NUREG-0649
Rev. 1

Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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Task Actions Plans for Unresolved Safety Issues Related to Nuclear Power Plants

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Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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ABSTRACT

This document contains Task Action Plans for generic tasks addressing Unresolved Safety Issues (USIs) related to nuclear power plants. Progress on USIs is reported to Congress each year in the NRC Annual Report pursuant to the requirements of Section 210 of the Energy Reorganization Act of 1974, as amended. In addition, the NRR issues NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" on a quarterly basis; this report provides current schedule information for each USI.

The Task Action Plans in this document include a description of the issue, a description of the NRC staff's approach to resolving the issue, a general discussion of the basis for continued operation and licensing pending resolution of the issue, a discussion of the technical organizations involved in the task, the requirements of manpower and program support funding, interactions with outside organizations and potential problems. This document does not include Task Action Plans for generic tasks addressing USIs for which reports providing the NRC staff resolution of the issue have been published. Those tasks for which reports have been published are ident'fied and the reports are referenced.

The Task Action Plans for active USIs are revised on a yearly basis and approved by the Director, Office of Nuclear Reactor Regulation. This report contains the 1984 revisions to the Task Action Plans.

CONTENTS

INTRODUCTION

As a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977, to include, among other things, a new Section 210 as follows:

UNRESOLVED SAFETY ISSUES PLAN

> SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

The joint Explanatory Statement of the House-Senate Conference Committee for the FY 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

SECTION 3 - UNRESOLVED SAFETY ISSUES

> The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned.

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report describing the NRC generic issues program (NUREG-0410).[1] The NRC program was already in place when PL 95-209 was enacted and was of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission

[1] NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," issued on January 1, 1978.

indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of over 130 generic technical activities addressed in the NRC program to determine which issues fit this description and qualified as Unresolved Safety Issues for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

This review is described in a report, NUREG-0510, entitled "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," dated January 1979. The report provides the following definition of an Unresolved Safety Issue:

> An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects.

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequcy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an Unresolved Safety Issue is potentially significant from a public safety standpoint <u>and</u> its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 Unresolved Safety Issues addressed by 22 tasks in the NRC program were originally identified.

An indepth and systematic review of generic safety concerns identified between January 1979 and March 1981 was performed by the staff to determine if any of these issues should be designated as Unresolved Safety Issues. The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident"; from ACRS recommendations;

from abnormal occurrence reports; and from other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD), and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional Unresolved Safety Issues be considered by the Commission. The Commission considered the above information and approved the four Unresolved Safety Issues A-45 through A-48. A description of the review process for candidate issues, together with a list of the issues considered, is presented in NUREG-0705, dated March 1981. An expanded discussion of each of the new Unresolved Safety Issues is also in NUREG-0705. In addition to the four issues identified above, in December 1981, the Commission approved another issue, A-49, Pressurized Thermal Shock, as an Unresolved Safety Issue.

Reports have been published which provide the NRC staff's resolution for those issues that have been technically resolved. These tasks are listed in Table 2 with a reference to the appropriate NRC document providing the staff's resolution of the issue.

The purpose of this document is to provide the latest revisions to Task Action Plans for those tasks listed in Table 1 that have not been completed. Further revisions to Task Action Plans, including those for any new USIs approved by the Commission, will be included as they are developed and approved.

Table 1. Unresolved Safety Issues and Applicable Generic Task Numbers

	Unresolved Safety Issue	Task No.
1.	Pressurized Water Reactor Steam Generator Tube Integrity*	A-3, A-4, A-5
2.	Systems Interactions in Nuclear Power Plants	A-17
3.	Seismic Design Criteria	A-40
4.	Containment Emergency Sump Reliability*	A-43
5.	Station Blackout	A-44
6.	Shutdown Decay Heat Removal Requirements	A-45
7.	Seismic Qualification of Equipment in Operating Plants	A-46
8.	Safety Implications of Control Systems	A-47
9.	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	A-48
10.	Pressurized Thermal Shock	A-49

*Updated Task Action Plans for Unresolved Safety Issues A-3, 4, 5 and A-43 are not included here since these Unresolved Safety Issues are near completion.

Table 2. NRC Documents Providing Staff's Resolution Of Unresolved Safety Issues

Task No. and Title	Document No. and Title	Document Date
A-1, Water Hammer	NUREG-0927, Revision 1, "Evaluation of Water Hammer in Nuclear Power Plants - Technical Findings Relevant to Unresolved Safety Issue A-1"	April 1984
	NUREG-0933, Revision 1, "Regulatory Analysis for USI A-1, 'Water Hammer'"	April 1984
	Standard Review Plan (NUREG-0800) Section 3.9.3, Revision 1, "ASME Code Class 1, 2 and 3 Component Supports and Core Support Structures"	April 1984
	Standard Review Plan (NUREG-0800) Section 3.9.4, Revision 2, "Control Rod Drive Systems"	April 1984
	Standard Review Plan (NUREG-0800) Section 5.4.6, Revision 3, "Reactor Core Isolation Cooling System (BWR)"	April 1984
	Standard Review Plan (NUREG-0800) Section 5.4.7, Revision 3, "Residual Heat Removal System (RHR)"	April 1984
	Standard Review Plan (NUREG-0800) Section 6.3, Revision 3, "Emergency Core Cooling System"	April 1984
	Standard Review Plan (NUREG-0800) Section 9.2.1, Revision 3, "Station Service Water System"	April 1984
	Standard Review Plan (NUREG-0800) Section 9.2.2, Revision 2, "Reactor Auxiliary Cooling Water Systems"	April 1984
	Standard Review Plan (NUREG-0800) Section 10.3, Revision 3, "Main Steam Supply Systems"	April 1984

Table 2. (Continued)

Task No. and Title	Document No. and Title	Document Date
A-1 (Continued)	Standard Review Plan (NUREG-0800) 10.4.7, Revision 3, Condensate and Feedwater Systems"	April 1984
A-2, Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems--Resolution of Generic Task Action Plan A-2"	January 1981
A-6, Mark I Short Term Program	NUREG-0408, "Mark I Containment Short Term Program Safety Evaluation Report	December 1977
A-7, Mark I Long Term Program	NUREG-0661, "Safety Evaluation Report, Mark I Containment Long Term Program--Resolution of Generic Technical Activity A-7"	July 1980
	NUREG-0661, Supplement No. 1, "Safety Evaluation Report - Mark I Containment Long Term Program--Resolution of Generic Technical Activity A-7"	August 1982
	Standard Review Plan (NUREG-0800) Section 6.2.1.1.C, Revision 4, "Pressure-Suppression Type BWR Containments"	July 1981*
A-8, Mark II Containment Pool Dynamic Loads	NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria"	August 1981
	Standard Review Plan (NUREG-0800) Section 6.2.1.1.C, Revision 4, "Pressure-Suppression Type BWR Containments"	July 1981*
A-9, ATWS	NUREG-0460, Volume 4, "Anticipated Transients Without Scram for Light Water Reactors"	March 1980
	Final Rule - 49FR57521	July 26, 1984

*Most current revision of the appropriate Standard Review Plan section is listed here.

Table 2. (Continued)

Task No. and Title	Document No. and Title	Document Date
A-10, BWR Feedwater Nozzle Cracking	NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking--Resolution of Generic Technical Activity A-10"	November 1980
A-11, Reactor Vessel Materials Toughness	NUREG-0744, Volumes I and II, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue"	October 1982
A-12, Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports"	October 1983
	Standard Review Plan (NUREG-0800) Section 5.3.4, Revision 0, For Comment, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports"	April 1984
A-24, Qualification of Class 1E Safety Related Equipment	NUREG-0588, "Interim Staff Position on Environmental Qualification of of Safety-Related Electrical Equipment"	December 1979
A-26, Reactor Vessel Pressure Transient Protection	NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors"	September 1978
	Standard Review Plan (NUREG-0800) Section 5.2., Revision 1, "Overpressure Protection," with BTP RSB 5-2, Revision 0, "Overpressurization Protection of Pressuzied Water Reactors While Operating at Low Temperatures"	July 1981*
A-31, Residual Heat Removal Requirements"	Regulatory Guide 1.139, "Guidance for Residual Heat Removal"	May 1978

*Most current revision of the appropriate Standard Review Plan section is listed here.

Table 2. (Continued)

Task No. and Title	Document No. and Title	Document Date
A-31 (Continued)	Standard Review Plan (NUREG-0800) Section 5.4.7, Revision 2, "Residual Heat Removal (RHR) System"	July 1981*
A-36, Control of Heavy Loads Near Spent Fuel	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants--Resolution of Generic Technical Activity A-36"	July 1980
	Standard Review Plan (NUREG-0800) Section 9.1.5, Revision 0, "Overhead Heavy Loads Handling Systems"	July 1981*
A-39, Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment	<u>Mark I Plants</u>	
	NUREG-0661, "Safety Evaluation Report, Mark I Containment Long Term Program" [Appendix A, NRC Acceptance Criteria and Section 2.13, Safety Relief Valve Discharge Loads]	July 1980
	<u>Mark II Lead Plants</u>	
	NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteira" [Appendix D, NRC Acceptance Criteria and Section II, SRV-Related Hydrodynamic Loads and Pool Temperature Limits]	October 1978
	<u>Mark II and III Plants</u>	
	NUREG-0802, "Safety/Relief Valve Quencher Loads Evaluation for BWR Mark II and III Containments"	October 1982
	<u>Applicable to Mark I, II and III</u>	
	NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharge for BWR Plants"	May 1981

*Most current revision of the appropriate Standard Review Plan section is listed here.

Table 2. (Continued)

Task No. and Title	Document No. and Title	Document Date
A-39 (Continued)	NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments"	November 1981
A-42, Pipe Cracks in Boiling Water Reactors	NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"	October 1979

TASK ACTION PLAN
(March 1984)

SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (TASK A-17)

Lead Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
Task Manager:	Dale Thatcher GIB, DST
Lead Manager:	K. Kniel, Chief, GIB, DST
NRR Principal Reviewers:	E. Chelliah, Systems Interaction Section, RRAB, DST
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Office of Nuclear Regulatory Research (RES)	D. Rasmuson Division of Risk Analysis
Office of Nuclear Material Safety and Safeguards (NMSS)	B. Mendelsohn Division of Safeguards
Office for Analysis and Evaluation of Operational Data (AEOD)	E. Imbro AEOD
Office of Inspection & Enforcement (IE)	(Later), Events Analysis Branch

Applicability:	Light Water Reactors (Pressurized and Boiling Water Reactors)
Projected Completion Date:	March 1986

1. DESCRIPTION OF PROBLEM

Background

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The review performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility, and where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The Advisory Committee on Reactor Safeguards (ACRS) identified a generic need to examine the matter of systems interactions in a letter to L. M. Muntzing dated November 8, 1974. The staff initiated a systems interaction program in May of 1978 with the definition of USI A-17 "Systems Interaction in Nuclear Power Plants." Subsequent events and follow up actions led to initiation of various programs to investigate the issue.

The specific objective of this Task Action Plan (TAP) is to assess the safety significance of potential adverse systems interactions, and the cost effectiveness of searching for and correcting safety significant systems interactions. If corrective measures are warranted, based on a regulatory analysis conducted by the staff, an implementation requirement will be developed.

Discussion

Many significant events at operating nuclear power plants have been traced to, or postulated to be the result of, a single common cause, as opposed to multiple independent causes, and as a result the required independence among the plant safety systems and the independence of the safety systems from the non-safety systems has been questioned.

Because some of these events are due to unexpected interdependencies among the various plant systems, generic issue A-17 was developed to address these "systems interactions." It has also been recognized that some of these single cause events resulted from common characteristics of the equipment which make up the plant systems. These common characteristics include inherent features such as single manufacturers, common maintenance practices, and common testing practices. For purposes of discussion this latter class of common cause events/failures will be referred to as common mode failures (CMFs). (For additional discussion of CMFs, see Reference 1.)

To proceed with a discussion of the broad subject area of "interactions" it is necessary to utilize some definitions. Although "perfect" definitions probably cannot be found some definitions are necessary in order to describe the scope of what this program will address and also to specifically exclude certain areas.

A definition is given here for adverse systems interactions and other common cause failures. For a diagram showing the interrelationship of these terms, see Figure 1. The objective of the A-17 program is to address adverse systems interactions. However, in the review of operating experience, all common cause failures will be initially considered.

Definitions

(a) Common Mode Failure (CMF)

Multiple failures resulting from a single common cause and typically characterized by the failure of identical components in redundant safety systems.

Such multiple failures are traceable to causes such as external events, common design, manufacturing and installation errors; or operation, testing and maintenance errors.

The usual design practice for safety systems is to satisfy the single failure criterion by providing identical, redundant safety systems which are subjected to common external events and made, installed, operated, tested and maintained by common individuals. Therefore, common mode failures are a recognized source of compromise in independence and are addressed in a number of ways, and in some cases without specific identification. The following is a discussion of some of the ways in which this class of failures/errors are addressed.

To obtain protection from possible failures, including failures resulting from external events, the components of the safety systems are designed, qualified and installed to be immune to such anticipated challenges. (For specific examples of external events covered in the review process, see Table 1.)

To obtain immunity to failures, including common mode failures, resulting from design, manufacturing and installation errors, the safety-related systems, structures and components are subjected to various independent design reviews and quality control and quality assurance programs which include comprehensive testing requirements at all phases of construction and preoperation. The concept of independent design reviews has been used by industry to varying degrees and specific reviews have been requested by NRC. The area of Quality Assurance has been undergoing major improvements both at utilities and within NRC.

Protection from failures, including common mode failures, attributed to errors by operators, technicians and maintenance personnel can be obtained through adequate training and good procedures for all aspects of operation, testing and maintenance. The Division of Human Factors Safety has major programs underway to address all of these areas.

Other provisions may be utilized for protection against common mode failures or discovered unreliability of specific types of components. One design technique which is utilized is diversity. An example of such an application by the staff is a portion of the requirements which resulted from the Salem Anticipated Transient Without Scram event. As part of the resolution, it was concluded that consideration should be given to providing a diverse breaker trip scheme. These cases have been addressed on an individual basis, however the concept of diversity is cited in the regulations (General Design Criterion 22).

The other class of common cause events which is defined here is adverse systems interactions events. Although it can be argued that these definitions may overlap, it is necessary to differentiate between the two so that a clear overall objective can be defined for USI A-17.

(b) Systems Interaction (SI)

Actions or inactions (not necessarily failures) of various systems (subsystems, divisions, trains), components or structures resulting from a single credible failure within one system, component or structure and propagation to other systems, components, or structures by inconspicuous or unanticipated interdependencies. Note: The major difference between this type of event and a classic single failure event is in the non-obvious aspects of the initiating failure and/or its propagation. Systems interactions also can involve both safety and non-safety systems.

(c) Adverse Systems Interaction (ASI)

A systems interaction which produces an undesirable result.

(d) Undesirable Result (due to systems interaction)

This is defined by a list of the types of events which will be considered in A-17.

The list was created to be general and conservative for the purpose of capturing potential adverse systems interactions and therefore terms such as "undesirable" instead of "unacceptable" and "degradation" instead of "failure" were used. After these potential events are found, the safety significance of the events will be determined and the question of acceptability addressed.

(1) Degradation of redundant portions of a safety system, including consideration of all auxiliary support functions. Redundant portions are those considered to be independent in the design and analysis (Chapter 15) of the plant.

(2) Degradation of a safety system by a non-safety system.

(3) Initiation of an "accident" (for example, Loss of Coolant Accident, Main Steam Line Break) and (1) the degradation of at least one redundant portion of any one of the safety systems required to mitigate that event (Chapter 15 analyses); or (2) degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed or incorrect action.

(4) Initiation of a "transient" (including reactor trip), and (1) the degradation of at least one redundant portion of any one of the safety systems required to mitigate the event (Chapter 15 analyses); or (2) degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed, or incorrect action.

(5) Initiation of an event which requires actions of the plant operators in areas outside the control room area (it may be due to Control Room evacuation or initiation of a plant shutdown) and

disruption of the access to these areas. (For example by disruption of the security system or isolation of an area by closure of fire doors or actuation of a suppression system.)

The intersystem dependencies [or systems interactions (SIs)] have been divided into three classes:

(a) Functionally coupled: Those SIs that result from sharing of common systems/components; or physical connections between systems including electrical, hydraulic, pneumatic or mechanical.

(b) Spatially coupled: Those SIs that result from sharing of common structures/locations, or spatial inteties such as Heating, Ventilating and Air Conditioning and drain systems.

(c) Induced-human-intervention-coupled: Those SIs where a plant malfunction (such as failed indication) inappropriately induces an operator action or a malfunction inhibits an operator's ability to respond. (Induced-human-intervention coupled systems interactions exclude random human errors and acts of sabotage.)

Staff Actions Related to System Interactions

The staff has addressed the issue of "systems interactions" in a number of ways:

(1) The SRP has a number of sections which specifically deal with the potential for adverse systems interactions. For a list of these sections, see Table 2.

(2) Similar to the SRP sections, the Systematic Evaluation Program (SEP) has utilized a number of review topics which address the potential for adverse systems interactions. For a list of topics, see Table 2.

(3) In response to events at operating reactors, the Office of Inspection and Enforcement (IE) has issued Bulletins and Information Notices which address the potential for adverse systems interactions. Two significant bulletins which are related to the issue of systems interaction are:

    (a) IEB-79-27, "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation." (Ref. 13)

    (b) IEB-80-11 "Masonry Wall Design." (Ref. 14)

    A significant Information Notice is:

    (a) IEIN-79-22 "Qualification of Control Systems." (Ref. 15)

(4) The significant operating events, and concerns raised after Three Mile Island (TMI), led the staff to identify a separate USI for the investigation of the potential for significant failures and adverse interactions in the area of control systems. This is USI A-47, "Safety Implications of Control Systems."

(5) Specific concerns including some related to common cause were identified in the area of direct current (dc) power systems and as a result a generic safety issue was instituted. This is A-30 "Adequacy of Safety Related DC Power Supplies.

(6) An initial task, identified as part of the resolution of A-17, involved a study of the Watts Bar-1 plant by Sandia National Laboratory with the use of fault trees (Ref. 2). Other tasks included a review of presently available systems interaction methodologies (see Refs. 3, 4 and 5).

(7) A number of plants have performed probabilistic risk assessments (PRAs) and have addressed the broad area of dependent failures including systems interactions.

There are also a number of ongoing inquiries into systems interactions.

The ongoing inquiries into systems interactions include: (1) Pacific Gas and Electric is completing their evaluation of the systems interactions discovered during their reviews of the Diablo Canyon units. (2) The study of Indian Point-3 by the licensee, the Power Authority of the State of New York (PASNY), using an analysis procedure developed by PASNY and its contractor is near completion (Ref. 11). (3) Lawrence Livermore National Laboratory (LLNL) completed its documentation and demonstration of the Digraph-Matrix Analysis (Ref. 6), a method by which potential systems interactions may be identified. They also evaluated, by a pilot study application of the technique, two modes of operation of the high pressure coolant injection systems at Watts Bar-1 (Ref. 12). (4) Consumers Power Company has initiated a systems interaction program on Midland-2.

The TMI accident led to issuance of the "NRC Action Plan Developed as a Result of the TMI-2 Accident" (NUREG-0660) which identified Action Item II.C.3, "Systems Interaction," "...to coordinate and expand ongoing staff work on systems interaction [Unresolved Safety Issue (USI) A-17] so as to incorporate it into an integrated plan for addressing the broader question of systems reliability in conjunction with IREP and other efforts." The TMI-2 Action Plan also stated that "As these programs go forward, there will be a conscious effort to coordinate these activities, including possible combination of resources, to eliminate unnecessary duplication." DST and the Division of Risk Analysis, RES have been developing and reviewing various techniques for addressing systems interactions. Some of this work is being done in conjunction with work in the area of PRA and some techniques have been included in the National Reliability Evaluation Program Procedures Guide (NUREG/CR-2815) (Ref. 18).

The work described in this TAP includes all systems interaction activities described in TMI Action Plan Item II.C.3. Resolution of USI A-17 will therefore constitute completion of action item II.C.3.

Approach

Based on the preceding considerations, this TAP outlines a program which is intended to:

(1) Evaluate discovered sources and potential sources of common cause events (the broader topic), identify significant actual and postulated common mode failures and adverse systems interactions and determine the safety significance of the adverse systems interactions;

(2) Compare and evaluate applicable adverse systems interaction search methods (past and ongoing) and determine the efficacy of the methods for current use;

(3) Evaluate regulatory criteria both from an adequacy viewpoint and an application viewpoint; and

(4) Develop proposed requirements, if any, based on 1, 2 and 3 above, perform a value/impact assessment and recommend implementation.

2. PLAN FOR RESOLUTION

There is a large amount of work which has been and is being performed in the area of "systems interaction." To best utilize all of the work, a program is outlined to integrate as much of this work as possible with the objective of resolving USI A-17 within approximately a 2-year schedule.

The overall program will review and evaluate past studies and preliminary conclusions, and also follow the ongoing studies. From this review and evaluation, it is anticipated that possible alternative resolutions can be defined in terms of the benefits and cost and from these possible resolutions, a cost effective solution could be chosen.

In general, the program will involve two significant efforts which will proceed in parallel, each with a number of tasks. One effort will focus on operating experience, various activities by utilities, and NRC studies. Its objective will be to search for common cause events and then evaluate them with emphasis on adverse systems interactions. The parallel effort will focus on a review of the methods that have been and are being, used to uncover adverse systems interactions. Its objective will be to determine the attributes of the methods so that guidelines can be developed for defining an acceptable search program in the event that it is determined to be necessary. For an overall diagram of the interrelationship of the tasks, see Figure 2.

Task Descriptions

(A) Task 1 Search for Common Cause Events

This activity will conduct a review of various sources of information on common cause events and compile a list of adverse systems interactions and other common cause events based on the definitions in this Task Action Plan. The sources for information will include:

(a) Systems interaction studies performed to date by the staff, laboratories, and utilities. This will include the Sandia PRA Study (Ref. 2), the Diablo Canyon Study (Ref. 8), the San Onofre Study (Ref. 9), the Grand Gulf study (Ref. 10), the Indian Point Study by PASNY (Ref. 11), the Watts Bar study by Digraph Matrix Method (Ref. 12), the Results on Zion (Ref. 13) and any other meaningful studies (including plant PRAs) performed by utilities. It will also consider the review of methodology performed by LLNL, Brookhaven National Laboratory (BNL) and Battelle (Refs. 3, 4 and 5).

(b) Evaluation done as a result of NRC requirements/requests in the area of common cause failures. This will consider responses to IE Bulletins 79-27 (Ref. 14) and 80-11 (Ref. 15) and IE Information Notice 79-22 (Ref. 16). As part of this effort the sections of the SRP and SEP topics, (see Table 2) which deal with some types of systems interactions will be considered to determine what impact (such as plant modifications) may have resulted from their application. In addition, the docket files for power plants undergoing licensing review will be evaluated to see what recent staff questions and/or positions have been issued in the area of systems interactions and common cause events.

(c) Other Generic Safety Issues. For example:

(1) USI A-47, "Safety Implications of Control Systems." It is recognized that the area of control systems is a potential contributor to adverse systems interactions. Some programs have been underway as part of the resolution of A-47. This subtask would evaluate the scope and content of that work to determine potential adverse systems interactions which have been discovered. Although this work will be investigated for examples of ASIs, it may be decided that such ASIs will not be addressed by A-17 but will be addressed by A-47.

(2) A-30, "Adequacy of Safety-Related DC Power Supplies." It has been recognized that the plant electrical system is an area of significant concern. This issue has been under investigation for a number of years, and resolution is expected within the next year. Therefore, the work on A-17 can possibly use the information from this program where applicable.

(3) New Generic Issues - for example, Issue 77, "Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains" and Issue 81, "Potential Safety Problems Associated with Locked Doors and Barriers in Nuclear Power Plants."

(d) Search of Operating Experience. There have been a number of events labelled "systems interactions." This search would apply the definitions of the previous section to these events and create a list of experienced common cause events including adverse systems interactions.

(e) ACRS information and meetings. ACRS has been a prime factor in the pursuit of systems interactions. This subtask will compile the examples, and postulated events, given by the ACRS, based on the preceding definitions. In addition, meetings will be scheduled with the ACRS for the purpose of keeping them informed of systems interaction activities.

(f) AEOD studies. AEOD has published a number of reports, some of which have discussed common cause events and systems interactions (for example, Potentially Damaging Failure Modes of High and Medium Voltage Electrical Equipment). This task will compile any results and conclusions from their work which could provide further information on potentially adverse systems interactions.

(g) Efforts by industry groups. This task will investigate the efforts in this area which have been undertaken by industry groups such as the Institute of Nuclear Power Operations (INPO), the Electric Power Research Institute (EPRI), the Institute of Electrical and Electronics Engineers (IEEE), the Atomic Industrial Forum (AIF) and the Nuclear Steam Supply System (NSSS) Owners Group. Any efforts which have uncovered common cause events and/or adverse systems interactions will be used to expand the list.

(B) Task 2 Trends/Patterns of Common Cause Events

Based on the results of the above task, this task will compile and evaluate the list of "common cause events." Where significant safety questions are raised due to the elimination of some issue as not to be considered further by A-17 (for example, common mode failures), this task will explain where such questions are, or will be, covered; or propose potential new generic issues to be processed in accordance with NRR Office Letter #40 (Ref. 19).

For the ASIs, complete (as possible) documentation will be compiled to include:

(a) What system, component, or structure failure initiated the event or could initiate the event? Is it considered a safety system, component, or structure?

(b) What system interdependency creates the coupling and is it functional or spatial or induced-human or some combination?

(c) What undesirable event or degraded function resulted? What was the plant mode of operation?

(d) What plant recovery actions were taken at that time and how much time was available?

(e) What subsequent corrective actions were taken?

(f) Is the event within the plant design bases?

(g) Could the review process (for example, SRP) be expected to identify this event?

(h) How was the ASI uncovered such as, by Licensee Event Report, PRA, SI study?

From this information, this task will search for patterns and trends based on similarities.

(C) Task 3 Indian Point Comparison

This task will involve the application to Indian Point 3, of

(1) Digraph Matrix Method by LLNL, and
(2) Interactive Fault Tree/Failure Mode Effects Analysis by BNL.

These studies are to be done in parallel with Tasks 1 and 2 above. The objective is to complete the candidate studies in a time frame which will allow an assessment of the efficacy of each methodology as part of

Task 5. In addition, any common cause events that are discovered will be included in Task 2.

(D) Task 4 Screen Events for Safety Significance

This task will evaluate the discovered events (Task 1); review the established trends/patterns, if any, (Task 2); review the results of the other SI studies (Task 3); to determine if there is a significant risk associated with the potential for adverse systems interactions.

The major emphasis of this task will be the development of screening criteria. To develop these criteria, the trends and patterns will be reviewed and screening categories will be formulated considering such factors as: the importance of the safety function affected, and the significance of time in the sequence. The resulting "fixes" will also be reviewed.

Based on the "screening criteria," the ASIs will be screened and an estimate of the potential for the occurrence of other safety significant ASIs will be made.

It is possible that a pattern of adverse systems interactions may indicate a weakness in criteria in a particular area. This task may conclude that certain types of ASIs appear to be resolved by certain classes of fixes. It is also possible that this task could conclude that certain types of ASIs should be resolved by implementing new criteria or new review procedures.

As part of this task, technical reports will be made available to all licensees and applicants.

(E) Task 5 Review-Define Search Methods

Considering the ASIs discovered in Task 1 and the evaluations performed in Tasks 2 and 3, this task will compile the various methods (proposed or used) to search for and find ASIs. The methods to be compiled and summarized will include:

(a) Operating Experience Searches
(b) Plant Walk Throughs
(c) Failure Mode and Effects Analyses
(d) Use of SRP Guidance
(e) Multi-discipline Review Team
(f) Candidate Studies on Indian Point-3
(g) PRA
(h) Combination of the above

(F) Task 6 Compare/Evaluate Methods

This task will compare the various methods to determine the most effective method or methods for discovering potential ASIs. It will consider both the effectivenss of the methods for finding the events and the costs involved.

This task will utilize applicable information developed previously by the National Laboratories. Specifically, the evaluations done in References 3, 4 and 5.

(G) Task 7 Technical Resolution

Based on the results of Task 4 and 6, a regulatory analysis will be performed as the basis to determine if:

(1) Plants need to do an ASI "study" to find ASIs and, if yes, what the "study" should involve; and/or
(2) New requirements are necessary; and/or
(3) New review criteria/procedures are necessary.

The objective of this task is to develop a proposed regulatory position from the technical findings of the program and to support that position with a regulatory analysis. Part of this task will be to develop any necessary regulatory guidance, such as a regulatory guide, SRP change and/or a rule.

Program Schedu'e

Complete and prepare draft resolution	03/30/85
Completed Package to NRR Director	05/30/85
Committee to Review Generic Requirements (CRGR) Review Complete	07/30/85
Issue for Public Comment	09/30/85
Resolve Comments and Re-issue Package to NRR Director	12/30/85
CRGR Review Complete	02/30/86
Issue Final Resolution	03/30/86

Manpower and Technical Assistance by Task

Task	FY84		FY85		FY86	
	NRC* (PSY)	TA ($)	NRC* (PSY)	TA ($)	NRC* (PSY)	TA ($)
1	1.0	230K	.3	-	-	-
2	1.1	(1)	.3	-	-	-
3	1.8	1350K(2)	.2	-	-	-
4	.5	(1)	1.0	-	-	-
5	.5	100K(3)	.7	(3)	-	-
6	.5	100K(3)	.8	100K(3)	-	-
7	-	-	1.0	200K(3)	.75	-
	5.4	1780K	4.3	300K	.75	

(1) Oak Ridge National Laboratory is performing Tasks 1, 2 and 4 as part of FIN No. B-0789
(2) BNL and LLNL are performing Task 3 under FIN Nos. A-3725 and A-0405.
(3) This work has not been scheduled.
*Most of the staff manpower will be from GIB and RRAB. Some smaller portion will be from other branches within NRR, AEOD, RES, NMSS, and IE . See Table 3.

3. BASES FOR LICENSING OR CONTINUED OPERATION PENDING COMPLETION OF PROGRAM

Although the occurrence of some events at Light Water Reactors that adversely affect plant safety justifies the present program on systems interactions, NRR is confident that current regulatory requirements and procedures provide an adequate degree of public health and safety.

Most applicants have not committed to implement a comprehensive program that separately evaluates all structures, systems, and components for adverse systems interactions. However, there is assurance that LWRs can be operated without endangering the health and safety of the public. Each application was evaluated against licensing requirements that were founded on the principle of defense-in-depth. Adherence to this principle and conformance to the regulations (for example, the General Design Criteria) results in design provisions such as physical separation and independence of redundant safety systems. The design provisions are also subject to review against the SRP which provides for multidisciplinary reviews of safety-related equipment and addresses some types of common cause events and potential adverse systems interactions (see Tables 1 and 2). Furthermore, the quality assurance program that is followed during design and construction contributes to the adherence to these provisions.

Therefore, it is concluded that the design and construction as well as the licensing process can provide for a significant degree of plant safety with respect to the potential for adverse system interactions.

The Systematic Evaluation Program (SEP) was initiated in 1977 to evaluate operating facilities to reconfirm and document their safety in light of the current regulatory requirements. The SEP derived a list of significant safety topics from existing issues. Although the 137 topics do not explicitly address systems interactions reviews, the acceptance criteria for some topics include reviews for hazards created by intersystem dependencies. The SEP also includes a systematic review of the operating experience of the plant under evaluation. The SEP is nearing completion of Phase II wherein eleven of the oldest plants are being evaluated (that is, those plants licensed most remotely from current requirements). The results of Phase II have been presented to the Commission. Concurrently, consideration is being given to a program which would follow SEP Phase II. Although the SEP objective was not intended to resolve USI A-17 on older plants, the acceptance criteria for the topics within SEP are derived from the acceptance criteria within the SRP. Some of the acceptance criteria inherently address potentially adverse systems interactions. The corrective actions resulting from the SEP reviews will help preclude adverse systems interactions for the operating plants reviewed, in the same way the SRP review provides protection against systems interactions. The follow-on program to SEP Phase II would similarly upgrade such protection for subsequently evaluated plants.

Operating reactor experience is continually monitored to detect precursors to serious event sequences. As such events occur, corrective actions are taken for all affected facilities. Thus, the performance of a systematic review of older plants against current requirements and the continuing generic reaction to isolated events contribute to the prevention of adverse systems interactions in operating plants.

An additional measure of safety has been taken on all plants (both those operating and those under licensing review) in the area of operator information. Specifically, Generic Letter 82-33 (Supplement 1 to NUREG-0737), dated December 17, 1982 provided "Requirements for Emergency Response Capability." As part of these requirements, utilities will be adding a Safety Parameter Display System as well as demonstrating the adequacy of their post-accident monitoring capabilities as outlined in Regulatory Guide 1.97 (Ref. 20). Both these requirements, and the other requirements of that letter, will enhance the ability of the operator to perform mitigating actions in response to events including those due to adverse systems interactions.

Based on the activities identified above and the ongoing activities in the area of adverse systems interactions, we conclude that licensing and operation of Pressurized Water Reactors and Boiling Water Reactors is acceptable pending completion of this program.

4. NRC TECHNICAL ORGANIZATIONS INVOLVED

A. Division of Licensing (DL)

Support from the DL is needed to continue the coordination with the participating utilities. The utilities' cooperation is needed to provide the detailed information used in a systems interactions analysis. The needed information includes engineering Piping and Instrumentation Diagrams (P&IDs), systems flow diagrams and manuals, electrical drawings, instrumentation and control drawings, plant procedures, and selected reports. DL will provide assistance to the contract Technical Monitor for setting up and coordinating with the utility personnel, informational meetings, documentation requests, and site visits that may be necessary. DL will also provide assistance to the Task Manager for integrating any relevant experience and any new requirements resulting from the activities identified in Task A-17. DL will contribute to the review and approval of any licensing requirements and guidelines developed as a result of this USI, and will provide review and comment on the technical evaluations provided by the Task Manager.

Manpower Requirements

	Total	FY84	FY85
Operating Reactors Branch No. 1	0.1 psy*	.05	0.05
Systematic Evaluation Program Branch	0.1 psy	.05	0.05
Licensing Branch No. 4	0.1 psy	.05	0.05
Licensing Branch No. 3	0.05 psy	.05	-
Operating Reactors Assessment Branch	0.05 psy	.05	-
	0.4 psy		

*Assumed 1 professional staff year = 40 person weeks.

B. Division of Systems Integration (DSI)

DSI will provide review and comment on technical evaluations provided by the Task Manager in the areas of instrumentation and control, electrical power, the reactor systems and auxiliary systems designs, and accident analysis. The Instrumentation and Control Systems Branch and the Power Systems Branch will provide assistance for the purpose of integrating relevant experience and any new requirements and guidelines stemming from the completion of the tasks described in Task A-17. The Reactor Systems Branch and the Auxiliary Systems Branch will assist in the development of the screening criteria to be used for establishing safety significance of discovered systems interactions. A large portion of the Auxiliary Systems Branch support will be determining the safety significance of systems interactions discovered at Indian Point-3 on the auxiliary feedwater systems. The Auxiliary Systems Branch will provide coordination with DST for completeness to assure that all sources of missiles and safety-related equipment that could be impacted by missiles were analyzed. The Auxiliary Systems Branch will also share the coordination responsibility with the Mechanical Engineering Branch, Division of Engineering for the consequences of High Energy Line Breaks (HELBs) since the Auxiliary Systems Branch has the primary responsibility for HELBs outside containment. The Containment Systems Branch will provide coordination with DST to assure that the effects of systems interactions on containment isolation and containment pressure/temperature analyses have been considered. In addition DSI will contribute to the formulation, review and approval of the recommendations, and guidelines developed at the completion of the tasks (described in Task A-17). DSI will also review and comment on the draft and final NUREG Report.

Manpower Requirements

	Total	FY84	FY85
Instrumentation and Control Systems Branch	.2 psy	.1	.1
Power Systems Branch	.1 psy	.05	.05
Reactor Systems Branch	.6 psy	.3	.3
Auxiliary Systems Branch	.4 psy	.3	.1
Containment Systems Branch	.10 psy	.05	.05
	1.4 psy		

C. Division of Engineering (DE)

DE will provide review and comment on technical evaluations provided by the Task Manager in the areas of (a) the qualification of equipment against spatially coupled adverse systems interactions, (b) the compatibility of fire detection and mitigation equipment with safety-related equipment including the adverse effects of inadvertent actuation, (c) HELBs and their consequential effects on control systems and safety-related equipment, and (d) generated missiles. The Equipment Qualifications Branch will provide support to establish the hostile-environment functionability of equipment identified to be within a harsh environment generated as part of a postulated systems interaction scenario. The Chemical Engineering Branch will provide coordination with DST for completeness to assure that fire protection equipment intended actuation, inadvertent actuation, or failure does not generate adverse systems interactions that are safety significant. The Mechanical Engineering Branch will provide coordination with DST for completeness to assure that the consequences of HELBs inside containment have been bounded in the safety analysis.

