UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 01 1977

MEMORANDUM FOR: R. Boyd, Director, DPM
H. Denton, Director, DSE
R. Mattson, Director, DSS
V. Stello, Director, DOR

FROM: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

SUBJECT: REVIEW OF DRAFT STATUS SUMMARY REPORT FOR NRR GENERIC TECHNICAL ACTIVITIES

Attached is a draft of the Status Summary Report for NRR Generic Technical Activities (the so-called Rainbow Book). This draft is being distributed for your comment.

Your comments should be restricted at this time to comments on the format and type of information presented in the book. Suggestions for modifications or additions should be provided to Mike Aycock by COB, December 16, 1977.

I am not requesting comments on the specific task schedules in the book at this time. As agreed to by the Technical Activities Steering Committee, branches participating in the tasks will get an opportunity to evaluate the impact of the combined manpower demands for these tasks before finalizing schedules. This will be done by providing each participating branch with Manpower Loading Reports reflecting upcoming milestones for the majority of the Category A tasks plus upcoming milestones for casework and other activities. Review of these Manpower Loading Reports should identify impacted branches and allow decisions to be made regarding the need for schedule adjustments. We expect to have Manpower Loading Reports available the week of December 12, 1977 following a further exchange of information between the Task Managers and OMIPC.

By copy of this memorandum, a copy of the draft schedule book is being provided to the Deputy Directors, Assistant Directors and Task Managers in your divisions.

Edson G. Case, Acting Director
Office of Nuclear Reactor
Regulation

Enclosure:
Draft Status Summary Report for
NRR Generic Technical Activities

cc: NRR Deputy Directors
NRR Assistant Directors
NRR Task Managers for
Category A Tasks
Advisory Group

DRAFT
TSu/mc
12/22/77

Gus - I want AB added to this since they have gone you ok on transient analysis on valve actuation for Part 21 item

39

MEMORANDUM FOR: R. Mattson, Director, Division of Systems Safety, ONRR

THRU: R. Tedesco, Assistant Director for Plant Systems, DSS

FROM: T. Su, Containment Systems Branch, DSS

SUBJECT: PROPOSED REVISIONS TO TASK ACTION PLAN A-39, "DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT"

Enclosed is our proposed Revision 1 to the Task Action Plan A-39. Included in the revision are additional tasks and changes of schedules due to slippage of tests and delays in the applicant's submittal of several key documents. All changes are indicated by lines shown in right margin. ~~Changes of~~ These ~~revised~~ tasks, however, face a severe shortage of manpower both in-house and from outside technical assistance sources.

We have revised ~~included in~~ A-39 to include additional ~~the~~ efforts for the resolution of GE Part 21 notification concerning multiple SRV's consecutive actuation ~~and associated load increase for all BWR containments~~. We have also included ~~the efforts~~ additional manpower required for the review and evaluation of the new Mark ~~II specific~~ quencher device ~~which was~~ being proposed for Mk II plant applications. ~~proposed recently by the applicant~~. This quencher device is being developed and will be tested in Germany by KWU.

We expect both of these efforts to be ~~substantiate.~~ substantial. In particular, the evaluation of the new Mark II quencher device is estimated to require ~~six months for~~ one-half to one-man year of outside technical assistance. ~~We will follow the normal procedure for that requested.~~ We are preparing a request for the additional technical assistance.

D-85

During the course of developing the revised TAP, we requested comments from all involved branches. ~~As a result~~, The Mechanical Engineering Branch indicated that they could not provide manpower for ~~the~~ th tasks during FY 1978 ~~and~~ in FY 1979, and the Reactor Systems Branch ~~said~~ stated that they have no manpower available ~~for~~ thru FY 1978; s~~ince their current projected manpower requirements for other Category A tasks does not allow them to commit this additional work.~~ The AID for PIS has contacted the AIDS for RS and E on this matter and is awaiting their response.

