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MFN 06-119, Supplement 4

Docket No. 52-010

January 02, 2008

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter Number 16 Related to ESBWR Design Certification Application – Piping Design – RAI Numbers 3.12-11 S01, 3.12-22 S01 and 3.12-27 S01**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) originally transmitted via the Reference 1 letter and supplemented by an NRC request for clarification in Reference 2. The GEH response to RAI Numbers 3.12-11 S01, 3.12-22 S01 and 3.12-27 S01 are addressed in Enclosure 1.

If you have any questions or require additional information, please contact me.

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

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NRO

References:

1. MFN 06-103, Letter from U.S. Nuclear Regulatory Commission to Mr. David H. Hinds, Manager, ESBWR, General Electric Company, *Request For Additional Information Letter No. 16 Related To ESBWR Design Certification Application*, dated March 30, 2006.

2. E-Mail from Amy Cubbage, U.S. Nuclear Regulatory Commission, to GE, dated May 20, 2007.

Enclosure:

1. Response to Portion of NRC Request for Additional Information Letter Number 16 Related to ESBWR Design Certification Application – Piping Design – RAI Numbers 3.12-11 S01, 3.12-22 S01 and 3.12-27 S01.

2. Attachment 1 – Proceedings of ASME-PVP 2007: 2007 ASME Pressure Vessel and Piping Division Conference, July 22-26, 2007, San Antonio, TX, USA. PVP2007-26143. "Application of Draft Regulatory Guide DG-1144 Guidelines For Environmental Fatigue Evaluation to a BWR Feedwater Piping System."

3. DCD Markups.

cc:	AE Cubbage	USNRC (with enclosure)
	DH Hinds	GEH/Wilmington (with enclosure)
	GB Stramback	GEH/San Jose (with enclosure)
	RE Brown	GEH/Wilmington (with enclosure)
	eDRF	0000-0075-9909

Enclosure 1

MFN 06-119, Supplement 4

Response to Portion of NRC Request for

Additional Information Letter No. 16

Related to ESBWR Design Certification Application

Piping Design

RAI Numbers 3.12-11 S01, 3.12-22 S01 and 3.12-27 S01

NRC RAI 3.12-11

DCD Tier 2, Appendix 3D, provides a description of the major computer programs used in the analysis and design of safety related components, equipment, and structures. According to this appendix, the quality of these programs and computer results is controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. To facilitate the staff review of the computer programs used in the ESBWR design, provide the following additional information:

(a) Identify which computer programs will be used during the design certification phase and which programs may be used in the future during the COL application phase.

(b) Identify which programs have already been reviewed by the NRC on prior plant license applications. Include the program name, version, and prior plant license application. As stated in SRP 3.9.1, this will eliminate the need for the licensee to resubmit, in a subsequent license application, the computer solutions to the test problems used for verification.

(c) Confirm that the following information is available for staff review for each program: the author, source, dated version, and facility; a description, and the extent and limitation of the program application; and the computer solutions to the test problems described above.

GE Response

(a) The programs used in the certification phase are:

PISYS07 It is a computer code for analyzing piping systems subjected to both static and dynamic piping loads.

ANSI713 The program is for calculating stresses and cumulative usage factors for Class 1, 2 and 3 piping components in accordance with articles NB, NC and ND-3650 of ASME Code Section III. ANSI7 is also used to combine loads and calculate combined service levels A, B, C and D load on piping supports and pipe-mounted equipment.

All of the programs in Appendix 3D.4 may also be used in the future during the COL application phase.

(b) PISYS05 has been benchmarked against NRC piping models. The results are documented in GE report NEDO 24210, dated August 1979 (Reference 3D 1 of Appendix 3D), for mode shapes and uniform support motion response spectrum analysis (USMA) options. The independent support motion response spectrum analysis (ISMA) option has been validated against NUREG/CR 1677.

The PISYS05 computer program has been reviewed by NRC, and the results are benchmarked with NUREG/CR-6049. PISYS07 USMA and ISMA analyses are the same as PISYS05. It has been benchmarked with NUREG/CR-6049.

(c) The computer programs listed in Appendix 3D are available for staff review. These programs are Level 2 programs. The author, source, dated version, and facility; a description, and the extent and limitation of the program application; and the computer solutions to the test problems are contained in the design record file of each program.

NRC RAI 3.12-11 S01

The issue involves the validation of the PISYS computer code used for the piping analysis. GE should verify that the PISYS computer code correctly implements the RG 1.92 procedure for mode combinations. In addition, GE should provide a technical justification for accepting the results at those locations that exceed the NUREG/CR-6049 acceptance criteria in the PISYS comparison with the NUREG/CR-6049 benchmark analysis.

GEH Response

GEH has modified the PISYS program to comply with RG 1.92 Rev. 2, 2006. The new version of the program is PISYS08. The PISYS08 program has been benchmarked with NUREG/CR-6049. The results are a 100% match with NUREG /CR-6049, except for a few values that are a 99% match. There were no locations that exceeded the NUREG/CR-6049 acceptance criteria in the PISYS08 comparison with the NUREG/CR-6049 benchmark analysis. Therefore, the requirements of RG 1.92 Rev. 2 have been met for the double sum of modal results and high frequency modes.

The detailed analysis and comparison are shown in GE-NE-0000-0070-1785-00, (eDRF 0000-0070-1785) "PISYS08 for Regulatory Guide 1.9R2 2006 and NUREG/CR-6049," a proprietary document, which is available for viewing in the GEH Washington office."

DCD Impact

No DCD changes will be made in response to this RAI.

NRC RAI 3.12-22

DCD Tier 1, Section 3.1, "Piping design," states that Class 1 piping systems will be analyzed for fatigue with environmental effects. Provide the analysis and design methods that will be used to perform the fatigue evaluation, including the environmental effects, for the ESBWR Class 1 piping systems.

GE Response

Requirements contained in ASME III NB-3653. The load combinations contained in Table 3.9-9, and the plant event cycles contained in Table 3.9-1 of the DCD, define the design conditions that are inputs to the fatigue analysis. Additionally, GE has additional design criteria for carbon steel and stainless steel materials that are intended to address environmental issues that have been applied to prior BWR applications, and are likewise being applied to the ESBWR piping design. Additionally, class 1 piping using a fatigue limit of 0.1 instead of the ASME Code acceptance limit of 1.0 in conjunction with a stress ratio limit of 0.80 for Equations 12 and 13 of the ASME Code in order to limit the number of pipe whip restraints within the containment. DCD paragraphs 3.9.3.3 and 3.9.3.4 will be revised in DCD Revision 2 to reflect this commitment as follows:

"Additionally, a fatigue usage limit of 0.10 is used as a design criteria for all Class 1 piping."