Manpower Requirements

	Total	FY84	FY85
Equipment Qualification Branch	.1 psy	.05	.05
Chemical Engineering Branch	.1 psy	.05	.05
Mechanical Engineering Branch	.1 psy	.05	.05
	.3 psy		

D. Division of Human Factors Safety (DHFS)

DHFS will provide review and comments on those technical evaluations involving man/machine interfaces. The scope of A-17 does not include random operator errors.

DHFS will contribute to the formulation, review and approval of the draft and final recommendations/requirements, and/or guidelines involving man/machine interfaces developed, as appropriate, during the program.

Manpower Requirements

	Total	FY84	FY85
Human Factors Engineering Branch	.2 psy	.1	.1
Procedures and Systems Review Branch	.2 psy	.1	.1
	.4 psy		

E. Division of Safety Technology (DST)

DST will provide overall management of USI A-17 and provide liaison between other Offices and NRR and provide coordination of activities performed within NRR.

DST will provide assistance to the Task Manager for the purpose of integrating relevant experience and any new requirements stemming from the completion of USIs A-44, A-47, and A-49.

The coordination between A-17 and A-47 is important and there will continue to be close coordination between these two programs.

RRAB will provide review of risk assessments associated with the regulatory analyses required to support A-17 proposed positions. The Safety Program Evaluation Branch will provide technical support on the cost/benefit evaluations associated with the recommendations and positions developed.

Manpower Requirements

	Total	FY84	FY85	FY86
Generic Issues Branch	2.7 psy	1.1	1.1	.5
Reliability and Risk Assessment Branch	4.0 psy	2.4	1.5	.1
Safety Program Evaluation Branch	.6 psy	.2	.3	.1
Research and Standards Coordination Branch	.15 psy	.05	.05	.05
	7.45 psy			

5. ASSISTANCE FROM OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS (NMSS), OFFICE OF ANALYSIS AND EVALUATION OF OPERATIONAL DATA (AEOD), OFFICE OF REGULATORY RESEARCH (RES), AND OFFICE OF INSPECTION AND ENFORCEMENT (IE)

AEOD will provide review and comments on the technical evaluations provided by the Task Manager. AEOD will also provide assistance to the Task Manager for the purpose of integrating relevant experience for which AEOD has responsibility.

Manpower Requirements

	Total	FY84	FY85
AEOD Plant Systems Unit	.1 psy	.05	.05

NMSS will provide assistance to the Task Manager in the area of nuclear power plant security systems including relevant experience at operating facilities. NMSS will provide review and comment on the technical evaluations provided by the Task Manager for those potential adverse systems interactions which involve the security systems. NMSS will also provide assistance to the formulation, review and comment of any recommendations and guidelines developed in the area of security systems.

Manpower Requirements

	Total	FY84	FY85
NMSS Division of Safeguards	.2 psy	0.1	0.1

RES will provide review and comments on those parts of the program which involve risk analysis and work for which they have related ongoing programs.

Manpower Requirements

	Total	FY84	FY85
Division of Risk Analysis	0.2 psy	0.1	0.1

IE will provide review and comments on the technical evaluations provided by the Task Manager. IE will also provide technical input regarding any inspection programs which relate to the area of systems interactions.

Manpower Requirements

	Total	FY84	FY85
Events Analysis Branch	.1 psy	.05	.05

6. TECHNICAL ASSISTANCE

Technical assistance to the program will be required for the activities identified in Tasks 1, 2, 3 and 4. Contracts have been initiated with the National Laboratories in these areas. Funding is to be provided by NRR. The estimated costs (in thousands) are as follows:

Contract	FY84
A-0445 - Application to Indian Point-3	650
A-3725 - Application to Indian Point-3	700
Survey and Evaluation of Systems Interaction Events and Sources	230

Additional technical assistance is anticipated for determination of "acceptable" search methods and estimation of the risk significance of discovered adverse systems interactions and the potential for risk reduction due to the implementation of possible solutions. Also, the cost of the solutions will be estimated. The estimated technical assistance cost (in thousands) are as follows:

Contracts	FY84	FY85
Methods Attributes	200	100
Value/Impact Assessment	-	200

7. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

The past program had benefited from a broad base of involvement with outside organizations due to the use of four National Laboratories in the program: Brookhaven National Laboratories, Livermore National Laboratories, Pacific

Northwest Laboratories, and Sandia National Laboratory. Three of the laboratories performed separate evaluations of methods that could be applied for near-term analysis of systems interactions. Many methods were evaluated including Fault Trees, Event Trees, Cause-Consequence Diagrams, Failure Modes and Effects Analysis, Phased Mission Analysis, Markov Modeling, GO, COMCAN-III, Visual Inspections, Operational Survey, Diversion Path Analysis, and Generic Cause Analysis (Refs. 3, 4 and 5). The laboratories concluded that no single method presently exists in a form adequate to perform analyses for adverse systems interactions (Ref. 17).

The staff will continue to maintain active interfaces with outside organizations. The staff has met annually since 1981 with the AIF Subcommittee on Systems Interaction. There have been discussions with NSSS vendors, applicants, licensees, and consultants on many occasions during the course of regular safety review activities, particularly those outside organizations involved in the systems interaction program tasks described in Section 2.

Additionally, informal exchanges have occurred with British and French individuals concerning their efforts on systems interactions.

The ACRS has continually pursued operating problems which it named systems interaction and has followed the progression of the systems interaction program. The ACRS interests led to meetings and memoranda and active interfaces between the staff and the ACRS. The activities of USI A-17 are scheduled to allow for keeping the ACRS informed of the program.

The cooperation of selected utilities is necessary for the resolution of of USI A-17. Utility cooperation is needed to provide the detailed information used in a systems interaction analysis on a plant. The needed information includes engineering P&IDs, systems flow diagrams and manuals, electrical line drawings, instrumentation and control drawings, plant procedures, and selected reports. In addition, utility cooperation is needed for informational meetings and site visits.

8. POTENTIAL PROBLEMS

A. A systems interaction analysis is basically a search for hidden safety problems at a nuclear power plant. It is not a process to engineer the solution to a well defined safety problem. As a result it may not be possible to assure that all such hidden problems have been uncovered. The A-17 program plan will try to answer whether we have gone far enough in the area of adverse systems interactions by reviewing operating experience for trends and patterns and considering the studies and the "fixes" which have been implemented. Based on the conclusion, it will then determine if additional requirements are necessary.

B. The cost of performing a systems interaction analysis is a potential problem. The analysis should be performed on the entire plant so as not to preclude the discovery of any intersystems dependencies. The analysis should be performed to the level of detail that would assure no hidden dependencies from supporting equipment. Both of the constraints on the analysis (broad scope and sufficient detail) contribute to the large costs of performing a systems interaction analysis. The decision to incur a large cost for the purpose of searching for adverse systems interactions is a potential problem in itself.

C. The need for detailed information about the plant creates a potential for a third problem. The utility is the organization possessing the needed detailed information. Considering that a requirement to perform a systems interaction analysis does not exist, the progress of the program will be depend upon voluntary cooperation from the involved utilities.

Table 1.

A. Some Reviews for External Events*

Hazard	Standard Review Plan Section	Systematic Evaluation Program Topic
Earthquakes	3.2, 3.7	III-1, III-6
Floodings	3.4	II-3B, III-3A
High Winds	3.3, 3.5.1.4	III-2, III-4A

B. Standard Review Plan Sections

Section 3.2.1, Seismic Classification
Section 3.2.2, System Quality Group Classification
Section 3.3.1, Wind Loadings
Section 3.3.2, Tornado Loadings
Section 3.4.1, Flooding Protection
Section 3.4.2, Analysis Procedures
Section 3.5.1.4, Missles Generated by Natural Phenomena
Section 3.7.2, Seismic System Analysis
Section 3.7.3, Seismic Subsystem Analysis

C. Systematic Evaluation Program Topics

Topic II-3B, Flooding Potential and Protection Requirements
Topic III-1, Classification of Structures, Components, and Systems, (Seismic and Quality)
Topic III-2, Wind and Tornado Loadings
Topic III-3A, Effects of High Water Level on Structures
Topic III-4A, Tornado Missles
Topic III-6, Seismic Design Considerations

*This should not be considered an exhaustive listing because there are other reviews (SRP, SEP) which deal with various aspects of external events.

Table 2. Some Present Staff Reviews for Adverse Systems Interaction*

Type	SEP Topic	SRP Section	Other
1. Spatial			
Fire	IX-6	9.5.1	
Flooding	II-3B.1 III-5B	3.4.1	
HELB	III-5A III-5B	3.6.2 3.6.1	IEIN-79-22
Missiles (Internal)	III-4.C	3.5.1.1, 3.5.1.2	
Missiles (Turbine)	III.4.B	3.5.1.3	
Masonry Walls	III-6	3.8.4	IEB-80-11
Overhead Heavy Handling Systems		9.1.5	Generic Letter
Ventilation Systems	IX-5	9.4	
2. Functional			
Reactor Protection	IV-2 VII-1.A	7.2	
Safe Shutdown	VI-10.B VII-3	7.4	
Station Service and Cooling Water Systems	IX-3	9.2.1, 9.2.2	
Circulating Water	IX-3	10.4.5	
Control Systems	VII-4	7.7	IEB-79-27 USI A-47

*This table should not be considered an exhaustive listing because there are other reviews (SRP, SEP) which deal with the potential for systems interactions.

Table 3. Resource Requirements Summary

		FY 84	FY 85	FY 86
Contract Dollars for Technical Assistance (in thousands)		--	--	--
A-0445		650	--	--
A-3725		700	--	--
B-0789		230	--	--
Not Scheduled		200	300	--
NRR Manpower in Person Years				
DST	GIB	1.1	1.1	.5
	SPEB	.2	.3	.1
	RRAB	2.4	1.5	.1
	RSCB	.05	.05	.05
DSI	RSB	.30	.30	
	ICSB	.10	.10	
	CSB	.05	.05	
	ASB	.30	.10	
	PSB	.05	.05	
	CPB	--	--	
	AEB	--	--	
	ETSB	--	--	
	RAB	--	--	
DE	MEB	.05	.05	
	SEB	--	--	
	GSB	--	--	
	HGEB	--	--	
	MTEB	--	--	
	CHEB	.05	.05	
	EQB	.05	.05	
DHFS	HFEB	.10	.10	
	OLB	--	--	
	LQB	--	--	
	PTRB	.10	.10	
DL (Total)		.25	.15	
RES (Total)		.1	.1	
AEOD (Total)		.05	.05	
NMSS (Total		.1	.1	
OIE (Total		.05	.05	
		5.45	4.35	.75

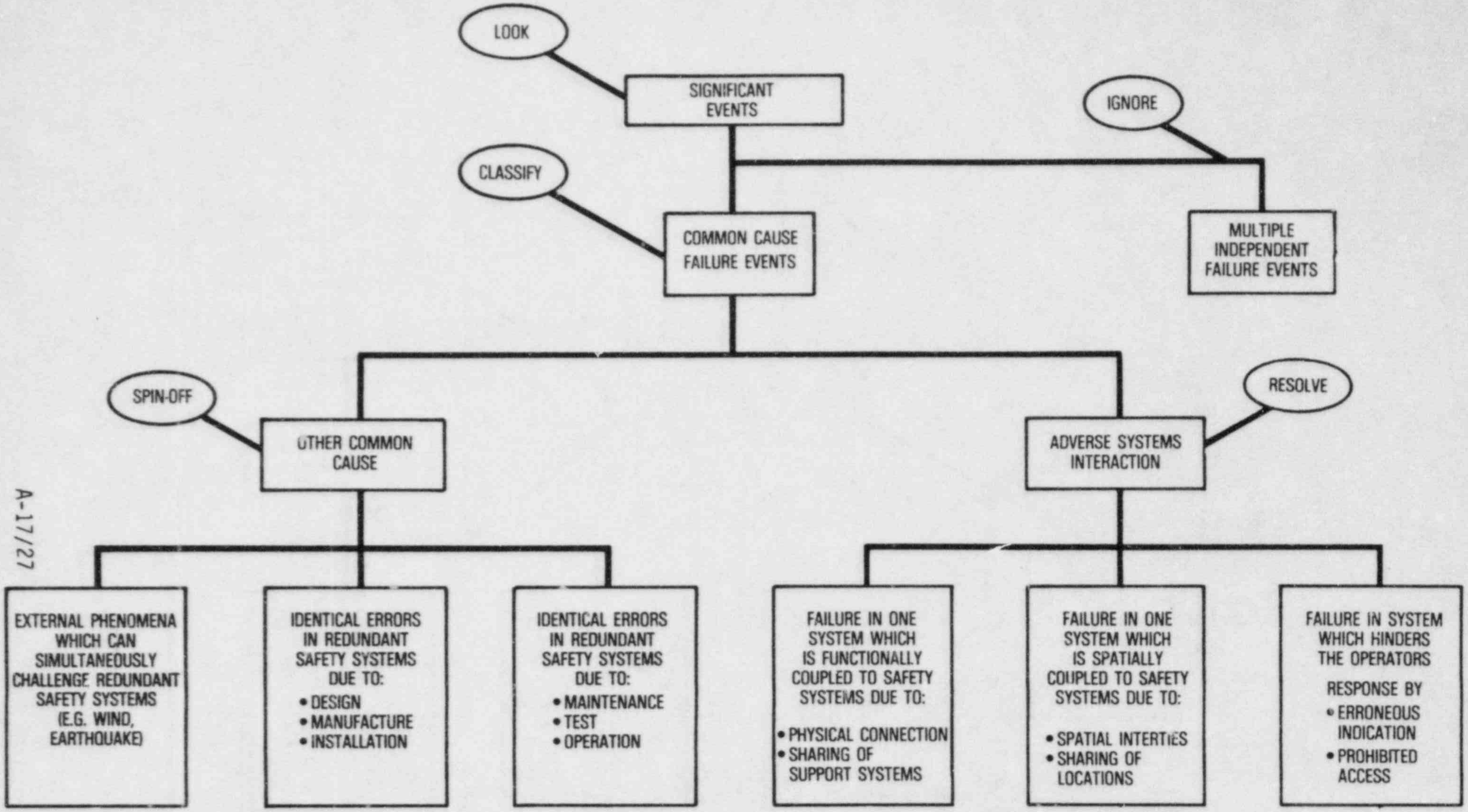

Figure 1. Definitions/Scope

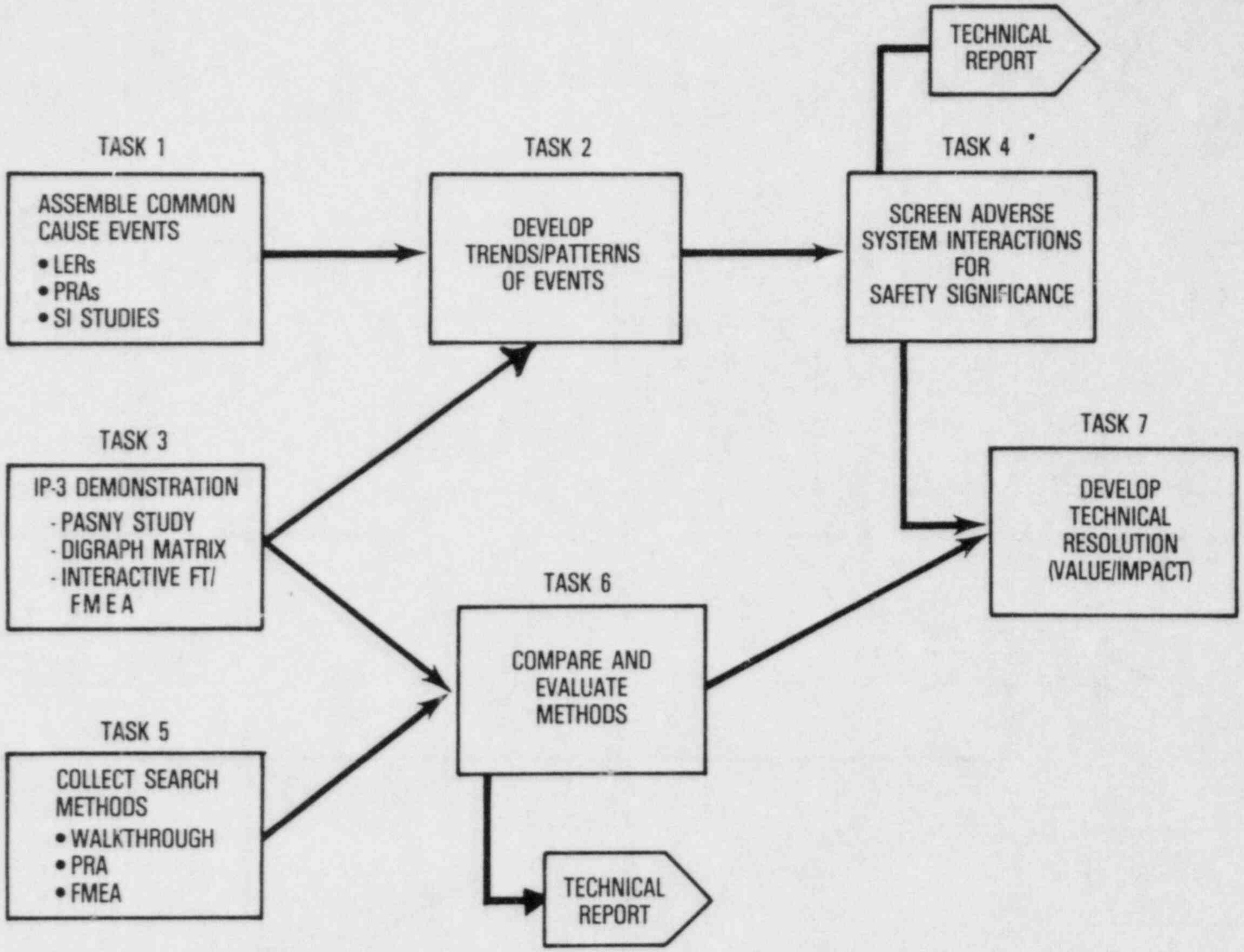

Figure 2. A-17 Task Flow

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13. Commonwealth Edison Company, "Zion Station Interaction Study," Zion Station, Docket 50-304.

14. IE Bulletin 79-27, "Loss of Non-Class-1-E Instrumentation and Control Power System Bus During Operation," November 30, 1979.

15. IE Bulletin 80-11, "Masonry Wall Design," May 8, 1980.

16. IE Information Notice No. 79-22, "Qualification of Control Systems," September 14, 1979.

17. Memorandum from T. E. Murley, NRC, to distribution, "The Approach to Systems Interactions in LWRs," June 25, 1981.

18. I. Papazoglou, et al., Brookhaven National Laboratory, Probabilistic Safety Analysis Procedures Guide, USNRC Report NUREG/CR-2815, January 1984.

19. Memorandum from H. R. Denton, NRC, to All NRR Employees, "NRR Office Letter No. 40 - Management of Proposed Generic Issues," March 19, 1983.

20. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following Accidents."

TASK ACTION PLAN
(March 1984)

SEISMIC DESIGN CRITERIA (TASK A-40)

Lead Organization:	Division of Engineering (DE) Structural & Geotechnical Engineering Branch (SGEB)
Technical Manager:	Nilesh C. Chokshi, SGEB, DE
Technical Supervisor:	David Jeng, Section Leader, SGEB, DE
Technical Consultant:	Professor M. Shinozuka Columbia University, New York
Lead Supervisor:	James P. Knight, Assistant Director for Components and Structures Engineering, DE
USI Task Manager:	Syed K. Shaukat Generic Issues Branch (GIB) Division of Safety Technology (DST)
USI Manager:	Karl Kniel, Chief, GIB, DST
NRR Principal Reviewers:	Goutam Bagchi Equipment Qualification Branch, DE
	Thomas Cheng Systematic Evaluation Program Branch Division of Licensing
	Phyllis Sobel Geosciences Branch, DE
Applicability:	All Reactor Types
Projected Completion Date:	January 1985

1. PROBLEM DESCRIPTION

The seismic design process required by current NRC criteria includes the following sequence of events.

A. Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).

B. Determine the free-field ground motion at the site that would result if the SSE occurred.

C. Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil for operating basis earthquake (OBE) and SSE.

D. Determine the motion of the plant equipment supported by the site structures for OBE and SSE.

E. Compare the seismic loads for OBE and SSE, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatisms for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRC's capability to evaluate the adequacy of seismic design of operating reactors and plants under construction, and (3) NRC's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

2. PLAN FOR PROBLEM RESOLUTION

The program for resolution of USI A-40 tasks consisted of two phases: (1) tasks concerning seismic input definitions and (2) tasks concerning the response of structures, systems and components. All technical work has been completed on both Phase I and Phase II. The only remaining task is to perform a value/impact analysis and develop supporting documents for presentation of the proposed requirements to the Committee to Review Generic Requirements (CRGR). The technical work accomplished on each of the tasks described below is summarized and evaluated in NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria." This report was prepared by Lawrence Livermore National Laboratory (LLNL) for the NRC. References to technical reports documenting completion of A-40 tasks and other pertinent technical work is included in NUREG/CR-1161. In developing the specific recommendations, the LLNL staff and their team of expert consultants considered all available and appropriate technical literature in addition to the results of USI A-40 tasks. In some cases, recommendations were developed by consensus of expert opinion and stem from unpublished data, research and experience.

An initial review of the recommendations by the NRC staff has been made and a proposed staff position developed. In some cases the recommendations have already been incorporated into the Standard Review Plan in a general revision in 1981.

A value/impact analysis has been completed for the assessment of the safety impact of implementing the Standard Review Plan changes proposed as a result of the technical findings. The supporting documents required for review by the CRGR will be prepared and the package which will include the technical findings (NUREG/CR-1161), the value/impact analysis and the proposed Standard Review Plan changes will be submitted to CRGR for review and approval. Following approval by CRGR, the proposed requirements will be submitted for public comment prior to implementing.

The USI A-40 tasks are presented below with a statement of the task objective. The reports and papers prepared as a result of these tasks are given in references to NUREG/CR-1161.

Task 1. <u>Quantification of Seismic Conservatism</u>

The objective of this task was to identify and quantify the conservatism in the following areas of the seismic design sequence:

- Regulatory Guide 1.60 response spectra
- Regulatory Guide 1.61 damping
- Soil-structure interaction
- Response to three components of motion
- Broadening of spectral peaks
- Structural and mechanical resistance
- Nonlinear structural response
- Sub-system response
- OBE vs. SSE response
- Overall conservatism of the seismic design process

The technical assistance contractor concluded, as A-40 tasks progressed, that much more research was needed to quantify the conservatism in the seismic design process.

Task 2. Elasto-Plastic Seismic Analysis

This study was undertaken to evaluate a typical braced steel frame of a power plant for reserve capacity from nonlinear effects and to determine the effect of supported equipment and piping on the overall response.

Task 3. Site-Specific Response Spectra

The objective of this task was to develop a method for developing spectral shapes that are realistic, not overly conservative, and that account for specific site characteristics.

Task 4. Seismic Aftershocks

The objective of this task was to assess the possibility that aftershocks, though less severe than the main earthquake, may result in additional damage to the structures, systems, or components that are allowed to respond inelastically during the SSE. Preliminary investigation indicated that available data are very limited, and it was decided that the inelastic SSE response will be limited to a small fraction of the available ductility for reevaluation of existing designs. As a result, this task was subsequently cancelled.

Task 5. Nonlinear Structural Dynamic Analysis Procedures for Category I Structures

This task consisted of an investigation of the feasibility of using simplified nonlinear dynamic analysis techniques for the design of typical Category I structures by comparing the results of various simplified techniques with those from more rigorous, nonlinear, time-history dynamic analyses.

Task 6. Soil-Structure Interaction

The soil-structure interaction (SSI) procedures and corresponding definition of seismic input now used in the seismic analysis of nuclear power plants were examined to determine limits and conditions of applicability as well as conservatism in currently used SSI procedures.

Task 7. Earthquake Source Modeling

The objective of this task was to develop criteria for determining the adequacy of modeling techniques proposed by applicants to assess ground motion near faults.

Tasks 8 and 9. Analysis of Near-Source Ground Motion

These tasks were to be separately carried out in the beginning, but as the work progressed they were combined into one.

The objective of these tasks was to develop methodology for determining ground motion response spectra in the strong motion, near-field region of earthquake sources.

Task 10. Review and Implementation

The objective of this task was to provide a technical review of the results of the first nine tasks in Task Action Plan A-40 and to recommend changes to existing NRC criteria based on that review.

Technical review of the recommendations is complete. Many of the recommendations were approved by the reviewing branches; some were rejected for various reasons (see NUREG/CR-1161). Those recommendations which have been approved have either been implemented in present Standard Review Plans or will be proposed for implementation.

3. BASIS FOR CONTINUED PLANT OPERATION PENDING IMPLEMENTATION OF PROPOSED REQUIREMENTS

All technical tasks have been completed and the contractor has made specific recommendations for changes to seismic design criteria. The intent of these recommendations is to clarify certain design procedures and to bring seismic design criteria up to the state-of-the-art. The results of these studies did not indicate that backfitting of older operating plants or those plants designed and/or constructed to current seismic design criteria was warranted except for the above ground free standing tanks. A 50.54f letter to all licensees and applicants is planned to be issued for collection of information about tanks to enable the staff to review and determine the adequacy of such tanks. The proposed changes are intended to apply only to new license applications.

4. OFFICE OF NUCLEAR REACTOR REGULATION (NRR) TECHNICAL ORGANIZATION INVOLVED

This section indicates the responsibilities of each NRR Branch in supporting USI A-40 until final disposition of the proposed requirements.

A. Structural and Geotechnical Engineering Branch, Division of Engineering, has lead responsibility for implementation of proposed changes and for maintaining schedule for completion of all work.

Manpower Estimate:	FY-83	0.5 psy*
	FY-84	0.5 psy
	FY-85	0.4 psy

B. Equipment Qualification Branch, Division of Engineering, will provide review of proposed implementation and assist in resolution of comments.

Manpower Estimate:	FY-83	0.1 psy
	FY-84	0.1 psy
	FY-85	0.02 psy

C. Geosciences Branch, Division of Engineering, will provide review of proposed implementation and assist in resolution of comments.

Manpower Estimate:	FY-83	0.2 psy
	FY-84	0.1 psy
	FY-85	0.1 psy

D. Systematic Evaluation Program Branch, Division of Licensing, will provide review of proposed implementation and assist in resolution of comments.

Manpower Estimate:	FY-83	0.1 psy
	FY-84	0.1 psy
	FY-85	0.03 psy

E. Generic Issues Branch, Division of Safety Technology will provide budgeting and scheduling and overall coordination of A-40 activities and assist Structural and Geotechnical Engineering Branch in management of all remaining work.

Manpower Estimates:	FY-83	0.5 psy
	FY-84	0.5 psy
	FY-85	0.4 psy

The following table presents a summary of staff resource requirements necessary to complete the USI A-40 task.

*Assumed 1 professional staff year (psy) = 40 person weeks.

Resource Requirements Summary

		FY-83	FY-84	FY-85
Contract Dollars For Technical Assistance (in thousands)		21.0	114.0	
NRR Manpower in person-years				
DST	GIB	0.5	0.5	0.4
	SPEB	0.05	0.03	
	RRAB	0.07	0.03	
DE	EQB	0.1	0.1	0.02
	SGEB	0.5	0.5	0.4
	GSB	0.2	0.1	0.1
DOL	SEPB	0.1	0.1	0.03

5. SCHEDULE

Task 1	Quantification of Seismic Conservatisms	Complete
Task 2	Elasto-Plastic Seismic Analysis	Complete
Task 3	Site-Specific Response Spectra	Complete
Task 4	Seismic Aftershocks	Cancelled
Task 5	Nonlinear Structural Dynamic Analysis Procedures for Category I Structures	Complete
Task 6	Soil-Structure Interaction	Complete
Task 7	Earthquake Source Modeling	Complete
Tasks 8&9	Analysis of Near Source Ground Motion	Complete
Task 10	Review and Implementation	Complete

Major Milestones

Value/Impact Analysis	08-01-83
Prepare Risk Assessment	08-01-83
Prepare Initial CRGR Package	09-16-83
NRR Review of Initial CRGR Package	10-20-83
CRGR Review of Initial Package	04-24-84
Issue for Public Comments	05-23-84
NRR Review of Resolved Public Comments	11-09-84
CRGR Review of Final Package and Decision	12-10-84
Issue Standard Review Plans and Final NUREG	01-10-85

TASK ACTION PLAN
(March 1984)

STATION BLACKOUT (TASK A-44)

Lead Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
Task Manager:	Alan M. Rubin, GIB, DST Patrick W. Baranowsky, RRBR, RES
Lead Supervisor:	Karl Kniel, Chief, GIB, DST
NRR Principal Reviewers:	Leon Engle Operating Reactors Branch No. 3 Division of Licensing
	J. T. Beard Operating Reactor Assessment Branch Division of Licensing
	P. Om Chopra Power Systems Branch Division of Systems Integration
	Raj Anand Auxiliary Systems Branch Division of Systems Integration
	David Langford/Sammy Diab Reactor Systems Branch Division of System Integration
	Scott Newberry Reliability and Risk Assessment Branch Division of Safety Technology
Applicability:	Pressurized Water Reactors and Boiling Water Reactors
Projected Completion Date:	May 1985

1. DESCRIPTION OF PROBLEM

A. Statement of Issue

The complete loss of alternating current (ac) electrical power to the essential and nonessential switchgear buses in a nuclear power plant is referred to as a "Station Blackout." Because many safety systems required for reactor core decay heat removal are dependent on ac power, the consequences of a station blackout could be a severe core damage accident. The technical issue involves the likelihood and duration of the loss of all ac power and the potential for severe core damage after a loss of all ac power.

B. Background

The issue of station blackout arose because of the historical experience regarding the reliability of ac power supplies. There have been numerous reports of emergency diesel generators failing to start and run in operating plants. In addition, a number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. In almost every one of these loss of offsite power events, the onsite emergency ac power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases there has been a complete loss of ac power, but during these events, ac power was restored in a short time without any serious consequences.

The results of the Reactor Safety Study (Ref. 1) showed that for one of the two plants evaluated, a station blackout accident could be an important contributor to the total risk from nuclear power plant accidents. Although this total risk was found to be small, the relative importance of station blackout accidents was established. This finding and the concern for diesel generator reliability based on operating experience raised station blackout to an Unresolved Safety Issue (USI).

C. Purpose

The purpose of this task is to evaluate the adequacy of current licensing design requirements to assure that nuclear power plants do not pose an unacceptable risk from a station blackout event.

The NRC safety design requirements applicable to station blackout can be grouped into three categories:

1. reliability of the offsite ac power supplies;
2. reliability of the onsite emergency ac power supplies; and
3. capability of plants to remove decay heat with ac power supplies unavailable.

Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence (Category 1 above). As such, it is required that an independent emergency onsite ac power supply be provided at nuclear plants. It is further required by NRC safety criteria that onsite electric power for safety systems at nuclear plants be supplied by at least two redundant and independent divisions (Categories 1 and 2). Each electrical division for safety systems includes an offsite ac power connection, an onsite emergency ac power supply (usually a diesel generator), and direct current (dc) power sources. Those safety systems required to remove decay heat from the reactor core following shutdown are required to have available these diverse ac power supplies. Surveillance requirements include periodic testing for emergency diesel generators (Category 2) and other related electrical equipment. Additional requirements are that diverse power drives and supporting systems independent of ac power must be provided for one emergency feedwater train in Pressurized Water Reactors (PWRs) (Category 3). The design practice for Boiling Water Reactors (BWRs) is to include at least one decay heat removal system (that is, reactor core isolation cooling) driven by a source independent of ac power (Category 3).

2. PLAN FOR RESOLUTION

A. Approach

Technical analyses in all three of the above categories are included in this task. However, the principal focus is on Category 2, reliability of emergency ac power supply. This is justified by several considerations. First, the questions raised about Category 2 were basically responsible for identifying station blackout as a safety issue. Second, if safety improvements are required, it is easier to analyze and identify them and implement them in Category 2 rather than in Categories 1 and 3. For example, offsite power reliability (Category 1) is dependent on a number of factors which are difficult to analyze and to control, such as regional electrical grid stability, weather phenomena, local industrial and population growth, and repair and restoration capability. Also, the capability of a plant to withstand a station blackout (Category 3) would require many decay heat removal-related systems, components, instruments and controls to be independent of ac power. These will vary from plant to plant, requiring considerable effort to analyze all of them and to assure that the plants indeed have that capability. Third, some progress has been made in Category 3. A significant improvement is under way for all operating PWRs by backfitting the auxiliary feedwater system to make it independent of ac power. In addition, USI A-45 is reviewing the adequacy of shutdown decay heat removal systems for nuclear power plants. Thus, the reliability of emergency ac power supplies is of principal importance to A-44.

During the development of this task action plan, a preliminary screening analysis was begun to identify plants most likely to suffer core damage due to a loss of all ac power supplies. The intent of this work was to survey the frequency and implication of station blackout accidents in operating plants and identify any especially high risk plants which might require further analysis or action on an urgent basis. Initial results showed no such plants. Completion of this task was the first step in resolving this issue.

A more detailed evaluation of station blackout concerns followed the completion of the preliminary analysis. It is recognized that this issue is centered around a concern for the adequacy of ac power supply reliability, especially for emergency onsite ac power supplies. As such, this area comprises the major program effort to resolve this issue. Typical offsite and emergency ac power supplies have been evaluated including a review of past operating (failure) experience. This effort is limited to power supply availability and does not include an evaluation of power distribution network adequacy or power capacity requirements.

In order to provide a consequence perspective, tasks to evaluate station blackout accident sequences and associated plant response analyses are included. The Interim Reliability Evaluation Program (IREP) was used as a primary information source in developing the shutdown cooling reliability models and accident scenarios needed to perform these tasks.

Upon completion of the technical evaluation tasks, a NUREG report documenting the results of the technical studies of this program will be published and a regulatory position will be developed for review and comment.

B. Technical Content of Major Subtasks

Task 1. <u>Preliminary Screening Analysis of Operating Plants</u>

A probabilistic safety assessment was performed and documented to provide a preliminary evaluation of station blackout accident sequences at operating nuclear power plants. The purpose of this work was to effectuate a screening analysis to identify any plants of unusually high susceptibility to station blackout and subsequent core damage. This task is complete (see Reference 2).

Task 2. <u>Alternating Current Power Supply Reliability Evaluation</u>

Failure modes and reliability analyses were performed for typical offsite and emergency ac power supplies. These analyses included an indepth examination of the potential causes, frequency and duration relationships for station blackouts. The ac power supply reliability subtasks include:

Subtask 2.1. Alternating Current Power Supply Design Review

Typical offsite and emergency ac power supply configurations have been identified and grouped generically. Consideration was given to type of power source, line diagrams showing redundancy and switching, plant systems supplied by each bus/division, ac power dependence on dc power, and operational characteristics.

Subtask 2.2. Alternating Current Power Supply Operating Experience Review

The operational experiences regarding loss of offsite power and emergency ac power supplies (particularly diesel generators) were reviewed. This included the identification of data needs and the collection of information. Knowledge gained from previous studies of offsite and emergency ac power supply reliability was included. The intent of this task was to obtain enough operational experience information to allow the construction of meaningful reliability models with due consideration to the limitations of such models.

Subtask 2.3. Reliability of Alternating Current Power Supplies

A reliability analysis of the typical ac power supply configuration was performed. Both offsite and onsite power supplies were modeled with special consideration given to interactive and common cause failure modes, including those induced by human error. The effect of regional and local factors on the loss and recovery of ac power were considered where possible. Aspects of design and operation which have the potential to improve ac power supply reliability have been identified and the amount of improvement estimated. Design and operational recommendations to assure ac power supply reliability were developed.

Task 2 has been completed (see References 3 and 4).

