NUREG-0410

RETURN TO
H H Scott

NRC PROGRAM FOR THE RESOLUTION OF GENERIC ISSUES RELATED TO NUCLEAR POWER PLANTS

cludes Plans for the Resolution of "Unresolved Safety Issues"
Pursuant to Section 210 of the Energy Reorganization Act of
1974, as Amended)

Report to Congress
January 1, 1978

U. S. Nuclear Regulatory Commission

7902090065

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NRC PROGRAM FOR THE RESOLUTION OF GENERIC ISSUES RELATED TO NUCLEAR POWER PLANTS

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Manuscript Completed: December 1977
Date Published: January 1978

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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PREFACE

As a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Action of 1974 was amended (PL 95-209) 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 Janaury 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 Bill S.1131 provided the following additional information regarding its 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) future 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."

This report provides a description of the Nuclear Regulatory Commission's Program for the Resolution of Generic Issues Related to Nuclear Power Plants. The NRC program is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. The NRC program does include plans for the resolution of "Unresolved Safety Issues"; however, in addition, it includes generic tasks for the resolution of environmental issues, for the development of improvements in the reactor licensing process and for consideration of less conservative design criteria or operating limitations in areas where over conservatisms may be unnecessarily restrictive or costly.

INTRODUCTION

The primary goals of the NRC in its regulation of nuclear power plants is to assure the health and safety of the public and the protection of the environment. These goals are achieved by means of a system of rules, regulations and regulatory guides coupled with a comprehensive licensing review and inspection process which encompasses all significant safety and environmental factors.

In achieving its goals the NRC is guided by a safety philosophy. This safety philosophy, termed the "defense-in-depth" approach or concept, acknowledges the fact that no single step can be made error-free and relies instead upon multiple lines of defense to provide the necessary level of safety. Thus, the concept is based on the assumption that all defects will not be eliminated and that men will err and materials will fail, despite our best efforts to the contrary.

Quite simply, the defense-in-depth concept requires that three levels of safety be incorporatored into the design of nuclear power plants.

(1) Design and build plants conservatively so that they will operate reliably without failures that could lead to accidents.

(2) Anticipate abnormalities and design back-up systems that will compensate automatically for the failure of essential equipment.

(3) Design multiple back-ups to provide additional margins to protect the public in the event of the occurrence of very unlikely accidents.

To assure that the defense-in-depth concept is fully implemented through conformance to the NRC's rules and regulations and the consideration of NRC's regulatory guidance, the NRC staff conducts thorough and comprehensive safety reviews of all license applications and conducts inspections during plant design, construction, testing and operation. In addition, an independent review of each application for a license is conducted by the Advisory Committee on Reactor Safeguards (ACRS).

This review process is supplemented by public hearings at various stages of the review process where members of the public, the applicant for a license and the NRC staff are afforded an opportunity to present their views to a NRC Atomic Safety and Licensing Board. The results of such hearings are encompassed in an initial decision issued by the Licensing Board. This decision is subject to review by an Atomic Safety and Licensing Appeal Board and by the Commission itself. The final decisions of the Commission may be appealed to an appropriate Federal Court.

The acceptance criteria and procedures for the NRC's safety reviews of applications for nuclear power plant licenses are provided in 224 Standard Review Plans containing over 1,400 pages. These Standard Review Plans provide a detailed statement of the NRC staff's safety requirements and were developed to improve the quality and uniformity of staff reviews and to provide a stabilizing effect on staff requirements.

The implementation of the Standard Review Plans does not, however, relieve the NRC staff from its responsibility to continuously evaluate the safety requirements utilized in its reviews against new information as it becomes available. This responsibility for evaluating the significance of new information is, of course, of immediate importance in continuously assuring the safety of operating reactors.

Information related to the safety of nuclear power plants comes from a variety of sources. Obvious sources of such information are experience from operating reactors, research results, NRC staff and ACRS safety reviews and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe plant operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken, to assure safety e.g., the derating of boiling water reactors as a result of the channel box wear problem in 1975. In other cases, interim measures, such as modifications to operating procedures may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria is not necessary. In all of these cases, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues", because they are related to a particular class or type of nuclear facility rather than a specific plant. These issues also are referred to as "unresolved safety issues." However, as discussed above, such issues are included in the NRC program only after the staff has made an initial assessment for individual plants and has made a determination that the safety significance of the issue does not prohibit continued operation or licensing actions while the longer term generic review is underway.

It is this group of "generic safety issues" or "unresolved safety issues" that are the subject of Section 210 of the Energy Reorganization Act of 1974, as amended and are included in the NRC program described herein.

1.0 Development of the NRC Program for Resolution of Generic Issues

On October 8, 1976, the Commission directed that a number of follow-up tasks be undertaken as a result of the FY 1978 NRC budget development effort. One task identified by the Commission was for the Office of Nuclear Reactor Regulation (NRR) to develop "a program plan for resolution of generic issues and completion of technical projects." The Commission further stated that "this plan should include: task schedules...task priority and manpower requirements (with proportions of staff contract efforts explicitly identified)."

In response to this request, the Office of Nuclear Reactor Regulation established a task force in February 1977 to develop a program plan. The task force recommended a basic framework of policy, organizational structure and procedures for the definition and management of generic technical activities related to nuclear power plants. NRR adopted the task force recommendations and began implementation of the program plan in early April 1977.

The Commission approved the program in the summer of 1977 and established the implementation of the program as an agency-wide objective, with full implementation scheduled for the end of Calendar Year 1977. A copy of the schedule for accomplishment of this agency objective is provided in Appendix A. The status of implementation of the agency objectives are routinely reviewed by NRC's Executive Director for Operations.

The elements of the NRC program are described in Section 2.0 of this report and the status of program implementation, including the issues identified and the projected costs of and schedules for resolution of the highest priority tasks is provided in Section 3.0.

2.0 Elements of the NRC Program

The Office of Nuclear Reactor Regulation has been charged with the responsibility for developing and implementing the NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants. The following basic program elements were developed early in 1977, although they have been refined somewhat during implementation of the program over the past nine months.

1. A set of uniform criteria (Appendix B) for grouping generic technical activities into categories indicative of their priority was developed. The program focuses primary attention on the highest priority activities (Category A activities). This is accomplished through the development of detailed Task Action Plans and scheduling networks, and continuous high level management oversight of these tasks. In addition, as their priority indicates, available resources will be utilized for these generic tasks before being allocated to lower priority generic efforts.

2. The Technical Activities Steering Committee was established to increase high level management involvement and improve management oversight of technical activities. The Steering Committee is chaired by the Deputy Director, ONRR and includes, as members, the four NRR Division Directors. The Committee's functions include assigning

proposed generic tasks to priority categories, assigning lead responsibility to an NRR division for defining and executing each generic task, approving Task Action Plans and regularly reviewing the progress of ongoing tasks. This progress review includes directing such actions as are necessary to recoupe or minimize task schedule slippages when they occur.

3. The concept of Task Managers with clearly identifiable authority and responsibility for the management of individual generic tasks was instituted.

4. Improved planning for NRC staff generic reviews has been provided by the use of detailed Task Action Plans for each generic issue. Task Action Plans include a description of the problem, the staff's approach to its resolution, the technical organizations involved in the review and estimates of the manpower required from each, a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations, an estimate of any funding required for contractor supplied technical assistance, a schedule for completing the task, and a description of any potential problems that could impact the plan. Each plan must be approved by the Technical Activities Steering Committee.

5. An improved scheduling and management control system and increased visibility of NRC generic technical tasks has been provided by the use of a management information book which displays scheduling networks and key information about each task and is updated monthly. Target dates for scheduled milestones are disseminated to all NRR participants in each task through a fully computerized Technical Assignments Control System. Schedules cannot be slipped without the explicit approval of the Chairman of the Technical Activities Steering Committee.

6. Public and industry awareness is provided by public dissemination of the issues, their priority category assignments, the approved Task Action Plans and, when completed, the results of each generic task.

3.0 Implementing the NRC Program

As indicated in Section 1.0 of this report, development of an NRC program related to generic issues was initiated by the Commission in October 1976 and implementation began in April 1977. Implementation of the NRC program has been a major effort that has required the participation of virtually every working and management level in NRR. Decisions regarding the relative priorities of the hundreds of generic issues that have been suggested, although based on agreed upon criteria (Appendix B), in the final analysis are the product of the collective judgments of the individuals making the decisions, in this case, the Technical Activities Steering Committee.

The Steering Committee's judgmental decisions regarding priorities and other matters, such as the assignment of an NRR division with lead responsibility and approval of the Task Action Plan for each task, are based upon the recommendations resulting from an extensive internal review process. This process begins in the NRR line organizations through their development, review, comment and concurrence on proposals for high priority tasks and Task Action Plans. In addition, specific recommendations regarding these proposals are provided by the Steering Committee's Advisory Group following its detailed review. The Advisory Group is made up of five senior technical staff representing each of the NRR divisions and the Director, NRR. The various steps of this review process are shown in Figure 1 [illegible]

Figure 3.1. The evolution of the process of implementing the NRC Program is discussed in the following paragraphs.

Implementation of the program began by making the judgments referred to above regarding the relative priority of hundreds of ongoing, planned or suggested generic efforts. The generic issues that were considered included those from the Advisory Committee on Reactor Safeguard's listing,[1] those listed in NRR's former Technical Safety Activities Report, the 27 issues discussed in NUREG-0138 and NUREG-0153,[2] and a number of other generic issues that have been identified from a variety of sources as described on page vii of this report.

Initially, each of the four NRR divisions described and proposed to the Technical Activities Steering Committee, those generic issues it considered to warrant the highest priority effort (Category A and Category B tasks). Proposals were received for over 130 Category A tasks and over 225 Category B tasks in April and May 1977, respectively. These proposals were reviewed in detail by the Steering Committee's Advisory Group. Following its review, the Advisory Group made recommendations to

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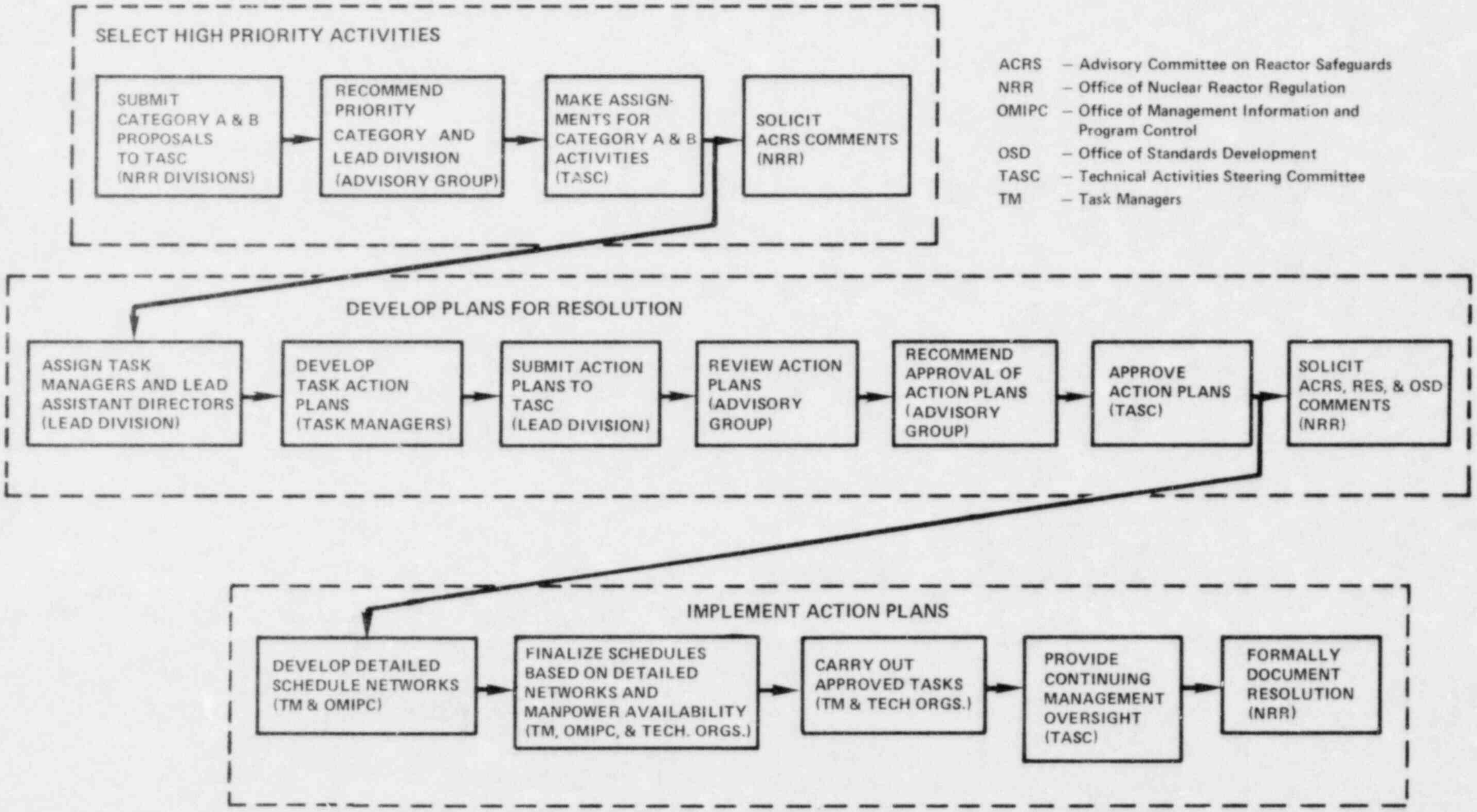
1/ The most recent ACRS status on its generic items (Report No. 6) and a cross index of the ACRS generic items and the NRC staff's generic tasks are provided as Appendix C.

2/ NUREG-0138 and NUREG-0153 published in November and December, 1976 respectively, provided the staff's discussion of 27 technical issues identified by one or more members of the NRR staff as problems whose priority, progress or resolution was, in their opinion, unsatisfactory.

FIGURE 3.1
NRR INTERNAL REVIEW AND APPROVAL PROCESS FOR GENERIC TASKS



ACRS – Advisory Committee on Reactor Safeguards
NRR – Office of Nuclear Reactor Regulation
OMIPC – Office of Management Information and Program Control
OSD – Office of Standards Development
TASC – Technical Activities Steering Committee
TM – Task Managers

the Steering Committee for each task regarding the Priority Category to which its should be assigned and the NRR division that should be assigned lead responsibility. The Steering Committee reviewed the division proposals and the recommendations of its Advisory Group, assigned each task to a Priority Category and designated an NRR division with lead responsibility (Lead Division) for each task. Appendix D provides listings of the issues assigned to Priority Categories A, B, C and D by the Steering Committee from those proposed as Category A and Category B tasks. This Steering Committee action was completed in July 1977.

As indicated in Appendix D, the Steering Committee has approved 41 Category A tasks, 72 Category B tasks, 17 Category C tasks and 3 proposed activities have been assigned to Category D. The disparity in the number of task proposals noted on the preceeding page and the number of approved tasks is the result of Steering Committee actions to combine identical or similar proposals into single tasks and to eliminate proposed tasks that were judged not to be within the scope of the program, e.g., issues requiring a policy decision rather than a generic technical solution were eliminated. Such issues are considered separate from the NRC generic issues program. Proposals for Category C tasks have not yet been considered by the Steering Committee.

Beginning with the Category A tasks, the Lead Division assigned individuals as Task Managers and individuals at the Assistant Director level as Lead Supervisors. Task Managers report directly to the Lead Supervisors for technical and project direction on their task.

Each Task Manager developed a Task Action Plan for his task which was subjected to extensive peer and management review within the line organizations. Following concurrence by the participating organizations, each Task Action Plan was provided to the Steering Committee for its approval. Prior to being considered by the Steering Committee, each Task Action Plan was again reviewed in detail by the Steering Committee's Advisory Group, which suggested modifications and provided its recommendations to the Steering Committee regarding approval.

To date, the Steering Committee has approved 32 of the 41 Task Action Plans for Category A tasks. These approved Task Action Plans are provided in Appendix F as NUREG-0371. Summary information from these Task Action Plans including projections of required manpower and technical assistance funding and preliminary schedules is provided in Appendix E. As indicated in the schedule summary, six Category A tasks are currently scheduled for completion in Fiscal Year 1978.

Subsequent to this comprehensive review and approval process within NRR, the approved Task Action Plans have been provided to other NRC offices for comment and to the Advisory Committee on Reactor Safeguards for its information and use in its interactions with the NRR staff. Pending receipt of these comments, activity on the tasks has begun or in some cases has continued for efforts that were ongoing prior to being incorporated into the program. Comments received from these organizations will be incorporated, as appropriate, as the tasks progress.

In parallel with the initial efforts of defining the issues and developing plans for their resolution, NRR and the NRC's Office of Management Information and Program Control initiated the development of a management information system for generic tasks. The system will include the following information for each Category A task in the form of the "Generic Technical Activities - Status Summary Report" which will be updated monthly.

1. A detailed critical path network containing the major activities necessary to accomplish the approved Category A tasks, the date for accomplishing each activity, and the organizations responsible for performing each activity.
2. A summary of the generic issue to be addressed in the review.
3. A summary of the status of applicable technical assistance contracts.

4. A summary of research activities that may have an impact on the outcome of the review.
5. A listing of potential problems that could prohibit the completion of the review on the projected schedule.
6. A summary of the current status.
7. A summary of actual and projected NRR manpower requirements to complete the review.
8. An explanation of any deviation in the task schedule from the original target schedule that impacted the final completion date of the review.

This information system has been designed to provide the management tools necessary for the Steering Committee, the division management and the Task Managers to monitor and control the NRR program.

Another important element of the program is keeping the public and industry aware of the program and it progresses. Copies of approved Task Action Plans for Category A tasks and listings of the issues assigned to Categories A, B, C and D were placed in the NRC Public Document Room (PDR) in Washington, D. C. in October 1977. Further, steps are underway to assure that the availability of new documents in the PDR is routinely announced and that documents related to the staff's generic tasks are routinely placed in the PDR as the tasks progress.

In addition, copies of the Task Action Plans were made available to members of the public and the nuclear industry in November 1977. Their availability was announced in the FEDERAL REGISTER. This was a one time offer to provide initial distribution of the staff's plans related to generic issues. Additional copies of NUREG-0371 and future updates, including newly approved Task Action Plans and Task Action Plan revisions, will be made available for sale.

The results of each task will be formally documented and placed in the NRC Public Document Room. As currently envisioned, the documentation will, in most cases, be in the form of a published NUREG report.

Since the program was initiated, one Category A issue has been resolved. This isssue was addressed by Task A-6, "Mark I, Short Term Program", which was completed with the issuance of NUREG-0408 in December 1977.

4.0 Future Actions

As indicated in Section 3.0 there are nine Task Action Plans for Category A activities that have not yet been approved by the Technical Activities Steering Committee. In addition, the projected schedules in all of the approved Task Action Plans (summarized in Appendix E) must be subjected to a final review and approval process. The initial schedule projections in the Task Action Plans were, out-of-necessity, developed and approved without considering the combined impact of all the tasks on the participating organizations. Using the detailed critical path schedule networks being developed for incorporation in the management information system, manpower loading projections are being developed for each of the participating NRR review branches. Some schedule adjustments no doubt will be necessary following this review to accommodate the workloads of severely impacted NRR review branches.

Other activities not yet complete, include completing the installation and refinement of the management information system described in Section 3.0. In addition, the activities relating to establishing the appropriate files in the NRC Public Document Room, installing the administrative system to routinely place information related to the generic tasks in these files, and providing for the routine announcement of the availability of this information are not yet complete.

It is anticipated that the activities described above will be completed by the end of February 1978. This is to say that, by the end of February all Category A Task Action Plans will be approved and active on schedules that were developed considering the combined impact of working all of the Category A tasks, the scheduling and management information systems will be in place, regular monitoring of the progress of Cagegory A tasks will have begun by the Technical Activities Steering Committee,[1/] and information regarding the generic tasks will be routinely and systematically placed in the PDR.

Task Action Plans for a number of the Category B generic tasks are currently under development. The Technical Activities Steering Committee will consider Task Action Plans for Category B tasks and proposals for Category C tasks following the assessment of the resource impact of Category A tasks. It is anticipated that a limited number of Category B tasks can be actively pursued in Fiscal Year 1978. This is because the Category A tasks are not evenly distributed across the technical disciplines available in NRR. While some NRR review branches may be severely impacted by the workload imposed by the Category A tasks, other branches may only be moderately impacted and accordingly, will have personnel available for lower priority (i.e., Category B or Category C) tasks. Nonetheless, current estimates indicate that probably less than 15 of the 70 plus Category B tasks can be initiated in Fiscal Year 1978

[1/] Progress monitoring has already begun by the Steering Committee. However, such monitoring has been somewhat sporadic, because of the press of other activities of the Steering Committee, i.e., the review and approval activities.

The end-products of the generic tasks that are encompassed by the NRC program will be generic technical positions. Implementation of these technical positions will be carried out by the NRR line organizations through their normal review, surveillance and licensing activities on individual plants, either as part of the construction permit review, the operating license review, or the continuous evaluation of operating reactors. A system to track the implementation of the generic technical positions that result from this program will be developed to aid the line organizations in monitoring the progress of implementation on the various facilities.

In addition, as required by the new Section 210 of the Energy Reorganization Act of 1974, as amended, reports on the progress of the NRC program and the progress on individual tasks will be provided in the Commission's Annual Report to Congress each year beginning with the 1978 Annual Report.

APPENDIX A

1 OF 2

ORGANIZATION	COMMISSION STAFF
STAFF	M. AYCOCK, D. CRUTCHFIELD
DATE	DECEMBER 15, 1977

GOAL	OBJECTIVE
A - IMPROVE NRC'S EFFECTIVENESS IN ACHIEVING ITS MISSION THEREBY INCREASING THE BASIS FOR PUBLIC CONFIDENCE IN NUCLEAR REGULATION	10 - IMPLEMENT A PROGRAM TO RESOLVE OUTSTANDING GENERIC SAFETY ISSUES

LEAD OFFICE . . . NRR

MILESTONES	FY 1978 O N D J F M A M J J A S · FY 1979 O N D J F M A M J J A S · FY 1980 O N D	LEAD STAFF ASSIGNED
1 ...ESTABLISH FRAMEWORK OF POLICY, ORGANIZATIONAL STRUCTURE AND PROCEDURES FOR RESOLUTION OF ISSUES (NRR)		
A - CREATE TASK FORCE TO DEVELOP RECOMMENDATIONS FOR A PROGRAM	COMP. 2/77	
B - FORMULATE RECOMMENDATIONS ON PRIORITY CATEGORIES AND RESOLUTION FRAMEWORK	COMP. 3/77	
C - TRANSMIT REPORT TO DIRECTOR (NRR)	COMP. 4/77	
D - FINALIZE PROGRAM PLAN (BASED ON NRR MANAGEMENT REVIEW OF TASK FORCE REPORT)	COMP. 4/77	
E - PROVIDE COMMISSION WITH DESCRIPTION AND DISCUSSION OF PROGRAM PLAN (NRR)	COMP. 4/77	
2 ...IMPLEMENT PROGRAM PLAN		
A - ESTABLISH TECHNICAL ACTIVITIES STEERING COMMITTEE (TASC) (NRR)	COMP. 4/77	
B - ESTABLISH FORMAT AND PROCEDURES FOR MAKING PROPOSALS TO THE COMMITTEE	COMP. 4/77	
C - SUBMIT INITIAL PROPOSALS TO THE COMMITTEE FOR:		
1) CATEGORY A ACTIVITIES	COMP. 4/77	
2) CATEGORY B ACTIVITIES	COMP. 5/77	
3) CATEGORY C ACTIVITIES	SCHEDULE TO BE DEVELOPED	
4) CATEGORY D ACTIVITIES	SCHEDULE TO BE DEVELOPED	
D - INITIATE RAINBOW BOOK DEVELOPMENT (NRR/MIPC)	COMP. 7/77	
3 ...PROCEED WITH RESOLUTION OF GENERIC ISSUES UNDER NEW FRAMEWORK (NRR/RES, SD)		
A - APPROVE OR MODIFY DIVISION PROPOSALS FOR:		
1) CATEGORY A ACTIVITIES (TASC)	COMP. 5/77	
2) CATEGORY B ACTIVITIES (TASC)	COMP. 7/77	

STATUS

Reviewed By:

LEGEND

△	CURRENT SCHEDULE
▲	COMPLETED
⟁	SLIPPED
↓	ADDITIONAL INFORMATION

AWO-29

AWO-30

APPENDIX A

2 OF 2

ORGANIZATION COMMISSION STAFF **STAFF** M. AYCOCK, D. CRUTCHFIELD **DATE** DECEMBER 15, 1977

GOAL

A - IMPROVE NRC'S EFFECTIVENESS IN ACHIEVING ITS MISSION THEREBY INCREASING THE BASIS FOR PUBLIC CONFIDENCE IN NUCLEAR REGULATION

OBJECTIVE

10 - IMPLEMENT A PROGRAM TO RESOLVE OUTSTANDING GENERIC SAFETY ISSUES

LEAD OFFICE . . . NRR

FY 1978: O N D J F M A M J J A S — FY 1979: O N D J F M A M J J A S — FY 1980: O N D

MILESTONES	FY 1978 / FY 1979 / FY 1980	LEAD STAFF ASSIGNED
3) CATEGORY C ACTIVITIES	SCHEDULE TO BE DEVELOPED	
4) CATEGORY D ACTIVITIES	SCHEDULE TO BE DEVELOPED	
B - SUBMIT TASC ASSIGNMENTS OF PRIORITIES AND SCHEDULES TO ACRS (NRR)	COMP. 9/77	M. AYCOCK
C - PROVIDE FOR INCORPORATION OF RESULTS INTO THE REGULATORY PROCESS (NRR)		
1) PUBLISH THE STAFF'S EVALUATION OF APPROPRIATE ISSUES IN NUREG FORM	CONTINUING	
2) NRR DIVISION DIRECTORS, RRRC TAKE APPROPRIATE ACTIONS BASED ON RESOLUTION	CONTINUING	
D - INFORM ACRS OF PROGRESS AND RESULTS OF ONGOING EFFORTS ON INDIVIDUAL ISSUES.	CONTINUING	
E - ADVISE COMMISSION THAT PROGRAM IS FULLY OPERATIONAL	⚠ △	M. AYCOCK

STATUS

E - 32 CATEGORY "A" TASK ACTION PLANS APPROVED. DRAFT SCHEDULE STATUS BOOK (RAINBOW BOOK) ISSUED FOR COMMENT. DEVELOPMENT OF CATEGORY "B" TASK ACTION PLANS HAS BEGUN.

Reviewed By:

LEGEND

- △ CURRENT SCHEDULE
- ▲ COMPLETED
- ⚠ SLIPPED
- ↓ ADDITIONAL INFORMATION

APPENDIX B

PRIORITY CATEGORY DEFINITIONS

Category A:

Those generic technical activities judged by the staff to warrant priority attention in terms of manpower and/or funds to attain early resolution. These matters include those the resolution of which could (1) provide a significant increase in assurance of the health and safety of the public, or (2) have a significant impact upon the reactor licensing process.

Category B:

Those generic technical activities judged by the staff to be important in assuring the continued health and safety of the public but for which early resolution is not required or for which the staff perceives a lesser safety, safeguards or environmental significance than Category A matters.

Category C:

Those generic technical activities judged by the staff to have little direct or immediate safety, safeguards or environmental significance, but which could lead to improved staff understanding of particular technical issues or refinements in the licensing process.

Category D:

Those proposed generic technical activities judged by the staff not to warrant the expenditure of manpower or funds because little or no importance to the safety, environmental or safeguards aspects of nuclear reactors or to improving the licensing process can be attributed to the activity.

APPENDIX C

CROSS INDEX OF ACRS GENERIC ITEMS VS NRR GENERIC TASKS

ACRS GENERIC ITEM[1]		NRR GENERIC TASK	
II-1	Turbine Missiles	A-32	Missile Effects
		A-37	Turbine Missiles
II-2	Effective Operation of Containment Sprays in a LOCA	C-10	Effective Operation of Containment Sprays in a LOCA
II-3	Possible Failure of Pressure Vessel Post-LOCA by Thermal Shock	A-11	Reactor Vessel Materials Toughness
II-4	Instruments to Detect (severe) Fuel Failures	Not yet considered by the TASC.[2] Will be considered as a Category C proposal.	
II-5A	Loose Parts Monitoring	B-60	Loose Parts Monitoring Systems
II-5B	Monitoring for Excessive Vibration	A recent clarification of ACRS Issue II-5. TASC will consider including this issue in the program as a Category B task.	
II-6	Common Mode Failures	C-13	Non-Random Failures
II-6A	Scram Systems	A-9	ATWS
II-6B	Alternating Current Systems	A-24	Qualification of Class IE Safety Related Equipment
		A-25	Non-Safety Loads on Class IE Power Sources
		A-35	Adequacy of Offsite Power Systems
		B-56	Diesel Reliability
		B-57	Station Blackout
II-6C	Direct Current Systems	A-24	Same as above
		A-25	Same as above
		A-30	Adequacy of Safety Related DC Power Supplies
		B-57	Same as above

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[1] From November 15, 1977 memo from Bender to Hendrie

[2] Technical Activities Steering Committee

ACRS GENERIC ITEM		NRR GENERIC TASK	
II-7	Behavior of Reactor Fuel Under Abnormal Conditions	B-22	LWR Fuel
II-8	BWR Recirculation Pump Overspeed During LOCA	B-68	Pump Overspeed During a LOCA
II-9	The Advisability of Seismic Scram	D-1	Advisability of Seismic Scram
II-10	ECCS Capability for Future Plants	D-2	ECCS Capability for Future Plants
II A-1	Ice Condenser Containments	B-54	Ice Condenser Containments
II A-2	PWR Pump Overspeed During a LOCA	B-68	Pump Overspeed During a LOCA
II A-3	Steam Generator Tube Leakage	A-3 W A-4 CE A-5 B&W	Steam Generator Tube Integrity
II A-4	ACRS/NRC Periodic 10-year Review of All Power Reactors	Not a generic technical task. Is being treated as a policy matter.	
II B-1	Computer Reactor Protection System	A-19	Digital Computer Protection System
II B-2	Qualification of New Fuel Geometries	B-22	LWR Fuel
II B-3	Behavior of BWR Mark III Containments	B-10	Behavior of BWR Mark III Containment
II B-4	Stress Corrosion Cracking in BWR Piping	Not a generic technical task. Involves implementation of existing staff technical positions on a case-by-case basis.	
II C-1	Locking Out of ECCS Power Operated Valves	B-8	Locking Out of ECCS Power Operated Valves
II C-2	Design Features to Control Sabotage	A-29	Design Features to Control Sabotage
II C-3A	Decontamination of Reactors	A-15	Chemical Decontamination
II C-3B	Decommissioning of Reactors	B-64	Decommissioning of Reactors
II C-4	Vessel Support Structures	A-2	Asymmetric Blowdown Loads on the Reactor Vessel

ACRS GENERIC ITEM		NRR GENERIC TASK	
II C-5	Water Hammer	A-1	Water Hammer
II C-6	Maintenance and Inspection of Plants	B-34	Occupational Radiation Exposure Reduction
II C-7	Behavior of BWR Mark I Containments	A-6 A-7	Mark I Short Term Program Mark I Long Term Program
II D-1	Safety Related Interfaces Between Reactor Island and Balance-of-Plant	Not a generic technical task. Is being treated as a policy matter.	
II D-2	Assurance of Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment
I E-1	Control Rod Drop Accident (BWRs)	D-3	Control Rod Drop Accident (BWRs)
I E-2	Rupture of High Pressure Lines Outside Containment	B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
I E-3	Isolation of Low Pressure From High Pressure Systems	B-63	Isolation of Low Pressure Systems Connected to RCPB

OTHER GENERIC ACRS CONCERNS[3/]

Source	Title	NRR Generic Task	
(Memo Bender to Gossick dtd 6/17/77)	Systems Interaction in Nuclear Power Plants	A-17	Systems Interaction in Nuclear Power Plants
(Memo Bender to Gossick dtd 3/15/77)	Auxiliary System Reliability	Not yet considered by the TASC. Will be considered as a Category C proposal.	

[3/] These issues are not addressed in the periodic ACRS Status Report on Generic Items, but they have been addressed in specific memoranda to the NRC staff.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 15, 1977

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: STATUS OF GENERIC ITEMS RELATING TO LIGHT-WATER REACTORS: REPORT NO. 6

Dear Dr. Hendrie:

The Advisory Committee on Reactor Safeguards has previously reported on the "Status of Generic Items Relating to Light-Water Reactors" in its letters of December 18, 1972, February 13, 1974, March 12, 1975, April 16, 1976 and February 24, 1977. Since the Committee limits its definition of generic items to those cited specifically in its letters pertaining to projects and related matters, the attached listing is not all-inclusive; the Nuclear Regulatory Commission Staff has additional generic items.

Groups I through ID of the attachments are a reiteration of the generic items considered resolved at the time the Committee issued its Report No. 5 on February 24, 1977. Group IE includes those items resolved since February 1977. Following each resolved item is a brief statement of the specific action that resulted in the resolution. Groups II through IID include items previously listed as those for which resolution on a generic basis is still pending. Group IIE includes those added in the present report. The ACRS and the NRC Staff will continue to consider the safety significance of items in Groups II through IIE on a case-by-case basis until generic resolution is reached. Formal actions, such as issuance of Regulations or Regulatory Guides, are anticipated for many of these items.

Owing to questions raised concerning the scope and intent of various generic issues, the Committee has incorporated into the attachments a brief description for all unresolved items cited in this report.

With regard to the status of generic issues, as they apply to each plant, the NRC Staff addresses the status of the pertinent issues in the applicable Safety Evaluation Report. The ACRS identifies those that it believes relevant in its reports on individual projects.

The ACRS has received requests concerning the priorities to be placed on the resolution of outstanding generic issues. Such priorities are shown in Table 1, attached.

"Resolved" as used in the Generic Items reports refers to the following: In some cases an item has been resolved in an administrative sense, recognizing that technical evaluation and satisfactory implementation are yet to be completed. Anticipated Transients Without Scram represents an example of this category. In other instances, the resolution has been accomplished in a narrow or specific sense, recognizing that further steps are desirable, as practical, or that different aspects of the problem require further investigation. Examples are the possibility of improved methods of locating leaks in the primary system, and of improved methods or augmented scope to inservice inspection of reactor pressure vessels.

Sincerely yours,

M. Bender

M. Bender
Chairman

Attachments:
(1) Group I; (2) Group IA; (3) Group IB; (4) Group IC; (5) Group ID; (6) Group IE; (7) Group II; (8) Group IIA; (9) Group IIB; (10) Group IIC; (11) Group IID; (12) Group IIE; and (13) Table 1, Priorities For Resolution of ACRS Generic Items.

GENERIC ITEMS

Group I - Resolved Generic Items

1. Net Positive Suction Head for ECCS Pumps: Covered by Regulatory Guide 1.1.

2. Emergency Power: Covered by Regulatory Guides 1.6, 1.9, and 1.32 and portions of IEEE-308 (1971).

3. Hydrogen Control After a Loss-of-Coolant Accident (LOCA): ACRS concurred in proposed Staff position, covered by NRC Standard Review Plan for Nuclear Power Plants.

4. Instrument Lines Penetrating Containment: Covered by Regulatory Guide 1.11 and Supplement.

5. Strong Motion Seismic Instrumentation: Covered by Regulatory Guide 1.12.

6. Fuel Storage Pool Design Bases: Covered by Regulatory Guide 1.13.

7. Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles: Covered by Regulatory Guide 1.14.

8. Protection Against Industrial Sabotage: Covered by Regulatory Guide 1.17.

9. Vibration Monitoring of Reactor Internals and Primary System: Covered by Regulatory Guide 1.20.

10. Inservice Inspection of Reactor Coolant Pressure Boundary: Covered by ASME Boiler and Pressure Vessel (BPV) Code, Section XI and Regulatory Guide 1.65.

11. Quality Assurance During Design, Construction and Operation: Covered by 10 CFR 50, Appendix B; ASME BPV Code, Section III; ANSI N-45.2-1971, Regulatory Guides 1.28, 1.33, 1.64, 1.70.6 and Proposed Standard ANS-3.2.

12. Inspection of BWR Steam Lines Beyond Isolation Valves: Covered by ASME BPV Code, Section XI.

13. Independent Check of Primary System Stress Analysis: Covered by ASME BPV Code, Section III.

14. Operational Stability of Jet Pumps: Test and operating experience at Dresden 2 and 3 and other jet pump BWRs have satisfied the ACRS concerns.

Group I Continued

15. Pressure Vessel Surveillance of Fluence and NDT Shift: Covered by 10 CFR 50, Appendix A and Appendix H; and ASTM Standard E-185.

16. Nil Ductility Properties of Pressure Vessel Materials: Covered by 10 CFR 50, Appendix A and Appendix G; ASME BPV Code, Section III; "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors," (WASH-1285) by the Advisory Committee on Reactor Safeguards dated January 1974.

17. Operation of Reactor With Less Than All Loops In Service: Covered by ACRS-Regulatory Staff position that manual resetting of several set points on the control room instruments under specific conditions and procedures is acceptable in taking one primary loop out of service. This position is based on the expectation that this mode of operation will be infrequent. Cited in Standard Review Plan Appendix 7-A, Branch Technical Position EICSB 12.

18. Criteria for Preoperational Testing: Covered by Regulatory Guide 1.68.

19. Diesel Fuel Capacity: Covered by ACRS-Regulatory Staff position requiring 7 days fuel (Standard Review Plan 9.5.4).

20. Capability of Biological Shield Withstanding Double-Ended Pipe Break at Safe Ends: Covered by ACRS-Regulatory Staff position cited in several letters that such a failure should have no unacceptable consequences.

21. Operating One Plant While Other(s) is/are Under Construction: Specific requirements have been established by ACRS-Regulatory Staff. Covered in Regulatory Guide 1.17, 1.70 Section 13.6.2; 1.101; ANSI N 18.17 and Standard Review Plan 13.3 Appendix A and 13.6.

22. Seismic Design of Steam Lines: Covered by Regulatory Guide 1.29.

23. Quality Group Classifications for Pressure Retaining Components: Covered by Regulatory Guide 1.26.

24. Ultimate Heat Sink: Covered by Regulatory Guide 1.27.

25. Instrumentation to Detect Stresses in Containment Walls: Covered by Regulatory Guide 1.18.

Group IA - Generic Items Resolved Since December 18, 1972

1. Use of Furnace Sensitized Stainless Steel: Covered by Regulatory Guide 1.44.

2. Primary System Detection and Location of Leaks: Covered by Regulatory Guide 1.45.

3. Protection Against Pipe Whip: Covered by Regulatory Guide 1.46.

4. Anticipated Transients Without Scram: Covered by Regulatory Position Document, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, September 1973.

5. ECCS Capability of Current and Older Plants: Covered by Rulemaking as a general policy decision, although acceptable detailed implementation remains to be developed. Docket RM-50-1, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Reactors," December 28, 1973.

Group IB - Generic Items Resolved Since February 13, 1974

1. Positive Moderator Coefficient: PWRs presently have or expect to have zero or negative coefficients. Where some Technical Specifications allow a slightly positive coefficient, the accident and stability analyses take this into account. Burnable poison provisions have been designed into PWRs to reduce otherwise excessive positive coefficients to allowable values.

2. Fixed Incore Detectors on High Power PWRs: Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

3. Performance of Critical Components (pumps, cables, etc.) in post-LOCA Environment: Qualification requirements of critical components are now covered by Regulatory Guides 1.40, 1.63, 1.73 and 1.89 and IEEE Standards 382-1972, 383-1974, 317-1972, 323-1974.

4. Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppression Containments: On designs prior to GE Mark III containment, reso[illegible]lies in surveillance and testing of vacuum relief valves. [illegible] Mark III containments, an additional requirement is that the design be capable of accommodating a bypass equivalent to one square foot for a given flow condition.

5. Emergency Power for Two or More Reactors at the Same Site: Resolved by issue of Regulatory Guide 1.81.

6. Effluents from Light-Water-Cooled-Nuclear Power Reactors: Resolved by issue of Appendix I to 10 CFR 50.

7. Control Rod Ejection Accident: Resolved for PWRs by Regulatory Guide 1.77.

Group IC - Generic Items Resolved Since March 12, 1975

1. Main Steam Isolation Valve Leakage of BWR's: Covered by Regulatory Guide 1.96.

2. Fuel Densification: Covered by 10 CFR 50 Appendix K plus case-by-case review of vendor fuel models.

3. Rod Sequence Control Systems: Covered by NRC Staff Review and Approval of NEDO-10527 and Presentation to ACRS.

4. Seismic Category I Requirements for Auxiliary Systems: Covered by Regulatory Guides 1.26 and 1.29.

Group ID - Generic Items Resolved Since April 16, 1976

1. Instruments to Detect (limited) Fuel Failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen, June, 1976 resolves issue for limited fuel failures, but not for severe failures (See II-4).

2. "Instrumentation to Follow the Course of an Accident" Regulatory Guide 1.97 Revision 1 resolves ACRS concerns.

3. Pressure in Containment Following LOCA - NRC document, "Containment Subcompartment Analysis" September 1976.

4. Fire Protection. Resolved by Branch Technical Position 9.5.1, and Regulatory Guide 1.120.

Group IE - Generic Items Resolved Since February 24, 1977

1. Control Rod Drop Accident (BWRs): Resolved through NRC review and documentation establishing such an event as not having severe consequences (Memorandum for M. Bender, Chairman ACRS, from Denwood F. Ross, Jr., Assistant Director for Reactor Safety, DSS, dated February 11, 1977.)

2. Rupture of High Pressure Lines Outside Containment: Resolved by positions in Standard Review Plan 3.6.1 and 3.6.2.

3. Isolation of Low Pressure from High Pressure Systems: Resolved by positions in Standard Review Plan 5.4.7.

Group II - Resolution Pending

1. Turbine Missiles: Turbine failures for past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discusses the problems.*

2. Effective Operation of Containment Sprays in a LOCA: Extensive documentation in topical reports. Review and evaluation are required.

3. Possible Failure of Pressure Vessel Post-LOCA By Thermal Shock: Regulatory Guide 1.2 covers current information. Ultimate position as to significance of thermal shock requires input of fracture mechanics data from the Heavy Section Steel Technology Program.

**4. Instruments to detect (severe) fuel failures - NRC document, "Fuel Failure Detection in Operating Reactors," B. L. Siegel and H. H. Hagen. Item ID covers limited failures. More work is required for the severe failure case to establish instrumentation criteria.

#5A. Monitoring for Loose Parts Inside the Reactor Pressure Vessel: State-of-the-Art results appear promising and some equipment has been installed.

#5B. Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel: Neutron Noise Analysis has been successful in detecting vibration of some components, however, additional work may be required concerning systems for detecting vibration in other components within the Reactor Pressure Vessel.

#6. Common Mode Failures: This heading covers a multiplicity of diverse components for which requirements should be established. Due to their diversity the ACRS feels that specific items should be separated into subsets under the general heading of common mode failures;

6A - Reactor Scram Systems
6B - Alternating Current Sources onsite and offsite
6C - Direct Current Systems

The above items are easily identified, other specific items may be added to this listing in the future.

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*Regulatory Guide is in preparation.

**Identified in the Committee's Report of April 16, 1976 as "Instruments to Detect Fuel Failures."

#These are a separation of items included under the same numbers in previous reports.

Group II Continued

7. Behavior of Reactor Fuel Under Abnormal Conditions: This includes: flow blockage; partial melting of fuel assemblies as it affects reactor safety; and transient effects on fuel integrity. The PBF program will address some of these items.

8. BWR Recirculation Pump Overspeed During LOCA: Decision required by ACRS-NRC Staff.

9. The Advisability of Seismic Scram: Further studies required to establish need.

10. Emergency Core Cooling System Capability for Future Plants: Partially resolved by amendments to 10 CFR 50 [50.34(a)(4), 50.34(b)(4), 50.46, and Appendix K]. LOCA evaluation model complete. ACRS feels new cooling approaches should be explored.

II-1 - Turbine Missiles

Turbine failures for the past 16 years have been evaluated and a statistical probability analysis has been completed. An ACRS letter (April 18, 1973) discuses the problem.

Three issues require answers to resolve the turbine missile problem: (1) The first relates to the appropriate failure probability value; based on historical failures the probability is about 10^{-4}. Industry predicts a much lower failure probability based on improvements in materials and design. To date the ACRS has accepted the more conservative value; (2) The second issue is strongly dependent on turbine orientation with respect to critical safety structures. Strike probabilities from high angle missiles are acceptably low for single units and may be acceptable for multi-unit plants, depending on plant layout; however, lower angle missiles with non-optimum (tangential) turbine orientation have unacceptably high strike probabilities; (3) The third issue is one of penetration and damage of structures housed in the containment. The limited experimental data pertaining to penetration of large irregularly shaped missiles are not sufficient to determine structural response to impingement of turbine disc segments. Most missile penetration formulas are not relevant to this case. Some experiments with irregular missiles might resolve this issue, particularly for older plants with non-optimum turbine orientations.

II-2 - Effective Operation Of Containment Sprays In a LOCA

Review and evaluation are required of the variety of experiments which have been conducted on the effectiveness of various containment sprays on the removal and retention of airborne radioactive materials anticipated to be present within containment following a LOCA. Such review should consider adequacy of definition of the physical and chemical forms of the anticipated airborne radionuclides, and quality of evaluative tests of the removal efficiencies of various sprays under the conditions of temperature, pressure, and radiation doses expected to exist under LOCA conditions. A desirable extension might be analyses of the use of sprays containing chemicals (such as NaOH) which have the potential for damaging equipment within containment. Studies using other spray additives, such as hydrazine, have been conducted. If compounds, such as this, have distinct advantages, insofar as minimizing equipment damage in the event of inadvertent actuation, action should be taken to encourage their use.

II-3 - Possible Failure Of Pressure Vessel Post-LOCA By Thermal Shock

Earlier nuclear reactor pressure vessels subjected to fluences of 1-4 x 10^{19} nvt, which are anticipated in the last 20 years of a 40-year life, may suffer severe radiation damage denoted by a pronounced shift in impact transition temperature at the inner surface. There will be a damage gradient which decreases sharply, so that the properties halfway through the wall are essentially those of the as-fabricated material. If a LOCA occurs near end-of-life, the injection of cold water on the region of degraded properties may initiate and propagate a crack because of high local stresses near the surface. Analytic procedures indicate the stresses drop rapidly with distance through the wall so the flaw should not propagate beyond some limiting point. The lack of experimental evidence and the relative width of the error band in the analytic results are such that some experiments are required to validate the analytic model. These are planned under the HSST program.

II-4 - Instruments To Detect (Severe) Fuel Failures

In the event of substantial fuel failure, including the possibility of fuel melt, large amounts of fission products could be rapidly released to the reactor coolant and possibly to the environment. Instrumentation capable of early warning and timely response may avert an incident becoming an accident.

Instrumentation related to such diagnostic purposes for limited fuel failure is being used on most power reactors. (See Item ID-1.) Further work is required to establish criteria for similar instrumentation for severe fuel failures.

II-5A - Monitoring For Loose Parts Inside The Pressure Vessel

Loose parts monitoring can provide early warning of potential mechanical problems or failures within the pressure vessel and throughout the primary coolant circuit. Reactor vendors have developed monitoring systems; however, requirements remain to be established.

II-5B - Monitoring For Excessive Vibration Inside The Reactor Pressure Vessel

Neutron noise analysis can detect vibration within specific components such as the core barrel. The detection of vibration in other reactor pressure vessel components is less well established.

II-6 - Non-Random Multiple Failures (Formerly "Common Mode Failure")

The term "common mode failures" has, in many instances, come to mean multiple failures of identical components exposed to identical or nearly identical conditions or environments, and the use of diversity in components has been proposed or required to avoid such failures. The concern of the ACRS is better expressed by the term "non-random multiple failures," which is intended to include not only the type of "common mode failure" discussed above but other types of multiple failures for which the consequences and probabilities cannot be predicted by application of the single-failure criterion. Examples include the use of the same sensors or components for both control and protection systems (a resolved matter); sequential multiple failures due to a "domino effect," and simultaneous multiple failures due to a single fault. Since designs usually do not knowingly incorporate features susceptible to such failures, techniques and criteria need to be developed to detect and avoid them in all systems important to safety. The following is a partial listing of systems whose common mode failure has been cited by the ACRS as a matter of safety concern:

II-6A - Scram Systems
II-6B - Alternating Current Sources
II-6C - Direct Current Sources

Other items may be added to this listing in the future.

II-7 - Behavior Of Reactor Fuel Under Abnormal Conditions

The behavior of reactor fuel under abnormal conditions is still considered unresolved due to the limited experimental data available. Partial melting of fuel assemblies due to flow blockage might lead to autocatalytic effects leading to more extensive fuel failure, pressure pulses, etc. Similar behavior might occur in the case of reactivity transients. The ACRS encourages analytic modeling but believes appropriate experimental data are necessary. It is anticipated that tests in the Power Burst Facility (PBF) should supply much of the required data.

II-8 - BWR Pump Overspeed During A LOCA

It is possible for a BWR recirculation pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components, with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as the use of decouplers between pump and motor is under study.

II-9 - The Advisability Of Seismic Scram

The ACRS has recommended that studies be made of techniques for seismic scram and of the potential safety advantages and potential disadvantages of prompt reactor scram in the event of strong seismic motion, say more than one-half the safe shutdown earthquake. Various suitable techniques have been identified and exist, but thus far only limited studies have been reported on the pros and cons of seismic scram. The principal potential advantage identified arises from the greatly improved coolability of a core in the unlikely event of a seismically induced LOCA, should scram precede the LOCA by several seconds. A principal reason given in opposition to seismic scram relates to a stated interest in keeping power stations on the line to provide power offsite should a severe earthquake occur.

II-10 - ECCS Capability For Future Plants

The ACRS has placed considerable emphasis on ECCS safety R&D so that the extent of the conservatism in the ECCS licensing requirements could be made more precise. With more experimental data a realistic and quantitative appraisal of ECC systems would lead to valid judgments on the changes in licensing which could be put on a firm basis.

Parallel approaches that seek to improve the reliability of ECC systems, to improve the monitoring of low power peaking, and to improve those fuel assembly designs which lower peaking factors, are encouraged. Further, changes in plant design which improve the reflooding of the reactor core should be sought and evaluated.

R&D efforts on analysis of core blowdown and reflood should be increased and combined with the results of the standard problems and the associated experiments. Improved analytical methods would provide a basis for optimized ECCS.

Group IIA - Resolution Pending - Items Since December 18, 1972

1. Ice Condenser Containments: Additional analyses are required to establish response during a LOCA, and to establish design margins.

2. PWR Pump Overspeed During a LOCA: Problem arises in similar manner to that of BWRs (Item 8 Group II).

3. Steam Generator Tube Leakage: Partially resolved by issuance of Regulatory Guide 1.83 which addresses the concern from a preventative point of view.

4. ACRS/NRC Periodic 10-Year Review of all Power Reactors: A more effective, continuous alternative approach to periodic reviews is being proposed. Pending ACRS review, this item is still considered unresolved.

IIA-1 - Ice Condenser Containments

The ice condenser containments have substantially smaller volume on the assumption that the ice will condense the steam during a LOCA, thus preventing system overpressurization. The rate of condensation is critical in the initial stages of the blowdown and is influenced by interaction of vapor with the ice. If the current analyses prove that the condensation model is suitably conservative, the problem may be resolved.

IIA-2 - PWR Pump Overspeed During a LOCA

It is possible for a PWR primary coolant pump to overspeed if a large break occurs at the appropriate position in specific piping. Conservative estimates indicate substantial overspeed and possible failure of components such as flywheels with the generation of missiles. The problem is being approached analytically and experimentally with scaled pumps. The reliability of such protective measures as electrical braking of the pump motor is under study.

IIA-3 - Steam Generator Tube Leakage

Normally the steam generator is not a critical component during a LOCA-ECCS. However, a special case exists where the steam generator tubes have been degraded due to corrosion, wastage, etc. If the shock loads imposed by the LOCA cause a critical number of tubes to fail, say by a double-ended (guillotine) break, the inflow from the secondary side can cause choking of flow during ECCS, preventing adequate cooling of the core. The critical number of tubes is relatively small. A position, such as one specifying a statistically significant level of nondestructive examination (NDE), might resolve this issue. The purpose of NDE would be to confirm that damage is not excessive; such examinations should minimize the possibility of catastrophic failure of a significant number of tubes.

IIA-4 - Periodic (10-Year) Review Of All Power Reactors

In its report of June 14, 1966, the ACRS recommended that periodic comprehensive reviews be conducted of operating licensed power reactors by the NRC Staff. These reviews would be preceded by a comprehensive report by the operator which evaluated the past experience and the safety of future operation of the plant.

The NRC Staff has maintained a continuing review of the safety of operating plants. In particular, as generic matters of potential safety significance arise, the appropriate operating reactors are asked to assess the relevance of the matter to each particular reactor. This is a necessary but different aspect of the continuing surveillance and review of the safety of operating reactors than was envisaged by the ACRS in its recommendation of June 1966.

The Committee continues to believe both approaches are desirable and awaits the development of a program of periodic comprehensive reviews.

Group IIB - Resolution Pending - Items Added Since February 13, 1974

*1. Computer Reactor Protection System: Systems should be qualified for reliability, particularly through in situ tests and under various environmental conditions, prior to use in reactor system.

2. Qualification of new fuel geometries: The 16x16 and 17x17 PWR, and 8x8 BWR fuels should undergo testing to meet Item 2 in Group IC and Item 7 in Group II.

3. Behavior of BWR Mark III Containments: Various aspects, including vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads and flow bypass should be resolved. This is an extension of Item 3 in Group ID.

4. Stress Corrosion Cracking in BWR Piping: Several failures have occurred in operating BWRs. The ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution and extensive programs are underway by industry, ERDA, and NRC.

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* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-1 - Computer Reactor Protection Systems*

The proposed systems would contain some types of components and subsystems not previously used for reactor protection. It is necessary that the required system reliability, both during normal operation and under postulated abnormal conditions, be established through an appropriate combination of tests and analyses. While the issue originated with the B&W Hybrid concept it is equally applicable to the proposed CE and W computer reactor protection systems.

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* Identified in the ACRS Report of April 16, 1976 as "Hybrid Reactor Protection System."

IIB-2 - Qualification Of New Fuel Geometries

New fuels proposed for both BWRs and PWRs include the 8x8 (BWR), and 16x16 and 17x17 PWR fuels. The Committee recognizes that these fuels are intended to operate at power densities lower than earlier fuel designs. However, testing programs are considered necessary to establish their densification behavior (IC-2) as well as their behavior under abnormal conditions (II-7). Appropriate experimental programs should be developed dealing with flow blockage, behavior of fuel after partial melting, and fuel response under transient conditions. It is anticipated that the solution of this item will include a synthesis of Power Burst Facility data, experiments on earlier fuel types, behavior of fuel in commercial reactors, and confirmatory experiments on these fuel designs.

IIB-3 - Behavior Of BWR Mark III Containment

The BWR Mark III Containment differs in many respects from the Mark I and II designs. Various aspects such as vent clearing, vent/coolant interaction, pool swell, pool stratification, pressure loads, and flow bypass must be evaluated and approved; ongoing experimental tests should develop much of the necessary data to confirm the conservatism in design.

IIB-4 - Stress Corrosion Cracking In BWR Piping

Several failures have occurred in operating BWRs. An ACRS letter of February 8, 1975, discusses possible actions that should lead to generic resolution, and extensive programs are underway by Industry, ERDA and NRC.

The austenitic stainless steels are commonly used as piping material in many of the smaller BWR lines. A combination of weld sensitization, residual stresses, superposed loads, and oxygen equal to or greater than 0.2 ppm in the BWR coolant can lead to cracking, initiating on the inner surface and propagating through the wall. In most cases there will be a leak well before pipe failure so there is adequate warning; however, one can postulate a LOCA caused by a guillotine break with minimal prior warning. Current efforts are to minimize stress corrosion by using other materials.

Group IIC - Resolution Pending - Items Added Since March 12, 1975

1. Locking Out of ECCS Power Operated Valves: The Committee suggests that further attention be given to procedures involving locking out electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS.

2. Design Features to Control Sabotage: Attention should be given to aspects of design that could improve plant security.

*3A. Decontamination of Reactors: As experience is gained in reactor decontamination it should be factored into future plants to optimize control of radioactivity levels.

*3B. Decommissioning of Reactors: Specific plans should be developed, including definitive codes and standards to cover the ultimate decommissioning of plants.

4. Vessel Support Structures: Questions that have arisen concerning the loads on pressure vessel support structures due to certain postulated loss-of-coolant accidents should be resolved.

5. Water Hammer: Several cases of water slugging or water hammer have occurred in both PWRs and BWRs. Corrective measures should be taken to minimize such events.

6. Maintenance and Inspection of Plants: Provisions should be included in the design of future plants which anticipate the maintenance, inspection and operational needs of the plant throughout its service life.

7. Behavior of BWR Mark I Containments: Various aspects relevant to the BWR Mark I Containment should be resolved. Included are such items as relief valve restraint, control of local dynamic loads in the torus, vent clearing and establishment of torus water temperature limits during a LOCA. This is an extension of Item 3 in Group ID.

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*This is a further separation of the issues identified as IIC-3 in previous reports.

IIC-1 - Locking Out Of ECCS Power-Operated Valves

The physical locking out of electrical sources to specific motor-operated valves required in the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; e.g., closed rather than open, or open rather than closed. While such an event has a finite probability another probability exists that the valves might be adversely positioned due to operator error.

The ACRS believes the matter should be studied using a systems approach, and considering such items as: (1) the evaluation of the probability of a spurious signal; (2) time required to reactivate the valve operator; (3) status of signal lights when the circuit breaker is open; (4) the possibility of locking out in an improper position due to a faulty indicator; (5) other designs with improved reliability without lock-out; (6) the advantages and disadvantages of corrective action by an alert operator in case of incorrect positioning vis-a-vis a system with power locked out.

IIC-2 - Design Features To Control Sabotage

Considerable attention has been devoted to control of industrial sabotage of nuclear power plants, particularly with regard to control of unauthorized access, and potential modes of sabotage by individuals or groups external to the operating organization. The ACRS believes that deliberate attention should be given to aspects of design that could improve plant security. With the emphasis being placed on standardized plant designs, it becomes especially important to introduce design measures that could protect against industrial sabotage, or mitigate the consequences thereof.

IIC-3A - Decontamination Of Reactors

The Committee believes that well developed plans, confirmed by appropriate experiments when necessary, should be available for the decontamination of primary reactor systems. At this time the information on full scale decontamination is limited. Examples of potential problems include such items as handling of decontamination solutions, potential hideout of radioactive products, enhanced corrosion and crud formation following decontamination, and the possible incompatibility of the different alloys in the pressure boundary with the decontamination solutions.

IIC-3B - Decommissioning Of Reactors

Experience is limited with regard to decommissioning operations, and particularly with rules for dismantling and for mothballing. Definitive plans and standards should be developed covering such items as adequacy of action, problems in restitution of site, mutual responsibility of State and Federal Government, etc.

IIC-4 - Vessel Support Structures

A possible consequence of the instantaneous double-ended pipe break postulated to occur in certain large pipes of PWRs is the asymmetric loading of the reactor pressure vessel support structures. The magnitude and effects of such loads on the pressure vessel should be determined to establish if such loads adversely affect the predicted course of a LOCA. If analysis indicates that the results are unacceptable, appropriate corrective action should be taken. A potential effect is pressure vessel movement due to blowdown jet forces at the location of the rupture, transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

IIC-5 - Water Hammer

Several instances of water slugging or water hammer have occurred in both BWRs and PWRs due to causes such as the trapping of water between two valves. This slug of water is accelerated by steam or water once the valves are opened. The stored energy is sufficient to damage piping, bend or break pipe restraints, and damage support structures. Water hammer may occur due to flow instabilities in steam generators in conjunction with water flowing into the feedwater inlets, resulting in comparable damage.

Corrective measures should be taken to minimize such occurrences after completion of analytic and experimental studies directed to an understanding of the causes.

IIC-6 - Maintenance And Inspection Of Plants

Experience with older plants has verified that appropriate modifications in piping layout, with respect to walls and structures, type of insulation used, and weld joint design, to cite some obvious items, lead to improved maintenance, more reliable inservice inspections, and a better meeting of the operational needs of the plant throughout its service life, including decontamination and eventual decommissioning. An additional benefit is the reduction in personnel exposures in plants, making them more amenable to maintenance and inspection. Appropriate changes should be considered in future designs to meet these criteria.

IIC-7 - Behavior Of BWR Mark I Containments

Recent tests on the BWR Mark I Containment design revealed phenomena not anticipated on the basis of earlier tests where pressure loads were imposed by insertion of air. Specific problems somewhat comparable to those under review for the Mark III Containment, include relief valve discharge, pipe restraints in the torus, local dynamic loads on the torus, vent clearing, and influence of torus temperature on the LOCA.

Ongoing experiments are expected to develop the necessary data to confirm the adequacy of the existing design or to permit necessary modifications.

Group IID - Resolution Pending - Items added since April 16, 1976

1. Safety related interfaces between reactor island and balance-of-plant: The nuclear steam suppliers and some architect-engineers have submitted standardized plant designs. The Committee wishes to be sure that adequate attention is devoted to the interface between the reactor island and balance-of-plant to minimize problems during design and construction. The development and use of interdisciplinary system analyses is an aspect of this problem.

2. Assurance of continuous long-term capability of hermetic seals on instrumentation and electrical equipment: The integrity of seals during post-accident conditions may be critical in controlling such an accident. The Committee believes appropriate test and maintenance procedures should be developed to assure long-term reliability.

IID-1 - Safety Related Interfaces Between Reactor Island And Balance-Of-Plant

Questions have been raised concerning both standardized balance-of-plant and nuclear steam supply systems on the one hand and custom-designed site-related structures and components on the other hand. The depth of detail required at the stage of Preliminary Design Approval may not be adequate for construction approval. Procedures for instituting quality assurance programs covering design, procurement, construction, and startup with emphasis on timely and appropriate interdisciplinary system analyses to assure functional compatibility across the interfaces as well as for other systems, are necessary to assure functional compatibility for the postulated design basis accident conditions.

IID-2 - Assurance Of Continuous Long-Term Capability Of Hermetic Seals On Instrumentation And Electrical Equipment

Certain classes of instrumentation incorporate hermetic seals. When safety related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The damage processes may fall within Item IB-3, "Performance of Critical Components in Post-LOCA Environment"; however, a special case requiring evaluation has to do with personnel errors in the maintenance of such equipment since such errors could lead to the loss of effective hermetic seals.

Group IIE - Resolution Pending - Items Added Since February 24, 1977

1. Soil-Structure Interactions: Several matters related to soil-structure interaction and the appropriate seismic response spectrum for use at foundation levels of nuclear plants are under review and reevaluation.

IIE-1 - Soil Structure Interactions

Ongoing studies by the NRC and the industry are reviewing and re-evaluating matters related to soil-structure interaction and to the appropriate seismic response spectrum to be used at the foundation level of a nuclear power plant. These reviews may lead to a modification of current criteria used in the seismic design of foundation structures.

Table 1 - Priorities For Resolution Of ACRS Generic Items

GENERIC ITEMS	RELEVANT TO PWR	BWR	PRIORITY FOR RESOLUTION 1/ ACRS	NRC
II-1 Turbine Missiles	X	X	A	A-37*
II-2 Containment Sprays	X		B	C-10
II-3 Pressure Vessel Failure By Thermal Shock	X		A	A-11
II-4 Instruments to Detect (Severe) Fuel Failure	X	X	C	-
II-5A Excessive Vibration	X	X	B	-
II-5B Loose Parts Monitoring	X	X	B	B-60

1/ The ACRS has adopted and uses the categorizations developed by the NRC Staff and paraphrased below:

A. Those items judged to warrant priority attention in terms of manpower and/or funds to attain early resolution. These items include those, the resolution of which could (1) provide a significant increase in assurance of the health and safety of the public, or (2) have a significant impact upon the reactor licensing process.

B. Those items judged to be important in assuring the health and safety of the public but for which early resolution is not required or which have a lesser safety significance than Category A matters.

(Continued on Page 2)

*The numerals have no significance regarding the priority for resolution but are included to identify the NRC program plans related to each item.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
II-6	Non-Random Multiple Failures	X	X	A	C-13
6A	Reactor Scram Systems	X	X	A	A-9
6B	Alternating Current Sources Onsite & Offsite	X	X	A	A-35 B-56 B-57
6C	Direct Current Systems	X	X	A	A-30
II-7	Behavior of Reactor Fuels Under Abnormal Conditions	X	X	A	B-22
II-8	BWR Recirculation Pump Overspeed During LOCA		X	B	B-68
II-9	Seismic Scram	X	X	C	D-1
II-10	ECCS Capability for Future Plants	X	X	A	D-2

Footnote 1 Continued:

C. Those items judged to have little direct or immediate safety significance but which could lead to improved understanding of particular technical issues or refinements in the licensing process.

D. Those items judged not to warrant the expenditure of manpower or funds because little or no importance to safety or improvements to the licensing process can be attributed to the item.

GENERIC ITEMS		RELEVANT TO PWR	RELEVANT TO BWR	PRIORITY FOR RESOLUTION 1/ ACRS	PRIORITY FOR RESOLUTION 1/ NRC
IIA-1	Ice Condenser Containments	X		B	B-54
IIA-2	PWR Pump Overspeed During a LOCA	X		B	B-68
IIA-3	Steam Generator Tube Leakage	X		A	A-3 A-4 A-5
IIA-4	ACRS/NRC Periodic 10-Year Review	X	X	C	Policy*
IIB-1	Computer Reactor Protection System	X		B	A-19
IIB-2	Qualification of New Fuel Geometry	X	X	C	B-22
IIB-3	BWR Mark III Containments		X	B	A-39 B-10
IIB-4	Stress Corrosion Cracking in BWR Piping		X	B	Policy*
IIC-1	Locking Out of ECCS Power Operated Valves	X	X	B	B-8

*The resolution of this item is to be effected through administrative means rather than by a specific technical activity.