In light of this manpower unavailability we can anticipate the schedule to be slipped for several tasks. Specifically ~~such as concerns of~~ Part 21 notification, (2) suppression ~~Mark II quencher~~ pool temperature limits and (3) the ... SRV loads on piping and components. All of these tasks are closely related to the safety evaluation of licensing BWR plants. Unless adequate manpower is provided, we expect the schedule for plant applying operating license will be slipped from six months to one year. The lead OL plant is the Zimmer plant which has a projected fuel loading date of October, 1978.

We believe that the schedule of plant case work should be maintained. Therefore, we suggest that a priority should be granted for this task, ~~so we could get this task~~ moving ~~to meet the licensing schedule~~.

T. M. Su
A-39 Task Manager
Containment Systems Branch

cc: Carl Rasso / Knight ...

Revision 1
~~November 14, 1977~~
~~[illegible]~~
[illegible]

TASK ACTION PLAN
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: T. M. Su (Containment Systems Branch) (1)

1. Program Description:

BWR plants are equipped with relief valves that discharge into the wetwell. 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.

Experience at several BWR plants with pressure suppression containments has shown that damage to certain wetwell internal structures can occur during safety/relief valve (SRV) blowdowns as a result of air clearing and steam quenching vibration phenomena.

In addition to the boundary loads; e.g., containment structures, reactor pedestal, 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.

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.

Recently, GE issued a Part 21 notification related to consecutive actuation of multiple safety/relief valves and concomitant load increases for BWR Mark III water pressure-suppression containments. This concern resulted from a recent study performed by GE of the primary system pressure response following an isolation event. The results showed that more than one safety relief valve could be actuated consecutively, as a result of a reactor isolation event. This SRV load combination has not been considered in the design. (1)

Discussions with GE have also revealed that this concern is generic to all BWR containments and really concerns itself with the GE code for transient analysis. AB is reviewing the code. This will have to be completed before any action can be taken. It also affects operating plants.

As a result of this finding, GE has developed a program for resolution specifically for Mark III. However, they believe this resolution can be applied generically to all BWR systems.

in a timely matter

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.

Plans to use a quencher device rather than a ramshead is being proposed for Mark II type plants. The type of quencher, however, will be different from that currently proposed for Mark III, ~~and some Mark II containments~~. The quencher design is now being developed and will be tested in Germany by KWU. ~~We have had discussions with the applicant and believe that the staff's effort for the review and evaluation of the test program, test data and methodology of predicting the quencher loads will be substantial.~~

With regard to the concern of multiple SRV's subsequent actuations, GE is currently proposing several alternatives which are aimed to (1)
eliminate the possibility of actuating more than one SRV consecutively following a most severe isolation event. These alternatives, however, have been discussed exclusively for the Mark III containments. As GE indicated, some of these alternatives could be also applicable for Mark I and II containments. GE will submit detailed descriptions and justifications for the approach selected from these alternatives which is considered to meet all design requirements. We will review and evaluate the information and determine its applicability for the Mark III containments.

For Mark I containments, we have established a short term and long term program for resolution of this particular concern. For the short term, efforts of review of the justification for continued operation are included in the task action plan, "Mark I Containment Long Term Program." The long term program for resolution of this concern, however, is included in this task action plan. We will review and evaluate the approach and justification supplied by GE and the Mark I owners group for plants in operation and plants which have not yet been licensed for operation. With respect to Mark II containment, we have requested each Mark II applicants to provide justification for their plant design to meet the load requirements for multiple SRVs subsequent actuation. (1

The proposed program conducted by GE to address the elevated pool temperature concern for the ramshead device is based on the 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 documen 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 Loads Criteria for Ramshead - (1)
   This task involves the review and evaluation of the analytical model and the supporting data base. Upon completion of the review, acceptance criteria will be established for ramshead loads on containment structures and components.

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. Establish Acceptance Criteria for Ramshead Load.