Task 3. Accident Sequence Analysis

An investigation into the probability and consequence of station blackout accidents was conducted through both generic and plant-specific studies. The insights gained from the IREP program were used to enhance the limited detail of the generic evaluations. These studies included the reliability of shutdown cooling systems given a loss of ac power supplies, an evaluation of the hazards posed by extended blackouts, and reactor coolant inventory requirements during station blackouts. These considerations were coupled with the results of Task 2 to identify a generic set of dominant station blackout accident scenarios. The subtasks for this evaluation include:

Subtask 3.1. Accident Sequence Review

Event and fault tree analyses were reviewed to identify dominant station blackout sequences, failure modes, and consequences. These include the Crystal River 3 analyses and IREP studies. This information supplemented that currently available from the Reactor Safety Study and follow-on studies.

Subtask 3.2. Shutdown Cooling Reliability

A generic review of systems and components used for shutdown cooling was performed to identify ac power dependencies and requirements, adequacy of ac-independent systems, and the reliability of these systems during a station blackout. The system reliability results obtained from accident sequence reviews were considered in this subtask.

Subtask 3.3. Generic Accident Sequence Evaluation

A set of generic event trees was developed and the dominant station blackout accident scenarios characterized. The probabilities and consequences of these scenarios were used to provide a simplified risk perspective. This information was used to establish acceptable requirements for ac power supply reliability and decay heat removal capability for station blackout.

Task 3 has been completed (see Reference 5).

Task 4. Plant Response to Station Blackout

Reactor coolant system response analyses were performed for station blackout accident scenarios. Typical Nuclear Steam Supply System

(NSSS) designs were analyzed to provide an estimate of the core damage times and to determine the important operational characteristics associated with these accidents. The subtasks for this work are:

Subtask 4.1. Develop Plant Response Models

Generic and plant-specific response characteristics were considered in the development of analysis models for each Light Water Reactor (LWR) vendor. A preliminary and simplified event tree and accident scenario list were used to determine the modeling requirements. Models included best estimates where possible using existing computer codes.

Subtask 4.2. Analysis Matrix

An initial accident analysis matrix was developed from simplified event trees. The accident sequence evaluations of Task 3 and initial accident sequence analysis results were used to revise the accident analyses matrix into a final set of plant response analyses which provide a characterization of reactor thermal response for station blackout accidents.

Subtask 4.3. Plant Response Analyses

Analyses were performed for each LWR NSSS vendor to assess the time dependence and consequences of station blackout accident sequences (such as mitigation by adequate core cooling or damage to the core and possible melting). These results were reviewed to identify important system or component availability and operational characteristics, including operator actions.

Task 4 has been completed (see Reference 5).

Task 5. Licensing Requirements

The results of Tasks 1 through 4 are being used to develop any licensing requirements which may be needed to resolve this issue. A value/impact analysis is being used to provide a basis for the recommendations made to resolve this issue. The development of a staff NUREG covering the technical findings of this program and appropriate internal and public review of the draft report are included in this task.

C. Management of Work

The responsibility for carrying out a program to resolve this issue was transferred to the Office of Nuclear Regulatory Research (RES) by memorandum dated July 13, 1979, from the Director of the Office of Nuclear Reactor Regulation (NRR) to the Director of RES. The Reactor Risk Branch (RRBR) of RES provided the program management through the completion of technical assistance contracts; however, NRR remained cognizant through assignment of liaison personnel and participation in subtasks as identified in this task action plan. NRR has program management responsibility for regulatory actions stemming from the results of the technical findings. NRR funded and contracted for the necessary technical assistance. In addition, NRR had the responsibility of obtaining and providing to the Task Manager operating experience information required from licensees as identified in this plan. NRR also has the responsibility of taking licensing related actions on station blackout issues during the conduct of this program.

D. Schedule

The following schedule has been developed for the completion of the major tasks of this program:

1. Interim Study

Final report	Complete

2. Alternating Current Power Reliability

Power supply design review	Complete
Operating experience evaluation	Complete
Reliability evaluation	Complete

3. Accident Sequence Analysis

IREP review	Complete
Shutdown cooling reliability	Complete
Accident sequence evaluation	Complete

4. Plant Response to Station Blackout

Plant response models	Complete
Analysis matrix	Complete
Plant response analyses	Complete

5. Licensing Position

Initial recommendations for staff comment	Complete
Issued for public comment	July 1984
Final technical resolution	May 1985

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

As stated in Section 1, Description of Problem, the purpose of this task is to evaluate the adequacy of current licensing design requirements regarding the risk of a station blackout accident resulting in unacceptable core damage. In particular, the adequacy of emergency ac power supplies reliability has been questioned. The current licensing criteria require licensees to provide redundant emergency ac power supplies, to demonstrate emergency ac power supply reliability (Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units as Onsite Electric Power Systems at Nuclear Power Plants"), and to include the capability of removing decay heat using at least one shutdown cooling train independent of ac power.

In the event of a total loss of ac power at PWRs, the auxiliary feedwater (AFW) system can provide a heat sink via the steam generators to remove the core decay heat. Since the Three Mile Island 2 accident and subsequent studies further highlighted the importance of the AFW systems, the Bulletins and Orders Task Force performed a review of these systems for operating Combustion Engineering (Reference 6) and Westinghouse-designed PWRs. The objectives of this study were to: (1) identify necessary changes in AFW system design or related procedures to assure continued safe operation, and (2) to identify other system characteristics in the AFW system design of these plants which, on a long term basis, may require system modification. Based on this study, the Bulletins and Orders Task Force made a number of recommendations to improve the reliability of the AFW systems. Some of these recommendations were specifically made to cover the concern for the total loss of offsite and onsite ac power. For the near term, the Bulletins and Orders Task Force required that as-built plants be capable of providing the required AFW flow for at least 2 hours from one AFW pump train independent of any ac power source. For the long term, it is required that this function be performed automatically in addition to various other improvements. Modifications to the AFW system in response to these requirements have been completed at all PWRs.

The reliability of the AFW systems for the Babcock and Wilcox operating PWRs was reviewed as part of the May 1979 shutdowns for these plants. This review resulted in various short-term system and emergency procedure modifications to improve the availability of these systems. A more systematic reliability review of these plants is now in progress. These plants will also be required to meet the long term requirements discussed above.

BWRs contain various systems to remove core decay heat following the total loss of ac power. These systems include the isolation condensers on BWR/1 through BWR/3 plants and the steam-driven high pressure coolant injection and reactor core isolation cooling system. For BWR/1, BWR/2 and early BWR/3 plants, the isolation condenser will provide an adequate heat sink for a minimum of 40 minutes. For other BWRs, adequate cooling can be maintained for approximately 2 hours. The Bulletins and Orders Task Force did not require any specific improvements for these systems following its review; however, a review of BWRs is included in this study.

In addition to the above, a preliminary study of operating plants was performed to assess plant vulnerability using probabilistic techniques. This study did not identify any plants of unusually high susceptibility to a severe core damage accident resulting from a station blackout. Accordingly, it is concluded that plants may continue to be licensed and operated while the evaluation of station blackout is ongoing. After completion of the evaluation, any additional interim licensing requirements will be identified.

4. ASSISTANCE FROM RES

The Division of Accident Evaluation (DAE) has provided assistance in developing reactor coolant system response characteristics identified in Task 4. The RRBR staff provided direction to DAE on appropriate accident scenarios to be analyzed. Funding and program management of contractor efforts in this area was provided by DAE. Continued technical assistance will be required from RRBR during the technical resolution phase of this issue.

Manpower Requirements (person-weeks)

FY 1984	FY 1985
50	30

5. ASSISTANCE REQUIRED FROM OTHER NRR DIVISIONS

A. Division of Licensing (DL). Provided the coordination necessary to expedite the collection of required operating reactor experience and design data. Information needs have been related to the reliability assessments for offsite power, emergency ac power (primarily emergency diesel generators), and shutdown cooling systems. DL will contribute to the review and approval of interim and final licensing positions.

Manpower Requirements (person-weeks)

	FY 1984	FY 1985
Operating Reactors Branch No. 3	4	4
Operating Reactors Assessment Branch	4	4

B. Division of Systems Integration (DSI). Provided review and comment on the technical evaluations provided by the Task Manager in the areas of instrumentation and control, electrical and power systems, reactor and auxiliary systems, containment systems, and systems interactions. DSI provided assistance in the identification of design and operational characteristics of ac power supplies and systems required for shutdown cooling. In addition, DSI will contribute to the review and approval of interim and final licensing positions.

Manpower Requirements (person-weeks)

	FY 1984	FY 1985
Power Systems Branch	6	6
Reactor Systems Branch	4	4
Auxiliary Systems Branch	4	4
Containment Systems Branch	4	4

C. Division of Human Factors Safety (DHFS). Provided review and comment on those technical evaluations involving man/machine interfaces. In this area, DHFS will contribute to the review and approval of interim and final licensing positions.

Manpower requirements (person-weeks)

	FY 1984	FY 1985
Human Factors Engineering Branch	2	2
Procedures and Systems Review Branch	2	2

D. Division of Safety Technology (DST). Provides liaison between NRR and RES, and provides general assistance in coordination of activities performed within NRR which are part of this Task Action Plan. DST has primary responsibility for the initial review of draft licensing recommenations and for coordination of the internal management and public review process required to adopt the final licensing positions. DST will also coordinate the formal revision and publication of licensing documents (that is, Regulatory Guides, Standard Review Plans, etc.).

Manpower Requirements (person-weeks)

	FY 1984	FY 1985
Generic Issues Branch	50*	30*
Reliability and Risk Assesment Branch	4	4
Safety Program Evaluation Branch	2	2

*Reflects Generic Issues Branch Task Management responsibility.

6. TECHNICAL ASSISTANCE

Direct technical assistance to the program was required for Tasks 2 and 3. Funding was provided by NRR. Technical assistance requirements for Task 4 were funded directly by the Division of Risk Analysis, RES and the Division of Safety Technology, NRR. The following is a brief description of the technical assistance for Tasks 2 and 3 for this program.

A. Offsite Power Reliability

1. Contractor - Oak Ridge National Laboratory (ORNL)

2. NRC managing organization - RRBR, RES.

3. Scope - Identify initiating events which can cause a loss of offsite power, evaluate the expected frequency, and determine dominant factors affecting the reliability of offsite power supplies and the recovery of offsite power. This includes consideration of power supply and circuit configurations, operational characteristics (technical specifications, limiting conditions of operation, operating procedures, human interactions), and location dependent factors (multiple unit sites, proximity to alternate power supplies, regional grid reliability). In the context of these considerations, operating experience data were evaluated, reliability models were developed, and reliability estimates were provided. Features which may improve the reliability of offsite power supplies were also evaluated.

4. Funds expended - $300K.

B. Emergency Alternating Current Power Reliability

1. Contractor - ORNL

2. NRC managing organization - RRBR, RES.

3. Scope - Identify range of emergency ac power supply design configurations used at nuclear power plants. Collect and analyze operating experience data. Quantify probabilities of dominant emergency power supply failure modes. Review experience at several operating nuclear plants. Review emergency power supply reliability experience from other applications such as the Department of Defense and the Federal Aviation Administration. Develop predictive reliability models for emergency ac power supplies including component and design differences, operational characteristics, and power supply recovery from failure. Identify practical reliability improvements and quantitative reliability goals. Earlier NRR qualitative studies and other studies were reviewed and incorporated. Estimate reliability increases possible and associated costs.

4. Funding requirements - $400K.

C. Station Blackout Accident Sequence Evaluation

1. Contractor - Sandia

2. NRC managing organization - RRBR, RES.

3. Scope - Develop generic event trees, characterize dominant accident scenarios, and provide a risk/consequence perspective for station blackout accidents. A review of IREP accident sequences and shutdown cooling systems reliability associated with a station blackout was conducted to supplement the generic evaluations. The results of the offsite and emergency ac power supply reliability studies were used in conjunction with the generic accident sequence and shutdown cooling reliability assessment to provide station blackout accident perspectives.

4. Funds expended - $300K.

7. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

Interaction with outside organizations could include the Electric Power Research Institute, Nuclear Safety Analysis Center, Institute of Nuclear Power Operations, Federal Energy Regulatory Commission, Federal Aviation Administration, utilities, Nuclear Steam Supply System vendors, Architect Engineers, and emergency diesel generator manufacturers. Peer review is conducted through periodic briefings of the Advisory Committee on Reactor Safeguards.

8. POTENTIAL PROBLEMS

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

A. Identification of reliability goals and translation of probabilistic results into licensing requirements.

B. Development of licensing requirements that take into account the many pertinent plant-specific variables that impact station blackout.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Reactor Safety Study," USNRC Report WASH-1400, NTIS, October 1975.

2. Memorandum from P. Baranowsky, NRC, to K. Kniel, NRC, "Completion of Station Blackout (USI A-44) Task 1," dated May 22, 1981.

3. U.S. Nuclear Regulatory Commission, "Reliability of Emergency AC Power Systems at Nuclear Power Plants," USNRC Report NUREG/CR-2989, July 1983.

4. ORNL Letter Report, "Loss of Offsite Power at Nuclear Power Plants," October 1983.

5. U.S. Nuclear Regulatory Commission, "Station Blackout Accident Analyses," USNRC Report NUREG/CR-3226, May 1983.

6. U.S. Nuclear Regulatory Commission, "Report of the Bulletins and Orders Task Force," USNRC Report NUREG-0645, January 1980.

Summary of Resource Requirements for USI A-44

		FY 84	FY 85
Contract Dollars ıor Technical Assistance		$85K	--
NRR Manpower (person-years)			
DST	GIB	1.0	0.6
	SPEB	.04	.04
	RRAB	.08	.08
DSI	RSB	.08	.08
	CSB	.08	.08
	ASB	.08	.08
	PSB	.10	.08
DHFS	HFEB	.04	.04
	PTRB	.04	.04
DL	ORB3	.08	.08
	ORAB	.08	.08
Other			
RES	RRBR	1.0	0.6

TASK ACTION PLAN, REVISION 3
(March 1984)

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (TASK A-45)

Lead Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
Task Manager:	Andrew R. Marchese GIB, DST
Lead Manager:	Karl Kniel, Chief, GIB, DST
NRR Principal Reviewers:	L. Marsh/C. Liang Reactor Systems Branch, DSI
	P. Hearn Auxiliary Systems Branch, DSI
	R. Ferguson Chemical Engineering Branch, DE
	G. Staley Environmental & Hydrologic Engineering Branch, DE
	D. DiIanni Operating Reactors Branch 3, DL
	S. Bryan Procedures & Systems Review Branch, DHFS
Office for Analysis & Evaluation of Operational Data	H. Ornstein AEOD
Office of Nuclear Regulatory Research	M. Cunningham Reactor Risk Branch, Division of Risk Analysis
Applicability:	Light Water Reactors (Pressurized and Boiling Water Reactors)
Projected Completion Date:	Contractor Final Report, February 1985 Final NUREG Report Issued by NRR, February 1986

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: (1) to maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel, and (2) to provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink. The reliability of a particular power plant to perform these functions depends on the frequency of initiating events that require or jeopardize decay heat removal operations and the probability that required systems will respond when needed to remove the decay heat.

One of the most crucial factors in the safety of nuclear reactors is the reliability of the systems used for decay heat removal following the shutdown of the reactor for any reason. The results of the Reactor Safety Study (WASH-1400) indicated that the overall probability of core meltdown in the first generation of large commercial LWRs was about 50 times higher than had been expected in WASH-1270 (about 5×10^{-5} as compared to 1×10^{-6} per reactor year). Insufficient reliability of the decay heat removal systems, particularly in response to small-break loss-of-coolant accidents (LOCAs), was shown to be responsible for a substantial portion of the overall probability of core meltdown.

The prinicipal means for removing the decay heat in a pressurized water reactor (PWR) under transient conditions (loss of main feed, loss of offsite power) 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 accident at Three Mile Island Unit 2 (TMI-2) 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 accident.

Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the NRC staff believes that providing an alternative means of decay heat removal could substantially increase the plant's capability to deal with a broader spectrum of transients and accidents and, therefore, could potentially reduce overall risk to the public. Accordingly, under Task A-45, the NRC staff is investigating alternative means of decay heat removal in PWR plants, using existing equipment or devising new methods. This Unresolved

Safety Issue (USI) will also entail investigation of 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 shutdown after operating at power for some time, the effect on the subsequent operating procedures for maintaining safe conditions of four 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 less than 350°F in a typical PWR).

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 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 and volume control system and depressurization systems, are also considered.

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

(a)	Reflood phase -	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 shutdown conditions.
(d)	Decay Heat - Removal (DHR) phase	SDHR and RHR phases combined.

To provide a clear understanding of the terms and distinction involving various stages of operation and shutdown, the following definitions, which are provided below in Tables 1 and 2, will be utilized in this plan.

Decay Heat Removal Systems (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 to an ultimate heat sink following shutdown of the reactor for normal events, off-normal transient events (for example, loss of offsite power, loss of main feedwater) and small-break LOCAs, described as "S2" in the Reactor Safety Study (that is, 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 10 minutes following medium or large LOCAs. However, it is necessary in Task A-45 to consider the supporting systems (for example, the chemical and volume control system, depressurization systems, component cooling water systems, and the essential service water 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 the above definitions are used rigorously in this Task Action Plan (for example, 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 LWR there are three broad groups of fault sequences which can lead to severe damage to the fuel, namely:

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.

Table 1. Typical PWR Operational Modes

	Mode	Reactivity Condition, K_{eff}	% Rated Thermal Power*	Average Coolant Temperature
1.	Power Operation	≥ 0.99	>5%	$\geq 350°F$
2.	Startup	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3.	Hot Standby	<0.99	0	$\geq 350°F$***
4.	Hot Shutdown	<0.99	0	$350°F > T_{avg} > 200°F$
5.	Cold Shutdown	<0.99	0	$\leq 200°F$
6.	Refueling**	≤ 0.95	0	$\leq 140°F$

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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

Table 2. Typical BWR Operational Conditions

	Condition	Mode Switch Position	Average Reactor Coolant Temperature
1.	Power Operation	Run	Any temperature
2.	Startup	Startup/Hot Standby	Any temperature
3.	Hot Shutdown	Shutdown	>200°F
4.	Cold Shutdown	Shutdown	≤200°F
5.	Refueling*	Shutdown or Refuel	≤140°F

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

2. Failure of the reactor to shut down correctly (that is, the Anticipated Transient Without Scram 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 (for example, due to loss of primary coolant or lack of main and 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 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. 2 and 3) 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 SDHRS (including those required in post-reflood conditions) alone, is about five. 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 significant 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 RHR systems 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 RHR systems 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 isolation 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; and in the third class of sequences, 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 the order of 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 generators, and from these to an ultimate heat sink.

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

(i) SDHR by the so-called "feed and bleed" procedure, and

(ii) SDHR by operation of the steam generators 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 severe environmental 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 and impact of upgrading in 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 (for example, 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 fault sequences described above. Resolution of this question is considered to be of sufficient importance to merit raising it to the status of an 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, USI 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 USI A-44 (Ref. 4). There are a number of other activities (Ref. 5) in which work conducted, or sponsored by NRC and by other organizations is proceeding that relates to the present Task Action Plan. Those activities have been taken into account in formulating this 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 and Objectives

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements to ensure that LWRs do not pose unacceptable risk as a result of DHR system failures. The specific objectives of the study include:

- Assess the safety adequacy of DHR systems in existing power plants for achieving both hot shutdown and cold shutdown conditions.

- Evaluate the feasibility of alternative measures for improving DHR system reliability, including diverse alternatives dedicated to the DHR function.

- Assess the value and impact (or cost-benefit) of the most promising alternative measures.

- Develop a plan for implementing any proposed new licensing requirements for DHR systems.

In order to accomplish these objectives, numerous tasks have been identified as part of the Task A-45 program, including system reliability assessments, DHR system engineering feasibility studies, thermal-hydraulic analyses, power plant characterization efforts, emergency DHR operating procedure reviews, and evaluations of the vulnerability of DHRS to special emergencies (for example, fire, flood, earthquake, sabotage).

3. PLAN FOR PROBLEM RESOLUTION

A. Approach to the Problem

Consistent with the above objectives, four major tasks and numerous supporting subtasks are identified. The approach taken to this problem comprises the following main elements:

- Development of acceptance criteria for assessment of DHR systems.

- Development of means for improvement of DHR function.

    - Assessment of adequacy of DHR systems in existing plants.

    - Development of a plan for implementing proposed new requirements for DHR systems.

Each of these elements constitutes a major task, the technical content of which is described below. The interrelation of each of the main elements is shown in Figure 1. The schedule for all work included in this plan is provided in Part D (Schedule) of this section.

B. Technical Content of Individual Tasks

A description of the individual tasks and subtasks is provided in this section. Figure 2 provides a flowchart of the Task A-45 program which shows the interrelation of the major subtasks, along with some key milestone dates in the program. For a better understanding, the reader may want to make frequent reference to Figure 2 while reading the description of each subtask below.

TASK 1. Development of Acceptance Criteria for Assessment of DHR Systems

Quantitative and qualitative acceptance criteria for DHR systems in existing and future LWR power plants will be developed.

Task 1.1 Development of Quantitative Acceptance Criteria for DHR Systems in Existing and Future Plants

Subtask 1.1.1 Identify DHR System Vulnerabilities to Random, Operator and Common-Mode Failures

Based on a review of completed U.S. and foreign probabilistic risk assessments (PRAs), systems analyses, and representative emergency operating procedures, those system and procedural characteristics which most often contribute to the unavailability of DHR systems will be identified for random, operator, and common-mode failures (for example, steam generator tube failures and pressurized thermal shock). This subtask will use the results of completed quantitative probabilistic analyses to identify, in a qualitative fashion, DHR system vulnerabilities.

Subtask 1.1.2 Establish a Quantitative Probabilistic Goal for DHR System Reliability

Overall quantitative probabilistic goals for core melt and radioactive material release will be evaluated, including consideration of similar goals proposed by the NRC, the Advisory Committee on Reactor Safeguards (ACRS), and the Atomic Industrial Forum. On the basis of

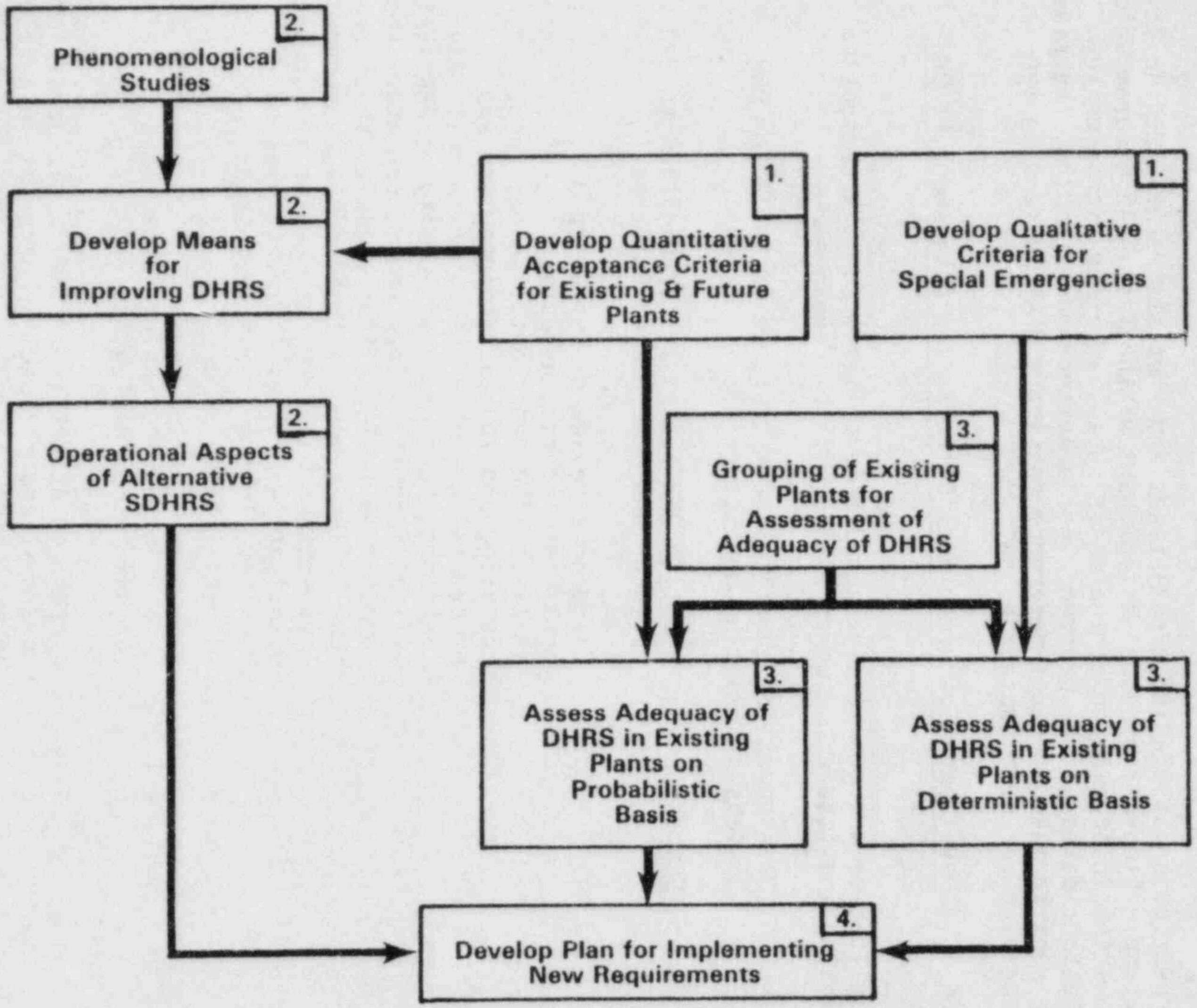
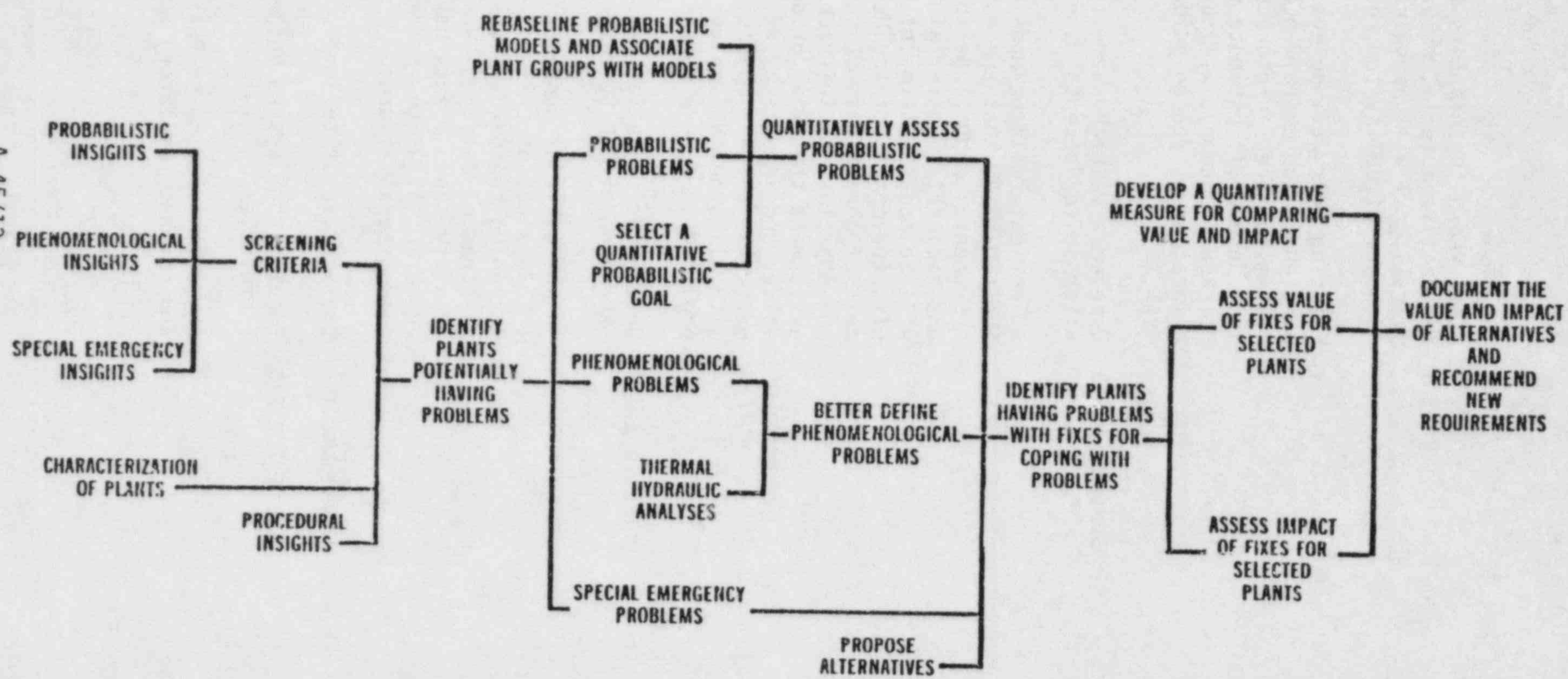


Figure 1. Inter-Relation of Sub-Tasks in Task Action Plan A-45

FIGURE 2. **FLOW CHART OF MAJOR SUBTASKS OF TASK A-45 PROGRAM**



other known contributors to core melt and radioactive material release, a portion of this overall goal will be selected for judging the adequacy of DHR system reliability.

Existing LWRs will be evaluated against the portion of the overall goal allocated to DHR system failures. Future plant systems, which lack an overall estimate of core melt or release frequency, will use this goal allocation as design guidance for the DHR systems.

Subtask 1.1.3 Develop a Value Measure for Comparing Alternative Concepts

An estimate will be made of averted costs and averted risks that are associated with incremental reductions in core melt frequency and radioactive material release. Estimates will be made in terms of both the onsite and offsite impacts of accidents involving core melt or the release of radioactive material. The objective of this subtask is to provide an assessment that will bridge the gap between values and impacts by determining what values may warrant what impacts. Close coordination with all other NRC value-impact work will be provided to ensure consistency.

Subtask 1.1.4 Develop Criteria for Probabilistically-Based Screening of DHR Systems

On the basis of Subtasks 1.1.1, 1.1.2, and 1.1.3, a set of criteria will be prepared for screening the DHR systems in a representative spectrum of existing LWR power plants to identify those plants having potential vulnerabilities to random, operator, and common-mode failures.

Task 1.2 Development of Qualitative Criteria for Special Emergencies

Subtask 1.2.1 Identify DHR System Vulnerabilities to Special Emergencies

Based on a review of PRAs, systems analyses, and current regulatory guidelines, those system and plant characteristics which most often contribute to the unavailability of DHR systems will be identified for special emergency threats (for example, fire, flood, earthquake, and sabotage). This will include an evaluation of the relative importance of special emergency accident contributions to core melt frequency, together with an assessment of how non-U.S. countries have analyzed special emergency threats.

Subtask 1.2.2 Develop Criteria for Special Emergency Screening of DHR Systems

On the basis of Subtask 1.2.1, a set of screening criteria will be prepared for assessing the DHR systems in a representative spectrum of existing LWR power plants to identify those plants having potential vulnerabilities to special emergency threats.

TASK 2. Development of Means for Improvement of DHR Function

Alternative methods for improving DHR systems will be identified, evaluated, and conceptually designed.

Task 2.1 Phenomenological Studies

Subtask 2.1.1 Identify Potential Phenomenological Uncertainties

A review will be made of existing test and analysis results (for example, LOFT and Semiscale) relative to the use of atypical modes of DHR (for example, feed and bleed, two-phase natural circulation, and reflux condensation). Emphasis will be placed on assessing the extent to which reliance can be placed on these modes of operation. Phenomenological unknowns or uncertainties which could require further analysis or testing will be identified. U.S. and non-U.S. test and analysis results will be contrasted.

Subtask 2.1.2 Develop Criteria for Phenomenological Screening of DHR Systems

On the basis of Subtask 2.1.1, a set of criteria will be prepared for assessing the DHR systems in a representative spectrum of existing LWR power plants to identify those plants adopting atypical DHR modes of operation which may lack a proven phenomenological basis.

Subtask 2.1.3 Perform Thermal/Hydraulic Analyses

For those DHR systems identified in Subtask 2.1.1, which employ unconventional use of existing DHR systems, thermal/hydraulic analyses will be performed to confirm the acceptability or to identify particular uncertainties associated with the systems. Also, thermal/hydraulic analyses using existing codes and models will be performed for those alternative DHR systems proposed in Subtask 2.2.1 which appear to apply unproven DHR techniques, including techniques (for example, feed and bleed) whose feasibility may be plant specific.

Task 2.2 Conceptual Design Studies

Subtask 2.2.1 Propose Alternative DHR Systems

Based on the DHR criteria and vulnerabilities identified in Subtasks 1.1.4, 1.2.2, and 2.1.2, a spectrum of simple and more elaborate alternative DHR systems will be proposed from both U.S. and foreign sources.

Subtask 2.2.2 Assess Engineering Feasibility of Alternatives

An engineering feasibility study will be made of the alternatives proposed in Subtask 2.2.1 to determine both major retrofit concerns and desirable features, including component availabilities and operational aspects.

Subtask 2.2.3 Develop Engineering Details

For those alternative DHR systems judged to be best from an engineering and phenomenological standpoint (Subtasks 2.1.3 and 2.2.2), a more detailed conceptual design will be developed for purposes of sizing components, identifying support system needs, and defining modes of operation.

Subtask 2.2.4 Apply the Alternatives to LWRs

For the less reliable LWR decay heat removal systems to be identified in Subtask 3.1.3, the most attractive alternative DHR systems from Subtask 2.2.1 will be selected. Electrical, piping, and structural points of interface will be identified for the alternatives, along with major areas of interference.

Subtask 2.2.5 Perform an Impact Assessment of Alternatives

An estimate will be made of the costs, plant downtime, and operational impacts associated with the alternatives and power plants considered in Subtask 2.2.4. Where particular features of an alternative have been dictated by special emergency requirements, an estimate will be made of the impact of providing the features. This subtask also included a near term cost-benefit evaluation of adding a depressurization capability to those Combustion Engineering plants that do not have power operated relief valves (PORVs). The final contractor report (NUREG/CR-3421) was issued in August 1983 and formed part of the staff's overall assessment of this issue (Draft NUREG-1044, dated February 1984).

Subtask 2.2.6 Perform a Value Assessment of Alternatives

A value assessment of alternative DHR systems will be performed by:

1. Establishing the probabilistic safety importance of DHR systems achieving cold shutdown.

2. Establishing the deterministic safety importance of DHR systems achieving cold shutdown using safety-grade equipment.

3. Assessing safety improvements in both hot and cold shutdown phases of DHR in terms of:

   a. Reduced frequency of core melt and/or release.
   b. Averted costs and averted public risk.
   c. Decreased susceptibility to special emergencies.

In order to avoid an extensive effort in assessing the probabilistic safety importance of achieving cold shutdown, this subtask will limit cold shutdown analyses to defining and quantifying, on a generic basis, those accident sequences for which failure to achieve cold shutdown most contribute to core melt frequency. For hot shutdown, the PRA models developed in Subtask 3.3.2 will be used to estimate the quantitative safety benefits of alternative DHR systems. This subtask also included a near term evaluation of the values or benefits achieved by adding a depressurization capability to those Combustion Engineering plants that do not have PORVs (see Subtask 2.2.5).

Task 2.3 Operational Aspects of Alternative DHR Systems

Subtask 2.3.1 Assess Emergency DHR Operating Procedures

A review will be made of several representative emergency DHR operating procedures from existing power plants to assess the practicality and feasibility of the procedures. The availability of time, instrumentation, and controls for operator actions will be evaluated. Thermal/hydraulic analyses from Subtask 2.1.3 will be used to help define time windows for operator actions.

Subtask 2.3.2 Develop Operating Guidelines for Alternative DHR Systems

Operating guidelines for the alternative DHR systems assessed in Subtask 2.2.2 will be prepared in outline form. Consideration will be given to the adequacy of time, instrumentation, and controls for operator actions. Also, the probability of covert and overt operator errors involving the alternative systems will be evaluated in the context of the existing plant system functions.

TASK 3. Assessment of Adequacy of DHR Systems in Existing LWRs

The adequacy of DHR systems in existing LWRs will be assessed on a probabilistic and deterministic basis.

Task 3.1 Assess Adequacy of DHR Systems in Existing Plants on a Deterministic Basis

Subtask 3.1.1 Establish DHR System Operating Modes

A review will be made of the DHR systems and support systems used in existing LWRs to determine modes of DHR system operation and functional requirements of the systems for ensuring successful DHR. Consideration will be given to the manner in which different types of plant conditions (for example, station blackout, loss of offsite power, loss of coolant accidents) can impact the functioning of the DHR systems.