Evaluations have also determined that the ASME Code has conservative methods that provide additional margins. Specifically, the ASME Code adds stresses that include P, Ma, Mb, Mc, DT1, DT2, and Dtab by absolute sum when in actuality the direction and signs of the stresses are different. Reference (1) has performed a detail finite element analysis to compare against the results of a NB-3600 analysis and found that the fatigue usage based on NB-3600 is about 10 times more conservative.

This design criteria that is being used for ESBWR is consistent with the design methods used on previous BWR product lines that have successfully operated for the last 40 years without piping fatigue issues. Data from fatigue usage monitors from operating plants have also confirmed that the design criteria specified by GE in the original plant design was conservative.

The simplified NB-3600 analysis has been used for last 40 years successfully. If newly developed environmental fatigue curves are used, high fatigue usage factors are predicted and pipe break locations will be postulated throughout the plant. The economical cost to the plant is huge, and any gain of safety is questionable.

It is recommended that the environmental fatigue design curves should not be used without substantial simultaneous changes in analytical methodology and the ASME Code.

Ref.1. "Fatigue Usage Factor Evaluation For An Integrally Reinforced Branch Connection Using NB-3600 And NB-3200 Analysis Methods" by Henry L. Hwang, PE, General Electric Nuclear Energy, Jack R. Cole, PE, David M. Bosi, PE, Design Engineering, Washington Public Power Supply System. PVP Vol. 313-2, page 139 through 156.

NRC RAI 3.12-22 S01

The RG on environmental effects in the fatigue calculations of Class 1 piping will be issued soon. GE committed to implement the criteria for evaluating environmental effects, but will request some relaxation in the pipe break criterion for fatigue usage. GE will provide the results of a study showing the impact of the new environmental fatigue criteria to support its request to relax the pipe break fatigue usage criterion. This item is open pending staff review of the GE submittal.

GEH Response

The environmental effects on fatigue in accordance with DG-1144 and NUREG/CR-6909 has been incorporated in GEH piping program ANSI7014; however, this incorporation is conditional to the NRC accepting a change from 0.1 to 0.4 fatigue usage as specified in BTP EMEB 3-1 to exempt piping components from pipe break consideration. Since this change has previously been discussed with the NRC staff, GEH will proceed to change DCD sections 3.6.2, and Table 3.9-9 to incorporate this change.

GEH's study of the impact of implementing the new environmental fatigue criteria is shown in Attachment 1, PVP2007-26143, "Application of Draft Regulatory Guide DG-1144 Guidelines for Environmental Fatigue Evaluations to a BWR Feedwater Piping System". This paper contains a detailed description of the methodology and output comparisons of fatigue usage factor with and without inclusion of environmental fatigue.

DCD Impact

DCD Tier #2, Table 3.9-9 will be revised in Revision 5 as shown in the attached markup 1 .

DCD Tier #2, Section 3.6.2, will be revised in Revision 5 as shown in the attached markup 2. .

NRC RAI 3.12-27

DCD Tier 2, Section 3.7.3.12, discusses the effect of differential building movement on piping systems that are anchored and restrained to floors and walls of buildings that may have differential movements during a dynamic event. SRP 3.9.2 Section II.2.g states that the responses due to the inertial effect and relative displacement for multiply-supported equipment and components with distinct inputs should be combined by the absolute sum method. Provide the combination methods that are to be used in the design of ESBWR piping systems for the inertial responses and SAM responses caused by relative displacements for all analysis methods (including ISM).

GE Response

DCD Tier 2, Section 3.7.3.12, discusses the effect of differential building movement on piping systems that are anchored and restrained to floors and walls of buildings that may have differential movements during a dynamic event. In general, the piping systems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event. The movements may range from insignificant differential displacements between rigid walls of a common building at low elevations to relatively large displacements between separate buildings at a high seismic activity site.

Piping system is different from multiply-supported equipment. For piping system, the induced displacements in compliance with NB 3653 are treated differently than the inertia displacements. The SRSS method is a standard industrial practice to combine the inertial responses and SAM responses caused by relative displacements.

NRC RAI 3.12-27 S01

SRSS combination of the inertial and SAM responses for USM method of analysis is not consistent with the staff position in the Standard Review Plan (SRP). GE should provide additional technical justification for this position.

GEH Response

During the NRC audit meeting held between January 9 through January 12, 2007 at San Jose, CA (Reference NRC "Audit Trip Report," ML070930012), the NRC staff found that the SRSS combination for the inertial and SAM responses is acceptable for the piping stress analysis, except for piping support designs. For piping support design, the DCD is being revised to show that the absolute sum method (ABS) is used.

DCD Impact

DCD Tier 2, Section 3.7.3.12 will be revised in Revision 5 as shown in the attached markup 3.

ATTACHMENT 1

Proceedings of ASME-PVP 2007:

2007 ASME Pressure Vessel and Piping Division Conference, July 22-26, 2007, San Antonio TX, USA.

PVP2007-26143, "Application of Draft Regulatory Guide DG-1144 Guidelines for Environmental Fatigue Evaluation to a BWR Feedwater Piping System"

Proceedings of ASME-PVP 2007:
2007 ASME Pressure Vessel and Piping Division Conference
July 22-26, 2007, San Antonio, TX, USA

PVP2007-26143

APPLICATION OF DRAFT REGULATORY GUIDE DG-1144 GUIDELINES FOR ENVIRONMENTAL FATIGUE EVALUATION TO A BWR FEEDWATER PIPING SYSTEM

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ABSTRACT
Recently published Draft Regulatory Guide DG-1144 by the NRC provides guidance for use in determining the acceptable fatigue life of ASME pressure boundary components, with consideration of the light water reactor (LWR) environment. The analytical expressions and further details are provided in NUREG/CR-6909. In this paper, the environmental fatigue rules are applied to a BWR feedwater line. The piping material is carbon steel (SA333, Gr. 6) and the feedwater nozzle material is low alloy steel (SA508 Class 2). The transients used in the evaluation are based on the thermal cycle diagram of the piping. The calculated fatigue usage factors including the environmental effects are compared with those obtained using the current ASME Code rules. In both cases the cumulative fatigue usage factors are shown to be less than 1.0.

BACKGROUND & INTRODUCTION
Since the early 1980s the effects of high temperature water on the fatigue cyclic life of light water reactor [LWR] components have been extensively discussed by numerous researchers. References 1 through 15 are some of the examples. The Subgroup on Fatigue Strength of the ASME Boiler & Pressure Vessel Code is currently working on a Code Case that would provide procedures for incorporating the reactor water environmental effects in the fatigue evaluation conducted per the guidelines in Section III Paragraphs NB-3200 and NB-3600 [16].