Based on the results of tasks 1.a and 1.b, and task 5, we will establish acceptance criteria for ramshead loads on containment structures and components. Included in the criteria are the following:

(1) Loads for single as well as multiple SRV's first actuation; (1)

(2) Loads for single as well as multiple SRV's, if any, for consecutive actuations;

(3) Development of SRV design load cases, which will include all representative SRV operational modes;

(4) Frequency and fatigue cycles for 40 years of plant life.

2. Evaluation of Loads Criteria for Quencher

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,

acceptance criteria for quencher loads will be (1)
established. 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.

*/ Caorso is a Mark II plant located in Northern Italy.

In addition, a 1/4 scale test program is being planned to perform sensitivity study of the [illegible] parameters for [illegible] loads.

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.

d. Establish Acceptance Criteria for SRV Load

Based on the results of tasks 2.a, b, c and f, and task 5, we will establish acceptance criteria for quencher loads on containment structures and components. Included in the criteria are the following:

(1) Loads for single as well as multiple SRVs first actuation; (1)

(2) Loads for single as well as multiple SRVs, if any, consecutive actuations;

(3) Load cases, which will include all representative SRV operational modes;

(4) Frequency and fatigue cycles for 40 years plant life.

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.

f. Evaluation of Mark II Quencher Design
The staff will review and evaluate the test programs, test data and methodology of predicting the quencher loads. The test data and methods of calculating the SRV loads will be supplied to the staff by the applicant as part of the Susquehanna licensing docket. This is the first plant to reference this design. Areas of review will include: (1)

- Data scatter
- Error band determination
- Fluid structure interaction effects on measured loads
- Adequacy of analytical methodology
- Applicability of test data to the real plant design
- Comparison of the Susquehanna analytical method with the generic quencher analytical method.

Results of this task will be incorporated into task 2.a. However, a report on the result of our evaluation and acceptance criteria will be issued as part of Susquehanna safety evaluation.

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 evalute 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 Normal Plant Transient and ATWS Pool Temperature Limits - (1
This task involves the staff's review and evaluation of GE-supplied supporting test data to confirm established design pool temperature limits for both normal plant transient and ATWS considerations. (1
Presently, GE has proposed a higher design pool temperature limit for the ATWS event, taking into account the low probability of occurrence. The adequacy of this reduced safety margin as well as the proposed pool temperature limit for the normal plant transient will be reevaluated (1)
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.

5. Evaluation of method for determining number of SRV operating consecutively.

Evaluation of this task will include BWR 4, 5 and 6. Areas of review include:

- Primary system pressure response to plant normal and abnormal transients which will result in primary system blowdown through the SRV's,
- Evaluation of SRV control logic,
- SRV operational sequence

Results of this task will be incorporated into task 1.c and 2.d. for establishing SRV load cases and load combination. A report of our evaluation will also be issued for this particular concern.

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

Evaluate the Monticello data excluding fluid structure interaction effects (FSI) and evaluate applicability of data (1
to Mark II.

4. Task 1c

A generic NUREG report will be issued summarizing the acceptance criteria for the ramshead load, and load cases (1
and load combination.

Manpower Requirements -

FY 77 - .06 Man-years

FY 78 - .8 Man-years

FY 79 - .1 Man-years

Total - 1.0 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 models.

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 1/4 Scale T-Quencher tests.

9. Task 2d

   Generic NUREG reports will be issued for the quencher load, and loads and load combination.

10. Task 2e

    Evaluate the Mark III confirmatory test plan and data (this effort will be part of a topical report evaluation).

10.1 Task 2f

Has overall responsibility for establishing acceptance criteria for SRV load cases and SRV load combinations. (1)

Manpower Requirements -

FY 77 - .1 Man-years

FY 78 - .7 Man-years

FY 79 - .4 Man-years (1)

Total - 1.2 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

16. Task 5

Has overall responsibility for integrating and coordinating the offers from several branches involved. Upon completion of the task, a report will be issued summarizing the result of our evaluation. Result of this task evaluation will be integrated into task 1C and 2d for establishing the SRV load cases. (1)

Manpower Requirements -

FY 78 - .1 man-years

Total - .1 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 appropriate 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 appropriate 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-year

   FY 1979 - .1 Man-year

   Total - .2 Man-years

F. Reactor Systems Branch, Division of Systems Safety

1. <u>Task 5</u>

   Has responsibility for reviewing and evaluating GE's analyses for primary systems pressure response and the sequence of SRV actuation. A report will be issued to the Task Manager summarizing the task results.