GENERIC ITEMS		RELEVANT TO		PRIORITY FOR RESOLUTION 1/	
		PWR	BWR	ACRS	NRC
IIC-2	Design Features to Control Sabotage	X	X	A	A-29
IIC-3A	Decontamination	X	X	B	A-15
IIC-3B	Decommissioning	X	X	B	B-64
IIC-4	Vessel Support Structures	X		B	A-2
IIC-5	Water Hammer	X	X	A	A-1
IIC-6	Maintenance and Inspection	X	X	B	B-34
IIC-7	BWR Mark I Containments		X	A	A-6 A-7 A-39
IID-1	Interfaces	X	X	A	Policy A-17
IID-2	Capability of Hermetic Seals	X	X	C	C-1
IIE-1	Soil-structure Interaction	X	X	C	A-40 A-41

APPENDIX D

CATEGORY A TECHNICAL ACTIVITIES

Task No.	Title
A-1	Water Hammer
A-2	Asymmetric Blowdown Loads on the Reactor Vessel
A-3	Westinghouse Steam Generator Tube Integrity
A-4	Combustion Engineering Steam Generator Tube Integrity
A-5	Babcock & Wilcox Steam Generator Tube Integrity
A-6	Mark I Short Term Program
A-7	Mark I Long Term Program
A-8	Mark II Program
A-9	ATWS
A-10	BWR Nozzle Cracking
A-11	Reactor Vessel Materials Toughness
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports
A-13	Snubbers
A-14	Flaw Detection
A-15	Decontamination
A-16	Steam Effects on BWR Core Spray Distribution
A-17	Systems Interaction in Nuclear Power Plants
A-18	Pipe Rupture Design Criteria
A-19	Digital Computer Protection Systems
*A-20	Impacts of Coal Fuel Cycle
A-21	Main Steam Line Break Inside Containment
A-22	PWR Main Steam Line Break - Core and Primary Coolant Boundary Response (MSLB Outside Containment)
A-23	Containment Leak Testing
A-24	Qualification of Class IE Safety-Related Equipment
A-25	Nonsafety Loads on Class IE Power Sources
A-26	Reactor Vessel Pressure Transient Protection (Overpressure)
A-27	Reload Application Guide
A-28	Increase in Spent Fuel Storage Capacity
A-29	Design Features to Control Sabotage
A-30	Adequacy of Safety-Related DC Power Supplies
A-31	RHR Shutdown Requirements
A-32	Evaluation of Overall Effects of Missiles
A-33	NEPA Reviews of Accident Risks
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents
A-35	Adequacy of Offsite Power Systems
A-36	Control of Heavy Loads Near Spent Fuel

Task No.	Title
A-37	Turbine Missiles
A-38	Tornado Missiles
A-39	Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments
A-40	Seismic Design Criteria - Short Term Program
A-41	Seismic Design Criteria - Long Term Program

CATEGORY B TECHNICAL ACTIVITIES

Task No.	Title
B-1	Environmental Technical Specifications
B-2	Forecasting Electricity Demand By State in the United States on an Annual Basis
B-3	Event Categorization
B-4	ECCS Reliability
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containment
B-6	Loads, Load Combinations, Stress Limits
B-7	Secondary Accident Consequence Modeling
B-8	Locking Out of ECCS' Power Operated Valves
B-9	Electrical Cable Penetrations of Containment
B-10	Behavior of BWR Mark III Containment
B-11	Subcompartment Standard Problems
B-12	Containment Cooling Requirements (Non-LOCA)
B-13	Marviken Test Data Evaluations
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA
B-15	CONTEMPT Computer Code Maintenance
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment
B-17	Criteria for Safety-Related Operator Actions
B-18	Vortex Suppression Requirements for Containment Sumps
B-19	Thermal-Hydraulic Stability
B-20	Standard Problem Analysis
B-21	Core Physics
B-22	LWR Fuel
B-23	LMFBR Fuel
B-24	Seismic Qualification of Electrical and Mechanical Components
B-25	Piping Benchmark Problems
B-26	Containment Penetrations
B-27	Implementation and Use of Subsection NF
B-28	Radionuclide/Sediment Transport Program
B-29	Effectiveness of Ultimate Heat Sinks
B-30	Design Basis Floods and Probability
B-31	Dam Failure Model
B-32	Ice Effects on Safety-Related Water Supplies
B-33	Dose Assessment Methodology
B-34	Occupational Radiation Exposure Reduction

Task No.	Title
B-35	Confirmation of Appendix I Models for "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors"
B-36	Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems
B-37	Chemical Discharges to Receiving Waters
B-38	Reconnaissance Level Investigations
B-39	Transmission Lines
B-40	Effects of Power Plant Entrainment on Plankton
B-41	Impacts on Fisheries
B-42	Socioeconomic Environmental Impacts
B-43	Value of Aerial Photographs for Site Evaluation
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants
B-45	Need for Power - Energy Conservation
B-46	Costs of Alternatives in Environmental Design
B-47	Inservice Inspection Criteria for Supports and Bolting of Class 1, 2, 3 and MC Components
B-48	BWR CRD Mechanical Failure (Collet Housing)
B-49	Inservice Inspection Criteria for Containment
B-50	Requirements for Post-OBE Inspection
B-51	Assessment of Inelastic Analysis Techniques
B-52	Fuel Assembly Seismic and LOCA Responses
B-53	Load Break Safety Switch
B-54	Ice Condenser Containments
B-55	Improved Reliability of Target-Rock Safety-Relief Valves
B-56	Diesel Reliability
B-57	Station Blackout
B-58	Passive Mechanical Failures
B-59	Review of (N-1) Loop Operation in BWRs and PWRs
B-60	Loose Parts Monitoring Systems
B-61	Allowable ECCS Equipment Outage Periods
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, etc.
B-63	Isolation of Low Pressure Systems Connected to RCPB
B-64	Decommissioning of Reactors
B-65	Iodine Spiking
B-66	Control Room Infiltration Measurements
B-67	Effluent and Process Monitoring Instrumentation
B-68	Pump Overspeed During a LOCA
B-69	ECCS Leakage Excontainment

Task No.	Title
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps
B-71	Incident Management
B-72	Development of Models for Assessing Risk of Health Effects and Life Shortening from Uranium and Coal Fuel Cycles

CATEGORY C TECHNICAL ACTIVITIES

Task No.	Title
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electric Equipment
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure
C-3	Insulation Usage Within Containment
C-4	Statistical Methods for ECCS Analysis
C-5	Decay Heat Update
C-6	LOCA Heat Sources
C-7	PWR System Piping
C-8	Main Steam Line Leakage Control System
C-9	RHR Heat Exchanger Tube Failures
C-10	Effective Operation of Containment Sprays in a LOCA
C-11	Assessment of Failure and Reliability of Pumps and Valves
C-12	Primary System Vibration Assessment
C-13	Nonrandom Failures
C-14	Storm Surge Model for Coastal Sites
C-15	NUREG Report for Liquid Tank Failure Analysis
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

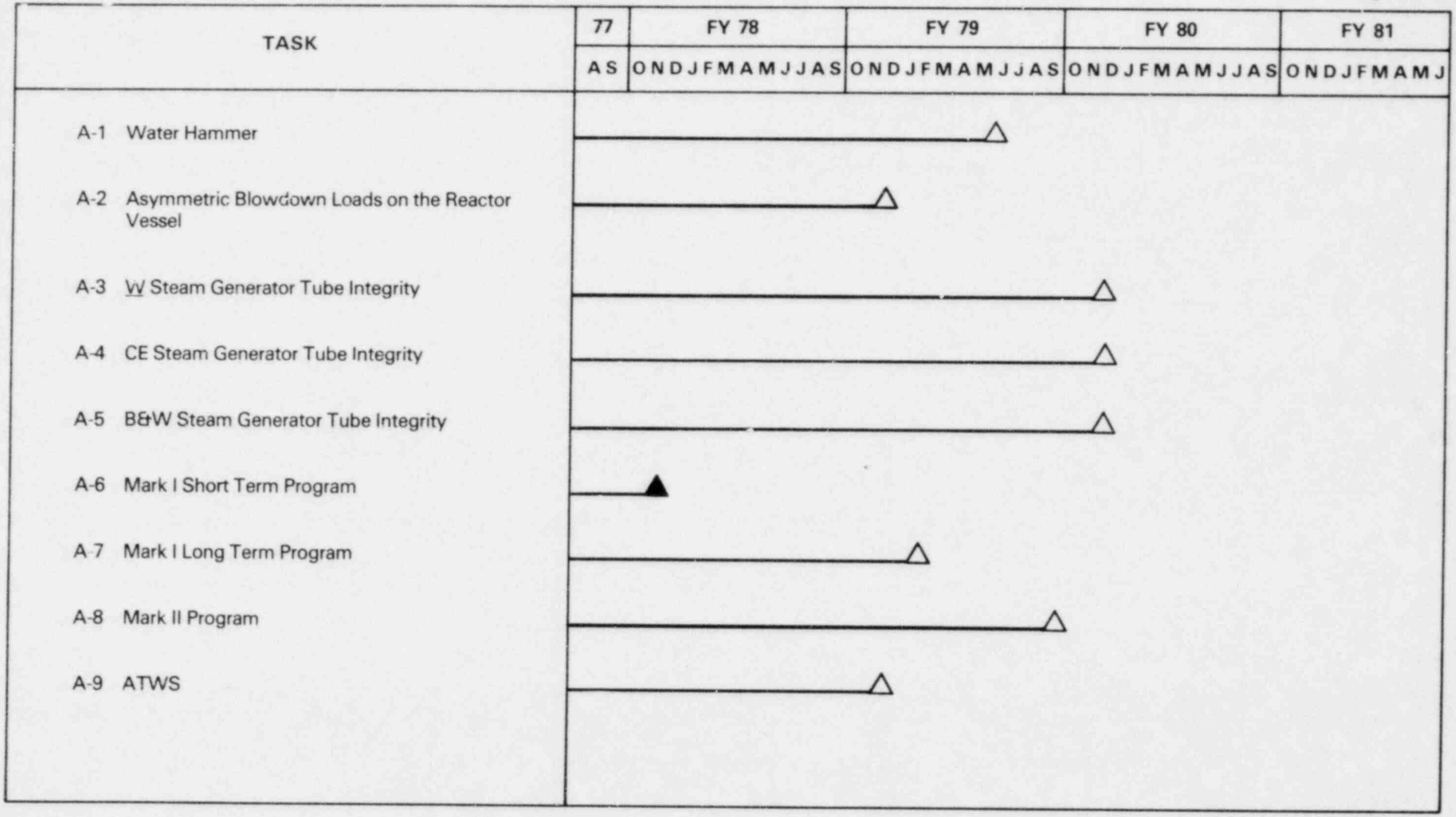
CATEGORY D TECHNICAL ACTIVITIES

Task No.	Title
D-1	Advisability of a Seismic Scram
D-2	Emergency Core Cooling System Capability for Future Plants
D-3	Control Rod Drop Accident (BWRs)

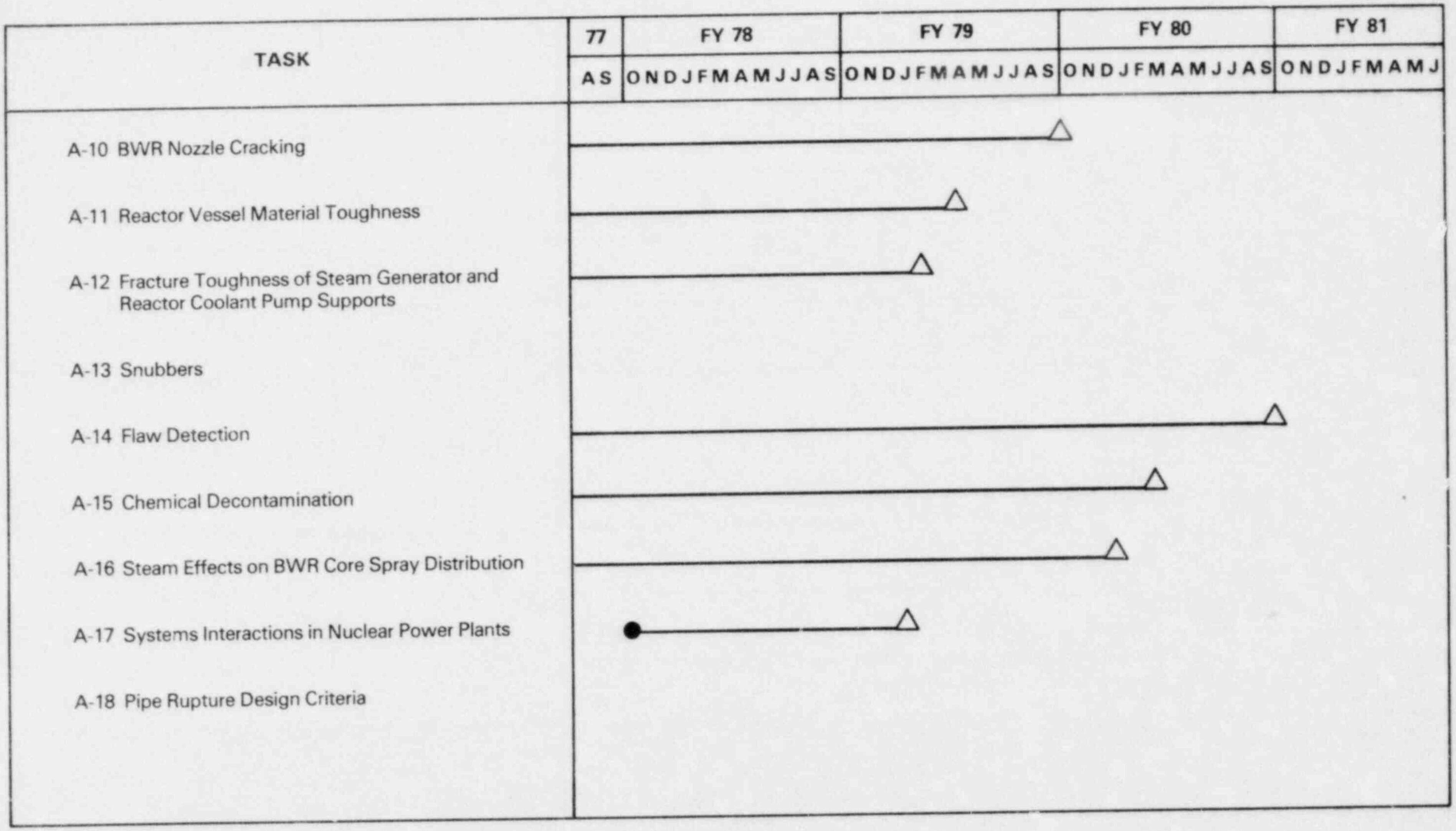
APPENDIX E

TASK ACTION PLAN SUMMARIES

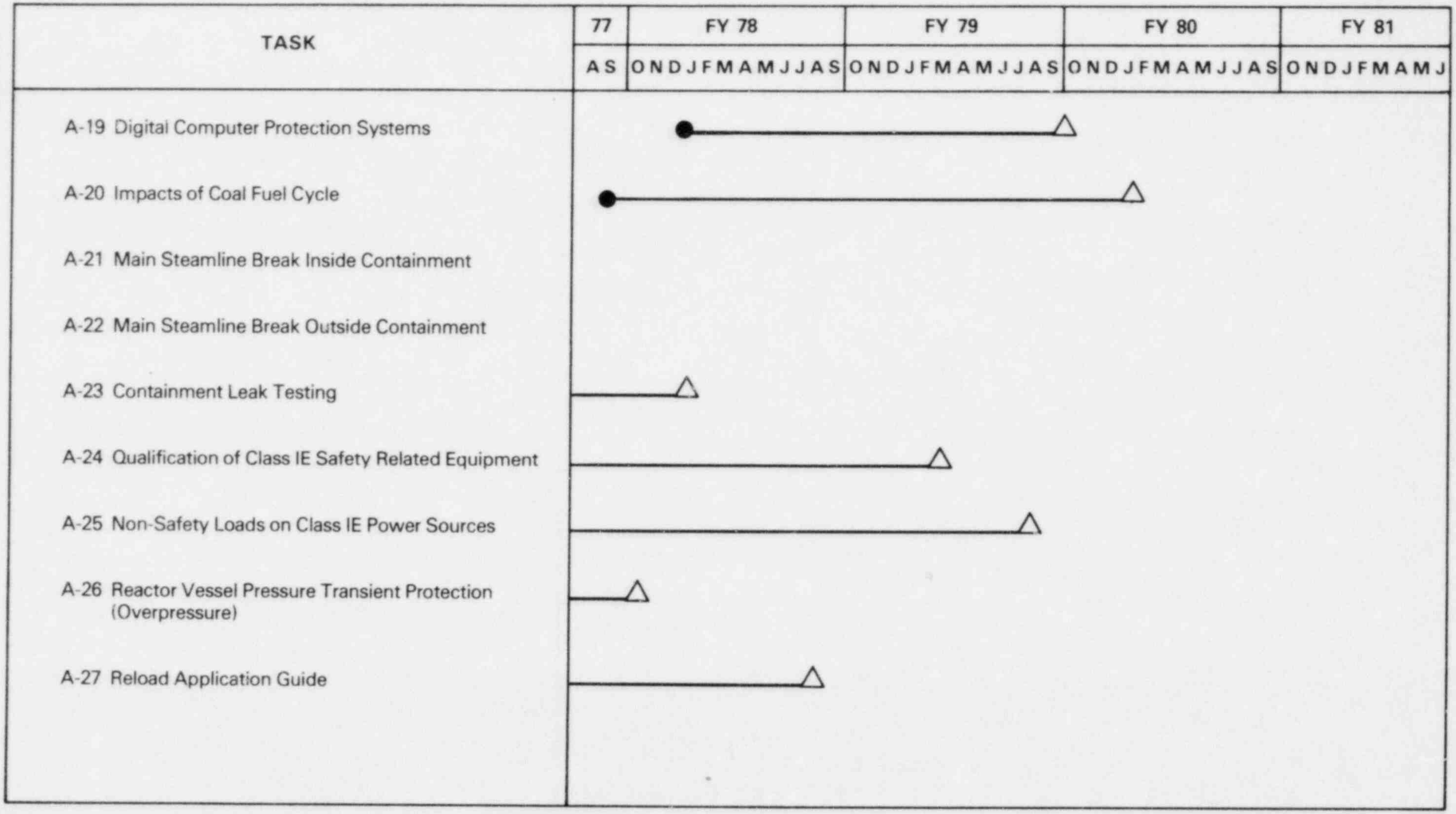
CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
SCHEDULE PROJECTIONS



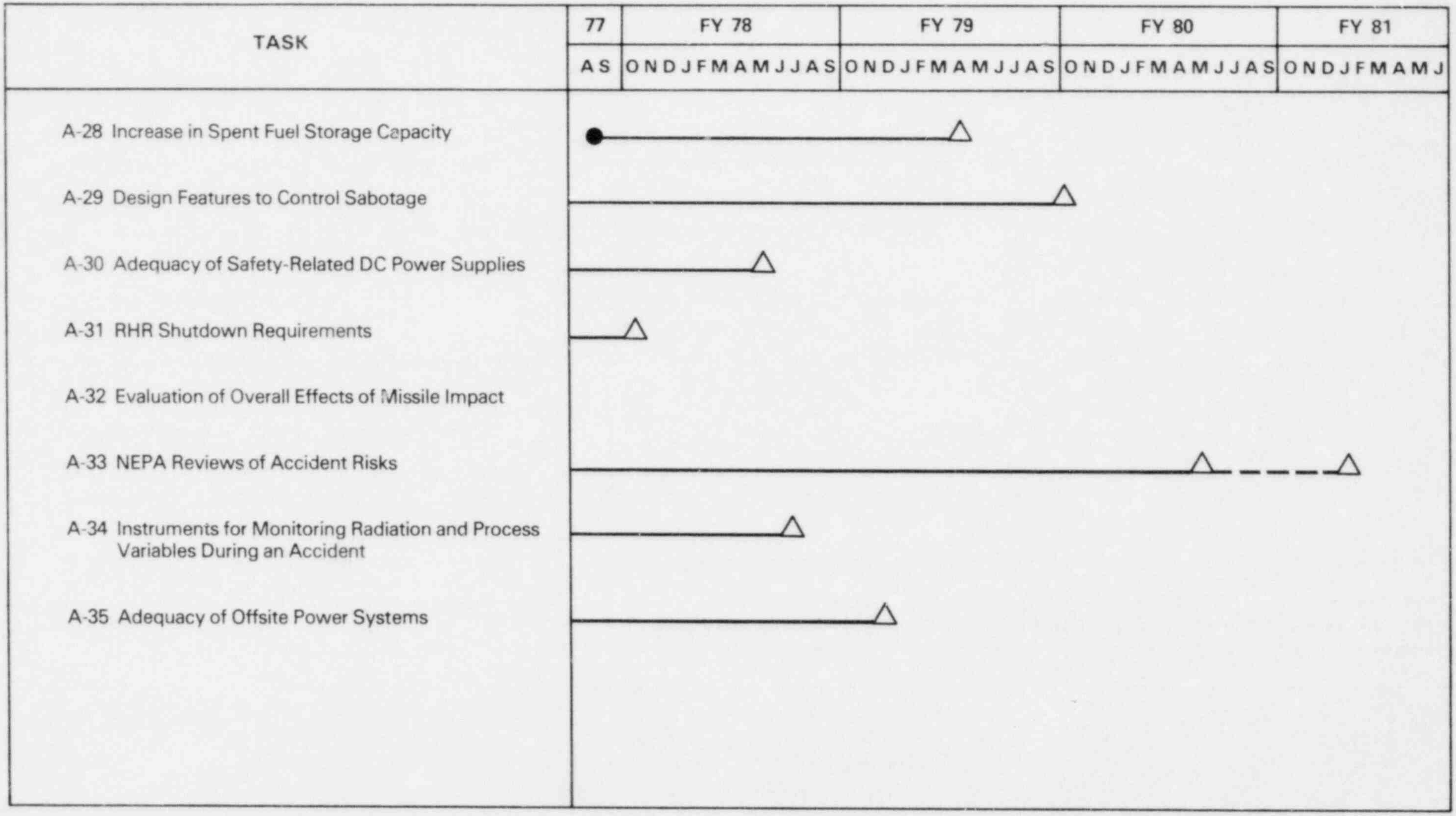
CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
SCHEDULE PROJECTIONS



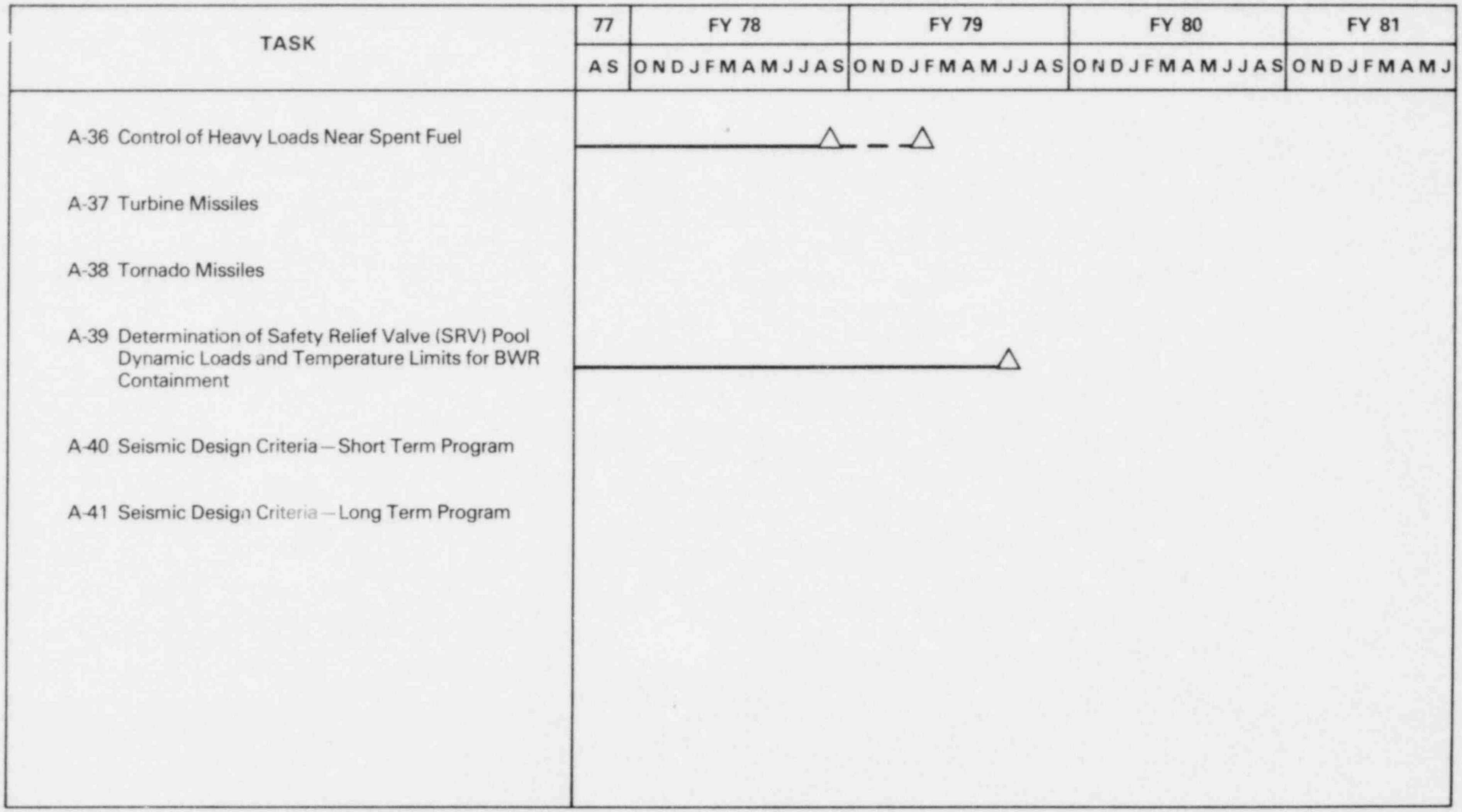
CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
SCHEDULE PROJECTIONS



CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
SCHEDULE PROJECTIONS



CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
SCHEDULE PROJECTIONS



CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
MANPOWER PROJECTIONS (PMY)*

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-1 Water Hammer	4.7		.15	.15			.7	2.1			.8	.8					
A-2 Asymmetric Blowdown Loads on the Reactor Vessel	10.25		.25	.6			1.7	3			1.7	3					
A-3 W Steam Generator Tube Integrity	5.66		.3	.2			1.33	.7	.2		1.33	.8				.8	
A-4 CE Steam Generator Tube Integrity	3.46		.3	.1			.83	.7			.83	.7					
A-5 B&W Steam Generator Tube Integrity	4.11		.35	.2			.83	.7			.83	.7				.5	
A-6 Mark I Short Term Program	0																
A-7 Mark I Long Term Program	7.75		.55	.3			3	1.5			2	.4					
A-8 Mark II Program	6.55					.1	.1	3.75		.1	.1	2.4					
A-9 ATWS	9.75						5.5	3.2	1.05								

*PMY-Professional Manyears

DPM - Division of Project Management
DOR - Division of Operating Reactors
DSS - Division of Systems Safety
DSE - Division of Site Safety and Environmental Analysis

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
MANPOWER PROJECTIONS (PMY)*

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-10 BWR Nozzle Cracking	9.3		2.3	.6			2.1	1.1			2.1	1.1					
A-11 Reactor Vessel Material Toughness	10		1.5	.5			3	1			3	1					
A-12 Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	3.1		.4	.3			1.1	.3			.7	.3					
A-13 Snubbers																	
A-14 Flaw Detection	20.8		2.1	1.8			3.1	2.9			3.2	2.7			2.5	2.5	
A-15 Chemical Decontamination	4.95		.5				1.85				2.6						
A-16 Steam Effects on BWR Core Spray Distribution	2.74		.33	.33			.33	.5			.33	.5			.16	.26	
A-17 Systems Interactions in Nuclear Power Plants	5.17					1.34	.64	2.02	.14	.33	.16	.51	.03				
A-18 Pipe Rupture Design Criteria																	

*PMY-Professional Manyears

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
MANPOWER PROJECTIONS (PMY)*

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-19 Digital Computer Protection Systems	1.06					.03		.36		.02	.04	.61					
A-20 Impacts of Coal Fuel Cycle	1.37				.07				.8				.5				
A-21 Main Steamline Break Inside Containment																	
A-22 Main Steamline Break Outside Containment																	
A-23 Containment Leak Testing	.92		.1	.4			.2	.22									
A-24 Qualification of Class IE Safety Related Equipment	8.33					.08		4.46	.18	.07		3.41	.13				
A-25 Non-Safety Loads on Class IE Power Sources	.8			.05				.2				.55					
A-26 Reactor Vessel Pressure Transient Protection (Overpressure)	.2		.14	.04				.02									
A-27 Reload Application Guide	1.7		.3				1.2	.2									

*PMY-Professional Manyears

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
MANPOWER PROJECTIONS (PMY)*

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-28 Increase in Spent Fuel Storage Capacity	2.02					.27	1.12	.45	.18								
A-29 Design Features to Control Sabotage	1.62						.3				1.1	.22					
A-30 Adequacy of Safety-Related DC Power Supplies	1.9						.33	1.57									
A-31 RHR Shutdown Requirements	.04			.04													
A-32 Evaluation of Overall Effects of Missile Impact																	
A-33 NEPA Reviews of Accident Risks	3.46							.38	1.15				1.06				.87
A-34 Instruments for Monitoring Radiation and Process Variables During an Accident	2.2				.1	.2	.4	.75	.75								
A-35 Adequacy of Offsite Power Systems	2.44						1.06	1.14			.14	.1					

*PMY-Professional Manyears

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
MANPOWER PROJECTIONS (PMY)*

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-36 Control of Heavy Loads Near Spent Fuel	1.56						1.02	.27	.27								
A-37 Turbine Missiles																	
A-38 Tornado Missiles																	
A-39 Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment	3.4		.1	.22		.1	.2	1.78		.1	.1	.8					
TOTALS (Numbers in parentheses represent total man years, i.e., PMY plus supervisory, clerical and administrative support.)	141.32 (199)	Total FY 77 15.68 (22)				Total FY 78 74.05 (104)				Total FY 79 44.00 (62)				Total FY 80 7.59 (11)			

(32 Task Action Plans)

*PMY-Professional Manyears

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
Technical Assistance Funds Projections
(Thousands of Dollars)

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-1 Water Hammer	365		75	10			85	145				50					
A-2 Asymmetric Blowdown Loads on the Reactor Vessel	465		105	80			180	100									
A-3 W Steam Generator Tube Integrity	948[1]		(50) 9	(75) 8			(100) 2	(100) 00			150 1	75					
A-4 CE Steam Generator Tube Integrity	150[1]														1	50	
A-5 B&W Steam Generator Tube Integrity	0[1]																
A-6 Mark I Short Term Program	0																
A-7 Mark I Long Term Program	365		155 2	5			40 1	20			2	5					
A-8 Mark II Program																	
A-9 ATWS	130							130									

[1] All programs under A-3 are applicable to Tasks A-3, A-4 and A-5.

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
Technical Assistance Funds Projections
(Thousands of Dollars)

Page 2 of 5

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-10 BWR Nozzle Cracking	245		60	25			80	80									
A-11 Reactor Vessel Material Toughness	510		120	140			50	75			50	75					
A-12 Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	150						150										
A-13 Snubbers																	
A-14 Flaw Detection	0																
A-15 Chemical Decontamination	350		50				150				150						
A-16 Steam Effects on BWR Core Spray Distribution	0																
A-17 Systems Interactions in Nuclear Power Plants	60						60										
A-18 Pipe Rupture Design Criteria																	

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
Technical Assistance Funds Projections
(Thousands of Dollars)

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-19 Digital Computer Protection Systems	30							10				20					
A-20 Impacts of Coal Fuel Cycle	150								100				50				
A-21 Main Steamline Break Inside Containment																	
A-22 Main Steamline Break Outside Containment																	
A-23 Containment Leak Testing	0																
A-24 Qualification of Class IE Safety Related Equipment	0																
A-25 Non-Safety Loads on Class IE Power Sources	130			65				65									
A-26 Reactor Vessel Pressure Transient Protection (Overpressure)	0																
A-27 Reload Application Guide	0																

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
Technical Assistance Funds Projections
(Thousands of Dollars)

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	E	DPM	DOR	DSS	DSE
A-28 Increase in Spent Fuel Storage Capacity	0																
A-29 Design Features to Control Sabotage	0																
A-30 Adequacy of Safety-Related DC Power Supplies	0																
A-31 RHR Shutdown Requirements	0																
A-32 Evaluation of Overall Effects of Missile Impact																	
A-33 NEPA Reviews of Accident Risks	150								150								
A-34 Instruments for Monitoring Radiation and Process Variables During an Accident	0																
A-35 Adequacy of Offsite Power Systems																	

CATEGORY A GENERIC ISSUES
TASK ACTION PLAN SUMMARY
Technical Assistance Funds Projections
(Thousands of Dollars)

TASK	TOTAL	FY 77				FY 78				FY 79				FY 80			
		DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE	DPM	DOR	DSS	DSE
A-36 Control of Heavy Loads Near Spent Fuel	0																
A-37 Turbine Missiles																	
A-38 Tornado Missiles																	
A-39 Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment	135			60				60				15					
A-40 Seismic Design Criteria—Short Term Program																	
A-41 Seismic Design Criteria—Long Term Program																	
TOTALS (32 Task Action Plans)	4,333	Total FY 77 1,193				Total FY 78 2,230				Total FY 79 760				Total Fy 80 150			

APPENDIX F

NUREG-0371
Vol. 1, No. 1
Revision 1

APPROVED TASK ACTION PLANS FOR CATEGORY A GENERIC ACTIVITIES

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

NUREG-0371
Vol. 1, No. 1
Revision 1

APPROVED TASK ACTION PLANS FOR CATEGORY A GENERIC ACTIVITIES

Manuscript Completed: November 1977
Date Published: December 1977

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Foreword

This document contains listings of generic technical activities as identified and placed in priority categories by the Office of Nuclear Reactor Regulation (NRR). In addition, it contains definitions of Priority Categories A, B, C and D and copies of thirty-two approved Task Action Plans for Category A activities.

This material was developed within the context of NRR's Generic Technical Activities Program. As part of this program, the assignment of identified issues to priority categories and the approval of Task Action Plans were made by NRR's Technical Activities Steering Committee, chaired by the Deputy Director, NRR.

The original document was published in November, 1977. Revision 1 added the Task Action Plan for Task No. A-17, Systems Interactions in Nuclear Power Plants. As additional Task Action Plans are approved by the Steering Committee and approved Task Action Plans are revised, this document will be updated.

NUREG-0371
Revision 1

December, 1977

APPROVED CATEGORY A
TASK ACTION PLANS

A-1	Water Hammer	
A-2	Asymmetric Blowdown Loads on the Reactor Vessel	
A-3	Westinghouse Steam Generator Tube Integrity	
A-4	Combustion Engineering Steam Generator Tube Integrity	
A-5	Babcock & Wilcox Steam Generator Tube Integrity	
A-6	Mark I Short Term Program	
A-7	Mark I Long Term Program	
A-8	Mark II Containment Pool Dynamic Loads	
A-9	ATWS	
A-10	BWR Nozzle Cracking	
A-11	Reactor Vessel Material Toughness	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	
A-14	Flaw Detection	
A-15	Chemical Decontamination	
A-16	Steam Effects on BWR Core Spray Distribution	
A-17	Systems Interactions in Nuclear Power Plants	1
A-19	Digital Computer Protection System	
A-20	Impacts of Coal Fuel Cycle	
A-23	Containment Leak Testing	
A-24	Qualification of Class IE Electrical Equipment	
A-25	Non-Safety Loads on Class IE Power Sources	
A-26	Reactor Vessel Pressure Transient Protection	
A-27	Reload Applications	
A-28	Increase in Spent Fuel Pool Storage Capacity	
A-29	Design Features to Control Sabotage	
A-30	Adequacy of Safety Related DC Power Supplies	

PRIORITY CATEGORIES

CATEGORY A:

THOSE GENERIC TECHNICAL ACTIVITIES JUDGED BY THE STAFF TO WARRANT PRIORITY ATTENTION IN TERMS OF MANPOWER AND/OR FUNDS TO ATTAIN EARLY RESOLUTION. THESE MATTERS INCLUDE THOSE THE RESOLUTION OF WHICH COULD (1) PROVIDE A SIGNIFICANT INCREASE IN ASSURANCE OF THE HEALTH AND SAFETY OF THE PUBLIC, OR (2) HAVE A SIGNIFICANT IMPACT UPON THE REACTOR LICENSING PROCESS.

CATEGORY B:

THOSE GENERIC TECHNICAL ACTIVITIES JUDGED BY THE STAFF TO BE IMPORTANT IN ASSURING THE CONTINUED HEALTH AND SAFETY OF THE PUBLIC BUT FOR WHICH EARLY RESOLUTION IS NOT REQUIRED OR FOR WHICH THE STAFF PERCEIVES A LESSER SAFETY, SAFEGUARDS OR ENVIRONMENTAL SIGNIFICANCE THAN CATEGORY A MATTERS.

CATEGORY C:

THOSE GENERIC TECHNICAL ACTIVITIES JUDGED BY THE STAFF TO HAVE LITTLE DIRECT OR IMMEDIATE SAFETY, SAFEGUARDS OR ENVIRONMENTAL SIGNIFICANCE, BUT WHICH COULD LEAD TO IMPROVED STAFF UNDERSTANDING OF PARTICULAR TECHNICAL ISSUES OR REFINEMENTS IN THE LICENSING PROCESS.

CATEGORY D:

THOSE PROPOSED GENERIC TECHNICAL ACTIVITIES JUDGED BY THE STAFF NOT TO WARRANT THE EXPENDITURE OF MANPOWER OR FUNDS BECAUSE LITTLE OR NO IMPORTANCE TO THE SAFETY, ENVIRONMENTAL OR SAFEGUARDS ASPECTS OF NUCLEAR REACTORS OR TO IMPROVING THE LICENSING PROCESS CAN BE ATTRIBUTED TO THE ACTIVITY.

November 1, 1977

CATEGORY A TECHNICAL ACTIVITIES

Task No.	Title
A-1	Water Hammer
A-2	Asymmetric Blowdown Loads on the Reactor Vessel
A-3	W Steam Generator Tube Integrity
A-4	CE Steam Generator Tube Integrity
A-5	B&W Steam Generator Tube Integrity
A-6	Mark I Short Term Program
A-7	Mark I Long Term Program
A-8	Mark II Program
A-9	ATWS
A-10	BWR Nozzle Cracking
A-11	Reactor Vessel Materials Toughness
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports
A-13	Snubbers
A-14	Flaw Detection
A-15	Chemical Decontamination
A-16	Steam Effects on BWR Core Spray Distribution
A-17	Systems Interaction in Nuclear Power Plants
A-18	Pipe Rupture Design Criteria
A-19	Digital Computer Protection Systems

Task No.	Title
A-20	Impacts of Coal Fuel Cycle
A-21	Main Steamline Break Inside Containment
A-22	Main Steamline Break Outside Containment
A-23	Containment Leak Testing
A-24	Qualification of Class IE Safety Related Equipment
A-25	Non-Safety Loads on Class IE Power Sources
A-26	Reactor Vessel Pressure Transient Protection (Overpressure)
A-27	Reload Application Guide
A-28	Increase in Spent Fuel Storage Capacity
A-29	Design Features to Control Sabotage
A-30	Adequacy of Safety Related DC Power Supplies
A-31	RHR Shutdown Requirements
A-32	Evaluation of Overall Effects of Missiles
A-33	NEPA Reviews of Accident Risks
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents
A-35	Adequacy of Offsite Power Systems
A-36	Control of Heavy Loads Near Spent Fuel
A-37	Turbine Missiles
A-38	Tornado Missiles
A-39	Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments

TASK NO.	TITLE
A-40	SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM
A-41	SEISMIC DESIGN CRITERIA - LONG TERM PROGRAM

CATEGORY B TECHNICAL ACTIVITIES

Task No.	Title
B-1	Environmental Technical Specifications
B-2	Forecasting Electricity Demand by State in the United States on an Annual Basis
B-3	Event Categorization
B-4	ECCS Reliability
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containment
B-6	Loads, Load Combindations, Stress Limits
B-7	Secondary Accident Consequence Modeling
B-8	Locking Out of ECCS' Power Operated Valves
B-9	Electrical Cable Penetrations of Containment
B-10	Behavior of BWR Mark III Containment
B-11	Subcompartment Standard Problems
B-12	Containment Cooling Requirements (Non-LOCA)
B-13	Marviken Test Data Evaluations
B-14	Study of Hydrogen Mixing Capability in Containmnet Post-LOCA
B-15	CONTEMPT Computer Code Maintenance
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Task No.	Title
B-17	Criteria for Safety-Related Operator Actions
B-18	Vortex Suppression Requirements for Containment Sumps
B-19	Thermal-Hydraulic Stability
B-20	Standard Problem Analysis
B-21	Core Physics
B-22	LWR Fuel
B-23	LMFBR Fuel
B-24	Seismic Qualification of Electrical and Mechanical Components
B-25	Piping Benchmark Problems
B-26	Containment Penetrations
B-27	Implementation and Use of Subsection NF
B-28	Radionuclide/Sediment Transport Program
B-29	Effectiveness of Ultimate Heat Sinks
B-30	Design Basis Floods and Probability
B-31	Dam Failure Model
B-32	Ice Effects on Safety Related Water Supplies
B-33	Dose Assessment Methodology
B-34	Occupational Radiation Exposure Reduction
B-35	Confirmation of Appendix I Models for "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors"

TASK NO.	TITLE
B-36	DEVELOP DESIGN, TESTING AND MAINTENANCE CRITERIA FOR ATMOSPHERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS FOR ENGINEERED-SAFETY-FEATURE SYSTEMS AND FOR NORMAL VENTILATION SYSTEMS
B-37	CHEMICAL DISCHARGES TO RECEIVING WATERS
B-38	RECONNAISSANCE LEVEL INVESTIGATIONS
B-39	TRANSMISSION LINES
B-40	EFFECTS OF POWER PLANT ENTRAINMENT ON PLANKTON
B-41	IMPACTS ON FISHERIES
B-42	SOCIOECONOMIC ENVIRONMENTAL IMPACTS
B-43	VALUE OF AERIAL PHOTOGRAPHS FOR SITE EVALUATION
B-44	FORECASTS OF GENERATING COSTS OF COAL AND NUCLEAR PLANTS
B-45	NEED FOR POWER - ENERGY CONSERVATION
B-46	COSTS OF ALTERNATIVES IN ENVIRONMENTAL DESIGN
B-47	INSERVICE INSPECTION CRITERIA FOR SUPPORTS AND BOLTING OF CLASS 1, 2, 3 AND MC COMPONENTS
B-48	BWR CRD MECHANICAL FAILURE (COLLET HOUSING)
B-49	INSERVICE INSPECTION CRITERIA FOR CONTAINMENT
B-50	REQUIREMENTS FOR POST-OBE INSPECTION
B-51	ASSESSMENT OF INELASTIC ANALYSIS TECHNIQUES
B-52	FUEL ASSEMBLY SEISMIC AND LOCA RESPONSES
B-53	LOAD BREAK SAFETY SWITCH

Task No.	Title
B-54	Ice Condenser Containments
B-55	Improved Reliability of Target-Rock Safety-Relief Valves
B-56	Diesel Reliability
B-57	Station Blackout
B-58	Passive Mechanical Failures
B-59	Review of (N-1) Loop Operation in BWRs and PWRs
B-60	Loose Parts Monitoring Systems
B-61	Allowable ECCS Equipment Outage Periods
B-62	Re-examination of Technical Bases for Establishing SLs, LSSSs, etc.
B-63	Isolation of Low Pressure Systems Connected to RCPB
B-64	Decommissioning of Reactors
B-65	Iodine Spiking
B-66	Control Room Infiltration Measurements
B-67	Effluent and Process Monitoring Instrumentation
B-68	Pump Overspeed During a LOCA
B-69	ECCS Leakage Ex-containment
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps
B-71	Incident Management
B-72	Development of Models for Assessing Risk of Health Effects and Life Shortening from Uranium and Coal Fuel Cycles

CATEGORY C TECHNICAL ACTIVITIES

Task No.	Title
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electric Equipment
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure
C-3	Insulation Usage Within Containment
C-4	Statistical Methods for ECCS Analysis
C-5	Decay Heat Update
C-6	LOCA Heat Sources
C-7	PWR System Piping
C-8	Main Steam Line Leakage Control System
C-9	RHR Heat Exchanger Tube Failures
C-10	Effective Operation of Containment Sprays in a LOCA
C-11	Assessment of Failure and Reliability of Pumps and Valves
C-12	Primary System Vibration Assessment
C-13	Non-Random Failures
C-14	Storm Surge Model for Coastal Sites
C-15	NUREG Report for Liquid Tank Failure Analysis

Task No.	Title
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

CATEGORY D TECHNICAL ACTIVITIES

Task No.	Title
D-1	Advisability of a Seismic Scram
D-2	Emergency Core Cooling System Capability for Future Plants
D-3	Control Rod Drop Accident (BWRs)

SEP 1 1977

REVISION 0

Title: Water Hammer (A-1)

Lead Responsibility: Division of Systems Safety

Lead Assistant Director: D. F. Ross, Jr., A/D for Reactor Safety

Task Manager: Charles C. Graves, DSS

1. Problem Description:

Since 1971 there have been about 50 incidents involving water hammers in BWR's and PWR's which have been cited in Licensee Event Reports. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

No water hammer incident has resulted in the release of radioactivity outside of the plant. However, because of the continuing incidence of water hammer events, the number of phenomena, and the potential safety significance of the systems involved, systematic review procedures should be developed to ensure that water hammer is given appropriate consideration in CP and OL licensing reviews and in reviews of operating reactors. There is also a need for systematic investigations of potential water hammer phenomena to obtain information to be used in providing guidance for the licensing review process and developing NRC positions on water hammer for use in the SRP. These investigations will also provide guidance and methods for understanding and resolving water hammer problems in existing plants.

2. Plan for Problem Resolution:

The overall program for resolution of the water hammer issue is divided into four tasks.

Task 1.0 Water Hammer Summary Reports

Under this task the initial and final summary reports on water hammer will be prepared.

Task 1.1 Water Hammer Report by DOR/DSS Technical Review Group

An interdivisional (DOR/DSS) Technical Review Group on Water Hammer Phenomena was established on March 10, 1977. In accordance with its charter, this group will prepare a report that will "review operating

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED
SEPTEMBER 1, 1977

experience and analytical investigations to date, place the safety significance of water hammer phenomena in nuclear plants in perspective, and summarize the current staff position regarding water hammer phenomena for CP and OL reviews and reviews of operating plants." A draft of this report has been prepared. Extensive revisions of the draft will be made prior to its submission for approval at the Assistant Director and Director levels. The report will provide input for Tasks 2.0, 3.0, and 4.0.

Task 1.2 Final Summary Report on Water Hammer

A NUREG report will be prepared which summarizes the results of this Category A task on water hammer.

Task 2.0 Revision of CP and OL Review Procedures

The objective of this task is to develop systematic review procedures concerning water hammer for use in the CP and OL review process.

The Standard Format and Standard Review Plan will be revised to require the applicant, as appropriate, to: (1) address potential water hammer problems in various systems; (2) demonstrate that there are adequate design features and operating procedures to prevent damaging water hammer events; and (3) expand the preoperational testing program to include verification that these design features and operating procedures do prevent damaging water hammer events. In addition, guidance for the licensing review process will be prepared in the form of Branch Technical Positions for steam generators, feedwater systems, and other systems, where required.

Requests for preparation or modification of Regulatory Guides and changes to the Standard Technical Specifications will also be made under this task. In view of the relatively short time scale of the overall water hammer task, performance of the task objectives will not be keyed to the issuance of new or modified Regulatory Guides. However, SD will be contacted at an early stage to permit them to make changes in manpower plans for work on the guides.

Work accomplished under this task will be based on the Task 1.1 report and the information developed under Task 4.0. Branches assigned primary review responsibility in the SRP will have the responsibility for all revisions to a given section of the SRP and the corresponding section of the Standard Format. This will include the responsibility for obtaining concurrence of any other branch assigned a secondary review or coordination responsibility in the given section of the SRP.

Task 3.0 Water Hammer Positions for Operating Reactors

Task 3.1 Short-Term Position

The DOR/DSS technical review group concluded that continued short-term plant operation is justified in view of the low probability of a water hammer resulting in unacceptable consequences. However, the staff also concluded that a particular type of water hammer, namely, those due to the rapid condensation of steam in feedwater lines of PWR's, represent the most immediate potential safety concern and that further actions by licensees were warranted to assure that an acceptably low risk to public safety is maintained. This is appropriate since steam generator feedwater line water hammers are well enough understood at this time to permit staff action. Accordingly, a generic position addressing this concern is being developed by DOR for operating plants and will be transmitted to affected licensees. A request for licensee proposed plant modifications to eliminate this concern and a more comprehensive reporting of water hammer events in the future will be included.

Task 3.2 Long-Term Position

Following completion of Tasks 2.0 and 4.0 and based on further data from operating plants, an assessment will be made of the need for any further requirements to be imposed on operating plants for other types of water hammer events. This assessment, which will include an impact/value appraisal, will consider all types of water hammers which are found to be significant to safety under Task 4.0.

Task 4.0 Water Hammer Safety Studies

The basic objective of this task is to obtain information and develop analytical methods and calculations regarding water hammer which will be used in completing the revisions of CP and OL review procedures under Task 2.0 and in implementing the long-term position paper of Task 3.0. The results of this task will also be used in implementing the revised CP and OL procedures developed under Task 2.0 and in the evaluation of water hammer incidents at operating reactors. The major part of the work will be done under technical assistance contracts.

Task 4.1 Review and Evaluation of Potential Water Hammer Problems

This task, which will be completed under a technical assistance contract, will involve the review and evaluation of those actual and potential water hammer problems considered to be significant in the Task 1.1 report. The first objective is to identify typical scenarios (e.g., basic initiating mechanisms, design features, operating procedures, anticipated transients, and single failures) that could result in water hammer events. The safety significance of the water hammer events will then be assessed in terms of probability of occurrence and consequences. Where necessary, recommendations will be made on possible design or procedural changes to prevent the occurrence or minimize the consequences of the postulated water hammer. Recommendations will also be made on criteria to be used in the licensing process. The second objective is to evaluate design features, operating procedures, and systems (e.g., BWR jockey pump system) which are used to prevent the occurrence of water hammer and to make recommendations on criteria to be used in the licensing process. This task will not be concerned with new PWR steam generators which are treated separately in Task 4.3. The interim and final reports on this task will be distributed to responsible branches for consideration in completion of Tasks 2.0 and 3.0.

Task 4.2 Development of Current Information on Water Hammer

The objectives of this task are (1) to provide a state-of-the-art review of experimental and analytical work reported in domestic and foreign literature which is pertinent to water hammer problems in nuclear plants, (2) to monitor Licensee Event Reports and experimental work on LOCA and ECC injection for information pertinent to water hammer, and (3) to ensure that information pertinent to water hammer which is obtained from licensees, vendors, and architect-engineers under Task 3.0 and given in applicant responses to questions raised during current CP and OL reviews will be brought to the attention of all responsible branches in DOR and DSS. The state-of-the-art review will be accomplished under a technical assistance contract. In support of the review, the Office of International Programs will be requested to obtain information from foreign sources on analyses and tests pertinent to water hammer in nuclear plants. Interim and final reports on the review will be sent to responsible branches. Information from the monitoring functions will be distributed when received via memoranda to responsible branches. The information obtained from the licensees and applicants will be maintained in control files for use by all responsible branches.

Task 4.3 Water Hammer in PWR Steam Generators

A. Current Steam Generator Designs

A number of damaging water hammer events have occurred which involve current steam generator designs with feedwater rings located near the top of the tube bundle and auxiliary feedwater lines connected to the main feedwater lines. A report (NUREG-0291) has been completed under a technical assistance contract in FY 1977 which deals with this water hammer problem.

B. New Steam Generator Designs

Some new steam generator designs incorporate bottom feed and preheater boxes. Recent tests have indicated that these designs may be susceptible to water hammer resulting from rapid steam condensation when cold auxiliary feedwater is added to the preheater. Potential water hammer problems for all new designs will be evaluated under this task. The major portion of the work will be done under a technical assistance program managed by the Auxiliary Systems Branch. Work during fiscal 1978 will cover review of scaling relationships presently available and the applicability of 1/8-scale test data in predicting results for full-scale steam generators. The FY 1979 work will involve review and evaluation of vendor design changes intended to prevent water hammer and consideration of other possible design changes and operating procedures for preventing water hammer. The results of this task will be used in defining an NRC position on new PWR steam generator designs under Task 2.0.

Task 4.4 Water Hammer Calculations

There is a currently funded contract at Lawrence Livermore Laboratory, managed by the Engineering Branch, which is concerned with calculations of pressure transients and stresses in PWR feedwater lines, using forcing functions assumed to represent those resulting from rapid condensation in the steam generator feedring. A final report on this work is scheduled for the end of FY 1977.

For FY 1978 a new technical assistance program is scheduled. A major objective of the new program is to provide analytical methods and calculations to be used in the evaluation of water hammer incidents at operating reactors. Flow closure functions representing the various initiating events will be formulated. Existing computer programs will then be used to establish the system loading due to water hammer from various initiating events and to establish the sensitivity of these loads to system design parameters and operating procedures. The structural response to the water hammer will be calculated.

3. NRR Technical Organizations Involved:

a) Reactor Systems Branch, Division of Systems Safety, has lead responsibility for Tasks 1.0, 2.0, and 4.0 and has responsibility for sub-tasks 4.1 and 4.2.

In Task 2.0, Revision of CP and OL Review Procedures, RSB has responsibility for 1) revising sections of SF and SRP for which it has primary review responsibility and 2) preparing, if required, branch positions, requests for changes in the Standard Technical Specifications and requests for preparation or modification of regulatory guides pertinent to these sections.

Manpower Estimate: .05 Man-years FY 1977; 0.8 Man-years FY 1978;
0.4 Man-years FY 1979; 1.25 Man-years Total

b) Auxiliary Systems Branch, Division of Systems Safety, has responsibility for Task 4.3.

In Task 2.0, Revision of CP and OL Review Procedures, ASB has responsibility for 1) revising sections of SF and SRP for which it has primary review responsibility, and 2) preparing, if required, branch positions, requests for changes in the Standard Technical Specifications, and requests for preparation or modification of regulatory guides pertinent to these sections.

Manpower Estimate: 0.1 Man-years FY 1977; 1.0 Man-years FY 1978;
0.3 Man-years FY 1979; 1.4 Man-years Total

c) Containment Systems Branch, Division of Systems Safety.

In Task 2.0, Revision of CP and OL Review Procedures, CSB has responsibility for 1) revising sections of SF and SRP for which it has primary review responsibility, and 2) preparing, if required, branch positions, requests for changes in the Standard Technical Specifications, and requests for preparation or modification of regulatory guides pertinent to these sections.

Manpower Estimate: -- Man-years FY 1977; 0.1 Man-years FY 1978;
-- Man-years FY 1979; 0.1 Man-years Total

d) Mechanical Engineering Branch, Division of Systems Safety.

In Task 2.0, Revision of CP and Ol Review Procedures, MEB has responsibility for 1) revising sections of SF and SRP for which it has primary review responsibility, and 2) preparing, if required, branch positions, requests for changes in the Standard Technical Specifications, and requests for preparation or modification of regulatory guides pertinent to these sections.

Manpower Estimate: - Man-years FY 1977; 0.2 Man-years FY 1978;
0.1 Man-years FY 1979; 0.3 Man-years Total

e) Plant Systems Branch, Division of Operating Reactors, has lead responsibility for Task 3.0, has responsibility for Task 4.4, has lead responsibility for collection of operating experience and for the maintaining of files on information from licensees under Task 4.2.

Manpower Estimate: 0.1 Man-years FY 1977; 0.5 Man-years FY 1978;
0.5 Man-years FY 1979; 1.1 Man-years Total

f) Engineering Branch, Division of Operating Reactors, has responsibility for evaluation and guidance of piping and structural response methods and calculations of Task 4.4 has responsibility for assisting in preparation of positions developed in Tasks 3.1 and 3.2.

Manpower Estimate: 0.05 Man-years FY 1977; 0.2 Man-years FY 1978;
0.3 Man-years FY 1979; 0.55 Man-years Total

4. Technical Assistance:

a) Contractor to be Selected

1) Title: Study of Fluid Flow Instabilities in PWR Steam Generators

2) Responsible Division/Branch: Division of Systems Safety, Auxiliary Systems Branch

3) Scope: This activity will provide technical assistance in the Task 4.3 work on evaluating water hammer problems for new PWR steam generator design. The work will involve review and evaluation of scaling relationships, 1/8-scale tests, proposed design changes to prevent water hammer and consideration of alternative approaches to prevent water hammer.

4) Funding: $10K FY 1977; $50K FY 1978; $50K FY 1979; $110K Total

b) Contractor to be Selected

1) Title: Evaluation of Water Hammer Problem in Nuclear Power Systems

2) Responsible Division/Branch: Division of Systems Safety, Reactor Systems Branch

3) Scope: This is a program under Task 4.1 to define scenarios resulting in water hammer in various plant systems, evaluate the safety significance and where necessary, recommend possible changes to prevent the occurrence and/or minimize the consequences of the water hammer. A contract requisition and detailed work plan will be prepared under Task 4.1.

4) Funding: $70K FY 1978; $70K Total

c) Contractor to be Selected

1) Title: State-of-the-Art Review of Experimental and Analytical Work Pertinent to Water Hammer in Nuclear Plant Systems.

2) Responsible Division/Branch: Division of Systems Safety, Reactor Systems Branch.

3) Scope: Experimental and analytical work in the domestic and foreign literature will be reviewed for information pertinent to Water Hammer in Nuclear Plant Systems. A contractor requisition and detailed work plan will be prepared under Task 4.2.

4) Funding: $35K FY 1978; $35K Total

d) Lawrence Livermore Laboratory

1) Title: Effect of Hydraulic Shock on Water Hammer

2) Responsible Division/Branch: Division of Operating Reactors, Engineering Branch

3) Scope: This currently funded project under Task 4.4 will utilize existing structural dynamic computer programs to calculate the stresses from which one can determine the integrity of piping elements, supports, and the operability of mechanical components. The project, which will be completed in September 1977, involves the following four tasks:

Task 1 - Characterization of shock waves in terms of parameters that are determined to be important in affecting the pipe integrity and component operability.

Task 2 - Development of a piping system model and the calculation of loads on components due to shock waves or water hammer.

Task 3 - Development of three-dimensional finite elements models for pipe bends, elbows, pumps, valves and supports and the calculation of stresses, strains, and deformations. If necessary, material and geometrical nonlinearities will be incorporated.

Task 4 - Definition of component operability in terms of component strain or deformation during and following the transient. This task will be limited to pumps and valves only.

4) Funding: $75K FY 1977; $75K Total

e) Contractor to be Selected

1) Title: Water Hammer Calculations

2) Responsible Division/Branch: Division of Operating Reactors, Plant Systems Branch

3) Scope: This project, which is part of the Task 4.4 effort, is concerned with the hydrodynamic/structural interactions in systems subject to water hammer loads. A major objective is to provide analytical methods and calculations to be used intthe evaluation of water hammer incidents at operating reactors. The effort will involve numerical studies to establish the sensitivity ff the structural consequences to the parameters of the initiating water hammer. The object is to provide a range of system design and/or operating procedures within which operating reactors may be judged to meet the intent of future NRC guidelines on water hammer. A contractor requisition and detailed work plan will be prepared under Task 4.4.

4) Funding: \$85K FY 1978; \$85K Total

5. Interaction with Outside Organizations:

Individual licensees, vendors, and architect-engineers may be asked to supply information concerning plant-specific design features, operating procedures pertinent to water hammer. This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

6. Assistance Requirements from Other NRC Offices:

a) Office of International Programs

The Office of International Programs will be requested to obtain information on foreign programs or specific tests dealing with water hammer in nuclear plants.

b) Office of Standards Development, Division of Engineering Standards

On the basis of the work done under Task 2.0, it is expected that some requests will be made to the Division of Engineering Standards for modification and/or preparation of regulatory guides.

c) Office of Nuclear Regulatory Research, Division of Reactor Safety Research

There are no programs currently funded under the Division of Reactor Safety Research which are concerned specifically with water hammer problems. However, work sponsored by RSR (e.g., LOCA and ECC injection tests) will be monitored for information pertinent to water hammer. In addition, requests for research on water hammer may be made as the result of work on Tasks 2.0, 3.0, and 4.0.

7. Schedule for Problem Resolution:

The major milestones for the Water Hammer Program are as follows:

1)	Completion of Task 1.1 Report:	11/30/77
2)	Completion of the NRC Short-Term Position on Water Hammer for Operating Reactors (Task 3.1):	11/30/77
3)	Contractor report on state-of-the-art review under Task 4.2 completed	5/30/78
4)	Contractor report on review and evaluation of potential water hammer problems under Task 4.1 completed.	9/30/78
5)	Contractor report on FY 1978 study of water hammer in steam generators under Task 4.3 completed.	9/30/78
6)	Contractor report on water hammer calculations under Task 4.4 completed.	9/30/78
7)	Contractor report on FY 1979 study of water hammer in steam generators under Task 4.3 completed.	3/30/79
8)	Approval by Director, NRR, of Branch Positions and Revisions to the SF and SRP (Task 2.0):	04/30/79
9)	Approval by Director, NRR, of the NRC Long-Term Position on Water Hammer for Operating Reactors (Task 3.2):	04/30/79
10)	Completion of Final Summary Report (Task 1.2):	05/30/79

8. Potential Problems:

1) The DOR/DSS Technical Review Group Report under Task 1.1 is input to Tasks 2.0, 3.0, and 4.0. Extensive revision to the current draft is needed and two more stages of review and approval have been previously scheduled.

2) There is a general problem of achieving systematic and consistent treatment in the revisions to the SF and SRP under Task 2.0 and the preparation of positions under Task 3.0. This arises because of (1) the large number of systems involved, (2) the fact that some components and potential water hammer problems are the same for systems under different branches, and (3) the different approaches of individual branches. The water hammer problem should be considered by branches under the three Assistant Directors in the Division of Systems Safety and by two branches under the Assistant Director for Operational Technology in the Division of Operating Reactors. The interdivisional DOR/DSS Technical Review Group was set up to achieve systematic coverage of water hammer. However, under this task plan, this group will be involved only in completion of the Task 1.1 report. This task plan is set up to achieve major objectives using normal line-management. The key to successful task completion is (a) coordination of work objectives and personnel assignments at the A/D level prior to and after the initiation of branch efforts, (b) several stages of coordinated review within each division, and (c) provision for interdivisional comments and concurrence at the A/D level.

3) The early development of an NRC position on water hammer in steam generators is of major importance.

4) Completion of the staff position on new steam generator designs in time to meet the Task 2.0 completion date of 04/30/79 is dependent on the submission of test results from the vendors.

August 31, 1977
REVISION 0

CATEGORY A TECHNICAL ACTIVITY TASK NO. A-2

Title:	Asymmetric Blowdown Loads on PWR Reactor Vessel
Lead Responsibility:	Division of Operating Reactors
Lead Assistant Director:	Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR
Task Manager:	Charles M. Trammell, DOR

1. Problem Description

On May 7, 1975, the NRC was informed by Virginia Electric & Power Company that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by Westinghouse or Stone and Webster in the original design of the reactor vessel support system for North Anna, Units 1 and 2. In the event of a postulated LOCA at the vessel nozzle, asymmetric LOCA loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more detailed analytical models, it became apparent to Westinghouse that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports, thereby affecting their integrity. Although first identified at the North Anna facility, this concern has generic implications for all PWRs.

Upon postulation of a break in a reactor coolant pipe, at the abovementioned locations, several rapidly occurring events could cause internal and external transient loads to act upon the reactor vessel. For the reactor vessel pipe break at the inlet nozzle, asymmetric pressure changes take place in the annulus between the core barrel and the vessel. Decompression could occur on the side of the vessel annulus nearest the pipe break before pressure on the opposite side changes. The momentary difference in pressure across the core barrel could induce lateral loads in opposite directions on the core barrel and the reactor vessel. Vertical loads could also be applied to the core internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, as fluid escapes through the break, the annulus between the reactor vessel and biological shield wall could become asymmetrically pressurized resulting in a difference in pressure across the vessel causing additional horizontal and vertical external loads on the vessel. In addition, the vessel could be loaded by the effects of initial tension release and blowdown

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED AUGUST 31, 1977

thrust at the pipe break. The loads occur simultaneously. For a reactor vessel outlet break the same type of loadings could occur, but the internal loads would be predominantly vertical due to more rapid decompression of the upper plenum.

For each of the above-mentioned postulated breaks, the time history of the reactor vessel support reactions due to the complete set of simultaneous horizontal and vertical loads should be calculated.

In the event that such loadings would result in a significant degree of failure within the reactor pressure vessel support system and consequent vessel movement, there is a potential that this could (1) result in damage to the ECCS lines connected to the coolant loops, (2) affect the capability of the control rods to function properly, and (3) result in damage to other reactor coolant system components (pump and steam generator supports). In addition, the differential pressures occurring during sub-cooled blowdown could result in stresses on fuel assemblies caused by lateral core barrel and core plate motion. This could degrade heat transfer capability if fuel spacer grids are deformed by impacting either each other or the core baffle. (This loading can occur independently of vessel support failure.)

The above-described phenomena also apply generally to BWRs, but the potential loads are not expected to be as large since the pressure in the reactor vessel is lower and the reactor coolant is less subcooled.

2. Plan for Problem Resolution

Background

Following disclosure of this problem during the OL review of North Anna Units 1 and 2, a survey of all operating PWR reactors was conducted in October 1975. That survey showed that neither of the above described transient differential pressures had been considered in the design of the reactor vessel supports for any operating PWR facility.

In June 1976, the NRC requested all operating PWR licensees to proceed to assess the adequacy of the reactor vessel supports at their facilities with respect to these newly-identified loads. Most licensees having a common NSSS vendor took identical or similar positions with respect to this request and did not respond as requested.

Most licensees with Westinghouse plants proposed an augmented inservice inspection program (ISI) of the reactor vessel safe-end-to-end pipe welds in lieu of providing the detailed analysis we requested. Licensees with Combustion Engineering plants submitted a probability study (prepared by Science Applications, Inc.) in support of a conclusion that a break at this location has such a low probability of occurrence that no further analysis is necessary. B&W licensees have engaged Science Applications, Inc. for a similar study (not yet submitted).

When the W and CE owners group reports were received in September 1976, DOR formed a review Task Group consisting of members from DOR, DSS and EDO to evaluate these alternate proposals. In addition, EG&G Idaho, Inc. was contracted to perform an independent review of the submitted probability study. A short review schedule was established since it appeared that most licensees would hold off on further analysis pending our consideration of their submittals.

Our review of the proposed alternates has been completed. The Task Group and EG&G independently reached the same conclusion: that the alternate proposals set forth in these reports should not be accepted in lieu of the requested analyses. The basis is that a sufficient data base does not exist within the nuclear industry to provide satisfactory answers to many information needs we identified. This information would be needed to support this "no-break" approach. Further investigation of pipe break probabilities is planned by the staff (see item d. below).

Plan

a. Letters will be sent to all licensees and applicants stating that an analysis must be undertaken to assess the design adequacy of the reactor vessel supports and other structures to withstand the loads when asymmetric LOCA forces are taken into account.* We will point out that it may be possible to group plants such that only a limited number of plants need be analyzed, and that it may be possible to provide a simple "fix" (e.g., pipe restraints) which will permit bounding the problem. Therefore, the letters will request licensees and applicants to submit their schedule for completion of the task.

b. The staff will meet with the licensees constituting both the W and the CE owners group to explain why the probability study reports could not be accepted. We will also provide them all the questions that have been generated to date as a result of our review of the W and CE topical reports. (We will not issue formal requests for additional information on these topicals to these groups of licensees.)

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*Including an assessment of asymmetric loads produced by large pipe breaks outside the reactor vessel cavity.

c. We will review and approve vendor models and codes prior to plant-specific application. (This has been completed for W analysis methods).

d. The staff will develop explicit guidelines and acceptance criteria for the asymmetric LOCA load analysis, including load combinations and acceptable alternatives where, depending on the construction or operating status of a given plant, application of the guidelines per se could require modifications that are judged to be a practical impossibility. Such alternative guidelines would be designed to provide an adequate and acceptable LOCA load generic issue consistent with safe plant shutdown requirements.

e. The staff will conduct a pipe break probability study that will encompass (1) advances that are being made in nondestructive examination techniques, (2) an improved flaw distribution data base of actual NSSS materials, and (3) development of a realistic break opening model to describe pipe break characteristics. The pipe break probability study will be used to confirm the adequacy of staff decisions related to the continued operation of plants for the interim period until an analysis of these loads is conducted.

f. The staff will perform a series of sensitivity studies to independently evaluate the effect of noding upon the magnitude and distribution of pressures within typical reactor cavity designs. Results of sensitity studies will be utilized to prepare guidelines for the evaluation of the volumes within the confines of the reactor cavity.

3. NRR Technical Organizations Involved

a. Analysis Branch, Division of Systems Safety. Has lead responsibili sibility for review of vendor hydrodynamic analysis methods and codes.

Manpower Estimates: 0.2 man-years in FY 1977, 1 man-years in FY 1978, and 1 man-years in FY 1979.

b. Core Performance Branch, Division of Systems Safety. Has lead responsibility for reviewing vendor analysis methods for calculating loads on fuel assemblies resulting from subcooled decompression for plants under CP and OL review (not yet licensed for operation).

Manpower Estimates: 0.1 man-years in FY 1977, .5 man-years in FY 1978, and .5 man-years in FY 1979.

c. Containment Systems Branch, Division of Systems Safety. Responsible for reviewing vendor models and methods for calculating asymmetric cavity loads for all plants, and associated vendor models.

Manpower Estimates: 0.1 man-years in FY 1977, .5 man-years in FY 1978, and .5 man-years in FY 1979.

d. Mechanical Engineering Branch, Division of Systems Safety. Responsible for review of structural aspects of vendor analysis methods and codes for plants not licensed for operation. Responsible for developing structural acceptance criteria (with Engineering Branch, DOR). MEB will investigate the applicability of this problem to BWRs (with Engineering Branch, DOR).

Manpower Estimates: 0.2 man-years in FY 1977, 1.0 man-years in FY 1978, and 1.0 man-years in FY 1979.

e. Engineering Branch, Division of Operating Reactors. Responsible for review of structural aspects of analysis methods and codes applicable to operating reactors (including loads on fuel assemblies). Responsible for development of structural acceptance criteria (with Mechanical Engineering Branch, DSS). EB will investigate the application of this problem to BWRs (with MEB, DSS).

Manpower Estimates: 0.2 man-years in FY 1977, 1.5 man-years in FY 1978, and 1.5 man-years in FY 1979.

f. Operating Reactors Branch #1, Division of Operating Reactors. Responsible for the coordination and management of this Technical Activity.

Manpower Estimates: 0.05 man-years in FY 1977, .20 man-years in FY 1978, and .20 man-years in FY 1979.

4. Technical Assistance Requirements

a. Managed by DOR (Engineering Branch):

Contractor: EG&G Idaho, Inc.

Funds Available: $105K in FY 1977 and $180K in FY 1978

This is an NRC program to independently model representative Westinghouse 4-loop (Indian Point 3), B&W (Arkansas Nuclear One Unit 1), and CE (not yet selected) plants for the purpose of assessing the loads on all major structures and components resulting from asymmetric LOCA loads. The purpose of this program is to develop an independent NRC capability for performing inelastic dynamic analyses. Sensitivity studies will be performed to evaluate the effects of various break opening times, effects of component stiffness, and three dimensional coupling effects.

b. Managed by DSS (Mechanical Engineering Branch):

Contractor: EG & G Idaho, Inc.

Funds Available: $80K in FY 1977 and $100K in FY 1978

This is a NRC/DSS program to provide the staff with the analytical tools necessary to independently verify the selection of design basis pipe rupture locations; and to verify that the criteria for assurance of integrity under LOCA & SSE loads for reactor coolant piping, the reactor vessel, steam generators, main coolant pumps and the supports for these components have been implemented correctly. Verification analyses for a CE plant (San Onofre 2), a B&W plant (Bellefonte 1), a BWR plant and a 4-loop Westinghouse plant will be run to verify results reported by the applicants. Support models will be designed to be revised as necessary to represent various support configurations utilized by Architect/Engineers of the plants under CP/OL review.

5. Interactions with Outside Organizations

a. W Owners Group of licensees

The W owners group of licensees is an ad hoc organization of most (but not all) owners of operating W plants, formed for the purpose of sponsoring and proposing the augmented inservice inspection program (WCAP-8802) in lieu of furnishing the detailed analysis requested by NRC. This group of licensees has engaged Westinghouse Electric Corporation as its principal consultant.

With the advent of the NRC decision to request all licensees for a detailed analysis and to set aside - at least for the present - the ISI proposal, the continued role of this licensee group is undetermined.

b. CE Owners Group of Licensees

The CE owners group of licensees is also an ad hoc organization of most owners of operating CE plants. This group sponsored the probability study prepared by Science Applications, Inc., which concluded that the probability of severe pipe breaks that could trigger the loads under consideration is below the threshold of concern. The future role of this licensee group is also undetermined.

c. B&W Owners Group of Licensees

This group is composed of owners of B&W plants having nuclear steam supply systems of the same design (177 fuel assemblies, skirt supported vessels.) This group has engaged SAI and B&W as its consultant for the preparation of a probability study similar to the one done by SAI for the CE owners group. This report has not yet been submitted.

d. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

6. Assistance Requirements from Other NRC Offices

None

7. Schedule for Problem Resolution

The major milestones for resolution of this generic issue are as follows:

1. Approval of W detailed analysis methods (MULTIFLEX, used on Beaver Valley and North Anna while in OL review)	May 1977 (complete)
2. Letter advising all licensees to proceed with some form of analysis and advise NRR of schedule	September 1977 (targeted)
3. Determination of whether BWRs should be included in generic review	December 1977 (targeted)
4. Approval of B&W detailed analysis methods	January 1978 (targeted)
5. Development of structural acceptance criteria	June 1978 (targeted)
6. Approval of CE detailed analysis methods	December 1978 (targeted)

8. Potential Problems

1. Three owners groups representing most operating PWRs have been formed and either will propose or have proposed solutions different from the requested analysis (augmented ISI, probability studies). Therefore, strong industry resistance to our request for some form of analysis is possible.

2. Rigorous application of the generic structural acceptance criteria may require modifications that are judged to be impossible for some older plants. For these cases, alternative solutions may be required.

REVISION 0
SEPTEMBER 23, 1977

CATEGORY A TECHNICAL ACTIVITY NO. A-3

Title: Westinghouse Steam Generator Tube Integrity

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: Richard J. Stuart, Section Leader, Engineering Branch, DOR

1. Problem Description

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in the tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been nominally arrested; however, degradation due to denting which leads to primary side stress corrosion cracks is the major problem at present, and the principal focus of this technical activity.

2. Plan For Problem Resolution

The major portion of the NRC staff efforts related to the resolution of the denting problem will consist of evaluation of the results of investigations by Westinghouse, EPRI, and EPRI supported contractors. In addition, NRC supported technical assistance and confirmatory research programs will be used as the basis for evaluation of applicant supplied data.

The specific activities directed at resolution of the denting problem in Westinghouse steam generators consist of the following issues and tasks:

A. Generic Evaluation of ISI Results

Review and evaluate the various eddy current inspection results; i.e., experience from operating reactors and evaluate these data as they relate to the generic determination of failure probability of degraded tubes. In addition, evaluate the test programs and analytical studies to provide staff understanding sufficient to continue to provide justification for continued safe operation of operating reactors.

APPROVED BY TASC, SEPTEMBER 6, 1977
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SEPTEMBER 23, 1977

B. Evaluation of Transients and Postulated Accidents

Evaluation of failure consequences under postulated accident conditions (LOCA and MSLB) to determine the acceptable levels of primary to secondary leakage rates and the effect on ECCS performance. The results will be used to define the acceptable number of tube failures that may be necessary as a licensing basis considering predicted fuel behavior and radiological dose during transients and postulated accident conditions.

C. Evaluation of Steam Generator Tube Structural Integrity

Review and evaluate the structural integrity of steam generator tubes under normal operating and postulated accident conditions (LOCA, SSE and MSLB) including licensee and Westinghouse analyses where appropriate to generic conclusions.

D. Establish Tube Plugging Criteria

Establish a generic tube plugging criteria that is consistent with the determined allowable leak rate, tube structural integrity and degradation rates. These results will allow assessment of the adequacy of the requirements defined in Regulatory Guide 1.121.

E. Secondary Coolant Chemistry Requirements

Evaluate the mechanism of tube degradation. The results will be used to define the requirements for secondary coolant chemistry control including considerations for condenser in-leakage.

F. Evaluation of ISI Methods

Review the development of improved eddy-current probes, coils and multi-frequency techniques to better quantify dents and growth of dents and increase sensitivity for detecting cracks in dented regions.

G. Establish Criteria for Revision of Regulatory Guide 1.83

Integrate experience from inservice inspection results, the results from the evaluation of various ISI improvements and the plugging and secondary water chemistry requirements into criterion for possible revision of Regulatory Guide 1.83.

H. Steam Generator Replacement (Prototype)

Review and evaluate plans for initial steam generator replacement as generic basis for subsequent replacement actions.

I. Review Design Criteria for Plants Not Yet Licensed

Review and evaluate design modifications proposed by applicants and Westinghouse to prevent denting in plants not yet licensed for operation.

3. NRR Technical Organizations Involved

a. Engineering Branch, Division of Operating Reactors, has the primary lead responsibility for the overall review and evaluation of steam generator tube integrity. This includes operational experiences, tube failure mechanisms and potential repairs, plugging criteria, ISI requirements, tube failure probability, leakage rate limits, and secondary coolant system control. This also includes the lead responsibility for determining the probability of LOCA and MSLB initiating events and the probability of tube failures during these events and responsibility for deteriming the number of tubes assumed to fail in LOCA and MSLB analyses. The Engineering Branch also has lead responsibility for the review of prototype steam generator tube replacement

Manpower Estimates: 0.1 manyear FY 1977, 1.0 manyear FY 1978,
and 1.0 manyear FY 1979.

b. Environmental Evaluation Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of the off site dosage related to the consequence or probability of a Main Steam Line Break (MSLB) accident or LOCA given the physical conditions determined in part a, above. EEB will also consult with EB and provide support for the probabilistic evaluation of MSLB and LOCA initiating events, the probability of tube failures during these postulated events and evaluation of environmental aspects of steam generator tube replacement.

Manpower Estimates: 0.1 manyear FY 1977, 0.2 manyear FY 1978,
and 0.2 manyear FY 1979.

c. Reactor Safety Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of: (1) the ECCS performance related to secondary-to-primary leakage as a consequence of a LOCA, and (2) the effect of primary-to-secondary leakage during a MSLB accident on fuel failures.

Manpower Estimates: 0.1 manyear FY 1977, 0.13 manyear FY 1978, and
0.13 manyear FY 1979.

d. Mechanical Engineering Branch/Materials Engineering Branch, Division of Systems Safety, has lead responsibility for the review of new design/material concepts and new system component requirements. This will apply to PWR facilities not yet licensed for operation.

The activities involved will include the review and evaluation of applicant's and Westinghouse's proposed improvements on the design and/or operation of the steam generators; for items such as secondary coolant chemistry, design modifications to avoid denting, condenser design to avoid inleakage, ISI requirements, recommendation for revision of Regulatory Guides, and provisions for access opening and space in the containment to facilitate steam generator inspections.

Manpower Estimates: 0.1 manyear for FY 1977, 0.5 manyear FY 1978, 0.5 manyear 1979, and 0.5 manyear FY 1980.

e. Analysis Branch, Division of Systems Safety, has the lead responsibility in developing analytical capabilities (computer codes, etc.) to evaluate the effects of steam generator tube rupture (s) concurrent with various reactor transients that include MSLB and LOCA accidents. The purpose is to determine the equivalent number of tube failures that can be tolerated during transient events. This information will then be factored into the overall program of determining an adequate sample plan for tube inspections.

Manpower Estimates: 0.1 manyear FY 1977, 0.2 manyear FY 1978, and 0.2 manyear FY 1979.

f. Reactor Systems Branch, Division of Systems Safety. Has the responsibility of implementing new procedures on CP/OL safety analyses for plants yet to be licensed, should any be required as the results of this technical activity.

Manpower Estimates: 0.1 manyear FY 1979, 0.3 manyear FY 1980.

g. Environment Project Branch No. 1, Division of Site Safety and Environmental Analysis. Responsible for the review of the non-radiological environmental aspect of steam generator replacement for the lead unit.

Manpower Estimeates: 0.2 manyear FY 1978

4. Technical Assistance Requirements:

a. Contactor: Brookhaven National Laboratory (BNL) - DOR, DSS

Funds Required: $98K FY 1977, $200K FY 1978 and $175K FT 1979

This effort is funded as part of an overall program at BNL applicable to the three Category A Technical Activities (A-3, A-4 and A-5) related to PWR steam generators. Funding values under DORSAT are not included.

This program is needed to obtain technical consultation and assistance to review information in areas of water chemistry and corrosion analysis, monitored jointly by EB/DOR and MTEB/DSS. Stress and/or burst strength calculations are funded in part under DORSAT contract on an as-needed basis. This program will provide assistance in accomplishing Tasks 2C, 2E and 2G.

b. Contactor: Idaho National Engineering Laboratory (INEL) - DSS

Funds Required: $75K FY 1977, $100K FY 1978.

This effort is generic in nature and will be applicable to the three Category A Technical Activities (A-3, A-4 and A-5) related to PWR steam generators.

The purpose of this program is to determine the effect of steam generator tube plugging on the predicted peak clad temperatures following a postulated LOCA. The primary activity is to produce a reliable computer code to aid the evaluation of the effects of tube plugging on the ECCS performance. An addition to the program will be needed to consider steam generator tube failures concurrent with MSLB or a LOCA. This program will provide assistance in accomplishing Tasks 2B and 2D.

c. Contactor: Sandia Laboratories, DOR proposed

Funds Required: $50K FY 1977, $100K FY 1978, and $150K FY 1979.

This work is of generic nature, and will be applicable to all PWR steam generators.

A program is needed for a statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. This is a new program to augment staff effort in steam generator safety reviews and will assist in addressing Tasks 2A, 2F and 2G.

5. Assistance Requirements from Other NRC Offices:

a. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Metallurgy and Materials Branch and Probabilistic Analysis Branch

RES has funded, at the request of NRR, a major confirmatory experimental program at Pacific Northwest Laboratory. The activity of this program consists of a series of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leakage rate data. This program is managed by Metallurgy and Materials Branch, (Task 2C).

RES has been requested to fund a new program, possibly starting this fiscal year, addressing the factors which determine Inconel 600 susceptibility to stress corrosion cracking in primary water. Metallurgical condition, chemistry, temperature, stress and environment will be considered, (Task 2E).

The Probabilistic Analysis Branch funded the program to assist EEB in probabilistic analyses, (Task 2B).

b. Office of Standards Development, Division of Engineering Standards, Structures and Components Standards Branch

- OSD has funded a confirmatory research program at Battelle Columbus laboratory to evaluate eddy current methods for inspecting steam generator tubes as a subcontract to Brookhaven National Laboratory, (Part of Task 2F).

c. Office of the Executive Director For Operations, Applied Statistic Group.

- Provide assistance to EB/DOR for statistical assessment of steam generator tube integrity, (Part of Tasks 2A, 2F, and 2G).

d. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the committee as the task progresses.

6. Interactions With Outside Organizations

a. Licensee(s) of Westinghouse (W) Nuclear Facilities

At present all W plants experiencing tube denting will be monitored for the progress of denting. Each licensee will submit an analysis of the consequences of tube denting on tube integrity and demonstrate that adequate safety margins exist for continued safe operation. The Turkey Point and Surry licensees will be closely monitored relative to steam generator replacement.

b. Westinghouse

The primary interaction with Westinghouse has been and continues to be on the investigation program for the resolution of the problems at Westinghouse designed plants and their generic implication such as the licensing bases or justifications for continued operation for Westinghouse plants with known tube degradations. For interim periods of operation before the cause of denting is identified and corrective measures implemented, the interaction will be needed to ensure that Westinghouse develops and improves capabilities for the evaluation of ECCS performance under postulated accidents concurrent with tube failures, should such a licensing basis become necessary. Review and evaluate new designs proposed to prevent denting in facilities not yet licensed for opreration.

c. EPRI, PWR Owner Group etc.

Interactions with other organizations such as the Electric Power Research Institute (EPRI) and the "ad hoc" organization of PWR owners may also be required because of mutual interests in the safe operation of steam generators in general and in particular, the various problems associated with the operation of steam generators.

The purpose for interactions with these organizations is to exchange information on the research works sponsored by NRC and these outside organizations in identifying potential problems or solutions to existing problems associated with the operation of steam generators. Current programs in this area include an EPRI sponsored steam generator program in conjunction with Combustion Engineering. One aspect of this program is designed to define the mechanism of tube denting, and its goal is to provide corrosion-related information for improved steam generator coolant system technology and operation. The technology will be applied to the operation of plant systems and components that affect the reliability of steam generators. Additionally, EPRI had underway an ISI round robin test program for steam generator tubes to determine the effectiveness of various ISI techniques and methods for tube inspection.