Recently, the U.S. Nuclear Regulatory Commission (NRC) has published for public comment the Draft Regulatory Guide DG-1144 to provide guidance for use in determining the acceptable fatigue life of ASME pressure boundary components, with consideration of LWR environment [17]. The associated detailed guidance document is NUREG/CR-6909 [18]. The NRC addressed the public comments and is expected to issue the final version as Regulatory Guide 207. NUREG/CR-6909 adopted the environmental fatigue correction factor method or F_{en} method to account for the environmental fatigue effects. F_{en} is defined as the ratio of fatigue initiation life in air at room temperature to that in reactor water at the service temperature. The regulatory guides are not mandatory. However, the NRC is likely to ask applicants for certification of new reactor designs for the technical approach they plan to follow to address environmental fatigue effects.

This paper describes the results of the application of DG-1144 methodology to a BWR plant piping system. The system chosen is feedwater piping inside the containment. This system is typically classified as Class 1 per the ASME Code classification.

DESCRIPTION OF PIPING SYSTEM
Figure 1 schematically shows the Feedwater piping system. The piping system delivers the feedwater to the reactor. It also receives water from Residual Heat Removal (RHR) and Reactor Core Isolation Cooling (RCIC) systems. The portion of the piping between the reactor nozzle and the header at the

containment penetration is designed to ASME Class 1 requirements. Piping thickness is per schedule 80. The specified design pressure and temperature for this piping are 1250 psi and 550°F, respectively. The feedwater temperature during normal operation is 420° F.

PIPING STRESS ANALYSIS BY CURRENT CODE RULES

Figure 2 shows the mathematical model of the feedwater piping system. The piping nominal diameters are 22-inches at the containment penetration (Node 26 at the right hand bottom of Figure 2) and 12-inches at the point where the risers connect through a safe end to the feedwater nozzles (Nodes 49, 67 and 85 in Figure 2).

A complete Class 1 piping stress analysis, including a fatigue evaluation, was first conducted according to the rules of Paragraph NB-3600 of ASME Section III [19]. A GE proprietary computer program, ANSI7 [20], was used in the analysis. A key input to the Code fatigue evaluation is the pressure/temperature duty cycles for the system. Figure 3 shows a part of the pressure/temperature duty cycle for the feedwater system considered in this evaluation. It defines the expected feedwater temperature changes during the Hot Standby event. The number in each of the diamonds on this figure represents a defined load state. In the load state 28 the temperature changes from 126°C (259°F) to 282°C (540°F) in 10 minutes. The load state 29 is defined as a step drop in temperature from 282°C to 126°C (259°F). In a single Standby event these load states occur 24 times. Since there are 166 hot standby events postulated, the number of cycles for events 28 and 29 are (166x24) or 3984.

For the load states that involve a temperature transient, one-dimensional heat transfer analyses were conducted to define the appropriate values of temperature parameters used in the fatigue evaluation. Specifically, the quantities calculated were average temperatures on each side of a node point (T_a on side A and T_b, on side B), ΔT_1 (linear thermal gradient) and ΔT_2 (non-linear thermal gradient).

Table 1 shows a partial listing of the load states (hereinafter called load sets) information at Node 048 (at Feedwater nozzle). This information is used in developing the load set pair information for fatigue usage calculation. A partial listing of the load set pairs information for the same node is shown in Table 2. The last column shows the partial fatigue usage factor. For example, for the load set pair 28-29 (indicated by bold in Table 2), the calculated fatigue alternating stress is 151 MPa (column 8 from left side) and the corresponding partial fatigue usage factor is 0.0626 based on the current Code fatigue curve for carbon and low alloy steels with ultimate tensile stress less than 80 ksi. The calculated cumulative fatigue usage factors are discussed later in this paper along with the environmental fatigue usage factors.

ENVIRONMENTAL FATIGUE EVALUATION METHODOLOGY

Appendix A of Reference 18 provides the equations to calculate the environmental correction factor F_{en}. Table 3 extracted from Reference 18 shows the equations for carbon and low alloy steels, the materials of interest for feedwater line. The cumulative fatigue usage factor, U_{en}, considering the effects of reactor coolant environments is calculated as the following:

$$U_{en} = U_1 \bullet F_{en,1} + U_2 \bullet F_{en,2}\ U_3 \bullet F_{en,3}\ U_4 \bullet F_{en,4} + \ldots + U_n \bullet F_{en,n}$$

where, U_i and $F_{en,i}$ are the partial fatigue usage factor and the environmental fatigue correction factor, respectively, for the "i"th load set pair. The partial fatigue usage factor U_i is to be based on air fatigue curves at room temperature. The F_{en} is defined as the ratio of fatigue life in air at room temperature ($N_{air,RT}$) to that in water at the service temperature (N_{water}). Reference 18 also provides alternating stress (S) versus $N_{air,RT}$ curves for carbon, low alloy and stainless steels. These curves are different than the current S-N curves in the ASME Code. For convenience, Table 4 gives the digitized S-N values for the current Code curves and the Reference 18 curves.

F_{en} Calculation for a Load Set pair

A review of the equations in Table 3 indicates that F_{en} is a function of four parameters: S*, T*, O* and ε'*. For the purpose of this evaluation, most conservative values of S* (=0.015), O* (=ln{12.5}) and ε'* (=ln{0.001}) were assumed.

The Appendix A of Reference 18 allows the use of average of the maximum and minimum temperatures in a transient in the determination of the appropriate temperature T for the calculation of parameter T*. This approach was followed in this evaluation as illustrated by a sample calculation described next.

A part of the partial cumulative fatigue usage factor calculation for Node 048 using the current ASME fatigue curves, is shown in Table 2. Temperature T for the calculation of F_{en} for the load set pair 28-29 (indicated by bold in the table) was determined as follows. As seen in Figure 3, the maximum and minimum temperatures during these two load sets are 282°C (540°F) and 126°C (259°F), respectively. Therefore, the average temperature during the transient is [(282+126)/2] or 204°C (399°F). Thus, T* = (204-150) or 54. For the nozzle side at node 048, the material is low alloy steel. The F_{en} for this load set pair is calculated as:

$$\begin{aligned} F_{en} &= \exp[0.702 - 0.101 x 0.015 x 54.0 x \{\ln(12.5)\} x \ln(0.001)] \\ &= \exp[0.702 + 0.101 x 0.015 x 54.0 x 2.5257 x 6.9078] \\ &= 8.409 \end{aligned}$$

The partial fatigue usage factor for load set pair 28-29 is shown as 0.0626 in Table 2. It is noted that this is based on the current Code fatigue curve. For the same level of alternating stress (151 MPa), the allowable number of cycles on the low alloy steel S-N curve given in NUREG/CR-6909 (see column 4 in Table 4) would be 159820. This would give air curve partial fatigue usage of (3964/159820) or 0.0248. It is seen that the use of NUREG/CR-6909 air fatigue curve results in a reduction of better than factor of two reduction in the partial fatigue usage value for this load set pair. The partial fatigue usage for this load set pair considering environmental fatigue effects is (0.0248x8.409) or 0.208.

