POLICY ISSUE
(Information)

September 25, 1995 SECY-95-245

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: COMPLETION OF THE FATIGUE ACTION PLAN

PURPOSE:

To inform the Commission that the staff has completed its work on the Fatigue Action Plan (FAP), and to give the Commission a summary of the staff actions.

BACKGROUND:

In performing rulemaking activities related to license renewal, the staff found that the licensing basis differed considerably between older and newer plants for two issues: 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 conjunction with future license renewal or whether they should be reassessed for the current license term. In a staff requirements memorandum of 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. On July 26, 1994, the staff reported to the Commission actions it had taken to resolve the fatigue issue for metal components (SECY 94-191). This memorandum discusses the subsequent actions taken by the staff to complete its work on the FAP. A copy of the FAP and a report summarizing the action taken to address each item are provided as Attachments 1 and 2, respectively.

DISCUSSION:

The FAP was developed to resolve three principal issues:

(1) Many older vintage nuclear power plants have components of the reactor coolant pressure boundary (RCPB) that were designed to industry codes that did not require the explicit fatigue analysis presently required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure

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Vessel Code. A concern was raised regarding the fatigue resistance of these components for the plant design life.

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

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

To address the technical issues, the FAP calls for the evaluation of a sample of RCPB components. Since the staff report in SECY 94-191, Idaho National Engineering Laboratory (INEL), under contract to the NRC Office of Nuclear Reactor Regulation (NRR), has completed its assessments of the sample components selected from seven power plants. NRC published the results of the INEL component evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components."

On the basis of the FAP work, the staff believes that no immediate staff or licensee action is necessary to deal with the fatigue issues addressed by the FAP. Although the component sample evaluation indicates that the current ASME Code fatigue limit may be exceeded prior to the end of the current design life at some piping component and piping nozzle locations using the fatigue curves that account for nuclear power plant primary system (reactor coolant) environments, the staff and its contractor believe that more detailed analyses or the use of measured plant transient data, or both, would show that the fatigue limit will not be exceeded for most of these components. However, even with additional plant transient information or analyses or both, there may be some piping component locations at which the ASME Code fatigue limit may be exceeded prior to the end of the current design life using fatigue curves that account for reactor coolant environments. The NRC Office of Nuclear Regulatory Research (RES) risk study indicates that a fatigue failure of piping is not a significant contributor to the core-melt frequency. This result is due to contributing reasons in the risk assessment which include the fact that while fatigue cracks may occur, they may not propagate to failure and, even if failure did occur, safety systems, such as the emergency core cooling system (ECCS), mitigate the consequences. On the basis of the RES risk study, the staff does not believe it can justify requiring a backfit of the environmental fatigue data to operating plants.

The conclusions in this paper are based on the results of the component sample evaluation for a current facility design life. Because fatigue is a time-dependent phenomenon whose effects increase with service life, the staff believes that the FAP fatigue issues should be evaluated for any proposed extended period of operation for license renewal. The staff will consider, as part of the resolution of GSI-166, "Adequacy of Fatigue Life of Metal Components," the need to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data.

This completes the staff's effort on the Fatigue Action Plan.

James L. Milhoan for
James M. Taylor
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Attachments: As stated

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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 in 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.

FATIGUE ACTION PLAN COMPLETION REPORT

In a staff requirements memorandum of June 28, 1993, the Commission directed the staff to treat fatigue as a potential safety issue within the existing regulatory process for operating reactors and to periodically inform the Commission of the staff's efforts. In response, the staff developed the Fatigue Action Plan (FAP), which was approved on July 13, 1993, and revised on June 2, 1994. The staff gave the Commission a copy of the revised FAP in a July 26, 1994, status report on the fatigue issue (SECY 94-191).

The FAP describes the activities to be undertaken by the Offices of Nuclear Reactor Regulation (NRR) and Nuclear Regulatory Research (RES). These activities comprise meetings with industry, evaluation of fatigue data and plant components, risk assessment, and development of a staff position.

The FAP was developed to resolve three principal issues:

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

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

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

To address the first two issues, the FAP calls for the evaluation of a sample of RCPB components. Since the last report was issued, Idaho National Engineering Laboratory (INEL), under contract to NRR, has completed its assessments of the sample components selected from seven power plants (FAP Items II.4 and II.5). While the components were being assessed, stainless steel fatigue data were also being evaluated under a separate contract with RES. As a result of this evaluation, stainless steel fatigue curves were revised (FAP Item III.1). INEL evaluated these revised curves in additional component assessments (FAP Item III.2). NRC published the results of the INEL component evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components."