2. <u>Task 4, 4a, 4b</u>

   Has responsibility for reviewing and evaluating the assumptions which are related to the primary systems response and used for the evaluation of pool temperature limits. A report will be issued to the Task Manager summarizing the task results. (1)

   Manpower Requirements -

   FY 78 - .2 Man-years

   Total - .20 Man-years

G. Instrumentation and Control Systems, Division of Systems Safety

1. Task 5

Has responsibility for reviewing and evaluating the control logic for SRV actuation. A report will be issued to the Task Manager summarizing the task results.

Manpower Requirements -

FY 78 - .3 Man-years

Total - .30 Man-years

H. Mechanical Engineering Branch, Division of Systems Safety

1. Task 1a, 1b, 2a, 2b, 2c and 2f

Has responsibility for reviewing and evaluating test data and analytical method relating to the SRV loads on SRV line, discharge device supports and components inside the containment. A report will be issued to the Task Manager summarizing the task results. (1)

Manpower Requirements -

FY 1978 - .2 Man-years

FY 1979 - .1 Man-years

Total - .3 Man-years

The review of vendor analyses is not an AB function. This should be performed by RSB. AB should be dropped.

I. Analysis Branch, Division of Systems Safety

1. Task 5

Has responsibility for reviewing and evaluating GE's analyses methods for primary ~~systems~~ system pressure response to isolation events. A report will be issued to the Task Manager summarizing the task results.

Manpower Requirements -

FY 1978 - .1 Man-years

Total - .1 Man-years

4. Technical Assistance Requirements

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, 1e, 2b, 2c, 2d, 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.

The contractor is also required to provide technical expertise in the evaluation of task 2.f (Evaluation of Susquehanna Test Data). Upon the completion of the evaluation of the task, the contractor will issue a letter report summarizing the result. It should be noted that this task is not included in the current contract. Funding for this task will be requested.

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 is 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	6/78	
1.2 Report of FSI Effects	8/78	
1.3 SER for Ramshead	11/78	
2.1 Report of FSI Effects	3/79	
2.2 SER for Quencher	7/79	(1)
3.1 Interim Submerged Structure Load Criteria	6/78	
3.2 Final Submerged Structure Load Criteria	6/79	
4.1 Reevaluation of Ramshead Pool Temperature Limits	4/78	
4.2 Final Criteria for Pool Temperature Limits	8/78	(1)
5.0 Report of SRV consecutive actuation	9/78	
6.0 Issue Revisions to Standard Review Plan	10/79	