A similar calculation for F_{en} on the safe end side, that is carbon steel, gives a value of 7.841. For the alternating stress level of 151 MPa for this load set pair, the allowable number of cycles on the low alloy steel S-N curve given in NUREG/CR-6909 (see column 3 in Table 4) would be 576820. This would give air curve partial fatigue usage of (3964/576820) or 0.0069. The use of NUREG/CR-6909 air fatigue curve for carbon steel results in a reduction of an almost an order of magnitude in the partial fatigue usage value for this load set pair. The partial fatigue usage for this load set pair considering environmental fatigue effects is (0.0069x7.841) or 0.054. It is seen that at least for this load set pair the reduction in partial air fatigue usage through the use of NUREG/CR-6909 curve more than offset the increase due to F_{en}.

A subroutine that calculates cumulative fatigue usage including reactor water effects according to DRG-1144, was added to the ANSI7 computer code used in the piping stress analyses. The calculation results for the subject feedwater piping are discussed in the next section.

ENVIRONMENTAL FATIGUE EVALUATION RESULTS

Table 5 provides a summary of the calculated values of cumulative fatigue usage factors at two locations. For the feedwater nozzle location, usage factors are provided for both the nozzle side (low alloy steel) and the safe end side (carbon steel). It is seen that there is a modest impact on the calculated fatigue usage factors when reactor water environmental effects are factored in. At the safe end location, the reduction is air fatigue usage through the use of NUREG/CR-6909 S-N curves, essentially offset the increase due to the use of F_{en}.

DISCUSSION

The increase in calculated fatigue usage when environmental fatigue effects are taken into account, was modest for the feedwater piping considered in this evaluation. One of the reasons is that the normal operating temperature for the feedwater line (240°C) is comparatively lower than the typical operating temperatures for the primary piping in LWRs. In the case of carbon and low alloy steel piping systems, the increase due to the use of F_{en} is significantly offset by the advantage gained through the use of air S-N curves provided in NUREG/CR-6909. This would not be the case for stainless steel piping systems where the air S-N curves in NUREG/CR-6909 predict higher usage factor than the Code curve.

In general one would expect several fold increase in the calculated fatigue usage factor when F_{en} is used. This would have implications in terms of number of locations where hypothetical pipe breaks need to be postulated. Currently, the NRC Branch Technical Position MEB 3-1 [21] is used for postulation of breaks in high energy lines. MEB 3-1 requires postulation of a break at an intermediate locations if the fatigue usage at a location exceeds 0.1 or the primary plus secondary stress range exceeds 2.4 S_m. The calculated primary plus secondary stress range is not impacted by the use of F_{en} but the fatigue usage factor is. The use of F_{en} results in more locations where cumulative fatigue usage factor would exceed 0.1. More break locations means more pipe whip restraints to meet the requirements of General Design Criterion 4 of 10CFR50 [22]. However, the presence of more pipe whip restraints adversely affects the ability to conduct piping in-service inspections and thus have a negative impact on piping reliability during operation. The 0.1 fatigue usage threshold was based on engineering judgment and perhaps can be revised upwards to say 0.4 or 0.6 to avoid this situation. The revision could be justified through a piping reliability analysis somewhat similar to that conducted in support of revised Appendix L in ASME Section XI Code [23].

SUMMARY AND CONCLUSIONS

This paper presents the results of a Class 1 stress analysis of a BWR Feedwater piping system in which the environmental fatigue effects due to reactor water per DG-1144 were included. The materials considered were carbon and low alloy steels. The results showed that there is a modest increase in the calculated fatigue usage factors but the values were found to be acceptable (i.e., less than 1.0). The increase in fatigue usage may result in more locations where CUF exceeds 0.1 thereby resulting in more locations with break postulations and requirement for installation of pipe whip restraints. An upward revision of 0.1 fatigue usage threshold is recommended through a piping reliability study. This paper did not include stainless steel piping system evaluation where the impact of DG-1144 procedures may result in a significant increase in the calculated cumulative fatigue usage factor.

REFERENCES

[1] Hale, D.A., et al., "Low Cycle Fatigue of Commercial Piping Steels in a BWR Primary Environment," J. Engineering Materials Technology, Vol. 103, pp. 15-25, 1981.

[2] Ranganath, S., J.N. Kass, and J.D. Heald, "Fatigue Behavior of Carbon Steel Components in High-Temperature Water Environments," ASTM STP 770, American Society for Testing and Materials, pp. 436-459, 1982.

[3] Higuchi, M. and K. Iida, "Fatigue Strength Correction Factors for Carbon and Low-Alloy Steels in Oxygen-Containing High-Temperature Water," Nuclear Engineering Design, Vol. 129, pp. 293-306, 1991.

[4] S. Majumdar, O.K. Chopra, and W.J. Shack, "Interim Fatigue Design Curves for Carbon, Low-alloy, and Austenitic Stainless Steels in LWR Environments," NUREG/CR-5999, April 1993.

[5] Mehta, H.S. and S.R. Gosselin, "Environmental Factor Approach to Account for Water Effects in Pressure Vessel and Piping Fatigue Evaluations," Nuclear Engineering and Design, Vol. 181, pp-175-197, 1998.

[6] Mehta, H.S., "An Update on the Consideration of Reactor Water Effects in Code Fatigue Initiation Evaluations for Pressure Vessels and Piping," ASME PVP Volume 410-2, pp. 45-51, 2000.

[7] Higuchi, M., "Fatigue Curves and Fatigue Design Criteria for Carbon and Low alloy Steels in High Temperature Water," ASME PVP Volume 386, pp. 161-169, 1999.

[8] Higuchi, M., "Revised Proposal of Fatigue Life Correction Factor F_{en} for Carbon and Low alloy Steels in LWR Environments," ASME PVP Volume 410-2, 2000.

[9] Higuchi, M., Iida, K., Hirano, A., Tsutsumi, K. and Sakaguchi, K., "A Proposal of Fatigue Life Correction Factor F_{en} for Austenitic Stainless Steels in LWR Water Environments," ASME PVP Volume 439, pp. 109-117, 2002.

[10] Chopra, O.K. and W.J. Shack, "Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels," NUREG/CR-6717, May 2001.

[11] Van Der Sluys, W.A., "PVRC's Position on Environmental Effects on fatigue Life in LWR Applications," Welding Research Council Bulletin 487, December 2003.

[12] Higuchi, M., Hirano, T. and Sakaguchi, K., "Evaluation of Fatigue Damage on Operating Plant Components in LWR Water," ASME PVP Volume 480, pp. 129-138, 2004.