The staff's survey of plant licensing documentation (FAP Item II.1) indicated that a significant number of older operating plants have RCPB piping designed to a code that did not require explicit fatigue analyses. Although the piping code for these older plants did not require such analyses, the code did

ATTACHMENT 2

provide for addressing cyclic thermal expansion stresses. These stresses are caused by the restraint of free thermal expansion at rigid support locations when the piping system heats up and cools down. The code required the allowable stress limit for thermal expansion/contraction stresses to be reduced if the number of full-temperature cycles exceeded a specified value. In addition, the allowable stress limits in the earlier piping code were generally more conservative than the ASME Code Class 1 allowable stress limits. (The RCPB piping components of newer plants are designed to the Class 1 criteria of the ASME Code.) However, the earlier piping code did not require an evaluation of local thermal stresses, such as those that result from the temperature gradients in the pipe wall during plant operational transients. INEL evaluated a sample of piping components from some of the older plants (FAP Item II.4) to assess the significance of these code differences. INEL relied on data obtained from the newer plant fatigue analyses to estimate the number and magnitude of the temperature and pressure transients for some components. The evaluation indicated that the current ASME Code fatigue CUF limit should not be exceeded for each component in the sample prior to the end of its current design life. (The ASME Code contains several other associated design limits. Not all of these limits were met for all component evaluations; however, INEL believed that a more detailed evaluation may show that these other limits can also be met.) On the basis of the sample evaluation results, using the current ASME fatigue curves, the staff does not believe that the lack of an explicit fatigue analysis in the design of piping system components at older plants constitutes a significant safety concern and, therefore, no backfit of fatigue analyses to older plants is recommended at this time.

For FAP Items II.4 and II.5, INEL evaluated a sample of components, including the reactor vessel shell, using fatigue curves developed from environmental test data. The concern was that fatigue damage might have been underestimated because the fatigue curves in the ASME Code were not based on temperature and dissolved oxygen conditions typical of light-water reactor (LWR) coolant environments. The evaluation, for the current design life, indicated that the current ASME Code CUF limit should not be exceeded for the majority of the components in the sample when fatigue curves that account for reactor coolant environments were used. The components for which, in some cases, the usage limit may be exceeded prior to the end of design life were piping components and piping nozzles, such as the boiling-water reactor (BWR) feedwater nozzle.

RCPB piping can be evaluated using either the "piping design" rules or the "design by analysis" rules in the ASME Code. Since the latter rules require a more detailed and time-consuming evaluation than the former rules, the former rules are generally used. The majority of the piping components in the INEL sample were evaluated using the piping design rules. INEL evaluated a pressurized-water reactor (PWR) charging nozzle and a PWR safety injection nozzle using both sets of rules to assess the conservatism in the piping design rules. The detailed evaluation using the design by analysis rules resulted in a significantly lower CUF for both nozzles. The evaluation indicated that the ASME Code CUF limit should not be exceeded prior to the end of the current design life for both nozzles using the design by analysis rules and the environmental fatigue curves.

Some utilities have installed fatigue monitoring equipment in their facilities at fatigue-critical locations. Some of the monitoring equipment uses plant process instrumentation to evaluate the severity of plant thermal transients. The data from this monitoring equipment can be used to assess the conservatism of the transients used in the evaluation of RCPB components. Although fatigue monitoring data were obtained for some components in the FAP sample, detailed thermal transient data were not available for most of the components selected. However, the results of a fatigue monitoring study of a BWR feedwater nozzle safe-end were reported in SAND94-0187, "Evaluation of Conservatisms and Environmental Effects in ASME Code, Section III, Class 1 Fatigue Analysis," published by Sandia National Laboratory. The study covered about 18 months of plant operational data and indicated that the fatigue usage based on the measured plant data was more than an order of magnitude lower than the fatigue usage predicted using the transients specified for design.

On the basis of this discussion, the staff and its contractor believe that additional detailed analyses or the use of data obtained from monitoring plant operating transients or both would show that most components in the FAP sample will not exceed the current ASME Code CUF limit prior to the end of the current design life using fatigue curves developed from environmental test data. Most of the piping components selected for the FAP sample (FAP Item II.2) were identified as locations for which high fatigue usage factors have been calculated. Therefore, the staff believes there are only a limited number of locations at a facility where the current ASME Code CUF may be exceeded at the end of the design life using environmental fatigue curves.