7. Schedule for Problem Resolution

The major milestone for each program task are as follows:

Task 2A - EB/DOR, MEB & MTEB/DSS

. Review and evaluation of tube denting at W plants - June, 1979

. Monitor ISI results of PWR facilities with W steam generators - June 1979

Task 2B - EEB & RSB/DOR

. Review and evaluate the consequence of MSLB for plants relevant to determination of allowable leakage rate - June 1977

. Review and evaluate plant systems at PWR facilities with W steam generators to ensure comprehensive generic coverage as required-FY 1978, FY 1979.

Task 2C - EB/DOR, MEB/DSS

. Review and evaluate generic integrity analysis related to denting (1) short term operation - July 1977, (2) long term operation - Spring 1978

. Review RES sponsored program at PNL - June 1979

Task 2D - EB/DOR, MEB/DSS

. Recommendations for revision of Regulatory Guide 1.121 - September, 1979

Task 2E - EB/DOR, MEB & MTEB/DSS

- Evaluate CE/EPRI Model Boiler Studies - December 1978
- Review RES sponsored program at BNL - June 1979
- Evaluate other PWR vendors test programs for resolving tube denting problems - Summer 1979
- Establish water chemistry criteria - September 1979

Task 2F - EB/DOR, MTEB/DSS

- Review and evaluate Battelle Columbus program of eddy current inspection - October 1977.
- Review W activities in inspection techniques - November 1978 (targeted)
- Review EPRI Round Robin - November 1978

Task 2G - EB/DOR, MTEB/DSS

- Recommendations for revision of Regulatory Guide 1.83 - Winter 1979

Task 2H - EB, RSB & EEB/DOR

- Review and evaluate Surry plans for prototype steam generator replacement (starting Summer 1977).
- Review and evaluate Turkey Point plans for prototype steam generator replacement (starting Summer 1977).
- Establish generic NRC Steam Generator Replacement Position - January 1979

Task 2I - AB, MEB & MTEB/DSS

- Review and evaluate new design/ material concepts and new system component requirements for safe operation of steam generators in new PWR facilities.
- Develop analytical capabilities to determine the tolerable number of tube ruptures during transient events in new PWR facilities.
- Establish NRC Criteria for new PWR facilities.

8. Potential Problems

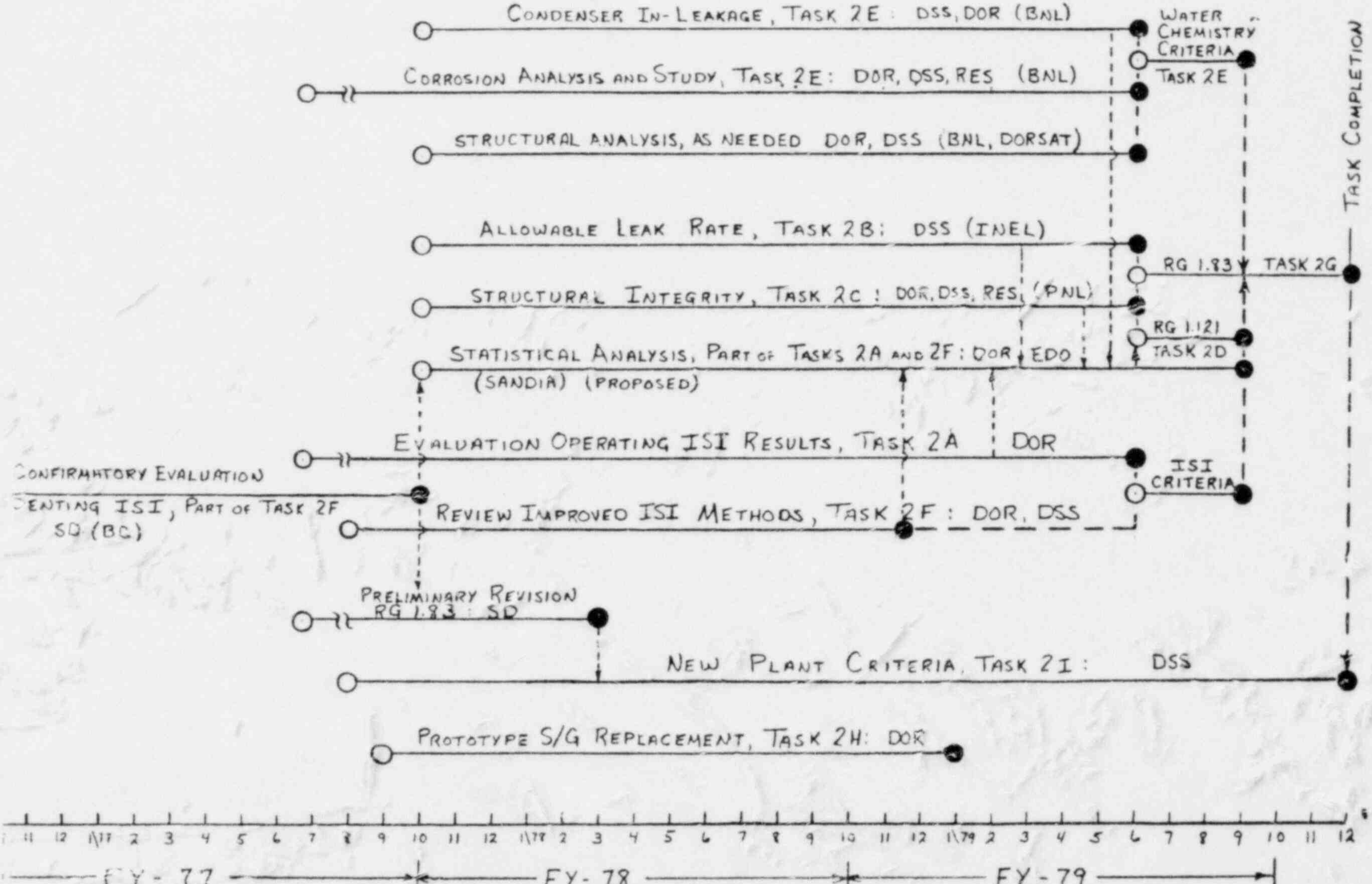
Except for steam generator replacement there is no apparent short term resolution of tube denting in affected Westinghouse plants. The many programs underway to resolve tube denting in presently operating plants may bring about a partial solution, by arresting denting through a cleaning program, sometime early in 1979.

However, by establishing quantitative plugging criteria for dented tubes, and requiring scheduled inspections, varying with the degree of denting observed, safety concerns can be minimized to the point where continued operation can be justified.

Finally, completion of many of the indicated tasks will depend on the scheduled completion of programs sponsored by organizations outside NRR. As with most experimental investigations, periodical delays can be expected, which may delay completion of some of the tasks indicated in the Task Action Plan.

WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY

TASK PHASE DIAGRAM



REVISION 0

CATEGORY A TECHNICAL ACTIVITY NO. A-4

OCT 3 1977

Title: Combustion Engineering Steam Generator Tube Integrity

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: Frank M. Almeter, Engineering Branch, DOR

1. Problem Description

Pressurized water reactor operating experience during the past five years has shown that steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. Since the steam generator tubes are an integrated part of the reactor coolant pressure boundary in the PWR system, the primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Palisades has been the only Combustion Engineering designed plant to experience tube degradation due to wastage and secondary side stress corrosion cracking with the use of a phosphate treatment for the secondary coolant. Both types of degradation have been nominally arrested by conversion to AVT chemistry control. However tube degradation due to denting (but to a lesser degree than the Westinghouse steam generators) occurred after the conversion to an AVT chemistry. Recent inservice inspections at two sea coast facilities with CE designed steam generators, which used an AVT chemistry for the secondary coolant since initial start up, have shown that the prior use of phosphates is not a necessary precursor to cause denting in steam generator tubing. Denting which leads to primary side stress corrosion cracking is the major problem at present, and the principal focus of this technical activity. However, as steam generator operating experience is accumulated and interpreted, it has become evident that condenser cooling water in-leakage resulting from the corrosion of condenser tubes can contaminate the secondary water of PWR steam generators and may be the principle source leading to all types of steam generator tube degradation. It has also become evident that the maintenance of secondary coolant water quality cannot be achieved if condenser in-leakage is allowed. Because the condenser is an important component of the PWR secondary system, an approach must be developed to minimize condenser in-leakage to ensure adequate steam generator tube integrity.

2. Plan for Problem Resolution

The problem will be resolved by reviewing the type and mechanism of tube degradation in operating reactors to evaluate the effects

APPROVED BY TASC, SEPTEMBER 6, 1977

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of tube structural integrity and failure probability under normal operation and accident conditions (LOCA & SSE and MSLB). Assessment of the effects of degraded tubes on postulated accident conditions will be factored into the development of new criteria for tube plugging, acceptable levels of primary to secondary leakage, and ISI requirements to ensure the safe operation of operating pressurized water reactors. To minimize tube degradation, priority areas where improvements in steam generator design and criteria for the secondary coolant system are needed will be identified to develop licensing positions for the CP/OL review of new plants.

The specific activities directed at resolution of the denting problem in Combustion steam generators consist of the following issues and tasks:

A. Generic Evaluation of ISI Results

Review and evaluate the various eddy current inspection results; i.e., experience from operating reactors and evaluate these data as they relate to the generic determination of failure probability of degraded tubes. In addition, evaluate the test programs and analytical studies to provide staff understanding sufficient to continue to provide justification for continued safe operation of operating reactors.

B. Evaluation of Transients and Postulated Accidents

Evaluation of failure consequences under postulated accident conditions (LOCA and MSLB) to determine the acceptable levels of primary to secondary leakage rates and the effect on ECCS performance. The results will be used to define the acceptable number of tube failures that may be necessary as a licensing basis considering predicted fuel behavior and radiological dose during transients and postulated accident conditions.

C. Evaluation of Steam Generator Tube Structural Integrity

Evaluation of licensees' and CE's analysis of structural integrity of tubes under normal operating and accident conditions (LOCA & SSE and MSLB). Information developed in this task will provide input for establishing a generic tube plugging criteria and recommendations for the revision of Regulatory Guide 1.121.

D. Establish Tube Plugging Criteria

Establish a generic tube plugging criteria that is consistent with the determined allowable leak rate, tube structural integrity and degradation rates. These results will allow assessment of the adequacy of the requirements defined in Regulatory Guide 1.121.

E. Secondary Coolant Chemistry Requirements

Evaluate the mechanism of tube degradation. The results will be used to define the requirements for secondary coolant chemistry control including considerations for condenser in-leakage.

F. Evaluation of ISI Methods

Review the development of improved eddy-current probes, coils and multi-frequency techniques to better quantify dents and growth of dents and increase sensitivity for detecting cracks in dented regions.

G. Establish Criteria for Revision of Regulatory Guide 1.83

Integrate experience from inservice inspection results, the results from the evaluation of various ISI improvements and the plugging and secondary water chemistry requirements into criterion for possible revision of Regulatory Guide 1.83.

H. Review Design Criteria for Plants Not Yet Licensed

Review and evaluate design modifications proposed by applicants and CE to prevent denting in plants not yet licensed for operation.

3. NRR Technical Organizations Involved

a. Engineering Branch, Division of Operating Reactors, has the primary lead responsibility for the overall review and evaluation of steam generator tube integrity in operating plants. This includes operational experiences, tube failure mechanisms and potential repairs, plugging criteria, ISI requirements, tube failure probability studies, leakage rate limits, and secondary coolant system control. This also includes the lead responsibility for determining the probability of LOCA and MSLB initiating events and the probability of tube failures during these events and responsibility for determing the number of tubes assumed to fail in LOCA and MSLB analyses.

Manpower Estimates: 0.1 manyears FY 1977, 0.5 manyears FY 1978,
0.5 manyears FY 1979

b. Environmental evaluation Branch, Division of Operating Reactors has the lead responsibility for the review and evaluation of the off site dosage related to the consequence or probability of a Main Steam Line Break (MSLB) accident or a LOCA should such evaluation become necessary. EEB will also consult with EB and provide support for the probabilistic evaluation of MSLB and LOCA initiating events and the probability of tube failures during these postulated events.

Manpower Estimates: 0.1 manyear FY 1977, 0.2 manyear FY 1978,
and 0.2 manyear FY 1979

c. Reactor Safety Branch, Division of Operating Reactors, has the lead responsibility for the review and evaluation of: (1) the ECCS performance related to secondary-to-primary leakage as a consequence of a LOCA, and (2) the effect of primary-to-secondary leakage during a MSLB accident on fuel failures should such evaluation prove necessary.

Manpower Estimates: 0.1 manyear FY 1977, 0.13 manyear FY 1978,
and 0.13 manyear FY 1979.

d. Mechanical Engineering Branch/Materials Engineering Branch, Division of Systems Safety, has responsibility in factoring all steam generator operating experience into the review of new design/material concepts and new system component requirements. This will apply to PWR facilities not yet licensed for operation.

The activities involved will include the review and evaluation of the applicant's and the NSSS's proposed improvements on the design and/or operation of the steam generators; for items such as secondary coolant chemistry, design modifications to avoid denting, ISI requirements, recommendations for revision of Regulatory Guides, condenser design to avoid in-leakage and provisions for access opening and space in the containment to facilitate steam generator inspections.

Manpower Estimates:	FY 1977	FY 1978	FY 1979
	0.1 manyear	0.5 manyear	0.5 manyear

e. Analysis Branch, Division of Systems Safety, has the lead responsibility in developing analytical capabilities (computer codes, etc.) to evaluate the effects of steam generator tube rupture(s) concurrent with various reactor transients that include MSLB and LOCA accidents. The purpose is to determine the equivalent number of tube failures that can be tolerated during transient events. This information will then be factored into the overall program of determining an adequate sample plan for tube inspections.

Manpower Estimates: 0.2 manyear FY 1978, 0.2 manyear FY 1979

f. Reactor Systems Branch, Division of Systems Safety has the responsibility of evaluating the design and performance of new associated auxiliary systems for CP/OL plants yet to be licensed, should any be required as the result of this technical activity; e.g., full flow condensate demineralization and etc. for PWR secondary Coolant. Manpower Estimates: 0.15 manyear FY 1979

4. Technical Assistance Requirements

a. Contactor: Brookhaven National Laboratory (BNL) - DOR, DSS

Funds Required: $98K FY 1977, $200K FY 1978 and $175K FT 1979

This effort is funded as part of an overall program at BNL applicable to the three Category A Technical Activities (A-3, A-4 and A-5) related to PWR steam generators. Funding values under DORSAT are not included.

This program is needed to obtain technical consultation and assistance to review information in areas of water chemistry and corrosion analysis, monitored jointly by EB/DOR and MTEB/DSS. Stress and/or burst strength calculations are funded in part under DORSAT contract on an as-needed basis. This program will provide assistance in accomplishing Tasks 2C, 2E and 2G.

b. Contactor: Idaho National Engineering Laboratory (INEL) - DSS

Funds Required: $75K FY 1977, $100K FY 1978.

This effort is generic in nature and will be applicable to the three Category A Technical Activities (A-3, A-4 and A-5) related to PWR steam generators.

The purpose of this program is to determine the effect of steam generator tube plugging on the predicted peak clad temperatures following a postulated LOCA. The primary activity is to produce a reliable computer code to aid the evaluation of the effects of tube plugging on the ECCS performance. An addition to the program will be needed to consider steam generator tube failures concurrent with MSLB or a LOCA. This program will provide assistance in accomplishing Tasks 2B and 2D.

c. Contactor: Sandia Laboratories, DOR proposed

Funds Required: $50K FY 1977, $100K FY 1978, and $150K FY 1979.

This work is of generic nature, and will be applicable to all PWR steam generators.

A program is needed for a statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. This is a new program to augment staff effort in steam generator safety reviews and will assist in addressing Tasks 2A, 2F and 2G.

5. Assistance Requirements from Other NRC Offices:

a. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Metallurgy and Materials Branch and Probabilistic Analysis Branch

RES has funded, at the request of NRR, a major confirmatory experimental program at Pacific Northwest Laboratory. The activity of this program consists of a series of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leakage rate data. This program is managed by Metallurgy and Materials Branch, (Task 2C).

RES has been requested to fund a new program, possibly starting this fiscal year, addressing the factors which determine Inconel 600 susceptibility to stress corrosion cracking in primary water. Metallurgical condition, chemistry, temperature, stress and environment will be considered, (Task 2E).

The Probabilistic Analysis Branch funded the program to assist EEB in probabilistic analyses, (Task 2B).

b. Office of Standards Development, Division of Engineering Standards, Structures and Components Standards Branch

- . OSD has funded a confirmatory research program at Battelle Columbus laboratory to evaluate eddy current methods for inspecting steam generator tubes as a subcontract to Brookhaven National Laboratory, (Part of Task 2F).

c. Office of the Executive Director For Operations, Applied Statistic Group.

- . Provide assistance to EB/DOR for statistical assessment of steam generator tube integrity, (Part of Tasks 2A, 2F, and 2G).

d. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the committee as the task progresses.

6. Interactions With Outside Organizations

a. Licensee(s) of Combustion Engineering Nuclear Facilities

At present all CE plants experiencing tube denting will be monitored to evaluate the progress of denting. Each licensee will submit an analysis of the consequences of tube denting on tube integrity and demonstrate that adequate safety margins exist for continued safe operation.

b. Combustion Engineering

The primary interactions with CE has been and continues to be related to their investigation program for the resolution of the tube denting problem at CE designed plants, and its generic implications, such as the licensing bases or justifications for continued operation of CE plants with known tube degradations. For interim periods of operation until the cause of tube denting is identified and corrective measure(s) implemented, this interaction will be needed to ensure that CE develops capabilities for the evaluation of ECCS performance for postulated accidents concurrent with tube failures, should such a licensing basis become necessary. In conjuction with licensees, CE will be requested to submit a test program and corrective action plan for Maine Yankee and Millstone Unit 2 and an analysis of the structural integrity of degraded tubes under normal operating and accident conditions (LOCA + SSE and MSLB).

In addition, CE will be requested to keep NRC informed of steam generator design changes and modifications in secondary water treatment systems to alleviate tube degradation in future CE plants This information will be incorporated into all Tasks of the program.

c. EPRI, PWR Owner Group etc.

Interactions with other organizations such as the Electric Power Research Institute (EPRI) and the "ad hoc" organization of PWR owners may also be required because of the mutual interests in the safe operation of steam generators in general and, in particular, the various problems associated with the operation of steam generators. Current programs sponsored by EPRI include the CE model boiler studies and the round robin program for ISI techniques.

7. Schedule for Problem Resolution

The major milestone for each program task are as follows:

Task 2A - EB/DOR, MEB & MTEB/DSS

- Review and evaluation of tube denting at Maine Yankee and Millstone Unit 2. October 1977
- Monitor and evaluate operating experience at other CE plants for generic application. September 1979

Task 2B - EEB & RSB/DOR

- Generic review of plant systems at PWR facilities with CE steam generators. September 1979
- Review and evaluate INEL studies December 1978

Task 2C - EB/DOR, MEB/DSS

. Review and evaluate generic integrity analysis related to denting (1) short term operation - July 1977, (2) long term operation - Spring 1978

. Review RES sponsored program at PNL - June 1979

Task 2D - EB/DOR, MEB/DSS

. Recommendations for revision of Regulatory Guide 1.121 - September, 1979

Task 2E - EB/DOR, MEB & MTEB/DSS

. Evaluate CE/EPRI Model Boiler Studies - December 1978

. Review RES sponsored program at BNL - June 1979

. Evaluate other PWR vendors test programs for resolving tube denting problems - Summer 1979

. Establish water chemistry criteria - September 1979

Task 2F - EB/DOR, MTEB/DSS

. Review and evaluate Battelle Columbus program of eddy current inspection - October 1977.

- Review CE activities in inspection techniques. - November 1978 (targeted)

. Review EPRI Round Robin - November 1978

Task 2G - EB/DOR, MTEB/DSS

- Recommendations for revision of Regulatory Guide 1.83 - Winter 1979

Task 2H - AB, MEB & MTEB/DSS

- Review and evaluate new design/ material concepts and new system component requirements for safe operation of steam generators in new PWR facilities.
- Develop analytical capabilities to determine the tolerable number of tube ruptures during transient events in new PWR facilities.
- Establish NRC Criteria for new PWR facilities.

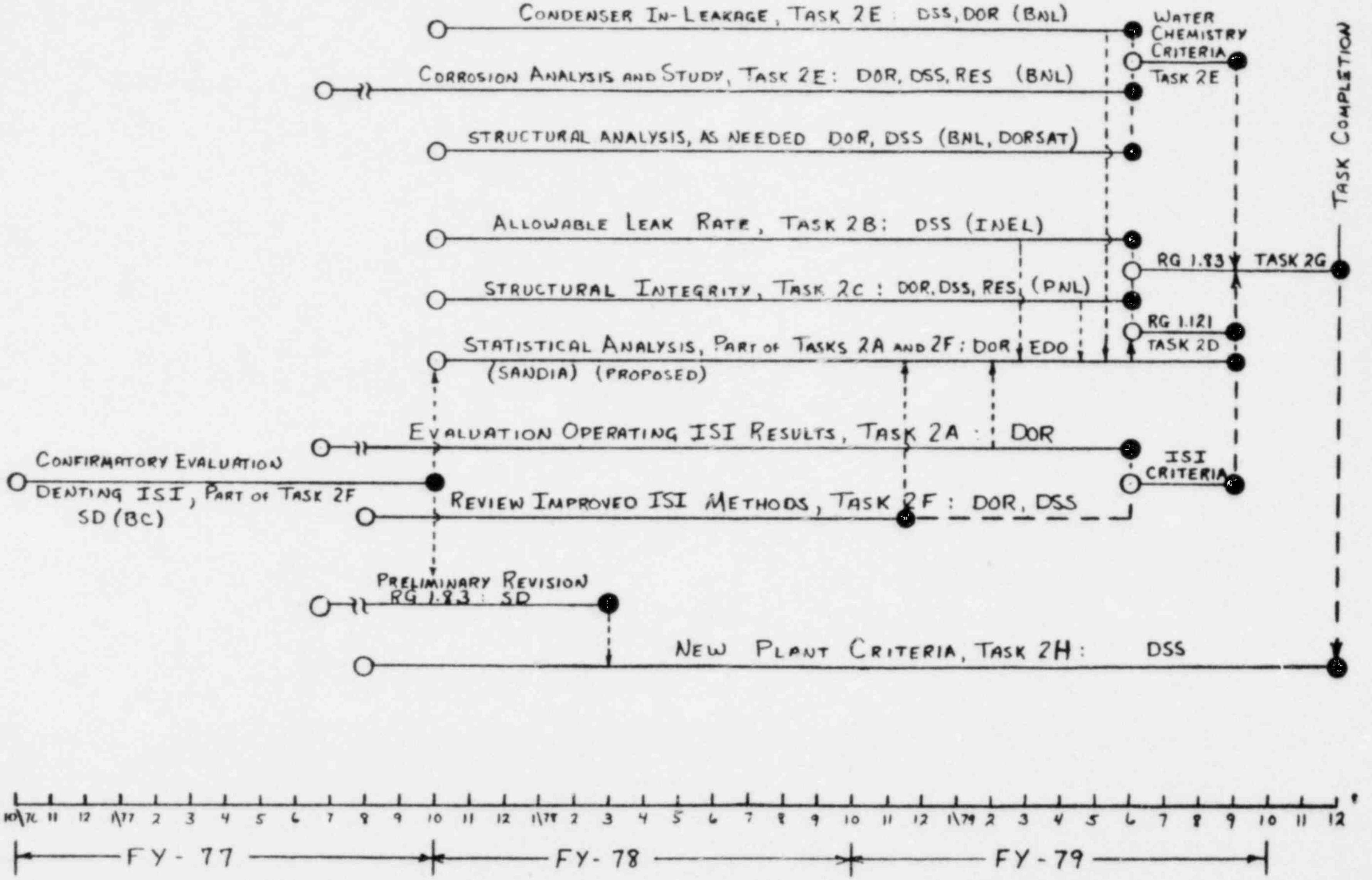
8. Potential Problems

It should be anticipated that required feedback from related programs funded by outside organizations may delay the timely completion of of certain subtasks. However, it is hoped that effective participation of NRC representatives at "ad hoc" organizational meetings will improve mutual interests in NRC goals. Any delays in submittals required by licensees and NSSS vendors would certainly delay the review and evaluation of tasks defined in the program. Timely input is required from all technical organizations involved.

Combustion Engineering STEAM GENERATOR TUBE INTEGRITY

OCT 3 1977

TASK PHASE DIAGRAM



REVISION 0
SEPTEMBER 21, 1977

CATEGORY A TECHNICAL ACTIVITY NO. A-5

Title: B&W Once-Through Steam Generator (OTSG) Tube Integrity

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: D. G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: B. D. Liaw, Division of Operating Reactors

1. Problem Description:

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting), and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

B&W's once-through steam generators (OTSG's) had been relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July, 1976. Since then, all three Oconee units have experienced tube leak incidents. In the last ten months, there have been a total of eight separate plant shutdowns due to leaks in four of the six steam generators in the three Oconee units. All leaking tubes and others with ECT indications were stablized and/or removed from service by plugging.

Laboratory examinations of a removed defective tube indicated that the tube failure was caused by the propagation of a crack, of unknown origin, in the circumferential direction due to flow induced vibration. B&W has been investigating the possible mechanisms that could initiate the crack. As part of the investigation program, Duke Power has conducted a series of turbine stop valve tests, with sensors mounted on steam generators and piping to measure pressures and velocities. The preliminary results indicated that the crack initiation, or propagation of existing crack, mechanism might be related to flow and/or pressure transients in the steam generators as a result of excessively frequent turbine stop valve testing.

2. Plan for Problem Resolution:

The major portion of the NRC staff efforts related to the resolution of the OSTG tube integrity problems will consist of review and evaluation of the results of B&W's investigation of the crack initiation and propagating mechanisms, Duke Power's turbine stop valve instrumented test program, and the inservice inspection (ECT) results of other B&W operating units. The following paragraphs describe briefly the background, the interim positions, and subtasks for the resolution of the problem:

APPROVED BY TASC, SEPTEMBER 6, 1977
TASC COMMENTS INCORPORATED SEPTEMBER 21, 1977

Background

During the interim period before B&W completes their investigation and implements any final fix or modification of the operating procedures, the NRC staff on March 17, 1977 specifically requested the licensee of the Oconee Units to address the following issues:

- Incorporate a primary to secondary leakage limit in the Technical Specifications, and the basis of the leakage limit to be proposed.
- Determine the consequences of failure of plugged, stabilized defective tubes at lower or unstabilized sections.
- Re-evaluate past ECT records for any tube defects that might have led to the initiation of tube cracking.
- Determine any change in operational procedures that may result in power transients and crack initiation.
- Perform ECT examinations of periphery tubes.
- Evaluate integrity of degraded tubes under normal operating and accident conditions.
- Perform tests to determine evidence of plastic cyclic straining.
- Develop capabilities to accurately assess accident consequences and the likelihood of LOCA and MSLB scenarios.
- Provide details of the B&W's programs for corrective actions.

On May 13, 1977 the licensee provided a partial response to address the issues stated above. The staff is continuing to review the information submitted and will need to most actively persue the evaluation of consequences of a postulated MSLB or LOCA to ensure that these are clearly understood should it become necessary to use this as part of the licensing bases. This is necessary because the tube degradations, either partially cracked or fatigued, are in the circumferential direction and are located in the upper, i.e., the super heat, region of the steam generators. Under a postulated MSLB condition, circumferentially degraded tubes could suffer complete severance because tubes in these regions are subjected to the maximum cross flow steam velocity. Under a LOCA condition, it can be postulated that tubes with this type of degradation could also suffer complete or partial severence under the LOCA shaking load in combination with the SSE loads.

Evidence obtained from examining the fracture surface of a defective tube removed from Oconee Unit 2 B-generator indicated that, from an initiation site of unknown origin, the crack propagated as a thru-wall defect due to the application of a high cycle fatigue loading. It was deduced that approximately 1 X 10^5 to 3 X 10^5 cycles were required for the crack to propagate its total observed length of roughly 240°. The only known source of loading which could involve this number of cycles is flow induced vibration. This would occur with most prominence in the fundamental mode of the tube, or at a frequency of about 40 Hz. From this information, it is apparent that an initial defect would propagate to a detectable leak in approximately 1 to 2 hours.

Interim Positions

Since the propagating mechanism is flow induced vibration, the defect is not "stable". It will rapidly progress around the circumference of the tube as long as there is flow of sufficient energy to drive it. However, the crack formed will produce an identifiable leak and the unit can be shut down promptly. Therefore, the probability of the occurrence of a major accident during the time between leak and shutdown is very low.

Pending a thorough evaluation of the effects of degraded tubes on the postulated accident conditions, the bases for permitting a limited period of operation of these steam generators is as follows:

(1) The low probabilities of these accidents to occur during (a) normal operation, and (b) the period between the detection of leak and the plant shutdown.

(2) The consequences of tube failures are acceptably small if these accidents were to occur during (a) normal operation, and (b) the period between the detection of leak and the plant shutdown.

In this context, the NRC staff requested on May 13, 1977 that the licensee develop the capabilities to assess the consequences and the probabilities of both the MSLB and the LOCA, concurrent with tube failures, as a basis of continued operation until the completion of the investigation and any subsequent fix, if appropriate. This information is scheduled to be provided to NRC during the first week of Sepetember 1977 along with a plan for the completion of the investigation of the crack initiating mechanism and the resolution of the problem.

Task Goals

The main goals of this task are to:

1. Establish a long-term licensing bases for continued operation of B&W units.
2. Formulate plant unique operational methods (related to ensuring steam generator tube integrity).
3. Establish tube integrity criteria and recommend revision of R.G. 1.121.
4. Establish primary to secondary leakage limits.
5. Establish improved ISI techniques and criteria and recommend revision of R.G. 1.83.

Subtask Descriptions

The specific activities directed toward achieving the above goals and resolving the problem(s) associated with B&W manufactured once-through steam generators consist of the following subtasks:

(1) Probabilities and Consequences of Accidents

Review and evaluate B&W's analyses and/or assessments of the probabilities and consequences of MSLB and LOCA concurrent with steam generator tube failure(s). The issue is to estimate the probability of having degraded tubes at incipient failure, and the number of such tubes that will fail during accident(s). Results of this subtask may be used to modify the licensing basis for continued operation of B&W units.

(2) Crack Initiation and Propagation Mechanisms - Plant Unique Operation

Review and evaluate B&W's on-going program for the investigation of the tube crack initiation and/or propagation mechanism(s). Results of this subtask may be used to formulate NRC's positions on the frequency and the power level for turbine stop valve testing, should it be confirmed to be the cause of the tube degradation.

(3) Primary to Secondary Leakage Limit

Review, evaluate, and approve B&W's proposal on the primary to secondary leakage limit and the basis, either analytical and/or experimental, for the limit in terms of crack size associated with it. Results of this subtask will be incorporated into Technical Specifications for B&W plants.

(4) Evaluation of ISI Methods

Review the development of improved eddy-current probes, coils and multi-frequency techniques, etc., for quantifying circumferential cracks.

(5) Evaluation of Steam Generator Tube Structural Integrity

Review and evaluate the structural integrity of degraded tubes under normal operating and accident (LOCA & MSLB) conditions including licensees' and B&W's analyses where appropriate to generic conclusions.

(6) Establishment of Tube Plugging Criteria

Based on experience from the continuing operation of B&W plants and results of Subtasks (4) and (5), a tube plugging criteria may be developed when appropriate or feasible. If tube plugging criteria are established, a recommendation for revision of Regulatory Guide 1.121 will be made as a result of this Subtask.

3. NRR Technical Organization Involved:

a. Engineering Branch (EB), Division of Operating Reactors. Has overall lead responsibility for review and evaluation of the information related to tube integrity. This includes the initiation and the propagation phases of cracks, leakage limit determination, stress and/or load calculations, and the instrumented turbine stop valve testing program. In addition, EB has the lead responsibility for determining the probability of tube failure during a postulated MSLB or LOCA.

Manpower Estimate: 0.15 manyear FY 1977, 0.5 manyear FY 1978, and 0.5 manyear FY 1979.

b. Reactor Safety Branch (RSB), Division of Operating Reactors. Has lead responsibility for the review and evaluation of: (1) the ECCS performance related to secondary-to-primary leakage as a consequence of a LOCA, and (2) the effect of primary-to-secondary leakage during a MSLB accident should such evaluations be needed for a licensing basis.

Manpower Estimate: 0.1 manyear FY 1977, 0.2 manyear FY 1978, and 0.2 manyear FY 1979.

c. Environmental Evaluation Branch (EEB), Division of Operating Reactors. Has the lead responsibility for review and evaluation of the consequences of the postulated accidents, related to the off site dosage aspect of the issue should such a licensing basis be used. In addition, EEB will consult with EB and provide analytical support for probability of tube failure during these postulated events.

Manpower Estimates: 0.10 manyear FY 1977, 0.13 manyear FY 1978, and 0.13 manyear 1979.

d. Mechanical Engineering Branch/Material Engineering Branch, Division of Systems Safety. Has the lead responsibility for monitoring this program as it progresses and for including operating experience into the review of new design/material concepts and new system operating requirements. This will apply to the review of B&W's designed facilities and proposed program to prevent tube leak occurrences for plants not yet licensed for operation. The technical activities involved may include the review of items such as the improvement made to prevent inleakage to steam generators, ISI requirements, recommendations for revision of Regulatory Guides, and provisions for access openings and space in the containment to facilitate steam generator inspections.

Manpower Estimates: 0.1 manyear FY 1977, 0.5 manyear FY 1978, 0.5 manyear FY 1979, and 0.5 manyear FY 1980.

e. Analysis Branch, Division of SystemsSafety. Has the lead responsibility for developing analytical capabilities (computer codes, etc.) to evaluate the effects of steam generator tube rupture(s) concurrent with various reactor transients that include MSLB and LOCA accidents sould such evaluations be needed as a licensing basis for B&W reactors. The purpose is to determine the equivalent number of tube failures that can be tolerated during transient events. This will then be used eventually to determine an adequate sample plan for tube inspection.

Manpower Estimates: 0.1 manyear FY 1977, 0.2 manyear FY 1978, and 0.2 manyear FY 1979.

4. Technical Assistance Requirements:

a. Contractor: Idaho National Engineering Laboratory (INEL) - DSS

Funds Required: $75K FY 1977, $100K FY 1978.

This effort is generic in nature and will be applicable to all three Category A Tasks for PWR steam generators.

The purpose of this program is to determine the effect of steam generator tube plugging on the predicted peak clad temperatures following a postulated LOCA. The primary activity is to produce a reliable computer code to aid the evaluation of the effects of tube plugging on the ECCS performance. An addition to the program will be needed to consider the concurrent steam generator tube failures and a MSLB or a LOCA.

b. Contractor: Sandia Laboratories, DOR proposed

Funds Requried: $50K FY 1977, $100K FY 1978, and $150K FY 1979.

This work is of generic nature, and will be applicable to all three Category A Tasks for PWR steam generators.

A program is needed for a statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. This is a new program to augment staff effort in steam generator safety reviews.

5. Assistance Requirements for Other NRC Offices:

a. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Metallurgy and Materials Branch and Probabilistic Analysis Branch

RES has funded, at the request of NRR, a major confirmatory experimental program at Pacific Northwest Laboratory. The activity of this program consists of a series of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leakage rate data. This program is managed by Metallurgy and Materials Branch. This program is related to, but may not be directly applicable to, this Task.

The Probabilistic Analysis Branch funded the program to assist EEB in its probabilistic analysis of tube failures concurrent with postulated accidents (i.e., MSLB, LOCA).

b. Office of Inspection and Enforcement

Assistance from I&E may be required to verify turbine stop valve test procedures at operating B&W facilities, should it become obvious that the frequency of turbine stop valve testing has contributed to the initiation of cracks in the steam generator tubes.

c. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

6. Interactions with Outside Organizations:

   a. Licensee(s) of B&W Nuclear Facilities

      At present, the three Oconee units owned by Duke Power Company are the only B&W plants that have experienced tube leaks. Interactions with other licensees of B&W plants than Duke Power will be limited to the discussions on the steam generator tube inspection results.

   b. Babcock & Wilcox (B&W)

      The primary interaction with B&W has been and continues to be related to the investigation program for the resolution of the problem at Oconee Station and its generic implication, such as the licensing bases or justifications for continued operation of B&W plants with known tube degradations. For interim periods of operation, before the cause of crack initiation is identified and corrective measure(s) implemented, interaction will be needed to ensure that B&W develops capabilities for the evaluation of ECCS performance under postulated accidents concurrent with tube failures.

   c. EPRI, PWR Owner Group, etc.

      Interactions with other organizations such as the Electric Power Research Institute (EPRI) and the "ad hoc" organization of PWR owners may also be required because of mutual interests in the safe operation of steam generators in general and, in particular, in the various problems associated with the operation of steam generators.

      The purpose for interactions with these organizations is to exchange information on the research works sponsored by NRC and these outside organizations in identifying potential problems or solutions to existing problems associated with the operation of steam generators.

7. Schedule for Problem Resolution:

   The major milestones for the resolution of the problem are identified and projected as follows:

   Subtaks 2.1 - EB, EEB & RSB/DOR

   An assessment of the probabilities and consequences of accidents (LOCA & MSLB) concurrent with tube failures submitted by the licensee - August 1, 1977

Review and evaluation of the above-mentioned analyses - November 1, 1977 (targeted)

Review and evaluation of INEL's program on the effects of tube plugging and tube failures concurrent with accidents - October, 1978

Complete study the consequences of a MSLB accident (in-house) - March, 1978

Review the results of Sandia's probabilistic analysis of tube failures - September, 1979 (targeted)

Subtask 2.2 - EB/DOR

Review licensee's and B&W's program for the investigation of the crack initiation and/or propagating mechanism(s) - November 1, 1977 (targeted)

Review and evaluate results of B&W's investigation program and any proposed modification of hardware and/or operational procedures for preventing future occurrences as appropriate - June, 1978 (targeted)

Review and evaluate the results of the instrumented turbine stop valve tests at Oconee Units, including a determination of their generic implication - March, 1978 (targeted)

Subtask 2.3 - EB & EEB/DOR, MEB/DSS

Review and evaluate, in a generic manner, the licensee's proposal on the primary to secondary leakage limit, and the basis associated with the proposed limit-December, 1977 (targeted)

Propose and/or incorporate changes into the Technical Specifications for primary to secondary leakage limits - February, 1978 (targeted)

Subtask 2.4 - EB/DOR, MTEB/DSS

Review and evaluate Batelle Columbus program of eddy current inspection - January 1979

Review and evaluate B&W's activities related to improvements in steam generator tube inspection techniques - November, 1978 (targeted)

- Review EPRI Round Robin - November 1978

Subtask 2.5 - EB/DOR, MEB/DSS

- Where appropriate and when necessary, review and evaluate B&W's analyses and/or tests on the structural integrity of degraded tubes and determine their generic implication - June, 1978

Subtask 2.6 - EB/DOR, MEB/DSS

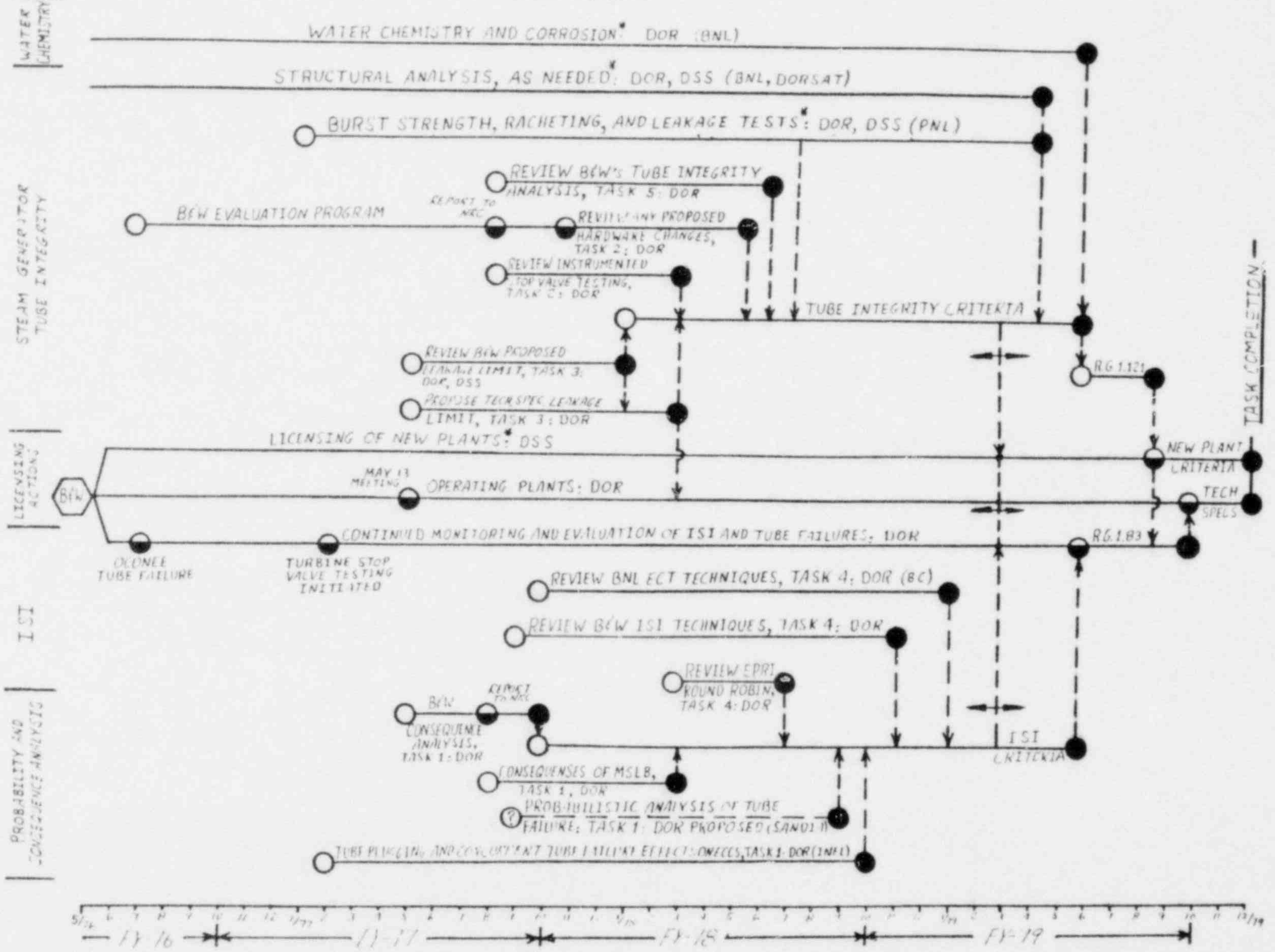
- Recommendations for revision of Regulatory Guide 1.121, where appropriate - September, 1979 (targeted)

8. Potential Problems:

Although progress has been made, B&W has not been able to pinpoint the cause(s) of the crack initiation as of May 13, 1977. To date, the tube leaking has only occurred in the Oconee generators. Should other B&W units (e.g., Arkansas Unit 1 and Three Mile Island) experience similar tube leaking occurrences as observed at Oconee, the scope of investigation may have to be expanded considerably.

B&W STEAM GENERATOR TUBE INTEGRITY

TASK PHASE DIAGRAM



* WORKS MAY BE RELATED, BUT CONSIDERED NOT DIRECTLY APPLICABLE TO THE RESOLUTION OF B&W STEAM GENERATOR PROBLEM(S), AS DESCRIBED IN

REVISION 0

CATEGORY A TECHNICAL ACTIVITY NO. A-6

Title: Mark I Containment Short Term Program (STP) AUG 3 1977

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: John Guibert, DOR

1. Problem Description:

During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated loss of coolant accident (LOCA) were identified which had not been explicitly included in the original design of the Mark I containment systems. These additional loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA event.

Consequently, it was determined that a reassessment of the Mark I containment system design would be required. This reassessment is being conducted in two phases, (1) a short-term program (STP) designed to confirm the adequacy of the containment system of each operating Mark I BWR facility to maintain its integrity and functional capability during a postulated LOCA event, and (2) a long-term program (LTP) designed to establish design basis loads appropriate for the intended life of each Mark I BWR facility and to restore the originally intended design safety margins for each Mark I containment system.

The primary objective of the Mark I containment STP is to verify that licensed Mark I BWR facilities may continue to operate safely, without undue risk to the health and safety of the public, while a methodical, comprehensive (LTP) is conducted. This short-term program evaluation has been conducted using a "most probable load" approach; the aim of this approach was to identify the load magnitudes and load combinations most likely to be encountered during the course of a postulated design basis LOCA. The STP structural acceptance criteria assure that, for the most probable loads induced by a postulated design basis LOCA, a safety factor to failure of at least two exists for the weakest structural or mechanical component in the containment system for each operating Mark I BWR facility.

2. Plan for Problem Resolution:

The major portion of the NRC staff efforts related to the STP have consisted of review and evaluation of the results of the analytical and testing programs conducted by the Mark I Owner's Group and by licensees of Mark I BWR facilities.

APPROVED BY TASC, AUGUST 19, 1977

Key elements of the short term program include: (1) an STP final report, including addenda, which has served as the STP load definition report and as the structural evaluation report for containment system components and elements other than the torus support system; (2) a one-twelth scale testing program, which was utilized to develop STP loading conditions on torus support systems; (3) drywell to torus differential pressure control procedures to mitigate postulated LOCA loadings on torus support systems; (4) structural acceptance criteria which were developed to assess the results of the plant unique analyses of torus support systems; and (5) a plant unique analysis of the torus support system of each operating Mark I BWR facility.

In addition to the above, the STP review by the staff has included an evaluation of the LTP program objectives proposed by the Mark I Owner's Group to assure that it is reasonably designed to provide resolution of issues raised during the STP and to meet the objectives of the LTP.

The Mark I Containment STP will be complete following issuance of a generic Mark I Containment STP safety evaluation report by the NRC staff and incorporation of technical specification requirements to assure that facility operation remains within the initial conditions assumed in the plant unique analyses.

3. NRR Technical Organizations Involved:

   a. Containment Systems Branch, Division of Systems Safety: Has had overall lead responsibility for STP load definition and has had lead responsibility for review and approval of upward and downward torus pressure loads.

      Manpower estimate: All work has been completed.

   b. Plant Systems Branch, Division of Operating Reactors: Has had lead responsibility for review and approval of all loading conditions other than upward and downward torus pressure loads (e.g., vertical reaction loads, drag loads on submerged components).

      Manpower estimate: All work has been completed.

   c. Engineering Branch, Division of Operating Reactors: Has had lead responsibility for the review and approval of the Mark I STP structural acceptance criteria, and has had lead responsibility for review and approval of all STP structural analyses, including plant-unique analyses of torus support systems.

      Manpower estimate: All work has been completed.

4. Technical Assistance Requirements:

No additional technical assistance work is required to complete the Short Term Program.

5. Interactions with Outside Organizations:

a. Mark I Owner's Group

The Mark I Owner's Group is an "ad hoc" organization of all utilities owning Mark I BWR facilities. They have engaged General Electric Company as their program manager for resolution of the Mark I Containment concerns and have designated General Electric as their primary contact with the NRC during the conduct of the STP and the LTP. Teledyne, Bechtel, and NUTECH have been engaged as the primary consultants to the Mark I Owner's Group. The majority of the technical exchanges with the NRC staff during the STP have been made by representatives of the above-mentioned organizations.

b. Individual licensees of Mark I BWR facilities

In addition to its participation as a member of the Mark Owner's Group, each licensee of a Mark I BWR facility has been involved in the primary correspondence with the NRC during the conduct of the STP. Each licensee was required to submit a plant unique analysis of the torus support system for his facility. In addition, each licensee was required to submit Technical Specification requirements which provide assurance that facility operation will remain within the conditions assumed in the plant unique analysis.

c. Brookhaven National Laboratory (BNL)

Representatives of BNL have served as consultants to the NRC staff during the conduct of the STP.

6. Assistance Requirements from Other NRR Offices: No assistance has been required during the STP.

7. Schedule for Problem Resolution:

The remaining major milestone for the Mark I Containment Short Term Program is the issuance of the Short Term Program Safety Evaluation Report - August 1977 (targeted).

8. Potential Problems:

No problems are anticipated.

REVISION 0

CATEGORY A TECHNICAL ACTIVITY TASK NO. A-7

AUG 29 1977

Title: Mark I Containment Long Term Program (LTP)

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: John Guibert, DOR

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 29, 1977

1. Problem Description:

During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated loss of coolant accident (LOCA) were identified which had not been explicitly included in the original design of the Mark I containment systems. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA event. In addition, recent experience at operating plants has indicated that the dynamic effects of safety-relief valve (SRV) discharges to the suppression pool could be substantial and should be reconsidered.

The results of the Mark I containment short-term program (STP) have provided assurance that the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA. However, the STP evaluation was conducted using a "most probable load" approach which was aimed at the identification of load magnitudes and load combinations which were most likely to be encountered during the course of a postulated design basis LOCA. In addition, the STP structural acceptance criteria were selected to assure that, for the most probable loads induced by a postulated design basis LOCA, a safety factor to failure of at least two existed for the weakest structural or mechanical component in the containment system for each operating Mark I BWR facility.

Consequently, since the design margin of safety for the containment systems of operating Mark I facilities has been reduced from the margin believed to be present at the time these facilities were originally reviewed and licensed, the need exists (1) to establish design basis LOCA loads which are appropriate for the life of the facility, and (2) to restore the originally-intended design safety margins for the containment systems. For those Mark I BWR facilities not yet licensed for operation, the need exists (1) to establish design basis LOCA loads which are appropriate for the life of the facility, and (2) to ensure that adequate design safety margin has been provided in the design of the containment system prior to issuance of an operating license.

In the event that the LTP evaluation results are not available before the issuance of an operating license for a Mark I BWR facility not yet licensed for operation, the utilization of "interim" loading requirements and/or "interim" structural acceptance criteria more conservative than those which were established for the STP evaluation will be considered on a case-by-case basis. These considerations will include value-impact assessments related to the timing (i.e., before or after initial reactor operation) for the implementation of necessary structural modifications, if any. However, in such cases, the containment system structural and mechanical elements will be subject to reanalysis when the LTP loading requirements and structural acceptance criteria become available.

2. Plan for Problem Resolution:

The major portion of the NRC staff's efforts related to the resolution of the Mark I Containment LTP concerns will consist of review and evaluation of the results of the Mark I Containment LTP which is being conducted by the Mark I Owner's Group. As documented in Revision 1 to the "Mark I Containment Program Action Plan" which was submitted to the NRC on February 11, 1977, the Mark I Owner's Group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

Key elements of the Mark I Owner's LTP are: (1) the submittal of a load definition report (LDR), which will contain design basis hydrodynamic pressure suppression loads and their possible combinations, and proper procedures as how to apply them for structural evaluation, and (2) the development of structural acceptance criteria, which will be used to assess the structural capability of each Mark I containment system to withstand the design basis loads.

The NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owners Group prior to the conduct of plant-unique structural evaluations. The results of this evaluation will be documented in a generic Safety Evaluation Report. Publication of this report will constitute the resolution of this Technical Activity.

Implementation of the results of this generic review, although not a part of this task, will be accomplished by an NRC requirement that each affected utility perform a plant-unique structural evaluation of the containment system for their facility using the loads, loading combinations, and structural acceptance criteria approved by the NRC staff.

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The NRC has initiated several confirmatory research programs related to the Mark I LTP. These programs, which are discussed in Section 4 below, are designed to provide the NRC staff with an independent source of information to evaluate the results of the Mark I Owner's program and to assist in providing a basis for regulatory decisions regarding the adequacy of the Mark I containment systems.

The Mark I Owner's LTP commenced in June 1976 with the in-plant SRV testing at Monticello and is currently scheduled for completion in 1979.

3. NRR Technical Organizations Involved:

   a. Plant Systems Branch, Division of Operating Reactors: Has overall lead responsibility for design basis load definition for the Mark I containment system and has lead responsibility for the review and approval of LOCA-related hydrodynamic loads for the Mark I BWR facilities.

      Manpower Estimates: .2 manyear remaining FY 1977, one manyear FY 1978, one manyear FY 1979.

   b. Containment Systems Branch, Division of Systems Safety: Has lead responsibility for review and approval of SRV-related hydrodynamic loads* and has responsibility for establishing, as appropriate, "interim" loading requirements for the purpose of issuing an operating license for a facility prior to the availability of the LTP LDR. At the present time, it is intended that the STP loads will be used as the "interim" loading requirements. ("Interim" loads will be subject to confirmation by LTP results.)

      Manpower Estimate: .2 manyear, remaining FY 1977, one manyear FY 1978, .3 manyear FY 1979.

   c. Engineering Branch, Division of Operating Reactors: Has lead responsibility for the review and approval of structural acceptance criteria for use in the LTP evaluation on all Mark I BWR containment systems.

* It should be noted that a separate Category "A" technical activity for "Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments" is currently under consideration by the Technical Activities Steering Committee. If such an activity is approved, it will be carefully coordinated with this activity.

Manpower Estimate: .3 manyears, remaining FY 1977, two manyears FY 1978, one manyear FY 1979.

d. Structural Engineering Branch/Mechanical Engineering Branch, Division of Systems Safety: Has responsibility to assist EB/DOR in the review and approval of the LTP structural acceptance criteria.

Manpower Estimate: .1 manyear remaining FY 1977, .5 manyear FY 1978, .1 manyear FY 1979.

e. Office of Assistant Director for Operational Technology, Division of Operating Reactors: The functions of the Task Manager will be provided by a member of this organization.

Manpower Estimate: .05 manyear remaining FY 1977, .3 manyear FY 1978, .1 manyear FY 1979.

4. Technical Assistance Requirements:

a. Contractor: Lawrence Livermore Laboratory
Funds Required: $100K FY 1977, $15K FY 1978

This is a program to study hydrodynamic/structural interactions in a Mark I containment system subject to hydrodynamic loading conditions. This effort should quantify the amplification, if any, of measured loads due to the structural interactions during pool swell, SRV discharge, and chugging loading conditions. Engineering Branch, DOR has responsibility for the management of this program and for the application of the program's results in the staff's review of the LTP.

b. Contractor: Lawrence Livermore Laboratory
Funds Required: $55K FY 1977, $25K FY 1978.

The purpose of this program is to assess the safety margins in containment structures subjected to rapidly applied dynamic loads. This program will provide information useful in the development of the LTP structural acceptance criteria and in NRC staff's review of the plant-unique analyses submitted by each affected utility. Engineering Branch, DOR, has responsibility for the management of this program and for the application of the program's results in the staff's review of the LTP.

c. Contractor: Brookhaven National Laboratory (BNL)
Funds Required: $25K FY 1977, $120K FY 1978, $25K FY 1979.

The purpose of this program is to obtain expert technical assistance in the review of the results of the Mark I Owner's LTP testing and analytical efforts related to hydrodynamic load definition. (Due to BNL's combined workload [Mark I, II and

III], the possibility exists that the NRC may have to obtain the services of an additional contractor to perform these functions.) Plant Systems Branch, DOR, has responsibility for the management of this program as it relates to the definition of LOCA-related hydrodynamic loads. Containment Systems Branch, DSS, has responsibility for the management of this program as it relates to the definition of safety-relief valve related hydrodynamic loads.

5. Interactions with Outside Organizations:

   a. Mark I Owner's Group

   The Mark I Owner's Group is an "ad hoc" organization of all utilities owning Mark I BWR facilities. They have engaged General Electric Company as their program manager for resolution of the Mark I containment concerns and have designated General Electric as their primary contact with the NRC during the conduct of this program. Teledyne, Bechtel and NUTECH have been engaged as the primary consultants to the Mark I Owner's Group. The majority of the technical exchanges with the NRC staff during the LTP will be made by representatives of the above-mentioned organizations.

   b. Individual licensee of Mark I BWR facilities.

   In addition to its participation as a member of the Mark I Owners Group, each licensee of a Mark I BWR facility is involved in the primary correspondence during the conduct of the Long Term Program.

   c. Advisory Committee on Reactor Safeguards

   This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

6. Assistance Requirements from Other NRC Offices:

   a. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Analysis Development Branch.

   RES has funded, at the request of NRR, a major confirmatory experimental research program at Lawrence Livermore Laboratory. The program involves the construction and operation of a 1/5 scale, 90° sector of a typical Mark I BWR containment system. The purpose of this program is to obtain data regarding the magnitude and character of hydrodynamic LOCA-related loads on the Mark I containment system in order to confirm the results obtained from the testing programs sponsored by the Mark I Owner's Group. RES and its contractor are responsible for interpretation of the data developed from this testing program. Plant Systems Branch, DOR, has responsibility for application of the results of this program in the Mark I Containment LTP review.

b. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Analysis Development Branch

RES has sponsored two additional research programs of possible applicability in the Mark I Containment Long Term Program:

(1) A program is currently underway at MIT to investigate the scaling relationships for hydrodynamic phenomena due to air discharge.

(2) A similar program is underway at UCLA to investigate scaling relationships for steam discharges.

7. Schedule for Problem Resolution

The major milestones for the conduct of the Mark I Owner's Long Term Program are as follows:

1. Submittal of the Long Term Program Action Plan - February 11, 1977 (complete).
2. Submittal of the LTP Load Definition Report - August 1978 (targeted).
3. Submittal of proposed LTP Structural Acceptance Criteria - October 1978 (targeted).
4. Issuance of a generic LTP Safety Evaluation Report by the NRC Staff - February 1979.

The NRC staff will contintually monitor the progress of the LTP to assure that its intended objectives are met in a timely manner.

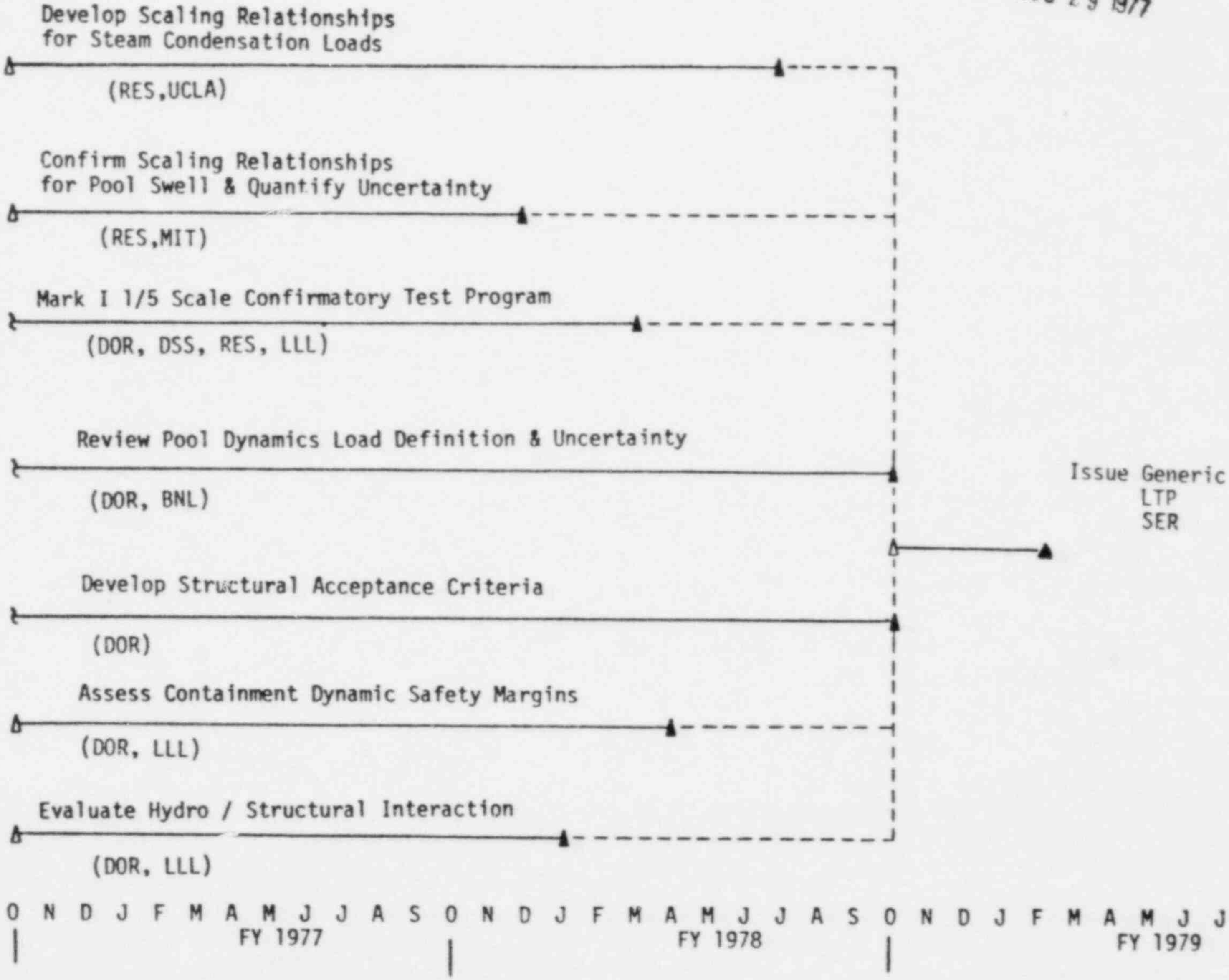
Attachment 1 is a phasing diagram which illustrates the integration of the schedules for completion of the NRC-sponsored LTP activities.

8. Potential Problems:

The schedule for completion of the LTP Load Definition Report by the Mark I Owners Group is dependent on the timely and successful completion of research and development efforts by industry. Although delays in the completion of these programs are not currently anticipated, the schedule for resolution of this generic task would be affected should such delays occur.

ATTACHMENT 1

AUG 29 1977



REVISION 0
SEPTEMBER 14, 1977

TASK ACTION PLAN
TASK NUMBER A-8

Title - Mark II Containment Pool Dynamic Loads

Lead Responsibility - Division of Systems Safety

Lead Assistant Director - Robert L. Tedesco (Plant Systems)

Task Manager - C. J. Anderson (Containment Systems Branch)

1. Problem Description:

As a result of the ongoing GE testing program for the Mark III pressure suppression containment program, new containment loads associated with a postulated loss of coolant accident (LOCA) were identified in 1975 which had not been explicitly included in the original design of Mark I and II containments. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. Other previously unaccounted for pool dynamic loads result from the actuation of safety relief valves (SRV) in the Mark II containment. The review and evaluation of the Mark I loads is assigned to Task A-7. Task A-39 will cover the loads for SRV valves for all pressure suppression type containments.

In view of the potential significance of these loads, it was determined that a reassessment of the Mark II containment system design would be required. A letter was then sent to each Mark II owner on April 11, 1975 notifying them of the need for this reassessment. Like the Mark I program, this reassessment is being conducted in two phases, (1) a short-term program (STP) designed to provide conservative pool dynamic loads, load combinations and design criteria that will be used in the

APPROVED BY TASC, SEPTEMBER 6, 1977
TASC COMMENTS INCORPORATED,
SEPTEMBER 14, 1977

licensing evaluation of the lead Mark II plants, and (2)a long-term program (LTP) designed to provide confirmation of specific pool dynamic loads and to provide an evaluation of new information related to these loads to determine appropriate loads to be considered in the licensing evaluation of later Mark II plants.

The pool dynamic loads, load combinations, and design criteria developed in this generic program form the bases for the plant unique evaluation of Mark II pool dynamic loads. The resulting plant unique evaluation using these generic dynamic loads for each plant is outside of this task and will be performed as a part of the Operating License review for each plant.

2. Plan for Problem Resolution:

A. Approach

As a result of our April, 1975 letter to each Mark II owner requiring a reassessment of their containment system considering pool dynamic loads, an "ad hoc" Mark II owner's group was formed which is an organization of all domestic utilities owning Mark II BWR facilities. They have engaged General Electric as their program manager for resolution of the Mark II containment pool dynamic concerns and have designated General Electric as their primary contact with the NRC during the conduct of this program. Sargent and Lundy and Bechtel have been engaged as the primary consultants to the Mark II owners group. The majority of the technical exchanges with the staff during the STP and LTP will be made with representatives of the above-mentioned organizations.

The Mark II owner's group developed the two phase STP and LTP approach to establish generic pool dynamic loads, load combinations and design criteria. The approach taken by the owner's group in the STP and LTP consists of a comprehensive experimental and analytical program to justify the Mark II pool dynamic design loads specified in the Mark II Dynamic Forcing Function Information Report (DFFR) - NEDO-21061. A preliminary list of the experimental and analytical programs included in the total program is provided in Attachment 1.A,B.

The major elements of the Mark II STP are listed in a Mark II lead plant topical report to be submitted August, 1977, A description of the LTP is to be submitted in a revision to NEDO-21297, "Mark II Containment Supporting Program Report" scheduled to be submitted in September, 1977.

The major portion of the staff's efforts relate to the review and evaluation of the results of the Mark II containment STP and LTP; however, we are also reviewing related foreign and domestic experimental programs. In addition, the NRC has initiated several confirmatory programs applicable to pressure suppression containments. The related foreign and domestic programs and the NRC confirmatory programs are discussed in Sections 5 and 6.

B. End Products

The end products of the Mark II owner's program consist of a number of topical reports which will contain the experimental and analytical bases to support design pool dynamic loads, load combinations and acceptance criteria. A preliminary list of these reports is provided in Attachment 1.A,B.

As a part of our review and evaluation program, we intend to issue two reports dealing with the preliminary Mark II safety evaluation report and the final Mark II confirmatory safety evaluation report.

Our first report will be issued at the completion of the Mark II STP and will contain an evaluation of the acceptability of the DFFR information for use in the plant unique analyses of the individual Mark II plants.

The Staff's second report will be issued at the completion of the Mark II LTP and will be a revision of the first safety evaluation report issued at the conclusion of the STP. In addition to the information contained in our first report, this report will include an evaluation of the LTP confirmatory experimental and analytical programs to assess the margin for selected loads. If reduced design loads are proposed compared with those in the DFFR, based on new information obtained in the LTP, an evaluation of the acceptability of the revised loads for use in the plant unique analyses of Mark II plants will be included in this report.

We anticipate initial licensing of a few lead plants will not include all information that will be developed from either the STP or the LTP. In these instances, licensing actions will be taken on a conservative basis utilizing available margins in the structural capability

C. Tasks

1. Pool Dynamic LOCA Loads

The primary purpose of this task is to review and evaluate the LOCA-related pool dynamic loads for Mark II containment systems. These loads are specified in Section 4.0 of the DFFR. The DFFR includes the original report submitted in September, 1975 along with a number of revisions, errata and amendments. In addition, several applications memos have been submitted by the Mark II owner's group describing pool dynamic loads that are to be incorporated in a future revision to the DFFR. This revision is to be submitted prior to completion of the STP. The pool dynamic loads specified in the DFFR include a combination of load models and specific loads that are to be applied directly to Mark II containment systems.

The review and evaluation of the DFFR loads includes the technical basis for these loads. It consists of a combination of experimental and analytical programs. A summary of the applicable supporting programs is provided in Attachment 1. This summary is to be revised in October, 1977 as additional information is supplied by the Mark II owners group.

It should be noted that in addition to the LOCA related pool dynamic loads the DFFR specifies methods for the prediction of the Mark II Safety Relief Valve (SRV) related pool dynamic loads. The review of the Mark II SRV loads is not a part of this task, but is being reviewed as a subtask of Task No. A-39, "Determination of Safety Relief Valve Pool Dynamic Loads." The method for prediction of submerged structure drag loads is common to both LOCA and SRV pool dynamic loads, and is also being reviewed as a subtask of Task A-39.

1.a Pool Swell

Review and evaluate the pool swell loads described in Section 4.4 of the DFFR including the supporting programs for these loads as shown in Attachment 1. The pool swell review items to be included in this subtask include: impact, drag, diaphragm and froth impingement loads; the pool swell model; and the maximum pool swell height criterion.

1.b Downcomer

Review and evaluate the downcomer loads described in Sections 4.2 and 4.3 of the DFFR including the supporting program for these loads as shown in Attachment 1. The review items associated with downcomer loads to be included in this subtask include: vent clearing loads, vertical loads, thrust loads, high mass flux condensation loads, medium mass flux condensation loads, and chugging loads.

1.c Pool Boundary

Review and evaluate the pool boundary loads that occur during pool swell as described in Section 4.4 of the DFFR including the supporting program for these loads as shown in Attachment 1. The review items included with the pool swell related pool boundary loads in this subtask include; vent clearing jet loads, air bubble loads and pool fall-back loads.

1.d Condensation

Review and evaluate the condensation loads on the pool boundary as described in Section 4.2 and 4.3 of the DFFR including the supporting program for these loads as shown in Attachment 1. The review items associated with the pool boundary condensation loads in this subtask include: high mass flux condensation loads, medium mass flux condensation loads and chugging loads.

1.e Safety Relief Valve Actuations

Review and evaluate the safety relief valve actuation methods as described in Section 5.3 of the DFFR. The review items associated with this subtask include: method for prediction of total number of SRV actuations; design considerations for performing fatigue analyses on structures; and determination of thermal cycles associated with SRV actuations. (These review items are outside the scope of Task A-39).

2. Load Combinations

The primary purpose of this task is to review and evaluate the load combination and load combination histories to be used for the design assessment of Mark II containment systems. The load combinations and load combination histories are provided in Section 5.0 and 6.0 of the DFFR report submitted in September, 1975 along with its revisions and amendments. Supplementary information to justify the DFFR load combinations and histories will be provided in August, 1977. A summary of the applicable supplementary information is listed in Attachment 1. This summary is to be revised in October, 1977 as additional information is made available by the Mark II owner's group.

2.a Containment and Containment Structures

Review and evaluate the load combinations and load combination histories described in Section 5.0 of the DFFR along with the supporting information listed in Attachment 1 for the design assessment of the Mark II containment and the containment structures.

2.b Piping and Components

Review and evaluate the load combinations and load combination histories as described in Section 6.0.of the DFFR along with the supporting information listed in Attachment 1 for the design assessment of piping and components in the Mark II containment.

3. Design Criteria

The primary purpose of this task is to review and evaluate the acceptance criteria for Mark II containment systems as found in Sections 5.0 and 6.0 of the DFFR.

3.a Containment and Containment Structures

Review and evaluate the acceptance criteria for Mark II containment and containment structures as described in Section 5.0 of the DFFR.

3.b Piping and Components

Review and evaluate the acceptance criteria for Mark II piping and components as described in Section 6.0 of the DFFR.

4. Plant Fluid-Structure Interaction

The purpose of this task is to review and evaluate the generic methods used in the analyses of fluid-structure interactions for chugging and SRV loads. This is to be done for steel, prestressed concrete and reinforced concrete Mark II containment designs. The methods will be described in future reports as shown in Attachment 1.

5. Safety Evaluation Report

Develop Safety Evaluation Report and LTP based on the results of tasks 1 through 4.

5.a Preliminary Safety Evaluation Report

Prepare a preliminary safety evaluation report of the Mark II DFFR at the end of the STP based on interim results of Tasks 1 through 4 of this program and tasks 1, 2 and 3 of the Task A-39 program. This report will contain our evaluation including bases of the Mark II pool dynamic loads, load combinations and acceptance criteria specified in the DFFR.

5.b Final Safety Evaluation Report

Prepare a final safety evaluation report of the Mark II DFFR at the end of the LTP based on the final results of Task 1 through 4 of this program and tasks 1, 2 and 3 of the Task A-39 program. In addition to the information contained in the preliminary evaluation report this report will include an evaluation of the LTP confirmatory experimental and analytical programs to assess the margin for selected loads. If reduced design loads are proposed based on new information obtained in the LTP, an evaluation with bases will be provided of the revised loads.

3. NRR Technical Organizations Involved:

   A. Division of Systems Safety, Containment Systems Branch

      1. Tasks No. 1.a through 1.d - The Containment Systems Branch has overall lead responsibility for design basis load definition for the Mark II containment system and has lead responsibility for the review and approval of LOCA-related pool dynamic loads for Mark II BWR facilities.

      2. Tasks No. 2.a and 2.b - The Containment Systems Branch will assist SEB and MEB in the review of load combination histories.

      3. Task No. 5.a and 5.b - The Containment Systems Branch has lead responsibility for the preparation of the preliminary and final Mark II safety evaluation reports based on results of the above tasks and the input received from the Structural Engineering and Mechanical Engineering Branches.

      4. Manpower requirements:

         FY 1978 - 1.75 man-years
         FY 1979 - 1.25 man-years
         Total - 3.0 man-years

B. Division of Systems Safety, Structural Engineering Branch

1. Tasks No. 1.b and 1.d - The Structural Engineering Branch has lead responsibility to review and evaluate the effects of fluid structure interaction on measured loads for the supporting tests used to establish downcomer and pool wall condensation and chugging loads.

2. Task No. 1.e - The Structural Engineering Branch has the responsibility to review and evaluate the design considerations for determining fatigue cycles on containment structures.

3. Task No. 2.a - The Structural Engineering Branch has lead responsibility for the review and evaluation of load combinations utilized in the evaluation of the Mark II containment and containment structures.

4. Task No. 3.a - The Structural Engineering Branch has lead responsibility for the review and evaluation of the acceptance criteria for the Mark II containment and containment structures.

5. Task No. 4 - The Structural Engineering Branch has lead responsibility to review and evaluate fluid-structure interactions in Mark II containment systems.

6. Task No. 5.a and 5.b - The Structural Engineering Branch has the responsibility of providing the results of their evaluation with bases to the task manager for each of the above tasks for incorporation in the preliminary and final Mark II safety evaluation reports.

7. Manpower Requirements:

   FY 1978 - 1.5 man-years

   FY 1979 - 0.9 man-years

   Total - 2.4 man-years

C. Division of Systems Safety, Mechanical Engineering Branch

   1. Task No. 1.e - The Mechanical Engineering Branch has the lead responsibility for the review and evaluation of methods used to predict SRV actuation.

   2. Task No. 2.b - The Mechanical Engineering Branch has the lead responsibility for the review and evaluation of load combinations utilized in the evaluation of the Mark II containment piping and components.

   3. Task No. 3.b - The Mechanical Engineering Branch has lead responsibility for the review and evaluation of the acceptance criteria for the Mark II containment piping and components.

4. Task No. 5.a and 5.b - The Mechanical Engineering Branch has the responsibility for providing the results of their evaluation with bases to the task manager for each of the above tasks to be incorporated in the preliminary and final Mark II safety evaluation reports.

5. Manpower Requirements:
   FY 1978 - .5 man-years
   FY 1979 - .25 man-years
   Total - .75 man-years

D. Division of Project Management

1. Tasks No. 1 through 5 - Provide coordination between the Division of Systems Safety, the Mark II applicants, and the Division of Project Management project managers for the individual Mark II BWR facilities. This includes meeting coordination and preparation of meeting minutes to document the actions of the generic Mark II review when the owners are involved.

2. Manpower Requirements:
   FY 1978 - .1 man-year
   FY 1979 - .1 man-year
   Total - .2 man-years

E. Division of Operating Reactors, Plant Systems Branch

1. Task 1 through 5 - Follow the activities of the STP and LTP Mark II programs and coordinate results with Mark I efforts.

2. FY 1978 - .1 man-years

FY 1979 - .1 man-years

Total - .2 man-years

4. Technical Assistance Requirements:

A. Brookhaven National Laboratory

1. Title: Pool Dynamic LOCA Loads - Task 1

2. Responsible Branch: Division of Systems Safety/Containment Systems Branch

3. Scope:

The contractor and his consultants are to provide expert technical assistance in the review of the Mark II owner's STP and LTP experimental and analytical efforts related to the definition of the Mark II pool dynamic loads.