[13] O'Donnell, W.J., W.J. O'Donnell, and T.P. O'Donnell, "Proposed New Fatigue Design Curves for Austenitic Stainless Steels, Alloy 600, and Alloy 800," Proc. Of the 2005 ASME Pressure Vessels and Piping Conference, July 17-21, 2005, Denver, CO, Paper # PVP2005-71409.

[14] O'Donnell, W.J., W.J. O'Donnell, and T.P. O'Donnell, "Proposed New Fatigue Design Curves for Carbon and Low-Alloy Steels in High Temperature Water," Proc. of the 2005 ASME Pressure Vessels and Piping Conference, July 17-21, 2005, Denver, CO, Paper # PVP2005-71410.

[15] Nakamura, T., M. Higuchi, T. Kusunoki and Y. Sugie, "JSME Codes on Environmental Fatigue Evaluation," Proc. of the 2006 ASME Pressure Vessels and Piping Conference, July 23-27, 2006, Vancouver, Canada, Paper # PVP2006-ICPVT11-93305.

[16] Letter from Subcommittee on Nuclear Power (Bill O'Donnell) to Richard Barnes, Chairman Subcommittee III, dated February 9, 2007, Subject: Implementation of February 1, '07 Plan to Resolve Environmental Fatigue Issue for Nuclear Power Plants.

[17] Draft Regulatory Guide DG-1144: Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light Water Reactor Environment for New Reactors; US Nuclear Regulatory Commission, July 2006.

[18] Chopra, O.K. and W.J. Shack, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," NUREG/CR-6909, Draft Report, February 2006; Final Report, February 2007.

[19] "Rules for Construction of Nuclear Power Plant Components," Section III, Division 1, ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers.

[20] Piping Stress Analysis Computer Code, ANSI7 (GE Proprietary).

[21] Standard Review Plan Section 3.6.2 of NUREG-0800: Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 2, June 1987.

[22] 10 CFR Part 50, appendix A, General Design Criterion 4, "Environmental and Missile Design Basis."

[23] *Materials Reliability Program: Recommended Improvements to ASME Section XI Appendix L (MRP-82)*, EPRI, Palo Alto, CA and U.S. department of Energy, Washington, D.C., 2002.

Table 1 Example of Load Set Information for Node 048

TRANSITION NEAR NODE 048.

LOAD SET NO.	NO. OF CYCLES	PRESSURE (MPa)	MOMENTS (N-M) MA/MPA	MB/MRB	MC/MRC	Temperatures (°C) TAS	TBS	DT1	DT2
23	41.	1.	-22326220	32425462	-7171240	144.20	144.67	-0.28	0.000
24	41.	1.	-3200050	2298608	-34176400	61.16	64.88	-1.91	-0.29
25	10.	8.	-39893680	54705392	-25590656	215.23	215.33	-7.36	-6.23
26	166.	7.	619056	-18744768	-76667952	25.49	26.65	-0.69	0.00
27	166.	7.	-10539570	2243589	-61756500	81.58	80.17	1.07	0.11
28	**3984.**	**7.**	**-27788684**	**42316516**	**10034424**	**133.00**	**132.00**	**7.60**	**1.60**
29	**3984.**	**7.**	**-20168404**	**19521136**	**-49962788**	**255.00**	**261.00**	**-43.00**	**-8.90**
30	35.	7.	-10539570	2243589	-61756500	196.00	203.00	-40.00	-8.10
31	35.	7.	-42417240	69191672	-6496594	277.55	266.83	4.95	1.13
32	35.	7.	-1888785	28477412	-71765568	139.00	144.00	-7.00	-1.40
33	166.	7.	-25786532	74809520	-42237400	161.00	159.00	1.70	0.40
34	70.	1.	-20134478	24867194	-31993436	129.25	129.84	-0.35	0.00
35	70.	1.	-30692016	44807892	-11857880	176.62	166.48	4.16	0.68
36	166.	0.	-2751758	1362957	-30320896	54.18	54.77	-0.27	0.00

Table 2 Fatigue Usage Calculation Process at Node 048 Using Current Code Curve

TRANSITION NODE 048.

STRESS RANGE AND FATIGUE USAGE

LOAD SETS I	J	STRESS EQUATIONS 11(SP)	10 (SN)	12 (SE)	13 (SX)	KE	14(ALTS)	RS/ 3SM	DT1/ ALWDT1	CYCLES EXPT	CYCLES ALLOW	USAGE FACTOR
38	42	483.	170.	147.	35.	1.00	242.	0.465	0.112	93.	13149.	0.0071
15	47	469.	212.	48.	90.	1.00	234.	0.580	0.077	5.	14612.	0.0003
40	46	425.	147.	89.	74.	1.00	213.	0.403	0.093	8.	20332.	0.0004
15	31	382.	154.	62.	94.	1.00	191.	0.422	0.086	35.	28291.	0.0012
35	40	380.	159.	89.	82.	1.00	190.	0.434	0.054	70.	28637.	0.0024
15	39	355.	126.	44.	89.	1.00	177.	0.345	0.102	30	35368.	0.0008
28	40	337.	113.	105.	27.	1.00	169.	0.310	0.060	15.	41391.	0.0004
28	30	335.	108.	93.	29.	1.00	167.	0.295	0.088	35.	42243.	0.0008
54	55	334.	185.	0.	20.	1.0	167.	0.507	0.000	50.	42743.	0.0012
28	**29**	**303.**	**85.**	**71.**	**28.**	**1.00**	**151.**	**0.231**	**0.094**	**3934.**	**62894.**	**0.0626**
1	18	297.	140.	62.	92.	1.00	149.	0.384	0.053	70.	68988.	0.0010
1	29	295.	110.	47.	79.	1.00	147.	0.301	0.080	50.	71470.	0.0007
21	45	286.	173.	92.	93.	1.00	143.	0.472	0.008	8.	83381.	0.0001
6	21	275.	169.	95.	88.	1.00	137.	0.463	0.005	27.	101378.	0.0003
10	16	256.	152.	97.	72.	1.00	128.	0.415	0.007	54.	131245.	0.0004

Table 3 F_{en} Equations from Appendix A of NUREG/CR-6909

The nominal environmental fatigue correction factor, $F_{en,nom}$, for carbon steels is expressed as

$$F_{en,nom} = \exp(0.632 - 0.101\ S^*\ T^*\ O^*\ \dot{\varepsilon}^*), \qquad (A.2)$$

and for low-alloy steels, it is expressed as

$$F_{en,nom} = \exp(0.702 - 0.101\ S^*\ T^*\ O^*\ \dot{\varepsilon}^*), \qquad (A.3)$$

where S^*, T^*, O^*, and $\dot{\varepsilon}^*$ are transformed S content, temperature, DO level, and strain rate, respectively defined as:

$S^* = 0.001$	$(S \le 0.001\ \text{wt.\%})$	
$S^* = S$	$(S \le 0.015\ \text{wt.\%})$	
$S^* = 0.015$	$(S > 0.015\ \text{wt.\%})$	(A.4)
$T^* = 0$	$(T < 150°C)$	
$T^* = T - 150$	$(T = 150–350°C)$	(A.5)
$O^* = 0$	$(DO \le 0.04\ \text{ppm})$	
$O^* = \ln(DO/0.04)$	$(0.04\ \text{ppm} < DO \le 0.5\ \text{ppm})$	
$O^* = \ln(12.5)$	$(DO > 0.5\ \text{ppm})$	(A.6)
$\dot{\varepsilon}^* = 0$	$(\dot{\varepsilon} > 1\%/s)$	
$\dot{\varepsilon}^* = \ln(\dot{\varepsilon})$	$(0.001 \le \dot{\varepsilon} \le 1\%/s)$	
$\dot{\varepsilon}^* = \ln(0.001)$	$(\dot{\varepsilon} < 0.001\%/s)$.	(A.7)

Table 4 S-N Values in Current Code and NUREG/CR-6909

Cycles	CS/LAS Current Code [UTS<80Ksi] (MPa)	CS Air NUREG/CR-6909 (MPa)	LAS/Air NUREG/CR-6909 (MPa)	SS Current Code (MPa)	SS NUREG/CR-6909 (MPa)
10	3999.2	5357.5	5467.8	4881.7	5998.8
20	2827.0	3833.7	3882.0	3530.3	4302.6
50	1896.2	2509.8	2440.9	2378.8	2751.2
100	1413.5	1820.3	1758.3	1799.6	1978.9
200	1068.7	1358.3	1303.2	1385.9	1441.1
500	724.0	937.7	903.3	1020.5	972.2
1000	572.3	730.9	717.1	820.5	744.7
2000	441.3	584.0	577.8	668.8	590.2
5000	331.0	453.0	435.1	524.0	450.3
10000	262.0	373.0	348.2	441.3	368.2
2.00E+04	213.7	304.8	277.9	382.7	299.9
5.00E+04	158.6	237.9	210.3	319.2	235.1
1.00E+05	137.9	201.3	171.7	281.3	195.8
2.00E+05	113.8	175.8	142.0	247.5	168.2
5.00E+05	93.1	153.8	115.8	213.7	142.0
1.00E+06	86.2	142.7	106.2	195.1	126.2
2.00E+06	83.2	138.0	102.5	157.2	113.1
5.00E+06	79.3	131.9	97.8	126.9	102.0
1.00E+07	76.5	127.6	94.5	113.1	99.3
2.00E+07	73.9	123.2	91.2	104.8	98.7
5.00E+07	70.7	117.8	87.1	98.6	97.8
1.00E+08	68.3	113.8	84.1	97.2	97.2
1.00E+09	60.7	101.4	75.2	95.8	95.8
1.00E+10	54.5	89.6	66.9	94.5	94.5
1.00E+11	48.3	80.0	59.3	93.8	93.8

Table 5 Current Code and Environmental Fatigue Usage Factors

Node/Location/Material	Fatigue Usage by Current Code	Fatigue Usage by NUREG/CR-6909
Node 26/Header/CS	0.083	0.117
Node 48/Nozzle/LAS	0.085	0.302
Node 48/Safe End/CS	0.085	0.086