The third issue of the FAP is the corrective action that should be taken if the licensing-basis criterion regarding the fatigue-allowable limit has been exceeded (FAP Item I.1). The staff believes that Generic Letter (GL) 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," provides guidance on this issue. The GL stipulates that the failure to meet a code criterion specified in the final safety analysis report (FSAR) is a nonconforming condition that requires a corrective action. Since the licensing-basis code is specified in the FSAR, the staff considers the guidance in GL 91-18 applicable. GL 91-18 describes actions that a licensee can take to resolve the nonconforming condition.

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. The transient monitoring concern in GSI-78 was subsumed into the FAP and the staff effort toward the resolution of GSI-78 was then redirected to evaluate risk for FAP Item II.6. The staff obtained available records of transient monitoring from the licensees at the seven plants selected for the FAP sample. These records, for the most part, contained the number of cycles associated with such plant operational changes as plant heatups and cooldowns. On the basis of the FAP sample, considering the available transient monitoring records and the conservatisms identified in the component analyses, the staff does not believe there is a significant safety concern that current licensing-basis fatigue criteria have been exceeded at operating plants.

Although the staff is not recommending additional licensee actions to address transient monitoring at this time, the staff does consider the fatigue adequacy of the RCPB to be important. The staff recently met with representatives of Nuclear Energy Institute (NEI) to discuss the fatigue issue. The industry is considering actions that may be appropriate to deal with the limited plant-specific information that is available regarding fatigue usage over plant life, and an NEI paper is expected in the near future.

RES completed the risk assessment for FAP Item II.6. Since a preliminary risk assessment identified the reactor vessel shell below the core as an important component, the staff selected locations of highest fatigue usage on the shell below the top of the core as part of the component sample for FAP Items II.4 and II.5. The evaluation of these locations, for the current design life, has produced acceptable fatigue results. The current RES risk assessment confirmed the findings of the preliminary risk assessment with regard to the importance of the reactor vessel shell location chosen for the component sample. The results of the current RES risk assessment also indicate that a fatigue failure of a piping component is not a significant contributor to the core-melt frequency. This result is due to contributing reasons in the risk assessment which include the fact that while fatigue cracks may occur, they may not propagate to failure and, even if failure did occur, safety systems, such as the emergency core cooling system (ECCS), mitigate the consequences. The staff believes that, although it is appropriate to consider the results of this risk assessment to determine whether new criteria should be backfit to operating plants, the results should not obviate the need for assuring the high integrity of RCBP components by taking appropriate corrective actions when fatigue cracks are detected.

On the basis of the FAP work, the staff believes that no immediate staff or licensee action is necessary to deal with the fatigue issues addressed by the FAP (FAP Item III.4). Although the component sample evaluation indicates that the current ASME Code fatigue limit may be exceeded prior to the end of the current design life of some components using the fatigue curves that account for reactor coolant environments, the staff and its contractor believe that more detailed analyses or the use of measured plant transient data or both would show that the fatigue limit will not be exceeded for most of these components. (The component evaluations were performed using interim fatigue curves which are also subject to change based on ongoing RES studies.) However, even with additional plant transient information or analyses or both, there may be some piping component locations at which the ASME Code fatigue limit will be exceeded prior to the end of the design life using the environmental fatigue curves (FAP Items I.2 and III.1). Since the RES risk study indicates that a fatigue failure of a piping component is not a significant contributor to the core-melt frequency, the staff does not believe it can justify requiring a backfit of the environmental fatigue data to operating plants. The staff also notes that the assessment of the reactor vessel shells below the core, using environmental fatigue curves, produced acceptable fatigue results. The staff will continue to work with industry groups and within the ASME Code committees to develop a long-term resolution regarding the ASME Code fatigue curves. If ASME adopts new fatigue curves, they would only be required for future component construction.