Subtask 3.1.2 Gather Design Information for DHR Systems

Design information for DHR systems of existing LWRs will be gathered in the form of flow diagrams, equipment arrangement drawings, piping and instrumentation diagrams, electrical one-line drawings, and emergency procedures. Where information is sketchy, unclear, or significantly out of date, plant walk-through inspections will be made.

Subtask 3.1.3 Identify LWRs Which Appear Inadequate

The sets of screening criteria from Subtasks 1.1.4, 1.2.2, and 2.1.2 for probabilistic-based, special emergency, and phenomenological vulnerabilities will be applied to existing LWRs on a deterministic basis. Those LWRs which appear on the basis of the screening criteria to have DHR weaknesses will be identified.

Task 3.2 Review Grouping of Existing Plants for Assessment of Adequacy of DHR Systems

A review has been made of the Brookhaven National Laboratories report on "Grouping of LWRs for Evaluation of DHR Capability" (NUREG/CR-3713, dated March 1984). This information, which categorizes all existing LWRs by similar DHRS design features and correspondence to reference plant probabilistic safety analyses, has been incorporated into Subtask 3.3.1.

Task 3.3 Assess Adequacy of DHR Systems in Selected Existing Plants on a Probabilistic Basis

Subtask 3.3.1 Modify PRA Models to Reflect Grouped LWRs

PRA logic models for the reference LWRs identified in Task 3.2 will be programmed. It is anticipated that as many as twelve different reference plant PRAs will need to be reviewed, placed in a consistent format, and computerized for analysis. The logic models of the programmed reference PRAs then will be modified to reflect those LWRs which were characterized and assessed in Subtasks 3.1.2 and 3.1.3 to be inadequate on the basis of the probabilistically-based screening criteria formulated in Subtasks 1.1.4 and 1.2.2. Once in place, the modified reference PRAs can be used to estimate core melt and release frequencies for LWRs for which a complete PRA has never been performed (Subtask 3.3.2). Also, core melt and release frequency reductions associated with alternative DHR systems can be estimated (Subtask 2.2.6).

Subtask 3.3.2 Estimate Core Melt and/or Release Frequencies

The modified PRA models from Subtask 3.3.1 will be requantified and solved to estimate the frequency of core melt and/or radioactive material release, together with an estimate of associated probabilistic uncertainties. On the basis of these estimates and the quantitative goal established in Subtask 1.1.2, those LWRs will be identified for which random, operator, common-mode, and special emergency failures prove to be probabilistically significant.

TASK 4. <u>Development of Plan for Implementing Proposed New Licensing Requirements for DHR Systems</u>

In addition to developing a plan for implementing any proposed new licensing requirements for DHR systems, this task will involve overall project management and integration.

Task 4.1 Integrate and Manage Program Subtasks and Peer Review Group Activities

A detailed schedule of milestones, interim milestones, and information transfer between program subtasks will be prepared and monitored throughout the program. Subcontractor efforts will be managed, and information from other related programs will be integrated. In addition, an industry peer review group will be established to comment on and make recommendations concerning the USI A-45 program approach to various issues. This includes reviewing selected USI A-45 milestone reports, and the results of the NRC research program entitled, "Study of the Value and Impact of Alternative Decay Heat Removal Concepts for Light Water Reactors" (NUREG/CR-2883, dated June 1983).

Task 4.2 Develop Implementation Plan

A NUREG report will be prepared which summarizes the technical studies performed under USI A-45. As appropriate, proposed revisions to Standard Review Plans and Regulatory Guides will be developed containing recommendations for any proposed new requirements, along with supporting technical and cost-benefit bases, to provide a consistent, integrated, and comprehensive set of DHR requirements which are based on the value and impact evaluations of various alternative DHR measures and the evaluations of the adequacy of existing DHR systems performed in Tasks 2 and 3, respectively. Interim reports and briefings will be given upon the completion of major milestones, together with appropriate recommendations for changes in regulatory requirements.

TASK 5. <u>Assessment of European Practice for Decay Heat Removal Criteria and Design</u>

Current European practice in nuclear plant criteria and design for decay heat removal systems will be evaluated. Specific DHR design information for dedicated and bunkered shutdown cooling systems will be obtained through foreign visits and its applicability to USI A-45 will be assessed.

As previously indicated, a flowchart of the major subtasks of USI A-45 is shown in Figure 2.

C. Management of Work

The responsibility for preparing, implementing and managing 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 the Office of Nuclear Regulatory Research (RES). The Task Manager will also provide close coordination with the ACRS. NRR will have the responsibility of taking licensing-related actions on decay heat removal issues during the conduct of this program.

D. Schedule

A milestone reporting schedule has been developed for the completion of the major blocks of work for this program and is shown in Table 3. A detailed schedule breakdown for all subtask work included in this plan is provided in Figure 3. Figure 3 also shows the due dates for milestone reports listed in Table 3

Table 3. Major Milestone Reporting Schedule

Milestone Description	Scheduled Start	Scheduled Completion*
1. Quantitative Probabilistic Goal for DHR System	July 1982	April 1984
2. Grouping of LWRs for Evaluation of DHR Capability	February 1982	March 1983
3. Criteria for Screening Systems in Existing LWRs	September 1982	April 1984
4. Probabilistic and Deterministic Safety Importance of Cold Shutdown DHR Systems	December 1982	August 1984
5. Identification of LWRs Having DHR System Features That Apparently Do Not Meet Screening Criteria (Deterministic Assessment)	September 1982	April 1984
6. Value Measure for Comparing Alternative Concepts	April 1983	July 1984
7. Thermal/Hydraulic Analysis of Unproven DHR Techniques	July 1983	April 1984
8. Engineering Feasibility of Alternatives	April 1984	November 1984
9. Identification of LWRs Having DHR System Features That Apparently Do Not Meet Screening Criteria (Probabilistic Assessment)	September 1982	October 1984

*Due date for final report from contractor; a draft report will be provided 1 month earlier for internal NRC staff review and comment.

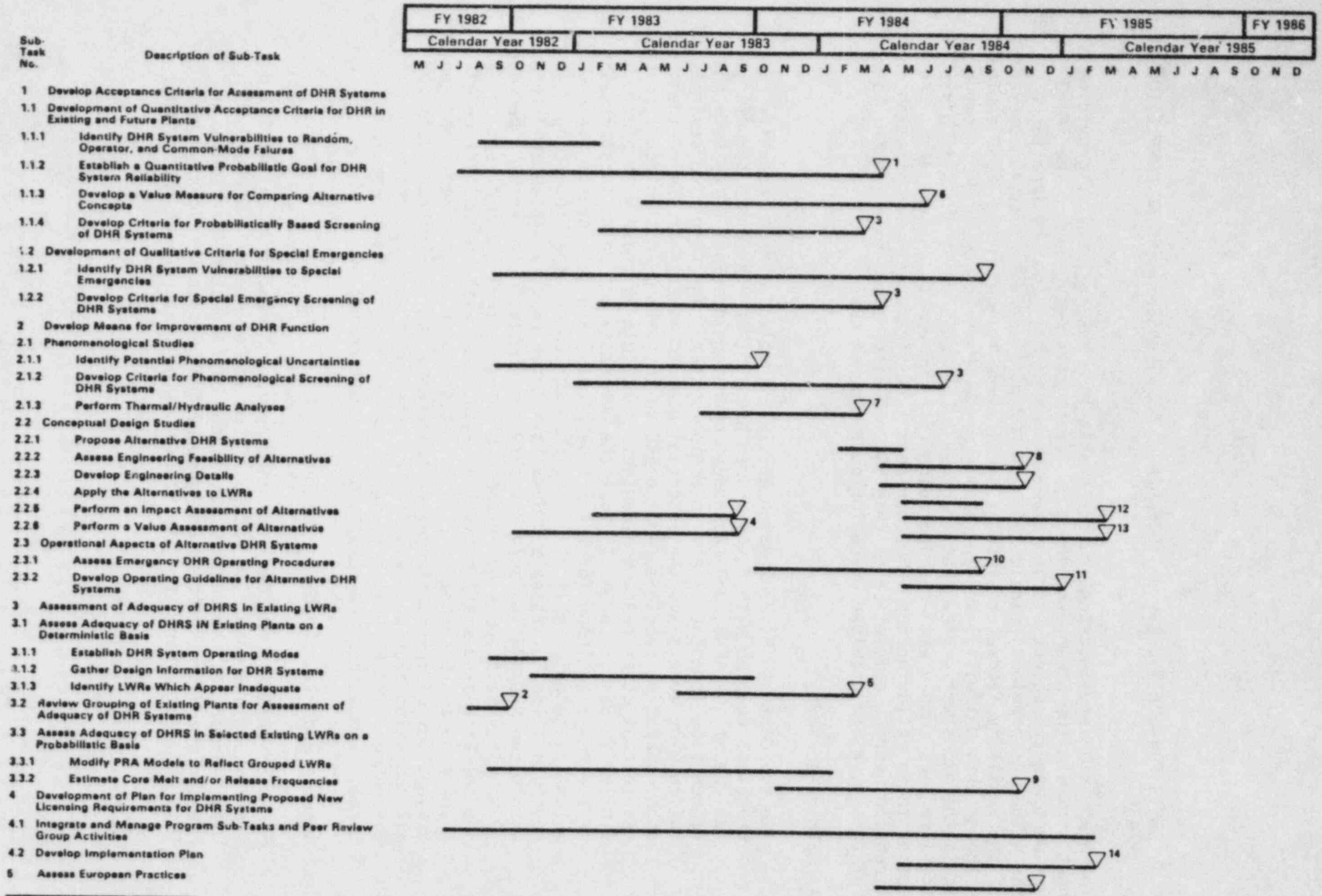
Table 3. Major Milestone Reporting Schedule

(Continued)

Milestone Description	Scheduled Start	Scheduled Completion*
10. Assessment of Emergency DHR Operating Procedures	October 1983	September 1984
11. Alternative DHR System Operating Guidelines	July 1984	January 1985
12. Impact Assessment of Alternative DHR Systems	May 1984	February 1985*
13. Value Assessment of Alternative DHR Systems	May 1984	February 1985*
14. Plan for Implementing Any Proposed New Requirements	April 1984	February 1985

*An interim value-impact assessment of the Combustion Engineering depressurization issue (see Subtasks 2.2.5 and 2.2.6) was provided in NUREG/CR-3421, dated August 1983.

FIGURE 3. SCHEDULE FOR TASK A-45 PROGRAM



NOTE: ▽-Milestone Report Numbers are Shown in Table 3.

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

The AFWS 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 AFWS. 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 AFWS, 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 PORV and a temporary failure of the AFWS, 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/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 LOCAs for all operating plants to assure their continued safe operation. Reference 6, NUREG-0645, "Report of the Bulletins and Orders Task Force," summarizes the results of the work performed.

B. Generic and Plant-Specific Studies

For Babcock & Wilcox (B&W) designed operating reactors, an initial NRC staff study was completed and published in Reference 7, 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 pursued.

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 8, 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, Combustion Engineering, 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 LOCAs, are published in Reference 9, NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant-Accidents in Westinghouse-Designed Operating Plants;" Reference 10, NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant-Accidents in Combustion Engineering-Designed Operating Plants;" and Reference 11, 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 plant's design and operation, and training of operators identified in Reference 6, 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 12, 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 6, NUREG-0645, "Report of the Bulletins and Orders Task Force."

Pressurized Water Reactors (PWRs)

The primary method for removal of decay heat from PWRs is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feedwater or AFWS, 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 AFWS was highlighted and a number of improvements were made to improve the reliability of the AFWS (see Reference 6, NUREG-0645). It was also required that operating plants be capable of providing the required AFWS flow for at least 2 hours from one AFWS pump train independent of any alternating current power source; that is, if both offsite and onsite alternating current (ac) power sources are lost.

As discussed in Reference 13, some PWRs potentially have at least one alternative means of removing decay heat if an extended loss of all 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 PORVs 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 lower primary system pressure (below about 400 psi), the long term decay heat is removed by the RHR system to achieve cold shutdown conditions.

Boiling Water Reactors (BWRs)

The principal means for removing decay heat in BWRs 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 pump of the 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 for high pressure coolant inventory control. 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 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 which opens the safety/relief valves and rejects energy to the suppression pool. At lower pressure, long term cooling in the RHR mode is initiated to achieve cold shutdown conditions.

In some earlier BWRs, a 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 plant's 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 and vulnerabilities to special emergencies 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 RHR system's reliability and risk assessments, design characteristics, and plant visits. DL will provide assistance to the Task Manager for USI A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to USI A-45 for which DL has responsibility, as identified in Reference 5. 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	0.02 person-year
Operating Reactors Branch No. 2	0.10 person-year
Operating Reactors Branch No. 3	0.20 person-year
Operating Reactors Branch No. 4	0.02 person-year

*All the manpower requirements provided below are estimates on an annual basis for FY84. Total FY85 and FY86 manpower estimates are provided in Table 4.

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 USI A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to USI A-45 for which DSI has responsibility, as identified in Reference 5. In addition, DSI will participate in selected plant visits for obtaining required plant design information and determining to what extent the selected plants comply with Branch Technical Position 5-1. DSI will also 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.

Manpower Requirements* -

Reactor Systems Branch (RSB)	0.50* person-year
Auxiliary Systems Branch (ASB)	0.50* person-year
Instrumentation and Control Systems Branch	0.05 person-year
Power Systems Branch	0.08 person-year
Containment Systems Branch	0.05 person-year

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

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 USI A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to USI A-45 for which DE has responsibility, as indicated in Reference 5. In addition, DE will contribute to the development of a consistent and comprehensive set of decay heat removal systems requirements.

Manpower Requirements -

Chemical Engineering Branch	0.10 person-year
Equipment Qualification Branch	0.10 person-year
Mechanical Engineering Branch	0.025 person-year
Structural and Geotechnical Engineering Branch	0.025 person-year
Materials Engineering Branch	0.025 person-year
Environmental and Hydrologic Engineering Branch	0.05 person-year

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 procedural guidelines developed, as part of Task 2.3, Operational Aspects of Alternative SDHR Systems; and DHFS will have a major role in this activity.

Manpower Requirements -

Human Factors Engineering Branch	0.10 person-year
Procedures and Test Review Branch	0.20 person-year

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 USI A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to USI A-45 for which DST has responsibility, as indicated in Reference 5. DST will also coordinate the formal revision and publication of licensing documents (that is, Rules, Regulatory Guides, Standard Review Plans) with RES.

Manpower Requirements* -

Generic Issues Branch (GIB)	2.0* person-year
Reliability and Risk Assessment Branch (RRAB)	0.25* person-year
Licensing Guidance Branch (LGB)	0.05 person-year
Safety Program Evaluation Branch (SPEB)	0.25* person-year
Research and Standards Coordination Branch (RSCB)	0.025 person-year

*Reflects GIB overall management responsibility, 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 decay heat removal measures.

A summary of the resource requirements, both technical assistance funding and NRC staff support, is provided in Table 4.

6. ASSISTANCE FROM RES DIVISIONS

Since RES has the lead role on related programs (for example, Reactor Safety Study Methodology Applications Program, Integrated Reliability Evaluation Program), very close coordination and cooperation will be required

Table 4. Resource Requirements Summary

		FY 84	FY 85	FY 86
Contract Dollars for Technical Assistance (in thousands)				
	A-1309	2174	725	515
	A-7282	200		
NRR Manpower in Person-Years				
DST	GIB	2.0	3.0	1.50
	SPEB	0.25	0.50	0.25
	RRAB	0.25	0.50	0.25
DSI	RSB	0.50	0.75	0.50
	ICSB	0.05	0.05	0.025
	CSB	0.05	0.05	0.025
	ASB	0.50	0.75	0.50
	PSB	0.08	0.08	0.04
	CPB			
	AEB			
	ETSB			
	RAB		0.05	0.025
DE	MEB	0.025	0.025	0.025
	SGEB	0.025	0.025	0.025
	GSB			
	EHEB	0.05	0.05	0.05
	MTEB	0.025	0.025	0.025
	CHEB	0.10	0.10	0.10
	EQB	0.10	0.10	0.10
DHFS	HFEB	0.10	0.10	0.10
	OLB			
	LQB			
	PTRB	0.20	0.20	0.20
DL (Total)		0.34	0.34	0.25
RES (Total)		0.60	0.60	0.25
AEOD (Total)		.10	.10	0.10
		5.35	7.40	4.34

on USI 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 National 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 USI A-44, "Station Blackout," relative to DHRS. The Division of Accident Evaluation will provide technical input relative to the response of existing and improved SDHRS to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support improved DHRS behavior under transient and accident conditions. The Division of Engineering Technology will provide assistance in the preparation and publication of documents (that is, Rules, Regulatory Guides, Standard Review Plans) providing 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 National Laboratories Program on Nuclear Power Plant Design Concepts for Sabotage Protection. RES will provide assistance to the Task Manager for USI A-45 for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to USI A-45 for which RES has responsibility, as identified in Reference 5.

Manpower Requirements -

Division of Risk Analysis	0.25 person-year
Division of Accident Evaluation	0.20 person-year
Division of Engineering Technology	0.10 person-year
Division of Facility Operations	0.05 person-year

7. ASSISTANCE FROM OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA (AEOD)

Provides review and comment on the technical evaluations provided by the Task Manager in those areas where systems operational experience is particularly important. AEOD will provide specific input on their ongoing review and evaluation of residual heat removal (RHR) system operating experience. In addition, AEOD will provide a specific review of the Electric Power Research Institute sponsored work on RHR operating experience and the USI A-45 contractor work on potential benefits and impacts of upgrading cold shutdown systems.

Manpower Requirements -

AEOD	0.10 person-year

8. TECHNICAL ASSISTANCE

Direct technical assistance contract work in support of the program will be required for all tasks. The funding will be provided by NRR. Table 4 provides a summary of the total estimated technical assistance funding requirements. A description of the technical assistance required for this program is provided above in Section 3.B. (Technical Content of Individual Tasks). Sandia National Laboratories has been selected to provide overall project management, technical direction and integration for the entire USI A-45 program, including selection and management of sub-contractors.

9. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

Interaction with outside organizations will be conducted by establishing an industry peer review group. The industry peer review group may include representatives from such organizations as the Atomic Industrial Forum, the Electric Power Research Institute, the Nuclear Safety Analysis Center, the Institute of Nuclear Power Operations, the Federal Energy Regulatory Commission, the Federal Aviation Administration, utilities, Nuclear Steam Supply System vendors, Architect/Engineers, and foreign development agencies, regulators, and manufacturers of nuclear power stations. The peer review group will be requested to comment on and make recommendations concerning the USI A-45 contractor milestone reports. Peer review will also be conducted through periodic ACRS briefings and issue of interim NUREG reports.

With regard to specific information requirements from the nuclear industry, approximately eight plant visits are planned during the period April to September 1984. The primary purpose of the visits is to obtain required design information that is not in the public domain. Each visit is expected to last 2 days. Each utility will be provided with a list of the required information before the visit. It is estimated that 2 person-weeks of utility effort will be required in preparing for and conducting each plant visits.

10. POTENTIAL PROBLEMS

The potential problem areas which have been identified are outlined below. Each of these potential problem areas could delay the program.

A. Development of appropriate reliability or quantitative goals for USI A-45 and translation into licensing requirements.

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

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

D. 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 because of significant design variation in the systems used for the DHR function.

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4. U.S. Nuclear Regulatory Commission, "Task Action Plan A-44, Station Blackout." July 1980.

5. Memorandum from A. R. Marchese, NRC, to T. E. Murley, NRC, "Activities Related to Task A-45, Shutdown Decay Heat Removal Requiements," dated April 8, 1981.

6. U.S. Nuclear Regulatory Commission, "Report of the Bulletins and Orders Task Force," USNRC Report NUREG-0645, January 1980.

7. U.S. Nuclear Regulatory Commission, "Staff Report on the Generic Assessment of Feed-Water Transients in PWRs Designed by B&W Co.," USNRC Report NUREG-0560, May 1979.

8. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants," USNRC Report NUREG-0565, January 1980.

9. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," USNRC Report NUREG-0611, January 1980.

10. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant-Accidents in Combustion Engineering-Design Operating Plants," USNRC Report NUREG-0635, January 1980.

11. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Design Operating Plants and Near-Term Operating License Applications," USNRC Report NUREG-0626, February 1980.

12. U.S. Nuclear Regulatory Commission, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," USNRC Report NUREG-0623, November 1979.

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

TASK ACTION PLAN
(March 1984)

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (TASK A-46)

Lead NRR Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
Task Manager:	T. Y. Chang, GIB, DST
Lead Supervisor:	Karl Kniel, Chief, GIB, DST
NRR Principal Reviewer:	Arnold Lee Equipment Qualification Branch Division of Engineering
	John Knox Power Systems Branch Division of Systems Integration
	Pei-Ying Chen Systematic Evaluation Program Branch Division of Licensing
	Frank Skopec Radiological Assessment Branch Division of Systems Integration
Applicability:	All Light Water Operating Reactors
Projected Completion Date:	December 1984

1. DESCRIPTION OF PROBLEM

There is a recognized need to demonstrate the functional capability of safety-related nuclear plant equipment subjected to a seismic event. The General Design Criteria (GDC) for nuclear power plants states that structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena, such as earthquakes, without a loss of capability to perform their safety functions (10 CFR Part 50, Appendix A, Criterion 2). Also the GDC states design control measures shall provide for verifying or checking the adequacy of design by the performance of a suitable testing program. Suitable qualification testing under the most adverse design conditions shall be included (10 CFR Part 50, Appendix B, Section III). Guidance on compliance with these provisions of 10 CFR Part 50 is contained in Revision 2 to Standard Review Plan Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment."

Today, equipment is seismically qualified by analysis and/or testing. Analyses alone are acceptable only if the necessary functional operability of the equipment is assured by its structural integrity alone. If not, some testing is required. Seismic input motion to equipment is specified by a required response spectrum or time history. Equipment that is small enough is subjected to a test response spectrum which envelopes the required response spectrum. The equipment should be tested in the operating condition. For equipment too large to fit on a test table, a combined analysis and test procedure is adopted.

Since commercial nuclear power plants were first introduced, significant changes in seismic qualification criteria have evolved. Also, the analytical and experimental methods used to qualify equipment have changed. Therefore, the seismic resistance of existing equipment installed in operating nuclear plants may vary considerably.

Operating plant equipment may not meet the current seismic qualification criteria. The seismic qualification of equipment in operating plants may have to be reassessed to ensure its performance during and after a seismic event.

The objective of this Unresolved Safety Issue (USI) is to develop seismic qualification methods and acceptance criteria that can be used to assess the capability of mechanical and electrical equipment in operating nuclear power plants to perform their intended safety function during and after a seismic event.

Technical work in support of USI A-46 will be provided by technical assistance contracts managed by the Office of Nuclear Reactor Regulation (NRR), from ongoing research programs in the seismic area, and possibly from the Systematic Evaluation Program (SEP).

The Equipment Qualification Branch (EQB) of NRR is supporting a program which includes (1) a risk sensitivity study of safety system components, which will form the basis for development of a minimum equipment list; (2) cost/benefit analyses of seismic qualification of equipment on the minimum equipment list; and (3) development of guidelines for generation of generic floor response spectra.

The Office of Nuclear Regulatory Research (RES) is supporting a research program which, in part, is an historical survey of methods used for seismic qualification of nuclear plant equipment and components and a comparison with current criteria.

The Generic Issues Branch (GIB) of NRR is supporting a program for correlation of seismic response of equipment in non-nuclear facilities to the qualification of nuclear plant equipment, and a program for the development of insitu test methods and the collection and correlation of test data from both laboratory tests and insitu tests.

2. PLAN FOR PROBLEM RESOLUTION

A. Approach

A minimum list of equipment to be qualified will be developed from a risk study conducted under contract to Brookhaven National Laboratory (BNL). The reliability of components in systems which perform important safety functions will be varied and the effect on risk computed. A sensitivity analysis will be used to identify equipment where changes in reliability result in large incremental risks, allowing a cost/benefit analysis to be made. Only those components whose failure significantly affects safety functions will be included on the minimum equipment list.

Mechanical and electrical components on the minimum equipment list will still be too numerous to consider on an individual basis. Generic groups of these components will be developed according to function and similarity of methods to be used for seismic qualification.

A review of past and present criteria and methods used to structurally and operationally qualify the various categories of equipment is being conducted. Both analytical and test methods are being considered. The conservatisms, disavantages, deficiencies, and anomalies of the methods will be determined. This review is part of a research program sponsored by RES and being performed at the Southwest Research Institute (SWRI). Activities of this research program in support of USI A-46 are an evaluation of past and present analytical and test methods of seismically qualifying operability of safety-related equipment and correlation of these methods with current criteria.

The SEP of the Systematic Evaluation Program Branch (SEPB) complements this USI program. In Phase I of SEP Topic III-6, a sampling of existing seismic design documents from five older plants was reviewed, and a limited amount of reevaluation was also made. Some structural retrofitting to ensure proper equipment anchoring was recommended for the five plants. Safety-related systems and components were reviewed in selected plants to some extent for operability. Some systems and components were found to require additional seismic evaluation. SEP plant owners have initiated a generic program to tabulate the equipment present in the SEP plants. If appropriate information can be developed in time, it will be reviewed and incorporated for consideration in developing USI A-46 resolution.

An effort has been initiated by the Seismic Qualification Utilities Group (SQUG) to survey mechanical and electrical equipment installed in non-nuclear plants built in high seismic areas. Non-nuclear power plants and many industrial facilities contain mechanical and electrical equipment similar in design and function to equipment used in nuclear power plants. A number of these non-nuclear power plants and industrial facilities have been subjected to seismic events. Experience with equipment in these plants and facilities can be useful in determining the seismic and dynamic response of comparable equipment in nuclear power plants. One task of USI A-46 is to monitor that survey, and if it is determined that the resulting information is useful, it will be integrated into the development of seismic qualification guidelines.

A program has been initiated for development of insitu test methods to assist in qualifying equipment in operating plants. In that program, a review and summary of existing methods for performing insitu testing will be made. In addition, operability and failure for various types of equipment will be defined, and a data base of laboratory test and insitu test information will be developed. Information on insitu and laboratory tests will be used in development of guidelines for qualification of equipment in operating plants.

Following completion of USI A-46 technical work and development of proposed regulatory requirements, a value/impact analysis will be performed to determine the cost effectiveness of implementing the proposed requirements. The USI A-46 technical findings, the implementation recommendations and the value/impact analysis will be submitted for review by the Committee to Review Generic Requirements

(CRGR). Following CRGR review, the proposed requirements with supporting documents will be issued for public comment prior to final approval by the NRC staff and CRGR.

B. Tasks

Task 1. Develop Minimum List of Equipment to be Qualified

It must be ensured that (1) modification of safety-related equipment provides substantial additional protection which is required for the public health and safety, and (2) equipment considered for upgrading be those that contribute most to risk.

Subtask 1(a). Perform Sensitivity Analysis

Using a list of systems essential to reactor shutdown and prevention of radioactive release, a sensitivity analysis will be performed using previously developed computer codes. The result is expected to be a list of equipment whose changes in reliability result in large effects on public risk.

Subtask 1(b). Perform Cost/Benefit Analysis of Seismic Upgrading of Equipment

Using the list of equipment developed in Subtask 1(a), cost will be estimated to upgrade the equipment. Benefit to the public will be estimated.

Task 2. Survey and Evaluation of Equipment Seismic Qualification Methods

This task involves a study sponsored by RES to evaluate past and present methods to qualify mechanical and electrical equipment to withstand seismic events. The structural adequacy of equipment subjected to seismic events is also being reviewed by SEPB. If this information can be developed by SEPB in time, it will be reviewed and incorporated into this task.

Subtask 2(a). Evaluation of Methods Used to Seismically Qualify Equipment

Past and current analytical and test methods used to qualify equipment will be cataloged, compared and evaluated. The contractor's developed equipment list will be used in this subtask.

Subtask 2(b). Comparison to Present NRC Requirements for Equipment Qualification

Methods to qualify equipment in operating plants will be compared to present requirements. Important differences will be determined and acceptability of qualification method will be recommended.

Task 3. Develop Methods of InSitu Testing to Assist in Qualification of Equipment

This task will involve surveying existing methods for performing insitu tests which may be used to assist in qualification of nuclear plant equipment. Also, analytical methods which would be used in conjunction with those insitu tests will be reviewed and summarized. The effects of component aging will be considered. The final part of this task will be an effort to improve insitu testing for use in seismic equipment qualification.

Subtask 3(a). Develop Preliminary InSitu Test Methods

Operability and failure of various types of equipment will be defined in the first part of this subtask. Existing methods for performing insitu tests will be surveyed. Equipment will be categorized according to which test procedures are appropriate. Limitations, shortcomings and nonconservatisms associated with the methods will be identified.

Subtask 3(b). Improve and Verify InSitu Methods

The limitations identified in Subtask 3(a) previously will be studied and recommendations made for improvement and verification of test methods.

Subtask 3(c). Develop Requirement and Acceptance Criteria for Insitu Test Method

Requirements and acceptance criteria for using insitu testing method in conjunction with experience data to qualify equipment will be developed.

Subtask 3(d). Prepare Program Report

A formal report, in NUREG format, will be written covering the results of Task 3.

Task 4. The Seismic Qualification of Equipment Using Non-Nuclear Plant InService Dynamic Response Information

A program has been developed by the SQUG to survey equipment in non-nuclear plants which has been subjected to seismic events. The equipment to be surveyed is similar to equipment installed in operating nuclear plants. The seismic events which the equipment survived were, in some instances, significant. The SQUG program will be closely monitored as part of this task and the results will be studied for possible use in development of qualification requirements. Other sources of information pertinent to response, damage and operability of equipment in non-nuclear facilities subjected to seismic events will be reviewed to determine if non-nuclear equipment experience can be used to predict equipment fragilities. If it is possible to predict equipment fragilities from non-nuclear equipment surveys, then methods will be developed for the use of seismic experience in non-nuclear facilities in developing guide-lines for equipment qualification in nuclear plants.

Subtask 4(a). Feasibility Study

To assess the feasibility of using data on equipment from non-nuclear plants which have been subjected to strong earthquakes, a significant amount of data will be assembled from known sources and from the SQUG program. It will be determined if a correlation exists or can be developed between structural and functional survival of equipment in non-nuclear plants and nuclear plants. To assist in assessing the feasibility, expert consultants will be provided by the contractor to review subtask results.

Subtask 4(b). Develop Guidelines for Application of Experience Data

Guidelines for the use of the experience data collected previously will be developed and recommendations will be made for criteria to be incorporated into the proposed guidelines on equipment qualification.

Task 5. Guidelines and Criteria for Development of Generic Floor Response Spectra

The feasibility of seismically qualifying equipment using a set of generic floor response spectra will be investigated in this task. Guidelines for developing these response spectra will be developed.

Subtask 5(a). Feasibility Investigation

The feasibility of seismically qualifying equipment by using a set of generic floor response spectra will be investigated. These response spectra will be derived by considering specific earthquakes zones in accordance with Uniform Building Codes, specific site geological conditions, specific plant installation configurations, or a combination of all of the above.

Subtask 5(b). Recommend Guidelines and Procedures to Develop Generic Floor Response Spectra

Guidelines and procedures to develop generic floor response spectra will be recommended.

Task 6. Establish Guidelines, Alternative Methods, and Acceptance Criteria for Seismic Qualification of Equipment in Operating Plants

Subtask 6(a). Develop Guidelines for Assessing Adequacy of Existing Seismic Qualification, Define Alternative Methods and Acceptance Criteria

From the conclusions reached during the continued performance of research programs on equipment qualification, the SEP on seismic qualification and this task action plan, a set of explicit guidelines will be written to assess the adequacy of equipment seismic qualification methods. Both structural and functional qualification requirements will be considered. If previously used qualification methods are found to be inadequate, alternative methods for requalifying equipment will be defined. Acceptance criteria for the alternative methods will also be developed.

Subtask 6(b). NUREG Final Report

In this task a final NUREG report will be written to summarize program accomplishment, conclusions, and recommendations. The justification for each guideline will be stated and limitations will be given. The NUREG report will be issued for public comment prior to final issuance.

Subtask 6(c). Licensing Changes

In addition to providing technical bases for the recommended guidelines and criteria, proposed changes to Standard Review Plans and/or Regulatory Guides or issuance of a generic letter will be recommended if needed, and issued for public comment prior to implementation.

C. End Product

In Task 6 of this study, proposed guidelines and criteria for requalification of equipment in operating plants will be developed. A NUREG report will be written summarizing the work performed, the conclusions reached, and recommendations regarding methods of requalifying equipment. Guidelines for the qualification of equipment in operating plants will be presented in detail. Also the logic behind these guidelines will be given. If a generic letter or changes to Standard Review Plans and/or Regulatory Guides are needed they will be prepared.

A value/impact analysis will be conducted and submitted to the CRGR along with the NUREG report and proposed regulatory requirement documents. Following CRGR review and public comment, the final regulatory position on implementation will be developed and again reviewed by CRGR prior to final issuance.

D. Program Schedule and Effort

The following schedule has been established for the tasks.

	Scheduled Completion	Actual Completion
Task 1	06/83	06/83
Task 2	08/83	08/83
Task 3	09/83	10/83
Task 4	12/83	02/84
Task 5	06/83	06/83
Task 6	04/84	---

Important milestones prior to Technical Resolution, their scheduled and actual completion dates are as follows.

	Scheduled Completion	Actual Completion
Important Milestones Prior to Technical Resolution		
Issue Interim Report	09/83	09/83
Draft Technical Resolution		
Issued by DST for Staff Comment	02/15/84	03/06/84
Staff Comments to DST	02/29/84	
Completed Package to Director, NRR	03/15/84	
Package to CRGR	04/15/84	
CRGR Review Complete	05/15/84	
Issued for Public Comment	06/15/84	

The level of NRC effort to complete USI A-46 is summarized below in staff years:

	FY82	FY83	FY84
GIB/DST	1.0	1.0	1.0
MSEB/RES	1.0	0.1	0.1
EQB/DE	1.0	0.3	0.5
SEPB/DL	---	0.1	0.1
ORAB/DL	---	0.1	0.1
PSB/DSI	---	0.08	0.08
RAB/DSI	---	0.05	0.05

E. Technical Assistance

Technical assistance funding is as follows:

	FY-82	FY-83	FY-84	
Task 1	$108K	$ 75K	0	BNL (T.A. Contract by NRR/EQB)
Task 2	--	--	--	SWRI (Funded by RES)
Task 3	$125K	$150K	$10K	INEL (T.A. Contract NRR/GIB)
Task 4	$ 75K		0	LLNL (T.A. Contract NRR/GIB)
Task 5	$ 99K	$ 15K	0	BNL (T.A. Contract NRR/EQB)
Task 6	--		--	

3. JUSTIFICATION FOR CONTINUED OPERATION

Although many operating plants were designed before the development of current licensing criteria, the design rules and procedures incorporated inherent conservatisms. These include: (1) the margins between allowable stresses and ultimate strength of engineering materials; (2) the methods used for combining loads; (3) the inherent ductility of materials; and (4) the seismic resistance of nonstructural elements which are not normally considered in design calculations.

An expanding data base of observations at large industrial facilities that have experienced strong ground motion suggests that these facilities have significant seismic resistance capabilities. From that data, it can be inferred that the inherent seismic resistance of engineered structures and equipment is usually greater than is generally assumed. When even the most modest attention is paid in design to providing lateral load carrying paths, significant capability results. Most nuclear power plants have been designed using more rigorous techniques; therefore, it is reasonable to expect high inherent margins.

Furthermore, Office of Inspection and Enforcement Information Notice No. 80-21, entitled, "Anchorage and Support of Safety-Related Electrical Equipment" was sent to all operating plants from the NRC on May 16, 1980. This Information Notice informed licensees of potential safety deficiencies in the design of safety-related electrical equipment supports in the SEP plants. They were requested to review the information for possible applicability to their facilities.

Because of the above cited reasons and the continued staff review of seismic issues, it is concluded that operating plants can continue to operate without endangering the health and safety of the public pending resolution of this USI.

4. TECHNICAL ORGANIZATIONS INVOLVED

A. Generic Issues Branch, Division of Safety Technology

GIB has the overall responsibility for the performance of this USI program.

(1) Task 3

GIB will establish a plan evaluating methods for insitu and laboratory qualification of equipment in operating plants. This will be done through a technical assistance program with Idaho National Engineering Laboratory (INEL) to study methods of requalifying equipment installed in operating plants. GIB will manage the performance of this technical assistance program and the publication of a final study report.