4. Funding:

FY 1977 - $185,000

FY 1978 - $195,000 (requested)*

FY 1979 - $210,000 (estimated)

* This includes funds for review of SRV loads.

5. Interactions with Outside Organizations:

   A. Mark II Owner's Group

      1. Title:

         Pool Dynamic Loads - Task 1

         Load Combinations - Task 2

         Design Criteria - Task 3

         Plant Fluid Structure Interaction - Task 4

      2. Scope:

         The Mark II owner's group has developed a program to establish Mark II pool dynamic loads, load combinations, acceptance criteria, and generic methods to evaluate plant fluid structure interactions. The major elements of the STP are listed in the Mark II lead plant topical report to be issued August, 1977. The LTP is to be described in a revision to NEDO-21297, "Mark II Containment Supporting Program Report." This revision is to be submitted September, 1977.

6. Assistance Requirements from Other NRC Offices:

   A. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Analysis Development Branch

      1. Pool Dynamic Loads - Task 1

      2. Scope:

         The NRC is a participating member in the Marviken containment tests being conducted in Sweden. The results of these tests are currently under review to determine their applicability to confirm the proposed Mark II containment condensation and chugging loads used in the STP.

         RES has sponsored two additional research programs of possible applicability to the Mark II containment Long Term Program:

a. A program is currently underway at MIT to investigate the scaling relationship for pool swell loads.
1977 - $85,000, 1978 - $100,000

b. A similar program is underway at UCLA to investigate scaling relationships for steam discharges.
1977 - $100,000, 1978 - $100,000

7. <u>Schedule for Problem Resolution</u>

A. Summary Schedule

1.	Submittal of the Mark II Owner's Long Term Program Action Plan.	9/2/77
2.	Receipt of complete documentation of the Mark II Owner's Group program for the STP (owners have indicated this date).	10/3/77
3.	Staff establishes interim acceptance criteria for the STP pool dynamic loads, load combinations and structural acceptance criteria.	2/1/78

4. Issue STP Mark II safety evaluation report 4/3/78

5. Receipt of complete documentation of the Mark II owner's group program for the LTP. 4/2/79

6. Issue Mark II final safety evaluation report 10/10/79

B. Detailed Schedule

See Attachment 2 A and B

C. Technical Assignment Control Number - TAC 3003

8. Potential Problems

A. The Mark II owner's group was restructured at the May 10, 1977, meeting with NRC management from a single program terminating in July, 1977, to the current two part program consisting of a short and long term program. Since that meeting we have requested on several occasions that the owner's group provide us with a complete description of the STP and LTP including: a clear description of all experimental and analytical programs to be included in each of the STP and LTP programs; a detailed schedule for program tasks; a description of the documents to be provided for each task and a schedule for transmitting task documents to the NRC. We have still not received this information.

An NRC management letter to the Mark II owner's group will be prepared to emphasize our urgent need for this information. It is difficult for us to develop a meaningful review schedule without this information. The attached review schedules were developed based on the limited information currently available to us.

B. Our current schedule for review of the Mark II STP calls for the preliminary Mark II load evaluation report to be issued in April, 1978. Up to this time, the results of this generic program will not have been factored into each plant's design assessment report.

Our schedule for the generic Mark II pool dynamic STP is not consistent with the current schedule for the lead Mark II plant. The lead plant schedule includes an SER issuance date of November 1, 1977, and a fuel loading date of April 1, 1978. The resolution of these schedule inconsistencies may result in an interruption of the generic STP program to allow our review efforts to concentrate on the capability of the lead Mark II plant or plants to accommodate pool dynamic loads. This could result in a significant change in our review schedule for the generic Mark II pool dynamic loads programs.

C. The Mark II owner's group program to resolve pool dynamic loads since its inception has suffered from changes in program direction and program delays. A continuation of these problems could affect our review schedule.

D. The current Mark II STP does not indicate a need for specific Mark II NRC-funded pool dynamic load experimental or analytical programs for Mark II containments. However, we are currently investigating the potential need for NRC-funded confirmatory Mark II pool dynamic analytical and experimental programs that might be included in the Mark II LTP. The determination of the need for NRC-funded programs depends on the Mark II owner's LTP and the applicability and availability of related domestic and foreign programs for pressure suppression containments. We should be able to establish our needs in this area by November, 1977.

MARK II POOL DYNAMIC LOADS ACTIVITIES (SIP) - ATTACHMENT 1A

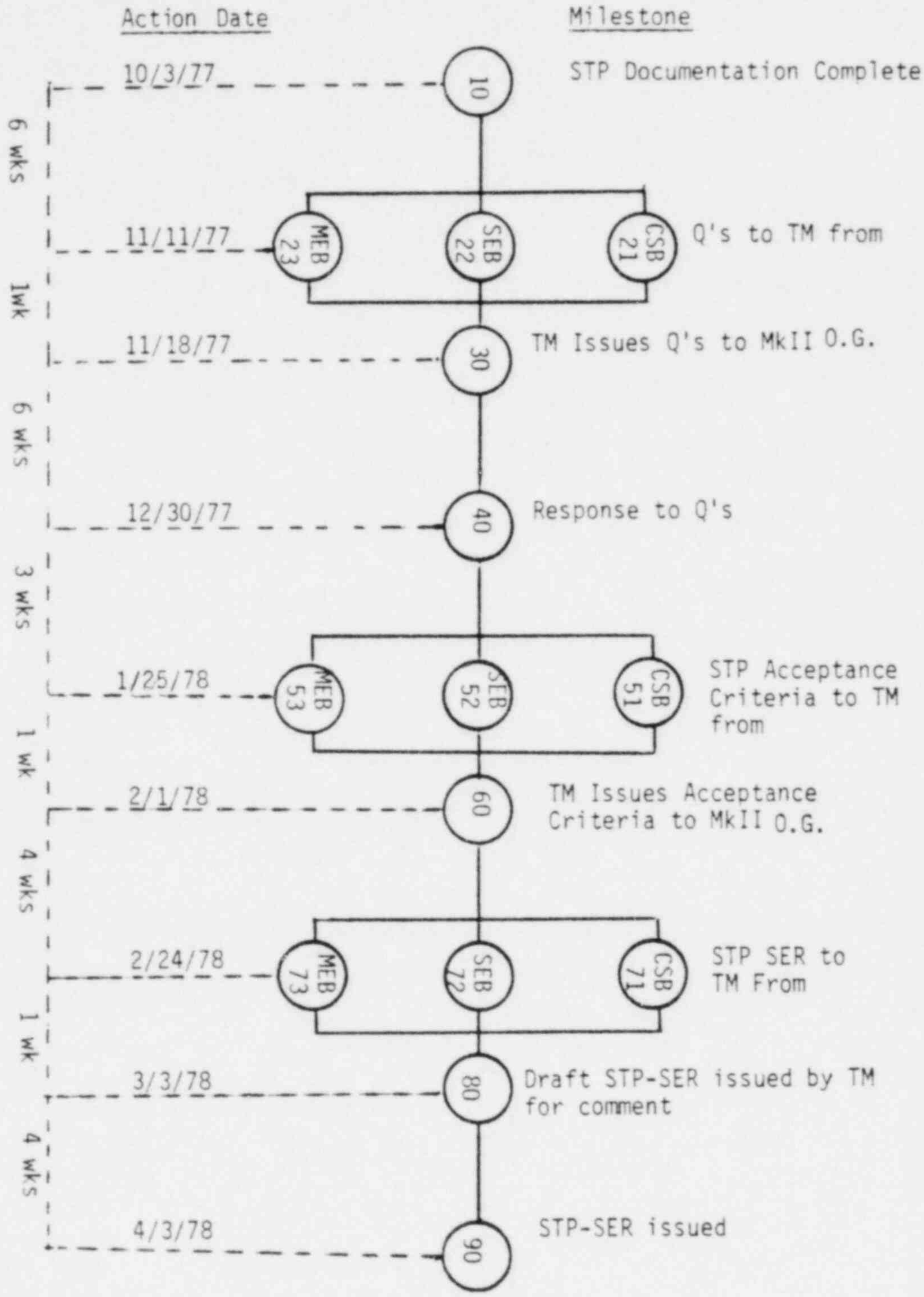
5. Reports			3. Design Criteria		2. Load Comb.		1. Loads					Tasks
.b Final Report	5.a Preliminary Report	4. Plant F/S Interaction	.b Piping and Components	3.a Containment Structures	.b Piping and Components	2.a Containment Structures	.e SRV Actuation	.d Condensation	.c Pool Boundary	.b Downcomer	1.a Pool Swell	Report or Activity
	X		X	X	X	X	X	X	X	X	X	1. DFFR, Rev. 1, 2 & 3, Am 1 (NEDE 21061)
	X							X	X	X	X	2. 4T Phase I Application Memo
	X							X	X	X	X	3. 4T Phase 2&3 Application Memo
	X	X						X	X	X	X	4. 4T Phase 1 Tests (NEDE 13442 P-01)
	X	X						X	X	X	X	5. 4T Phase 2&3 Tests (NEDE 13468P)
	X	X						X	X	X	X	6. MkII Supporting Program Report (NEDO 21297)
	X										X	7. Pool Swell Model & Confirmation (NEDE 21544-P)
	X				X	X						8. SRSS Load Combination Justification (NEDE 24010)
	X	X						X		X		9. GE Licensee Test Results for Vent Loads (NEDE 21078-P)
	X										X	10. EPRI 3D, 1/13 scale MkII Air Test Report (EPRI NP-441)
	X										X	11. Impact Tests on Pool Structures (NEDE 13426 P, NEDC 20989-2P)
	X										X	12. Impact Model (NEDE 13426 P, NEDC 20989-2P)
	X	X						X		X		13. MkII Load Plant Report
	X							X		X		14. Single Vent Chugging Model
	X							X				15. Multivent Hydrodynamic Chugging Model
	X	X						X				16. 4T Facility Fluid-Structure Interaction Study (Anamet lab. 1078.57-8)
	X										X	17. Evaluation of EPRI 3D air tests (Bechtel Report)
	X	X						X				18. Plant Fluid-Structure Interactions for Chugging loads (Bechtel Report)
	X	X						X				19. Plant Fluid- Structure Interactions for SRV loads (Bechtel Report)
	X	X			X	X		X		X	X	20. MkII Supporting Program Report (Revised to include LTP)
	X	X						X		X		21. Overseas Licensing Approach Chugging Loads
	X	X								X		22. Single Vent Lateral Loads

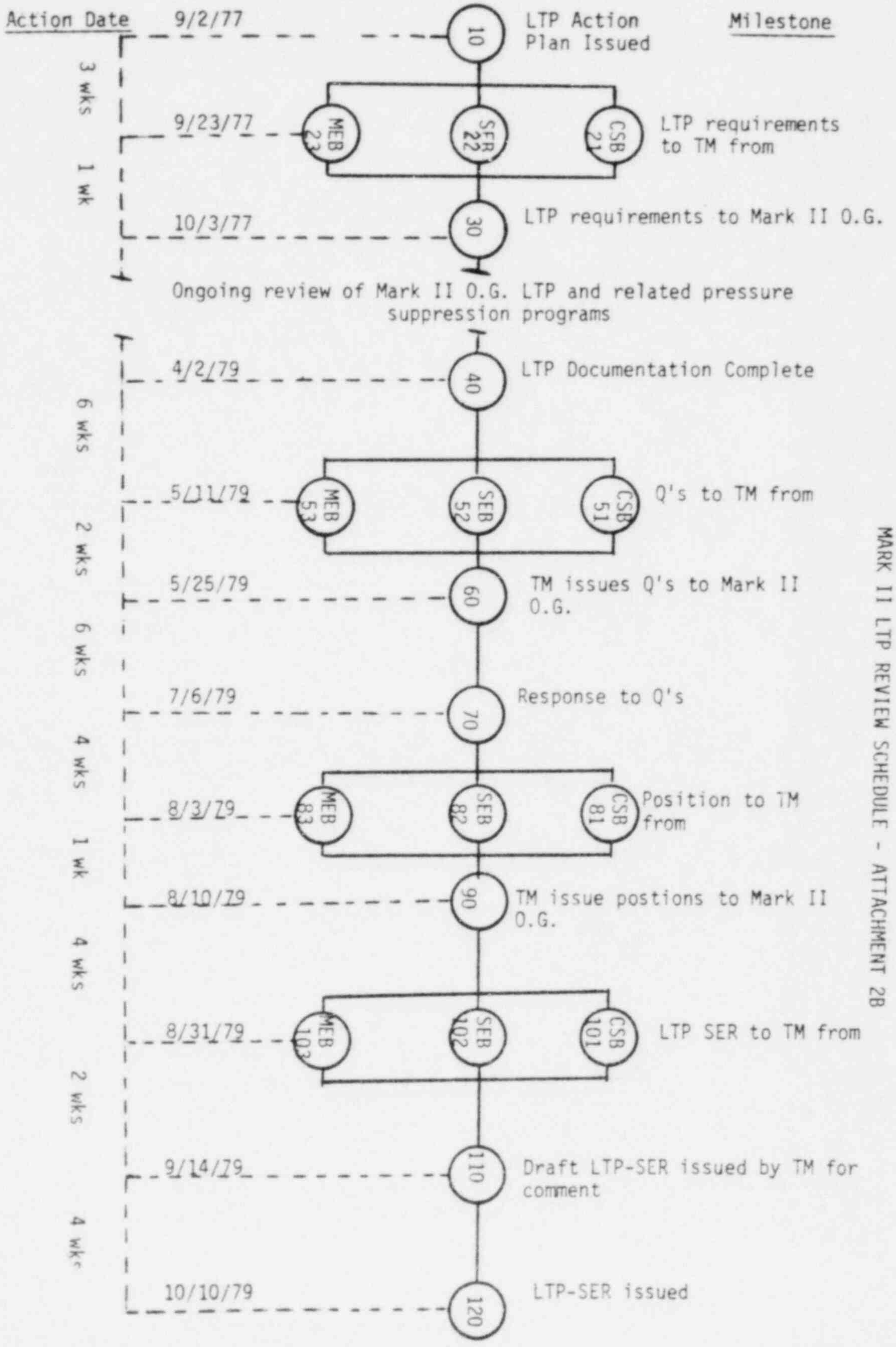
MK II Pool Dynamic Loads [illegible]

(Attachment B)

Tasks		Report or Activity: 1. DFFR, Rev. 4+	2. CREARI Single Vent Steam Scaling Test Report	3. Analytical Steam Scaling Study	4. Mark II multivent Scaled Steam Test Report	5. Multivent Monte Carlo Model Description	6. EPRI Single Vent, MkII 1/13 scale, Air Test Report	7. MkII Owners Group Evaluation of EPRI Single Vent, Scaled, Air Test	8. Related Pressure Supression Steam Tests - foreign & Domestic	9. Mark II Multivent Facility Fluid Structure Interaction Study	10. Confirmation of Multivent Analytical Model
1. Loads	1a Pool Swell	X	X	X	X		X	X			
	b Downcomer								X		
	c Pool Boundary										
	d Condensation	X	X	X	X	X			X	X	X
	e SRV Actuation										
2. Load Comb.	2.a Containment Structures										
	b Piping and Components										
3. Design Criteria	3.a Containment Structures										
	b Piping and Components										
	4. Plant F/S Interaction	X	X		X				X	X	
5. Reports *	5.a Preliminary Report										
	b Final Report	X	X	X	X	X	X	X	X	X	X

MARK II STP REVIEW SCHEDULE - ATTACHMENT 2A





MARK II LTP REVIEW SCHEDULE - ATTACHMENT 2B

REVISION 0

AUG 30 1977

Title: ATWS (A-9) (TAC #4019)

Lead Responsibility: DSS

Lead Assistant Director: D. F. Ross, Jr., A/D for Reactor Safety

Task Manager: A. C. Thadani

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 30,1977

1. Problem Description

The Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors, WASH-1270, discussed the probability of an ATWS event and an appropriate safety objective for these events. After several years of discussions with the vendors and evaluation of vendor models and analyses, the staff published in late 1975 its status report on each vendor analysis including detailed guidelines on analysis models, and ATWS safety objectives. The available information on consequences consists of vendor calculations based on essentially realistic models, but with some conservatism in the input data. The calculated consequences using the staff status report indicate design changes ranging from minor to significant are needed to meet the safety objective. The industry has argued that the staff requirements in the status reports are excessive and that inadequate attention has been given to ATWS probabilities and consequences.

2. Plan for Resolution:

The staff is presently rereviewing the whole ATWS program to reassess the impact of the status report requirements. In this regard, a task force, chaired by Dr. Stephen Hanauer, is developing and evaluating different approaches to determine the probability of an ATWS exceeding acceptance limits, if the status report fixes are incorporated. The task force intends to recommend the publishing of a staff technical report which would include a review of recent vendor calculations, probability of an ATWS, discussion of various options, and a recommended course of action.

The following provides an estimate of the manpower effort needed to complete this generic review. This estimate on manpower and schedule would have to be revised if one or more of the potential problems discussed in Section 8 materialize.

I. TECHNICAL REPORT.

The Hanauer task force is expected to complete a technical report on ATWS in September 1977. The report will include the following considerations.

1. Safety Goal

Develop rationale for safety goal using WASH-1270, WASH-1400 and conservative versus realistic risk calculations.

2. Frequency of ATWS

Identify transients of concern and their recurrence frequency and the reliability of scram systems. The data and methods would use the information presented in WASH-1270, WASH-1400, EPRI reports and other publications. The scram system unreliability estimates will include the consideration of the rods and drives.

3. Course of ATWS Events

The discussion would cover evaluation of each vendor analysis assumptions, evaluation models and transient analyses. This evaluation will also include a discussion of system availability.

4. Conclusions

Using the ATWS safety goal, estimate frequency of events and the calculated consequences,and decide which design modifications are necessary.

5. Criteria for Acceptable Fix

Having defined the kinds of design changes indicated to meet the ATWS goals, the staff criteria for acceptability of any required design modifications will be provided.

II. OPTION SELECTION

The technical report would provide a basis for the selection of an option for implementation of ATWS requirements. It may be necessary to have different requirements for operating reactors than those being designed or yet to be designed. The present judgment is that an option that requires design modifications to meet ATWS limits likely will be selected. In view of the low probability of an ATWS event and a small number of nuclear power plants operating, the time period before ATWS fixes are implemented does not contribute significantly to the risk.

III. PRESSURIZER SAFETY VALVE INTEGRITY AND WATER RELIEF RATE

ATWS events result in conditions for which the valves are not tested and some concerns as to their integrity and the relief rate need to be resolved. The Germans may have a water relief test plan and perhaps we should particupate to obtain information of interest to us. In any case, a decision to participate in this task is necessary to allocate manpower requirements.

IV. BWR POOL TEMPERATURE LIMIT

Establishment of the suppression pool temperature limit is needed to completely identify the necessary design changes for Boiling Water Reactors. The pool temperature limit is part of a Category A item (this activity number not yet assigned) and an interim acceptance limit is expected to be available in November 1977 and the final acceptance limit to be available in June 1978.

V. COMPLETION OF EVALUATION MODELS (See Item 3D)

VI. STANDARD REVIEW PLAN

Following completion of the technical report and selection of an alternative for CPs and OLs, it would be necessary to develop a plan for review of individual license applications.

VII. OPERATING REACTORS

A) Short Term Fix

DOR has begun an effort to require implementation of a recirculation pump trip on all operating BWRs. This task involves development of criteria and implementation procedures.

B) Long Term ATWS Fix

Following selection of an option for satisfying ATWS requirements for CPs and OLs, develop requirements for the operating reactors.

VIII. REVIEW OF FINAL VENDOR ATWS GENERIC ANALYSES

In conformance with the selected option, the vendors will be required to provide ATWS analyses. The analyses will be reviewed to insure that the assumed modifications satisfy the ATWS limits.

3. NRR Technical Organizations Involved: (Manpower Estimates for FY '78 only)

A. DSS/Reactor Systems Branch

(1) Coordination of ATWS Program

(2) This effort would involve coordination of internal reviews, meetings with vendors and consultants.

(3) Manpower requirements: Four Man Months

B. DSS/RSB

(1) Option Selection (Task II)

(2) Technical Report Discussions with ACRS with other NRR Divisions

(3) Manpower Requirements: EDO: ½ Man Month
RSB: 1 Man Month

C. DSS/AB/I&CSB for GE Model (Task V)

(1) Evaluation Models

(2) Complete Review of B&W and GE models

(3) Manpower Requirements: AB: Two Man Months
I&CSB: ½ Man Month

D. DSS/RSB/CPB/I&CSB/DSE/AAB (Task VI)
CSB

(1) Develop Standard Review Plans including considerationof value impact and obtain RRRC approval.

(2) RSB with support from other branches will develop review guidelines.

(3) Manpower Requirements: RSB: 3 Man Months
I&CSB: 4 Man Months
Other Branches: 3 Man Months

E. DSS/RSB/I&CSB/CSB/MEB/AB/CPB (Task VIII)

(1) Generic vendor analyses for staff guidelines

(2) The staff will review the vendor analyses to assure that the analyses are performed in accordance with the staff guidelines. Design changes necessary to meet the limits would also be identified.

(3) Manpower Requirements:

Branch	MMS Vendor				Total
	GE	W	B&W	CE	
RSB	1	1	1	1	4 MM
MEB	½	¼	3/4	1	2½ MM
AB	¼	¼	¼		3/4 MM
AAB	½	½	½	½	4 MM
CPB	¼	¼	¼	¼	1 MM
CSB	¼	1/8	1/8	1/8	5/8 MM
I&CSB	1	1	1	1	4 MM

F. AAB/DSE

Review vendor analyses and calculate radiological consequences.

Four Man Months

G. DOR/PSB/RSB/ORB (Task VII A)

(1) Short Term Fix on BWRs

(2) This effort involves development of criteria for recirculation pump trip and development of implementation procedures.

(3) Manpower Requirements:

Plant Systems Branch: 6 Man Months
Reactor Safety Branch: 6 Man Months
Operating Reactors Branch: 6 Man Months

H. DOR/PSB/RSB/ORB (Task VII B)

(1) Long Term ATWS Program

(2) Depending on DSS findings, develop criteria and implementation procedures for required fixes on operating reactors.

(3) Based on presently available information, the projected required manpower is as follows:

Plant Systems Branch: 6 Man Months
Reactor Safety Branch:12 Man Months
Operating Reactors Branch: . 6 Man Months

I. DOR/PSB/RSB/ORB

(1) Contributions to other subtasks

(2) Liason and review efforts that DOR will supply

(3) Manpower Requirements:

Plant Systems Branch: 6 Man Months
Reactor Safety Branch: 6 Man Months
Operating Reactors Branch: 3 Man Months

4. Technical Assistance Requirements:

A. BNL: Perform computer runs for B&W 177FA plant to obtain sensitivity values for changes in initial conditions. This task has essentially been completed.

Management: Analysis Branch

B. Sandia: Perform Monte Carlo calculations using vendors' and BNL calculations.

Support ATWS probability studies

Management: EDO

EDO/RSB Effort - Two Man Weeks

Funding: $6K

C. Reactor Pressure Vessel Closure (TAC-3932)

Finite Element Analysis of B&W Vessel

Report Completed and Reviewed 11/15

Management: Mechanical Engineering Branch

MEB Effort - 2 Man Weeks

Funding: $ 30K

D. Three-Dimensional Inelastic RPV Closure Analysis

The decision to contract this analysis will be made following review of the two-dimensional analysis.

Funding: $ 100K

MEB Manpower - Five Man Weeks

5. Interactions with Outside Organizations

A. ACRS

"This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses."

B. EPRI

The staff has been reviewing the EPRI probabilistic studies and intends to document its review in the technical report.

C. KWU

As noted in 8.E.

D. Standard Development

It is anticipated that an ATWS ANSI standard would be developed. NRC and vendor participation in this task is anticipated. The standard would be a useful tool in the implementation stages. Therefore, a decision to participate in the standard development effort must be made.

6. Assistance Requirements from other NRC Offices:

Nuclear Regulatory Research/Probabilistic Analysis Branch Support on ATWS statistical effort.

7. Schedule for Problem Resolution:

A.	Technical Report	10/30/77
B.	Option Selection	1/78
C.	Safety Valve Requirements Decision	11/77
D.	Vendor Analysis Provided in Conformance with 7.B.	3/78
E.	Standard Review Plans	4/78
F.	GE SER	6/78
G.	W SER	9/78
H.	CE SER	11/78
I.	B&W SER	1/79

8. Potential Problems

A. Rulemaking Hearings

If rulemaking is eventually chosen as the method of generic resolution of this problem, hearings would likely be requested. If so, it is difficult to assess the length of time and manpower the hearings would require.

B. Plant Hearings

Extensive effort is expected for hearings on some plants. For example, the Black Fox hearing (possibly this fall), would require significant effort because of the type and the details of contentions. Three or more man months from RSB and two man months effort from other branches may be needed.

C. Role of Hanauer Task Force

If the completion of the technical report is delayed or if the recommendation of the task force is to do additional studies, this action plan would have to be revised.

D. Reactor Safety Study

Possible differing conclusions between NRR and RES on ATWS contribution to the overall risk.

E. Pressurizer Safety Valve Integrity and Water Relief Rate

If the decision is made to obtain this information experimentally, the staff could continue ATWS generic review with an interim statement on safety valve integrity and water relief rate. However, significant effort from the Reactor Systems Branch to coordinate this experimental program would be required.

F. Long-Term Detailed Probabilistic Studies:

In the present simplified probabilistic study, a large number of assumptions, necessarily made to get some quick results, may cast a doubt on the study. These concerns relate to inadequate selection of parameters, their distributions, nonlinear effects of parameters, interdependencies between parameters, etc., and the staff may recommend in the technical report to perform a more detailed study.

Management: Reactor Systems Branch

RSB: Significant Efforts (Support from ASG) FY '78 and FY '79

Estimated Cost: $ 2 ½ M

G. Standard Development

If it is decided to participate in the ANSI ATWS standard development effort, approximately two man months of RSB effort would be anticipated and support from other branches may be necessary.

REVISION 0

August 11, 1977

CATEGORY A TECHNICAL ACTIVITY NO. A-10

Title: BWR Feedwater Nozzle Cracking (Including Non-Destructive Examination Techniques for Inservice Inspection)/BWR Control Rod Drive Return Line Nozzle Cracking

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for
Operational Technology, DOR

Task Manager: Dick Snaider, DOR

1. Problems Description

A. BWR Feedwater Nozzle Cracking

Of the 22 operating BWR's with feedwater nozzle/sparger systems (normally 4 nozzles/spargers per BWR, nominal nozzle diameter being 10" - 12"), 19 have been inspected to date (6/30/77), resulting in the discovery of blend radius and bore cracking in 18 vessels. Although most cracks have been in the range of 1/2" to 3/4" total depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 1.50 inches. The initiation of cracking is due to high cycle fatigue caused by fluctuations in water temperature within the vessel in the sparger-nozzle region during periods of low feedwater temperature when the flow may be unsteady and intermittent. Once initiated, the cracks are driven deeper by the larger pressure and thermal cycles associated with startup and shutdown.

Fracture analyses indicate that the cracks found to date in the feedwater nozzles constitute a potential safety problem because the observed rate of crack growth with time in service is such that the margin of safety against fracture will be reduced below acceptable values unless the cracks are detected and ground out every few years. Obviously, repair by grindout can be repeated only a few times before ASME Code limits for nozzle reinforcement are exceeded. However, repair by welding buildup of the grindout has not been demonstrated to be acceptable. In addition, the inspection and removal of cracks by grinding has caused enough radiation exposure to personnel to be deemed unacceptable as a long-term solution.

APPROVED BY TASC, AUGUST 19, 1977

B. Control Rod Drive Hydraulic Return Line Nozzle Cracking (CRDRL Nozzle)

Each of the applicable BWR's has one CRDRL nozzle of 3" - 4" diameter, which is normally located approximately four feet below the level of the feedwater nozzles (In the Oyster Creek and Nine Mile Point vessels, the CRDRL nozzle is located at the same level as the feedwater nozzles). Thermal fatigue cracks have been found by dye penetrant (PT) inspection of CRDRL nozzle at 3 of the 4 domestic units inspected to date (6/30/77). These cracks resemble those found in the BWR feedwater nozzles, and the cause of cracking appears to be thermal fatigue. All but 2 of the operating domestic BWRs have some sort of thermal sleeve (there are several designs) in the CRDRL nozzle, but because of the limited number of inspections of nozzles with sleeves, the efficacy of the sleeves is not known.

To date, the principal activity by licensees has been to plan for the ultimate re-routing of the CRDRL, to be relocated following recommendations made by the General Electric Company (GE) in Services Information Letter (SIL) 200 (October 29, 1976), and SIL 200 Supplement 1 (March 25, 1977). Some licensees have chosen to implement a temporary "fix" by simply valving out the CRDRL (thus shutting off cold water flow to the nozzle), and simultaneously increasing CRD system pressures to force return water to the vessel through the CRD seals. This modification results in decreasing the CRD "settle" margin and could ultimately result in failure of a rod (or rods) to settle. Consequently, this temporary fix procedure and the changes brought about by re-routing of the CRDRL are under active review by the staff.

In the interim, to increase assurance of safety for continued operation, the staff is recommending inspection of the CRDRL nozzle blend radius and bore at each BWR during its next scheduled refueling outage. As in the case of feedwater nozzles, we are especially concerned, particularly in the case of older units, that a potential safety problem could arise from deep cracks which would necessitate weld repair.

2. Plan for Problem Resolution

Briefly stated, the plan for generic resolution of the BWR feedwater nozzle and CRDRL nozzle cracking problems will involve the following:

(a) Issue interim guidance to operating units. Such guidance will include criteria for inspection based upon present knowledge of crack growth and available techniques and will be issued as a NUREG report.

(b) DOR and DSS follow advancements in the following areas:

- (i) Development, by a consultant to DSS, of a mathematical model of the reactor-feedwater nozzle area including its thermal-hydraulic, heat transfer, and stress/fracture mechanics conditions. Although the mixing flow (and possible stratification) problem seems three-dimensional, a reasonable solution might be obtained from a one-dimensional thermal/hydraulic model coupled to a two-dimensional plane model (to capture the non-axisymmetric temperature distribution). The stress analyses, including the fatigue calculations and crack growth estimates, are more straight forward.

  The model will consider loadings and transients from normal and abnormal plant operations and is essential for evaluation of the generic design modifications. DOR will actively participate in such modeling as a means of verifying GE test data.

- (ii) Development and testing of effective feedwater nozzle thermal sleeves and spargers to protect the nozzle bore and blend radius from thermal cycling and thus minimize or remove the source of crack initiation.

- (iii) Development of viable ultrasonic test (UT) techniques by the nuclear industry to allow reliable and consistent early determination of cracking (and credible claims for the absence of cracking) from positions exterior to the reactor vessel. Such development of UT is important to both DOR and DSS final positions and the staff effort will be supplemented by two consultants listed below. This portion of the program will be coordinated with Task No. A-14, Flaw Detection.

- (iv) Development of various feedwater system and CRD system modifications as part of the generic effort toward problem resolution.

- (v) Issuance of Branch Technical Position paper (CP and OL plants) and final NUREG document (operating plants) upon satisfactory completion of subtasks (i) through (iv) above.

3. NRR Technical Organizations Involved

A. Engineering Branch, Division of Operating Reactors. Has overall lead responsibility for review of all generic inspection, repair, in-service inspection technique development, weld-repair/annealing study, and modification (such as clad removal and new design thermal sleeves/spargers) efforts. Will gather and disseminate critical information (fluid flows and temperatures) on operating plants. Will manage UT and fracture mechanics consultants as listed in section 4 below. Issue interim and final NUREG documents.

Manpower estimates: 2.0 manyear FY 1977, 1.7 man year, FY 1978,
1.7 manyear FY 1979.

B. Plant Systems Branch, Division of Operating Reactors. Has lead responsibility for review and approval of any proposed generic feedwater or CRD system modifications. Will assist in development of NUREG documents.

Manpower estimates: .1 manyear FY 1977, .2 manyear FY 1978,
.2 manyear FY 1979.

C. Mechanical Engineering Branch, Division of Systems Safety. Will work with DOR on development of criteria and will issue BTP for CP/OLs similar to NUREG guidance issued for operating facilities. Will manage consultant on model development.

Manpower estimates: .5 manyear FY 1977, .8 manyear FY 1978,
.8 manyear FY 1979

D. Materials Engineering Branch, Division of Systems Safety. Will assist DSS-MEB as necessary, in the development of criteria.

Manpower estimates: .1 manyear FY 1977, .3 manyear FY 1978,
.3 manyear FY 1979

E. Task Manager, Division of Operating Reactors. Has overall responsibility for coordination of DOR and DSS technical tasks and for the development and issuance of criteria documents.

Manpower estimates: .2 manyears FY 1977, .2 manyears FY 1978,
.2 manyears FY 1979

4. Technical Assistance Requirements

Contractor	Amount FY 1977	 FY 1978	Program Objectives
A. ORNL - Ken Klindt (Managed by DOR)	$15K	$30K	Monitor UT development efforts when requested to do so by DOR, and provide consultation in evaluating results from field inspections and related developmental work. DOR will disseminate such test data. Such information is necessary in determining the largest flaw which could remain undetected in the complex nozzle geometry. This flaw size will be used in the fracture mechanics crack growth calculations.
B. Sandia Laboratories - John Gieske (Managed by DOR)	$25K	$30K	Same as for (a) above.
C. Washington University - Paul Paris (Managed by DOR)	$20K	$20K	Perform fracture analyses of feedwater nozzle cracks detected in operating reactors. This is necessary for generic crack growth calculations.
D. Contractor to be Selected (Managed by DSS)	$25K	$80K	Perform mathematical and thermal-hydraulic evaluations of various designs as outlined in Paragraph 2(b)(i).

5. Interactions With Outside Organizations

A. General Electric Company

The NRC staff has followed all GE generic testing and developmental work, especially those tests designed to determine the cause of cracking and those developments related to UT enhancement. This coordination will continue.

B. Electric Power Research Institute

The NRC staff will follow closely EPRI UT optimization development work for the complex nozzle geometry. This work has other generic implications (see Task No. A-14).

C. Individual Licensees and Applicants of BWR Facilities

Each licensee has already been involved in discussions and written correspondence with the NRC concerning inspections to be performed. This interaction, as well as discussions on a generic basis, will continue until problem resolution, although the NRC position shall be spelled out clearly in the forthcoming interim position paper. Applicants for BWR OLs will also be involved in similar interaction with DSS.

6. Assistance Requirements From Other NRR Offices

Office of Nuclear Regulatory Research (RES). RES is responsible for the Heavy Section Steel Technology (HSST) program. Information obtained from this program will be useful in the development of generic fracture analysis methods for a flaw at a geometric discontinuity.

7. Schedule for Problem Resolution

The major milestones for the generic feedwater and CRDRL nozzle issues are as follows:

(i) Issue interim NRC guidance to licensees - August 1977.

(ii) Start review of completed GE testing - June 1978

(iii) DSS approve generic designs for new plants and issue interim guidance - October 1978

(iv) With assistance of consultants and input from Task A-14, resolve UT issue, evaluating techniques for use on complex geometry - June 1979.

(v) Issue final guidance to applicants (Branch Technical Position) and licensees (NUREG Document) - October 1979.

8. Potential Problems

The most serious potential problem facing the NRC staff and licensees at this point is the discovery of a crack large enough to exceed the ASME code criteria for required reinforcement area. This would result in the need for a vessel repair (other than grinding) which would be an undertaking of potentially large proportions and of safety significance.

A generic contingency plan is presently being outlined by DOR. As scoping of such a contingency plan develops, we will document the plan as Appendix A to this report.

The schedule may be lengthened by extension of UT analysis in the performance of related task A-14.

CATEGORY A TECHNICAL ACTIVITY NO. A-11

REVISION 0
AUG 30 1977

Title: Reactor Vessel Materials Toughness

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: Ronald M. Gamble, DOR

1. Problem Description:

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. At service times and operating conditions typical of current operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure; however, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWR's will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10 CFR Part 50, after comparatively short (approximately 10 EFPY) periods of operation. To ensure adequate toughness margins for operating plants it will be necessary to (1) establish a suitable safety criterion for low toughness materials, (2) define and identify critical materials in reactor vessels and (3) monitor and evaluate operational materials surveillance program results relevant to establishing a suitable generic toughness criterion. For those facilities not yet licensed for operation, current licensing criteria are adequate to ensure suitable safety margins throughout design life with the materials currently employed for reactor vessel fabrication. However, the need exists to reconsider these current criteria in light of new methods that may be developed in the evaluation of low toughness materials and to appropriately augment or refine these present criteria to include these new aspects and maintain NRC licensing consistency.

2. Plan For Problem Resolution:

The determination of an appropriate licensing criterion for low toughness reactor vessel materials in currently licensed plants and the evaluation of material degradation resulting from neutron irradiation demands a broad, integrative effort encompassing several aspects

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 30, 1977

of materials and fracture technology. To successfully establish a suitable licensing criterion for low toughness reactor vessel materials in currently licensed plants and to enable an accurate assessment of neutron irradiation damage, the following tasks need be completed.

A. Development of Advanced Fracture Mechanics Methods

The measurement of fracture toughness for reactor vessel and other materials at temperatures corresponding to the upper shelf region is complicated by the presence of material plastic flow. Current toughness testing and analytical methods based on linear elastic fracture mechanics are not adequate to account for plastic flow. New testing and analytical techniques must be developed to allow evaluation of low toughness in reactor vessel materials for normal operating and faulted conditions.

B. Fracture Toughness Evaluation For Postulated Accident Conditions

In addition to normal operating conditions, the generic criterion for low toughness materials must be sufficiently comprehensive to include postulated accident conditions. To ensure that adequate material fracture toughness is available during postulated accident conditions, the following conditions must be reviewed.

1. Thermal Shock: It has been postulated that the thermal shock to the reactor vessel caused by ECCS operation could, near end of life where accumulated radiation damage is significant, cause the vessel to fail in such a manner that it could not hold the cooling water.

2. Effect of Main Steam Line Break: A main steam line rupture could produce a repressurization of the reactor vessel following initiation of the ECCS. The NRC staff has been advised by various NSSS vendors that this repressurization would produce a significant temperature and stress transient and could, near end of life where accumulated radiation damage is significant, result in significantly reduced safety margins for the reactor vessel.

C. Radiation Damage Technology Improvements

The amount of radiation damage incurred by the reactor vessel must be predicted and then checked by extrapolation of the results from the surveillance program during the service life. Uncertainties in the predicting methodology can be significant. The variables that are relevant to this study are steel chemical composition and microstructure, neutron spectra variations, uncertainties in dosimetry and dose rate. As older vessels become more highly irradiated, our capability to predict the associated, apparent reduction in toughness must improve. During this time, more information will be available from the surveillance programs. We must develop better ways to evaluate this information to improve our predictive capability.

D. Reactor Vessel Annealing Feasibility

Because the possibility exists that severe material toughness degradation due to neutron irradiation damage may eventually preclude some reactor vessels from meeting fracture toughness licensing criteria, it may be necessary to anneal these vessels to regain the required toughness levels. A reactor vessel annealing feasibility study will be conducted to define the annealing parameters and the procedures necessary to ensure that adequate safety margins are regained and maintained.

E. Identification and Evaluation of Reactor Vessel Welds

It has recently been determined that the welds in various reactor vessels are not always represented by identical welds in the surveillance program associated with any given plant. This makes accurate evaluation of the surveillance data, relative to the toughness degradation for any given operating plant, difficult. Consequently to obtain sufficient comprehensive information for a generic evaluation, each reactor vessel and surveillance program weld material must be identified, located and categorized to ensure effective utilization of the surveillance program and effective evaluation of the reactor vessel material fracture toughness. Letters requesting information relevant to this task have been sent to PWR licensees.

F. Development of Surveillance Information System

Because of the large number of possible combinations of reactor vessel and surveillance materials and the large number of variables involved in evaluating these materials, it is necessary to develop an information system for the storage and retrieval of these data. This system will be utilized particularly to maintain up-to-date, accurate data for the generic and plant specific evaluation of operating facilities.

3. NRR Technical Organizations Involved:

A. Engineering Branch, Division of Operating Reactors. Has overall lead responsibility in the identification of relevant reactor vessel material in licensed plants, evaluation of operating experience with neutron irradiation damage, determination of the associated degradation in reactor vessel material toughness and the evaluation and determination of an appropriate safety criterion for low toughness reactor vessel materials.

Manpower Estimates: 0.3 manyears FY 1977, 2.5 manyears FY 1978, 2.5 manyears FY 1979

B. Materials Engineering Branch, Division of Systems Safety. Has lead responsibility for the review of information developed during the evaluation of material toughness in licensed facilities for possible inclusion into material toughness criteria currently used for facilities not yet licensed for operation, where appropriate.

Manpower Estimates: 0.1 manyear FY 1977, 1 manyear FY 1978, 1 manyear FY 1979.

4. Technical Assistance Requirements

Technical assistance from organizations outside the NRC will be required to complete tasks 2A, Development of Advanced Fracture Mechanics Methods; 2B, Fracture Toughness Evaluation During Faulted Conditions; 2C, Radiation Damage Technology Improvements and 2D, Reactor Vessel Annealing Feasibility. The contractors assisting in these tasks are as follows:

A. Contractor: Washington University,(EB/DOR)

Funds Required: $120K FY 1977, $50K FY 1978, $50K FY 1979.

This program is directed specifically at tasks 2A, Development of Advanced Fracture Mechanics Methods and 2B, Fracture Toughness Evaluation During Faulted Conditions. The results of the program will allow advanced fracture mechanics techniques to be used to establish a technical basis for NRC's development of a suitable licensing criterion for low toughness materials. Associated with this is the determination of simplified analytical techniques to evaluate normal operating conditions, postulated accident conditions and assistance in plant specific analyses.

B. Contractor: Naval Research Laboratory, (EB/DOR, MTEB/DSS)

Funds Required: $140K FY 1977, $75K FY 1978, $75K FY 1979.

This program will investigate neutron irradiation of reactor vessel steels and is directed specifically at tasks 2C, Radiation

Damage Technology Improvements and 2D, Reactor Vessel Annealing Feasibility. The results should provide improved means to quantitatively describe the effects of material microstructure, chemical composition, neutron spectra and dose rate and allow suitable evaluation, prediction and monitoring of irradiation damage to reactor vessel steels. Included in this program is a study of the feasibility of in-place annealing of reactor vessels to restore fracture toughness to levels that will provide adequate safety margins should the material toughness degradation be sufficient to preclude meeting licensing requirements. Funding for this program is now shared by DSS and DOR.

5. Interaction With Other Outside Organizations:

A. Licensees

Intermittent interaction with licensees is expected for the purpose of obtaining required materials data.

B. NSSS Vendors

Some plant specific analyses have been conducted by the NSSS Vendors. Review of the portions of these analyses relevant to completion of the generic task will be required. Some NSSS Vendors have first hand knowledge of fabrication and materials data relevent to low material toughness; review of these data will be required.

C. EPRI

EPRI is currently funding a number of programs related to reactor vessel materials toughness. These programs include studies for neutron irradiation damage of pressure vessel steels and the development of fundamental failure criteria based on elastic plastic fracture mechanics. Interaction with EPRI to remain informed on the direction and results of these programs and to ensure that appropriate NRC licensing concerns are addressed will be required.

D. ACRS

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

6. Assistance Requirements from Other NRC Offices

A. Office of Nuclear Regulatory Research, Division of Reactor Safety Research, Metallurgy and Materials Branch

RES is funding a major experimental research program (Heavy Section Steel Technology, HSST) through Oak Ridge National Laboratory to determine the fracture toughness of reactor vessel steels and the safety margins for reactor vessels. At the request of NRR, RES recently modified this program to include materials with low toughness that are representative of those at operating facilities.

RES has just initiated a comprehensive research program to experimentally validate neutron irradiation damage in pressure vessel steels and the associated calculational schemes used to predict radiation damage. This effort is to be part of an overall program being conducted in cooperation with research groups in the US and Europe.

B. Office of Standards Development, Division of Engineering Standards, Structures and Components Standards Branch

SD has assisted NRR in the study of the effects of neutron irradiation and the evaluation of low toughness reactor vessel steels over the past year by providing the services of Dr. P. N. Randall, who is on loan to the Engineering Branch, DOR.

C. Office of Management Information and Program Control, Division of Regulatory Information Systems, Processing and Programming Branch.

MIPC has been assisting NRR in establishing a computer based information system for the storage and retrieval of materials surveillance data.

7. Schedule for Problem Resolution

The major milestones for the Reactor Vessel Materials Toughness Program are:

A. Determination of a preliminary engineering fracture toughness criterion for low toughness reactor vessel materials and appropriate operating conditions, (Tasks 2A and 2B). - December 1977.

B. Obtain information from licensees concerning neutron irradiation surveillance materials, (Part of Tasks 2E and 2F). - December, 1977.

C. Complete generic evaluation of licensee surveillance materials, (Part of Task 2E). - Ocotber, 1978.

D. Completion of the experimental program to determine the fracture toughness of irradiated, low toughness reactor vessesl steels, (RES Task) - November, 1978.

E. Define neutron irradiation effects for reactor vessel materials, (Task 2C) - January, 1979.

F. Determination of the feasibility of reactor annealing, (Task 2D). - January, 1979

G. Complete development of NRC criterion for low toughness reactor vessel materials for operating facilities and revise, if appropriate, the fracture toughness criterionfor facilities not yet licensed for operation. Complete NUREG report presenting results and conclusions of program including management review. - April, 1979.

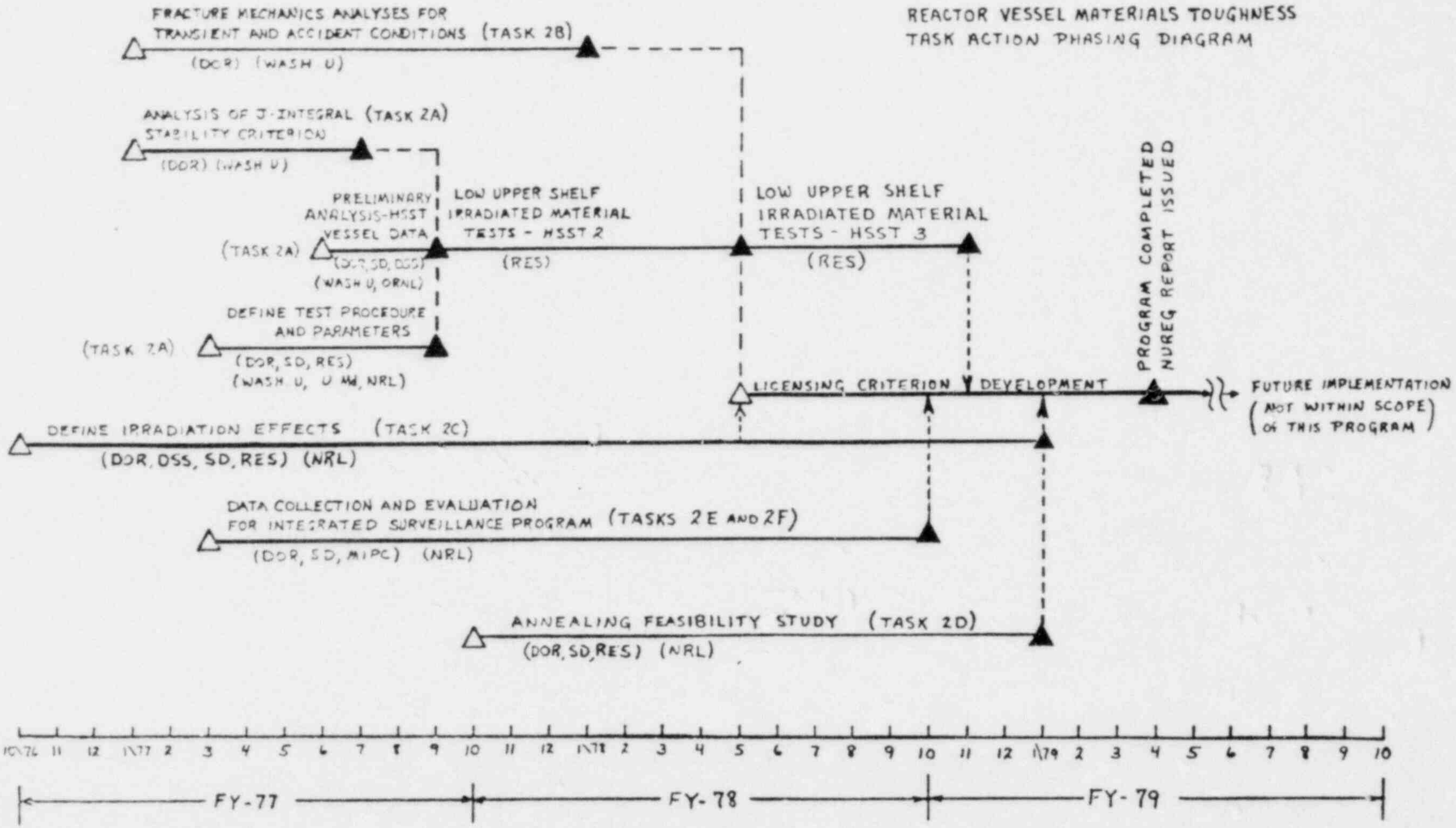
8. Potential Problems

Critical path items for the development and implementation of a licensing criterion for low toughness reactor vessel materials include completion of the fracture mechanics toughness criterion analysis, the definition of appropriate experimental techniques for testing irradiated materials and the subsequent completion of the HSST experimental program for irradiated low toughness materials.

Because the experimental techniques required for completion of the HSST irradiation materials testing have not been used previously for this type testing, there is reason to expect that short periodic delays will be encountered during this program.

Information supplied by some PWR NSSS vendors indicate that because of neutron irradiation damage some reactor vessels will not satisfy current NRC fracture toughness criteria for the postulated main steam line break accident after approximately 20 years of operation. If the results from the analyses described in Tasks 2A and 2B indicate that newly proposed criteria cannot be satisfied, then additional analyses will be necessary and a new task will be defined to consider equipment modifications for certain operating reactors. These equipment modifications will be employed to mitigate the impact of the postulated main steam line break accident and ensure that NRC fracture toughness requirements are satisfied for the postulated accident conditions.

AUG 30 1977



REACTOR VESSEL MATERIALS TOUGHNESS
TASK ACTION PHASING DIAGRAM

8/2/77

CATEGORY A TECHNICAL ACTIVITY NO. A-12

REVISION 0

Title: Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: Dick Snaider, DOR

1. Problem Description:

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2 a number of questions were raised as to the potential for lamellar tearing [1/] and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of North Anna Unit Nos. 1 and 2, the applicant has agreed to raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F prior to reactor coolant system pressurization to levels above 1000 psig. Auxiliary electrical heat will be supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities may be applicable for other PWR plants. It is therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Lamellar tearing may also be a problem in those support structures similar in design to North Anna. This possibility will be investigated on a generic basis.

[1/] Lamellar tearing is a cracking phenomenon which occurs beneath welds and is principally found in rolled steel plate fabrications. The tearing always lies within the parent plate, often outside the transformed (visible) heat-affected zone (HAZ) and is generally parallel to the weld fusion boundary. Lamellar tearing occurs at certain critical joints usually within large welded structures involving a high degree of stiffness and restraint. Restraint may be defined as a restriction of the movement of the various joint components that would normally occur as a result of expansion and contraction of weld metal and adjacent regions during welding. ("Lamellar Tearing in Welded Steel Fabrication", The Welding Institute).

APPROVED BY TASC, AUGUST 19, 1977

The scope of this program is presently limited to PWR steam generator and reactor coolant pump supports. Another program, ASYMMETRIC LOCA LOADS (A-2) will investigate vessel supports as part of its scope. As part of that effort, a review of the need for including BWR vessel supports is being undertaken. As noted in Section 8 of this report, activity A-12 can be expanded to include BWR supports and other PWR support structures if warranted.

2. Plan for Problem Resolution:

A preliminary survey of operating PWR plants was made in May, 1976 to determine the initial scope of this problem. Results indicate that five units have designs similar to North Anna and that 12 units use A36 materials. No plants which were surveyed used the A572 material.

The staff concluded that, depending on the heat treatment of the A36 material, a potential material toughness problem may exist. In addition, it was determined that other materials used in the design of steam generator and pump supports have never been tested to determine toughness properties. Therefore, the potential "toughness problem" may exist for operating plants that did not use A36/A572. As noted above, the potential for lamellar tearing may also exist for certain support structures.

Based on the above, the continuing action plan for resolution of this concern for operating PWRs is as follows:

a. Send a generic letter to all PWR licensees stating NRC concerns and requesting information on the design, materials, fabrication and inspection of the steam generator and reactor coolant pump supports for each plant. (A follow-on letter to BWR licensees may be necessitated by information developed in program A-2).

b. Based on information supplied by the licensees and with the aid of the consultant, categorize the support design and materials as far as practical and select typical designs for further study. DSS/MTEB will concurrently review fracture toughness and possibility of lamellar tearing for PWRs in the CP and OL stages, based on information gathered from the DOR review.

c. Complete preliminary review of typical designs and inform each applicable PWR licensee of the concerns on their particular support system.

d. Utilizing input from consultant, develop and issue specific guidance for resolution of the problems discovered. This will be a joint DSS/DOR task and will result in the issuance of a NUREG document and/or other appropriate document.

   Subsequent case-by-case resolution (implementation) will involve requiring those applicants or licensees for whose facility(ies) a problem exists to either: (1) demonstrate that safety margins are not lower than anticipated or; (2) propose a solution to the problem in accordance with the criteria developed in step d above.

3. NRR Technical Organizations Involved:

   a. Engineering Branch, Division of Operating Reactors. Has lead responsibility for review of data generated from licensee responses, control of and coordination with consultant organization, and will coordinate with DSS in development and issuance of criteria.

      Manpower Estimates: .3 manyears FY 1977, 1.0 manyears FY 1978, .6 manyears FY 1979.

   b. Materials Engineering Branch, Division of Systems Safety. Review information received from operating units and problems identified during review. Coordinate with DOR in development and issuance of criteria.

      Manpower Estimates: .3 manyears FY 1977, .3 manyears FY 1978, .3 manyears FY 1979

   c. Task Manager, Division of Operating Reactors. Has overall responsibility for coordination of DOR and DSS technical tasks and for the developement and issuance of criteria documents.

      Manpower Estimates: .1 manyears FY 1977, .1 manyears FY 1978
      .1 manyears FY 1979

4. Technical Assistance Requirements:

   Technical assistance for the DOR program is required to provide expertise in evaluating the potential for lamellar tearing and low fracture toughness of the support materials. The work will include:

   a. Categorizing the support designs and materials (as far as practical) and selecting typical designs for further study.

   b. Performing a literature search for fracture toughness and lamellar tearing data on the materials in question.

   c. Evaluating typical designs and selecting those designs which may have low fracture toughness or a potential for lamellar tearing.

   d. Evaluating any proposed solutions to problems which may be identified.

      Bids were received from contractors in September and October, 1976 ranging from $122,095 to $170,000. These bids are being evaluated.

5. Interactions With Outside Organizations:

   Individual licensees of PWR facilities and applicants for PWR licenses. All PWR licensees will be contacted to gather information at the commencement of the program. Some licensees will become more deeply involved in this study due to the need for site visits and/or the discovery of material problems at their particular facility(ies). Further interaction will be a function of the results of our review.

   DSS will perform information review during CP and OL stages of review in order to resolve issues prior to licensing.

6. Interaction With Other NRC Offices:

The Office of Standards Development intends to commence, in FY 1979, work on a program involving Fabrication and Examination of Component Supports. Although an effort is presently being made to incorporate specific guidance in the ASME Code, this new program may result in issuance of a Regulatory Guide.

7. Schedule for Problem Resolution:

The major milestones for this program are as follows:

a.	Send generic letter to operating reactors	September 16, 1977
b.	Obtain consultant	October 1, 1977
c.	Select typical designs for further study	February 3, 1978
d.	Complete preliminary review of operating units	May 12, 1978
e.	Receive input on generic resolution from consultant	September 29, 1978
f.	Issue branch technical position paper/NUREG Document	February 25, 1979

8. Potential Problems:

Although this program is presently aimed at a problem known to exist for PWRs, the scope of review could uncover similar problems at BWR facilities or additional PWR component support problems, necessitating a major change in the program.

REVISION 0
SEPTEMBER 16, 1977

Task No.: A-14
Title: Flaw Detection
Lead Responsibility: Division of Systems Safety
Lead Assistant Director: J. P. Knight, Assistant Director for Engineering
Task Manager: Uldis Potapovs, MTEB/DSS

1. Problem Description

The failure probability of a reactor pressure vessel is considered to be sufficiently low to exclude it from consideration as a design basis accident. The rationale for this low probability relies heavily on the maintenance of rigorous manufacturing and quality control standards, adherence to conservatively derived operating limits and effective, regularly repeated inservice inspection. The inspection method must be sufficiently sensitive to assure that all flaws approaching the severity levels used as basis for establishing the margin against fracture during normal operating and transient conditions will be reliably detected particularly in the later stages of plant life, where reduction in fracture toughness of the vessel materials may occur.

Similarly, the integrity of the entire primary pressure boundary and of important safety system components must be assured throughout the plant lifetime. General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary" requires that the design reflect consideration of uncertainties in determining the size of flaws and General Design Criterion 32 requires that the reactor coolant pressure boundary be designed so as to permit periodic inservice inspection.

Flaw detection methods and procedures specified in the present American Society for Mechanical Engineers (ASME) inservice inspection (ISI) rules leave uncertainties concerning the smallest size defect which can be reliably detected by NDT in various parts of the pressure boundary. Similarly, significant uncertainties are known to be associated with dimensional characterization of identified defects. The ability to detect and adequately size flaws is essential in assuring continued integrity of the reactor coolant pressure boundary and in assessing the margin against failure under various plant conditions throughout the full life of the plant.

2. Plan for Problem Resolution

a. Approach

The problem will be resolved by assessing the flaw detection limits which can be achieved using current ASME Code Inservice Inspection (ISI) rules, defining priority areas where improvements

APPROVED BY TASC, SEPTEMBER 6, 1977
TASC COMMENTS INCORPORATED, SEPTEMBER 16, 1977

are needed, following development of new and improved flaw detection methods, and implementing procedures and inspection requirements capable of providing the necessary improvement. The technical bases for the anticipated revisions in inspection requirements will be developed through currently existing and planned NRC sponsored programs in combination with feedback from related programs funded by Energy Research and Development Administration (ERDA) Electric Power Research Institute (EPRI), Pressure Vessel Research Council (PVRC) and similar work conducted by nuclear steam system supply (NSSS) vendors. The results of these programs will be evaluated as they become available, selecting those parameters which offer significant improvements to current practice and effecting appropriate changes to applicable ISI rules. In addition to the information gained from these programs, field experience from past preservice and inservice inspections will be analyzed and factored into the planned procedure and rule revisions.

It is anticipated that two parallel efforts, one aimed at providing recommendations for the revision of existing ASME Code rules, the other at issuing a series of appropriate regulatory guides will be required. The main purpose of the regulatory guides will be to achieve timely implementation of the necessary improvements since revisions to the ASME Code may take considerably longer to accomplish. The regulatory guides will be periodically revised as new information is developed or as specific requirements are incorporated into the ASME Code.

b. End Product

The end product will be NUREG report summarizing accomplishments under this task and recommending any future action based on reassessment of current needs in the light of these accomplishments. The report will be issued approximately three years from the task initiation.

Yearly summary reports will also be issued to critically evaluate the task progress.

Additionally, at least two branch position will be issued and technical input will be provided for two basic regulatory guides to achieve timely and effective utilization of significant task findings in the licensing process.

c. Tasks

C-1 Evaluation of ultrasonic testing limitations and potential improvements.

A critical assessment of the ASME Code UT requirements will be made with emphasis on parameters affecting flaw detection and characterization capabilities. Comparisons with other state-of-the-art procedures supplemented by confirmatory laboratory investigations will be utilized in developing the basis for conclusions concerning licensing requirements and their relation to existing code rules. Of specific concern are equipment standardization and calibration, scanning requirements and defect evaluation criteria as percent of signal amplitude or background level response.

Supporting investigations in this area are currently under way at Oak Ridge National Laboratory (ORN) as part of a larger material investigation program funded by the Structures and Components Branch of Office of Standards Development. Three subtasks are identified based on this program:

C1.1 Review of ASME Code specified inspection equipment and procedures variables.

C1.2 Confirmatory laboratory evaluation of testing parameters.

C1.3 Assessment of foreign ISI codes and related standards.

Information developed under this program will provide input to development of licensing requirements for flaw detection and characterization (Task C-6) as well as recommendations to industry code writing groups.

The specific effort will include evaluation of technical information developed under the identified subtasks in the light of existing licensing requirements and making appropriate recommendations where changes in these requirements are indicated. Specific review assignments will be made consistent with the scheduled completion dates of the above subtasks (paragraph 7.2).

C-2 Development of licensing criteria for improved and alternate flaw detection and characterization techniques - NRC sponsored programs.

This task will evaluate and analyze useful information from several ongoing research programs currently funded by RSR and assure that such information is integrated into the development of licensing requirements under Task C-6. At the present time three subtasks are identified based on ongoing or planned activities.

C2.1 Synthetic apperture imaging - technique development and adaption for field use.

C2.2 Improved UT penetration and signal to noise ratio.

C2.3 Acoustic emission development.

Additional subtasks may be identified as this program proceeds and new needs and priorities develop.

Work under this task will include a critical evaluation of information generated under the various subtasks, identification of findings which have a potential for improving flaw detection capability and recommending specific changes of or additions to existing licensing criteria to accomplish these improvements. Specific review assignments will be made consistent with the estimated completion dates of the above subtasks (paragraph 7.b).

C-3 Monitoring and assessment of results from ERDA/industry sponsored programs.

A significant amount of research and development in flaw detection is being funded through other agencies and private organizations or corporations. The projects range from general optimization of testing methods and procedures to development of highly specialized techniques for specific testing needs and applications. The progress of the more significant of these programs will be monitored and the results reviewed as they become available through published reports or thru direct communications with the organizations involved. Significant findings will be assessed in relation to ongoing NRC studies and factored into the development of licensing requirements in this area (Task C-6). The more important of these projects are identified under the following subtasks:

C3.1 Ultrasonic testing optimization programs - ERDA/EPRI.

C3.2 Development of acoustic emission - ERDA/EPRI.

C3.3 Development of NDE techniques for BWR feedwater nozzle inspection - EPRI.

C3.4 Evaluation of current UT practice (round robin pipe crack evaluation) - EPRI.

3.5 Evaluation of current UT practice (heavy plate round robin tests) - PVRC.

Specific review assignments will be made consistent with the data availability from the above programs.

C-4 Evaluation of field experience in flaw detection.

Preservice and inservice inspections of nuclear systems and components have been routinely conducted in operating plants since the publication of ASME Section XI in 1970. In addition many special or augmented examinations have been made in order to comply with NRC requests based on identified generic problems such as BWR pipe and feedwater nozzle cracking.

These inspections have disclosed some significant defects and have failed to disclose others. Information gained from these inspections in combination with similar data being currently generated can provide a good basis for the assessment of flaw detection capabilities, limitations and identify areas for potential improvement. Similarly such service data should be helpful in providing technical basis for establishing realistic indication recording and evaluation levels and provide important contribution to the development of licensing requirements.

As a part of this task, the data from scheduled or special examinations which have produced significant flaw indications will be evaluated to establish a data base for analysis of inspection parameters. Emphasis will be placed on those instances where flaw size/test method correlation is possible because of removal and destructive examination of identified defects. This review will include the known instances of BWR feedwater/CRDM return nozzle cracking and pipe cracking where defects have been detected by NDE techniques and subsequently verified by alternative means or grind-outs. Close coordination with Technical Activity A-10, "BWR Nozzle Cracking" will be necessary to accomplish these objectives.

The results from this task will provide a significant input to Task C-6 (development of licensing requirements). Specific work assignments will be made consistent with the schedule shown in paragraph 7.d.

C-5 Development of criteria for evaluating conditions of limited inspectability and requests for relief from applicable inspection requirements.

This task is aimed at developing a rationale for evaluating flaw detection limits and inspection capabilities for specific RCPB component types/arrangements and defining acceptable alternate inspection techniques when code-required examinations are not feasible or are only partially effective.

The task will include developing of a standard list of components and weld joint/material configurations which fall into this category and establishing alternate basis for acceptance. This list will be used in the evaluation of initial or updated ISI programs where specific relief is requested from applicable code requirements.

C-6 Development of licensing requirements for flaw detection and characterization.

The purpose of this task is to develop specific licensing requirements in the area of flaw detection based on recognized needs, potential improvements identified under other subtasks of this program or deficiencies in currently applicable ISI rules. Technical basis for the proposed licensing requirements will be developed under other parallel tasks of the overall flaw detection problem. Currently identified subtasks are as follows:

C6.1 Technical input for Regulatory Guide ___, "Supplementary Procedures for Ultasonic Testing of Reactor Vessel Welds During Inservice Inspection."

C6.2 Technical input for Regulatory Guide ___, "Supplementary Procedures for Ultrasonic Examination of Piping."

C6.3 Technical input for Revision 1 of Regulatory Guide issued under C6.1.

C6.4 Technical input for Revision 1 of Regulatory Guide issued under C6.2.

C6.5 Basis for NRC endorsement of ASME Section XI rules.

C6.6 Materials Engineering Branch Position on Augmented Inspection Requirements for BWR Feedwater Nozzle Radii.

C6.7 Materials Engineering Branch Position on Augmented Inservice Inspection of BWR Coolant Pressure Boundary Piping.

Subtask C6.5 is intended to utilize the information developed from the parallel studies and state-of-the-art reviews and make specific recommendations concerning ASME Code rule endorsement by NRC. The ASME Section XI ISI standards must be incorporated by reference into the Federal Register before they can be used by an applicant or licensee as a basis for satisfying licensing requirements. Since revisions to ASME Code are made bi-annually, changes in the rules relating to flaw detection must be evaluated by NRC on about the same frequency and appropriate recommendations made concerning endorsement of these revisions. This subtask describes a current NRC activity which has been integrated under this Task for the Task duration.

C-7 Input to ASME Code

In parallel with the regulatory guides and branch positions discussed under Task C-6, information derived from the various R & D activities and surveys carried out under this program will be made available to ASME Section V and XI working groups and committees through the individual NRC representatives serving on these committees. Specific recommendations will be made regarding revisions of existing code rules or the need for additional inspection requirements supported by technical basis developed under this program. The extent to which these recommendations are adopted by the ASME Code and incorporated into referenced rules and procedures will determine the scope of the regulatory guides proposed under Task C-6.

C-8 Task management.

Because of the broad and far-reaching scope of the overall program, and extensive NRC and industry organizational interface requirements, the need of continuous assessment or evaluation of task outputs and anticipated need for redirection of effort based on such evaluation, effective management of this task will require considerable expenditure of technical and administrative manpower. The results of each task will be evaluated as they become available and related to the overall problem objectives. The scope and emphasis of the tasks will be reassessed in the light of these evaluations and appropriate recommendations made to expedite the achievement of problem objectives.

In addition to periodically furnishing specific recommendations and directions to organizations involved in the individual tasks based on analysis of the overall technical output, yearly progress reports will be prepared summarizing the status of all tasks and relating them to the overall problem solution. A complete summary report will be issued three years from the problem initiation. This report will assess the current state-of-the-art in flaw detection based on output from the individual tasks, summarize progress made towards the problem resolution and make specific recommendations for future needs.

3. NRR Technical Organizations Involved

Division of Systems Safety/Materials Engineering Branch
Division of Operating Reactors/Engineering Branch

a. Task C-1 Evaluation of UT limitations and potential improvements, including subtasks C1.1 thru C1.3.

Most of the investigative work under this task will be performed by ORNL under existing contract to the Office of Standards Development. Input from MTEB/DSS and Eng. Br./DOR will be required to identify specific NRR needs in this area and to recommend priorities. Input will be provided thru participation in meetings with OSD and ORNL and thru review progress reports submitted by ORNL. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.2	.3	.2	.1 (man-years)
Eng. Br./DOR:	.2	.3	.2	.1

b. Task C-2 Development of improved or alternate flaw detection methods, including subtasks C2.1 thru C2.3.

The investigative work under this task will be performed under existing contracts administered by the Office of Reactor Safety Research. MTEB/DSS and Eng. Br./DOR will provide input to this task by participation in the NDE Research Review Group which provides direction to the individual research projects. In addition progress reports from these programs will be objectively reviewed and related to licensing requirements. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.2	.3	.3	.3
Eng. Br./DOR:	.3	.4	.4	.4

c. Task C-3 Monitoring ERDA/industry sponsored programs, including subtasks C3.1 thru C3.9.

MTEB/DSS and Eng. Br./DOR will evaluate the results of the ongoing programs identified under subtasks C3.1 thru C3.9 thru review of results published in open literature (specific review assignments will be made) and information exchange meetings with organizations involved. Results will be related to identified licensing needs. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.2	.4	.4	.4
Eng. Br./DOR:	.3	.5	.5	.5

d. Task C-4 Evaluation of field experience.

Eng. Br./DOR will have the lead reponsibility for this task. The scope of this task includes quantitative assessment of flaw detection capability based on currently available inservice inspection records at operating plants. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.2	.2	.2	.2
Eng. Br./DOR:	.3	.5	.5	.5

e. Task C-5 Criteria for conditions of limited inspectability - basis for relief from ISI requirements.

The responsibility for this task will be shared by MTEB/DSS and Eng. Br./DOR. The scope will include developing a basis for exempting systems, components or individual welds from specified inservice inspection requirements and defining acceptable alternate means of assuring continued integrity of these items. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.2	.3	.2	.1
Eng. Br./DOR:	.4	.5	.2	.1

f. Task C-6 Development of licensing requirements, including subtasks C6.1 thru C6.4.

Based on current requests from NRR, OSD has initiated work on two regulatory guides in the area of flaw detection. Additional guides may be necessary as the work proceeds. It is expected that MTEB/DSS and Eng. Br./DOR will work very closely with OSD in developing the technical basis for regulatory positions stated in these guides and draw heavily from the other tasks under this problem for the necessary information. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.3	.5	.5	.5
Eng. Br./DOR:	.4	.6	.6	.6

g. Task C-7 Input to ASME Code.

MTEB/DSS, MEB/DSS and Eng. Br./DOR have representatives on working groups, task groups, subgroups and subcommittees of ASME which are directly involved in writing flaw detection procedures, specifying examination requirements and establishing acceptance criteria. A coordinated effort by all NRC representatives will be necessary to provide constructive input to these groups and assure that our concerns are understood and are properly considered at all stages of the ASME rule making process. Incorporation of important flaw detection requirements into the code rules at this stage is the most effective and efficient means of achieving the overall objectives of this problem. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.2	.3	.3	.3
MEB/DSS:	.1	.1	.1	.1
Eng. Br./DOR:	.2	.3	.3	.3

h. Task C-8 Task management.

DSS has the lead responsibility for the overall problem management. Included in the scope of the problem management will be continuous monitoring of the individual tasks and subtasks, coordination of results towards uniform licensing positions, initiation of new subtasks based on identified needs and redirection of effort of the ongoing activities if required. The scope also includes preparation of yearly progress reports and a final report at the problem completion. Manpower estimates are as follows:

	1977	1978	1979	1980
MTEB/DSS:	.3	.5	.5	.5

4. Technical Assistance Requirements:

There are no currently active technical assistance contracts under the cognizance of NRR which are directly related to the scope of this problem. Extensive use will be made of output from existing contracts administered by OSD and RSR which are described in other sections of this outline. Eng. Br./DOR has "on-call" technical assistance agreements with Sandia and ORNL which may be utilized for the resolution of specific short range problems. It is likely that a need for additional technical assistance will develop as the work progresses under the various tasks of this problem.

5. Interactions with Outside Organizations

Because of the broad scope of this problem and the large number of organizations involved, extensive interactions between these organizations will be necessary in working toward effective resolution of this problem. Most of these interactions will develop through implementation of Tasks C-3, "Monitoring and Assessment of Results from ERDA/Industry Sponsored Programs," C-4, "Evaluation of Field Experience in Flaw Detection," and C-7, "Input to ASME Code." As this work proceeds it may also be desirable to develop an effective information exchange with foreign regulatory and inspection organizations. Such information change would be useful to explore alternate approaches specifically identified problem areas in flaw detection.

6. Assistance Requirements from Other NRC Offices

a. Office of Reactor Safety Research

Task C-2 Development of licensing criteria for improved and alternate flaw detection and characterization techniques.

The Metallurgy and Materials Branch/RSR is currently funding four separate contracts with different organizations which are expected to contribute directly towards the resolution of this problem. Other research needs may be identified as the work progresses. In that case work and funding requests will be initiated under a new subtask. The existing contracts are identified below:

Battelle Pacific Northwest Laboratories

Laboratory program to develop acoustic emission - flaw relationships for inservice monitoring of nuclear pressure vessels.

University of Michigan

Development of synthetic apperture focusing technique for ultrasonic testing.

National Bureau of Standards

Development of deeper penetration and high sensitivity UT flaw evaluation equipment.

Southwest Research Institute

Adoption of synthetic apperture focusing technique to field use.

b. Office of Standards Development

Task C-1 Evaluation of UT limitations and potential improvement.

The Structures and Components Standards Branch of OSD is currently funding at Oak Ridge National Laboratory analytical and investigative work aimed at defining the significant variables which affect flaw detection capabilities using current ASME Code procedures. Specific recommendations from these studies will provide input in developing augmented requirements for flaw detection.

In addition, OSD is responsible for preparing the regulatory guides specifically identified under Task C-6.

c. Office of Inspection and Enforcement

Input will be required from OIE Division of Reactor Construction and Reactor Operation for Tasks C-1, C-2, C-4 and C-6 to assure full and effective utilization of field experience in assessing the flaw detection limits which can be achieved using the currently specified NDE requirements and in development of new licensing requirements in this area. This input will be in the form of OIE participation in planning and information meetings for the identified tasks, their review and comment on proposed licensing positions and their response to specific requests for ISI information from operating plants.

7. Schedule for Problem Resolution (see attachment 1 for task summary schedule)

a. Task C-1 Evaluation of UT limitations and potential improvements.

Subtask C1.1	Complete April 1978
Subtask C1.2	Progress Report April 1978 Complete January 1979
Subtask C1.3	Progress Report April 1978 Complete January 1979

b. Task C-2 Development of licensing criteria for improved and alternate test methods (NRC sponsored)

Subtask C2.1

Demonstrate UT imaging in thick sections	October 1978
Perform flaw detection probability study on large specimens	October 1979
Test system on full size mockup or actual component	October 1980

Subtask C2.2

Demonstrate capability of high-power UT system in field	October 1978

Subtask C2.3

Propose AE-material property-flaw size severity model	October 1977
Establish model and differentiate defect signals from other noises	October 1978
Validate model in lab for application to reactor monitoring	October 1979

c. Task C-3 Monitoring ERDA/industry sponsored programs

Because of the large number of subtasks involved, redundancy and continuous nature of many of these programs, specific milestones can not be established at this time. Completion dates for specific research reviews will be established as such assignments are made after initiation of work on the overall problem. The review schedule will be consistent with the milestones of Task C-6.

d. Task C-4 Evaluation of field experience

Evaluate effectiveness of UT techniques in detecting FW nozzle radii and bore cracks in BWR's	December 1977
Evaluate UT effectiveness in examining vessel closure studs	February 1978
Evaluate the results from IE independent measurements program	July 1978
Based on review of preservice and inservice inspection information propose recommended weld configurations and geometry tolerances for optimum inspectability	December 1978

e. Task C-5 Develop criteria for conditions of limited inspectability - basis for relief from ISI requirements

Complete a table of typical ISI relief requests and provide standard basis for waiving inspection requirements or accepting alternate examinations	February 1978
Update table	July 1978

f. Task C-6 Development of licensing requirements

Subtask C6.1	Complete July 1978
Subtask C6.2	Complete December 1978
Subtask C6.3	Complete July 1979
Subtask C6.4	Complete December 1979
Subtask C6.5	Continuous throughout task completion
Subtask C6.6	Complete January 1978
Subtask C6.7	Complete March 1978

g. Task C-7 Input to ASME Code - Continuous throughout task completion.