The FAP addresses the technical concerns regarding the original licensing-basis code criteria used to evaluate the fatigue resistance of RCPB components in operating plants. The loads used in the licensing-basis fatigue analyses are generally based on events that consist of a limited number of cycles, such as plant startups and shutdowns. As discussed in SECY 94-191, no cracking has occurred that is attributable to the issues addressed by the FAP. However, as also discussed in SECY 94-191, fatigue cracking has occurred from loads that were not considered in the original design. In some cases, these cracks were caused by a relatively large number of cycles of local temperature fluctuations. These local temperature fluctuations were unanticipated and, consequently, could not have been dealt with by an explicit fatigue analysis in the original design. The staff has taken, and will continue to take, appropriate actions to deal with these events when they occur. The staff has dealt with these occurrences of fatigue cracking by such actions as issuing bulletins (e.g., NRC Bulletin No. 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems"). In response to this bulletin, licensees have assessed the potential for piping stresses due to local temperature fluctuations. The staff dealt with another related issue in NRC Bulletin No. 88-11, "Pressurizer Surge Line Thermal Stratification." This bulletin addressed the unexpected movements of a PWR pressurizer surge line due to temperature stratification of the fluid in the horizontal section of the piping. All surge lines in the FAP sample, including surge lines of the older plants, had detailed fatigue analyses of this phenomenon.

The conclusions discussed above are based on the results of the component sample evaluation for a current facility design life. For facility operation beyond the current design life, the conclusions are less certain. It would be more difficult to demonstrate that all components in the FAP sample, including components from both old and new plants, will not exceed the ASME Code fatigue usage factor limit using environmental fatigue curves for an extended plant life. In addition, the RES risk study used the results of previous technical studies to obtain estimates of the probability of through-wall cracking or failure of the reactor vessel and reactor coolant loop piping. These previous estimates were developed considering a current facility design life. The staff also notes that the evaluation of one vessel location below the top of the core indicated that the current ASME Code CUF limit may be exceeded for a 60-year design life when the environmental fatigue data were used. Therefore, the staff believes that the FAP fatigue issues should be evaluated further, focusing mainly on components in the RCPB with high fatigue usage for any proposed extended period of operation. The staff will consider, as part of the resolution of GSI-166, "Adequacy of Fatigue Life of Metal Components," the need to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data, to ensure that RCPB components will continue to perform their intended functions and maintain a high level of reliability during the extended period of operation for license renewal. If GSI-166 has not been resolved before the issuance of a renewal license, the applicant would have to submit (60 FR 22484, May 8, 1995) its technical rationale for concluding that the effects of fatigue are adequately managed for the extended period or until the resolution of GSI-166 becomes available.

Efforts to resolve GSI-166, being managed by NRR, are continuing and are periodically reported under the RES Generic Issues Management Control System.

The following summary lists the FAP items and describes the staff actions on each item.

Phase I - Short-Term Actions

1. Action Plan Item:

   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 facilities 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.

   Completion Date: September 1995

   Results:

   The staff guidance on piping system operability determinations, GL 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," addresses the issue in general terms. The GL stipulates that the failure to meet a code criterion specified in the FSAR is a nonconforming condition which requires a corrective action. Since the licensing-basis code is specified in the FSAR, the staff considers the guidance in GL 91-18 applicable. GL 91-18 describes actions that a licensee can take to resolve the nonconforming condition.

   Section XI of the ASME Boiler and Pressure Vessel Code is developing a non-mandatory Appendix to specify actions that should be taken if the CUF exceeds unity. When this Appendix is published, the staff will determine whether it provides an acceptable approach. Meanwhile, the staff will continue to review licensee corrective actions using the guidance in GL 91-18.

2. Action Plan Item:

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

   Completion Date: April 1993

Results:

The interim fatigue curves were used for the evaluations described in FAP Items II.4 and II.5.

Phase II - Long-Term Actions

1. Action Plan Item:

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

Results:

The licensing documentation survey indicated that about 40 percent of the operating plants have primary system piping components designed to a piping code that did not require a formal fatigue analysis.

2. Action Plan Item:

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

Results:

The staff used a list contained in Electric Power Research Institute (EPRI) Report TR-100252, "Metal Fatigue in Operating Nuclear Power Plants," April 1992, to select some of the critical components. The EPRI report lists typical areas in pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) for which high fatigue usage factors have been calculated.

3. Action Plan Item:

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 owners' groups.

Completion Date: December 1993

Results:

The prioritization was based on preliminary work by RES on the risk study. A location on the reactor vessel shell below the top of the core was selected as the most risk-significant component location to include in the evaluation.

4. Action Plan Item:

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.

Completion Date: July 1994

Results:

Three plants with different nuclear steam supply system (NSSS) vendors were selected for this evaluation. Based on the review in Item II.1, a Babcock & Wilcox (B&W) plant was not selected because all B&W primary piping systems had fatigue analyses. Later, it was discovered that most of the primary piping from the Combustion Engineering plant selected also had existing fatigue analyses. The surge lines of all PWRs selected also had recent fatigue analyses. For each plant, six separate locations including the vessel shells were evaluated.