(2) Task 4

GIB will develop a program plan to review and correlate available information on the inservice response of non-nuclear plant equipment that has been subjected to seismic or severe dynamic events.

This will be accomplished by close cooperation between an in-place SQUG program which is collecting data on equipment in non-nuclear plants which have been subjected to earthquakes, and the technical assistance program with Lawrence Livermore National Laboratory (LLNL) and INEL. GIB will coordinate these programs and manage the performance of the technical assistance program.

(3) Task 6

GIB, in conjunction with EQB, will establish the appropriate guidelines and acceptance criteria for the seismic qualification of equipment in operating plants. The technical bases of these guidelines will be documented in a final NUREG report. This report will also summarize the work performed in this USI program and the conclusions reached.

B. Mechanical and Structural Engineering Branch, Division of Engineering Technology, RES

(1) Task 2

The Mechanical and Structural Engineering Branch has a contract entitled, "Seismic Qualification of Nuclear Plant Mechanical and Electrical Equipment," with SWRI. This research program will be coordinated with the USI program. The research program will survey existing knowledge and develop a basis and the methodology for evaluating conservatisms, limitations, and anomalies related to current and past methods used to qualify equipment.

C. Systematic Evaluation Program Branch, Division of Licensing

(1) Task 2

SEPB is conducting a program to review and evaluate the seismic design criteria and the ability of safety-related mechanical and electrical equipment to retain their structural integrity during and after a seismic event. The functional operability of the equipment is not being considered. This SEP branch program will complement the USI study. If appropriate information can be generated in time, it will be integrated into the USI program.

D. Equipment Qualification Branch, Division of Engineering

(1) Task 1

EQB is developing a program to (a) identify equipment that contributes most to risk during and after a seismic event, and (b) perform a cost/benefit analysis to establish the extent to which safety-related equipment needs to be upgraded. This will be accomplished by a technical assistance program with BNL.

(2) Task 5

EQB is developing a program to investigate the feasibility of seismically qualifying equipment by employing a set of generic enveloping response spectra. This will be done through a technical assistance program with BNL.

(3) Task 6

EQB, in conjunction with GIB, will establish the appropriate guidelines and acceptance criteria for the seismic qualification of equipment in operating plants.

5. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

In Task 4 of this program, a program to review and correlate available information on the inservice response of non-nuclear plant equipment that has been subjected to severe seismic or dynamic events and to define data base and establish guidelines for use of non-nuclear experience data, will be developed with technical assistance from LLNL and INEL. A concurrent program is being sponsored by SQUG to collect data on equipment in non-nuclear plants which have been subjected to earthquakes. The owner's group program will be closely monitored by GIB so that data from that program can be used in the LLNL/INEL program.

As this task progresses, it is anticipated that meetings and information exchange with industry groups such as the Atomic Industrial Forum and the Electric Power Research Institute will take place.

6. ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

Requirements for assistance from NRC Offices other than input through RES sponsored work at SWRI discussed in Task 2 are not anticipated at this time.

7. POTENTIAL PROBLEMS

None expected at this time.

Resource Requirements Summary

		FY 83	FY 84
Contract Dollars for Technical Assistance (in thousands)		240	50
NRR Manpower in Person Years			
DST	GIB	1.0	1.0
	SPEB		0.02
	RRAB		0.02
DSI	RSB		0.01
	ICSB		
	CSB		
	ASB		
	PSB		
	CPB		
	AEB		
	ETSB		
	RAB	0.05	0.05
DE	MEB		
	SGEB		0.02
	GSB		0.05
	EHEB		
	MTEB		
	CHEB		
	EQB	0.3	0.5
DHFS	HFEB		.01
	OLB		
	LQB		
	PSRB		
DL	SEPB	0.1	0.1
RES Manpower in Person-Years			
RES		0.1	0.1

TASK ACTION PLAN
(April 1984)

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (TASK A-47)

Lead Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
Task Manager:	A. J. Szukiewicz, GIB, DST
Lead Supervisor:	Karl Kniel, Chief, GIB, DST
NRR Principal Reviewers:	Jose Calvo Instrumentation and Control Systems Branch Division of Systems Integration
	Sammy S. Diab Reactor Systems Branch Division of Systems Integration
	A. S. Gill Power Systems Branch Division of Systems Integration
	James T. Beard Operating Reactors Assessment Branch Division of Licensing
	Chelliah Erulappa Reliability and Risk Assessment Branch Division of Safety Technology
	William G. Kennedy Procedures and Test Review Branch Division of Human Factors Safety
Office of Analysis and Evaluation of Operational Data (AEOD)	Matthew Chiramal Plant Systems Unit
Office of Nuclear Regulatory Research (RES)	Demetrios Basdekas Division of Facility Operations

Applicability:	Light Water Reactors (Pressurized and Boiling Water Reactors)
Projected Completion Date:	Last Contractor Final Evaluation Draft Reports Submitted, May 1985 Draft NUREG Report Issued by Office of Nuclear Reactor Regulation (NRR) for Public Comment, September 1985 Final NUREG Report Issued by NRR, April 1986

1. DESCRIPTION OF PROBLEM

Non-safety grade control systems are used to maintain the plant within the necessary pressure and temperature limits during normal shutdown, startup, and load varying power operation. The control systems are not relied upon to perform any safety functions following postulated accidents but are required to control plant processes that could have a significant impact on plant safety. Those control systems include the reactivity control systems, and reactor coolant pressure, temperature, level, flow and inventory controls (that is, borated water controls). In addition, they include secondary system pressure and flow controls [pressurized water reactor (PWR)] as well as the associated support systems such as electric, hydraulic and/or pneumatic power supply systems.

During the licensing process, the staff performs an audit review of the non-safety grade control systems, on a case-by-case basis, to assure that an adequate degree of separation and independence is provided between these non-safety grade systems and the safety systems, and that effects of the operation or failure of these systems are bounded by the accident analysis in Chapter 15 of the plant's Safety Analysis Report. Typical events that are addressed by the licensees, and are evaluated by the staff in the audit review include, but are not limited to: (1) the feedwater system malfunctions that result in a decrease or an increase in the feedwater flow (including the loss of the normal feedwater flow); (2) the steam pressure regulator malfunctions or failures that result in an increase or a decrease in the steam flow (including the turbine trip event); (3) a spectrum of reactivity addition events; and (4) chemical and volume control malfunctions that increase the reactor coolant inventory or decrease the boron concentration.

On this basis it is generally believed that control system failures are not likely to result in loss of safety functions that could lead to serious events or result in conditions that the safety systems are not able to mitigate. Indepth studies for all the non-safety grade systems have not been performed however, and there exists some potential for accidents or transients being made more severe than previously analyzed, as a result of some of these control system failures or malfunctions.

The control system failures or malfunctions may occur independently or as a result of an accident or transient under consideration. Failures or malfunctions may also occur as a result of a common mode or a system interaction that could make recovery to normal safe shutdown conditions difficult.

Two potential concerns have already been identified in which a failure or malfunction of the non-safety grade control system can (1) potentially cause a steam generator or reactor vessel overfill, or (2) can lead to a transient (in PWRs) in which the vessel could be subjected to severe overcooling. In addition, there is the potential for an independent event like a single failure, (such as a loss of power supply, a short circuit, open circuit, control sensor failure) or a common mode event (such as a harsh environment caused by an accident or a seismic event) to cause a malfunction of one or several control systems which would lead to an undesirable control action, or provide misleading information to the plant operator. These concerns will be reviewed and evaluated as part of the tasks discussed in the following sections. It should be recognized that the effects of control system failures during accident or normal plant operation may differ from plant to plant, and therefore it may not be possible to develop generic solutions to these concerns. It is possible, however, to develop generic criteria that can be used for the plant-specific reviews.

The purpose of this Unresolved Safety Issue (USI) is to perform an indepth evaluation of the control systems that are typically used during normal plant operation and to verify the adequacy of current licensing design requirements or propose additional guidelines and criteria to assure that nuclear power plants do not pose an unacceptable risk due to inadvertent non-safety grade control system failures.

2. PLAN FOR PROBLEM RESOLUTION

In order to best utilize NRC's capabilities and resources, the resolution of the activities described in detail in the following sections will be conducted under contract with the National Laboratories. The responsibility for resolution of this safety issue rests with NRR, but will involve both NRR and RES staff effort to manage and review the adequacy of the evaluations conducted. To scope the issue to a manageable level and bound the generic review to a reasonable completion schedule, USI A-47 will evaluate the non-safety grade systems of three PWR designs and one Boiling Water Reactor (BWR) design.

The task will review the plant designs of the manual and/or automatic control systems for each of the four Nuclear Steam System Supplier (NSSS) designs [Babcock and Wilcox (B&W), Combustion Engineering (CE), General Electric (GE), and Westinghouse (W)] and will include the review of any manual and/or automatic control system that interfaces with the NSSS design or dynamically interacts with the primary reactor fluid system and the secondary steam system. These associated control systems may be supplied or designed by different manufacturers or architect/engineers than the NSSS. Two PWR non-safety grade control system plant designs (that is, B&W and CE) will be evaluated by Oak Ridge National Laboratory (ORNL) under contract with RES (FIN No. B-0467). The GE BWR and the W PWR designs will be evaluated by Edgerton, Germeshausen & Grieg-Idaho (EG&G-Idaho) under contract with NRR (FIN No. A-6477). It is recognized that developing generic resolutions based on plant-specific reviews has certain limitations. Engineering judgments by the staff and the National Laboratories, based on experience and general knowledge of control and plant systems for plants other than those being studied, will be utilized wherever possible. These judgments, together with existing or future insights that can be obtained from licensees or NSSS vendors, will provide a basis for reaching generic conclusions.

The task will, for each type design: (1) identify the non-safety grade control system(s) whose failure or misoperation can, (a) cause transients or accidents identifyed in Chapter 15 of the Final Safety Analysis Report (FSAR) to be made potentially more severe than previously analyzed, (b) create the potential to negate the timely action of the automatic protection system or the manual operation of any equipment required to achieve a safe shutdown condition; (2) establish and define the order of importance of the control system(s) identified as having safety significance; (3) describe the mechanism(s) contributing to the credible failure modes, (that is, loss of power supply or the environmental effects on the control systems); (4) verify the adequacy of the existing design criteria, described in Standard Review Plan Section 7.7, "Control Systems," or develop and propose additional criteria and guidelines to improve system reliability or minimize the consequences of the control system failures that have been identified as safety significant.

To evaluate control system actions that have safety implications, the work effort will focus on the following activities.

- Evaluate control system failures that could lead to a steam generator or a reactor vessel overfill transient. (Subtask 1 of Task 7)

- Evaluate control system failures that could lead to a reactor overcooling transient. (Subtask 2 of Task 7)

- Evaluate (all other) non-safety grade control systems that have safety implications. (Overall task)

- Evaluate the effect of loss of power supplies to the control systems. This would include the electrical alternating current (ac) and direct current (dc) supplies and also the pneumatic and hydraulic supplies. (Task 4)

The major activity will be to identify and evaluate non-safety grade control systems that have safety implications. The tasks associated with the activity are outlined below. Subtasks 1 and 2 focus on specific areas of concern identified as part of the overall activity. Additional tasks or subtasks may be identified as the program develops; if other tasks are developed, the Task Action Plan will be revised. Should these reviews indicate that additional criteria for control system designs are necessary or that specific problems require resolution, appropriate action will be taken for plants in the licensing process and for plants now in operation.

Task Action Plan A-47 has been developed to utilize, whenever possible, any applicable data developed by the following current ongoing activities.

- Resolution of USI A-49, "Pressurized Thermal Shock" (PTS).
- RES activities with ORNL regarding Safety Implications of Control Systems (FIN No. B-0467).
- Steam Generator Tube Rupture Overfill Study - A study conducted by Los Alamos for the Reactor Systems Branch of the Division of Systems Integration (RSB/DSI) FIN No. A-7281 (formally FIN No. A-7276).
- Systems Interaction Program - A study conducted by the Reliability and Risk Assessment Branch (RRAB) of DST. TMI Action Plan Item II.C.3 and USI A-17.
- RES activities with ORNL evaluating plant electrical systems interactions (FIN No. B-0816).

The interface between the USI A-47 program and these activities is discussed in more detail in the appropriate tasks.

Task Description

Evaluate Non-Safety Grade Control Systems that Have Safety Implications

This activity will evaluate non-safety grade control systems and identify any non-safety grade control systems whose failure may lead to transients or accidents more severe than those analyzed in Chapter 15 of the plant FSAR and to identify non-safety grade control system failures which could produce a high frequency of occurrence of those transients bounded by Chapter 15. The control systems evaluation will review the designs of each of the four nuclear steam system (NSS) suppliers (B&W, CE, W, GE) and will include the control systems which may be designed by other suppliers but interface with the NSS control system design or dynamically interact with the reactor primary or secondary system. This activity will consist of the tasks listed below. The flow diagram (Figure 1) illustrates the interactions between these tasks.

Task 1 Identify the Systems Whose Failure Can Lead to Significant Primary System Transients

Conduct a review of the automatic and manual control systems that are used during startup, shutdown and normal load varying operations and identify all systems whose failure or malfunction has the potential for causing pressure, temperature flow and power transients in the primary reactor system. Identify also any control systems whose failure or malfunction before, during or after any transient or accident analyzed in Chapter 15 of the FSAR could cause more severe consequences then presently analysed. Gross analysis based on tools such as failure mode effects analysis (FMEA), dependency tables or diagrams, functional and system event trees and fault trees and/or any other analytical tools judged to be adequate will be used initially on a system level basis for the purpose of identifying the significant control systems. During this phase, non-mechanistic "worst-case" failure modes of the control systems will be assumed. The major components (such as valves, pumps, control drives, etc.) whose failure can cause a system malfunction will be identified.

The criteria that will be used for selecting and categorizing the safety significant control systems will be identified. A review of the applicable Licensee Event Reports (LERs), Office of Inspection and Enforcement (IE) Bulletin and Orders, and NSS emergency procedures and operating guidelines will be conducted. The results of this review will be factored into the criteria selection process and will help to identify safety significant systems. The control systems identified will be compared with those systems described in (1) the Integrated Reliability Evaluation Program (IREP) study, (2) the applicable studies conducted by selected Near-Term Operating License (NTOL) applicants in response to the Instrumentation and Control Systems Branch control system concerns identified during the NTOL review, and (3) the probability and risk assessment studies conducted by the utilities on similar designs.

The control systems identified via the activities described above will be compared with the systems identified in the analysis in Chapter 15 of the FSAR. The safety impact and the order of importance of the systems identified will be described and categorized to define for example, system whose failures initiate significant transients by themselves (that is, spills, blowdown, etc.) or systems whose failures can occur concurrent with transients resulting from other initiators. Failures will be limited to independent single failures or multiple failures resulting from a common initiator. An additional independent single failure may also be included if, as part of a specific scenario analysis, it is apparent that such failure is highly likely and the attendant consequences significant. For these scenarios, one train of the existing redundant protection systems is assumed to be available. The treatment of undetected failures will be conducted in accordance with the Institute of Electrical & Electronics Engineers (IEEE) Standard 379-1977, "IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems," Section 5.2. Operator misoperation of control systems is outside the scope of this task if existing procedures, the information available to the operator, and the time for the operator to accomplish a required action is sufficient. The control systems whose failure or malfunction may be considered less important or inconsequential or highly unlikely to warrant further study will be identified and the basis for such conclusions will be documented. For example, there may be control systems whose failure produce transients that are enveloped by the limiting transients assumed in Chapter 15 analyses, and therefore, failure of these systems would be of little relative consequence. There may also be failures whose probability of occurrence in a given sequence or at a particular point in time may be so unlikely as not to warrant further study.

As a result of these activities a set of control systems potentially significant to safety will be identified for further computer study in order to identify important failure sequences and to investigate the dynamic plant behavior as a result of these failures (see Task 2). Applicable information data developed by other ongoing NRC activities conducted by (1) RES through contracts with ORNL; (2) Instrumentation and Control Systems Branch (ICSB) case reviews; (3) the RRAB Systems Interaction Study for Indian Point Unit 3; and (4) the IREP Study for Calvert Cliffs 1, Millstone 1, Arkansas Nuclear One Unit 1 and Browns Ferry Unit 1 will be assessed as part of this task. The data developed from these activities that identifies significant control systems and assesses their reliability will be considered in the evaluation of this task.

Task 2 Conduct Computer Simulation Studies for Evaluating Combination of Systems Failures

Develop an analytical model to simulate the reactor transients, as a result of control system failures or malfunctions, using existing codes whenever possible. The model should include the plant characteristics of the primary reactor fluid and the secondary steam system and the feedwater system as well as the major elements of the control systems. The objective of these simulations will complement the system level FMEA activity (described in Task 1) in identifying and evaluating the sequences and combinations of control system failures and assessing their importance to safety. It is anticipated that the plant dynamic simulator will minimize the need for extensive use of the analytical techniques (described in Task 1) to study the interactive control system failures resulting from simultaneous and/or sequential faults.

As part of the activities conducted at ORNL through NRR/RES (FIN No. B-0467), ORNL will develop a hybrid computer model to simulate the behavior of a B&W plant. To study the effects of control system failures for the CE design, ORNL through NRR/RES will utilize Baltimore Gas and Electric Company's newly developed simulator which models the Calvert Cliffs plant design. Concurrently, as part of the activities conducted at EG&G-Idaho (FIN No. A-6477), EG&G will develop a digital computer model to simulate the dynamic behavior of a GE BWR-type plant and a digital model to simulate a W PWR design. The models will be oriented toward identification and evaluation of the impact of system interaction and failure dependencies of control systems identified in Task 1. The models will employ the use of different codes. EG&G will utilize existing RELAP 5 codes and ORNL will utilize a hard-wired analog computer for modeling the control systems and a modified RETRAN and RELAP code for the plant dynamic model. Extensive use of existing and verifiable codes and models will be utilized. Additional modeling will be developed for the control systems and for the necessary secondary flow loops. We plan to develop the models as necessary to simulate as close as possible the plant-specific characteristics of the four plants under review. Computer simulations of postulated scenarios will be performed to determine if plant operating or safety limits (identified in the specific Technical Specifications and in NUREG-0800) are exceeded. When plant operating or safety limits are exceeded then the respective event sequences will be identified as requiring further analysis to determine the risk involved in the particular event sequence and considered in Tasks 5 and/or 6. As a result of this task it is anticipated that the lists of systems identified in Task 1 will be modified. During this phase an assessment will be made as to the possibility of utilizing any other dynamic models in part or in whole, already developed by others to simulate the plant-specific characteristics of the plants under review or for verification testing of the models that will be developed. The benefits of using the models developed for the LOFT project, or the use of the Tennessee Valley Authority simulators, or the capability to use the NSSS vendor engineering simulators will be evaluated.

Task 3 Identify the Failure Modes of the Safety Significant Systems

Identify the potential failure mechanisms (that is, root causes) of the control systems that have been identified as a result of the collective activities described in Tasks 1 and/or 2. The information learned as a result of the LER reports, IE Bulletins and Orders and other applicable documents (such as failure rate data) will be considered in the evaluation to identify credible failure modes and to assess the likelihood of their occurrence. Additional FMEA and fault tree analysis may need to be performed on a sub-system (that is, component) level on selected systems to identify the mechanistic failure modes that can occur and to assess methods for corrective actions. The need for additional analysis will be evaluated on a case-by-case basis. The relative importance of the control system, its complexity and its dependence on environmental conditions and on other systems will be a factor for implementing any additional analysis. During this phase failure modes due to short or open circuits, loss of environmental support systems, loss of power supply, abnormal environmental or seismic effects will be considered. As part of the activities conducted by the contractors, consideration of control system failures resulting from seismic or harsh environment will be limited to control systems identified as particularly safety significant. For such systems (which will be identified via Tasks 1 and 2) consideration will be given to assess the effect of their multiple failures such as might result from seismic events, and from harsh environments caused by accidents. For these systems, an evaluation will be made to determine whether or not seismic or harsh environmental qualification needs to be recommended, and to what extent the qualification requirements used for safety systems should be imposed. The contractors should perform a value/impact assessment to support any recommendations that are to be considered by the staff. It should be noted that a systematic investigation of the effects of all seismically or environmentally-induced control system failures is not within the scope of this program. Also, sabotage is not within the scope of this activity. Operator action will be addressed to the extent of assessing if credit can be given to the operator in mitigating certain selected transients caused by control system failures. This assessment will be limited to assuring that the procedures to mitigate these limited transients are adequately written and relatively simple for the operator to correctly accomplish the task in the time allowed, and that sufficient information and time is available to the operator to assess the conditions that exist. The USI A-47 program will also perform a data gathering function and document the type and number of non-safety grade control system failures that have occurred (based on the LER or operating history reviews) or could occur (based on the FMEA or simulator studies conducted in Tasks 1 and 2) to cause the number of those design transients bounded by the FSAR to exceed the number of trips allowed by the design basis.

Task 4 Evaluate the Effects of Loss of Power Supply to the Control Systems. [Including electric (ac and dc) pneumatic, and hydraulic power sources.]

Numerous incidents have occurred in nuclear generating plants involving loss of power in the non-safety grade instrumentation and control systems. These incidents resulted in reactor and turbine trip; the opening of the pressurizer power operated relief valves, and code safety valves; discharge of a significant amount of primary coolant into the containment building; and, the loss of display instrumentation in the control room. The transients and the loss of equipment function produced as a result of these incidents significantly impact the operator's ability to proceed to safe shutdown conditions in an orderly manner. The purpose of this task is to evaluate the effects of loss or degradation of the safety grade or non-safety grade power supplies which provide power to the non-safety grade instrumentation and control system identified in Tasks 1 and 2. The evaluation will include the effects of the loss of ac and dc electrical power sources and loss of any applicable pneumatic and hydraulic power sources that operate any important valves. The evaluation will be limited to the loss or degradation of a single power supply and multiple power supply failures that result from a single (source) failure or event. The control systems of the four plant designs will be reviewed. The review of this task will be integrated as part of a review effort associated with the other tasks identified in this plan, and will consist of the following:

(a) Coordinate activities with the findings of USI A-44, "Station Blackout," and NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants," April 1981, and integrate any applicable requirements and information developed as a result of that activity.

(b) Consider the licensees' evaluations and responses to IE Bulletin 79-27, "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation," November 30, 1979. This subtask will complement the review of IE Bulletin 79-27 and evaluate ac and dc bus power supply failures of the relevant power distribution systems (not limited to 120V systems) on important non-safety equipment and systems. If the non-safety grade equipment is powered from a safety bus, the effects of bus degradation on the safety loads connected on that bus will also be evaluated.

(c) Identify and document the control systems that have a significant safety impact due to power supply failures (this will be a specific sub-group of the systems identified in Tasks 1 and 2). Evaluate the effects of a loss of power to the display instrumentation of these systems. Using the criteria and guidance proposed in Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environment Conditions During and Following an Accident," determine to what extent the problems found would be resolved by implementing this guide. Verify the adequacy of existing criteria or develop additional criteria (if ncessary) to minimize the consequence of such power failures. Assess the reliability of the non-safety grade electrical bus, by evaluating the existing operating history. The effects of the non-safety grade bus failures during startup, shutdown, normal power operation and during accident and transient modes of operation will be considered in the evaluation.

(d) Develop and propose criteria (or guidelines) to improve the reliability of non-safety grade power supplies (if necessary) and propose recommendations to improve the capability of the systems to cope with the effects of the system failures identified in subtask (c). Integrate the applicable requirements and information developed as a result of the IREP studies conducted on Calvert Cliffs 1, Millstone 1, Arkansas Nuclear One Unit 1 and Browns Ferry 1, and those identified in subtask (a). In addition, integrate the applicable information that is developed as a result of the ORNL plant electrical systems studies (FIN No. B-0816).

Task 5 Determine the Need for Control or Protection System Improvements

Verify the adequacy of the existing criteria for control systems, defined in (a) the Standard Review Plan Section 7.7 (NUREG-0800) and (b) applicable Branch Technical Positions. Review the activities and approaches used by the International community to (1) minimize control system failure, and (2) improve control system reliability. Evaluate the need for additional non-safety grade control systems or the need for additional safety grade protection systems. During this phase, assessing the need for improved or additional operator action to recognize and to mitigate specific transients resulting from control system failures will be made. Recommendations concerning improvements to the existing control, protection and power systems, and the need for additional equipment, such as high level alarms, level controls or interlocks to minimize postulated faults will be justified on the basis of cost effectiveness and risk to safety. The adequacy of existing staff positions regarding certain design requirements for control systems such as the sharing of common sensor lines between safety and non-safety systems will be evaluated in light of the knowledge gained through the operating history (that is, via LERs and IE Bulletins, etc.). The need for improved or additional surveillance testing to improve the reliability of the non-safety systems will also be evaluated and proposed if warranted.

Task 6 Provide Design Criteria for the Evaluation of Control Systems

Develop and propose (if necessary) additional criteria or guidelines to improve system reliability and minimize control system failures that (1) could lead to transients more severe than predicted in the plant FSAR accident analysis, and (2) could cause transients that could frequently and severely challenge safety grade systems.

As a result of this study and at the completion of this task, a report will be issued describing the conduct and conclusions of tasks identified above. Recommendations (if any) for control system or protection system modifications will be provided separately as proposed revisions or additions the to Standard Review Plan, the Regulatory Guides, or the NRC Regulations.

Task 7 Identify Control Systems That Could Lead to Steam Generator Reactor Vessel Overfill and Overcooling Transients

As part of the overall review effort, the initial focus will be to:

- Evaluate Control Systems that could lead to a steam generator or reactor vessel overfill transient. (Subtask 1)
- Evaluate control system failures that could lead to a reactor overcooling transient. (Subtask 2)
- Identify the lessons that have been learned from past control system failures from the LERs, the IE Bulletins and Orders, the applicable applicant responses and from independent utility studies.

The objective of Subtask 1 is to identify automatic and manual control systems whose failure have the potential for causing steam generator or reactor vessel overfill. The objective of Subtask 2 is to identify those control systems whose failure or malfunction can contribute to an overcooling transient in the primary system of sufficient magnitude to initiate repressurization via the automatic initiation of the safety injection systems. The criteria that will be used for selecting and categorizing significant control systems for these tasks will be defined. A candidate criteria for identifying significant systems for Subtask 1 may be one whose failure or malfunction may lead to water ingress (or significantly increase moisture carryover or steam quality in the main steam line steam space). This water ingress may lead to a loss of existing safety systems (that is, the loss of auxiliary feed pump turbines) or cause undue stress to the steam lines. The screening criteria for Subtask 2 will be developed with assistance from USI A-49. This assistance will be in the form of defining important event sequences and describing unacceptable pressure-temperature conditions that may occur as a result of selected control failures. The approach and methodology outlined in Tasks 1 through 6 will be utilized for resolution of these subtasks.

As part of a separate subtask conducted for USI A-49, RES has contracted ORNL (FIN No. B-0468) to perform a study of PTS, including as one subtask, the control and safety system design for each of the three PWR vendors (the same plants will be studied for this task.) One purpose of the contract is to provide details of the control and safety functions that could contribute to PTS events. We plan to utilize the control system information developed on that subtask and include their findings in our evaluation. At the same time, we expect that the results from efforts related to USI A-47, including those under FIN Nos. B-0467 and B-0816 at ORNL (see Section 5), to contribute to the resolution of USI A-49.

Proposed recommendations in the form of guidelines or criteria will be developed (if necessary) for control system modification or for additional protection system functions which would minimize the impact of control system failures or malfunctions that could contribute to significant steam generator or reactor vessel overfill transients and/or pressurized overcooling transients.

As a result of these studies and at the completion of Subtasks 1 and 2 a report will be issued describing the technical results and findings. A report will also be issued to summarize the lessons learned from the study of the applicable LERs, IE Bulletins and Orders and from the other information identified in Task 1. Recommendations for new or modifications to existing requirements (if any) will be provided separately as proposed revisions or additions to the Standard Review Plan or the Regulatory Guides.

3. BASIS FOR CONTINUED OPERATION OR LICENSING PENDING COMPLETION OF PROGRAM

As previously noted, the NRC staff has performed instrumentation and control system reviews on licensed plants and is currently reviewing, on a case-by-case basis, the NTOL plants. The goal of the reviews is to verify that the control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety protection system equipment required to trip the plant or maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident." These reviews are performed utilizing, in whole or in part, the guidelines and criteria identified in Standard Review Plan Section 7.7.

With the recent emphasis on the availability of post-accident instrumentation (Regulatory Guide 1.97), the staff reviews evaluate the designs to assure that control system failures will not deprive the operator of information required to maintain the plant in a safe shutdown condition after any "anticipated operational occurrence or accident." For the NTOL reviews, the applicants are requested to evaluate their control systems and identify any control system whose malfunction could impact plant safety. The licensees are requested to identify the use (if any) of common power supplies, and the use of common sensors or common sensor impulse lines whose failure could have potential safety significance. The results of these reviews and the staff's evaluation for the NTOLs are documented in the Safety Evaluation Reports on a case-by-case basis.

In addition, a specific set of "accidents" has been analyzed to demonstrate that plant trip and/or safety system equipment actuation occurs with sufficient capability and on a time scale such that the potential consequences to the health and safety of the public are within acceptable limits. In these analyses, conservative assumptions have been used. The conservative analyses performed and the "accidents" chosen for the analyses are intended to demonstrate that the potential consequences to the health and safety of the public are within acceptable limits for a wide range of postulated events even though specific actual events might not follow the same assumptions made in the analyses.

Several activities that have been completed or are still ongoing which address the effects of control system failures have been conducted by the NSSS vendors. B&W has completed a failure modes and effects analysis and a review of operating experience for their Integrated Control System (ICS) and reported the results in B&W Report BAW-1564, "Integrated Control System Reliability Analysis," August 1979. The staff completed its review of BAW-1564 through a technical assistance contract with ORNL (Memorandum from R. Satterfield to P. S. Check, "Assessment of B&W Report 1564, 'Integrated Control System Reliability Analysis'," dated May 9, 1980). As a result of this review, both the staff and ORNL concluded that the ICS itself had a relatively low failure rate and did not appear to initiate a significant number of plant upsets. Failure statistics revealed that only approximately 6 of 162 hardware malfunctions resulted in reactor trip. ORNL has further concluded that the B&W analysis shows that anticipated failures of and within the ICS are adequately mitigated by the plant safety systems and many potential failures would be mitigated by cross-checking features of the control system without challenging the plant safety systems. In BAW-1564, B&W recommended six actions regarding control system improvements which could be made to improve overall plant performance. In November 1979, the licensees with B&W plants (except Three Mile Island Unit 1) were requested to evaluate the B&W recommendations and report their followup actions. Subsequently, the responses have been reviewed and found acceptable by ICSB.

Also, the licensees have been requested (IE Information Notice 79-22, "Qualification of Control Systems," September 14 and 17, 1979) to review the possibility of consequential control system failures which exacerbate the effects of high energy line breaks (HELB) and adopt design changes or new operator procedures where needed, to assure that the postulated events would be adequately mitigated. All licensees responded to the request and the responses were screened. On the basis of the review, no specific event leading to unacceptable consequences was identified and, in general, control equipment locations were such that consequential failures would be unlikely. Some licensees did make changes to their operating procedures to address the possibility of control failures. As part of the staff's ongoing review of the adequacy of the equipment qualification program on NTOLs, and in response to IE Bulletin 79-01, "Environmental Qualification of Class IE Equipment," February 8, 1979, for all operating reactors, the staff is re-evaluating the qualification programs to assure that equipment that may potentially be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile environment.

The equipment qualification evaluations are conducted on a case-by-case basis. The staff reviews for all operating plants will be documented in the supplemental Safety Evaluation Reports. For NTOLs, the staff reviews will be completed before operating licenses are granted.

In addition, IE Bulletin 79-27 was issued to licensees requesting that evaluations be performed to ensure the adequacy of plant procedures for accomplishing shutdown upon loss of power to any electrical bus supplying power for instruments and controls. In their responses to the IE Bulletin, licensees have indicated that corrective action has been taken including hardware changes and revised procedures, where required, to assure that the loss of any single instrument bus would not result in the loss of instrumentaton required to mitigate such an event. As part of Operating License licensing reviews, ICSB is requesting that similar reviews be conducted by the NTOL applicants.

Based on the activities identified above and the ongoing NTOL case review activities, continued licensing and operation of PWRs and BWRs is acceptable pending completion of this program.

4. NRC TECHNICAL ORGANIZATIONS INVOLVED

A. Division of Licensing (DL)

DL will provide the coordination necessary to expedite and collect system design information on four operating reactors. The information needs will be to procure system piping and instrumentation designs and flow and logic diagrams for the non-safety grade control systems. Associated control equipment support system design schematics, such as power supply systems, will also be needed. DL will provide assistance to the Task Manager for setting up and coordinating with the utility personnel, information meetings and site visits that may be necessary. DL will also provide assistance to the Task Manager for integrating any relevant experience and any new requirements resulting from the activities identified in USI A-47. DL will contribute to the review and approval of any licensing requirements and guidelines developed as a result of this USI, and will provide review and comment on the technical evaluations provided by the Task Manager.

Manpower Requirements

	Total	FY83	FY84	FY85
Operating Reactors Branch No. 1	0.25 psy*	.05	.15	0.05
Operating Reactors Branch No. 2	0.25 psy	.15	.05	0.05
Operating Reactors Branch No. 3	0.25 psy	.05	.15	0.05
Operating Reactors Branch No. 4	0.25 psy	.10	.10	0.05
Operating Reactors Assessment Branch	0.375 psy	.15	.15	0.075

*Assumed 1 professional staff year (psy) = 40 person weeks.

B. Division of Systems Integration (DSI)

DSI will provide review and comment on technical evaluations provided by the Task Manager in the areas of instrumentation and control, electrical power, the reactor and auxiliary plant designs, and accident analysis. The Instrumentation and Control Systems Branch and the Power Systems Branch will provide assistance for the purpose of integrating relevant experience and any new requirements and guidelines stemming from the completion of the subtasks described in USI A-47. The Reactor Systems Branch and the Auxiliary Systems Branch will assist in the development of the selection criteria to be used for establishing safety significant control systems (described in Task 1) and will verify completeness of non-safety grade control systems that may be needed in mitigating the accidents and transients analyzed in Chapter 15 of the plant FSAR. In addition DSI will contribute to the formulation, review and approval of the recommendations, criteria and guidelines developed during at the completion of the tasks (described in USI A-47). DSI will also review and comment on the draft overfill and overcooling event evaluations and final NUREG Report.

Manpower Requirements

	Total	FY83	FY84	FY85
Instrumentation and Control Systems Branch	0.50 psy	.20	.15	0.15
Power Systems Branch	0.35 psy	.15	.15	0.05
Reactor Systems Branch	0.65 psy	.25	.25	0.15
Auxiliary Systems Branch	0.225 psy	.075	.10	0.05

C. Division of Human Factors Safety (DHFS)

DHFS will provide review and comment on those technical evaluations involving man/machine interfaces. DHFS will contribute to the formulation, review and approval of recommendations, criteria and guidelines involving man/machine interfaces developed during the completion of the tasks. In this area DHFS will contribute in the development of maintenance or testing requirements (if warranted) for non-safety control systems.

Manpower Requirements

	Total	FY83	FY84	FY85
Human Factors Engineering Branch	0.20 psy	.05	.10	0.05
Procedures and Test Review Branch	0.20 psy	.05	.10	0.05

D. Division of Safety Technology (DST)

DST will provide overall management of the program to resolve this USI. DST will provide liaison between NRR and RES and provide coordination of activities performed within NRR which are part of this Task Action Plan. DST has primary responsibility for the review of the draft recommendations and guidelines and for coordination of the internal management and the public review process required to adopt the recommendations and guidelines into licensing requirements. 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 non-safety control systems.

DST will provide assistance to the Task Manager for the purpose of integrating relevant experience and any new requirements stemming from the completion of those activities related to USI A-47 for which DST has responsibility. Those activities include RRAB system interaction studies, and the USI A-49 and USI A-44 activities referenced in previous sections of this plan.

In addition, RRAB will provide technical support in the area of reliability and risk assessments on non-safety control systems that have been identified as safety significant. The Safety Program Evaluation Branch will provide techni l support on the cost/benefit evaluations associated with the recommendations and positions developed on each of the subtasks. DST will also coordinate the revision and publication of the NUREG report and coordinate the issuance of other licensing documents such as

Regulatory Guides, Rules, and the Standard Review Plan with the Division of Engineering Technology.