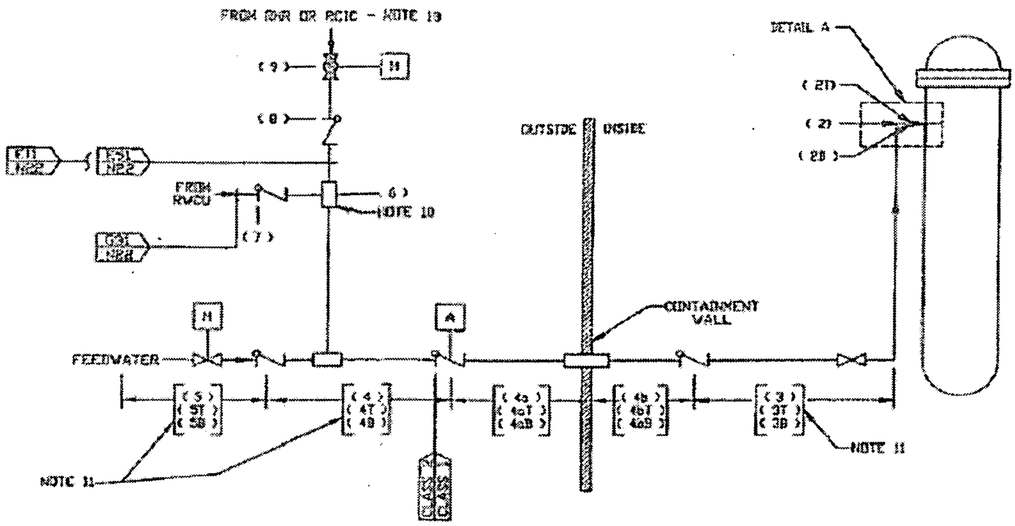

Figure 1 Schematic of Feedwater Piping System

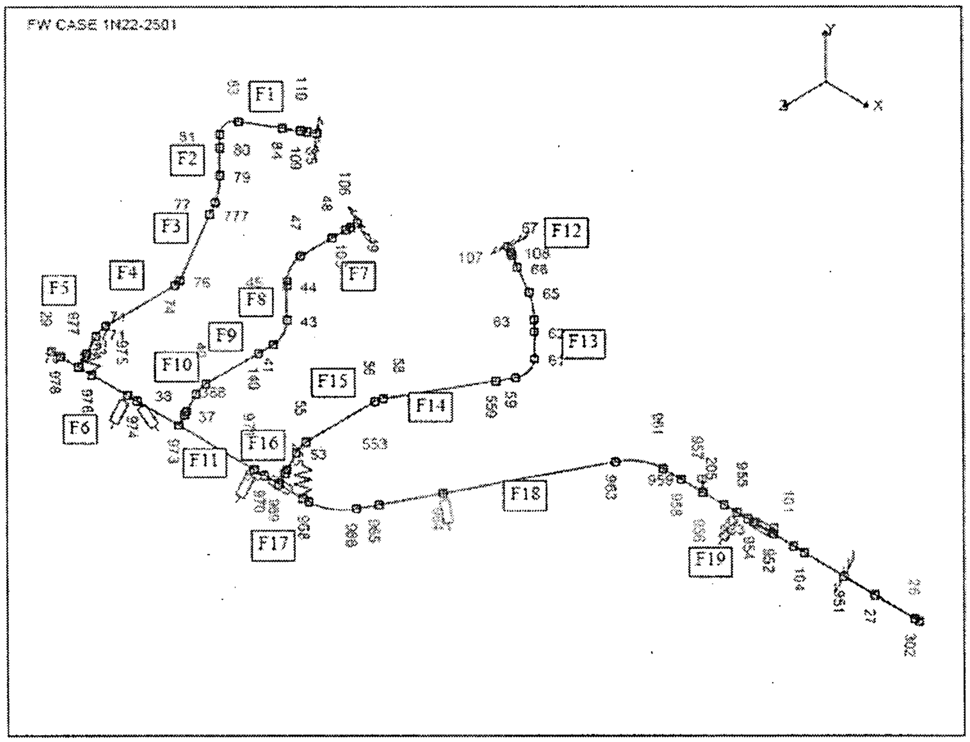

Figure 2 Mathematical Model of Feedwater Piping System

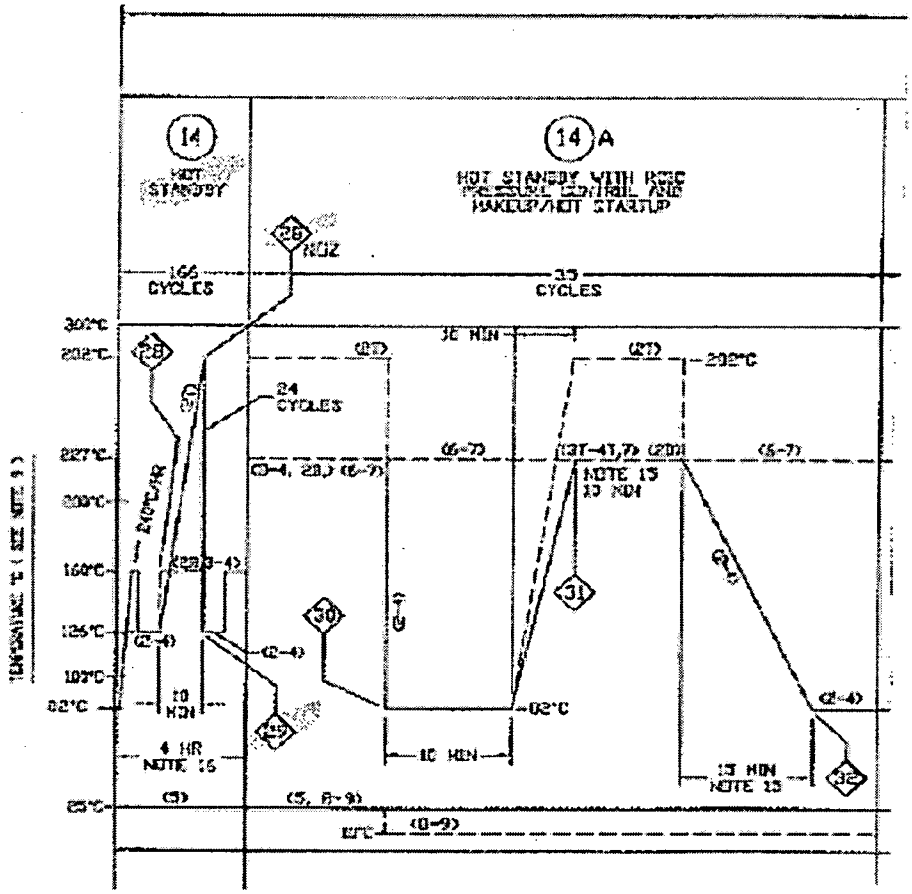

Figure 3 An Extract from Pressure/Temperature Cycle Diagram for feedwater System

DCD Tier 2
Revision 5 Markups

Table 3.9-9

Load Combinations and Acceptance Criteria for Class 1 Piping Systems

Condition	Load Combination for all terms[(1) (2)(3)]	Acceptance Criteria
Design	PD + WT	Eq 9 ≤ 1.5 S_m NB-3652
Service Level A & B	PP, TE, ΔT1, ΔT2 , TA-TB, RV_1, RV_2I, RV_2D, TSV, SSEI, SSED	Eq 12 & 13 ≤ 2.4 S_m Fatigue - NB-3653: U < 0.40[(4)]
Service Level B	PP + WT + (TSV) PP + WT + (RV_1) PP + WT + (RV_2I)	Eq 9 ≤ 1.8 S_m, but not greater than 1.5 S_y Pressure not to exceed 1.1P_a (NB-3654)
Service Level C	PP + WT + $[(CHUGI)^2 + (RV_1)^2]^{1/2}$ PP + WT + $[(CHUGI)^2 + (RV_2I)^2]^{1/2}$	Eq 9 ≤2.25 S_m, but not greater than 1.8 S_y Pressure not to exceed 1.5 P_a (NB-3654)
Service Level D	PP + WT + $[(SSEI)^2 + (TSV)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CHUGI)^2 + (RV_1)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CHUGI)^2 + (RV_2I)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CONDI)^2 + (RV_1)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (CONDI)^2 + (RV_2I)^2]^{1/2}$ PP + WT + $[(SSEI)^2 + (API)^2]^{1/2}$	Eq 9 ≤ 3.0 S_m but not greater than 2.0 S_y Pressure not to exceed 2.0 P_a (NB-3654)

(1) RV1 and TSV loads are used for MS Lines only

(2) RV2 represents RV2 ALL (all valves). RV2SV (single Valve) and RV2 AD (Automatic Depressurization operation)

(3) For the SRV discharge piping, all direct loads for SRV and LOCA loads are evaluated for submerged piping.

(4) In conjunction with compliance with RG 1.207, the fatigue usage limit of ≤ 0.40 will be used as the criteria for piping locations exempt from pipe break consideration.