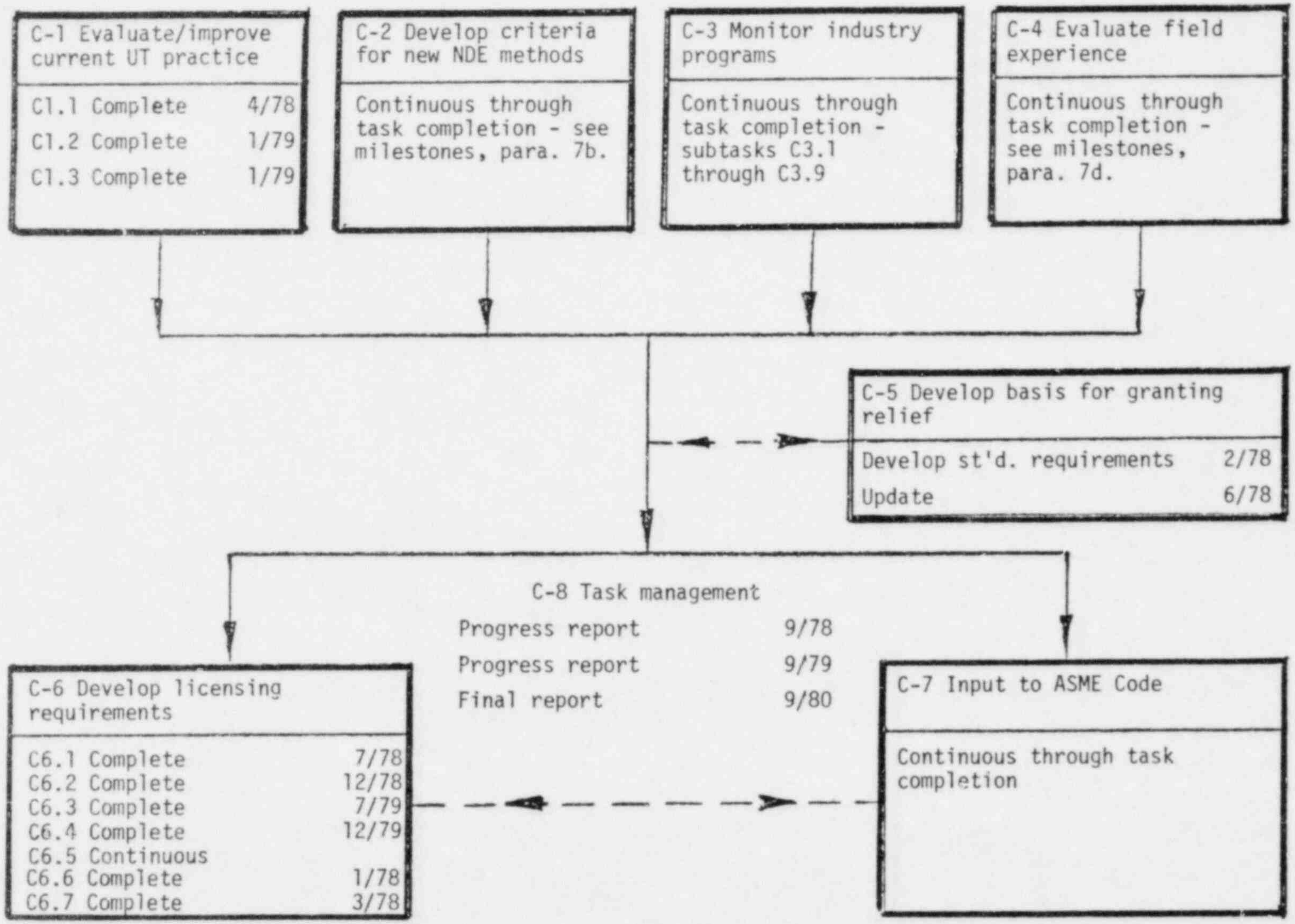
h. Task C-8 Problem management

First progress report	September 1978
Second progress report	September 1979
Final report	September 1980

8. Potential Problems

Difficulties are anticipated in achieving timely NRC endorsement of ASME Section XI revisions. At the present time there is a 2-year lag between published Section XI addenda and NRC endorsement of these documents. It is hoped that improvement in this area can be made thru effective participation of NRC representatives in the ASME rule-making process, early identification of unacceptable rule revisions and close coordination of outstanding issues between the NRC technical organizations involved in the endorsement action.

FLAW DETECTION - TASK SUMMARY/SCHEDULE



Attachment 1

AUG 31 1977

REVISION 0

CATEGORY A TECHNICAL ACTIVITY NO. A-15

Title: Decontamination

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

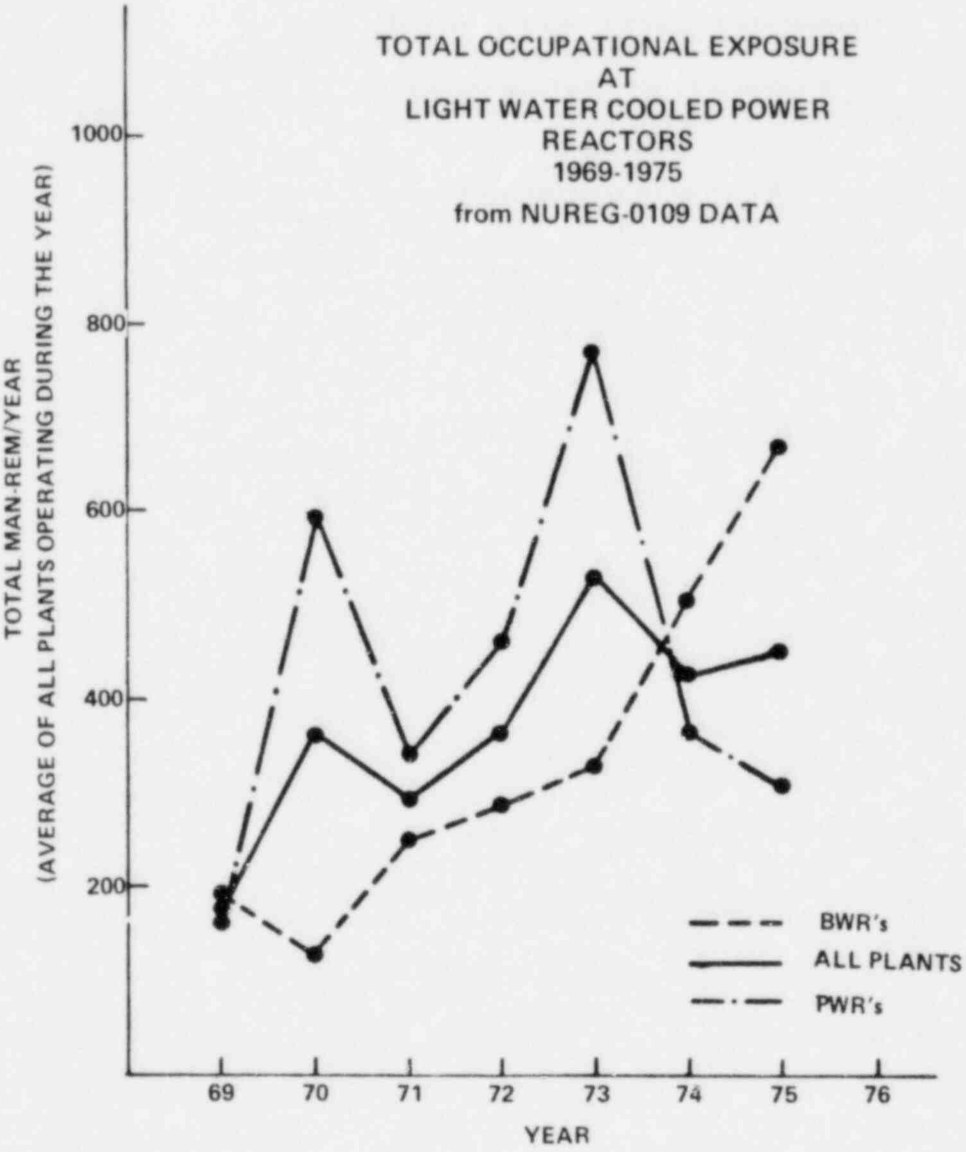
Task Manager: Paul W. O'Connor, DOR

1. Problem Description:

The presence of a layer of highly radioactive corrosion products adhering to the interior surfaces of the primary cooolant system has, in some cases, prevented licensees from carrying out some of the in-service inspections required by their technical specifications. These inspections and the prompt repair of any defects discovered during their conduct, have provided continuing assurance that incipient failures in facility components important to safety will be detected and repaired long before they develop to an extent that would constitute a significant hazard to the health and safety of the public. Because of the importance of the system and components being inspected, an approach must be developed to permit continued inspections while at the same time taking proper consideration of personnel exposure considerations.

The benefits achieved due to the Commission's requirements relating to inspection and repair are clearly evident and these activities will continue to be required of all licensees because the integrity of the primary system pressure boundary is foremost in our safety approach. However, as experience is accumulated and interpreted on operating reactors it has become evident that the radiation levels in the vicinity of the primary coolant system of operating reactors are generally increasing with the age of these facilities. It has also become evident that the occupational radiation exposures received by personnel conducting required inspections, repairs and maintenance on primary system piping and components are increasing and may eventually limit the efficacy of these actions. Figure 1 indicates this trend toward higher average personnel exposures through 1975 as reported in NUREG-0109 (Occupational Radiation Exposure at Light Water Cooler Power Reactors 1969 - 1975).

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED AUGUST 31, 1977



TOTAL OCCUPATIONAL EXPOSURE
AT
LIGHT WATER COOLED POWER
REACTORS
1969-1975
from NUREG-0109 DATA

A major portion of this occupational exposure is received during planned maintenance, inspections and unanticipated repair operations, many of which are carried out to satisfy Commission requirements. These personnel exposures can be expected to increase in the future as a result of augmented inservice inspection requirements and major generic repair projects such as BWR nozzle crack repairs and PWR steam generator repair and replacement.

Exposures from such required functions are often caused by the presence of long-lived activated corrosion products, for example, Co and Co , which have been deposited in a tightly adhering layer of metallic oxide at various locations throughout the entire primary coolant system. This layer of contamination is present as a fairly uniform deposit. Some "hot spots" of concentrated activity are present at various locations in the primary system, but in general the dispersed nature of the radioactivity makes exposure reduction by shielding impractical.

The increased occupational exposure caused by activated corrosion products present throughout the primary coolant system is a matter of significant concern because of the following:

a. Increased exposure rates in conjunction with poor accessability on older reactors may prevent licensees from carrying out required inservice inspections,

b. Repairs and modifications carried out in high radiation fields could limit the availability of specially qualified employees such as welders and inspectors when they reach their quarterly or annual radiation exposure limits,

Some reduction in the exposure received by plant personnel may be achieved by traditional radiation protection methods of reducing the time of the exposure, shielding contaminated components during the operation of concern, through the use of limited localized chemical or mechanical decontamination, or by employing remote means to carry out various reactor operations. In general, utilization of these traditional methods of exposure control has not stopped the trend towards increased exposure to operational personnel as they are often not viable methods of accomplishing the safety objectives.

In consideration of these developments, considerable interest has been shown in the development of methods to remove the contamination from reactor primary systems prior to inspection, testing, maintenance or other facility functions. The NRC will be faced with making decisions in several areas which relate to the appropriateness, the effectiveness and the overall safety considerations of the decontamination of operating light water reactors.

NRC is currently reviewing a licensee submittal for the decontamination of a boiling water reactor primary coolant system utilizing a strong chemical decontaminant (Dresden 1). A test program has been proposed that will try 4 diversified methods of chemical decontamination on the 4 primary coolant loops of a pressurized water reactor (Indian Point 1). The chemical cleaning process being proposed for the Dresden 1 facility is indicative of the different technical questions that the staff must address; e.g., what are the effects of the chemical substance on the materials of the primary system?

In the near future licensee requests are anticipated for the replacement, retubing, or other major maintenance of PWR steam generators that have been damaged due to the denting phenomena (Turkey Point and Surry). The primary side of these steam generators is highly contaminated and any maintenance or removal operation will be a complex task with extensive radiological considerations related to occupational exposures and a potential for significant off site considerations related to waste storage, transportation, and disposal.

Inasmuch as these decontamination programs may be proposed in order to gain access for repair or modifications related to safety, NRC review of decontamination requests will need to be accomplished on a timely basis. Since this is a technical area where the NRC staff has limited expertise and experience with commercial nuclear power plants, it will be difficult to establish the necessary meaningful guidance and criteria for the decontamination of operating reactors in advance of these anticipated licensee submittals.

2. Plan for Problem Resolution:

There are currently at least five methods of decontamination that may be proposed for use in the primary coolant systems or components of operating reactors. These include:

a. Strong Chemical Decontamination (proposed by Commonwealth Edison for the decontamination of Dresden Unit 1 provides high decontamination factor (DF) - long outage time).

b. Weak Chemical Decontamination (developed and used by Atomic Energy of Canada Ltd., requires very little outage time - low DF).

c. On line chemical decontamination additions to the primary coolant system (hydrazine and peroxide additions).

d. Hydraulic methods (High pressure jets) can be very useful for local decontamination of components prior to maintenance and repair if the contamination is not a fixed oxide scale.

e. Mechanical Decontamination (sand blasting using zirconium oxide).

Each of these methods of decontamination has advantages and disadvantages that must be identified and weighed by the staff before approval can be granted to decontaminate.

The staff, by the use of technical assistance contracts, will have these proposed decontamination methods reviewed to assure their efficacy and safety. The program is being phased so as to assure that the information necessary to assist the staff in making its decisions is available within the time frame tentatively identified by licensees as to possible future decontamination requests.

The initial phase of the Decontamination task action plan requires the completion of the staff's review of the strong solution decontamination of the lead Boiling Water Reactor, Dresden Unit No. 1 and staff review of the radiological considerations related to steam generator decontamination and removal. In order to complete this first phase before licensing decisions are needed, FY 1977 technical assistance contracts must be initiated immediately to identify and resolve any staff concerns relative to materials compatibly and radiological impact of strong chemical decontamination and steam generator decontamination. The proposed technical assistance contracts for FY 1977 and early FY 1978 will provide this support as outlined in Section 4.

In later FY 1978 and FY 1979 the emphasis in the program will shift to a review of alternative methods of decontamination that may be used in place of strong chemical decontamination or used after an initial strong decontamination.

The primary NRC concern related to the decontamination is to assure that the decontamination method does not degrade the integrity of the primary coolant system boundary. This consideration involves both immediate degradation during the decontamination and latent effects that could cause degradation during subsequent operation of the reactor. The assurance of compatibility of the materials present in the reactor primary coolant system boundary with the decontamination method will require various staff actions:

a. An inventory of all materials that are to be decontaminated must be compiled from fabrication records, inspection of the primary coolant system and a review of the as-built facility drawings,

b. The availability of applicable data to determine the effect of the actual decontamination procedure to be used on each of the materials to be decontaminated,

c. The development of guidance for an inspection program that includes a "baseline" inspection and appropriate followup inspections to provide a high degree of confidence that no degradation has occurred.

In addition to the materials compatibility of a proposed decontamination method, the staff must be assured that each approved method is acceptable from the standpoint of occupational exposure during decontamination and radwaste handling. Each process must also be determined acceptable in assuring that the health and safety of the public is not adversely affected by subsequent onsite waste processing or offsite waste transportation for disposal.

The resolution of this Category A Task will be the publication of a NUREG Document on Decontamination and recommendation of preparation of a Regulatory Guide which identifies the methods of decontamination that are acceptable to the staff and which establishes the materials testing criteria that must be satisfied to qualify each decontamination method for licensing approval.

3. NRR Technical Organizations Involved:

   a. Operating Reactors Branch #2. Task management for this Technical Activity will be provided by Paul W. O'Connor.

      Manpower Estimates: FY 77 - 0.1 man-year
      FY 78 - 0.1 man-year
      FY 79 - 0.1 man-year

   b. Engineering Branch, Division of Operating Reactors. Has lead responsibility for evaluating decontamination methods for materials compatibility and for reviewing and approving proposed pre-service inspection and operational surveillance programs to assure that primary system integrity is not compromised by decontamination.

      Manpower Estimates: FY 77 - 0.3 man-year
      FY 78 - 1.0 man-year
      FY 79 - 2.0 man-year

   c. Environmental Evaluation Branch, Division of Operating Reactors. Has responsibility for evaluating the acceptability of radiation exposures associated with decontamination methods and onsite and off-site considerations radioactive waste processing, storage and transportation. The Environmental Evaluation Branch is also responsible for resolving radiological concerns related to steam generator decontamination, repair, replacement and disposal. (This effort will result in an input to the Task Action Plan for Westinghouse Steam Generator Tube Integrity, A-3).

      Manpower Estimates: FY 77 - 0.1 man-year
      FY 78 - 1.25 man-year
      FY 79 - 0.75 man-year

   d. Effluent Treatment Systems Branch, DSE
      Materials Engineering Branch, DSS
      Radiological Assessment Branch, DSE

These branches have related long term licensing interests in Technical Activity A-15 due to their Category B Technical Activities related to their ongoing Construction Permit (CP) and Operating License (OL) reviews. A description of each branch's area of interest follows:

The Effluent Treatment Systems Branch will be involved in Task A-15 to the extent necessary to determine whether methods of decontamination proposed for operating reactors will be compatible with the effluent treatment systems under review by ETB for CP and OL plants. (Technical Activity B-34).

The Materials Engineering Branch will review decontamination techques proposed for operating reactors to assure that these techniques are compatible with materials proposed for use in plants under CP and OL review.

The Radiological Assessment Branch will monitor Technical Activity A-15 to determine whether the decontamination of operating reactors suggests any design concepts related to contamination reduction or ease of decontamination, that should be incorporated in the design of plants during their CP and OL review. (Technical Activity B-34).

4. Technical Assistance Requirements:

a. Contractor: Brookhaven National Laboratory
Lead DOR Branch: EB
Funds required: FY 77 - $25 K
FY 78 - $100 K (projected)
FY 79 - $100 K (projected)

Carry out a broad review of the materials testing programs conducted by General Electric Company, Dow Chemical Company, Commonwealth Edison in support of Strong Chemical Decontamination of Boiling Water Reactors.

Review AECL weak chemical decontamination methods to determine materials compatibility and decontamination factors associated with this method.

Review EPRI and ERDA programs aimed at developing strong and weak decontamination methods and any other available information to assure that adequate consideration is being given to materials compatible with decontamination solvents.

b. Contractor: Battelle-Pacific Northwest Laboratories
Lead DOR Branch: EEB
Funds required: FY 77 - $25 K
FY 78 - $50 K (projected - contractor may change)
FY 79 - $50 K (projected - contractor may change)

Carry out a review of current decontamination methods and steam generator decontamination, repair, and replacement techniques to determine advantages and disadvantages of existing methods in relation to onsite personnel exposures during decontamination, solution and waste handling radiological problems, offsite releases associated with decontamination and waste processing, radiological considerations of ultimate disposal of decontamination residues.

5. Interaction with Outside Organizations

a. Electric Power Research Institute (EPRI)

EPRI has an extensive 3 to 4 year program underway to develop methods of increasing reactor availability, reduce radiation exposure, and assure materials compatibility of decontamination methods. This program is aimed at developing an acceptable online or weak chemical decontamination method that can provide a decontamination factor of 3-10 with a minimum of down time. NRC should follow the EPRI program closely inasmuch as it is the method that would be most attractive to licensees if it can be developed and licensed.

b. Atomic Energy of Canada, Limited

An extensive body of successful weak chemical decontamination data has been developed in the Canadian reactor program. AECL has licensed a private firm, London Associates, to market its developed decon process. NRC should review the Canadian method so that we can be ready to make licensing decisions should it be proposed by any NRC licensees.

c. Other Foreign Decontamination Experiences

A review of Japanese and European decontamination experience should be carried out to take advantage of any existing background data available.

d. Energy Research and Development Administration (ERDA)

ERDA is funding $8,000,000 of the Commonwealth Edison Strong Chemical decontamination of Dresden 1 and is also funding studies of PWR decontamination and steam generator decontamination at Indian Point. These programs will have both generic relevance and case related licensing actions associated with them and should be monitored closely by NRC.

e. Naval Reactors Program

An extensive amount of information exists in classified form in the Naval Reactors Program. Access to this data would be beneficial in allowing NRC to meet its licensing commitments on a timely basis and a NRC - Naval Reactor technical information exchange should be undertaken to allow NRC to make maximum use of this data consistent with national security constraints.

f. Advisory Committee on Reactor Safeguards

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

6. Assistance from Other NRC Offices

a. Division of Reactor Safety Research

No additional assistance is necessary at this time from other NRC Offices for this technical activity. However, as the activity progresses an integral part of the activity will be to identify any areas of additional confirmatory research that may be necessary. To this end the Division of Reactor Safety Research will be kept fully informed of all activities carried out under this task and their input relative to the need for additional research will be solicited.

b. NRC Office Standards Development

Since the end product of this program will be to provide guidelines to industry relative to acceptable methods of decontamination, it is anticipated that the assistance of the Office of Standards Development will eventually be required. Consequently, OSD will also be kept informed of all activities carried out under this task.

The Office of Standards Development (OSD) has met with the Division of Operating Reactors and has expressed a desire to assist DOR in developing Standards and Regulatory guides on the subject of decontamination. Since the end product of this program will be to provide guidelines to industry relative to acceptable methods of decontamination; it is anticipated that the assistance of the Office of Standards Development will eventually be required. The OSD will also be kept informed of activities carried out under this task.

c. Office of International Programs

Interaction with the Canadian Program and other foreign decontamination programs will require input from the Office of International programs to assure that we receive maximum cooperation from the foreign governments involved.

7. Schedule for Problem Resolution

The major milestones for this technical activity are as follows:

1.	Dresden Materials Testing Report Submitted	7-77
2.	Dresden Decontamination Licensing Resubmittal	8-77
3.	BNL Completes Preliminary Review of Dresden Decontamination Program and Materials of Construction	10-77
4.	EPRI "Evaluation of Hydrogen Peroxide Additions to PWRs Prior to Refueling" completed	10-77
5.	BNL Completes Review of Dresden Decontamination Procedures and Materials Testing Results	12-77
6.	PNL Completes Review of Decontamination Procedures to Identify Onsite Radiological Concerns	1-78
7.	PNL Completes Review of Waste Handling and Disposal Considerations related to Decontamination	2-78
8.	BNL Completes Preliminary Review of Materials Aspects of EPRI-ERDA Programs in Decontamination	2-78
9.	PNL Completes Review of Radiological Considerations of Steam Generator Decontamination, Repair, and Replacement	3-78
10.	Staff Position on Radiological considerations Related to Steam Generator Decontamination	3-78
11.	NRC Approval of Dresden Materials Test Program	5-78
12.	Tentative ERDA - Consolidated Edison Four Loop Decontamination Studies at Indian Point 1	7-78
13.	NRC Approval of Dresden Pre-operational Inspection and Inservice Inspection Programs	9-78
14.	Dresden Strong Chemical Decontamination	11-78
15.	BNL-PNL Final Report "Operating Reactor Decontamination"	1-79

16.	Possible Turkey Point Steam Generator Mechanical Decontamination	1-79
17.	Development of Staff Position on Decontamination of LWRs	12-79
18.	Request to Office of Standards Development to issue Regulatory Guide	12-79
19.	Publication of a NUREG REPORT on Decontamination	3-80

Non-NRC Milestones

20.	General Electric Completes EPRI Study Re: BWR Radiation Assessment and Control	5-80
21.	Babcock & Wilcox Completes EPRI Study Re: PWR Radiation Control	6-80
22.	Westinghouse Completes EPRI Study Re: PWR Radiation Control	10-80

8. Potential Problems:

As indicated in Section 7, the industry programs designed to develop methods of reactor decontamination are scheduled for completion in 1980. The results of these programs will not be available to assist NRC in making licensing decisions such as steam generator replacement that may be needed in 1979. Therefore it is important that the staff develop its own guidelines on decontamination prior to any requests for licensing action.

JUL 22 1977
REVISION 0

CATEGORY A TECHNICAL ACTIVITY NUMBER A-16

Title: Steam Effects on BWR Core Spray Distribution

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: Roy Woods, DOR

1. Problem Description:

The core spray (CS) systems are one component of the Emergency Core Cooling System (ECCS) for all BWRs. CS systems have a nozzle or a set of nozzles arranged to distribute water over the top of the core following a postulated loss-of-coolant-accident (LOCA). Each fuel bundle must receive a specified minimum amount of coolant (i.e., flow) from the CS system to provide the post-LOCA spray cooling assumed in the LOCA analyses.

During tests conducted in Europe (the results of which were later confirmed by tests conducted by the General Electric Company), it was discovered that the presence of steam and/or increased pressures in and above the upper core region could adversely affect the distribution of flow from certain types of core spray nozzles.

Prior to this discovery, GE had conducted full scale spray distribution tests in air at atmospheric pressure for all BWR/2 and later designs to ensure that the necessary minimum coolant would be provided to each fuel bundle. Those tests were performed in a full scale test facility which used a mockup of the core spray nozzle geometry (spacing, type, arrangement, and alignment) spraying water over a mockup of the top of the reactor core. Core spray flow into each mockup "fuel bundle" was collected and measured.

Prior to the European tests in steam and at higher pressure, such tests in air were accepted as an adequate demonstration that sufficient flow would be delivered to each fuel assembly to provide adequate cooling. However, the new test data in a steam environment and at various pressures raise questions regarding the safety margin previously thought to exist in the spray flow to individual fuel assemblies.

APPROVED BY TASC, AUGUST 19, 1977

The new data in steam and at increased pressures were from a single nozzle spraying vertically downward. Depending upon the type of nozzle tested, various significant effects on spray distribution were noted. These included partial or complete collapse of the spray cone and/or a shift in the average direction of flow (i.e., in the spray axis). These effects were most severe for nozzles which produce a small, high velocity droplet. Some BWR's do not utilize such nozzles, but others have a combination of such nozzles and larger droplet, lower velocity nozzles.

In contrast to the vertically oriented single nozzle tests, spray flow in most domestic BWR core spray systems comes from many nozzles spraying approximately horizontally over the core from a sparger (or spargers) surrounding the core. Therefore, with the exception of the Big Rock Point (BRP) reactor where one of the two spray systems has a single nozzle directed vertically downward, it is not known how applicable the new data is for domestic BWRs. In an initial attempt to quantify steam and increased pressure effects on spray distribution for geometries more typical of domestic BWRs, GE conducted a series of single nozzle tests in steam with different types of nozzles, typical of those used in their BWR/2 through BWR/5 plants. These tests quantified the amount of cone collapse and spray axis shift due to the steam environment that would be expected in the upper plenum of a BWR following a LOCA. These effects were then simulated in the full scale testing (air only) facility previously described. That is, each nozzle in the air testing facility was modified so that it would reproduce, in air, the spray pattern that the single nozzle steam tests showed would be produced by an identical nozzle in a steam environment. The full scale air tests, with the nozzles so modified, were then repeated. GE contends that these "Air Mockup of Steam Environment" tests present the actual distribution that would be measured if a full scale test were conducted in a steam environment (no facility exists to actually conduct such tests for BWR/2 and later plant designs).

Results of those tests indicate that adequate spray flow would be present in BWR/2 through BWR/5 plants following a LOCA. For most of these plants, the limiting break with the worst single failure leaves two core spray systems available, plus one or more flooding systems in certain plants. Flow typical of the minimum flow to any group of fuel bundles (at a given radius) from only one-spray-system-operation was present in tests previously run to measure spray heat transfer coefficients (Full Length Emergency Core Heat Transfer tests -

FLECHT tests). Thus, availability of two systems contributes to the spray flow safety margin still believed to be present even when steam effects on spray distribution are considered. However, it is not known how much of the margin previously thought to exist above minimum required spray flows actually would be present. The principal areas identified to date requiring more effort in order to resolve this question are:

(1) The single-nozzle-in-steam tests do not include possible effects due to steam quality. Water droplets entrained in the steam may change the interaction of the steam and the spray cone.

(2) There have been no full scale "Air Mockup of Steam Environment" tests for the geometries and nozzles of BWR/1 plants (i.e., Humboldt Bay Unit 3, Dresden 1, Big Rock Point, and LaCrosse). These older plants are particularly sensitive to effects of reduced spray flow since bottom breaks can be postulated which would prevent reflooding (Humboldt Bay can be reflooded since it has no credible large bottom break; however, it has only one spray system). That is, all cooling in the short and long term must be provided by the core spray system(s).

(3) There is no experimental verification that the "Air Mockup of Steam Environment" tests that were conducted for BWR/2 through BWR/5 plants actually predict the spray distribution that would exist in the real steam environment following a LOCA. Any core-wide phenomena such as gross upper plenum flow effects (swirling, vortex formation, redistribution) would not be discovered without actual large scale, multi-nozzle experiments in steam at pressures typical of a BWR upper plenum after a LOCA. To provide early verification of the "Air Mockup" method, the staff has taken the position that an "Air Mockup of Steam Environment" test should be conducted to predict certain planned steam tests (see Section 2 below).

(4) The "Air Mockup of Steam Environment" tests do not include effects of two phase froth buildup on top of the core due to counter-current-flooding phenomena. Such phenomena could further affect core spray distribution.

2. Plan for Problem Resolution:

GE has recently (May, 1977) submitted documentation of the above described single-nozzle-in-steam and multi-nozzle-in-air tests (Amendment No. 3 to NEDO-20566, "Effect of Steam Environment on BWR Core

Spray Distribution," April 1977, General Electric Co.). The report ("Amendment 3") also describes GE's application of the test results to show that "Appendix K" ECCS-LOCA limiting analyses previously performed for BWR/2 through BWR/5 plants are still applicable. The initial staff effort will involve the review of the tests and arguments presented in "Amendment 3" including any possible applications to older plants not specifically covered by the document. The four principal areas of concern already identified during preliminary review of "Amendment 3" were noted above. As noted, one concern was with lack of experimental verification for the "Air Mockup of Steam Environment" tests. The staff has taken the position that preliminary experimental verification for this method should be provided as described in the third paragraph below.

The applicability of the tests and analyses contained in "Amendment 3" to BWR 1 plants will be evaluated by the licensees and by the NRC staff, and this review may result in requests for further information from the licensees and/or from GE. If it can be demonstrated that certain "Amendment 3" conclusions are applicable to older plants and can be referenced by the older plants even though the older plants were not specifically considered in "Amendment 3", the conclusions would be considered in the evaluation regarding the adequacy of core spray distributions for those older plants. For example, large-droplet spray distributions are much less affected by steam effects than small-droplet sprays. Therefore, if it can be demonstrated that older plants have large-droplet sprays, then this conclusion from a review of "Amendment 3" would be considered in evaluating older plant spray distributions, requirements for further testing, etc.

The staff will also continue to follow progress of full scale steam environment tests being performed by Consumer's Power Corp. (CPC) for their Big Rock Point (BRP) plant. A full scale, steam environment test facility was constructed by CPC (and their subcontractors and consultants) for the initial purpose of testing a full scale mockup of the unique BRP single nozzle spray system. CPC was granted an exemption from certain of the requirements of 10 CFR 50.46 to allow plant operation for the present cycle without demonstration of the adequacy of the single nozzle CS system. However, a condition attached to granting of that exemption was that the single nozzle spray system be demonstrated to be acceptable with appropriate tests before operation beyond the present cycle would be authorized. The present tests are to fulfill that exemption condition by performing a set of full scale, steam environment tests of the BRP single nozzle core spray system.

In addition, although not required by the exemption, CPC has committed voluntarily to perform full scale, steam environment tests of their other spray system which utilizes multiple nozzles on a ring sparger above the core. We will take the position that GE should perform an "Air Mockup of Steam Environment" test for the BRP ring spray system, in conjunction with the CPC steam environment tests. Comparison of the steam and the air tests will provide an experimental basis for judging the interim acceptability of the GE "Air Mockup of Steam Environment" method which was used for BWR/2 through BWR/5 plants. Those "Air Mockup of Steam Environment" tests, as described above, constitute GE's primary justification for acceptability of core spray systems, with consideration of steam effects on spray distribution. Such a comparison of steam and air tests is considered essential to justify continued BWR operation and licensing in the interim period prior to completion of the extensive review and/or further tests beyond the BRP tests that will be needed to finalize resolution of this generic concern.

The staff will observe the CPC single nozzle and ring sparger steam environment tests for BRP and will evaluate the test facility to determine the facility's potential for any possible use in later, large scale, multi-nozzle spray distribution tests for other plants. If such tests are considered necessary as the result of detailed review of the "Amendment 3 - Air Mockup of Steam Environment" tests, knowledge of the CPC facility's capabilities will be useful in determining the optimum set of tests to be recommended or required.

All NRR technical organizations involved will review "Amendment 3" (already submitted) to better define further information requirements. The extent and exact nature of further tests (beyond the BRP steam tests) and/or analyses which may be recommended will be determined when the review has progressed to a point at which a meaningful consensus among all reviewers can be determined by the Task Manager. This will be possible following review of "Amendment 3" including responses to first round questions, and including consideration of the CPC-BRP ring sparger steam tests and the GE air prediction of those tests. When a consensus is reached, the proposed course of action will be submitted by the Task Manager to NRC management for approval.

Following the review of "Amendment 3" and all subsequent submittals by all Technical Review Branches involved, and following successful completion and review of additional analyses and tests, if required, it is anticipated that a Safety Evaluation will be published in the form of a NUREG report on this generic issue. The NUREG report can then be referenced as covering this generic item in future Safety Evaluation Reports (SER's) on NEDO-20566 including "Amendment 3" and in future SER's on the GE-ECCS "Appendix K" Evaluation Model. The NUREG report will state the NRR conclusions regarding generic resolution of the problem (i.e., acceptability of the analytical techniques used to model the phenomena demonstrated by the test results). The NUREG report will also cover the acceptability of applying those techniques to all operating plants and all plants under construction or being planned.

3. NRR Technical Organizations Involved:

   a. Reactor Safety Branch, Division of Operating Reactors

      RSB/DOR has overall lead responsibility for the conduct of this generic review. RSB/DOR will be primarily concerned with effects on operating reactors, but will review "Amendment 3" generically, relative to plants in all stages of licensing in cooperation with the other two branches involved. All three branches will cooperate with the Task Manager in review of "Amendment 3", since there is at least one experienced reviewer familiar with steam effects on spray distribution in each of the three branches. This will include consideration of the GE air prediction of the CPC BRP ring spray system steam test, and comparison to the BRP steam test, as a preliminary justification for the GE "Air Mockup" method. This Branch will also evaluate the "BRP test facility" to determine its potential for use in any later experiments deemed necessary to resolve outstanding concerns for operating plants.

      Manpower Estimates: 0.33 man-year FY 1977, 0.33 man-year FY 1978, 0.33 man-year FY 1979, 0.16 man-year FY 1980.

   b. Reactor Systems Branch, Division of Systems Safety

      RSB/DSS will be primarily concerned with effects on reactors not yet licensed for operation, but will review "Amendment 3" in cooperation with the other two branches involved. This will include consideration of the GE air prediction of the CPC BRP ring spray system steam test, and comparison to the BRP steam test, as a preliminary justification for the GE "Air Mockup" method. This Branch will also evaluate the "BRP test facility" to determine its potential for use in any later experiments deemed necessary to resolve outstanding concerns for plants in the licensing process.

Manpower Estimates: 0.16 man-year FY 1977, 0.16 man-year FY 1978, 0.16 man-year FY 1979, 0.10 man-year FY 1980.

c. Analysis Branch, Division of Systems Safety

AB/DSS will evaluate and compare test results to analytical results to determine the adequacy of current analytical techniques, and will review any proposed changes in analytical techniques as a result of the tests reviewed. This Branch will review "Amendment 3" in cooperation with the other two branches. This will include consideration of the GE air prediction of the CPC BRP ring spray system steam test, and comparison to the BRP steam test, as a preliminary justification for the GE "Air Mockup" method. Principle review subjects will include but not necessarily be limited to analysis techniques used to predict spray vaporization, countercurrent-flooding phenomena (i.e., liquid-vapor interaction), droplet entrainment, channel and fuel quenching, parallel channel effects, and modeling of any new phenomena discovered in future tests.

Manpower Estimates: 0.16 man-year FY 1977, 0.33 man-year FY 1978, 0.33 man-year FY 1979, 0.16 man-year FY 1980.

4. Technical Assistance Requirements:

It is reasonable to expect that additional testing will be recommended, (i.e., as close as possible to full scale testing of complete multiple nozzle spray systems in steam for plants other than BRP). This decision will be made following review by all concerned branches of the "Amendment 3" material, including round one question responses and with consideration of the CPC BRP ring sparger steam tests and the GE air prediction of those tests. At that time, the extent of the tests and the necessity for TA funding and/or the extent of RES or International Programs involvement will be recommended.

5. Interactions with Outside Organizations:

a. General Electric Company (GE)

Requests for additional information resulting from NRC staff review of the "Amendment 3" document will be addressed to GE, as will the staff position recommending prediction of the BRP sparger-system-in-steam tests. All recommendations or staff positions regarding additional tests that would be generic to all BWRs or to a large group of BWRs will also be addressed to GE.

b. BWR Licensees

In certain cases, requests for information regarding design of certain plant specific or unique spray systems will be addressed to the licensee. For example, design or droplet size distribution data for a certain nozzle or spray system used on a specific plant would be addressed to the individual licensee. This would be most likely the case for older, unique plants such as BWR/1's.

c. Consumer's Power Corp. (CPC)

CPC is currently performing full scale, steam environment spray distribution tests for the BRP single nozzle spray system. Questions regarding the design and capabilities of the test apparatus will be directed to CPC.

6. Assistance Requirements from Other NRR Offices:

If additional testing beyond the BRP test is recommended as a result of review of the "Amendment 3" material or other material, it is possible that other offices such as RES and/or International Programs will be involved.

Such determination will be made following review by the branches involved of the "Amendment 3" and related material.

7. Schedule for Problem Resolution:

The major milestones for the Steam Effects on BWR Core Spray Distribution task are as follows:

1) Submittal by GE of "Amendment 3" to NEDO-20566-May, 1977 (Complete).

2) Transmit to GE results of preliminary review of "Amendment 3", including the staff position that an "Air Mockup of Steam Environment" test should be conducted as soon as possible to predict the Consumer's Power Corp. full scale ring spray tests (to be run in the steam test facility for Big Rock Point) - 7/22/77.

3) Review of "Amendment 3" by the three specified NRR branches and submittal of requests for additional information to the Task Manager for transmittal to GE and/or licensees - 08/15/77.

4) Response received from GE and/or licensees to additional information requests - 11/01/77.

5) Receive data from air and steam tests conducted for BRP ring spray system (see 2 above) - 12/31/77.

6) Submittal of positions from all three specified NRR branches, including additional testing beyond the BRP tests recommended (if any) to Task Manager for transmittal to GE and/or licensees - 02/01/78.

7) Planning complete for additional tests beyond BRP tests (if any) (GE/NRC/Licensees/Contractors coordinate plans) - 09/01/78.

8) Submittal to NRC of final test report (if any) - 09/01/79.

9) Review of all information complete - SER (NUREG report) issued - 12/31/79.

10) Requests issued for modifications to plant hardware and/or Technical Specifications (if necessary as a result of conclusions in the SER) - 12/31/79.

The above schedule assumes that further tests are recommended in step 6. If such tests are not needed, then steps 7 and 8 will be eliminated. Instead, responses would be received to any step 6 positions on 03/01/78 and steps 9 and 10 would be accomplished 06/30/78.

8. Potential Problems:

The above schedule assumes that since any additional testing and analysis will be conducted to answer questions regarding margins of safety, safe plant operation and orderly licensing procedures can continue while the program is completed. If questions regarding the safety of continued plant operation arise during this review that indicate an urgent need for further test results (such is not anticipated at this time), then the critical path item would be design and construction of the necessary test facility, or utilization of the CPC facility if it proved adaptable to the test requirements. If such concerns regarding safety of continued plant operation are found, it might become necessary to grant exemption to certain of the requirements of 10 CFR 50.46 if plant operation is to continue while the Plan is completed.

The future course of this task is not well enough defined to warrant further speculation concerning potential problems at this time. After review of the "Amendment 3" material, further problem definition will be possible.

REVISION 0

NOVEMBER 15, 1977

TASK ACTION PLAN

TASK NO. A-17

TITLE: Systems Interaction in Nuclear Power Plants

LEAD RESPONSIBILITY: Division of Project Management

LEAD ASSISTANT DIRECTOR: R. C. DeYoung, Deputy Director, DPM

TASK MANAGER: John Angelo

1. Problem Description

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines such as civil, electrical, mechanical, structural, chemical, hydraulic, and nuclear, and into scientific disciplines such as geology, seismology, and meteorology. The reviews 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 to assure that the single failure criterion is met.

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 such as containment systems, reactor systems, etc., or by disciplines such as mechanical engineering, materials engineering, and structural engineering. 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

APPROVED BY TASC, NOVEMBER 2, 1977

TASC COMMENTS INCORPORATED, NOVEMBER 15, 1977 account for systems interaction to a large extent. Furthermore, many of our 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 problem to be resolved by this task is to establish a systematic process to review plant systems to determine their impact on various other plant systems. For purposes of this task, systems interaction is defined as actions or consequences in one system that could adversely affect the redundancy or independence of safety systems in another system or systems.

2. Plan for Problem Resolution

The plan for resolution of this task is to develop and implement, to the extent that a study indicates the need, a method of review that will extend the present review techniques in sufficient breadth and depth to assure a systematic and comprehensive review of systems interaction.

The plan will also include the development of criteria and procedures to assure that applicants incorporate appropriate systems interaction considerations into their design and review process.

Qualified personnel shall be assigned to accomplish this plan. The major tasks to be performed are:

(a) Establish a uniform designation of plant systems and their associated functional inputs and outputs, and determine the interface points or boundaries where interactions can occur, including identification of the types of interactions.

The subtask will be accomplished by use of the SRP, selected Safety Analysis Reports, WASH-1400, and other documents, as well as discussions with reactor manufacturers, architect-engineers, and utilities to derive the designation of plant systems and functional inputs and outputs. Review of the results of this and the other subtasks by each of the technical review branches and project management branches will provide further assurance that all plant systems are properly accounted for and correctly and clearly described and defined. This review is scheduled to be accomplished at two specific milestones identified in paragraph 7 of this task action plan. The identification of the types of interactions will be an identification by broad categories such as electro-mechanical, thermal-hydraulic, and pneumatic-mechanical.

This subtask is divided further into four elements as follows: (1) designation of systems, (2) designation of system functions, (3) designation of interaction points, and (4) designation of interaction types. These elements follow in a logical sequence of development but are somewhat interdependent. For example, the breakdown of a plant into systems is dependent upon the functions to be performed. Ideally, a collection of components should be assigned to a system on the basis of performing one specific function that is readily identifiable. Also, the subtask elements have been chosen so that a discrete product can be produced; e.g., a list of plant systems. This procedure allows for assignment of work and control of output and provides for review and concurrence control by all review branches.

In order to accomplish this subtask and in order to establish a uniform basis for review by cognizant review branches, it will be necessary to develop criteria for bounding the extent of definition of systems, functions, interaction points, and interaction types. The criteria must define the items that will be retained in the matrix of systems and functions; otherwise, the matrix will become unmanageable and the review will not proceed on a uniform basis. The criteria will serve as the basis to eliminate systems interactions of little or no safety significance. These criteria will be developed early in the execution of the task in order to give purposeful direction to the task and to its review. The development of these criteria is shown as scheduled item (b) in paragraph 7 of this task action plan.

(b) Compare the Standard Review Plan (SRP) against item (a) above to determine the extent to which the SRP already adequately addresses interdisciplinary review areas and systems interactions. Also, determine the extent to which the SRP includes consideration of systems interactions during postulated accidents, events, and transients. Measure the SRP against licensee event reports to determine if a review conducted according to the SRP would predict systems interactions of the type identified in licensee event reports.

Subtask (b) is divided into three work elements: (1) comparison of subtask (a) to the SRP, (2) comparison of the SRP for design basis events, transients, and accidents, and (3) comparison of the SRP for licensee event reports. Again, the elements of the subtask have been selected so that each element can be accomplished as a discrete work unit with an identifiable end product. It is anticipated that a matrix will be constructed, for example, to display the interdisciplinary review areas of the SRP against the plant systems so that there is a readily identifiable method to ascertain the extent to which the SRP now provides for interdisciplinary review.

A specific milestone is provided for the review and concurrence by all review branches as indicated in paragraph 7 of this plan.

Also, at this point in the task, an assessment will be made to determine the extent of effort necessary to complete the task and make appropriate adjustments to the schedule.

(c) Develop, to the extent necessary, revisions of the SRP based on the results of task (b). The task action plan provides for review and concurrence control by all review branches at the completion of this task as indicated in paragraph 7 of this plan.

(d) Develop criteria and procedures, including information requirements, for use by applicants in their design and review of plant designs for systems interaction. The task action plan provides for review and concurrence control by all review branches at the completion of this task as indicated in paragraph 7 of this plan.

One of the end products of this task will be additions, where necessary, to the SRP to assure that our review procedures adequately address considerations for systems interaction. The other end product will also be a recommendation that a Regulatory Guide be issued to provide guidance on the criteria, procedures, and information required related to applicants' analyses and review of systems interaction.

During the accomplishment of this task, consideration will be given to the use of the end products for operating reactors. The most effective method of accomplishing this objective will be by review of the task by the technical branches in DOR at several identified milestones. For example, DOR will review the results of each of the major subtasks for applicability to operating reactors. An additional measure of assurance that the end products will be applicable to operating reactors would be to assign a qualified individual from DOR to maintain a continuous working relationship with the other assigned personnel during the development of the task. Since some of the elements of this systems interaction task are common to the elements that have been and will be used in the Systematic Evaluation Program (SEP) currently being conducted by DOR, the assignment of personnel should be from the SEP Branch.

3. NRR Technical Organizations Involved

This task should be accomplished by assignment of qualified personnel from within DSS, DSE, DOR, and DPM. A total effort of about 55 man-months expended over 13 calendar months is anticipated. Approximately five persons should be assigned to work with the task manager approximately 60% of full time during the 13-month interval. This estimate includes 40 man-months expended by assigned personnel and 15 man-months by review branches. An estimate of the review effort by each branch is shown in Table 1 attached to this plan.

Since the majority of interactions involve electrical systems interactions with mechanical and hydraulic systems, at least one person with a strong background in instrumentation and control systems and the relationship of these systems to other plant systems should be assigned to this task. A second individual with background in auxiliary plant systems and a third individual with background in reactor systems should also be assigned. A fourth person with strong general background in boiling water reactors should be assigned. The task manager presently assigned has a broad background in pressurized water reactors. As stated in Section 2 of this plan, an additional measure of assurance that the end product of this task will be applicable to operating reactors would be to assign a qualified person from DOR to maintain continuous working relationship with other assigned personnel and with the cognizant review branches and to provide direct input from DOR. This is particularly important in view of the current activities being conducted by DOR under the SEP for selected operating reactors and the technical assistance program being conducted by ORNL. These assignments should provide a well balanced group capable of probing into the more specialized areas of systems interactions and will assure effective interfacing with the SEP.

Accordingly, the assignment of personnel from DSS to accomplish this task should be from (a) Auxiliary Systems Branch, (b) Reactor Systems Branch, and (c) Instrumentation and Control Systems Branch. The fourth person should be selected from among the Licensing Project Managers in the Light Water Reactors project branches, and the fifth person should be selected from the Systematic Evaluation Program Branch in DOR. The estimated manpower for these assigned persons, including the presently assigned Task Manager, is as follows:

Unit	Man-Months
Auxiliary Systems Branch, DSS	7
Reactor Systems Branch, DSS	6
Instrumentation & Control Systems Branch, DSS	8
AD for Light Water Reactors (Undesignated Branch), DPM	7
Systematic Evaluation Program Branch, DOR	4
Light Water Reactors Branch #1 (Task Manager), DPM	8
Total	40

In addition to these individuals, virtually all technical branches within DSS, DSE, DOR, and DPM will be requested to review and critique the end products of the task and provide a nominal level of time for consultation in selected areas. The requirements of specific branches will vary as a function of their involvement with systems. This time is anticipated to require about 15 man-months and will vary from one-half man-week to four man-weeks per branch. This time will be expended over the span of the task at the specific milestones indicated in paragraph 7 of this report.

In addition to the review and critique by cognizant review branches within NRR, the assistance of the AD for Reactor Safeguards, DOR, will be requested for consultant assistance to aid in using the techniques for plant and systems reviews that was developed by the workshop group for Industrial Security.

4. Technical Assistance Requirements

At appropriate points during the execution of this task, and as the results become available, the results of the ongoing technical assistance program with Oak Ridge National Laboratory (ORNL) now being conducted by DOR will be used in the task. In order to accomplish this objective, cognizance of the ongoing technical assistance program will be developed and maintained by review of published information, attendance at meetings, and conferences with personnel who are active in the program in DOR and ORNL.

The scope of the task at ORNL is (1) to identify and evaluate the safety significance of possible interactions between control and protection systems, (2) provide recommendations for possible design modifications or operational requirements, (3) perform a detailed analysis, including a failure mode analysis, of auxiliary control systems specified by the NRC for the purpose of identifying any dependence between these systems and the reactor protection system, (4) assess the possibility of control system failures resulting in a challenge to the reactor protection system, and (5) evaluate the significance of possible adverse interactions between protection and control systems, and the capability of the reactor protection systems to mitigate the consequences resulting from these interactions or from control systems failures. The task is further described as follows:

1. For auxiliary systems specified by the DOR staff:
    a. Identify the possible failure modes in auxiliary system controls.
    b. Identify the protection systems provided to mitigate the consequences resulting from the control system failure and an evaluation of their capability to do so.
    c. Identify possible adverse interactions between protection and control systems as a result of the assumed failure.
    d. Define the effect of each failure on fuel integrity and the reactor coolant pressure boundary.
2. Provide an assessment of the probability of occurrence of failures resulting in a challenge to the protection system.
3. Identify areas where modifications to the auxiliary control systems or the protection systems would result in a significant increase in protection.

4. Based on the studies performed in Tasks 1-3, provide an assessment of possible adverse system interaction effects that may exist in operating power plants and recommend a course of action to correct any deficiencies defined.

Manpower and funding estimates for this task at ORNL are 15 man-months of support effort during FY 1978 at a cost of $60,000.

5. Interactions with Outside Organizations

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses.

Meetings are anticipated with NSSS vendors, A/Es, and utilities to assess the extent to which these organizations conduct reviews and analyses for systems interaction, and to keep these organizations informed of our developments. It is not intended to conduct a formal review process through these organizations, however. It is intended to develop a free exchange of information so that the task can take advantage of existing methods of review.

Commonwealth Edison Company has developed and will implement a somewhat limited systems interaction study for the Zion Station. The Commonwealth Edison Company study will consist of a detailed review of Licensee Event Reports of those events which have occurred that involve undesirable systems interactions. Both physical and electrical interactions will be covered in the event review but will be approached on a case-by-case basis rather than from a more general standpoint. We agree that this study should proceed, recognizing that it may or may not be the final effort for the Zion facility in that additional techniques may be developed at a later time.

6. Assistance Requirements from Other NRC Offices

Assistance will be requested from the Probabilistic Analysis Branch, Office of Nuclear Regulatory Research, to provide consultant assistance in the detailed development and execution of this task action plan. It is estimated that this total assistance from RES will be about one man-month of effort. It is anticipated that this group can provide valuable insights into the task because of its involvement with the Reactor Safety Study (WASH-1400). Additionally, this group would be requested to review and critique the results of this task action plan.

7. Schedule for Problem Resolution

The following schedule is proposed assuming that personnel are assigned by November 15, 1977.

(a) Assignment of Personnel and Assignment of Task Items - DECEMBER 1, 1977

(b) Identify Plant Systems and Functional Requirements and Criteria for Selection of Systems and Interactions - JANUARY 30, 1978

(c) Review of Item (b) by All Cognizant Branches - MARCH 1, 1978

(d) Identify Interaction Points and Types of Interactions - MARCH 17, 1978

(e) Review of Item (d) by All Cognizant Branches - APRIL 1, 1978

(f) Compare SRP to Results of (a) and (b) Above - MAY 28, 1978

(g) Compare SRP for Systems Interaction for DBAs and DBEs - JUNE 12, 1978

(h) Compare SRP for Systems Interaction for LERs - JUNE 26, 1978

(i) Review of Items (f), (g), and (h) by All Cognizant Branches - JULY 16, 1978

(j) Develop Revisions to SRP Based on Results of (d), (e), and (f) - AUGUST 28, 1978

(k) Review of Item (j) by All Cognizant Branches - SEPTEMBER 18, 1978

(l) Develop Criteria and Procedures for Systems Interaction Analysis and Reviews by Applicants and Licensees - OCTOBER 1, 1978

(m) Develop Information Requirements for SARs - OCTOBER 15, 1978

(n) Review of Task Results by DSS, DSE, DPM, and DOR - NOVEMBER 13, 1978

(o) Complete Task - DECEMBER 30, 1978

8. Potential Problems

One of the problem areas is that systems interaction cuts across all disciplines and technical branch review areas and cuts across all groups and divisions. Consequently, in order to effectively perform a review for systems interaction, it is necessary to either define more clearly and more extensively the primary and secondary review responsibilities in the SRP or organize a new element to perform the review. The real problem is where to place this new organizational element if one is formed. Consideration will be given during execution of this task to the resolution of this problem.

A second potential problem area is related to estimating the scope and extent of effort required to complete subtasks (c) and (d) concerning the potential revisions to the SRP and the development of criteria and procedures for use by applicants in their design and review of plant designs for systems interaction. Therefore, it is anticipated that at the completion of subtasks (a) and (b), a reassessment will be made of the follow-on effort to complete subtasks (c) and (d). It is expected that the information generated by completion of subtasks (a) and (b) will provide a valid basis for a reassessment of the balance of effort to complete the task.

ATTACHMENT

ESTIMATE OF TASK REVIEW

TASK A-17

SYSTEMS INTERACTION IN NUCLEAR POWER PLANTS

	Man-Weeks
DIVISION OF PROJECT MANAGEMENT	
Quality Assurance Branch	2
Emergency Planning Branch	½
Operator Licensing Branch	2
Light Water Reactors Branch No. 1	1
Light Water Reactors Branch No. 2	1
Light Water Reactors Branch No. 3	1
Light Water Reactors Branch No. 4	1
Liquid Metal Fast Breeder Branch	2
DIVISION OF OPERATING REACTORS	
Systematic Evaluation Program Branch	4
Engineering Branch	1
Plant Systems Branch	2
Reactor Safety Branch	2
Environmental Evaluation Branch	1
Operating Reactors Branch No. 1	1
Operating Reactors Branch No. 2	1
Operating Reactors Branch No. 3	1
Operating Reactors Branch No. 4	1
Standard Technical Specification Group	1
Reactor Safeguards Licensing Branch	2
Reactor Safeguards Development Branch	2

	Man-Weeks
DIVISION OF SYSTEMS SAFETY	
Mechanical Engineering Branch	1
Materials Engineering Branch	1
Structural Engineering Branch	1
Auxiliary Systems Branch	4
Containment Systems Branch	2
Instrumentation & Control Systems Branch	4
Power Systems Branch	4
Analysis Branch	1
Core Performance Branch	1
Reactor Systems Branch	4
DIVISION OF SITE SAFETY & ENVIRONMENTAL ANALYSIS	
Geosciences Branch	½
Hydrology-Meteorology Branch	1
Accident Analysis Branch	4
Effluent Treatment Systems Branch	1
Radiological Assessment Branch	1
TOTAL	60

SEP 14 1977

TASK ACTION PLAN

TASK NO. A-19

REVISION 0

Title: Digital Computer Protection System
Lead Responsibility: Division of Systems Safety
Lead Assistant Director: R. Tedesco, DSS:PS
Task Manager: L. Beltracchi, DSS:ICSB

1. Problem Description

Current design trends are for reactor protection systems to incorporate digital computer technology and components. The staff is currently reviewing an operating license application for a plant design in which digital computers are utilized as initiation logic devices for two reactor trips.[1] Additional applications for protection system designs using digital computers have been docketted and are under review by the staff.

A need exists to standardize the safety review of reactor protection systems incorporating digital computers. Since digital technology is considerably different from the analog technology previously used for protection systems, the criteria appropriate for the safety review of digital-computer-based systems are different from those used for analog based systems. Although the ANO-2 digital computer based protection system has been reviewed, the technology is rapidily changing, especially in the area of software, and needs to be assessed.

The benefits of standardizing the review are:

a) The design, development, and qualification information required by the staff to conduct the safety review are

APPROVED BY TASC, OCTOBER 19, 1977

defined to the applicant.

b) Documentation requirements for safety review criteria to be used by the staff in the evaluation of digital computer hardware and software are uniformly stated.

c) A standard Review Plan will define uniform and consistent guidelines for the conduct of the safety review. --

2. Plan for Problem Resolution

A. Approach

Criteria and procedures for safety review of digital-computer-based protection systems will be developed drawing upon:

1. The experience of the Regulatory Staff gained in conducting the safety review for ANO-2 (1),

2. The advice of consultants to the staff,

3. A survey of criteria used in evaluating digital computers by other Government Agencies,

4. A review of industry standards and international standards established for digital computers.

B. End Products

The end products of this technical activity and their use in the licensing process are:

1) A recommended revision to the Standard Format[3] which defines the design, development, and qualification information required by the staff to conduct a safety review of reactor protection systems incorporating digital computer technology.

2) A Branch Technical Position (BTP) is to be written that will contain the safety review criteria for the evaluation of digital computer based protection systems. In the licensing process, the BTP will serve as the basis of the safety review.

3) A revision to the Standard Review Plan[4] which defines the guidelines for the execution of the safety review will be developed.

C. Tasks

1) Document the information required, guidelines/criteria, and methodology used in conducting the safety review for the digital computers proposed in the ANO-2 application.

2) Each technical consultant to the staff used in conducting the safety review for the digital computers proposed in the ANO-2 application is to submit a report documenting the guidelines/criteria and methodology used for the review.

3) Conduct a survey of other Goverment Agencies (specified in Section 5) to determine the criteria and methodology used in evaluating the performance of digital computer hardware and software.

4) Review industry standards and international standards established for digital computers.

5) Analyze the results of Tasks 1-4 above and write revisions to the Standard Review Plan and recommend revisions of the Standard Format to the Office of Standards Development. Submit revisions for management review, address comments and update revisions.

6) Incorporate requirements for digital computer based protection systems into the Standard Technical Specificatiosn and bases.

3. <u>NRR Technical Organizations Involved</u>

A. Division of Systems Safety/Instrumentation and Control Systems Branch

1) Task 1 - Document the information required, guidelines/ criteria, and methodology used in conducting the review of the design and qualification of the digital computers proposed in the ANO-2 protection system. The manpower estimate for this task is 10 man-days.

2) Task 3 - Conduct a survey of other goverment agencies (specified in Section 5) to obtain methodology and techniques used to evaluate computer systems. Generate a report on each survey conducted. The manpower estimate for this task is 12 man-days.

3) Task 4 - Conduct a review of available industry standards and international stardards used for design and qualification of digital computers. Generate a report for each review conducted. The manpower estimate for this task is 12 mandays.

4) Task 5 - Analyze the results of Tasks 1, 3, and 4 to establish a Branch Technical Position and the revisions to the Standard Format and the Standard Review Plan. These documents are then to be submitted to management for review. The estimates for these efforts are:

Analyses:	13 man-days
Addendum to Standard Format:	13 man-days
NUREG-Review Criteria	20 man-days
Addendum to Standard Review Plan:	33 man-days
Total:	79 man-days

5) Task 6 - Revise Standard Technical Specification. The manpower estimate for this task is 5 man-days.

6) Total Manpower -- FY-78 -- 0.17
FY-79 -- 0.35
Total -- 0.52 man-years

B. Division of Systems Safety/Analysis Branch

1) Task 1 - Document the information required, guidelines/criteria, and methodology used in conducting the review of the design and the qualification of the thermal-hydraulic protection algorithms for the digital computers proposed in the ANO-2 protection system. The manpower estimate for this task is 10 man-days.

2) Task 5 - Analyze the results of Task 1, and of the technical surveys conducted and document the revisions to the Standard Format and the Standard Review Plan for the conduct of future reviews. These documents are than to be submitted to management for review.

The manpower estimate for this task is 40 man-days.

3) Task 6 - Revise Standard Technical Specifications. The manpower estimate for the task is 5 man-days.

4) Total Manpower -- FY-78 -- 0.07 man-years
FY-79 -- 0.15 man-years
Total -- 0.22 man-years

C. Divison of Systems Safety/Mechanical Engineering Branch

1) Task 1 - Document the information required, guidelines/criteria and methodology used in conducting the seismic design and qualification review for the digital computers proposed in the ANO-2 protection system. The manpower estimate for this task is 10 man-days.

2) Task 5 - Analyze the results of Task 1, and of the technical surveys conducted and document the revisions to the Standard Format and the Standard Review Plan for the conduct of future reviews. These documents are then to be sumitted to management for review. The manpower estimate for this task is 32 man-days.

3) Total Manpower -- FY-78 -- 0.06 man-years
FY-79 -- 0.13 man-years
Total -- 0.19 man-years

D. Division of Project Management/Quality Assurance Branch

1) Task 1 - Document the information required, guidelines/criteria, and methodology used in evaluating the Quality Assurance Program for the digital computers proposed in the ANO-2 protection system.

   Specific emphasis should be placed upon documentation for the design, development, and qualification of computer programs and configuration change control for computer programs. The manpower estimate for this task is 5 man-days.

2) Task 5 - Aanlyze the results of Task 1, and of the technical surveys conducted and document the revisions to the Standard Format and the Standard Review Plan for the conduct of future reviews. These documents are then to be submitted to management for review. The manpower estimate for this task is 7 man-days.

3) Total Manpower -- FY-78 -- 0.02 man-years
   FY-79 -- 0.03 man-years
   Total -- 0.05 man-years

E. Division of Systems Safety/Core Performance Branch

1) Task 5 - Review and comment on revised Standard Review Plan and Standard Format

2) Task 6 - Revise Standard Technical Specifications

3) Total Manpower -- FY-78 -- none
FY-79 -- 0.04 man-years

F. Division of Operating Reactors/Standard Technical Specification Group

1) Task 6 - Revise Standard Technical Specifications to include requirements for computer-based protection systems. The manpower estimate for this task is 10 man-days

2) Total Manpower -- FY-78 -- none
FY-79 -- 0.04 man-years

4. Technical Assistance Requirements

A. Contractor: Oak Ridge National Laboratory

1) Task 2 - Document the information required, guidelines/criteria and methodology used in conducting the review of the design and qualification of the digital computers proposed in the ANO-2 protection system. The manpower estimate for this task is 10 man-days.

2) Task 3 - Provide support for a survey of government agencies (Section 5) to obtain methodology and techniques used to evaluate computer systems. Provide a report for each survey supported. The manpower estimate for this task is 10 man-days.

3) Task 4 - Provide support for survey and review of available industry standard and international standards (Section 5) used for design and qualification of digital computers. Provide a report for each survey and review supported. The manpower estimate for this task is 10 man-days.

4) Task 5 - Analyze the results of Tasks 2, 3, and 4 and document recommendations for technical positions and recommendations for revisions to the Standard Review Plan. The recommendations are to be forwarded to the staff in report form. The manpower estimate for this task is 40 man-days.

B. Responsible Division/Branch -- Division of Systems Safety/Instrumentation and Control Systems Branch

C. Funding

FY-78	$10K (estimate)
FY-79	$20K (estimate)
(Total)	$30K (estimate)

5. Interactions With Outside Organizations

A. National Bureau of Standards

1) Task 3 - Survey of Government Agencies. Conduct a survey of National Bureau of Standards methodology and criteria employed for evaluating computers and computer programs.

B. Rome Air Development Center

1) Task 3 - Survey of Government Agencies. Conduct a survey of Rome Air Development Center methodology and criteria employed for evaluating computers and computer programs.

C. IEEE Computer Society

1) Task 4 - Survey of Technical Societies. Conduct a survey of the IEEE Computer Society, and specifically of the "Subcommittee on Software Standards" for design and qualification standards for digital computers and computer programs.

D. International Electrotechnical Commission

1) Task 4 - International Standards.

Technical Committee No. 45: Nuclear Instrumentation
Sub-Committee 45A: Reactor Instrumentation

Application of Digital Computers to Nuclear Reactor Instrumentation, Control and Protection. Obtain and evaluate the most recent draft of this standard.

E. American Nuclear Society

1) Task 4 - Survey of Technical Societies. Obtain and evaluate the most recent draft of the APLPHA System Standard. The ALPHA Class digital computers are used in protection system applications in Nuclear Power Plants.

6. Assistance Requirements From Other NRC Offices

None required

7. Schedule for Problem Resolution

A. Summary Schedule

Because the current resources are sufficient only for the review of ANO-2, this effort will not be initiated until the ANO-2 review is completed in December, 1977.

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B. Task Schedule

A line item schedule for the tasks defined in Section 2.C. is provided as Table I. A summary of manpower estimates is presented in Table II.

8. Potential Problems

A. Case work, CP's and OL's have a higher priority on resources than Category A tasks. This is the basic reason for a January 1978 start date.

B. The effort described in this report embraces many branches. within the Nuclear Regulatory Commission. Effective progress will be difficult, as priorities will vary from branch to branch.

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REFERENCES

1. Arkansas Nuclear One, Unit 2, Final Safety Analysis Report, Docket Number 50-368.

2. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director NRR to NRR Staff", November 1976.

3. NUREG-75/094, Regulatory Guide 1.70, Revision 2, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", LWR Edition, September 1975.

4. NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", LWR Edition, September 1975.

TABLE 1 - SCHEDULE

A-19 DIGITAL COMPUTER PROTECTION SYSTEM

Dates

Tasks

1. Provide documentation for guidelines, criteria, methodology used in conducting ANO-2 review
2. Consultants documentation for task 1.
3. Survey other government agencies, generate report.
4. Survey technical societies, generate report.
5. Analyze results, generate:
   - Standard Format
   - NUREG-Review Criteria
   - Standard Review Plan
   - Address review comments and update documents.

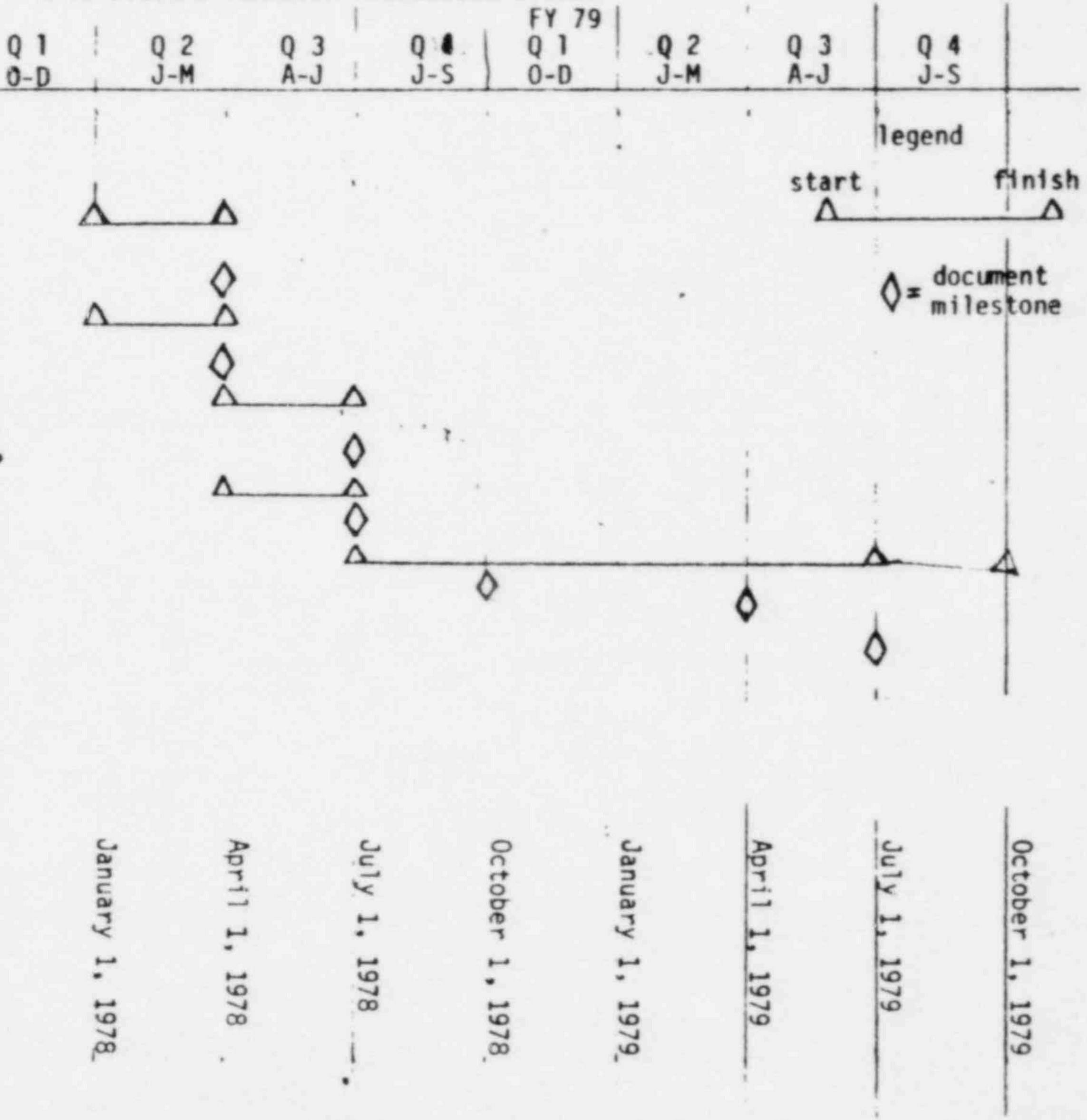

TABLE 2

MANPOWER ESTIMATES

A-19 DIGITAL COMPUTER PROTECTION SYSTEM

TASKS	ICSB	CPB	AB	QAB	ORNL	MEB	STS
1. Document Staff Experience	10 M-D		10 M-D	5 M-D		10 M-D	
2. Document Consultant Experience					10 M-D		
3. Conduct Survey Gov't. Agencies	12 M-D				10 M-D		
4. Conduct Survey Tech Societies	12 M-D				10 M-D		
5. Analyze results of 2C1--2C4, Establish end products	79 M-D	5 M-D	40 M-D	7 M-D	40-M-D	32 M-D	
6. Revise STS	5 M-D	5 M-D					10 M-D
7. Quarterly progress reports	4 M-D						
TOTALS	117 M-D	10 M-D	50 M-D	12 M-D	70 M-D	42 M-D	10 M-D

GRAND TOTAL = 341 M-D (1.51 Man-Years)

REVISION 0

August 12, 1977

Title: A-20 Impacts of the Coal Fuel Cycle

Lead Responsibility: Division of Site Safety and Environmental Analysis

Lead Assistant Director: Malcolm L. Ernst
Assistant Director for Environmental Technology

Task Manager: Richard V. Watkins
Environmental Projects Branch 1

1. Problem Description:

Need for Study:

Compliance with the National Environmental Policy Act (NEPA) requires that alternatives to a proposed Federal action be considered and that required alternatives be balanced against the base case in terms of their associated environmental impacts. NRC has established through its rulemaking authority a generic description and evaluation of the environmental impacts of the uranium fuel cycle in WASH-1248, NUREG-0116 and NUREG-0216. Based on these studies, a summary table, Table S-3, has been prepared and promulgated as regulation in 10 CFR Part 51.20(e).

A coal fired plant is currently the only realistic alternative to a nuclear power plant. Present treatment of the coal alternative is aimed essentially at economics and public health impacts. It is relatively incomplete in other areas of impact. The comparison of the coal alternative to the proposed nuclear facility would be significantly improved, if a study was conducted for the coal fuel alternative that augmented the work already done by ANL in the area of health effects. Such a study would provide a comprehensive summary which evaluates the environmental effects of the coal fuel cycle in a form directly comparable to that for the uranium fuel cycle. In the absence of such a generic treatment of the effects of using coal for generating electric power, it is necessary for the staff to develop an analysis de novo for each licensing action, to present this individual analysis in detail in the EIS, and to defend it throughout the hearing process. This repetitive staff effort could be avoided by preparing a generic statement suitable to support rulemaking proceedings. After the rulemaking procedure, such a statement would have the force of law necessary to avoid repetitive staff effort.

APPROVED BY TASC, AUGUST 19, 1977

A thorough analysis of alternatives to a proposed nuclear power plant requires an evaluation of the environmental effects of the coal fuel cycle to the same extent as the nuclear cycle. The environmental effects of the coal fuel cycle have long been recognized as being significant. There are deleterious effects to human health due to burning coal, but there are other significant socioeconomic and other environmental impacts at each stage of the cycle. For example, mining coal exacts a penalty in human health and safety, may require modification of large areas of land use requiring expensive reclamation and habitat restoration, and frequently produces polluting liquid and solid mine wastes. Environmental, social, economic and health effects also accompany the transportation, storage, treatment, combustion, and waste management and disposal aspects of the fuel cycle. Failure to treat these factors has been criticized by ASLB's and the ASLAB in the past, necessitating increased staff efforts in this direction.

Description of Work to be Performed:

The purpose of this task is to provide a technical basis for the detailed generic assessment of the environmental effects of using coal for generating electricity, and to provide for a comparison of these effects with those of using uranium. It will be necessary to identify and evaluate the impacts associated with each phase of the fuel cycle such as mining, transportation, storage, treatment, combustion, and waste management and disposal. The major impacts of concern include but are not limited to land use and reclamation, gaseous (including radon) and particulate emissions, point and nonpoint source liquid discharges, acid mine drainage, acid precipitation, toxic effects of effluents and solid wastes, effects of heavy metals and organic substances, solid waste disposal and management, effects on ambient air and water quality, effects on the integrity of natural habitats, effects of noise and vibration, impacts on transportation systems, and other impacts such as possible "green house effects." It is anticipated that the literature review will reveal other impacts that should be included to provide a consideration of environmental, social, economic and health effects associated with the coal fuel cycle comparable to our present consideration of the effects of the uranium cycle.

The scope of work should also include consideration of the major variables associated with the coal fuel cycle that determine the kinds and degree of impacts that will occur. Some examples of these variables are sulfur and trace metal content; heat content of the coal; technological options for mining, treatment, transportation, storage, combustion, waste management and disposal, and emission control; land reclamation practices; habitat restoration; and other impact mitigation measures and differences in siting practices. The basic purpose of this study will be to develop information for the coal cycle to the same level and scope as that of the S-3 table. However, the study could develop information on some significant coal cycle impacts for which the corresponding nuclear impacts are negligible. The discovery of such impacts could lead to modifications of the scope of the S-3 table so that these negligible impacts would at least be identified.

While it is anticipated that no new basic research will be necessary, synthesis of information from a variety of disciplines may be needed to attain the goals of this study. It is expected that many impacts will not be quantifiable to a high degree of accuracy, and others will not be quantifiable at all (such as the "greenhouse" effect). The proposed generic assessment of environmental, social, economic and health impacts will base its findings on available state of the art technology; it will not postulate or depend on the development of new or experimental technology. Numerous recent technical reports are available that should help provide the necessary data. A partial list of references is attached.