Manpower Requirements

	Total	FY83	FY84	FY85
Generic Issues Branch	3.25 psy	1.00	1.25	1.00
Reliability and Risk Assessment Branch	.30 psy	.10	.10	0.10
Licensing Guidance Branch	.15 psy	.05	.10	0.00
Safety Program Evaluation Branch	.30 psy	.10	.20	0.00
Research & Standards Coordination Branch	.15 psy	.05	.10	0.00

E. Office of Analysis and Evaluation of Operational Data (AEOD)

AEOD will provide review and comment on the technical evaluations provided by the Task Manager. AEOD will provide assistance to the formulation, review and comment of the recommendations and guidelines developed (primarily on Subtask 1). AEOD will also provide assistance to the Task Manager for the purpose of integrating relevant experience for which AEOD has responsibility.

Manpower Requirements

	Total	FY83	FY84	FY85
Plant Systems Unit	0.25 psy	.05	.10	.10

5. ASSISTANCE FROM RES DIVISIONS

Close coordination and cooperation will be required on USI A-47 between NRR and RES. RES assistance will be required from the Division of Facility Operations, Instrumentation and Control Branch (ICB). ICB through contracts with ORNL, will develop the PWR simulator models (discussed in Tasks 1 through 3) as a specific input for the activities outlined in the USI A-47 Task Action Plan. In addition, RES (FIN No. B-0467) will conduct a review on two PWR designs discussed in this Task Action Plan and will perform the activities identified in Tasks 1 through 7 on each of these plants in conformance with the schedule identified in Table 2. RES will also provide a draft report on each of the plants reviewed. The report will include the content of the information described in Tasks 1 through 7.

Any control systems identified by RES to be generic will be identified in USI A-47. In addition the Division of Risk Analysis will provide technical input from USI A-44, "Station Blackout" relative to loss of power to the vital buses associated with non-safety control systems. Also, any applicable information developed by the ORNL plant electrical systems study (FIN No. B-0816) that would enhance a more complete understanding of significant interactions between the electrical power and the electrical control systems will be factored into the overall evaluation if the information is available and compatible with the schedule for resolution of this task.

Manpower Requirements

	Total	FY83	FY84	FY85
Instrumentation and Control Branch	2.30 psy	.800	.85	.65
Division of Risk Analysis	.475 psy	.075	.20	.20

(The manpower requirements for RES/ORNL activities are summarized in Table 1).

6. TECHNICAL ASSISTANCE

Technical assistance to the program will be required for the activities identified in Tasks 1 through 7. Contracts will be made with the National Laboratories to conduct the studies and activities described in Section 2 of this plan. Funding will be provided by NRR and RES. The estimated costs are shown in Table 1. The proposed schedule for task resolution is shown in Table 2. Should additional evaluations of other plant designs be needed, a significant cost increase will take place. Such costs are not included in the cost estimates shown in Table 1.

The funding associated with the RES activities related to USI A-47, (specifically FIN Nos. B-0467, B-0816 and B-0468) are funded directly by the Division of Facility Operations, RES. These related activities are a part of a large overall research program which is beyond the scope of Task Action Plan A-47.

The funding associated with the RSB activities at Los Alamos related to USI A-47 on the steam generator tube rupture evaluations (specifically FIN A-7281) are funded by the Division of Systems Integration, NRR.

7. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

Interaction with outside organizations will include the NSSS vendors, utilities, the architect/engineers, the Electric Power Research Institute, Institute for Nuclear Power Operations, ORNL, Sandia National Laboratories, and EG&G-Idaho.

The activities of USI A-47 will be coordinated with the appropriate Advisory Committee on Reactor Safeguards (ACRS) subcommittee. Significant information will be provided to the subcommittee as it becomes available and meetings will be scheduled at appropriate times. Peer review will be conducted through ACRS briefings and by establishing a peer review panel (if necessary) selected from outside NRC having appropriate expertise. In addition, as Task 5 progresses, it will be necessary to establish a strong interaction and information exchange with the International community. Attendance at International conferences and/or site visits to selected foreign utility agencies and consultants is anticipated.

8. POTENTIAL PROBLEMS

A. Traditionally, the licensees were not required to provide design and operating experience on non-safety grade control systems, and therefore complete information on the final "as-built design" for these systems (that is, schematics, flow logic diagrams and system descriptions) and operating experience is difficult to obtain. The information gathering task is a substantial and an important one. An adequate and timely resolution of this USI relies on obtaining this needed information.

B. Performance of selected tasks described in Tasks 1 through 7 by NRR will require participation from members of DSI, DL and RES at various intervals throughout the program. Assignments of selected personnel, at specific intervals, will be required. Close coordination and cooperation is needed within NRR (for example, USI A-49) and between NRR and RES (for example, ORNL).

C. Development of appropriate reliability/safety goals for specific non-safety grade control systems and translation of these goals into licensing requirements.

D. Uncertainty as to the applicability or compatability of the information that will be available from IREP, systems interaction studies, and other ongoing reliability and risk assessment studies for use on USI A-47. The completion schedules of these activities may not be compatible with USI A-47. Uncertainty as to whether the information obtained from these activities can be used for a generic study.

E. Availability of the Baltimore Gas and Electric simulator for the Calvert Cliffs-1 evaluation (scheduled to begin in October 1984).

F. Uncertainty as to whether the information gained from the evaluation of the four plant-specific NSS designs can be utilized to formulate generic conclusions.

Table 1. Resource Requirements Summary

		FY82	FY83	FY84	FY85	FY86
Contract Dollars for Technical Assistance (in thousands) at INEL (FIN No. A-6477)		142	779	444	100	0
Contract Dollars for Technical Assistance (in thousands) at Los Alamos for the Reactor Systems Branch (FIN No. A-7281)		--	--	--	100	--
Contract Dollars for Technical Asistance (in thousands at ORNL)						
	FIN B-0467	636	950	1200	1200	0
	FIN B-0816	350	300	400	400	0
NRR Manpower in Person-Years						
DST	GIB		1.00	1.25	1.00	0.5
	SPEB		0.10	0.20	0.00	
	RRAB		0.10	0.10	0.10	
DSI	RSB		0.25	0.25	0.15	
	ICSB		0.20	0.15	0.15	
	CSB					
	ASB		0.075	0.10	0.05	
	PSB		0.15	0.15	0.05	
	CPB					
	AEB					
	ETSB					
	RAB					
DE	MEB					
	SEB					
	GSB					
	HGEB					
	MTEB					
	CHEB					
	EQB					
DSHF	HFEB		0.05	0.10	0.05	
	OLB					
	LQB					
	PTRB		0.05	0.10	0.05	

Table 1. Resource Requirements Summary

(Continued)

		FY82	FY83	FY84	FY85	FY86
DL	ORAB		0.15	0.15	0.075	
	ORB (total)		0.35	0.45	0.20	
RES	ICB		0.80	0.85	0.65	
	DRA		0.075	0.20	0.20	
AEOD	PSU		0.05	0.10	0.10	
	MANPOWER TOTAL	0	3.40	4.15	3.725	0.5

Figure 1. Flow Diagram for Resolution of USI A-47

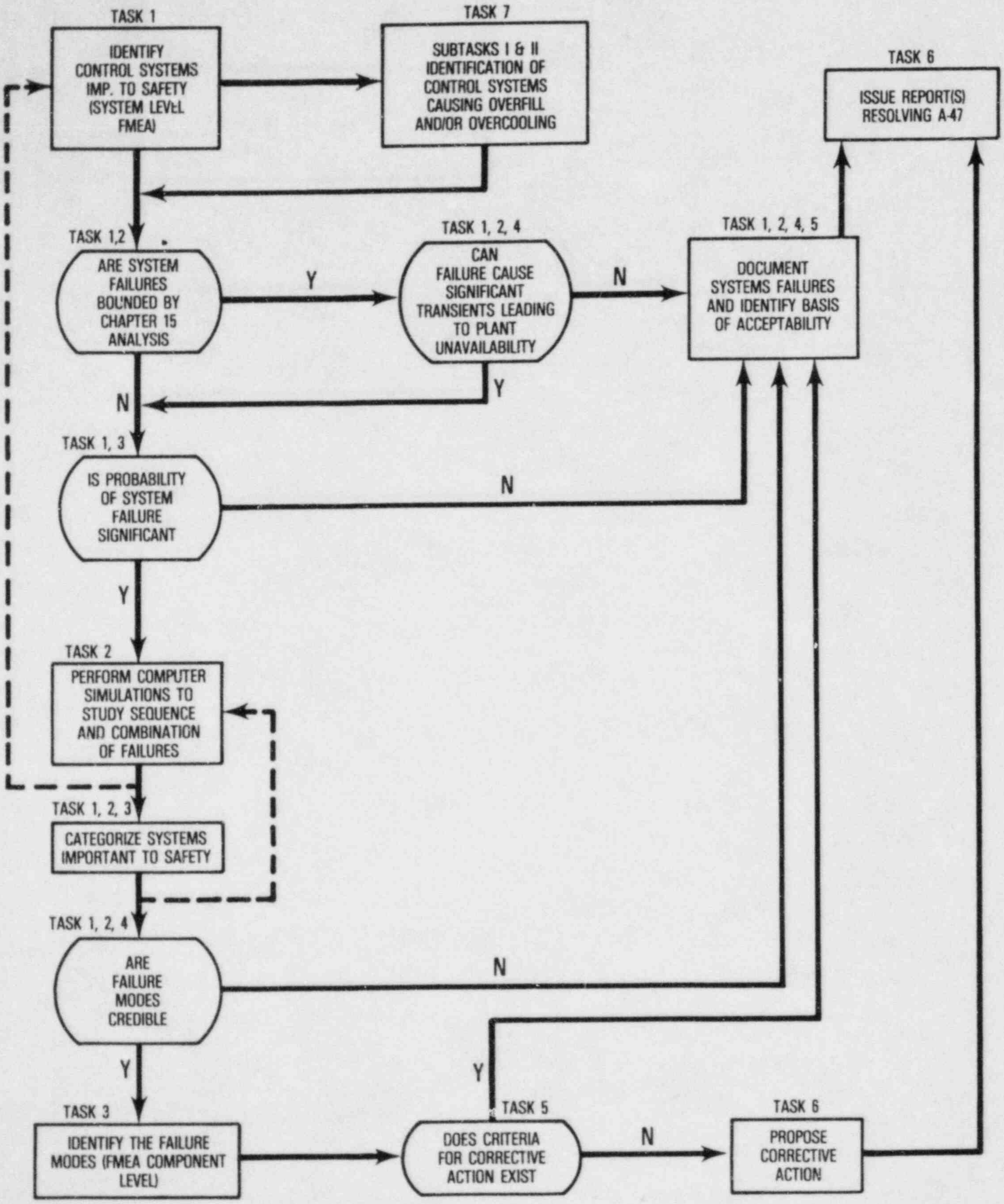

Table 2. Proposed Schedule for A-47

Receive Draft Final Reports from Contractors (INEL and ORNL)	
GE Plant Review	04/04/84
B&W Plant Review	09/84
W Plant Review	07/01/84
CE Plant Review	05/01/85
Draft Technical Resolution DST Draft Complete	05/15/85
Staff Comments to DST	06/01/85
Completed Package to Director NRR	06/15/85
Package to CRGR	07/15/85
CRGR Review Complete	08/15/85
Issue for Public Comments	09/15/85
Technical Resolution for Staff Comments Complete (Incorpration of Public Comments)	12/02/85
Staff Comments to DST Complete Division Review	12/16/85
Complete Package to Director, NRR	01/01/86
Package to CRGR	02/01/86
CRGR Review Complete	03/01/86
Issue Final NUREG	04/01/86

Resource Requirements Ssummary

		FY 83	FY 84	FY 85	FY 86
Contract Dollars for Technical Assistance [in thousands at INEL (because of W late start some carryover $ from 83 to 84 is expected)].		865K	125K		
Contract Dollars for Technical Assistance (in thousands at ORNL)		936K	1435K		
NRR Manpower in Person-Years					
DST	GIB	1.00	1.25	1.00	
	SPEB	0.10	0.20		
	RRAB	0.10	0.10	0.10	
DSI	RSB	0.25	0.25	0.075	
	ICSB	0.20	0.15	0.075	
	CSB				
	ASB	.075	0.10		
	PSB	0.15	0.15	0.05	
	CPB				
	AEB				
	ETSB				
	RAB				
DE	MEB				
	SEB				
	GSB				
	HGEB				
	MTEB				
	CHEB				
	EQB				
DHFS	HFEB	0.05	0.10		
	OLB				
	LQB				
	PTRB	0.05	0.10		
RES or other;	ICB	0.80	0.85		
	DRA (RES)	0.075	0.15		
	AEOD	0.05	0.10		
	ORAB	0.15	0.15		
	ORB	0.20	0.60		

TASK ACTION PLAN, REVISION 1
(March 1984)

HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (TASK A-48)

Lead Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
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Office of Nuclear Regulatory Research (RES)	John Larkins Division of Accident Technology
	Morton Fleishman Division of Risk Analysis
	William Farmer Division of Engineering Technology
Applicability:	Light Water Reactors (PWRs and BWRs)
Projected Completion Dates:	Mark I & II Containments - Complete
	Lead Ice Condenser Containment (Sequoyah) - September 31, 1984
	Final Hydrogen Rule for Ice Condenser/ Mark III - September 1984
	Lead Mark III Containment (Grand Gulf) - December 1985
	Ice Condenser/Mark III Summary Report - June 1986

1. DESCRIPTION OF PROBLEM

Following a loss-of-coolant accident (LOCA) in a light water reactor plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coatings and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of a LOCA, Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors," and General Design Criterion 41, "Containment Atmosphere Cleanup," in Appendix A to 10 CFR Part 50 require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated LOCA to ensure that containment integrity is maintained. 10 CFR Section 50.44 requires that the amount of hydrogen contributed by the core metal-water reactions as a result of degradation of the emergency core cooling system, be assigned to be either five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

Conventional hydrogen control systems (for example, hydrogen recombiners) have historically been installed to provide the capability to control the hydrogen accumulation as a result of radiolytic decomposition of water, corrosion of metals inside containment, and hydrogen producing reactions of coatings and insulation. The design capability or margin to control the contribution of hydrogen accumulation resulting from a metal-water reaction involving the fuel cladding has historically been provided in pressurized water reactors (PWR) facilities by the net free volume inside the containment structure. Boiling water reactor (BWR) facilities with small pressure suppression containments have utilized inerted containments. For the PWR plants, the containment volume was large enough such that hydrogen generated and released from the cladding reaction would not reach a uniform concentration approaching the lower limit of flammability. The reason for this approach is that the amount of metal-water reaction that had to be postulated was small (that amount consistent with a design basis accident). Also, the rate of hydrogen release as a result of cladding reaction is assumed to be rapid following a postulated accident (on the order of minutes). This corresponds to a release rate beyond the capability of conventional hydrogen control systems. However, the containment net free volume was found to be sufficient for providing the initial protection, and hydrogen control systems (recombiners) would be actuated later to control hydrogen accumulation from the other sources and gradually reduce the hydrogen concentration inside containment.

The accident at Three Mile Island-2 (TMI-2) on March 28, 1979 resulted in a metal-water reaction which involved hydrogen generation well in excess of the amounts specified in 10 CFR Section 50.44. As a result, it became apparent to the NRC that additional hydrogen control and mitigation measures would have to be considered for nuclear power plants with small containments. This topic was first addressed in the Lessons Learned report (NUREG-0578) and subsequently included in the TMI Action Plan, NUREG-0660 (Item II.B.7). As part of--and as a result of--these considerations, it was determined that rulemaking proceedings should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core must be taken into account in plant design.

The rulemaking proceedings take the form of three rules that provide prompt resolution of the hydrogen issue for those small and intermediate volume containments which have limited ability to accommodate the large quantities of hydrogen associated with a severe degraded core accident. The NRC rulemaking proceedings also include a long term severe accident rule.

Beginning early in 1980, a number of technical programs were initiated to investigate the control of large amounts of hydrogen in small volume Shortly thereafter, the owners of the ice condenser plants (Tennessee Valley Authority (TVA), Duke Power and American Electric Power Company) formed the Ice Condenser Owners Group (ICOG) to conduct a joint research and development program. Later, owners of BWR-6/Mark III containments formed a similar owners group, the Hydrogen Control Owners Group (HCOG) to jointly sponsor hydrogen research and development efforts for the Mark III containment design. In addition to these programs, Industry Degraded Core Rulemaking (IDCOR), the Electric Power Research Institute (EPRI), the Department of Energy (DOE) and NRC have initiated hydrogen related research and development programs.

The Office of Nuclear Reactor Regulation (NRR) has been conducting an evaluation of the hydrogen control systems proposed for the lead ice condenser and Mark III plants since 1980. These reviews have been conducted on an individual basis as a part of the licensing review for the lead plants. In 1981 the Commission designated the hydrogen issue as an Unresolved Safety Issue (USI) to ensure integration of the various case-specific NRC review efforts into a unified program with NRC management attention. The purpose of Task Action Plan A-48 is to describe the NRC program to reach this goal.

2. PLAN FOR PROBLEM RESOLUTION

A. General Approach

Following the TMI accident a number of hydrogen-related rulemaking, research, and plant-specific review efforts were initiated to address hydrogen control in plants with small and intermediate volume containments. The NRC determined that for these containments, priority consideration should be given to safety questions associated with the control of large quantities of hydrogen that could result from a severely degraded core accident. The function of this USI program is to provide a focus for the NRC rulemaking and technical review efforts associated with this safety concern.

This program is concerned with the technical resolution of the hydrogen issue for each of the following categories of containment:

- small volume Mark I and II pressure suppression containments;

- intermediate volume ice condenser and Mark III pressure suppression containments.

The major elements of this program include:

- near term hydrogen rulemaking tasks;

- ice condenser and Mark III lead plant implementation reviews;

- issuance of a Summary Report describing the generic resolution of this issue for Mark III and ice condenser containments.

The NRC initiated several hydrogen related rulemaking efforts following the TMI accident including three near term rules and the longer term severe accident rule. The scope of this USI is limited to the near term rulemaking efforts. These rules are currently either final rules or they are proposed and anticipated to be published as a final rule by September 1984. Possible hydrogen control requirements resulting from the severe accident rule have been excluded from this USI. The rulemaking subtasks follow three NRC hydrogen rulemaking activities through to their publication as final rules. The three rules include: the Mark I and II containments hydrogen inerting rule; the ice condenser/Mark III containment hydrogen control rule; and the near term construction permit/manufacturing license (CP/ML) rule. The CP/ML rule specifies hydrogen licensing requirements for pending CP and ML applications.

In addition to these rules the Commission published in the Federal Register, an advance notice of proposed rulemaking (45FR65474) on October 2, 1980. This rule is commonly referred to as the Severe Accident Rule. The Severe Accident Rule may address additional measures for the control of large amounts of combustible gases beyond those specified in the three rules described above. This rule goes beyond the near term hydrogen concerns for small containments growing out of TMI. Two to four years may elapse before the hydrogen control requirements of this rule are developed and implemented. Furthermore, the applicability of this rule to current plants is uncertain at this time. Considering the above, the hydrogen requirements of the severe accident rule have been excluded from the scope of USI A-48.

The plant-specific hydrogen review subtasks are included in this program to follow the plant-specific technical review of hydrogen control systems for the lead ice condenser containment and the lead Mark III containment plants. Shortly after the TMI accident, the Commission determined that measures should be taken for the control of large quantitites of hydrogen in small Mark I and II containments and intermediate volume ice condenser and Mark III containments. Operating Mark I and II containments have installed containment inerting systems to satisfy the requirements of the inerting rule. For the intermediate volume containments, a significant amount of work has already been conducted on a plant-specific basis to establish that ice condenser and Mark III containments can accommodate the large quantities of hydrogen associated with a degraded core accident. This work laid the groundwork for the proposed Mark III/ice condenser hydrogen control rule. Plants with these containment designs have elected to install a distributed ignition system for hydrogen control. The staff has completed the interim review of these systems for initial operation of the lead ice condenser and Mark III plants. Licensing conditions have been imposed on the lead plants requiring that a utility-funded hydrogen control research program be conducted to confirm the acceptability of the installed control systems. These licensing conditions may be modified or new conditions may be developed in the future. The USI A-48 program will follow these lead plant programs through to the point that the hydrogen-related licensing conditions are removed on Sequoyah and Grand Gulf.

Dry containments have been excluded from USI A-48. In addition, the staff is considering dropping the requirement of the proposed hydrogen rule that dry containments be analyzed for degraded core hydrogen combustion. This is because dry containments have a much greater ability to accommodate the large quantitites of hydrogen associated with a degraded core accident than the small Mark I, II, III and ice condenser containments. Most dry containments have about two million or more ft^3 of net free volume. Assuming 50% metal-water reaction in the core, the resulting uniformly mixed concentration of hydrogen in the containment is about 10%. This is well below the concentration for detonation and even below the limits for combustion if there were more than 50% steam in the containment atmosphere. Sixty percent is the upper bound estimate of the amount of metal-water reaction generated in the TMI-2 accident. The design pressure for these large containments ranges from about 45 to 60 psi. Analyses performed on the Zion and Indian Point plants show that the pressure capabilities are greater than twice the design pressures.

Preliminary calculations have been performed to determine the pressure in a dry containment resulting from the combustion of hydrogen corresponding to a 75% metal-water reaction, following onset of a large loss of coolant accident (LOCA) and while the containment is still near its peak pressure.

These calculations for a typical dry containment indicate a peak total pressure below the failure pressure. If the metal-water reaction were to occur well after onset of the large LOCA, when the containment heat removal systems has been able to condense most of the steam, the containment pressure would be reduced. Under these conditions, a substantial margin would exist for the hydrogen generated by 75% metal-water reaction.

With regard to the issue of equipment survivability in dry containments under degraded core hydrogen combustion conditions, a survey of equipment damage in the TMI-2 containment following the containment hydrogen burn during the accident indicates that critical safety-related equipment survived the hydrogen burn.

Based on the above, the staff has determined that degraded core hydrogen control problems in dry containments were not serious enough to warrant their consideration in either the proposed hydrogen control rule or in USI A-48. Rather the staff has determined that dry containment degraded core hydrogen considerations should be deferred to the long term severe accident rulemaking* at which time a substantially greater understanding of hydrogen combustion in dry containments will exist as a result of the completion of numerous government and industry sponsored hydrogen research programs. Notwithstanding the above, the significance of hydrogen control for dry containments is not entirely trivial when one considers the volume and pressure capacity of dry containments. In addition the equipment survivability issue for dry containments needs some attention, though not in the context of this USI program. Accordingly, the staff is considering that the issue of hydrogen control in dry containments be treated as part of the severe accident programs.

At the conclusion of the Mark III/ice condenser lead plant hydrogen control review, the staff will issue a generic report as a task in the USI A-48 program, that documents the resolution of the hydrogen issue for the Sequoyah and Grand Gulf lead ice condenser and Mark III plants. In this report these plants will be treated as typical of the ice condenser and Mark III plants.

Since substantial progress has been made in both the development and implementation of hydrogen control methods for the small containments, the resolution of this issue will treat each containment category in separate tasks of USI A-48, with a staggered resolution schedule. Resolution of this issue for the small Mark I and II BWR pressure suppression containments was given the highest priority due to the limited ability of these small volume containments to accommodate large releases of hydrogen and the large number of plants that are in operation that utilize these types of containment. Hydrogen control in the intermediate volume ice condenser containments was next in priority. Several PWR plants (such as Sequoyah, McGuire and D. C. Cook) with the

*Hydrogen control matters for the dry containments may be considered further on the basis of upcoming NRC decisions on the direction of the severe accident rule.

ice condenser containments are currently in operation. Final resolution of this issue for these types of containments is anticipated by the end of calendar year 1984. The first domestic BWR plant with a Mark III containment is Grand Gulf. It is currently scheduled for commercial operation in 1984. The NRC has satisfactorily completed the interim review of the Grand Gulf hydrogen control system. The final resolution of the hydrogen issue for Mark III containments is scheduled for December 1985.

B. Technical Content of Major Tasks

Task 1. Near Term Hydrogen Rulemaking

A number of hydrogen-related rulemaking activities were initiated following the TMI accident, to assure that nuclear pcwer plants can safely accommodate the large hydrogen releases that accompany a severe degraded core accident. This task consists of following the near term hydrogen rulemaking proceedings through to their publication as a final rule.

Subtask 1.1 Inerting of Mark I and II Containments

This subtask is complete. This rule was published on December 2, 1981 in the Federal Register. Requirements of this rule (only the USI A-48 related hydrogen provisions of this rule are identified) include an inerted containment atmosphere for Mark I and II containments; provision for either an internal recombiner or the ability to install an external recombiner. The publication of this rule and the established technical feasibility of its inerting requirements resolves this USI A-48 hydrogen issue for plants with Mark I and II containments. Additional non-USI review work on a case-specific basis is being conducted in the line review branches to implement the hydrogen recombiner capability portion of this rule.

Subtask 1.2 Ice Condenser/Mark III Hydrogen Control Rule

This proposed rule was published for comment on December 23, 1981 in the Federal Register (46FR62281). The comment period for this rule expired April 8, 1982. The final rule was submitted for the Commission approval on August 26, 1983. Additional information to justify the staff's position on the final rule was also provided for the Commission on December 28, 1983. Publication of this rule as a final rule is projected for June 1984. The major elements (only the A-48 hydrogen related requirements are identified) of this proposed rule include:

- hydrogen control measures for BWR Mark III and PWR ice condenser containments;

- survivability and qualification requirements for safety systems and components that must function during and after a hydrogen burn;

- analyses of BWR Mark III and PWR ice condenser containments for the consequences of large hydrogen releases;

- containment structural integrity requirements.

All of the above are based on a 75% metal-water reaction of the active fuel cladding.

The staff is currently imposing or will impose hydrogen licensing requirements similar to those in the proposed rule as a part of the licensing of ice condenser and Mark III plants.

Subtask 1.3 Rule for Near Term Construction Permits and Manufacturing Licenses

This subtask is complete. The Near Term Construction Permits and Manufacturing Licenses Rule (NTCP/ML), was published as a final rule on January 15, 1982 in the Federal Register (47FR2286). This rule is not limited to hydrogen issues, it addresses other issues that are an outgrowth of the TMI accident. The major hydrogen-related requirements of this rule for Near Term Construction Permit (NTCP)/ML plants include:

- a hydrogen control system to accommodate hydrogen resulting from metal-water reaction of 100% of the active fuel cladding;

- a maximum, uniformly-distributed post accident hydrogen concentration of 10%;

- facility design to prevent combustible gas mixing problems;

- containment structural integrity requirements;

- a 45 psig minimum pressure capacity for the containment;

- analyses to assure that equipment essential to safe shutdown and containment integrity will function in the environment resulting from the large hydrogen releases specified by this rule.

The requirements of this rule are currently being applied to all plants in the NTCP/ML category. The staff's review of the hydrogen control systems for plants in the NTCP/ML category has been excluded from the USI A-48 program. This was done on the basis that this issue is not a priority safety concern since there are only a few plants in this category and these plants are a number of years away from commercial operation. Furthermore, the few plants in this category vary in their containment designs. The review of the hydrogen control systems for the NTCP/ML plants will be conducted on a case-by-case basis by the line review branches.

Task 2. Plant-Specific Hydrogen Reviews

Shortly following the TMI accident, the Commission determined that hydrogen control systems should be installed in the ice condenser and Mark III containments. These systems were to be designed to handle a large hydrogen evolution, similar to the amount generated in the TMI accident. The purpose of these systems is to maintain containment integrity and to assure that necessary safety equipment will function in the environment resulting from large hydrogen releases. The ice condenser plants and the Mark III plants have proposed and installed similar hydrogen control systems (that is the distributed ignition system). A substantial amount of work has been done since 1980 by both industry and the NRC to establish the acceptability of these hydrogen control systems. This work to support the plant-specific hydrogen control systems plays an important role in the resolution of this USI.

The following technical review areas are included in the NRC review of the hydrogen control systems for the ice condenser and Mark III containment designs in Tasks 2 and 3:

A. Definition of hydrogen sources, release rates, and amounts generated.

- definition of degraded core accident scenarios;

- development of analysis procedures and computer codes.

B. Definition of hydrogen distribution and mixing characteristics in the containment including stratification and pocketing and their effect on equipment survivability.

C. Investigation of hydrogen mitigation systems.

- design basis and qualification of the hydrogen control systems.

D. Capability of containment to withstand hydrogen burns and detonation.

E. Survivability and qualification of safety equipment.

Subtask 2.1 Sequoyah Ice Condenser Review

2.1.1 Interim Ignition System

The TVA proposed and installed within the Sequoyah Unit 1 and 2 containments a system of igniters and ancillary equipment. This system is referred to as the interim distributed ignition system (IDIS). The system was designed to provide a controlled burning of hydrogen in the event that large quantities of hydrogen are generated as a result of a severely degraded core accident. This interim system was installed by TVA and reviewed and approved by the NRC as a condition that they be allowed to exceed 5% power.

The staff's approval of the IDIS was an interim approval to be followed by a staff review of the final system prior to startup following the first refueling outage. As a part of this interim review, industry and NRC studies were conducted to investigate the effectiveness of igniter systems to handle large hydrogen releases. The bases for evaluating the igniter system was the preliminary testing and analyses performed by TVA. This was augmented by NRC confirmatory analyses and testing.

A detailed description of the staff's review of the Sequoyah IDIS is provided in Supplements 2 through 5 to the Sequoyah Safety Evaluation Report, NUREG-0011. In these supplements, the staff concluded that there was reasonable assurance that the IDIS was adequate for an interim period to control hydrogen from a TMI-type degraded core accident. This subtask of USI A-48 is complete.

2.1.2 Final Ignition System

The staff has completed its final review of the Sequoyah permanent hydrogen control system. This has resulted in the following Sequoyah NRC license conditions:

(1) Prior to startup following the first refueling outage, the Commission must confirm that an adequate hydrogen control system for the plant is installed and will perform its intended function in a manner that provides adequate safety margins.

(2) During the interim period of operation, TVA shall continue a research program on hydrogen control measures and the effects of hydrogen burns on safety functions and shall submit to the NRC quarterly reports on that research program.

(a) TVA shall amend its research program on hydrogen control measures to include, but not be limited to, the following items:

(1) Improved calculational methods for containment temperature and ice condenser response to hydrogen combustion.

(2) Research to address the potential for local detonation.

(3) Confirmatory tests on selected equipment exposed to hydrogen burns.

(4) New calculations to predict differences between expected equipment temperature environments and containment temperatures.

(5) Evaluate and resolve any anomalous results occurring during the course of its ongoing test program.

(b) A schedule for confirmatory tests shall be provided by TVA.

Following the staff's review of the Sequoyah interim system, the staff identified several technical issues requiring additional work prior to the staff's final approval of the Sequoyah permanent hydrogen control system. These issues are:

- consideration of a spectrum of accidents beyond the base case scenario assumed in the interim evaluation;

- design criteria for the permanent hydrogen control system;

- revised containment atmosphere pressure and temperature analyses;

- equipment survivability;

- combustion phenomena:

  °containment mixing
  °local detonations
  °deflagration transition to detonation
  °inadvertent inerting
  °continuous burning

The ICOG, EPRI and the NRC are conducting extensive hydrogen research programs to address these issues. The staff issued a supplement to the

Sequoyah Safety Evaluation Report on December 15, 1982. As stated in the Safety Evaluation Report, the staff concluded that the final ignition system proposed by the licensee for Sequoyah Units 1 and 2 is acceptable subject to the following:

- additional igniters shall be installed in the upper compartment;
- additional testing of the Tayco ignitor in a simulated spray environment is required.

Completion of the testing and evaluation of the test results is currently scheduled for the second quarter of 1984. The staff also identified a number of confirmatory items. These items are:

- local detonations;
- analytical code development;
- equipment survivability for a spectrum of accidents;
- combustion effects at large scale; and
- combustion phenomena including flame acceleration in the upper ice bed.

The confirmatory items will be carried out as part of the NRC-sponsored research programs for the hydrogen control issue.

Subtask 2.2 Grand Gulf Mark III Reviews

2.2.1 Ignition System Interim Review

As in the case of the first ice condenser containment, the applicant for the first Mark III containment to be licensed for operation [Mississippi Power and Light (MP&L) for Grand Gulf] has proposed use of a hydrogen ignition system. The staff's review of this system for the Mark III containment lags the ice condenser review. The NRC recently completed the interim review of the Grand Gulf hydrogen control system. The Grand Gulf hydrogen control system is a hydrogen igniter system similar to that used in the Sequoyah ice condenser containment. The system is designed to provide a controlled burning of hydrogen in the event that large quantitites of hydrogen are generated as a result of a severely degraded core accident. The system was installed by MP&L as a condition that they be allowed to exceed 5% power. The staff's interim review of the Grand Gulf hydrogen igniter system is provided in Supplement No. 3 to NUREG-0831, the Grand Gulf Units 1 and 2 Safety Evaluation Report (July 1982). In that report the staff concluded that the Grand Gulf igniter system was acceptable to reduce the consequences of a severely degraded core accident. This subtask of USI A-48 is complete.

2.2.2 Ignition System - Final Review

The staff is currently conducting a final review of the Grand Gulf hydrogen control system.

Hydrogen-related licensing conditions have been issued by the staff similar to those discussed in Section 2.1.2 for the Sequoyah Ice Condenser containment. This task of USI A-48 will follow the Grand Gulf hydrogen review through to the point that all hydrogen-related licensing conditions are removed.

MP&L, supported by the BWR/Mark III Containment Hydrogen Control Owners Group, has initiated a research and development program to confirm the adequacy of their hydrogen control system. The staff's schedule for issuing a Supplement to the Grand Gulf Safety Evaluation Report addressing the final review of the Grand Gulf igniter system is December 1985. This report will address the same technical issues identified in Section 2.1.2 of this report for the Sequoyah plant.

Task 3. NRC Generic Summary Report, "The Control of Hydrogen in Ice Condenser and Mark III Containments"

The staff's interim evaluation of the lead ice condenser and Mark III hydrogen control systems is documented in the form of Supplements to the Safety Evaluation Reports for the Sequoyah and Grand Gulf plants, respectively. This same mechanism will be used to document the staff's final review of the hydrogen control systems for these plants. The staff's review of these lead plants to remove the current or future hydrogen-related licensing conditions and to implement the proposed hydrogen rule will set the pattern for the staff's review of the hydrogen control systems for the following ice condenser and Mark III facilities. The staff considers that the satisfactory removal of the hydrogen-related licensing conditions will constitute the USI A-48 technical resolution of the hydrogen issue for the ice condenser and Mark III plants.

In consideration of the above, as a part of the USI A-48 program, the staff will issue a generic summary report documenting the technical resolution of the hydrogen issue for the ice condenser and Mark III containments. This document will be prepared at the conclusion of the staff's final review of the lead plant ice condenser and Mark III containment hydrogen control system. The staff's schedule for publication of this summary report is June 1986. This report will generically address the technical issues identified in Section 2.1.2 of this report for ice condenser and Mark III containments. The generic resolution will be patterned after the staff's Sequoyah and Grand Gulf hydrogen control system reviews.

C. Management of Work

The overall responsibility for preparing a generic program to resolve this USI is with the Generic Issues Branch (GIB), Division of Safety Technology (DST), NRR. The lead responsibility for conducting and coordinating most of the tasks of the USI A-48 program lies outside of DST, as discussed below and in Section 4. However, the Task Manager will maintain awareness of those NRC and industry activities required to complete Tasks 1 and 2. The Task Manager will also stay cognizant of all research efforts needed to support degraded core hydrogen control and identify for NRR reviewers and management any results of potential significance to the licensing process.

The three tasks comprising USI A-48 along with the lead responsibilities for each subtask are identified below.

- Near Term Hydrogen Rulemaking

  NTCP/ML Rule: The Division of Licensing, NRR was responsible for the coordination and publication of this rule.

  Mark I and II Inerting Rule: The Division of Risk Analysis, Office of Nuclear Regulatory Research was responsible for the coordination and publication of this rule.

  Ice Condenser/Mark III Hydrogen Control Rule: The Division of Risk Analysis, Office of Nuclear Regulatory Research (RES) is responsible for the coordination and publication of this rule. The Research and Standards Coordination Branch, DST is responsible for coordinating NRR input to RES regarding this rule.

- Plant-Specific Hydrogen Reviews (Sequoyah/Grand Gulf Hydrogen Control Systems)

  The Containment Systems Branch, Division of Systems Integration, NRR is the lead technical branch for technically coordinating the NRR line branch reviews.

  The Division of Licensing provides program management for the plant specific review of the Sequoyah and Grand Gulf plants.

