POLICY ISSUE
(Information)

July 26, 1994

SECY-94-191

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: FATIGUE DESIGN OF METAL COMPONENTS

PURPOSE:

To keep the Commission informed of staff actions regarding resolution of the fatigue issue for metal components.

BACKGROUND:

In performing rulemaking activities related to the license renewal, the staff identified two issues for which the licensing bases differed notably between older and newer plants: equipment qualification (EQ) of electrical equipment and fatigue design of metal components. In SECY 93-049, the staff questioned whether these two issues should be reassessed in connection with future license renewal or whether they should be reassessed for the current license term. In a staff requirements memorandum (SRM) dated June 28, 1993, the Commission directed the staff to treat EQ and fatigue as potential safety issues within the existing regulatory process for operating reactors and to periodically inform the Commission of the staff's efforts. The EQ issue was addressed in an April 8, 1994, memorandum.

DISCUSSION:

Subsequent to the SRM dated June 28, 1993, the staff identified Generic Safety Issue 166 (GSI-166), "Adequacy of Fatigue Life of Metal Components." To address this GSI, the staff developed a Fatigue Action Plan (FAP) that was approved by NRR on July 13, 1993, and revised on June 2, 1994 (Enclosure 1). At the completion of the FAP, the staff will determine if additional regulatory actions are needed to ensure continued fatigue adequacy. If such actions are determined to be necessary, a regulatory licensing action plan that addresses the implementation of staff actions will be developed.

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS FROM THE DATE OF THIS PAPER

Contact: Terence L. Chan, NRR
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The FAP discusses the actions and schedule required to expeditiously resolve three principal issues:

(1) Many older vintage nuclear power plants have components of the reactor coolant pressure boundary that were designed to codes that did not require the explicit fatigue analysis required by the current American Society of Mechanical Engineers (ASME) Code. A concern regarding the fatigue resistance of these components for the plant design life was raised.

(2) Current test data show that the design fatigue curves of the ASME may not be conservative for nuclear power plant primary system environments. A concern regarding the fatigue resistance of components designed using these ASME curves was also identified.

(3) The appropriate corrective action to be taken when the calculated fatigue allowable limits have been exceeded (cumulative usage factor (CUF) >1) is the subject of controversy. The staff identified a need to develop a staff position on this subject.

The FAP addresses the technical concerns regarding the original licensing-basis code criteria used to evaluate the fatigue resistance of components in operating plants. Since the fatigue phenomenon is a time-dependent issue, its cumulative effects increase with the number of stress or strain cycles induced in service. Significant fatigue damage would eventually result in cracks in components. If fatigue cracks were to occur as a result of the stated principal issues addressed by the action plan, they would be expected to show up at older operating plants first. However, considering the service experience to date, fatigue cracks related to the issues addressed by the action plan have not been identified, even at the older plants. This offers some assurance that the concerns addressed by the action plan do not present an immediate problem. On the other hand, fatigue cracking has occurred due to loads that were not considered in the original design. These occurrences of fatigue cracking have been dealt with by staff actions such as the issuance of bulletins (e.g., NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems"). Even if fatigue damage results in pipe cracking, the contribution of the pipe cracking to core melt is considered small based on a study of the phenomenon of intergranular stress-corrosion cracking (IGSCC) at BWR facilities. The IGSCC phenomenon resulted in cracking in piping and piping nozzles at some facilities. Earlier risk studies concluded that higher pipe failure rates due to IGSCC were not a major contributor to core melt (Reference: NUREG-1061, Volume 1).

The staff also believes that the fatigue issues addressed in this action plan will primarily impact the piping and piping component nozzles. Therefore, the staff believes that the previous risk studies serve as an adequate basis for a preliminary assessment of the action plan concerns. Additional risk assessments directly applicable to the action plan concerns are an ongoing effort pursuant to FAP Action II.6. The results of that effort will be used to determine the need for any additional staff actions.

Another issue related to the fatigue evaluation of components, Generic Safety Issue (GSI)-78, "Monitoring of Design Basis Transient Fatigue Limits for Reactor Coolant System," was developed to determine whether transient monitoring (cycle counting) is necessary at operating plants. The goal of the transient monitoring addressed by GSI-78 is to provide assurance that components do not exceed their licensing basis during the lifetime of the plant. This licensing basis, with respect to fatigue design, is to ensure that the component CUF is less than unity for its design life. The design-basis requirement to have the CUF below unity is to assure that component fatigue failures will not occur. However, the simple cycle counting procedures used at some operating plants do not provide a direct measure of the CUF. One reason is that the licensing-basis CUF may be below unity for the number of cycles specified in the design. Therefore, the component could experience additional cycles without exceeding its licensing basis. Another reason is the design usually specifies bounding transients for the CUF evaluations, whereas the actual plant transients may be much less severe; therefore, the fatigue damage per transient cycle is actually less than that calculated in the design. This technical action plan will attempt to assess the margins in the licensing-basis analyses. These evaluations will be used to determine whether any new requirements, such as additional or more detailed transient monitoring of components, are necessary. Such additional actions will then be developed in the regulatory action plan.