The recent report prepared for NRC by Argonne National Laboratory (NUREG-0252) entitled, "The Environmental Effects of Using Coal for Generating Electricity," is especially relevant as a source document. However, it is not as comprehensive as is proposed for this study. For example, this report does not include comprehensive information on the externalized economic and social costs. Moreover, it does not consider the likelihood or frequency of occurrence of specific impacts. However, this report is an excellent point of departure for organizing the outline of the anticipated study because of its broad recognition of the phases of the fuel cycle and the kinds of environmental impacts that are associated with each.

In developing its assessment of impacts, the study will assume that reasonable and legally required mitigative measures are being used at every portion of the fuel cycle as applicable. The study will further assume that the economic costs of these mitigative measures will be reflected in the capital costs of plant and equipment and in fuel prices paid by the user. Assessment of these internalized costs is not a part of this study. The focus of this study is directed toward an evaluation of the externalized costs and related impacts expressed where possible in quantifiable terms.

Selected References

1. USAEC, December 1972, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," WASH-1238, Washington, D.C.
2. USAEC, April 1974, "Environmental Survey of the Uranium Fuel Cycle," WASH-1248, Washington, D.C.
3. USAEC, December 1974, "Competitive Risk-Cost-Benefit Study of Alternate Sources of Electrical Energy," WASH-1224, Washington, D.C.
4. USNRC, June 1977, "The Environmental Effects of Using Coal for Generating Electricity," NUREG-0252, Washington, D.C.
5. USNRC, October 1976, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," NUREG-0116, Washington, D.C.
6. USNRC, March 1977, "Public Comments and Task Force Responses Regarding the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," NUREG-0216, Washington, D.C.
7. Pennsylvania Department of Education, 1975, "The Environmental Impact of Electrical Power Generation: Nuclear and Fossil," ERDA-69, Harrisburg, Pennsylvania.
8. Hittman Associates, Inc. (1974 and 1975), Environmental Impacts Efficiency and Cost of Energy Supply and End Use, Final Report: Vol. I, 1974; Vol. II, 1975, Columbia, Maryland.
9. Teknekron, Inc. (1973), Fuel Cycles for Electrical Power Generation, Phase 1: Towards Comprehensive Standards: The Electric Power Case, Report for the Office of Research and Monitoring, EPA, Berkeley, California.
10. Energy Alternatives: A Comparative Analysis, University of Oklahoma, The Science and Public Policy Program Prepared for CEQ, ERDA, EPA, FEA, DOI and NSF. May 1975.
11. Gotchy, R., July 16, 1977, "Supplemental Testimony Regarding Health Effects Attributed to the Coal and Nuclear Fuel Cycle Alternatives." Testimony delivered at Oswego, N.Y. re: Sterling Nuclear Power Station.

2. Plan for Problem Resolution:

Results of the Study:

The study will provide the technical basis for a staff paper that could be used generically by the NRC as necessary. After a rulemaking action, it could be incorporated into every EIS by reference in the same manner as Table S-3 of 10 CFR Part 51.20(e). The anticipated end product would be a table or set of tables patterned after Table S-3. However, because of the nature of the coal cycle, it may be that quantified impacts suitable for inclusion in tables might be difficult to obtain in all areas of impact. Coal reserves are widely dispersed and there are many different types of coal, coal mining techniques, transportation methods, and combustion technologies in use, each of which has different properties and problems.

The new tables would be supported by full documentation of all inputs and detailed breakdowns of relevant influencing factors. The table entries would be the identified environmental impacts characteristic of the coal fuel cycle. The tables developed must permit ready comparison of the environmental impacts of the coal fuel cycle with the environmental impacts of the uranium fuel cycle.

Since the results will have to accommodate the variant types of coal, locations of mines, combustion techniques, and waste management techniques, impacts presented in the tables will likely have wide ranges of values. Supporting documentation will be necessary to fully explain the variability of each of the entries in the tables so as to permit proper selection of a table for inclusion in a specific EIS.

As stated earlier, the results of this study will be presented in terms which facilitate direct comparisons between the impacts of the coal and uranium fuel cycles, assuming that all reasonable and regulatory mitigative actions are implemented at each stage of the fuel cycle. Because of the fact that Table S-3 presents impacts on the basis of a standardized power plant, the table developed for coal should also present information on the basis of impacts annualized and normalized to a model 1,000 MWe base load plant operating at capacity factors of 50, 60, 70 and 80 percent. Possible sources of error inherent in the normalization process due to scaling should be identified and discussed so that they may be incorporated into the comparative process.

All results and documentation must be tailored to facilitate direct incorporation into the licensing process. The amount of information concerning the coal fuel cycle is vast, requiring judicious selection of information for inclusion in the results of this study. The prime criterion for evaluation of the study results is that they should be applicable to the decision making requirements of NEPA as implemented by the NRC in its preparation of EISs and the subsequent hearing process. The results should focus on significant factors relevant to the balancing of irreversible and irretrievable commitment of resources, the relationships between local short term uses of the human environment, and the maintenance and enhancement of long term productivity. The primary application of the results of this study will be resolution of NEPA issues arising out of licensing of a nuclear power plant.

General Approach and Methodology:

For the purposes of this study and to describe environmental impacts, a model or reference system for the coal fuel cycle has been selected. This is shown in Figure 1. Selection of the model system was constrained by the availability of requisite technologies (i.e., require no major scientific or engineering advances).

Each step within the coal fuel cycle has associated environmental impacts. These are frequently separate in both place and time and will be discussed separately and sequentially.

The project effort will be accomplished principally through outside contractual effort under the general overview of the responsible Assistant Director and designated Task Manager with technical assistance of the Task Committee to be organized within NRC. The Task Committee, in addition to the responsible Assistant Director, will consist of representatives of each of the branches designated in 3. and 6., below. Coordination and cooperation of other Federal and State agencies will be maintained throughout with assistance of these agencies being sought in the data and information gathering stage of the work.

The project is proposed to be conducted in the following general stages or principal project tasks. Project tasks are more fully described within the overall project framework and timeframe given in Section 7, Schedule for Problem Resolution. The principal project tasks are given here for general reference and orientation to the reader.

FIGURE 1

GENERAL ENVIRONMENTAL EFFECTS OF THE COAL RESOURCE SYSTEM

Exploration	Mining and Reclamation	Coal	Within or Near Mine Transportation	Beneficiation (crushing, screening, cleaning and drying)	Processing[a]
Physical Disturbances	Solid Waste		Liquid Wastes	Liquid Wastes	
Noise	Liquid Wastes		Air Pollutants	Air Pollutants	
	Air Emissions		Haul Road or Land Use	Solids	
	Water Discharge		Noise	Land Use	
	Mine Safety		Accidents	Health	
	Land Consumption		Health		
	Noise				
	Aesthetics				

Transportation (Rail, Barge, Truck, Slurry Pipeline)	Electric Power Generation	Electric Energy Transmission
Rights-of-way	Liquid Discharge	Land Use
Liquid Waste	Thermal from blowdown	Surface Runoff
Air Pollutants	Cooling Tower Plume	Noise
Solids	Stack Gaseous Emissions	AM/TV Interference
Land Utilization	Dust	Shock
Health	Gaseous Pollutants (SO_x, NO_x, particulates)	Health Effects
	Solids	
	Land Utilization	
	Health	
	Combustion Emissions (fly ash, sulfur and trace elements)	

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[a] Various and numerous alternatives for coal processing to produce gaseous, liquid and and solvent refined solids are not to be included in this project.

TASK	EFFORT
1.	Organization of NRC Task Committee
2.	Establishment of contacts with Federal and state agencies, review of available information, and determination of specific study needs.
3.	Review and assess internal information on health effects of coal and nuclear fuel cycles and National Academy of Sciences report.
4.	Preparation, review of proposals and selection of qualified contractor.
5.	Contractor to submit detailed study approach and methodology
6.	Preparation of contractor report and review by NRC task committee.
7.	Final draft and review by Federal and other outside comment.
8.	Staff paper on coal/nuclear environmental effects and recommendations for use in EIS preparation process

3. <u>NRR Technical Organizations Involved</u>:

a. Environmental Projects Branch 1, Division of Site Safety and Environmental Analysis:

Task Manager will service in the principal management function for the project. The Task Manager will have primary responsibility for maintaining coordination, work progress and general monitoring of the work effort.

Estimated Manpower = 6 man-months

b. Other technical organizations within NRR will have representation on the NRC Task Committee to provide technical review of the project, and assistance to the Task Manager and responsible Assistant Director.

Assistance will include periodic review of contractual work, review of status reports submitted by the selected contractor, assistance to the Task Manager in preparation of the RFP, evaluation of contractor submitted study proposals, review of study report drafts, and general technical advice and assistance to the Task Manager.

The Task Committee will meet quarterly to review the status of the study progression. More frequent meetings may be necessary over short time periods should problems develop, and as requested by the responsible Assistant Director and/or Task Manager. The following organizations will be represented on the Task Committee with the total estimated manpower levels as indicated:

(1) Division of Site Safety and Environmental Analysis, Environmental Specialists Branch

Estimated Manpower = 3 man-months

(2) Division of Site Safety and Environmental Analysis, Cost Benefit Analysis Branch

Estimated Manpower = 3 man-months

(3) Division of Site Safety and Environmental Analysis, Radiological Assessment Branch

Estimated Manpower = 3 man-months

With respect to ongoing Radiological Assessment Branch activities, the staff has recommended to the Commission preparation of a draft NUREG by July 1977 that revises the present assessment of health effects set forth in the supplemental testimony prepared by Dr. Reginald Gotchy, suggested agency comment on the draft and preparation of a final NUREG by October 1977, and a determination of the advisability of proceeding with rulemaking in the health effects area by November 1977. A final rule, if judged appropriate, could result by May-July 1978.

Also, RAB is presently sponsoring an ongoing study under contract with Argonne National Laboratory. The ANL study will develop improved models for estimating risks of mortality, disease and consequent life-shortening due to the coal and nuclear fuel cycles. The first phase of this study is scheduled for completion in October 1977, with additional modules due in FY 1978 and FY 1979.

Results of the ANL study as well as any comments on the draft NUREG will be incorporated into the present coal fuel cycle study to the greatest extent possible as they become available. Close coordination will be maintained.

4. Technical Assistance Requirements:

   a. Contractual Effort:

      The nature of the task will require a contractual effort to address the technical aspects of the problem. The contractual work performed under the general direction of the Task Manager and the NRC Task Committee for technical guidance is needed to perform the detailed work required, i.e., the study reconnaissance, literature search, analysis of available information, and development of the study conclusions and results. Performing this work under contract allows the staff time for independent study and evaluation, so as to best direct and monitor the contractual work. The actual magnitude of the task will be contingent on the scope and depth of a related study currently being performed by the National Academy of Sciences (NAS). The results of the NAS study are expected in August 1977. The NRC Task Committee should collectively prepare the final contract scoping documents and review any proposals.

      Estimated Contract Cost = \$50,000-\$250,000* for work to be performed over a 12-18-month period

5. Interaction with Outside Organizations:

   Interaction with other Federal and State agencies will be established by the Task Manager early in the study to develop an understanding of the study effort among these agencies, and to solicit data, information and literature sources related to the effort. The result of this interaction and preliminary gathering and evaluation of information will guide the Task Committee in determining the specific direction and need for emphasis in the study in light of presently available information. Of particular interest is the study presently being conducted by the National Academy of Sciences in the evaluation of coal, nuclear, and other alternatives. The results of National Academy of Sciences study is now anticipated to be released in August 1977.

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*Does not include the cost of closely related and ongoing ANL contractual study on health effects model development for radioactive and nonradioactive pollutants described under paragraph 3b.(3). Nor does the estimated cost include the cost of the related work being sponsored by NMSS and described under paragraph 6(2) below. The \$200,000 range reflects the uncertain usefulness of the NAS study to NRC.

Other than general interaction and coordination with other Federal and state agencies, the technical evaluation and assessment of this study will be carried out by the selected contractor and the NRC staff. This is judged to be the most practical and productive approach. After completion of the final draft of the study document with the formulation of specific proposed study results and conclusions, increased interagency involvement will occur through the request for comments on this draft.

6. Assistance Requirements from Other Than NRR Offices:

Coordination and technical assistance will be required from the following technical organizations.

a. Assistance will be required from these technical organizations only as necessary to assure general conformity and consistency with present or planned activities and the specific responsibilities of these groups. Overall technical review of the project will be sought. Representatives of these groups will be requested to review and comment upon the proposed contract, quarterly status report, and final study report drafts. Advisory assistance may be solicited in the assessment of any contractor submitted study proposals which are generally reviewed by the Task Committee, as well as lending minor technical advice and assistance to the Task Committee and Task Manager. These activities are included within Tasks 3, 4, 6 and 7 of the Schedule for Problem Resolution (Section 7).

(1) Office of Nuclear Regulatory Research, Division of Safeguards, Fuel Cycle and Environmental Research, Health and Environmental Research Branch

Estimated Manpower = 0.5 man-months

(2) Office of Standards Development, Division of Siting, Health and Safeguards, Standards, Environmental Standards Branch

Estimated Manpower = 0.5 man-months

(3) Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle and Material Safety, Technology Assessment Branch

Estimated Manpower = 0.5 man-months

The Division of Fuel Cycle and Material Safety has recently developed an action plan for conducting a complete revision and update of the uranium fuel cycle survey to assess the environmental impacts of the front-end of the LWR fuel cycle. Basic studies are scheduled for completion in February 1978 with preparation of revised WASH-1248, Environmental Survey of the Uranium Fuel Cycle, scheduled to begin in July 1978 and extending to November 1978. These study results will be fully considered in the present coal fuel cycle study to assure a comparable consideration of impacts.

7. Schedule for Problem Resolution:

Major milestones and target dates for completion of the study are given below. The work progression should proceed as described. Time 0 is the date of award of the contract.

The contractor is to submit monthly letter progress reports to the Task Manager and more detailed status reports for review by the Task Committee on a quarterly basis. All work is to be performed subsequent to approval of the contractor's detailed plan. Changes or additions in the contractor's work plan (but within the sense of work described within the contract) must be requested and approved by the Task Manager and responsible Assistant Director operating through the Task Committee.

Interim presentations of project work status and results will be made to the NRC staff by the contractor upon request where conditions reasonably justify this action.

8. Potential Problems:

Some readily recognized problems that could affect the successful outcome of the study within the Task scope and/or adherence to the anticipated study schedule are as indicated below:

a. The availability of information and data. Considerable data and information presently exists on the environmental, social, economic and health impacts of the coal fuel cycle--some quantitative and some only descriptive. The study will attempt to quantify all impact factors and variables where possible, thoroughly describe and assess other nonquantifiable factors, and identify areas where sufficient information does not exist or where basic research may be needed.

		Task Description	Completion Target In Cumulative Months
	a.	Pre-contract award	
TASK	1)	Organize NRC Task Committee	-8
TASK	2)	Establish contacts with Federal and state agencies, review and assess available information, and determine specific study needs with available information and data	-6
TASK	3a)	Review comments on draft NUREG on health effects of the coal and nuclear fuel cycles [1]	-5
TASK	3b)	Review and assessment of National Academy of Sciences report	-5
TASK	4a)	Preparation of detailed contract scope by Task Committee	-4
	4b)	Advertisement and solicitation of bids and proposals [2]	-3
	4c)	Task Committee review of proposals [2]	-2
	4d)	Selection of most qualified contractor and award of contract [2]	-1
	b.	Post-contract award	
TASK	4e)	Award of contract	0
TASK	5)	Contractor submit detailed plan of approach and methodology for the study for NRC Task Committee review and approval (more detailed than included in contract)	1

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[1] See ongoing work described for ANL and procedure proposed by NRR to the Commission for generic rulemaking and expanded distribution and formal requests for comments on the staff assessment of the comparative health effects of the coal and nuclear fuel cycles [Item 3.b)2) of this study proposal].

[2] If accomplished under RFP.

	Task Description	Completion Target In Cumulative Months
TASK 6a)	Incorporation of Division of Fuel Cycle and Material Safety uranium fuel cycle environmental impact revision results	2-11
TASK 6b)	Completion of thorough literature review by contractor, compilation of available information and data, and analysis and evaluation of all input	9[2]
TASK 6c)	Contractor completion of initial draft report for presentation to NRC	11[2]
TASK 6d)	Formal Review and Comment on initial Draft Report by NRC Staff (Task Committee)	12[2]
TASK 7a)	Final draft prepared for Federal agency and other outside comment	14[2]
TASK 7b)	Comments received from Federal agencies and others	16[2]
TASK 7c)	Completion of Final Report to NRC by Contractor	18[2]
TASK 8)	Staff Paper on Coal/Nuclear Fuel Cycle Environmental Effects & Recommendation For Use in EIS Process	21[2]

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[2] Depending on the scope and depth of the National Academy of Sciences report, these milestones could be reduced by as much as 3 to 6 months.

b. The study must review past findings and results of previous studies in the perspective of changing technology and revised cost structure. This is an inherent difficulty in conducting any major study of this nature which could result in technical difficulties and delay in the work effort.

c. Reliance upon interagency cooperation in part to ascertain the existence of available data and information and to comment on the final draft report. Lack of full cooperation by other governmental and private agencies could result in project delay and difficulties in obtaining full and complete input information.

Task Action Plan

REVISION 0
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Task No. A-23

Title: Containment Leak Testing (TAC-4585)

Lead Responsibility: Division of Systems Safety

Lead Assistant Director: R. L. Tedesco, AD/PS, DSS

Task Manager: J. W. Shapaker, CSB/DSS

1. Problem Description

Appendix J to 10 CFR Part 50 was issued February 14, 1973. Since that time, certain requirements of the appendix have been found to be conflicting, impractical for implementation, or subject to a variety of interpretations by the NSSS vendors, architect-engineers, utilities and the staff. These requirements make it difficult to determine if applicants and licensees have developed uniformly acceptable containment leak testing programs and for field inspectors to judge the acceptability of a licensee's containment leak testing practices. This also leads to increases in the time devoted to leak testing and can unnecessarily delay the return of a plant to service following a refueling outage. Therefore, it is necessary that Appendix J be revised to clarify existing requirements, as well as incorporate containment leak testing experience. This will make Appendix J a more easily implemented regulation and provide greater assurance that containment integrity is being effectively monitored.

2. Plan for Problem Resolution

A. Approach

The Containment Systems Branch (CSB) is developing a list of proposed changes to Appendix J based on information obtained from utility applicants and licensees, architect-engineering firms, the ANS work group engaged in the development of an industry standard on containment leak testing, the Applied Statistics Branch (ASB) in the Office of the Executive Director for Operations (EDO), the Plant Systems Branch (PSB) in the Division of Operating Reactors (DOR) and the Division of Operating Reactors Inspection (DORI) of the Office of Inspection and Enforcement (OIE), regarding containment leak testing experience and data analysis, and the interpretation of Appendix J requirements. Concurrence on the proposed changes to Appendix J will be obtained from cognizant groups within the Office of Nuclear Reactor Regulation, the Office of Inspection and Enforcement and the Applied Statistics Branch, prior to forwarding them to the Office of Standards Development for use in revising Appendix J to 10 CFR Part 50.

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B. End Product

The end product for this task effort will be the forwarding of a memorandum to the Office of Standards Development from the Office of Nuclear Reactor Regulation detailing the changes that should be made to Appendix J to 10 CFR Part 50. The memorandum will address the technical changes sought and will present sufficient value/impact information for the OSD to complete the assessment required for all regulation changes.

C. Tasks:

The tasks required to resolve this problem include the following:

(1) Resolution and incorporation of comments, received from cognizant groups, into initial draft of proposed changes to Appendix J.

(2) Review of final draft of proposed changes to Appendix J by PSB (DOR) and DORI (OIE) and receipt of comments.

(3) Statistical evaluation of the methods of leak testing and data reduction for the containment integrated leak rate test by ASB (EDO) and incorporation of comments into proposed changes to Appendix J.

(4) Resolution and incorporation of PSB (DOR) and DORI (OIE) comments and submitting requested changes to Appendix J, with associated background material to the Office of Standards Development (OSD) for use in revising Appendix J to 10 CFR Part 50.

3. NRR Technical Organizations Involved

A. Division of Systems Safety, Containment Systems Branch

(1) Task No. 1 - The Containment Systems Branch will have primary responsibility for developing a list of proposed changes to Appendix J based on information obtained from organizations outside the Commission and from cognizant groups within the Commission. This will entail the soliciting of comments from cognizant groups within the Commission and resolving with them the comments received.

(2) Task No. 4 - The Containment Systems Branch will resolve and incorporate the comments that were obtained from PSB (DOR) and DORI (OIE), develop implementation criteria, and submit a final draft to the task manager for transmittal to OSD.

(3) It is estimated that about 0.4 man-years will be required in FY 77 to accomplish Task No. 1 and about 0.22 man-years will be required to accomplish Task No. 4 in FY 78.

B. Division of Operating Reactors, Plant Systems Branch

(1) Task No. 1 - Initial comments have been received from PSB (DOR) and are being reviewed. The Plant Systems Branch (DOR) will assist in the resolution of comments.

(2) Task No. 2 - The Plant Systems Branch (DOR) will review and comment on a redraft of the proposed changes to Appendix J.

(3) Task No. 3 - The Plant Systems Branch (DOR) will review and comment on the ASB (OED) evaluation of methods of leak testing and data reduction for the containment integrated leak rate test (CILRT) with a view towards establishing CILRT acceptance criteria for inclusion in Appendix J.

(4) Task No. 4 - The Plant Systems Branch (DOR) will assist in the resolution of comments, and the development of implementation criteria.

(5) It is estimated that about 0.1 man-year will be required in FY 77 for Task No. 1 and about 0.2 man-year for Task Nos. 2, 3 and 4 in FY 78.

4. Technical Assistance Requirements

None required.

5. Interactions with Outside Organizations

No formal interaction with outside organizations is required. The information previously obtained from outside organizations has been through normal working relations with them.

6. Assistance Requirements from Other NRC Offices

A. Office of Inspection and Enforcement, Division of Operating Reactors Inspection.

(1) Task No. 2 - The Division of Operating Reactors Inspection (OIE) will review and comment on the proposed changes to Appendix J, as well as identify additional changes which they feel should be included to improve the practicability of the regulation.

(2) Task No. 3 - DORI (OIE) will review and comment on the ASB (OED) evaluation of methods of leak testing and data reduction for the containment integrated leak rate test (CILRT) with a view towards establishing CILRT acceptance criteria for inclusion in Appendix J.

(3) Task No. 4 - DORI (OIE) will assist in the resolution of comments.

(4) Responsibility - Submit comments to the task manager in accordance with the attached schedule and assist in the resolution of the comments.

(5) It is estimated that about 0.18 man-years will be required in FY 78 to accomplish Task No. 2, 3 and 4.

B. Office of the Executive Director for Operations, Applied Statistics Branch

(1) Task No. 3 - The Applied Statistics Branch will provide technical support in the evaluation of methods of leak testing and data reduction for the containment integrated leak rate test (CILRT). As a result of this evaluation, the CILRT acceptance criteria will be established.

(2) Responsibility - Submit evaluation to the task manager in accordance with the attached schedule.

(3) It is estimated that about 0.53 man-years will be required in FY 77/78 to accomplish Task No. 3.

7. Schedule for Problem Resolution

A. Summary of Schedule

Comments on the initial draft of the proposed changes to Appendix J have been received from within ONRR. These comments will be incorporated, and a final draft issued to the Plant Systems Branch (DOR) and the Division of Operating Reactors Inspection (OIE) for review and comment. Their comments on the final draft and on the ASB (OED) evaluation will be incorporated, and the proposed changes to Appendix J will be forwarded to the Office of Standards Development requesting that Appendix J be revised accordingly. Completion dates for the major milestones are as follows:

Milestone	Task No. Completed	Date
30	1	9/23/77
40	2	11/04/77
50	3	12/16/77
60	4	12/30/77

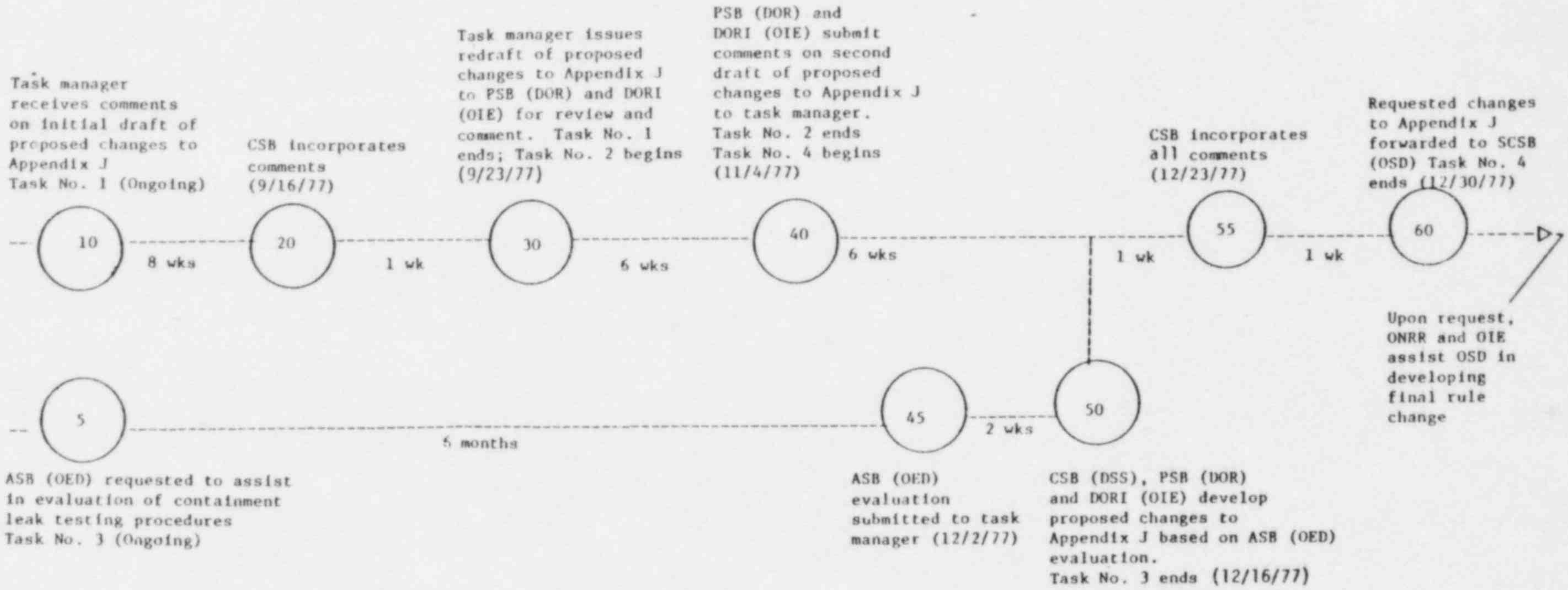
B. Detailed Schedule

(See attached chart).

C. Technical Assignment Control Form Number for Major Task - The number to which all effort should be charged is 4584.

8. Potential Problems

Since all cognizant groups have a deep interest in the outcome of the forthcoming revision to Appendix J, achieving a consensus among them may be difficult.


Task manager receives comments on initial draft of proposed changes to Appendix J Task No. 1 (Ongoing)
10
8 wks
CSB incorporates comments (9/16/77)
20
1 wk
Task manager issues redraft of proposed changes to Appendix J to PSB (DOR) and DORI (OIE) for review and comment. Task No. 1 ends; Task No. 2 begins (9/23/77)
30
6 wks
PSB (DOR) and DORI (OIE) submit comments on second draft of proposed changes to Appendix J to task manager. Task No. 2 ends Task No. 4 begins (11/4/77)
40
6 wks
1 wk
CSB incorporates all comments (12/23/77)
55
1 wk
Requested changes to Appendix J forwarded to SCSB (OSD) Task No. 4 ends (12/30/77)
60
Upon request, ONRR and OIE assist OSD in developing final rule change
5
6 months
ASB (OED) requested to assist in evaluation of containment leak testing procedures Task No. 3 (Ongoing)
45
2 wks
ASB (OED) evaluation submitted to task manager (12/2/77)
50
CSB (DSS), PSB (DOR) and DORI (OIE) develop proposed changes to Appendix J based on ASB (OED) evaluation. Task No. 3 ends (12/16/77)

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TASK ACTION PLAN

Task No. A-24

TITLE: Qualification of Class IE Safety Related Equipment - TAC 4586

LEAD RESPONSIBILITY: Division of Systems Safety

LEAD ASSISTANT DIRECTOR: Robert L. Tedesco, AD for Plant Systems, DSS

TASK MANAGER: A. Szukiewicz, ICSB, DSS

1. Problem Description

It is the NRC position that construction permit applicants for which a Safety Evaluation Report was issued after July 1, 1974, are required to qualify all safety related equipment to the requirements established in IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations."

From the conception of the standard, industry has been developing methods that will be used to qualify their equipment in order to satisfy the objectives of the standard. Certain proposed concepts and methods used by industry in addressing equipment qualification, such as testing margins, aging effects on materials and equipment, and adequacy of testing simulators, which simulate the worst case environment for the equipment have not yet been resolved.

In order to expedite the review and assess the adequacy of the qualification methods, on a case by case basis, a generic approach to review the equipment qualification methodology and the acceptance criteria used by the major Nuclear Steam Supply (NSS) and Balance of Plant (BOP) equipment suppliers must be conducted, and resolution of the above actions for equipment qualification must be established.

2. Plan for Problem Resolutions

A. Approach

The staff will request the nuclear steam system suppliers and the standard balance of plant equipment suppliers to submit their safety related equipment qualification programs that describe the

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methods, acceptance criteria, and test procedures that are used or will be used to qualify their equipment to the requirements established in IEEE Standard 323-1974 (as augmented by Regulatory Guide 1.89). The enclosed appendix provides the current status of this effort.

The individual nuclear steam supplier and the balance of plant supplier responses will be evaluated by the staff in order to assess if the degree of conformance of these programs to the requirements established in IEEE Standard 323-1974 (as augmented by Regulatory Guide 1.89) are acceptable.

B. End Product

The evaluations for each of the nuclear steam system and balance of plant equipment suppliers will be in the form of a topical report evaluation. It is envisioned that once the criteria, methodology, and equipment scope is defined in the topical reports, and accepted by the staff, these topical reports will serve as an acceptable basis for equipment qualification review on a case by case basis. The results of the review could be used by DOR where applicable.

C. Tasks (General)

1) Request qualification methods, test plans and test procedures from all standard plant nuclear steam supply system suppliers. (i.e., Westinghouse, Babcock and Wilcox, General Electric and Combustion Engineering).

2) Request qualification methods, test plans and test procedures from all standard balance of plant architect engineers. (i.e., Stone and Webster - SWESSAR, Fluor Pioneer - BOPSSAR, Gibbs and Hill - GIBBSSAR).

3) Review and evaluate the qualification methods, test plans and test procedures for each major equipment type (e.g., sensors, valves, motors and logic modules) and assess the degree of conformance of the design to IEEE Standard 323-1974 and the Standard Review Plan Sections 3.10 and 3.11.

4) An audit of final test results on selected safety related equipment which were tested in accordance with IEEE Standard 323-1974 will be conducted when this task is completed. This review of the final test results will be conducted on a case basis during the OL review, to verify that the design as implemented conforms to the requirements established in the topical reports in accordance with the standard review plan.

3. NRR Technical Organizations Involved

A. DSS/Containment Systems Branch

1) Required Safety Related Equipment Scope

Assure that all safety related equipment inside containment, required for design basis events and reactor shutdown, that have been identified by RSB are evaluated.

2) Validity of Environment

a) Evaluate the validity of normal, abnormal, and accident environments inside containment.

b) Establish environmental envelope requirements (e.g. pressure, temperature, and time of exposure) to which safety related equipment must be qualified to.

3) Qualification Methodology

a) Evaluate the methodology used to simulate environments inside containment and confirm that the simulated environments provide a sufficient margin envelope for design basis events for LOCA and MSLB. And, assist in evaluating aging concepts utilized in the qualification of safety related equipment to assure conformance to IEEE Standard 323-1974.

b) Evaluate proposed analyses which justify qualification acceptability.

4) Report

Submit equipment qualification evaluation to Task Manager.

5) Manpower Requirements

FY-78 -- 12 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 9 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 21 man-weeks

B. DSS/Auxiliary Systems Branch

1) Required Safety Related Equipment Scope

a) Confirm that the safety related equipment outside containment, required for high and moderate energy line breaks and other design basis events such as flooding and pipe whip that have been identified are adequately enveloped environmentally.

b) Assure that all safety related equipment outside containment required for design basis events and reactor shutdown, that have been identified by RSB, are evaluated.

2) Validity of Environment

a) Evaluate the validity of normal, abnormal and accident environments that have been identified based on accident condition input by AB.

b) Establish environmental requirements outside containment (with required margin) to which safety related equipment must be qualified to (including time of exposure).

3) Qualification Methodology

Evaluate the methodology used to identify and simulate environments outside containment and confirm that the simulated and/or analyzed environments provide sufficient margin to envelope the expected range of operating conditions. Also, assist in evaluating aging concepts used in the qualification of safety related equipment to assure conformance to IEEE Standard 323-1974.

4) Report

Submit equipment qualification evaluation to Task Manager.

5) Manpower Requirements

FY-78 -- 24 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 18 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 42 man-weeks

C. DSE/Accident Analysis Branch

1) Required Safety Related Equipment Scope

Evaluate the validity of normal, abnormal and accident radiation environments that have been identified for inside and outside containment.

2) Validity of Environment

a) Establish radiation environment envelope requirements (with required margin) to which safety related equipment for inside and outside containment must be qualified to.

b) Verify the adequacy of the designs' conformance to Regulatory Guide 1.89, with regard to radiation doses and methodology used to simulate the required environments on safety related equipment inside and outside containment considering the postulated conditions of the event being evaluated.

3) Evaluate aging concepts (due to radiation) addressed in the qualification of equipment.

4) Report

Submit equipment qualification evaluations to Task Manager.

5) Manpower Requirements

FY-78 -- 8 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 6 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 14 man-weeks

D. DSS/Reactor Systems Branch

1) Required Safety Related Equipment Scope

Confirm that the safety related equipment identified as being required for certain design basis events and reactor shutdown both inside and outside containment is correct.

2) Validity of Environment

RSB will verify time spans necessary for equipment availability.

3) Report

Submit equipment qualification to the Task Manager.

4) Manpower Requirements

FY-78 -- 4 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 3 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 7 man-weeks

E. DSS/Analysis Branch

1) Reviews Accident Flow Rates

Evaluate the adequacy of the mass and energy flow rates for postulated breaks in piping systems either inside or outside of the containment and provide their input data to CSB and ASB.

2) Report

Submit evaluation and/or requirements to CSB and ASB.

3) Manpower Requirements

FY-78 -- 8 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 6 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 14 man-weeks

F. DSS/Power Systems Branch

1) Required Safety Related Equipment Scope

Assure that all safety related equipment inside and outside containment required for design basis events and reactor shutdown that have been identified by RSB are evaluated.

2) Qualification Methodology

Evaluate the methodology used to qualify safety related power systems equipment (i.e., motors, pumps, valves, fans, switchgear, breakers, batteries, electrical penetrations, inverters, cables, etc.), and confirm that functional operability of the safety related power systems equipment has been verified.

3) Qualification Evaluation

a) Confirm that the applicable safety related equipment adequately conforms to the requirements established in IEEE Standard 382 (augmented by Regulatory Guide 1.73, IEEE Standard 334-1974 (augmented by Regulatory Guide 1.40), IEEE Standard 317-1976 (augmented by Regulatory Guide 1.63), and IEEE Standard 383-1974 (for cable qualification), and satisfies all the requirements of IEEE Standard 323-1974 and IEEE Standard 344-1975 applicable to their equipment scope of review (which includes aging).

4) Report

Submit evaluation and/or requirements to the Task Manager.

5) Manpower Requirements

FY-78 -- 64 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 48 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 112 man-weeks

G. DSS/Instrumentation and Control Systems Branch

1) Required Safety Related Equipment Scope

Verify that all safety related equipment inside and outside containment, required for design basis events and reactor shutdown, that have been identified by RSB are evaluated.

2) Qualification Methodology

Evaluate the methodology used to qualify safety related equipment (i.e., protection system logic, relays, limit switches and sensors) and confirms that functional operability of safety related instrumentation and control equipment has been verified.

3) Qualification Evaluation

a) Verify that the qualification methodology used, conforms to the requirements established in IEEE Standard 323-1974 for their equipment scope of review including aging.

b) Verify that the qualification methods adequately simulate the normal and accident environmental conditions required by the Containment Systems Branch, Auxiliary Systems Branch, Reactor Systems and Accident Analysis Branches.

c) Verify that the qualification methods adequately demonstrate that functional operability of safety related equipment under seismic and/or accident environment conditions has been satisfactorily addressed.

4) Report

Submit evaluation and/or requirements to Task Manager.

5) Manpower Requirements (later)

FY-78 -- 64 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 48 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 112 man-weeks

H. DSS/Mechanical Engineering Branch

1) Qualification Methodology

a) Evaluate test methodology and analysis used to simulate required response spectra for Class IE equipment and verify that the methods used adequately demonstrate that the equipment conforms to the requirements of IEEE Standard 344-1975 as augmented by Regulatory Guide 1.100 for normal, abnormal and accident conditions.

2) Required Safety Related Equipment Scope

Verify that all safety related equipment required for design basis events and reactor shutdown, both inside and outside containment, that have been identified by RSB and ASB are evaluated.

3) Report

Submit seismic equipment qualification evaluation and/or requirements to the Task Manager.

4) Manpower Requirements

FY-78 -- 28 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 21 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 49 man-weeks

I. Division of Project Management

1) Provide liason between DSS and NSS suppliers, standard BOP suppliers (as identified in item 2).

2) Manpower Requirements

FY-78 -- 4 man-weeks (Based on 4 Topical Report Evaluations)
FY-79 -- 3 man-weeks (Based on 3 Topical Report Evaluations)
Total -- 7 man-weeks

J. Division of Operating Reactors - Plant Systems Branch

1) Provide coordination on program evaluation with task manager to become familiar with vendor program.

4. Technical Assistance Requirements

Not required or anticipated at this time.

5. Interactions with Outside Organizations

A. Sandia Laboratories

1) Research and Development

2) Scope

Task 1: Testing to evaluate synergistic effects from LOCA environments.

Task 2: Accelerated aging modes for Class I components.

Task 3: LOCA radiation source evaluation.

3) Responsibility

Office of Nuclear Regulatory Research

4) The results of this effort will be evaluated by the task group to determine our need with regard to this task action plan (A-24).

6. Assistance Requirements from Other NRC Offices

Not anticipated or required at this time.

7. Schedule for Problem Resolution

A. Summary

1) The overall schedule will require a period of 18 months, three weeks for each evaluation. This includes requesting qualification programs, first round questions, staff positions and final evaluation.

B. Detail

1) See attached enclosure.

C. Technical Assignment Control Form Number for Major Task (all time to be charged to that number by all groups in NRC)

TAC 4586

8. Potential Problems

In order to finalize this task, resolution and finalization of Task A-21 (Main Steam Line Break Inside Containment) and Task A-34 (Instruments for Monitoring Radiation and Process Variables During Accidents) should preceed this task and staff positions regarding these items should be established.

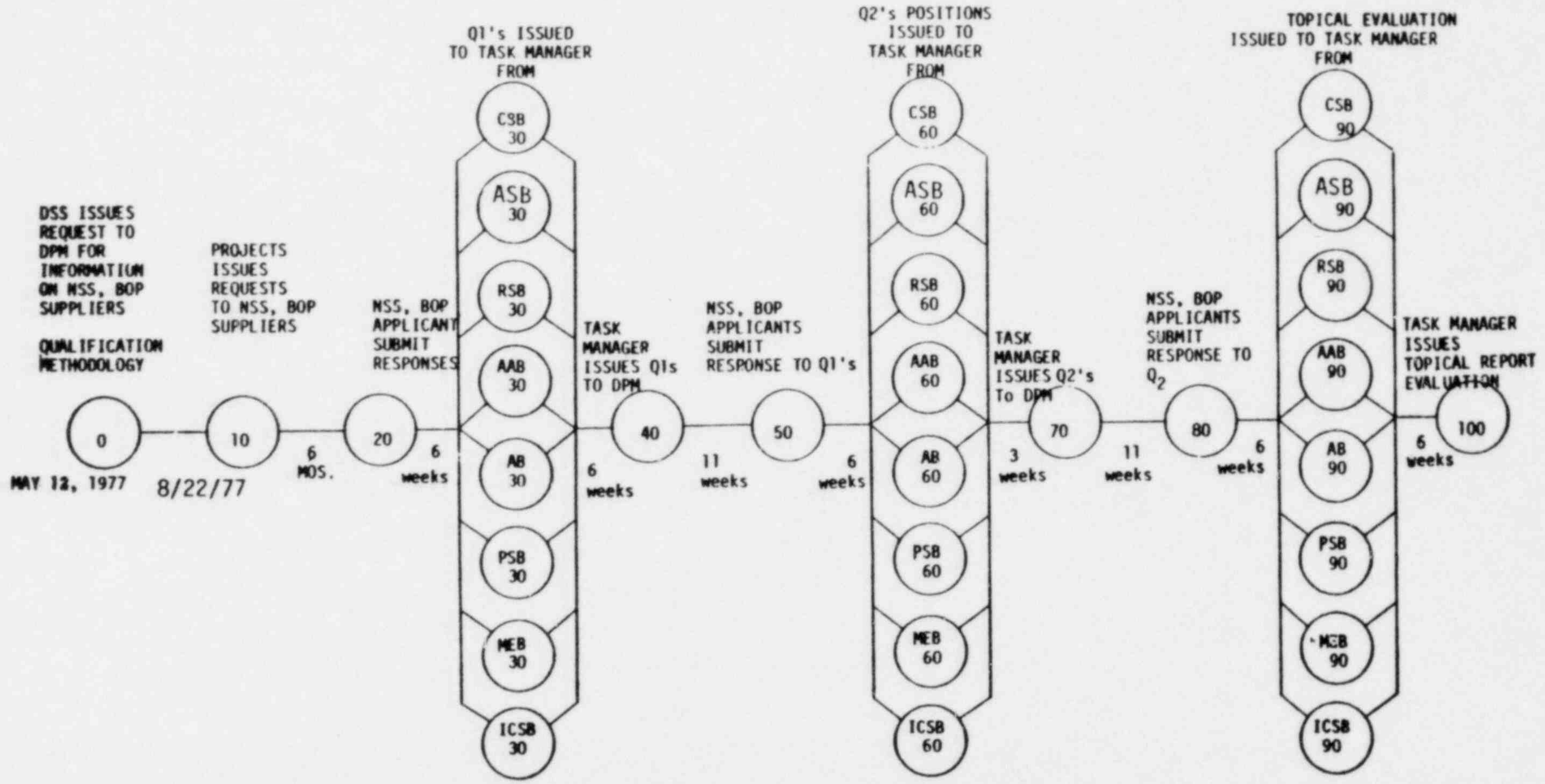
Task A-24

ESTIMATED NRR MANPOWER (IN MAN YEARS)

NRR Organization	Estimate Man Years in Fiscal Year			
	1978		1980	Total
DSS/MEB	0.6	0.47		1.07
DSS/ASB	0.53	0.4		0.93
DSS/RSB	0.08	0.07		0.15
DSS/CSB	0.27	0.2		0.47
DSE/AAB	0.18	0.13		0.31
DSS/AB	0.18	0.13		0.31
DSS/ICSB	1.4	1.07		2.47
DSS/PSB	1.4	1.07		2.47
DPM	0.08	0.07		0.15
Total	4.72	3.61		8.33

ENCLOSURE

PROPOSED ACTION PLAN REVIEW SCHEDULE FOR EACH TOPICAL REPORT



APPENDIX

Status of Equipment Qualification

1. Memo from R. Tedesco to D. Vassallo (dated May 12, 1977) Requests the information identified in Items 2.C.(1), 2.C.(2).

2. The status of item 2.C.(3) is as follows:

   a. Westinghouse Topical Report WCAP 8587 currently under review.

   b. Babcock & Wilcox Co. BAW-10082 Part I (Seismic Qualification) has been reviewed and approved by ICSB. Additional concerns by MEB have not yet been resolved. B&W response to MEB scheduled to be submitted in last quarter of 1977.

      Part II and III of the Topical Report (Environmental Qualification) yet to be submitted.

   c. Combustion Engineering

      1) CENPD-182 (Seismic Qualification) currently under review.

      2) Environmental Qualification Topical Report to be submitted shortly. (i.e., August 1977)

   d. General Electric Company
      Topical Reports yet to be submitted

   e. SWESSAR
      (Status?)

   f. BOPSSAR
      (Status?)

   g. GIBBSSAR
      (Status?)

REVISION 0

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED AUGUST 30,1977

AUG 3 0 1977

TASK ACTION PLAN
TASK NUMBER A-25

Title - Non-Safety Loads on Class 1E Power Sources

Lead Responsibility - Division of Systems Safety/NRR

Lead Assistant Director - R. L. Tedesco (Plant Systems)

Task Manager - J. Calvo (Power Systems Branch)

1. Problem Description:

The Class 1E power sources are part of the onsite emergency power system and provide the electric power for the equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal or are otherwise essential in preventing significant release of radioactive material to the environment. Present regulatory practice allows the connection of non-safety loads in addition to the required safety loads to Class 1E power sources by imposing some restrictions. The question is whether or not the reliability of the Class 1E power sources is significantly affected by allowing the sharing of these sources by loads that perform safety functions and loads that perform normal plant functions (non-safety loads).

An integral problem associated with investigating the practice of connecting non-safety loads on Class 1E power sources arises because there are no existing acceptance criteria governing the application of circuit breakers and fuses as isolation devices. Proper application of these devices could considerably improve the reliability of Class 1E power systems which supply power to non-safety loads.

2. Plan for Problem Resolution:

A. Approach

The approach selected for problem resolution is that of a reliability analysis of typical plant on-site Class 1E power systems. This analysis is to be carried out under technical assistance contract. Staff participation will be limited to technical direction, contract management, end product evaluation, and incorporation of results into the licensing process.

B. End Products

The program calls for two major reports to be supplied by the contractor and each in turn will be factored into the licensing process in the form of branch technical positions and revisions to regulatory guides and parts of standard review plans. In addition to the major reports, a number of status reports on the various milestones will also be submitted by the contractor.

The first report will address the acceptability of using circuit breakers and fuses as isolation devices. This report will establish acceptance criteria and will delineate acceptable circuit configurations. This input can be factored directly into a branch technical position for the short term and can be factored into Regulatory Guide 1.75 "Physical Independence of Electric Systems" and the Standard Review Plan for the long term. This portion of the task will be considered to be complete upon completion of the branch technical position and revision of the Standard Review Plans and the forwarding of recommendations for revision of Regulatory Guide 1.75 to the Office of Standards Development.

The second report will provide numerical results of a reliability assessment and detailed criteria for the practice of connecting non-safety loads on the Class 1E power sources. This input will be factored into a branch technical position and an updating of the Standard Review Plan. This portion of the task will be considered to be complete upon completion of the branch technical position and revision of the Standard Review Plan.

C. Tasks

To assure overall electric power systems reliability, this task action plan will be coordinated to the extent necessary with other electrical related Task Action Plans (A-24, "Oualification of Class 1E Safety Related Equipment;" A-30, "Adequacy of Safety Related D.C. Power Supplies;" A-35, "Adequacy of Offsite Power Systems".

1. Investigate present practices - A questionnaire was sent to applicants and architect engineering firms requesting them to delineate their present practice and to comment on how changes in regulatory requirements in this area would affect safety.

2. Compile failure rate and reliability data -

   Obtain available failure rate data and reliability data for over current protective devices (circuit breaker and fuses) from manufacturers on present day equipment. Purpose is to gather data for task item number 4.

3. Methods for treating reliability problems -

   Investigate methods for treating reliability problems of operational load shedding and faulted load shedding. This task will provide the reliability analysis tool to be used to persue task item numbers 5 and 9.

4. Evaluate isolation device configurations -

   Evaluate various configurations of isolation devices and recommend minimum acceptable designs for the various categories of circuits. This will determine if a single circuit breaker, or a circuit breaker and a fuse, or two circuit breakers, or other arrangement are acceptable for given circuits requiring isolation.

5. Develop isolation device criteria -

   Develop acceptance criteria for the use of circuit isolation devices (circuit breaker and fuses) for both power and control, between safety and non-safety portions of the onsite power distribution system.

6. Compare IEEE Std 384 and Regulatory Guide 1.75 -

   Evaluate the work done in IEEE Std 384-1977 on circuit breakers and fuses as isolation devices using the acceptance criteria of task item number 5 and provide recommendations for changes to Regulatory Guide 1.75.

7. Branch technical position on isolation devices -
   Develop branch technical position and changes to the Standard Review Plan based upon results of task item number 5 for short term applications and forward recommendations to Office of Standards Development for changes to Regulatory Guide 1.75.

8. Reliability assessment -
   Provide the numerical reliability assessments associated with the practice of connecting non-safety loads on the Class 1E busses. (The effects of non-safety loads on the reliability of the Class 1E D.C. Power Supply System will be coordinated with the Task Manager of Action Plan A-30, "Adequacy of Safety Related D.C. Power Supplies.")

9. Develop non-safety load criteria -
   Develop criteria for the sharing of Class 1E power sources by safety and non-safety loads.

10. Branch technical position on non-safety loads -
    Develop branch technical position based upon results of task item number 9 and prepare changes to the Standard Review Plan.

3. NRR Technical Organizations Involved

   A. Power Systems Branch, Division of Systems Safety

      1. Task No. 7 - Develop branch technical position and revision to the Standard Review Plan based upon input from contractor concerning acceptance criteria for the use of circuit breakers and fuses as isolation devices. Recommend changes to Regulatory Guide 1.75 to the Office of Standards Development.

      2. Manhour requirements: FY 1978 - 80 hours
         FY 1979 - 320 hours

   B. Power Systems Branch, Division of Systems Safety

      1. Task No. 10 - Develop branch technical position and revisions to the Standard Review Plan based upon input from contractor concerning criteria for the sharing of the Class 1E power sources by safety and non-safety loads.

      2. Manhour requirements: FY 1978 - 80 hours
         FY 1979 - 320 hours

   c. Plant Systems Branch, Division of Operating Reactors

      1. Task Nos. 7 and 10 - Provide assistance and concurrence in the development of the two branch technical positions.

      2. Manhour requirements: FY 1978 - 20 hours
         FY 1979 - 80 hours

4. Technical Assistance Requirements

A. Oak Ridge National Laboratory

1. Title: Risk Assessment of Non-Safety Loads Connected to Class 1E Power Systems

2. Responsible Division/Branch: Division of Systems Safety/ Power Systems Branch

3. Scope

The contractor is to provide a reliability assessment of the practice of connecting non-safety loads on the Class 1E power systems. He is responsible for all necessary research, data accumulation, methodology development, and evaluation required to complete this task.

A major sub-task included in the contract is to evaluate the acceptability of using circuit breakers and fuses as isolation devices between Class 1E sources and non-Class 1E loads. Contractor is responsible for doing all necessary research, data collection, methodology development and evaluation required to complete this sub-task. As a logical extension of this sub-task, there is to be an evaluation of IEEE Std 384-1977 based upon the acceptance criteria developed.

It is anticipated that the contractor will need to pursue the following items as a minimum to successfully complete the assigned work. In order to establish a base-line as to what is the current regulatory position and the various methods used by industry to comply, a questionnaire was sent to applicants and architect engineering firms asking detailed questions of design philosophy, design implementation, reaction to regulatory practice, and appraisal of possible changes in regulatory practice brought about by changes based upon the results of this study (Task No. 1). In order to obtain a data base for circuit breaker and fuse reliability, requests for reliability data will be requested of the various manufacturers (Task No. 2). Reliability assessment analyses are required tools for evaluating major portions of this task and it is expected that the contractor must familiarize himself with these tools and develop as necessary the methodology to apply these tools to the assigned task (Task No. 3). Given the data and methodology, various configurations of circuit breakers and fuses can be analyzed for their acceptability as isolation devices (Task No. 4).

Design criteria governing the use of circuit breaker and fuses as isolation devices are to be developed and submitted as one of the major required outputs (Task No. 5). A follow-on to this required output is the evaluation of IEEE Std 384-1977 which in part addresses the use of circuit breakers and fuses as isolation devices (Task No. 6). This evaluation is to be based upon the criteria established in Task No. 5.

Following the above assessment of the major components that deal with isolation of non class 1E loads and their Class 1E sources, the final step is to evaluate on a systems basis the overall reliability associated with the practice in question (Task No. 8). The results of the reliability assessment are then to be used to develop criteria for the sharing of Class 1E power sources by safety and non-safety loads (Task No. 9). Formal submittal of this second major report including the evaluation and attendant recommended design criteria completes the contractors obligations.

4. Funding: FY 1977 - $65,000
FY 1978 - $65,000 (requested)
FY 1979 - $49,000

5. Interactions With Outside Organizations

No direct interactions are anticipated. Contractor will be contacting applicants, architect engineering firms, and equipment vendors as necessary to get needed information to complete task.

6. Assistance Requirements from other NRC Offices

Requirements for assistance are not anticipated at this time.

7. Schedule for Problem Resolution

A. Summary of major schedule milestones

Status report on present industry practices (Task 1)	9/1/77
Final report on recommended acceptance criteria for breakers and fuses as isolation devices (includes Tasks 2, 3, 4, and 5)	7/1/78
Status report on evaluation of IEEE Std 384-1977 (Task 6)	10/1/78
Proposed branch technical position on use isolation devices (circuit breakers and fuses) ready for RCCC (Task 7)	1/1/79
Final report on reliability assessment of connecting non-safety loads on Class 1E power sources and recommended design criteria (Tasks 8 and 9)	5/1/79
Proposed branch technical position on practice of connecting non-safety loads on Class 1E power sources ready for RCCC (Task 10)	8/1/79
TOTAL	25 months

B. Detailed Schedule

Bar Chart Enclosed

C. Technical Assignment Control Number - TAC 2140

8. Potential Problems

A. ORNL has only one man assigned to this work. Any contingencies that should affect him (personal or business) will directly affect scheduled events. Further, this individual is now approximately one year up on the specific learning curve for this task a good portion of which would not be recoverable should a reassignment occur. ORNL reports that their manpower allocation is such that this potential problem will probably exist for the duration of the contract.

B. Willingness and responsiveness of those requested by contractor to supply needed information. So far this has not been a problem but not all have responded. Should trouble occur we will need to assist through NRR organizations.

DETAILED SCHEDULE

Status Report Present Practices		9/1/77
Compilation of failure rate data		2/1/78
Investigate applicable reliability methodology		2/1/78
Evaluate various isolation device configurations		4/1/78
Report with Acceptance criteria for circuit breakers and fuses as isolation devices	(draft) (final)	6/1/78 7/1/78
Status report on evaluation IEEE Std 384-1977	(draft) (final)	9/1/78 10/1/78
Branch technical position on use of circuit breakers and fuses as isolation devices ready for RRRC		1/1/79
Report containing criteria for the sharing of Class 1E power sources by safety and non-safety loads and numerical results of reliability analysis	(draft) (final)	4/1/79 5/1/79
Branch technical position on practice of connecting non-safety loads to Class 1E power sources ready for RRRC		8/1/79

REVISION 0

August 16, 1977

CATEGORY A TECHNICAL ACTIVITY NO. A-26

Title: Reactor Vessel Pressure Transient Protection (Overpressure Protection)

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

Task Manager: Gary G. Zech, DOR

1. Problem Description

Since 1972, there have been over 30 reported incidents of pressure transients in pressurized water reactors which have exceeded the pressure temperature limits of the reactor vessels involved. These limits were those identified in the technical specifications for each facility and were based on the requirements of Appendix G to 10 CFR Part 50. The majority of these events occurred while in a water solid condition, during startup or shutdown operations, and at relatively low reactor vessel temperatures. Since the reactor vessel material has less toughness at these lower temperatures, it is much more susceptible to failure through brittle fracture at lower temperatures than at normal operating temperatures; and therefore, the margin of safety to vessel failure under low temperature conditions is reduced.

Reactor vessel pressure transients have been initiated by a variety of causes which can be grouped into the following categories: personnel error, procedural deficiencies, component random failure and spurious valve actuation. The resultant pressure transients are of basically two types: a mass input type from charging pumps, safety injection pumps or safety injection accumulators, or a thermal expansion type caused by the feedback of heat from the secondary side of steam generators. The magnitude of the pressure transients varied from minor violations of the Appendix G limits (500 to 1000 psig peak pressure) to pressure increases up to the safety valve setpoint (2450 psig).

APPROVED BY TASC, AUGUST 19, 1977

Although a new nuclear reactor pressure vessel could in all likelihood withstand pressures considerably greater than the safety valve setpoint, even at lower temperatures, increased neutron irradiation can cause the existing safety margins to significantly decrease due to a reduction in the toughness properties of the vessel. The immediate safety concern is therefore the older operating facilities.

In view of the frequency of these transients and the associated potential for pressure vessel damage, the staff has concluded that measures should be taken to minimize the number of occurrences of pressure transients in the future and to reduce the severity of such transients should they occur.

The problem addressed by this Task Action Plan is the identification of those actions that will assure that adequate overpressure protection is provided for both operating PWR facilities and those that have yet to receive their operating licenses.

2. Plan for Problem Resolution

Due to the frequency of occurrence of pressure transients since 1972, NRR conducted a review of the safety concerns and existing safety margins at operating reactor facilities. On November 1, 1976, a Technical Report on Reactor Vessel Pressure Transients was issued which summarized the various considerations relevant to this matter. It was concluded that adequate protection exists for the health and safety of the public by immediately reducing the likelihood of future pressure transients through improved administrative measures and by further reducing the likelihood of such events through design changes that will be implemented over the next year.

At Congressional hearings held in October 1976, the NRR Office Director committed to a schedule for implementation of any design objectives by the end of 1977.

The licensees of operating PWR reactors were requested to provide an analysis of the reactor coolant system response to pressure transients that can occur during startup and shutdown and to identify the design changes determined to be necessary to preclude exceeding the Appendix G limits for their plant. In November 1976, separate meetings were held with the licensees of each of the three PWR NSSS-designed plants to discuss their planned approach to resolve the presssure transient problem. At these meetings, specific criteria were identified that the licensees should apply in the design of equipment intended to prevent pressure transients that might exceed the limits of Appendix G to 10 CFR 50. These criteria were:

1. Credit of Operator Action - No credit can be taken for operator action until 10 minutes after the operator is aware that a pressure transient is in progress.

2. Single Failure Criteria - The pressure protection system should be designed to protect the vessel given a single failure in addition to a failure that initiated the pressure transient. In this area, redundant or diverse pressure protection systems would be considered as meeting the single failure criteria.

3. Testability - The equipment design should include some provision for testing on a schedule consistent with the frequency that the system is used for pressure protection.

4. Seismic Design and IEEE 279 Criteria - Ideally, the pressure protection system should meet both seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

Subsequent discussions with licensees and between NRR divisions has caused a reconsideration of certain aspects of the above criteria, particularly as they apply to the instrumentation, control and power areas of the proposed design changes.

Because of the large safety margins to vessel failure that exist in unirradiated reactor pressure vessels, it has been determined that new plants can continue to be licensed under existing safety criteria. However, administrative procedures and overpressure protection devices to reduce the likelihood of future pressure transients in a new plant are being required on a timely basis (prior to second cycle). The Reactor Systems Branch (DSS) is developing a Branch Technical Position on Reactor Coolant System Overpressure Protection. This Branch

Position will apply to all CP and OL applications, with certain qualifications, and will provide the guidance for continued DSS, DPM and DSE review of the adequacy of the design of the overpressure protection system. Comments have been received on the draft Branch Position which are being evaluated prior to incorporation. The major aspects of the proposed Branch Position are as follows:

1. A system shall be designed and installed which will prevent the exceeding of the applicable Technical Specifications and Appendix G limits for the reactor pressure vessel during plant cooldown or startup. The system shall be capable of relieving pressure during all potential overpressurization events at a rate sufficient to satisfy the Technical Specification limits, particularly while the Reactor Coolant System is in a water-solid condition.

2. The system must be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single failure. The cause for initiation of the event, i.e., operator error, component malfunction, etc., will not be considered as the single active failure. The analysis should assume the most limiting allowable operating conditions (e.g., one RHR train operating or available for letdown, other components in normal operation when the system is water solid such as pressurizer heaters and charging pumps). All potential overpressurization events must be considered when establishing the worst case event.

3. The system must operate automatically, providing a completely independent backup protective feature for the operator. The design must not include manual actions to enable or "turn on" the system or to mitigate the consequences of a potential overpressure event.

4. To assure operational readiness, the overpressure protection system must be tested in the following manner:

    a. A test must be performed to assure operability of the system electronics prior to each shutdown.

    b. A test for valve operability must be conducted as specified in the ASME Code Section XI.

   c. Subsequent to system, valve, or electronics maintenance, a test on that portion(s) of the system must be performed prior to declaring the system operational.

5. The system must meet the design requirements of IEEE-279. The design must be of at least the same quality as those system(s) to which it is connected, such that no portion of the plant design is compromised. The requirements of Regulatory Guide 1.26 must be satisfied.

6. The protection system does not have to meet Seismic Category I requirements if it can be shown that an earthquake would not initiate an overpressure transient. The postulated earthquake should be of magnitude equivalent to the SSE. If the earthquake can initiate an overpressure transient, then it should be assumed that loss of offsite power is an expected consequence of the event and the protection system should be designed to Seismic Category I requirements and not require the availability of offsite power to perform its function. Should the applicant show that a postulated earthquake could not cause an overpressure event, the overpressure protection system design must not compromise the design criteria of any other safety-grade system with which it would interface. The requirements of Regulatory Guide 1.29 must be satisfied.

7. The loss of offsite power shall be considered as an anticipated transient which could occur while in a shutdown condition. If this event can initiate an overpressure transient, the overpressure protection system must be independent of offsite power, in addition to performing its function assuming any single active failure.

8. Plant designs which take credit for an active component(s) to mitigate the consequences of an overpressurization event must include additional analysis considering inadvertent initiation or provide justification to show that existing analyses bound such an event.

The proposed implementation of the Branch Position would be that it should apply to all CP and OL applications, with the exception of the requirement for the system to meet IEEE-279. OL applicants would be allowed to justify reasonable deviations from the requirements of IEEE-279. For those applicants expected to receive an operating expected to receive an operating license this year, installation of all equipment would occur no later than the first refueling outage. For any plant receiving an operating license in 1978 or later, installation of equipment should be made prior to plant startup.

The basic suggested differences in the criteria that would be applied by DOR to design changes in operating reactors and by DSS, DSE and DPM to applications for a CP or OL are as follows:

1. <u>System Alignment for Operation (Enabling)</u>

   DOR: Operator action to align the system for operation is sufficient when accompanied by alarms and procedural verification.

   DSS: Fully automatic operation.

2. <u>Administrative Controls</u>

   DOR: Administrative controls may be used to eliminate from consideration transients from certain specific sources. Technical Specification controls will be allowed on accumulators, maximum T between steam generators and the Reactor Coolant System, and one of two trains of high pressure safety injection.

   DSS: Administrative controls are not specifically identified as acceptable means for protection.

3. <u>Seismic Design</u>

   DOR: The system should meet seismic category I requirements to the extent that an event which causes a pressure transient does not also cause a failure of equipment needed to terminate the transient. The staff, however, will evaluate a licensee's rationale for not fully meeting the seismic Category I criteria.

   DSS: The system does not have to meet Seismic Category I requirements if it can be shown that an earthquake would not initiate an overpressure transient.

4. Electrical Criteria

DOR: IEEE-279 equipment required at an interface with existing safety systems. The balance of the system must be of good quality, have redundancy in actuation channels and function with a loss of offsite power.

DSS: The system must meet the design requirements of IEEE-279.

The task items that require accomplishment for completion of the generic solution are:

1. A finalization of the criteria to be applied in the review of design changes to operating PWR reactors. This will include the resolution of, or justification for, the differences that exist between the DOR criteria and those contained in the Branch Position, as discussed above.
2. The submittal of the Branch Position for approval.
3. Approval of the Branch Position by the Regulatory Requirements Review Committee.

3. NRR Technical Organizations Involved

A. Reactor Safety Branch, Division of Operating Reactors

Has overall lead responsibility for the finalization of the design criteria to be applied to overpressure protection systems in operating reactors .

Manpower Estimates: .08 man years FY 1977

B. Plant Systems Branch, Division of Operating Reactors

Has lead responsibility for the criteria and design requirements to which the instrumentation, control and power aspects of the proposed overpressure protection system in operating reactors must conform.

Manpower Estimates: .04 man years FY 1977

C. Reactor Systems Branch, Division of Systems Safety

Has lead responsibility for DSS in justifying (or resolving) the differences that exist between the criteria in the Branch Position and those that have been used by DOR. Has lead responsibility for development of the Branch Technical Position identifying the review criteria for overpressure protection systems by applicants for CP's and OL's, and for initiating subsequent changes to Standard Review Plans.

Manpower Estimates: .04 man years FY 1977

D. Task Manager

Has overall responsibility for the coordination between NRR Branches in the accomplishment of Task Items to complete the generic solution as identified in this Task Action Plan.

Manpower Estimates: .02 man years FY 1977
.02 man years FY 1978

4. Technical Assistance Requirements

Significant work is in progress under an interagency agreement between the NRC and NRL to evaluate the radiation effects, analytical techniques and advanced testing methods for the analysis of radiation damage of materials in operating reactor vessels. Although not required for this Task Action Plan, information from this program could ultimately affect the acceptance criteria applied to the design of future systems to provide overpressure protection. Engineering Branch of DOR has management responsibility for this program which is described in Category A Technical Activity No. A-11.

5. Interactions with Outside Organizations

A. Westinghouse Owners' Group

Most of the licensee's with Westinghouse-designed operating PWR facilities have formed an ad hoc owners' group to evaluate the problems of reactor vessel overpressurization. These licensees have engaged Westinghouse to perform a transient analysis to include consideration of both mass input and heat input induced overpressurizations. The range of system and component physical parameters, performance characteristics and operating limits applicable to Westinghouse designed plants are to be used to bound the analysis. The final results of this analysis were submitted in late July 1977.

B. Combustion Engineering Owners' Group

Five of the six operating Combustion Engineering designed PWR facilities have also formed an owners' group to evaluate the generic aspects of the overpressurization problem. Combustion Engineering has performed an analysis similar to that conducted by Westinghouse and has been submitted by the licensees for the staff's review.

6. Assistance Requirements from Other NRC Offices

A. Office of Nuclear Regulatory Research, Division of Reactor Safety Research

RES, at their own initiative, has developed the OCTAVIA computer code capability of determining the reactor vessel failure probability of operating PWRs relative to the pressure transient events that have occurred. The Engineering Branch (DOR) has used OCTAVIA to compile a listing of these probabilities based on information currently available. This listing has been used by the staff in ordering its review schedules. Ongoing efforts in this area will provide additional analyses with RES continuing to perform in an advisory capacity.

7. Schedule for Problem Resolution

The major milestones are as follows:

	Item	Date
A.	DOR and DSS resolve or justify differences in the criteria that are to be applied by DSS (as defined in the proposed RSB Branch Technical Position) and by DOR (as defined in meetings with licensees on November 3, 4 and 5, 1976). NOTE: A determination will be made as to whether different criteria are to be applied to the overpressure protection systems from operating plants and those undergoing an OL review.	August 31, 1977
B.	Reactor Systems Branch (DSS) submit Branch Technical Position for approval.	September 23, 1977

Item	Date
C. Receive final approval of BTP from the Regulatory Requirements Review Committee.	October 28, 1977

8. Potential Problems

A delay in the finalization of the criteria to be applied to design changes in operating reactors would delay the review of the proposed system modifications from PWR licensees.

It should be noted that the Branch Technical Position uses Appendix G to 10 CFR 50 as the limit for all postulated initiating events, regardless of the probability of that event or combination of events occurring. It is recognized that the probability of a safe shutdown earthquake and a resultant overpressurization event is less probable than an anticipated operational occurrence and therefore the application of the upset criteria of Appendix G to these less probable events may represent an excessive degree of conservatism. The same is true for a loss of offsite power if it is shown to result in the initiation of an overpressure transient. A less conservative pressure vessel bittle fracture limit may prove to be appropriate for such events. There is no effort presently planned to develop other limits and therfore no attempt will be made in the near future to define any additional criteria less consevative than Appendix G. The Branch Technical Position and the criteria to be utilized by DOR for Operating Reactors require that Appendix G be applied to all possible overpressure events including those associated with the safe shutdown earthquake or loss of offsite power. Some licensees have expresed viewpoints which question the validity of the Appendix G limits, similar to the discussion above, and may challenge the criteria we have identified.

SEP 15 1977

REVISION 0

CATEGORY A TECHNICAL ACTIVITY TASK NO. A-27

TITLE: Reload Applications

LEAD RESPONSIBILITY: Division of Operating Reactors

LEAD ASSISTANT DIRECTOR: D. Eisenhut, Assistant Director for Operational Technology, DOR

TASK MANAGER: David Jaffe, DOR

1. Problem Description

By letter dated June 18, 1975, licensees of operating reactor facilities were sent a preliminary copy of a staff paper, "Guidance for Proposed License Amendments Relating to Refueling", and "Refueling Information Request Form". The purpose was to provide guidance, although preliminary, to licensees as to information the staff considers to be essestial for the conduct of its review of core reload submittals. In order to add more predictability to the review process and to improve the staff scheduling of such reviews, licensees were asked to submit the Refueling Information Request data within 30 days after receipt of the letter and were requested to update the information annually thereafter (or more often if appropriate).

The purpose of Task A-27 is to (1) update the preliminary guidance issued to licensees in the June 18, 1975 letter to assure conformance with the latest staff technical positions that relate to core reloads, and (2) prepare formal review procedures to assure prompt and uniform review of the licensee reload submittals. Revision of procedures for review of reloads is an important task in order to assure that projected staffing levels will be sufficient to accommodate future reload reviews. Under the present system of individualized reload reviews, the staff level for reload reviews alone would have to grow proportional to the number of facilities being licensed.

With regard to updating our guidance to licensees, providing licensees with uniform and up-to-date information on our criteria will help to make the review process more orderly and predictable. Ultimately, standardizing the review process will encourage licensees to plan reloads which do not require prior NRC approval and thus will serve to reduce our staffing commitment to reload reviews. Once uniform criteria in the form of the BTP have been developed for use with operating reactors then a reexamination of the OL stage of licensing will be made to determine if any incentives to licensees exist which would encourage evaluation of reloads prior to receipt of the OL. This would have the effect of allowing the licensee to perform reloads according to certain specifications without NRC approval beyond granting of the OL. In addition, the revised guidance will further underscore our interest in early identification of non-reload related activities which often take place during refueling outages and which require Commission review.

APPROVED BY TASC, SEPTEMBER 6, 1977
TASC COMMENTS INCORPORATED,
SEPTEMBER 15, 1977

2. Plan for Problems Resolution

Updated guidance on reload applications will be made available to licensees in two phases. The first phase involves preparation of a DOR Branch Technical Position (BTP) to revise the preliminary guidance issued to licensees. The revisisions that have been made will (a) ensure completeness of the safety analyses performed by the licensee, (b) ensure completeness of the application for license amendment relating to core refueling, (c) provide sufficient time for NRC review of proposed license amendments and/or unreviewed safety questions, and (d) improve efficiency and scheduling of NRC review of proposed license amendments. The BTP, DOR-1, was presented to the Regulatory Requirements Review Committee (RRRC) which indicated that additional development of the BTP was required.

Completion and issuance of the BTP to licensees will provide updated interim guidance to licensees until completion of the Regulatory Guide.

The second phase of the task, already underway in the Office of Standards Development, involves preparation of a Regulatory Guide on reload reviews which will incorporate the BTP. Following approval by the RRRC, BTP-DOR-1 will be sent to the Office of Standards Development as our response to their Request for Technical Assistance dated April 9, 1976. The April 9, 1976 request transmitted a copy of their draft Regulatory Guide on refueling for our comments. In addition, the BTP will be sent to licensees as discussed above. NRR will continue to support this effort as required. It is estimated that the Regulatory Guide will be issued in final form one year following approval of BTP-DOR-1 by the RRRC.

Documentation of NRC internal procedures related to the review of core reloads will be provided by an addition to the Standard Review Plan (SRP). This will be undertaken as a parallel effort with the development of the Regulatory Guide. The Division of Operating Reactors will be the lead organization for this effort. It is estimated that a SRP for core reload reviews will be issued one year following approval of BTP-DOR-1.

DSS will conduct a feasibility study in order to investigate the possibility of considering reloads prior to OL issuance. Based upon the results of this study, additional actions, if necessary, will be scheduled. A target date of March 1978 is set for completion of the DSS, pre-OL, reload feasibility study.

3. NRR Technical Organizations Involved

The Reactor Safety Branch, Division of Operating Reactors will be responsible for providing the technical input necessary (1) to obtain RRRC approval of BTP-DOR-1, (2) to support the preparation of the Regulatory Guide by the Office of Standards Development, and (3) to undertake preparation of a SRP for core reload reviews.

The Core Performance Branch and the Reactor Systems Branch, Division of Systems Safety, will participate in the above areas by providing review and comments as requested. In addition, the Core Performance Branch will conduct a feasibility study to investigate the possibility of considering reloads as part of the OL review.

4. Manpower Estimates

We estimate that .3 manyears of DOR technical effort is required for the remainder of FY 1977, 1.2 manyears for FY 1978, for developing this guidance. We estimate that .2 manyears of DSS technical effort in the Core Performance Branch will be required to complete Milestone "d" below.

5. Technical Assistance Requirements:

None

6. Interaction with Outside Organizations:

None

7. Assistance Requirements from Other NRC Offices:

None

8. Schedule for Problems Resolution:

The following are the major milestones for this task:

a. Initial considerations of BTP-DOR-1 by RRRC - June 1977 (complete)

b. Reconsideration of modified BTP-DOR-1, September 1977 T

c. Issuance of BTP-DOR-1 to licensees, September 1977 T

d. DSS completes Pre-OL Reload Feasibility Study, March 1978 T

e. Presentation of Reload Regulatory Guide to ACRS, June 1978 T

f. Issuance of Reload Regulatory Guide, July 1978 T

g. Issuance of SRP for Reload Applications for Internal Reivew, June 1978 T

h. SRP for Reload Applications to RRRC, July 1978 T

i. Issuance of SRP for Reload Applications, August 1978 T

9. Potential Problems

No potential problems can be foreseen at this time.

October 28, 1977
REVISION 0

CATEGORY A TECHNICAL ACTIVITY NO. A-28

APPROVED BY TASC,
OCTOBER 19, 1977

TITLE: Increase in Spent Fuel Pool Storage Capacity

LEAD RESPONSIBILITY: Division of Operating Reactors

LEAD ASSISTANT DIRECTOR: Darrell G. Eisenhut, Assistant Director for Operational Technology, DOR

TASK MANAGER: H. Elliot Chakoff, DPM

1. Problem Description

With the present "no-reprocessing" posture throughout the nuclear power industry, a considerable increase in onsite spent fuel storage will be required in order to permit continued operation of many nuclear power plants. On September 16, 1975, the Commission announced (40 FR 42801) its intent to prepare a generic environmental impact statement on handling and storage of light water power reactor spent fuel (GEIS) and also identified five factors to be applied in licensing decisions in the interim. The Commission concluded that licensing activities should proceed on a "case-by-case" basis and, pending the outcome of the GEIS, rulemaking proceedings on more definitive standards and criteria would be instituted on or about the time of issuance of the draft GEIS by the NRC staff.

To achieve the Commission's stated objective, we must review the GEIS to determine what, if any, change to the rules and procedures under which NRR licenses power reactors are required. While such an effort is underway, NRR must be prepared to continue to review and conclude on the many spent fuel pool expansion requests expected in the near future. Consistent with and in support of these objectives, we must develop uniform guidelines and procedures for licensees and the staff alike.