B. Detailed Schedule

Bar chart enclosed

C. Technical Assignment Control Number - TAC 4671.

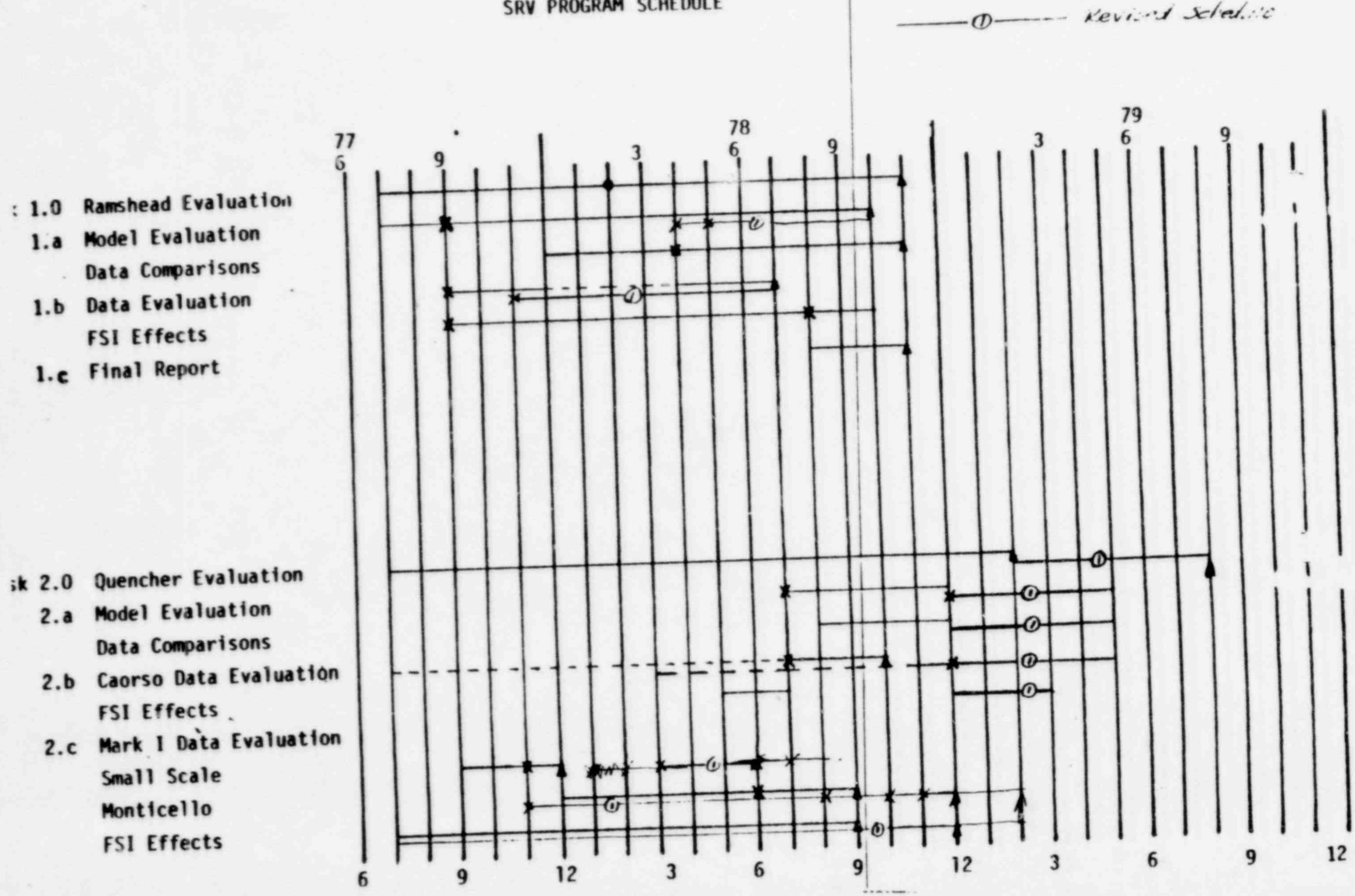
8. Potential Problems

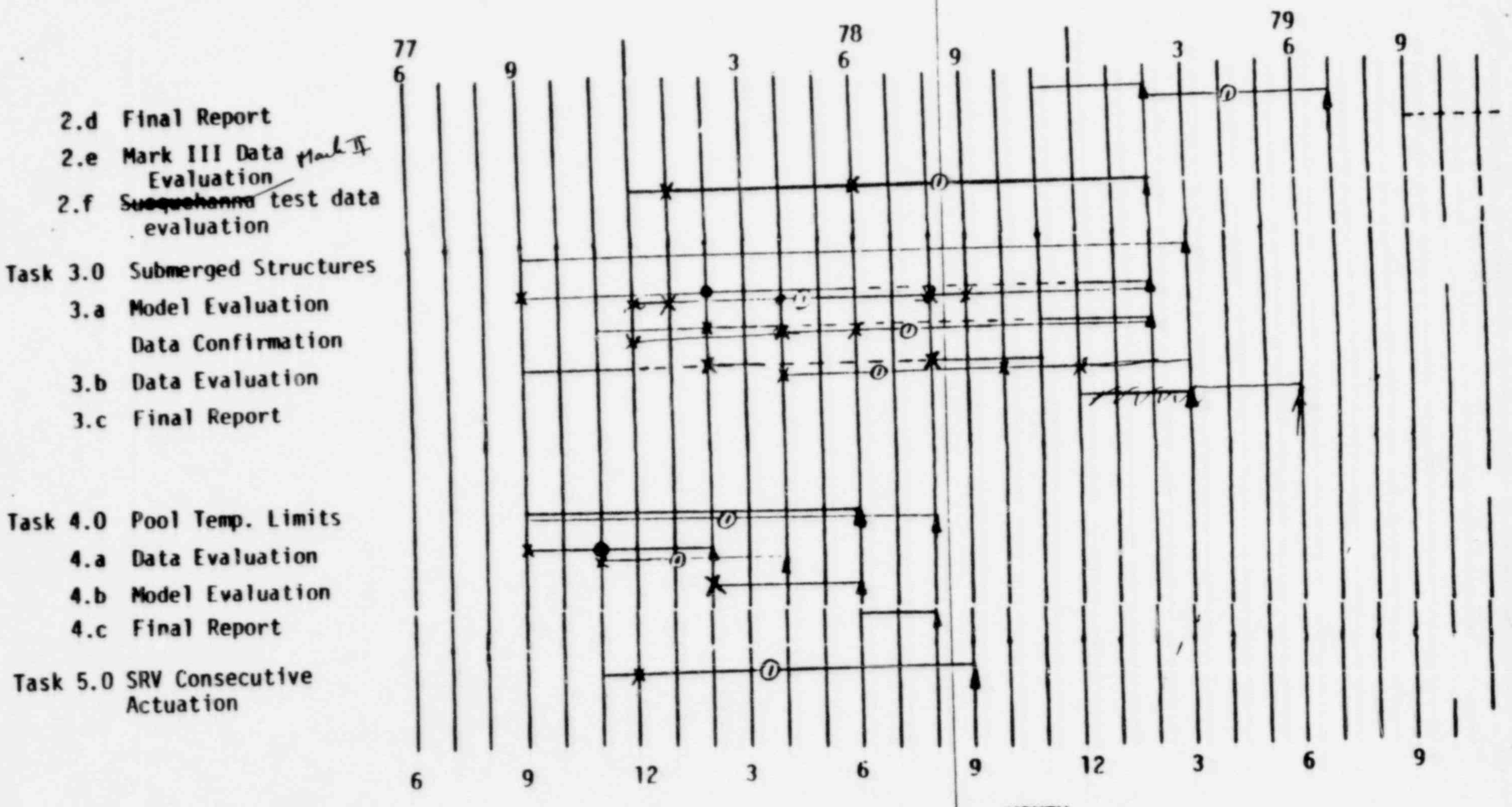
A. The proposed schedules have been based to a large part on the current estimates of receipt of key documents from both the Mark I and Mark II owner's programs. Since there are several test programs involved, past performance would indicate a good possibility in schedule slippages in one or two tasks. 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.

C. The current funding for technical assistance has not included the effort for reviewing and evaluating the Susquehanna test program, test data and method of predicting SRV load (Task 2.f). As we have pointed out, this effort is expected to be substantial. Additional funding is needed. We will finalize the amount of funding needed when the final scope of responsibility becomes available, and will follow the normal procedure to submit our request for approval.