To date, no cracking has occurred that is attributed to the technical issues addressed by the fatigue action plan. The staff is assessing whether the technical issues will lead to a concern with eventual fatigue cracking in components as the operating plants continue to age. Because previous studies concluded that pipe cracks did not contribute significantly to core-melt probabilities, and this conclusion has been borne out by actual field experience with cracking of piping and components due to other concerns, the staff believes that no immediate safety concern exists, while pursuing the resolution of the issues addressed by this action plan.

Regarding FAP issues 1 and 2, the staff has completed its survey of the plant design codes, identified fatigue-sensitive components for review with consideration of safety significance, and completed plant visits to gather design and operation information in support of performing independent fatigue analyses on selected primary system components. These visits included two plants from each reactor vendor type for Westinghouse (W), General Electric (GE), and Combustion Engineering (CE). Only one Babcock & Wilcox (B&W) plant was visited, since all B&W reactor pressure boundary piping components were designed to the same code. The seven plants selected are: Browns Ferry (GE), Clinton (GE), Comanche Peak (W), Ft. Calhoun (CE), Oconee (B&W), San Onofre (CE), and Turkey Point (W). This constitutes completion of FAP Actions II.1, II.2 and II.3.

The interim fatigue design curves have been obtained (FAP Action I.2), and an NRC contractor is currently performing fatigue analyses consistent with FAP Actions II.4 and II.5. The staff plans to publish the results of the these fatigue analyses in a NUREG/CR-series report. Since the analysis effort did

not begin as originally scheduled, we expect this activity to be completed in July 1994.

As to FAP issue 3, the staff is developing an interim position paper (FAP Action I.1) which proposes that corrective actions be implemented when a licensee discovers that its licensing-basis fatigue criterion has been exceeded. This interim position paper, which will be made available in the form of a generic communication, will be approved through established review and approval processes.

In Phase III of the FAP, the staff will develop its final resolution of fatigue issues. Actions to be taken are described in Phase III of the FAP. A graphical representation of the staff's progress is presented in Enclosure 2.

The staff sent a copy of the FAP to the former Nuclear Management and Resources Council (NUMARC) on July 30, 1993, and met with NUMARC on September 17, 1993, to discuss implementation of the FAP. A copy of the revised FAP was sent to Nuclear Energy Institute (NEI), the successor to NUMARC, on June 16, 1994. More meetings with NEI will be arranged once the staff has progressed enough with the objectives of the FAP so that these meetings will be mutually beneficial.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Fatigue Action Plan
2. Fatigue Adequacy Technical
   Action Plan — Milestones

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Encl 1

~~Revision 1~~

ENCLOSURE

FATIGUE ACTION PLAN

DEFINITION OF ISSUES

In developing criteria for the evaluation of applications for license renewal, the staff developed a draft branch technical position on fatigue evaluation procedures. Subsequent discussions within the staff and between the staff and the industry identified three major issues regarding the fatigue evaluation of candidate plants for license renewal (and current operating plants). These issues are:

1. Many older vintage nuclear power plants have components of the reactor coolant pressure boundary that were designed to codes that did not require the explicit fatigue analysis required by the current ASME Code. A concern regarding the fatigue resistance of these components for the plant design life was identified.

2. Current test data show that the ASME design fatigue curves may not be conservative for nuclear power plant primary system environments. A concern regarding the fatigue resistance of components designed using these ASME curves was also identified.

3. The appropriate corrective action to be taken when the calculated fatigue allowable limits have been exceeded (CUF>1) is the subject of controversy. A staff position regarding this issue is needed.

DISCUSSION OF ISSUES

For older vintage plants, components of the reactor coolant pressure boundary were designed to codes, such as ANSI B31.1, that did not require an explicit fatigue analysis of the components. Because the ASME Code currently requires a fatigue evaluation of the components of the reactor coolant pressure boundary, this leads to a question regarding the fatigue resistance of these older vintage plants. In order to assess the fatigue resistance of the older vintage plants, an actual fatigue evaluation of a sample of the components in these plants is planned. This sample will be selected using the results of fatigue analyses from similar systems or components in plants for which the fatigue analyses have been performed as a guide in selecting critical locations.