Seven separate piping components in the selected sample did not have an existing fatigue analysis. INEL performed a fatigue analysis of these components using current ASME Code fatigue criteria. The evaluation of these seven components indicated that the ASME Code fatigue usage factor limit should not be exceeded for each component in the sample prior to the end of its current design life when the current ASME Code fatigue curves were used. (The ASME Code contains several other associated design limits. Not all of these ASME Code limits were met for all component evaluations; however, INEL believed that a more detailed evaluation may show that these other limits can also be met.) The current code usage factor limit should also not be exceeded for a 60-year design life for most components. (On the basis of INEL's evaluation, it may be exceeded for one BWR location using a shutdown cooling transient defined by the licensee. However, if a shutdown cooling transient definition recommended by General Electric (GE) is used, the usage factor limit should not be exceeded at this location. INEL used the transient recommended by GE because it was considered more representative for older BWRs.)

An industry study of two piping systems, published in EPRI Report TR-102901, "Comparison of Piping Designed to ANSI B31.1 and ASME Section III, Class 1," produced results similar to the INEL study.

INEL evaluated all components in the sample, including the components that did not have existing fatigue analyses, using the interim fatigue curves (FAP Item I.2). The results of those evaluations are discussed in Item II.5.

5. Action Plan Item:

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.

Completion Date: July 1994

Results:

Four plants were selected for this evaluation, one plant from each NSSS supplier. INEL evaluated six locations at each plant using the interim fatigue curves (FAP Item I.2). INEL also evaluated the components selected in FAP Item II.4 using the interim fatigue curves.

The evaluation, for the current design life, indicated that the current ASME Code usage limit should not be exceeded for the majority of the components in the sample when the interim fatigue curves were used. The components for which, in some cases, the usage limit may be exceeded prior the end of the design life, were piping components and piping nozzles, such as the BWR feedwater nozzle. (A number of the components for which the usage limit may be exceeded prior to the end of the design life included thermal stratification events.) INEL believed that additional detailed analyses and/or the use of data obtained from monitoring plant operating transients in the analyses could show that most of these components also meet the current ASME Code fatigue usage factor limit. An evaluation for a 60-year projected life indicates that the usage limit may be exceeded at some additional locations, the most significant being at a lower vessel head core support block in the older Westinghouse plant.

6. Action Plan Item:

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.

Completion Date: September 1994

Results:

The evaluations of reactor pressure vessel (RPV) locations below the level of the core, for the current design life, in Tasks II.4 and II.5 produced acceptable fatigue results. Therefore, these locations are not a concern. The results of the RES risk study were transmitted by a September 23, 1994, memorandum from E. S. Beckjord to A. C. Thadani. The risk study showed that the risk from fatigue failure of reactor coolant piping systems is negligible compared to the RPV contribution. This result is based, in part, on an earlier study of the probability of primary loop piping failure (NUREG/CR-2189, Vol. 5, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant").

Phase III - Develop Staff Position on Fatigue Issues

1. Action Plan Item:

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.

Completion Date: October 1994

Results:

RES provided preliminary results of the stainless steel test data evaluations performed by Argonne National Laboratory (ANL) for Item II.6. NRC will publish the final results of the ANL stainless steel test data evaluations in NUREG/CR-6335. In addition, RES still has an ongoing effort to update the interim fatigue curves.

2. Action Plan Item:

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.

Completion Date: January 1995

Results:

The sample analyses were updated using fatigue curves developed from preliminary work by ANL on the stainless steel data evaluations. The final ANL statistical data evaluations resulted in a slightly different equation for stainless steel, but this difference should not significantly impact the results obtained by INEL. Also, some of the component analyses were updated on the basis of additional plant information.

3. Action Plan Item:

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.

Completion Date: June 1994

Results:

The staff met with industry representatives at the 1994 Pressure Vessels and Piping (PVP) Conference in Minneapolis, Minnesota to discuss technical issues related to fatigue. The staff also met with representatives of the Nuclear Energy Institute (NEI) in April 1995 to discuss the fatigue issues. The staff expects NEI to issue a white paper on the fatigue issues in the near future.

4. Action Plan Item:

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 RCPB should now be required to perform a fatigue assessment of the RCPB components; and (2) whether plants with ASME Code fatigue analyses of the RCPB should be required to reassess the RCPB 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.