- NRC Generic Ice Condenser/Mark III Hydrogen Control Summary Report

  GIB, DST, NRR is responsible for coordinating and editing the line branch input to the generic Ice Condenser/Mark III summary report.

D. Summary Schedule

A summary of the task completion dates is provided below.

Schedule

Task	Estimated Completion Date
1. Near Term Hydrogen Rulemaking	
1.1 Mark I and II Containments Hydrogen Inerting Rule	12/02/81 - Complete
1.2 Mark III/Ice Condenser Hydrogen Control Rule	09/30/84
1.3 Rule for NTCPs and MLs	01/15/82 - Complete
2. Lead Plant Hydrogen Reviews (Sequoyah and Grand Gulf)	
2.1 Sequoyah Ice Condenser Review	
2.1.1 Interim Ignition System	06/30/81 - Complete
2.1.2 Final Ignition System	09/31/84
2.2 Grand Gulf Mark III Review	
2.2.1 Ignition System Interim Review	07/30/82 - Complete
2.2.2 Ignition System Final Review	12/85
3. Generic Ice Condenser/Mark III Hydrogen Control Summary Report	06/30/86

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION

A. Background

The accident at TMI-2 resulted in a fuel cladding metal-water reaction which involved hydrogen generation well in excess of the design basis amounts specified in 10 CFR 50.44. As a result it became apparent that additional hydrogen control measures would have to be taken for nuclear power plants with small containment buildings.

This topic was first addressed in the Lessons Learned report (NUREG-0578) and subsequently included in the TMI Action Plan, NUREG-0660 (Item II.B.7). As a result, the "Short Term Lessons Learned" from the TMI-2 accident have been implemented at all operating plants and will be implemented at all the other plants before issuance of the operating licenses.

Containment designs for all nuclear plants can generally be placed in three categories on the basis of their capability for accommodating large hydrogen releases and the subsequent burning of hydrogen without loss of containment integrity. These three categories are defined in terms of the relative containment volume as small, intermediate, and large. Additional actions for hydrogen management, pending rulemaking and resolution of this USI, are under consideration by the staff. These actions will reflect the capability of containments in each category.

A final rule has been published requiring inerting of the small Mark I and II containments. In addition, hydrogen control systems are required as a licensing condition for the intermediate volume ice condenser and Mark III containments. A rule has been proposed relating to the requirement for the installation of hydrogen control systems to control the hydrogen associated with a severe degraded core accident for the ice condenser/Mark III containments. No additional hydrogen control requirements or requirements for hydrogen analyses have been imposed for operating plants or plants currently undergoing operation licensing reviews where the plants have large dry containments. This is because of their large containment volumes and relatively large capability to accommodate the generation and combustion of large quantities of hydrogen. However, the NRC and IDCOR are conducting programs outside of this USI A-48 program to address the hydrogen management in PWR dry containments following a severe degraded core accident. These programs will form the basis for the NRC's Severe Accident Rulemaking.

The staff's bases for plant operation and licensing of Mark I, II, III and ice condenser containments is discussed below.

B. Mark I and II Containments

Metal-water reactions in the range of what occurred at TMI-2 (30 to 50%) can produce hydrogen concentrations in Mark I and Mark II air-filled containments that are well within the range of rapid combustion and detonation. Inerting these containments, as currently required by the inerting rule, published in the Federal Register (46FR58484), will eliminate the hydrogen concern relative to combustion and detonation. The peak containment pressures, considering the effect of the noncondensible hydrogen gas from a postulated 75% metal-water reaction and the associated exothermic reaction energy, will approach twice the design pressure for the worst case assumption of an uncooled core immediately following a reactor shutdown. The staff believes Mark I and Mark II containments can withstand, without failure, a slowly applied pressure of at least two times the design pressure. Accordingly, the staff considers USI A-48 resolved for the small Mark I and II containments.

Therefore, the staff concluded that plants with Mark I and II containment can be licensed and operated without jeopardizing the health and safety of the public.

C. Ice Condenser and Mark III Containments

Ice Condenser Containments

Metal-water reactions in the range of 30 to 50% in ice condenser containment plants can produce hydrogen concentrations in the range of 9 to 15%. At these concentrations, containment-wide detonation is not expected.

Moreover, combustion will be inhibited for steam concentrations above 50%, which is also expected in the event of a LOCA. However, operation of the containment spray system and/or the effects of passive heat sinks will condense the steam and produce mixtures that are combustible.

Assuming that there is combustion of hydrogen gas and considering the effect of the noncondensible hydrogen gas and the energy associated with its formation, the estimated amount of metal-water reaction needed to reach the containment design pressure and failure pressure are 15% and 25%, respectively. The design pressures for ice condenser plants are between 12 and 15 psig, and the corresponding pressure capability are estimated to be between 36 and 47 psig.

The Commission has directed that a hydrogen control system be installed in ice condenser containments as a prerequisite for licensing. These systems have been proposed to accommodate cladding metal-water reactions of 75%. Inerting was considered as a mitigative measure for ice condensers. It was concluded that although it might improve the hydrogen management capability, certain important maintenance functions would be restricted and inerting was therefore omitted from further consideration.

The ice condenser plants (Sequoyah, McGuire and D. C. Cook) proposed the use of a distributed ignition system for hydrogen control. These igniter systems provide further assurance that containment integrity would be maintained in the event of a degraded core accident. The NRC has reviewed and approved interim ignition systems for the ice condenser plants. A licensing condition was imposed on the lead ice condenser facility, Sequoyah, requiring that the Commission confirm that a hydrogen control system be installed and shown to perform its intended function in a manner that provides adequate safety margins. This is to be accomplished prior to startup following the first refueling outage. The staff concluded its review of the final hydrogen control systems for Sequoyah to remove this license condition on December 15, 1982 subject to the new conditions described in Subtask 2.1.2.

Therefore the staff concludes that, pending completion of this USI, continued operation and licensing of nuclear plants with ice condenser containments is justified and will not jeopardize the health and safety of the public.

Mark III Containment

If it is assumed that the hydrogen gas produced by a 30 and 50% metal-water reaction does not burn, the resulting containment pressure in a Mark III containment will be between 15 and 20 psig, respectively. In arriving at these containment pressures, the noncondensible hydrogen gas and its associated energy of formation are assumed to enter the containment along with the other LOCA mass and energy sources. If it is assumed that the hydrogen gas does burn, the Mark III containment can accommodate the burning of the hydrogen produced by about 17% metal-water reaction without exceeding its design pressure and about 23% metal-water reaction without exceeding twice the design pressure. The design pressure for the Mark III containments is 15 psig. The pressure capacity for a Mark III containment is estimated at about four times the design pressure (56 psig).

The Commission has directed that hydrogen control systems be installed in Mark III containments as a prerequisite for licensing. As in the case of ice condenser containments, these systems are to be designed to accommodate 75% cladding metal-water reactions. The Mark III plants have proposed the use of a distributed ignition system for hydrogen control. The NRC has reviewed and given interim approval to the ignition system installed in the lead Mark III facility, Grand Gulf. As in the case of the ice condenser plants, the Commission plans to impose licensing conditions for the Grand Gulf facility requiring that the Commission confirm that a hydrogen control system be installed and shown to perform its intended function in a manner that provides adequate safety margins. This is to be accomplished prior to startup following the first refueling outage. The staff anticipates concluding their review of the Grand Gulf hydrogen control system to remove this licensing condition by December 1985. The owners of the other Mark III facilities have committed to the installation of similar hydrogen ignition systems.

Therefore the staff concludes that licensing and operation of nuclear power plants with Mark III containments is justified prior to the ultimate resolution of this issue and will not jeopardize the health and safety of the public.

4. NRC TECHNICAL ORGANIZATIONS INVOLVED

This USI includes three major tasks: (1) Near Term Hydrogen Rulemaking; (2) Plant-Specific Hydrogen Reviews; and (3) Publication of a Mark III/Ice Condenser Generic Summary Report. While the first two tasks play an important role in the resolution of this USI, the direct management of these tasks falls outside of the USI A-48 program. As a result, the identification of branch review responsibilities and manpower requirements related to these two tasks have been excluded from this Task Action Plan. This section is limited to describing the Task 3 work to issue a generic summary report relating to the control of hydrogen in ice condenser and Mark III containments. The line branch manpower projections provided in this section are based on the assumption that a generic Mark III/Ice Condenser summary report can be written based on the Task 2 line branch reviews of the lead ice condenser and Mark III containment. A general description of the Task 3 line branch responsibilities and manpower requirements is provided in Sections 4.A and 4.B below.

A. Generic Issues Branch (GIB)

GIB provides the necessary overall coordination needed to assemble, edit and publish a generic summary report. This report will include a discussion of issues relating to degraded core hydrogen control in the ice condenser and Mark III containments. GIB will maintain awareness of those NRC and industry activities required to complete Tasks 1 and 2. The Task Manager will also stay cognizant of all research efforts needed to support degraded core hydrogen control and identify for NRR reviewers and management any results of potential significance to the licensing process.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
GIB/DST	.8 psy*	.8 psy	.5 psy

B. <u>Other NRC Branches</u>

(1) Containment Systems Branch, NRR

The Containment Systems Branch is responsible for preparing Task 3 input to the generic Ice Condenser/Mark III summary report dealing with the following technical areas:

- containment temperature and pressure conditions;
- reliability and availability of the hydrogen control and monitoring systems;
- combustion phenomena (containment mixing, local detonations, deflagration transition to detonation, inadvertent burning and continuous burning);
- identification of equipment essential to maintain containment integrity;
- technical specifications and operating procedures governing use of the hydrogen control system.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
CSB/DSI	.05 psy	.05 psy	.25 psy

*Assumed 1 professional staff year (psy) = 40 person weeks.

(2) Chemical Engineering Branch, NRR

The Chemical Engineering Branch is responsible for preparing Task 3 input to the generic Ice Condenser/Mark III summary report dealing with the following technical areas:

- local temperature conditions around essential equipment;

- equipment thermal response;

- secondary fire considerations.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0 CEB/DE	.05 psy	.05 psy	.2 psy

(3) Equipment Qualification Branch, NRR

The Equipment Qualification Branch is responsible for preparing Task 3 input to the generic Ice Condenser/Mark III summary report dealing with equipment survivability/qualification.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0 EQB/DE	.05 psy	.05 psy	.2 psy

(4) Reactor Systems Branch, NRR

The Reactor Systems Branch is responsible for preparing Task 3 input to the generic Ice Condenser/Mark III summary report dealing with the following technical areas:

- accident scenarios and the associated time release of hydrogen/steam;

- identification of equipment essential to core cooling and safe shutdown.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
RSB/DSI	.02 psy	.02 psy	.15 psy

(5) Structural and Geotechnical Engineering Branch, NRR

The Structural Engineering Branch is responsible for preparing Task 3 input to the generic Ice Condenser/Mark III summary report dealing with the structural integrity of these containment vessels.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
SGEB/DSI	.02 psy	.02 psy	.15 psy

(6) Containment Systems Research Branch, RES

The Containment Systems Research Branch is responsible for preparing Task 3 input to the generic Ice Condenser/Mark III summary report describing the RES degraded core hydrogen programs that have had a direct bearing on the licensing of the Mark III/Ice Condenser facilities.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
CSRB/RES	.05 psy	.05 psy	.15 psy

(7) Division of Licensing

The Division of Licensing will review plans, reports, and assist in licensing related issues as they arise.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
DL	.05 psy	.05 psy	.05 psy

(8) Procedures and Systems Review Branch, NRR

The Procedures and Systems Review Branch is responsible for review of the revisions to emergency operating procedure guidelines which may be recommended in Task 3.

Manpower Requirements

	FY84	FY85	FY86
Task 3.0			
PSRB/DHFS	.05 psy	.05 psy	.2 psy

Summary of the manpower requirements is shown in Table 1.

Staff manpower required to complete USI A-48 is defined in the resource requirements summary attached to this Task Action Plan.

5. TECHNICAL ASSISTANCE

The following technical assistance contract will be required to resolve this USI.

A. <u>Contractor</u> (Undesignated)

(1) Title: A Review of Generic Ice Condenser/Mark III Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment in Support of USI A-48 Resolution.

(2) NRC Managing Organization: Generic Issues Branch, Division of Safety Technology, Office of Nuclear Reactor Regulation.

(3) Scope: Assist the Generic Issues Branch in writing a generic Mark III/Ice Condenser hydrogen control summary report to document the resolution of the USI A-48 Hydrogen Safety Issue.

(4) Funding: Funding for this technical assistance program will be provided by the Office of Nuclear Reactor Regulation, Division of Safety Technology:

FY1984 - $50,000

6. INTERACTIONS WITH OUTSIDE ORGANIZATIONS

Technical work currently underway or planned by outside organizations plays an important role in the resolution of the USI A-48. Specifically, the staff relied primarily on the ICOG hydrogen program to provide the technical bases for hydrogen control in the ice condenser containment. Similarly, the work of the HCOG will play a dominant role in resolving the hydrogen issue for the Mark III containment.

A number of other groups are conducting hydrogen research programs that may yield results useful in the resolution of the hydrogen issue for the ice condenser and Mark III containments. These groups include IDCOR, EPRI, DOE and foreign organizations. The staff will interact with these organizations to integrate these efforts into the resolution of the hydrogen issue.

The staff's hydrogen review activities associated with the review of lead ice condenser and the Mark III hydrogen control system has been coordinated and will continue to be coordinated with the appropriate Advisory Committee on Reactor Safeguards (ACRS) subcommitee. Significant information will be provided tc the subcommittee as it becomes available and meetings will be scheduled at appropriate times. Peer review will be conducted through ACRS briefings.

7. ASSISTANCE FROM THE OFFICE OF NUCLEAR REGULATORY RESEARCH (RES)

RES has an extensive number of hydrogen combustion and control research programs in place. Many of these research programs have been going on since 1980 and a number of these programs will continue through 1986. The goals of the various programs vary. Portions of these programs provide direct support to the staff's current licensing efforts. A significant fraction of these programs will provide the staff with an improved understanding of the hydrogen combustion phenomenon and an evaluation of alternative containment hydrogen control in support of the NRC's long-term severe accident rule proceeding. A subtask of USI A-48 will be to identify those NRC research programs that are needed to close out the hydrogen-related licensing conditions on the Sequoyah and Grand Gulf facilities. This subtask will be coordinated by the Research and Standards Coordination Branch. A list of the NRC Hydrogen Research Programs is provided below:

- Hydrogen Behavior Program (FIN A-1246) at Sandia National Laboratory
- Combustible Gas in Containment (FIN A-1255) at Sandia National Laboratory
- Equipment Survival Experiments (FIN A-1270) at Sandia National Laboratory

° Hydrogen Combustion Mitigative and Preventative Schemes (FIN A-1336) at Sandia National Laboratory

° RALOC Code Assessment and Applications (FIN A-1205) at Sandia National Laboratory

° Hydrogen Mitigation Study (FIN A-7027) at Los Alamos National Laboratory

° COBRA Applications Study (FIN B-2391) at Pacific Northwest Laboratory

° Development of Methodology for Evaluating Equipment Survivability in an Environment of Burning Hydrogen (FIN A-1306) at Sandia National Laboratory (Complete, 1983)

° Review of Grand Gulf Ignition System (FIN A-1308) at Sandia National Laboratory

° Continue Hydrogen Analysis Review (FIN A-7274) at Los Alamos National Laboratory

8. POTENTIAL PROBLEMS

A. Close coordination is needed between the NRR line review branches that are conducting the lead ice condenser/Mark III hydrogen reviews and the NRR/DST USI A-48 program. Timely completion and adequate documentation of the lead plant ice condenser and Mark III reviews is important to a timely generic resolution of the hydrogen issue for ice condenser and Mark III containments.

B. Early and clear identification of the specific technical problems* that must be solved to resolve the hydrogen issue for the lead Mark III plant and identification of the tests/analyses needed to address these problems.

C. Close cooperation and coordination of internal NRC programs and external generic hydrogen programs (such as IDCOR, ICOG, HCOG, EPRI, and DOE) and timely completion of these research and development programs.*

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*NRC line branches have been requested to address this issue by identifying specific RES programs that must be complete to close out the Sequoyah and Grand Gulf hydrogen review (see Memorandum from S. Hanauer to R. Mattson, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment," dated November 4, 1982). A future revision to this Task Action Plan will cite specific technical problem areas based on the information supplied by the line review branches.

Resource Requirements Summary

	FY-84	FY-85	FY-86
I. Technical Assistance Contract Dollars (in thousands)			
RES: A-1246 (SNL)	2050	2000	
A-1255 (SNL)	88	Complete	
A-7274 (SNL)	250	250	
A-1336 (SNL)	670	790	
A-1270 (SNL)	450	400	
NRR: A-1301 (SNL)	70	30	
A-7274 (LANL)	70	30	
II. Manpower in person-years			
NRR DST GIB	.8	.8	.5
DSI RSB	.02	.02	.15
ICSB			
CSB	.05	.05	.25
DE MEB			
SGEB	.05	.05	.2
CHEB	.05	.05	.2
EQB	.05	.05	.20
DHFS PSRB	.05	.05	.2
DL OR4	.05	.05	.05
RES DAE CSRB	.02	.02	.15

TASK ACTION PLAN
(March 1984)

PRESSURIZED THERMAL SHOCK (TASK A-49)

Lead Organization:	Division of Safety Technology (DST) Generic Issues Branch (GIB)
Task Manager:	Roy Woods, GIB, DST
Lead Supervisor:	Karl Kniel, Chief, GIB, DST
NRR Principal Reviewers:	Brian Sheron and Edward Throm Reactor Systems Branch Division of Systems Integration
	Raymond Klecker and Warren Hazelton Materials Engineering Branch Division of Engineering
	Lambros Lois Core Performance Branch Division of Systems Integration
	Richard E. Johnson and Felix Litton Generic Issues Branch Division of Safety Technology
	Guy S. Vissing Operating Reactors Branch No. 4 Division of Licensing
	James Clifford Procedures and Test Review Branch Division of Human Factors Safety
	Sanford Israel Reliability and Risk Assessment Branch Division of Safety Technology

Office of Nuclear Regulatory Research (RES)	Carl Johnson and Patrick Baranowsky Reactor Risk Branch Division of Risk Analysis Jack E. Strosnider, Milton Vagins, Pryor N. Randall, and Charles Serpan Materials Engineering Branch Division of Engineering Technology
Applicability:	Pressurized Water Reactors
Projected Completion Date:	December 31, 1985

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 of 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 end-of-life. The program that will be conducted to provide firm bases and confirmation 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 prototype plant-specific analyses for three "lead" plants. These analyses will aid NRC in developing acceptance criteria for the plant-specific PTS risk analyses to be required from all PWR licensees before their plants have reached a specified material condition (see Section 2).

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 vs. 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 end-of-life for several vessels. The results of this program are needed to aid NRC development of acceptance criteria for plant-specific analyses. (Plant-specific analyses of systems responses, material properties, risks, and needed corrective actions will be required before a PWR has reached a specified material condition as described in Section 2).

Potential corrective actions are discussed in Section 2.B.(10) 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 with PTS, as described in this TAP, 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 LOCA. 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.

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

This section gives an outline of the program. Further detail of each subtask is given in Section 2.B, including a listing of which organization is responsible for the subtask. Also in Section 2.B is Table 1 showing each organization's manpower and funding requirements for each incomplete task for each fiscal year.

An outline of the integrated program being 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. This approach will result in promulgation of a new PTS rule requiring that plant-specific PTS risk analyses be performed before material properties reach a specified value, and that reasonably practicable neutron flux reduction be made to avoid or delay material properties reaching that value. The plant-specific PTS safety analyses will identify potential corrective actions to reduce the PTS risk. Those potential corrective actions will be reviewed by the NRC staff, and, if necessary, recommendations will be made to the Commission regarding any corrective actions that must be required. Those corrective actions could then be ordered by future, plant-specific Commission action. 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 report periodically to the Advisory Committee on 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). 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 subtasks. The first three tasks, designated by lower case letters, are considered to be part of the short term NRC program. Tasks (a) and (c) have been completed as noted and Task (b) is scheduled to be completed during calendar year 1984. Short Term Program - Review of Industry Responses and Promulgation of a New PTS Rule (Figure 1)

(a) Review of information requested by the August 21, 1981 letter to industry groups and eight selected utilities. This review, including supplemental information obtained from the selected utilities and their owners' groups, provided the bases for a reassessment of the PTS issue that was presented to the Commission on December 9, 1982 (see memorandum for the Commissioners from William J. Dircks, "Pressurized Thermal Shock," dated November 23, 1982, referred to here as SECY-82-465) (Ref. 7). That reassessment presented the following staff conclusions:

(1) The risk from PTS events for reactor vessels with RT_{NDT} values less than the proposed screening criterion (270°F for axial welds, and 300°F for circumferential welds) is acceptable. On the basis of presently available information, no reactor vessel will exceed the screening criterion for the next few years, therefore there is no need to shut down or anneal any operating PWR in the next few years.

(2) Most plants can avoid reaching the screening criterion throughout their service life by timely implementation of flux reduction programs. Such flux reduction programs should be implemented on a time schedule that will avoid foreclosure of this option.

(3) Any plant for which the value of RT_{NDT} is projected to reach the screening criterion before the end of service life, using the conservative method of RT_{NDT} determination described in Section 5 and Appendix E of Enclosure A to SECY-82-465 should submit plant-specific evaluations (of the type described in Section 9 of Enclosure A to SECY-82-465) to determine what, if any, modifications to equipment, systems, and procedures should be required to provide acceptable protection against vessel failure from PTS events for the remainder of plant life. These evaluations should be submitted 3 years before the vessel is projected to reach the screening criterion.

(4) The staff will develop more detailed guidance for these evaluations and acceptance criteria for determining whether plant modifications are needed based on the evaluations.

(5) Some of the Commission's regulations (Appendix G to 10 CFR Part 50, 10 CFR 50.46, and possibly others) may not appropriately reflect current understanding of the state of reactor vessel embrittlement and the potential for vessel failure as a result of PTS. Timely consideration should be given to the need for amendments to the regulations (as discussed in Section 8.6 of Enclosure A to SECY-82-465).

This subtask (a) was completed upon publication of SECY-82-465, November 23, 1982.

(b) Promulgation of a New PTS Rule. Acting on the basis of the above stated staff conclusions [(a)(1) and (a)(3)], the Commission directed the staff to develop a Notice of Proposed Rulemaking that would establish a RT_{NDT} screening criterion, require licensees to submit present and projected values of RT_{NDT}, require early analysis and implementation of such flux reduction programs as are reasonably practicable to avoid reaching the screening criterion, and require plant-specific PTS safety analyses before plants are within 3 calendar years of reaching the screening criterion, including analyses of proposed alternatives to minimize the PTS problem. The Notice of Proposed Rulemaking was submitted for Commission approval on July 15, 1983 (SECY-83-288)(Ref. 8) and was approved on January 13, 1984 (memo from S. Chilk to W. Dircks) (Ref. 9). The Federal Register Notice was published on February 7, 1984. Public comments received by May 7, 1984 will be addressed and a final rule proposed to the Commission. The scheduled objective is to publish the final rule by December 1984.

Guidance and acceptance criteria (staff conclusion (a)(4) above) will be developed by the staff for issuance as Regulatory Guides as described under "Long Term Program" below. These Regulatory Guides will be completed before any licensee would have to begin performing the analyses required to be submitted 3 years before exceeding the RT_{NDT} screening criterion.

Throughout the rule development and Regulatory Guide development described in this TAP, consideration will be given to the need to revise other NRC regulations (conclusion (a)(5) above). If the need to make such changes becomes apparent, they will be proposed.

(c) Consideration of Flux Reduction Options for Lead Plants. Acting on the basis of the above stated staff conclusion (a)(2), the staff met with the licensees of plants for which near term flux reductions greater than a factor of 2 would ensure that the screening criterion would not be exceeded throughout service life, to determine the licensees' plans for such programs.

This accelerated effort was necessary so that flux reduction changes for these plants will be made earlier (when they are most effective) instead of later after the PTS rule is in effect. The delay until the PTS rule is promulgated could be critical to the success of flux reduction efforts for plants in this category.

The staff concluded that reasonably practicable flux reduction options have been implemented or are planned at each of these lead plants, and that the efforts have been effective in delaying the first plants' reaching the screening criterion until the late 1990's. The final report on this task was issued on October 28, 1983 (SECY-83-443) (Ref. 10).

Long Term Program

The eighteen step long term program listed below is directed toward providing additional guidance to the licensees for carrying out provisions of the new PTS rule, and toward providing confirmatory information concerning appropriateness of the RT_{NDT} screening criterion.

(1) Draft revision of the trend curves in Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This revision has been 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).

During the development of the generic PTS screening criterion, the staff selected a conservative trend curve equation that differs from those in Regulatory Guide 1.99. The bases for selection of this method are presented in Appendix E of Enclosure A to SECY-82-465. Since the purpose of the screening criterion established in the proposed PTS rule is to provide a defined and consistent threshold for triggering the submittal of plant-specific PTS analyses, the proposed rule requires licensees to calculate current and projected values of RT_{NDT} for comparison with the screening criterion using the method described in SECY-82-465. Further refinements in the method are appropriate, however, and these have been incorporated in the draft revision to Regulatory Guide 1.99. The guide can then be used in the plant-specific PTS evaluations required once the plant is within 3 years of reaching the screening criterion, and for other licensing reviews (for example, compliance with Appendix G of Part 50). Completion of this Draft Revision to Regulatory Guide 1.99 is not considered essential for completion of USI A-49. We anticipate that certain licensees will present their own data justifying use of a different RT_{NDT} as part of the plant-specific analyses due 3 years before the screening criterion is exceeded, so this plant specific information (if approved) or the present Regulatory Guide will be used if this item is not complete.

(2) The longer range improvement program for procedures and training will continue.

The remaining items in the long term program lead to development of a Regulatory Guide regarding plant-specific analyses required by the new, proposed PTS rule (see item (b) above). Prototype plant-specific analyses are being performed for three lead plants under the program described below. The understanding of how to do such analyses that will be gained during this program will constitute the basis for Regulatory Guide development. The Regulatory Guide will define the scope and methodology required for these analyses and will provide the acceptance criteria for the results. The steps in these prototype analyses and development of a Regulatory Guide are discussed below.

(3) PTS transients to be analyzed have been selected based on systems studies, human factors studies, and probabilistic and risk assessment analyses for three lead plants.

(4) This task has accomplished selection, model improvement and verification of transient codes for use in calculation of the selected transients.

(5) This task has involved 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).

(6) This task has involved 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 included input from near term fracture mechanics experiments performed by the Heavy Section Steel Technology (HSST) group at ORNL.

(7) This task will accomplish calculation of through-wall crack propagation frequency vs. irradiation embrittlement (that is, neutron fluence from the operating history) of the pressure vessels at the three plants for the selected PTS event sequences using the pressure and temperature vs. time histories from item (5) as input to the item (6) codes. These analyses assume pre-existence of a range of crack sizes of various depths.

(8) This task will accomplish integration of results, taking into account results from the above steps.

(9) This task was for Babcock & Wilcox (B&W) plants and involved performance of plant-specific sensitivity studies for the B&W prototype plant to determine changes in through wall crack penetration probabilities due to uncertainties in such parameters as copper content of the weld, initial crack size, lowest temperature of cooldown, etc. This task included development of an understanding regarding 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. This task was terminated by a report submitted by the contractor (ORNL) for the B&W prototype plant.

(10)(11) These tasks are the same as Task 9 above except they are for Combustion Engineering (CE) and Westinghouse (W) plants, respectively (they are not yet completed).

(12) This task will utilize the VISA code to perform generic, non-plant specific sensitivity studies, on a probabilistic basis, involving sensitivity of through wall crack penetration probability to variations in input parameters.

(13) This task will utilize the outputs of Tasks 9, 10, 11, and 12 and will convert the results of those tasks from a through wall crack probability basis to the probability of a LOCA due-to-vessel-failure basis.

(14) This task will utilize the outputs of Task 13 and will convert the results of that task from a LOCA-due-to-vessel-failure probability basis to a core-melt and man-rem (that is, risk) basis.

(15) This task will perform a value-impact study of various corrective actions. Costs of various corrective action options will be calculated, to give the "impact"; value will be determined by Tasks 9 through 14 which will calculate results as described under those respective tasks both with and without the various corrective actions.

(16) This task will utilize results of all previous tasks to formulate the Regulatory Position giving requirements to be met by plant specific analyses and corrective actions before the plant can be operated beyond the screening criterion.

(17) This task involves internal review of the Task 16 result.

(18) This task will result in development and publication of a Regulatory Guide by NRC NRR/Generic Issues Branch with input from other participating Divisions and Offices recommending acceptance criteria for the PTS plant-specific analyses to be required by the new PTS rule. This will include deciding what safety margins are acceptable for the overall analyses including the fracture analysis.

Each of these items constitutes a major subtask in the Regulatory Guide development. Many of the subtasks are planned to proceed concurrently, but some must be sequential. The accompanying Figure 2 is provided to show an overview of the subtasks, including their relationship and schedule. More details of each subtask are given in the discussion below.

B. Technical Content of Major Subtasks

This section gives further details of each subtask that was summarized in Section A above, and notes the organization within the NRC staff primarily responsible for the subtask. Also, tabulated in Table 1 at the end of this Section, for each incomplete task, are the manpower and funding requirements for each participating organization for FY84, FY85 and FY86.

Short Term Program

(a) Review of information requested by the August 21, 1981 letters to industry groups and eight selected utilities. (NRR, with GIB coordinating, was responsible for this subtask.) This item was completed as described in Section 2.A.

This review also resulted in a proposed rule (see item (b) below) which will set up the mechanism and requirements to allow determination of what corrective actions might be required, and when they might be required, through end-of-life for all PWRs.

The original TAP also stated that a near term review of plant PTS and procedures and training was underway. That review has been completed, and the near term changes recommended by the NRC staff have been implemented in all cases.

(b) Promulgation of a New PTS Rule. The staff (NRR with the Generic Issues Branch coordinating) has developed a Notice of Proposed Rulemaking (SECY-83-288, July 15, 1983, "Proposed PTS Rule"). That Notice was approved by the Commission and published as stated in Section 2.A. The proposed rule will: establish an RT_{NDT} screening criterion of 270°F for axial welds, and 300°F for circumferential welds; require that licensees of all operating PWRs submit a determination of the present RT_{NDT} values for their reactor vessels, and the estimated date at which the RT_{NDT} value will exceed the screening criterion; require licensees to implement such flux reduction programs as are reasonably practicable to avoid reaching the screening criterion before expiration of the operating license, and require licensees of operating PWRs for which the RT_{NDT} value is projected to exceed the screening criterion before the expiration of the operating license to submit a plant-specific

PTS safety analysis of the type outlined in Section 9 of Enclosure A to SECY-82-465, 3 years before the screening criterion is exceeded, or 1 year after the effective date of the regulation, whichever is later. For purposes of comparison with the screening criterion, the rule would require calculations of RT_{NDT} values in the manner described in Section 5 of Enclosure A to SECY-82-465. During the longer range PTS program described in this TAP, the staff will develop guidance on the plant-specific analyses to be required and on the acceptance criteria to be used in judging the acceptability of the results.

(c) Consideration of Flux Reduction Options for Lead Plants. (NRR, with the Generic Issues Branch coordinating, is responsible for this task.) This item was completed as described in Section 2.A.

Long Term Program

(1) A revised draft Regulatory Guide 1.99 has been developed. (RES has the lead responsibility for this item.) 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. In addition, the draft revision to Regulatory Guide 1.99, Revision 1 updated the data base and put the trend curves on a statistical basis from which both mean curves and upper bound curves are derived. [As already noted in the previous description of this subtask, the new PTS rule will require licensees to use a conservatively prescribed method of determining RT_{NDT} (Appendix E to Enclosure A of SECY-82-465) for purposes of comparison with the screening criterion. The revision to Regulatory Guide 1.99 developed under this subtask will be used for the plant-specific PTS analysis required to be submitted 3 years before the plant is projected to exceed the screening RT_{NDT}.] Completion of this Draft Revision to Regulatory Guide 1.99 is not considered essential for completion of USI A-49. We anticipate that certain licensees will present their own data justifying use of a different RT_{NDT} as part of the plant-specific analyses due 3 years before the screening criterion is exceeded, so this plant specific information (if approved) or the present Regulatory Guide will be used if this item is not complete.

(2) The ongoing program to improve procedures and operator training regarding prevention and mitigation of PTS events will continue as described in Appendix C of Enclosure A of SECY-82-465, and in Generic Letter 82-33. (NRR/Division of Human Factors Safety has lead responsibility for this subtask.)

The staff has audited training and procedures at eight older PWRs. Emphasis during these audits was on procedural adequacy and the operators' understanding of PTS events and the potentially conflicting requirements of avoiding PTS situations while at the same time assuring adequate cooling to the core. The audit reports are summarized in Appendix C to Enclosure A of SECY-82-465. Generally, it was found that adequate procedures and training exist at the eight plants reviewed although longer range improvements are desirable. The exceptions were noted, and were corrected promptly by the licensees.

The industry is pursuing a major revision of emergency operating procedures as part of TMI Action Plan Item II.C.1. Longer range improvements to the PTS-related procedures and training will result from the integrated, long-range reassessment of procedures in this program which is aimed at adopting "symptom oriented" instead of "event oriented" procedures. That program is also discussed in Appendix C to Enclosure A of SECY-82-465.

The staff believes that it is important to avoid quick and/or frequent changes to procedures with consideration being focused on a particular concern.

The remaining items below all lead to development of guidance and acceptance criteria for plant-specific PTS analyses to be required by the rule 3 years before the RT_{NDT} screening criterion is exceeded. The rule will specify in general what must be considered, but additional guidance is needed so that the licensee knows how thorough the analyses must be, what must be considered, and how the staff will evaluate the acceptability of operation with RT_{NDT} values in excess of the screening criterion, should that be proposed.

The program described below is underway in order to gain the experience needed to generate the needed guidance and acceptance criteria. The program consists of pilot or prototype plant-specific analyses of three plants, one representing each PWR vendor's product. These analyses are being performed for the NRC by various National Laboratory contractors. Major parts of the program are being coordinated by RES for use by NRR (Generic Issues Branch) in developing a Regulatory Guide which will be used in the licensing process. Steps in the analyses leading to publication of a Regulatory Guide are given below.

(3) Determination of Event Sequences to be Considered (RES)

Two major subtasks were involved in selecting the transients to be considered.

(3-a) Development and Quantification of Event Trees for PTS Events Including Review of Control and Safety Systems (RES)

The RES program with ORNL included a study of control and safety system design at the three prototype plants. That program provided details of control and safety system functions and failure modes that may lead to PTS event sequences. Owners of the three prototype plants provided to ORNL control, feedwater, and safety system functions pertinent to PTS event sequences. ORNL defined twelve event sequences in sufficient detail to provide input to Los Alamos National Laboratory (LANL) and Idaho National Engineering Laboratory (INEL) calculations of reactor coolant pressure and temperature vs. time in the downcomer region. The event sequences specified included consideration of multiple failures and multiple operator errors.

(3-b) Human Factors Studies (RES)

An additional ORNL research project addressed required operator actions for the transients being considered and resulted in an assessment of the probability and the effect of human errors on the likelihood of occurrence and severity of overcooling transients.

The above results were used by ORNL, with NRC/RES and NRR review, to determine which PTS events are the major risk contributors, and these events were used in later subtasks (refer to Figure 2).

(4) Transient Model Development and Verification (RES)

Concurrent with Subtask 3, LANL and INEL developed and obtained data to verify the TRAC, RELAP5, and SOLA codes which were used to calculate pressure as a function of time (P(t)) and temperature as a function of position and time ($T(\bar{r},t)$) for the selected PTS events. The three codes needed some model improvement and verification by comparison with data. Code improvements were 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 was obtained from an ongoing EPRI program. Brookhaven National Laboratory (BNL) performed a quality assurance function for plant-specific information on geometry, facility operation during transients, and instrument location and response as they affect the input decks and completed calculations.

(5) Calculation of P(t) and $T(\bar{r},t)$ (RES)

These calculations were performed at LANL and INEL for the transient event sequences identified in Subtask 3 using the improved codes developed and verified in Subtask 4.