In addition, some recent test data indicate that the effects of the LWR environments could significantly reduce the fatigue resistance of materials. The ASME Code design fatigue curves were based primarily on strain-controlled fatigue tests of small polished specimens at room temperature in air. Although factors of safety were applied to the best-fit curves to cover effects such as size and data scatter, some of the recent test data indicate that these factors of safety may not be adequate to encompass the environmental effects. In order to assess the significance of the recent test data, an actual fatigue evaluation of a sample of components in plants where Code fatigue analyses have been performed is planned. These evaluations will

use interim or proposed fatigue curves that account for the environmental test data. The sample will be selected based on the most critical locations identified by the existing Code fatigue analyses. The new fatigue evaluations will remove conservatism, where appropriate, contained in the original fatigue analyses. This evaluation is intended to determine the impact on existing plant components of a proposed revision of the Code design fatigue curves that would account for the environmental effects.

Another major issue that has evolved from the discussions relating to the environmental effects on the fatigue curves is the appropriate corrective action required when the Code fatigue allowable limits have been exceeded (CUF>1). The staff needs to develop a regulatory position on this issue.

SAFETY SIGNIFICANCE

This action plan addresses the technical concerns regarding the original licensing basis code criteria used to evaluate the fatigue resistance of components in operating plants. Since the fatigue phenomenon is a time-dependent issue, its cumulative effects increase with the number of stress or strain cycles induced in service. Significant fatigue damage would eventually result in cracks in components. If fatigue cracks were to occur as a result of issues addressed by the action plan, they would be expected to show up at older operating plants first. However, considering the service experience to date, fatigue cracks related to the issues covered by the action plan have not been identified, even at the older plants. This provides some assurance that the concerns addressed by the action plan do not present an immediate problem. On the other hand, fatigue cracking has occurred due to loads that were not considered in the original design. These occurrences of fatigue cracking have been dealt with by staff actions such as the issuance of bulletins (e.g., NRC Bulletin No. 88-08). Even if fatigue damage results in pipe cracking, the contribution of the pipe cracking to core melt is considered small based on a study of the phenomenon of intergranular stress-corrosion cracking (IGSCC) at BWR facilities. The IGSCC phenomenon resulted cracking in piping and piping nozzles at some facilities. Previous risk studies concluded that higher pipe failure rates due to IGSCC were not a major contributor to core melt (Reference: NUREG-1061 Volume 1).

The staff also believes that the fatigue issues identified in this action plan will primarily impact the piping and piping component nozzles. Therefore, the staff believes that the previous risk studies provide an adequate basis for a preliminary assessment of the action plan concerns. Additional risk assessments directly applicable to the action plan concerns are an ongoing effort in action plan Item II.6. The results of that effort will be used to determine the need for any additional staff actions.

Another issue related to the fatigue evaluation of components, Generic Issue (GI) 78, "Monitoring of Design Basis Transient Fatigue Limits for Reactor Coolant System," was developed to determine whether transient monitoring (cycle counting) is necessary at operating plants. The goal of the transient monitoring addressed by GI-78 is to provide assurance that components do not exceed their licensing basis during the lifetime of the plant. This licensing basis, with respect to fatigue design, is to ensure that the component CUF is

less than unity. The design basis requirement to have the CUF below unity is to assure that component fatigue failures will not occur. However, the simplistic cycle counting procedures used at some operating plants do not provide a direct measure of the CUF. One reason is that the licensing basis CUF may be below unity for the number of cycles specified in the design. Therefore, the component could experience additional cycles without exceeding its licensing basis. Another reason is the design usually specifies bounding transients for the CUF evaluations whereas the actual plant transients may be much less severe; therefore, the fatigue damage per transient cycle is actually less than that calculated in the design. This technical action plan will attempt to assess the margins in the licensing basis analyses. These evaluations will be used to determine whether any new requirements, such as additional or more detailed transient monitoring of components, are necessary. Such additional actions will then be developed in the regulatory action plan.

To date, no cracking has occurred that is attributed to the technical issues addressed by the fatigue action plan. The staff is assessing whether the technical issues will lead to a concern with eventual fatigue cracking in components as the operating plants continue to age. Because previous studies concluded that pipe cracks did not contribute significantly to core melt probabilities and this conclusion has been borne out by actual field experience with cracking of piping and components due to other concerns, the staff believes that no immediate safety concern exists, while pursuing the resolution of the issues addressed by this action plan.