Completion Date: September 1995

Results:

On the basis of the FAP work, the staff believes that no immediate staff or licensee action is necessary to deal with the fatigue issues addressed by the FAP. Although the component sample evaluation indicates that the current ASME Code fatigue limit may be exceeded prior to the end of the current design life of some components using the fatigue curves that account for reactor coolant environments, the staff and its contractor believe that more detailed analyses or the use of measured plant transient information or both would show that the fatigue limit will not be exceeded for most of these components. (The component evaluations were performed using interim fatigue curves which are also subject to change based on ongoing RES studies.) However, even with additional plant transient information or analyses or both, there may be some piping component locations at which the ASME Code fatigue limit may be exceeded prior to the end of the current design life using the environmental fatigue curves. Since the RES risk study indicates that a fatigue failure of a piping component is not a significant contributor to the core-melt frequency, the staff does not believe it can justify requiring a backfit of the environmental fatigue data to operating plants. The staff also notes that the assessment of the reactor vessel shells below the core, using environmental fatigue curves, produced acceptable fatigue results. The staff will continue to work with industry groups and within the ASME Code committees to develop a long-term resolution regarding the ASME Code fatigue curves. If ASME adopts new fatigue curves, they would only be required for future component construction.

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WITS 0135

September 25, 1995

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: COMPLETION OF THE FATIGUE ACTION PLAN

PURPOSE:

To inform the Commission that the staff has completed its work on the Fatigue Action Plan (FAP), and to give the Commission a summary of the staff actions.

BACKGROUND:

In performing rulemaking activities related to license renewal, the staff found that the licensing basis differed considerably between older and newer plants for two issues: 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 conjunction with future license renewal or whether they should be reassessed for the current license term. In a staff requirements memorandum of 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. On July 26, 1994, the staff reported to the Commission actions it had taken to resolve the fatigue issue for metal components (SECY 94-191). This memorandum discusses the subsequent actions taken by the staff to complete its work on the FAP. A copy of the FAP and a report summarizing the action taken to address each item are provided as Attachments 1 and 2, respectively.

DISCUSSION:

The FAP was developed to resolve three principal issues:

(1) Many older vintage nuclear power plants have components of the reactor coolant pressure boundary (RCPB) that were designed to industry codes that did not require the explicit fatigue analysis presently required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure

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Vessel Code. A concern was raised regarding the fatigue resistance of these components for the plant design life.

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

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

To address the technical issues, the FAP calls for the evaluation of a sample of RCPB components. Since the staff report in SECY 94-191, Idaho National Engineering Laboratory (INEL), under contract to the NRC Office of Nuclear Reactor Regulation (NRR), has completed its assessments of the sample components selected from seven power plants. NRC published the results of the INEL component evaluations in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components."

On the basis of the FAP work, the staff believes that no immediate staff or licensee action is necessary to deal with the fatigue issues addressed by the FAP. Although the component sample evaluation indicates that the current ASME Code fatigue limit may be exceeded prior to the end of the current design life at some piping component and piping nozzle locations using the fatigue curves that account for nuclear power plant primary system (reactor coolant) environments, the staff and its contractor believe that more detailed analyses or the use of measured plant transient data, or both, would show that the fatigue limit will not be exceeded for most of these components. However, even with additional plant transient information or analyses or both, there may be some piping component locations at which the ASME Code fatigue limit may be exceeded prior to the end of the current design life using fatigue curves that account for reactor coolant environments. The NRC Office of Nuclear Regulatory Research (RES) risk study indicates that a fatigue failure of piping is not a significant contributor to the core-melt frequency. This result is due to contributing reasons in the risk assessment which include the fact that while fatigue cracks may occur, they may not propagate to failure and, even if failure did occur, safety systems, such as the emergency core cooling system (ECCS), mitigate the consequences. On the basis of the RES risk study, the staff does not believe it can justify requiring a backfit of the environmental fatigue data to operating plants.

The conclusions in this paper are based on the results of the component sample evaluation for a current facility design life. Because fatigue is a time-dependent phenomenon whose effects increase with service life, the staff believes that the FAP fatigue issues should be evaluated for any proposed extended period of operation for license renewal. The staff will consider, as part of the resolution of GSI-166, "Adequacy of Fatigue Life of Metal Components," the need to evaluate a sample of components with high fatigue usage, using the latest available environmental fatigue data.

This completes the staff's effort on the Fatigue Action Plan.

James L. Milhoan for
James M. Taylor
Executive Director
for Operations

Attachments: As stated