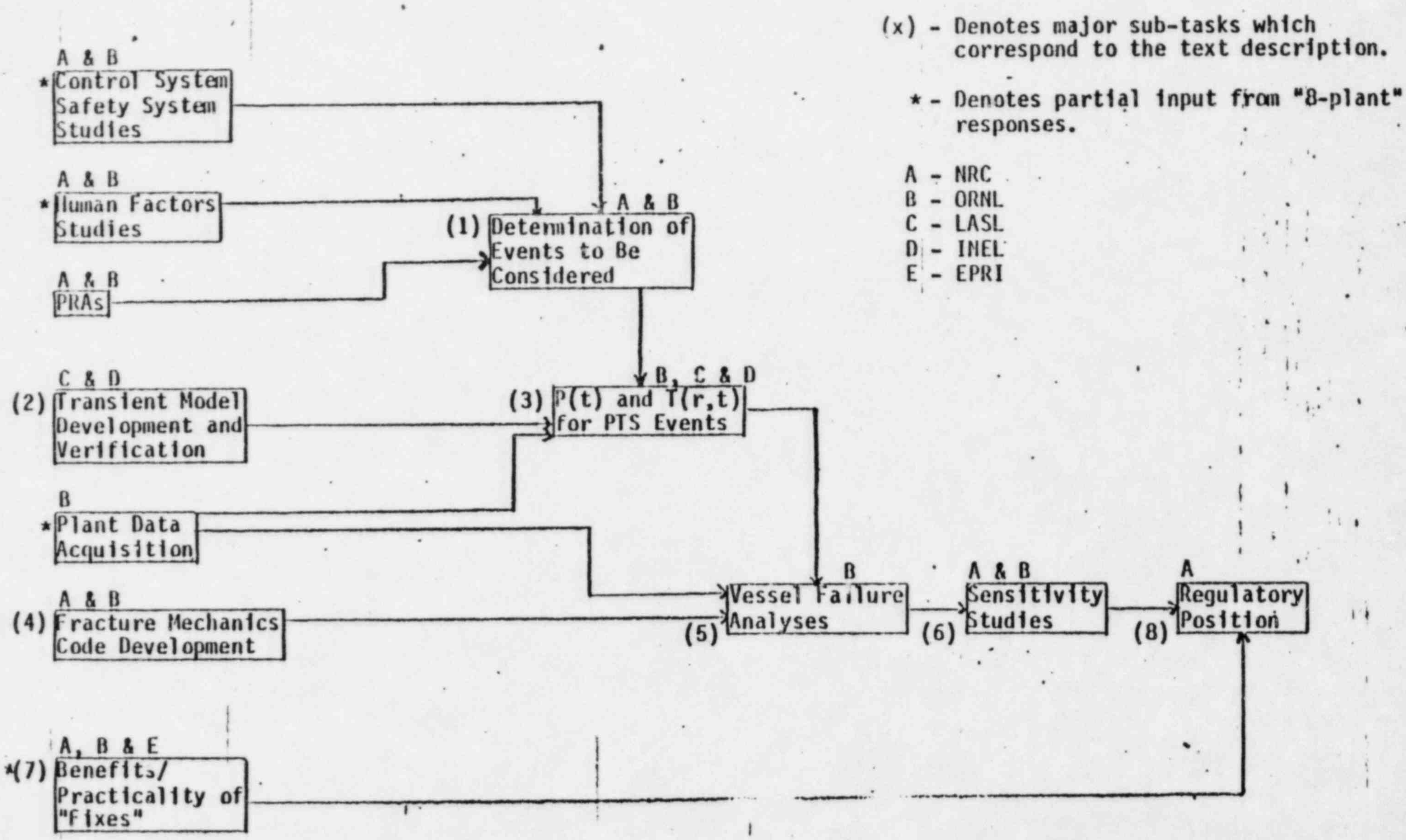
(x) - Denotes major sub-tasks which correspond to the text description.
* - Denotes partial input from "8-plant" responses.
A - NRC
B - ORNL
C - LASL
D - INEL
E - EPRI
A & B
* Control System Safety System Studies
A & B
* Human Factors Studies
A & B
PRAs
A & B
(1) Determination of Events to Be Considered
C & D
(2) Transient Model Development and Verification
B, C & D
(3) P(t) and T(r,t) for PTS Events
B
* Plant Data Acquisition
A & B
(4) Fracture Mechanics Code Development
B
(5) Vessel Failure Analyses
A & B
(6) Sensitivity Studies
A
(8) Regulatory Position
A, B & E
*(7) Benefits/ Practicality of "Fixes"

FIGURE 1