ACTION PLAN

Phase I - Short Term Actions

1. Develop a proposed staff position paper on licensee required actions for CUF>1.0. The paper will clarify the staff's position regarding exceeding the licensing basis Code criteria and the position will only apply to those facility's where the current licensing basis includes Code required fatigue analyses. If the staff decides to implement new requirements as a result of the evaluations performed in this action plan, then the backfit analysis discussed in Phase III Item 4 will be required. In developing the position paper regarding required actions for a CUF>1.0, past staff actions regarding exceeding licensing basis Code criteria will be researched. For example, the staff has issued several bulletins regarding piping analysis which contained required corrective actions for cases where calculated stresses exceed Code-allowable stresses. In addition, the staff has recently issued a generic position covering piping system operability determinations.

   Estimated Completion Date: June 1994

   Estimated Level of Effort: 12 staff weeks

2. Obtain a set of interim fatigue design curves from RES. This effort has been completed. The interim curves were published in NUREG/CR-5999.

Phase II - Long Term Actions

1. Perform a survey of current plants to determine the number of operating plants that have a fatigue analysis of the vessel, primary system components and piping. Based on the results of this survey, select representative plants from each reactor vendor that have components of the reactor coolant system that were designed without a fatigue analysis and representative plants for which similar components were designed using an ASME fatigue analysis. This effort will be performed by a review of the available NRC licensing documentation.

   Completion Date: September 1993

   Level of Effort: 5 staff weeks

2. Obtain a list of the critical components in terms of fatigue usage factors from the plants that have performed the ASME fatigue analyses. This effort may require coordination with the reactor vendor owners' groups.

   Completion Date: October 1993

   Level of Effort: 7 staff weeks

3. Prioritize the critical components identified in Task 2 in terms of safety significance of the components. This effort may require coordination with the reactor vendor owner's groups.

   Completion Date: December 1993

   Level of Effort: 7 staff weeks

4. Select example reactor coolant system components from plants designed without fatigue analyses and perform an ASME Section III fatigue analysis on these systems. The plants will include one from each reactor vendor and the components selected will be based on the results of task 3. Use both the current ASME Code and the interim fatigue design curves to perform the analysis. In addition, the fatigue usage factors will be computed for both a 40 and 60 year projected life. The results of the analyses from plants that currently have fatigue analyses will be used as a guide to select appropriate component examples for this analysis.

   Estimated Completion Date: July 1994

   Estimated Level of Effort: 32 contractor professional staff weeks
   4 staff weeks

5. Select example reactor coolant system components from plants designed using the ASME Code current fatigue curves to assess the impact of the interim fatigue curves. The plants will include one from each reactor vendor and the components selected will be based on the results of task

3. This evaluation will include a removal, when appropriately justified, of the conservatism in the assumptions used in the current analysis. An example of a conservative assumption may be in the heat transfer coefficient used in the original analysis. This evaluation is intended to assess the impact on the design of a change in the design fatigue curves. This evaluation will also consider both a 40 and 60 year projected life.

Estimated Completion Date: July 1994

Estimated Level of Effort: 32 contractor professional staff weeks
4 staff weeks

6. Obtain the Generic Issue 78 PRA parametric study from Research. Use these results in combination with the results of tasks 3, 4 and 5 to assess the impact of the fatigue concerns. Research originally estimated that the studies would be complete by 12/31/93.

Estimated Completion Date: August 1994

Estimated Level of Effort: 4 staff weeks

Phase III - Develop Staff Position on Fatigue Issues

1. Obtain the latest fatigue data from all sources including foreign sources (i.e., the Germans and the Japanese). Since the development of fatigue data is an ongoing effort, the latest available data will be obtained prior to developing the staff position.

Estimated Completion Date: August 1994

Estimated Level of Effort: 4 staff weeks

2. Update the interim fatigue curves using the latest available test data. The significance of any changes between these revised curves and the original interim curves will be assessed in terms of the results of the Phase II example analyses.

Estimated Completion Date: August 1994

Estimated Level of Effort: 4 staff weeks

3. Meet with the current industry working groups (PVRC, ASME, etc) and obtain the latest data available from these groups. Also obtain their input regarding the results of the staff's analysis.

Estimated Completion Date: June 1994

Estimated Level of Effort: 2 staff weeks

4. Develop a staff position using the available input from the fatigue studies and the industry efforts. The staff position will address: (1) whether older plants for which ASME Code fatigue analyses were not required at the time of plant licensing for the reactor coolant pressure boundary should now be required to perform a fatigue assessment of the reactor coolant pressure boundary components; and (2) whether plants with ASME Code fatigue analyses of the reactor coolant pressure boundary should be required to reassess the reactor coolant pressure boundary components for the impact of the new data on environmental concerns. This staff position will be supported by a backfit analysis using the results of the PRA parametric study obtained from Research, if appropriate.