2. Plan for Problem Resolution

To achieve the aforementioned objectives two concurrent task plans have been developed. Task I, "Development of a Standard Review Plan and Regulatory Guide for Review of Increase in Spent Fuel Pool Storage Capacity" provides for the development of an SRP and recommendations for a R.G. based upon an OT branch technical position (currently in draft form). This draft BTP is a compendium of insights gained by DOR in their "case-by-case" reviews of spent fuel pool capacity increases. The development of these documents would serve to ensure the uniformity of reviews for the sixty-five operating plants and for plants under licensing alike, as well as to inform applicants and licensees in advance of preparation of applications, as to the staff's informational needs. For additional details see Attachment 1. The outcome of this activity may require a reevaluation of existing SRPs to determine if they require changes as a result of this effort.

TASC COMMENTS INCORPORATED, OCTOBER 23, 1977

Task II, "Policy Development" provides a mechanism for the development of information to determine the necessity for changes to existing NRR licensing criteria based on the conclusions of the GEIS. The results of Task II will be the development of a factual basis for determining the need for rulemaking and the development of proposed rules if they are found to be required. For additional details see Attachment 2.

A Critical Path Milestone-Summary is presented as Attachment 3. This summary identifies the relationship of each task and key milestones in the individual tasks to the overall effort.

3. NRR Technical Organizations Involved

   a. Lead responsibility for Task I, "Development of a Standard Review Plan and Regulatory Guide for Review of Increase in Spent Fuel Pool Storage Capacity" will reside with the Task Manager and the Plant Systems Branch of DOR. A significant support effort will be required by the EB and EEB of DOR. Additional assistance will be required from DSS/DSE/DPM. The total estimated DOR/Task Manager effort is estimated to be 41 man-weeks, 28 man-weeks for DSS/DSE and 3 man-weeks for DPM. Detailed work assignments and manpower estimates are provided in Attachment 4.

   b. Lead responsibility for Task II, "Policy Development," will reside with the Task Manager with assistance from DPM, DOR, and OELD. The estimated effort for the selected DPM and DOR individuals will be 9 man-weeks each. No attempt has been made to estimate the OELD manpower effort. Detailed work assignments and manpower estimates are provided in Attachment 4.

4. Technical Assistance Requirements

   a. For Task I, "Development of a Standard Review Plan and Regulatory Guide for Review of Increase in Spent Fuel Pool Storage Capacity" no assistance is anticipated.

   b. For Task II, "Policy Development," Brookhaven National Lab may be requested to perform a technical study of licensing alternatives identified by the staff. The need for this assistance will be determined pending the NRR staff review of the draft GEIS. Potentially this effort could involve development of alternative criteria to determine the acceptable degree of spent fuel pool expansion at each site. The cost of the contractor effort is estimated to be approximately $10,000.

5. Interactions with Outside Organizations

   a. For Task I, "Development of a Standard Review Plan and Regulatory Guide for Review of Increase in Spent Fuel Storage Capacity," no interaction with outside organizations is anticipated.

   b. For Task II, "Policy Development," no interaction is anticipated with outside organizations.

6. Assistance Requirements from Other NRC Offices

   The Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle and Material Safety, Fuel Reprocessing and Recycle Branch has requested that they continue to be informed of progress and developments by ONRR in the review of spent fuel pool storage capacity. NRR will review and comment on the draft GEIS which has been prepared by NMSS. In addition, we anticipate that, for Task II, "Policy Development" OELD participation will be required. OSD will be requested to develop rules, stemming from the findings of the GEIS, for NRR if they are found to be required.

7. Schedule for Problem Resolution

   A detailed Critical Path Milestone Summary is presented for each task in their respective attachments. A Critical Path Milestone - Summary is presented as Attachment 3. This summary identifies the relationship of each task and the key milestones in the individual tasks to the overall effort.

8. Potential Problems

   a. For Task I, "Development of a Standard Review Plan and Regulatory Guide for Review of Increase in Spent Fuel Pool Storage Capacity" no problems are anticipated.

   b. For Task II, "Policy Development," the proposed schedule is very tight (20-weeks) which introduces the uncertainties of meeting it. OELD has indicated that a drawn out schedule would be unacceptable and that we should be in a position to promulgate proposed rules, if any, within sixty days of publication of the draft GEIS.

      The task may also be impacted by evolving administration plans for the Department of Energy regarding spent fuel storage.

Attachments:
1. Task I - Detailed Summary
2. Task II - Detailed Summary
3. Critical Path Milestones - Summary
4. Manpower Estimates

ATTACHMENT 1

TASK I - DETAILED SUMMARY

TASK I

DEVELOPMENT OF A STANDARD REVIEW PLAN AND REGULATORY GUIDE FOR REVIEW OF INCREASE IN SPENT FUEL POOL STORAGE CAPACITY

BASIS FOR TASK

An increased demand for onsite storage of spent fuel assemblies has prompted licensees to propose the use of high density storage racks in existing spent fuel storage pools. Similar proposals can reasonably be anticipated to be submitted by applicants for operating licenses. The review of these proposals and of the acceptance criteria which have been applied to these proposals indicates that to provide sufficient consistency of staff reviews, a standard review plan and standard format should be developed. Experience to date indicate that there is sufficient similarity among the proposals that much of the review can be "standardized."

TASK PLAN

DOR/OT, Engineering Branch, has developed a draft BTP entitled, "Proposed Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" which describes the staff review, acceptance criteria, and development of findings for the Environmental Impact Appraisal and the Safety Evaluation Report. The insights gained by DOR in their "case-by-case" reviews are reflected in this document. Due to the immediate industry need for guidance as to the staff's information requirements for approval of spent fuel pool modifications the draft BTP should be finalized, reviewed by RRRC, and promulgated to the licensees prior to the development of a Standard Review Plan and transmittal of recommendations for Regulatory Guide to OSD. The development of a Standard Review Plan for the NRR staff and a Regulatory Guide with information requirements based on this BTP would serve to ensure the uniformity of reviews for operating plants and plants not yet licensed for operation alike, as well as to inform applicants and licensees, in advance of preparation of applications, as to the staff's information needs. The first publication of a Standard Review Plan and Regulatory Guide should be titled "Interim" to reflect that it will be reevaluated subsequent to completion of Task A-36, i.e., "Control of Heavy Loads Near Spent Fuel." New acceptance criteria or changes in staff review emphasis which result from this generic task, i.e., Task A-36, will be factored into a revised version of the Standard Review Plan and Regulatory Guide. Consequently, we can promulgate our findings from the generic effort, i.e., Task A-36 in a consistent and orderly fashion while still providing the best guidance available to the staff, licensees and applicants in the interim.

NRR Technical Organizations Involved

After finalization of the draft paper a value/impact statement will be prepared for RRRC by the EB and Task Manager. It is estimated that this effort will require four man-weeks with approximately one-man week each of consultation by EEB and PSB. Following the issuance of the finalized and approved BTP the lead responsibility for this task will be transferred from the EB of DOR to the Plant Systems Branch of DOR and the Task Manager.

The development of the first draft of the Standard Review Plan (SRP) is estimated to require about seven man-weeks total for EB, PSB, and EEB of DOR. Review and comment by DSS (MEB, Matl. EB, SEB, CPB, ASB), DPM, DSE (AB, ETSB), and DOR on the first draft should require about two man-weeks per branch. Incorporation of comments by PSB and the Task Manager should require two man-weeks with approximately one man-week of consultation each by EEB and EB. The review of the final draft by management and resolution of comments is estimated to be two man-weeks for PSB and the Task Manager with approximately one man-week each of consultation by EB and EEB. At this stage about one man-week per branch should be allocated by involved DSS/DSE branches for consultation.

The completion of Task A-36, "Control of Heavy Loads Near Spent Fuel" will be a decision point for this task, i.e., Task I. At this time we will determine what, if any, modifications are required to the SRP and to the proposed RG to reflect the results of Task A-36. In the event that further action is required, the PSB and the Task Manager would require approximately three man-weeks to draft changes to the interim SRP and recommendations for the RG to be sent to OSD with two man-weeks each for EEB and EB required for their comments. The involved DSS, DOR, DPM and DSE branches should require one man-week for each branch to review these changes. The incorporation of comments by PSB and the Task Manager and the preparation of a value/impact statement for RRRC should require four man-weeks with approximately one-man week each of consultation by EEB and EB. The review of the final draft by management and the resolution of comments is estimated to be two man-weeks for PSB and the Task Manager with approximately one man-week each of consultation by EEB and EB.

The total estimated DOR/Task Manager effort for this task is about 41 man-weeks, 28 man-weeks for DSS/DSE and three man-weeks for DPM. Detailed manpower estimates are provided in Attachment 4.

Technical Assistance Requirements

None anticipated

Interactions with Outside Organizations

None anticipated

Assistance Requirements from Other NRC Offices

The Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle and Material Safety, Fuel Reprocessing and Recycle Branch has requested that they continue to be informed of progress and developments by ONRR in the review of spent fuel pool storage capacity.

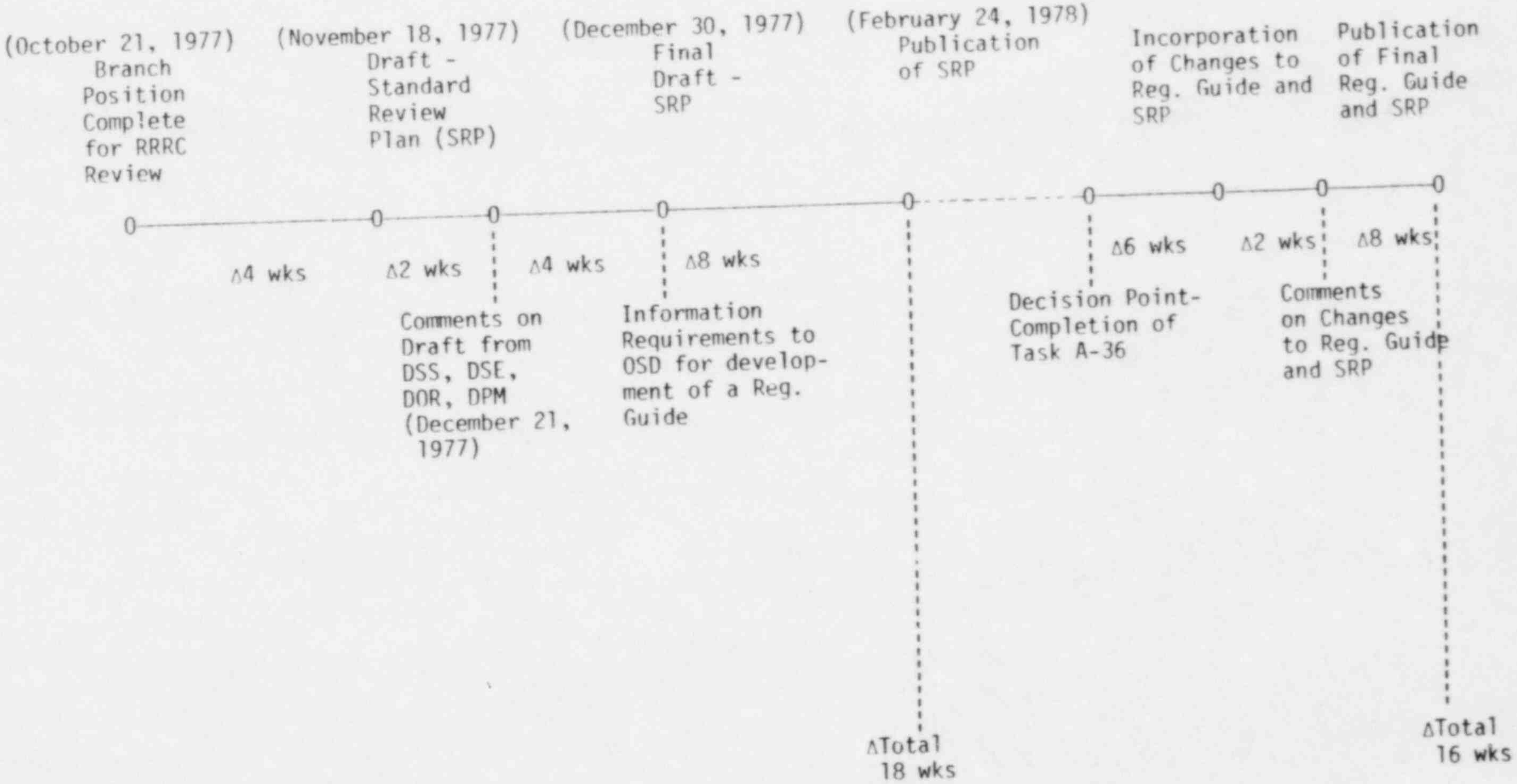
Schedule for Problem Resolution

We anticipate that eighteen weeks after RRRC approval of the Branch Technical Position, an interim Standard Review Plan would be available for use by the staff. We estimate that ten weeks after RRRC approval of the Branch Technical Position, we will have developed the information necessary to request OSD development of a Regulatory Guide. At the completion of the task, a reevaluation of existing SRPs may be necessary to determine if any changes are necessary to them. For additional details see the Critical Path Milestones in the attached Figure.

Potential Problems

None anticipated.

TASK I DEVELOPMENT OF STANDARD REVIEW PLAN AND REGULATORY GUIDE FOR REVIEW OF INCREASE IN SPENT FUEL POOL STORAGE CAPACITY



ATTACHMENT 2

TASK II - DETAILED SUMMARY

TASK II

POLICY DEVELOPMENT

BASIS FOR TASK

On September 16, 1975 the Commission announced (40 F.R. 42801) its intent to prepare a generic environmental impact statement (GEIS) on handling and storage of spent light water power reactor fuel. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement. The Commission provided "five factors" to be applied, weighed, and considered within the staff's statements or appraisals in reaching licensing determinations. Consequently, in the last two years the staff has proceeded on a case-by-case basis in the review of these licensing actions.

The Commission stated in part, that upon completion of the draft GEIS, rulemaking proceedings on more definitive standards and criteria would be instituted on or about the time of issuance of the draft GEIS. Possible amendments to 10 CFR 51.20(e) would also be considered at this time. This task is to develop the technical information and legal documentation for NRR to support this action with respect to expansion of spent fuel storage at reactor facilities. We also wish to assure that the rules applicable to reactor facilities are developed consistent with and in conjunction with rules for non-reactor storage facilities. Consequently, close coordination with NMSS in this regard will be required.

TASK PLAN

The thrust of this plan is to review the draft GEIS and determine what, if any, technical and legal changes to the rules and procedures under which NRR licenses power reactors are required to implement the findings of the draft GEIS. This will require a parallel legal and technical review effort. The results of this effort will be a Summary Staff Report documenting the NRR staff's review of the draft GEIS and a presentation of conclusions and recommendations for rulemaking proceedings, if necessary, for reactor facilities. OSD will be requested to develop proposed rules if they are necessary. The results of this task effort may provide input to Task I "Development of a Standard Review Plan and Regulatory Guide for Review of Increase in Spent Fuel Pool Storage Capacity", in the event that policy issues are identified which lead to technical requirements, either in terms of information required from licensees or staff technical criteria.

NRR TECHNICAL ORGANIZATIONS INVOLVED

The lead responsibility for this task should reside with the Task Manager. Upon receipt of the draft GEIS for NRR review in August 1977, a review of the draft GEIS should commence by the Task Manager and appropriate representatives from each NRR office. The objective of this initial effort would be to develop NRR comments on the GEIS for transmittal to NMSS, to develop a series of relevant questions for OELD to consider in their review, and to develop a work plan for a contractor if as a result of the NRR review this is found to be necessary. No attempt has yet been made to estimate NRR manpower required for the review of the draft GEIS. The contractor effort could involve the preparation of technical bases for various licensing alternatives identified by the NRR staff. Potentially this effort could involve development of alternative criteria to determine the acceptable degree of spent fuel pool expansion at each site. The work plan development effort could be full time for the Task Manager and require about 3 man-weeks each for DPM and DOR (EEB) support. The proposed contractor is Brookhaven National Lab. The estimated cost of the contractor effort is $10,000.

The contractor effort, if necessary, and the OELD support effort would begin upon publication of the draft GEIS for public comment. It is currently anticipated that a statement will be included in the draft GEIS that the NRR staff is considering the following:

1. Issuance of an interim Regulatory Guide and Standard Review Plan for Increases in Spent Fuel Storage Capacity at Nuclear Power Reactors.

2. The development of definitive criteria to implement the draft GEIS findings in the licensing process for increases in reactor spent fuel storage capacity.

Upon NRR receipt of the draft reports from OELD and the contractor, if necessary, the selected DPM and DOR staff members, the Task Manager and the Director of NRR would review and comment. This effort would require about 2 man-weeks each for DOR (EEB) and DPM and full time effort for the Task Manager.

It is anticipated that a coordinated effort between NMSS and NRR will assure that NRR implements, as necessary, the findings of the GEIS, and that NMSS appropriately reflects the experience gained to date by NRR in its review of spent fuel pool modifications at reactors. This will assure that the findings presented in the GEIS will reflect NRR experience and future plans.

Upon receipt of the final contractor report and final OELD recommendations, the Task Manager and the selected DPM and DOR staff members will prepare a draft Staff Report on definitive criteria to implement the draft GEIS findings in the licensing process. At this time OSD will be requested to develop rules for NRR if they are found to be necessary. This effort will be full time for the Task Manager and 2 man-weeks each for selected DPM and DOR (EEB) staff members. A management review of the draft will follow.

A final Staff Report will be written by the selected DPM and DOR staff members and the Task Manager and will include proposed rules developed by OSD if they are required. This effort will be full time for the Task Manager and require 2 man-weeks each for DPM and DOR (EEB). OELD will be requested to review and approve the final Staff Report and proposed rules if they are necessary.

The total anticipated manpower for this effort is 9 man-weeks each for DOR and DPM selected personnel with a full time effort by the Task Manager. No attempt has been made to estimate the OELD effort. Detailed manpower estimates are provided in Attachment 4.

TECHNICAL ASSISTANCE REQUIREMENTS

Brookhaven National Lab may be requested to provide assistance in the development of technical basis for licensing alternatives.

INTERACTION WITH OUTSIDE ORGANIZATIONS

None anticipated.

ASSISTANCE REQUIREMENTS FROM OTHER NRC OFFICES

The Office of Nuclear Material Safety and Safeguards, Division of Fuel Cycle and Material Safety, Fuel Reprocessing and Recycle Branch has requested that they be included in the review process for the generic NRR effort on Increase in Spent Fuel Pool Storage Capacity. It is anticipated that a coordinated effort between NMSS and NRR will assure that NRR implements, as necessary, the findings of the GEIS, and that NMSS appropriately reflects the experience gained to date by NRR in its review of spent fuel pool modifications at reactors. This will assure that the findings presented in the GEIS will appropriately reflect NRR experience and future plans.

OELD effort will be required as described in the section of this task entitled, "NRR Technical Organizations Involved."

OSD will be requested to assist in the development of rules for NRR if they are found to be required.

SCHEDULE FOR PROBLEM RESOLUTION

The estimated time from the beginning of the contractor and OELD review effort of the draft GEIS to the publication of proposed rules and a Summary Staff Report is about 20 weeks. For additional details see the attached CPM.

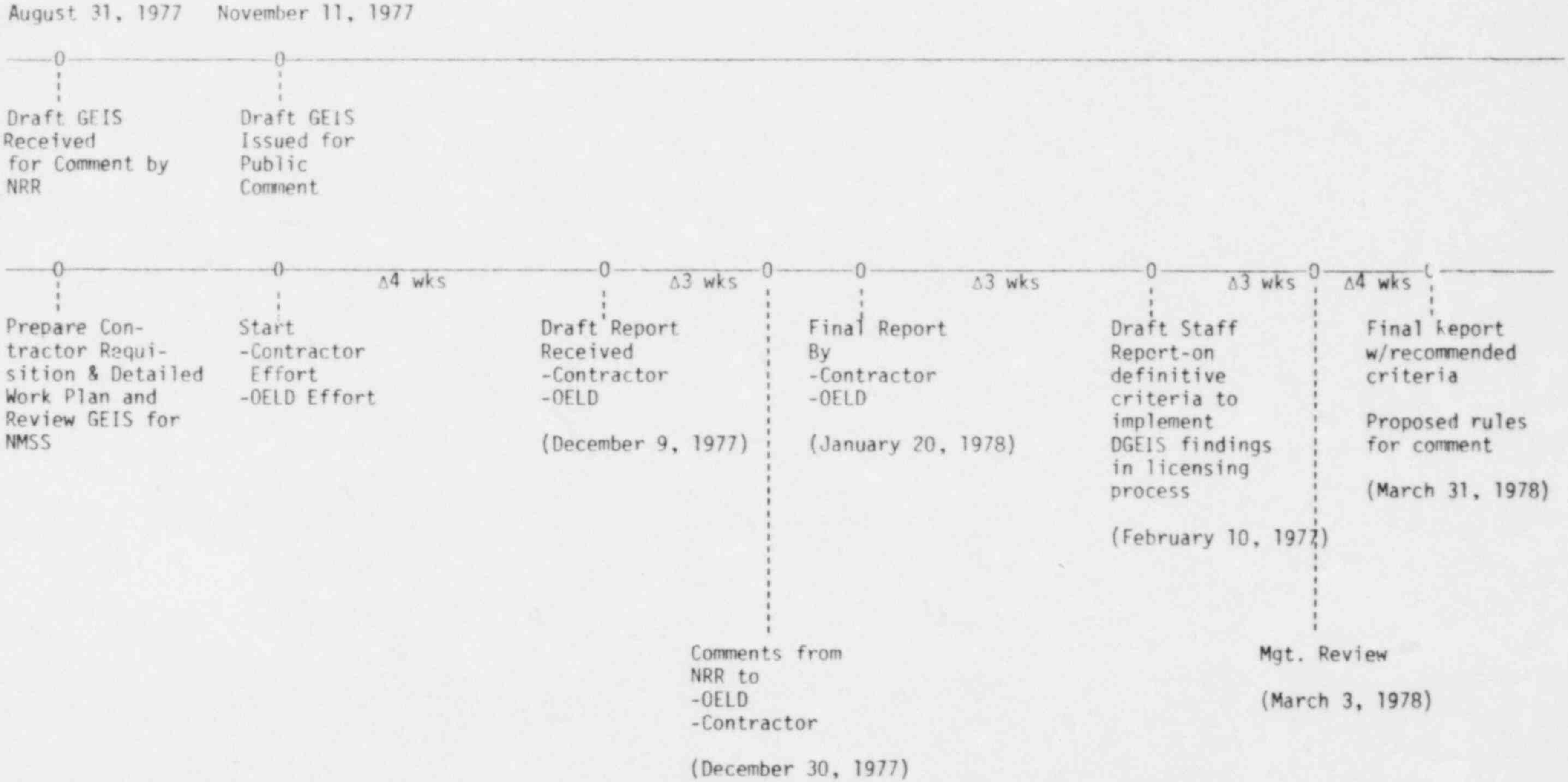
POTENTIAL PROBLEMS

The proposed schedule is very tight which introduces the uncertainties of meeting it. OELD indicated that a drawn out schedule is unacceptable, and we should be promulgating our proposed rules ideally within sixty days of publication of the draft GEIS.

The task may also be impacted by evolving administration plans for the Department of Energy regarding spent fuel storage.

TASK II

POLICY DEVELOPMENT



August 31, 1977

Draft GEIS
Received
for Comment by
NRR

November 11, 1977

Draft GEIS
Issued for
Public
Comment

Prepare Contractor Requisition & Detailed
Work Plan and
Review GEIS for
NMSS

Start
-Contractor
Effort
-OELD Effort

Δ4 wks

Draft Report
Received
-Contractor
-OELD

(December 9, 1977)

Δ3 wks

Comments from
NRR to
-OELD
-Contractor

(December 30, 1977)

Final Report
By
-Contractor
-OELD

(January 20, 1978)

Δ3 wks

Draft Staff
Report-on
definitive
criteria to
implement
DGEIS findings
in licensing
process

(February 10, 1977)

Δ3 wks

Mgt. Review

(March 3, 1978)

Δ4 wks

Final Report
w/recommended
criteria

Proposed rules
for comment

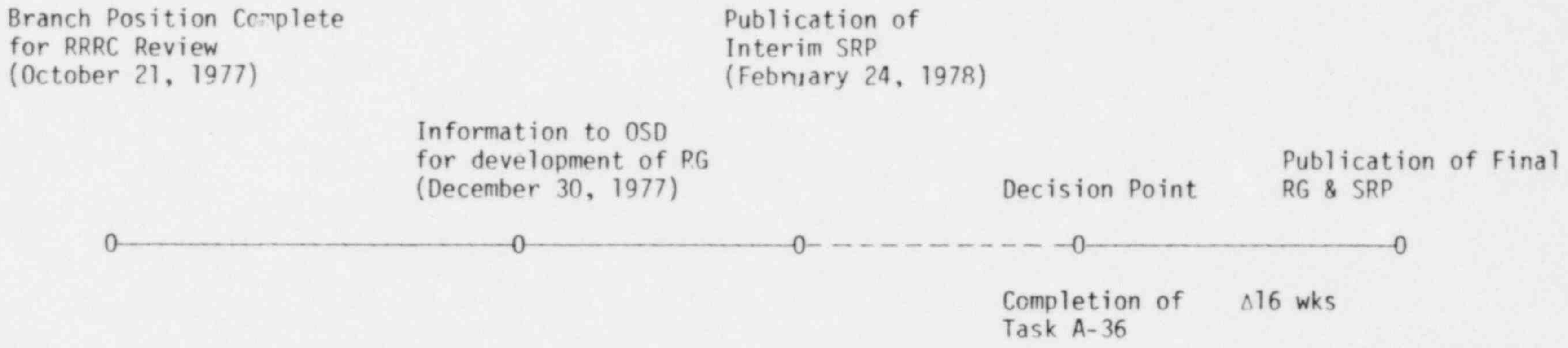
(March 31, 1978)

ATTACHMENT 3

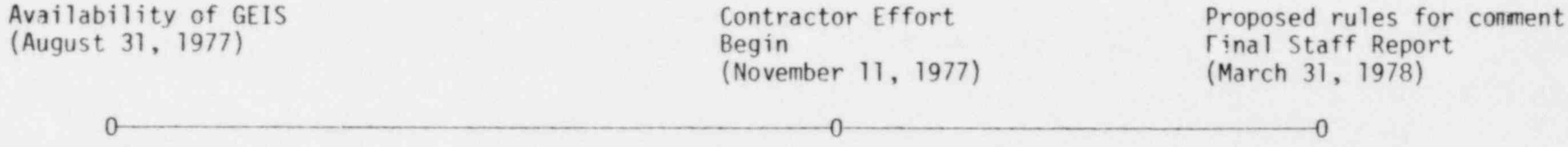
CRITICAL PATH MILESTONES - SUMMARY

CRITICAL PATH MILESTONE - SUMMARY

Task I - Development of Standard Review Plan and Regulatory Guide



Task II - Policy Development



ATTACHMENT 4

MANPOWER ESTIMATES FOR FY 1978
(MAN-WEEKS)

	DOR			DSS					DSE		DPM
	EB	PSB	EEB	ASB	SEB	MEB	Mat'l. EB	Core PB	AB	ETSB	
Task I	12	20	9	4	4	4	4	4	4	4	3
Task II (*)	-	-	9	-	-	-	-	-	-	-	9

(*) No attempt has been made to evaluate the NRR manpower required to review the draft GEIS - the scope and manpower of the task could vary depending upon the results of the NRR review of the draft GEIS.

AUG 30 1977

CATEGORY A TECHNICAL ACTIVITY NO. A-29

REVISION 0

Title: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Lead Responsibility: Division of Operating Reactors

Lead Assistant Director: James R. Miller, Assistant Director for
Reactor Safeguards

Task Manager: J. Mark Elliott

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 30,1977

1) Problem Description:

Extensive efforts and resources are expended in designing nuclear power plants to minimize the risk to the public health and safety from equipment or system malfunction or failure. However, reduction of the vulnerability of reactors to industrial sabotage is currently treated as a plant physical security function and not as a plant design requirement. Although present reactor designs do provide a great deal of inherent protection against industrial sabotage, extensive physical security measures are still required to provide an acceptable level of protection. An alternate approach would be to more fully consider reactor vulnerabilities to sabotage along with economy, operability, reliability, maintainability, and safety during the preliminary design phase. Since emphasis is being placed on standardizing plants, it is especially important to consider measures which could reduce the vulnerability of reactors to sabotage. Of course, any design features to enhance physical protection must be consistent with present and future system safety requirements.

2) Plan for Problem Resolution:

The NRR staff efforts concerning the resolution of sabotage protection design considerations will be to (1) continue lead-role participation in the on-going Inter-Office Working Group on Reactor Design Provisions for Protection Against Industrial Sabotage, (2) recommend a confirmatory research program to RES, (3) monitor and review the results of the RES studies, and (4) recommend changes in licensing design criteria to include consideration of sabotage protection at the CP review stage.

The Inter-Office Working Group, comprised of respresentatives from NRR, NMSS, RES, IE, and SD, will prepare a report which will represent an identification of candidate alternative design approaches to be used as the principal input to the research program. Previously completed reactor vulnerability studies will also serve as inputs to the research program.

The proposed research will provide the staff with (1) detailed feasibility and cost-effectiveness analyses of alternative design features for protection against sabotage, (2) impact analyses of those design features on safety, operability, reliability, and maintainability of the plant, (3) recommendations for any additional design changes which would reduce the vulnerability of reactors to industrial sabotage, and (4) evaluation methodologies for incorporating the consideration of those design features into the review of license applications. After a review by NRR safety and safeguards staff, the recommendations of the research studies could be incorporated into revisions to the "General Design Criteria" of Appendix A to Part 50.

The final product which implements NRC's resolution of this concern could be manifested in one of the following ways:

1) A change to the regulations of Appendix A to Part 50 could be developed which would invoke certain design criteria to reduce the vulnerability of a plant to industrial sabotage. This approach would provide a firm basis to ensure that designs would have a higher level of inherent protection against sabotage than many present designs have. On the other hand, stringent design criteria could limit an applicants flexibility to make other design improvements. The Standard Review Plan (SRP) would be modified to accommodate changes in the regulations.

2) A Division 1 Regulatory Guide could be published which provides the staff's position on how a proposed design should include proper consideration of sabotage protection needs. This method would leave some flexibility with an applicant and does ensure that the necessary consideration would be given. However, this method provides a weaker regulatory basis. Modifications to the SRP would also be required in this case.

3) At a minimum, the standard review plan alone could be modified to explain the methodology used to evaluate proposed designs to assure that adequate consideration has been given to sabotage protection. This approach would allow the licensee maximum flexibility to develop a cost-effective mix of design features and physical security measures in order to achieve adequate physical security. However, the regulatory basis for obtaining advances in design which enhance inherent security would be somewhat tenuous.

Once the staff has received and evaluated the results of the research studies, we will be in a better position to choose between the above three alternatives.

3) <u>NRR Technical Organizations Involved</u>:

a. Reactor Safeguards Development Branch, DOR:
Is chairing the inter-office working group. Has lead responsibility for monitoring the research effort being funded by RES. Has lead responsibility for coordinating review and evaluation of research results. Will propose preliminary changes to regulations or new Regulatory Guides, for drafting by the Office of Standards Development and will draft preliminary modifications to the SRP. Manpower requirements are .25 man-year in FY 78 and 1.0 man-years in FY 79.

b. Reactor Safeguards Licensing Branch, DOR:
Will be consulted to assure adequate feedback from §73.55 review program, 2 man-weeks in each FY 78 and 79.

c. Several systems oriented technical branches will be called upon to review the research results to determine the potential impact of the research recommendations on plant safety. These branches include:

Plant Systems Branch, DOR
Reactor Safety Branch, DOR
Reactor Systems Branch, DSS
Auxiliary Systems Branch, DSS
Containment Systems Branch, DSS
Instrumentation and Control Systems Branch, DSS
Power Systems Branch, DSS

In addition, each of these Branches will be called upon to review and comment on proposed changes to the regulations, Regulatory Guides and SRP.

Manpower effort required by each branch is estimated at 2 man-weeks.

4) Technical Assistance Requirements:

None.

5) Interactions With Outside Agencies:

ERDA is developing evaluation methodologies that should be useful in determining the effectiveness of alternate design schemes. Interaction with ERDA, primarily through RES may be desirable during the course of this task although the extent of such interactions is unknown at this time. Also, AE's and vendors may be involved in this task but such involvement would be effected through the RES effort.

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the Committee as the task progresses."

6) Assistance Requirements from Other NRC Offices:

a. Office of Nuclear Regulatory Research, Division of Safeguards, Fuel Cycle and Environmental Research, Systems Analysis Branch. Will fund and manage research contract for approximately $1 million for FY 78.

b. Office of Standards Development, Division of Engineering Standards, Reactor Systems Standards Branch and Division of Siting, Health and Safeguards Standards, Materials Protection Standards Branch. Should NRR decide that rulemaking is the proper choice for implementation, these branches would coordinate the development of changes to the regulations to include design requirements for enhanced protection from industrial sabotage. If new Regulatory Guides are proposed, these Branches would also be involved. SD has requested $120,000 for FY 79 in support of this effort should rulemaking or Regulatory Guide development be adopted as implementation schemes.

c. RES, IE, SD and NMSS participation in the Inter-Office Working Group.

7) Schedule for Problem Resolution:

09/77 - Report of Inter-Office Working Group
11/77 - Initiation of Research Effort
10/78 - Completion of Research Study
01/79 - Decision point to identify implementation schemes
04/79 - Publication of proposed amendments or draft Regulatory Guides (if appropriate), and preliminary modifications to SRP.
10/79 - Publication of effective amendments or Regulatory Guides (if appropriate), and final modifications to SRP.

8) Potential Problems:

Although implementation of the results of this activity are not considered to be part of this task, some question may develop regarding the desirability or necessity of "backfitting" design improvements into licensed plants or those already in construction.

No Technical Assistance requirements have been identified for this task. It is possible, however, that some funds may be required in FY 79 to assist in preparation of modifications to the SRP, although it is intended that close coordination with RES during the course of the research effort will preclude the need for Technical Assistance.

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 30, 1977

REVISION 0
AUG 3 0 1977

Task Action Plan
Task No. A-30

Title: Adequacy of Safety Related D.C. Power Supplies

Lead Responsibility: Division of Systems Safety/NRR

Lead Assistant Director: R.L. Tedesco (Plant Systems)

Task Manager: R. Fitzpatrick, PSB, DSS

1. Problem Description:

The minimum acceptable D.C. power system is comprised of two physically independent Divisions which supply D.C. power for control and actuation of redundant safety related systems. The staff has recently issued a report, NUREG-0305, "Technical Report on D.C. Power Supplies in Nuclear Power Plants" which provides a summary of the current staff position and its bases for use in our review of D.C. power systems.

Recently questions have been raised concerning the current staff position, including the application of the single failure criterion for assuring a reliable D.C. power supply. These concerns stem from (1) the dependence on D.C. power of the decay heat removal systems which are required for long term heat removal, (2) the fact that failure of one D.C. division of a two divisions redundant system would generally result in a reactor scram which then would require removal of decay heat and therefore depends upon the remaining division for D.C. power supply, and (3) questions raised regarding the frequency of reported single D.C. divisions failures including those resulting from human error. The staff believes that these considerations require reexamination of the staff's design requirements for D.C. power systems; however, preliminary studies utilizing the results and methods of WASH 1400, indicate that the failure of DC power supplies leading to a loss of heat removal capability is a small contribution to the core melt probability.

In addition, for some time (since May) the staff has publically stated that this task would be completed in less than a year.

2. Plan for Problem Resolution

A. Approach

Analyses will be performed to quantify the reliability of various D.C. power system designs, that meet the staff criteria, for assuring adequate decay heat removal capability. The relative reliability of the various system designs will be intercompared and the reliability of D.C. power will be compared with the reliability of other redundant vital systems. The analyses will be based on reliability data for various systems and components, using probability methods. These analyses will be performed by NRR staff with technical assistance provided, as required, by RES (Probabilistic Analysis Branch).

B. End Product

The end product of this program is a NUREG report which will provide complete documentation of the analyses performed, develop a staff position regarding the adequacy of the existing acceptance criteria for D.C. power systems, recommend any new or revised criteria, and proposed changes to Section 8.3 of the Standard Review Plan (SRP) deemed necessary for licensing review of nuclear power plants.

This task will be considered to be complete on completion of the branch technical positions and the forwarding of recommendations to the Office of Standards Development for new or revised regulatory guides, and on NRR completion of revisions to the SRP.

C. Tasks

To assure overall electric power systems reliability, this task action plan will be coordinated to the extent necessary with other electrical related task action plans (A-24, "Qualification of Class 1E Safety Related Equipment"; A-25, "Non-Safety Loads on Class 1E Power Sources"; A-35 "Adequacy of Offsite Power Systems".

1. Detailed Scoping of Issue - Clearly define and scope the concern including identification of all the factors that must be considered in the assessment of D.C. power systems reliability (Task task is essentially complete). Define the bases for acceptance of D.C. power systems reliability.

2. Establish Data Base - Development of a data base using existing information on nuclear operating experience (e.g., Licensee Event Reports and the NPRDS system). Supplement this data base as necessary from equivalent industrial sources, e.g., IEEE, Chemical Industry. To the extent that is absolutely necessary, requests for additional information from licensees and applicants will be considered.

3. Establish Time Available for Manual Actions - Perform analyses to assess the time available following reactor scram within which decay heat removal capability must be made available to successfully cool the core. These analyses will expand those already performed in support of the present staff position as defined in NUREG-0305.

4. Establish Capability of Manual Actions - Evaluate proposed and existing plant designs and procedures (on a generic basis) to determine the capability for operation of decay heat removal systems assuming total loss of power (A.C. and D.C.) with manual operator actions. Human factors reliability will be included.

5. Quantify Reliability of Present Minimum System - Provide a best estimate quantitative assessment of the reliability of D.C. power system designs (for assuring decay heat removal

capability) which meet present regulatory criteria. This assessment will supplement that which was already performed in support of the present staff position as defined in NUREG-0305 for such a minimum system.

6. Assessment of Detailed Design Features - Define the present detailed design features of D.C. Power Systems, assess their contribution to system reliability, and identify any new or revised features that would enhance reliability. (Considerations of the effects of non-safety related loads on the reliability of the D.C. power supply system will be coordinated with the Task Manager for Category A Technical Activity No. A-25, "Non Safety Loads on Class IE Power Sources.")

7. Quantify Reliability of Other Selected System Designs - Provide a best estimate quantitative assessment of the reliabilities of selected D.C. power system designs (for assuring decay heat removal capability) which provide features which exceed present regulatory criteria.

8. Development of Criteria - Develop staff position regarding whether or not NRC's present criteria are acceptable. If criteria are determined to require modification, this task will include the development of any modified criteria. This effort will include an impact/value determination. Following development of the staff position, regardless of its conclusion, it will be discussed by the RRRC.

9. Prepare Staff Report - Prepare a staff report in the form of a NUREG document which will provide complete documentation of the analyses performed and staff conclusion reached from Items 1 through 8.

10. Develop Branch Technical Position - if determined necessary develop a modified branch technical position and changes to the Standard Review Plan and develop in form for NRR transmittal to the Office of Standards Development for new or revised regulatory guides.

3. NRR Technical Organizations

   A. Power Systems Branch, Division of Systems Safety

      1. Task No. 1 - Explicitly define the safety concern with regard to reliability of the minimum D.C. power system. Define the present minimum requirements for D.C. power systems; Describe the D.C. power system just meeting minimum requirements; Identify all elements of the minimum D.C. power system design that contribute significantly to system reliability, and whether they are covered explicitly by present requirements; Define the bases for acceptance of D.C. power systems reliability.

         Manhour requirement - 160 hours

2. Task No. 2 - In conjunction with DOR, review currently available information related to the reliability of D.C. power systems and develop a data base for use in other subtasks.

   Manhour requirements - 160 hours

3. Task No. 4 - Provide support to Reactor Systems Branch in its evaluation of capability of manual actions in event of loss of D.C. power.

   Manhour requirements - 80 hours

4. Task No. 6 - Define the present detailed design features of D.C. power systems and provide support to the Probabilistic Analysis Branch, Office of Nuclear Regulatory Research in its evaluation of the reliability of those features.

   Manhour requirements - 240 hours.

5. Task No. 8 - Develop position regarding adequacy of present criteria and it necessary, develop new or augmented criteria.

   Manhour requirements - 240 hours.

6. Task No.. 9 - Prepare a staff report in the form of a NUREG which documents all work done and conclusions drawn from task numbers 1 through 8. Obtain concurrence from all contributing organizations and forward to R^3C for consideration.

   Manhour requirements - 880 hours

7. Task No. 10 - If determined necessary develop draft branch technical positions and prepare any necessary changes to the SRP for RRRC review. If necessary, draft NRR recommendations for forwarding to the Office of Standards Development for the long term incorporation ~~of~~ such actions into new or revised regulatory guides. Obtain all appropriate concurrences in the formulation of any new or modified BTPs and SRPs.

   Manhour requirements - 240 hours

8. Total manpower FY 78 - 1.12 manyears

B. Analysis Branch, Division of Systems Safety

1. Task No. 1 - provide assistance to Power Systems Branch for definition and scoping of the concern. This will include definition of the safe plant condition which must be achieved to terminate the postulated sequence of events.

   Manpower requirements - 40 hours

2. Task No. 3 - Perform analyses to assess the time available following reactor scram within which decay heat removal capability must be made available to successfully cool the core. This must be done for the various types of boiling and pressurized water reactor designs.

   Prepare instructions and data tablulations to be completed for standardized analyses to assess the time available following reactor scram within which decay heat removal capability must be made available to successfully cool the core. These instructions are to be provided to the Task Manager who will be responsible for selection of the representative types of boiling and pressurized water reactor designs to be evaluated for this and other tasks. The Task Manager will be responsible for acquiring the necessary plant data and calculations from the appropriate licensees or vendors in accordance with the prescribed instructions. AB will consult with those performing the calculations as required. Upon receipt of the plant data and calculations, AB will review the results and prepare a summary report.

   Manhour requirements - 200 hours

3. Total manpower FY 78 - 0.14 manyears.

C. Reactor Systems Branch, Division of Systems Safety

1. Task No. 1 - Provide assistance to Power Systems Branch for definition and scoping of the concern. This will include definition of the safe plant condition which must be achieved to terminate the postulated sequence of events including identification of those systems which must be operational.

   Manpower requirements - 40 hours.

2. Task No. 4 - Provide support to Power Systems Branch in its evaluation of plant designs to determine the capability to achieve and maintain hot standby assuming total loss of power (A.C. and D.C.). This would include (1) an evaluation of the natural circulation capability and the behavior of boron in maintaining an adequate shutdown margin, and (2) an assessment of the adequacy of the pressure relief system for discharging the quantities of steam and water calculated in Task 3.

As a minimum evaluate two plants for each NSSS Vendor. In the event this provides insufficient information to be used in the analyses, provide additional evaluations as necessary.

Manhour requirements - 160 hours

3. Total manpower FY 78 - 0.11 manyears

D. Auxiliary Systems Branch, Division of Systems Safety

1. Task No. 4 - Provide support to Reactor Systems Branch in its evaluation of capability for manual action. This will include evaluation of various plant systems to determine whether fluid systems could remain open with loss of power that might lead to an excessive loss of primary coolant.

   Manhour requirements - 340 hours

2. Total manpower FY 78 - 0.19 manyears

E. Plant Systems Branch, Division of Operating Reactors

1. Tasks No. 1, 5, 6, 7 - Provide general input to the Power Systems Branch to ensure operating reactor considerations are appropriately reflected in task being evaluated.

   Manhour requirements - 120 hours

2. Task No. 2 - If necessary, provide support to Power Systems Branch for the collection of data to support overall Task.

   Manpower requirements - 120 hours

3. Task No. 8 - Provide support as requested by Power Systems Branch for development of licensing criteria.

   Manhour requirements - 240 hours

4. Task No. 10 - If it is necessary, provide assistance in the development of any revised NRR Branch technical position.

   Manhour requirements - 120 hours

4. Technical Assistance Requirements

Technical assistance requirements are not anticipated at this time.

5. Interactions with Outside Organizations

Outside organizations may be contacted as necessary to complete the data base.

This task is closely related to one of the generic items identified by the ACRS and, accordingly will be coordinated with the committee or the task progresses.

6. Assistance Requirements From Other NRC Offices

   A. Probabilistic Analysis Branch, Office of Nuclear Regulatory Research

      1. Task No. 5 - Provide a best estimate quantitative assessment of the reliability of a typical D.C. power system design (for assuring decay heat removal capability) which just meets present regulatory criteria. The typical design to be assessed will be provided by the Task Manager from the PSB/DSS output resulting from Task No. 1.

         Manhour requirements - 200 hours

      2. Task No. 6 - Assess the contribution to system reliability of the detailed design features of the D.C. power system which provide reliability in excess of that provided by the system defined in Task No. 5. The design features to be assessed will be identified by the Task Manager. Identify any new design features suggested by your analyses that could enhance reliability.

         Manhour requirements - 160 hours.

      3. Task No. 7 - Provide a best estimate quantitiative assessment of the reliabilities of selected D.C. power system designs (for assuring decay heat removal capability) which provide features exceeding present

regulatory criteria. Decisions on what design will be analyzed will be influenced by the results of Task No. 6. It is expected that at least five selected D.C. power system designs will be evaluated.

Manhour requirements - 240 hours.

4. Total manpower FY 78 - 0.34 manyears

B. Division of Reactor Operations Inspection, Office of Inspection and Enforcement

1. Task No. 2 - Provide assistance as requested by the task manager to verify reliability data and to verify that decay heat removal capability at selected Operating Plants is as documented by actual operating experience.

Manhour requirements - 180 hours

2. Manpower FY 78 - 0.1 manyear

7. <u>Schedule for Problem Resolution</u>

A. Summary of Schedule

1. Detailed Scoping of Issue 10/01/77
2. Establish data base 11/01/77
3. Establish time available for manual action . . . 11/30/77
4. Establish capability for manual actions 01/15/78
5. Quantify Reliability of Present Minimum System . 02/15/78

6. Assessment of detailed design features 02/28/78
7. Quantify Reliability of Other Selected System Designs. 03/15/78
8. Development of criteria 04/30/78
9. Prepare Staff Report 05/15/78
10. Develop Branch Technical Position 05/15/78

B. Detailed Schedule

Bar Chart Table 1 Attached. A summary of manpower estimates is presented in Table 2.

C. Technical Assignment Control Number - TAC 4587 (R-56)

8. Potential Problems

A. The availability of failure data. The analysis will use reliability data derived from operating plant experience, to the extent practicable. The availability of such data and data from other sources in both the quantity and detail required to establish the validity of the data base may be a problem .

B. Schedule completion. Completion of this task was scheduled for May 15, 1978 in accordance with the commitment made to the Commission during the briefing on June 27, 1977, and which is also contained in NUREG-0305 (Pg 7). This schedule does not reflect the availability of manpower in the participating branches which will be required for implementation. Therefore, in all likelihood, this schedule cannot be met unless additional trained manpower is made available.

Table 1
DETAILED SCHEDULE

1.	Detail Scoping of Issue	10/1/77
2.	Establish data base	11/1/77
3.	Establish time available for manual action	11/30/77
4.	Establish capability for manual actions	1/15/78
5.	Quantify Reliability of Present Minimum System	2/15/78
6.	Assessment of detailed design features	2/28/78
7.	Quantify Reliability of Other Selected System Designs	3/15/78
8.	Development of criteria	4/30/78
9.	Prepare Staff Report	5/15/78
10.	Develop Branch Technical Position	5/15/78

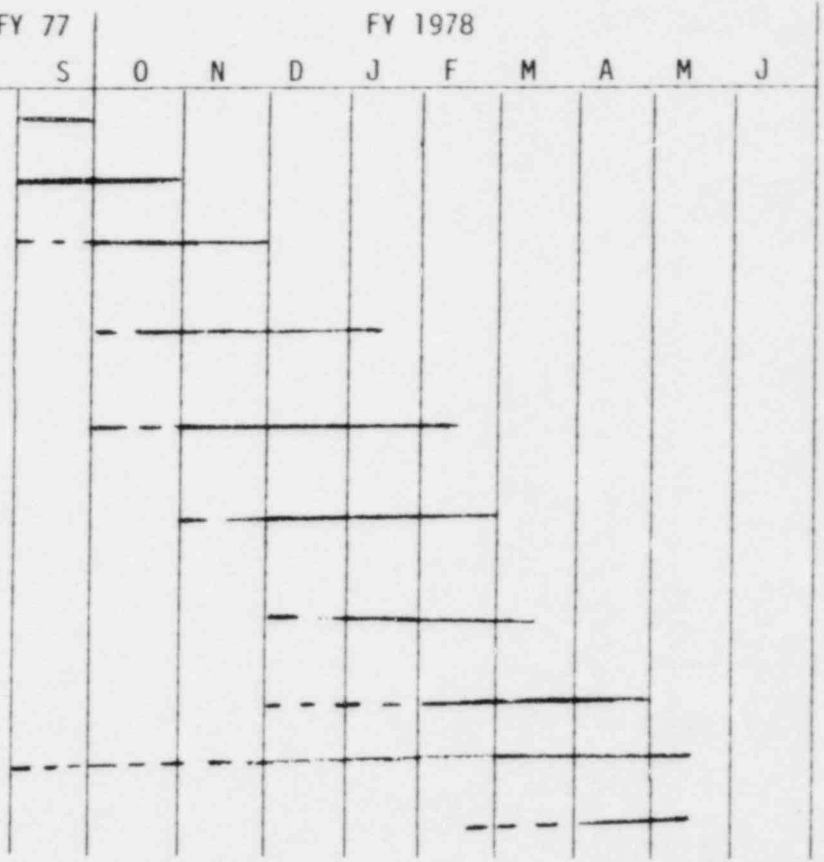


TABLE 2

MANPOWER ESTIMATES
TASK A-30

TASK				BRANCH/DIVISION				Div. (ROI/I
	PSB/DSS	AB/DSS	RSB/DSS	ASB/DSS	PSB/DOR	OR/DOR	PAB/RES	
1. Detail Scoping of Issue	.09	.03	.02		.07*			0.10
2. Establish data base	.09				.07			
3. Establish time available for manual action		.11						
4. Establish capability for present minimum system	.05		.09	.19				
5. Quantify reliability of present minimum system					*		.11	
6. Assessment of detailed design features	.14				*		.09	
7. Quantify reliability of other selected system designs					*		.14	
8. Development of criteria	.14				.14			
9. Prepare staff report	.49							
10. Develop branch technical position	.14				.07			
TOTAL (MAN YEARS)	1.14	.14	.11	.19	.35		.34	0.10

GRAND TOTAL = 2.37 man years

* Manpower indicated for Task 1 (PSB/DOR) also includes manpower for tasks 5, 6 and 7

AUG 19 1977

REVISION 0

Title: RHR Shutdown Requirements (A-31)

Lead Responsibility: Division of Systems Safety

Lead Assistant Director: D. F. Ross, Jr., A/D for Reactor Safety

Task Manager: Charles C. Graves, DSS

1. Problem Description:

On June 24, 1976 (Ref. 1), the RRRC approved a proposed revision of the standard review plan dealing with the RHR system (Section 5.4.7) and the Branch Technical Position RSB 5-1. On September 27, 1976, Westinghouse gave a detailed presentation of the impact of this position on the RESAR-3S standard nuclear steam supply system (Ref. 2). Westinghouse's conclusion was that the branch position requiring a capability to go from hot to cold shutdown without offsite power had a significant impact on (a) the chemical and volume control system (CVCS) in terms of additional valves required and associated seismic and environmental tests, and (b) interface requirements on balance of plant design, including a safety-grade instrument air system.

Concern has been expressed [illegible] impact of the position was not fully treated in the material presented to RRRC (Ref. 3). Accordingly, a review was made of three PWR plants (RESAR-3S, CESSAR-80, and B-SAR-205) to obtain a more thorough assessment of all systems affected by the requirement to go to cold shutdown without offsite power. As the result of this new review, it was proposed that the functional requirements of the position be retained, but the RSB 5-1 be modified to (a) establish minimum requirements for the size of the condensate water storage tank, (b) require procedures for cooldown following loss of offsite power, and (c) require justification that the procedures are adequate for reaching cold shutdown following loss of offsite power.

Drafts of the revised SRP 5.4.7 and the Reactor Systems Branch proposed BTP 5-1 have been circulated to DPM, DOR, and DSS for comment and were discussed in a meeting on July 7, 1977 (Ref. 4). Some changes to the revised SRP and BTP 5-1 will be made before submittal to RRRC.

References:

1. Memorandum from E. G. Case to L. V. Gossick, dtd. July 15, 1976.
2. Summary of Meeting held on Sept. 22, 1976 to discuss "Capability to Achieve Cold Shutdown Using Safety-Grade Systems and Equipment," C. O. Thomas, re Docket No. STN 50-545, dtd. Oct. 5, 1976.

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 19, 1977

3. Memorandum from R. C. DeYoung to B. C. Rusche on RHR System - Revision to Standard Review Plan, dtd. Sept. 24, 1976.

4. Memorandum from D. F. Ross, Jr., to R. E. Heineman on "Requirements for Capability to Achieve Cold Shutdown with Loss-of-Offsite Power," dtd. May 26, 1977.

2. Plan for Problem Resolution:

The remaining work involves 1) changes to the proposed SRP 5.4.7 and the proposed BTP 5-1 prior to submittal to RRRC, 2) review and approval of the proposed package by RRRC, 3) preparation of a request, with supporting documentation, to the Office of Standards Development for preparation of a regulatory guide, and 4) NRR approval and transmittal of documentation for publication.

3. NRR Technical Organizations Involved:

Reactor Systems Branch, Division of Systems Safety, has responsibility for completion of this task.

Manpower Estimate: 80 Man-hours FY 1977; 80 Man-hours Total

4. Technical Assistance:

None required

5. Interaction with Outside Organizations:

None required

6. Assistance Requirements from Other NRC Offices:

The Office of Standard Development will be requested to prepare regulatory guide based on the BTP 5-1. Work on the guide and issuance of the guide is not part of this task.

7. Schedule for Problem Resolution:

Completion of changes to SRP 5.4.7 and BTP 5-1: Oct. 1977

Approval by RRRC: Nov. 1977

Request for preparation of new regulatory guide: Nov. 1977

Transmittal of documentation: Nov. 1977

8. Potential Problems:

None

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 14, 1977

REVISION 0

TASK NO. A-33

TASK ACTION PLAN

Title:	NEPA Reviews of Accident Risks
Lead Responsibility:	Division of Site Safety and Environmental Analysis
Lead Assistant Director:	Richard H. Vollmer, Assistant Director for Site Analysis
Task Manager:	Oliver D. T. Lynch, Jr. Environmental Projects Branch 2

1. Problem Description:

In 1971 the AEC determined that, consistent with NEPA, the environmental assessments of requests for construction permits and operating licenses should include consideration of the possible impacts from accidents. An Annex to 10 CFR 50 Appendix D was proposed which provided guidance to applicants in this regard. This guidance was included in Regulatory Guide 4.2 and has constituted, since 1971, the basis for the staff reviews.

Since 1971, a considerable number of "realistic" accident assessments have been made. In substance these reviews have uniformly shown that the risks associated with potential accidental releases are very low.

The approach in these assessments, typically, is limited to preparation of a two page narrative summary that qualitatively describes accident probabilities and the rational for concluding that accident risks are low and a one page table that provides numerical

APPROVED BY TASC, OCTOBER 19, 1977

estimates of the consequences of various categories of accidents (excluding Class 9 events). The approach to developing these consequence estimates also involves a largely simplistic analysis; minor adjustments are made from case to case (basically to account for variations in power level, exclusion boundary distance and population density). These numerical estimates are also limited to airpathway consequences.

The staff's environmental statement by its nature is typically the only document concerning NRC license reviews that receives wide public and government agency attention and exposure even though other documents are circulated and available. It is evident from the comments received on the staff's statements that the present approach does not adequately inform the public regarding the sustance and depth of NRC's safety reviews nor adequately respond to public and various government agency questions dealing with the risks of accidents.

The Environmental Protection Agency (EPA) and the Department of the Interior (DOI) expressed the need for an improved treatment of accident risks and an expansion of the staff assessments to include quantitative estimates of Class 9 events.

Beginning in 1973, in response to EPA concerns, the staff augmented its assessments to discuss the then-ongoing Rasmussen Study as it related to Class 9 risks. For its part EPA agreed that updating of the standard assessment was not warranted until after the Rasmussen Study results were made available. This dialogue was renewed in 1976, with EPA recommending that a generic environmental statement be prepared on accident risks. After extended discussions, the NRC staff reiterated its 1973 commitment to update the standard assumptions in the proposed Annex A. As a precursor to this update, the staff committed to an extension of the WASH-1400 study to include a more in-depth evaluation of Class 3-8 accidents and to further explore the significance of variations in site and plant design characteristics.

The Department of the Interior has routinely suggested that more attention be given to the site risks associated with the liquid pathway. In mid-1977, DOI and NRC staff met to discuss the DOI's generic concerns. DOI was informed of the staff's programs to augment the generic studies in WASH-1400, but no commitments were made to revise the current approach (which, as noted above, includes no discussion of the impacts of accidental releases to the liquid pathway).

Another issue that has surfaced during the last few years relates to the lack of guidelines on situations which require substantially different (more detailed/different scope) treatment. In the cases of CRBRP,* OPS/ AGS** and Fulton/Summit substantial departures from the standard assumptions were required; for example requests were made for analyses to support the claimed conclusion that Class 9 risks were low, comparable to typical LWRs. In the cases of CRBRP and OPS, substantial schedular delays have resulted, apparently because the applicants were not aware that the standard guidelines regarding the treatment of Class 9 accidents applied to a limited design/site envelope. Additional guidelines need to be developed to help avoid future situations as these.

2. Plan for Problem Resolution

Based on a review of the various comments received over the last six years, and the commitments to EPA, the recommended approach is to conduct limited additional analyses (described below) and prepare a summary survey document which could be used as a standard reference regarding accident risks in the context of the staff's NEPA reviews (much as WASH-1400 is now cited in our statements). This same document would serve as the principal basis for a decision on the disposition of the proposed Annex to 10 CFR 50 Appendix D.

The area of quantitative risk assessment is very active and there are a variety of study efforts within and outside of NRC. Similarly, there are a multitude of efforts related to case reviews and generic issues. Both groups of activities may provide useful input to any revision of the guidelines for treating accidents in environmental statements. To the extent that results from these other efforts are available, consistent with the major milestones in this Task Action Plan, they will be considered. This plan includes only those tasks which must be conducted to fulfill existing commitments to EPA regarding the update of the proposed annex, and will not result in a full description of any and all possible accident risks from Light Water Reactors. The plan, however, does provide for efforts to assure that they are consistent between principal arguments and conclusions in the safety and environmental reviews (note, for example, 3e through 3g which call for DSS input on event analyses and review of contractor work; both activities are aimed at assuring consistency of approach and use of best available information).

The structure of this program is expected to involve four major subtasks: 1) extension of the WASH-1400 methodology to types of those events currently analyzed in the staff's environmental statements (Class 3-8 accidents)*** to develop a consistent, integrated set of generic risk analyses; 2) conduct of limited sensitivity studies of variations in estimated risks due to plant and site variation; 3) preparation and issuance of a for-comment report leading to a decision to revise or reissue as a Regulatory Guide (or some other

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* CRBRP (Clinch River Breeder Reactor Project)

** OPS/AGS (Offshore Power Systems/Atlantic Generating Station)

*** The study will consider only those accidents already defined as Class 3-8. No effort will be made to decide that those accidents adequately represent the entire spectrum of possible events.

action) the proposed Annex A to 10 CFR 50 Appendix D;4) developing the draft Regulatory Guide or draft revised Annex A, as appropriate, including a revision to the Environmental Standard Review Plan (ESRP) accident analysis (Section 7.1).

2.1 The Surry/Peach Bottom Class 3-8 Study

WASH-1400 methodology will be extended to a spectrum of non-core-melt accidents (Class 3 thru 8). The principal objective will be to develop risk estimates for the general class of events that are based on comparable analytical methods. Sub-objectives will be to suggest modified standard assumptions for Class 3-8 accidents and to develop improved estimates of the relative risks of various categories of accidents.

2.2 Sensitivity Studies

The main product of the sensitivity studies will be estimates of the range of risks that may be attributed to variations in plant/site characteristics.

2.3 Preparation of Report and Decision Regarding Annex A to 10 CFR 50 Appendix D

After the tasks in 2.1 and 2.2 have been performed the results need to be documented, together with a presentation and discussion of the background (including comments received) on staff assessments made during the period 1971-1977. This report will also be used to solicit views to modify the proposed annex. It must be resolved whether to revise the proposed Annex A and submit to the Commission the necessary rule making recommendations or it be more appropriate to issue a Regulatory Guide on the subject. Quite possibly, the study will reveal that some other action, rather than proposed Annex A revision or Regulatory Guide issuance, is more appropriate. This decision must be effected by NRC management after consideration of the staff's recommendations and input from interested parties on the study report.

2.4 Revision of Proposed Annex A or Issuance of Regulatory Guide

Subsequent to the decision indicated in Subtask 2.3, above, the appropriate action would be effected. Proposed rule making to revise Annex A would be developed and submitted to the Commission for action, or a draft Regulatory Guide would be developed and provided to the Office of Standards Development for action.

Revision of the Environmental Standard Review Plan

It is expected that, based on the results of the above tasks, a revision to the current treatment will be developed in the

ESs. This may include an analog to Table S-3, Summary of Environmental Considerations for the Uranium Fuel Cycle, which would include plant boundary conditions such that specific analyses would not have to be included.

It is recognized that Task A-33, NEPA Reviews of Accident Risks, should be integrated with an overall NRR reactor risk study. The Risk Assessment Methodology Application Plan is still under development. At such time as that plan is finalized, this Task Action Plan will be revised to indicate the interfaces.

3. <u>NRR Technical Organizations Involved</u>:

a. Environmental Projects Branch 2, Division of Site Safety and Environmental Analysis

Task Manager will serve in the principal management function for the task. The Task Manager will have primary responsibility for maintaining coordination, task progress, and general monitoring of the task effort within NRR, as well as managing preparation of the study report.

Estimated manpower:	FY 78	2 man months
	FY 79	4 man months
	FY 80	3 man months
	FY 81	1/2 man months
	TOTAL	9 1/2 man months

b. Accident Analysis Branch, Division of Site Safety and Environmental Analysis

AAB will carry a major task load within NRR, performing the following activities:

i. compilation of survey of LWR ES accident risk assessments vs. principal site and design features (Subtask 2.3)

ii. preparation (with PAB) of initial scoping assessment of accident risks (risk curve) for use in guiding the detailed studies by the contractors.* (Subtask 2.1)

iii. preparation of survey of major comments offered on LWR accident discussions in LWR ESs. (Subtask 2.3)

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* (See 3.c and 6.b. below)

iv. compilation of models used in case reviews of inplant releases and event summaries for use of the contractors. (Subtask 2.1)

v. compilation and limited extension of staff design sensitivity studies to guide detailed contractor studies. (Subtask 2.2)

vi. provision of miscellaneous technical assistance to laboratory contractors of RAB and RES participating in study. (Subtask 2.1)

vii. provision of major input to the study report. (Subtask 2.3)

viii. compilation of draft Regulatory Guide, (if required) or revisions to Annex A. (Subtask 2.4)

Estimated manpower:

FY 78	3 man months
FY 79	3 man months
FY 80	3 man months
FY 81	2 man months
TOTAL MANPOWER	11 man months

c. Radiological Assessment Branch, Division of Site Safety and Environmental Analysis

RAB will be a major participant in the study by providing:

i. planning of exercises for consequences model studies; this will require some effort to:

  a. review CRAC model (Calculations of Reactor Accident Consequences-computer code) used in WASH-1400 Appendix VI. (Subtask 2.1)

  b. review modeling and sensitivity studies of consequence dose predictions. (Subtask 2.2)

ii. input for consequences to the liquid pathway based on the Liquid Pathway Generic Study results.* (Subtasks 2.1, 2.2)

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* Current staff practices in environmental reviews do not consider risks resulting from releases to the liquid pathway. The bases for this practice will be one factor included in this task. The effort will largely draw on the Staff's LPGS and RES's program at Sandia.

iii. review of PNL assessment of recommended consequence models for Classes 3 thru 8. (Subtask 2.1)

iv. administration of contract with Battelle Memorial Institute - Pacific Northwest Laboratories. (Subtask 2.1) See Item 4.

v. input to the study report. (Subtask 2.3)

Estimated manpower:	FY 78	4 man months
	FY 79	4 man months
	FY 80	3 man months
	TOTAL MANPOWER	11 man months, including 6 man months contract administrator time.

d. Hydrology-Meteorology Branch, Division of Site Safety and Environmental Analysis

HMB will compile or review site meteorology variability, assessments of significance of variation in meteorology from site to site and provide input and assistance to contractor effort, as required. (Subtask 2.1)

Estimated manpower: 3 man months (FY-78)

e. Reactor Systems Branch, Division of Systems Safety

RSB will provide input on accident event classification and will review AAB assessment of event scenarios and accident models developed by contractors. (Subtask 2.1)

Estimated manpower: 2 man months (FY-78)

f. Auxiliary Systems Branch, Division of Systems Safety

ASB will provide input on accident event classification and will review AAB assessment of event scenarios and accident models developed by contractors. (Subtask 2.1)

Estimated manpower: 1 man month (FY-78)

g. Various other branches in Division of Systems Safety (as appropriate)

Provide secondary reviews of event scenarios and accident models developed by contractors. (Subtask 2.1)

Estimated manpower: 1 man month total manpower (FY-78)

4. Technical Assistance Requirements:

The Radiological Assessment Branch Division of Site Safety and Environmental Analysis will administer a contract with Battelle Memorial Institute - Pacific Northwest Laboratories. This contract is to provide the following effort:

a. review Appendix I to 10 CFR (ALARA) models for applicability to evaluate consequences of accidental radioactive releases,

b. review CRAC model for applicability to estimate latent health effects.

c. if necessary, develop and recommend standard consequence models for accident Classes 3 thru 8.

d. conduct sensitivity studies on design, site parameters and model variations,

e. issue reports.

The PNL contract study is to be executed in three phases, described as follows:

Phase	Activity
I	Definition and Scope of Work and development of Work Plan. Perform sensitivity study of consequence model for WASH-1400, ALARA and NRC accident analysis models
II	Evaluation of ALARA releases using new model at typical site. Integration of BNL/ASI and BCL source terms and probability into PWR evaluation. Rerun WASH-1400 releases with new model for Pressurized Water Reactor and Boiling Water Reactor
III	Address sensitivity studies of Phase II results to determine critical parameters.

This contract effort will involve approximately 6 man months of contract administrator time and 60 man months of laboratory effort over a period of 18 months with a FY-78 funding effort of $150,000 to be provided by DSE.

5. Interactions with Outside Organizations:

Interaction with outside organizations will be primarily with the Environmental Protection Agency and the Department of the Interior stemming from these agencies' interest and comments on NRC environmental statements consistent with Commission commitments to these agencies.* Briefings will be held early in the study to inform EPA and Interior of the NRC plan of action and intentions. These and other governmental agencies and the general public will have the opportunity to comment on the study results and planned revisions in our NEPA review practices.

6. Assistance Requirements from Other NRC Offices:

Assistance outside of NRR will be provided by the Office of Executive Legal Director, Regulations Division and the Office of Nuclear Reactor Research (RES), Probabilistic Analysis Branch (PAB). RES PAB is administering contracts with Battelle Columbus Laboratories (BCL), and Brookhaven National Laboratories (BNL)/Science Applications, Inc. (SAI), which will provide a major input to the study as indicated below:

a. Office of Executive Legal Director, Regulations Division

OELD will assist in preparation of a Federal Register Notice announcing the proposed rulemaking regarding any revision to the proposed Annex. Assistance will also be provided in review of the draft study report prior to publication. (Subtasks 2.3 and 2.4)

Estimated manpower:	FY 78	0.25 man months
	FY 79	0.75 man months
	TOTAL MANPOWER	1 man month

b. Office of Nuclear Reactor Research, Administration and Project Control, Probabilistic Analysis Branch

PAB will direct work at Battelle Columbus Laboratories and Brookhaven National Laboratories/Science Applications, Inc. These contracts are to provide assessment of probabilities and magnitudes of releases which would be associated with LWR reactor accident Classes 3 thru 8 consistent with the assessments of potential Class 9 accidents in WASH-1400. (Subtask 2.1)

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* Letter: L. V. Gossick to W. D. Rowe, USEPA, dated April 5, 1977.

The BNL-SAI and BCL contracts study is to be executed in four or five phases (Phase V is optional*), described as follows:

Phase	Activity
I	Definition and Scope of Work for Pressurized Water Reactor
II	WASH-1400 extension - SURRY for Pressurized Water Reactor
III	Definition and Scope of Work for Boiling Water Reactor
IV	WASH-1400 extension for Boiling Water Reactor
V (optional)	Review design and sensitivity of results for Pressurized Water Reactor and Boiling Water Reactor

This effort is expected to involve approximately 5 man months of contract administrator time and 97 man months of laboratory effort over a period of 18 months, with a funding effort of $759,000 for both BNL-SAI and BCL provided by RES.

7. Schedule for Problem Resolution:

Major milestones and target dates for completion of the described work are given herein. The work progression should proceed as below.

Subtask 2.1 Surry/Peach Bottom Class 3-8 Study

Milestone Description	Target Date
a. Committee approval of A-33 Task Action Plan (TASC)	September 1977
b. Survey of ES comments on accident analysis (AAB)	October 1977
c. Scoping assessment of accident risks (AAB, RAB)	November 1977

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* The need for Phase V will be determined based on the results of Phases II and IV. If these earlier phases show that detailed design sensitivity studies will not materially improve the estimated risks associated with the selected Class 3-8 events, Phase V will, in all likelihood, not be performed.

Milestone Description	Target Date
d. Completion of Survey of Risk Assessment and Event Scenarios (AAB, PAB) Review by RSS, RSB	January 1978
e. Complete BNL/SAI and BCL Phase I, Definition and Scope of Work (RES)	January 1978
f. Complete PNL Phase I, Definition and Scope of Work, Work Plan, Sensitivity Study of Consequence Model for WASH-1400, ALARA and NRC Accident Analysis models (RAB)	March 1978
g. Complete BNL/SAI and BCL Phase II, WASH-1400 Extension-Surry for PWR (RES)	September 1978
h. Complete BNL/SAI and BCL Phase III, Definition and Scope for BWR (RES)	January 1979
i. Complete PNL Phase II, Evaluation of ALARA releases using new model at typical site, Integration of BNL/SAI and BCL source terms and probability into PWR evaluation. Rerun WASH-1400 releases with new model for PWR and BWR (RAB)	March 1979
j. Complete BNL/SAI and BCL Phase IV, WASH-1400 Extension for BWR (RES)	April 1979

Subtask 2.2 Sensitivity Studies

Milestone Description	Target Date
a. Initiate PNL Phase III, Address sensitivity studies of Phase II results to determine critical parameters for Classes 3-8 and Class 9 and man-rem doses within 50 miles (RAB)	March 1979
b. Initiate BNL/SAI and BCL Phase V (Optional), Review design and sensitivity of results, PWR and BWR (RES)	April 1979

Milestone Description	Target Date
c. Complete PNL Phase III, Address sensitivity studies of Phase II results to determine critical parameters (RAB)	January 1980
d. Complete BNL/SAI and BCL Phase V (Optional), Review design and sensitivity of results, PWR and BWR (RES)	March 1980

Subtask 2.3 Report on NEPA Reviews of Accident Risks and Decision Regarding Annex A to 10 CFR 50 Appendix D

Milestone Description	Target Date
a. Receive Survey of ES comments for AAB	October 1977
b. Receive Scoping assessment of accident risks from AAB and RAB	November 1977
c. Receive Survey of Risk Assessment and Event Scenarios from AAB and PAB	January 1978
d. Develop Scope and Outline of Report and Author Assignments (TM, AAB)	February 1978
e. Receive BNL/SAI and BCL Phase II report (RES)	September 1978
f. Receive BNL/SAI and BCL Phase III report (RES)	January 1979
g. Receive PNL Phase II report (RAB)	March 1979
h. Issue Draft Report on NEPA reviews of Accident Risks (TM, AAB, RAB, OELD)	April 1979
i. Receive BNL/SAI and BCL Phase IV report (RES)	April 1979
j. End of Public Comment Period on Report	June 1979
k. Receive PNL Phase III Report	January 1980

Milestone Description	Target Date
l. Receive BNL/SAI Phase V Report	March 1980
m. Issue Final Report on Accident Risks (TM, AAB, RAB, OELD)	March 1980
n. Staff recommendations submitted to NRR management (TM, AAB, RAB)	March 1980
o. Initiate Summary of study results for all ES use (TM, AAB, RAB)	March 1980
p. NRR management decision	May 1980
q. Complete Survey of Study Report for all ES use (TM, AAB, RAB)	May 1980

Subtask 2.4 Revision of Proposed Annex A or Issuance of Regulatory Guide

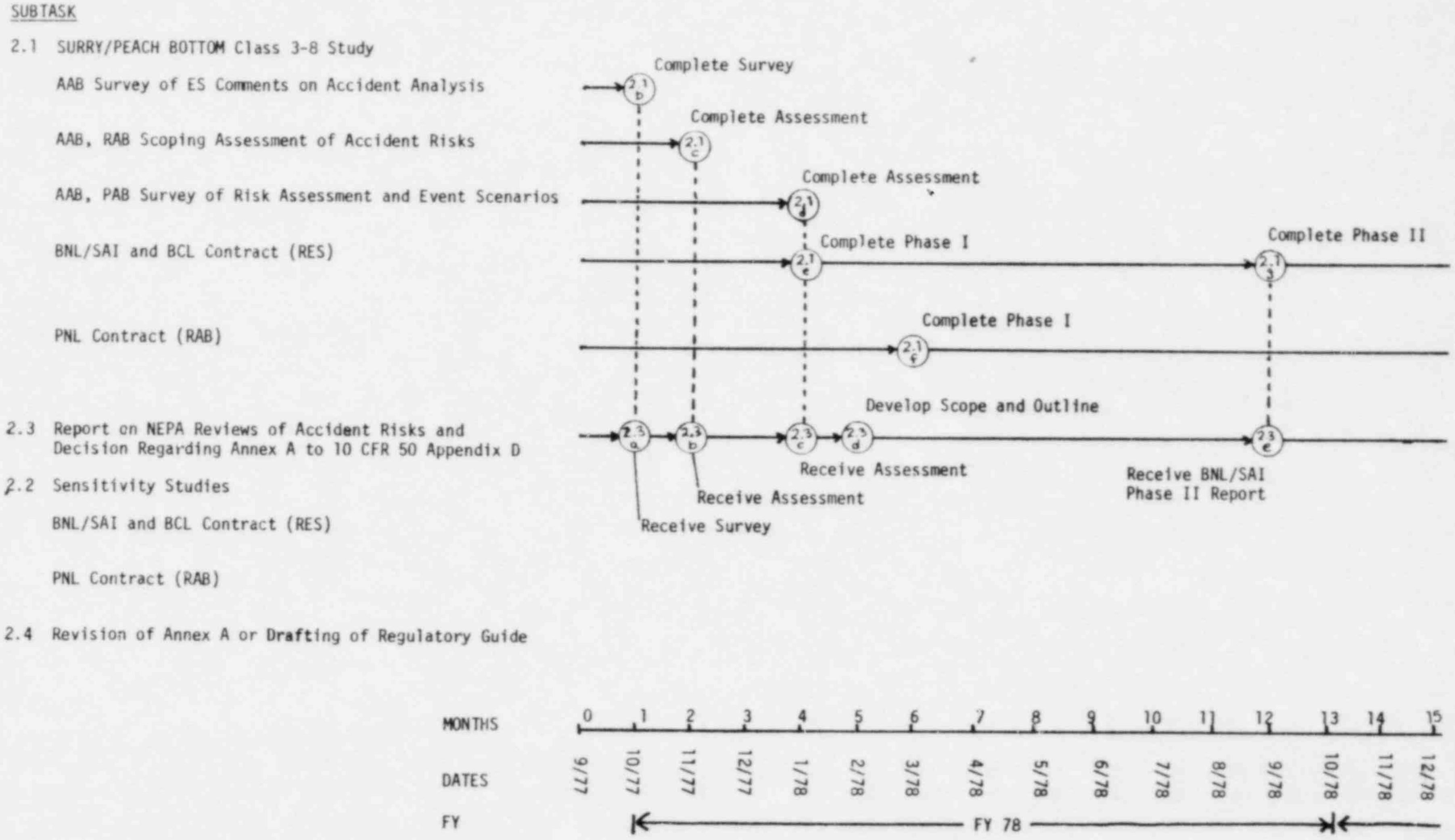
Milestone Description	Target Date
a. NRR management decision on Annex A (See Subtask 2.3)	May 1980
b. Initiate Proposed Revision of Annex A or development of draft Regulatory Guide (TM, AAB)	May 1980
c. Complete proposed Revision of Annex A or draft Regulatory Guide and provide to Commission or OSD for further action (TM, AAB)	January 1981

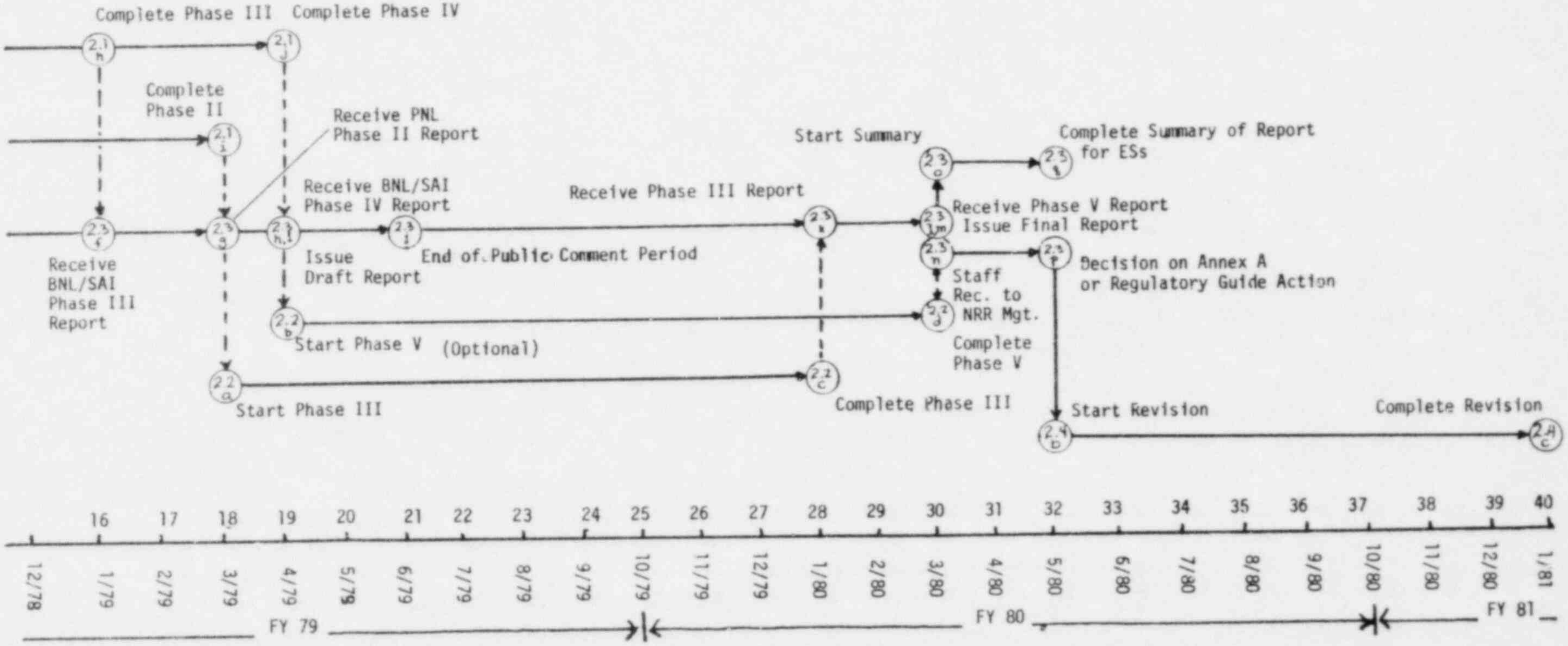
8. Potential Problems:

RES may require additional funding to complete all related contract work as the schedule progresses. Failure to obtain these funds could affect the full benefit of the project.