(6) Improvements in Methods and Data for Fracture Mechanics Calculations (RES)

Several different types of experiments were used to provide data needed for methods improvement. These tests were part of the HSST program at ORNL. The experiments were designed to improve our understanding of flaw initiation, propagation, and arrest so that fracture mechanics calculations will be more relevant to PTS conditions. The tests included a series designed to further our understanding of the warm prestress phenomenon and the limits of its applicability. Ultimately it is hoped that the methods can be extended beyond the presently accepted linear elastic fracture mechanics methodology to include elastic-plastic fracture mechanics methods. In particular, these programs continue to focus on obtaining theoretical and empirical information on the effects of cladding and the potential benefits of warm prestressing. Consideration was also given to crack propagation into material still on the upper shelf, utilizing analysis procedures developed in USI A-11.

Fracture mechanics codes (OCA-1 at ORNL and the NRC codes) were further developed utilizing 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 were obtained and developed for use as input to these calculations. This subtask validated the fracture mechanics codes by using results of thermal shock experiments.

(7) Vessel Analyses (RES)

Calculations will be performed using the methods and data from Subtask 6 and the P(t) and $T(\bar{r},t)$ results from Subtask 5 for PTS events. These results will be used to calculate frequency of crack growth (initiation) without arrest (that is, through-vessel-wall crack penetration) as a function of effective full power years of operation given assumed occurrence of the PTS events. A range of crack depths are assumed to pre-exist for these calculations.

(8) Integration of Results (RES)

The objective of the above steps is to determine the event sequences that significantly contribute to the probability of a PTS event and to quantitatively estimate the expected frequency of through-wall crack penetration due to such events. This step combines the frequency of the event sequences determined in Subtask 3 with the resulting vessel through-wall crack propagation probability calculated in Subtask 7 using the methods developed in Subtask 6. The result from all of the above is the expected vessel through-wall crack propogation frequency due to each event sequence, and to the extent that the chosen event sequences represent all significant PTS sequences, the sum is the total PTS-related vessel through-wall crack propagation frequency.

(9) Plant-Specific Sensitivity Studies, Benefits/Practicality of Corrective Actions, and Draft Final Report for B&W Plant (RES)

There are many uncertainties in the overall analysis (Subtasks 3 through 8). The effect of those uncertainties on Subtask 8 results will be evaluated. Examples are: specific control and/or safety system features of the prototype plant, plant design features, 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 to credible variations in these parameters (individual or varying in multiple combinations simultaneously) must be assessed before a Regulatory Guide can be written. A series of determiniatic $P(t)$, $T(\bar{r},t)$ and fracture mechanics calculations for several combinations of different input parameters will be performed to determine the effects on outputs of Subtask 8 of variations in the input. The intent is to identify regimes where small variations in input parameters cause qualitatively different event sequences to occur.

Results will be utilized to evaluate the uncertainties in the estimated frequency of through-wall crack propagation from PTS events at the representative B&W prototype plant.

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 partial 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.

(c) Reducing the thermal shock that would result from certain transients by raising the temperature of the emergency core cooling system injection water.

(d) Reducing the probability of PTS events 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.

This task will be concluded when the lead contractor, ORNL, provides a report documenting all methods and results upon completion of work for the B&W prototype plant. (This report will refer to analyses and input data provided by other organizations.)

The programs described will determine the change in vessel through-wall crack penetration frequency due to the various corrective actions. That is, calculations as described in the subtasks above will be performed for the B&W prototype plant with and without the corrective actions. The difference in the results will define the change due to the corrective action.

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.

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 W 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. This information will be factored into development of the Regulatory Guide (Task 18) along with similar information developed by NRC in the Tasks descriped in this TAP.

(10)(11) Same as (9) except for CE plant and W plant, respectively. (RES)

(12) This task will utilize the VISA code to perform generic sensitivity studies, on a probabilistic basis, involving sensitivity of through-wall crack penetration probability to variations in input parameters. Pacific Northwest Laboratory (PNL)

PNL will provide evaluations of the sensitivity or though-wall crack probability to variations and uncertainties in selected input parameters. This study will utilize the VISA code as developed under FIN Number B-2853. The selected parameters are:

- flaw shape (length and depth)
- flaw orientation
- composition of plate and weld metals (copper, nickel, phosphorus contents)
- fracture toughness properties of reactor vessel materials
- shell course orientation
- fluence (azimuthal and radial)
- RT_{NDT} (initial and shift)
- inservice inspection procedures

(13) Results of all of the above tasks are in terms of vessel through-wall crack frequency due to PTS events for each of the three prototype plants. In order to determine PTS risk, it is first necessary to determine the consequences of the through-wall cracks resulting from the calculated PTS events. This is a two step process. This subtask will determine the probability of various size and location LOCAs due to vessel failure resulting from the through-wall cracks. This subtask will also estimate the size and momentum of potential missiles generated by the LOCAs for each of the three prototype plants. These results will be used in subtask (14), the second step in determining PTS risk. (PNL)

(14) Results of subtask (13) are in terms of frequency of various size and location LOCAs. A LOCA is a "consequence" of the PTS event, but it is not the type of consequence most readily used to determine risk. Hence, this task will utilize the output of Task 13 to determine consequences of PTS events in terms of core melt frequency and man-rem exposure to the population. Since frequency of the PTS events was determined in the subtasks up to and including 9, 10 and 11, and since this subtask will determine consequences, the final output of this subtask will be risk for each of the three prototype plants (which is defined as frequency times consequences). (Contract to be let or NRR)

(15) This task will perform a value-impact study involving the various corrective actions. (PNL) The value (that is, safety-benefit) of the corrective actions will be taken from previous tasks 9 through 14

which have calculated risk, both with and without the various corrective actions, as a function of plant type (B&W, CE, or W) and age (that is, RT_{NDT}). The impact (that is, cost) will be calculated directly including a breakdown of total cost into categories including engineering costs, training costs, procurement of equipment, and down time.

The results (ratio of cost in dollars per man-rem or dollars per core melt averted) will be in terms of cost per PTS induced risk averted at each of the threee prototype plants. <u>It will not represent a thorough plant-specific exploration of total plant risk changes due to all causes. Such studies will be the responsibility of the licensee when his plant-specific analyses are submitted.</u>

(16) This task will utilize results of all previous tasks to formulate the Regulatory Position, giving requirements to be met by plant-specific analysis and corrective actions before operation can be considered beyond the screening criterion. (NRR)

The studies described above will be utilized both for the numerical results obtained (total risk, improvements possible, costs) and for what can be learned about desirable and undesirable methodologies to perform the studies.

(17) This task involves internal review of the Task 16 result. (NRR)

(18) Regulatory Guide (NRR)

Utilizing all of the above described information, the NRR staff (NRR/Generic Issues Branch with extensive input from other NRC Divisions and Offices and the ORNL prototype analyses provided by RES) will develop and issue a Regulatory Guide for public and industry comment. After resolution of the public and industry comments, an implementation position will be recommended to the Commission. We anticipate that the implementation position will contain: (1) guidance on plant-specific analyses that must be performed; (2) suggested corrective actions that must be considered; (3) a guide for preparation of a justification of the acceptability of continued operation above the RT_{NDT} screening criterion; and (4) numerical acceptance criteria for the plant's PTS risk.

Table 1. FY84 Technical Support Contracts

Dollar Costs

	FY 84	FY 85
CPB (A-3701, A-3744):	$ 50K	0
Task (c) and post Task (c), plant specific	$200K	0
RSB (A-7272):		
Task 5	0	0
GIB (B-2510):		
Task 12	$ 50K	0
Task 13	$210K	0
Task 15	$150K	0

Manpower Requirements (PSY) Breakdown per Task for all branches with non-zero manpower in Table 3.

	FY 84	FY 85	FY 86
DST/GIB			
Tasks 9 thru 14	1-3/4		
Task 15 thru 17		1-3/4	
Task 18			1/2
DST/RRAB			
Task 9 thru 11	3/16		
Task 12	1/16		
Task 16		1/4	
DSI/RSB			
Task 9 thru 11	1/4		
Task 16		1/4	
DSI/ICSB			
Task 9 thru 11	1/4		
Task 16		1/8	
DSI/CPB			
Task C and post	1/4		

Table 1. FY84 Technical Support Contracts

(Continued)

	FY84	FY85	FY86
Task C plant-specific Task 16			1/8 (Total)
DE/MTEB			
Task 9 thru 11		3/8	1/8
Task 13		3/8	1/4
Task 16			1/8
DHFS/HFEB			
Task 9 thru 11	1/8		
Task 16		1/8	
DHFS/PSRB			
Task 9 thru 11	1/10		
Task 16		1/10	
DL/ORB4			
Admin support for all tasks	1/12	1/6	1/8
DL/All PWR ORBs	0	(1)*	(1/2)*
RES			

General support of effort and management of ORNL effort (tasks 3-8). See Table 3 for total.

*Represents review of RT_{NDT} submittals and flux reduction submittals required by proposed PTS rule. This is considered as casework, not work on this USI.

C. Management of Work

The responsibility for preparing and implementing a program to resolve this USI is with the Generic Issues Branch, DST, NRR. A Task Manager in the Generic Issues Branch will provide overall management of all work identified in this TAP, 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 PTS issues during the conduct of this program.

D. Schedule

Table 2 contains the schedule estimates which have been developed for the completion of the major tasks of this program.

Table 2. Schedule

	Subtask	Estimated Completion Date
(a)	Review of Information Requested	Complete
(b)	Promulgation of PTS Rule	12/31/84
(c)	Consideration of Flux Reduction Options for Lead Plants	Complete
1.	Draft Revision of Reg. Guide 1.99	01/01/86*
2.	Longer Range Procedures and Training Program	**

	Subtask	B&W Plant	CE Plant	W Plant
3.	Plant-Specific Events	Complete	Complete	Complete
4.	Transient Model Development	Complete	Complete	Complete
5.	P(t) and $T(\bar{r},t)$ Calc.	Complete	Complete	Complete
6.	Fracture Mechanics Code Development	Complete	Complete	Complete
7.	Fracture Mechanics Calc.	Complete	Complete	04/01/84
8.	Integration of Results	Complete	04/15/84	06/01/84

	Subtask	Estimated Completion Date
9.	Plant-Specific Sensitivity Studies, Benefits of Corrective Actions, and Draft Final Report, for B&W plant	Complete
10.	Task 9, for CE plant	06/15/84
11.	Task 9, for W plant	08/01/84
12.	Generic Probabilistic Sensitivity Studies	04/30/84
13.	Determine LOCA Probability	09/30/84
14.	Determine Risk	09/30/84
15.	Value Impact Study	10/15/84
16.	Formulate Draft Regulatory Position	01/30/85
17.	Internal Review	02/28/85
18.	Issue Final Technical Resolution Including Regulatory Guide	12/31/85

*As stated in Section 2.B above, this task's completion is not considered essential to completion of this TAP.

**Will be completed according to a plant-specific schedule to be determined as specified in Generic Letter 82-33, from D. Eisenhut, NRC, to all licensees, dated December 27, 1982.

3. BASIS FOR CONTINUED PLANT OPERATION AND LICENSING PENDING COMPLETION

The issue of PTS arises because in PWRs transients and accidents can occur that result in severe overcooling (thermal shock) of the reactor pressure vessel, concurrent with or followed by repressurization. In these PTS events, rapid cooling of the reactor vessel internal surface 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 profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses.

Severe reactor system overcooling events simultaneous with or followed by pressurization of the reactor vessel (PTS events) can result from a variety of causes. These include system transients, some of which are initiated by instrumentation and control systems malfunctions (including stuck open valves in either the primary or secondary system), and postulated accidents such as small break LOCAs, MSLBs, and feedwater line breaks.

The PTS issue is a concern for PWRs only after the reactor vessel has lost some of its fracture toughness due to neutron irradiation. As long as the fracture toughness of the reactor vessel material is relatively high, overcooling events are not expected to cause vessel failure. However, the fracture toughness of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture toughness of the vessel has been reduced sufficiently by neutron irradiation, severe overcooling events could cause propagation of small flaws that might exist near the inner surface. The assumed initial flaw might be enlarged into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

For the reactor pressure vessel to fail and constitute a risk to public health and safety, a number of contributing factors must be present. These factors are (1) a reactor vessel flaw of sufficient size to initiate and propagate; (2) a level of irradiation (fluence) and material properties and composition sufficient to cause significant embrittlement (the exact fluence depends on materials present, such as high copper content causes embrittlement to occur more rapidly); (3) a severe overcooling transient with repressurization; and (4) the crack resulting from the propagation of initial cracks must be of such size and location that the vessel fails.

As a result of the evaluation of the PTS issue, the staff recommended to the Commission (in SECY-82-465) actions to minimize risk due to PTS events in operating reactors. The Commission accepted the staff recommendations and has directed the staff to develop a Notice of Proposed Rulemaking that would establish an RT_{NDT} screening criterion (below which PTS risk is considered acceptable); require licensees to submit present and project values of RT_{NDT};

require early analysis and implementation of such flux reduction programs as are reasonably practicable to avoid reaching the screening criterion; and require plant-specific PTS safety analysis before plants are within 3 calendar years of reaching the screening criterion including analyses of proposed alternatives to minimize the PTS problem. Such a proposed rule has been drafted (SECY-83-288) (Ref. 8) and accepted by the Commission for publication for public comment (January 13, 1984 memo from S. Chilk to W. Dircks) (Ref. 9). It was published on February 7, 1984 in the Federal Register with a request for public comments to be submitted by May 7, 1984. This rule is scheduled to be finalized by the end of calendar year 1984, well before any plant exceeds the RT_{NDT} screening criterion.

All new PWRs, and all currently operating PWRs now have maximum RT_{NDT} values below the RT_{NDT} screening criterion proposed in the rulemaking. On the basis of the studies and analyses presented in SECY-82-465, the staff has concluded that risk due to PTS events from such plants (with RT_{NDT} below the screening limit) is acceptable. Furthermore, plant-specific PTS risk analyses will be required by the rule well before the plant's RT_{NDT} exceeds the screening criterion. These analyses will allow determination of PTS risk, which will allow identification of necessary corrective actions, in time for such actions to be implemented before operation above the screening RT_{NDT} would be considered.

The NRC staff does not believe BWRs have a significant PTS concern, for several reasons. 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 is unlikely. 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.

On the basis of the above considerations, the staff concludes that new reactors can be licensed and operating reactors can continue to be operated before complete resolution of this issue and completion of the proposed rulemaking without undue risk to the health and safety of the public.

4. TECHNICAL ORGANIZATIONS INVOLVED

Manpower requirements discussed in this section are summarized in Table 1.

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

Manpower Requirements: FY 1983 1-3/4 staff-years
FY 1984 1-3/4 staff-years
FY 1985 1-3/4 staff-years
FY 1986 1/2 staff-years

(See Section 2.C) - Overall coordination and direction of the effort will be provided by GIB. In the short term program described in Section 2.B, items (a), (b) and (c), this will include writing and coordinating promulgation of the new PTS rule, and coordinating review of the licensee's flux reduction plans (both in the near term). In the long term (Section 2.B, items 1 through 18), GIB will coordinate, through RES, the ORNL program directed toward the new Regulatory Guide. GIB will coordinate review of the various technical areas covered in the proposed Regulatory Guide with the appropriate technical review branches within NRR.

B. Office of Nuclear Reactor Regulation (Branches Other Than GIB)

In the short term program, the appropriate technical review branches have reviewed parts of the PTS rule (drafted by GIB) within these technical areas, and they have reviewed parts of the flux reduction plans (presented by the licensees by request from GIB, coordinated through the Division of Licensing) within their technical area.

In the long term program, a significant portion of the work leading to the Regulatory Guide will be performed by contractors as discussed throughout this TAP and as summarized in Figure 2. The contracts will be administered by RES, but appropriate NRR personnel in the various technical disciplines involved in the contract work will closely monitor the 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 reports will be reviewed when submitted.

Manpower estimates are given below in the form (x, y, z) where x is the branch's professional staff year estimate for FY83, y for FY84, and z for FY85 including all short and long term program work. It is not anticipated that branches other than GIB will have work ongoing into FY86, as the project will have progressed to the stage of incorporation of public comment, etc. by that time, work which is within GIB.

This TAP will involve: the Materials Engineering Branch (1-1/2, 3/4, 1/2) (that is, 1-1/2 staff-years in FY83, 3/4 staff-year in FY84, and

1/2 staff-year in FY85) for materials properties and fracture mechanics direction and support; the Reliability and Risk Assessment Branch (1/8, 1/4, 1/4) for support in the estimation of probabilities for several PTS events and quantification of the event trees; the Reactor Systems Branch (1/2, 1/4, 1/4) for review of proposed system-oriented corrective actions and transient code development, verification and application; the Instrumentation and Control Systems Branch (1/6, 1/4, 1/8) for review of control system-oriented correction actions; Core Performance Branch (1/2, 1/4, 1/8) for review of fluence studies and review of corrective actions involving fuel removal or rearrangement to reduce flux at the vessel wall; the Division of Human Factors Safety (1/6, 1/8, 1/8) for review of training and procedure-oriented corrective actions and for PTS aspects of the long-range procedure improvements program; and the Division of Licensing (1/2, 1/12, 1/6) for coordination of requests and new requirements to licensees. As noted on Table 1, review of the RT_{NDT} and flux reduction submittals required by the proposed rule will be coordinated by DL in FY85. This will be considered as casework, and not as part of this USI and so is not shown here.

C. Office of Nuclear Regulatory Research (3, 3, 1/2)

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 Technology. See description under Subtask 9 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 $T(\bar{r},t)$ 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. BNL will also audit LANL and INEL thermal-hydraulic code validation and input data for the PTS transients analyzed.

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.

The HSST program at ORNL is also a part of the RES program being applied to the PTS concern.

D. <u>NRR Technical Assistance Contracts</u>

The Reactor Systems Branch of the Division of Systems Integration, NRR has used technical assistance at 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, and to perform peer review of the RES thermal-hydraulic analysis programs.

The Core Performance Branch of the Division of Systems Integration, NRR will utilize technical assistance contract at BNL to collect plant-specific geometry and source data, perform plant-specific fluence reduction review and verification, and make calculational improvements.

The Generic Issues Branch of the Division of Safety Technology, NRR will utilize NRR contract FIN No. B-2510 with PNL to accomplish Task 14 (determine risk) and Task 15 (value-impact), and to provide the probabilistic sensitivity studies (Task 12) and the relative frequencies of vessel failure resulting from through-wall crack penetration due to PTS events (Task 13).

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.

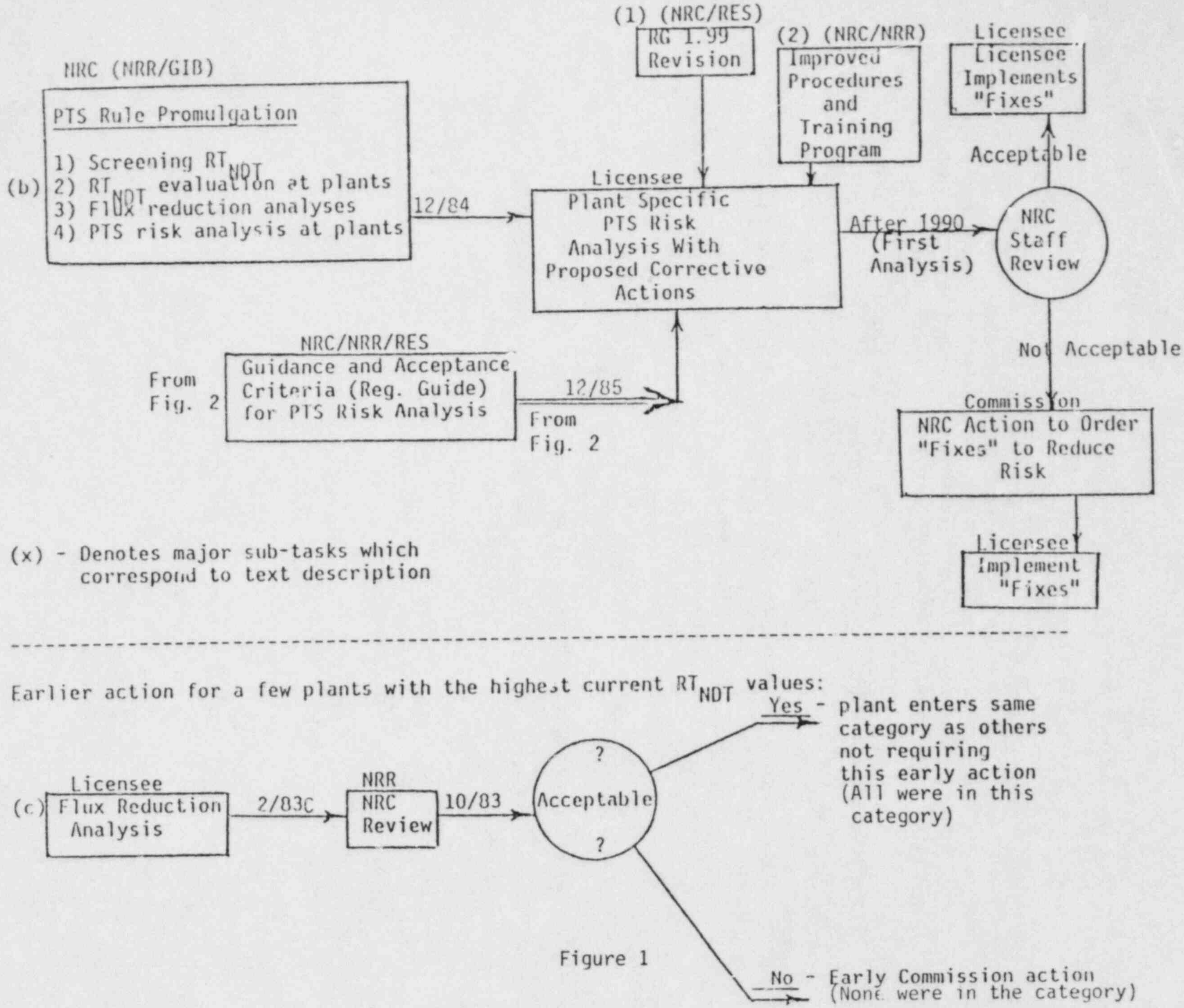
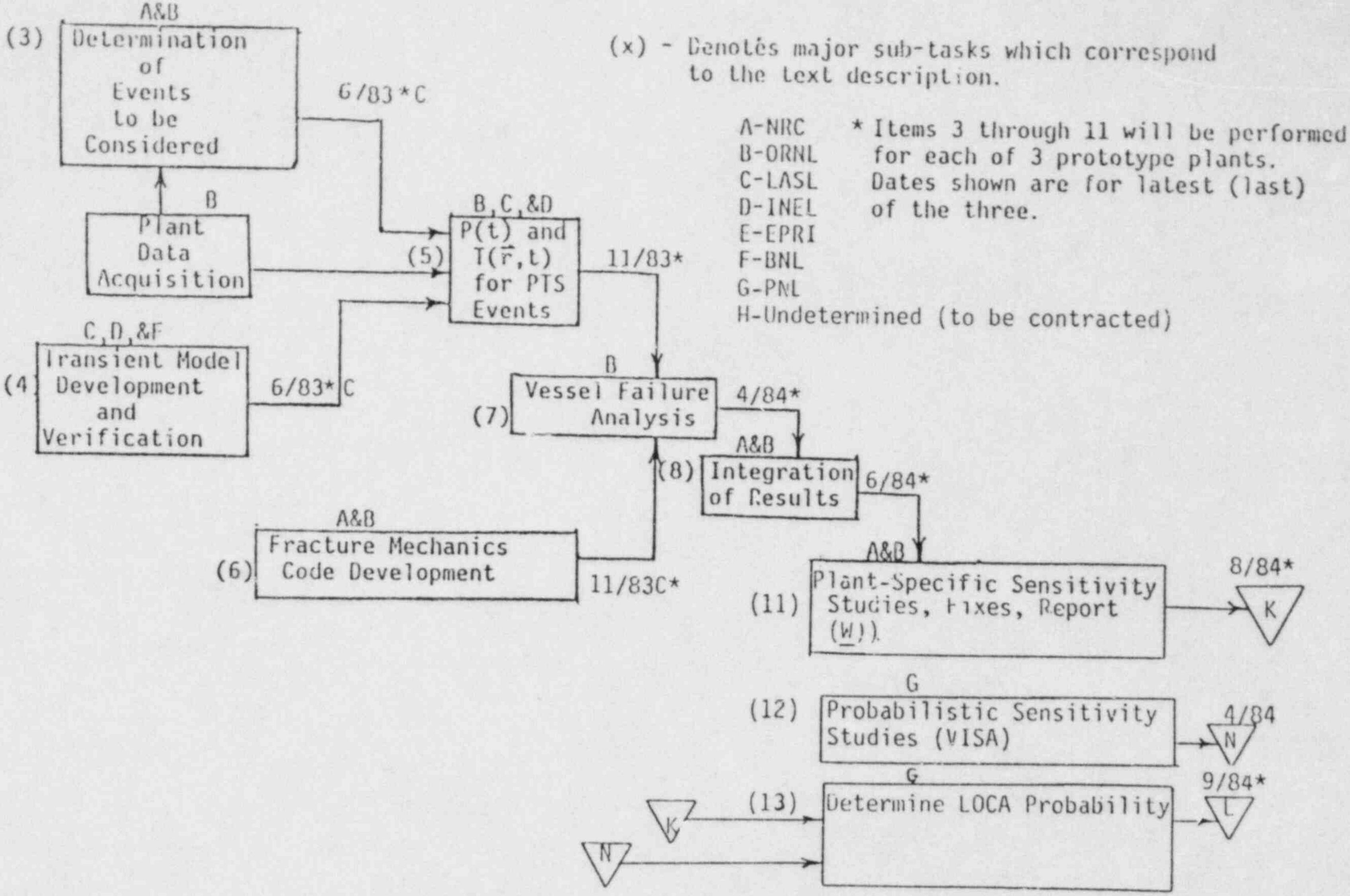

(1) (NRC/RES)
RG 1.99 Revision
(2) (NRC/NRR)
Improved Procedures and Training Program
Licensee
Licensee Implements "Fixes"
NRC (NRR/GIB)
PTS Rule Promulgation
1) Screening RT_{NDT}
2) RT_{NDT} evaluation at plants
3) Flux reduction analyses
4) PTS risk analysis at plants
(b)
12/84
Licensee
Plant Specific PTS Risk Analysis With Proposed Corrective Actions
Acceptable
After 1990 (First Analysis)
NRC Staff Review
NRC/NRR/RES
Guidance and Acceptance Criteria (Reg. Guide) for PTS Risk Analysis
From Fig. 2
12/85
From Fig. 2
Not Acceptable
Commission
NRC Action to Order "Fixes" to Reduce Risk
Licensee
Implement "Fixes"
(x) - Denotes major sub-tasks which correspond to text description
Earlier action for a few plants with the highest current RT_{NDT} values:
Yes - plant enters same category as others not requiring this early action (All were in this category)
Licensee
(c) Flux Reduction Analysis
2/83C
NRR
NRC Review
10/83
? Acceptable ?
No - Early Commission action (None were in the category)

Figure 1



(x) - Denotes major sub-tasks which correspond to the text description.
A-NRC
B-ORNL
C-LASL
D-INEL
E-EPRI
F-BNL
G-PNL
H-Undetermined (to be contracted)
* Items 3 through 11 will be performed for each of 3 prototype plants. Dates shown are for latest (last) of the three.
A&B
(3) Determination of Events to be Considered
6/83*C
B
Plant Data Acquisition
C,D,&F
(4) Transient Model Development and Verification
6/83*C
B,C,&D
(5) P(t) and T(r,t) for PTS Events
11/83*
B
(7) Vessel Failure Analysis
4/84*
A&B
(8) Integration of Results
6/84*
A&B
(6) Fracture Mechanics Code Development
11/83C*
A&B
(11) Plant-Specific Sensitivity Studies, Fixes, Report (W)
8/84*
K
G
(12) Probabilistic Sensitivity Studies (VISA)
4/84
N
G
(13) Determine LOCA Probability
9/84*
L
K
N

Figure 2

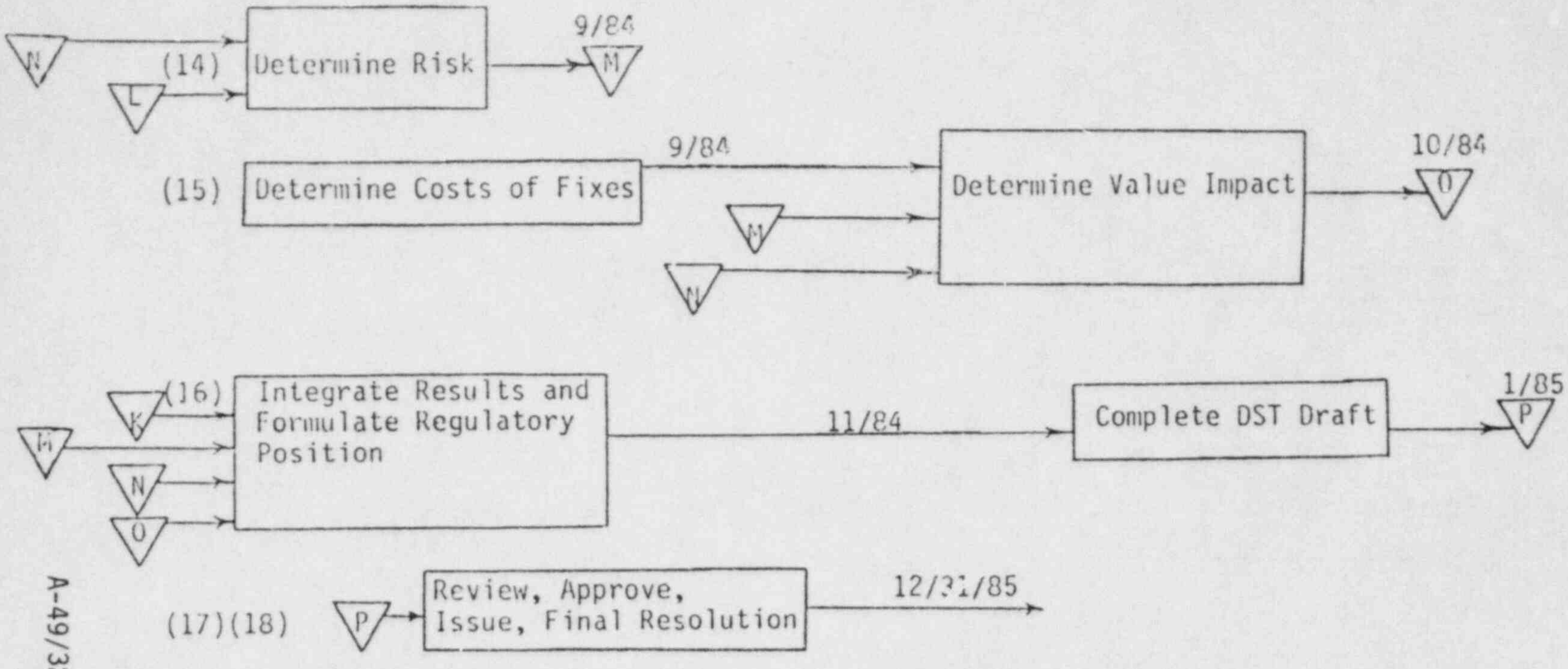


Figure 2. Continued

Table 3. Resource Requirements Summary (PSY)

NOTE: Total dollar value applicable to PTS except where dollar value followed by (.x) in which case .x is fraction needed for PTS.

	Contract Dollars in Thousands*			
	FY 83	FY 84	FY 85	FY 86
RES:				
G-1047 (Purdue)		101	0	
A-4070 (Creare)		124	0	
A-3266 (BNL)		200	0	
A-7306 (LASL, SOLA)		60	50	
A-7315 (LASL, TRAC)		360	0	
A-6047 (INEL, RELAP)		777(.375)	0	
B-0468 (ORNL, Int. PTS)		600	200	
B-0119 ((ORNL, HSST)		6020	6559	
B-8900 (ENSA, St. Integ)		1575(.5)	1600(.5)	
B-6290 (NSRDC, Spec. Shape)		150(.5)	185(.5)	
B-7026 (USNA, Rapid J-R)		70(.5)	70(.5)	
B-8942 (Gundremming)		600	200	
B-2853 (PNL, Visa)		135	0	
B-5988 (HEDL, Dosimetry)		722	597	
B-0415 (ORNL, PVSim)		310	330	
B-6224 (NBS, Dosim)		200	200	
B-2289 (PNL, NDE)		1320(.5)	1400(.5)	
B-2467 (PNL, NDE)		660(.5)	800(.5)	
B-2088 (PNL, Accoustic)		770(.5)	800(.5)	
NRR:				
A-3744 (CPB, casework)	--	200	0	
A-3701 (CPB, dosimetry)	200	50	0	
A-7272 (RSB, TH)	340	0	0	
B-2510 (GIB-PNL)	121	410**	0	

*Entire cost of contract shown for contracts related to PTS. See Table 1 for costs of contract per FY supporting each task, for NRR sponsored work. Such a breakdown is not given for RES sponsored work, as it is not possible to define what fraction of a basic research project (HSST, for example) is in support of any particular task.

**In addition to $180K carryover from FY83.

Table 3. Manpower in Years

(Continued)

		FY 83	FY 84	FY 85	FY 86
DST	GIB	1-3/4	1-3/4	1-3/4	1/2
	SPEB	0	0	0	
	RRAB	1/8	1/4	1/4	
DSI	RSB	1/2	1/4	1/4	
	ICSB	1/6	1/4	1/8	
	CSB				
	ASB				
	PSB				
	CPB	1/2	1/4	1/8	
	AEB				
	ETSB				
	RAB				
DE	MTEB	1-1/2	3/4	1/2	
	SEB				
	GSB				
	HGEB				
	MTEB				
	CHEB				
	EQB				
DHFS	HFEB	1/6	1/8	1/6	
	OLB				
	LQB				
DHFS	PTRB				
	PSRB	1/10	1/10	1/10	
DL*	ORB4	1/2	1/12	1/6	1/8
RES		3	3	1/2	

*See note on Table 1 for casework, not detailed here.

Information Requirements from Industry

Information	Source	Needed by
Plant Design and Operating Information for ORNL Study	Duke Power Co., BG&E, and CP&L	Bulk of information has been provided, Continuing need to review input data and assumptions through completion of study in June 1984.
Generic PRA (PTS Risk) for B&W plants	B&W OG	
Preliminary		05/01/83C
Final		08/01/83C
Flux Reduction Plans	Licensees with FRF>2 Req'd to avoid RT_{sc}	Fall 1983C
Flux Reduction Plans	All other PWR Licensees	When PTS rule promulgated
Evaluation of Plant-Specific RT_{NDT} and future projection	All PWR licensees	When PTS rule promulgated
Plant-Specific PTS PRA	All PWR licensees that will exceed RT_{sc}	Three years before exceed RT_{sc} or 1 year after rule promulgated, whichever is later

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

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

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

4. Letter from R. D. Cheverton, ORNL to M. Vagins, NRC, "Parametric Analysis of Rancho Seco Overcooling Accident," dated March 3, 1981.

5. U.S. Nuclear Regulatory Commission, "Evaluation of Pressurized Thermal Shock," ORNL, USNRC Report NUREG/CR-2083, October 1981.

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

7. SECY-82-465, "Pressurized Thermal Shock (PTS)," dated November 23, 1982.

8. SECY-83-288, "Proposed Pressurized Thermal Shock (PTS) Rule," dated July 15, 1983.

9. Memorandum for William J. Dircks, EDO, from Samuel J. Chilk, Secretary, SECY-83-288, "Proposed Pressurized Thermal Shock (PTS) Rule," dated January 13, 1984.

10. SECY-83-443, "Flux Reduction Programs Related to PTS at Selected (Lead) Plants," dated October 28, 1983.

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16. ABSTRACT (200 words or less)

This document contains Task Action Plans for generic tasks addressing Unresolved Safety Issues (USIs) related to nuclear power plants. Progress on USIs is reported to Congress each year in the NRC Annual Report pursuant to the requirements of Section 210 of the Energy Reorganization Act of 1974, as amended. In addition, the NRR issues NUREG-0606, "Unresolved Safety Issues Summary, Aqua Book" on a quarterly basis; this report provides current schedule information for each USI.

The Task Action Plans in this document include a description of the issue, a description of the NRC staff's approach to resolving the issue, a general discussion of the basis for continued operation and licensing pending resolution of the issue, a discussion of the technical organizations involved in the task, the requirements of manpower and program support funding, interactions with outside organizations and potential problems. This document does not include Task Action Plans for generic tasks addressing USIs for which reports providing the NRC staff resolution of the issue have been published. Those tasks for which reports have been published are identified and the reports are referenced.

The Task Action Plans for active USIs are revised on a yearly basis. This report contains the 1984 revisions to the Task Action Plans.

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