Where: API = Annulus Pressurization Loads (Inertia Effect)

CHUGI = Chugging Load (Inertia Effect)

ONDI = Condensation Oscillation (Inertia Effect)

PD = Design Pressure

PP = Peak Pressure or the Operating Pressure Associated with that transient

RV_1 = SRV Opening Loads (Acoustic Wave)

- The pressure, water level, and flow sensor instrumentation for those safety-related systems, which are required to function following a pipe rupture, are protected.
- High-energy fluid system pipe whip restraints and protective measures are designed so that a postulated break in one pipe could not, in turn, lead to a rupture of other nearby pipes or components, if the secondary rupture could result in consequences that would be considered unacceptable for the initial postulated break.
- For any postulated pipe rupture, the structural integrity of the containment structure is maintained. In addition, for those postulated ruptures classified as a loss of reactor coolant, the design leaktightness of the containment fission product barrier is maintained.
- Safety relief valves (SRVs) are located and restrained so that a pipe failure would not prevent depressurization.
- Protection for the FMCRD scram insert lines is not required, because the motor operation of the FMCRD can adequately insert the control rods even with a complete loss of insert lines (Subsection 3.6.2.1.3).
- The escape of steam, water, combustible or corrosive fluids, gases, and heat in the event of a pipe rupture do not preclude:
  - accessibility to any areas required to cope with the postulated pipe rupture;
  - habitability of the control room; or
  - the ability of safety-related instrumentation, electric power supplies, components, and controls to perform their safety-related function.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis for dynamic effects are discussed in this Subsection in accordance with NUREG-0800 Draft Rev. 2, April 1996, SRP 3.6.2. This includes location criteria and methods of analysis needed to evaluate the dynamic effects associated with postulated breaks and cracks in high and moderate-energy fluid system piping inside and outside of the primary containment. This information provides the basis for the requirements for the protection of safety-related structures, systems, and components defined in the introduction of Section 3.6, which includes meeting the requirements of GDC 4 as it relates to safety-related structures, systems and components (SSC) being designed to accommodate the dynamic effects of postulated pipe rupture, including postulation of pipe rupture locations; break and crack characteristics; dynamic analysis of pipe-whip; and jet impingement loads.

The plant meets the relevant requirements of GDC 4 as follows:

(1) Criteria defining postulated pipe rupture locations and configurations inside containment are in accordance with Branch Technical Position (BTP) EMEB 3-1. For the piping system with reactor water, if the environmental fatigue is included in accordance with RG. 1.207, the fatigue usage limit should be $\leq$ 0.40 as the criteria instead of $\leq$ 0.10 for determining pipe break locations.

(2) Protection against postulated pipe ruptures outside containment is provided in accordance with BTP EMEB 3-1.

(3) Detailed acceptance criteria covering pipe-whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement are in accordance with Section III of SRP 3.6.2. The general bases and assumptions of the analysis are in accordance with BTP EMEB 3-1.

Piping in Containment Penetration Areas

No pipe breaks or cracks are postulated in those portions of piping from the containment wall penetration to and including the inboard or outboard isolation valves which meet the following requirements in addition to the requirement of the ASME Code, Section III, Subarticle NE-1120:

- The following design stress and fatigue limits are not exceeded:

  For ASME Code, Section III, Class 1 Piping

  – The maximum stress range between any two load sets (including the zero load set) does not exceed 2.4 S_m, and is calculated by Equation (10) in NB-3653, ASME Code, Section III. If the calculated maximum stress range of Equation (10) exceeds 2.4 S_m, the stress ranges calculated by both Equation (12) and Equation (13) in paragraph NB-3653 shall meet the limit of 2.4 S_m.

  – The cumulative usage factor is less than 0.1.

  For the piping system with reactor water, if the environmental fatigue effect is included in accordance with RG 1.207, the fatigue usage limit should be ≤ 0.40 as the criteria instead of ≤ 0.10 for determining pipe break locations.

The maximum stress as calculated by Equation (9) in NB-3652 under the loadings resulting from a postulated piping failure beyond those portions of piping, does not exceed the lesser of 2.25 S_m and 1.8 S_y except that, following a failure outside containment, the pipe between the outboard isolation valve and the first restraint may be permitted higher stress, provided a plastic hinge is not formed and operability of the valves with such stresses is assured in accordance with the requirement identified in Subsection 3.9.3. Primary loads include those that are deflection limited by whip restraints.

ASME Code Section III Class 1 Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping identified above, breaks in ASME Code, Section III, Class 1 piping are postulated at the following locations in each piping and branch run:

- At terminal ends.
- At intermediate locations where the maximum stress range as calculated by Equation (10) in NB-3653, ASME Code, Section III exceeds 2.4 S_m, and either Equation (12) or Equation (13) in Paragraph NB-3653 exceeds 2.4 S_m.
- At intermediate locations where the cumulative usage factor exceeds 0.1. As a result of piping reanalysis caused by differences between the design configuration and the as-built configuration, the highest stress or cumulative usage factor locations may be shifted;

however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:

- The dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.
- A change is required in pipe parameters, such as major differences in pipe size, wall thickness, and routing.

For the piping system with reactor water, if the environmental fatigue effect is included in accordance with RG 1.207, the fatigue usage limit should be ≤ 0.40 as the criteria instead of ≤ 0.10 for determining pipe break locations.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

Definition of High-Energy Fluid Systems

High-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.6.1.1), are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- maximum operating temperature exceeds 93.3°C (200°F); or
- maximum operating pressure exceeds 1.9 MPaG (275 psig).

Definition of Moderate-Energy Fluid Systems

Moderate-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.6.1.1), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

- maximum operating temperature is 93.3°C (200°F) or less; and
- maximum operating pressure is 1.9 MPaG (275 psig) or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period, qualify as moderate-energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2% of the total time that the system operates as a moderate-energy fluid system.

Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or a sudden longitudinal split without pipe severance, and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe failures are limited to postulation of cracks in piping and branch runs; these cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe. High-energy fluid systems are also postulated to have cracks for

In place of the response spectrum analysis, the ISM time history method of analysis is used for multi-supported systems subjected to distinct support motions, in which case both inertial and relative displacement effects are already included.

3.7.3.10 Use of Equivalent Vertical Static Factors

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.7.2.1.3 are satisfied.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are included for subsystems similar to that for the piping systems discussed in Subsection 3.7.3.3.1.

3.7.3.12 Effect of Differential Building Movements

In most cases, subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event. The movements may range from insignificant differential displacements between rigid walls of a common building at low elevations to relatively large displacements between separate buildings at a high seismic activity site.

Differential endpoint or restraint deflections cause forces and moments to be induced into the system. The stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the system satisfy the condition which caused the stress to occur.

For the piping stress analysis, SRSS combination for the inertial and the SAM (Seismic Anchor Motion, including Effect of Differential Building Movements) responses is acceptable. For the piping support design, the absolute sum method (ABS) is used.

3.7.3.13 Seismic Category I Buried Piping, Conduits and Tunnels

There is no directly buried Seismic Category I (C-I) piping or conduits that are directly buried underground.

Fire Protection System (FPS) yard piping with a C-I classification are installed in covered reinforced concrete trenches near surface with removable covers to facilitate maintenance and inspection access.

There are C-I conduits in four electrical duct banks from the CB to the RB. The duct banks are installed in closed concrete trenches covered with backfill.

There are no C-I tunnels in the ESBWR design. The access tunnel (AT), which includes walkways between and access to RB, CB, Turbine Building (TB), and Electrical Building (EB) is classified Seismic Category II (C-II). Since C-II structures are designed to the same criteria as C-I structures there is no impact to adjacent C-I structures.