In all cases and at all levels timely completion of schedule performance will be required to assure the study does not bog down at the man points of interaction.

NEPA REVIEWS OF ACCIDENT RISKS, TASK A-33
GRAPHICAL TASK ANALYSIS




Complete Phase III
Complete Phase IV
Complete Phase II
Receive PNL Phase II Report
Receive BNL/SAI Phase IV Report
Receive Phase III Report
Start Summary
Complete Summary of Report for ESs
Receive Phase V Report
Issue Final Report
Decision on Annex A or Regulatory Guide Action
Staff Rec. to NRR Mgt.
Complete Phase V
Receive BNL/SAI Phase III Report
Issue Draft Report
End of Public Comment Period
Start Phase V (Optional)
Start Phase III
Complete Phase III
Start Revision
Complete Revision
2.1 h
2.1 j
2.1 i
2.3 f
2.3 g
2.3 h.1
2.3 i
2.3 k
2.3 o
2.3 q
2.3 l,m
2.3 n
2.3 p
2.2 b
2.2 d
2.2 a
2.2 c
2.4 b
2.4 c
16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40
12/78 1/79 2/79 3/79 4/79 5/79 6/79 7/79 8/79 9/79 10/79 11/79 12/79 1/80 2/80 3/80 4/80 5/80 6/80 7/80 8/80 9/80 10/80 11/80 12/80 1/81
FY 79
FY 80
FY 81

AUGUST 23, 1977

Title: Instruments for Monitoring Radiation and Process Variables During Accidents, TASK A-34

Lead Responsibility: Division of Site Safety and Environmental Analysis

Lead Assistant Director: Richard H. Vollmer, Assistant Director for Site Analysis

Task Manager: Frederick J. Hebdon, Project Manager, Environmental Projects Branch 1

1. Problem Description

To develop criteria and guidelines to be used by applicants, licensees and staff reviewers to support implementation of Regulatory Guide 1.97, Revision 1 (Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident).

Such criteria and guidelines would provide specific guidance on functional and operational capabilities required of the various classes of instruments, including in-plant and ex-plant instruments. Where such guidance cannot be provided, the rationale to be applied to derive requirements for specific situations will be provided.

2. Planned Staff Approach

a. Detailed guidance and acceptance criteria concerning implementation of Regulatory Guide 1.97 has not yet been developed. Therefore, the members of this Task Group will answer questions that arise before and during the development of the required proposals for implementation of Regulatory Guide 1.97 for the lead plants described below. In this way, the Task Group will develop the necessary guidance as it is needed by the lead plant applicants. The Task Group will also be responsible for the review of submittals made by the lead plant applicants.

b. There are two aspects of the implementation of Regulatory Guide 1.97, Revision 1 (Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident) that must be considered.

(1) Position 3 of RG 1.97 requires the installation of specific instrumentation to follow the course of an accident (IFCA). The staff has determined that this requirement should be satisfied in as timely a manner as possible. The Task Group established by this Task Action Plan will identify

APPROVED BY TASC, AUGUST 19, 1977
TASC COMMENTS INCORPORATED, AUGUST 23, 1977

lead plants (at least one BWR and one PWR) for implementation of Position 3 and will answer questions raised by the lead plant applicants, assume responsibility for the review of the proposals for implementation of Position 3 that are submitted. Based on the experience gained during this review, the Task Group will prepare uniform review procedures and acceptance criteria to be used by the staff for the review of subsequent implementation proposals.

(2) Full implementation of RG 1.97 requires the applicant/licensee to prepare a Safety Analysis which is reviewed by the staff. Lead plants (at least one BWR and one PWR) for full implementation of RG 1.97 will be designated. The Task Group established by this Task Action Plan will assist the lead plant applicants in the development of the required Safety Analyses by answering questions from the applicants. The Task Group will review the Safety Analyses when they are submitted. Based on the experience gained during the development and review of the Safety Analyses for the lead plants, the Task Group will prepare guidance to assist other applicants/licensees in the development of the required Safety Analysis and acceptance criteria to be used by the staff to review the Safety Analyses submitted.

C. Description of the End Product of Task Group

(1) A letter to all applicants and license containing guidance to facilitate the preparation of Safety Analyses required by RG 1.97.

(2) Revision of various Standard Review Plans to provide for the uniform review of required Safety Analyses and Proposals for Implementation of Position 3.

(3) Recommendation for revision of RG 1.70, Standard Format and Content of SAR's for Nuclear Power Plants

(4) Recommendations for confirmatory research as required.

(5) Recommendations for revisions to RG 1.97.

3. NRR Technical Organizations Involved

These branches will carry out their responsibilities through participation on the Task Group.

a. Accident Analysis Branch (DSE) - review the Safety Analyses required by RG 1.97 for the lead plants to ensure that variations in plant variables are adequately defined, from a consequences viewpoint, for the Design Basis Accidents analyzed. This review will also include evaluation of operator interactions (e.g., procedures, actions, timing) for utilizing instrumentation to follow the course of an accident (IFCA) to assess and minimize risk. Develop guidance for applicants/licensees and uniform review procedures for the staff to support the implementation of RG 1.97 on other plants. Review the plans for implementation of Position 3 for lead plants and develop uniform review procedures for the staff to use to review implementation proposals for other plants. (Manpower Requirements: 1 reviewer, 2MM per reviewer.)

b. Reactor Systems Branch (DSS)
Containment Systems Branch (DSS)
Auxiliary Systems Branch (DSS)
Power Systems Branch (DSS)

Review the Safety Analyses for the lead plants to ensure that significant process variables required to monitor the course of Design Basis Accidents, from a systems performance viewpoint, are identified. This review will also include evaluation of operator interactions (e.g., procedures, actions, timings) for utilizing IFCA to optimize system performance. Develop guidance for applicants/licensees and uniform review procedures for the staff to use to implement RG 1.97 on other plants. (Manpower requirements: 1 reviewer per branch, 3MM per reviewer in RSB, 1MM per reviewer in CSB, and PSB.)

c. Radiological Assessment Branch (DSE) and Effluent Treatment Systems Branch (DSE) - develop criteria for application of inplant and explant radioactivity monitoring systems to follow the course of an accident during various accident situations and accident scenarios. Review the Safety Analyses for the lead plants to ensure that plant radiation sources are adequately defined and that radiation monitoring is adequate from the viewpoint of protection of the health and safety of utility staff personnel, of emergency program personnel and of the public outside the immediate plant environs. (Manpower requirements: 1 reviewer, 2 MM per reviewer for RAB and 1 reviewer, 1 MM per reviewer for ETSB).

d. Instrumentation and Control Systems Branch (DSS) - review the Safety Analyses for the lead plants to ensure that IFCA is appropriately designed, will remain operable as required, and will accurately represent the information required by the operator.

This review will include consideration of maintenance and testing of instrumentation. Develop guidance for applicants/licensees and review procedures for the staff to use to implement RG 1.97 on other plants. Review the plans for implementation of Position 3 for lead plants and develop uniform review procedures for the staff to support the review of implementation proposals for other plants. (Manpower Requirements: 1 reviewer, 2MM per reviewer.)

e. Operator Licensing Branch (DPM) - assist in evaluating operator interactions and expected operator responses to identify the instrumentation required and the procedures to be followed to deal with Design Basis Accidents. Develop guidance for applicants/licensees and uniform review procedures for the staff to support implementation of RG 1.97 on other plants. (Manpower Requirements: 1 reviewer, 1MM per reviewer.)

f. Emergency Planning Branch (DPM) - review the Safety Analyses for lead plants and the applicant's Emergency Plan to ensure that the operator will be supplied with the information needed to permit him to provide authorities responsible for implementation of Emergency Plan with accurate and timely recommendations concerning implementation of all or part of the plan. Develop guidance for applicants/licensees and uniform review procedures for the staff to support implementation of RG 1.97 on other plants. Review the plan of Position 3 for lead plants and develop uniform review procedures for the staff to support the review of implementation proposals for other plants. (Manpower Requirements: 1 reviewer, 1MM per reviewer.)

g. Environmental Projects Branch 1 (DSE) - provide a Task Manager to serve in the principle management function for the project. (Manpower requirements: 1 project manager, 3MM per project manager.)

h. Operating Technology (DOR) - Review and comment on materials developed by the Task Group. Adapt the criteria and guidance developed by the Task Group for use by reviewers and licensees of operating reactors. (Manpower Requirements: 1 reviewer per branch (4 branches), 1 MM per reviewer.)

i. Other Branches in NRR may be called upon to provide technical support to the Task Group as needed on a consultation basis. (Manpower Requirements: Total 1MM.)

4. Technical Assistance Funds and Confirmatory Research Funding Required

It is not presently anticipated that technical assistance funding or confirmatory research funding will be required to directly support this Task Group. Two projects (described below) may produce data that will support the activities of this Task Group.

a. DOR has an existing technical assistance contract with BNL to evaluate certain operating plants to determine the capability of existing effluent radiation monitors to measure radioactivity releases through anticipated release paths from postulated accidents. The funding level for this program is $25K for FY 1977 and FY 1978.

b. DSE has an existing technical assistance contract with Allied Chemical Company (INEL) to develop bases for the specification of gaseous effluent accident monitoring instrumentation. The funding level for this program is $40K for FY 1977.

5. Interaction with Outside Organizations

The Task Group will maintain close contact with applicants for the lead plants.

6. Assistance Requirements from Other NRC Offices

Office of Standards Development - issue RG 1.97 Rev 1 and assist in the development of subsequent revisions of RG 1.97 and other Regulatory Guides based on experience gained during the review of the lead plants.

7. Schedule for Problem Resolution

		Schedule (Months)	Expected Completion Date
a.	Development of a Task Action Plan		June 29, 1977
b.	Approval of the Task Action Plan		August 1977
c.	Issuance of Reg Guide 1.97 Rev 1		August 15, 177
d.	Identification and notification of lead plants for full implementation of RG 1.97		August 31, 1977

Schedule for Problem Resolution (Cont'd)

		Schedule (Months)	Expected Completion Date
e.	Identification and notification of lead plants for implementation of Position 3 of RG 1.97	0	August 31, 1977
f.	Submission of lead plant proposals for implementation of Position 3 (3 months)	+3	December 1, 1977
g.	Completion of review of lead plant proprosal for implementation of Position 3 (3 months)	+ 6	March 1, 1978
h.	Development of uniform review procedures for implementation of Position 3 on other plants (1 month)	+ 7	April 1, 1978
i.	Submission of the Safety Analyses for the lead plants for full implementation (4 months)	+4	January 1, 1978
j.	Completion of review of the lead plant Safety Analyses (4 months)	+ 8	April 1, 1978
k.	Revise and develop guidance to licensee/ applicants and the staff to support implementation of RG 1.97 on other plants (See Section 2c for specific end products anticipated). (2 months)	+ 10	July 1, 1978

8. Potential Problems

Based on preliminary studies, as exemplified in BNWL-1635, it is anticipated that many plant evaluations particularly those for operating plants, will show the need for monitoring equipment not commercially available and, therefore, a lead time of six months to two years may be necessary for development, procurement, and installation of monitoring equipment.

REVISION 0

TASK ACTION PLAN

TASK NUMBER A-35

OCT 12 1977

Title: Adequacy of Offsite Power Systems

Lead Responsibility: Division of Systems Safety/NRR

Lead Assistant Director - D. G. Eisenhut, DOR:OT

Task Manager - D. G. McDonald, DOR:PSB

APPROVED BY TASC,
OCTOBER 19, 1977

1. Problem Description

Recent events at Millstone 2, Turkey Point 3 and 4, and Indian Point 2 and 3 involving the offsite power system have provided additional indication that the reliability of the preferred source of emergency power may be less than what has been expected. A study is needed to assess this matter.

General Design Criterion 17 (GDC 17) "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants, of 10 CFR Part 50 requires the offsite power source to be available and to have sufficient capacity and capability to assure that: (1) the fuel and reactor pressure boundary are maintained within specified acceptable limits; and (2) core cooling, containment integrity, and other vital safety functions are maintained during accident conditions.

The Commission has in the past, accepted the results of transient and steady state stability analyses documented in the Safety Analysis Reports for license applications which indicate that the offsite power source remains stable and meets the requirements of GDC-17. The disturbances on the Florida Power and Light (FP&L) system in 1973 and 1974 resulted in the examination by the staff of the offsite power system and the stability analyses performed by FP&L in greater detail. In addition, the abnormal occurrences at Millstone, Unit 2, reported to Congress in NUREG-0090-5, July-September 1976 indicated that sustained degradation of the offsite power source could result in failure of redundant safety-related electrical equipment or components.

NUREG-0138, "Staff Discussion of Fifteen Technical Issues," Issue 10, defined the staff's concerns relating to: (1) the reliability of the offsite power system as the preferred emergency source; (2) vulnerability of safety-related equipment to sustained degraded voltage; (3) adequacy of design interfaces of offsite and onsite power sources; and (4) adequacy of testing the onsite power sources. In addition, Issue 9 of NUREG-0138 defined a concern relating to a rapid rate of frequency decay on the offsite power system; a rapid rate of frequency decay could provide an electrical braking effect on the reactor coolant pump motors resulting in a flow coastdown in excess of that analyzed in the accident analysis portion of the Safety Analysis Reports.

The recent blackouts in Florida and New York provide additional emphasis on the concerns expressed relating to the availability of the preferred offsite power sources.

2. Plan for Problem Resolution

A. Approach

The emergency power systems of selected operating nuclear facilities and those in the licensing process will be evaluated to determine the adequacy of existing criteria in relation to the susceptibility of redundant safety-related electric equipment to: (1) sustained degraded voltage condition on the offsite power source; (2) interaction of the offsite and onsite power sources; and (3) adequacy of existing testing requirements. In addition, the results of the Technical Assistance Program with Oak Ridge National Laboratory (described in detail in Section 4) will identify those conditions affecting offsite power sources which may require that additional safety measures, Technical Specification changes, or design changes be taken to complement those already implemented as a result of our prior reviews.

The tasks identified in Section 2C are required to determine the adequacy of the offsite power source and its interface with the onsite power system. The results of these tasks will provide the input and bases for modifying existing criteria, if required, relating to: (1) monitoring grid conditions to identify when a grid would be vulnerable to a subsequent contingency (failure); (2) additional procedural actions or requirements to be taken within the nuclear plant when the grid is vulnerable to natural events or grid system equipment failures, e.g., start onsite diesel generators; (3) design changes which can provide a dedicated offsite power source to some nuclear plants; (4) design changes to provide additional protection for redundant safety-related equipment from sustained voltage degradation of the offsite source; and (5) determination of the adequacy of existing testing requirements for the onsite power sources.

B. End Product

The end product of this program will be a NUREG report which will document: (1) the details and results of the program; (2) the basis for the development of staff positions regarding the adequacy of the existing review procedures; (3) acceptance criteria that will be used for evaluation of offsite power system; and (4) recommendations for new or revised review procedures and criteria determined to be necessary as the result of the program.

Any new or revised criteria resulting from this program will be factored into the licensing process by the preparation of new or revised Technical Specifications, Branch Technical positions, standard review plan and recommendations for Regulatory Guides including

R. G. 1.70. This program will be considered complete when the above documents have been developed and are available for use in the licensing process.

C. Tasks

To assure overall electrical power systems reliability, this Task Action Plan will be coordinated to the extent necessary with other electrical-related Task Action Plans (A-25, "Qualification of Class IE Safety-Related Equipment," A-24, "Non-Safety Loads on Class IE Power Systems," A-30, "Adequacy of Safety-Related DC Power Supplies."

The following tasks are required to determine the adequacy of the in-plant AC power system design and the interface of the offsite source and the onsite power distribution system:

1) Define the requirements and criteria for under or over-voltage detection systems to protect the redundant safety-related loads, their control circuitry and the associated electrical components from sustained degradation of the offsite power system voltage.

2) Define the technical specification requirements for under or over-voltage monitors, loss of power and degraded conditions, include the limiting conditions for operation, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values.

3) Define "stable offsite power system conditions" as related to the voltage supplied at all safety-related buses. Include the effect of the various voltage transformations and bus loads from the normal operating offsite power systems voltage to and including the 480/120 volt control transformers at the safety loads.

4) Define the time duration that transient conditions outside the conditions of offsite power system stability defined in Task (3) can exist without degrading the performance of safety-related equipment or result in common mode failure of redundant safety-related equipment.

5) Determine the relative reliability of the various designs, considering the effects of transient and degraded grid conditions, for connecting the offsite power from the switchyard to the emergency buses.

6) Determine the adequacy of the existing test requirements to demonstrate the full functional operability and independence of the onsite power sources and verify the absence of adverse system interactions with the offsite power source.

The following tasks are required to determine the adequacy of the offsite power source during all operating conditions.

7) Develop a generic list of questions the responses to which would be used to provide a greater degree of assurance that current offsite power source stability analyses encompass the worst case conditions, that the utilities' methods of performing offsite power system stability analyses are acceptable, and that the results of the analysis are valid.

8) Prepare a final report of the generic aspects of the Florida Power and Light Company's system disturbance, May 16, 1977, which resulted in loss of offsite power to the St. Lucie and Turkey Point facilities. Include recommendations relative to criteria, operational restrictions, and/or technical specification requirements to improve the assurance of the availability and capability of all nuclear power plant offsite power systems.

9) Prepare the final staff report of the generic aspects of the Consolidated Edison Company system disturbance, July 13-14, 1977, which resulted in loss of offsite power to the Indian Point facilities. Include recommendations relating to criteria, operational restrictions and/or technical specification requirements to improve the assurance of the availability and capability of all nuclear power plant offsite power systems.

10) Determine the generic maximum credible grid frequency decay rate and provide the bases for the determination.

11) Study the offsite power systems of selected plants to determine which perturbations and grid configurations can lead to unacceptable power, voltage, and frequency conditions at the grid/plant interface.

12) Provide a description of the methodology and a computer program(s) for use in identifying critical system parameters and components relative to maintaining a stable offsite power source and configuration.

13) Perform a survey to determine the number and the types of events that have occurred which resulted in offsite power system conditions outside of the normal limits.

14) Provide the methodology used to determine when an offsite power system would be in a normal alert condition, e.g., those operating conditions that are within one contigency of system instability.

The results of all the tasks identified above will be factored into the following:

15) Develop new or augment existing criteria, if required, for the offsite and onsite ac power systems. The criteria will be processed for inclusion in the licensing process and the Systematic Evaluation Program for operating plants. This task will include the recommendations for new or revised regulatory guides, technical specifications, standard review plan (including branch technical positions), standard format, value/impact assessment, and assessment of the need for backfitting on a generic basis.

16) Prepare a staff report in the form of a NUREG document which will provide complete documentation of the details, conclusions, and any new or augmented criteria developed as the result of the staff's implementation of this task action plan relating to offsite power systems.

3. NRR Technical Organizations Involved

A. Plant Systems Branch, Division of Operating Reactors

1) Task 1 and 2 - Develop the requirements and criteria (technical specification and basis) for an under and over voltage detection system to protect the redundant safety-related loads, their control circuitry and the associated electrical components from sustained degradation of the offsite power system voltage. The report for this task has been completed by PSB/DOR and concurred in by PSB/DSS.

The staff positions and basis are documented in a memorandum from D. G. Eisenhut to K. R. Goller, dated April 20, 1977. These positions have been forwarded to all licensees of operating reactors.

Manpower requirements - none. (complete)

2) Task 3 and 4 - Define the design voltage requirements for the safety loads, their control circuitry, and the associated electrical components at the redundant safety-related buses. Determine the time duration that transients or sustained degradations of the common offsite power source can be reflected onto the onsite systems redundant safety-related buses without resulting in failure of the safety-related electrical equipment.

The information provided by the licensees in response to DOR's request for additional information relative to the electrical power systems of currently operating facilities, dated August 12 and 13, 1976, and applicable industry standards will be used to determine the acceptable voltage and time.

Manhpower requirements - 160 hours.

3) Task 5 - Work in conjunction with PSB/DSS to determine what effects transients or sustained degradation of the offsite power source will have on the various designs for connecting offsite power from the switchyard to the emergency buses.

Manpower requirements - 240 manhours.

4) Task 8 and 9 - Prepare final reports relating to the generic aspects of the system disturbance on the Florida (FP&L) and New York (CECo) system. Each report will include the sequence of events, effect on safety of the nuclear plants, causes of events and recommendations for augmenting or changing existing criteria, procedures, or

technical specifications to improve the assurance of the availability and capability of nuclear plant's offsite power systems. Oak Ridge National Laboratory (ORNL) will provide assistance in this area.

Manpower requirements - 640 manhours.

5) Task 10 - Determine the maximum credible frequency decay rate as a function of time and provide the bases for establishing the value. The established value will be used as input for Task Action Plan B-70, "Power Grid Frequency Degradation and Its Affect on Primary Coolant Pumps."

Manpower requirements - 240 manhours.

6) Task 15 and 16 - The development of new or augmented criteria to be used in the licensing process and for operating reactors will be based upon the input provided from all the other tasks. These tasks will require the joint effort of PSB/DSS and PSB/DOR. The joint effort will also be required in preparing the staff report in the form of a NUREG.

Manpower requirements - 1040 manhours.

B. Power Systems Branch, Division of Systems Safety

1) Task 1 and 2 - Revise as necessary for use in CP and OL licensing reviews the staff position (developed by PSB/DOR for operating plants with concurrence by PSB/DSS) regarding the requirements and criteria for under and over voltage detection, and for protection of safety-related electrical systems and equipment from these conditions.

The staff positions have been revised as required for use in the licensing process. Documentation of the revised positions and their application in the OL review of Three Mile Island, Unit 2 is contained in a memorandum from R. L. Tedesco to D. Vassallo dated August 16, 1977.

Manpower requirements - none

2) Task 3 and 4 - Work in conjunction with PSB/DOR to define the design voltage/time requirements for the safety-related systems and components.

Manpower requirements - 120 manhours.

3) Task 5 - Perform reliability analyses to determine the relative reliability of the various designs for connecting offsite power from the switchyard to the emergency buses; consider the effect of transient or degraded grid conditions and possible faults in the plant electric power generation system. The designs to be addressed include:

   a) Normal power feed through unit auxiliary transformer with fast transfer to station service transformer.

   b) Normal power feed through a continuously connected station service transformer(s).

   c) Normal power feed through a unit auxiliary transformer with generator breaker(s) disconnect.

   d) Normal power feed through a unit auxiliary transformer with generator load switch(s) disconnect.

   Manhour requirements - 440 manhours.

4) Task 6 - Determine the adequacy of the existing test requirements to demonstrate the full functional operability and independence of the onsite power sources and verify the absence of adverse system interaction with the offsite power source.

   Manhour requirements - 200 manhours.

5) Task 7 - Develop a list of generic questions the responses to which will be used to provide a greater degree of assurance that the results of current offsite power stability analysis encompass the worse case conditions. ORNL has provided some input to assist in defining the required information.

   Manhour requirements - 120 manhours.

6) Task 10 - Work in conjunction with PSB/DOR to determine the maximum credible frequency decay rate as a function of time and provide the bases for establishing the value. The established value will be used as input for Task Action Plan B-70, "Power Grid Frequency Degradation and Its Affect on Primary Coolant Pumps."

   Manhour requirements - 240 manhours.

7) Task 15 and 16 - Develop new or augmented criteria to be used in the licensing process and for operating reactors based upon input from all the other tasks. These tasks will require the joint effort of PSB/DSS and PSB/DOR. The joint effort will also be required in preparing the staff NUREG report.

Manpower requirements - 880 manhours.

4. Technical Assistance Requirements

A. Oak Ridge National Laboratory

1) Title: Electrical Systems Analysis

2) Responsible Division/Branch: Division of Operating Reactors/ Plant Systems Branch

3) Objective: To define characteristics of the offsite power systems and their relationship to the nuclear power plant safety. Provide recommendations and basis for establishing guidelines, licensing positions, or operational requirements for improving the assurance of plant safety in the event of offsite power system transients and sustained degradation.

4) Work Scope

a. Task 7 - Provide input to assist PSB/DSS in developing the list of questions for determining the adequacy of current offsite power stability analyses.

The memorandum from F. Clark (ORNL) to D. McDonald, dated June 9, 1977, includes the input requested. This information has been provided to PSB/DSS.

b. Task 8 and 9 - ORNL will provide assistance to PSB/DOR in the evaluation and preparation of reports relating to the generic aspects of the system disturbances in Florida and New York.

c. Task 10 - Assist PSB/DSS and PSB/DOR in determining the maximum credible grid frequency decay rate and the bases for the determination.

ORNL has provided a critique of the Westinghouse report, "WCAP-8424, An Evaluation of Loss of Flow Accidents Caused By Power System Frequency Transient in Westinghouse PWR's", in a memorandum to the Director, Division of Operating Reactors, from L. C. Oakes (ORNL), dated June 20, 1977. In addition, information has been provided to PSB/DSS in a memorandum dated July 12, 1977 from F. Clark (ORNL) to F. Rosa.

d. Task 11 - Study the offsite power systems of selected plants to determine which perturbations and grid configurations can lead to unacceptable power, voltage, and frequency conditions at the grid/plant interface.

   The memorandum to the Director, Division of Operating Reactors from L. C. Oakes (ORNL) dated July 27, 1977, includes the input requested and will be utilized by PSB/DOR in implementing Task 1 and 2.

e. Task 12 - Provide a description of the methodology and a computer program(s) for use in identifying critical system parameters and components (sensitivity matrix) relative to maintaining stable offsite power sources and configurations.

f. Task 13 - Perform a survey to determine the number and types of events that have occurred which resulted in grid conditions outside the normal limits for nuclear facilities offsite power sources.

g. Task 14 - Provide the methodology used to determine when a grid system would be in a "Normal Alert" condition, e.g., those operating conditions that are within one contingency of system instability.

h. Task 15 and 16 - ORNL will provide a final report summarizing their efforts and provide recommendations for any operational restrictions or criteria related to:

   (1) The offsite power system;

   (2) Design modifications for plant protection from transient and sustained degraded conditions; and

   (3) Procedural or technical specifications requirements relating to the plant or offsite power operation.

   The information provided in the report will be used to assist the staff in completing Tasks 15 and 16.

Funding: FY 1977 - $120,000
FY 1978 - $120,000 (requested)

5) Interactions with Outside Organizations

The Division of Operating Reactors and the Technical Assistance Program contractor will be contacting various utilities, Electric Power Research Institute, Federal Power Commission and other industry-related organizations as necessary to obtain required information to complete the task.

6) Assistance Requirements from the NRC Offices

Assistance from other NRC offices is not anticipated at this time.

7) Schedule for Problem Resolution

A. Summary of Schedule - NRR

Task 1 - Criteria for the under or over voltage monitors; Protection for sustained degradation of the offsite power source.	Complete
Task 2 - Define Technical Specification requirements for the under/over voltage monitors.	Complete
Task 3 - Define the voltage requirements at the safety-related buses.	Nov 1, 1977
Task 4 - Define the maximum time duration that a degraded voltage condition can be allowed to exist on safety-related buses.	Nov 1, 1977
Task 5 - Comparison of the reliability of various designs for connecting offsite power from the switchyard to emegency buses.	May 1, 1978
Task 6 - Determine the adequacy of existing test requirements relating to the onsite power source.	Dec 1, 1977
Task 7 - Develop a list of questions to determine the adequacy of current offsite power source stability analyses.	Dec 1, 1977

Task 8 -	Report on the generic aspects of the Florida Power and Light Company system disturbance.		Dec 15, 1977
Task 9 -	Report on the generic aspects of the Consolidated Edison Company system disturbance.		Jan 15, 1978
Task 10 -	Determine the maximum credibel frequency decay rate and the bases for the determination.		Feb 1, 1978
Task 15 -	Development of new or augmented criteria for offsite power systems. (TS., BTP & SRP)	Draft Final	Oct 1, 1978 Nov 1, 1978
Task 16 -	Preparation of a NUREG report	Draft Final	Nov 1, 1978 Dec 1, 1978

B. Summary of Schedule - Technical Assistance Program (ORNL)

Task 7 -	Provide input to assist PSB/DSS in development of list of questions to determine adequacy of current stability analyses.	Oct 1, 1977
Task 8 and 9 -	ORNL will provide assistance to PSBDOR in the evaluation and preparation of reports relating to the generic aspects of the system disturbances in Florida and New York.	Nov-Dec 1977
Task 10 -	Determine the maximum credible frequency decay rate and the bases for the determination.	Jan 1, 1978
Task 11 -	Determine the perturbations and system disturbances that can lead to unacceptable conditions at the grid/plant interface.	Jul 1, 1977 (complete)
Task 12 -	Prepare the sensitivity matrix for use in identifying critical system components relative to system stability.	Aug 1, 1977
Task 13 -	Document results of a survey of events leading to unacceptable offsite power source conditions.	Sep 1, 1977
Task 14 -	Document methodology for determining the normal alert conditions of the offsite power source.	Apr 1, 1978

Tasks 15 and 16 - Provide input and recommendations to assist the staff in developing criteria or operational restrictions relating to the offsite power source.

Draft Sep 1, 1978
Final Oct 1, 1978

C. Detailed Schedule

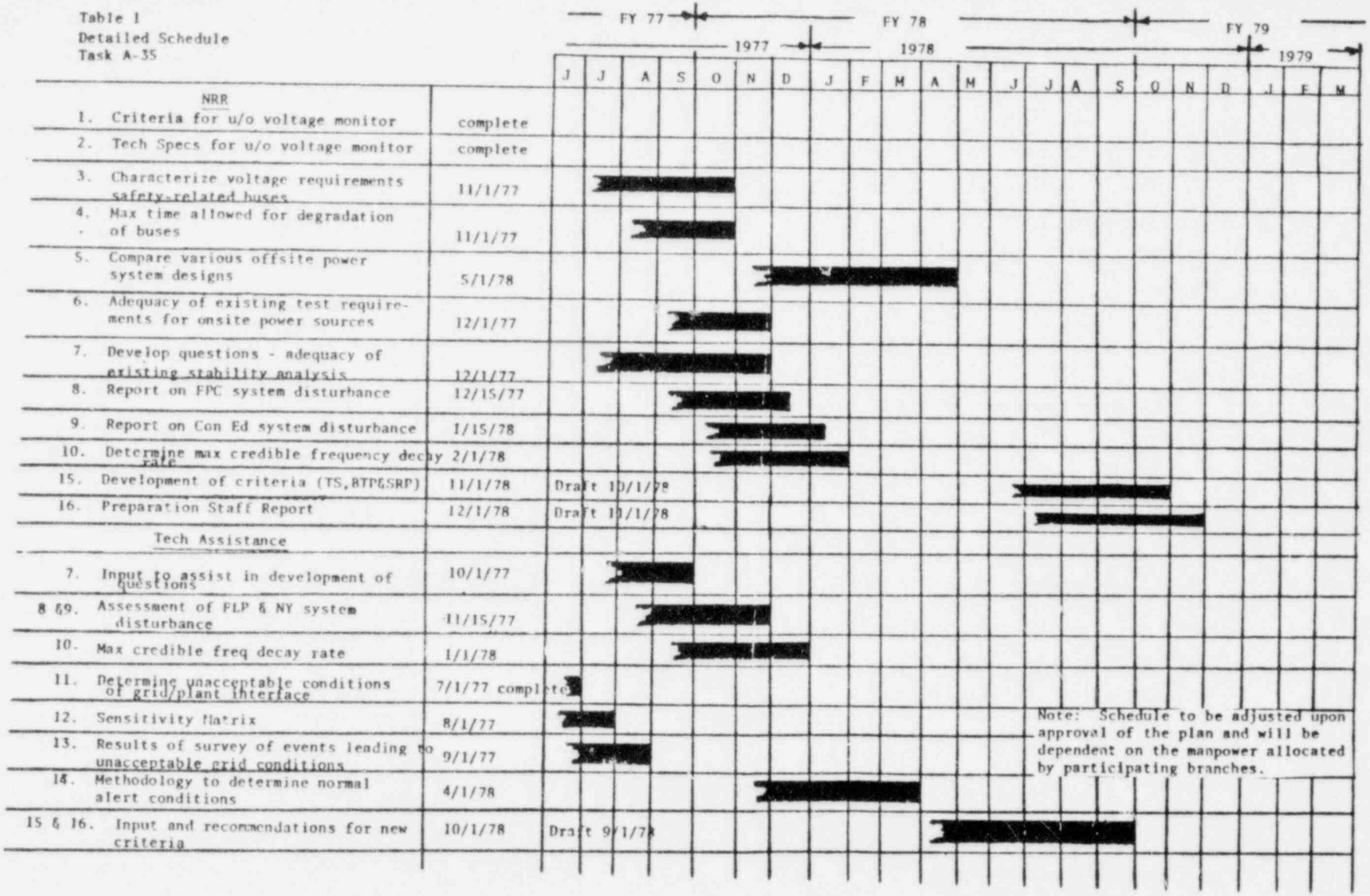
A bar chart and summary of manpower estimates, Tables 1 and 2, are attached.

D. The Technical Assignment Control Number is TAC 06289 (R52) and TAC 06015 (R38).

8. Potential Problems

A. The completion of this task is scheduled for December 1, 1978. This schedule is dependent on the availability of manpower in the participating branches, which will be required for its implementation.

Table I
Detailed Schedule
Task A-35



NRR		
1. Criteria for u/o voltage monitor	complete	
2. Tech Specs for u/o voltage monitor	complete	
3. Characterize voltage requirements safety-related buses	11/1/77	
4. Max time allowed for degradation of buses	11/1/77	
5. Compare various offsite power system designs	5/1/78	
6. Adequacy of existing test requirements for onsite power sources	12/1/77	
7. Develop questions - adequacy of existing stability analysis	12/1/77	
8. Report on FPC system disturbance	12/15/77	
9. Report on Con Ed system disturbance	1/15/78	
10. Determine max credible frequency decay rate	2/1/78	
15. Development of criteria (TS,BTP&SRP)	11/1/78	Draft 10/1/78
16. Preparation Staff Report	12/1/78	Draft 11/1/78
Tech Assistance		
7. Input to assist in development of questions	10/1/77	
8 &9. Assessment of FLP & NY system disturbance	11/15/77	
10. Max credible freq decay rate	1/1/78	
11. Determine unacceptable conditions of grid/plant interface	7/1/77 complete	
12. Sensitivity Matrix	8/1/77	
13. Results of survey of events leading to unacceptable grid conditions	9/1/77	
14. Methodology to determine normal alert conditions	4/1/78	
15 & 16. Input and recommendations for new criteria	10/1/78	Draft 9/1/78

Note: Schedule to be adjusted upon approval of the plan and will be dependent on the manpower allocated by participating branches.

Table 2
Manpower Estimates
Task A-35

NRR Tasks	Branch/Division			
	PSB/DOR	PSB/DSS		
1. Critieria for u/o voltage monitor	complete			
2. Tech Specs for u/o voltage monitor	complete			
3. Characterize voltage requirements at safety-related buses	0.04	0.03		
4. Max. time allowed for degradation on buses	0.04	0.03		
5. Compare various offsite power system designs	0.12	0.22		
6. Adequacy of existing test requirements for onsite power sources		0.10		
7. Develop questions - adequacy of existing stability analysis		0.06		
8. Report on FPL system disturbances	0.16			
9. Report on Con Ed system disturbances	0.16			
10. Determine max. credible frequency decay rate	0.06	0.12		
15. Development of criteria	0.26	0.24		
16. Preparation of staff report	0.26	0.20		
	1.10	1.00		total 2.10 manyears

NOTES:
1) 2000 manhours/manyear
2) Tasks 11 through 14 are Technical Assistance tasks and are not included NRR manpower estimates

OCT 5 1977

REVISION 0

CATEGORY A TECHNICAL ACTIVITY NO. A-36

TITLE: Control of Heavy Loads Near Spent Fuel

LEAD RESPONSIBILITY: Division of Operating Reactors

LEAD ASSISTANT DIRECTOR: Darrell G. Eisenhut, Assistant Director
for Operational Technology, DOR

TASK MANAGER: James A. Long, Program Support Branch, NRR

1. Problem Description

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of spent fuel in PWRs and BWRs. If a heavy object, e.g., a spent fuel shipping cask or a shielding block, were to fall or tip on to spent fuel in the storage pool or the reactor core during refueling and damage the fuel, there could be a release of radioactivity to the environment and a potential for radiation over-exposures to inplant personnel. If the dropped object is large, and the damaged fuel contained a large amount of undecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines. These concerns are currently considered in the licensing review. However, with the advent of increased and longer term storage of spent fuel assemblies in the spent fuel pools, there is a need to systematically review NRC requirements, facility designs and technical specifications regarding the movement of heavy loads to assess safety margins and to improve those margins where warranted.

2. Plan for Problem Resolution

The staff actions required to assess and if warranted, to improve existing safety margins are divided into three subtasks as follows:

Sub Task 1 - Evaluation of Current NRC Requirements and Available Licensee Procedures

A staff evaluation will be performed of existing NRC requirements, available licensee procedures and technical specifications associated with the movement of heavy loads on the refueling floors inside containment and near the spent fuel pool outside containment of operating reactor facilities. The review of existing NRC requirements will include an evaluation of the staff's acceptance criteria for overhead crane handling systems as delineated in Reg. Guide 1.104. A review of existing cask drop analyses will also be performed to determine the extent to which postulated cask tipping accidents have been considered.

APPROVED BY TASC, OCTOBER 19, 1977

The staff evaluation of available licensee procedures and technical specifications will include a review of the facility specific data provided by OI&E for six BWR's and six PWR's on the movement of heavy loads at those facilities. A determination will be made of the adequacy of the data currently available to evaluate the licensee procedures for heavy load movements. If it is determined that insufficient information exists, a generic letter will be written to all applicants and licensees requesting the data necessary to perform this evaluation. This letter, if required, will be coordinated with the generic letter currently planned by Plant Systems Branch to determine the degree to which licensees comply with the guidance in the final version of Regulatory Guide 1.104.

Based on the results of these evaluations, the adequacy of the measures currently in effect to protect the spent fuel in the storage pool or the fuel in the reactor during refueling will be determined.

Sub Task 2 - Accident Assessment

If the conclusions of Sub Task 1 indicate that existing NRC requirements and licensee procedures are inadequate, an evaluation will be performed of the probability and consequences of an accident wherein a spent fuel shipping cask tips into the storage pool and similarily for an accident wherein a heavy load is dropped or tips into the storage pool or the reactor core during refueling. The consequences to be considered will include the radiological releases due to ruptured fuel assemblies as well as potential for the creation of a critical configuration of fuel due to dropped loads and the potential for degrading the decay heat removal system capabilities. For the purpose of defining the scope of this evaluation it is assumed that the existing staff procedures for assessing fuel cask drop accidents are adequate. Upon completion of the evaluation, the probabilities and consequences will be combined to assess whether regulatory action is required and, if so, what action is appropriate.

Sub Task 3 - Documentation of Safety Criteria

Utilizing as a basis (1) the NRC requirements, licensee procedures or designs found adequate during the first subtask and (2) the regulatory actions found appropriate during the second subtask, prepare a revision to the SRP which will provide guidance to the staff and the industry on the criteria which must be satisfied to reduce to an acceptably low level the potential for heavy loads causing unacceptable damage to spent fuel in a storage pool or in the reactor core during refueling. The revised SRP will provide the basis for implementing additional requirements and procedures in existing plants where warranted and can be utilized in future reviews of new plants.

3. NRR Technical Organizations Involved

Overall project management for this task will remain with the Task Manager. Technical assistance will be provided during all three subtasks by EEB, PSB and EB of DOR, by AAB of DSE and by ASB of DSS. The activities performed under this task will be coordinated with those planned in Task Action Plan A-28, "Increase in Spent Fuel Pool Storage Capacity."

Sub Task 1 - Evaluation of Current NRC Requirements and Available Licensee Procedures

EEB, PSB, EB, AAB and ASB will be responsible for providing one representative to work with the Task Manager to evaluate the NRC requirements in effect regarding the movement of heavy loads at operating facilities and at facilities undergoing licensing review. After the review of available licensee procedures, this review group will prepare, if deemed necessary, the generic letter to all applicants and licensees requesting additional information on the movement of heavy loads at their facilities. The evaluation of the responses to such a generic letter would also be performed by this group.

Total estimated effort to complete this subtask is 24 man weeks with each technical branch representative contributing approximately 4 man weeks.

Sub Task 2 - Accident Assessment

Utilizing as a basis the results of the evaluations performed in Sub Task 1, EB, PSB, AAB and ASB will be responsible for determining the probability and extent of the spent fuel damage resulting from a heavy load falling or tipping into the storage pool or the reactor core during refueling. EEB and AAB will be responsible for performing an analysis of the potential radiological consequences both on and off site due to the accident parameters identified. Upon completion of these evaluations, the review group established in Sub Task 1 will be responsible for combining the probabilities and consequences to determine whether regulatory action is required and, if so, what action is appropriate.

Estimated effort required from each technical branch to perform the evaluations identified is 4 man weeks plus 2 additional man weeks to account for the review group representative. Total effort for this sub task including that of the task manager is 34 man weeks.

Sub Task 3 - Documentation of Safety Criteria

EEB, PSB, EB, AAB and ASB will be responsible for providing one representative to work with the Task Manager to determine how the regulatory actions found to be necessary and appropriate in Sub Tasks 1 and 2 can be incorporated into the existing SRP's. After this determination is made, the review group will prepare the SRP revision for management review and approval.

Total estimated effort to complete this sub task is 12 man weeks with each technical branch representative contributing approximately 2 man weeks.

4. Technical Assistance Requirements

None anticipated.

5. Interactions with Outside Organizations

If the generic letter to all licensees and applicants is found to be necessary, interactions could be considerable during the first sub task. If the generic letter is not necessary, it is anticipated that some minimal interaction may still be required with crane vendors, architect engineers, spent fuel pool rack manufacturers, as well as licensees. In either case, we expect this interaction will provide the detailed design data, accident frequency data, and operational procedures required for this task.

6. Assistance Requirements From Other NRC Offices

For Sub Task 1, I&E has provided a survey of the procedures in effect at twelve operating plants for the movement of heavy loads near spent fuel pools and the reactor during refueling. If the data currently available on licensee procedures for the movement of heavy loads is found to be inadequate, certain I&E assistance may be required to compile the additional information. During the second subtask, the Probabilistic Analysis Branch, RES, and the Applied Statistics Group, EDO, will be requested to provide assistance in managing and reviewing the probability assessment effort. The assistance required from each branch should not exceed one man week of effort.

7. a) Schedule for Problem Resolution (Generic Letter not Required)

Sub Task 1 - Evaluation of Current NRC Requirements and Available Licensee Procedures

. I&E inspection data received	August 26, 1977
. Compilation of existing inhouse data on licensee procedures completed	November 11, 1977
. Review group evaluation of the licensee procedures completed	November 30, 1977
. Review group evaluation of current NRC requirements completed	December 30, 1977
. Review group conclusions distributed to applicable technical branches and NRR management	January 6, 1978

Sub Task 2 - Accident Assessments*

. Accident probabilities and associated fuel damage defined	March 17, 1978
. Radiological consequences identified	April 28, 1978
. Appropriate regulatory actions identified	May 19, 1978

Sub Task 3 - Documentation of Safety Criteria

. Draft SRP revisions prepared	June 23, 1978
. Staff and management review completed	July 21, 1978
. Proposed SRP Revision submitted to RRRC	August 25, 1978

* Accident assessments performed under Sub Task 2 may be commenced prior to the completion of Sub Task 1 if the need and required input data have been identified.

7. b) Schedule for Problem Resolution (Generic Letter Required)

Sub Task 1 - Evaluation of Current NRC Requirements and Available Licensee Procedures

I&E inspection data received	August 26, 1977
Compilation of existing inhouse data on licensee procedures completed	November 11, 1977
Generic letter requesting data transmitted to all licensees/applicants	January 20, 1978
Licensee/applicants responses received	March 31, 1978
Review group evaluation of licensee procedures completed	April 21, 1978
Review group evaluation of current NRC requirements completed	May 19, 1978
Review group conclusions distributed to applicable technical branches and NRR management	May 26, 1978

Sub Task 2 - Accident Assessments*

Accident probability and associated fuel damage defined	August 4, 1978
Radiological consequences identified	September 15, 1978
Appropriate regulatory actions identified	October 13, 1978

Sub Task 3 - Documentation of Safety Criteria

Draft SRP revision prepared	November 10, 1978
Staff and management review completed	December 15, 1978
Proposed SRP Revision submitted to RRRC	January 19, 1979

* Accident assessments performed under Sub Task 2 may be commenced prior to the completion of Sub Task 1 if the need and required input data have been identified.

8. Potential Problems

It should be recognized that the results of this review, particularly Sub Task 2, are highly dependent on plant design characteristics and the specific procedures in effect at a particular plant. It is anticipated that similarities between facilities will justify the selection of two or three representative facilities which should provide sufficient information to bound the assessments being performed by the technical branches involved. If many plant specific assessmentsare necessary, completion of the second and thus the third subtask will be delayed.

REVISION 0
TASK ACTION PLAN SEPTEMBER 20, 1977
TASK NUMBER A-39

Title - Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment

Lead Responsibility - Division of Systems Safety/NRR

Lead Assistant Director - R. L. Tedesco (Plant Systems)

Task Manager: J. A. Kudrick (Containment Systems Branch)

1. Program Description:

Experience at several BWR plants with pressure suppression containments has shown that damage to wetwell internal structures can occur during safety/relief valve (SRV) blowdowns as a result of air clearing and steam quenching vibration phenomena.

Upon relief valve actuation, the initial air column within the SRV discharge line is accelerated by the high pressure steam flow and expands as it is released into the pool as a high pressure air bubble. The high rate of air and steam injection flow in the pool followed by expansion and contraction of the bubble as it rises to the pool surface produces pressure oscillations on the pool boundary. This effect is referred to as the air-clearing phenomenon.

In addition to the boundary loads, the air injection and subsequent bubble motion produces pressure waves and water movement within the pool that produce drag loads on components in the pool.

APPROVED BY TASC, SEPTEMBER 6, 1977
TASC COMMENTS INCORPORATED,
SEPTEMBER 20, 1977

Following the air-clearing phase, pure steam is injected into the pool. Condensation oscillations occur during this time period. However, the amplitudes of these vibrations are relatively small at low pool temperatures. Continued blowdown into the pool will increase the pool temperature until a threshold temperature is reached. At this point, steam condensation becomes unstable. Vibrations and forces can increase by a factor of 10 or more if the SRV continues to blow down. This effect is referred to as the steam quenching vibration phenomenon. Current practice for BWR operating plants is to restrict the allowable operating temperature envelope via technical specifications such that the threshold temperature is not reached.

In response to the concern on relief valve loads, letters were sent in 1975 to all licensees of operating BWR plants requesting that they report on the potential magnitude of relief valve loads, and on the structural capability of the suppression chamber and internal structures to tolerate such loads. In addition, consideration of these loads has become an integral part of our review of CP and OL plant applications for all BWR pressure suppression containments (i.e., Mark I, II and III). As a result of the generic concerns, owner's groups were formed by both Mark I and II utilities. Through these groups, integrated generic analytical and experimental programs have been developed to address the subject of SRV loads.

2. Plan for Problem Resolution:

A. Approach

The staff will review and evaluate the results from the Mark I and II programs conducted by the owner's groups and related programs conducted by General Electric Co. (GE).

The approach taken by the owner's groups consists of a number of comprehensive experimental and analytical programs to establish and justify the SRV-related pool dynamic loads for BWR Mark I and II designs. In addition, prototypical in-plant testing is proposed to confirm Mark III SRV loads

For both the air-clearing-induced loads and the drag loads on submerged structures, the Mark I and II programs are based on the development of analytical models which will be confirmed with test data. A series of experimental programs are underway to provide this data base for model verification. Because of differences between the Mark I, II and III designs, the composite program which will be reviewed by the staff consists of both programs common to all BWR designs and programs unique to particular SRV discharge line configurations.

With respect to drag loads on submerged structures for both SRV and LOCA events, a generic analytical model is under development by GE which will be used for all BWR designs. For loads induced by air clearing, separate analytical models are under development to describe the two different types of discharge nozzles of the relief valve discharge lines; a ramshead model and a quencher model. The ramshead is a "Tee" fitting, whereas the quencher is a multi-branch diffuser type of nozzle.

The ramshead model under development by GE is jointly sponsored by both the Mark I and Mark II owner's groups. In-plant tests at Monticello will provide the necessary confirming data base.

The basic quencher analytical model also under development by GE will be common to both Mark I and II programs. However, the confirming data bases are different. This is due to configurational differences in the SRV end device. In-plant tests to be conducted at the Caorso facility in Italy are proposed by the Mark II owner's group as the confirming data base, while, in-plant tests to be conducted at Monticello are proposed by the Mark I owner's group as the confirming data base.

The proposed program conducted by GE to address the elevated pool temperature concern for the ramshead device is based on experimental determination of the threshold temperature. Current technical specifications for operating Mark I plants restricting plant operation below this limit would be sufficient to satisfy this concern. GE plans to document these additional data to support the current temperature limit in the near future for staff review.

B. End Products

The program as outlined consists of four major tasks, described below. Upon completion of each task, a NUREG report will be issued. In some cases, this may take the form of input into a more general report (e.g., input into the overall Mark II NUREG report prepared as part of Task A-8). Each NUREG report will be generic in nature outlining the acceptable methodology to be used for computation of plant specific loads.

In addition to the final report, interim acceptance criteria may be necessary to properly interface with both the Mark I and Mark II generic programs. Reports will be issued to the appropriate task manager if such action is necessary. The enclosed detailed schedule indicates those areas where such an intermediate report may be required. The actual need will be determined when more definite schedules are established on the individual programs.

As part of the SRV program, revisions as required to the Standard Review Plan will be prepared to properly reflect the program results.

C. Tasks

1. Evaluation of the ramshead air clearing load methodology - This task involves the review and evaluation of the analytical model and the supporting data base. Upon completion of the review, an acceptable methodology for computation of design bases loads associated with air clearing will be developed.

a. Evaluation of analytical model - the GE developed analytical model will be reviewed by the staff from both a theoretical and experimental viewpoint. The model will be evaluated for analytical completeness and experimental comparisons made considering the data base from both Monticello and Quad Cities in-plant tests. The actual experimental comparisons will be provided to the staff in topical reports supplied by GE.

b. Evaluation of test data - Evaluation of the Monticello test data, to be supplied by GE in a topical report, will be performed by the staff within this subtask. Areas of consideration will include;

- data scatter
- error band determination
- degree of variations of principal parameters
- fluid structure interaction effects on measured loads
- applicability of test data to plant specific conditions (i.e., applicability to other Mark I designs as well as Mark II designs).

Results of this investigation will be incorporated in the model-data comparisons evaluation conducted in Task 1.a.

c. Develop air clearing load methodology

Based on the results of tasks 1.a and 1.b, load acceptance criteria will be developed by the staff for ramshead air clearing induced loads for both Mark I and Mark II designs.

2. Evaluations of the Quencher Air Clearing Load Methodology - Evaluation and review by the staff of the analytical model with the supporting data base will be performed in this task. Currently, the various industry programs indicate that the quencher arm configuration will differ between Mark I and II designs. However, the bubble pattern associated with each arm will be the same. Therefore, it is assumed that the analytical model will remain essentially the same for both the Mark I and II designs. Upon completion of the staff's review, an acceptable methodology for computation of design basis loads will be determined. It should be noted that as part of the overall testing program, prototypical in-plant testing is planned for the Mark III quencher. This program is considered as confirmatory. The staff effort for review of this program is included in this task but will not impact on the development of the load acceptance criteria since it is confirmatory in nature.

a. Evaluation of Analytical Model -

The analytical model will be reviewed by the staff both from an analytical and empirical viewpoint. Model-to-data comparisons performed and reported by GE will form the basis of the staff's review, since the basic approach is anticipated to be similar to the methodology used in the ramshead model (see Task 1.a).

b. Evaluation of Caorso* Test Data

Caorso test data will be reviewed and evaluated by the staff to determine the adequacy of the data base for confirmation of the analytical model (Task 2.a). These data will be supplied to the staff by GE in the form of a topical report. Areas of consideration will include:

- Data scatter
- Error band determination
- Degree of variation of principal parameters
- Fluid structure interaction effects on measured loads
- Applicability of test data to Mark II designs.

Results of this task will be incorporated into task 2.a.

c. Evaluation of Mark I related test data -

The staff will review and evaluate two separate test programs; a small scale test program recently completed to determine relative performance between various quencher designs and an in-plant test program to be conducted at the Monticello plant. The results of these programs will be documented by GE in the form of topical reports. Similar considerations as outlined in task 2.b will be included in this task.

The results of this task will be integrated into Task 2.a.

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*/ Caorso is a Mark II plant located in Northern Italy.

d. Develop Air Clearing Load Methodology -
Based on the results of tasks 2.a, b and c, load acceptance criteria will be developed by the staff for quencher air clearing loads for both Mark I and II designs.

e. Evaluate Confirming Mark III In-Plant Test Program and Data -
The staff will review and evaluate the test plans, instrumentation and data of the prototypical in-plant test program. This information will be supplied to the staff by GE in a topical report. Similar considerations as delineated in task 2.b will be included.

3. Evaluation of Submerged Structure Load Methodology -
This task involves the staff's review and evaluation of a generic analytical model to be developed by GE to compute the loads on submerged structures due to SRV actuation and LOCA. A portion of the review will involve the evaluation of supporting test data to be supplied to the staff in a topical report. Acceptable load criteria will be developed by the staff as a result of this effort.

a. Evaluation of Analytical Model -
The staff will review and evaluate the generic model developed by GE to compute induced loads on components located within the suppression pool. Particular attention will be directed toward the analytical considerations of the following:

- Development of transient flow fields
- Presence of components within the flow field affecting the field
- Supporting experimental data
- Applicability to LOCA induced loads

b. Evaluation of Supporting Data Base -

The staff will review and evaluate the applicability of the data provided by GE for confirmation of the analytical program. It is anticipated that the data base will consist of experimentally derived drag coefficients, recent data obtained from the 1/3 scale pressure suppression test facility tests and possible future tests which will be documented as part of the Mark I and II owner's group programs.

c. Develop Submerged Structure Load Methodology -

Based on the results of tasks 3.a and b, load acceptance criteria will be developed by the staff. These criteria will be applicable for all BWR designs.

4. Determination of LOCA and ATWS Pool Temperature Limits -

This task involves the staff's review and evaluation of GE-supplied supporting test data to confirm established design pool temperature limits for both LOCA and ATWS considerations. Presently, GE has proposed a higher design pool temperature limit for the ATWS event, taking into account the low probability of occurrance. The adequacy of this reduced safety margin as well as the proposed pool temperature limit for the design basis LOCA will be reevaluated within this task. Although the primary emphasis will be directed towards the ramshead device, the limits for the quencher device will also be included. In addition, minimum pool temperature monitoring requirements will be determined by the staff. Upon completion of this task, a final report will be issued by the staff summarizing our review and evaluation.

a. Evaluate Supporting Data Base -

The staff will evaluate the adequacy of the data base to be provided by GE in the form of a letter report from operating experience, Moss Landing tests and tests conducted at General Electric's San Jose facility as well as GE's licensee data (NEDE-21078). Based on the staff's review, the currently recommended pool temperature limits will be reevaluated for the ramshead device. A similar review will be conducted for the Mark I quencher device.

b. Evaluate Thermal Mixing Model -

The staff will review and evaluate the thermal mixing model with its supporting data base to be provided by GE. Based on results of this review, pool temperature limits will be reevaluated and minimum temperature monitoring requirements will be established.

3. NRR Technical Organizations Involved

A. Containment Systems Branch, Division of Systems Safety

1. Task 1

Has overall responsibility for establishing an acceptable methodology to calculate ramshead air clearing loads.

2. Task 1a

Review and evaluate the analytical model.

3. Task 1b

and evaluate the Monticello data excluding fluid structure interaction effects (FSI) and evaluate applicability of data to Mark II.

4. Task 1c

A generic NUREG report will be issued summarizing the acceptance criteria for the ramshead load.

Manpower Requirements -

FY 77 - .05 man-years

FY 78 - .5 Man-years

FY 79 - .1 Man-years

Total - .70 Man-years

5. Task 2

Has overall responsibility for establishing an acceptable methodology to compute quencher air clearing loads.

6. Task 2a

Review and evaluate the analytical model.

7. Task 2b

Review and evaluate the Caorso test plan and data (excluding FSI effects).

8. Task 2c

Review and evaluate the Mark I small scale tests and the Monticello in-plant tests (excluding FSI effects) and,

9. Task 2d

Generic NUREG reports will be issued for both Mark I and Mark II designs.

10. Task 2e Review

Evaluate the Mark III confirmatory test plan and data (this effort will be part of a topical report evaluation).

Manpower Requirements -

FY 77 - 0 Man-years

FY 78 - .5 Man-years

FY 79 - .4 Man-years

Total - 0.9 Man-years

11. Task 3

Has total responsibility for establishing an acceptable methodology to compute submerged structure drag loads due to SRV actuation and LOCA.

12. Task 3a

Review and evaluate the analytical model.

13. Task 3b

Review and evaluate the supporting data.

14. Task 3c

A generic NUREG report will be issued for all BWR designs.

Manpower Requirements -

FY 77 - .05 Man-years

FY 78 - .25 Man-years

FY 79 - .10 Man-years

Total - .40 Man-years

15. Task 4, 4a, 4b

Has total responsibility for the review and evaluation of supporting information supplied by GE to confirm the current pool temperature limits for both ramshead and Mark I load mitigating devices. Input will be provided for the ATWS evaluation report. A generic NUREG report will be issued summarizing the minimum pool temperature monitoring requirements and the acceptable temperature limits for SRV devices. This report will in large part be based on the review of the GE thermal mixing model.

Manpower Requirements -

FY 77 - .02 Man-years

FY 78 - .23 Man-years

Total - .25 Man-years

B. Plant Systems Branch, Division of Operating Reactors

1. Task 1 through 4 - Follow the progress of the SRV Program to insure correct application of generic resolutions to specific plant applications.

2. Manpower Requirements -

FY 77 - .1 Man-years
FY 78 - .2 Man-years
FY 79 - .1 Man-years
Total - .4 Man-years

C. Engineering Branch, Division of Operating Reactors

1. Task 1b

Has responsibility for determining the fluid structure interaction effects (FSI) associated with the Monticello tests. If FSI effects are significant, methods will be developed by which the pure forcing function can be obtained. A report will be issued to the Task Manager summarizing the results of this task.

2. Task 2c

Has responsibility for determining the fluid structure interaction effects associated with the Monticello in-plant load mitigating tests. If FSI effects are significant, methods will be developed by which the pure forcing function can be obtained. A report will be provided to the Task Manager summarizing the results of this task. (Due to the similarity of this task with SEB s task associated with the Caorso test FSI evaluation, coordination between these efforts will be needed).

Manpower Requirements -

FY 77 - .04 Man-years

FY 78 - .6 Man-years

FY 79 - .3 Man-years

Total - .94 Man-years

D. Structural Engineering Branch, Division of Systems Safety

1. Task 2b

Has responsibility for determining the FSI effects associated with the Caorso test series. If the FSI effects are significant, methods will be developed by which the pure forcing function can be obtained. A report will be issued to the Task Manager summarizing the task results. (Coordination with EB will be made with respect to the FSI investigation of Monticello tests).

Manpower Requirements -

FY 77 - .1 Man-years

FY 78 - .3 Man-years

FY 79 - .2 Man-years

Total - .6 Man-years

E. Division of Project Management

1. Tasks No. 1 through 4

Provide coordination between the Division of Systems Safety, the Mark I and Mark II licensees/applicants, and the Division of Project Management project managers for the individual Mark I, II and III BWR facilities. This includes meeting coordination and preparation of meeting minutes to document the actions of the generic SRV review when the owners are involved.

2. Manpower Requirements -

FY 1978 - .1 Man-years

FY 1979 - .1 Man-years

Total - .2 Man-years

4. <u>Technical Assistance Requirements</u>

A. Brookhaven National Laboratory

1. Title: BWR Pool Dynamic Technical Assistance Program

2. Responsible Division/Branch: Division of Systems Safety/ Containment Systems Branch

3. Scope

The contractor is to provide technical expertise in the evaluation of all analytical models provided for review in all four major tasks. (Tasks 1a, 2a, 3a, 4b). In addition, he will provide an independent assessment of the available test data. (Tasks 1b, 2b, 2c, 2e, 3b, 4a). Upon the completion of each specific model or test review, a letter report will be issued to the staff for each of the above noted task items. During the course of the review, requests for additional information will also be issued as required.

4. Funding: FY 1977 - $60,000

   FY 1978 - $60,000 (requested)

   FY 1979 - $15,000 (estimated)

   Total - $135,000

B. Lawrence Livermore Laboratory

1. Title: Structural Hydrodynamic Interactions Technical Assistance Programs

2. Responsible Division/Branch: Division of Operating Reactors/ Engineering Branch.

3. Scope

   This is a program to study hydrodynamic/structure interactions in a Mark I containment system subject to hydrodynamic loading conditions. This effort should quantify the amplification, if any, of measured loads due to the structural interactions during pool swell, SRV discharge, and chugging loading conditions. This is a common technical assistance program for Mark I, Mark II and the SRV task action plans.

4. Funding: FY 1977 - 100K (NOTE: This funding represents the total program which are reflected also in Task A-7).
FY 1978 - 15K

5. Interactions with Outside Organizations:

Mark I and Mark II Owner's Groups

These groups are "ad hoc" organizations of utilities owning either Mark I or Mark II BWR facilities. They have engaged GE as their program manager for resolution of the BWR containment concerns and have designated GE as their primary contact with the NRC during the conduct of these programs.

Advisory Committee on Reactor Safeguards (ACRS)

This task is closely related to one of the generic items identified by the ACRS and, accordingly, will be coordinated with the committee as the task progresses.

6. Assistance Requirements from Other NRC Offices:

Requirements for assistance are not anticipated at this time.

7. Schedule for Problem Resolution

1.1 Interim Ramshead Load Criteria	2/78
1.2 Report of FSI Effects	8/78
1.3 SER for Ramshead	11/78
2.1 Report of FSI Effects	9/78
2.2 SER for Quencher	2/79
3.1 Interim Submerged Structure Load Criteria	2/78
3.2 Final Submerged Structure Load Criteria	3/79
4.1 Reevaluation of Ramshead Pool Temperature Limits	2/78
4.2 Final Criteria for Pool Temperature Limits	6/78
5.0 Issue Revisions to Standard Review Plan	6/79

B. Detailed Schedule

Bar chart enclosed

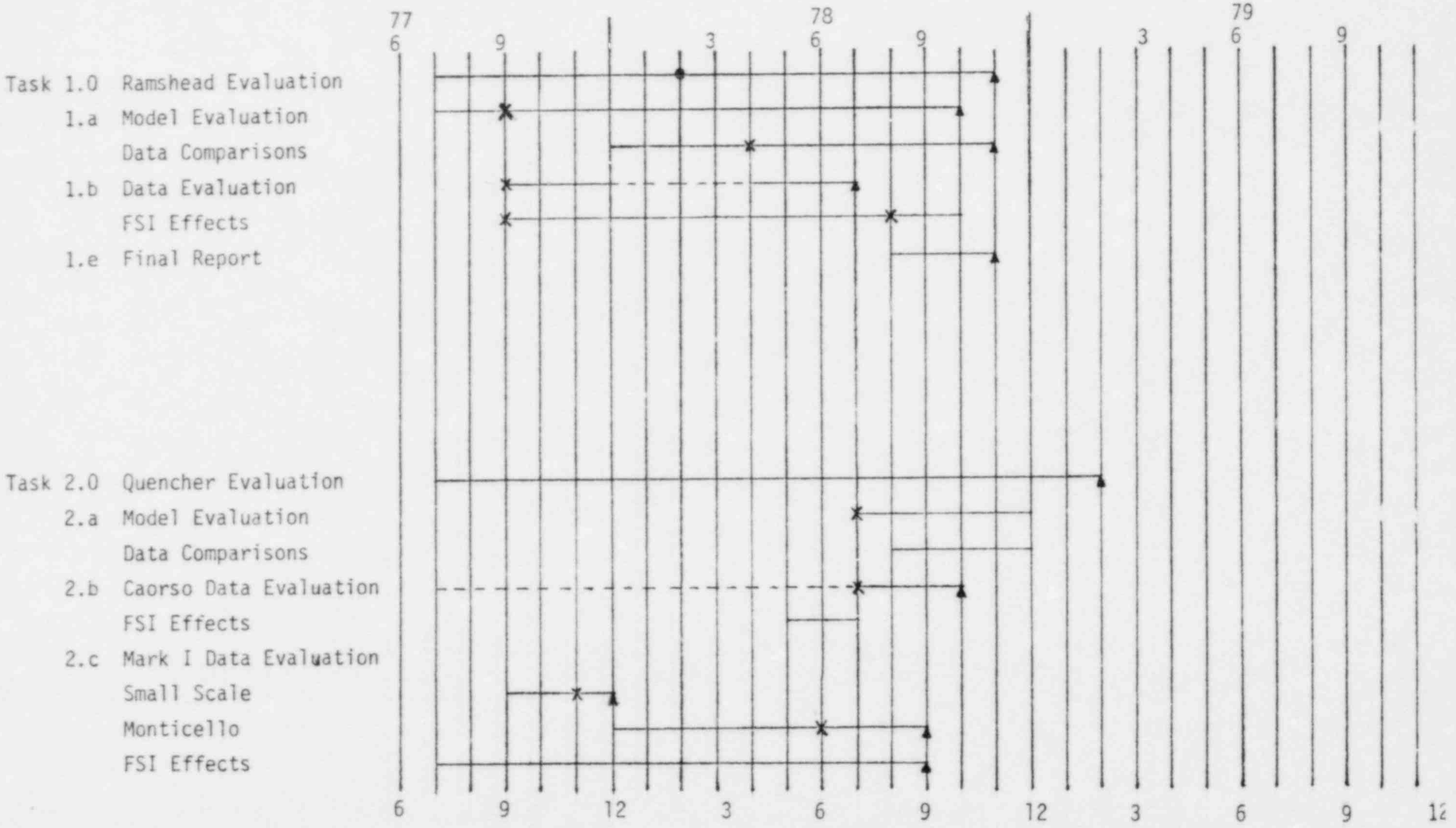
C. Technical Assignment Control Number - TAC 4671.

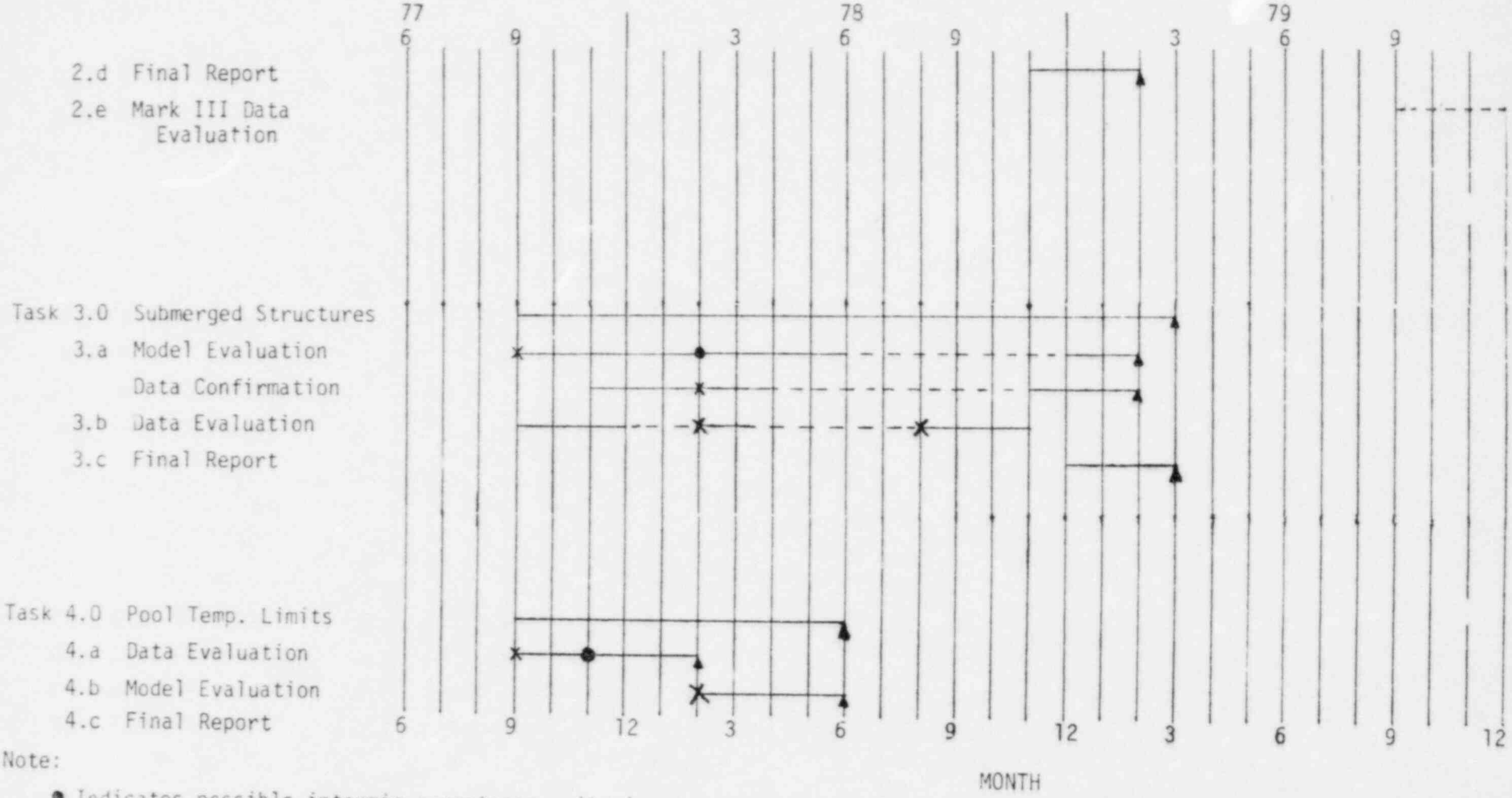
8. <u>Potential Problems</u>

A. The proposed schedules have been based to a large part on the current estimates of receipt of key documents from both the Mark I and Mark II owner's programs. Since there are several test programs involved, past performance would indicate a good possibility in schedule slippages in one or two tasks. This may necessitate additional in-plant testing on lead Mark II plants prior to completion of the SRV generic program.

B. Fluid structure interaction effects are an important consideration in the evaluation of both ramshead and quencher test data. A technical assistance program has been initiated for Mark I related tasks. However, efforts to develop a similar program for Mark II considerations have just begun. Early initiation of this program or incorporation into the existing program is required if successful completion of task 2 is to be realized.

SRV PROGRAM SCHEDULE





Note:

- ● Indicates possible intermin acceptance criteria
- X Indicates receipt of key documentation from either the Mark I or Mark II owner's programs or GE

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