SRV PROGRAM SCHEDULE

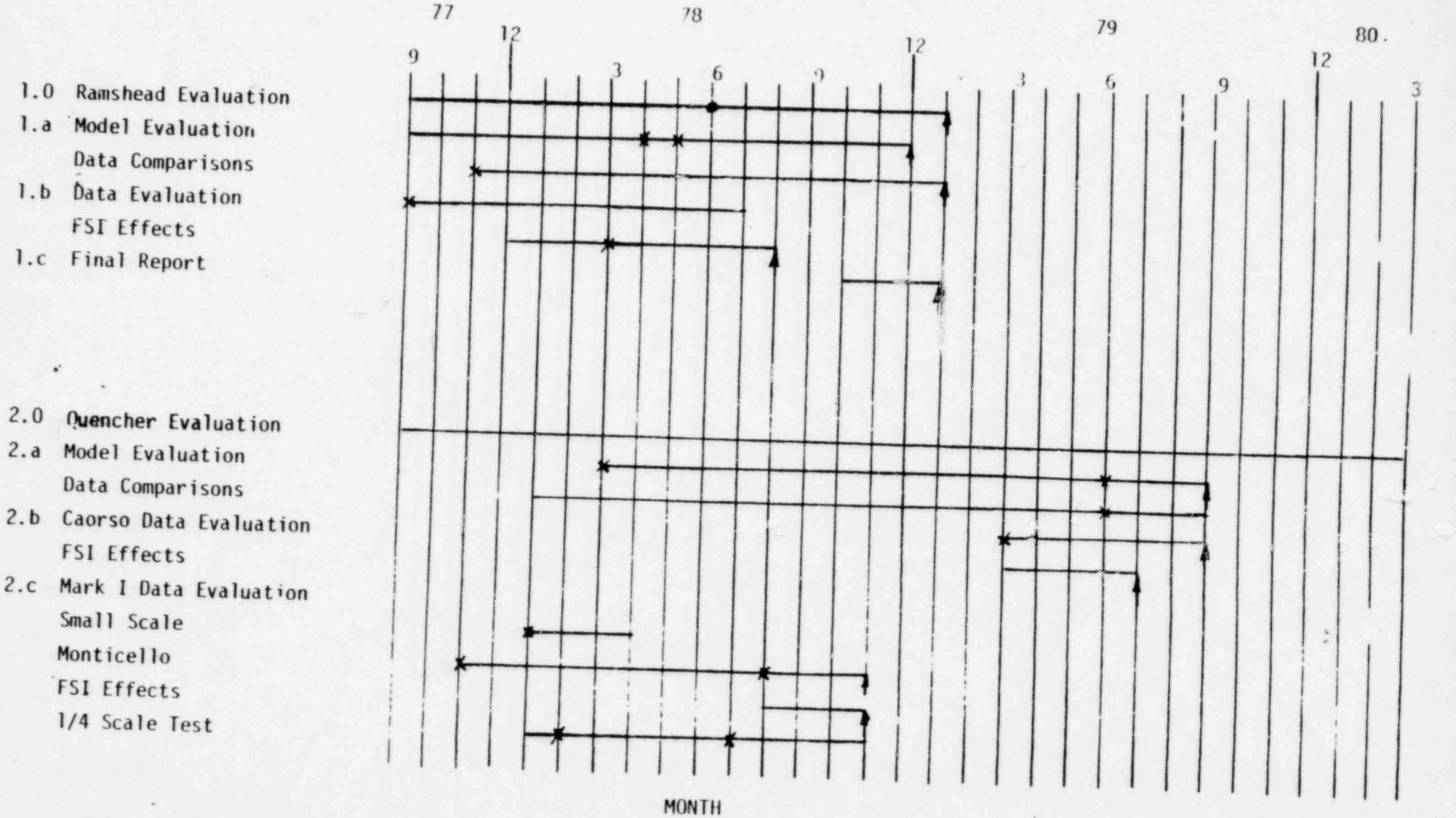


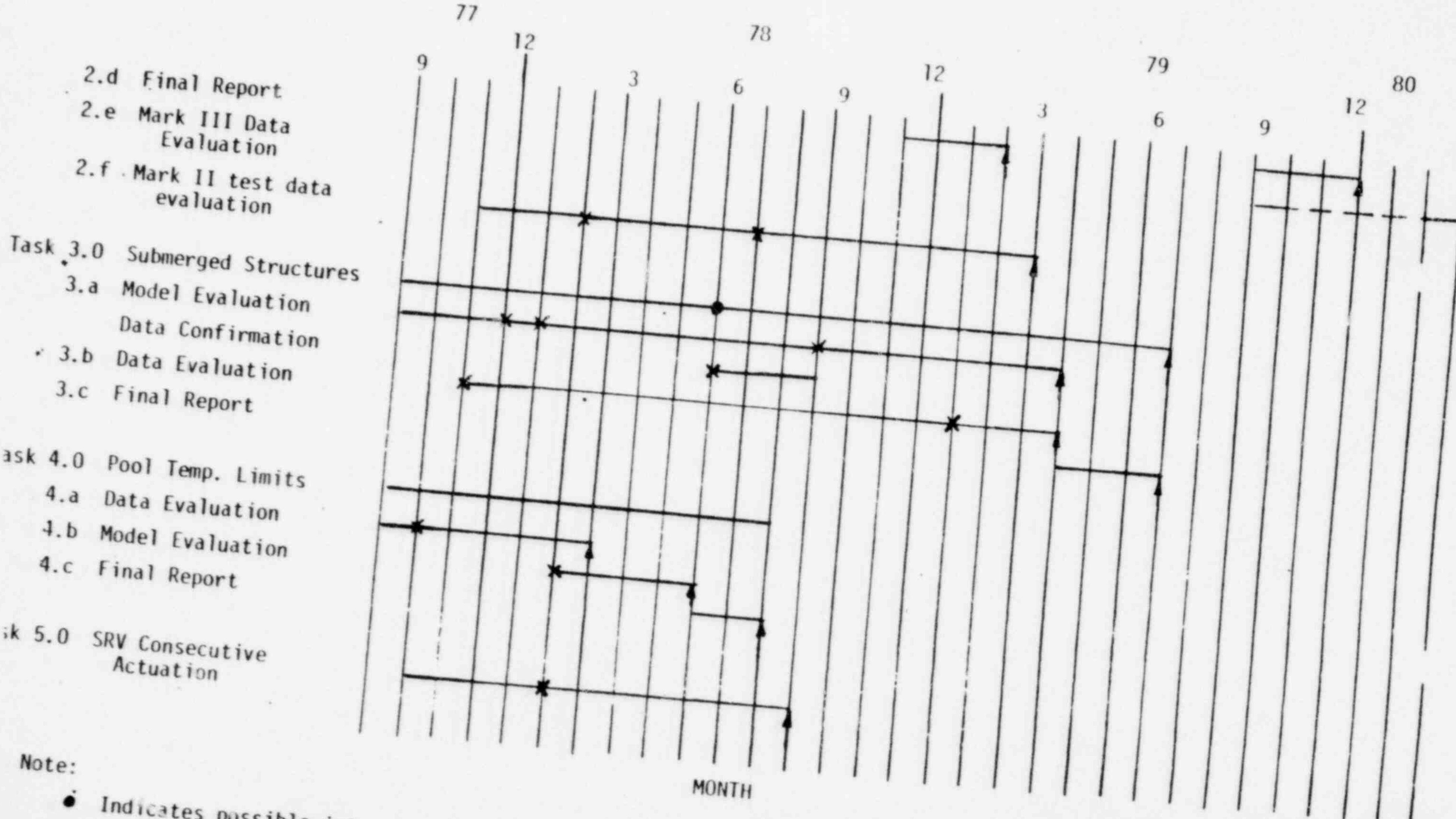


Note:

- ● Indicates possible intermin acceptance criteria
- X Indicates receipt of key documentation from either the Mark I or Mark II owner's programs or GE

SRV PROGRAM SCHEDULE (Revision 1)





Note:

● Indicates possib'e intermin acceptance criteria

X Indicates receipt of key documentation from either the Mark I or Mark II owner's programs or GE