   Estimated Completion Date: October 1994

   Estimated Level of Effort: 10 staff weeks

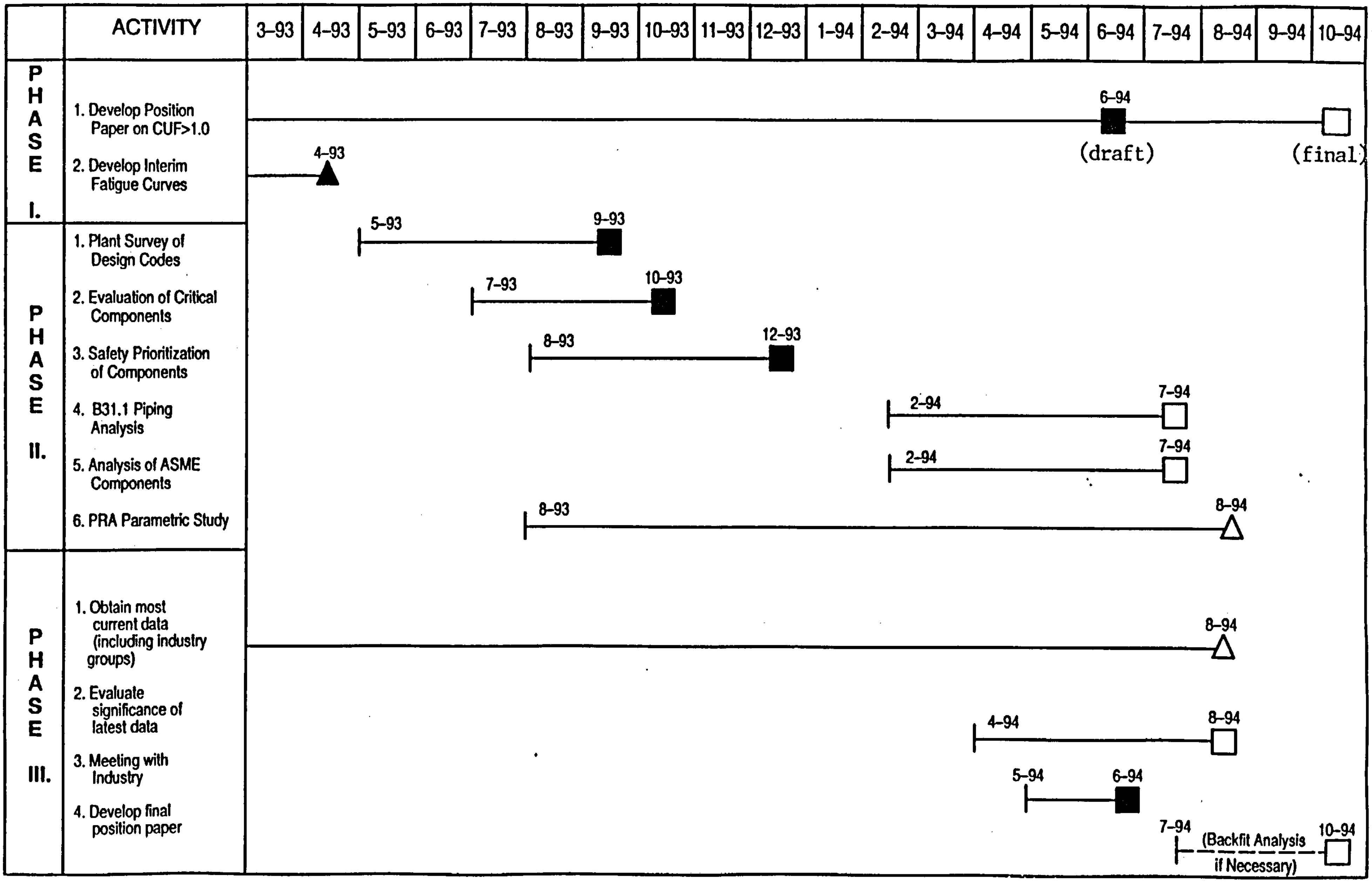
OTHER CONSIDERATIONS

This is a technical action plan that is necessary to determine the scope of the problem. A regulatory licensing action plan will be developed to address the implementation of the final staff position if required.

ENCLOSURE 2

FATIGUE ACTION PLAN – MILESTONES

Date: 6/7/94



Legend: □ NRR; △ RES; ● Complete